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15.0 ACCIDENT ANALYSES

A new main feedwater Leading Edge Flow Meter (LEFM) system was installed in SQN Units 1 and 2. The LEFM resulted in a 1.3% reduction in the calorimetric uncertainty of the secondary side power measurement. TVA took advantage of this reduction by making an equivalent 1.3% upgrade in rated thermal power. The new rated thermal power is 3455 MWt and the new calorimetric uncertainty is 0.7%. However, the value of rated thermal power plus calorimetric uncertainty is identical before and after the power level upgrade. Before the power level upgrade, the rated thermal power plus calorimetric uncertainty was 102% of 3411 MWt = 3479 MWt and after the power level upgrade it is 100.7% of 3455 MWt = 3479 MWt.

All FSAR Chapter 15 LOCA and non-LOCA safety analyses were evaluated relative to the power level upgrade. Transient analyses that assumed an initial core power of 102% of rated thermal power or greater were unaffected by the power level upgrade. Safety analyses performed at zero power conditions were also unaffected by the power level upgrade. The remainder of the Chapter 15 safety analyses were either insensitive to power level considerations or were bounded by other events. The key factor in the evaluation of the FSAR Chapter 15 events is a 1.3% power level upgrade coincident with an equivalent reduction in calorimetric uncertainty. Therefore, none of the Chapter 15 events were reanalyzed for the power level upgrade.

FSAR Chapter 15 LOCA and non-LOCA analyses were evaluated for the Unit 1 replacement steam generators. The analyses (see Reference 28) demonstrate that the acceptance criteria continue to be met subsequent to the steam generator replacement. The analysis for the OSGs continues to be applicable to the RSGs for tube plugging level up to 15 percent.

15.1 CONDITION I - NORMAL OPERATION AND OPERATIONAL TRANSIENTS

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

A typical list of Condition I events is listed below:

1. Steady state and shutdown operations

    a. Power operation (~15 to 100 percent of full power)

    b. Start up (or standby) (critical, 0 to 15 percent of full power)

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

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

    e. Refueling

2. Operation with permissible deviations

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

a. Operation with components or systems out of service

b. Leakage from fuel with cladding defects

c. Activity in the reactor coolant

   i. Fission products

   ii. Corrosion products

   iii. Tritium

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

e. Operation with loss of one (redundant) DC system as permitted by Technical Specifications.

3. Operational transients

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

b. Step load changes (up to ± 10 percent)

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

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

15.1.1 Optimization Of Control Systems

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

For each mode of plant operation, a group of optimum controller setpoints was determined. In areas where the resultant setpoints were different, compromises based on the optimum overall performance were made and verified. A consistent set of control system parameters was derived satisfying plant operational requirements throughout the core life and for power levels between 15 and 100 percent. The study comprised an analysis of the following control systems: Rod cluster control assembly, steam dump, steam generator level, pressurizer pressure and pressurizer level.

15.1.2 Initial Power Conditions Assumed In Accident Analyses

15.1.2.1 Power Rating

Table 15.1.2-1 lists the principal power rating values which are assumed in analyses performed in this section. Two ratings are given:

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

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

Where initial power operating conditions are assumed in accident analyses, the "guaranteed Nuclear Steam Supply System thermal power output" plus allowance for errors in steady state power determination is assumed. Where demonstration of adequacy of the containment and Engineered Safety Features are concerned, the "Engineered Safety Features design rating" plus allowance for error is assumed. The thermal power values for each transient analyzed are given in Table 15.1.2-2.

15.1.2.2 Initial Conditions

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

1.	Core power	+ 0.7 percent allowance calorimetric error based on LEFM on feedwater header (Reference 27)
2.	Average Reactor Coolant System temperature	+ 4°F allowance for deadband and measurement error*
3.	Pressurizer pressure	+ 30/-42 psi allowance for steady state fluctuations and measurement error

*A uniform temperature distribution between 578.2°F and 583°F was used in the realistic large break loss-of-coolant accident analysis consistent with the realistic analysis methodology.

Installation of the LEFM effectively reduces the 2% calorimetric error to 0.7%. However, the results presented in Chapter 15 continue to be based on 2% error because they are based on the original rated thermal power of 3411 MWt. 102% of the original rated thermal power of 3411 MWt is equivalent to 100.7% of the upgraded rated thermal power of 3455 MWt.

The text of Chapter 15 contains several references to a rated thermal power of 3411 MWt in combination with a calorimetric uncertainty of 2%. Unless indicated otherwise, plots and results related to reactor power are reported in terms of the original rated thermal power and calorimetric uncertainty. In the interest of brevity, the references to the original rated thermal power and calorimetric uncertainty were left unchanged since the power level upgrade is exactly balanced by a reduction in calorimetric uncertainty.

The magnitude of the errors, not the absolute temperatures, are about the same.

The outer surface of the fuel rod at the hot spot operates at a temperature of approximately 660°F for steady state operation at rated power throughout core life due to the onset of nucleate boiling. Initially (beginning of life), this temperature is that of the cladding metal outer surface. During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the cladding surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel cladding and the reactor coolant so that the core thermal output is not limited by considerations of the cladding temperature. These temperatures are calculated using the Westinghouse fuel rod model (Reference 1) which has been reviewed and approved by the NRC.

15.1.2.3 Power Distribution

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

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.1.3-1. All transients that may be DNB limited are assumed to begin with $F_{\Delta H}$ consistent with the initial power level defined in the technical specifications. The axial power shape used in the DNB calculation is the 1.55 chopped cosine as discussed in Subparagraph 4.4.3.2.2.

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

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

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.1.3-1. Reference is made in that table to overtemperature and overpower ΔT trip shown in Figure 15.1.3-1.

Accident analyses which assume the S/G Low-Low Water Level trip signal to initiate protection functions may be affected by the Environmental Allowance Modifier (EAM) and the Trip Time Delay (TTD) (References 18 and 19) systems, which were developed to reduce the incidence of unnecessary feedwater related reactor trips.

The EAM system permits plant operation with a relatively low setpoint for the S/G Low-Low Water Level trip, which does not include the full environmental error allowance. The EAM will automatically enable a higher Low-Low level trip setpoint, which includes the full environmental error allowance, whenever an adverse containment environment is indicated by a rise in containment pressure.

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

The overtemperature ΔT setpoints shown in Figure 15.1.3-1 along with all other evaluated DNBR's were calculated assuming approximately 15 percent margin in the critical heat flux calculation, as discussed in Paragraph 4.4.2.1.

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. During preoperational start-up tests, it was demonstrated that actual instrument errors and time delays are equal to or less than the assumed values.

High and low power range neutron flux trip setpoints allow for a 2% calorimetric uncertainty and the installation of the LEFM effectively reduces the calorimetric uncertainty to 0.7%. As a result of the reduction in calorimetric uncertainty and equivalent upgrade in rated thermal power, the safety analysis high and low power range neutron flux trip setpoints will be redefined based on the new power level. The power range neutron flux (high setting) will be defined as 116.5% of 3455 MWt and the power range neutron flux (low setting) will be defined as 34.6% of 3455 MWt. Redefining these setpoints makes additional analysis unnecessary because the new values are equivalent, in terms of total megawatts, to the current licensing basis at 3411 MWt (116.5% of 3455 MWt = 118% of 3411 MWt and 34.6% of 3455 MWt = 35% of 3411 MWt).

The text of Chapter 15 contains references to high and low power range neutron flux trip setpoints of 118% and 35% of the original rated thermal power, respectively. Unless indicated otherwise, plots and results of transients that modeled these trips used the original setpoint definitions. In the interest of brevity, the references to the original setpoints were left unchanged since the new setpoints represent identical values in terms of total megawatts.

15.1.4 Instrumentation Drift And Calorimetric Error - Power Range Neutron Flux

The instrumentation drift and calorimetric errors used in establishing the maximum overpower setpoint are presented in References 17 and 27.

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis. The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generators and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow.

15.1.5 Rod Cluster Control Assembly Insertion Characteristic

The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod cluster control assemblies and the variation in rod worth as a function of rod position.

With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel. For accident analyses it is conservatively assumed that the insertion time to dashpot entry is 2.7 seconds. The rod cluster control assembly position versus time assumed in accident analyses is shown in Figure 15.1.5-1.

Figure 15.1.5-2 shows the fraction of total negative reactivity insertion for a core where the axial distribution is skewed to the lower region of the core. An axial distribution which is skewed to the lower region of the core can arise from a xenon oscillation or can be considered as representing a transient axial distribution which would exist after the rod cluster control assembly bank had already traveled some distance after trip. This lower curve is used as input to all point kinetics core models used in transient analyses.

There is inherent conservatism in the use of this curve in that it is based on a skewed distribution which would exist relatively infrequently. For cases other than those associated with xenon oscillations significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized rod cluster control assembly negative reactivity insertion versus time is shown in Figure 15.1.5-3. The curve shown in this figure was obtained from Figures 15.1.5-1 and 15.1.5-2. A total negative reactivity insertion following trip of 4 percent $\Delta k/k$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Table 4.3.2-3.

The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.1.5-3) is used in transient analyses. Where special analyses require use of three dimensional or axial one dimensional core models, the negative reactivity insertion resulting from reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of Figure 15.1.5-1 is used as code input.

15.1.6 Reactivity Coefficients

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Chapter 4.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses such as loss of reactor coolant from cracks or ruptures in the Reactor Coolant System do not depend on reactivity feedback effects. The values used are given in Table 15.1.2-2; reference is made in that table to Figure 15.1.6-1 which shows the upper and lower Doppler power coefficients, as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event by event basis. To facilitate comparison, individual sections in which justification for the use of large or small reactivity coefficient values is to be found are referenced below:

Condition II Events		Section
1.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition	15.2.1
2.	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power	15.2.2
3.	Rod Cluster Control Assembly Misalignment	15.2.3
4.	Uncontrolled Boron Dilution	15.2.4
5.	Partial Loss of Forced Reactor Coolant Flow	15.2.5
6.	Startup Of An Inactive Reactor Coolant Loop	15.2.6
7.	Loss of External Electrical Load And/Or Turbine Trip	15.2.7
8.	Loss of Normal Feedwater	15.2.8
9.	Loss Of All Off-Site Power To The Station Auxiliaries (Station Blackout)	15.2.9
10.	Excessive Heat Removal Due to Feedwater System Malfunctions	15.2.10
11.	Excessive Load Increase Incident	15.2.11
12.	Accidental Depressurization of the Reactor Coolant System	15.2.12
13.	Accidental Depressurization of Main Steam System	15.2.13
14.	Spurious Operation of the Safety Injection System At Power	15.2.14

Condition III Events

1.	Complete Loss of Forced Reactor Coolant Flow	15.3.4
2.	Single Rod Cluster Control Assembly Withdrawal, At Full Power	15.3.6

Condition IV Events

1.	Rupture of A Steam Pipe Rupture of A Feedwater Pipe	15.4.2.1 15.4.2.2
2.	Single Reactor Coolant Pump Locked Rotor	15.4.4
3.	Rupture Of A Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	15.4.6

15.1.7 Fission Product Inventories

15.1.7.1 Activities in the Core

The design basis LOCA source terms are based on an average 1000 EFPD reactor core with an enrichment of 5% U235. The reactor core inventory analysis was performed with the ORIGIN-S computer code (Ref. 19a) by Oak Ridge National Laboratory. The fraction of reactor core releases into the containment are based on TID-14844 (Ref. 2) methodology which consists of an instantaneous release of 100% of the noble gases, 50% of the halogens, and 1% of the solids in the fission product inventory. These isotopes are given in Table 15.5.3.8. The subset of these isotopes which are important from a health hazards point of view are given in Table 15.5.3-5. The isotopes included in Table 15.5.3-5 are the isotopes controlling from considerations of inhalation dose (iodines) and from external dose due to immersion (noble gases).

15.1.7.2 Activities in the Fuel Pellet Cladding Gap

The computed gap activities (Table 15.1.7-1) are based on buildup in the fuel from the fission process and diffusion to the gap at rates dependent on the operating temperature. The temperature dependence is accounted for by determining the core fuel fraction operating within each of nine temperature regions (Table 15.1.7- 2), each with a release rate to the gap dependent of the mean fuel temperature within that region. Since the temperature distribution changes during core life, the highest expected values are used. The temperature dependence of the diffusion coefficient, D′, for Xe and Kr in UO_2, follows the Arrhenius law:

$$D'(T) = D'(1673)\exp - \left[\frac{E}{R}\left(\frac{1}{T} - \frac{1}{1673}\right)\right]$$

where

D′(T) = diffusion coefficient at temperature T, sec^{-1}

D′(1673) = 1×10^{-11} sec^{-1}, diffusion coefficient at 1673°K

E	=	82 kilocalories/mole, activation energy
R	=	1.99×10^{-3} kilocalories/mole - °K, gas constant
T	=	temperature, °K

The above expression is valid for temperatures above 1100°C. Below 1100°C fission gas release occurs mainly by two temperature independent mechanisms, recoil and knock out, and is predicted by using D' at 1100°C. The value used for D' (1673°K), based on data at burnups greater than 10^{19} fissions/cc, accounts for possible fission gas release by other mechanisms as well as pellet cracking during irradiation.

The diffusion coefficient for iodine isotopes was conservatively assumed to be the same as for Xe and Kr. Toner and Scott (Reference 4) observed that iodine diffuses in UO_2 at about the same rate as Xe and Kr and has about the same activation energy. Data reported by Belle (Reference 5) indicate that the iodine diffuses at slightly slower rates than Xe and Kr.

With the diffusion coefficient determined for the fuel temperature region of interest, the fraction of radioactive fission gas which crosses the fuel boundary into the fuel rod gap is found from:

$$f = 3\sqrt{\frac{D'}{\lambda}} \quad \mathrm{Coth}\sqrt{\frac{\lambda}{D'}} - \sqrt{\frac{D'}{\lambda}}$$

Where:

f	=	fraction of a given radioactive fission gas in fuel rod gap
λ	=	fission gas decay constant, sec^{-1}
D'	=	diffusion coefficient, Sec^{-1}

The above expression is the steady-state solution of the diffusion equation in spherical geometry as given by Booth (Reference 6).

Table 15.1.7-1 lists the total core activities as well as activities present in the gap for each pertinent isotope obtained using the above equations and the fuel temperature distribution given in Table 15.1.7-2.

The activities in the reactor coolant, as well as in the volume control tank, pressurizer and gaseous waste processing system, are given in Chapter 11 including the data on which the computation of these activities are based.

15.1.8 Residual Decay Heat

Residual heat in a subcritical core consists of:

1. Fission product decay energy,

2. Decay of neutron capture products, and

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

These constituents are discussed separately in the following paragraphs.

15.1.8.1 Fission Product Decay

For short times (< 10^3 seconds) after shutdown, data on yields of short half life isotopes is sparse. Very little experimental data is available for the γ-ray contributions and even less for the β-ray contribution. Several authors have compiled the available data into a conservative estimate of fission product decay energy for short times after shutdown, notably Shure (Reference 7), Dudziak (Reference 8), and Teage. Of these three selections, Shure's curve is the highest and it is based on the data of Stehn and Clancy (Reference 9) and Obenshain and Foderaro (Reference 10).

The fission product contribution to decay heat which has been assumed in the accident analyses is the curve of Shure increased by 20 percent for conservatism. This curve with the 20 percent factor included is shown in Figure 15.1.8-1.

15.1.8.2 Decay of U-238 Capture Products

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

$$P_1/P_o = \frac{E_{\gamma 1} + E_{\beta 1}}{200\ \text{Mev}} \quad c(1+a)\, e^{-\lambda_1 t} \quad \text{watts/watts}$$

$$P_2/P_o = \frac{E_{\gamma 2} + E_{\beta 2}}{200\text{Mev}}\, c(1+a) \quad \left[\frac{\lambda_2}{\lambda_1 - \lambda_2} \left(e^{-\lambda_2 t} - e^{-\lambda_1 t}\right) + e^{-\lambda_2 t} \quad \text{watts / watts} \right]$$

Where:

P_1/P_o is the energy from U-239 decay

P_2/P_o is the energy from Np-239 decay

t is the time after shutdown (seconds)

c (1+a) is the ratio of U-238 captures to total fissions = 0.6(1+.2)

λ_1 = the decay constant of U-239 = 4.91 x 10^{-4} seconds $^{-1}$

λ_2 = the decay constant of Np-239 decay = 3.41 x 10^{-6} seconds $^{-1}$

$E_{\gamma 1}$ = total γ-ray energy from U-239 decay = .06 Mev

$E_{\gamma 2}$ = total γ-ray energy from Np-239 decay = .30 Mev

$E_{ß1}$ = total ß-ray energy from U-239 decay = 1/3 x 1.18 Mev

$E_{ß2}$ = total ß-ray energy from Np-239 decay = 1/3 x 0.43 Mev

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

This expression with a margin of 10 percent is shown in Figure 15.1.8-1. The 10 percent margin, compared to 20 percent for fission product decay, is justified by the availability of the basic data required for this analysis. The decay of other isotopes, produced by neutron reactions other than fissions, is neglected.

15.1.8.3 Residual Fissions

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

For the purpose of illustration only a one delay neutron group calculation, with a constant shutdown reactivity of -4 percent $\Delta k/k$, is shown in Figure 15.1.8-1.

15.1.8.4 Decay Heat Following Loss of Coolant Accident

For a large break loss-of-coolant accident the core is rapadily shut down by void formation such that heat generation comes from fission product decay. The decay heat assumed by the analysis is based on the 1979 ANSI/ANS standard with an infinite operating time at full power. For each case, the decay heat value is randomly sampled within (plus or minus) one standard deviation. For the S-RELAP5 calculation, the heat is conservatively assumed to be generated within the fuel pellet.

15.1.9 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, very specialized codes in which the modeling has been developed to simulate one given accident, which consequently has a direct bearing on the analysis of the accident itself, are either summarized or referenced in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.1.2-2.

15.1.9.1 FACTRAN

FACTRAN 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, density). The code uses a fuel model which exhibits the following features simultaneously:

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

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

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

The gap heat transfer coefficient is calculated according to an elastic pellet model (refer to Figure 15.1.9-1). The thermal expansion of the pellet is calculated as the sum of the radial (one-dimensional) expansions of the rings. Each ring is assumed to expand freely. The cladding diameter is calculated based on thermal expansion and internal and external pressures.

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

FACTRAN is further discussed in Reference 11.

15.1.9.2 MARVEL

The MARVEL code is used to determine the detailed transient behavior of multi-loop pressurized water reactor systems caused by prescribed initial perturbations in process parameters. The code is useful in predicting plant behavior when different conditions are present in the loops. For analytical purposes, the physical, thermal and hydraulic characteristics of a multi-loop plant are represented by two "equivalent" loops. The perturbation is considered to occur in one of the equivalent loops which may represent one or more physical loops. The other equivalent loop thus represents in lumped form, the remaining loops in the plant.

The code simulates the coolant flow through the reactor vessel, hot leg, cold leg, steam generator plus the pressurizer surge line. Neutron kinetics, fuel-clad heat transfer and the rod control system characteristics are modeled. Simulation of the Reactor Trip System, Engineered Safety Features (Safety Injection) and Chemical and Volume Control System is provided.

MARVEL also has the capability of calculating the transient value of DNB ratio based on the input from the core limits illustrated on Figure 15.1.3-1. The core limits represent the minimum value of DNBR as calculated for a typical or thimble cell.

MARVEL is further discussed in Reference 12.

15.1.9.3 LOFTRAN

The LOFTRAN program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multi-loop system by a lumped parameter single loop 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 neutron flux, overpower and overtemperature reactor coolant delta-T, high and low pressurizer pressure, low RCS flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. The Safety Injection System including the accumulators is also modeled.

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

LOFTRAN also has the capability of calculating the transient value of DNB ratio based on the input from the core limits illustrated on Figure 15.1.3-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell. LOFTRAN is further discussed in Reference 13.

15.1.9.4 LEOPARD

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

LEOPARD is further described in Reference 14.

15.1.9.5 TURTLE

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

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

TURTLE is further described in Reference 15.

15.1.9.6 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady-state codes presently used for reactor core design. The code uses an implicit

finite-difference method to solve the two-group transient neutron diffusion equations in one, two and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise doppler and moderator feedback effects. The code handles up to 2000 spatial points, and performs its own steady state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits provide channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, fuel temperatures, and so on.

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

TWINKLE is further described in Reference 16.

15.1.9.7 THINC

The THINC code is described in Section 4.4.3.1.

15.1.9.8 RELAP5/MOD2-B&W

RELAP5/MOD2-B&W is a BWNT adaptation of the Idaho National Engineering Laboratory code RELAP5/MOD2. The code, developed for best-estimate transient simulation of pressurized water reactors, has been modified to include models required for licensing analysis. Modeling capabilities are associated with the analysis of large and small break LOCAs, as well as operational transients defining the safety envelope of a reactor. The latter class of transients include Anticipated Transient Without Scram, Loss of Offsite Power, Loss of Feedwater, and Loss of RCS Flow transients. The code has been benchmarked extensively to existing experimental data for regulatory approval of its use in analyzing LOCA and Non-LOCA transients. RELAP5/MOD2-B&W is documented in topical BAW-10164 (Reference 20).

15.1.9.9 S-RELAP5

This code is used for the system calculation for analysis of large break LOCAs. The field equations are basically the same form as RELAP5/MOD2 with the addition of full two-dimensional momentum equations. This two dimensional capability is only applied within the reactor vessel in the Realistic Large Break LOCA methodology, but can be applied anywhere in the reactor coolant system through input. The S-RELAP5 code structure was modified to be essentially the same as RELAP5/MOD3. The coding for reactor kinetics, control systems, and trip systems was also replaced from RELAP5/MOD3. Initial fuel conditions are supplied by the realistic fuel performance code, RODEX3A. To be consistent, the fuel deformation and conductivity models from RODEX3A were included in S-RELAP5. Capability to interface with a concurrent calculation of containment backpressure using the ICECON code was added. S-RELAP5 is documented in topical report EMF-2100 (P) (Reference 21).

15.1.9.10 RODEX3A

RODEX3A calculates fuel rod performance for Realistic Large Break LOCA analysis. In particular, the initial operating temperature of the fuel pellets (as stored energy) and the internal fuel rod gas pressure are provided as functions of fuel exposure and power history. RODEX3A is documented in topical report ANF-90-145(P)(A) (Reference 22).

15.1.9.11 LYNXT

LYNXT is approved by the NRC and provides the capability for single-pass core thermal-hydraulic analysis for both steady state and transient conditions. It also has the capability to analyze

conditions with high lateral flow and/or recirculating flow, such as encountered in the analysis of a steamline break with reactor coolant pumps off. The single pass LYNXT model has been extensively benchmarked to multi-pass analyses and appropriate experimental data. LYNXT is used almost exclusively for determining core flow redistribution and for predicting the DNB performance of various fuel designs.

LYNXT has been qualified for the BWC, BWCMV, BWCMV-A, B&W2 and W3 correlations by data base analysis. In each case, where this evaluation has been performed, LYNXT supported the licensed DNBR limit for the respective CHF correlation. Some of the features of LYNXT include:

1) Reverse/recirculating flow option
2) Exit pressure profile boundary condition and transient pressure drop boundary condition
3) Generalized DNBR subroutine
4) Internal code generation of the axial power shape
5) Transient radial and axial power shapes input capability
6) Dynamic gap conductance fuel model
7) ANSI Fortran 77 and self-contained
8) Enhancements to the conducting wall model to allow rectangular and cylindrical walls

LYNXT is described in topical report BAW-10156 (Reference 23).

15.1.9.12 TACO3

The TACO3 code, with its Fuel Rod Gas Pressure Criteria, is a state-of-the-art methodology for fuel rod thermal performance analysis. This package applies to fuel rod burnups to 62,000 MWd/mtU, with possible extrapolation to 65,000 MWd/mtU. The TACO3 fuel performance code is a major evolution in the prediction of fuel rod performance. TACO3 uses best-estimate models benchmarked to an extensive data base of fuel performance data from numerous industry sponsored experimental programs. TACO3 uses a complete set of new thermal and mechanical models, as well as new fuel and cladding material relations. Several models represent advances in the state-of-the-art. The TACO3 fuel temperature predictions have less uncertainty than other comparable codes. The NRC has reviewed and approved TACO3. TACO3 predicts the following as a function of burnup:

- Centerline Fuel Melt
- Fuel Rod Internal Gas Pressure
- LOCA Analysis Initialization Parameters
- Cladding Strain
- Creep Collapse Analysis Initialization Parameters

TACO3 uses best-estimate inputs to provide best-estimate predictions. Statistical evaluations are performed to estimate uncertainties and provide conservative results for use in licensing evaluations. Code and power prediction uncertainties and manufacturing variations are considered for internal gas pressure uncertainties. Statistical parameters obtained from the analysis of an extensive code benchmark database evaluate fuel temperature uncertainties. Transient fission gas release and cladding oxide effects are also represented to provide appropriate conservatism. TACO3 is described in topical report BAW-10162 (Reference 24).

15.1.9.13 NEMO (BAW-10180-A)

NEMO is a nodal neutronics code used to calculate power distributions and perform reactivity analyses of pressurized water reactors (PWRs). The nodal balance equation is solved to determine the neutron flux, source, relative power density (which includes pin power reconstruction to obtain detailed pin power profiles), and reactivity of the core. NEMO employs a two-group nodal expansion method to determine the currents and fluxes at the surface of each node in the core. Two or three dimensional problems can be analyzed with thermal-hydraulic feedback and isotopic depletion. Microscopic cross sections are required for the isotopic depletion calculation and are obtained from pre-calculated cross section database files. Interpolation of these cross section tables, one for each fuel enrichment/burnable poison combination, is performed versus a six-dimensional space of independent variables: burnup, boron, xenon, moderator specific volume, fuel temperature, and a spectral parameter.

NEMO has been applied to a wide array of problem solutions, including: development of reload fuel assembly loading patterns, calculation of startup physics control rod worths and reactivity coefficients and defects, core maneuvering analyses, core follow, provide input parameters to safety analysis evaluation, and generation of neutron flux signal to power factors for online measurement systems. A detailed description of NEMO is given in topical report BAW-10180 (Reference 25).

15.1.10 Loss Of One (Redundant) DC System

15.1.10.1 Identification of Causes

The plant DC System serves as a power source for DC pump motors, controls, and instrumentation. A description of this system and its redundant design are presented in Subsection 8.3.2. The loss of one DC System will be defined for the purposes of this analysis as the loss of one battery and one battery charger.

15.1.10.2 Analysis of Effects and Consequences

As discussed in Subsection 8.3.2, the plant has been designed so that the loss of one DC System will not affect the safe operation of the plant.

15.1.11 References

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

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TABLE 15.1.2-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Guaranteed Core Thermal Power	3455 MWt**
Thermal power generated by the reactor coolant pumps	12 MWt
Guaranteed Nuclear Steam Supply System thermal power output (Core Thermal Power + thermal power guaranteed from RCPs)	3467 MWt**
The Engineered Safety Features design rating (maximum calculated turbine rating)*	3577 MWt

* See Westinghouse Letter TVA-97-078 (dated July 24, 1997) for explanation of NSSS power ratings in FSAR Chapter 15.

** See Section 15.0 for a discussion of the 1.3% power level upgrade. Rated thermal power was increased from 3411 MWt to 3455 MWt.

TABLE 15.1.2-2 (Sheet 1)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR[1] TEMPERATURE (Δk/°F)	MODERATOR[1] DENSITY (Δk/gm/cc)	DOPPLER [2]	INITIAL CORE THERMAL POWER ASSUMED (MWT)
CONDITION II					
Uncontrolled RCC Assembly Bank Withdrawal from a Subcritical Condition	RELAP5/MOD2 - B&W LYNXT	Opcm/°F	--	Lower	0
Uncontrolled RCC Assembly Bank With-drawal at Power	RELAP5/MOD2 - B&W LYNXT	See Section 15.2.2.2		Lower and upper	3479
RCC Assembly Misalignment	THINC, TURTLE LOFTRAN	-	0	Upper	3411[3]
Uncontrolled Boron Dilution	NA	NA	NA	NA	NA
Partial Loss of Forced Reactor Coolant Flow	LOFTRAN, THINC FACTRAN	-	0	Upper	3479
Start-up of an Inactive Reactor Coolant Pump	LOFTRAN, FACTRAN,THINC	-	0.43	Lower	2456
Loss of External Electrical Load and/or Turbine Trip	RELAP5/MOD-2 - B&W	+7.0 pcm/°F		Lower	3479 and 1774
Loss of Normal Feedwater	RELAP5/MOD-2 - B&W	-	NA	NA	3479
Loss of Off-Site Power to the Plant Auxiliaries (Plant Blackout)	RELAP5/MOD-2 - B&W	-	NA	NA	3479
Excessive Heat Removal Due to Feedwater System Malfunctions	RELAP5/MOD-2 - B&W LYNXT	-45pcm/°F	---	Lower	0 and 3479

TABLE 15.1.2-2 (Sheet 2)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR[1] TEMPERATURE (Δk/°F)	MODERATOR[1] DENSITY (Δk/gm/cc)	DOPPLER [2]	INITIAL CORE THERMAL POWER ASSUMED (MWT)
Excessive Load Increase	LOFTRAN	-	0 and 0.43	Lower	3491
Accidental Depressurization of the Reactor Coolant System	LOFTRAN	-	0	Upper	3479
Accidental Depressurization of the Main Steam System	LOFTRAN	-	Function of Moderator Density See Subsection 15.2.13 (Figure 15.2.13-1)	-2.9 pcm/PF	0 (Subcritical)
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	-	0	Lower	3494
CONDITION III					
Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipe which Actuate Emergency Core Cooling	RELAP5/MOD2 - B&W				3479
Inadvertent Loading of a Fuel Assembly into an Improper Position	LEOPARD, TURTLE	-	NA	NA	3411[3]
Complete Loss of Forced Reactor Flow	RELAP5/MOD2 - B&W LYNXT	+7.0 pcm/°F	-	Lower	3479
Waste Gas Decay Tank Rupture	NA	-	NA	NA	3582
Single RCC Assembly Withdrawal at Full Power	TURTLE, THINC LEOPARD	-	NA	NA	3411[3]

TABLE 15.1.2-2 (Sheet 3)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR[1] TEMPERATURE (Δk/°F)	MODERATOR[1] DENSITY (Δk/gm/cc)	DOPPLER [2]	INITIAL CORE THERMAL POWER ASSUMED (MWT)
CONDITION IV					
Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the Reactor Coolant System (Loss of Coolant Accident)	S-RELAP5 RODEX3A	Function of Moderator Density. See Subsection 15.4.1		Function of Fuel Temp. See Subsection 15.4.1	3479
Major secondary system pipe rup-ture up to and including double-ended rupture (Rupture of a Steam Pipe)	RELAP5/MOD2 - B&W LYNXT,NEMO	Function of Moderator Density See Subsection 15.4.2 (Figure 15.4.2-1)		-2.9 pcm/°F	0 (Critical)
Steam Generator Tube Rupture	NA	NA	NA	NA	3479
Single Reactor Coolant Pump Locked Rotor	RELAP5/MOD2 - B&W LYNXT	+7.0 pcm/°F		Lower	3479
Fuel Handling Accident	NA	NA		NA	NA
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN	+5.2pcm/°F BOL -23pcm/°F EOL (Isothermal Temperature Coefficent)	----- See Table 15.4.6-1	Least negative Doppler defect.	0 and 3479

Notes:

(1) Only one is used in an analysis i.e. either moderator temperature or moderator density coefficient.
(2) Reference Figure 15.1.6-1
(3) These events, performed at a rated thermal power of 3411 MWt, have been evaluated in Reference 26. The evaluations demonstrate that the analysis of record for each of these events continue to be applicable to the uprated power of 3455 MWt.

TABLE 15.1.2-2 (Sheet 4)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED MODERATOR(1) TEMPERATURE (Δk/°F)	MODERATOR(1) DENSITY (Δk/gm/cc)	DOPPLER (2)	INITIAL CORE THERMAL POWER ASSUMED (MWT)
Fuel Handling Accident	NA	NA		NA	NA
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE, FACTRAN	+5.2pcm/°F BOL -23pcm/°F EOL (Isothermal Temperature Coefficent)	----- See Table 15.4.6-1	Least negative Doppler defect.	0 and 3479

Notes:

(1) Only one is used in an analysis i.e. either moderator temperature or moderator density coefficient.
(2) Reference Figure 15.1.6-1
(3) These events, performed at a rated thermal power of 3411 MWt, have been evaluated in Reference 26. The evaluations demonstrate that the analysis of record for each of these events continue to be applicable to the uprated power of 3455 MWt.

TABLE 15.1.3-1 (Sheet 1)

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay (Seconds)
Power Range High Neutron Flux, High Setting	116.5%*	0.5
P-8 (Three-loop operation)	84%	0.5
Power Range High Neutron Flux, Low Setting	34.6%*	0.5
Overtemperature ΔT	Variable, see Figure 15.1.3-1	8.0**
Overpower ΔT	Variable, see Figure 15.1.3-1	8.0**
High pressurizer pressure	2445 psig	2.0
Low pressurizer pressure	1845 psig***	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage Trip (17x17)	68% nominal	1.2
Turbine Trip	Not applicable	1.0

* These values were adjusted for the 1.3% power level upgrade to coincide, in terms of megawatts, with the setpoints for the original rated thermal power level.

** 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.

***Except Subsection 15.2.14, Spurious Operation of the Safety Injection System at Power, which uses 1760 psig.

TABLE 15.1.3-1 (Sheet 2)
(Continued)

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay (Seconds)
Low-Low steam generator level	0% of narrow range level span	2.0 + TTD****
High steam generator level trip of the feedwater pumps and closure of feedwater system valves, and turbine trip	93% of narrow range level span	2.0

**** The Trip Time Delay (TTD) is applicable only below 50% RTP.

TABLE 15.1.7-1

CORE AND GAP ACTIVITIES
BASED ON FULL POWER OPERATION FOR 1000 DAYS
FULL POWER: 3582 MWt

Isotope	Curies in Core ($x 10^7$)*	Percent of Core Activity in Gap	Curies in Gap ($x 10^5$)
I-131	9.449	0.822	7.767
I-132	13.851	0.0901	1.248
I-133	19.500	0.271	5.284
I-134	21.708	0.0557	1.209
I-135	18.616	0.154	2.866
Xe-131m	0.104	1.0	0.104
Xe-133	19.145	0.667	12.770
Xe-133m	0.615	0.437	0.269
Xe-135	6.426	0.180	1.156
Xe-135m	4.053	0.0303	0.122
Xe-138	16.675	0.0316	0.526
Kr-83m	1.150	0.0824	0.094
Kr-85	0.103	16.7	1.724
Kr-85m	2.393	0.124	0.296
Kr-87	4.805	0.0668	0.321
Kr-88	6.658	0.0988	0.657
Kr-89	8.279	0.0137	0.113

* These activities are derived from TVA Calculation SQN-APS3-067 (Ref. 16., section 15.5.8) based on 193 nuclear fuel assemblies.

TABLE 15.1.7-2

CORE TEMPERATURE DISTRIBUTION

Percent of Core Fuel Within Given Temperature Range	Power, MWt	Fuel Temperature Range, °F
0.0	.1961	>3400
0.1	3.1373	3400 - 3200
0.3	10.3922	3200 - 3000
0.7	25.1	3000 - 2800
1.6	58.333	2800 - 2600
2.9	104.61	2600 - 2400
4.3	152.55	2400 - 2200
5.9	211.275	2200 - 2000
84.1	2999.02	<2000

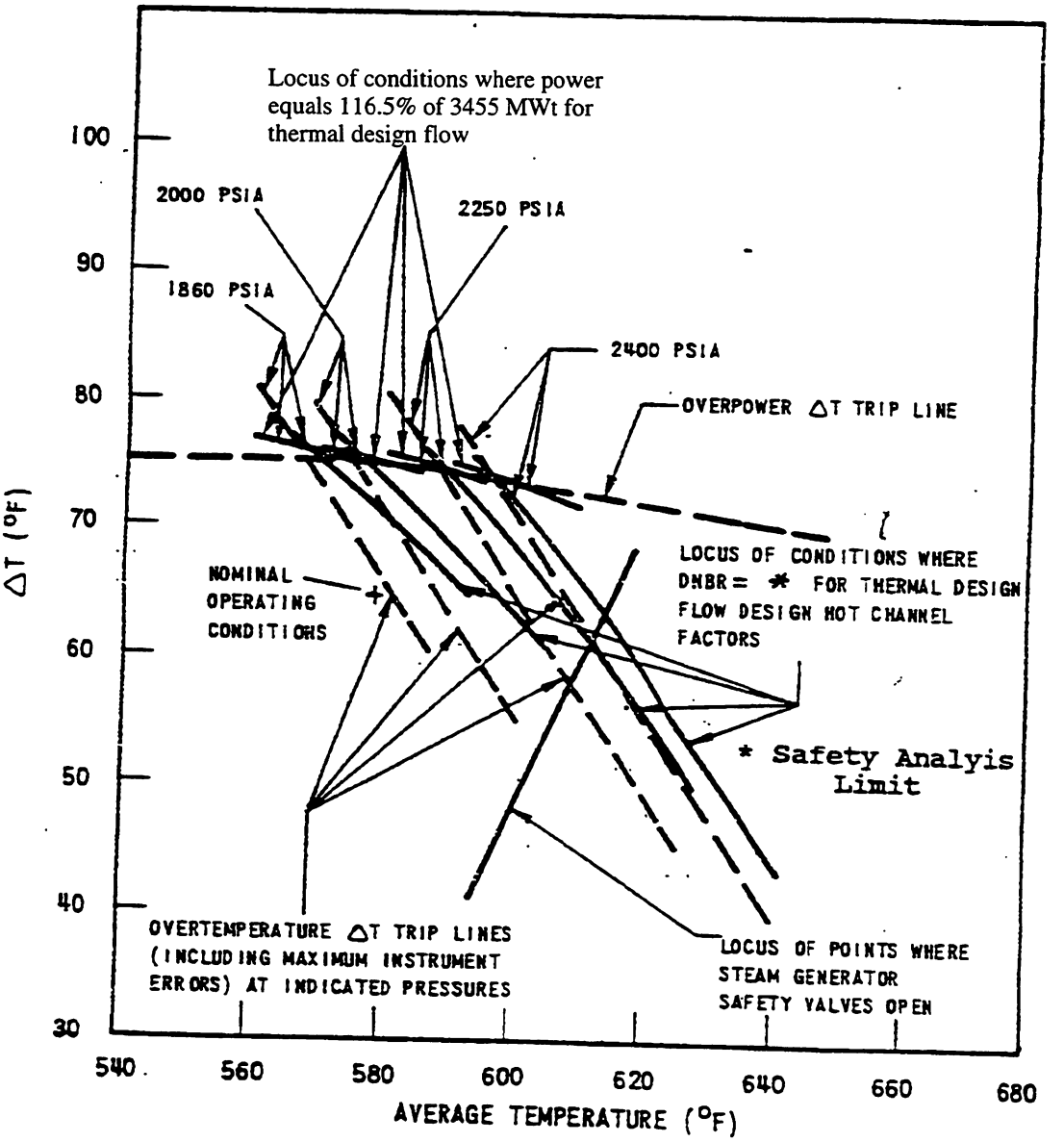


Figure 15.1.3-1 Illustration of Overtemperature and Overpower ΔT Protection

Revised by Amendment 17

Figure 15.1.5–1

RCCA POSITION vs. TIME ON REACTOR TRIP

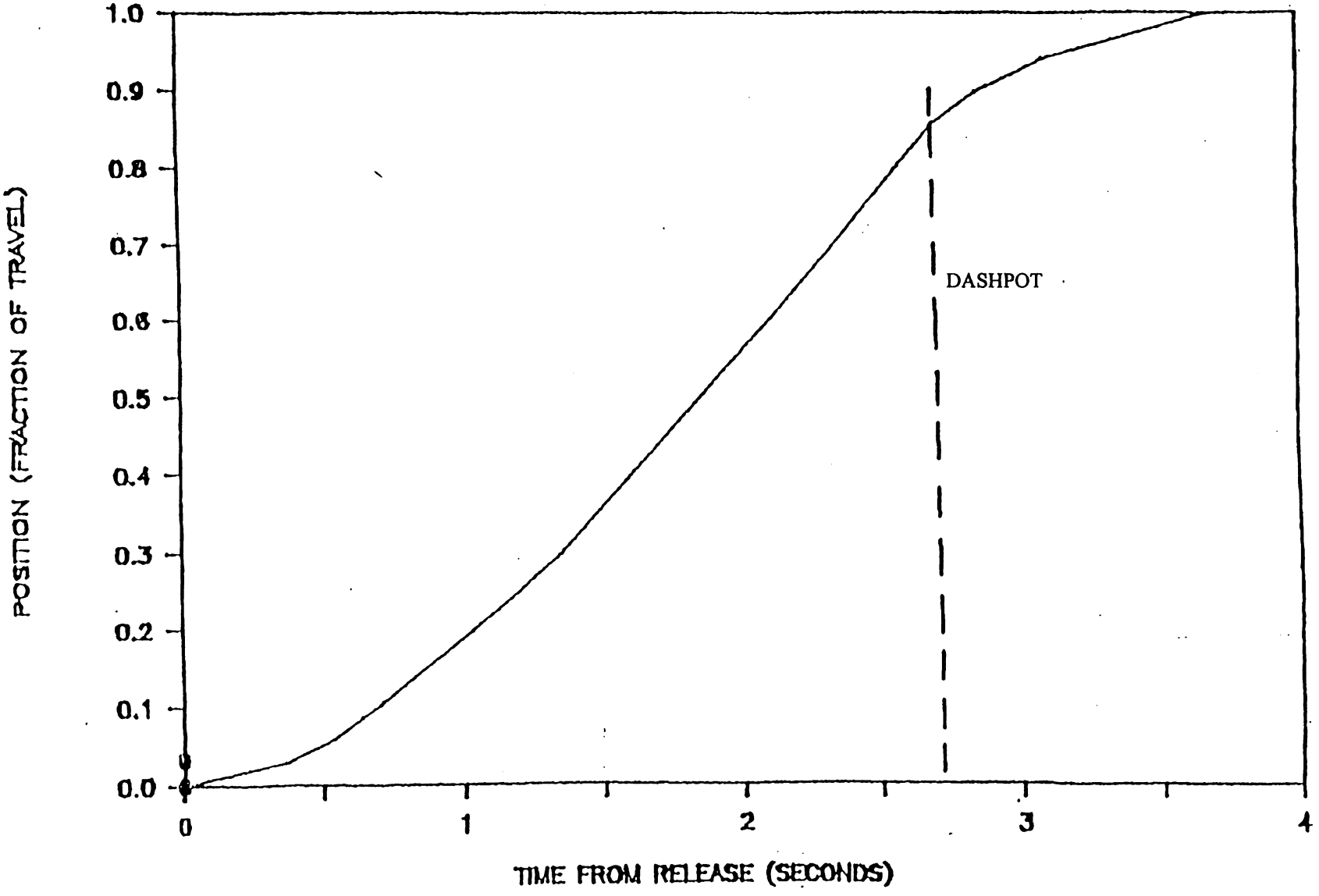


Figure 15.1.5–2

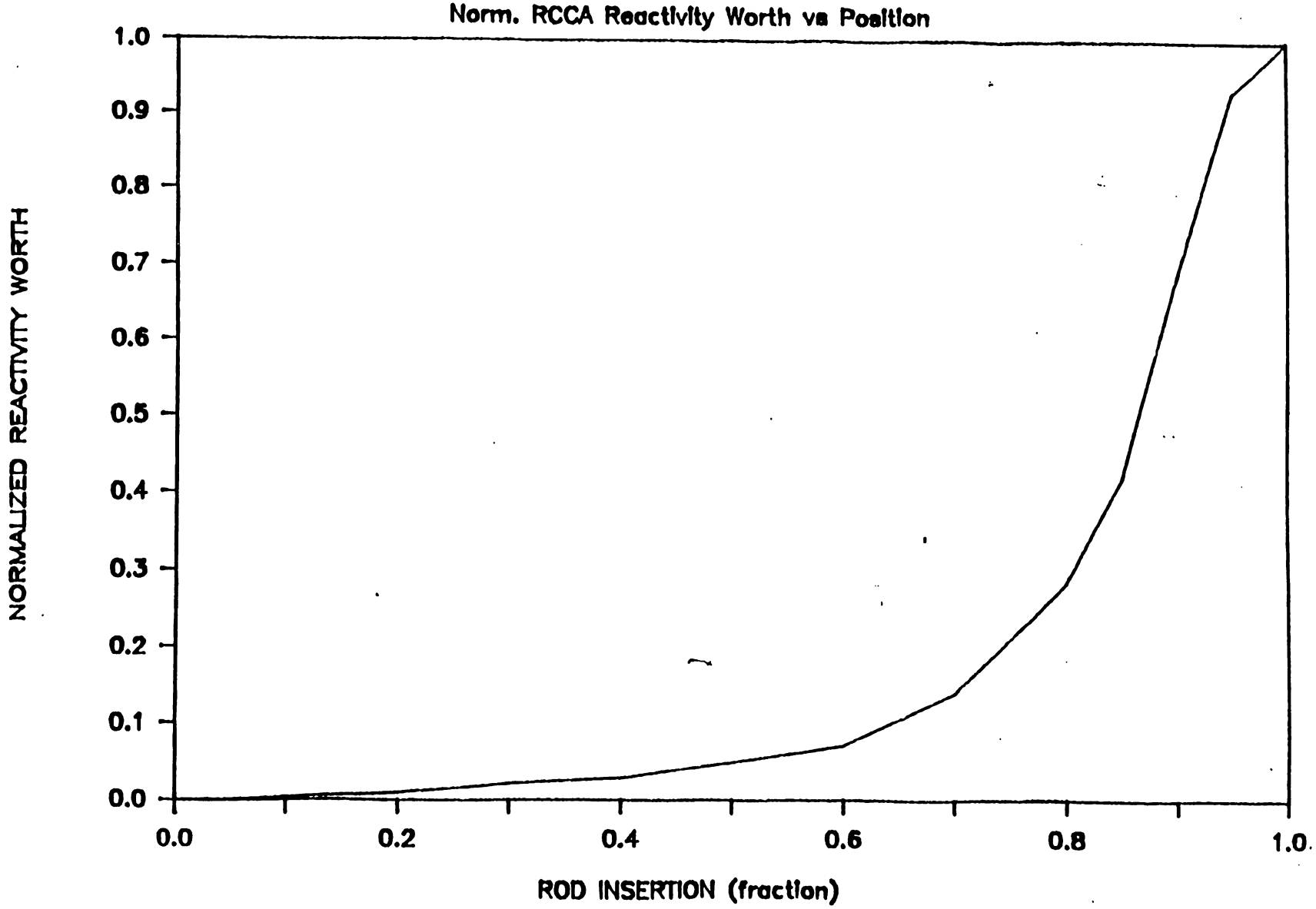
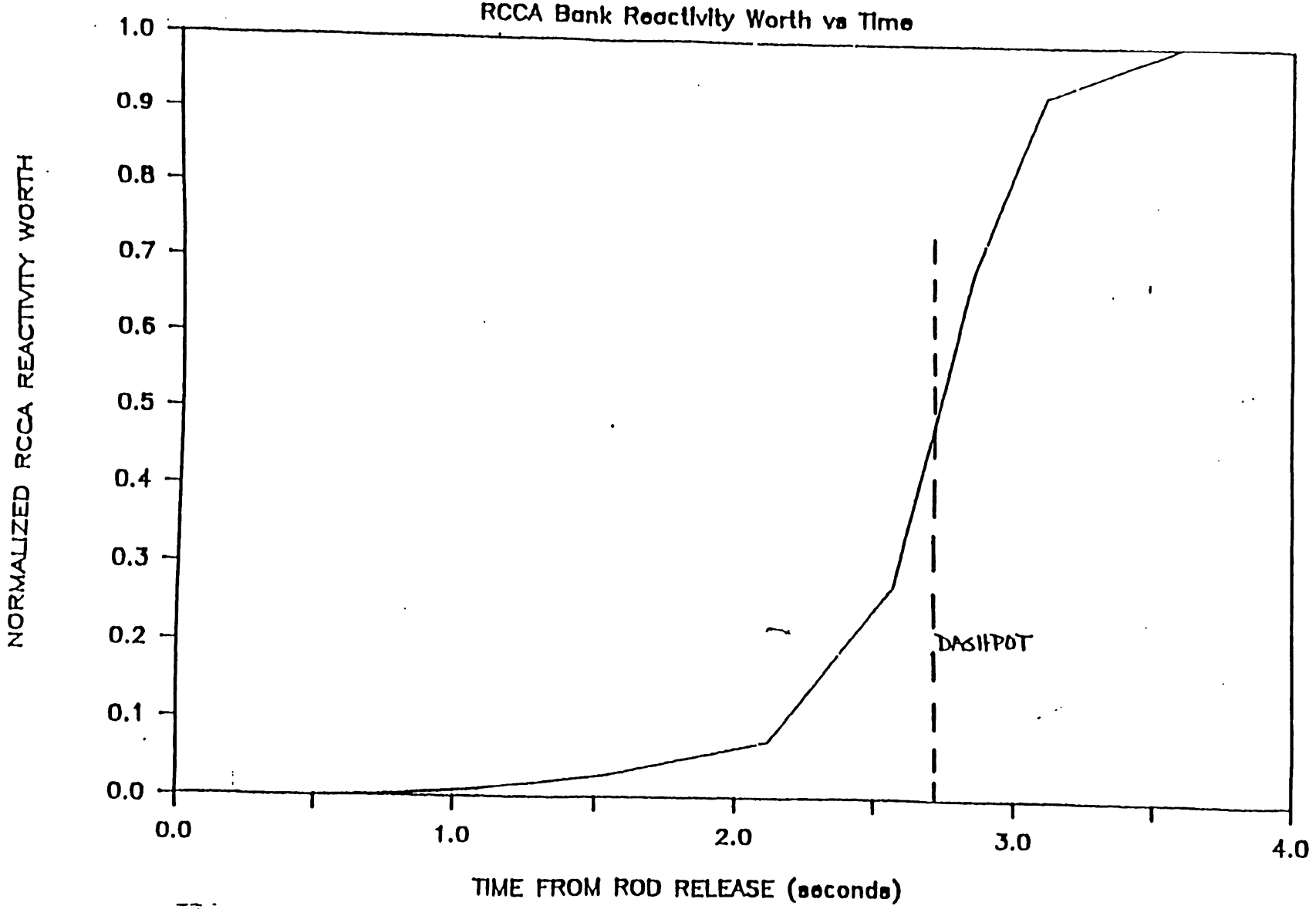


Figure 15.1.5–3

RCCA Bank Reactivity Worth vs Time



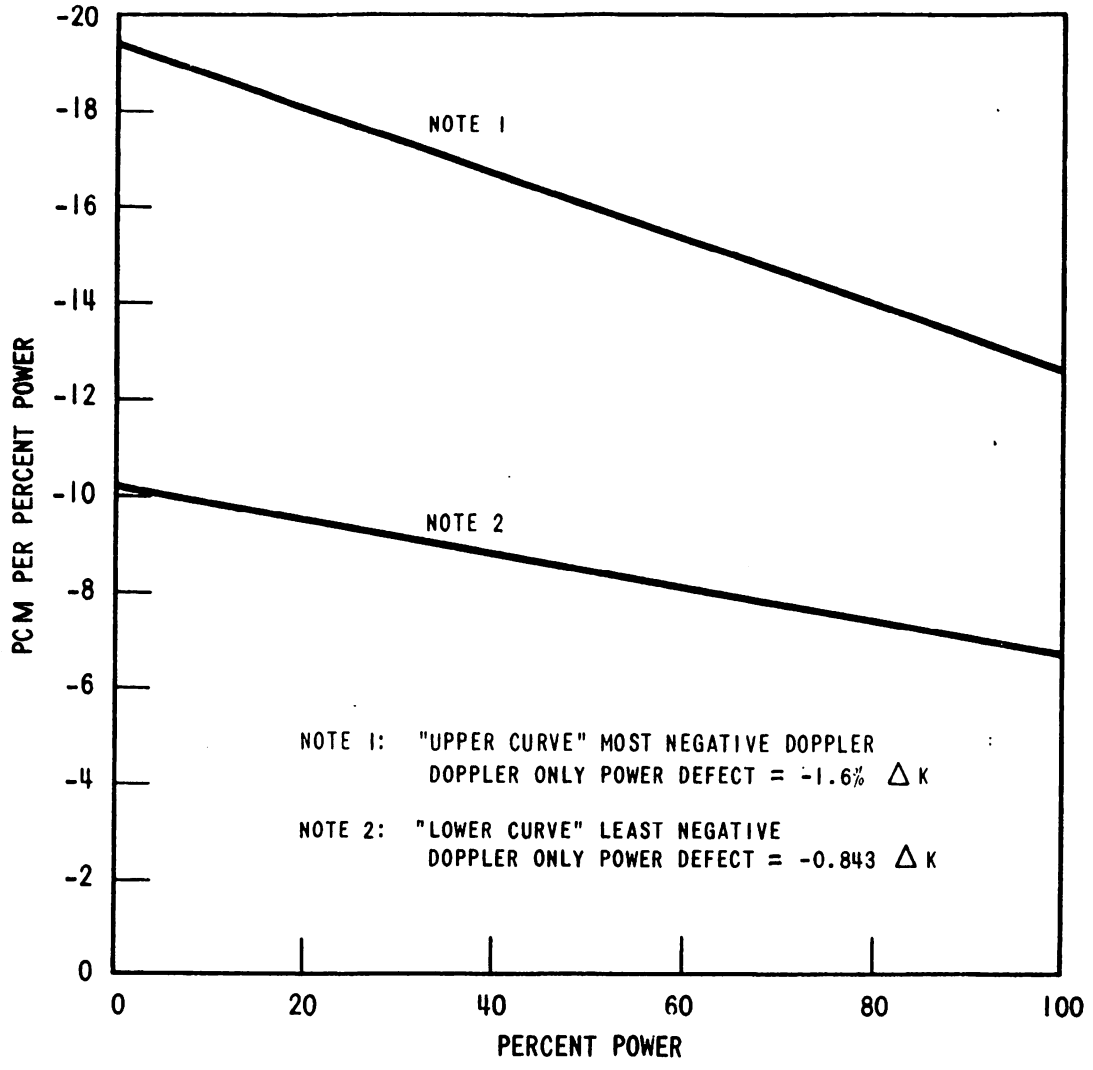


Figure 15.1.6-1 **Doppler Power Coefficient Used in Accident Analysis** | 1

Revised by Amendment 1

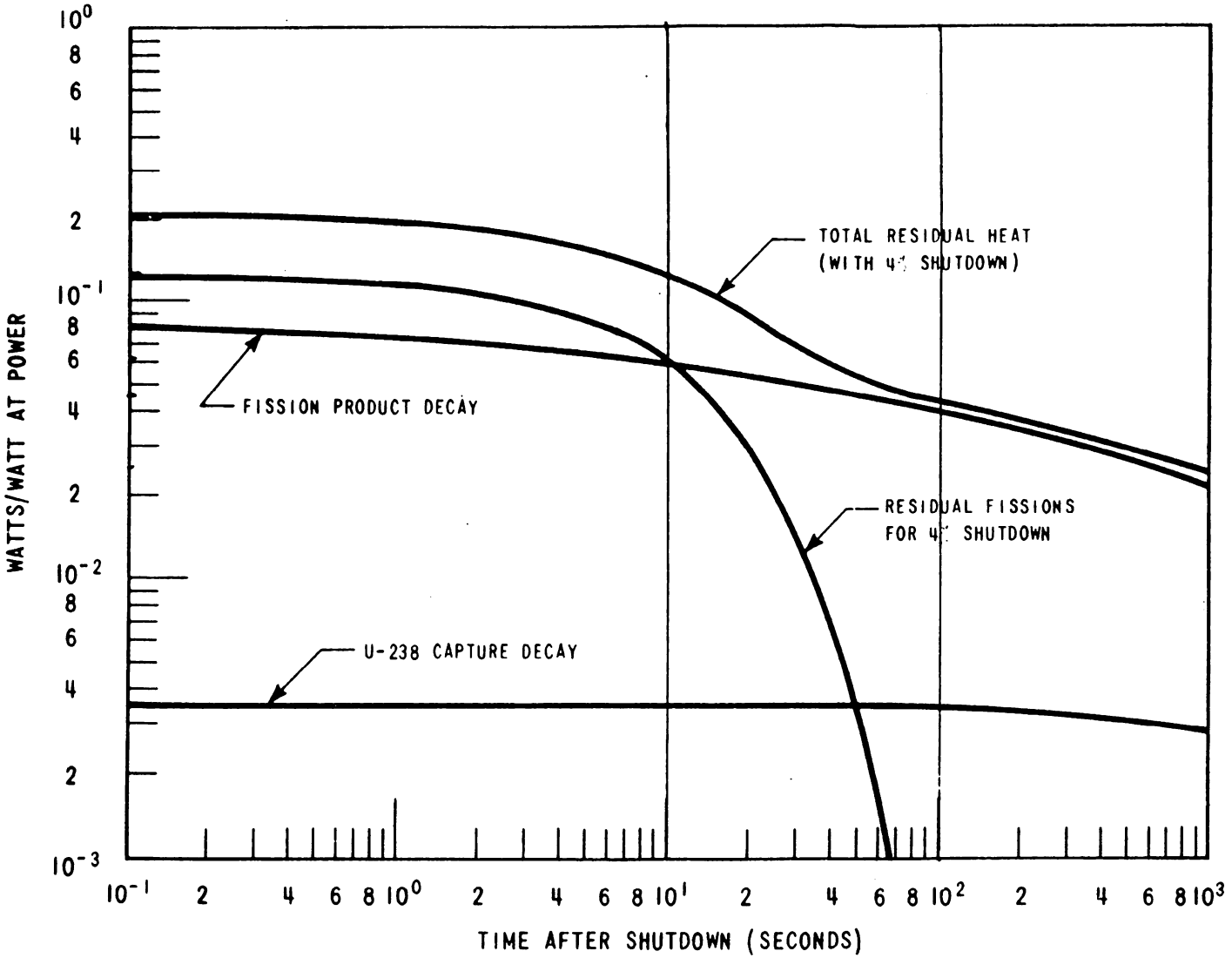


Figure 15.1.8-1 Residual Decay Heat

Revised by Amendment 1

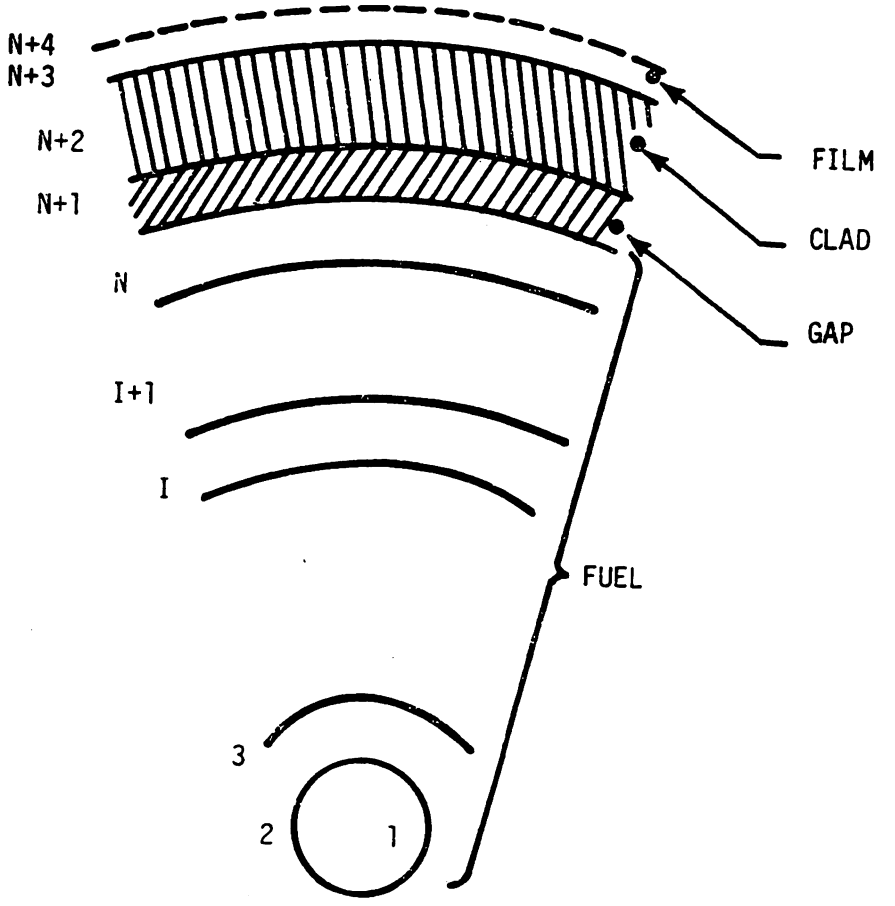


Figure 15.1.9-1 Fuel Rod Cross Section

15.2 CONDITION II - FAULTS OF MODERATE FREQUENCY

These faults at worst result in the reactor shutdown with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV category. In addition, Condition II events are not expected to result in fuel rod failures or Reactor Coolant System overpressurization.

For the purposes of this report the following faults have been grouped into this category:

1. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition.
2. Uncontrolled rod cluster control assembly bank withdrawal at power.
3. Rod cluster control assembly misalignment.
4. Uncontrolled boron dilution.
5. Partial loss of forced reactor coolant flow.
6. Start-up of an inactive reactor coolant loop.
7. Loss of external electrical load and/or turbine trip.
8. Loss of normal feedwater.
9. Loss of offsite power to the station auxiliaries.
10. Excessive heat removal due to feedwater system malfunctions.
11. Excessive load increase.
12. Accidental depressurization of the Reactor Coolant System.
13. Accidental depressurization of the Main Steam System.
14. Spurious Operation of Safety Injection System at power.

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

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

Hence, because of the high reliability of the protection system no special provision is proposed to be taken in the design to cope with the consequences of Condition II events without trip.

15.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition

15.2.1.1 Identification of Causes and Accident Description

A rod cluster control assembly withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of rod cluster control assemblies resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical, at hot zero power or at power. The "at power" case is discussed in Subsection 15.2.2.

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

The rod cluster control assembly drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the assemblies from being 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. The rod cluster control assembly 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 the two control banks having the maximum combined worth at maximum speed.

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

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

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

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

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

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

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

15.2.1.2 Analysis of Effects and Consequences

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in two stages. First a systems analysis is performed which includes the prediction of core power response. In the second stage, forcing functions generated in the systems analysis are subsequently used in sub-channel thermal-hydraulics analyses to determine minimum DNBR for the event.

System and core nuclear response is performed using the RELAP5/MOD2- B&W computer code (Reference 12). The RELAP5/MOD2-B&W code simulates the neutron kinetics, reactor coolant system and steam system thermal-hydraulics. The power response for the uncontrolled RCCA bank withdrawal from subcritical event is generated with the RELAP5/MOD2-B&W point kinetics model utilizing conservatively bounding reactivity feedback inputs.

System flow, pressure, core inlet temperature, and core kinetics responses resulting from the system analysis are transferred to the LYNXT computer code (Reference 13). LYNXT is used to generate core sub-channel fluid response, hot-pin heat flux, and location-specific DNBR. LYNXT studies determine a radial peaking limit that, in conjunction with the core power response, yields an acceptable margin to DNB for the RCCA bank withdrawal from subcritical event. Adherence to the peaking limit is assured by the use of three-dimensional neutronics computer simulation - NEMO (Reference 16) as part of each fuel cycle design.

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

1. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values are used. The least negative, or lower, Doppler curve of Figure 15.1.6-1 is utilized in this analysis.

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. After the initial neutron flux peak, the succeeding rate of power increase can be affected by the moderator reactivity coefficient. Since the moderator reactivity coefficient must be negative at all times in plant life any reactivity feedback to a core heatup would be negative, a mitigative response. A conservative value of 0 pcm/^{o}F is therefore utilized in this analysis.

3. The reactor is assumed to be initially critical at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher 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.

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod cluster control assembly release, is taken into account. A 10 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 rod cluster control assembly is stuck in its fully withdrawn position. See Subsection 15.1.5 for rod cluster control assembly insertion characteristics.

5. The maximum positive reactivity insertion rate assumed (57 pcm/s) is greater than that for the simultaneous withdrawal of the combination of the two control banks having the greatest combined worth at maximum speed (45 inches/min). Control rod drive mechanism design is discussed in Subsection 4.2.3.

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

7. Two reactor coolant pumps are assumed to be operational.

Results

Use of a BOL delayed neutron fraction effectively slows the neutron flux response, allowing closer coupling to the increase in lagging core thermal power. Use of BOL parameters, therefore, result in the most limiting RCCA withdrawal from subcritical event response. Figures 15.2.1-1 through 15.2.1-3 show the transient behavior for this event.

Figure 15.2.1-1 is a graph of the nuclear power transient predicted in the systems analysis with RELAP5/MOD2-B&W. Nuclear power is normalized to the rated thermal power of 3455 MWth. The nuclear power overshoots the rated thermal power for a very short time period. As a result, the fuel thermal response is limited.

Core thermal power predicted in the systems analysis with RELAP5/MOD2-B&W is normalized to rated thermal power and plotted in Figure 15.2.1-2. The benefit of the fuel thermal lag behind the nuclear power response is demonstrated in this plot. Thermal power peaks significantly below rated thermal power.

An adequate margin to DNB is indicated by the results of the LYNXT DNB study. In addition, the study demonstrates that the hot-pin fuel temperature response is acceptable and that the peak centerline temperature is significantly less than the fuel melt temperature. A plot of the peak fuel and cladding temperature response is shown in Figure 15.2.1-3.

The time sequence of events for the RCCA withdrawal from subcritical event is shown in Table 15.2-1.

15.2.1.3 Conclusions

The system analysis for the RCCA withdrawal accident from subcritical event demonstrates that the systems responses are well within relevant material and component limits. The corresponding core sub-channel thermal-hydraulics analysis results in a minimum DNBR above the safety limit. In addition, fuel thermal responses are well within material limits. All acceptance criteria are, therefore, met for this event.

The RCCA Withdrawal from Subcritical event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power

15.2.2.1 Identification of Causes and Accident Description

Uncontrolled rod cluster control assembly bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power

mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, in order to avert damage to the cladding the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the safety limit.

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

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

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

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

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

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

In addition to the above listed reactor trips, there are the following rod cluster control assembly withdrawal blocks:

1. High neutron flux (one out of four)

2. Overpower ΔT (two out of four)

3. Overtemperature ΔT (two out of four)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of Reactor Coolant System conditions is described in Chapter 7. This includes a

plot (also shown as Figure 15.1.3-1) representing typical allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety limit. All points below and to the left of DNB line for a given pressure have a DNBR greater than the safety limit. The diagram shows that DNB is prevented for all cases if the area enclosed within the maximum protection lines is not traversed by the applicable DNBR line at any point. This diagram is valid also for Cycle 9 reload fuel supplied by the B&W Fuel Company.

The area where DNBR is greater than the safety limit, and power less than the overpower limit (power, pressure and temperature) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

Power distribution analyses of rod withdrawal accidents from reduced power conditions are described in WCAP-8403 (Reference 10). Radial peaking factors under various rodded conditions used in the accidents are shown in Table 3-1 from Reference 10. The axial power shapes preceding and during the withdrawal accidents are not included here explicitly because of the very large number involved. However, the results from the axial calculations are synthesized with the radial peaking factors using the techniques described in Section 4.3, and then plotted in the form of a "flyspeck." The results indicate that the control bank malfunction flyspeck and the boration/dilution flyspeck reach a smaller maximum linear heat generation rate than the limiting value of 21 kW/ft. Considerable margin is available for conservatism.

15.2.2.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed using the RELAP5/MOD2-B&W code (Reference 12). The RELAP5/MOD2-B&W code is a thermal-hydraulic code that simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, steam generators and steam generator safety valves. The code calculates the system parameters by performing a semi-implicit solution of conservation of mass, energy and momentum over two fluids (liquid and vapor). The system and core power responses generated by RELAP5/MOD2-B&W are used to determine the hot channel DNBR by applying the statistical core design methodology (Reference 14) with the LYNXT computer code (Reference 13).

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

1. Initial conditions of maximum core power (including instrumentation errors) and use of statistical core design methodology that accounts for measurement and control band uncertainties on all critical parameters.

2. Reactivity Coefficients - Two cases are analyzed:

    a. Minimum Reactivity Feedback. The most positive moderator coefficient is assumed, corresponding to the beginning of core life. The least negative Doppler power coefficient is assumed, consistent with the beginning of core life.

b. Maximum Reactivity Feedback. The largest negative moderator coefficient allowed by the Technical Specifications of the plant is assumed. The largest negative Doppler power coefficient is assumed, consistent with the end of core life.

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

4. The rod cluster control assembly 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.

Results

Figures 15.2.2-1 through 15.2.2-3 show the transient response to a rapid rod cluster control assembly withdrawal incident (75 pcm/sec) starting from full power. Reactor trip on high neutron flux occurs shortly after start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and a large margin to DNB is maintained.

The transient response for a slow control rod assembly withdrawal (0.82 pcm/sec) from full power is shown in Figures 15.2.2-4 through 15.2.2-6. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid rod cluster control assembly withdrawal.

Figure 15.2.2-7 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for the minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and the overtemperature ΔT channels. The minimum DNBR is never less than the safety limit. The minimum DNBR values shown in Figure 15.2.2-7 are from RELAP5/MOD2-B&W calculations.

A typical sequence of events for both a large and a small reactivity insertion rate may be found in Table 15.2-1.

15.2.2.3 Conclusion

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

The RCCA Withdrawal at Power has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.3 Rod Cluster Control Assembly Misalignment

15.2.3.1 Identification of Causes and Accident Description

Rod cluster control assembly misalignment accidents include:

1. A dropped full-length assembly;

2. A dropped full-length assembly bank;

3. Statically misaligned full length assembly.

Each rod cluster control assembly 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 bottom light. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank.

A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Mechanical failures are in the direction of insertion or immobility.

A dropped assembly or assembly bank is detected by:

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

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

3. Rod bottom light(s);

4. Rod deviation alarm;

5. Rod position indication.

Misaligned assemblies are detected by:

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

2. Rod deviation alarm;

3. Rod position indicators.

The resolution of the rod position indicator channel is ± 5 percent of span (± 7.2 inches). Deviation of any assembly from its group by twice this distance (10 percent of span, or 14.4 inches) will not cause

power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to group demand position in excess of 5 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 non-indicated assemblies. The operator is also required to take action, as required by the Technical Specifications.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

1. One or More Dropped RCCAs from the Same Group

   For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code (Reference 4). The LOFTRAN code is described in section 15.1.9.3.

   Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNBR design basis is shown to be met using the THINC code (See Section 4.4.3.4.1). The transient response, nuclear peaking factor analysis, and DNBR design basis confirmation are performed in accordance with the methodology described in Reference 5.

2. Statically Misaligned RCCA

   Steady state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the THINC code to calculate the DNBR.

Results

1. One or More Dropped RCCAs

   Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion that may be detected by the power range negative neutron flux rate trip circuitry. If detected, the reactor is tripped within approximately 2.7 seconds following the drop of the RCCAs. The core is not adversely affected during this period since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

   For those dropped RCCAs that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

   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 15.2.3-1 and 15.2.3-2 show a typical transient response to a dropped RCCA (or RCCAs) in automatic control. In all cases, the minimum DNBR remains above the safety analysis limit value.

2. Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion of greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.7 seconds following the drop of a RCCA bank. The core is not adversely affected during this period since power is decreasing rapidly. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 minutes following the incident.

3. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where Bank D is fully inserted with one

RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the safety analysis limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. The full power insertion limits on control Bank D must be chosen to be above that position (which meets minimum DNBR and peaking factors. The full power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with Bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the safety analysis limit value. This case is analyzed assuming 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.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming 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.

DNBR does not occur from 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 group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

15.2.3.3 Conclusions

For all cases of dropped RCCAs or dropped banks, for which the reactor is tripped by the power range negative neutron flux rate trip, there is no reduction in the margin to core thermal limits and, consequently, the DNB design basis is met. It is shown for all cases which do not result in reactor trip that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For all cases of any RCCA 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 safety analysis limit value.

The RCCA Misalignment event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.4 Uncontrolled Boron Dilution

15.2.4.1 Identification of Causes and Accident Description

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

The opening of the primary water makeup control valve provides makeup to the RCS which can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve or stopping the primary makeup water pump. Makeup water can be added to the RCS at pressure when at least one charging pump and a primary makeup water pump are running.

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

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board.

In order to dilute, two separate operations are required:

1. The operator must switch from the automatic makeup mode to the dilute mode, and

2. The boric acid to blender flow control switch must be turned to the start position.

Omitting either step would prevent dilution. Information on the status of the reactor coolant makeup is continuously available to the operator by:

1. Status lights on the control board to indicate CVCS operating conditions.

2. CVCS deviations in flow from programmed levels at the boric acid and demineralized water blender.

3. Pressurizer level and pressure would be increasing from prescribed values (at higher than planned dilution flows).

4. Volume control tank level deviation from programmed level.

Thus there are a number of diverse indications available to the operator to indicate inadvertent or excessive dilutions.

15.2.4.2 Analysis of Effects and Consequences

Methods of Analysis

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

Dilution During Refueling

An uncontrolled boron dilution accident is not credible during refueling. This accident is prevented by administrative controls which isolate the RCS from significant sources of unborated water.

Various valve combinations that are required to be verified closed during refueling operations are specified in Technical Specification 3.9.1. These valves will block the significant dilution flow paths which could allow unborated makeup to reach the RCS. Dilution flow paths, such as instrument and sampling sense lines, where the physical size of the connection to the RCS allows sufficient time for operator response based on source range count rate, are considered insignificant. Any makeup which is required during refueling will be borated water supplied either from the refueling water storage tank by the low head safety injection pumps or the centrifugal charging pumps, or from the boric acid tanks via a boric acid transfer pump and a centrifugal charging pump.

Dilution During Startup

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

1. Dilution flow is the maximum capacity of the makeup water pumps, 300 gpm.

2. A minimum RCS water volume of 9019 ft^3. This corresponds to the active RCS volume excluding the pressurizer and the reactor vessel upper head.

3. The initial boron concentration is assumed to correspond to a conservative maximum value for the critical concentration at the condition of hot zero power, rods to insertion limits, and no Xenon.

4. The critical boron concentration following reactor trip is assumed to correspond to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition.

Dilution Following Reactor Shutdown

Following reactor shutdown, when in hot standby, hot shutdown, and subsequent cold shutdown condition, and once below the P-6 interlock setpoint, and 10^4 counts per second, the high flux at shutdown alarm setting will be automatically adjusted downward to a nominal value of 3 times the background count rate as the background count rate reduces.

Surveillance testing will ensure that the alarm setpoint is operable. The operator does not depend entirely on this alarm setpoint but has audible indication of increasing neutron flux from the audible count rate drawer and visual indication from counts per second meters for each channel on the main control board and source range drawer.

Dilution at Power

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

1. Dilution flow at power is the maximum capacity of the makeup water pumps, 300 gpm.

2. A minimum RCS water volume of 9019 ft^3. This corresponds to the active RCS volume excluding the pressurizer and the reactor vessel upper head.

3. The initial boron concentration is the conservative maximum value for the critical concentration at the condition of hot full power, rods to insertion limits, and no Xenon.

4. The critical boron concentration following reactor trip is assumed to correspond to the hot zero power, all rods inserted (minus the most reactive RCCA), no Xenon condition

15.2.4.3 Conclusions

The time sequence of events during these transients are shown in Table 15.2-1.

For dilution during refueling:

Sufficient dilution of the RCS boron concentration during refueling cannot occur due to administrative controls (see Section 15.2.4.2) and operator response.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. At all times during fuel movement, the source range audible count rate is on the refueling floor of reactor containment and the main control room. In addition, a high source range flux level at shutdown is alarmed in the control room and in the reactor containment. The count rate increase is proportional to the subcritical multiplication factor.

For dilution during startup:

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

For dilution during startup, a cycle specific check demonstrates that the initial and final borons result in more than 15 minutes for operator action from the time of alarm (reactor trip on P-6) to loss of shutdown margin.

For dilution during full power operation:

With the reactor in automatic rod control, the power and temperature increase from boron dilution results in insertion of the control rods and a decrease in the available shutdown margin. The rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator that a dilution event is in process. A cycle specific check demonstrates that the initial and final borons result in more than 15 minutes for operator action from the time of alarm (LOW-LOW rod insertion limit) to loss of shutdown margin.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the Overtemperature ΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to be within the range of insertion rates analyzed for the RCCA bank withdrawal at power. A cycle specific check demonstrates that the initial and final borons result in more than 15 minutes for operator action from the time of alarm (overtemperature ΔT) to loss of shutdown margin.

For Dilution Following Reactor Shutdown:

In providing a description of a boron dilution event initiated immediately after scram, it is appropriate to analyze two initial conditions. These are:

1. BOL, Equilibrium Xe

   This will result in the longest time following scram until the Source Range Nuclear Instrumentation System (NIS) is available to provide an indication of a dilution event.

2. BOL, Clean Core

   This will result in a very short time following scram for the source range NIS to become available, however, it yields the most rapid boron dilution (return to criticality) case.

Figure 15.2.4-1 shows the relative change in boron concentration with time for the two cases. The dilution rates are consistent with the time for the two cases. Figure 15.2.4-2 shows the condition of the core consistent with the boron concentrations of Figure 15.2.4-1 and Xe build-up following trip for the Eq Xe cases.

Figure 15.2.4-3 shows the information available to the operator on the core relative power based on the Nuclear Instrumentation System for the Eq Xe case. As shown there is essentially an instantaneous decrease in nuclear power from 100% to 7.5% (< 5 seconds). From 7.5% the standard 80 second period is used until the precursor isotopes have been depleted. From the point shown, an 18-day half life is assumed. For the case without Eq Xe, the NIS stable reading on source range is achieved very rapidly, < 5 minutes as opposed to 21 minutes for the Eq Xe case.

The sequence of events Table 15.2-1 show that for both cases > 15 minutes of operator action time is available. Therefore the acceptance criteria for this event is met. In addition to the High Flux at Shutdown Alarm, there is also the High Pressurizer Level Trip and alarm available. In order to return critical a very large total dilution volume is required. The only means of accommodating this large volume is to allow the pressurizer to start filling. As shown, however, this results in a High Pressurizer Level Alarm very early in the transient.

These two alarms would provide the operator an adequate set of indications that a boron dilution event was in progress and also allow adequate time for operator corrective action.

The Uncontrolled Boron Dilution event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.5 Partial Loss of Forced Reactor Coolant Flow

15.2.5.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump. 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 necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip which is actuated by two out of three low flow signals in any reactor coolant loop. Above approximately 35% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. A reactor trip signal from the pump undervoltage relay is provided as an anticipatory signal which serves as a backup to the low flow signal. It functions essentially identically to the low flow trip so that above Permissive 7 an undervoltage relay trip signal from any two pumps will actuate a reactor trip.

The RCPs are constantly supplied from the Common Station Service Transformers (CSSTRs) or are supplied for 30 seconds from the Unit Station Service Transformers (USSTRs) provided there are no electrical faults to separate the generator from the network. This operational logic is discussed in Section 8.2. The following analysis is bounding for the condition described above.

15.2.5.2 Analysis of Effects and Consequences

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 Code (Reference 4) 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 Code (Reference 3) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC Code is used to calculate the departure from DNBR during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The WRB-1 correlation is used for DNBR calculation. The DNB transients presented represent the minimum of the typical or thimble cell.

Typical Initial Conditions

Initial operating conditions assumed are the most adverse with respect to the margin to DNB, i.e., maximum steady state power level, minimum steady state pressure, and maximum steady state coolant average temperature. See Subsection 15.1.2 for explanation of initial conditions. In addition to the initial average temperature condition in Subsection 15.1.2, 1.5°F was added to the initial average temperature for conservatism.

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (See Table 15.1.2-2). The total integrated Doppler reactivity from 0 to 100% power is assumed to be 0.016 Δk. The lowest absolute magnitude of the moderator temperature coefficient (0.0 $\Delta k/°F$) is assumed since this results in the maximum hot-spot heat flux during the initial part of the transient when the minimum DNBR is reached.

Flow Coastdown

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

Results

The calculated sequence of events is shown on Table 15.2-1 for the case analyzed. Figures 15.2.5-1 through 15.2.5-3 show the resulting transient conditions for the 2/4 Partial Loss of Flow analysis. Included in these figures are total RCS flow, faulted loop flow, average and hot channel heat flux, nuclear power, and DNBR, each as a function of time. The minimum DNBR is not less than the safety analysis limit.

15.2.5.3 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit at any time during the transient. Thus there will be no cladding damage and no release of fission products to the Reactor Coolant System.

The Partial Loss of Forced Reactor Coolant Flow event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.6 Startup Of An Inactive Reactor Coolant Loop

15.2.6.1 Identification of Causes and Accident Description

The SQN Technical Specification requires that all reactor coolant loops be in operation during plant's startup and power operations except for special test conditions. The following analysis is for a 3 loop operation and will address startup of an inactive reactor coolant loop which may be created due to an operational error.

If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, 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 causes a rapid reactivity insertion and subsequent power increase. This event is classified as an ANS Condition II incident (a fault of moderate frequency).

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 would have been previously reset for three loop operation.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by three digital computer codes. The LOFTRAN Code (4) is used to calculate the loop and core flow, nuclear power and core pressure and temperature transients following the startup of an idle pump. FACTRAN (3) is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC Code (see Section 4.4) is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

Plant characteristics and initial conditions are discussed in Section 15.1.2. In order to obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

1. Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to DNB. These values are consistent with the maximum steady state power level allowed with three loops in operation. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

2. Following initiation of startup of the idle pump, flow in the inactive loop reverses and accelerates to its nominal full flow value in approximately 7 seconds.

3. A conservatively large moderator density coefficient (see Section 15.1.6).

4. A conservatively small (absolute value) negative Doppler power coefficient (see Section 15.1.6).

5. The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

6. The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.2.6-1 through 15.2.6-4. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average water temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit. See Section 4.4 for a description of the DNBR design basis.

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient, this decrease is due both to the increase in reactor coolant flow and, as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.2.6-2.

The calculated sequence of events for this accident is shown on Table 15.2-1. The transient results illustrated in Figures 15.2.6-1 through 15.2.6-4 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.2.6.3 Conclusions

The transient results show that the core is not adversely affected, i.e., there is considerable margin to the DNB safety analysis limit.

Operation of the Sequoyah Nuclear Plant with fewer than 4 RCP's in operation is prohibited (except during special test conditions) by plant Technical Specifications. The startup of an inactive reactor coolant loop event, therefore, is not considered as a safety issue with replacement steam generators.

15.2.7 Loss Of External Electrical Load And/Or Turbine Trip

15.2.7.1 Identification of Causes and Accident Description

Major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case off site power remains available for the continued operation of plant components such as

the reactor coolant pumps. The case of loss of all AC power (station blackout) is analyzed in Subsection 15.2.9. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a turbine trip, the reactor would be tripped directly (unless below approximately 50% power) from a signal derived from the turbine autostop oil pressure (Westinghouse Turbine) and turbine stop valves. The turbine stop valves close on loss of autostop oil pressure actuated by one of a number of possible turbine trip signals. Turbine-trip initiation signals include:

1. Generator Trip
2. Low Condenser Vacuum
3. Loss of Lubricating Oil
4. Turbine Thrust Bearing Failure
5. Turbine Overspeed
6. Manual Trip

Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors associated with the stop valves detect the turbine trip and initiate the turbine trip and initiate steam dump and if above 50 percent power, a reactor trip. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure with a resultant primary system transient.

The automatic steam dump system would accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to atmosphere through the steam generator relief and safety valves. Additionally, main feedwater flow would be lost if the turbine condenser was not available. For this situation feedwater flow would be maintained by the Auxiliary Feedwater System.

The Sequoyah plant is designed to accept a load rejection of 50 percent of its rated electrical load, and signals from the reactor protection system will trip the plant for load rejections in sufficient excess of 50 percent of rated load.

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

The steam generator safety valve capacity is sized to remove the original maximum calculated turbine steam flow from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the Reactor Coolant System pressure within 110 percent of the Reactor Coolant System design pressure without direct or immediate reactor trip action.

A more complete discussion of overpressure protection can be found in Reference 8.

Normal power for the reactor coolant pumps is supplied through busses from a transformer connected to the preferred off-site power system. In the alternate or maintenance power supply configuration, when a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator, thus ensuring flow for 30 seconds before any transfer is made. The analysis of effects and consequences for this condition is based on the alternate power supply configuration and is bounding.

Should the network bus transfer fail at 30 seconds, a complete loss of forced reactor coolant flow would result. This assumption is made for the analysis of a complete loss of load at approximately 50% power without direct reactor trip. The immediate effect of loss of coolant flow is a rapid increase in the coolant temperature in addition to the increased coolant temperature as a result of the turbine trip. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly.

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

1. Reactor coolant pump power supply undervoltage or underfrequency

2. Low reactor coolant loop flow

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

The reactor trip on reactor coolant pump underfrequency is provided to open the reactor coolant pump breakers and trip the reactor for an underfrequency condition, resulting from frequency disturbances on the major power grid. The trip disengages the reactor coolant pumps from the power grid so that the pump's kinetic energy is available for full coastdown.

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. 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.

15.2.7.2 Analysis of Effects and Consequences

Methods of Analysis

The minimum DNBR for a total loss of load transient is bounded by the value calculated for a complete loss of forced reactor coolant flow (15.3.4). Consequently, the analysis of total loss of load is performed to show the adequacy of the pressure relieving devices on the primary and secondary systems. Two loss of load cases are analyzed. These are a loss of load from 102 percent of full power and a total loss of load from 52 percent of full power.

BWFC reanalyzed the total loss of load from 102 percent of full power for the Cycle 9 reload with BWFC fuel because this event is the limiting overpressure event for the primary and secondary systems. The transient was analyzed using the RELAP5/MOD2-B&W computer program (Reference 12) to generate the time responses of the primary system, secondary system and core average power. The reactor was not tripped on the turbine trip but tripped later on a high pressurizer pressure trip. 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. In addition, no credit is taken for steam dump and worst case lift tolerance of six percent is assumed for the steam generator safety valves. Two of three pressurizer safety valves, each with a lift tolerance of five percent, are assumed to be available to mitigate the primary system pressure transient. Although this is not the limiting event for minimum DNBR, the system and core power responses predicted by RELAP5/MOD2-B&W are used to determine the minimum DNBR for the event by employing the statistical core design methodology of BAW-10170 (Reference 14).

The total loss of load transient from 52 percent of full power is analyzed by employing the detailed digital computer program LOFTRAN. The program simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes the pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.1.3-1 are used as input to LOFTRAN to determine the minimum DNBR of the typical or thimble cell during the transient. In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 52 percent of full power without reactor trip. The turbine is assumed to trip without actuating all the sensors for reactor trip on the turbine stop valves. The assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst transient. In addition, no credit is taken for steam dump. Main feedwater is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

A fast bus transfer is attempted 30 seconds following the loss of steam load from 52 percent power. The transfer to an external power source is assumed to fail which results in a complete loss of flow transient initiated from the loss of load condition. The loss of flow transient, due to the assumed failure of the fast bus transfer, is analyzed by employing the detailed digital computer codes LOFTRAN (4), FACTRAN (3), and THINC Subparagraph 4.4.3.4. The FACTRAN code calculates the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code calculates the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN.

Typical assumptions are:

1. Initial Operating Conditions - The initial reactor coolant pressure is set to the nominal value for the 102 percent power case. To maximize the pressure response, the initial primary system average temperature was set to the nominal value minus four degrees for control band and measurement uncertainty. For the loss of load from 52 percent power, the initial reactor power and Reactor Coolant System temperatures are assumed at their maximum values consistent with the power level including allowances for calibration and instrument errors. The initial Reactor Coolant System pressure for this case is assumed at a minimum value that includes allowances for calibration and instrument errors. Table 15.2.7-1 summarizes the initial conditions assumed.

2. Moderator and Doppler Coefficients of Reactivity - Sensitivity calculations show that the maximum primary system pressure is obtained for beginning of life conditions. Consequently, for the loss of load from 102 percent power, a moderator coefficient of +7.0 pcm/F and the least negative

   Doppler power coefficient are used. The total loss of load from 52 percent power is analyzed for both beginning of life and end of life conditions. Moderator temperature coefficients of zero at beginning of life and a large (absolute value) negative value at end of life are used. A conservatively large (absolute value) Doppler power coefficient is used for all cases.

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

4. Steam Release - No credit is taken for the operation of the steam dump system or steam generator power operated relief valves. The steam generator pressure rises to the safety valve setpoint (plus lift tolerances, where applicable) where steam release through safety valves limits secondary steam pressure near the lift value.

5. Pressurizer Spray and Power Operated Relief Valves - To maximize the primary system pressure response to a total loss of load from 102 percent power, no credit is taken for the pressurizer spray or pressurizer power operated relief valves. For the loss of load from 52 percent power, two cases for both the beginning and end of life are analyzed:

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

   b. No credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Pressurizer heater operation is assumed since heater operation on high pressurizer water level will tend to increase the maximum surge rate through the pressurizer safety valves.

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 on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

7. Reactor Trip - Reactor trip is actuated by the first Reactor Protection System trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, low reactor coolant loop flow, reactor coolant pump power supply undervoltage, and low-low steam generator water level.

Except as discussed above, normal reactor control system and Engineered Safety Systems are not required to function.

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.

Results

The transient response for a total loss of load from 102 percent of full power is shown in Figures 15.2.7-1 through 15.2.7-6. Following closure of the turbine stop valves, the primary pressure increases causing a reactor trip on high pressurizer pressure. Control rod insertion terminates the transient. Actuation of the pressurizer safety valves ensures that the primary system pressure remains below the acceptance criterion. Likewise, the main steam safety valves ensure that the secondary system pressure does not exceed 110 percent of design. The DNBR remains above the limit at all times. The sequence of events is shown in Table 15.2-1.

The transient responses for a turbine trip from 52% of full power operation are shown for four cases: two cases for minimum reactivity feedback (beginning of core life) and two cases for maximum reactivity feedback (end of core life). These results are shown in Figures 15.2.7-9 through 15.2.7-16. The calculated sequence of events for this accident is shown on Tables 15.2.7-2 and 15.2.7-3.

Figures 15.2.7-9 and 15.2.7-10 show the transient responses for the total loss of steam load with a least negative moderator temperature coefficient assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The fast bus transfer is attempted and assumed to fail 30 seconds after the total loss of steam load. The transfer failure results in an undervoltage trip of the reactor and the initiation of the loss of flow transient. The minimum DNBR remains well above the safety analysis limit. The steam generator safety valves limit the secondary steam conditions to saturation at the safety valve setpoint.

Figures 15.2.7-11 and 15.2.7-12 show the responses for the total loss of steam load with a large negative moderator temperature coefficient. All other plant parameters are the same as the above. The minimum DNBR remains well above the safety analysis limit throughout the transient. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively.

The turbine trip accident was also studied assuming the plant to be initially operating at 52% of full 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. With the plant in the alternate or maintenance auxiliary power supply configuration (generator supplying station power), the

fast bus transfer for this case is assumed to fail at 30 seconds after the total loss of load. Figures 15.2.7-13 and 15.2.7-14 show the transients with a least negative moderator coefficient. The neutron flux remains essentially constant at 52% of full power until the reactor is tripped. The DNBR remains above the safety analysis limit throughout the transient. In this case the pressurizer safety valves are actuated and maintain system pressure below 110% of the design value.

Figures 15.2.7-15 and 15.2.7-16 are the transients with maximum reactivity feedback with the other assumptions being the same as in the preceding case. Again, the minimum DNBR remains above the safety analysis limit throughout the transient. In this case, the pressurizer safety valves are momentarily actuated.

Reference 8 presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This report shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

The time sequence of events during this transient is shown in Table 15.2-1.

15.2.7.3 Framatome ANP Safety Evaluation With Replacement Steam Generators in Unit 1

The parameters that affect the peak secondary pressure are the primary-to-secondary heat transfer rate, MSSV characteristics, and the pressure differential between the MSSVs and the peak steam generator pressure location. The peak RCS pressure is a function of the initial secondary pressure, primary-to-secondary heat transfer rate, pressurizer safety valve capacity, and high pressurizer pressure reactor trip setpoint.

The heat transfer capacity of the RSG is greater than the OSGs, allowing for greater primary-to-secondary heat transfer rates. The pressurizer safety valve capacity and pressurizer pressure reactor trip setpoint are not affected by the RSG. The initial secondary pressure is greater for the RSGs. Therefore, since the greater primary-to-secondary heat transfer rate would be greater with the RSGs for primary over-pressurization, and the initial secondary pressure is greater for the RSGs, the current licensing analysis for primary over-pressurization is applicable to Unit 1 with the RSGs.

An analysis was performed by Framatome ANP to demonstrate that the peak secondary pressure would remain below the acceptance criteria following a loss of electrical load event at Sequoyah Unit 1 with RSGs. The analysis was performed with the RELAP5/MOD2 computer code using a Sequoyah specific plant model.

The RELAP5 peak secondary pressure was obtained using a maximum RCS average temperature and 0% tube plugging in the RSGs. This resulted in a peak steam generator pressure of 1197 psia, which is less than the pressure limit of 1208 psia. Because the acceptance criteria would not be exceeded following a loss of electrical load event at Sequoyah Unit 1 using the RSGs, there is no reduction in the margin of safety associated with the replacement.

To validate Technical Specification Table 3.7-1, partial power cases were also analyzed with different numbers of MSSVs out of service. These analyses yielded secondary pressure below the acceptance limit. Therefore, the Technical Specification table is still valid with the RSGs.

15.2.7.4 Conclusions

Results of the analyses, including those in Reference 8 show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the Reactor Protection System, i.e., the DNBR will be maintained above the safety analysis limit. Thus there will be no cladding damage and no release of fission products to the Reactor Coolant System.

15.2.8 Loss of Normal Feedwater

15.2.8.1 Identification of Causes and Accident Description

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

The worst postulated loss of normal feedwater event is one initiated by a loss of offsite AC power which is described in Section 15.2.9. This is due to the decreased capability of the reactor coolant to remove residual core heat as a result of the RCP coastdown.

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 available, the steam generator 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 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.

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

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

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

    a. Low-low level in any steam generator

b. Trip of both feedwater pumps

c. A safety injection signal

d. Loss of offsite power

e. Manual actuation

f. Loss of one main feedwater pump with turbine load greater than 77% (Unit 2) and 76.6% (Unit 1).

3. One turbine driven auxiliary feedwater pump which utilizes steam from the steam generators is started on:

a Low-low level in any two steam generators, or

b. Trip of both feedwater pumps

c. Loss of offsite power

d. Safety injection signal

e. Manual actuation

f. Loss of one main feedwater pump with turbine load greater than 77% (Unit 2) and 76.6% (Unit 1).

The motor driven auxiliary feedwater pumps are supplied with power by the diesel generators if a loss of offsite power occurs and the turbine-driven pump utilizes steam from the main steam system. Both type pumps are designed to supply at least, minimum required flow within one minute of the initiating signal even if a loss of all non-emergency AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the used steam to the atmosphere. The auxiliary pumps take suction from the condensate storage tank for delivery to the steam generators.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat, thus, preventing either overpressurization of the Reactor Coolant System or loss of water from the reactor core.

The analysis takes into account a maximum of 15 percent uniform tube plugging in the steam generators.

15.2.8.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using the RELAP5/MOD2 (Reference 12) is performed to obtain the plant parametric response due to a loss of normal feedwater. The digital computer simulation of RELAP5 includes plant nuclear kinetics, reactor coolant system (with pressurizer and steam generators), main

feedwater, auxiliary feedwater, and safety injection systems. The code computes the resultant system parameters including the steam generator level, pressurizer water level, and the reactor coolant average temperature.

Major assumptions are:

1. The plant is initially operating at 102% of 3423 MWt.

2. Core decay heat is conservatively calculated based upon 1.0 times ANS 5.1 of 1971 decay heat curve with B&W heavy actinides contribution to decay heat.

3. Only one motor driven auxiliary feedwater pump is available one minute after the low-low steam generator level signal is initiated in any steam generator.

4. Auxiliary feedwater flow rate of 410 gpm is split uniformly between two steam generators.

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

6. The initial reactor coolant average temperature is 5.5°F higher than the nominal value since this results in a greater expansion of Reactor Coolant System water during the transient and, thus, in a higher water level in the pressurizer.

Results

Figures 15.2.8-1 through 15.2.8-4 show 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. One minute following the initiation of the low-low level trip, at least one auxiliary feedwater pump is automatically started, reducing the rate of water level decrease in two steam generators.

The capacity of one motor driven auxiliary feedwater pump is such that the water level in two steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves. Figure 15.2.8-2b 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 15.2.8-1 through 15.2.8-4, the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation.

15.2.8.3 Conclusions

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

The Loss of Normal Feedwater event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.9 Loss of Off-Site Power to the Station Auxiliaries

15.2.9.1 Identification of Causes and Accident Description

In the event of a complete loss of offsite power and a turbine trip there will be a loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc.

The events following a loss of AC power with turbine and reactor trip are described in the sequence listed below:

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

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

3. As the no load temperature is approached, the steam generator power-operated relief valves (or 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.

4. The Class 1E standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The auxiliary feedwater system is started automatically as discussed in the loss of normal feedwater analysis. The turbine driven auxiliary feedwater pump utilizes steam from the main steam system and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by power from the diesel generators. The auxiliary feedwater pumps take suction from the condensate storage tank for delivery to the steam generators.

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

The analysis presented takes into account a maximum of 15 percent uniform tube plugging in the steam generators.

15.2.9.2 Analysis of Effects and Consequences

Method of Analysis

A detailed analysis using RELAP5/MOD2 (Reference 12) is performed to obtain the plant parametric response due to a loss of offsite power. The digital computer simulation of RELAP5 includes plant

nuclear kinetics, reactor coolant system (with pressurizer and steam generators), main feedwater, auxiliary feedwater, and safety injection systems. The code computes the resultant system parameters including the steam generator level, pressurizer water level, and the reactor coolant average temperature.

Major assumptions are:

1. The plant is initially operating at 102% of 3423 MWt.

2. Core decay heat is conservatively calculated based upon 1.0 times ANS 5.1 of 1971 decay heat curve with B&W heavy antinides contribution to decay heat.

3. Only one motor-driven auxiliary feedwater pump is available one minute after the low-low steam generator level signal is initiated in any steam generator.

4. Auxiliary feedwater flow rate of 410 gpm is split uniformly between two SGs.

5. The initial reactor coolant average temperature is 5.5°F lower than the nominal value since this results in a greater expansion of Reactor Coolant System water during the transient and a higher water level in the pressurizer.

Results

The transient response of the RCS following a loss of AC power is shown in Figures 15.2.9.1 through 15.2.9.4. The calculated sequence of events for this event is listed in Table 15.2-1.

The first few seconds of the transient following receipt of a reactor trip signal will closely resemble a simulation of the complete loss of flow incident (see Subsection 15.3.4), 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.

15.2.9.3 Framatome Safety Evaluation With Replacement Steam Generators in Unit 1

Core cooling following a Loss of Offsite Power event is ensured by heat removal via auxiliary feedwater injection and by boiling in the steam generator liquid inventory. Steam is relieved through the MSSVs. Steam generator replacement will not affect the AFW flow rate, the AFW delay time, or the steam relief capacity of the MSSVs. However, differences between physical OSG and RSG parameters such as primary-to-secondary heat transfer area, tube bundle height, and primary flow resistance can affect long-term heat removal capabilities. The Loss of Offsite Power event was reanalyzed by Framatome.

The reanalysis was performed using the identical methods described in Section 15.2.9.2. Results of the analysis demonstrate that the relevant acceptance criteria are met for this event. Both adequate long-term heat removal and natural circulation characteristics of Sequoyah Unit 1 with RSGs subsequent to a Loss of Offsite Power event are demonstrated by the analysis.

15.2.9.4 Conclusions

Results of the "Complete Loss of Forced Reactor Coolant Flow" analysis (Section 15.3) and the loss of non-emergency AC power to the station auxiliaries show that no adverse conditions occur in the reactor core. The DNBR is maintained above the safety limit. The Reactor Coolant System is not overpressurized and no water relief will occur through the pressurizer relief valves. The pressurizer safety valves do not lift. Thus there will be no cladding damage and no release of fission products to the RCS.

15.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions

15.2.10.1 Identification of Causes and Accident Description

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower - overtemperature protection (high neutron flux, overtemperature and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than the safety analysis limit.

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

15.2.10.2 Analysis of Effects and Consequences

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer codes, RELAP5 and LYNXT (References 12 and 13). The codes are described in section 15.1.9.8 and section 15.1.9.11, respectively. The core limits as illustrated in Figure 15.1.3-1 are used as input to the RELAP5 and LYNXT codes to determine the minimum DNBR during the transient for the full power case. The RELAP5 model used incorporated a detailed representation of the condensate and feedwater system piping including the feedwater heaters, condensate booster pumps, and the main feedwater pumps.

Operation With Analog Feedwater Control

The system is analyzed to evaluate plant behavior in the event of a feedwater system malfunction. Based on the differences in failure modes, a separate analysis has been performed for the analog feedwater control system and the digital feedwater control system. The analog control system contains a single point failure mode which results in a multi-loop feedwater malfunction event. The multi-loop failure mode is analyzed for the analog control system. The multi-loop failure mode was eliminated in the digital control system. However; the single loop failure mode in the digital control system will also cause the main feedwater pump speed demand signal to both pumps to increase to a maximum value. This will cause the speed of both feed pumps to exceed the maximum program speed modeled by the analog system analysis. The digital control system analysis addresses this failure mode by modeling the maximum feed pump speed allowed by the pump overspeed protection equipment.

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

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

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

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

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

Operation With Digital Feedwater Control

3. Accidental opening of all feedwater regulating valves and feedwater regulating bypass valves to the full open position in a single loop with the reactor at zero load.

4. Accidental opening of all feedwater regulating valves and feedwater regulating bypass valves to the full open position in a single loop with the reactor at full power.

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

1. Reactor at zero load

   a. The reactor is assumed to be just critical in the hot shutdown condition.

   b. For case 1.a., an increase in feedwater flow to one steam generator from zero flow to full flow assuming 2 condensate booster pumps and 1 main feed pump are operating at maximum rpm.

      For case 1.b., each steam generator experiences an increase in feedwater flow from zero flow to full flow assuming 2 condensate booster pumps and 1 main feed pump are operating at maximum rpm.

      For case 3, an increase in feedwater flow to one steam generator from zero flow to full flow assuming 2 condensate booster pumps are operating at maximum rpm and 1 main feed pump is operating at the maximum speed allowed by the pump overspeed trip with maximum uncertainty applied.

   c. The feedwater temperature is assumed at a conservatively low value of 58°F, corresponding to lowest condenser hotwell conditions.

2. Reactor at full power

   a. Initial operating conditions are assumed at extreme values consistent with the steady state full power operation allowing for calibration and instrument errors. This results in minimum margin to DNB at the start of the accident.

b. Both automatic and manual rod control was assumed for each of the full power cases. The results from the most limiting scenario are presented.

c. For case 2.a., a step increase in feedwater flow to one steam generator from nominal flow to full flow assuming all 3 condensate booster pumps and both main feed pumps are operating at maximum rpm.

For case 2.b., a step increase to each of the four steam generators from nominal flow to full flow assuming all 3 condensate boosters and both main feed pumps are operating at maximum rpm.

For case 4, a step increase in feedwater flow to one steam generator from nominal flow to full flow assuming all 3 condensate booster pumps are operating at maximum rpm and 2 main feed pumps are operating at the maximum speed allowed by the pump overspeed trip with maximum uncertainty applied.

3. For cases 1 through 4,

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

b. No credit is taken for the heat capacity of the Reactor Coolant System in attenuating the resulting plant cooldown.

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

d. A conservatively large moderator coefficient of reactivity characteristic of end of life core conditions is used.

Results

For the cases of an accidental full opening of one or more feedwater control valves with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Subsection 15.2.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition, and therefore, the results of the analyses are not presented. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 35 percent.

The full power cases give the largest reactivity feedback and result in the greatest power increases. Transient results for operation with the analog feedwater control (see Figures 15.2.10-1 thru 15.2.10-10) and the digital feedwater control (see Figures 15.2.10-11 thru 15.2.10-15) show the pressurizer pressure, T_{avg}, and DNBR, as well as the increase in nuclear power and ΔT_{avg}, associated with the increased thermal load on the reactor. A turbine trip is actuated when the steam generator level reaches the high-high level setpoint. The DNB ratio does not drop below the safety analysis limit.

For all excessive feedwater cases continuous addition of cold feedwater is prevented by closure of all feedwater control valves and a trip of the feedwater pumps on steam generator high-level.

The time sequence of events during this transient is shown in Table 15.2-1.

15.2.10.3 Conclusions

It has been shown that the reactivity insertion rate which occurs at no load following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition. Also, the DNB ratios encountered for excessive feedwater addition at power are well above the safety analysis limit.

The Excessive Heat Removal due to Feedwater System Malfunctions event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.11 Excessive Load Increase

15.2.11.1 Identification of Causes and Accident Description

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

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

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

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

1. Overpower ΔT

2. Overtemperature ΔT

3. Power range high neutron flux

15.2.11.2 Analysis of Effects and Consequences

Method of Analysis

This accident is analyzed using the LOFTRAN (Reference 4) code. This code is described in section 15.1.9.3. The core limits as illustrated in Figure 15.1.3-1 are used as input to LOFTRAN to determine the minimum DNBR of the typical or thimble cell during the transient.

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

1. Manually controlled reactor at beginning-of-life.

2. Manually controlled reactor at end-of-life.

3. Reactor in automatic control at beginning-of-life.

4. Reactor in automatic control at end-of-life.

At beginning of life the core has the least negative moderator temperature coefficient of reactivity and therefore the least inherent transient capability. At end of life the moderator temperature coefficient of reactivity has its highest absolute value. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters. Initial operating conditions are assumed at extreme values consistent with the steady state full power operation allowing for calibration and instrument errors. This results in minimum margin to DNB at the start of the event.

Results

Figures 15.2.11-1 through 15.2.11-4 illustrate the transient with the reactor in the manual control mode. As expected, for the beginning of life case there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the end of life manually controlled case there is a much larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above the safety analysis limit.

Figures 15.2.11-5 through 15.2.11-8 illustrate the transient assuming the reactor is in the automatic control mode. Both the beginning of life and the end of life cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both the beginning of life and end of life cases, the minimum DNBR remains above the safety analysis limit. The time sequence of events for this transient is shown in Table 15.2-1.

15.2.11.3 Conclusions

The Excessive Load Increase has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

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

15.2.12 Accidental Depressurization of the Reactor Coolant System

15.2.12.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the Reactor Coolant System (RCS) are associated with an inadvertent opening of a pressurizer safety valve. Initially the event results in a rapidly decreasing RCS pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At that time, the pressure decrease is slowed considerably. The pressure continues to decrease, however, throughout the transient. The effect of the pressure decrease would be to decrease the neutron flux via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until the reactor trip.

The reactor will be tripped by one of the following Reactor Protection System signals:

1. Pressurizer low pressure

2. Overtemperature ΔT

This transient represents the limiting analysis for the RCS accidental depressurization and imposes the worst temperature/pressure profile onto the reactor core for Condition II events. The inadvertent opening of a pressurizer safety valve event does not establish the design basis piping load condition to be adhered for RCS piping (Reference 11).

15.2.12.2 Analysis of Effects and Consequences

Methods of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN (Reference 4). This code is described in section 15.1.9.3. The core limits as illustrated in Figure 15.1.3-1 are used as input to LOFTRAN to determine the minimum DNBR of the typical or thimble cell during the transient.

In calculating the DNBR the following conservative assumptions are made:

1. Initial conditions of maximum core power and reactor coolant temperatures and minimum reactor coolant pressure resulting in the minimum initial margin to DNB (See Section 15.1).

2. A zero moderator coefficient of reactivity conservative for beginning of life operation in order to provide a conservatively low amount of negative reactivity feedback due to changes in moderator density. The spatial effect of void due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

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

4. The pressurizer safety valve capacity is assumed to be 10% greater to increase the depressurization rate.

It should also be noted that in the analysis power peaking factors are kept constant at the design values while, in fact, the core feedback effects would result in considerable flattening of the power distribution. This would significantly increase the calculated DNBR; however, no credit is taken for this effect.

Results

Figure 15.2.12-1 illustrates the nuclear power transient following the event. Reactor trip on overtemperature ΔT occurs as shown in Figure 15.2.12-1. The pressure and vessel average coolant temperature transients during the accident, including a 10% increase in capacity, are given in Figure 15.2.12-2. The resulting DNBR never goes below the safety analysis limit as shown in Figure 15.2.12-3.

15.2.12.3 Conclusions

The pressurizer low pressure and the overtemperature ΔT Reactor Protection System signals provide adequate protection against this accident, and the minimum DNBR remains greater than the safety analysis limit.

The Accidental Depressurization of the RCS event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.13 Accidental Depressurization of the Main Steam System

15.2.13.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief or safety valve. The analyses performed assuming a rupture of a main steam pipe are given in Section 15.4.

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

The analysis is performed to demonstrate that the following criterion is satisfied: Assuming a stuck rod cluster control assembly and a single failure in the Engineered Safety Features the DNB design basis will be met 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.

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

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

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

    b. Two-out-of three low pressurizer pressure signals.

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

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

15.2.13.2 Analysis of Effects and Consequences

Method of Analysis

The following analyses of a secondary system steam release are performed for this section.

1. A full plant digital computer simulation, LOFTRAN (Reference 4) code, to determine RCS temperature and pressure during cooldown.

2. An evaluation to determine that the DNB design basis is met.

The following conditions are assumed to exist at the time of a secondary system depressurization.

1. End of life shutdown margin at no load, equilibrium xenon conditions, and with the most reactive assembly stuck in its fully withdrawn position. Operation of rod cluster control assembly banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system 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 rod cluster control assembly in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The K_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used plus the Doppler temperature effect, is shown in Figure 15.2.13-1.

3. Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the Safety Injection System. The injection curve assumed is shown in Figure 15.2.13-2. This corresponds to the flow delivered by one charging pump delivering its full contents to the cold leg header. Subsequent to this analysis, the minimum charging pump performance requirements were reduced from those shown in Figure 15.2.13-2. However, the flow reduction was more than offset by the minimum flow available from one safety injection pump. While not specifically modeled in the analysis, the flow from one safety injection pump is also credited in Reference 19 to establish that the results of the analysis remain conservative and bounding for the current charging pump minimum performance requirements in Figure 6.3.2-7. The analysis conservatively assumes that the safety injection lines downstream of the RWST contain no borated water (0 ppm). This water must be swept from the safety injection lines prior to the delivery of boric acid (1950 ppm) from the RWST to the reactor coolant loops.

4. The case studied is an initial total steam flow of 228 lbs/second at 1015 psia from all steam generators with offsite power available. This is the maximum capacity of any single steam dump or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most pessimistic initial condition.

   Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power the RCS 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 are reached. After the additional stored energy is removed, cooldown proceeds in the same manner as in the analysis which assumes no load condition at time zero. However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the RCS cooldown are less for steam line breaks occurring at power.

5. In computing the steam flow the Moody Curve for fL/D = 0 is used.

6. Perfect moisture separation in the steam generator is assumed.

7. The auxiliary feedwater system provides a flow rate of 588 GPM to each steam generator. This auxiliary feedwater is not required to mitigate the transient and is modeled to increase the severity of the core cooldown.

Results

The calculated time sequence of events for this accident is listed in Table 15.2-1. The results presented are a conservative indication of the events which would occur assuming a secondary system steam release since it is postulated that all of the conditions described above occur simultaneously.

Figure 15.2.13-3 shows the transients arising as a result of a steam release having an initial steam flow of 228 lb/second at 1015 psia with steam release from one safety valve. The assumed steam release is the maximum capacity of any single steam dump or safety valve. Safety injection is initiated automatically by low pressurizer pressure. Operation of only one train of ECCS pumps is assumed. Boron solution at 1950 ppm enters the RCS providing sufficient negative reactivity to assure that the DNB design basis is met.

The transient is quite conservative with respect to cooldown, since no credit is taken for the energy stored in the system metal other than that of the fuel elements or energy stored in the other steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

15.2.13.3 Conclusions

The analysis has shown that the criteria stated earlier in this section is satisfied. Since the minimum DNBR remains above the limiting value, no consequential damage to the core or reactor system occurs.

The Accidental Depressurization of the Steamline event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.14 Spurious Operation of the Safety Injection System at Power

15.2.14.1 Identification of Causes and Accidents Descriptions

NOTE: The BIT terminology will be used here since this analysis of record includes the Boron Injection Tank (now the Centrifugal Charging Pump Injection Tank) and provides bounding results.

Spurious SIS operation at power could be caused by operator error or a false electrical actuating signal. A spurious signal in any of the following channels could cause this incident.

1. High containment pressure

2. Low pressurizer pressure

3. Low steam line pressure

Following the actuation signal, the suction of the centrifugal charging pumps is diverted from the volume control tank to the refueling water storage tank. The valves isolating the injection tank from the charging pumps and the injection header then automatically open. The charging pumps then provide RWST water through the header and injection line and into the cold legs of each loop. The safety injection pumps also start automatically but provide no flow when the RCS is at normal pressure. The passive injection system and the low head system also provide no flow at normal RCS pressure.

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

Case A Trip occurs at the same time spurious injection starts

Case B The reactor protection system produces a trip later in the transient.

For Case A the operator should determine if the spurious signal was transient or steady state in nature, i.e., an occasional occurrence or a definite fault. The operator must also determine if the safety injection system must be defeated for repair. For the former case the operator would stop the safety injection and bring the plant to the hot shutdown standby conditions. If the safety injection system must be disabled for repair, boration should continue through the normal boration mode and the plant brought to cold shutdown.

For Case B the reactor protection system does not produce an immediate trip and the reactor experiences a negative reactivity excursion causing a decrease in reactor power. The power unbalance causes a drop in T_{avg} and consequent coolant shrinkage. Pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load after the electro-hydraulic governor fully opens the turbine throttle valve. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressure trip or by manual trip.

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

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

Reference 17 addresses a PWR transient condition where a spoulous safety injection signal with an immediate reactor trip could challenge the non-escalation criteria. A spurious safety injection signal, a Condition II event, can become a Condition III event (Small Break LOCA), if the resulting safety injection flow fills the pressurizer and a pressurizer relief or safety valve opens, discharges water, and then fails to close. This condition can be precluded if operator actions to terminate safety injection flow can be completed before the pressurizer becomes water solid or the pressurizer relief and safety valves are fully qualified to close following a water discharge.

15.2.14.2 Analysis of Effects and Consequences

Method of Analysis

The spurious operation of the SIS system is analyzed by employing the detailed digital computer program LOFTRAN (Reference 4). This code is described in section 15.1.9.3.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Analysis of several cases shows the results are relatively independent of time to trip.

A transient is presented representing conditions at beginning of core life. Results at end of life are similar except that moderator feedback effects result in a slower transient.

The assumptions are:

1. Initial Operating Conditions - the initial reactor power and Reactor Coolant System temperatures are assumed at their maximum values consistent with the steady state full power operation including allowances for calibration and instrument errors.

2. Moderator and Doppler Coefficients of Reactivity - A low beginning of life moderator temperature coefficient was used. A low absolute value Doppler power coefficient was assumed.

3. Reactor Control - The reactor was assumed to be in manual control.

4. Pressurizer Heaters - Pressurizer heaters were assumed to be nonoperable in order to increase the rate of pressure drop.

5. Injection - At time zero, two charging pumps inject 20,000 ppm borated water from the BIT and 1950 ppm of borated water from the RWST water into the cold legs of each loop. Even though the BIT is functionally removed, Westinghouse analysis conservatively assumes that the BIT is still intact. This assumption lends to a larger power mismatch with subsequent RCS cooling and inventory shrinkage.

6. Turbine Load - Turbine load was assumed constant until the electro- hydraulic governor drives the throttle valve wide open. Then turbine load drops as steam pressure drops.

7. Reactor Trip - Reactor Trip was initiated by low pressurizer pressure assumed at a conservatively low value of 1775 psia.

To address the pressurizer overfill transient in Reference 17, a calculation of the pressurizer fill time for the spurious safety injection transient was performed using a Sequoyah plant specific evaluation model developed with the RELAP5/MOD2-B&W computer code. The calculation was based on a number of best-estimates assumptions and input parameters to provide a realistic pressurizer fill time for the transient condition (see Reference 18).

Results

The transient response is shown in Figures 15.2.14-1 and 15.2.14- 2. Nuclear power starts decreasing immediately due to boron injection but steam flow does not decrease until 15 seconds into the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The low pressure trip set point is reached at 64 seconds and rods start moving into the core at 66 seconds.

After trip, pressures and temperatures slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat. The time sequence of events during this transient is shown in Table 15.2-1.

The time required to fill the pressurizer was demonstrated to exceed the required operator action time of 15 minutes to terminate safety injection flow (see Reference 18).

15.2.14.3 Conclusions

Results of the analysis show that spurious safety injection with or without immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System.

DNB ratio is never less than the initial value. Thus there will be no cladding damage and no release of fission products to the reactor coolant system.

If the reactor does not trip immediately, the low pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident.

The degradation of a spurious safety injection event to a SBLOCA is precluded for Sequoyah since the time required to fill the pressurizer during the event is greater than the maximum 15 minute operator action time required to terminate the transient (see Reference 18).

The Spurious Operation of the Safety Injection at Power event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.2.15 References

1. W. C. Gangloff, "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486, May 1971.

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

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

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11. DIM-SQN-DC-V-27.4-21 (B38 940506 800).

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13. BAW-10156, R01, "LYNXT-Core Transient Thermal Hydraulic Program," August 1993.

14. BAW-10170, "Statistical Core Design for Mixing Vane Cores," December 1988.

15. Document No. 77-5016198, “Framatome ANP, Replacement Steam Generator Report for Sequoyah Unit 1.”

16. BAW-10180, “NEMO - Nodal Expansion Method Optimized,” March 1993.

17. NRC Regulatory Issue Summary 2005-29, Anticipated Transients that Could Develop into More Serious Events.

18. SQN-APS2-139, Time for the Pressurizer to Become Water Solid During a Spurious SI at Power Event.

19. Document No. 51-9076350-000, “SQN Non-LOCA Disposition of Events for ECCS Flow Modification.”

TABLE 15.2-1 (Sheet 1)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (Sec.)
Uncontrolled RCCA Withdrawal from a Subcritical Condition	Initiation of uncontrolled rod withdrawal	0
	Power range high neutron flux low setpoint reached	13.4
	Peak nuclear power occurs	13.7
	Rods begin to fall into core	13.9
	Minimum DNBR Occurs	15.9
	Peak heat flux occurs	16.2
	Peak average fuel temperature occurs	16.5
Uncontrolled RCCA Withdrawal at Power	Initiation of uncontrolled RCCA withdrawal at a large reactivity insertion rate (75 pcm/sec)	0
1. Case A	Power range high neutron flux high trip point reached	0.9
	Rods begin to fall into core	1.4
	Minimum DNBR occurs	3.2
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (0.82 pcm/sec)	0
	Over temperature ΔT reactor trip signal initiated	590.2
	Rods begin to fall into core	598.2
	Minimum DNBR occurs	598.3

TABLE 15.2-1 (Sheet 2)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (Sec.)
Uncontrolled Boron Dilution		
1. Dilution during refueling and startup	Dilution begins	0
	Operator isolates source of dilution; minimum margin to criticality occurs	refueling - precluded by admin controls) startup - >1140
2. Dilution During Full Power Operation		
a. Automatic Reactor Control	Shutdown margins lost	2520
b. Manual Reactor Control	Dilution begins	0
	Reactor trip setpoint reached for over temperature ΔT	<120
	Shutdown margin is lost (if dilution continued after trip)	>2400
3. Dilution Following Shutdown (Equilibrium Xe Case)		
	Reactor Trip	0
	Reactor Power = 7.5% of nominal 80 sec. reactor period - Intermediate NIS reads ~10% power	10
	Source Range NIS Available	930
	Source Range NIS no longer decreasing (without dilution event, flux would stabilize at this point- an 18 day half life decay of	

TABLE 15.2-1 (Sheet 3)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (Sec.)
	flux would be normal - Source Range Count Rate would change from 200 cps to 197 cps).	1,250
	High Flux at Shutdown Alarm nominally 3 times stabilized flux level. For this example, the value is » 700 cps.	1,250
	High pressurizer Level Trip and Alarm	1,800
	Source Range High Flux at Shutdown Alarm	7,400
	Keff = 1.0	12,960
4. Dilution Following Shutdown (BOL Clean Case)		
	Reactor Trip	0
	Source Range NIS available and no longer decreasing count rate. Operator need not reset count rate since refueling/previous shutdown value conservative	60
	High Pressurizer Level Trip and Alarm	1,800
	Source Range High Flux at Shutdown Alarm	3,500
	Keff = 1.0	5,220
Partial Loss of Forced Reactor Coolant Flow		
All loops operating, two pumps coasting down	Coastdown begins	0
	Low flow reactor trip	1.4
	Rods begin to drop	2.4
	Minimum DNBR occurs	3.8

TABLE 15.2-1 (Sheet 4)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (Sec.)
Startup of an Inactive Reactor Coolant Loop	Initiation of pump startup	0
	Power reaches P-8 trip setpoint	2.79
	Rods begin to drop	3.29
	Minimum DNBR occurs	4.0
Loss of External Electrical Load Without Pressurizer Control	Loss of electrical load	0
	High pressurizer pressure reactor reactor trip set point reached	5.1
	Rods begin to drop	7.1
	Minimum DNBR occurs	8.6
	Peak primary pressure reached	9.8
	Main steam safety valves lift	10.4
	Peak secondary pressure reached	16.0

TABLE 15.2-1 (Sheet 5)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (Sec.)
Loss of Normal Feedwater	Main feedwater flow stopped	0.0
	Low-Low steam generator water level trip-setpoint reached	8.6
	Rods begin to drop	10.6
	Peak water level in the pressurizer occurs	14.6
	Two steam generators begin to receive auxiliary feedwater from one motor-driven auxiliary feedwater pump	70.7
	Cold auxiliary feedwater is delivered to the steam generators	507.7
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~6240

TABLE 15.2-1 (Sheet 6)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (Sec.)
Loss of Off-Site Power to the Station Auxiliaries	Main feedwater flow stopped	0
	Low-Low steam generator water level trip setpoint reached	11
	Rods begin to drop	13
	Power lost to the reactor coolant pumps	13
	Two steam generators begin to receive auxiliary feedwater from one motor-driven auxiliary feed-water pump	71
	Cold auxiliary feedwater is delivered to the steam generators	508
	Core decay heat decreases to auxiliary feedwater heat removal capacity	1430
	Peak water level in pressurizer occurs	3027

(1) DNBR does not decrease below its initial value

TABLE 15.2-1 (Sheet 7)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time (Sec.)
Excessive Feedwater at Full Load in One Steam Generator - With Analog Feedwater Control	One main feedwater regulator valve fails fully open	0
	Minimum DNBR occurs	49.34
	High-high steam generator water level signal generated	50.27
	Turbine trip occurs due to high-high steam generator water level	52.78
	Reactor trip occurs due to turbine trip	53.79
	Feedwater isolation valves fully closed	65.78
Excessive Feedwater at Full Load in Four Steam Generators - With Analog Feedwater Control	Four main feedwater regulator valves fail fully open	0
	Minimum DNBR occurs	170.00
	High-High steam generator water level signal generated	209.45
	Turbine trip occurs due to high-high steam generator water level	211.96
	Rod motion occurs due to turbine trip	212.97
	Feedwater isolation valves fully closed	224.97

TABLE 15.2-1 (Sheet 8)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (Sec.)
Excessive Feedwater at Full Load in One Steam Generator - Digital Feedwater Control	All main feedwater regulator valves and all regulating bypass valves in a single loop fail fully open	0
	High-High steam generator water level signal generated	37.39
	Turbine trip occurs due to High-High steam generator water level	39.90
	Minimum DNBR occurs	40.5
	Reactor trip occurs due to turbine Trip	40.91
	Feedwater isolation valves fully closed	52.91
Excessive Load Increase		
1. Manual Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	200
2. Manual Reactor Control (EOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	50
3. Automatic Reactor Control (BOL)	10% step load increase	0
	Equilibrium conditions reached	(3)
4. Automatic Reactor Control (EOL)	10% step load increase	0
	Equilibrium conditions reached (approximate times only)	50

(3) Did not reach equilibrium within the time scale of Figure 15.2.11-2

TABLE 15.2-1 (Sheet 9)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (Sec.)
	Accidental depressurization of the Reactor Coolant System	Inadvertent
	Opening of one RCS Safety Valve	0
	Reactor Trip on Overtemperature ΔT	33.8
	Rods begin to drop	35.3
	Minimum DNBR occurs	35.8
Accidental depressurization of the Main Steam System	Inadvertent Opening of one main steam safety or relief valve	0
	Pressurizer empties	161
	Boron reaches core	227
	Criticality attained	305
Inadvertent Operation of ECCS during Power Operation	Charging pumps begin injecting borated water	0
	Low pressure trip point reached	64
	Rods begin to drop	66

TABLE 15.2.7-1

INITIAL CONDITIONS FOR A COMPLETE LOSS OF LOAD FROM 52% POWER*

	52% POWER
Core Power, Mwt	1780
Thermal Design Flow (TOTAL) GPM	354000
Reactor Coolant Temperature	
Vessel Outlet, °F	587
Vessel Inlet, °F	551
Steam Generator Steam	
Temperature, °F	542
Pressure, PSIA	977

*This power is based on 3423 MW, NSSS thermal power output and includes thermal power generated by the RCPs.

TABLE 15.2.7-2

TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP AT 52% POWER WITH PRESSURIZER PRESSURE CONTROL

	EVENT	52% POWER
1. Minimum Feedback (BOL)		
	Turbine Trip	0
	Initiation of steam release from steam generator safety valves	11
	Peak pressurizer pressure occurs	12
	Fast bus transfer failure, flow coast-down begins	30
	Low flow reactor trip occurs and rods begin to fall	33
2. Maximum Feedback (EOL)		
	Turbine Trip	0
	Initiation of steam release from stream generator safety valves	11
	Peak pressurizer pressure occurs	12
	Fast bus transfer failure, flow coastdown begins	30
	Low flow reactor trip occurs and rods begin to fall	33

TABLE 15.2.7-3

TIME SEQUENCE OF EVENTS FOR A TURBINE TRIP AT 52% POWER WITHOUT PRESSURIZER PRESSURE CONTROL

	EVENT	52% POWER
1. Minimum Feedback (BOL)		
	Turbine Trip	0
	Initiation of steam release from steam generator safety valves	11
	High pressurizer pressure trip occurs, and rods begin to fall	12.4
	Peak pressurizer occurs	14
	Fast bus transfer failure, flow coastdown begins	30
2. Maximum Feedback (EOL)		
	Turbine Trip	0
	Initiation of steam release from steam generator safety valves	11
	High pressurizer pressure trip occurs, and rods begin to fall	13
	Peak pressurizer pressure occurs	15.5
	Fast bus transfer failure, flow coastdown begins	30

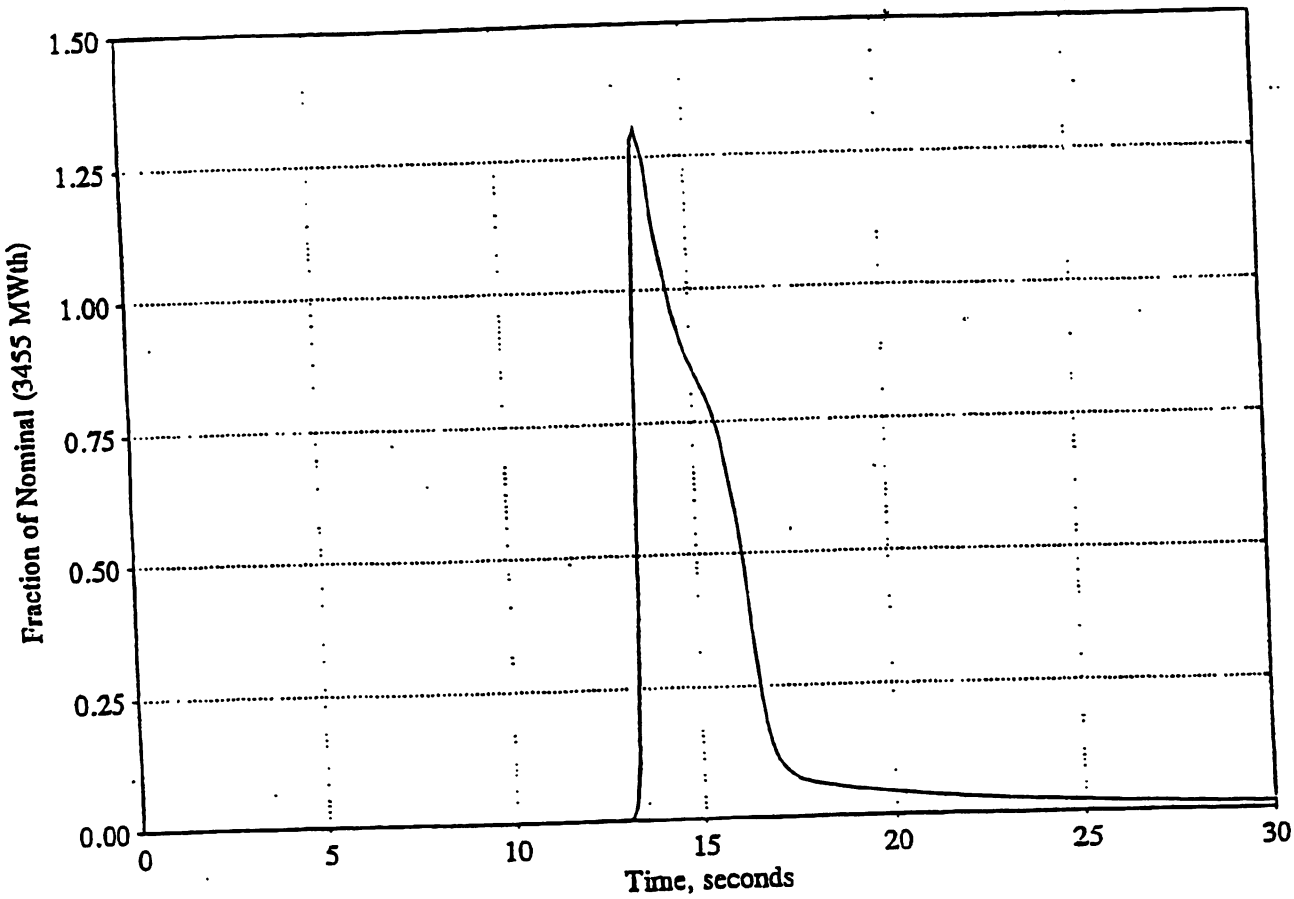


Figure 15.2.1-1 RCCA Withdrawal from Subcritical - Nuclear Power

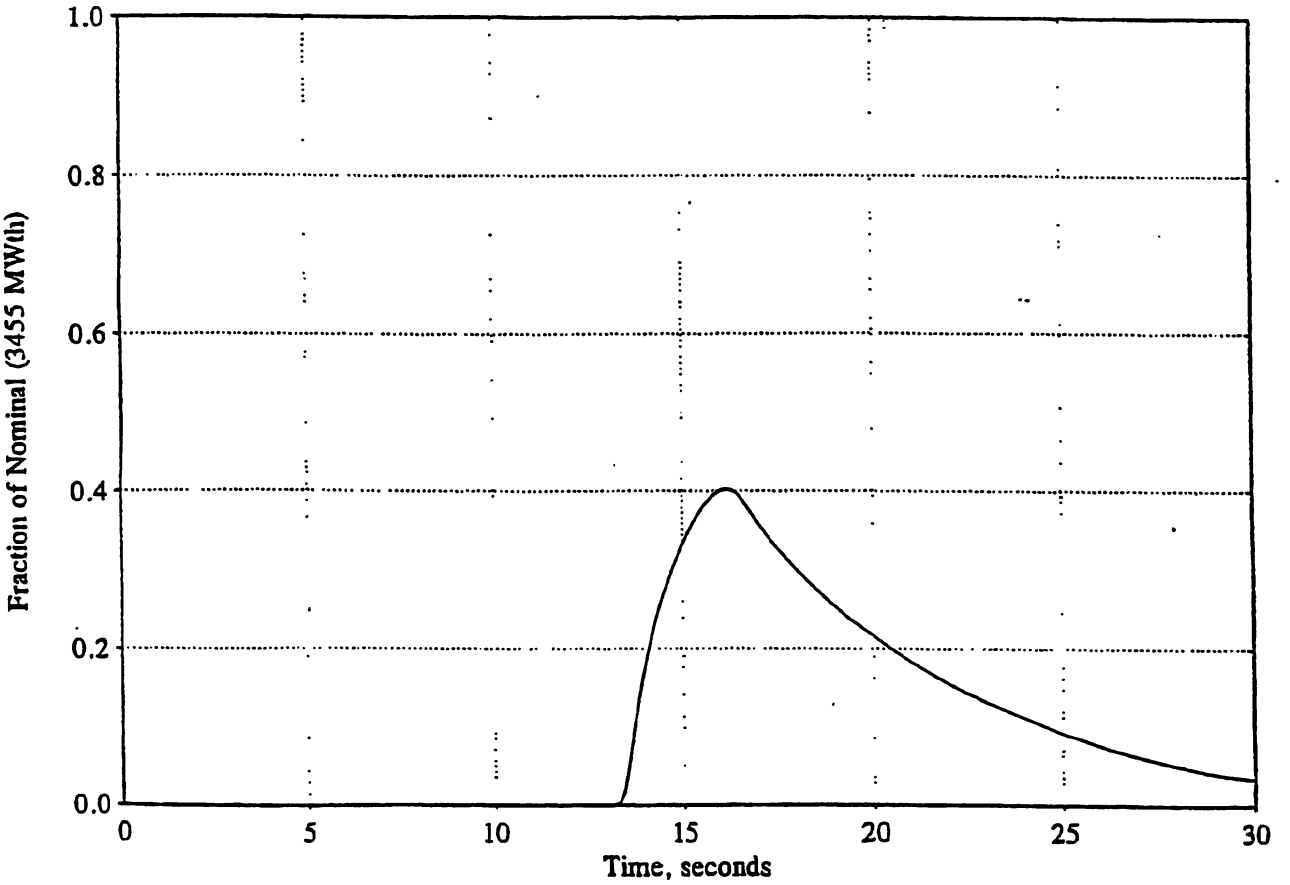


Figure 15.2.1-2 RCCA Withdrawal from Subcritical - Thermal Power

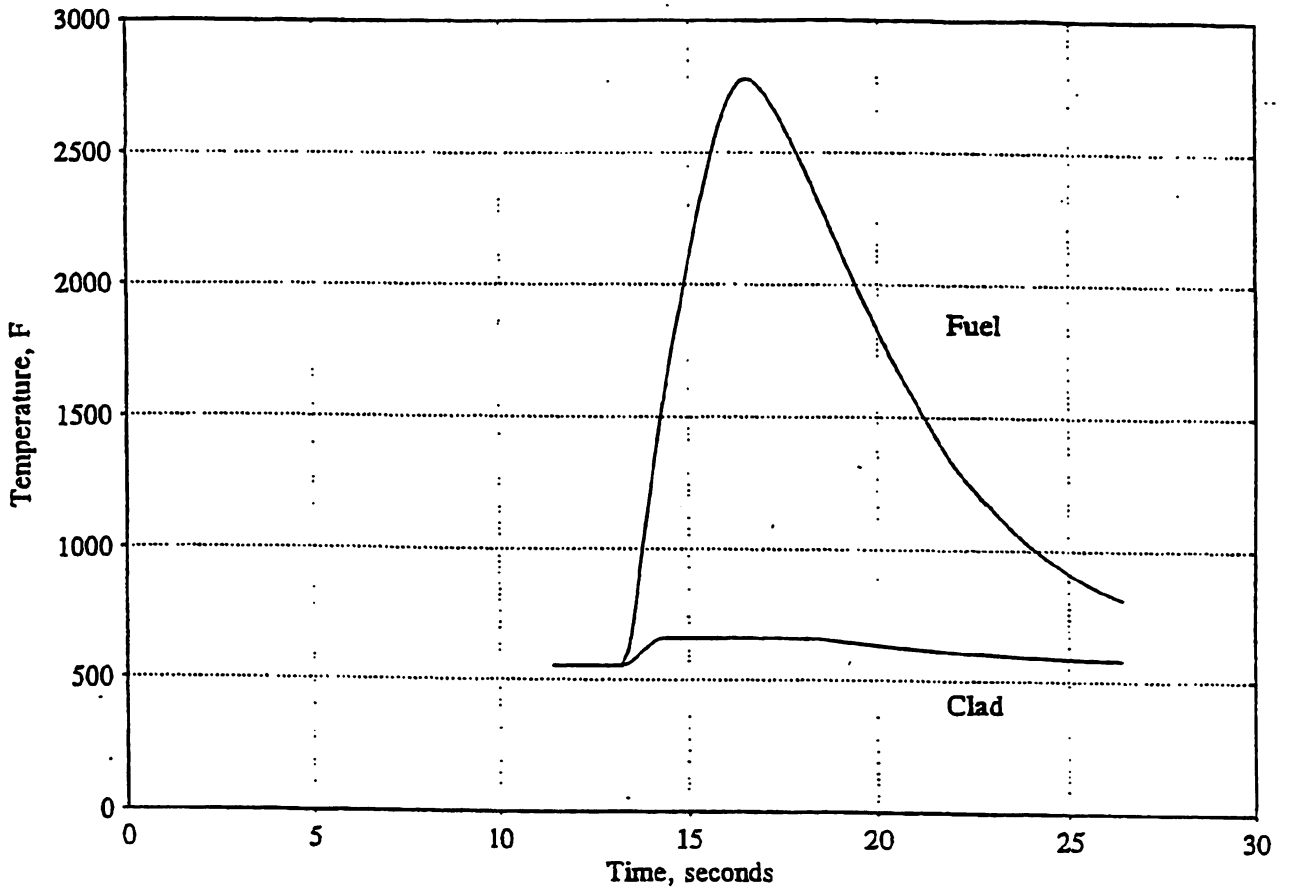
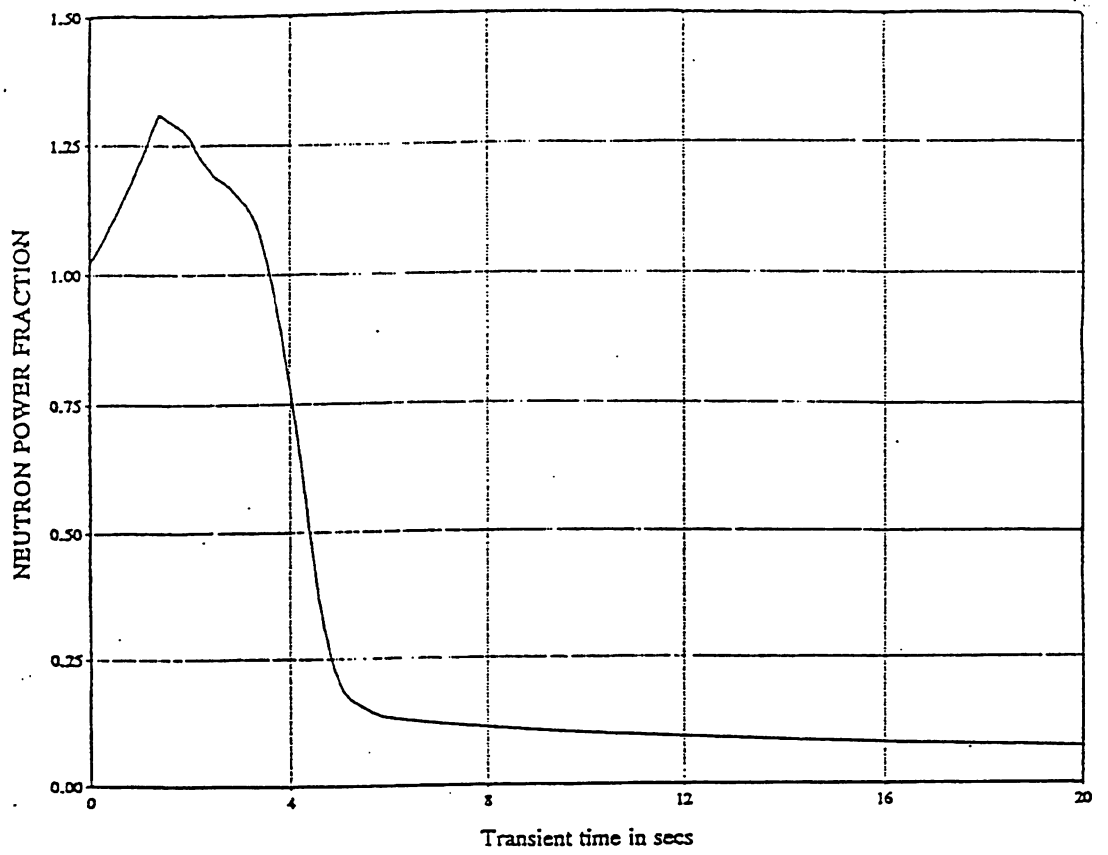


Figure 15.2.1-3 RCCA Withdrawal from Subcritical - Fuel/Clad Temperature



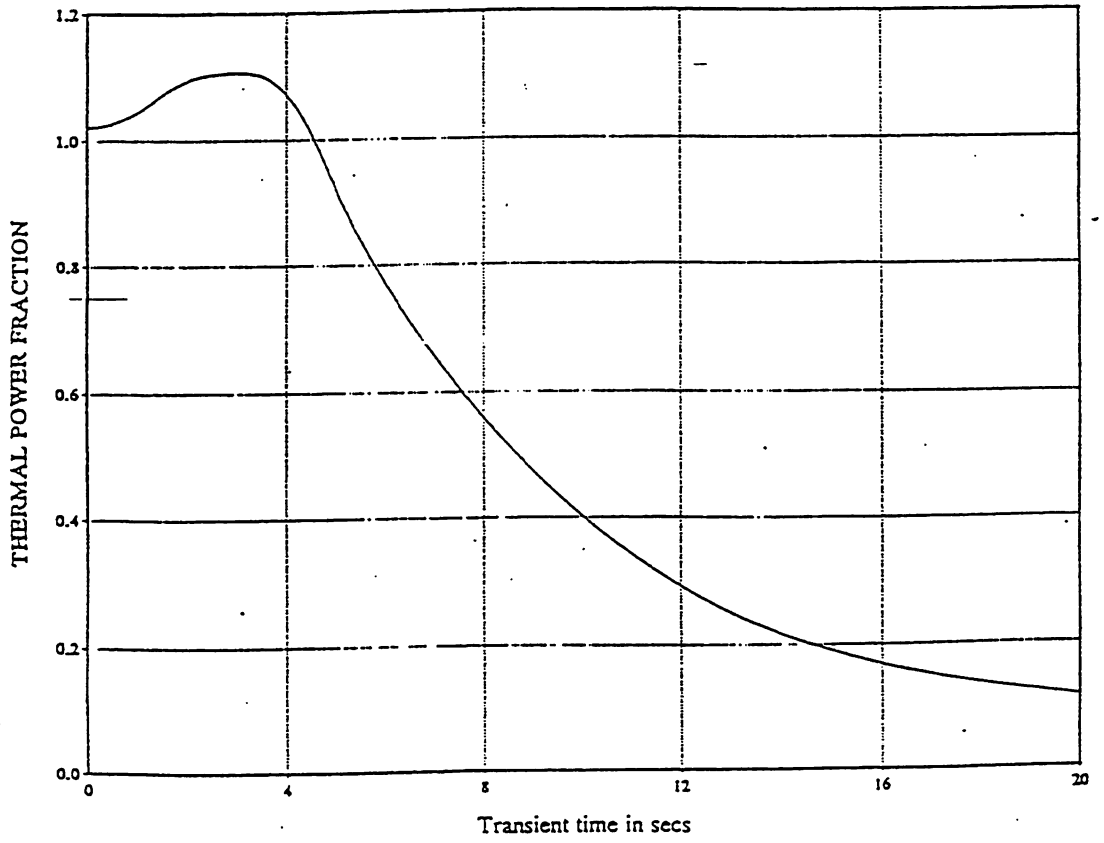
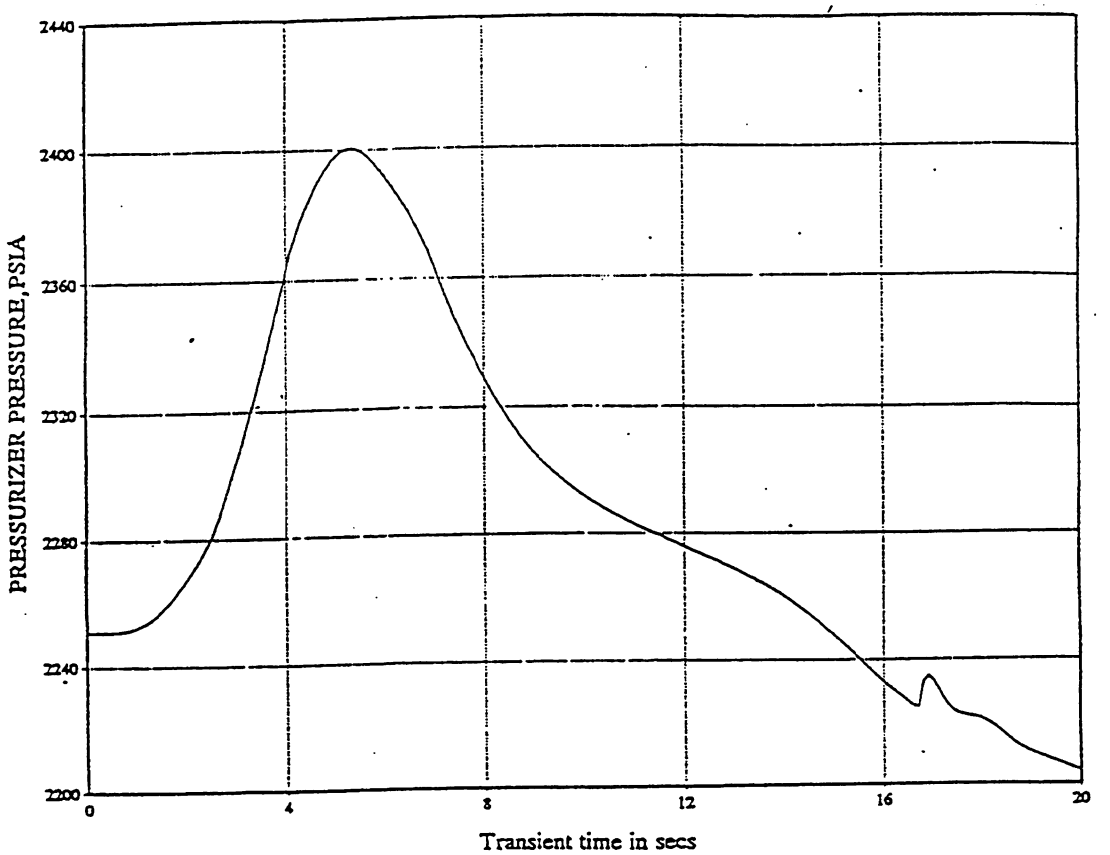


Figure 15.2.2-1
Uncontrolled Rapid Rod Withdrawal From
100% Power Terminated by
High Neutron Flux Trip
Neutron Power, Thermal Power
vs Time



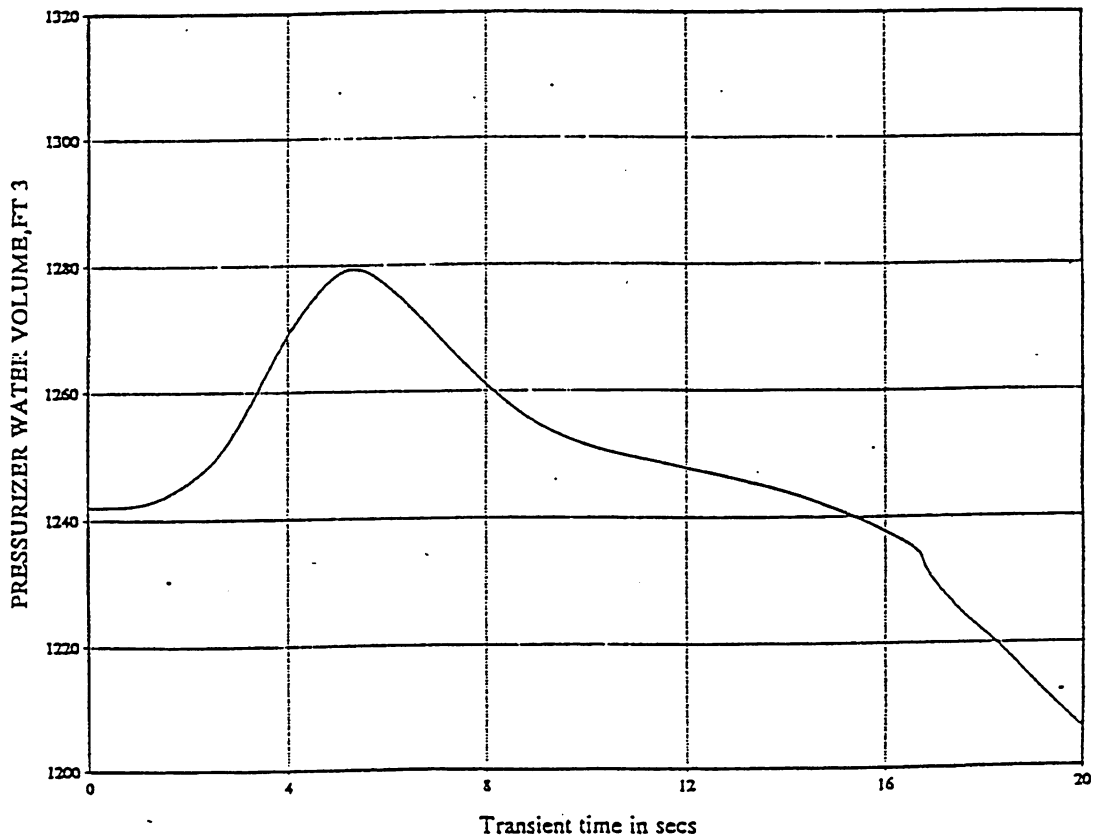
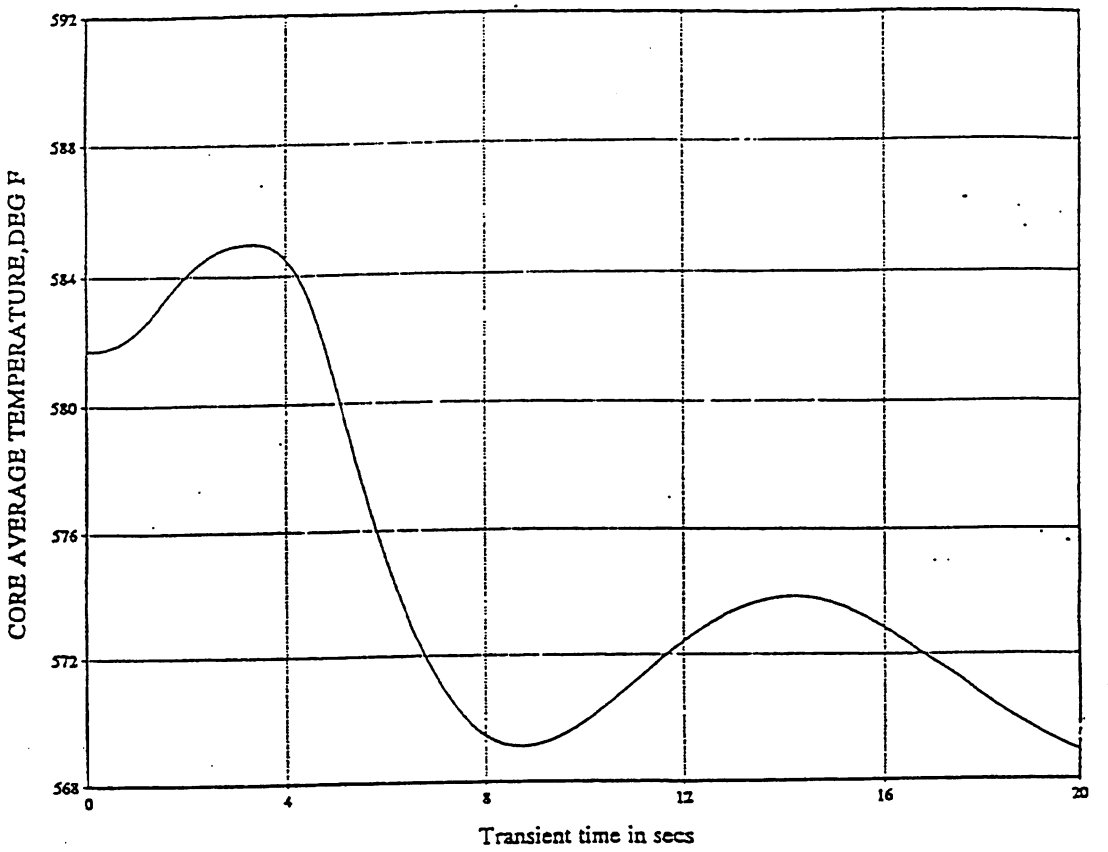


Figure 15.2.2-2
Uncontrolled Rapid Rod Withdrawal Fror
100% Power Terminated by
High Neutron Flux Trip
Pressurizer Pressure,
Pressurizer Water Volume
vs Time



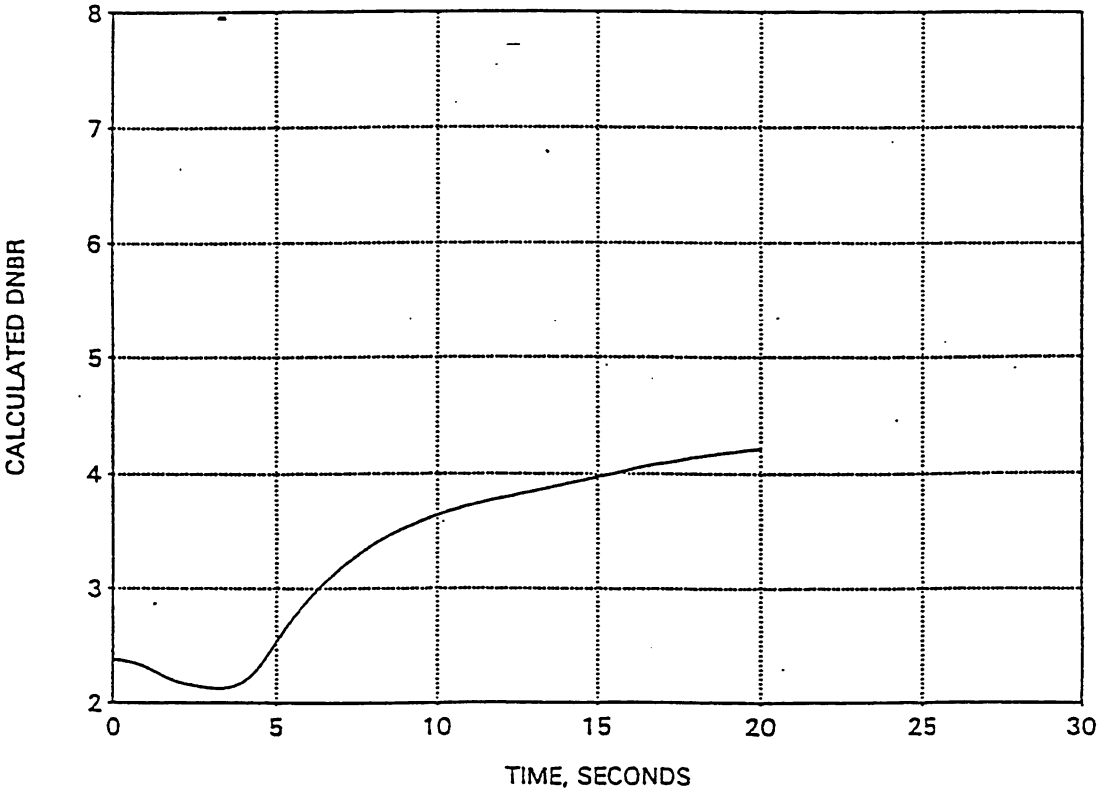
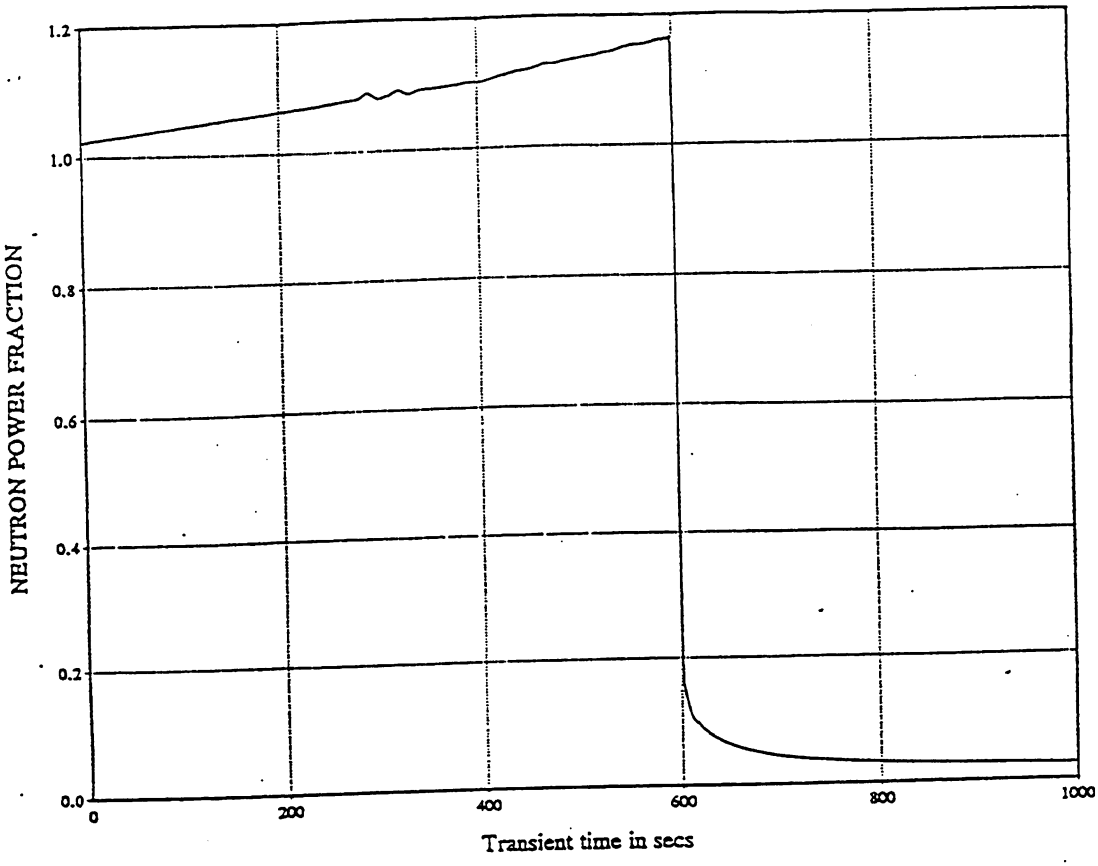


Figure 15.2.2-3
Uncontrolled Rapid Rod Withdrawal From
100% Power Terminated by
High Neutron Flux Trip
Vessel Average Temperature, DNBR
vs Time



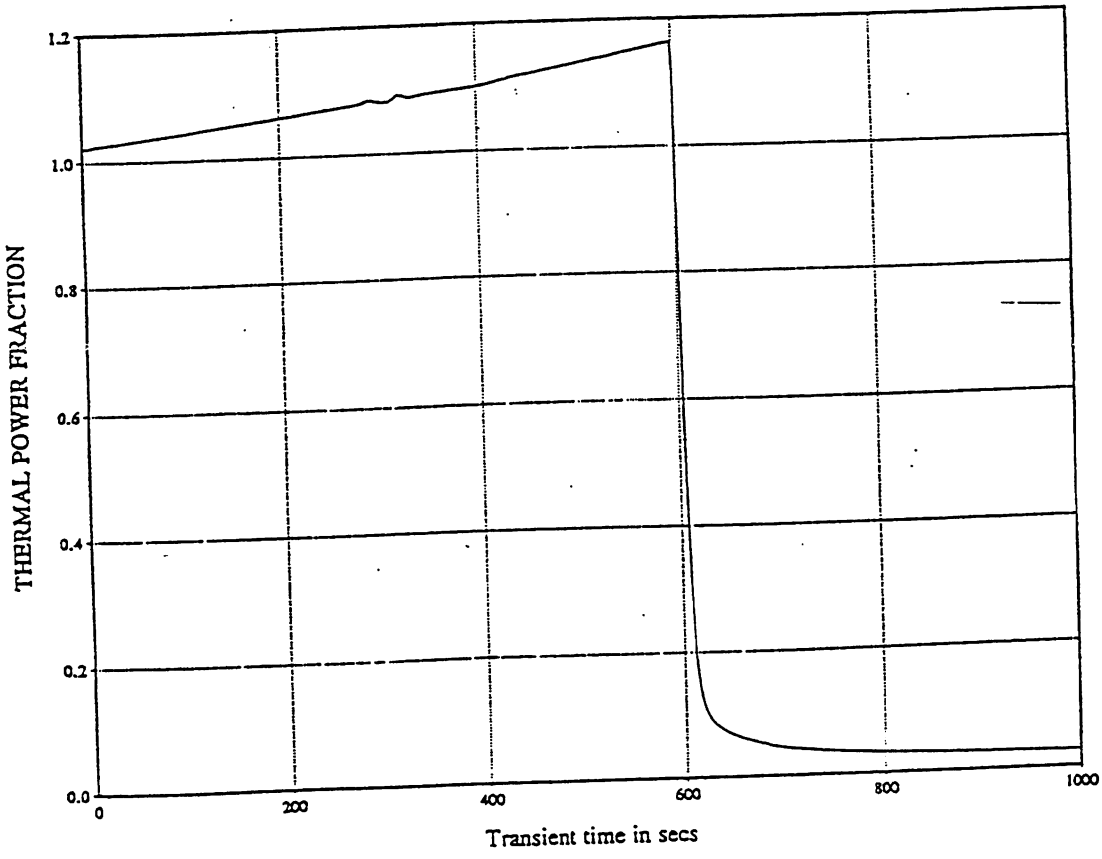
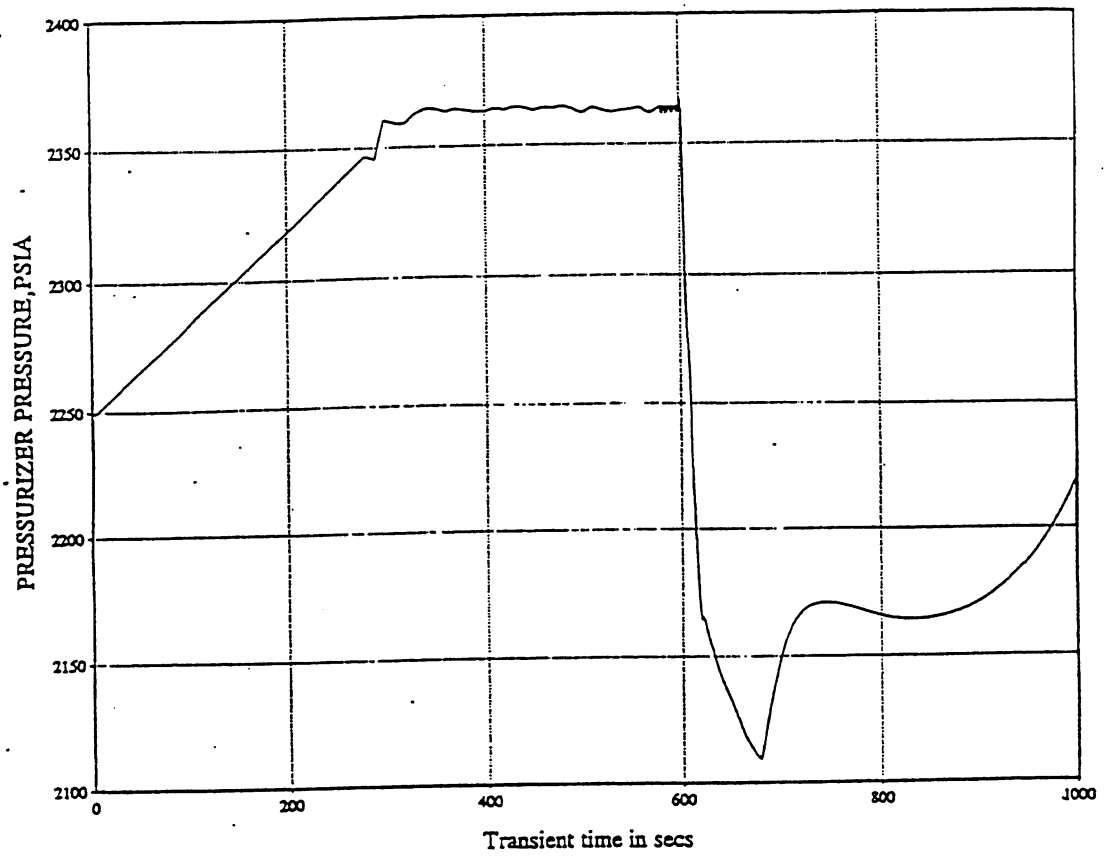


Figure 15.2.2-4
Uncontrolled Slow Rod Withdrawal From
100% Power Terminated by
Overtemperature ΔT Trip
Neutron Power, Thermal Power
vs Time



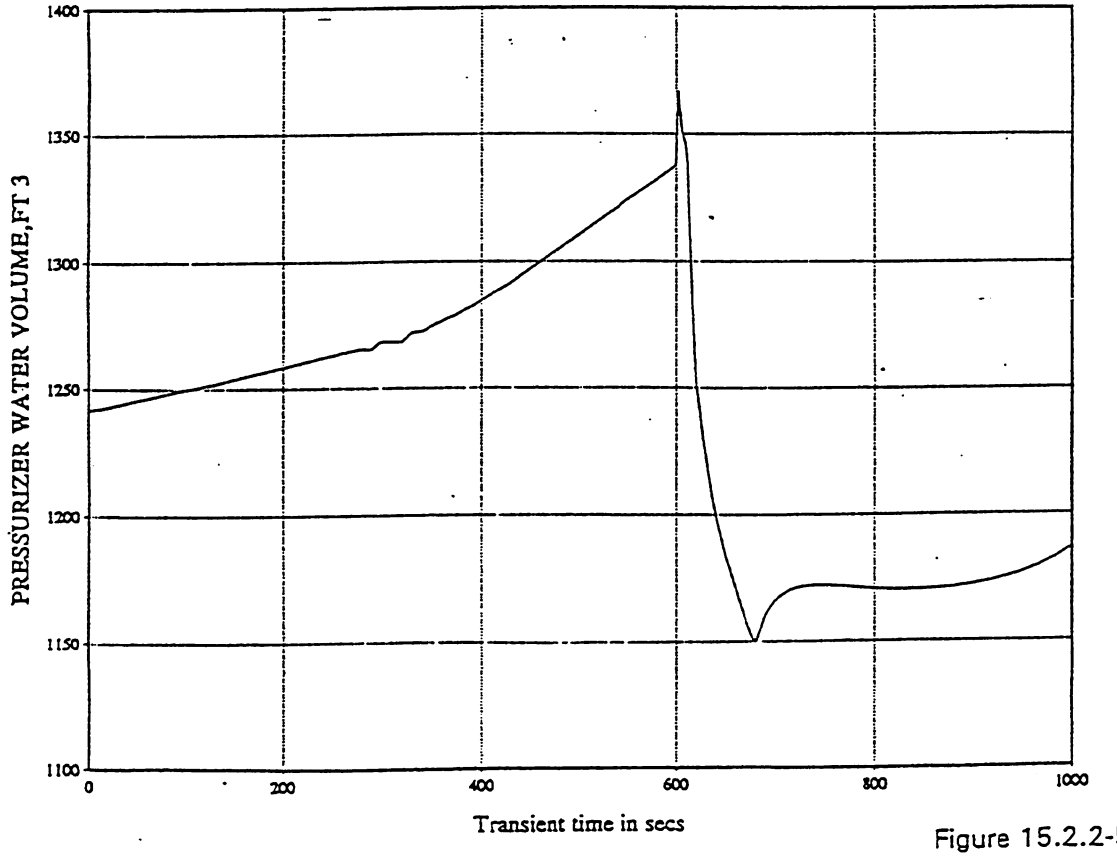
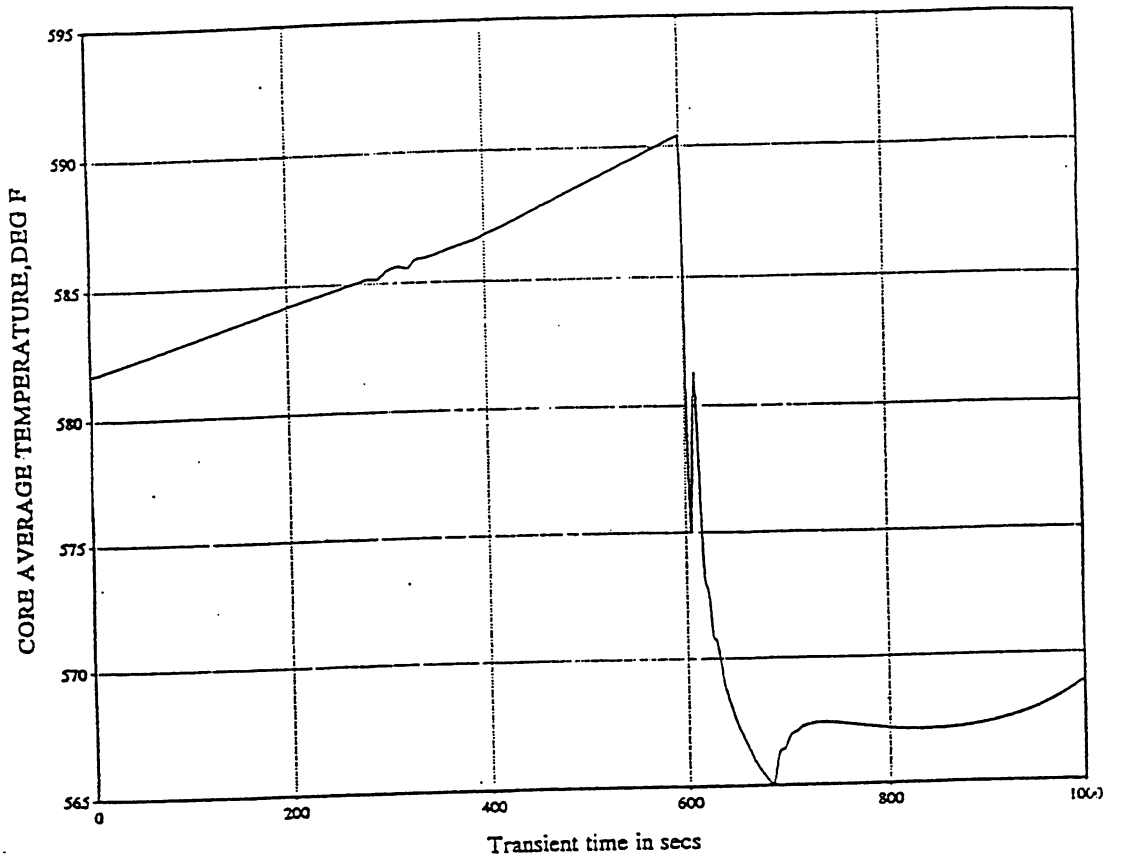


Figure 15.2.2-5
Uncontrolled Slow Rod Withdrawal From 100% Power Terminated by Overtemperature ΔT Trip Pressurizer Pressure, Pressurizer Water Volume vs Time



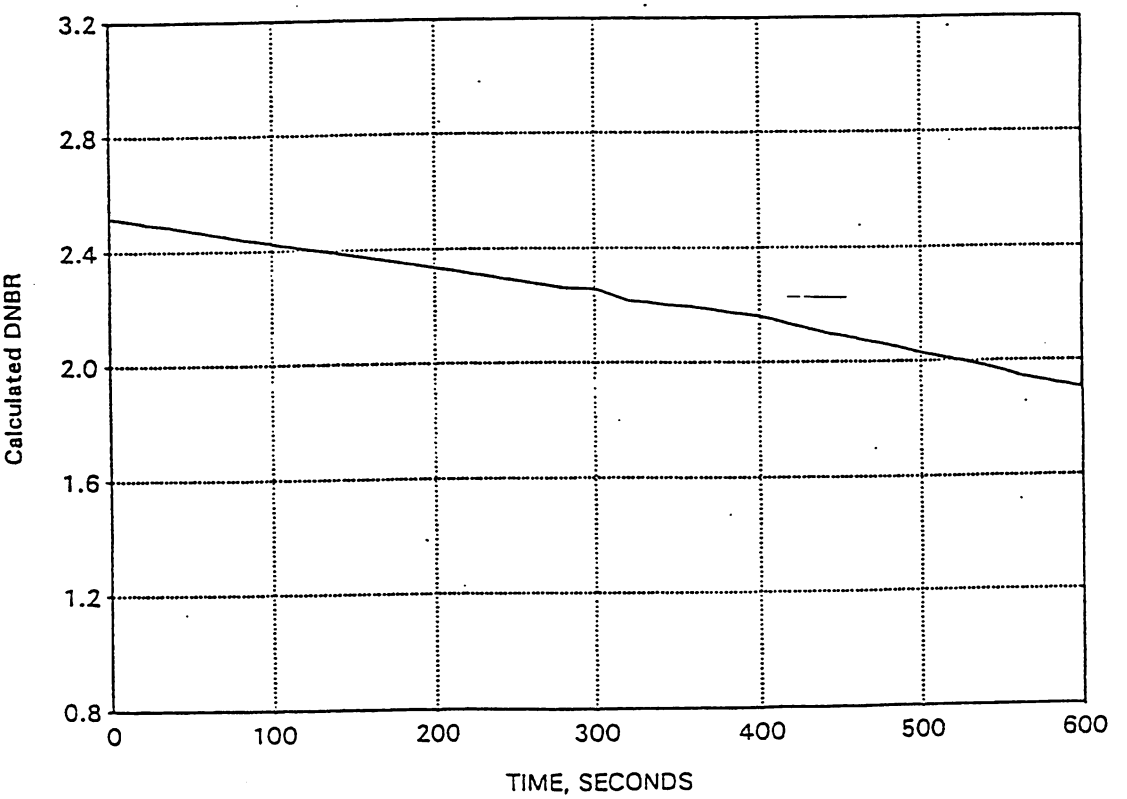


Figure 15.2.2-6
Uncontrolled Slow Rod Withdrawal From
100% Power Terminated by
Overtemperature ΔT Trip
Vessel Average Temperature, DNBR
vs Time

Minimum DNBR vs Rod Withdrawal Rate

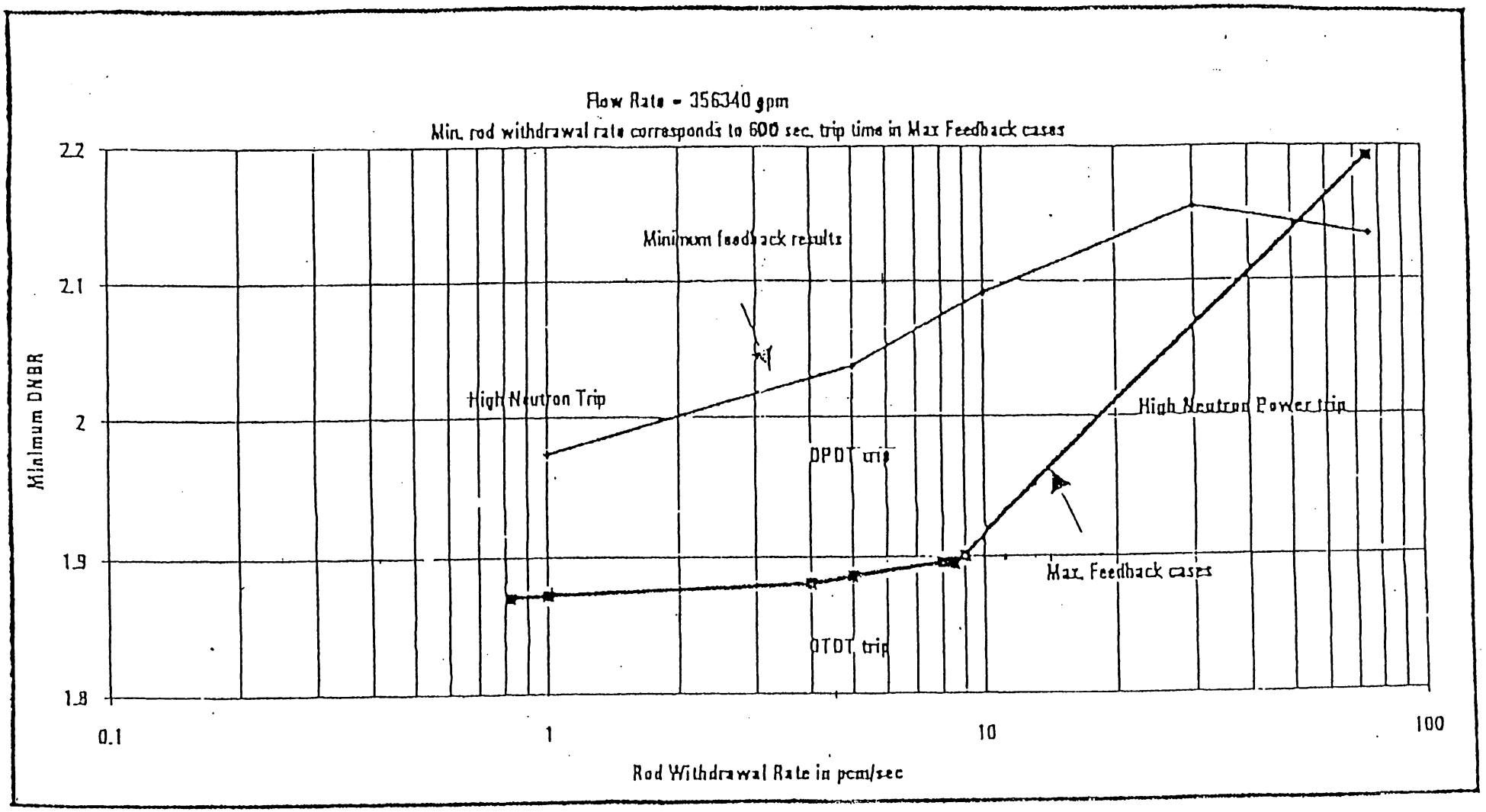
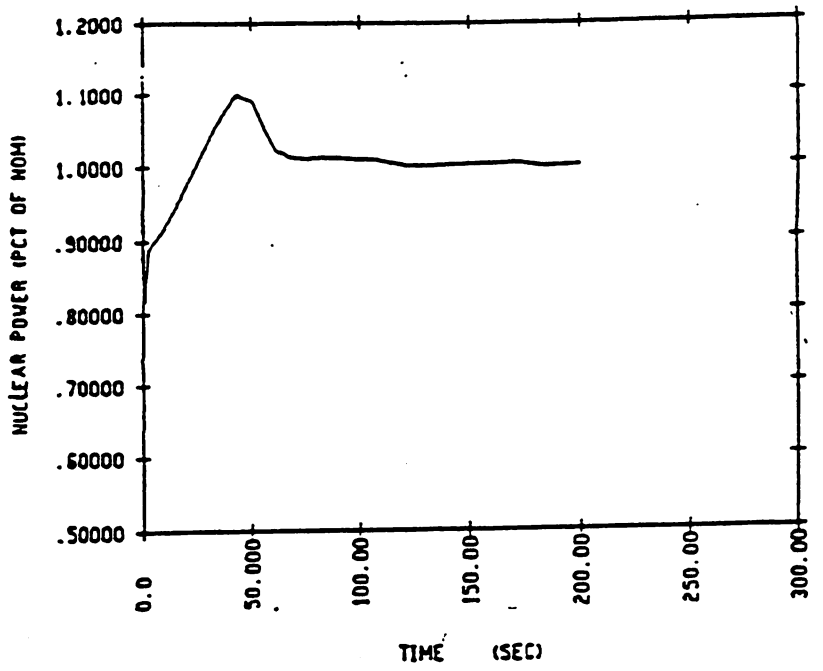


Figure 15.2.2-7

Effect of Reactivity Insertion Rate
on Minimum DNBR for a
Rod Withdrawal Accident at
100 % Power

Best Available Historical Image



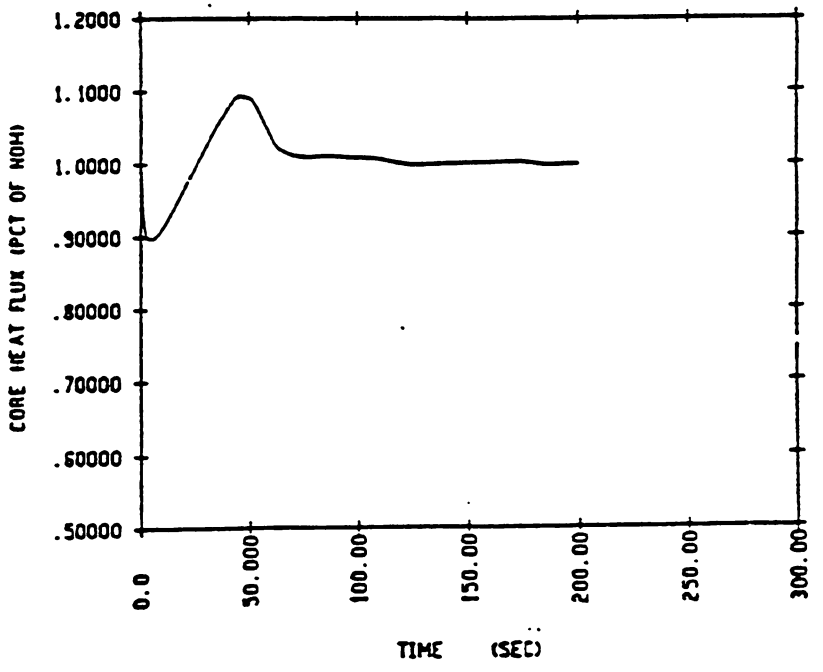
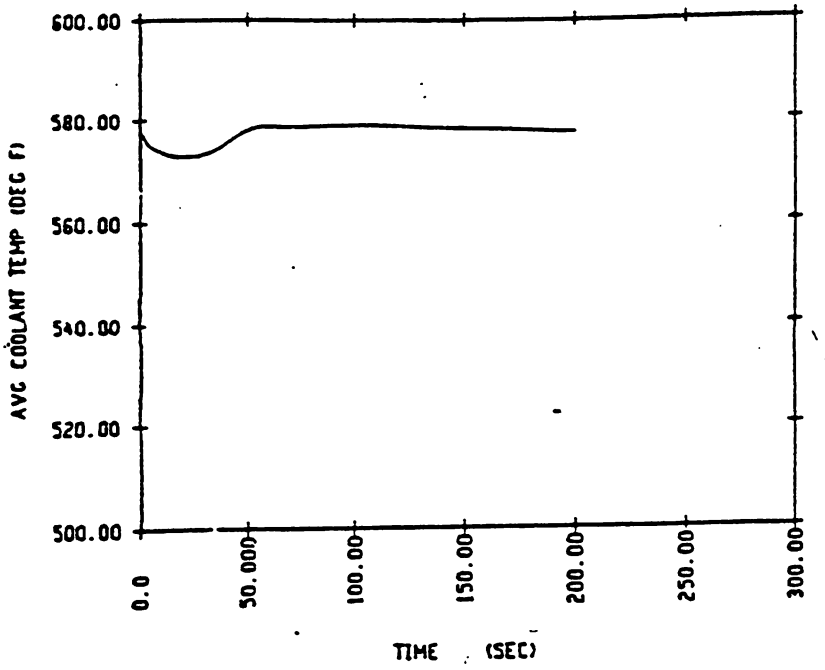


Figure 15.2.3-1 Nuclear Power and Core Heat Flux Transient Response to Dropped Rod Cluster Control Assembly

Best Available Historical Image



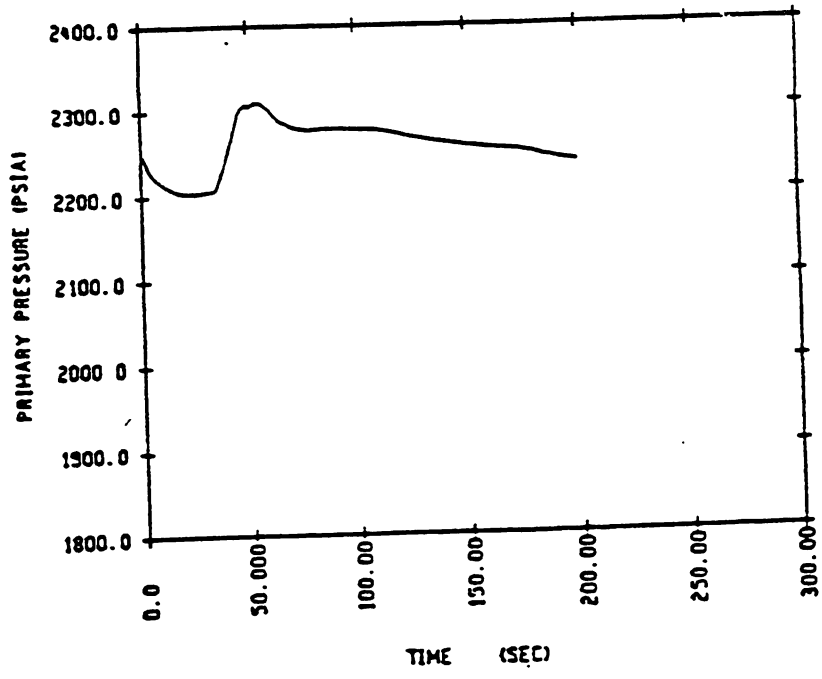


Figure 15.2.3-2 Coolant Temperature and Primary Pressure Transient Response to Dropped Rod Cluster Control Assembly

RCS BORON CONCENTRATION VS TIME

BOL EQUILIBRIUM XE AND CLEAN INITIAL CONDITIONS

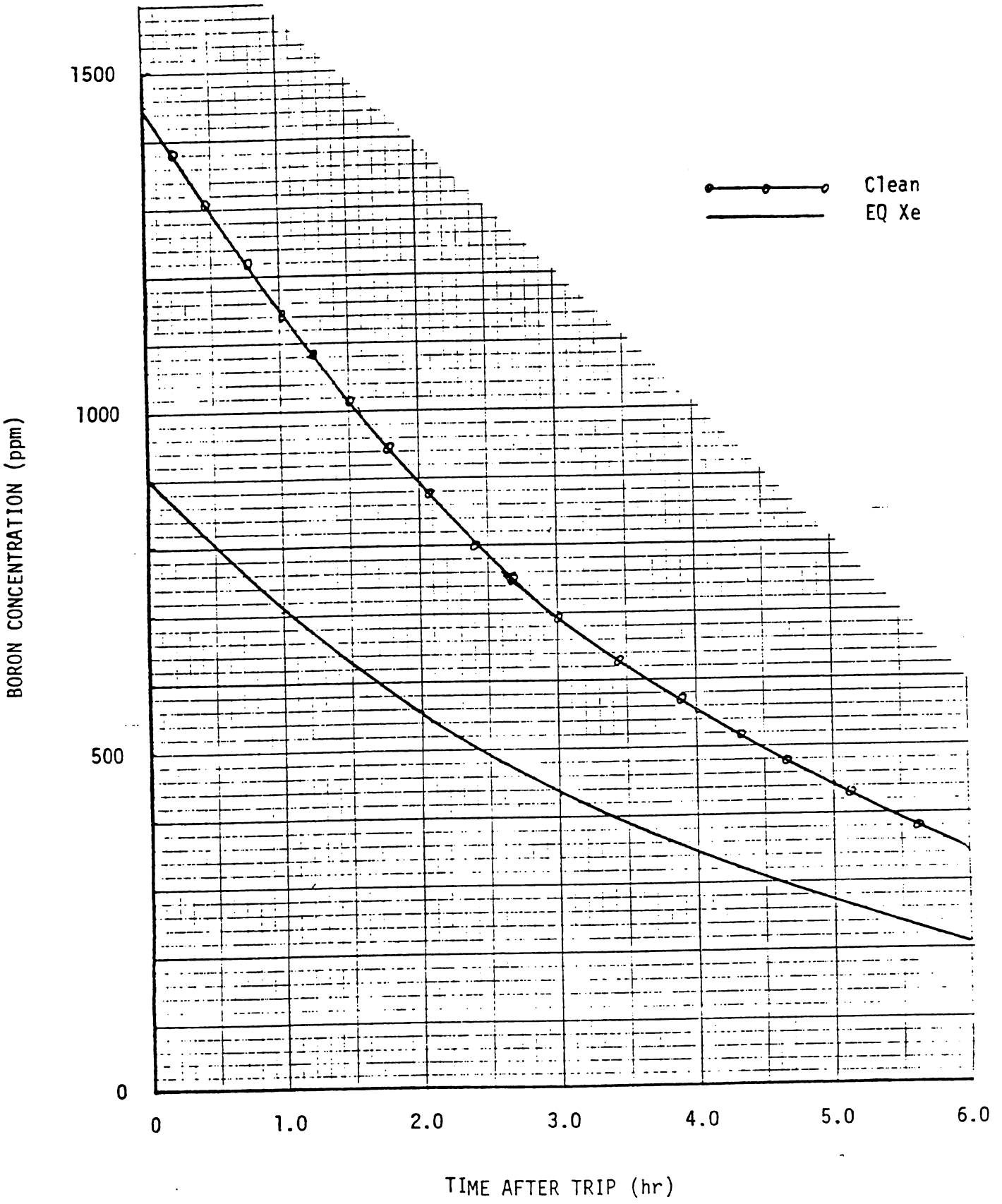


FIGURE 15.2.4-1

Keff vs Time

Equilibrium Xe and Clean Condtions
Following Trip from Full Power

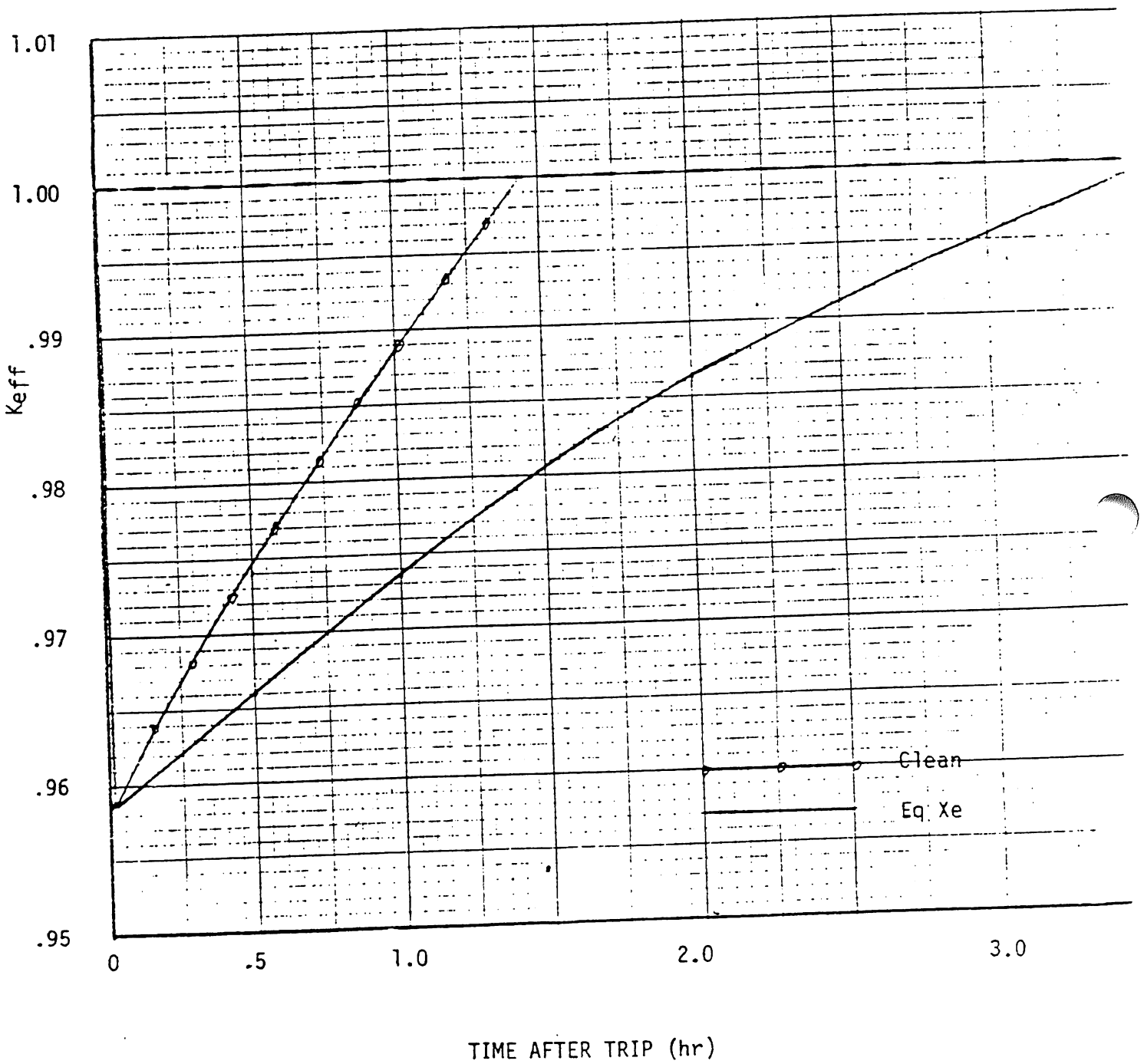
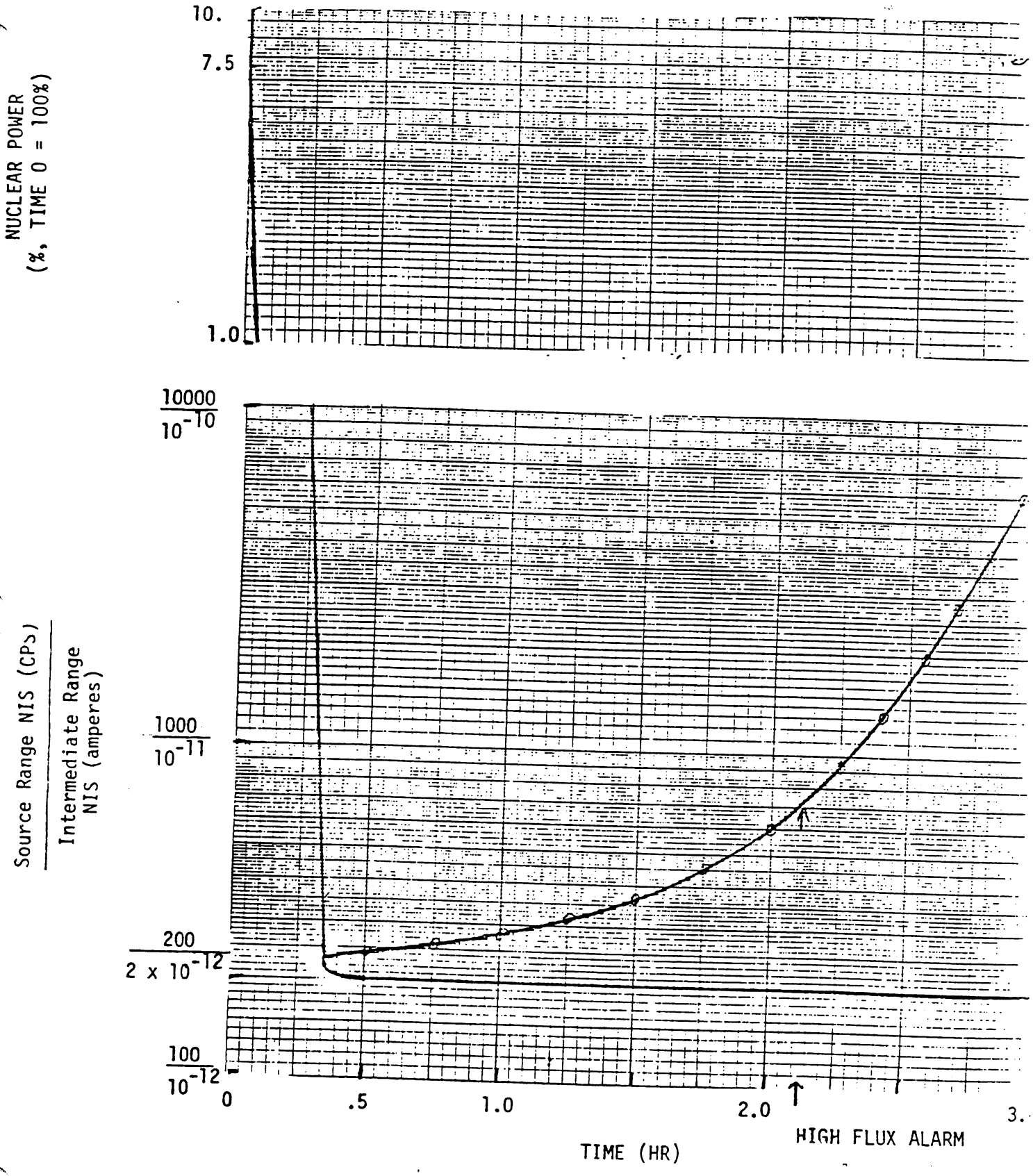


FIGURE 15.2.4-2

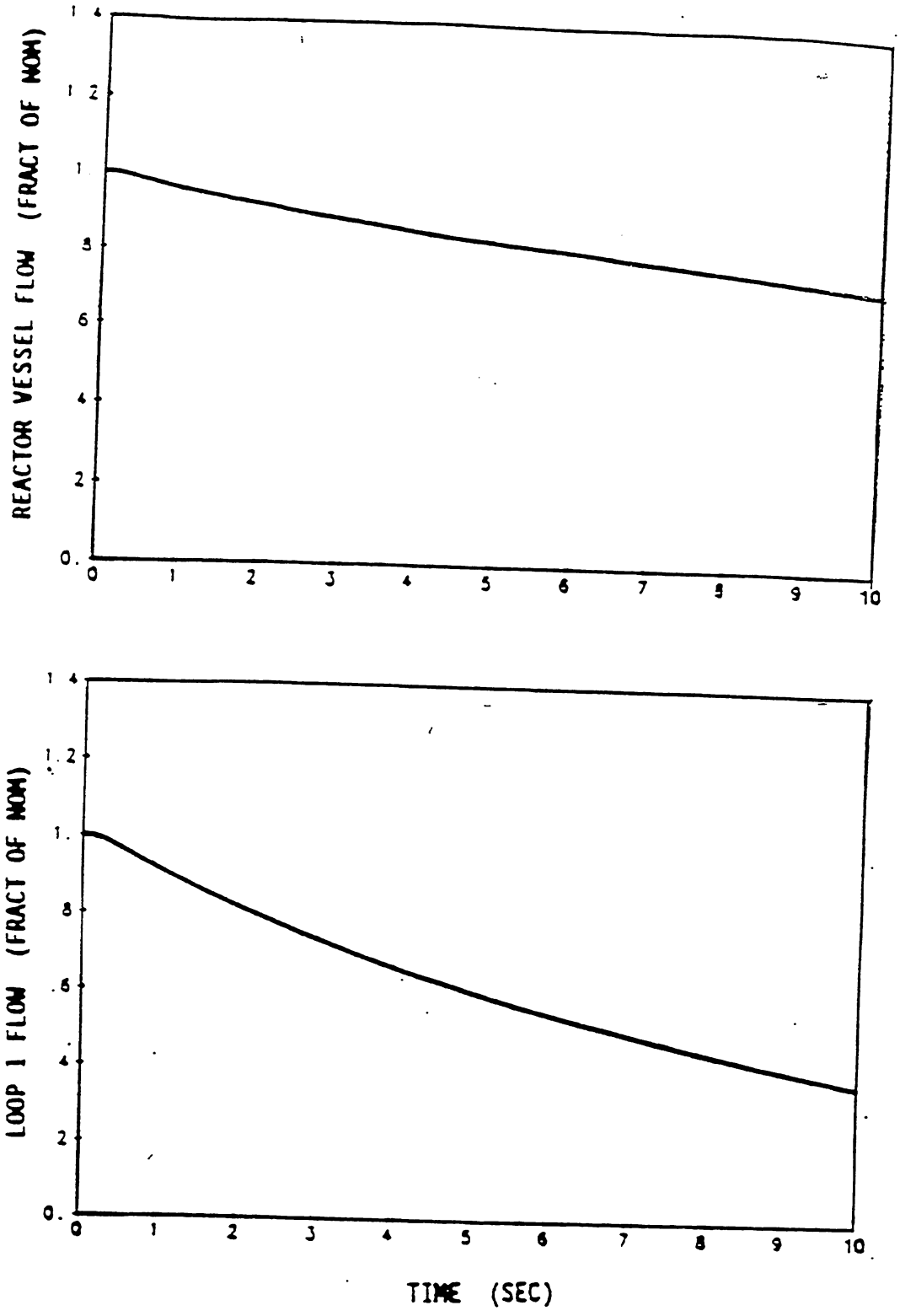
NUCLEAR POWER (DETECTOR INDICATION) AFTER TRIP VS TIME



EQ. Xe, no dilution

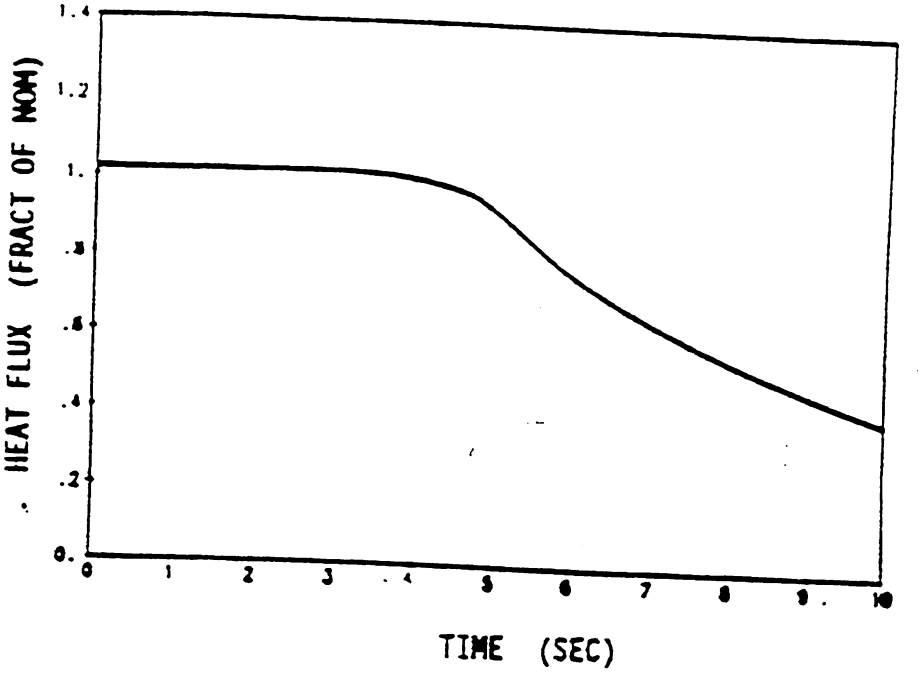
Figure 15.2.4-3

Dilution event occuring simultaneously with flux reduction due to trip



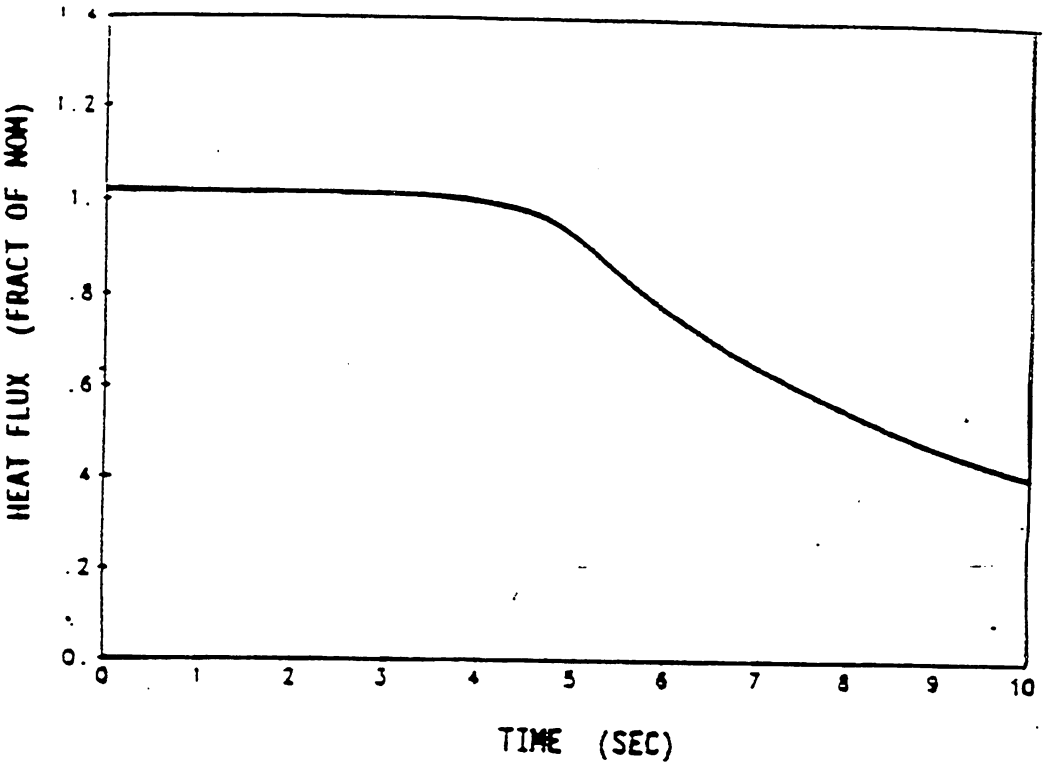
SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Partial Loss of Forced Reactor Coolant Flow, Reactor Vessel and Loop Flow vs. Time
Figure 15.2.5-1

Revised by Amendment 10



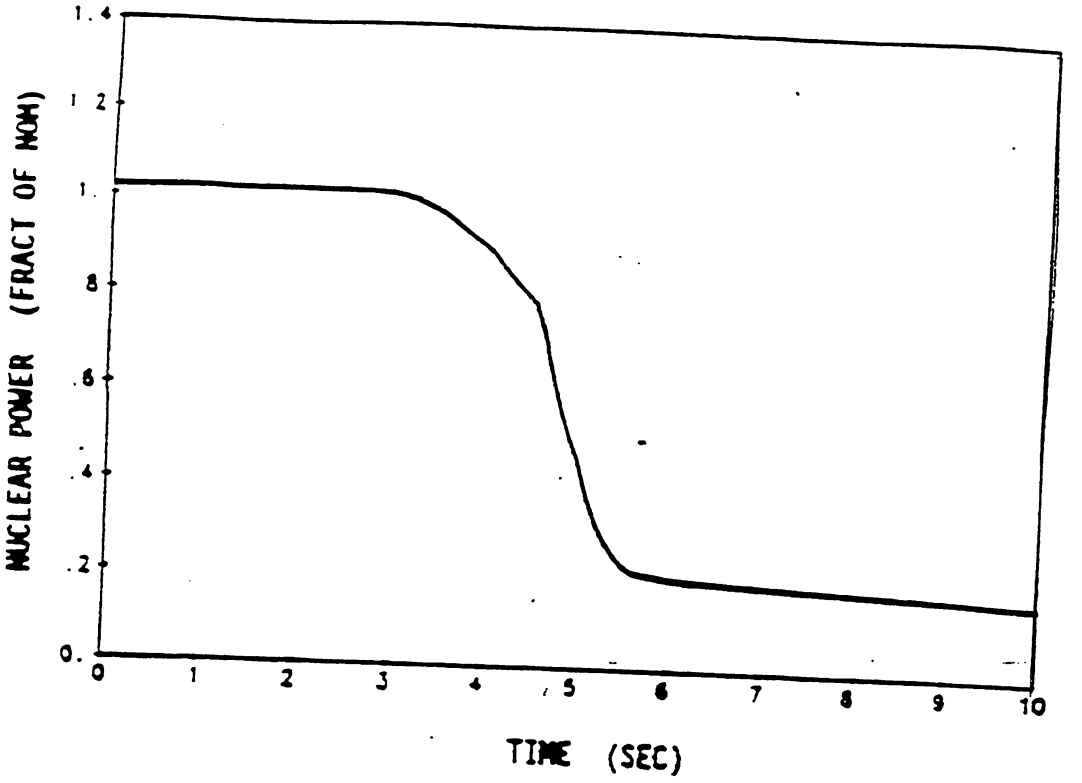
SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Partial Loss of Forced Reactor Coolant Flow, Heat Flux vs. Time (Hot Channel)
Figure 15.2.5-2a

Revised by Amendment 10



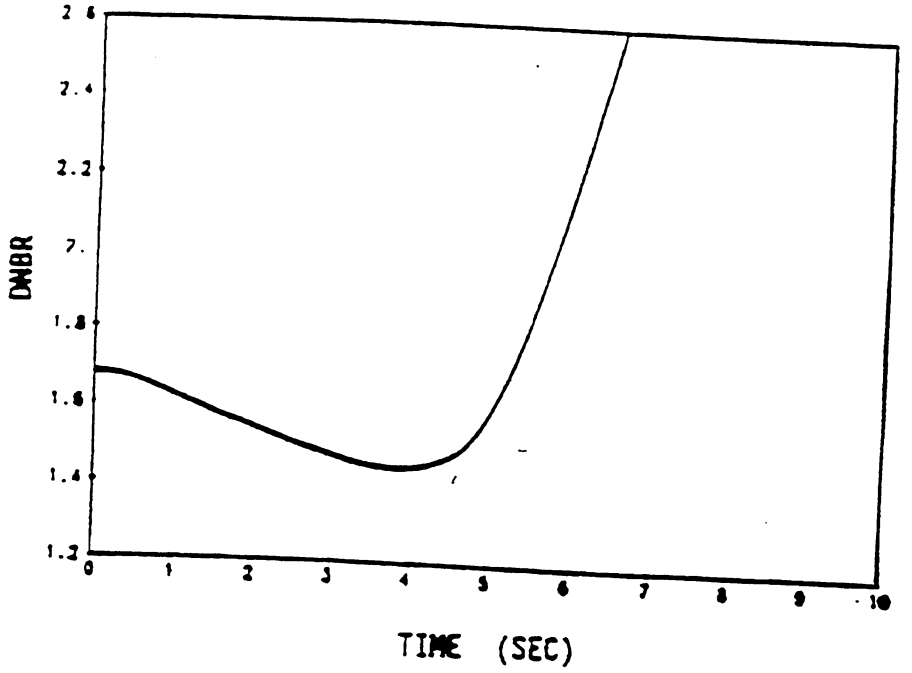
SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Partial Loss of Forced Reactor Coolant Flow, Heat Flux vs. Time (Average Channel)
Figure 15.2.5-2b

Revised by Amendment 10



SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Partial Loss of Forced Reactor Coolant Flow, Nuclear Power vs. Time
Figure 15.2.5-2c

Revised by Amendment 10

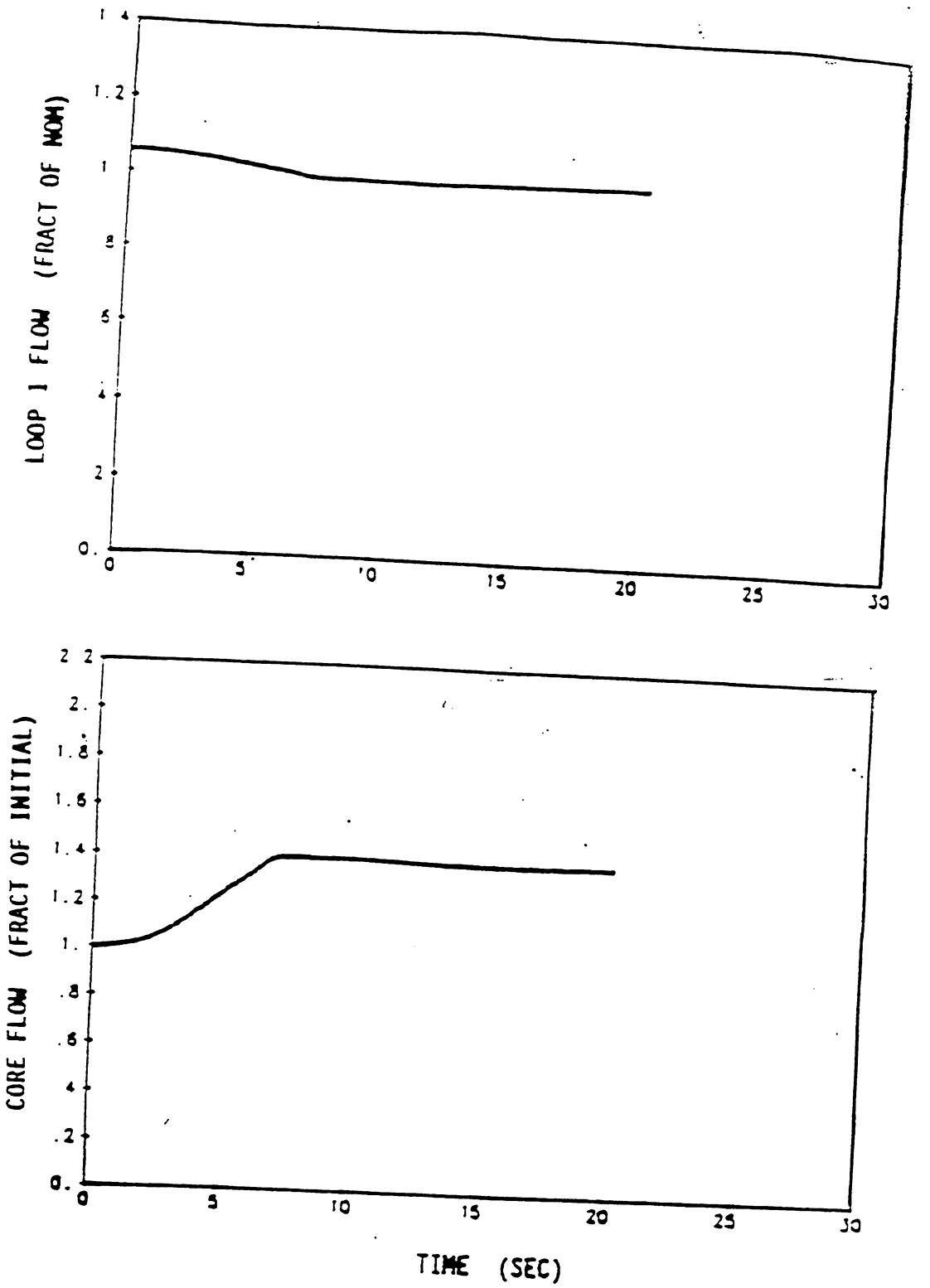


SEQUOYAH
FINAL SATETY ANALYSIS REPORT
UNITS 1 and 2

Partial Loss of Forced Reactor Coolant Flow, DNBR vs. Time

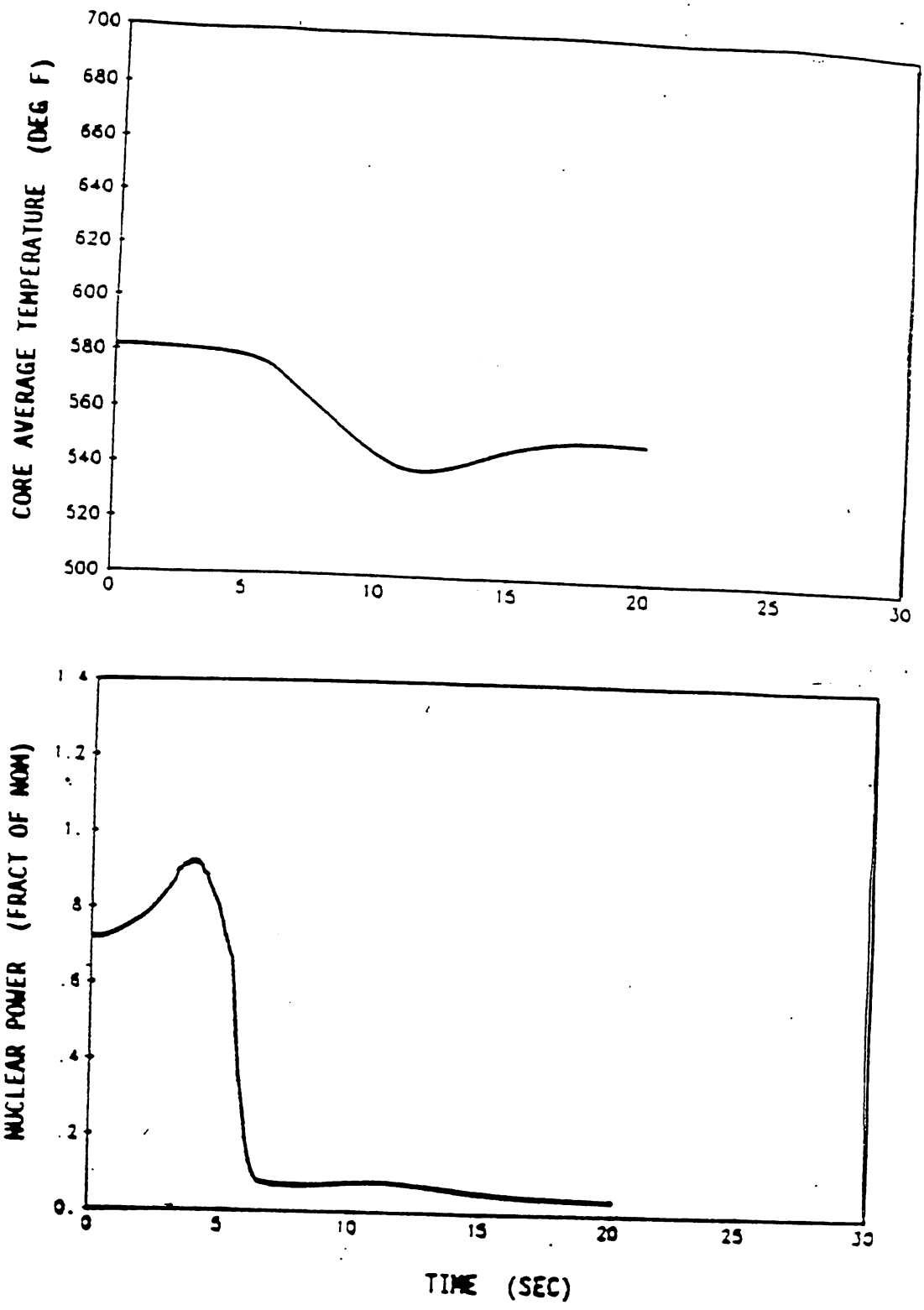
Figure 15.2.5-3

Revised by Amendment 10



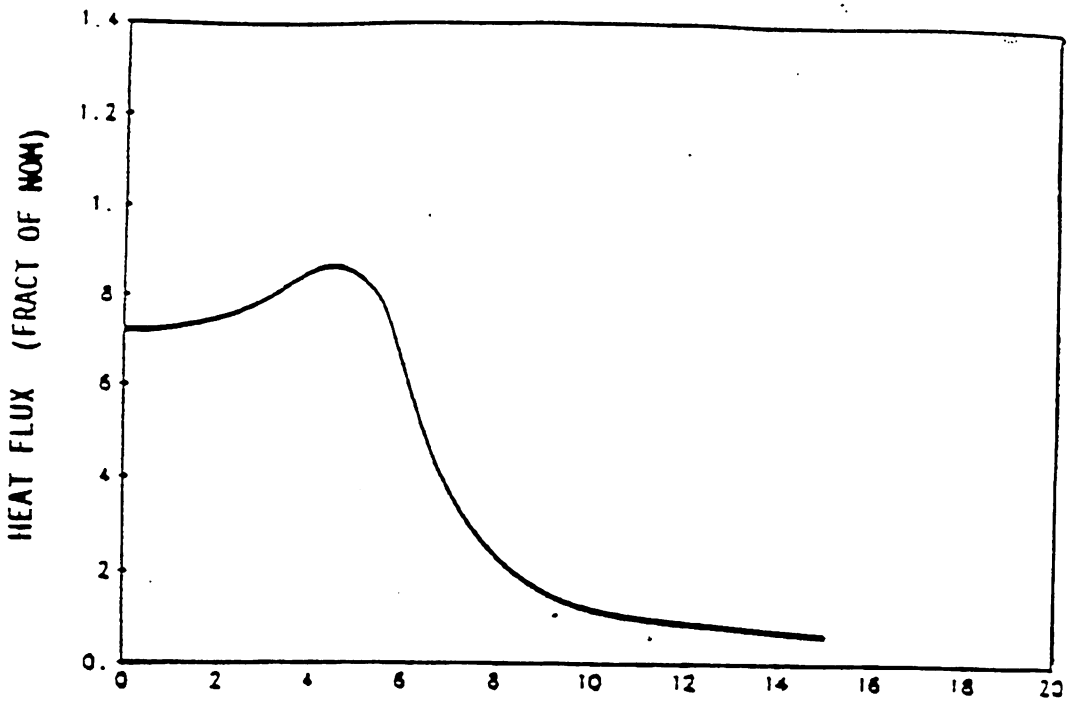
SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Startup of an Inactive Reactor Coolant Pump, Active Loop Flow vs. Time; Core Flow vs. Time
Figure 15.2.6-1

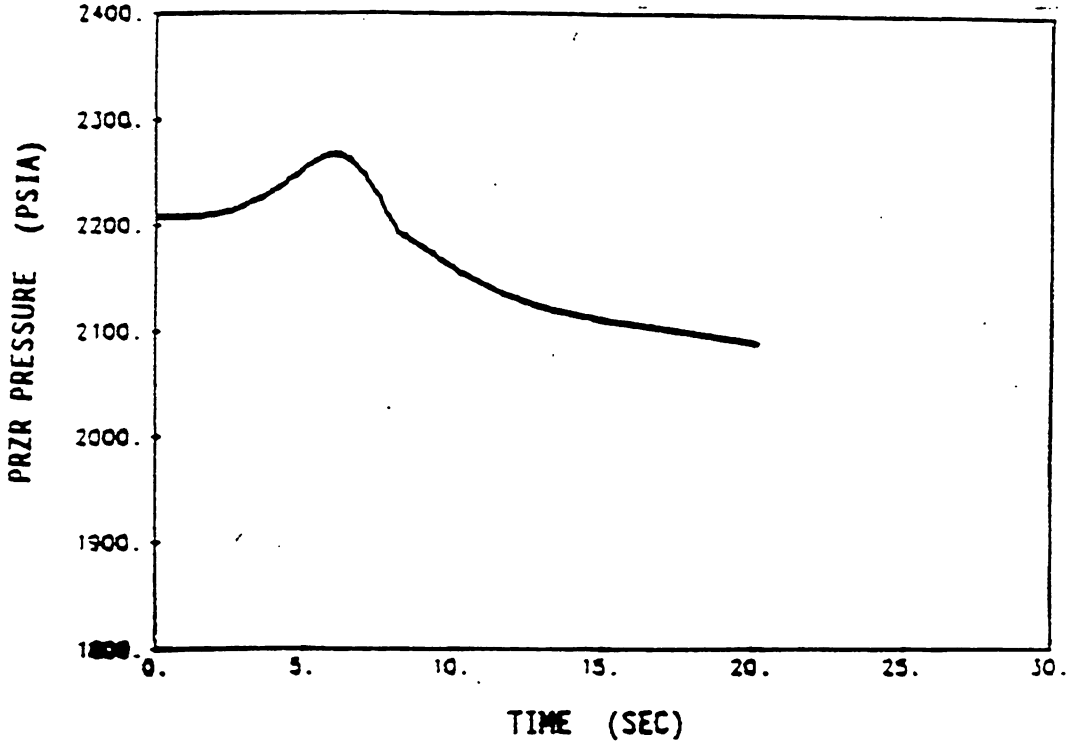
Revised by Amendment 10



SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Startup of an Inactive Reactor Coolant Pump, Core Average Temperature vs. Time; Nuclear Power vs. Time
Figure 15.2.6-2

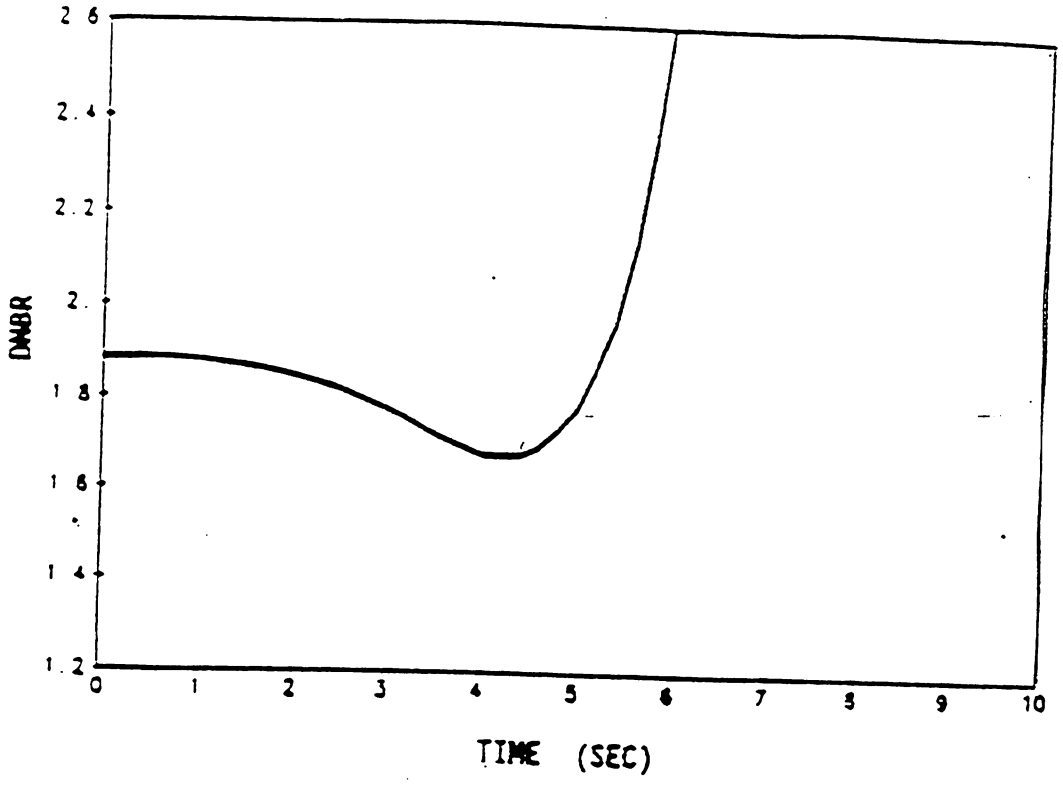
Revised by Amendment 10





SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Startup of an Inactive Reactor Coolant Pump, Heat Flux vs. Time; Pressurizer Pressure vs. Time
Figure 15.2.6-3

Revised by Amendment 10



SEQUOYAH FINAL SATETY ANALYSIS REPORT UNITS 1 and 2
Startup of an Inactive Reactor Coolant Pump, DNBR vs. Time
Figure 15.2.6-4

Revised by Amendment 10

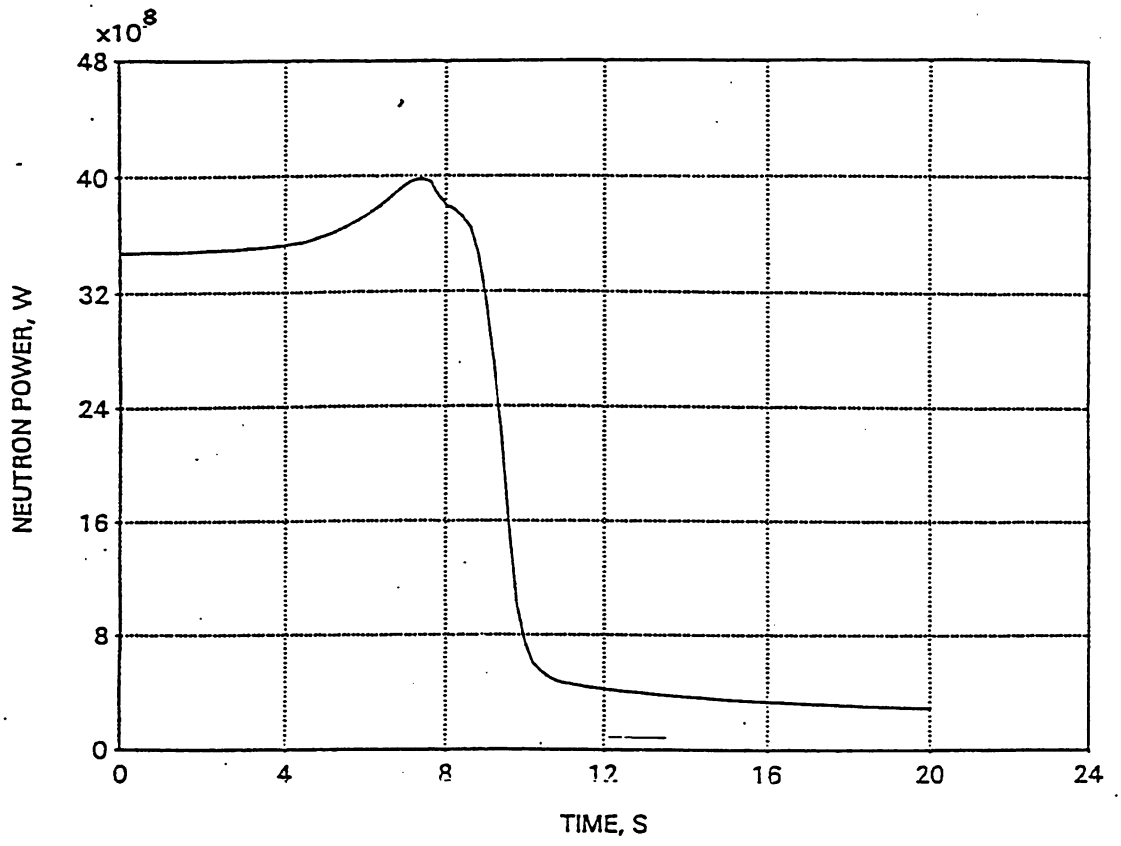


FIGURE 15.2.7-1

Loss of Electric Load
Neutron Power

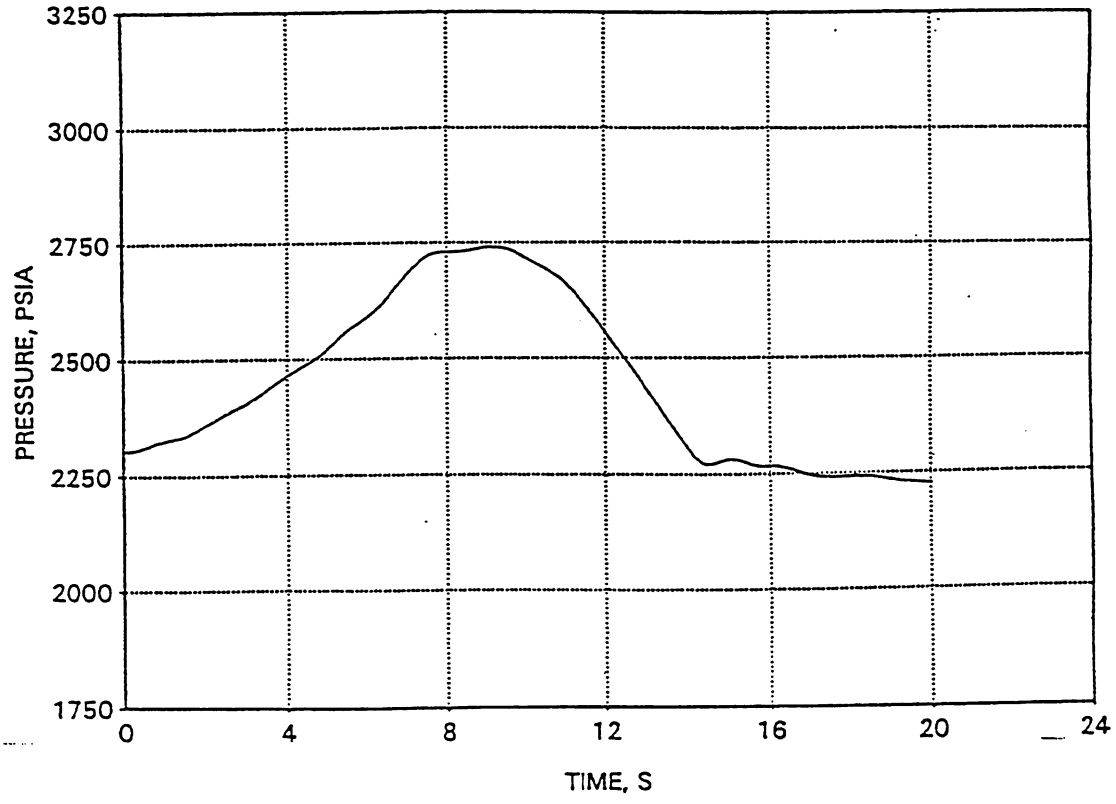


FIGURE 15.2.7-2

Loss of Electric Load
Reactor Vessel Lower Plenum Pressure

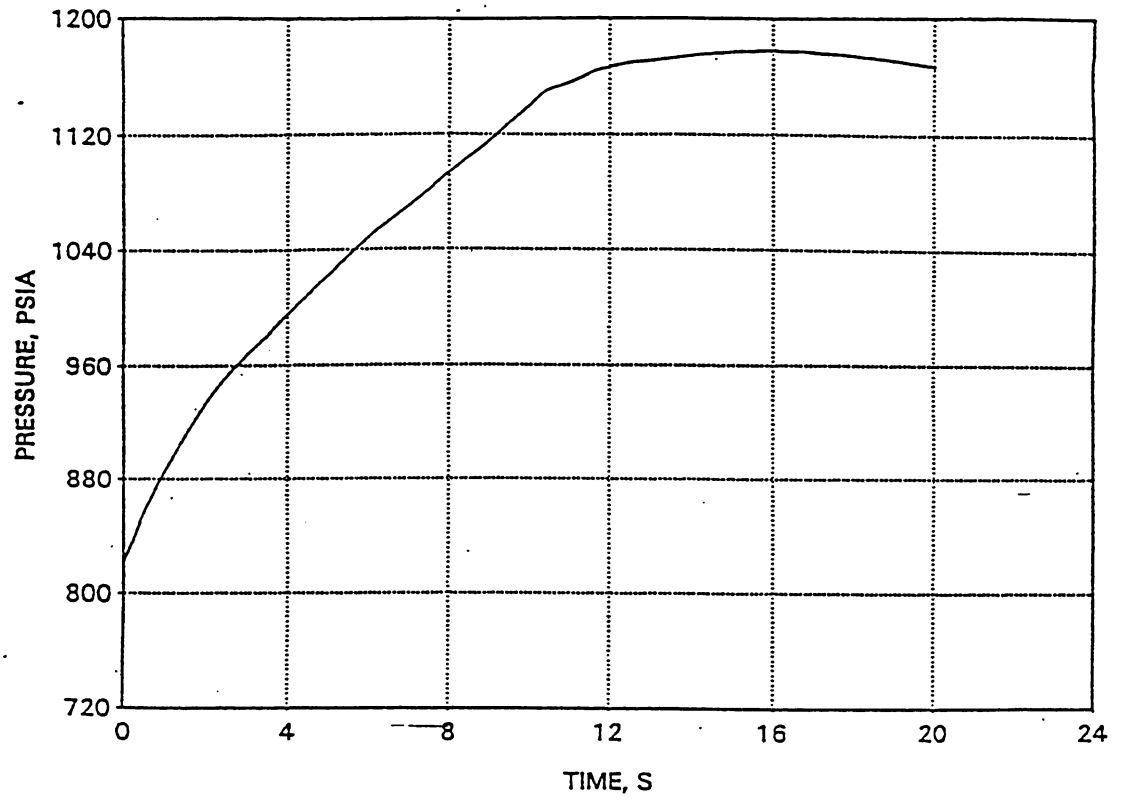


FIGURE 15.2.7-3

Loss of Electric Load
Steam Generator Downcomer Pressure

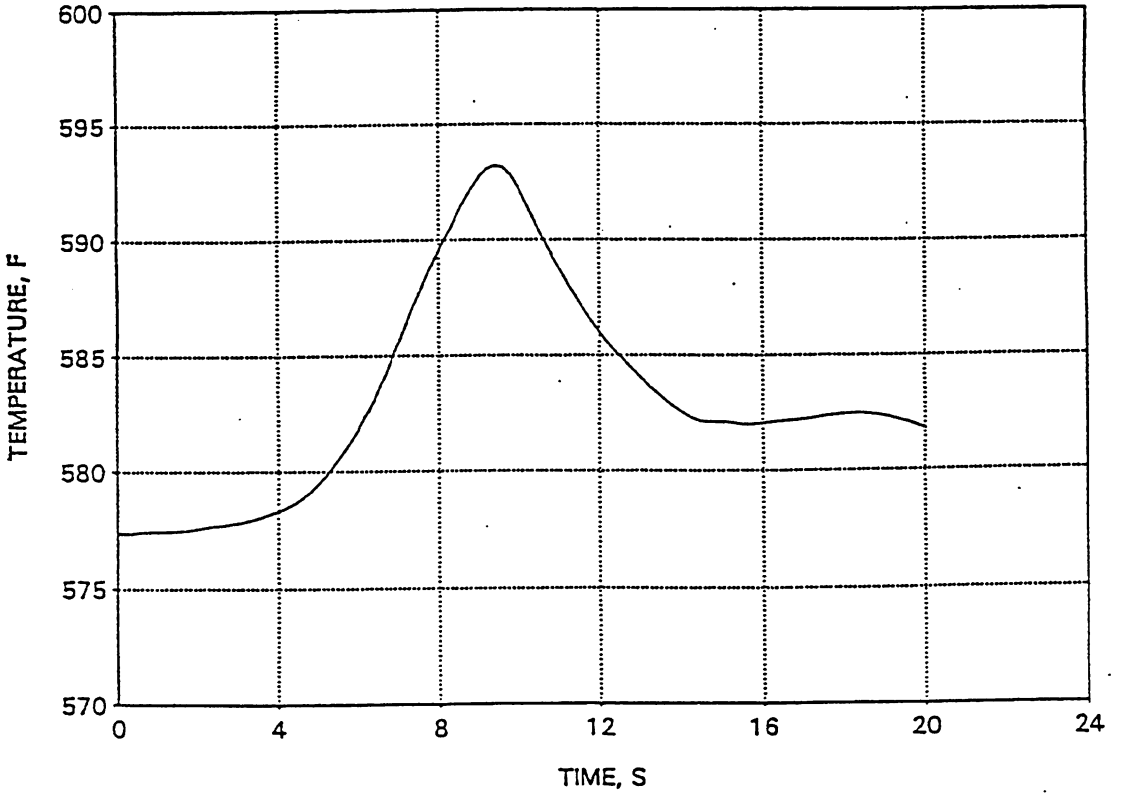


FIGURE 15.2.7-4

Loss of Electric Load
Average Core Fluid Temperature

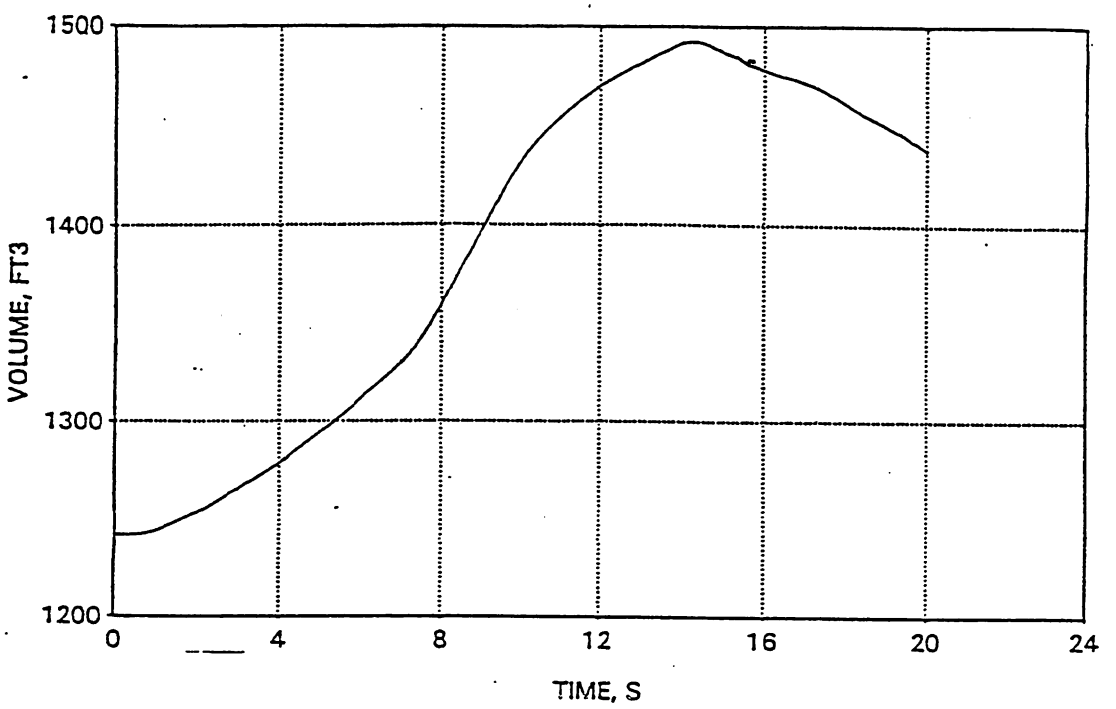


FIGURE 15.2.7-5

Loss of Electric Load
Pressurizer Water Volume

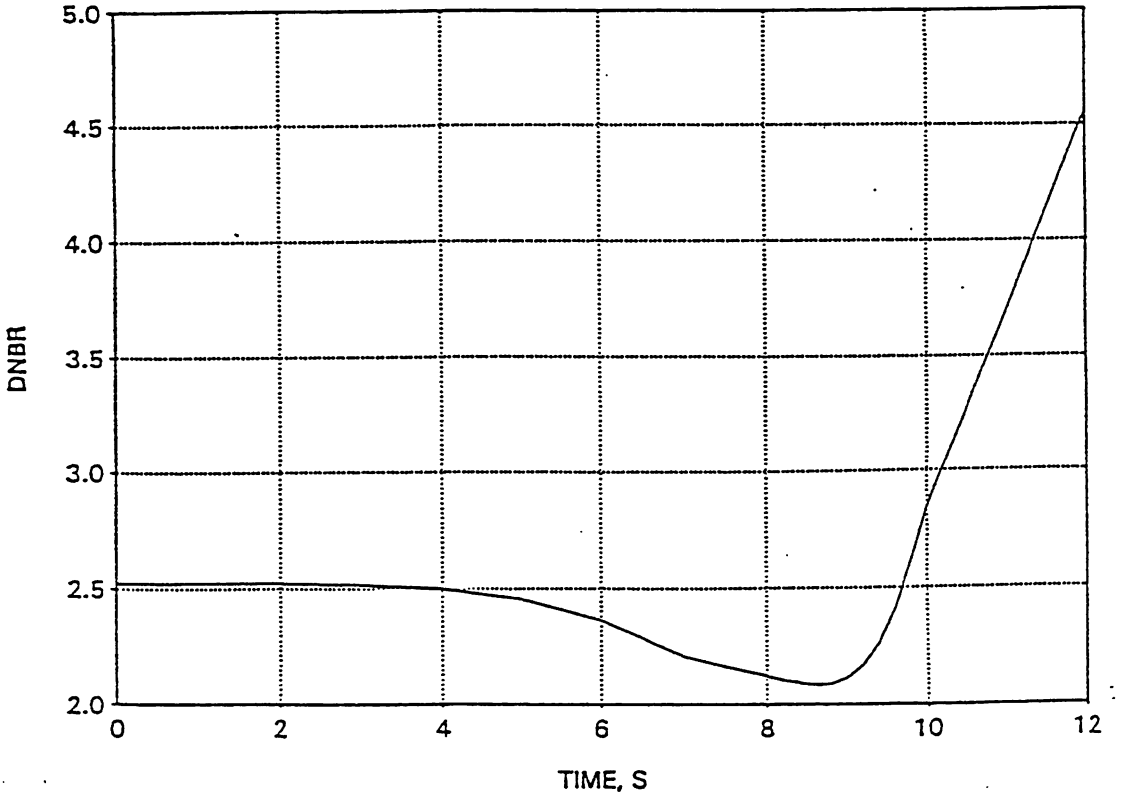


FIGURE 15.2.7-6

Loss of Electric Load
DNBR

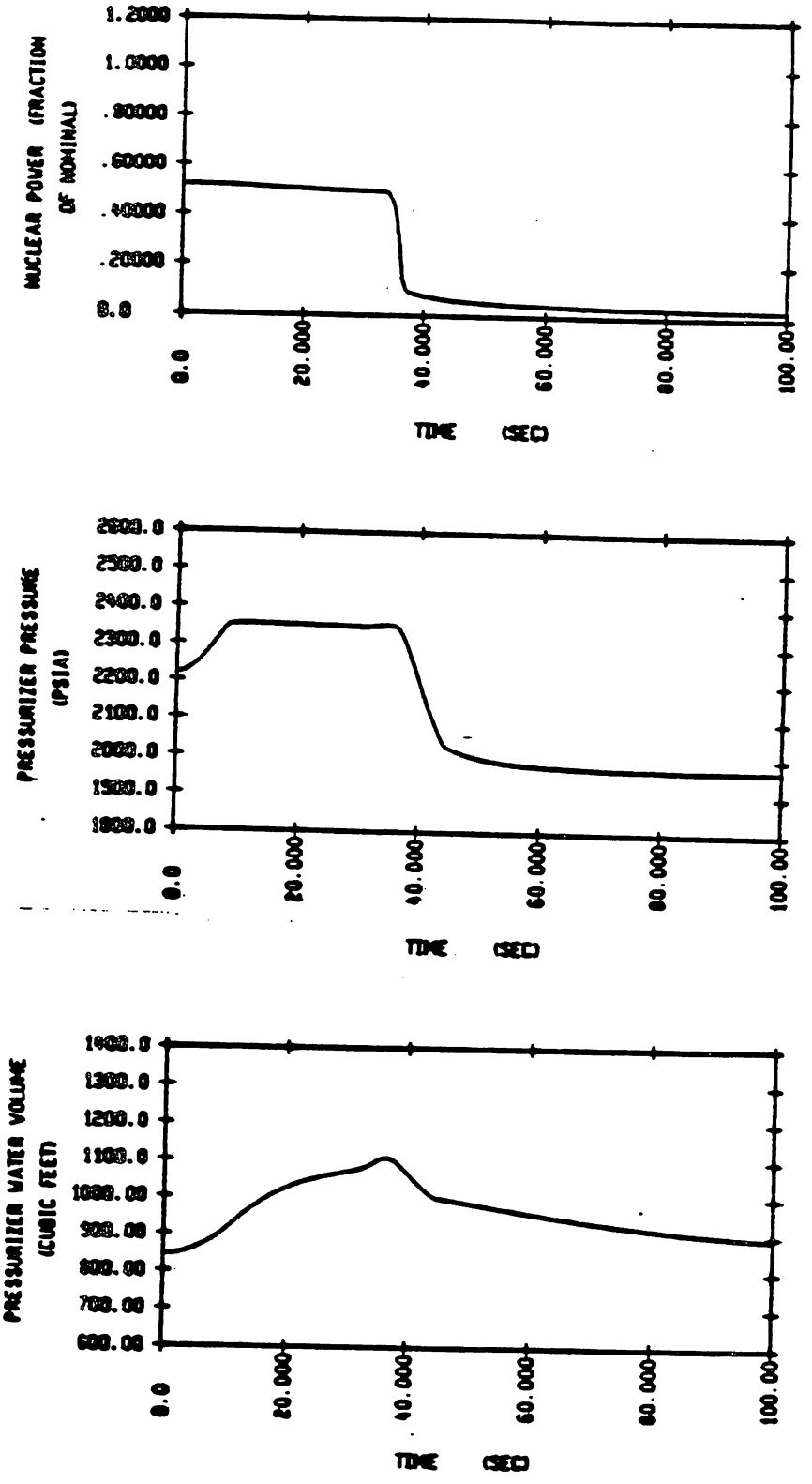


FIGURE 15.2.7-9 LOSS OF LOAD FROM 52% POWER WITH PRESSURE CONTROL, MINIMUM FEEDBACK

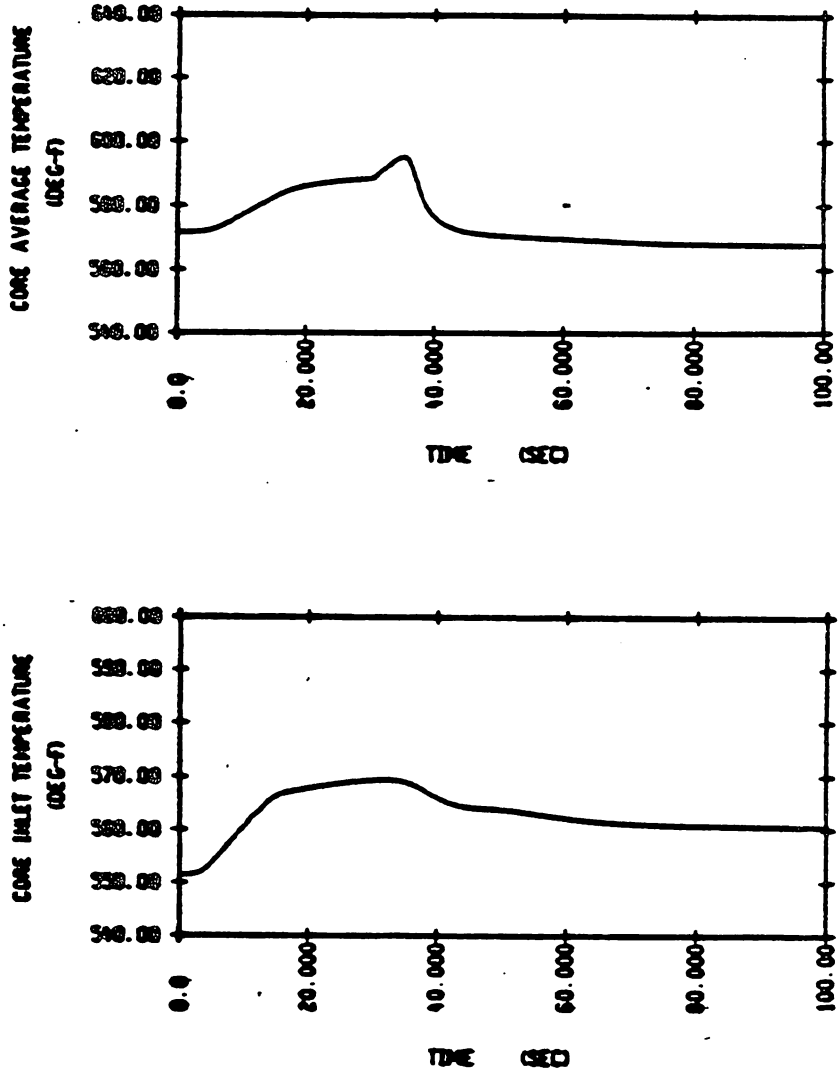


FIGURE 15.2.7-10 LOSS OF LOAD FROM 52% POWER WITH PRESSURE CONTROL, MINIMUM FEEDBACK

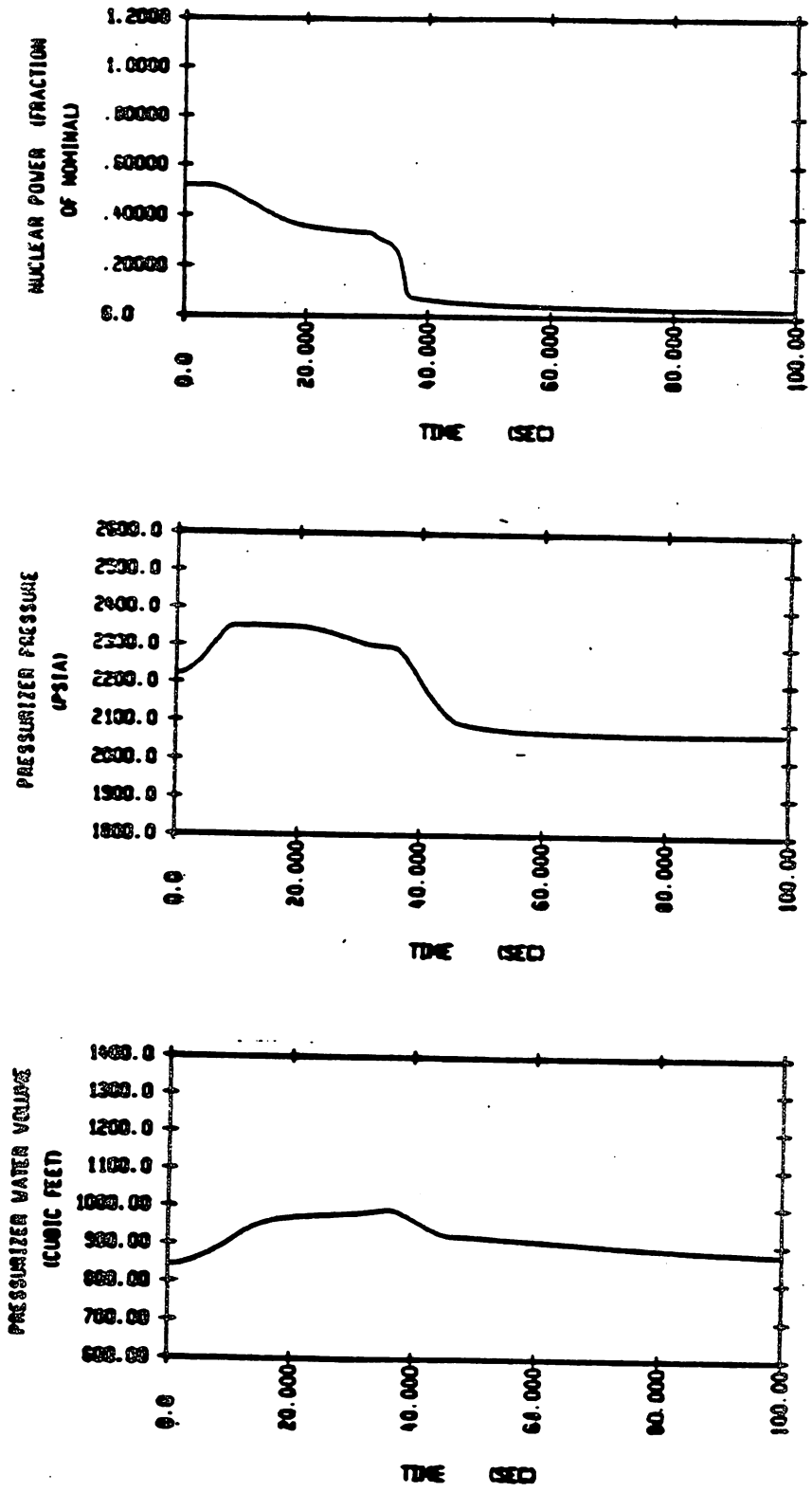


FIGURE 15.2.7-11 LOSS OF LOAD FROM 52% POWER WITH PRESSURE CONTROL, MAXIMUM FEEDBACK

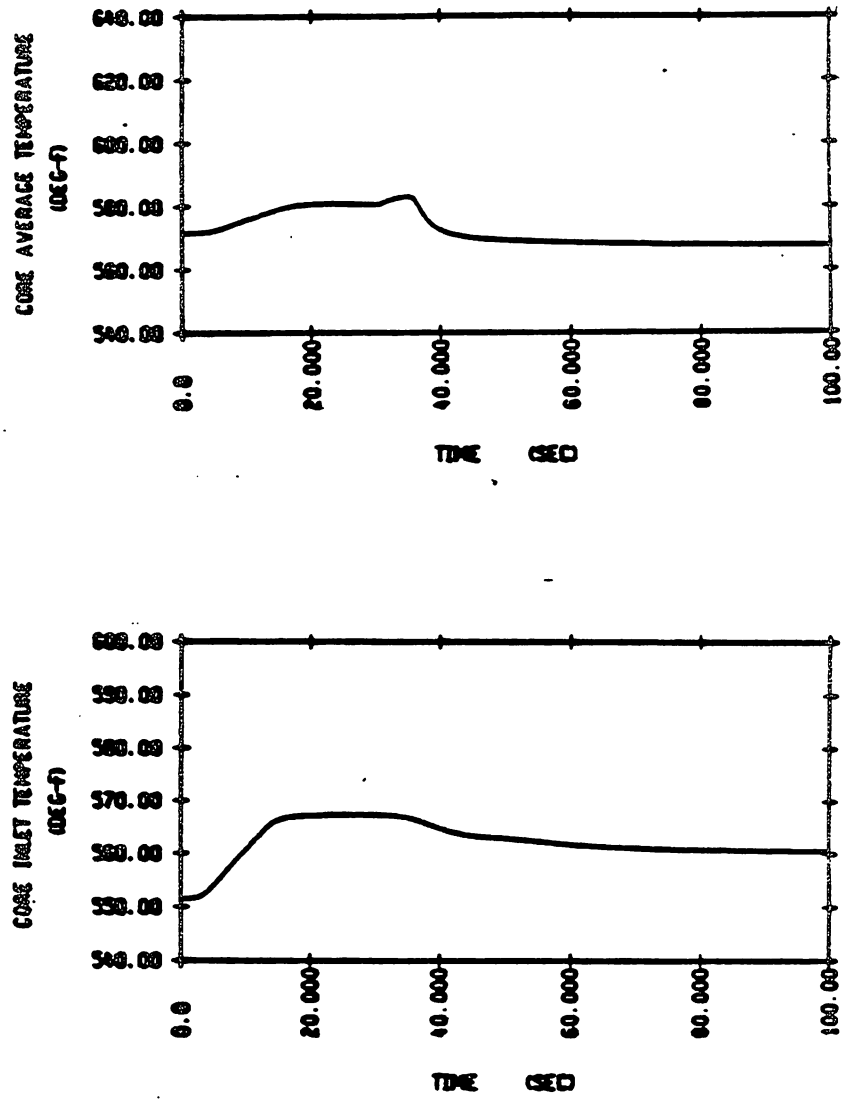


FIGURE 15.2.7-12 LOSS OF LOAD FROM 52% POWER WITH PRESSURE CONTROL, MAXIMUM FEEDBACK

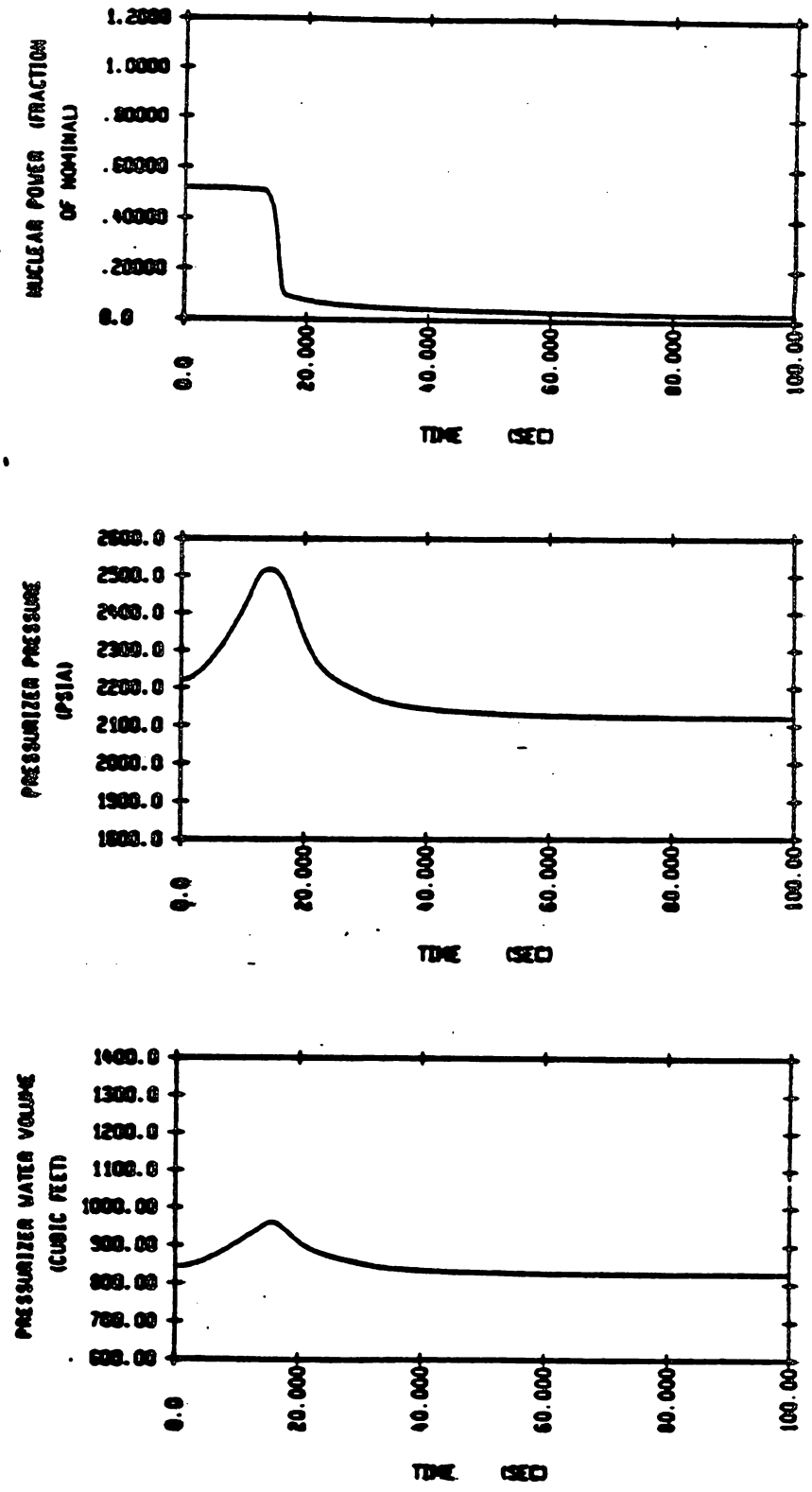
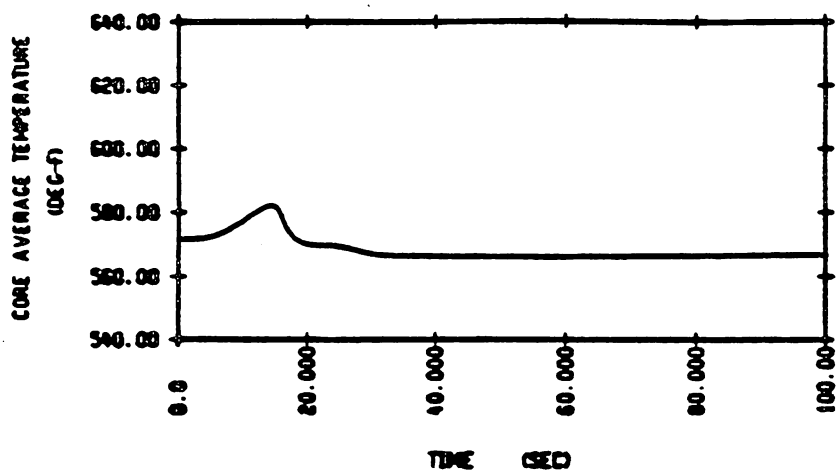


FIGURE 15.2.7-13 LOSS OF LOAD FROM 52% POWER WITHOUT PRESSURE CONTROL, MINIMUM FEEDBACK

Revised by Amendment 1

1



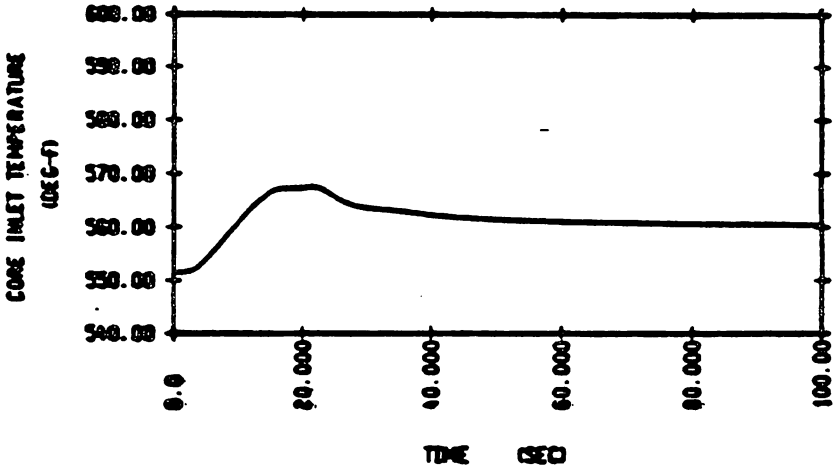


FIGURE 15.2.7-14 LOSS OF LOAD FROM 52% POWER WITHOUT PRESSURE CONTROL, MINIMUM FEEDBACK | 1

Revised by Amendment 1

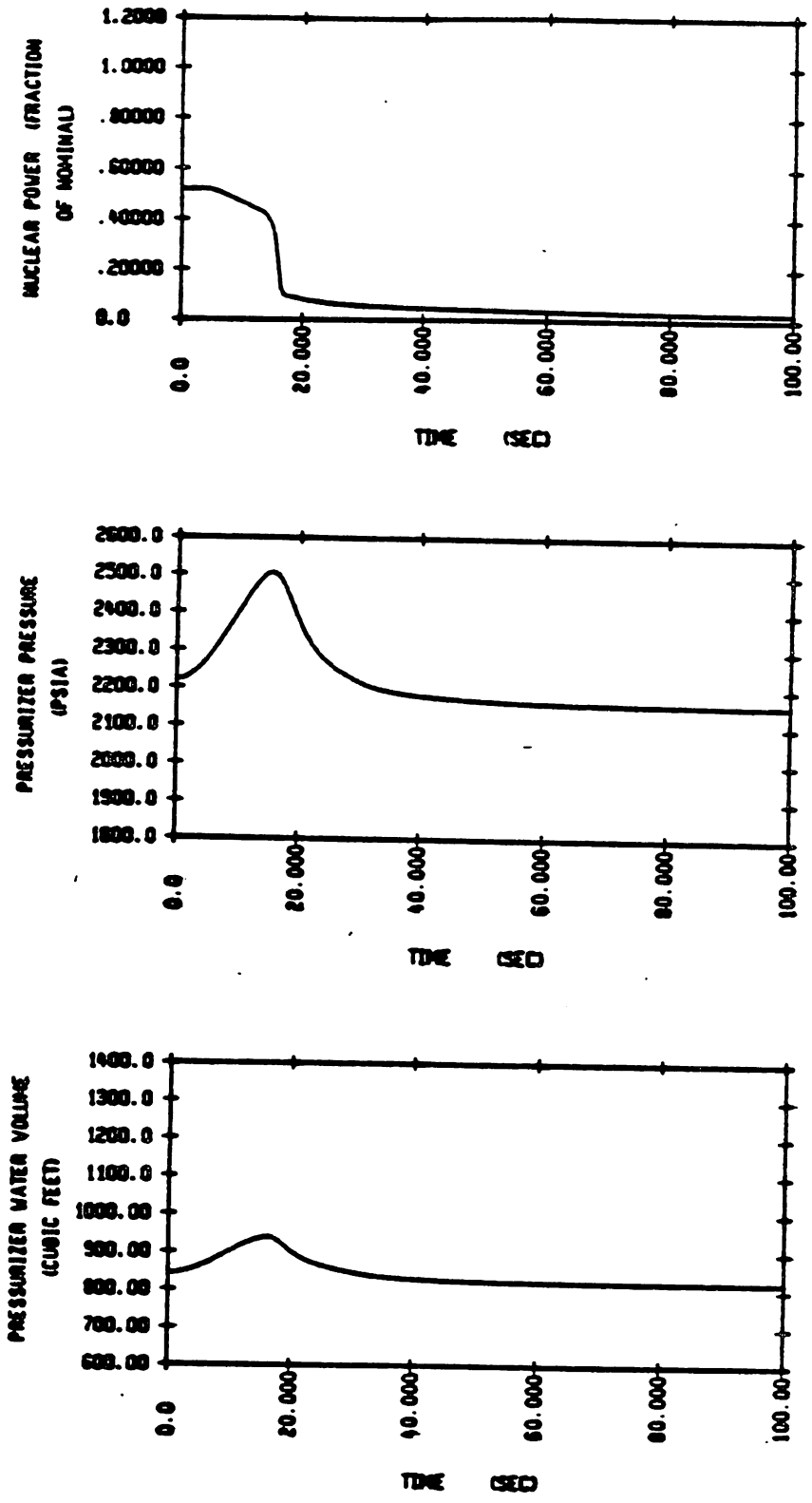
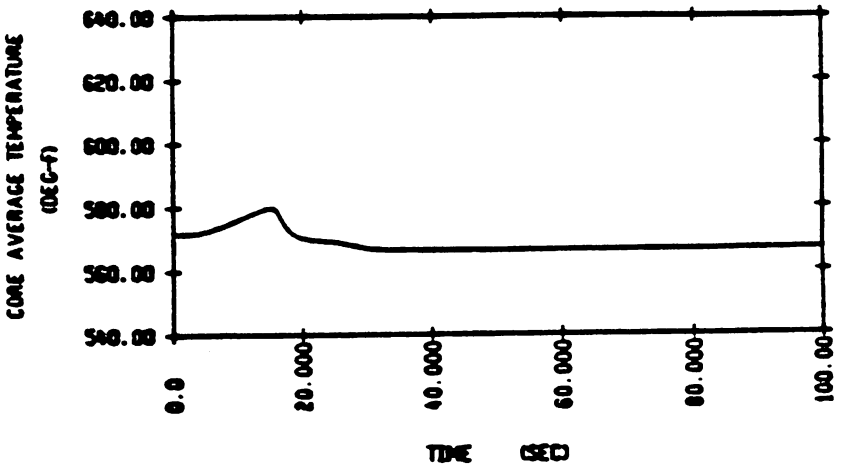


FIGURE 15.2.7-15 LOSS OF LOAD FROM 52% POWER WITHOUT PRESSURE CONTROL, MAXIMUM FEEDBACK 1

Revised by Amendment 1



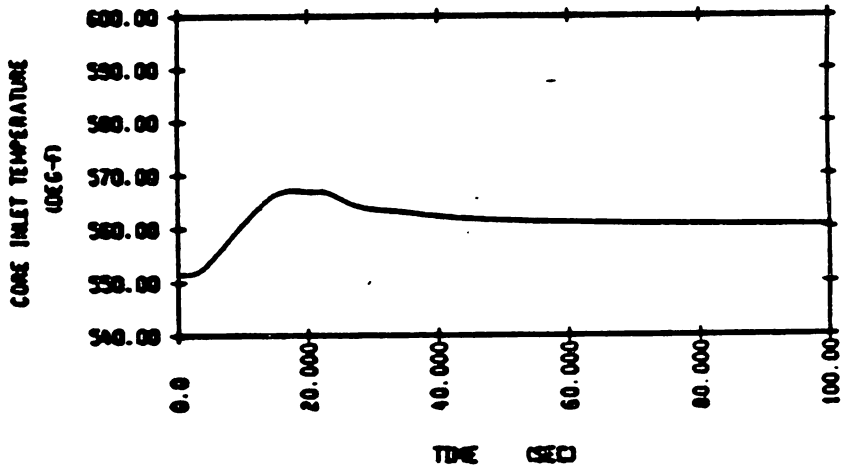


FIGURE 15.2.7-16 LOSS OF LOAD FROM 52% POWER WITHOUT PRESSURE CONTROL, MAXIMUM FEEDBACK | 1

Revised by Amendment 1

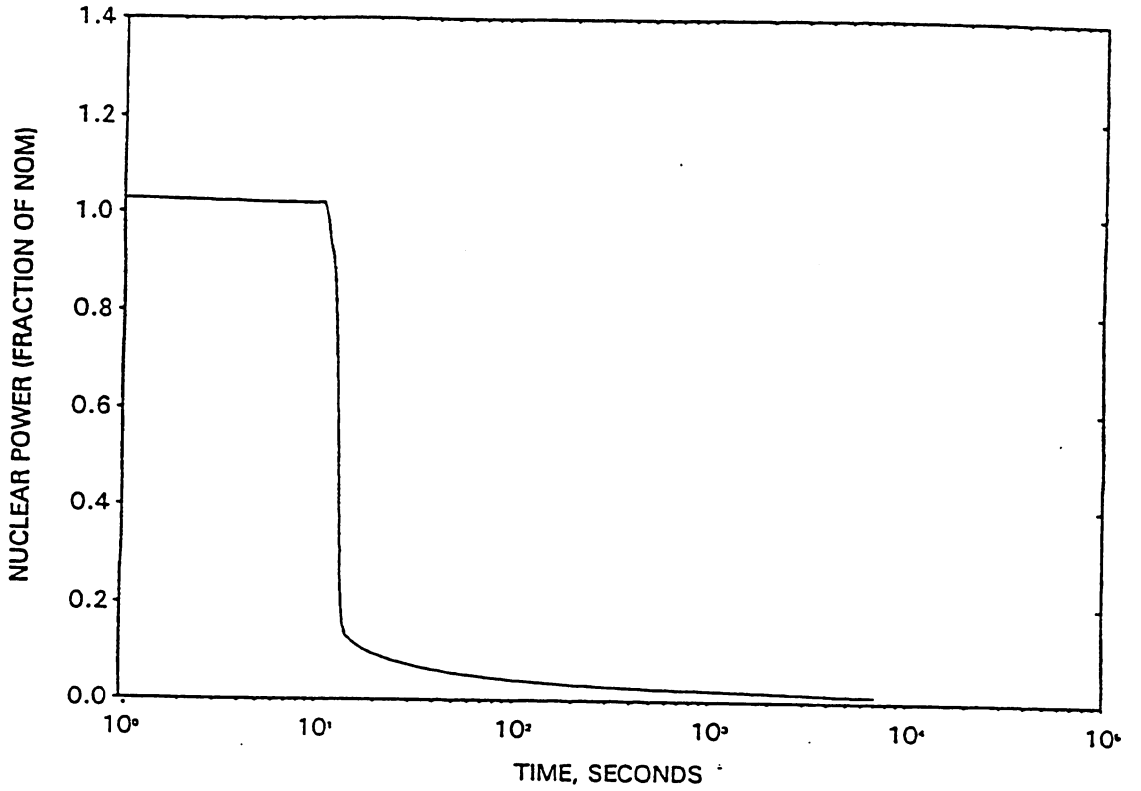


FIGURE 15.2.8-1a FULL POWER LOSS OF NORMAL FEEDWATER - NUCLEAR POWER

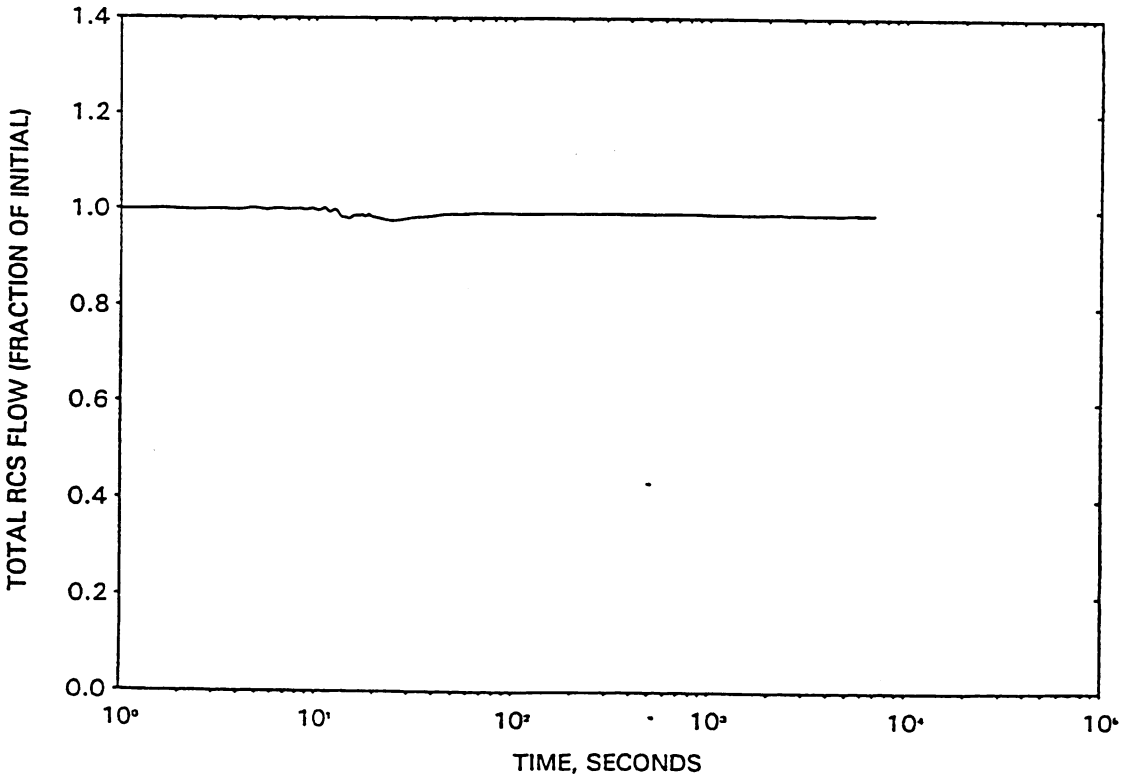


FIGURE 15.2.8-1b FULL POWER LOSS OF NORMAL FEEDWATER - RCS FLOW

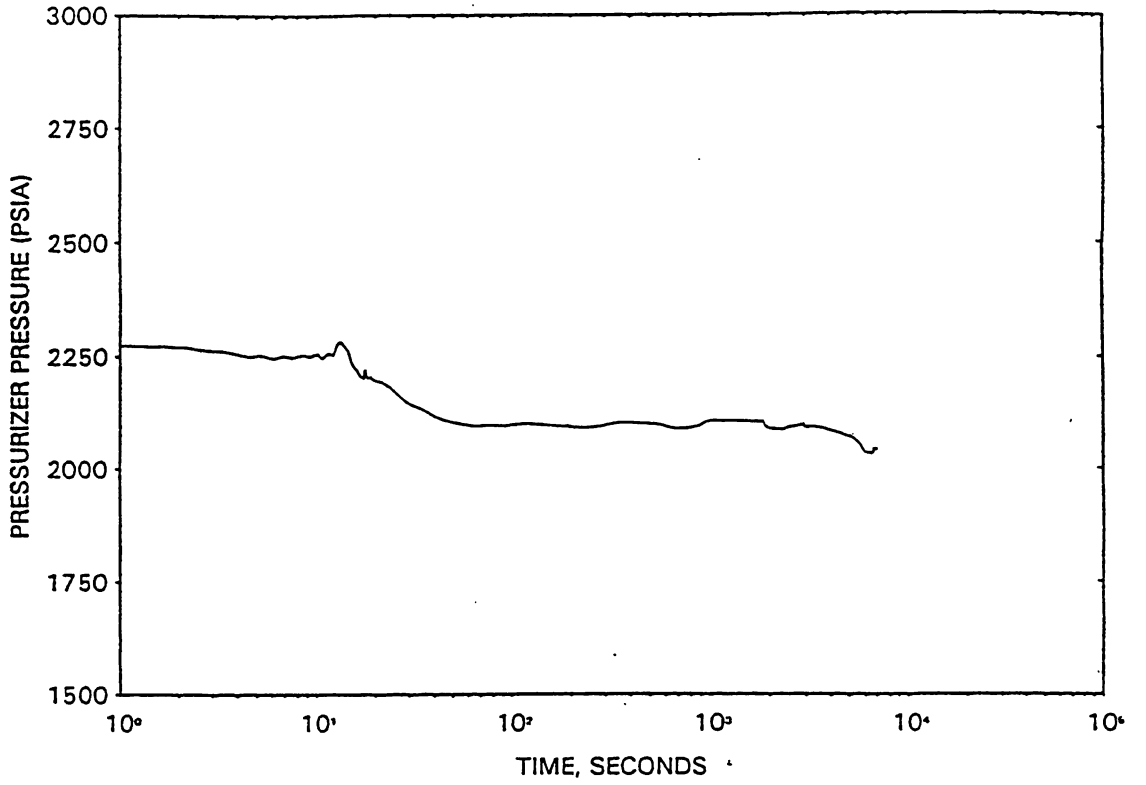


FIGURE 15.2.8-2a FULL POWER LOSS OF NORMAL FEEDWATER - PRESSURIZER PRESSURE

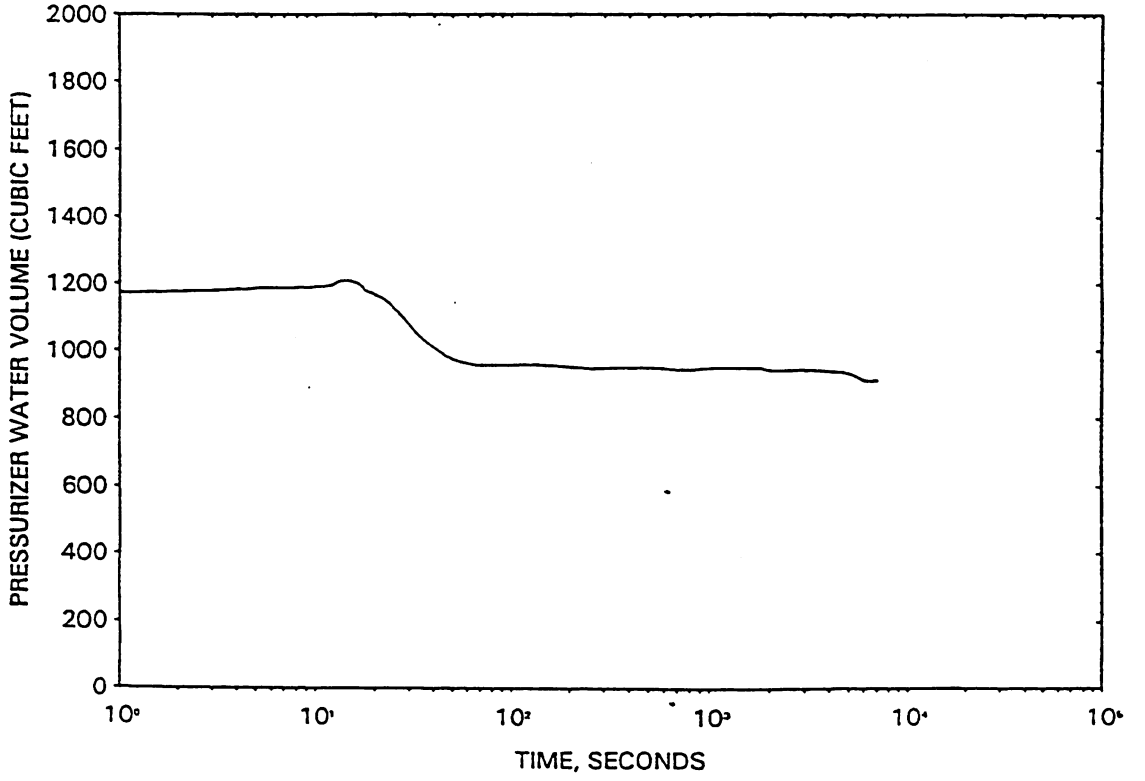


FIGURE 15.2.8-2b FULL POWER LOSS OF NORMAL FEEDWATER - PRESSURIZER WATER VOLUME

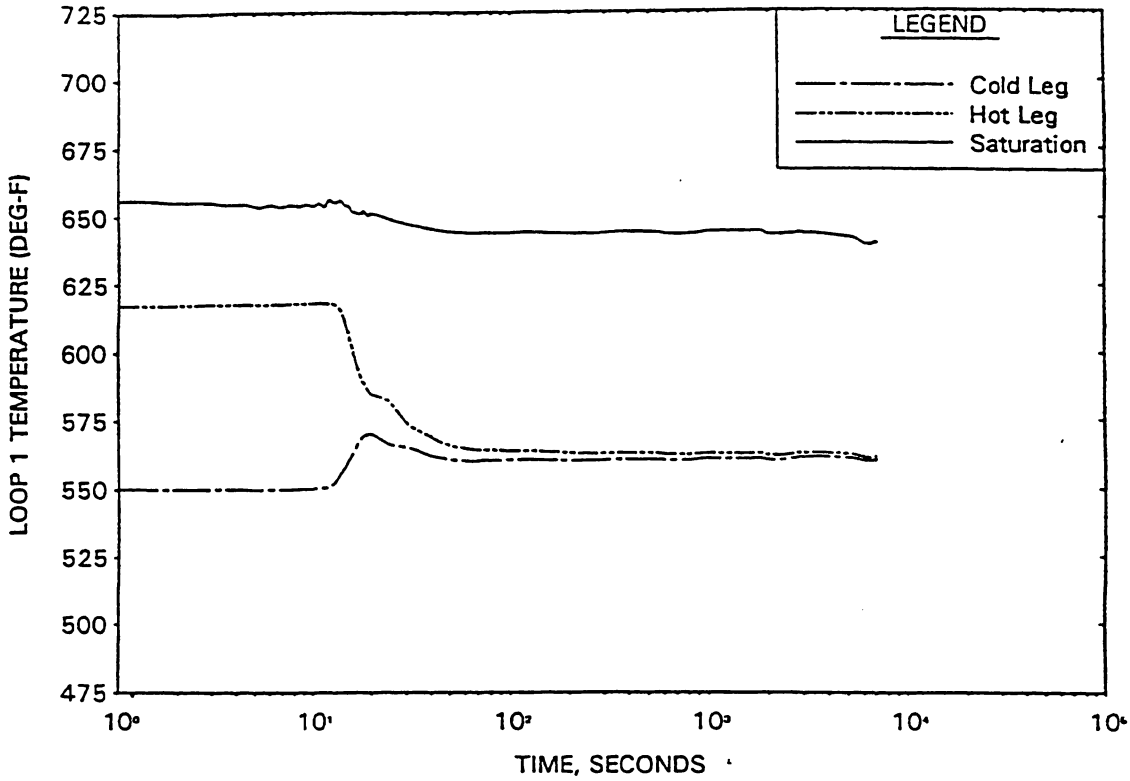


FIGURE 15.2.8-3a FULL POWER LOSS OF NORMAL FEEDWATER - LOOP 1 COLD LEG, HOT LEG AND SATURATION TEMPERATURES

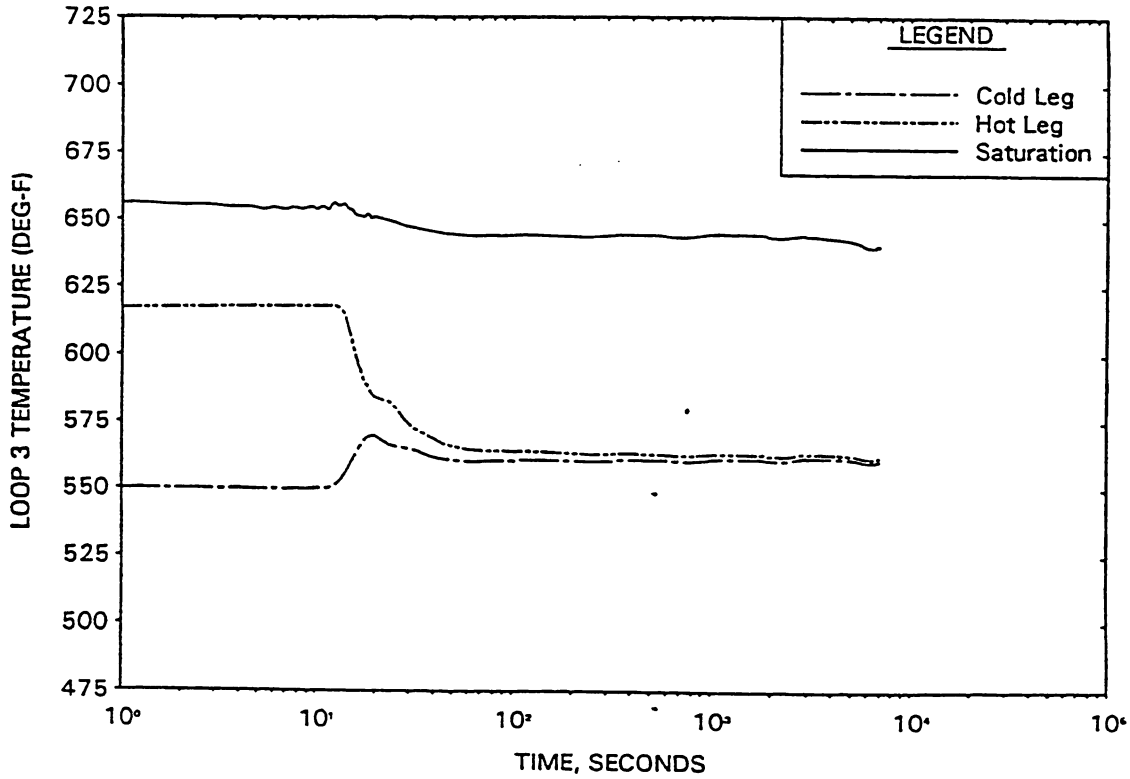


FIGURE 15.2.8-3b FULL POWER LOSS OF NORMAL FEEDWATER - LOOP 3 COLD LEG, HOT LEG AND SATURATION TEMPERATURES

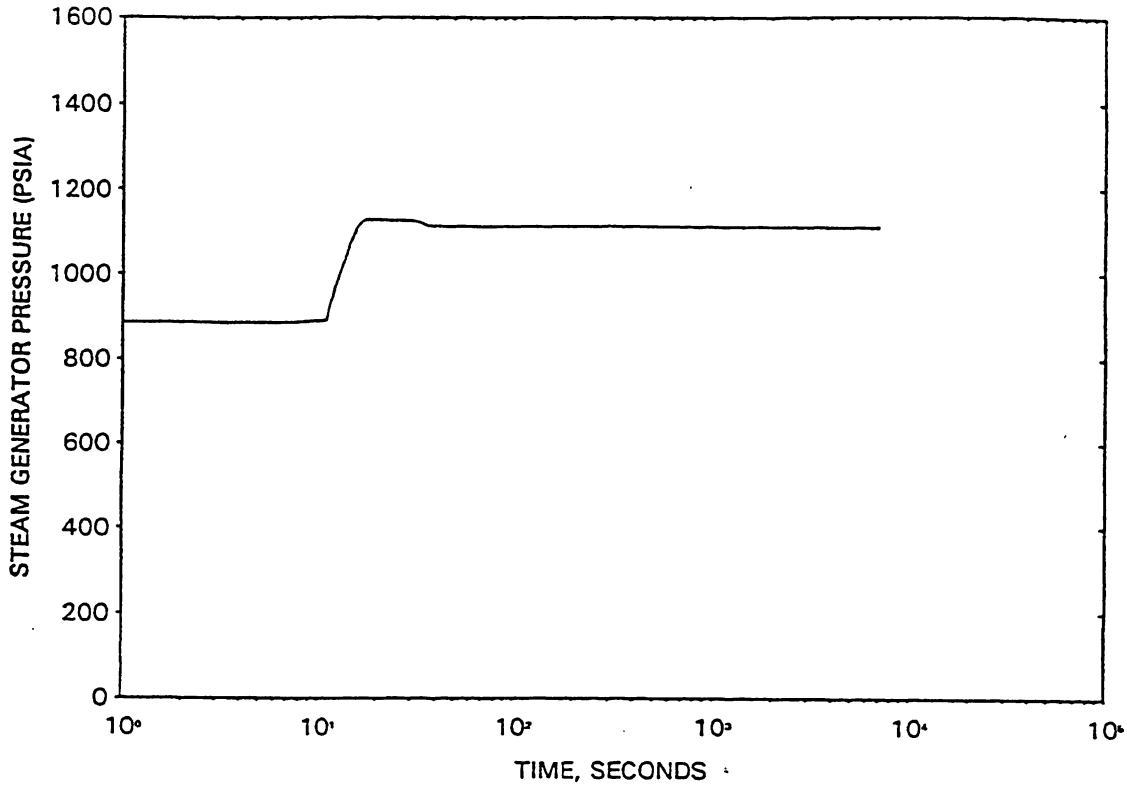


FIGURE 15.2.8-4a FULL POWER LOSS OF NORMAL FEEDWATER - STEAM GENERATOR PRESSURE

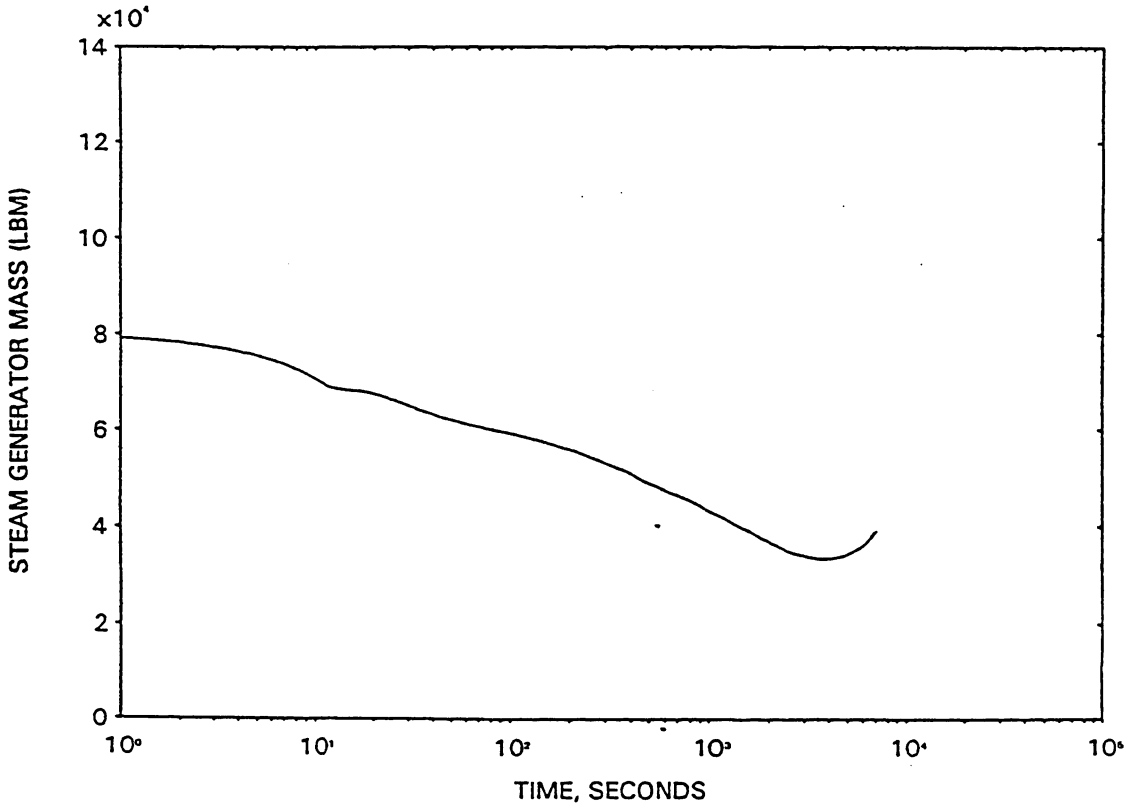


FIGURE 15.2.8-4b FULL POWER LOSS OF NORMAL FEEDWATER - STEAM GENERATOR MASS

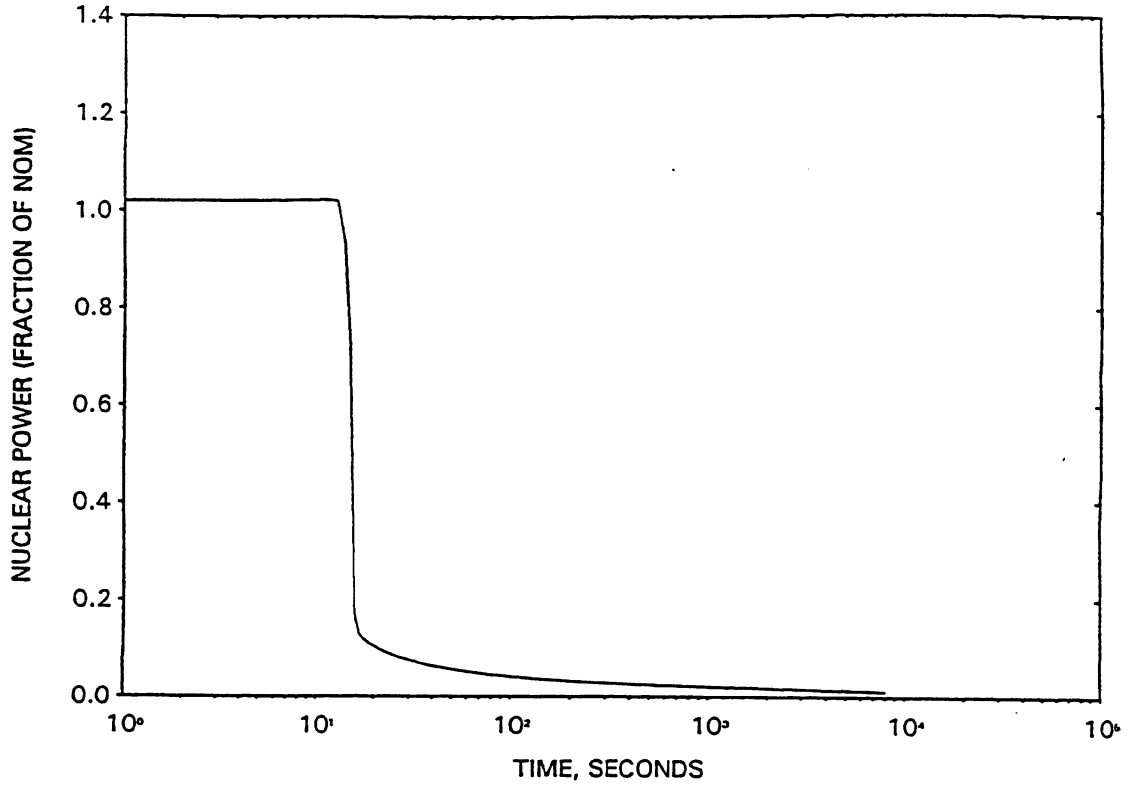


FIGURE 15.2.9-1a LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - NUCLEAR POWER

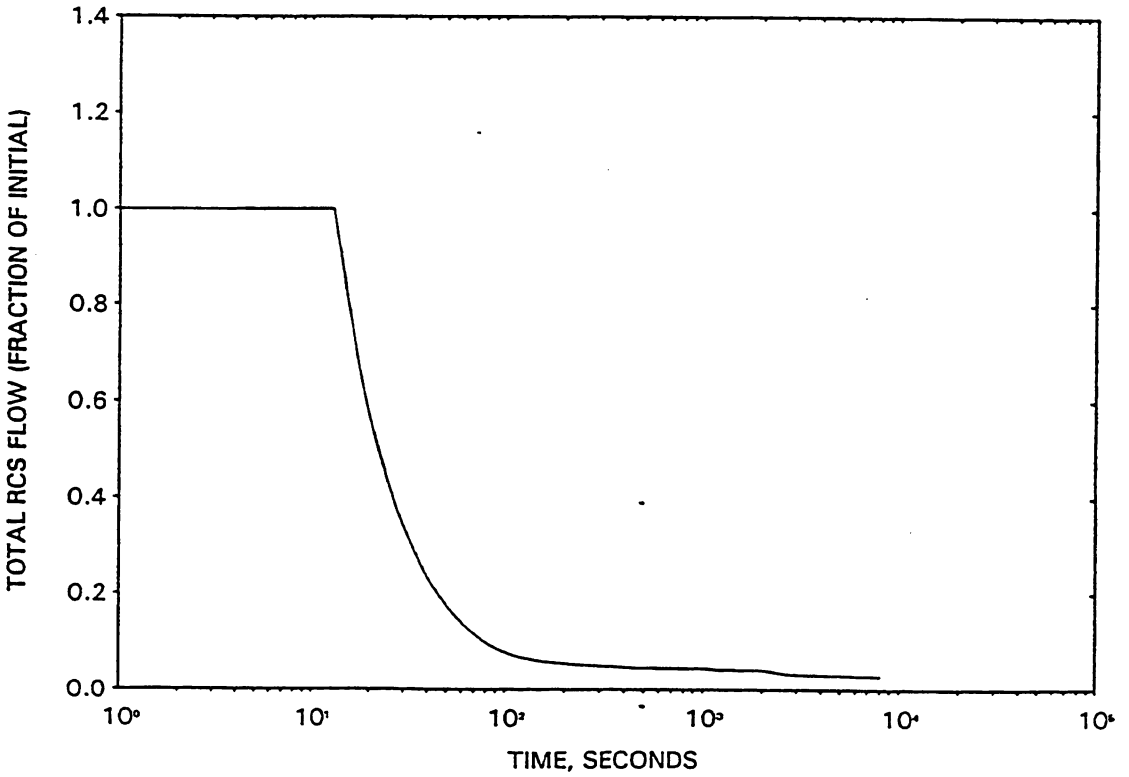


FIGURE 15.2.9-1b LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - RCS FLOW

Revised by Amendment 15

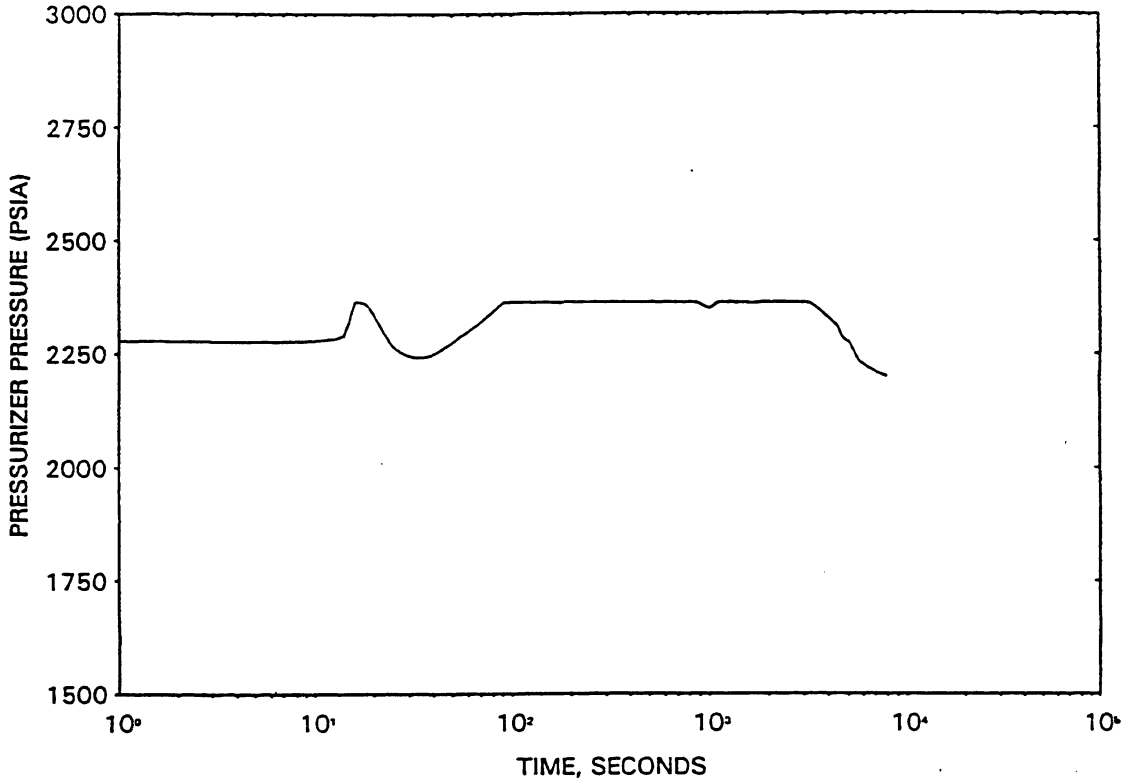


FIGURE 15.2.9-2a LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - PRESSURIZER PRESSURE

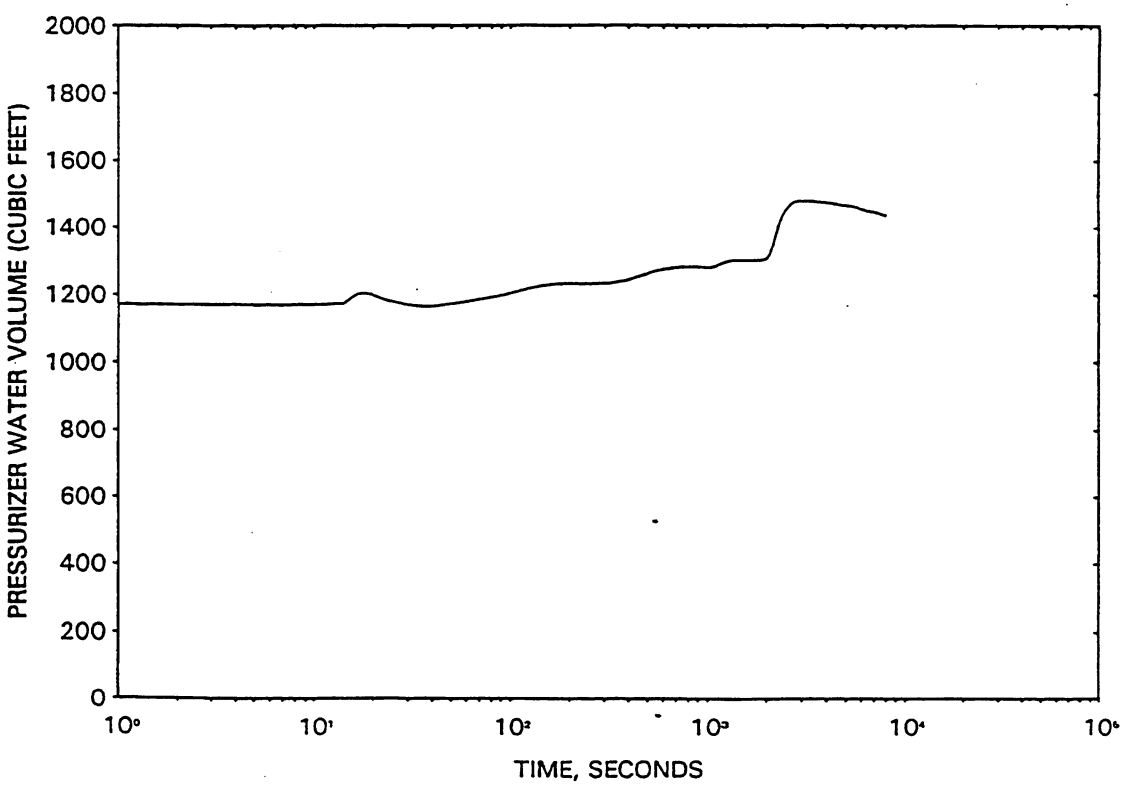


FIGURE 15.2.9-2b LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - PRESSURIZER WATER VOLUME

Revised by Amendment 15

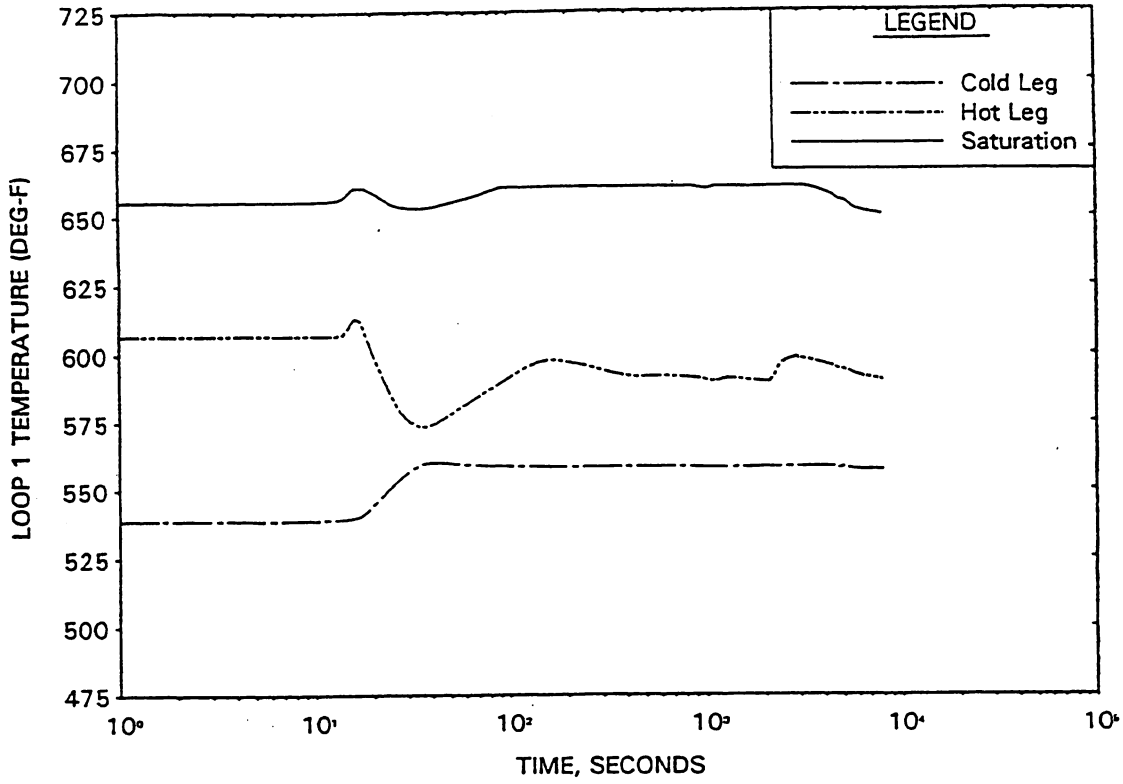


FIGURE 15.2.9-3a LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - LOOP 1 COLD LEG, HOT LEG AND SATURATION TEMPERATURES

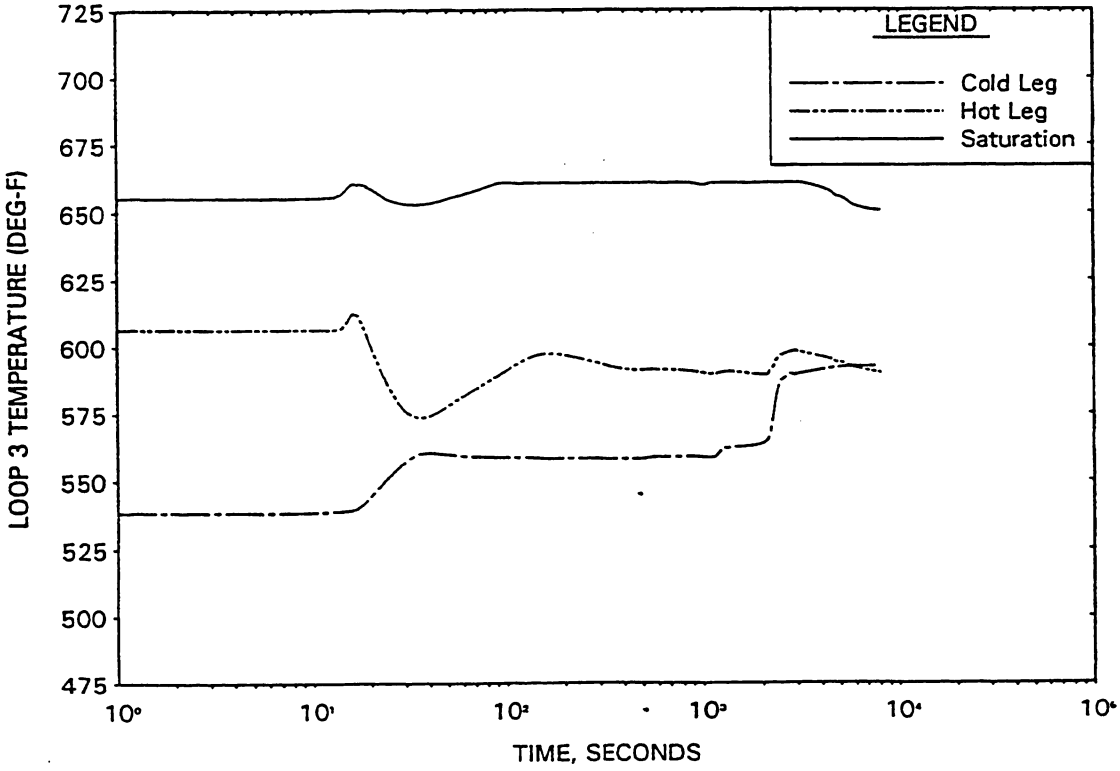


FIGURE 15.2.9-3b LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - LOOP 3 COLD LEG, HOT LEG AND SATURATION TEMPERATURES

Revised by Amendment 15

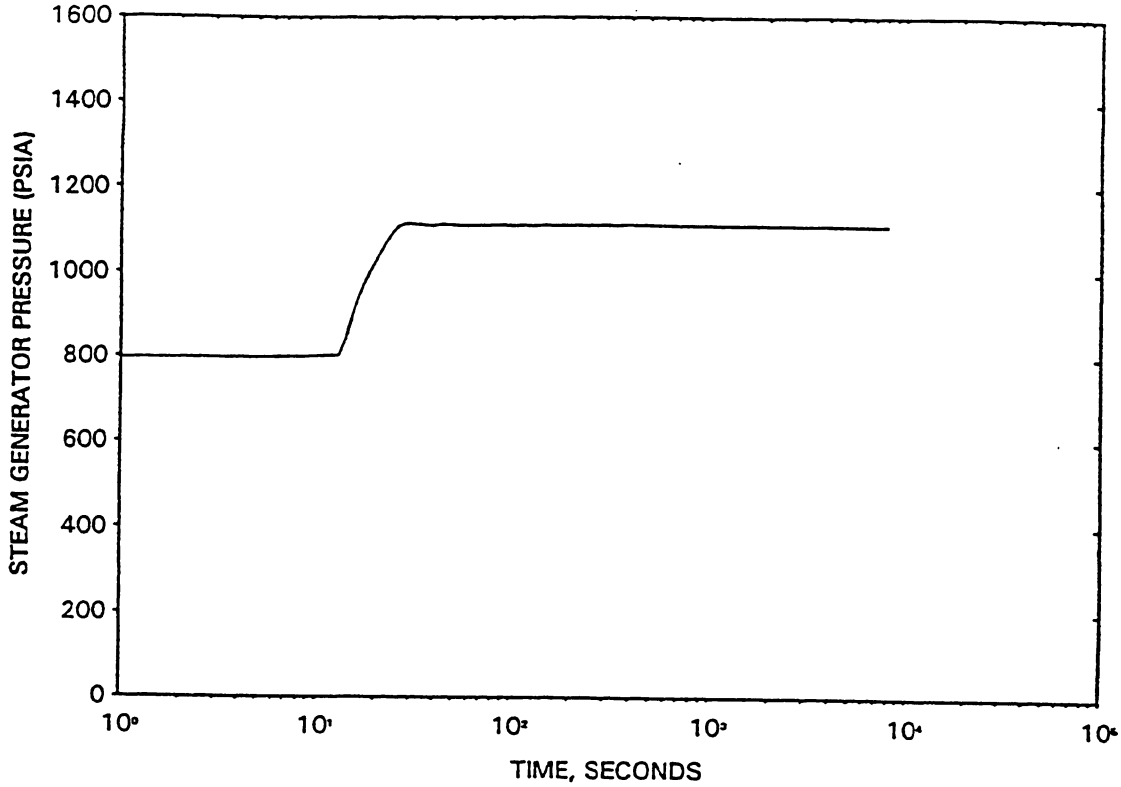


FIGURE 15.2.9-4a LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - STEAM GENERATOR PRESSURE

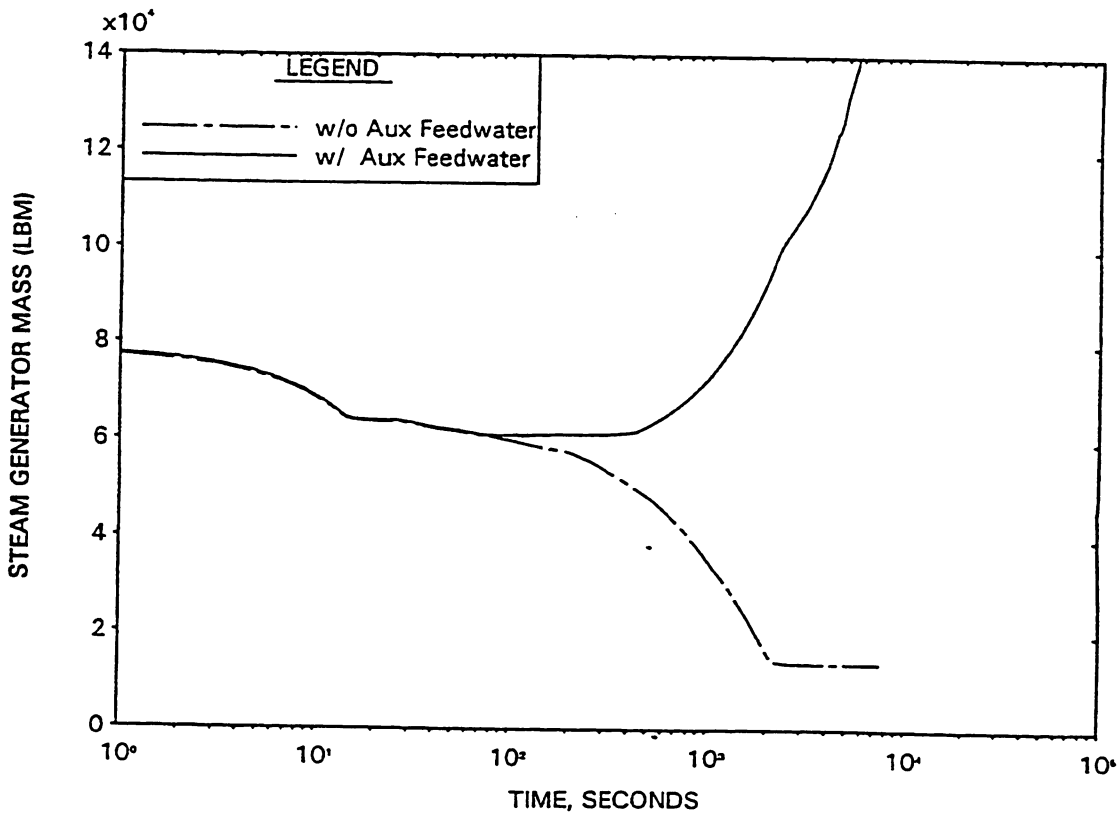
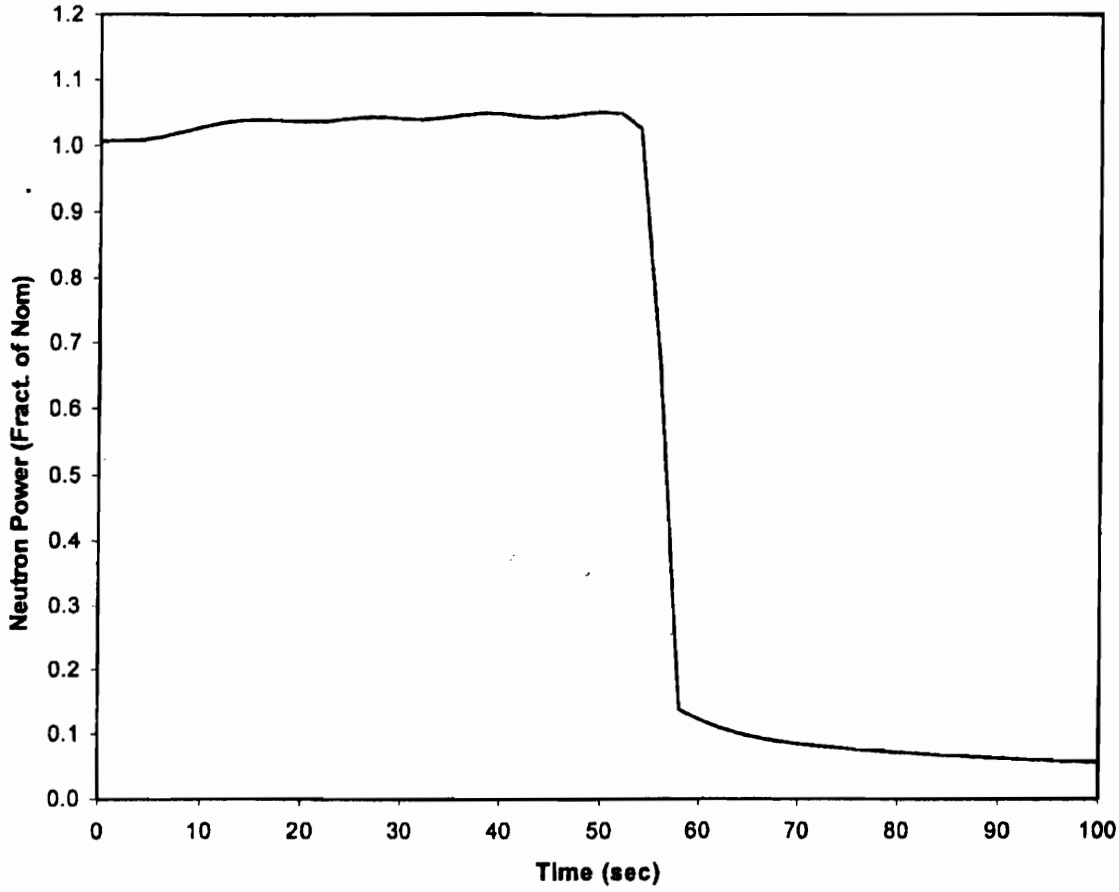


FIGURE 15.2.9-4b LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES - STEAM GENERATOR MASS

Revised by Amendment 15

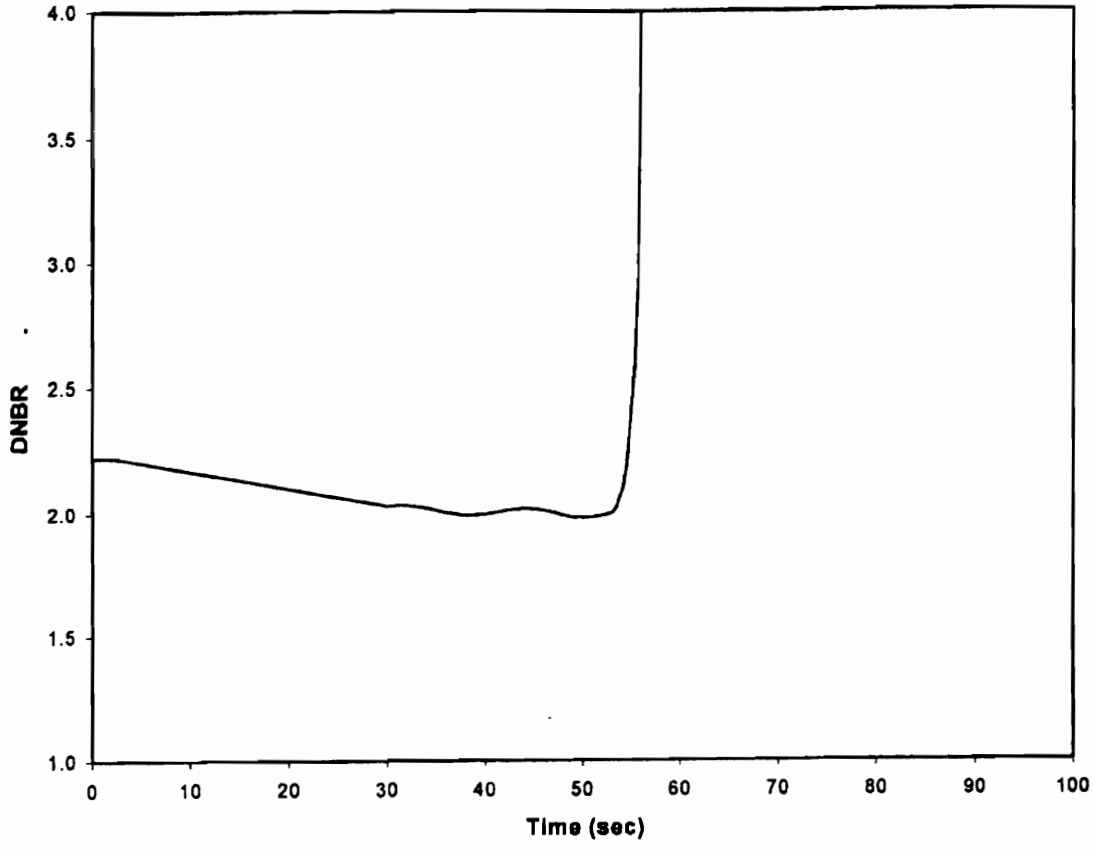


15.2.10-1 SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Nuclear Power verses time

Revised by Amendment 22

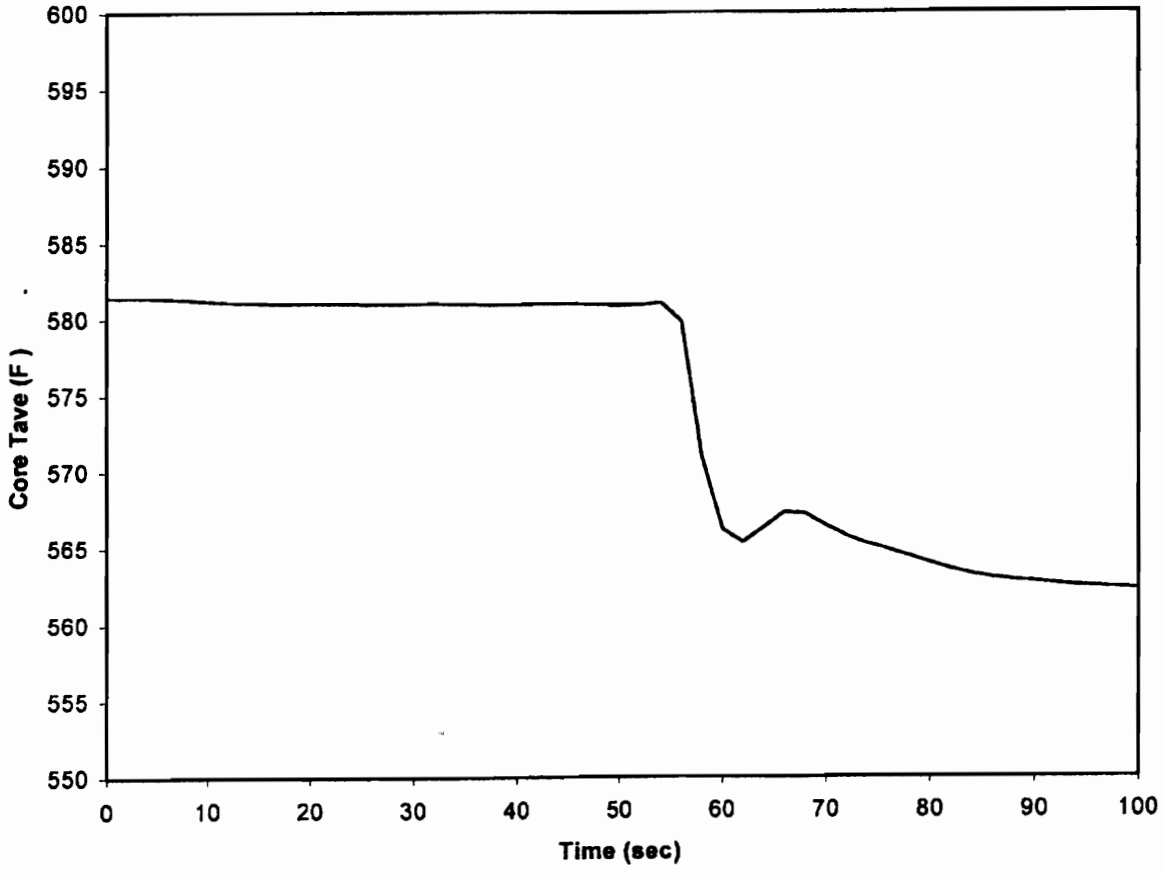
Best Available Historical Image



15.2.10-2 SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

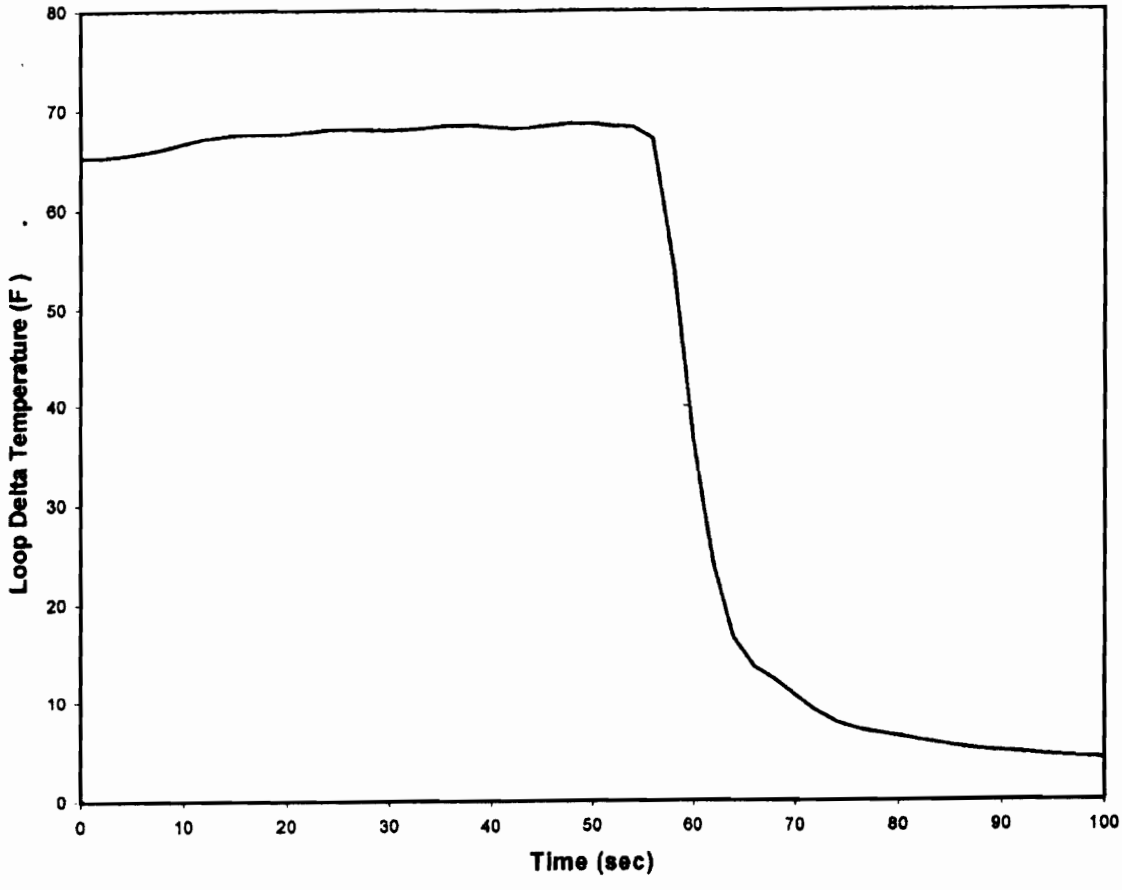
DNBR verses time



15.2.10-3 SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

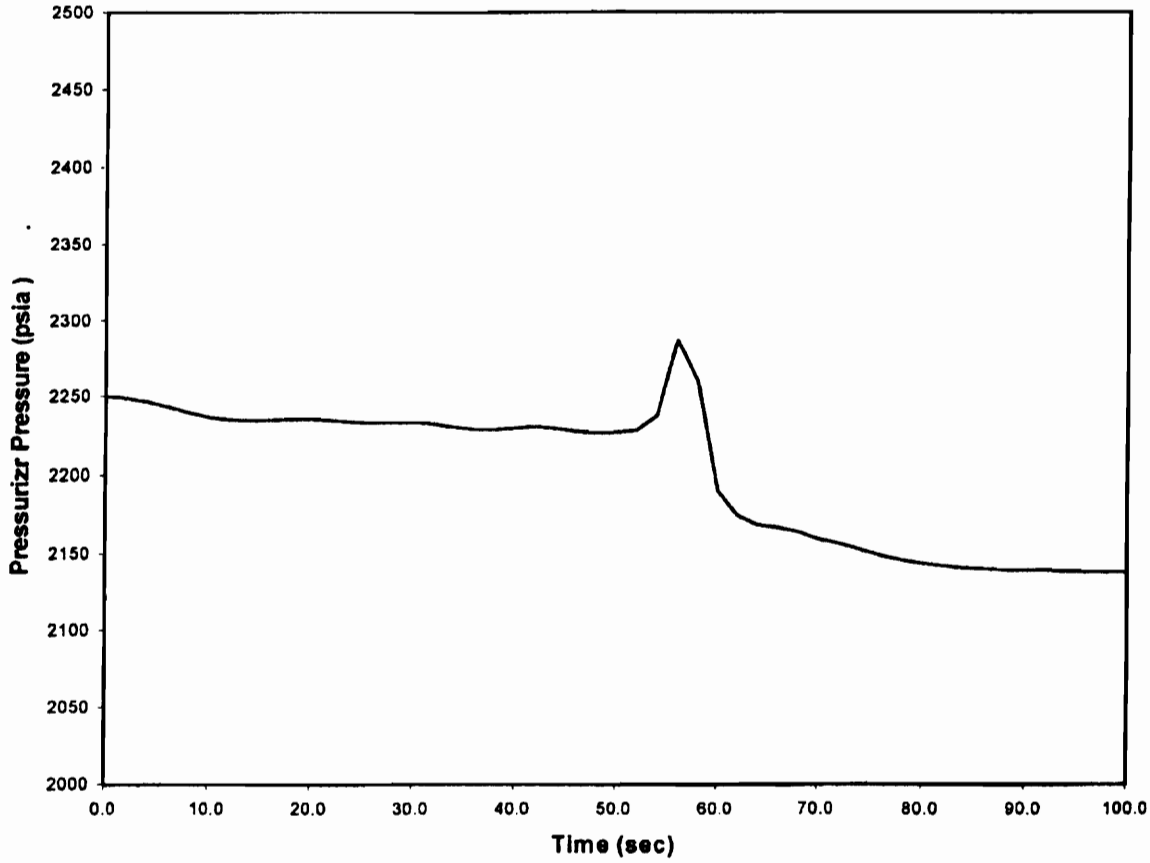
Core Tavg verses time



15.2.10-4 SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

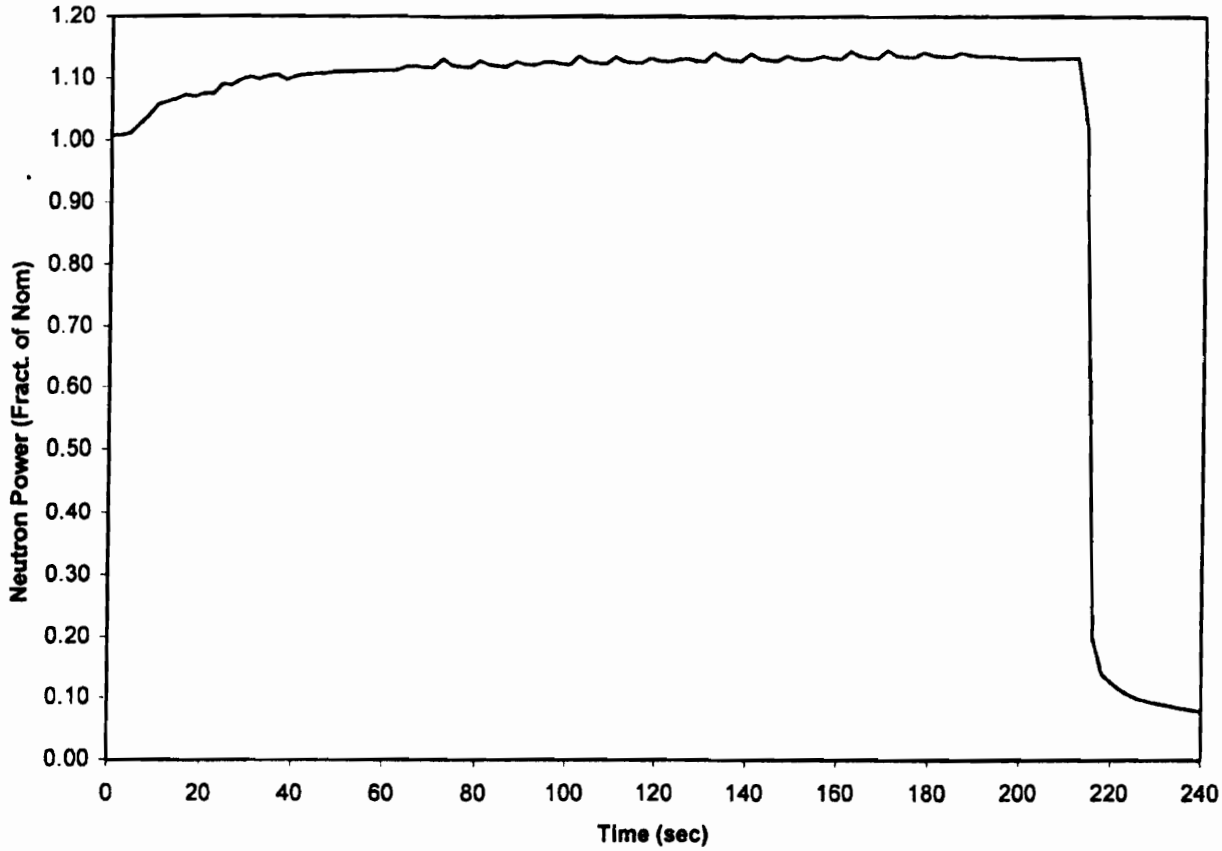
Loop Delta-T verses time



15.2.10-5 SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

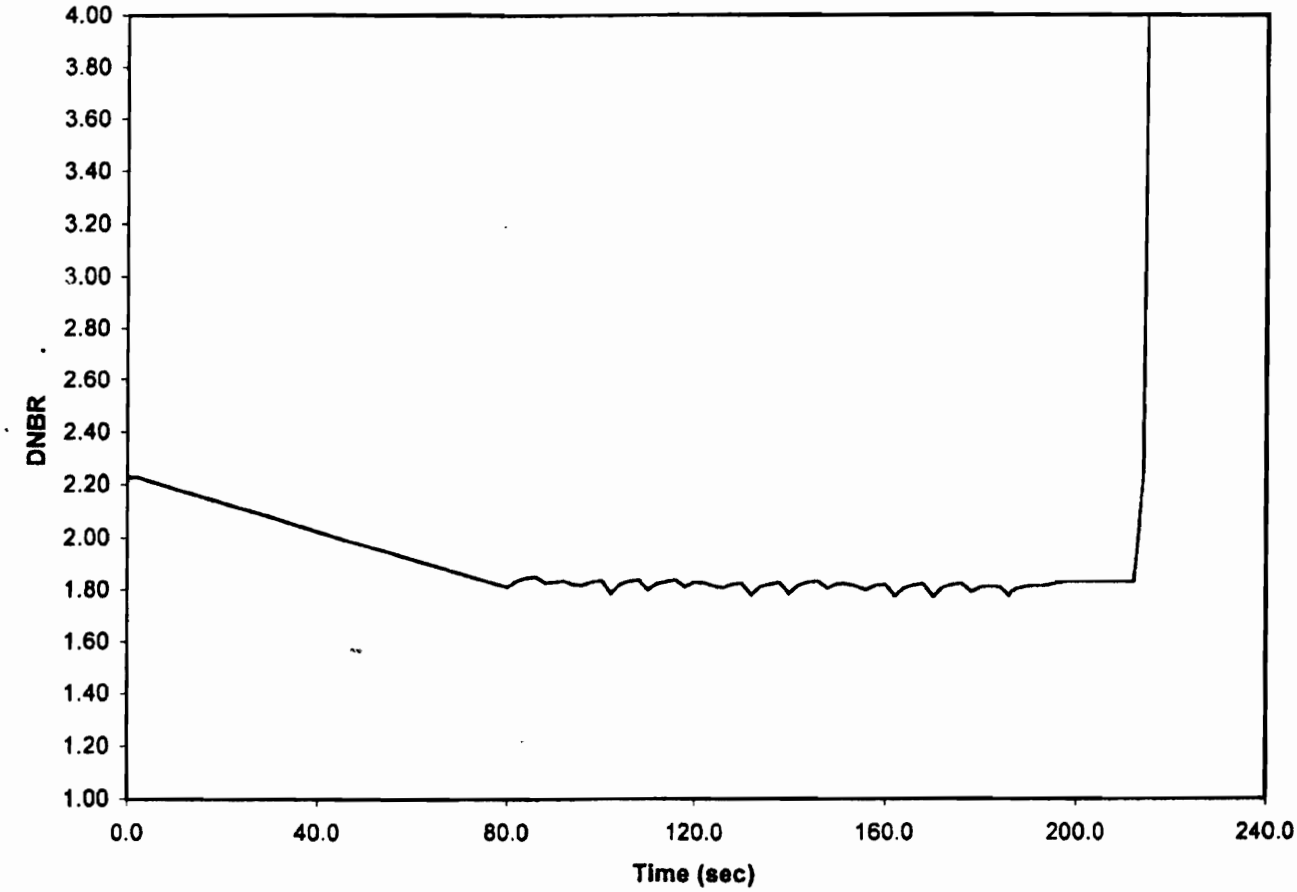
Pressurizer Pressure verses time



15.2.10-6 MULTIPLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

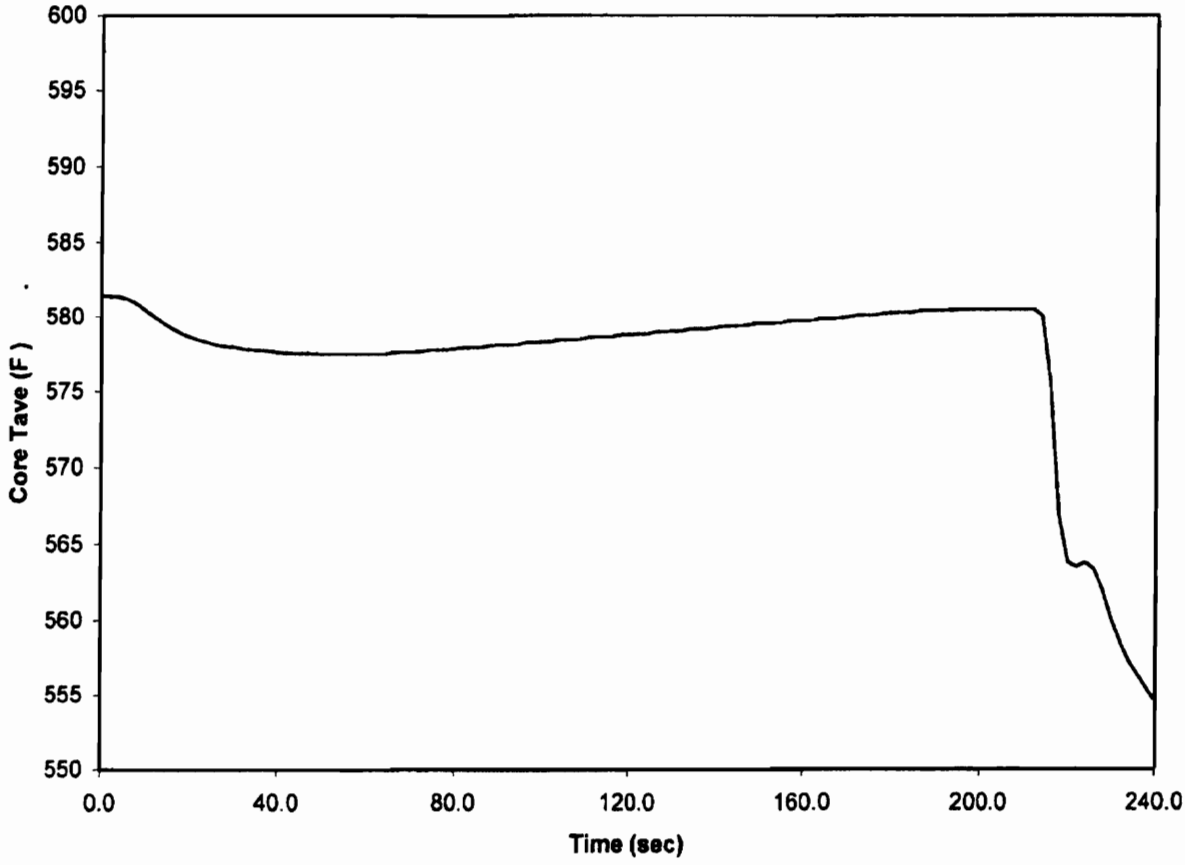
Nuclear Power verses time



15.2.10-7 MULTIPLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

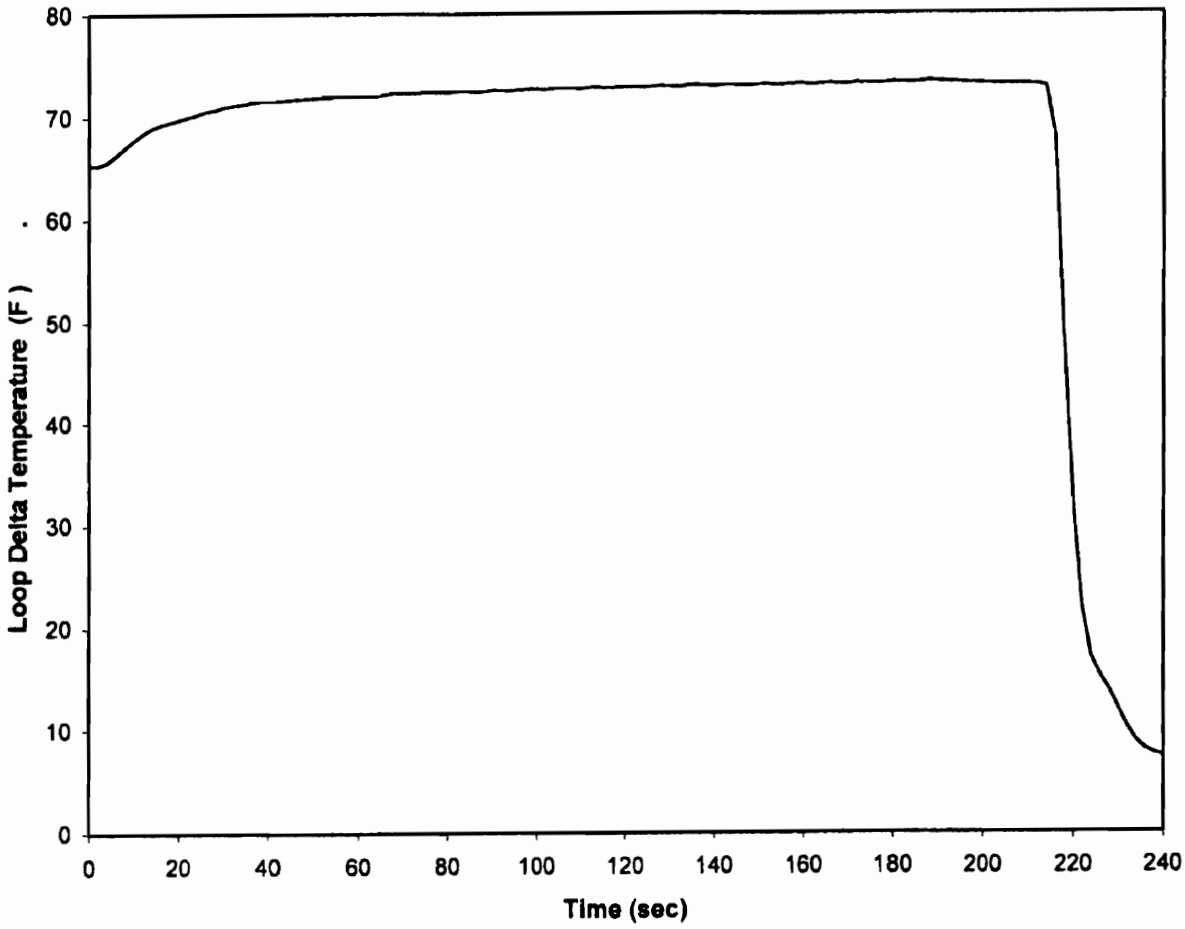
DNBR verses time



15.2.10-8 MULTIPLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

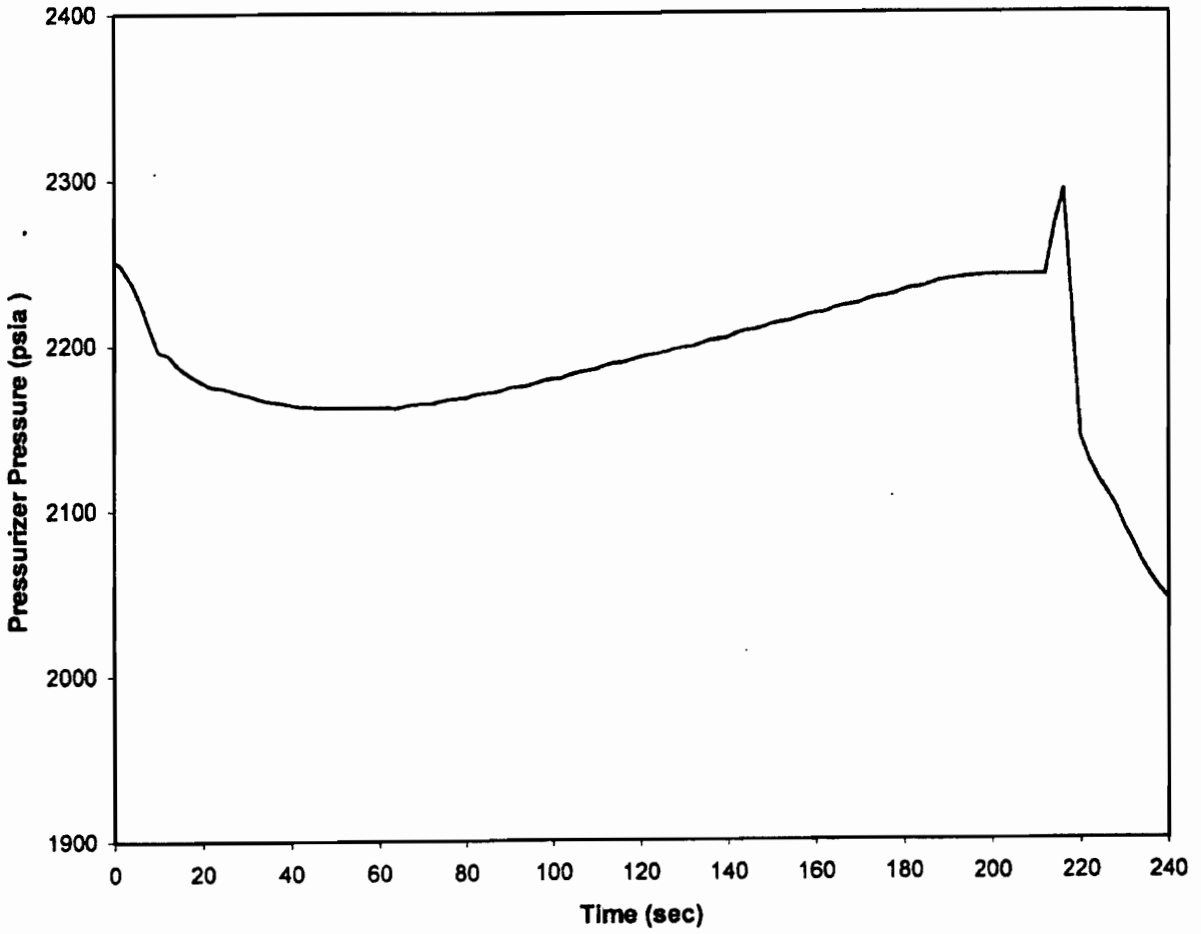
Core Tavg verses time



15.2.10-9 MULTIPLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

Loop Delta-T verses time



15.2.10-10 MULTIPLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - ANALOG FWC

Revised by Amendment 22

Pressurizer Pressure verses time

Figure 15.2.10-11 - SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - DIGITAL FWC

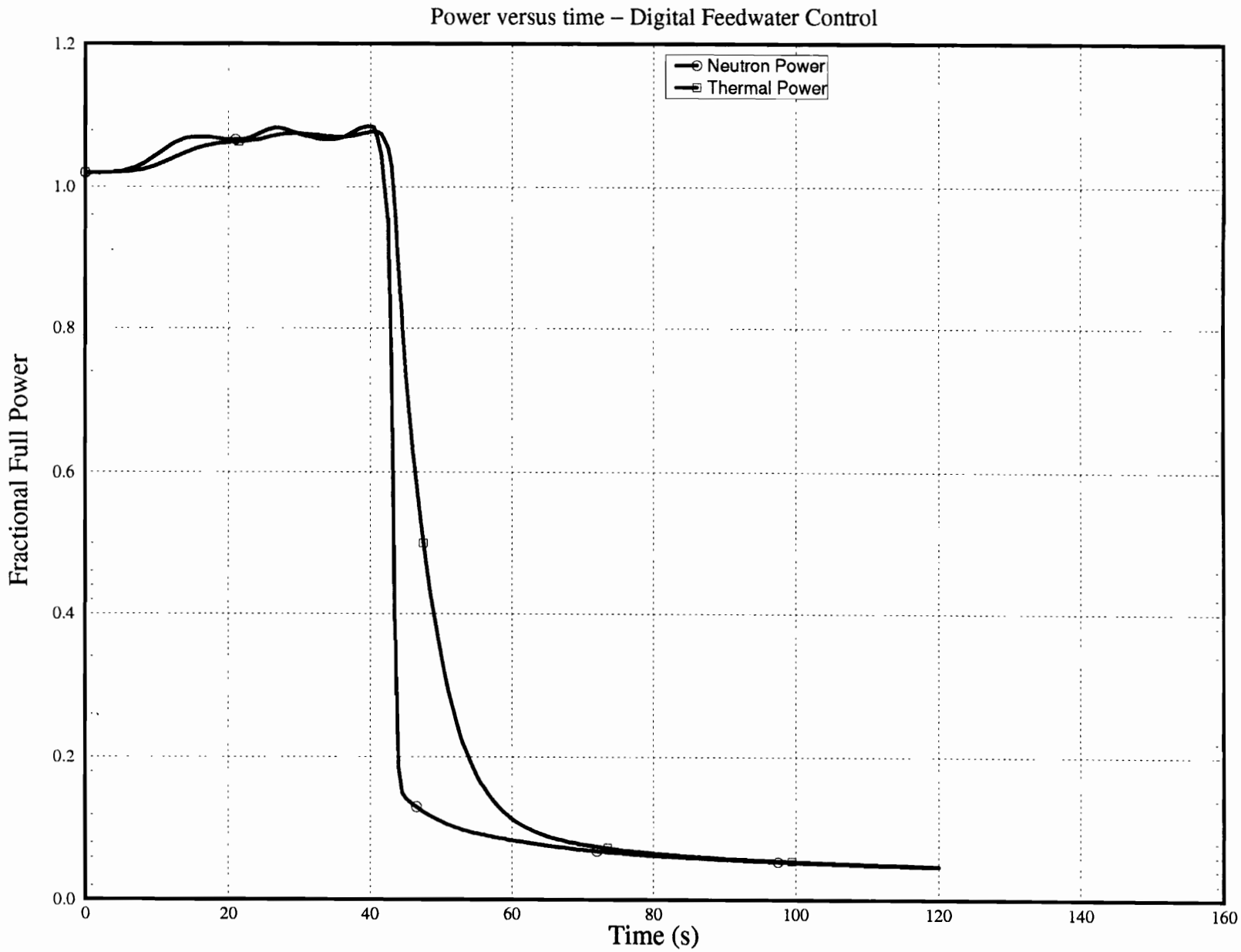


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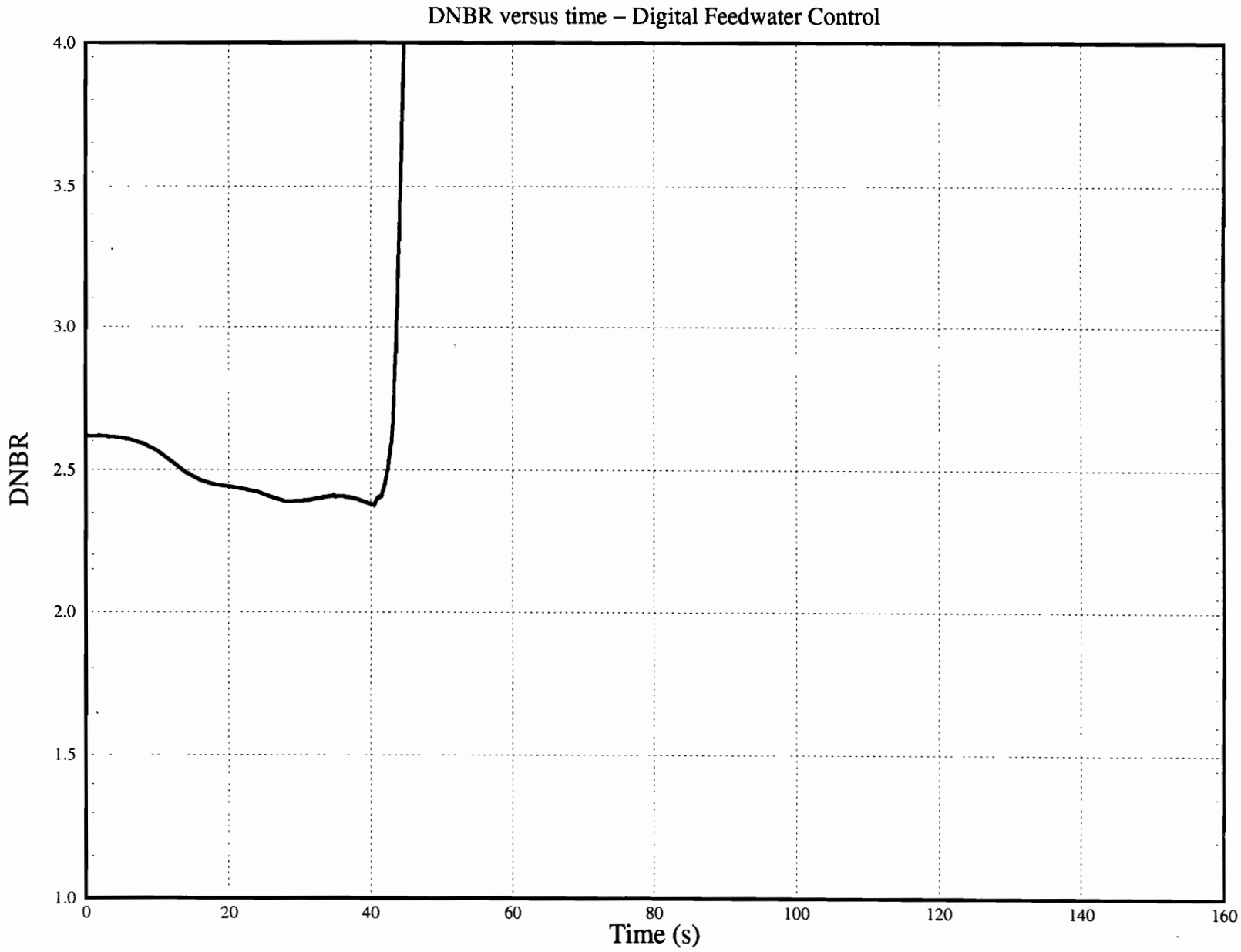
Figure 15.2.10-12 - SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - DIGITAL FWC



Revised by Amendment 22

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Figure 15.2.10-13 - SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - DIGITAL FWC

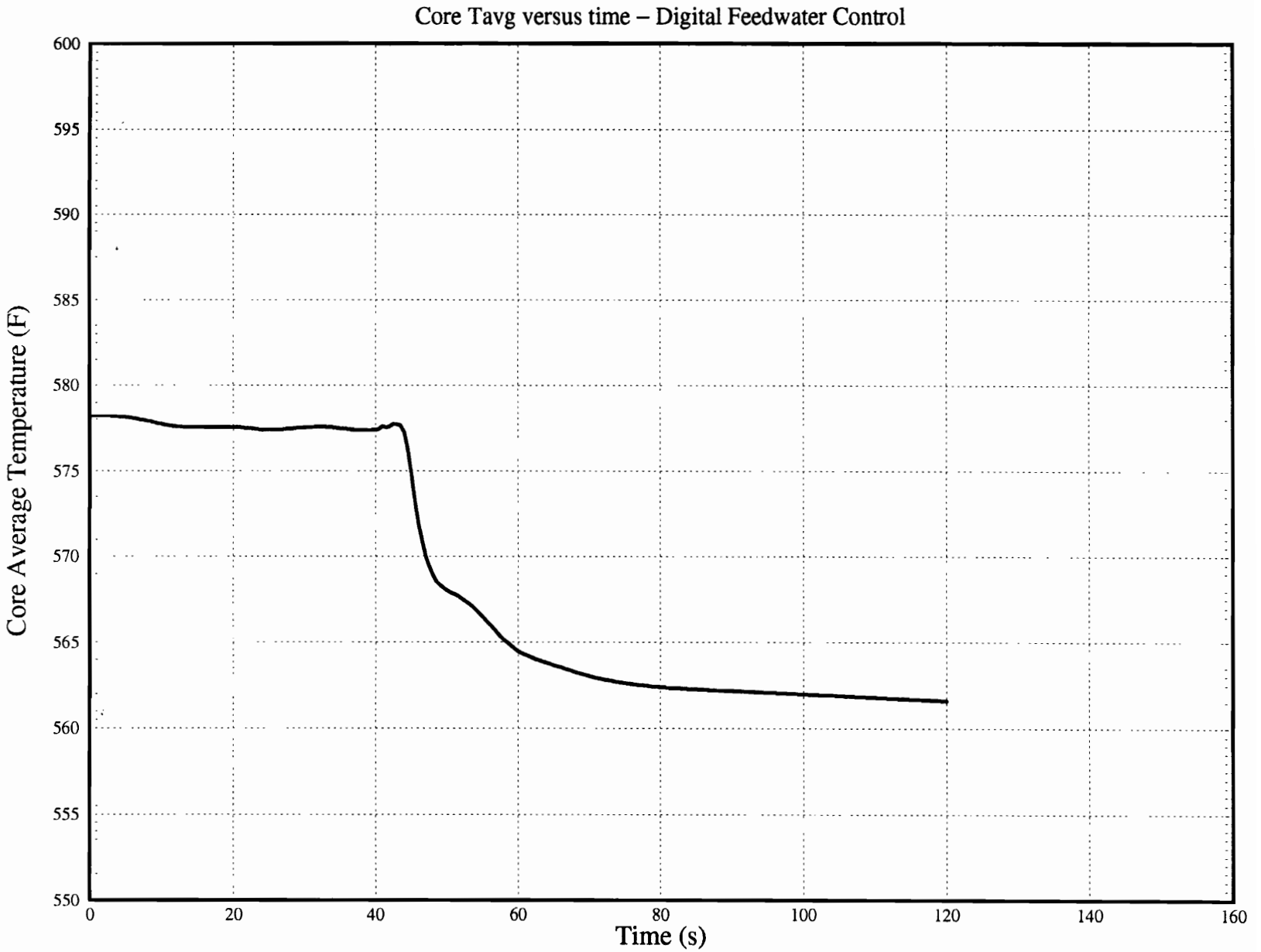

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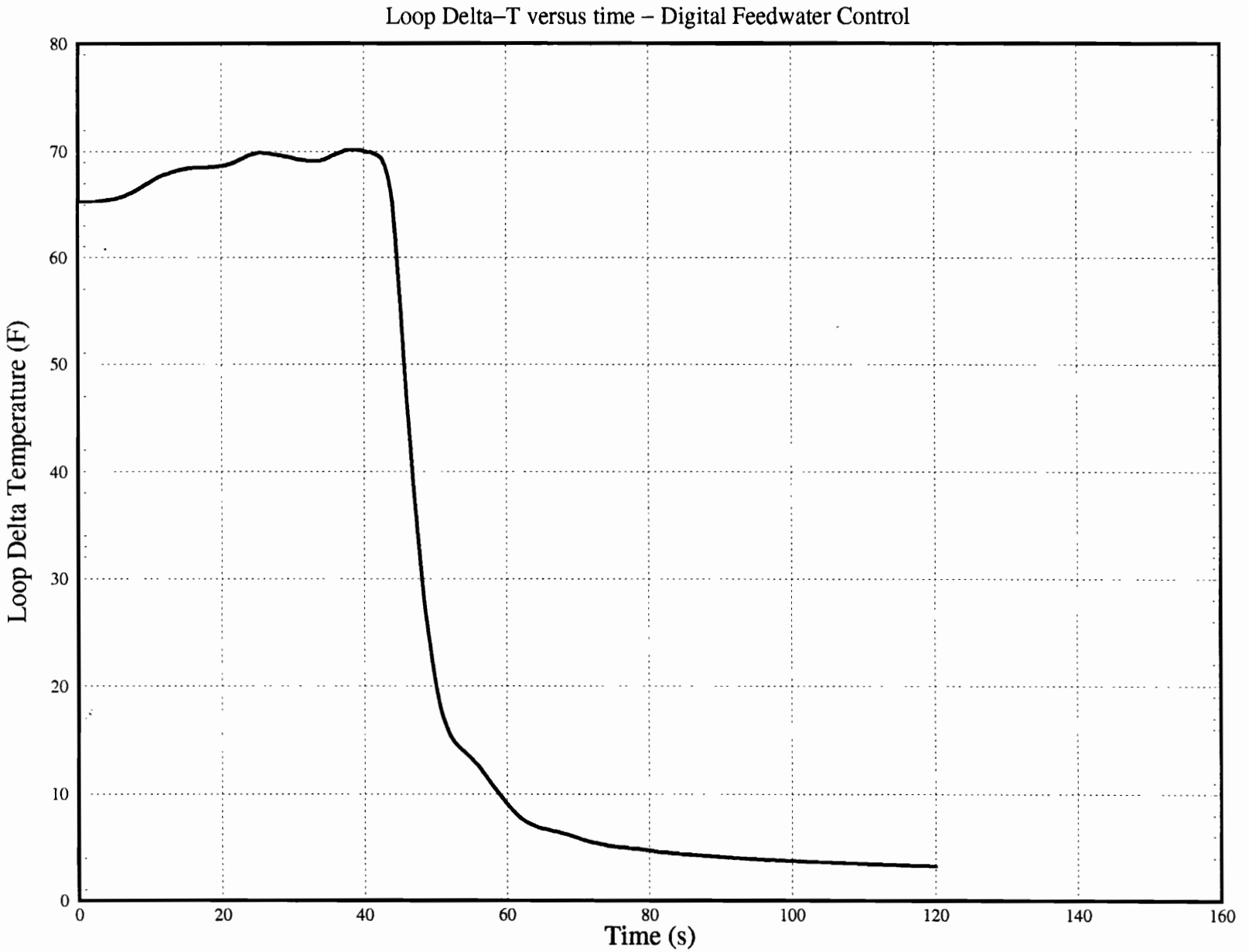
Figure 15.2.10-14 - SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - DIGITAL FWC



Revised by Amendment 22

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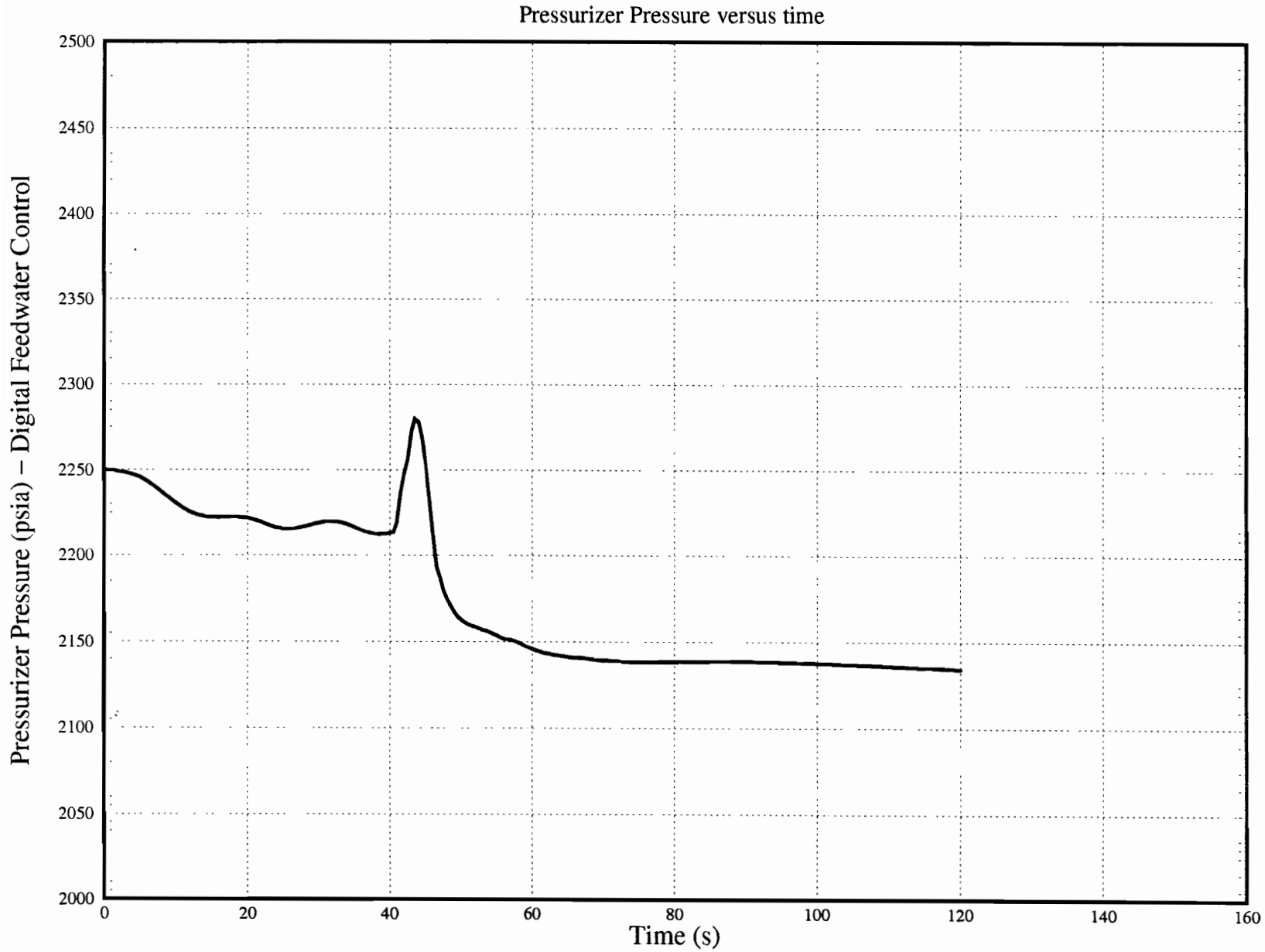
Figure 15.2.10-15 - SINGLE LOOP FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL AT FULL POWER - DIGITAL FWC



Revised by Amendment 22

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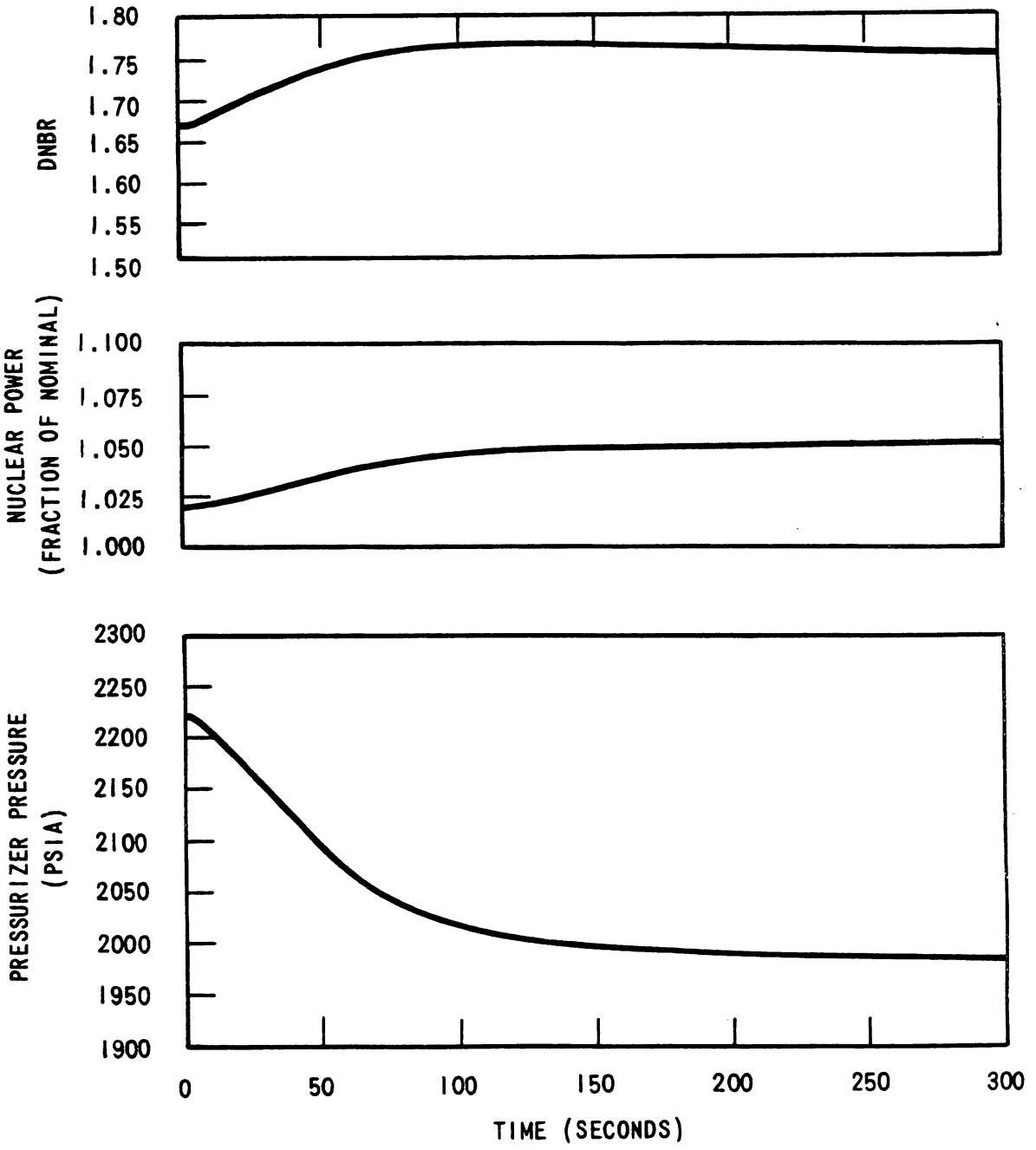


Figure 15.2.11-1 Excessive Load Increase with Manual Reactor Control, at Beginning of Life. DNBR, Nuclear Power and Pressurizer Pressure as a Function of Time

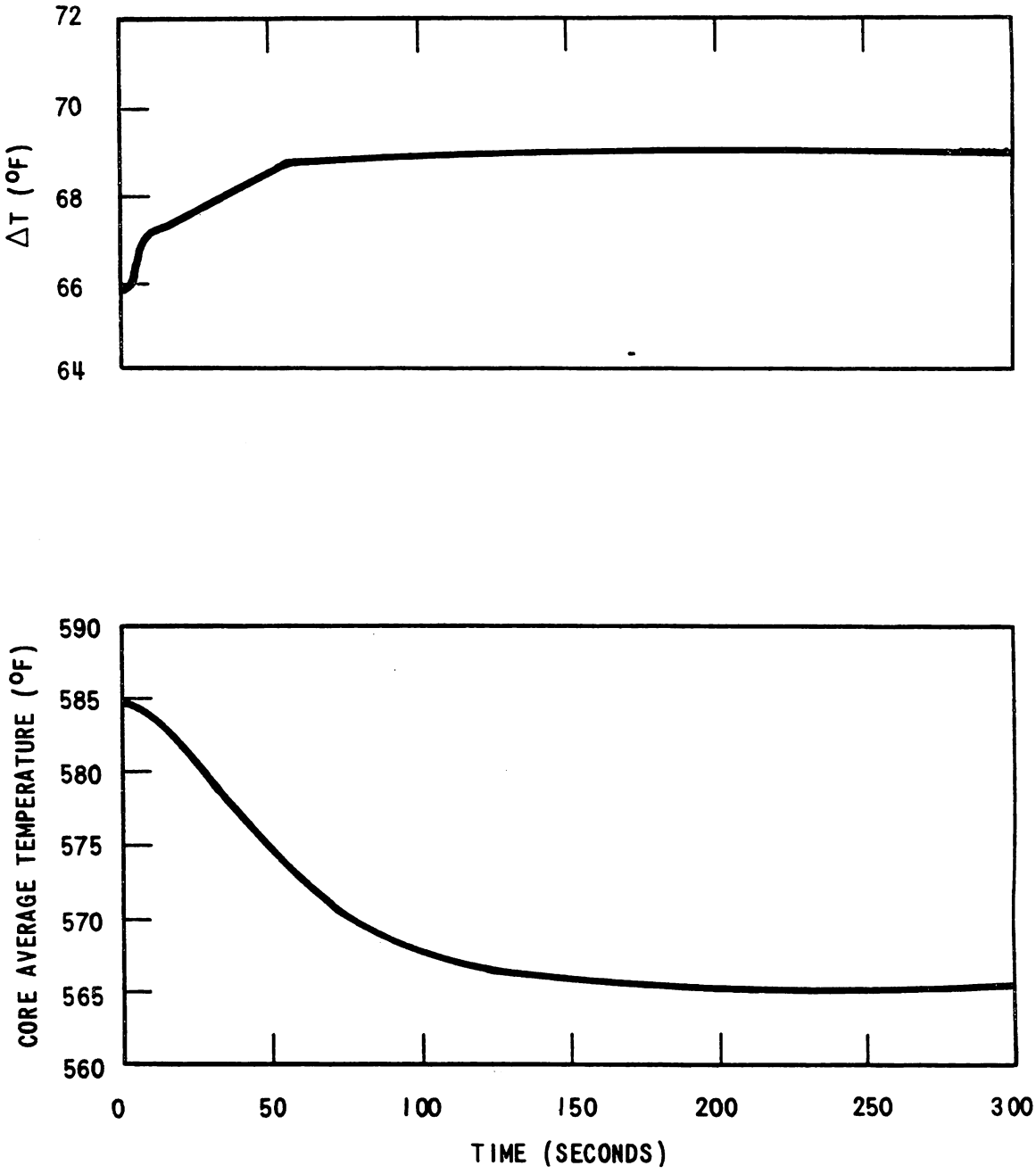


Figure 15.2.11-2 Excessive Load Increase with Manual Reactor Control, at Beginning-of-Life. ΔT and T avg as a Function of Time

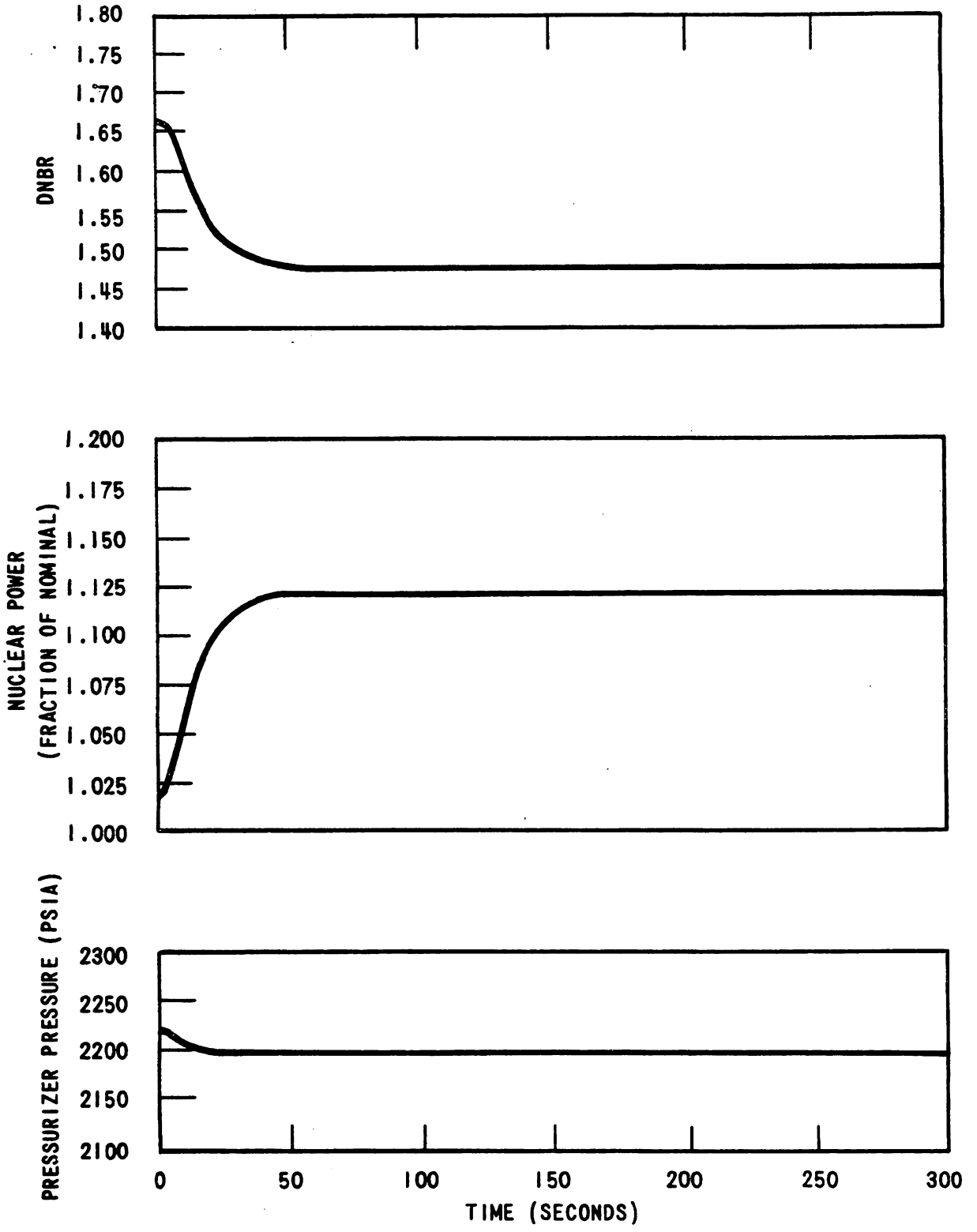


Figure 15.2.11-3 Excessive Load Increase with Manual Reactor Control at End-of-Life. DNBR, Nuclear Power and Pressurizer Pressure as a Function of Time

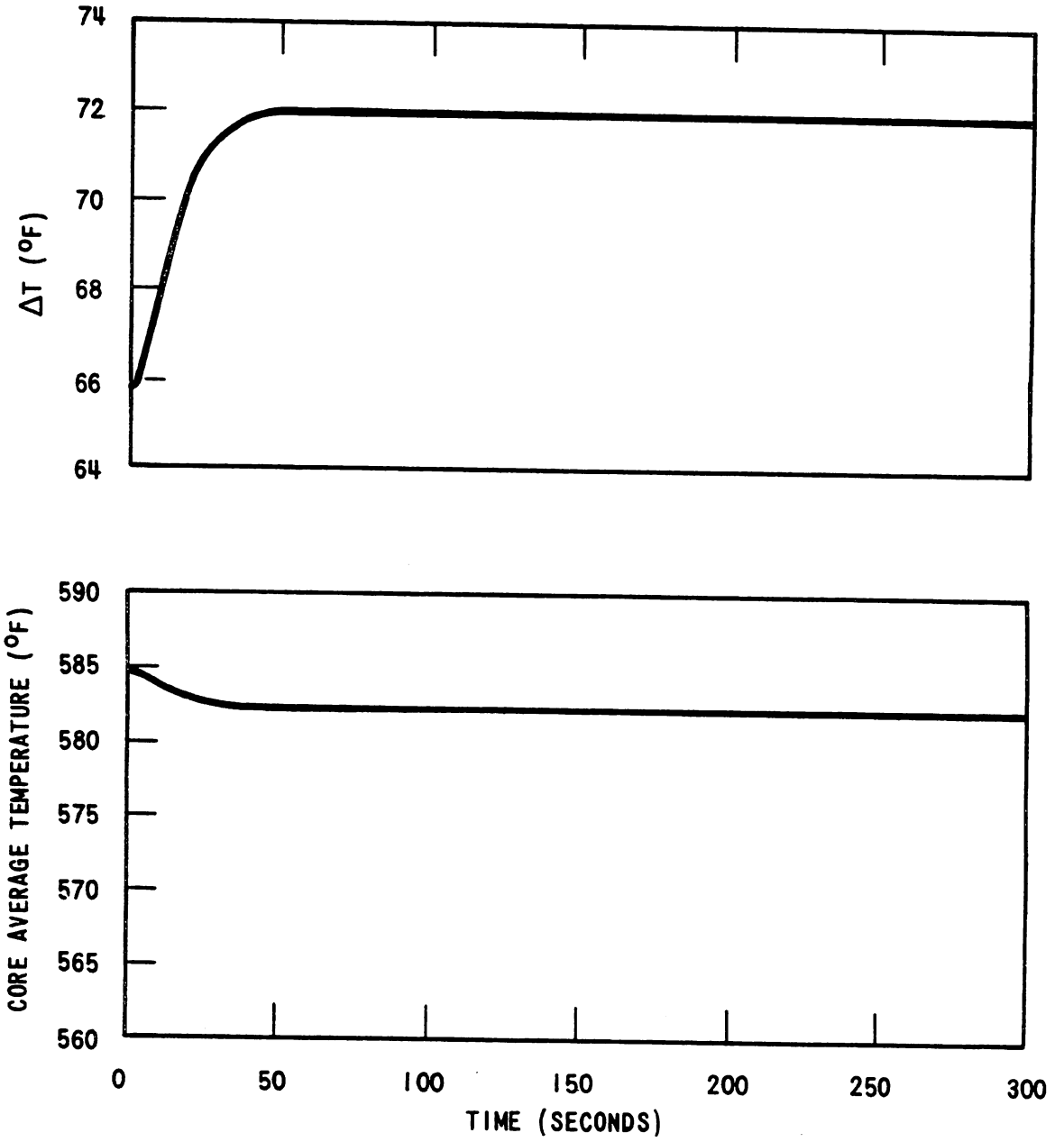


Figure 15.2.11-4 Excessive Load Increase with Manual Reactor Control, at End-of-Life. ΔT and T avg as a Function of Time

Best Available Historical Image

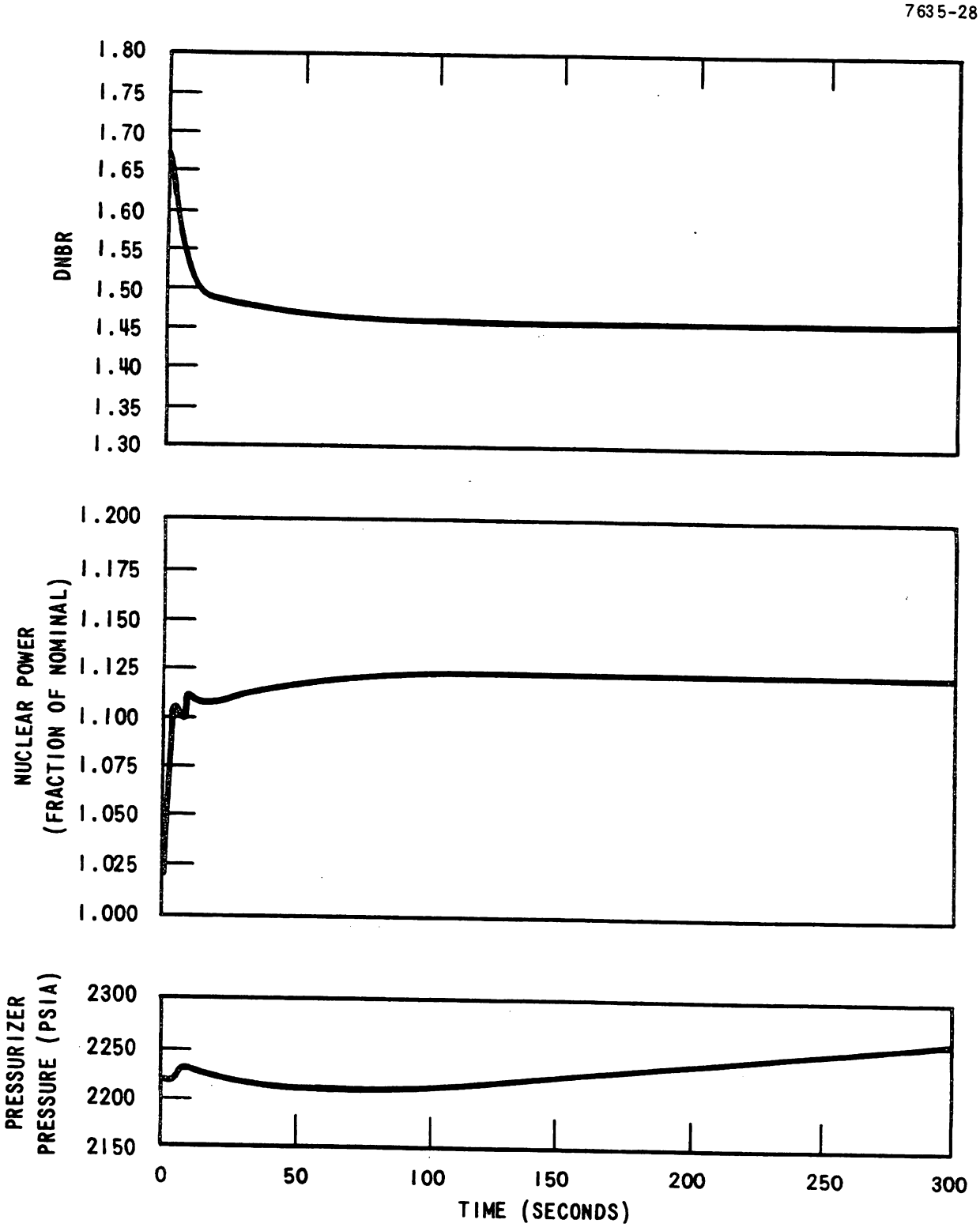


Figure 15.2.11-5 Exessive Load Increase with Automatic Reactor Control, at Beginning-of-Life. DNBR, Nuclear Power, and Pressurizer Pressure as a Function of Time.

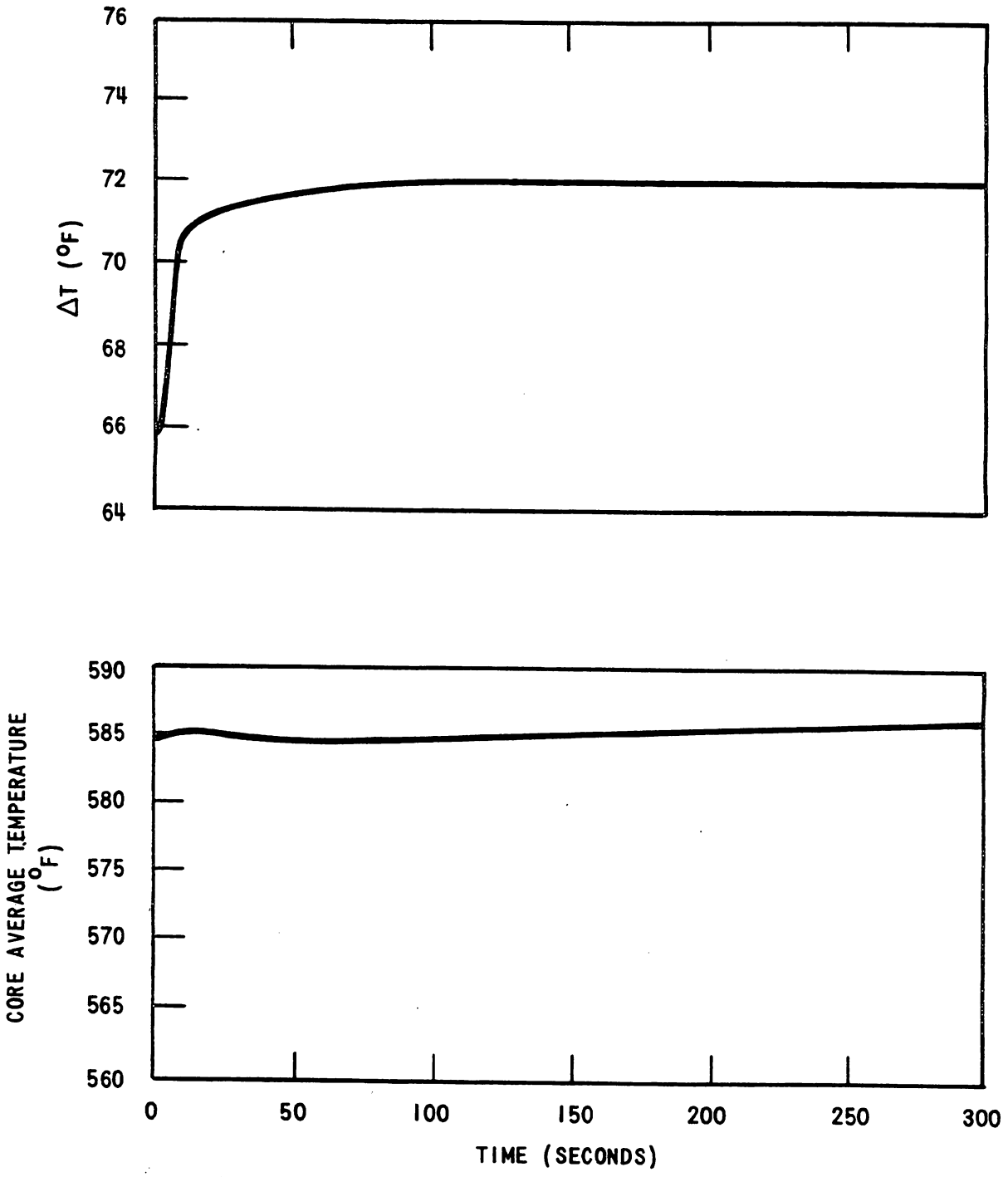


Figure 15.2.11-6 Excessive Load Increase with Automatic Reactor Control, at Beginning of Life. ΔT and T avg as a Function of Time

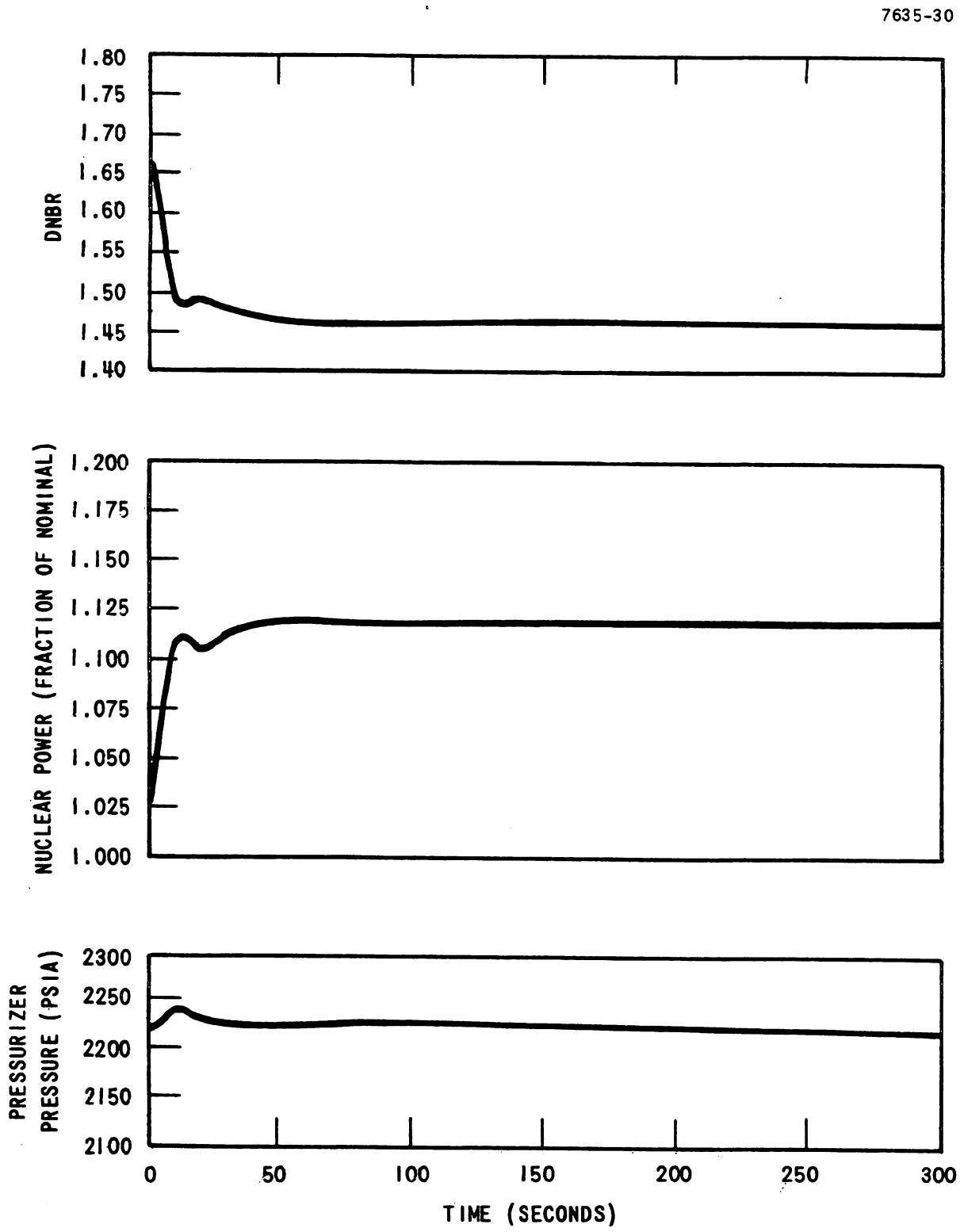


Figure 15.2.11-7 **Excessive Load Increase with Automatic Reactor Control, at End-of-Life, DNBR, Nuclear Power and Pressurizer Pressure as a Function of Time**

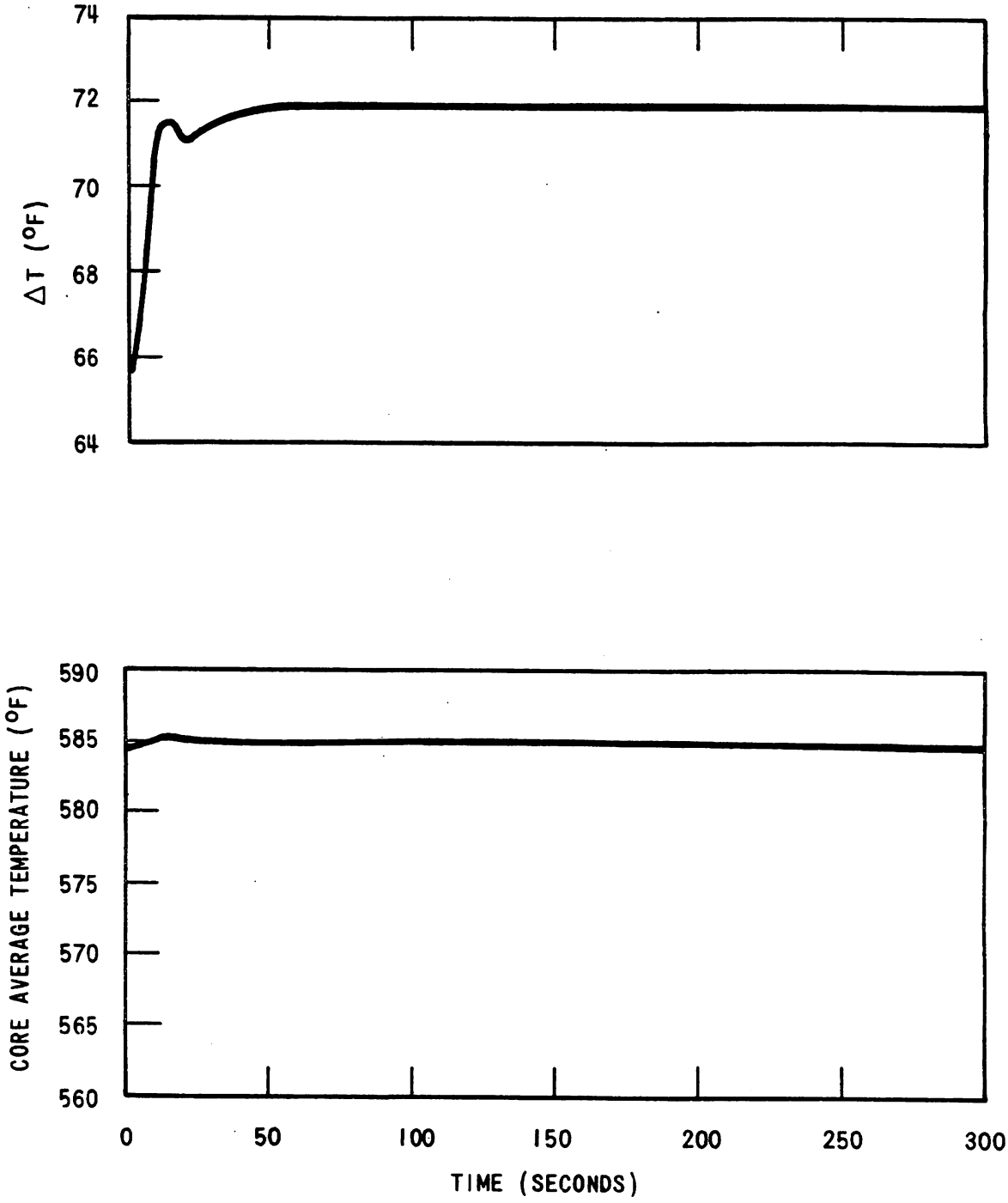


Figure 15.2.11-8 Excessive Load Increase with Automatic Reactor Control, at End-of-Life. ΔT and T (avg) as a Function of Time

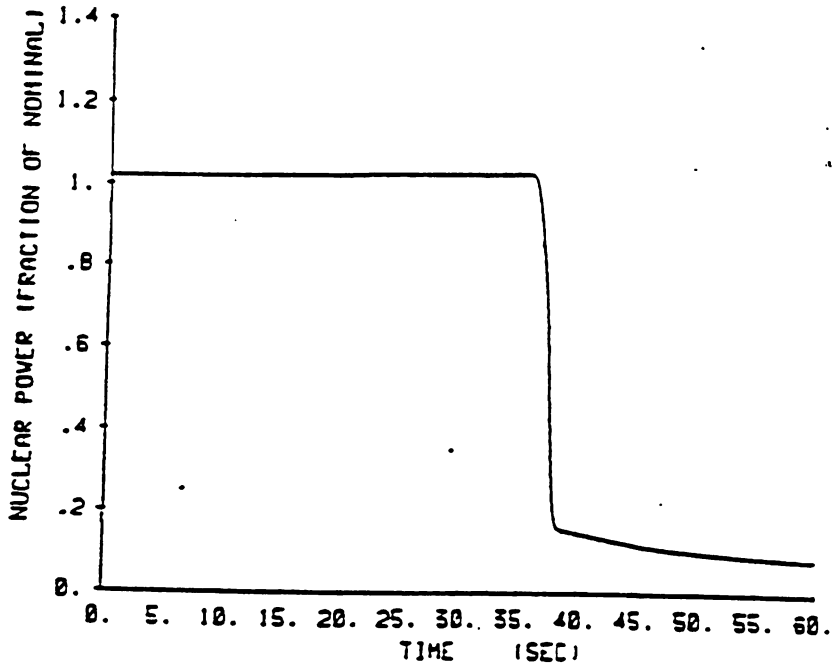
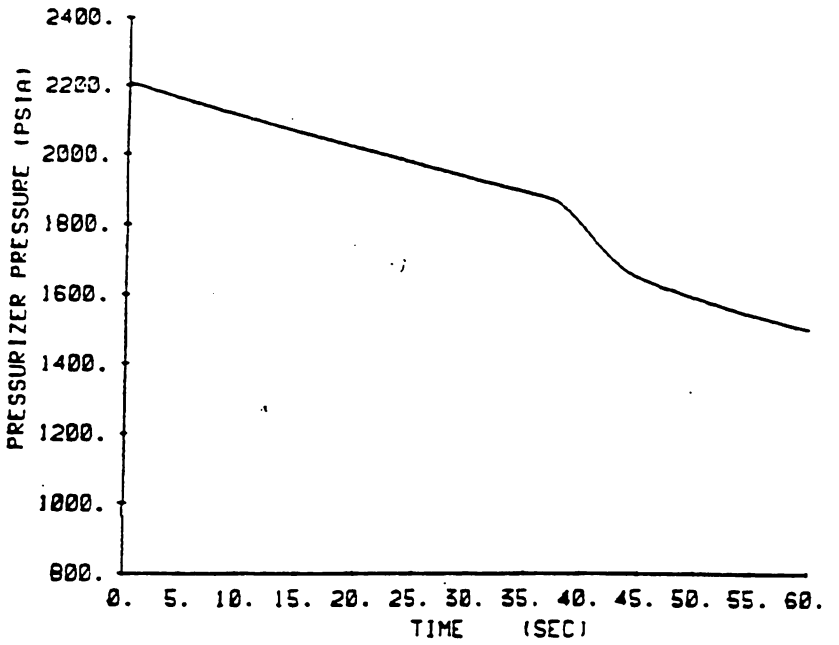


FIGURE 15.2.12-1

Nuclear Flux Transient for Accidental Depressurization



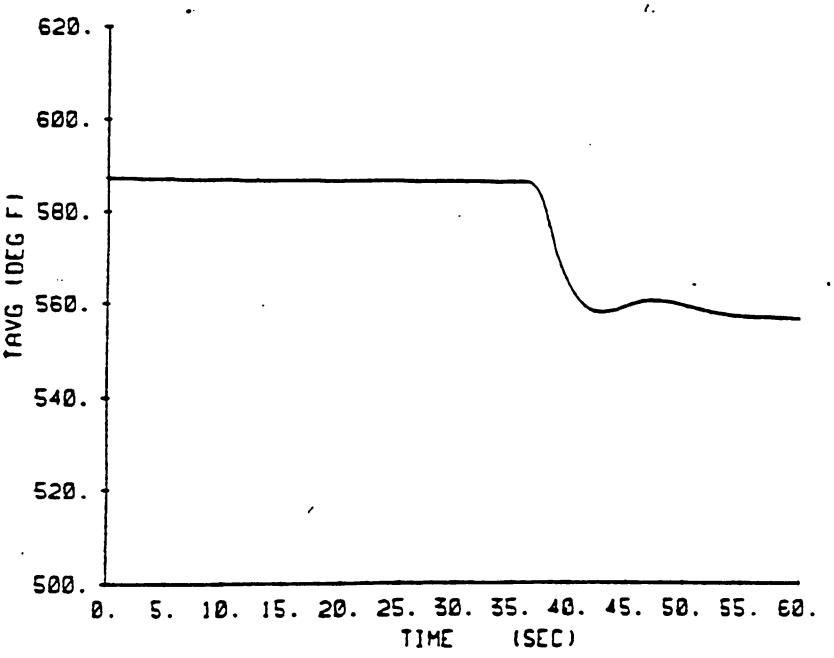


FIGURE 15.2.12-2

Pressurizer Pressure and Coolant Temp
Transients for Accidental Depressurization

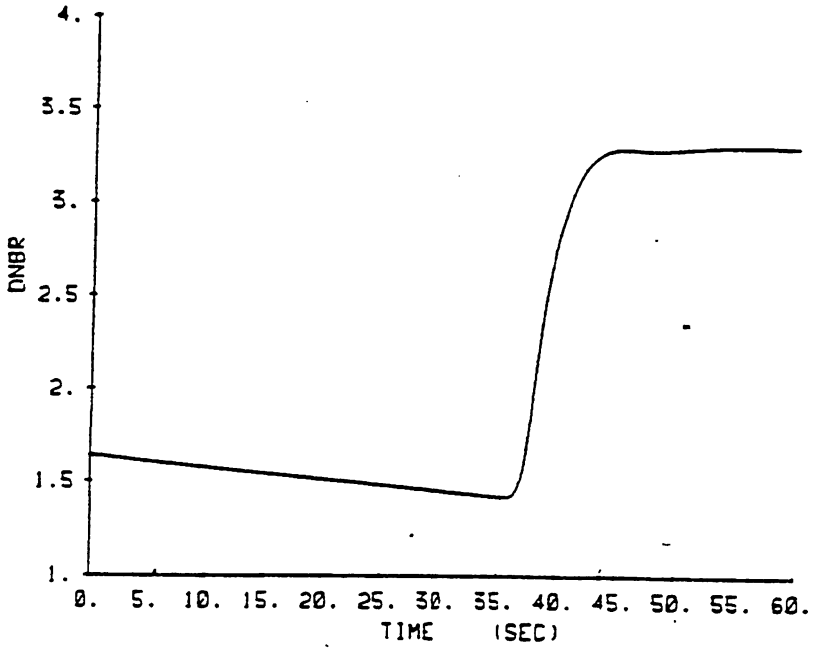


FIGURE 15.2.12-3

DNBR Transient for Accidental Depressurization

SQN

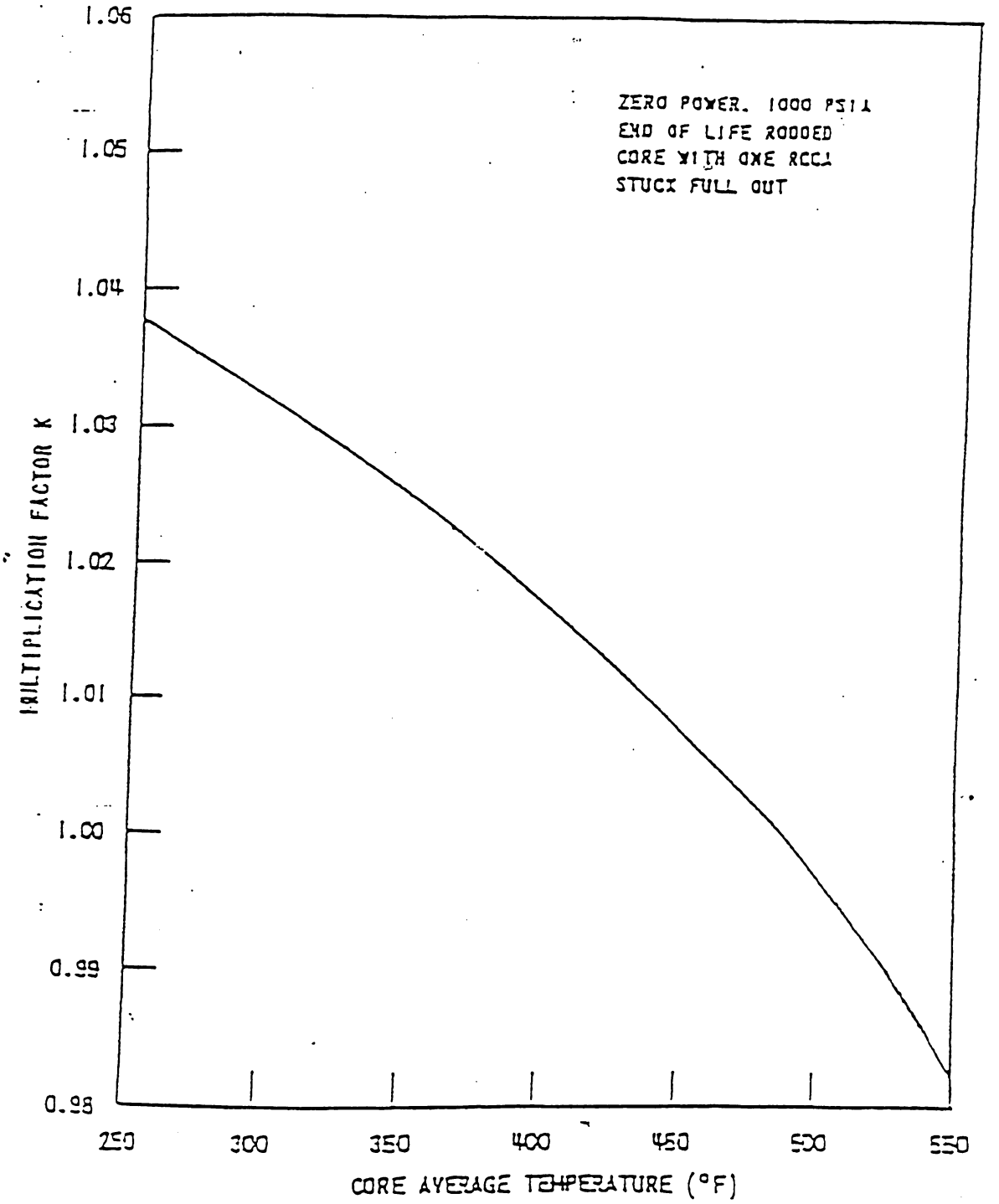


FIGURE 15.2.13-1 VARIATION OF Keff with CORE TEMPERATURE

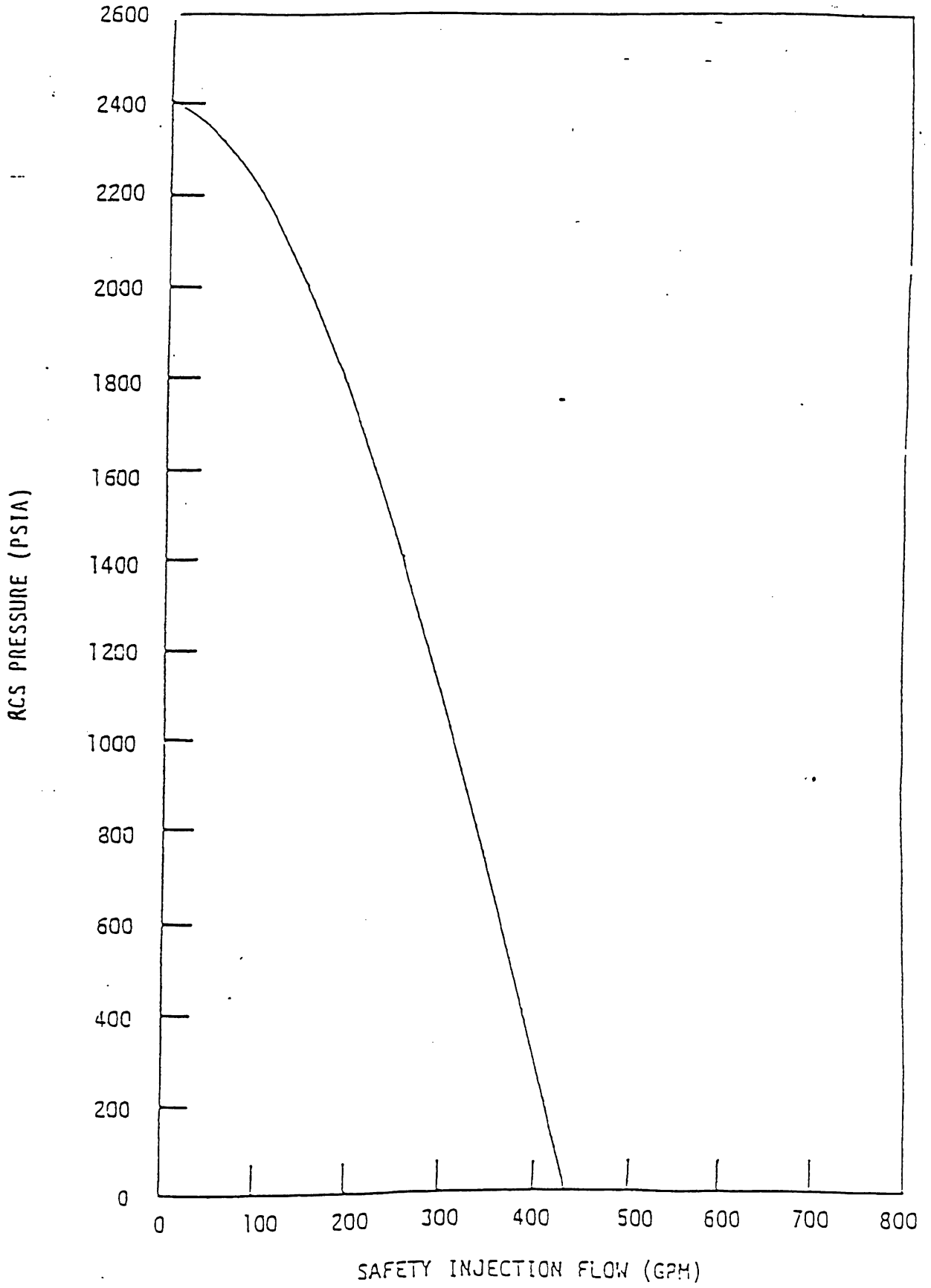


FIGURE 15.2.13-2 SAFETY INJECTION CURVE

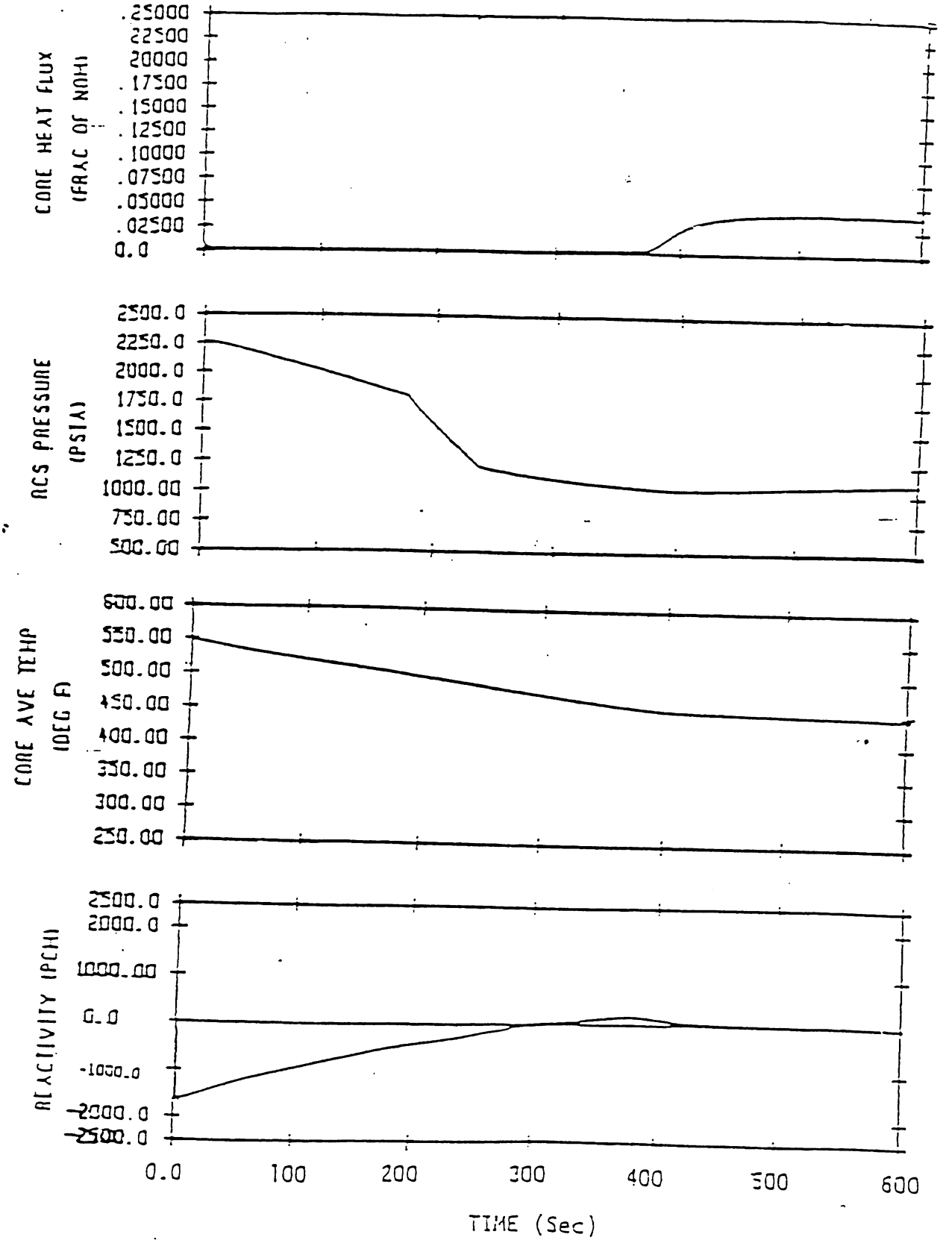


FIGURE 15.2.13-3 TRANSIENT RESPONSE FOR A STEAMLINE BREAK EQUIVAENT TO 228 LBS./SEC.AT 1015 PSIA WITH OUTSIDE POWER AVAILABLE

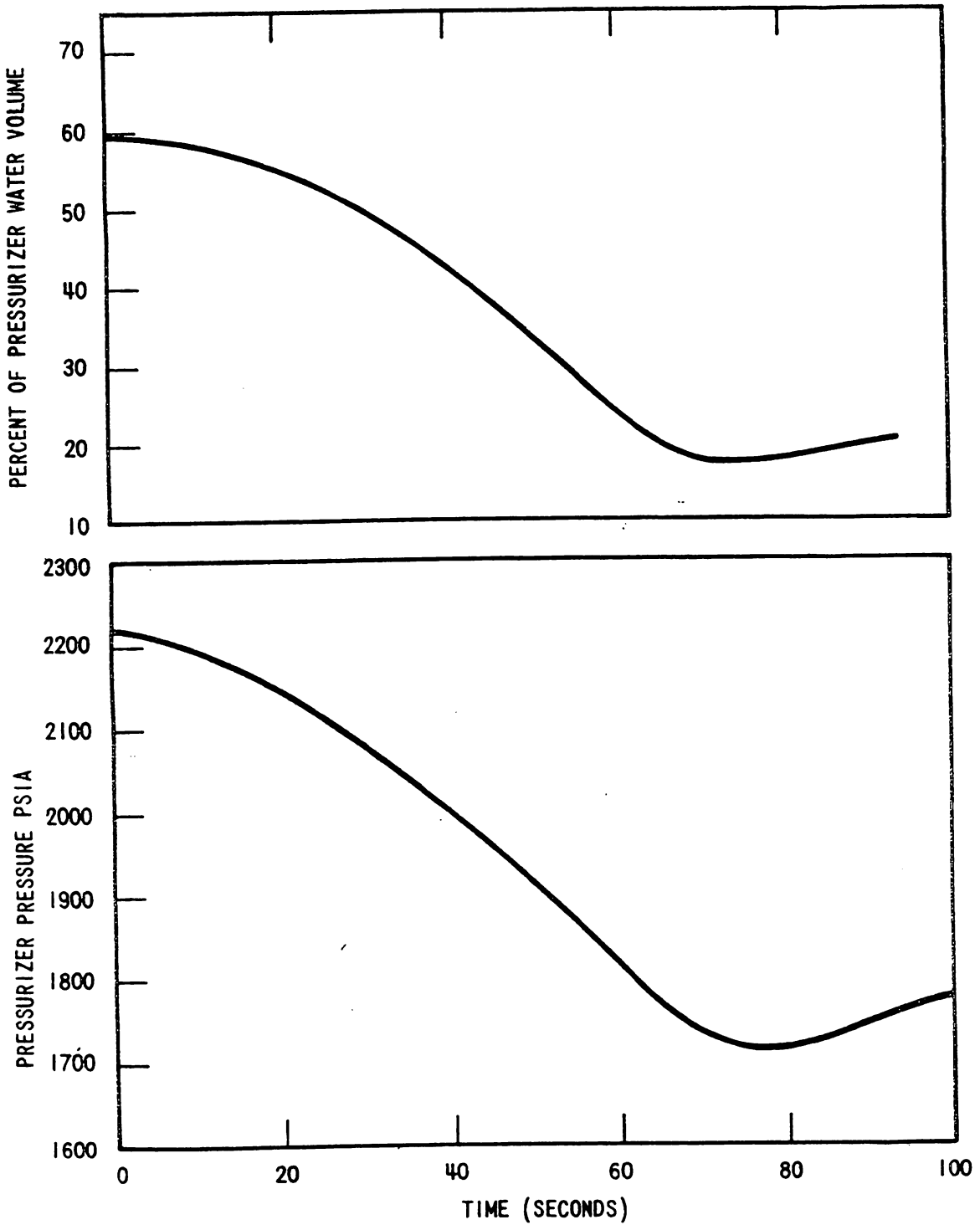
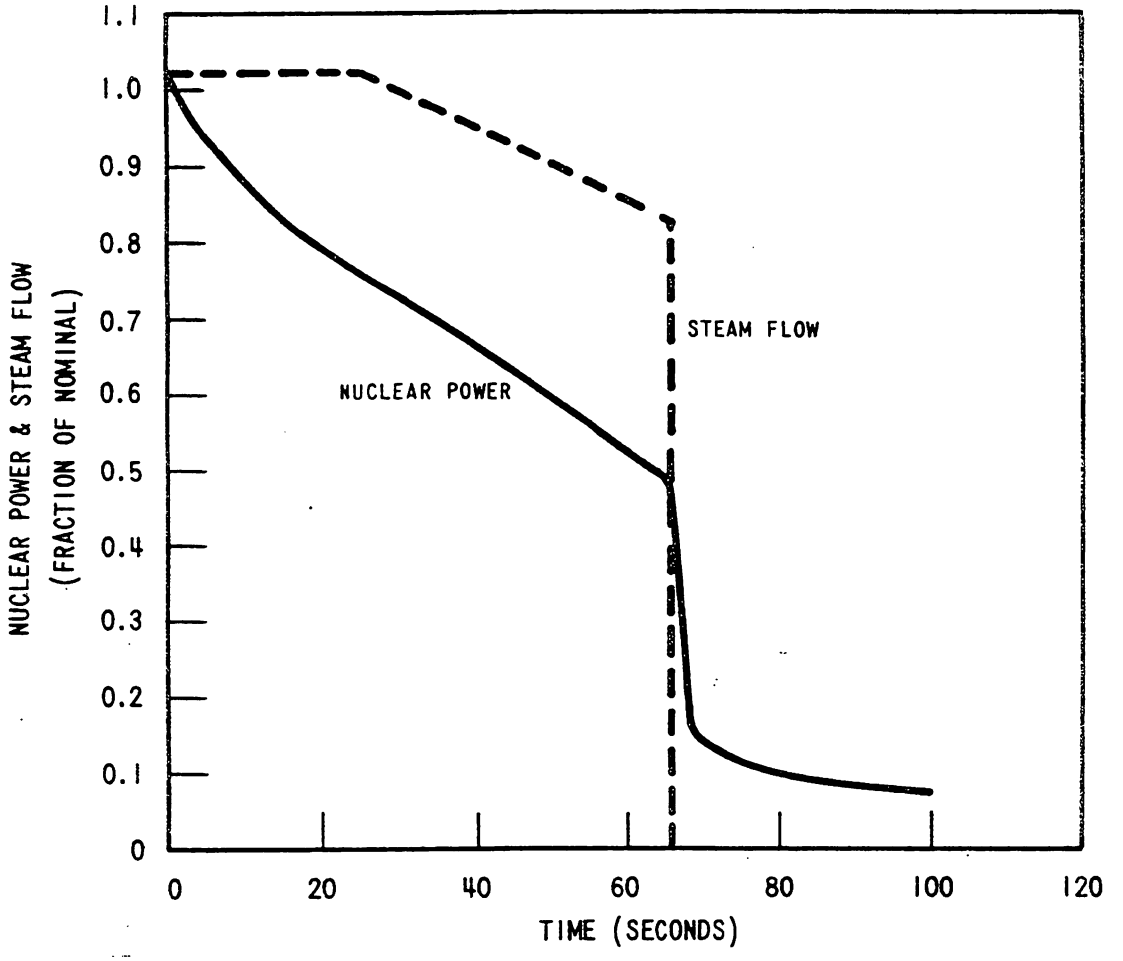


Figure 15.2.14-1 **Spurious Actuation of Safety Injection System at Power**

6937-129



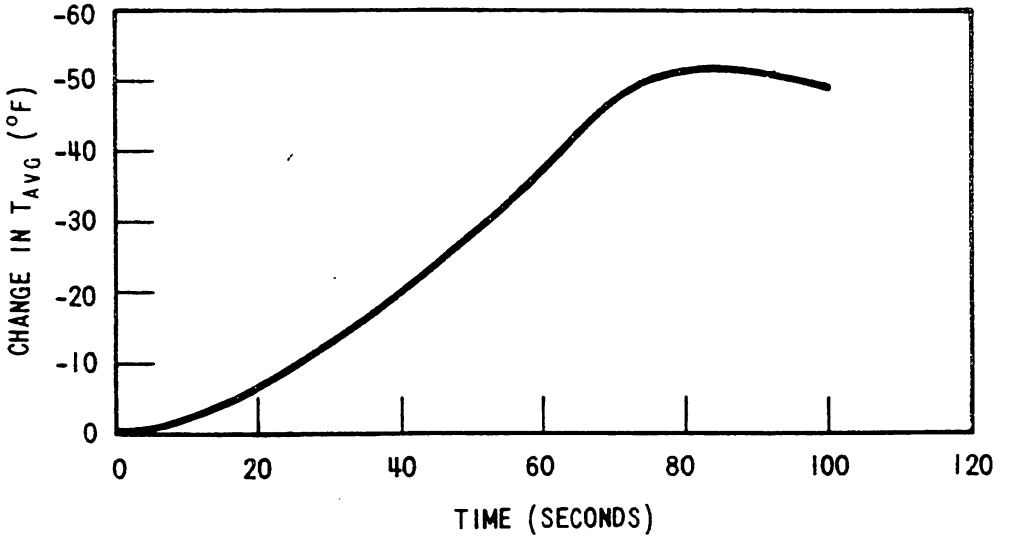


Figure 15.2.14-2 **Spurious Actuation of Safety Injection System at Power**

15.3 CONDITION III - INFREQUENT FAULTS

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

1. Loss of Reactor Coolant, from Small Ruptured Pipes or from Cracks in Large Pipes, which actuates Emergency Core Cooling.

2. Minor Secondary System Pipe Break.

3. Inadvertent Loading of a Fuel Assembly into an Improper Position.

4. Complete Loss of Forced Reactor Coolant Flow.

5. Waste Gas Decay Tank Rupture.

6. Single Rod Cluster Control Assembly Withdrawal at Full Power.

7. Steam Line Break Coincident With Rod Withdrawal at Power.

The time sequence of events during applicable Condition III faults 1, 4, and 7 above is shown in Tables 15.3.1-1, 15.3.4-1, and 15.3.7-1.

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

15.3.1.1 Identification of Causes and Accident Description

A loss of coolant accident (LOCA) is defined as a rupture of the reactor coolant system (RCS) piping or of any line connected to the system. See Section 3.6 for a more detailed description of the loss of reactor coolant accident boundary limits. Ruptures of small cross section will cause a loss of coolant at a rate that can be accommodated by the charging pumps. Pumped flow would maintain an operational water level in the pressurizer and permit the operator to execute an orderly shutdown.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging pump makeup flow at normal RCS pressure. Makeup flow rate from one centrifugal charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.375-inch diameter hole. Should a larger break occur, inventory loss through the break results in a reduction in pressurizer level and pressure. These breaks are considered small breaks of consequence and are examined in the analysis of this section.

At the time of break initiation, the plant is assumed to be operating at 102% of full power. Heat produced in the reactor core is exactly balanced by the heat transferred to the secondary side of the steam generator. As the RCS depressurizes, safety injection is initiated and the reactor is tripped on low pressurizer pressure. After reactor trip, heat continues to be added to the reactor coolant system by core decay heat and the hot passive heat structures.

The reactor coolant pumps are assumed to trip coincident with reactor trip. The safety injection signal isolates the main feedwater. Secondary inventory is subsequently controlled by auxiliary feedwater. The turbine will trip on reactor trip.

An important phenomenon during the small break LOCA transient is the occurrence of loop seal clearing, which is accompanied by a momentary core level depression. The momentary core depression causes a brief heatup in the upper core region. After loop seal clearing, the break mass flow rate is reduced with a phase change (predominantly liquid to predominantly steam) at the break. As a result of loop seal clearing, the RCS depressurizes; the rate of depressurization is primarily a function of break size and the number of loop seals clearing. The occurrence of loop seal clearing also marks the initiation of the core boildown phase of the transient.

During the boildown phase of the transient, a prolonged core heatup may occur depending on the ECCS pump performance. The ECCS injection rate is dependent on the system depressurization rate. Smaller breaks result in lower depressurization rates and lower ECCS flow but the mass lost to the break is also low, reducing the possibility of core uncovery. Larger small breaks cause a rapid depressurization of the RCS and the ECCS flow is sufficiently high to maintain the core in a liquid-covered state.

Intermediate small breaks are typically the worst in terms of core heatup. The break is large enough to cause a significant RCS mass loss. The depressurization rate is slow enough for these breaks to minimize pumped injection and the core may become uncovered.

Ultimately, the small break transient is mitigated by the pumped ECCS injection and/or the passive (accumulator) injection. Reactor power is reduced rapidly by reactor trip and the injection of borated water by the ECCS. The ECCS injection also serves to make up for the mass lost to the break and floods the core, ensuring prolonged decay heat removal.

15.3.1.2 Small Break LOCA Evaluation Model

The RELAP5/MOD2-B&W code is used to predict the RCS thermal-hydraulic and core heatup responses to a small break LOCA. The code has been approved by the NRC for licensing application, and is documented in detail in Reference 1. Methods and models associated with the application of RELAP5/MOD2-B&W to small break LOCA analysis are described at length in Volume II of Reference 2.

The Sequoyah reactor core model is divided radially into two regions. One region representing the hot channel and the other representing the average channel (remainder of the core). The core is further segmented axially to allow detailed computation of hot assembly cladding and vapor temperatures and provides resolution of core mixture level to within approximately 0.5 feet. Fuel pin initial condition parameters are calculated with the TACO3 computer code (Reference 9). The RCS model contains sufficient detail to represent the coolant void distribution that affects the system hydrostatic balance during a small break.

The steam generator tube region is divided into two radial regions. One region for the shortest half of the tubes and the other region for the remainder of the tubes. This provides sufficient modeling accuracy to simulate tube length-induced draining effects.

15.3.1.3 Small Break LOCA Spectrum Analysis

Break spectrum results in Reference 2 (Volume II, Appendix A) indicate that break areas from 2- to 6-inch in diameter produce the most severe core depression. Six breaks in the bottom of the pump discharge piping with break areas equivalent to 2.5-, 2.75-, 3-, 3.25-, and 5- inch diameters were, therefore, included in the Sequoyah small break analysis. A 1.34-inch diameter centrifugal charging line break, located in the top of the pump discharge pipe, was also analyzed.

The total core peaking distribution assumed in all of the small breaks is shown in Figure 15.3.1-1. The power peaks at the 10.0-ft core elevation, maximizes local power in the upper region of the core, and is conservative for small break analysis because of the top-down core uncovering process. As the core uncovers, the cladding in the upper elevations of the core heats up and is sensitive to the local power at that elevation. The cladding temperature in the lower elevation of the core, below the two-phase mixture height, remains near the coolant temperature. The peak power used in the analysis is based on a total peaking factor, Fq, of 2.65.

A single failure in pumped ECCS injection is assumed that minimizes RCS liquid inventory during the small break transient. One train of charging, SI, and RHR pumps are assumed operable and aligned for injection. The total flow rates associated with these systems are provided in Figure 15.3.1-2. For the charging line break, the charging flow to the intact loops is conservatively neglected.

Table 15.3.1-1 presents a time sequence of events predicted in the small break LOCA spectrum studies and Table 15.3.1-2 summarizes the results of the hot rod heatup calculations. Principal parameters of interest resulting from the 2.75-inch break case are presented in Figures 15.3.1-3 through 15.3.1-7. These figures depict primary system pressure, leak flow rate, mixture level, loop seal levels, and hot spot cladding temperature.

15.3.1.4 Conclusions

The results of the small break LOCA spectrum studies performed with the BWNT evaluation model show that the small break LOCA is not limiting with respect to large break LOCA results. The predicted peak cladding temperature is 1403° F for the pump discharge break with an area equivalent to 2.75-inch diameter pipe. With the limited temperature excursion, the local and whole-core metal-water reaction percentages are negligible. The hot pin thermal transient is insufficient to cause significant fuel pin deformation and the core remains amenable to cooling. Further, recovery of the core is demonstrated and continued operation of the ECCS will guarantee long-term cooling. The analysis, therefore, demonstrates that a significant safety margin exists between the calculated results and the 10CFR50.46 limits and that compliance with regulatory acceptance criteria is met.

The Small Break LOCA event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.3.2 Minor Secondary System Pipe Breaks

15.3.2.1 Identification of Causes and Accident Description

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

15.3.2.2 Analysis of Effects and Consequences

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

The analysis of the more probable accidental opening of a secondary system steam dump, relief or safety valve is presented in Subsection 15.2.13. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

15.3.2.3 Conclusions

The analysis presented in Subsection 15.4.2 demonstrate that the consequences of a minor secondary system pipe break are acceptable since a DNBR of less than 1.3 does not occur even for a more critical major secondary system pipe break.

15.3.3 Inadvertent Loading of a Fuel Assembly into an Improper Position

15.3.3.1 Identification of Causes and Accident Description

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

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. The incore system of moveable flux detectors which is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

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

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with in-core flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. In-core flux measurements are taken during the startup subsequent to every refueling operation.

15.3.3.2 Analysis of Effects and Consequences

Method of Analysis

Steady state power distribution in the x-y plane of the core are calculated using the TURTLE (Reference 5) code based on macroscopic cross section calculated by the LEOPARD (Reference 6) code. A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The power distributions in the x-y plane for a correctly loaded core assembly are also given in Chapter 4 based on enrichments given in that section.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown at all-in-core detector locations (see Figures 15.3.3-1 to 15.3.3-5, inclusive).

Results

The following core loading error cases have been analyzed:

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange of two adjacent assemblies near the periphery of the core (see Figure 15.3.3-1).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case (see Figures 15.3.3-2 and 15.3.3-3).

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

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

Case C:

Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.3.3-4).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.3.3-5).

15.3.3.3 Conclusions

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

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

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

This event is characterized by the misplacement of a fuel assembly in the reactor core. The event is statically examined using various neutronics codes. No system interaction or effects are included in the event analyses. The results of the analyses are completely independent of steam generator design. Thus, results with OSGs are equally applicable to operation of Sequoyah Unit 1 with the RSGs.

15.3.4 Complete Loss of Forced Reactor Coolant Flow

15.3.4.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. 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. The following provide necessary protection against a loss of coolant flow accident:

1. Undervoltage or underfrequency on reactor coolant pump power supply busses.

2. Low reactor coolant loop flow.

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

The reactor trip on reactor coolant pump underfrequency is provided to open the reactor coolant pump breakers and trip the reactor for an underfrequency condition, resulting from frequency disturbances on the major power grid. The trip disengages the reactor coolant pumps from the power grid so that the pumps' kinetic energy is available for full coastdown.

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. It also serves as a backup to the undervoltage and underfrequency trips. This function is generated by two out of three low flow signals per reactor coolant loop. Above approximately 35 percent power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10 percent power and 35 percent power (Permissive 7 and Permissive 8), low flow in any two loops will actuate a reactor trip.

Normal power for the reactor coolant pumps is supplied through start buses from common station service transformers connected to the 161-kV external power lines. Each reactor coolant pump is attached to a separate unit board bus and its supplied power is not interrupted for a turbine or generator trip. These pumps will continuously supply coolant flow to the core.

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

Although the original analysis was performed under the assumptions of a Condition III event, for the Cycle 9 reload, this event is analyzed with Condition II acceptance criteria.

15.3.4.2 Analysis of Effects and Consequences

Method of Analysis

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

The transient is analyzed using two digital computer codes. First, the RELAP5/MOD2-B&W code (Reference 11) 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 system and core power responses from RELAP5/MOD2-B&W are used to determine the DNBR using the LYNXT code (Reference 12) and a statistical core design methodology (Reference 11). The DNBR transients presented represent the thermal-hydraulic conditions of the average channel. The DNBR curve shown is the minimum value at the hot spot of the hot channel.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2, except that following the loss of supply to all pumps at power, a reactor trip is actuated by either bus undervoltage or bus underfrequency. To maximize the power response during the event, the lease negative Doppler power coefficient and +7.0 pcm/F moderator coefficient are assumed.

The calculated sequence of events is shown in Table 15.3.4-1 for the case analyzed. Figures 15.3.4-1 and 15.3.4-2 show the core flows, thermal and neutron power and DNB ratios as a function of time for the case. The reactor is assumed to trip on the undervoltage signal. The DNBR was greater than the safety analysis limit for the duration of the event.

15.3.4.3 Conclusions

The analysis demonstrates that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit during the transient and thus there is no clad damage or release of fission products to the Reactor Coolant System.

Analysis has shown that for frequency decay rates less than 6.8 Hz/second, no reactor coolant pump trip is necessary. A grid analysis was provided for the Sequoyah Nuclear Plant which determined that, for the worst case, the grid decay rate is less than 5.0 Hz/second.

The Complete Loss of Forced Reactor Coolant Flow event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.3.5 Waste Gas Decay Tank Rupture

15.3.5.1 Identification of Causes and Accident Description

The Gaseous Waste Processing System is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, hydrogen analyzers, waste gas decay tanks for service at power and other waste gas decay tanks for service at shutdown and startup.

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

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

15.3.5.2 Analysis of Effects and Consequences

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

15.3.6 Single Rod Cluster Control Assembly Withdrawal at Full Power

15.3.6.1 Identification of Causes and Accident Description

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single rod cluster control assembly from the inserted bank at full power operation.

The operator could deliberately withdraw a single rod cluster control assembly in the control bank. This feature is necessary in order to retrieve an assembly should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures which could result in single rod cluster control assembly withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single rod cluster control assembly by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications.

Each bank of rod cluster control assemblies in the system is divided into two groups of 4 mechanisms each, except control bank A which is comprised of two groups of two mechanisms each and group 2 of control bank D which consists of 5 mechanisms. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation and deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the rod cluster control assembly attached to the mechanism. Since the four stationary gripper, movable gripper, and lift coils associated with the four rod cluster control assemblies of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group, or 4 rod cluster control assemblies except for control banks A (groups 1 and 2) and D (group 2). Mechanical failures either are in the direction of insertion, or immobility.

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

15.3.6.2 Analysis of Effects and Consequences

Method of Analysis

Power distributions within the core are calculated by the TURTLE (Reference 5) code based on macroscopic cross section generated by LEOPARD (Reference 6). The peaking factors calculated by TURTLE are then used by THINC to calculate the minimum DNB for the event. The case analyzed was for the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power. FDH for this case was 1.71 including appropriate allowances for calculational uncertainties.

Results

Two cases have been considered as follows:

1. If the reactor is in the manual control mode, continuous withdrawal of a single rod cluster control assembly results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the failed rod cluster control assembly. In

terms of the overall system response, this case is similar to those presented in Subsection 15.2.2; however, the increased local power peaking in the area of the withdrawn rod cluster control assembly results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn rod cluster control assembly, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNB ratio from falling below the safety analysis limit. 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 safety analysis limit is 5 percent.

2. If the reactor is in automatic control mode, withdrawal of a single rod cluster control assembly will result in the immobility of the other rod cluster control assemblies in the controlling bank. The transient will then proceed in the same manner as Case 1 described above. For such cases as above a trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNB ratio in the core of less than the safety analysis limit.

15.3.6.3 Conclusions

For the case of one rod cluster control assembly fully withdrawn, with the reactor in the automatic or the 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 a DNBR of less than 1.3 is 5 percent of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before DNB could occur. For case 2 discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard.

The Single Rod Cluster Control Assembly Withdrawal at Full Power event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.3.7 Steam Line Break Coincident with Rod Withdrawal at Power (SLB c/w RWAP)

15.3.7.1 Identification of Causes and Accident Description

In September of 1979, IE-79-22 entitled "Qualification of Control Systems" was issued by the United States Nuclear Regulatory Commission (USNRC) identifying a potential unreviewed safety question resulting from Control and Protection Systems interactions. One of the postulated scenarios that was identified was the operation of the non-safety grade automatic rod control system following a steam line break inside or outside of containment. The automatic rod control system derives signals from the Nuclear Instrumentation System (specifically the excore power range neutron detectors) and the turbine impulse pressure, among other inputs, to determine if control rod motion is required. Since a steam line break may occur inside containment in the vicinity of the excore detectors (which are classified as Category "C" equipment per NUREG-0588, Revision 1, Appendix E), or outside containment in the vicinity of the turbine impulse pressure transmitters, the automatic rod control

system may be exposed to an adverse environment. This equipment is not qualified to preclude the steam line break from causing a control rod (bank) withdrawal due to an adverse environment. In addition to the potential rod withdrawal, the Power Range High Neutron Flux and OTΔT reactor protection trip functions may not be available as a result of the harsh environmental conditions which may exist.

15.3.7.2 Method of Analysis

This transient is simulated using the RELAP5/MOD2 and LYNXT codes (Reference 1 and 12) by modeling a steam line break in coincidence with the withdrawal of control bank D at hot full power (HFP) conditions. A spectrum of steam line break sizes were analyzed to determine the limiting condition. The reactivity assumption associated with the rod withdrawal was 15 pcm per second, which is based on the maximum speed of the rod speed controller (45 inches per minute) and the maximum differential rod worth of control bank D at HFP conditions (20 pcm per inch). Reactivity assumptions are verified each fuel cycle as part of the reload safety evaluation.

In the RELAP5/MOD2 system analysis model, the Doppler Power Coefficient was allowed to vary with power and the Moderator Temperature Coefficient was allowed to vary with core average temperature. Conservative values of Doppler Coefficient vs. Power and Moderator Coefficient vs. Temperature were used.

The following reactor trip functions may actuate during this postulated transient:

- Overpower ΔT (OPΔT): typically actuated for the small-to-intermediate breaks
- Low Steamline Pressure - Safety Injection (LSP-SI): typically actuated for the large breaks

15.3.7.3 Results

With respect to the minimum Departure from Nucleate Boiling Ratio (DNBR), the limiting case was for ton OPΔT having two-out-of-four coincidence logic. Furthermore, the measured vessel ΔT lead/lag values associated with this trip function were 5 seconds and 3 seconds, respectively. The results demonstrate that the DNB design basis was met. The sequence of events for this case is presented in Table 15.3.7-1. Plots associated with this transient are presented in Figures 15.3.7-1 thru -3.

Since the RSG has a higher heat transfer area than the OSGs, this event was analyzed with the RSGs. The results of the steam line break coincident with rod withdrawal at power analysis performed by Framatome ANP show that the analysis results with OSGs bound the RSG response.

15.3.8 References

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2. BAW-10168, R03, "BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," December 1996.

3. W. A. Bezalla, C. L. Caso, A. C. Spencer, "LOCTRA-R2 Program Loss-of-Coolant Transient Analysis," WCAP 7835, January 1972.

4. S. Altomare and R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758, September 1971.

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

6. F. M. Bordelon, "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP 7969, September 1972.

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

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9. BAW-10162, "TACO3-Fuel Pin Thermal Analysis Code," October 1989.

10. M.Y. Young, et. Al., “Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection,” WCAP-8479 Rev. 2 (Proprietary), WCAP-8480 Rev. 2 (Non-Proprietary), January 1975.

11. BAW-10170, "Statistical Core Design for Mixing Vane Cores," December 1988.

12. BAW-10156, "LYNXT-Core Transient Thermal Hydraulic Program," August 1993.

13. Document No. 77-5016198, “Framatome ANP, Replacement Steam Generator Report for Sequoyah Unit 1.”

TABLE 15.3.1-1

SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

Event, seconds	2.0”	2.5”	2.75”	3.0”	3.25”	3.5”	4.0”	5”	Charging Line
Break Initiation	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Scram	49.5	32.1	26.7	22.7	19.7	16.9	13.9	10.0	110.1
RC Pump Coastdown	49.5	32.1	26.7	22.7	19.7	16.9	13.9	10.0	110.1
MSIV Closed	49.5	32.1	26.7	22.7	19.7	16.9	13.9	10.0	110.1
SI Signal	62.7	41.6	34.8	29.5	25.1	21.2	17.6	10.6	133.2
MFW Isolation	72.7	51.6	44.8	39.5	35.1	31.2	27.6	20.6	143.2
Pumped ECCS Injection	99.7	78.6	71.8	66.5	62.1	58.2	54.6	47.6	170.2
Loop Seal Clearing	2994	1259	984	790	637	512	388	250	NA
Top of Core Uncovers	2620	2145	1760	1495	1430	1150	890	415	NA
Peak Clad Temperature	2987	2763	2310	1933	1718	1384	NA	517	NA
Accumulator Injection	NA	NA	2500	1975	1700	1360	950	485	NA

TABLE 15.3.1-2

SMALL BREAK LOCA RESULTS

Results	2.75 inch	3.0 inch
Peak Cladding Temperature, °F	11621403	8281400
Peak Temperature Location, ft	10.911.64	11.64
Rupture Time, seconds	NA	NA
Rupture Location, ft	NA	NA
*Maximum Local Metal / Water Reaction, %	0.12130	0.10730
Total Metal / Water M/W Reaction, %	<0.0016	<0.06~0

* M/W = metal/water reaction

TABLE 15.3.4-1

TIME SEQUENCE OF EVENTS FOR COMPLETE LOSS OF FLOW EVENT

Accident	Event	Time (sec.)
Complete loss of Forced Reactor Coolant Flow		
All loops operating, all pumps coasting down		
	Coastdown begins	0
	Rod motion begins	1.5
	Minimum DNBR occurs	2.8

TABLE 15.3.7-1

TIME SEQUENCE OF EVENTS FOR THE

STEAM LINE BREAK COINCIDENT WITH

ROD WITHDRAWAL AT POWER ANALYSIS

EVENT	TIME (sec.)
Steam line breaks / Rod withdrawal occurs	0.0
Overpower ΔT reactor trip setpoint reached	9.7
Rods begin to fall	17.7
Peak Core Heat Flux occurs	18.3
Minimum RELAP5 DNBR occurs	18.4

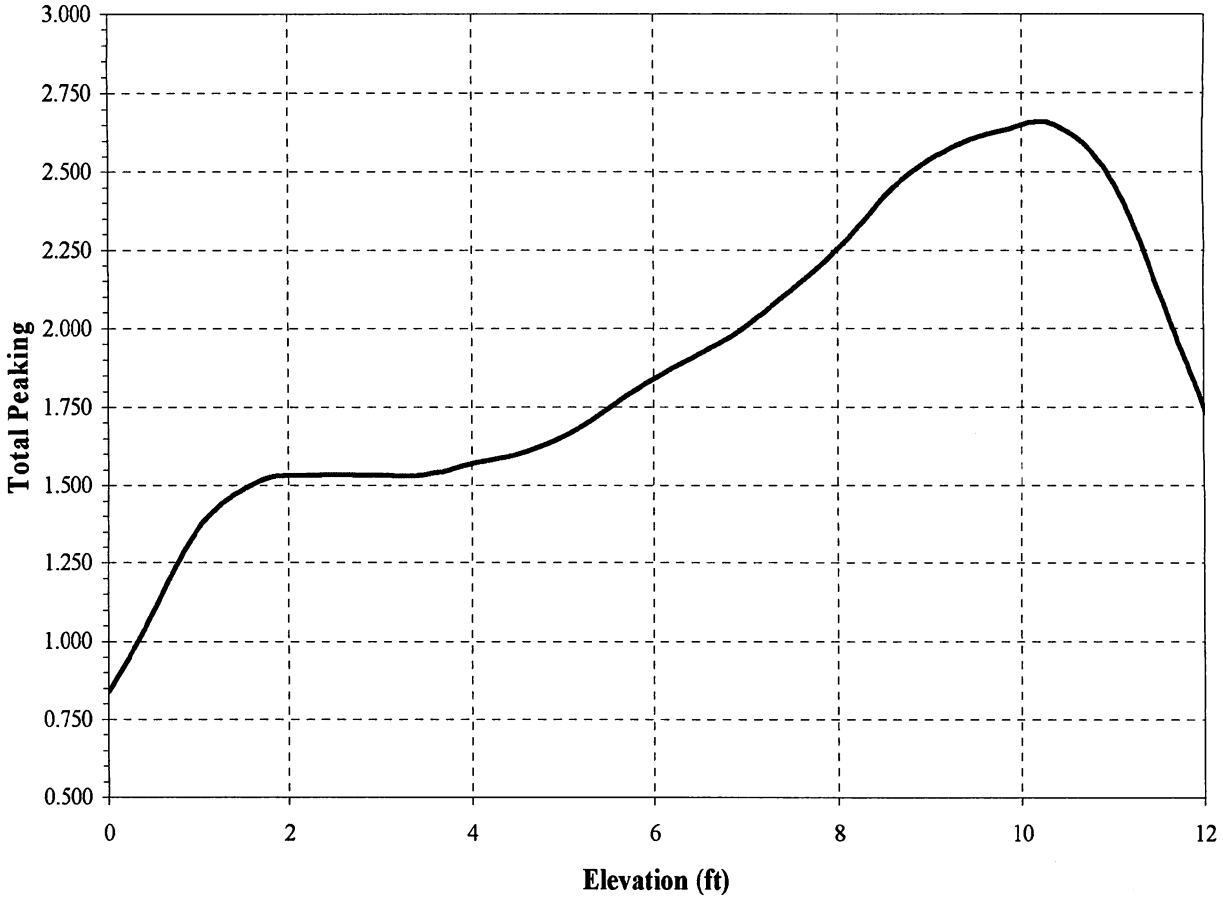


Figure 15.3.1-1

Small Break LOCA Study
Total Peaking Profile

Revised by Amendment 21

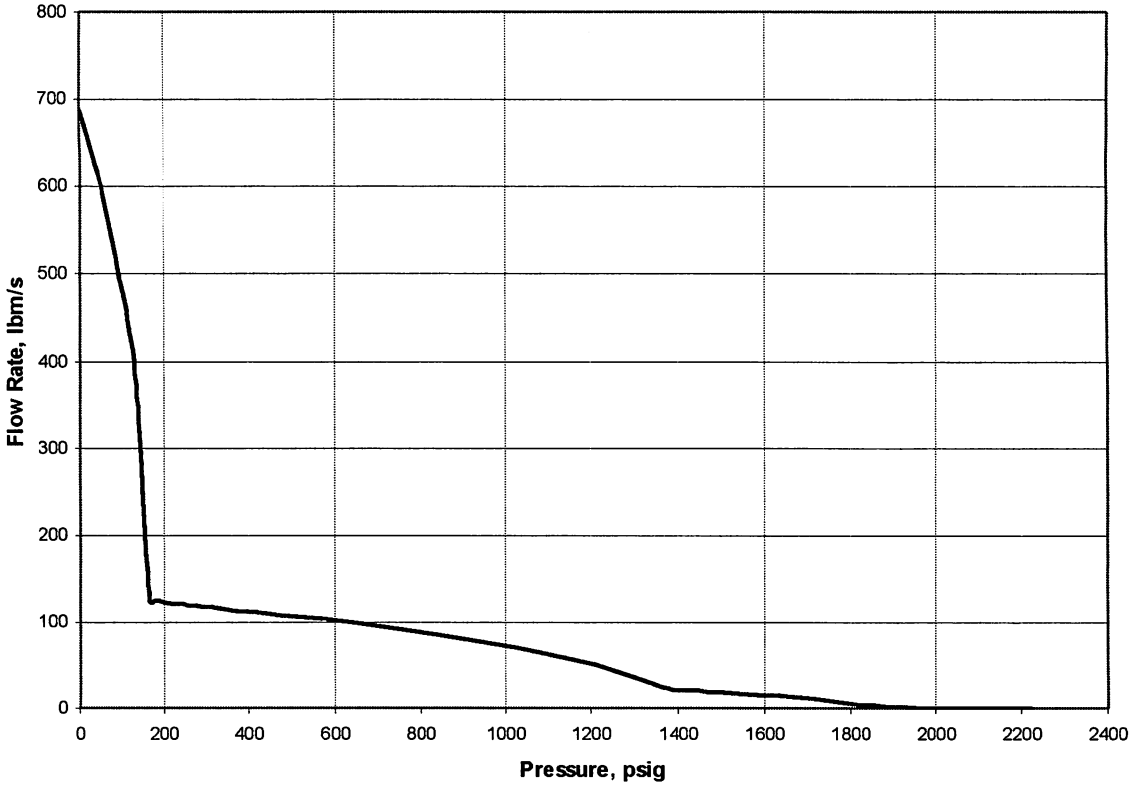


Figure 15.3.1-2

Small Break LOCA Study
Pumped ECCS Injection Flow vs. RCS Pressure

Revised by Amendment 21

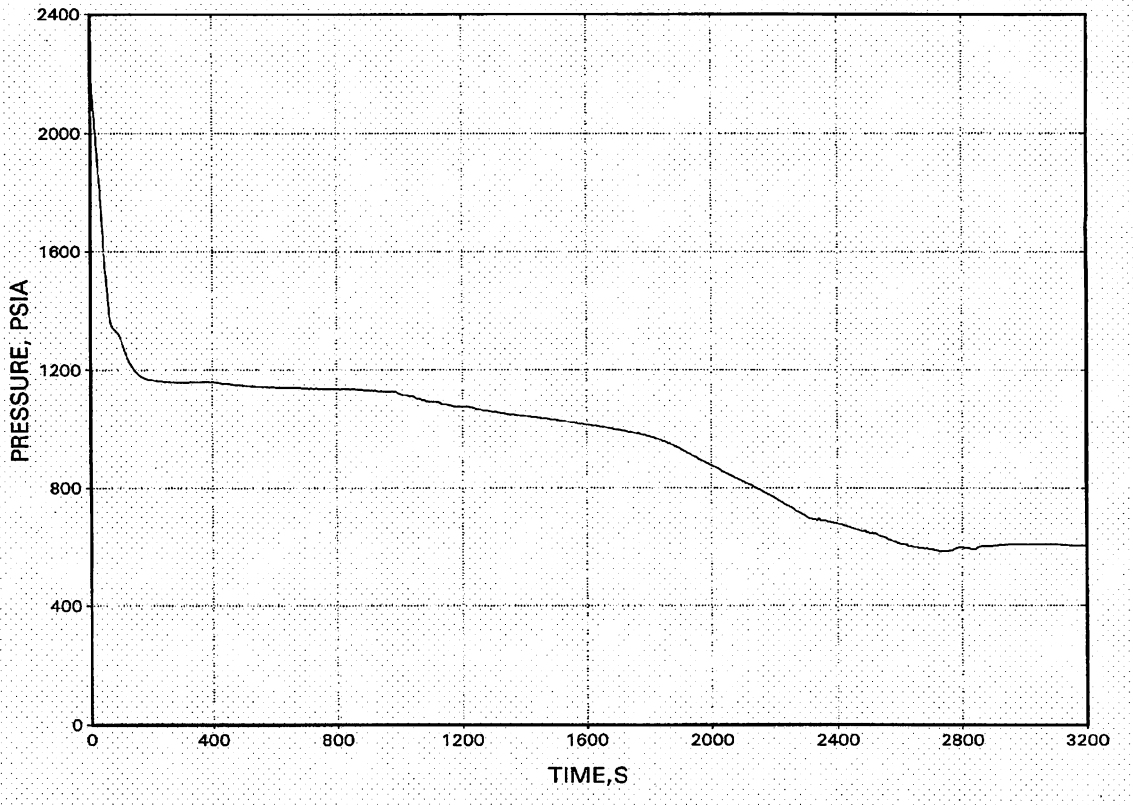


Figure 15.3.1-3

2.75-Inch Pump Discharge Break
Primary System Pressure

Revised by Amendment 21

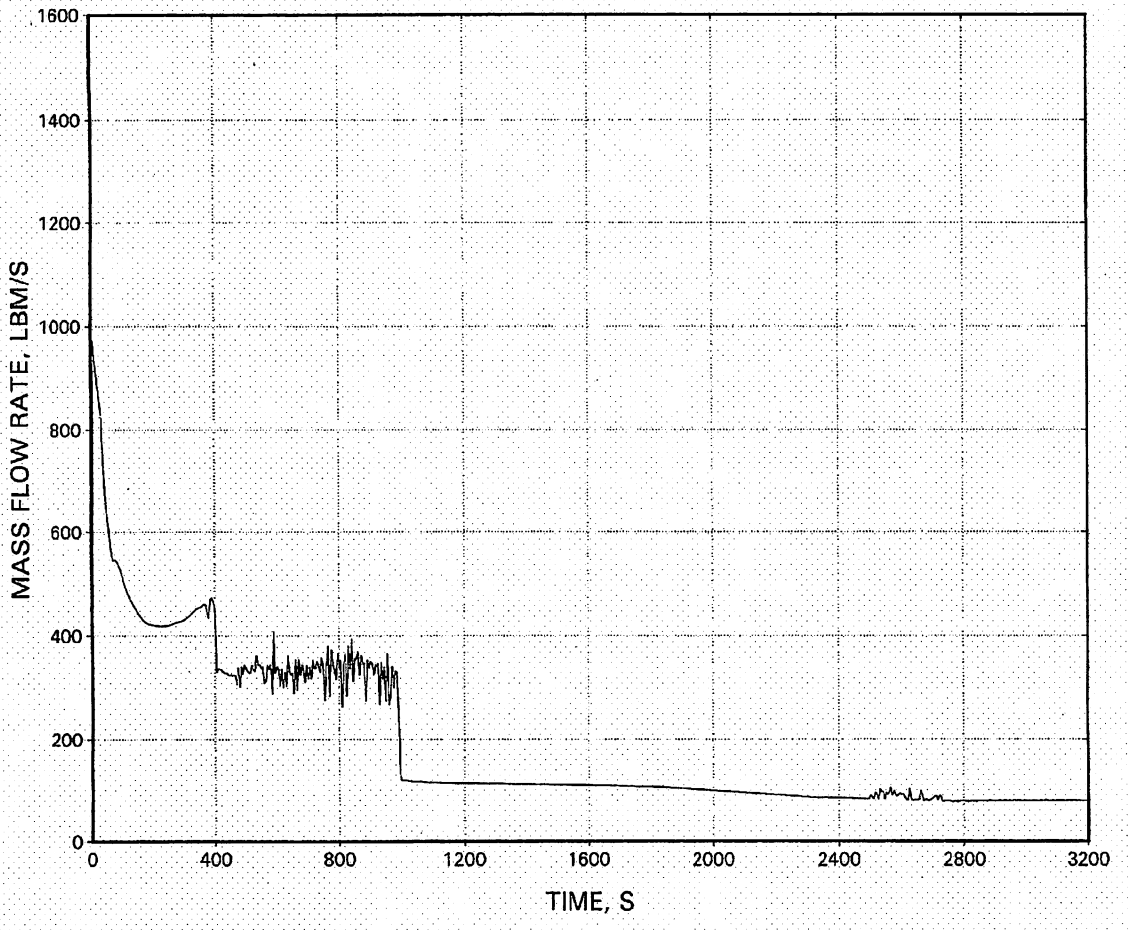


Figure 15.3.1-4

2.75-Inch Pump Discharge Break Leak Flow Rate

Revised by Amendment 21

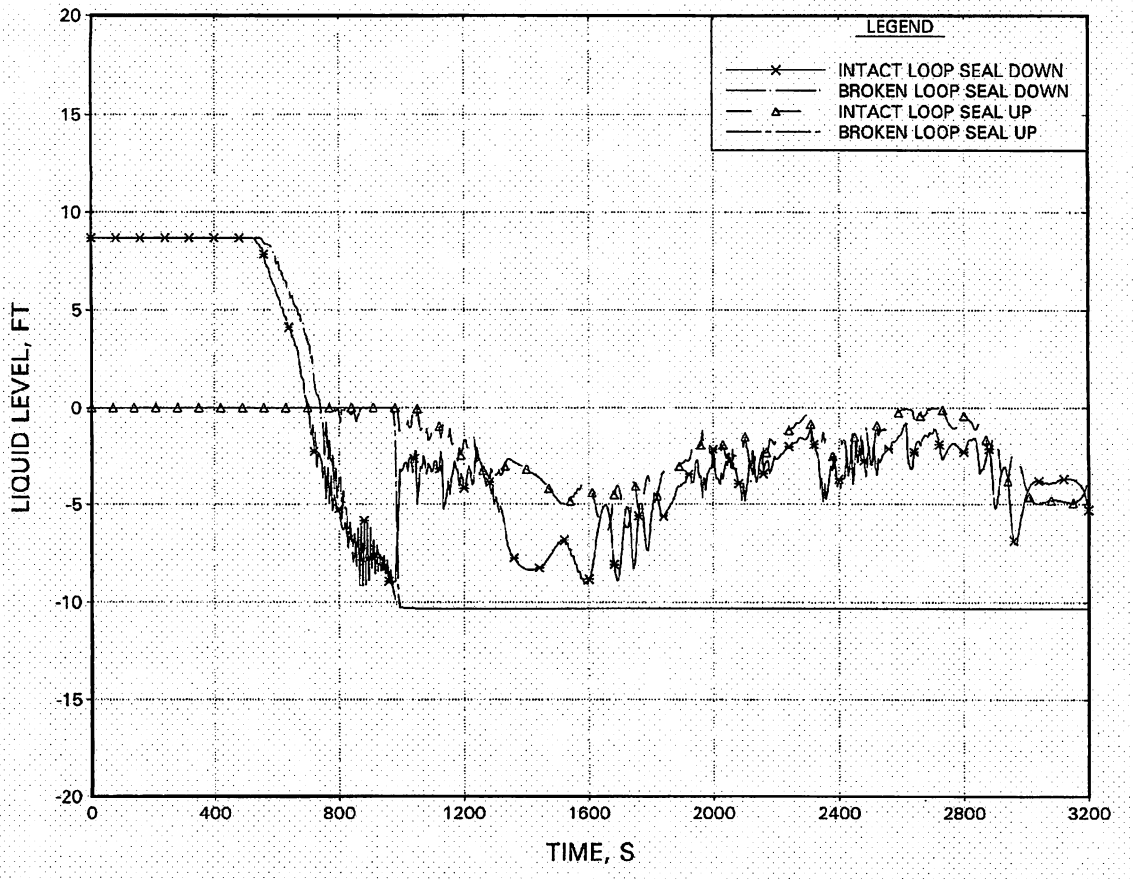


Figure 15.3.1-5

2.75-Inch Pump Discharge Break
Pump Suction Loop Seal Levels

Revised by Amendment 21

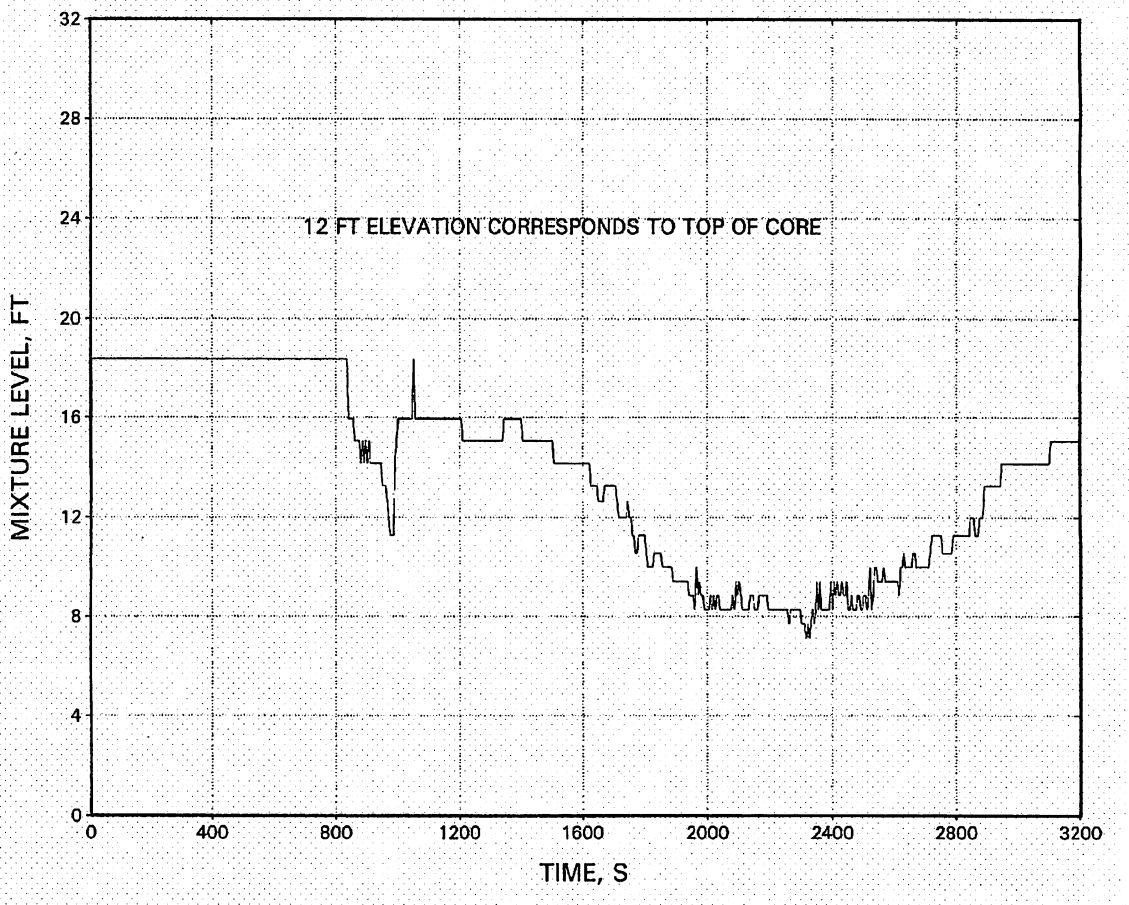


Figure 15.3.1-6

2.75-Inch Pump Discharge Break
Mixture Level Above Bottom of Core

Revised by Amendment 21

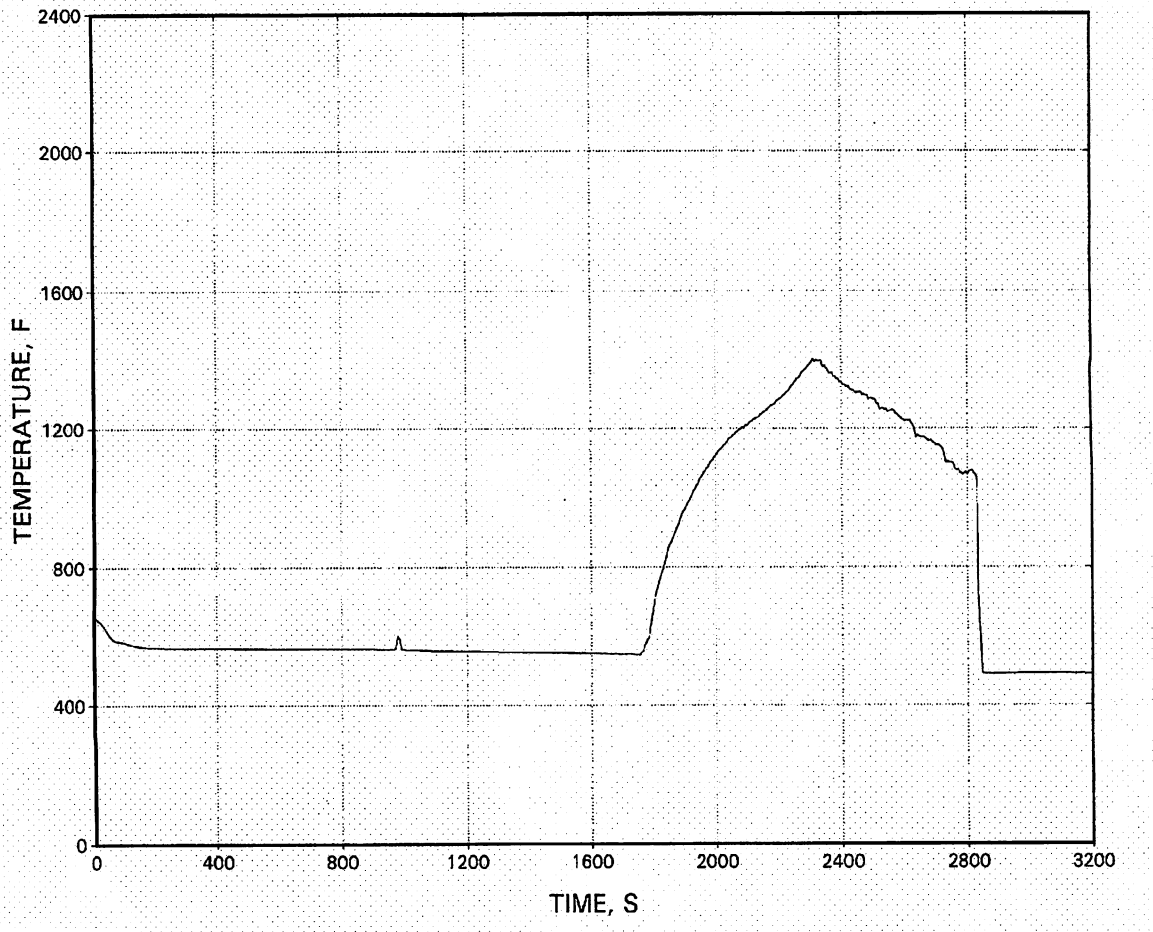


Figure 15.3.1-7

2.75-Inch Pump Discharge Break
Hot Pin Clad Temperature

Revised by Amendment 21

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

CASE A

Figure 15.3.3-1 Interchange Between Region 1 and Region 3 Assembly

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					0.3			1.5							
2			3.2				0.8			3.2		6.0			
3			1.2		0.0			1.6				-		10.3	
4						0.0			2.9			6.5			
5	-2.2				-1.0			2.2			6.9				6.6
6					-1.7		0.5					8.8			
7		-3.2							5.2		16.7				5.4
8	-3.5		-3.4		-2.6		-0.7			11.4		11.3	5.8	4.4	
9				-3.6			-2.0			-2.3			2.2		
10	-3.8		-3.8			-3.6		-2.9						0.5	
11					-3.9				-4.3		-4.6		-1.5		
12		-2.8	-3.1					-4.5							
13								-4.8		-4.4		-2.6		1.4	
14			-0.4			-4.8		-4.8							
15							-4.8			-4.5					

CASE B-1

Figure 15.3.3-2 Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Retained by the Region 2 Assembly

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

CASE B-2

Figure 15.3.3-3 Interchange Between Region 1 and Region 2 Assembly, Burnable Poison Rods Being Transfered to the Region 1 Assembly

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

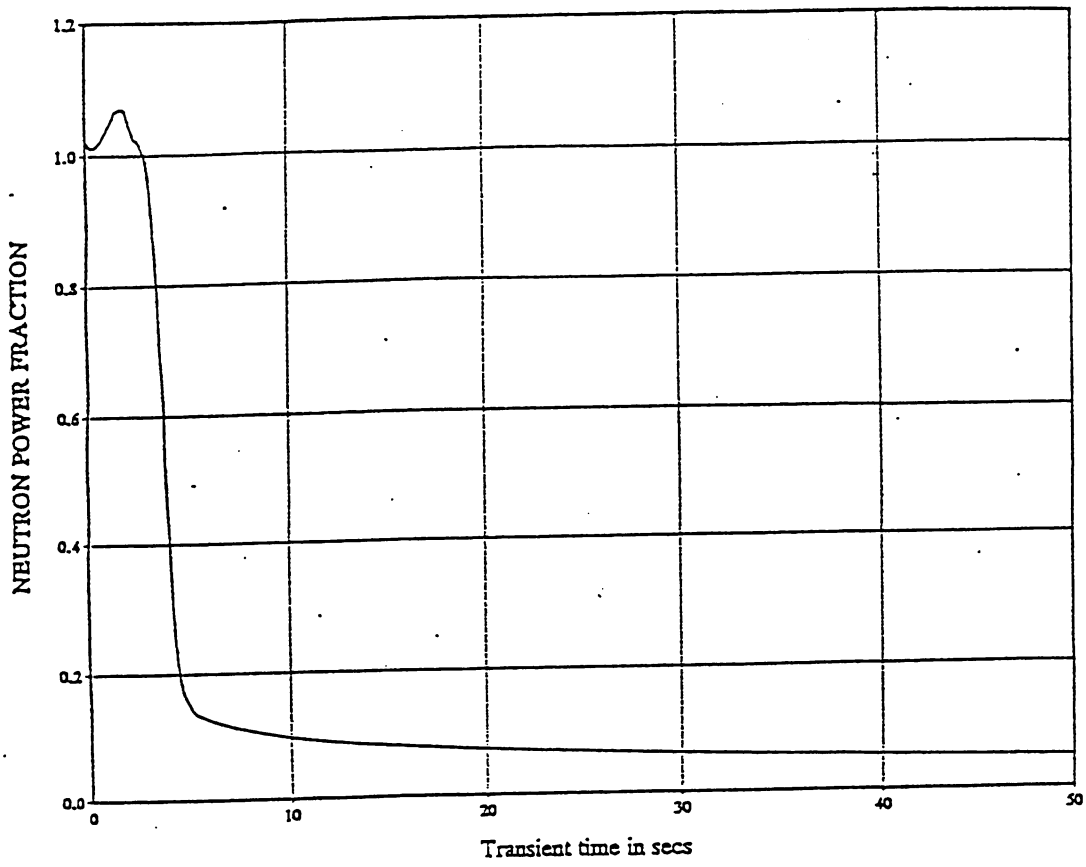
CASE C

Figure 15.3.3-4 Enrichment Error: A Region 2 Assembly Loaded into the Core Central Position

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

CASE D

Figure 15.3.3-5 Loading a Region 2 Assembly into a Region 1 Position Near Core Periphery



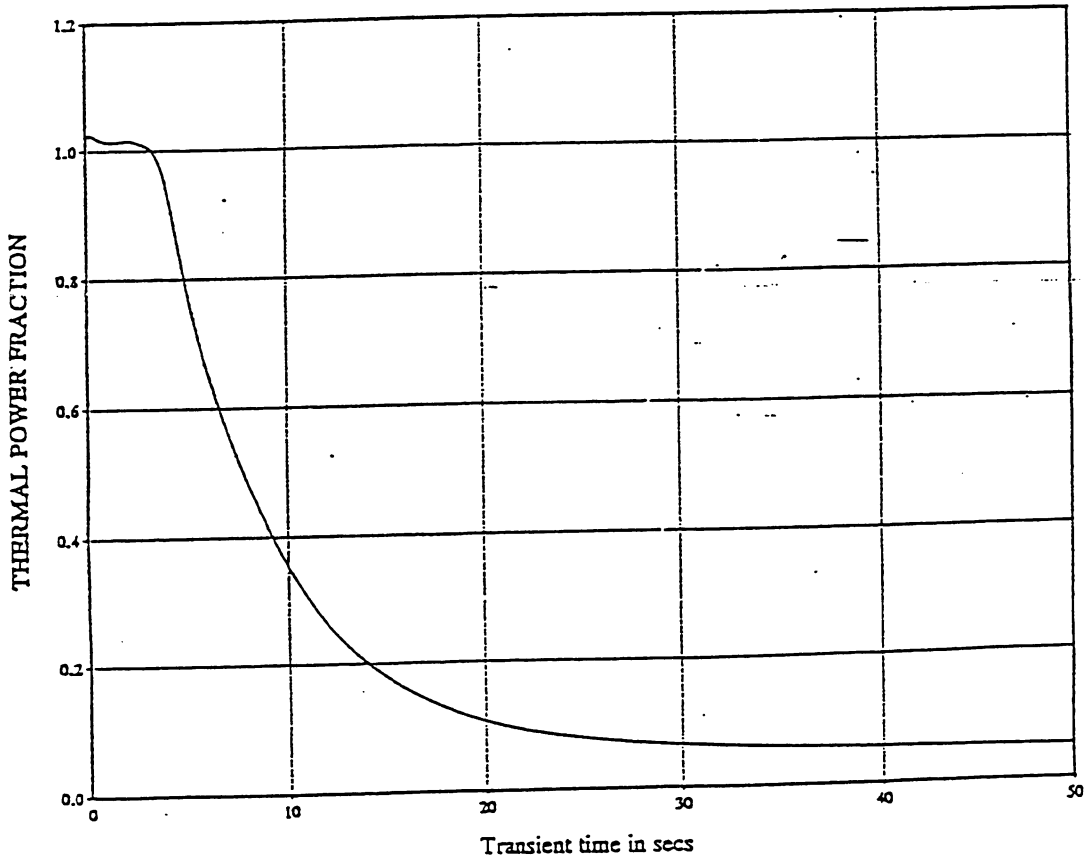
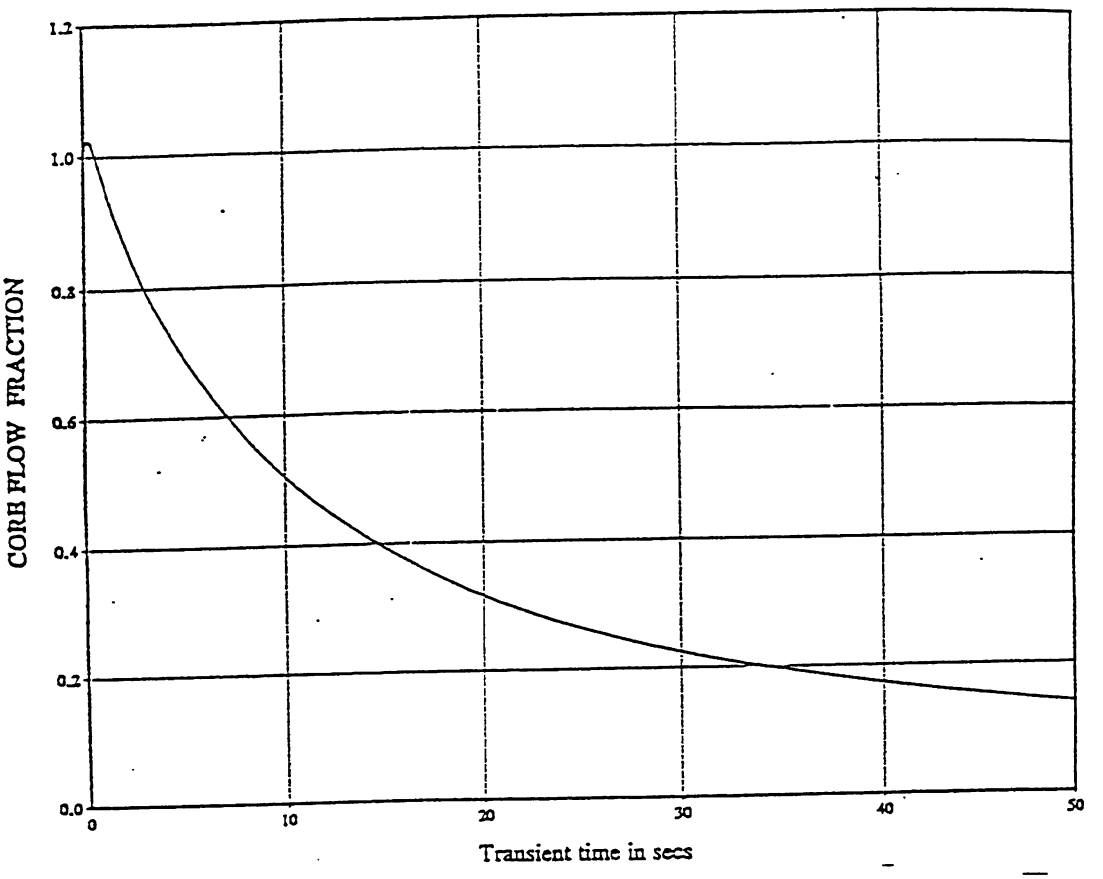


Figure 15.3.4-1
Complete Loss of Forced Reactor Coolant Flow
Neutron Power and Thermal Power
vs Time



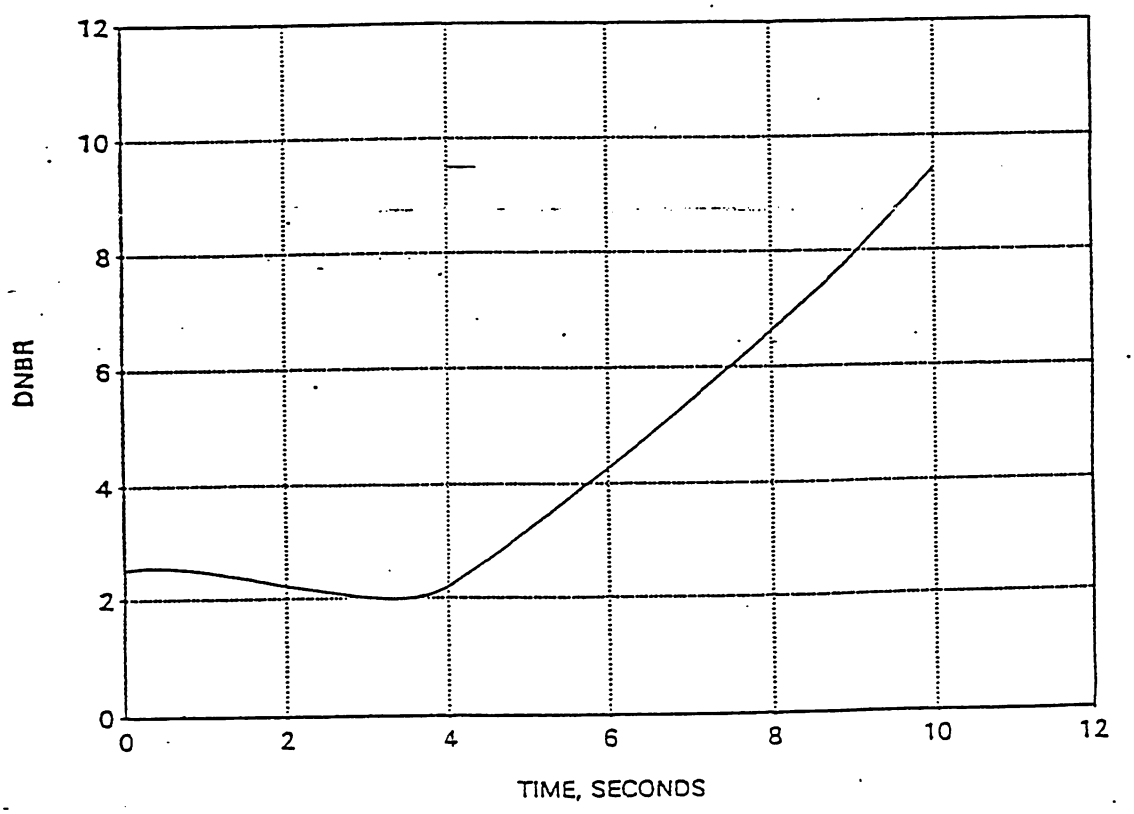
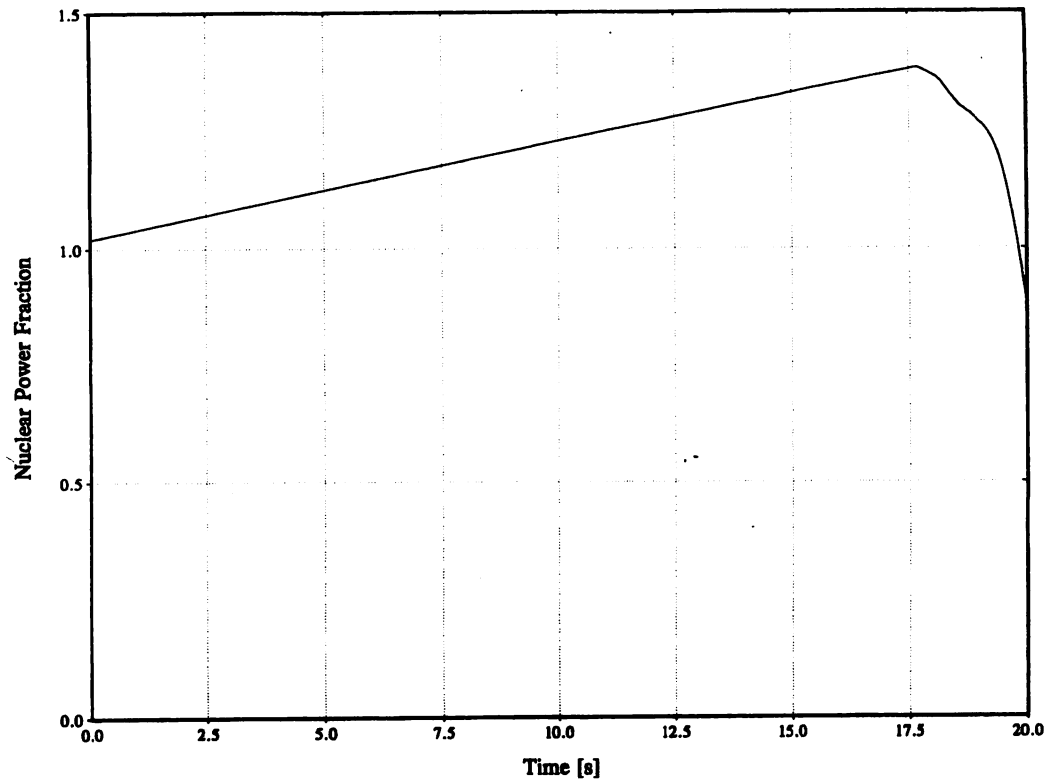
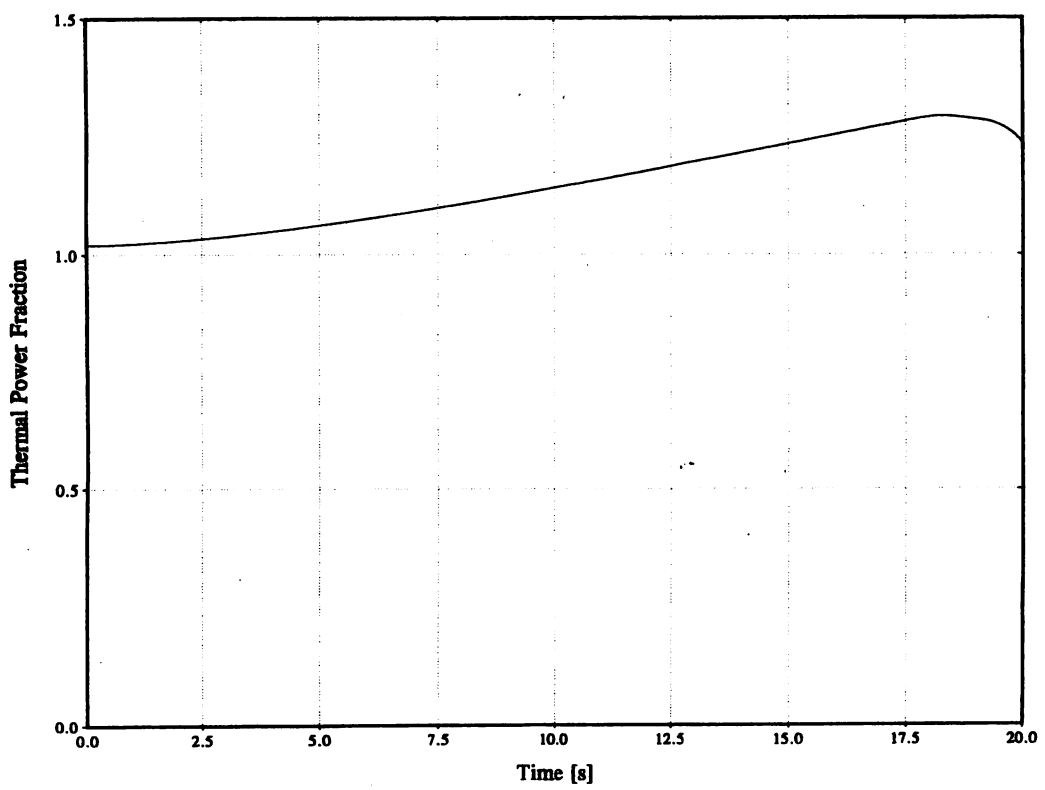


Figure 15.3.4-2
Complete Loss of Forced Reactor Coolant Flow
Core Flow Fraction and DNBR
vs Time

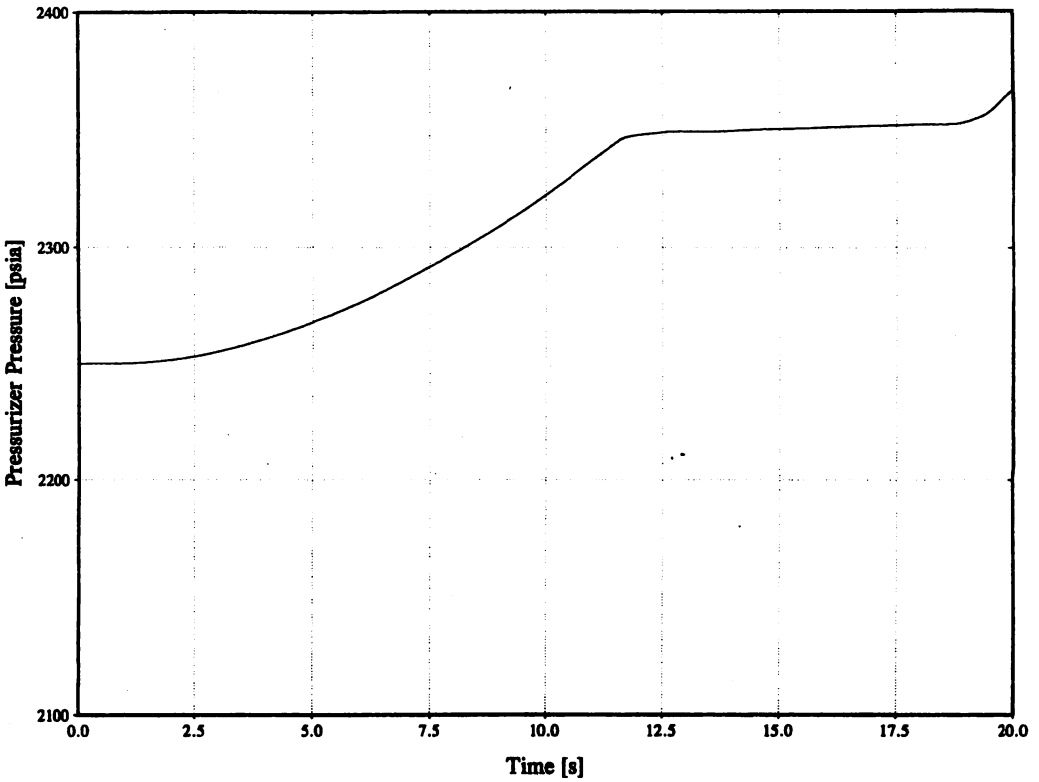


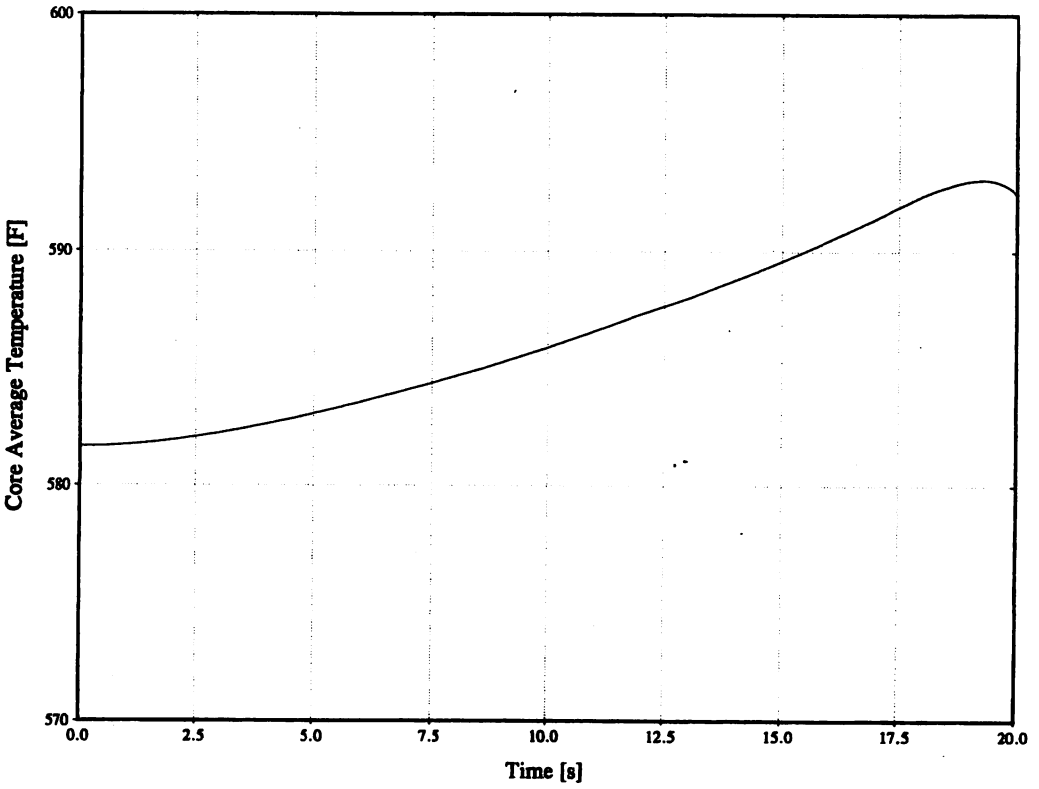


Revised by Amendment 21

Figure 15.3.7-1

Sequoyah Steam Line Break Coincident with Rod Withdrawal
Neutron Power, Thermal Power vs Time

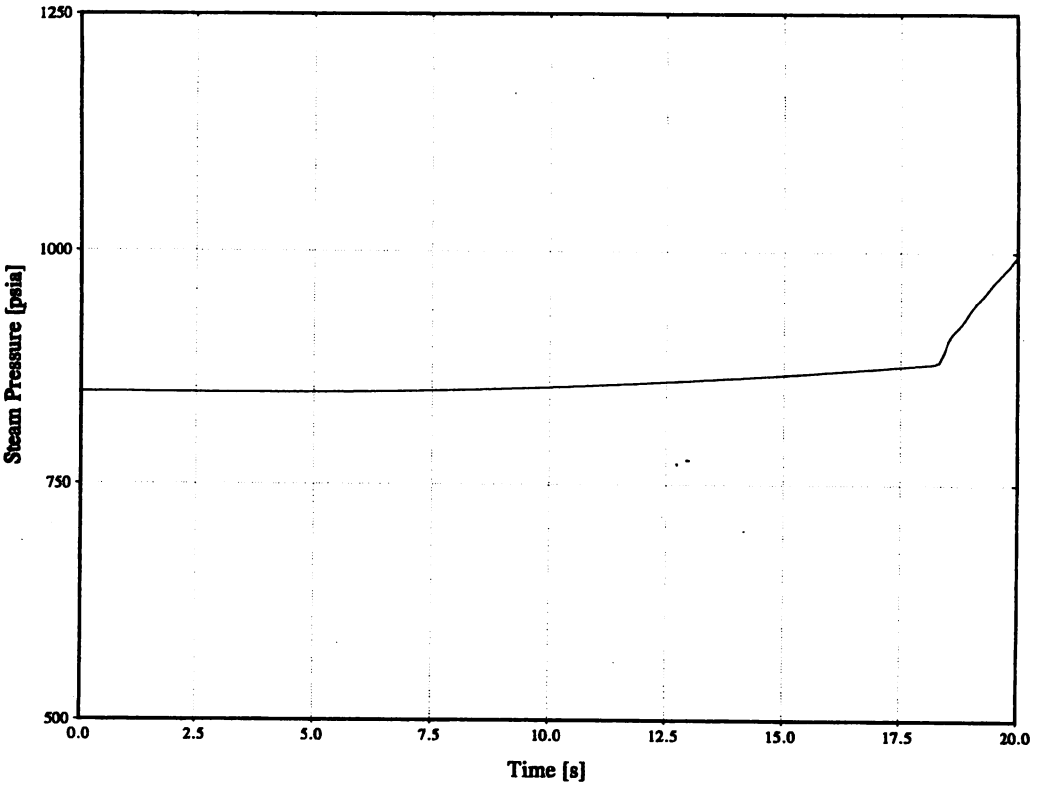


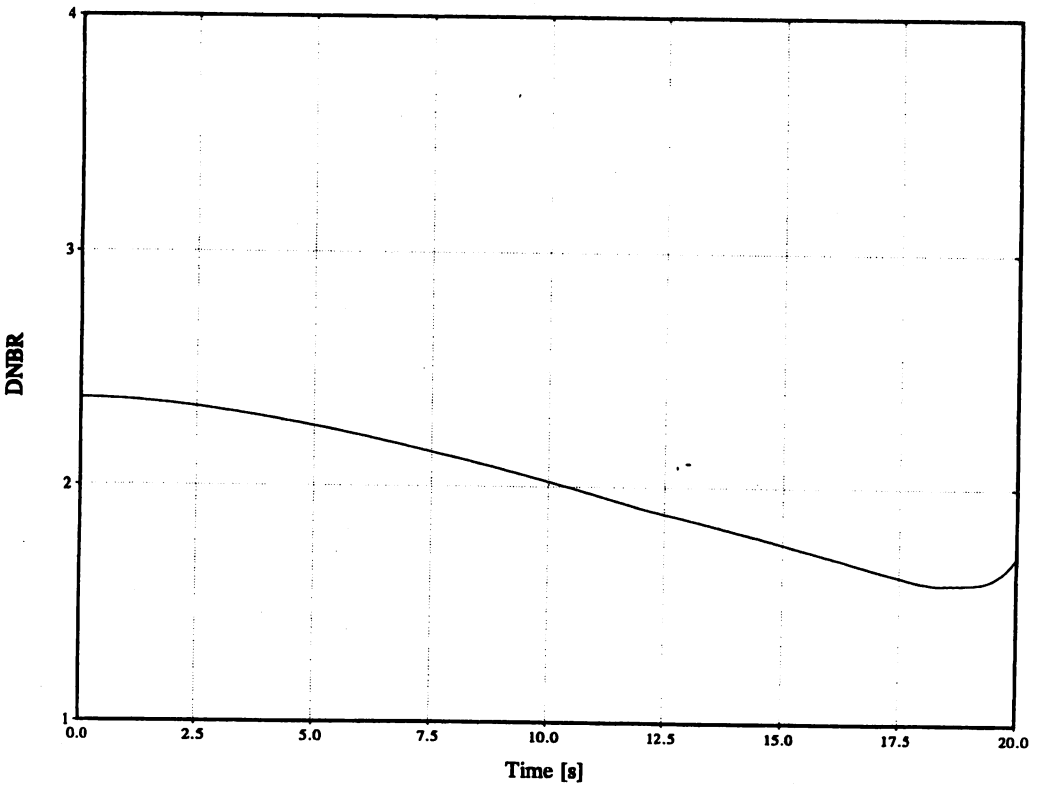


Revised by Amendment 21

Figure 15.3.7-2

Sequoyah Steam Line Break Coincident with Rod Withdrawal
Pressurizer Pressure, Core Average Temperature vs Time





Revised by Amendment 21

Figure 15.3.7-3

Sequoyah Steam Line Break Coincident with Rod Withdrawal
Steam Line Pressure, DNBR vs Time

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic which must be designed against and thus represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

1. Major rupture of pipes containing reactor coolant up to and including double ended rupture of the largest pipe in the Reactor Coolant System (loss of coolant accident).

2. Major secondary system pipe ruptures.

3. Steam generator tube rupture.

4. Single reactor coolant pump locked rotor.

5. Fuel handling accident.

6. Rupture of a control rod mechanism housing (rod cluster control assembly ejection).

The analysis of thyroid and whole body doses, resulting from events leading to fission product release, appears later in the Safety Analysis Report. The fission product inventories which form a basis for these calculations are presented in Chapter 11 and Section 15.1. The Safety Analysis Report also includes the discussion of systems interdependency contributing to limiting fission product leakages from the containment following a Condition IV occurrence.

15.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

The analysis specified by 10CFR50.46 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors", is presented in this section. The results of the loss of coolant accident analysis are shown in Table 15.4.1-15 and show compliance with the acceptance criteria.

A Large Break LOCA is initiated by a postulated rupture of the Reactor Coolant System (RCS) primary piping. Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. The break initiates a rapid depressurization of the RCS. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, reactor trip is conservatively neglected in the analysis. The reactor is shut down by coolant voiding in the core.

The plant is assumed to be operating normally at full power prior to the accident. The cold leg break is assumed to open instantaneously. For this break, a rapid depressurization occurs, along with a

core flow stagnation and reversal. This causes the fuel rods to experience departure from nucleate boiling (DNB). Subsequently, the limiting fuel rods are cooled by film convection to steam. The coolant voiding creates a strong negative reactivity effect and core fission ends. As heat transfer from the rods is reduced, the cladding temperature rises.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate, and leads to a period of positive core flow or reduced downflow as the reactor coolant pumps in the intact loops continue to supply water to the vessel. Cladding temperatures may be reduced and some portions of the core may rewet during this period. The positive core flow or reduced downflow period ends as two-phase conditions occur in the reactor coolant pumps, reducing their effectiveness. Once again, the core flow reverses as most of the vessel mass flows out through the broken cold leg.

Mitigation of the Large Break LOCA begins when the Safety Injection System signal is actuated. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst single-failure be considered. This single-failure has been determined to be the loss of one ECCS pumped injection train. The Realistic Large Break LOCA methodology conservatively assumes an on-time start and normal lineups of the containment spray to conservatively reduce containment pressure and increase break flow. Hence, the analysis assumes that one charging pump, one Safety Injection pump, one RHR pump and two containment spray pumps are operating.

When the RCS pressure falls below the accumulator pressure, fluid from the accumulators is injected into the cold legs. In the early delivery of accumulator water, high pressure and high break flow will drive some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core; thus, core heat transfer improves and cladding temperatures decrease. Eventually, the relatively large volume of accumulator water is exhausted and core recovery must rely on pumped ECCS coolant delivery alone. As the accumulators empty, the nitrogen gas used to pressurize the accumulators exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas has been expelled, the ECCS temporarily may not be able to sustain full core cooling because of the core decay heat and the higher steam temperatures created by quenching in the lower portions of the core. Peak fuel rod cladding temperatures may increase for a short period until more energy is removed from the core by the charging, Safety Injection and RHR while the decay heat continues to fall. Steam generated from fuel rod rewet will entrain liquid and pass through the core, vessel upper plenum, the hot legs, the steam generator, and the reactor coolant pump before it is vented out the break. Some steam may flow to the upper head and pass through the spray nozzles which would provide a vent path to the break. The resistance of this flow path to the steam flow is balanced by the driving force of water filling the downcomer. This resistance may act to retard the progression of the core reflood and postpone core wide cooling. Eventually (within a few minutes of the accident), the core reflood will progress sufficiently to ensure core wide cooling. Full core quench occurs within a few minutes after core wide cooling. Long-term cooling is then sustained with the RHR system.

15.4.1.1 Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss of coolant accident including the double ended severance of the largest Reactor Coolant System Pipe. The reactor core and internals, together with the ECCS are designed so that the reactor can be shutdown safely and the essential heat transfer geometry of the core can be preserved following the accident.
The ECCS even when operating during the injection mode with the most severe single active failure, is designed to meet the Acceptance Criteria, Reference 1.

15.4.1.2 Method of Thermal Analysis

The RLBLOCA methodology is documented in Reference 67. The methodology follows the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 68). This method outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a LOCA analysis.

The RLBLOCA methodology consists of the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance.
- S-RELAP5 for the system thermal-hydraulic calculation.

The governing two-fluid (plus noncondensibles) model with conservation equations for mass, energy, and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heating. The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction, and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components accounts for physical dimensions and the dominant phenomenon expected during the LBLOCA event. The basic building blocks for modeling are the hydraulic volume for fluid paths and the heat structure for a heat transfer surface. In addition, special purpose components exist to represent specific components such as the pumps or the steam generator separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations. The S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and accumulator lines. The charging injection flows are connected to the RCS and the SI and RHR injection flows are connected to the accumulator lines, consistent with the plant layout. This model also describes the secondary-side steam generator that is instantaneously isolated (closed Main Steam Isolation Valves and feedwater trip) at the time of the break. A symmetric steam generator tube plugging level of 15% per steam generator was assumed. The break is modeled in the same loop as the pressurizer, as directed by the RLBLOCA methodology. The RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water (i.e. Recirculation) for CCS pumped injection need not be considered.

As described in the AREVA RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. The LBLOCA

phenomenological uncertainties are provided in Reference 67. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 15.4.1-13. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analysis. Two parameters, Refueling Water Storage Tank (RWST) temperature for pumped ECCS flows and diesel start time, are set at conservative bounding values for all calculations. Where applicable, the sampled parameter ranges are based on technical specification limits or supporting plant calculations that provide more bounding values.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops (specifically, the loop with the pressurizer). The evolution of the transient through blowdown, refill and reflood is computed continuously using S-RELAP5. Containment pressure is also calculated by S-RELAP5 using containment models derived from ICECON (Reference 69) which is based on the CONTEMPT-LT code (Reference 70) and has been updated for modeling ice condenser containments.

The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the Peak Cladding Temperature (PCT) at a high probability level.

15.4.1.3 Containment Analysis

The concurrent containment transient pressure calculation is performed by the ICECON module within the S-RELAP5 code. For the Realistic Large Break LOCA analysis, dominant containment parameters, as well as NSSS parameters, were established via a Phenomena Identification and Ranking Table (PIRT) process. Other model inputs are generally taken as nominal or conservatively biased. The PIRT outcome yielded two important (relative to Peak Cladding Temperature) containment parameters-containment pressure and temperature. In many instances, the conservative guidance of Containment Systems Branch Technical Position 6-1 (Reference 71) was used in setting the remainder of the containment model input parameters. As indicated in Table 15.4.1-13, containment temperature is a sampled parameter. Containment pressure response is indirectly ranged by sampling the upper containment volume. The minimum value is carried over from use in the long-term containment integrity analysis of record for Sequoyah. The maximum value is a simplified value computed as the volume available within the upper dome of the containment and within the crane wall above the control rod drive missile shield with no accounting for internal structures and the volumes of the refueling canal and the annular region separating the ice compartments neglected. This volume is maximized by neglecting the volume of internal structures. The lower compartment volume is biased low in order to promote flow through the ice baskets. In accordance with Reference 67, the condensing heat transfer coefficient is intended to be closer to a best-estimate instead of a bounding high value. In the ice compartment, the water formed by melted ice and condensed steam flows to the lower ice compartment sump where it accumulates, if the ice bay drains are not large enough to accommodate the rate of water production. When the water level in the lower ice compartment sump rises above the bottom of the lower doors, water spillage through the lower doors occurs in addition to flow through the drain ports. The water drainage (spillage plus drainage) from the ice compartment falls through the lower compartment vapor. This condenses steam and reduces the containment pressure. The ice compartment drainage flow is treated as a 100 percent efficient spray during the post-blowdown period of the transient.

The initial conditions and boundary conditions are given in Table 15.4.1-16. The building spray is modeled at maximum heat removal capacity. While there is an option within the computer code model to deliver spray to the lower compartment, this option is not applicable to Sequoyah. All spray

flow is delivered to the upper compartment. Because the start time for the recirculation fan is 600 seconds, forced flow from the upper compartment to the lower compartment is not likely to occur during the time period analyzed. The flow of steam or air, from the lower compartment to the upper compartment, backwards through the back draft dampers, is not modeled (no reverse direction flow). This approach is conservative in that no bypass of the ice beds (from lower to upper compartments) is allowed, and all flow from the lower compartment is directed through the ice beds. The passive flow of air and steam, from the upper compartment to the lower compartment, is modeled however. This is a passive flow, which is only a function of the excess pressure of the upper compartment compared to the lower compartment, the flow area of the recirculation fan back draft dampers, and the loss coefficient of the dampers. The back draft dampers are designed such that reverse flow from the lower to the upper compartment is prevented. However, when the upper compartment pressure is at least 0.5 psi greater than the lower compartment, the dampers open and allow flow from the upper compartment to the lower compartment. Flow in this manner, from the upper to lower compartment, is modeled without this minimum pressure difference, i.e. any excess pressure is modeled as resulting in flow.

Passive heat sink parameters are listed in Table 15.4.1-17. Surface coatings, where they existed, were incorporated as an equivalent thickness of base material in order to eliminate any insulating effects on the exposed surfaces of the heat structures. Because the original basis for the size of each heat sink was biased low (for a different application), the values listed in Table 15.4.1-17 reflect a 10 percent increase in heat transfer surface area as compensation. Passive heat sinks were added to the lower containment to represent the advanced design containment sump strainers recently installed over the sump intake (17 ft^3 of steel). Additionally, all heat structure exposed surfaces remain available for condensing steam, even when they may become covered by ice melt or condensate.

The mass and energy release rates used for the containment backpressure calculation as a function of time during blowdown are given in Figures 15.4.1-41 and 15.4.1-42 for the limiting case.

15.4.1.4 Results of Large Break Spectrum

Two case sets of 59 transient calculations were performed, one with Loss of Offsite Power and another with offsite power continuing to be available. For each transient calculation, Peak Cladding Temperature (PCT) was calculated for a $U0_2$ rod and for Gadolinia bearing rods with concentrations of 2, 4, 6 and 8 w/o Gd_20_3. The limiting case set, that contained the PCT, was the set with offsite power available. The limiting PCT (2002°F) occurred in Case 20 for a $U0_2$ rod. As the results of random sampling within defined limits, a few of the characteristics defining the limiting case include a fuel assembly burnup of 21 GWd/MTU, a top skewed axial power shape, and a split break configuration with an area of 2.726 ft^2 per side (relative to an intact cold leg cross sectional flow area of 4.125 ft^2).

The time sequence of accident milestones for the limiting transient is characterized in Table 15.4.1-14. Table 15.4.1-15 lists the results of the limiting case. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1 percent limit. A nominal best estimate PCT case was identified as Case 14, which corresponded to the median case out of the 59-case set with offsite power available. The nominal PCT was 1514°F. This result can be used to quantify the relative conservatism in the limiting case result. In this analysis, it was 488°F.

Key parameters for the limiting PCT case are shown in Figures 15.4.1-37 through 15.4.1-50. Figure 15.4.1-46 is the plot of PCT independent of elevation; and this figure clearly indicates that the transient exhibits a sustained and stable quench.

15.4.1.5 Effect of Containment Purging

To assess the impact of purging on the calculated post-LOCA Sequoyah containment pressure, a calculation was first performed to obtain the amount of mass which exits through three available sets of purge lines during the initial portion of a postulated LOCA transient. Purge-line isolation closure time is assumed at 4.0 seconds after receipt of signal; during this interval, the full flow area is presumed to be available. In addition, the time to reach the SI signal setpoint and the delay necessary to generate the SI signal are conservatively assessed as 1.5 seconds total. Thus, flow through three pairs of fully open available purge lines was evaluated from 0.0 to 5.5 seconds for the postulated double-ended cold leg break.

The calculation employed the 50-node TMD computer code model which is described in Section 6.2.1.3.4. The 24-inch purge supply lines are connected to Volumes 34, 37, and 25; purge exhaust lines are connected to 36 and 25. Possible combinations of supply lines and exhaust lines open to the atmosphere were considered. Each of these purge lines is represented by a flow path of cross section area equal to 2.948 ft^2 and a total flow resistance factor equal to 3.98 (entrance and exit loss, three fully open butterfly valves and a debris screen). The most conservative two pairs of 24-inch purge and supply lines were assumed to be open in this calculation. In addition, two 12-inch lines connected to TMD node 29 were modeled as open.

In a computation for ECCS performance, the greatest impact on containment pressure occurs for the purge case of maximum air mass loss, which is based upon the two 12-inch lines being open and involves three open purge lines in the lower compartment (TMD elements 34, 36, and 37) and one purge line open in the upper compartment together with a cold leg break in TMD Volume 1. A total of 2620 pounds of air are calculated to be lost in this case. The maximum air loss case is the limiting case because any steam lost through purging in an ECCS backpressure elevation would otherwise be calculated to condense in the ice bed. Therefore, any steam lost through purging is ultimately of no consequence in the containment pressure determination, while any air loss directly reduces calculated pressure.

The impact of the reduced containment pressure on ECCS performance is accommodated in the calculated peak cladding temperature. While the containment backpressure calculation does not directly incorporate the loss of 2620 pounds of air at the beginning of the accident, the input data is sufficiently biased to produce an estimate of pressure that is lower than "best" estimate. Relative to the historical Westinghouse calculation of containment backpressure, the RLBLOCA analysis is based on a smaller value for the free volume of the Lower Compartment and a larger volume of the Upper Compartment. This produces the same result of blowing less air from the Lower Compartment into the Upper and also results in lower initial compression in the Upper Compartment. On this basis, the Realistic Large Break LOCA analysis permits purging of the Sequoyah containment during normal operation to be conducted through three sets of purge lines.

15.4.1.6 Additional Items Impacting Peak Clad Temperature

Variations in the PCT from the analysis of record as reported herein (due to model assessments, sensitivity analysis, and margin allocations) are reported to the NRC by means of 10CFR50.46.

15.4.1.7 Conclusions - Thermal Analysis

For cases considered, the ECCS will meet the Acceptance Criteria as presented in 10CFR50.46. That is:

1. The calculated peak fuel element clad temperature provides margin to the requirement of 2200°F, based on an Fq value of 2.65.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of zirconium alloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The clad oxidation limits of 17 percent are not exceeded during or after quenching.

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

15.4.2 Major Secondary System Pipe Rupture

15.4.2.1 Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

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

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

> Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safeguards there is no consequential damage to the primary system and the core remains in place and intact.

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

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

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

   a. Two-out-of-three low steam line pressure signals in any one loop
   b. Two-out-of-three low pressurizer pressure signals
   c. Two-out-of-three high containment pressure signals

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

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

4. Trip of the fast acting steam line stop valves on:

   a. Two-out-of-three low steam line pressure signals in any one loop.
   b. Two-out-of-four high-high containment pressure signals
   c. Two-out-of-three high steam line pressure rate signals in any one loop.

Fast-acting isolation valves are provided in each steam line that are assumed to fully close 8 seconds after a steam line isolation signal setpoint is reached. 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 blowdown even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

Steam flow is measured by monitoring dynamic head in flow restrictors located either in the steam generator nozzles (Unit 1) or in the main steam piping near the steam generators (Unit 2). The flow restrictors are of considerably smaller diameter than the main steam piping and the steam generator nozzles and thereby also serve to limit the maximum steam flow for any downstream break.

15.4.2.1.2 Analysis of Effects and Consequences

Method of Analysis

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

1. The core heat flux, RCS temperature, and pressure resulting from the cooldown following the steam line break. The RELAP5/MOD2-B&W code (Reference 56) is used to calculate system response and the NEMO code (Reference 59) is used to calculate limiting core power distributions.

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal-hydraulic digital-computer calculation (LYNXT code, Reference 60) has been used to determine whether DNB occurs for the core conditions computed in (1) above.

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 rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse conditions 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 variation of the coefficient with temperature and pressure has been included. The effect of power generation in the core on overall reactivity is shown in Figure 15.4.2-1.

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 calculations in RELAP5. Further, it was conservatively assumed that the core power distribution was uniform in RELAP5. These two conditions cause under-prediction 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 with NEMO for the limiting statepoints shown on Table 15.4.2-1. The NEMO 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 nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high heat flux regions of the core, the effect of void formation was also included. A 3-D statepoint analysis with NEMO confirmed that the reactivity employed in the kinetics analysis was larger than the reactivity calculation including the above local effects for the statepoints in Table 15.4.2-1. These results verify conservatism, i.e., under-prediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of boric acid (1950 ppm from the RWST) solution corresponding to the most restrictive single failure in the Safety Injection System. The Emergency Core Cooling System consist of three systems: 1) the passive accumulators, 2) the Residual Heat Removal System, and 3) the Safety Injection System. Only the Safety Injection System and the accumulators are modeled for the steam line break accident analysis.

   The injection curve used is shown in Figure 15.4.2-2. The flow corresponds to that delivered by one SI pump delivering its full flow to the cold leg header. Subsequent to this analysis, the minimum safety injection pump performance requirements were reduced from those shown in Figure 15.4.2-2. However, the flow reduction was more than off-set by the minimum flow available from one centrifugal charging pump. While not specifically modeled in the analysis, the flow from one charging pump is also credited in Reference 72 to establish that the results of the analysis remain conservative and bounding for the current safety injection pump minimum performance requirements in Figure 6.3.2-6. The analysis conservatively assumes that the safety injection lines downstream of the second check valve removed from the cold leg, has a boron concentration of 0 ppm. Modeling an unborated purge volume accounts for possible dilution via the diffusion of RCS coolant back into the lines plus potential leakage of the first set of check valves. Once the unborated water is purged, boron is assumed to be injected into the RCS at a concentration of 1950 ppm via a single intermediate head pump. This is a reasonable method of modeling postacident boron injection in that it both is conservative and reflective of the actual plant configuration. Subsequent to the original Framatome analysis, sensitivity studies have been performed related to the assumption of boron delivery to the RCS for the main steam line break event. The sensitivity studies examine the extent that leakage/diffusion of RCS coolant can occur in the safety injection lines before the results of the limiting case are adversely affected. The studies show that coolant with a boron concentration equivalent to that of the End of Life RCS (0 ppm) can exist in the safety injection system piping back as far as three check valves removed from the cold leg up to the discharge of the safety injection pump. A boron concentration of 1950 ppm is assumed from the RWST to the discharge of the SI pumps with 0 ppm from the pump discharge to the RCS. Based on these conditions, the limiting break (a complete severance of the steam line upstream of the pipe flow reducer with offsite power available), remains valid and limiting. The sensitivity studies performed prove that the original Framatome analysis remains bounding and the return to power associated with the other nonlimiting steam line break cases will also remain less than the

return to power associated with the limiting case assuming 0 ppm boron back to the discharge of the SI pump. Based on the system design and the Technical Specification required venting surveillance (with a 31 day frequency) of the SI pump and piping, it is technically acceptable to allow the assumption of a 1950 ppm boron concentration in the SI pump and suction piping for the main steam line break event.

For the cases with offsite power assumed, the following alignment sequence takes place prior to coolant injection via the Safety Injection System. After the generation of the safety injection signal (with appropriate delays for instrumentation, logic, and signal transport included), the makeup tank is isolated and the SI pump suction is aligned with the refueling water storage tank. In 28 seconds the pump is aligned and considered to be at full speed. The volume of unborated water in the pump discharge piping is swept into the RCS before the 1950 ppm water from the RWST reaches the core. The "purge" delay, described above, is explicitly modeled.

In the cases where offsite power is not available, an additional 30 second delay is assumed to allow for starting and loading the necessary safety injection equipment on to the diesel generators.

4. Four combinations of break sizes and initial plant conditions have been considered in determining the core power RCS transients:

   A. Complete severance of a pipe outside the containment (downstream of the steam flow measuring nozzle) with the plant initially at no load conditions, full reactor coolant flow with offsite power available.

   B. Complete severance of a pipe inside the containment at the outlet of the steam generator (upstream of the steam flow measuring nozzle) with the plant initially at no load conditions, full reactor coolant flow with offsite power available. Note: The design of the replacement steam generators on Unit 1 includes an integral flow limiter in the main steam nozzle, which eliminates the potential for a main steam line break upstream of the flow limiter.

   C. Case (A) above with loss of offsite power. Loss of offsite power results in coolant pump coastdown.

   D. Case (B) above with the loss of offsite power.

5. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined for end-of-life core conditions. 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 fluid conditions in the region of the stuck control assembly during the return to power phase following the steam line break. These fluid conditions 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 limiting statepoint conditions of the four steam line break accidents analyzed are given in Table 15.4.2-1. The limiting case is selected on the basis of hot channel factors, core power, core inlet temperature, and reactor coolant pressure. The core parameters evaluated for each of the cases correspond to values determined from the respective transient analysis.

The reactor is protected in at-power conditions by the normal overpower protection system. All the cases assume initial hot zero power conditions, however, since this represents the most pessimistic initial condition. The lack of stored energy in the core, high initial steam generator liquid inventory, and high steam pressure associated with this operational mode all contribute to a conservative over-cooling event.

6. In computing the steam flow during a steam line break, the Moody Curve (Reference 22) for fL/D = 0 is used.

7. Non-homogeneous flow in the steam generator and high moisture separation in the moisture separator is assumed. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water carryover would reduce the magnitude of the temperature decrease in the core and the pressure increase in the containment.

8. A conservatively high auxiliary feedwater flow rate (2350 gpm) is assumed to be delivered to the faulted steam generator. This auxiliary feedwater is not required to mitigate the transient and is modeled to increase the severity of the core cooldown.

Results

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.

Core Power and Reactor Coolant System Transient

Figures 15.4.2-3 through 15.4.2-10 show the RCS transient following a main steam line rupture (complete severance of a pipe) outside the containment, downstream of the flow measuring nozzle at initial no load conditions (Case A). The break assumed is the largest break which can occur anywhere outside the containment either upstream or downstream of the isolation valves.

Offsite power is assumed available such 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 signal by low steam line pressure will trip the reactor and isolate the steam lines. Even with the failure of one steam line valve, release is limited to no more than 10 seconds for the other steam generators while the one steam generator blows down. The steam line isolation valves are required to be closed within the time specified by the technical specifications.

As shown in the reactivity plot (Figure 15.4.2-7) the core attains criticality with the rod cluster control assemblies inserted (with the design shutdown of 1600 pcm assuming one stuck assembly) before a boron solution of approximately 1950 ppm enters the RCS from the Safety Injection System. The delay time associated with boron delivery consists of system alignments and discharge line purging described above. Core power peaks well below the nominal full power value.

The accumulators provide an additional source of borated water (1950 PPM) after the RCS pressure decreases to below 641.5 psig. The calculation assumes the 1950 ppm 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 from the Safety Injection System and the accumulator due to changes in the RCS pressure. The core boron concentration vs. time for each of the four cases analyzed is shown in Figures 15.4.2-8, 15.4.2-16, 15.4.2-24, and 15.4.2-32.

Figures 15.4.2-11 through 15.4.2-18 show Case B, a steam line rupture at the exit of a steam generator (upstream of the flow measuring nozzles) at no load. The sequence of events is similar to those of Case A except that criticality is attained earlier due to more rapid cooldown and a higher peak core average power is attained.

Figures 15.4.2-19 through 15.4.2-26 and 15.4.2-27 through 15.4.2-34 show the responses of the salient parameters for Case C and Case D, respectively. These cases correspond to the cases discussed above with the added complication of a loss of offsite power at the time of break initiation; Case C assumes a break downstream from the flow measuring device and Case D assumes an upstream break. Safety injection begin 60 seconds after the break; delays associated with system alignment, diesel generator startup, and pump discharge line purging are accounted for. In both Case C and Case D, criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. For both these cases the peak core power remains well below the nominal full power value.

The sequence of events tables for the Main Steam Line Break analyses are included in Table 15.4.1-12, Sheets 1 and 2. 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 long-term core cooling after the initial transient is over. Operators are instructed to maintain the intact steam generator levels within the narrow range level scale. In cases assuming a loss of offsite power core decay heat is removed to the atmosphere via the steam line safety valves which have been adequately sized for this purpose.

Generic thermal stress analyses and subsequent fracture mechanics analyses of reactor vessels have been performed for 4-Loop plants. These analyses were applied to a 4-Loop reactor vessel having material properties and end of life (40 years) accumulated fluence similar to the Sequoyah vessel. The fracture mechanics analysis utilized linear elastic fracture mechanics method in the evaluation of the reactor vessel integrity. The fracture mechanics analysis results show that the reactor vessel integrity under large steam line break conditions would be maintained over the design life of the vessel.

Steam pressure from the steam generators is relieved by the steam dump system, secondary system atmospheric safety valves, or secondary system relief valves. The operator is instructed to terminate auxiliary feedwater flow to the faulted steam generator as soon as he determines which steam generator is faulted. As soon as an indicated water level returns to the pressurizer the operator is instructed to turn off the safety injection pumps and restrict the charging pumps as required.

Following a steam line break incident, a steam line isolation signal will be generated almost immediately, causing the steam line isolation valves to close within a few seconds. If the break is downstream of the isolation valves, all of which subsequently close, the break will be isolated. If the

break is upstream of the isolation valves or one valve fails to close, the break will be isolated to three steam generators while the faulted one will continue to blow down. Only the case in which one steam generator continues to blow down is discussed here since the break followed by isolation of all steam generators will terminate the transient.

A safety injection signal (generated a few seconds after the break) will cause main feedwater isolation to occur. The only source of water available to the faulted steam generator is then the auxiliary feedwater system. Following steam line isolation, steam pressure in the steam line with the faulted steam generator will continue to fall rapidly, while the pressure stabilizes in the remaining three steam lines. The indication of the different steam pressures will be available to the operator within a few seconds of steam line isolation. This will provide the information necessary to identify the faulted steam generator so that auxiliary feedwater to it can be isolated. Manual controls are provided in the control room for start and stop of the auxiliary feedwater pumps and for the control valves associated with the auxiliary feedwater system. The means for detecting the faulted steam generator and isolation of auxiliary feedwater to it requires only the use of safety grade equipment available following the break. The removal of decay heat in the long term (following the initial cooldown) using the remaining steam generators requires only the Auxiliary Feedwater System as a water source and the secondary system safety valves and/or the power operated relief valve to relieve steam. Power to the motor driven auxiliary feedwater pumps is supplied by the onsite diesel generator units. The turbine driven AFW pump has redundant steam supplies. Flow (440 gpm) from one motor-driven auxiliary feedwater pump to one steam generator is sufficient for long term cooling.

The operator has available, in the control room, an indication of pressurizer water level from the instrumentation used in the reactor protection system. Indicated water level returns to the pressurizer in approximately five to seven minutes following the steam line break. To maintain the indicated water level the operator can start and stop the charging pumps as necessary. No operator action is assumed in the analysis. The pressurizer level instrumentation and manual controls for operation of the charging pumps meet the required standards for safety systems.

As indicated, the information for terminating auxiliary feedwater is available to the operator within one minute of the break while the information required for stopping the charging pumps and safety injection pump becomes available within five to seven minutes following the break. The requirements to terminate auxiliary feedwater flow to the faulted steam generator and stop the charging pumps and safety injection pumps can be met by simple switch actions by the operators, i.e., closing auxiliary feed discharge valves and stopping charging pumps and safety injection pumps. Thus, the required simple actions to limit the cooldown and depressurization can be easily recognized, planned, and performed within ten minutes. For the longer time requirements for decay heat removal and plant cooldown the operator has time on the order of hours to respond.

The worst case condition for long term cooling following a steam line break is loss of offsite power with failure of one emergency power train, since the condition requires the greatest amount of operator action and the longest time to achieve cold shutdown. However, since the plant can be maintained safely at hot standby conditions for extended periods of time, there is no safety requirement which dictates rapid achievement of cold shutdown conditions.

With only onsite power available, the plant can be maintained in a safe hot standby condition using the intact steam generators by supplying feedwater with the auxiliary feedwater system, and venting steam through the secondary side, power-operated relief valves. The relief valves will be controlled to gradually reduce pressure and temperature as the core residual heat decays. If the relief valves

are not available, the safety valves will be used for steam dump. In this case, the primary system pressure would be controlled such that adequate subcooling is maintained. Primary system temperature would be maintained at that value necessary to lift the steam generator safety valves as necessary to match the decay heat from the core. This temperature would be approximately 553 F which corresponds to the lowest steam generator safety valve setpoint of 1064 psig. For either means of steam relief, the steam generator water level will be maintained within the span of the narrow range indicators.

Margin to Critical Heat Flux

A complete set of statepoints are reviewed to determine the most limiting condition. Past experience in performing DNB analyses for steam line breaks for Westinghouse cores has shown that Case B (inside break with offsite power) is always worse than Case A. Cases A and B generally have very similar temperatures and pressures, but Case B returns to a power level of 1.5 to 2 percentage points greater than Case A. It is this higher power level that makes Case B the worse of the two.

A detailed nuclear and thermal-hydraulic analysis of the limiting steam line break, Case B, statepoint (see Table 15.4.2-1) was performed. The results of the analysis show that minimum DNBR is very high. This assures that DNB will not occur and that the DNB design basis is met for the steam line break event.

15.4.2.1.3 Framatome ANP Safety Evaluation With Replacement Steam Generators in Unit 1

The thermal response characteristics of the transient discount a challenge to the RCS and main steam system pressure limit. RCS pressure is reduced throughout the transient in response to the excessive heat removal from the steam generators. Therefore, this event is analyzed for DNB concerns.

Two of the critical parameters that effect the system and core responses to the steam line break are (primary-to-secondary) heat transfer area and break size. The OSG has a maximum area of 4.6 ft^2 whereas, the RSG has a flow restrictor at the exit of the SG with an area of 1.42ft^2. The OSG represents an increased heat transfer area relative to the RSG. It is expected that the greater heat transfer area will increase primary heat removal but the flow restrictor will limit the increase in over-cooling. It follows that there is a potential for an increase in the return to power associated with the RSG. The Steam Line Break, therefore was reanalyzed for the RSGs utilizing identical methods described in Section 15.4.2.1.2.

The RELAP5/MOD2 analyses show that the return to power with the RSGs is slightly greater than that with the OSGs. However, the RCS pressure remains higher at the time of peak power. As a result, the LYNXT analysis showed a slightly higher margin to DNB. Therefore, the analysis with OSGs remains bounding and applicable to Sequoyah Unit 1 with RSGs. All acceptance criteria for this event continue to be met subsequent to the installation of the RSGs.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shellside fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break.

Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. If the break is postulated in a feedline upstream of the check valve, it would affect the Nuclear Steam Supply System only as a loss of feedwater (see Subsection 15.2.8). If the break occurs upstream or downstream of the feedline check valve in the main steam valve vaults, the main feedwater isolation valves may become submerged due to flooding and fail to respond to automatic or manual control/isolation signals and/or may spuriously fully or partially close. However, since main feedwater isolation is not necessary to mitigate the consequences of a feedwater line break (see Subsection 15.4.2.2.2) or loss of feedwater event (see Subsection 15.2.8), the loss of operability of the main feedwater isolation valves is an acceptable consequence of a main feedwater line break in the main steam valve vaults (Reference 57). Similarly, the main steam isolation bypass valves and their operators would also be submerged, the MSIBVs are closed and their control circuits deenergized during power operation and so they remain closed following submergence. In addition, the MFIVs and MSIBVs are designated as containment isolation valves and have post-accident monitoring valve position indication in the MCR which would also be lost or be unreliable. However, since valve position indication is not utilized to mitigate a loss of MFW nor MFLB event, loss of position indication for these valves is an acceptable consequence of a main feedwater line break in the main steam valve vaults (again, see Reference 57).

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break), or a RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Paragraph 15.4.2.1, "Rupture of a Main Steam Line." Therefore, only the RCS heatup effects are evaluated for a feedline rupture.

A feedline rupture reduces the ability to remove heat generated by the core from the RCS because of the following reasons:

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

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

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

An Auxiliary Feedwater System is provided to assure that adequate feedwater will be available such that:

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

2. Liquid in the RCS shall be sufficient to cover the reactor core at all times.

The severity of the feedwater line break 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. Based on sensitivity studies, it has been shown that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line, occurring at full power with and without loss of offsite power, with no credit taken for pressurizer spray. The method of analysis, results, and conclusions for these cases are discussed below. A number of analyses have also been performed based on the functioning of the EAM and TTD safety systems. These analyses are discussed in Reference 53.

The following provides the necessary protection against a main feedwater rupture;

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

    a. High pressurizer pressure,
    b. Overtemperature delta-T
    c. Low-low steam generator water level in one or more steam generator,
    d. Safety injection signals from either of the following:

        i. Low steam line pressure
        ii. High containment pressure

        (Refer to Chapter 7 for a description of the actuation system.)

2. An Auxiliary Feedwater System to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to Section 10.4.7.2 for description of the Auxiliary Feedwater System.)

15.4.2.2.2 Analysis of Effects and Consequences

Analysis of the effects and consequences following a main feedwater line break have been conducted by the licensee and by Framatome Technologies Inc. Both analyses determine the plant transient upon such an event. The licensee analysis examines the pressure response inside containment using the MONSTER code (Reference 55) and is documented in Reference 56. The licensee analysis focuses on a reactor trip initiated by a safety-injection signal upon receipt of High Containment pressure. The analysis conducted by Framatome as follows focuses on a reactor trip following a low-low steam generator water level in one or more steam generators. The steam generators are assumed to have 15% of the tubes plugged.

Method of Analysis

A detailed analysis using RELAP5/MOD2 (Reference 60) is performed to obtain the plant parametric response due to a feedline rupture. The digital computer simulator of RELAP5 includes plant nuclear kinetics, reactor coolant system (with pressurizer and steam generators), main feedwater, auxiliary feedwater, and safety injection systems. The code computes the resultant system parameters including the steam generator level, pressurizer water level, and the reactor coolant average temperature.

Major assumptions are:

1. The plant is initially operating at 102% of the engineered safeguards design rating.

2. Initial reactor coolant average temperature is 5.5°F above the nominal value, and the initial pressurizer pressure is 30 psi above its nominal value.

3. No credit is taken for the pressurizer spray.

4. No credit is taken for the high pressurizer pressure reactor trip.
   Note: This assumption is made for calculational convenience. Pressurizer power-operated relief valves and spray could act to delay the high pressure trip. Assumptions 3 and 4 permit evaluation of one hypothetical, limiting case rather than two possible cases: one with a high pressure trip and no pressure controls; and one with a pressure control but no high pressure trip.

5. Main feed water to all steam generators is assumed to stop at the time the break occurs.

6. Discharge through the break in the affected steam generator is not restricted to liquid only. Two-phase discharge is modeled at the break.

7. No credit is taken for the low-low water level trip on the affected steam generator until the steam generator level reaches 0% of the narrow range span and after the expiration of any applicable delays imposed by the TTD System.

8. The worst possible break area is assumed; i.e., one that ensures that the initial reactor coolant system depressurization is maximized. This assumption minimizes subcooling margin during the post trip reactor coolant heatup period.

9. No credit is taken for heat energy deposited in reactor coolant system metal during the Reactor Coolant System heatup.

10. No credit is taken for charging or letdown.

11. Loss of offsite electrical power is assumed after the reactor trip, and reactor coolant flow decreases to natural circulation.

12. The RELAP5 code realistically calculates the appropriate heat transfer based upon the prevailing physical conditions in the generator.

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

14. The auxiliary feedwater is actuated by the low-low steam generator water level signal. The auxiliary feedwater system is assumed to supply a total of 410 gpm to two unaffected steam generators as follows:

    a. The turbine-driven pump is assumed to fail.

    b. The motor-driven pump supplying the faulted steam generator is assumed to conservatively spill all its flow out the break. The intact steam generator aligned to that pump is therefore assumed to receive no flow.

    c. The remaining motor-driven pump supplies flow to two intact steam generators.

    A 60 second delay was assumed following the low-low level signal to allow time for startup of the emergency diesel generators and the auxiliary feedwater pumps.

Note: An auxiliary feedwater system failure scenario involving a motor-driven pump was also analyzed by Westinghouse with 5% of the tubes plugged in the steam generators. See discussion under the "Results" section for a description of the scenario, assumptions, and results. Framatome has confirmed that the discussion is also applicable to the 15% steam generator tube plugging condition.

Results

Comparing the analyses conducted by the licensee and Framatome, the Framatome analysis bounds the licensee analysis, with regard to RCS heatup. Results from the Framatome analysis are presented herein.

Figures 15.4.2-35 and -36 show the calculated plant parameters following a feedline rupture for the case with offsite power. Figures 15.4.2-37 and -38 show similar results from the case with loss of offsite power. The calculated sequence of events for both cases analyzed are presented in Table 15.4.1-9.

The system response following the feedwater line rupture is similar for both cases analyzed. The results show that the pressures in the RCS and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure remains at or slightly below the steady-state pressure before the reactor trip on low-low steam generator level. When the turbine trips on reactor trip, the primary pressure shows a rapid increase due to the mismatch between the heat generated and the heat removed by the steam generators. This is followed by reduction in heat generation in the core due to rod drop and the event becomes an overcooling event. The primary pressure drops rapidly due to cooldown and the pressurizer level drops. When the low steamline pressure SI signal actuates isolation of the steam generators from the affected generator, the primary system heats up. The primary safety valves do not open in this event. Safety injection is actuated by the low steamline pressure signal and the reactor vessel and the pressurizer start to refill to original levels. When the intact steam generators fill back up to normal levels, the steam produced in the steam generators is relieved through the MSSVs and a viable heat removal mechanism is maintained via the auxiliary feedwater.

In both of the cases, the core remains fully covered with water throughout the transient. The auxiliary feedwater is capable of removing the decay heat and cooling the primary system. Bulk boiling does not occur in the RCS at any time in the transient.

An evaluation of the full power feedline rupture and the part-power feedline rupture cases analyzed for the TTD system was performed by Westinghouse (for the 5 percent plugged tubes) considering an auxiliary feedwater system failure scenario involving a motor-driven pump. This evaluation is valid for the 15 percent tube plugging levels since the conclusions are not dependent upon the tube plugging levels. Under this scenario, the auxiliary feedwater system is assumed to provide 1070 gal/min to 3 unaffected steam generators 10 minutes after the low-low steam generator water level signal. Assuming a feedline rupture to loop 4, the following assumptions were made:

1. Steam Generators 1 and 2 receive no auxiliary feedwater from the Train A motor-driven auxiliary feedwater pump (single failure).

2. Steam Generators 3 and 4 receive no auxiliary feedwater from the Train B motor-driven auxiliary feedwater pump (all flow out of the break).

3. The turbine-driven auxiliary feedwater pump flow spills out of the break (Steam Generator 4) and thereby starves the remaining unfaulted steam generators (1, 2, and 3).

4. Operator action isolates the auxiliary feedwater line from the turbine-driven auxiliary feedwater pump and the Train B motor-driven auxiliary feedwater pump. Ten minute operator action is taken credit for isolating the auxiliary feedwater system from the faulted steam generator.

The turbine-driven auxiliary feedwater pump provides flow to Steam Generators 1, 2, and 3, and the motor-driven auxiliary feedwater pump provides flow to Steam Generator 3. A total of 1070 gal/min auxiliary feedwater is available after the 10 minute operator action time to the three unfaulted steam generators. The secondary side steam pressure in the unfaulted steam generators, which drives the turbine-driven AFW pump, reaches and remains at or near the Main Steam safety valve (MSSV) set point pressure during the critical transient time (AFW initiation to event turnaround) and after event turnaround. After event turnaround, less AFW is required to continue plant cooldown.

The single failure of the turbine-driven pump proved more limiting for the full power feedline break cases. The single failure case of the motor-driven auxiliary feedwater pump does not invalidate the TTD analysis presented in Reference 53.

15.4.2.2.3 Conclusion

Results of the analysis show that for the postulated feedline rupture, the assumed Auxiliary Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core.

The Rupture of a Main Feedwater Pipe event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.4.3 Steam Generator Tube Rupture

15.4.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power operated relief valves.

In view of the fact that the steam generator tube material is Inconel 600 and is highly ductile material, it is considered that the assumption of a complete severance is somewhat conservative. The more probable mode of tube failure would be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance and an accumulation of minor leaks which exceed the limits established in the Technical Specifications is not permitted during the unit operation.

The operator is expected to determine that a steam generator tube rupture has occurred, and to identify and isolate the faulty steam generator on a restricted time scale in order to minimize

contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The recovery procedure can be carried out on a time scale which ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the isolation procedure can be completed within 30 minutes of accident initiation.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

a. Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow.

   Mismatch before trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that steam generator.

b. Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated by low pressurizer pressure, and follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates auxiliary feedwater addition.

c. The steam generator blowdown liquid monitor, main steamline monitor, and the condenser off gas radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system. The steam generator blowdown liquid monitor will automatically terminate steam generator blowdown.

d. The reactor trip automatically trips the turbine and if offsite power is available the steam dump valves open permitting steam dump to the condenser. In the event of a coincident station blackout, the steam dump valves would automatically close to protect the condenser. In this case, the steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam generator safety and/or power operated relief valves.

e. Following reactor trip, the continued action of auxiliary feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser, or in the case of loss of offsite power, steam relief to atmosphere, is attenuated during the 30 minutes in which the recovery procedure leading to isolation is being carried out.

f. Safety injection flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment.

15.4.3.2 Analysis of Effects and Consequences

Method of Analysis

In estimating the mass transfer from the RCS through the broken tube the following conservative assumptions are made:

a. Reactor trip occurs automatically as a result of low pressurizer pressure.

b. Following the initiation of the safety injection signal, all centrifugal charging SI pumps are actuated and continue to deliver flow for 30 minutes.

c. After reactor trip the break flow reaches equilibrium at the point where incoming safety injection flow is balanced by outgoing break flow. The resultant break flow persists for 30 minutes beyond initiation of the accident.

d. The steam generators are controlled at the safety valve setting rather than the power operated relief valve setting. Auxiliary feedwater flowrate equivalent to approximately 2% of the nominal main feedwater flowrate (660 gal/min) is assumed to be available to all steam generators.

e. The operator identifies the accident type and terminates break flow to the faulty steam generator within 30 minutes of accident initiation. Included in this 30 minute time period would be an allowance of 5 minutes to trip the reactor and actuate the safety injection system, 10 minutes to identify the accident as a steam generator tube rupture and 15 minutes to isolate the faulty steam generator.

Mass and energy balance calculations are performed to determine primary to secondary mass release and to determine amount of steam vented from each of the steam generators.

Recovery Procedure

Immediately apparent symptoms of a tube rupture accident such as falling pressurizer pressure and level and increased charging pump flow are also symptoms of small steam line breaks and loss of coolant accident. It is therefore important for the operator to determine that the accident is a rupture of a steam generator tube in order that he may carry out the correct recovery procedure. The accident under discussion is uniquely identified by a main steam line radiation alarm, condenser air ejector radiation alarm and/or a steam generator blowdown radiation alarm and the operator will proceed with the following recovery procedures if one of these alarms is received. In the event of a relatively large rupture, it will be clear soon after trip that the level in one steam generator is rising more rapidly than in the other. This too is a unique indication of a tube rupture accident.

The operator carries out the following procedures subsequent to reactor trip which lead to isolation of the ruptured steam generator and to unit cooldown.

With Offsite Power Available:

a. Identify ruptured steam generator by rising water level or high radiation indication in shell side fluid.

b. Isolate flow from ruptured steam generator by closing the main steam isolation valve and bypass valve, steam generator blowdown valves, and turbine-driven auxiliary feedwater pump steam supply from ruptured steam generator.

c. When water level in ruptured steam generator is above a minimum level, then isolate auxiliary feedwater flow to ruptured steam generator.

d. Initiate RCS cooldown by dumping steam from intact steam generators to condenser. This action is required to establish adequate subcooling to permit reducing RCS pressure.

e. When adequate subcooling is established, then reduce RCS pressure using pressurizer sprays or a power-operated relief valve to terminate break flow and restore pressurizer level.

f. Verify SI termination criteria is met and then terminate ECCS flow.

g. Resume RCS cooldown using intact steam generators until RHR system is placed in service.

Without Offsite Power:

a. Identify ruptured steam generator by rising water level or high radiation indication in shell side fluid.

b. Isolate flow from ruptured steam generator by closing the main steam isolation valve and bypass valve, steam generator blowdown valves, and turbine-driven auxiliary feedwater pump steam supply from ruptured steam generator.

c. When water level in ruptured steam generator is above a minimum level, then isolate auxiliary feedwater flow to ruptured steam generator.

d. Initiate RCS cooldown using atmospheric relief valves on intact steam generators. This action is required to establish adequate subcooling to permit reducing RCS pressure.

e. When adequate subcooling is established, then reduce RCS pressure using a pressurizer power-operated relief valve to terminate break flow and restore pressurizer level.

f. Verify SI termination criteria is met and then terminate ECCS flow.

g. Resume RCS cooldown using intact steam generators until RHR system is placed in service.

After the Residual Heat Removal System is placed in operation, the condensate accumulated in the secondary system can be examined and processed as required.

Section 15.4.3.1 describes the accident sequence as analyzed. The flow from a broken tube is assumed to reach an equilibrium at the point where safety injection flow is balanced by break flow. This break flow is conservatively assumed to persist until 30 minutes following accident initiation at which time the operator will have terminated the break flow through the faulty steam generator. Section 15.4.3.2 outlines operations which the operator could perform to terminate flow through the faulty steam generator.

Following isolation of the faulty steam generator, the primary pressure is reduced by either pressurizer spray or operation of the pressurizer power operated relief valve. It should be noted that the reduction in primary pressure will result in a decrease in break flow. When the primary pressure has been reduced to ruptured steam generator pressure, excess makeup flow is stopped and break flow is terminated.

The core will remain completely covered by liquid throughout the accident, thus clad temperatures will remain very near the saturation temperature of the coolant, even if DNB was postulated to occur.

There is ample time available to carry out the above recovery procedures such that isolation of the affected steam generator is established before water level rises into the main steam pipes. This analysis used 30 minutes as the time that operators stop flow from the primary to the secondary side of the faulted steam generator; however, at least 40 minutes is available for operators to stop this flow before the water level rises into the main steam pipes.

Results

The previous assumptions lead to a conservative estimate for the total amount of reactor coolant transferred to the secondary side of the ruptured steam generator as a result of a tube rupture accident. Approximately 172,700 lbs. of reactor coolant is discharged to the secondary side of the ruptured steam generator before break flow is isolated at 30 minutes (8400 lbs. before reactor trip and at 164,300 lbs. after reactor trip). Reactor trip occurs at 65 seconds into the accident. A fraction of the break flow flashes directly to steam while the remainder mixes with the secondary coolant in the steam generator. This flashing fraction is conservatively determined based on hot leg temperature to be 18.0% prior to reactor trip and 4.74% after reactor trip.

Also, approximately 138,900 lbs. of steam are released from the ruptured steam generator to the atmosphere during the 30-minute period (76,588 lbs. before reactor trip and 62,312 lbs. after reactor trip). The steam releases from the intact steam generators were conservatively calculated to be:

0 - 65 sec.	232,000 lbs.
65 - 1800 sec.	170,000 lbs.
0.5 - 2.0 hr.	360,000 lbs.
2 - 8 hr.	1,237,000 lbs.

The accident radiological consequences are reported in Section 15.5.5 based on the above values for break flows, flashing fractions, and steam releases.

15.4.3.3 Conclusions

A steam generator tube rupture will cause no subsequent damage to the Reactor Coolant System or the reactor core. An orderly recovery from the accident can be completed even assuming simultaneous loss of offsite power.

The Steam Generator Tube Rupture event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.4.4 Single Reactor Coolant Pump Locked Rotor

15.4.4.1 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 a pressure increase throughout the Reactor Coolant System. The insurge into the pressurizer compresses the steam volume, and may open the pressurizer safety valves in the analysis. For conservatism in peak pressures, the power-operated relief valves are not included in the analysis.

15.4.4.2 Analysis of Effects and Consequences

Method of Analysis

Two digital computer codes are used to analyze this transient. The RELAP5/MOD2-B&W code (Reference 60) 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 transient, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the LYNXT code (Reference 64) and the statistical core design methodology of BAW-10170 (Reference 57a).

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 nominal steady-state conditions, i.e., 102% steady state power level, steady state pressure and steady state coolant average temperature. The DNB calculations are performed according to a statistical core design methodology that incorporates calibration and measurement uncertainties. Consequently, nominal conditions are adequate for transient analysis. Only the primary safety valves are allowed to maintain the primary pressure in the transient, thus maximizing the peak primary pressures. The pressure response is shown in Figure 15.4.4-4. For the DNB calculations, the pressure is assumed constant at the initial value. To maximize the power response during the event, the least negative Doppler power coefficient and +7.0 pcm/F moderator coefficient are assumed.

Evaluation of the Pressure Transient

After pump seizure, reactor coolant system flow is reduced and the system heats up and pressurizes. A reactor trip occurs as a consequence of low flow. The neutron flux is rapidly reduced by control rod insertion. Loss of off-site power is assumed to occur simultaneously with the reactor trip.

No credit is taken for the pressure reducing effect of pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after the plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect. The pressurizer safety valves are actuated at 2500 psia and their capacity for relief is as stated in Section 5.2.2.

Evaluation of the Effects of DNB in the Core During the Accident

The minimum DNBR calculated in the hot channel for this event is less than the DNBR limit of 1.5. Consequently, a clad temperature excursion of short duration is predicted. Less than 10% of the fuel pins could experience DNB during the accident.

Locked Rotor Results

Table 15.4.4-1 gives a summary of the results for the transient analysis of the reactor coolant pump locked rotor event. Transient values of pressurizer pressure, reactor vessel flow coastdown, nuclear power and thermal power are shown in Figures 15.4.4-1 through 15.4.4-5.

15.4.4.3 Conclusions

1. Since the peak RCS pressure reached during any of the transients 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.

2. Since the peak fuel temperature is well below the 5080°F fuel temperature limit and the peak cladding temperature is well below the 1800°F cladding temperature limit, the core will remain intact with no consequential loss of core cooling capability. Typically, for the purpose of dose calculations, all pins that experience DNB are assumed to fail. The evaluation of DNB effects for this accident showed that less than 10% of the fuel pins experience DNB. This is bounded by the conclusions presented in Section 15.5.3.

The Single Reactor Coolant Pump Locked Rotor event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.4.5 Fuel Handling Accident

15.4.5.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

15.4.5.2 Analysis of Effects and Consequences

For the analysis and consequences of the postulated fuel handling accident, refer to Subsection 15.5.6.

15.4.6 Rupture Of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

15.4.6.1 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 rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Design Precautions and Protection

Certain features in the Sequoyah Nuclear Plant 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 rod cluster control assemblies and minimizes the number of assemblies inserted at power.

Mechanical Design

The mechanical design is discussed in Section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of rod cluster control assembly drive mechanism housing failure sufficient to allow a rod cluster control assembly to be rapidly ejected from core are listed below:

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

2. The mechanism housings are individually hydrotested as they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed RCS.

3. 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 ASME Code, Section III, for Class 1 components.

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

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy seal welds. These welds are inspected in accordance with the plant's Inservice Inspection Program (Section 5.2.8) and a discussion on non-welding repair is provided in FSAR Section 4.2.3.2.2.

Nuclear Design

Even if a rupture of a rod cluster control assembly drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected rod cluster control assembly is inherently limited. In general, the reactor is operated with the rod cluster control changes caused by core depletion and xenon transients are compensated by boron changes.

Further, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of a rod cluster control assembly ejection accident. Therefore, should a rod cluster control assembly be ejected from its normal position during high 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 rod cluster control assemblies above this limit guarantees shutdown capability and acceptable power distribution. The position of all rod cluster control assemblies is continuously indicated in the control room. An alarm will occur if a bank of rod cluster control assemblies approaches its insertion limit or if one assembly deviates from its bank. There are low and low-low level insertion monitors with visual and audio signals. Operating instructions require boration at low level alarm and emergency boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 26. The protection for this accident is provided by the power range high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a rod cluster control assembly mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking is not expected to cause damage to adjacent housings leading to increased severity of the initial accident.

Limiting Criteria

Due to the extremely low probability of a rod cluster control assembly ejection accident, limited fuel damage is 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 27). 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 28) results, which indicted that this threshold decreases by about 10% 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, conservative 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 uniradiated fuel and 200 cal/gm for irradiated fuel.

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

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

15.4.6.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the RCCA ejection accident is performed in two stages, first an average core nuclear power transient calculation and then a hot spot heat transfer 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 29.

Average Core Analysis

The spatial kinetics computer code, TWINKLE (Reference 30), is used for the average core transient analysis. This code uses cross sections generated by LEOPARD (Reference 31) to solve the two group neutron diffusion theory kinetic equations 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 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 special effects of axial moderator feedback and rod cluster control assembly movement and the elimination of axial feedback weighting factors. 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. Further description of TWINKLE appears in Subsection 15.1.9.

Hot Spot Analysis

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN (Reference 25). 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 parabolic radial power generation is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong correlation (Reference 32) to determine the film boiling coefficient after DNB. The DNB heat flux 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 that predicted by design fuel heat transfer codes presently used by Westinghouse.

For full power cases, the design initial hot channel factor (F_QT) is input to the code. The hot channel factor during the transient is assumed to increase from the steady state design value to the maximum transient value in 0.1 seconds, and remain at the maximum for the duration of the transient. This is conservative, since detailed spatial kinetics models show that the hot channel factor decreases shortly after the nuclear power peak due to power flattening caused by preferential feedback in the hot channel (Reference 29). Further description of FACTRAN appears in Subsection 15.1.9.

System Overpressure Analysis

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

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer spray and pressure relief valves. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure.

Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of calculated values for this type of core. The more important parameters are discussed below. Table 15.4.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using a synthesis of 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 and part length rod positions are considered in the calculations.

The total transient hot channel factors F_qT, is then obtained by combining the axial and radial factors.

Appropriate margins are added to the results to allow for calculational uncertainties, including an allowance for nuclear power peaking due to fuel densification.

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

single channel analysis. Physics calculations were 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 used. 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 were shown to be conservative compared to three dimensional analysis (Reference 29).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life were 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 the one dimensional steady state computer code with a Doppler weighting factor of 1.0. The resulting curve is conservative compared to design predictions for this plant. The Doppler weighting factor should be larger than 1.0 (approximately 1.3), just to make the present calculation agree with design predictions before ejection. This weighting factor will increase under accident conditions, as discussed above. The transient weighting factor used in the analysis is presented in Table 15.4.6-1.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values of 0.70% at beginning of life and 0.50% at end of life for the first cycle. The accident is sensitive to the ejected rod when its worth is nearly equal to or greater than β_{eff} as in zero power transients. In order to allow for future fuel cycles, pessimistic estimates were used in the analysis (0.55% at beginning of cycle and 0.45% β_{eff} at end of cycle).

Trip Reactivity Insertion

The trip reactivity insertion is assumed to be 4% from hot full power and 2% from hot zero power including the effect of one stuck 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 seconds after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for the coil to release the rods. The curve of rod insertion versus time which was used is shown in Figure 15.1.5-1. The time to full insertion assumed together with the 0.5 second delay overestimates the time for significant insertion of shutdown reactivity into the core. This is particularly important conservatism for hot full power accidents.

Results

The values of the parameters used in the analysis, as well as the results of the analysis, are presented in Table 15.4.6-1 and discussed below.

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 0.20% ΔK/K and 7.11 respectively. The peak hot spot fuel center temperature reached the beginning of life melt temperature of 4900°F. However, melting was restricted to less than 10% of the pellet.

Beginning of Cycle, Zero Power

For this condition, control bank D was assumed to be fully inserted and C was at its insertion limit. The worst ejected rod is located in control bank D and has a worth of 0.75% Δk/k and a hot channel factor of 14.05.

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 0.21% Δk/k and 7.88 respectively. The peak hot spot fuel temperature exceeded the end of life melt temperature of 4800°F. However, melting was restricted to less than 10% of the pellet. The variation in melt temperature with burnup is discussed in Paragraph 4.4.1.2.

End of Cycle, Zero Power

Original analysis - The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and bank C at its insertion limit. The results were 1.01% Δk and 22.2, respectively. The peak fuel center temperature was 4203°F.

Reanalysis - This transient was reanalyzed in Reference 65 to address an increase in the Cycle 8 ejected rod hot channel factor. The reanalysis was performed consistent with the original analysis except that the bounding Cycle 8 values of 0.91%Δk for ejected rod worth and 24.8 for the hot channel factor were assumed. The results of the reanalysis were bounded by the original analysis.

A summary of the cases presented above is given in Table 15.4.6-1. The nuclear power and fuel clad temperature transients for the worst case in terms of fuel melt (EOL full power) are presented in Figures 15.4.6-1 and 15.4.6-2. The same transients for the worst case in terms of clad temperature (EOL zero power) are presented in Figures 15.4.6-3 and 15.4.6-4.

Fission Product Release

It is assumed that fission products are released from the gaps of all rods having a DNBR of less than the safety analysis limit. In all cases considered, less than 10% of the rods entered DNB based on a detailed 3 dimensional THINC analysis. Although limited fuel melting at the hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth 1 dollar at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Reference 29). Since the severity of the present analysis does not exceed this "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

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 force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. 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 ration, 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 conservatively ignored in the analyses.

15.4.6.3 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 primary system. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10%.

The rupture of a Control Rod Drive Mechanism Housing event has been evaluated with respect to the CENP-Westinghouse steam generator replacement at Sequoyah Unit 1. The evaluation concludes that the parameters important to the consequences of this event are not adversely affected by the RSG.

15.4.7 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50.

2. Deleted by Amendment 21

3. Deleted by Amendment 21

4. Deleted by Amendment 21

5. Deleted by Amendment 21

6. Deleted by Amendment 8

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15. Deleted by Amendment 21

16. Deleted by Amendment 21

17. Deleted by Amendment 21

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20. Deleted by Amendment 21

21. Deleted by Amendment 21

22. F. S. Moody, Transactions of the ASME, Journal of Heat Transfer, February 1965, Figure 3, page 134.

23. Deleted by Amendment 21

24. Deleted by Amendment 21

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

26. T. W. T. Burnett, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactor," WCAP-7306, April, 1969.

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TABLE 15.4.1-1 (Sheet 1)

PLANT OPERATING RANGE SUPPORTED BY THE REALISTIC LARGE BREAK LOCA ANALYSIS

Plant Physical Description

Fuel	
Cladding Outside Diameter	0.374 in.
Cladding Inside Diameter	0.326 in.
Cladding Thickness	0.024 in.
Pellet Outside Diameter	0.3195 in.
Pellet Density	96 percent of theoretical
Active Fuel Length	144 in.
Gd_2O_3 Concentrations	2, 4, 6, 8, w/o
Reactor Coolant System	
Flow Resistance	Calculated by Model
Pressurizer Location	Faulted Loop
Hot Assembly Location	Any Core Location
Hot Assembly Type	17 x 17
SG Tube Plugging	≤15 percent

Plant Initial Operating Conditions

Reactor	
Nominal Power	3455 MWt
Initial Power	3479 MWt[1]
Peaking Factor (F_q)	≤ 2.65[2]
Hot Channel Factor ($F_{\Delta H}$)	≤ 1.706[3]
Moderator Temperature Coefficient	≤ 0 at Hot Full Power
Fluid Conditions	
Loop Flow	131.6 Mlbm/hr ≤ M ≤ 152.8 Mlbm/Hr
RCS Average Temperature	578.2ºF ≤ T ≤ 583ºF
Upper Head Temperature	Tcold Temperature[4]
Pressurizer Pressure	1859.7 psia ≤ P ≤ 2459.7 psia
Pressurizer Level	57 percent ≤ L ≤ 95 percent
Accumulator Pressure	614.7 psia ≤ P ≤ 697.7 psia
Accumulator Volume	1004.6 ft^3 ≤ V ≤ 1095.4 ft^3
Accumulator Temperature	95ºF ≤ T ≤ 130ºF
Accumulator Flow Resistance (fl/D)	Calculated by Model
Minimum ECCS Boron Concentration	≥ 2400 ppm

Notes -

1. Includes uncertainties.
2. Ensure that a minimum 7 percent peaking margin is maintained to the F_q limits when operating at the positive or negative AFD limit.
3. Includes 4 percent measurement uncertainty.
4. Upper head temperature will change based on sampling of RCS temperature.

TABLE 15.4.1-1 (Sheet 2)

PLANT OPERATING RANGE SUPPORTED BY THE REALISTIC LARGE BREAK LOCA ANALYSIS

Accident Boundary Conditions

Break Location	Any RCS piping location
Break Type	Double-ended guillotine or split
Break Size	0.33 ≤ A ≤ 1.0 full pipe area (split)
	0.33 ≤ A ≤ 1.0 full pipe area (guillotine)
Single Failure	Loss of one train of ECCS
Offsite Power	Both available and not available
Charging Pump Flow	Minimum Safeguards
Safety Injection Pump Flow	Minimum Safeguards
Residual Heat Removal Pump Flow	Minimum Safeguards
ECCS Injection Temperature	120ºF
Charging Pump Delay	37 sec. (w/ off-site power)
	27 sec. (w/o off-site power)
Safety Injection Pump Delay	37 sec. (w/ off-site power)
	27 sec. (w/o off-site power)
Residual Heat Removal Pump Delay	37 sec. (w/ off-site power)
	27 sec. (w/o off-site power)
Containment Pressure	14.3 psia, nominal value
Upper Compartment Temperature	80ºF ≤ T ≤ 110ºF
Lower Compartment Temperature	95ºF ≤ T ≤ 130ºF
Containment Spray Delay	8 sec.
Containment Spray Temperature	55ºF

TABLE 15.4.1-2

REALISTIC LARGE BREAK LOCA TIME SEQUENCE OF EVENTS

Event	Time (s)
Break Opened	0.00
RCP Trip	N/A[1]
Safety Injection Signal	00.1
Accumulator Injection (Faulted Loop)	12.8
Accumulator Injection (Intact Loops)	14.8
Start of Charging Flow	37.1
SI/RHR Flow Delivery Begins (All Loops)	37.1
Start of Core Reflood	50.2
All Accumulators Empty	84.6
Peak Cladding Temperature Occurs	130.7
Transient Analysis terminated	501.2

Notes -

1. Offsite power was available for the limiting case set.

TABLE 15.4.1-3

REALISTIC LARGE BREAK LOCA RESULTS

Fuel Assembly Cladding	
Peak Temperature	2002ºF
Peak Temperature Time	130.7 seconds
Peak Temperature Elevation	9.831 ft
Metal-Water Reaction	
Maximum Oxidation	3.4199 percent
Total Oxidation	0.0200 percent

TABLE 15.4.1-4

REALISTIC LARGE BREAK LOCA
ICE CONDENSER CONTAINEMNT INITIAL AND BOUNDARY CONDITIONS

Containment Net Free Volume	
Upper Compartment	651,000 - 692,600 ft^3
Lower Compartment (minimum)	248,500 ft^3
Ice Condenser	181,400 ft^3
Dead Ended Compartments	129,900 ft^3
Initial Conditions	
Initial Mass of Ice	2.448 x 106 lbm
Containment Pressure (nominal)	14.3 psia
Upper Containment Temperature	80 °F -110 °F
Lower Containment Temperature	95 °F -130 °F
Humidity	100 percent
Containment Spray	
Maximum Total Flow	2 x 7700 = 15,400 gpm
Minimum Spray Temperature	55°F
Fastest Post-LOCA initiation of spray	10 sec (ramped to full flow between 8 and 10 s)
Containment Air Return Fans[1]	
Post-LOCA initiation	600 sec
Total Flow	120,000 cfm

Notes -

1. Due to the relatively late start of the recirculation fan, it is not modeled in this analysis.

TABLE 15.4.1-5

REALISTIC LARGE BREAK LOCA
PASSIVE STRUCTURAL HEAT SINKS IN CONTAINMENT

Heat Sink	Area ft^2	Thickness ft	Inside Radius ft	Thickness ft	Height ft	Material	Left Side	Right Side
Reactor Cavity Walls	6438	2.02				Concrete	Lower Comp.	Insulated
Concrete Floor	4444	2.00				Concrete	Lower Comp.	Insulated
Interior Concrete	8464	1.00				Concrete	Lower Comp.	Insulated
Reactor Vessel Biological Shield Wall			11	6.0	19.88	Concrete	Lower Comp.	Lower Comp.
Steel Lined Refueling Canal in LC			13.	0.02083	21.48	Stainless Steel	Lower Comp.	
				4.0	21.48	Concrete		Lower Comp.
Crane Wall between LC & DE			41.5	3.0	33.72	Concrete	Lower Comp.	Dead End
Crane Wall in LC			41.5	3.0	29.37	Concrete	Lower Comp.	Insulated
Crane Wall in UC			41.5	3.0	32.44	Concrete	Upper Comp.	Insulated
Refueling Canal in Contact with Upper and Lower Compartment	2551	0.02083				Stainless Steel	Upper Comp.	
		3,87				Concrete		Lower Comp.
Refueling Canal in Contact with Annular Region	1,260	0.02083				Stainless Steel	Upper Comp.	
		3.0				Concrete		Annulus
Concrete Structure between Upper and Lower Compartment	13,081	2.34				Concrete	Upper Comp.	Lower Comp.
Interior Concrete	2278	3.0				Concrete	Upper Comp.	Insulated
Containment Shell	24,646	0.05417				Carbon Steel	Upper Comp.	Annulus
LC Steel Heat Sink	24,999	0.03674				Carbon Steel	Lower Comp.	Insulated
UC Steel Heat Sink	11669	0.4229				Carbon Steel	Upper Comp.	Insulated
Dead-End Steel Heat Sink	8610	0.074375				Carbon Steel	DE Comp.	Insulated

Material Properties	Thermal Conductivity (BTU/hr-ft-°F)	Volumetric Heat Capacity (BTU/ft^3-°F)
Concrete	0.84	30.24
Carbon Steel	27.3	59.2
Stainless Steel	9.87	59.22

TABLE 15.4.1-9 (Sheet 1)

TIME SEQUENCE OF EVENTS FOR CONDITION IV EVENTS

Accident	Event	Time (Sec)
Major Secondary System Pipe Rupture		
1. Case a	Steam line ruptures	0.0
	SIS Low Steamline Pressure Setpoint Reached	0.6
	Steam line isolation occurs	8.6
	Feedwater isolation occurs	9.6
	Pressurizer empties	14.0
	Safety Injection Flow Initiated	28.6
	Boron Reaches the Core	36.1
	Criticality Attained	50.1
2. Case b	Steam Line Ruptures	0.0
	SIS Low Steamline Pressure Setpoint Reached	0.2
	Steam Line Isolation Occurs	8.2
	Feedwater Isolation Occurs	9.2
	Pressurizer Empties	16.0
	Safety Injection Flow Initiated	28.2
	Criticality Attained	32.1
	Boron Reaches The Core	36.1
3. Case c	Steam line ruptures	0.0
	SIS Low Steamline Pressure Setpoint Reached	0.6
	Steam line isolation occurs	8.6
	Feedwater isolation occurs	9.6
	Pressurizer empties	16.0
	Safety Injection Flow Initiated	58.6
	Criticality Attained	66.1
	Boron Reached the Core	72.1
4. Case d	Steam Line Ruptures	0.0
	SIS Low Steamline Pressure Setpoint Reached	0.2
	Steam Line Isolation Occurs	8.2
	Feedwater Isolation Occurs	9.2
	Pressurizer Empties	18.0
	Criticality Attained	40.1
	Safety Injection Flow Initiated	58.2
	Boron Reaches The Core	70.1

TABLE 15.4.1-9 (Sheet 2)

TIME SEQUENCE OF EVENTS FOR CONDITION IV EVENTS

Accident	Event	Time (Sec)
a. End of Cycle, Zero Power	RCCA ejected	0
	Reactor trip setpoint reached (High Neutron Flux, high setting	0.16
	Rods begin to drop	0.66
	Peak clad average temperature reached	1.42
	Peak fuel center temperature reached	2.79
b. End of Cycle, Full Power	RCCA ejected	0
	Reactor trip setpoint reached (High Neutron Flux, high setting)	0.05
	Rods begin to drop	0.55
	Peak clad average temperature reached	2.36
	Peak fuel center temperature reached	3.99
c. Beginning of Cycle, Full Power	RCCA ejected	0
	Reactor trip setpoint reached (High Neutron Flux, high setting)	0.05
	Rods begin to drop	0.55
	Peak clad average temperature reached	2.29
	Peak fuel center temperature reached	4.36

TABLE 15.4.1-9 (Sheet 3)

TIME SEQUENCE OF EVENTS FOR CONDITION IV EVENTS

Accident	Event	Time (Sec)
Major rupture of a Main Feedwater Pipe with off-site power available	Main feedline rupture occurs	0.0
	Low-Low steam generator level reactor trip	4.9
	Rods begin to drop	7.0*
	Auxiliary feedwater started	65.0
	Low steamline pressure signal	178.7
	Minimum reactor vessel level reached	198.0
	Primary system starts cooling down	>7000
Major Rupture of a Main Feedwater Pipe without offsite power	Main feedline rupture occurs	0.0
	Low-Low steam generator level reactor trip	4.9
	Rods begin to drop	7.0*
	Auxiliary feedwater started	65.0
	Low steamline pressure reached	137.6
	Minimum pressurizer level reached	194.0
	PORV opens	900.0
	Primary system starts cooling down	3680.0

*There are no trip delays imposed by the TTD system at power levels greater than 50%.

Table 15.4.1-12 (Sheet 1)

TIME SEQUENCE OF EVENTS FOR STEAM LINE BREAK

Accident	Event	Time (Sec)
Major Secondary System Pipe Rupture		
1. Case A	Steam line rupture	0.0
	Steam line isolation setpoint reached	2
	Steam line isolation occurs	10
	Feedwater isolation occurs	22
	Safety Injection Flow initiated	30
	Criticality attained	35
	Pressurizer empties	57
	Boron reaches the core	98
2. Case B	Steam line rupture	0.0
	Steam line isolation setpoint reached	2
	Steam line isolation occurs	10
	Feedwater isolation occurs	22
	Pressurizer empties	30
	Safety Injection Flow initiated	30
	Criticality attained	32
	Boron reaches the core	97

Table 15.4.1-12 (Sheet 2)
(continued)

TIME SEQUENCE OF EVENTS FOR STEAM LINE BREAK

Accident	Event	Time (Sec)
Major Secondary System Pipe Rupture		
3. Case C	Steam line rupture	0.0
	Steam line isolation setpoint reached	2
	Steam line isolation occurs	10
	Feedwater isolation occurs	22
	Pressurizer empties	35
	Criticality attained	53
	Safety Injection Flow initiated	60
	Boron reaches the core	125
4. Case D	Steam line rupture	0.0
	Steam line isolation set point reached	2
	Steam line isolation occurs	10
	Feedwater isolation occurs	22
	Pressurizer empties	43
	Criticality attained	48
	Safety Injection Flow initiated	60
	Boron reaches the core	127

Table 15.4.2-1

LIMITING CORE PARAMETERS USED IN STEAM LINE BREAK DNB ANALYSIS

Case	Inside break with power (case B)
Reactor vessel inlet temperature	399.4 F (Faulted SG Loop) 492.4 F (Intact SG Loop)
RCS pressure	742.3 psia
RCS flow	106% (of nominal HZP)
Average Heat flux	16.4% (of nominal)
Time	80.0 seconds

Table 15.4.4-1

Summary of Results for Locked Rotor Transient

Maximum reactor Coolant System Pressure (psia)	2581.4
Maximum Clad Temperature (F)	1104
Maximum Fuel Temperature (F)	3264

TABLE 15.4.6-1

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	Beginning	Beginning	End	End#	End##
Power Level	102 pct	0 pct	102 pct	0 pct	0 pct
Ejected rod worth, %Δk/k	.20	.75	.21	1.01	0.91
Delayed neutron fraction, %	.55	.55	.44	0.45	0.45
Feedback reactivity weighting	1.3	2.4	1.6	3.55	3.55
Trip Reactivity, %Δk/k	4.0	2.0	4.0	2.0	2.0
F_q before rod ejection	2.62	--	2.62	--	--
F_q after rod ejection	7.11	14.05	7.88	22.2	24.8
Number of operational pumps	4	2	4	2	2
Max. fuel pellet average temperature, °F	4121	3156	4056	3760	3493
Max. fuel center temperature, °F	4971	3610	4879	4203	3940
Max. fuel stored energy, cal/gm	181	132	177	162	148.2

- Original
- Reanalysis

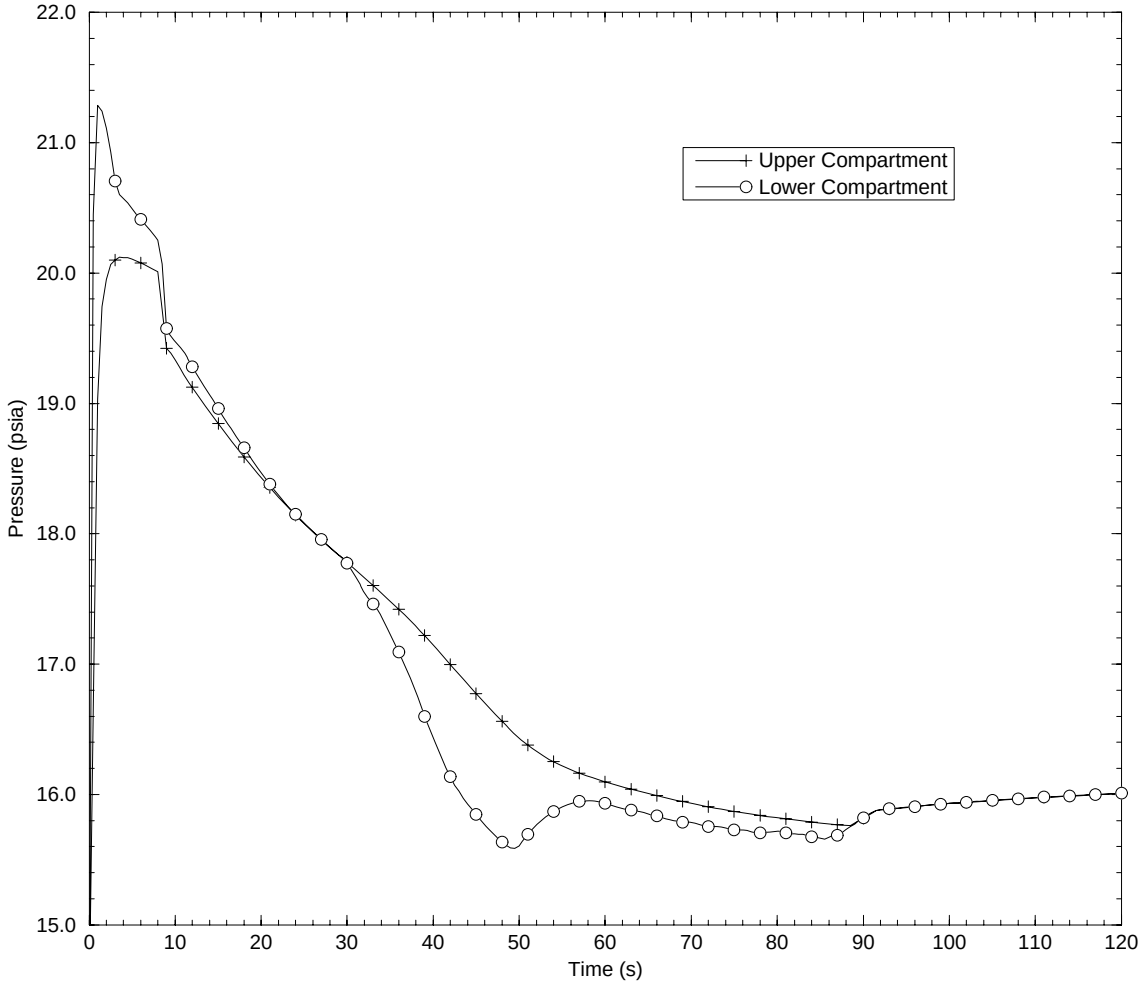

Figure 15.4.1-1

Realistic Large Break LOCA
Containment Compartment Pressure for Limiting PCT Case

Revised by Amendment 21

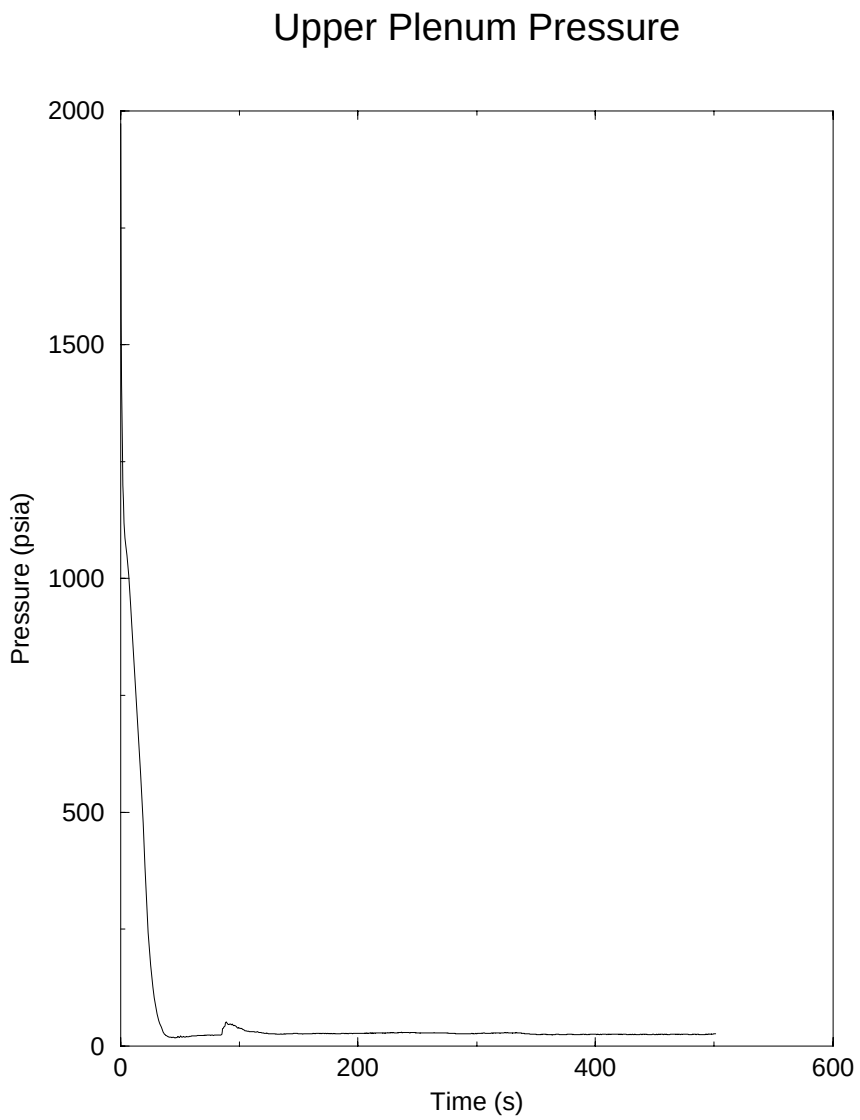

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Figure 15.4.1-2

Realistic Large Break LOCA

Reactor Vessel Upper Plenum Pressure for the Limiting PCT Case

Revised by Amendment 21

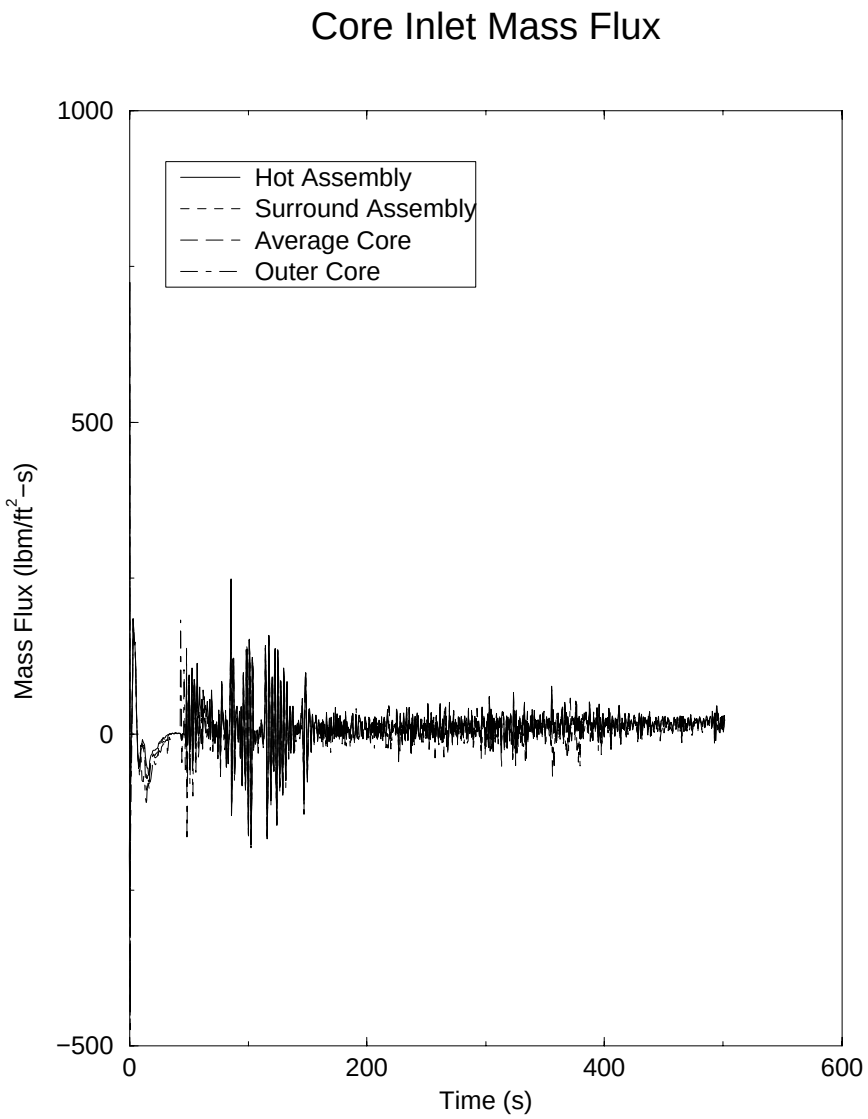

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Figure 15.4.1-3

Realistic Large Break LOCA
Core Inlet Mass Flux for the Limiting Case

Revised by Amendment 21

Best Available Historical Image

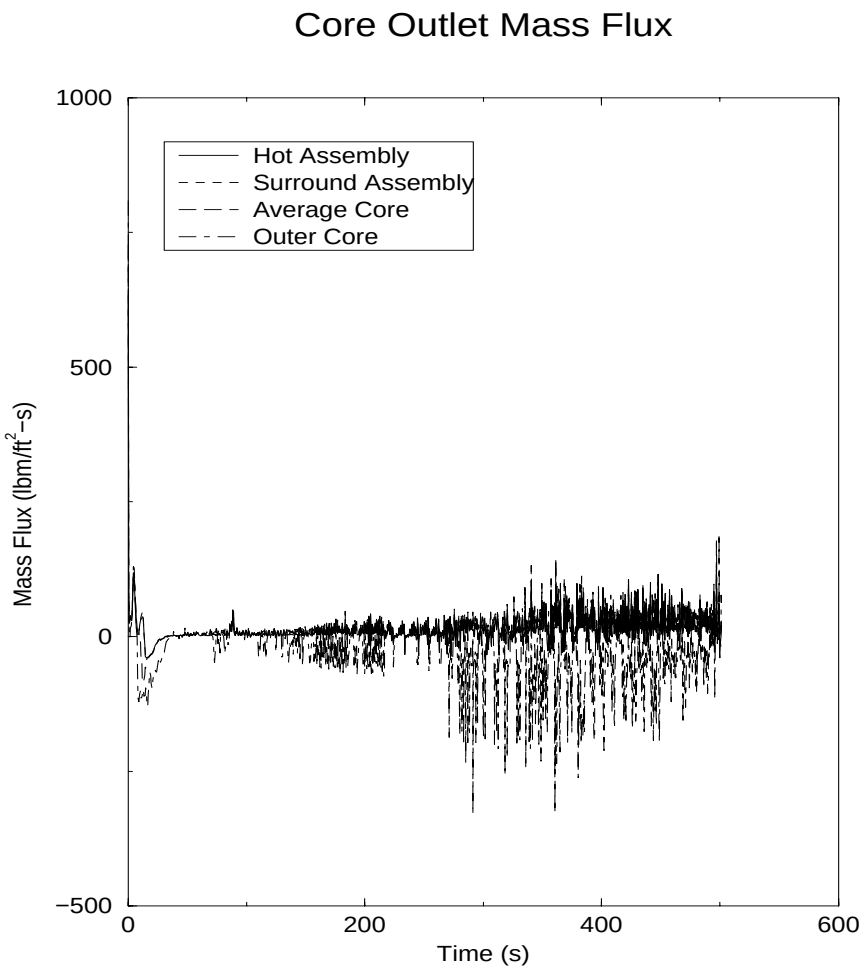

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Figure 15.4.1-4

Realistic Large Break LOCA
Core Outlet Mass Flux for the Limiting Case

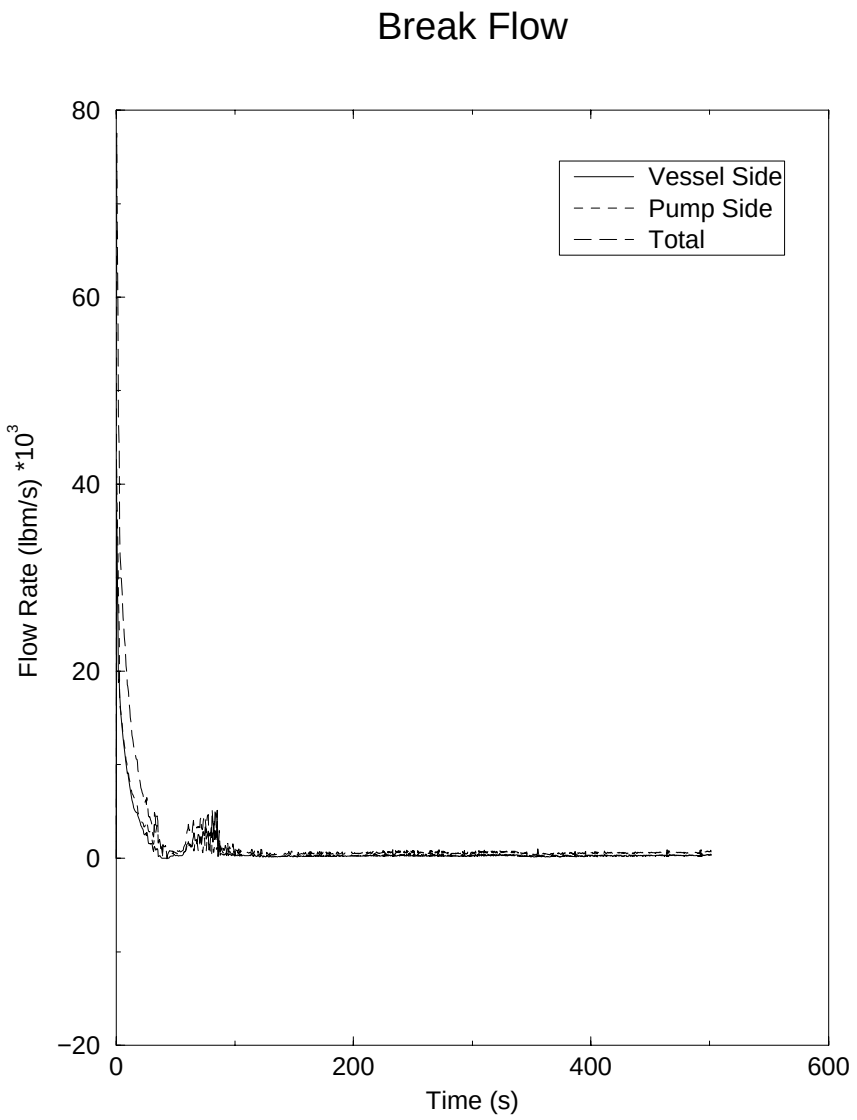

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Figure 15.4.1-6

Realistic Large Break LOCA
Break Mass Flowrate for the Limiting Case

Revised by Amendment 21

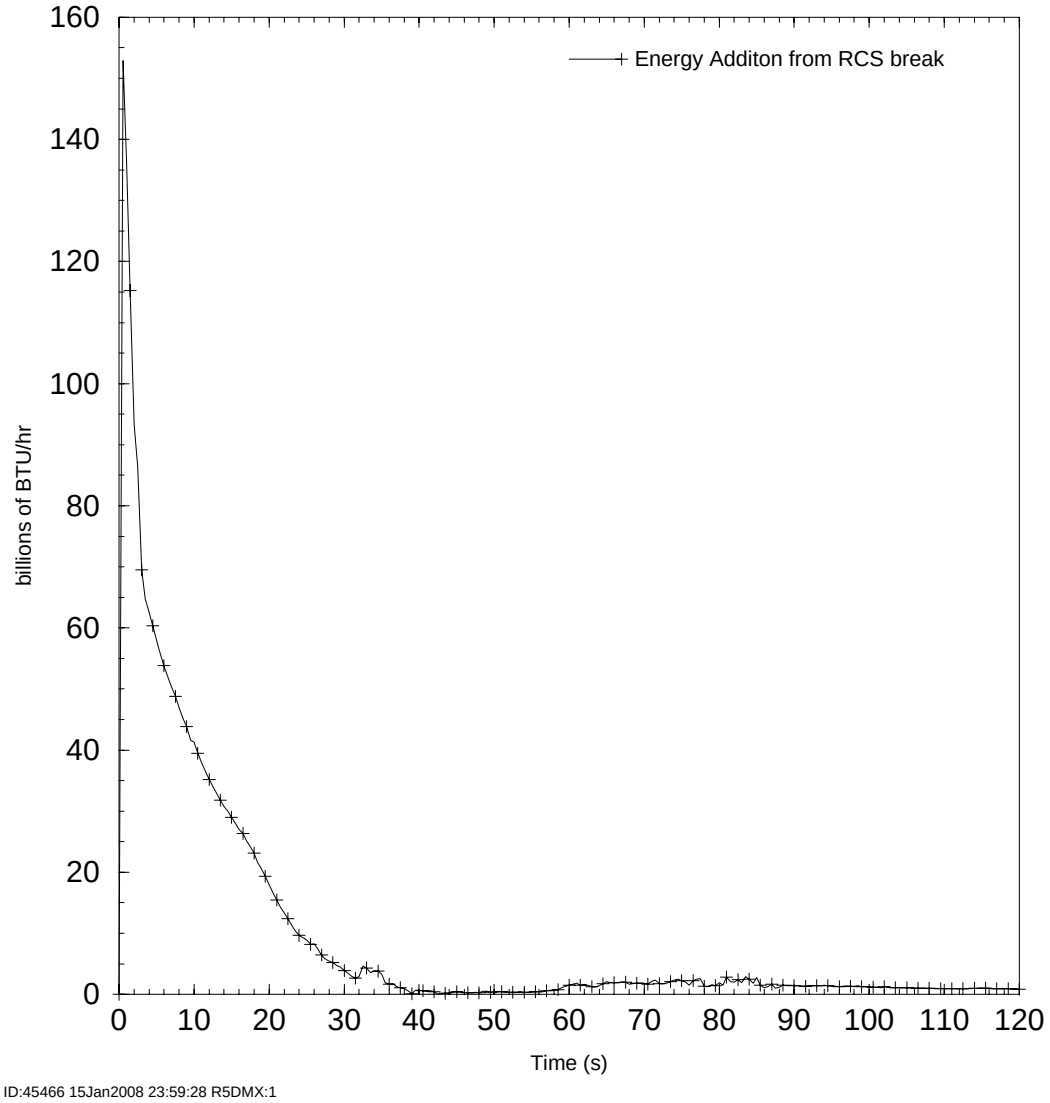

Figure 15.4.1-7

Realistic Large Break LOCA

Break Energy Flowrate for the Limiting Case

Revised by Amendment 21

Best Available Historical Image

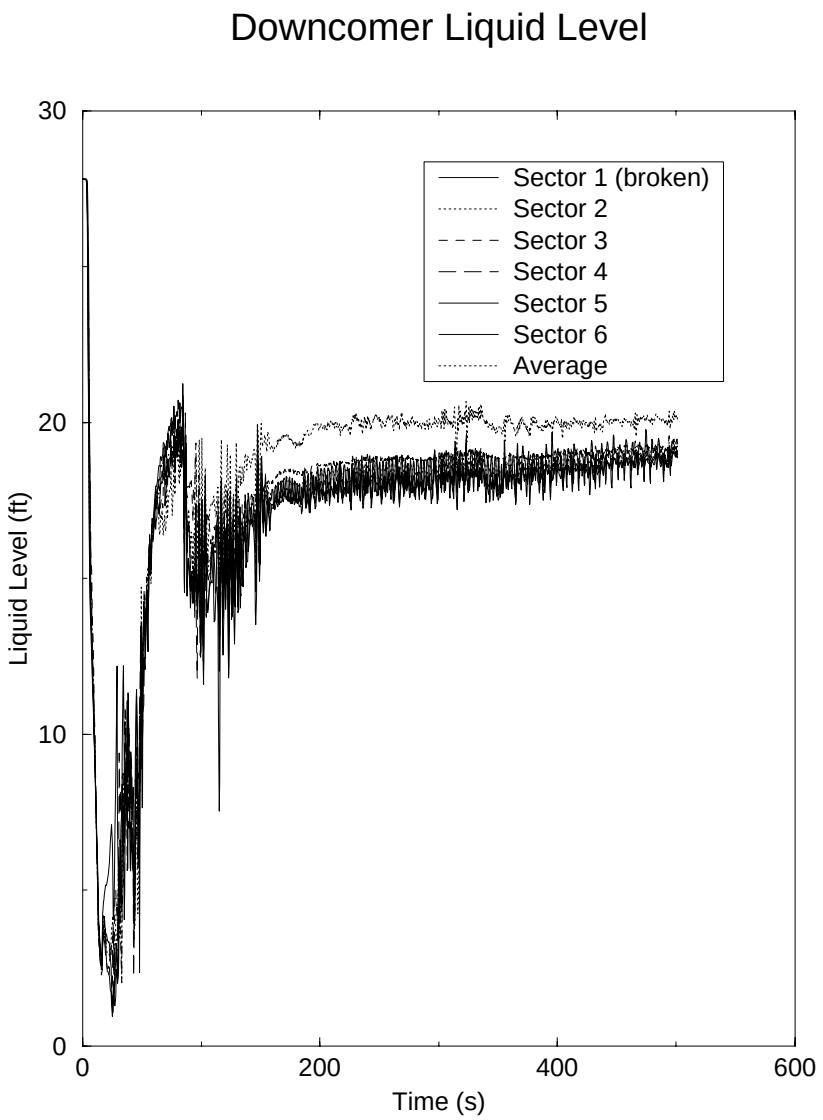

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Figure 15.4.1-9

Realistic Large Break LOCA
Downcomer Liquid Level for the Limiting Case

Revised by Amendment 21

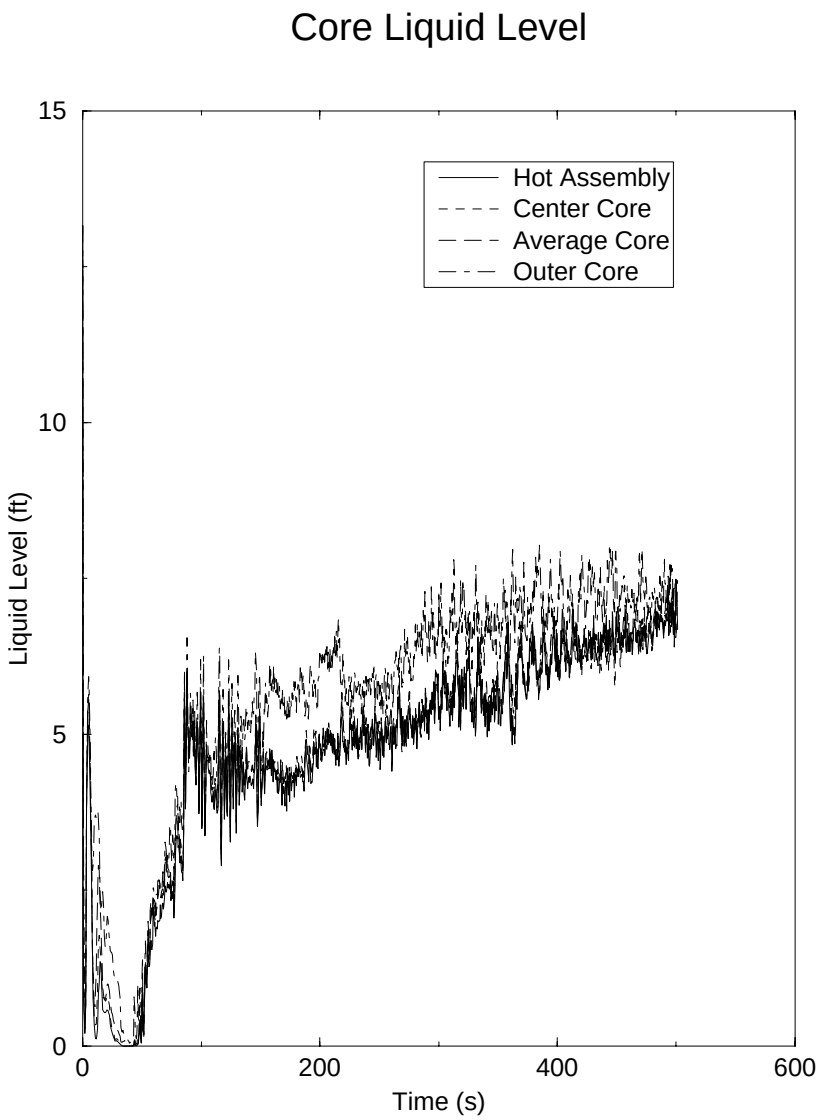

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Figure 15.4.1-10

Realistic Large Break LOCA
Core Liquid Level for the Limiting Case

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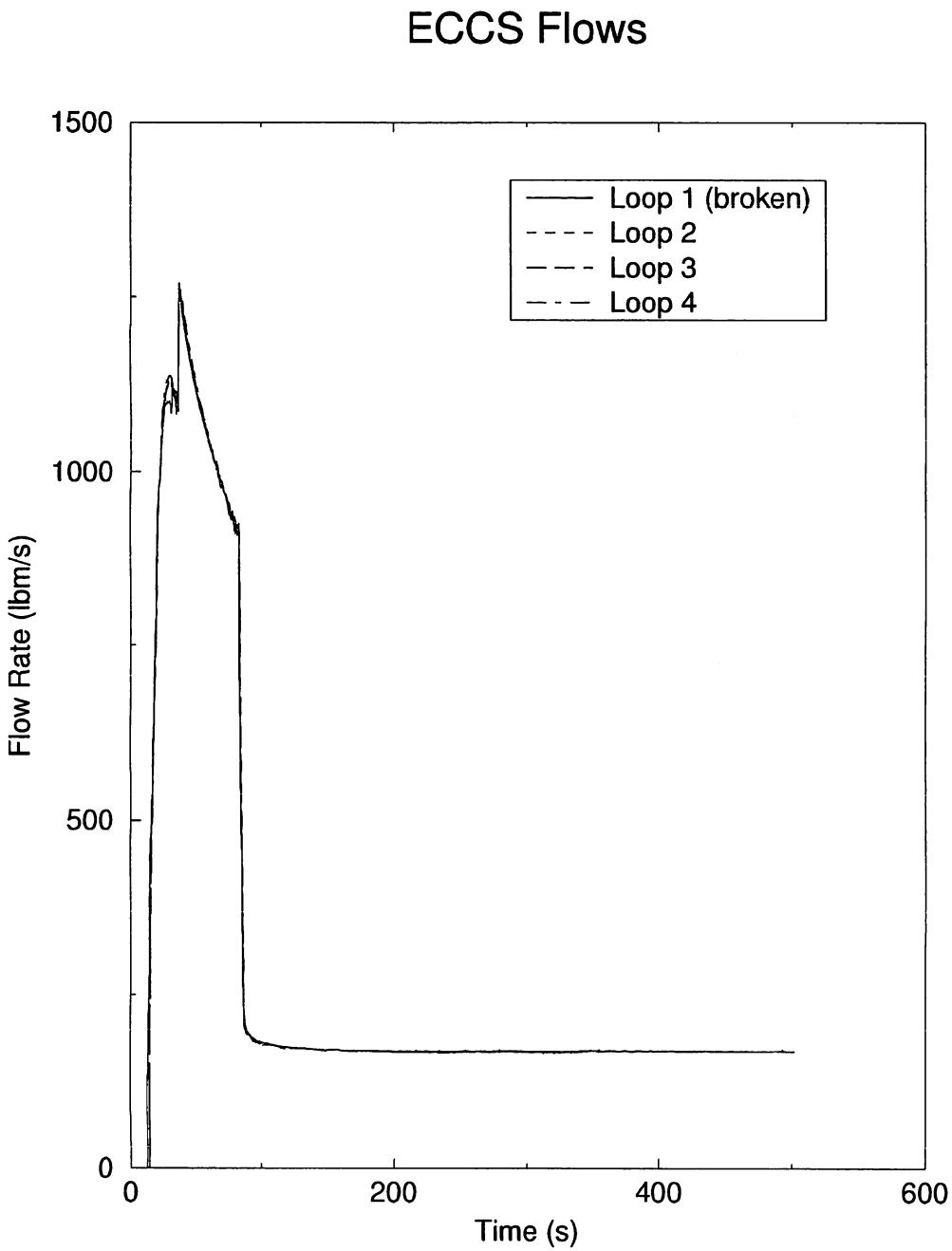

Figure 15.4.1-11

Realistic Large Break LOCA

Accumulator and Pumped SI Flowrate for the Limiting Case

Revised by Amendment 21

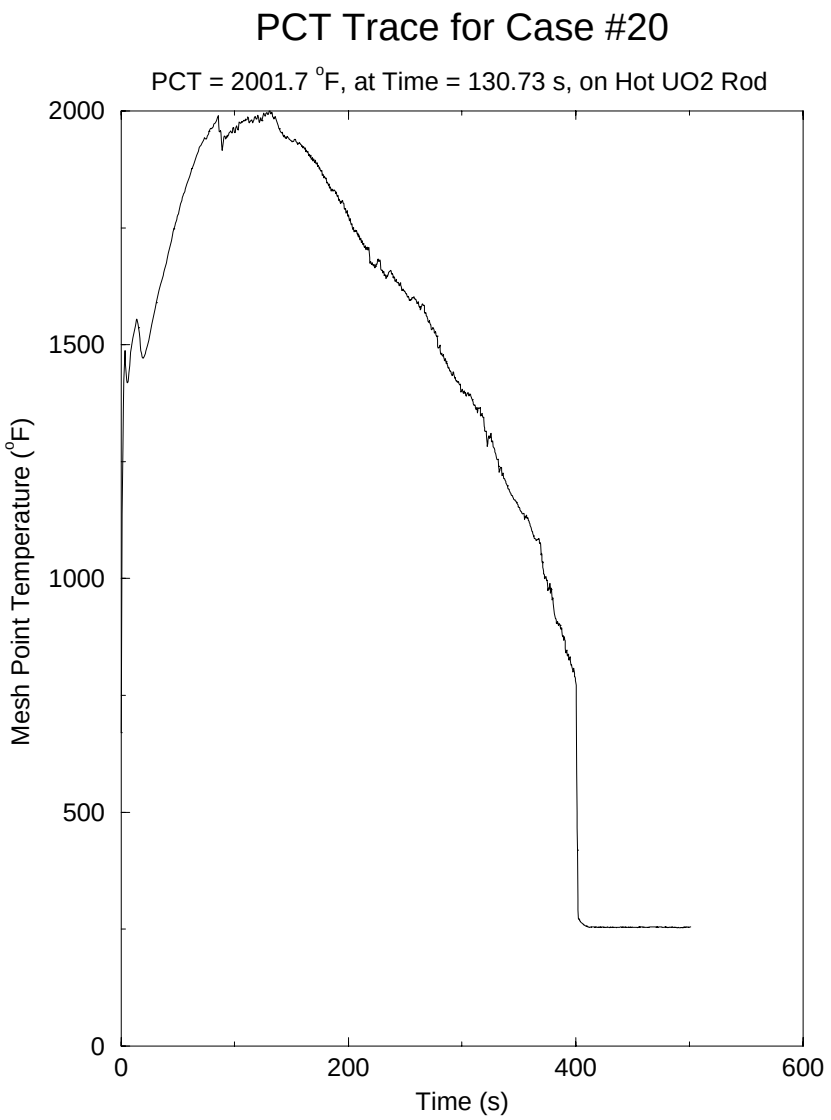

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Figure 15.4.1-14

Realistic Large Break LOCA
Peak Clad Temperature for the Limiting Case

Revised by Amendment 21

Best Available Historical Image

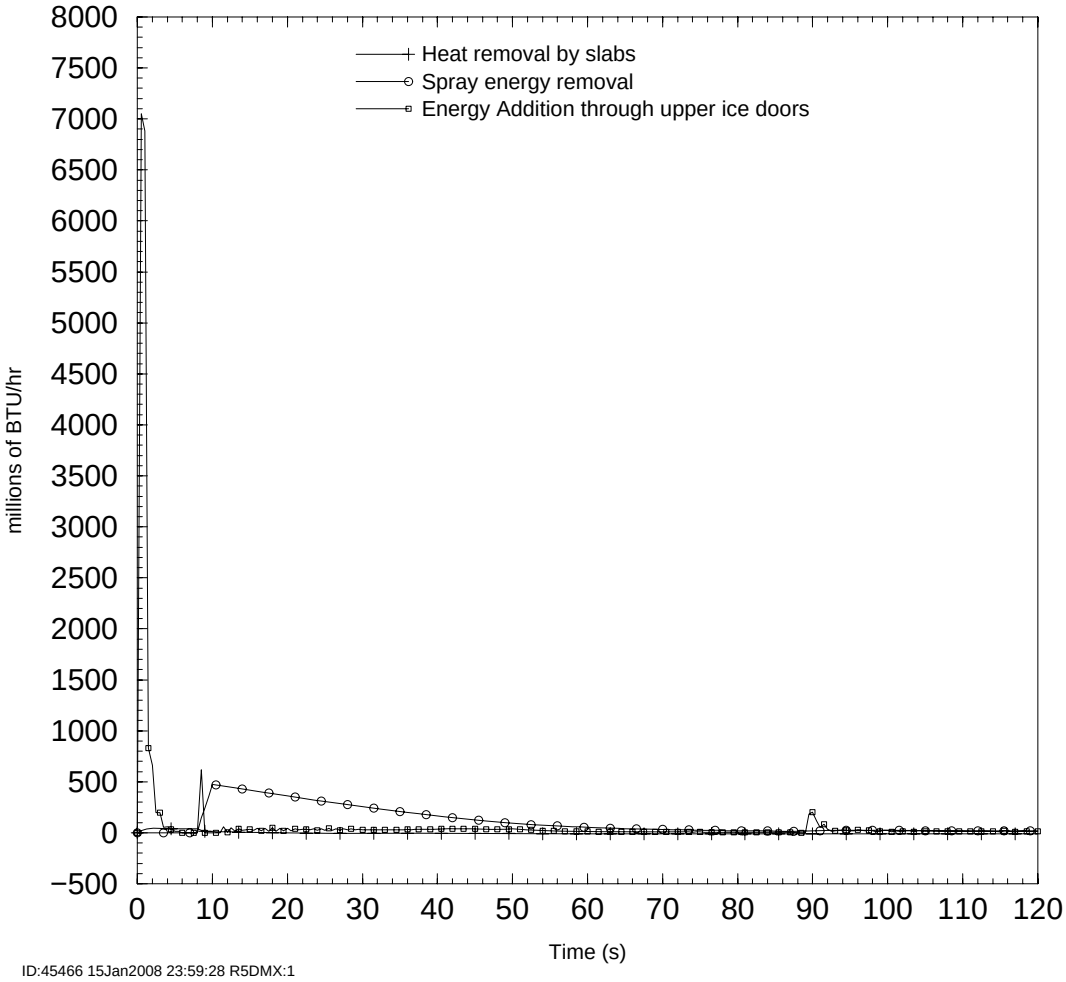

Figure 15.4.1-15

Realistic Large Break LOCA
Upper Compartment Energy Addition & Removal - Limiting Case

Revised by Amendment 21

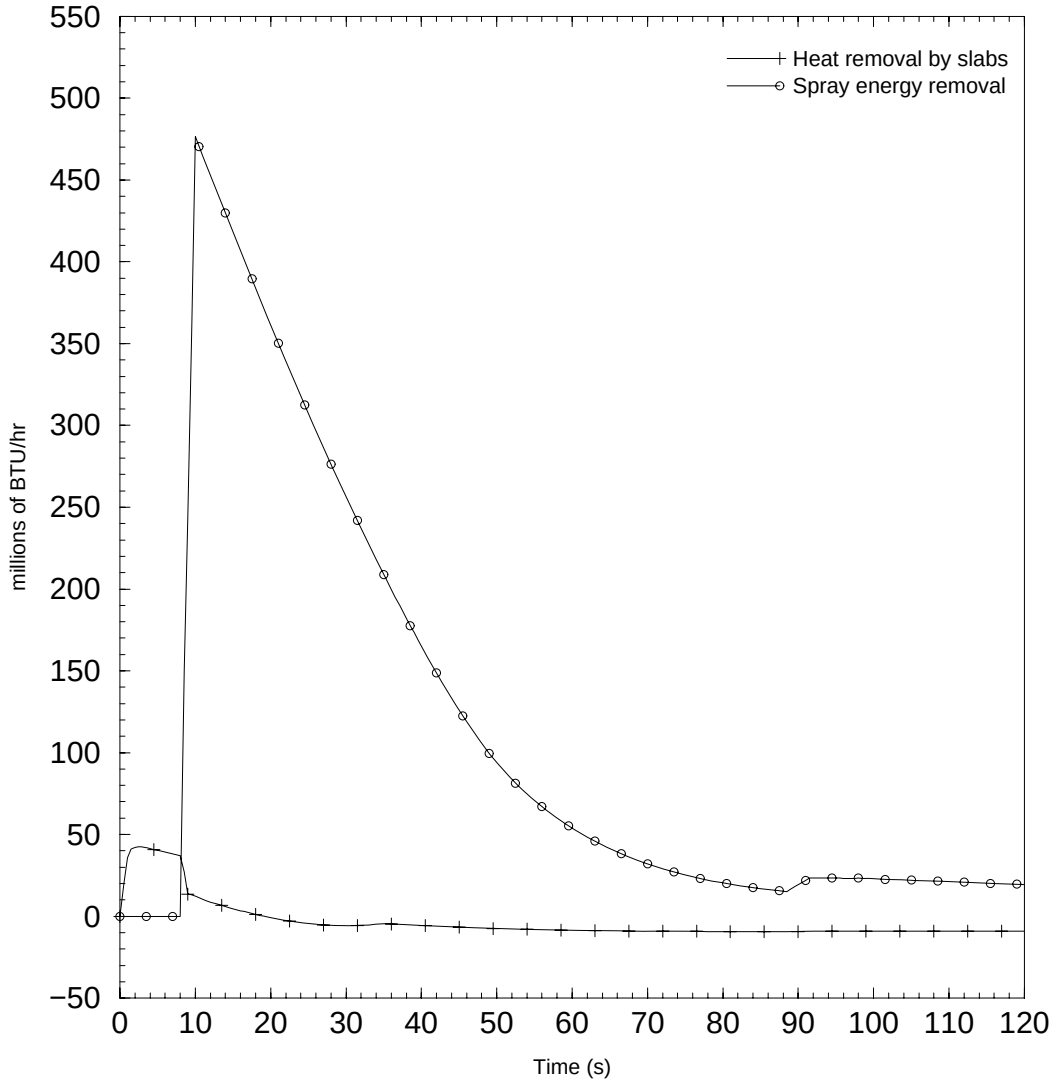

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Figure 15.4.1-16

Realistic Large Break LOCA
Upper Compartment Heat Removal Rate for the Limiting Case

Revised by Amendment 21

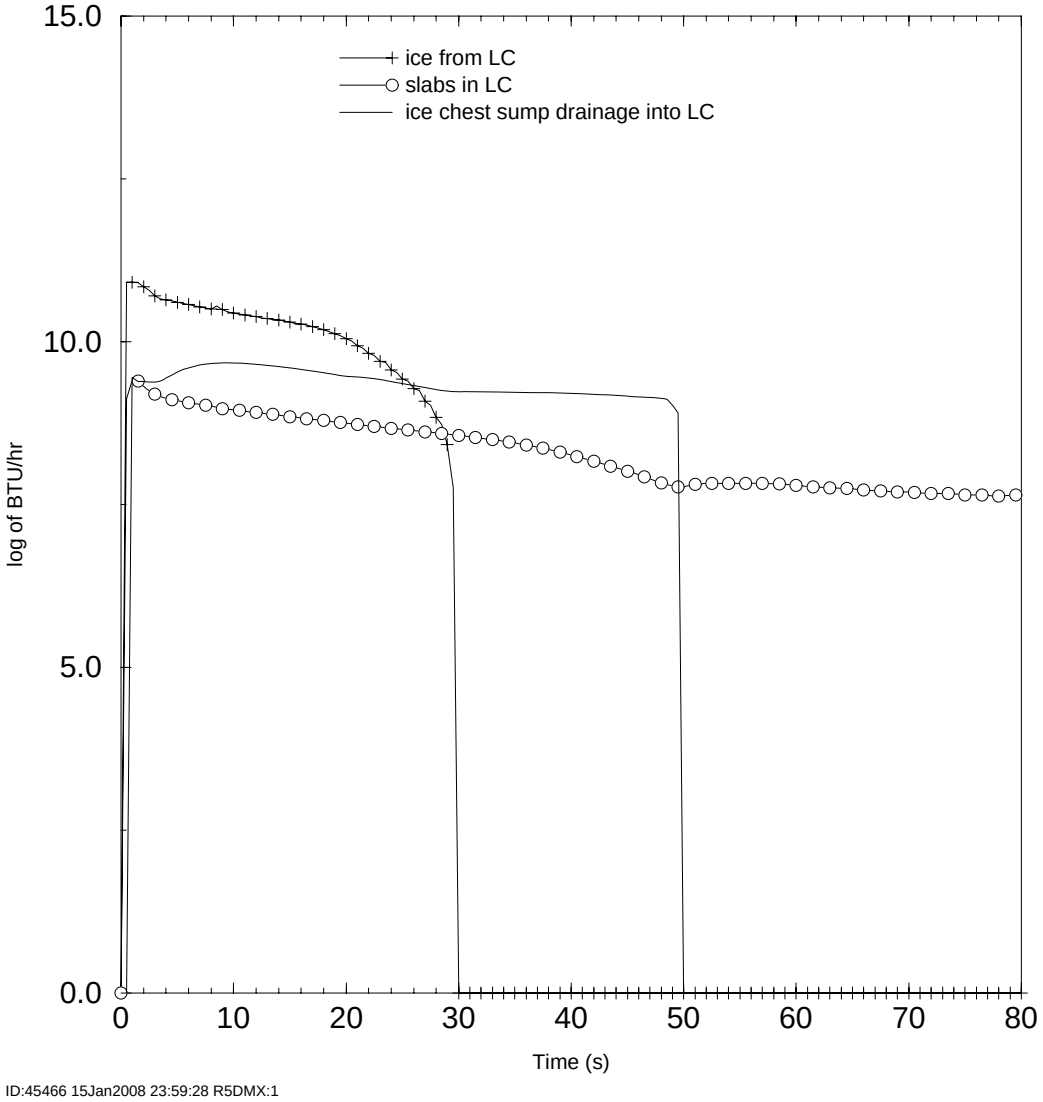


Figure 15.4.1-17

Realistic Large Break LOCA

Lower Compartment Heat Removal Rate for the Limiting Case

Revised by Amendment 21

Best Available Historical Image

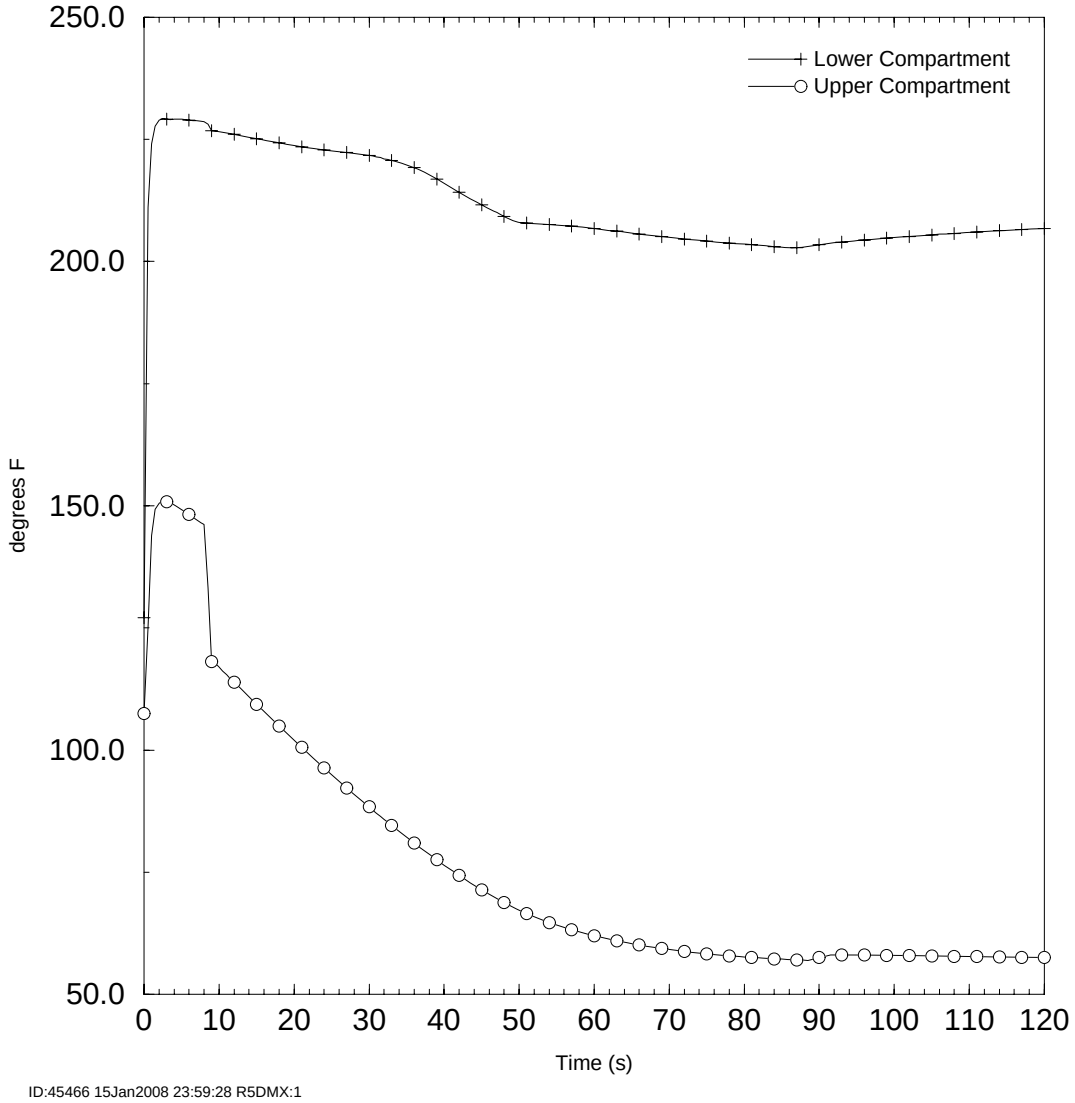


Figure 15.4.1-18

Realistic Large Break LOCA
Compartment Temperature for the Limiting Case

Revised by Amendment 21

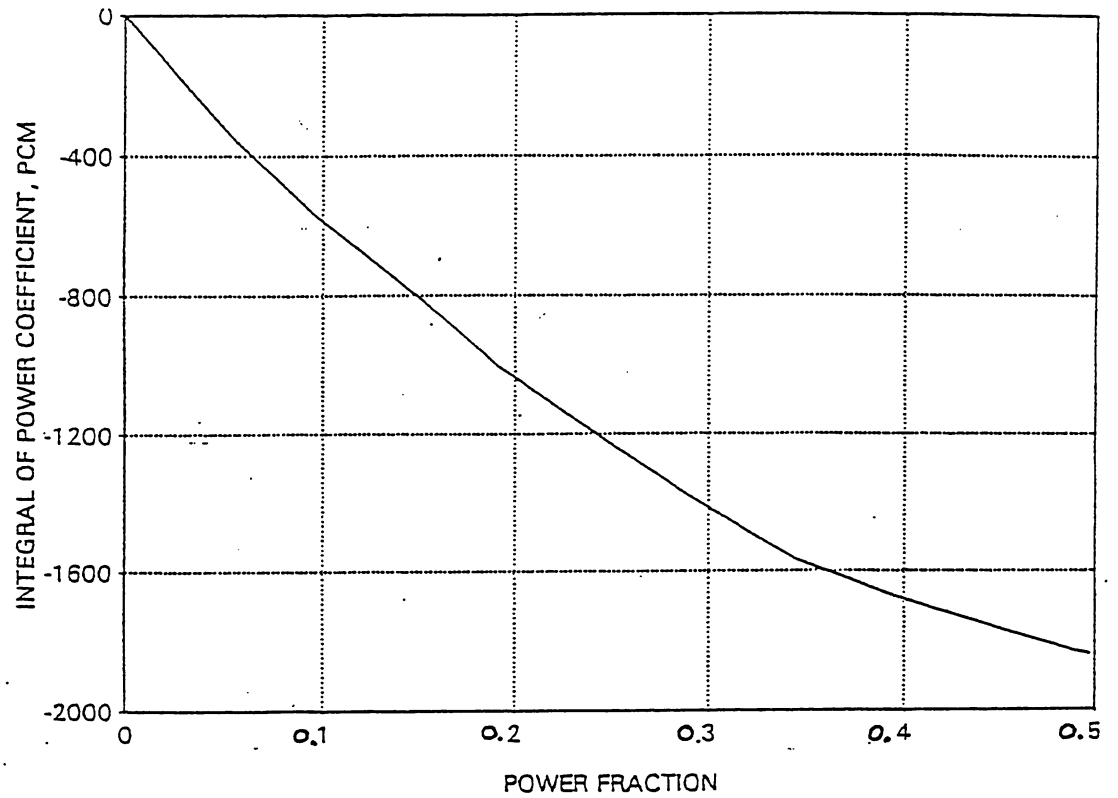


FIGURE 15.4.2-1

Variation of Reactivity with Power at Constant Core Average Temperature
Power Doppler Reactivity

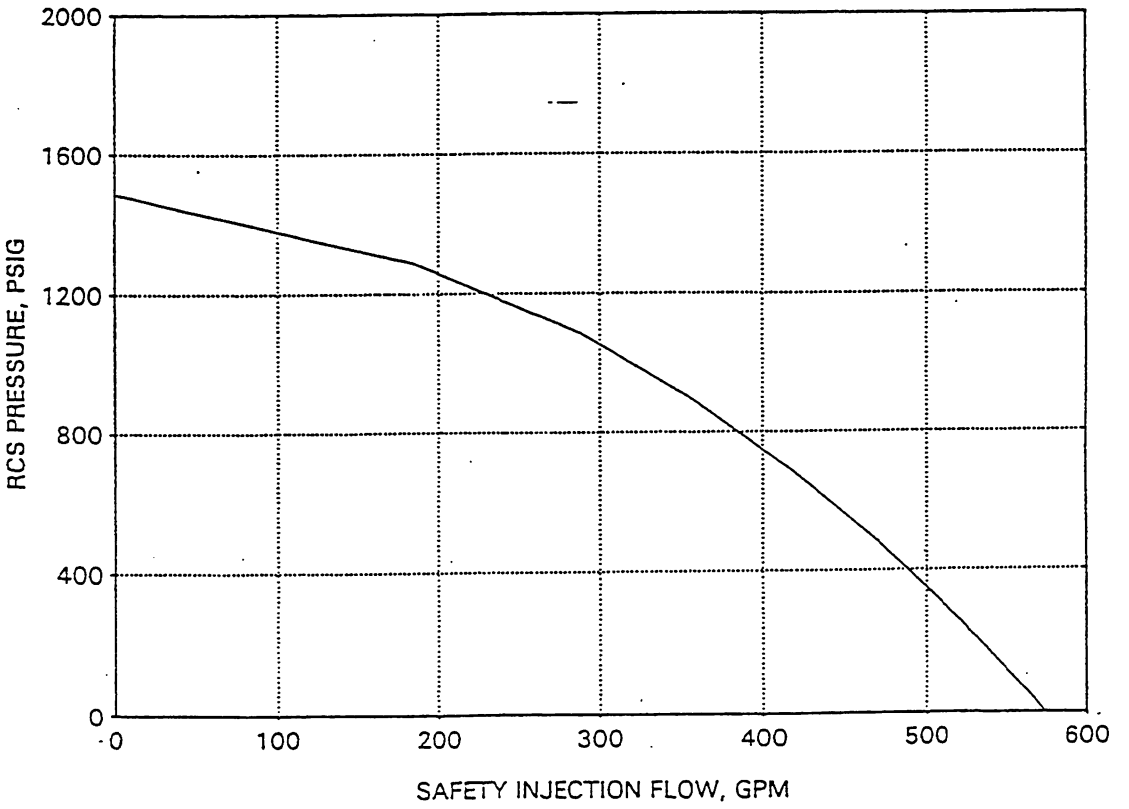


FIGURE 15.4.2-2

Safety Injection Curve
Minimum Safeguards

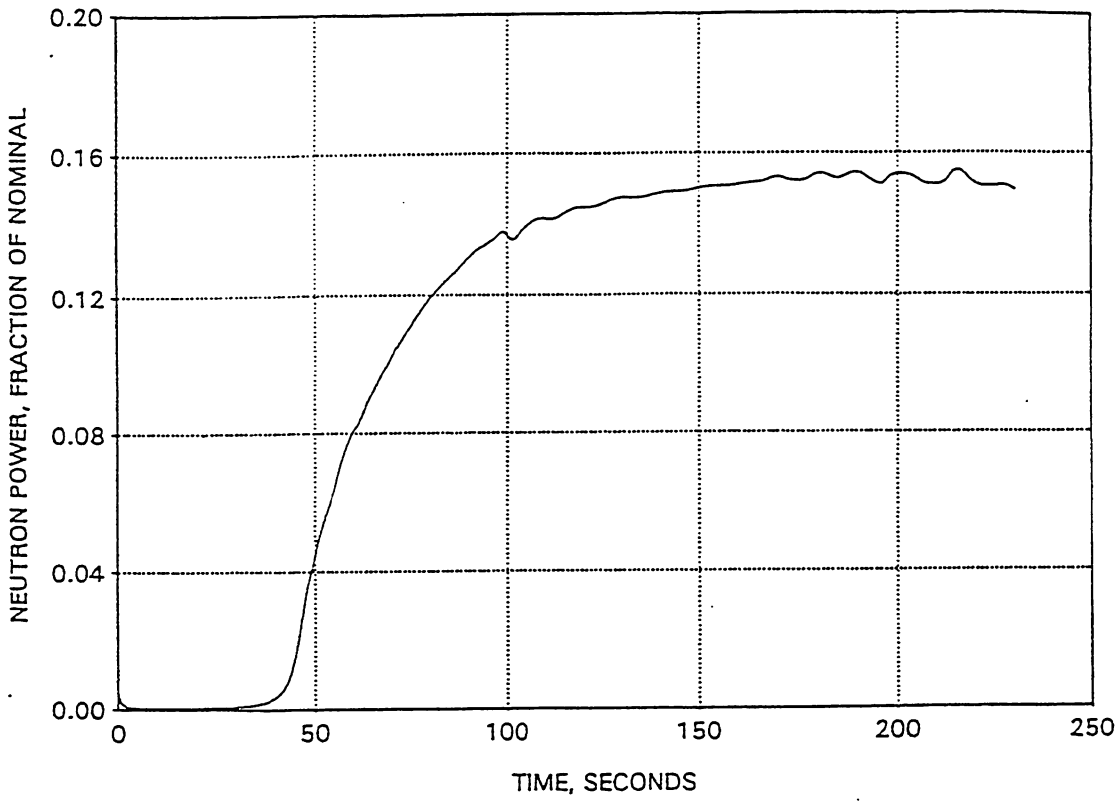


FIGURE 15.4.2-3

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
With Off-Site Power (Case A)
Neutron Power

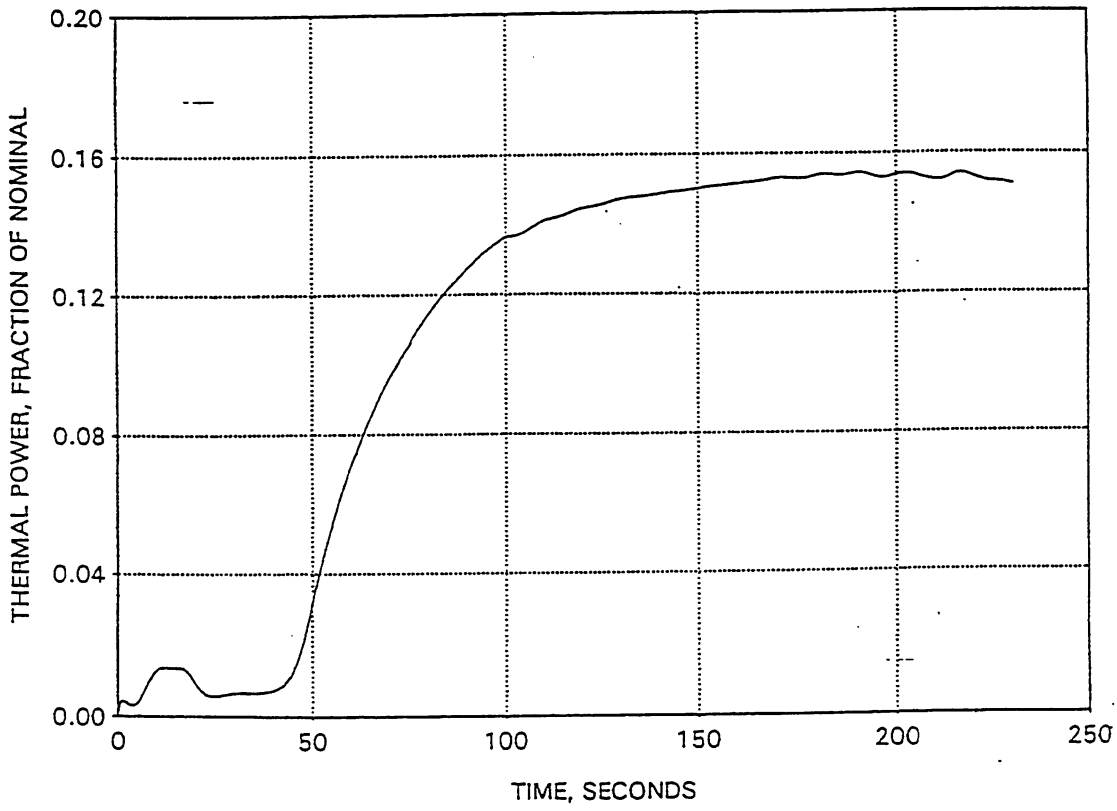


FIGURE 15.4.2-4

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
With Off-Site Power (Case A)
Thermal Power

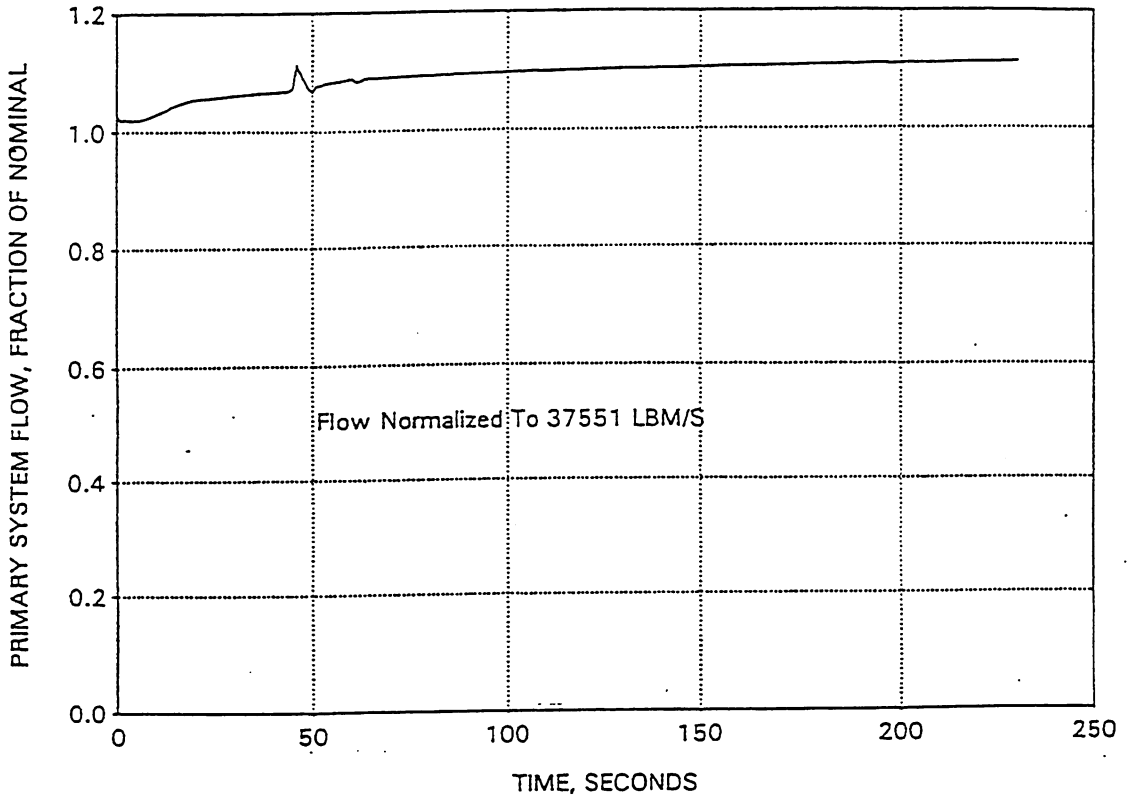


FIGURE 15.4.2-5

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
With Off-Site Power (Case A)
RCS Flow

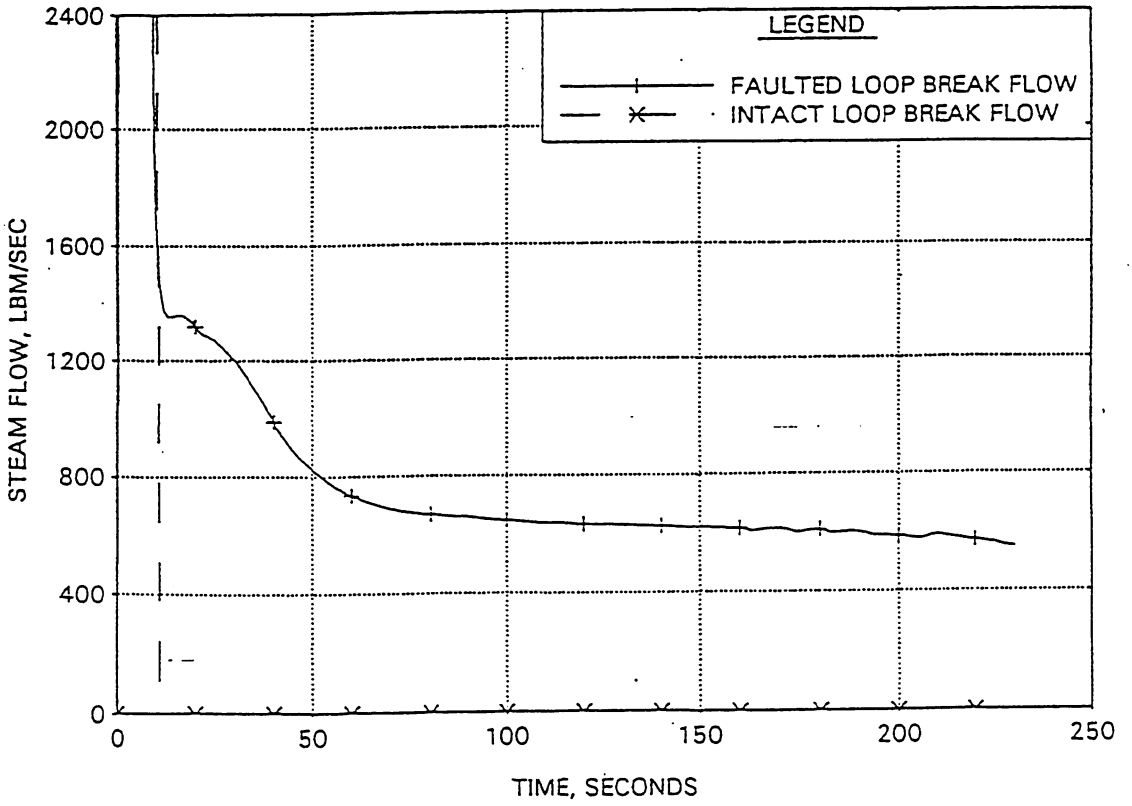


FIGURE 15.4.2-6

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
With Off-Site Power (Case A)
Steam Flow

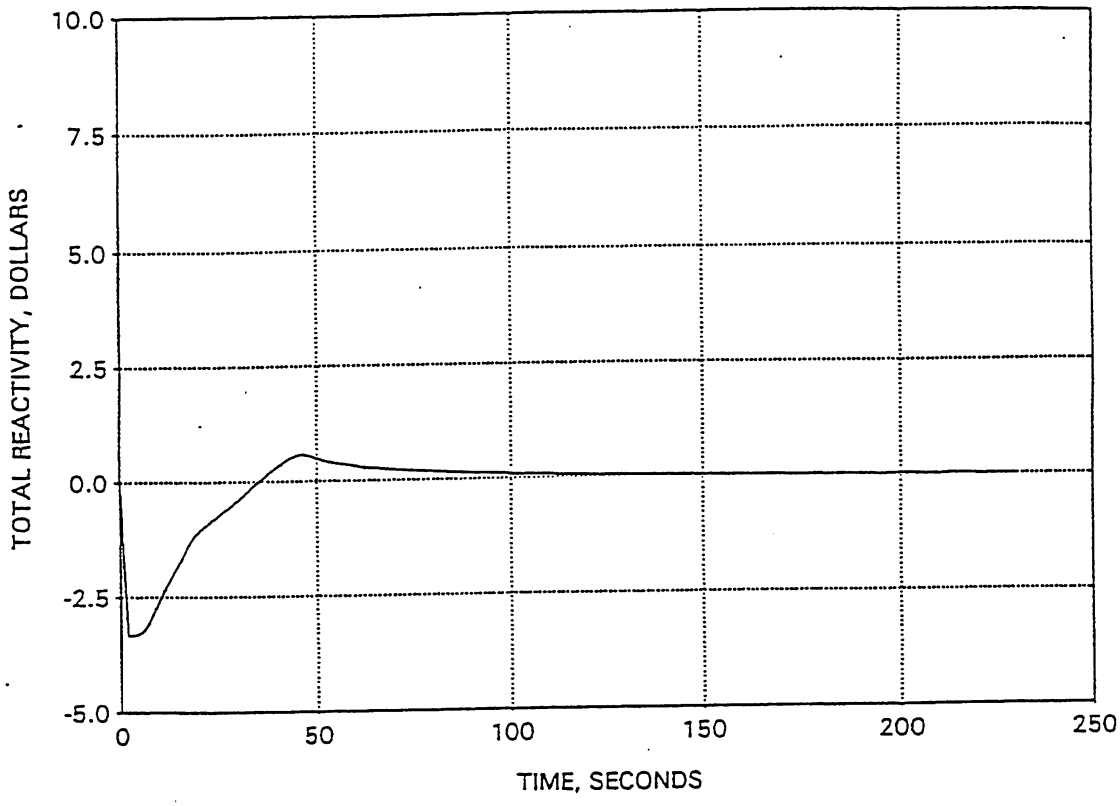


FIGURE 15.4.2-7

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle With Off-Site Power (Case A) Reactivity

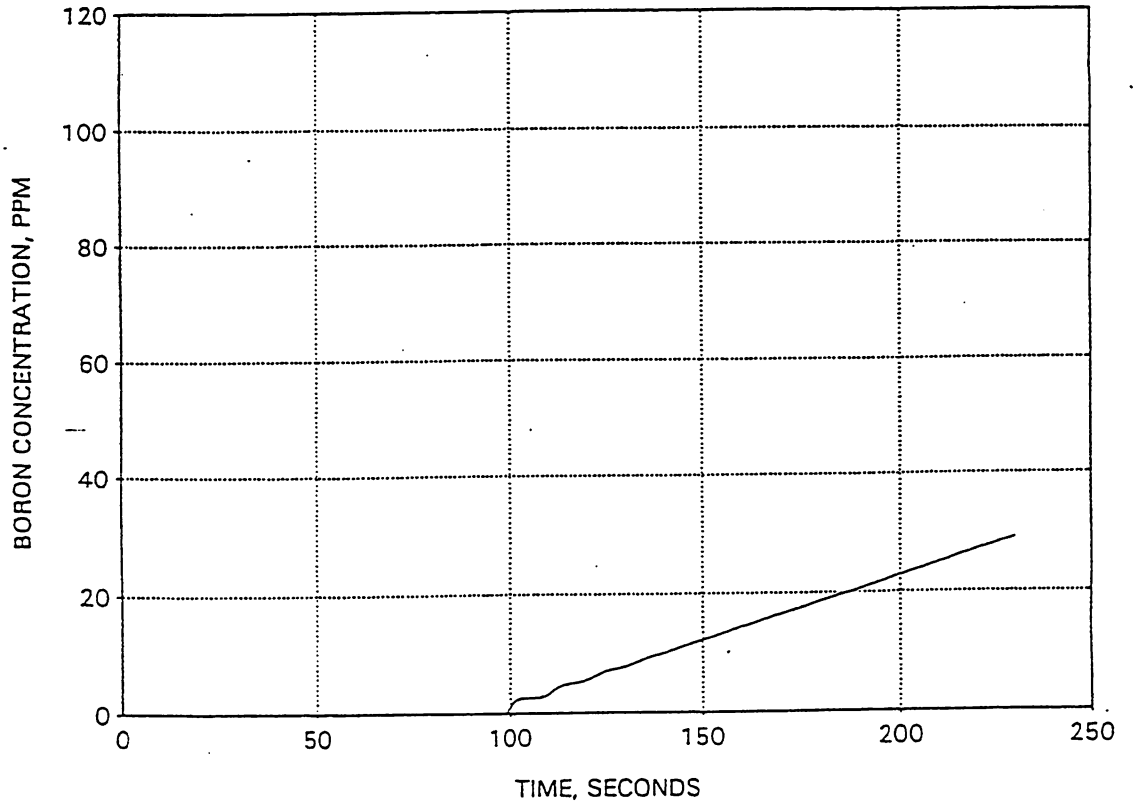


FIGURE 15.4.2-8

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
With Off-Site Power (Case A)
Core Boron

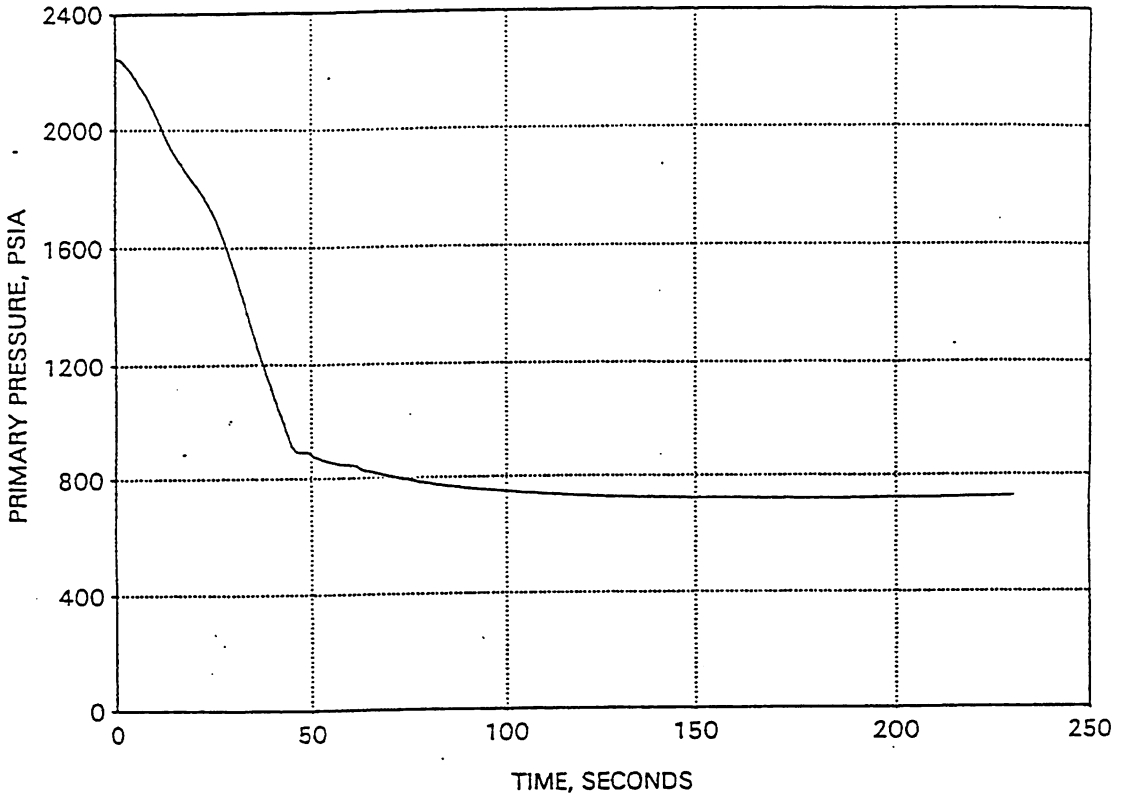


FIGURE 15.4.2-9

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
With Off-Site Power (Case A)
System Pressure

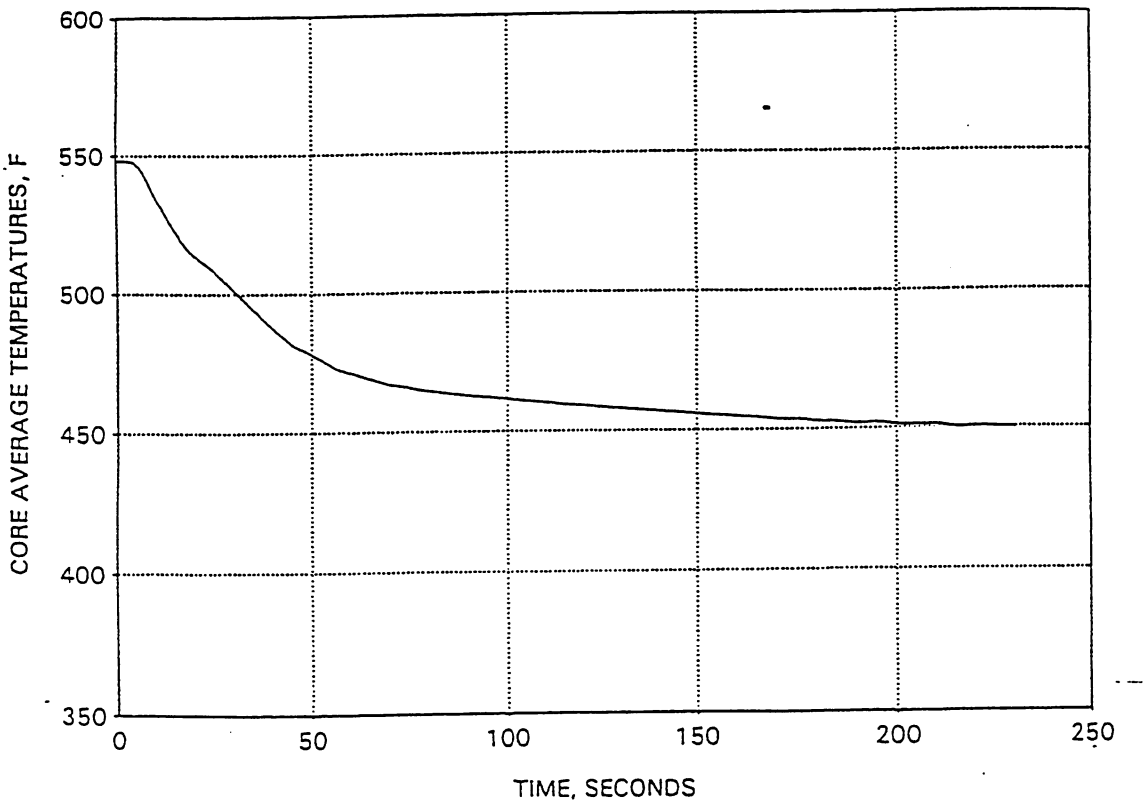


FIGURE 15.4.2-10

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
With Off-Site Power (Case A)
Core Average Temperature

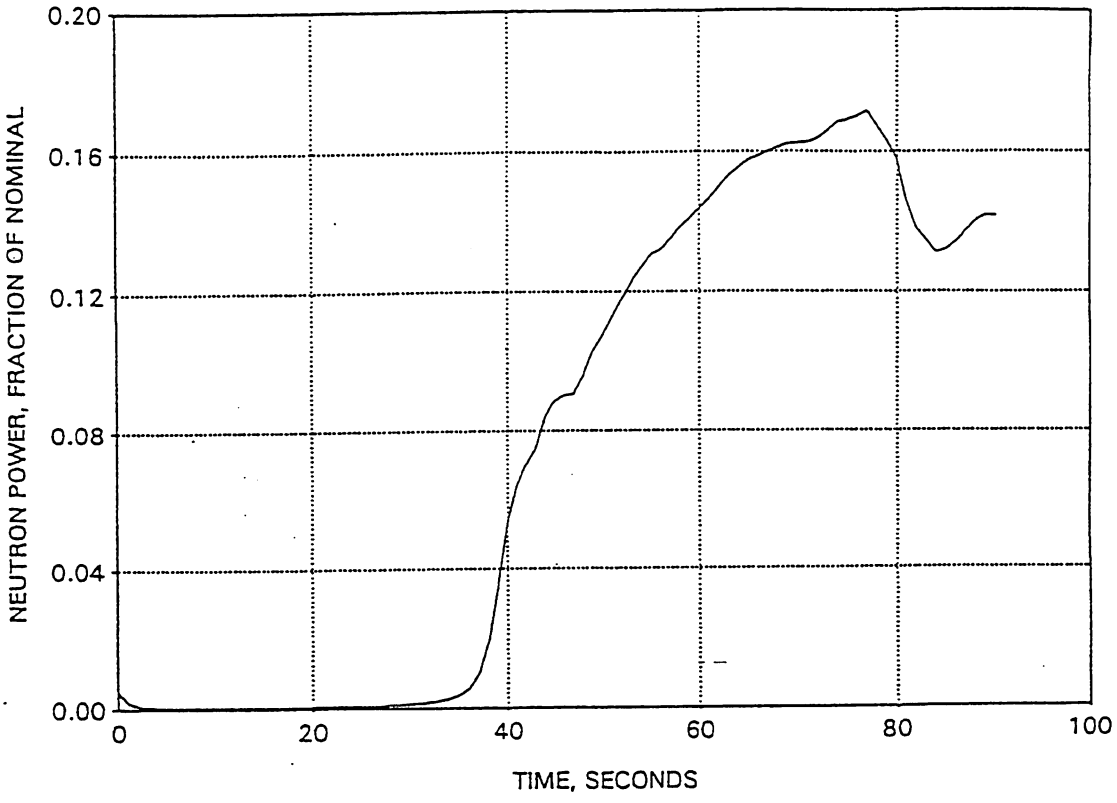


FIGURE 15.4.2-11

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle
With Off-Site Power (Case B)
Neutron Power

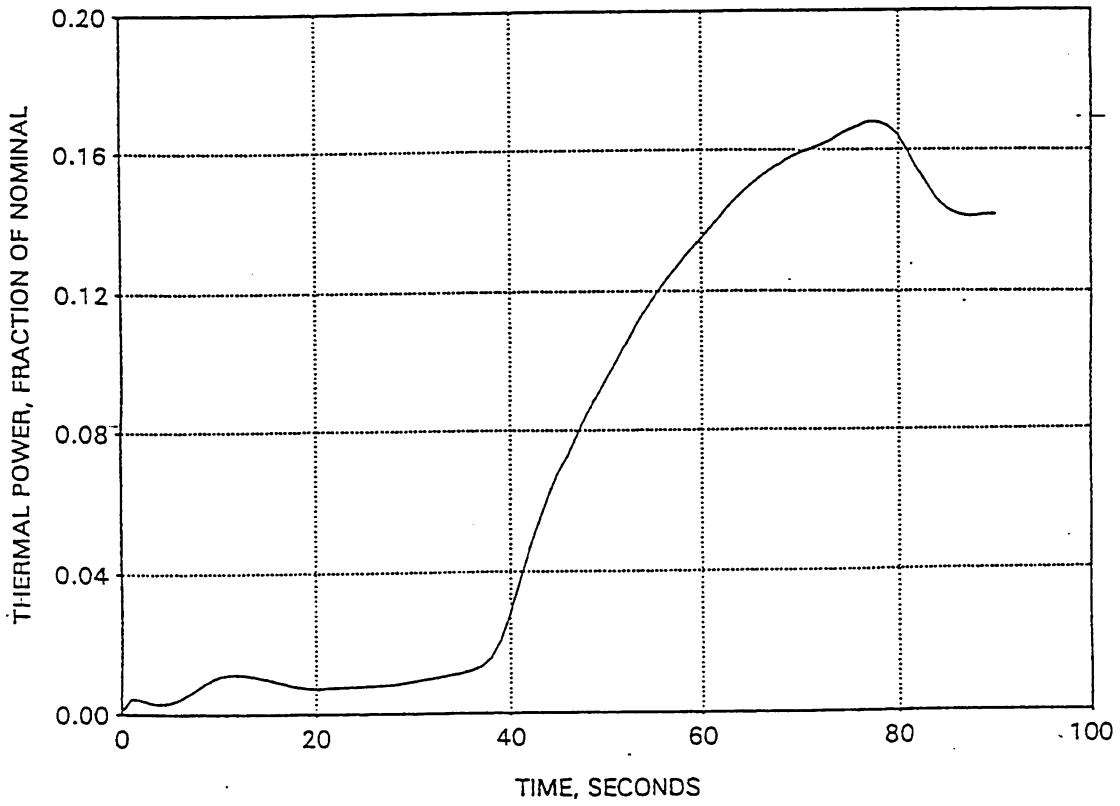


FIGURE 15.4.2-12

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle
With Off-Site Power (Case B)
Thermal Power

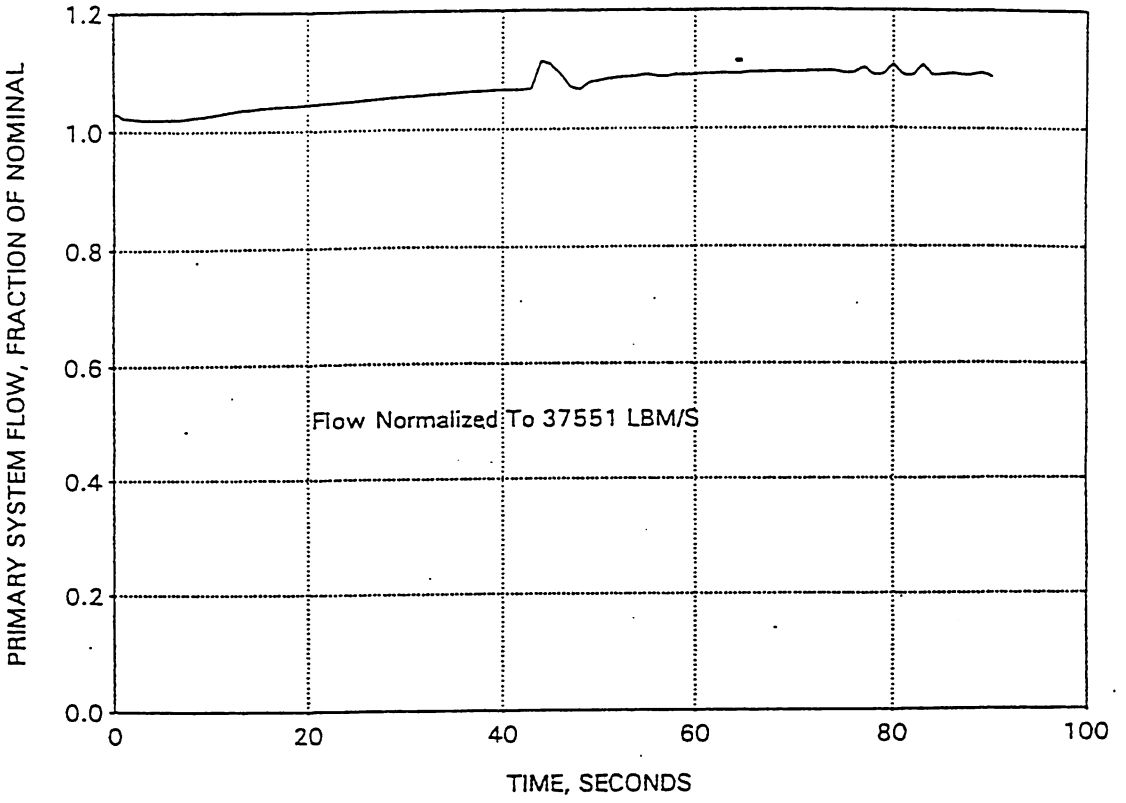


FIGURE 15.4.2-13

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle
With Off-Site Power (Case B)
RCS Flow

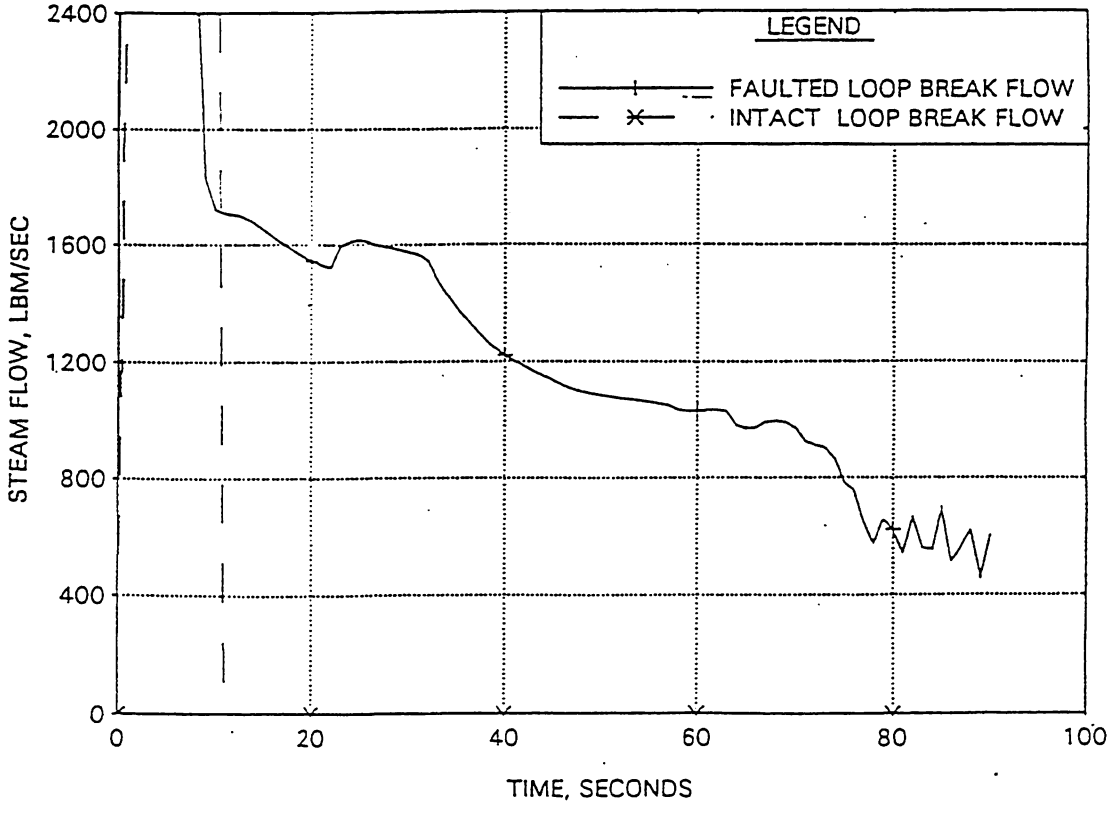


FIGURE 15.4.2-14

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle
With Off-Site Power (Case B)
Steam Flow

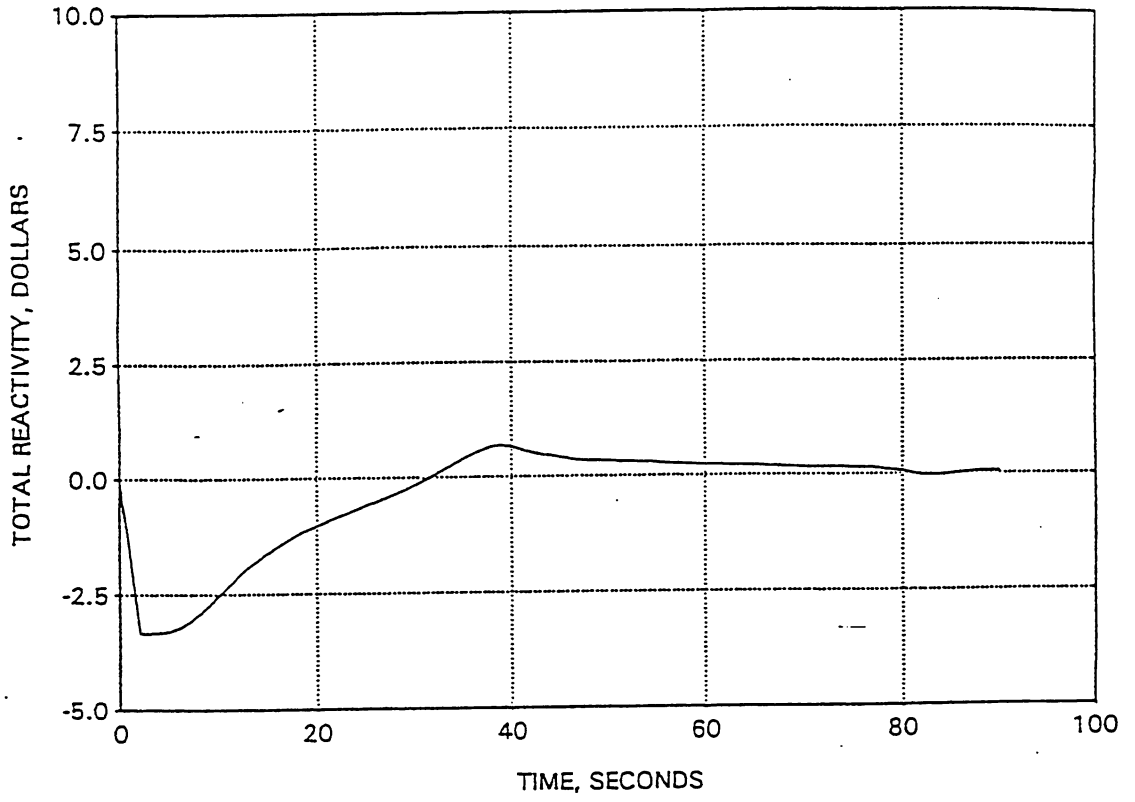


FIGURE 15.4.2-15

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle With Off-Site Power (Case B) Reactivity

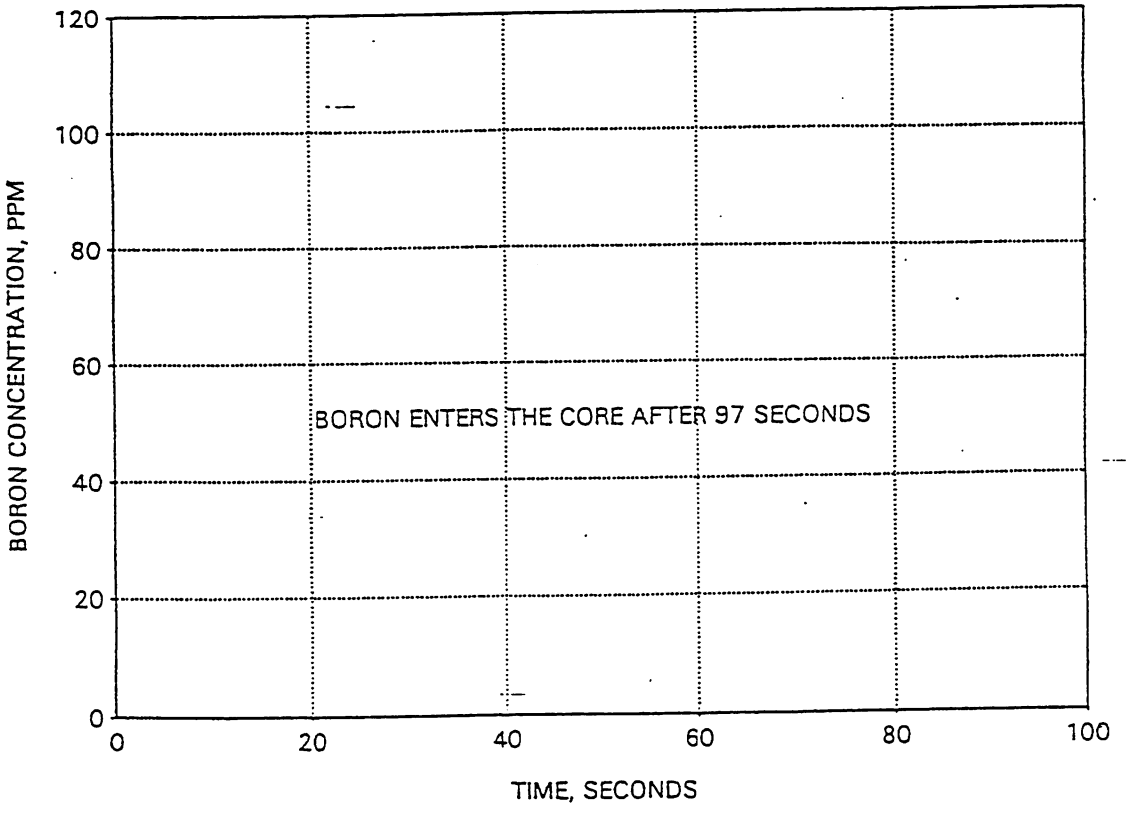


FIGURE 15.4.2-16

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle
With Off-Site Power (Case B)
Core Boron

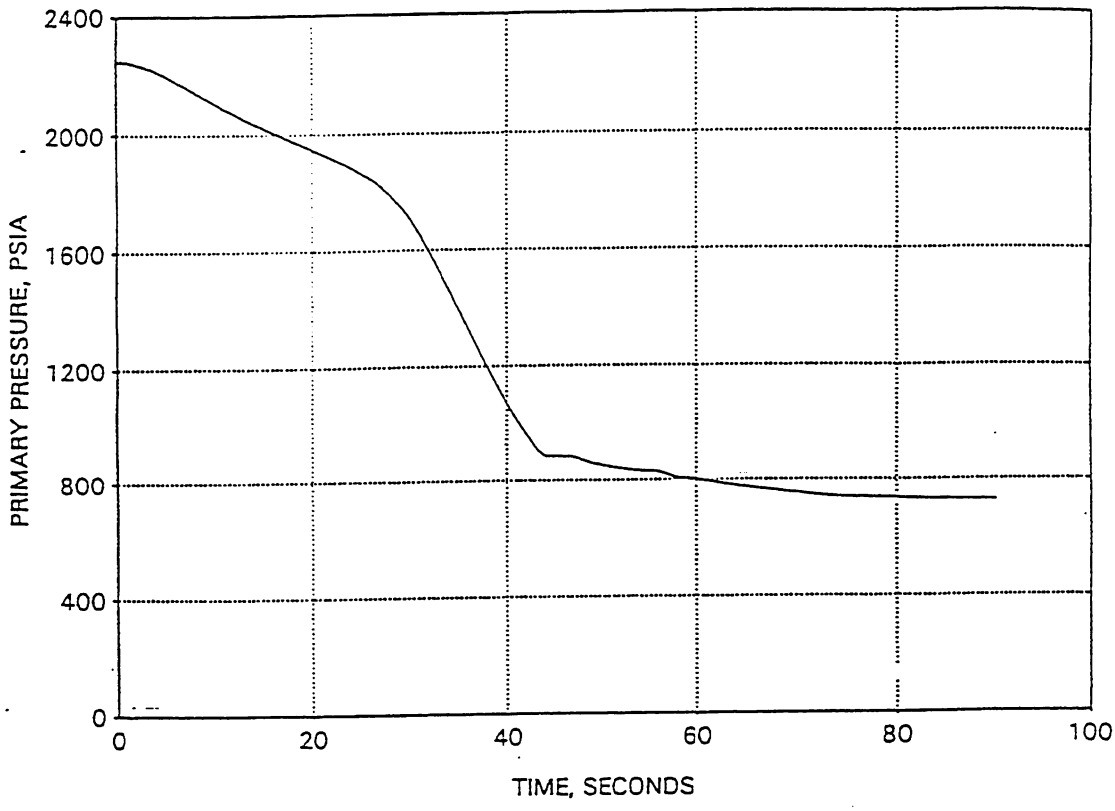


FIGURE 15.4.2-17

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle
With Off-Site Power (Case B)
System Pressure

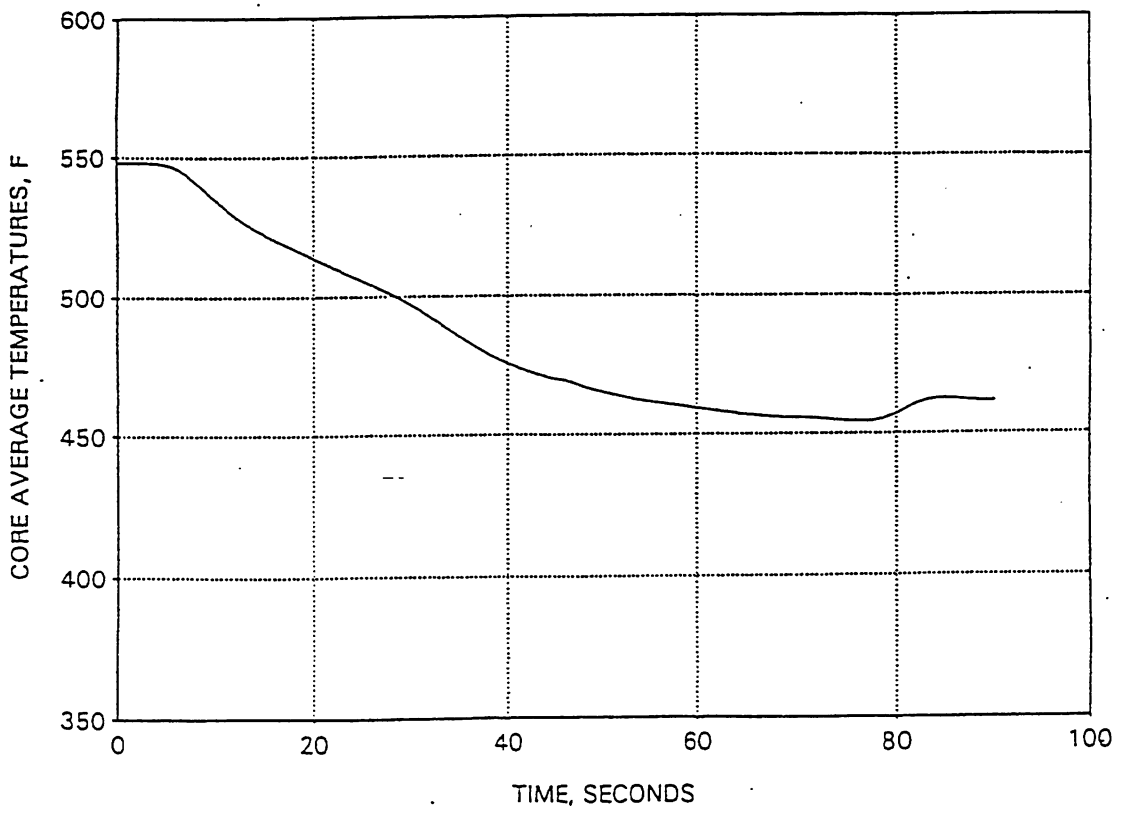


FIGURE 15.4.2-18

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle
With Off-Site Power (Case B)
Core Average Temperature

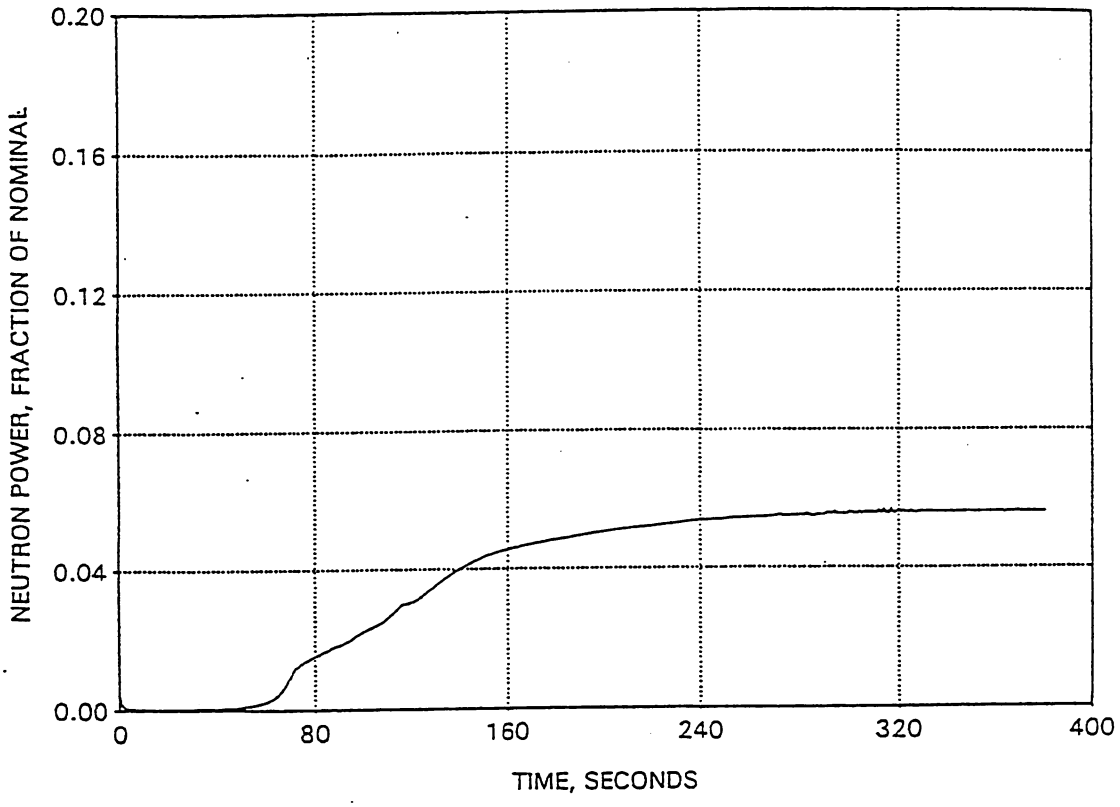


FIGURE 15.4.2-19

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle
Without Off-Site Power (Case C)
Neutron Power

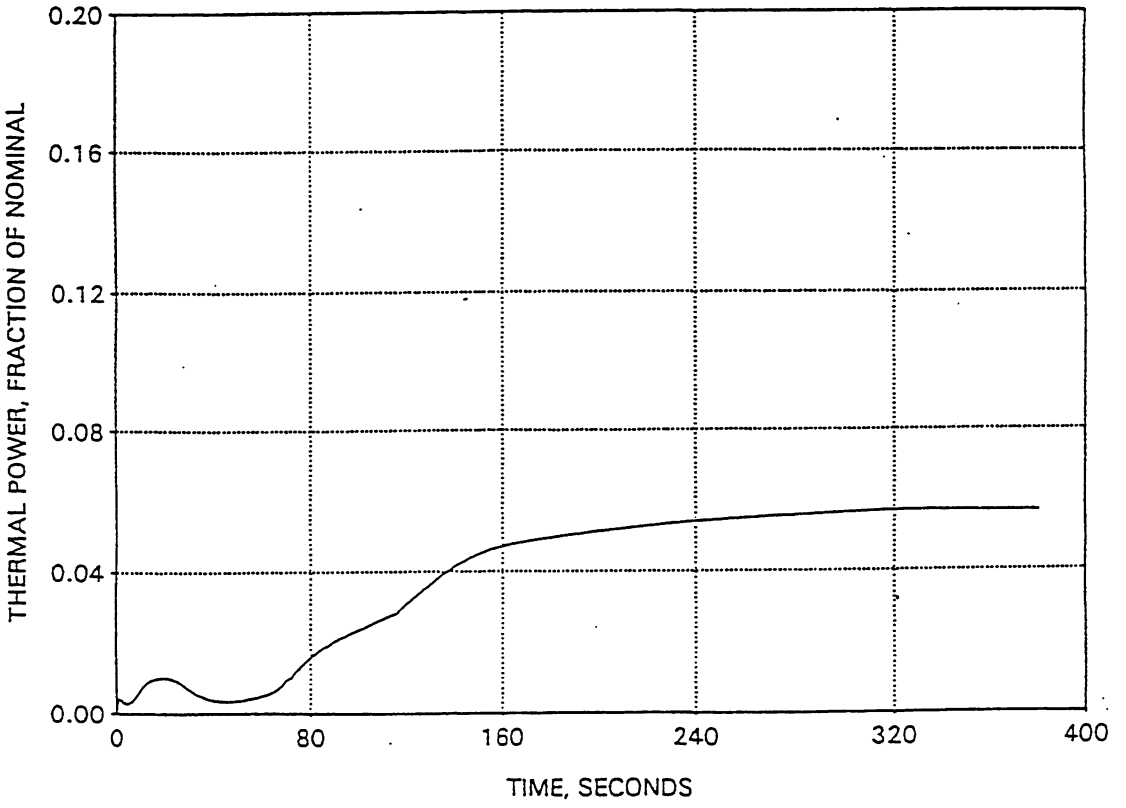


FIGURE 15.4.2-20

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle Without Off-Site Power (Case C) Thermal Power

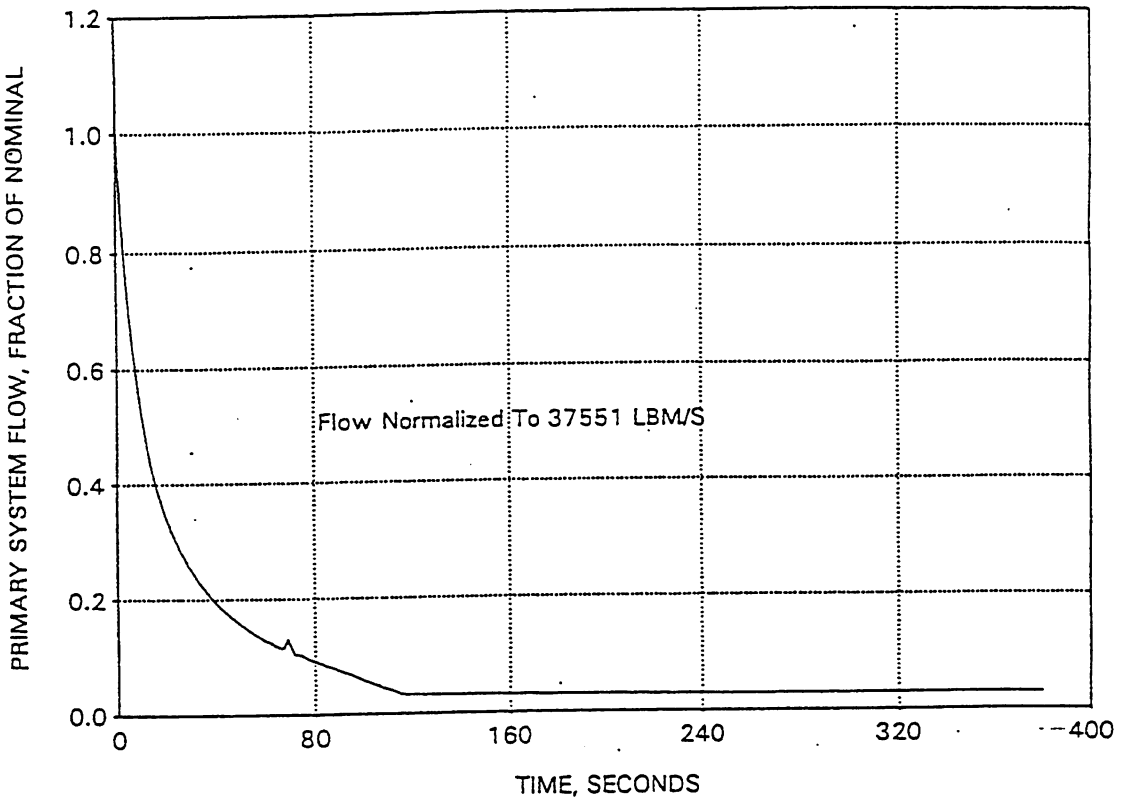


FIGURE 15.4.2-21

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle Without Off-Site Power (Case C) RCS Flow

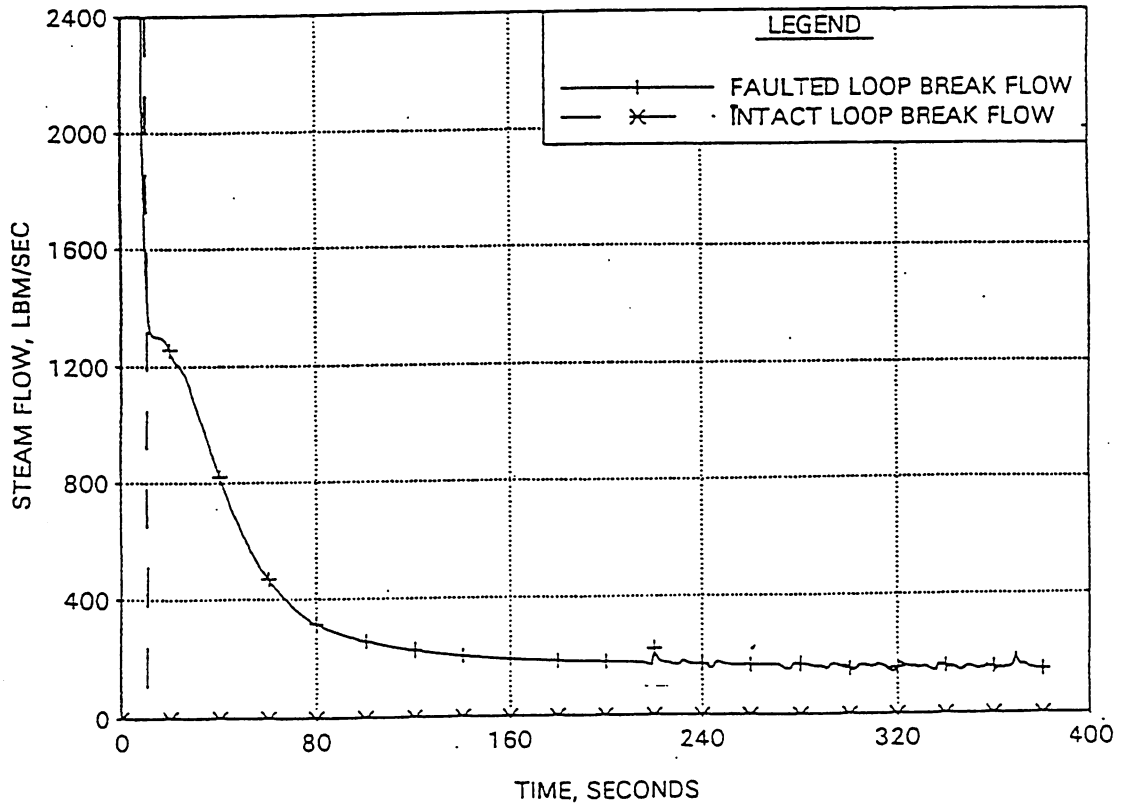


FIGURE 15.4.2-22

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle Without Off-Site Power (Case C) Steam Flow

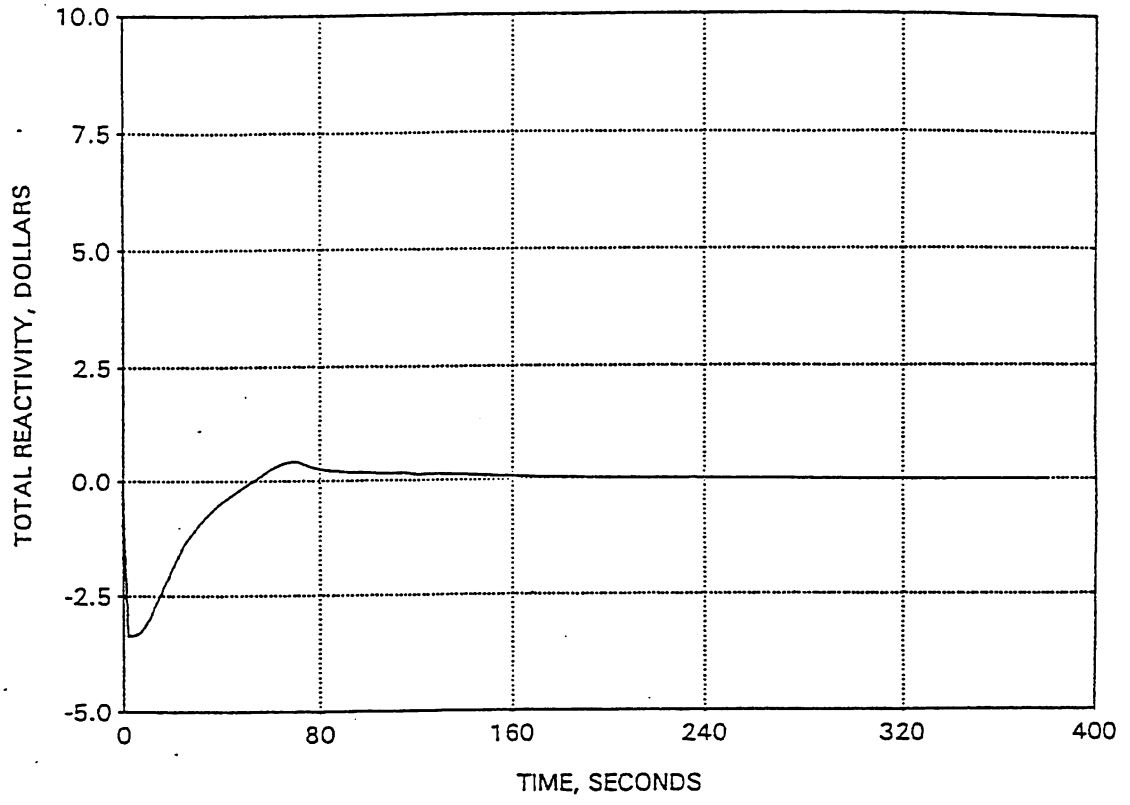


FIGURE 15.4.2-23

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle Without Off-Site Power (Case C) Reactivity

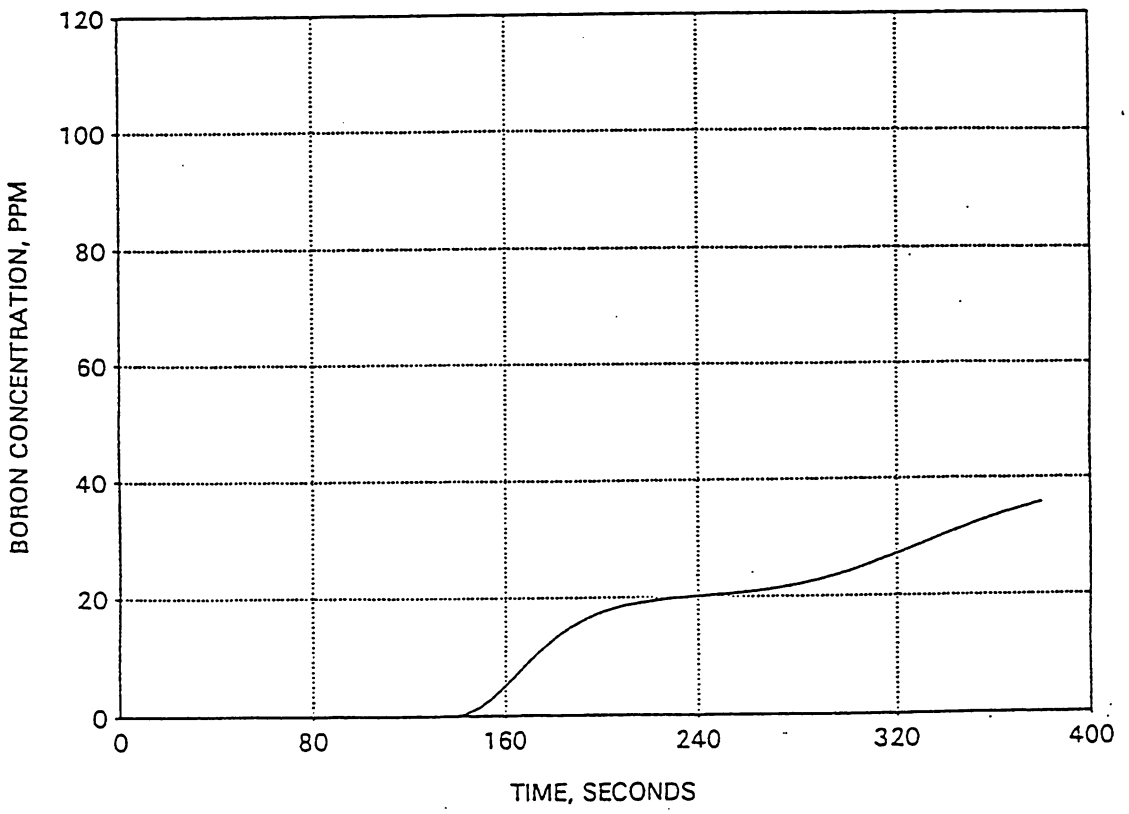


FIGURE 15.4.2-24

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle Without Off-Site Power (Case C) Core Boron

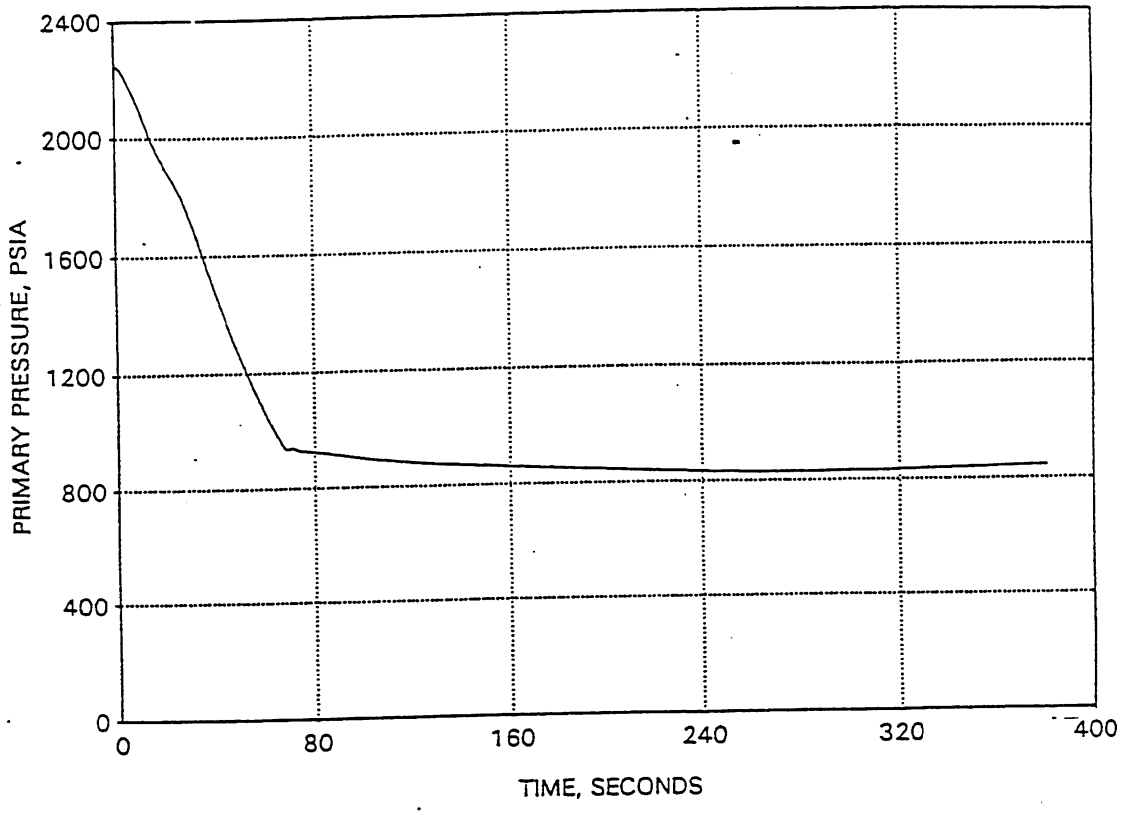


FIGURE 15.4.2-25

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle Without Off-Site Power (Case C) System Pressure

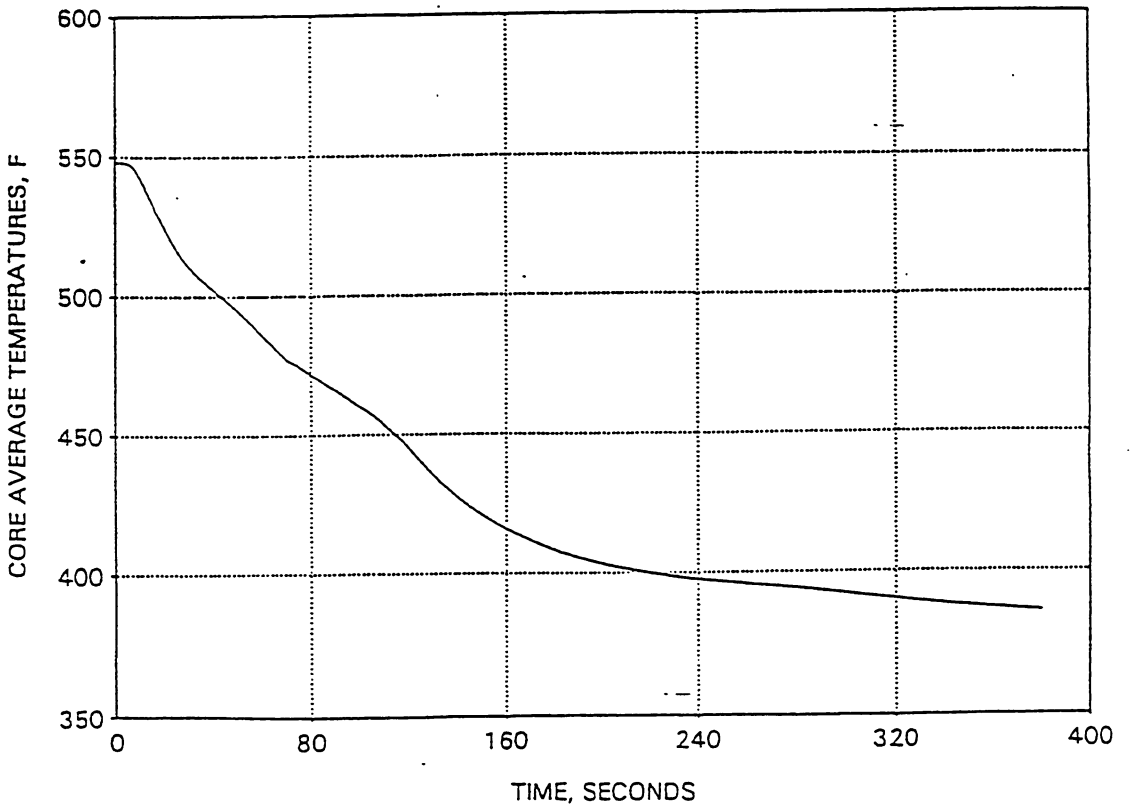


FIGURE 15.4.2-26

Sequoyah Steam Line Break Downstream of Flow Measuring Nozzle Without Off-Site Power (Case C) Core Average Temperature

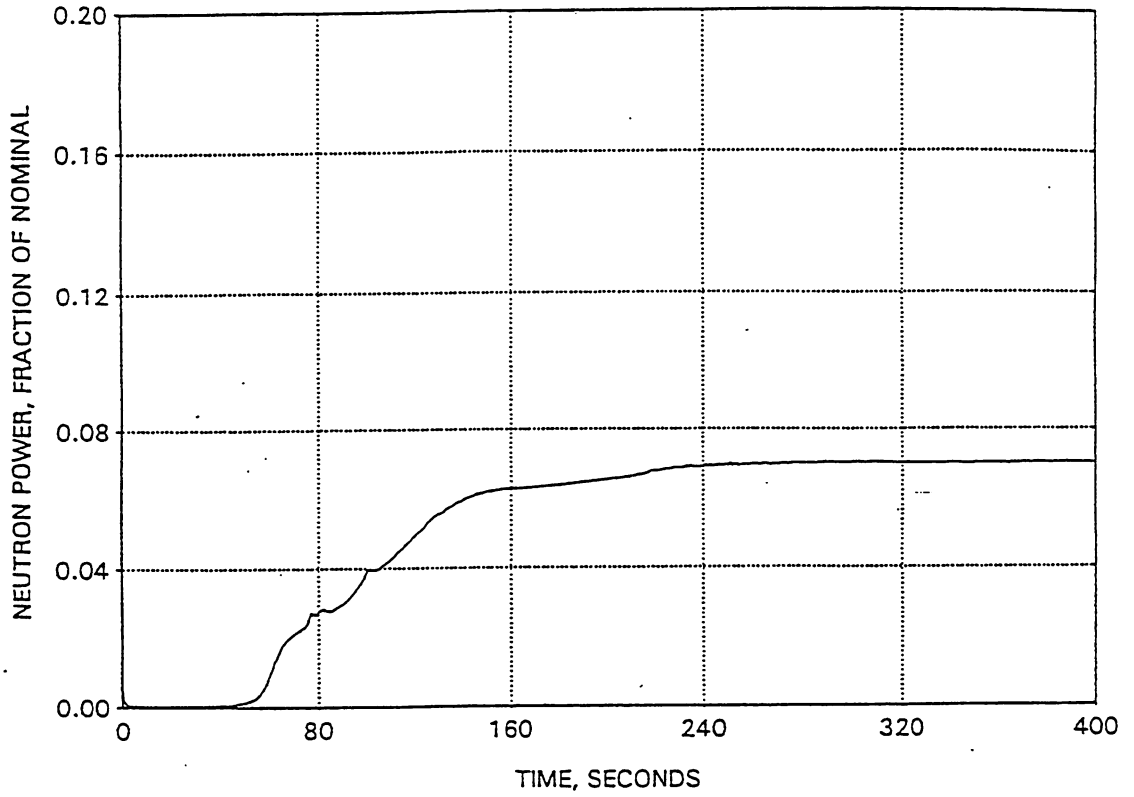


FIGURE 15.4.2-27

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) Neutron Power

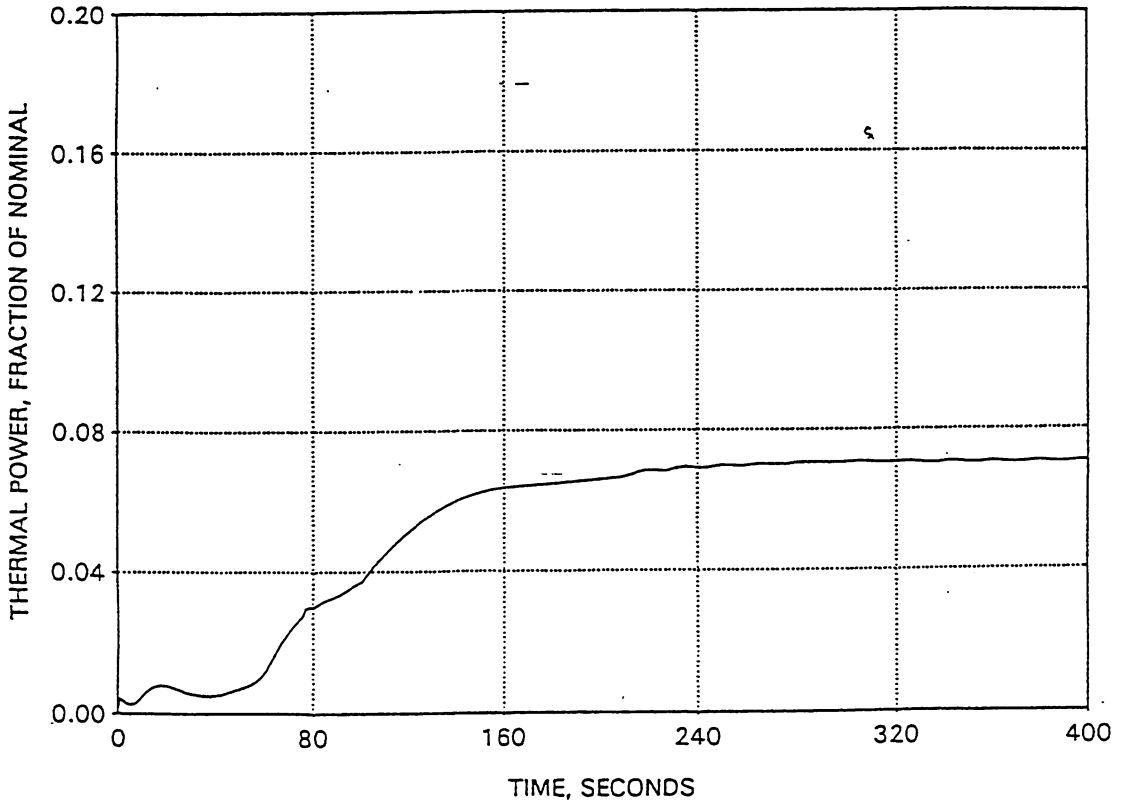


FIGURE 15.4.2-28

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) Thermal Power

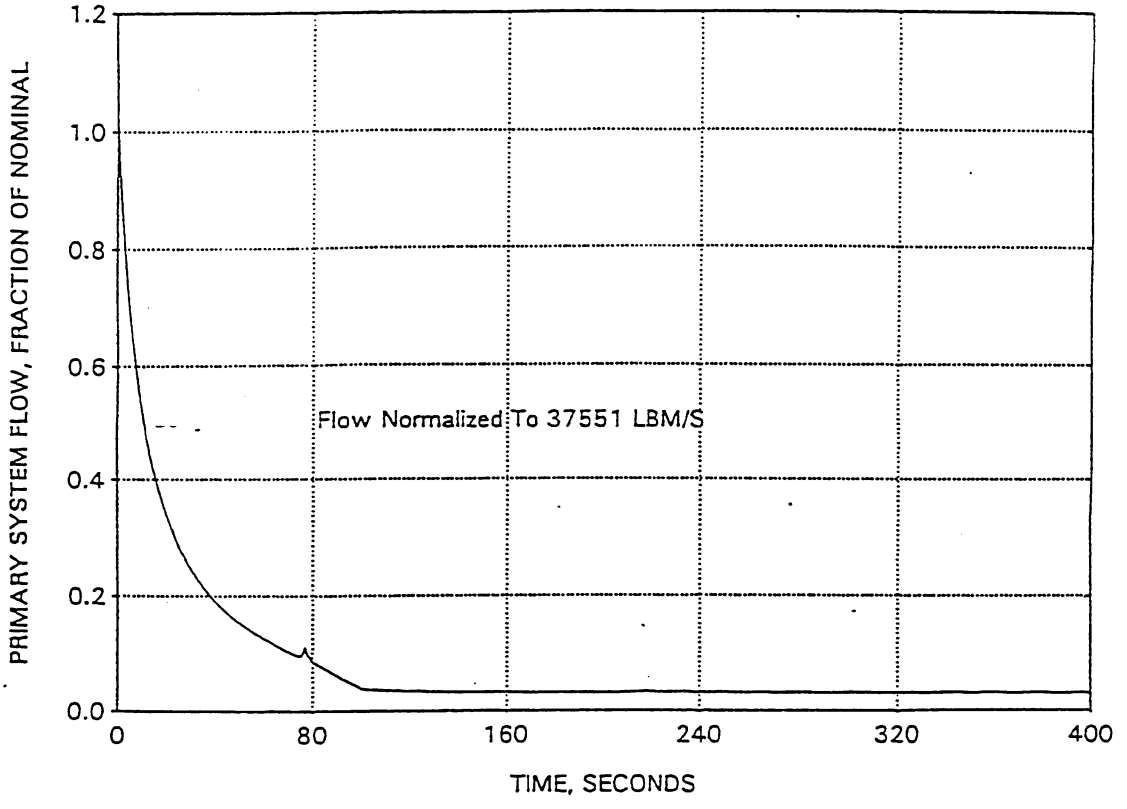


FIGURE 15.4.2-29

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) RCS Flow

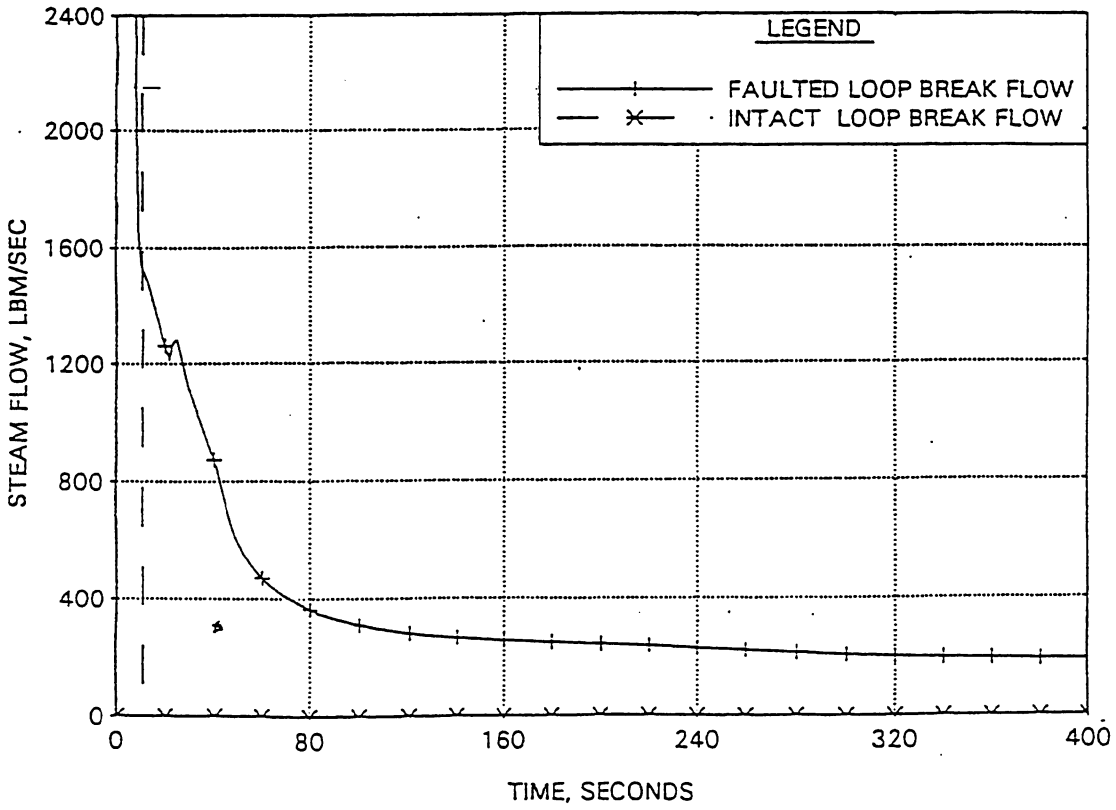


FIGURE 15.4.2-30

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) Steam Flow

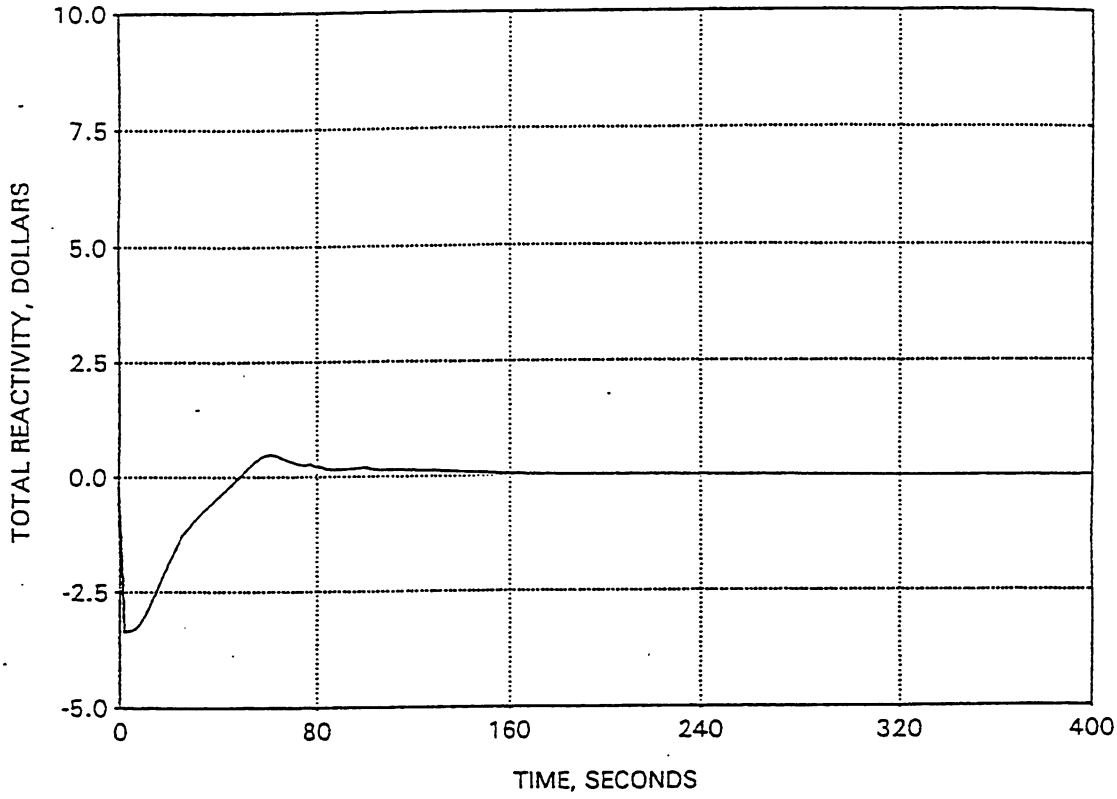


FIGURE 15.4.2-31

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) Reactivity

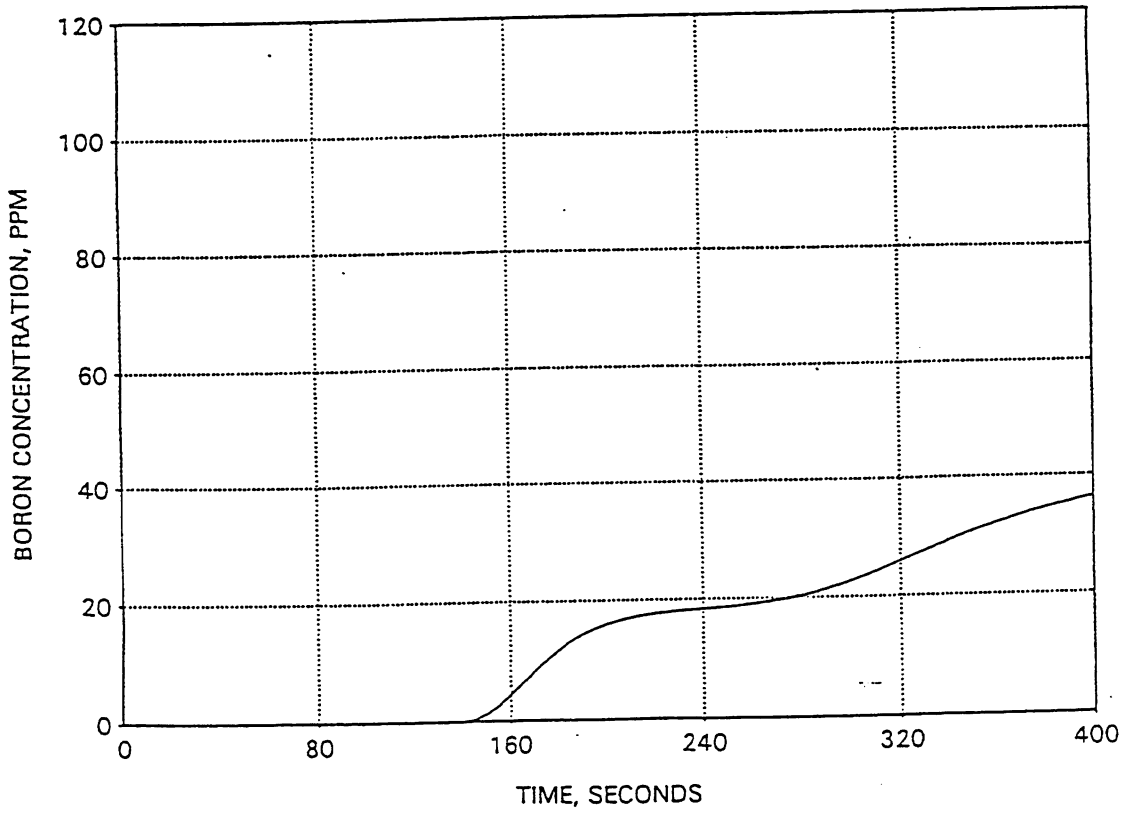


FIGURE 15.4.2-32

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) Core Boron

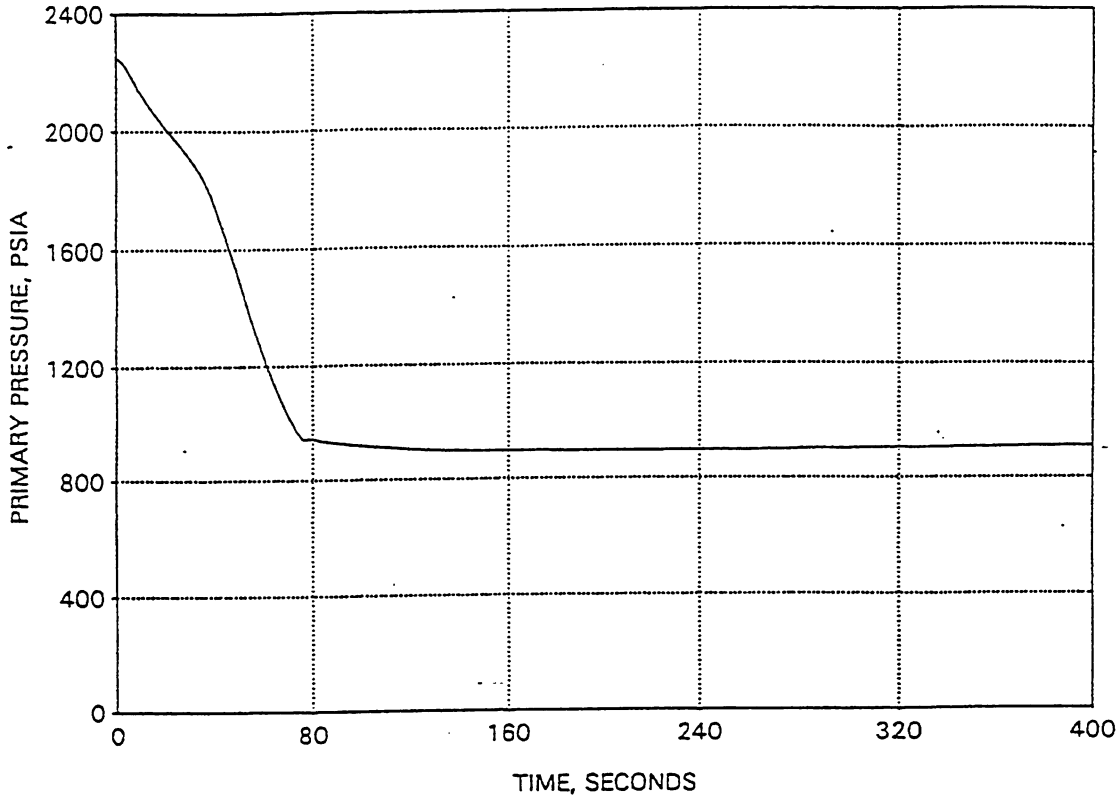


FIGURE 15.4.2-33

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) System Pressure

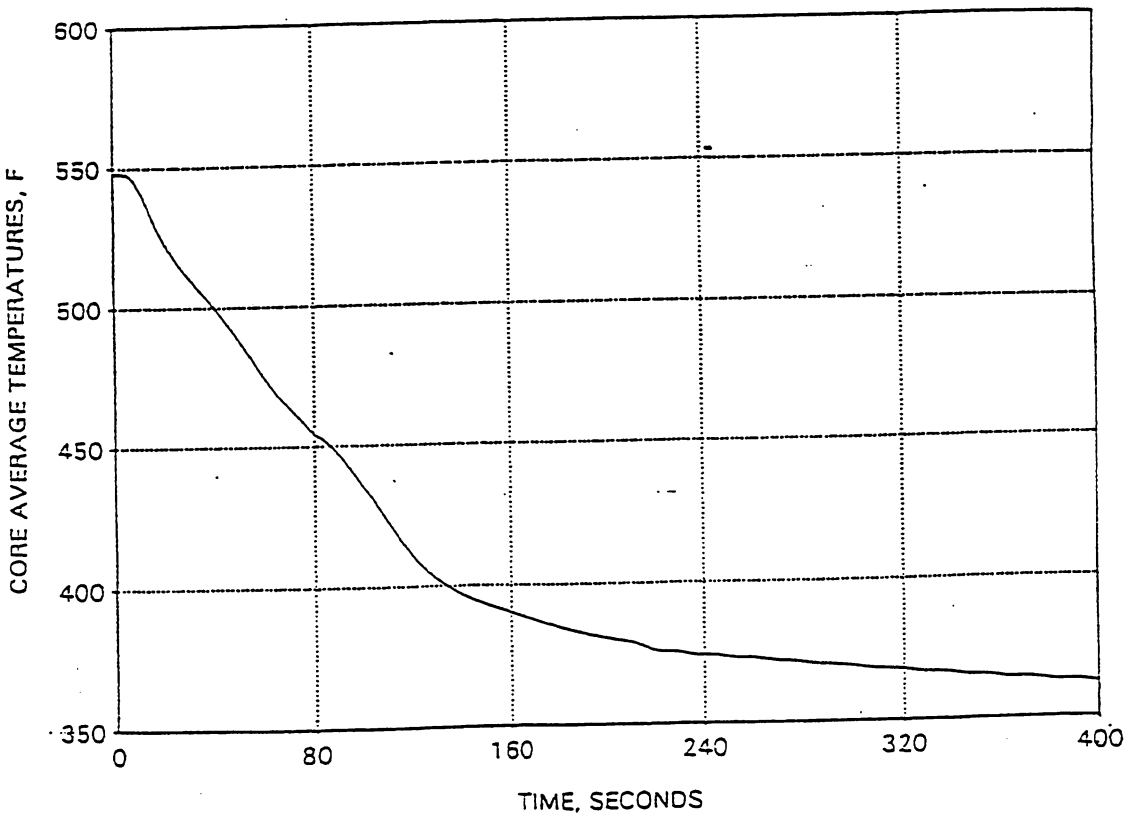


FIGURE 15.4.2-34

Sequoyah Steam Line Break Upstream of Flow Measuring Nozzle Without Off-Site Power (Case D) Core Average Temperature

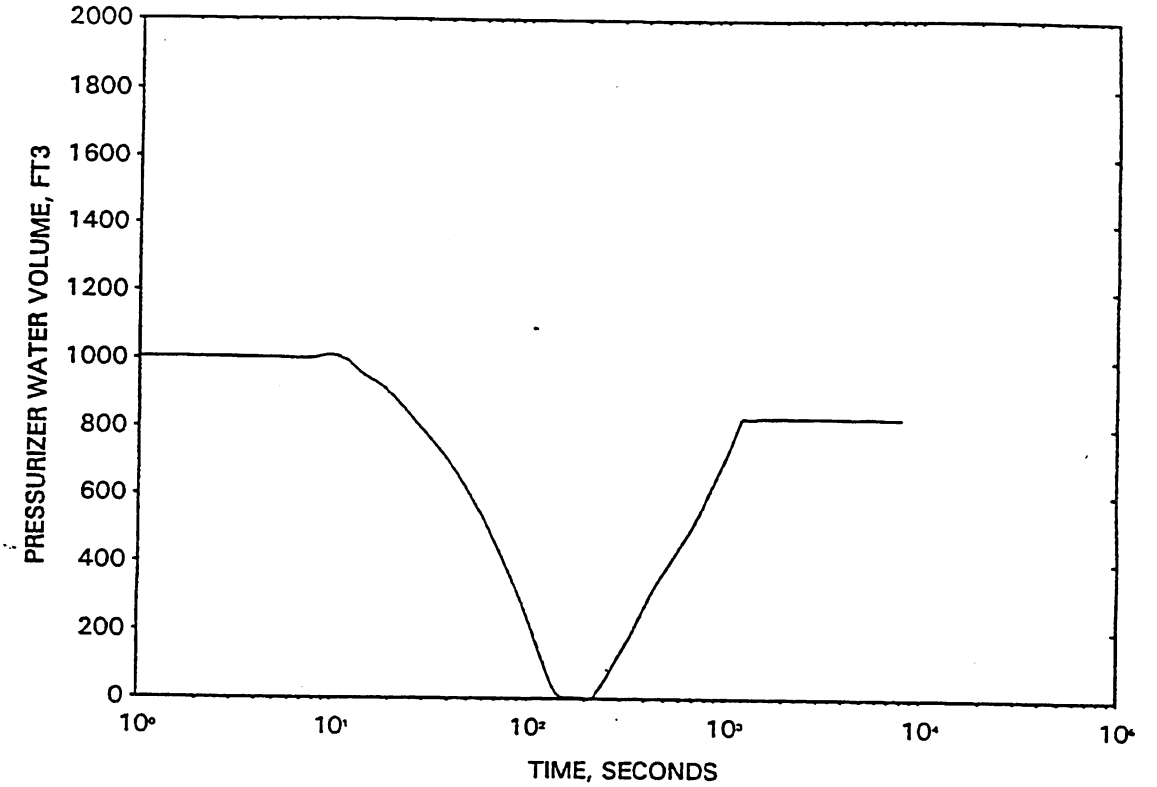


FIGURE 15.4.2-35a MAJOR RUPTURE OF A MFW PIPE - NO LOOP
PRESSUREIZER WATER VOLUME

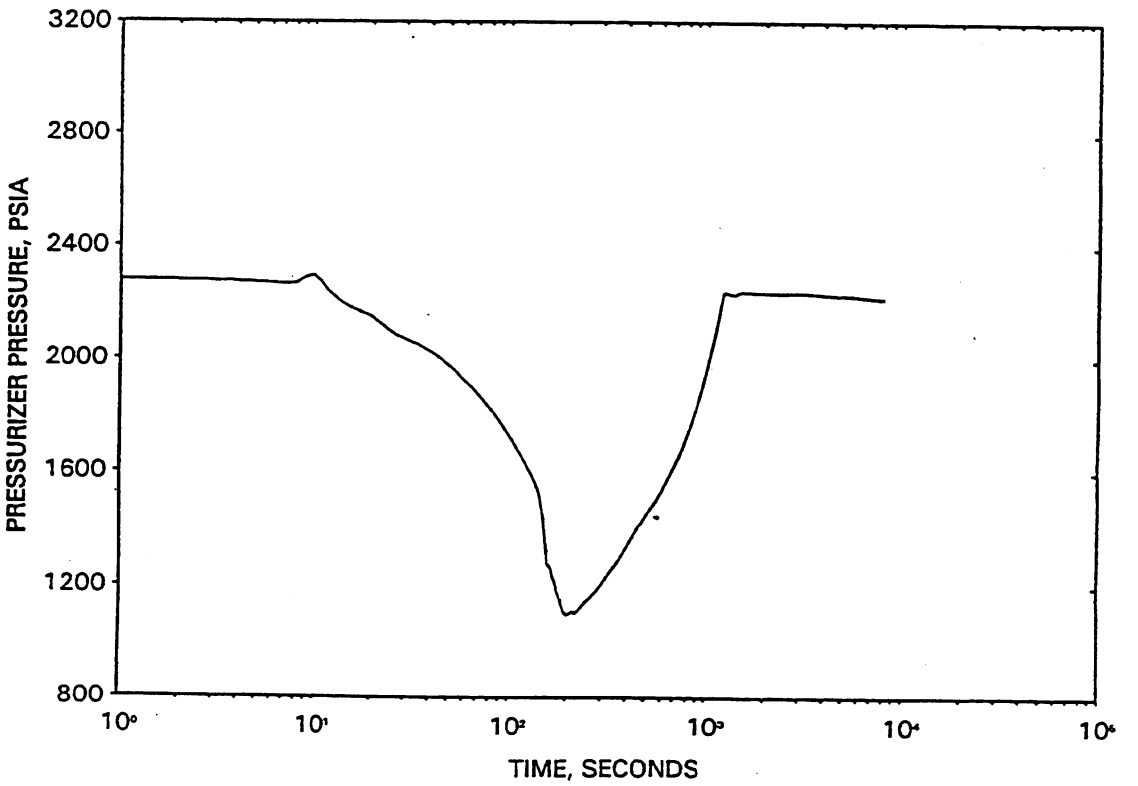


FIGURE 15.4.2-35b MAJOR RUPTURE OF A MFW PIPE - NO LOOP
PRESSURIZER PRESSURE

Revised by Amendment 15

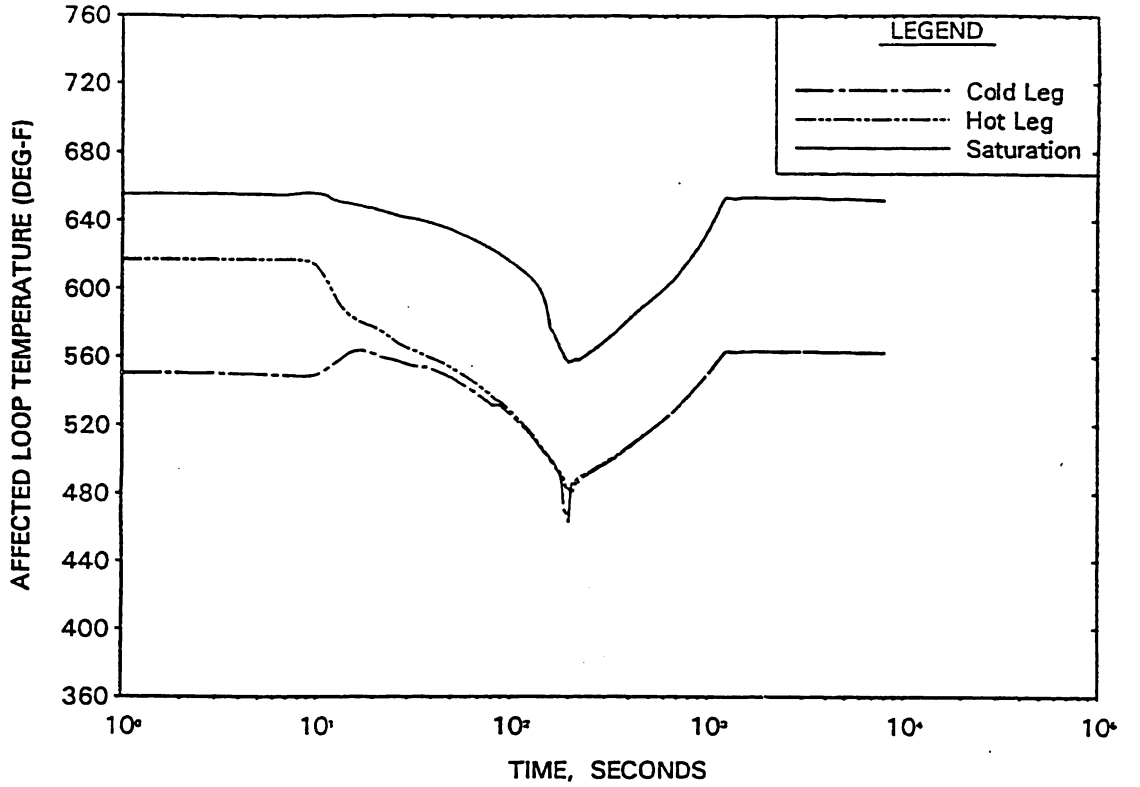


FIGURE 15.4.2-36a MAJOR RUPTURE OF A MFW PIPE - NO LOOP
LOOP 1 COLD LEG, HOT LEG AND SATURATION TEMPERATURES

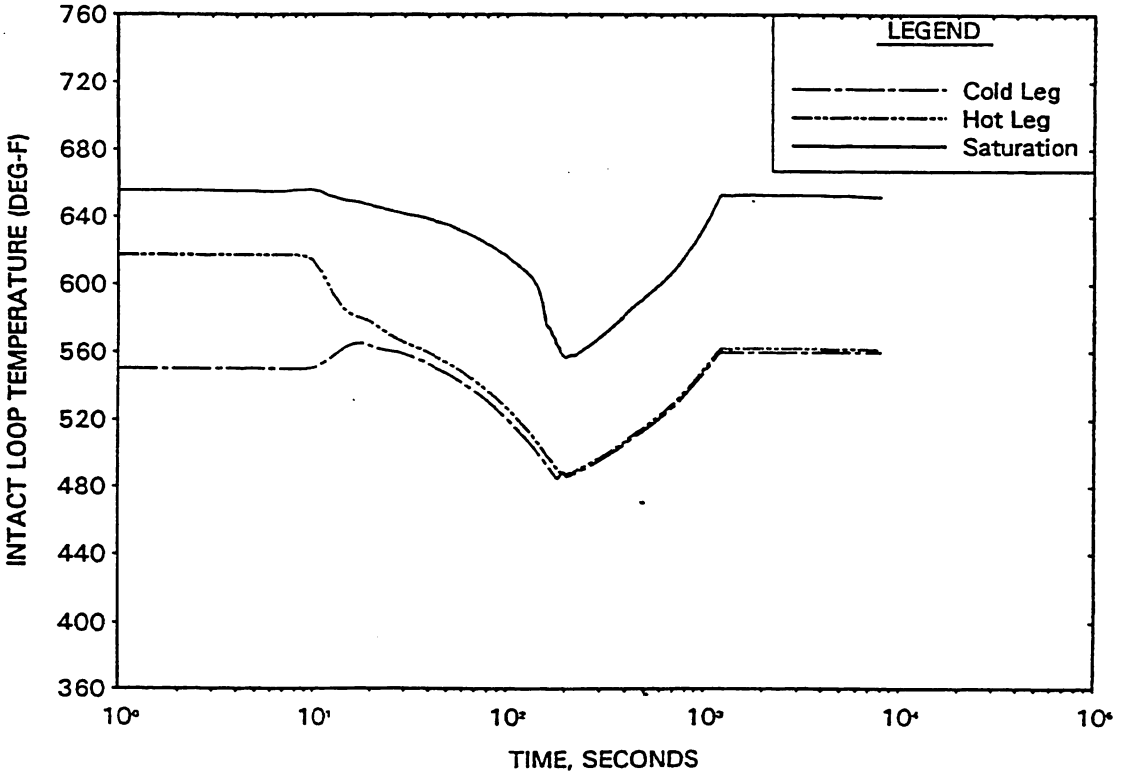


FIGURE 15.4.2-36b MAJOR RUPTURE OF A MFW PIPE - NO LOOP
INTACT LOOP COLD LEG, HOT LEG AND SATURATION TEMPERATURES

Revised by Amendment 15

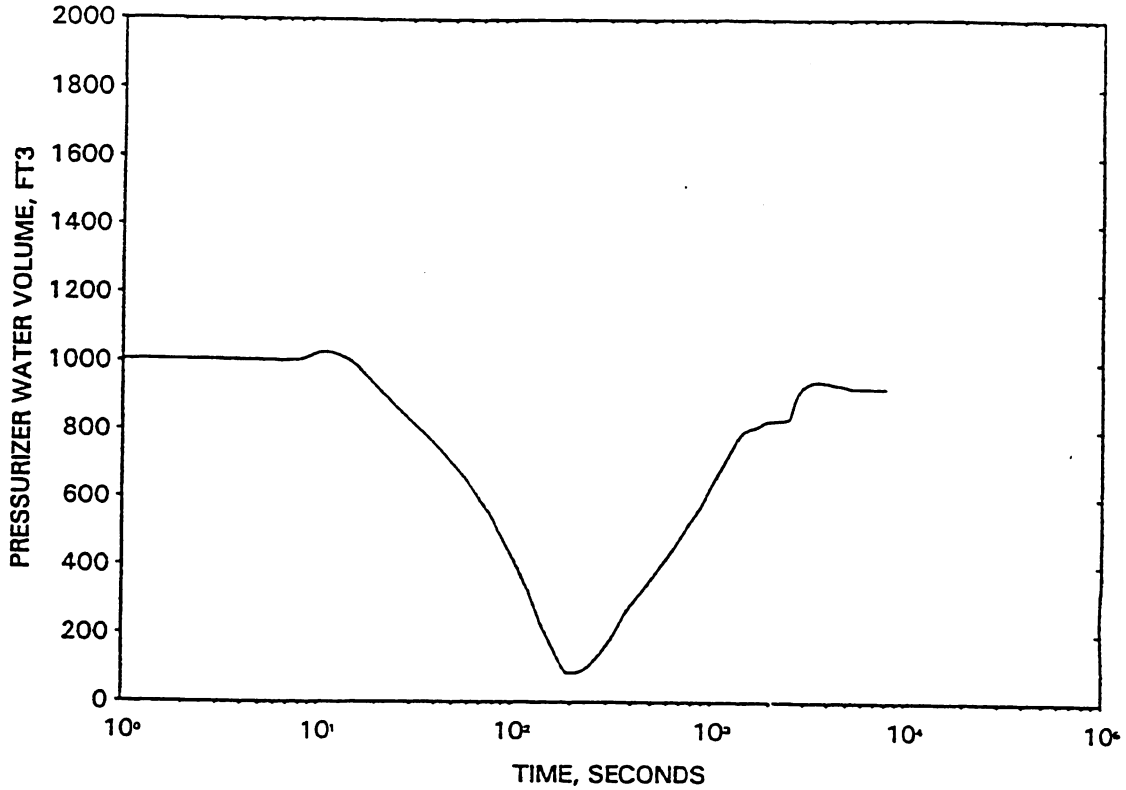


FIGURE 15.4.2-37a MAJOR RUPTURE OF A MFW PIPE - LOOP PRESSURIZER WATER VOLUME

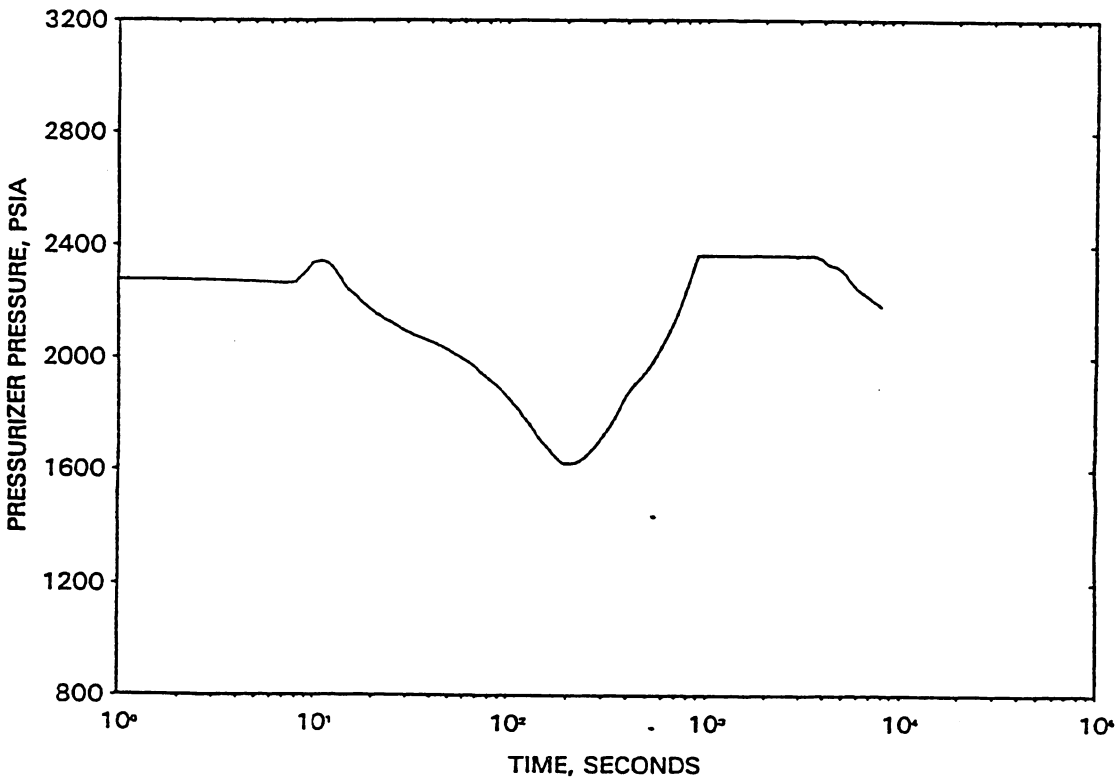


FIGURE 15.4.2-37b MAJOR RUPTURE OF A MFW PIPE - LOOP PRESSURIZER PRESSURE

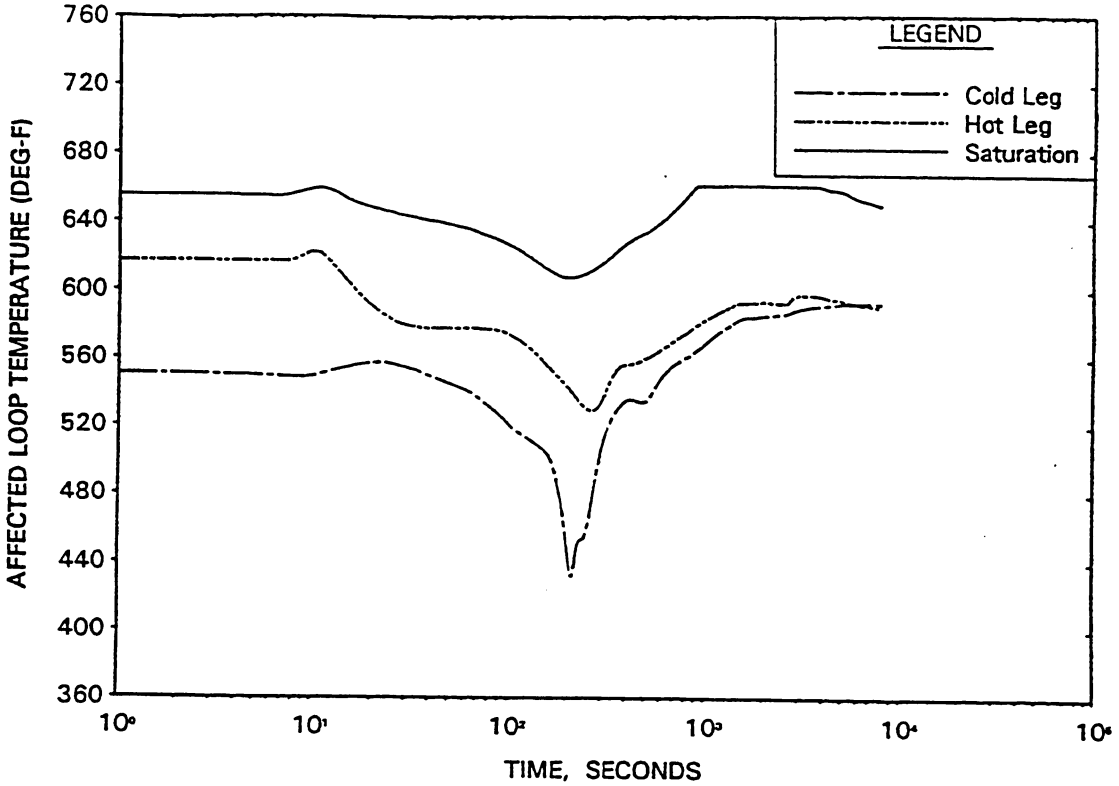


FIGURE 15.4.2-38a MAJOR RUPTURE OF A MFW PIPE - LOOP
AFFECTED LOOP COLD LEG, HOT LEG AND SATURATION TEMPERATURES

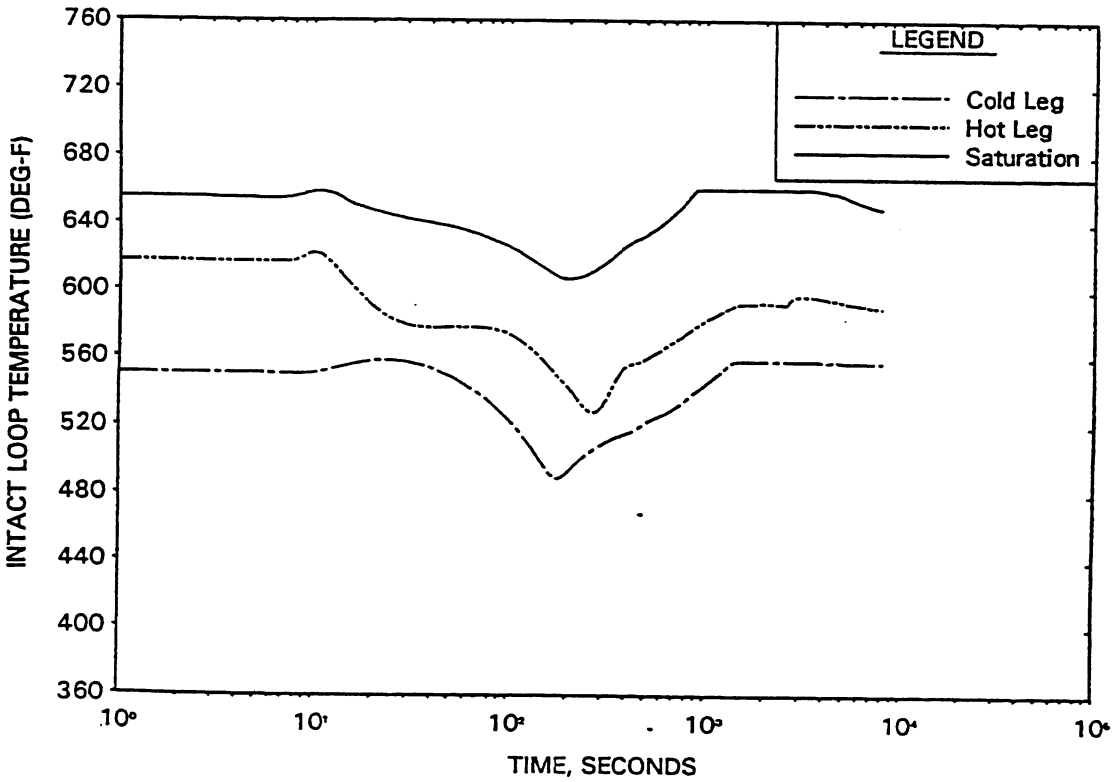


FIGURE 15.4.2-38b MAJOR RUPTURE OF A MFW PIPE - LOOP
INTACT LOOP COLD LEG, HOT LEG AND SATURATION TEMPERATURES

Revised by Amendment 15

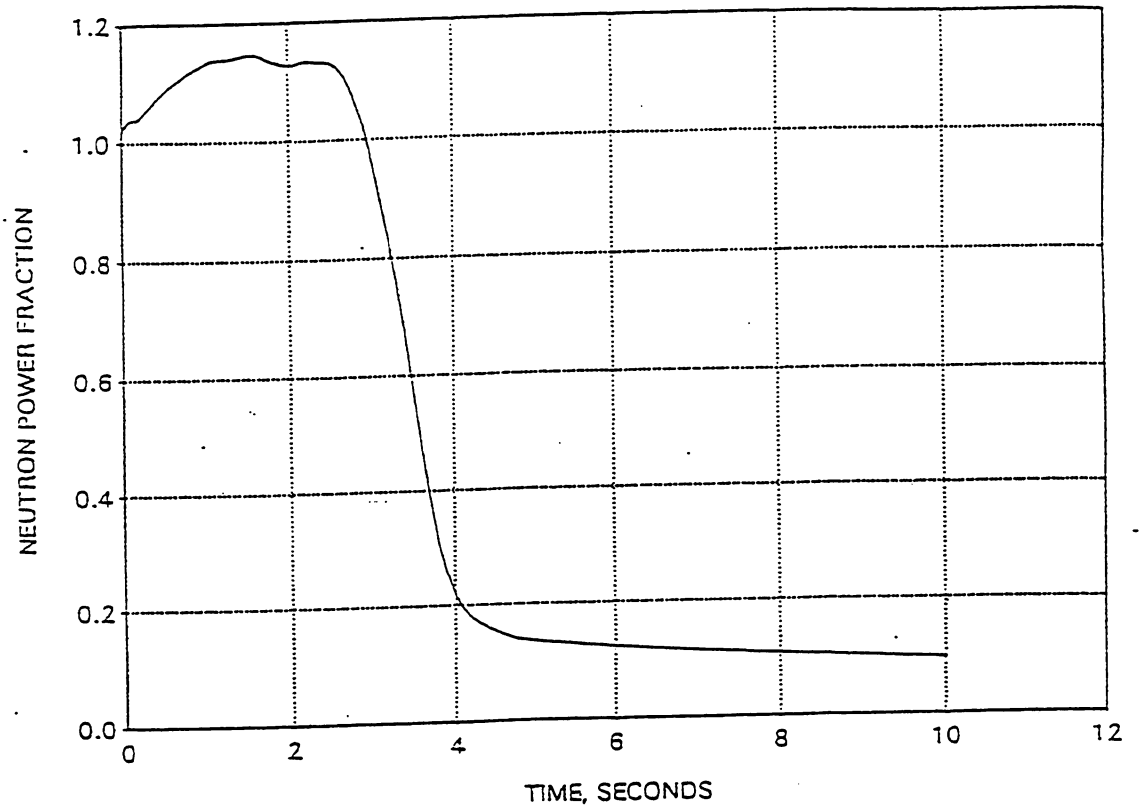


FIGURE 15.4.4-1

Locked Rotor
Neutron Power vs. Time

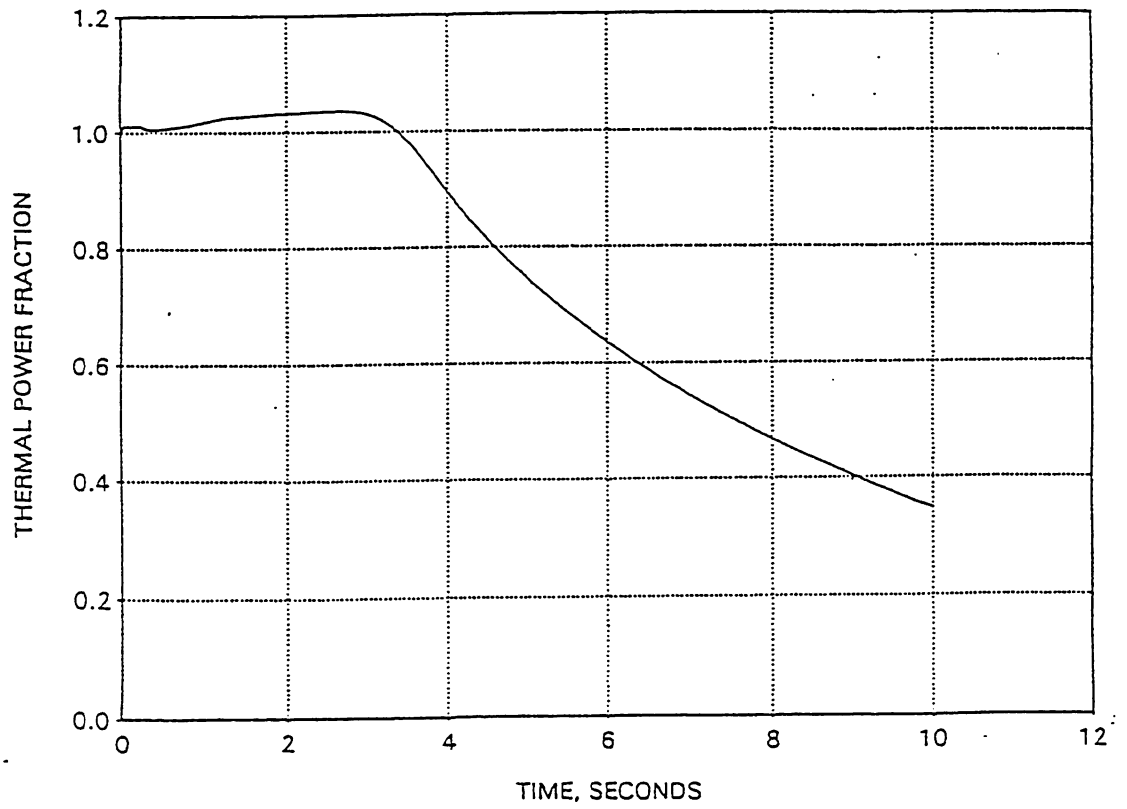


FIGURE 15.4.4-2

Locked Rotor
Thermal Power vs. Time

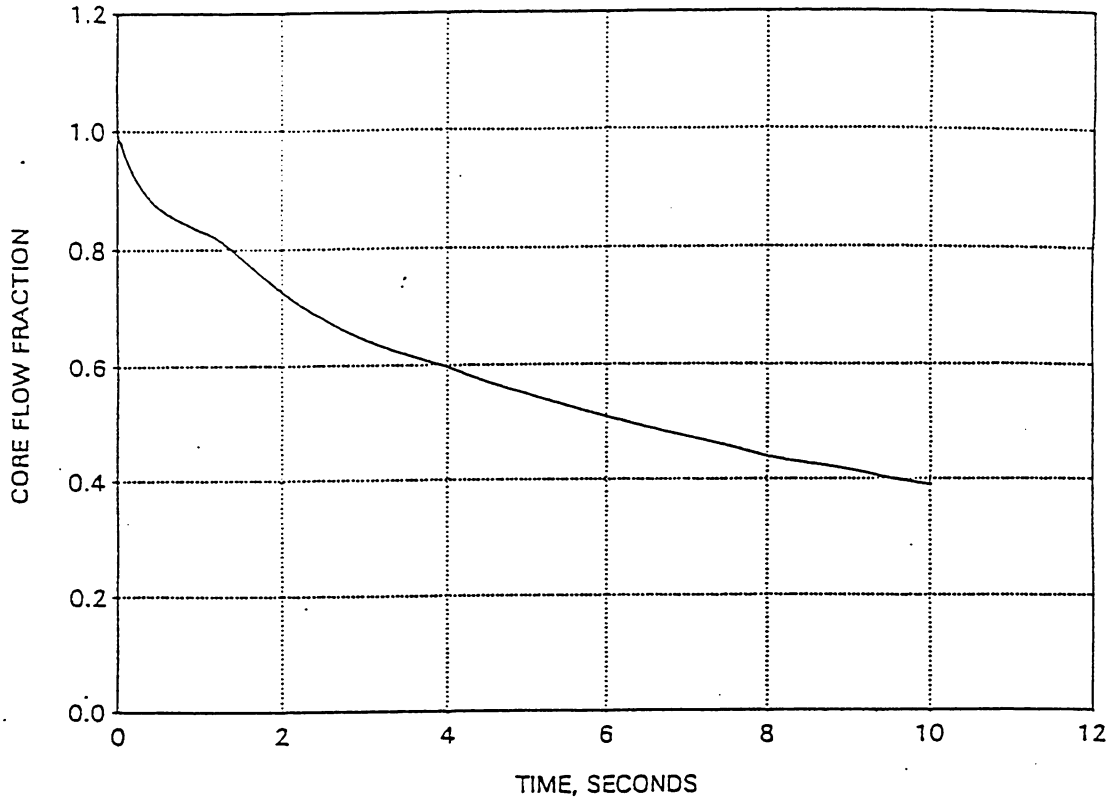


FIGURE 15.4.4-3

Locked Rotor
Core Flow Fraction vs. Time

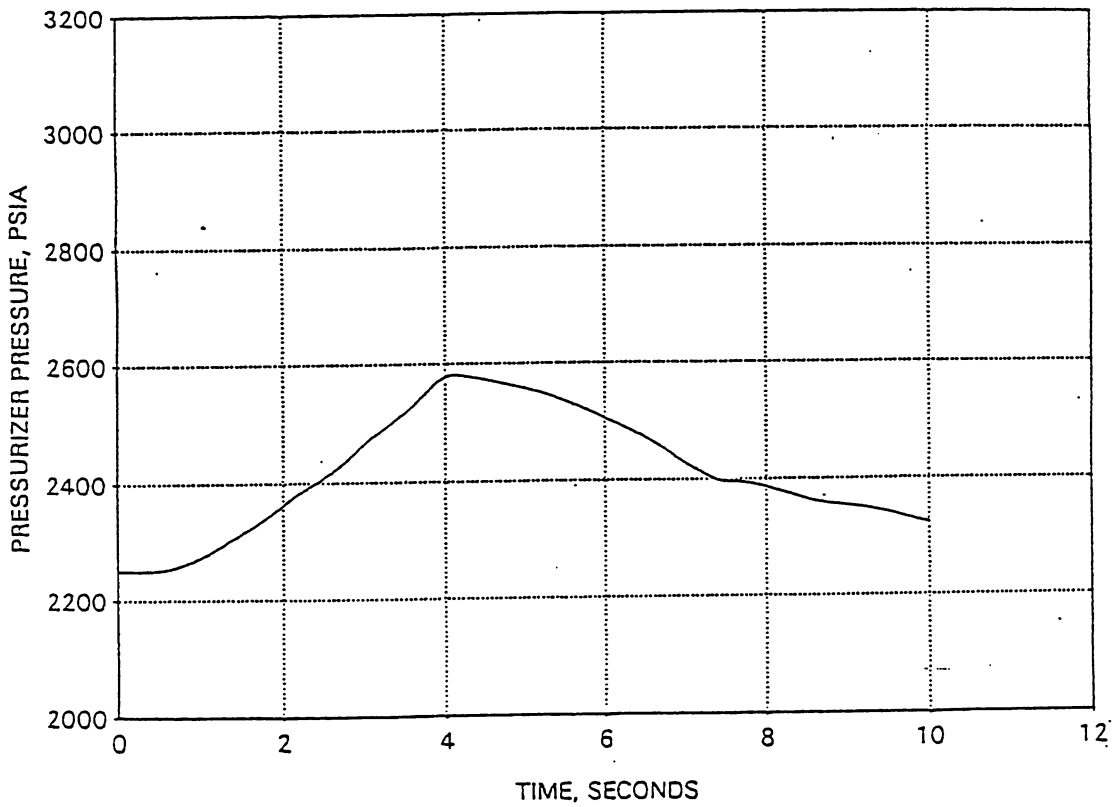


FIGURE 15.4.4-4

Locked Rotor
Pressurizer Pressure vs. Time

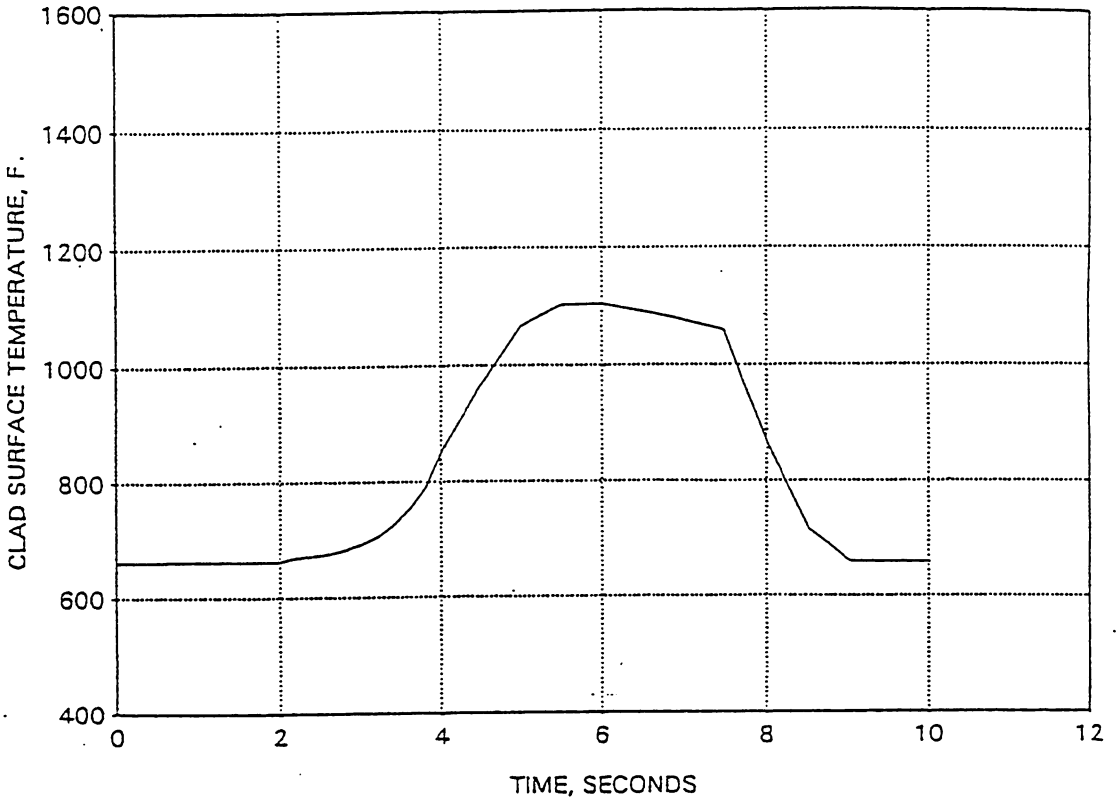
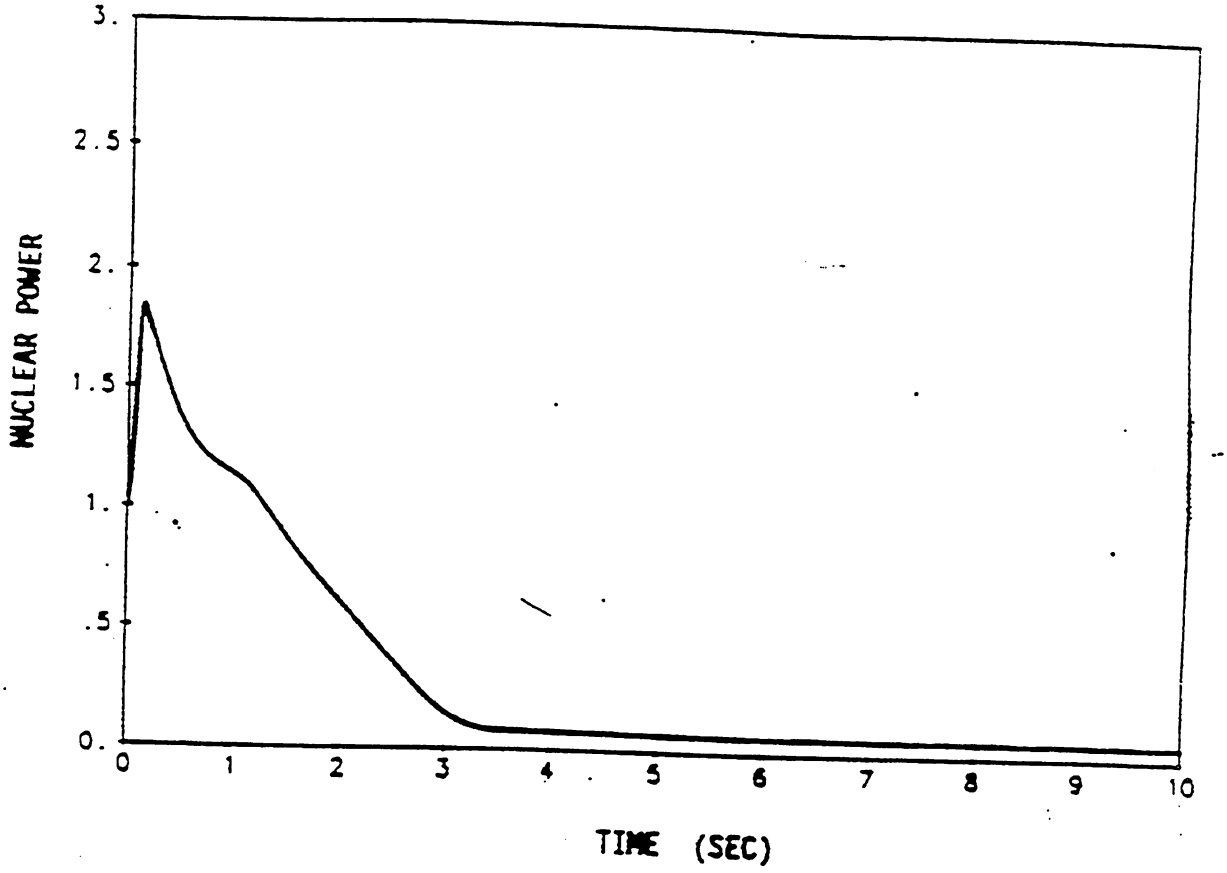


FIGURE 15.4.4-5

Locked Rotor
Clad Surface Temperature vs. Time

Best Available Historical Image

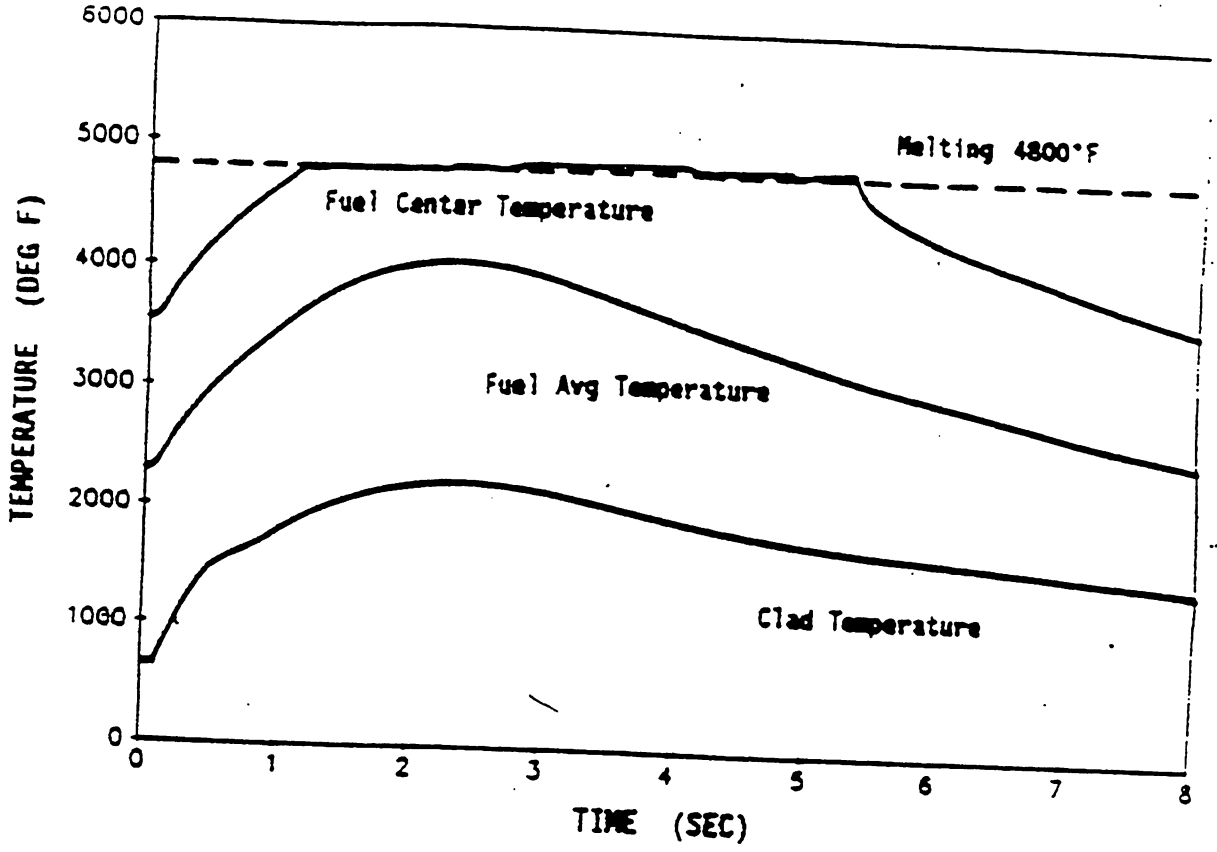


SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Rod Cluster Control Assembly Ejection
Nuclear Power vs. Time (EOL, HFP)

Figure 15.4.6-1

Revised by Amendment 10

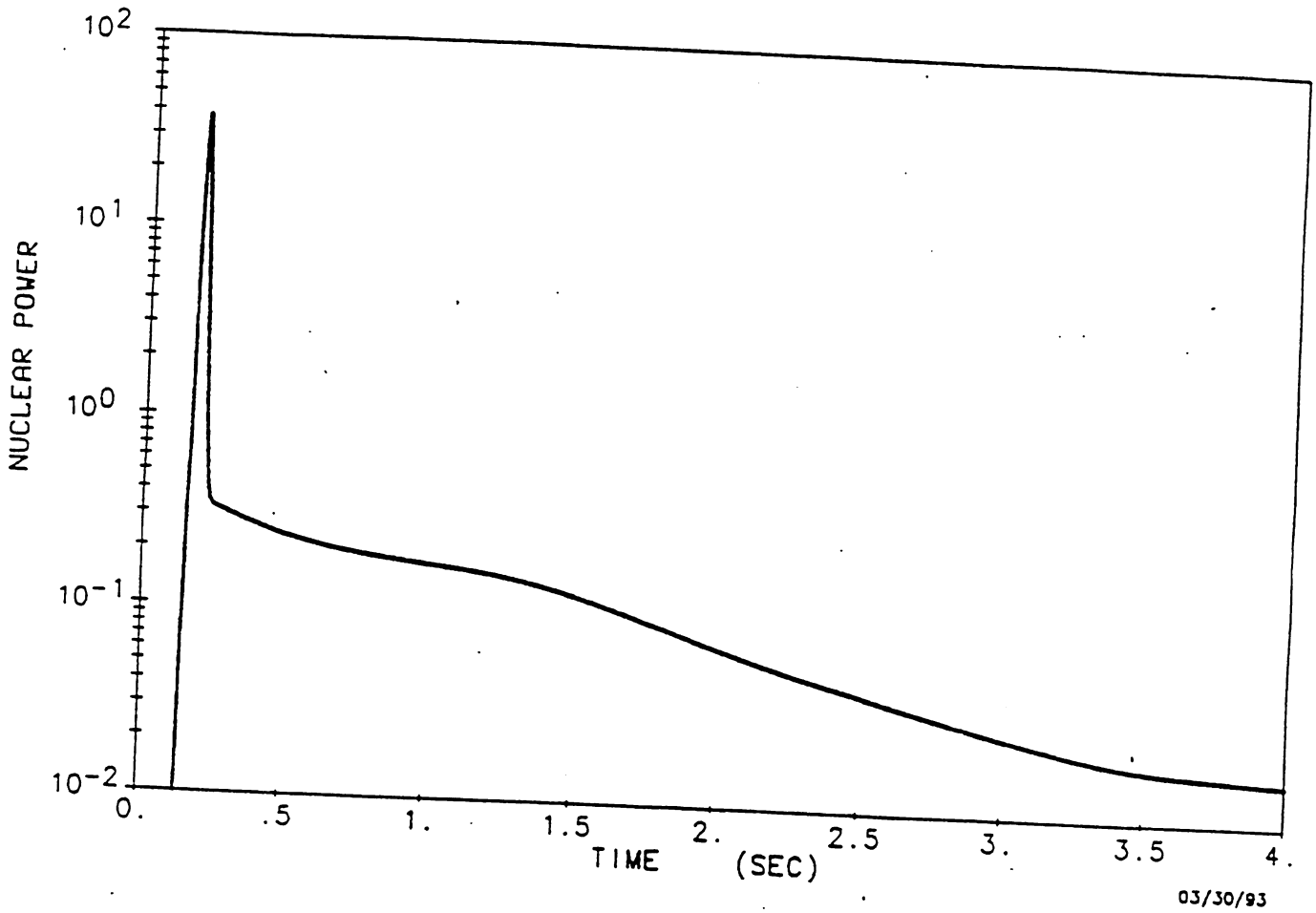


SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Rod Cluster Control Assembly Ejection
Fuel and Clad Temperature vs. Time (EOL, HFP)

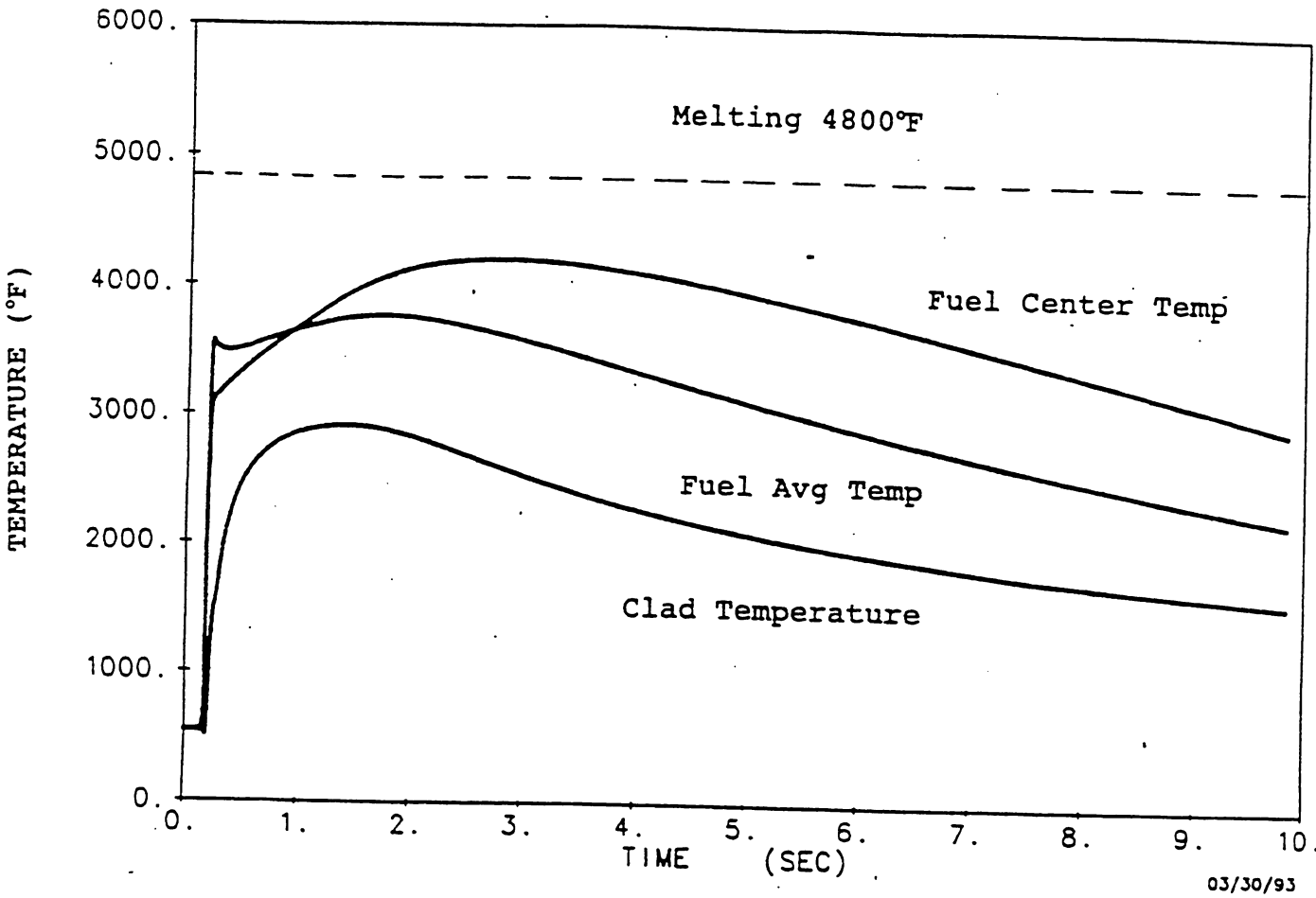
Figure 15.4.8-2

Revised by Amendment 10



SEQUOYAH FINAL SAFETY ANALYSIS REPORT UNITS 1 and 2
Rod Cluster Control Assembly Ejection Nuclear Power vs. Time (EOL, HZP)
Figure 15.4.6-3

Revised by Amendment 10



SEQUOYAH FINAL SAFETY ANALYSIS REPORT UNITS 1 and 2
Rod Cluster Control Assembly Ejection Fuel and Clad Temperature vs. Time (EOL, HZP)
Figure 15.4.6-4

Revised by Amendment 10

15.5 ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

Each unit specific cycle nuclear fuel reload analysis will verify the consequences of an accident previously evaluated in the FSAR have not increased. The reload core design does not have a direct role in mitigating the consequences of any design basis accident and does not affect any of the conclusions for the current 10CFR50 Appendix A, GDC-19 or 100CFR100 analyses described in this section. The reload core design ensures that all applicable design criteria and licensing basis acceptance criteria are met. Adherence to applicable standards and criteria ensures that the fission product barriers maintain an adequate design margin relative to the applicable safety margins.

15.5.1 Environmental Consequences of a Postulated Loss of A.C. Power to the Plant Auxiliaries

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented assuming steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. Parameters used in the conservative analysis are listed in Table 15.5.1-1.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated loss of offsite power to the plant auxiliaries:

1. Offsite power is lost and the main steam condensers are not available for steam dump.

2. Eight hours after the accident the residual-heat removal system starts operation to cool down the plant.

3. After eight hours following the accident, no steam and activity are released to the environment.

4. No air ejector release and no steam generator blowdown during the accident.

5. The primary to secondary leakage is evenly distributed in steam generators.

6. Initial primary coolant activity (iodine and noble gas) is consistent with the model discussed in Appendix 15D. This model is consistent with Technical Specification limits on primary coolant activity limits.

7. The analysis assumes an iodine spike with two separate cases considered. For one case there is assumed to be a pre-existing iodine spike. For the second case the initial primary coolant iodine concentration is at the limit set in the Technical Specifications for equilibrium operation and an iodine spike is assumed to be initiated by the reactor trip associated with the event. The iodine spiking models are described in Appendix 15D.

8. Initial secondary coolant activity (iodine) is consistent with the model discussed in Appendix 15D. This model is consistent with Technical Specification limits on primary coolant activity limits. There is no noble gas activity in the secondary coolant.

9. The iodine partition factor is $\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}}$ = 0.01 in steam generators.

10. During the postulated accident, iodine transferred to the secondary side in the three good steam generators is uniformly mixed with the water in the steam generators.

11. The steam release for cooling down the plant is equally contributed by all steam generators.

12. The 0-2 and 2-8 hour atmospheric diffusion factors given in Appendix 15A and the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable.

The resulting doses for the case with the accident-initiated iodine spike are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	0.69 rem	0.085 rem	0.18 rem
Low Population Zone	0.21 rem	0.013 rem	0.026 rem

These doses are well within the limits of 25 rem whole body and 300 rem thyroid defined in 10 CFR 100.

[For information only, the TEDE doses are 0.11 rem at the EAB and 0.02 at the LPZ.]

For the case with the pre-existing iodine spike the doses are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	0.73 rem	0.078 rem	0.17 rem
Low Population Zone	0.18 rem	0.012 rem	0.024 rem

These doses are well within the limits of 25 rem whole body and 300 rem thyroid defined in 10 CFR 100.

[For information only, the TEDE doses are 0.10 rem at the EAB and 0.017 at the LPZ.]

Framatome ANP Safety Evaluation With Replacement Steam Generators in Unit 1

The primary-to-secondary leak rate primary coolant activity, iodine activity in the secondary side liquid, and iodine partition factor are set by Technical Specification limits and are not affected by SG design.

The steam release to cool the plant to the RHR cut-in temperature is approximately 5600 lbm less with the RSDs than with the OSGs. Also, the secondary mass at full power and at RHR cut-in is less with the RSGs. Therefore, the mass of secondary steam released to the atmosphere to cool the plant to the RHR cut-in temperature is bounded by the calculation with the OSG.

Since all parameters affecting the Loss of A.C. Power to the Plant Auxiliaries for environmental consequences are not adversely affected by the RSG, the results of the existing analysis are applicable and all acceptance criteria continue to be met with the steam generator replacement.

15.5.2 Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture

The analysis of the postulated waste gas decay tank rupture is performed based on Regulatory Guide 1.24, 1972 (Reference 2).

The parameters used for waste gas decay tank rupture analysis are listed in Table 15.5.2-1. The bases for the analysis are:

1. The reactor has been operating at full power with one percent defective fuel and a shutdown to cold condition has been conducted near the end of an equilibrium core cycle. As soon as possible after shutdown, all noble gases have been removed from the Reactor Coolant System and transferred to the gas decay tank that is assumed to fail.

   The iodine inventory of the tank is based on plant operating procedures (degassing of volume control tank every 3 hours for 21 hours after shutdown). At 21 hours after shutdown the summation of the product of the iodine isotopic inventories in the tank and their respective dose conversion factors is a maximum.

2. The maximum content of the decay tank assumed to fail is used for the purpose of computing the noble gas inventory in the tank. Radiological decay is taken into account in the computation only for the minimum time period required to transfer the gases from the Reactor Coolant System to the decay tank. The noble gas and iodine inventories of the tank are given in Table 15.5.2-2.

3. The tank rupture is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank at ground level to the outside atmosphere. The assumption of the release of the noble gas inventory from only a single tank is based on the fact that all gas decay tanks will be isolated from each other whenever they are in use.

4. The short-term, i.e., 0-2 hour, diffusion factor at the site boundary given in Appendix 15A is used to evaluate the doses from the released activity.

The resulting doses are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	0.039 rem	1.8 rem	4.7 rem
Low Population Zone	0.005 rem	0.22 rem	0.56 rem

These doses are well within the limits of 25 rem whole body and 300 rem thyroid defined in 10 CFR 100.

[For information only, the TEDE doses are 1.8 rem at the EAB and 0.22 at the LPZ.]

Framatome ANP Safety Evaluation With Replacement Steam Generators in Unit 1

The tank activity assumed at the event initiation are conservatively determined based on the reactor coolant system volume. The RCS volume is unaffected by SG replacement and the assumed tank activity, based on OSGs, is unchanged.

15.5.3 Environmental Consequences of a Postulated Loss of Coolant Accident

The results of the analysis presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a Loss-of-Coolant Accident (LOCA) do not result in doses which exceed the guideline values specified in a 10 CFR 100.

An analysis based on Regulatory Guide 1.4, 1973, (Reference 3) was performed. The parameters used for the analysis are listed in Table 15.5.3-1. In addition, an evaluation of the dose to control room operators and an evaluation of the offsite dose resulting from the operation of the Post-Accident Sampling Facility are presented.

Fission Product Release to the Containment

Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the Emergency Core Cooling System keeps cladding temperatures well below melting, and limits zirconium-water reactions to an insignificant level, ensuring that the core remains intact and in place.

As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the Reactor Coolant System and thereby to the primary containment.

In order to conservatively evaluate the radiological consequences of a fission product release, the offsite doses were calculated for a core inventory fission product release case.

Core Activity Release (Regulatory Guide 1.4 Analysis)

The offsite doses resulting from a hypothetical accident such as a large LOCA assuming core activity releases have been analyzed. Activity releases of these magnitudes have a considerably lower probability than those associated with a gap release. For the analysis of this hypothetical case, it is assumed that of the entire core-fission product inventory, 100 percent of the noble gases, 50 percent of the halogens, and 1% of the solids in the fission product inventory are released to the containment. Of the fission product iodine released to the containment, 50 percent is considered to be available for leakage, while the remaining 50 percent is assumed to condense on the various structural surfaces in the containment.

Thus, a total of 100 percent of the noble gas core inventory and 25 percent of the core iodine inventory are assumed to be immediately available for leakage from the primary containment. Of the halogen activity available for release, it is further assumed that 91 percent is in elemental form, 4 percent in methyl form, and 5 percent in particulate form.

The fission product inventories used for the core activity release cases are listed in Table 15.5.3-5. Post LOCA radiation doses at the site boundary and low population zone are provided in Table 15.5.3-4.

Modeling of Removal Process

For fission products other than iodine, the only removal process considered is radioactive decay.

The fission product iodine is assumed to be present in the containment atmosphere in elemental, organic, and particulate form. It is assumed that 91 percent of the iodine available for leakage from the containment is in elemental (i.e., iodine vapor) form, 4 percent is assumed to be in the form of organic iodine compounds (e.g., methyl iodine), and 5 percent is assumed to be adsorbed on airborne particulate matter. In this analysis it was conservatively assumed that the organic form of iodine is not subject to any removal processes other than radioactive decay and leakage from the containment.

The effectiveness of the ice condenser for elemental iodine removal is described in Section 6.2.3.3.4. For the calculation of doses, the ice condenser was treated as a removal process proportional to the amount of elemental and particulate iodine airborne in the containment, with time dependent removal constants. The time dependent ice condenser iodine removal efficiencies for the conservative Regulatory Guide 1.4 (1973) analysis are given in Table 15.5.3-2.

Ice Condenser

The ice condenser is designed to limit the leakage of airborne activity from the containment in the event of a LOCA. This is accomplished by the removal of heat released to the containment during the accident to the extent necessary to initially maintain that structure below design pressure and then reduce the pressure to near atmospheric. The addition of an alkaline solution such as sodium tetraborate enhances the iodine removal qualities of the melting ice to a point where credit can be assumed in the radiological analyses.

The operation of the containment deck fans is delayed for 10 minutes following the LOCA. This delay in fan operation yields an initial inlet steam-air mixture into the ice condenser of greater than 90 percent steam by volume which results in more efficient iodine removal by the ice condenser.

As a result of experimental and analytical efforts, the ice condenser system has been proven to be an effective passive system for removing iodine from the containment atmosphere following a LOCA. (Reference 4)

With respect to iodine removal by the ice condenser, the following assumptions were made:

1. The ice condenser is only effective in removing airborne elemental and particulate iodine from the containment atmosphere.

2. The ice condenser is modeled as a time dependent removal process.

3. The effectiveness of the ice condenser in removing iodine is lost after all of the ice has melted using the most conservative assumptions.

Primary Containment Leak Rate

The primary containment leak rate used in the conservative Regulatory Guide 1.4 (1973) analysis is the design basis leak rate guaranteed in the technical specifications regarding containment leakage. For the first 24 hours following the accident, the leak rate was assumed to be 0.25 percent per day and the leak rate was assumed to be 0.125 percent per day for the remainder of the 30-day period.

The leakage from the primary containment can follow either of two paths: (1) leakage into the annulus volume, or (2) through-line leakage to rooms in the Auxiliary Building (see Figure 15.5.3-1). The environmental effects of the core-release source event have been analyzed on the basis that 25 percent of the total primary containment leakage goes to the Auxiliary Building.

Auxiliary Building Release Path

The Auxiliary Building allows holdup and is normally ventilated by the Auxiliary Building ventilation system. However, upon initiation of the SIS signal following a LOCA, the normal ventilation systems to all areas of the Auxiliary Building are shutdown and isolated. Upon Auxiliary Building isolation, the Auxiliary Building Gas Treatment System (ABGTS) is activated to provide ventilation of the area and filtration of the exhaust to the atmosphere. This system is described in Subsection 6.2.3.

Fission products which leak from the primary containment to areas of the Auxiliary Building will be diluted in the room atmosphere and will travel via ducts and other rooms to the fuel handling area or the waste packaging area where the suctions for the (ABGTS) are located. The mean holdup time for airborne activity in the Auxiliary Building areas other than the fuel handling area is greater than one hour with the Auxiliary Building isolated and both trains of the ABGTS operating. For the reference case, it has been conservatively assumed in the estimation of activity releases that activity leaking to the Auxiliary Building is immediately released without filtration to the environment for the first 5 minutes after which it is held up for 0.3 hours and then released through the ABGTS filter system. This assumption is based on an initial delay of 4 minutes to establish the ABSCE and 1 minute to draw down the ABSCE to a negative 1/4-inch water gauge. In the Regulatory Guide 1.4 analysis, the ABGTS filter system is assumed to have a removal efficiency of 95 percent for all forms of iodine.

The Auxiliary Building internal pressure will be maintained at less than atmospheric during normal operation (See Subsection 9.4.2), thereby preventing release to the environment without filtration following a LOCA. The annulus pressure will be maintained less than the Auxiliary Building internal pressure during normal operation, therefore, any leakage between the two volumes following a LOCA will be into the annulus. It has been assumed conservatively that there is no leakage via this route.

Shield Building Releases

The presence of the annulus between the containment vessel and the shield building reduces the probability of direct leakage from the containment vessel to the atmosphere and allows holdup, dilution, mixing, and plate- out of fission products in the shield building. Seventy five percent of the primary containment leakage is assumed to go into the annulus volume in the reference case.

The initial pressure in the annulus is less than atmospheric. After blowdown, the annulus pressure will increase rapidly due to expansion of the containment vessel as a result of primary containment

atmosphere temperature and pressure increases. The annulus pressure will continue to rise due to heating of the annulus atmosphere by conduction through the containment vessel. At 46 seconds following a LOCA, the annulus pressure will rise above negative 1/4 inch water gauge for approximately eight (8) seconds. After a delay of 38 seconds from the start of the accident, the Emergency Gas Treatment System (EGTS) will be operating at full flow to return the annulus pressure to negative 1/2 inch water gauge. See Reference 21.

The EGTS is essentially an annulus recirculation system with pressure activated valves which allow part of the system flow to be exhausted to atmosphere to maintain a "negative" annulus pressure. The system includes absolute and impregnated charcoal filters for removal of halogens. After the initial delay of the ABGTS, the EGTS combined with ABGTS ensures that all primary containment leakage is filtered before release to the atmosphere.

The EGTS suction points in the annulus are located at the top of the containment dome, while nearly all penetrations are located near the bottom of the containment (see Subsection 6.2.3.2.2), thereby minimizing the probability of leakage directly from the primary containment into the EGTS. However, it has been conservatively assumed for the reference case that, after the initial 30 second period, 100 percent of the primary containment leakage to the annulus volume goes directly to the EGTS suction.

The holdup time is a function of the EGTS flow and exhaust rates as well as the annulus volume. The mean holdup time (t_H) before release to the atmosphere is defined as:

$$t_H = \frac{0.5 \times \text{Volume of the Annulus}}{\text{Exhaust Flow from the EGTS to Atmosphere}}$$

It is conservatively assumed that only 50 percent of the annulus free volume is available for mixing of activity.

Nearly all of the leakage is expected to occur in the area of the penetrations near the base of the annulus where it would be diluted by the EGTS flow and slowly travel to the EGTS suction.

Table 15.5.3-3 shows the variation of EGTS exhaust and recirculation flow rates, with time after the LOCA, which was used for calculation of activity releases for the conservative Regulatory Guide 1.4 analysis. The flow path of fission products which are drawn into the air handling systems is shown schematically in Figure 15.5.3-1.

Effectiveness of Double Containment Design

The analysis has demonstrated clearly the benefits of the double containment concept. As would be expected for a double barrier arrangement, the second barrier acts as an effective holdup tank, resulting in substantial reduction in the two-hour inhalation and whole body immersion doses. The expected offsite doses for the 30-day period at the low population zone are also substantially reduced, since the holdup process is effective for the duration of the accident.

The EGTS exhaust flow rate is dependent on the rate of air inleakage to the annulus. Studies (Reference 5) made of leak rates from typical concrete buildings of this type have resulted in leak

rates from 4 percent to 8 percent per day at a pressure differential of 14 inches of water. Although the pressure differential in this case will be much lower than this value, it has been assumed that a shield building inleakage flow of 500 SCFM exists throughout the 30-day period. This inleakage flow includes leakage past ventilation system primary containment isolation valves assuming that a single isolation valve fails in the open position.

Emergency Gas Treatment System Filter Efficiencies

The EGTS takes suction from the annulus, and the exhaust gases are drawn through two banks of impregnated charcoal filters in series. Sufficient filter capacity is provided to contain all iodines, inorganic, organic, and particulate available for leakage. Since the air in the annulus is dry, filter efficiencies of greater than 99 percent are attainable. Tests reported in ORNL-NSIC-4 (Reference 6) have demonstrated that inorganic halogen removal efficiencies greater than 99.99 percent can be expected with low relative humidity.

For this analysis, the overall filter system efficiency for the two filter banks of 95 percent was assumed for all forms of iodine.

Discussion of Results

The gamma, beta, and thyroid doses for the LOCA at the exclusion area boundary and the low population zone are given in Table 15.5.3-4 for the analyses presented in this section. The dose limits for this accident are defined in 10 CFR 100 (25 rem deep dose equivalent and 300 rem thyroid). Even for this conservative analysis, the doses are well within the 10 CFR 100 guidelines.

The major factor in the effectiveness of the secondary containment is its inherent capability to collect the containment leakage for filtration of the radioactive iodine prior to release to the environment. This effect is greatly enhanced by the recirculation feature of the air handling systems, which forces repeated filtration passes for the major fraction of the primary containment leakage before release to the environment.

Loss of Coolant Accident - Control Room Operator Doses

In accordance with General Design Criterion 19, the control room ventilation system and shielding have been designed to limit deep dose equivalent during an accident period to 5 rem. Thyroid dose is limited to 30 rem and beta skin dose should not exceed 30 rem.

The doses to personnel during a postaccident period originate from several different sources. Exposure within the control room may result from airborne radioactive nuclides entering the control room via the ventilation system. In addition, personnel are exposed to direct gamma radiation penetrating the control room walls, floor, and roof from:

1. Radioactivity within the primary containment atmosphere.

2. Radioactivity released from containment which may have entered adjacent structures.

3. Radioactivity released from containment which passes above the control room roof.

Further exposure of control room personnel to radiation may occur during ingress to the control room from exclusion area boundary and during egress from the control room to site boundary.

In the event of a radioactive release incident, the control room will be isolated automatically by a safety injection system signal and/or by radiation signal from beta detectors located in the air intake stream common to the air intake ports at either end of the Control Building. These redundant signals are routed to redundant controls which actuate air-operated isolation dampers. Operation of the emergency pressurizing fans with inline HEPA filters and charcoal adsorbers is also initiated by these signals. Simultaneously, 4000 cfm of recirculation and makeup air is rerouted automatically through the HEPA filters and charcoal adsorbers. Approximately 200 to 1000 cfm of outside air bypasses the intake isolation dampers through a flow controlled duct routed to the emergency recirculation system upstream of the HEPA filters and charcoal adsorbers. This flow of outside air provides the control room with a slight positive pressure relative to the atmosphere outside and to surrounding structures. Isolation dampers located in each intake line may be selectively closed by control room personnel. The selection between the two would be based on the objective of admitting a minimum of airborne activity to the control room via the makeup airflow. The control room ventilation flow system is shown in Figure 9.4.1-1. The equivalent of 51 cfm of unfiltered outside air enters through the main control room doors as a result of personnel entering or leaving and through leakage from ducts and dampers. To evaluate the ability of the control room to meet the requirements of General Design Criterion 19, a time dependent model of the control room was developed. In this model, the outside air concentration enters the control room via the isolation damper bypass line and the HEPA filters and charcoal adsorbers. The concentration in the room is reduced by decay, leakage out, and by recirculation through the HEPA filters and charcoal adsorbers. Credit for filtration is taken during two passes through the charcoal adsorbers. Using these assumptions the following equations for the rate or change of the control room concentrations are obtained:

$$\frac{dM}{dt} = C_o(1 - K_1)\,L/V - L/V\,M - \frac{R_c}{V}M - \lambda M \qquad (1)$$

$$\frac{dN}{dt} = \frac{R_c}{V}(1 - K_2)\,M - L/V\,N - \lambda N \qquad (2)$$

$$C(t) = M(t) + N(t) \qquad (3)$$

Where

M(t) = Once filtered time dependent concentration

N(t) = Twice filtered time dependent concentration

C(t) = Total time dependent concentration in control room

C_o = Concentration of isotope entering air intake

K_1 = Effective filter efficiency for a particular isotope during first pass (corrected for bypass leakage)

K_2 = Filter efficiency for a particular isotope during second pass

L = Flow rate of outside air into control room and leakage out of control room

R_c = Recirculated air flow rate through filters

λ = Decay constant

V = Control room free volume

These equations are readily solvable if C_o is constant or a simple function of time during a time interval. Since C_o consists of a number of terms involving exponentials, it was assumed to be constant during particular time intervals corresponding to the average concentration during each interval as described below. Solving equations (1), (2), and (3) yields:

$$C(t) = \left[\frac{(1-K_1)(1-K_2)C_o}{W_m V}\right]\left[\frac{L}{(1-K_2)(1-e^{-W_m t})} + \frac{R_c L}{W_n V}(1-e^{-W_n t}) - L(1-e - e^{-W_m t})\right] \quad (4)$$

Where

$$W_m = \frac{(L + R_c + \lambda V)}{V}$$

$$W_n = \frac{(L + \lambda V)}{V}$$

The value of C_o used in equation (4) is determined as follows:

$$C_{oi} = \frac{(X/Q)_i}{t_{i+1} - t_i} \int_{t_i}^{t_{i+1}} R dt \quad (5)$$

Where

C_{oi} = Average concentration of activity outside control room during ith time period (Ci/m^3)

$(X/Q)_i$ = Atmospheric dilution factor (sec/m^3) during the ith time period

R = Time dependent release rate of activity from containment (Ci/sec)

The atmospheric dilution factors were determined using the accumulated meteorological data on wind speed, direction, and duration of occurrence obtained from the Sequoyah plant site applied to a building wake dilution model. The dilution factors are calculated by the ARCON96 methodology (Reference 11). The values used are given in Table 15.5.3-6.

Equation (4) is used to determine the concentration at any time within a time period and upon integrating and dividing by the time interval gives the average concentration during the time interval due to inflow of radioactivity with outside air as shown:

$$\overline{C_i} = \frac{\int_O^T C_i(t)dt}{T - O}$$

Where

T	= t- t_{i-1}
t	= Time after accident
t_{i-1}	= Time at end of previous time period

Further contributions to the concentration during time period are due to the concentrations remaining from prior time periods. These contributions are obtained from the following equations:

$$C_{R(i+j)} = M_{R(i+j)} + N_{R(i+j)} \quad (7)$$

$$\frac{dM_{R(i+j)}}{dt} = -(L/V + R_c/V + \lambda)\, M_{R(i+j)} \quad (8)$$

$$\frac{dN_{R(i+j)}}{dt} = R_c/V\,(1-K_2)\, M_{R(i+j)} - (L/V + \lambda)\, N_{R(i+j)} \quad (9)$$

With initial conditions:

$M_{R(i-j)}(O) = M_{R0(i)}$ = (Once filtered concentration at the end of the <u>ith</u> time period.)

$N_{R(I+j)}(O) = N_{RO(i)}$ = (Twice filtered concentration at end of the <u>ith</u> time period.)

Solving equations (8) and (9) and substituting certain initial condition relations equation (7) becomes:

$$C_{R(i+j)} = C_{Ro(i)}\, e^{-W_n(t-\tau)} - M_{RO(i)}\, K_2\, e^{-W_n(t-\tau_i)} - e^{-W_m(t-\tau_i)} \quad (10)$$

Integrating equation (10) for each of the prior time periods gives the contribution from these time periods to the present time period. The average concentration is determined for these contributions using the method of equation (6).

Filter efficiencies of 95 percent for elemental and particulate iodine and 95 percent for organic iodine were deemed appropriate for the first filter pass. Since the concentration of iodine in the Main Control Room is much reduced as a result of this filtration, the efficiencies were reduced for the second pass to 70 percent for elemental and particulate iodine, and 70 percent for organic iodine.

To account for the unfiltered inleakage, 51 cfm were added to the make-up flow (L in equation 1) of 200 cfm, and the filter factor for the first pass was decreased to an equivalent value of 75.7 percent for elemental and particulate iodine, and 75.7 percent for organic iodine. For a 1000 cfm makeup flow, the filter-factor for the first pass was decreased to equivalent values of 90.4 percent for both species. In both cases, the filter efficiencies for the second pass are not affected by the unfiltered inleakage.

The filter efficiency for noble gases was taken as zero for all cases.

The above equations were incorporated into a computer program together with appropriate equations for computing gamma dose, beta dose, and thyroid dose using these average nuclide concentrations and time periods. The whole body gamma dose calculation consists of an incremental volume summation of a point kernel over the control room volume. The principal gammas of each isotope are used to compute the dose from each isotope. The dose computations for beta activity were based on a semi-infinite cloud model. Doses to thyroid were based on activity to dose conversion factors and the breathing rates presented in Table 15A-1. (The equations and various data are given below.) The doses from these calculations are presented in Table 15.5.3-7. Gamma dose contributions from shine through the control room roof due to the external cloud and from shine through the control room walls from adjacent structures and from containment are computed using an incremental volume summation of a point kernel which includes buildup factors for the concrete shielding. For the calculation of shine through the control room roof, an atmospheric, rectangular volume several thousand feet in height and several control room widths was used. The control room roof is a 2-foot, 3-inch-thick concrete slab and is the only shielding considered in this calculation. The average isotope concentrations at the control bay for each time period were used as the source concentrations. For the shine from adjacent structures the shielding consists of the 3-foot-thick (5-foot in certain areas) control room walls. The doses are calculated similarly to the shine dose through the roof. The average isotope concentrations at the control bay intake for each time period are also used for these calculations.

The shine from the spreading room below the control room is also computed in the same manner as adjacent structures. The shielding for this computation consists of the 8 inch thick concrete floor. The summation of the incremental elements is performed over the volume of each room or structure of interest.

In addition to the dose due to shine from surrounding structures and from the passing cloud, the shine from the Reactor Containment Building also contributes to the deep dose equivalent to personnel. This contribution is computed in the same manner as the methods used above. Due to the location of the Auxiliary Building between the Reactor Buildings and the control room and the thicker control room Auxiliary Building wall near the roof, the minimum ray path through concrete from the containment into the control room below 10 feet above the control floor, is 8 feet. All nuclides released to containment are assumed uniformly distributed and their time dependent concentrations were used to compute dose. The dose computed from this source is negligible and is not included in Table 15.5.3-7.

Several doors penetrate the control room walls, and the dose at these areas would be larger than the doses calculated as described above. The potential shine at these doors and at other penetrations has been evaluated. As a result, hollow steel doors filled with No. 12 lead shot have been incorporated into the design of the shield wall between the control room and the turbine building. These doors will provide shielding comparable to the concrete walls. Shine through other penetrations was found to be negligible.

Another contribution to the total dose of control room personnel is the exposure incurred during ingress from and egress to the exclusion area boundary. The doses to the Control Room personnel were computed based on the following assumptions:

1. Five minutes are required to leave the control room and arrive at car or vice versa.

2. Five minutes are required to drive along the access road.

3. The radioisotope concentrations at the control room arrival or departure point are those computed for the control bay air intakes.

4. The concentration as a function of distance along the access road was determined from the atmospheric diffusion model.

5. One one-way trip first day, one round-trip/day 2nd through 30th day.

6. Other parameters used in the calculation were taken from Murphy and Campe[10]. They are:

1)	Occupancy Adjustment Factors:	100 percent occupancy 0-24 hours 60 percent occupancy 1-4 days 40 percent occupancy 4-30 days
2)	Wind Speed Factors:	5th percentile wind speed 0-8 hours 10th percentile wind speed 8-24 hours 20th percentile wind speed 1-4 days 40th percentile wind speed 4-30 days

It was also assumed that initially the make-up air intake would be through the vent admitting the highest radioisotope concentration, but that the Main Control Room personnel could switch intake vents 2 hours after the accident in order to admit a minimum of Airborne Activity to the Main Control Room via the make-up airflow.

The whole body and thyroid doses from the radiation sources discussed above are presented in Table 15.5.3-7. The dose to whole body is below the General Design Criterion 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 300 rem. The total calculated doses are within acceptable limits.

Dose Equations, Data, and Assumptions

The dose from gamma radiation originating within the control room is given by:

$$D\gamma = 1.696 \text{ X } 10^4 \sum_{j=1}^{\alpha} [\sum_{i=1}^{\beta} Q_{ij} (\sum_{k=1}^{\gamma} \{E_{ik} f_{ik} \frac{\mu_e}{(p)_{ki}} \sum_{m=1}^{\varepsilon} \sum_{n=1}^{\omega} \sum_{q=1}^{\sigma} \frac{e^{-\mu ak \sqrt{x_m^2 + y_n^2 + z_q^2}}}{(x_m^2 + y_n^2 + z_q^2)} \bullet \Delta x \Delta y \Delta z\})] \ (t_j - t_{j-1}) \qquad (11)$$

Where

D_γ = Absorbed dose in flesh in mrads

Q_{ij} = Concentration of isotope i during time period j in curies/m^3

E_{ik} = Energy of gamma k isotope i in MeV

f_{ik} = Number of the k gammas of isotope i given off per disintegration

$\frac{\mu_e}{(p)_{ki}}$ = Mass attenuation coefficient for flesh determined at the energy of the kth gamma of isotope i in cm^2/gram.

$_\mu ak$ = Linear attenuation coefficient for air determined at the energy of the kth gamma of isotope i in inverse meters.

x,y,z = Coordinate distances from the dose point to the source volume element (m,n,q) in meters.

t_{j-1}, t_j = Times at the beginning and end of time period j in hours.

This equation may also be used for shine through concrete shields by the inclusion of two additional factors within the innermost summation.
These factors are:

1. $\exp(-\mu_{ck} t_c \sec \Theta)$

2. $B_c (\mu_{ck} t_c \sec \Theta)$

Where

μ_{ck} = The linear attenuation coefficient of concrete determined at the energy of the kth gamma of isotope i in inverse meters

t_c = Concrete shield thickness

B_c = Buildup factor for concrete

Θ = Angle between a vector normal to the shield and a vector from the dose point to the source point

Terms (1) and (2) are the linear attenuation exponential and the buildup factors for concrete respectively.

The dose from beta radiation is given by the semi-infinite cloud immersion dose:

$$D_B = (0.230)\ (X/Q) \left[\sum_{i=1}^{a} Q_i \left(\sum_{k=1}^{\gamma} E_{ik} f_{ik} \right) \right]$$

Where

D_B = Dose due to beta in rem

X/Q = Atmospheric dispersion factor during time period in sec/m^3

Q_i = Accumulated activity release of isotope i during time period (12)

E_{ik} = Average energy of beta k of isotope i

f_{ik} = Number of k betas of isotope i per disintegration

For beta dose in the control room equation (12) becomes:

$$D_B = (0.230) \sum_{j=1}^{\delta} \sum_{i=1}^{\alpha} \overline{C_{ij}} \sum_{k=1}^{\beta} E_{ik} f_{ik}\ (t_j - t_{j-i})$$

Where

$\bar{C}_{ij}$ = Average concentration of isotope i during time period j.

Inhalation Dose (Thyroid)

The inhalation dose for a given period of time has the general form:

$$D_I = (X/Q)(B)\left[\sum_{i=1}^{n} (Q_{ij})(DCF_i)\right](t_j - t_{j-i}) \qquad (13)$$

Where

D_I = Thyroid inhalation dose, rem

X/Q = Site dispersion factor during time period, sec/m^3

B = Breathing rate during time period, m^3/hr

Q_{ij} = Average activity release rate during time period j of iodine isotope i

DCF_i = Dose conversion factor for iodine isotope i, rem/microcurie inhaled

t_j = Total time at end of period j, hours

For inhalation dose within the control room equation (13) becomes:

$$D_i = (B)\left[\sum_{i=1}^{n} \overline{C_{ij}}(DCF_i)\right](t_j - t_{j-1}) \qquad (14)$$

In this expression $\bar{C}_{ij}$, the average concentration of isotope i during time period j, has replaced the following factor:

$(X/Q)\ Q_{ij}$

The C_{ij}'s are those determined by equations (4) and (6).

The assumed breathing rates, B, are shown on Table 15A-1.

Environmental Consequences Due to the Operation of the Postaccident Sampling Facility

Section 9.5.10 discusses the Postaccident Sampling Facility (PASF) at Sequoyah Nuclear Plant (SQN). The "worst case" offsite doses resulting from the operation of the PASF are calculated in this section. NUREG-0737 recommends the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guide 1.4 (i.e., 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant). For this "worst case" analysis, the primary system remains intact and pressurized; consequently, the noble gases will stay in the reactor coolant and, in addition, there is no dilution by the Emergency Core Cooling System (ECCS) which would occur during a LOCA.

The assumptions used in determining the offsite doses are listed below:

1. The initial reactor coolant sample is drawn one hour postaccident, followed by daily samples for the next seven days. After the first week, one sample a week is taken. The offsite doses are based on the samples taken during the first 30 days after the accident.

2. A core release of 100 percent of the noble gases and 50 percent of the iodines is assumed to be mixed with the reactor coolant (94,350 gallons).

3. The postaccident sample collector drain tank is assumed to accumulate 60 gallons of water during each sampling mission. Twenty of the 60 gallons are reactor coolant and the rest are demineralized flush water.

4. All of the noble gases contained in the reactor coolant that are held in the sample collector drain tank are liberated to the tank's air space and exhausted to the atmosphere.

5. The vapor-water partition coefficients of the iodines in the sample collector drain tank are based on Reference 12.

6. All of the iodine liberated to the tank's air space are exhausted to the atmosphere through charcoal filters.

7. A charcoal filter iodine removal efficiency of 95 percent is used for both the organic and inorganic iodines.

8. Two-hour atmospheric dilution factors from FSAR Table 15A-2 are used for each release since each release occurs over short periods of time.

9. Radioactivity decay of the noble gases and iodines is taken into account for the reactor coolant only; there is no holdup in the sample collector drain tank assumed.

Discussion of Results

The offsite doses at the exclusion area boundary resulting from the operation of PASF for this "worst-case" evaluation are 15 rem deep dose equivalent, 9.4 rem beta, and 16.6 rem thyroid; at the low population zone, they are 1.8 rem deep dose equivalent, 1.2 rem beta, and 2.2 rem thyroid. It should

be noted that doses this large cannot occur simultaneously with the doses calculated for a Regulatory Guide 1.4 LOCA. The post-LOCA operation of the PASF would result in much lower offsite doses since most of the noble gases and other volatile nuclides would be released in the primary containment before they could get into the sample collector drain tank, FSAR 15.5.8, Reference 18. The doses resulting from the operation of the PASF are within the limits prescribed in 10 CFR 100.

Plant Accessibility Post LOCA

The Sequoyah Nuclear Plant was designed so that access is generally not required outside the main control room for 30 days after an accident. Access to areas within the auxiliary building and structures away from the main complex for the performance of specified tasks are examined individually. Approval of such missions is based on control room personnel performing the required task and not exceeding the limit of 10CFR50, Appendix A, General Design Criterion 19. SQN-DC-V-21.0 (Reference 17) addresses the design basis post accident mission regulatory requirements, provides the basis for determining which design basis accidents would require a post accident mission following a design basis event, and establishes the criteria used to determine when a post accident mission dose evaluation is required outside the main control room.

A study was performed to evaluate the Auxiliary Building shielding design to determine accessibility after an accident and to be used to calculate doses to equipment in the Auxiliary Building subsequent to an accident.

The study assumed the design LOCA with subsequent recirculation of contaminated water from the containment sump through the residual heat removal (RHR), containment spray, and safety injection systems. The source terms used for this accident are the Regulatory Guide 1.4 sources diluted by the full volume of injection water available plus the melted ice.

The dose rates through the Reactor Shield Building from activity released into the containment atmosphere were also calculated.

The sources of available water are:

Refueling Water Storage Tank	340,000 gallons
4 SIS Accumulators	27,676 gallons (negligible)
Ice Condenser	357,314 gallons

The reactor coolant volume is 12,612 cubic feet or 94,249 gallons; therefore, the total volume of water is 829,064 gallons.

For source volumes containing contaminated liquid, the activity in the liquid was assumed to consist of:

1. 100 percent of the noble gases.
2. 50 percent of the halogens with noble gas daughter products.
3. One percent of the solid fission products.

Dose rates were calculated using these sets separately as the source term. The results of this analysis are reported as the sum of these separate dose rates. Source terms used in this analysis are provided in FSAR Table 15.5.3-8. For source volume containing gaseous activity, source terms were calculated for 100 percent of the noble gases and 25 percent of the halogens. Post LOCA radiation doses in the Reactor Building are addressed in FSAR 15.5.8., Reference 17.

Flow diagrams, equipment, and field-routed piping layouts were reviewed to determine the flow paths which might be used after an accident and to determine the volume and physical locations of contaminated fluids in the Auxiliary Building. The layout of the shield walls and equipment within the rooms were conservatively modeled and source terms were calculated at various times after an accident. Dose rates were then calculated at several positions in the Auxiliary Building with respect to the contained sources and at various times after an accident. The locations where dose rates were calculated were chosen to conservatively calculate the dose rates in corridors outside equipment cubicles, in adjacent rooms, and within the equipment cubicles. The source volume in each pump room was modeled as a volume of water in a rectangular box equivalent to the volume of water contained in the pump and piping in that room. The smallest wall thickness of the pipes carrying significant volumes of contaminated liquid was used as an iron cladding around the source. Heat exchangers were modeled as equivalent volume rectangular boxes with a square cross-section and with an iron cladding equal to the shell thickness. Tanks are modeled similarly to heat exchangers. All of the piping in a pipe chase or pipe shaft was modeled as a single equivalent volume of water in a rectangular box extending the length of the pipe chase or shaft with the minimum pipe wall thickness as an iron cladding around the source. Post LOCA radiation doses for the Auxiliary Building are addressed in FSAR 15.5.8., Reference 17.

Of the vital areas, none is near an area containing postaccident sources. The discussion of the exposure of control room operators is therefore still valid without modification. The Technical Support Center (TSC) is located in the Control Building on the same floor as the control room, hence within the habitability zone. The exposure in the TSC should therefore be the same as in the control room. Table 15.5.3-7 provides the integrated 30-day operator dose. The continuous occupancy exposure rate in those two areas is less than the 15 mrem per hour limit suggested in NUREG 0578. Note that the calculated dose in the Control Building habitability zone is due almost entirely to airborne contamination of the zone and neighboring spaces due to the introduction of filtered outside air for maintaining a pressurized condition in the habitability zone. The contribution to the operator whole-body dose from the plume released at the containment vent is negligible.

The shutdown board room may require occupancy briefly to accomplish a single control function. Access restriction would be due only to airborne activity. Analyses shows that required access can be obtained with dose consequences within the limits of GDC19. TVA has also determined that it may be necessary to perform certain actions in the auxiliary building following a LOCA. Analyses have shown that required access can be obtained for all missions currently examined with dose consequences within the limits of GDC 19. Future missions will be analyzed as the need to perform such missions are identified. In no event will a mission be approved which exceeds the limits of GDC 19. FSAR Section 15.5.8, Reference 17, addresses the GDC 19 analyses and plant procedures associated with these analyses.

The RSGs have no effect on the assumptions and inputs used in the LOCA environmental consequences analyses. Therefore, the conclusions of the existing analyses are equally applicable to Sequoyah Unit 1 with the RSGs.

15.5.4 Environmental Consequences of a Postulated Steam Line Break

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented assuming steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. Parameters used in the analysis are listed in Table 15.5.4-1.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated steam line break.

1. Prior to the accident, an equilibrium activity of fission products exists in the primary and the secondary systems due to a primary to secondary leakage in steam generators.

2. Offsite power is lost, main steam condensers are not available for steam dump.

3. Eight hours after the accident, the residual heat removal system starts operation to cool down the plant.

4. The primary to secondary leakage is evenly distributed in steam generators.

5. After eight hours following the accident, no steam and activity are released to the environment.

6. No air ejector release and no steam generator blowdown during the accident.

7. Initial primary coolant activity (iodine and noble gas) is consistent with the model discussed in Appendix 15D. This model is consistent with Technical Specification limits on primary coolant activity limits.

8. The analysis assumes an iodine spike with two separate cases considered. For one case there is assumed to be a pre-existing iodine spike. For the second case the initial primary coolant iodine concentration is at the limit set in the Technical Specifications for equilibrium operation and an iodine spike is assumed to be initiated by the reactor trip associated with the event. The iodine spiking models are described in Appendix 15D.

9. Initial secondary coolant activity (iodine) is consistent with the model discussed in Appendix 15D. This model is consistent with Technical Specification limits on primary coolant activity limits. There is no noble gas activity in the secondary coolant.

10. The iodine partition factor is $\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}} = 0.01$ in the good steam generators

11. During the postulated accident, iodine transferred to the secondary side in the three good steam generators is uniformly mixed with the water in the steam generators.

12. In the affected steam generator, all the water boils off and releases through the break immediately after the accident. The partition factor for the iodine released is assumed to be 1.0. After this initial release, further iodine is released due to the primary to secondary leakage in the affected steam generator. A partition factor of 1.0 is also assumed for this iodine release.

13. The primary pressure remains constant at 2235 psig for 0-2 hours and decreases linearly to atmospheric from 2235 psig during the period of 2-8 hours.

14. The 0-2 and 2-8 hour atmospheric diffusion factors given in Appendix 15A and the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable.

The resulting doses for the case with the accident-initiated iodine spike are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	5.4 rem	0.073 rem	0.11 rem
Low Population Zone	0.69 rem	0.01 rem	0.014 rem
Control Room	0.22 rem	.046 rem	0.009 rem

These doses are well within the limits of 25 rem whole body and 300 rem thyroid defined in 10 CFR 100 and the control room dose limits of 5 rem whole body, 30 rem thyroid, and 30 rem skin defined in Section 6.4 of the Standard Review Plan.

[For information only, the TEDE doses are 0.25 rem at the EAB, 0.032 at the LPZ and 0.053 rem in the control room.]

For the case with the pre-existing iodine spike the doses are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	5.3 rem	0.061 rem	0.09 rem
Low Population Zone	0.67 rem	0.008 rem	0.011 rem
Control Room	0.20 rem	0.037 rem	0.008 rem

These doses are well within the limits of 25 rem whole body and 300 rem thyroid defined in 10 CFR 100 and the control room dose limits of 5 rem whole body, 30 rem thyroid, and 30 rem skin defined in Section 6.4 of the Standard Review Plan.

[For information only, the TEDE doses are 0.24 rem at the EAB, 0.03 at the LPZ and 0.043 rem in the control room.]

Sequoyah Nuclear Plant has implemented alternate steam generator tube plugging criteria as provided for in NRC Generic Letter (GL) 95-05 (Reference 22). The GL allows an increase in primary to secondary leakage during a main steam line break based on the offsite and control room operator dose staying well within the guidelines of 10 CFR Part 100 and 10 CFR Part 50, GDC 19,

respectively. In the analyses discussed earlier in this section, the input parameters were set and an offsite and operator dose were calculated. For this evaluation, the EAB dose was set at 30 rem thyroid based on an accident initiated iodine spike. The allowable primary to secondary leakage rate was then back calculated. Once this leakage rate was established, the other dose calculations were performed based on the calculated leakage rate. The allowable primary to secondary leakage rate for the steam generator on the faulted loop was determined to be 3.7 gpm. Table 15.5.4-1 provides a summary of the major parameters used in this analysis. The allowable leakage rate is a function of the assumed reactor coolant activity. The allowable leakage rate may be increased, if the allowable reactor coolant activity is decreased to limit the offsite and control room operator doses to the limits calculated in this analysis as specified in GL 95-05.

With the application of the 3.7 gpm primary to secondary leak rate in the faulted loop, the resulting doses for the case with the accident-initiated iodine spike are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	30.0 rem	0.67 rem	0.97 rem
Low Population Zone	5.0 rem	0.09 rem	0.14 rem
Control Room	2.1 rem	0.52 rem	0.084 rem

[For information only, the TEDE doses are 1.7 rem at the EAB, 0.26 at the LPZ and 0.59 rem in the control room.]

For the case with the pre-existing iodine spike the doses are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	25.8 rem	0.25 rem	0.38 rem
Low Population Zone	4.0 rem	0.034 rem	0.051 rem
Control Room	1.6 rem	0.20 rem	0.067 rem

[For information only, the TEDE doses are 1.1 rem at the EAB, 0.16 at the LPZ and 0.24 rem in the control room.]

The parameters that effect the environmental consequences of a steam line break are not adversely affected by the steam generator. Therefore, all acceptance criteria continue to be met on Sequoyah Unit 1 with the RSGs.

15.5.5 Environmental Consequences of a Postulated Steam Generator Tube Rupture

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generators. A conservative analysis of the postulated steam generator tube rupture assumes the loss of offsite power and hence involves the release of steam from the secondary system. A conservative analysis of the potential offsite doses resulting from this accident is presented assuming steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. Parameters used in the analysis are listed in Table 15.5.5-1.

The following conservative assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated steam generator tube rupture:

1. Prior to the accident, an equilibrium activity of fission products exists in the primary and the secondary systems due to primary to secondary leakage in steam generators.

2. Offsite power is lost, main steam condensers are not available for steam dump.

3. Eight hours after the accident, the residual heat removal system starts operation to cool down the plant.

4. The primary to secondary leakage is evenly distributed in the steam generators.

5. After eight hours following the accident, no steam and activity are released to the environment.

6. No air ejector release and no steam generator blowdown during the accident.

7. Initial primary coolant activity (iodine and noble gas) is consistent with the model discussed in Appendix 15D. This model is consistent with Technical Specification limits on primary coolant activity limits.

8. The analysis assumes an iodine spike with two separate cases considered. For one case there is assumed to be a pre-existing iodine spike. For the second case the initial primary coolant iodine concentration is at the limit set in the Technical Specifications for equilibrium operation and an iodine spike is assumed to be initiated by the reactor trip associated with the event. The iodine spiking models are described in Appendix 15D.

9. Initial secondary coolant activity (iodine) is consistent with the model discussed in Appendix 15D. This model is consistent with Technical Specification limits on primary coolant activity limits. There is no noble gas activity in the secondary coolant.

10. The iodine in the flashed portion of the break flow entering the ruptured steam generator is assumed to be released directly to the environment.

11. With the exception of the iodine in the flashed portion of the break flow entering the ruptured steam generator, the iodine entering the secondary side is assumed to uniformly mix in the steam generator water.

12. The iodine partition factor is $\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}} = 0.01$ in the steam generators.

13. Steam release to atmosphere and the associated activity release from the nondefective steam generators is terminated at eight hours after the accident when the residual heat removal system takes over in cooling down the plant.

14. Thirty minutes after the accident, the pressure between the defective steam generator and the primary system is equalized. The defective unit is isolated. No steam and fission product activities are released from the defective steam generator thereafter.

15. The 0-2 and 2-8 hour atmospheric diffusion factors given in Appendix 15A and the 0-8 hour breathing rate of 3.47 x 10^{-4} m^3/sec are applicable.

The resulting doses for the case with the accident-initiated iodine spike are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	18.5 rem	1.6 rem	2.8 rem
Low Population Zone	2.3 rem	0.19 rem	0.4 rem

These doses are well within the limits of 25 rem whole body and 300 rem thyroid defined in 10 CFR 100.

[For information only, the TEDE doses are 2.2 rem at the EAB, 0.26 at the LPZ.]

For the case with the pre-existing iodine spike the doses are:

	Thyroid	Whole Body	Skin
Exclusion Area Boundary	41.1 rem	1.4 rem	2.6 rem
Low Population Zone	5.0 rem	0.16 rem	0.4 rem

These doses are well within the limits of 25 rem whole body and 300 rem thyroid defined in 10 CFR 100.

[For information only, the TEDE doses are 2.7 rem at the EAB, 0.32 at the LPZ.]

The safety evaluation for the replacement steam generators concludes that parameters important to the consequences of this event are not adversely affected by the RSG.

The doses given above are based on operators stopping the steam release from the ruptured steam generator by 30 minutes. Should operators take 45 minutes to stop the steam release from the ruptured steam generator, then the maximum offsite doses will be 62 rem to the thyroid and 2.4 rem whole body.

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

Two analyses of a postulated fuel handling accident are performed: 1) an event occurring in the Auxiliary Building and 2) an event occurring inside the primary containment. The analyses use methodology from Regulatory Guide 1.183 (Reference 13) associated with the use of alternative source term modeling and do not use the dose calculation methodology presented in Appendix 15A. The parameters used for each of these analyses are listed in Table 15.5.6-1.

The basic analysis assumptions for the fuel handling accident in the Auxiliary Building are:

1. The accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pool is taken into account.

2. All rods in one assembly rupture. Additionally, the damaged fuel assembly is assumed to contain 24 Tritium Producing Burnable Absorber Rods (TPBARs) and all of these are assumed to rupture.

3. The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. A radial peaking factor of 1.70 is used.

4. The maximum fuel rod pressurization is 1200 psi.

5. The minimum water depth between the top of the damaged fuel rods and the spent fuel pool surface is 23 feet.

6. All of the gap activity in the damaged rods is released to the spent fuel pool and consists of 5 percent of the total noble gases other than Kr-85, 10 percent of the Kr-85, 5 percent of the total radioactive iodine other than I-131, and 8 percent of the I-131, in the rods at the time of the accident.

7. Noble gases released to the spent fuel pool do not experience retention in the water pool.

8. The iodine gap inventory is composed of inorganic species (99.85 percent) and organic species (0.15 percent).

9. The spent fuel pool decontamination factor for iodine is 200.

10. No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmospheric.

11. All of the tritium contained in the failed TPBARs is conservatively assumed to be immediately released to the water pool. No credit is taken for retention of the tritium in the water. The tritium is assumed to enter the environment in the form of tritiated water vapor rather than in the elemental form.

12. The activity released from the pool (iodine, noble gas, and tritium) is all assumed to be released to the environment over a two-hour period.

If a fuel handling accident should occur inside the primary containment, the analysis model differs from the above. During refueling operations, the containment may be ventilated through a single train of Reactor Building Purge Ventilation System (RBPVS). The RBPVS is described in Section 9.4.7. The containment equipment hatch and personnel air locks may be open during fuel handling operations, and although the purge line would be quickly isolated, activity release is assumed to continue through these open penetrations.

The dose analysis for a fuel handling accident inside the reactor containment is the same as for the fuel handling accident inside the Auxiliary Building, with the following exceptions:

1. Initially, activity is assumed to be exhausted by one train of containment purge directly to the environment at a rate of 16,000 cfm.

2. The containment purge is assumed to be isolated in 30 seconds by the purge line radiation monitor. This response time bounds the demonstrated response time; Reference 14.

3. No iodine removal credit is taken for the Reactor Building Purge Ventilation System filters.

4. The volume of containment assumed to participate in mixing is 32,550 ft^3 (less than 5% of the containment free volume).

5. The efficiency of the refueling canal perimeter exhaust is assumed to be 100%, i.e., none of the activity released from the pool is assumed to escape the mixing volume.

6. After the containment purge line is isolated, the activity remaining in the containment is assumed to be released by way of the penetrations. All activity is assumed to be released within two hours of the fuel damage occurring.

Doses were determined at the exclusion area boundary (EAB) and at the low population zone boundary (LZP) for the two-hour interval over which releases are assumed to take place and in the control room for an extended period of time after termination of releases in order to address the continued presence of activity in the control room atmosphere.

The accident doses were calculated using the dose model consistent with the use of the alternate source term methodology (Regulatory Guide 1.183) and are reported as Total Effective Dose Equivalent (TEDE).

The TEDE dose is the sum of the Committed Effective Dose Equivalent (CEDE) and the Effective Dose Equivalent (EDE) which are calculated using the following equations:

$$D_{CEDE} = (A)(X/Q)(BR)(DCF_{CEDE})$$

$$D_{EDE} = (A)(X/Q)(DCF_{EDE})$$

where: A = Activity of the nuclide released (Ci)
X/Q = atmospheric dispersion factor (sec/m^3)
BR - breathing rate (m^3/sec)
DCF_{CEDE} = CEDE dose conversion factor (rem/Ci inhaled)
DCF_{EDE} = dose conversion factor (rem-m^3/Ci-s)

	FHA in Auxiliary Building	FHA inside Primary Containment
Exclusion Area Boundary	4.5 rem TEDE	4.5 rem TEDE
Low Population Zone Outer Boundary	0.8 rem TEDE	0.8 rem TEDE
Control Room	4.1 rem TEDE	4.2 rem TEDE

These doses are within the dose acceptance limits of 6.3 rem TEDE for offsite doses (Regulatory Guide 1.183) and 5.0 rem TEDE for the control room dose (GDC 19) associated with the use of alternate source term methodology.

15.5.7 Environmental Consequences of a Postulated Rod Ejection Accident

The consequences of a postulated rod ejection accident are bounded by the results of the loss-of-coolant accident analysis evaluated in Section 15.5.3.

15.5.8 References

1. Not used.

2. Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure," Division of Reactor Standards, U.S. Atomic Energy Commission, March 23, 1972.

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4. D. D. Malinowski, "Iodine Removal in the Ice Condenser System," WCAP-7426, April 1970.

5. NAA-SR-10100, Conventional Buildings for Reactor Containment.

6. ORNL-NSIC-4, Behavior of Iodine in Reactor Containment Systems, February 1965.

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8. Not used.

9. James Halitsky, Jack Golden, Paul Halpern, and Pal Wu "Wind Tunnel Tests of Gas Diffusion from a Leak in the Shell of a Nuclear Power Reactor and from a Nearby Stack," New York University College of Engineering Research Division, Department of Meteorology and Oceanography Geophysical Sciences Laboratory Report No. 63-2, (1963).

10. K. G. Murphy and K. M. Campe "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," (13th NRC Air Cleaning Conference Proceedings, August 1974, Vol 1, pages 401-430).

11. Ramsdell, J.V. and C.A. Simonen, “Atmospheric Relative Concentration in Building Wakes. Prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, PNL-10521, NUREG/CR-6331, Rev. 1, May, 1997.

12. Y. Nishizawa, et al., "Vapor Water Partition Coefficient of Iodine and Organic Iodines," ORNL-tr-2255, Nippon Genshiryoku Gakkaishi, 1969.

13. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, July 2000.

14. Safety Assessment/Evaluation (10CFR50.59) [B38 931027 802].

15. Not used.

16. NE Calculation SQN-APS3-067, "Offsite and Control Room Operator Doses Due to a MHA LOCA With Maximum Allowable Annulus Inleakage."

17. SQN-DC-V-21.0, "Environmental Design Criteria."

18. NE Calculation SQN-TI- RPS-158, "Maximum Offsite Doses Due to the Post Accident Sampling Facility."

19. Not used.

20. Not used.

21. NE Calculation SQN-APS2-44, "Annulus Pressure Transient Analysis Following A LOCA."

22. NRC Generic Letter 95-05, “Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.”

23. Document No. 77-5016198, Framatome ANP, Replacement Steam Generator Report for Sequoyah Unit 1.”

TABLE 15.5.1-1

PARAMETERS USED IN LOSS OF A.C. POWER ANALYSIS

Steam generator tube leak rate during accident	150 gpd per SG
Offsite power	Lost
Reactor Coolant Activity	
Noble Gas Activity Prior to Accident	See Appendix 15D
Iodine Activity Prior to Accident (without spike)	See Appendix 15D
Iodine Activity (pre-existing spike)	21 μCi/g D.E. I-131 (see Appendix 15D)
Equilibrium Iodine Appearance Rates	See Appendix 15D
Increase in Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rates
Duration of Accident-Initiated Spike	1 hour
Secondary Coolant Activity	
Iodine Activity Prior to Accident	0.1 μCi/g D.E. I-131 (see Appendix 15D)
Noble Gas Activity	None
Iodine Chemical Form	Elemental
Iodine partition factor in steam generator during accident	0.01
Duration of plant cooldown by Secondary System after accident	8 hr
Steam release from 4 steam generators	487,000 lb (0-2 hr) 1,090,000 lb (2-8 hr)
Meteorology	Accident (see Appendix 15A)

(1) Based on operating experience of Westinghouse PWR's.

TABLE 15.5.2-1

PARAMETERS USED IN WASTE GAS DECAY TANK RUPTURE ANALYSIS

Plant load factor	1.0
Fuel defects	1 percent
Activity released from GWPS	Noble gas inventory of one Reactor Coolant System volume[1]
Stripping fraction in volume control tank	1.0
Iodine partition factor in volume control tank	0.01
Time of accident	Immediately after shutdown at end of equilibrium core cycle
Meteorology	Accident (see Appendix 15A)

(1) Iodine releases are based on plant operating procedures (degassing of volume control tank every 3 hours for 21 hours after shutdown). At 21 hours after shutdown the summation of the product of the iodine isotopic inventories times their respective dose conversion factor is a maximum.

TABLE 15.5.2-2

WASTE GAS DECAY TANK INVENTORY (one unit)

(Regulatory Guide 1.24 (1972) Analysis)

Isotope	Activity (Curies)
Xe-133	8.46×10^{4}
Xe-133m	9.35×10^{2}
Xe-135	2.13×10^{3}
Xe-135m	4.80×10^{1}
Xe-138	1.70×10^{2}
Kr-85	4.30×10^{3}
Kr-85m	5.83×10^{2}
Kr-87	3.25×10^{2}
Kr-88	1.02×10^{3}
I-131	5.49×10^{-2}
I-132	4.00×10^{-5}
I-133	4.73×10^{-2}
I-134	- -
I-135	7.09×10^{-3}

TABLE 15.5.3-1 (Sheet 1)

PARAMETERS USED IN LOCA ANALYSES

	Regulatory Guide 1.4 Analysis
Core thermal power	3582 MWt
Primary containment free volume	1.241×10^6 ft^3
Upper primary containment free volume	7.16×10^5 ft^3
Lower primary containment free volume	4.0×10^5 ft^3
Ice condenser free volume	1.25×10^5 ft^3
Annulus free volume	3.75×10^5 ft^3
Primary containment deck fan flow rate	40,000 cfm
Number of deck fans assumed operating	1 of 2
Activity released to primary containment and available for release	
noble gases	100 percent of core inventory
iodines	25 percent of core inventory
Plateout of iodine activity released to primary containment	50 percent
Form of iodine activity in primary containment available for release	
elemental iodine	91 percent
methyl iodine	4 percent
particulate iodine	5 percent
Ice condenser removal efficiency for elemental iodine	See Table 15.5.3-2
Primary containment leak rate	0.25 percent per day (0-24 hours)
	0.125 percent per day (1-30 days)

TABLE 15.5.3-1 (Sheet 2)

PARAMETERS USED IN LOCA ANALYSES

	Regulatory Guide 1.4 Analysis
Percent of primary containment leakage to Auxiliary Building	25 percent
ABGTS filter efficiencies	
elemental iodine	95 percent
methyl iodine	95 percent
particulate iodine	95 percent
Delay time of activity in Auxiliary Building	0.3 hour
Time ABGTS filters are bypassed following LOCA	5 minutes
EGTS total intake flow	4000 cfm
Percent of annulus free volume available for mixing of activity	50 percent
EGTS exhaust flow rate	See Table 15.5.3-3
Number of EGTS air cleanup units assumed operating	1 of 2
EGTS filter efficiencies	
elemental iodine	95 percent
methyl iodine	95 percent
particulate iodine	95 percent
Percent of primary containment leakage to the annulus to EGTS air cleanup units suction	100 percent
Meteorology	Accident (see Appendix 15A)

TABLE 15.5.3-2

ICE CONDENSER IODINE REMOVAL EFFICIENCY[1]

Time Interval Post LOCA (Hours)	Fractional Iodine Removal Efficiency
0.0 to 0.106	0.96
0.106 to 0.133	0.84
0.133 to 0.244	0.71
0.244 to 0.383	0.67
0.383 to 0.522	0.64
0.522 to 0.578	0.62
0.578 to 0.606	0.30
0.606 to 720	0.0

[1] The ice condenser removal efficiencies given in the above table are used for the conservative Regulatory Guide 1.4 analyses. The inlet steam-air mixture coming into the ice condenser is greater than 90 percent steam by volume initially due to the delaying of the operation of the deck fans. Without the delay of operation of the deck fans the amount of steam by volume in the inlet mixture initially would be much lower and the ice condenser iodine removal efficiencies would be reduced. The zero value after 0.606 hours results from all of the ice being melted.

TABLE 15.5.3-3

EMERGENCY GAS TREATMENT SYSTEM FLOW RATES: LARGE BREAK LOCA

Time Interval Post LOCA (Hours)	Recirculation Flow Rate (cfm)	Exhaust Flow Rate (cfm)
0 to 0.00833	0	0
0.00833 to 0.0128	3600	0
0.0128 to 0.01306	3525	75
0.01306 to 0.01333	3060	540
0.01333 to 0.01361	2387	1213
0.01361 to 0.01389	1662	1938
0.01389 to 0.01417	967	2633
0.01417 to 0.0144	340	3260
0.0144 to 0.02083	0	3600
0.02083 to 0.025	295	3305
0.025 to 0.0333	266	3334
0.0333 to 0.05	474	3126
0.05 to 0.0667	913	2687
0.0667 to 0.106	1337	2263
0.106 to 0.133	1857	1743
0.133 to 0.16111	2427	1173
0.16111 to 0.244	2647	953
0.244 to 0.3111	3600	0
0.3111 to 0.33889	3455	145
0.33889 to 0.383	3346	254
0.383 to 0.41667	3253	347
0.41667 to 720.0	3213	387

TABLE 15.5.3-4

OFFSITE DOSES FROM LOSS OF COOLANT ACCIDENT

	Thyroid Dose (Rem)	
	Site Boundary (0-2 hours) 556 meters	Low Population Zone (0-30 days) 4828 meters
Regulatory Guide 1.4 Analysis (Ref. 16)	83.1	16.5
10 CFR 100 Guidelines	300	300

	Gamma and Beta Doses (Rem)			
	Site Boundary (0-2 hours) 556 meters		Low Population Zone (0-30 days) 4828 meters	
	Gamma Dose	Beta Dose	Gamma Dose	Beta Dose
Regulatory Guide 1.4 Analysis (Ref. 16)	7.68	4.52	1.5	1.4
10 CFR 100 Guidelines	25*		25*	

*Whole Body Dose

[For information only, the TEDE doses are
9.8 rem at the EAB and 1.9 rem at the LPZ]

TABLE 15.5.3-5

LOCA SOURCE TERM ACTIVITIES FOR CONTROL ROOM AND OFFSITE DOSE ANALYSES

(Reference 16)

NUCLIDE			CONTAINMENT SOURCE ACTIVITY
1	Kr	83m (1)	5.960E+04
2	Kr	85m	1.240E+05
3	Kr	85	5.350E+03
4	Kr	87	2.490E+05
5	Kr	88	3.450E+05
6	Kr	89	4.290E+05
7	Xe	131m	5.430E+03
8	Xe	133m	3.190E+04
9	Xe	133	9.920E+05
10	Xe	135m	2.100E+05
11	Xe	135	3.330E+05
12	Xe	138	8.640E+05
13	I	131	4.700E+05
14	I	132	6.890E+05
15	I	133	9.700E+05
16	I	134	1.080E+06
17	I	135	9.260E+05
18	I* (2)	131	1.960E+04
19	I*	132	2.870E+04
20	I*	133	4.040E+04
21	I*	134	4.480E+04
22	I*	135	3.860E+04

The above activities represent 1 of 193 nuclear fuel assemblies within the reactor core at 1000 EFPD and having a U235 enrichment of 5%. These activities are in accordance with Regulatory Guide 1.4 consisting of 100% of the Noble Gases and 25% of the Iodine. The Iodine inventory consist of 96% elemental and 4% methyl.

(1) m denotes metastable state for the given isotope.
(2) I* denotes an organic Iodine.

TABLE 15.5.3-6

ATMOSPHERIC DILUTION FACTORS AT CONTROL BUILDING

Time Period (hour)	Diluting Factor[1] (sec/m^3)
0 - 2	5.63E-04
2 - 8	3.78E-04
8 - 24	1.12E-04
24 - 96	9.38E-05
96 - 720	6.96E-05

[1]Onsite meteorology for April 2, 1971 - March 31, 1972.

TABLE 15.5.3-7

CONTROL ROOM PERSONNEL DOSE FOR DBA POST ACCIDENT PERIOD

(Reference 16)

Source	Personnel Dose With Immediate Control Room Isolation		
	Whole Body Gamma Dose (rem)*	Beta Dose (rem)*	Thyroid (rem)
Control room airborne activity	0.65	5.73	3.28
External cloud shine	0.45	0	0
Ingress - Egress	0.027	0.057	0.51
Total	1.2	5.8	3.8

*Includes occupancy factor: 100 percent occupancy 0-24 hours
60 percent occupancy 1-4 days
40 percent occupancy 4-30 days

[For Information only, the TEDE dose is 1.4 rem]

TABLE 15.5.3-8
(Sheet 1)

CONTAINMENT ATMOSPHERE AND SUMP
LOCA SOURCE TERM ACTIVITIES

NUCLIDE	CI PER ASSEMBLY	NUCLIDE	CI PER ASSEMBLY	NUCLIDE	CI PER ASSEMBLY
1 KRM 83	5.960E+04	21 I 134	1.080E+06	42 SE 84	1.080E+05
2 KRM 85	1.240E+05	22 I 135	9.260E+05	43 SR 89	4.840E+05
3 KR 85	5.350E+03	23 IM 136	2.070E+05	44 SR 90	4.630E+04
4 KR 87	2.490E+05	24 BR 83	5.960E+04	45 SR 91	6.030E+05
5 KR 88	3.450E+05	25 BRM 84	2.950E+03	46 SR 92	6.410E+05
6 KR 89	4.290E+05	26 BR 84	1.110E+05	47 SR 93	7.240E+05
7 KR 90	4.590E+05	27 BR 85	1.230E+05	48 SR 94	7.220E+05
8 XEM 131	5.430E+03	28 BR 87	1.960E+05	49 Y 90	4.910E+04
9 XEM 133	3.190E+04	29 CS 134	8.610E+04	50 YM 91	3.500E+05
10 XE 133	9.920E+05	30 CS 135	0.0	51 Y 91	6.270E+05
11 XEM 135	2.100E+05	31 CS 136	3.050E+04	52 Y 92	6.460E+05
12 XE 135	3.330E+05	32 CS 137	6.040E+04	53 Y 93	4.910E+05
13 XE 137	9.230E+05	33 CS 138	9.360E+05	54 Y 94	7.800E+05
14 XE 138	8.640E+05	34 CS 139	8.720E+05	55 Y 95	8.110E+05
15 XE 139	6.410E+05	35 CS 140	7.830E+05	56 Y 96	7.830E+05
16 XE 140	4.460E+05	36 CS 141	5.970E+05	57 ZR 95	8.660E+05
17 I 130	9.590E+03	37 RB 88	3.540E+05	58 ZR 97	8.350E+05
18 I 131	4.700E+05	38 RB 89	4.620E+05	59 NB 95	8.740E+05
19 I 132	6.890E+05	39 RBM 90	1.340E+05	60 NBM 97	7.920E+05
20 I 133	9.700E+05	40 RB 90	4.250E+05	61 NB 97	8.400E+05
		41 RB 91	5.620E+05	62 MO 99	9.220E+05

The above activities represent 1 of 193 nuclear fuel assemblies within the reactor core at 1000 EFPD and having a U235 enrichment of 5%. Reference 17.

TABLE 15.5.3-8
(Sheet 2)

CONTAINMENT ATMOSPHERE AND SUMP
LOCA SOURCE TERM ACTIVITIES

NUCLIDE	CI PER ASSEMBLY	NUCLIDE	CI PER ASSEMBLY	NUCLIDE	CI PER ASSEMBLY
63 TCM 99	8.110E+05	84 TEM 129	3.010E+04	105 PR 143	7.460E+05
64 TC 99	0.0	85 TE 129	1.490E+05	106 PR 144	6.730E+05
65 TC 101	8.360E+05	86 TEM 131	9.620E+04	107 PR 145	5.210E+05
66 RU 103	7.660E+05	87 TE 131	4.140E+05	108 NP 239	9.690E+06
67 RU 106	5.200E+05	88 TE 132	7.050E+05	109 I* 130	4.000E+02
68 RU 106	2.590E+05	89 TEM 133	4.520E+05	110 I* 131	1.960E+04
69 RU 107	2.990E+05	90 TE 133	5.490E+05	111 I* 132	2.870E+04
70 REM 103	7.650E+05	91 TE 134	8.950E+05	112 I* 133	4.040E+04
71 RHM 105	1.480E+05	92 BAM 137	5.740E+04	113 I* 134	4.480E+04
72 RH 105	4.950E+05	93 BA 139	8.940E+05	114 I* 135	3.860E+04
73 RH 106	2.7600E+05	94 BA 140	8.980E+05	115 IM* 136	8.640E+03
74 RH 107	2.990E+05	95 BA 141	8.100E+05		
75 SN 130	1.650E+05	96 BA 142	7.710E+05		
76 SB 127	4.170E+04	97 LA 140	9.290E+05		
77 SB 129	1.570E+05	98 LA 141	8.170E+05		
78 SBM 130	2.210E+05	99 LA 142	7.990E+05		
79 SB 130	5.200E+04	100 LA 143	7.590E+05		
80 SB 133	3.200E+05	101 CE 141	8.250E+05		
81 TEM 125	1.000E+02	102 CE 143	7.650E+05		
82 TEM 127	6.870E+03	103 CE 144	6.690E+05		
83 TE 127	4.110E+04	104 CE 145	5.200E+05		

The above activities represent 1 of 193 nuclear fuel assemblies within the reactor core at 1000 EFPD and having a U235 enrichment of 5%. Reference 17.

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TABLE 15.5.4-1

PARAMETERS USED IN STEAM LINE BREAK ANALYSIS

Reactor Coolant Activity	
Noble Gas Activity Prior to Accident	See Appendix 15D
Iodine Activity Prior to Accident (without spike)	See Appendix 15D
Iodine Activity (pre-existing spike)	21 μCi/g D.E. I-131 (see Appendix 15D)
Equilibrium Iodine Appearance Rates	See Appendix 15D
Increase in Iodine Appearance Rate Increase Due to the Accident-Initiated Spike	500 times equilibrium rates
Duration of Accident-Initiated Spike	1 hour
Secondary Coolant Activity	
Iodine Activity Prior to Accident	0.1 μCi/g D.E. I-131 (see Appendix 15D)
Noble Gas Activity	None
Iodine Chemical Form	Elemental
Primary to Secondary Leak Rate (per steam generator)	150 gpd
Steam Release from Intact Steam Generators	
0-2 hours	4.79E5 lb
2-8 hours	1.03E6 lb
>8 hours	none
Blowdown of Faulted Steam Generator to Release the Initial Inventory of Activity	1.48E5 lb over 10 minutes
Intact Steam Generator Iodine Partition Factor	0.01
Faulted Steam Generator Iodine Partition Factor	1.0
Offsite Power	Lost
Atmospheric Dispersion Factor	Accident (see Appendix 15A)
Breathing Rate	3.74E-4m^3/sec
Control Room Parameters	
Volume	2.6E5 cubic feet
Filtered intake flow	1000 cfm
Filtered recirculation flow	2600 cfm
Filter efficiency for iodine	95%
Unfiltered inleakage	51 cfm
Atmospheric dispersion factor (X/Q)	
0-2 hours	1.93E-3 sec/m^3
2-8 hours	7.02E-4 sec/m^3
Occupancy factor	1.0
Breathing rate	3.47E-4m^3/sec

TABLE 15.5.5-1 (Sheet 1)

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSIS

Reactor Coolant Activity	
Noble Gas Activity Prior to Accident	See Appendix 15D
Iodine Activity Prior to Accident (without spike)	See Appendix 15D
Iodine Activity (pre-existing spike)	21 μCi/g D.E. I-131 (see Appendix 15D)
Equilibrium Iodine Appearance Rates	See Appendix 15D
Increase in Iodine Appearance Rate due to the Accident-Initiated Spike	500 times equilibrium rates
Duration of Accident-Initiated Spike	1 hour
Secondary Coolant Activity	
Iodine Activity Prior to Accident	0.1 μCi/g D.E. I-131 (see Appendix 15D)
Noble Gas Activity	None
Iodine Chemical Form	Elemental
Steam generator tube leak rate	150 gpd per SG
Off-site power	Lost on reactor trip
Time of reactor trip	65 sec
Iodine partition factor in non-defective steam generators prior to and during accident	0.01
Iodine partition factor in defective steam generator prior to accident	0.01
Iodine partition factor in condenser prior to reactor trip	0.01
Time to isolate ruptured steam generator (generic assumption for dose analysis)	30 min
Duration of plant cooldown by Secondary System after accident	8 hour
Tube Rupture Break Flow	
0 - 65 seconds	8400 lb
65 - 1800 seconds	164,300 lb
Percentage of Break Flow that Flashes (average over time interval)	
0 - 65 seconds	18.0%
65 - 1800 seconds	4.74%

TABLE 15.5.5-1 (Sheet 2)

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSIS

Steam Release from Ruptured Steam Generator (not including contribution from flashed break flow)	
0 - 65 seconds	76,588 lb
65 - 1800 seconds	62.312 lb
Steam Release from Intact Steam Generators	
0 - 65 seconds	232,000 lb
65 - 1800 seconds	170,000 lb
0.5 - 2 Hours	360,000 lb
2 - 8 Hours	1,237,000 lb
Meteorology	Accident (See Appendix 15A)

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TABLE 15.5.6-1 (Sheet 1)

PARAMETERS USED IN FUEL HANDLING ACCIDENT ANALYSES

Time between plant shutdown and accident	100 hours
Average fuel assembly active (curies) after 100 hour decay[1]	
I-131	3.536E5 Ci
I-132	3.472E5 Ci
I-133	3.610E4 Ci
I-135	2.7E1 Ci
Kr-85	5.346E3 Ci
Xe-131m	4.260E3 Ci
Xe-133m	8.838E3 Ci
Xe-133	6.897E5 Ci
Xe-135	1.437E3 Ci
Radial peaking factor	1.7
Fuel rod gap fraction	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0. 05
Fuel damage	One assembly with 24 TPBARS
Iodine species split	
Elemental	99.85%
Organic	0.15%
Tritium release from 24 damaged TPBARs	84,000 Ci
Pool scrubbing factor	
Iodine	200
Noble gases	1
Tritium	1
Breathing rate	3.47E-4 m^3/sec

[1] Only the iodines and noble gases having a significant presence after 100 hours are included in the list.

TABLE 15.5.6-1 (Sheet 2)

PARAMETERS USED IN FUEL HANDLING ACCIDENT ANALYSES

Atmospheric dispersion factor	
EAB	8.59E-4 sec/m^3
LPZ outer boundary	1.39E-4 sec/m^3
FHA Outside Containment	
Release path filter efficiency for iodines	None
Isolation of release path	None
Duration of releases	2 hours
FHA Inside Containment	
Mixing volume	32,550 ft^3
Purge flow rate	16,000 cfm
Release path filter efficiency for iodines	None
Isolation of purge release path	30 seconds
Duration of releases via the equipment hatch	30 sec - 2 hr
Control Room Dose Analysis Parameters	
Volume	2.6E5 cubic feet
Normal operation inflow (unfiltered)	3200 cfm
Air intake high radiation setpoint to actuate HVAC emergency mode	400 cpm
Time to switch to emergency mode after signal	5 min
Emergency mode filtered intake flow	1000 cfm
Emergency mode filtered recirculation flow	2600 cfm
Filter efficiency for iodine	95%
Unfiltered inleakage	51 cfm
Atmospheric dispersion factor (X/Q)	
FHA outside containment (0 - 2 hr)	1.80E-3 sec/m^3
FHA inside containment	
0 - 30 sec	5.63E-4 sec/m^3
30 sec - 2 hr	1.80E-3 sec/m^3
Occupancy factor	
0 -24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4
Breathing rate	3.47E-4 m^3/sec

TABLE 15.5.6-1 (Sheet 3)

PARAMETERS USED IN FUEL HANDLING ACCIDENT ANALYSES

Nuclide Decay Constants (hr^1)	
I-131	3.5833E-3
I-132	3.0401E-1
I-133	3.3320E-2
I-135	1.0486E-1
Kr-85	7.3692E-6
Xe-131m	2.4269E-3
Xe-133m	1.2836E-2
Xe-133	5.4594E-3
Xe-135	7.5755E-2
H-3 (tritium)	6.407E-6
Te-131m	2.3105E-3
Te-132	8.8638E-3
Committed Effective Dose Equivalent DCF (rem/Ci inhaled)	
I-131	3.29E-4
I-132	3.81E2
I-133	5.85E3
I-135	1.23E3
Kr-85	N/A
Xe-131m	N/A
Xe-133m	N/A
Xe-133	N/A
Xe-135	N/A
H-3 (tritium)	64.01
Effective Dose Equivalent DCF (rem-m^3/Ci-s)	
I-131	6.734E-2
I-132	4.144E-1
I-133	1.088E-1
I-135	2.953E-1
Kr-85	4.403E-4
Xe-131m	1.439E-3
Xe-133m	5.069E-3
Xe-133	5.772E-3
Xe-135	4.403E-2
H-3 (tritium)	1.225E-6

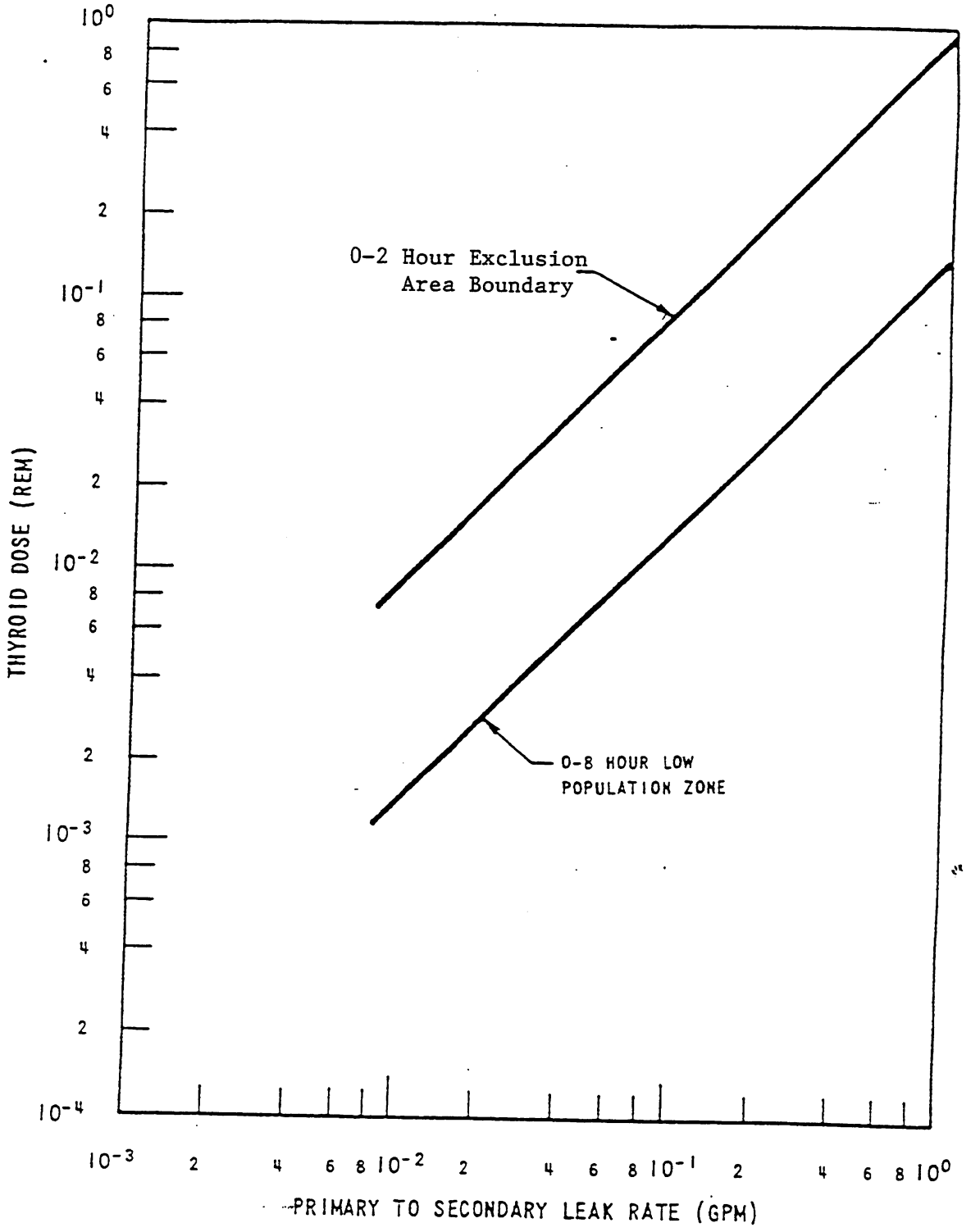


Figure 15.5.1-1 Loss of AC Power - Thyroid Dose

Best Available Historical Image

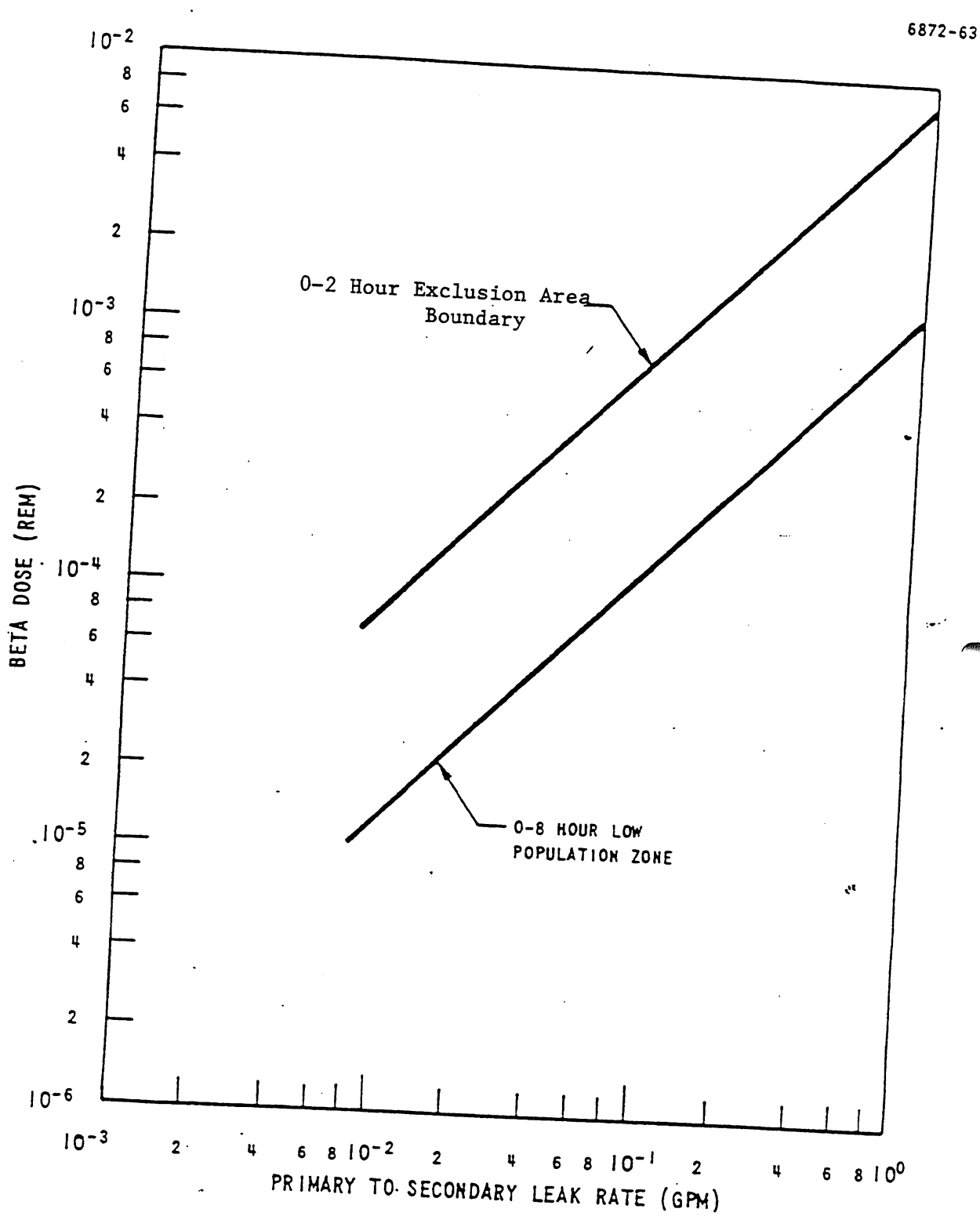


Figure 15.5.1-2 Loss of AC Power - Beta Dose

Revised by Amendment 10

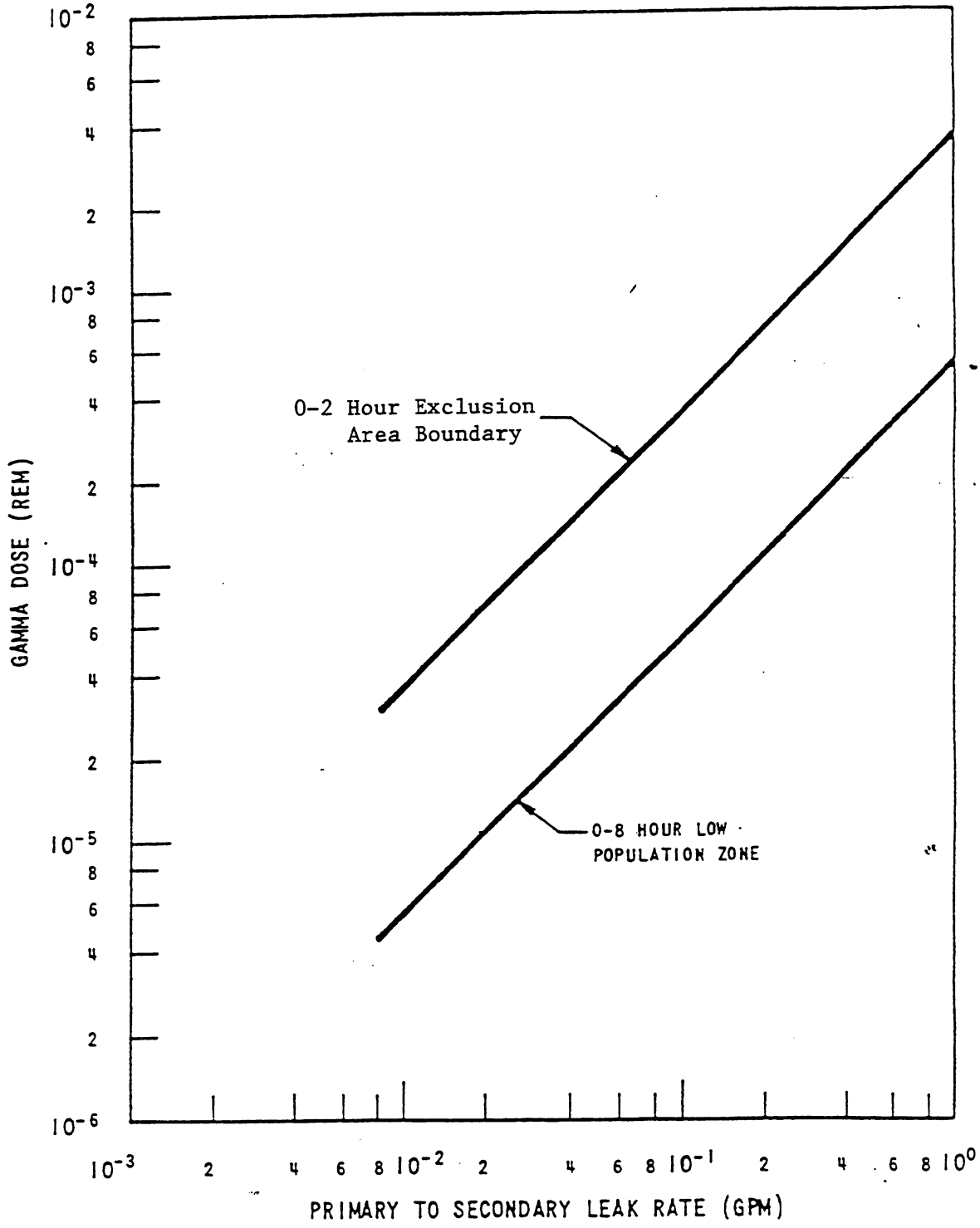
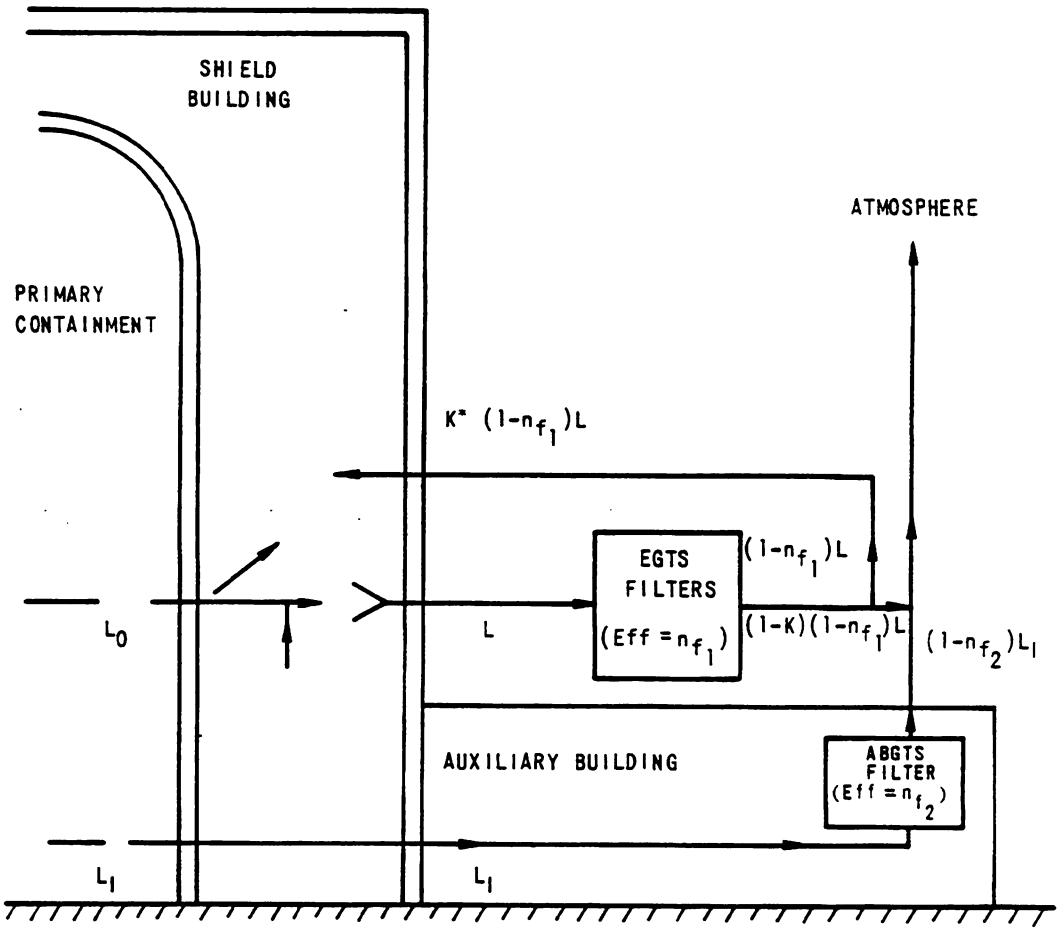


Figure 15.5.1-3 Loss of AC Power - Gamma Dose

Revised by Amendment 10



*RECIRCULATION FLOW RATE

Figure 15.5.3-1 Schematic of Leakage Path

APPENDIX 15A

DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

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APPENDIX 15A

DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15A.1 Introduction

This Appendix identifies the models used to calculate the offsite radiological doses that would result from releases of radioactivity due to various postulated accidents. The postulated accidents are:

1. Waste Gas Decay Tank Rupture
2. Steam Generator Tube Rupture
3. Steam Line Break
4. Loss of A.C. Power
5. Rod Ejection Accident
6. Loss-of-Coolant Accident

The radiological consequences analysis for the Fuel Handling Accident does not use the modeling described here but instead uses the modeling associated with the use of alternate source term methodology as discussed in Section 15.5.6.

15A.2 Assumptions

The following assumptions are basic to both the model for the whole body and skin doses due to immersion in a cloud of radioactivity and the model for the thyroid dose due to inhalation of radioactivity.

1. Direct radiation from the source point is negligible compared to gamma and beta radiation due to submersion in the radioactivity leakage cloud.

2. All radioactivity releases are treated as ground level releases regardless of the point of discharge.

3. The dose receptor is a standard man as defined by the International Commission on Radiological Protection (ICRP II) (Reference 1).

4. Radioactive decay from the point of release to the dose receptor is neglected.

5. Isotopic data such as decay constants and dose conversion factors are as provided in Table 15A-1.

15A.3 Whole Body Dose and Skin Dose

The whole body and skin doses delivered to a dose receptor are obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane, i.e., an "infinite semispherical cloud." The concentration of radioactive material within this cloud is taken to be uniform and equal to the maximum centerline ground level concentration that would exist in the cloud at the appropriate distance from the point of release. Equations describing an infinite semispherical cloud were used to calculate the doses for a given time period as follows: (Reference 3).

Skin Dose (Rem) = $\left(\frac{X}{Q}\right)_t \sum_1 A_i \left(DCF_{SDE}\right)_i$

and

Gamma Dose (Rem) = $\left(\frac{X}{Q}\right)_t \sum_1 A_i \left(DCF_{EDE}\right)_i$

Where:.

A_i is the activity of isotope i released during a given time period t, curies

$\left(\frac{X}{Q}\right)_t$ is the atmospheric dilution factor for a given time period t, sec/m^3

$(DCF_{SDE})_i$ is the shallow dose equivalent Dose Conversion Factor for isotope i, rem-m^3/Ci-sec (Table 15A-1)

$DCF_{EDE})_i$ is the effective dose equivalent Dose Conversion Factor for isotope i, rem-m^3/Ci-sec (Table 15A-1)

15A.4 Thyroid Dose

The thyroid dose for a given time period t, is obtained from the following expression:

$$D = \left(X/Q\right)_t B_t \sum_i A_i \left(DCF_{thy}\right)_i$$

where:

D	= thyroid dose, rem
$(X/Q)_t$	= atmospheric dispersion factor for time interval t, sec/m^3
B_t	= breathing rate for time interval t, m^3/sec (Table 15A-1)
A_i	= total activity of iodine isotope i released in time period t, curies
$(DCF_{thy})_i$	= dose conversion factor for iodine isotope, i, rem/curie inhaled (Table 15A-1)

The breathing rates are consistent with the guidance of Regulatory Guide 1.4 (Reference 2):

0 – 8 hours	3.47E-4 m^3/sec
8 – 24 hours	1.75E-4 m^3/sec
>24 hours	2.32E-4 m^3/sec

15A.5 Total Effective Dose Equivalent

The accident radiological consequences are also calculated in the form of Total Effective Dose Equivalent (TEDE) for information only. The TEDE dose is defined as being the sum of the whole body dose plus the Committed Effective Dose Equivalent (CEDE) dose. The CEDE dose is obtained from the following expression:

$$D_{CEDE} = (X/Q)_t \, B_t \, \Sigma_i \, A_i \, (DCF_{CEDE})_i$$

Where:

	$(X/Q)_t$	= atmospheric dispersion factor for time interval t, sec/m^3
	B_t	= breathing rate for time interval t, m^3/sec
	A_i	= activity of isotope i released during time interval t, Ci
	$(DCF_{CEDE})_i$	= CEDE DCF for isotope i, rem/Ci (Table 15A-1)

15A.6 References

1. "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)," Health Physics Vol. 3, pp. 30, 146-153, 1960.

2. Regulatory Guide 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," USAEC, June 1973.

3. EPA Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion” EPA-520/1-88-020, September 1988.

4. EPA Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” EPA 402-R-93-081, September 1993.

TABLE 15A-1

PHYSICAL DATA FOR ISOTOPES

Nuclide	Decay Constant (hr^{-1})	Thyroid DCF(1) (rem/Ci inhaled)	Effective Dose Equivalent DCF(2) ($rem\text{-}m^3/Ci\text{-}s$)	Shallow Dose Equivalent DCF(2) ($rem\text{-}m^3/Ci\text{-}s$)	Committed Effective Dose Equivalent DCF(1) (rem/Ci inhaled)
1-131	3.5833E-3	1.080E6	6.734E-2	1.103E-1	3.29E4
1-132	3.0401E-1	6.438E3	4.144E-1	5.846E-1	3.81E2
1-133	3.3320E-2	1.798E5	1.088E-1	2.157E-1	5.85E3
1-134	7.9067E-1	1.066E3	4.810E-1	6.919E-1	1.31E2
1-135	1.0486E-1	3.130E4	2.953E-1	4.107E-1	1.23E3
Kr-85m	1.5472E-1	N/A	2.768E-2	8.288E-2	N/A
Kr-85	7.3692E-6	N/A	4.403E-4	4.884E-2	N/A
Kr-87	5.4508E-1	N/A	1.524E-1	5.069E-1	N/A
Kr-88	2.4755E-1	N/A	3.774E-1	4.995E-1	N/A
Xe-131m	2.4269E-3	N/A	1.439E-3	1.783E-2	N/A
Xe-133m	1.2836E-2	N/A	5.069E-3	3.848E-2	N/A
Xe-133	5.4594E-3	N/A	5.772E-3	1.839E-2	N/A
Xe-135m	2.6574E0	N/A	7.548E-2	1.099E-1	N/A
Xe-135	7.5755E-2	N/A	4.403E-2	1.154E-1	N/A
Xe-138	2.9350E0	N/A	2.135E-1	3.959E-1	N/A

Note: 1. From EPA Federal Guidance Report No. 11 (Reference 3)

2. From EPA Federal Guidance Report No. 12 (Reference 4)

TABLE 15A-2

ACCIDENT ATMOSPHERIC DILUTION FACTORS[1]
(sec/m^3)

SEQUOYAH NUCLEAR PLANT

Time Period (hours)	Exclusion Area Boundary 556 Meters	Low Population Zone 4828 Meters
0 - 2	1.64×10^{-3}	1.96×10^{-4}
2 - 8		2.46×10^{-5}
8 - 24		2.02×10^{-5}
24 - 96		1.03×10^{-5}
96 - 720		4.77×10^{-6}

[1] Based on onsite meteorology for April 2, 1971 - March 31, 1972. The method used in calculating the values is presented in Section 2.3.4.

APPENDIX 15B

SENSITIVITY ANALYSIS OF LOCA DOSE CALCULATIONS

HISTORICAL INFORMATION)

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APPENDIX 15B

SENSITIVITY ANALYSIS OF LOCA DOSE CALCULATIONS

(HISTORICAL INFORMATION)

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APPENDIX 15B

SENSITIVITY ANALYSIS OF LOCA DOSE CALCULATIONS

(HISTORICAL INFORMATION)

15B.1 Introduction

The critical parameters which affect offsite doses from the postulated LOCA were investigated to determine the sensitivity of the doses to these parameters. The effect of the ice condenser iodine removal is given as well as an analysis of the key conservatisms in the dose model.

15B.2 Parametric Study

Table 15B-1 gives the results of 9 cases and the assumptions used. Assumptions not listed in Table 15B-1 are identical to those used in Subsection 15.5.3 for the core release source. Case 1 is the base case which is discussed in Subsection 15.5.3 and is used as the reference case when relative doses are presented. The cases given in Table 15B-1 along with other calculations are the basis for the discussion of sensitivity.

15B.3 The Effect of Ice Condenser Iodine Removal

The removal of elemental iodines by the ice bed following blowdown is discussed in Subparagraph 6.2.3.3.4. The effect of the ice condenser, (along with the charcoal filters) using the efficiencies given in Table 15.5.3-2 is to reduce the offsite doses due to elemental iodines to the same magnitude as that due to organic iodines, at times in excess of about 0.2 hours. Therefore, increased ice condenser removal of elemental iodines (or any other removal system for elemental iodines) will not significantly reduce thyroid doses.

Table 15B-2 shows the two hour thyroid exclusion area boundary doses calculated for several cases with and without ice condenser iodine removal credit using the TID-14844 source term. Dose reduction factors (DRFs) are given for total thyroid doses as well as for the release components (via the annulus and via the Auxiliary Building). All DRFs are in the neighborhood of 3.75. The DRFs for the releases via the annulus are lowest and reflect the importance of the high initial EGTS exhaust rate on these releases. Over the period when the exhaust flow is high the ice condenser has not yet had time to fully reduce the elemental iodine inventory.

Table 15B-3 gives similar information for the 30-day doses at the LPZ distance. The DRFs are in the neighborhood of 4.25 for the 30-day doses. The DRFs for the annulus releases would be expected to be lower than those for the Auxiliary Building releases due to the importance of the first half hour on annulus releases and because multiple passes through the EGTS charcoal filters reduce the relative importance of elemental iodines slightly.

15B.4 Sensitivity of Releases to Holdup in the Auxiliary Building

The model used to describe transport of radioactivity through the Auxiliary Building to the Auxiliary Building Gas Treatment System (ABGTS) is based on a constant delay time between leakage and exhaust. This delay time is used to calculate the reduction in activity release due to decay, but is neglected when considering the time of exposure to the release, i.e., the 2-hour dose is calculated on the basis of the release from the EGTS for the first two hours plus the release from the primary containment to the Auxiliary Building for the first two hours reduced by decay.

APPENDIX 15B

SENSITIVITY ANALYSIS OF LOCA DOSE CALCULATIONS

(HISTORICAL INFORMATION)

The decay time was determined by investigating probable leakage and transport paths to and through the Auxiliary Building. The Auxiliary Building free volume is approximately 3.5 million cubic feet. The ABGTS design flow rate is 9,000 cfm per train. Each train is capable of maintaining the Auxiliary Building at negative pressure, since inleakage is expected to be less than 8,000 cfm. However, even with both systems operating at capacity, the exhaust rate is about 0.3 volumes per hour, providing a mean holdup time of more than 3 hours. For conservatism no delay time was used for the base case.

The ABGTS will reduce the pressure in the Auxiliary Building in less than 10 minutes to the point where no outleakage is assured. Case 9 of Table 15B-1 shows the doses to be expected at the exclusion area boundary and the LPZ distance under the assumptions of ice condenser removal credit, 90 percent containment leakage directly to the EGTS suction, with the remaining 10 percent going directly to the environment for the first 10 minutes following the LOCA and to the Auxiliary Building thereafter where the activity will experience a 0.3 hour average delay.

Figure 15B-1 shows the relative dose for the first two hours at the exclusion area boundary as a function of decay time in the Auxiliary Building. Figure 15B-2 gives similar information for the 30-day doses at the LPZ distance. Note that delay has little effect on the thyroid dose. This is to be expected since about 90 percent of the thyroid dose is due to the long lived isotopes I-131 and I-133. The ß and γ doses are principally due to relatively short-lived noble gases and therefore, show a significant reduction with delay in the Auxiliary Building.

15B.5 Sensitivity of Releases to the Assumed Fraction Entering the EGTS Suction Directly

No significant amount of primary containment leakage is expected to enter the EGTS suction at the top of the dome directly. However, the conservative assumption was made for the base case that 10 percent of the leakage to the annulus would enter the suction directly. Dose calculations for the exclusion area boundary were performed for EGTS direct leakage fraction from 0.0 up to 100 percent and the results are shown in Figure 15B-3. Even if the very conservative assumption is made that all primary containment leakage into the annulus enters the EGTS suction directly, the resulting two-hour thyroid dose is only 10 percent of the 10 CFR 100 guideline value, the beta dose is less than 12 percent of the 10 CFR 100 guideline value, and the gamma dose is less than 27 percent of the 10 CFR 100 guideline value.

Figure 15B-4 gives similar information for the 30-day doses at the LPZ distance.

The case analyzed in which none of the primary containment leakage enters the EGTS suction directly is the complete mixing model. In this model, the secondary containment acts similar to a large hold-up tank which collects the primary containment leakage. All fission products escaping from the primary containment to the secondary containment are assumed to mix with the air in the secondary containment. Ninety percent of the primary containment leakage is assumed to go into the secondary containment while the remaining 10 percent goes to the Auxiliary Building. The Auxiliary Building release path is discussed in Subsection 15.5.3.

APPENDIX 15B

SENSITIVITY ANALYSIS OF LOCA DOSE CALCULATIONS

(HISTORICAL INFORMATION)

The initial pressure in the annulus is less than atmospheric. After blowdown, the annulus pressure will increase rapidly due to expansion of the containment vessel as a result of primary containment atmosphere temperature and pressure increases. The annulus pressure will continue to rise due to heating of the annulus atmosphere by conduction through the containment vessel. After a delay of 38 seconds, the Emergency Gas Treatment System (EGTS) will be operating at full flow to maintain the annulus pressure to below atmospheric pressure.

During the initial 30 second period following the LOCA, primary containment leakage is assumed to mix uniformly with the secondary containment air. No activity is released to the environment during this period.

After the initial 30 second period following the LOCA, the EGTS draws air and fission products at uniform concentration from the secondary containment volume, passes it through HEPA-charcoal filters and exhausts a portion to the atmosphere. The remainder is exhausted back into the secondary containment where it is assumed to mix uniformly with the secondary containment air; it is then available for reentry into the EGTS in a similar manner to the primary containment leakage. The EGTS exhaust and recirculation flow rates as a function of time are given in Table 15.5.3-3.

The case analyzed in which all of the primary containment leakage enters the EGTS directly is the no mixing model. During the initial 30 second period following the LOCA, the no mixing model is similar to that used for the complete mixing case, discussed above in that no activity is released to the environment during this period.

After the initial 30 second period following the LOCA, all of the primary containment leakage to the secondary containment is drawn immediately into the EGTS where it is filtered. A portion is then exhausted to the atmosphere (see Table 15.5.3-3) while the remainder is returned to the secondary containment where it is assumed to mix uniformly in that volume.

15B.6 Sensitivity of Releases to Annulus Holdup Effectiveness

The holdup time in the annulus is a function of both the exhaust rate and the active annulus volume. The geometry of the annulus volume, as well as the locations of the EGTS suctions and recirculation exhausts, should provide uniform and complete circulation over essentially all the annulus volume. To demonstrate the capabilities of the containment design, doses were calculated assuming that large regions of the annulus (50 percent or more of the annulus free volume) do not contribute to fission product holdup. The results of this analysis are shown in Figure 15B-5, and demonstrate that offsite doses are still within 10 CFR 100 guidelines even for the unrealistic assumption of large reductions in the active annulus volume.

15B.7 Dose Models

The dose model discussed in Appendix 15A and used in the analyses in this appendix is conservative in three principle areas as conventionally applied. First, the decay of fission products downwind is neglected; second, the actual transport time (time to reach the receptor)

is neglected; and third, the dose is based on a uniform cloud which may, in fact, be quite nonuniform. All three factors have only a small effect when wind speeds are greater than 1 meter per second, however, for wind speeds less than 1 meter per second the first considerations tend to result in the "two-hour" doses being over estimated, while the third consideration overestimates the long term doses.

Figure 15B-6 shows the relative effect of decay of the activity releases during downwind transport on the two hour LOCA doses given in case 1. Due to the importance of long lived isotopes, the thyroid doses are not significantly affected. However, the beta and gamma doses show a stronger decrease with increasing transport time since the short lived kryptons contribute significantly to the doses.

Figures 15B-7 and 15B-8 show the total effect of the wind speed on the doses given in case 1 for Pasquill G and Pasquill F meteorological conditions. The curves take into account the effect of changing wind speed on the atmospheric dilution factor and the activity reduction due to decay during transport to the receptor. Four curves are presented in each figure. One curve shows the relative dose as a function of wind speed taking into account only the variance in the x/Q. The other three curves show the combined effect of the wind speed on the relative doses taking into account decay during transport and a variance in the x/Q.

Figure 15B-9 shows the effect of the receptor exposure time on the relative dose. The intent of the figure is to show the relative dose a receptor at the exclusion area boundary would receive under very low wind speed conditions during the first two hours following the initiation of a postulated LOCA. If the wind speed were such that it would take 1-1/2 hours for the radioactive cloud to reach the exclusion area boundary, the receptor would only be exposed to the cloud for 1/2 hours during the initial two hours following a postulated LOCA. Therefore, in the initial 2 hours he would only be exposed to that portion of the radioactivity that was released from the plant during the first 1/2 hour.

The information in Figures 15B-6 through 15B-9 is used to construct Figures 15B-10 and 15B-11. Figures 15B-10 and 15B-11 show the relative dose at the minimum exclusion distance during the first two hours of the accident for various wind speeds. The curves include (1) the effect of decay during transport, (2) the dependence of the dispersion factor on wind speed, and (3) the transport time required to reach the exclusion area boundary. This analysis demonstrates that the dose models used in Appendix 15A for the two-hour thyroid dose are conservative by a factor of 1.11 for Class G stability conditions and 1.33 for Class F stability conditions.

The conclusion to be drawn on beta and gamma doses is complicated by the assumption of a semi-infinite cloud, which ignores doses before the arrival of the plume at the receptor location. A detailed calculation of doses due to travel of the nonhomogenous plume would be required to determine the level of conservatism of the semi-infinite cloud model and to determine the effect of wind speed on doses.

The dispersion factor used for the initial two-hour period following the LOCA corresponds to Pasquill Class F and a wind speed of 0.25 meters/ second. At this low wind speed considerable

meandering of the wind can be expected; thus increasing the dispersion at the receptor location, thereby reducing doses. Actual measurements of dispersion factors at other sites have shown that the equation used in this analysis for the dispersion factor is quite conservative for low wind speeds.*

15B.8 Conclusions

Three conclusions can be drawn from the analyses in this appendix.

1. The effectiveness of the double containment concept utilizing the Shield Building augmented by treatment of the Auxiliary Building exhaust is relatively independent of a wide range of conservative assumptions which might be postulated. The major factor in the effectiveness of the secondary containment is its inherent capability to collect the containment leakage for filtration of the radioactive iodine prior to release to the environment. This effect is greatly enhanced by the recirculation feature of the air handling systems, which forces repeated filtration passes for the major fraction of the primary containment leakage before release to the environment.

2. The offsite doses for the hypothetical LOCA with the TID-14844 activity release will be less than those presented for the Regulatory Guide 1.4 (1973) (TID-14844 activity release) evaluation presented in Subsection 15.5.3 and considerably below the guideline values of 10 CFR 100.

3. The containment concept used in the Sequoyah design is highly effective in minimizing the release of radioactivity to the environment as a result of a loss-of-coolant.

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*Atmospheric Diffusion Experiments with SF_6 Tracer Gas at Three Mile Island Nuclear Station Under Low Wind Speed Inversion Conditions, Pickard, Lowe, and Associates, Inc., The Research Corporation of New England General Public Utilities Service Corporation, January 1972.

TABLE 15B-1

PARAMETRIC STUDY OF LOCA

(HISTORICAL INFORMATION)

Case	Ice Condenser Iodine Removal Credit	Containment Leakage Directly To EGTS Suction(%)	Auxiliary Building Delay Time (Hr)	Thyroid Dose (Rem) SB	Thyroid Dose (Rem) LPZ	Beta Dose (Rem) SB	Beta Dose (Rem) LPZ	GammaDose (Rem) SB	GammaDose (Rem) LPZ
1	yes	10	0.0	2.32×10^{1}	5.69	1.89	7.81×10^{-1}	4.26	7.33×10^{-1}
2	yes	10	0.3	2.30×10^{1}	5.65	1.68	7.55×10^{-1}	3.57	6.51×10^{-1}
3	yes	10	1.0	2.26×10^{1}	5.58	1.43	7.08×10^{-1}	2.87	5.56×10^{-1}
4	no	10	0.0	8.75×10^{1}	2.41×10^{1}	1.93	7.88×10^{-1}	4.42	7.59×10^{-1}
5	no	10	0.3	8.64×10^{1}	2.40×10^{1}	1.75	7.46×10^{-1}	3.78	6.76×10^{-1}
6	no	10	1.0	8.47×10^{1}	2.36×10^{1}	1.46	7.14×10^{-1}	2.98	5.79×10^{-1}
7	yes	100	0.0	2.88×10^{1}	6.51	2.82	8.92×10^{-1}	6.69	1.03
8	yes	0	0.0	2.23×10^{1}	5.61	1.77	7.61×10^{-1}	3.94	6.94×10^{-1}
9	yes	100	0.3*	9.70×10^{1}	183×10^{1}	3.20	9.6×10^{-1}	6.49	1.01

*Auxiliary Building by-passed for first 10 min. following LOCA.

TABLE 15B-2

ICE CONDENSER IODINE REMOVAL EFFECTS

2 HOUR THYROID DOSES (REM)

(HISTORICAL INFORMATION)

WITH REMOVAL CREDIT		WITHOUT REMOVAL CREDIT		
Case	Total Tyroid Dose	Case	Total Thyroid Dose	Dose Reduction Factor
1	23.2	4	87.5	3.77
2	23.0	5	86.4	3.76
3	22.6	6	84.7	3.75
Case	Annulus Release Tyroid Dose	Case	Annulus Release Thyroid Dose	Dose Reduction Factor
1,2,3	3.05	4,5,6	9.53	3.12
Case	Aux. Build. Release Tyroid Dose	Case	Aux. Building Release Thyroid Dose	Dose Reduction Factor
1	20.1	4	78.0	3.88
2	19.9	5	76.9	3.86
3	19.5	6	75.2	3.86

TABLE 15B-3

ICE CONDENSER IODINE REMOVAL EFFECTS

30 DAY THYROID DOSES (REM)

(HISTORICAL INFORMATION)

WITH REMOVAL CREDIT		WITHOUT REMOVAL CREDIT		
Case	Total Thyroid Dose	Case	Total Thyroid Dose	Dose Reduction Factor
1	5.69	4	24.1	4.24
2	5.65	5	24.0	4.25
3	5.58	6	23.6	4.23

Case	Annulus Release Thyroid Dose	Case	Annulus Release Thyroid Dose	Dose Reduction Factor
1,2,3	0.599	4,5,6	2.40	4.01

Case	Aux. Build. Release Thyroid Dose	Case	Aux. Building Release Thyroid Dose	Dose Reduction Factor
1	5.09	4	21.7	4.26
2	5.05	5	21.6	4.28
3	4.98	6	21.2	4.26

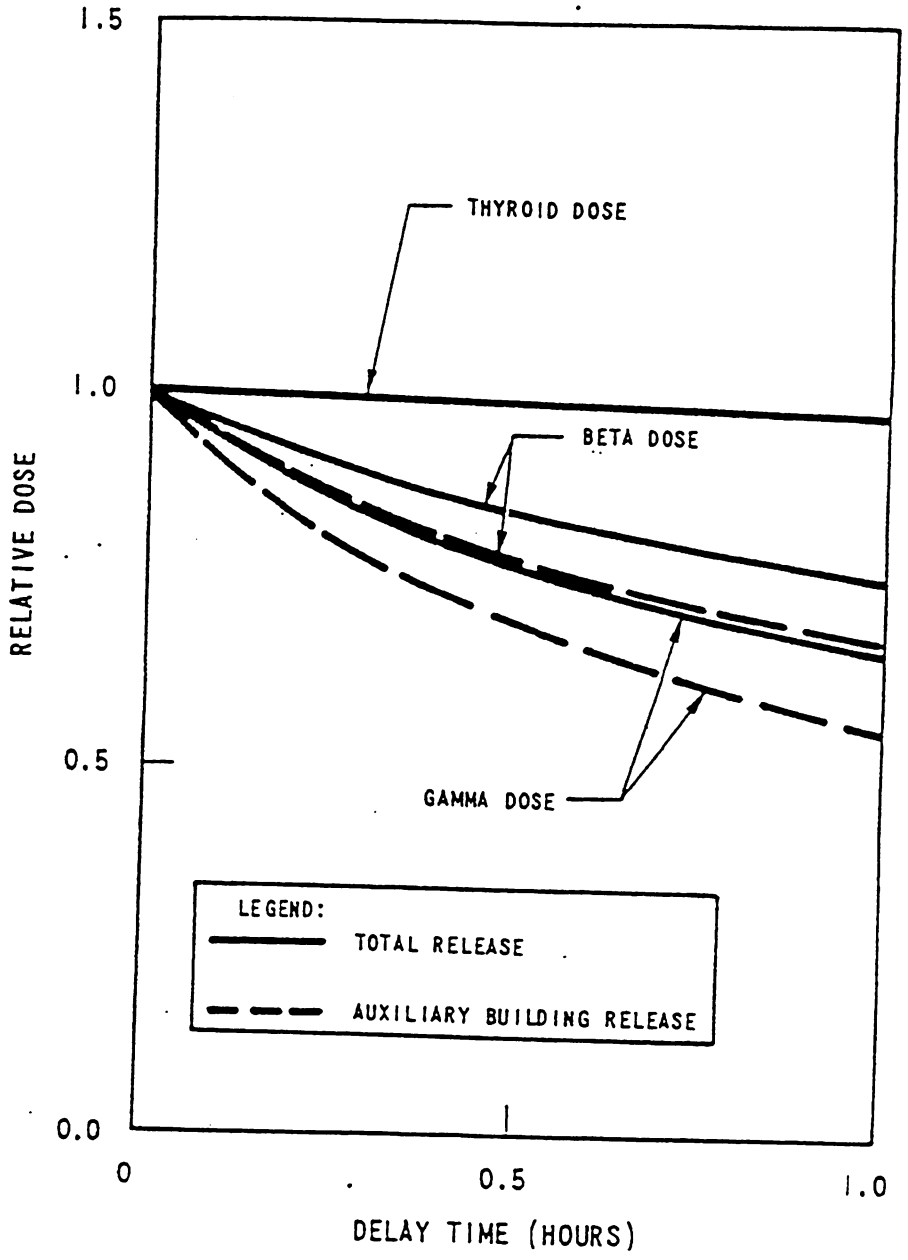

Figure 15B-1 Effect of Delay in the Auxiliary Building Release On 2 Hour Doses at the Exclusion Area Boundary

Revised by Amendment 10

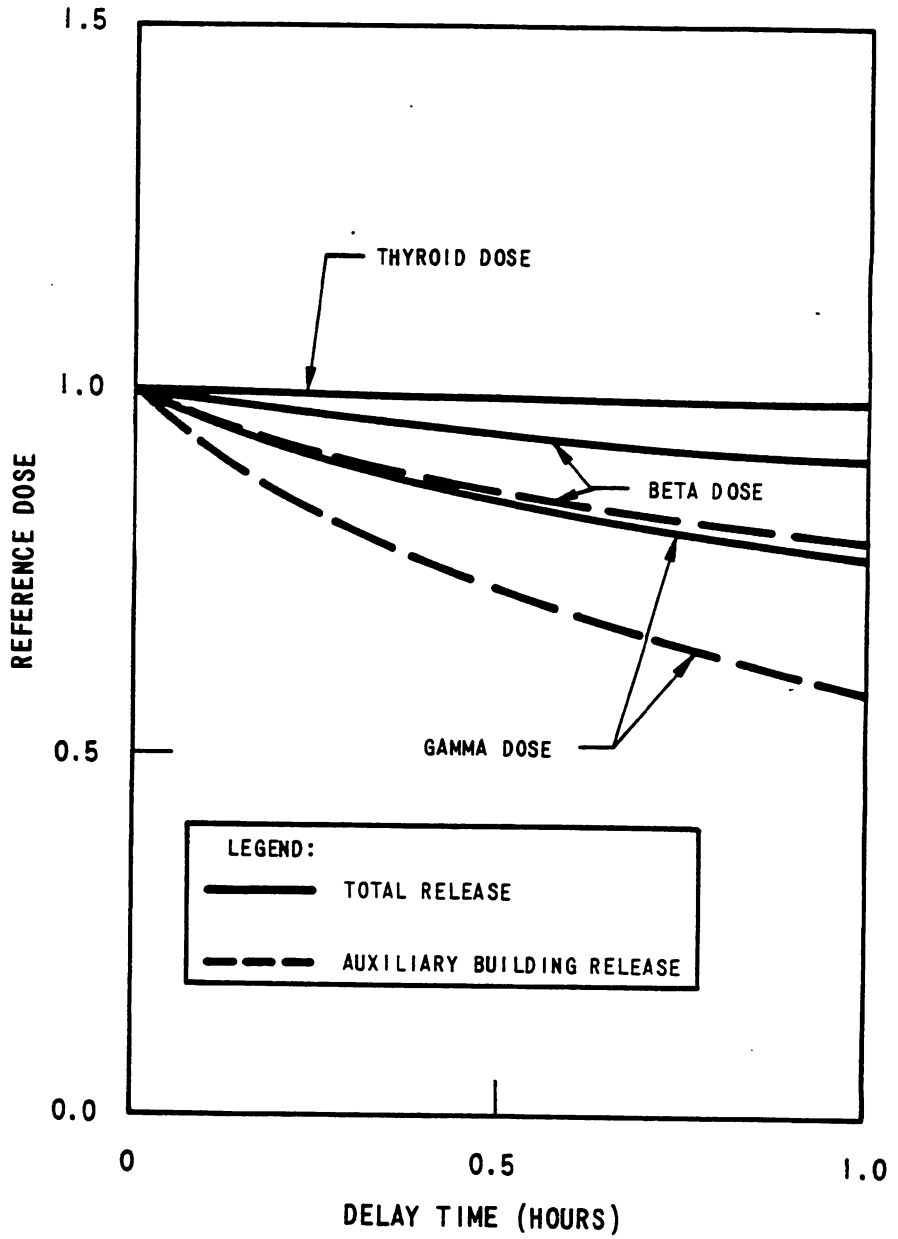

Figure 15B-2 Effect of Delay in the Auxiliary Building Release On 30 Day Doses at the Low Population Zone

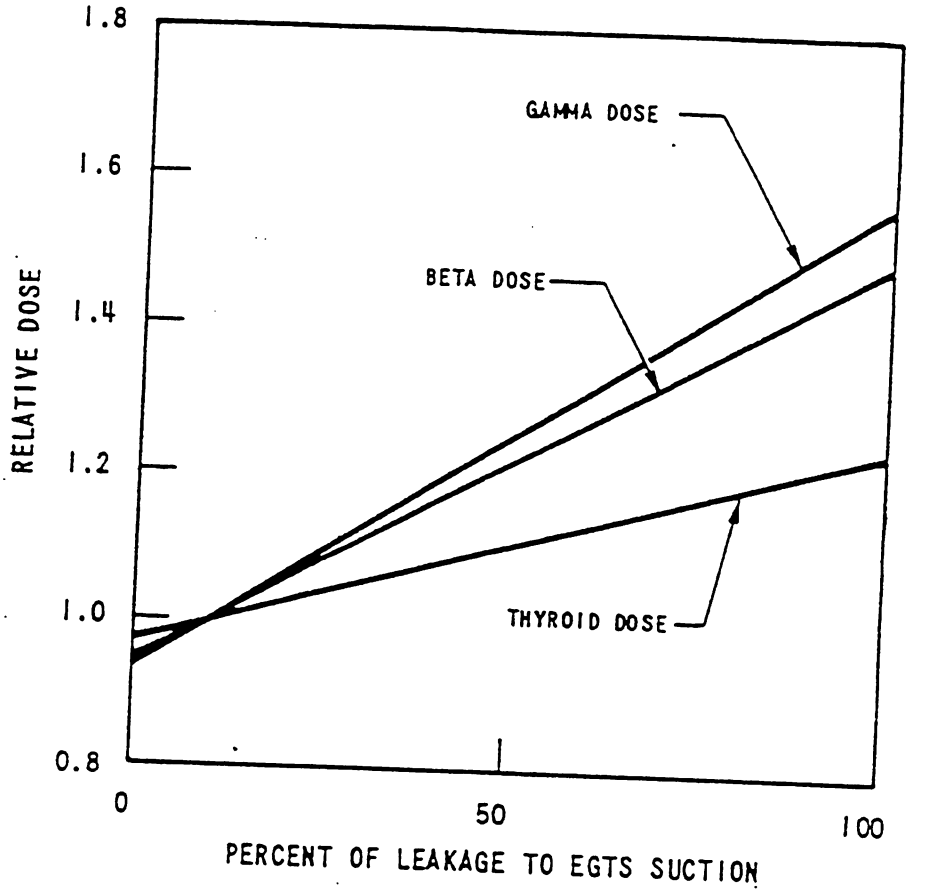

Figure 15B-3 Effect of Leakage Directly Into the EGTS Suction Header on 2 Hour Doses at the Exclusion Area Boundary

Revised by Amendment 10

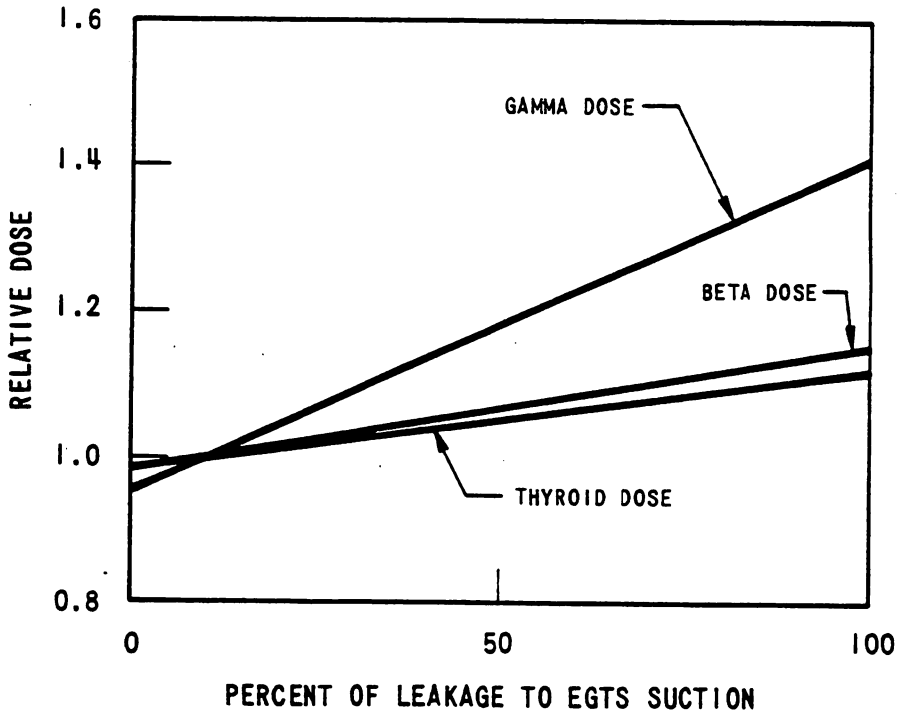

Figure 15B-4 Effect of Leakage Directly Into the EGTS Suction Header On 30 Day Doses at the Low Population Zone

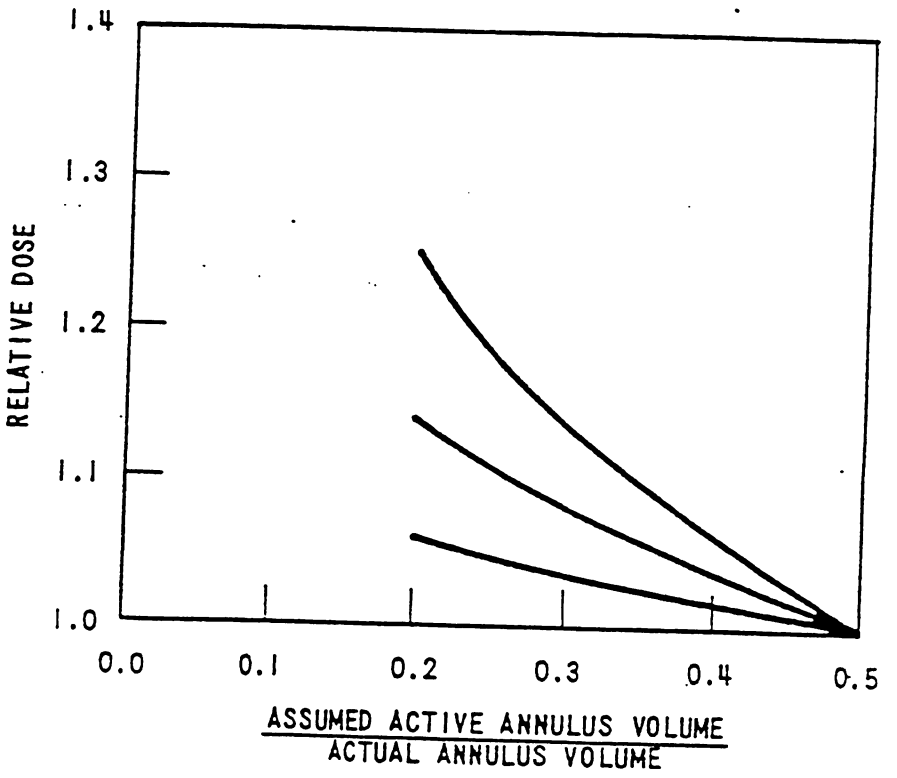

Figure 15B-5 Effect of Assumed Annulus Volume on 2 Hour Dose at Exclusion Area Boundary

Revised by Amendment 10

Best Available Historical Image

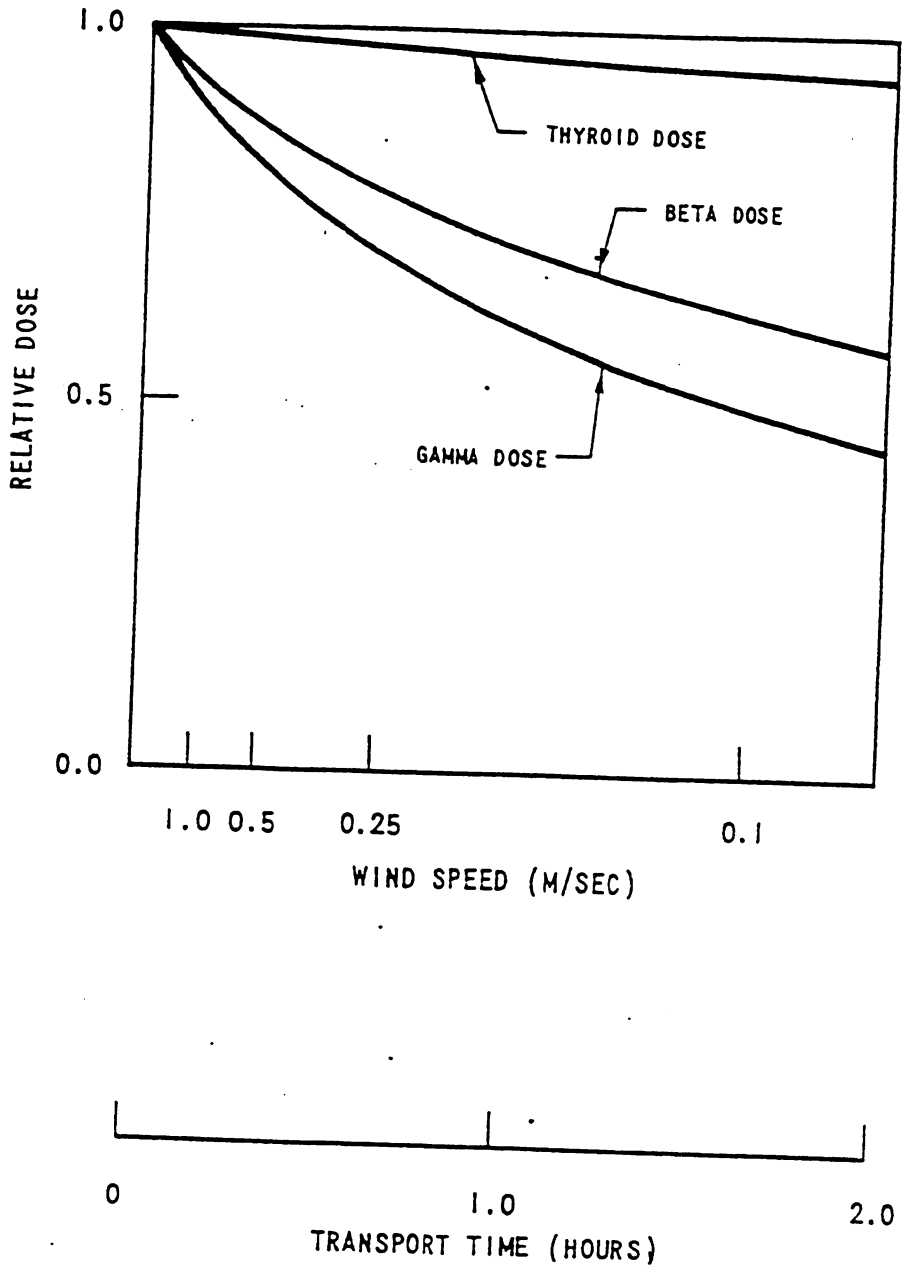

Figure 15B-6 Effect of Decay During Downwind Transport on 2 Hour Doses at Exclusion Area Boundary

Revised by Amendment 10

Best Available Historical Image

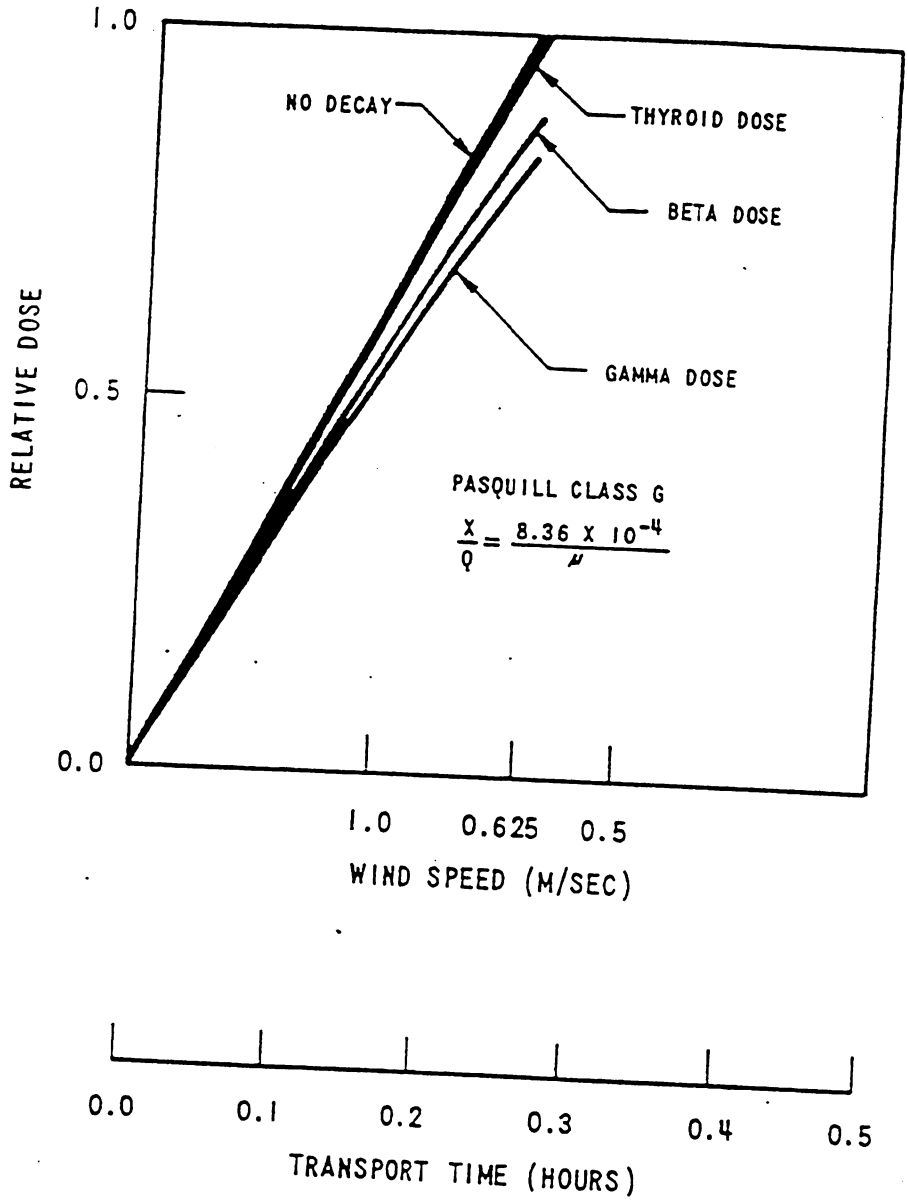

Figure 15B-7 Effect of Decay During Downwind Transport for Various Wind Speeds on 2 Hour Doses at Exclusion Area Boundary

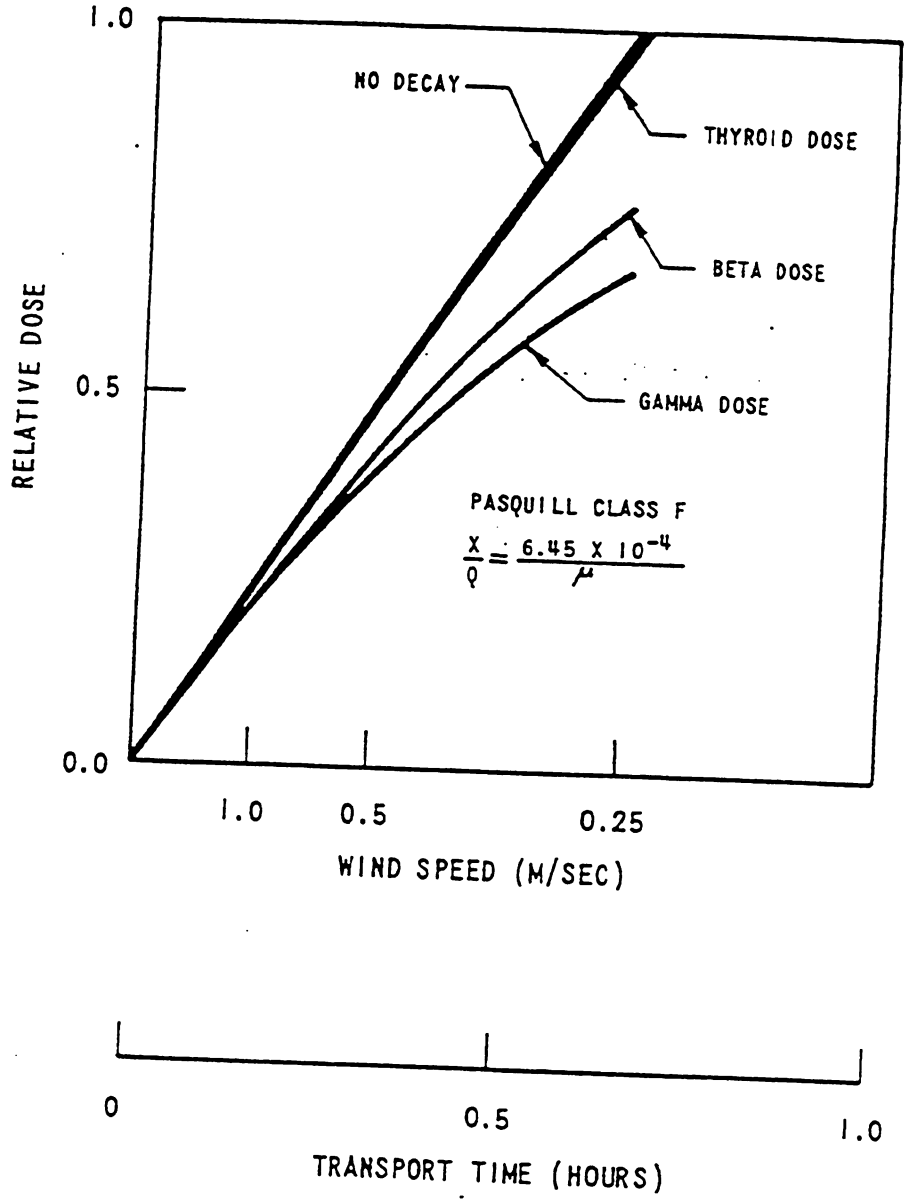

Figure 15B-8 Effect of Decay During Downwind Transport for Various Wind Speeds on 2 Hour Doses at Exclusion Area Boundary

Revised by Amendment 10

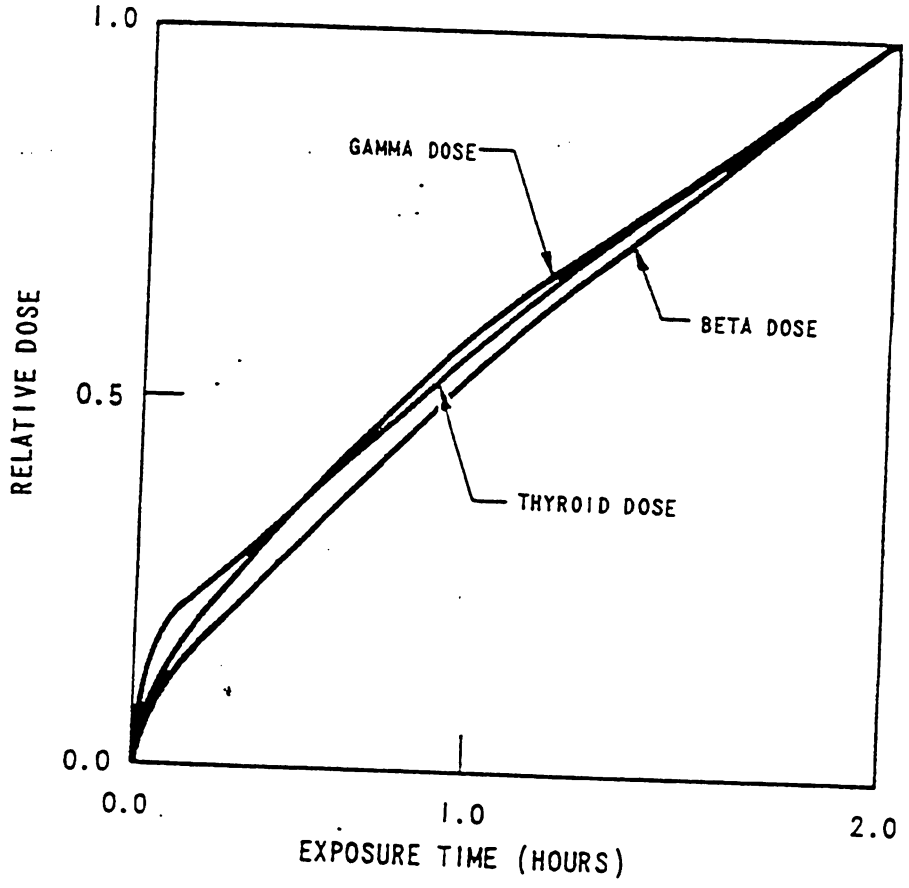

Figure 15B-9 Effect of Exposure Time on Exclusion Area Boundary Doses

Best Available Historical Image

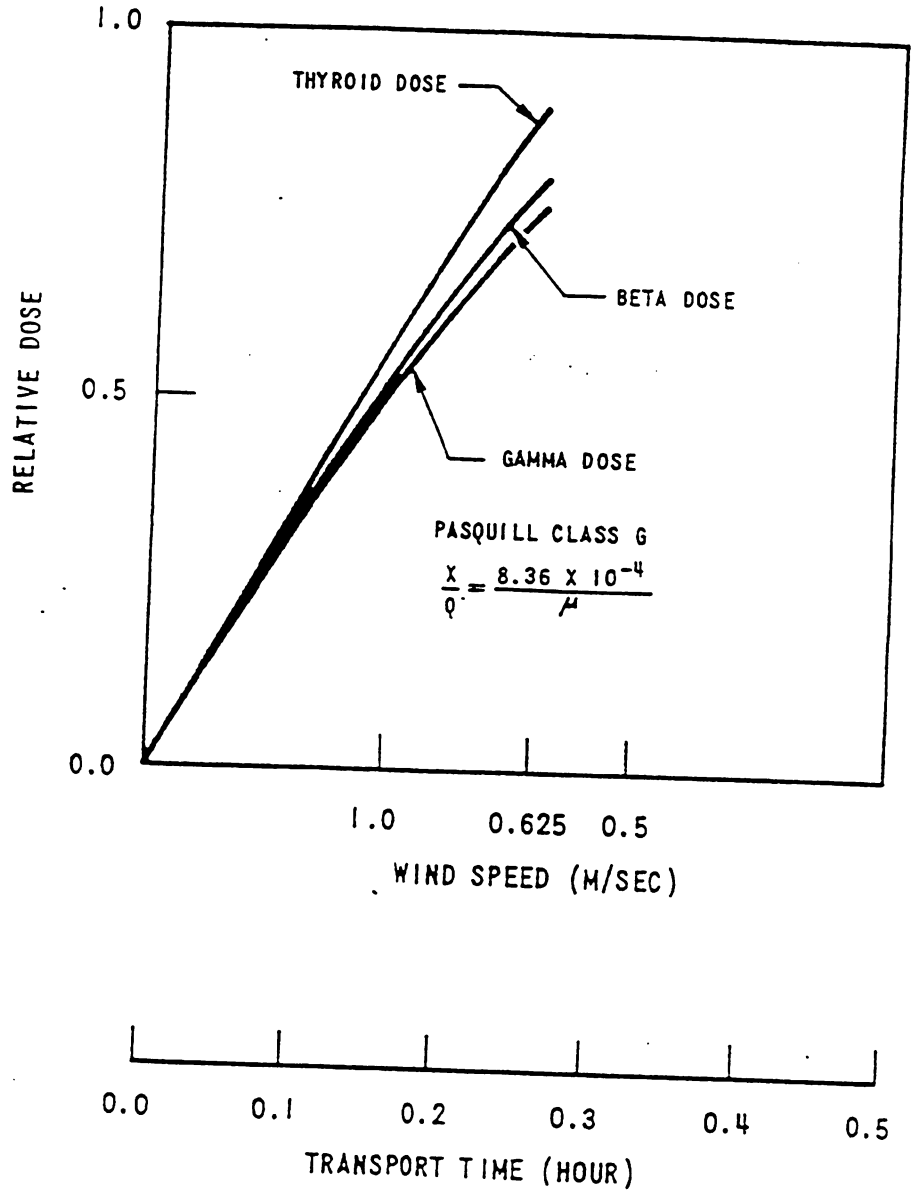

Figure 15B-10 Effect of Decay During Downwind Transport for Various Wind Speeds on Doses at the Exclusion Area Boundary in the First Two Hours of Exposure

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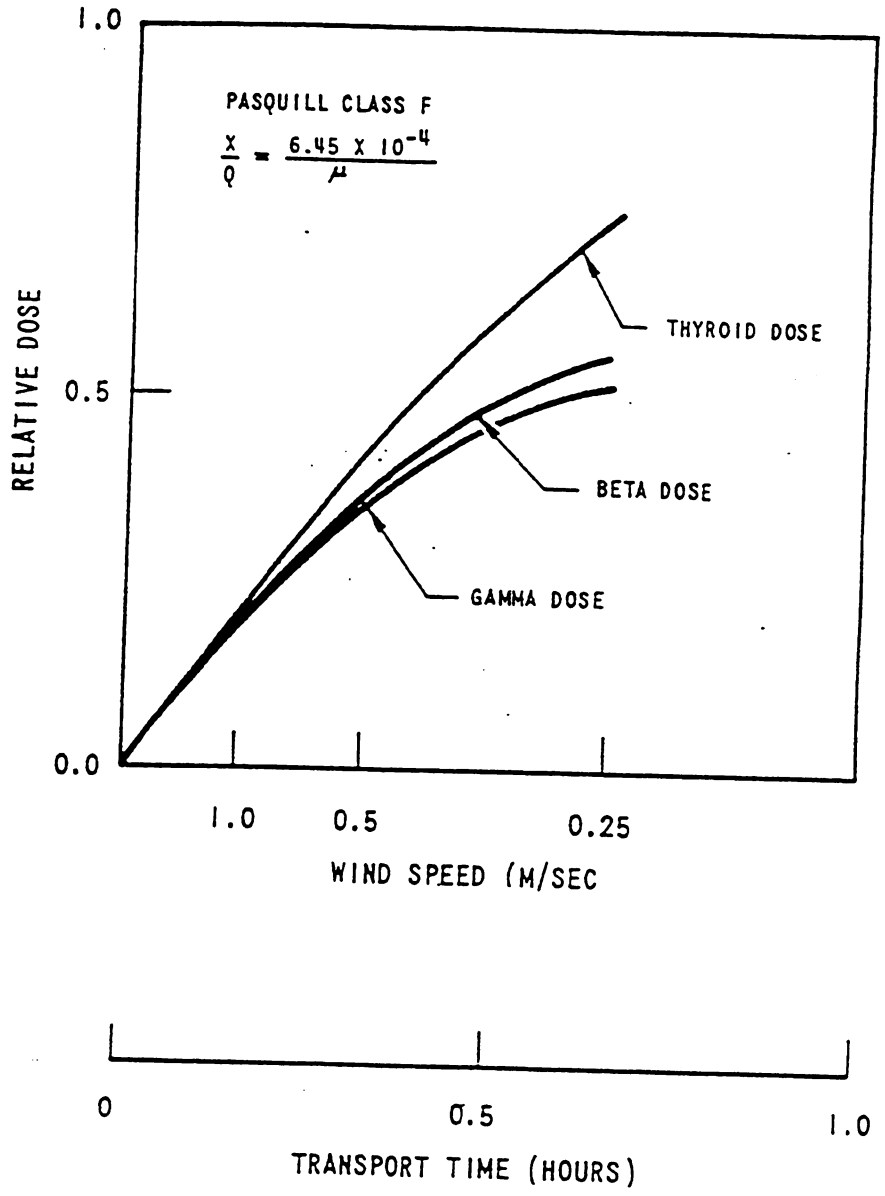

Figure 15B-11 Effect of Decay During Downwind Transport for Various Wind Speeds on Doses at the Exclusion Area Boundary in the First Two Hours of Exposure

Revised by Amendment 10

APPENDIX 15C

FUEL ROD MODEL DISCUSSION

(HISTORICAL INFORMATION)

LIST OF FIGURES

Number	Title
15C-1	Engineering Hoop Stress (KPSI)

APPENDIX 15C

FUEL ROD MODEL DISCUSSION

(HISTORICAL INFORMATION)

The U.S. Nuclear Regulatory Commission (NRC) issued a letter dated November 9, 1979 [1] to operators of light water reactors regarding fuel rod models used in Loss of Coolant Accident (LOCA) ECCS evaluation models. This letter describes a meeting called by the NRC on November 1, 1979 to present draft report NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" [2]. At the meeting, representatives of NSSS vendors and fuel suppliers were asked to show how plants licensed using their LOCA/ECCS evaluation model continued to conform to 10 CFR Part 50.46 in view of the new fuel rod models presented in reference 2. Westinghouse representatives presented information on the fuel rod models used in analyses for plants licensed with the Westinghouse ECCS evaluation model and discussed the potential impact of fuel rod model changes on results of those analyses. That information was formally documented in letter NS-TMA-2147, dated November 2, 1979 [3], and formed the basis for the Westinghouse conclusion that the information presented in Reference 2 did not constitute a safety problem for Westinghouse plants and that all plants conformed with NRC regulations. The NRC requested [1] that operators of light water reactors provide, within sixty (60) days, information which would enable the staff to determine, in light of the fuel rod model concerns, whether or not further action was necessary.

This Appendix provides information on the LOCA analysis of Sequoyah Nuclear Plant unit 1 required to respond to such a request. Note, however, that a significant amount of discussion and information exchange between Westinghouse and the NRC has transpired since Reference 3 was prepared and the basis for demonstrating compliance with 10 CRF Part 50 has been modified. The following is an outline of the significant events that have occurred since November 2, 1979 and provides an update on this situation.

As a result of compiling information for Reference 3, Westinghouse recognized a potential discrepancy in the calculation of fuel rod burst for cases having clad heatup rates (prior to rupture) significantly lower than 25°F per second. This issue was reported to the NRC staff, by telephone, on November 9, 1979, and although independent of the NRC fuel rod model concern, the combined effect of this issue and the effect of the NRC fuel rod models had to be studied. Details of the work done on this issue were presented to the NRC on November 13, 1979 and documented in letter NS-TMA-2163 dated November 16, 1979 [4]. That work included development of a procedure to determine the clad heatup rate prior to burst and re-evaluation of operating Westinghouse plants with consideration of a modified Westinghouse fuel rod burst model. As part of this re-evaluation, the Westinghouse position on NUREG-0630 was reviewed and it was still concluded that the information presented in Reference 1 did not constitute a safety problem for plants licensed with the Westinghouse ECCS evaluation model.

On December 6, 1979, NRC and Westinghouse personnel discussed the information thus far presented. At the conclusion of that discussion, the NRC staff requested Westinghouse to provide further detail on the potential impact of modifications to each of the fuel rod models used in the LOCA analysis and to outline analytical model improvements in other parts of the

analysis and the potential benefit associated with those improvements. This additional information was compiled from various LOCA analysis results and documented in letter NS-TMA-2174 dated December 7, 1979 [5].

Another meeting was held in Bethesda on December 20, 1979 where NRC and Westinghouse personnel established:

1. The currently accepted procedure for assessing the potential impact on LOCA analysis results of using the fuel rod models presented in Reference 1 and

2. Acceptable benefits resulting from analytical model improvements that would justify continued plant operation for the interim until differences between the fuel rod models of concern are resolved.

The information following on pages 15C-3 through 15C-6 is expected to satisfy the NRC request for information on SQN which will enable the NRC to determine whether or not further action is necessary.

Part of the Westinghouse effort provided to assist in the resolution of these LOCA fuel rod model differences is documented in letter NS-TMA-2175, dated December 10, 1979 [6], which contains Westinghouse comments on draft NUREG-0630. As stated in that letter, Westinghouse believes the current Westinghouse models to be conservative and to be in compliance with Appendix K.

A. Evaluation of the potential impact of using fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for Sequoyah.

This evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT - 0.6 IMP MIXING

WESTINGHOUSE ECCS EVALUATION MODEL VERSION - FEB '78

CORE PEAKING FACTOR - 2.25

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 1705°F = PCT_B

ELEVATION - 6.5 Feet

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2143°F = PCT_N

ELEVATION - 7.5 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION	10	Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION	10	Percent

Maximum temperature for this non-burst node occurs when the core reflood rate is less than 1.0 inch per second and reflood heat transfer is based on the (STEAM COOLING) calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - 7.25 Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - 34.9 Percent

1. BURST NODE

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate increases significantly at temperatures above 2200°F, individual effects (such as WPCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges, but a simultaneous change in FQ which causes the PCT to remain in the neighborhood of 2200°F justifies use of this evaluation procedure.

From NS-TMA-2174:

For the Burst Node of the clad:

- 0.01 ΔFQ 150°F BURST NODE ΔPCT
- Use of the NRC burst model and the revised Westinghouse burst model could require an FQ reduction of 0.027
- The maximum estimated impact of using the NRC strain model is a required FQ reduction of 0.03.

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (0.027 + .03)\,(150°F/.01) = 855°F$$

Margin to the 2200°F limit is:

$$\Delta PCT_2 = 2200°F - 1705\ PCT_B = 495\ °F$$

The FQ reduction required to maintain the 2200°F clad temperature limit is:

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\left(\frac{.01\ \Delta FQ}{150°F}\right)$$

$$= \left(\frac{855 - 495}{}\right)\left(\frac{.01}{150}\right)$$

$$= .024 \text{ (but not less than zero)}$$

2. NON-BURST NODE

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspect of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated.

The effect of the change in pellet-clad gap conductance in the non-burst node was determined by performing a sensitivity study using various rate dependent burst curves. The results of these studies are presented in Table I. It is seen that the most representative case, as defined by the relationship between the actual clad heatup rate and the heatup rate used in the burst mode, is the 8.4°F/sec ramp rate. The increase in calculated peak clad temperature, ΔPCTs, for this case was 19°F. It is also noted that hot rod burst occurs 6.9 seconds earlier and the blockage increases by 9.7% over the base case results.

The sensitivity study included the effect of blockage as well as the effect of the change in gap conductance on the peak clad temperature. The value of ΔPCT_3 includes an increased blockage penalty. Thus in the evaluation of the increase in PCT, ΔPCT_4, due to blockage, the 45% blockage value from the sensitivity rather than the 34.9% value from the limiting break LOCA analysis identified at the beginning of section A will be used in order to prevent a double accounting of the 9.7% blockage increase noted in the sensitivity study.

A comparison of the ramp dependent burst curves used in this study with the NUREG-0630 is shown in Figure 15C-1. It can be seen that the burst temperatures predicted by the Westinghouse rate dependent models predict higher burst temperatures than the ORNL correlations.

Detailed discussion of these differences are provided in the December 10 letter (NS-TMA-2175), including an explanation of the conservatism of the model and demonstration of compliance with Appendix K.

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (see above discussion) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore,

$$\Delta PCT_4 = 1.25°F\ (50 - \text{PERCENT CURRENT BLOCKAGE})$$

$$+ 2.36°F\ (75\text{-}50)$$

$$= 1.25\ (50 - 45) + 2.36\ (75\text{-}50)$$

$$= 65.25°F$$

If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 19.0 + 65.25 = 84.25°F$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200°F - PCT_N = 2200 - 2143 = 57°F$$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$$\Delta FQ_N = (\Delta PCT_5 - \Delta PCT_6)\left(\frac{.01\Delta FQ}{10°F\Delta PCT}\right) = (84.25 - 57)\left(\frac{.01}{10}\right) = .027$$

$\Delta FQ_N = .027$ but not less than zero.

The peaking factor reduction required to maintain the 2000°F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_N.

B. The NRC has recently approved the removal of the 65°F uncertainty on hot rod fuel pellet temperature for ECCS analysis. The effect of removing this uncertainty on the calculated PCT has been determined based on previously established sensitivities performed to quantify this effect (WCAP-9180). From these, it is estimated that this reduction in applied model uncertainty would result in a decrease in calculated PCT of 15°F for UHI plants. Applying the same sensitivity used in calculating ΔFQ_N.

$$\Delta FQ_{CREDIT} = 15°F\left(\frac{.01\Delta FQ}{10°F\ \Delta PCT}\right) = .015$$

C. The peaking factor limit adjustment required to justify plant operation for this interim period is determined as the appropriate WFQ credit identified in Section B above, minus the WFQ[PENALTY] calculated in Section A above (but not greater than zero).

FQ ADJUSTMENT = .015 - .027 = .012

D. The revised peaking factor is then FQ FSAR minus the FQ adjustment, or:

FQ = 2.25 - .012 = 2.237

References

1. U.S. Nuclear Regulatory Commission letter, D. G. Eisenhut to Utilities with Operating Light Water Reactors, November 9, 1979.

2. NUREG-0630, (Draft) Powers, D. A., Meyer, R. O., November 8, 1979, Cladding Swelling and Rupture Models for LOCA Analysis.

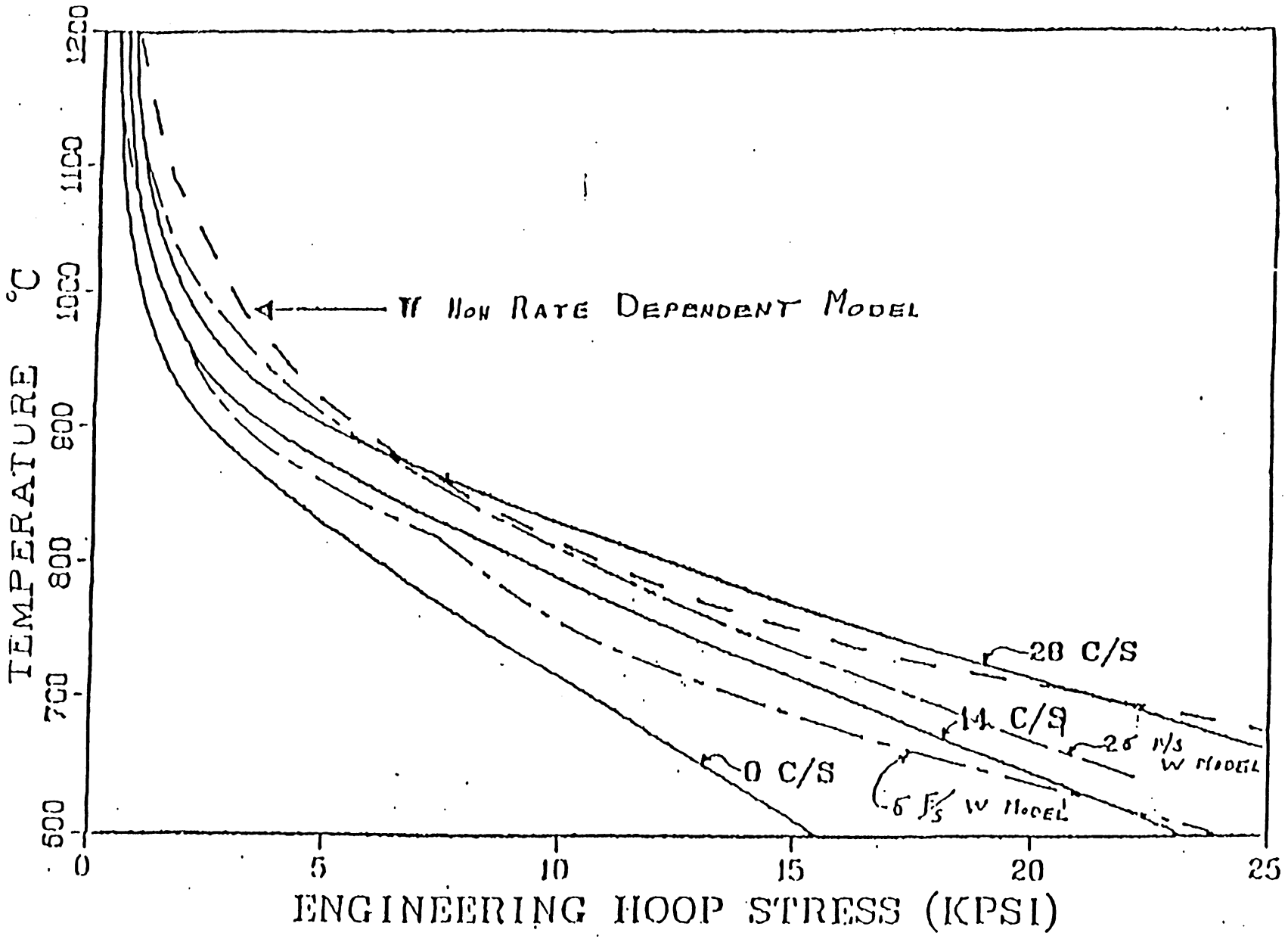

FIGURE 15C-1 H large-break model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate. (Historical Information)

APPENDIX 15D
PRIMARY COOLANT ACTIVITY MODEL USED TO EVALUATE THE RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

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15D.3	Iodine Spike Modeling	15D-2
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15D-2	Iodine Appearance Rates at Equilibrium Primary Coolant Concentration of 0.35 (micro-curies/gram) D.E. I-131 (Ci/min)
15D-3	Secondary Coolant Activity (micro-curies/gram)

APPENDIX 15D

PRIMARY COOLANT ACTIVITY MODEL USED TO EVALUATE THE RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

15D.1 Introduction

This appendix describes the model used to define the primary and secondary coolant fission product concentrations that are used in the design basis accident radiological consequences analyses. The accidents incorporating this modeling are:

1. Loss of A.C. Power
2. Steam Line Break
3. Steam Generator Tube Rupture

15D.2 Primary Coolant Equilibrium Activity

There is a Technical Specification limit of 100/E-Bar µCi/gm for the gross activity in the primary coolant (excluding iodines and nuclides having a half-life shorter than 15 minutes). This translates to 210 µCi/gm and it is conservatively assumed that all of this activity exists as noble gases. Table 15.D-1 lists the primary coolant noble gas concentrations when operating at the Technical Specification limit. The individual nuclide concentrations were determined based on the isotopic distribution that is expected during normal operation and then factoring up to reach a total noble gas concentration of 210 µCi/gm.

For the primary coolant iodines, the Technical Specification limit is 0.35 µCi/gm of Dose Equivalent I-131 for equilibrium operation. Table 15D-1 lists the iodine concentrations associated with operation at the Technical Specification limit. The individual isotope concentrations were determined based on the isotopic distribution that is expected during normal operation and using the thyroid dose conversion factors listed in Appendix 15A.

15D.3 Iodine Spike Modeling

Iodine spikes (periods of time during which the iodine appearance rate in the primary coolant is elevated) are part of the accident analysis modeling for the steam line break, the steam generator tube rupture, and the loss of A.C. power events. These accidents are analyzed considering both the situation of a pre-existing iodine spike and the situation in which the event initiates an iodine spike (accident-initiated spike).

For these accident analyses the pre-existing iodine spike is assumed to be the maximum iodine concentration allowed by Technical Specification Figure 3.4-1 when at full power (21 µCi/gm Dose Equivalent I-131). This is the limit applied for a transient duration of <48 hours. The pre-existing spike coolant concentrations are a factor of 60 greater than the concentrations reported in Table 15D-1 for equilibrium operation at 0.35 µCi/gm Dose Equivalent I-131.

For analysis modeling the accident-initiated spike it is assumed that the initial iodine concentration in the primary coolant is at the Technical Specification value of 0.35 µCi/gm Dose Equivalent I-131 and that the equilibrium iodine appearance rates are those which support this coolant concentration. As a result of the accident, an iodine spike is initiated which increases the iodine appearance rate by a factor of 500 resulting in an increasing primary coolant iodine concentration. The iodine appearance rates associated with maintaining the primary coolant concentration of 0.35 µCi/gm are provided in Table 15D-2.

15D.4 Secondary Coolant Activity

There is no noble gas activity retained in the secondary coolant.

The iodine activity concentration of the secondary coolant at the time an accident occurs is assumed to be at the Technical Specification limit of 0.1 µCi/gm Dose Equivalent I-131 (see Table 15D-3). Iodine spiking phenomena are assumed.

TABLE 15D-1

PRIMARY COOLANT ACTIVITY (μCi/gm)

Noble gas (100/E-Bar)	
Kr-85m	6.74E0
Kr-85	1.05E1
Kr-87	6.35E0
Kr-88	1.18E1
Xe-131m	2.58E1
Xe-133m	2.83E0
Xe-133	9.98E1
Xe-135m	5.48E0
Xe-135	3.56E1
Xe-138	5.09E0
Iodines (Equilibrium concentration of 0.35 μCi/gm D.E. I-131, no spike)	
I-131	2.03E-1
I-132	9.57E-1
I-133	6.34E-1
I-134	1.55E0
I-135	1.18E0

SQN-18

TABLE 15D-2

IODINE APPEARANCE RATES AT EQUILIBRIUM PRIMARY COOLANT CONCENTRATION OF 0.35 (μCi/gm) D.E. I-131 (Ci/min)

I-131	1.127E-1
I-132	1.646E0
I-133	4.248E-1
I-134	5.583E0
I-135	1.121E0

TABLE 15D-3

SECONDARY COOLANT ACTIVITY (μCi/gm)

Noble gas	none
Iodine (0.1 μCi/gm D.E. I-131)	
I-131	0.058
I-132	0.274
I-133	0.181
I-134	0.443
I-135	0.338