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15.0 ACCIDENT ANALYSES

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

1. Condition I: Normal Operation and Operational Transients

2. Condition II: Faults of Moderate Frequency

3. Condition III: Infrequent Faults

4. Condition IV: Limiting Faults

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

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

The specific accident sequences analyzed in this chapter include those required by Revision 1 of Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, and others considered significant for V. C. Summer. Because the V. C. Summer design differs from other plants, some of the accidents identified in Table 15-1 of Regulatory Guide 1.70, Revision 1, are not applicable to this plant. Some comments on these items are as follows:

(Item 10) - There are no pressure regulators or regulating instruments in the Westinghouse PWR design whose failure could cause heat removal greater than heat generation.

(Item 11) - Reactor coolant flow controller is not a feature of the Westinghouse PWR designs. Treatment of the performance of the reactivity controller in a number of accident conditions is offered in this chapter.

(Item 12) - The analysis of specific effects of internal and external events such as major and minor fires, floods, storms, or earthquakes are generally discussed in Chapter 3. Refer to Section 3.1.2.1 for guidance on which FSAR sections specifically address GDCs 2, 3, and 4.

(Item 22) - No instrument lines from the Reactor Coolant System boundary in the Westinghouse PWR design penetrate the Reactor Building [(1)].

(Item 26) - Control room habitability is discussed. Chapter 7 contains an analysis showing that the plant can be brought to, and maintained in, the hot shutdown condition from outside the control room.

(Item 27) - Overpressurization of the Residual Heat Removal System is not considered credible due to the isolation valve interlocks described in Section 7.6.

(Item 28) - Loss of condenser vacuum is covered by the analyses of Section 15.2.7, Loss of External Electric Load and/or Turbine Trip.

(Item 29) - Turbine trip is covered by the analyses of Section 15.2.7, Loss of External Electric Load and/or Turbine Trip.

(Item 30) - Loss of the Service Water System is discussed in Section 9.2.

(Item 31) - Loss of one D-C system is discussed in Chapter 8.

(Item 33) - Turbine trip with failure of generator breakers to open is discussed in Chapter 10. The effect of turbine trip on the Reactor Coolant System are presented in Section 15.2.7.

(Item 34) - Loss of the Instrument Air System is discussed in Chapter 9.

(Item 35) - Loss of the turbine gland seal is of no significance for PWRs.

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(1) For the definition of the Reactor Coolant System boundary, refer to ANSI-N18.2, “Nuclear Safety Criteria for the Design of Stationary PWR Plants,” Section 5, 1973.

The Nuclear Steam Supply System (NSSS) is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. Chapter 17 discusses the Quality Assurance Program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the NSSS, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-1 will be available for mitigation of the events discussed in Chapter 15. In determining which systems are necessary to mitigate the effects of these postulated events, Westinghouse utilizes the classification system of ANSI-N18.2, 1973. The design of “systems important to safety” (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53 in the application of the single failure criterion.

TABLE 15.0-1

EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
1.	Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	Power range high flux (low s.p.), manual	-	-	-
2.	Uncontrolled RCCA Bank Withdrawal at Power	Power range high flux, OTΔT, hi pressurizer pressure, manual	-	Pressurizer safety valves, steam generator safety valves	-
3.	RCCA Misalignment	Manual	-	-	-
4.	Uncontrolled Boron Dilution	Source range high flux, power range high flux, OTΔT, manual	Low insertion limit annunciators for boration	-	-
5.	Startup of an Inactive Reactor Coolant Loop	Power range high flux, manual	-	-	-
6.	Loss of External Electrical Load and/or Turbine Trip	High pressurizer pressure, OTΔT, manual	-	Pressurizer safety valves, steam generator safety valves	-
7.	Loss of Normal Feedwater	Steam generator lo-lo level, manual	Steam generator lo-lo level	-	One motor driven emergency feedwater pump
8.	Loss of Offsite Power to the Station Auxiliaries	Same as 7	Same as 7	Same as 7	Same as 7

TABLE 15.0-1 (Continued)

EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment	
9.	Excess Heat Removal due to Feedwater System Malfunctions	Power range high flux, high steam generator level, OTΔT, OPΔT, manual	High Steam generator level, low pressurizer pressure	Feedwater isolation valves, turbine trip	-	RN 10-033
10.	Excessive Load Increase Incident	Power range high flux, OTΔT, OPΔT, manual	-	Pressurizer safety valves, steam generator safety valves	-	00-01
11.	Accidental Depressurization of the RCS	Pressurizer low pressure, OTΔT, manual	-	-	-	
12.	Major Rupture of Main Steam Line	SIS, manual	Low pressurizer pressure, low compensated steam line pressure, hi-1 containment pressure, manual	Feed line isolation valves, steam line isolation valves	Emergency feedwater system, SI equipment minus either one SI charging pump, or one diesel generator	
13.	Complete Loss of Forced Reactor Coolant Flow	Low flow, underpower, underfrequency, manual	-	-	-	
14.	Rupture of a Control Rod Drive Mechanism Housing	Power range high flux, manual	-	-	-	
15.	Single RCCA Withdrawal at Full Power	OTΔT, manual	-	-	-	
16.	Major Rupture of a Main Feedwater Line	Low steam generator level plus steam/feed mismatch, SIS, manual	High containment pressure, high pressurizer pressure, steam generator low-low water level, low compensated steam line pressure	Steam line isolation valves, feed line isolation, pressurizer safety valves, steam generator safety valves	Emergency feedwater pumps	00-01

TABLE 15.0-1 (Continued)

EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment	
17. Large Break LOCA	Reactor Trip system	Engineered safety features actuation system	Service water system, component cooling water system	Emergency core cooling system, containment heat removal system, emergency power system.	
18. Small Break LOCA	Reactor trip system	Engineered safety features actuation system	Service water system, component cooling water system, generator safety and/or relief valves	Emergency core cooling system, emergency feedwater system, containment heat removal system, emergency power system.	
19. Steam Generator Tube Rupture	Reactor trip system	Engineered safety features actuation system	Service water system, component cooling water system, steam generator shell side fluid operating system, steam generator safety and/or relief valve, steam line isolation valves.	Emergency core cooling system, emergency feedwater system, emergency power system	02-01 00-01

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. Since 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 follows:

1. Steady state and shutdown operations

   Mode 1 - Power operation (> 5% of rated thermal power)

   Mode 2 - Startup ($K_{eff} \geq 0.99$, $\leq 5\%$ of rated thermal power)

   Mode 3 - Hot standby ($K_{eff} < 0.99$, $T_{avg} \geq 350°F$)

   Mode 4 - Hot shutdown (subcritical, residual heat removal system in operation, $K_{eff} < 0.99$, $200°F < T_{avg} < 350°F$)

   Mode 5 - Cold shutdown (subcritical, residual heat removal system in operation, $K_{eff} < 0.99$, $T_{avg} \leq 200°F$)

   Mode 6 - Refueling ($K_{eff} \leq 0.95$, $T_{avg} \leq 140°F$)

2. Operation with permissible deviations

   Various deviations which may occur during continued operation as permitted by the plant Technical Specifications[1] 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 clad defects

c. Activity in the reactor coolant

    (1) Fission products

    (2) Corrosion Products

    (3) Tritium

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

e. Testing as allowed by the Technical Specifications

3. Operational transients

    a. Plant heatup and cooldown (up to 100°F/hour for the Reactor Coolant System; 200°F/hour cooldown for the pressurizer, 100°F/hour heatup 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 control system setpoint study[2] has been 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 is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for 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.

Due to the revised parameters associated with replacement steam generators, the control system setpoint study described in Reference [2] was reviewed for applicability. Several of the setpoints required revision in order to provide acceptable control system performance for the entire core life and over the entire design range of Reactor Coolant System operating conditions. The control systems requiring revision included the steam generator level control (due to the replacement steam generator), rod control, steam dump, and pressurizer level control systems (due to the revised range of the full power T_{avg}).

15.1.2 INITIAL POWER CONDITIONS ASSUMED IN ACCIDENT ANALYSES

15.1.2.1 Power Rating

Table 15.1-1 lists the principal power rating values which are assumed in analyses performed in this section. This rating is the guaranteed Nuclear Steam Supply System (NSSS) thermal power output. 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 for some accidents) is assumed. The thermal power values used for each transient analyzed are given in Table 15.1-4.

15.1.2.2 Initial Conditions

For most accidents which are Departure from Nucleate Boiling limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit Departure from Nucleate Boiling Ratio, as described in Reference [3]. This procedure is known as the "Revised Thermal Design Procedure" (RTDP) and these accidents utilize the WRB-1 and WRB-2 Departure from Nucleate Boiling correlations (References [4] and [5]). RTDP allowances may be more restrictive than non-RTDP allowances. The initial conditions for other key parameters are selected in such a manner to maximize the impact on Departure from Nucleate Boiling Ratio. Minimum measured flow is used in all RTDP transients.

For accident evaluations that are not Departure from Nucleate Boiling-limited, or for which the Revised Thermal Design Procedure is not employed, the initial conditions are obtained by adding maximum steady state errors to rated values. The following steady state errors are considered.

(1)	Core power	± 2.0% allowance calorimetric error
(2)	Average RCS temperature	+ 4.0°F / - 5.3°F allowance for deadband and measurement error
(3)	Pressurizer pressure	± 50 psi allowance for steady state fluctuations and measurement error
(4)	RCS Flow	2.1% allowance on flow measurement error

15.1.2.3 <u>Power Distribution</u>

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operating instructions. 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 and the Core Operating Limits Report (COLR).

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 15.1-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is discussed in Section 4.4.3.

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

The value of peak kW/ft can be directly related to fuel temperature using steady state fuel rod performance predictions. For transients that are slow with respect to the fuel rod thermal time constant (approximately 5 seconds), the fuel temperatures follow the steady state predictions. For transients that are fast with respect to the fuel rod thermal time constant (for example, rod ejection), a detailed heat transfer calculation is made. 00-01

15.1.3 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

A reactor trip signal acts to open 2 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 15.1-2.

Reference is made in that table to the overtemperature and overpower ΔT trip shown on Figure 15.1-1. This figure presents the allowable reactor coolant loop average temperature and ΔT for the design flow and the NSSS Design Thermal Power distribution as a function of primary coolant pressure. The boundaries of operation defined by the Overpower ΔT trip and the Overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, a trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit values (1.43 and 1.42 for typical cell, and thimble cell respectively) for RTDP accidents. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit values. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

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

The limit values, which were used as the DNBR limits for all accidents analyzed with the Revised Thermal Design Procedure, are conservative compared to the actual design DNBR values required to meet the DNB design basis.

The difference between the limiting trip point, assumed for the analysis, and the normal trip point represents an allowance for instrumentation channel error and setpoint error. During startup tests, it is demonstrated that actual instrument errors and time delays are equal to or less than the assumed values.

15.1.4 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS - POWER RANGE NEUTRON FLUX

The instrumentation drift and calorimetric uncertainties used in establishing the Technical Specification power range neutron flux (High) reactor trip setpoint are included in the report "Westinghouse Setpoint Methodology for Protection Systems - Virgil C. Summer", WCAP-11770 (Class 2)[19] and WCAP-11814 (Class 3)[20]. These uncertainties are based on the installed instrumentation, on the calibration limits, and on the calibration methods for the power range neutron flux channels. The Technical Specification Power Range Neutron Flux (High) reactor trip setpoint preserves the analyzed limit used in the safety analysis. 99-01

The calorimetric uncertainty is the uncertainty determined for the core thermal power measurement as obtained from secondary side plant measurements. The power range neutron flux channels (sum of the top and bottom sections) are required by Technical Specifications to be calibrated (set equal) to the measured power on a daily basis. The calorimetric (power measurement) uncertainty is included in the report "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology - Virgil C. Summer", WCAP-13812 (Class 2)[21] and WCAP-13813 (Class 3)[22]. 99-01

A high accuracy calorimetric Reactor Coolant System flow measurement is performed at the beginning of each fuel cycle to calibrate (set equal to) the Reactor Coolant System elbow tap flow channels. This calibration or normalization allows a simplified periodic Reactor Coolant System flow measurement that is independent of feedwater venturi fouling. For the high accuracy Reactor Coolant System flow measurement, feedwater venturi fouling is accounted for by inspecting and cleaning (if necessary) each venturi at each refueling outage. 02-01 RN 00-078

15.1.5 ROD CLUSTER CONTROL ASSEMBLY INSERTION CHARACTERISTICS

The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod cluster control assemblies and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85 percent of the rod cluster travel. For accident analyses the insertion time to dashpot entry is conservatively taken as 2.7 seconds. The rod cluster control assembly position versus time assumed in accident analyses is shown in Figure 15.1-2.

Figure 15.1-3 shows the fraction of total negative reactivity insertion versus rod position for a core where the axial distribution is skewed to the lower region of the core. This curve is used to compute the negative reactivity insertion versus time following a reactor trip which is input to all point kinetics core models used in transient analyses.

There is inherent conservatism in the use of this curve in that it is based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with 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 15.1-4. The curve shown in this figure was obtained from Figures 15.1-2 and 15.1-3. A total negative reactivity insertion following a trip of 4.8 percent ΔK is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Tables 4.3-2 and 4.3-3.

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

15.1.6 REACTIVITY COEFFICIENTS

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

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses such as loss of reactor coolant from cracks or ruptures in the reactor coolant system do not depend on reactivity feedback effects. The values used are given in Table 15.1-4. Reference is made in that table to Figure 15.1-5 which shows the upper and lower bound Doppler power coefficients as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values are treated on an event by event basis.

15.1.7 FISSION PRODUCT INVENTORIES

15.1.7.1 Radioactivity in the Core

The fission product inventories for all isotopes, which are important from a health hazards standpoint, were calculated using the ORIGEN code[17] and are given in Table 15.1-5. This code uses a data base of fission product yields, cross sections, and decay constants taken from the ENDFB-IV/V fission product library. The calculation of the core iodine fission product inventory is consistent with TID-14844[18]. The ORIGEN code takes into account fuel burnup as well as fission product buildup and decay. Continuous operation at full power is assumed during the fuel residence time to provide an upper limit estimate of the fission product inventory. The isotopes included in Table 15.1-5 are the isotopes controlling from considerations of inhalation dose (iodines) and from direct dose due to immersion (noble gases).

The isotopic yields used in the calculations are from the data of APED-5398 [23], utilizing 00-01
the isotopic yield data for thermal fissioning of U-235 as the sole fissioning source. The change in fission product inventory resulting from the fissioning of other fissionable atoms has been reviewed. The results of this review indicated that inclusion of all fission source data would result in small (less than 10 percent) change in the isotopic inventories due to the overall conservatism.

15.1.7.2 Radioactivity in the Fuel Pellet Clad Gap

This section was deleted by Amendment 96-02.

15.1.8 RESIDUAL DECAY HEAT

Residual heat in a subcritical core consists of:

1. Fission product decay energy,

2. Decay of neutron capture products, and

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

These constituents are discussed separately in the following paragraphs.

15.1.8.1 Fission Product Decay Energy

For short times (< 10^3 seconds) after shutdown, data on yields of short half life isotopes are sparse. Very little experimental data are available for the γ-ray contributions and even less for the ß-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[6], Dudziak[7], and Teage[8]. Of these 3 selections, Shure's curve is the highest and it is based on the data of Stehn and Clancy[9] and Obenshain and Foderaro[10].

The fission product contribution to decay heat that has been assumed in the LOCA accident analyses is the curve of Shure increased by 20% for conservatism. This curve with the 20% factor included is shown on Figure 15.1-6. For the non-LOCA analyses the 1979 ANS decay heat curve is used[11]. Figure 15.1-7 presents this curve as a function of time after shutdown.

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:

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

$$\frac{P_2}{P_0} = \frac{E_{\gamma 2} + E_{\beta 2}}{200\,\text{Mev}} c(1+\alpha) \left[\frac{\lambda_2}{\lambda_1 - \lambda_2} \left(e^{-\lambda_2 t} - e^{-\lambda_1 t} \right) + e^{-\lambda_2 t} \right] \text{watts / watt}$$

Where:

P_1/P_o	=	the energy from U-239 decay
P_2/P_o	=	the energy from Np-239 decay
t	=	the time after shutdown (seconds)
$c(1+\alpha)$	=	the ratio of U-238 captures to total fissions = 0.6 (1+0.2)
λ_1	=	the decay constant of U-239 = 4.91×10^{-4} seconds^{-1}
λ_2	=	the decay constant of Np-239 = 3.41×10^{-6} seconds^{-1}
$E_{\gamma 1}$	=	total γ-ray energy from U-239 decay = 0.06 Mev
$E_{\gamma 2}$	=	total γ-ray energy from Np-239 decay = 0.30 Mev
$E_{\beta 1}$	=	total β-ray energy from U-239 decay = 1/3 x 1.18 Mev
$E_{\beta 2}$	=	total β-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 15.1-6 as it is used in the LOCA analysis. 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 fission, is neglected. For the non-LOCA analysis, the decay of U-238 capture products is included as an integral part of the 1979 decay heat curve presented as Figure 15.1-7.

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 6 delayed neutron groups.

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

15.1.8.4 Distribution of Decay Heat Following Loss of Coolant Accident

During a loss of coolant accident the core is rapidly shut down by void formation or rod cluster control assembly insertion, or both, and long term shutdown is assured by borated ECCS water. A large fraction of the heat generation to be considered comes from fission product decay gamma rays.

This heat is not distributed in the same manner as steady-state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady-state factor of 97.4 percent which represents the fraction of heat generated within the clad and pellet drops to 95 percent for the hot rod in a loss of coolant accident.

For example, consider the transient resulting from the postulated double ended break of the largest reactor coolant system pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or three percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining two percent being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

15.1.9 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, such as the SATAN-VI code used in the analysis of the reactor coolant system pipe rupture (see Section 15.4), and which consequently have a direct bearing on the analysis of the accident itself, are summarized in their respective accident analyses sections. The codes used in the analyses of each transient have been listed in Table 15.1-2.

15.1.9.1 FACTRAN

FACTRAN calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod (see Figure 15.1-8) and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which exhibits the following features simultaneously:

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

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

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

The gap heat transfer coefficient is calculated according to an elastic pellet model. 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 clad 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 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 sufficient to reduce the gap to zero by elastic deformation of both. This contact pressure determines the heat transfer coefficient.

FACTRAN is further discussed in Reference [12].

15.1.9.2 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 multiloop system by modeling the reactor core and vessel, hot and cold leg piping, steam generator (tube and shell-sides), reactor coolant pumps, and the pressurizer with up to 4 reactor coolant loops. 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 pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control and pressurizer pressure control. The Safety Injection System, including the accumulators, is also modeled.

LOFTRAN is 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 DNBR based on the input from the core limits illustrated on Figure 15.1-1. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference [13].

15.1.9.3 LEOPARD

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

LEOPARD is further described in Reference [14].

15.1.9.4 TURTLE

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

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

TURTLE is further described in Reference [15].

15.1.9.5 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 1, 2, and 3 dimensions. The code uses 6 delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating point wise 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 are provided, e.g., channel wise power, axial offset, enthalpy, volumetric surge, point wise power, and fuel temperatures.

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

TWINKLE is further described in Reference [16].

15.1.9.6 THINC

The THINC Code is described in Section 4.4.3.

15.1.10 REFERENCES

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TABLE 15.1-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Licensed Core Thermal Power	2900 MWt
Analyzed Core Thermal Power (unless otherwise noted)	2900 MWt
Thermal power generated by the reactor coolant pumps	12 MWt
Nuclear steam supply system thermal power output	2912 MWt

TABLE 15.1-2

TRIP POINTS AND THE TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delay (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
Overtemperature ΔT	Variable see Fig. 15.1-1	8.5[(1)]
Overpower ΔT	Variable see Fig. 15.1-1	8.5[(1)]
High pressurizer pressure	2450 psig	2.0
Low pressurizer pressure	1775 psig	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage Trip	(2)	1.5
Turbine Trip	Not applicable	1.0
Low-low steam generator level	16.7% of narrow range span	2.0
High-high steam generator level trip of the feedwater pumps and closure of the feedwater system valves *, and turbine trip	100% of narrow range span	2.0 * 13.0 (for Feedwater isolation)

1. Total time delay (including RTD time response, and trip circuit channel electronics delay) from the time the temperature difference in coolant loops exceeds the trip setpoint until the rods are free to fall. 02-01

2. A specific undervoltage setpoint was not assumed in the safety analysis.

TABLE 15.1-4

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED 02-01

Event	Computer Codes Utilized	Assumed Reactivity Coefficients: MTC [a] pcm/°F [d]	MDC [a] Δk/gm/cc	Doppler [b]	Initial NSSS Thermal Power Output Assumed, [c] MWt	
CONDITION II						
Uncontrolled RCCA Bank Withdrawal from a subcritical Condition	TWINKLE FACTRAN THINC	+7	-	Consistent with lower limit on figure 15.1-5	0	
Uncontrolled RCCA bank Withdrawal at power	LOFTRAN	+7	0.50	Lower and upper	2915	
RCCA Misoperation	LOFTRAN THINC	-	-	-	2912	
Uncontrolled Boron Dilution					0 and 2912	
Partial Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC	+5	-	Upper	2915	
Startup of an Inactive Reactor Coolant Loop	LOFTRAN FACTRAN THINC	-	0.50	Lower	1805	
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	+7	0.50	Lower and Upper	2912	
Loss of Normal Feedwater	LOFTRAN	0	-	Upper	2915	
Loss of Offsite Power to the Station Auxiliaries (station blackout)	LOFTRAN	0	-	Upper	2915	
Excessive Heat Removal due to Feedwater System Malfunctions	LOFTRAN THINC	-	0.50	Lower	0 and 2912	RN 10-033
Excessive Load Increase	LOFTRAN	-	0 and 0.50	Lower and Upper	2912	02-01

TABLE 15.1-4 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED 02-01

Event	Computer Codes Utilized	Assumed Reactivity Coefficients MTC [a] pcm/°F [d]	MDC [a] Δk/gm/cc	Doppler [b]	Initial NSSS Thermal Power Output Assumed, [c] MWt
CONDITION II, Continued					
Accidental Depressurization of the Reactor Coolant System	LOFTRAN	+7	-	Lower	2915
Accidental Depressurization of the Main Steam System	LOFTRAN	-	Function of the moderator density. See Sec. 15.2.13 (Fig. 15.2-46)	See Fig. 15.4-75	0 (subcritical) 02-01
Spurious Operation of the SIS at power	LOFTRAN	+7	0.50	Lower and Upper	2915
CONDITION III					
Complete Loss of Forced Reactor Coolant Flow	LOFTRAN FACTRAN THINC	+5	-	Upper	2915
Inadvertent loading of a fuel assembly into an improper position	LEOPARD, TURTLE	-	-	-	2915 [e]
Single RCCA withdrawal at full power	TURTLE, THINC, LEOPARD	-	-	-	2915 [e] 00-01
Loss of reactor coolant from small ruptured pipes or from cracks in large pipe which actuate emergency core cooling	NOTRUMP, SBLOCTA	-	-	-	2915 [e]

TABLE 15.1-4 (Continued)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

02-01

Event	Computer Codes Utilized	Assumed Reactivity Coefficients: MTC [a] pcm/°F [d]	MDC [a] Δk/gm/cc	Doppler [b]	Initial NSSS Thermal Power Output Assumed, [c] 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)	Function of evaluation model see section 15.4.1	Function of moderator density see section 15.4.1	-	Function of fuel temp see section 15.4.1	2900 [e]
Major Secondary System Pipe Ruptures up to and including Double-Ended-Rupture (rupture of a steam pipe)	LOFTRAN THINC	-	Function of the moderator density see Section 15.2.13 (Fig. 15.2-46)	See Fig. 15.4-75	0 (subcritical)
Major Secondary System Pipe Ruptures up to and including Double-Ended-Rupture (rupture of a feedline)	LOFTRAN	-	0.50	Upper	2915
Single Reactor Coolant Pump Locked Rotor	LOFTRAN FACTRAN THINC	+5	-	Upper	2915
Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	TWINKLE FACTRAN	+7.1 BOL -23 EOL	-	Consistent with lower limit on Fig 15.1-5	0 and 2900 [e]

(a) Only one is used in analysis, i.e., either moderator temperature or moderator density coefficient.

(b) Reference Figure 15.1-5.

(c) Appropriate calorimetric error considered where applicable.

(d) pcm means percent mille.

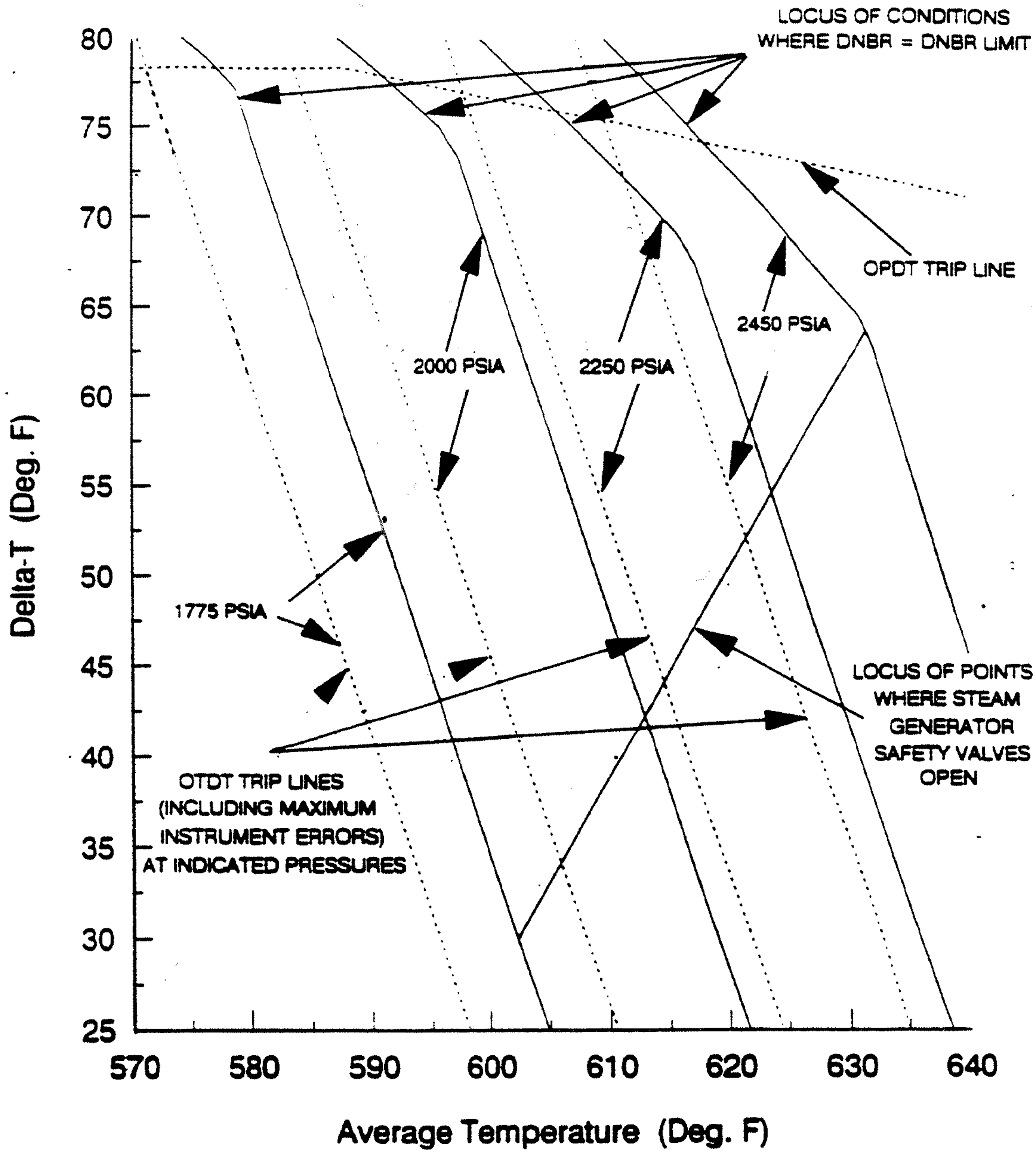
(e) Core power.

TABLE 15.1-5

CORE ACTIVITIES (IODINE AND NOBLE GASES)
BASED ON FULL POWER OPERATION FOR 480 DAYS

Isotope	Curies in Core ($x 10^7$)	Curies in Core ($x 10^6$)
I-131	8.2	8.2
I-132	12.0	12.0
I-133	16.8	17.0
I-134	18.0	18.0
I-135	15.4	15.0
Xe-131m	0.056	0.056
Xe-133	17.0	17.0
Xe-133m	2.4	2.4
Xe-135	3.7	3.7
Xe-135m	3.4	3.4
Xe-138	13.0	13.0
Kr-83m	0.95	0.95
Kr-85	0.083	0.25
Kr-85m	2.1	2.1
Kr-87	3.8	3.8
Kr-88	5.4	5.4
Kr-89	6.6	6.6

98-01



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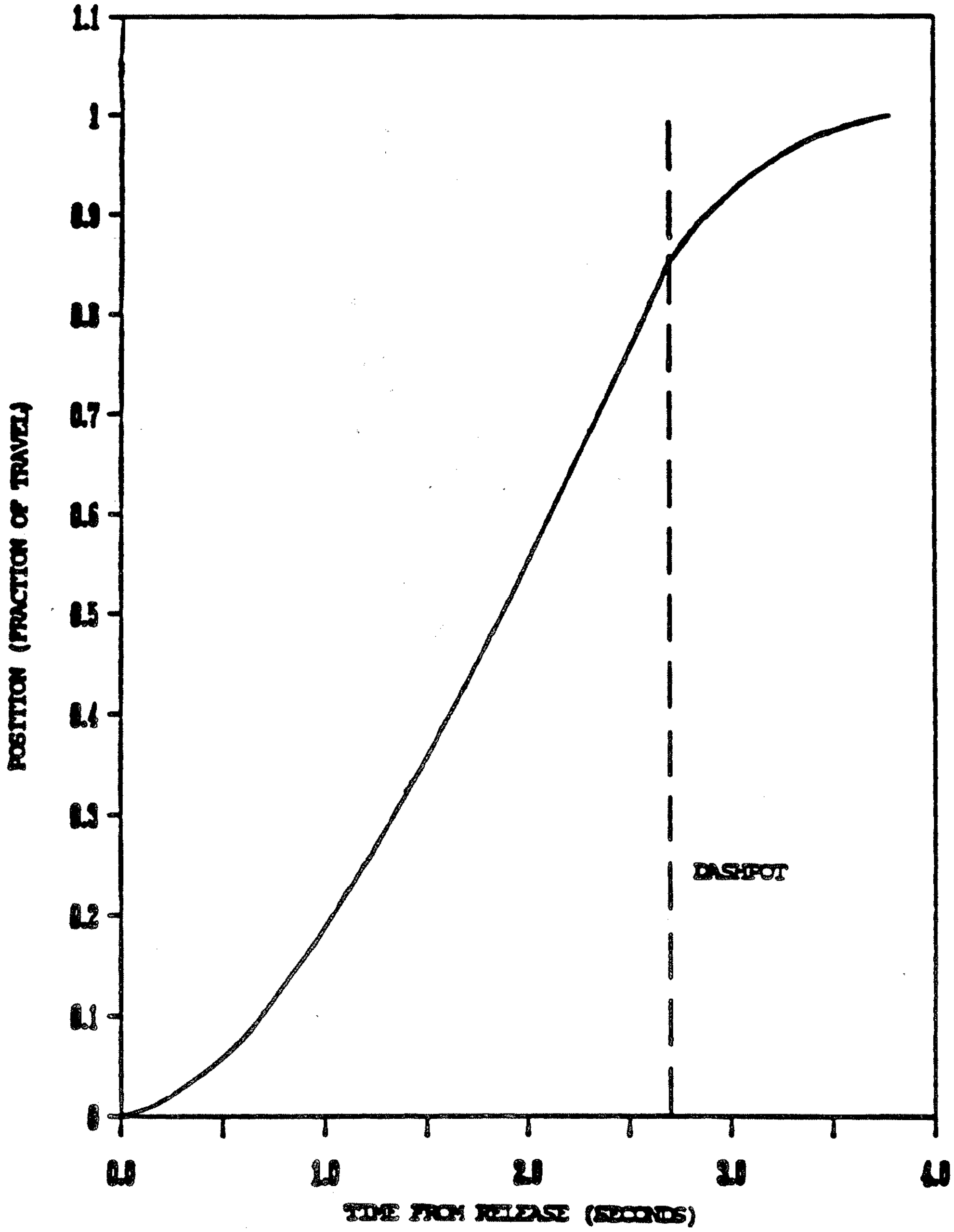
Overtemperature and Overpower
Delta-T Protection
Figure 15.1-1



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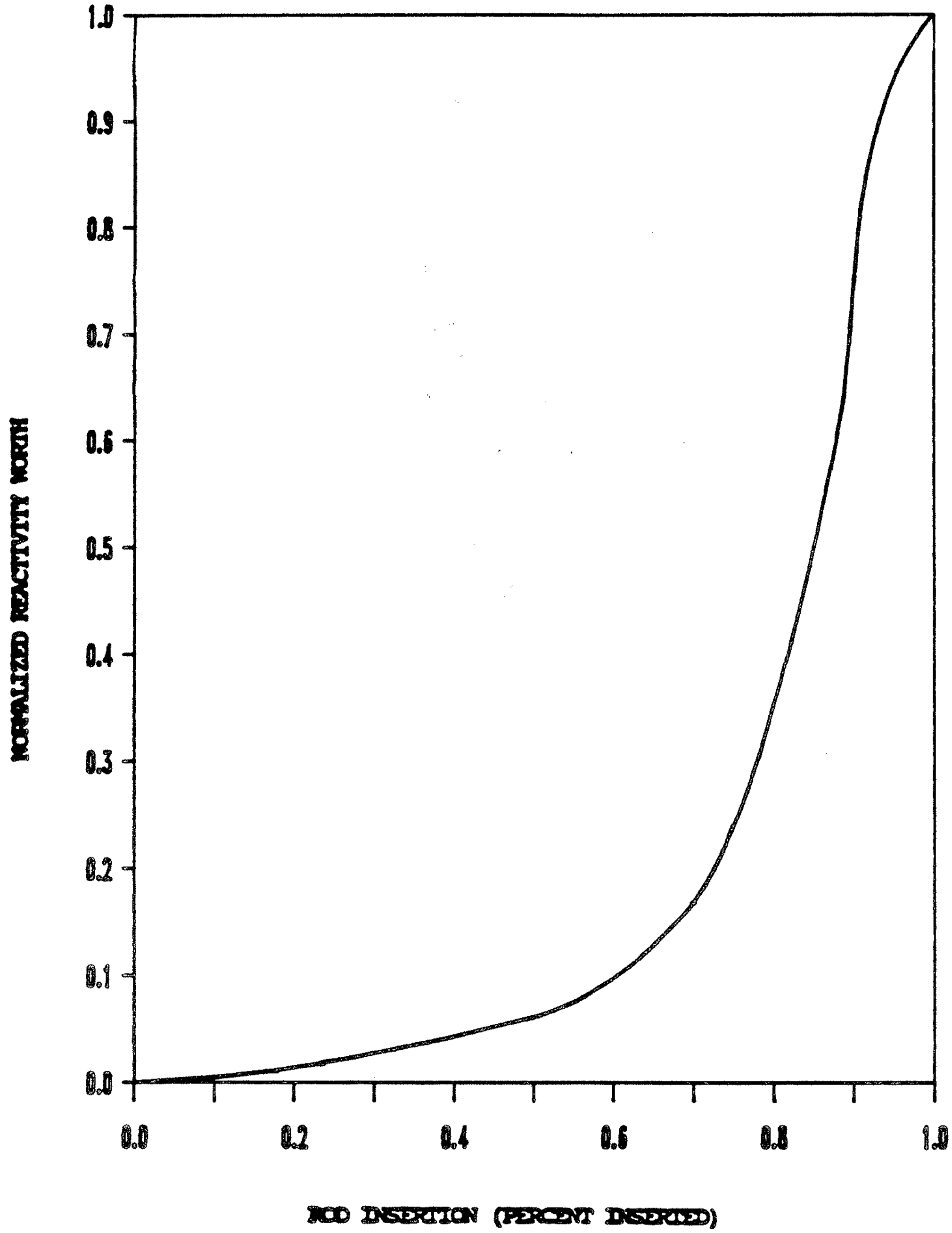
Rod Position vs.
Time On Reactor Trip

Figure 15.1-2



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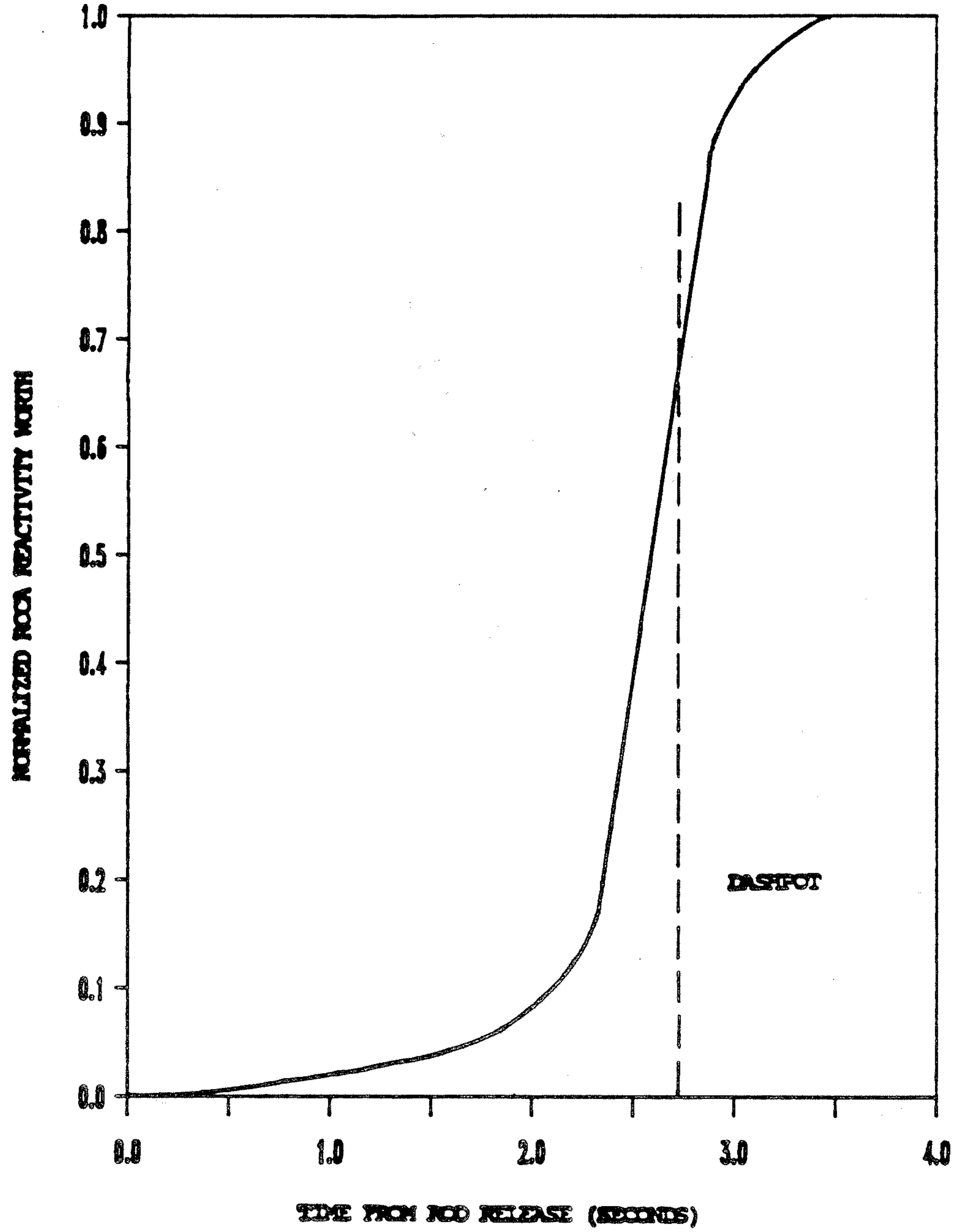
Normalized RCCA Reactivity
Worth vs. Percent Insertion

Figure 15.1-3



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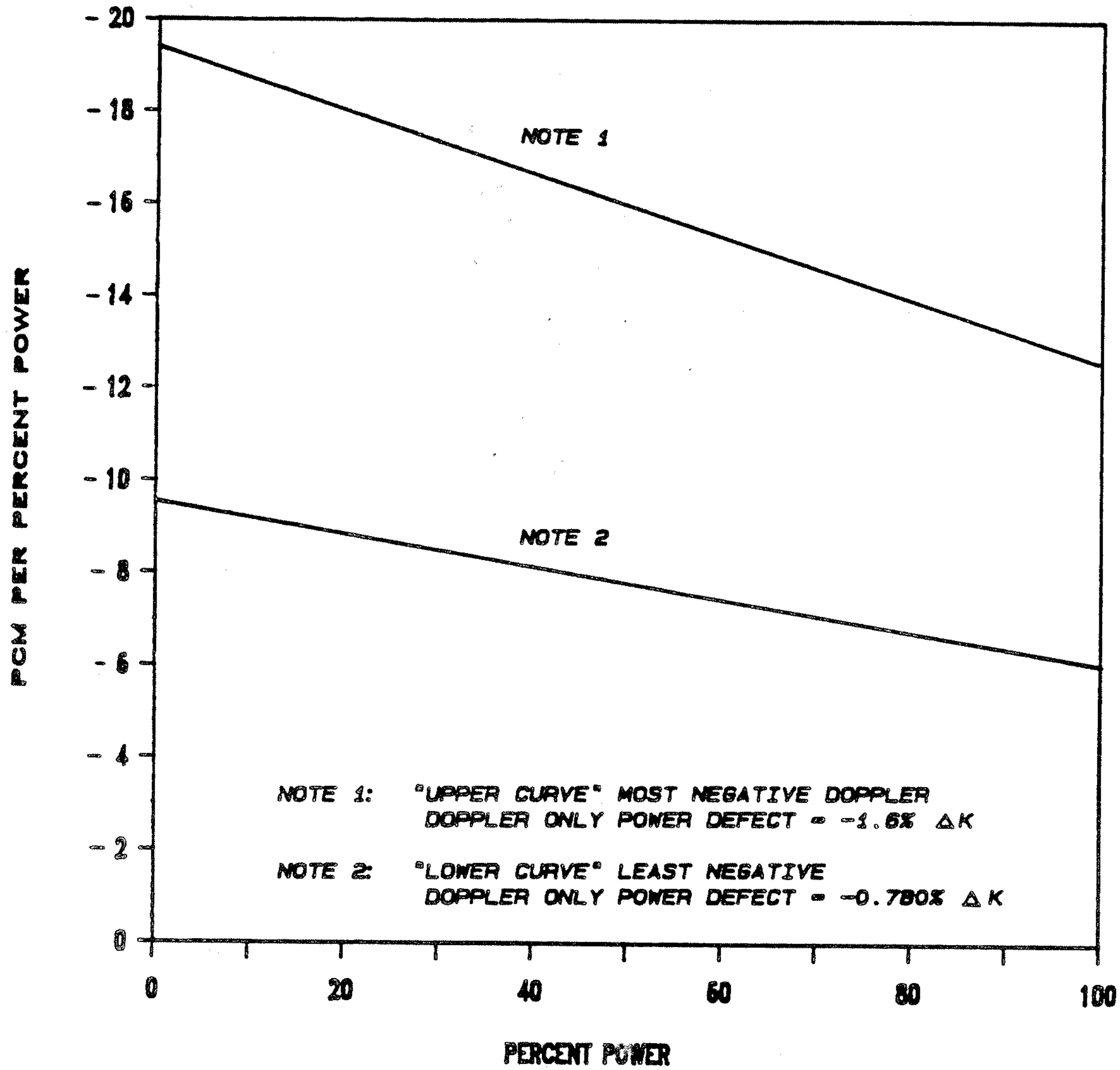
Normalized RCCA Bank
Worth vs. Time After Trip

Figure 15.1-4



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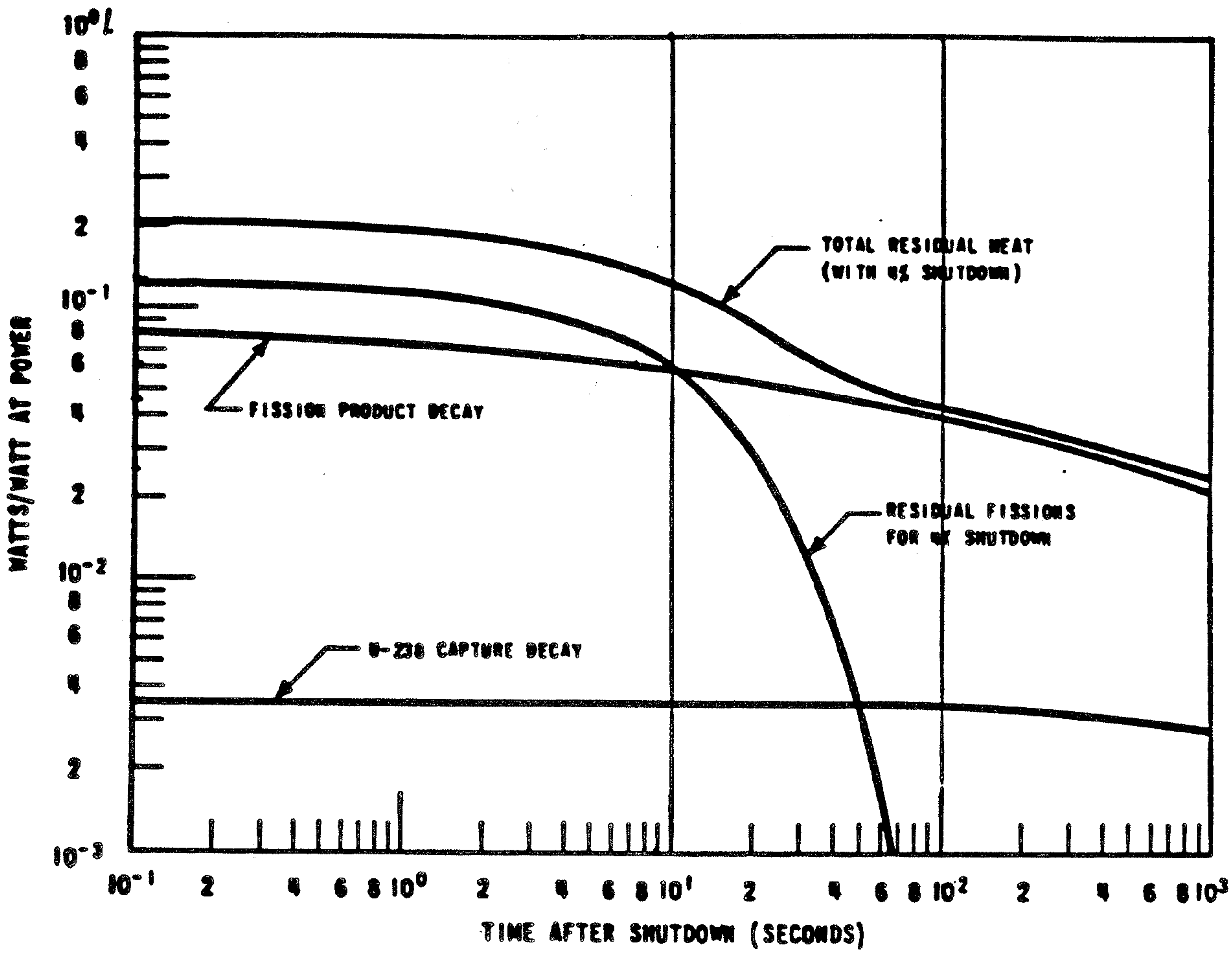
Doppler Power Coefficient Used
In Accident Analysis

Figure 15.1-5



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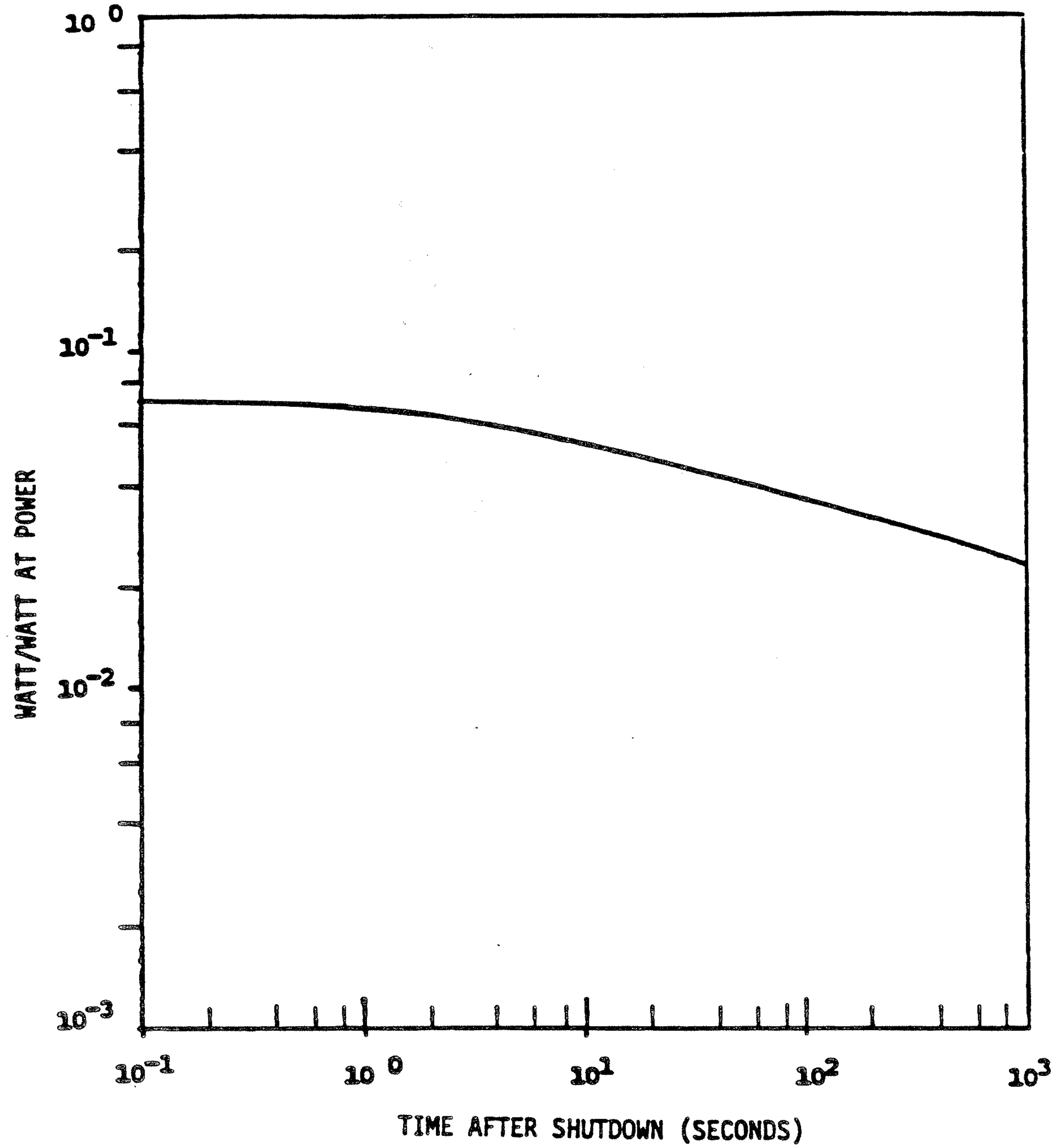
Residual Decay Heat

Figure 15.1-6



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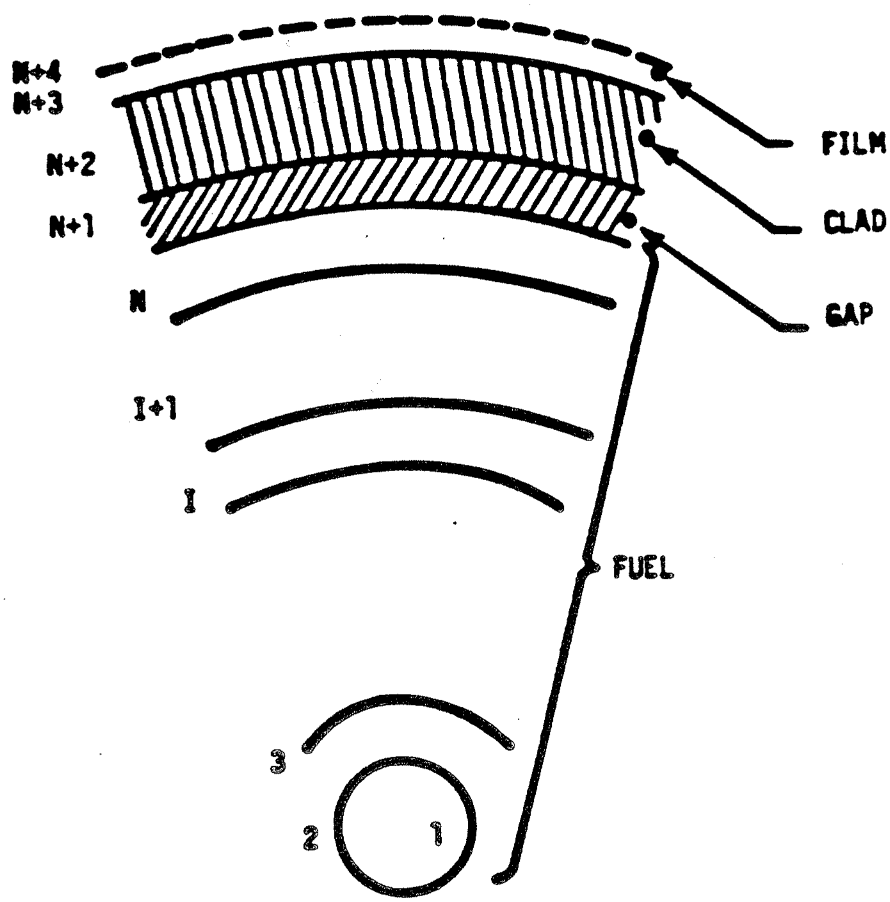
1979 ANS Decay Heat

Figure 15.1-7



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Fuel Rod Cross Section

Figure 15.1-8

15.2 CONDITION II - FAULTS OF MODERATE FREQUENCY

These faults, at worst, result in a reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not 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. Startup of an inactive reactor coolant loop.
7. Loss of external electrical load and/or turbine trip.
8. Loss of normal feedwater.
9. Loss of offsite power to the station auxiliaries (station blackout).
10. Excessive heat removal due to feedwater system malfunctions.
11. Excessive load increase incident.
12. Accidental depressurization of the Reactor Coolant System.
13. Accidental depressurization of the Main Steam System.
14. Inadvertent operation of the Safety Injection System (SIS) during power operation.

Each of these faults of moderate frequency are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

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.

In order to satisfy the positions set forth in WASH-1270, Appendix A, Part 11.13, anticipated transients without trip have been analyzed for Westinghouse Pressurized Water Reactors (PWR's). The results of these analyses are presented in References [11], [12], and [13] and demonstrate that a Westinghouse PWR is inherently self-limiting within well defined safety limits.

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

15.2.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL CONDITION

15.2.1.1 Identification of Causes and Accident Description

Branch Technical Position (BTP), ICSB 14, discusses spurious withdrawals of control rods in Pressurized Water Reactors. General Design Criteria (Appendix A to 10CFR50) 20 and 25 require protection systems which at all times limit the consequences of anticipated operational occurrences or malfunctions, and for reactivity control systems the fuel design limits may not be exceeded. The following discussion provides evidence that the Rod Control System in conjunction with the Reactor Protection System at VCSNS assures the GDC requirements are met. This BTP is also applicable to Section 15.2.2 and 15.3.6. 00-01

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

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

The RCCA drive mechanisms are wired into preselected bank configurations, which are not altered during reactor life. These circuits prevent the RCCA's from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than 2 banks can be withdrawn at the same time. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of 2 control banks having the maximum combined worth at maximum speed.

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

1. Source Range High Neutron Flux Reactor Trip

   The source range high neutron flux reactor trip is actuated when either of 2 independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

2. Intermediate Range High Neutron Flux Reactor Trip

   The intermediate range high neutron flux reactor trip is actuated when either of 2 independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after 2 of the 4 power range channels are reading above approximately 10% of full power and is automatically reinstated when 3 of the 4 channels indicate a power level below this value.

3. Power Range High Neutron Flux Reactor Trip (Low Setting)

   The power range high neutron flux trip (low setting) is actuated when 2 out of the 4 power range channels indicate a power level above approximately 25% of full power. This trip function may be manually bypassed when 2 of the 4 power range channels indicate a power level above approximately 10% of full power and is automatically reinstated only after 3 of the 4 channels indicate a power level below this value.

4. Power Range High Neutron Flux Reactor Trip (High Setting)

   The power range high neutron flux reactor trip (high setting) is actuated when 2 out of the 4 power range channels indicate a power level above a preset setpoint. This trip function is always active.

5. High Positive Neutron Flux Rate Trip

   The high positive neutron flux rate trip is actuated when the rate of change in power exceeds the setpoint in two-out-of-four power range channels. This function is always active.

In addition, control rod stops on high intermediate range flux level (1 of 2) and high power range flux level (1 out of 4) serve to discontinue rod withdrawal and prevent actuation of the intermediate range flux level trip and the power range flux level trip, respectively.

15.2.1.2 Analysis of Effects and Consequences

15.2.1.2.1 Method of Analysis

The analysis of the uncontrolled rod withdrawal from subcritical accident is performed in 3 stages: first a core average nuclear power transient calculation is performed, followed by an average core heat transfer calculation, and finally a DNBR calculation. The core average nuclear power transient calculation is performed using a spatial neutron kinetics code, TWINKLE [2], to determine the average power generation with time including the various total core feedback effects, i.e., Doppler and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN [3]. The average heat flux is next used in THINC [9] for the transient DNBR calculation.

The core axial power distribution is severely peaked to the bottom of the core for the limiting transient. The W-3 DNB correlation is used to evaluate DNBR in the span between the lower non-mixing vane grid and the first mixing vane grid. The WRB-2 correlation remains applicable for the rest of the fuel assembly.

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

1. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservative values (low absolute magnitude) as a function of temperature are used. See Section 15.1.6 and Table 15.1-4.

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 moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value, given in Table 15.1-4, is used in the analysis to yield the maximum peak heat flux.

3. The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect, thereby, increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1.0 since this results in the worst nuclear power transient.

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod cluster control assembly release, is taken into account. A 10% increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25% to 35%. Previous results, however, show that rise in the neutron flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. 98-01

5. The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the 2 control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.2.3.

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

7. Two (2) reactor coolant pumps are assumed to be operating.

8. A 1% core flow statepoint reduction was applied to incorporate the effects of asymmetric loop flow, assuming a maximum RCS loop-to-loop flow asymmetry of 5%. RN 01-125

15.2.1.2.2 Results

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

Figures 15.2-1 and 15.2-2 show the transient behavior for the indicated reactivity insertion rate with the accident terminated by reactor trip at 35% nominal power. This insertion rate is greater than that for the 2 highest worth control banks, both assumed to be in their highest incremental worth region.

Figure 15.2-1 shows the nuclear power transient. The nuclear power overshoots the full power nominal value but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increases are relatively small. The heat flux response, of interest for DNB considerations, is shown in Figure 15.2-1. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full power nominal value.

Figure 15.2-2 shows the response of the hot spot average fuel and clad temperature. The average fuel temperature increases to a value lower than the nominal full power value.

15.2.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the Reactor Coolant System are not adversely affected, since the combination of thermal power and the coolant temperature result in a departure from nucleate boiling ratio (DNBR) greater than the design limit value. Thus, no fuel or clad damage is predicted as a result of DNB.

15.2.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

15.2.2.1 Identification of Causes and Accident Description

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

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

1. Power range neutron flux instrumentation actuates a reactor trip if 2 out of 4 channels exceed a high flux setpoint, or a high positive flux rate setpoint. RN 09-021

2. Reactor trip is actuated if any 2 out of 3 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 2 out of 3 ΔT channels exceed an overpower ΔT setpoint, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

4. A high pressurizer pressure reactor trip actuated from any 2 out of 3 pressure channels that are 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 2 out of 3 level channels, is set at a fixed point.

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

1. High neutron flux (1 out of 4)

2. Overpower ΔT (2 out of 3)

3. Overtemperature ΔT (2 out of 3)

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of Reactor Coolant System conditions is described in Chapter 7. Figure 15.1-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions the trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

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

15.2.2.2 Analysis of Effects and Consequences

The analyses described below demonstrate conformance to fuel design limits. Reference 18 documents that a conservative analysis has also been performed that verifies that the RCS overpressure limit will not be exceeded. RN 09-021

15.2.2.2.1 Method of Analysis

This transient is analyzed by the LOFTRAN[4] Code. This code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.1-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Revised Thermal Design Procedure as described in Reference [5]. In order to obtain conservative results, the following assumptions are made:

1. Initial conditions of nominal core power and reactor coolant average temperatures and nominal reactor coolant pressure are assumed. Uncertainties in initial conditions are included in the limit DNBR as described in Reference [5];

2. Reactivity Coefficients - 2 cases are analyzed:

    a. Minimum reactivity feedback. A positive moderator coefficient of reactivity of + 7 pcm/°F is assumed. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

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

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

4. The RCCA trip insertion characteristics 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 which would be obtained for the simultaneous withdrawal of the 2 control banks having the maximum combined worth at maximum speed.

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

15.2.2.2.2 Results

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

Figures 15.2-3 and 15.2-4 show the transient response to a rapid RCCA withdrawal starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and a large margin to DNB is maintained.

The transient response for a slow control rod assembly withdrawal from full power is shown on Figures 15.2-5, and 15.2-6. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is never less than the safety analysis limit values.

Figures 15.2-7, 15.2-8 and 15.2-9 illustrate the minimum DNBR calculated for minimum and maximum reactivity feedbacks.

Figure 15.2-7 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that 2 reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT trip channels. The minimum DNBR is never less than the safety analysis limit values.

Figures 15.2-8 and 15.2-9 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10% power respectively. The results are similar to the 100% power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the safety analysis limit values.

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

Referring to Figure 15.2-9, for example, it is noted that:

1. For high reactivity insertion rates of about ~15 pcm/sec, reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNBR's during the transient. As the reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNBR during the transient thus decreases with decreasing insertion rate.

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

3. For reactivity insertion rate below ~ 15 pcm/sec the overtemperature ΔT trip terminates the transient.

   For reactivity insertion rates between ~ 20 pcm/sec and ~ 5 pcm/sec the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant. 02-01

Referring to Figure 15.2-8, it is noted that:

For reactivity insertion rates less than ~ 50 pcm/sec, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load on the Reactor Coolant System, sharply decreases the rate of increase of Reactor Coolant System average temperature. This decrease in rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation causing the overtemperature ΔT trip setpoint to be reached later with resulting lower minimum DNBR.

For transients initiated from higher power levels (for example, see Figure 15.2-7) the effect described above, which results in the sharp peak in minimum DNBR at approximately 50 pcm/sec, does not occur since the steam generator safety valves are not actuated prior to trip.

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

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

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

The calculated sequence of events for this accident is shown on Table 15.2-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.2.2.3 Conclusions

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

15.2.3 ROD CLUSTER CONTROL ASSEMBLY MISOPERATION

This section discusses RCCA misoperation that can result from system malfunction or operator error.

15.2.3.1 Identification of Causes and Accident Description

Rod cluster control assembly (RCCA) misalignment accidents include:

1. One(1) or more dropped RCCAs within the same group;
2. A dropped RCCA bank;
3. Statically misaligned RCCA.

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator’s convenience. Fully inserted assemblies are further indicated by a rod bottom signal, which actuates a local alarm and a control room annunciator. 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 2 groups. The rods comprising a group operate in parallel through multiplexing thyristors. The 2 groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the 4 RCCAs of a rod group are driven in parallel, any single failure that would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

A dropped RCCA or RCCA bank is detected by:

1. A sudden drop in the core power level as seen by the nuclear instrumentation system;

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

3. Rod at bottom signal;

4. Rod deviation alarm;

5. Rod position indication.

Misaligned assemblies are detected by:

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

2. Rod deviation alarm;

3. Rod position indicators.

The deviation alarm alerts the operator whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications[6].

If 1 or more rod position indicator channels should be out of service, detailed operating instructions are followed to ensure the alignment of the nonindicated RCCAs. The operator is also required to take action as required by the Technical Specifications.

15.2.3.2 Analysis of Effects and Consequences

15.2.3.2.1 Method of Analysis

1. One (1) or more dropped RCCAs from the same group.

   For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN [4] code. The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

   Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code[9]. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference [10]. 02-01 99-01

2. Dropped RCCA Bank

   A dropped RCCA bank results in a symmetric power change in the core. As discussed in Reference 10, assumptions made for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

3. 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.

15.2.3.2.2 Results

1. One (1) or more Dropped RCCAs

   Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will re-establish power.

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

For a dropped RCCA event in the automatic rod control mode, the Rod Control System detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 15.2-10 and 15.2-11 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 results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described in Part 1; however, the return to power will be less due to the greater worth of the entire bank. The power distribution during a dropped bank transient is symmetric. Following plant stabilization, normal procedures are followed.

3. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which 1 RCCA is fully inserted, or where bank D is fully inserted with 1 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. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

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

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been done as required for the fully withdrawn analysis. Inspection of the power shapes shows that the DNB

and peak kw/ft situation is less severe than the bank D case discussed above, assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For RCCA misalignments with 1 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.

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

Following the identification of a 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

It is shown for all cases of dropped RCCAs or dropped banks 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 fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value. Therefore, the DNB design basis is met.

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 (RCS) via the reactor makeup portion of the Chemical and Volume Control System (CVCS). Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The 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. In order for makeup water to be added to the RCS at pressure, at least 1 charging pump must be running in addition to a primary makeup water pump.

The rate of addition of unborated makeup water to the RCS is limited by a flow limiting orifice between the reactor makeup water pumps and the boric acid blender. As demonstrated by tests at the plant, flow is within the bounds of unborated water used in analyses in this section.

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 2 separate operations are required:

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

2. The start/stop switch is placed in the start position.

Omitting either step would prevent dilution.

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

1. Indication of the boric acid and blended flow rates,

2. CVCS and RMWS pump status lights,

3. Deviation alarms if the boric acid or blended flow rates deviate by more than 10% from the preset values.

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Indication of a dilution event is available to the operator by:

1. Source Range Neutron Flux - when the reactor is subcritical:

    a. High flux at shutdown alarm. A separate main control board (MCB) and computer generated alarm is provided for each channel.

    00-01

    b. Indicated source range neutron flux count rates.

    c. Audible source range neutron flux count rate.

2. With the reactor critical:

    a. Axial flux difference alarm (reactor power ≥ 50% RTP),

b. Control rod insertion limit low and low-low alarms,

c. Overtemperature ΔT alarm (at power),

d. Overtemperature ΔT turbine runback (at power),

e. Overtemperature ΔT reactor trip, and

f. Power range neutron flux - High, both high and low setpoint reactor trips.

15.2.4.2 Analysis of Effects and Consequences

To cover all phases of the plant operation, boron dilution during refueling, cold shutdown, hot standby, startup, and power operation are considered in this analysis. Table 15.2-1 contains the time sequence of events for this accident.

1. Dilution During Refueling

An uncontrolled boron dilution accident based on a failure in the primary water makeup system cannot occur during refueling. This accident is prevented by administrative controls which isolate the RCS from the potential source unborated water.

Valves 8454, 8441, 8430, and 8439 will be locked closed during refueling operations. These valves will block the flow paths which could allow unborated makeup water to reach the RCS. Any makeup which is required during refueling will be added to the Reactor Coolant System by unlocking these valves as appropriate and initiating the required blended makeup water flow. After the required volume of blended makeup flow has been added, these valves will again be locked closed. An alternate source of borated water that could be used is from the refueling water storage tank to the charging pump suction.

Demineralized water may be used to decontaminate the refueling cavity walls while draining the refueling cavity. When the cavity water level is above the RV flange, water is pumped back to the refueling water storage tank through the Residual Heat Removal System. Administrative controls preclude an uncontrolled boron dilution accident by limiting the amount of demineralized water used such that the boron concentrations within the refueling cavity, Reactor Coolant System, and refueling water storage tank remain above minimum Technical Specification limits.

The most limiting alternate source of unborated water is from the Boron Thermal Regeneration System (BTRS). For this case, highly borated RCS water is depleted of boron as it passes through the BTRS and is returned via the volume control tank. The following conditions are assumed for an uncontrolled boron dilution during refueling.

Technical Specifications require the reactor to be borated to at least 2,000 ppm or shutdown by at least 5.0% Δk at refueling.

If an inadvertent dilution from the BTRS occurs during refueling, with the reactor vessel head off, and the refueling cavity filled with borated water (i.e., in a condition to move fuel), the maximum dilution capability of the BTRS is insufficient to cause a return to criticality.

The maximum dilution capability of the BTRS at these conditions is conservatively estimated to be 250 ppm. However, the minimum change in boron concentration necessary to bring the reactor critical at these conditions is conservatively estimated to be 590 ppm. An initial boron concentration of 2000 ppm is assumed. Therefore, a dilution to criticality from the BTRS at these refueling conditions cannot occur. RN 00-073

The most limiting conditions for an inadvertent boron dilution from the BTRS during refueling occur when the reactor coolant level is at the vessel/head junction. The dilution capability of the BTRS at these conditions is sufficient to cause a return to criticality. The minimum volume in the Reactor Coolant System corresponding to this condition is conservatively estimated to be 3075 ft^3. The critical boron concentration is conservatively estimated to be 1410 ppm. RN 00-073

2. <u>Dilution During Cold Shutdown</u>

Technical Specifications specify the required shutdown margin as a function of RCS boron concentration during cold shutdown. The specified shutdown margin ensures sufficient time for the operator to terminate the dilution. If the reactor is in cold shutdown and on the Residual Heat Removal System with RCS piping filled and vented, the following conditions are assumed for an uncontrolled boron dilution. Dilution flow is assumed to be a maximum of 150 gpm, which is the capability of 1 primary water makeup pump to deliver unborated water to the RCS. Mixing of the reactor coolant is accomplished by the operation of 1 residual heat removal pump.

A volume of 5240.5 ft^3 in the Reactor Coolant System is used. This corresponds to the active volume of the Reactor Coolant System minus the pressurizer volume, while on the Residual Heat Removal System.

If the reactor is in cold shutdown and the RCS water level is drained down from a filled and vented condition while on RHR, an inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water. Valves 8454, 8441, 8430, and 8439 will be locked closed during operation in these conditions. These valves block all flow paths that could allow unborated makeup water to reach the RCS. Any makeup, which is required, will be added to the Reactor Coolant System by unlocking these valves, as

appropriate, and initiating the required blended makeup water flow. After the required volume of blended makeup flow has been added, these valves will again be locked closed. An alternate source of borated water which may be used is from the refueling water storage tank to the charging pump suction.

3. <u>Dilution During Hot Shutdown</u>

   Technical Specifications specify the required shutdown margin as a function of RCS boron concentration. The analysis assumes that the RCS is filled and vented on the Residual Heat Removal System. This is more conservative than if 1 or more reactor coolant pumps are assumed operating. The following conditions are assumed for a continuous boron dilution during hot standby. Dilution flow is assumed to be a maximum of 150 gpm, which is the capacity of 1 primary water makeup pump to deliver unborated water to the RCS. A minimum water volume of 5240.5 ft^3 in the Reactor Coolant System is used. This corresponds to the active volume of the Reactor Coolant System minus the pressurizer volume, while on the Residual Heat Removal System.

4. <u>Dilution During Hot Standby</u>

   Technical Specifications specify the required shutdown margin as a function of RCS boron concentration. The following conditions are assumed for a continuous boron dilution during hot standby. Dilution flow is assumed to be a maximum of 150 gpm, which is the capability of one primary water makeup pump to deliver unborated water to the RCS. A minimum RCS water volume of 8126 ft^3 is assumed. This is a conservative estimate of the active RCS volume with one reactor coolant pump operating.

5. <u>Dilution During Startup</u>

   Prior to startup, the RCS is filled with borated water at a boron concentration of 2200 ppm. This is a conservative estimate with the reactor at a 1.77% Δk shutdown margin at 557°F. Dilution flow is assumed to be a maximum of 150 gpm, which is the capability of one primary water makeup pump to deliver unborated water to the RCS. A minimum volume of 8673 ft^3 in the Reactor Coolant System is used. This is a conservative estimate of the active volume of the RCS excluding the pressurizer.

6. <u>Dilution During Full Power Operation</u>

   During power operation, the plant may be operated 2 ways, under manual operator control or under automatic T_{avg}/rod control. The Technical Specifications require 3 reactor coolant pumps operating and a shutdown margin of at least 1.77% Δk. The RCS is conservatively assumed to be filled with borated water at a boron concentration of 2200 ppm.

While the plant is in manual control, the dilution flow is assumed to be a maximum of 150 gpm, which is the capacity of one reactor makeup water pump to deliver unborated water to the RCS. When in automatic control, the dilution flow is limited by the maximum letdown flow (approximately 125 gpm). A minimum RCS water volume of 8673 ft^3 is used. This is a conservative estimate of the active volume of the RCS excluding the pressurizer.

15.2.4.3 Results 00-01

Dilution During Refueling

During refueling, an inadvertent dilution from the Reactor Makeup Water System is prevented by administrative controls which isolate the RCS from the potential source of unborated makeup water. The most limiting conditions for an inadvertent dilution from the BTRS occur when the reactor vessel head is unbolted and the vessel water level is at the vessel/head junction. The high flux at shutdown alarm, set at twice the background flux level measured by the source range nuclear instrumentation, is available at these conditions to alert the operator that a dilution event is in progress. For this case, the operator has more than 30 minutes from the high flux at shutdown 00-01 alarm to recognize and terminate the dilution before shutdown margin is lost and the reactor becomes critical.

Dilution During Cold Shutdown

While in cold shutdown the high flux at shutdown alarm, set at twice the background flux level measured by the source range nuclear instrumentation, is available to alert the operator that a dilution event is in progress. During the cold shutdown mode while operating on the Residual Heat Removal System (RHRS) with the RCS piping filled and vented, the shutdown margin requirement ensures that the operator has at least 13.6 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost. During the cold shutdown mode, while operating on the RHRS with the RCS drained down from a filled and vented condition, an inadvertent dilution is precluded by administrative controls which isolate the RCS from the potential source of unborated water.

Dilution During Hot Shutdown

While in hot shutdown, the high flux at shutdown alarm, set at twice the background flux level measured by the source range nuclear instrumentation, is available to alert the operator that a dilution event is in progress. During hot shutdown, the shutdown margin requirement ensures that the operator has at least 13.6 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost.

Dilution During Hot Standby

While in hot standby, the high flux at shutdown alarm, set at twice the background flux level measured by the source range nuclear instrumentation, is available to alert the operator that a dilution event is in progress. During hot standby, the shutdown margin requirement ensures that the operator has at least 13.4 minutes from the high flux at shutdown alarm to recognize and terminate the uncontrolled reactivity insertion before shutdown margin is lost.

Dilution During Startup

In the event of an unplanned approach to criticality or dilution during power escalation while in the startup mode, the operator is alerted to an uncontrolled reactivity insertion by a reactor trip at the Power Range Neutron Flux-High, low setpoint (nominally 25% RTP). After reactor trip there is more than 15 minutes for operator action prior to return to criticality.

Dilution at Power

During the at power mode with manual control, the operator is alerted to an uncontrolled reactivity insertion by an overtemperature ΔT trip. More than 15 minutes are available from the trip for the operator to recognize and terminate the uncontrolled dilution before shutdown margin is lost. The sensitivity and alarm thresholds are already assumed to be degraded to the maximum extent allowable for the overtemperature ΔT trip function (see Section 15.2.2).

During the at power mode with automatic control, the operator is alerted to an uncontrolled reactivity insertion by the rod insertion limit alarms. Two (2) insertion limit alarms are available: The first occurs when the rods are 10 steps above the insertion limit (Lo Insertion Limit Alarm) and the second occurs at the insertion limit (Lo-Lo Insertion Limit Alarm). The analysis assumed that the operator is alerted to the need for action by the Lo-Lo alarm although action would be taken when the first alarm occurs. Thus the analysis already assumes a 10 step allowance for rod position indicator inaccuracies. Even with this conservatism, more than 15 minutes are available from the time of alarm until all shutdown margin is lost. In addition to the above, other indications are available. The main indication would be a violation of the axial offset control band which could result in a reactor trip (reduction in overtemperature ΔT setpoint).

15.2.4.4 Event Detection Immediately Following Reactor Shutdown

Following a reactor trip or shutdown, the high flux at shutdown alarms for the MCB are implemented via a manual calibration of the alarm bistable based on the existing source range counts. Following the initial calibration, the MCB alarms are recalibrated at approximately 12, 24, and 48 hours after shutdown to compensate for the anticipated decrease in source range counts to stable levels. Consistent with Generic Letter 85-05, 00-01

computer generated alarms, which account for decreasing source range counts, and administrative controls are also utilized to provide increased assurance that an inadvertent dilution event, should it occur, will be terminated prior to a loss of shutdown margin. 00-01

15.2.4.5 Conclusions 02-01

The analysis presented above shows that, for an inadvertent boron dilution, the operator has sufficient time to recognize and terminate the dilution before shutdown margin is lost and the reactor becomes critical. 00-01

15.2.5 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

15.2.5.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps. 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 2 out of 3 low flow signals in any reactor coolant loop. Above approximately 38% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and power level corresponding to Permissive 8, low flow in any 2 loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive 7.

A reactor trip signal from the pump breaker position is also provided. When operating above Permissive 7, a breaker open signal from any 2 pumps will actuate a reactor trip. This serves as a backup to the low flow trip. Reactor trip on reactor coolant pump breakers open is blocked below Permissive 7.

Normal power for each pump is supplied through individual buses connected to the isolated phase bus duct between the generator circuit breaker and the main transformer. Faults in the substation may cause a trip of the main transformer high side circuit breaker leaving the generator to supply power to the reactor coolant pumps. When a generator circuit breaker trip occurs because of electrical faults, the pumps are automatically transferred to an alternate power supply and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults the generator circuit breaker is tripped and the reactor coolant pumps remain connected to the network through the transformer high side breaker. Continuity of power to the pump buses is achieved without motoring the generator since means are provided to isolate the generator without isolating the pump buses from the external power lines (e.g., a generator output breaker is provided as well as station output breaker).

15.2.5.2 Analysis of Effects and Consequences

15.2.5.2.1 Method of Analysis

The following case has been analyzed:

All loops operating, one loop coasting down.

This transient is analyzed by 3 digital computer codes. First the LOFTRAN code is used to calculate the loop and core flow during the transient. The LOFTRAN code is also used to calculate the time of reactor trip, based on the calculated flows and the nuclear power transient following reactor trip. The FACTRAN code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code [9] is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN.

1. Initial Conditions

Initial operating conditions assumed are the most adverse with respect to the margin to DNB (i.e., nominal steady-state power level, nominal steady-state pressure, and nominal steady-state coolant average temperature). See Section 15.1.2 for an explanation of initial conditions. The accident is analyzed using the Revised Thermal Design Procedure as described in Reference 5.

2. Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 15.1-4). The total integrated Doppler reactivity from 0 to 100% power is assumed to be -0.016 ΔK.

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

3. Flow Coastdown

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

15.2.5.2.2 Results

The calculated sequence of events is shown in Table 15.2-1. Figures 15.2-12 and 15.2-13 show the vessel flow coastdown, the faulted loop flow coastdown, the nuclear power, and heat flux transient. The minimum DNBR is not less than the safety analysis limit value. (See Figure 15.2-14).

15.2.5.3 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit value at any time during the transient. Thus, no core safety limit is violated.

15.2.6 STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

In accordance with Technical Specification 3/4.4.1, V. C. Summer operation during startup and power operation with less than 3 loops operating is not permitted. Therefore, this section is not applicable. 99-01

15.2.7 LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE TRIP

15.2.7.1 Identification of Causes and Accident Description

A major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of offsite power to the station auxiliaries is analyzed in Section 15.2.9.

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 and turbine stop valves.

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 main condenser was not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the main condenser was not available. For this situation, steam generator level would be maintained by the Emergency Feedwater System.

For a loss of external electrical load without subsequent turbine trip, a direct reactor trip signal may be generated. However, with full load rejection capability the plant may be expected to continue without a reactor trip. A continued steam load of approximately 5% would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

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

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

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the Reactor Coolant System (RCS) and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power operated relief valves, automatic control of the rod cluster control assembly nor direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the ESF rating (104.5% steam flow at rated power) from the steam generator without exceeding 110% of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load (along with operation of the steam generator safety valves). The pressurizer safety valves are then able to maintain the RCS pressure within 110% of the RCS design pressure without direct or immediate reactor trip action. 02-01

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

15.2.7.2 Analysis of Effects and Consequences

15.2.7.2.1 Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. This is done to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. The turbine is assumed to trip without actuating all the turbine stop valve limit switches. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for emergency feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The total loss of load transients are analyzed with the LOFTRAN computer program (see Section 15.1). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

Major assumptions are summarized below:

1. Initial Operating Conditions

   This accident is analyzed using the Revised Thermal Design Procedure (RTDP). For the DBN (pressure control) case, initial core power, reactor coolant temperature, and pressurizer pressure are assumed to be at their nominal full-power values consistent with steady-state full power operation. Uncertainties in initial conditions and instrument errors are included in the limit departure from nucleate boiling ratio (DNBR) as described in Reference [5]. For the pressure transient case, the initial core power and pressurizer pressure are assumed to be at their nominal full-power values. The initial reactor coolant temperature is assumed to be the nominal full-power value minus uncertainty. RN 03-042

2. Moderator and Doppler Coefficients of Reactivity

   The turbine trip is analyzed with minimum reactivity feedback which assumes a minimum moderator temperature coefficient and the least negative Doppler coefficient.

3. Reactor Control

From the standpoint of the maximum pressures attained and DNBR, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient. RN 03-042

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 where steam release through safety valves limits secondary steam pressure at the setpoint value.

5. Pressurizer Spray and Power Operated Relief Valves

Two (2) cases for the beginning of life are analyzed:

a. For evaluating the minimum DNBR, full credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available. RN 03-042

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

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 emergency feedwater flow since a stabilized plant condition will be reached before emergency feedwater initiation is normally assumed to occur; however, the emergency feedwater pumps would be expected to start on a trip of the main feedwater pumps. The emergency feedwater flow would remove core decay heat following plant stabilization.

7. 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 and overtemperature ΔT. RN 03-042

15.2.7.2.2 Results

The transient responses for a total loss of load from full power operation are shown for 2 cases for the beginning of core life and 2 cases for the end of core life, in Figures 15.2-19 through 15.2-21 and 15.2-25 through 15.2-27. The calculated sequence of events for the accident is shown in Table 15.2-1. RN 03-042

Figures 15.2-19, 15.2-20 and 15.2-21 show the transient responses for the total loss of steam load at beginning of life assuming full credit for the pressurizer spray and pressurizer power operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure trip channel. The minimum DNBR is well above the limit value. The pressurizer safety valves are actuated for this case and maintain system pressure below 110% of the design value. The steam generator safety valves open and limit the secondary steam pressure increase.

RN 03-042

The total loss of load accident was also studied assuming the plant to be initially operating at 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. Figures 15.2-25, 15.2-26 and 15.2-27 show the beginning of life transients. The neutron flux remains constant at full power until the Reactor is tripped. The DNBR increases throughout the transient. In this case, the pressurizer safety valves are actuated and maintain system pressure below 110% of the design value.

RN 03-042

Reference [8] presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This analysis shows the overpressure protection that is provided by the pressurizer and steam generator safety valves. RN 03-042

15.2.7.3 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 2 systems are adequate to limit the maximum pressures to within the design limits. RN 03-042

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 values. Thus, no core safety limit will be violated.

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 power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss of heat sink. If an alternative supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs. Significant loss of water from the 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 following provides the necessary protection against a loss of normal feedwater:

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

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

3. Two (2) motor driven emergency feedwater (EFW) pumps which are started on:

   a. Low-low level in any steam generator.

   b. Trip of all main feedwater pumps.

   c. Any safety injection signal.

   d. Loss of offsite power (automatic transfer to diesel generator).

   e. Manual actuation.

4. One (1) turbine driven emergency feedwater pump is started on:

   a. Low-Low level in any 2 steam generators.

   b. Loss of offsite power.

   c. Manual actuation.

The motor driven emergency feedwater pumps are connected to vital buses and are supplied by Class 1E electric power from the diesels if a loss of offsite power occurs. The turbine driven pump utilizes steam from the safety class portions of the Main Steam System and exhausts it to the atmosphere. Both type pumps are designed to start within 1 minute even if a loss of offsite power occurs simultaneously with loss of normal feedwater. The emergency feedwater pumps take suction from the condensate storage tank for delivery to the steam generators.

The analysis shows that following a loss of normal feedwater, the Emergency Feedwater System is capable of removing the stored and residual heat thus preventing either overpressurization of the RCS or loss of water from the reactor core.

15.2.8.2 Analysis of Effects and Consequences

15.2.8.2.1 Method of Analysis

A detailed analysis using the LOFTRAN[4] Code is performed in order to obtain the plant transient following a loss of normal feedwater. The code describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators and feedwater system, and computes pertinent variables including the steam generator level, pressurizer water level, pressurizer pressure, and reactor coolant average temperature.

Major assumptions are:

1. Reactor trip occurs on steam generator low-low level at 16.7% of narrow range span.

2. The plant is initially operating at 102% of the NSSS design rating.

3. Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 decay heat ANSI 5.1 + 2 SIGMA was used for calculation of residual decay heat levels.

4. The Emergency Feedwater System is actuated by the low-low steam generator water level signal.

5. The worst single failure in the Emergency Feedwater System occurs (turbine-driven pump) and 1 motor-driven pump is assumed to be unavailable. The emergency feedwater system is assumed to supply a total of 400 gpm equally split to all 3 steam generators from the motor-driven pump.

6. The pressurizer sprays and PORVs are assumed operable. Per the plant specific analysis performed in support of NSAL-07-10[17], PORV operability does not challenge pressurizer filling. RN 08-025 RN 14-026

7. 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.

8. The initial reactor coolant average temperature is 4.0°F higher than the nominal value to allow for uncertainty on nominal temperature. The initial pressurizer pressure uncertainty is 50 psi.

15.2.8.2.2 Results

Figures 15.2-31 and 15.2-32 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One (1) minute following the initiation of the low-low level trip, the motor driven EFW pump is automatically started, reducing the rate of water level decrease.

The capacity of the motor driven EFW pump is such that the water level in the steam generator being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves.

From Figure 15.2-32 it can be seen that at no time is there water relief from the pressurizer. If the emergency feed delivered is greater than that of 1 motor driven pump, the initial reactor power is less than 102% of the NSSS design rating, or if the steam generator water level in 1 or more steam generators is above the low-low level trip point at the time of trip, then the result for this transient will be less limiting.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2-31 and 15.2-32, the plant approaches a stabilized condition following reactor trip and emergency feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.8.3 <u>Conclusions</u>

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

15.2.9 LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES

15.2.9.1 Identification of Causes and Accident Description

During 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 offsite power with turbine and reactor trip are described in the sequence listed below:

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

2. 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 power relief valves are not available, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

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

4. The emergency diesel generators start on loss of voltage on the plant ESF buses and begin to supply plant vital loads.

The Emergency Feedwater System is started automatically as discussed in the loss of normal feedwater analysis. The steam driven emergency feedwater pump utilizes steam from the safety class portions of the Main Steam System and exhausts to the atmosphere. The 2 motor driven emergency feedwater pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

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

15.2.9.2 Analysis of Effects and Consequences

A detailed analysis using the LOFTRAN code [4] is performed in order to determine the plant transient following a station blackout. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature. Major assumptions differing from those in a loss of normal feedwater are:

1. No credit is taken for immediate response of control rod drive mechanisms caused by a loss of offsite power.

2. A heat transfer coefficient in the steam generator associated with RCS natural circulation is assumed following the reactor coolant pump coastdown.

The time sequence of events for the accident is given in Table 15.2-1. The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Section 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 heat must be removed to prevent damage to either the RCS or the core. The LOFTRAN code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

15.2.9.3 Conclusions

Results of the analysis show that, for the loss of offsite power to the station auxiliaries event, all safety criteria are met. Since the DNBR remains above the safety analysis limit, the core is not adversely affected. EFW capacity is sufficient to prevent water relief through the pressurizer relief and safety valves; this assures that the RCS is not overpressurized.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long-term heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.

If it is assumed that there is leakage from the RCS to the secondary system in the steam generators and that radioactivity will be released to the atmosphere through the relief or safety valves. Parameters used in determining the amount of radioactivity released are given in Table 15.2-5.

15.2.9.4 Environmental Consequences of a Postulated Loss of Offsite Power to the Station Auxiliaries

The postulated incidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the RCS to the secondary system within the steam generators. A realistic and conservative analysis of the potential offsite doses resulting from this accident is presented.

These analyses incorporate assumptions for operating with defective fuel and steam generator leakage for a sufficient time prior to the postulated accident to establish equilibrium specific activity levels in the secondary system. Parameters used in both the realistic and conservative analyses are listed in Table 15.2-5.

The assumptions used to determine the equilibrium concentrations of isotopes in the secondary system are as follows:

1. Primary to secondary leakage in steam generators occurs when the reactor is started and the leakage remains constant during the plant operation. For the conservative analysis, the Technical Specification limit on leakage rate of 1 gpm is used whereas 100 lbs/day is used in the realistic analysis.

2. Primary to secondary leakage is evenly distributed in the steam generators.

3. For the unconservative analysis, the reactor coolant activity is 1% defective fuel (see Table 11.1-2). For the realistic analysis, 12% of these values are used.

4. No noble gas is dissolved or contained in the steam generator water, i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser air removal system.

5. The blowdown rate from steam generators is continuous at 12,756 and 42,000 lbs/hr per steam generator for the conservative and realistic cases, respectively.

Secondary system equilibrium concentrations are provided in Table 15.2-6.

The following additional assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated loss of offsite power to plant auxiliaries:

1. Offsite power is lost; main condensers are not available for steam dump.

2. Eight (8) hours after the accident the Residual Heat Removal System commences operation to cooldown the plant.

3. After 8 hours following the accident, no steam and activity are released to the environment.

4. No condenser air removal system release and no steam generator blowdown during the accident.

5. No noble gas is dissolved in steam generator water.

6. The iodine partition factor in the steam generators is 0.01 between the steam generator steam and water phases.

7. During the postulated accident, iodine carryover from the primary side is uniformly mixed with the water in the steam generators and is diluted by the incoming feedwater.

8. The steam release for cooling down the plant is equally contributed by all steam generators.

9. 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.

10. Dose model used to evaluate the environmental consequence of this accident is given in Appendix 15A.

11. Steam releases to the atmosphere for the loss of offsite power are given by Table 15.2-5.

Using the previously listed assumptions, isotopic releases to the environment are summarized by Tables 15.2-7 and 15.2-8 for realistic and conservative analyses, respectively.

Gamma, beta, and thyroid doses in the first 2 hours of the loss of offsite power to plant auxiliaries for the realistic analysis at the site boundary are 9.51×10^{-7} rem, 1.85×10^{-6} rem and 1.77×10^{-5} rem, respectively. The corresponding doses at the low population zone are 2.20×10^{-7} rem, 4.30×10^{-7} rem, and 3.11×10^{-6} rem, respectively.

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The gamma, beta, and thyroid doses in the first 2 hours of the loss of offsite power to plant auxiliaries for the conservative analysis at the site boundary are 9.69×10^{-4} rem, 1.86×10^{-3} rem, and 4.68×10^{-2} rem, respectively. Corresponding doses at the low population zone are 2.23×10^{-4} rem, 4.31×10^{-4} rem and 7.91×10^{-3} rem, respectively, for the duration of the accident.

The doses for this accident are well within the limits defined in 10 CFR 100 (25 Rem, whole body and 300 Rem, thyroid).

15.2.10 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTION

15.2.10.1 Identification of Causes and Accident Description

Excessive feedwater additions or reductions in feedwater enthalpy are a 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 and overtemperature protection (high neutron flux, overtemperature ΔT, and overpower ΔT trips) prevent any power increase that could lead to fuel melting or a DNBR that is less than the DNBR limit. RN 10-033

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

An example of a reduction in feedwater enthalpy is the transient associated with a postulated failure of the feedwater digital control system that causes the loss of feedwater heating. This results in a sudden reduction in feedwater temperature at the inlet to the steam generators and a consequential greater load demand on the RCS due to increased subcooling in the steam generator. In the presence of a negative moderator temperature coefficient, positive reactivity will be inserted casing an increase in power. RN 10-033

15.2.10.2 Analysis of Effects and Consequences

15.2.10.2.1 Method of Analysis

The excessive heat removal due to feedwater system malfunction transient is analyzed with the LOFTRAN code. This code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The detailed thermal-hydraulic THINC code is used for the limiting case to calculate the DNBR using transient information from LOFTRAN. RN 10-033

The system is analyzed to evaluate plant behavior in the event of a feedwater system malfunction.

Excessive feedwater addition due to a control system malfunction or operator error which allows a feedwater control valve to open fully is considered. Three (3) cases are analyzed as follows:

1. Accidental opening of one (1) feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large negative moderator temperature coefficient characteristic of end of core life conditions.

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

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

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The transient response following an excessive feedwater addition event is calculated with the following assumptions:

1. For the feedwater control valve accident at full power, one (1) feedwater control valve is assumed to malfunction resulting in a step increase to 250% of nominal feedwater flow to one (1) steam generator.

2. For the feedwater control valve accident at zero load condition, one (1) feedwater control valve is assumed to malfunction resulting in a step increase in flow from zero to the nominal full load value for one (1) steam generator.

3. For the zero load condition, feedwater temperature is at a conservatively low value of 70°F.

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

5. No credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

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

A feedwater enthalpy reduction due to the failure of the feedwater digital control system that results in the loss of all feedwater heaters is considered. Two (2) cases are analyzed as follows:

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1. Loss of all feedwater heaters with the reactor in manual control at full power.

2. Loss of all feedwater heaters with the reactor in automatic control at full power.

The failure of the digital control system is limiting at full power; thus, a zero load case is not analyzed.

The transient response following a feedwater enthalpy reduction event is calculated with the following assumptions:

1. The feedwater temperature cooldown profile resulting from the failure of the feedwater digital control system is conservatively modeled with two successive step changes. The first step decrease from the nominal full power feedwater temperature to the nominal outlet temperature of the deaerator downstream of the low pressure feedwater heater string, occurs at time zero. The second step decrease to the nominal outlet temperature of the condensate system (inlet to the low pressure heaters) occurs after the water downstream of the low pressure feedwater heaters at the time of the event has been purged.

2. The feedwater flow rate to the steam generators remains at the nominal full power value throughout the transient. RN 10-033

3. The transient is terminated by a reactor trip from an overpower ΔT signal, and by a low pressurizer pressure safety injection system actuation signal, which closes all feedwater control valves, closes all feedwater bypass valves, trips the main feedwater pumps, and shuts the feedwater isolation valves.

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

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system will actuate to trip the reactor due to an overpower condition (feedwater enthalpy reduction) or trip the turbine (excessive feedwater addition). No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

15.2.10.2.2 Results

<u>Excessive Feedwater Addition</u>

In the case of an accidental full opening of one (1) feedwater control valve 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 Section 15.2.1 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 25% of nominal full power. RN 10-033

The full power excessive feedwater flow case with the reactor in manual control results in the greatest power increase. Assuming the reactor to be in the automatic control mode results in a slightly less severe transient. The rod control system is not required to function for an excessive feedwater flow event. A turbine trip and feedwater isolation are actuated when the steam generator level reaches the high-high level setpoint. For convenience, reactor trip is assumed to be initiated upon turbine trip. However, this function is not necessary for core protection. Should turbine trip not initiate a reactor trip signal, reactor trip will eventually occur on low-low steam generator level following feedwater isolation. RN 10-033

For excessive feedwater flow cases, continuous addition of cold feedwater is prevented by closure of feedwater control valves, a trip of the feedwater pumps, and closure of the feedwater pump bypass and isolation valves on steam generator high-high level signal. RN 10-033

Transient results (see Figures 15.2-33 and 15.2-34) show the core heat flux, pressurizer pressure, T_{avg}, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Steam generator level rises until the feedwater is terminated as a result of the high-high steam generator level trip. The DNBR does not drop below the safety analysis limit DNBR value. RN 10-033

Feedwater Enthalpy Reduction

For the feedwater enthalpy reduction cases, the manual reactor control case results in the greatest power increase. This case is also more severe than the limiting excessive feedwater addition case. The reduction in feedwater temperature caused by the failure of the feedwater digital control system increases the thermal load on the primary system. The resultant temperature and power transients cause a reactor trip on an overpower ΔT signal. When the pressurizer pressure reaches the safety injection low pressure setpoint, the feedwater control and isolation valves are closed and feedwater isolation occurs. Following reactor trip and feedwater isolation, the plant will approach a stabilized condition at which point normal operating procedures may be followed. RN 10-033

Transient results (see Figures 15.2-33a and 15.2-34a) show the core heat flux, pressurizer pressure, T_{avg}, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Since the power level increases during the feedwater enthalpy reduction event, the fuel temperature will also increase until reactor trip occurs. However, the peak linear heat rate produced in the fuel rods remains below a value which would result in exceeding the fuel melting temperature. Hence, fuel melting is precluded for this event. The transient results also show that the DNBR does not fall below the safety analysis DNBR limit value.

15.2.10.3 Conclusion

The reactivity insertion rate that 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 DNBRs encountered for the full power cases (excessive feedwater addition and feedwater enthalpy reduction) are above the safety analysis limit DNBR value and fuel melting is precluded. RN 10-033

15.2.11 EXCESSIVE LOAD INCREASE INCIDENT

15.2.11.1 Identification of Causes and Accident Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The 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

15.2.11.2.1 Method of Analysis

This accident is analyzed using the LOFTRAN[4] Code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four (4) cases are analyzed to demonstrate the plant behavior following a 10% step load increase from rated load. These cases are as follows:

1. Reactor control in manual with BOL minimum moderator reactivity feedback,

2. Reactor control in manual with EOL maximum moderator reactivity feedback,

3. Reactor control in automatic with BOL minimum moderator reactivity feedback,

4. Reactor control in automatic with EOL maximum moderator reactivity feedback.

For the BOL minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore the least inherent transient response capability. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

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

This accident is analyzed with the Revised Thermal Design Procedure as described in Reference [5]. Initial reactor power, RCS pressure and temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference [5].

Plant characteristics and initial conditions are further discussed in Section 15.1.

15.2.11.2.2 Results

The calculated sequence of events for the excessive load increase incident are shown on Table 15.2-1.

Figures 15.2-35 through 15.2-38 illustrate the transient with the reactor in the manual control mode. As expected, for the BOL minimum moderator feedback 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 EOL maximum moderator feedback 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 limit value.

Figures 15.2-39 through 15.2-42 illustrate the transient assuming the reactor is in the automatic control mode. Both the BOL minimum moderator feedback and the EOL maximum moderator feedback 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 limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power.

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

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

15.2.11.3 <u>Conclusions</u>

The analysis presented above shows that for a 10% step load increase, the DNBR remains above the safety analysis limit value, thereby precluding fuel or clad damage. The plant reaches a stabilized condition rapidly, following the load increase.

15.2.12 ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

15.2.12.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flowrate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure, which could reach hot leg saturation conditions without Reactor Protection System intervention. If saturated conditions are reached, the rate of depressurization is slowed considerably. However, the pressure continues to decrease throughout the event. The effect of the pressure decrease is to increase power via moderator density feedback. However, if the plant is in the automatic mode, the rod control system functions to maintain the power essentially constant throughout the initial stages of the transient. The average coolant temperature remains approximately the same, but the pressurizer level increases until reactor trip because of the decreased reactor coolant density.

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

1. Overtemperature ΔT.

2. Pressurizer low pressure.

15.2.12.2 Analysis of Effects and Consequences

15.2.12.2.1 Method of Analysis

The accidental depressurization transient is analyzed with the LOFTRAN code[4]. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the Revision Thermal Design Procedure as described in Reference [5].

In calculating the DNBR the following conservative assumptions are made:

1. Plant characteristics and initial conditions are discussed in Section 15.1. Uncertainties and initial conditions are included in the limit DNBR as described in Reference [5].

2. A positive moderator temperature coefficient of reactivity for BOL operation in order to provide a conservatively high amount of positive reactivity feedback due to changes in moderator temperature. The spatial effect of voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. These voids would tend to flatten the core power distribution.

3. A low (absolute value) Doppler coefficient of reactivity such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

15.2.12.2.2 Results

Figure 15.2-43 illustrates the nuclear power transient following the RCS depressurization accident. The flux increases until the time reactor trip occurs on overtemperature ΔT, thus resulting in a rapid decrease in the nuclear flux. The time of reactor trip is shown in Table 15.2-1.

The pressure decay transient following the accident is given on Figure 15.2-44. The resulting DNBR never goes below the safety analysis limit value as shown on Figure 15.2-45.

15.2.12.3 Conclusion

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

15.2.13 ACCIDENTAL DEPRESSURIZATION OF THE MAIN STEAM SYSTEM

15.2.13.1 Identification of Causes and Accident Description

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

The steam release as a consequence of this accident results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant

temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.

The 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 limit DNBR value 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 mitigation of an accidental depressurization of the main steam system.

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

    a. Two (2) out of 3 low pressurizer pressure signals.

    b. Two (2) out of 3 high-1 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, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater isolation valves.

15.2.13.2 Analysis of Effects and Consequences

15.2.13.2.1 Method of Analysis

The following analyses of a secondary system steam release are performed for this section based on methodologies documented in Reference [16].

1. A full plant digital computer simulation (LOFTRAN [4]) to determine Reactor Coolant System temperature and pressure during cooldown, and the effect of safety injection.

2. An analysis to ascertain that the reactor does not exceed the limit DNBR value.

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

1. End of life shutdown margin at no load, equilibrium xenon conditions, and with the most reactive rod cluster control assembly stuck in its fully withdrawn position. 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 steam release 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 1150 psia corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown in Figure 15.2-46.

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 is shown on Figure 15.2-47. This corresponds to the flow delivered by 1 charging pump delivering its full contents to the cold leg header. No credit has been taken for the low concentration boric acid that must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) isolation valves prior to the delivery of high concentration boric acid (2300 ppm) to the reactor coolant loops.

4. The case studied is an initial steam flow of 255 pounds per second at 1100 psia from 1 steam generator with offsite power available. This is the maximum capacity of any single steam dump, relief or safety valve. Initial hot shutdown conditions at time 0 are assumed since this represents the most conservative initial condition.

   Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches a trip point. Following a trip at power, the Reactor Coolant System contains more stored energy than at no load, the average coolant temperature is higher than at no load, and there is appreciable energy stored in the fuel.

   Thus, the additional stored energy is removed via the cooldown caused by the steam release before the no load conditions of Reactor Coolant System temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no load condition at time zero. However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the Reactor Coolant System cooldown are less for steam line release 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. No credit was taken for secondary side safety injection actuation on low steam pressure.

15.2.13.2.2 Results

This 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.

Figures 15.2-48 and 15.2-49 show the transient arising as the result of an initial steam flow of 255 lbs/second at 1100 psia with steam release from 1 safety valve. The assumed steam release is the maximum capacity of any single steam dump or safety valve. In this case safety injection is initiated automatically by low pressurizer pressure. Operation of 1 centrifugal charging pump is considered. Boron solution at 2300 ppm from the refueling water storage tank enters the Reactor Coolant System providing sufficient negative reactivity to terminate the transient. The reactivity transient for the case shown in Figure 15.2-49 is more severe than the case of a failed steam generator safety or relief valve, which is terminated by steam line differential pressure, or a failed condenser dump valve that is terminated by low pressurizer pressure and level.

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 the 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 are satisfied. For an accidental depressurization of the Main Steam System, the DNB design basis is met. This case is less limiting than the rupture of a main steam pipe case presented in Section 15.4.

15.2.14 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

15.2.14.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuating signal. A spurious signal in any of the following channels could cause this accident:

1. High containment pressure.

2. Low pressurizer pressure.

3. High steam line differential pressure.

4. Low steam line pressure.

5. Manual actuation.

Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the refueling water storage tank. The charging pumps then force highly concentrated (2300 ppm) boric acid solution from the RWST, through the header and injection line, and into the cold legs of each loop. The low head safety injection pumps also start automatically but provide no flow when the Reactor Coolant System is at normal pressure.

A Safety Injection System signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the safety injection system will also produce a reactor trip. Therefore, 2 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.

1. Case A

   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 will determine this by following approved procedures. In the transient case, the operator would stop the safety injection and bring the plant to the hot shutdown condition. If the SIS must be disabled for repair, boration should continue and the plant brought to cold shutdown.

2. Case B

   For Case B, the Reactor Protection System does not produce an immediate trip and the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in the reactor power. At beginning of life, the power mismatch causes a drop in T_{avg} and consequent coolant shrinkage, and pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load when the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the Reactor Protection System low-pressure trip or by manual trip.

   Results at end of life are similar except that moderator feedback effects result in a slower transient. The pressurizer pressure and level increase slowly and the coolant T_{avg} decreases slowly. The transient is eventually terminated by the

Reactor Protection System high pressurizer pressure or high pressurizer level trip or by manual trip.

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

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

15.2.14.2 Analysis of Effects and Consequences

15.2.14.2.1 Method of Analysis

The spurious operation of the Safety Injection System is analyzed with the LOFTRAN[4] code. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the safety injection system. The program computes pertinent plant variables including temperatures, pressures, and power level.

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

A typical transient is considered representing conditions at beginning of core life.

This accident is analyzed with the Revised Thermal Design Procedure as described in Reference [5]. The assumptions made in the analysis are:

1. Initial Operating Conditions

   The initial reactor power, pressure, and reactor coolant system temperatures are assumed at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5.

2. Moderator and Doppler Coefficients of Reactivity

   A positive 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 inoperable in order to increase the rate of pressure drop.

5. Boron Injection

   At time 0, 2 charging pumps inject 2300 ppm borated water into the cold legs of each loop.

6. Turbine Load

   Turbine load was assumed constant until the 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. The trip was conservatively assumed to be delayed until the pressure reached 1775 psia.

15.2.14.2.2 Results

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

The transient response for the minimum feedback case is shown in Figures 15.2-50 and 15.2-51. Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until 36 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 setpoint is reached at 52 seconds and rods start moving into the core at 54 seconds.

15.2.14.3 <u>Conclusions</u>

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.

DNBR 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.

15.2.15 ANTICIPATED TRANSIENTS WITHOUT TRIP

The worst common mode failure which is postulated to occur is the failure to scram the reactor after an anticipated transient has occurred. A series of generic studies (References [11] and [14]) on Anticipated Transients Without Scram (ATWS) showed acceptable consequences would result provided that the turbine trips and emergency feedwater flow is initiated in a timely manner. The effects of ATWS events are not considered as part of the design basis for transients analyzed in Chapter 15. The final NRC ATWS rule (Reference [15]) requires that Westinghouse designed plants install ATWS Mitigation System Actuation Circuitry (AMSAC) to initiate a turbine trip and actuate emergency feedwater flow independent of the Reactor Protection System. The V. C. Summer AMSAC design is described in Section 7.8 of the FSAR.

15.2.16 REFERENCES

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16. Reactor Core Response to Excessive Secondary Steam Releases, WCAP-9226-R1, January 1978.

17. "Response to CAPs IR 06-184-M004.03 - Examination of PORV Operability for LONF-LOAC," Westinghouse Calculation CN-TA-07-43, Revision 2. RN 14-026

18. "Rod Withdrawal at Power Overpressure Analysis for V. C. Summer," Westinghouse Engineering Letter, LTR-TA-09-145, dated July 21, 2009 and transmitted by CGE-09-028. RN 09-021

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, sec
Uncontrolled RCCA Withdrawal from a Subcritical Condition	Initiation of uncontrolled rod withdrawal 90 pcm/sec reactivity insertion rate from 10^{-9} fraction of nominal power	0.0
	Power range high neutron flux setpoint reached	8.8
	Peak nuclear power occurs	8.9
	Rods begin to fall into core	9.3
	Peak heat flux occurs	11.8
	Peak hot spot average clad temperature occurs	11.8
	Peak hot spot average fuel temperature occurs	12.1
	Peak hot spot fuel centerline temperature occurs	12.5

TABLE 15.2-1 (continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, sec
Uncontrolled RCCA Withdrawal at Power			99-01
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (70 pcm/sec)	0.0
	Power range high neutron flux high trip setpoint reached	1.6
	Rods begin to fall into core	2.1
	Minimum DNBR occurs	3.1
2. Case B	Initiation of uncontrolled RCCA withdrawal at a low reactivity insertion rate (5 pcm/sec)	0.0
	Overtemperature ΔT reactor trip signal initiated	265.9
	Rods begin to fall into core	267.9
	Minimum DNBR occurs	268.0

TABLE 15.2-1 (continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, Sec	
Uncontrolled Boron Dilution			
1. Dilution during refueling	Dilution begins	0	
	Operator receives high flux at shutdown alarm, set at twice background	2123	RN 00-073
	Operator isolates source of dilution; minimum margin to criticality occurs	3923	RN 00-073
2. Dilution during cold shutdown	Operator receives high flux at shutdown alarm set at twice background	0	
	Operator isolates source of dilution; shutdown margin is lost	816	
3. Dilution during hot shutdown	Operator receives high flux at shutdown alarm set at twice background	0	
	Operator isolates source of dilution; shutdown margin is lost	816	
4. Dilution during hot standby	Operator receives high flux at shutdown alarm set at twice background	0	
	Operator isolates source of dilution; shutdown margin is lost	804	99-01
5. Dilution during startup	Power Range-low setpoint reactor trip due to dilution	0	
	Shutdown margin lost (if dilution continues after trip)	1080	
6. Dilution during full power operation			
a. Automatic reactor control	Operator receives lo-lo rod insertion limit alarm due to dilution	0	
	Shutdown margin is lost	1176	
b. Manual reactor control	Overtemperature ΔT reactor trip due to dilution	0	
	Shutdown margin lost (if dilution continues after trip)	942	

TABLE 15.2-1 (continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, Sec
Partial Loss of forced Reactor Coolant Flow		
All loops operating, one pump coasting down	Coastdown begins	0.0
	Low flow reactor trip	1.47
	Rods begin to drop	2.47
	Minimum DNBR occurs	3.5
Loss of External Electrical Load		
1. With pressure control (BOL)	Loss of electrical load	0.0
	Initiation of steam release from steam generator safety valves	8.0
	High pressurizer pressure trip setpoint reached	10.5
	Rods begin to drop	12.5
	Minimum DNBR occurs	14.0
	Peak pressurizer pressure occurs	14.0
2. Without pressure control (BOL)	Loss of Electrical load	0.0
	High pressurizer pressure trip setpoint reached	5.1
	Rods begin to drop	7.1
	Peak pressurizer pressure occurs	8.7
	Initiation of steam release from steam generator safety valves	9.4
	Minimum DNBR occurs	(a)

99-01

99-01

RN 03-042

(a) DNBR does not decrease below its initial value

TABLE 15.2-1 (continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, Sec w/ power	Time, Sec w/o power
Loss of Normal Feedwater and Loss of Offsite Power to the Station Auxiliaries (Station Blackout)	Main feedwater flow stops	10	10
	Low-low steam generator water level reactor trip setpoint	68.7	66.3
	Rods begin to drop	70.7	68.3
	Reactor coolant pumps begin to coast down	-	70.3
	All steam generators begin to receive emergency feedwater from one motor driven emergency feedwater pump	128.7	126.3
	Cold emergency feedwater is delivered to the steam generators	146	143
	Peak water level in pressurizer occurs	3060	368
	Core decay heat plus pump heat decreases to emergency feedwater heat removal capacity	~3200	--
	Core decay heat decreases to emergency feedwater heat removal capacity	--	~370

TABLE 15.2-1 (continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, Sec.	
Excessive Feedwater Flow at Full Load	One main feedwater control valve fails fully open	0	
	High-high steam generator level signal generated	22.2	
	Turbine trip occurs due to high-high steam generator level	24.2	
	Minimum DNBR occurs	26.0	
	Reactor trip due to turbine trip[1]	26.2	
	Feedwater isolation valves fully closed	32.2	
Feedwater Enthalpy Reduction at Full Load	Feedwater digital control system fails and a loss of all feedwater heaters occurs	0.0	
	Overpower ΔT reactor trip setpoint reached	53.6	
	Rod motion occurs	55.1	RN 10-033
	Minimum DNBR occurs	56.0	
	Peak linear heat rate occurs	56.0	
	Low pressurizer pressure safety injection setpoint reached	72.0	
	Feedwater isolation occurs	97.3	
Excessive Load Increase			
1. Manual reactor control (BOL minimum moderator feedback)	10% step load increase	0.0	
	Equilibrium conditions reached (approximate times only)	125	

[1] Not a required safety function.

TABLE 15.2-1 (continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, Sec
Excessive Load Increase		
1. Manual reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	125
2. Manual reactor control (EOL maximum moderator feedback)	10% Step load increase	0.0
	Equilibrium conditions reached (approximate times only)	50
3. Automatic reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	150
4. Automatic reactor control (EOL maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	75
Accidental Depressurization of the Reactor Coolant System	Inadvertent opening of one RCS safety valve	0.0
	Overtemperature ΔT trip setpoint reached	25.2
	Rods begin to drop	26.7
	Minimum DNBR occurs	27.2
Accidental Depressurization of the Main Steam System	Inadvertent opening of one main steam safety or relief valve	0.0
	Pressurizer empties	189
	Boron from the RWST reaches RCS loops	253
Inadvertent Operation of the ECCS During Power Operation	Charging pumps begin injecting borated water	0.0
	Low-pressure trip setpoint reached	52
	Rods begin to drop	54

02-01

TABLE 15.2-5

PARAMETERS USED IN LOSS OF OFFSITE POWER ANALYSIS

	Realistic Analysis	Conservative Analysis
Core thermal power	2958 MWt	2958 MWt
Steam generator tube leak rate prior to and during accident	100 lbs/day [(1)]	1.0 gpm
Fuel defects	0.12% [(1)]	1%
Iodine partition factor in steam generators prior to and during accident	0.01	0.01
Blowdown rate per steam generator prior to accident	42,000 lbs/hr	12,756 lbs/hr
Duration of plant cooldown by secondary system after accident	8 hr	8 hr
Steam release from three steam generators	447,900 lbs (0-2hr) 868,300 lbs (2-8hr)	447,900 lbs (0-2hr) 868,300 lbs (2-8hr)
Feedwater flow to three steam generators	375,500 lbs (0-2hr) 841,800 lbs (2-8hr)	375,500 lbs(0-2hr) 841,800 lbs (2-8hr)
Meteorology	Annual average	Accident

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(1) American National Standards Institute, “Source Term Specification, ANS/ANSI 18.1-1984.

TABLE 15.2-6

SECONDARY SYSTEM EQUILIBRIUM CONCENTRATION (μCi/lb)

Isotopes	Realistic	Conservative
I-131	1.57×10^{-2}	4.87×10^{1}
I-132	4.94×10^{-3}	6.29×10^{0}
I-133	1.96×10^{-2}	4.35×10^{1}
I-134	4.55×10^{-4}	5.05×10^{-1}
I-135	7.07×10^{-3}	1.14×10^{1}

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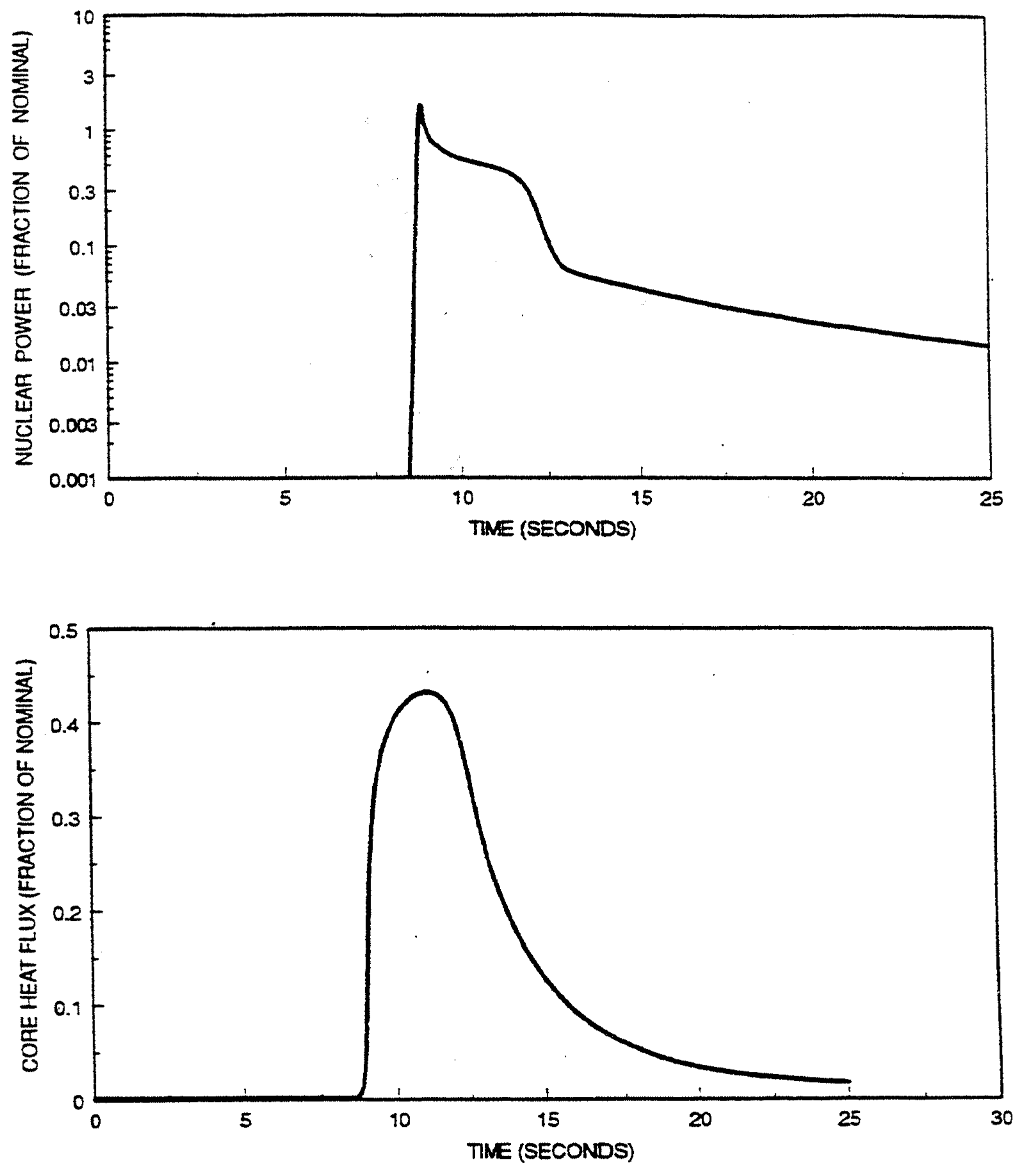
06-022

TABLE 15.2-7

LOSS OF OFFSITE POWER ACCIDENT
ISOTOPIC RELEASE TO ENVIRONMENT
REALISTIC ANALYSIS

Isotope	Activity Released to Environment by Accident (Ci) (0-2 hr)	(2-8 hr)
I-131	9.3×10^{-5}	1.9×10^{-4}
I-132	3.9×10^{-5}	8.8×10^{-5}
I-133	1.2×10^{-4}	2.4×10^{-4}
I-134	5.0×10^{-6}	1.2×10^{-5}
I-135	4.7×10^{-5}	9.8×10^{-5}
Xe-131m	1.0×10^{-3}	3.1×10^{-3}
Xe-133	1.3×10^{-1}	3.9×10^{-1}
Xe-133m	8.6×10^{-3}	2.6×10^{-2}
Xe-135	3.9×10^{-3}	1.2×10^{-2}
Xe-135m	2.4×10^{-4}	7.1×10^{-4}
Xe-138	2.9×10^{-4}	8.7×10^{-4}
Kr-83m	2.0×10^{-4}	5.9×10^{-4}
Kr-85	3.5×10^{-3}	1.0×10^{-2}
Kr-85m	8.2×10^{-4}	2.5×10^{-3}
Kr-87	5.0×10^{-4}	1.5×10^{-3}
Kr-88	1.5×10^{-3}	4.4×10^{-3}
Kr-89	4.0×10^{-5}	1.2×10^{-4}

RN
06-022



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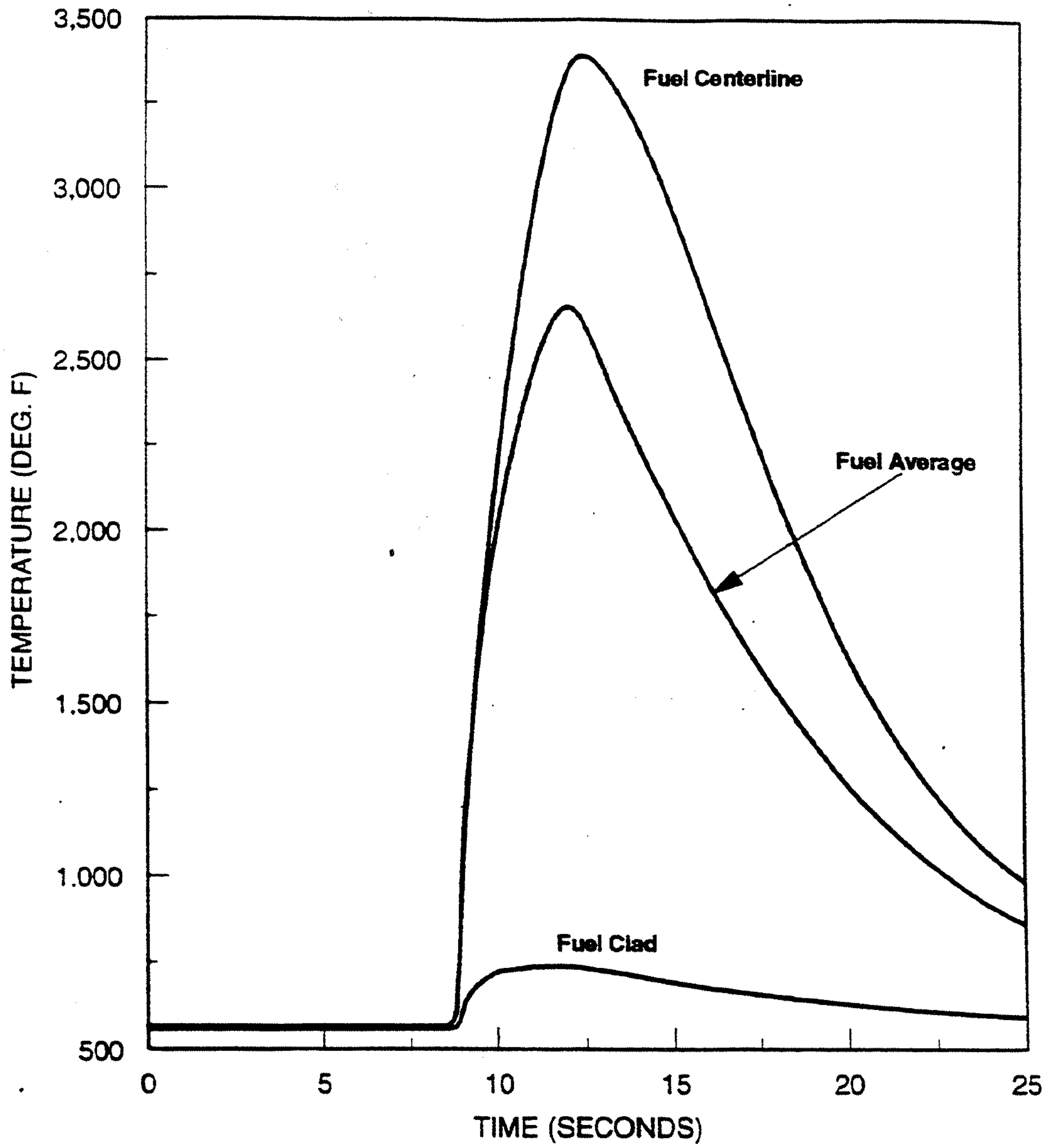
JULY 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Uncontrolled RCCA Withdrawal from Subcritical: Nuclear Power and Core Heat Flux vs. Time

Figure 15.2-1



SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Uncontrolled RCCA Withdrawal from Subcritical: Fuel Centerline, Average, and Clad Temperature vs. Time Figure 15.2-2

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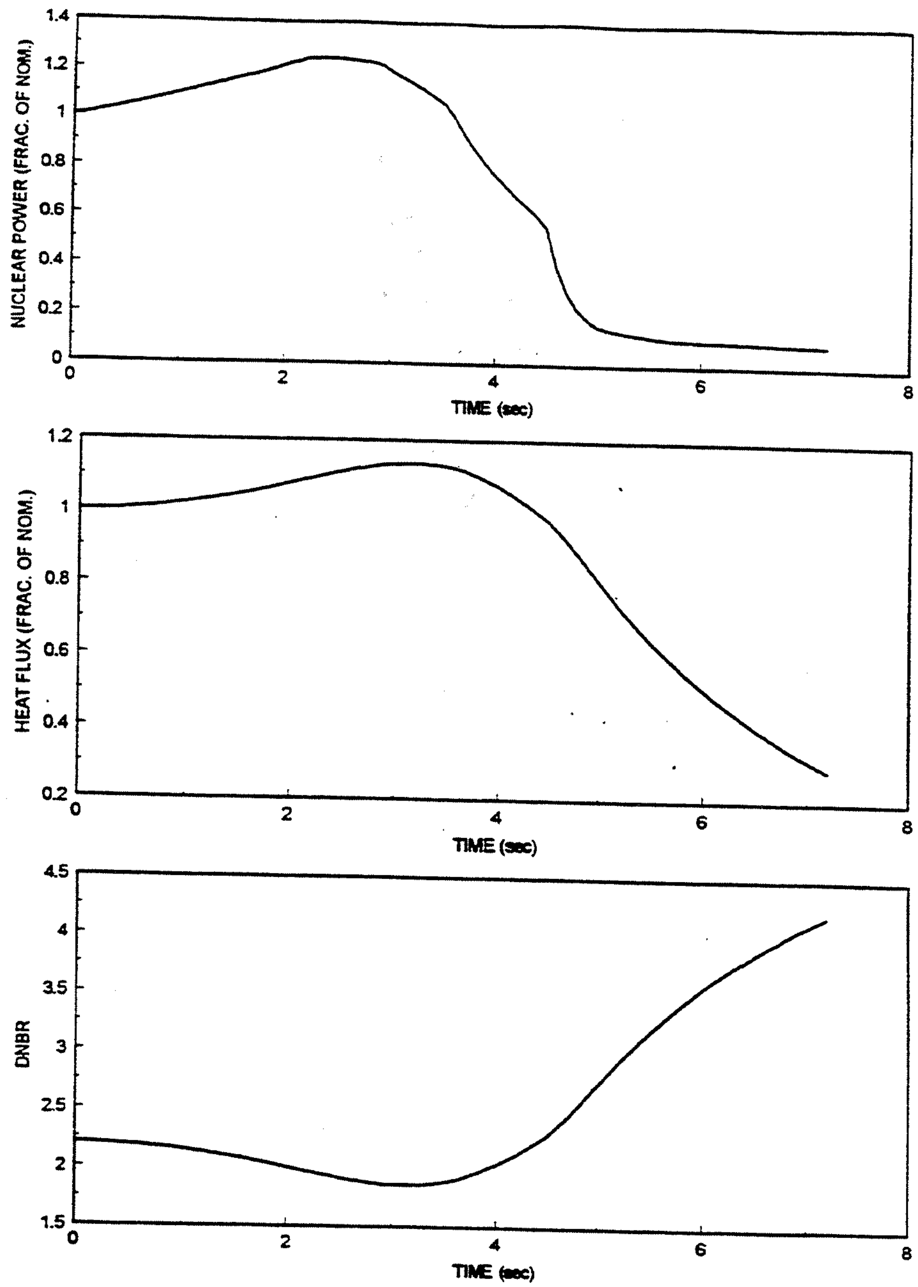
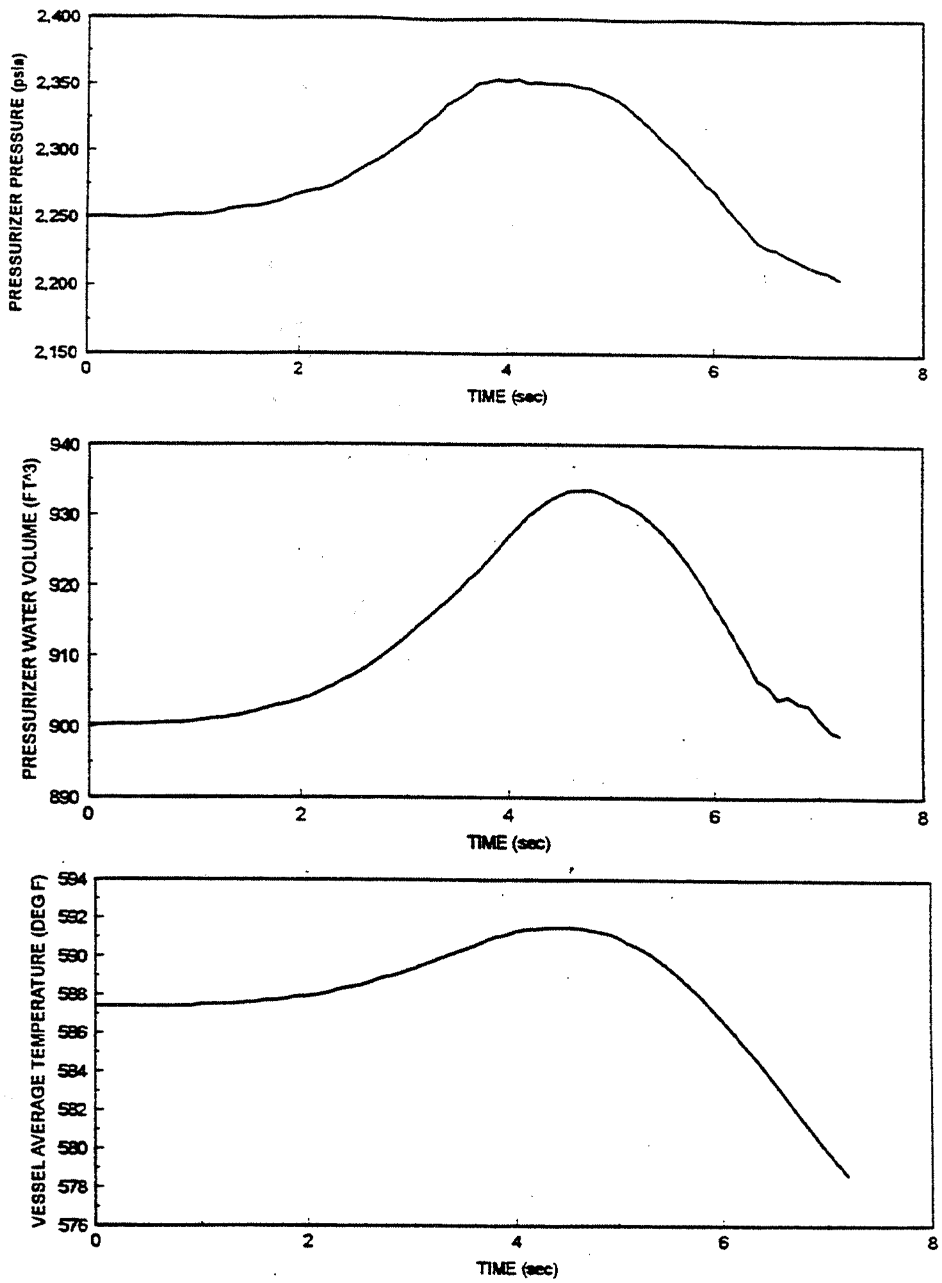
JULY 1996



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Uncontrolled Rod Withdrawal From 100% Power Terminated by High Neutron Flux Trip Nuclear Power, Heat Flux and DNBR vs. Time
Figure 15.2-3



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Uncontrolled Rod Withdrawal From
100% Power Terminated by
High Neutron Flux Trip
Pressurizer Pressure and Water Volume
and Tavg vs Time
Figure 15.2-4

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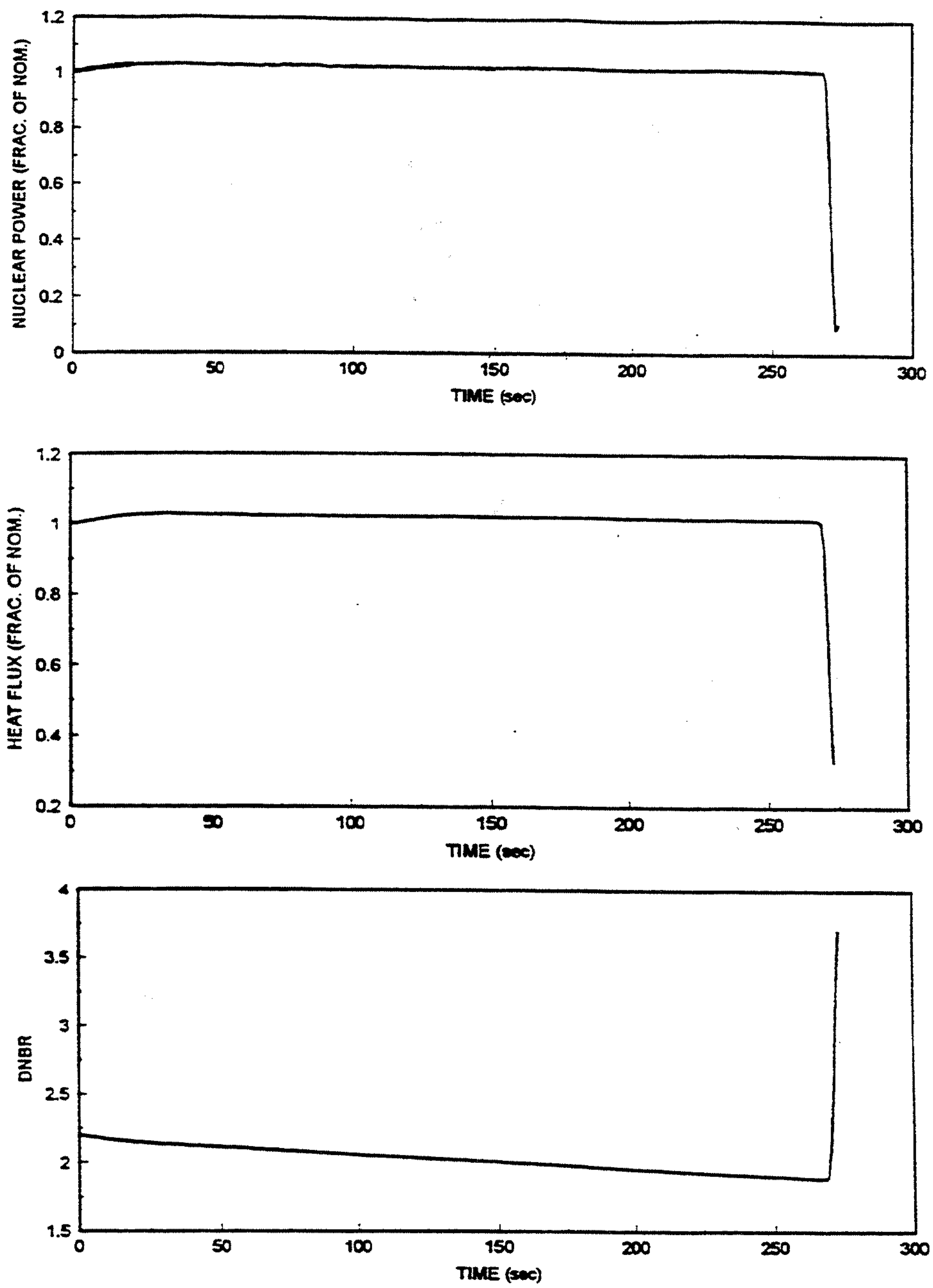
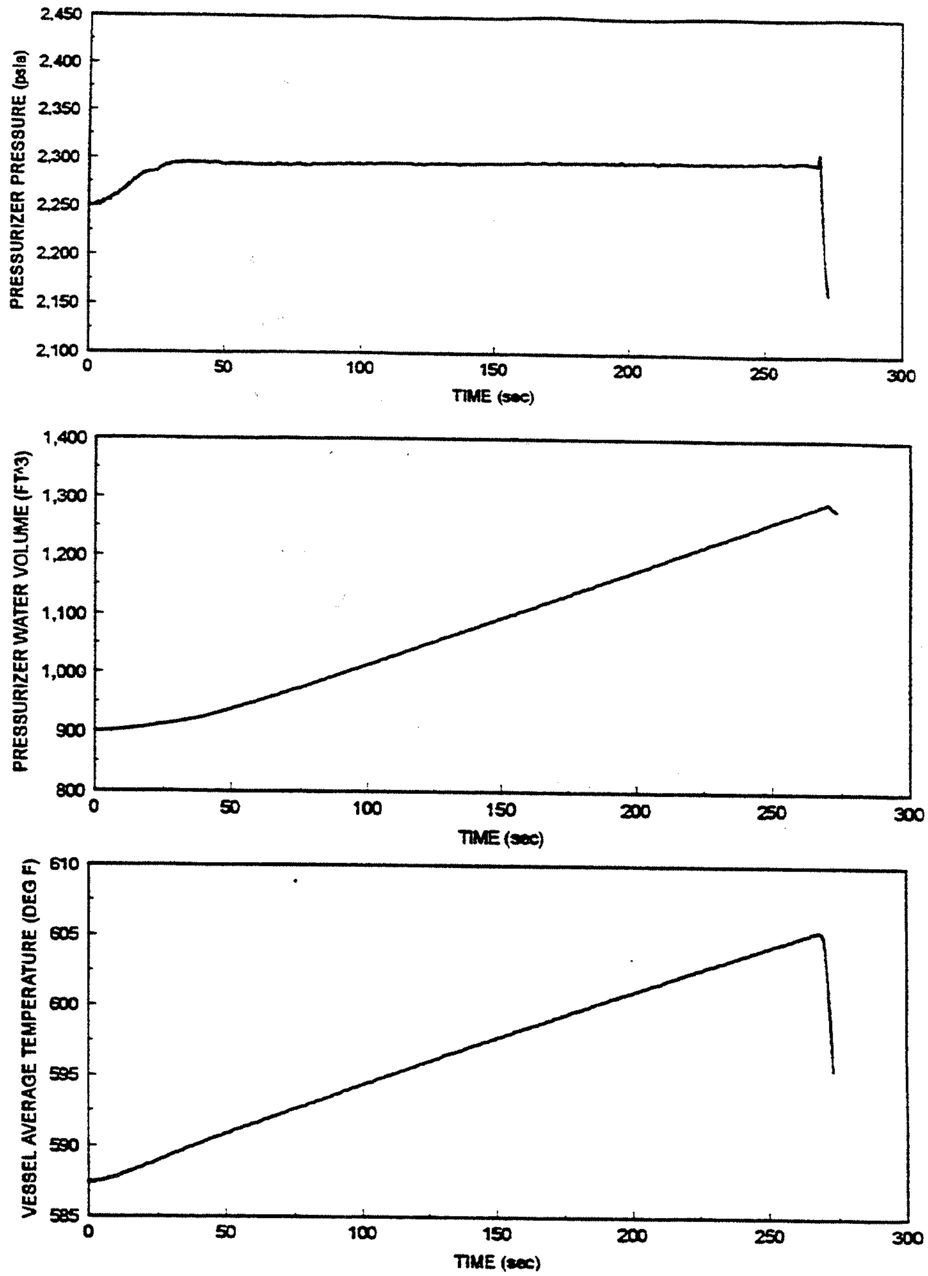
JULY 1996



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Uncontrolled Rod Withdrawal From 100% Power Terminated by Overtemperature ΔT Trip Nuclear Power, Heat Flux and DNBR vs. Time
Figure 15.2-5



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Uncontrolled Rod Withdrawal From
100% Power Terminated by
Overtemperature ΔT Trip
Pressurizer Pressure and Water Volume
and Tavg vs Time
Figure 15.2-6

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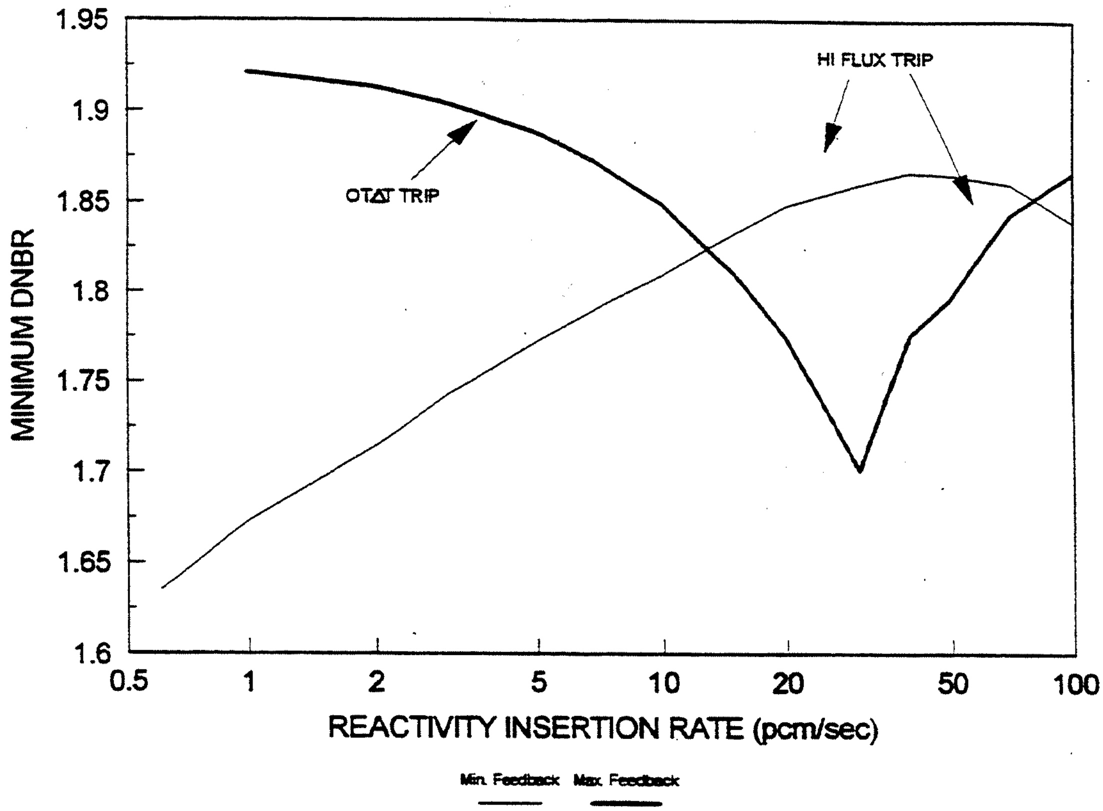

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

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

Figure 15.2-7

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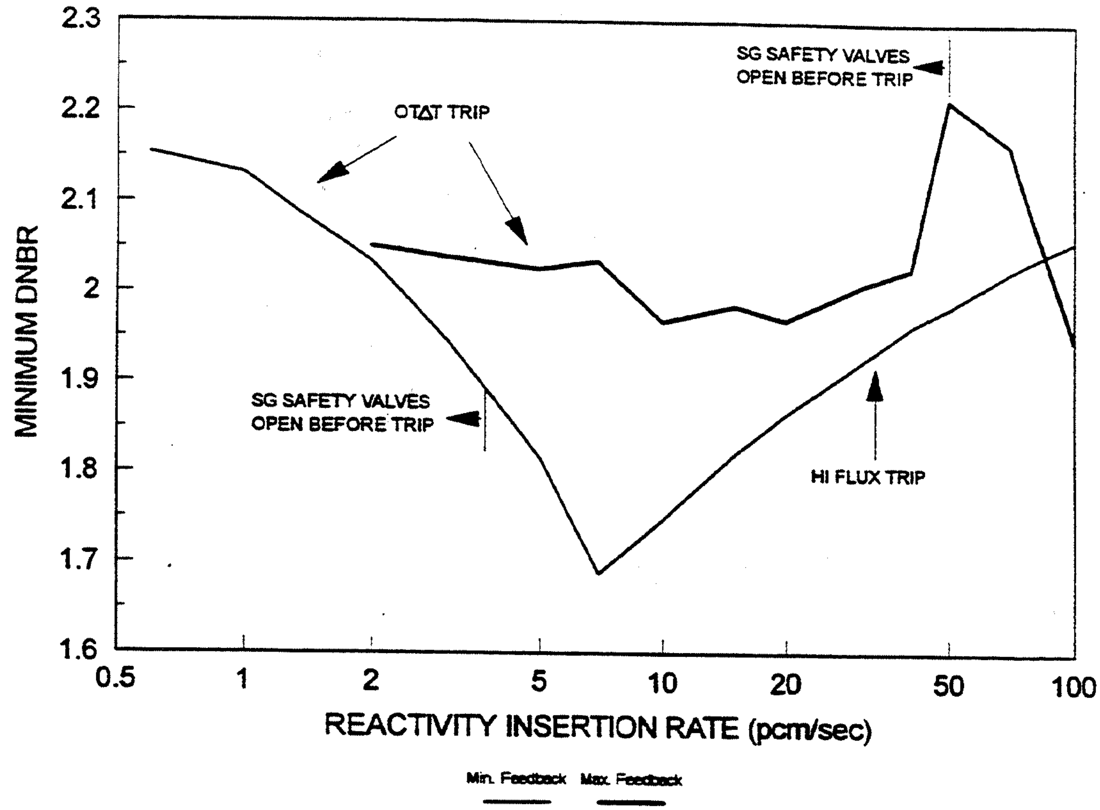

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Effect of Reactivity Insertion Rate on Minimum DNBR for a Rod Withdrawal Accident at 60% Power

Figure 15.2-8

AMENDMENT 96-02
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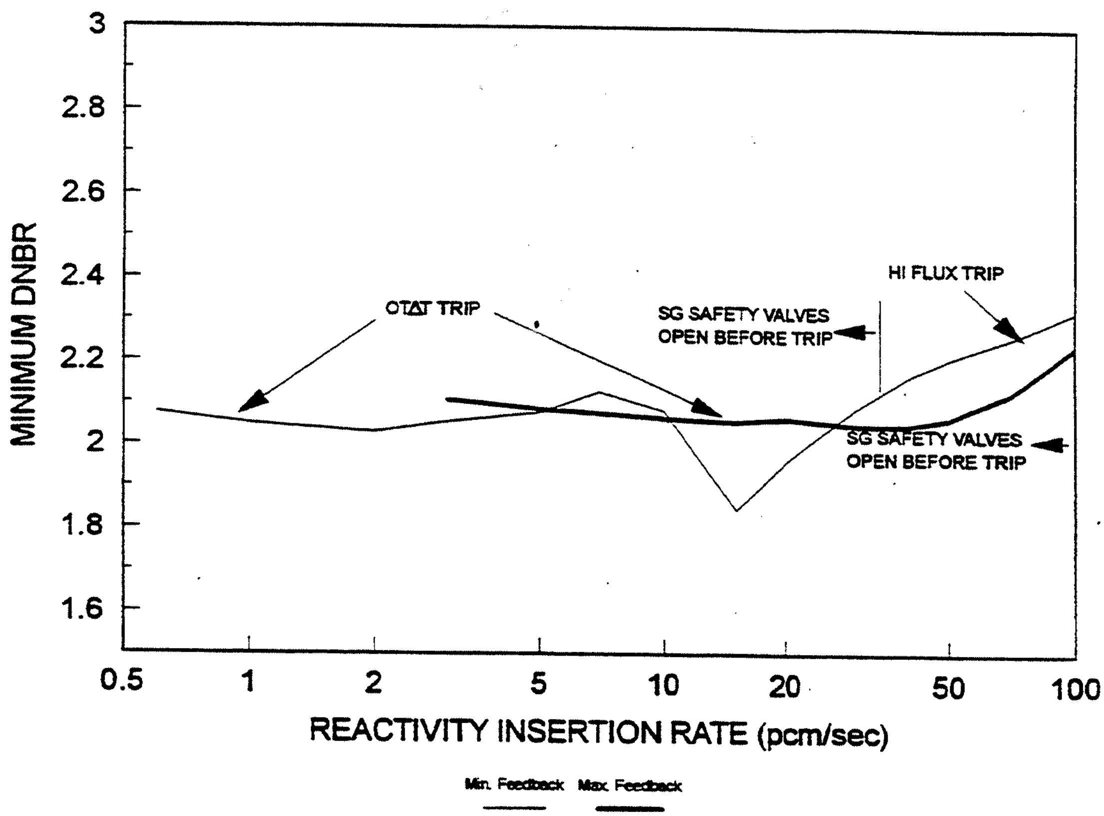

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Effect of Reactivity Insertion Rate
on Minimum DNBR for a
Rod Withdrawal Accident at
10% Power

Figure 15.2-9

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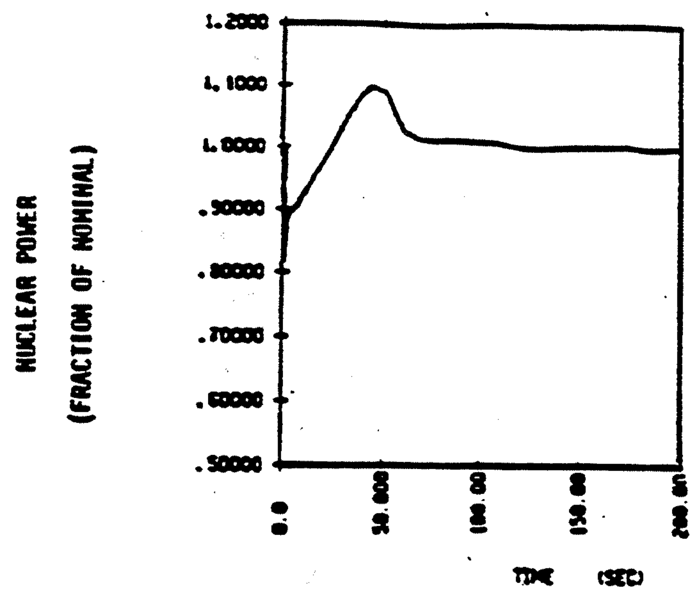

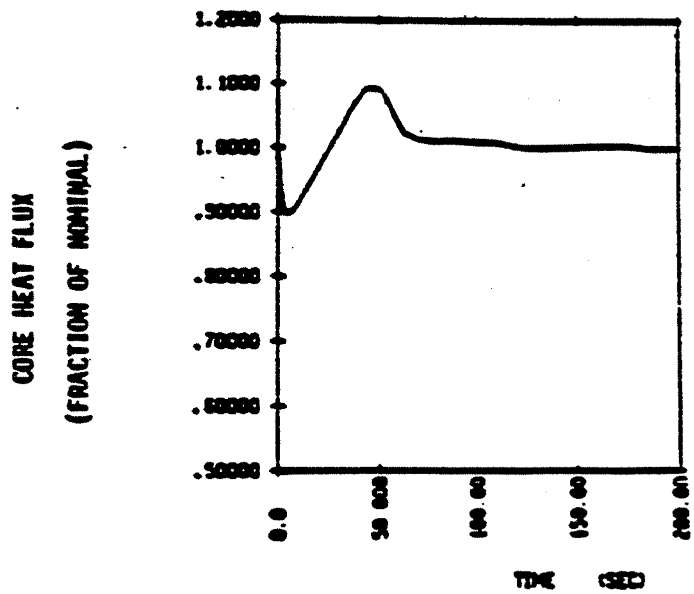

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Transient Response To A Dropped RCCA
Nuclear Power and Heat Flux vs. Time

Figure 15.2-10

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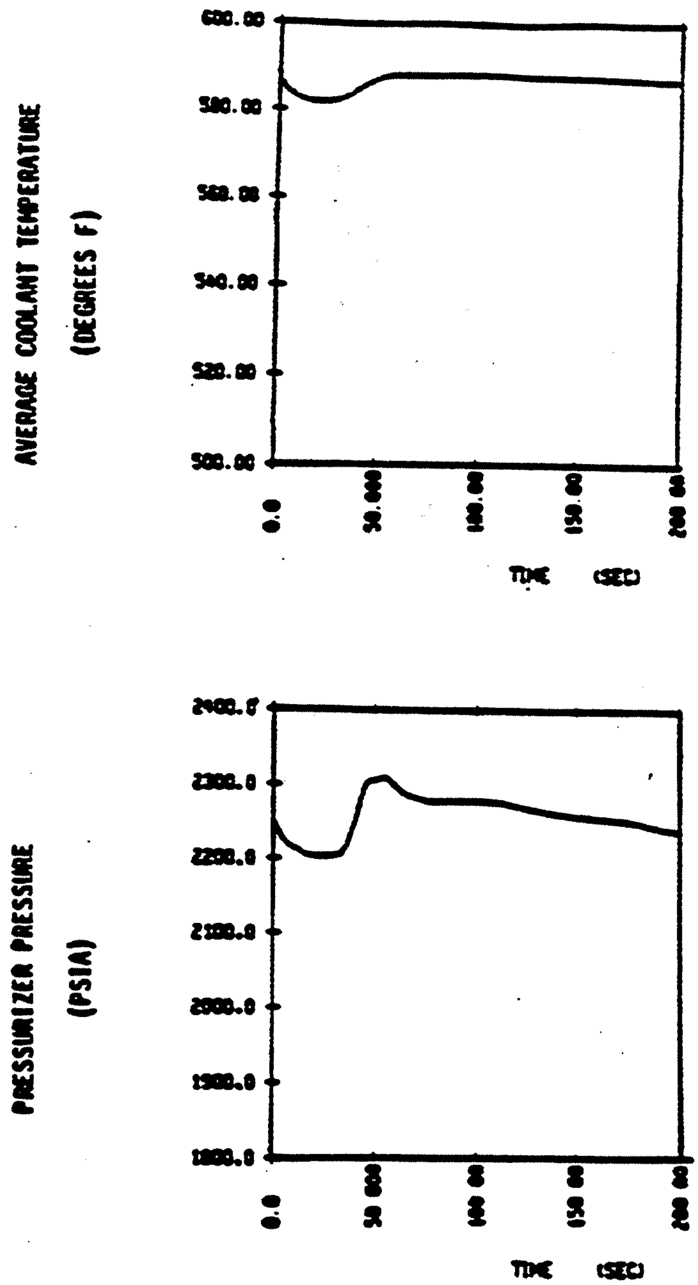

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Transient Response To A Dropped RCCA
Tavg and Pressurizer Pressure vs. Time

Figure 15.2-11

AMENDMENT 96-02
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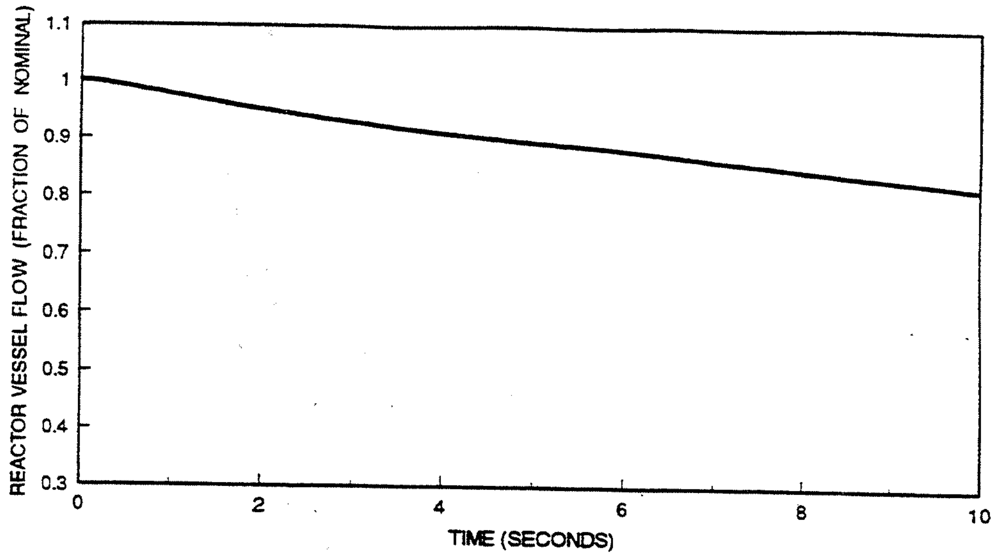
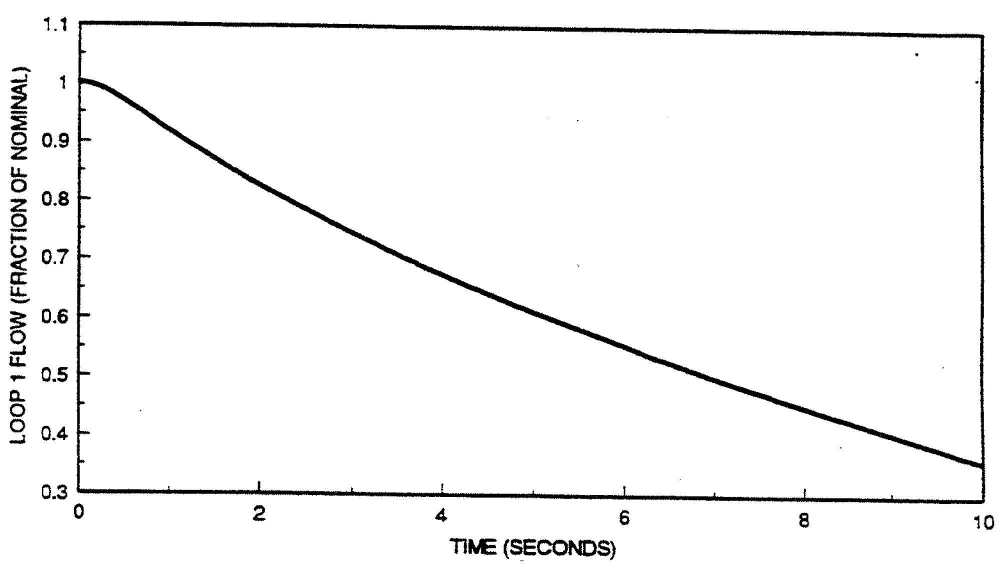
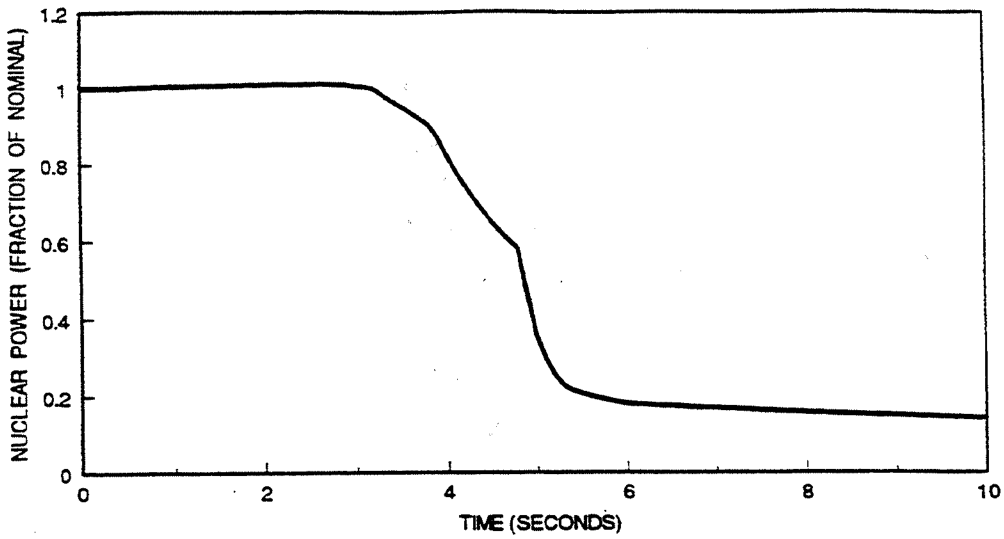
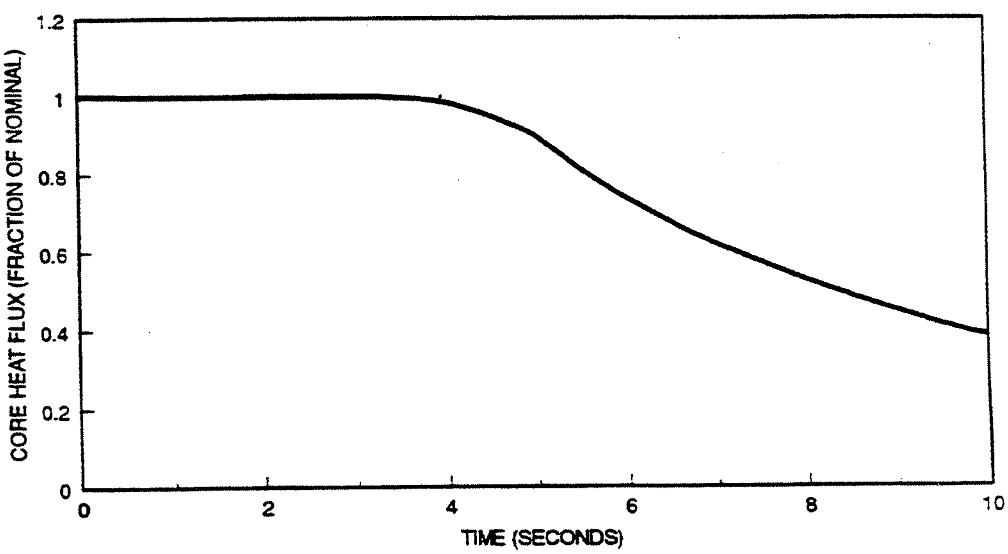




SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
All Loops Operating, One Loop Coasting Down: Vessel Flow and Faulted Loop Flow vs. Time Figure 15.2-12





SOUTH CAROLINA ELECTRIC & GAS CO. **VIRGIL C. SUMMER NUCLEAR STATION**
All Loops Operating, One Loop Coasting Down: Nuclear Power and Core Heat Flux vs. Time **Figure 15.2-13**

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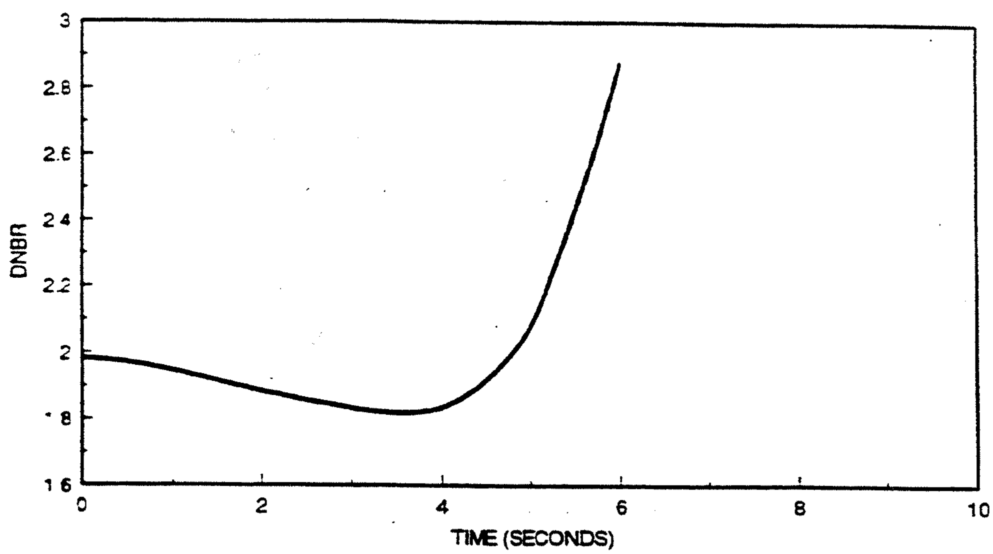

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

All Loops Operating, One Loop Coasting Down: DNBR vs. Time

Figure 15.2-14

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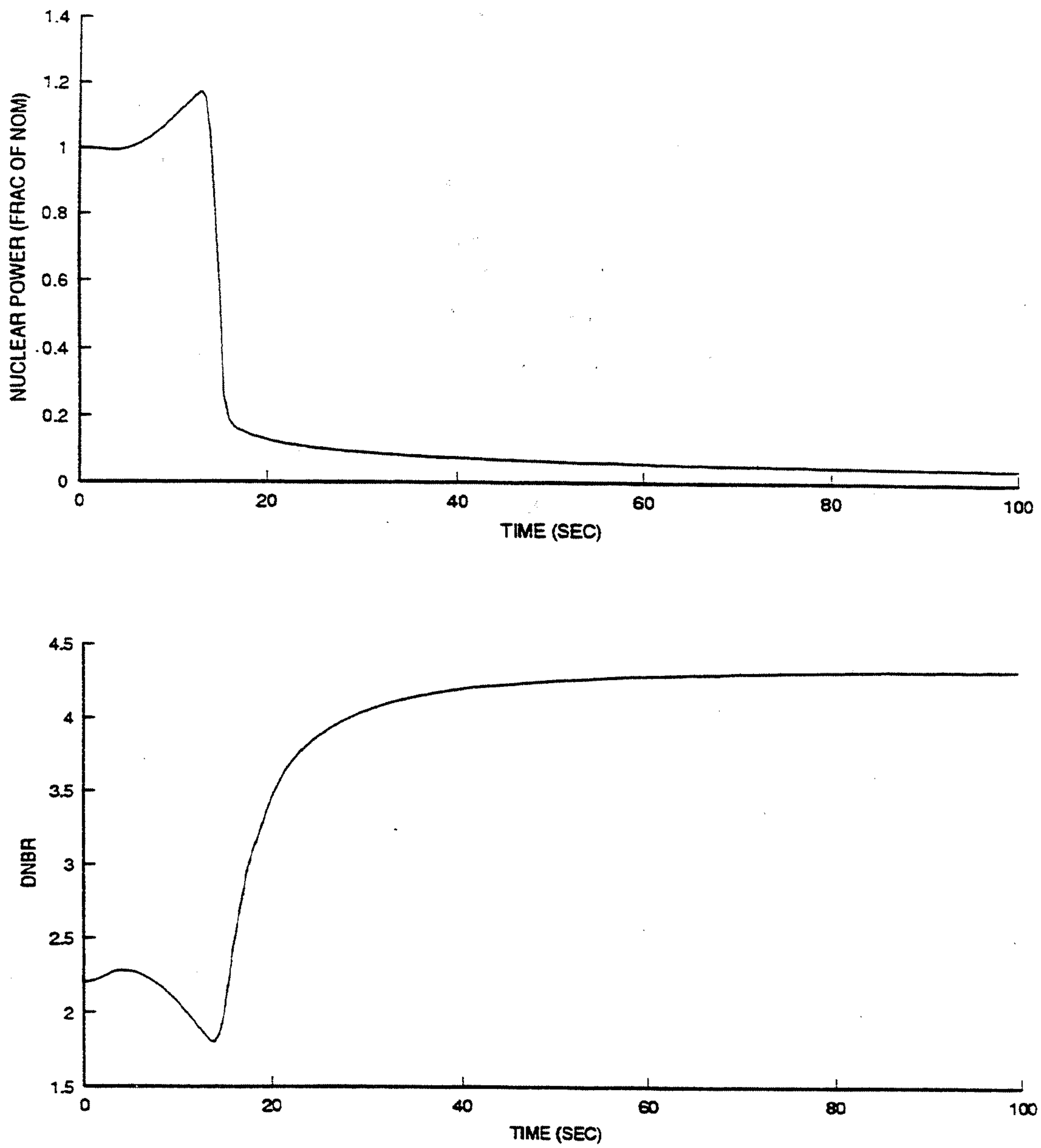
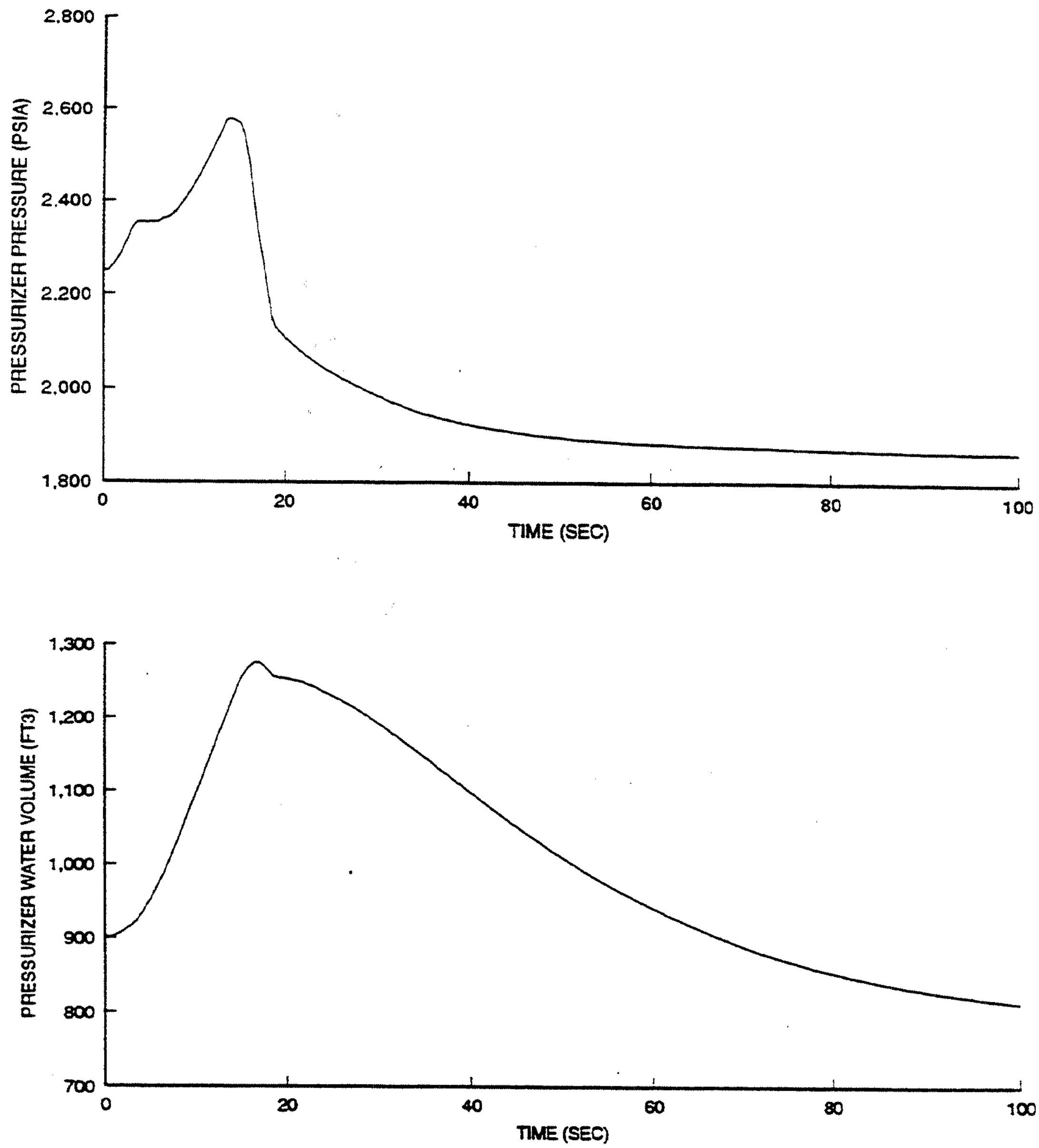
Figures 15.2-15 Through 15.2-18
Deleted by Amendment 99-01



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

LOSS OF LOAD/TURBINE TRIP WITH AUTO PRESSURE CONTROL MINIMUM FEEDBACK (BOL) NUCLEAR POWER AND DNBR VS. TIME

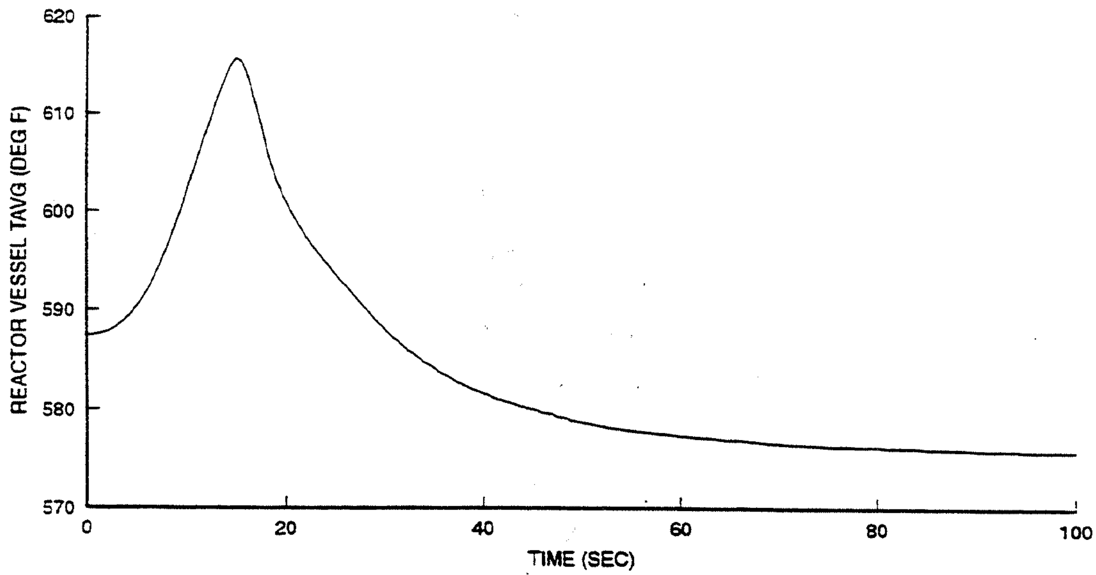
FIGURE 15.2-19

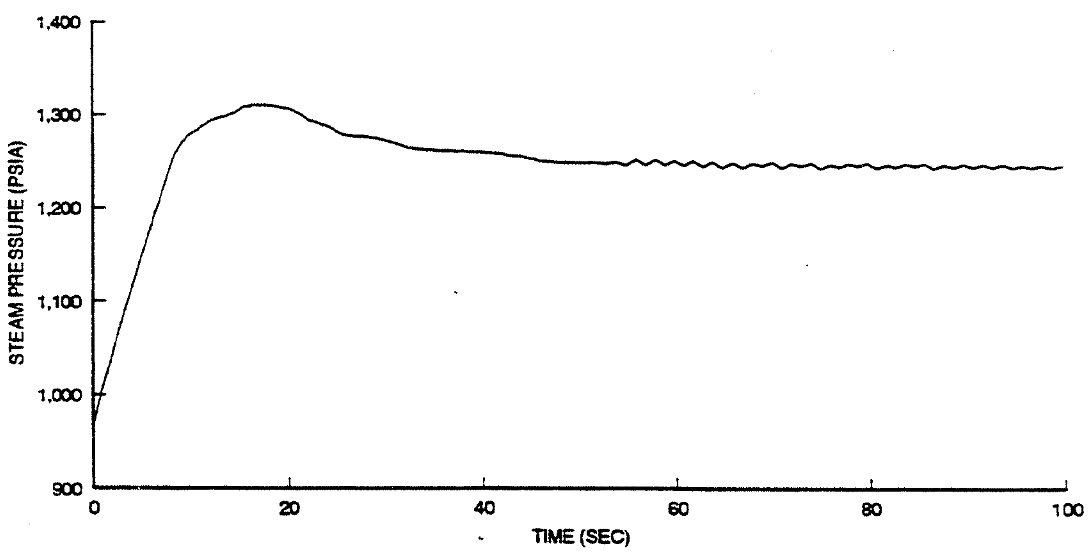


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

LOSS OF LOAD/TURBINE TRIP WITH AUTO PRESSURE CONTROL MINIMUM FEEDBACK (BOL) PRESSURIZER PRESSURE AND WATER VOLUME VS. TIME

FIGURE 15.2-20





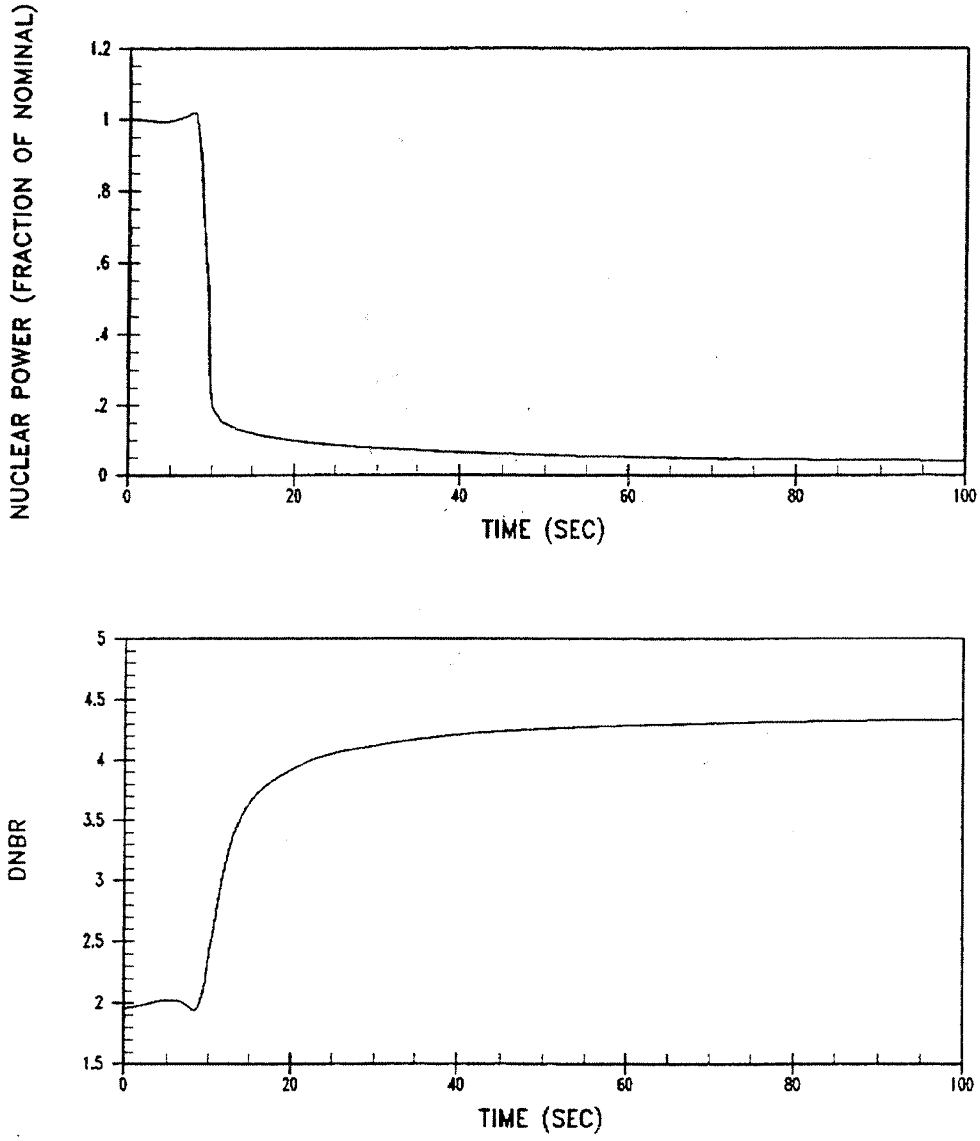
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

LOSS OF LOAD/TURBINE TRIP WITH AUTO PRESSURE CONTROL MINIMUM FEEDBACK (BOL) REACTOR VESSEL TAVG AND STEAM PRESSURE VS. TIME

FIGURE 15.2-21

Figures 15.2-22 Through 15.2-24
Deleted per RN 03-042

RN
03-042



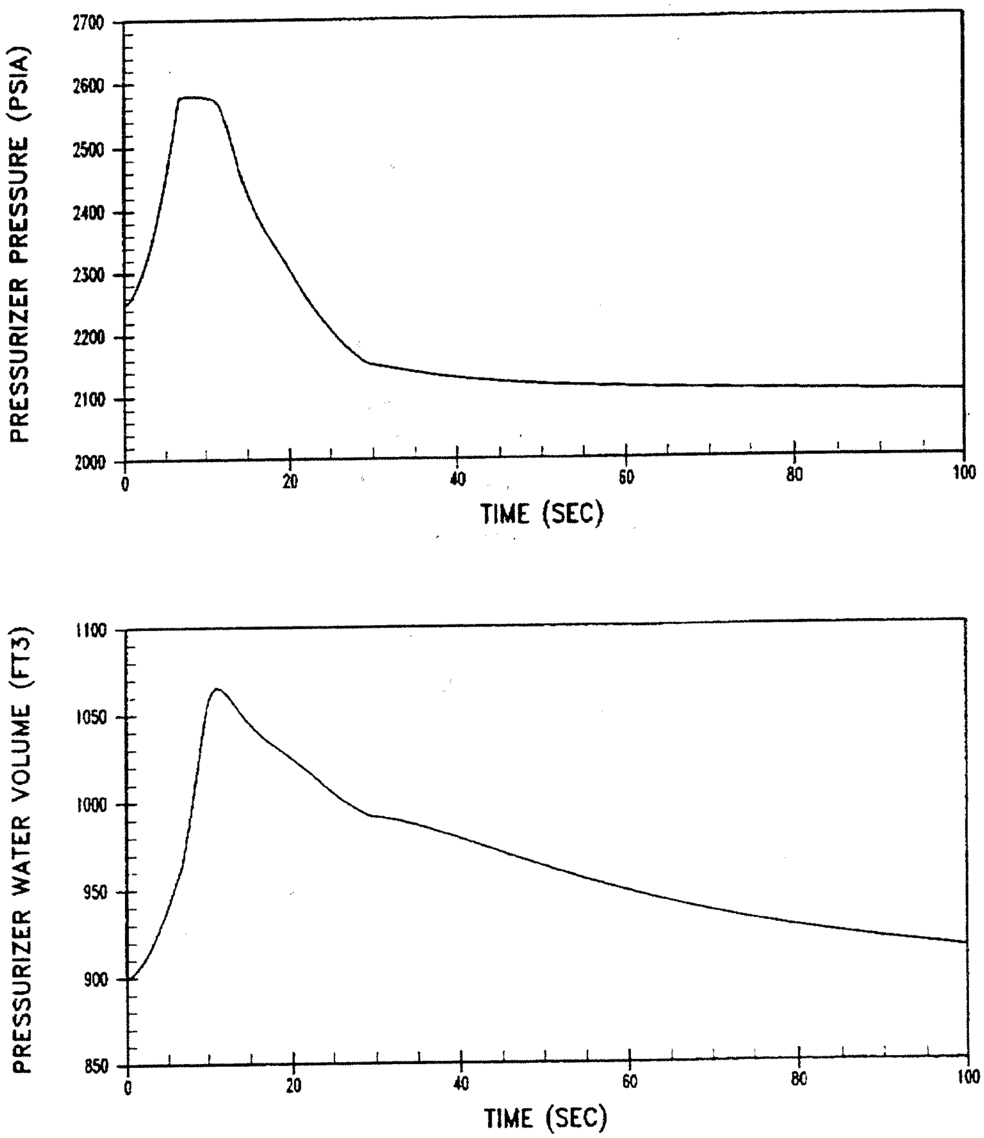
RN
03-042

RN 03-042
October 2003

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Loss Of Load/Turbine Trip With No Pressure Control
Minimum Feedback (BOL)
Nuclear Power and DNBR Vs. Time

Figure 15.2-25



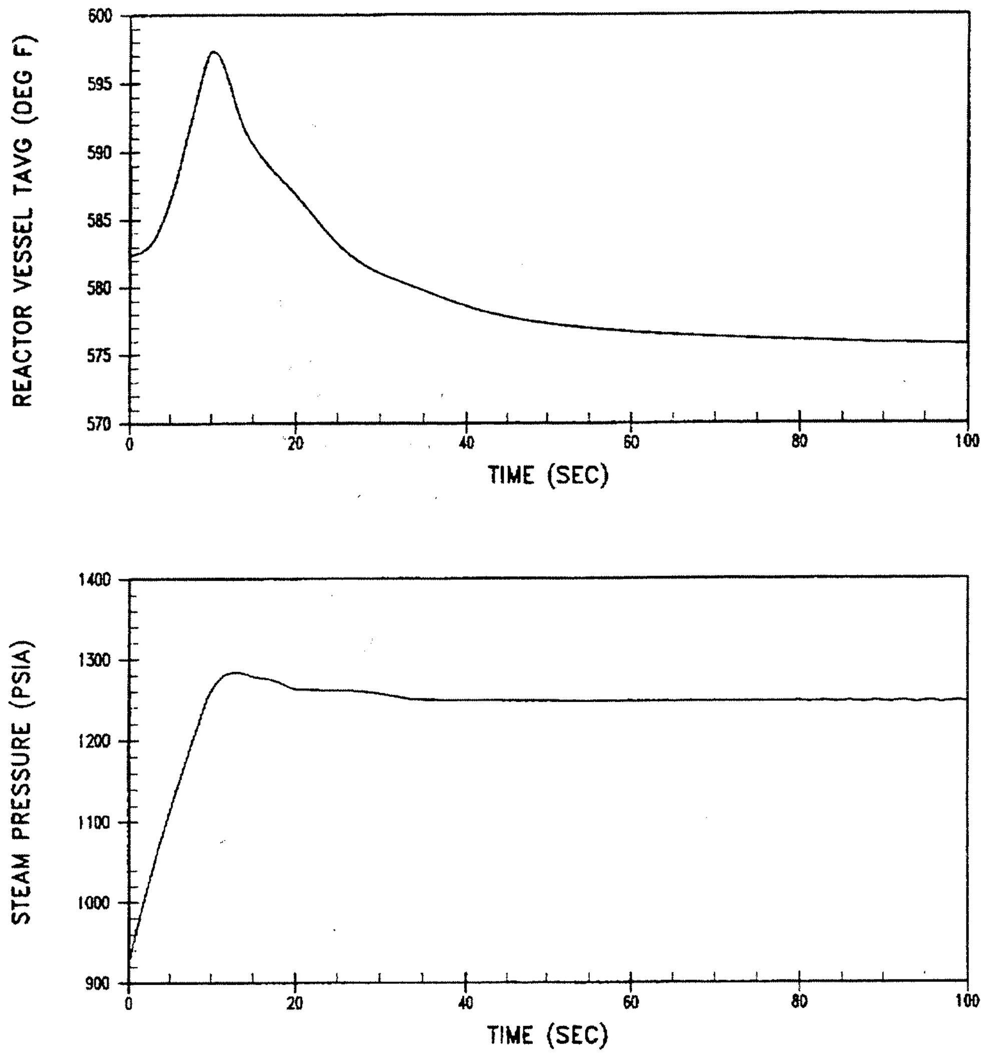
RN 03-042

RN 03-042
October 2003

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Loss Of Load/Turbine Trip With No Pressure Control
Minimum Feedback (BOL)
Pressurizer Pressure and Water Volume Vs. Time

Figure 15.2-26



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October 2003

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Loss Of Load/Turbine Trip With No Pressure Control
Minimum Feedback (BOL)
Reactor Vessel T_{AVG} And Steam Pressure Vs. Time

Figure 15.2-27

Figures 15.2-28 Through 15.2-30
Deleted per RN 03-042

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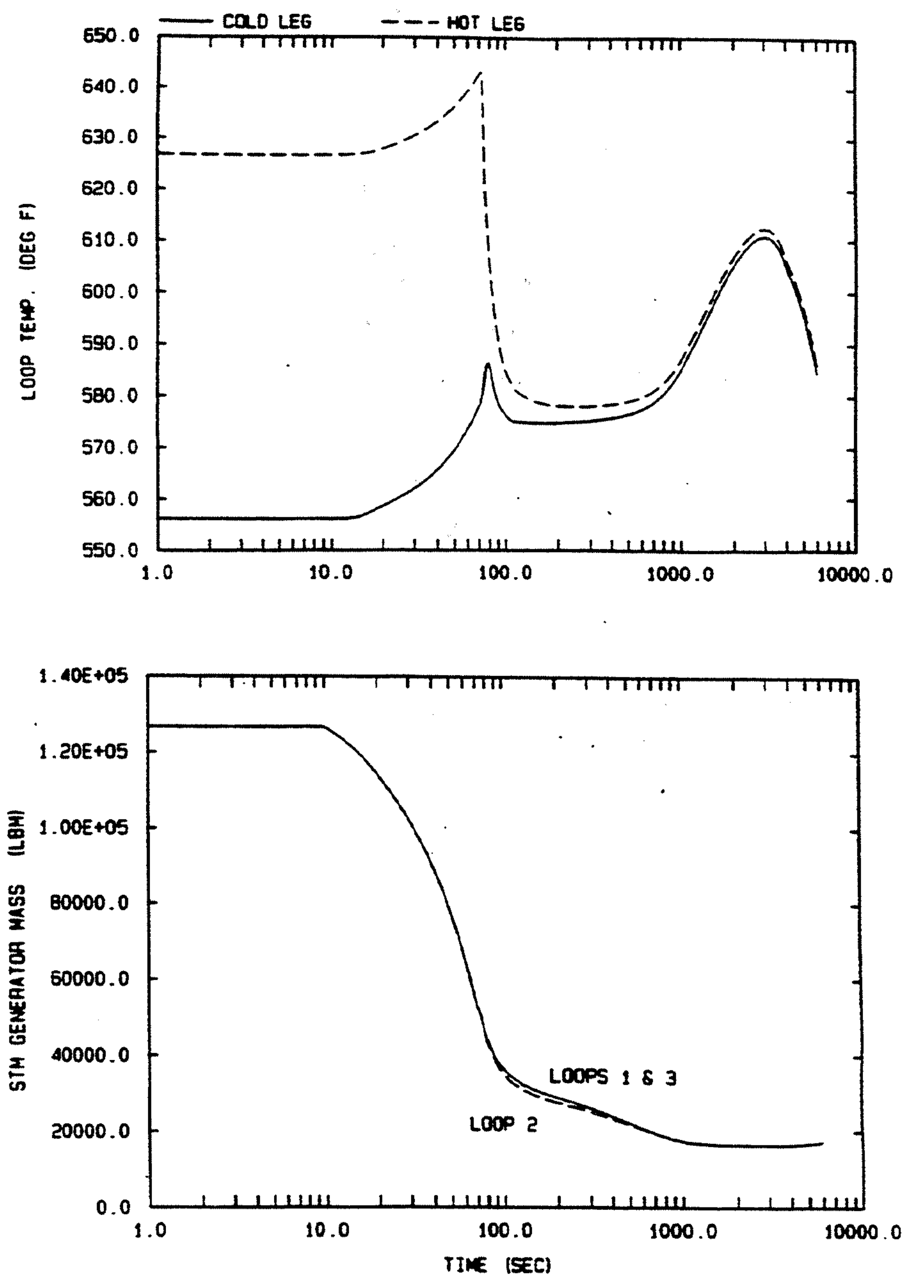
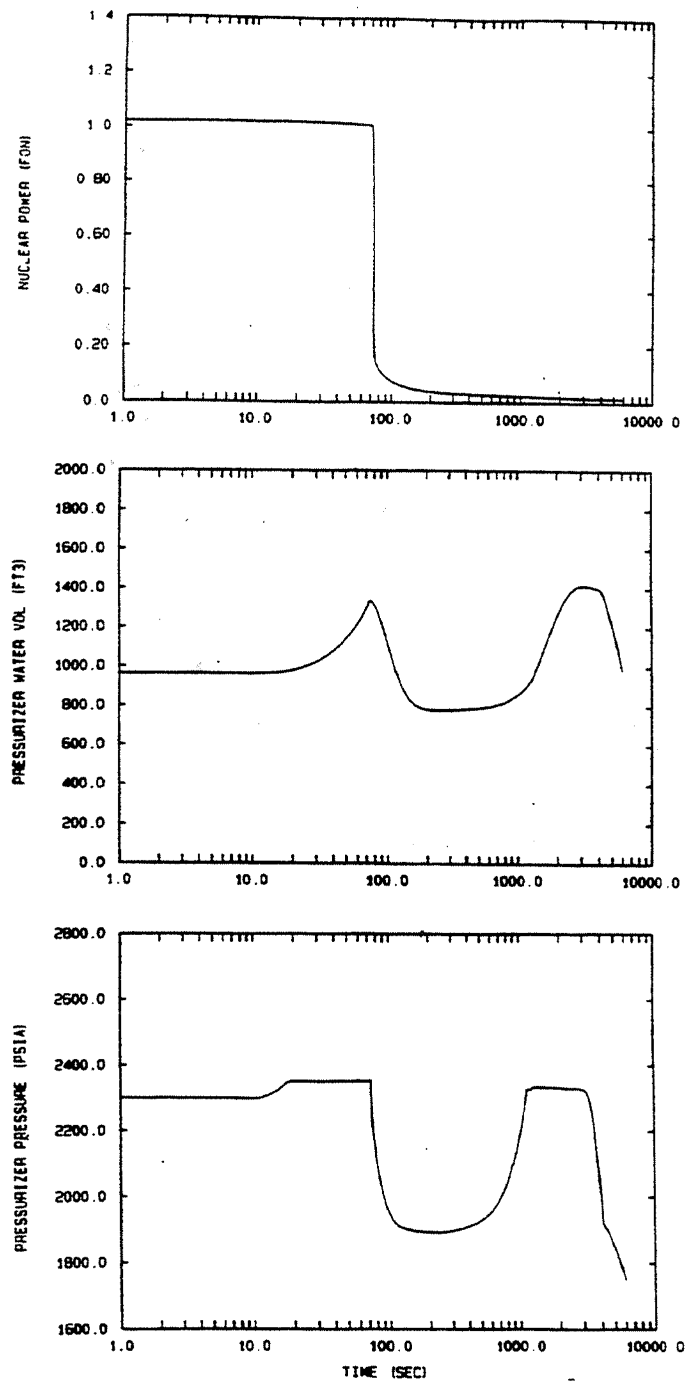
03-042



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loss of Normal Feedwater Loop Temperatures, Steam Generator Water Mass versus Time

Figure 15.2-31



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loss of Normal Feedwater Nuclear Power, Pressurizer Water Volume and Pressurizer Pressure versus Time

Figure 15.2-32

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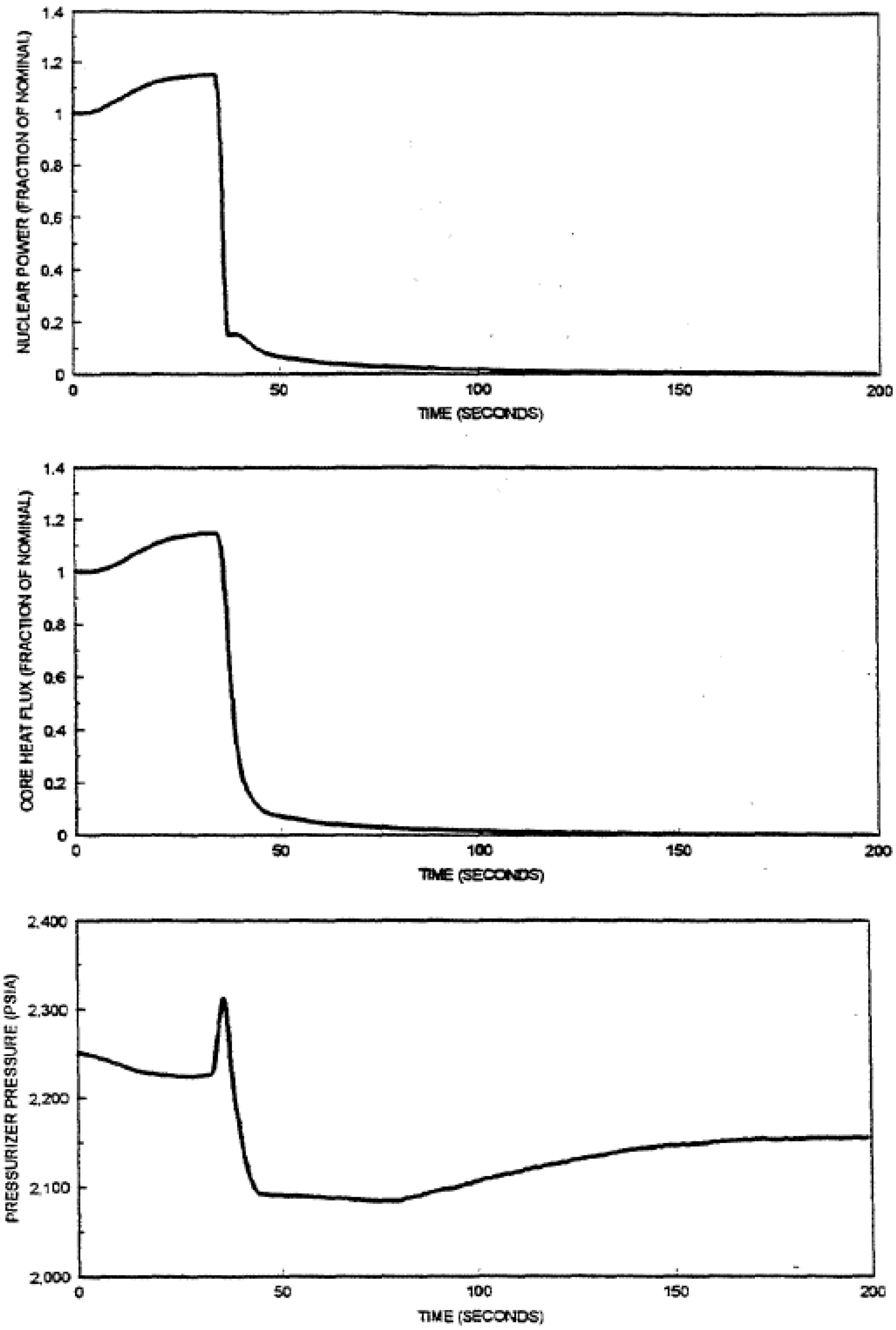
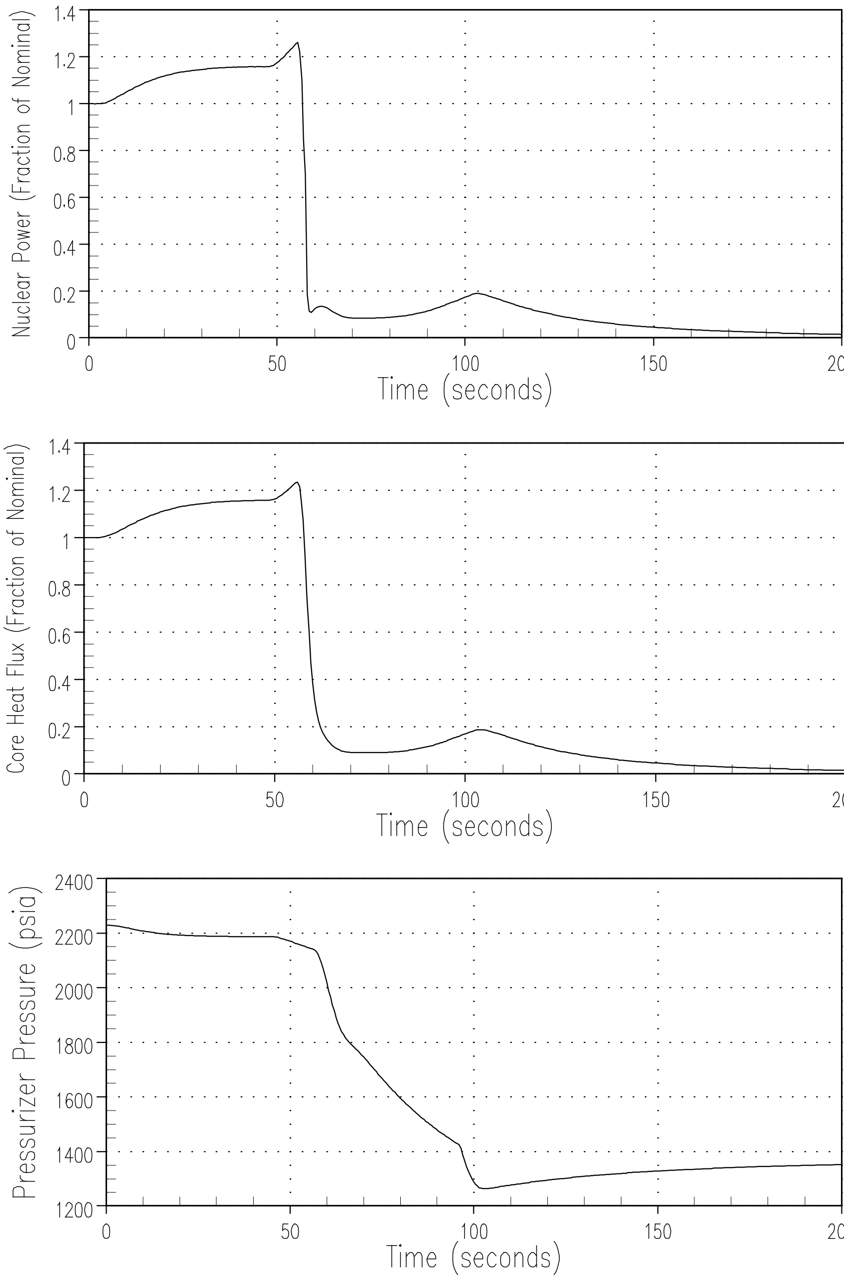


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Feedwater System Malfunction (Excessive Feedwater Addition): Nuclear Power, Core Heat Flux and Pressurizer Pressure vs. Time

Figure 15.2-33

RN 10-033

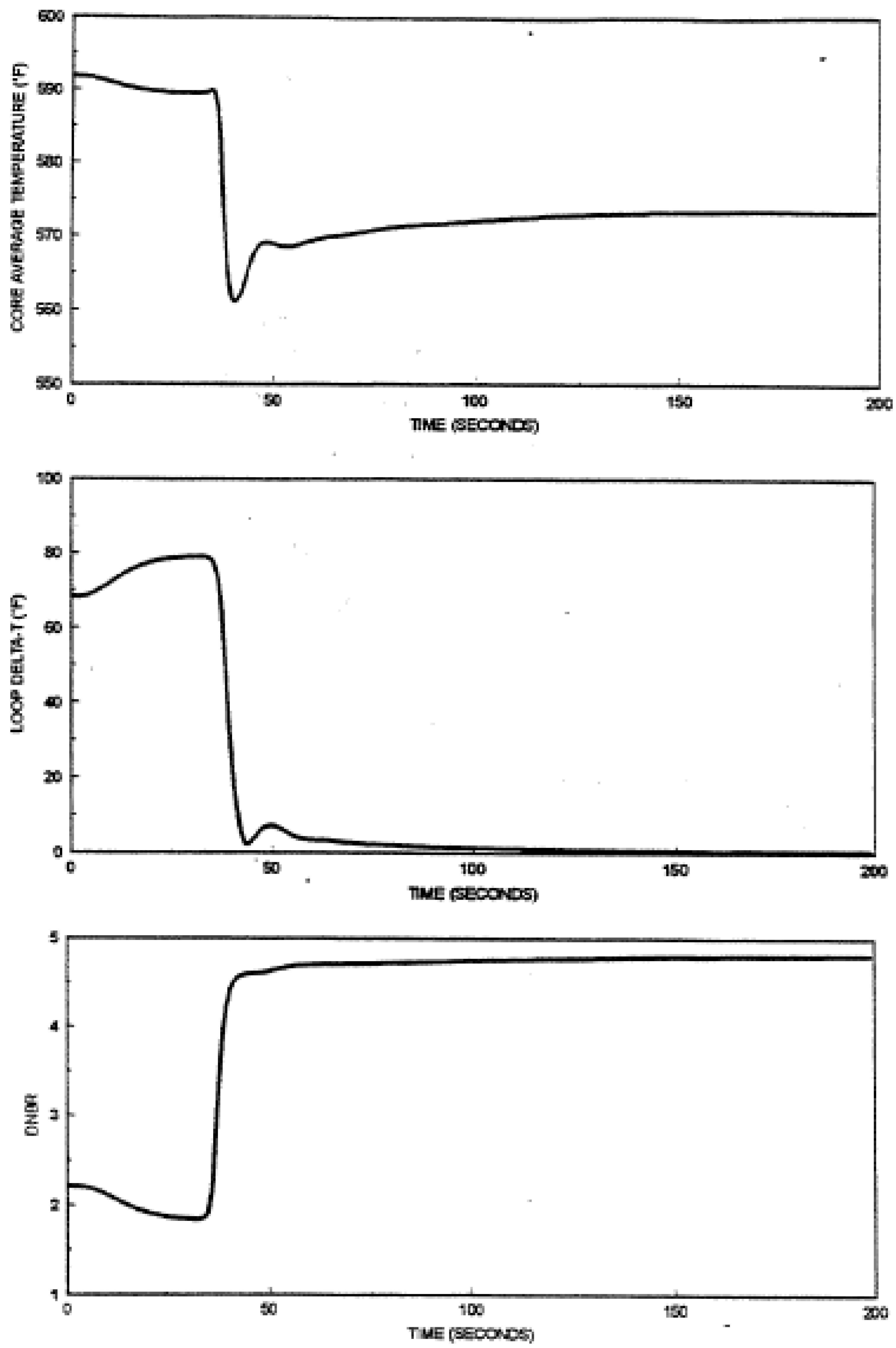


RN 10-033

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Feedwater System Malfunction (Feedwater Enthalpy Reduction): Nuclear Power, Core Heat Flux and Pressurizer Pressure vs. Time

Figure 15.2-33a

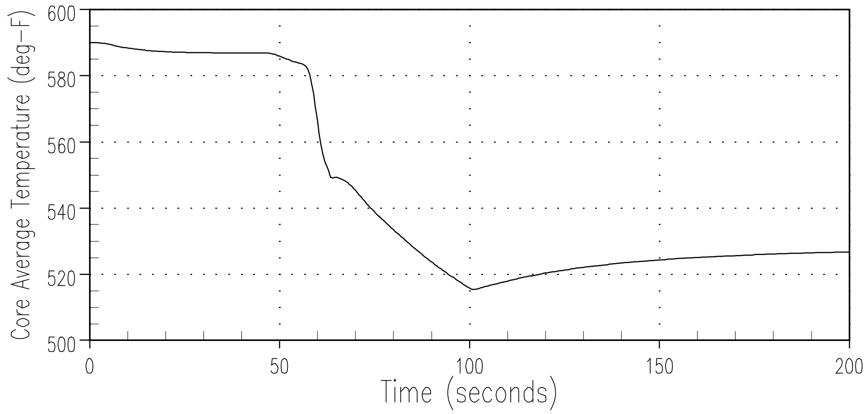


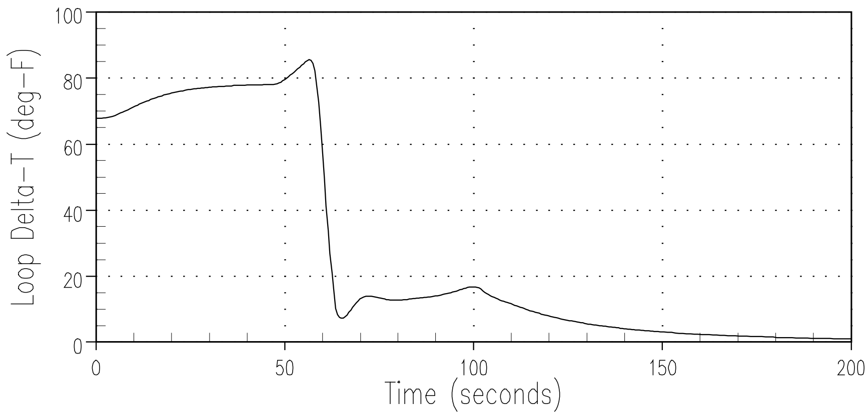
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

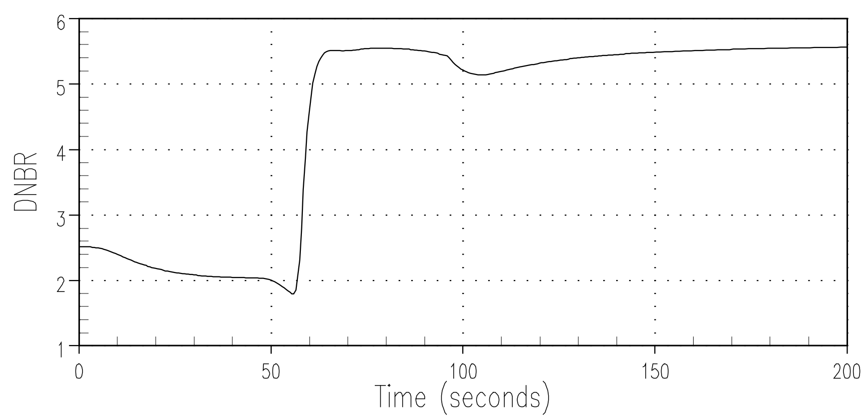
Feedwater System Malfunction (Excessive Feedwater Addition): Loop Delta-T, Core Average Temperature and DNBR vs. Time

Figure 15.2-34

RN 10-033







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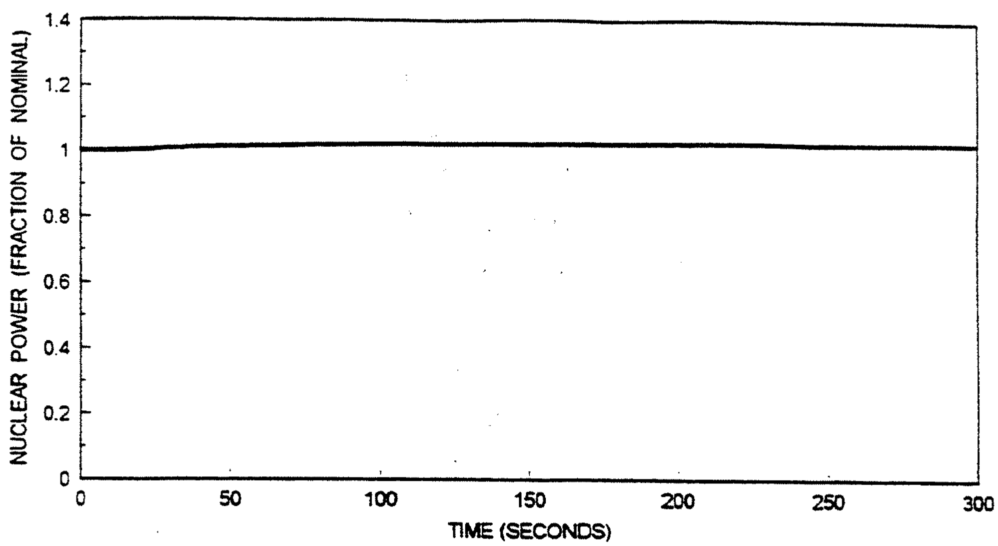
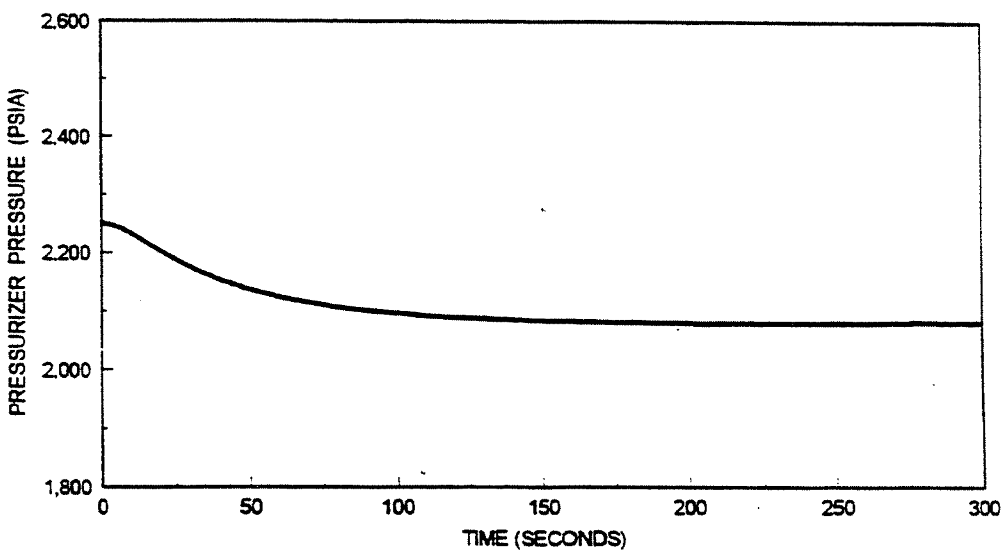
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Feedwater System Malfunction (Feedwater Enthalpy Reduction): Loop Delta-T, Core Average Temperature and DNBR vs. Time

Figure 15.2-34a





SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Excessive Load Increase w/o Control, Minimum Feedback Nuclear Power and Pressurizer Pressure vs. Time

Figure 15.2-35

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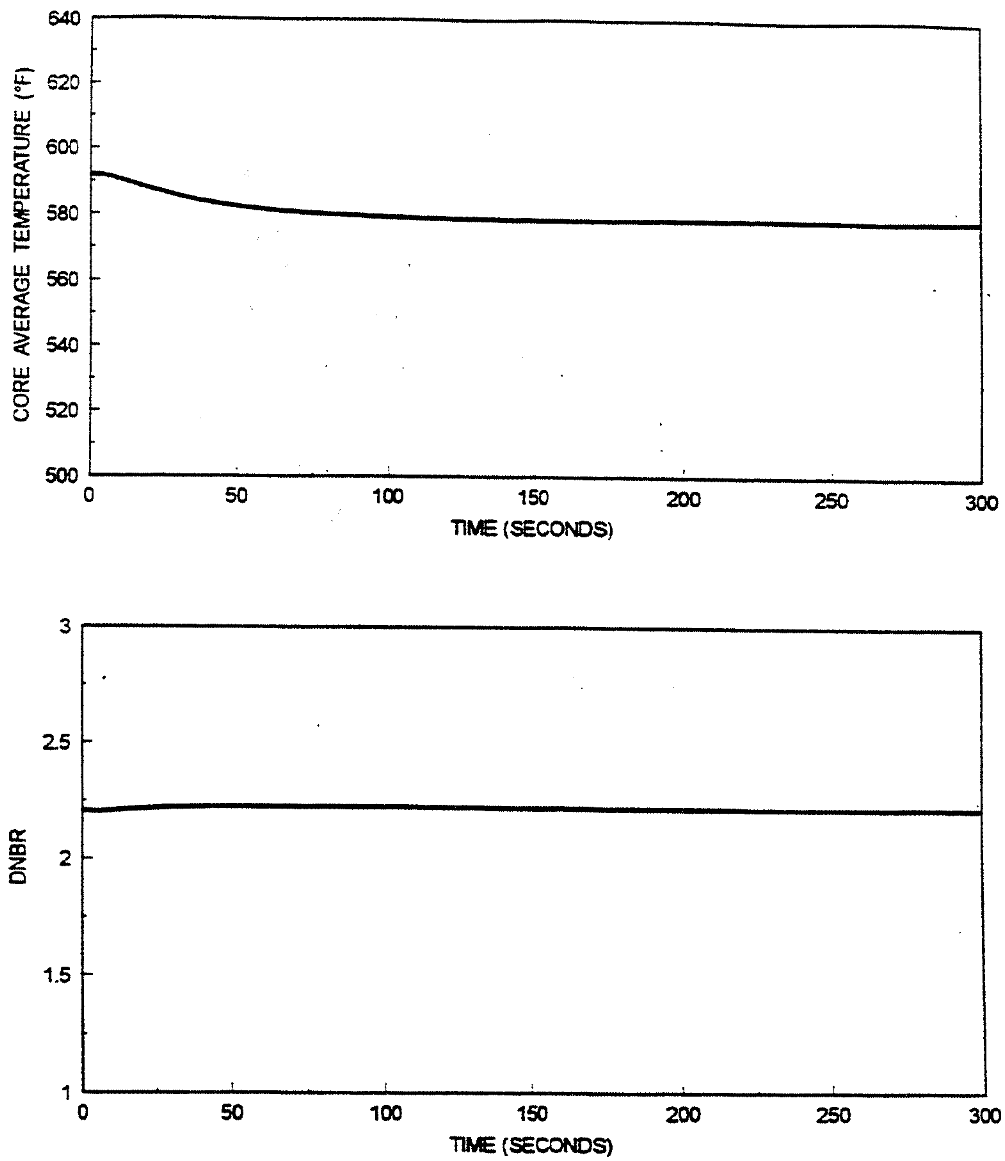

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Excessive Load Increase w/o Control, Minimum Feedback Core Average Temperature and DNBR vs. Time

Figure 15.2-36

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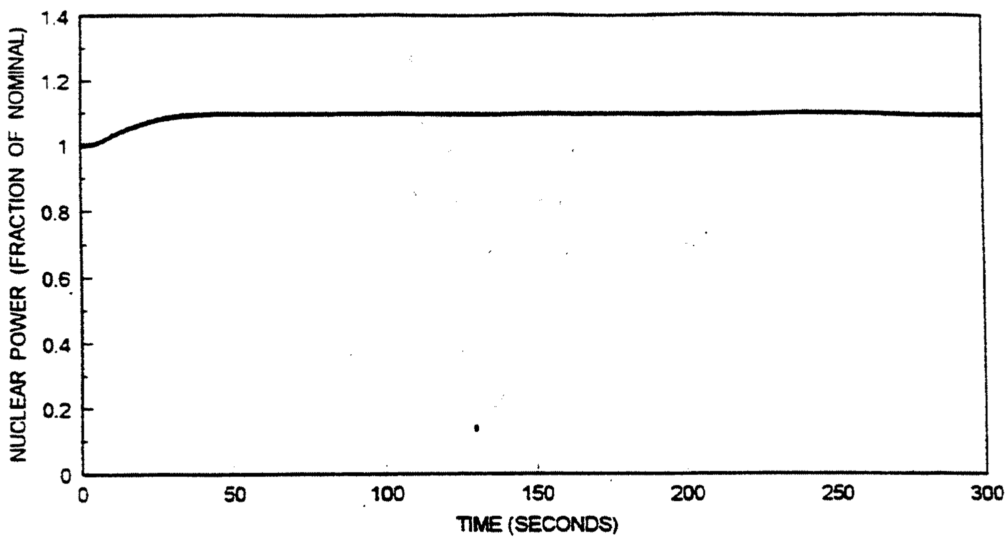
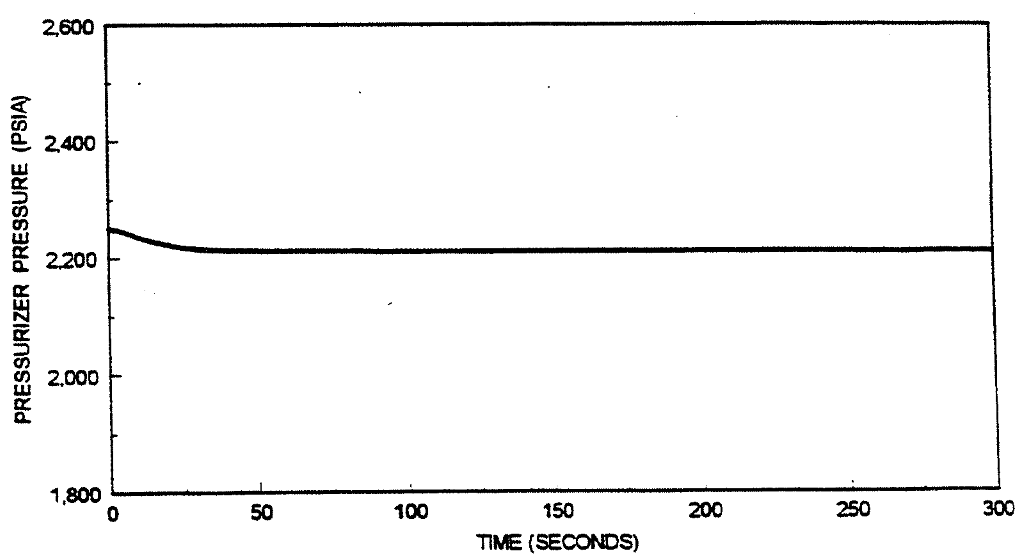
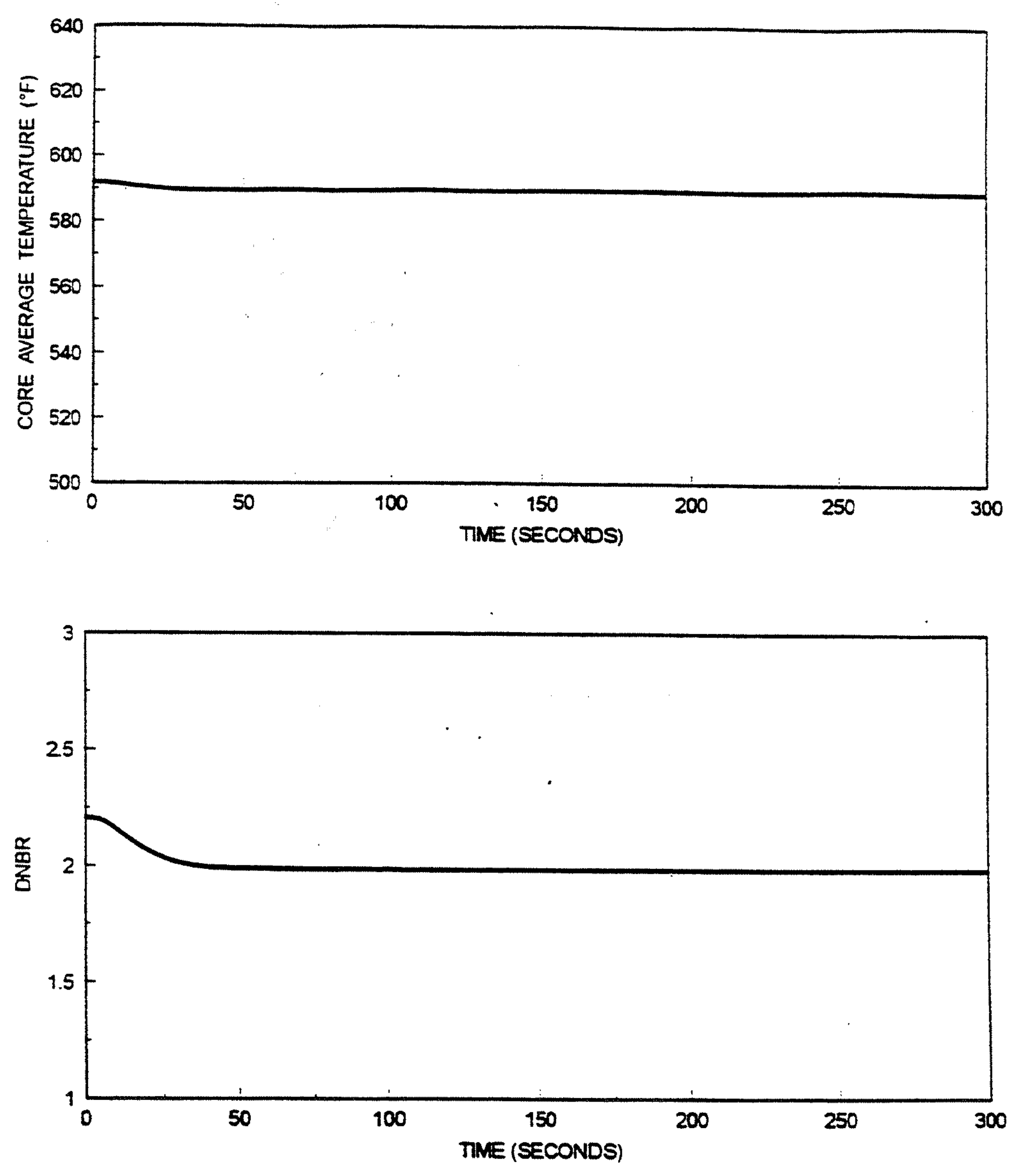




SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Excessive Load Increase w/o Control, Maximum Feedback Nuclear Power and Pressurizer Pressure vs. Time Figure 15.2-37



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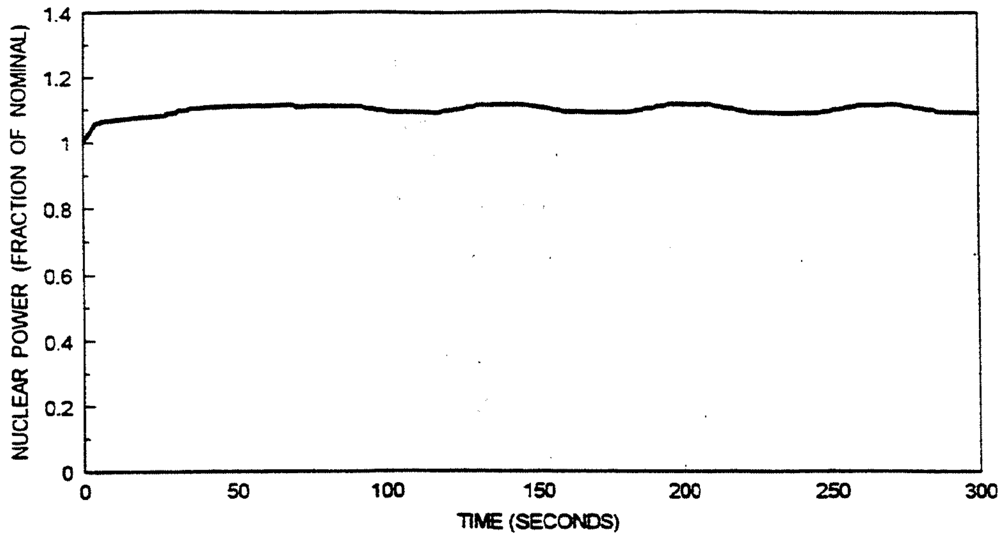
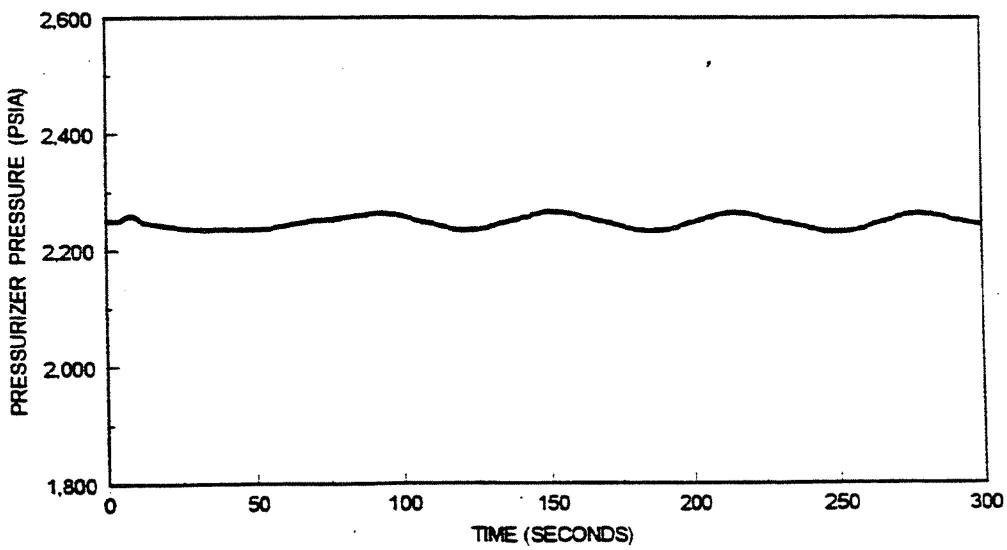
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

**Excessive Load Increase w/o Control,
Maximum Feedback Core Average
Temperature and DNBR vs. Time**

Figure 15.2-38





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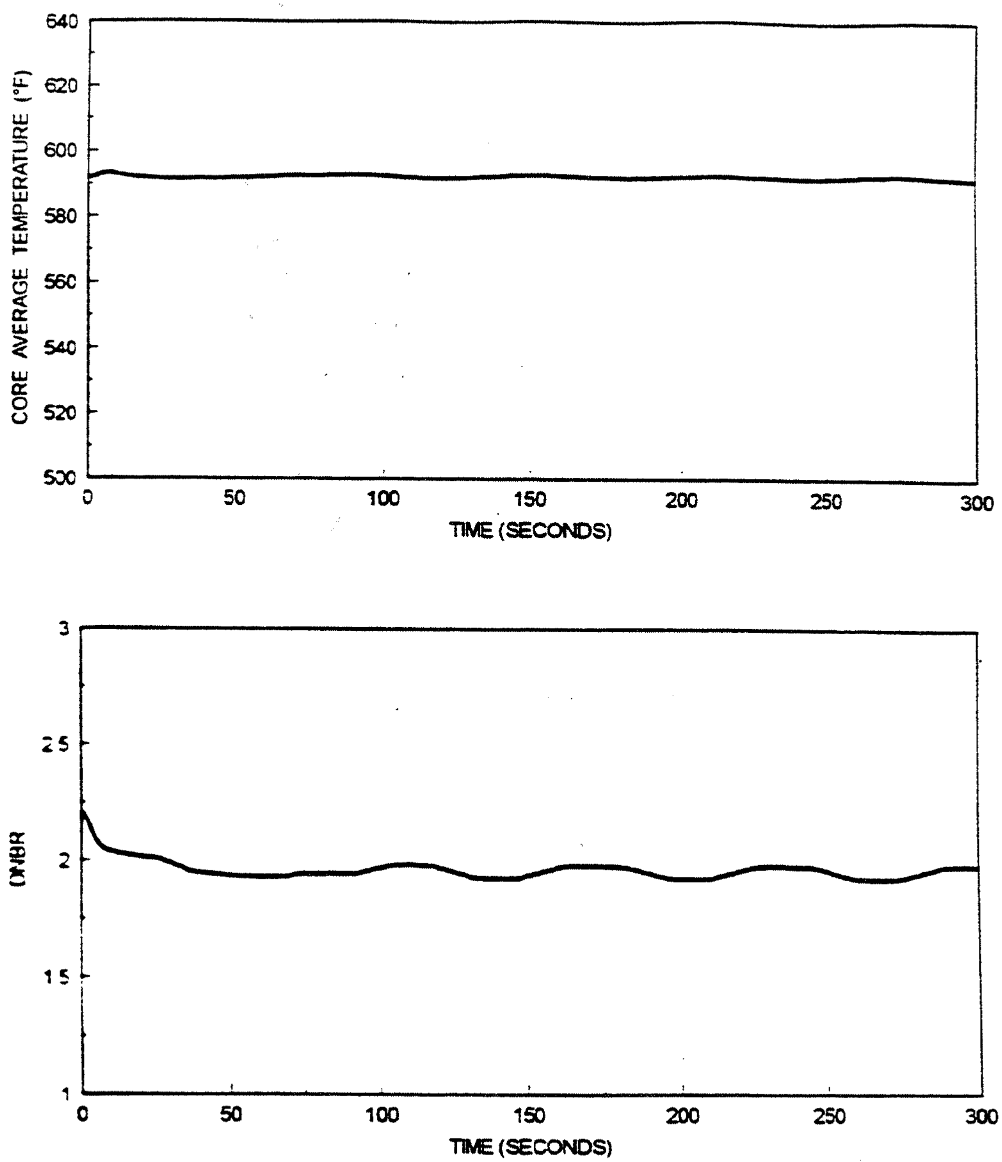
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Excessive Load Increase w/ Control, Minimum Feedback Nuclear Power and Pressurizer Pressure vs. Time **Figure 15.2-39**



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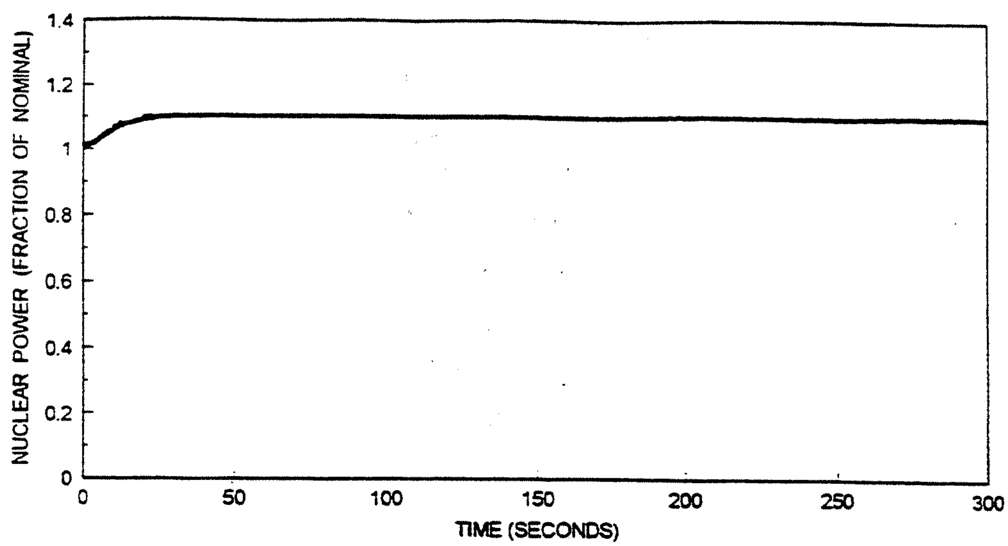
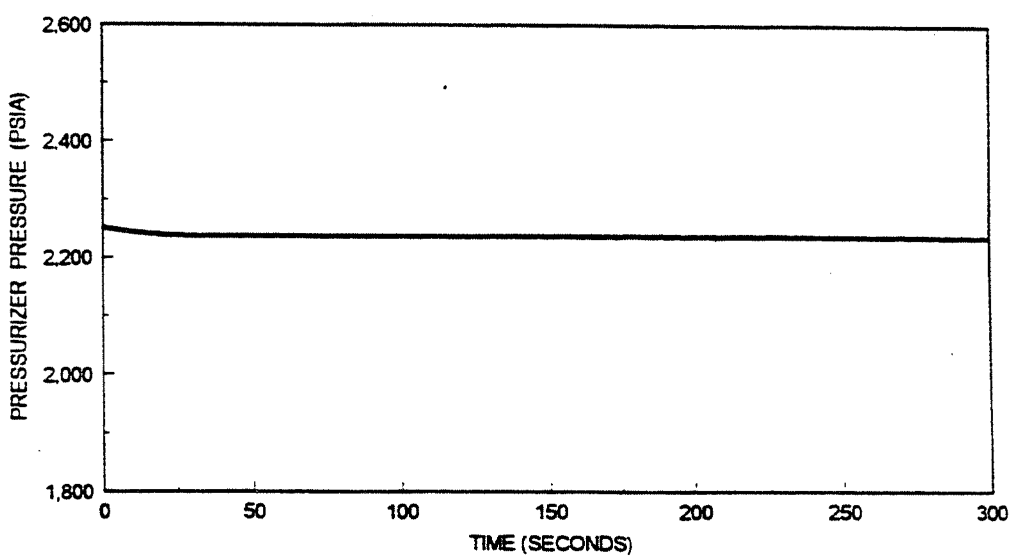
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Excessive Load Increase w/ Control, Minimum Feedback Core Average Temperature and DNBR vs. Time

Figure 15.2-40





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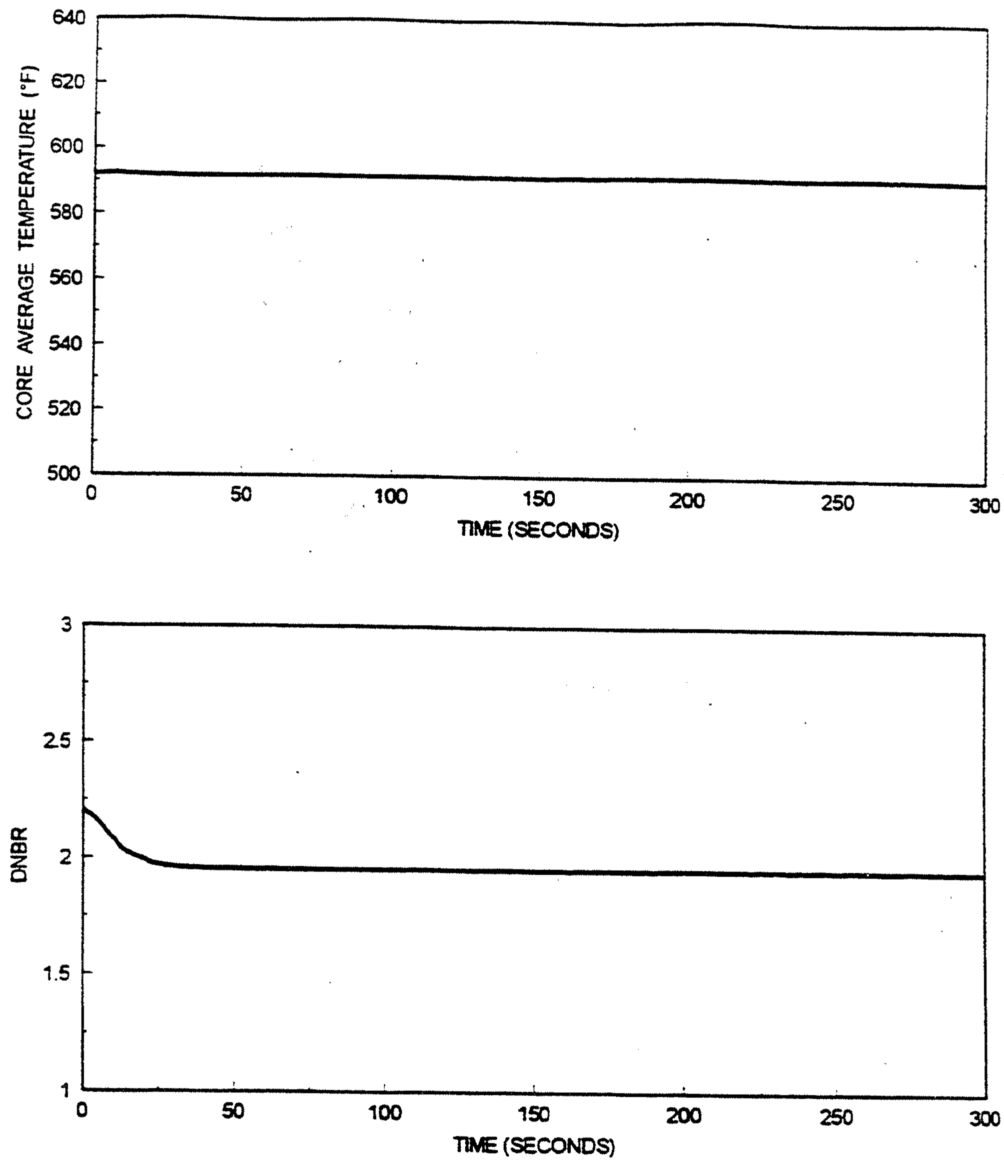
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Excessive Load Increase w/ Control, Maximum Feedback Nuclear Power and Pressurizer Pressure vs. Time

Figure 15.2-41



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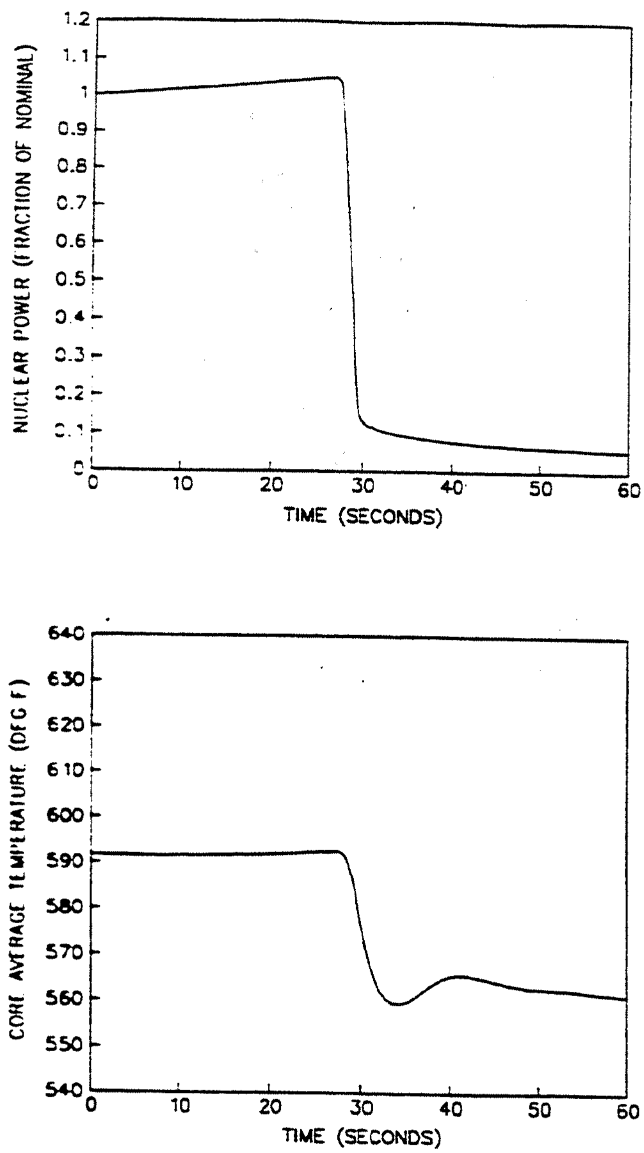
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Excessive Load Increase w/ Control, Maximum Feedback Core Average Temperature and DNBR vs. Time

Figure 15.2-42



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Accidental Depressurization of the Reactor Coolant System Nuclear Power and Core Average Temperature vs. Time

Figure 15.2-43

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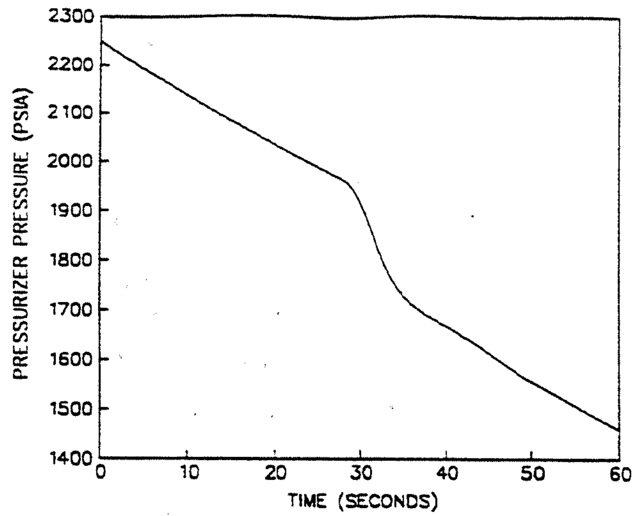
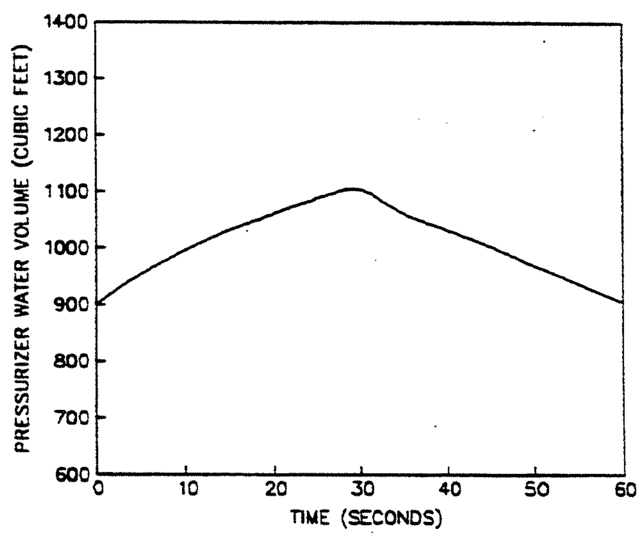


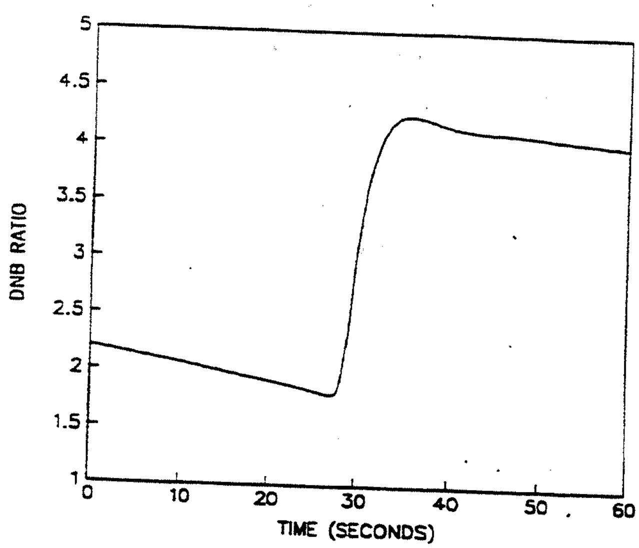


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Accidental Depressurization of the Reactor Coolant System Pressurizer Pressure and Water Volume vs. Time

Figure 15.2-44

AMENDMENT 96-02
JULY 1996



SOUTH CAROLINA ELECTRIC & GAS CO.
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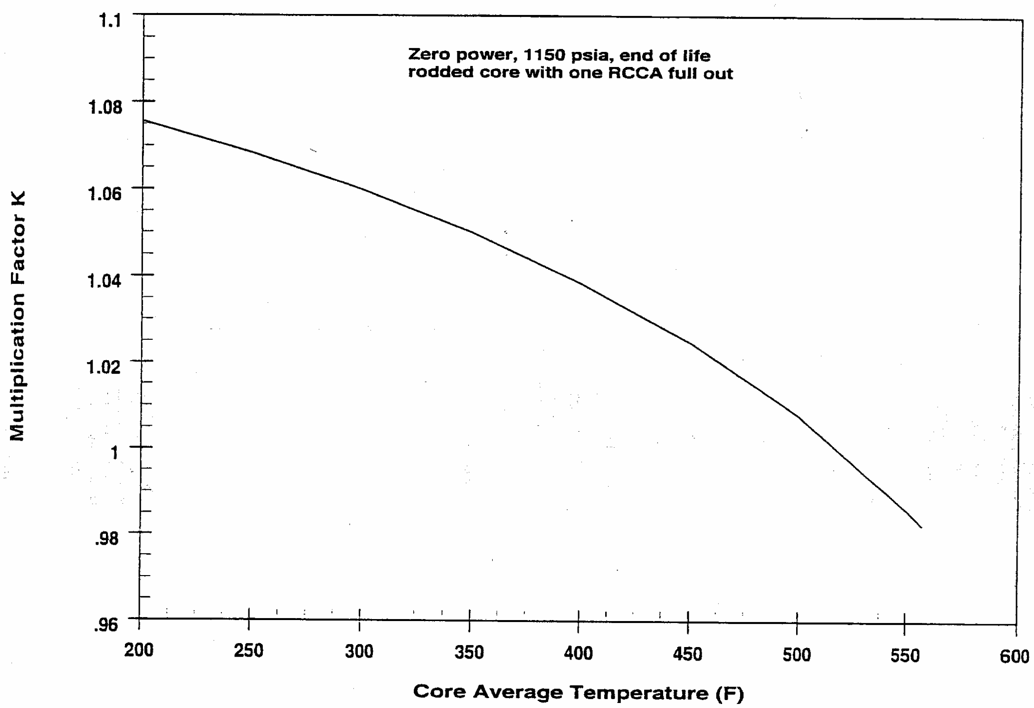
Accidental Depressurization of the Reactor Coolant System DNBR vs. Time

Figure 15.2-45

AMENDMENT 96-02
JULY 1996

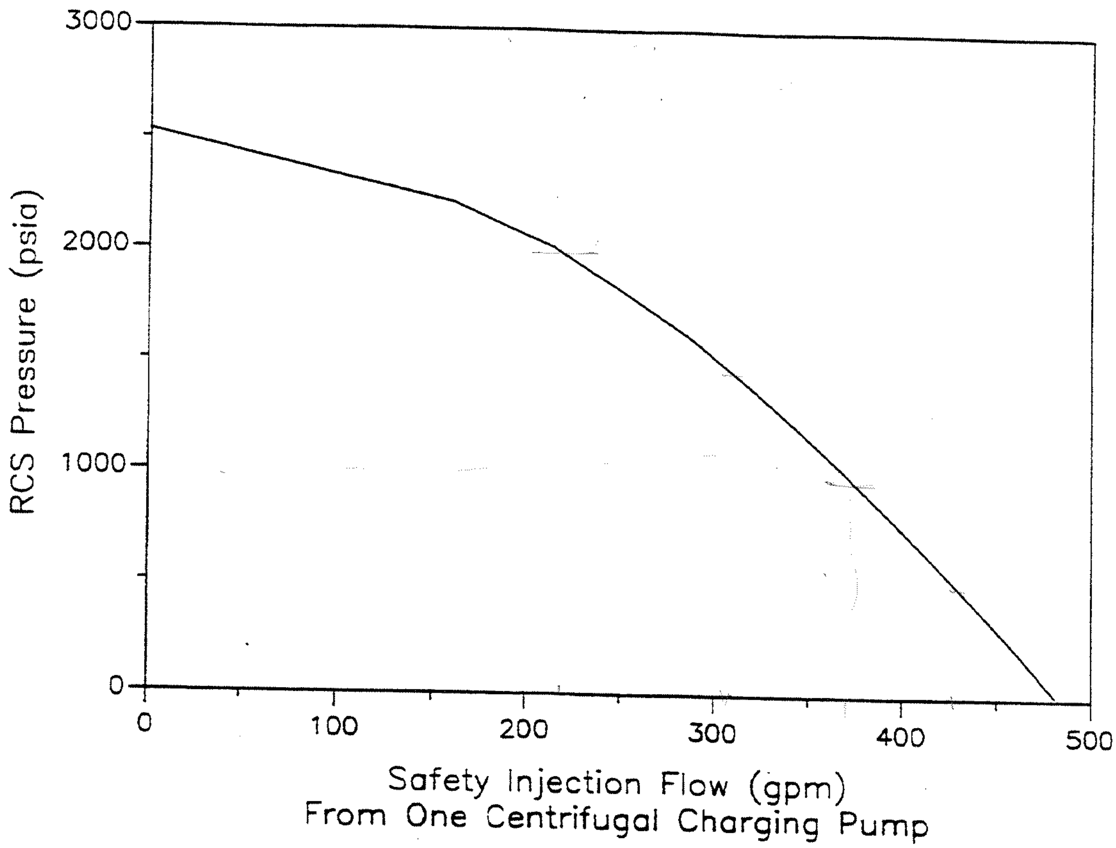


Amendment 00-01
December 2000

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam Depressurization
Variation of Keff With Core Temperature

FIGURE 15.2-46

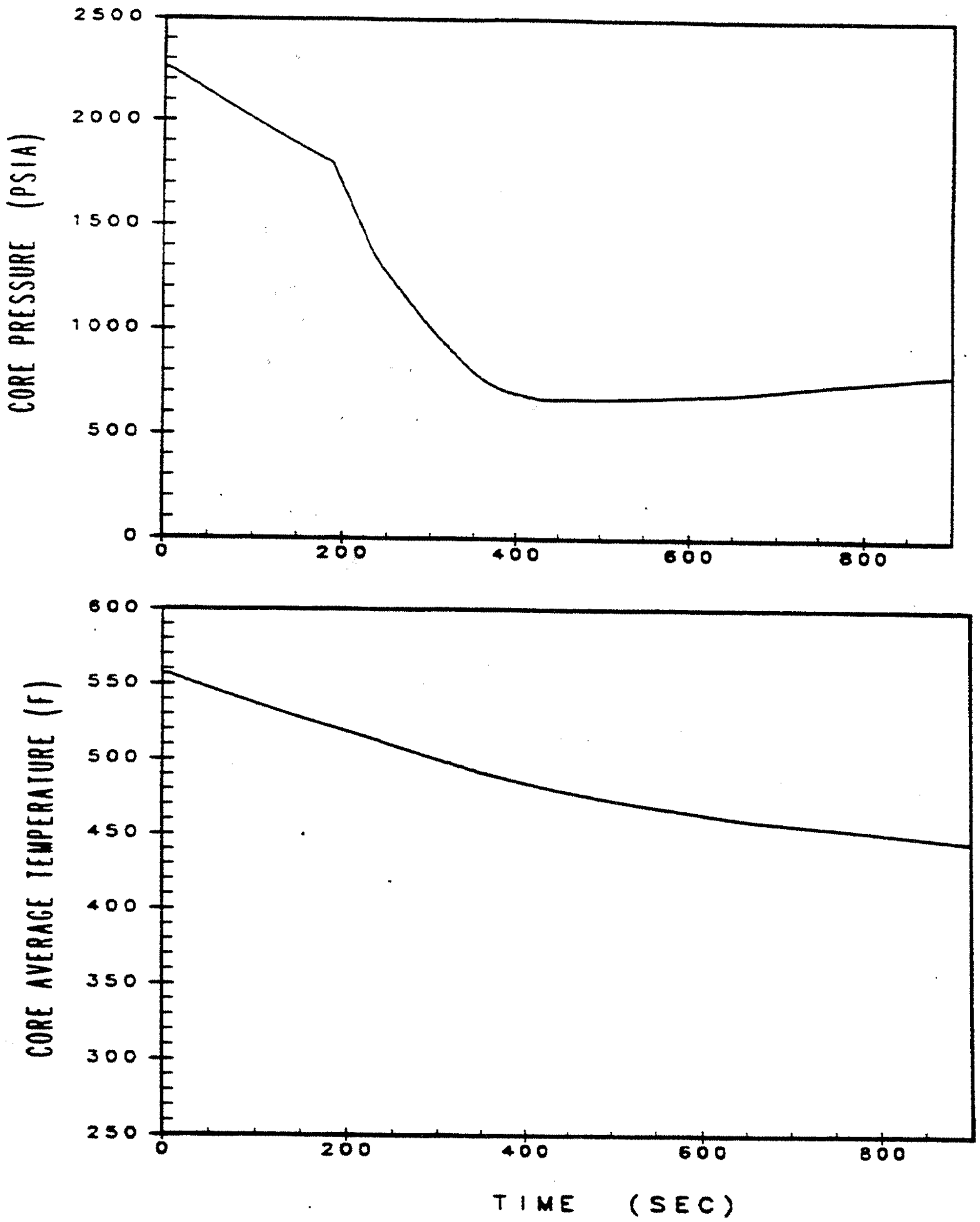


AMENDMENT 96-02
JULY 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam Depressurization Safety Injection Flowrate

Figure 15.2-47

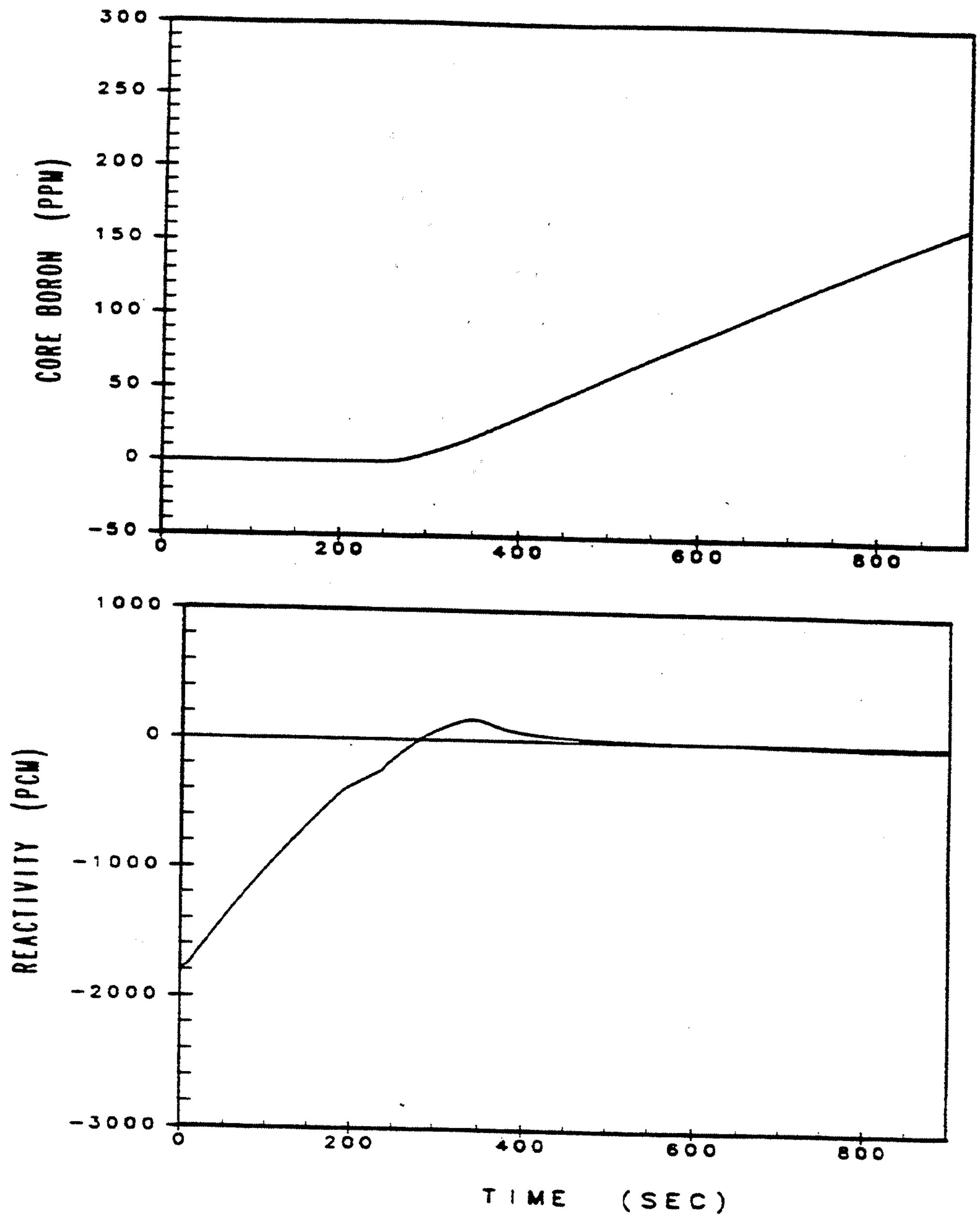


AMENDMENT 96-02
JULY 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Transient Response for a Steam Line Break Equivalent to 255 lb/sec at 1100 psia w/Offsite Power Available

Figure 15.2-48

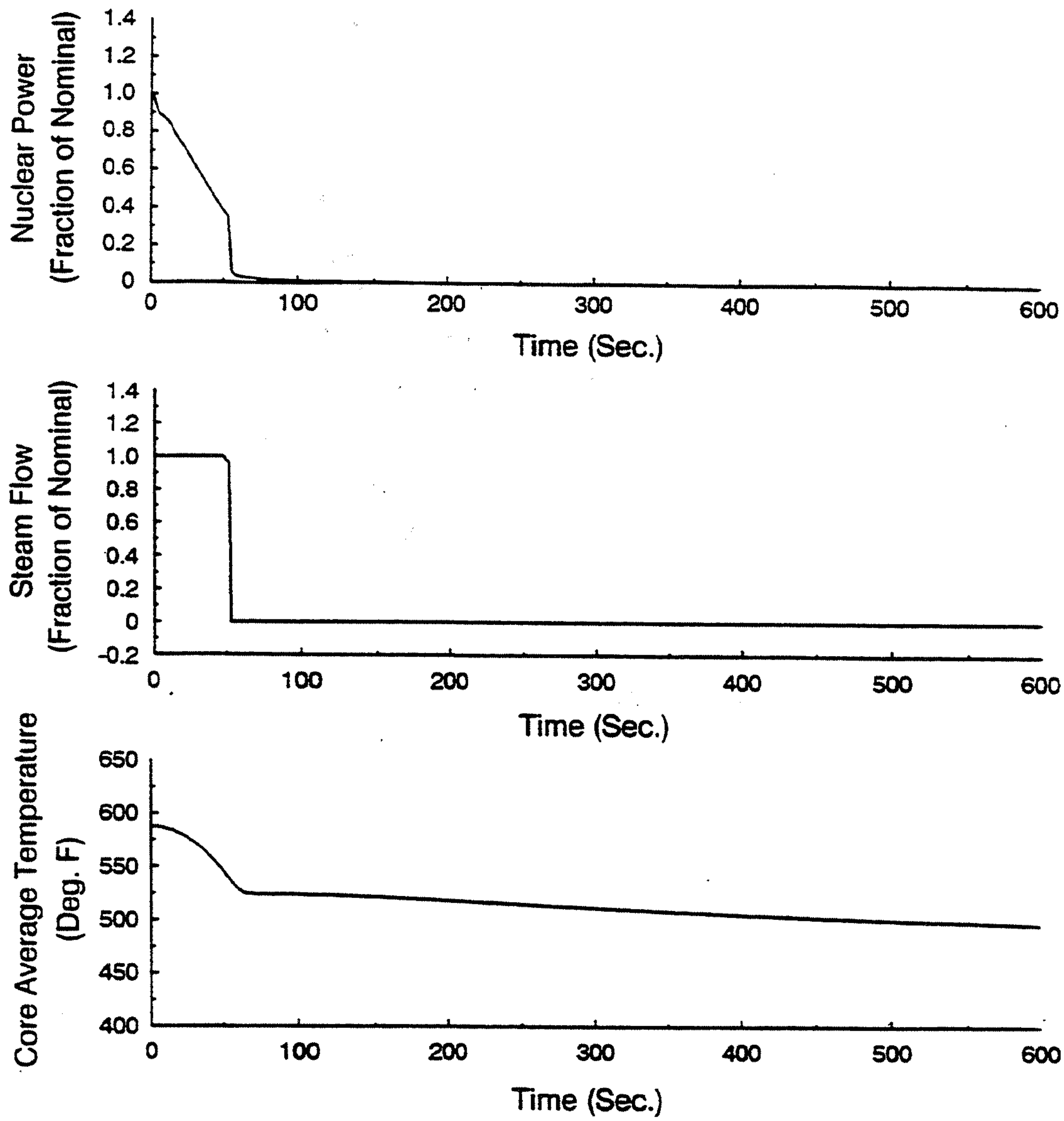


AMENDMENT 96-02
JULY 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

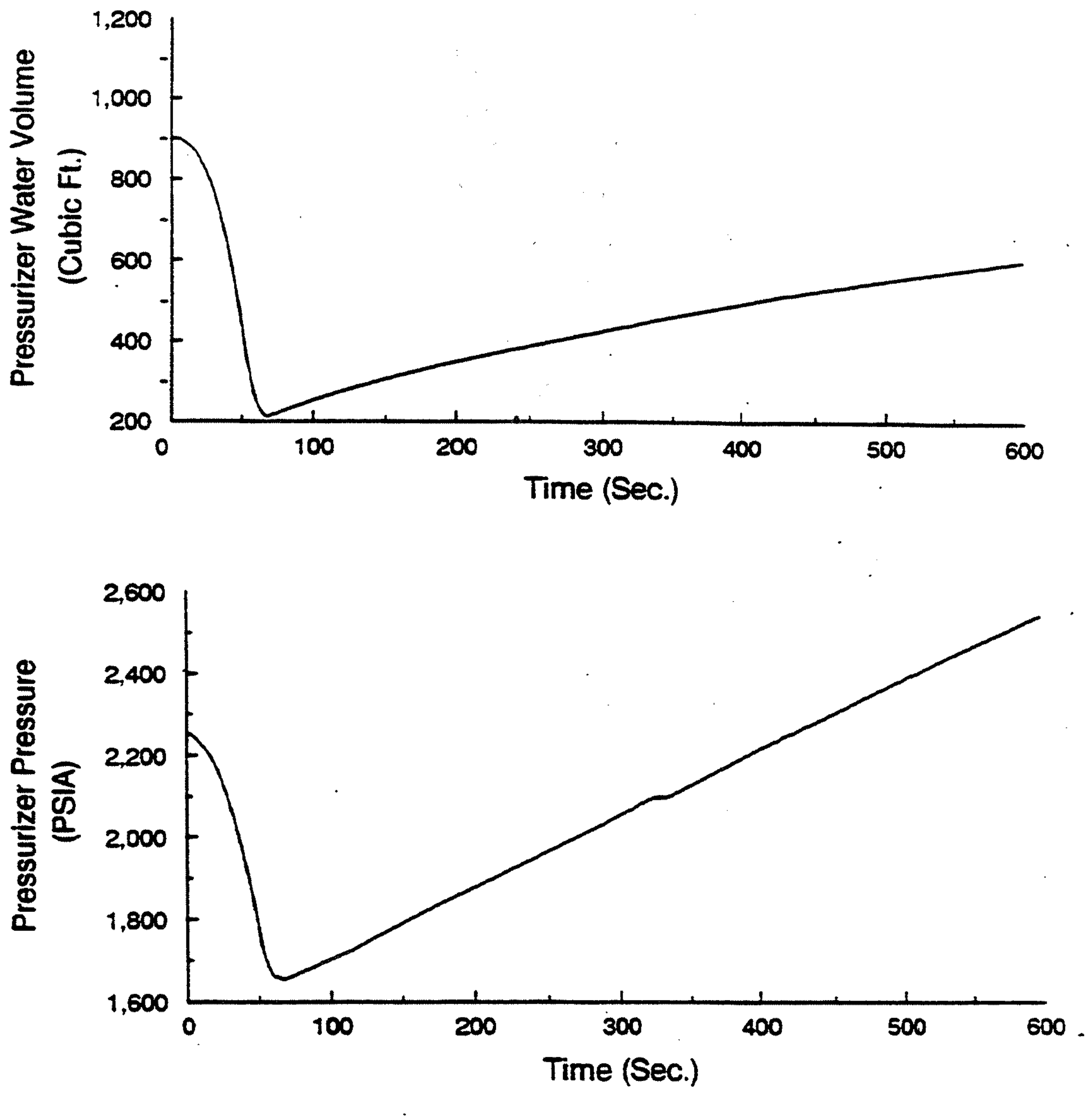
Transient Response for a Steam Line Break Equivalent To 255 lb/sec at 1100 psia w/Offsite Power Available

Figure 15.2-49



SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Spurious Actuation Of The Safety Injection System Nuclear Power, Steam Flow and Core Tavg vs. Time Figure 15.2-50

AMENDMENT 96-02
JULY 1996



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spurious Actuation Of The Safety Injection System Pressurizer Water Volume and Pressurizer Pressure vs. Time
Figure 15.2-51

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 the Emergency Core Cooling System (ECCS).

2. Minor secondary system pipe breaks.

3. Inadvertent loading of 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.

Each of these infrequent faults are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The time sequence of events during applicable Condition III faults 1 and 4 above is shown in Table 15.3-1 and Table 15.3-3.

15.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATES THE EMERGENCY CORE COOLING SYSTEM

15.3.1.1 Identification of Causes and Accident Description

A loss of coolant accident is defined as a rupture of the Reactor Coolant System piping or of any line connected to the system. See Section 5.2 for a more detailed description of the loss of reactor coolant accident boundary limits. Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer, permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing in it.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the Reactor Coolant System through the postulated break against the charging pump makeup flow at normal Reactor Coolant System pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 3/8 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec.

Should a larger break occur, depressurization of the Reactor Coolant System causes fluid to flow to the Reactor Coolant System from the pressurizer, resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The Safety Injection System is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the Reactor Coolant System. The heat transfer between the Reactor Coolant System and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the emergency feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressure. When the Reactor Coolant System depressurizes below the accumulator discharge pressure, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initiation of the accident, and effects of pump coastdown are included in the blowdown analyses.

99-01

RN 06-038

15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis

For loss-of-coolant accidents due to small breaks less than 1 square foot, the NOTRUMP computer code[13,14,28] is used to calculate the transient depressurization of the Reactor Coolant System as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features. Among these are calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. Also, safety injection into the broken loop is modeled using the COSI condensation model[28]. The NOTRUMP small-break LOCA ECCS evaluation model was developed to determine the Reactor Coolant System response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants." RN 06-038

The Reactor Coolant System model is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, while the two intact loops are lumped into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components; which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. Detailed descriptions of the NOTRUMP code and the evaluation model are provided in References [13], [14] and [28]. RN 06-038

Safety injection systems consist of gas pressurized accumulator tanks and pumped injection systems. Minimum ECCS availability is assumed for the analysis. Assumed pumped safety injection characteristics as a function of Reactor Coolant System pressure used as boundary conditions in the analysis are shown in Figure 15.3-1a for break sizes less than 6 inches and Figure 15.3-1b for the 6 inch break case. For the break sizes less than 6 inches the broken loop safety injection flow shown in Figure 15.3-1a is assumed to spill to RCS pressure and for the 6 inch break case the broken loop safety injection flow shown in Figure 15.3-1b is assumed to spill to the containment back pressure of 0 psig. The injection rate is based upon the pump performance curves, but degraded for conservatism and to account for possible reduced injection rates due to pump cooling recirculation miniflow operation. The safety injection was assumed to be delivering to the RCS 27 seconds after the generation of the injection signal as indicated in Table 15.3-1. This delay is assumed to account for diesel generator startup and emergency power bus loading in case of a loss of offsite power coincident with an accident. RN 06-038

Peak clad temperature calculations are performed with the LOCTA-IV[1] code using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow, and mixture height as boundary conditions. Figure 15.3-2 depicts the hot rod axial power shape used to perform the small break analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation in the uncovered elevations. Figure 15.3-3 presents the normalized core power curve as a function of time after reactor trip. The scram delay times denoted in Table 15.3-1 reflect the assumption that the core is assumed to continue to operate at full rated power until the control rods are completely inserted.

15.3.1.2.2 Results

This section presents the results of the limiting break size analysis as determined by the highest peak fuel rod clad temperature for a range of break sizes. The limiting break size at beginning-of-life (BOL) conditions was found to be a 2.75-inch diameter cold leg break. A burnup study was performed to determine the hot rod peak cladding temperature (PCT) and maximum local oxidation at the limiting time-in-life for the 2.75-inch break. The maximum temperature attained during the transient was 1952° F at 12,000 MWD/MTU burnup. The maximum transient local oxidation was 14.34% at 14,000 MWD/MTU burnup. Important calculation results and input parameters are summarized in Table 15.3-2a and 15.3-2b respectively, while key transient event times are listed in Table 15.3-1. Figures 15.3-4 through 15.3-9 show for the limiting 2.75-inch break transient, respectively: RN 06-038

- Reactor Coolant System Pressure
- Core mixture height
- Clad temperature transient at Peak Clad Temperature elevation
- Steam mass flow rate out top of core
- Clad surface heat transfer coefficient at Peak Clad Temperature elevation
- Fluid temperature at Peak Clad Temperature elevation

During the initial period of the small-break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor recirculation cooling pumps as they coast down. Normal upward flow is maintained through the core and core heat is adequately removed. At the low heat generation rates following shutdown the fuel rods continue to be well cooled as long as the core is covered by a two-phase mixture level. From the core mixture level and clad temperature transients for the 2.75-inch break calculation shown in Figures 15.3-5 and 15.3-6, it is seen that the peak clad temperature occurs near the time at which the core is most deeply uncovered when the top of the core is steam cooled. This time is also accompanied by the highest vapor superheating above the mixture level. RN 06-038

RN 06-038

15.3.1.2.3 Additional Break Sizes 00-01

Studies documented in References 9 and 10 determined that the limiting small-break size occurred for breaks less than 10 inches in diameter. To insure that the worst possible small break size has been identified, calculations were performed for a spectrum of breaks (2.0, 2.25, 2.5, 3.0, 3.25, 4 and 6 inches) in addition to the 2.75-inch break. The results of these calculations are shown in the Sequence of Events Table 15.3-1, and the Results Table 15.3-2. Plots of the following parameters are shown in Figures 15.3-10a through 15.3-16c for the 2.0, 2.25, 2.5, 3.0, 3.25, 4 and 6 inch break sizes: RN 06-038

- Reactor Coolant System Pressure
- Core mixture height
- Clad temperature transient at Peak Clad Temperature elevation

15.3.1.2.4 Additional Analysis

NUREG-0737[11], Section II.K.3.31, required plant-specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC Generic Letter 83-85[12], generic analyses using NOTRUMP[13,14] were performed and are presented in WCAP-11145[15]. Those results demonstrate that in a comparison of cold leg, hot leg, and pump suction leg break locations, the cold leg break location is limiting.

Analyses of a LOCA in the pressurizer vapor space such as that caused by opening a pressurizer relief valve or a safety valve were provided in WCAP-9600[10]. The conclusion presented in WCAP-9600 is that these breaks are not limiting since little or no core uncovery will take place. WCAP-9600 states that the analyses reported therein apply to all Westinghouse designed plants. RN 06-038

RCS loop-to-loop flow imbalance of 5% was considered, to incorporate the effects of RCS Flow Asymmetry, per Westinghouse Nuclear Safety Advisory Letter (NSAL) 00-008. The results show that the effect of 5% Loop Flow Asymmetry is inconsequential for LOCA PCT calculations. RN 01-125

15.3.1.2.5 Impact of ECCS Evaluation Model Changes

The October 17, 1988, revision to 10CFR50.46 requires applicants and holders of operating licenses or construction permits to notify the NRC of errors and changes in the ECCS Evaluation Models, which are not significant, on an annual basis. Reference [18] defines a significant error or change as one which results in a calculated peak fuel cladding temperature (PCT) different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is an accumulation of changes and errors such that the sum of the absolute temperature change is greater than 50°F. The current ECCS evaluation model changes for Virgil C. Summer Nuclear Station that affect the small break LOCA PCT and their effect are identified in the latest 10CFR50.46 annual or 30-day report. 02-01

15.3.1.2.6 Supplemental Calculations to Support HHSI Throttle Valve Replacement

SBLOCA calculations were performed to determine the effect of increased high head safety injection (HHSI) flow at the Virgil C. Summer Nuclear Station (VCSNS) resulting from the replacement of the HHSI throttle valves. The increased safety injection (SI) flow as a function of Reactor Coolant System pressure used in the analysis is shown in Figure 15.3-17. The broken loop SI flow shown in the figure is assumed to spill to RCS pressure for the 2.75 inch and 3 inch break sizes analyzed for these supplemental calculations. To insure that the worst possible small break size has been identified, these calculations considered the 2.75-inch break size as well as the 2.5-inch, 3.0-inch and 3.25-inch breaks. Based on the results of the NOTRUMP runs performed for the above break sizes, it was determined that LOCTA-IV calculations were necessary only for the 2.75-inch and 3.0-inch break cases. Therefore, results of these two cases are presented below.

The results at beginning-of-life (BOL) conditions showed less than one degree peak cladding temperature difference between the 2.75-inch and 3.0-inch diameter cold leg breaks. Therefore, burnup studies were performed for both cases to determine the hot rod peak cladding temperature (PCT) and maximum local oxidation at the limiting time-in-life. The maximum temperature attained during the transient was 1775°F at 15,000 MWD/MTU burnup for the 3.0-inch break case. The maximum transient local oxidation was 6.92% at 17,500 MWD/MTU burnup for the 2.75-inch break case. Important calculation results and input parameters are summarized in Table 15.3-2c and 15.3-2d respectively, while key transient event times are listed in Table 15.3-1a. Plots of the following parameters are shown in Figures 15.3-18a through 15.3-18f for the limiting PCT 3.0-inch break transient at 15,000 MWD/MTU: RN 06-019

- Reactor Coolant System Pressure
- Core mixture height
- Clad temperature at Peak Clad Temperature elevation
- Steam mass flow rate out top of core
- Clad surface heat transfer coefficient at Peak Clad Temperature elevation
- Fluid temperature at Peak Clad Temperature elevation

Figures 15.3-19a through 15.3-19c show the following results for the limiting PCT 2.75-inch break transient at 17,000 MWD/MTU:

- Reactor Coolant System Pressure
- Core mixture height
- Clad temperature at Peak Clad Temperature elevation

15.3.1.2.7 Supplemental Calculations to Support the Upflow-Conversion

SBLOCA calculations using the NOTRUMP-EM (described in Section 15.3.1.2.1) were performed to determine the effect of converting the barrel/baffle region from a downflow configuration to an upflow configuration at VCSNS. To ensure that the worst possible small break size has been identified, these calculations considered quarter inch break sizes between 2.25-inches and 3.25-inches. Based on the NOTRUMP runs performed (results summarized In Table 15.3-1b), beginning-of-life (BOL) fuel rod heat up calculations were performed for the 2.50-inch, 2.75-inch, 3.00-inch, and 3.25-inch break sizes. From the BOL results, it was clear that the 2.50-inch break size would produce limiting results, therefore, a burnup study was performed for this break and the results are summarized in Table 15.3-2e.

The majority of the inputs for the upflow conversion remain unchanged from the V. C. Summer small break LOCA analysis of record. Key changes include identifying the plant as an upflow barrel/baffle configuration and changes to the vessel hydraulic and geometric data due to the upflow conversion. Additionally, credit for a reduction in the hot assembly peaking factor (P_{HA}) at high bumup steps (> 50,000 MWD/MTU) was taken to ensure that the 10 CFR 50.46 17% oxidation limit was met at all times-in-life. RN 09-022

A limiting PCT of 1923°F was calculated at 15,000 MWD/MTU, a limiting maximum HR transient oxidation of 12.13% was calculated at 16,000 MWD/MTU, and a limiting total maximum HR oxidation (pre-transient plus transient) of 16.63% was calculated at 50,000 MWD/MTU. The core wide oxidation was calculated to be less than 1% for the entire bumup study.

Plots of the following parameters are shown in Figures 15.3-19d through 15.3-19f for the limiting 2.50-inch break transient.

- Reactor Coolant System Pressure
- Core Mixture Height
- Cladding Temperature at Peak Cladding Temperature Elevation

Overall, the results of the upflow conversion calculations show that the requirements of 10 CFR 50.46 are still met: total oxidation is less than 17%, core wide oxidation is less than 1%, and peak cladding temperature is less than 2200°F.

15.3.1.3 Conclusions

Analyses presented in this section show that the high head portion of the ECCS together with accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the ECCS in the event of a small break loss of coolant accident.

15.3.2 MINOR SECONDARY SYSTEM PIPE BREAKS

15.3.2.1 Identification of Causes and Accident Description

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

15.3.2.2 Analysis of Effects and Consequences

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

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

15.3.2.3 Conclusions

The analyses presented in Section 15.4.2 demonstrate that the consequences of a minor secondary system pipe break are acceptable since a DNBR of less than the design limit 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 Acceptance Criteria

This event is classified as a Condition III event. Condition III events are defined as those events that do not cause more than a small fraction of fuel rods to fail, although sufficient fuel damage might occur to preclude immediate resumption of operation. The specific acceptance criteria for this event are as follows:

a. To meet the requirements of General Design Criteria 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations. RN 10-015

b. In the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 guidelines.

15.3.3.2 Identification of Causes and Accident Description

The Inadvertent Loading Event comprises core misloading scenarios such as the loading of one or more fuel assemblies into improper positions, the loading of 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. In addition to these scenarios, misloading events involving burnable absorbers are theoretically possible, scenarios such as the placement of a cluster of 20 burnable absorbers into a core location slated to have 24 burnable absorbers. All of these misloading scenarios potentially result in a core reactivity distribution that differs from the intended core reactivity distribution. As a result, the core power distribution and peaking factors may differ from predictions. Specifically, misloading errors can lead to increased local power peaking at the location of the misloading if the misloading results in a local reactivity increase relative to the intended pattern. If the misloading results in a local reactivity decrease, power peaking increases away from the location of the misloading are possible due to unintended power tilts. These kinds of increases, however, are generally distributed over a large core volume and are small relative to those where the local reactivity is increased.

Fuel misloads are prevented by the manufacturing controls employed to build the fuel and the core loading controls used to assemble the core. The manufacturing controls include checks on fuel rod weight to confirm the uranium loading in the fuel rod, active and passive gamma scans of individual fuel rods to confirm fuel enrichments, pellet stack lengths, pellet types, and the absence of pellet gaps during fuel manufacturing, and bar coding of each fuel rod to confirm its proper placement in the fuel assembly. RN 10-015

To reduce the probability of core loading errors during fuel loading, each fuel assembly and core component is marked with an identification number and loaded in accordance with a core loading diagram. During core loading, the identification numbers are checked before the assembly is moved into the core. Identification numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed. These procedures make the likelihood of core misloadings very small.

The severity and detectability of fuel misloads are influenced by several factors: the local reactivity perturbation relative to the intended core loading pattern, the core position of the misload, the local environment of the misloaded fuel assembly, and the number of operable incore detector locations and their proximity to the misload location. Should misloadings occur, the incore system of movable flux detectors, which is used to verify power distributions during startup and throughout the operating cycle, is capable of revealing enrichment errors or misloadings which would cause the kind of substantial power distribution perturbation that would be necessary to induce large numbers of fuel rod failures. In addition, thermocouples and excore detectors can provide additional indications of power distribution anomalies. This instrumentation, along with the startup testing performed each cycle, make the detection of severe misloadings highly likely.

15.3.3.3 Evaluation

The incore moveable detector system is used to search for potential fuel misloads at the start of each operating cycle. Following fuel loading and low power physics testing, an initial core power distribution measurement is made. The core power level of this initial flux map is typically between ~30% and ~50% of rated thermal power. This initial power distribution measurement is used to confirm that the measured power distribution is consistent with the predicted power distribution. Observed flux map deviations in excess of the flux map review criteria (see Table 15.3-9) would prompt an investigation of a possible core anomaly. This satisfies the first acceptance criterion given in Section 15.3.3.1a above.

In Reference 15.3.8.29, a large number of misloads were evaluated for representative core designs employing current fuel types and fuel features. The simulated misloads, involving one or two fuel assemblies, covered a wide range of local reactivity perturbations and core positions. The resulting hot full power (HFP) $F_{\Delta H}$ peaking factors ranged from benign to very severe. Severe misloads with peaking factors that exceed the $F_{\Delta H}$ limit for DNB at normal operation conditions have the potential for fuel failure if they remain undetected. The simulated misloads were assessed with respect to severity and detectability.

The detectability assessments of Reference 15.3.8.29 demonstrated that the incore detector system is very robust with respect to detection of misloads severe enough to fail fuel during normal operation. By examining a large number of moveable detector thimble patterns, the detectability assessments considered the effect of inoperable moveable detector thimbles on misload detectability. Even when the minimum number of operable detector locations allowed per the plant licensing bases was assumed, the incore detector system was capable of reliably detecting misloads severe enough to fail fuel during normal operation. RN 10-015

Fuel misloads involving a single fuel rod or fuel pellet were not evaluated as part of Reference 15.3.8.29. Such misloads, in general, will not be detectable using the incore detector system due to the very small power distribution perturbation. In terms of increased peaking factors and reduced DNBR values, however, the consequences of such misloads will be very small and limited to the affected fuel rod and the immediately adjacent fuel rods.

Detection of fuel misloads is, in part, a function of the number of available incore detector locations. Reference 15.3.8.29 demonstrated that the flux map review criteria of Table 15.3-9 are effective in detecting fuel misloads that could lead to fuel failures during normal operation. To enhance the probability that significant misloads will be detected, tighter review criteria are employed when the number of available detector locations is reduced. These review criteria will be used for startup and subsequent at-power flux maps.

The detectability assessments of Reference 15.3.8.29 confirm that the moveable detector system can reliably detect fuel misloads that could fail fuel during normal operation when the Table 15.3-9 review criteria are employed. Specifically, Reference 15.3.8.29 demonstrated that only a small fraction of 1% of misloads severe enough to fail fuel during normal operation would be undetected at startup using these limited review criteria. Furthermore, it was judged that even these "undetected" misloads would very likely be detected if other attributes of the startup power distribution measurement (e.g., tilts and reaction rate error contours) were considered along with the results of low power physics testing. Given that detection of >99% of misloads severe enough to fail fuel is expected using these review criteria, a radiological consequences analysis is deemed unnecessary. Failures in fresh fuel during startup would have negligible radiological consequences since there is only a small fission product inventory. Following startup, any fuel rod failures would occur gradually and would be detected by coolant activity monitoring. Since the number of potential fuel rod failures due to a core misload would be extremely small and such failures would occur gradually, any coolant activity releases would initially be well within the cleanup capacity of the plant. Any trend in increased coolant activity would warrant further investigation and evaluation. Therefore, the second acceptance criterion for this event would be satisfied since failures would be gradual, detectable, and the operations would be maintained within Technical Specification coolant activity guidelines. RN 10-015

15.3.3.4 Conclusions

Fuel misloads are prevented by manufacturing controls and core loading controls. In the unlikely event that a fuel misload should occur, the incore moveable detector system is capable of reliably detecting misloads that could fail fuel at normal operation conditions. Exceeding the review criteria herein would initiate an investigation to identify potential core anomalies. Any failures associated with an undetected fuel misload would be gradual, detectable, and the operations would be maintained within Technical Specification coolant activity guidelines.

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

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

2. Low reactor coolant loop flow.

The reactor trip on reactor coolant pump bus undervoltage is provided to protect against conditions that can cause a loss of voltage to all reactor coolant pumps, i.e., station blackout. 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' flywheel kinetic energy is available for full coastdown. Both trips are blocked below approximately 10% power (Permissive 7).

The reactor trip on low primary coolant loop flow is provided to protect against loss-of-flow conditions that 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 38% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10 and 39% power (Permissive 7 and Permissive 8), low-flow in any two loops will actuate a reactor trip.

Normal power for each pump is supplied through individual busses connected to the isolated phase bus duct between the generator circuit breaker and the main transformer. Faults in the substation may cause a trip of the main transformer high side circuit breaker leaving the generator to supply power to the reactor coolant pumps. When a generator circuit breaker trip occurs because of electrical faults, the pumps are automatically transferred to an alternate power supply and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults, the generator circuit breaker is tripped and the reactor coolant pumps remain connected to the network through the transformer high side breaker. Continuity of power to the pump buses is achieved without motoring the generator since means are provided to isolate the generator without isolating the pump buses from the external power lines (e.g., a generator output breaker is provided as well as a station output breaker).

15.3.4.2 <u>Analysis of Effects and Consequences</u>

15.3.4.2.1 Method of Analysis

This transient is analyzed by three digital computer codes. First the LOFTRAN[16] Code is used to calculate the loop and core flow during the transient. The LOFTRAN Code is also used to calculate the time of reactor trip, based on the calculated flows, and the nuclear power transient following reactor trip. The FACTRAN[8] Code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC[17] Code is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN.

The following case has been analyzed:

All loops operating, all loops coasting down.

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.

15.3.4.2.2 Results

The calculated sequence of events is shown in Table 15.3-3. Figures 15.3-27 and 15.3-28 show the flow coastdown, nuclear power, and heat flux transients and minimum DNBR for the limiting complete loss of flow event. The reactor is assumed to trip on the undervoltage signal.

15.3.4.3 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit value during the transient, and thus, no core safety limit is violated.

15.3.5 WASTE GAS DECAY TANK RUPTURE

15.3.5.1 Identification of Causes and Accident Description

The Gaseous Waste Processing System, as discussed in Chapter 11, is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, hydrogen recombiners, 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

98-01

Nonvolatile fission product concentrations are greatly reduced as the coolant being let down is passed through the purification demineralizers. An iodine removal factor of 10 is expected in the mixed bed demineralizers, and an iodine partition factor of the order of 10,000 is expected between the liquid and vapor phases. Based on the above analysis and operating experience at Yankee-Rowe and Saxton, activity stored in a gas decay tank consists of that from the noble gases released from the processed coolant and only negligible quantities of less volatile isotopes.

98-01

15.3.5.3 Environmental Consequences

The quantity of noble gases contained in each gas decay tank is restricted to assure that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 Rem. Using the methodology outline in Section 5.6.1 of NUREG-0133, the maximum allowable limit is 131,000 curies of noble gases (considered as ^{133}Xe). The corresponding beta/gamma skin dose will also not exceed 2.05 Rem. 98-01

To supplement the above evaluation, a realistic analysis and a conservative analysis of potential environmental consequences is performed using the NUREG-0133 methodology with the maximum expected noble gas inventory shown in Table 15.3-5. These supplemental dose projections resulting from the postulated rupture of a gas decay tank are based on the following assumptions:

1. The decay tank rupture is assumed to occur immediately after isolation of the decay tank from the gaseous waste processing system, releasing the entire contents of the tank to the outside atmosphere at ground level. The assumption of the release of the noble gas inventory from only a single tank is based upon the fact that the valving of the decay tanks in the gaseous waste processing system has been designed so that a release from one decay tank by any means does not result in any additional release of radioactivity stored in any of the other decay tanks.

2. The 0-1 hour atmospheric diffusion factor given in Appendix 15A is applicable for the conservative offsite dose analysis. The average annual atmospheric diffusion factors of 3.25×10^{-6} and 3.65×10^{-7} sec/m^3 at the site boundary and low population zone are used for the realistic offsite dose analysis.

The site boundary gamma whole body and gamma/beta skin doses for the realistic analysis are 2.1×10^{-3} Rem and 1.22×10^{-2} Rem respectively. 98-01

The low population zone gamma whole body and gamma/beta skin doses for the realistic analysis are 2.29×10^{-4} Rem and 1.37×10^{-3} Rem respectively.

The site boundary gamma whole body and gamma/beta skin doses for the conservative analysis are 2.56×10^{-1} Rem and 1.54 Rem respectively.

The low population zone gamma whole body and gamma/beta skin doses for the conservative analysis are 6.34×10^{-2} Rem and 0.38 Rem respectively.

Since there is no significant amount of the iodine radioisotope within the gas decay tank, a thyroid dose is not calculated.

The calculated doses are well within the limits of 10CFR100 and fall below the 10CFR20 yearly instantaneous dose rate limits (≤ 500 mrem total body & ≤ 3000 mrem skin) for unrestricted areas.

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

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

15.3.6.2 Analysis of Effects and Consequences

15.3.6.2.1 Method of Analysis

Power distributions within the core are calculated by the ANC[20] Code based on macroscopic cross section generated by PHOENIX-P[20]. The peaking factors calculated by ANC are then used by THINC to calculate the minimum DNB for the event. The plant was analyzed for the worst rod withdrawn from Bank D inserted at the insertion limit, with the reactor initially at full power.

15.3.6.2.2 Results

Two (2) cases have been considered as follows:

1. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the failed RCCA. In terms of the overall system response, this case is similar to those presented in Section

15.2.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBR's than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNBR from falling below the safety limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip would be expected to trip the plant, shows that an upper limit for the number of rods with a DNBR less than the safety limit value is 5 percent.

2. If the reactor is in automatic control mode, continuous withdrawal of a single RCCA will result in the immobility of the other RCCA's 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 DNBR in the core of less than the safety limit.

15.3.6.3 Conclusions

For the case of one RCCA fully withdrawn, with the reactor in the automatic or manual control mode, and initially operating at full power with Bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNBR less than the design limit is 5 percent or less 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 1 discussed above, the insertion limit alarms (low and low-low alarms) would also serve to alert the operator.

15.3.7 BREAK IN INSTRUMENT LINE OR OTHER LINES FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT

There are no instrument lines connected to the reactor coolant system that penetrate containment. However, grab sample lines from reactor coolant loop 2 and loop 3 hot legs, the pressurizer steam and liquid spaces, and the 3 inch chemical and volume control system (CVCS) letdown line do penetrate containment. The grab sample lines are equipped with normally closed isolation valves both inside and outside containment and are designed in accordance with General Design Criterion 55.

The most severe pipe rupture, with regard to release of radioactivity during normal plant operation, would be complete severance of the 3 inch CVCS letdown line just outside containment, upstream of the outer containment isolation valve, (see Figure 9.3-16, Sheet 1) at rated power. Complete severance of the letdown line would result in the loss of reactor coolant at the rate of up to 165 gpm based on the use of two (60 gpm) letdown orifice prior to the accident. Following such a break, the RCS pressure decreases due to the loss of reactor coolant. When the pressurizer pressure has reached the low pressure setpoint, a reactor trip is initiated. A turbine trip follows a reactor trip and results in an increase in secondary side pressure to the steam generator safety valve set pressure. The safety injection signal on low pressurizer pressure terminates the break flow by isolating the letdown line inside containment. The reactor coolant inventory is replenished by the charging pumps. Operation of these pumps ensure that the core will not be uncovered and prevents any significant increases in clad temperature. For smaller breaks within the capability of the Reactor Makeup System, Engineered Safety Features actuation would not occur. Frequent automatic operation of the Reactor Makeup System would provide some indication of reactor coolant loss to the operator. 99-01

For this event, environmental consequences are evaluated for both a realistic and a conservative case. Assumptions and parameters used to calculate the activity released and the offsite doses resulting from this event for both the realistic and conservative cases are summarized in Table 15.3-6 and are listed as follows:

1. The operator will be able to detect the rupture and isolate the break within 10 minutes by use of the following:

   a. Low flow alarm on the reactor coolant letdown monitor, RM-L1 (see Section 11.4.2).

   b. Leakage greater than 45 gpm which actuates an alarm in the control room. The alarm drain system is discussed in Section 9.3.3.3 and illustrated schematically by Figure 9.3-7.

   c. Indication of letdown line flow on the main control board.

   However, for this offsite dose analysis it is conservatively assumed that the time required for the operator to identify the accident and isolate the break is 30 minutes.

2. Reactor coolant is lost through the break at a rate of 165 gpm. No activity is released from the break after the isolation time of 30 minutes. RN 01-125

3. The iodine partition factor for activity released from the break is 0.4. RN 01-125

4. The concentration of radioactive nuclides in the reactor coolant is listed in Table 11.1-2. For the realistic case these values were multiplied by a factor of 0.12. For the conservative case it is assumed that concurrent iodine spikes occur as a result of the accident. RN 01-125

5. For the concurrent iodine spike, the increase in the reactor coolant iodine concentration is estimated using a spiking model which assumes that the iodine release rate from the fuel rods to the coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value associated with reactor coolant activities at 1 μCi/gm dose equivalent I-131. The iodine release rate during the concurrent spike is based on a maximum letdown flow of 143 gpm which includes the maximum normal letdown of 120 gpm plus 12 gpm to account for uncertainty in flow and 11 gpm primary coolant leakage. 99-01 RN 01-125

Using the previously listed assumptions, isotopic releases to the environment are determined to be those listed in Tables 15.3-7 and 15.3-8 for the realistic and conservative cases, respectively.

Gamma, beta and thyroid doses at the site boundary for the realistic case are 4.27×10^{-5} Rem, 7.11×10^{-5} Rem, and 3.17×10^{-3} Rem, respectively. Corresponding doses at the low population zone are 4.82×10^{-6} Rem, 8.03×10^{-6} Rem, and 3.58×10^{-4} Rem, respectively. RN 01-125

Gamma, beta and thyroid doses at the site boundary for the conservative case with a concurrent iodine spike are 2.31×10^{-1} Rem, 1.35×10^{-1} Rem, and $2.17 \times 10^{+1}$ Rem, respectively. Corresponding doses at the low population zone are 1.34×10^{-2} Rem, 7.83×10^{-3} Rem, and 1.26 Rem, respectively. RN 01-125

Doses resulting from this accident are well within the limits defined in 10 CFR 100 (25 Rem whole body and 300 Rem thyroid).

15.3.8 REFERENCES

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TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR SMALL BREAK LOCA

Break Size	2-Inch	2.25-Inch	2.5-Inch	2.75-Inch	3-Inch	3.25-Inch	4-Inch	6-Inch
Break Initiation (sec)	0	0	0	0	0	0	0	0
Reactor Trip Signal (sec)	105	81	60	47	39	32	22	7.9
Core Power Shutdown (sec)	110.7	86.7	65.7	52.7	44.7	37.7	27.7	13.6
S-Signal (sec)	115	91	69	56	48	41	30	14
Safety Injection Begins[(1)] (sec)	142	118	96	83	75	68	57	41
Accumulator Injection (sec)	N/A	N/A	2322	1596	1312	1072	640	304
Top of Core Uncovered (sec)	2125	1571	994	722	553	447	311	300
PCT Time[(3)] (sec)	3879.6	3355.8	2772.9	1871.9	1573.6	1315.6	816.8	391
Top of Core Recovered	(2)	(2)	(2)	(2)	(2)	(2)	(2)	436

RN 06-038

Notes:

(1) Safety Injection (SI) is assumed to begin 27 seconds after the SI signal.

(2) For the cases where core recovery is greater than the transient time, basis for transient termination can be concluded based on some or all of the following: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, and (3) Core mixture level has begun to increase and is expected to continue for the remainder of the accident.

(3) The limiting time-in-life for the 2.75-inch break case for PCT was determined to be at 12,000 MWD/MTU. All other PCT times are for beginning-of-life (BOL).

TABLE 15.3-1a

TIME SEQUENCE OF EVENTS FOR SMALL BREAK LOCA
HHSI THROTTLE VALVE REPLACEMENT

Break Size	2.75-inch	3-inch
Break Initiation (sec)	0	0
Reactor Trip Signal (sec)	46.9	39.2
Core Power Shutdown (sec)	52.6	44.9
S-Signal (sec)	56.3	48.3
Safety Injection Begins[(1)] (sec)	83.3	75.3
Accumulator Injection (sec)	1644	1307
Top of Core Uncovered (sec)	699	522
PCT Time [(3)] (sec)	1939.4	1628.6
Top of Core Recovered (sec)	(2)	7565

RN 06-019

Notes:

(1) Safety Injection (SI) is assumed to begin 27 seconds after the SI signal.

(2) For the cases where core recovery is greater than the transient time, basis for transient termination can be concluded based on some or all of the following: (1) The RCS system pressure is decreasing which will increase SI flow, (2) Total RCS system mass is increasing due to SI flow exceeding break flow, and (3) Core mixture level has begun to increase and is expected to continue increasing for the remainder of the accident.

(3) The limiting time-in-life for the 2.75-inch break case for PCT was determined to be at 17,000 MWD/MTU and the limiting time-in-life for the 3.0-inch break case for PCT was determined to be at 15,000 MWD/MTU.

TABLE 15.3-1b

TIME SEQUENCE OF EVENTS FOR SMALL BREAK LOCA
UPFLOW CONVERSION

Break Size	2.25-Inch	2.50-Inch	2.75-Inch	3.00-Inch	3.25-Inch
Break Initiation (sec)	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal (sec)	79.2	61.2	50.9	40.8	34.2
Core Power Shutdown (sec) (1)	84.9	66.9	56.6	46.5	39.9
S-Signal (sec)	89.3	71.0	60.2	49.9	43.1
Safety Injection Begins (sec) (2)	116.3	98.0	87.2	76.9	70.1
Top of Core Uncovered (sec)	1109	864	653	551	587
Accumulator Injection Begins (sec)	2940	2008	1631	1308	1088
Top of Core Recovered (sec)	N/A (3)	6627	7454	7495	6874

RN
09-022

Notes:

(1) The core is assumed to shut down 5.7 seconds after the Reactor Trip Signal.

(2) Safety Injection (SI) is assumed to begin 27 seconds after the S-Signal.

(3) The core does not recover during the length of the transient; however, the core mixture level is steadily increasing and the transient is considered terminated since all transient termination criteria are met.

TABLE 15.3-2a

SMALL BREAK LOCA CALCULATION RESULTS
FUEL CLADDING DATA

Break Size (in)	2	2.25	2.5	2.75		3	3.25	4	6
Burn Up	BOL	BOL	BOL	12,000(1) MWD/MTU	14,000(2) MWD/MTU	BOL	BOL	BOL	BOL
PCT (°F)	1309	1582	1778	1952	1909	1837	1776	1544.5	957
PCT Elevation (ft)	11.75	11.75	12	12	12	11.75	11.75	11.75	10.5
Hot Rod Burst Time (sec)	N/A	N/A	N/A	1869.4	1728.1	N/A	N/A	N/A	N/A
Hot Rod Burst Elevation (ft)	N/A	N/A	N/A	12	11.75	N/A	N/A	N/A	N/A
Max. Local transient ZrO_2 Reaction (%)	0.43	1.71	4.55	13.8	14.34(3)	4.24	2.98	0.68	0.01
Max. Local ZrO_2 Elevation (ft)	11.75	11.75	12	12	11.75	11.75	11.75	11.5	10.5
Core-Wide Avg. ZrO_2 (%)	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0	< 1.0

RN
06-038

1. The limiting time-in-life for the 2.75 inch break case for PCT was determined to be at 12,000 MWD/MTU.
2. The limiting time-in-life for the 2.75 inch break case for transient oxidation was determined to be at 14,000 MWD/MTU.
3. With pre-transient oxidation, the maximum local ZrO_2 reaction is 16.5%.

TABLE 15.3-2b

SMALL-BREAK LOCA ANALYSIS
SIGNIFICANT INPUT PARAMETERS

Reactor core power, (MWt)	2900
Peak linear heat generation rate	13.80 kW/ft
Accumulator water volume, nominal (ft^3/accumulator)	1013.5
Accumulator gas pressure, minimum (psia)	584.7
Hot Assembly Peaking Factor, $\overline{P}_{HA}$	1.42(1)

1. A $\overline{P}_{HA}$ value of 1.443 was used for all breaks other than the 2.75-inch break case, which used a $\overline{P}_{HA}$ value of 1.42. The $\overline{P}_{HA}$ value of 1.42 is the current limit and will remain the limit going forward.

TABLE 15.3-2c

SMALL BREAK LOCA CALCULATION RESULTS
FUEL CLADDING DATA
HHSI THROTTLE VALVE REPLACEMENT

Break Size (in)	2.75		3	
Burn up (MWD/MTU)	17,000(1)	17,500(2)	15,000(1)	15,500(2)
PCT (°F)	1747	1739	1775	1764
PCT Elevation (ft)	11.75	11.75	12	12
Hot Rod Burst Time (sec)	1937.7	1926.6	1626.7	1549.5
Hot Rod Burst Elevation (ft)	11.75	11.75	12	12
Max. Local Transient ZrO_2 Reaction (%)	6.91	6.92	5.83	6.23
Max. Local ZrO_2 Elev. (ft)	11.75	11.75	12	12
Core-Wide Avg. ZrO_2 (%)	< 1.0	< 1.0	< 1.0	< 1.0

1. The limiting time-in-life for the 2.75-inch break case for PCT was determined to be at 17,000 MWD/MTU and the limiting time-in-life for the 3.0-inch break case for PCT was determined to be at 15,000 MWD/MTU. RN 06-019
2. The limiting time-in-life for the 2.75-inch break case for transient oxidation was determined to be at 17,500 MWD/MTU and the limiting time-in-life for the 3.0-inch break case for transient oxidation was determined to be at 15,500 MWD/MTU.

TABLE 15.3.2d

SMALL BREAK LOCA ANALYSIS
SIGNIFICANT INPUT PARAMETERS
HHSI THROTTLE VALVE REPLACEMENT

Reactor core power, (MWt)	2900
Peak linear heat generation rate	13.80 kW/ft
Accumulator water volume, nominal (ft^3/accumulator)	1013.5
Accumulator gas pressure, minimum (psia)	584.7
Hot Assembly Peaking Factor, $\overline{P}_{HA}$	1.42(1)

1. A $\overline{P}_{HA}$ value of 1.42 was used in the HHSI throttle valve replacement calculations, which is the current limit and will remain the limit going forward.

TABLE 15.3-2e

FUEL CLADDING RESULTS FOR SMALL BREAK LOCA
UPFLOW CONVERSION - 2.50-INCH BREAK

Burnup (MWD/MTU)	15,000[(1),(3)]	16,000[(2),(3)]	50,000[(4)]
PCT (°F)	1922.5	1867.3	1832.9
PCT Elevation (ft)	12.00	12.00	12.00
Hot Rod Burst Time (sec)	2362.6	2307.3	1787.8
Hot Rod Burst Elevation (ft)	12.00	12.00	11.75
Maximum HR Transient ZrO_2 (%)	11.73	12.13	8.72
Maximum HR Transient ZrO_2 Elevation (ft)	12.00	11.75	11.75
Total Maximum HR ZrO_2 (%)	14.20	14.70	16.63
Core-Wide Average ZrO_2 (%)	<1.0	<1.0	<1.0

RN 09-022

Notes:

(1) The limiting time-in-life PCT for the Upflow Conversion was calculated at 15,000 MWD/MTU.

(2) The limiting time-in-life maximum HR transient oxidation for the Upflow Conversion was calculated at 16,000 MWD/MTU.

(3) The effects of annular pellets were evaluated for the limiting PCT case (15,000 MWD/MTU) and the limiting transient oxidation case (16,000 MWD/MTU) and were found to have a negligible impact on the results.

(4) The total maximum HR oxidation for the Upflow Conversion was calculated to occur at 50,000 MWD/MTU.

TABLE 15.3-3

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Accident	Event	Time (sec)	
Complete Loss of Forced Reactor Coolant Flow			
All loops operating, all pumps coasting down	Coastdown begins	0.0	02-01
	Rod motion begins	1.5	
	Minimum DNBR occurs	3.0	

TABLE 15.3-5

ESTIMATED GAS DECAY TANK INVENTORY

Isotope	Gas Decay Tank Inventory (Curies)
Kr-83m	4.3
Kr-85	53000
Kr-85m	47
Kr-87	4.6
Kr-88	45
Xe-131m	530
Xe-133	56000
Xe-133m	2900
Xe-135	500
Xe-135m	2.1
Xe-138	0.13

Basis: 1% Defective Fuel and the Purge System Operation; VCT Purge Rate = 0.7 sc/m

TABLE 15.3-6

PARAMETERS USED IN ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM LETDOWN LINE RUPTURE

Parameter	Realistic Case	Conservative Case	
Core Thermal Power	2958 MWt	2958 MWt	
Fuel Defects	0.12 percent (1)	1 percent (2)	
Iodine Spiking Basis	None	Concurrent Spike (3)	
Break Flow Rate	165 gpm	165 gpm	
Break Flow Isolation Time	30 min	30 min	99-01
Partition Factors			
Iodine	0.4	0.4	99-01
Noble Gases	1.0	1.0	
Meteorology	Annual Average	Accident	
Dose Model	Appendix 15A	Appendix 15A	

NOTE:

1. American National Standards Institute, "Source Team Specification", ANS/ANSI 18.1-1984.
2. For noble gases.
3. Initial Reactor Coolant iodine activities based on 1 μCi/gm dose equivalent I-131. 99-01

TABLE 15.3-7

CHEMICAL AND VOLUME CONTROL SYSTEM
LETDOWN LINE RUPTURE - ISOTOPIC RELEASE
TO THE ENVIRONMENT - REALISTIC CASE
NO IODINE SPIKE

Isotope	Activity Released (Ci)
I-131	1.99×10^{0}
I-132	2.06×10^{0}
I-133	3.06×10^{0}
I-134	3.99×10^{-1}
I-135	1.59×10^{0}
Kr-83m	9.05×10^{-1}
Kr-85	1.60×10^{1}
Kr-85m	3.78×10^{0}
Kr-87	2.31×10^{0}
Kr-88	6.73×10^{0}
Kr-89	1.87×10^{-1}
Xe-131m	4.84×10^{0}
Xe-133	6.10×10^{2}
Xe-133m	3.99×10^{1}
Xe-135	1.81×10^{1}
Xe-135m	1.09×10^{0}
Xe-138	1.34×10^{0}

99-01

TABLE 15.3-8

CHEMICAL AND VOLUME CONTROL SYSTEM
LETDOWN LINE RUPTURE - ISOTOPIC RELEASE
TO THE ENVIRONMENT - CONSERVATIVE CASE
WITH IODINE SPIKE

Isotope	Activity Released (Ci)	
I-131	9.87×10^{1}	RN 00-056
I-132	4.82×10^{2}	
I-133	2.01×10^{2}	
I-134	1.85×10^{2}	
I-135	1.80×10^{2}	
Kr-83m	7.54×10^{0}	
Kr-85	1.33×10^{2}	
Kr-85m	3.16×10^{1}	RN 00-056
Kr-87	1.93×10^{1}	
Kr-88	5.61×10^{1}	
Kr-89	1.56×10^{0}	
Xe-131m	4.03×10^{1}	
Xe-133	5.09×10^{3}	RN 00-056
Xe-133m	3.33×10^{2}	
Xe-135	1.51×10^{2}	RN 00-056
Xe-135m	9.12×10^{0}	
Xe-138	1.12×10^{1}	

TABLE 15.3-9

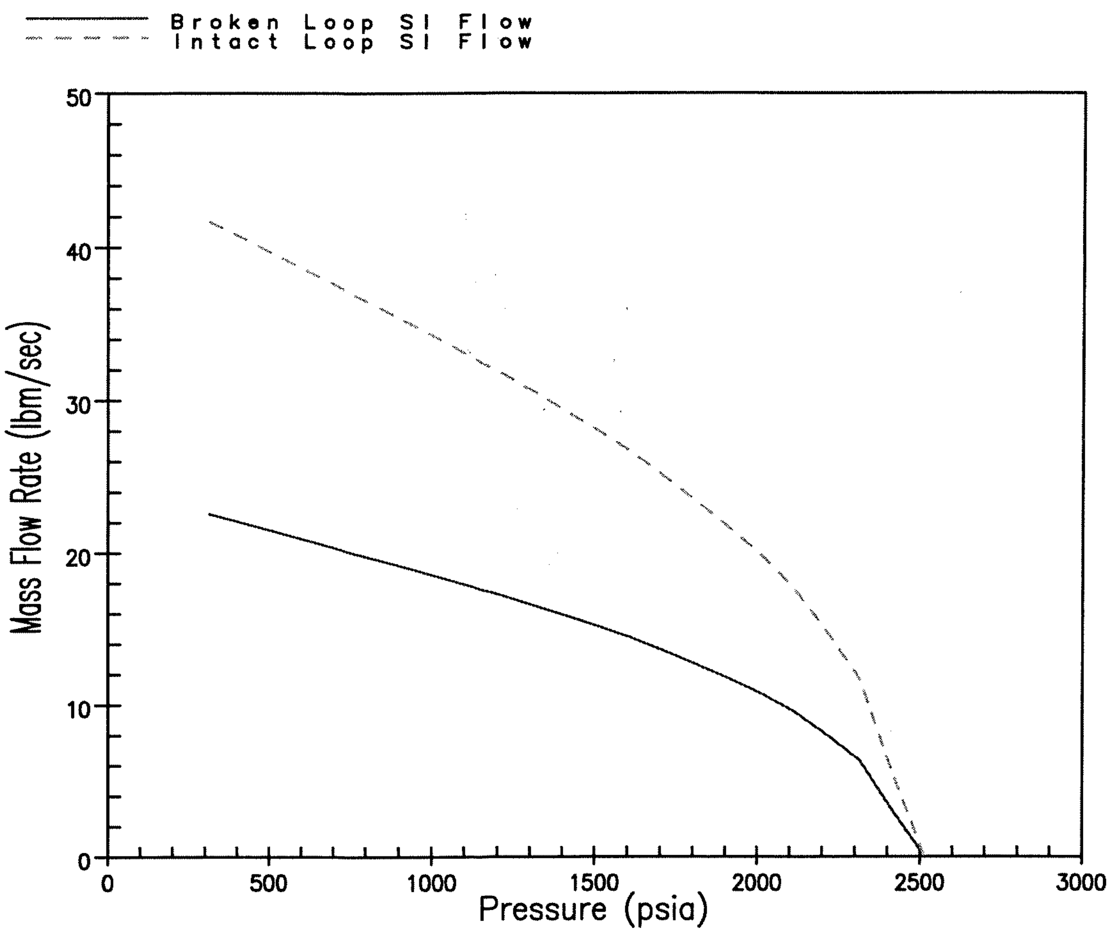
FLUX MAP REVIEW CRITERIA FOR INADVERTENT CORE LOADING

Number of Available Incore Detector Locations	**Measured vs. Predicted Detector Reaction Rate Comparison ***	**Symmetric Thimble Reaction Rate Comparison****
47 to 50	10%	7%
42 to 46	8%	5%
38 to 41	6%	5%

RN 10-015

* The review criterion is the table value (%) or an absolute normalized reaction rate difference equal to the table value divided by 100% (e.g., 10% / 100% = 0.1), whichever is greater.

** Applicable to symmetric thimbles with normalized reaction rates above 0.7. The review criterion is relative to the expected reaction rate difference.



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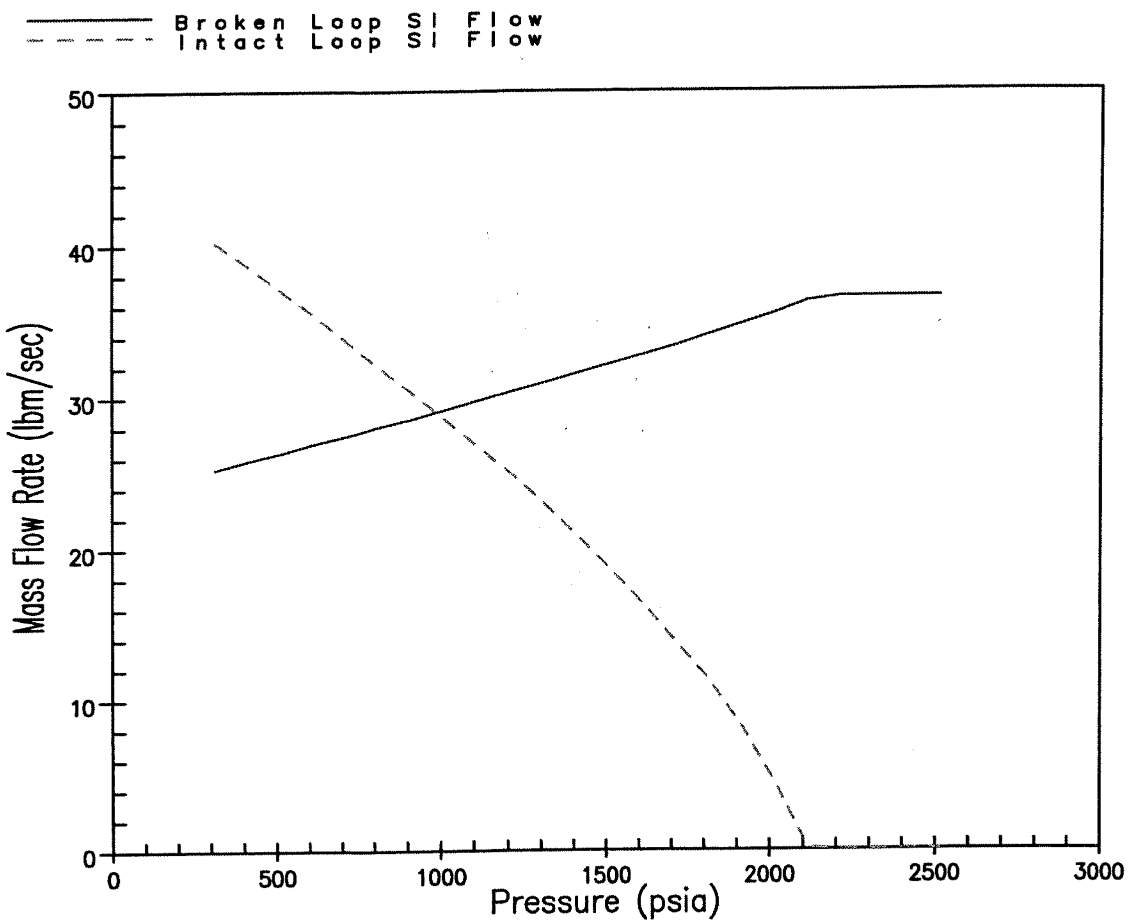
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Pumped Safety Injection Flow Rate
Faulted Loop Injects to RCS Pressure

(2 - 4 Inch Break)

Figure 15.3-1a



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Pumped Safety Injection Flow Rate
Faulted Loop Injects to Containment Pressure (0 psig)

(6 Inch Break)

Figure 15.3-1b

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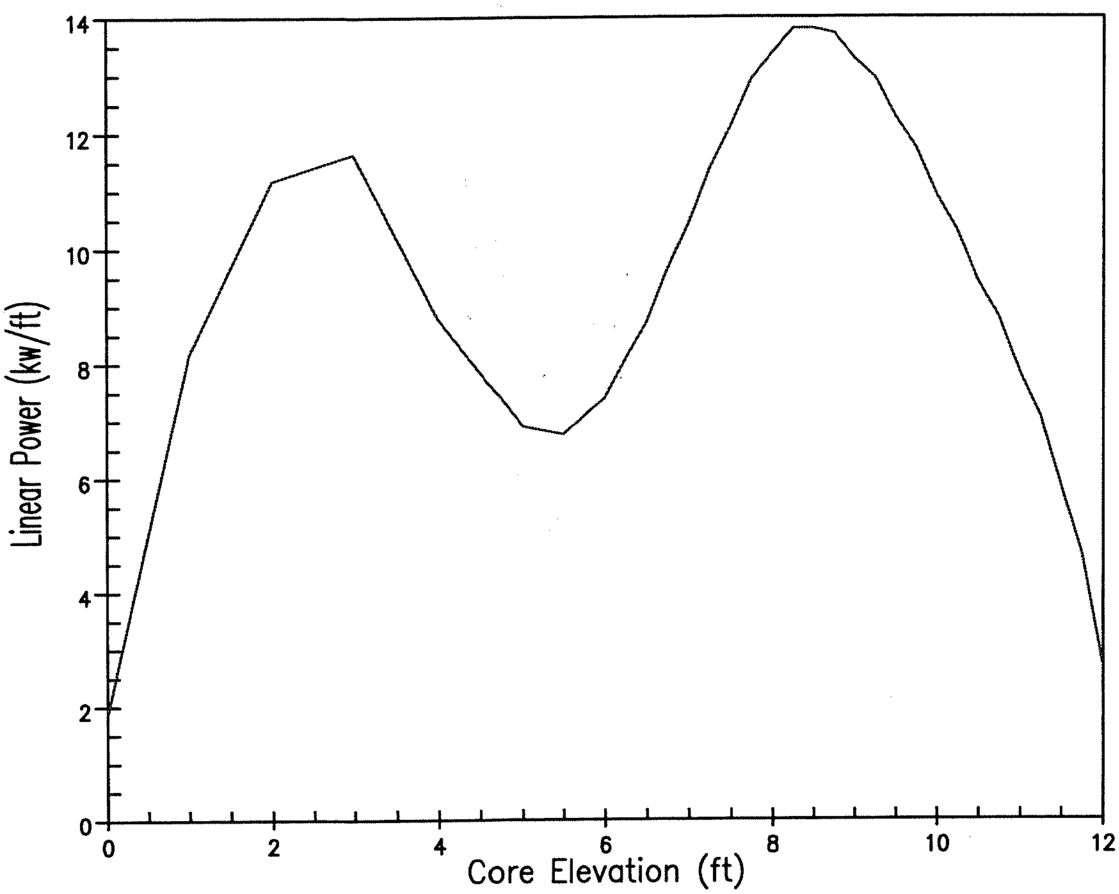

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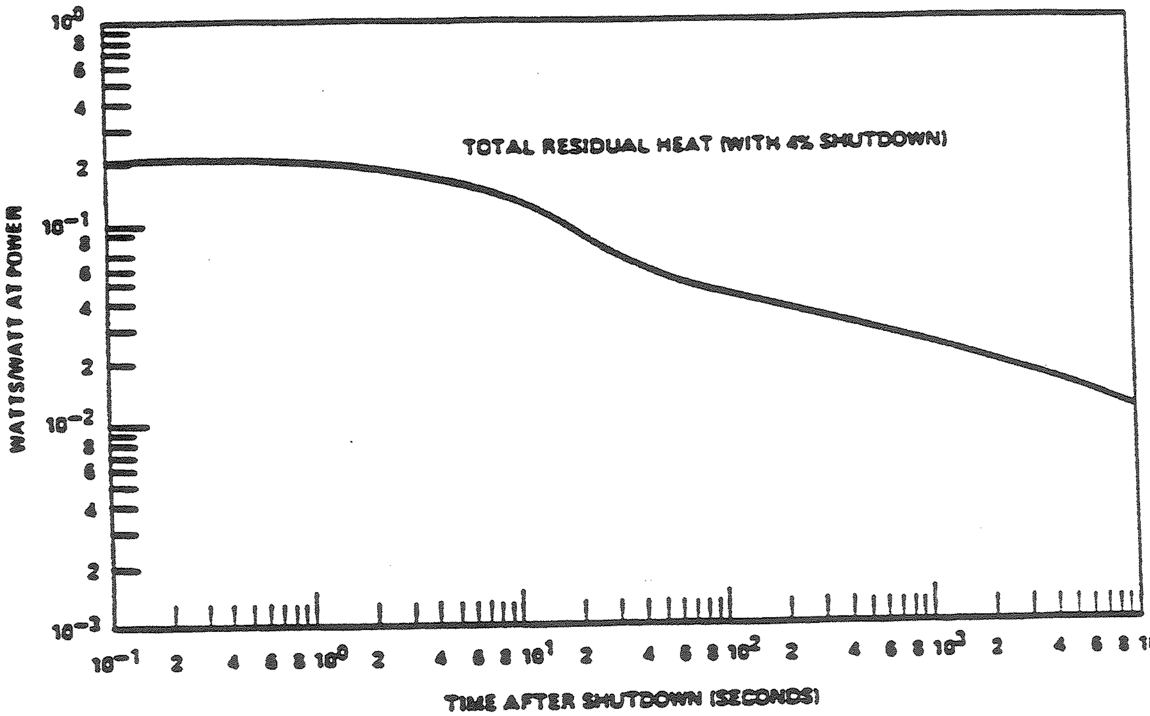
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Peak Rod Axial Power Shape Figure 15.3-2

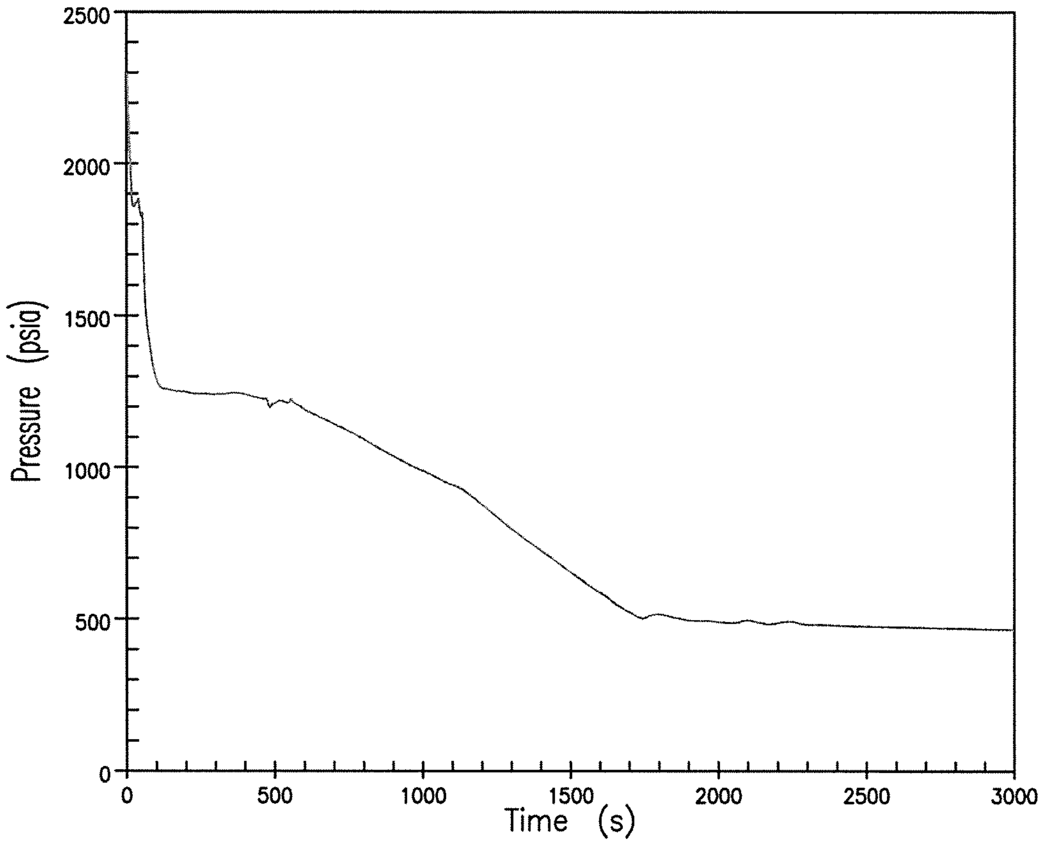


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Normalized Core Heat Generation Rate Following Shutdown (full rod insertion)

Figure 15.3-3

AMENDMENT 96-02
JULY 1996



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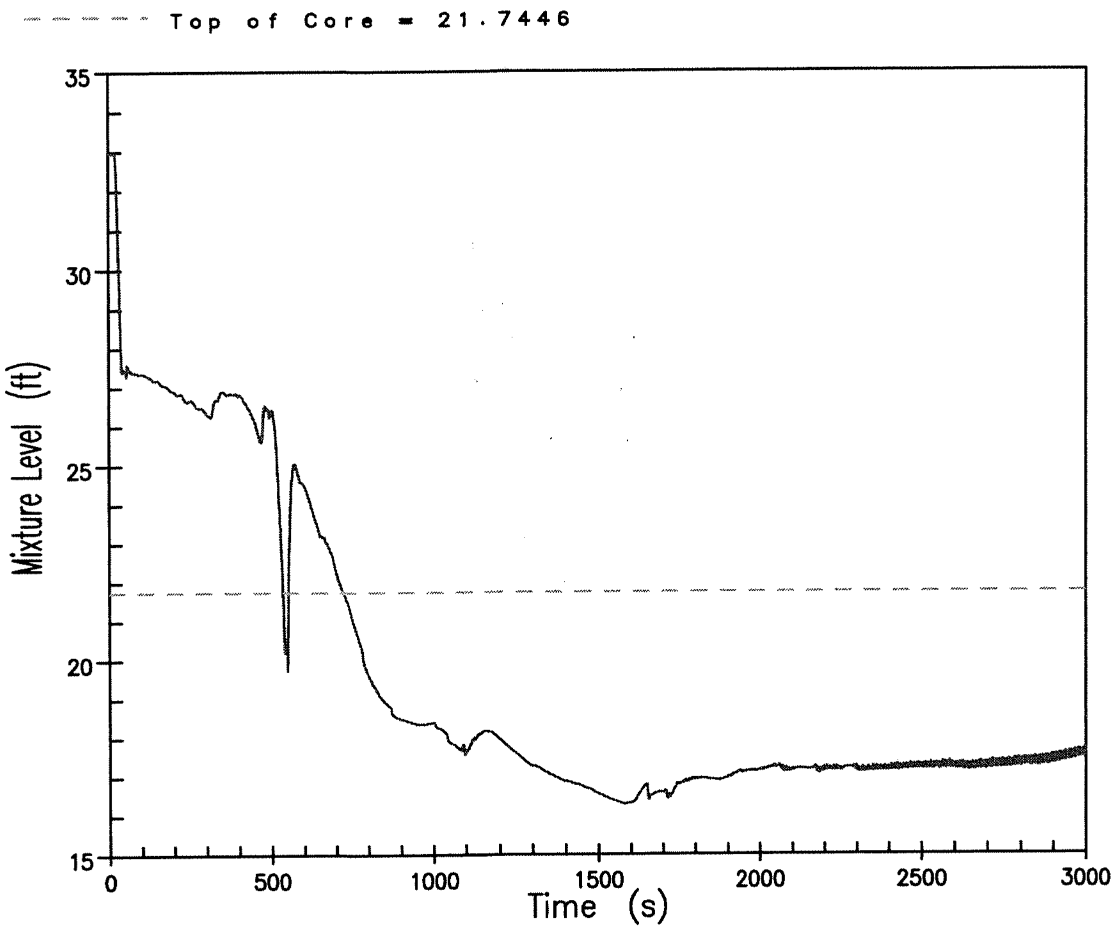
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure

(2.75-inch break)

Figure 15.3-4



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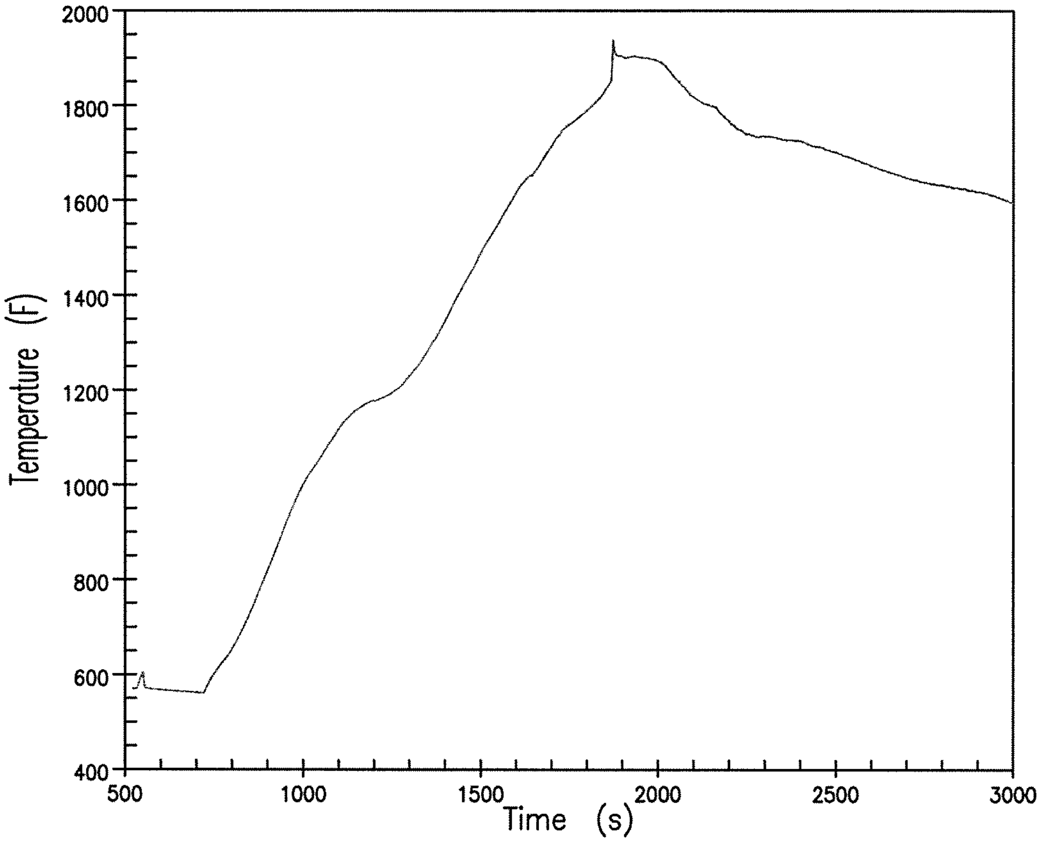
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Core Mixture Height (2.75-inch break) Figure 15.3-5



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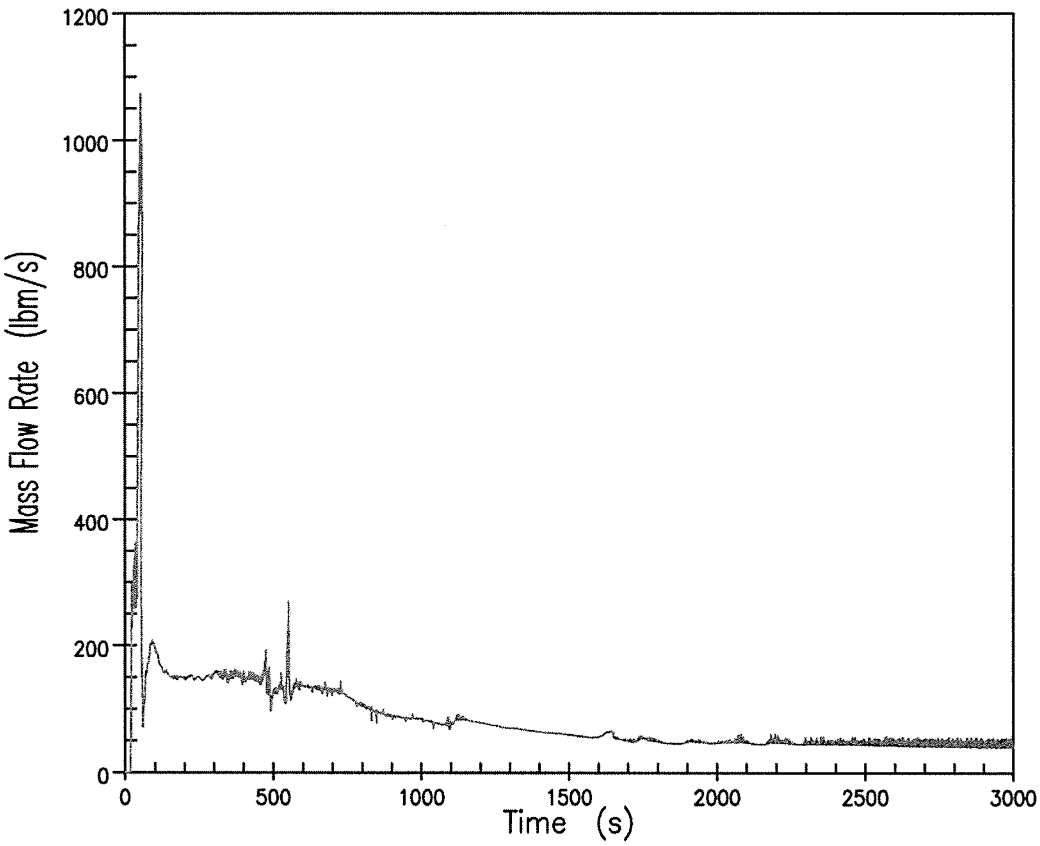
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(2.75 inch break)

Figure 15.3-6



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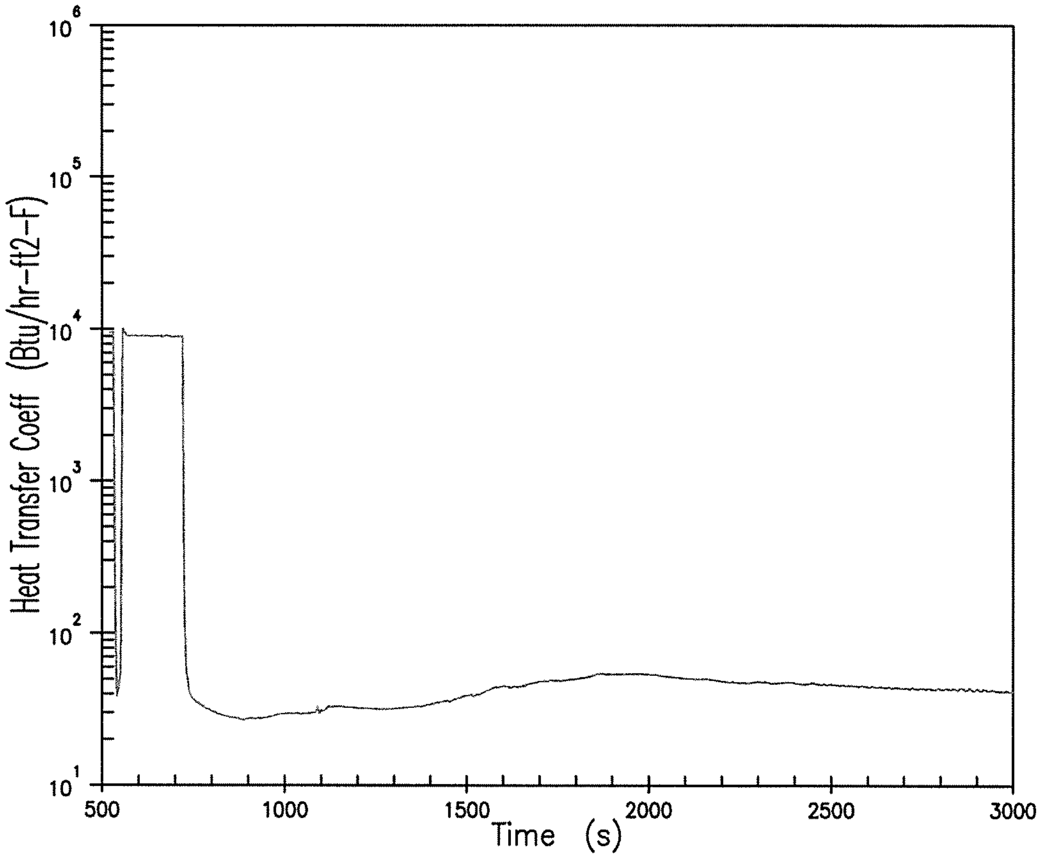
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Steam Mass Flowrate Out Top of Core (2.75 inch break) Figure 15.3-7



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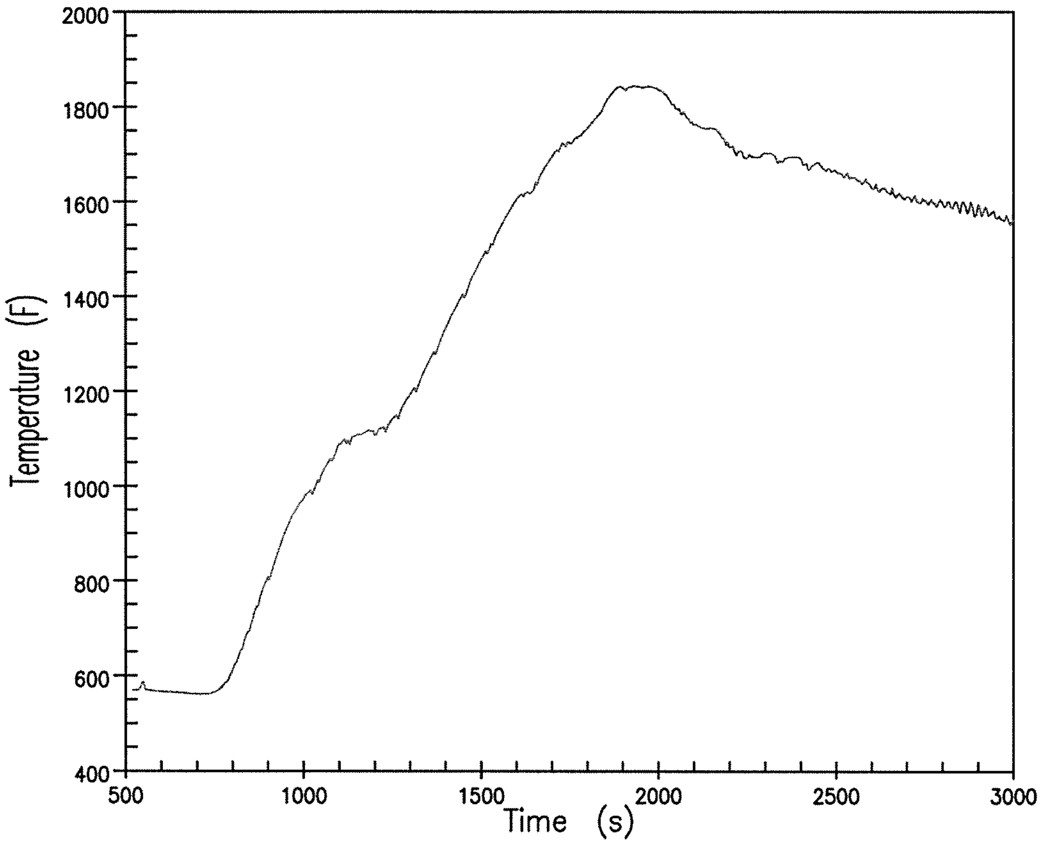
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Surface Heat Transfer Coefficient at Peak Temperature Elevation

(2.75-inch break)

Figure 15.3-8



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Fluid Temperature at Peak Clad Temperature Elevation

(2.75-inch break)

Figure 15.3-9

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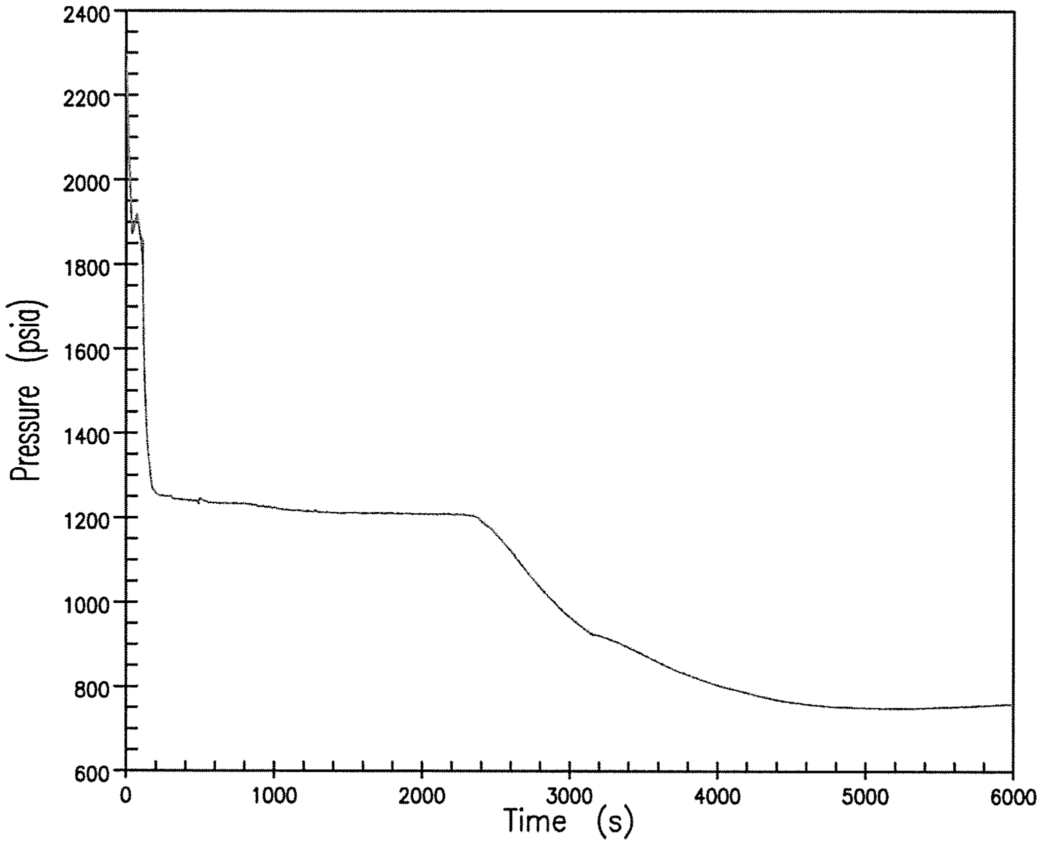

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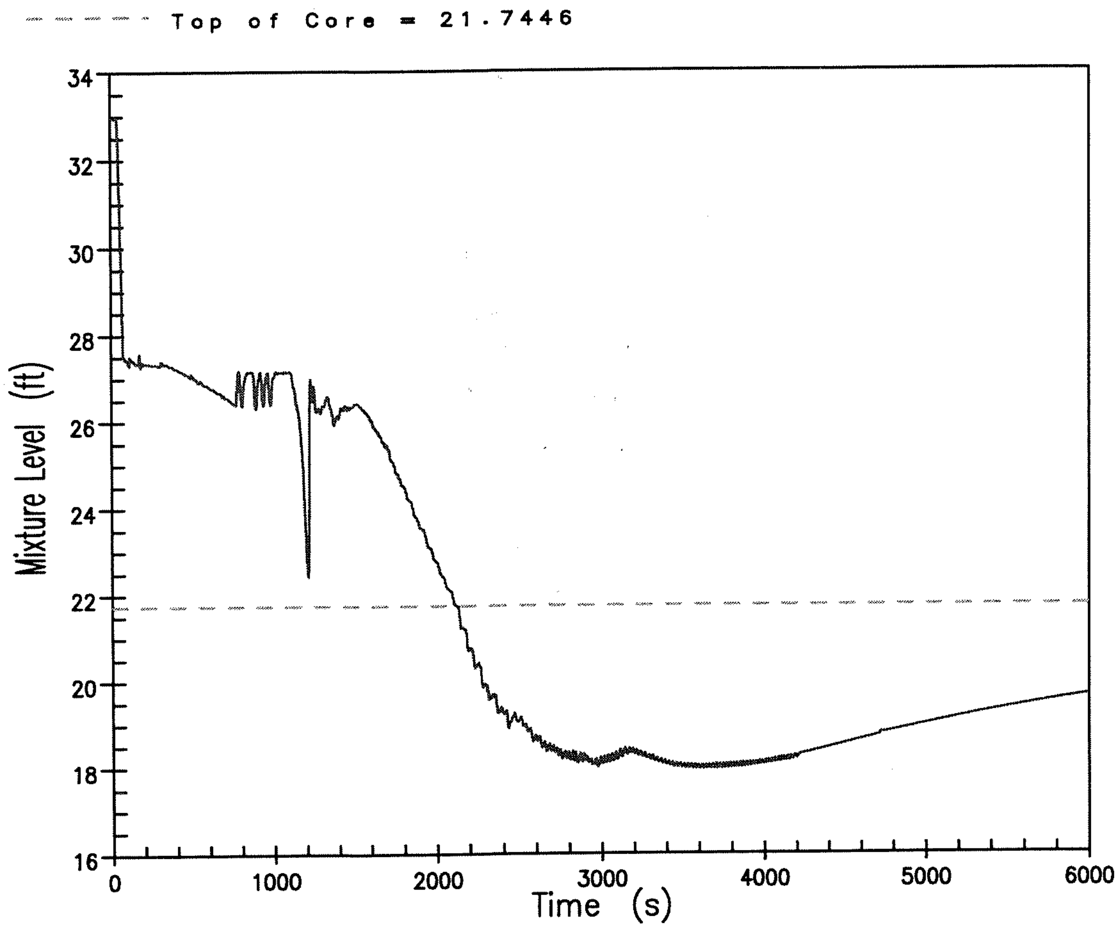
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Reactor Coolant System Pressure (2-inch break) Figure 15.3-10a



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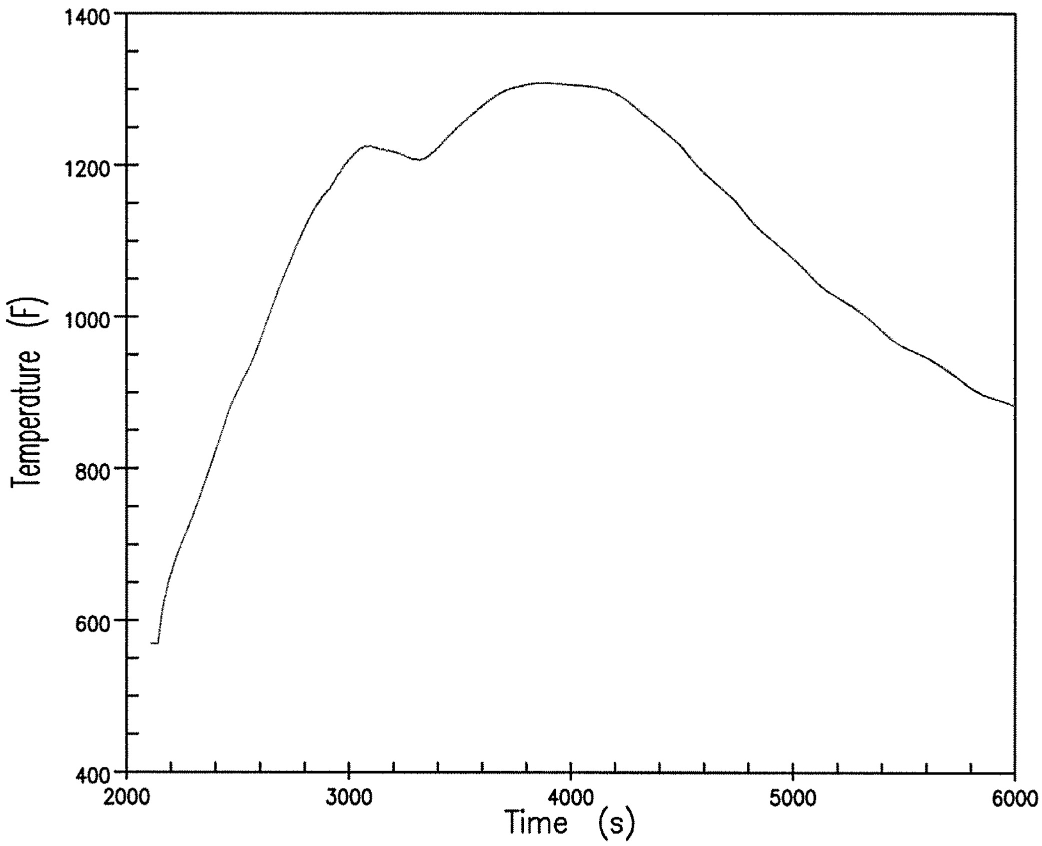
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height

(2-inch break)

Figure 15.3-10b



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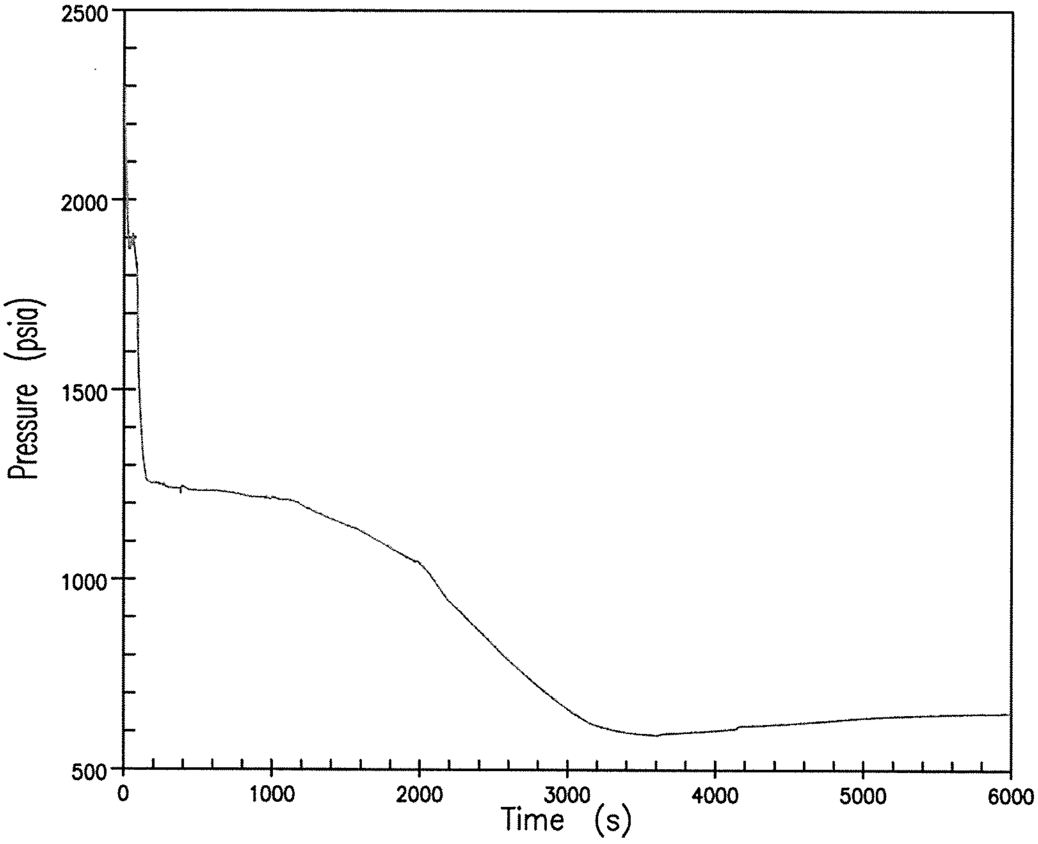
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(2-inch break)

Figure 15.3-10c



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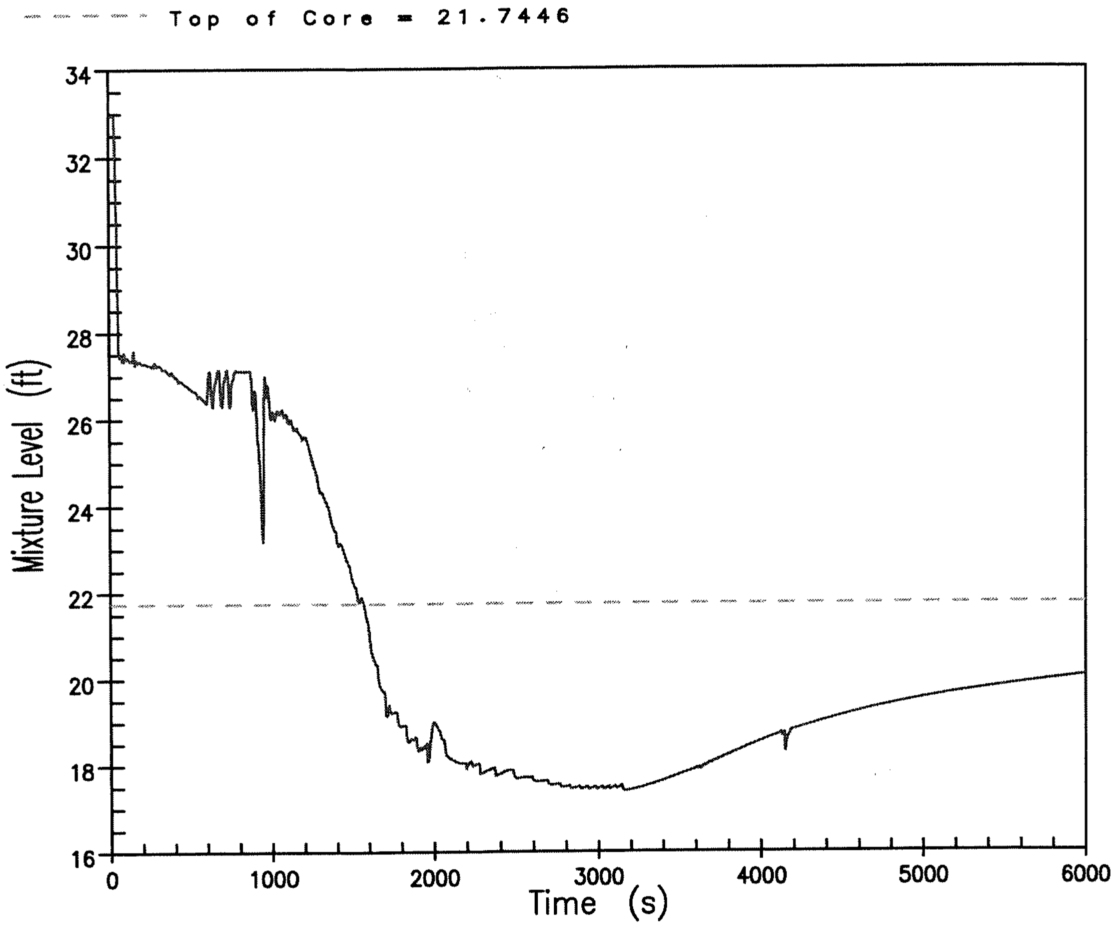
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure

(2.25-inch break)

Figure 15.3-11a



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height

(2.25-inch break)

Figure 15.3-11b

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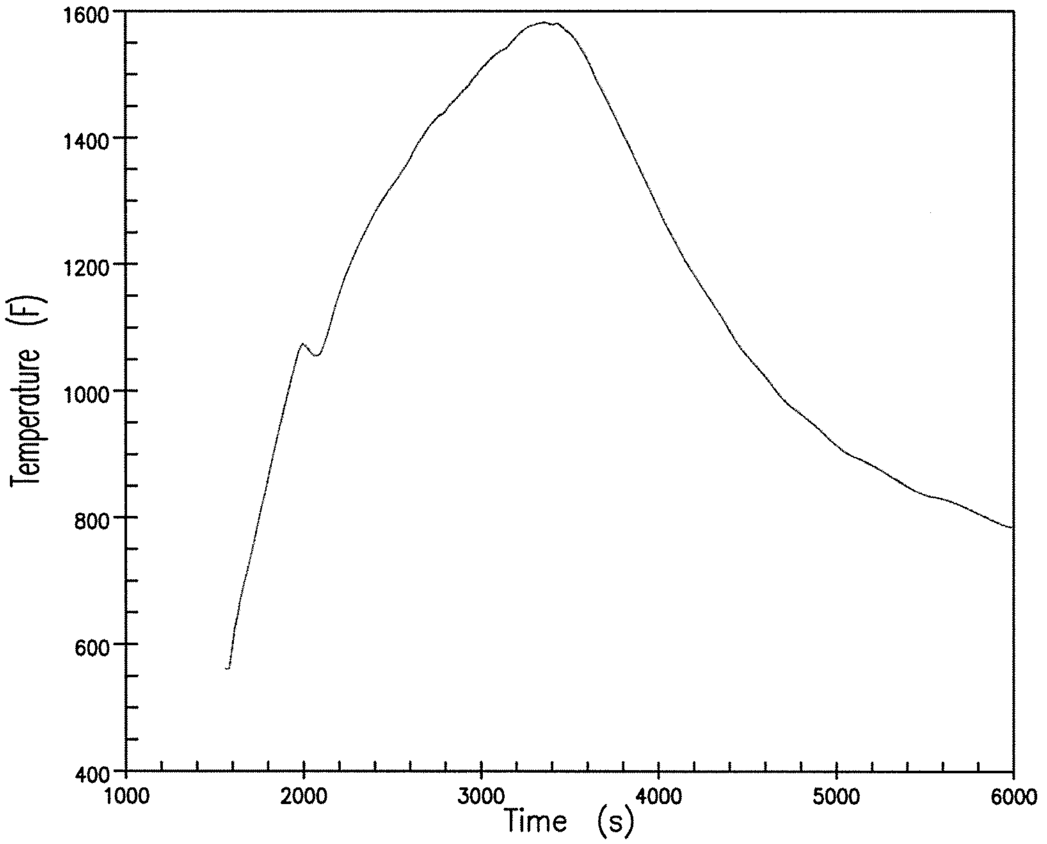

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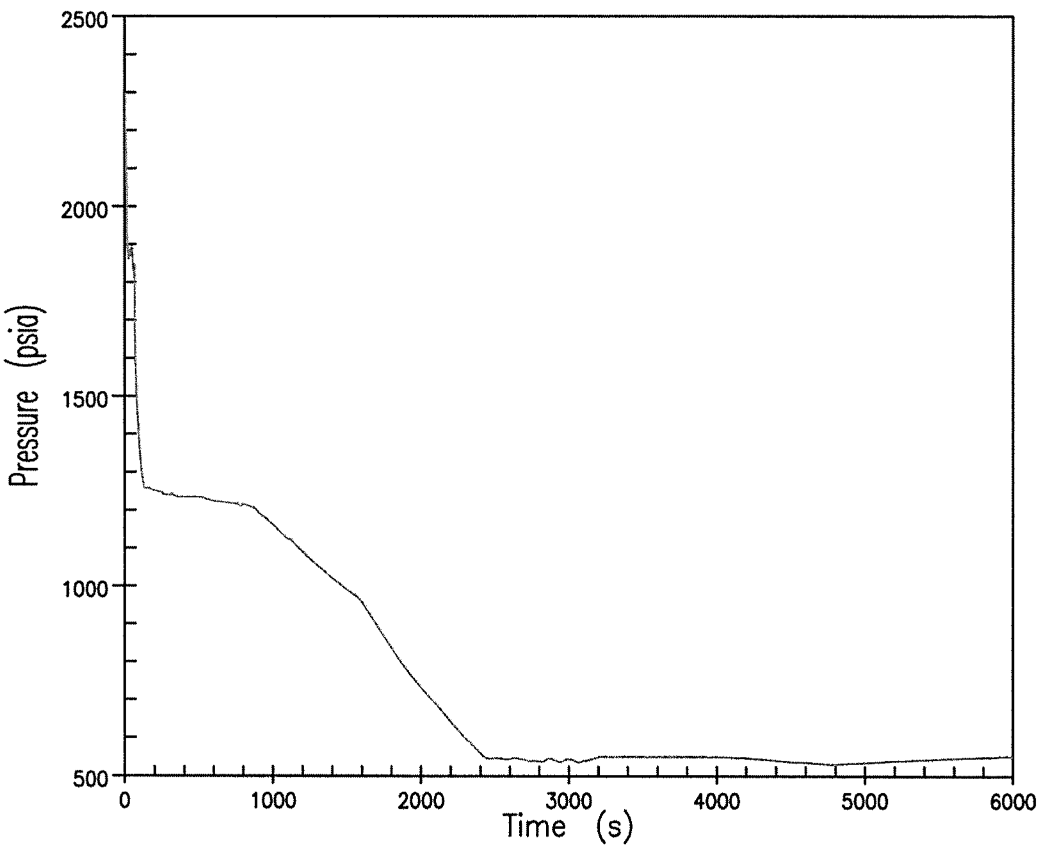
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(2.25-inch break)

Figure 15.3-11c



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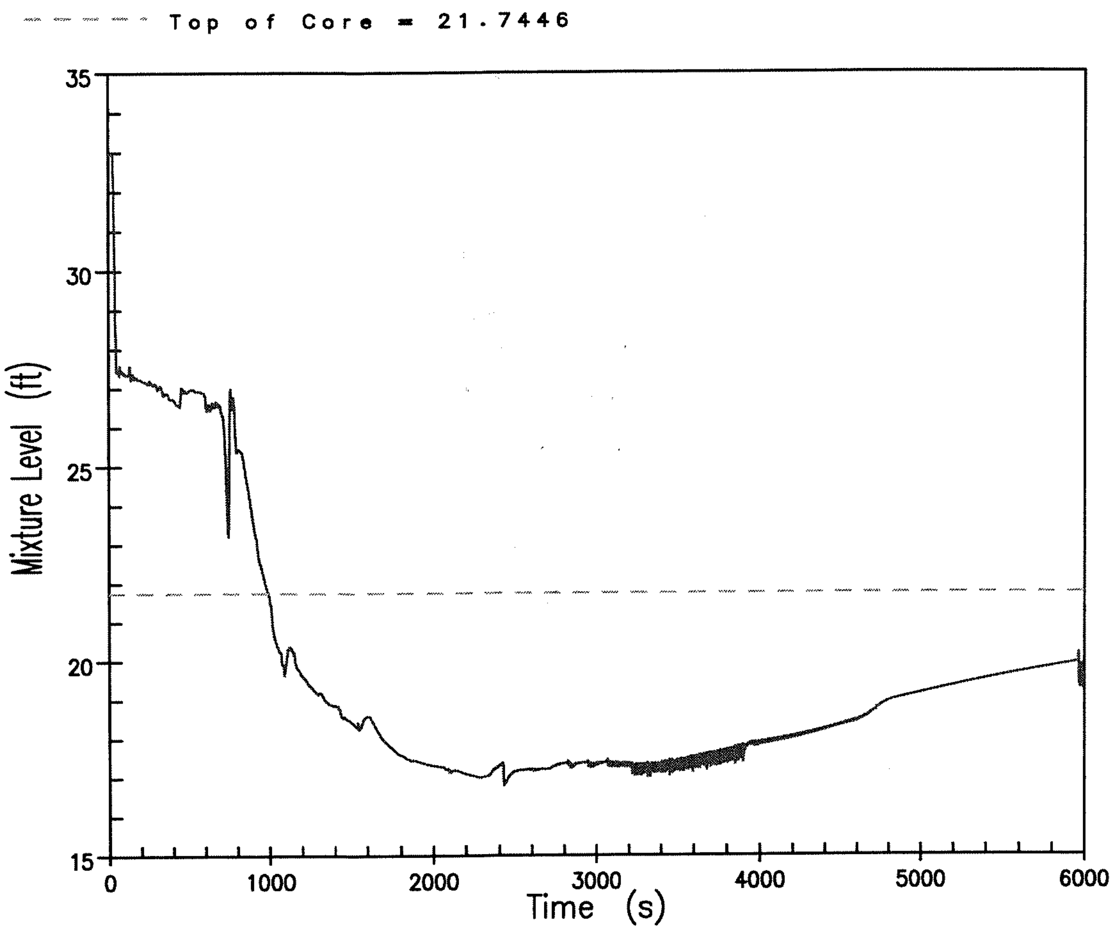
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure

(2.5-inch break)

Figure 15.3-12a



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height

(2.5-inch break)

Figure 15.3-12b

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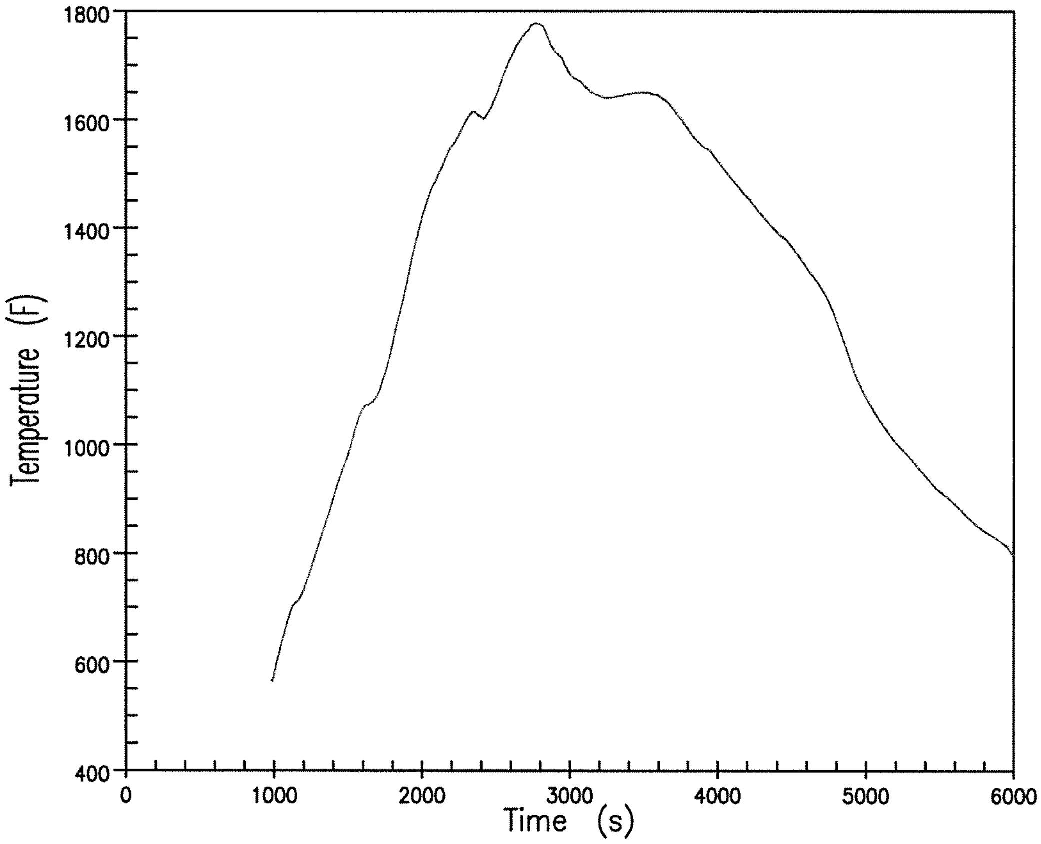

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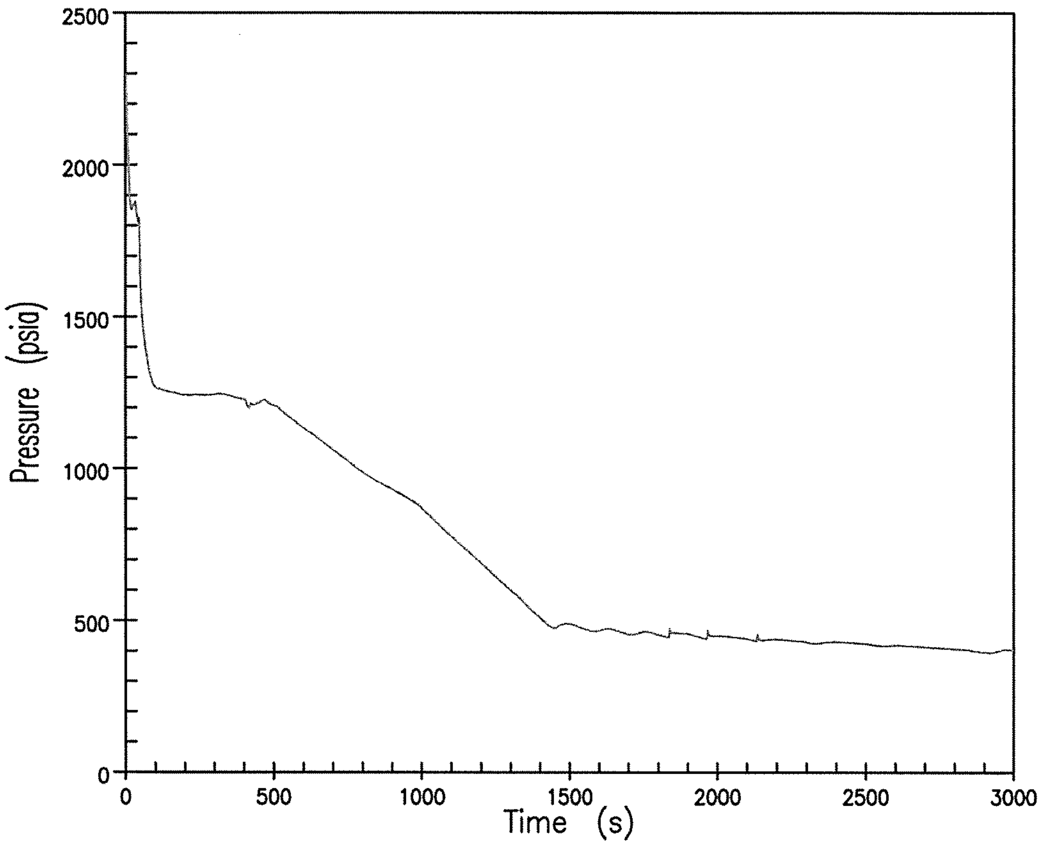
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(2.5-inch break)

Figure 15.3-12c



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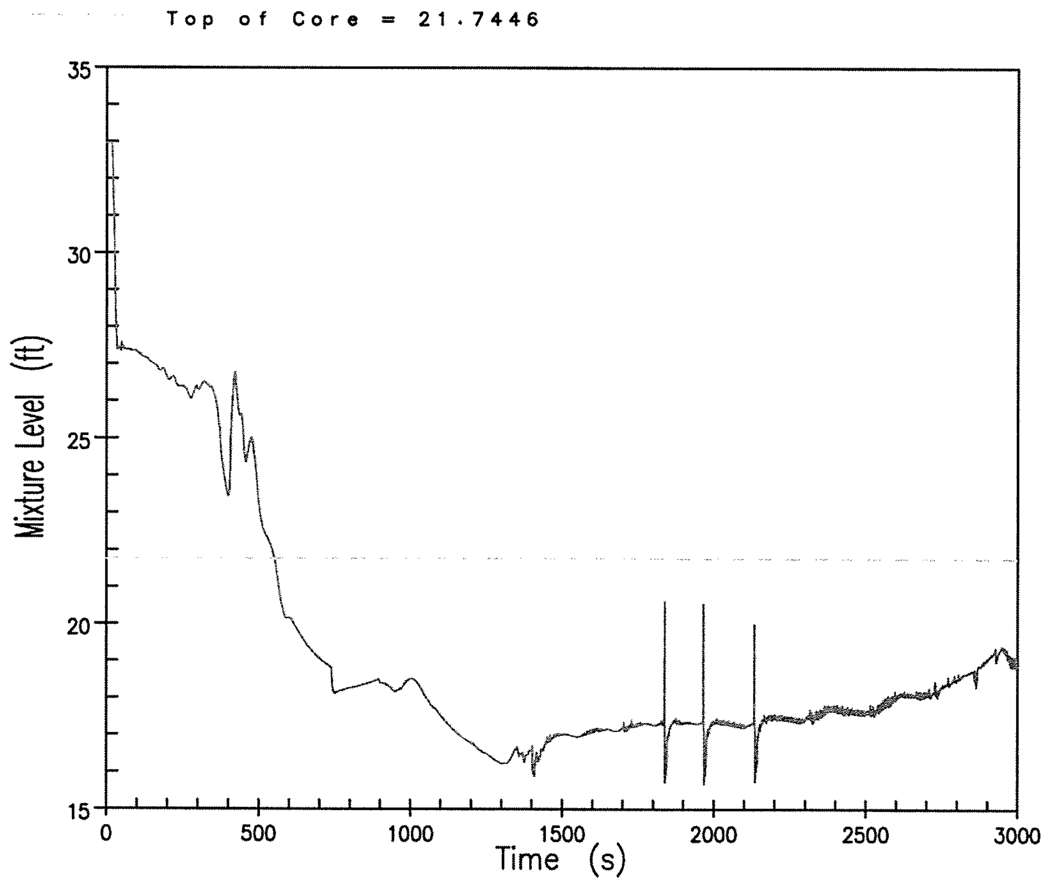
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Reactor Coolant System Pressure (3.0-inch break) Figure 15.3-13a



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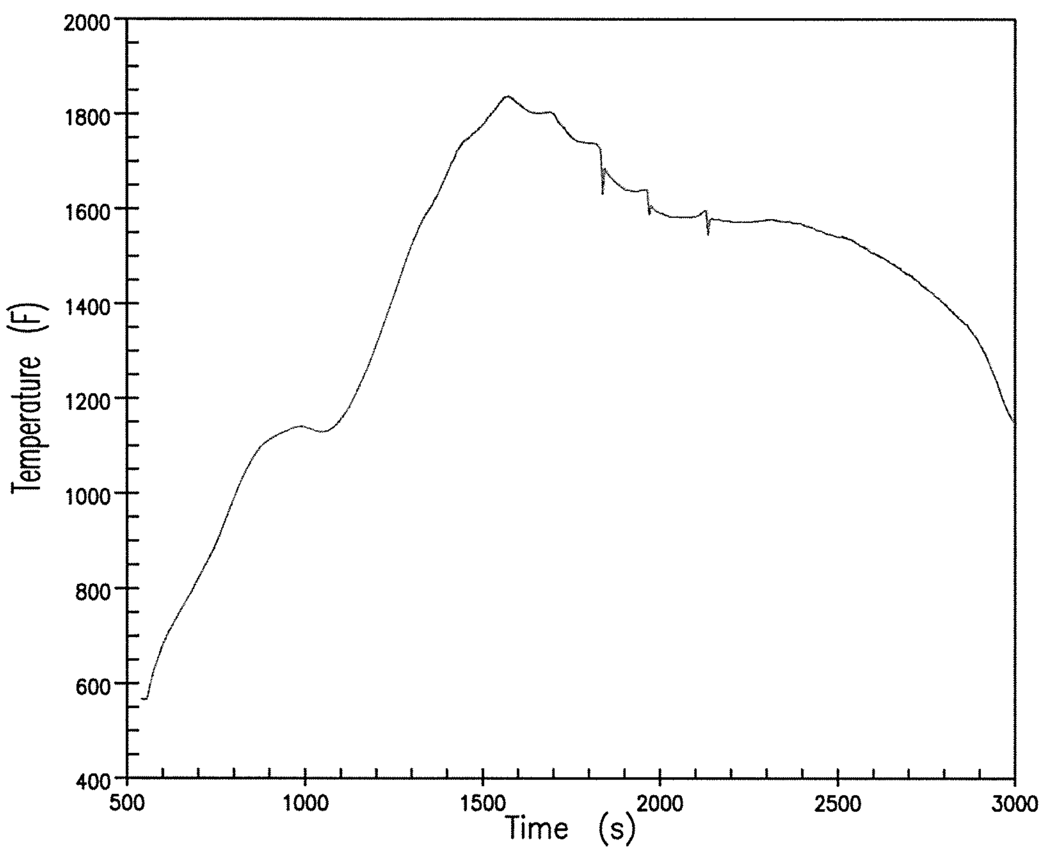
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Core Mixture Height (3.0-inch break) Figure 15.3-13b



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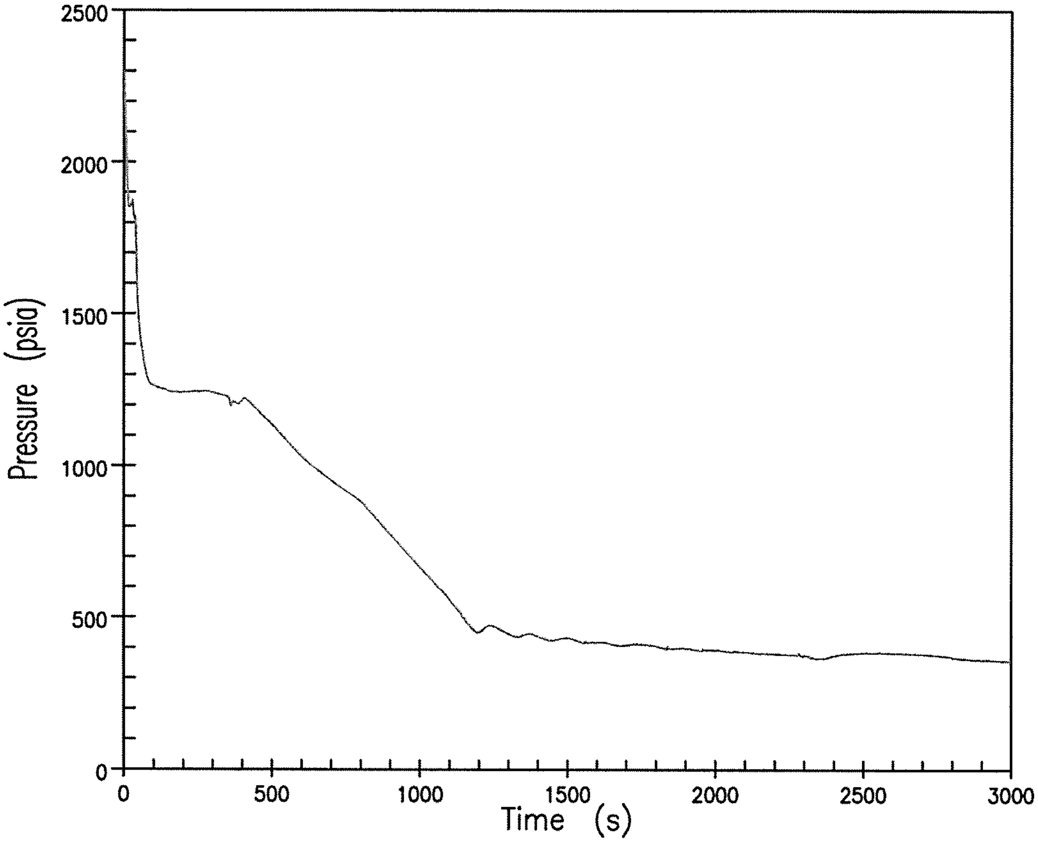
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(3.0-inch break)

Figure 15.3-13c



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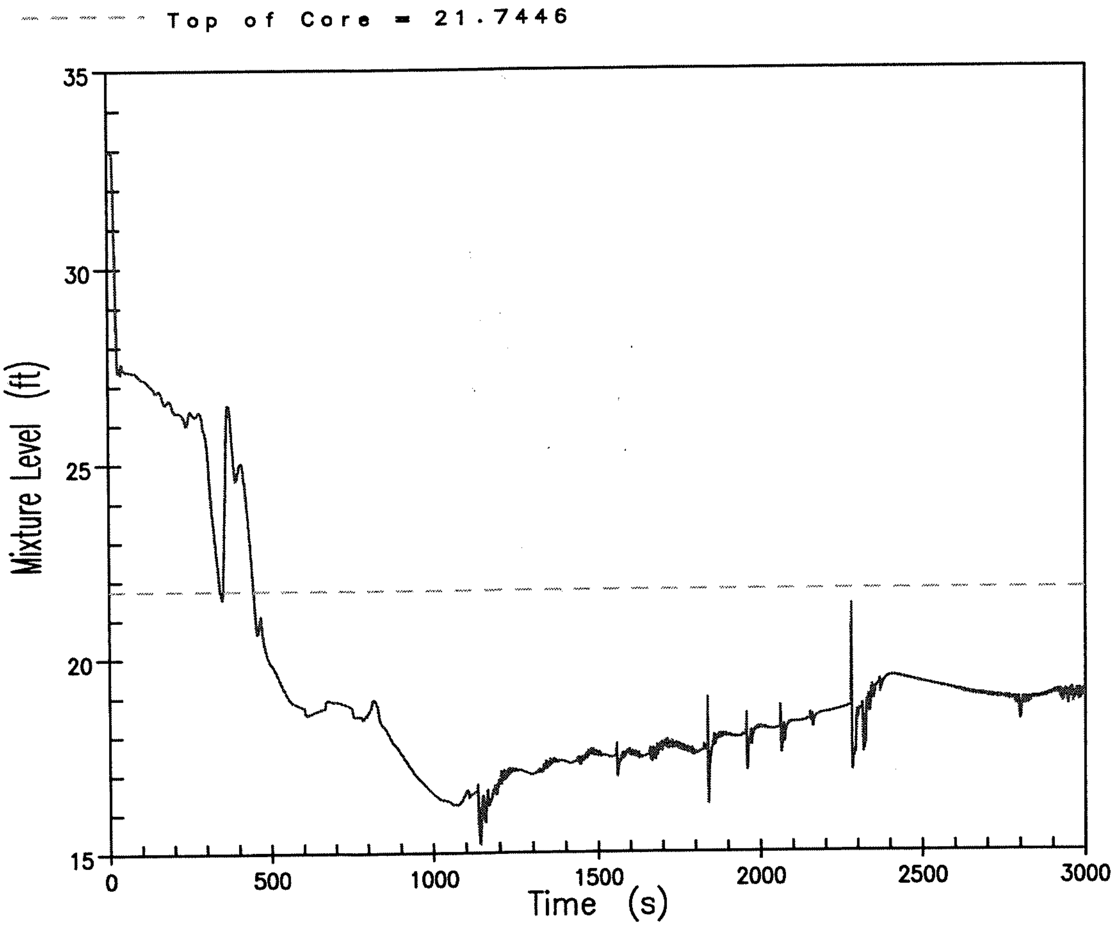
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure

(3.25-inch break)

Figure 15.3-14a



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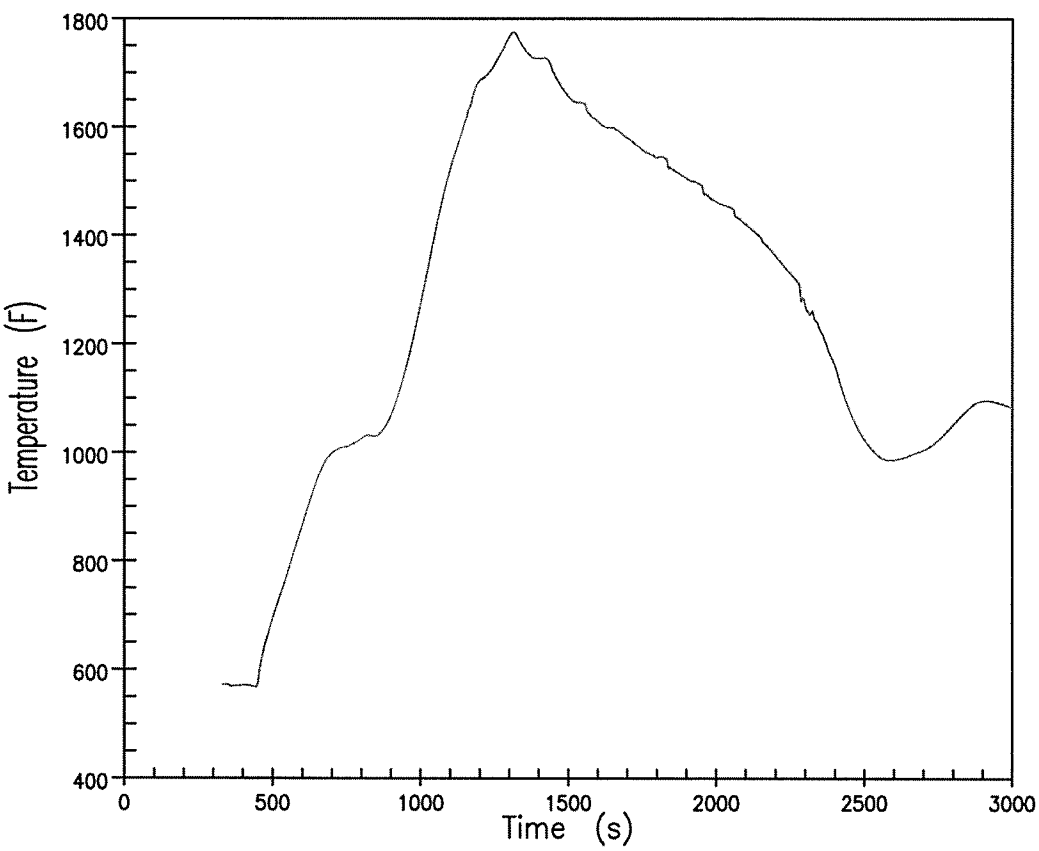
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height

(3.25-inch break)

Figure 15.3-14b



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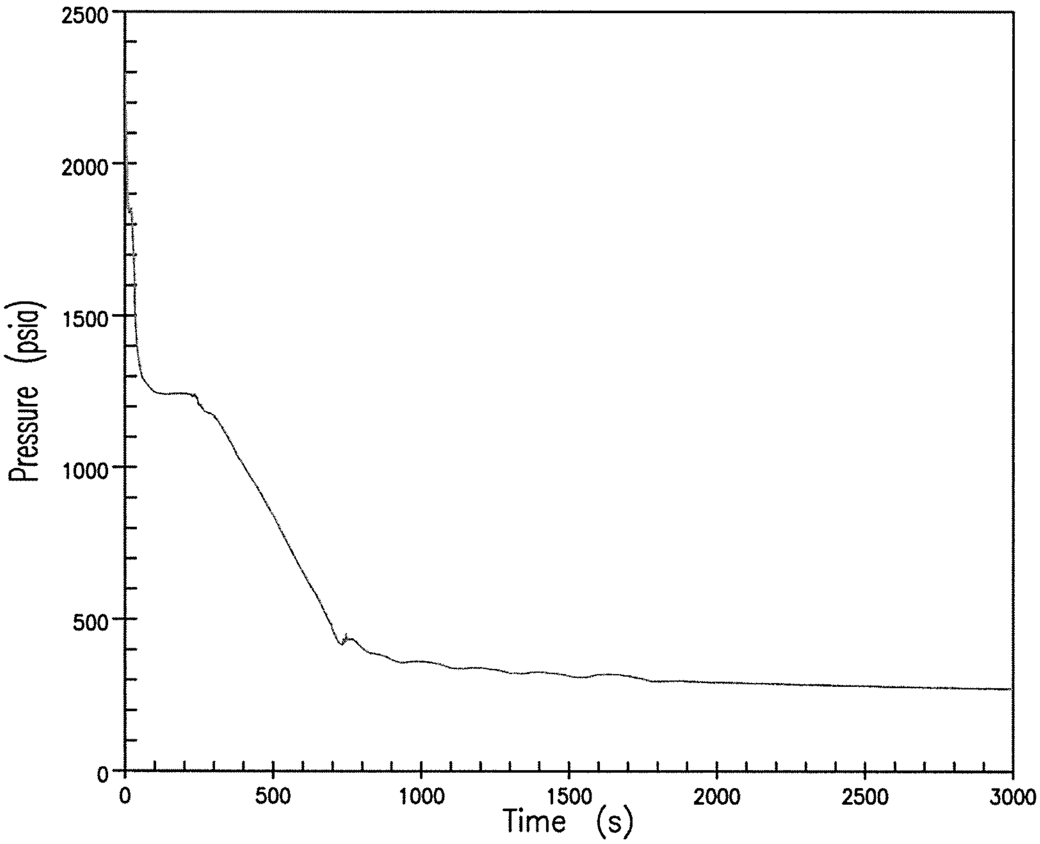
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(3.25-inch break)

Figure 15.3-14c



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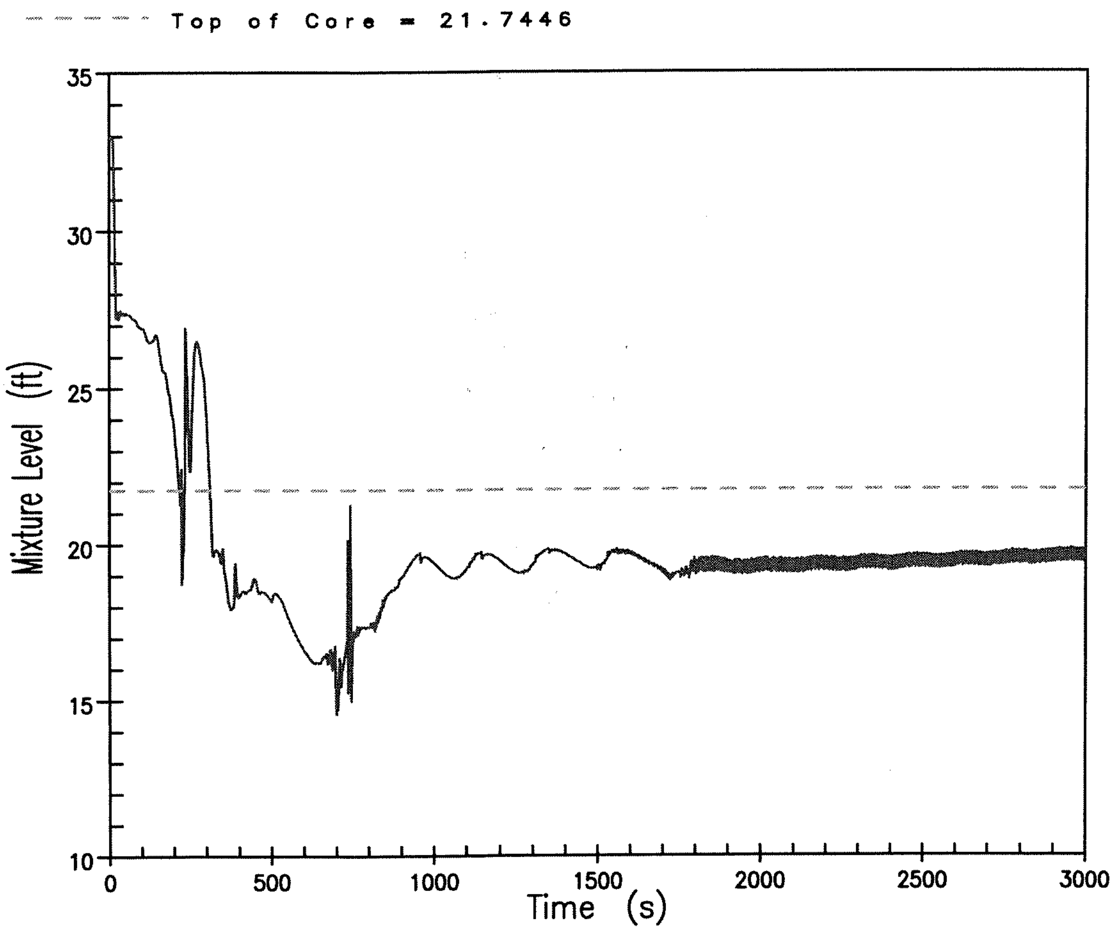
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure

(4.0-inch break)

Figure 15.3-15a



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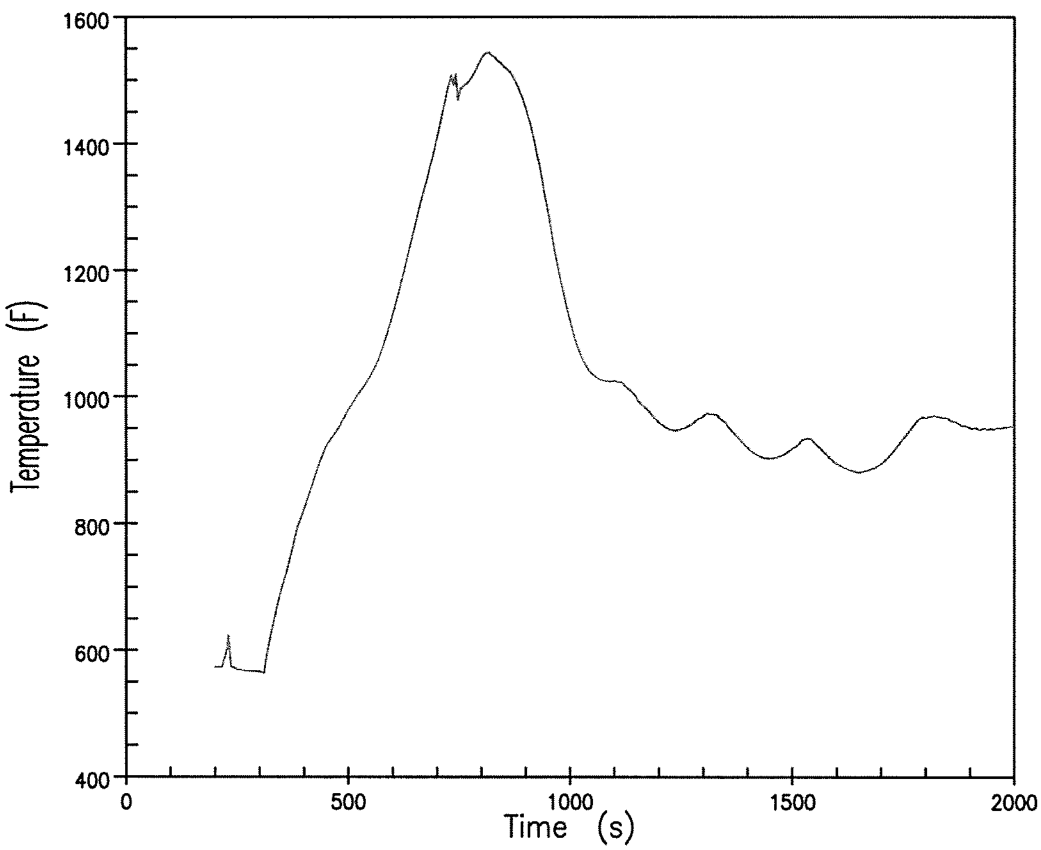
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height

(4.0-inch break)

Figure 15.3-15b



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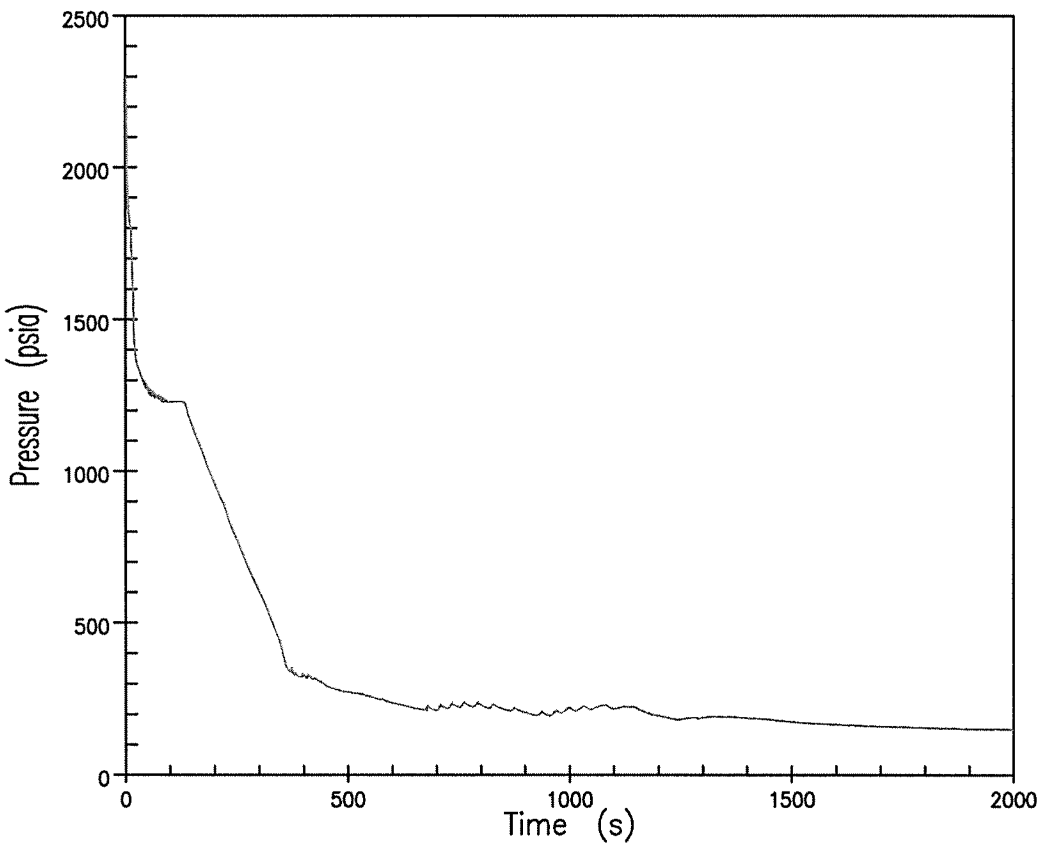
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(4.0-inch break)

Figure 15.3-15c



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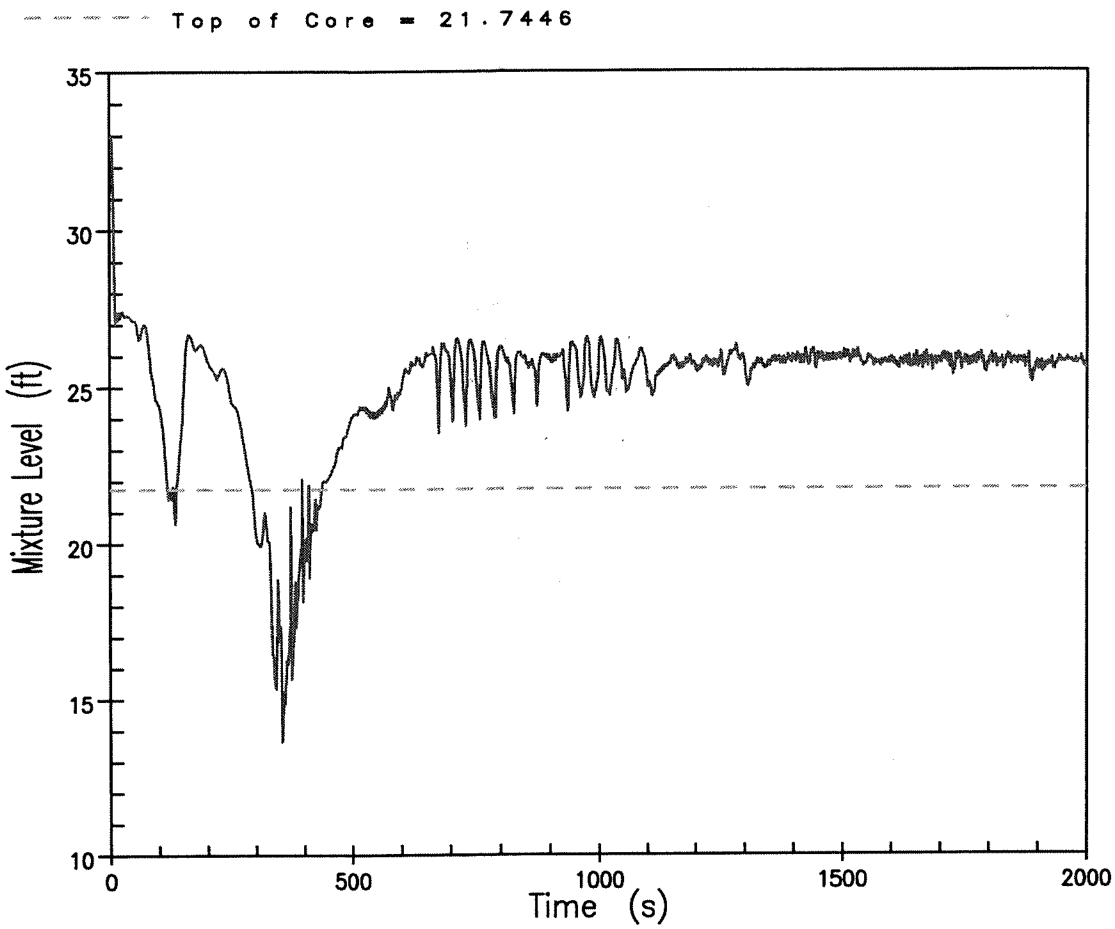
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure

(6.0-inch break)

Figure 15.3-16a



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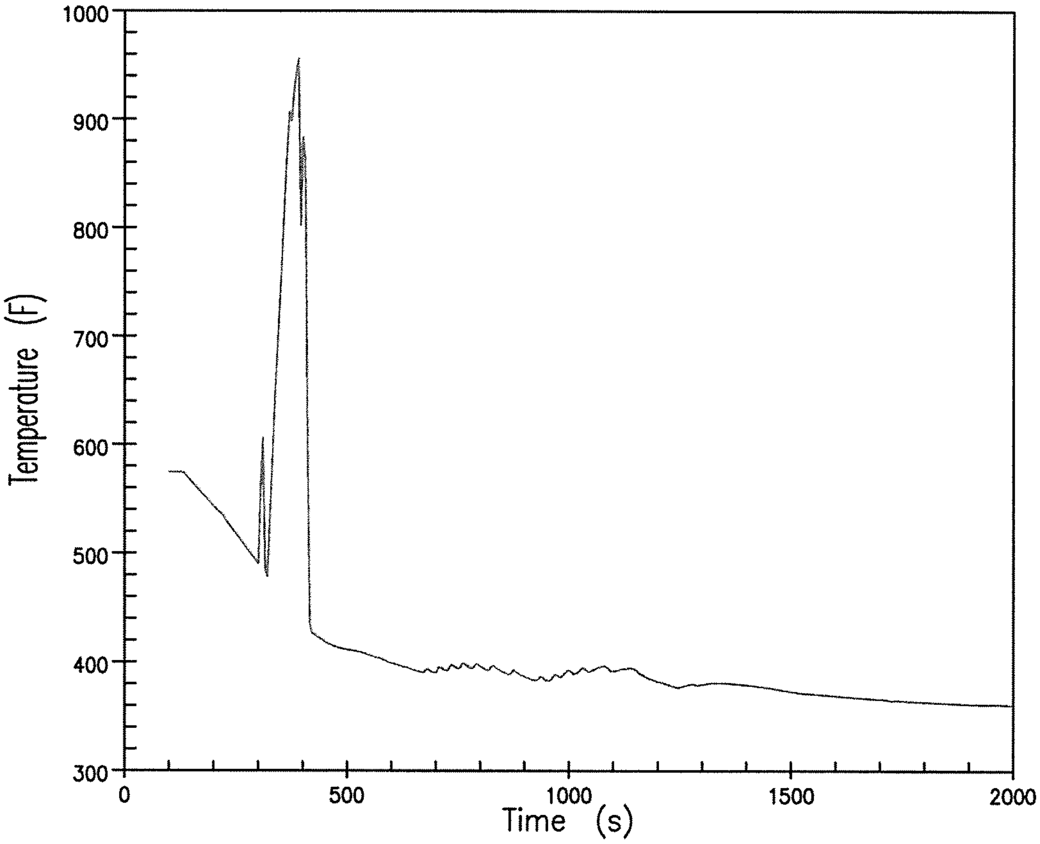
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height

(6.0-inch break)

Figure 15.3-16b



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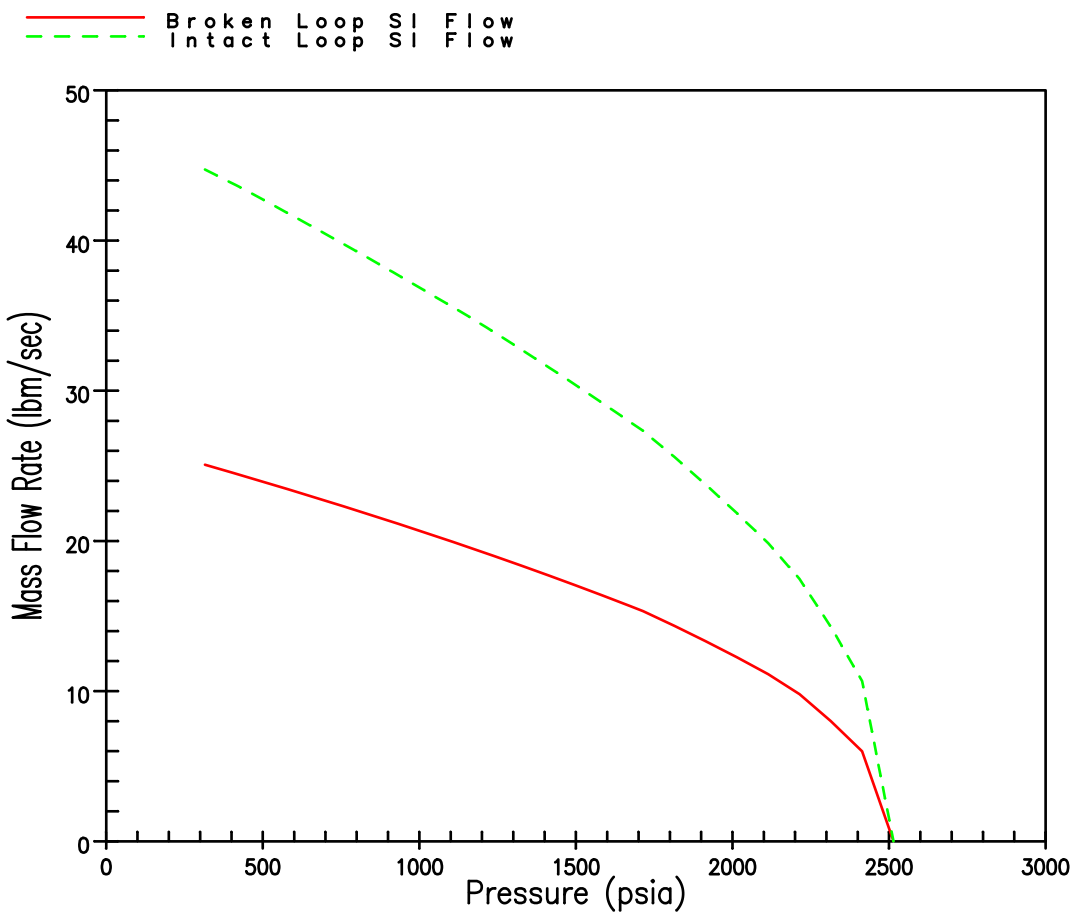
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation

(6.0-inch break)

Figure 15.3-16c



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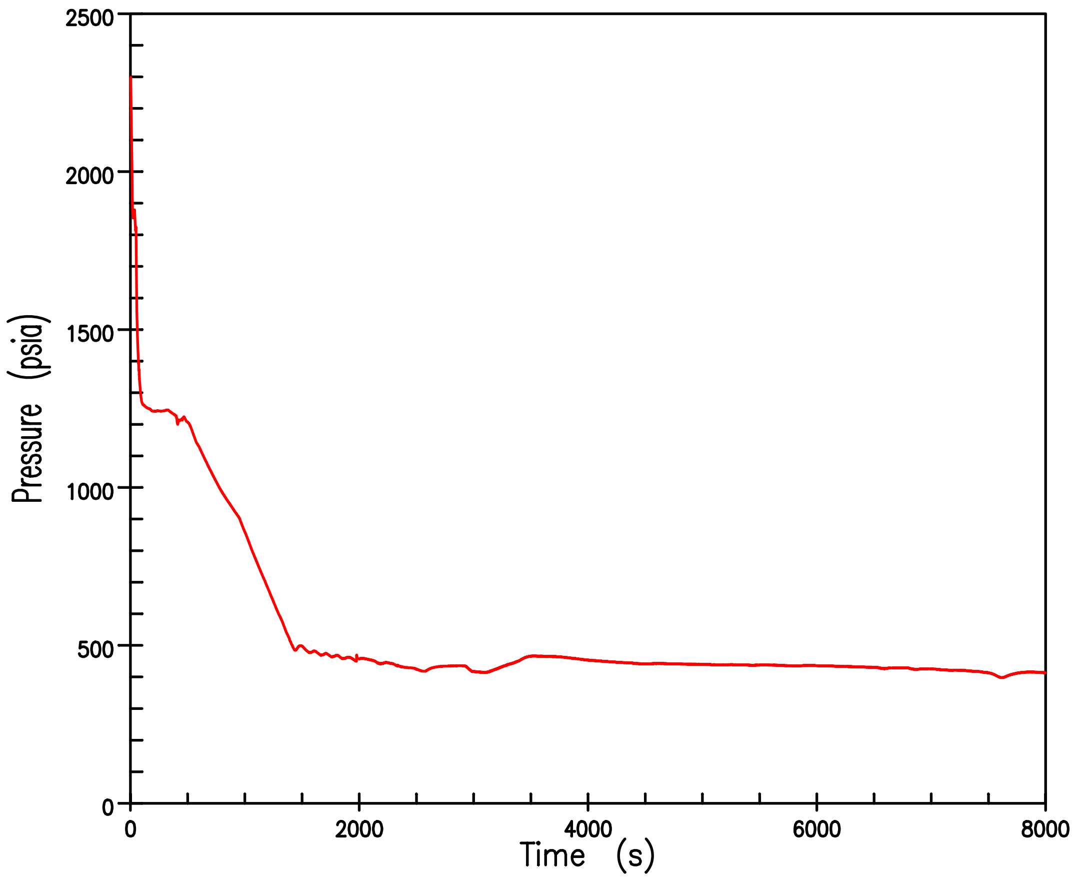
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Pumped Safety Injection Flow Rate
HHSI Throttle Valve Replacement

Figure 15.3-17



SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Reactor Coolant System Pressure HHSI Throttle Valve Replacement (3.0-inch break) Figure 15.3-18a

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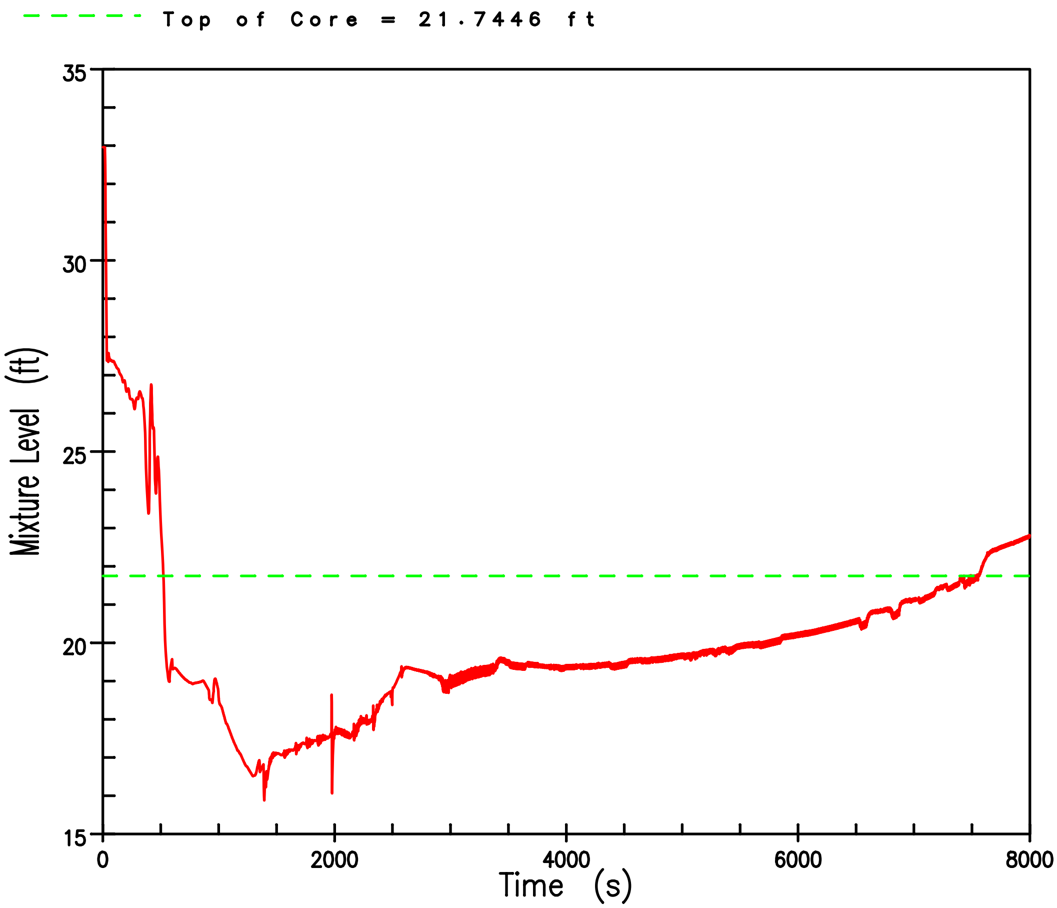

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height
HHSI Throttle Valve Replacement

(3.0-inch break)

Figure 15.3-18b

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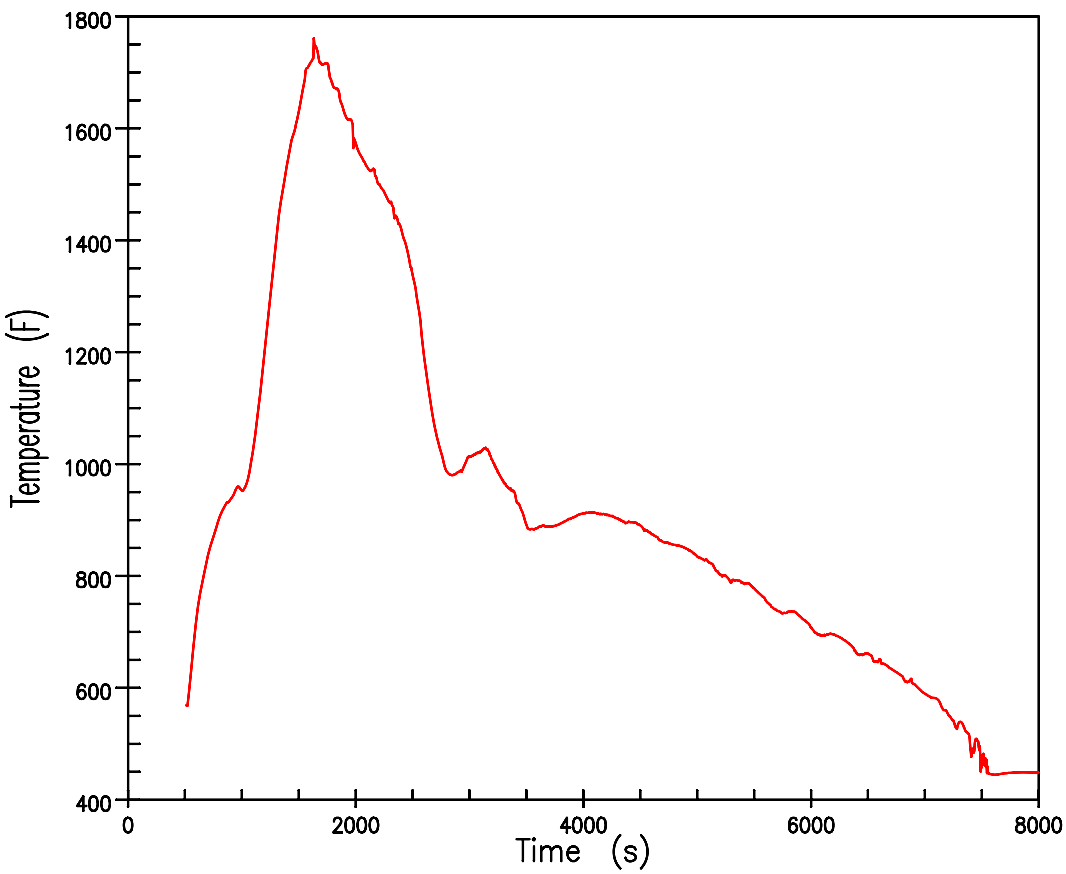

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation
HHSI Throttle Valve Replacement

(3.0-inch break)

Figure 15.3-18c

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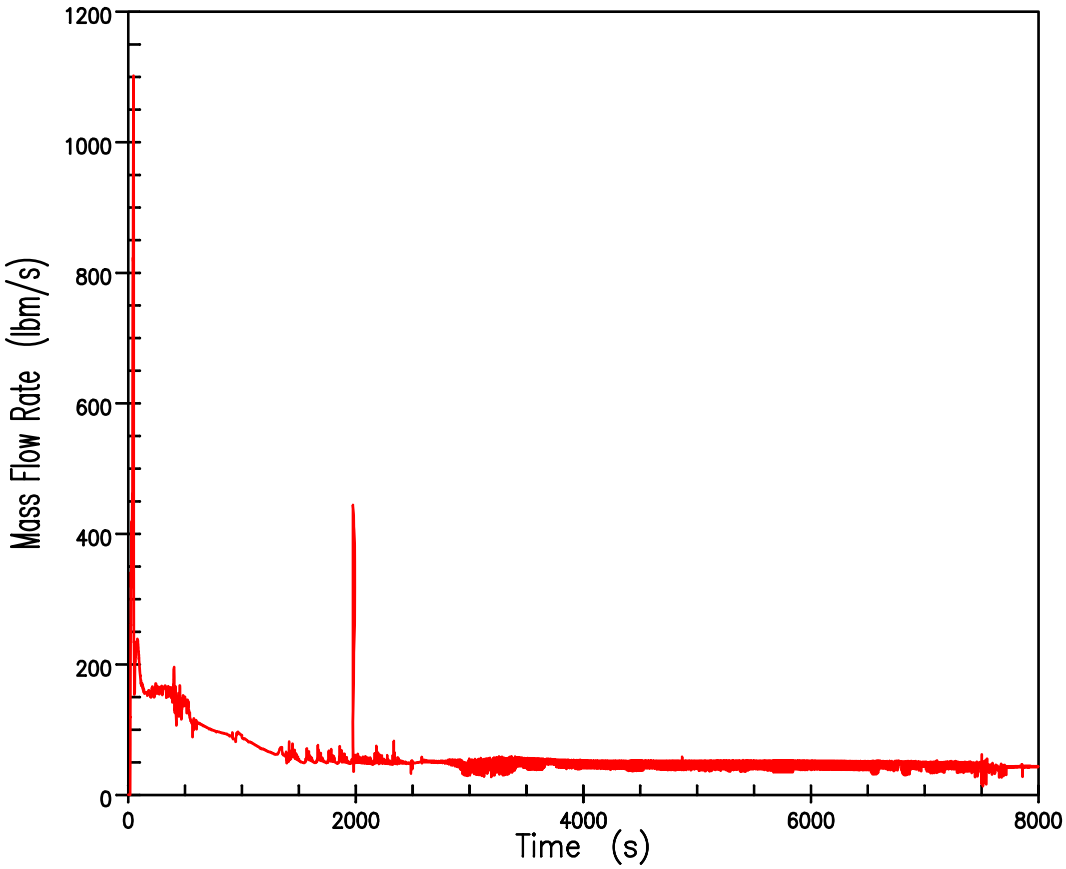

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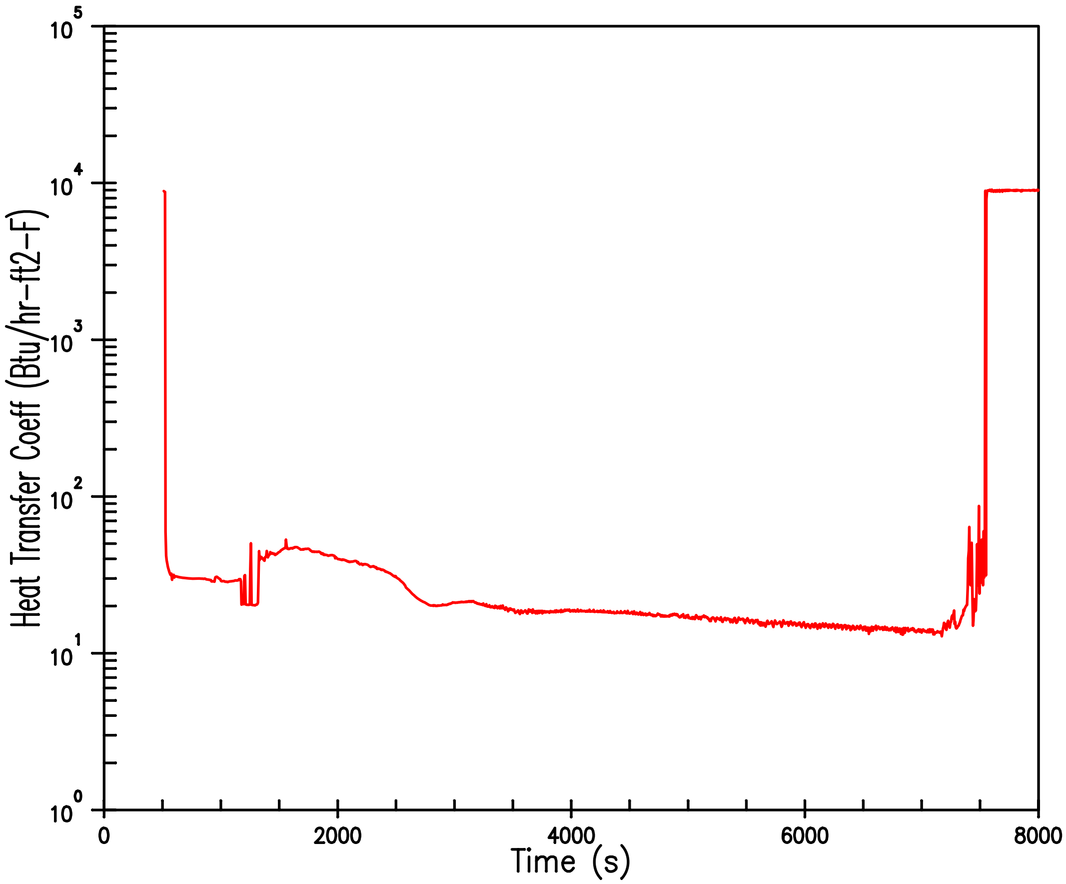
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Mass Flow Rate Out Top of Core
HHSI Throttle Valve Replacement

(3.0-inch break)

Figure 15.3-18d



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Surface Heat Transfer Coefficient at
Peak Temperature Elevation
HHSI Throttle Valve Replacement

(3.0-inch break)

Figure 15.3-18e

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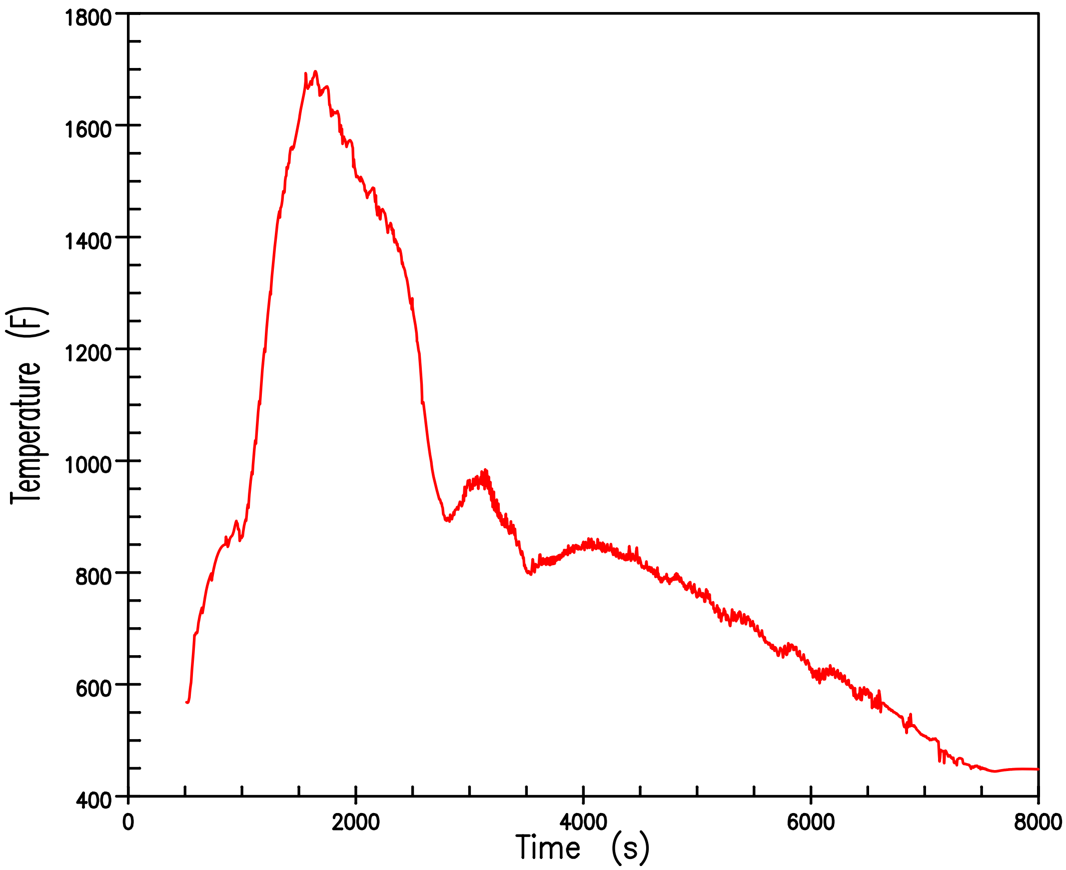

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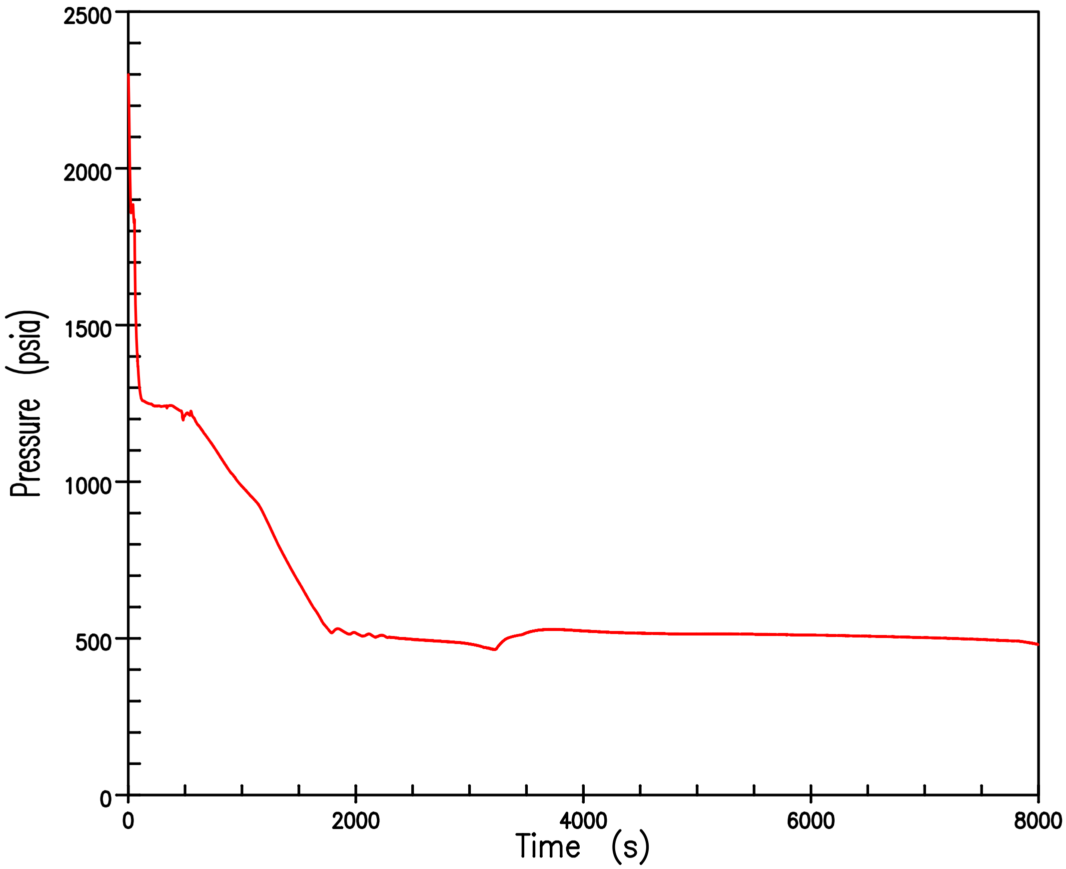
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Fluid Temperature at Peak Clad
Temperature Elevation
HHSI Throttle Valve Replacement

(3.0-inch break)

Figure 15.3-18f



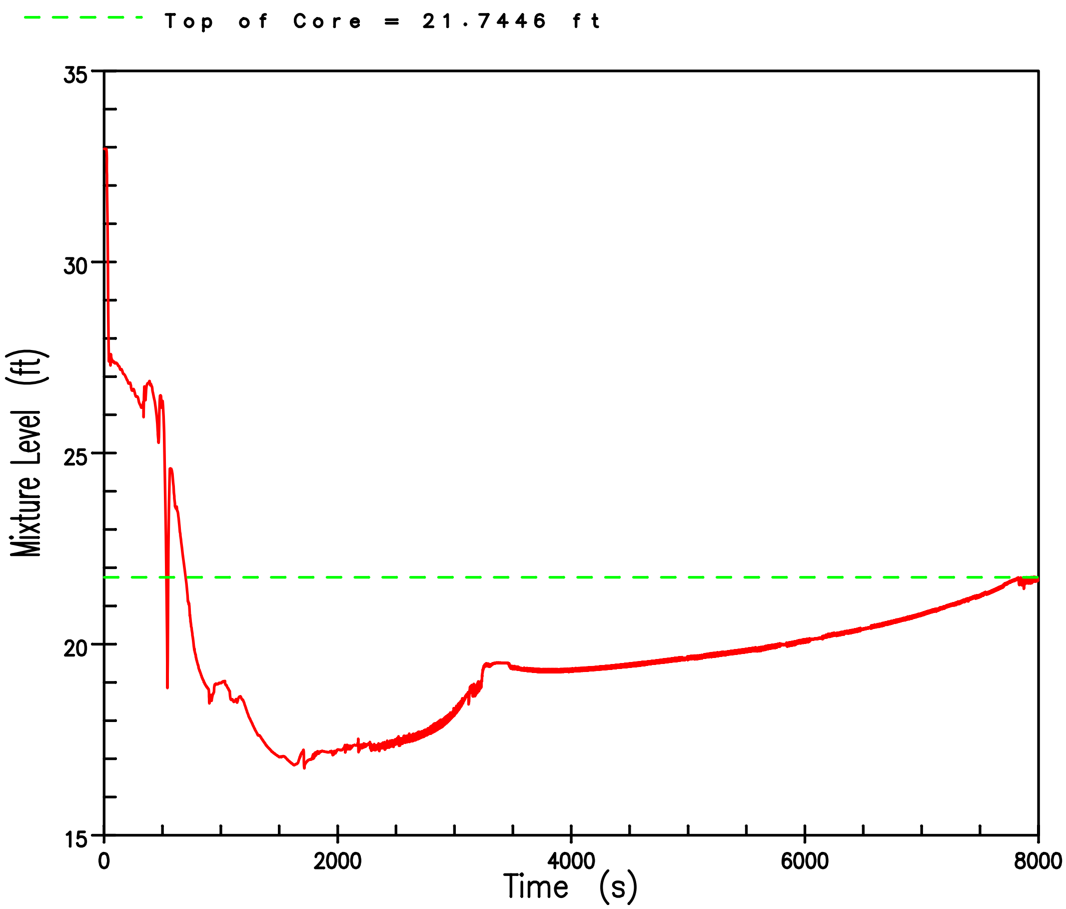
RN 06-019
February 2007

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure
HHSI Throttle Valve Replacement

(2.75-inch break)

Figure 15.3-19a



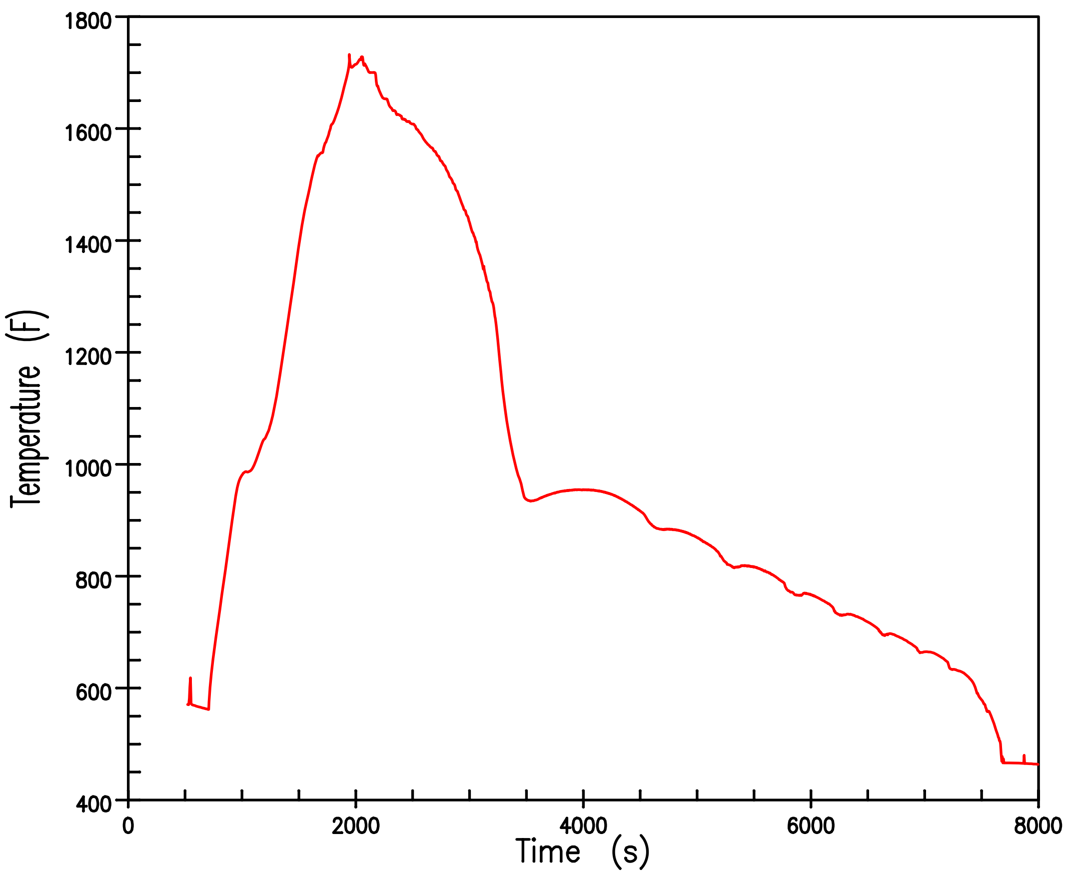
RN 06-019
February 2007

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height
HHSI Throttle Valve Replacement

(2.75-inch break)

Figure 15.3-19b



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Temperature Transient at Peak Temperature Elevation
HHSI Throttle Valve Replacement

(2.75-inch break)

Figure 15.3-19c

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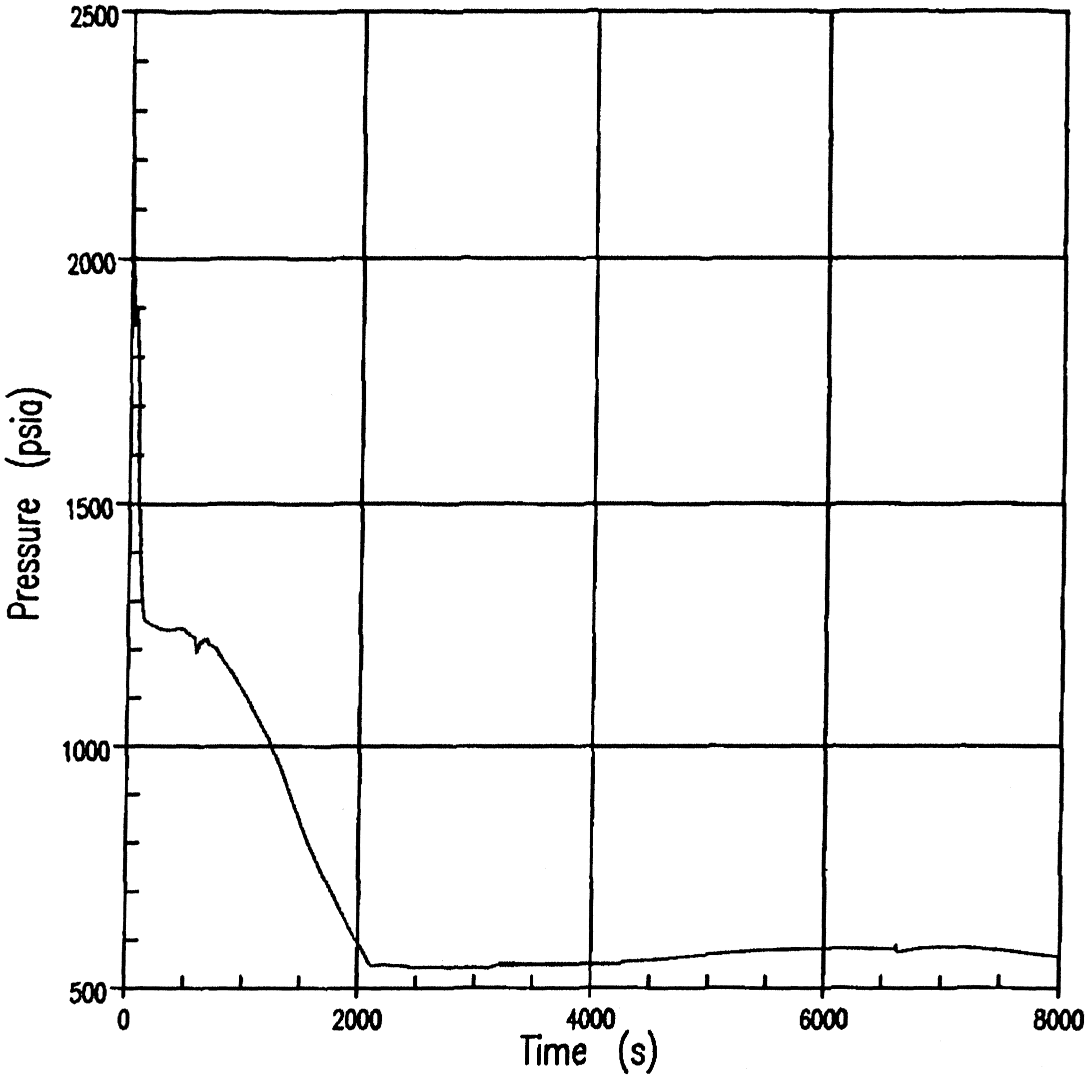
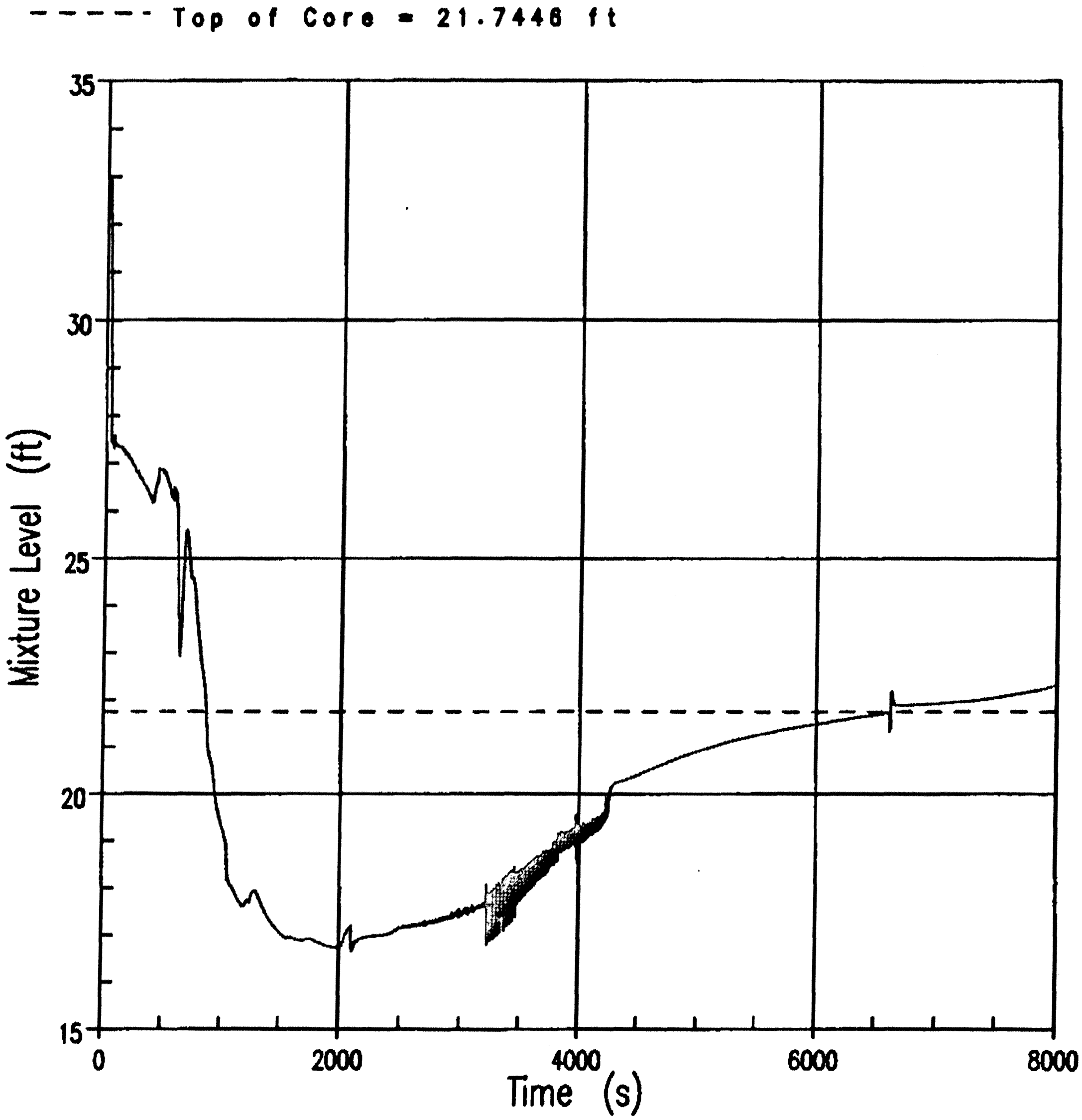


RN 09-022

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Coolant System Pressure
Upflow Conversion
(2.50-Inch Break)

Figure 15.3-19d

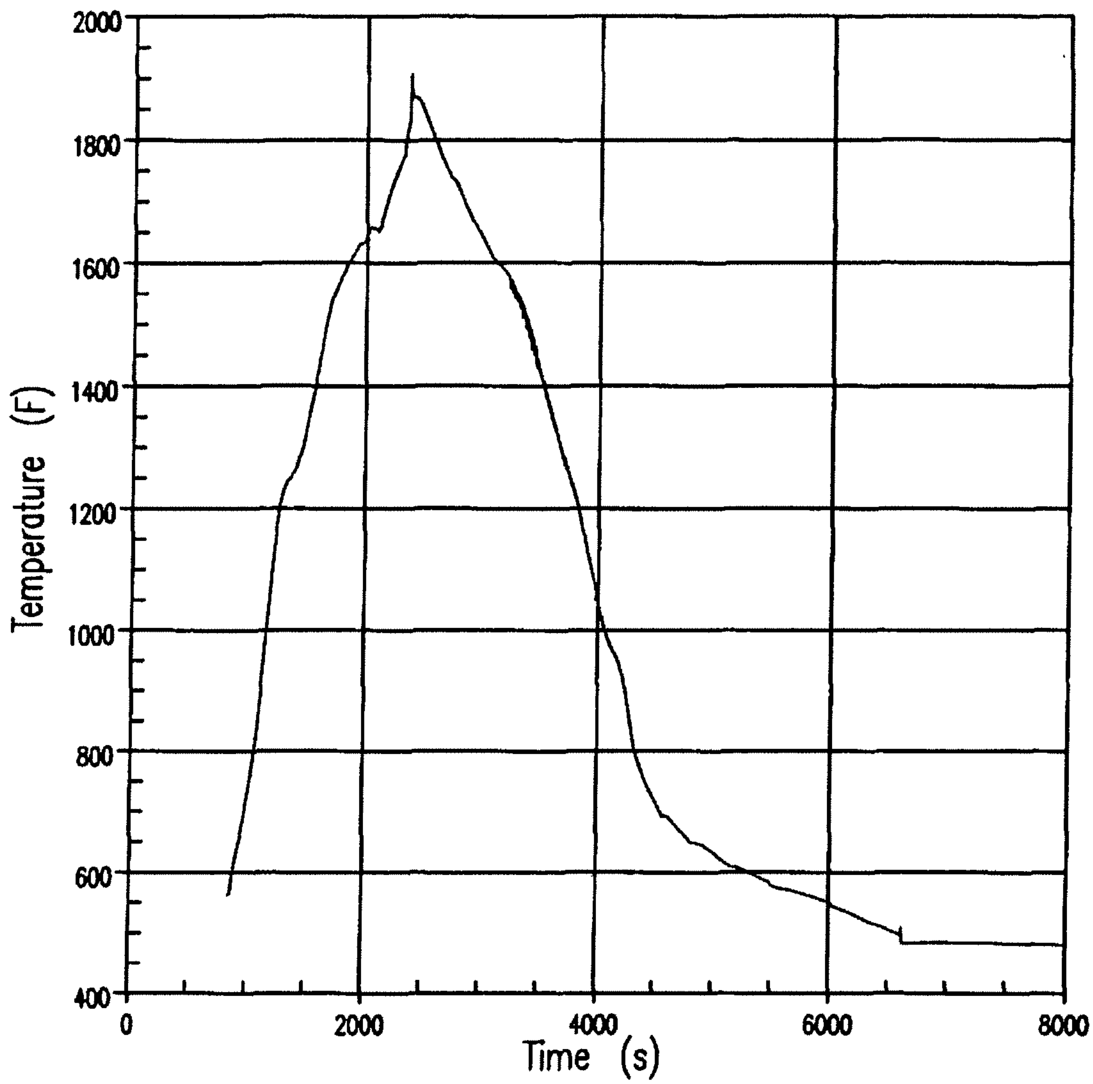


RN 09-022

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Core Mixture Height
Upflow Conversion
(2.50-Inch Break)

Figure 15.3-19e

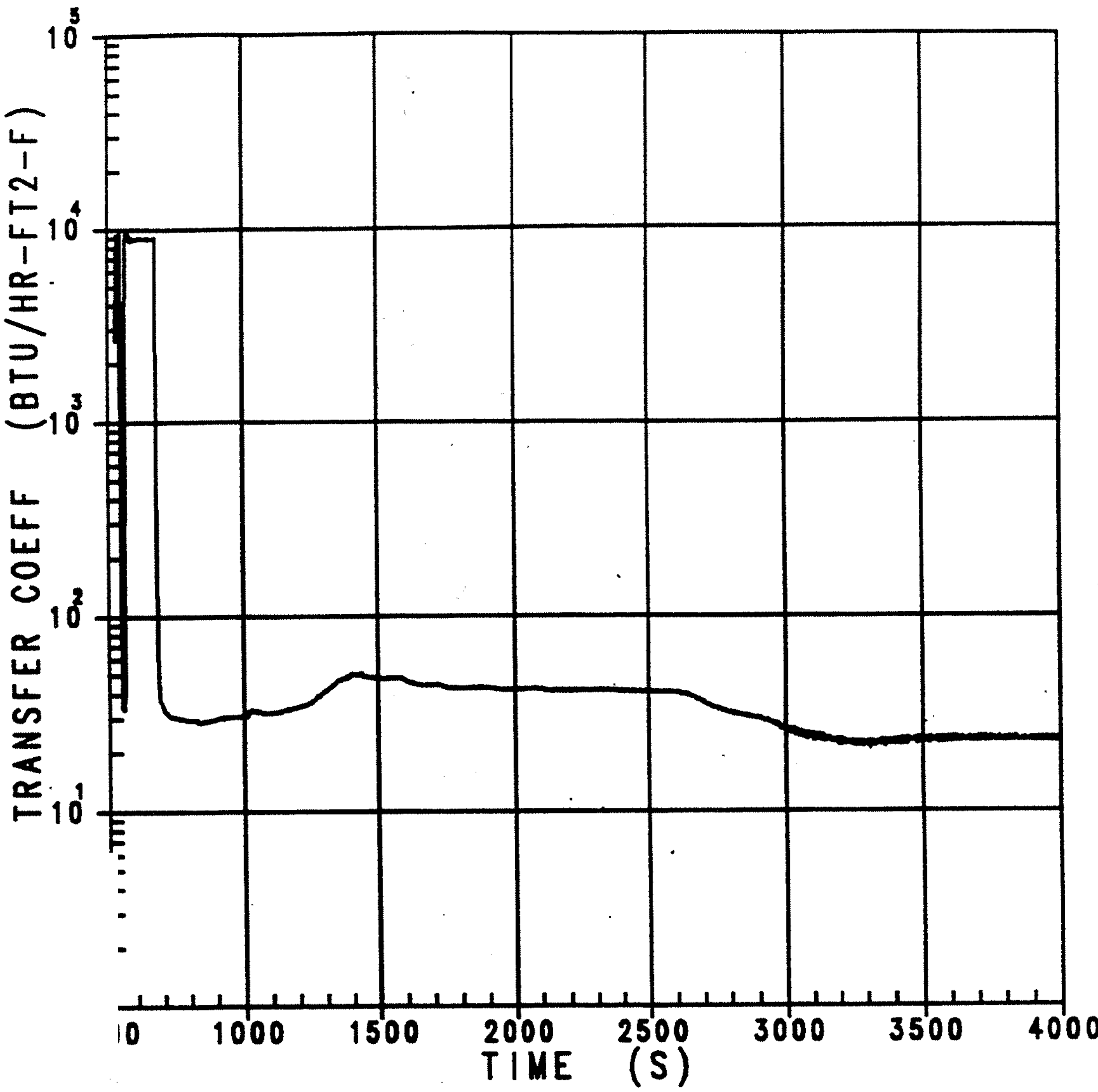


RN 09-022

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Cladding Temperature at Peak Cladding Temperature Elevation
Upflow Conversion
(2.50-Inch Break)

Figure 15.3-19f

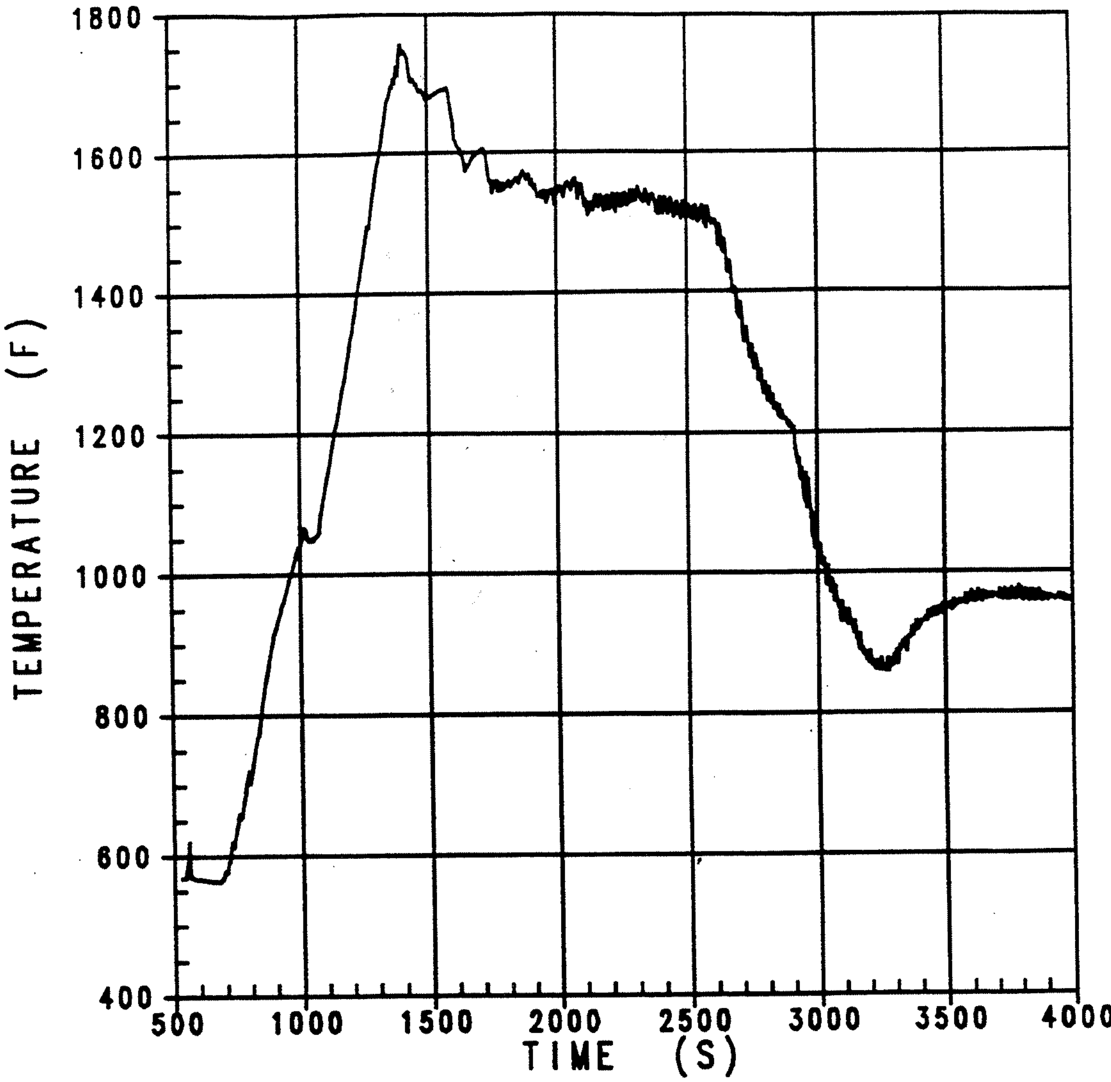


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Clad Surface Heat Transfer Coefficient
at Peak Temperature Elevation
(3-inch break)

Figure 15.3-20

AMENDMENT 96-02
JULY 1996

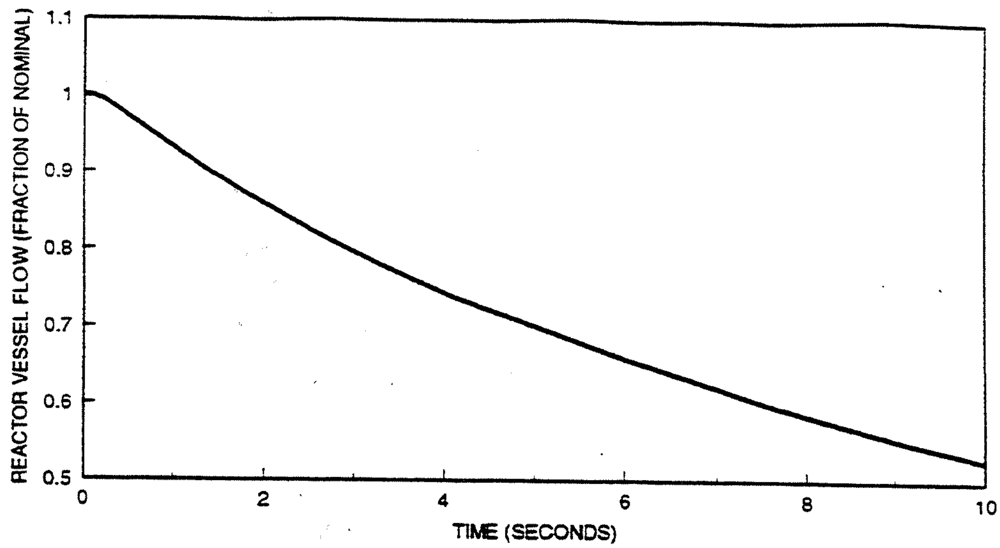


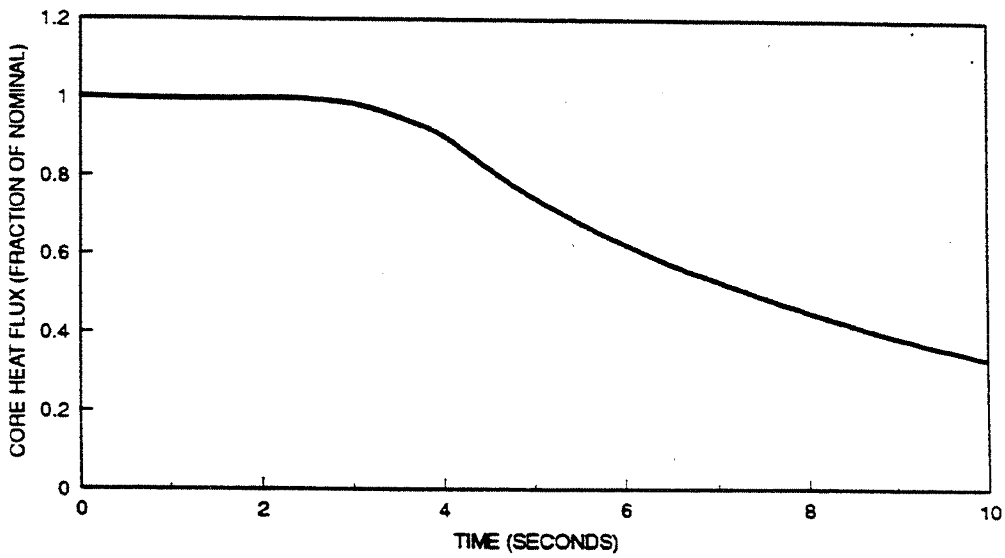
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Fluid Temperature
at Peak Clad Temperature Elevation
(3-inch break)

Figure 15.3-21

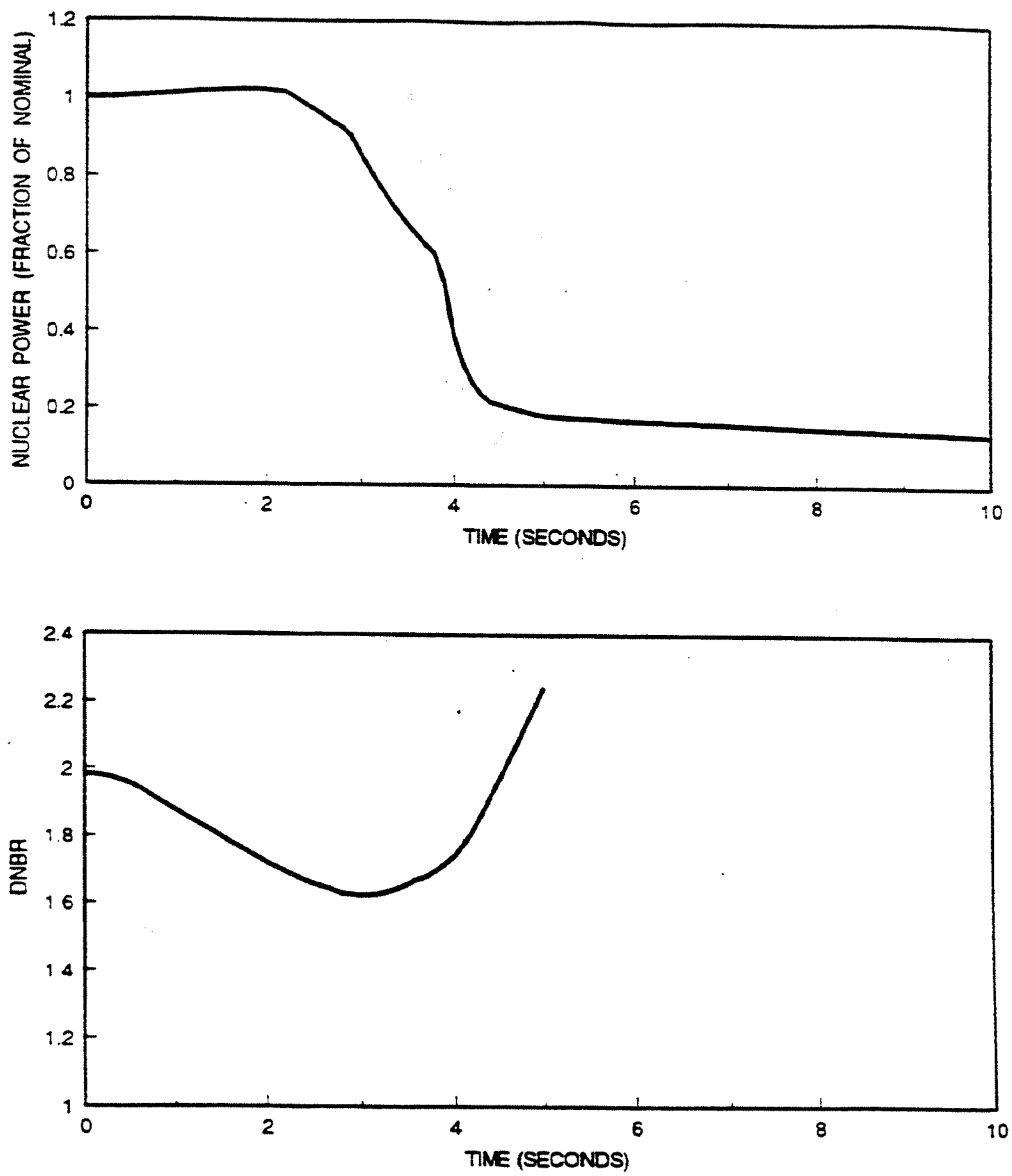
AMENDMENT 96-02
JULY 1996





AMENDMENT 96-02
JULY 1996

SOUTH CAROLINA ELECTRIC & GAS CO. **VIRGIL C. SUMMER NUCLEAR STATION**
All Loops Operating, All Loops Coasting Down: Vessel Flow and Core Heat Flux vs. Time **Figure 15.3-27**



AMENDMENT 96-02
JULY 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

All Loops Operating, All Loops Coasting Down: Nuclear Power and DNBR vs. Time

Figure 15.3-28

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. They are the most drastic which must be designed against and 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 50.67. 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 and the containment. For the purposes of this report the following faults have been classified in this category: RN 12-034

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

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 drive mechanism housing (rod cluster control assembly ejection).

Each of these six limiting faults is analyzed in Section 15.4 in accordance with USNRC Regulatory Guide 1.183 guidelines. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions. RN 12-034

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS-OF-COOLANT-ACCIDENT)

The analysis specified by 10 CFR 50.46 (Reference 1), "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors", is presented in this section. The results of the Best-Estimate large break loss-of-coolant accident (LOCA) analysis are summarized in Table 15.4-1f, and show compliance with the acceptance criteria. The results for the small break loss-of-coolant accident are presented in Section 15.3.1. RN 06-040

For the purpose of ECCS analyses, Westinghouse (W) defines a large break loss-of-coolant accident (LOCA) as a rupture 1.0 ft^2 or larger of the reactor coolant system piping including the double ended rupture of the largest pipe in the reactor coolant system or of any line connected to that system.

Should a major break occur, rapid depressurization of the Reactor Coolant System (RCS) to a pressure nearly equal to the containment pressure occurs in approximately 40 seconds, with a nearly complete loss of system inventory. Rapid voiding in the core shuts down reactor power. A safety injection system signal is actuated when the low pressurizer pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Borated water injection complements void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. An average RCS / sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. However, no credit is taken for the insertion of control rods to shut down the reactor in the large break analysis.

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Before the break occurs, the reactor is assumed to be in a full power equilibrium condition, i.e., the heat generated in the core is being removed through the steam generator secondary system. At the beginning of the blowdown phase, the entire RCS contains sub-cooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. After the break develops, the time to departure from nucleate boiling is calculated. RN 06-040

Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes voided, both transition boiling and forced convection are considered as the dominant core heat transfer mechanisms. Heat transfer due to radiation is also considered.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of the large break LOCA, the primary pressure rapidly decreases below the secondary system pressure and the steam generators are an additional heat source. In this analysis using the WCOBRA/TRAC methodology, the steam generator secondary is conservatively assumed to be isolated (main feedwater and steam line) at the initiation of the event to maximize the secondary side heat load.

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 cooling system cold leg pipe. The reactor core and internals together with the Emergency Core Cooling System (ECCS) are designed so that the reactor can be safely shut-down and the essential heat transfer geometry of the core preserved following the accident. Long-term coolability is maintained.

When the RCS depressurizes to approximately ~642.7 psia, the accumulators begin to inject borated water into the reactor coolant loops. Borated water from the accumulator in the broken loop is assumed to spill to containment and be unavailable for core cooling for breaks in the cold leg of the RCS. Flow from the accumulators in the intact loops may not reach the core during depressurization of the RCS due to the fluid dynamics present during the ECCS bypass period. ECCS bypass results from the momentum of the fluid flow up the downcomer due to a break in the cold leg, which entrains ECCS flow out toward the break. Bypass of the ECCS diminishes as mechanisms responsible for the bypassing are calculated to be no longer effective.

The blowdown phase of the transient ends when the liquid level in the lower plenum reaches its minimum. After the end of the blowdown, refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the active fuel region of the fuel rods (called bottom of core (BOC) recovery time).

The reflood phase of the transient is defined as the time period lasting from BOC recovery until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and on into the beginning of reflood, the intact loop accumulator tanks rapidly discharges borated cooling water into the RCS. Although a portion injected prior to end of bypass is lost out the cold leg break, the accumulators eventually contributes to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head safety injection (LHSI) and high head safety injection (HHSI) pumps aid in the filling of the downcomer and core and subsequently supply water to help maintain a full downcomer and complete the reflooding process. RN 06-040

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching from the injection mode to the sump recirculation mode of ECCS operation. Spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the LHSI pumps (also called the Residual Heat Removal pumps, or RHR pumps) and returned to the RCS cold legs. Figure 15.4-1a contains a schematic of the bounding sequence of events.

For the Best-Estimate large break LOCA analysis, one ECCS train, including one HHSI pump and one LHSI pump, starts and delivers flow through the injection lines. The accumulator and safety injection flows from the broken loop were assumed to spill to containment. Both emergency diesel generators (EDGs) are assumed to start in the modeling of the containment spray pumps. Modeling full containment heat removal systems operation is required by Branch Technical Position CSB 6-1 (Reference 61) and is conservative for the large break LOCA.

To minimize delivery to the reactor, the HHSI and LHSI branch line chosen to spill is selected as the one with the minimum resistance.

15.4.1.1.1 Large Break LOCA Analytical Model

In 1988, as a result of the improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs, the NRC staff amended the requirements of 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," so that a realistic evaluation model may be used to analyze the performance of the ECCS during a hypothetical LOCA (Reference 55). Under the amended rules, best-estimate thermal-hydraulic models may be used in place of models with Appendix K features. The rule change also requires, as part of the analysis, an assessment of the uncertainty of the best-estimate calculations. It further requires that this analysis uncertainty be included when comparing the results of the calculations to the prescribed acceptance limits. Further guidance for the use of best-estimate codes was provided in Regulatory Guide 1.157 (Reference 56).

To demonstrate use of the revised ECCS rule, the NRC and its consultants developed a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 57). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

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A LOCA evaluation methodology for three- and four-loop PWR plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison and was approved by the NRC (Reference 58). The methodology is documented in WCAP-12945-P-A "Code Qualification Document (CQD) for Best Estimate LOCA Analysis" (Reference 59).

The thermal-hydraulic computer code which was reviewed and approved for the calculation of fluid and thermal conditions in the PWR during a large break LOCA is WCOBRA/TRAC Version MOD7A Rev. 1 (Reference 59).

WCOBRA/TRAC combines two-fluid, three-field, multi-dimensional fluid equations used in the vessel with one-dimensional drift-flux equations used in the loops to allow a complete and detailed simulation of a PWR. This best-estimate computer code contains the following features:

- Ability to model transient three-dimensional flows in different geometries inside the vessel
- Ability to model thermal and mechanical non-equilibrium between phases
- Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes

- Ability to represent important reactor components such as fuel rods, steam generators, reactor coolant pumps, etc.

The reactor vessel is modeled with the three-dimensional, three-field fluid model, while the loop, major loop components, and safety injection points are modeled with the one-dimensional fluid model.

The basic building block for the vessel is the channel, a vertical stack of single mesh cells. Several channels can be connected together by gaps to model a region of the reactor vessel. Regions that occupy the same level form a section of the vessel. Vessel sections are connected axially to complete the vessel mesh by specifying channel connections between sections. Heat transfer surfaces and solid structures that interact significantly with the fluid can be modeled with rods and unheated conductors.

One-dimensional components are connected to the vessel. Special purpose components exist to model specific components such as the steam generator and pump.

A typical calculation using WCOBRA/TRAC begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are discussed in the next section. RN 06-040

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood follows continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code (Reference 3) and mass and energy releases from the WCOBRA/TRAC calculation. The parameters used in the containment analysis to determine this pressure curve are presented in Tables 15.4-1a and 15.4-1b.

The methods used in the application of WCOBRA/TRAC to the large break LOCA are described in Reference 59. A detailed assessment of the computer code WCOBRA/TRAC was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. Modeling of a PWR introduces additional uncertainties which are identified and quantified in the plant-specific analysis (Reference 60). The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the PCT at the 95th percentile ($PCT^{95\%}$). The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Plant Model Development

   In this step, a WCOBRA/TRAC model is developed. A high level of noding detail is used, in order to provide an accurate simulation of the transient. However, specific guidelines are followed to assure that the model is consistent with models used in the code validation. This results in a high level of consistency among plant models, except for specific areas dictated by hardware differences such as in the upper and lower plenums of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions

   In this step, the expected or desired range of the plant operating conditions to which the analysis applies is established. The parameters considered are based on a "key LOCA parameters" list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the "initial transient." Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. The results of these calculations are discussed in Section 5 of Reference 60. The most limiting input conditions, based on these confirmatory runs, are then combined into a single transient, which is then called the "reference transient." RN 06-040

3. PWR Sensitivity Calculations

   A series of PWR transients are performed in which the initial fluid conditions and boundary conditions are ranged around the nominal conditions used in the reference transient. The results of these calculations form the basis for the determination of the initial condition bias and uncertainty discussed in Section 6 of Reference 60.

   Next, a series of transients are performed which vary the power distribution, taking into account all possible power distributions during normal plant operation. The results of these calculations form the basis for the determination of the power distribution bias and uncertainty (response surface) discussed in Section 7 of Reference 60.

   Finally, a series of transients are performed which vary parameters that affect the overall system response ("global" parameters) and local fuel rod response ("local" parameters). The results of these calculations form the basis for the determination of the model bias and uncertainty (response surface) discussed in Section 8 of Reference 60.

4. Response Surface Calculations

   The results from the power distribution and global model WCOBRA/TRAC runs performed in Step 3 are fit by regression analyses into equations known as response surfaces. The results of the initial conditions run matrix are used to generate a PCT uncertainty distribution.

5. Uncertainty Evaluation

   The total PCT uncertainty from the initial conditions, power distribution, and model calculations is derived using the approved methodology (Reference 59). The uncertainty calculations assume certain plant operating ranges which may be varied depending on the results obtained. These uncertainties are then combined to determine the initial estimate of the total PCT uncertainty distribution for the guillotine and limiting split breaks. The results of these initial estimates of the total PCT uncertainty are compared to determine the limiting break type. If the split break is limiting, an additional set of split transients are performed which vary overall system response ("global" parameters) and local fuel rod response ("local" parameters). The results of these calculations form the basis for the determination of the model bias and uncertainty discussed in Section 9 of Reference 60. Finally, an additional series of runs is made to quantify the bias and uncertainty due to assuming that the above three uncertainty categories are independent. The final PCT uncertainty distribution is then calculated for the limiting break type, and the 95th percentile PCT ($PCT^{95\%}$) is determined. RN 06-040

6. Plant Operating Range

   The plant operating range over which the uncertainty evaluation applies is defined. Depending on the results obtained in the above uncertainty evaluation, this range may be the desired range established in step 2, or may be narrower for some parameters to gain additional margin.

There are three major uncertainty categories or elements:

- Initial condition bias and uncertainty
- Power distribution bias and uncertainty
- Model bias and uncertainty

Conceptually, these elements may be assumed to affect the reference transient PCT as shown below:

$$PCT_i = PCT_{REF,i} + \Delta PCT_{IC,i} + \Delta PCT_{PD,i} + \Delta PCT_{MOD,i} \qquad (15.4.1.1.1\text{-}1)$$

where,

$PCT_{REF,i}$ = **Reference transient PCT:** The reference transient PCT is calculated using WCOBRA/TRAC at the nominal conditions identified in Table 15.4-1c, for the blowdown, first reflood and second reflood periods.

$\Delta PCT_{IC,i}$ = **Initial condition bias and uncertainty:** This bias is the difference between the reference transient PCT, which assumes several nominal or average initial conditions, and the average PCT taking into account all possible values of the initial conditions. This bias takes into account plant variations which have a relatively small effect on PCT. The elements which make up this bias and its uncertainty are plant-specific.

$\Delta PCT_{PD,i}$ = **Power distribution bias and uncertainty:** This bias is the difference between the reference transient PCT, which assumes a nominal power distribution, and the average PCT taking into account all possible power distributions during normal plant operation. Elements which contribute to the uncertainty of this bias are calculational uncertainties, and variations due to transient operation of the reactor. RN 06-040

$\Delta PCT_{MOD,i}$ = **Model bias and uncertainty:** This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict important phenomena which affect the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT taking into account all possible values of global and local parameters.

The separability of the bias and uncertainty components in the manner described above is an approximation, since the parameters in each element may be affected by parameters in other elements. The bias and uncertainty associated with this assumption is quantified as part of the overall uncertainty methodology and included in the final estimates of $PCT^{95\%}$.

15.4.1.1.2 Large Break LOCA Analysis Results

A series of WCOBRA/TRAC calculations were performed to determine the effect of variations in several key LOCA parameters on peak cladding temperature (PCT). From these studies, an assessment was made of the parameters that had a significant effect as will be described in the following sections.

15.4.1.1.2.1 LOCA Reference Transient Description

The plant-specific analysis performed indicated that the double-ended cold leg guillotine (DECLG) break is more limiting than the split break. The plant conditions used in the reference transient are listed in Table 15.4-1c. The following is a description of the final reference transient.

The LOCA transient can be conveniently divided into a number of time periods in which specific phenomena are occurring. For a typical large break, the blowdown period can be divided into the critical heat flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, first reflood, second reflood, and long term cooling phases. The important phenomena occurring during each of these phases are discussed for the DECLG break with a discharge coefficient (CD) of 1.0 and offsite power available. The results are shown in Figures 15.4-1b through 15.4-1n. RN 06-040

Critical Heat Flux (CHF) Phase (20 - 22 seconds) [Note: 0-20 sec is steady state run]

Immediately following the cold leg rupture, the break discharge rate is subcooled and high, the core flow reverses, the fuel rods go through departure from nucleate boiling (DNB) and the cladding rapidly heats up while core power shuts down. Figure 15.4-1b shows the maximum cladding temperature in the core, as a function of time. The hot water in the core and upper plenum flashes to steam during this period. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture swells and the intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

Upward Core Flow Phase (22 - 27 seconds)

Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, and the break discharge rate is low because the fluid is saturated at the break. Figures 15.4-1c and 15.4-1d show the break flowrate from the vessel and loop sides of the break. This phase ends as lower plenum mass is depleted, the loops become two-phase, and the pump head degrades. If pumps are highly degraded or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 15.4-1e shows the void fraction for one intact loop pump and the broken loop pump. The intact loop pump remains in single-phase liquid flow for several seconds, while the broken loop pump is in two-phase and steam flow soon after the break.

Downward Core Flow Phase (27 - 43 seconds)

The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core. Figure 15.4-1f shows the vapor flow at the mid-core of channels 11 and 13. While liquid and entrained liquid flows also provide core cooling, the vapor flow entering the core best illustrates this phase of core cooling. This period is enhanced by flow from the upper head. As the system pressure continues to fall, the break flow and consequently the core flow, are reduced. The core begins to heat up as the system reaches containment pressure and the vessel begins to fill with Emergency Core Cooling System (ECCS) water.

Refill Phase (43 - 51 seconds)

The core experiences a nearly adiabatic heatup as the lower plenum fills with ECCS water, as shown in Figure 15.4-1g. This phase ends when the ECCS water enters the core and entrainment begins, with a resulting improvement in heat transfer. Figures 15.4-1h and 15.4-1i show the liquid flows from the accumulator and the safety injection to one of the two intact loops.

First Reflood Phase (51 - 58 seconds)

The accumulators begin to empty and nitrogen enters the system. This forces water into the core which then boils as the lower core region begins to quench, causing repressurization. The re-pressurization is best illustrated by the reduction in pumped SI flow (Figure 15.4-1i, ~53 sec). During this time, core cooling may be increased. It is typical for the nitrogen injection induced core insurge to result in two closely spaced PCT peaks during this brief period. For the Reference transient, these two peaks occur at 52 and 56 seconds as illustrated in Figure 15.4-1b. RN 06-040

Second Reflood Phase (58 seconds - end)

The system then settles into a gravity driven reflood which exhibits lower core heat transfer and gradual PCT heatup. Figures 15.4-1j and 15.4-1k show the core and downcomer liquid levels. Figure 15.4-11 shows the vessel fluid mass. As the quench front progresses further into the core, the peak cladding temperature (PCT) location moves higher in the top core region. Figure 15.4-1m shows the movement of the PCT location. As the vessel continues to fill, the PCT location is cooled and the heatup PCT transient is terminated, at approximately 115 seconds for the reference transient. This is the limiting PCT peak for the reference transient. Very late in the transient, at about 200 seconds for the reference transient, the reflood phase is characterized by boiling in the downcomer and lower plenum. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel metal, reduces the subcooling of ECCS water in the lower plenum and downcomer.

This boiling can be observed from the reduction in lower plenum collapsed liquid level (Figure 15.4-1g). However, this boiling is rather mild and, as illustrated by Figure 15.4-1b, does not produce a late reflood PCT excursion.

Long Term Core Cooling

At the end of the WCOBRA/TRAC calculation, the core and downcomer levels are increasing as the pumped safety injection flow exceeds the break flow. The core and downcomer levels would be expected to continue to rise, until the downcomer mixture level approaches the loop elevation. At that point, the break flow would increase, until it roughly matches the injection flowrate. The core would continue to be cooled until the entire core is eventually quenched.

15.4.1.1.2.2 Confirmatory Sensitivity Studies

A number of sensitivity calculations were carried out to investigate the effect of the key LOCA parameters, and to develop the required data for the uncertainty evaluation. In the sensitivity studies performed, LOCA parameters were varied one at a time. For each sensitivity study, a comparison between the base case and the sensitivity case transient results was made.

The results of the sensitivity studies are summarized in Tables 15.4-1d and 15.4-1e. A full report on the results for all sensitivity study results is included in Sections 5 and 8 of Reference 60. The results of these analyses lead to the following conclusions: RN 06-040

1. Modeling minimum steam generator tube plugging (0%) results in a higher PCT than maximum steam generator tube plugging (10%).

2. Modeling offsite power available results in a higher PCT then loss-of-offsite power (LOOP).

3. Modeling the minimum value of vessel average temperature (T_{avg} = 572.0 °F) results in a higher PCT than the maximum value of vessel average temperature (T_{avg} = 587.4 °F).

4. Modeling the minimum power fraction (PLOW = 0.2) in the low power/periphery channel of the core results in a higher PCT than the maximum power fraction (PLOW = 0.8).

5. The limiting break type is a double ended cold leg guillotine (DECLG) break. This transient then becomes the reference transient for the determination of uncertainties.

15.4.1.1.2.3 Initial Conditions Sensitivity Studies

Several calculations were performed to evaluate the effect of change in the initial conditions on the calculated LOCA transient. These calculations analyzed key initial plant conditions over their expected range of operation. These studies included effects of ranging RCS conditions (pressure and temperature), safety injection temperature, and accumulator conditions (pressure, temperature, volume, and line resistance). The results of these studies are presented in Section 6 of Reference 60. Some of these cases are performed with the approved code version and some are performed with the MOD7A predecessor version as permitted by the Reanalysis Work Plan (Reference 62).

The calculated results were used to develop initial condition uncertainty distributions for the blowdown and reflood peaks. These distributions are then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from initial conditions uncertainty ($\Delta PCT_{IC,i}$).

15.4.1.1.2.4 Power Distribution Sensitivity Studies

Several calculations were performed to evaluate the effect of power distribution on the calculated LOCA transient. The power distribution attributes which were analyzed are the peak linear heat rate relative to the core average, the maximum relative rod power, the relative power in the bottom third of the core (P_{BOT}), and the relative power in the middle third of the core (P_{MID}). The choice of these variables and their ranges are based on the expected range of plant operation. RN 06-040

The power distribution parameters used for the reference transient are biased to yield a relatively high PCT. The reference transient uses the maximum $F_{\Delta H}$, a skewed to the top power distribution, and a F_Q at the midpoint of the sample range.

A run matrix was developed in order to vary the power distribution attributes singly and in combination. The calculated results are presented in Section 7 of Reference 60. The sensitivity results indicated that power distributions with peak powers shifted towards the top of the core produced higher PCTs. All of these cases are performed with the MOD7A predecessor version as permitted by the Reanalysis Work Plan (Reference 62).

The calculated results were used to develop response surfaces, as described in Step 4 of Section 15.4.1.1.1 which could be used to predict the change in PCT for various changes in the power distributions for the blowdown and reflood peaks. These were then used in the uncertainty evaluation, to predict the PCT uncertainty component resulting from uncertainties in power distribution parameters, ($\Delta PCT_{PD,i}$).

15.4.1.1.2.5 Global Model Sensitivity Studies

Several calculations were performed to evaluate the effect of broken loop resistance, break discharge coefficient, and condensation rate on the PCT for the guillotine break. As in the power distribution study, these parameters were varied singly and in combination in order to obtain a data base which could be used for response surface generation. The run matrix and ranges of the break flow parameters are described in Reference 59. The limiting split break was also identified using the methodology described in Reference 59. The plant specific calculated results are presented in Section 8 of Reference 60. The results of these studies indicated that the double ended cold leg guillotine break resulted in the highest PCT.

The calculated results were used to develop response surfaces as described in Section 15.4.1.1.1, which could be used to predict the change in PCT for various changes in the flow conditions. These were then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from uncertainties in global model parameters ($\Delta PCT_{MOD,i}$).

15.4.1.1.2.6 Uncertainty Evaluation and Result

The PCT equation was presented in Section 15.4.1.1.1. Each element of uncertainty is initially considered to be independent of the other. Each bias component is considered a random variable, whose uncertainty and distribution is obtained directly, or is obtained from the uncertainty of the parameters of which the bias is a function. For example, $\Delta PCT_{PD,i}$ is a function of F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID}. Its distribution is obtained by sampling the plant F_Q, $F_{\Delta H}$, P_{BOT}, and P_{MID} distributions and using a response surface to calculate $\Delta PCT_{PD,i}$. Since PCT_i is the sum of these biases, it also becomes a random variable. Separate initial PCT frequency distributions are constructed as follows for the guillotine break and the limiting split break size: RN 06-040

1. Generate a random value of each ΔPCT element.

2. Calculate the resulting PCT using Equation 15.4.1.1.1-1.

3. Repeat the process many times to generate a histogram of PCTs.

The results of this assessment showed the split break to be non-limiting.

A final verification step is performed in which additional calculations (known as "superposition" calculations) are made with WCOBRA/TRAC, simultaneously varying several parameters which were previously assumed independent (for example, power distributions and models). Predictions using Equation 15.4.1.1.1-1 are compared to this data, and additional biases and uncertainties are applied.

The estimate of the PCT at 95 percent probability is determined by finding that PCT below which 95 percent of the calculated PCTs reside. This estimate is the licensing basis PCT, under the revised ECCS rule.

Results are given in Table 15.4-1f. As shown, limiting results occur for the second reflood peak with a 95th percentile PCT ($PCT^{95\%}$) of 1988°F, and the difference between the 95 percent value and the average value increases with increasing time, as more parameter uncertainties come into play. RN 06-040

15.4.1.1.2.7 Evaluations

An additional calculation was performed to assess IFBA fuel. The base analysis is for non-IFBA fuel. An analysis of IFBA fuel was performed independently, utilizing the HOTSPOT code and the high PCT case identified in Appendix A of Reference 60. The analysis results indicated that IFBA fuel is bounded by non-IFBA fuel.

Another Virgil C. Summer Nuclear Station specific evaluation was performed to assess the impact of a conversion of the barrel/baffle region to an upflow configuration.

The WCOBRA/TRAC reference steady state and transient decks documented in Section 15.4.1.1.3.1 were modified to reflect the conversion of the barrel/baffle region to an upflow configuration; the detailed thermal-hydraulic parameters associated with the modified barrel/baffle region flow were modeled, and the reference steady state and transient were executed using the same approved version of WCOBRA/TRAC. RN 09-022

This evaluation resulted in a 7°F reduction in PCT during the Blowdown period, a 44°F reduction in PCT during the Reflood 1 period, and a 29°F reduction in PCT during the Reflood 2 period. Based on these results, it is concluded that the Table 15.4-1f PCT values will remain bounding upon the conversion to an upflow configuration.

15.4.1.1.3 Large Break LOCA Conclusions

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

1) There is a high level of probability that the peak cladding temperature (PCT) shall not exceed 2200°F. The results presented in Table 15.4-1f indicate that this regulatory limit has been met with a reflood $PCT^{95\%}$ of 1988°F. RN 06-040

2) The maximum calculated local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. The approved Best Estimate LOCA methodology assesses this requirement using a plant-specific transient which has a PCT in excess of the estimated 95 percentile PCT ($PCT^{95\%}$). Based on this conservative calculation, a maximum local oxidation of 5.0 percent is calculated, which meets the regulatory limit.

3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel were to react. The total amount of hydrogen generated, based on this conservative assessment is 0.72 percent, which meets the regulatory limit. RN 06-040

4) Calculated changes in core geometry shall be such that the core remains amenable to cooling. This requirement is met by demonstrating that the PCT does not exceed 2200°F, the maximum local oxidation does not exceed 17% , and the seismic and LOCA forces are not sufficient to distort the fuel assemblies to the extent that the core cannot be cooled. The BE methodology (Reference 59) specifies that the effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crush extends to in-board assemblies. Fuel assembly structural analyses indicate that this condition does not occur. Therefore, this regulatory limit is met.

5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core. The conditions at the end of the WCOBRA/TRAC calculations indicate that the transition to long term cooling is underway even before the entire core is quenched. RN 06-040

15.4.1.1.4 SER Requirements

The SER requirements for three-loop plants (Reference 58) have been met for this analysis.

15.4.1.1.5 Plant Operating Range

The expected PCT and its uncertainty developed above are valid for a range of plant operating conditions. In contrast to Appendix K calculations, many parameters in the base case calculation are at nominal values. The range of variation of the operating parameters has been accounted for in the estimated PCT uncertainty. Table 15.4-1g summarizes the operating ranges considered.

15.4.1.1.6 Long Term Cooling

In the long term, the core remains subcritical and amenable to cooling. For the limiting cold leg break, evaluations show:

1. At initiation of cold leg recirculation, the sump boron concentration is sufficient to keep the core subcritical (K_{eff} less than 1.0) with all control rods out and no Xenon. RN 02-052

2. Boron levels within the core are limited to less than the precipitation level by supplying ECCS water to the hot legs within 8 hours after initiation of the LOCA.

3. At the initiation of hot leg recirculation, the sump boron concentration is sufficient to keep the core subcritical (K_{eff} less than 1.0) with N-1 rods inserted (per WCAP-15704[50]) and Xenon. RN 02-052

15.4.1.1.7 Impact of ECCS Evaluation Model Changes RN 06-040

The October 17, 1988 revision to 10CFR50.46 requires applicants and holders of operating licenses or construction permits to notify the NRC of errors and changes in the ECCS Evaluation Models, which are not significant, on an annual basis. Reference [47] defines a significant error change as one which results in a calculated peak fuel cladding temperature (PCT) different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute temperature change is greater than 50°F. The current ECCS evaluation model changes that affect the large break LOCA PCT are identified in the latest 10CFR50.46 annual or 30-day report. 02-01

15.4.1.2 <u>Hydrogen Production and Accumulation</u>

Hydrogen accumulation in the containment atmosphere following a design basis accident can be the result of production from several sources. The potential sources of hydrogen are the zirconium-water reaction, corrosion of construction materials, and radiolytic decomposition of the emergency core cooling solution. The latter source, solution radiolysis, includes both core solution radiolysis and sump solution radiolysis.

15.4.1.2.1 Method of Analysis

Methods and assumptions used in the analysis of post-accident hydrogen generation and control are specified by Regulatory Guide 1.7 [53]. In determining the amount of hydrogen generated by reaction of the core Zircaloy cladding with water, Regulatory Guide 1.7 states that the amount of hydrogen generated by zirc-water reaction should be the greater of either five times the maximum amount calculated by ECCS evaluation or the amount that would result from reaction of all the metal in the outside surfaces of the fuel cladding to a depth of 0.00023 inches. The 10CFR50.46 acceptance criteria for ECCS performance requires that total hydrogen generated by cladding oxidation be less than 0.01 times (1.0%) the amount generated by oxidation of the whole core cladding. The amount of cladding oxidation following the design basis LOCA is shown by the ECCS evaluation to be less than the 1.0% acceptance criteria (Table 15.4-2a). Five times that amount, then, is 5.0%. Since 0.00023 inches is approximately 1% of the cladding thickness, the conservative assumption for Zirc-water reaction under Regulatory Guide 1.7 is 5.0% of the whole core cladding Zircaloy. RN 03-005

The use of aluminum inside the containment is limited, and it is not used in safety-related components which are in contact with the recirculating core cooling fluid. Aluminum is much more reactive with the containment spray alkaline borate solution than other plant materials such as galvanized steel, copper, and copper nickel alloys. RN 03-005

By limiting the use of aluminum and zinc, the aggregate source of hydrogen over the long term is primarily due to that arising from radiolytic decomposition and can be predicted with reasonable certainty to permit the design of effective countermeasures.

It should be noted that the zirconium-water reaction and aluminum-zinc corrosion with containment spray are chemical reactions and thus essentially independent of the radiation field inside the containment following a loss of coolant accident. Radiolytic decomposition of water is dependent on the radiation field intensity. The radiation field inside the containment is calculated for the maximum credible accident in which the fission product activities given in TID-14844 [15] are used.

The hydrogen generation calculations are performed using the NRC model discussed in Regulatory Guide 1.7 and NRC Standard Review Plan, Section 6.2.5 with no recombiner operating; also, a calculation is made using the NRC model and recombiner operation after the first day following a LOCA. Recombiner operation is discussed in Section 6.2.5. 98-01

15.4.1.2.2 Assumptions

The following discussion outlines the assumptions used in the calculations.

1. Zirconium-water reaction

   The zirconium-water reaction is described by the chemical equation:

   $Zr + 2H_2O \rightarrow Zr\ O_2 + 2H_2 + \text{Heat}$

   The hydrogen generation due to this reaction will be completed during the first day following the loss of coolant accident. The quantity of hydrogen generated from the zirconium-water reaction is 14,987 scf, and is based on 5% of the total zirconium mass reacted. The hydrogen generated is assumed to be released immediately to the containment atmosphere following the loss of coolant accident. RN 03-005

2. Hydrogen from the Reactor Coolant System

   The maximum quantity of hydrogen contained in the primary coolant system is 909 scf. This includes both hydrogen dissolved in the coolant water [35 cc (STP)/kg] and the corresponding equilibrium hydrogen in the pressurizer gas space. The 909 scf of hydrogen is assumed to be released immediately to the containment atmosphere following the loss of coolant accident. 98-01 98-01

3. Corrosion of plant materials

Oxidation of metals in aqueous solution results in the generation of hydrogen gas as one of the corrosion products. Extensive corrosion testing has been conducted to determine the behavior of the various metals used in the containment in the emergency core cooling solution at design basis accident conditions. Metals tested include Zircaloy, Inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel, and copper.

Tests conducted at Oak Ridge National Laboratory (ORNL) [17], [18] have also verified the compatibility of the various materials (exclusive of aluminum) with alkaline borate solution. As applied to the quantitative definition of hydrogen production rates, the results of the corrosion tests have shown that only aluminum and zinc will corrode at a rate that will significantly add to the hydrogen accumulation in the containment atmosphere.

The corrosion of aluminum may be described by the overall reaction:

$$2\ Al + 3\ H_2O \rightarrow Al_2O_3 + 3\ H_2$$

Therefore, three moles of hydrogen are produced for each two moles of aluminum oxidized. This corresponds to 20 scf hydrogen produced for each pound of aluminum corroded. 98-01

The corrosion of zinc may be described by the overall reaction:

$$Zn + 2\ H_2O \leftrightarrows Zn(OH)_2 + H_2$$ 02-01

Therefore, one mole of hydrogen is produced for each mole of zinc oxidized. This corresponds to 5.5 scf hydrogen produced for each pound of zinc corroded.

The time-temperature cycle (Table 15.4-6) used in the calculation of aluminum and zinc corrosion is based on a conservative representation of the postulated post accident containment transient. The corrosion rates at the various steps were determined from the aluminum and zinc corrosion rate design curve shown in Figures 15.4-68 and 15.4-69. The corrosion data points include the effects of temperature, alloy, and spray solution conditions. Based on these corrosion rates and the aluminum and zinc inventory given in Table 15.4-7, the contribution of aluminum and zinc corrosion to hydrogen accumulation in the containment following a design basis accident has been calculated. For conservative estimation, no credit was taken for protective shielding effects of insulation or enclosures from the spray, and complete and continuous immersion was assumed. 98-01

4. Radiolysis of core and sump water

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the reaction:

$$2\ H_2O \leftrightarrows 2\ H_2 + O_2$$ 02-01

Of interest here is the quantitative definition of the rates and extent of radiolytic hydrogen production following a design basis accident.

As the emergency core cooling solution flows through the core, it is subjected to gamma radiation by decay of fission products in the fuel. This energy deposition results in solution radiolysis and the production of molecular hydrogen and oxygen. The initial production rate of these species will depend on the rate of energy absorption and the specific radiolytic yields. Another source of hydrogen for the post-accident period arises from water contained in the reactor containment sump being subjected to radiolytic decomposition by fission products.

For the purposes of this analysis, the calculations are based on Regulatory Guide 1.7 assumptions.

Calculations based on Regulatory Guide 1.7 assume a hydrogen yield value of 0.5 molecules per 100 ev, 10% of the gamma energy produced from fission products in the fuel rods is absorbed by the solution in the region of the core, 100% of the gamma and beta energy produced from fission products intimately mixed with coolant.

15.4.1.2.3 Results

Figures 15.4-71 and 15.4-72 show the hydrogen production and accumulation in the containment following a loss of coolant accident for the NRC model. Figure 15.4-73 shows the volume percent of hydrogen in the containment for the NRC model assuming no recombiner and recombiner start at one day. 98-01

15.4.1.3 <u>Deleted</u>

15.4.1.4 <u>Environmental Consequences of a Postulated Loss of Coolant Accident</u>

The results of analyses presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a LOCA do not result in doses which exceed guideline values.

Two analyses have been performed:

1. An analysis based upon Regulatory Guide 1.183 (see Appendix 3A). RN 12-034

2. A realistic analysis.

15.4.1.4.1 Fission Product Release to the Containment

To evaluate the radiological consequences of a postulated LOCA, the following 2 cases of fission product releases to the containment have been analyzed.

1. Core Activity Release (Regulatory Guide 1.183 Analysis)

   Acceptable assumptions regarding core inventory and the release of radionuclides from the core are provided in Regulatory Position 3 of Regulatory Guide 1.183.

   The inventory of fission products in the reactor core available for release from the containment is based on a core thermal power of 2958 MWt. A list of the 60 isotopes used in the Regulatory Guide 1.183 analysis is given in Table 15.4-11. The release fractions and timing associated with these isotopes are taken from RG 1.183 and reproduced in Table 15.4-12. RN 12-034

2. Primary Coolant Activity Release (Realistic Case)

   The inventory of fission products available for release from the containment are based on the 1% failed fuel primary coolant activities. In addition, the iodines and bromines in the coolant are conservatively assumed to be increased by a factor of 50 and the noble gases by a factor of 1.5 due to the spiking phenomenon. The resulting values available for release from containment are listed in Table 15.4-13.

15.4.1.4.2 Radioactive Releases from Recirculation Loops

In addition to the fission product release to the containment, the effect of leakage of recirculating sump fluid from engineered safety features equipment located outside the Reactor Building needs to be considered in evaluating the radiological consequences of a LOCA. The following assumptions are utilized in estimating the radiological consequences of post LOCA radioactive releases from these systems (i.e., emergency core cooling and Reactor Building spray) to the Auxiliary Building.

1. With the exception of noble gases, the fission products released to the containment listed in Table 15.4-11 and 15.4-13 are assumed to instantaneously and homogeneously mix in the primary containment sump water. RN 12-034

2. The leakage from these systems is conservatively assumed to be 12,000 cc/hr which is approximately 2 times the operational leakage estimated for the RHR and RB Spray systems as indicated in Tables 6.2-52b and 6.3-4. The release from these systems is assumed to start at the earliest time that the recirculation mode is initiated and continue for the duration of the accident. RN 03-019 12-034 RN 12-034

3. It is conservatively assumed that the recirculation loop leakage flashes and that 10% of the iodine is released to the Auxiliary Building atmosphere.

4. No credit is taken for the holdup or filtration of this leakage in the auxiliary building, i.e., the iodine released by the recirculation loop leakage is assumed to be immediately available for release to the environs.

15.4.1.4.3 Radiological Consequences Offsite

The RADTRAD[64] computer code was used to calculate the offsite doses for the LOCA event. Schematics showing the code modeling of the activity flow paths are shown on Figures 15.4-74 and 15.4-74A. Parameters and assumptions used in evaluating the offsite doses for the LOCA are summarized in Table 15.4-15. RN 12-034

The calculated offsite doses for LOCA are provided in Table 15.4-16. The dose limits as defined in 10 CFR 50.67 are included in Table 15.4-16. The doses resulting from this accident are well within the limits defined in 10 CFR 50.67.

15.4.1.4.4 Radiological Consequences to the Control Room

The radiological effects that could exist in the control room as a consequence of this event would be due to direct radiation from the radioactive cloud in the atmosphere which results from leakage of fission products from the containment and from the postulated recirculation loop leakage in the Auxiliary Building; direct exposure from, the containment; and exposure to radioactive materials which might leak into the control room from these sources.

For the Regulatory Guide 1.183 Case the integrated dose to the control room operators was calculated using the following methods and assumptions:

1. The RADTRAD[64] computer code was used to calculate the control room doses for the Regulatory Guide 1.183 case. Schematics showing the code modeling of the activity flow paths are shown on Figures 15.4-74 and 15.4-74A. Parameters and assumptions used in evaluating the control room doses for the LOCA are summarized in Table 15.4-17. RN 12-034

2. The analytical model used to calculate the direct dose from the external radioactive cloud in the atmosphere is calculated as the unprotected control room dose utilizing the RADTRAD computer code and is adjusted to reflect the shielding effects of the control building structure.

3. The direct whole body dose from the radioactivity in the reactor building is calculated utilizing the MicroShield[65] shielding computer code based on a cylindrical source model, applicable post-accident source terms, and reactor/control building shielding.

4. The atmospheric diffusion parameters for the control room are provided in Table 2.3-123. The limiting control room release point for the LOCA is the Reactor Building nearest point (intake A). RN 12-034

5. As described in Section 9.4.1, the Control Room Ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a Safety Injection (SI) or high radiation signal from the gaseous activity channel of RM-A1. If both trains are operating, one train will be isolated within 30 minutes by the Operator in accordance with the Emergency Operating Procedures to minimize dose consequences. This condition is bounded by the analysis which only credits one train control room ventilation for the duration of the accident.

6. The modeling of the Control Room Ventilation assumed for the Regulatory Guide 1.183 case is shown on Figures 15.4-74 and 15.4-74A. The emergency mode of operation is assumed to occur at time zero since a SI signal occurs quickly after the accident.

The resulting control room doses for this LOCA are summarized in Table 15.4-18.

15.4.2 MAJOR SECONDARY SYSTEM PIPE RUPTURE

Two (2) major secondary system pipe ruptures are analyzed in this section: rupture of a main steam line and rupture of a main feedwater pipe. The time sequence of events for each of these events is provided in Table 15.4-19.

15.4.2.1 Major 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 would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in 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 limiting main steam line break was selected based upon the sensitivity studies performed in "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226, January, 1978 [42].

The analysis of a main steam line rupture is performed to demonstrate that, assuming a stuck RCCA (with and without offsite power), and a single failure in the engineered safety features (ESF) there is no consequential damage to the primary system and the core remains in place and intact.

Energy release to containment for the worst steam pipe break is discussed in FSAR Chapter 6.

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

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

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

    a. Two (2) out of 3 low pressurizer pressure signals.

    b. Two (2) out of 3 high-1 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, a safety injection signal will rapidly close all feedwater control valves and feedwater isolation valves that backup the control valves.

4. Trip of the main steam line isolation valves (See Technical Specification[24] Table 3.3-5) on:

    a. High-2 containment pressure.

    b. High steam line flow in 2 out of 3 steam lines in coincidence with 2 out of 3 low-low RCS average temperature.

c. Two (2) out-of-3 low steam line pressure.

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 nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe and thus the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.4-21 lists the equipment required in the recovery from a high energy line rupture. Not all equipment is required for any one particular break, since it will vary depending upon postulated break location and details of balance of plant design and pipe rupture criteria. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in FSAR Section 3.6.

15.4.2.1.2 Analysis of Effects and Consequences

15.4.2.1.2.1 Method of Analysis

The analysis of the steam pipe rupture has been performed based on the methodologies documented in Reference [42] to determine:

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN[25] Code has been used.

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC[26], has been used to determine if DNB occurs for the core conditions computed in item 1 above.

The following conditions were assumed to exist at the time of a main steam line break accident.

1. End of life shut down 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 condition than the case analyzed.

2. The 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 k_{eff} versus temperature coefficient used is shown in Figure 15.2-46. The effect of power generation in core on overall reactivity is shown in Figure 15.4-75.

   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. Further, it was conservatively assumed that the core power distribution was uniform. These 2 conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked. These core analyses considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution, and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the true reactivity. These results verified conservatism; i.e., underprediction of negative reactivity feedback from power generation.

3. Minimum capability for injection of high concentration boric acid (2300 ppm) solution corresponding to the most restrictive single failure in the Safety Injection System (SIS). The characteristics of the injection unit used are shown on Figure 15.2-47. This corresponds to the flow delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration of boric acid that must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) isolation valves prior to the delivery of highly concentrated boric acid to the reactor coolant loops. This effect has been allowed for in the analysis. The modeling of the SIS in LOFTRAN is described in Reference [2].

   When offsite power is available, the sequence of events in the high head injection system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the charging pump starts. In 27 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept before the 2300 ppm water from the refueling water storage tank reaches the core. This delay is inherently included in the modeling. RN 01-125

When offsite power is not available, an additional ten second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them. That is, after a total of 37 seconds following a safety injection signal, safety injection is assumed to be capable of delivering flow to the RCS. For conservatism, both the with and without offsite power available cases were analyzed using 37 seconds. RN 01-125 RN 01-125

The choice of having loss of power near the beginning of the transient is made to maximize the delay of boron reaching the core. Having full RCS flow and immediate safety injection startup during the early stage of the transient facilitates boron delivery by minimizing pump startup time, thereby immediately purging unborated water from the safety injection lines, and minimizing RCS transport time. Loss of power after the break occurs, therefore, will allow a partial or total purge of unborated water from the safety injection lines, thus reducing the amount of time before which boron can be injected into the core, and loss of power prior to the break will allow for immediate safety injection startup and thus almost immediate purging of unborated water from the safety injection lines. Furthermore, since the steamline break with loss of offsite power is less severe than the case with offsite power available, the latter will be more conservative and bound the case of loss of offsite power.

4. Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 square feet, regardless of location would have the same effect on the Nuclear Steam Supply System (NSSS) as the 1.4 square foot break. The following cases have been considered in determining the core power and RCS transients:

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

   b. Complete severance of a pipe with the plant initially at no-load conditions with offsite power unavailable.

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

Both the cases assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the 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 temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumed 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 than steam line breaks occurring at power.

6. In computing the steam flow during a steam line break, the Moody Curve[27] for fl/D = 0 is used. The Moody Multiplier is 1 with a discharge at dry saturated steam conditions.

7. Perfect moisture separation in the steam generator 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. No credit was taken for secondary side safety injection actuation on low steam pressure.

9. A flow imbalance of 5% between loops (faulted vs. intact) was assumed to incorporate the effects of loop-to-loop RCS flow asymmetry in the limiting steamline rupture event (with offsite power available).

10. To maximize primary-to-secondary heat transfer, 0 percent (0%) steam generator tube plugging is assumed in the limiting steamline rupture event (with offsite power available). RN 01-125

15.4.2.1.2.2 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. The calculated sequence of events is listed in Table 15.4-19. RN 01-125

Figures 15.4-76 and 15.4-77 show the response of pertinent system parameters following a main steam pipe rupture. Offsite power is assumed to be available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator, with all 3 steam generators blowing down through the break until steamline isolation. RN 01-125

As can be seen, the core attains criticality with RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boric acid solution at 2300 ppm enters the RCS from the SIS which is drawing from the Refueling Water Storage Tank (RWST). The delay time consists of the time to receive and actuate the safety injection signal and the time to completely open valve trains in the safety injection lines. The safety injection pumps are then ready to deliver flow. At this stage, a further delay is incurred before 2300 ppm boron solution can be injected to the RCS due to the low concentration solution being swept from the safety injection lines. Should a partial loss of offsite power occur such that the power is lost to the ESF functions while the reactor coolant pumps remain in operation, an additional safety injection delay of 10 seconds would occur while the diesel generators startup and the necessary safety injection equipment is loaded onto them. This additional 10 second delay is included in the limiting analyzed case (with offsite power available). A peak core power well below the nominal full power value is attained. RN 01-125

The calculation assumes the boric acid is mixed with, and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the high head 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 high head injection system and the accumulator due to changes in the RCS pressure. The high head injection system flow calculation includes the line losses in the system as well as the pump head curve. The accumulators provide an additional source of borated water after the RCS pressure has decreased to below 600 psia.

Should the core be critical at near zero power when the rupture occurs low steam line pressure will trip the reactor. Automatic trip of the isolation valves in the steam lines by low steam line pressure or the high steam flow signal in coincidence with low-low RCS temperature will prevent continued steam release from more than one steam generator. The steam line isolation valves are designed to be fully closed in less than 7 seconds after receipt of closure signal. The analysis assumes a 10 second closing time for conservatism. RN 01-125

Figures 15.4-78 and 15.4-79 show the responses of the salient parameters for the case with a total loss of offsite power at the time of the rupture. This results in a coastdown of the reactor coolant pumps. In this case, the core power increases at a slower rate and reaches a lower peak value than in the cases in which offsite power is available to the reactor coolant pumps. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. RN 01-125

It should be noted that following a steam line break only one steam generator blows down completely while the intact steam generators only blow down for a short time. Thus, the intact steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power this heat is removed to the atmosphere via the steam line safety valves. RN 01-125

15.4.2.1.3 Conclusions

A DNB analysis was performed for both these cases. It was found that the DNB design basis[39] is met.

15.4.2.1.4 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 RCS to the secondary system in the steam generators. A realistic and a conservative analysis based upon Regulatory Guide 1.183 of the potential offsite and control room doses resulting from a steam line break outside containment is presented below. The analyses incorporate defective fuel and steam generator leakage assumptions prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. In addition, the conservative analysis includes considerations of iodine spiking effects, both pre-existing and concurrent iodine spike occurrences. Parameters used in both the realistic and conservative analyses are listed in Table 15.4-23. RN 12-034

The RADTRAD[64] computer code was used to calculate the offsite and control room doses for the main steam line break (MSLB). A schematic showing the code modeling of the activity flow paths is shown on Figure 15.4-143. RN 12-034

For the conservative analysis, the primary and secondary coolant activity equilibrium concentrations, prior to the accident, are based upon 1% failed fuel and are conservative relative to technical specification limits of 1 μc/gm and 0.1 μc/gm dose equivalent (DE) I-131, respectively. In addition, 2 iodine spike scenarios are addressed as follows: RN 06-022

a. Pre-accident Iodine Spike

A reactor transient has occurred prior to the postulated Main Steam Line Break (MSLB) and has raised the primary coolant iodine concentration to the maximum value permitted by Figure 3.4-1 of the VCSNS technical specifications or 60 μc/gm DE I-131. Based upon the assumed 1 gpm primary to secondary leakage, the secondary system coolant specific activity, during this iodine spike, is assumed to increase to an equilibrium value based on the 60 μc/gm DE I-131 as provided in Table 15.4-24a.

b. Concurrent Iodine Spike

A reactor trip and/or primary system depressurization associated with the MSLB creates an iodine spike in the primary system. The increase in the reactor coolant iodine concentration is estimated using a spiking model which assumes that the iodine release rate from the fuel rods to the coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value associated with reactor coolant activities at 1 μCi/gm dose equivalent I-131. 99-01

The iodine release rate during the concurrent spike is based on a maximum letdown flow of 143 gpm which includes the maximum normal letdown of 120 gpm plus 12 gpm to account for uncertainty in flow and 11 gpm primary coolant leakage and is shown in Table 15.4-24b. RN 00-056

The following additional assumptions and parameters are used to calculate the activity releases and offsite doses for a steam line break:

1. The faulted steam generator was assumed to be isolated 30 minutes after the accident.

2. The primary-to-secondary leak rate in the steam generators is assumed to be the leak-rate-limiting condition for operation specified in the plant requirements of 1 gpm for all three steam generators. The leakage is apportioned between the steam generators in such a manner that the calculated dose is maximized. Prior to the accident this leakage is assumed to be distributed throughout the three steam generators. To be consistent with preferred plant procedures for shutdown, the shutdown cooling rate following this type of event is taken as 25 ^{o}F /hr. Therefore, to go from 557 ^{o}F to 212 ^{o}F would require about 14 hours. The MSLB is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is assumed to be released to the environment via the faulted steam generator at a rate of 0.35 gpm for the 24 hour duration with no credit taken for any reduction or mitigation, i.e. a partition factor of 1.0. This is conservative in that the actual maximum value allowed by TS 3.4.6.2.c for any one steam generator is 150 gpd (approximately 0.104 gpm). The remaining 0.65 gpm is assumed to be released to the environment via the two intact steam generators for the 24 hour duration crediting a partition factor of 100 in accordance with Regulatory Guide 1.183, Appendix E. In addition, the secondary coolant activity initially contained in the faulted steam generator is released to the environment with a partition factor of 1.0.

RN
12-034

3. Offsite power is lost and the main condenser is not available for steam dump in the conservative analysis but is available for the realistic case.

4. As a result of the accident, no fuel failures occur.

5. Steam and reactor coolant releases to the faulted and the intact steam generators are given in Table 15.4-23.

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12-034

6. No steam generator blowdown during the accident.

7. No noble gas is dissolved in the steam generator secondary system water. All noble gases are assumed to be released via the condenser air removal system.

The RADTRAD[64] computer code was used to calculate the control room doses for the main steam line break (MSLB). The following assumptions and parameters are used to calculate the control room doses for the MSLB.

1. The atmospheric diffusion parameters for the control room are provided in Table 2.3-123. The limiting control room release point for the MSLB is the Main Steam Safety Relief Valve “A” (Reliefs B, C, D, E), Control Room Intake B.

2. The control room emergency filtration system is credited after 30 minutes. RN 12-034

3. The control room ventilation system flow rates are shown on Figure 15.4-143.

4. The control room breathing rate is assumed as 3.47E-04 m^3/sec for the duration of the accident.

5. Additional control room parameters are shown on Table 15.4-17.

Isotopic releases to the environment using these assumptions are summarized by Tables 15.4-25 and 15.4-26.

The resultant doses calculated at the site boundary, low population zone and control room for the steam line break accident, based upon the realistic and conservative analysis assumptions are given in Table 15.4-27. The doses resulting from this accident are well within the limits defined by 10 CFR 50.67. RN 12-034

15.4.2.2 Major Rupture of a Main Feedwater Line

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedline check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Section 15.2.8.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break), or a RCS heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in Section 15.4.2.1. 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 Emergency 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 following provides the necessary protection for a main feedwater rupture:

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

    a. High pressurizer pressure.

    b. Overtemperature Delta-T.

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

    d. Low steam generator level plus steam/feed flow mismatch in any steam generator.

    e. Safety injection signals from any of the following:

        (1) Low steam line pressure.
        (2) High containment pressure (Hi-1).
        (3) High steam line differential pressure.

    (Refer to Chapter 7 for a description of the actuation system)

2. An emergency feedwater system to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to Section 10.4.9 for a description of the emergency feedwater system.)

15.4.2.2.2 Analysis of Effects and Consequences

15.4.2.2.2.1 Method of Analysis

A detailed analysis using the LOFTRAN [25] Code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics, reactor coolant system including natural circulation, pressurizer, steam generators and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions are:

1. The plant is initially operating at 102% of the nominal NSSS design rating.

2. Initial reactor coolant average temperature is 4.0°F above the nominal value, and the initial pressurizer pressure is 50 psi below its nominal value. RN 01-125

3. A conservatively high initial pressurizer level is assumed; initial steam generator water level is at the nominal value plus 5% in the faulted steam generator, and at the nominal value minus 5% in the intact steam generators.

4. No credit is taken for the pressurizer spray.

5. Cases with and without pressurizer power operated relief valves are analyzed. RN 01-125

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

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

8. The worst possible break area is assumed which minimizes the steam generator fluid inventory at the time of trip and maximizes the blowdown discharge rate following the time of trip, and thereby maximizes the resultant heatup of the reactor coolant. RN 01-125

9. A conservative feedline break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a conservatively low blowdown quality is assumed until all water inventory is discharged from the affected steam generator. A low blowdown quality after trip results in a relatively small amount of energy release through the break. This increases the amount of energy which must be removed via the emergency feedwater system. RN 01-125

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10. Reactor trip is assumed to be initiated when the low-low level trip setpoint in the ruptured steam generator is reached. A low-low level setpoint of 0% narrow range span is assumed.

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11. The emergency feedwater system is actuated by the low-low steam generator water level signal and is assumed to supply a total 380 gpm to the unaffected steam generators within 108 seconds. The assumed time delay includes the following allowances: 60 seconds to account for instrumentation delays and startup of the diesels and emergency feed pumps and 48 seconds to automatically isolate emergency feed flow to the faulted SG. Although substantial emergency feedwater will be added to the steam generators prior to 108 seconds, no credit is taken for emergency feedwater flow prior to completion of the automatic actions to isolate flow to the faulted SG. Before the relatively cold (120°F) emergency feedwater enters the unaffected steam generators, additional time is also modeled to allow for the purging of 5 cubic feet of hot water contained in the emergency feedwater system lines.

12. No credit is taken for heat energy deposited in RCS metal during the RCS heatup. RN 01-125

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

14. Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 ANS 5.1 [28] decay heat standard plus uncertainty was used for calculation of residual decay heat levels. RN 01-125

15. No credit is taken for charging or letdown.

16. A flow imbalance of 5% between loop (faulted vs. intact) was assumed to incorporate the effects of loop-to-loop RCS flow asymmetry for the limiting cases (with PORVs only). RN 01-125

17. The maximum steam generator tube plugging level (10%) is assumed.

15.4.2.2.2.2 Results

Results for two feedline break cases are presented. Results for a case in which offsite power is assumed to be available are presented in Section 15.4.2.2.2.2.1. Results for a case in which offsite power is assumed to be lost following reactor trip are presented in Section 15.4.2.2.2.2.2. The calculated sequence of events for both cases is listed in Table 15.4-19. Both of the presented feedline break cases credit the power operated relief valves, because these cases were found to be the most limiting. RN 01-125

15.4.2.2.2.2.1 Feedline Rupture with Offsite Power Available

The system response following a feedwater line rupture, assuming offsite power is available, is presented in Figures 15.4-83 through 15.4-86. Results presented in Figures 15.4-84 and 15.4-86 show that pressures in the RCS and Main Steam System remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to the pressurizer power operated relief valve setpoint. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the steam generators. Figure 15.4-84 shows that the water volume in the pressurizer increases in response to the heatup and pressurizer water relief begins at 399 seconds. At approximately 2300 seconds, decay heat generation decreases to a level such that the RN 01-125

total RCS heat generation (decay heat plus pump heat) is less than emergency feedwater heat removal capability, and RCS pressure and temperature begin to decrease.

The results show that the core remains covered at all times and that no boiling occurs in the reactor coolant loops.

15.4.2.2.2.2.2 Feedline Rupture with Offsite Power Unavailable

The system response following a feedwater line rupture without offsite power available is similar to the case with offsite power available and is presented in Figures 15.4-87 through 15.4-90. However, as a result of the loss of offsite power (assumed to occur at reactor trip), the reactor coolant pumps coast down. This results in a reduction in total RCS heat generation by the amount produced by pump operation. RN 01-125

The reduction in total RCS heat generation produces a milder transient than in the case where offsite power is available. Results presented in Figures 15.4-88 and 15.4-90 show that pressure in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases as a result of coolant expansion caused by the reduction in heat transfer capability in the steam generators. Figure 15.4-88 shows that the water volume in the pressurizer increases in response to the heatup and pressurizer water relief begins at 415 seconds. At approximately 1000 seconds, a level decay heat generation decreases to less than the emergency feedwater heat removal capability, and RCS temperatures begin to decrease. The results show that the core remains covered at all times since the pressurizer does not empty. RN 01-125

15.4.2.2.3 Conclusions

Results of the analysis show that for the postulated feedline rupture, the assumed Emergency Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. Radioactivity doses from the postulated feedline rupture are less than those previously presented for the postulated steam line break.

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 reactor coolant system. In the event of a coincident loss of offsite power, or failure of the condenser steam 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-690 and is a highly ductile material, it is considered that the assumption of a complete severance is 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:

1. Pressurizer low pressure and low level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that unit.

2. 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, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates emergency feedwater addition.

3. The steam generator blowdown liquid radiation monitor and the condenser air removal system radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system and will automatically divert the steam generator blowdown flow to the Nuclear Blowdown Processing System.

4. The reactor trip automatically trips the turbine and if offsite power is available the steam dump valves open permitting steam dump to the condenser. In the event of a coincident loss of offsite power, the condenser steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the main steam safety and/or power operated relief valves.

5. Following reactor trip, the continued action of emergency feedwater supply and borated high head 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.

6. High head 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

15.4.3.2.1 Method of Analysis

In estimating the mass transfer from the RCS through the broken tube the following assumptions are made:

1. Reactor trip occurs automatically as a result of low pressurizer pressure.

2. Following the initiation of the safety injection signal, 2 centrifugal charging pumps are actuated and continue to deliver flow for 30 minutes.

3. After reactor trip, the break flow reaches equilibrium at the point where incoming safety injection flow is balanced by outgoing break flow as shown in Figure 15.4-91. The resultant break flow persists from plant trip until 30 minutes after the accident.

4. The steam generators are controlled at the safety valve setting rather than the power operated relief valve or steam dump setting.

5. 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 the amount of steam vented from each of the steam generators.

15.4.3.2.2 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 accidents. 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 can be identified by the following method. In the event of a complete tube rupture, it will be clear soon after trip that the level in one steam generator is rising more rapidly than in the others. This is a unique indication of a tube rupture accident. Also this accident could be identified by either a condenser air removal system radiation alarm, or main steamline radiation alarm, or a steam generator blowdown radiation alarm.

The operator carries out the following procedures subsequent to reactor trip which lead to isolation of the faulty steam generator and to unit cooldown.

These actions are based on the Revision 1 of the Westinghouse Owner's Group Emergency Response Guidelines (ERGs).

With Offsite Power Available:

1. Identify the ruptured steam generator by one or more of the following means: increasing steam generator water level, high radiation from any steam generator sample, high radiation from any steam generator steam line, and/or high radiation from any steam generator blowdown line.

2. Close the main steam isolation valve and isolate all other steam paths from the ruptured steam generator.

3. Minimize emergency feedwater flow to the ruptured steam generator to control secondary water level.

4. Dump steam to the condenser from the intact steam generator at a maximum rate to establish subcooling margin for subsequent RCS depressurization. The amount of RCS cooldown is determined by the ruptured steam generator pressure.

5. Decrease RCS pressure by use of normal pressurizer spray until the water level returns in the pressurizer and RCS pressure and the ruptured steam generator pressure are equal, or high pressurizer water level is attained, or minimum subcooling is attained.

6. Based on the pressurizer water level, secondary heat sink(s), RCS subcooling and increasing RCS pressure, stop all but one charging pump to minimize break flow to the secondary system. At this point, RCS pressure and the ruptured steam generator pressure should be maintained approximately equal.

7. Continue dumping steam to the condenser from the intact steam generators and simultaneously decrease RCS pressure by use of normal pressurizer spray. Decrease pressure in the ruptured steam generator by backfill, blowdown or steam release.

8. Initiate operation of residual heat removal system at the appropriate RCS conditions.

Without Offsite Power:

1. Identify the ruptured steam generator by one or more of the following means: increasing steam generator water level, high radiation from any steam generator blowdown line, and/or high radiation from any steam generator steam line.

2. Close the main steam isolation valve and isolate all other steam paths from the ruptured steam generator.

3. Minimize emergency feedwater flow to the ruptured steam generator to control secondary water level.

4. Dump steam through the intact steam generator power relief valves at a maximum rate to establish RCS subcooling margin for subsequent RCS depressurization. The amount of RCS cooldown is determined by the ruptured steam generator pressure.

5. Open pressurizer relief valve to reduce RCS pressure until water level returns in the pressurizer and RCS pressure and the ruptured steam generator pressure are equal, or high pressurizer water level is attained, or minimum subcooling is attained.

6. Based on the pressurizer water level, secondary high sink(s), RCS subcooling, and increasing RCS pressure, stop all but one charging pump to minimize break flow to the secondary system. At this point, RCS pressure and the ruptured steam generator pressure should be maintained approximately equal.

7. Continue dumping steam through the intact steam generator power relief valves and simultaneously decrease RCS pressure. Decrease pressure in the ruptured steam generator by backfill, blowdown or steam release. RN 11-007

8. Initiate operation of Residual Heat Removal System at the appropriate RCS conditions.

After the residual heat removal system is placed in operation, the condensate accumulated in the secondary system can be examined and processed as required. Table 15.4-28 presents the balance of plant equipment required for the steam generator tube rupture accident.

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.

15.4.3.2.3 Results

Figure 15.4-91 illustrates the flow rate that would result through the ruptured steam generator tube. For conservatism, the equilibrium break flow rate is based on the maximum safety injection flow rate with three charging pumps operating. The previous assumptions lead to a conservative estimate of 92,900 pounds for the total amount of reactor coolant transferred to the secondary side of the faulty steam generator as a result of a tube rupture accident.

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. Parameters used in determining the radioactivity released to the atmosphere for a steam generator tube rupture are listed in Table 15.4-29.

15.4.3.4 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 RCS to the secondary system in the steam generators. A realistic analysis and a conservative analysis based on Regulatory Guide 1.183 of the potential offsite and control room doses from a postulated Steam Generator Tube Rupture (SGTR) is presented below. RN 12-034 These analyses incorporate assumptions for defective fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activity levels in the secondary system. In addition, the conservative analysis includes the consideration of iodine spiking effects caused by the SGTR. Both pre-existing and concurrent iodine spikes are evaluated coincident with the postulated tube rupture. Parameters used in both the realistic and conservative analyses are listed in Table 15.4-29.

In the safety analysis to evaluate the radiological consequences of a steam generator tube rupture, calculations are performed to determine the primary to secondary mass transfer and steam releases from the faulted and non-faulted steam generators. The RCS depressurization rate due to the primary to secondary break flow is calculated to determine the time of reactor trip and actuation of the safety injection system. The average break flow rate during the time period from the tube rupture initiation to reactor trip and safety injection was used to calculate the integrated break flow for this period. After reactor trip and safety injection, it was assumed that the RCS stabilizes at a pressure where the incoming safety injection flow equals the flow out the broken tube, with the assumption of maximum safeguards and no spillage (which gives the highest primary to secondary flow). The resultant break flow rate is assumed to persist from reactor trip and safety injection actuation until 30 minutes after the accident. The integrated break flow values for these two time periods were then summed to yield a total primary to secondary break flow of 92,900 lb for the 30 minutes.

The steam released to the atmosphere from the faulted and non-faulted steam generators is calculated for 30 minutes following the event by a mass and energy balance between the primary and secondary system. It is assumed that the faulted steam generator is isolated after 30 minutes.

For the remainder of the transient, after faulted steam generator isolation, steam releases from the non-faulted steam generators are obtained by an energy balance between the primary and secondary systems. The energy in the primary system is calculated, including energy of the fluid, decay heat, and metal energy. This is equated with the amount of secondary steam which must be generated and released in order to remove this energy from the primary system and decrease the primary system pressure to the desired levels.

In summary, then, although a detailed analysis is not performed, the current analysis is a conservative estimate of the radiological consequences of the steam generator tube rupture.

The RADTRAD[64] computer code was used to calculate the offsite and control room doses for the steam generator tube rupture (SGTR). A schematic showing the code modeling of the activity flow paths is shown on Figure 15.4-144. RN 12-034

The following additional assumptions and parameters are used to calculate the activity releases and offsite doses for the postulated steam generator tube rupture.

1. For the realistic analysis, it is assumed that prior to the accident an equilibrium fission product activity exists in the primary and secondary systems due to primary to secondary leakage in the steam generators. In the conservative analyses, the activity concentrations are assumed in combination with a pre-accident iodine spike as well as an accident initiated iodine spike.

2. Offsite power is available throughout the accident in the realistic case and not available in the conservative analysis.

3. The reactor coolant released to the defective steam generator and associated steam releases and feedwater flows are given in Table 15.4-29.

4. No failed fuel.

5. No steam generator blowdown during the accident. RN 12-034

6. Thirty minutes after the accident, the pressure between the faulted steam generator and primary system is equalized. The faulted steam generator is isolated. No steam and fission product activities are released from the defective steam generator after this time.

7. A portion of the primary to secondary leakage through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. The leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG and enter the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited. All leakage that does not immediately flash is assumed to mix with the bulk water. The radioactivity within the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 is assumed.

8. The primary-to-secondary leak rate in the steam generators is assumed to be the leak-rate-limiting condition for operation specified in the plant requirements of 1 gpm for all three steam generators. The leakage is apportioned between the steam generators in such a manner that the calculated dose is maximized. To be consistent with preferred plant procedures for shutdown, the shutdown cooling rate following this type of event is taken as 25 ^{o}F /hr. Therefore, to go from 557 ^{o}F to 212 ^{o}F would require about 14 hours. The SGTR is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is assumed to be released to the environment via the intact steam generators for the 24 hour duration with a partition factor of 100 for iodine and particulates and 1.0 for noble gases. In addition, the secondary coolant activity initially contained in the faulted SG is released to the environment with a partition factor of 1.0. RN 12-034

9. This analysis assumes that the RCS activity conservatively remains constant throughout this pre-existing iodine spike SGTR event; i.e., no dilution of the RCS activity from the safety injection system is considered. Additionally, this evaluation assumes that the RCS mass remains constant throughout the SGTR event; i.e., no change in the RCS mass is assumed as a result of the rupture flow within the SGTR or from the safety injection system.

For the Concurrent Iodine Spike SGTR event, a similar assumption is made with the exception that the iodine activity increases during 8 hours of the transient as a result of release from the defective fuel at a rate of 335 times the iodine equilibrium appearance rate.

The following assumptions and parameters are used to calculate the control room doses for the SGTR:

1. The atmospheric diffusion parameters for the control room are provided in Table 2.3-123. The limiting control room release point for the SGTR is the Main Steam Safety Relief Valve “A” (Reliefs B, C, D, E), Control Room Intake B.

2. The control room emergency filtration system is credited after 30 minutes.

3. The control room ventilation system flow rates are shown on Figure 15.4-144.

4. The control room breathing rate is assumed as 3.47E-04 m^3/sec for the duration of the accident. RN 12-034

5. Additional control room parameters are shown on Table 15.4-17.

Isotopic releases to the environment based upon these assumptions are summarized in Tables 15.4-30 through 15.4-32.

The offsite and control room doses resulting from the postulated steam generator tube rupture accident based upon the realistic and conservative analysis assumptions are summarized in Table 15.4-33. The doses from this accident are within the limits defined in 10 CFR 50.67. RN 12-034

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 such as is discussed in Section 5.5. 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 RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power operated relief valves, and opens the pressurizer safety valves, in that sequence.

The three power operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

15.4.4.2 Analysis of Effects and Consequences

15.4.4.2.1 Method of Analysis

Two digital-computer codes are used to analyze this transient. The LOFTRAN Code is used to calculate the resulting loop and core coolant flow following the pump seizure. The LOFTRAN Code is also used to calculate the time of reactor trip, based on the calculated flow, 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 using the FACTRAN[29] Code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN Code includes the use of a film boiling heat transfer coefficient.

The following case is analyzed:

All loops operating, one locked rotor.

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

When the peak pressure is evaluated, the initial pressure is conservatively estimated as 50 psi above nominal pressure of 2250 psia to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. A flow imbalance of 5% between loops (faulted vs. intact) was assumed to incorporate the effects of loop-to-loop RCS flow asymmetry. RN 01-125

15.4.4.2.2 Evaluation of the Pressure Transient

After pump seizure and reactor trip, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin one second after the flow in the affected loop reaches 87% of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

Although these 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 assumed to achieve rated flow at 2580 psia. No relief is credited prior to reaching this pressure. This conservatively accounts for uncertainties in the nominal setpoint of 2500 psia. RN 01-125

15.4.4.2.3 Evaluation of DNB in the Core During the Accident

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

In the evaluation, the rod power at the hot spot is conservatively assumed to be approximately 2.6 times the average rod power at the initial core power level.

15.4.4.2.4 Film Boiling Coefficient

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

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

15.4.4.2.5 Fuel Clad Gap Coefficient

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

15.4.4.2.6 Zirconium Steam Reaction

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

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left(-\frac{45{,}500}{1.968T}\right)$$

Where:

w = amount reacted, mg/cm^2.

t = time, sec.

T = temperature, ° Kelvin. RN 01-125

The reaction heat is 1510 cal/gm.

15.4.4.2.7 Results

Transient values of RCS flow, faulted loop flow, nuclear power, core heat flux, RCS pressure, and hot spot clad temperature are shown in Figures 15.4-95, 15.4-96A, and 15.4-96B. RN 01-125

Maximum RCS pressure, maximum clad average temperature, and amount of zirconium-water reaction are contained in Table 15.4-34. RN 01-125

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 clad average temperature calculated for the hot spot during the worst transient remains considerably less than the 2700°F limit for ZIRLO® material and the 2375°F limit for Optimized ZIRLO™ material, the core will remain in place and intact with no consequential loss of core cooling capability. RN 11-013

3. The results of the transient analysis show that less than 15.0% of the fuel rods will have DNBR's below the safety analysis limit values.

If it is assumed that there is leakage from the RCS to the secondary system in the steam generators and that offsite power is lost following the reactor coolant pump locked rotor accident, radioactivity will be released to the atmosphere through the relief or safety valves. Parameters used in determining the amount of radioactivity released are given in Table 15.4-34a.

15.4.4.4 Environment Consequences of a Postulated Reactor Coolant Pump Locked Rotor

The postulated accidents involving release of steam from the secondary system will not result in release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generators. A realistic analysis and a conservative analysis based on Regulatory Guide 1.183 of the potential offsite and control room doses resulting from a reactor coolant pump locked rotor is presented. This analysis incorporates assumptions of defective fuel and 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 both the realistic and conservative analyses are listed in Table 15.4-34a. RN 12-034

The RADTRAD[64] computer code was used to calculate the offsite and control room doses for the reactor coolant pump locked rotor accident (RCPLRA). A schematic showing the code modeling of the activity flow paths is shown on Figure 15.4-145. RN 12-034

The following assumptions and parameters are used to calculate the activity releases and offsite doses for a reactor coolant pump locked rotor:

1. Prior to the accident, an equilibrium activity of fission products exists in the primary and secondary systems due to primary to secondary leakage in the steam generators.

2. Offsite power is lost and the main condenser is not available for steam dump.

3. Twenty four hours after the accident the Residual Heat Removal System starts operation to cool down the plant. RN 12-034

4. After 24 hours following the accident, no steam and activity are released to the environment.

5. Primary to secondary leakage is 1 gpm and is evenly distributed in the steam generators.

6. Defective fuel prior to the accident is 1%.

7. As a result of the accident, 15% of the fuel rods in the core are considered to be failed and their gap activity is considered to be released to, and instantaneously mixed with, the reactor coolant. Per Regulatory Guide 1.183, Table 3, the non-LOCA fraction of fission products inventory in the gap is acceptable for use if the peak fuel burnup does not exceed 62,000 MWD/MTU and the maximum linear heat generation rate does not exceed 6.3 kw/ft. peak rod average power for burnups exceeding 54 GWD/MTU. To account for possible variation in burnup and rod power, the Table 3 non-LOCA fraction of fission products inventory in the gap is conservatively doubled. The Regulatory Guide 1.183, Table 3 gap fractions and the gap fractions used in the RCPLRA analysis are listed as follows: RN 12-034

Group	Regulatory Guide 1.183 Fraction	RCPLRA Analysis Fraction
I-131	0.08	0.16
Kr-85	0.10	0.20
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.24

8. No Condenser Air Removal System release and no steam generator blowdown occurs during the accident.

9. No noble gas is dissolved in the steam generator water.

10. The primary-to-secondary leak rate in the steam generators is assumed to be the leak-rate-limiting condition for operation specified in the plant requirements of 1 gpm for all three steam generators. The leakage is apportioned between the steam generators in such a manner that the calculated dose is maximized. To be consistent with preferred plant procedures for shutdown, the shutdown cooling rate following this type of event is taken as 25 ^{o}F /hr. Therefore, to go from 557 ^{o}F to 212 ^{o}F would require about 14 hours. The RCPLRA is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is assumed to be released to the environment via the steam generators for the 24 hour duration with a partition factor of 100 for iodine and particulates and 1.0 for noble gases. In addition, the secondary coolant activity initially contained in the steam generators is released to the environment with a partition factor of 100. RN 12-034

Steam releases to the atmosphere for the reactor pump locked rotor accident are given in Table 15.4-34a. Assumptions for the realistic analysis are also presented in Table 15.4-34a. Isotopic releases to the environment using these assumptions are summarized by Tables 15.4-34b through 15.4-34c.

In addition, the following assumptions and parameters are used to calculate the control room doses for the RCPLRA:

1. The atmospheric diffusion parameters for the control room are provided in Table 2.3-123. The limiting control room release point for the LRA is the Main Steam Safety Relief Valve "A" (Reliefs B, C, D, E), Control Room Intake B.

2. The control room emergency filtration system is credited after 120 minutes.

3. The control room ventilation system flow rates are shown on Figure 15.4-145. RN 12-034

4. The control room breathing rate is assumed as 3.47E-04 m^3/sec for the duration of the accident.

5. Additional control room parameters are shown on Table 15.4-17.

The offsite and control room doses resulting from the RCPLRA are listed in Table 15.4-34d.

15.4.5 FUEL HANDLING ACCIDENTS

A fuel handling accident (FHA) during refueling could release a fraction of the fission product inventory in the plant to the environment. Two (2) accident scenarios are considered: (1) a refueling accident occurring inside containment and (2) a refueling accident occurring outside containment.

The RADTRAD[64] computer code was used to calculate the offsite and control room doses for both of the fuel handling accident (FHA) scenarios. A schematic showing the code modeling of the activity flow paths is shown on Figure 15.4-146. RN 12-034

15.4.5.1 Fuel Handling Accident Inside of Containment

The postulated fuel handling accident inside containment is the dropping of a spent fuel assembly onto the core during refueling which results in damage to the fuel assemblies. For this postulated accident, two analyses bases are evaluated: (1) a realistic case and (2) a conservative case. The conservative case analysis is based on Regulatory Guide 1.183 assumptions. The assumed analysis parameters and radiological consequences associated with these cases are discussed below. RN 12-034

15.4.5.1.1 Identification of Causes and Accident Description

There are numerous administrative controls and physical limitations which are imposed to prevent a fuel handling accident from occurring during refueling operations. Nevertheless, an accident sequence has been postulated with the objective of assessing the potential risk to the public health and safety.

It is postulated that a spent fuel assembly is dropped onto the core during refueling resulting in breaching of the fuel rod cladding. As a result of the damage, a portion of the volatile fission gases are released to the water pool covering the core. Subsequently, a fraction of the water soluble gases are absorbed in the pool with the remainder being transported through the water and into the Reactor Building atmosphere. The escaped gases are assumed to be released instantaneously to the environment via the Reactor Building Purge System and dispersed into the atmosphere.

15.4.5.1.2 Analysis of Effects and Consequences

15.4.5.1.2.1 Method of Analysis

The following assumptions are postulated in the calculation of the radiological consequences of a fuel handling accident inside containment:

Realistic Analyses

1. The accident occurs at 100 hours after reactor shutdown, which is the minimum time after shutdown that refueling operations could commence. Radioactive decay of the fission product inventory for this time period is taken into account.

2. A total of 314 pins are assumed to be damaged as a result of this event. All 264 pins in the dropped spent fuel assembly and 50 pins in the impacted assembly are assumed to be ruptured.

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.

4. All activity in the clad gap of the damaged fuel, as given in Table 15.4-36, is released to the reactor cavity pool.

5. The minimum water depth between the top of the damaged fuel rods and the reactor cavity pool surface is 23 feet.

6. Noble gases released to the reactor cavity pool are immediately released to the reactor building atmosphere.

7. The pool water retains a large fraction of the iodine species activity by virtue of their solubility and hydrolysis. An iodine/pool decontamination factor of 500 is used for this analysis, and is based upon the results of an experimental test program (Reference 40) conducted to evaluate the extent of the removal of iodines released from a damaged irradiated fuel assembly. The iodine/pool decontamination factor is discussed in Section 15.4.5.1.2.2.

Conservative Analysis

1. The accident occurs at 72 hours after reactor shutdown, which is a conservative minimum time after shutdown. Radioactive decay of the fission product inventory for this time period is taken into account. RN 01-125

2. A total of 314 pins are damaged as a result of this event. This includes all 264 pins in the dropped spent fuel assembly and 50 pins in the impacted fuel assembly. This is equivalent to 1.19 assemblies.

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 fuel are calculated using ORIGEN-S/ARP program [53] and the calculation assumes full-power operation at the end of core life immediately preceding shutdown. The fission product inventory in the highest-rated assembly for the conservative case is given in Table 15.4-37 and is based on a peaking factor of 1.7. Per Regulatory Guide 1.183, Table 3, the non-LOCA fraction of fission products inventory in the gap is acceptable for use if the peak fuel burnup does not exceed 62,000 MWD/MTU and the maximum linear heat generation rate does not exceed 6.3 kw/ft. peak rod average power for burnups exceeding 54 GWD/MTU. To account for possible variation in burnup and rod power, the Table 3 non-LOCA fraction of fission products inventory in the gap is conservatively doubled. The Regulatory Guide 1.183, Table 3 gap fractions and the gap fractions used in the FHA analyses are listed as follows: RN 03-017 RN 12-034

Group	Regulatory Guide 1.183 Fraction	FHA Analysis Fraction
I-131	0.08	0.16
Kr-85	0.10	0.20
Other Noble Gases	0.05	0.10
Other Halogens	0.05	0.10
Alkali Metals	0.12	0.24

4. All activity in the clad gap of the damaged fuel is released to the reactor cavity pool.

5. Noble gases released to the reactor cavity pool are immediately released to the reactor building atmosphere.

6. Per Regulatory Guide 1.183, Appendix B.1.3, the chemical form of radioiodine released from the fuel is 95% aerosol (CsI), 4.85% elemental and 0.15% organic. Due to the low pH of the water, CsI instantaneously disassociates and the iodine re-evolves as elemental, resulting in 99.85% elemental (4.85% + 95%) and 0.15% organic iodine within the water. The minimum fuel depth over the reactor core when handling fuel and over the spent fuel in the FHB is 23 feet. Therefore, RN 12-034

consistent with Regulatory Guide 1.183, Appendix B.2, the overall effective iodine decontamination factor is 200, for the reactor core pool, with a resulting chemical species released from the water of 57% elemental and 43% organic iodine. RN 12-034

15.4.5.1.2.2 Iodine Decontamination Factors

An experimental test program[40] was performed to evaluate the extent of removal of iodine released from a damaged irradiated fuel assembly. Iodine removal from the released gas takes place as the gas rises through the body of solution in the fuel storage area to the pool surface. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble in the solution.

In order to obtain all the necessary information regarding this mass transfer process, a number of small scale tests were conducted, using trace iodine and carbon dioxide in an inert carrier gas. Iodine testing was performed at the design basis solution conditions (temperature and chemistry) and data were collected for various bubble diameters and solution depths. This work resulted in the formulation of a mathematical expression for iodine decontamination factor in terms of bubble size and bubble rise time.

Similar tests were conducted with carbon dioxide in an inert carrier, except that the solution temperature and chemistry were patterned after that of a deep pool where large scale tests were also performed with carbon dioxide. The small scale carbon dioxide tests also resulted in a mathematical expression for decontamination factor in terms of bubble size and bubble rise time through the solution.

To complete the experimental program, a full size fuel assembly simulator was fabricated and placed in a deep pool for testing, where gas released would be typical of that from the postulated damaged assembly. Tests were conducted with trace carbon dioxide in an inert carrier gas and overall decontamination factors were measured as a function of the total gas volume released. These measurements, combined with the analytical expression derived from small scale tests with carbon dioxide, permitted an in situ measurement of both the effective bubble diameter and rise time, both as a function of the volume of gas released. Having measured the characteristics of large scale gas releases, the decontamination factor for iodine was obtained, using the analytical expression from small scale iodine testing.

$$\text{Decontamination Factor} = 7.3\ e^{0.313\ t/d}$$

Where:

t = rise time

d = effective bubble diameter

The overall test results clearly indicate that iodine will be readily removed from the gas rising through the spent fuel pool and that the efficiency of removal will depend on the volume of gas released instantaneously from the full void space.

With consideration given to the total quantity of gas released from a fuel assembly, i.e., 6.9 scf for the 17 x 17 array, the pool decontamination factor for iodine is indicated to be a minimum of 760 for the 26 foot depth. Thus, a decontamination factor of 760 constitutes one parameter of the expected case. In the realistic case presented here, a lower decontamination factor is selected to provide for reasonable deviation in the factors that control iodine absorption by the pool water. For the realistic analysis, a decontamination factor value of 500 is used, which is a reduction of 66% of the expected value. For the conservative analysis, the decontamination factor is further reduced to a value of 200, or less than 30% of the value that would be expected. RN 03-017 RN 12-034

15.4.5.1.3 Conclusions

A summary of the parameters and assumptions used to evaluate the consequences of a fuel handling accident are given in Table 15.4-38 for both the realistic and conservative analyses.

Activities released to the environment as a result of this event, are given in Table 15.4-40 for both the realistic and conservative cases. RN 12-034

15.4.5.1.4 Environmental Consequences of a Postulated Fuel Handling Accident Inside Containment

Following the postulated accident inside the Reactor Building, the activity released to the Reactor Building atmosphere is assumed to be released over a two hour duration to the environment through the Reactor Building Purge System. In both the realistic and conservative analyses, no credit is taken for a reduction in the amount of activity released due to filtration or radioactive decay due to holdup in the containment. A summary of the pertinent parameters used to evaluate the consequences are presented in Table 15.4-38. RN 12-034

The following additional assumptions and parameters are used to calculate the control room doses for the FHA inside containment.

1. The atmospheric diffusion parameters for the control room are provided in Table 2.3-123. The limiting control room release point for the FHA inside of containment is the nearest point to the reactor building control room intake “A”. RN 12-034

2. The control room emergency filtration system is credited after 30 minutes.

3. The control room ventilation system flow rates are shown on Figure 15.4-146.

4. The control room breathing rate is assumed as 3.47E-04 m^3/sec for the duration of the accident.

5. Additional control room parameters are shown on Table 15.4-17. RN 12-034

The offsite and control room radiation doses resulting from both the realistic and conservative analyses of a postulated fuel handling accident inside containment are presented in Table 15.4-41.

15.4.5.2 Fuel Handling Accident Outside Containment

The fuel handling accident outside containment is postulated as the dropping of a spent fuel assembly into the Spent Fuel Pool which results in damage to the fuel assemblies and the release of the volatile gaseous fission products. Similar to the evaluation of the FHA inside containment, realistic and conservative analyses are performed for this postulated accident.

The identification of causes and description of this accident is identical to the FHA inside containment with the exception that the escaped gaseous fission products are released to the spent fuel pool and subsequently to the Fuel Handling Building. These gases are released to the environment via the fuel handling building charcoal exhaust system.

The conditions and parameters assumed in analyzing the effects and consequences of this accident are identical to those utilized in the FHA inside containment except that the activity released to the environment is treated by the HEPA and charcoal filters of the Fuel Handling Building Exhaust System. No credit is taken for these filters in the analysis of the environmental doses. Accordingly, the activity released to the environment is identical to that presented for the FHA inside containment and shown in Table 15.4-40 for the realistic and conservative cases respectively. RN 12-034

15.4.5.2.1 Environmental Consequences of a Postulated Fuel Handling Accident Outside of Containment

Following a postulated FHA outside containment, a quantity of airborne radioactivity would be released to the environment via the fuel handling building charcoal exhaust system.

In both the realistic and conservative analyses, no credit is taken for the mixing of the activity released with the fuel building atmosphere nor for radioactive decay due to holdup in the building or transit time after release to the environs. A summary of the pertinent parameters used to evaluate the consequences are presented in Table 15.4-38. Isotopic releases to the environment are presented in Table 15.4-40. RN 12-034

The following additional assumptions and parameters are used to calculate the control room doses for the FHA outside containment.

1. The atmospheric diffusion parameters for the control room are provided in Table 2.3-123. The limiting control room release point for the FHA outside of containment is the Main Plant Vent.

2. The control room emergency filtration system is credited after 30 minutes. RN 12-034

3. The control room ventilation system flow rates are shown on Figure 15.4-146.

4. The control room breathing rate is assumed as 3.47E-04 m^3/sec for the duration of the accident.

5. Additional control room parameters are shown on Table 15.4-17.

The offsite and control room radiation doses resulting from both the realistic and conservative analyses of the postulated fuel handling accident outside containment are presented in Table 15.4-50.

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 (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.6.1.1 Design Precautions and Protection

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

15.4.6.1.1.1 Mechanical Design

The mechanical design is discussed in FSAR Section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of a RCCA drive mechanism housing failure are listed below:

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

2. The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed 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-type rod welds which are subject to periodic inspections.

15.4.6.1.1.2 Nuclear Design

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

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

15.4.6.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference [30]. The protection for this accident is provided by high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

15.4.6.1.1.4 Effects on Adjacent Housings

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

The operating coil stack assembly of this mechanism has a 10.718 inch by 10.718 cross section and 39.875 inch length. The position indicator coil stack assembly is located above the operating coil stack assembly. It surrounds the rod travel housing over nearly its entire 163.24 inch length. The rod travel housing outside diameter is 3.75 inches and the position indicator coil stack assembly inside and outside diameters are 3.75 inches and 7.0 inches, respectively. This assembly consists of a steel tube surrounded by a continuous stack of copper wire coils. The assembly is held together by two end plates, an outer sleeve, and four axial tie rods.

15.4.6.1.1.5 Effects of Rod Travel Housing Longitudinal Failures

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

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

15.4.6.1.1.6 Effect of Rod Travel Housing Circumferential Failures

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

jet from the break would continue to push the broken-off piece against the missile shield.

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

15.4.6.1.1.7 Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCA's are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal, however this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.6.1.1.8 Summary

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

15.4.6.1.2 Limiting Criteria

Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation[31]. 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[32] results, which indicated a failure threshold at 280 cal/gm. Limited results have indicated 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 [49] are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

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

2. Peak reactor coolant pressure less than that which 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

15.4.6.2.1 Method of Analysis

The analysis of the RCCA ejection accident is performed in two stages, first an average core nuclear power 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 [33].

15.4.6.2.2 Average Core Analysis

The spatial kinetics computer code, TWINKLE[34], is used for the average core transient analysis. This code solves the 2 group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for 6 delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects.

In this analysis, the code is used as a 1 dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA 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 Section 15.1.9.

15.4.6.2.3 Hot Spot Analysis

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

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN[29]. 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 distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter[35] or Jens-Lottes[36] correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong correlation[37] 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 pellet temperature distribution to agree with the fuel heat transfer design codes.

For full power cases, the design initial hot channel factor (F_Q) is input to the code. The hot channel factor during the transient is assumed to increase from 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. Further description of FACTRAN appears in Section 15.1.9.

15.4.6.2.4 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 RCS and heat transfer to the

steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.[38]

15.4.6.2.5 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 15.4-42 presents the parameters used in this analysis.

15.4.6.2.6 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using three dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in 98-01
the calculations.

Appropriate margins are added to the results to allow for calculational uncertainties, including an allowance for nuclear power peaking due to densification.

15.4.6.2.7 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 have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, the axial weighting is not necessary. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to 3 dimensional analysis.

15.4.6.2.8 Moderator and Doppler Coefficient

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

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1. The resulting curve is conservative compared to design predictions for this plant. The Doppler weighting factor should be larger than 1.0 (approximately 1.2), just to make the present calculation agree with design predictions before ejection. This weighting factor will increase under accident conditions, as discussed above. The Doppler defect used as an initial condition is 900 pcm at the beginning of life (BOL) and 840 pcm at the end of life (EOL).

15.4.6.2.9 Delayed Neutron Fraction

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 β_{eff} if the ejected rod worth is nearly equal to or greater than β_{eff} as in zero power transients. In order to allow for future cycles, pessimistic estimates of β_{eff} of 0.54% at beginning of cycle and 0.44% at end of cycle were used in the analysis.

15.4.6.2.10 Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-42 and includes the effect of one stuck RCCA. 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 analyses presented are applicable for a rod insertion time of 2.7 seconds from coil release to entrance to the dashpot. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, and adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations for this plant have shown that the effect of 2 stuck RCCA's (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1% Δk. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed for the Virgil C. Summer Nuclear Station assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The safety injection system is actuated on low pressurizer pressure within one minute after the break. The RCS pressure continues to

drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about 2 to 3 minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the RCS temperature below no-load by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2% Δk due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than ten minutes after the break. The addition of borated safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

15.4.6.2.11 Results

The values of the parameters used in the analysis, as well as the results of the analysis, are presented in Table 15.4-42 and discussed below.

1. Beginning of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively assumed to be 0.20% Δk and 6.0 respectively. The peak hot spot clad average temperature was 2520°F. The peak hot spot fuel center temperature exceeded the BOL melting temperature of 4900°F. However, melting was restricted to less than 10% of the pellet.

2. Beginning of Cycle, Zero Power

   For this condition, control bank D was assumed to be fully inserted and bank C was at its insertion limit. The worst ejected rod is located in control bank D and was conservatively assumed to have a worth of 0.855% Δk and a hot channel factor of 13. The peak hot spot clad temperature reached 2580°F. The peak hot spot fuel center temperature was 4825°F.

3. End of Cycle, Full Power

   Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively assumed to be 0.21% ΔK and 6.5 respectively. This resulted in a peak hot spot clad temperature of 2419°F The peak hot spot fuel center temperature exceeded the EOL melting temperature of 4800°F. However, melting was restricted to less than 10% of the pellet.

4. End of Cycle, Zero Power

   Control Bank D was assumed to be fully inserted and C was at its insertion limit. The ejected rod worth and hot channel factor were conservatively assumed to be 0.90% Δk and 22.5 respectively. The peak clad average and fuel center temperature was 2415°F and 4198°F, respectively.

A summary of the cases presented above is given in Table 15.4-42. The nuclear power and hot spot fuel and clad temperature transients for the worst cases (beginning of life full power and zero power) are presented in Figures 15.4-100 through 15.4-101.

15.4.6.2.12 Fission Product Release

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

15.4.6.2.13 Pressure Surge

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

15.4.6.2.14 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 ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.6.3 Conclusions

Even on a conservation basis, the analyses indicate that the described fuel and limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses

to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10%[38] .

15.4.6.4 Environmental Consequences of a Postulated Rod Ejection Accident

Two analyses of a postulated rod ejection accident are performed: a realistic analysis and a conservative Regulatory Guide 1.183 analysis. The parameters used for each of these analyses are listed in Table 15.4-43. RN 12-034

The realistic analysis of the doses resulting from a rod ejection accident is based upon the release to the containment of all volatile fission products in the RCS.

Prior to the accident it is assumed that the plant has been operating with 0.12% fuel defects and steam generator tube leakage of 100 lb./day for a sufficient period of time to establish equilibrium levels of activity in the primary and secondary systems. Release from contaminated steam dumped to the condenser is considered negligible.

For the conservative Regulatory Guide 1.183 analysis, it is assumed that the plant is operating at equilibrium levels of radioactivity in the primary and secondary systems prior to the postulated rod ejection accident as a result of coincident fuel defects (1%) and steam generator tube leakage (1 gpm). Following a postulated rod ejection accident, two activity release paths are considered. The first release path is via containment leakage of activity released to the containment from the reactor coolant. The second path is via the contaminated steam from the secondary system which is released through the relief valves since it is assumed that offsite power is lost. RN 12-034

15.4.6.4.1 Model

The RADTRAD[64] computer code was used to calculate the offsite and control room doses for the control rod ejection accident (CREA). Schematics showing the code modeling of the activity flow paths are provided on Figures 15.4-147 (steam generator release path) and 15.4-148 (containment release path). Parameters and assumptions used in evaluating offsite doses for this event are summarized in Table 15.4-43. RN 12-034

Prior to the accident it is assumed that the plant has been operating with simultaneous fuel defects and steam generator tube leakage for a sufficient period of time to establish equilibrium levels of activity in the primary and secondary systems. 02-01

Following a postulated rod ejection accident, the activity released is assumed to be instantaneously mixed uniformly throughout the reactor coolant. Thus, the total activity released from the fuel rod gaps is assumed to be immediately available for release from the RCS.

Of the activity released with the reactor coolant to the containment by the postulated failure of the RCCA mechanism pressure housing, 100% is assumed to be mixed

instantaneously throughout the containment and to be available for leakage from the containment at the design leak rate. The only removal processes considered within containment are radioactive decay and leakage from the containment.

The model for the activity available for release to the atmosphere from the relief valves assumes that the release consists of the activity in the secondary coolant prior to the accident plus that fraction of the activity leaking from the reactor coolant through the steam generator tubes following the accident. The leakage of reactor coolant to the secondary side of the steam generators is assumed to continue at its initial rate, i.e., the same leakage rate that existed prior to the accident, until primary and secondary system pressure is equalized. No mass transfer from the primary to the secondary system through steam generator tube leakage is assumed after system pressures are equalized. RN 12-034

15.4.6.4.2 Assumptions for Conservative Analysis

The following conservative assumptions were used in the analysis of the release of radioactivity to the environment in the event of a postulated rod ejection accident. A summary of parameters used in the analysis is given by Table 15.4-43.

1. The primary-to-secondary leak rate in the steam generators is assumed to be the leak-rate-limiting condition for operation specified in the plant requirements of 1 gpm for all three steam generators. To be consistent with preferred plant procedures for shutdown, the shutdown cooling rate following this type of event is taken as 25 ^{o}F /hr. Therefore, to go from 557 ^{o}F to 212 ^{o}F would require about 14 hours. The CREA is conservatively analyzed using 24 hours for the cooldown time. The activity associated with the 1 gpm leak is assumed to be released to the environment via the intact steam generators for the 24 hour duration with a partition factor of 100 for iodine and particulates and 1.0 for noble gases. In addition, the secondary coolant iodine and particulate activity initially contained in the secondary system is released to the environment with a partition factor of 100.

2. Per Regulatory Guide 1.183, Table 3, the non-LOCA fraction of fission products inventory in the gap is acceptable for use if the peak fuel burnup does not exceed 62,000 MWD/MTU and the maximum linear heat generation rate does not exceed 6.3 kw/ft. peak rod average power for burnups exceeding 54 GWD/MTU. To account for possible variation in burnup and rod power, the Table 3 non-LOCA fraction of fission products inventory in the gap is conservatively doubled. The Regulatory Guide 1.183, Table 3 gap fractions and the gap fractions used in the CREA analysis are listed as follows: RN 12-034

Group	Regulatory Guide 1.183 Fraction	CREA Analysis Fraction
Noble Gases	0.10	0.20
Halogens	0.10	0.20
Alkali Metals	0.12	0.24

3. Instantaneous mixing of all activity released from the reactor coolant into the containment is assumed.

4. No credit is assumed for removal of iodine within containment by the reactor building sprays. RN 12-034

Design criteria applied to ensure that fuel dispersal into the reactor coolant will not occur include:

> "Fuel melting limited to less than the innermost
> 10% of the fuel pellet at the hot spot . . . " [38]

Even though centerline melting in a small fraction of the core is not expected, a conservative upper limit of fission product release from the core as a result of a postulated rod ejection accident can be estimated. This limit would include the release of 100% of the noble gases and 50% of the iodines from that portion of the fuel which could experience centerline melting under the above criteria.

The upper limit of fission product release from the core for this very conservative case is determined using the following assumptions:

1. It is assumed that 100% of the noble gases and iodines in the clad gaps of the fuel rods experiencing clad damage (assumed to be 10% of the rods in the core[38]) is assumed to be released to the reactor coolant.

2. It is assumed that 50% of the iodines and 100% of the noble gases in the fuel that melts are released to the reactor coolant. This is a very conservative assumption since only centerline melting could occur for a maximum time period of six seconds.

3. The fraction of fuel melting is conservatively assumed to be one quarter of one percent of the core, determined by the following method:

    a. A conservative upper limit of 50% of the rods experiencing clad damage may experience centerline melting (a total of 5% of the core).

    b. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10% of the rod volume will actually melt (equivalent to 0.5% of the core that could experience melting).

    c. A conservative maximum of 50% of the axial length of the rod will experience melting due to the power distribution (0.5 of the 0.5% of the core equals 0.25% of the core).

The release fraction from the melted fuel is based on Attachment H and Regulatory Position 3.2, Table 2 of Regulatory Guide 1.183, 100% for the nobles, 25% for the iodines, and 30% for the alkali metals for the containment release pathway and 100% for the nobles, 50% for the iodines, and 50% for the alkali metals for the steam generator release pathway.

The following additional assumptions and parameters are used to calculate the control room doses for the CREA.

1. The atmospheric diffusion parameters for the control room are provided in Table 2.3-123. The limiting CR release point for the CREA is the RB nearest point – Control Room Intake A for the containment release pathway and the Main Steam Safety Relief Valve “A” (Reliefs B, C, D, E), Control Room Intake B for the SG release pathway. RN 12-034

2. The control room emergency filtration system is credited after 30 minutes for the containment release case and 120 minutes for the SG release case.

3. The control room ventilation system flow rates are shown on Figures 15.4-147 and 15.4-148.

4. The control room breathing rate is assumed as 3.47E-04 m^3/sec for the duration of the accident.

5. Additional control room parameters are shown on Table 15.4-17.

   The remainder of the assumptions and parameters used to calculate the activity release from the plant and the subsequent offsite doses for the ultraconservative analysis are identical to those used for the conservative analysis.

15.4.6.4.3 Results

Isotopic releases are summarized in Tables 15.4-44 through 15.4-46. The calculated offsite and control room doses are provided in Table 15.4-47. RN 12-034

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TABLE 15.4-1a

LARGE BREAK LOCA CONTAINMENT DATA USED FOR CALCULATION OF CONTAINMENT PRESSURE

Net Free Volume	1,900,000 ft^3
Initial Conditions Pressure Temperature RWST temperature (Spilling SI and Spray) Temperature outside containment	 14.7 psia 90° F 55° F 19° F
Spray System Post-accident spray system initiation delay without LOOP Maximum spray system delivered flow (2 pumps operating)	 39.6 sec 6000 gpm
Containment Fan Coolers Post-accident initiation fan coolers without LOOP Number of post-accident fan coolers operating Heat Removal Rate	 33 sec 2 Table 15.4-1b
Wall Data	Table 6.2-60

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TABLE 15.4-1b

LARGE BREAK LOCA FAN COOLER PERFORMANCE DATA

Containment Temperature (°F)	Performance per Fan Cooler (BTU/sec)
80	8250
150	8250
175	13028
200	20389
225	27778
250	37222

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TABLE 15.4-1c

KEY LOCA PARAMETERS AND REFERENCE TRANSIENT ASSUMPTIONS

Parameter	Reference Transient	Uncertainty or Bias
1.0 Plant Physical Description		
a. Dimensions	Nominal	ΔPCT_{MOD} [1]
b. Flow resistance	Nominal	ΔPCT_{MOD} [1]
c. Pressurizer location	Intact Loop	Bounded
d. Hot assembly location	Under limiting location	Bounded
e. Hot assembly type	17x17 VANTAGE PLUS w/IFMs and ZIRLO™ clad	Bounded
f. SG tube plugging level	Minimum (0%) (4)	Bounded (4)
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate (AFLUX)	Nominal - Based on 100% of power (2900 MWt)	ΔPCT_{PD} [2]
b. Hot Rod Peak linear heat rate (PLHR)	Conservatively derived from intended Tech Spec (TS) limit F_Q = 2.50 and maximum baseload F_Q= 2.0. Analyzed F_Q is 2.215.	ΔPCT_{PD} [2]
c. Hot rod average linear heat rate (HRFLUX)	Derived from intended Tech Spec $F_{\Delta H}$ = 1.70. Analyzed $F_{\Delta H}$ is 1.733	ΔPCT_{PD} [2]
d. Hot assembly average heat rate (HAFLUX)	HRFLUX/1.04	ΔPCT_{PD} [2]
e. Hot assembly peak heat rate (HAPHR)	PLHR/1.04	ΔPCT_{PD} [2]
f. Axial power distribution (PBOT, PMID)	Power Shape 10 Figure 7.2-10 (7)	ΔPCT_{PD} [2]
g. Low power region relative power (PLOW)	minimum (0.2) (4)	Bounded(4)
h. Hot assembly burnup	BOL	Bounded
i. Prior operating history	Equilibrium decay heat	Bounded
j. Moderator Temperature Coefficient (MTC)	Tech Spec Maximum (0)	Bounded
k. HFP boron	800 ppm	Generic

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TABLE 15.4-1c (Continued)

KEY LOCA PARAMETERS AND REFERENCE TRANSIENT ASSUMPTIONS

Parameter	Reference Transient	Uncertainty or Bias
2.2 Fluid Conditions		
a. T_{avg}	Low Tavg Window Nominal (572.0° F) (4)	$\Delta PCT_{IC}{}^{3,4}$
b. Pressurizer pressure	Nominal (2250.0 psia)	$\Delta PCT_{IC}{}^{3}$
c. Loop flow	Equivalent to TDF = 92,600 gpm (6)	$\Delta PCT_{MOD}{}^{1,5}$
d. T_{UH}	T_{COLD}	0
e. Pressurizer level	Nominal (60%)	0
f. Accumulator temperature	Nominal (100° F)	$\Delta PCT_{IC}{}^{3}$
g. Accumulator pressure	Nominal (642.7 psia)	$\Delta PCT_{IC}{}^{3}$
h. Accumulator liquid volume	Nominal (1014 ft^3)	$\Delta PCT_{IC}{}^{3}$
i. Accumulator line resistance	Nominal	$\Delta PCT_{IC}{}^{3}$
j. Accumulator boron	Minimum (2200 ppm)	Bounded
3.0 Accident Boundary Conditions		
a. Break location	Cold leg	Bounded
b. Break type	Guillotine	$\Delta PCT_{MOD}{}^{1}$
c. Break Size	Nominal (cold leg area)	$\Delta PCT_{MOD}{}^{1}$
d. Offsite power	On (RCS pumps running) (4)	Bounded (4)
e. Safety injection flow	Minimum (Table 15.4-1i)	Bounded
f. Safety injection temperature	Nominal (75.0° F)	$\Delta PCT_{IC}{}^{3}$
g. Safety injection delay	Max delay (22.0 sec - No LOOP value) (4)	Bounded (4)
h. Containment pressure	Minimum based on COCO containment pressure calculation results (Figure 15.4-1o) using plant conditions supplied in Tables 15.4-1a and 15.4-1b and the Reference Transient mass and energy release.	Bounded
i. Single failure	ECCS: Loss of 1 SI train Containment pressure: no failures, two trains in operation	Bounded
j. Control rod drop time	No control rods	Bounded

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TABLE 15.4-1c (Continued)

KEY LOCA PARAMETERS AND REFERENCE TRANSIENT ASSUMPTIONS

Parameter	Reference Transient	Uncertainty or Bias
4.0 Model Parameters		
a. Critical flow	Nominal (C_D = 1.0)	ΔPCT_{MOD}[1]
b. Resistance uncertainties in broken loop	Nominal (as coded)	ΔPCT_{MOD}[1]
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	ΔPCT_{MOD}[1]
d. Core heat transfer	Nominal (as coded)	ΔPCT_{MOD}[1]
e. Delivery and bypassing of ECC	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Non-condensible bases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	ΔPCT_{MOD}[1]

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Notes:

1. PCT_{MOD} indicates this uncertainty is part of code and global model uncertainty.
2. PCT_{PD} indicates this uncertainty is part of power distribution uncertainty.
3. PCT_{IC} indicates this uncertainty is part of initial condition uncertainty.
4. Confirmatory Parameter (Items 1.0f, 2.1g, 2.2a, 3.0d / 3.0g): value confirmed limiting in Section 5 of Reference 60.
5. Item 2.2c Loop Flow: Uncertainty / Bias assumed to be result of loop resistance uncertainty.
6. Item 2.2c Loop Flow: Cases @ Hi SGTP model 92,600 gpm/loop flow resistance. Reference Case @ 0% SGTP maintains same flow resistance and hence loop flow increases accordingly.
7. Item 2.0f. Hot Rod and Hot Assembly Rod values are at Figure 7.2-10 / 1.04 as noted in detail in Section 3.2.1 text of Reference 60.

TABLE 15.4-1d

CONFIRMATORY CASES PCT RESULTS SUMMARY

Case	Confirmatory Configuration				PCT Results (°F)		
	Offsite Power	**SGTP**	**PLOW**	**RCS T_{avg}**	**Blowdown**	**2nd Reflood (middle reflood)**	**2nd Reflood (late reflood)**
1. Final Reference Transient (Section 4.0 of Reference 60) (2)	No LOOP	Low	Low	Low	1634	1735	Non-existent
2. Interim Reference Transient : Combinative Case (1)	No LOOP	Low	Low	Low	1634	1820	1953
3. Offsite Power	LOOP	High	Low	High	1590	1595	Non-existent
4. Reduced SGTP	No LOOP	Low	Low	High	1651	1676	Non-existent
5. Increased PLOW	No LOOP	High	High	High	1552	1480	Non-existent
6. Low RCS T_{avg}	No LOOP	High	Low	Low	1641	1768	1796
7. Base	No LOOP	High	Low	High	1642	1664	Non-existent

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Values: Low Tavg=572.0, High Tavg = 587.4, Low PLOW = 0.2, High PLOW = 0.8, Low SGTP = 0%, High SGTP = 10%.

(1) Results of Individual Cases 3,4,5,6 versus 7 determines No-LOOP, Low SGTP, Low PLOW, Low T_{avg} limiting respectively. Hence Combinative Case 2 is run at No-LOOP, Low SGTP, Low PLOW, Low T_{avg}. Case 2 is more limiting than any of Cases 3-7 and thus establishes the Reference Transient configuration.

(2) Differences between Case 1 (Final Reference Transient) and Case 2 (interim reference transient).
- Improved RHR Performance. Cases 2-7 modeled slightly reduced SI in comparison to the Table 15.4-1i values employed in the Reference Case. At 20 psig, the difference is ~6%. The difference is associated with the assumed containment backpressure for the spilling broken loop line.
- Minor change to the containment backpressure inputs in the BREAK components. The final reference transient utilized inputs based on Figure 15.4-1o COCO calculation. The values in the final reference transient are 1-2 psi lower.
- Minor updates to a couple initial condition, accumulator line friction and vessel geometry terms

TABLE 15.4-1e

SUMMARY OF SPLIT BREAK STUDIES

CD [(a)]	Overall PCT (°F)	Time Period
0.6	1164	Middle Reflood
0.8	1286	Middle Reflood
1.0	1430	Middle Reflood
1.2	1516	Middle Reflood
1.4	1588	Middle Reflood
1.6 (Limiting Case)	1647	Middle Reflood
1.8	1576	Blowdown
2.0	1564	Blowdown

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(a) CD = Split Flow Area/Cold Leg Cross-Sectional Area (=4.125 ft^2)

Middle Reflood is the early time period of Second Reflood as described in Section 4.3 of Reference 60, namely the period prior to downcomer / lower plenum boiling.

TABLE 15.4-1f

OVERALL PCT RESULTS* FOR BEST ESTIMATE LARGE BREAK LOCA ANALYSIS

Component	Blowdown Peak (°F)	First Reflood Peak (°F)	Second Reflood Peak (°F)
$PCT^{50\%}$	< 1596	< 1488	< 1608
$PCT^{95\%}$	< 1860	< 1808	< 1988

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* Refer to the latest 10 CFR 50.46 annual or 30-day report for the current values.

TABLE 15.4-1g

PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

Parameter			Operating Range
1.0	**Plant Physical Description**		
	a)	Dimensions	No in-board assembly grid deformation during LOCA + SSE
	b)	Flow resistance	N/A
	c)	Pressurizer location	N/A
	d)	Hot assembly location	Anywhere in core interior (129 locations) [a]
	e)	Hot assembly type	17X17 V+ w/ IFMs & ZIRLO™ clad non-IFBA and IFBA [b]
	f)	SG tube plugging level	≤ 10%, SG Model Δ75
2.0	**Plant Initial Operating Conditions**		
	2.1	Reactor Power	
	a)	Core avg linear heat rate	Core power ≤ 102 % of 2900 MWt @ 2.0 % Calorimetric
	b)	Peak linear heat rate	$F_Q \leq 2.5$
	c)	Hot rod average linear heat rate	$F_{\Delta H} \leq 1.7$
	d)	Hot assembly average linear heat rate	$\bar{P}_{HA} \leq 1.70 / 1.04$
	e)	Hot assembly peak linear heat rate	$F_{QHA} \leq 2.50 / 1.04$
	f)	Axial power dist (PBOT, PMID)	Figure 15.4-1q (dashed lines)
	g)	28 assembly peripheral region relative power (PLOW)	0.2 ≤ PLOW ≤ 0.8 (see Figure 15.4-1p for peripheral locations)
	h)	Hot assembly burnup	≤ 75000 MWD / MTU, lead rod
	i)	Prior operating history	All normal operating histories
	j)	MTC	≤ 0 at HFP
	k)	HFP boron	≥ 800 ppm (BOC)
	l)	Rod power census	Table 15.4-1h

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TABLE 15.4-1g (Continued)

PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

Parameter		Operating Range
	2.2 Fluid Conditions	
	a) T_{avg}	$572.0 \pm 5.3 \leq T_{avg} \leq 587.4 \pm 5.3°$ F
	b) Pressurizer pressure	P_{RCS} = 2250 psia ± 100 psi
	c) Loop flow	≥ 92,600 gpm / loop
	d) T_{UH}	Current upper internals, Tcold UH
	e) Pressurizer level	Normal 60% level, automatic control
	f) Accumulator temperature	$85 \leq T_{ACC} \leq 115°$ F
	g) Accumulator pressure	$570 \leq P_{ACC} \leq 686$ psig
	h) Accumulator volume	$994 \leq V_{ACC} \leq 1034$ ft^3
	i) Accumulator fL/D	Current line configuration (based on as-tested 1981 pre-operational test results)
	j) Minimum accumulator boron	≥ 2200 ppm
3.0	**Accident Boundary Conditions**	
	a) Break location	N/A
	b) Break type	N/A
	c) Break size	N/A
	d) Offsite power	Available or LOOP
	e) Safety injection flow	Table 15.4-1i
	f) Safety injection temperature	55 ≤ SI Temp ≤ 95° F
	g) Safety injection delay	≤ 22 seconds (with offsite power) ≤ 32 seconds (with LOOP)
	h) Containment pressure	Bounded, see Figure 15.4-1o; Raw Data Tables 15.4-1a and 15.4-1b
	i) Single failure	Loss of one train of pumped ECCS
	j) Control rod drop time	N/A
	k) Hi-1 Containment Pressure (for SI actuation)	≤ 18.9 psia

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(a) 28 Peripheral locations (Figure 15.4-1p) will not physically be lead power assembly.
(b) The analysis was performed considering ZIRLO® cladding; an evaluation was performed addressing Optimized ZIRLO™ cladding.

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TABLE 15.4-1h

ROD CENSUS USED IN BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

Rod Group	**Power Ratio (Relative to HA Rod Power)**	**% of Core**
1	1.0	10
2	0.912	10
3	0.853	10
4	0.794	10
5	0.735	10
6	0.676	10
7	< 0.65	40

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TABLE 15.4-1i

BEST ESTIMATE LARGE BREAK LOCA TOTAL MINIMUM INJECTED SI FLOW
(TOTAL CHG / SI AND RHR INTO 2 INTACT LOOPS)

RCS Pressure (psig)	Flow Rate (gpm)
0	2709
15	2561
20	2467
30	2270
40	2064
50	1848
60	1612
70	1356
80	1071
90	699
100	305
200	295
300	285
400	274

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TABLE 15.4-6

POST ACCIDENT CONTAINMENT TEMPERATURE TRANSIENT USED IN THE CALCULATION OF ALUMINUM AND ZINC CORROSION

Time Interval (seconds)	Temperature (°F)
0	278.0
1,996	278.0
4,000	251.6
8,001	245.7
9,996	225.5
20,002	206.9
40,003	194.2
79,998	189.2
99,999	181.0
199,999	166.8
399,997	156.1
800,004	156.1
999,994	153.1
3,440,016	140.3
3,879,965	130.0
8,640,000	130.0

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TABLE 15.4-7

PARAMETERS USED TO DETERMINE HYDROGEN GENERATION

Core Thermal Power	2958 MWt	98-01
Containment Free Volume	1.84×10^6 ft^3	
Containment Temperature at Accident	120°F	
Weight Zirconium (Active Fuel Clad)	35,244 lb	
Hydrogen Generated Zirconium-Water Reaction Based on 5.0% value	14,987 SCF	RN 03-005
Hydrogen from Reactor Coolant System	909 SCF	98-01
Corrodable Metal	Aluminum and Zinc	

INVENTORY OF ALUMINUM AND ZINC IN CONTAINMENT

	Weight (lbs)	Surface Area (ft^2)	
Aluminum	2,286	1,281	RN 03-005
Zinc	32,747	87,386	

TABLE 15.4-11

ACTIVITY AVAILABLE FOR RELEASE
CORE RELEASE CASE (REGULATORY GUIDE 1.183)

Isotope	Curies	Isotope	Curies	Isotope	Curies
Co-58	7.55E+05	Ru-103	1.06E+08	Cs-136	3.08E+06
Co-60	5.78E+05	Ru-105	6.92E+07	Cs-137	5.66E+06
Kr-85	8.30E+05	Ru-106	2.42E+07	Ba-139	1.47E+08
Kr-85m	2.72E+07	Rh-105	4.79E+07	Ba-140	1.46E+08
Kr-87	4.96E+07	Sb-127	6.53E+06	La-140	1.49E+08
Kr-88	6.71E+07	Sb-129	2.31E+07	La-141	1.37E+08
Rb-86	4.43E+04	Te-127	6.31E+06	La-142	1.32E+08
Sr-89	8.41E+07	Te-127m	8.35E+05	Ce-141	1.32E+08
Sr-90	4.54E+06	Te-129	2.17E+07	Ce-143	1.29E+08
Sr-91	1.08E+08	Te-129m	5.72E+06	Ce-144	7.98E+07
Sr-92	1.13E+08	Te-131m	1.10E+07	Pr-143	1.26E+08
Y-90	4.87E+06	Te-132	1.09E+08	Nd-147	5.65E+07
Y-91	1.02E+08	I-131	8.20E+07	Np-239	1.51E+09
Y-92	1.13E+08	I-132	1.20E+08	Pu-238	8.58E+04
Y-93	1.28E+08	I-133	1.68E+08	Pu-239	1.94E+04
Zr-95	1.29E+08	I-134	1.80E+08	Pu-240	2.44E+04
Zr-97	1.35E+08	I-135	1.54E+08	Pu-241	4.11E+06
Nb-95	1.22E+08	Xe-133	1.70E+08	Am-241	2.72E+03
Mo-99	1.43E+08	Xe-135	3.70E+07	Cm-242	1.04E+06
Tc-99m	1.23E+08	Cs-134	1.01E+07	Cm-244	6.08E+04

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TABLE 15.4-12

FRACTION OF FISSION PRODUCT INVENTORY RELEASED

Group	Isotopes	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	Xe, Kr	0.05	0.95	1.00
Halogens	I, Br	0.05	0.35	0.40
Alkali Metals	Cs, Rb	0.05	0.25	0.30
Tellurium Metals	Te, Sb, Se	0.00	0.05	0.05
Ba, Sr	Ba, Sr	0.00	0.02	0.02
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co	0.00	0.0025	0.0025
Cerium Group	Ce, Pu, Np	0.00	0.0005	0.0005
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0.00	0.0002	0.0002

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LOCA RELEASE PHASES

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

TABLE 15.4-13

PRIMARY COOLANT ACTIVITY AVAILABLE FOR RELEASE
REALISTIC CASE

Isotope	Curies	Isotope	Curies	Isotope	Curies
H-3	6.30E+02	Y-90	1.03E-02	I-130	2.97E+02
Cr-51	9.90E-01	Y-91	9.72E-02	I-131	2.70E+04
Mn-54	7.38E-02	Y-91m	5.22E-01	I-132	2.79E+04
Mn-56	3.96E+00	Y-92	1.98E-01	I-133	4.14E+04
Fe-59	9.36E-02	Y-93	6.84E-02	I-134	5.40E+03
Co-58	2.52E+00	Zr-95	1.21E-01	I-135	2.16E+04
Co-60	2.34E-01	Nb-95	1.21E-01	Xe-131m	6.21E+02
Br-83	8.01E+02	Mo-99	1.42E+02	Xe-133	7.83E+04
Br-84	3.78E+02	Tc-99m	1.51E+02	Xe-133m	5.13E+03
Kr-83m	1.16E+02	Ru-103	1.15E-01	Xe-135	2.32E+03
Kr-85	2.05E+03	Ru-106	3.78E-02	Xe-135m	1.40E+02
Kr-85m	4.86E+02	Ag-110m	5.40E-01	Cs-134	7.92E+02
Kr-87	2.97E+02	Te-125m	8.46E-02	Cs-136	8.10E+02
Kr-88	8.64E+02	Te-127	2.70E+00	Cs-137	3.78E+02
Rb-86	6.48E+00	Te-127m	6.48E-01	Cs-138	1.75E+02
Rb-88	6.84E+02	Te-129	3.60E+00	Ba-140	7.92E-01
Rb-89	3.24E+01	Te-129m	3.78E+00	La-140	2.52E-01
Sr-89	7.20E-01	Te-131	2.88E+00	Ce-141	1.24E-01
Sr-90	3.60E-02	Te-131m	5.22E+00	Ce-143	9.36E-02
Sr-91	9.54E-01	Te-132	5.22E+01	Ce-144	8.46E-02
Sr-92	2.16E-01	Te-134	5.04E+00	Pr-143	1.12E-01

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TABLE 15.4-15

PARAMETERS USED TO EVALUATE OFFSITE DOSES
FOR THE LOSS OF COOLANT ACCIDENT

Parameter	Core Release Case	Realistic Case
Core Thermal Power	2958 MWt	2958 MWt
Fuel Damaged	100%	0%
Activity Available for Release from Containment	Table 15.4-11	Table 15-4-13
Form of Containment Iodine Release		
Particulate Iodine	95% as CsI	95% as CsI
Elemental Iodine	4.85%	4.85%
Organic Iodine	0.15%	0.15%
Activity Available for Release from Recirculation Loop Leakage	Table 15.4-11	Table 15-4-13
Activity Released to Environment from Recirculation Loop Leakage	10% of the Iodine Activity in Table 15.4-11	10% of the Iodine Activity in Table 15.4-13
Form of Recirculation Loop Iodine Release		
Elemental Iodine	97%	97%
Organic Iodine	3%	3%
Number of Spray Pumps Operating	1 of 2	1 of 2
Spray Removal Coefficients		
Elemental Iodine	20 hr^{-1}	20 hr^{-1}
Particulates	5.68 hr^{-1} (0-98% removal) 0.568 hr^{-1} (98-100% removal)	5.68 hr^{-1} (0-98% removal) 0.568 hr^{-1} (98-100% removal)
Effective Decontamination Factor for Elemental Iodine Spray Removal	200	200
Containment Free Volume	1.84E+06 ft^3	1.84E+06 ft^3
Containment Leak rate	0.2% per day (0-24hr) 0.1% per day (1-30 days)	0.2% per day (0-24hr) 0.1% per day (1-30 days)
Containment Recirculation Flow	54,200 cfm	54,200 cfm

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TABLE 15.4-15 (Continued)

PARAMETERS USED TO EVALUATE OFFSITE DOSES
FOR THE LOSS OF COOLANT ACCIDENT

The following parameters are common to both cases:

Offsite Atmospheric Dispersion Factors

Time Period	Site Boundary	Low Population Zone
0 – 2 hours	1.24E-04	-
0 – 8 hours		2.42E-05
2 – 8 hours		-
8 – 24 hours		1.68E-05
1 – 4 days		7.55E-06
4 – 30 days		2.40E-06

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Offsite Breathing Rate for EAB and LPZ (common to all accidents)

Time	Breathing Rate (m^3/sec)
0-8 hr	3.5E-04
8-24 hr	1.8E-04
1-30 days	2.3E-04

TABLE 15.4-16

OFFSITE DOSES FROM LOSS OF COOLANT ACCIDENT

	Doses in rem TEDE	
	Site Boundary [1] 1609 meters	Low Population Zone (0-30 days) 4827 meters
Core Release Case	1.5E+00	8.3E-01
Realistic Case	5.3E-04	9.1E-04
10 CFR 50.67 Limit	25	25

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(1) Core Release Case – Dose for time 0.4 – 2.4 hr (worst 2 hour time period)
Realistic Case – Dose for time 0 – 2 hr

TABLE 15.4-17

PARAMETERS USED IN ANALYSIS OF CONTROL ROOM DOSE FOLLOWING DESIGN BASIS ACCIDENTS

	Parameters
Control Room Free Volume	226,040 ft^3
Filtered Recirculation Flow	19,125 cfm [1]
Recirculation Filter Efficiencies	95% for all species of iodine
Maximum Control Room Filtered Air Infiltration Rate per Operating Train	1265 cfm
Control Room Unfiltered Air infiltration Rate	243 cfm including 10 cfm for ingress/egress
Maximum Control Room Outleakage	Equal to total inleakage (1508 cfm)
Meteorology	Table 2.3-123
Percent of Time Operator Is in Control Room Following Accident	0-24 hrs 100% 1-4 days 60% 4-30 days 40%
Duration of Accident	30 days
Breathing Rate of Operators in Control Room	3.47×10^{-4} m^3/sec
Activity Release Parameters	Table 15.4-15
Method of Dose Calculation	RADTRAD [64]

Time to Initiate Control Room Emergency Ventilation System [2,3] RN 12-034

Loss of Coolant Accident	0
Fuel Handling Accidents	30 minutes
Main Steam Line Break	30 minutes
Steam Generator Tube Rupture	30 minutes
Reactor Coolant Pump Locked Rotor	120 minutes
Control Rod Ejection	
Steam Generator Release Case	120 minutes
Containment Release Case	30 minutes

Notes:

1. 90% of 21,250 cfm.
2. The Control Room Ventilation system is automatically placed in the emergency mode, with filtration of incoming and recirculated air, following receipt of a Safety Injection or high radiation signal from the gaseous activity channel of RM-A1. If both trains are operating, one train will be isolated within 30 minutes by the Operator in accordance with the Emergency Operating Procedures to minimize dose consequences. This condition is bounded by the analysis which only credits one train control room ventilation for the duration of the accident.
3. Within the analysis, the emergency mode of operation is assumed to occur at the following times by either automatic isolation or manual initiation.

TABLE 15.4-18

CONTROL ROOM DOSES FOLLOWING A LOCA

	Doses in rem TEDE
Core Release Case	
Containment Leakage	7.0E-01
Containment Shine	2.4E-03
External Cloud from Containment Leakage	2.0E-02
Recirculation Loop Leakage	2.7E-01
External Cloud from Recirculation Loop-Leakage	1.4E-02
Total	1.0E+00
Realistic Case	
Containment Leakage	1.2E-04
Containment Shine	4.6E-08
External Cloud from Containment Leakage	3.9E-06
Recirculation Loop Leakage	7.5E-04
External Cloud from Recirculation Loop-Leakage	4.0E-05
Total	9.1E-04
10 CFR 50.67 Limit	5

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TABLE 15.4-19

TIME SEQUENCE OF EVENTS FOR
MAJOR SECONDARY SYSTEM PIPE RUPTURES

Accident	Event	Time (sec)	
Major Steam Line Rupture			
Offsite Power available	Steam line ruptures	0	
	Criticality attained	25	RN 01-125
	Boron from RWST reaches core	57	
	Accumulators actuate	74	
	Peak heat flux attained	76	
	Core becomes subcritical	~ 108	
Without offsite power	Steam line ruptures	0	
	Criticality attained	22	02-01
	Boron from RWST reaches core	72	
	Peak heat flux attained	~ 261	
	Core becomes subcritical	~ 313	
Rupture of Main Feedwater Pipe			
(Offsite Power Available)	Feedline rupture occurs	10	
	Low-low SG water level setpoint reached	32.5	RN 01-125
	Rods begin to drop	34.5	
	Low steamline pressure setpoint simulated	44	
	Steamline and feedline isolation occurs	54	
	Emergency Feedwater is started	122	RN 01-125
	Feedwater lines are purged and emergency feedwater is delivered to two intact SGs	134	
	First steam generator safety valve lifts in intact loop	365	
	Pressurizer water relief begins	399	
	Total RCS heat generation (decay heat + pump heat) decreases to emergency feedwater heat removal capability	~ 2300	

TABLE 15.4-19 (Continued)

TIME SEQUENCE OF EVENTS FOR
MAJOR SECONDARY SYSTEM PIPE RUPTURES

Accident	Event	Time (sec)	
Rupture of Main Feedwater Pipe (Continued)			
(Offsite Power Unavailable)	Feedline rupture occurs	10	
	Low-low SG water level setpoint reached	32.5	RN 01-125
	Rods begin to drop	34.5	
	Reactor coolant pump coastdown	36.5	
	Low steamline pressure setpoint simulated	44	
	Steamline and feedline isolation occurs	54	
	Emergency Feedwater is started	122	
	Feedwater lines are purged and emergency feedwater is delivered to two intact SGs	134	RN 01-125
	Pressurizer water relief begins	415	
	First steam generator safety valve lifts in intact loop	467	
	Total RCS heat generation (decay heat + pump heat) decreases to emergency feedwater heat removal capability	~ 1000	

TABLE 15.4-21

EQUIPMENT REQUIRED FOLLOWING A HIGH ENERGY LINE BREAK

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage, underfrequency, and turbine trip may be excluded)	Emergency Feedwater System including pumps, water supply, and system valves and piping (Emergency Feedwater System automatically supplies water, see section 10.4.9).	Steam generator power operated relief valves (can be manually operated locally).
Safety Injection System including the pumps, the refueling water storage tank, and the systems valves and piping	Reactor Building emergency cooling units.	Controls for defeating automatic safety injection actuation during a cooldown and depressurization. 02-01
Diesel generators and emergency power distribution equipment	Capability for obtaining a Reactor Coolant System sample	Residual Heat Removal System including pumps, heat exchanger, and system valves and piping necessary to cool and maintain the Reactor Coolant System in a cold shutdown condition.
Service Water System		
Reactor Building emergency cooling units.		
Emergency Feedwater System including pumps, water supplies, piping, valves		
Main feedwater control valves[(1)] (trip closed feature)		
Primary and secondary safety valves		

TABLE 15.4-21 (Continued)

EQUIPMENT REQUIRED FOLLOWING A HIGH ENERGY LINE BREAK

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT)	HOT STANDBY	REQUIRED FOR COOLDOWN
Circuits and/or equipment required to trip the main feedwater pumps(1)		
Main feedwater isolation valves(1) (trip closed feature)		
Main steam line isolation valves(1) (trip closed feature)		
Main steam line isolation valve bypass valves(1) (trip closed feature)		
Steam generator blowdown isolation valves (automatic closure feature)		
Batteries (Class IE)		
Control room ventilation		
Control room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.		
Emergency lighting		
Post Accident Monitoring System(2)		

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TABLE 15.4-21 (Continued)

EQUIPMENT REQUIRED FOLLOWING A HIGH ENERGY LINE BREAK

SHORT TERM (REQUIRED FOR MITIGATION OF ACCIDENT	HOT STANDBY	REQUIRED FOR COOLDOWN
Wide range T_{hot} or T_{cold} (preferably T_{hot}) for each reactor coolant loop		
Pressurizer water level		
Wide range Reactor Coolant System pressure		
Steam line pressure for each steam generator		
Wide range or narrow range steam generator level for each steam generator		
Containment pressure		

(1) Require for steam line, feed line, and steam generator blowdown line break only.

(2) See Section 7.5 for a discussion of the Post Accident Monitoring System.

TABLE 15.4-23

PARAMETERS USED IN STEAM LINE BREAK ANALYSIS

Parameter	Conservative Case	Realistic Case
Core thermal power	2958 MWt	2958 MWt
Fuel defects	1%	0.12%
Fuel damaged	0%	0%
Steam generator tube leak	1.0 gpm	100 lb/day
Iodine spiking basis	Pre-accident/concurrent	None
Noble gas partition factor faulted and intact steam generator	1	1
Iodine and particulate partition factor faulted steam generator	1	1
Iodine and particulate partition factor intact steam generator	100	100
Form of iodine release		
Particulate iodine	97%	97%
Organic iodine	3%	3%

The following parameters are common to both cases: RN 12-034

Integrated Mass of Steam and Reactor Coolant Release Rates

Time Period	Integrated Steam Release from Faulted SG (lbm)	Integrated Steam Release from Intact SGs (lbm)	Reactor Coolant Release to Faulted SG (gpm) (1)	Reactor Coolant Release to Intact SGs (gpm)
0 – 30 min	406,000			
0 – 2 hours		343,700	0.35	0.65
2 – 8 hours		733.900	0.35	0.65
8 – 24 hours		1,200,000	0.35	0.65

(1) For the realistic case the 100 lb/day reactor coolant release is assumed to be released to the faulted steam generator.

Offsite Atmospheric Dispersion Factors

Time	Site Boundary	Low Population Zone
0 – 2 hours	1.24E-04	-
0 – 8 hours		2.42E-05
8 – 24 hours		1.68E-05

TABLE 15.4-24a

SECONDARY SYSTEM EQUILIBRIUM CONCENTRATION FOR PRE-ACCIDENT IODINE SPIKE

SECONDARY SYSTEM ACTIVITY (μCi/lb)

Isotope	1 μCi/gm DE I-131	60 μCi/gm DE I-131
I-131	1.27E+01	7.62E+02
I-132	1.61E+00	9.66E+01
I-133	1.13E+01	6.80E+02
I-134	1.26E-01	7.55E+00
I-135	2.97E+00	1.78E+02
Cs-134	2.04E+01	2.04E+01
Cs-136	1.97E+01	1.97E+01
Cs-137	9.74E+00	9.74E+00
Cs-138	1.27E-01	1.27E101
Rb-88	2.78E-01	2.78E-01
Br-83	4.73E-02	2.84E+00
Br-84	5.43E-03	3.26E-01

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TABLE 15.4-24b

POST ACCIDENT IODINE RELEASE RATE
CONCURRENT IODINE SPIKE

Isotope	Iodine Release Rate (Ci/Sec)	500 x Iodine Release Rate (Ci/Sec)
I-131	7.21×10^{-3}	3.61
I-132	3.33×10^{-2}	16.64
I-133	1.39×10^{-2}	6.93
I-134	1.53×10^{-2}	7.66
I-135	1.29×10^{-2}	6.46

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TABLE 15.4-25

STEAM LINE BREAK ISOTOPIC RELEASE TO ENVIRONMENT REALISTIC ANALYSIS

	Activity Released to Environment (Ci)	
Isotope	(0 - 2 hr)	(0 - 24 hr)
Kr-85m	8.16E-04	9.83E-03
Kr-85	3.45E-03	4.15E-02
Kr-87	4.99E-04	6.01E-03
Kr-88	1.45E-03	1.75E-02
I-131	7.81E-03	2.29E-02
I-132	3.42E-03	1.90E-02
I-133	1.01E-02	3.33E-02
I-134	4.51E-04	3.46E-03
I-135	3.98E-03	1.61E-02
Xe-131m	1.04E-03	1.26E-02
Xe-133m	8.61E-03	1.04E-01
Xe-133	1.31E-01	1.58E+00
Xe-135m	2.36E-04	2.84E-03
Xe-135	3.90E-03	4.70E-02
Xe-138	2.90E-04	3.50E-03
Cs-134	1.17E-02	3.39E-02
Cs-136	1.18E-02	3.45E-02
Cs-137	5.60E-03	1.62E-02
Cs-138	6.27E-04	5.49E-03
Rb-88	2.14E-03	2.12E-02
Br-83	9.91E-05	5.46E-04
Br-84	2.71E-05	2.38E-04

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TABLE 15.4-26

STEAM LINE BREAK ISOTOPIC RELEASE TO ENVIRONMENT
CONSERVATIVE ANALYSIS

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Isotope	Pre-Accident Iodine Spike (Ci) (0 - 2 hr)	Pre-Accident Iodine Spike (Ci) (0 - 24 hr)
Kr-85m	2.13E-01	2.55E+00
Kr-85	8.98E-01	1.08E+01
Kr-87	1.30E-01	1.56E+00
Kr-88	3.78E-01	4.54E+00
I-131	3.19E+02	4.11E+02
I-132	4.74E+01	1.37E+02
I-133	2.90E+02	4.26E+02
I-134	4.61E+00	2.18E+01
I-135	7.91E+01	1.49E+02
Xe-131m	2.72E-01	3.26E+00
Xe-133m	2.25E+00	2.69E+01
Xe-133	3.43E+01	4.11E+02
Xe-135m	6.14E-02	7.37E-01
Xe-135	1.02E+00	1.22E+01
Xe-138	7.56E-02	9.07E-01
Cs-134	8.53E+00	1.08E+01
Cs-136	8.26E+00	1.05E+01
Cs-137	4.07E+00	5.15E+00
Cs-138	9.28E-02	5.56E-01
Rb-88	2.74E-01	2.09E+00
Br-83	1.39E+00	3.95E+00
Br-84	2.40E-01	1.44E+00

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TABLE 15.4-26 (Continued) RN 12-034

STEAM LINE BREAK ISOTOPIC RELEASE TO ENVIRONMENT
CONSERVATIVE ANALYSIS

	Concurrent Iodine Spike (Ci)		
Isotope	(0 - 2 hr)	(0 - 24 hr)	02-01
Kr-85m	2.13E-01	2.55E+00	
Kr-85	8.98E-01	1.08E+01	
Kr-87	1.30E-01	1.56E+00	
Kr-88	3.78E-01	4.54E+00	
I-131	2.72E+01	6.45E+02	
I-132	4.82E+01	1.68E+03	
I-133	2.47E+01	1.19E+03	
I-134	1.53E+01	3.46E+02	
I-135	1.97E+01	9.15E+02	
Xe-131m	2.72E-01	3.26E+00	
Xe-133m	2.25E+00	2.69E+01	RN 12-034
Xe-133	3.43E+01	4.11E+02	
Xe-135m	6.14E-02	7.37E-01	
Xe-135	1.02E+00	1.22E+01	
Xe-138	7.56E-02	9.07E-01	
Cs-134	2.60E+01	2.86E+01	
Cs-136	2.52E+01	2.77E+01	
Cs-137	1.24E+01	1.36E+01	
Cs-138	2.02E-01	6.67E-01	
Rb-88	5.13E-01	2.33E+00	
Br-83	6.36E-02	1.07E-01	
Br-84	8.65E-03	2.88E-02	

TABLE 15.4-27

OFFSITE AND CONTROL ROOM DOSES
MAIN STEAM LINE BREAK ACCIDENT

	Dose (rem TEDE)		
Case	EAB	LPZ	CHRE
Pre-existing Iodine Spike Design CR Leakage	0.60	0.14	1.15
Concurrent Iodine Spike Design CR Leakage	0.24	0.20	0.37
Realistic Design CR Leakage	0.000053	0.000024	0.00012

RN 12-034

The dose acceptance criteria provided in 10 CFR 50.67 and Regulatory Guide 1.183 are given as follows:

Pre-existing Iodine Spike MSLB – 25 TEDE for the EAB and LPZ and 5 Rem TEDE for the CR.

Concurrent Iodine Spike MSLB – 2.5 TEDE for the EAB and LPZ and 5 Rem TEDE for the CR.

TABLE 15.4-28

BALANCE OF PLANT EQUIPMENT REQUIRED FOR RECOVERY FROM STEAM GENERATOR TUBE RUPTURE ACCIDENT 02-01

1. Emergency Feedwater System including pumps, water supply, and system valves and piping.
2. Steam generator shell side fluid sampling system.
3. Steam generator safety valves
4. Steam generator power operated relief valves (can be manually operated locally).
5. Refueling Water Storage Tank.
6. Main steam line isolation and bypass valves
7. Diesel generators and emergency power distribution equipment.
8. Residual Heat Removal System

Note: Range of acceptable values are discussed in the recovery procedure.

TABLE 15.4-29

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE ANALYSES

	Realistic Analysis	Conservative Analysis	
Core thermal power	2958 MWt	2958 MWt	
Steam generator tube leak rate prior to and during accident	100 lbs/day [(1)]	1.0 gpm	02-01
Offsite power	Available	Lost	
Fuel defects	0.12% [(1)]	1% for Noble Gases	02-01
Iodine Spiking Basis	None	Pre-accident / Concurrent	
Failed Fuel	0.0	0.0	
Initial RC Iodine Activities	At 0.12% failed fuel levels	60 μCi/gm and 1 μCi/gm Dose Equivalent I-131 for Pre-accident / Concurrent Spike	99-01
Iodine partition factors in steam generators	100	100	RN 12-034
Time to isolate defective steam generator	30 min	30 min	
Duration of plant cooldown by secondary system after accident	24 hr	24 hr	RN 12-034
Steam release from defective steam generator	56,800 lbs (0 - 30 min)	56,800 lbs (0 - 30 min)	
Steam release from 2 unaffected steam generators	381,400 lbs (0 - 2 hr) 924,900 lbs (2 - 8 hr) 1,200,000 lbs (8 – 24 hr)	381,400 lbs (0 - 2 hr) 924,900 lbs (2 - 8 hr) 1,200,000 lbs (8 – 24 hr)	RN 12-034
Reactor coolant released to defective steam generator	92,900 lbs	92,900 lbs	

Meteorology

The following parameters are common to both cases: RN 12-034

Offsite Atmospheric Dispersion Factors

Time Period	Site Boundary	Low Population Zone
0 – 2 hours	1.24E-04	-
0 – 8 hours		2.42E-05
8 – 24 hours		1.68E-05

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(1) American National Standards Institute, "Source Term Specification," ANS/ANSI 18.1 - 1984.

TABLE 15.4-30

STEAM GENERATOR TUBE RUPTURE ISOTOPIC RELEASE TO ENVIRONMENT REALISTIC ANALYSIS

Isotope	Activity Released to Environment (Ci) (0 - 2 hr)	Activity Released to Environment (Ci) (0 - 24 hr)
Kr-85m	9.11E+00	9.12E+00
Kr-85	3.85E+01	3.85E+01
Kr-87	5.57E+00	5.57E+00
Kr-88	1.62E+01	1.62E+01
I-131	2.26E+00	2.26E+00
I-132	2.34E+00	2.34E+00
I-133	3.47E+00	3.47E+00
I-134	4.53E-01	4.53E-01
I-135	1.81E+00	1.81E+00
Xe-131m	1.16E+01	1.17E+01
Xe-133m	9.62E+01	9.63E+01
Xe-133	1.47E+03	1.47E+03
Xe-135m	2.63E+00	2.63E+00
Xe-135	4.35E+01	4.36E+01
Xe-138	3.24E+00	3.24E+00
Cs-134	3.32E+00	3.32E+00
Cs-136	3.40E+00	3.40E+00
Cs-137	1.59E+00	1.59E+00
Cs-138	7.32E-01	7.32E-01
Rb-88	2.87E+00	2.87E+00
Br-83	6.72E-02	6.72E-02
Br-84	3.17E-02	3.17E-02

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TABLE 15.4-31

STEAM GENERATOR TUBE RUPTURE ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS

	Pre-Accident Iodine Spike (Ci)	
Isotope	(0 - 2 hr)	(0 - 24 hr)
Kr-85m	2.00E+01	2.23E+01
Kr-85	8.43E+01	9.42E+01
Kr-87	1.22E+01	1.36E+01
Kr-88	3.55E+01	3.97E+01
I-131	2.98E+02	3.11E+02
I-132	3.05E+02	3.12E+02
I-133	4.56E+02	4.71E+02
I-134	5.91E+01	6.03E+01
I-135	2.37E+02	2.43E+02
Xe-131m	2.55E+01	2.85E+01
Xe-133m	2.11E+02	2.36E+02
Xe-133	3.22E+03	3.60E+03
Xe-135m	5.77E+00	6.45E+00
Xe-135	9.54E+01	1.07E+02
Xe-138	7.10E+00	7.93E+00
Cs-134	7.30E+00	7.64E+00
Cs-136	7.46E+00	7.79E+00
Cs-137	3.49E+00	3.64E+00
Cs-138	1.59E+00	1.62E+00
Rb-88	6.23E+00	6.36E+00
Br-83	8.77E+00	8.97E+00
Br-84	4.13E+00	4.22E+00

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TABLE 15.4-32

STEAM GENERATOR TUBE RUPTURE ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS

02-01

Isotope	Concurrent Iodine Spike (Ci) (0 - 2 hr)	Concurrent Iodine Spike (Ci) (0 - 24 hr)
Kr-85m	2.00E+01	2.23E+01
Kr-85	8.43E+01	9.42E+01
Kr-87	1.22E+01	1.36E+01
Kr-88	3.55E+01	3.97E+01
I-131	7.54E+01	1.02E+02
I-132	3.67E+02	4.37E+02
I-133	1.53E+02	2.03E+02
I-134	1.41E+02	1.55E+02
I-135	1.37E+02	1.76E+02
Xe-131m	2.55E+01	2.85E+01
Xe-133m	2.11E+02	2.36E+02
Xe-133	3.22E+03	3.60E+03
Xe-135m	5.77E+00	6.45E+00
Xe-135	9.54E+01	1.07E+02
Xe-138	7.10E+00	7.93E+00
Cs-134	7.49E+00	8.22E+00
Cs-136	7.65E+00	8.36E+00
Cs-137	3.58E+00	3.92E+00
Cs-138	1.59E+00	1.63E+00
Rb-88	6.23E+00	6.37E+00
Br-83	1.47E-01	1.51E-01
Br-84	6.89E-02	7.05E-02

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TABLE 15.4-33

OFFSITE AND CONTROL ROOM DOSES
STEAM GENERATOR TUBE RUPTURE ACCIDENT

Case	Dose (Rem TEDE) EAB	LPZ	CHRE
Pre-existing Iodine Spike Design CR Leakage	0.63	0.13	1.18
Concurrent Iodine Spike Design CR Leakage	0.22	0.05	0.37
Realistic Design CR Leakage	0.017	0.0034	0.03

RN 12-034

The dose acceptance criteria provided in 10 CFR 50.67 and Regulatory Guide 1.183 are given as follows:

Pre-existing Iodine Spike MSLB – 25 TEDE for the EAB and LPZ and 5 Rem TEDE for the CR.

Concurrent Iodine Spike MSLB – 2.5 TEDE for the EAB and LPZ and 5 Rem TEDE for the CR.

TABLE 15.4-34

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENT

	Three Loops Operating Initially 1 Locked Rotor	
Maximum Reactor Coolant System pressure (psia)	< 2710[1]	RN 09-017
Maximum clad average temperature at core hot spot (°F)	2013	RN 01-125
Amount of $Zr\text{-}H_20$ reaction at core hot spot (% by weight)	0.7	

TABLE 15.4-34a

PARAMETERS USED IN REACTOR COOLANT PUMP LOCKED ROTOR ACCIDENT ANALYSIS

RN 12-034

	Realistic Analysis	Conservative Analysis	
Core thermal power	2958 MWt	2958 MWt	
Steam Generator Tube Leak Rate Prior to Accident and for First Twenty Four Hours Following Accident	100 lbs/day [(1)]	1.0 gpm	RN 12-034
Offsite power	Lost	Lost	
Fuel defects	0.12 percent	1 percent	
Failed Fuel	0.0	15 percent	
Activity Released to Reactor Coolant from Failed Fuel	0.0	15 percent of gap inventory	
Iodine and Alkali Metals Partition Factor for Steam Generators	100	100	RN 12-034
Duration of Plant Cooldown by Secondary System After Accident	24 hours	24 hours	
Steam Release from Three Steam Generators	447,900 lbs (0 - 2 hr) 868,300 lbs (2 - 8 hr) 1,200,000 lbs (8 – 24 hr)	447,900 lbs (0 - 2 hr) 868,300 lbs (2 - 8 hr) 1,200,000 lbs (8 – 24 hr)	
Meteorology			

The following parameters are common to both cases:

RN 12-034

Offsite Atmospheric Dispersion Factors

Time Period	Site Boundary	Low Population Zone
0 – 2 hours	1.24E-04	-
0 – 8 hours		2.42E-05
8 – 24 hours		1.68E-05

(1) American National Standards Institute, "Source Term Specification," ANS/ANSI 18.1 - 1984. 02-01

TABLE 15.4-34b

REACTOR COOLANT PUMP LOCKED ROTOR
ISOTOPIC RELEASE TO ENVIRONMENT REALISTIC ANALYSIS

Isotope	Activity Released to Environment (Ci) (0 - 2 hr)	Activity Released to Environment (Ci) (0 - 24 hr)
Kr-85	3.45E-03	4.14E-02
Kr-85m	8.17E-04	9.81E-03
Kr-87	4.99E-04	5.99E-03
Kr-88	1.45E-03	1.74E-02
Rb-86	1.02E-06	6.80E-06
I-131	8.42E-05	5.60E-04
I-132	3.61E-05	2.93E-04
I-133	1.09E-04	7.44E-04
I-134	4.68E-06	4.37E-05
I-135	4.26E-05	3.09E-04
Xe-133	1.32E-01	1.58E+00
Xe-135	3.90E-03	4.69E-02
Cs-134	1.26E-04	8.38E-04
Cs-136	1.27E-04	8.46E-04
Cs-137	6.04E-05	4.00E-04

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TABLE 15.4-34c

REACTOR COOLANT PUMP LOCKED ROTOR ACCIDENT
ISOTOPIC RELEASE TO ENVIRONMENT CONSERVATIVE ANALYSIS

Isotope	Activity Released to Environment (Ci) (0 - 2 hr)	(0 - 24 hr)
Kr-85	1.09E+02	1.31E+03
Kr-85m	1.74E+03	2.08E+04
Kr-87	3.17E+03	3.80E+04
Kr-88	4.28E+03	5.14E+04
Rb-86	1.38E-01	1.64E+00
I-131	1.67E+02	2.00E+03
I-132	1.52E+02	1.83E+03
I-133	2.14E+02	2.56E+03
I-134	2.29E+02	2.74E+03
I-135	1.96E+02	2.35E+03
Xe-133	1.10E+04	1.32E+05
Xe-135	2.37E+03	2.84E+04
Cs-134	3.12E+01	3.72E+02
Cs-136	9.77E+00	1.15E+02
Cs-137	1.74E+01	2.08E+02

RN
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TABLE 15.4-34d

OFFSITE AND CONTROL ROOM DOSES
REACTOR COOLANT PUMP LOCKED ROTOR ACCIDENT

	Dose (Rem TEDE)		
Case	EAB	LPZ	CHRE
Regulatory Guide 1.183 Case	0.67	0.66	2.43
Realistic Case	0.0000008	0.0000007	0.0000032

RN 12-034

The dose acceptance criteria provided in 10 CFR 50.67 and Regulatory Guide 1.183 are given as follows:

2.5 Rem TEDE for the EAB and LPZ.

5 Rem TEDE for the CR.

TABLE 15.4-35

NUCLEAR CHARACTERISTICS OF HIGHEST RATED DISCHARGED ASSEMBLY

Core Power, MWt	2958
Number of Assemblies	157
Core Average Assembly Power at 102% of Full power, MWt	19.22
Highest Power Discharged Assembly	
Axial Peak to Average Ratio	1.70
Radial Peak to Average Ratio	1.70

TABLE 15.4-36

REALISTIC CASE
ACTIVITIES IN HIGHEST RATED ASSEMBLY AT 100 HOURS AFTER REACTOR SHUTDOWN

Isotope	Percent of Activity in Gap	Curies in Gap (1)
I-131	0.16	1.38E+05
I-132	0.10	2.01E+01
I-133	0.10	9.01E+03
I-134	0.10	0
I-135	0.10	6.26E+00
Kr-85	0.20	3.37E+03
Kr-85m	0.10	6.05E-03
Kr-87	0.10	0
Kr-88	0.10	2.19E-06
Xe-131m	0.10	1.71E+03
Xe-133	0.10	1.69E+05
Xe-133m	0.10	3.16E+03
Xe-135	0.10	3.23E+02
Xe-135m	0.10	1.07E+00

02-01

RN 12-034

(1) Gap activity in 314 fuel rods (1.19 assemblies).

TABLE 15.4-37

CONSERVATIVE CASE ACTIVITIES IN HIGHEST RATED ASSEMBLY AT 72 HOURS AFTER REACTOR SHUTDOWN

RN 03-017

Isotope	Percent of Activity in Gap	Curies in Gap (1)
I-131	0.16	1.52E+05
I-132	0.10	9.30E+04
I-133	0.10	2.29E+04
I-134	0.10	0.00E+00
I-135	0.10	1.18E+02
Kr-85	0.20	3.37E+03
Kr-85m	0.10	4.61E-01
Kr-87	0.10	0.00E+00
Kr-88	0.10	2.03E-03
Xe-131m	0.10	1.76E+03
Xe-133	0.10	1.95E+05
Xe-133m	0.10	4.39E+03
Xe-135	0.10	2.55E+03
Xe-135m	0.10	1.92E+01

RN 12-034

(1) Gap activity in 314 fuel rods (1.19 fuel assemblies).

TABLE 15.4-38

PARAMETERS USED TO EVALUATE THE CONSEQUENCES OF A FUEL HANDLING ACCIDENT

Parameter	Realistic Analysis	Conservative Regulatory Guide 1.183 Analysis	
			RN 12-034
Time between plant shutdown and accident	100 hours	72 hours	RN 03-017
Maximum fuel rod pressurization	≤ 1200 psig	≤ 1200 psig	
Minimum water depth between top of damaged fuel rods and spent fuel pool surface	≥ 23 feet	≥ 23 feet	
Damaged to fuel assembly	All rods ruptured in 1.19 assemblies	All rods ruptured in 1.19 assemblies	
Fuel assembly activity	Highest powered fuel assembly in core region discharged	Highest powered fuel assembly in core region discharged	
Activity released to fuel pool	Gap activity in ruptured rods	Gap activity in ruptured rods	
Form of iodine activity released from fuel pool			
Elemental Iodine	57%	57%	
Organic Iodine	43%	43%	
Decontamination factor (DF) in fuel pool			
Iodine	500	200	
Noble gases	1	1	RN 12-034
Filter efficiencies in Fuel Handling and Reactor Building			
Elemental Iodine	0	0	
Methyl Iodine	0	0	
Meteorology			

The following parameters are common to both cases:

Offsite Atmospheric Dispersion Factors

Time Period	Site Boundary	Low Population Zone
0 – 2 hours	1.24E-04	5.06E-05

TABLE 15.4-40

ACTIVITY RELEASES TO THE ENVIRONMENT
FROM A FUEL HANDLING ACCIDENT

Isotope	Activity Released Conservative (Regulatory Guide 1.183) Case (Curies)	Activity Released Realistic Case Curies
I-131	7.62E+02	2.76E+02
I-132	4.65E+02	4.02E-02
I-133	1.14E+02	1.80E+01
I-134	0	0
I-135	5.90E-01	1.25E-02
Kr-85	3.37E+03	3.37E+03
Kr-85m	4.61E-01	6.05E-03
Kr-87	0	0
Kr-88	2.03E-03	2.19E-06
Xe-131m	1.76E+03	1.71E+03
Xe-133	1.95E+05	1.69E+05
Xe-133m	4.39E+03	3.16E+03
Xe-135	2.55E+03	3.23E+02
Xe-135m	1.92E+01	1.07E+00

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TABLE 15.4-41

OFFSITE AND CONTROL ROOM DOSES
FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

	Dose (Rem TEDE)		
Case	EAB	LPZ	CHRE
Regulatory Guide 1.183 Case	1.30	0.53	0.76
Realistic Case	0.53	0.21	0.31

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The dose acceptance criteria provided in 10 CFR 50.67 and Regulatory Guide 1.183 are given as follows:

6.3 Rem TEDE for the EAB and LPZ.

5 Rem TEDE for the CR.

TABLE 15.4-42

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	Beginning	Beginning	End	End	
Power level, %	102	0.0	102	0.0	
Ejected rod worth, %Δk	0.20	0.855	0.21	0.90	
Delayed neutron fraction, %	0.54	0.54	0.44	0.44	
Feedback reactivity weighting	1.30	2.07	1.30	3.55	
Trip reactivity, %Δk	4	2	4	2	
FQ before rod ejection	2.635	-	2.635	-	
FQ after rod ejection	6.0	13	6.5	22.5	
Number of operating pumps	3	3	3	3	
Maximum fuel pellet average temperature, °F	4195	4015	4067	3550	
Maximum fuel center temperature, °F	*	4825	*	4198	99-01
Maximum clad average temperature, °F	2520	2580	2418	2415	
Maximum fuel stored energy, cal/gm	185	175	178	151	02-01

* Less than 10% fuel melt.

TABLE 15.4-43

PARAMETERS USED IN ROD EJECTION ACCIDENT ANALYSES

Parameter	Conservative Case Containment Release Path	Conservative Case Steam Generator Release Path	Realistic Case
Core Thermal Power	2958 MWt	2958 MWt	2958 MWt
Fuel Defects	1%	1%	0.12%
Fuel Damaged			0
Activity Available for Release from Containment	Table 15.4-45	Table 15.4-46	Table 15.4-44
Form of Iodine Release			
Particulate Iodine	95% as CsI	0%	95% as CsI
Elemental Iodine	4.85%	97%	4.85%
Organic Iodine	0.15%	3%	0.15%
Containment Leak Rate	0.2% per day (0-24 hours) 0.1% per day (1-30 days)	N/A	0.2% per day (0-24 hr) 0.1% per day (1-30 days)
SG Release Rate	N/A	447,900 lbs (0-2 hours) 868,300 lbs (2-8 hours) 1,200,000 lbs (8-24 hours)	N/A

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The following parameters are common to the above cases:

Offsite Atmospheric Dispersion Factors

Time Period	Site Boundary	Low Population Zone
0 – 2 hours	1.24E-04	-
0 – 8 hours		2.42E-05
2 – 8 hours		-
8 – 24 hours		1.68E-05
1 – 4 days		7.55E-06
4 – 30 days		2.40E-06

TABLE 15.4-44

CONTROL ROD EJECTION ACCIDENT
ISOTOPIC RELEASE TO CONTAINMENT REALISTIC ANALYSIS RN 06-022

Isotope	Activity Released to Containment (Ci)
Kr-85	1.66E+02
Kr-85m	3.92E+01
Kr-87	2.40E+01
Kr-88	6.97E+01
Rb-86	7.85E-01
I-131	6.54E+01
I-132	6.76E+01
I-133	1.00E+02
I-134	1.31E+01
I-135	5.23E+01
Xe-133	6.32E+03
Xe-135	1.87E+02
Cs-134	9.59E+01
Cs-136	9.81E+01
Cs-137	4.58E+01

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TABLE 15.4-45

CONTROL ROD EJECTION ACCIDENT
ISOTOPIC RELEASE TO CONTAINMENT CONSERVATIVE ANALYSIS

RN 06-022

Isotope	Activity Released to Containment (Ci)
Kr-85	3.31E+04
Kr-85m	1.04E+06
Kr-87	1.90E+06
Kr-88	2.57E+06
Rb-86	1.87E+03
I-131	2.88E+06
I-132	4.21E+06
I-133	5.89E+06
I-134	6.31E+06
I-135	5.40E+06
Xe-133	6.56E+06
Xe-135	1.42E+06
Cs-134	4.26E+05
Cs-136	1.30E+05
Cs-137	2.39E+05

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TABLE 15.4-46

CONTROL ROD EJECTION ACCIDENT ISOTOPIC RELEASE TO STEAM GENERATOR CONSERVATIVE ANALYSIS

	Activity Released to Environment (Ci)	
Isotope	(0 - 2 hr)	(0 - 24 hr)
Kr-85	8.29E+01	9.95E+02
Kr-85m	2.60E+03	3.13E+04
Kr-87	4.75E+03	5.70E+04
Kr-88	6.42E+03	7.71E+04
Rb-86	9.78E-02	1.16E+00
I-131	1.48E+02	1.77E+03
I-132	2.16E+02	2.59E+03
I-133	3.02E+02	3.63E+03
I-134	3.24E+02	3.89E+03
I-135	2.77E+02	3.33E+03
Xe-133	1.64E+04	1.97E+05
Xe-135	3.55E+03	4.25E+04
Cs-134	2.20E+01	2.62E+02
Cs-136	6.96E+00	8.14E+01
Cs-137	1.23E+01	1.46E+02

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TABLE 15.4-47

OFFSITE AND CONTROL ROOM DOSES
CONTROL ROD EJECTION ACCIDENT

Case	Dose (Rem TEDE) EAB	LPZ	CHRE
Regulatory Guide 1.183 Containment Release	1.31	1.46	1.71
Regulatory Guide 1.183 SG PORV Release	0.77	0.68	2.38
Realistic Containment Release	0.000072	0.000091	0.000108

RN
12-034

The dose acceptance criteria provided in 10 CFR 50.67 and Regulatory Guide 1.183 are given as follows:

6.3 Rem TEDE for the EAB and LPZ.

5 Rem TEDE for the CR.

TABLE 15.4-50

OFFSITE AND CONTROL ROOM DOSES
FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT

	Dose (Rem TEDE)		
Case	EAB	LPZ	CHRE
Regulatory Guide 1.183 Case	1.30	0.53	0.41
Realistic Case	0.53	0.21	0.17

RN 12-034

The dose acceptance criteria provided in 10 CFR 50.67 and Regulatory Guide 1.183 are given as follows:

6.3 Rem TEDE for the EAB and LPZ.

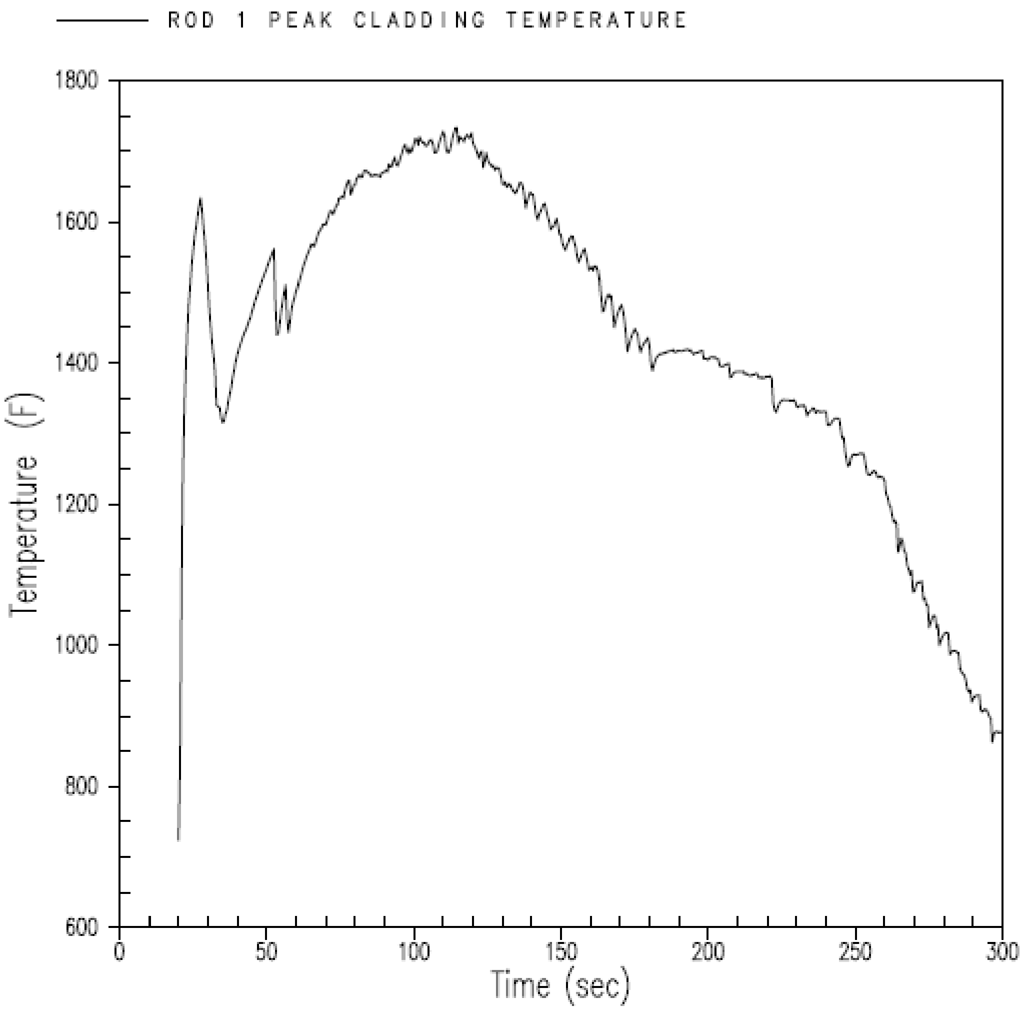
5 Rem TEDE for the CR.

BLOWDOWN	0 second	BREAK OCCURS
		REACTOR TRIP
		PUMPED SI SIGNAL (PRESSURIZER PRESSURE)
		ACCUMULATOR INJECTION BEGINS
		PUMPED ECCS INJECTION BEGINS (ASSUMING OFFSITE POWER AVAILABLE)
	20-30 seconds	CONTAINMENT HEAT REMOVAL SYSTEM STARTS (ASSUMING OFFSITE POWER AVAILABLE)
		END OF BYPASS
REFILL	30-40 seconds	END OF BLOWDOWN
		BOTTOM OF CORE RECOVERY
REFLOOD	10 minutes	ACCUMULATOR EMPTY
		CORE QUENCHED
LONG TERM COOLING	24 hours	SWITCH TO COLD LEG RECIRCULATION ON RWST LOW LEVEL ALARM
		SWITCH TO HOT LEG/COLD LEG RECIRCULATION

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Typical Time Sequence of Events for the Virgil C. Summer Nuclear Station BELOCA Analysis

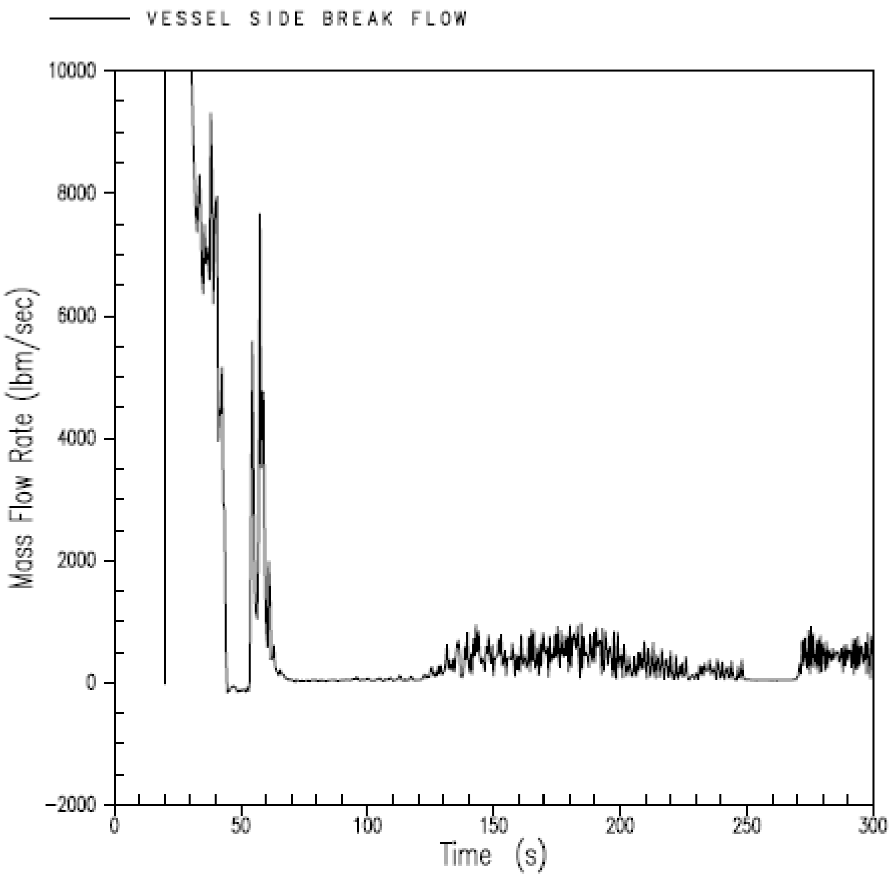
Figure 15.4-1a



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Peak Cladding Temperature for Reference Transient

Figure 15.4-1b



SOUTH CAROLINA ELECTRIC & GAS CO.
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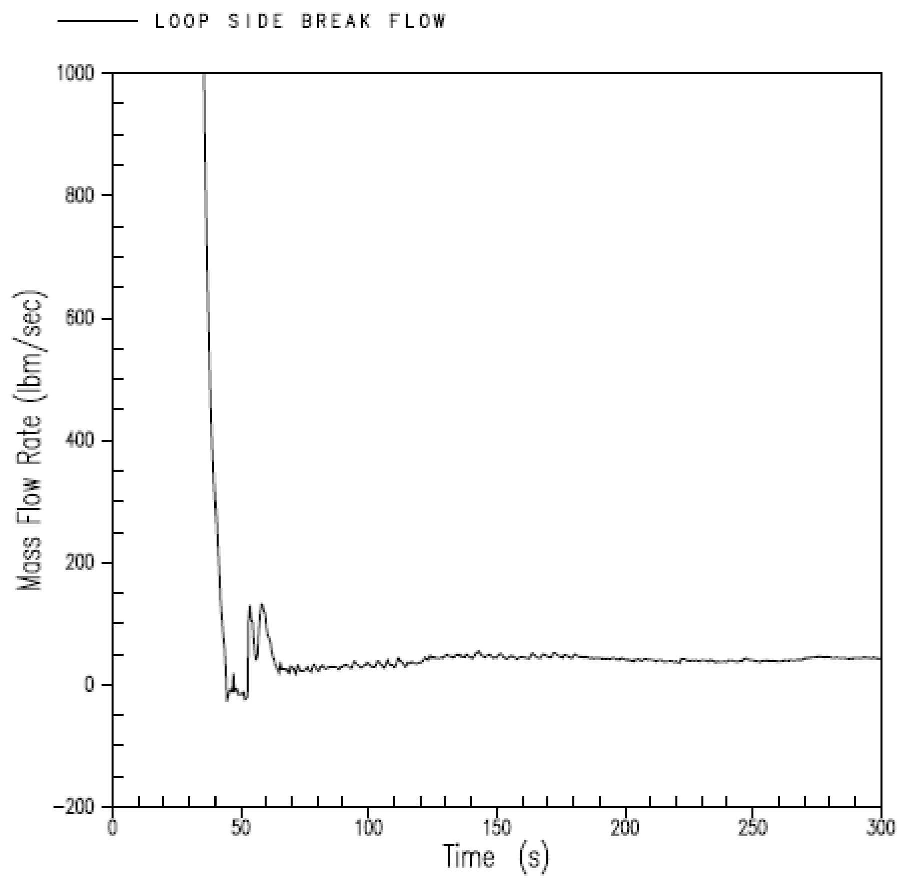
Break Flow on Vessel Side of Broken Cold Leg for Reference Transient

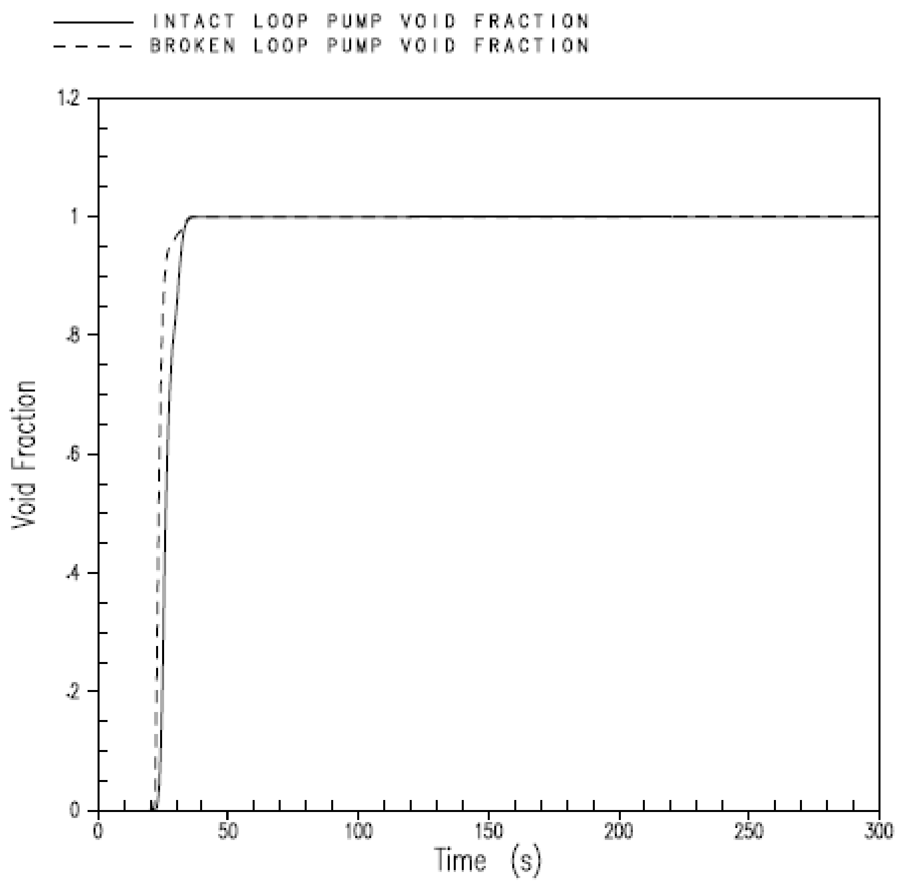
Figure 15.4-1c



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Break Flow on Loop Side of Broken Cold Leg for Reference Transient

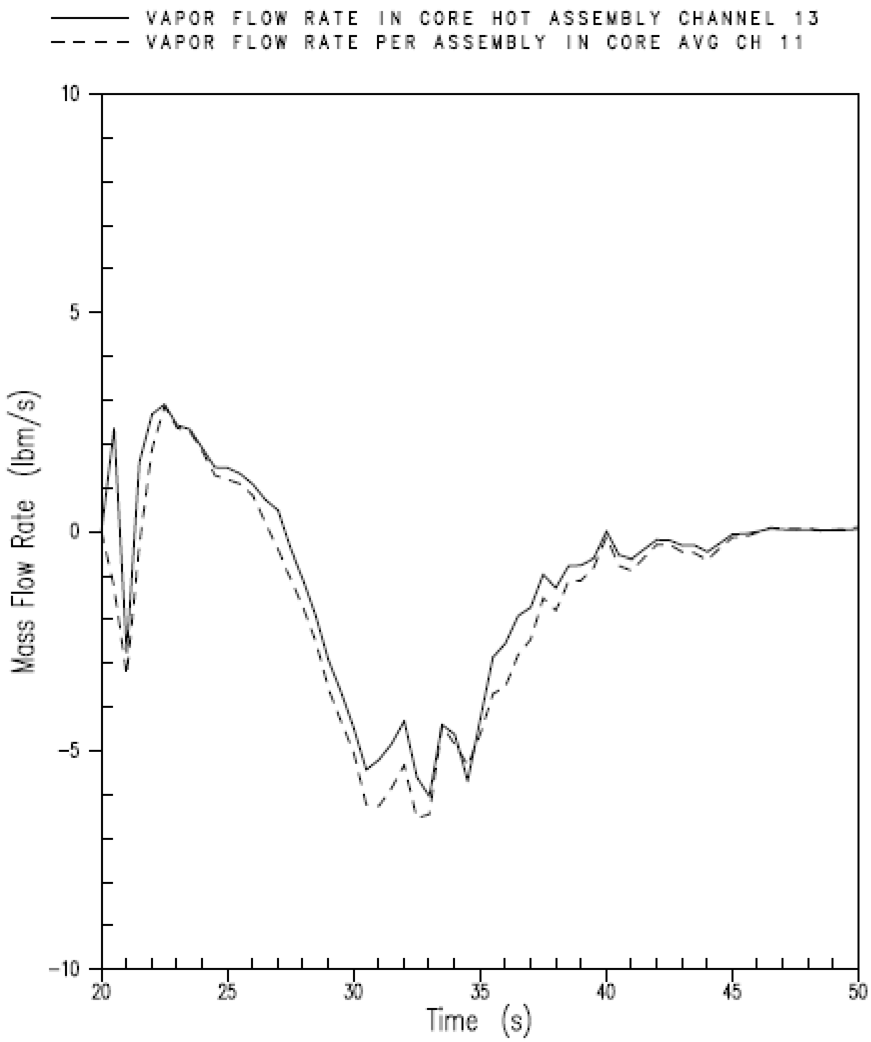
Figure 15.4-1d



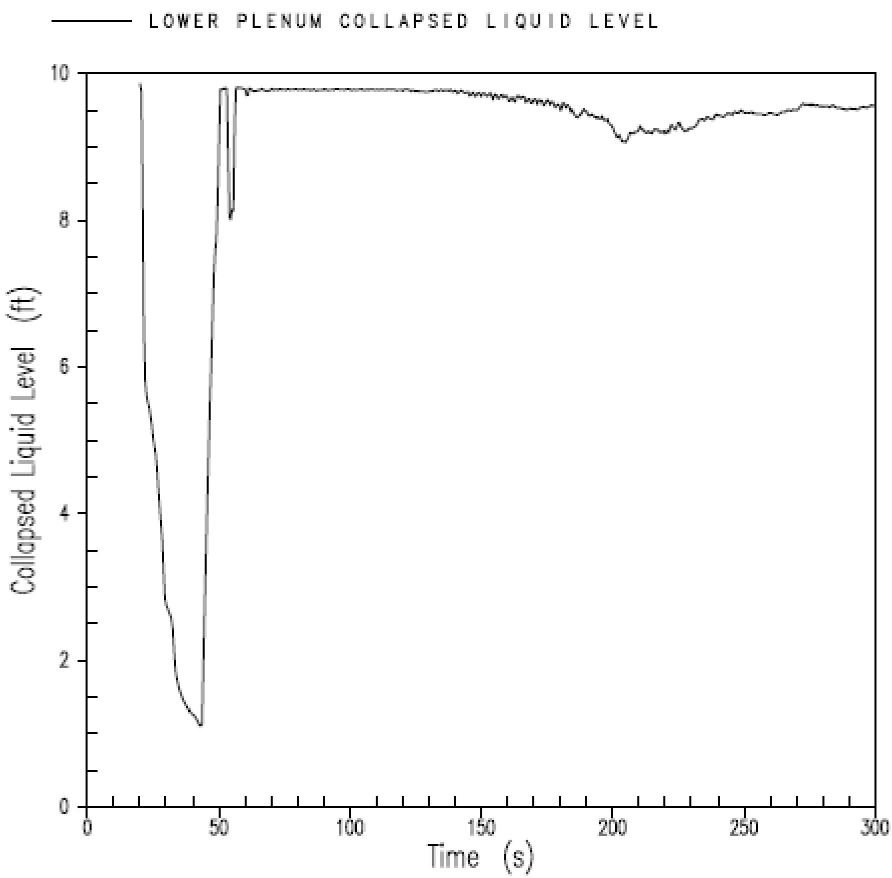
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Void Fraction at the Intact and Broken Loop Pump Inlet for Reference Transient

Figure 15.4-1e



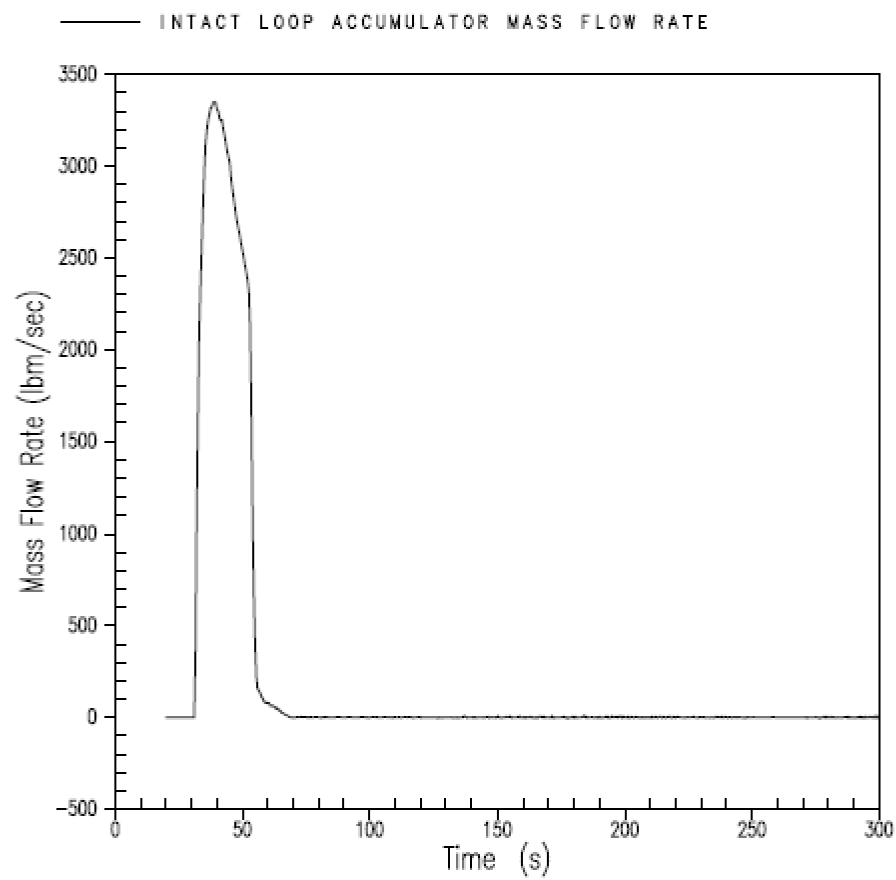
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Vapor Flow Rate at Midcore in Channels 11 and 13 During Blowdown for Reference Transient Figure 15.4-1f



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Collapsed Liquid Level in Lower Plenum for Reference Transient

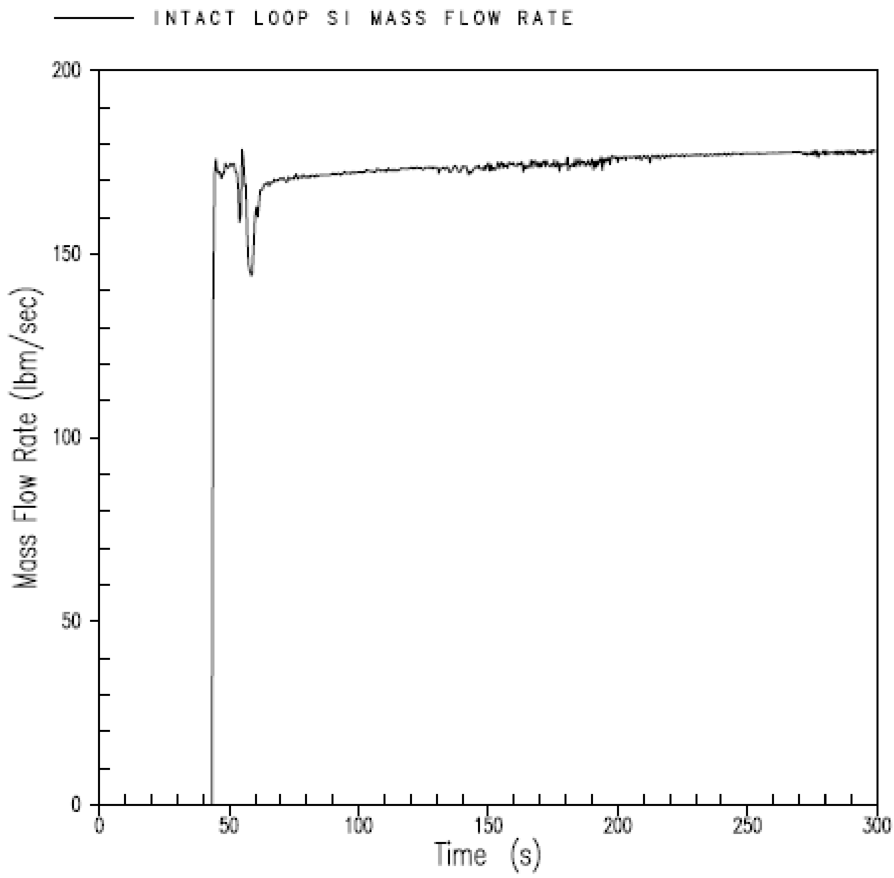
Figure 15.4-1g



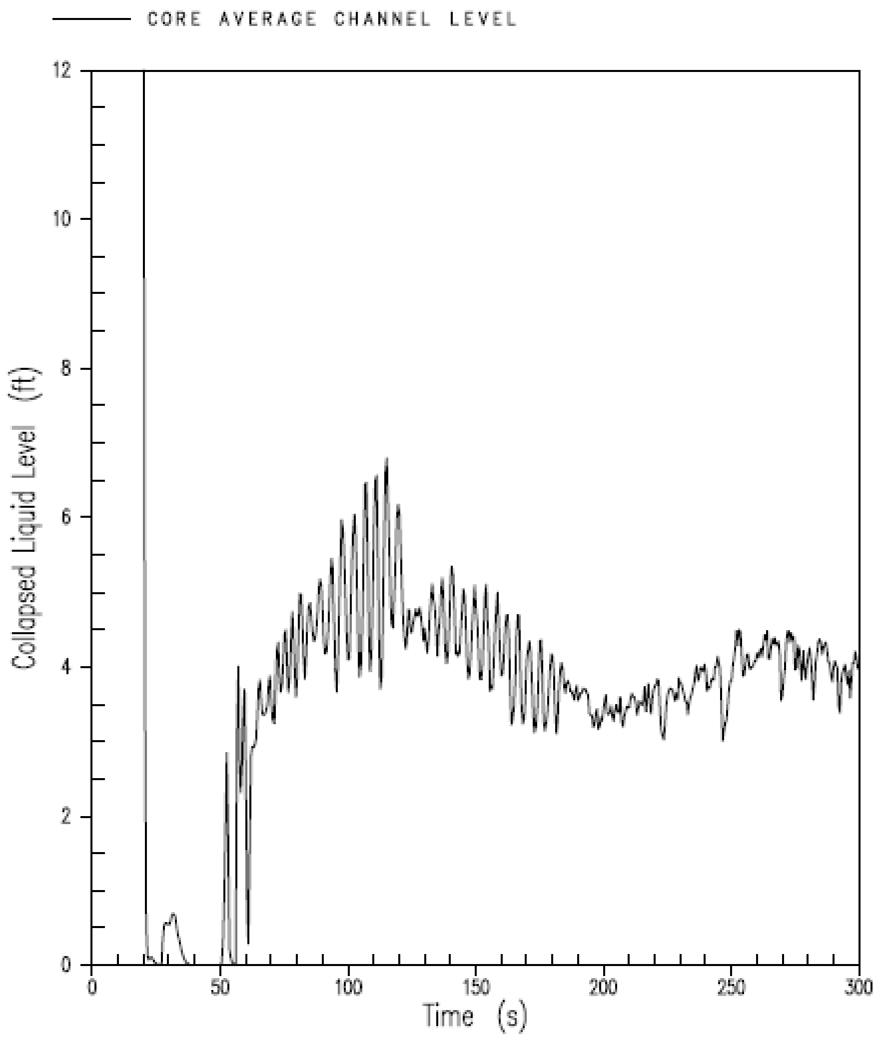
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

One Intact Loop Accumulator Mass Flow Rate for Reference Transient

Figure 15.4-1h



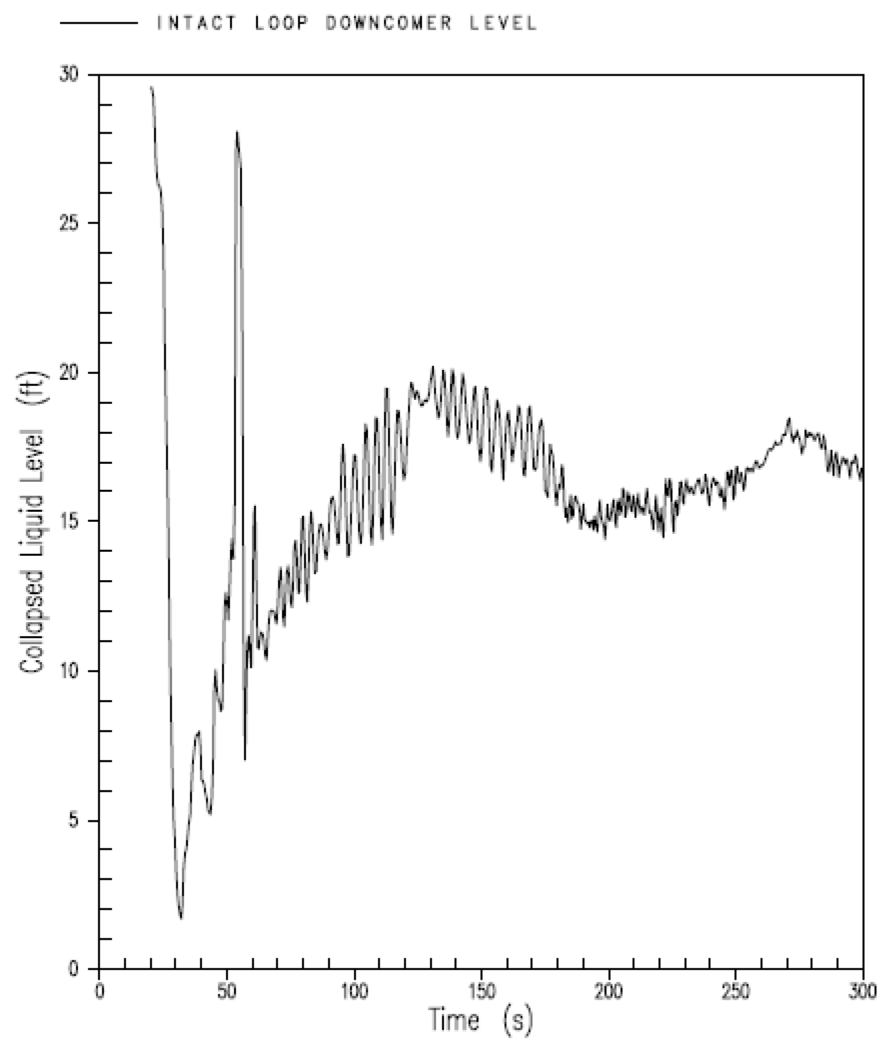
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
One Intact Loop Safety Injection Mass Flow Rate for Reference Transient Figure 15.4-1i



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Collapsed Liquid Level in Core for Reference Transient

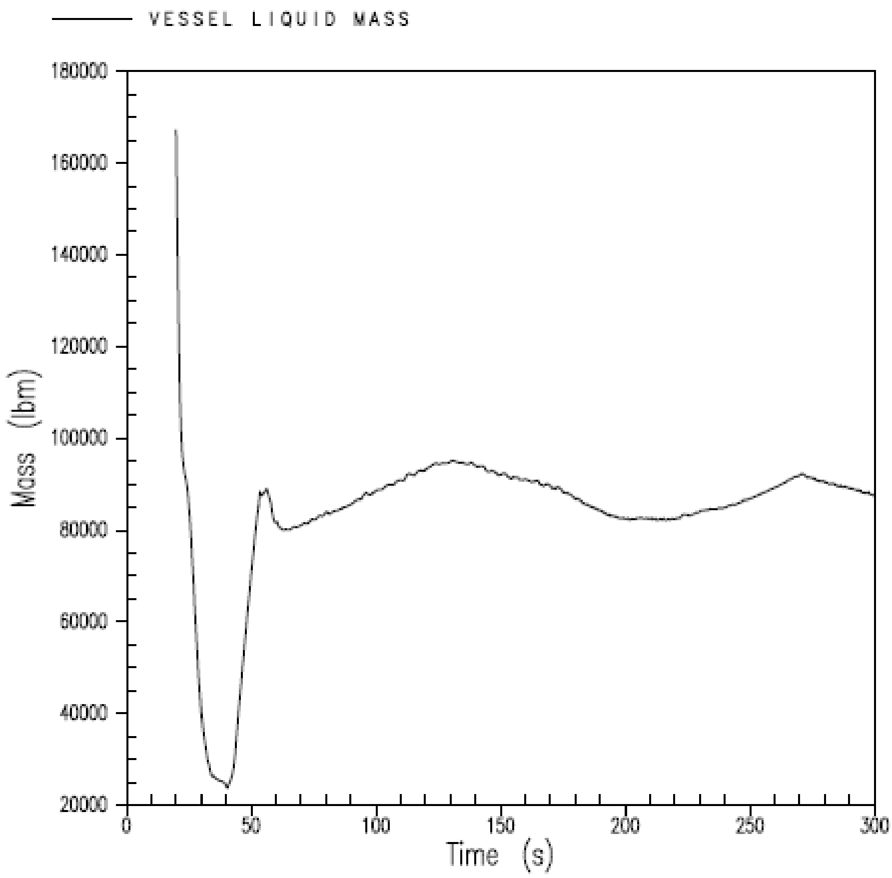
Figure 15.4-1j



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Collapsed Liquid Level in Downcomer for Reference Transient

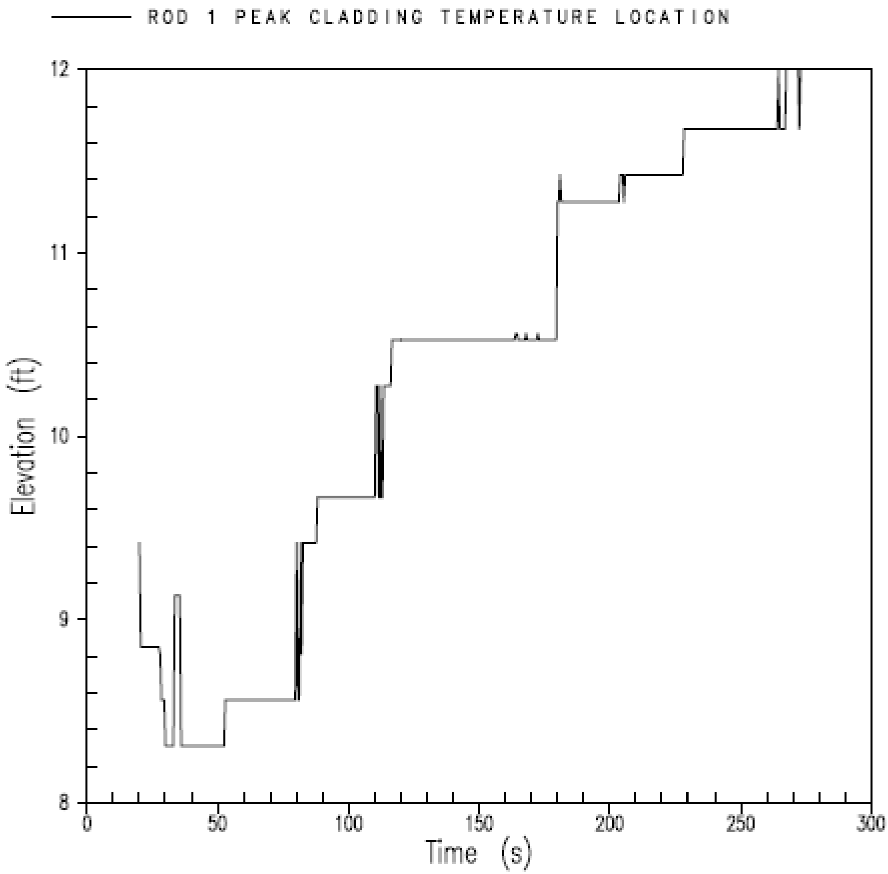
Figure 15.4-1k



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Vessel Fluid Mass for Reference Transient

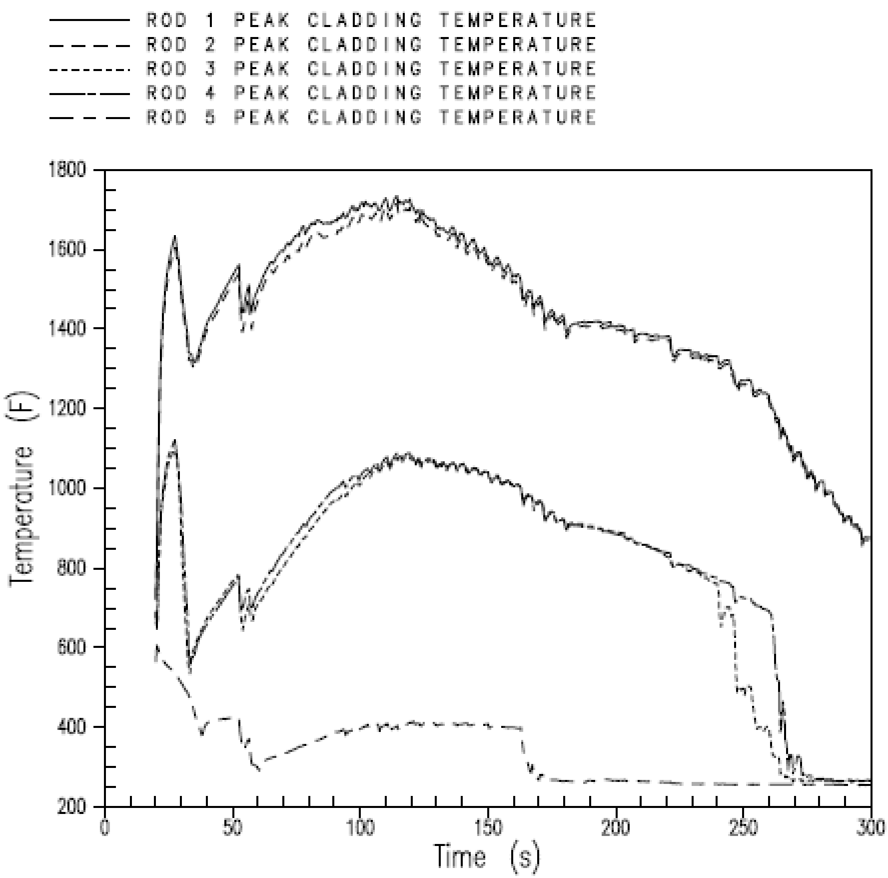
Figure 15.4-1I



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Peak Cladding Temperature Location for Reference Transient

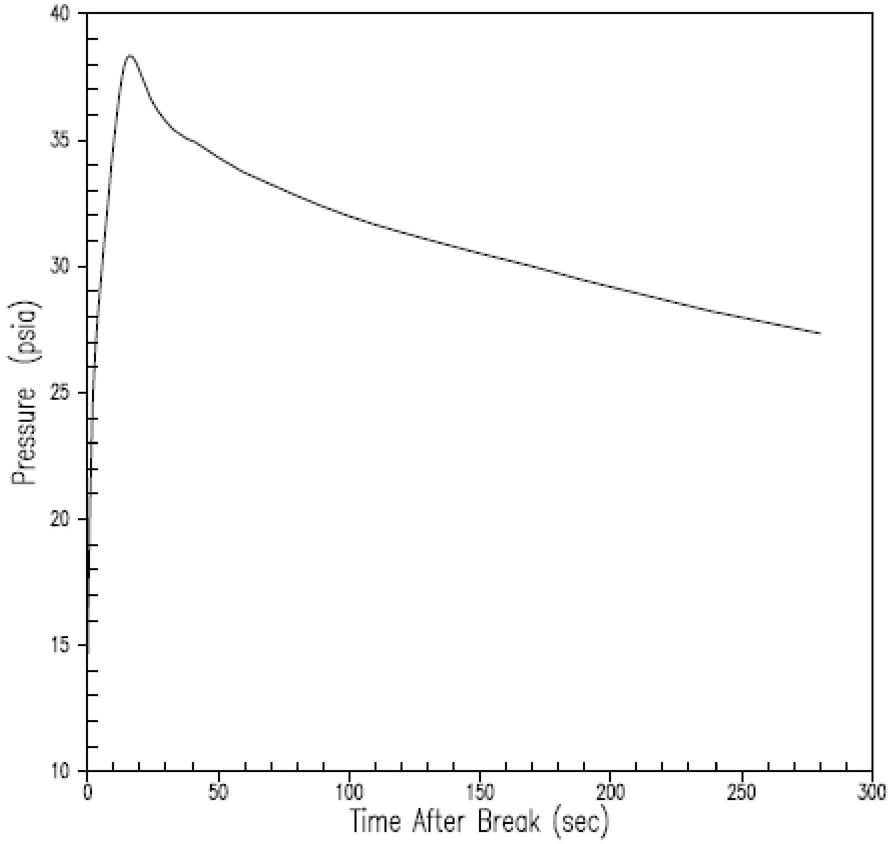
Figure 15.4-1m



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Peak Cladding Temperature Comparison for Five Rods for Reference Transient

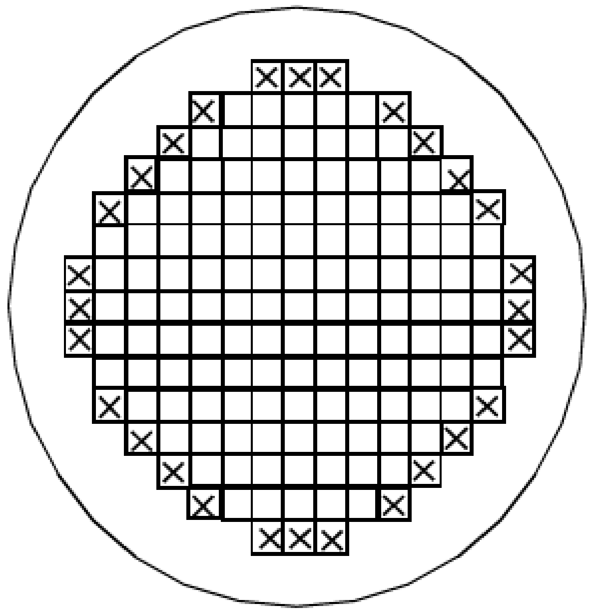
Figure 15.4-1n



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Lower Bound Containment Pressure for Best Estimate Large Break LOCA

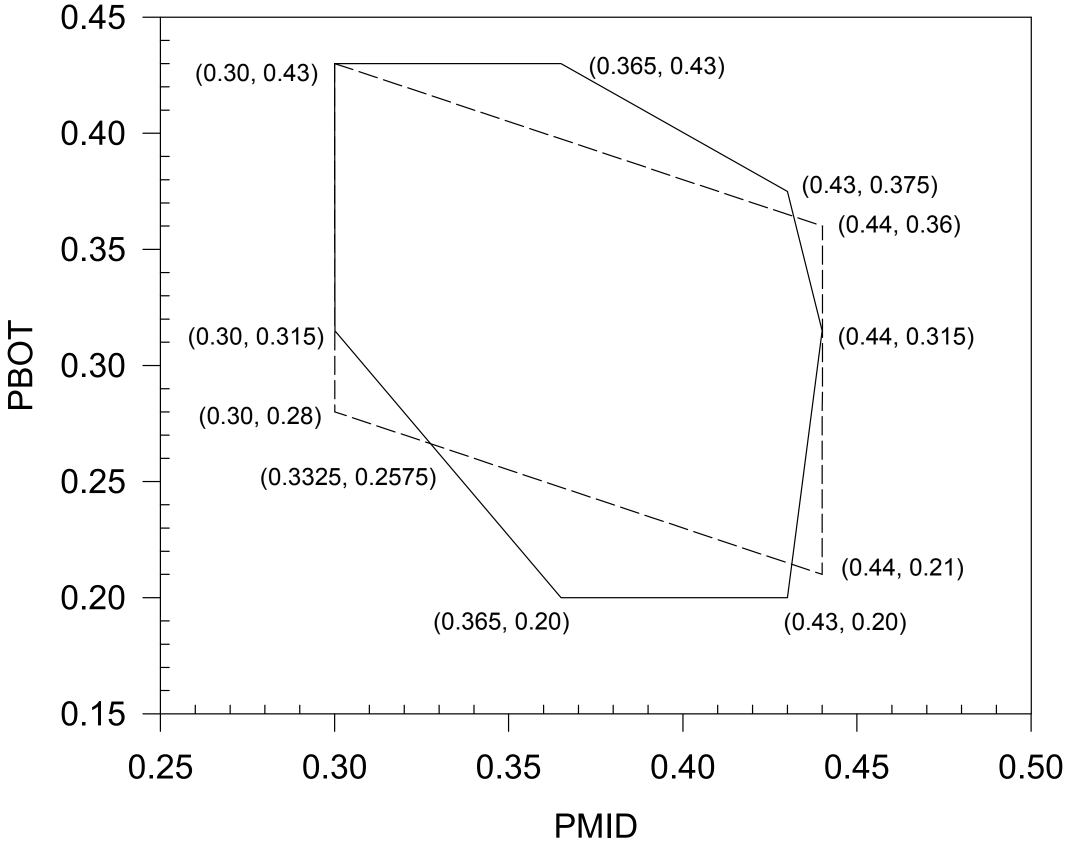
Figure 15.4-1o



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

28 Core Peripheral Assembly Locations

Figure 15.4-1p

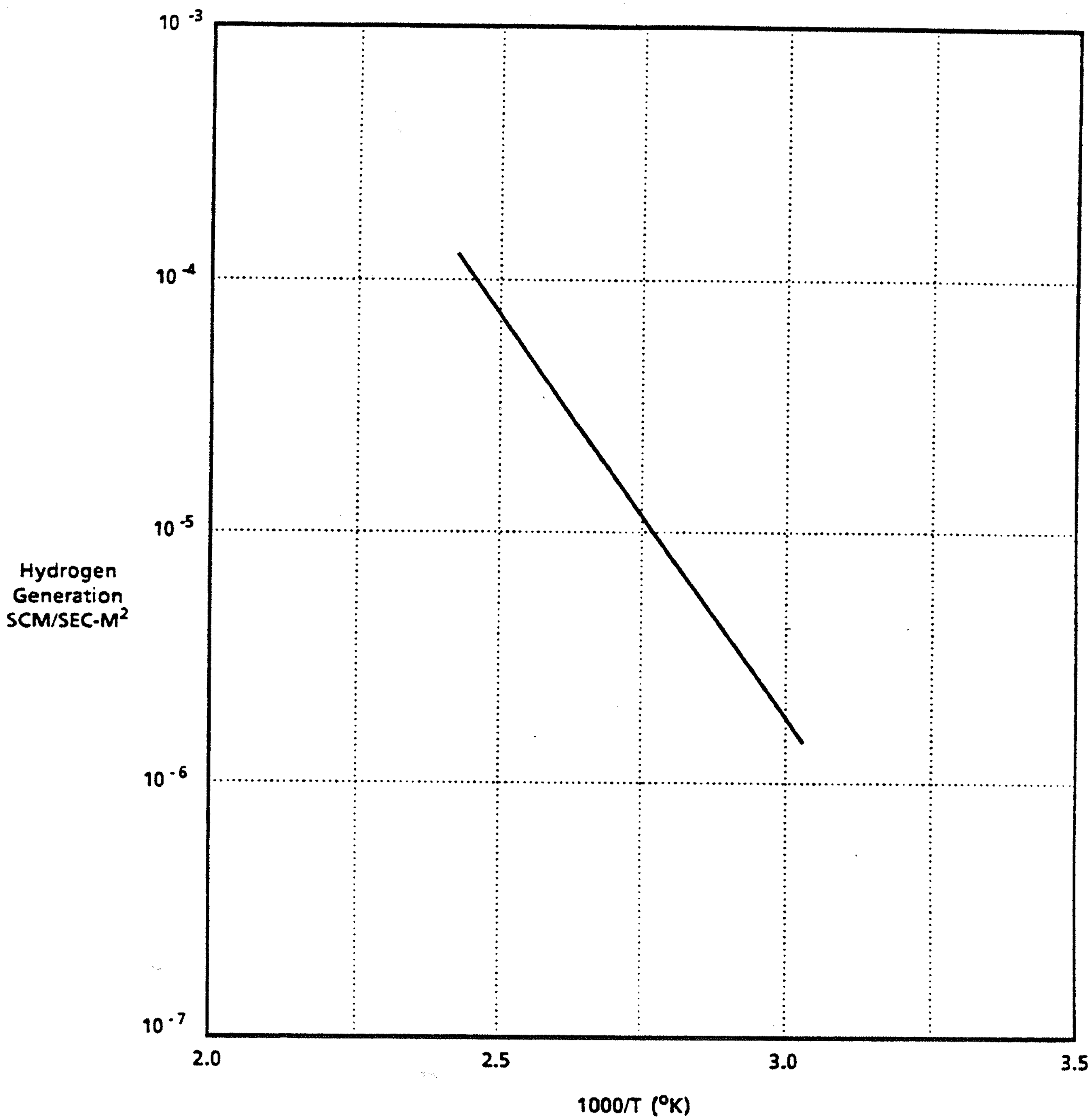


Note: MONTEC sampling range indicated by dashed line is Analysis Limit.
Associated WCOBRA/TRAC response surface range is indicated by solid line.

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
PBOT/PMID Core Design Limit Figure 15.4-1q

Figures 15.4-2 Through 15.4-67
Deleted per RN 06-040

RN
06-040

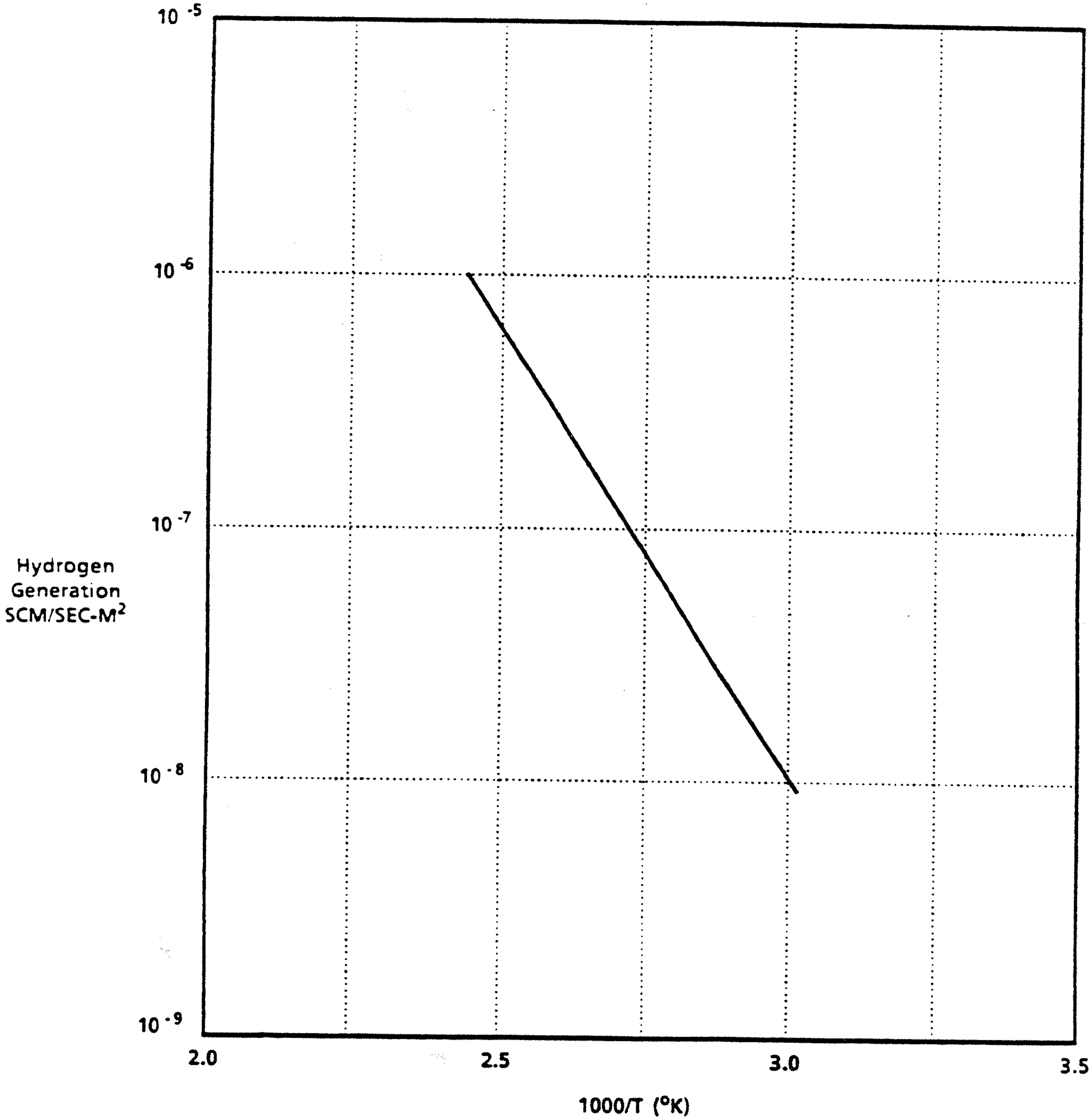


AMENDMENT 96-03
SEPTEMBER 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Hydrogen Generation from Aluminum Corrosion - ANS 56.1

Figure 15.4-68

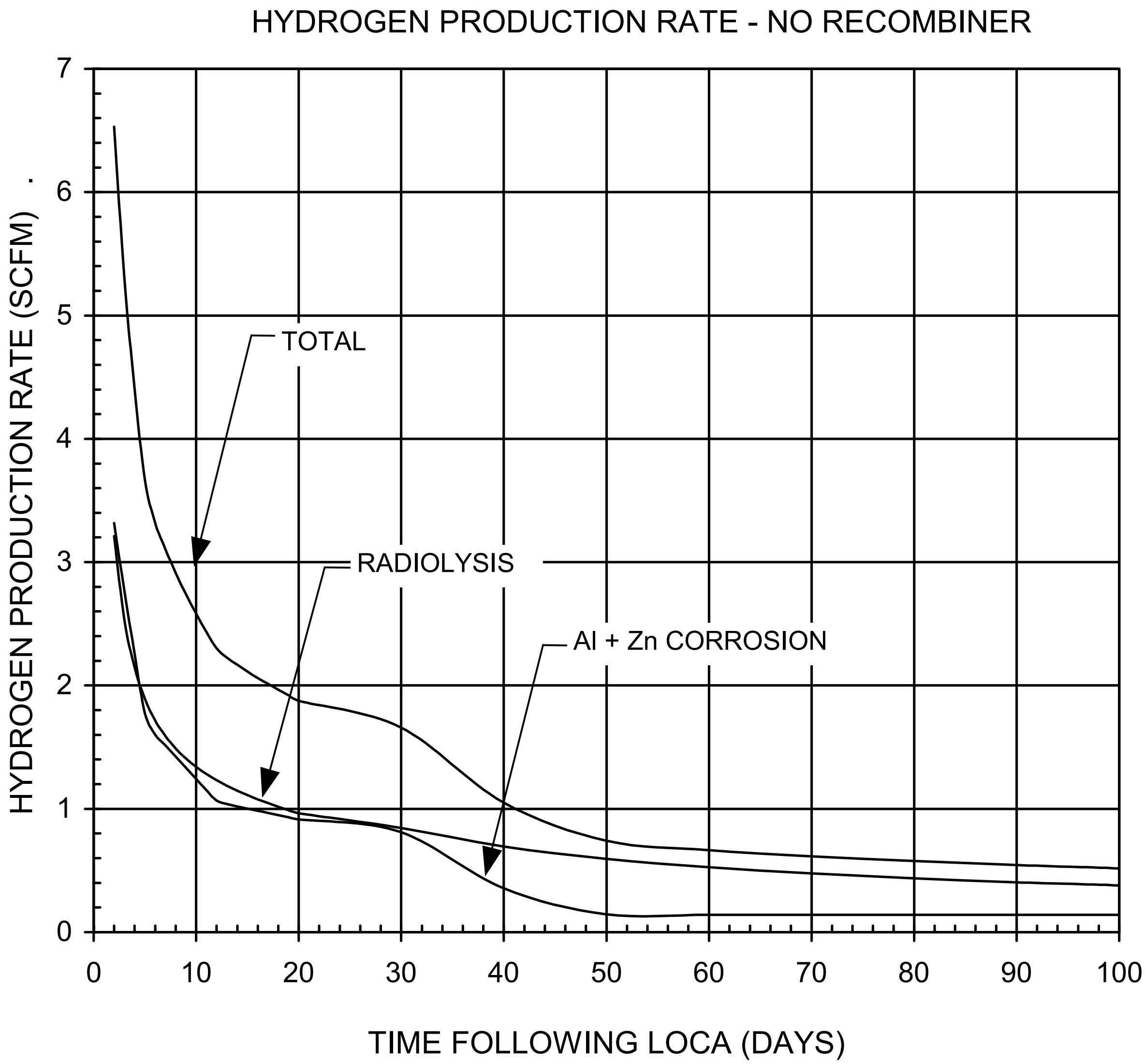


AMENDMENT 96-03
SEPTEMBER 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Hydrogen Generation from Zinc Corrosion - ANS 56.1

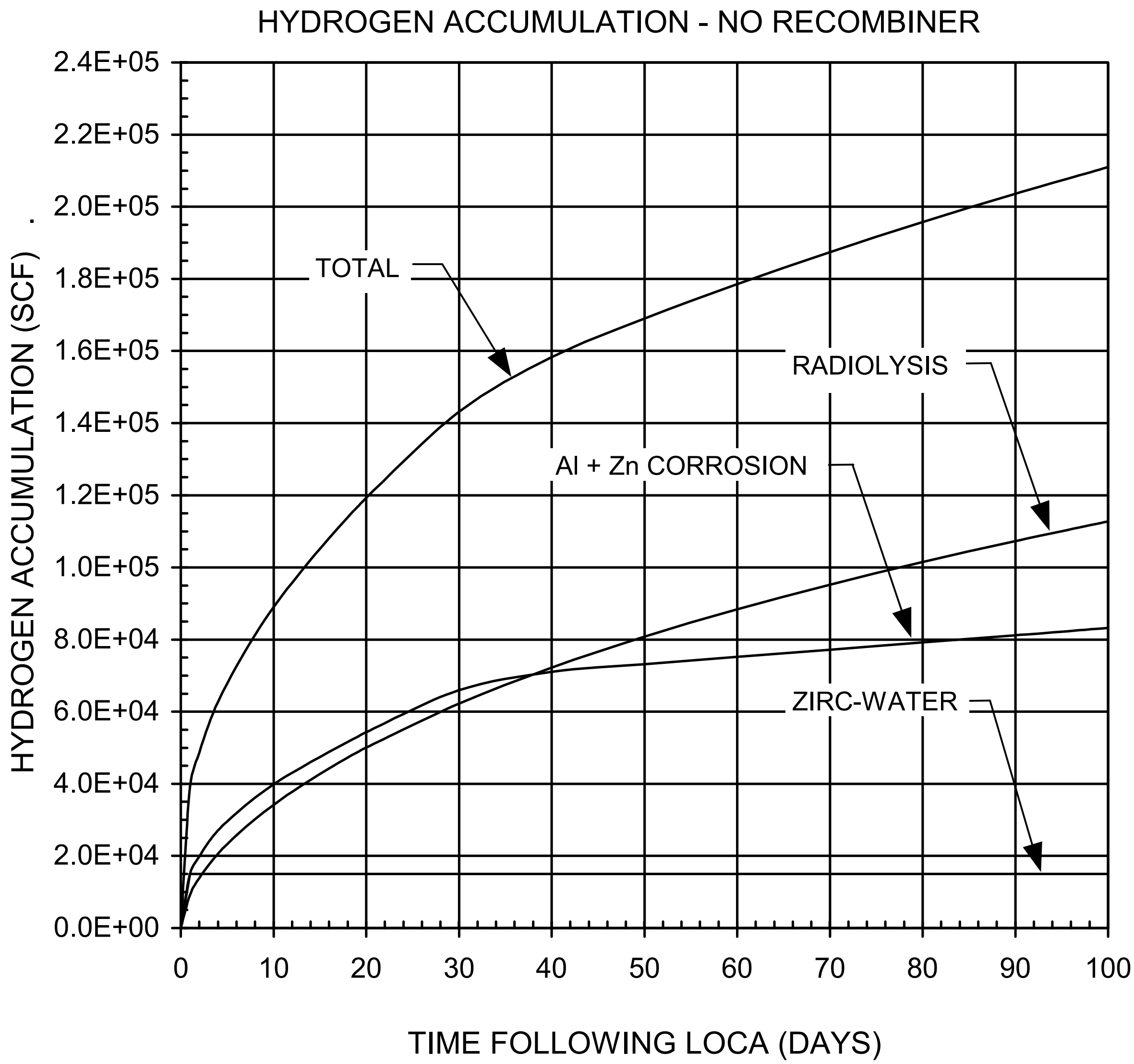
Figure 15.4-69



RN 03-005

RN 03-005
June 2003

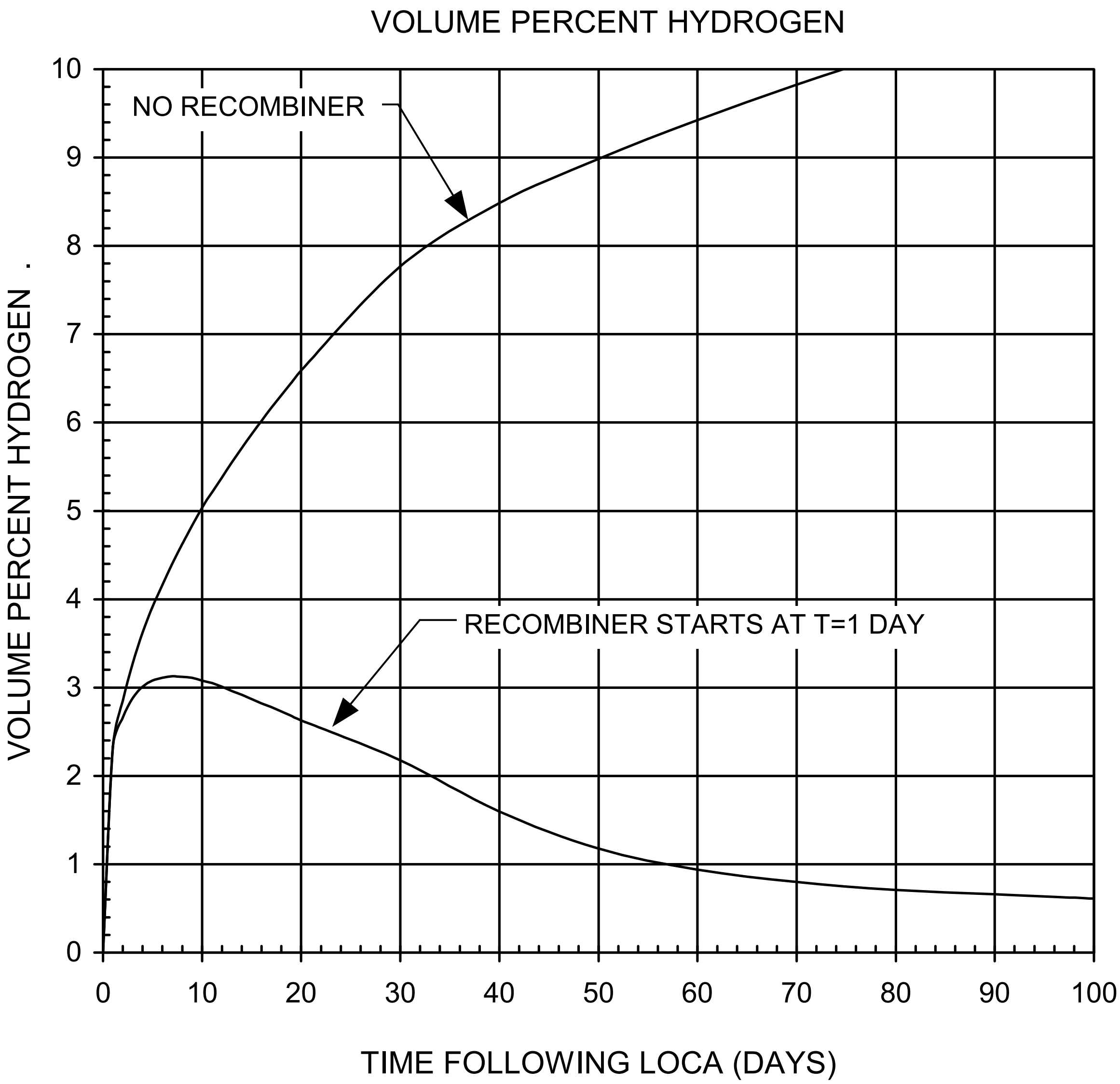
SOUTH CAROLINA ELECTRIC & GAS CO VIRGIL C. SUMMER NUCLEAR STATION
Hydrogen Production Rate No Recombiner Figure 15.4-71



RN 03-005

RN 03-005
June 2003

SOUTH CAROLINA ELECTRIC & GAS CO VIRGIL C. SUMMER NUCLEAR STATION
Hydrogen Accumulation No Recombiner Figure 15.4-72



RN
03-005

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June 2003

SOUTH CAROLINA ELECTRIC & GAS CO
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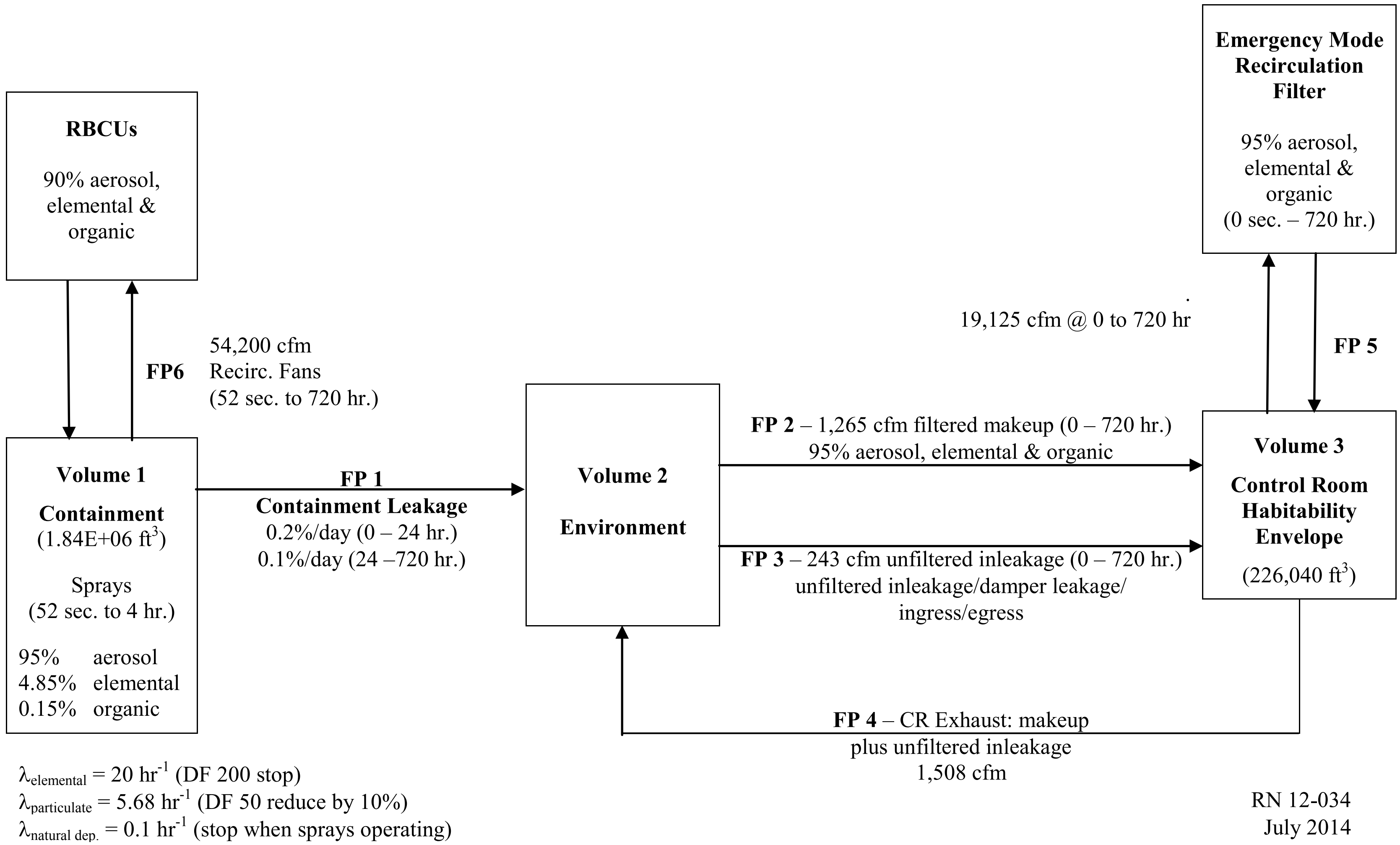
Volume Percent Hydrogen
In Containment

Figure 15.4-73

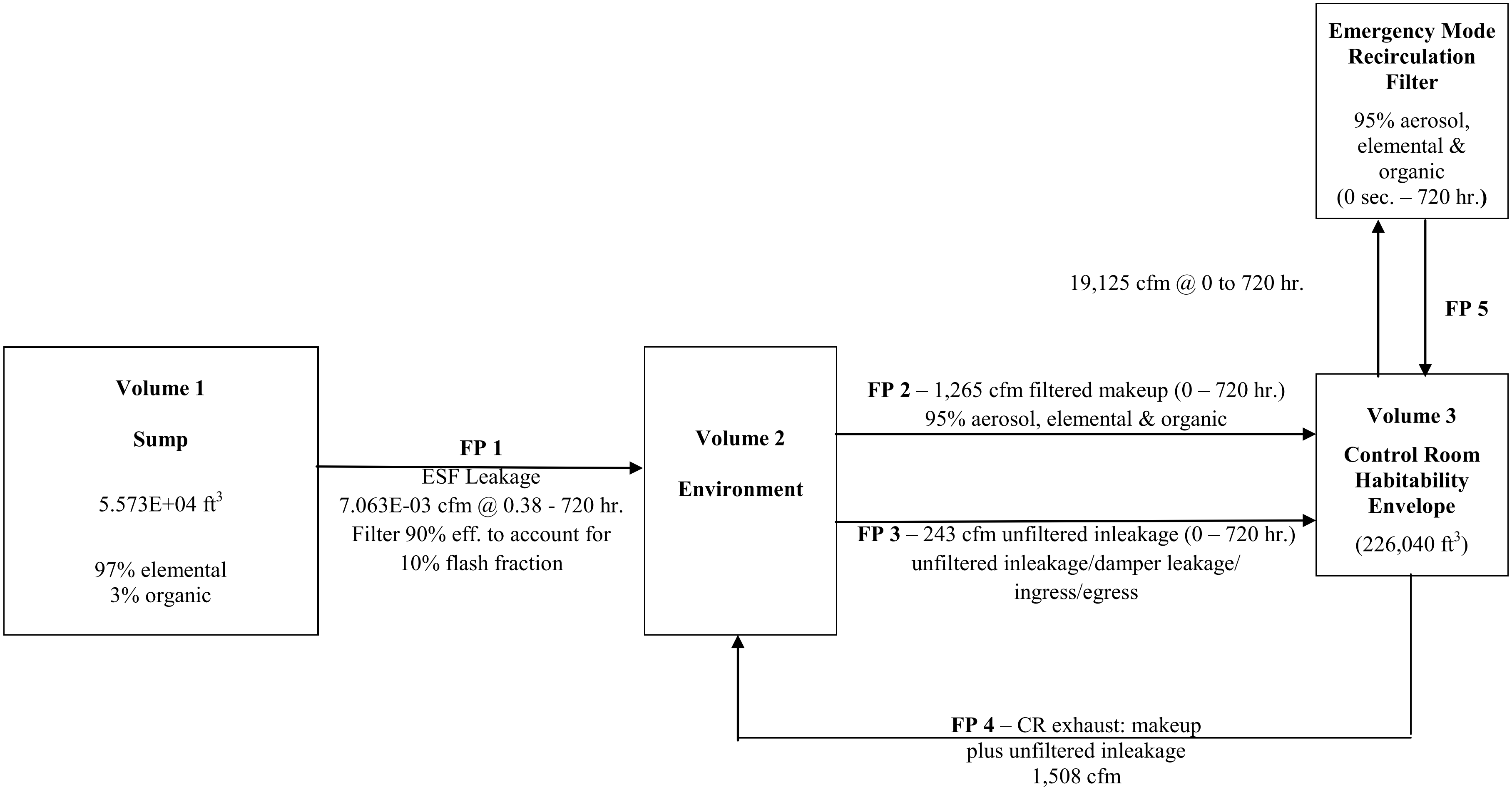
Figure 15.4-74
RADTRAD Model for Containment Source Release - LOCA



RN 12-034
July 2014

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
RADTRAD Model for Containment Source Release - LOCA Figure 15.4-74

Figure 15.4-74A
RADTRAD Model for ESF Leakage Source Release - LOCA



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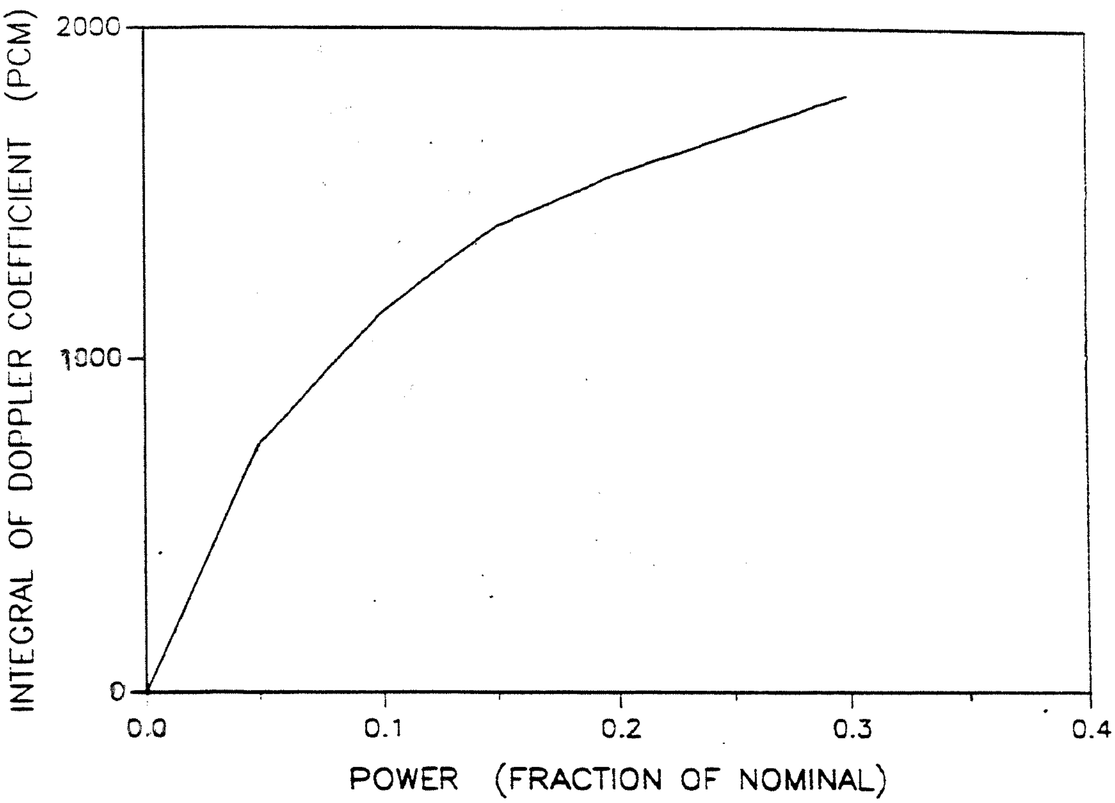
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RADTRAD Model for
ESF Leakage Source Release - LOCA

Figure 15.4-74A

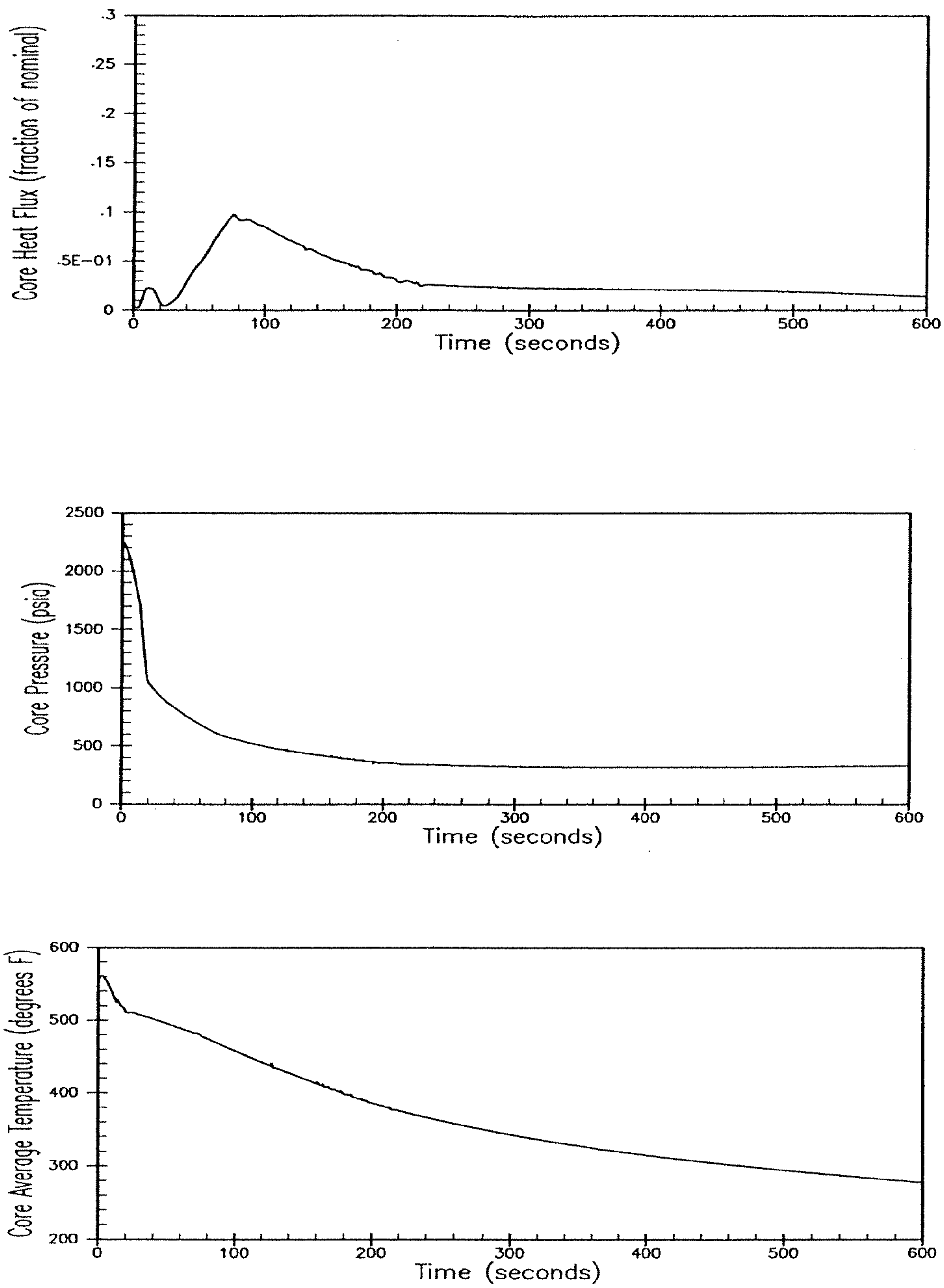


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Variation of Reactivity With Power At Constant Core Average Temperature

Figure 15.4-75

Amendment 96-02
July 1996



RN
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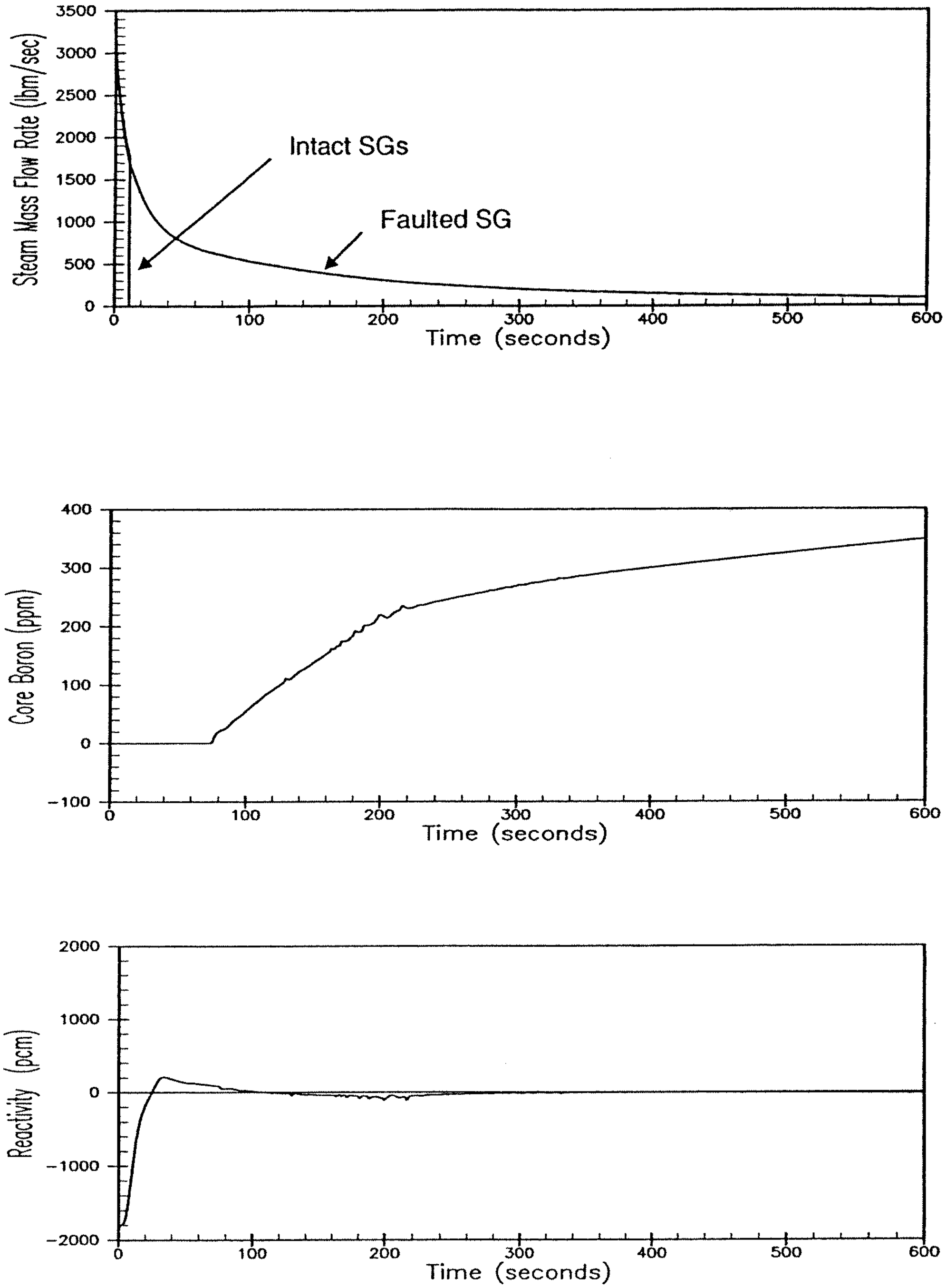
May 2002

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Transient Response to a Steam Line Break Double Ended Rupture With Offsite Power Available (Case A)

Figure 15.4-76



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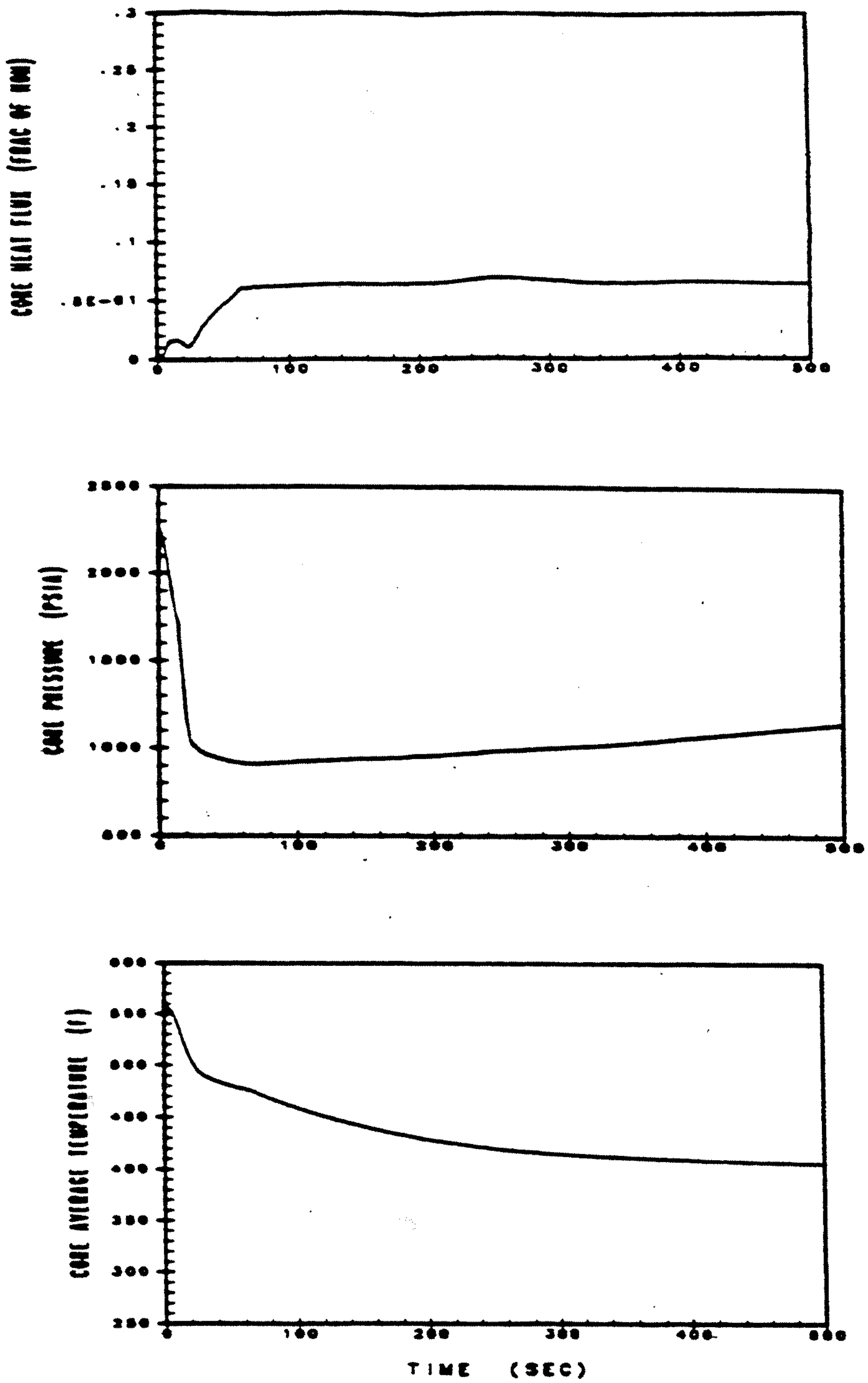
May 2002

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Transient Response to a Steam Line Break Double Ended Rupture With Offsite Power Available (Case A)

Figure 15.4-77



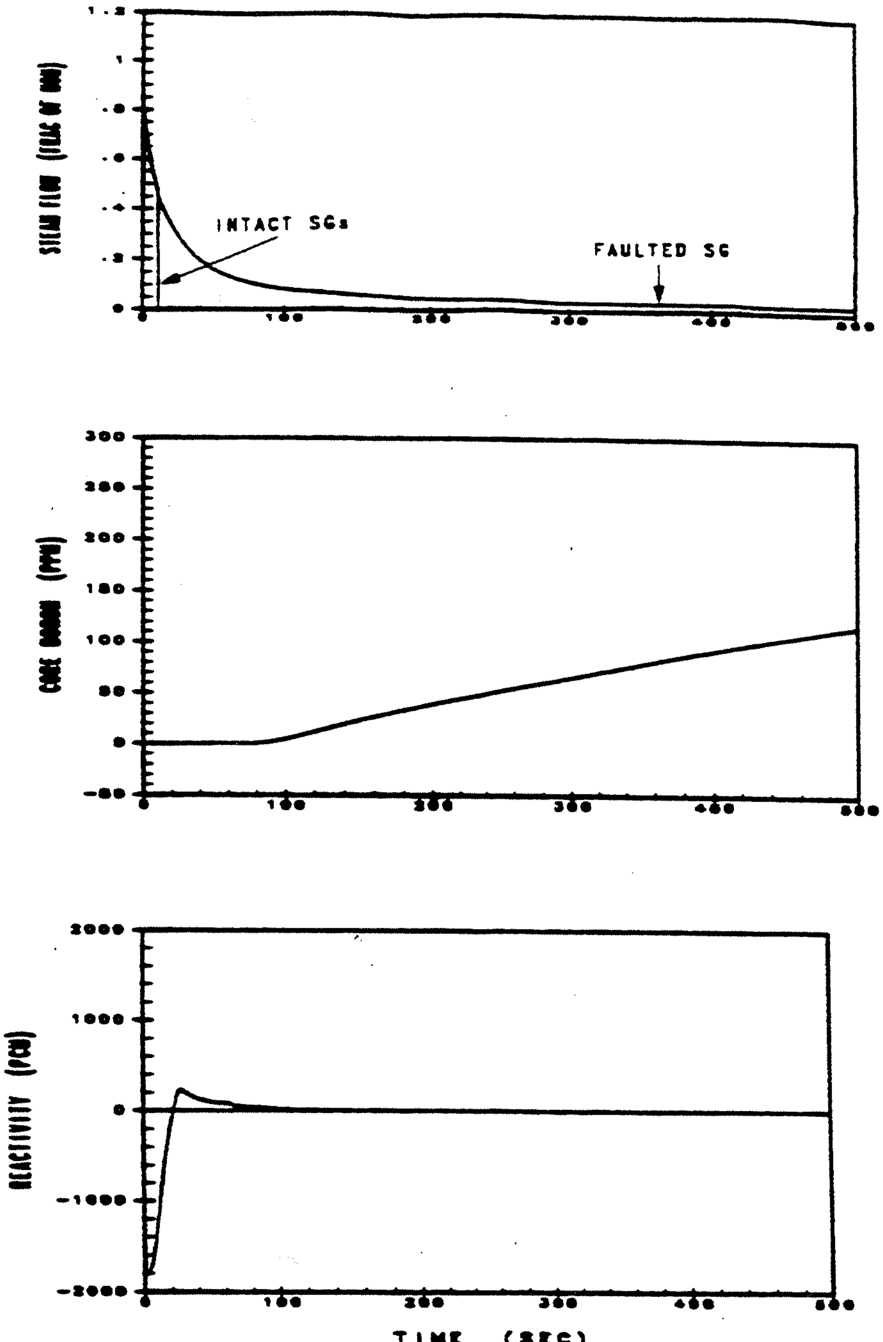
AMENDMENT 99-01
JUNE 1999

SOUTH CAROLINA ELECTRIC & GAS CO.

VIRGIL C. SUMMER NUCLEAR STATION

Transient Response To A Steam Line Break Double Ended Rupture Without Offsite Power Available (Case B)

Figure 15.4-78



AMENDMENT 99-01
JUNE 1999

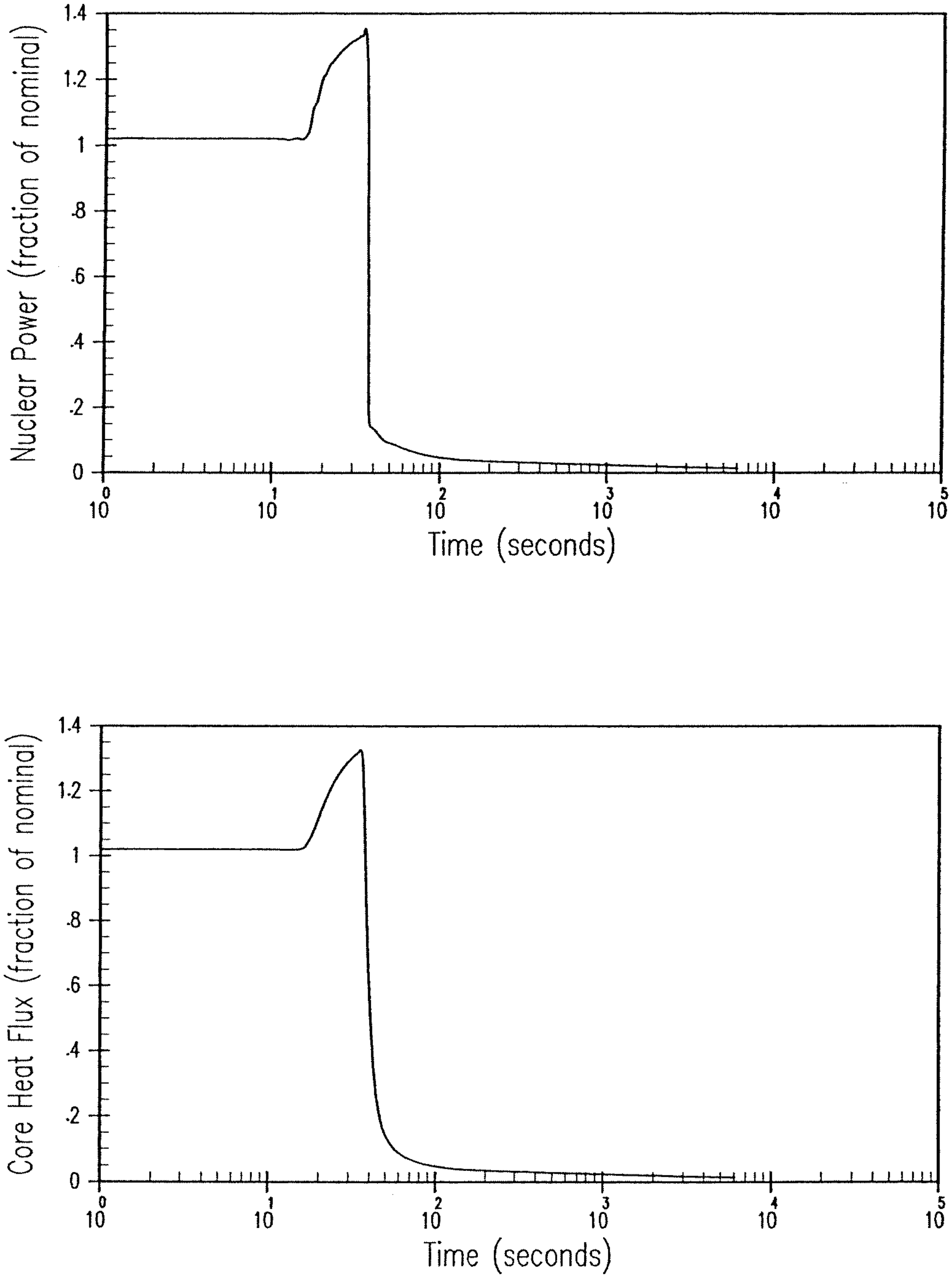
SOUTH CAROLINA ELECTRIC & GAS CO.

VIRGIL C. SUMMER NUCLEAR STATION

Transient Response To A Steam Line Break Double Ended Rupture Without Offsite Power Available (Case B)

Figure 15.4-79

Figures 15.4-80 Through 15.4-82
Have Been Deleted



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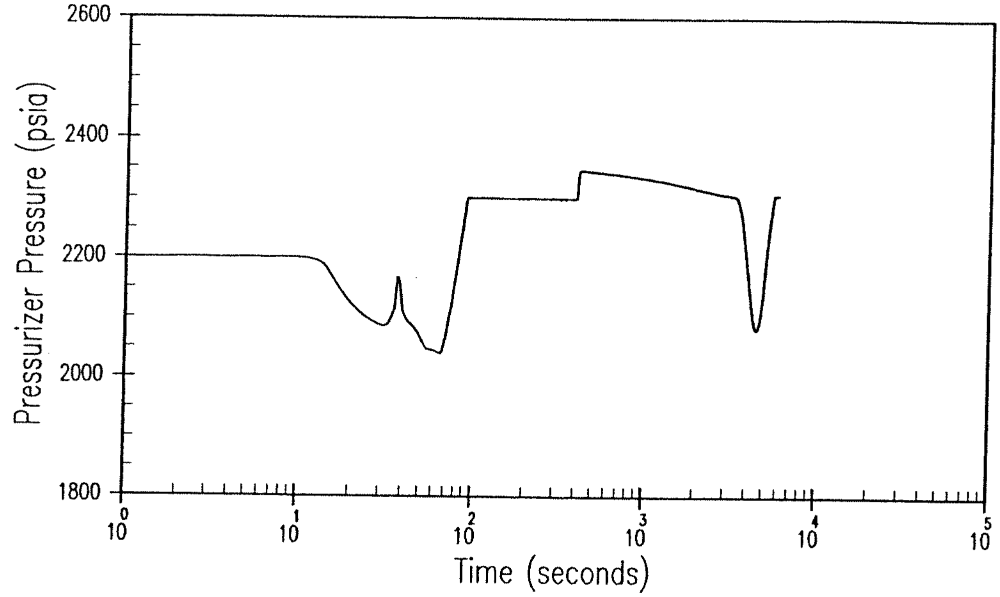
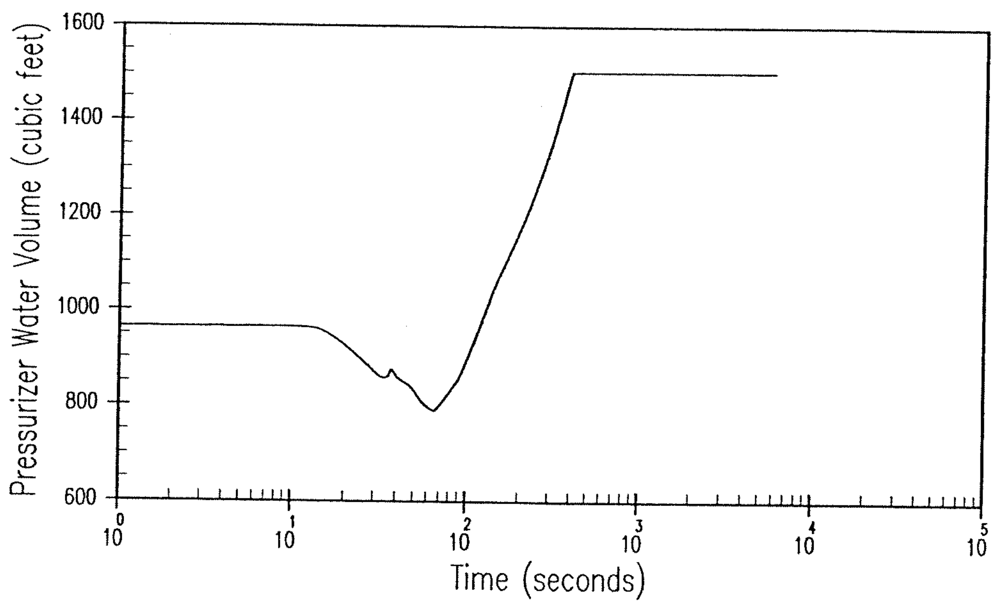
May 2002

SOUTH CAROLINA ELECTRIC & GAS CO VIRGIL C. SUMMER NUCLEAR STATION
Main Feedwater Rupture With Offsite Power Nuclear Power and Core Heat Flux vs. Time Figure 15.4-83





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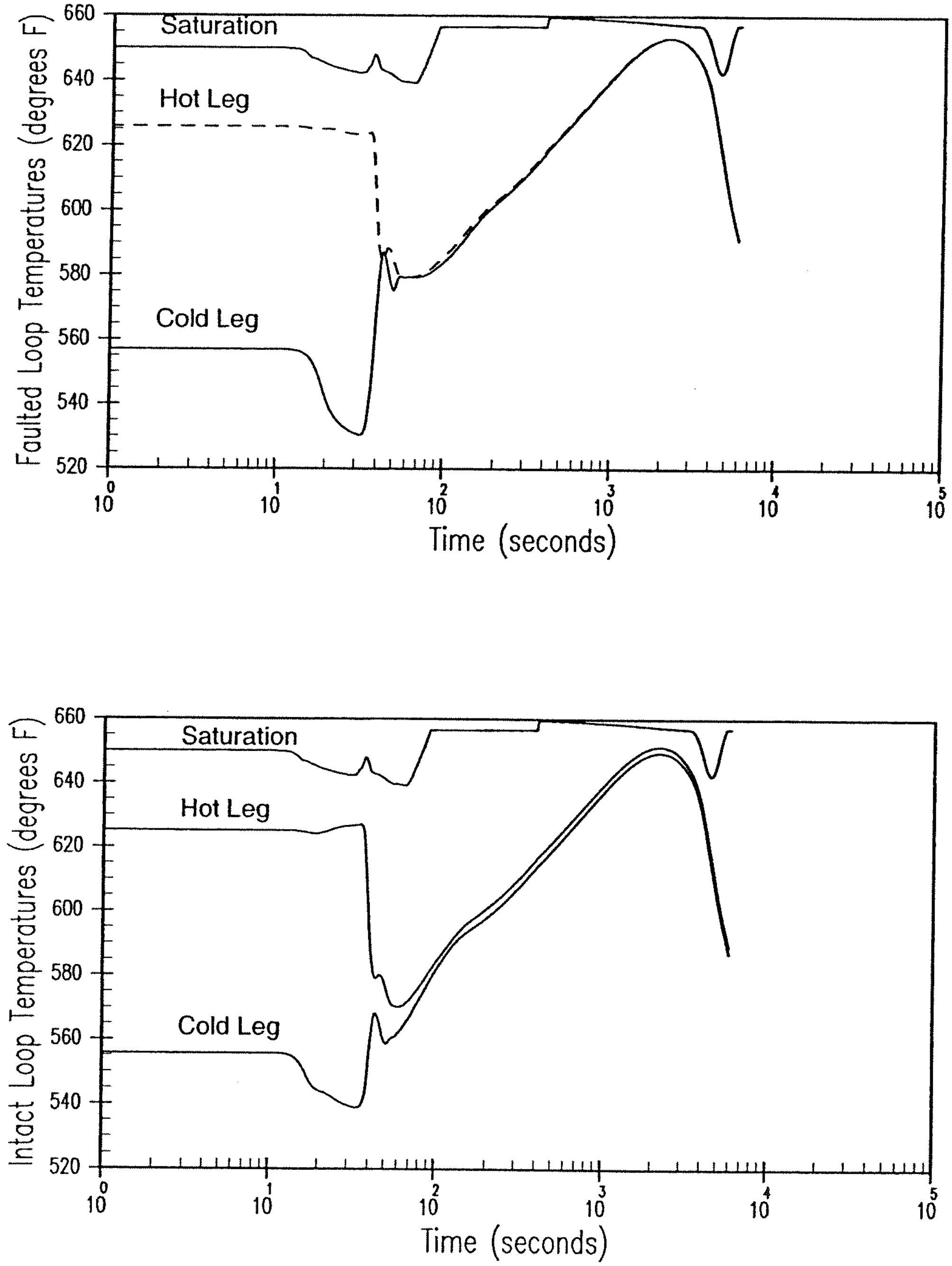
May 2002

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedwater Rupture With Offsite Power
Pressurizer Pressure and Water Volume vs. Time

Figure 15.4-84



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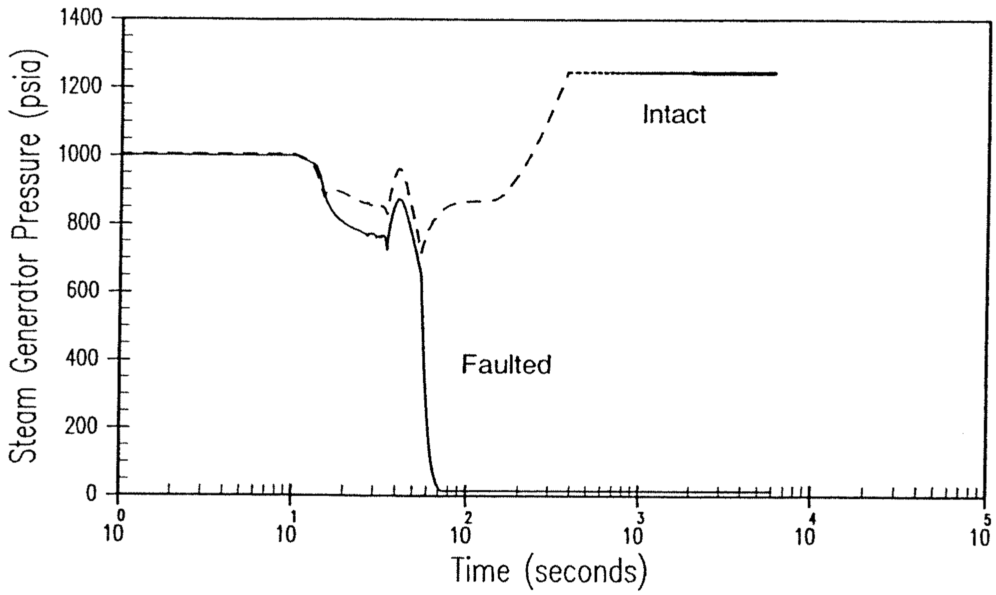
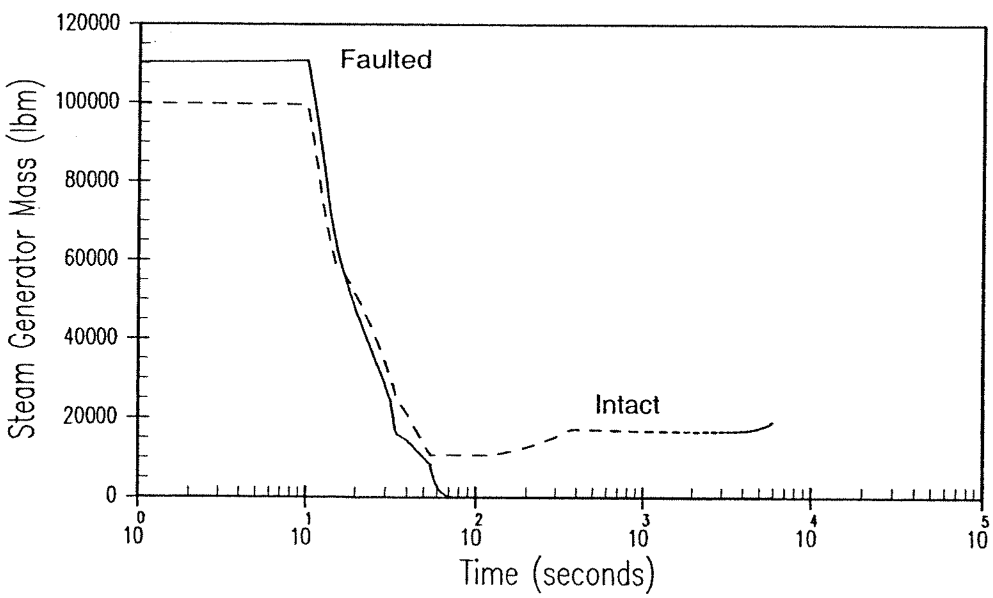
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedwater Rupture With Offsite Power
Faulted and Intact loop Coolant
Temperature vs. Time

Figure 15.4-85





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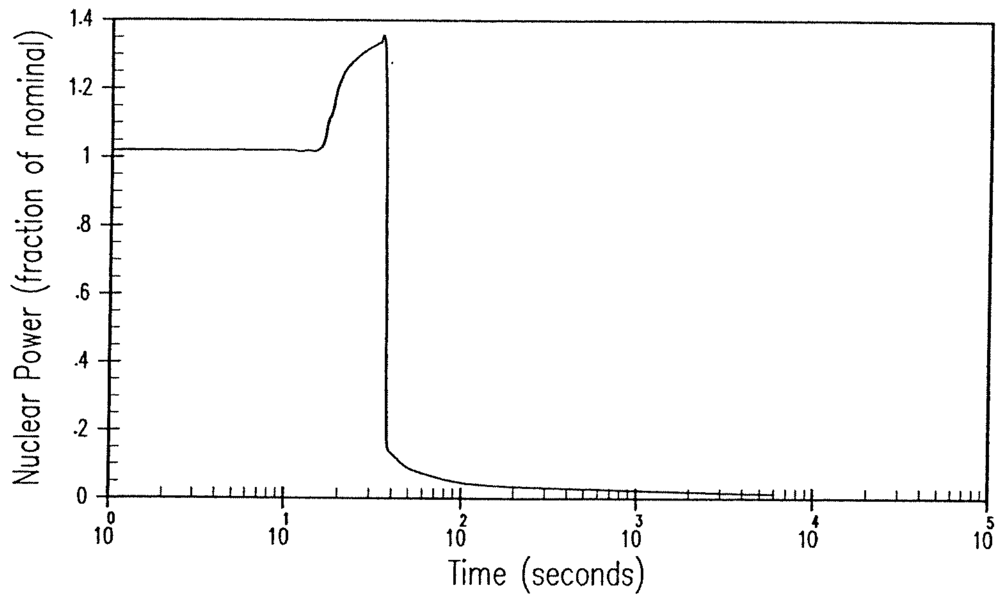
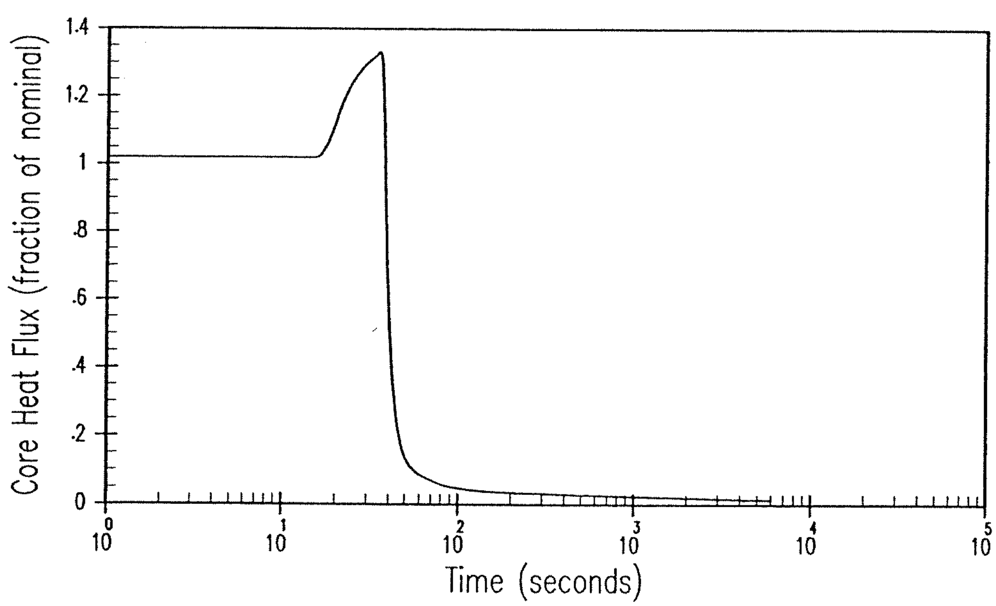
SOUTH CAROLINA ELECTRIC & GAS CO VIRGIL C. SUMMER NUCLEAR STATION
Main Feedwater Rupture With Offsite Power Steam Generator Pressure and Water Mass vs. Time Figure 15.4-86





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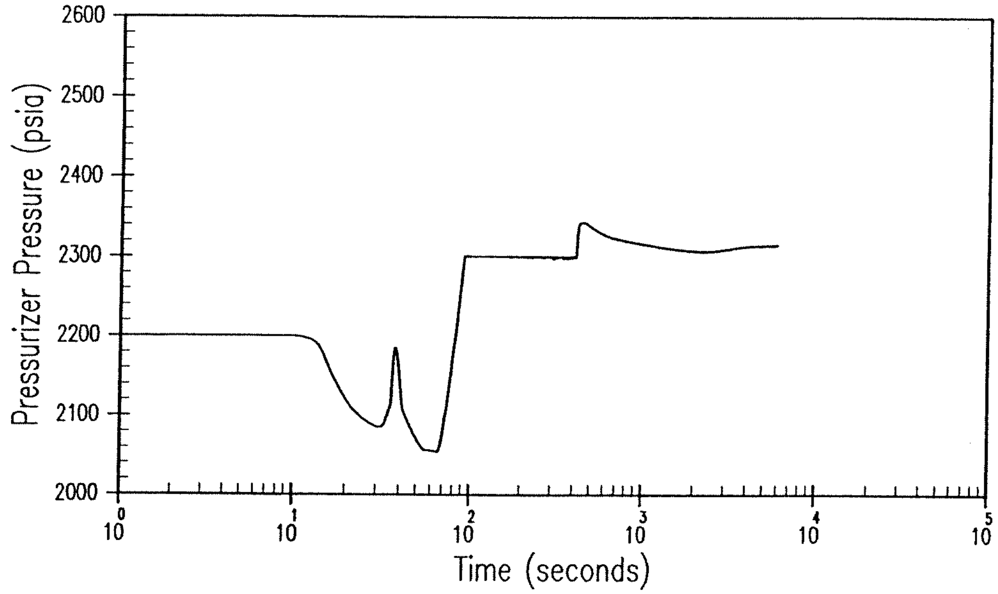
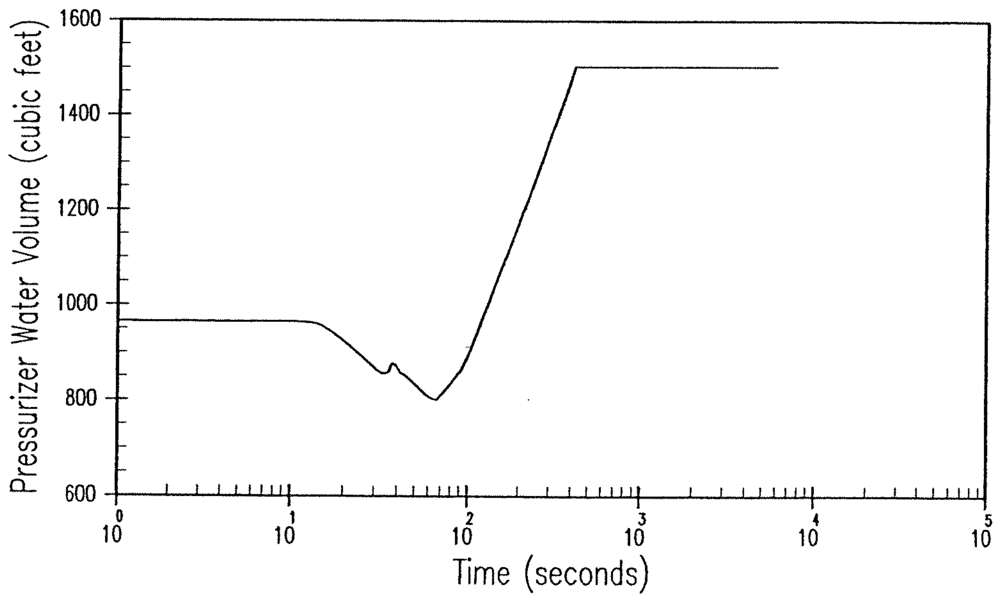
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedwater Rupture Without Offsite Power
Nuclear Power and Core Heat Flux vs. Time

Figure 15.4-87





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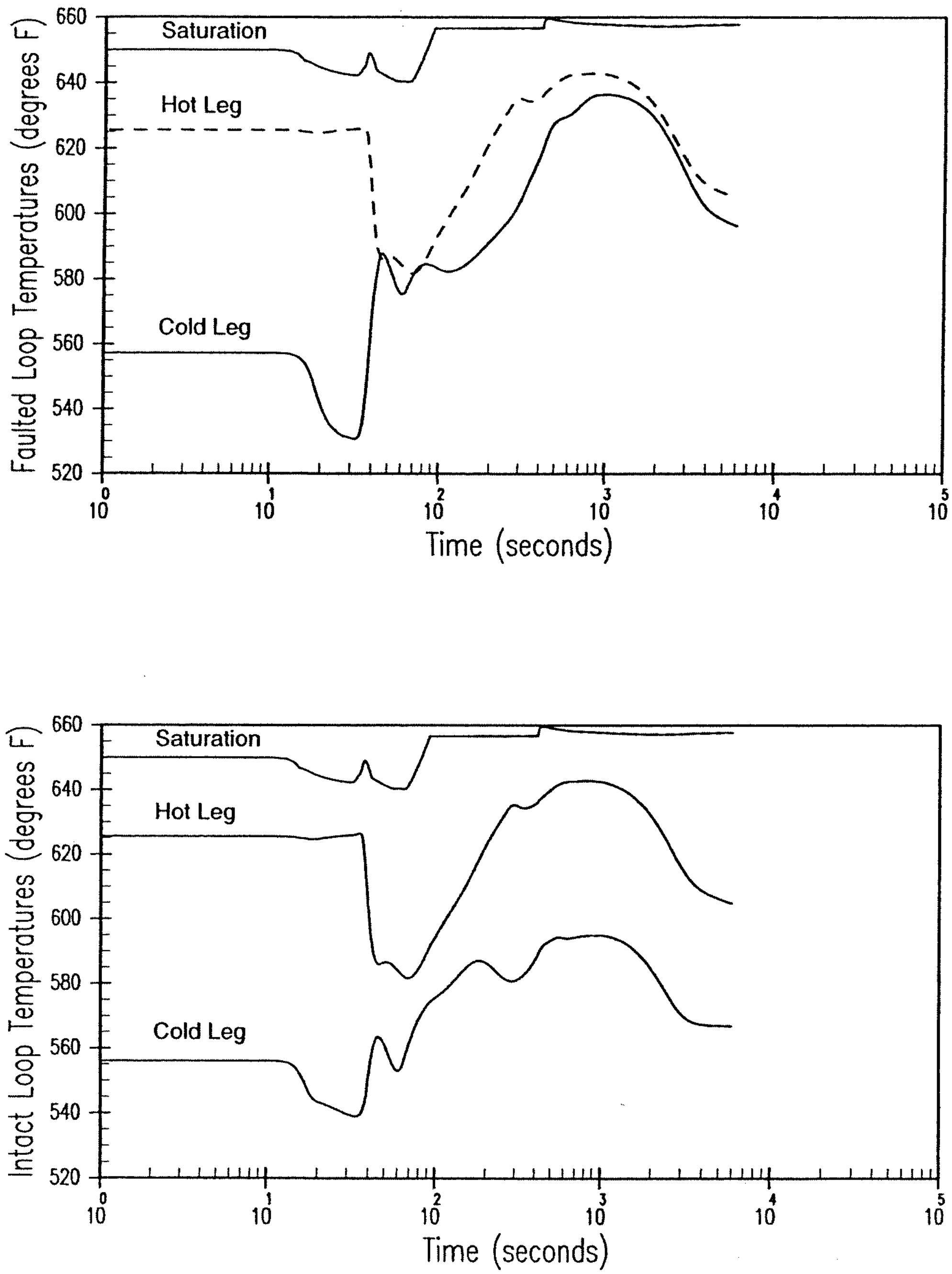
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedwater Rupture Without Offsite Power Pressurizer Pressure and Water Volume vs. Time

Figure 15.4-88



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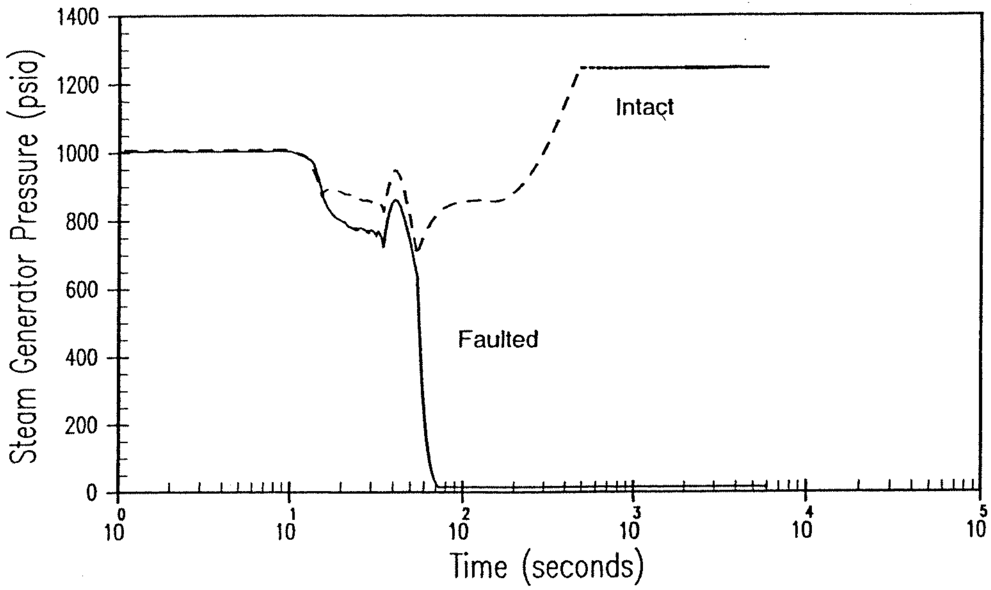
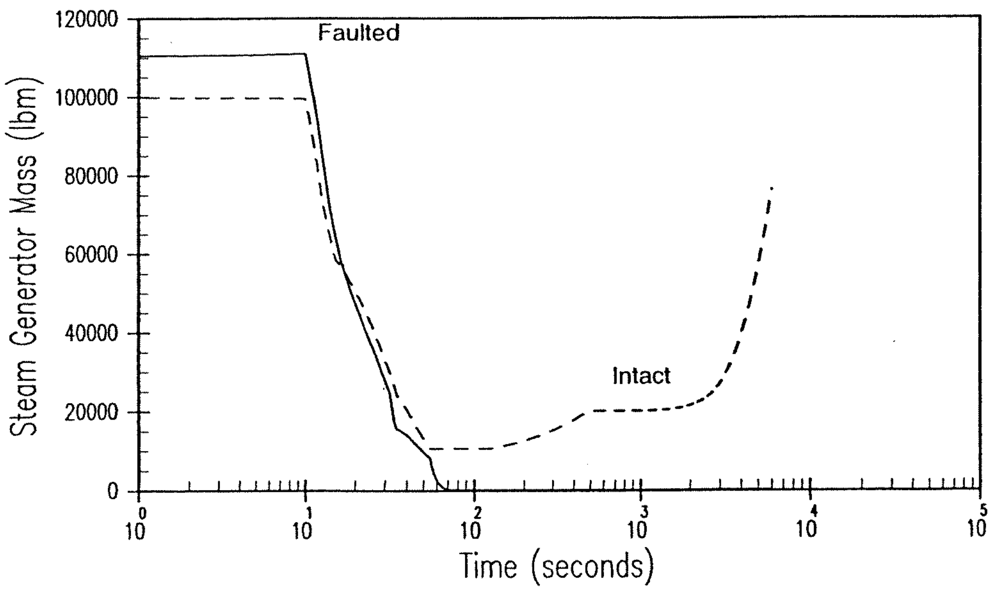
May 2002

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedwater Rupture Without Offsite Power
Faulted and Intact loop Coolant
Temperature vs. Time

Figure 15.4-89





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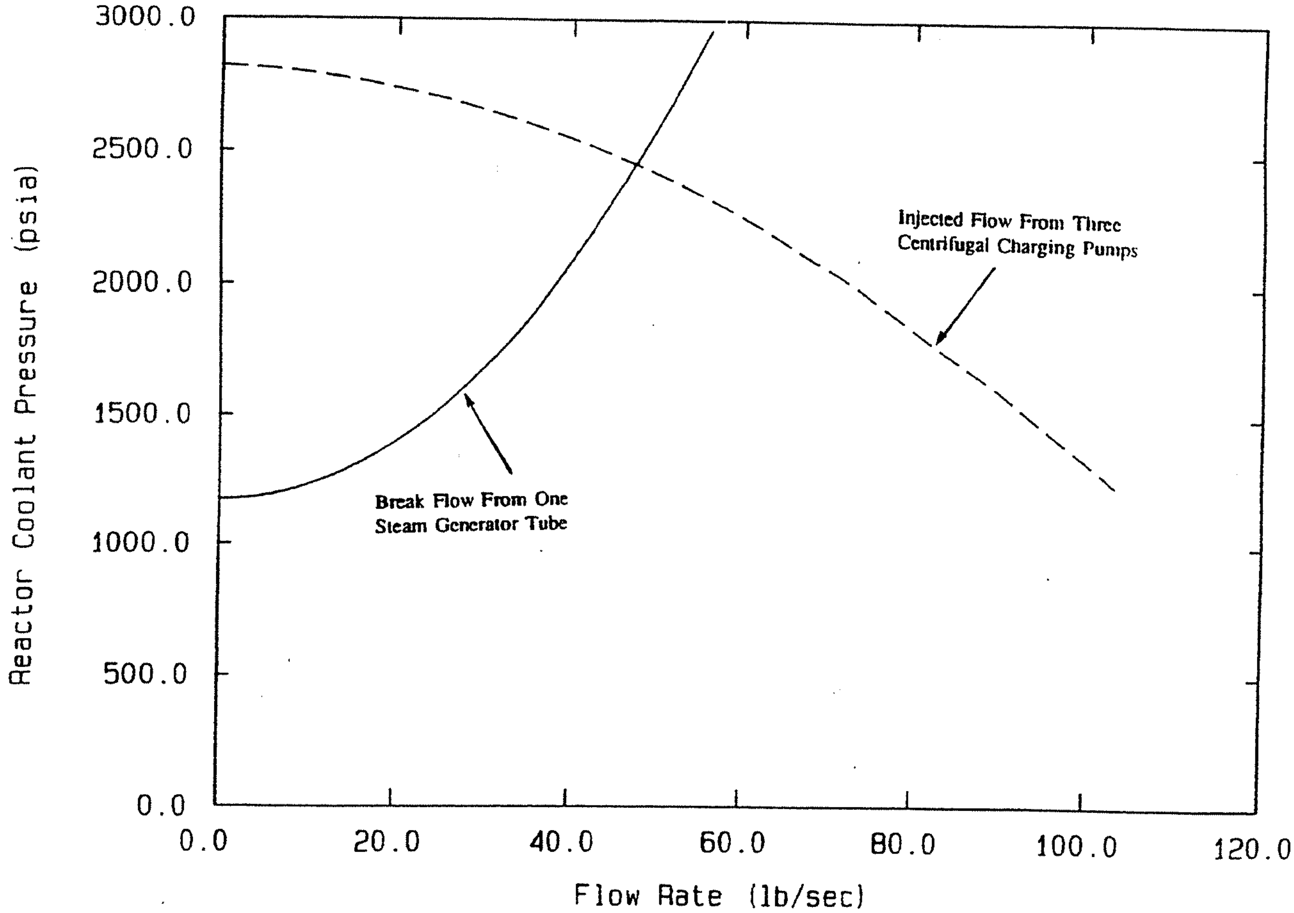
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedwater Rupture Without Offsite Power
Steam Generator Pressure and
Water Mass vs. Time

Figure 15.4-90

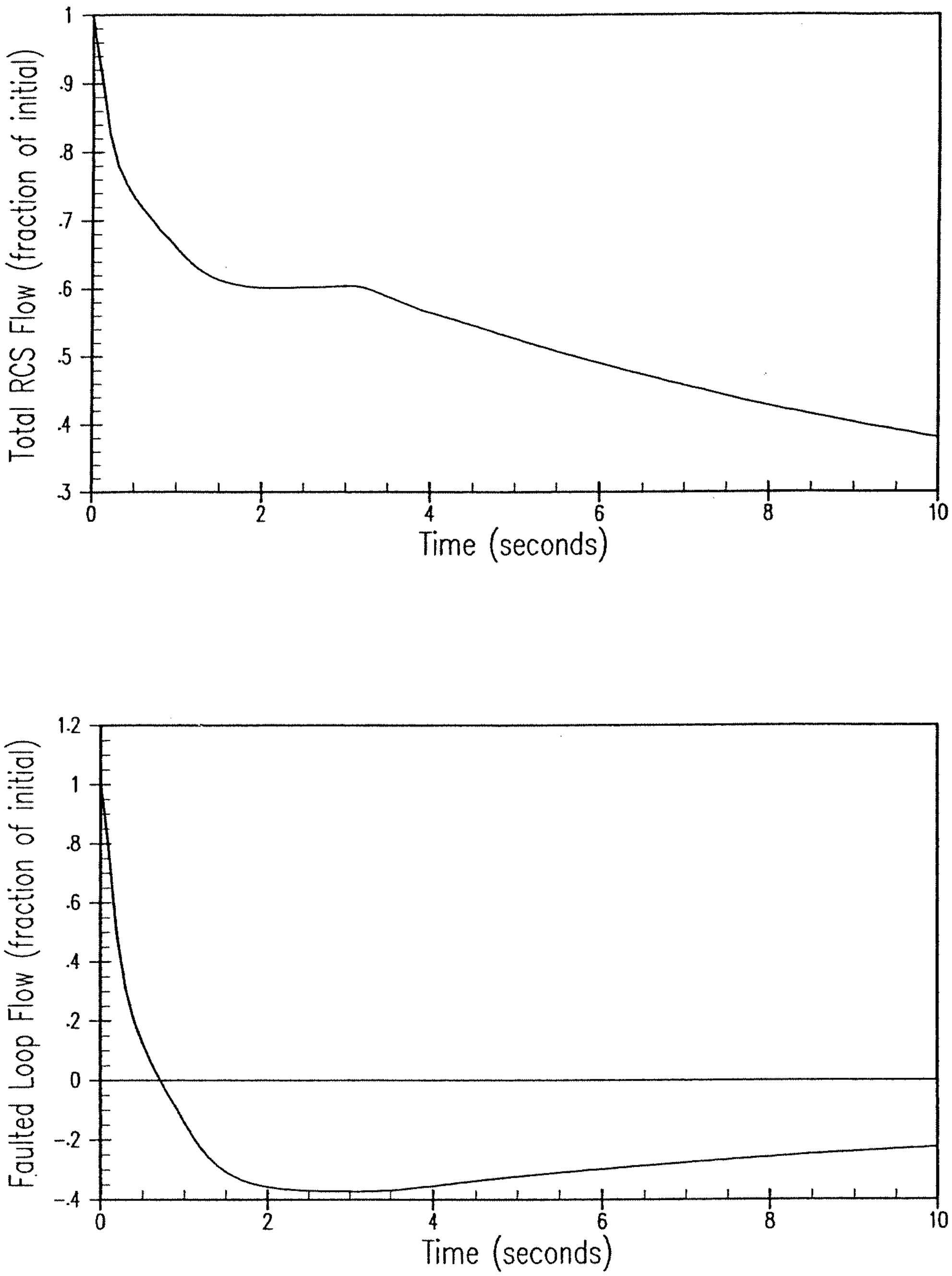


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Break Flow and Injected Mass Flow
Steam Generator Tube Rupture

Figure 15.4-91

Figures 15.4-92 Through 15.4-94
Have Been Deleted



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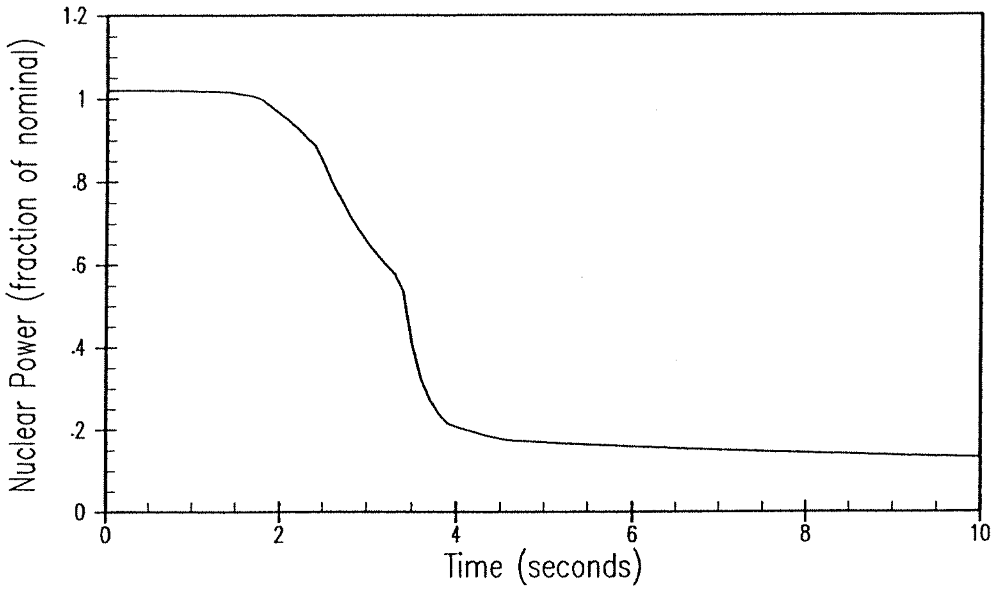
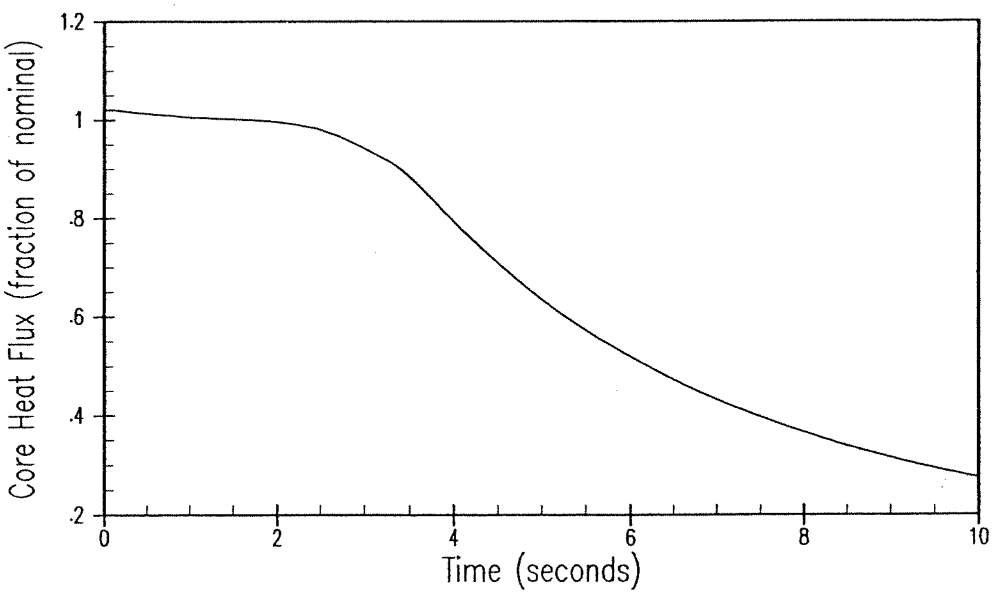
SOUTH CAROLINA ELECTRIC & GAS CO VIRGIL C. SUMMER NUCLEAR STATION
All Loops Operating, One Locked Rotor RCS Flow and Faulted Loop Flow vs. Time Figure 15.4-95





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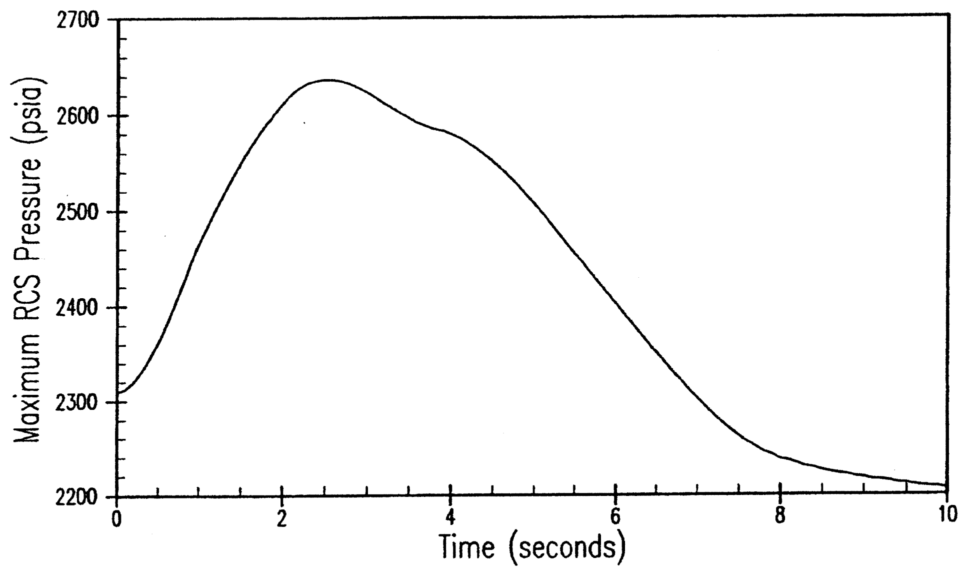
May 2002

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

All Loops Operating, One Locked Rotor
Nuclear Power and Core Heat Flux vs. Time

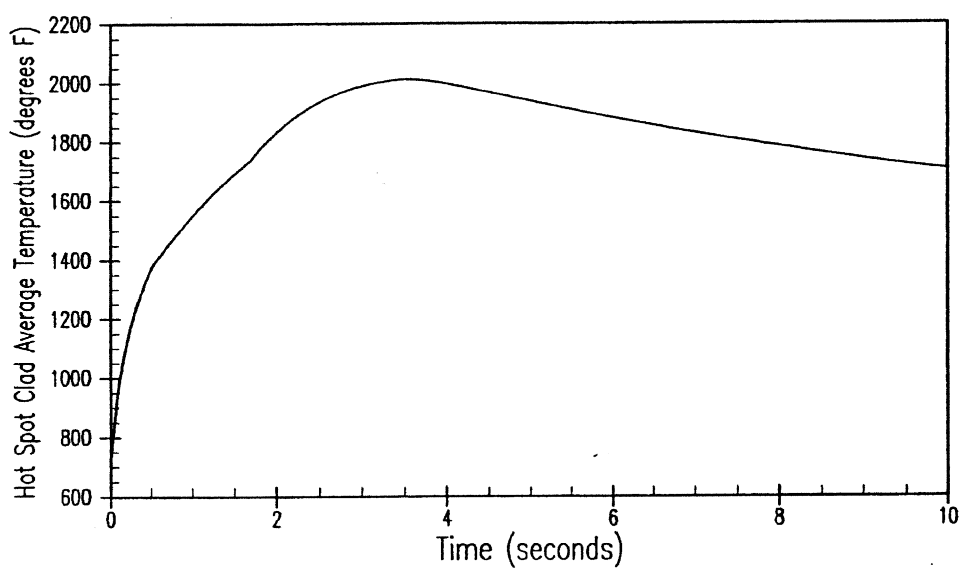
Figure 15.4-96A



RN 01-125

Peak pressure in this figure will increase less than 45 psi[55] when a more limiting, maximum initial pressurizer level is considered.

RN 09-017



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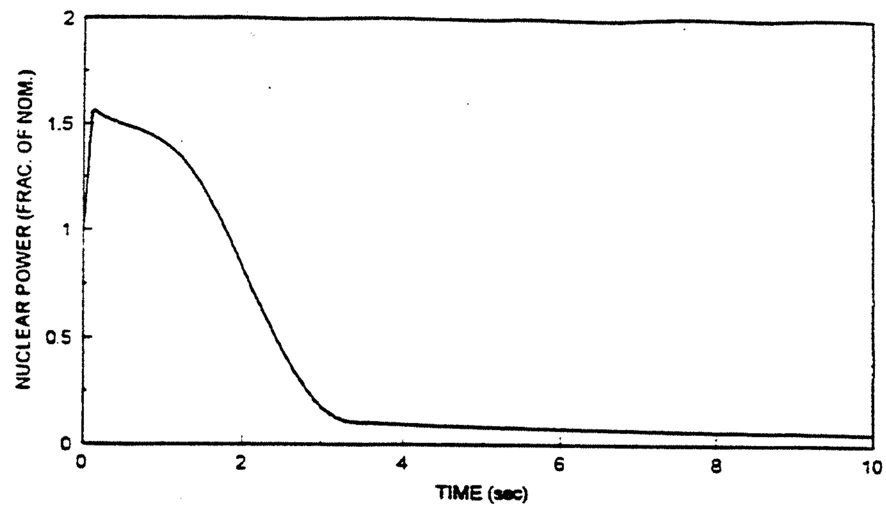
RN 09-017
August 2009

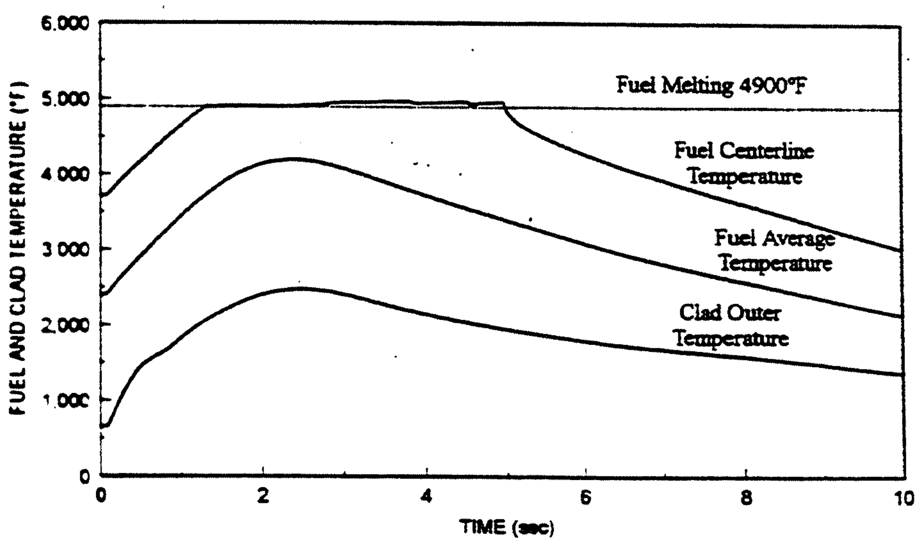
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

All Loops Operating, One Locked Rotor Maximum RCS Pressure and Hot Spot Clad Temperature vs. Time

Figure 15.4-96B

Figures 15.4-97 Through 15.4-99
Have Been Deleted





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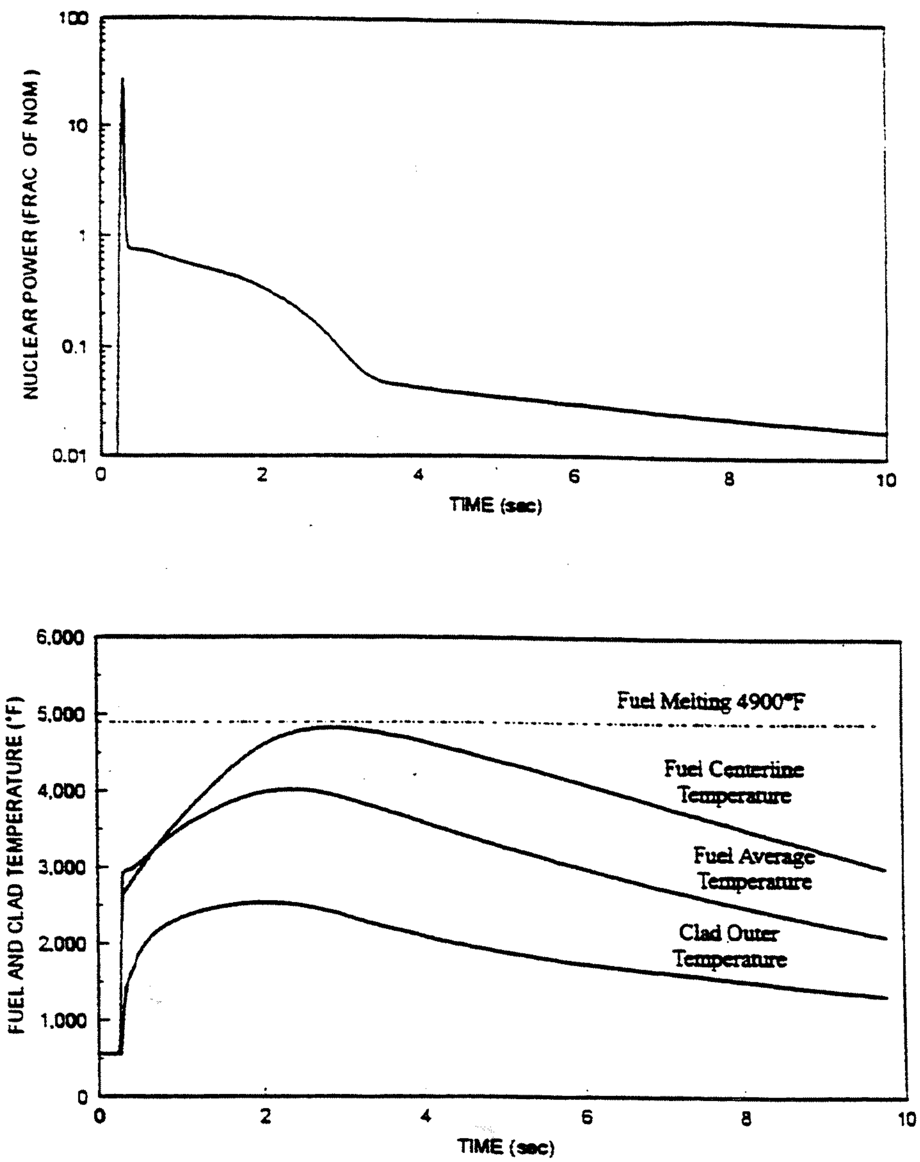
VIRGIL C. SUMMER NUCLEAR STATION

Rod Ejection Accident
BOL HFP
Nuclear Power, Hot Spot Fuel and
Clad Temperature vs. Time

Figure 15.4-100

Amendment 96-02
July 1996



Amendment 96-02
July 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Rod Ejection Accident
BOL HZP
Nuclear Power, Hot Spot Fuel and
Clad Temperature vs. Time

Figure 15.4-101

Figures 15.4-102 Through 15.4-104
Have Been Deleted

Figures 15.4-105 Through 15.4-122
Deleted per RN 06-040 June 2009

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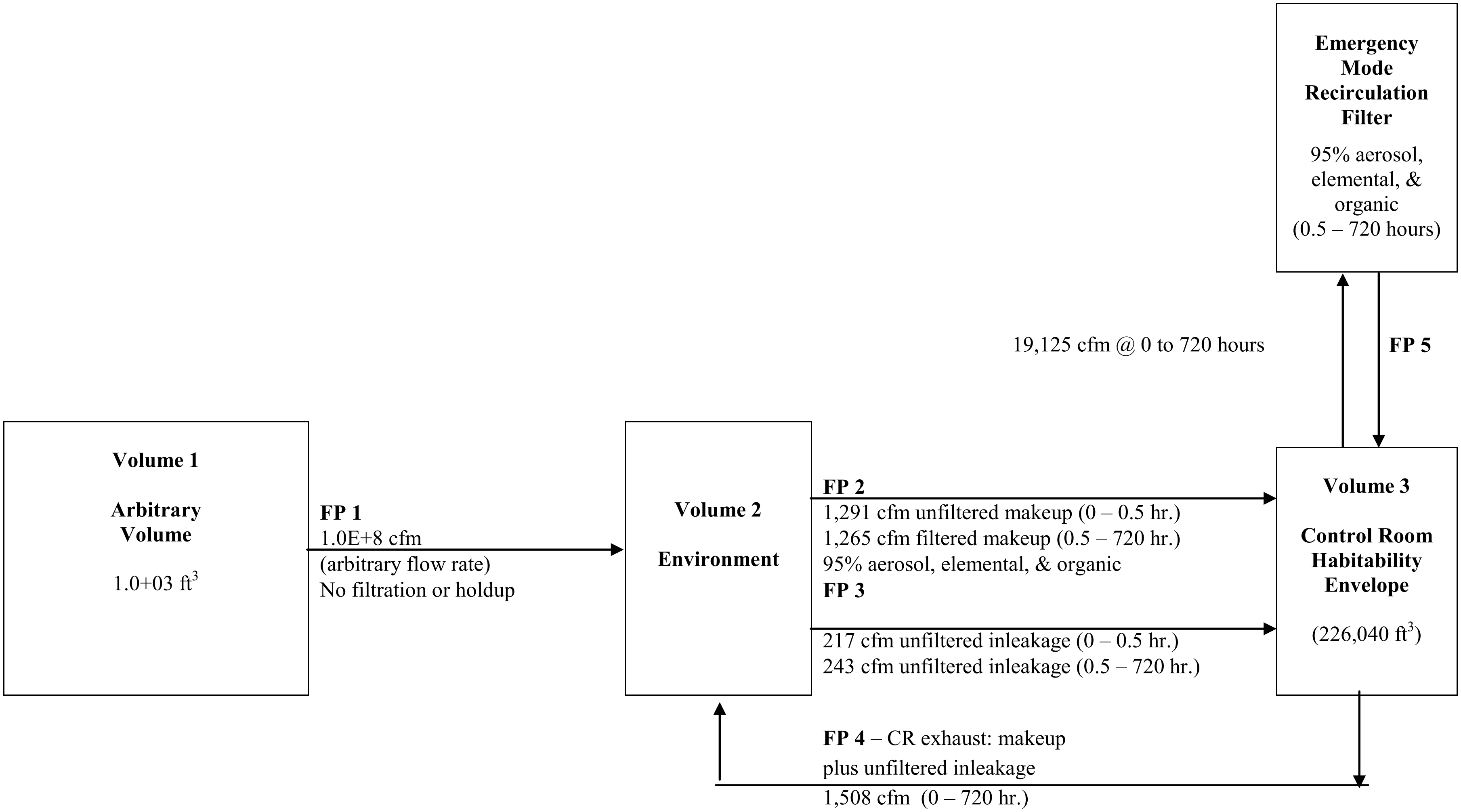
06-040

Figures 15.4-123 Through 15.4-142
Deleted per Amendment 96-02

Figure 15.4-143
RADTRAD Model for MSLB



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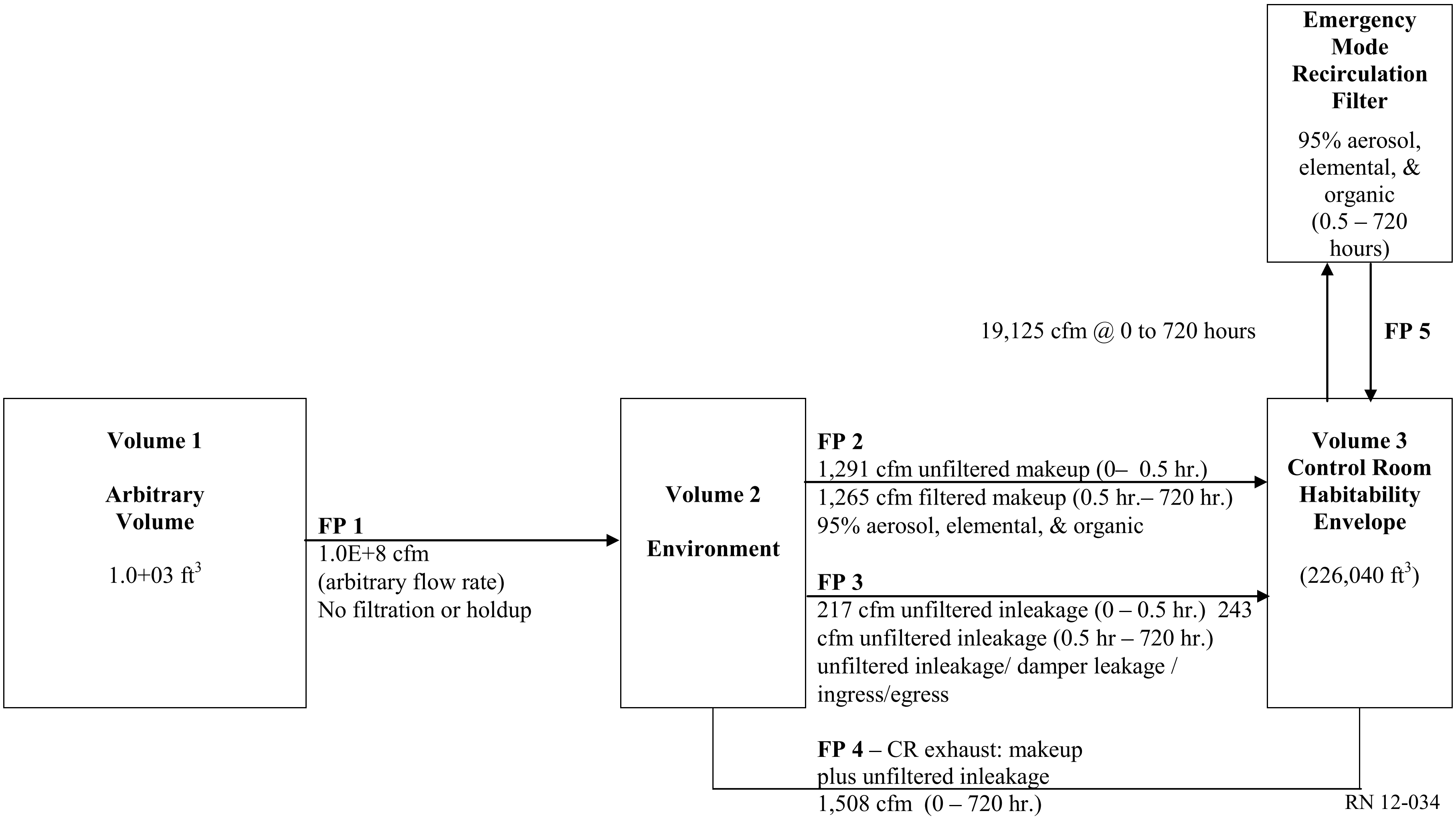
RADTRAD Model for MSLB Figure 15.4-143

Figure 15.4-144
RADTRAD Model for SGTR



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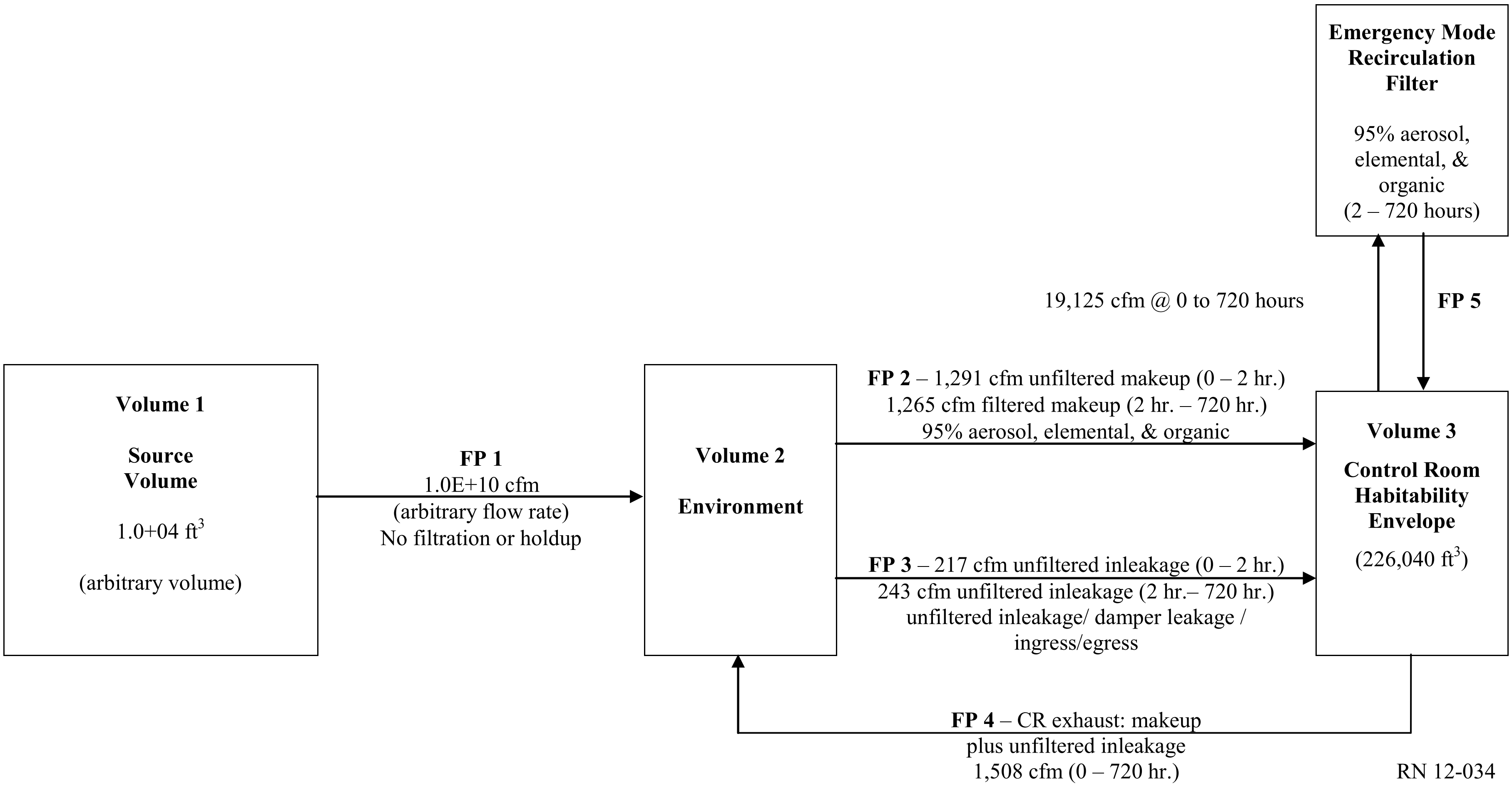
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RADTRAD Model for SGTR

Figure 15.4-144

Figure 15.4-145
RADTRAD Model for RCPLRA



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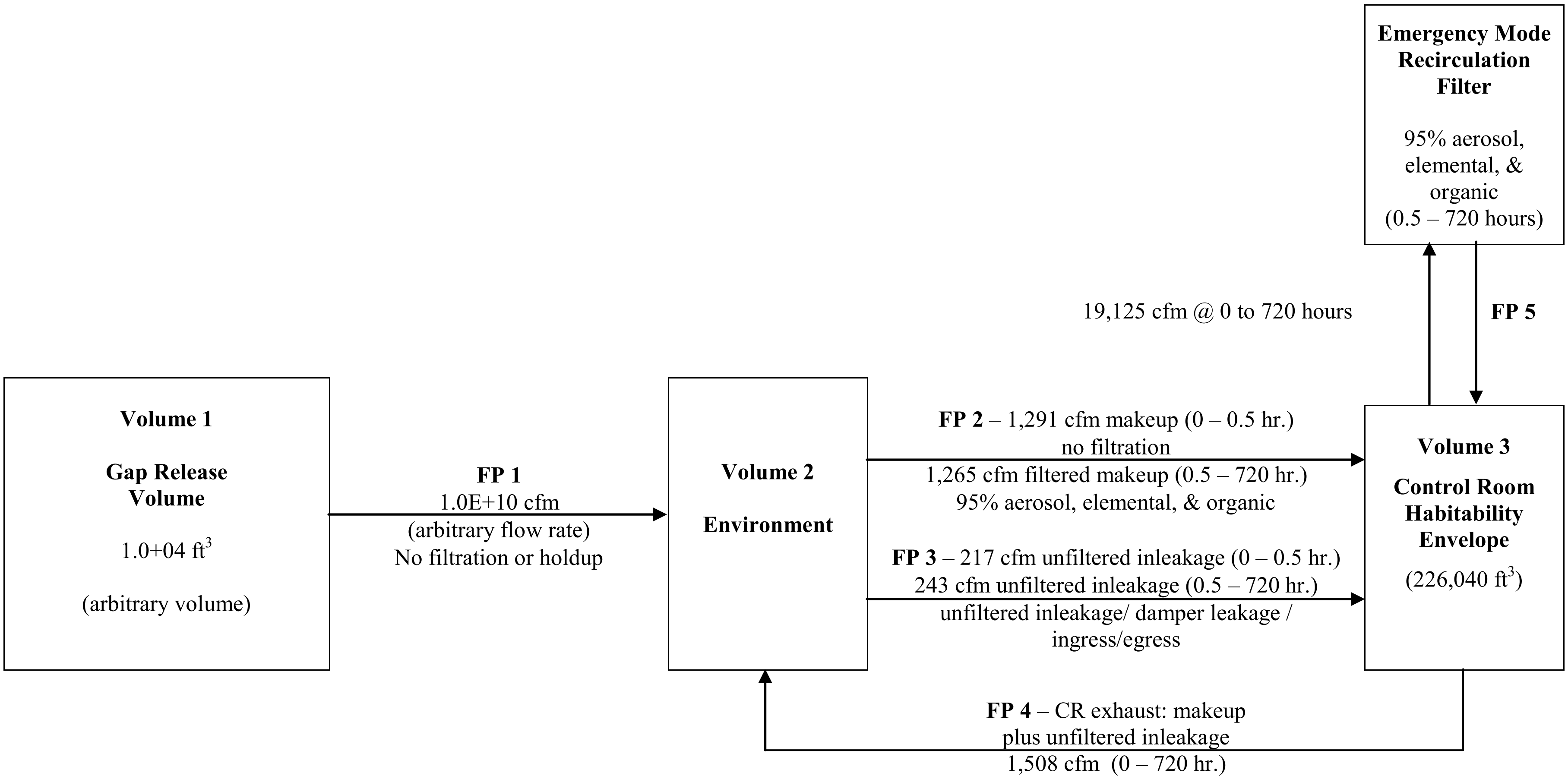
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RADTRAD Model for RCPLRA

Figure 15.4-145

Figure 15.4-146

RADTRAD Model for Inside/Outside Containment FHA



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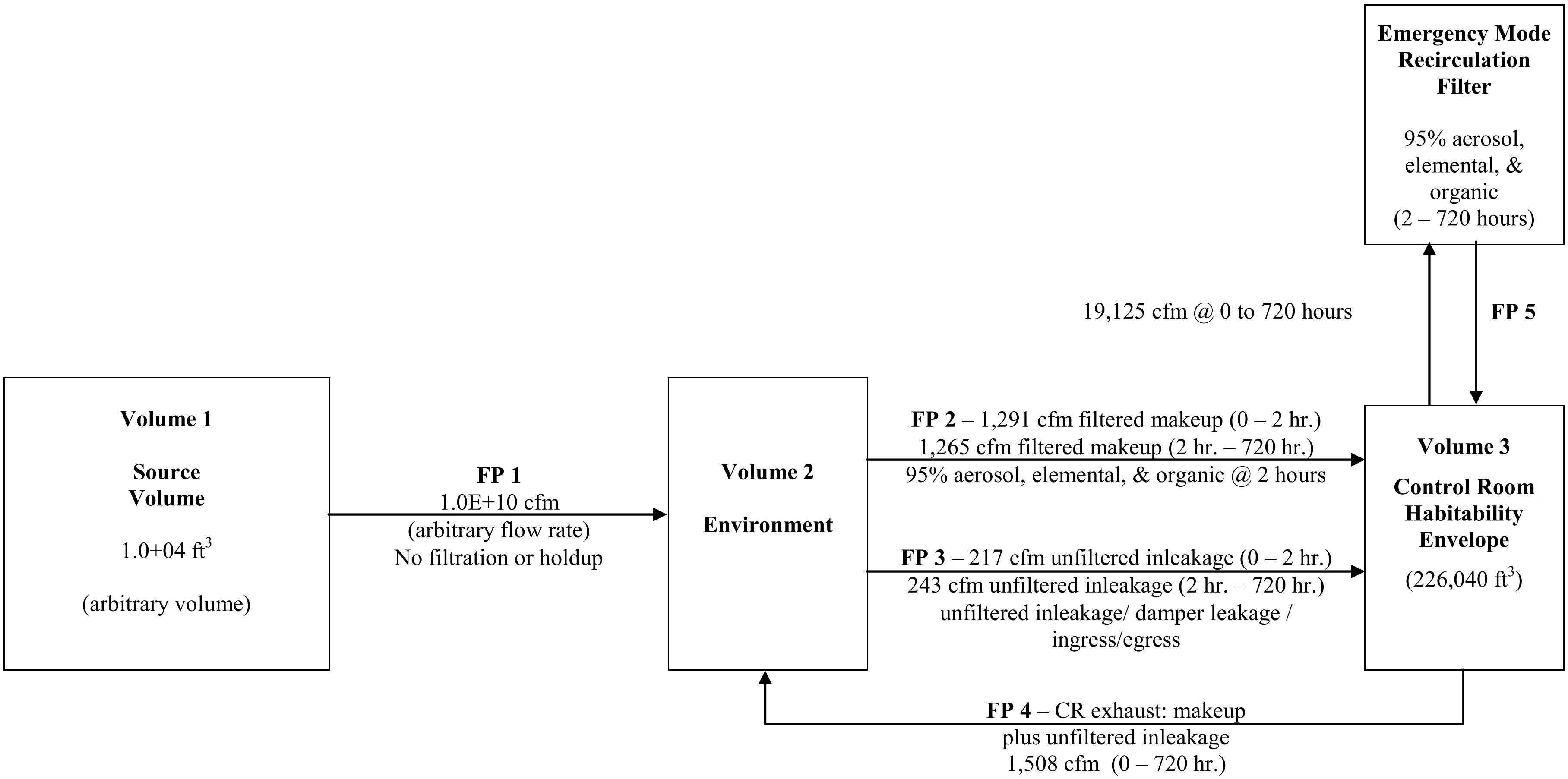
VIRGIL C. SUMMER NUCLEAR STATION

RADTRAD Model for Inside/Outside Containment FHA

Figure 15.4-146

Figure 15.4-147
RADTRAD Model for CREA (Steam Generator Release)



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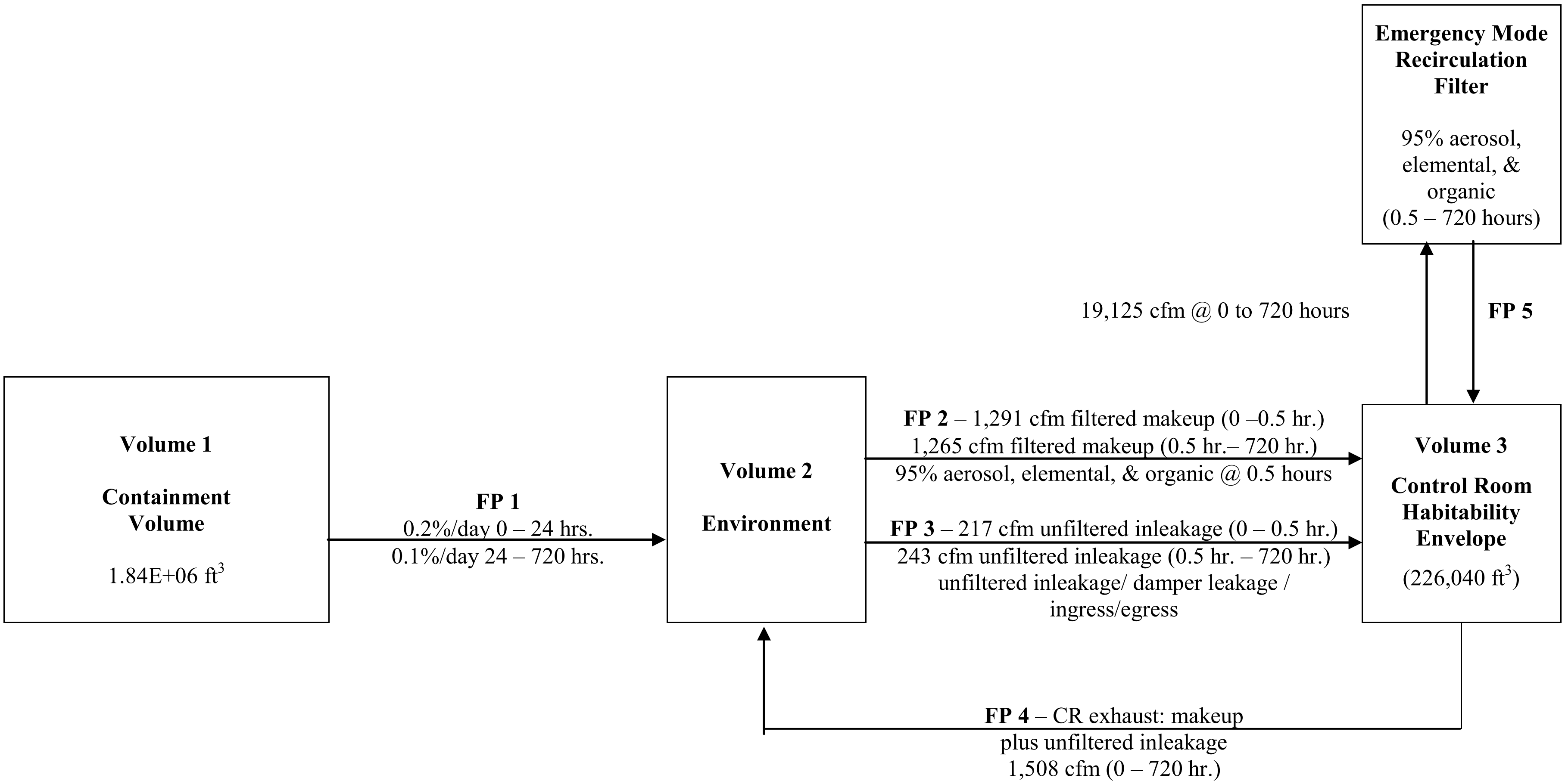
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RADTRAD Model for CREA
(Steam Generator Release)

Figure 15.4-147

Figure 15.4-148
RADTRAD Model for CREA (Containment Release)



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SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

RADTRAD Model for CREA (Containment Release)

Figure 15.4-148

15A DOSE MODELS USED TO EVALUATE THE ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15A.1 INTRODUCTION

This appendix identifies the models used to calculate offsite radiological doses that would result from releases of radioactivity due to various postulated accidents. The postulated accidents using this Appendix are as follows: RN 12-034

1. Loss of Offsite Power.

2. CVCS Line Rupture.

The following accidents are evaluated using the methods and models based on USNRC Regulatory Guide 1.183:

1. Loss of Coolant Accident.

2. Steam Line Break. RN 12-034

3. Steam Generator Tube Rupture.

4. Fuel Handling Accident.

5. Rod Ejection Accident.

6. Reactor Coolant Pump Locked Rotor Accident.

15A.2 ASSUMPTIONS

The following assumptions are basic to both the model for the gamma and beta 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) [1].

4. Isotopic data, such as decay rates and decay energy emissions, are taken from Table of Isotopes[2] and Kocher [6].

5. No correction is made for depletion of the effluent plume due to deposition on the ground or for the radiological decay in transit.

6. The receptor is located at a point on or beyond the site boundary where the maximum ground level concentration is expected to occur.

15A.3 GAMMA DOSE AND BETA SKIN DOSE

External whole body doses are calculated using "infinite cloud" assumptions (i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel). Such a cloud is considered an infinite cloud for a receptor at the center because any additional gamma and beta emitting material beyond the cloud dimensions does not alter the flux of gamma rays and beta particles to the receptor. Under these conditions, the rate of energy absorption per unit volume is equal to the rate of energy release per unit volume. The dose at any distance from the reactor is calculated based on the maximum ground level concentration at that distance. The infinite cloud methods for calculating whole body beta and gamma doses are as follows:[3]

1. For the infinite uniform cloud containing χ Ci of beta radioactivity per cubic meter, the beta dose rate in air at the cloud center is:

$$ {}_{\beta}D'_{\infty} = 0.457 E_{\beta} \chi $$

Where:

${}_{\beta}D'_{\infty}$ = Beta dose rate from an infinite cloud (rem/sec).

E_{β} = Average beta energy per disintegration (Mev/dis).

χ = Concentration of beta emitting isotope in the cloud (Ci/m^3).

Because of the limited range of beta particles in tissue, the surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount or:

$$ {}_{\beta}D'_{\infty} = 0.23 E_{\beta} \chi $$

Thus, in terms of integrated dose, the total beta dose to an individual located at the center of the cloud path is approximated as:

$_{\beta}D_{\infty}$ = $0.23E_{\beta}A\chi/Q$

Where:

$_{\beta}D_{\infty}$ = Total beta dose to an individual located at the center of the cloud path (rem).

E_{β} = Average beta energy per disintegration (Mev/ dis).

A = Isotopic activity released (Ci).

χ/Q = Atmospheric diffusion factor at receptor location (sec/m^3). RN 12-034

2. For an infinite uniform cloud containing χ curies of gamma radioactivity per cubic meter, the gamma dose rate in tissue at the cloud center is:

$_{\gamma}D'_{\infty}$ = $0.507E_{\gamma}\chi$

Where:

$_{\gamma}D'_{\infty}$ = Gamma dose rate from an infinite cloud (rem/sec).

E_{γ} = Average gamma energy per disintegration (Mev/dis).

χ = Concentration of gamma emitting isotope in the cloud (Ci/m^3).

However, because of the presence of the ground, the receptor is assumed to be exposed to only one-half of the cloud (semi-infinite) and the equation becomes:

$_{\gamma}D'$ = $0.25E_{\gamma}\chi$

Where:

$_{\gamma}D'$ = Gamma dose rate from a semi-infinite cloud (rem/sec).

Thus, in terms of integrated dose, the total gamma dose to an individual located at the center of the cloud path is approximated at:

$_{\gamma}D$ = $0.25E_{\gamma}A\chi/Q$

Where:

$_{\gamma}D$ = Total gamma dose to an individual located at the center of the cloud path (rem).

E_{γ} = Average gamma energy per disintegration (Mev/dis)

A = Isotopic activity released (Ci).

χ/Q = Atmospheric diffusion factor at receptor location (sec/m^3).

15A.4 THYROID INHALATION DOSE

The integrated inhalation dose to the thyroid from a release of activity is approximated by use of the following equation[5]:

$_{I}D$ = $AKBR\chi/Q$

Where:

$_{I}D$ = Inhalation dose to the thyroid (rem).

A = Isotopic activity released (Ci).

K = Adult thyroid dose conversion factor for the isotope of interest (rem/Ci).

BR = Breathing rate (m^3/sec)

χ/Q = Atmospheric diffusion rate at receptor location (sec/m^3).

The isotopic data and "standard man" data are given in Table 15A-2. The atmospheric dilution factors can be used in the analysis of the environmental consequences of accidents are given in Table 15A-3.

15A.5 REFERENCES

1. "Permissible Dose for Internal Radiation," ICRP Publication 2, Report of Committee II, 1959.

2. Lederer, C. M., et al., "Table of Isotopes," Sixth Edition, 1968.

3. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," USNRC, Revision 2, June, 1974.

4. Deleted by RN 12-034. RN 12-034

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

6. Kocher, ORNL/NUREG/TM-102, August 1977.

7. Deleted by RN 12-034. RN 12-034

TABLE 15A-2

PHYSICAL DATA FOR ISOTOPES

RN 00-056

Isotope	Decay (1) Constant (sec^{-1})	Average (1) Gamma Energy (Mev/dis)	Average (1) Beta Energy (Mev/dis)	Thyroid Dose Conversion Factor (Rem/Curie)	
I-131	9.96×10^{-7}	0.3708	0.202	1.48×10^{6} (2)	RN 12-034
I-132	8.26×10^{-5}	2.337	0.706	5.35×10^{4} (2)	
I-133	9.20×10^{-6}	0.477	0.423	4.00×10^{5} (2)	
I-134	2.20×10^{-4}	1.85	0.81	2.50×10^{4} (2)	
I-135	2.86×10^{-5}	1.77	0.47	1.24×10^{5} (2)	
Kr-83m	1.04×10^{-4}	0.00081	-	-	
Kr-85	2.04×10^{-9}	0.0021	0.223	-	
Kr-85m	4.41×10^{-5}	0.1507	0.273	-	
Kr-87	1.48×10^{-4}	1.374	1.27	-	
Kr-88	6.95×10^{-5}	1.744	0.933	-	
Kr-89	3.63×10^{-3}	1.9 (3)	1.33	-	00-01
Xe-131m	6.80×10^{-7}	0.0033	-	-	
Xe-133	1.52×10^{-6}	0.03	0.115	-	00-01
Xe-133m	3.49×10^{-6}	0.0326	-	-	
Xe-135	2.11×10^{-5}	0.246	0.307	-	
Xe-135m	7.40×10^{-4}	0.422	-	-	
Xe-137	2.96×10^{-3}	0.1502	1.37		00-01
Xe-138	6.60×10^{-4}	1.127 (3)	0.80	-	

Breathing Rates

Time Period (Hours)	Breathing Rates (M^3/sec)
0-8	3.47×10^{-4}
8-24	1.75×10^{-4}
24-720	2.32×10^{-4}

(1) See Reference [2]
(2) See Reference [5]
(3) See Reference [6] 00-01

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TABLE 15A-3

ACCIDENT ATMOSPHERIC DILUTION FACTORS

Averaging Period (Hours)	(sec/m^3) 1 Mile	(sec/m^3) 3 Miles
1	4.08×10^{-4}	1.01×10^{-4}
8	8.43×10^{-5}	2.37×10^{-5}
16	1.34×10^{-5}	2.44×10^{-6}
72	6.12×10^{-6}	1.11×10^{-6}
624	3.52×10^{-6}	6.28×10^{-7}

02-01