15.0 GENERAL

15.0.1 Classification of Plant Conditions

Since 1970 the American Nuclear Society (ANS) classification of plant conditions has been used 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:

- Condition I: Normal Operation and Operational Transients
- Condition II: Faults of Moderate Frequency
- Condition III: Infrequent Faults
- Condition IV: Limiting Faults.

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

15.0.1.1 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. Inasmuch as Condition I occurrences happen 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 adverse conditions which can occur during Condition I operation.

A typical list of Condition I events is identified below:

a. Steady state and shutdown operations:

1. Power operation (> 5 to 100 percent of rated thermal power)
2. Startup ($K_{eff} \geq 0\text{-}99$, ≤ 5 percent of rated thermal power)
3. Hot standby (subcritical, residual heat removal system isolated)
4. Hot shutdown (subcritical, Residual Heat Removal System in operation)
5. Cold shutdown (subcritical, Residual Heat Removal System in operation)
6. Refueling.

b. Operation with permissible deviations:

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

1. Operation with components or systems out of service
2. Leakage from fuel with clad defects
3. Radioactivity in the reactor coolant
   (a) Fission products
   (b) Corrosion products
   (c) Tritium
4. Operation with steam generator leaks up to the maximum allowed by the Technical Specifications
5. Testing as allowed by the Technical Specifications.

c. Operational transients:

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

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

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

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

15.0.1.2 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 events. In addition, Condition II events are not expected to result in fuel rod failures or Reactor Coolant System or secondary system overpressurization.

For the purposes of this report, the following faults are included in this category:

a. Feedwater System malfunction causing a decrease in feedwater temperature (Subsection 15.1.1) or an increase in feedwater flow (Subsection 15.1.2)

b. Excessive increase in secondary steam flow (Subsection 15.1.3)

c. Accidental depressurization of the Main Steam System (Subsection 15.1.4)

d. Loss of external load (Subsection 15.2.2)

e. Turbine trip (Subsection 15.2.3)

f. Inadvertent closure of main steam isolation valves (Subsection 15.2.4)

g. Loss of condenser vacuum and other events resulting in turbine trip (Subsection 15.2.5)

h. Loss of nonemergency AC power to the station auxiliaries (Subsection 15.2.6)

i. Loss of normal feedwater flow (Subsection 15.2.7)

j. Partial loss of forced reactor coolant flow (Subsection 15.3.1)

k. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (Subsection 15.4.1)

l. Uncontrolled rod cluster control assembly bank withdrawal at power (Subsection 15.4.2)

m. Control rod misalignment - Dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly (Subsection 15.4.3)

n. Startup of an inactive reactor coolant pump at an incorrect temperature (Subsection 15.4.4)

o. Chemical and Volume Control System (CVCS) malfunction that results in a decrease in the boron concentration in the reactor coolant (Subsection 15.4.6)

p. Inadvertent operation of the Emergency Core Cooling System during power operation (Subsection 15.5.1)

q. CVCS malfunction causing an increase in reactor coolant inventory (Subsection 15.5.2)

r. Inadvertent opening of a pressurizer safety or relief valve (Subsection 15.6.1)

s. Failure of small lines outside Containment (Subsection 15.6.2).

15.0.1.3 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 plant 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 are included in this category:

a. Minor steam system piping failure (Subsection 15.1.5)

b. Complete loss of forced reactor coolant flow (Subsection 15.3.2)

c. Control rod misalignment - single Rod Cluster Control Assembly withdrawal at full power (Subsection 15.4.3)

d. Inadvertent loading and operation of a fuel assembly in an improper position (Subsection 15.4.7)

e. Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuate the Emergency Core Cooling System (Subsection 15.6.5)

f. Waste Gas System failure (Subsection 15.7.1)

g. Radioactive Liquid Waste System leak or failure (atmospheric release) (Subsection 15.7.2)

h. Liquid containing tank failure (Subsection 15.7.3)

i. Spent fuel cask drop accidents (Subsection 15.7.5). [Historical]

15.0.1.4 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 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and the Containment. For the purposes of this report the following faults have been classified in this category:

a. Steam system piping failure (Subsection 15.1.5)

b. Feedwater system pipe break (Subsection 15.2.8)

c. Reactor coolant pump shaft seizure (locked rotor) (Subsections 15.3.3 and 15.3.4)

d. Reactor coolant pump shaft break (Subsection 15.3.5)

e. Spectrum of Rod Cluster Control Assembly ejection accidents (Subsection 15.4.8).

f. Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary (Subsection 15.6.5)

g. Steam generator tube rupture (Subsection 15.6.3)

h. Fuel handling accidents (Subsection 15.7.4).

15.0.2 Optimization of Control Systems

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

Nominal protection system setpoints on which the accident analysis is based are also used in the control system setpoint study. Instrumentation errors are calculated consistent with the method used in the accident analysis. These errors are applied in an adverse direction with respect to maintaining system stability and transient performance. The accident analysis and setpoint study combine to show that the plant can be operated and meet both safety and operability requirements.

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 various levels of power operation.

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

15.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.0.3.1 Design Plant Conditions

Table 15.0-1 lists the principal power rating values which are assumed in analyses performed in this report.

Where initial power operating conditions are assumed in accident analyses, the "guaranteed nuclear steam supply system thermal power output" plus allowance for errors in steady state power determination is assumed. The thermal power values used for each transient analyzed are given in Table 15.0-3.

The values of other pertinent plant parameters utilized in the accident analyses are given in Table 15.0-2.

15.0.3.2 Initial Conditions

Table 15.0-2 and Table 15.0-3 provides a list of conditions representing nominal plant parameters. These parameters also represent a set of initial conditions for the accidents and transients. Uncertainties in these parameters are accounted for either through RTDP or in the initial conditions selected for the transient cases. The following uncertainties are considered:

a.	Core power [+]	0 percent allowance for calorimetric error
b.	Average RCS temperature	± 3.0°F random with a –3.0°F bias[*] allowance for controller deadband and measurement error and steam generator fouling penalty
c.	Pressurizer pressure	±50 psi allowance for steady-state fluctuations and measurement error

[+] Analysis performed at a reactor thermal power of 3659 MWt +0% calorimetric uncertainty. This value reflects the licensed power of 3648 MWt +0.3% uncertainty starting in Cycle 12.

[*] A negative bias means that the indication is lower than actual.

15.0.3.3 Power Distribution

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

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

15.0.3.4 Component Response Times and Capacities

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

15.0.3.5 Non-LOCA Accidents

This section summarizes the non-LOCA analyses and evaluations performed to support the SPU program at Seabrook Unit 1.

15.0.3.5.1 Fuel Features

The fuel features which were evaluated are:

a. $ZIRLO_{TM}$ Intermediate Flow Mixing (IFMs) grids;

b. $ZIRLO_{TM}$ mid grids;

c. $ZIRLO_{TM}$ fuel clad;

d. $ZIRLO_{TM}$ instrument and thimble tubes;

e. Removable top nozzles;

f. Protective bottom grids;

g. Debris filter bottom nozzles

15.0.3.5.2 Other Major Assumptions

a. An NSSS power level of 3678 MWt

b. A reactor thermal power of 3659 MWt

c. A reactor coolant system Thermal Design Flow (TDF) of 93,600 gpm/loop

d. A reactor coolant system Minimum Measured Flow (MMF) of 95,950 gpm/loop

e. An average vessel average coolant temperature of between 571.0°F and 589.1°F

f. An average reactor coolant pressure of 2250 psia

g. An average steam generator tube plugging (SGTP) of 10%

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

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

a. For reactor power, a 0%

b. For average RCS temperature, a $\pm$ 3.0°F random and –3.0°F bias*

c. For pressurizer pressure, a $\pm$ 50 psi

* A negative bias means that the indication is lower than actual.

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

15.0.3.5.3 Overtemperature-ΔT and Overpower-ΔT

The overtemperature-ΔT and overpower-ΔT setpoints were recalculated for the power uprate program based on the most conservative core limits. The core limits used to calculate the OTΔT/OPΔT setpoints are provided in the COLR. All of the FSAR events which rely on OTΔT and OPΔT for protection were analyzed to reflect the setpoint changes, as provided in the COLR. It has been confirmed that these OTΔT and OPΔT setpoints protect the core safety limits as shown in Figure 15.0-1.

15.0.3.5.4 RCCA Reactivity Characteristics

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

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

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

15.0.4 Reactivity Coefficients Assumed in the Accident Analyses

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

Maximum and minimum integrated DPCs assumed in the safety analyses are provided in Figure 15.0-2. The formulas for calculating the DPCs used are $[(.034Q^2)\ -19.4Q]\times10^{-5}$ for the maximum and $[(.0175Q^2)-9.55Q]\times10^{-5}$ for the minimum, where Q is the power level. Note that Steamline Break Core Response uses a different DPC based on a stuck RCCA.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. The values used for each accident are given in Table 15.0-3. Conservative combinations of parameters are used for each event selected on a case-by-case basis.

15.0.5 Rod Cluster Control Assembly Insertion Characteristics

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

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

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

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

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

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. Opening either trip breaker initiates a turbine trip. The loss of power to the mechanism coils causes the mechanisms to release the Rod Cluster control Assemblies, which then fall by gravity into the core. There are various delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached at the sensor to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the total time delay assumed for each trip function are given in Table 15.0-4. The Overtemperature ΔT trip functions are illustrated in Figure 15.0-1.

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

In the analysis of the Chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident.

15.0.7 Instrumentation Drift and Calorimetric Errors - Power Range Neutron Flux

Instrumentation drift and calorimetric errors are considered when establishing the power range high neutron flux setpoint.

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

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

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 of the postulated accidents. 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 15.0-5 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, the classification system of ANSI N18.2-1973 is utilized. 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.

In the analysis of the Chapter 15 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case.

A functional analysis of the plant systems, in response to the various accidents, has been conducted. Results of the analysis, in the form of protection sequence diagrams, are presented in Figure 15.0-7, Figure 15.0-8, Figure 15.0-9, Figure 15.0-10, Figure 15.0-11, Figure 15.0-12, Figure 15.0-13, Figure 15.0-14, Figure 15.0-15, Figure 15.0-16, Figure 15.0-17, Figure 15.0-18, Figure 15.0-19, Figure 15.0-20, Figure 15.0-21, Figure 15.0-22, Figure 15.0-23, Figure 15.0-24, Figure 15.0-25, Figure 15.0-26, Figure 15.0-27, Figure 15.0-28, Figure 15.0-29, Figure 15.0-30, and Figure 15.0-31.

15.0.8.1 Effects of Operator Actions

For most of the events analyzed in Chapter 15, the plant will be in a safe and stable hot standby condition following the automatic actuation of reactor trip. This condition will in fact be similar to plant conditions following any normal, orderly shutdown of the reactor. At this point, the actions taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time these actions would occur, will depend on what systems are available (e.g., Steam Dump System, Main Feedwater System, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam generators. The Main Feedwater System and the Steam Dump or Atmospheric Relief System could be used for this purpose. Alternatively, the Emergency Feedwater System and the steam generator safety valves may be used, both of which are safety grade systems. Although the Emergency Feed System may be started manually, it will be automatically actuated if needed by one of the signals shown on Figure 7.2-1, Sheet 15, such as low-low steam generator water level. If hot standby conditions are maintained for an extended period of time, operator action may be required to transfer the emergency feedwater source. The time at which such action is required will be sufficiently long after initiation of the event to permit operator action. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 hours), operator action may be required to add boric acid via the CVCS to compensate for the xenon decay and maintain shutdown margin. Again, the actions taken by the operator would be no different than during normal plant shutdown.

For several events involving breaks in the Reactor Coolant System or secondary system piping, additional requirements for operator action can be identified. (Additional information about the impact of equipment failures or erroneous operator actions may be found in WCAP-9691, "NUREG-0578 2.1.9.C, Transient and Accident Analysis," Reference 14.

15.0.9 Fission Product Inventories

15.0.9.1 Activities in the Core

The Alternate Source Term (AST) for activities in the core is provided in Appendix 15C.

15.0.9.2 Activities in the Fuel Pellet Cladding Gap

The Alternate Source Term (AST) for activities in the Fuel Pellet Cladding Gap is provided in Table 15C-4.

15.0.9.3 Activities in the Secondary Side Coolant

The Alternate Source Term (AST) for activities in the Secondary Side Coolant is provided in Table 15C-3.

15.0.10 Residual Decay Heat

15.0.10.1 Total Residual Heat

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

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

During a loss-of-coolant accident the core is rapidly shut down by void formation or Rod Cluster Control Assembly insertion, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady-state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady-state factor of 97.4 percent which represents the fraction of heat generated within the clad and pellet drops to 95 percent for the hot rod in a loss-of-coolant accident.

For example, consider the transient resulting from the postulated double-ended break of the largest reactor coolant system pipe; 1/2 second after the rupture about 30 percent of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady-state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10 percent of the gamma ray contribution or 3 percent of the total. Since the water density is considerably reduced at this time, an average of 98 percent of the available heat is deposited in the fuel rods, the remaining 2 percent being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of 0.95 rather than 0.974, to be applied to the heat production in the hot rod.

15.0.11 Computer Codes Utilized

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

15.0.11.1 LOFTRAN

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

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

15.0.11.2 RETRAN-02

The RETRAN-02 program is used for studies of transient response of a PWR system to specified perturbations in process parameters. RETRAN-02 simulates a multi-loop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides) and the pressurizer. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics and reactivity effects of the moderator, fuel, boron and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The Reactor Protection System is simulated to include reactor trips on high neutron flux, Overtemperature ΔT. Overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure and level control. The Emergency Feedwater and Emergency Core Cooling System (except accumulators) are also modeled.

RETRAN-02 is a versatile program which is suited to both accident evaluation and control studies, as well as parameter sizing.

RETRAN-02 is further discussed in Reference 10.

15.0.11.3 SIMULATE-3 and CASMO-3

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

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

SIMULATE-3 is further described in Reference 11.

15.0.11.4 VIPRE

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

15.0.11.5 FACTRAN

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

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

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

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

15.0.11.6 TWINKLE

The TWINKLE[17] program is a multi-dimensional spatial neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2,000 spatial points and performs steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. The code provides various output, e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

15.0.11.7 ANC

ANC[20] is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

15.0.12 Radiological Consequences

Radiological consequences have been calculated for each hypothetical accident which can potentially result in radioactivity releases in excess of those expected to be experienced during normal plant operating conditions. In general, two hour TEDE doses are presented at the 914 meter site exclusion area boundary and duration of accident doses for the outer boundary of the low-population zone (2012 meters). Parameters and assumptions used to evaluate the radiological consequences are presented in the following discussions of each hypothetical accident, and are summarized in Appendix 15C.

The physical and mathematical models used in calculating radioactivity source terms are discussed in Section 11.1. Core fission products (halogens and noble gases) used to calculate accident doses are given in Appendix 15C.

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

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

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

15.0.13 References

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

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15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of events have been postulated which could result in an increase in heat removal from the Reactor Coolant System by the secondary system. Detailed analyses are presented for several such events which have been identified as limiting cases.

Discussions of the following reactor coolant system cooldown events are presented in this section:

a. Feedwater system malfunction causing a reduction in feedwater temperature

b. Feedwater system malfunction causing an increase in feedwater flow

c. Excessive increase in secondary steam flow

d. Inadvertent opening of a steam generator relief or safety valve

e. Steam system piping failure.

The above are considered to be ANS Condition II events, with the exception of a major steam system pipe break, which is considered to be an ANS Condition IV event. Subsection 15.0.1 contains a discussion of ANS classification and applicable acceptance criteria.

15.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

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

A reduction in feedwater temperature may be caused by the accidental opening of a feedwater bypass valve which diverts flow around a portion of the feedwater heaters. In the event of an accidental opening of a bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. At power, this increased subcooling will create a greater load demand on the RCS.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient of reactivity. However, the rate of energy change is reduced as load and feedwater flow decrease, so the no-load transient is less severe than the full power case. The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow, i.e., the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT.

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

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive steam flow increase, as discussed in Subsection 15.0.8 and listed in Table 15.0-5.

15.1.1.2 Analysis of Effects and Consequences

a. Method of Analysis

This transient is analyzed by computing conditions at the feedwater pump inlet following opening of the heater bypass valve. These feedwater conditions are then used to recalculate a heat balance through the high pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet.

The following assumptions are made:

1. Plant initial power level corresponding to guaranteed NSSS thermal output.
2. Low pressure heater bypass valve opens, resulting in condensate flow splitting between the bypass line and the low pressure heaters; the flow through each path is proportional to the pressure drops.
3. Heater drain pumps trip; this increases the effect of the cold bypass flow.

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

b. Results

Opening of a low pressure heater bypass valve and trip of the heater drain pumps cause a reduction in feedwater temperature which increases the thermal load on the primary system. The calculated reduction in feedwater temperature is less than 35°F, resulting in an increase in heat load on the primary system of less than 10 percent of full power. The increased thermal load, due to opening of the low pressure heater bypass valve, thus would result in a transient very similar (but of reduced magnitude) to that presented in Subsection 15.1.5 for a steam system piping failure initiated at full power conditions. Therefore, the transient results of this analysis are not presented.

15.1.1.3 Radiological Consequences

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

15.1.1.4 Conclusions

The decrease in feedwater temperature transient is less severe than the increase in feedwater flow event (Subsection 15.1.2), and the steam system piping failure initiated at full power conditions (Subsection 15.1.5). Based on results presented in Subsections 15.1.2 and 15.1.5, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

15.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

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

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

Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which activates the feedwater isolation. Pre-trip alarm of high steam generator level is available in the control room.

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

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

15.1.2.2 Analysis of Effects and Consequences

a. Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer RETRAN[1] code. This code simulates the neutron kinetics of the reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate acceptable consequences in the event of an excessive feedwater addition, due to a control system malfunction or operator error which allows a feedwater control valve to open fully. Three cases are analyzed as follows:

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

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

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

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

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

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

c. For the zero load condition, feedwater temperature is at a conservatively low value of 100°F.

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

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

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

b. <u>Results</u>

The calculated sequence of events for this accident is shown in Table 15.1-1.

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

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

Transient results for the full power and zero power cases are provided in Figure 15.1-1. The DNBR does not fall below the limit value. Following reactor trip (full power cases), the plant approaches a stabilized condition; standard plant shutdown procedures may then be followed to further cool down the plant.

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

The transient results have shown that the DNBR does not go below the limit value at any time during the excessive feedwater flow incident; thus, the ability of the primary coolant to remove heat from the fuel rod is not reduced.

15.1.2.3 Radiological Consequences

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

15.1.2.4 Conclusions.

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

15.1.3 <u>Excessive Increase in Secondary Steam Flow</u>

15.1.3.1 Identification of Causes and Accident Description

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

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

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

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

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

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

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

15.1.3.2 Analysis of Effects and Consequences

a. Method of Analysis

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

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

A conservative limit on the turbine valve opening was assumed, and all cases were analyzed without credit being taken for pressurizer heaters.

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

Normal reactor control systems and engineered safety systems were not required to function for this event. The reactor protection system was assumed to be operable; however, reactor trip was not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure would prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control were analyzed to ensure that the worst case with respect to minimum DNBR is presented. The automatic rod control function is not required for core protection.

Given the non-limiting nature of this event with respect to the DNBR safety analysis criterion, an explicit analysis was not performed as part of the Power Uprate Program. Instead, a detailed evaluation of this event was performed. The evaluation model consists of the generation of statepoints based on generic conservative data. The statepoints are in the form of changes to the initial conditions and then applied to the actual operating conditions of the plant. The statepoints are then compared to the core thermal limits to ensure that the DNBR limit is not violated. Four cases are evaluated for both manual rod control and automatic rod control. These cases are:

- Reactor in manual rod control with BOL (minimum moderator) reactivity feedback.
- Reactor in manual rod control with EOL (maximum moderator) reactivity feedback.
- Reactor in automatoc rod control with BOL (minimum moderator) reactivity feedback.
- Reactor in automatic rod control with EOL (minimum moderator) reactivity feedback.

15.1.3.3 **Results and Conclusions**

An evaluation of this event was performed to support the Power Uprate Program. The evaluation determined that the DNB design basis for a 10% step load increase continues to be met.

15.1.4 **Inadvertent Opening of a Steam Generator Relief or Safety Valve**

15.1.4.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the Main Steam System are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed assuming a rupture of a main steam line are given in Subsection 15.1.5.

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

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

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

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

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

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

   1. Two out of four pressurizer pressure signals

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

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

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

c. Redundant isolation of the main feedwater lines.

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

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

1. High-2 containment pressure

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

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

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

15.1.4.2 Analysis of Effects and Consequences

a. Method of Analysis

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

b. Results

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

15.1.4.3 Radiological Consequences

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

15.1.4.4 Conclusions

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

15.1.5 Steam System Piping Failure

15.1.5.1 Identification of Causes and Accident Description

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

The limiting steam line break presented in this section corresponds to a double-ended rupture of the main steam line at the steam generator nozzle at zero power with offsite power available.

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

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

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

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

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events.

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

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

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

   1. Two out of four low pressurizer pressure signals

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

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

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

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

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

   1. High-2 containment pressure

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

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

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

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

15.1.5.2 Analysis of Effects and Consequences

a. Method of Analysis

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

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

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

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

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

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

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

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

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

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

When offsite power is assumed, the sequence of events in the safety injection system is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 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 into core before the 2,400 ppm borated water from the refueling water storage tank reaches the core. This delay, described above, is inherently included in the modeling.

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

5. Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, would have the same effect on the NSSS as the 1.4 ft^2 break.

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

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

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

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

7. In computing the steam flow during a steam line break, the Moody curve[15] for fL/D = 0 is used.
8. Emergency Feedwater flow is limited by passive flow restrictors to protect the pumps against a runout condition during main steam line rupture.

The following cases have been considered in determining the core power and RCS transients:

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

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

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

b. Results

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

1. Core Power and Reactor Coolant System Transient

Figure 15.1-6, sheets 1 through 6, show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition (case a).

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast-acting isolation valves in the steam lines, by high containment pressure signals, or low steam line pressure. Even with the failure of one isolation valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steam line isolation valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in sheets 1 through 6 of Figure 15.1-6, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2,400 ppm enters the RCS. A peak core power less than the nominal full power value is attained.

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

Once the pressure in the RCS falls below the pressure in the accumulators, boron solution at 2,300 ppm also enters the RCS from the accumulators.

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

2. Margin to Critical Heat Flux

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

15.1.5.3 Radiological Consequences Using Alternate Source Term Methodology

a. Background

This event consists of a double-ended break of one main steam line outside of containment. The radiological consequences of such an accident bound those of a MSLB inside containment. The affected steam generator (SG) rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cool down is achieved via the remaining unaffected SGs.

b. Compliance with RG 1.183 Regulatory Positions

The MSLB dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," as discussed below:

1. Regulatory Position 1 – No fuel damage is postulated to occur for the Seabrook MSLB event.

2. Regulatory Position 2 – No fuel damage is postulated to occur for the Seabrook MSLB event. Two cases of iodine spiking are evaluated.

3. Regulatory Position 2.1 – One iodine spiking case assumes a reactor transient prior to the postulated MSLB that raises the primary coolant iodine concentration to the maximum allowed by Tech Specs, which is a value of 60.0 μCi/gm DE I-131 for the analyzed conditions. This is the pre-accident spike case.

4. Regulatory Position 2.2 – One case assumes the transient associated with the MSLB causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Tech Spec value of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

5. Regulatory Position 3 – The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

6. Regulatory Position 4 – Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 – The primary-to-secondary leak rate is apportioned between the SGs as specified by Tech Specs (1.0 gpm total, 500 gpd to any one SG). The tube leakage is conservatively apportioned as 500 gpd to the faulted SG and 940 gpd total to the other three SGs in order to maximize dose consequences.

8. Regulatory Position 5.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS. For the intact Steam Generators, the primary to secondary leak rate is based on a density of 1.0 gm/cc (cold liquid).

9. Regulatory Position 5.3 – The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 48 hours. The release of radioactivity from the unaffected SGs is assumed to continue until the steam release is terminated due to RHR initiation at 8 hours.

10. Regulatory Position 5.4 – All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

11. Regulatory Position 5.5.1 – In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the unaffected steam generators used for plant cooldown, tube bundle uncovery is not postulated; therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

12. Regulatory Position 5.5.2 – Tube bundle uncovery is not postulated for the unaffected SGs; therefore, this section does not apply. In the faulted SG, all of the fluid is assumed to flash and be released without mitigation.

13. Regulatory Position 5.5.3 – All leakage that does not immediately flash is assumed to mix with the bulk water.

14. Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%. No reduction in the release is assumed from the faulted SG.

15. Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for the intact SGs for Seabrook.

c. <u>Other Assumptions</u>

1. RG 1.183 does not address secondary coolant activity. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the Tech Spec limit of 0.1 μCi/gm Dose Equivalent I-131.

2. The steam mass release rates for the intact SGs are provided in Table 15.1-3.

3. This evaluation assumes that the RCS mass remains constant throughout the MSLB event (no change in the RCS mass is assumed as a result of the MSLB or from the safety injection system).

4. The SG secondary side mass in the unaffected SGs is assumed to remain constant throughout the event.

5. Releases from the faulted main steam line (and associated SG) are postulated to occur from the main steam line associated with the most limiting atmospheric dispersion factors. Releases from the unaffected SGs are postulated to occur from the MSSV or ASDV with the most limiting atmospheric dispersion factors.

d. Methodology

Input assumptions used in the dose consequence analysis of the MSLB are provided in Table 15.1-2. The postulated accident assumes a double-ended break of one main steam line outside containment. The radiological consequences of such an accident bound those of a MSLB inside of containment. Upon a MSLB, the affected SG rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cooldown is achieved via the remaining unaffected SGs.

The analysis assumes that the entire fluid inventory from the affected SG is immediately released to the environment. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 μCi/gm DE I-131 permitted by Tech Specs. Primary coolant is also released into the affected steam generator by leakage across the SG tubes based on the Tech Spec primary to secondary leakage limits. Activity is released to the environment from the affected steam generator, as a result of the postulated primary-to-secondary leakage and the postulated activity levels of the primary and secondary coolants, until the affected steam generator is completely isolated at 48 hours (primary system temperature less than 212°F). Additional activity, based on the Tech Spec primary-to-secondary leakage limits (SG tube leakage), is released via the unaffected SGs via steaming from the unaffected SGs MSSVs/ASDVs for 8 hours (time of RHR initiation). These release assumptions are consistent with the requirements of RG 1.183.

Fuel damage is not postulated for the MSLB event. Consistent with Regulatory Guide 1.183, Appendix E, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity released is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike; and (2) maximum accident-induced or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated MSLB event. The primary coolant iodine concentration is increased to the maximum value of 60 μCi/gm DE I-131 permitted by Technical Specification 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 15.1-4.

For the case of the accident-induced spike, the postulated MSLB event induces an iodine spike. The RCS activity is initially assumed to be 1.0 μCi/gm DE I-131 as allowed by Tech Specs. Iodine is released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 15.1-6.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially, the ventilation system is assumed to be operating in normal mode. The air intake to the Control Room during this mode is 1000 cfm of unfiltered fresh air.

- After the start of the event, the Control Room normal air intake is isolated on a CR intake radiation monitor signal. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time, load sequencing and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution consists of 600 cfm of filtered makeup flow through the worst of the two emergency intakes, 150 cfm of unfiltered inleakge and 390 cfm of filtered recirculation flow.

- 20 cfm of unfiltered inleakage was assumed to enter the Control Room via the CR fire exit and 130 cfm was assumed to enter via the Diesel Building.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95% for elemental and organic iodine.

e. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room does are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 2R-2 and 2R-3.

Releases from the intact SGs are assumed to occur from the MSSV/ASDV that produces the most limiting *X/Qs*. Releases from the faulted SG are assumed to occur from the location on a steam line that produces the most limiting *X/Qs*.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Cases for MSLB pre-accident and concurrent iodine spikes are analyzed. As shown in Table 15.1-7, the results of both cases for EAB dose, LPZ dose, and Control Room dose are within the appropriate regulatory acceptance criteria.

15.1.6 References

1. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D.S. Huegel, et al., April 1999
2. WCAP-7908-A, "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," H. G. Hargrove, December 1989
3. WCAP-7979-A, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," D. H. Risher, and R. F. Barry, January 1975
4. WCAP-14565, "Vipre-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., April 1997
5. WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1984
6. WCAP-9226-P-A. "Reactor Core Response to Excessive Secondary Steam Releases," S. D. Hollingsworth, et al., January 1998
7. WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture," G. E. Lang and J. P. Cunningham, January 1978
8. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990
9. WCAP-8424, Revision 1, An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs," M. S. Baldwin, et al., June 1975
10. WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," D. H. Risher, January 1975

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

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

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

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

15. Journal of Heat Transfer, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," F.J. Moody, February 1965

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of transients and accidents have been postulated in this section which could result in a reduction of the capacity of the secondary system to remove heat generated in the Reactor Coolant System (RCS). Detailed analyses are presented for the most limiting of these events.

Discussions of the following RCS coolant heatup events are presented in this section:

a. Steam pressure regulator malfunction

b. Loss of external load

c. Turbine trip

d. Inadvertent closure of main steam isolation valves

e. Loss of condenser vacuum and other events resulting in turbine trip

f. Loss of nonemergency AC power to the station auxiliaries

g. Loss of normal feedwater flow

h. Feedwater system pipe break.

The above items are considered to be American Nuclear Society (ANS) Condition II events, with the exception of a feedwater system pipe break, which is considered to be an ANS Condition IV event. Subsection 15.0.1 contains a discussion of ANS classifications and applicable acceptance criteria.

15.2.1 Steam Pressure Regulator Malfunction Or Failure that Results In Decreasing Steam Flow

There are no steam pressure regulators in the Seabrook plant whose failure or malfunction could cause a steam flow transient.

15.2.2 Loss of External Load

15.2.2.1 Identification of Causes and Accident Description

A major plant load loss can result from the loss of external electrical load due to some electrical system disturbance. Offsite alternating current power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite emergency diesel generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. The plant would be expected to trip from the Reactor Protection System if a safety limit were approached. A continued steam load of approximately 5 percent would exist after total loss of external electrical load because of the steam demand of plant auxiliaries.

In the event that a safety limit is approached, protection would be provided by the high pressurizer pressure, Overtemperature ΔT and steam generator low-low water level trips. Voltage 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 percent, resulting in an overfrequency of less than 6 hertz (hz). This resulting overfrequency is not expected to damage the sensors (Nonnuclear Steam Supply System) in any way. However, it is noted that frequent testing of this equipment is required by the Technical Specifications. Any degradation in their performance could be ascertained at that time. Any increased frequency to the reactor coolant pump motors will result in slightly increased flow rate and subsequent additional margin to safety limits. Safeguards loads are supplied from offsite power or alternatively, from emergency diesels. Reactor protection system equipment is supplied from the 118 volt AC instrument Power Supply System, which in turn is supplied from the inverters; the inverters are supplied from a direct current bus energized from batteries or by a rectified AC voltage from safeguards buses.

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 steam generator low-low 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 RCS and steam generator against overpressure for all load losses without assuming the operation of the Steam Dump System, pressurizer spray, pressurizer power-operated relief valves, automatic rod cluster control or direct reactor trip on turbine trip.

The steam generator safety valve capacity is capable of removing the steam flow at 100 percent of the analyzed core power from the steam generator without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to relieve sufficient steam to maintain the RCS pressure within 110 percent of the RCS design pressure.

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

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

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

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

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

15.2.2.2 Analysis of Effects and Consequences

a. Method of Analysis

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

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

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

15.2.2.3 Radiological Consequences

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

15.2.2.4 Conclusions

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

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Accident Description

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

a. Electrical faults associated with the generator or transformers

b. Low condenser vacuum

c. Loss of lubricating oil

d. Turbine thrust bearing failure

e. Turbine overspeed

f. Main steam reheat high level

g. Manual trip.

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

The automatic Steam Dump System would normally accommodate the excess steam generation when the unit is operating below the P-9 setpoint. Reactor coolant temperatures and pressure do not significantly increase if the Steam Dump System and Pressurizer Pressure Control System are functioning properly.

If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the Auxiliary Feedwater System to ensure adequate residual and decay heat removal capability. Should the Steam Dump System fail to operate, the steam generator safety valves may lift to provide pressure control. See Subsection 15.2.2.1 for a further discussion of the transient.

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

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

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

15.2.3.2 Analysis of Effects and Consequences

a. Method of Analysis

In this analysis, two cases are analyzed. In one case, the behavior of the unit is evaluated for a complete loss of steam load from 100 percent or rated thermal power without direct reactor trip, primarily to show the adequacy of the pressure relieving devices to limit the maximum RCS pressure to 110 percent of its design value. The second case analyzes the accident with respect to determining the minimum DNBR. This second case typically also represents the limiting transient with respect to peak steam generator pressure because it usually results in a longer time to reactor trip. In both cases the turbine trip is assumed to trip without actuating any of the sensors for reactor trip on the turbine stop valves. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient.

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

The turbine trip transients are analyzed by employing the detailed digital computer program RETRAN[6]. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

The following turbine trip cases are analyzed:

A. Minimum reactivity feedback, with RCS pressure control

B. Minimum reactivity feedback, with no RCS pressure control

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

Major assumptions are summarized below:

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

   For Case B, an uncertainty of 50 psi is applied in the most limiting direction to the initial reactor coolant pressure. No uncertainty is applied to the core power as it is already accounted for in the conservatively high nominal core power level assumed. No uncertainty is applied to the nominal full-power value for reactor coolant temperature as this yields more conservative results.

2. Moderator and Doppler Coefficients of Reactivity - The turbine trip is analyzed with minimum reactivity feedback, which assumes a 0 pcm/°F moderator temperature coefficient and a least negative Doppler Power coefficient.

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

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

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

   a. For evaluating the minimum DNBR, full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Pressurizer safety valves are also available. This case results in a delayed reactor trip on overtemperature ΔT which results in a more limiting steam generator pressure transient.

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

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

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

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

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

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

b. Results

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

Figure 15.2-1, sh. 1, Figure 15.2-1, sh. 2, and Figure 15.2-1, sh. 3 show the transient responses for the turbine trip with minimum reactivity feedback assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip signal. The minimum DNBR remains well above the limit value. The steam generator safety valves limit the secondary side pressure below 110 percent of the design value.

The turbine trip accident was also studied assuming the plant to be initially operating at 100 percent of rated thermal power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figure 15.2-1, sh. 4 and Figure 15.2-1, sh. 5 show the transients with minimum reactivity feedback without pressure control. In this case, the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

15.2.3.3 Radiological Consequences

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

15.2.3.4 Conclusions

Results of the analyses show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the Reactor Protection System, i.e., the DNBR will be maintained above the safety analysis limit value. The above analysis demonstrates the ability of the Nuclear Steam Supply System to safely withstand a full load rejection.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

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

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

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

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

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

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

15.2.6.1 Identification of Causes and Accident Description

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

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

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

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

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

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

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

The Emergency Feedwater System is started automatically as described below.

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

a. Low-low level in any steam generator

b. Any safety injection signal (SIS)

c. Manual actuation.

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

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

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

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

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

When a loss of nonemergency AC power is the initiating event, the first few seconds of the transient will closely resemble the simulation of the complete loss of reactor coolant flow event (Section 15.3.2), where DNB and core damage due to rapidly increasing core temperature is prevented by promptly tripping the reactor. For the loss of nonemergency AC power scenario, the DNBR results would be less limiting since the reactor is already tripped when RCP coastdown begins. Thus, the DNBR is not evaluated for this event since it would be bounded by the loss of reactor coolant flow analysis.

In addition, the maximum RCS and main steam system (MSS) pressures for this event are bounded by the loss of external electrical load analysis, which demonstrates that the peak pressures remain below 110 percent of the respective design limit values. For the loss of nonemergency AC power event, turbine trip occurs after reactor trip, whereas for loss of external electrical load analysis the turbine trip is the initiating fault. Therefore, the primary/secondary power mismatch and resultant RCS and main steam system heatup and pressurization transients are always more severe for loss of external electrical load than for loss of nonemergency AC power.

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

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

The above assumptions are more conservative than an actual loss of nonemergency AC because the reactor power is maintained following the loss of AC and loss of feedwater. This minimizes the steam generator heat transfer capability and increases the amount of RCS stored energy at the time of reactor trip and loss of primary coolant flow.

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

15.2.6.2 Analysis of Effects and Consequences

a. Method of Analysis

A detailed analysis using the RETRAN computer code (Reference 6) is performed to determine the plant transient following a loss of nonemergency AC power to the plant auxiliaries. The code simulates the core neutron kinetics, reactor coolant system including natural circulation, pressurizer, pressurizer power operated relief valves and safety valves, pressurizer heaters and spray, steam generators, main steam safety valves, and the emergency feedwater system, and computes pertinent variables, including pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

The assumptions used in the analysis are as follows:

1. The plant is initially operating at an NSSS power level of 3678 MWt.
2. Core residual heat is based on the 1979 version of ANS 5.1[(4)]. ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long term operation at the initial power level preceding the reactor trip is assumed.
3. Reactor trip occurs on steam generator low-low level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.
4. Emergency feedwater at a temperature of 100°F is delivered by one emergency feedwater pump. A total flow of 650 gpm is assumed to be delivered equally to all four steam generators 77 seconds after the steam generator low-low level setpoint is reached.
5. Secondary system steam relief is achieved through the steam generator safety valves.
6. The initial pressurizer pressure is assumed to be 50 psi higher and lower than the nominal value to determine the limiting case.

7. Cases are analyzed assuming initial hot full power reactor vessel average coolant temperatures at the upper and lower ends of the operating range with uncertainty applied in both the positive and negative direction. The vessel average temperature assumed at the upper end of the range is 589.1°F with an uncertainty of +6/-5 °F. The average temperature assumed at the lower end of the range is 571.0°F with an uncertainty of +6/-5 °F. Results for the limiting case are presented.

8. A moderator temperature coefficient of 0 pcm/°F, the least negative Doppler temperature coefficient, and the most negative Doppler-only power were assumed for conservatism.

9. Cases are analyzed assuming initial feedwater temperatures of 452.4°F and 390°F.

10. Analysis with both minimum (0%) and maximum (10%) steam generator tube plugging was performed to conservatively bound potential operating conditions.

11. The pressurizer relief valves, sprays, and heaters are assumed to function.

b. Results

The transient responses of the RCS and the secondary side following a loss of nonemergency AC power are shown in Figure 15.2-5 sheets 1 through 4.

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

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

As noted previously, the DNBR result for this event is bounded by the complete loss of flow event, and the maximum RCS and MSS pressures for this event are bounded by the loss of external electrical load analysis. The sole acceptance criterion for this analysis is that the pressurizer does not become water solid. Figure 15.2-5, sheet 2 demonstrates that this criterion is met. The calculated sequence of events for this accident is listed in Table 15.2-1.

15.2.6.3 Radiological Consequences

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

15.2.6.4 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. The radiological consequences of this event would be less severe than the steam line break event analyzed in Subsection 15.1.5.3.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description

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

The following events occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

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

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

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

Reactor trip on low-low water level in any steam generator provides protection for a loss of normal feedwater.

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

The DNBR consequences of this event are bounded by the Loss of load/turbine trip (LOL/TT) event. Both of these events represent a reduction in the heat removal capability of the secondary system. For the loss of normal feedwater event, the RCS temperature increases gradually as the steam generators boil down to the low-low level trip setpoint, at which time reactor trip occurs, followed by turbine trip. For the LOL/TT event, the turbine trip is the initiating event, and the loss of heat sink is much more severe. Therefore, the initial RCS heatup will be much more severe for the LOL/TT event than for the loss of normal feedwater event, and the LOL/TT event will always be more severe with respect to the minimum DNBR criterion.

With respect to system overpressure concerns, the loss of normal feedwater event is also bounded by the LOL/TT event analysis (minimum reactivity feedback, without pressure control). For the loss of normal feedwater event, turbine trip occurs after reactor trip, whereas for LOL/TT the turbine trip is the initiating fault. Therefore, the primary/secondary power mismatch and resultant RCS and main steam system heatup and pressurization transients are always more severe for LOL/TT than for loss of normal feedwater.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the Emergency Feedwater System is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition. This is demonstrated by showing that the pressurizer does not become water solid.

15.2.7.2 Analysis of Effects and Consequences

a. Method of Analysis

A detailed analysis using the RETRAN computer code (Reference 6) is performed to determine the plant transient following a loss of normal feedwater. The code simulates the core neutron kinetics, reactor coolant system, pressurizer, pressurizer power operated relief valves and safety valves, pressurizer heaters and spray, steam generators, main steam safety valves, and the emergency feedwater system, and computes pertinent variables, including pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. The plant is initially operating at an NSSS power level of 3678 MWt.
2. Core residual heat is based on the 1979 version of ANS 5.1[4]. ANSI/ANS-5.1-1979 is a conservative representation of decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.
3. Reactor trip occurs on steam generator low-low level.
4. Emergency feedwater at a temperature of 100°F is delivered by one emergency feed pump. A total flow of 650 gpm is assumed to be delivered equally to all four steam generators 77 seconds after the steam generator low-low level setpoint is reached.
5. Secondary system steam relief is achieved through the steam generator safety valves.
6. The initial pressurizer pressure is assumed to be 50 psi higher and lower than the nominal value to determine the limiting case.
7. Cases are analyzed assuming initial hot full power reactor vessel average coolant temperatures at the upper and lower ends of the operating range with uncertainty applied in both the positive and negative direction. The vessel average temperature assumed at the upper end of the range is 589.1°F with an uncertainty of +6/-5 °F. The average temperature assumed at the lower end of the range is 571.0°F with an uncertainty of +6/-5 °F. Results for the limiting case are presented.
8. A moderator temperature coefficient of 0 pcm/°F, the least negative Doppler temperature coefficient, and the most negative Doppler-only power were assumed for conservatism.
9. Cases are analyzed assuming initial feedwater temperatures of 452.4°F and 390°F.
10. Analysis with both minimum (0%) and maximum (10%) steam generator tube plugging was performed to conservatively bound potential operating conditions.
11. The pressurizer relief valves, sprays, and heaters are assumed to function.

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

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

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

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

b. Results

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

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Within 75 seconds following the initiation of the low-low level trip, the emergency feedwater pumps are automatically started, reducing the rate of water level decrease. The capacity of the emergency feedwater pumps is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the pressurizer safety valves. Figure 15.2-6, sheet 2 shows that at no time is there water relief from the pressurizer.

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

As shown in Figure 15.2-6, sheets 1 through 4, the plant approaches a stabilized condition following reactor trip and emergency feedwater initiation at hot standby with the emergency feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the emergency feed flow. The operating procedures would also call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the Emergency Feedwater System. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

15.2.7.3 Radiological Consequences

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

15.2.7.4 Conclusions

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

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description

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

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

This event is analyzed in order to evaluate the capacity of the emergency feedwater system to remove core decay heat, and to ensure that the core remains in a coolable geometry. In order to demonstrate this, a more limiting criterion is imposed such that the maximum hot leg temperature remains below the saturation temperature until the EFW heat removal capability exceeds the RCS heat generation, which demonstrates that the core remains covered with water.

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

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

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

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

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

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

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

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

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

The severity of the feedwater line rupture transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. Sensitivity studies presented in WCAP-9230[5] illustrate that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. Analyses were performed at full power with and without loss of offsite power. The pressurizer power-operated relief valves were modeled, as their modeling results in more limiting conditions.

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

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

   1. High pressurizer pressure

   2. Overtemperature ΔT

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

   4. Safety injection signals from any of the following:

      (a) Two out of three low steam line pressure in any one loop,

      (b) Two out of three high containment pressure (hi-1), or

      (c) Low pressurizer pressure.

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

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

15.2.8.2 Analysis of Effects and Consequences

a. Method of Analysis

   A detailed analysis using the RETRAN[6] code is performed in order to determine the plant transient following a feedwater line rupture. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

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

1. The plant is initially operating at an NSSS power of 3678 MWt, which includes calorimetric uncertainties.

2. Initial reactor coolant average temperature is 6.0 degrees F above the nominal value, and the initial pressurizer pressure is 50 psi below its nominal value.

3. Normal reactor control systems are not assumed to function unless their function results in more severe consequences. Therefore, the pressurizer PORVs are assumed to operate normally in order to minimize the RCS pressure.

4. Initial pressurizer level is at the nominal programmed value plus 5% uncertainty, initial steam generator water level is at the nominal value plus 8% in the faulted steam generator, and at the nominal value minus 14.5 percent in the intact steam generators.

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

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

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

8. Choked flow is assumed at the break.

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

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

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

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

13. Minimum high head ECCS pump performance and maximum ECCS temperature (98°F) are assumed. The flow rates assumed conservatively account for flow from only one centrifugal charging pump with 10% head degradation.

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

15. No credit is taken for charging or letdown.

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

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

18. One of the redundant EFW flow control valves leading to the faulted steam generator is assumed to close on a high flow rate signal with a bounding stroke time of 23 seconds to terminate EFW flow through the break. This stroke time is conservatively modeled as an additional delay over and above the EFW signal delay (2 seconds) and the delay for EFW pump start (77 seconds).

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

    (a) High pressurizer pressure

    (b) Overtemperature ΔT

    (c) High pressurizer level.

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

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

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

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

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

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

b. Results

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

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

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

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

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

15.2.8.3 Radiological Consequences

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

15.2.8.4 Conclusions

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

15.2.9 References

1. Cooper, L., Miselis, V. and Starek, R.M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June, 1972 (also letter NS-CE-622, dated April 16, 1975, C Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), additional information on WCAP-7769, Revision 1)

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

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

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

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

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

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

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

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

a. Partial Loss of Forced Reactor Coolant Flow

b. Complete Loss of Forced Reactor Coolant Flow

c. Reactor Coolant Pump Shaft Seizure (Locked Rotor)

d. Reactor Coolant Pump Shaft Break.

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

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

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

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

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

The necessary protection against a partial loss-of-coolant flow accident is provided by the low primary coolant flow reactor trip signal which is actuated in any reactor coolant loop by two out of three low flow signals, in any reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. Above Permissive 7, either power supply low voltage on both buses or opening of one reactor coolant pump breaker on each bus will actuate the corresponding undervoltage relays, resulting in a reactor trip. Additionally, underfrequency on the two buses will actuate a reactor trip above P-7. These trips serve as a backup to the low flow trip.

15.3.1.2 Analysis of Effects and Consequences

a. Method of Analysis

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

This transient is analyzed by two digital computer codes. The RETRAN[1] code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE[4] code is then used to calculate the heat flux and the Departure from Nucleate Boiling Ratio (DNBR) transients based on the nuclear power and RCS flow calculated by RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

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

1. Initial Conditions

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

2. Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used. The most positive moderator temperature coefficient allowed by the Technical Specifications at full power conditions, 0.0 pcm/°F, is assumed. This results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

3. Flow Coastdown

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

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

b. Results

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

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

15.3.1.3 Radiological Consequences

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

15.3.1.4 Conclusions

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

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

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

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

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

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

a. Reactor coolant pump power supply undervoltage or underfrequency

b. Low reactor coolant loop flow.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., loss of offsite power. Channel response time includes consideration of the bus voltage decay time due to generated Electro-Motive Force (EMF) from motors connected to the bus as the motors coast down. This function is blocked below approximately 10 percent power (Permissive 7).

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

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 Hz/second the low flow trip function will protect the core from underfrequency events. This effect is fully described in Reference 9.

15.3.2.2 Analysis of Effects and Consequences

a. Method of Analysis

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

This transient is analyzed by two digital computer codes. The RETRAN (Reference 1) Code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE Code (see Section 4.4) is used to calculate the heat flux and DNBR transients based on the RETRAN calculated nuclear power and RCS flow. The DNBR transients presented represent the minimum of the typical or thimble cell.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Subsection 15.3.1.2, except that following the loss of power supply to all pumps at power, a reactor trip is actuated by either reactor coolant pump power supply undervoltage or underfrequency.

b. Results

Figure 15.3-3, Figure 15.3-4, and Figure 15.3-5 show the transient response for the loss of power to all reactor coolant pumps. The reactor is assumed to be tripped on an undervoltage signal. Figure 15.3-5 shows the DNBR to be always greater than the limit value. Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

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

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

15.3.2.3 Radiological Consequences

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

15.3.2.4 Conclusions

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

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor

15.3.3.1 Identification of Causes and Accident Description

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

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

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

15.3.3.2 Analysis of Effects and Consequences

a. Method of Analysis

Two digital computer codes are used to analyze this transient. The RETRAN Code (Reference 1) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot are investigated using the VIPRE Code (Reference 4), using core flow and nuclear power calculated by RETRAN. The VIPRE code includes the use of a film boiling heat transfer coefficient.

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

1. Evaluation of the Pressure Transient

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

2. Evaluation of DNB in the Core During the Accident

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

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

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

Film Boiling Coefficient

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

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

Fuel Clad Gap Coefficient

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

Zirconium Steam Reaction

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

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

where:

w = amount reacted (mg/cm^2)

t = time (seconds)

T = temperature (Kelvin)

The reaction heat is 1,510 cal/g.

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

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

b. Results

The transient results for the most limiting conditions of the locked rotor and pump shaft break (Subsection 15.3.4) accidents are shown in Figure 15.3-6, sh.1, Figure 15.3-6, sh. 2, and Figure 15.3-6, sh. 3. The results of these calculations are also summarized in Table 15.3-1. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700°F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient.

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

15.3.3.3 Radiological Consequences Using Alternate Source Term Methodology

The limiting radiological consequences for the locked rotor and shaft break event are associated with a loss of offsite power and are presented in Section 15.3.4.3.

15.3.3.4 Conclusions

a. Since the peak reactor coolant system 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.

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

c. The doses which have been calculated for the locked rotor accident are below regulatory limits.

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

15.3.4.1 Identification of Causes and Accident Description

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

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

Nevertheless, a bounding evaluation of a locked rotor is provided in the analysis presented in Subsection 15.3.3, which assumed that offsite power is lost coincident with turbine trip. The transient is postulated to occur in the following manner:

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

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

c. Turbine-generator trips.

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

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

15.3.4.2 Analysis of Effects and Components

Method of Analysis

The method of analysis used is the same as presented in Subsection 15.3.3. A bounding value of maximum reactor coolant pressure is calculated by assuming offsite power is lost coincident with turbine trip. This assumption is conservative because grid stability analyses show offsite power will not be lost.

15.3.4.3 Radiological Consequences

a. Background

This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through the steam generator via the ASDVs and MSSVs. In addition, radioactivity is contained in the primary and secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

b. Compliance with RG 1.183 Regulatory Positions

The revised Locked Rotor dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1. Regulatory Position 1 – The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 15C-1. The inventory provided in Table 15C-1 is then adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The fraction of fission product inventory in the gap available for release due to fuel breach is consistent with Table 3 of RG 1.183.

2. Regulatory Position 2 – Fuel damage is assumed for this event.

3. Regulatory Position 3 – Activity released from the damaged fuel is assumed to mix instantaneously and homogeneously throughout the primary coolant.

4. Regulatory Position 4 – The chemical form of radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to equilibrium iodine concentrations in the RCS and secondary system.

5. Regulatory Position 5.1 – The primary-to-secondary leak rate is apportioned between the steam generators, as specified by Technical Specification 3.4.6.2, as 1 gpm total and 500 gallons per day to any one SG.

6. Regulatory Position 5.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate Technical Specification. For the intact Steam Generators, the primary to secondary leak rate is based on a density of 1.0 gm/cc (cold liquid).

7. Regulatory Position 5.3 – The release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the steam generators are terminated.

8. Regulatory Position 5.4 – The analysis assumes a coincident loss of offsite power.

9. Regulatory Position 5.5 – All noble gas radionuclides released from the primary system are assumed released to the environment without reduction or mitigation.

10. Regulatory Position 5.6 – The steam generator tubes are assumed to remain covered throughout this event for Seabrook. Therefore, the iodine and transport model for release from the SGs is as follows:

    - Appendix E, Regulatory Position 5.5.1 – All four steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix instantaneously and homogeneously with the secondary water without flashing.

    - Appendix E, Regulatory Position 5.5.2 – None of the SG tube leakage is assumed to flash for this event.

    - Appendix E, Regulatory Position 5.5.3 – All of the SG tube leakage is assumed to mix with the bulk water.

    - Appendix E, Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SG is limited by the moisture carryover from the SG. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

    - Appendix E, Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

c. Other Assumptions

1. RG 1.183, Section 3.6 – The assumed amount of fuel damage caused by the non-LOCA events is analyzed to determine the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and to determine the fraction of fuel elements for which fuel clad is breached. This analysis assumes DNB as the fuel damage criterion for estimating fuel damage for the purpose of establishing radioactivity releases. For the Locked Rotor event, Table 3 of RG 1.183 specifies noble gas, alkali metal, and iodine fuel gap release fractions for the breached fuel.

2. The initial RCS activity is assumed to be at the TS limit of 1.0 μCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The ratio of radioiodines to other radionuclides, provided in Table 11.1-1, is assumed to be constant and the activities are scaled up to produce the TS limit of 1.0 μCi/gm Dose Equivalent I-131 and 100/E-bar gross activity. The initial SG activity is assumed to be at the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131. This analysis also conservatively assumes that the initial secondary coolant activity includes 10% of the primary coolant equilibrium concentration of alkali metals.

3. This analysis assumes that the DNB fuel damage is limited to 10% breached fuel assemblies.

d. Methodology

Input assumptions used in the dose consequence analysis of the Locked Rotor event are provided in Table 15.3-3. This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Following the reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will rise. The rapid rise in primary system temperatures during the initial phase of the transient results in a reduction in the initial DNB margin and fuel damage.

For the purpose of this dose assessment, a total of 10% of the fuel assemblies are assumed damaged. A radial peaking factor of 1.65 is assumed. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant with source term from and release fractions per Appendix G of RG 1.183. Primary coolant is released to the SGs as a result of postulated primary-to-secondary leakage. Activity is released to the atmosphere via steaming from the steam generator ASDVs and MSSVs until the decay heat generated in the reactor core can be removed by the shutdown cooling system 8 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 150 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 150 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

e. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the location of the release and the pathway for ingress into the CR. These *X/Qs* are summarized in Tables 2R-2 and 2R-3.

The EAB and LPZ dose consequences are determined using the *X/Q* factors provided in Appendix 2Q for the appropriate time intervals.

The radiological consequences of the Locked Rotor event are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.3-5, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.3.4.4 Conclusion

The transient analysis performed in Subsection 15.3.3 assumes a loss of offsite power coincident with turbine trip. Thus, the conclusions of Subsection 15.3.3.4 apply for a reactor coolant pump shaft seizure followed by a loss of offsite power accident.

Grid stability analyses show that offsite power will not be lost following a turbine trip. However, a conservative radiological dose calculation was performed assuming offsite power is lost at the time of turbine trip and assuming primary to secondary allowable leakage is apportioned between the steam generators as 1 gpm total and 500 gallon per day to any one steam generator.

15.3.5 Reactor Coolant Pump Shaft Break

15.3.5.1 Identification of Causes and Accident Description

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

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

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

15.3.5.2 Radiological Consequences

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

15.3.5.3 Conclusion

The conclusions of Section 15.3.3.4 apply for a reactor coolant pump shaft break accident.

15.3.6 References

1) WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999

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

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

4) WCAP-14565-A (Proprietary) and WCAP-15306 (Non-Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, et al., October 1999

5) WCAP-11397-P-A, "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989

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

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

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

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

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

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

Discussions of the following incidents are presented:

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

b. Uncontrolled Rod Cluster Control Assembly bank withdrawal at power

c. Rod Cluster Control Assembly misalignment

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

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

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

g. Spectrum of Rod Cluster Control Assembly ejection accidents.

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

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

15.4.1.1 Identification of Causes and Accident Description

A Rod Cluster Control Assembly (RCCA) withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs, resulting in a power excursion. Such a transient could be caused by a malfunction of the Reactor Control or Rod Control Systems. This could occur with either the reactor subcritical, at Hot Zero Power or at power. The "at power" case is discussed in Subsection 15.4.2.

Procedural controls restrict rod motion if the power range nuclear instruments are inoperable. With RCA Tave less than 551ºF and power range NIs inoperable, the motor generator sets can only be energized if the RCS is borated to greater than the all rods out value or if alternate means have been established to ensure that the control and shutdown rods are not capable of being withdrawn.

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

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

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

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

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

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

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

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

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

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

15.4.1.2 Analysis of Effects and Consequences

a. Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: (1) an average core nuclear power transient calculation, (2) an average core heat transfer calculation, and (3) the DNBR calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods, TWINKLE (Reference 3), to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). The average heat flux is next used in VIPRE (described in Reference 4) for the transient DNBR calculation. Plant characteristics and initial conditions are discussed in Subsection 15.0.3.

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

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

2. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. The analysis assumes a moderator temperature coefficient of at least +5 pcm/°F at the zero power nominal temperature.

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

4. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10 percent increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 percent to 35 percent. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Subsection 15.0.5 for RCCA insertion characteristics.

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

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

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

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

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

b. Results

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

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

15.4.1.3 Radiological Consequences

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

15.4.1.4 Conclusions

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the Reactor Coolant System are not adversely affected, since the DNBR is greater than the limit value for all regions of the core. Thus, no fuel or clad damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

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

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

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

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

b. Reactor trip is actuated if any two-of-four channels exceed a rate lag setpoint on the high positive neutron flux rate setpoint.

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

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

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

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

Figure 15.0-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the Overpower ΔT trip and the Overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

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

15.4.2.2 Analysis of Effects and Consequences

a. Method of Analysis

This transient is analyzed by the RETRAN Code (Reference 15). This code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressure, and power level. The core limits as illustrated in Figure 15.0-1 are used as input to RETRAN to determine the minimum DNBR during the transient.

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

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

2. Reactivity coefficients - Two cases are analyzed:

    (a) Minimum reactivity feedback

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

    (b) Maximum reactivity feedback

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

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

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

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

The effect of RCCA movement on the axial core power distribution is accounted for by the f(ΔI) penalty function, which decreases the overtemperature ΔT setpoint proportional to the decrease in margin to DNB.

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

b. Results

Figure 15.4-4, sh.1, Figure 15.4-4, sh.2, and Figure 15.4-4, sh.3, the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since the neutron flux increase is rapid with respect to the thermal time constant, small changes in coolant temperature and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figure 15.4-5, sh.1, Figure 15.4-5, sh.2, and Figure 15.4-5, sh.3. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

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

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

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

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

1. For high reactivity insertion rates (i.e., between ~ 12 pcm/sec and 110 pcm/sec) reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux; minimum DNB ratio during the transient thus decreases with decreasing insertion rate.

2. The overtemperature ΔT channels initiate a reactor trip when measured coolant ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. It is important in this context to note that the average temperature contribution to the circuit as well as the measured ΔT that is compared to the setpoint are lead-lag compensated in order to decrease the effect of the thermal capacity of the RCS in response to power increases.

3. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient (i.e. at ~ 12 pcm/sec reactivity insertion rate).

For reactivity insertion rates less than ~ 12 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.

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

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

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

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

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

15.4.2.3 Radiological Consequences

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

15.4.2.4 Conclusions

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

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

15.4.3.1 Identification of Causes and Accident Description

Rod Cluster Control Assembly (RCCA) misalignment accidents include:

a. One or more dropped RCCAs within the same group

b. A dropped RCCA bank

c. Statically misaligned RCCA

d. Withdrawal of a single RCCA.

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

Full length RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

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

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

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

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

A dropped RCCA or RCCA bank is detected by:

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

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

c. Rod at bottom signal

d. Rod deviation alarm or

e. Rod position indication.

Misaligned RCCAs are detected by:

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

b. Rod deviation alarm or

c. Rod position indicators.

The resolution of the rod position indicator channel is ± 1.7 percent of span (± 2.5 inches). Deviation of any RCCA from its group by twice this distance (3.4 percent of span, or 5 inches) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5.1 percent of span. If the rod deviation alarm is not operable, the operator is required to take action as required by the technical specifications.

If one or more rod position indicator channels should be out of service, detailed operating instructions shall be followed to assure the alignment of the nonindicated RCCA. The operator is also required to take action as outlined by the Technical Specifications.

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

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

15.4.3.2 Analysis of Effects and Consequences

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

1. Method of Analysis

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

The LOFTRAN[1] is used to calculate the transient system response for the evaluation of the dropped RCCA event. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Transient statepoints (temperature, pressure and power) are calculated by LOFTRAN and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient analysis and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using dropped rod limit lines developed with the Westinghouse version of VIPRE-01 code (VIPRE)[4]. The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are performed in accordance with the methodology described in WCAP-11394[8]. Note that the analysis does not take credit for the negative flux rate reactor trip.

A generic statepoint analysis for this event, which was performed in 1986 to bound a number of four-loop PWRs, was evaluated and determined to remain applicable to Seabrook. With the generic statepoints being applicable, the effects of the power uprate are accounted for in the DNB analysis, which is performed on a cycle specific basis.

(b) Dropped RCCA Bank

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

(c) Statically Misaligned RCCA

Steady-state power distributions are analyzed using appropriate nuclear physics computer codes. The peaking factors are then compared to peaking factor limits developed using the VIPRE code, which are based on meeting the DNBR design criterion. The analysis examines the following cases: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out, all with the reactor at full power. The analysis assumes this incident to occur at the time in core life with maximum predicted peaking factors.

2. Results

(a) One or More Dropped RCCAs

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

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

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

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

(b) Dropped RCCA Bank

A dropped RCCA bank results in a large negative reactivity insertion. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described for a dropped RCCA above. In most cases, the negative reactivity worth of a dropped RCCA bank is greater than the available positive reactivity worth associated with the automatic withdrawal of control bank D from its full power insertion limit. However, in the case of a relatively low worth dropped RCCA bank, such as Shutdown Bank A, the available positive reactivity worth from automatic control bank D withdrawal may exceed the negative reactivity worth of the dropped bank by a small amount. Therefore, a power overshoot may still occur in the case of a dropped RCCA bank, due to the combined effects of both automatic withdrawal of bank D and moderator temperature reactivity feedback. However, the magnitude of the possible power overshoot is smaller with a dropped RCCA bank, than it is with single or multiple dropped RCCAs, due to the greater worth of the entire dropped bank, when compared to the available D-bank worth. In addition, the power distribution associated with a dropped RCCA bank is symmetric, resulting in lower peaking factors, when compared to the asymmetric power distributions associated with single or multiple dropped RCCAs. Therefore, the minimum DNBR for a dropped RCCA bank event is bounded by the DNBR associated with single or multiple dropped RCCAs. Following plant stabilization, normal procedures are followed.

(c) Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where bank D is inserted to its insertion limit with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following the incident.

The insertion limits in the technical specifications may vary from time to time depending on a number of limiting criteria. The full power insertion limits on control bank D must be chosen to meet minimum DNBR and peaking factor criterion under normal and misaligned rod conditions. However, the actual insertion limit is usually dictated by other criterion. Detailed results will vary from cycle to cycle depending on fuel arrangements.

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

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

Calculations have not been performed specifically for RCCAs misaligned from other control banks, which are permitted to be either fully or partially inserted at part power conditions. However, it has been determined on a generic basis that the increase in radial peaking factor necessary to reach the DNBR limit at reduced power conditions, is greater than the credible increase in radial peaking factors associated with reduced thermal power levels and deeper permitted control bank insertion. Therefore, the full power case discussed above with bank D at the insertion limit is more limiting than any credible part power RCCA misalignment scenario involving rods at the insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value.

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

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

b. Single RCCA Withdrawal

1. Method of Analysis

Core power distributions simulating a single RCCA withdrawal event are calculated using the computer code ANC. The case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power, is identified and analyzed. The purpose of this calculation is to confirm that the number of fuel rods that go through DNB is less than the safety analysis limit of 5%. The ANC calculated peaking factors are compared to the design peaking factor used to set the overtemperature ΔT trip. Overtemperature ΔT trip setpoints are established to prevent exceeding DNBR limits. If the calculated peaking factors are above the design peaking factor limit, including appropriate calculational uncertainty, a fuel census is generated for the most limiting case to determine the percentage of rods in the core which exceed the design peaking factor. All rods which exceed the design peaking factor are assumed to undergo DNB prior to reaching the power and coolant conditions that would trip the plant on overtemperature ΔT.

The ANC calculations are performed at the time in core life which has the highest peak $F_{\Delta H}$. Power distributions are generated for all unique combinations of bank D inserted to the full power insertion limit, with one bank D RCCA fully withdrawn. Xenon reconstruction is used to skew the axial flux difference to the upper allowable limit. The most limiting configuration is determined by the case that produces the highest peaking factors under these conditions.

2. Results

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

(a) If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the withdrawing RCCA. In terms of the overall system response, this case is similar to those presented in Subsection 15.4.2; however, the increased local power peaking in the area of the withdrawn RCCA may result in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNB ratio from falling below the limit value. Evaluation of this case at the power and coolant conditions at which the

Overtemperature ΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the limit value is 5 percent.

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

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

15.4.3.3 Radiological Consequences

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

15.4.3.4 Conclusions

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

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

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

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

15.4.4.1 Identification of Causes and Accident Description

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

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

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

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

15.4.4.2 Analysis of Effects and Consequences

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

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

Not applicable to Seabrook.

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

15.4.6.1 Identification of Causes and Accident Description

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

The inadvertent opening of the Reactor Makeup Water (RMW) control valve in conjunction with a failure in the blend system permitting 0 ppm water to flow from the discharge of a single RMW pump to the charging pump suction is considered the limiting ANS Condition II boron dilution event for all modes of operation. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to an RMW pump.

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

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

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

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

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

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

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

15.4.6.2 Method of Analysis

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

a. Dilution during Refueling

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

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of All Rods In (ARI) most reactive time in life, no xenon, with $T_{avg} \leq 140°F$.
2. The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:
    a. A K_{eff} of 0.95 or less; or
    b. A boron concentration of greater than or equal to 2,100 ppm.
3. Dilution flow is assumed to be 150 gpm.
4. Mixing of the reactor coolant is accomplished by the operation of at least one residual heat removal pump.
5. A minimum water volume (3,395 ft^3) in the Reactor Coolant System is used. This is the minimum volume of the RCS for residual heat removal system operation. The water in the reactor vessel is assumed to be drained so that the nozzles are half-filled. The total volume includes the reactor vessel up to the nozzle centerline, one hot leg half filled up to the RHR connection, two cold legs half filled up to the RHR connections, and the active volume of one RHR loop.
6. The density of RCS fluid is assumed to be 61.4 lb/ft^3.

b. Dilution during Cold Shutdown (with Filled Loops)

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

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

3. The assumed dilution flowrate is 150 gpm.

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

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

6. The density of RCS fluid is assumed to be 60.1 lb/ft^3.

c. Dilution during Cold Shutdown (with Drained Loops)

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

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

3. The assumed dilution flowrate is 150 gpm.

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

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

6. The density of RCS fluid is assumed to be 60.1 lb/ft^3.

d. Dilution during Hot Shutdown

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of zero power, ARI less the highest worth rod, most reactive time in life, no xenon, with 200°F ≤ T_{avg} ≤ 350°F.
2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.
3. The assumed dilution flowrate is 150 gpm.
4. A minimum water volume of 3,992 ft^3 in the RCS is used.
5. The density of RCS fluid is assumed to be 55.6 lb/ft^3 (350°F, saturated condition conservatively used. RCS is maintained 50°F subcooled).

e. Dilution during Hot Standby

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

1. The maximum boron concentration to lose all shutdown margin conservatively bounds the condition of zero power, ARI less the highest worth rod, most reactive time in life, no xenon, with 350°F ≤ T_{avg} ≤ 557°F.
2. Technical Specifications and the Core Operating Limits Report require a minimum shutdown margin. The assumed initial boron concentration is the minimum boron concentration required to meet this shutdown margin.
3. Dilution flow is assumed to be limited to the capacity of one RMW pump (150 gpm).
4. A minimum water volume (8,645.9 ft^3) in the Reactor Coolant System is used. This volume corresponds to the active volume of the Reactor Coolant System, minus the pressurizer and surge line volumes.
5. Mixing of the reactor coolant is accomplished by operation of the reactor coolant pumps.
6. The density of the RCS fluid is assumed to be 46.4 lb/ft^3 (557°F and 2250 psia).

f. Dilution During Startup (Mode 2)

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

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

2. A minimum water volume (9818.3 ft^3) in the RCS is used. This is a conservative estimate of the active volume of the RCS minus the pressurizer volume, and accounts for 10% steam generator tube plugging.

3. The initial condition in the analysis is assumed to be during the dilution, corresponding to a critical, hot zero power condition with the control rods at the rod insertion limits. A reactor trip on source range high neutron flux is assumed to occur at this condition, alerting the operator to the dilution in progress. The maximum boron concentration at which the reactor will again attain criticality with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition at the rod insertion limits is taken as 200 ppm.

g. Dilution During Power Operation (Mode 1)

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

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

2. A minimum water volume (9818.2 ft^3) in the RCS is used. This is a conservative estimate of the active volume of the RCS minus the pressurizer volume, and accounts for 10% steam generator tube plugging.

3. For the case of manual reactor control, the initial condition in the analysis is assumed to correspond to a critical, hot full power condition with the control rods at the rod insertion limits. Dilution causes the power and RCS temperature to rise, resulting in a reactor trip on overtemperature ΔT or high neutron flux, alerting the operator to the dilution in progress. The maximum boron concentration at which the reactor will again attain criticality at hot zero power with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition of hot full power at the rod insertion limits is taken as 200 ppm.

   For the case of automatic reactor control, the initial condition in the analysis is assumed to correspond to a critical, hot full power condition with the control rods at the rod insertion limits. The operator will be alerted to the dilution in progress by the low-low rod insertion limit alarm. The maximum boron concentration at which the reactor will attain criticality at hot zero power with all rods inserted less the most reactive RCCA stuck out of the core is taken as 1750 ppm. The minimum change from this condition to the initial condition of hot full power at the rod insertion limits is taken as 200 ppm.

15.4.6.3 Results of Analysis

a. Dilution during Refueling

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

b. Dilution during Cold Shutdown (with Filled Loops)

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

c. Dilution during Cold Shutdown (with Drained Loops)

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

d. Dilution during Hot Shutdown

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

e. Dilution during Hot Standby

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

f. Dilution During Startup (Mode 2)

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

g. Dilution During Power Operation (Mode 1)

During full power operation with the reactor in manual control, the operator has at least 15 minutes following reactor trip on overtemperature ΔT until the loss of shutdown margin. The maximum reactivity insertion rate resulting from the boron dilution is 1.4 pcm/sec.

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

15.4.6.4 Radiological Consequences

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

15.4.6.5 Conclusions

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

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

15.4.7.1 Identification of Causes and Accident Description

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

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

To reduce the probability of core loading errors, strict administrative controls are placed on the entire core loading sequence. Then, using the core loading patterns from the just completed cycle and the ensuing cycle along with the spent fuel pool map, a core loading sequence is developed. The core loading sequence provides the step-by-step instructions necessary to end up with the desired core configuration. Lastly, as part of the reload process, fuel assembly identification numbers and component types are verified during the reload process. This positive identification along with the multiple independent verification programs utilized during the placement in the reactor vessel provides additional assurance that the fuel is loaded in accordance with the core loading pattern.

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

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

15.4.7.2 Radiological Consequences

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

15.4.7.3 Conclusions

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

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

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

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

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

a. Design Precautions and Protection

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

1. Mechanical Design

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

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

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

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

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

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

2. Nuclear Design

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

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

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

3. Reactor Protection

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

Procedural controls restrict rod motion if the power range nuclear instruments are inoperable. With RCA Tave less than 551ºF and power range NIs inoperable, the motor generator sets can only be energized if the RCS is borated to greater than the all rods out value or if alternate means have been established to ensure that the control and shutdown rods are not capable of being withdrawn.

4. Effects on Adjacent Housings

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

5. Effects of Rod Travel Housing Longitudinal Failures

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

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

6. Effects of Rod Travel Housing Circumferential Failures

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

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

7. Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause the RCCA not to fall on receiving a trip signal; however, this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

8. Summary

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

b. Limiting Criteria

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

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

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

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

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

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

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

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

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

15.4.8.2 Analysis of Effects and Consequences

a. Method of Analysis

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

b. Average Core Analysis

The spatial kinetics computer code, TWINKLE[3], is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 8000 spatial points. The computer code includes a detailed multiregion, transient fuel clad coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor.

c. Hot Spot Analysis

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

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

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

d. System Overpressure Analysis

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

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a thermal hydraulic calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in the NSSS 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. The system overpressure is generically addressed in Reference 10.

Calculation of Basic Parameters

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

a. Ejected Rod Worths and Hot Channel Factors

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

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties.

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

b. Reactivity Feedback Weighting Factors

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

c. Moderator and Doppler Coefficient

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

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

d. Delayed Neutron Fraction, βeff

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

e. Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-2 and includes the effect of one stuck RCCA adjacent to the ejected rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 second after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breaker to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time was used which assumed that insertion to the dashpot does not occur until 2.4 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over one second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

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

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

f. Reactor Protection

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

g. Results

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

(1) Beginning of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.25%$\Delta\rho$ and 6.0 respectively. The maximum fuel stored energy was 164 cal/gm. The peak hot spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the fuel pellet.

(2) Beginning of Cycle, Zero Power

For this condition, control bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in control bank D and has a worth of 0.78%$\Delta\rho$ and a hot channel factor of 11.5. The maximum fuel stored energy was 141 cal/gm. The peak fuel center temperature was 3835°F.

(3) End of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.25%$\Delta\rho$ and 7.0 respectively. The maximum fuel stored energy was 164 cal/gm. The peak fuel center temperature was 4850°F.

(4) End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and banks B and C at their insertion limits. The results were 0.85%$\Delta\rho$ and 26.0 respectively. The maximum fuel stored energy was 149 cal/gm. The peak fuel center temperature was 3938°F. The Doppler weighting factor for this case is significantly higher than that of the other cases due to the very large transient hot channel factor.

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

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

h. Fission Product Release

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

i. Pressure Surge

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

j. Lattice Deformations

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

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

15.4.8.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event consists of the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly and drive shaft. This event is the same as the Rod Ejection event referred to in RG 1.183. The RCCA Ejection results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Following the reactor trip, plant cooldown is effected by steam release from the SG MSSVs/ASDVs. Two RCCA Ejection cases are considered. The first case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. The second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system.

b. Compliance with RG 1.183 Regulatory Positions

The RCCA Ejection dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," as discussed below:

1. Regulatory Position 1 – The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 15C-1. The inventory provided in Table 15C-1 is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The release fractions provided in RG 1.183 Table 3 are adjusted to comply with the specific RG 1.183 Appendix H release requirements. For both the containment and secondary release cases, the activity available for release from the fuel gap for fuel that experiences DNB is assumed to be 10% of the noble gas and iodine inventory in the DNB fuel. For the containment release case for fuel that experiences fuel centerline melt (FCM), 100% of the noble gas and 25% of the iodine inventory in the melted fuel is assumed to be released to the containment. For the secondary release case for fuel that experiences FCM, 100% of the noble gas and 50% of the iodine inventory in the melted fuel is assumed to be released to the primary coolant.

2. Regulatory Position 2 – Fuel damage is assumed for this event.

3. Regulatory Position 3 – For the containment release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the containment atmosphere. For the secondary release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the primary coolant and be available for leakage to the secondary side of the SGs.

4. Regulatory Position 4 – The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide.

5. Regulatory Position 5 – The chemical form of radioiodine released from the SGs to the environment is assumed to be 97% elemental iodine, and 3% organic iodide.

6. Regulatory Position 6.1 – For the containment leakage case, natural deposition in the containment is credited. In addition, the Secondary Containment ventilation filtration is credited. Containment spray is not credited.

7. Regulatory Position 6.2 – The containment is assumed to leak at the TS maximum allowable rate of 0.15% for the first 24 hours and 0.075% for the remainder of the event.

8. Regulatory Position 7.1 – The primary-to-secondary leak rate is apportioned between the SGs as specified by proposed TS 3.4.6.2 (1.0 gpm total, 500 gallons per day to any one SG).

9. Regulatory Position 7.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS.

10. Regulatory Position 7.3 – All of the noble gas released to the secondary side is assumed to be released directly to the environment without reduction or mitigation.

11. Regulatory Position 7.4 – Compliance with Appendix E Sections 5.5 and 5.6 is discussed below:

    - Appendix E, Regulatory Position 5.5.1 – All four steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

    - Appendix E, Regulatory Position 5.5.2 – None of the SG tube leakage is assumed to flash for this event.

    - Appendix E, Regulatory Position 5.5.3 – All of the SG tube leakage is assumed to mix with the bulk water.

- Appendix E, Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

- Appendix E, Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

c. Other Assumptions

1. RG 1.183 does not address secondary coolant activity. This analysis assumed that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131. This analysis also conservatively assumes that the initial secondary coolant activity includes 10% of the primary coolant equilibrium concentration of alkali metals.

2. The steam mass release rates for the SGs are provided in Table 15.4-4.

3. This evaluation assumed that the RCS mass remains constant throughout the event.

4. The SG secondary side mass in the SGs is assumed to remain constant throughout the event.

5. Steam releases from the SGs are postulated to occur from the MSSV or ASDV with the most limiting atmospheric dispersion factors. For the RCCA Ejection inside of containment release case, releases are assumed to leak out of the containment via the same containment release points as used for the LOCA.

d. Methodology

Input assumptions used in the dose consequence analysis of the RCCA Ejection are provided in Table 15.4-3. The postulated accident consists of two cases. One case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere, and the second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system.

For the containment release case, 100% of the activity is released instantaneously to the containment. The releases from the containment correspond to the same leakage points as used for the LOCA. Natural deposition of the released activity inside of containment is credited. In addition, the secondary containment building ventilation and filtration system is credited. Removal of activity via containment spray is not credited.

For the secondary release case, primary coolant activity is released into the SGs by leakage across the SG tubes. The activity on the secondary side is then released via steaming from the SG MSSVs/ASDVs until the decay heat generated in the reactor core can be removed by the Shutdown Cooling (SDC) system 8 hours into the event. Additional activity, based on the secondary coolant initial iodine concentration is assumed to be equal to the maximum value of 0.1 μCi/gm DE I-131 permitted by TS 3.7.1.4. Activity is released to the environment from the steam generator as a result of the postulated primary-to-secondary leakage and the postulated activity levels of the primary and secondary coolants, until the steam generator steam release is terminated (at 8 hours for SDC initiation). These release assumptions are consistent with the requirements of RG 1.183

The RCCA Ejection is evaluated with the assumption that 0.375% of the fuel experiences FCM and 15.0% of the fuel experiences DNB. The activity released from the damaged fuel corresponds to the requirements set out in Regulatory Position 1 of Appendix H to RG 1.183. A radial peaking factor of 1.65 is applied in the development of the source terms.

For this event, the Control Room ventilation system modes of operation are:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 150 cfm of unfiltered inleakage assumed for the secondary side release or 190 cfm unfiltered inleakage assumed for the primary release.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 150 cfm of unfiltered inleakage assumed for the secondary side release or 190 cfm unfiltered inleakage assumed for the primary release and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

e. <u>Radiological Consequences</u>

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the pathway into the control room. These *X/Qs* are summarized in Tables 2R-2 and 2R-3.

For the RCCA secondary side release case, releases from the SGs are assumed to occur from the MSSV/ASDV that produces the most limiting *X/Qs*. For the RCCA Ejection containment release case, the *X/Qs* for containment leakage are assumed to be identical to those for the LOCA.

For the EAB and the LPZ dose the *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the RCCA Ejection are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.4-5, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.4.8.4 Conclusions

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

15.4.9 Spectrum of Rod Drop Accidents in a BWR

Not applicable to Seabrook.

15.4.10 References

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

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

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

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

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

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

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

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

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

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

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

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

13) Letter from W. J. Johnson (Westinghouse) to R. C. Jones (USNRC), "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, October 1989

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

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

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events is presented in this section:

a. Inadvertent operation of Emergency Core Cooling System during power operation.

b. Chemical and volume control system malfunction that increases reactor coolant inventory.

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Subsection 15.0.1 contains a discussion of ANS classifications.

15.5.1 Inadvertent Operation of Emergency Core Cooling System during Power Operation

15.5.1.1 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 actuation signal. A spurious signal may originate from any of the safety injection actuation channels as described in Section 7.3.

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 (2600 ppm) boric acid solution into the cold leg of each loop. The safety injection pumps also start automatically but provide no flow when the Reactor Coolant System (RCS) is at normal RCS pressure.

A Safety Injection System (SIS) signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS signal, the operator should determine if the spurious signal was transient or steady-state in nature. The operator must also determine if the safety injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, future plant operation will be in accordance with the Technical Specifications.

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

The time of 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 second case is made in the same manner as described for the case where the SIS signal results directly in a reactor trip. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. Since the negative reactivity from the injected boron causes reactor power to decrease, the time at which reactor trip occurs has little effect on DNBR.

A second issue associated with this event is the possibility of the pressurizer overfilling, especially a possible condition where the pressurizer is water-solid and its pressure reaches the setpoint of the pressurizer safety relief valves. In this condition, water would pass through these safety relief valves, which could damage the valves and challenge the ability to ensure the RCS boundary can be isolated. The analysis focuses on the pressurizer filling aspects of the event and demonstrates that the pressurizer does not become water-solid, and therefore, there is no water flow through the PORV's or pressurizer safety valves.

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

15.5.1.2 Analysis of Effects and Consequences

a. Method of Analysis

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits for DNBR. The spurious operation of the ECCS is analyzed by employing the RETRAN computer code. The RETRAN computer code is discussed in UFSAR Section 15.0.11.

The analysis employs several plant parameters at maximum and minimum values to maximize the rate for pressurizer filling. For example, the analysis employs maximum reactivity feedback, maximum initial pressurizer water level, and the minimum initial reactor coolant temperature to maximize the rate of pressurizer filling.

The assumptions are as follows:

1. Initial Operating Conditions

   The impact of the full power RCS Tavg window was considered. The upper end of the Tavg window was determined to be more limiting. The higher corresponding pressurizer level turned out to be more limiting than the benefit gained by the lower initial mass.

   The impact of the feedwater temperature window was also analyzed and the upper end of the feedwater temperature window was determined to be slightly more limiting.

   Initial reactor power is assumed to be at the maximum value, initial reactor coolant temperature is assumed to be at the minimum value, initial pressurizer pressure is assumed to be at its minimum value, and the initial pressurizer water level is assumed to be at its maximum value, consistent with steady-state full power operation including allowances for calibration and instrument errors.

2. Moderator and Doppler Coefficients of Reactivity

   A most-negative moderator temperature coefficient was used. A high (absolute value) Doppler power coefficient was assumed.

3. Reactor Control

   The reactor was assumed to be in manual control.

4. Pressurizer Control

   Pressurizer heaters and spray are assumed to be operable to increase the rate of pressurizer filling. The pressurizer sprays act to reduce the RCS pressure, thus increasing ECCS injection. The pressurizer heaters act to add energy to the pressurizer fluid, thus increasing the pressurizer fluid volume through thermal expansion.

5. Boron Injection

   At time zero, two charging pumps conservatively inject 2,600 ppm borated water into the cold leg of each loop.

6. Reactor Trip

   The reactor and turbine are assumed to trip upon receipt of the SI signal. Assuming reactor and turbine trip on SI minimizes the heat removal capability of the RCS, thereby maximizing the RCS inventory increase through SI flow and thermal expansion of the RCS fluid.

7. Pressurizer PORVs

   No credit is taken for any pressurizer PORV operation.

Plant systems and equipment, which are available to mitigate the effects of the accident, are discussed in Subsection 15.0.8 and listed in Table 15.0-5. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident. Safety injection termination is defined as the stopping of all mass injection into the RCS.

b. Results

Figure 15.5-1 shows the transient response to inadvertent operation of the ECCS during power operation. The calculated sequence of events is shown on Table 15.5-1.

Reactor trip occurs at event initiation followed by a rapid initial cooldown of the RCS. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes the pressure and level transient to rapidly turn around. Pressurizer water level then increases throughout the transient. Per Emergency Operating Procedures, RCS temperature (T_{avg}) is maintained at 557ºF through heat removal from the steam generators using the steam generator atmospheric steam dump valves, and flow from all but one centrifugal charging pump is terminated early in the event. The analysis credits heat removal through the steam generators using the atmospheric steam dump valves, stopping of all but one centrifugal charging pump at 9 minutes after the beginning of the event, and termination of all charging flow at 13 minutes into the event based on analyzed maximum fill rate. The results of the revised analysis indicate that at no time does the pressurizer become water-solid.

15.5.1.3 Radiological Consequences

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

15.5.1.4 Conclusions

Results of the analysis of the inadvertent ECCS initiation at power event demonstrate that there is no hazard to the integrity of the RCS. The approach to terminating this event is consistent with Option II of Westinghouse NSAL 93-013.

For this event, the DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the RCS.

Operator action terminating safety injection flow is sufficient to preclude a pressurizer water-solid condition and prevent actuation of the pressurizer PORVs and safety valves. By demonstrating that sufficient time is available for the appropriate operator actions to preclude a pressurizer water-solid condition, the pressurizer valve integrity can be maintained for the inadvertent ECCS initiation at power event. No credit for operation of the pressurizer PORVs is assumed. Therefore, the ability to isolate the RCS and maintain the integrity of the RCS pressure boundary confirms that this event does not lead to a more serious plant condition, hence demonstrating acceptability of the Condition II acceptance criteria.

15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Transients due to CVCS malfunctions that increase the reactor coolant inventory can be divided into three categories:

Category 1	CVCS malfunctions that result in the injection of water with a boron concentration greater than the RCS boron concentration.
Category 2	CVCS malfunctions that result in the injection of water with a boron concentration less than the RCS boron concentration.
Category 3	CVCS malfunctions that result in the injection of water with a boron concentration equal to the RCS boron concentration.

There are two possible criteria for evaluating these transients: core integrity and overfilling of the pressurizer. Transients of the type listed in Category 1 are bounded by the "inadvertent operation of emergency core cooling system analysis" presented in Subsection 15.5.1. Transients of the type listed in Category 2 are bounded by the "CVCS malfunction that results in a decrease in boron concentration in the reactor coolant" presented in Subsection 15.4.6.

CVCS malfunctions of the type described under Category 3 will not result in any significant nuclear power or RCS temperature transient; this type of transient may result in filling the pressurizer. An analysis of the CVCS malfunction that results in injection of water with a boron concentration equal to the RCS boron concentration are presented in this section.

<u>CVCS Malfunctions that Result in the Injection of Water with a Boron Concentration Equal to the RCS Boron Concentration</u>

a. <u>Identification of Causes and Accident Description</u>

The most limiting case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow would be isolated. The worst single failure for this event would be another pressurizer level channel failing in an as is condition or a low condition. This will defeat the reactor trip on 2 out of 3 high pressurizer level channels. To prevent filling the pressurizer, the operator must be relied upon to terminate charging. This event is classified as a Condition II incident (an incident of moderate frequency).

b. <u>Analysis of Effects and Consequences</u>

The CVCS malfunction is analyzed by employing the detailed digital computer program RETRAN[1]. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the ECCS. The program computes pertinent plant variables, including temperatures, pressures, and power level.

The assumptions incorporated in the analyses were as follows:

1. Initial Operating Conditions

Pressurizer pressure is assumed to be at its minimum value. Pressurizer water level is assumed to be at the high end of the range of the values consistent with its programmed level. The initial reactor power and RCS temperature are at their full power values with uncertainties.

The impact of the full power RCS T_{avg} window was considered. The upper end of the T_{avg} window was determined to be more limiting. The higher corresponding pressurizer level turned out to be more limiting than the benefit gained by the lower initial mass.

The impact of the feedwater temperature window was also analyzed and the upper end of the feedwater temperature window was determined to be slightly more limiting.

2. Reactivity Coefficients

   Maximum reactivity feedback case

   The most negative moderator temperature coefficient and a most negative Doppler coefficient.

3. Reactor Control

   Both manual and automatic control have been analyzed.

4. Charging System

   Maximum charging system flow based on RCS back pressure from one centrifugal pump is delivered to the RCS.

5. Reactor Trips

   The transient is initiated by the pressurizer level channel which is used for control purpose failing low. As a worst single failure, another pressurizer level channel fails low, defeating the two out of three high pressurizer level trip. No reactor trips are used.

c. Results

Figure 15.5-2 shows the transient response due to the charging system malfunction. In all the cases analyzed, core power and RCS average temperature remain relatively unchanged.

The calculated sequence of events is shown in Table 15.5-2.

d. Conclusions

The sequence of events presented in Table 15.5-2 shows that the operator has sufficient time to take corrective action.

15.5.3 References

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

2. Westinghouse Nuclear Safety Advisory Letter NSAL-93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory, as discussed in this section, are as follows:

a. Inadvertent opening of a pressurizer safety or relief valve

b. Break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate Containment

c. Steam generator tube failure

d. Loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

15.6.1.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System (RCS) 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 flow rate 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 the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback (positive MTC), but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

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

a. Overtemperature ΔT

b. Pressurizer low pressure.

An inadvertent opening of a pressurizer safety or relief valve is classified as an ANS Condition II event, a fault of moderate frequency. See Subsection 15.0.1 for a discussion of Condition II events.

15.6.1.2 Analysis of Effects and Consequences

a. Method of Analysis

The accidental depressurization transient is analyzed by employing the detailed digital computer code RETRAN (Reference 4). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the revised thermal design procedure described in WCAP-11397 (Reference 36).

In order to give conservative results in calculating the Departure from Nucleate Boiling Ratio (DNBR) during the transient, the following assumptions are made:

1. Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397 (Reference 36).
2. The most positive MTC is assumed.
3. The least negative Doppler coefficient of reactivity is assumed such to maximize power increase prior to the reactor trip.
4. The pressurizer safety valve flowrate is assumed to be 120% of the design capacity of the valve.

Plant systems and equipment which are necessary to mitigate the effects of RCS depressurization caused by an inadvertent safety valve opening are discussed in Subsection 15.0.8 and listed in Table 15.0-5.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the manual mode in order to prevent rod insertion due to an increase in RCS temperature prior to reactor trip. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

b. Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figure 15.6-1 and Figure 15.6-2. Figure 15.6-1 illustrates the nuclear power transient following the depressurization. Nuclear power increases slowly from the initial value until reactor trip occurs on overtemperature ΔT. The pressure transient and average coolant temperature transient following the accident are given in Figure 15.6-2. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 15.6-1. The DNBR remains above the limit value throughout the transient.

The calculated sequence of events for the inadvertent opening of a pressurizer safety valve incident is shown on Table 15.6-1.

15.6.1.3 Radiological Consequences

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

15.6.1.4 Conclusions

The results of the analysis show that the Low Pressurizer Pressure and the Overtemperature ΔT reactor protection system signals provide adequate protection against the RCS depressurization event. No fuel or clad damage is predicted for this accident.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes and Accident Description

The sample lines from the hot legs of reactor coolant loops 1 and 4, and from the steam and liquid space of the pressurizer, and the chemical and volume control system (CVCS) letdown and excess letdown lines penetrate the Containment. The sample lines are provided with normally closed isolation valves on both sides of the containment wall, as sampling requirements dictate only intermittent daily use. The CVCS letdown line is provided with normally open containment isolation valves on both sides of the containment wall that are designed in accordance with the requirements of General Design Criteria 55. The excess letdown line is normally isolated and is also provided with two normally open containment isolation valves. The temperature of this fluid leaving the Containment is a maximum of 380°F.

The most severe pipe rupture with regard to radioactivity release during normal power operation is a complete severance of the 3-inch letdown line outside the Containment between the outboard containment isolation valve and the letdown heat exchanger (see Figure 9.3-26, Figure 9.3-27, Figure 9.3-28, Figure 9.3-29 and Figure 9.3-31).

15.6.2.2 Analysis of Effects and Consequences

The occurrence of a complete severance of the letdown line would result in a loss-of-reactor coolant at the rate of about 140 gpm (referenced to a density of 62 lb/ft^3). This release rate would not result in actuation of any Engineered Safety Features Systems. Area radiation and leakage detection instrumentation provide the primary means for detection of a letdown line rupture (see Subsection 5.2.5). Frequent operation of the CVCS Reactor Makeup Control System and the other CVCS instrumentation will aid the operator in diagnosing a letdown line rupture.

The time required for the operator to identify the accident and isolate the rupture is expected to be within 30 minutes of the rupture. Once the rupture is identified, the operator would isolate the letdown line rupture by closing the high pressure letdown valves, followed by closing the pressurizer low level isolation valves. Alternatively, the operator could close the letdown line containment isolation valves to isolate the rupture. All valves are provided with control switches at the main control board. There are no single failures that would prevent isolation of the letdown line rupture.

15.6.2.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event is a rupture of a primary coolant letdown line outside of containment. Since RG 1.183 does not provide specific guidance to the analysis of this type of event, the general guidance of the Regulatory Guide will be supplemented with guidance of the Standard Review Plan (SRP) section 15.6.2 and consideration of the current licensing basis for this event. In accordance with SRP 15.6.2, the source term for this calculation will assume an accident-generated or concurrent iodine spike. A reactor trip is not predicted for this event. The dose assessment for this event is comprised of two separate release paths. Path 1 defines the leakage from the double ended rupture of the Letdown line in the Plant Auxiliary Building (PAB) outside of containment. Path 2 defines the release of activity through the secondary side steam release from the condenser.

b. Compliance with RG 1.183 Regulatory Positions

Since Regulatory Guide 1.183 does not provide any direct guidance regarding analysis of a Letdown Line Rupture, Standard Review Plan (SRP) Section 15.6.2 is used as the primary source of guidance for this analysis. In accordance with SRP 15.6.2, this analysis assumes an accident-generated or concurrent iodine spike in combination with the maximum leakage of primary fluid through the SG tubes into the secondary side. The RG 1.183 guidance provided for other events is applied to this event as applicable and appropriate.

The revised Letdown Line Rupture event dose consequence analysis is consistent with the guidance provided in RG 1.183, as discussed below:

1. Regulatory Position 2.2 of Appendix E – This guidance is used to define the concurrent iodine spike of 500 times the release rate corresponding to the iodine concentration at the equilibrium value (1.0 μCi/gm DE I-131).

2. Regulatory Position 3 of Appendix E – The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

3. Regulatory Position 4 of Appendix E – The chemical form of radioiodine released from the fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the faulted SG and the unaffected SG to the environment (or containment) are assumed to be 97% elemental and 3% organic.

4. Regulatory Position 5.1 of Appendix E – The SGs are modeled as a single component with all SG tube leakage modeled into that component.

5. Regulatory Position 5.2 of Appendix E – The density used is converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS. For the intact Steam Generators, the primary to secondary leak rate is based on a density of 1.0 gm/cc (cold liquid).

6. Regulatory Position 5.3 of Appendix E – Since a reactor trip is not predicted for this event, the primary-to-secondary leak rate is assumed to continue throughout the 30 day duration of the analysis.

7. Regulatory Position 5.4 of Appendix E – All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation. All of the noble gas released from the primary system to the SGs is assumed to be released directly to the environment.

8. Regulatory Position 5.5.1 of Appendix E – For the steam generators used for plant cooldown, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

9. Regulatory Position 5.5.4 of Appendix E – It is conservatively assumed that the decontamination prescribed for the SGs in Regulatory Guide 1.183 is not applicable to the SGs under power operation. Therefore, no partition factor is applied to the activity as it is transferred from the SG to the turbine. Consistent with the pre-trip treatment of the secondary steam release during the current Steam Generator Tube Rupture at Seabrook, an iodine Decontamination Fraction of 99% will be assigned for the release from the condenser. It is similarly reasonable to assume that the 99% is equally applicable to all particulate released from the condenser. Therefore, the SG tube leakage will be modeled as a release from the RCS to the environment at the condenser location with a 99% filter efficiency for all particulates, and elemental and organic iodine.

10. Regulatory Position 5.6 of Appendix E – Steam generator tube bundle uncovery is not postulated for the SG's for Seabrook.

c. <u>Other Assumptions</u>

1. RG 1.183 does not address secondary coolant activity. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2. For a Letdown Line Rupture event outside of containment, releases from the faulted line are postulated to occur from the Primary Auxiliary Building at the location with the most limiting atmospheric dispersion factors. Releases from the secondary side are postulated to occur from the condenser.

d. Methodology

The dose assessment for this event is comprised of two separate release paths. Path 1 defines the leakage from the double ended rupture of the letdown line in the Primary Auxiliary Building outside of containment with a direct release to the environment. Path 2 defines the release of RCS tube leakage through the secondary side via steam release through the condenser. Since RG 1.183 does not provide any direct guidance regarding analysis of a Letdown Line Rupture, Standard Review Plan (SRP) Section 15.6.2 is used as the primary source of guidance for this analysis. In accordance with SRP 15.6.2, this analysis assumes an accident-generated or concurrent iodine spike in combination with the maximum leakage of primary fluid through the SG tubes into the secondary side.

The accident generated appearance rate for the concurrent iodine spike is computed using the input in Table 15.6-3, with a 500 times multiplier on the normal appearance rate. The modeling of this spike is identical to that modeled for the MSLB concurrent spike case.

The Letdown Line Rupture flow rate is modeled as 140 gpm (at 62 lb_m/ft^3) for 30 minutes with a flashing fraction of 0.1815 as computed using the RG 1.183 guidance from position 5.4 of Appendix A for ECCS leakage for leakage at 380°F and 2235 psia. All of the noble gas in the letdown line rupture flow is released to the environment and the non-noble gas activity in the 0.1815 flashing fraction is assumed to be released (consistent with SRP 15.6.2 guidance).

For this event, the Control Room ventilation system modes of operation are:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. In this emergency mode, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

e. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the pathway into the control room. These *X/Qs* are summarized in Tables 2R-2 and 2R-3.

For the secondary side release case, releases from the SGs are assumed to occur from the condenser.

For the EAB and the LPZ dose the *X/Q* factors are provided in Appendix 2Q.

Reg. Guide 1.183 does not provide any direct guidance for the acceptance criteria for this event. However, the SRP states that the acceptance criteria is "a small fraction" of the 10 CFR 100 values which is further described as 10% of the limit. In applying the AST methodology to the letdown line break that same 10% interpretation is applied to the 10 CFR part 50.67 limits for the LPZ and EAB dose. The acceptable dose limit for the Control Room (CR) is that specified in 10 CFR 50.67. For a Letdown Line Rupture, these are interpreted as:

Area	Dose Criteria	
EAB	2.5 rem TEDE	(for the worst two hour period)
LPZ	2.5 rem TEDE	(for 30 days)
Control Room	5 rem TEDE	(for 30 days)

The radiological consequences of the Letdown Line Rupture event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 15.6-5, the radiological consequences of the Letdown Line Rupture event are all within the appropriate acceptance criteria.

15.6.2.4 Conclusion

The doses which have been calculated for the accident of a small line break outside the Containment are within regulatory limits.

15.6.3 Steam Generator Tube Rupture

15.6.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. This event is considered an ANS Condition IV event, a limiting fault (see Subsection 15.0.1). The accident is assumed to take place at power with the reactor coolant activity corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in activity in 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 dump system, discharge of activity to the atmosphere takes place via the steam generator safety and/or power-operated relief valves.

Since the steam generator tube material is Inconel, a highly ductile material, 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 unit operation.

The operator will determine that a steam generator tube rupture has occurred, and will identify and isolate the ruptured steam generator to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured 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, (References 38, 39). Sufficient indications and controls are provided so the operator can carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

a. Pressurizer low-pressure and low-level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the affected steam generator is reduced due to the additional break flow which is now being supplied to that unit.

b. Continued loss of reactor coolant inventory leads to a reactor trip signal generated by Overtemperature ΔT, low pressurizer pressure, or manual operator action. Resultant plant cooldown following reactor trip leads to a rapid change of pressurizer level, and the safety injection signal, initiated manually by the operator or automatically by low pressurizer pressure, follows soon after the reactor trip. The safety injection signal automatically terminates normal feedwater supply and initiates emergency feedwater addition.

c. The steam generator blowdown liquid monitor and the condenser off-gas radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system.

d. The reactor trip automatically trips the turbine and, if offsite power is available, the steam dump valves open permitting steam dump to the condenser and atmosphere. In the event of a coincident station blackout, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere (when the pressure reaches to the setpoint) through the steam generator safety and/or power-operated relief valves.

e. Following reactor trip, the continued action of emergency feedwater supply and borated safety injection flow (supplied from the refueling water storage tank) provide a heat sink which absorbs some of the decay heat. Thus, steam bypass to the condenser, or in the case of loss of offsite power, steam relief to the atmosphere, is attenuated during the time interval in which the recovery procedure leading to identification of the ruptured steam generator is being carried out.

f. Safety injection flow results in increasing pressurizer water level. The time after trip at which the operator can clearly see returning level in the pressurizer is dependent upon the amount of operating auxiliary equipment.

The sequence of events for the steam generator tube rupture thermal and hydraulics analysis is given in Table 15.6-7.

The time dependent parameters for the Steam Generator Tube Rupture thermal and hydraulics analysis which is bounded by the radiological analysis presented in Section 15.6.3.3 are listed in the following figures:

15.6.3.2 Analysis of Effects and Consequences

a. Method of Analysis

Two scenarios are considered, one leading to minimum margin to overfill of the ruptured steam generator, and the other to maximum radiological consequences.

In estimating the mass transfer from the Reactor Coolant System through the broken tube for the scenario with maximum radiological consequences the following assumptions are made:

1. Reactor trip occurs automatically as a result of an OTDT signal.
2. Following the initiation of the safety injection signal, two centrifugal, high head safety injection and two charging pumps are actuated and continue to deliver flow until the emergency instructions for a tube rupture accident indicate that the operator should switch off all but one pump when he has identified the accident and has pressurizer level indication. The analysis considers high head safety injection and charging pumps.
3. The power-operated relief valve on the main steam line from the ruptured steam generator fails in the full open position during the initial attempt by the operator to isolate steam flow from the ruptured steam generator.

4. The operators identify the open power-operated relief valve and manually isolate it by locally closing the upstream block valve within 20 minutes of the initial attempt to close the valve. The implementation of further recovery procedure actions is delayed until the power-operated relief valve on the ruptured steam generator has been isolated.

5. The break flow is terminated by cooldown of the reactor coolant system opening the power-operated relief valves on the intact steam generators, reducing reactor coolant system pressure to a pressure below the pressure of the ruptured steam generator at the end of the cooldown by opening one of the power-operated relief valves on the pressurizer, and stopping safety injection flow.

The assumptions for the overfill scenario are similar except that no degradation of the high head safety injection or charging pumps is assumed. A power-operated relief valve on a main steam line from one of the intact steam generators is assumed to fail to open on demand, reducing the rate of reactor coolant system cooldown. The power-operated relief valve on the main steam line from the ruptured steam generator is assumed to function normally and close upon operator demand when the operators attempt to isolate steam flow from the ruptured steam generator.

b. Recovery Procedure

Symptoms of a tube rupture such as falling pressurizer pressure and level and increased charging pump flow are also symptoms of small steamline breaks and loss-of-coolant accidents. It is therefore important to determine that the accident is a rupture of a steam generator tube to 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, the level in one steam generator will rise more rapidly than in the others. This is a unique indication of a tube rupture accident. Also, this accident could be identified by a steam generator blowdown radiation alarm. The recovery procedure includes isolation of the ruptured steam generator and unit cooldown.

After the Residual Heat Removal System is placed in operation, the condensate accumulated in the secondary system can be analyzed and processed as required.

There is ample time available to carry out the above recovery procedures so that isolation of the affected steam generator is established before water level rises into the main steam pipes. The available time scale is improved by the termination of emergency feedwater flow to the ruptured steam generator and the regulation of pressurizer water level with only one charging pump operating. Normal operator vigilance therefore assures that excessive water level will not be attained.

c. Results

The results of the scenario leading to minimum margin to overfill of the ruptured steam generator show that operator implementation of the steam generator tube rupture recovery procedures results in termination of the break flow before water level in the ruptured steam generator rises into the main steam pipes. The results of the radiological consequences analysis are described in Section 15.6.3.3 below. The thermal hydraulic results of this accident are less severe than that for a LOCA small break (see Subsection 15.6.5).

15.6.3.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event is assumed to be caused by the instantaneous rupture of a Steam Generator tube that relieves to the lower pressure secondary system. No melt or clad breach is postulated for the Seabrook SGTR event.

A single ASDV is assumed to stick open in the Seabrook SGTR analysis. Two stuck open ASDV scenarios are considered. Case 1 assumes that a single ASDV fails open when level reaches 33% in the affected SG. Case 2 assumes that a single ASDV fails open 3 minutes following reactor trip. The failed open ASDV is assumed to be reclosed 20 minutes after failing open.

b. Compliance with RG 1.183 Regulatory Positions

The SGTR dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," as discussed below:

1. Regulatory Position 1 – No fuel damage is postulated to occur for the Seabrook SGTR event.

2. Regulatory Position 2 – No fuel damage is postulated to occur for the Seabrook SGTR event. Two cases of iodine spiking are assumed.

3. Regulatory Position 2.1 – One case assumes a reactor transient prior to the postulated SGTR that raises the primary coolant iodine concentration to the maximum allowed by Tech Specs, which is a value of 60.0 μCi/gm DE I-131 for the analyzed conditions. This is the pre-accident spike case.

4. Regulatory Position 2.2 – One case assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Tech Spec limit of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

5. Regulatory Position 3 – The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

6. Regulatory Position 4 – Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 – The primary-to-secondary leak rate is apportioned between the SGs as specified by Tech Specs (1.0 gpm total, 500 gpd to any one SG). The tube leakage is conservatively apportioned as 313.33 gpd to the faulted SG and 1126.67 gpd total to the other three SGs in order to maximize dose consequences.

8. Regulatory Position 5.2 – The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS. For the intact Steam Generators, the primary-to-secondary leak rate is based on a density of 1.09 gm/cc (cold liquid).

9. Regulatory Position 5.3 – The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 48 hours. The release of radioactivity from the SGs is assumed to continue until shutdown cooling is in operation and steam release from the SGs is terminated (RHR initiation at 8 hours).

10. Regulatory Position 5.4 – The release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power (LOOP).

11. Regulatory Position 5.5 – All noble gases released from the primary system are assumed to be released to the environment without reduction or mitigation.

12. Regulatory Position 5.6 – Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:

    - Appendix E, Regulatory Position 5.5.1 – Tube uncovery is not postulated for this event; therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing for all steam generators.

    - Appendix E, Regulatory Position 5.5.2 – A portion of the primary-to-secondary ruptured tube flow through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary. The portion that flashes immediately to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.

    - Appendix E, Regulatory Position 5.5.3 – All of the SG tube leakage and ruptured tube flow that does not flash is assumed to mix with the bulk water.

    - Appendix E, Regulatory Position 5.5.4 – The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionucludes in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

    - Appendix E, Regulatory Position 5.6 – Steam generator tube bundle uncovery is not postulated for this event for Seabrook.

f. Other Assumptions

1. RCS and SG volume are assumed to remain constant throughout both the pre-accident and the accident-induced iodine spike SGTR events.

2. Data used to calculate the iodine equilibrium appearance rate are provided in Table 15.6-10.

g. Methodology

Input assumptions used in the dose consequence analysis of the SGTR event are provided in Table 15.6-6. This event is assumed to be caused by the instantaneous rupture of a steam generator tube releasing primary coolant to the lower pressure secondary system. In the unlikely event of a concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum. Valves in the condenser bypass lines would automatically close to protect the condenser thereby causing steam relief directly to the atmosphere from the ASDVs or MSSVs. This direct steam relief continues until it is terminated by initiation of RHR cooling (8 hours).

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the faulted SG relief valves. Additional activity, based on the proposed primary-to-secondary leakage limits, is released via the MSSVs or ASDVs until the RHR system is placed in operation to continue heat removal from the primary system.

No fuel melt or clad breach is postulated for the SGTR event. Consistent with RG 1.183 Appendix F, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike, and (2) maximum accident-induced, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated SGTR event. The primary coolant iodine concentration is increased to the maximum value of 60 μCi/gm DE I-131 permitted by Technical Specification 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 15.6-9. Primary coolant is released into the ruptured SG by the tube rupture and by a fraction of the total proposed allowable primary-to-secondary leakage. Activity is released to the environment from the ruptured SG via direct flashing of a fraction of the released primary coolant from the tube rupture and also via steaming from the ruptured SG ASDVs. The unaffected SGs are used to cool down the plant during the SGTR event. Primary-to-secondary tube leakage is also postulated into the intact SGs. Activity is released via steaming from the SG MSSVs/ASDVs until the decay heat generated in the reactor core can be removed by the RHR system at 8 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-induced spike, the postulated SGTR event induces an iodine spike. The RCS activity is initially assumed to be 1.0 μCi/gm DE I-131 as allowed by Technical Specifications. Iodine is released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 15.6-11. All other release assumptions for this case are identical to those for the pre-accident spike case.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake to the Control Room during this mode is 1000 cfm of unfiltered fresh air.

- After the start of the event, the Control Room normal air intake is conservatively assumed to isolate on a CR intake radiation monitor signal. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time, load sequencing, and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution consists of 600 cfm of filtered makeup flow through the worst of the two emergency intakes, 300 cfm of unfiltered inleakage and 390 cfm of filtered recirculation flow.

- 20 cfm of unfiltered inleakage was assumed to enter the Control Room via the CR fire exit and 280 cfm was assumed to enter via the Diesel Building.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95% for elemental and organic iodine.

h. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

Releases from the intact and faulted SGs are assumed to occur from the MSSV/ASDV that produces the most limiting *X/Qs* when combined with the limiting applicable control room intake.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the SGTR Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the RCS maximum pre-accident iodine spike and the accident-induced iodine spike, based on Tech Spec limits, are analyzed. In addition, two ASDV failure cases are analyzed. As shown in Table 15.6-12, the radiological consequences of the Seabrook SGTR event for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.6.3.4 Conclusions

The offsite doses from a postulated steam generator tube rupture at Seabrook Station are well within the exposure guideline values. Thus, the occurrence of this postulated accident will not result in an undue hazard to the general public.

15.6.4 Spectrum of BWR Steam System Piping Failures Outside of Containment

Not applicable to Seabrook.

15.6.5 Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary (see Section 5.2). For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an American Nuclear Society (ANS) Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see Section 15.0.1).

Ruptures of small cross-section will cause expulsion of the coolant at a rate which can be accommodated by the high head safety injection pumps and 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 containment contains the fission products present in it.

A minor pipe break (small break), as considered in this section, is defined as a rupture of the RCS piping with a cross-sectional area less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is considered an American Nuclear Society (ANS) Condition III event, which is a fault which may occur very infrequently during the life of the plant.

It must be demonstrated that there is a high level of probability that the Acceptance Criteria for the LOCA as described in the 10 CFR 50.46 (Reference 1) are met.

1) There is a high level of probability that the peak cladding temperature (PCT) shall not exceed 2200°F.

2) 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.

3) The maximum calculated local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

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.

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.

These criteria were established to provide a significant margin in Emergency Core Cooling System (ECCS) performance following a LOCA. Reference 2 presents a study in regards to the probability of occurrence of RCS pipe ruptures.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the Acceptance Criteria limits than large breaks.

15.6.5.2 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 15.6-32, and show compliance with the acceptance criteria.

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 27.5 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 (except for pressurizer, which is at T_{sat}) 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. 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 the Seabrook Station Large Break LOCA analysis using the Best-Estimate 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.

15.6.5.2.1 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 634.7 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 RCS pressure reaches approximately 40 psia. 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 safety injection from the centrifugal charging pump (CCP), the safety injection pump (SIP) and the residual heat removal pump (RHR) aids in the filling of the downcomer and core and subsequently supply water to help maintain a full downcomer and complete the reflooding process.

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 RHR pumps and returned to the RCS cold legs. Figure 15.6-3 contains a schematic of a representative sequence of events for the Seabrook Station Best-Estimate large break LOCA transient.

For the Best-Estimate large break LOCA analysis, one ECCS train, including one centrifugal charging pump (CCP), one safety injection pump (SIP) and one low head pump (RHR), starts and delivers flow through the injection lines. The accumulator and safety injection flows from the broken loop were assumed to spill to containment. All 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 9) and is conservative for the large break LOCA.

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

15.6.5.2.2 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 1). 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 5).

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 6). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis.

A LOCA evaluation methodology for three- and four-loop 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 7). The methodology is documented in WCAP-12945, "Code Qualification Document (CQD) for Best Estimate LOCA Analysis" (Reference 8).

A 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 8).

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. The fuel parameters are generated using the Westinghouse fuel performance code (PAD, 4.0, Reference 4).

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.

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 (References 3 and 11) 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 15.6-26 through 15.6-28.

The methods used in the application of WCOBRA/TRAC to the large break LOCA are described in Reference 8. 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. 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, WCOBRA/TRAC model of the Seabrook Station 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 plenum of the reactor vessel or the ECCS injection configuration.

2. Determination of Plant Operating Conditions

   In this step, the expected or desired range or 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 turns are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. The most limiting input conditions, based on these confirmatory runs, are then combined into a single transient, which is then called the "reference transient."

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.

   Next, a series of transients are performed which vary the power distribution, taking into account all possible power distributions during normal plant operation.

   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).

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 8). 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). 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, as described in Section 15.6.5.2.3.6 (Uncertainty Evaluation).

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 PCT margin.

There are three major uncertainty categories or elements:

- Initial conditions 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

$$PCT_i = PCT_{REF,i} + \Delta PCT_{IC,i} + \Delta PCT_{PD,i} + \Delta PCT_{MOD,i} \qquad (15.6.5.2\text{-}1)$$

Where,

$PCT_{REF,i}$ = **Reference transient PCT:** The reference transient PCT is calculated using WCOBRA/TRAC at the nominal conditions identified in Table 15.6-29, for the blowdown, first 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.

$\Delta PCT_{MOD,i}$ = **Model bias and uncertainty:** This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict 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 the $PCT^{95\%}$.

15.6.5.2.3 Large Break LOCA Analysis Results

A series of WCOBRA/TRAC calculations were performed using the Seabrook Station input model, 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.

15.6.5.2.3.1 Large Break LOCA Reference Split Break Transient Description

The plant-specific analysis performed for the Seabrook Station confirmed that the split break is more limiting than the double-ended cold leg guillotine (DECLG) break. Because split break is limiting, this split break transient (CD = 2.0) will now be known as the reference split break transient. The plant conditions used in the reference split break transient are listed in Table 15.6-29. Since many of these parameters are at their bounded values, the calculated results are a conservative representation of the response to a large break LOCA. The following is a description of the reference split break 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 down-ward core flow phase. These are followed by the refill, reflood 1, reflood 2 and long term cooling phases. The important phenomena occurring during each of these phases are discussed for the reference split break transient. The results are shown in Figures 15.6-7 through 15.6-19.

<u>Criteria Heat Flux (CHF) Phase (~ 0-2 seconds)</u>

Immediately following the cold leg rupture, the break discharge is subcooled and high flow rate, 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 15.6-7 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.

<u>Upward Core Flow Phase (~ 2-11.5 seconds)</u>

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. Figure 15.6-8 shows the break flowrate for the reference split break transient. 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 15.6-9 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 (~ 11.5-28 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. Figures 15.6-10 and 15.6-11 show the vapor flow at the mid-core of channels 13 and 15. While liquid and entrained liquid flows also provide core cooling, the vapor flow in 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 (~ 28-33 seconds)

The core experiences a nearly adiabatic heatup as the lower plenum fills with ECCS water. Figure 15.6-12 shows the lower plenum liquid level. This phase ends when the ECCS water enters the core and entrainment begins, with a resulting improvement in heat transfer. Figure 15.6-13 and Figure 15.6-14 show the liquid flows from the accumulator and the safety injection from an intact loop (Loop 1).

First Reflood Phase (~ 33-50 seconds)

The accumulators are emptying and nitrogen enters into the system (Figure 15.6-13). This forces water into the core which then boils as the lower core region begins to quench, causing repressurization. The repressurization is best illustrated by the reduction in pumped SI flow (Figure 15.6-14). During this time, core cooling may be increased.

Second Reflood Phase (~ 50 seconds – end)

The system then settles into a gravity driven reflood which exhibits lower core heat transfer. Figures 15.6-15 and 15.6-16 show the core and downcomer liquid levels. Figure 15.6-17 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 15.6-18 shows the movement of the PCT location. As the vessel continues to fill, the PCT location is cooled and the PCT heatup is terminated (Figures 15.6-7 and 15.6-9).

Long Term Core Cooling

At the end of the WCOBRA/TRC 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 levels 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.

The reference split break transient resulted in a first reflood PCT of 1570°F and a second reflood PCT of 1567°F.

15.6.5.2.3.2 Confirmatory Sensitivity Studies

A number of sensitivity calculations were carried out to investigate the effect of the key LOCA parameters, and to determine the reference transient. 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 15.6-30 and 15.6-31. The results of these analyses lead to the following conclusions:

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

2. Modeling loss-of-offsite-power (LOOP) results in a higher PCT than no loss-of-offsite-power (no-LOOP).

3. Modeling the maximum value of vessel average temperature (T_{avg} = 589.1°F) results in a higher PCT than minimum value of vessel average temperature (T_{avg} = 571.0°F).

4. Modeling the minimum power fraction (P_{LOW} = 0.2) in the low power/periphery channel of the core results in a higher PCT than maximum power fraction. (P_{LOW} = 0.6).

5. For the split break confirmatory study, it was determined that the limiting split break area is 2 times the area of a cold leg pipe (C_D = 2.0).

15.6.5.2.3.3 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 and water volume).

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 ($\Delta PCT_{IC,i}$).

15.6.5.2.3.4 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, 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.

The power distribution parameters used for the reference transient are biased to yield a relatively high PCT. The reference transient uses a lightly higher $F_{\Delta H}$ value (1.683) than the Tech Spec $F_{\Delta H}$ value (1.65), a skewed to the top power distribution, and a F_Q (2.2) 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 sensitivity results indicated that power distributions with peak powers shifted towards the top of the core produced higher PCTs.

The calculated results were used to develop response surfaces, as described in Step 4 of Section 15.6.5.2.2, 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, ($\Delta PCT_{PD,i}$).

15.6.5.2.3.5 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 database which could be used for response surface generation. The run matrix and ranges of the break flow parameters are described in Reference 8. The limiting split break was also identified using the methodology described in Reference 8. The results of these studies indicated that the split break calculation with an area equal to 2 times the cold leg area results in the highest PCT. This requires that the effect of broken loop resistance and condensation must be reevaluated for the limiting split area.

The calculated results were used to develop response surfaces as described in Section 15.6.5.2.2, 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 ($\Delta PCT_{MOD,i}$).

15.6.5.2.3.6 Uncertainty Evaluation and Results

The PCT equation was presented in Section 15.6.5.2.2. 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 ΔPCT_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:

1. Generate a random value of each ΔPCT element.

2. Calculate the resulting PCT using Equation 15.6.5.2-1.

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

For the Seabrook Station, the results of this assessment showed the split break to potentially be limiting. Additional split break calculations were then performed, a more detailed description of $\Delta PCT_{MOD,i}$ was developed, and steps 1 through 3 repeated for the limiting split break. This analysis confirmed the split break to be the limiting break type. As the result of this analysis, the split break is used in the final verification step and the limiting split break transient (CD = 2.0) becomes the reference split break transient for the Seabrook Station.

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 global models). Predictions using Equation 15.6.5.2-1 are compared to this data, and additional biases and uncertainties are applied.

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

The results for the Seabrook Station are given in Table 15.6-32, which shows the limiting first reflood 95th percentile PCT ($PCT^{95\%}$) of 1789°F. As expected, the difference between the 95th percent value and the average value increases with increasing time, as more parameter uncertainties come into play.

15.6.5.2.3.7 Evaluation

The base analysis discussed in Sections 15.6.5.2.3.1 to 15.6.5.2.3.6 is for non-IFBA fuel. An analysis of IFBA fuel was performed independently, utilizing the HOTSPOT code and a high PCT case. The analysis result indicated that IFBA fuel is bounded by non-IFBA fuel.

15.6.5.2.4 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 for the Seabrook Station 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 15.6-32 indicate that this regulatory limit has been met with a reflood $PCT^{95\%}$ of 1789°F.

2) 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 0.003 times the maximum hypothetical amount, which meets the regulatory limit.

3) 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 3.53 percent is calculated, which meets the regulatory limit.

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 8) 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 performed for Seabrook Station 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.

15.6.5.2.5 Plant Operating Range

The expected PCT and its uncertainty developed above are valid for a range of plant operating conditions. In contrast to current 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 15.6-33 summarizes the operating ranges for the Seabrook Station. If operation is maintained within these ranges, the LOCA analysis is considered to be valid.

15.6.5.3 Small Break LOCA

a. Sequence of Events and Systems Operations

The Seabrook Small Break Loss-of-Coolant Accident (SBLOCA) analysis was performed to support the Power Uprate[29]. Pertinent analysis assumptions include: licensed core power of 3659 MWt (including calorimetric uncertainty), 10% uniform SGTP, maximum peaking factor ($F_Q(Z)$) envelope of 2.50, hot channel enthalpy rise factor $F_{\Delta H}$ of 1.65 and RFA (w/IFMs) fuel. A break spectrum of 3 inch, 4 inch, and 6 inch breaks was analyzed, resulting in the 4 inch case being limiting with a peak cladding temperature (PCT) of 1373°F[29].

The results of this Small Break ECCS analysis, utilizing the currently approved NOTRUMP Evaluation Model [16, 17, 30], have shown that Seabrook remains in compliance with the requirements of 10 CFR 50.46.

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. Loss-of-Offsite Power (LOOP) is assumed to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-low-pressure setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

- Reactor trip and borated water injection, together with void formation, cause a rapid reduction of nuclear power to a residual level corresponding to delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. However, an average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical. In addition, in the small break LOCA analysis, credit is taken for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position.

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

For small break LOCAs, the most limiting single active failure is the one that results in the minimum emergency core cooling system (ECCS) flow delivered to the RCS. This has been determined to be the loss of an emergency power train which results in the loss of one complete train of ECCS components. This means that credit can be taken for only one centrifugal charging pump (CCP), one safety injection pump (SIP), and one residual heat removal (RHR) (or low head) pump. During the small break transient, one ECCS train is assumed to start and deliver flow through the injection lines (one of each loop) with one branch injection line (SIP and CCP) spilling to the RCS backpressure. RHR flow is not modeled for small break LOCAs because the pressure will not fall below the RHR cut-in pressure before the end of the transient. To minimize delivery to the reactor, the branch line chosen to spill is selected as the one with the minimum resistance. In addition, the SIP and CCP performance curves were degraded by 10%.

1. Description of Transient

   The sequence of events following a small break LOCA are presented in Table 15.6-1.

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. After the small break LOCA is initiated, reactor trip occurs due to a low pressurizer signal. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as the pumps coast down following LOOP. Upward flow through the core is maintained. However, the core flow is not sufficient to prevent a partial core uncovery. Subsequently, the ECCS provides sufficient core flow to cover the core.

During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. Continued heat addition to the secondary system results in increased secondary system pressure which leads to steam relief via the main steam safety valves. Makeup to the secondary is automatically provided by the emergency feedwater pumps. The safety injection signal isolates normal feedwater flow by closing the main feedwater isolation, control, and bypass valves and initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to approximately 600 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. However, for most small breaks the vessel mixture level starts to increase, covering the fuel with ECCS pumped injection before accumulator injection begins.

b. Core and System Performance

1. Evaluation Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Reference 1).

For small breaks (less than 1.0 ft^2) the NOTRUMP digital computer code (References 16, 17, and 30) is employed to calculate the transient depressurization of the RCS as well as to describe the mass and energy of the fluid 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 drift flux calculations with multiple-stacked fluid nodes and regime-dependent heat transfer correlations. Also, safety injection into the broken loop is modeled using the COSI condensation model [30]. The NOTRUMP small break LOCA ECCS evaluation model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 18).

The RCS model is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, while the intact loops are lumped into a single second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multimode 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 LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Clad thermal analyses are performed with the LOCTA-IV code (Reference 19) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. (Figure 15.6-6).

Figure 15.6-4 depicts the hot rod axial power shape used to perform the small break LOCA analysis presented here. The 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 LOCA because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break analysis assumes that the core continues to operate at full power until the control rods are completely inserted. For conservatism, it is assumed that the most reactive RCCA does not insert.

The safety injection performance, as modeled in the small break analysis, is presented in Figure 15.6-49. Conservatively, 10% head degradation is assumed for the charging and safety injection pumps.

Schematic representation of the computer code interface is given in Figure 15.6-6.

2. Input Parameters and Initial Conditions

Table 15.6-13 lists important input parameters and initial conditions used in the analysis.

The bases used to select the numerical values that are important parameters to the analysis have been conservatively determined from extensive sensitivity studies (See References 13, 14, and 15). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

3. Results

NUREG-0737 Section II.K.3.31 (Reference 20) requires a plant specific small break LOCA analysis using an Evaluation Model revised per Section II.K.3.30. In accordance with NRC generic letter 83-35 (Reference 21), generic analyses using, NOTRUMP (References 16 and 17) were performed and are presented in Reference 22. Those results demonstrate that in a comparison of cold leg, hot leg and pump suction leg break locations, the cold leg break of less than 10 inches in diameter is limiting.

Therefore, a range of small break analyses is presented which establishes the limiting break size. The results of these analyses are summarized in Table 15.6-1 and Table 15.6-15.

It was determined that, because of the low calculated PCT, rod burst and blockage effects would not have a significant effect on the small break results for Seabrook Station. Therefore, a fuel assembly burnup sensitivity study was not required.

Figures 15.6-4, 15.6-34 through 15.6-45, 15.6-49 present the principal parameters of interest for the small break LOCA ECCS analyses. For all cases analyzed the following transient parameters are presented:

(a) RCS pressure

(b) Core mixture height

(c) Hot spot clad temperature

For the limiting break analyzed, the following additional transient parameters are presented:

(a) Core steam flow rate

(b) Core heat transfer coefficient

(c) Hot spot fluid temperature

The limiting break PCT is 1373°F, which is less than the Acceptance Criteria limit of 2200°F of 10 CFR 50.46.

The small break LOCA results are well below all Acceptance Criteria limits of 10 CFR 50.46 and in all cases are not limiting when compared to the results presented for large breaks.

15.6.5.4 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event is assumed to be caused by an abrupt failure of the main reactor coolant pipe and the ECCS fails to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity is released from the containment and from there, released to the environment by means of containment leakage and leakage from the ECCS.

b. Compliance with RG 1.183 Regulatory Positions

The LOCA dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," as discussed below:

1. Regulatory Position 1 – The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided in Table 15C-1. The core inventory release fractions for the gap release and early in-vessel damage phases of the LOCA are consistent with Regulatory Position 3.2 and Table 2 of RG 1.183.

2. Regulatory Position 2 – The sump pH is controlled at a value greater than 7.0 per UFSAR Section 6.5.2.2. Therefore, the chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.

3. Regulatory Position 3.1 – The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase.

4. Regulatory Position 3.2 – Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.23 hr^{-1}. This removal is credited in the sprayed and unsprayed regions. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed (based on the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983) for all aerosols in the unsprayed regions with no credit of natural deposition of aerosols in the sprayed regions. No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3 – Containment spray provides coverage to 85.4% of the containment. Therefore, the Seabrook containment building atmosphere is not considered to be a single, well-mixed volume. The containment is divided into sprayed and unsprayed regions. A mixing rate of two turnovers of the unsprayed region per hour is assumed.

The SRP limits the spray removal coefficient for elemental iodine to 20 hr^{-1}; therefore, although a higher value was calculated, 20 hr^{-1} was used for the elemental iodine spray removal coefficient. In addition, the SRP and Reg. Guide 1.183 specify a maximum decontamination factor of 200 for spray removal of elemental iodine. The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is based on the maximum airborne elemental iodine concentration in the containment. The time for the containment sprays to reach an elemental iodine decontamination factor of 200 was determined by running a containment leakage case without environment leakage paths.

Radioactive decay and natural deposition of iodine were conservatively left on as removal mechanisms contributing to the decontamination factor. Due to mixing between the sprayed and unsprayed regions of containment, the iodine activity in both containment regions was included in the determination of the time required to reach a decontamination factor of 200. The decontamination factor for elemental iodine reaches 200 at just over 2.92 hours.

The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 5.75 hr^{-1}, the time of a DF of 50 is computed with the same model used to determine the elemental iodine DF of 200. The time for containment spray to produce an aerosol decontamination factor of 50 with respect to the containment atmosphere is just over 3.56 hours.

6. Regulatory Position 3.4 – Reduction in airborne radioactivity in the containment by filter recirculation systems is not assumed in this analysis.

7. Regulatory Position 3.5 – Not applicable to Seabrook.

8. Regulatory Position 3.6 – Not applicable to Seabrook.

9. Regulatory Position 3.7 – A containment leak rate of 0.15% per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.075% per day of the containment air.

10. Regulatory Position 3.8 – Routine containment purge is considered in this analysis. The purge release evaluation assumes that 100% of the radionuclide inventory in reactor coolant system liquid (based on Technical Specification RCS equilibrium activity) is released to the containment at the initiation of the LOCA. The purge system is isolated before the onset of the gap release phase.

11. Regulatory Position 4.1 – Leakage from containment collected by the secondary containment is processed by ESF filters prior to an assumed ground level release.

12. Regulatory Position 4.2 – Leakage into the secondary containment is assumed to be released directly to the environment as a ground level release prior to drawdown of the secondary containment at 8 minutes.

13. Regulatory Position 4.3 – The containment enclosure emergency air cleaning system is credited as being capable of maintaining a negative pressure with respect to the outside environment considering the effect of high windspeeds and LOCA heat effects on the annulus as described in Sections 6.5.1.1 and 6.5.1.3.

14. Regulatory Position 4.4 – No credit is taken for dilution in the secondary containment volume.

15. Regulatory Position 4.5 – 60% of the primary containment leakage is assumed to bypass the secondary containment. This bypass leakage is released from containment without filtration.

16. Regulatory Position 4.6 – The containment enclosure emergency air cleaning system is credited as meeting the requirements of RG 1.52 and Generic Letter 99-02 per Section 6.5.1.3 and Table 6.5-1.

17. Regulatory Position 5.1 – Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.

18. Regulatory Position 5.2 – Leakage from the ESF system is taken as two times 24 gallons per day for a total leakage rate of 48 gallons per day. The leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continues for the 30-day duration. Backleakage to the Refueling Water Storage Tank is also considered separately as two times the measured leakage value of 0.47975 gpm for a total leakage rate of 0.9595 gpm.

19. Regulatory Position 5.3 – With the exception of the iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

20. Regulatory Position 5.4 – A flashing fraction of 4.7% was determined based on the temperature of the containment sump liquid at the time recirculation begins. The iodine available for release at the time recirculation begins is based on expected sump pH history and temperature (see the Release Inputs in the Methodology section below). All of the iodine available for release is assumed to become airborne and leak directly to the environment from the initiation of recirculation through 30 days. For ECCS leakage back to the RWST, the analysis demonstrates that the temperature of the leaked fluid will cool below 212°F prior to release from the tank.

21. Regulatory Position 5.5 – The iodine available for release at the time recirculation begins is based on expected sump pH history and temperature (see the Release Inputs in the Methodology section below). The amount of iodine that becomes airborne is assumed to be 10% of the total iodine available and leak directly to the environment from the initiation of recirculation through 30 days. For the ECCS leakage back to the RWST, the sump and RWST pH history and temperature are used to evaluate the amount of iodine that enters the RWST air space.

22. Regulatory Position 5.6 – The temperature and pH history of the sump and RWST are considered in determining the radioiodine available for release and the chemical form. Credit is taken for hold-up and dilution of activity in the RWST as allowed by Regulatory Position 5.6. No credit for ESF filtration of the RWST leakage is taken. Filtration of non-RWST ECCS leakage is credited.

23. Regulatory Position 6 – Not applicable to Seabrook.

24. Regulatory Position 7 – Containment purge is not considered as a means of combustible gas or pressure control in this analysis; however, the effect of routine containment purge before isolation is considered.

c. Methodology

For this event, the Control Room ventilation system cycles through two modes of operation. Inputs and assumptions fall into three main categories: Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

For the purposes of the LOCA analyses, a major LOCA is defined as a rupture of the RCS piping, including the double-ended rupture of the largest piping in the RCS, or of any line connected to that system up to the first closed valve. Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection system signal is actuated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. The following measures will limit the consequences of the accident in two ways:

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

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

d. Release Inputs

The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and is provided as Table 15C-1. The source term represents end of cycle conditions assuming enveloping initial fuel enrichment and an average core burnup of 45,000 MWD/MTU.

For the first 24 hours, the containment is assumed to leak at a rate of 0.15% of the containment air per day. Per RG 1.183, Regulatory Position 3.7, the primary containment leakage rate is reduced by 50% at 24 hours into the LOCA to 0.075%/day based on the post-LOCA primary containment pressure history.

The ESF leakage to the auxiliary building is assumed to be 48 gpd based upon two times the current value of 24 gpd. The temperature of the leakage is based on the sump temperature at and after the time recirculation begins (255°F in the sump at the time recirculation begins). The leakage is assumed to start at 26 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that all of the non-particulate iodine available for release is released from the leaked liquid. Based on sump pH history and pH control (pH is greater than 7 at the time recirculation begins), the iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form during the time of interest for this analysis. This analysis conservatively assumes a 10% iodine release with a chemical form of 97% elemental and 3% organic.

The ECCS backleakage to the RWST is assumed to be 0.9595 gpm. The leakage is assumed to start at 26 minutes into the event when recirculation starts and continue throughout the 30-day period. Note that based on the leakage rate and the size of the piping, the leakage would not reach the RWST for an extended period of time after recirculation begins. This time period is conservatively not credited for determining when the leakage reaches the RWST (i.e., the leakage is assumed to reach the RWST instantaneously). Based on sump pH history and pH control (pH is greater than 7 at the time recirculation begins), the iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form during the time of interest for this analysis.

Based upon the initial RWST pH of 7.1 at the start of recirculation, and based on information provided in NUREG-5950, it is expected that no elemental iodine will be regenerated in the RWST. However, for this analysis it was conservatively assumed that 1% of the particulate iodine would be converted to elemental iodine in the RWST. This conversion fraction is conservatively assumed to exist throughout the event even though the pH of the RWST would increase during the course of the event.

The elemental iodine generated in the RWST is assumed to become volatile and partition between the liquid and vapor space in the RWST based upon the temperature dependent partition coefficient for elemental iodine as presented in NUREG-5950. The particulate portion of the leakage is assumed to be retained in the liquid phase of the RWST since no boiling occurs in the RWST. The release of the activity from the vapor space within the RWST is calculated based upon the displacement of air by the incoming leakage and the expansion due to the daily heating and cooling cycle of the contents (both air and liquid) of the RWST. The average daily temperature swing of 18.2°F is applied for every 24-hour period for 30 days and no credit is taken for daily cooling. The final iodine release rate from the RWST is implemented via an adjustment to the leakage flow rate from the containment sump, which is applied to the entire iodine inventory in the containment sump, then released directly to the environment. The adjusted release rate is determined as follows:

$$Adjusted\ release\ rate = \left[\frac{(Leaked\ Volume\ x\ Iodine\ Fraction) / RWST\ Liquid\ Volume}{Partition\ Coefficient\ (I_2)}\right] x\ Air\ Flow\ Rate$$

where:

Iodine Fraction = 0.01 (Elemental Iodine fraction available for release from the leaked water)

RWST Liquid Volume = Time dependent RWST liquid volume

Partition Coefficient (I_2) = Temperature dependent elemental iodine partition coefficient

Air Flow Rate = Time dependent air flow from RWST based on expansion and displacement

The adjusted release rate presented in Table 15.6-8 is then applied to the entire iodine inventory in the containment sump.

Containment purge is also assumed coincident with the beginning of the LOCA. Since the purge is isolated prior to the initial release of fission products from the core at 30 seconds, only the initial RCS activity (at an assumed 1.0 microcuries per gram DE I-131 and 100/E-bar gross activity) is available for release via this pathway. The release is modeled as an unfiltered release for 5 seconds until isolation occurs.

e. Transport Inputs

During the LOCA event, the activity collected by the secondary containment is assumed to be a filtered ground level release from the plant vent. The activity that bypasses the secondary containment is identified as being leaked via a ground level release from the containment without filtration. The activity from the ECCS leakage enters the secondary containment and is released to the environment via the plant vent after filtration. The activity from the RWST is modeled as an unfiltered ground level release from the RWST.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake during this mode is 1000 cfm of unfiltered fresh air.

- After the start of the event, the Control Room normal air intake is isolated due to a high containment pressure signal. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution consists of 600 cfm of filtered makeup flow through the worst of the two emergency intakes, 150 cfm of unfiltered inleakage and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulates/aerosols, and 95% for elemental and organic iodine.

f. LOCA Removal Inputs

Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.23 hr^{-1}. This removal is credited in the sprayed and unsprayed regions. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region. No natural deposition removal of aerosols is credited in the sprayed regions. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 85.4% of the containment. Therefore, the Seabrook containment building atmosphere is not considered to be a single, well-mixed volume. A mixing rate of two turnovers of the unsprayed region per hour is assumed.

The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is 200 based on the maximum airborne elemental iodine concentration in the containment. The time for the containment sprays to reach an elemental iodine decontamination factor of 200 was determined by running a containment leakage case without environment leakage paths. Radioactive decay and natural deposition of iodine were conservatively left on as removal mechanisms contributing to the decontamination factor. Due to mixing between the sprayed and unsprayed regions of containment, the iodine activity in both containment regions was included in the determination of the time required to reach a decontamination factor of 200. The decontamination factor for elemental iodine reaches 200 at just over 2.92 hours.

The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 5.75 hr^{-1}, the time of a DF of 50 is computed with the same model used to determine the elemental iodine DF of 200. The time for containment spray to produce an aerosol decontamination factor of 50 with respect to the containment atmosphere is just over 3.56 hours.

Filter removal in the Control Room Emergency Mode is simulated using conservative assumptions based on plant design data as listed in Table 15.6-16.

g. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Table 2R-2 and Table 2R-3.

Three pathways for unfiltered inleakage to the control room were considered; inleakage via the diesel building, inleakage via the primary control room entrance (double air lock configuration), and inleakage via the emergency fire exit (two doors in series). A value of 10 cfm is typically assumed for door leakage for normal ingress/egress. However, this flow would be reduced or eliminated by a two-door vestibule. It was conservatively assumed that 20 cfm of total door leakage occurs via the most limiting door. The *X/Qs* for the fire exit are always more limiting than those for the primary control room entrance; therefore, all of the unfiltered inleakage via the doors was assumed to occur at the fire exit. For most release locations, the *X/Qs* for the fire exit are more limiting than the *X/Qs* for the diesel building inleakage. For these cases, the fire exit was considered as a separate path for unfiltered inleakage. In cases where the diesel building is more limiting than the fire exit, all of the unfiltered inleakage was assumed to enter via the diesel building.

For the EAB and LPZ dose analysis, the *X/Q* factors for the appropriate time intervals are used. These *X/Q* factors are provided in Appendix 2Q.

The radiological consequences of the design basis LOCA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. In addition, the MicroShield code, Version 5.05, Grove Engineering, is used to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis.

The post accident doses are the result of four distinct activity releases:

- Containment leakage
- ESF system leakage into the Primary Auxiliary Building
- ESF leakage into the RWST
- Containment Purge at event initiation

The dose to the Control Room occupants includes terms for:

1. Contamination of the Control Room atmosphere by intake and infiltration of radioactive material from the containment and ESF.
2. External radioactive plume shine contribution from the containment and ESF leakage releases. This term takes credit for Control Room structural shielding.

3. A direct shine dose contribution from the Containment's contained accident activity. This term takes credit for both Containment and Control Room structural shielding.

4. A direct shine dose contribution from the activity collected on the Control Room ventilation filters.

As shown in Table 15.6-20, the sum of the results of all four activity releases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.6.6 BWR Transients

Not applicable to Seabrook.

15.6.7 References

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15.7 RADIOACTIVE RELEASE FROM A SYSTEM OR COMPONENT

15.7.1 Radioactive Gaseous Waste System Leak or Failure

15.7.1.1 Identification of Causes and Accident Description

The most limiting waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in the five carbon delay beds. Although there is no credible mechanism by which this could occur, it is considered a limiting fault since it includes the potential for significant amounts of radioactive releases. The Radioactive Gaseous Waste System (RGWS) is discussed in Section 11.3.

Each low pressure (0-2 psig) carbon delay bed in the RGWS is designed to provide 17 hours of krypton (Kr) delay and 12 days of xenon (Xe) delay, which results in a total system delay of 85 hours for krypton and 60 days for xenon. It is assumed that there is no delay time of the noble gases before they reach the carbon delay beds. The gas volume of each carbon delay bed is approximately 20.4 ft^3. The decontamination factor of the beds is discussed in Subsection 12.2.1. The maximum expected inventory in the five carbon delay beds is shown on Table 12.2-27, and is based on 1 percent failed fuel. The design basis is further discussed in Subsection 12.2.1.

The RGWS is designated NNS, nonseismic Category I, in accordance with ANSI/ANS 51.1-1983 and NRC Regulatory Guide 1.143. Also, the system is designed to withstand a hydrogen explosion. The portion of the Waste Processing Building (WPB) which houses the RGWS is seismic Category I. It should be noted that the gas chillers, iodine guard beds, dryers, carbon delay beds and some valves were designed and fabricated as safety Class 3 seismic Category I, prior to the declassification of the RGWS to NNS.

15.7.1.2 Analysis of Effects and Consequences

In the event of a pipe or carbon delay bed failure, noble gases would be released from the carbon delay beds, since the RGWS operates at a slight positive pressure. The quantity of radioactivity released would depend on the failure location, but in all cases would be a small fraction of the total system inventory.

The sequence of events following this failure is shown in Table 15.7-1. The RGWS process stream is monitored continuously for radioactivity and oxygen upstream of the carbon delay beds. Outleakage of hydrogen would be detected and alarmed by the hydrogen monitors in the Waste Processing Building.

The ventilation system for the areas housing the carbon delay beds operates continuously. The WPB ventilating system would remove the radioactive waste gases and exhaust them to the atmosphere via the plant unit vent (see Subsection 9.4.4).

For the conservative analysis, it is assumed that 100 percent of the carbon delay bed's inventory is released to the environment in a two-hour period.

For the realistic case, it is assumed that a failure occurs upstream of the first carbon delay bed resulting in depressurization of the carbon delay beds and release of their inventory to the WPB atmosphere. Should this occur, the operator would take the actions described in Table 15.7-1, and the WPB ventilating system would operate as described in Subsection 9.4.4.

A leak in the hydrogen surge tank in the RGWS is considered an infrequent incident, since it could occur during the lifetime of the plant. The hydrogen surge tank has a volume of 44 ft^3, operates at 150 psig, and is nonnuclear safety class and nonseismic Category I. The WPB ventilating system also includes a separate exhaust system to purge the area housing the hydrogen surge tank, if the hydrogen gas level in the area approaches the low flammable limit. The Purge Exhaust System would dilute the abnormal hydrogen gas release in the hydrogen surge tank cubicle.

15.7.1.3 Radiological Consequences using Alternate Source Term

a. Background

This event involves a major rupture of one of the Radioactive Gaseous Waste System (RGWS) components. This analysis assumes that the ruptured RGWS component contains an inventory equivalent to the activity limit specified in Table 15.7-3. The entire source term is applied to this RGWS component at the beginning of the event. The leak rate from the RGWS to the environment is conservatively modeled to be very high to simulate a major tank rupture, which releases essentially all of the activity to the environment within 2 hours. No hold-up, dilution or filtration of the tank release is assumed. The impact of the release is then computed as it disperses to the offsite receptors. The dose to Control Room operators is computed as the release is modeled to be treated by the Control Room Air Conditioning and Emergency Cleanup system during the 30-day period following the accident.

b. Compliance with RG 1.183 Regulatory Positions

RG 1.183 does not provide direct guidance relative to the Waste Gas system failures. Therefore, this analysis will rely primarily upon the current UFSAR licensing basis for guidance on performance of this event.

c. Methodology

The dose assessment model releases the above-prescribed inventory from the RGWS at a high rate of release. Per the existing analysis assumptions, the contents are released to the environment without any hold up, dilution or filtration over a 2 hour period.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially, the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of unfiltered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

d. Radiological Consequences

The release-receptor point locations are chosen to model the distance from the release point to the Control Room intake. The X/Q values for the various combinations of release points and receptor locations are presented in Tables 2R-2 and 2R-3.

For the EAB and LPZ dose analyses, the X/Q factors are provided in Appendix 2Q.

Reg. Guide 1.183 does not provide any requirement or dose limits for a RGWS failure; therefore, the acceptance criteria are set by the current Seabrook Licensing basis. Therefore, the off-site dose acceptance criteria are established as 10% of the 10 CFR 50.67 limits. The control room dose limits are specified in 10 CFR 50.67. Therefore, the dose limits are:

Area	Dose Criteria	
EAB	2.5 rem TEDE	(for the worst two hour period)
LPZ	2.5 rem TEDE	(for 30 days)
Control Room	5 rem TEDE	(for 30 days)

The radiological consequences of the RGWS failure event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 15.7-4, the radiological consequences of the Radioactive Gaseous Waste System failure are all within the appropriate acceptance criteria.

15.7.1.4 Conclusions

The doses which have been calculated for the radioactive gaseous waste system accident are below regulatory limits

15.7.2 Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)

This analysis evaluates the radiological consequences of the release to the atmosphere of radioactive fission gases, resulting from an unexpected and uncontrolled release of radioactive liquids that are stored or transferred in waste systems, to determine that they are small fractions of the 10 CFR 100 guideline values. The primary sources evaluated consisted of the Liquid Waste System that normally contains and processes the waste liquid before final disposal, as discussed in Section 11.2, and the below-listed systems that may store or handle a radioactive liquid:

- Boron Recovery System (Subsection 9.3.5)
- Steam Generator Blowdown System (Subsection 10.4.8)
- Equipment and Floor Drain System (Subsection 9.3.3)
- Chemical and Volume Control System (Subsection 9.3.4).

A leak or failure of a component in one of the above systems can release some fission gases and/or iodine-contaminated liquids into the building housing the particular component. The impact of the releases is evaluated in this section. For the purpose of these analyses, it is assumed that at the time of the leak or failure, there exist excessive fuel cladding defects on one (1.0) percent with no decay, as discussed in Subsection 11.1.1 and Section 12.2.

15.7.2.1 Identification of Causes and Accident Description

The above-mentioned systems are located in the Primary Auxiliary Building (PAB) and the Waste Processing Building (WPB), both seismic Category I structures.

For the purpose of release to atmosphere, the components that contain undergasified liquid are of major significance. The liquids in the Chemical and Volume Control System and the Boron Recovery System before degasification contain significant fission-gas inventories (see Section 12.2). There are also significant fission gas concentrations in the boron waste storage tank and spent resin sluice tank. In all these systems, components such as valve-stems, pump-seals, etc., are designed with double seals to minimize leakage. At strategic locations, manual bypass valves have been provided in case the main control valve fails and goes under repair. Moreover, the construction materials of the components (as discussed in specific sections) are of high allowable stresses and proper corrosion resistance, or allowance is made.

In spite of the safety features mentioned above, some design basis failures are postulated due to operator error, instrumentation/controls failure, or seismic loads beyond design limits. The leakage of a valve or pump-seal can release only minimal amounts of radioactivity which can be cleaned up and terminated quickly. The rupture of a pipe or tank can release considerable amounts of fission gases into the buildings where they are located. All the liquid-containing tanks (outside Containment) are listed in Table 15.7-15. The tanks which may contain significant quantities of fission gases, along with the appropriate table in Subsection 12.2.1, which gives their radionuclide inventory, are listed below.

- Letdown Degasifier (Subsection 9.3.3) in PAB - Table 12.2-5
- Primary Drain Tank Degasifier (Subsection 9.3.5) in WPB - Table 12.2-13
- Boron Waste Storage Tank (Subsection 9.3.5) in WPB - Table 12.2-11
- Spent Resin Sluice Tank (Section 11.4) in WPB - Table 12.2-15

For the purpose of this analysis, the rupture of only one tank (irrespective of seismic design category) is considered at a time. The backup tank is assumed to be available for the process. All the systems considered in this section are provided with Overpressure relief protection. Therefore, the rupture of a tank or adjacent pipe due to Overpressure is possible only if the relief capability becomes inoperative. Some of the system components are nonseismic Category I and could, therefore, rupture in the event of a major earthquake. Unexpected corrosion beyond the design allowance could also cause failure. However, due to the safety features, plant inspection and maintenance provided, it is highly improbable that failure of a tank will occur more than once during the expected plant life. Therefore, the frequency of occurrence of such a leak or failure is not of any significance.

15.7.2.2 Analysis of Effects and Consequences

All the tanks of the Liquid Waste System are contained within shielded enclosures, the floors of which slope toward drains. Some of the piping containing radioactive liquid waste is routed outside the shielded enclosures, but inside the buildings. Upon failure of a liquid waste system component, the majority of the liquid will be contained within the enclosure or will drain to the building sump. The liquid will flow into the local drains for eventual processing by one of the evaporators. There is a backup capacity available in the evaporators from the Boron Recovery System.

The building vents and area radiation (airborne) levels are monitored and high activity or radiation level is alarmed (see Subsection 12.3.4).

Upon receiving an alarm, the operator will verify the cause of the alarm. The following sequence of events will ensue:

a. The affected area will be evacuated of unnecessary personnel

b. Accessibility to the affected area will be limited under proper protective measures

c. Proper respirators will be made available to the operating personnel

d. The radioactive fission gases released in the buildings will be removed by the ventilation system, filtered and cleaned up to the capacity of the system, as described in Subsections 9.4.3 and 9.4.4.

The consequences of final releases to the atmosphere are discussed in Subsection 15.7.2.3. The operation of the failed system/component will not be continued without cleaning the area and repairing the affected component. The system is under administrative control during this period.

15.7.2.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event involves a major rupture of one of the Radioactive Liquid Waste System (RLWS) components. This analysis considers the two separate cases of the rupture of either the Boron Waste Storage Tank or the Letdown Degasifier. This analysis assumes that the ruptured RLWS component contains an inventory equivalent to the activity limit specified in Table 15.7-8. The entire release source term is applied to this RLWS component at the beginning of the event. The leak rate from the RLWS to the environment is conservatively modeled to be very high to simulate a major tank rupture, which releases essentially all of the activity to the environment within 2 hours. No hold-up, dilution or filtration of the tank release is assumed. The impact of the release is then computed as it disperses to the offsite receptors. The dose to Control Room operators is computed as the release is modeled to be treated by the Control Room Air Conditioning and Emergency Cleanup system during the 30-day period following the accident.

b. Compliance with RG 1.183 Regulatory Positions

RG 1.183 does not provide direct guidance relative to the Liquid Waste system failures. Therefore, this analysis will rely primarily upon the current licensing basis for guidance on performance of this event.

c. Methodology

The dose assessment model releases the above-prescribed inventory from the RLWS at a high rate of release. Per the existing analysis assumptions, the contents are released to the environment without any hold up, dilution or filtration over a 2 hour period.

For this event, the Control Room ventilation system cycles through three modes off operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air make up and an assumed value of 300 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

d. Radiological Consequences

The release-receptor point locations are chosen to model the distance from the release point to the Control Room intake. The X/Q values for the various combinations of release points and receptor locations are presented in Tables 2R-2 and 2R-3.

For the EAB and LPZ dose analyses, the X/Q factors are provided in Appendix 2Q.

Reg. Guide 1.183 does not provide any requirement or dose limits for a RLWS failure. The dose limits for this event are based on 10 CFR Part 20. The acceptance criteria for this event is ≤ 100 mrem for offsite dose and ≤ 5 Rem for the control room.

The radiological consequences of the RLWS failure event are analyzed using the RADTRAD-NAI code and the inputs and assumptions previously discussed. As shown in Table 15.7-9, the radiological consequences of the Radioactive Liquid Waste System failure are all within the appropriate acceptance criteria.

15.7.2.4 Conclusions

The doses which have been calculated for the failure of either the boron waste storage tank or the letdown degasifier are within regulatory limits.

15.7.3 Postulated Radioactive Releases Due to Liquid Containing Tank Failures

15.7.3.1 Identification of Causes and Accident Description

A rupture of a liquid-containing tank or associated component inside the Containment would release radioactive liquid to the containment sump. The collected liquid would be processed in the Radioactive Liquid Waste Disposal Systems; i.e., the Boron Recovery System (Subsection 9.3.5) and the Liquid Waste Processing System (Section 11.2). The Containment has a steel-lined interior structure; therefore, there is no credible pathway for spilled fluid due to a ruptured tank in Containment to affect water in unrestricted areas. Thus, tanks in the Containment are not considered here.

All liquid containing tanks outside containment were evaluated for postulated radioactive releases due to liquid containing tank failures. The Waste Concentrate Tank is the bounding source term and is based on a condition of 1% failed fuel, as shown in Table 15.7-14. An accident involving the rupture of a tank (irrespective of seismic design category) resulting in leakage of liquid outside the tank is considered.

All the systems considered are provided with proper instrumentation, controls and over-pressure relief protection. If the pressure inside a tank rises above the design pressure of the tank due to controls not functioning properly or due to an operator error, pressure relief devices will relieve the overpressure. Therefore, the rupture of a tank due to overpressure is possible only if the relief devices are inoperative.

A seismic event beyond the design capacity of the tank, or unanticipated corrosion beyond the design corrosion allowance, could also result in failure. The frequency of a possible rupture of a specific tank is not anticipated to be more than once during the expected plant life. As such, the frequency of occurrence is not significant.

The failure of a component associated with a tank could, at the worst, evacuate the tank within the diked/cubicle area. Therefore, rupture of a tank resulting in complete evacuation of the tank is conservatively considered as the design basis.

Significant loading of the liquid waste management systems can only be caused by the failure of those tanks considered which have a large volume or a high concentration of radioactivity.

The refueling water storage tank, the waste concentrates tank, and the floor drain tank are considered for purposes of the analysis. The waste concentrates tank is the bounding source term. The wastes from the failed tanks will be disposed under controlled monitored conditions.

15.7.3.2 Analysis of Effects and Consequences

All the tanks considered are located in areas protected by concrete walls and floors (cubicles). The tank farm adjacent to the Waste Processing Building has some tanks. The tank farm is divided into two sections. These sections have concrete floors and surrounding dikes as high as the rupture level of the largest tank in a section. In one section, the largest tank is the refueling water storage tank and in the other, the reactor makeup water tank. The floors of the tank farm slope toward local drains. On rupture of a tank, the liquid gets drained and routed into the Liquid Waste Processing System. There is no flooding outside the diked areas. Control panel alarms alert the operator to liquid inside diked areas so that the liquid can be drained out under proper controlled conditions to suit the capacity of the drainage and processing equipment. Supply lines to the ruptured tank are then isolated by the operator, and all the leaked liquid is processed by the liquid waste system evaporators before final disposal (Section 11.2). Normal methods of disposal follow concentration and solidification. Effluents within the limits of 10 CFR 20 are discharged offshore via the circulating water tunnels. This is the only discharge into the surface waters. The failed tank is then under administrative control and the unfailed tank is under close supervision.

Because of the potential for cracks in concrete, no credit is taken for liquid retention by unlined building foundations. This means that there is infiltration into the ground until the spilled liquid has drained into the Liquid Waste Processing System. Therefore, for purposes of analysis, a conservative case of all spilled liquid of a tank seeping instantaneously into the ground is considered.

As described in Subsection 2.4.13.3, accidental liquid discharge seeping into the ground will not contaminate any well supplies in the area.

15.7.3.3 Radiological Consequences

The following analysis has been performed to determine the radiological consequences of the rupture of the worst case liquid radwaste tank.

The worst case tank failure was developed by considering the tank with the highest expected radionuclide level and individual radionuclide dose conversion factors. The spent resin sluice tank had the highest curie inventory but was not considered in the analysis. Due to the physical properties of the spent resin material, negligible quantities of the sludge would be able to diffuse through a hypothesized crack in the concrete wall and into the water table aquifer.

The waste concentrates tank had the second highest curie content, based on the conservative 1 percent failed fuel level, with the total activity equal to 1.5×10^3 µCi and liquid volume equal to 6.9×10^3 gallons. For conservatism, it is assumed that the waste concentrates tank has a tritium concentration of 1 µCi/ml.

Based on information presented in Subsection 2.4.13, the maximum rate of groundwater movement along a flow path moving southward from the southern portion of the site is 0.7 ft/day. Radionuclides released from the tank rupture are assumed to instantaneously enter the groundwater system and travel along the groundwater path to the adjoining marsh area. All radionuclides are assumed to travel at the groundwater flow rate with the exception of cesium and strontium isotopes. Since the shortest distance from the southern boundary of the site to the marsh is about 200 feet, it will require at least 290 days for a contaminant released at the site to reach the marsh. With this minimum decay time, all radionuclides with half-lives less than 22 days need not be considered in the dose calculations since they will have decayed to at least .01 percent of their original release concentrations enroute to the marsh. The cesium and strontium isotopes will travel considerably slower than the groundwater due to adsorption and ion exchange with the soil particles. The retardation factor associated with cesium and strontium isotopes traveling through soil columns is well established in literature (References 2, 3) and may be approximated by the following equation (Reference 4):

$$V_i = V_w/(1 + rK_d),$$

where:

V_i = The average velocity of the ionic species

V_w = The average velocity of the groundwater

r = The ratio of the weight of mineral to volume of water per unit volume of aquifer material

K_d = The distribution coefficient of the given ionic species for the prevailing conditions

The quantity $(1 + K_d)$ is referred to as the retardation factor.

The distribution coefficient (K_d) assumed in this evaluation (70 ml/g) is that value used for ^{90}Sr in Reference 5. This assumption is conservative in that cesium isotopes are more tightly bound by soil than strontium isotopes and will exhibit a larger distribution coefficient. Seabrook Station and the standard site used in Reference 5 are both coastal sites with similar soil parameters and groundwater flow rates.

With the above assumptions and parameters listed in Table 15.7-16, the average velocity of Sr and Cs isotopes is calculated to be 1.2×10^{-3} ft/day. The time required to travel 200 feet to the marsh area is 457 years. Based on 457 years decay time for cesium and strontium isotopes, and 290 days decay time for all other radionuclides released, specific nuclide concentrations at the marsh are calculated and listed in Table 15.7-17. These values are based on the assumption that 80 percent of the maximum liquid volume of the affected tank is released. No credit is taken for dilution in the groundwater or liquid retention by unlined building foundations or leakage barriers.

The marsh concentration of radioisotopes is subject to tidal flushing as well as wind and wave action into Hampton Harbor. The discharge into the marsh will be quickly diluted and mixed in the intertidal zone or tidal prism of Hampton Harbor. A value of 2.24×10^{8} ft^3 has been conservatively used in determining the extent of radionuclide dilution, since no credit is taken for dilution within the tidal prism of the marsh. Within the entire Hampton Harbor and estuary, the volume of the tidal prism is approximately 4.70×10^{8} ft^3.

Water is lost from the entire Hampton estuary at an average rate of 9850 ft^3/sec. Expressed on a percentage basis, about 88 percent of the estuary volume leaves and returns on each ebb and flood tide cycle. At ebb slack tide, the estuarine residual is approximately 12 percent of the total volume of the basin. These figures indicate that the Hampton Harbor estuary exhibits substantial tidal exchange rate under natural conditions.

Radionuclide concentrations in Hampton Harbor can be found in Table 15.7-17, and have been used to calculate doses to individuals by the ingestion of finfish and invertebrates. Doses have been calculated based on methodology and dose conversion factors, bioaccumulation factors and maximum usage factors delineated in Regulatory Guide 1.109, Revision 1. The highest organ dose was to the lower large intestine of an adult and was calculated to be 18.2 mrem. The adult total body dose is 1.4 mrem.

15.7.3.4 Conclusions

The radioactive liquid release from a tank rupture will not result in any uncontrolled surface release. The liquid will be processed in the Liquid Waste and/or Boron Recovery System and the final effluent will be controlled and monitored before discharging into the circulating water tunnels or disposed offsite. The seepage of tank contents through cracks in the concrete cubicles will not significantly impact any potable water supply or possible ingestion pathways.

15.7.4 Fuel Handling Accident

15.7.4.1 Identification of Causes and Accident Description

Subsequent to plant start-up, a Licensing Amendment Request (LAR), LAR 94-06, "Revision to Technical Specification 3.9.4," was submitted and accepted by the staff. LAR 94-06 proposed two changes to the Seabrook Station Technical Specifications that address containment building penetrations. The first change is to allow the use of alternate closure methodologies for containment building penetrations during core alterations or movement of irradiated fuel within containment. The second change would allow both personnel airlocks to be open during core alterations or movement of irradiated fuel within containment. Consequently, the most limiting fuel handling accident is defined as the dropping of a spent fuel assembly within an open containment, resulting in the rupture of the cladding of all the fuel rods in the assembly, despite administrative controls and physical limitations imposed on fuel handling operations (see UFSAR Subsection 9.1.4). This potential fuel handling accident is considered an ANS Condition IV event, a limiting fault, since it includes the potential for significant amounts of radioactive releases. All refueling operations are conducted in accordance with prescribed procedures.

Dropping or damaging an assembly within the Fuel Storage Building (FSB) is another postulated accident addressed in this analysis. Dropping an assembly within the FSB is evaluated assuming a minimum of 23 feet of water above the assembly is available for iodine scrubbing (effective iodine decontamination factor of 100) prior to release of fuel assembly gap activity to the FSB atmosphere. Damaging an assembly (i.e., during fuel assembly maintenance or inspection) assumes a minimum of 10 feet of water above the assembly release point (effective iodine DF of 37) prior to release of fuel assembly gap activity to the FSB atmosphere.

15.7.4.2 Analysis of Effects and Consequences

a. Method of Analysis

The following assumptions are postulated in the calculation of the radiological consequences of a fuel handling accident:

1. The accident occurs 80 hours following reactor shutdown, the earliest time when spent fuel would be first moved from the reactor vessel.

2. The accident results in the rupture of the cladding of all fuel rods in the assembly.

3. The damaged assembly was the one operating at the highest power level in the core region to be discharged. The power in this assembly, and the corresponding fuel temperatures, establish the total fission product inventory and the fraction of this inventory present in the fuel pellet-cladding gap at the time of reactor shutdown.

4. The fuel pellet-cladding gap inventory of fission products is released to the refueling cavity or spent fuel pool water at the time of the accident.

5. Refueling cavity or spent fuel pool water will retain a large fraction of the gap activity of halogens by virtue of their solubility and hydrolysis. Noble gases are not retained by the water as they are not subject to hydrolytic reactions.

Additional assumptions are given in Table 15.7-18.

b. Fission Product Inventories

The fission product gap inventory in a fuel assembly is dependent on the power rating of the assembly and the temperature of the fuel. The parameters used for the calculation of the fission product inventory of the highest rated assembly to be discharged are summarized in Table 15.7-19. Table 15.7-20 shows the activity of the highest rated fuel assembly at the time of reactor shutdown and after 80 hours decay.

The conservative parameters are based on Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," dated March 23, 1972.

c. Iodine Decontamination Factors

An experimental test program (Reference 6) was conducted by Westinghouse 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 solution in the spent fuel pool 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.

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.

Decontamination factor = $73\ e^{0.313\ t/d}$

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 solution, 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 full assembly, that is, 6.9 SCF for the 17x17 array, the pool decontamination factor for iodine is indicated to be a minimum of 589 for a 23 foot depth and 181 for a 10 foot depth. In the conservative analyses, a lower decontamination factor is selected to provide for reasonable deviation in the factors which control iodine adsorption by the pool water. For the dropped assembly analysis, an overall effective iodine decontamination factor of 100 is used, as discussed in Regulatory Guide 1.25. In both the realistic and conservative evaluations, a decontamination factor of one is used for noble gas isotopes. The activity released from the spent fuel pool surface or refueling canal by isotope is shown in Table 15.7-21.

Potential damaging of a fuel assembly in the FSB during maintenance or inspection could occur with a minimum of 10 feet of water above the assembly. As discussed above the calculated iodine DF for 10 feet of water is 181 using the analytical expression above. For the analysis described below for the potential damaging of an assembly while suspended with only 10 feet of water above the assembly, a reduced iodine DR of 37 is used to provide for reasonable deviation in the factors which control iodine adsorption by the pool water. This value was determined by appropriate normalization of the experimental test case values.

15.7.4.3 Radiological Consequences using Alternate Source Term Methodology

a. Background

This event consists of the drop of a single fuel assembly either in the Fuel Storage Building (FSB) or inside of Containment. All of the fuel rods in a single fuel assembly are damaged. In addition, a minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and FSB release locations.

This analysis bounds dropping a fuel assembly either inside the containment (with the personnel hatch open) or inside the FSB. Although filtration can be credited for the accident in the FSB there is sufficient margin to allow the analysis to be performed without crediting FSB filters. The source term released from the overlying water pool is the same for both the FSB and the containment cases. RG 1.183 imposes the same 2-hour criteria for the direct unfiltered release of the activity to the environment for either location. With the containment personnel hatch assumed open and filtration of the Fuel Storage Building exhaust not credited, the analyses are essentially identical for either the containment or the FSB release point except that the dispersion factors from the containment are slightly greater than the dispersion factors from the FSB.

To ensure that this analysis bounds the FHA in Containment or in the Fuel Storage Building, the most limiting combination of release point dispersion factors (X/Q) from the containment personnel hatch or the Fuel Storage Building release points is used. Use of the most limiting dispersion factors with no credit for FSB filtration assures the event results bound a Fuel Handling accident in either the containment or the Fuel Storage Building.

b. Compliance with RG 1.183 Regulatory Positions

The FHA dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix B, “Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident,” as discussed below:

1. Regulatory Position 1.1 – The amount of fuel damage is assumed to be all of the fuel rods in a single fuel assembly.

2. Regulatory Position 1.2 – The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of RG 1.183. A listing of the FHA source term is provided in Table 15C-4. The gap activity available for release is specified by Table 3 of RG 1.183. This activity is assumed to be released from the fuel instantaneously.

3. Regulatory Position 1.3 – The chemical form of radioiodine released from the damaged fuel into the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The cesium iodide is assumed to completely dissociate in the spent fuel pool resulting in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine.

4. Regulatory Position 2 – A minimum water depth of 23 feet is maintained above the damaged fuel assembly. Therefore, an overall decontamination factor of 200 is applied to the elemental and organic iodine based upon the composition specified in Regulatory Position 1.3

5. Regulatory Position 3 – All of the noble gas released is assumed to exit the pool without mitigation. All of the non-iodine particulate nuclides are assumed to be retained by the pool water.

6. Regulatory Position 4.1 – The analysis models the release from the FSB to the environment over a 2-hour period.

7. Regulatory Position 4.2 – No credit is taken for filtration of the release from the FSB.

8. Regulatory Position 4.3 – No credit is taken for dilution of the release in the FSB.

9. Regulatory Position 5.1 – The containment personnel hatch is assumed to be open at the time of the fuel handling accident.

10. Regulatory Position 5.2 – No automatic isolation of the containment is assumed for the FHA.

11. Regulatory Position 5.3 – The release from the containment fuel pool is assumed to leak to the environment over a two-hour period.

12. Regulatory Position 5.4 – No ESF filtration of the containment release is credited.

13. Regulatory Position 5.5 – No credit is taken for dilution or mixing in the containment atmosphere.

c. Methodology

The input assumptions used in the dose consequence analysis of the FHA are provided in Table 15.7-18. The limiting accident bounds a FHA inside of containment with the containment personnel hatch open or in the Fuel Storage Building without exhaust filtration. It is assumed that the fuel handling accident occurs at 80 hours after shutdown of the reactor per Licensing Submittal 02-06. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from a single fuel assembly into the fuel pool. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. 100% of the noble gas released from the damaged fuel assembly is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the pool. The iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. A DF of 285 for elemental iodine and 1 for organic iodine is applied to the pool to accomplish the overall DF of 200 for the iodine release. The activity released from the pool is then assumed to leak to the environment over a two-hour period.

The FHA source term meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. The source term is listed in Table 15C-4.

For this event, the Control Room ventilation system cycle through three modes of operation:

- Initially, the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air makeup and an assumed value of 300 cfm of unfiltered inleakage.

- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 30-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 600 cfm of filtered makeup flow from the outside, 300 cfm of unfiltered inleakage, and 390 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

d. Radiological Consequences

The atmospheric dispersion factors (X/Qs) used for this event for the Control Room dose are based on the location of the containment personnel hatch (bounds the FSB release) and the operational modes of the control room ventilation system. These X/Qs are summarized in Tables 2R-2 and 2R-3.

The EAB and LPZ dose is determined using the X/Q factors provided in Appendix 2Q.

The radiological consequences of the FHA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.7-19, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

15.7.4.4 Conclusions

The doses calculated for both the fuel handling accident occurring within the containment and for the fuel handling accident occurring in the fuel storage building are well within the values specified in regulatory limits. The fuel handling accident occurring within the containment with an open release path to the environment is the bounding fuel handling event.

15.7.5 Spent Fuel Cask Drop Accident [Historical]

Note: The spent fuel cask drop accident represents assumptions used in the original plant design. This event contains historical information that is not relevant at this time.

15.7.5.1 Identification of Causes and Accident Description [Historical]

As discussed in Subsection 9.1.4, an isolation gate installed between the spent fuel pool and the transfer canal will prevent a loss of spent fuel pool water due to a postulated cask drop accident. This gate is closed during cask handling operations. The cask handling crane cannot be passed over the isolation gate or any part of the spent fuel storage area; hence, the spent fuel shipping cask cannot be transported over these areas. Consequently, in the event that a heavy cask were dropped, the spent fuel storage area integrity would not be compromised nor any stored fuel damaged. The limited travel of the cask handling crane prevents it from traveling over any safety-related equipment.

The cask is lifted in and out of the cask loading pool in two steps. The first step is from elevation (-) 23'-10½" to a shelf at elevation 4'-5¼," a lift of 28'-3¾". The second step is from elevation 4'-5¼" to clear the operating floor at the 25' elevation, a lift of 21'-6¾". The Engineered Safety Features Filter System (Subsection 6.5.1) is in operation during handling of a loaded cask. Operation of the Fuel Storage Building Ventilation System in the emergency mode is further discussed in Subsection 6.5.1.

15.7.5.2 Radiological Consequences [Historical]

The radiological consequences for the postulated spent fuel cask drop accident have been calculated based on no impact limiting devices in the designs of the Seabrook Station cask handling equipment. The cask handling crane cannot be passed over the fuel pool isolation gate or any part of the spent fuel storage area; hence, the only source of fission products available for release are those contained within the spent fuel cask and the contained fuel assemblies. For the purpose of this analysis, it is conservatively assumed that all of the fuel pins are breached, releasing all of the halogens and noble gases contained in the gap area of the fuel pins.

The following assumptions are postulated in the calculation of the radiological consequences of the spent fuel cask drop accident; additional parameters are given in Table 15.7-26.

a. Conservative Analysis

1. The maximum number of fuel assemblies contained within one shipping cask is seven assemblies, which have been stored and decayed for a minimum of 150 days. This is a conservative estimate based on methodology used in WASH 1238 "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," USAEC, December 1972, and current practices of storing spent fuel assemblies. Conservative case core (193 assemblies) and gap activities for iodines and noble gases are given in Table 15.0.6.

2. The postulated cask drop occurs within the Spent Fuel Storage Building. The Engineered Safety Features Filter System is in operation, providing an iodine DF of 20.

3. It is assumed, for the purpose of providing offsite dose consequences, that all of the fission products released within the cask are instantaneously released to the Fuel Storage Building environment and ultimately to the environment via the ESF filter system. This is very conservative in view of the stringent testing criteria (10 CFR 71, Appendix B) that spent fuel shipping casks must comply with. The activity released as a result of the postulated accident is given in Table 15.7-27.

   Offsite doses resulting from the spent fuel cask drop accident were evaluated based on the dose methodology given in Appendix 15A and the one hour atmospheric dispersion parameters presented in Appendix 15B. Results are given in Table 15.7-28.

b. Realistic Analysis

The realistic analysis assumes the realistic gap fractions (Table 15.7-22) and that the Fuel Storage Building Engineered Safety Features Filter System is 99 percent efficient for the removal of elemental iodine. Realistic doses are evaluated based on the dose methodology presented in Appendix 15A and the realistic one-hour atmospheric dispersion factor (X/Q) given in Appendix 15B. Results are given in Table 15.7-28.

15.7.6 References

1. Underhill, D.W., "Effect of Rupture in a Fission Gas Holdup Bed," Nuclear Safety, 13(6), November-December 1972.

2. S. Iwai, Y. Inove and K. Nishimaki, "Movement Through Soil of Radioactive Nuclides Contained in Chemical Processing Waste," Kyoto University, 1968.

3. "Comments on the Content of AEC's Proposed Environmental Impact Statement on the Hanford Site," US EPA, Office of Radiation Programs, 1973.

4. H. B. Levy, "On Evaluating the Hazards of Groundwater Contamination by Radioactivity from an Underground Nuclear Explosion," Lawrence Livermore Laboratory, Rept. UCRL-51278, 1972.

5. Y. Inove and S. Morisawa, "On the Selection of a Ground Disposal Site by Sensitivity Analysis," Health Phys. 26, 251-261 (1973).

6. D. D. Malinowski, et al., "Radiological Consequences of a Fuel Handling Accident," WCAP-7518-L (Proprietary), July 1971 and WCAP-7828 (Nonproprietary), December 1971.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

An Anticipated Transient Without Scram (ATWS) is a postulated anticipated operational occurrence (such as loss of feedwater, loss of load, or loss of off-site power) that is accompanied by a failure of the Reactor Protection System (RPS) to shutdown the reactor.

The final ATWS rule, 10 CFR 50.62 (c) (Reference 1), requires that Westinghouse designed plants install NRC approved ATWS Mitigating System Actuation Circuitry (AMSAC) to initiate a turbine trip and actuate emergency feedwater flow independent of the Reactor Protection System. The basis for this rule and the AMSAC design is supported by Westinghouse analyses documented in Reference 2 and 3. The information presented in References 2 and 3 is applicable to Seabrook. The Seabrook ATWS mitigation system is described in Subsection 7.6.12.

The basis for the final ATWS rule and the AMSAC design are supported by Westinghouse analyses documented in NS-TMA-2182 (Reference 3). These analyses were performed based on the guidelines published in NUREG-0460 (1978) (Reference 4). Appendix A of WASH-1270 (Reference 5) states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the "emergency conditions" as defined in the ASME Nuclear Power Plant Components Code, Section III" (recently termed Service Limit C). Based on a review of reactor vessels for 2-, 3- and 4-loop plants, the maximum allowable pressure for the reactor vessel is 3200 psig. This value corresponds to the maximum allowable pressure for the weakest component in the reactor pressure vessel (the nozzle safe end); thus, the Reference 3 analyses were performed to demonstrate that the RCS pressure did not exceed 3200 psig (3215 psia). Reference 3 describes the methods used in the analyses and provides reference analyses for 2-loop, 3-loop and 4-loop plant designs with several different steam generator models available in plants at that time.

The loss of normal feedwater (LONF) and loss of load (LOL) ATWS events are the two most limiting RCS overpressure transients reported in Reference 3. To address the Power Uprate for the Seabrook Station, these two events were reanalyzed at the Power Uprate conditions to ensure that the basis for the final ATWS rule continues to be met.

The primary input to the loss of normal feedwater and loss of load ATWS analyses performed in support of the Seabrook Station Power Uprate are the 4-loop reference LONF and LOL ATWS models with Model F steam generators supporting Reference 3. The nominal and initial conditions were updated to reflect an analyzed NSSS power of 3678 MWt corresponding to the Power Uprate.

Seabrook Station has two emergency feedwater pumps, 1 motor-driven and one turbine-driven pump. The design flow capacity of each pump is 710 gpm for a total emergency feedwater flow (EFW) of 1420 gpm. The emergency feedwater flow assumed in the Reference 3 analysis was a best-estimate value of 1760 gpm. In support of the Power Uprate, three cases were run for both the LONF and LOL ATWS events, one with Reference 3 best-estimate feedwater flow of 1760 gpm and two sensitivity cases, one with the Seabrook Station emergency feedwater design flow capacity of 1420 gpm and one with an assumed maximum emergency feedwater flow capacity of 1337 gpm.

To address the Seabrook Station Power Uprate, the two most limiting RCS overpressure transients reported in Reference 3, loss of normal feedwater and loss of load, were analyzed at the Power Uprate conditions to ensure that the analytical basis for the final ATWS rule (10 CFR 60.62) (c) continues to be met. These analyses were based on the Reference 3, 4-loop loss of normal feedwater and loss of load ATWS model with Model F steam generators. The nominal and initial conditions were appropriately revised to reflect a NSSS power level of 3678 MWt consistent with the Power Uprate.

The results of the loss of normal feedwater and loss of load ATWS analyses at 3678 MWt demonstrates that the analytical basis for the final ATWS rule continues to be met for operation of the Seabrook Station with the Power Uprate.

15.8.1 References

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

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

3. Anderson, T. M., "ATWS Submittal," Westinghouse Letter NS-TMA-2182 to S. H. Hanauer of the NRC, December 1979.

4. NUREG-0460, Anticipated Transients Without Scram for Light Water Reactors, USNRC, December 1978.

5. WASH-1270, Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors, USNRC, September 1973.

15.9 STATION BLACKOUT

10 CFR 50.63 (Reference 1), Regulatory Guide 1.155 (Reference 2) and NUMARC 87-00 (Reference 3) require that each light-water-cooled nuclear power plant be able to withstand and recover from a loss of all alternating current power or Station Blackout (loss of both offsite power and onsite emergency power). The effects of Station Blackout are not considered as part of the design basis for the transients analyzed in Chapter 15. The capability for Seabrook to cope with Station Blackout is described in Section 8.4.

15.9.1 References

1. 10 CFR 50.63, "Loss of All Alternating Current Power (Station Blackout)"
2. Regulatory Guide 1.155, "Station Blackout"
3. NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors"

TABLE 15.0-1 NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

Nuclear Steam Supply System Thermal power output	3678 MWt
Thermal power generated by the reactor coolant pumps	19 MWt
Core thermal power	3659 MWt

TABLE 15.0-2 NOMINAL VALUES OF PERTINENT PLANT PARAMETERS UTILIZED IN THE ACCIDENT ANALYSES

Core Thermal Power (MWt)	3659[(1)]
Reactor Coolant System Pressure (psia)	2250
Reactor Coolant Flow per Loop (gpm)	95,950[(2)]
Assumed Feedwater Temperature at Steam Generator Inlet (°F)	390.0, 452.4
Average Core Heat Flux (Btu/hr-ft^2)	203,551

Parameter	Low Tavg	High Tavg
Vessel Average Temperature (°F)	571.0	589.1
Core Inlet Temperature (°F)	537.7	556.8
Steam Flow from NSSS (lb/hr)		
@ 390.0°F FW Temperature, 0% SGTP	15,080,000	15,170,000
@ 390.0°F FW Temperature, 10% SGTP	15,070,000	15,160,000
@ 452.4°F FW Temperature, 0% SGTP	16,420,000	16,520,000
@452.4°F FW Temperature, 10% SGTP	16,410,000	16,510,000
Steam Pressure (psia)		
@ 0% Steam Generator Tube Plugging	815	962
@10% Steam Generator Tube Plugging	797	943

(1) Analyzed power which conservatively bounds licensed power plus uncertainties.

(2) Minimum measured flow used in RTDP.

TABLE 15.0-3 SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature	Reactivity Coefficients Assumed: Doppler Power	Thermal Power Output Assumed %
15.1	Increase in Heat Removal by the Secondary System Feedwater System				
-	Feedwater System Malfunction Causing an Increase in Feed- water Flow	RETRAN	Most Negative	Least Negative	0 and 100
-	Excessive Increase in Secondary Steam Flow	N/A	N/A	N/A	100
-	Accidental Depressurization of the Main Steam System	Bounded by Steam System Piping Failure Analysis	---	---	---
-	Steam System Piping Failure	RETRAN	Most Negative	Least Negative	0

	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature	Reactivity Coefficients Assumed: Doppler Power	Thermal Power Output Assumed %
15.2	Decrease in Heat Removal by the Secondary System				
-	Loss of External Load and/or Turbine Trip	RETRAN	0[b]	Least Negative	100[a]
-	Loss of Nonemergency AC Power to the Station Auxiliaries	RETRAN	0[b]	Most Negative	100
-	Loss of Normal Feedwater Flow	RETRAN	0[b]	Most Negative	100
-	Feedwater System Pipe Break	LOFTRAN, RETRAN	0[b] and Most Negative	Most and Least Negative	100
15.3	Decrease in Reactor Coolant System Flow Rate				
-	Partial and Complete Loss of Forced Reactor Coolant Flow	RETRAN, VIPRE	0[b]	Most Negative	100

			Reactivity Coefficients Assumed		Thermal Power Output Assumed
	Faults	Computer Codes Utilized	Moderator Temperature	Doppler Power	%
-	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	RETRAN, VIPRE	0[(b)]	Most Negative	100
15.4	Reactivity and Power Distribution Anomalies				
-	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low power Startup Condition	TWINKLE, FACTRAN, VIPRE	Most Positive	Least Negative	0
-	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	RETRAN	Most and Least Negative	Most and Least Negative	10, 60,100
-	Control Rod Misalignment	VIPRE, LOFTRAN	---	N/A	100
-	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	N/A	N/A	N/A	0 and 100

			Reactivity Coefficients Assumed		Thermal Power Output Assumed
	Faults	Computer Codes Utilized	Moderator Temperature	Doppler Power	%
-	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	N/A	N/A	N/A	N/A
-	Spectrum of Rod Cluster Control Assembly Ejection Accidents	TWINKLE, FACTRAN	Predicted values plus uncertainty	Predicted values plus uncertainty	0 and 100
15.5	Increase in Reactor Coolant Inventory				
-	Inadvertent Operation of ECCS During Power Operation	RETRAN, VIPRE	Most Negative	Most Negative	100

	Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature	Reactivity Coefficients Assumed: Doppler Power	Thermal Power Output Assumed %
15.6	Decrease in Reactor Coolant Inventory				
-	Inadvertent Opening of a Pressurizer Safety or Relief Valve	RETRAN	0[b]	Least Negative	100
-	Steam Generator Tube Rupture	RETRAN	Least Negative	Most Negative	100
-	Loss-of-Coolant Accident Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant System	NOTRUMP W COBRA	See Subsection 15.6.5	See Subsection 15.6.5	100

[a] A conservative (high) NSSS power is assumed; no additional power uncertainty is modeled.

[b] Modeling a zero moderator temperature coefficient is sufficient for analyses at full power. A zero MTC at full power bounds a positive MTC at part power.

TABLE 15.0-4 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
Power Range Neutron Flux High Positive Rate	6.9%	0.65
Power Range Neutron Flux, P-8	50%	0.5
Power Range Neutron Flux, P-10	10%	N/A
Overtemperature ΔT	Variable	6.0*
Overpower ΔT	Variable	6.0*
High pressurizer pressure	2425 psia	2.0
Low pressurizer pressure	1935 psia	2.0
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0
Undervoltage Trip	70% nominal (9660 volts)	1.5
Underfrequency Trip	55 HERTZ	0.6
Turbine Trip	Not applicable	1.0
Low-low steam generator level	0% of narrow range level span	2.0
High-high steam generator level, P-14	100% of narrow range level span	2.0

Note that P-4 is implicitly assumed in all reactor trip scenarios.

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

TABLE 15.0-5 PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.1	Increase in Heat Removed by the Secondary System				
--	Feedwater System Malfunction Causing an Increase in Feedwater Flow	Power range high flux, high steam generator level, Manual	High steam generator level--produced feedwater isolation & turbine trip	Feedwater isolation valves, turbine stop and control valves	--
--	Excessive Increase in Secondary Steam Flow	Power range high flux, OTΔT, OPΔT, Manual	--	Pressurizer self--actuated safety valves; steam generator safety valves, turbine stop and control valves	--
--	Accidental Depressur ization of the Main Steam System	Low pressurizer pressure, Manual, SIS	Low pressurizer pressure, low compensated steam line pressure, Hi--1 containment pressure, Manual	Feedwater isolation valves, Steam line stop valves, turbine stop and control valves	Emergency Feed System, Safety Injection System
--	Steam System Piping Failure	SIS, Low pressurizer pressure, Manual, OPΔT	Low pressurizer pressure, low compensated steam line pressure, Hi--1 containment pressure, Manual, high neg. steam line pressure rate	Feedwater isolation valves, Steam line stop valves, turbine stop and control valves	Emergency Feed System, Safety Injection System

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.2	Decrease in Heat Removal by the Secondary System				
--	Loss of External Load/ Turbine Trip	High pressurizer pressure, OTΔT, steam generator low--low level, Manual	--	Pressurizer safety valves, steam generator safety valves, turbine stop and control valves	--
--	Loss of Nonemergency AC Power to the Station Auxiliaries	Steam generator low--low level, Manual	Steam generator low--low level	Steam generator safety valves, turbine stop and control valves	Emergency Feed System
--	Loss of Normal Feedwater Flow	Steam generator low--low level, Manual	Steam generator low--low level	Steam generator safety valves, turbine stop and control valves	Emergency Feed System
--	Feedwater System Pipe Break	Steam generator low--low level, High Pressurizer Pressure, SIS, Manual	High containment pressure, steam generator low--low water level, low compen sated steam line pressure, high neg. steam line pressure rate	Steam line isolation valves, feedline isolation, Pressur izer self--actuated safety valves, steam generator safety valves, turbine stop and control valves	Emergency Feed System, Safety Injection System
15.3	Decrease in Reactor Coolant System Flow Rate				
--	Partial and Complete Loss of Forced Reactor Coolant Flow	Low flow, Undervoltage, Underfrequency, Manual	--	Steam generator safety valves, P-8, turbine stop and control valves	--

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
--	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Low flow, Manual	--	Pressurizer safety valves, steam generator safety valves, turbine stop and control valves	--
15.4	Reactivity and Power Distribution Anomalies				
--	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Sub--critical or Low Power Startup Condition	Power range high flux (low setpoint), Manual	--	--	--
--	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Power range high flux, OTΔT, Hi pressurizer pressure, Manual	--	Pressurizer safety valves, steam generator safety valves, turbine stop and control valves	--
--	Control Rod Misalignment	OTΔT, Manual	--	Turbine stop and control valves	--
--	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Source range high flux, power range high flux, OTΔT, Manual	--	Low insertion limit annunciators for boration, turbine stop and control valves	--
--	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Power range high flux, High positive flux rate, Manual, low pressurizer pressure, P--10	--	Turbine stop and control valves	--

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment
15.5	Increase in Reactor Coolant Inventory				
--	Inadvertent Operation of ECCS During Power Operation	Low pressurizer pressure, Manual, SI trip	--	Turbine stop and control valves	Safety Injection System
15.6	Decrease in Reactor Coolant Inventory				
--	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Pressurizer low pressure, OTΔT, Manual	--	Turbine stop and control Valves	--
--	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 safety and/or relief valve, steam line stop valves, turbine stop and control valves	Emergency Core Cooling System, Emergency Feed Water System, Emergency Power Systems
--	Loss--of--Coolant Accident from Spectrum of Postulated Piping Breaks within the Reactor Coolant System	Reactor Trip System	Engineered Safety Features Actuation System	Service Water System, Component Cooling Water System, steam generator safety and/or relief valves, turbine stop and control	Emergency Core Cooling System, Emergency Feed-- water System, Containment Heat Removal System, Emergency Power System Valves

TABLE 15.0-6 IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE AND FUEL ROD GAPS [HISTORICAL]

Isotope	Core (Ci)	Percentage of Core Activity in Gap*	Assumed Fuel Rod Gap Activity (Ci)
I-127	0 (2.80 Kg)	30	0 (0.84 Kg)
I-129	2.0E+0	30	5.9E-0.1
I-130	1.8E+06	10	1.8E+05
I-131	1.0E+08	10	1.0E+07
I-132	1.4E+08	10	1.4E+07
I-133	2.1E+08	10	2.1E+07
I-134	2.3E+08	10	2.3E+07
I-135	2.0E+08	10	2.0E+07
Kr-83m	1.2E+07	10	1.2E+06
Kr-85m	2.8E+07	10	2.8E+06
Kr-85	6.8E+05	30	2.0E+05
Kr-87	5.0E+07	10	5.0E+06
Kr-88	7.2E+07	10	7.2E+06
Kr-89	8.9E+07	10	8.9E+06
Xe-131m	7.2E+05	10	7.2E+04
Xe-133m	3.0E+07	10	3.0E+06
Xe-133	2.0E+08	10	2.0E+07
Xe-135m	4.1E+07	10	4.1E+06
Xe-135	4.3E+07	10	4.3E+06
Xe-138	1.6E+08	10	1.6E+07

Note: The information presented in Table 15.0-6 represents assumptions used in the original accident analysis. The information presented in table is retained for historical purposes.

Power level 3654 MWt.

Three-region equilibrium cycle core at end of life. The three regions have operated at a specific power of 40.03 MW/Mtu for 300, 600, and 900 EFPD respectively.

* Based on Regulatory Guides 1.25 and 1.77.

Table 15.0-7 Component Times And Capacities

COMPONENT	RESPONSE TIME	CAPACITY	TEST PROVISIONS
Main Steam line Isolation Valves	1 second logic and delay 5 second closure	--	See Table 14.2-3, item 13
Main Feedwater Isolation Valves	2 second logic and delay 10 second closure	--	See Table 14.2-3, item 14
Main Feedwater Regulating Valves FWIS Response	7 second logic and delay 5 second closure	--	
Pressurizer Power--Operated Relief Valves	--	2 Valves @ 210000 lbm/hr	See Table 14.2-3, item 2
Pressurizer Safety Valves(2)	--	3 Valves @ 420000 lbm/hr	See Table 14.2-3, item 40
Steam Generator Safety Valves(2)	--	14,588,456 lbm/hr	See Table 14.2-3, item 40

(2) Allowance of 3% setpoint tolerance for all code safety valves is assumed.

COMPONENT	RESPONSE TIME	CAPACITY	TEST PROVISIONS
Emergency Feedwater[1]	2 second logic and delay, 75 second delay for pump start, and 23 second control valve closure time on high flow during an FLB	Feedline rupture -- EFW flow is based on 470 gpm minimum total flow at a steam generator back pressure of 1236 psia to two intact steam generators with two EFW pumps operational, or 470 gpm minimum total flow to three intact steam generators with one EFW pump operational Loss of feedwater with AC power: 650 gpm total minimum flow at a steam generator back pressure of 1236 psia to all steam generators with one EFW pump operational. Loss of feedwater without AC power: 650 gpm total minimum flow at a steam generator back pressure of 1236 psia to all steam generators with one EFW pump operational.	See Table 14.2-3, item 14
Atmospheric Steam Dump Valves	--	400,000 lbm/hr @ 1135 psia (SGTR) -- Minimum 583,000 lbm/hr @ 1135 psia (SGTR -- Maximum)	

[1] For Steam line Rupture, see Subsection 15.1.5

TABLE 15.0-8 STEAM GENERATOR TUBE RUPTURE

Component	Response Time	Capacity
Main Steam Isolation Valves	Manual operation on indicated level	--
Intact Steam Generator ASDVs		Nominal flow rate of 530,908 lb/hr/valve @ 1125 psig
Ruptured Steam Generator ASDV		Maximum flow rate of 583,385 lb/hr/valve @ 1125 psig

TABLE 15.1-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE AN INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
Excessive Feedwater flow malfunction at full power		
	One main feedwater control valve fails fully open	0.0
	High-high steam generator water level setpoint reached	45.1
	Minimum DNBR occurs	48.0
	Feedwater isolation valves fully closed	57.1
Steam System piping failure (with offsite power available)		
	Steam line ruptures	0.0
	Criticality attained	16.5
	SI flow begins	27.55
	Minimum DNBR occurs	158.2
	Accumulators actuate	425.3

TABLE 15.1-2 MAIN STEAM LINE BREAK (MSLB) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (includes uncertainty)
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 15C-3
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	Faulted SG – 500 gallons/day Intact SGs – 940 gallons/day
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	48 hours
RCS Mass	539,037 lb_m
SG Secondary Side Mass	Maximum (Hot Zero Power) – 166,000 lb_m (used for faulted SG to maximize release) Minimum (Hot Full Power) – 99, 304 lb_m per SG for a total of 297,912 lb_m (used for intact SGs to maximize concentration)
Release from Faulted SG	Instantaneous
Steam Release from Intact SGs	Table 15.1-3
Secondary Coolant Iodine Activity prior to accident	0.1 μCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG – none Intact SGs – 100
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3

Input/Assumption	Value
Control Room Ventilation System Time of automatic control room normal intake isolation and switch to emergency mode	30 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183, Section 4.2.6

TABLE 15.1-3 INTACT SGS STEAM RELEASE RATE

Time (Hours)	Intact SGs Steam Release Rate* (lb_m/min)
0-2	3383.3
2-8	2563.9
8-720.0	0

*Total release rate for all three (3) intact SGs.

TABLE 15.1-4 60 µCi/gm D.E. I-131 ACTIVITIES

Isotope	**Activity (µCi/gm)**
Iodine-131	46.36
Iodine-132	16.88
Iodine-133	74.18
Iodine-134	10.76
Iodine-135	40.80

TABLE 15.1-5 IODINE EQUILIBRIUM APPEARANCE ASSUMPTIONS

Input Assumption	Value
Maximum Letdown Flow	120 gpm
Assumed Letdown Flow *	132 gpm at 115°F, 2235 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	505,000 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* maximum letdown flow plus 10% uncertainty

TABLE 15.1-6 CONCURRENT IODINE SPIKE (500 X) ACTIVITY APPEARANCE RATE

Isotope	**Appearance Rate (CI/min)**
Iodine-131	209.029344
Iodine-132	235.958907
Iodine-133	404.536383
Iodine-134	317.823719
Iodine-135	315.402448

TABLE 15.1-7 MSLB DOSE CONSEQUENCES

Case	EAB Dose[1] (rem TEDE)	LPZ Dose [2] (rem TEDE)	Control Room Dose[2] (rem TEDE)
MSLB pre-accident iodine spike	0.08	0.15	1.00
Acceptance Criteria (pre-accident iodine spike)	25 [3]	25 [3]	5 [4]
MSLB concurrent iodine spike	0.39	0.82	3.23
Acceptance Criteria (concurrent iodine spike)	2.5 [3]	2.5 [3]	5 [4]

(1) Worst 2-hour dose

(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 150 cfm

(3) RG 1.183, Table 6

(4) 10 CFR 50.67

TABLE 15.2-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident		Event	Time (seconds)
a. Turbine Trip / Loss of Load			
	A. With pressure control	Turbine trip, loss of main feedwater flow	0.0
		Initiation of steam release from pressurizer relief valves	N/A
		Overtemperature ΔT reactor trip setpoint reached	10.6
		Rods begin to drop	12.6
		Minimum DNBR occurs	13.0
		Maximum steam generator pressure occurs	17.0
	B. Without pressure control		
		Turbine trip, loss of main feedwater flow	0.0
		High pressurizer pressure reactor trip setpoint reached	5.9
		Initiation of steam release from pressurizer safety valves	7.7
		Rods begin to drop	7.9
		Peak RCS pressure occurs	8.5
		Maximum steam generator pressure occurs	13.5
b. Loss of Nonemergency AC Power to the Station Auxiliaries			
		AC power is lost; main feedwater flow stops	20
		Low-low steam generator water level setpoint reached	59
		Rods begin to drop	61
		RCPs begin to coast down	63
		Four SGs begin to receive emergency feedwater flow from one emergency feedwater pump	136
		Minimum SG inventory occurs	640

Accident	Event	Time (seconds)
	Long term peak pressurizer water level occurs	1300
c. Loss of Normal Feedwater		
	Main feedwater flow stops	20
	Low-low steam generator water level setpoint reached	59
	Rods begin to drop	61
	Four SGs begin to receive emergency feedwater flow from one emergency feedwater pump	136
	Long term peak pressurizer water level occurs	1055
	Minimum SG inventory occurs	1165
d. Feedwater System Pipe Break		
	Main feedwater line rupture occurs	0.0
	Low-low steam generator water level setpoint reached in broken loop	5.0
	Rods begin to drop	7.0
	Low Pressurizer Pressure reached for SIS injection	76.3
	Safety injection flow is started	103.3
	Emergency feedwater flow is started	105.0
	Low steamline pressure isolation setpoint is reached	114.5
	All main steam line isolation valves are closed	118.5
	Steam generator safety valves open in steam generators of intact loops	810.1
	Core decay heat plus pump heat decrease to emergency feedwater heat removal capacity	~ 4000

TABLE 15.2-2 DELETED

TABLE 15.2-3 DELETED

TABLE 15.2-4 DELETED

TABLE 15.3-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW

Accident	Event	Time (seconds)
Partial Loss of Forced Reactor Coolant Flow	Two of four operating RCPs begin coasting down	0.0
	Low flow reactor trip setpoint reached	1.5
	Rods begin to drop	2.5
	Minimum DNBR occurs	2.95
Complete Loss of Forced Reactor Coolant Flow - Undervoltage	All operating RCPs lose power and coastdown begins	0.0
	RCP undervoltage setpoint reached	0.0
	Rods begin to drop	1.5
	Minimum DNBR occurs	2.7
Complete Loss of Forced Reactor Cooland Flow – Underfrequency	Frequency decay begins	0.0
	RCP underfrequency setpoint reached and all pumps begin to coast down	1.0
	Rods begin to drop	1.6
	Minimum DNBR occurs	3.1
Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Rotor on one pump locks	0
	Low flow reactor trip setpoint reached in the affected loop	0.045
	Rods begin to fall into core; Turbine trip, loss of offsite power, unaffected reactor coolant pumps begin to coast down	1.045
	Maximum clad temperature occurs	3.35
	Maximum RCS pressure occurs	4.55

TABLE 15.3-2 SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENT

Maximum Reactor Coolant System Pressure (psia)	2544
Maximum Cladding Temperature (F°) Core Hot Spot	1650
Zr-H2O Reaction At Core Hot Spot (% by weight)	0.20

TABLE 15.3-3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Core Fission Product Inventory	Table 15C-1
RCS Equilibrium Activity (1.0 μCi/gm DE I-131)	Table 15C-2
Release Fraction from Breached Fuel	RG 1.183, Section 3.2, Table 3
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
Fuel Failure	10.0%
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for fuel failure dose contribution to maximize SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.
Primary-to-Secondary Leakage Rate	1.0 gpm total (500 gpd maximum to any one SG)
Time to establish shutdown cooling and terminate release	8 hours
SG Mininum Mass (per SG)	99,304 lb_m
Secondary Side Iodine Activity prior to accident	Table 15C-3
Seconday Side Mass Releases to environment	Table 15.3-4
Steam Generator Secondary Side Partition Coefficient	100

Input/Assumption	Value
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Control Room Ventilation System Time of automatic control room isolation	30 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183, Section 4.2.6

TABLE 15.3-4 LOCKED ROTOR STEAM RELEASE RATE

Time (hours)	Intact SG Steam Release Rate (lb_m/min)
0.0 – 2.0	3392
2.0 – 8.0	2675
8.0 – 720.0	0

TABLE 15.3-5 LOCKED ROTOR DOSE CONSEQUENCES

Case	**EAB Dose [1] (rem TEDE)**	**LPZ Dose [2] (rem TEDE)**	**Control Room Dose [2] (rem TEDE)**
Locked Rotor	1.01	0.76	1.83
Acceptance Criteria	2.5	2.5	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 150 cfm

TABLE 15.3-6 DELETED

TABLE 15.3-7 DELETED

TABLE 15.3-8 DELETED

Table 15.3-9 DELETED

TABLE 15.4-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident		Event	Time (seconds)
a.	Uncontrolled RCCA Bank withdrawal from a subcritical or Low-Low Power Startup Condition.	Initiation of uncontrolled rod Control withdrawal from 10^{-9} of nominal power	0.0
		Power range high neutron flux low setpoint reached	10.4
		Peak nuclear power occurs	10.6
		Rods begin to fall into core	10.9
		Minimum DNBR occurs	12.6
		Peak heat flux occurs	12.6
		Peak average clad temperature occurs	12.9
		Peak average fuel temperature occurs	13.1
b.	Uncontrolled RCCA Bank Withdrawal at Power (minimum feedback)		
	Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (75 pcm/sec)	0.0
		Power range high neutron flux high setpoint reached	1.9
		Rods begin to fall into core	2.4
		Minimum DNBR occurs	2.8

Accident		Event	Time (seconds)
	Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
		Overtemperature ΔT reactor trip setpoint reached	78.9
		Rods begin to fall into core	80.9
		Minimum DNBR occurs	81.0
c.	CVCS Malfunctions that Result in a Decrease in the Boron Concentration in the Reactor Coolant		
	1. Dilution during refueling	Dilution begins	0
		Shutdown Monitor alarm occurs	T
		Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 1800
	2. Dilution during cold shutdown (filled loops)	Dilution begins	0
		Shutdown Monitor alarm occurs	T
		Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
	3. Dilution during cold shutdown (drained loops)	Dilution begins	0
		Shutdown Monitor alarm occurs	T
		Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
	4. Dilution during hot shutdown	Dilution begins	0

Accident		Event	Time (seconds)
		Shutdown Monitor alarm occurs	T
		Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
5.	Dilution during hot standby		
		Dilution begins	0
		Shutdown Monitor alarm occurs	T
		Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
6.	Dilution during startup	Dilution begins	0
		Source range high neutron flux trip/alarm	T
		Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
7.	Dilution during power operation		
	(a) Automatic reactor control	Dilution begins	0
		Rod insertion limit alarms occur at	T
		Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
	(b) Manual reactor control	Dilution begins	0
		Rod insertion limit alarms, or OTΔT or other trip alarm at	T

Accident			Event	Time (seconds)
			Operator isolates source of dilution prior to loss of all shutdown margin at time ≥	T + 900
d.	Rod Cluster Control Assembly Ejection			
	1.	Beginning-of-Life, Full Power	Initiation of rod ejection	0.0
			Power range high neutron flux setpoint reached	0.04
			Peak nuclear power occurs	0.135
			Rods begin to fall into core	0.54
			Peak fuel average temperature occurs	2.10
			Peak clad temperature occurs	2.15
			Peak heat flux occurs	2.16
	2.	End-of-Life, Zero Power	Initiation of rod ejection	0.0
			Power range high neutron flux low setpoint reached	0.19
			Peak nuclear power occurs	0.22
			Rods begin to fall into core	0.69
			Peak heat flux occurs	1.51
			Peak clad temperature occurs	1.51
			Peak fuel average temperature occurs	1.72

TABLE 15.4-2 PARAMETERS USED IN THE RCCA EJECTION ACCIDENT

Time In Life – Power	BOL-HFP	BOL-HZP	EOL-HFP	EOL-HZP
Power level, %	100	0	100	0
Ejected rod worth (% $\Delta\rho$)	0.25	0.78	0.25	0.85
Delayed neutron fraction, %	0.54	0.54	0.44	0.44
Feedback reactivity weighing	1.355	2.081	1.486	3.765
Trip reactivity (% $\Delta\rho$)	4.0	2.0	4.0	2.0
F_q before rod ejection	2.5	--	2.5	--
F_q after rod ejection	6.0	11.5	7.0	26.0
Number of Operation Pumps	4	2	4	2
Max. Fuel C/L Temperature, °F	4929	3835	4850	3938
Max. Fuel Avg. Temperature, °F	3795	3340	3796	3516
Max. Fuel stored energy, cal/gm [Btu/lb]	163.7 [294.7]	140.7 [253.2]	163.8 [294.8]	149.4 [268.9]
Fuel Melt (%)	0.31	0	1.79	0

TABLE 15.4-3 ROD CLUSTER CONTROL ASSEMBLY (RCCA) EJECTION – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
% DNB Fuel	15%
% Fuel Centerline Melt	0.375%
LOCA Containment Leakage Source Term	Table 15C-1
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium Activity (0.1 μCi/gm DE I-131)	Table 15C-3
Release from DNB Fuel	Section 1 of Appendix H to RG 1.183
Release from Fuel Centerline Melt Fuel	Section 1 of Appendix H to RG 1.183
Steam Generator Secondary Side Partition Coefficient	100
Steam Generator Tube Leakage	1.0 gpm total
Time to establish shutdown cooling and terminate steam release	8 hours
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for fuel failure dose contribution to maximum SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution.
SG Secondary Side Mass	minimum – 99,304 lb_m (one SG) Minimum mass used for SGs to maximize steam release nuclide concentration.
Chemical Form of Iodine Released to Containment	Particulate – 95% Elemental – 4.85% Organic – 0.15%

Input/Assumption	Value
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97% Organic – 3%
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Time of Control Room Ventilation System Isolation	30 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183, Section 4.2.6
Containment Volume Containment Leakage Rate 0 to 24 hours after 24 hours	2.704E+06 ft^3 0.15% (by weight)/day 0.075% (by weight)/day
Secondary Containment Filter Efficiency	Particulate – 95% Elemental – 95% Organic – 85%
Secondary Containment Drawdown Time	8 minutes
Secondary Containment Bypass Fraction	60%
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 2.2 hr^{-1} Organic Iodine - None

TABLE 15.4-4 RCCA EJECTION STEAM RELEASE RATE

Time (hours)	**SG Steam Release Rate (lb_m/min)**
0.0 – 2.0	3392
2.0 – 8.0	2675
8.0 – 720.0	0

TABLE 15.4-5 RCCA EJECTION DOSE CONSEQUENCES

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
RCCA Ejection – Containment Release (3)	1.73	1.97	4.96
RCCA Ejection – Secondary Release (4)	2.57	1.99	3.22
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) Based on an unfiltered control room inleakage rate of 190 cfm
(4) Based on an unfiltered control room inleakage rate of 150 cfm

TABLE 15.4-6 DELETED

TABLE 15.4-7 DELETED

TABLE 15.4-8 DELETED

TABLE 15.4-9 DELETED

TABLE 15.4-10 DELETED

TABLE 15.4-11 DELETED

TABLE 15.4-12 DELETED

TABLE 15.5-1 TIME SEQUENCE OF EVENTS FOR INCREASE IN REACTOR COOLANT INVENTORY EVENTS

Accident	Event	Time (seconds)
Inadvertent Actuation of ECCS During Power Operation	Spurious SI signal generated; two charging pumps begin injecting borated water	0.0
	Terminate flow from all but one centrifugal charging pump	540.0
	Terminate all charging flow	780.0
	Peak pressurizer water volume occurs (6.0 cu. ft. below pressurizer fill)	1712.1

TABLE 15.5-2 TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec)
CVCS Malfunction	Two pressurizer level channels fail low; maximum charging is begun; letdown is isolated; low pressurizer level alarm	0.0
	High pressurizer level alarm	684.10
	Pressurizer fills	794.10

TABLE 15.5-3 DELETED

TABLE 15.5-4 DELETED

TABLE 15.5-5 DELETED

TABLE 15.6-1 TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT INVENTORY

Accident			Event	Time (Seconds)
a.	Inadvertent Opening of a Pressurizer Safety Valve		Safety valve opens fully	0
			Low Pressurizer Pressure reactor trip setpoint reached	25.6
			Rods begin to fall into core	27.6
			Minimum DNBR occurs	27.6
b.	Large Break LOCA		See Figure 15.6-3	
c.	Small Break LOCA			
	1.	3 inch	Start	0
			Reactor trip signal	17
			Top of core uncovered	669
			Accumulator injection begins	N/A
			Peak clad temperature occurs	1319
			Top of core covered	2361
	2.	4 inch	Start	0
			Reactor trip signal	9.9
			Top of core uncovered	523
			Accumulator injection begins	806
			Peak clad temperature occurs	884
			Top of core covered	1507
	3.	6 inch	Start	0
			Reactor trip signal	5.5

Accident	Event	Time (Seconds)
	Top of core uncovered	222
	Accumulator injection begins	363
	Peak clad temperature occurs	416
	Top of core covered	457

TABLE 15.6-2 LETDOWN LINE RUPTURE – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (including uncertainty)
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium Iodine Activity (0.1 μCi/gm DE I-131)	Table 15C-3
Iodine spike appearance rate	500 times (see Table 15.6-4 for values)
Duration of accident initiated spike	8 hrs
Condenser Decontamination Factor	100
Steam Generator Tube Leakage	1 gpm (total for all SGs)
RCS Mass	minimum – 434,044 lb_m maximum – 539,037 lb_m Minimum mass used for iodine spike dose contribution to maximum SG tube leakage activity. Maximum mass used for RCS initial activity dose contribution
Letdown Line Rupture flow rate	140 gpm (1160 lb/min) for 30 minutes
Letdown Line Flashing Fraction	0.1815 at 380°F and 2235 psia
Letdown Line Rupture Release Point	Worst release point (to the CR) from the PAB
Secondary Side Release Point	Worst release point (to the CR) from the Turbine Building (condenser)
Control Room Ventilation System Time of automatic control room isolation	 30 seconds
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Breathing rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

TABLE 15.6-3 IODINE EQUILIBRIUM APPEARANCE ASSUMPTIONS

Input Assumption	**Value**
Maximum Letdown Flow	120 gpm
Assumed Letdown Flow*	132 gpm at 115°F, 2235 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	505,000 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* maximum letdown flow plus 10% uncertainty

TABLE 15.6-4 CONCURRENT IODINE SPIKE (500X) ACTIVITY APPEARANCE RATE

Isotope	Appearance Rate (Ci/min)
Iodine-131	209.029344
Iodine-132	235.958907
Iodine-133	404.536383
Iodine-134	317.823719
Iodine-135	315.402448

TABLE 15.6-5 LETDOWN LINE RUPTURE DOSE CONSEQUENCES

Case	EAB Dose [1] (rem TEDE)	LPZ Dose [2] (rem TEDE)	Control Room Dose [2] (rem TEDE)
Letdown Line Rupture	0.46	0.28	1.66
Acceptance Criteria	2.5 [3]	2.5 [3]	5 [4]

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered control room inleakage of 300 cfm
(3) RG 1.183, Table 6
(4) 10 CFR 50.67

TABLE 15.6-6 STEAM GENERATOR TUBE RUPTURE (SGTR) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	3659 MW_{th} (include uncertainty)
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100 E-bar gross activity)	Table 15C-2
Initial Secondary Side Equilibrium iodine Activity (0.1 μCi/gm DE I-131)	Table 15C-3
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate (the selected split between SGs maximizes dose)	Faulted SG – 313.33 gallons/day Intact SGs – 1126.67 gallons/day
Time to establish shutdown cooling and terminate steam release	8 hours
Time for RCS to reach 212°F and terminate SG tube leakage	48 hours
RCS Mass	539,037 lb_m
SG Secondary Side Mass	99,304 lb_m per SG (minimum mass used to maximize concentration)
Release Rates	Table 15.6-8
Secondary Coolant Iodine Activity prior to accident	0.1 μCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG (flashed tube flow) – none Faulted SG (non-flashed tube flow) – 100 Intact SGs – 100
Break Flow Flash Fraction	Table 15.6-8
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Control Room Ventilation System Time of automatic control room normal intake Isolation and switch to emergency mode	30 seconds
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183, Section 4.2.6

TABLE 15.6-7 TIME SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE WITH A LOSS OF OFFSITE POWER

Event	Time (Seconds)
Tube rupture occurs	0
Reactor trip signal, Loss of offsite power	206
Main feedwater flow terminated *	206
Safety injection signal	214
Safety injection	214
Auxiliary feedwater flow commences	283
ASDV on steam line from ruptured steam generator fails open	1166
ASDV block valve closed	2366
Break flow terminated	6098

* Sequence of events provided is for the scenario leading to maximum radiological consequences. This analysis conservatively assumed instantaneous isolation of Main Feedwater coincident with reactor trip actuation to minimize water inventory and partitioning of radioactive gases in the ruptured steam generator.

TABLE 15.6-8 SGTR RELEASE INFORMATION

Tube Break Flow - ASDV Failure Case 1

Time (hours)	Break Flow (lb_m/sec)
0.000000	12.5
0.002778	46.2
0.274167	34.9
0.500000	36.7
0.753611	42.7
1.000000	43.8
1.253611	41.4
1.461944	37.2
1.712778	37.3
1.762778	34.1
1.778333	26.2
1.793611	3.9
1.825278	4.6
1.901944	12.7
2.000000	12.6
2.777778	0

Tube Break Flow Flashing Fraction – ASDV Failure Case 1

Time (hours)	Flashing Fraction
0.000000	0.17688
0.002778	0.17864
0.274167	0.07193
0.500000	0.06080
0.753611	0.12432
1.000000	0.11501
1.253611	0.03959
1.461944	0.00229
1.712778	0.00000

Tube Break Flow – ASDV Failure Case 2

Time (hours)	Break Flow (lb_m/sec)
0.000000	12.5
0.002778	46.2
0.274167	38.1
0.416667	43.1
0.555556	43.3
0.694444	43.8
0.825000	40.1
1.032500	36.6
1.197222	37.5
1.381389	39.0
1.431389	17.3
1.495833	8.3
1.777778	2.9
1.888889	1.5
2.000000	0.0

Tube Break Flow Flashing Fraction – ASDV Failure Case 2

Time (hours)	Flashing Fraction
0.000000	0.17688
0.002778	0.17864
0.274167	0.11334
0.416667	0.14772
0.555556	0.13800
0.694444	0.12804
0.825000	0.13375
1.032500	0.05482
1.197222	0.01417
1.381389	0.00000

Intact Steam Generator Steam Release – ASDV Failure Case 1

Time (hours)	Steam Release from Unaffected SGs (lb_m/min)
0.000000	217,542
0.002778	216,967
0.274167	3,630
1.461944	9,959
1.778333	1,934
2.0	3,056
8.0	0.0
720.0	0.0

Intact Steam Generator Steam Release – ASDV Failure Case 2

Time (hours)	Steam Release from Unaffected SGs (lb_m/min)
0.0	217,542
0.002778	216,967
0.274167	4,752
0.825000	2,361
1.032500	15,738
1.197222	4,393
1.888889	4,772
2.0	3,056
8.0	0.0
720.0	0.0

Faulted Steam Generator Steam Release – ASDV Failure Case 1

Time (hours)	Steam Release from Faulted SG (lb_m/min)
0.000000	72,393
0.002778	73,140
0.274167	2,743
0.500000	11,860
0.753611	7,032
1.000000	4,843
1.253611	13.9
1.461944	0
2.0	42.6
8.0	0.0
720.0	0.0

Faulted Steam Generator Steam Release – ASDV Failure Case 2

Time (hours)	Steam Release from Faulted SG (lb_m/min)
0.000000	72,393
0.002778	73,140
0.274167	7,782
0.416667	6,446
0.555556	5,547
0.694444	4,819
0.825000	0.00
2.0	42.6
8.0	0.0
720.0	0.0

TABLE 15.6-9 60 µCi/gm DE I-131 ACTIVITIES

Isotope	Activity (µCi/gm)
Iodine-131	46.36
Iodine-132	16.88
Iodine-133	74.18
Iodine-134	10.76
Iodine-135	40.80

TABLE 15.6-10 IODINE EQUILIBRIUM APPEARANCE ASSUMPTIONS

Input Assumption	Value
Maximum Letdown Flow	120 gpm
Assumed Letdown Flow*	132 gpm at 115°F, 2235 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	505,000 lb_m
I-131 Decay Constant	5.986968E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

* maximum letdown flow plus uncertainty

TABLE 15.6-11 CONCURRENT IODINE SPIKE (335 X) ACTIVITY APPEARANCE RATE

Isotope	Appearance Rate (CI/min)
Iodine-131	140.04966
Iodine-132	158.092468
Iodine-133	271.039377
Iodine-134	212.941892
Iodine-135	211.31964

TABLE 15.6-12 SGTR DOSE CONSEQUENCES

ASDV Fails Open at 33% Level in Faulted Steam Generator (Case 1)

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
SGTR pre-accident iodine spike	3.78	1.85	2.02
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	2.21	1.10	1.35
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

ASDV Fails Open 3 Minutes Following Reactor Trip (Case 2)

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
SGTR pre-accident iodine spike	3.78	1.85	2.02
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	2.03	1.00	1.27
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm
(3) RG 1.183, Table 6
(4) 10 CFR 50.67

TABLE 15.6-13 INPUT PARAMETERS USED IN THE SBLOCA ECCS ANALYSIS

Analyzed core power (MWt)	3659
Peak linear power (kW/ft.)	13.94
Power shape	See Figure 15.6-4 (+20% axial offset)
Fuel assembly array	17x17 ZIRLO
Accumulator water volume, nominal (ft^3/ accumulator)	850
Accumulator gas pressure, minimum (psia)	600
Safety injection pump flow	See Figure 15.6-49
Nominal vessel average temperature range	571.0 – 589.1°F
Reactor coolant pressure (psia)	2250
Steam generator tube plugging level (%)	10

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TABLE 15.6-15 SMALL BREAK LOCA RESULTS - FUEL CLADDING DATA

Results	3 Inch	4 Inch	6 Inch
Peak clad temperature (°F)	1114	1373	1156
Peak clad temperature location (ft)	11.25	11.25	11.0
Local Zr/H_20 reaction, maximum (%)	0.06	0.20	0.02
Local Zr/H_20 reaction location (ft)	11.25	11.25	11.0
Total Zr/H_20 reaction (%)	<1.0	<1.0	<1.0
Hot rod burst	None	None	None

TABLE 15.6-16 LOSS OF COOLANT ACCIDENT (LOCA) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Release Inputs:	
Core Power Level	3659 MW_{th} (include uncertainty)
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Initial RCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 15C-2
RCS Mass	539,037 lb_m
Core Fission Product Inventory	Table 15C-1
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.15% (by weight)/day 0.075% (by weight)/day
LOCA release phase timing and duration	Table 15.6-17
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	RG 1.183, Sections 3.1, 3.2, and Table 2
<u>ECCS Systems Leakage (from 26 minutes to 30 days)</u> Sump volume (minimum) ECCS Leakage (2 times allowed value) Flashing Fraction Chemical form of the iodine in the sump water at the time of recirculation (based on pH history) after 26 minutes Released via plant vent after filtration	 69,159.75 $ft.^3$ 48 gpd All available elemental and organic iodine assumed to be released 98.85% aerosol, 1.0% elemental, and 0.15% organic
<u>RWST Back-leakage</u> Sump volume (minimum) ECCS Leakage to RWST (2 times allowed value) Flashing Fraction (elemental iodine assumed to be released into tank air space based upon partition factor) RWST liquid/vapor elemental iodine partition factor Chemical form of iodine in the RWST (based on Sump	 69,159.75 $ft.^3$ 0.9595 gpm 0% assumed See Table 15.6-19 99% aerosol, 1.0% elemental

and RWST pH history)	
Input/Assumption	**Value**
Initial RWST Liquid Inventory (minimum at time of recirculation)	47,000 gallons
Release from RWST Vapor Space	Table 15.6-18
Containment Purge Release (unfiltered)	1,000 cfm for 5 seconds
Removal Inputs:	
Containment Aerosol/Particulate Natural Deposition (only credited in unsprayed regions)	0.1/hour
Containment Elemental Iodine Wall Deposition	2.23/hour
Containment Sprayed Region Volume	2,309,000 ft^3
Containment Unsprayed Region Volume	395,000 ft^3
Flowrate Sprayed and Unsprayed Volumes (Based on two turnovers per hour of unsprayed volume)	13,000 cfm
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Particulate Iodine Time to reach DF of 50	 20/hour 2.92 hours 5.75/hour 3.56 hours
Spray Initiation Time	65 seconds (0.018 hours)
Control Room Ventilation System Time to automatic control room normal intake isolation and switch to emergency mode	 30 seconds
Containment enclosure emergency air cleaning system filter efficiency	Particulate – 95% Elemental – 95% Organic – 85%
Containment enclosure emergency air cleaning system drawdown time	8 minutes
Containment enclosure emergency air cleaning system bypass fraction	60%
Containment Purge Filtration	0%
Transport Inputs:	
Containment Leakage Release	Plant vent (filtered by CEVA) and closest containment point (CEVA bypass)
ECCS Leakage	Plant vent
RWST Backleakage	RWST tank

Containment Purge	Plant vent
Input/Assumption	**Value**
Personal Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183, Section 4.2.6

TABLE 15.6-17 LOCA RELEASE PHASES

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

* From RG 1.183, Table 4

TABLE 15.6-18 ADJUSTED RELEASE RATE FROM RWST

Time (hours)	Release Rate (cfm)
0	1.0403E-05
22	2.7185E-05
24	6.4831E-05
100	1.0186E-04
200	1.3059E-04
300	1.5290E-04
400	1.7027E-04
500	1.8411E-04
600	1.8538E-04
700	1.8044E-04

TABLE 15.6-19 RWST ELEMENTAL IODINE PARTITION FACTOR

Time (hours)	Partition Factor
0	47.42
22	45.34
24	45.21
100	42.30
200	41.12
300	40.74
400	40.70
500	40.82
600	40.92
700	41.01

TABLE 15.6-20 LOCA DOSE CONSEQUENCES

Dose Component	EAB Dose [1] (rem TEDE)	LPZ Dose [2] (rem TEDE)	Control Room Dose [2] (rem TEDE)
Containment Purge	4.2391E-04	2.0602E-04	3.8398E-04
Containment Leakage	4.6199	3.2319	3.8024
ECCS Leakage	9.5305E-03	4.7662E-02	2.7304E-02
RWST Backleakage	1.0728E-02	0.14140	0.47062
Radiation Shine			0.45
Total	4.64	3.42	4.75
Acceptance Criteria	25	25	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 150 cfm

TABLE 15.6-21 **DELETED**

TABLE 15.6-22 **DELETED**

TABLE 15.6-23 **DELETED**

TABLE 15.6-24 **DELETED**

TABLE 15.6-25 **DELETED**

TABLE 15.6-26 SEABROOK STATION LARGE BREAK LOCA CONTAINMENT DATA USED FOR CALCULATION OF CONTAINMENT PRESSURE

Net Free Volume	2.974×10^6 ft^3
Initial conditions	
Pressure	14.6 psia
Temperature	90°F
RWST temperature (Spilling SI and Spray)	50°F
Temperature outside containment	50°F
Spray System	
Post-accident spray system initiation delay	
with LOOP	65 sec
without LOOP	39 sec
Maximum spray system delivered flow (both pumps operating)	7000 gal/min
Containment Fan Coolers	N/A

TABLE 15.6-27 SEABROOK STATION LARGE BREAK LOCA CONTAINMENT WALL DATA USED FOR CALCULATION OF CONTAINMENT PRESSURE

Wall Description	Area (ft^2)	Thickness/Material (ft)
1. Containment Cylinder	70,151	0.00075 / Paint 0.03125 / Carbon Steel 4.5 / Concrete
2. Containment Dome	33,856	0.00075 / Paint 0.04167 / Carbon Steel 3.5 / Concrete
3. Misc. Concrete	106,165	0.001417 / Paint 4.46 / Concrete
4. Refueling Canal	7,873	0.01658 / Stainless Steel 2.76 / Concrete
5. Ducts and Trays	74,419	0.0145 / Carbon Steel
6. Structural Steel	71,157	0.0015 / Paint 0.4667 / Carbon Steel
7. Polar Crane	21,912	0.0015 / Paint 0.1192 / Carbon Steel
8. Equipment Steel	2,580	0.00075 / Paint 0.1292 / Carbon Steel
9. Concrete Floor and Sump	12,804	0.00075 / Paint 0.02083 / Carbon Steel 13.9 / Concrete
10. Equipment Hatch	3,575	0.00075 / Paint 0.07033 / Carbon Steel
11. Personnel Hatch	396	0.00075 / Paint 0.068 / Carbon Steel

MATERIAL PROPERTIES

Material	Thermal Conductivity (BTU/hr-°F-ft)	Volumetric Heat Capacity (BTU/ft.^{3}-°F)
Concrete	0.92	22.62
Carbon Steel	27.0	58.80
Stainless Steel	10.0	58.80
Paint	10.54	24.12

TABLE 15.6-28 SEABROOK STATION LARGE BREAK LOCA MASS AND ENERGY RELEASES FROM BCL USED FOR COCO CALCULATION AT SELECTED TIME POINTS

Time (sec)	M&E from Loop Side BCL		M&E from Vessel Side BCL	
	Mass Flow (lbm/s)	Energy Flow (Btu/s)	Mass Flow (lbm/s)	Energy Flow (Btu/s)
0	9713	5396220	-9	0
0.5	25491	14073548	52706	29096774
1	24688	13833361	47023	25944024
1.5	22606	12960468	39639	21865331
2	18961	11125391	33780	18644770
4	9216	6562060	25721	14366584
6	6566	5355779	21480	12373222
8	5738	4802553	18094	10719466
10	5417	4305995	14745	8949362
12	4573	3682444	10461	6957006
14	3383	2862811	7301	5127627
16	1807	1797630	8365	3944964
18	895	1024780	7357	2732203
20	495	593970	7278	2014597
25	68.5	87109	-79	0
30	-3	0	-109	0
35	105	134248	141	96278
40	44	56073	-52	0
45	122	155258	5637	796322
50	170	211272	2631	731484
60	65	83082	347	241723
70	46	59553	65	62391
80	45	57101	45	46792
90	51	65611	67	67140
100	59	75872	114	104642
110	62	78906	110	103744
120	80	98704	333	179209
130	62	78531	236	126023
140	64	80990	414	168110

Time (sec)	M&E from Loop Side BCL		M&E from Vessel Side BCL	
	Mass Flow (lbm/s)	Energy Flow (Btu/s)	Mass Flow (lbm/s)	Energy Flow (Btu/s)
150	65	81994	399	166301
160	66	83364	756	231416
170	65	81052	574	196391
180	64	80468	584	192434
190	66	82776	240	154844
200	68	84175	232	162589

TABLE 15.6-29 **SEABROOK STATION LARGE BREAK LOCA KEY PARAMETERS AND REFERENCE SPLIT BREAK TRANSIENT ASSUMPTIONS**

		Parameter	Reference Split Break Transient	Uncertainty or Bias
1.0	Plant Physical Decription			
	a.	Dimensions	Nominal	ΔPCT_{MOD}[1]
	b.	Flow resistance	Nominal	ΔPCT_{MOD}[1]
	c.	Pressurizer location	Opposite broken loop	Bounded
	d.	Hot assembly location	Under limiting location	Bounded
	e.	Hot assembly type	17x17 RFA with IFMs, non-IFBA and ZIRLO™ clad	Bounded
	f.	SG tube plugging level	High (10%)	Bounded*
2.0	Plant Initial Operating Conditions			
	2.1	Reactor Power		
		a. Core average linear heat rate (AFLUX)	Nominal – Based on 100% of power (3659 MWt)	ΔPCT_{PD}[2]
		b. Hot Rod Peak linear heat rate (PLHR)	FQ = 2.2; Derived from desired Tech Spec (TS) limit FQ = 2.5 and maximum baseload FQ = 2.0	ΔPCT_{PD}[2]
		c. Hot rod average linear heat rate (HRFLUX)	$F_{\Delta H}$ = 1.683; Derived from TS $F_{\Delta H}$ = 1.65	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)	Figure 15.6-5	ΔPCT_{PD}[2]
		g. Low power region relative power (PLOW)	0.2	Bounded*
		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

Parameter	Reference Split Break Transient	Uncertainty or Bias
2.2 Fluid Conditions		
a. T_{avg}	High Nominal T_{avg} (589.1°F)	ΔPCT_{IC}^{3*}
b. Pressurizer pressure	Nominal (2250.0 psia)	ΔPCT_{IC}^{3}
c. Loop flow	93600 gpm	ΔPCT_{MOD}^{1**}
d. T_{UH}	T_{cold}	0
e. Pressurizer level	Nominal at high T_{avg} (60% span)	0
f. Accumulator temperature	Nominal (85.0°F)	ΔPCT_{IC}^{3}
g. Accumulator pressure	Nominal (634.7 psia)	ΔPCT_{IC}^{3}
h. Accumulator liquid volume	Nominal (850 ft^3)	ΔPCT_{IC}^{3}
i. Accumulator line resistance	Nominal	ΔPCT_{IC}^{3}
j. Accumulator boron	Minimum (2300 ppm)	Bounded
3.0 Accident Boundary Conditions		
a. Break location	Cold leg	Bounded
b. Break type	Split	ΔPCT_{MOD}^{1}
c. Break Size	2.0 times cold leg area	ΔPCT_{MOD}^{1}
d. Offsite power	Loss-Of-Offsite-Power (LOOP)	Bounded*
e. Safety injection flow	Minimum	Bounded
f. Safety injection temperature	Nominal (75.0°F)	ΔPCT_{IC}
g. Safety injection delay	Max delay with LOOP (30.0 sec)	Bounded
h. Containment pressure	Bounded – Slightly lower than pressure curve shown in Figure 15.6-21	Bounded
i. Single failure	ECCS: Loss of 1 SI train Containment press: all trains operations	Bounded
j. Control rod drop time	No control rods	Bounded

	Parameter	Reference Split Break Transient	Uncertainty or Bias
4.0	Model Parameters		
	a. Critical flow	Nominal ($C_D = 2.0$)	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]

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

* Confirmed to be limiting

** Assumed to be result of loop resistance uncertainty

TABLE 15.6-30 SEABROOK STATION LARGE BREAK LOCA CONFIRMATORY CASES PCT RESULTS SUMMARY

Case	PCT (°F)		
	Blowdown	1st Reflood	2nd Reflood
Initial Transient	1337	1489	1462
Reduced SGTP (0%)	1322	1424	1376
Offsite Power Available	1296	1327	1364
Low Nominal RCS T_{avg} (571.0°F)	1310	1349	1323
Increased PLOW (0.6)	1337	1462	1450

TABLE 15.6-31 SEABROOK STATION LARGE BREAK LOCA SPLIT BREAK CONFIRMATORY CASES PCT RESULTS SUMMARY

Case	PCT (°F)	
	1st Reflood	2nd Reflood
CD = 1.4	1337	1190
CD = 1.6	1418	1369
CD = 1.8	1501	1499
CD = 2.0 (Limiting)	1570	1567

TABLE 15.6-32 SEABROOK STATION LARGE BREAK LOCA RESULTS

Component	First Reflood	Second Reflood	Criteria
50th Percentile PCT (°F)	< 1520	< 1412	N/A
95th Percentile PCT (°F)	< 1789	< 1724	< 2200
Maximum Local Oxidation (%)	< 3.53		< 17.0
Maximum Total Hydrogen Generation (%)	< 0.30		< 1.0

TABLE 15.6-33 SEABROOK STATION PLANT OPERATING RANGE ALLOWED BY THE BEST-ESTIMATE LARGE BREAK LOCA ANALYSIS

	Parameter	Operating Range
1.0	Plant Physical Decription	
	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 (1)
	e. Hot assembly type	Fresh 17x17 RFA with IFMs, ZIRLO™ clad, and IFBA or Non-IFBA (2)
	f. SG tube plugging level	≤ 10%
2.0	Plant Initial Operating Conditions	
	2.1 Reactor Power	
	a) Core avg linear heat rate	Core power ≤3659 MWt (3)
	b) Peak linear heat rate	$F_Q \leq 2.5$
	c) Hot rod average linear heat rate	$F_{\Delta H} \leq 1.65$
	d) Hot assembly average linear heat rate	$P_{HA} \leq 1.65 / 1.04$

(1) Peripheral locations will not physically be lead power assembly.

(2) Analysis models thimble plugs removed which is judged to bound thimble plug installed. Hence any combination of thimble plugs installed/removed is supported.

(3) Core power analyzed = 3659 MWt with 0% calorimetric uncertainty (bounding).

Parameter	Operating Range
e) Hot assembly peak linear heat rate	$F_{QHA} \leq 2.5/1.04$
f) Axial power dist (PBOT, PMID)	Figure 15.6-20
g) 44 assembly peripheral region relative power (PLOW)	$0.2 \leq PLOW \leq 0.6$
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 (minimum)	800 ppm (at BOL)
l) Rod power census	See Table 15.6-34
2.2 Fluid Conditions	
a) T_{avg}	$571.0 - 2.9 \leq T_{avg} \leq 589.1 + 5.7$°F (4)
b) Pressurizer pressure	$P_{RCS} = 2250 \pm 50$ psia
c) Loop flow	≥ 93600 gpm/loop
d) T_{UH}	Current upper internals, T_{cold} UH
e) Pressurizer level	Normal level, automatic control
f) Accumulator temperature	$70 \leq T_{ACC} \leq 100$°F

(4) 571°F and 589.1°F are nominal values. The +/- values reflect bias and uncertainty.

	Parameter	Operating Range
	g) Accumulator pressure	$585 \leq P_{ACC} \leq 664$ psia
	h) Accumulator volume	$818 \leq V_{ACC} \leq 882$ ft^3
	i) Accumulator fL/D	Current line configuration
	j) Minimum accumulator boron	≥ 2300 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 15.6-35
	f) Safety injection temperature	50 ≤ SI Temp ≤ 100°F
	g) Safety injection delay	≤ 27 seconds (with offsite power available) ≤ 30 seconds (with Loss-Of-Offsite-Power)
	h) Containment pressure	See Figure 15.6-21; and Tables 15.6-26, 27 & 28
	i) Single failure	Loss of one train of pumped ECCS
	j) Control rod drop time	N/A

TABLE 15.6-34 **SEABROOK STATION 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

TABLE 15.6-35 **SEABROOK STATION BEST-ESTIMATE LARGE BREAK LOCA TOTAL MINIMUM INJECTED FLOW FROM CCP, SIP, AND RHR**

RCS Pressure (psig)	Flow Rate (lbm/sec)
0	519.22
20	457.24
40	393.36
60	327.13
80	263.52
100	199.89
120	123.15
200	89.42
300	84.99
400	80.61
500	75.95
600	71.02
700	65.62
800	59.75
900	53.66
1000	47.19
1100	39.75
1200	30.26
1300	20.34
1400	18.48
1500	16.32
1600	14.16
1700	11.77
1800	9.37
1900	6.44
2000	3.50
2100	0.00

TABLE 15.7-1 SEQUENCE OF EVENTS FOR RGWS FAILURE

Approximate Elapsed Time	Events
0 second	Event begins - one carbon delay bed ruptures
0 second	Noble gases are released (Ref. Table 12.2-27)
<1 minute	Radiation and hydrogen alarms alert plant personnel
>1 minute	Operator actions begin with: 1. Purge RGWS with nitrogen 2. Isolate system 3. Evacuate unnecessary personnel from the area

TABLE 15.7-2 RADIOACTIVE GASEOUS WASTE SYSTEM FAILURE – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
RGWS release inventory	Table 15.7-3
RGWS component volume (arbitrary)	10,000 ft^3
Tank leak rate (arbitrarily high)	Entire inventory released within 2 hours
Control Room Ventilation System Time of automatic control room isolation	30 seconds
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

TABLE 15.7-3 RGWS SOURCE TERM (Ci)

Isotope	1st Carbon Delay Bed	2nd Carbon Delay Bed	3rd Carbon Delay Bed	4th Carbon Delay Bed	5th Carbon Delay Bed	Total RGWS Inventory (Curies)
Kr-83m	8.2E+01	1.5E-01	-	-	-	8.2E+01
Kr-85m	7.5E+02	5.2E+01	3.5E+00	2.4E-01	1.7E-02	8.1E+02
Kr-85	1.8E+02	1.8E+02	1.8E+02	1.8E+02	1.8E+02	9.0E+02
Kr-87	1.5E+02	1.4E-02	<1.0E-02	-	-	1.5E+02
Kr-88	1.0E+03	1.5E+01	2.2E-01	<1.0E-02	-	1.0E+03
Xe131m	1.1E+03	5.7E+02	2.8E+02	1.4E+02	7.0E+01	2.2E+03
Xe-133m	3.5E+03	8.8E+01	2.2E+00	5.6E-02	<1.0E-02	3.6E+03
Xe-133	3.0E+05	6.1E+04	1.3E+04	2.6E+03	5.4E+02	3.8E+05
Xe-135m	1.2E+01	-	-	-	-	1.2E+01
Xe-135	3.2E+03	-	-	-	-	3.2E+03
Xe-137**	2.1E-01	-	-	-	-	2.1E-01
Xe-138	8.3E+00	-	-	-	-	8.3E+00

** not included in analysis (insignificant)

TABLE 15.7-4 RGWS FAILURE DOSE CONSEQUENCES

Case	EAB Dose [1] (rem TEDE)	LPZ Dose [2] (rem TEDE)	Control Room Dose [2] (rem TEDE)
RGWS	0.89	0.43	1.02
Acceptance Criteria	2.5	2.5	5

(1) Worst 2-hour dose
(2) Integrated 30-day does based on an unfiltered Control Room inleakage of 300 cfm

TABLE 15.7-5 **DELETED**

TABLE 15.7-6 **DELETED**

TABLE 15.7-7 RADIOACTIVE LIQUID WASTE SYSTEM FAILURE – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
RLWS release inventory	Table 15.7-8
RLWS component volume (arbitrary)	10,000 ft^3
RLWS leak rate (arbitrarily high)	Entire inventory released within 2 hours
Control Room Ventilation System Time of automatic control room isolation	30 seconds
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
CR Occupancy Factors	RG 1.183, Section 4.2.6

TABLE 15.7-8 RLWS SOURCE TERM

Isotope	Letdown Degasifier Release (Curies)	Boron Waste Tank Release (Curies)
I-131	1.7E-01	7.8E-00
I-132	5.5E-02	1.0E-00
I-133	2.2E-01	8.8E-00
I-134	3.0E-02	3.3E-01
I-135	1.4E-01	4.0E-00
Kr-83m	2.6E+01	-
Kr-85m	1.1E+02	-
Kr-85	9.2E+00	-
Kr-87	7.3E+01	-
Kr-88	2.2E+02	-
Xe-131m	4.7E+00	1.3E-01
Xe-133m	4.0E+01	1.5E-01
Xe-133	1.8E+03	4.9E+00
Xe-135m	2.6E+01	1.2E+01
Xe-135	2.1E+02	7.5E+00
Xe-137**	2.0E+00	-
Xe-138	2.1E+01	-

** not included in analysis (insignificant)

TABLE 15.7-9 RLWS FAILURE DOSE CONSEQUENCES

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Boron Waste Tank	0.04	0.02	0.92
Letdown Degasifier	0.04	0.02	0.46
Acceptance Criteria	0.1	0.1	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered control room inleakage of 300 cfm

TABLE 15.7-10 **DELETED**

TABLE 15.7-11 **DELETED**

TABLE 15.7-12 **DELETED**

Deleted

TABLE 15.7-14 SOURCE TERM FOR LIQUID CONTAINING TANK FAILURES [(1)]

Nuclide	Source Term (μCi/gm)
Sr-89	3.0E-02
Sr-90	2.0E-03
Y-91	4.4E-02
Zr-95	5.3E-03
Nb-95	6.7E-03
Cs-134	5.9E+00
Cs-137	3.2E+00
Ce-144	4.4E-03
Mn-54	8.0E-03
Co-60	8.5E-03
Fe-59	7.5E-03
Cr-51	5.3E-03
H-3	4.5E+02

[(1)] Source term is for bounding waste concentrates tank and based on 1% fuel failure.

TABLE 15.7-15 **DELETED**

TABLE 15.7-16 PARAMETERS USED TO CALCULATE GROUNDWATER FLOW RATE (E) AND SR, CS RETARDATION FACTORS

Type of soil(aquifer)		Silty sand
K,	permeability of aquifer	25 gpd/ft^2
P,	porisity of aquifer	0.3
I,	hydraulic gradient	0.06 feet/foot
A,	soil density	2.07 g/cc
B,	volume ratio of water to soil	1 to 4
K_d,	distribution coefficient	70 ml/g
r,	ratio of the weight of mineral to volume of water per unit volume of aquifer material (A/B)	8.3 g/cc

TABLE 15.7-17 RADIONUCLIDE CONCENTRATION

Isotope	Concentration in Marsh (μCi/ml)	Concentration in Hampton Harbor (μCi/ml)
Sr-90	1.7E-08*	6.9E-14
Y-91	1.1E-03	4.6E-09
Zr-95	2.0E-04	8.2E-10
Nb-95	1.7E-05	6.9E-11
Cs-137	6.9E-04	2.8E-09
Ce-144	1.8E-03	7.4E-09
Mn-54	3.4E-03	1.4E-08
Co-60	6.2E-03	2.5E-08
Fe-59	6.6E-05	2.7E-10
Cr-51	3.0E-06	1.2E-11
H-3	8.0E-01	3.3E-06

* 1.7E-08 = 1.7×10^{-8}

TABLE 15.7-18 FUEL HANDLING ACCIDENT (FHA) – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level Before Shutdown	3659 MW_{th} (including uncertainty)
Core Average Fuel Burnup	45,000 MWD/MTU
Discharged Fuel Assembly Burnup	45,000 – 62,000 MWD/MTU
Fuel Enrichment	1.6 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
Number of Fuel Assmblies in the Core	193
Number of Fuel Assemblies Damaged	1
Delay Before Spent Fuel Movement	80 hours
FHA Source Term for a Single Assembly	Table 15C-4
Water Level Above Damaged Fuel Assembly	23 feet minimum
Iodine Decontamination Factors	Elemental – 285 Organic - 1
Noble Gas Decontamination Factor	1
Chemical Form of Iodine in Pool	Elemental – 99.85% Organic – 0.15%
Duration of Release to Environment	2 hrs
Atmospheric Dispersion Factors Offsite Control Room	 Appendix 2Q Tables 2R-2 and 2R-3
Time of Control Room Ventilation System Isolation	30 seconds Includes diesel start time, damper actuation time, instrument delay, and detector response time
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183, Section 4.2.6

TABLE 15.7-19 FUEL HANDLING ACCIDENT DOSE CONSEQUENCES

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
FHA	1.41	0.69	2.39
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose

(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 300 cfm

TABLE 15.7-20 ACTIVITY IN HIGHEST RATED ASSEMBLY AT TIME OF REACTOR SHUTDOWN AND 80 HOURS AFTER SHUTDOWN

	Total Activity	Fraction In	Activity in Cladding Gap (Ci)	
Radionuclide	At Shutdown (Ci)	Cladding Gap	At Reactor Shutdown	80 Hours After Shutdown
I-131	9.0E+05*	0.08	7.2E+03	2.7E+02
I-132	1.3E+06	0.05	6.5E+04	1.6E+02
I-133	1.7E+06	0.05	8.5E+04	3.0E+01
I-134	1.8E+06	0.05	9.0E+04	Negl.**
I-135	1.6E+06	0.05	8.0E+04	Negl.
Kr-85m	2.1E+05	0.05	1.1E+04	Negl.
Kr-85	1.1E+04	0.10	1.1E+03	1.1E+03
Kr-87	4.1E+05	0.05	2.1E+04	Negl.
Kr-88	5.7E+05	0.05	2.9E+04	Negl.
Xe-131m	1.0E+04	0.05	5.0E+02	5.4E+02
Xe-133m	5.6E+04	0.05	2.8E+03	1.4E+03
Xe-133	1.7E+06	0.05	8.5E+04	6.5E+04
Xe-135m	3.6E+05	0.05	1.8E+04	3.0E+00
Xe-135	4.3E+05	0.05	2.2E+04	4.8E+02
Xe-138	1.4E+06	0.05	7.0E+04	Negl.

* $9.0E+05 = 9.0 \times 10^5$
** Negligible (<0.1 Ci)

TABLE 15.7-21 ACTIVITY RELEASED FROM WATER SURFACE AND MAXIMUM BUILDING AIR CONCENTRATION

Radionuclide	Release From Water Surface (Ci)	Max. Containment Air Concentration[a] (μCi/cc)	Max. Fuel Storage Bldg. Air Concentration[b] (μCi/cc)
I-131	2.7E+02*	3.5E-03	3.2E-02
I-132	1.6E+02	2.1E-03	1.9E-02
I-133	3.0E+01	3.9E-04	3.5E-03
Kr-85	1.1E+03	1.4E-02	1.3E-01
Xe-131m	5.4E+02	7.0E-03	6.4E-02
Xe-133m	1.4E+03	1.8E-02	1.6E-01
Xe-133	6.5E+04	8.5E-01	7.7E+00
Xe-135m	3.0E+00	3.9E-05	3.5E-04
Xe-135	4.8E+02	6.2E-03	5.7E-02

[a] Containment Air Volume: 2.715×10^6 ft^3
[b] Fuel Storage Building Air Volume: 3.0×10^5 ft^3 – dropped assembly case
* 6.8E+02 = 6.8×10^2

TABLE 15.7-22 DELETED

TABLE 15.7-23 DELETED

TABLE 15.7-24 DELETED

TABLE 15.7-24A DELETED

TABLE 15.7-25 **DELETED**

TABLE 15.7-25A **DELETED**

TABLE 15.7-26 SUMMARY OF PARAMETERS AND ASSUMPTIONS USED FOR THE SPENT FUEL CASK DROP ACCIDENT [HISTORICAL]

			Conservative Analysis	Realistic Analysis
I.	Data and assumptions used to estimate radioactive source from postulated accident			
	A.	Power level	Appendix 15B	Appendix 15B
	B.	Burnup	Appendix 15B	Appendix 15B
	C.	Percent of fuel perforated	100 (7 assemblies)	100 (7 assemblies)
	D.	Release of activity by nuclide	Table 15.7-27	Table 15.7-27
	E.	Iodine fractions elemental, organic and particulate)	All elemental	All elemental
	F.	Reactor coolant and secondary coolant activity before the accident	NA	NA
II.	Data and assumptions used to estimate activity released			
	A.	Primary containment leak rate	NA	NA
	B.	Secondary containment leak rate	NA	NA
	C.	Valve movement times	NA	NA
	D.	Absorption and filtration efficiency (%)		
	E.	Recirculation system parameters (flow rates versus time, missing factor, etc.)	95	99
	F.	Containment spray parameters	NA	NA
	G.	Containment volumes	NA	NA
	H.	All other pertinent data and assumptions	Subsection 15.7.5.2	Subsection 15.7.5.2

Note: The spent fuel cask drop accident represents assumptions used in the original plant design. This event contains historical information that is not relevent at this time.

			Conservative Analysis	Realistic Analysis
III.	Dispersion data			
	A.	Location of points of release	Appendix 15B	Appendix 15B
	B.	Distances to applicable receptors (e.g., control room, exclusion boundary, and LPZ)	Appendix 15B	Appendix 15B
	C.	χ/Qs at control room, exclusion boundary and LPZ (for time intervals of 2 hrs., 8 hrs., 24 hrs., 4 days and 30 days)	Appendix 15B	Appendix 15B
IV.	Dose data			
	A.	Method of dose calculation	Appendix 15A	Appendix 15A
	B.	Dose conversion assumptions	Appendix 15A	Appendix 15A
	C.	Peak (or f(t)) concentrations in Containment	NA	NA
	D.	Doses	Table 15.7-28	Table 15.7-28

TABLE 15.7-27 ACTIVITY RELEASED TO ENVIRONMENT FROM SPENT FUEL CASK DROP ACCIDENT[*] [HISTORICAL]

	Activity Released (curies)	
Radionuclide	Conservative	Realistic
I-131	8.7E-02[**]	1.6E-03
Kr-85	7.2E+03	6.6E+03
Xe-131M	4.2E-01	9.5E-02
Xe-133	1.8E-03	2.8E-04

Note: The spent fuel cask drop accident represents assumptions used in the original plant design. This event contains historical information that is not relevant at this time.

[*] Based on gap activity of seven fuel assemblies and 150 days decay.
[**] $8.7E-02 = 8.7 \times 10^{-2} = 0.087$

TABLE 15.7-28 OFFSITE DOSES FROM SPENT FUEL CASKHANDLING ACCIDENT (REM) [HISTORICAL]

	Thyroid (rem)	Whole Body (rem)	Skin (rem)
Conservative Analysis			
EAB (0-2 Hours)	1.2E-02*	9.9E-04	8.3E-02
LPZ (0-2 Hours)	5.9E-04	4.9E-04	4.1E-02
Realistic Analysis			
EAB (0-2 Hours)	3.1E-05	1.3E-04	1.1E-02
LPZ (0-2 Hours)	1.1E-05	4.6E-05	3.8E-03

Note: The spent fuel cask drop accident represents assumptions used in the original plant design. This event contains historical information that is not relevant at this time.

* 1.2E-02 = 1.2×10^{-2} = .012

Thyroid Dose Conversion Factors
(Rem/Ci Inhaled)

Iodine Isotope	**R. G. 1.109 DCF Rem/Ci**	**ICRP-30 DCF Rem/Ci**
I-131	1.490E+06	1.080E+06
I-132	1.430E+04	6.438E+03
I-133	2.690E+05	1.798E+05
I-134	3.730E+03	1.066E+03
I-135	5.600E+04	3.130E+04

The Dose Conversion Factors (DCFs) used to determine the radiological consequences of UFSAR Chapter 15 Design Bases Accidents (DBAs) have evolved considerably since the plant's inception in the 1970's when TID14844 values were considered. In general, the thyroid and skin DCFs used for the Seabrook Station DBAs are based on Regulatory Guide 1.109, and the whole body DCFs are from an internally generated document "Gamma Dose Correction Factors for Finite Hemispherical Clouds," model by J. Hamawi, May 31, 1977.

Recent reanalysis of the radiological consequences of certain DBAs (for instance, *the Steam Generator Tube Rupture Event, Containment Fuel Handling Accident and the LOCA Control Room Habitability)* have used the more recent ICRP-30 DCFs. The ICRP-30 dose conversion values (as taken from Federal Guidance Report 11) are used for thyroid, skin, and whole body dose evaluation.

The table above illustrates the differences between the more conservative Regulatory Guide 1.109 DCFs and the more recent ICRP-30 thyroid DCFs. The ICRP-30 values are used industry wide and by the Regulatory staff as referenced in NRC Regulatory Issue Summary 2001-19; "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," October 18, 2001.

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

Reactor Coolant Activity Concentrations Corresponding To Technical Specification Levels For Pre-Existing Iodine Spike

Radionuclide	Activity Concentration μCi/gm	Total Activity Ci
I-131	4.53E+01*	1.04E+04
I-132	1.65E+01	3.78E+03
I-133	7.25E+01	1.66E+04
I-134	1.05E+01	2.40E+03
I-135	3.99E+01	9.14E+03
Kr-83M	2.17E+00	4.97E+02
Kr-85M	8.60E+00	1.97E+03
Kr-85	6.58E-01	1.51E+02
Kr-87	6.58E+00	1.51E+03
Kr-88	1.72E+01	3.94E+03
Xe-131M	3.38E-01	7.74E+01
Xe-133M	2.87E+00	6.57E+02
Xe-133	1.26E+02	2.89E+04
Xe-135M	4.14E+00	9.48E+02
Xe-135	1.57E+01	3.60E+03
Xe-137	8.60E-01	1.97E+02
Xe-138	3.59E+00	8.22E+02

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

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* 4.53E+01 = 4.53×10^1

Reactor Coolant Isotopic Iodine Activity Concentrations For 1 μCi/Gm Dose Equivalent I-131

Radionuclide	Reactor Coolant Concentration (μCi/gm) Equivalent to 1 Dose Equivalent I-131
I-131	7.6E-01*
I-132	2.7E-01
I-133	1.2E+00
I-134	1.7E-01
I-135	6.6E-01

Note: This table represents information used in the original accident analysis. The information presented in this Table is retained for historical purposes.

* 7.6E-01 = 7.6×10^{1}

Reactor Coolant Iodine Activity Concentrations During Coincident Iodine Spike

Conservative Case

	Activity Concentration (µCi/gm) f(t)			
Radionuclide	t = 15 mins	t = 30 mins	t = 2 hours	t = 4 hours
I-131	5.7E+00[*]	1.1E+01	3.8E+01	7.2E+01
I-132	1.2E+01	2.2E+01	6.9E+01	1.0E+02
I-133	1.4E+01	2.6E+01	9.4E+01	1.7E+02
I-134	1.6E+01	3.0E+01	7.0E+01	8.3E+01
I-135	1.3E+01	2.5E+01	8.9E+01	1.5E+02

Realistic Case

	Activity Concentration (µCi/gm) f(t)			
Radionuclide	t = 15 mins	t = 30 mins	t = 2 hours	t = 4 hours
I-131	2.0E+00	3.7E+00	1.3E+01	2.5E+01
I-132	4.3E+00	8.2E+00	2.5E+01	3.SE+01
I-133	4.5E+00	8.6E+00	3.IE+01	5.7E+01
I-134	6.7E+00	1.2E+01	2.8E+01	3.4E+01
I-135	3.8E+00	7.3E+00	2.6E+01	4.4E+01

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

[*] $5.7E+00 = 5.7 \times 10^{0}$

Summary Of Dilution Factors At The Exclusion Radius (Sec/M^3) 914 Meters, Apr 79 - Mar 80 Onsite Meteorology

			Time Interval	Maximum (ESE) Sector Values[(a)]	Overall-Site Values[(b)]
I.	Concentration CHI/Q Values				
	A.	Conservative Estimates	0-1 Hour	$2.67x10^{-4}$	$2.32x10^{-4}$
			1-2 Hours	$1.88x10^{-4}$	$1.72x10^{-4}$
			2-8 Hours	$1.02x10^{-4}$	$9.35x10^{-5}$
			8-24 Hours	$2.58x10^{-5}$	$2.64x10^{-5}$
			1-4 Days	$1.43x10^{-5}$	$1.49x10^{-5}$
			4-30 Days	$7.78x10^{-6}$	$7.57x10^{-6}$
	B.	Realistic Estimates	0-1 Hour	$3.53x10^{-5}$	$3.78x10^{-5}$
			1-2 Hours	$2.66x10^{-5}$	$2.83x10^{-5}$
			2-8 Hours	$1.44x10^{-5}$	$2.26x10^{-5}$
			8-24 Hours	$5.97x10^{-6}$	$1.06x10^{-5}$
			1-4 Days	$5.21x10^{-6}$	$7.45x10^{-6}$
			4-30 Days	$5.74x10^{-6}$	$5.81x10^{-6}$
II.	Effective Gamma CHI/Q Values				
	A.	Conservative Estimates	0-1 Hour	$2.98x10^{-5}$	$3.00x10^{-5}$
			1-2 Hours	$2.05x10^{-5}$	$2.13x10^{-5}$
			2-8 Hours	$1.14x10^{-5}$	$1.12x10^{-5}$
			8-24 Hours	$6.02x10^{-6}$	$6.21x10^{-6}$
			1-4 Days	$3.71x10^{-6}$	$3.74x10^{-5}$
			4-30 Days	$2.37x10^{-6}$	$2.31x10^{-6}$
	B.	Realistic Estimates	0-1 Hour	$6.19x10^{-6}$	$7.23x10^{-6}$
			1-2 Hours	$4.75x10^{-6}$	$5.73x10^{-6}$
			2-8 Hours	$2.66x10^{-6}$	$4.30x10^{-6}$
			8-24 Hours	$1.91x10^{-6}$	$3.39x10^{-6}$
			1-4 Days	$1.48x10^{-6}$	$2.21x10^{-6}$
			4-30 Days	$1.61x10^{-6}$	$1.63x10^{-6}$

(a) The maximum sector conservative CHI/Q values represent the ESE sector's values which are exceeded 0.5 percent of the total time; the maximum sector realistic CHI/Q values represent the ESE sector's median values.

(b) The overall-site conservative CHI/Q values represent the overall-site values which are exceeded 5 percent of the total time; the overall-site realistic CHI/Q values represent the overall-site median values.

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

Summary Of Dilution Factors At The Low Population Zone (Sec/M^3) 2012 Meters, Apr 79 - Mar 80 Onsite Meteorology

		Time Interval	Maximum (ESE) Sector Values[a]	Overall-Site Values[b]
I. Concentration CHI/Q Values				
A.	Conservative Estimates	0-1 Hour	$1.31x10^{-4}$	$1.10x10^{-4}$
		1-2 Hours	$9.17x10^{-5}$	$8.31x10^{-5}$
		2-8 Hours	$4.82x10^{-5}$	$4.49x10^{-5}$
		8-24 Hours	$7.21x10^{-6}$	$7.50x10^{-6}$
		1-4 Days	$4.25x10^{-6}$	$4.32x10^{-6}$
		4-30 Days	$2.25x10^{-6}$	$2.20x10^{-6}$
B.	Realistic Estimates	0-1 Hour	$1.25x10^{-5}$	$1.35x10^{-5}$
		1-2 Hours	$9.68x10^{-6}$	$1.08x10^{-5}$
		2-8 Hours	$5.54x10^{-6}$	$8.53x10^{-6}$
		8-24 Hours	$1.80x10^{-6}$	$3.25x10^{-6}$
		1-4 Days	$1.52x10^{-6}$	$2.23x10^{-6}$
		4-30 Days	$1.70x10^{-6}$	$1.71x10^{-6}$
II. Effective Gamma CHI/Q Values				
A.	Conservative Estimates	0-1 Hour	$1.15x10^{-5}$	$1.20x10^{-5}$
		1-2 Hours	$8.19x10^{-6}$	$8.26x10^{-6}$
		2-8 Hours	$4.43x10^{-6}$	$4.39x10^{-6}$
		8-24 Hours	$2.33x10^{-6}$	$2.43x10^{-6}$
		1-4 Days	$1.44x10^{-6}$	$1.45x10^{-6}$
		4-30 Days	$9.05x10^{-7}$	$8.81x10^{-7}$
B.	Realistic Estimates	0-1 Hour	$2.35x10^{-6}$	$2.77x10^{-6}$
		1-2 Hours	$1.82x10^{-6}$	$2.19x10^{-6}$
		2-8 Hours	$9.99x10^{-7}$	$1.66x10^{-6}$

(a) The maximum sector conservative CHI/Q values represent the ESE sector's values which are exceeded 0.5 percent of the total time; the maximum sector realistic CHI/Q values represent the ESE sector's median values.

(b) The overall-site conservative CHI/Q values represent the overall-site values which are exceeded 5 percent of the total time; the overall-site realistic CHI/Q values represent the overall-site median values.

Time Interval	Maximum (ESE) Sector Values[a]	Overall-Site Values[b]
8-24 Hours	$7.20x10^{-7}$	$1.30x10^{-6}$
1-4 Days	$5.54x10^{-7}$	$8.53x10^{-7}$
4-30 Days	$6.15x10^{-7}$	$6.21x10^{-7}$

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

(a) The maximum sector conservative CHI/Q values represent the ESE sector's values which are exceeded 0.5 percent of the total time; the maximum sector realistic CHI/Q values represent the ESE sector's median values.

(b) The overall-site conservative CHI/Q values represent the overall-site values which are exceeded 5 percent of the total time; the overall-site realistic CHI/Q values represent the overall-site median values.

Control Room Dilution Factors

Time Interval	Accident Control Room CHI/Q Values (sec/m^3) Concentration[a]	Gamma
1 hr	4.08×10^{-3}	2.49×10^{-4}
2 hrs	3.18×10^{-3}	1.92×10^{-4}
8 hrs	2.04×10^{-3}	1.24×10^{-4}
24 hrs	1.44×10^{-3}	8.85×10^{-5}
96 hrs	9.78×10^{-4}	6.02×10^{-5}
720 hrs	7.51×10^{-4}	4.63×10^{-5}

Time Interval	5% Concentration CHI/Q Values (sec/m^3) East CR Intake[b]	West CR Intake
1 hr	1.42×10^{-3}	1.57×10^{-3}
2 hrs	1.14×10^{-3}	9.81×10^{-4}
8 hrs	6.95×10^{-4}	4.59×10^{-4}
24 hrs	4.67×10^{-4}	2.53×10^{-4}
96 hrs	3.05×10^{-4}	1.49×10^{-4}
720 hrs	2.00×10^{-4}	7.77×10^{-5}

Note: This table represents information used in the original accident analysis. The information presented in this table is retained for historical purposes.

[a] These values are appropriate for infiltration or leakage of air into the control room from either containment leakage or a primary vent stack

[b] These values are appropriate for a containment leakage pathway only.

TABLE 15C-1 LOCA CONTAINMENT LEAKAGE SOURCE TERM

Nuclide	Curies	Nuclide	Curies
Kr-85	1.263E+06	**Pu-239**	3.991E+04
Kr-85m	2.492E+07	**Pu-240**	7.068E+04
Kr-87	4.765E+07	**Pu-241**	1.865E+07
Kr-88	6.703E+07	**Am-241**	1.847E+04
Rb-86	3.028E+05	**Cm-242**	8.770E+06
Sr-89	9.245E+07	**Cm-244**	2.598E+06
Sr-90	1.002E+07	**I-130**	7.585E+06
Sr-91	1.134E+08	**Kr-83m**	1.186E+07
Sr-92	1.231E+08	**Xe-138**	1.609E+08
Y-90	1.048E+07	**Xe-131m**	1.179E+06
Y-91	1.198E+08	**Xe-133m**	6.446E+06
Y-92	1.237E+08	**Xe-135m**	4.221E+07
Y-93	1.434E+08	**Cs-138**	1.787E+08
Zr-95	1.644E+08	**Cs-134m**	8.436E+06
Zr-97	1.613E+08	**Rb-88**	6.819E+07
Nb-95	1.664E+08	**Rb-89**	8.726E+07
Mo-99	1.892E+08	**Sb-124**	3.729E+05
Tc-99m	1.657E+08	**Sb-125**	2.414E+06
Ru-103	1.879E+08	**Sb-126**	2.109E+05
Ru-105	1.510E+08	**Te-131**	9.249E+07
Ru-106	1.000E+08	**Te-133**	1.184E+08
Rh-105	1.344E+08	**Te-134**	1.609E+08
Sb-127	1.390E+07	**Te-125m**	5.259E+05
Sb-129	3.781E+07	**Te-133m**	7.129E+07
Te-127	1.381E+07	**Ba-141**	1.585E+08
Te-127m	1.874E+06	**Ba-137m**	1.294E+07
Te-129	3.719E+07	**Pd-109**	6.296E+07
Te-129m	5.518E+06	**Rh-106**	1.115E+08
Te-131m	1.598E+07	**Rh-103m**	1.692E+08
Te-132	1.460E+08	**Tc-101**	1.764E+08
I-131	1.051E+08	**Eu-154**	2.003E+06
I-132	1.491E+08	**Eu-155**	1.387E+06
I-133	1.988E+08	**Eu-156**	4.767E+07
I-134	2.152E+08	**La-143**	1.466E+08
I-135	1.872E+08	**Nb-97**	1.628E+08

Nuclide	Curies	Nuclide	Curies
Xe-133	1.994E+08	**Nb-95m**	1.177E+06
Xe-135	5.012E+07	**Pm-147**	1.379E+07
Cs-134	3.258E+07	**Pm-148**	2.968E+07
Cs-136	8.347E+06	**Pm-149**	6.826E+07
Cs-137	1.365E+07	**Pm-151**	2.409E+07
Ba-139	1.747E+08	**Pm-148m**	3.426E+06
Ba-140	1.684E+08	**Pr-144**	1.350E+08
La-140	1.750E+08	**Pr-144m**	1.609E+06
La-141	1.593E+08	**Sm-153**	7.818E+07
La-142	1.538E+08	**Y-94**	1.448E+08
Ce-141	1.623E+08	**Y-95**	1.560E+08
Ce-143	1.476E+08	**Y-91m**	6.581E+07
Ce-144	1.340E+08	**Br-82**	8.712E+05
Pr-143	1.464E+08	**Br-83**	1.183E+07
Nd-147	6.392E+07	**Br-84**	2.044E+07
Np-239	2.922E+09	**Am-242**	1.285E+07
Pu-238	5.151E+05	**Np-238**	7.062E+07
		Pu-243	1.266E+08

TABLE 15C-2-1 REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES – 1% CLAD DEFECTS

Nuclide	µCi/gm	Nuclide	µCi/gm
I-131	2.5	Sr-91	0.031
I-132	0.91	Y-90	0.00022
I-133	4.0	Y-91	0.0058
I-134	0.58	Y-92	0.001
I-135	2.2	Zr-95	0.00067
H-3	5.0	Nb-95	0.00068
Kr-83m	0.43	Mo-99	3.3
Kr-85m	1.7	Cs-134	0.44
Kr-85	0.13	Cs-136	0.22
Kr-87	1.3	Cs-137	2.2
Kr-88	3.4	Ba-140	0.0045
Xe-131m	0.067	La-140	0.0014
Xe-133m	0.57	Ce-144	0.00044
Xe-133	25	Mn-54	0.00031
Xe-135m	0.82	Co-58	0.016
Xe-135	3.1	Co-60	0.002
Xe-138	0.71	Fe-59	0.001
Sr-89	.0041	Cr-51	0.0019
Sr-90	.00018	Fe-55	0.0016

TABLE 15C-2-2 PARAMETERS USEDIN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES – 1% CLAD DEFECTS

Parameter	Value	
Ultimate Core Thermal Power, MWt	3654	
Clad Defects, as a percent of rated thermal power being generated by rods with clad defects	1.0%	
Reactor Coolant Liquid Volume, ft^3	12,100	
Reactor Coolant Full Power Average Temperature, °F	590	
Purification Flow Rate, Normal gpm	80	
Effective Cation Demineralizer Flow, gpm	7.5	
Fission Product Escape Rate Coefficients	Noble Gas Isotopes, sec^{-1}	$6.5x10^{-8}$
	Br, Rb, I and Cs isotopes, sec^{-1}	$1.3x10^{-8}$
	Te, Se, Tc, Sn, Sb Isotopes, sec^{-1}	$1.0x10^{-9}$
	Mo Isotopes, sec^{-1}	$2.0x10^{-9}$
	Sr, Ba Isotopes, sec^{-1}	$1.0x10^{-11}$
	Y, Zr, Nb, Ru, Rh, La, Ce, Pr, Nd, and Pm Isotopes, sec^{-1}	$1.6x10^{-12}$
Mixed Bed Demineralizer Decontamination Factors	Noble Gases and Cs, Y, and Mo	1.0
	All Other Isotopes Including Corrosion Products	10.0
Cation Bed Demineralizer Decontamination Factor	Cs, Y, and Mo	10.0
Degassifier Noble Gas Stripping factor	1.0	

TABLE 15C-2 PRIMARY COOLANT ACTIVITIES*

Nuclide	µCi/gm	Nuclide	µCi/gm
I-131	0.7727	**Sr-91**	1.341E-01
I-132	0.2813	**Y-90**	9.515E-04
I-133	1.2363	**Y-91**	2.509E-02
I-134	0.1793	**Y-92**	4.325E-03
I-135	0.6800	**Zr-95**	2.898E-03
H-3	2.163E+01	**Nb-95**	2.941E-03
Kr-83m	1.860E+00	**Mo-99**	1.427E+01
Kr-85m	7.353E+00	**Cs-134**	1.903E+00
Kr-85	5.623E-01	**Cs-136**	9.515E-01
Kr-87	5.623E+00	**Cs-137**	9.515E+00
Kr-88	1.471E+01	**Ba-140**	1.946E-02
Xe-131m	2.898E-01	**La-140**	6.055E-03
Xe-133m	2.465E+00	**Ce-144**	1.903E-03
Xe-133	1.081E+02	**Mn-54**	1.341E-03
Xe-135m	3.547E+00	**Co-58**	6.920E-02
Xe-135	1.341E+01	**Co-60**	8.650E-03
Xe-138	3.071E+00	**Fe-59**	4.325E-03
Sr-89	1.773E-02	**Cr-51**	8.218E-03
Sr-90	7.785E-04	**Fe-55**	6.920E-03

* 1.0 µCi/gm Dose Equivalent I-131 and 100/E-bar gross activity

TABLE 15C-3 **SECONDARY SIDE SOURCE ACTIVITIES** *

Isotope	µCi/gm
I-131	0.07727
I-132	0.02813
I-133	0.12363
I-134	0.01793
I-135	0.06800

* 0.1 µCi/gm Dose Equivalent I-131

TABLE 15C-4 FUEL HANDLING ACCIDENT SOURCE TERM

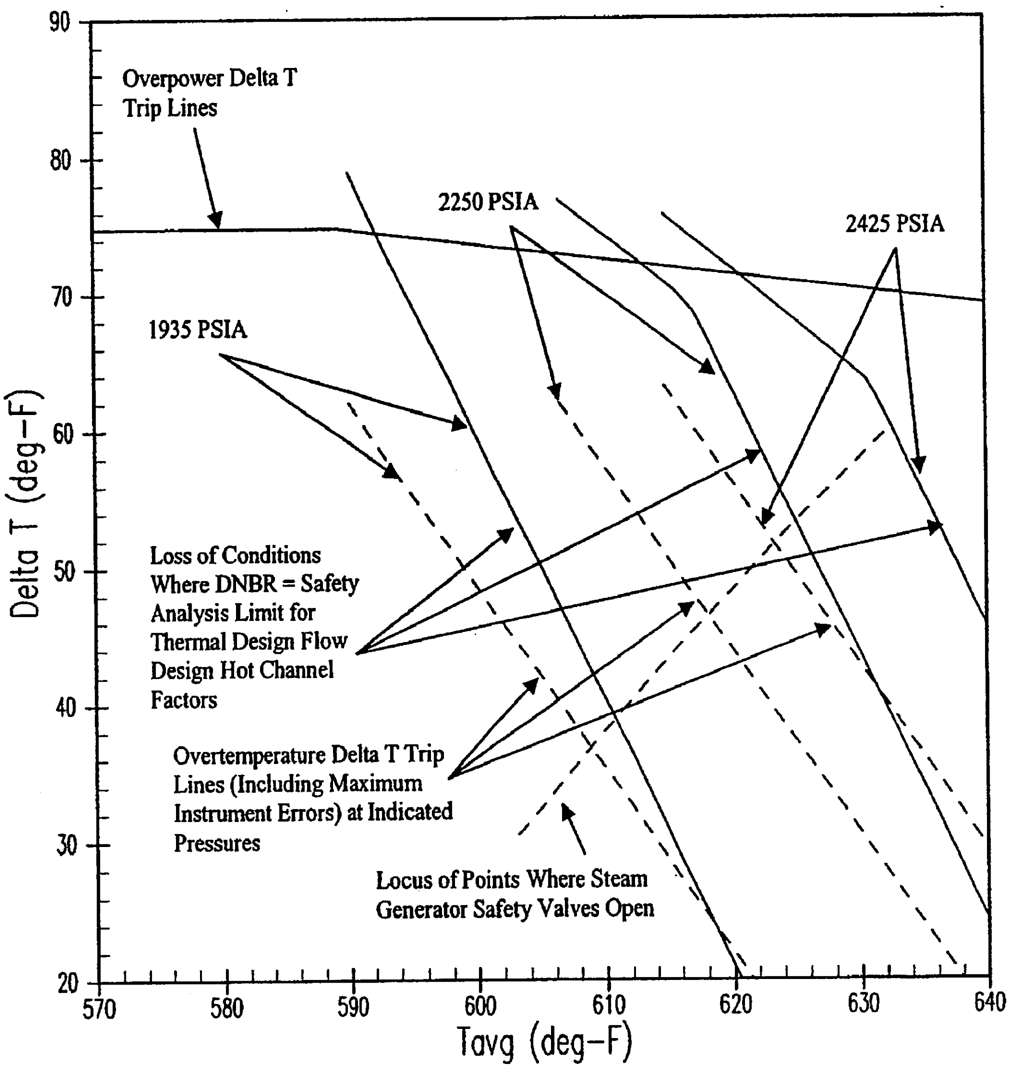
Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)
Kr-85	1.080E+04	**Xe-133**	1.705E+06	**Sb-126**	1.803E+03
Kr-85m	2.130E+05	**Xe-135**	4.285E+05	**Te-131**	7.907E+05
Kr-87	4.074E+05	**Cs-134**	2.785E+05	**Te-133**	1.012E+06
Kr-88	5.731E+05	**Cs-136**	7.136E+04	**Te-134**	1.376E+06
Rb-86	2.589E+03	**Cs-137**	1.167E+05	**Te-125m**	4.496E+03
Sr-89	7.904E+05	**Ba-139**	1.494E+06	**Te-133m**	6.095E+05
Sr-90	8.566E+04	**Ba-140**	1.440E+06	**Ba-141**	1.355E+06
Sr-91	9.695E+05	**La-140**	1.496E+06	**Ba-137m**	1.106E+05
Sr-92	1.052E+06	**La-141**	1.362E+06	**Pd-109**	5.383E+05
Y-90	8.960E+04	**La-142**	1.315E+06	**Rh-106**	9.532E+05
Y-91	1.024E+06	**Ce-141**	1.388E+06	**Rh-103m**	1.447E+06
Y-92	1.058E+06	**Ce-143**	1.262E+06	**Tc-101**	1.508E+06
Y-93	1.226E+06	**Ce-144**	1.146E+06	**Eu-154**	1.712E+04
Zr-95	1.405E+06	**Pr-143**	1.252E+06	**Eu-155**	1.186E+04
Zr-97	1.379E+06	**Nd-147**	5.465E+05	**Eu-156**	4.075E+05
Nb-95	1.423E+06	**Np-239**	2.498E+07	**La-143**	1.253E+06
Mo-99	1.618E+06	**Pu-238**	5.425E+03	**Nb-97**	1.392E+06
Tc-99m	1.417E+06	**Pu-239**	3.412E+02	**Nb-95m**	1.006E+04
Ru-103	1.606E+06	**Pu-240**	6.043E+02	**Pm-147**	1.179E+05
Ru-105	1.291E+06	**Pu-241**	1.594E+05	**Pm-148**	2.537E+05
Ru-106	8.549E+05	**Am-241**	1.579E+02	**Pm-149**	5.836E+05
Rh-105	1.149E+06	**Cm-242**	7.498E+04	**Pm-151**	2.060E+05
Sb-127	1.188E+05	**Cm-244**	3.232E+04	**Pm-148m**	2.929E+04
Sb-129	3.232E+05	**I-130**	6.485E+04	**Pr-144**	1.154E+06
Te-127	1.181E+05	**Kr-83m**	1.014E+05	**Pr-144m**	1.376E+04
Te-127m	1.602E+04	**Xe-138**	1.376E+06	**Sm-153**	6.684E+05
Te-129	3.179E+05	**Xe-131m**	1.008E+04	**Y-94**	1.238E+06
Te-129m	4.717E+04	**Xe-133m**	5.511E+04	**Y-95**	1.334E+06
Te-131m	1.366E+05	**Xe-135m**	3.609E+05	**Y-91m**	5.626E+05
Te-132	1.248E+06	**Cs-138**	1.528E+06	**Br-82**	7.448E+03
I-131	8.985E+05	**Cs-134m**	7.212E+04	**Br-83**	1.011E+05
I-132	1.275E+06	**Rb-88**	5.830E+05	**Br-84**	1.747E+05
I-133	1.700E+06	**Rb-89**	7.460E+05	**Am-242**	1.099E+05
I-134	1.840E+06	**Sb-124**	3.188E+03	**Np-238**	6.347E+05
I-135	1.600E+06	**Sb-125**	2.064E+04	**Pu-243**	1.082E+06

TABLE 15C-5 CONTROL ROOM VENTILATION SYSTEM PARAMETERS

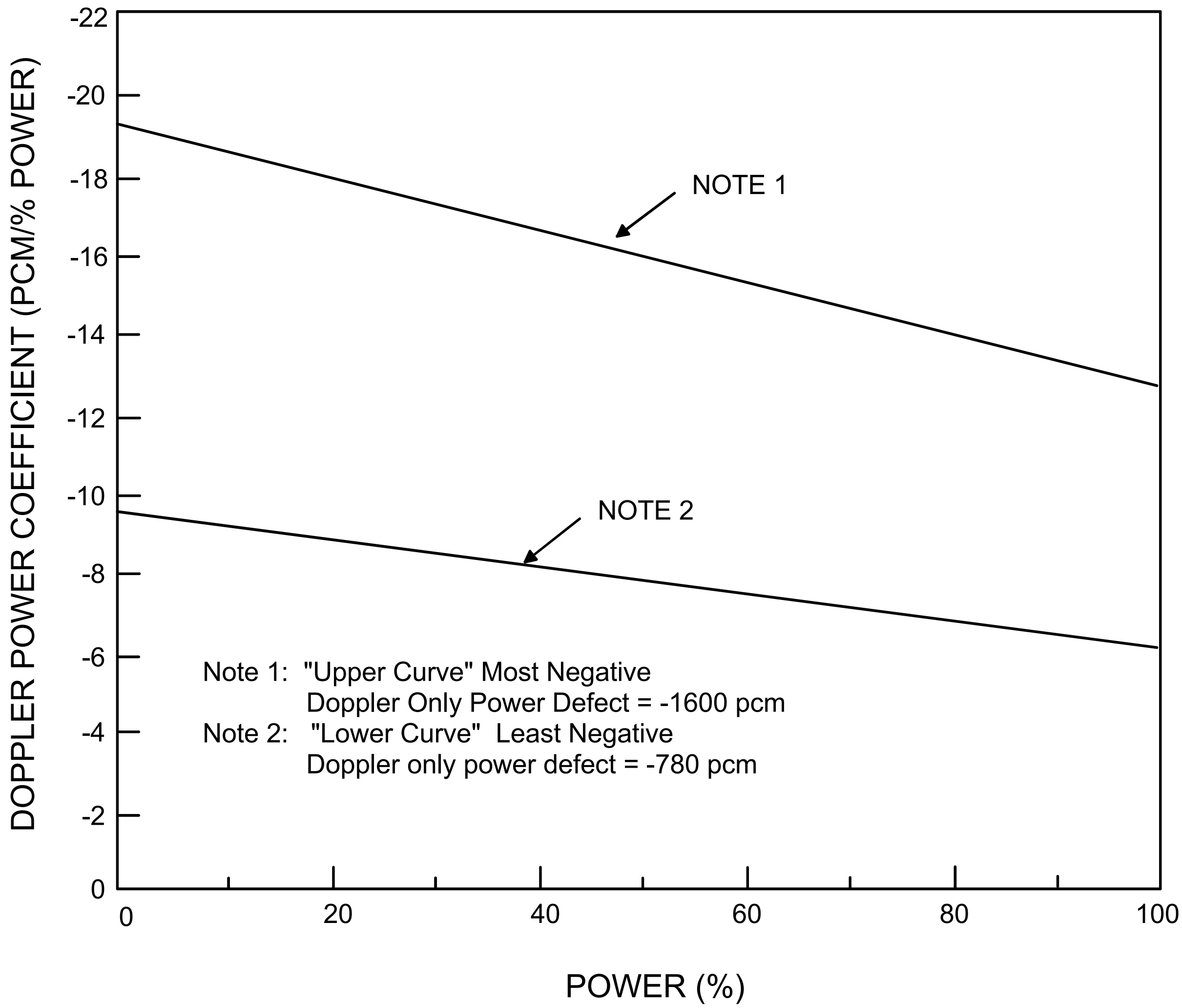
Parameter	**Value**
Control Room Volume	246,000 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	1000 cfm
Unfiltered Inleakage (Total)	
LOCA, MSLB, Locked Rotor, RCCA Ejection – Secondary Release	150 cfm
SGTR, Small Line Break Outside Containment (Letdown Line), Radioactive Gaseous Waste System Failure, Radioactive Liquid Waste System Failure and Fuel Handling Accident	300 cfm
RCCA Ejection – containment release	190 cfm
Emergency Operation	
Filtered Make-up Flow Rate	600 cfm
Filtered Recirculation Flow Rate	390 cfm (entire 10% tolerance on total CR filter flow conservatively applied to reduce recirculation flow)
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage (Total)	
LOCA, MSLB, Locked Rotor, RCCA Ejection – Secondary Release	150 cfm
SGTR, Small Line Break Outside Containment (Letdown Line), Radioactive Gaseous Waste System Failure, Radioactive Liquid Waste System Failure and Fuel Handling Accident	300 cfm
RCCA Ejection – containment release	190 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%

TABLE 15C-6 LOCA DIRECT SHINE DOSE

Source	**Direct Shine Dose (rem)**
Containment	0.001
Filters	0.429
External Cloud	0.020
Total	**0.450**



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Illustration of Overtemperature ΔT and Overpower ΔT Protection	
		Figure 15.0-01



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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Doppler Power Coefficient Assumed in Analyses	
		Figure 15.0-02

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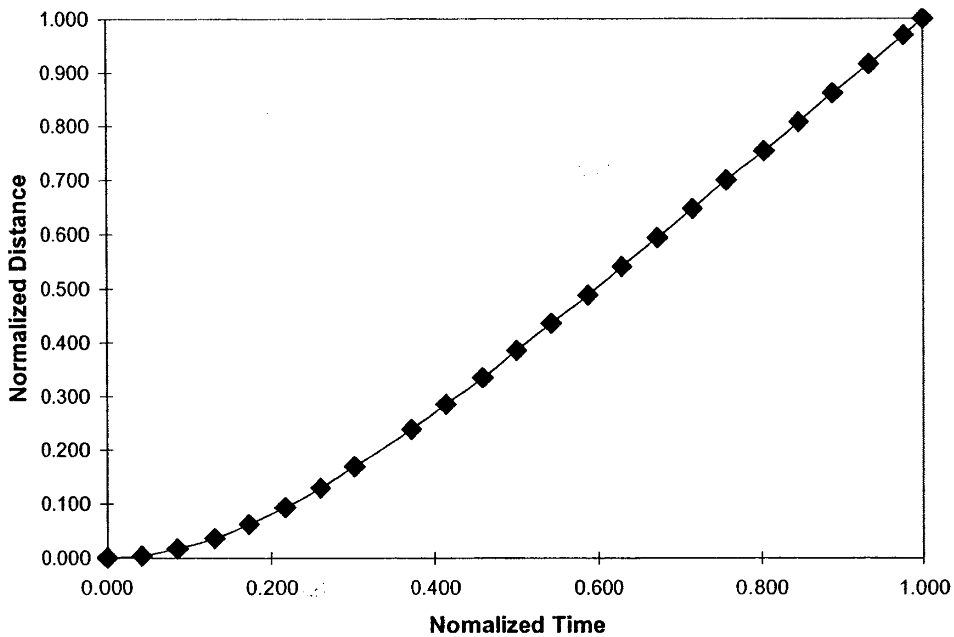
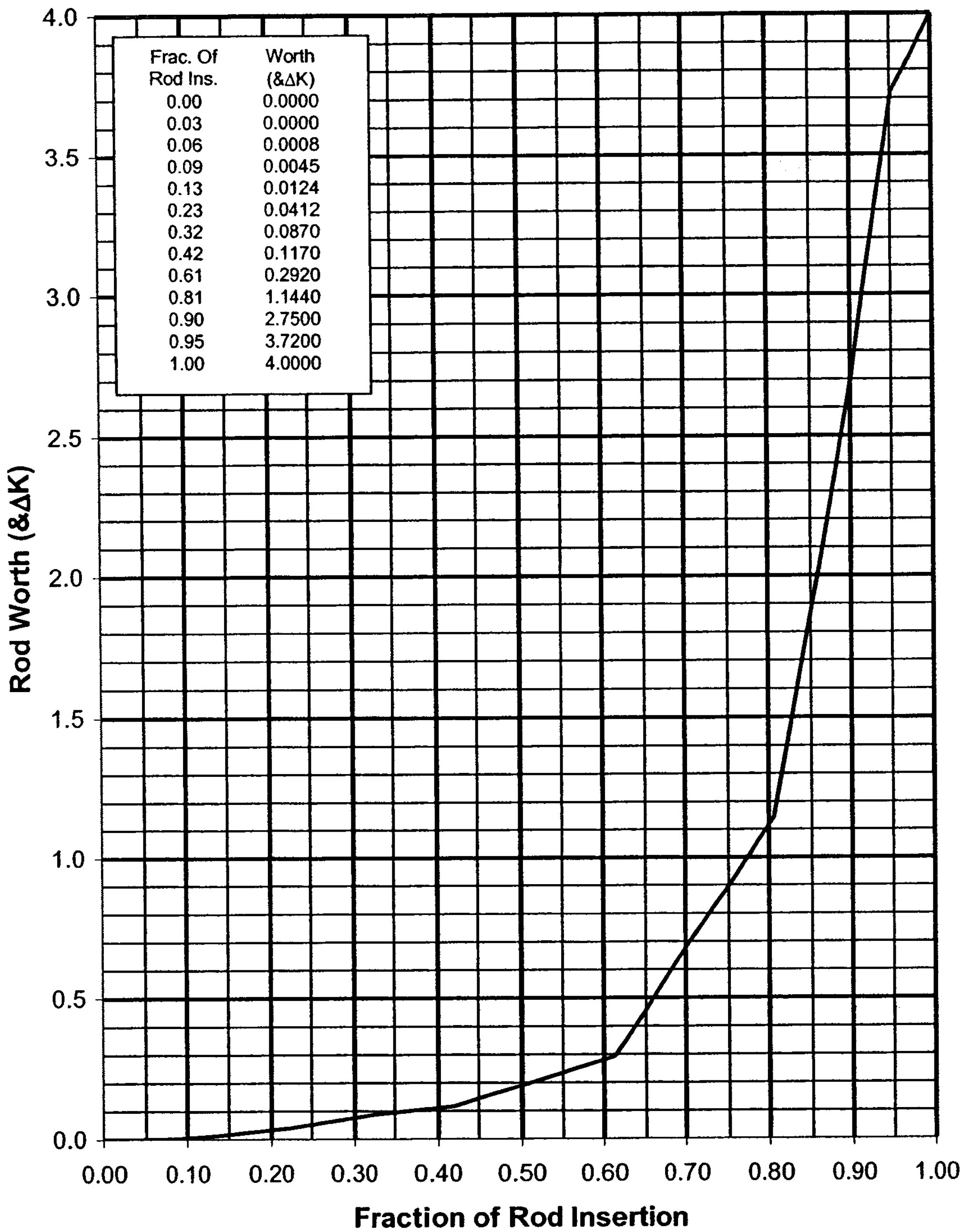
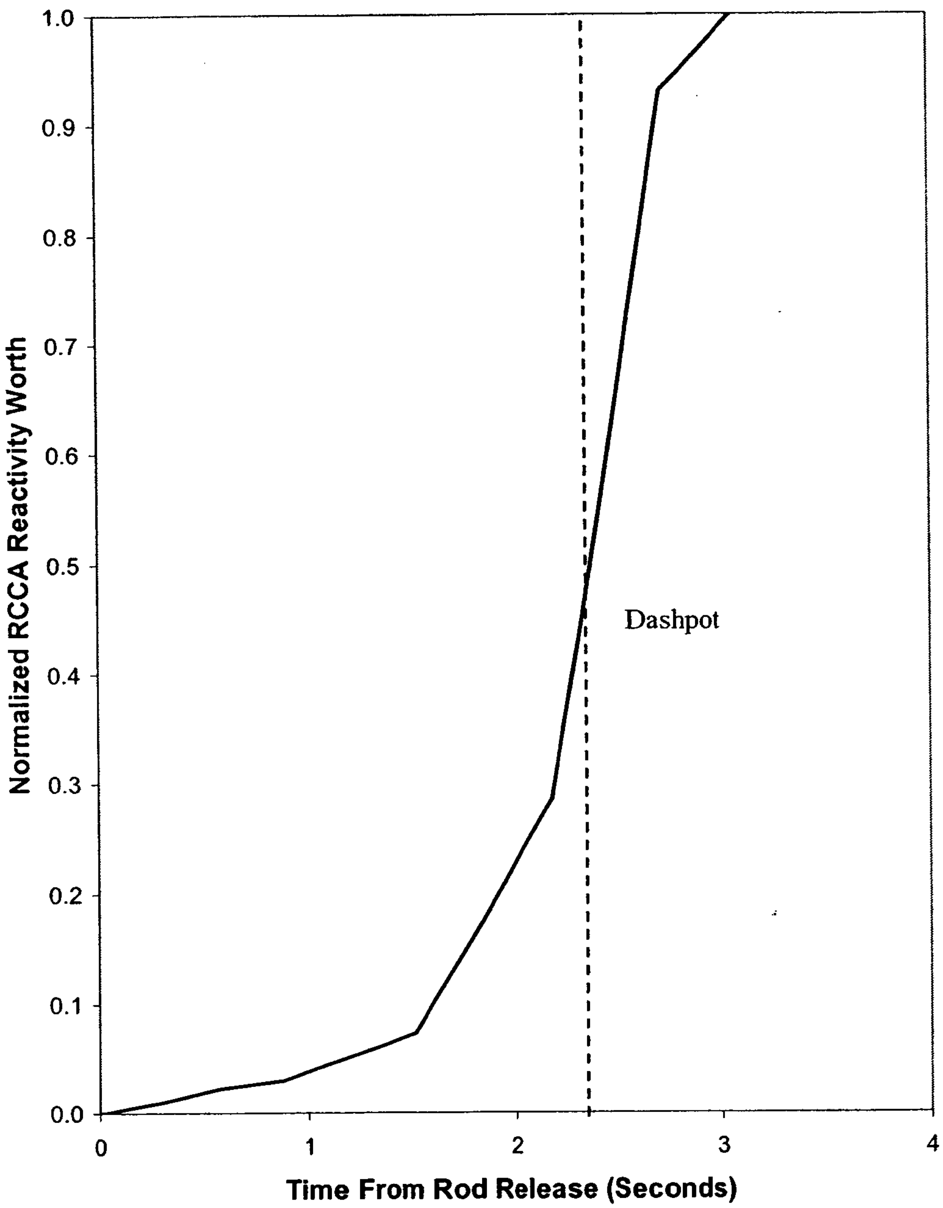


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized RCCA Position (Fraction insertion) vs. Normalized RCCA Drop Time After Release	
		Figure 15.0-04


Frac. Of Rod Ins.
Worth (&ΔK)
0.00 0.0000
0.03 0.0000
0.06 0.0008
0.09 0.0045
0.13 0.0124
0.23 0.0412
0.32 0.0870
0.42 0.1170
0.61 0.2920
0.81 1.1440
0.90 2.7500
0.95 3.7200
1.00 4.0000
Rod Worth (&ΔK)
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
Fraction of Rod Insertion



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Normalized RCCA Reactivity Worth versus Time from Start of Rod Motion	
		Figure 15.0-06

Abbreviations Used:

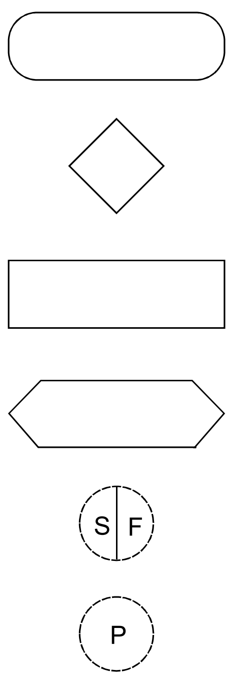
EFWS	- Emergency Feedwater System
CVCS	- Chemical and Volume Control System
ESFAS	- Engineered Safety Features Actuation System
FW	- Feedwater
RTS	- Reactor Trip System
SIS	- Safety Injection System
SI	- Safety Injection
RT	- Reactor Trip
CS	- Containment Spray
ECCS	- Emergency Core Cooling System
HL	- Hot Leg
CL	- Cold Leg
CCWS	- Component Cooling Water System
RCS	- Reactor Coolant System
SWS	- Service Water System
HPI	- High Pressure Injection
LPI	- Low Pressure Injection
CI	- Containment Isolation
SG	- Steam Generator

Notes:

1. For trip initiation and safety system actuation, multiple signals are shown but only a single signal is required. The other signals are backups.

2. No timing sequence is implied by position of various branches. Refer to event timing sequences presented in tabular form in pertinent accident analysis section of Chapter 15.0 of the FSAR.

Diagram Symbols:



- Event Title

- Branch Point for Different Plant Conditions

- Safety System

- Safety Action

- System Required To Meet Single-Failure Criteria (System is Single Failure Proof)

- Manual Action Required During System Operation

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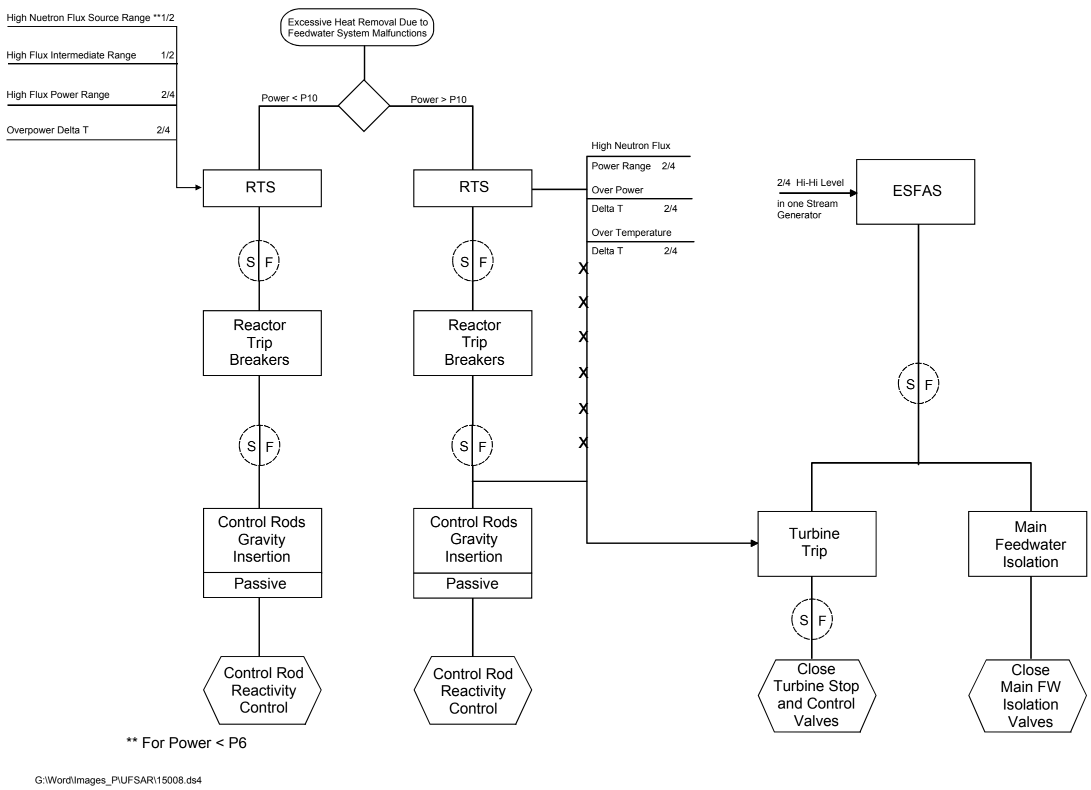
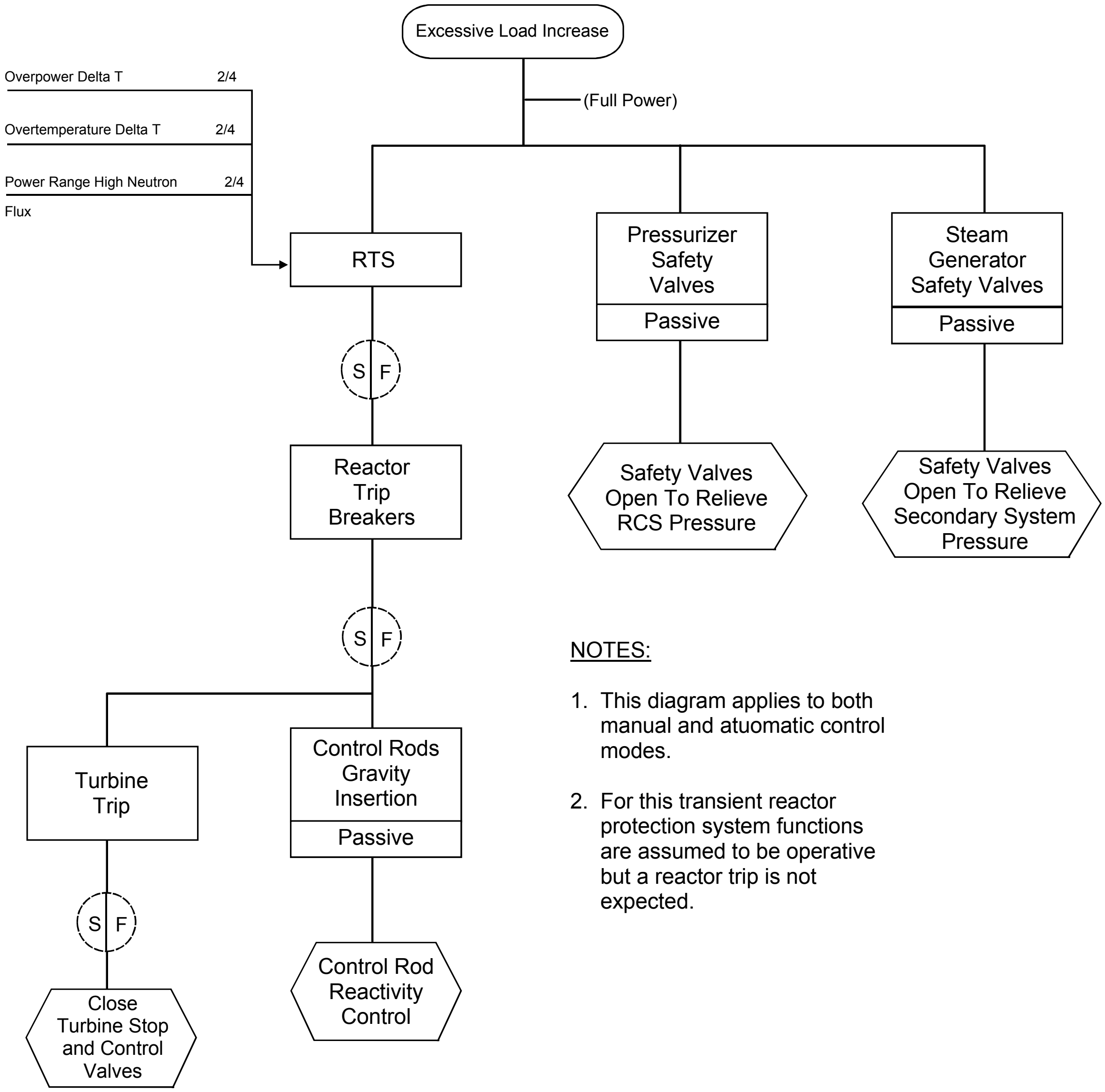


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater System Malfunction	
		Figure 15-0-08



NOTES:

1. This diagram applies to both manual and atuomatic control modes.

2. For this transient reactor protection system functions are assumed to be operative but a reactor trip is not expected.

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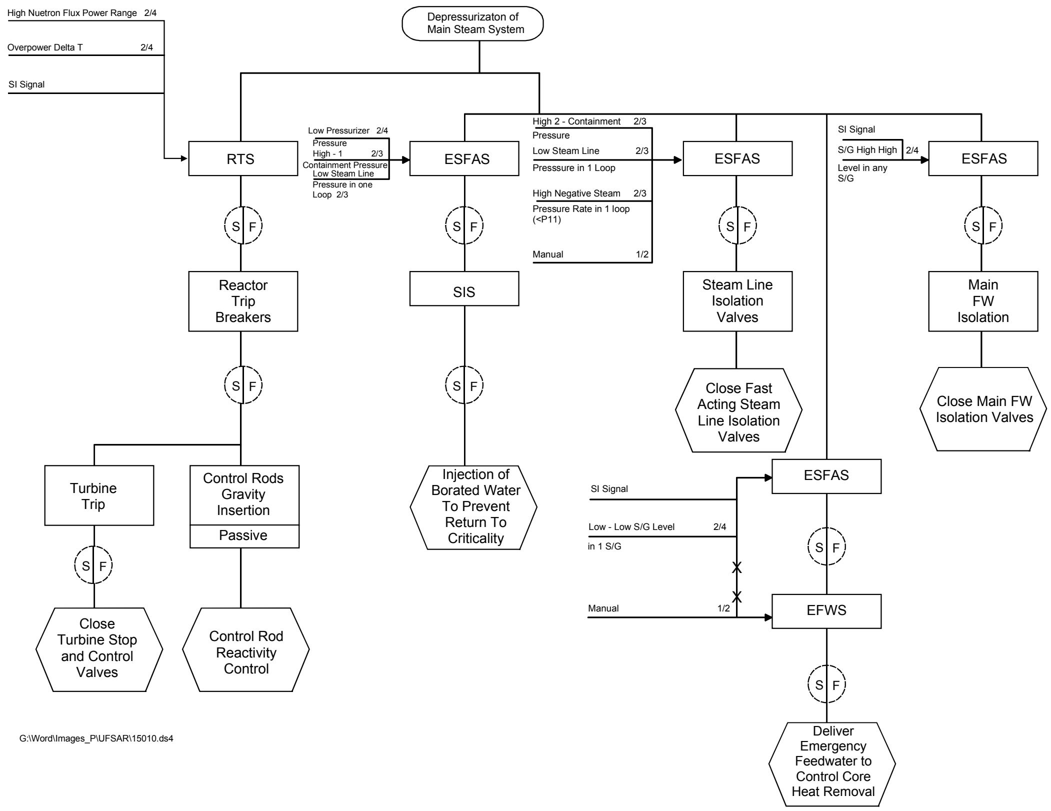
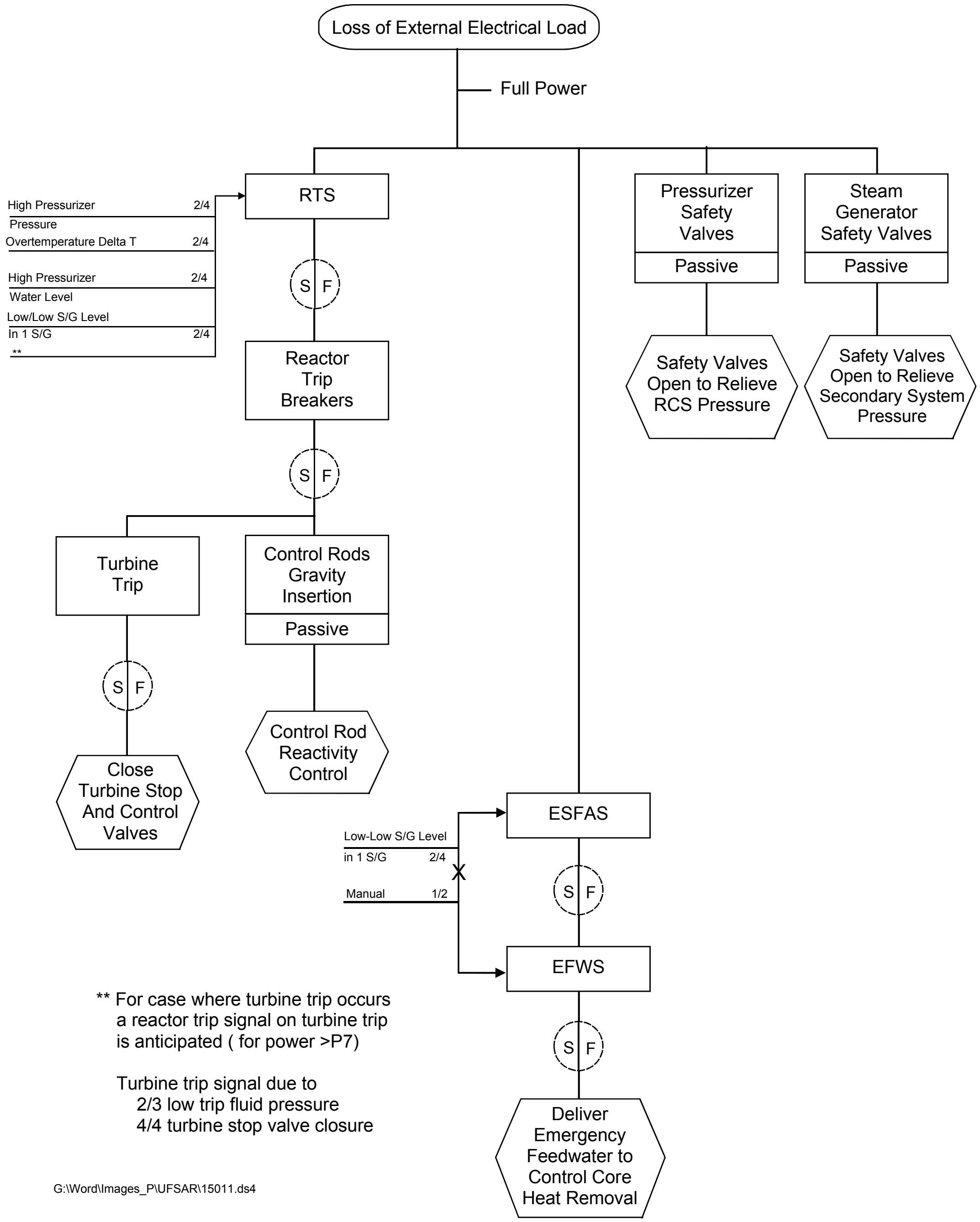
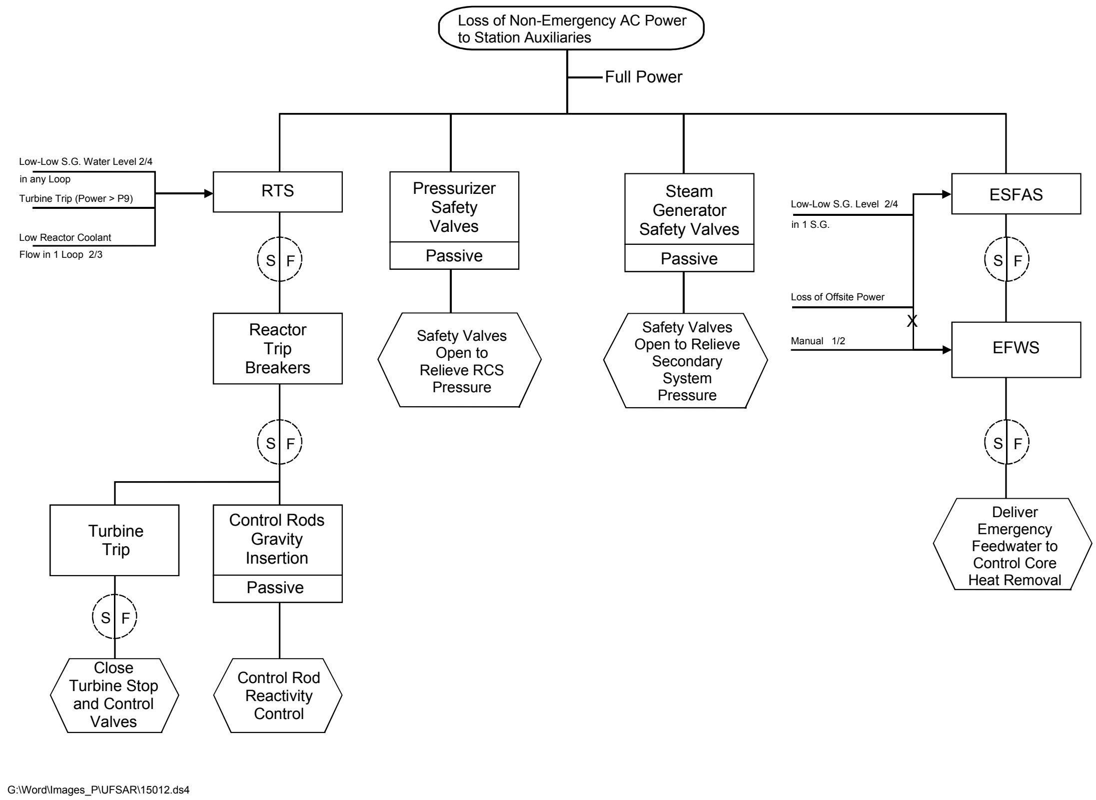


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Depressurization of Main Steam System	
		Figure 15-0-10

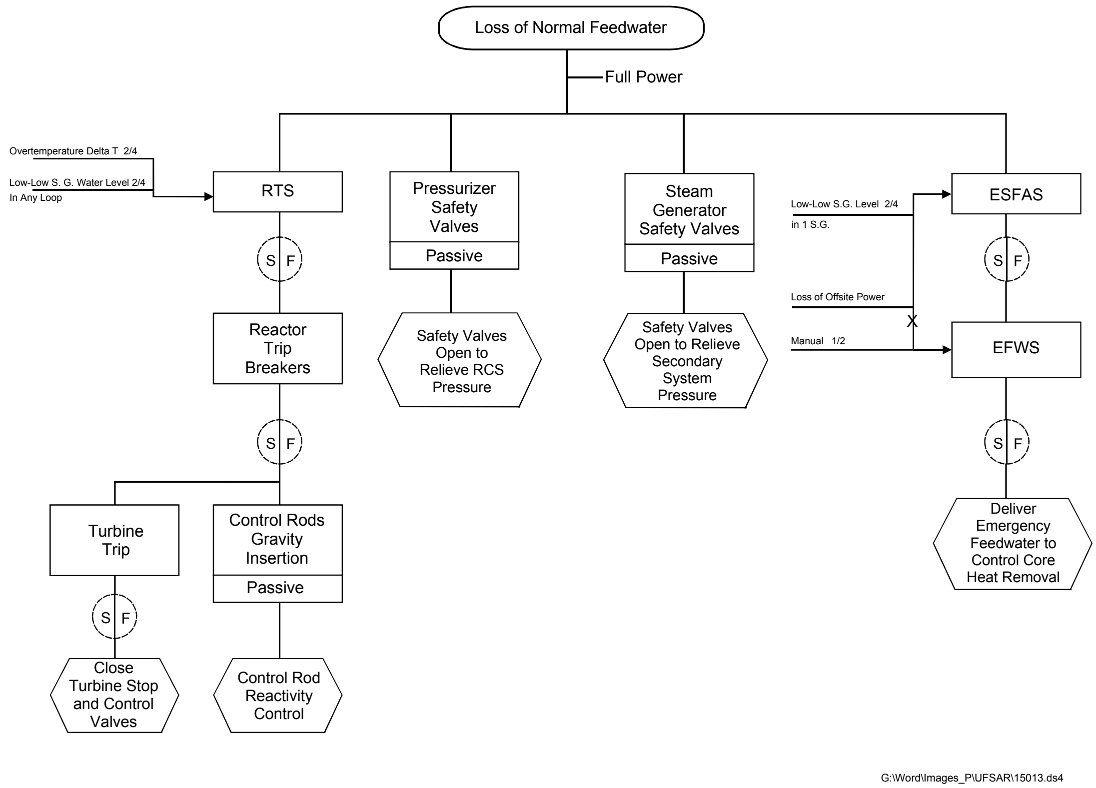


Loss of External Load

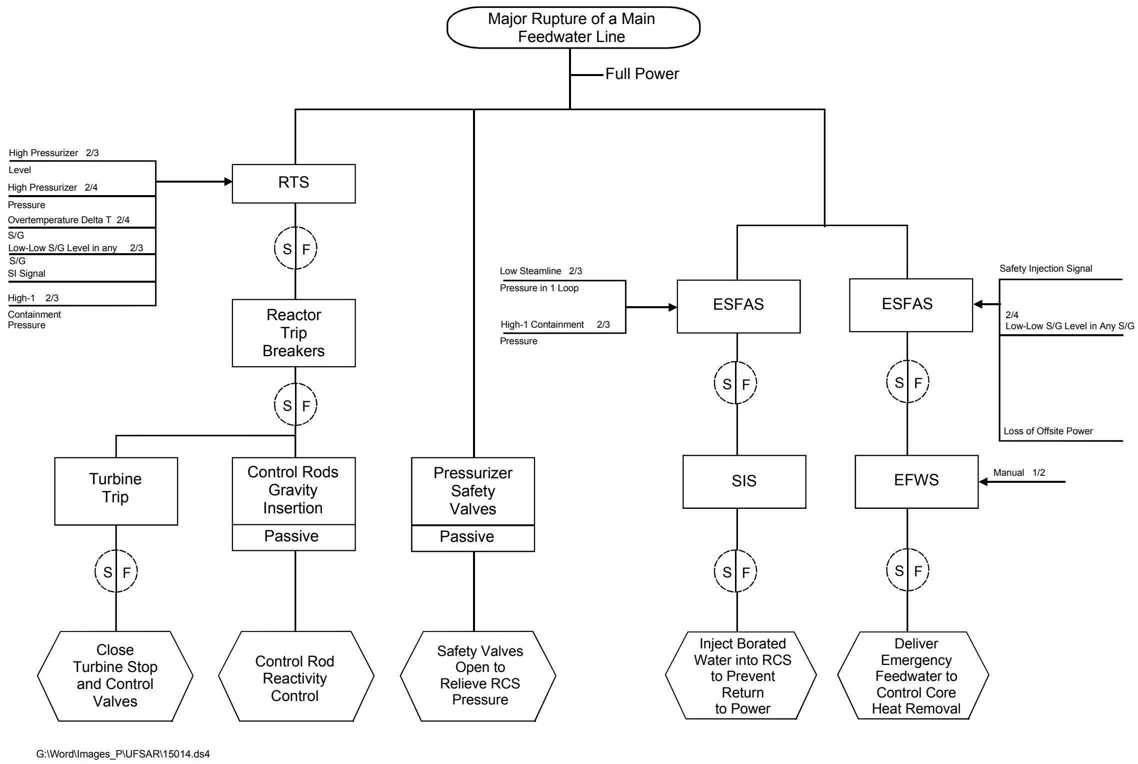
Figure 15-0-11



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Offsite Power to Station Auxiliaries	
		Figure 15-0-12

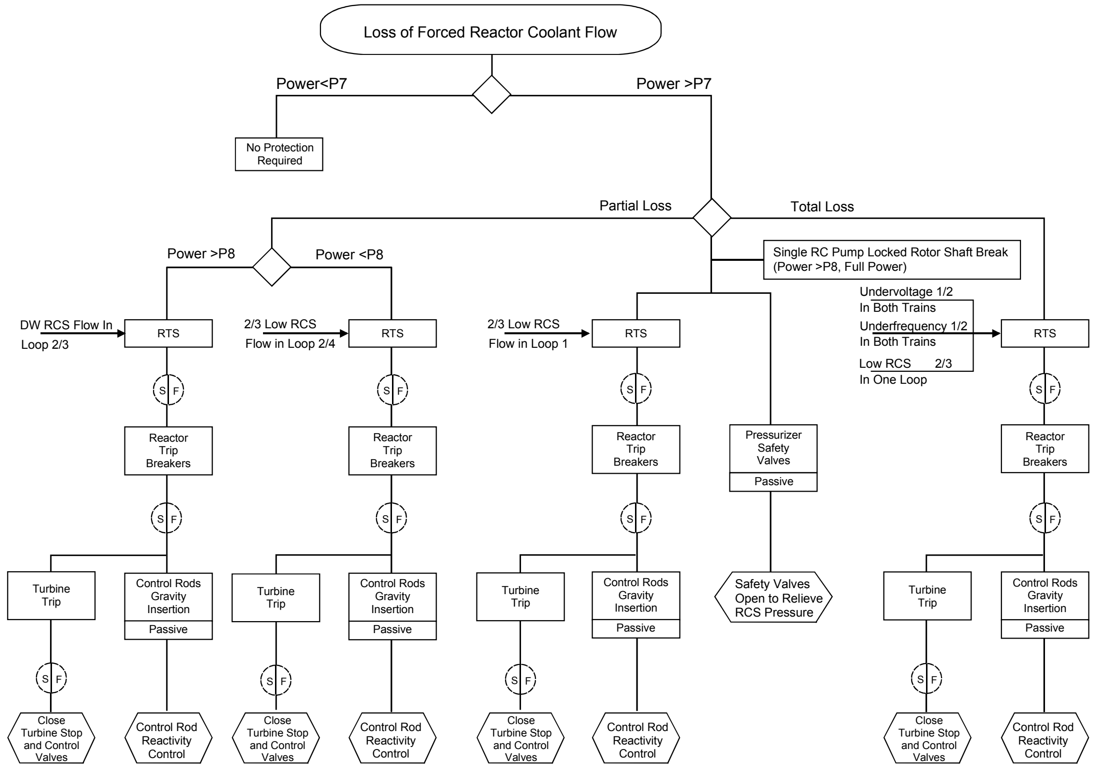


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Normal Feedwater
	Figure 15-0-13



Major Rupture of a Main Feedwater Line

Figure 15-0-14



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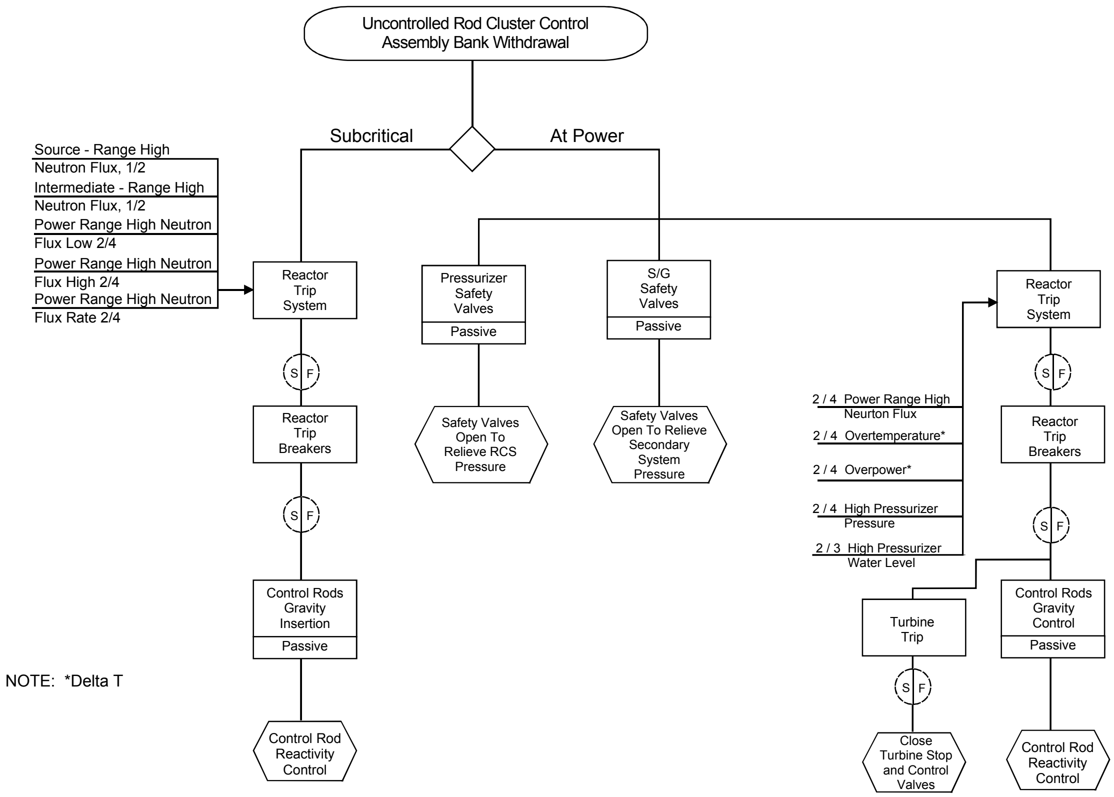
Seabrook Station Updated Final Safety Analysis Report	Loss of Forced Reactor Coolant Flow	
		Figure 15-0-15



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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal Figure 15-0-16

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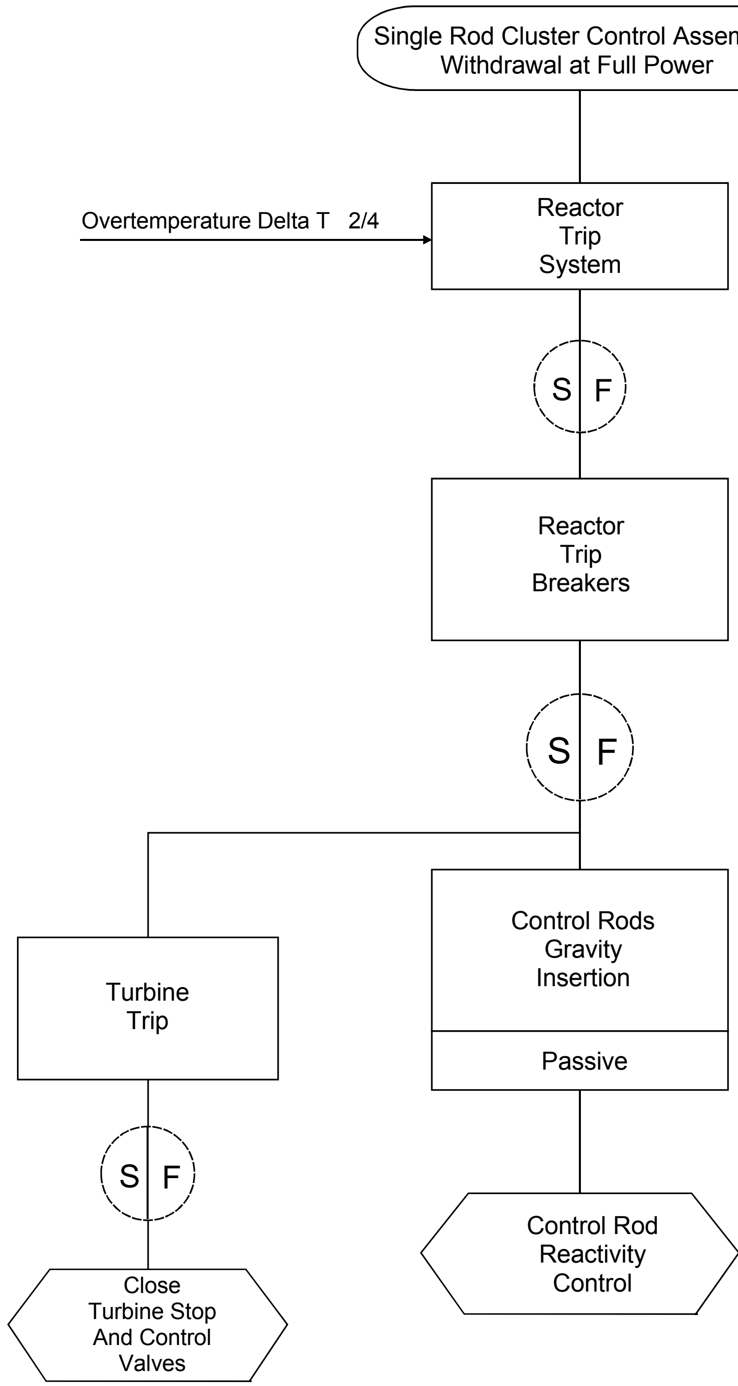
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Dropped Rod Cluster Control Assembly	
		Figure 15-0-17



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Single Rod Cluster Control Assembly Withdrawal at Full Power

Figure 15-0-18

Figure 15.0-19
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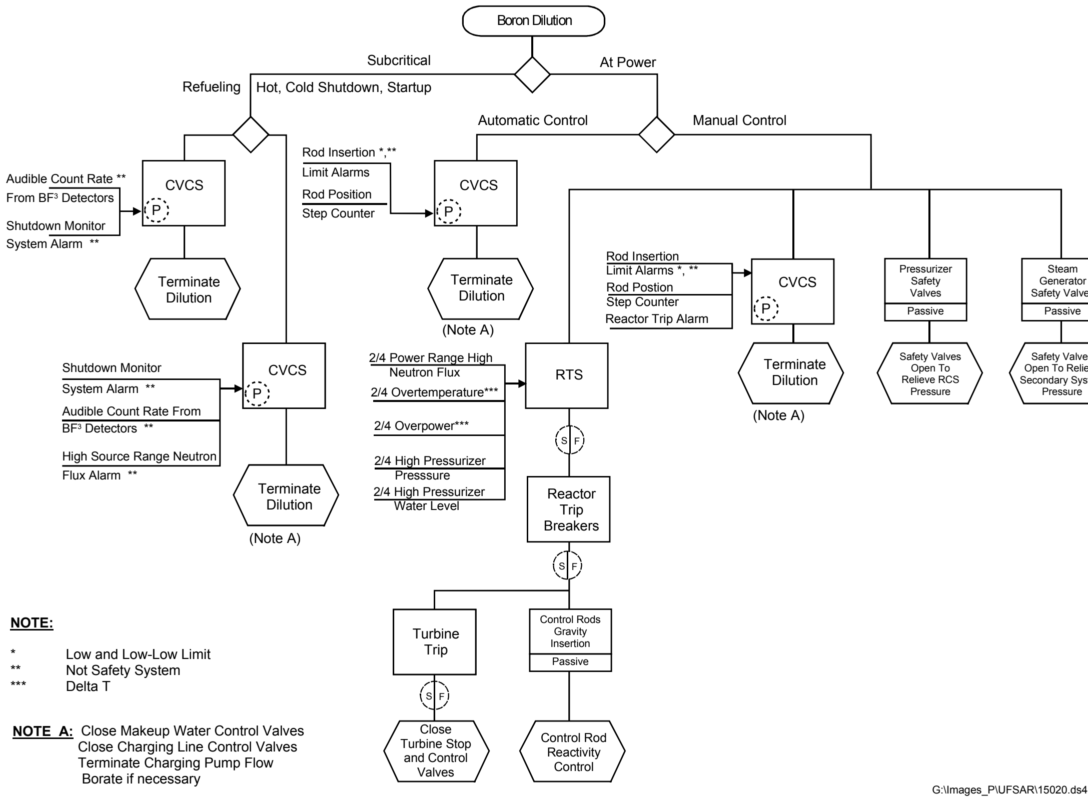
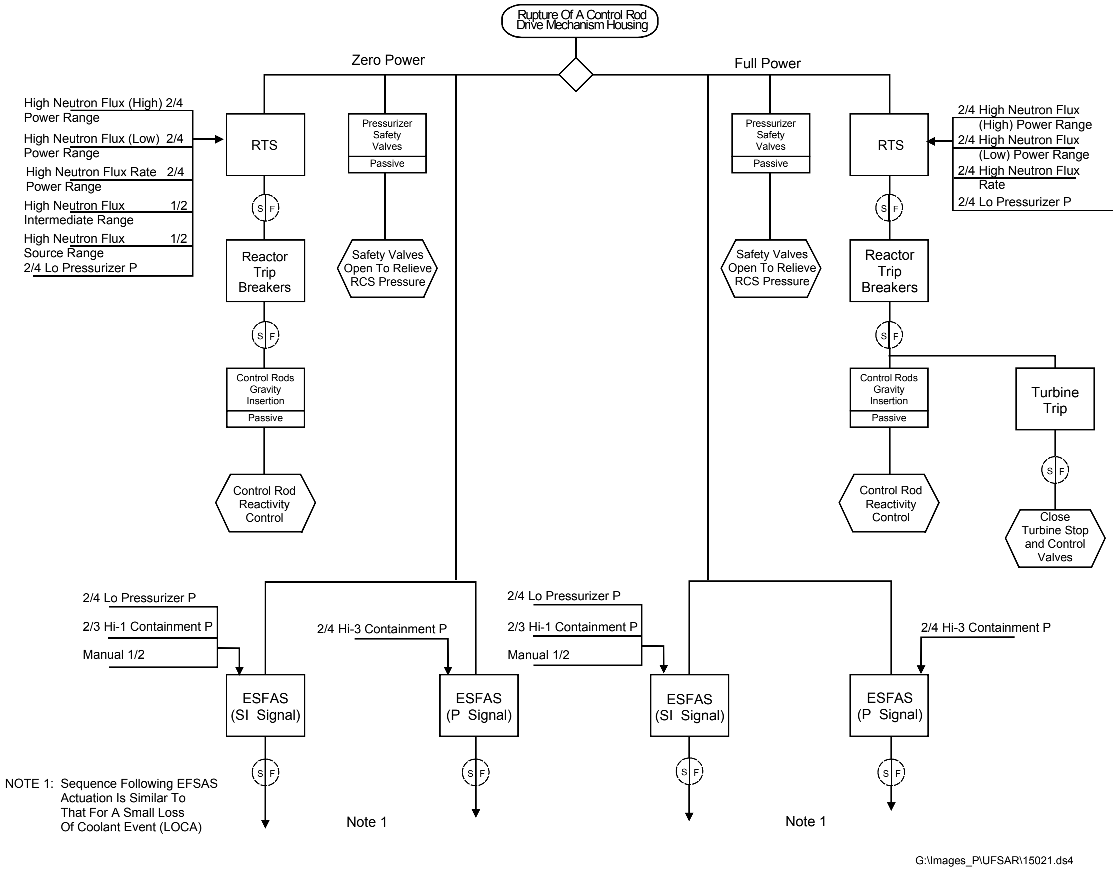
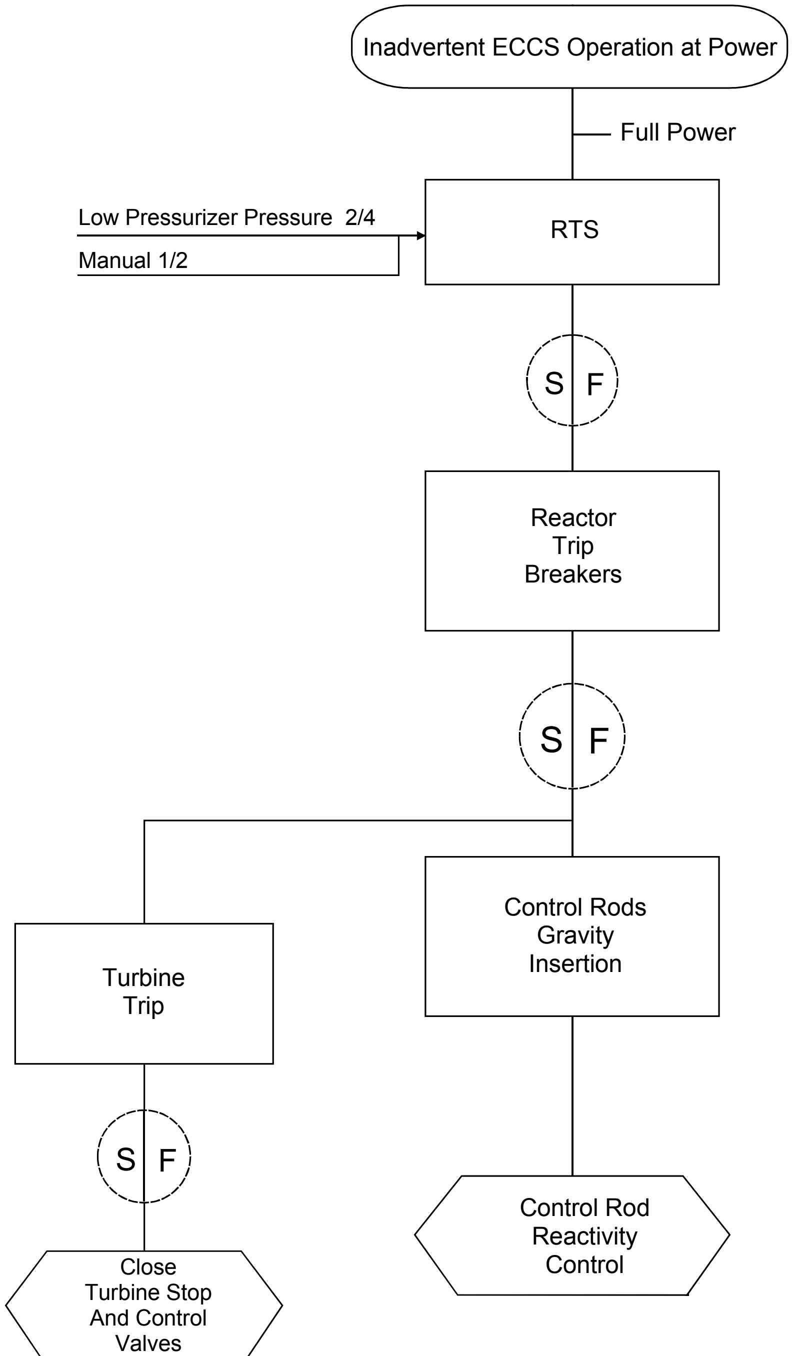


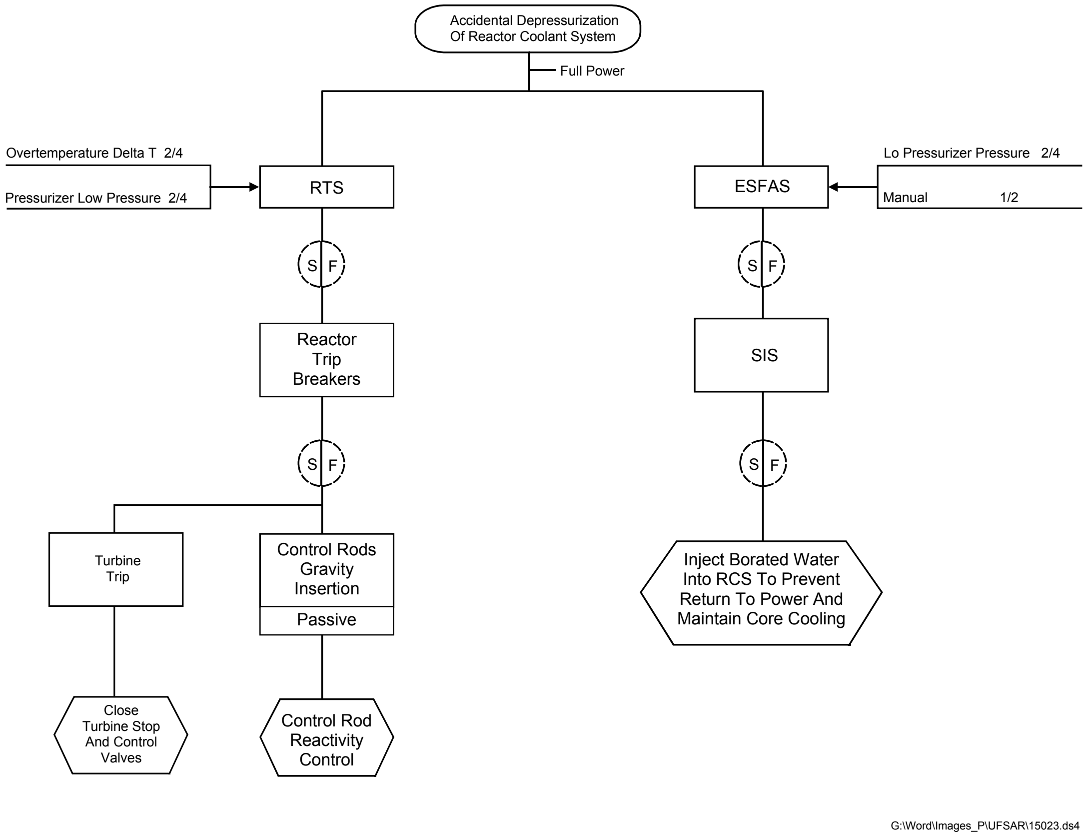
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Boron Dilution
	Figure 15-0-20



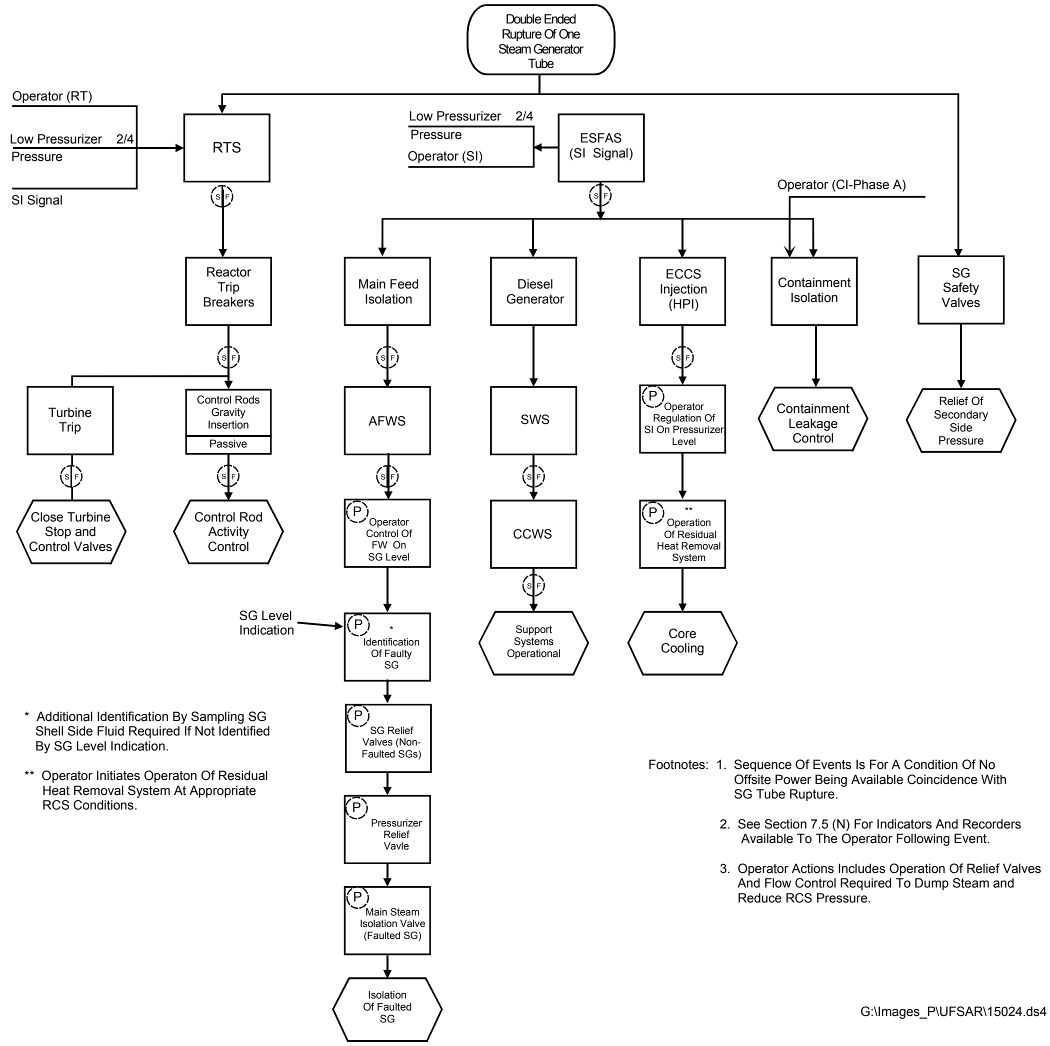
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Rupture of Control Rod Drive Mechanism Housing
	Figure 15-0-21



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Inadvertent ECCS Operation at Power
	Figure 15-0-22



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Accidental Depressurization of Reactor Coolant System	
		Figure 15-0-23



* Additional Identification By Sampling SG Shell Side Fluid Required If Not Identified By SG Level Indication.

** Operator Initiates Operaton Of Residual Heat Removal System At Appropriate RCS Conditions.

Footnotes:
1. Sequence Of Events Is For A Condition Of No Offsite Power Being Available Coincidence With SG Tube Rupture.
2. See Section 7.5 (N) For Indicators And Recorders Available To The Operator Following Event.
3. Operator Actions Includes Operation Of Relief Valves And Flow Control Required To Dump Steam and Reduce RCS Pressure.

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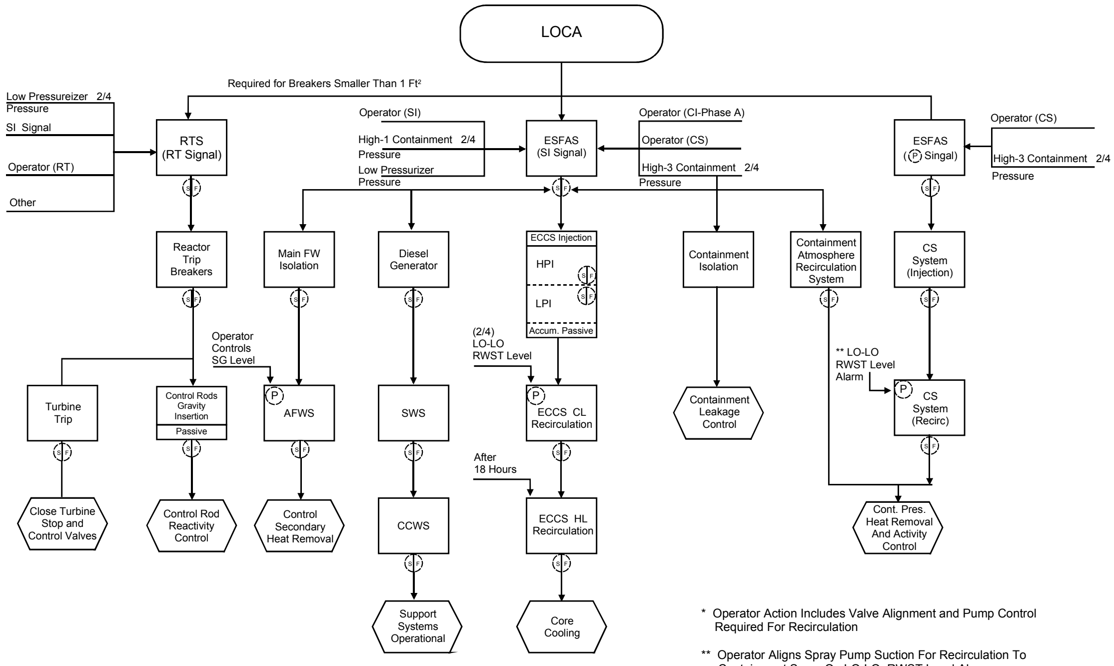
Steam Generator Tube Rupture

Figure 15-0-24



* Operator Action Includes Valve Alignment and Pump Control Required For Recirculation

** Operator Aligns Spray Pump Suction For Recirculation To Containment Sump On LO-LO RWST Level Alarm

Footnote: See Section 7.5 (N) For Indicators and Recorders Available To The Operator Following Event

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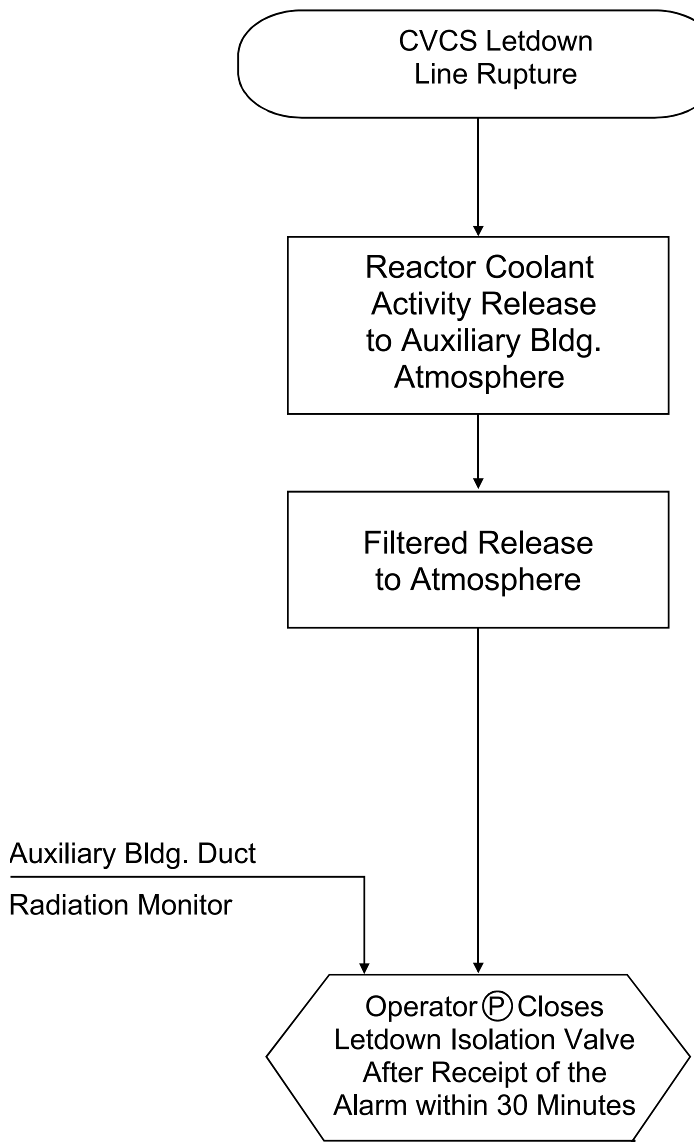
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loss of Coolant Accident
	Figure 15-0-25



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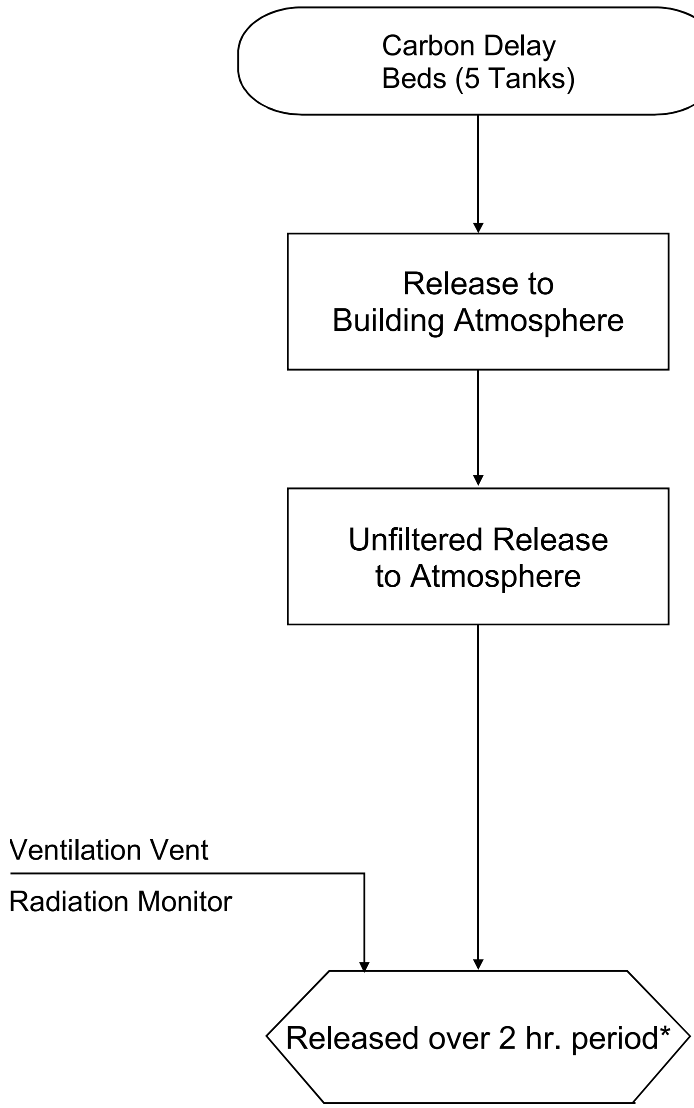
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	CVCS Letdown Rupture	
		Figure 15-0-26



* See Section 15.7.1.2

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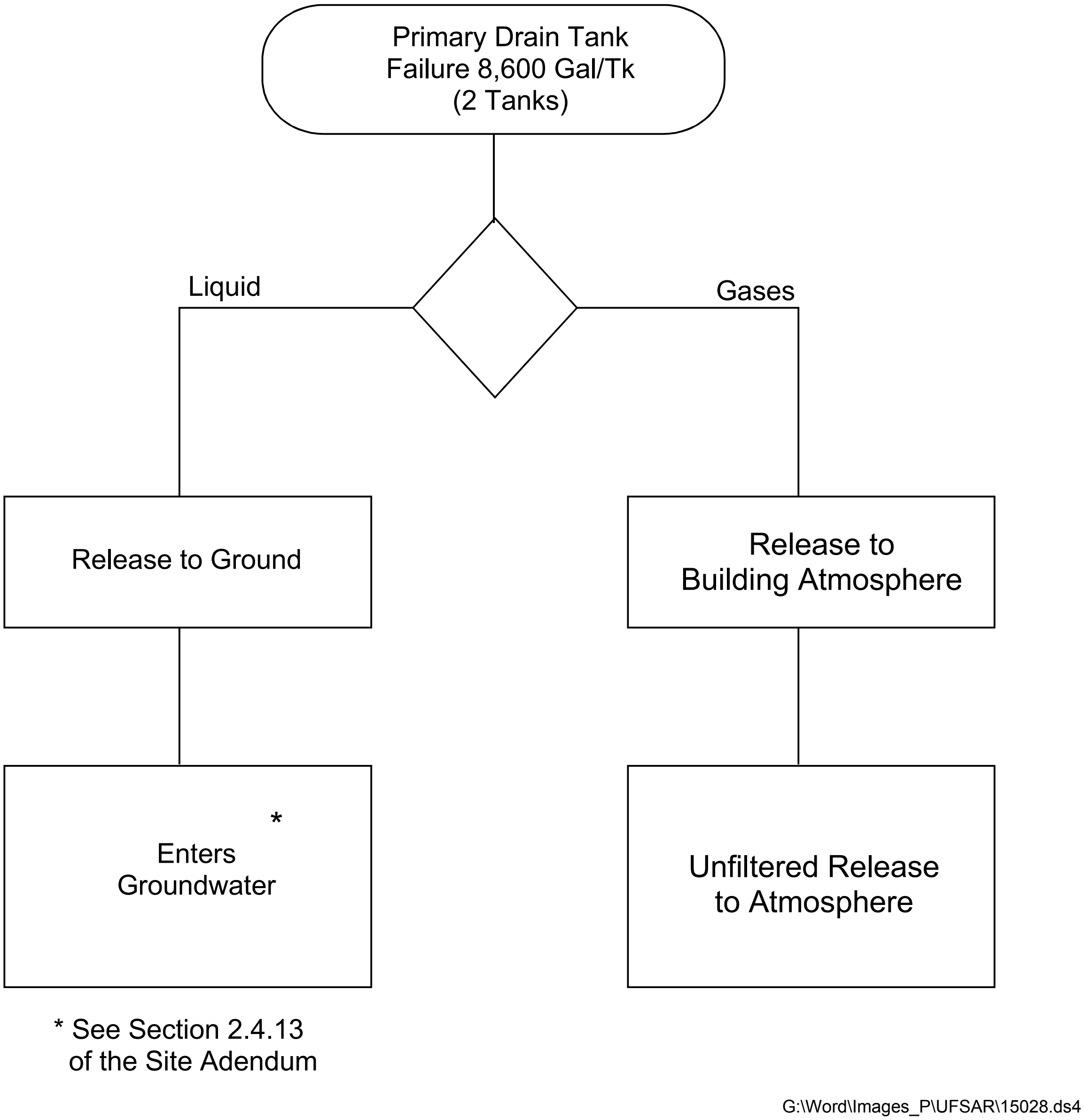
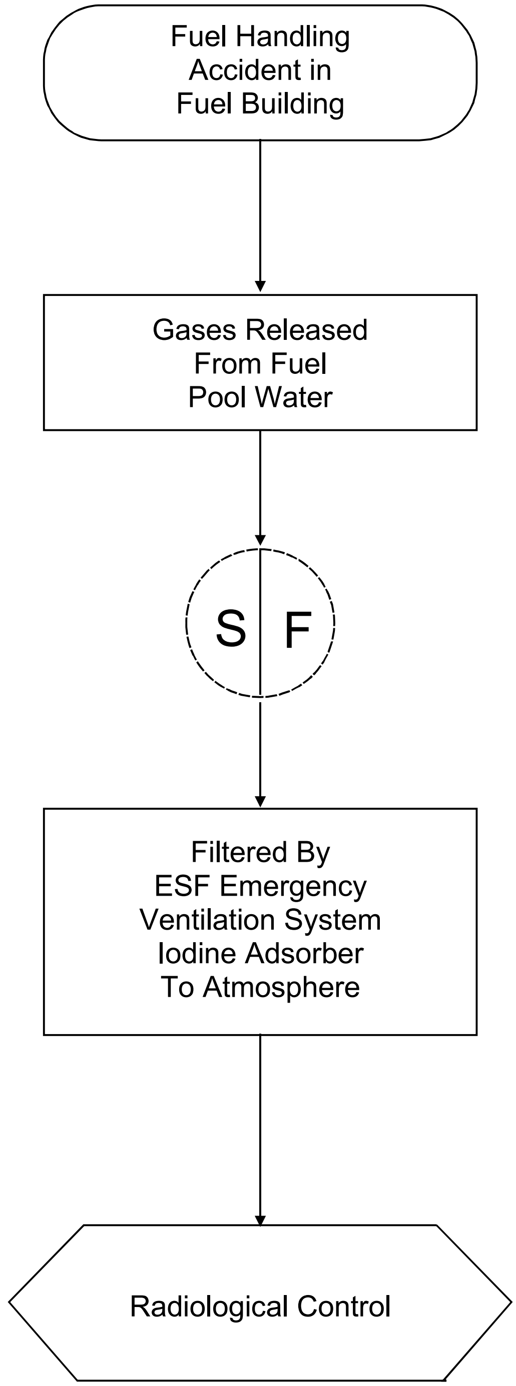


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Floor Drain Tank Failure Figure 15-0-28



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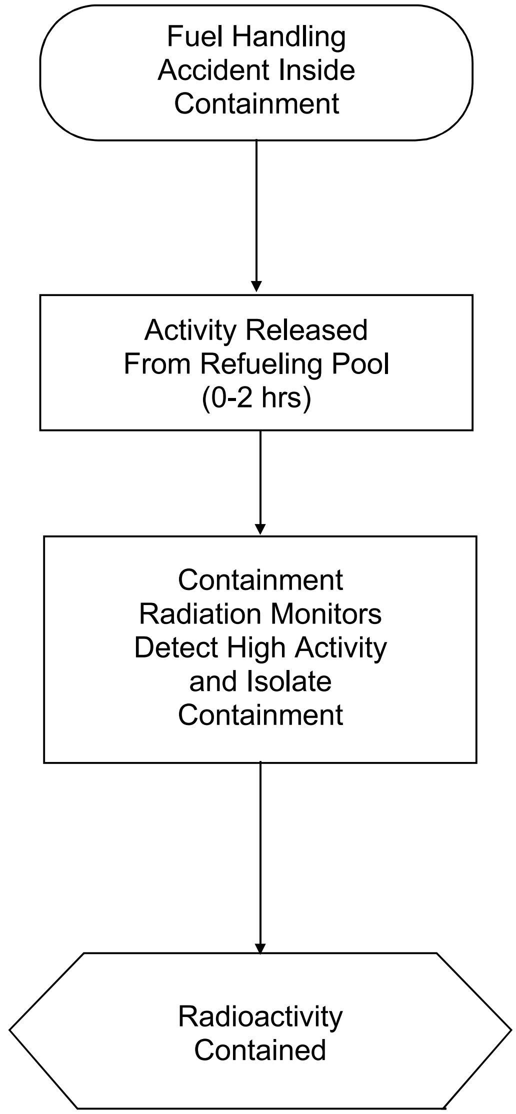

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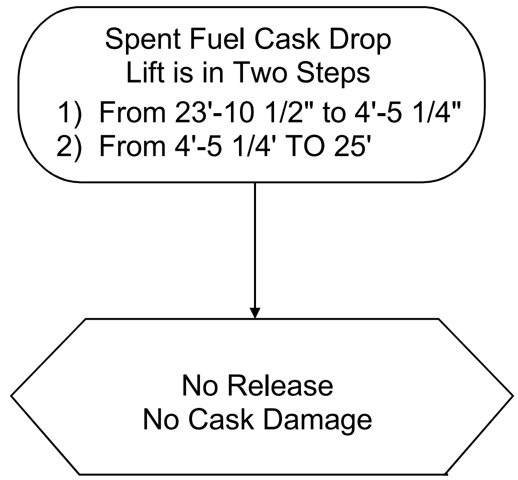
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Handling Accident Inside Containment	
		Figure 15-0-30



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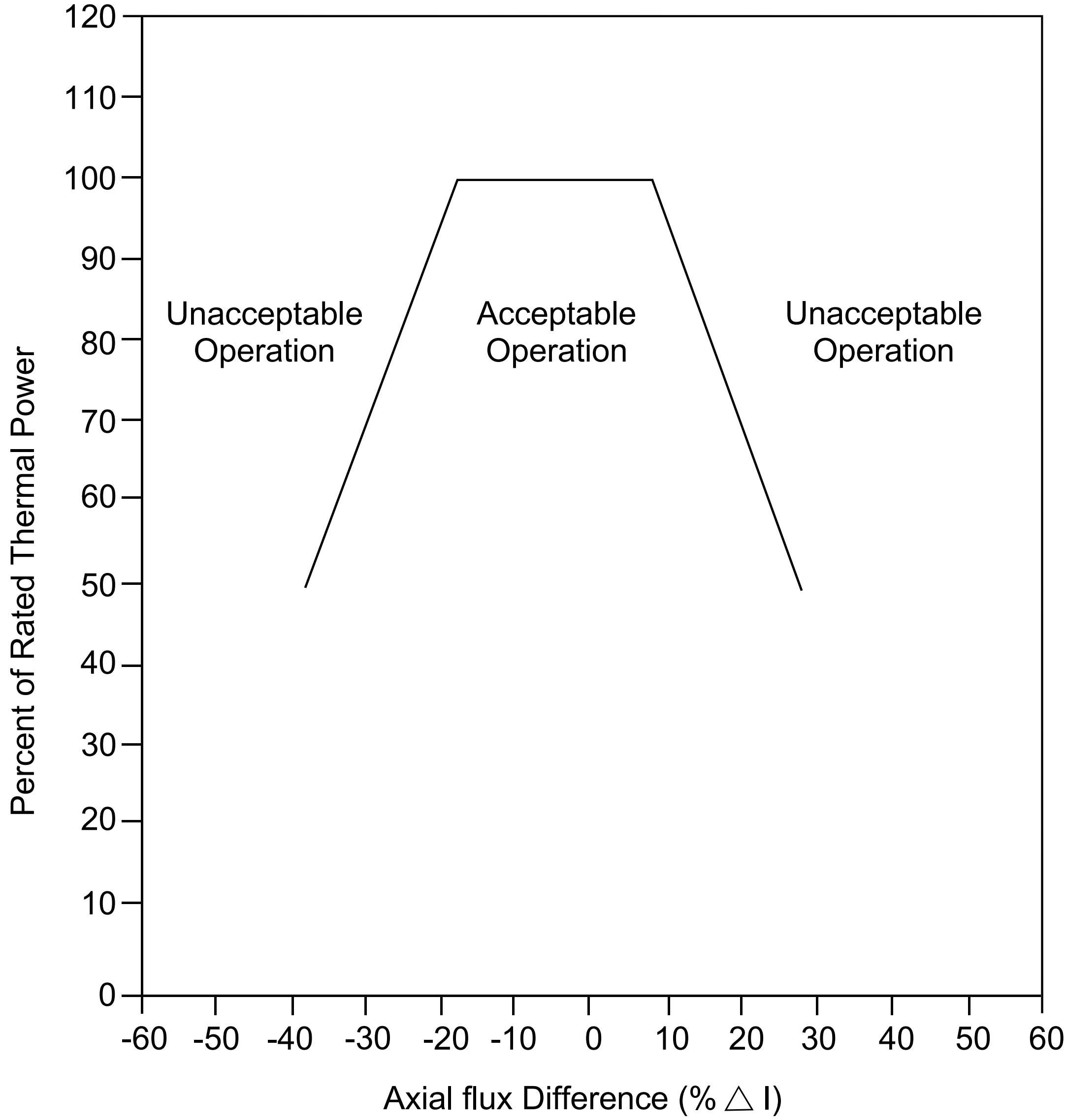
Note: The spent fuel cask drop accident represents assumptions used in the original plant design. The event contains historical information that is not relevant at this time.

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Spent Fuel Cask Drop Accident [Historical]	
		Figure 15.0-31



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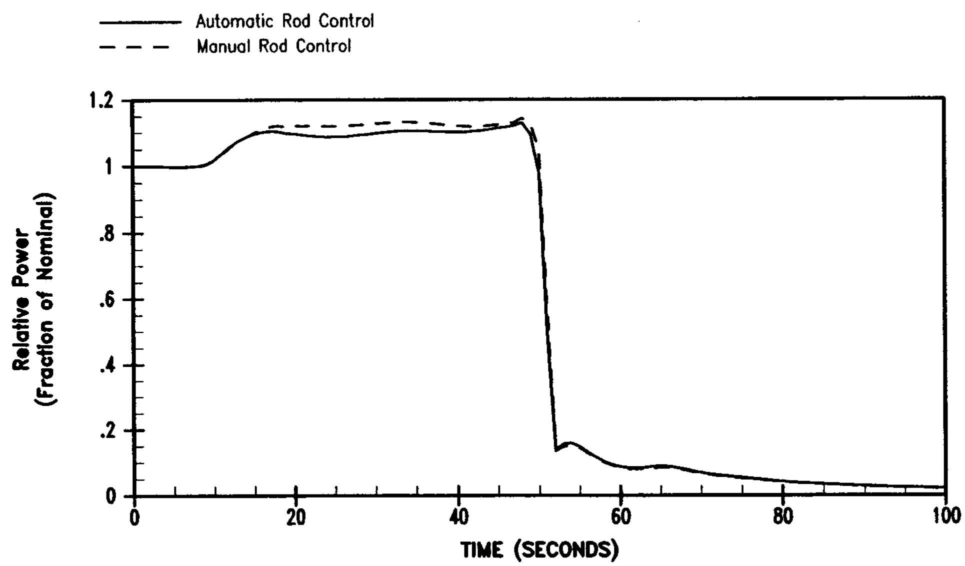
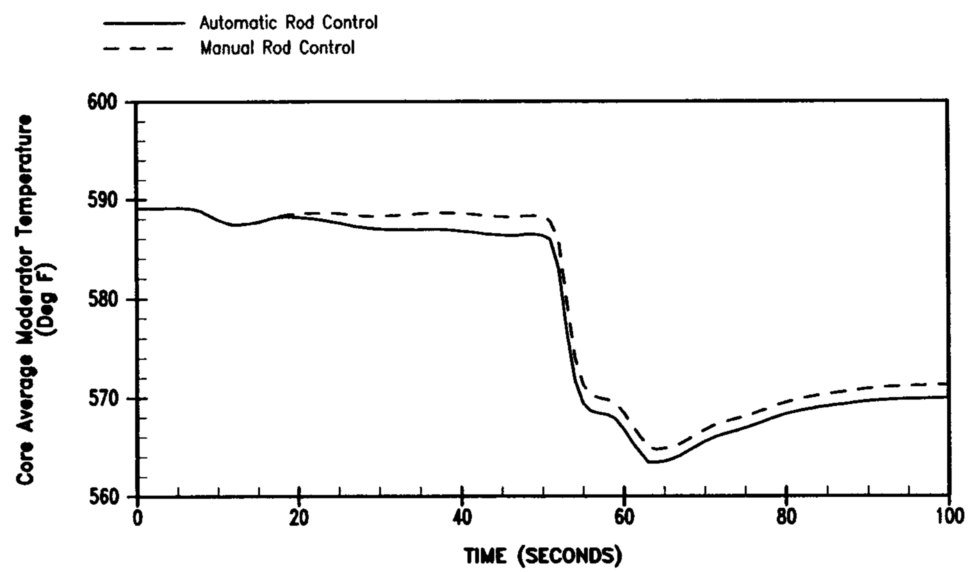
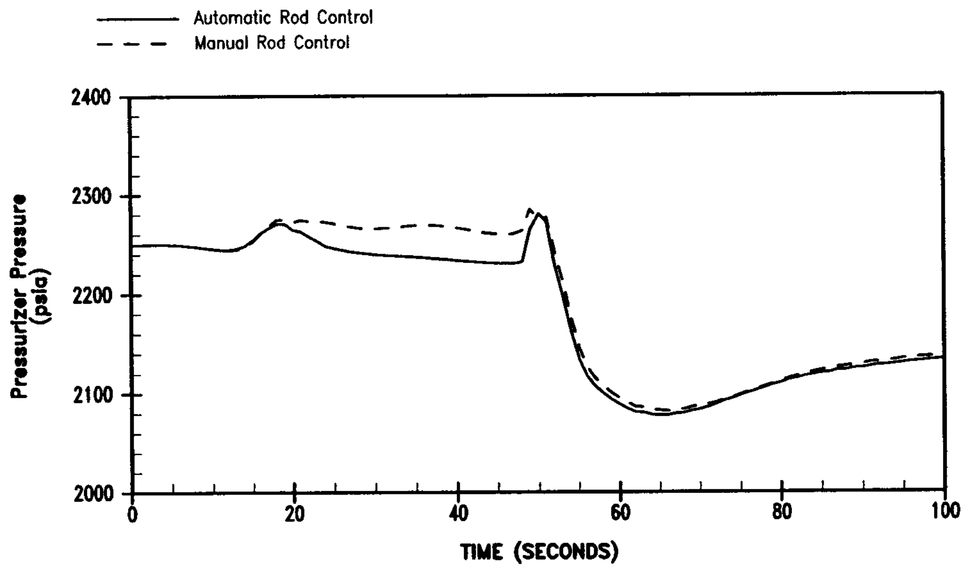
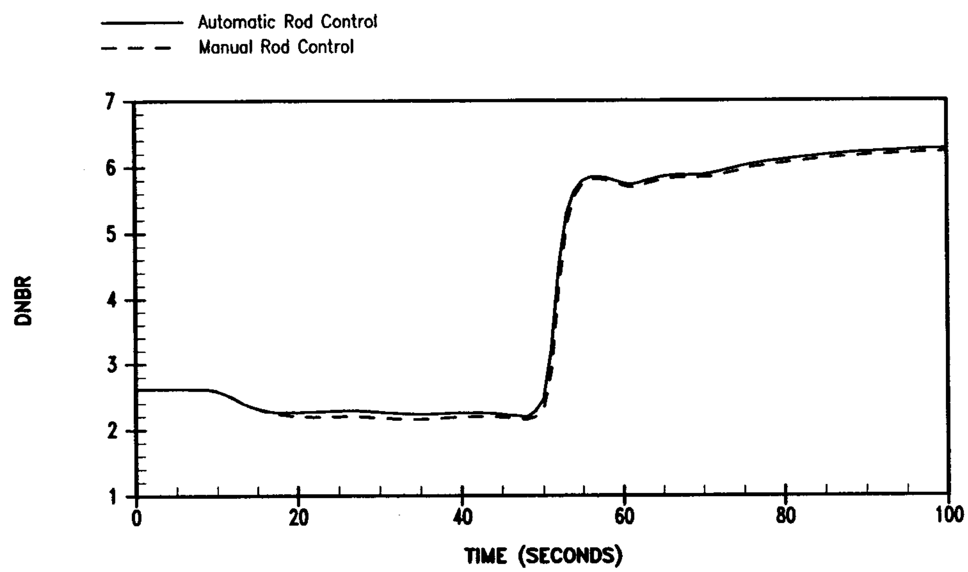
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Illustration of Allowable Axial Flux Difference Limits as a Function of Rated Thermal Power
	Figure 15-0-32



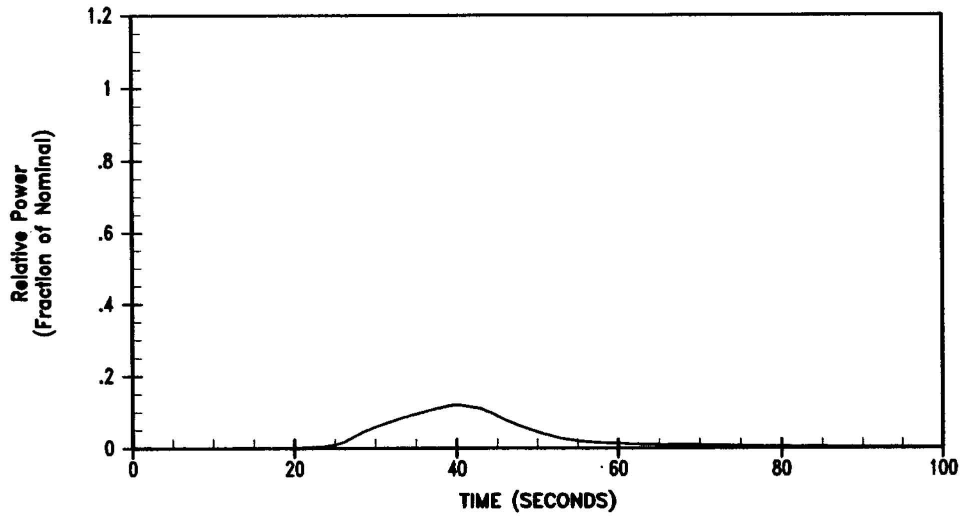


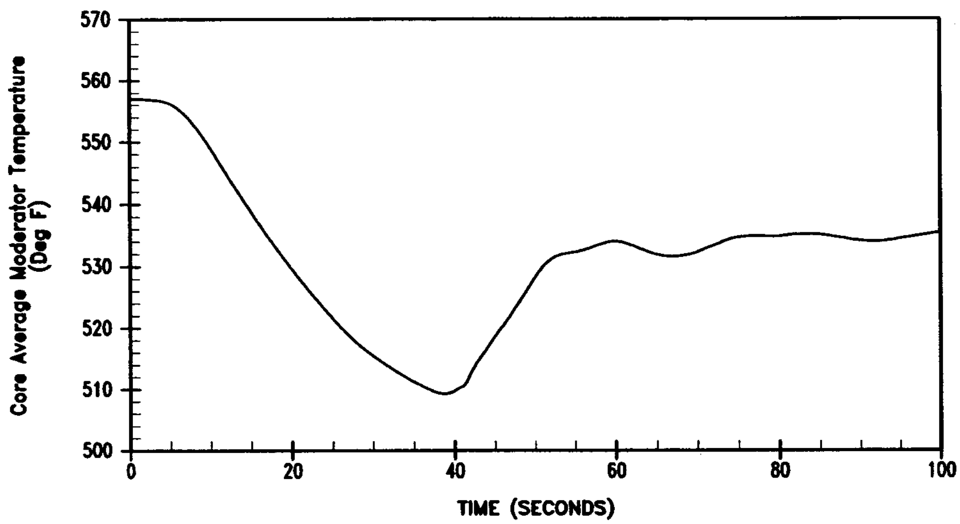
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Core Average Moderator Temperature Transients for a Feedwater Control Valve Malfunction at Full Power	
		Figure 15.1-1 Sh. 1 of 4



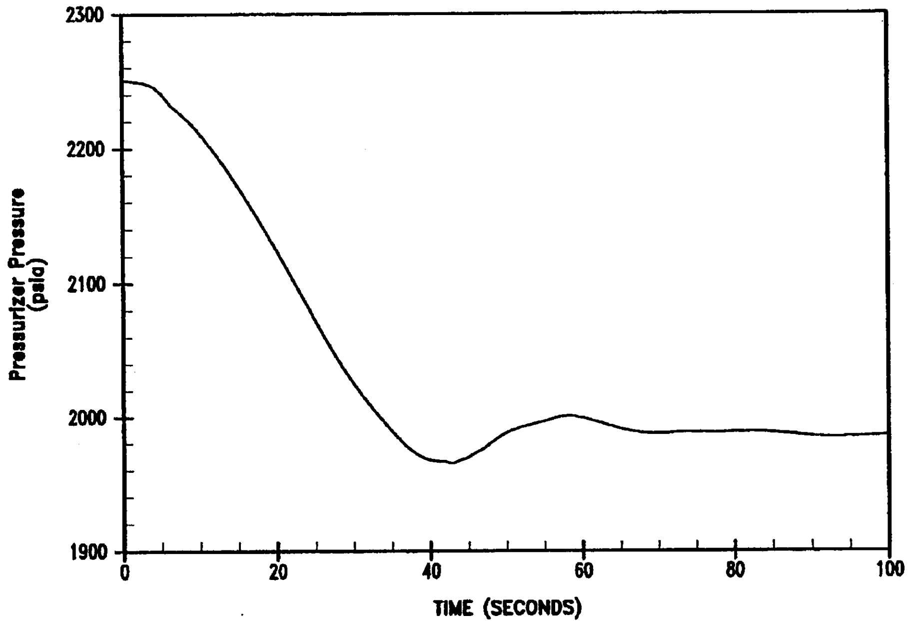


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and DNBR Transients for a Feedwater Control Valve Malfunction at Zero Power	
		Figure 15.1-1 Sh. 2 of 4





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Core Average Moderator Temperature Transients for a Feedwater Control Valve Malfunction at Zero Power	
		Figure 15.1-1 Sh. 3 of 4



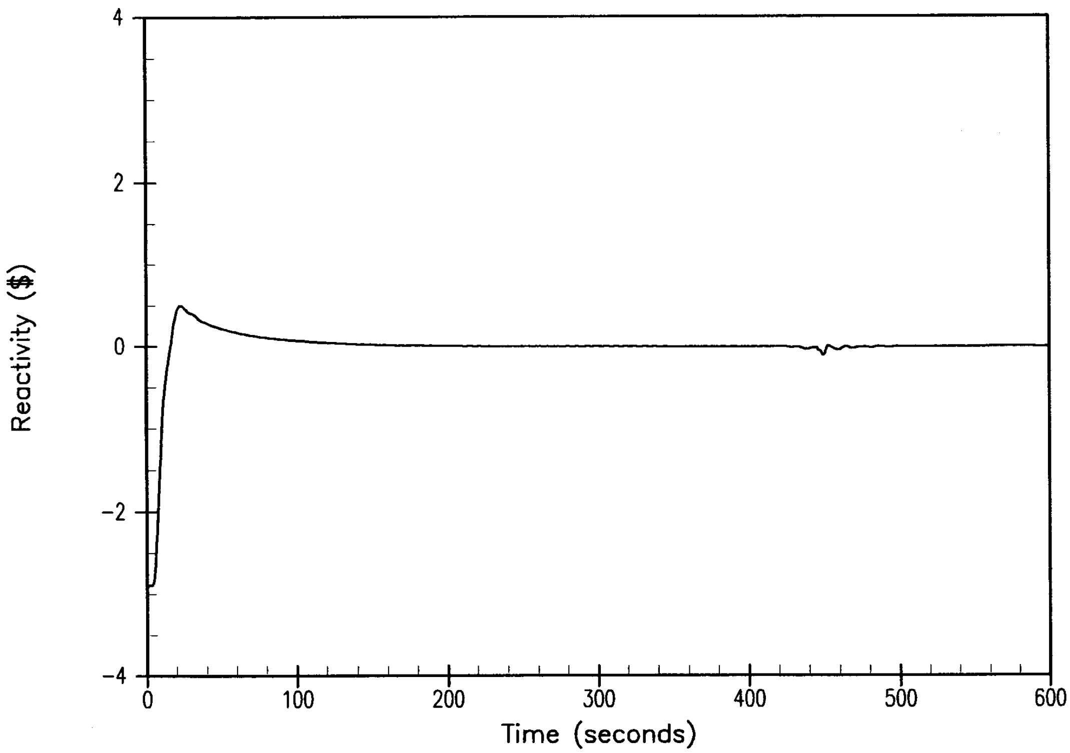
2300
2200
2100
2000
1900
Pressurizer Pressure (psia)
0
20
40
60
80
100
TIME (SECONDS)

DELETED

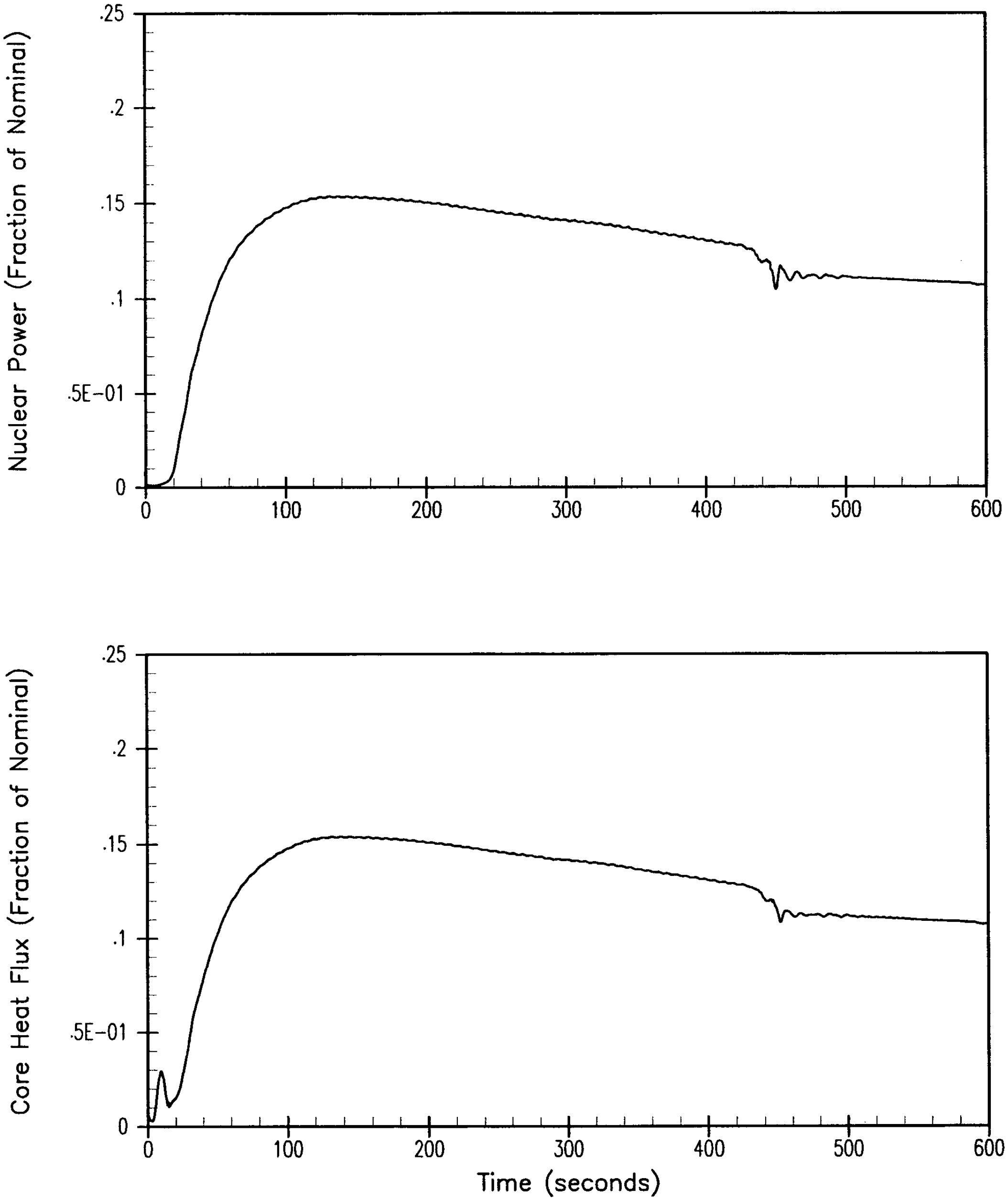
DELETED

DELETED

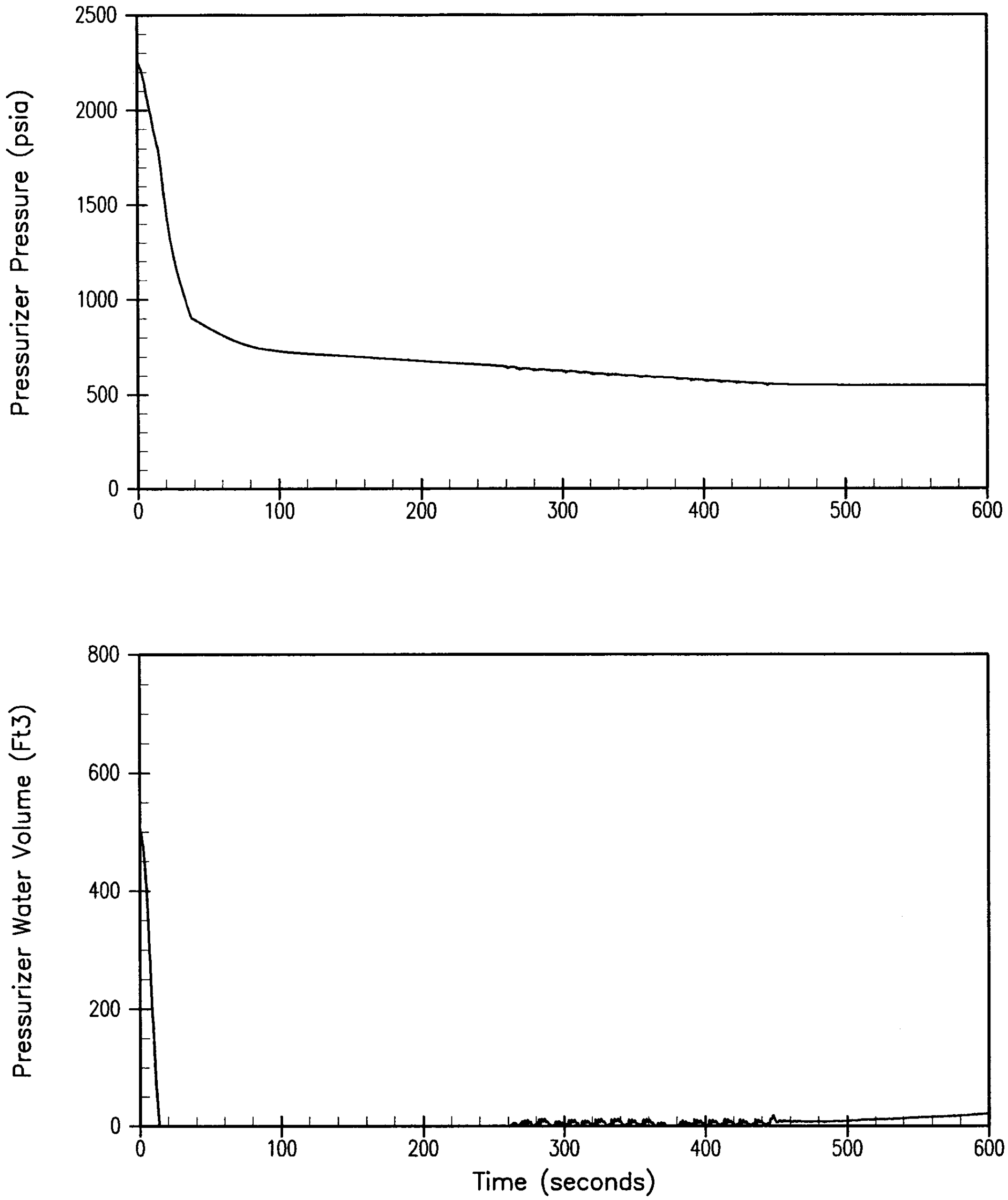
DELETED



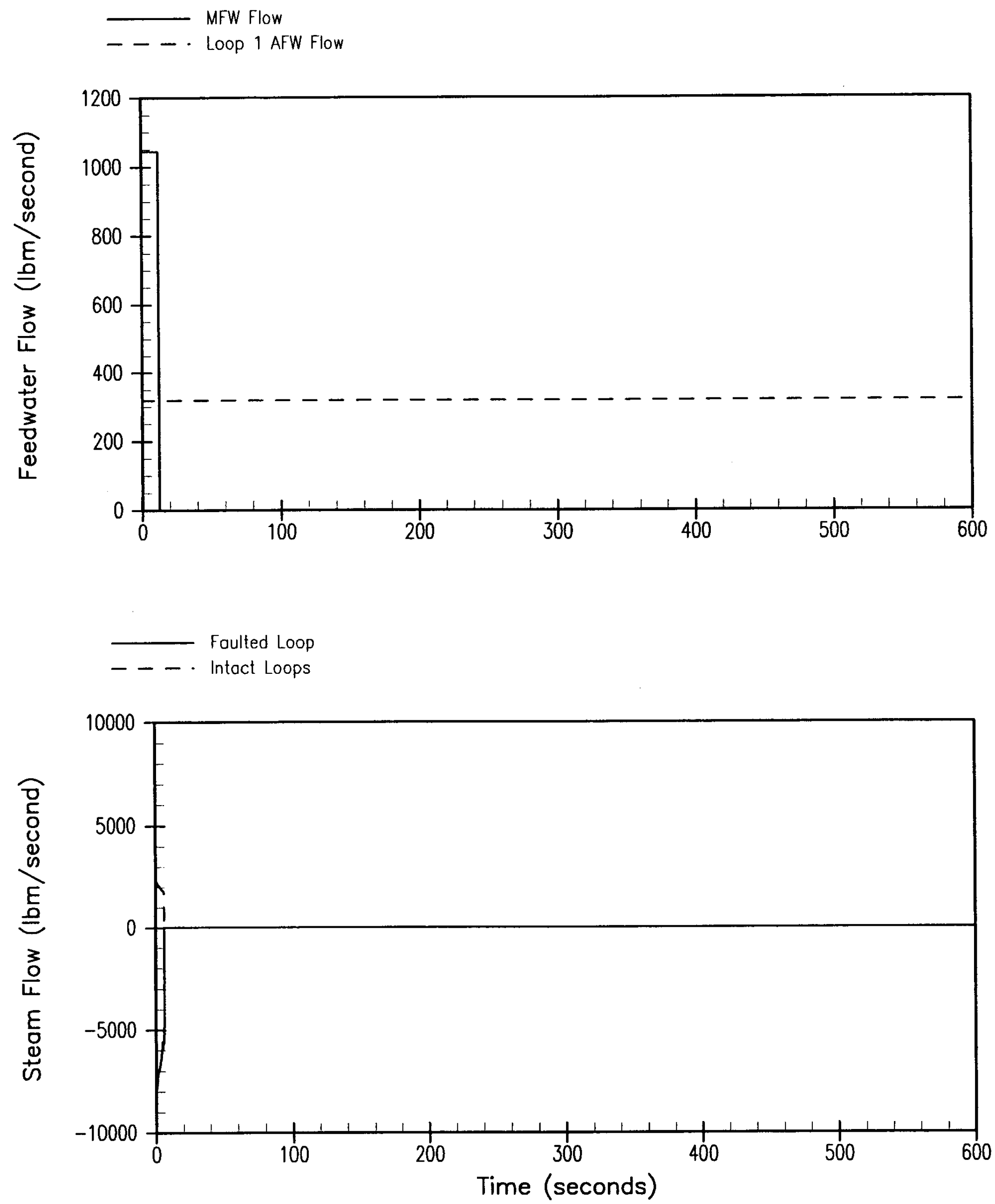
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactivity Transient for a Steam Rupture at Zero Power with Offsite Power Available	
		Figure 15.1-6 Sh. 1 of 6



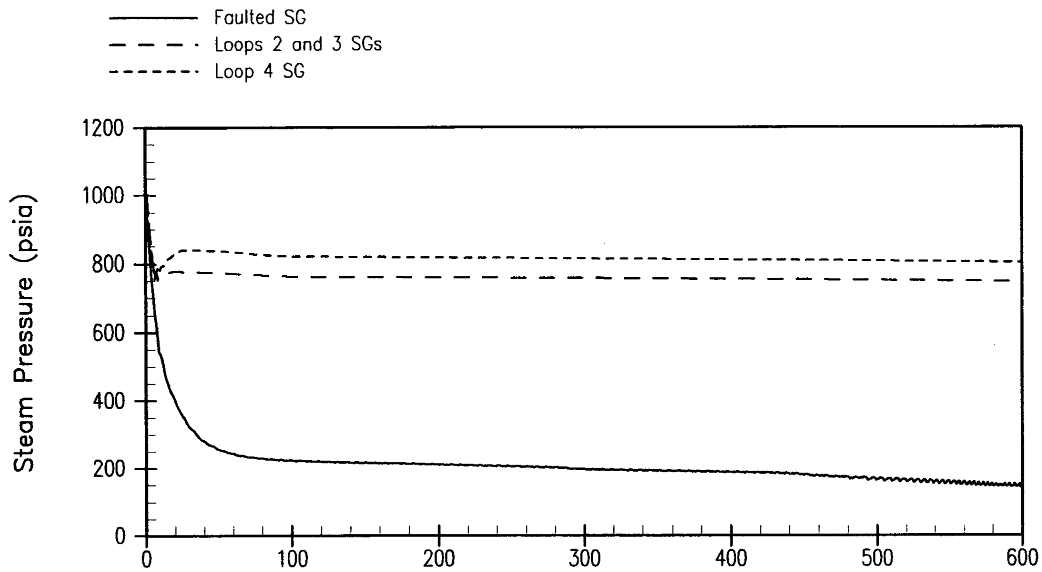
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
		Figure 15.1-6 Sh. 2 of 6

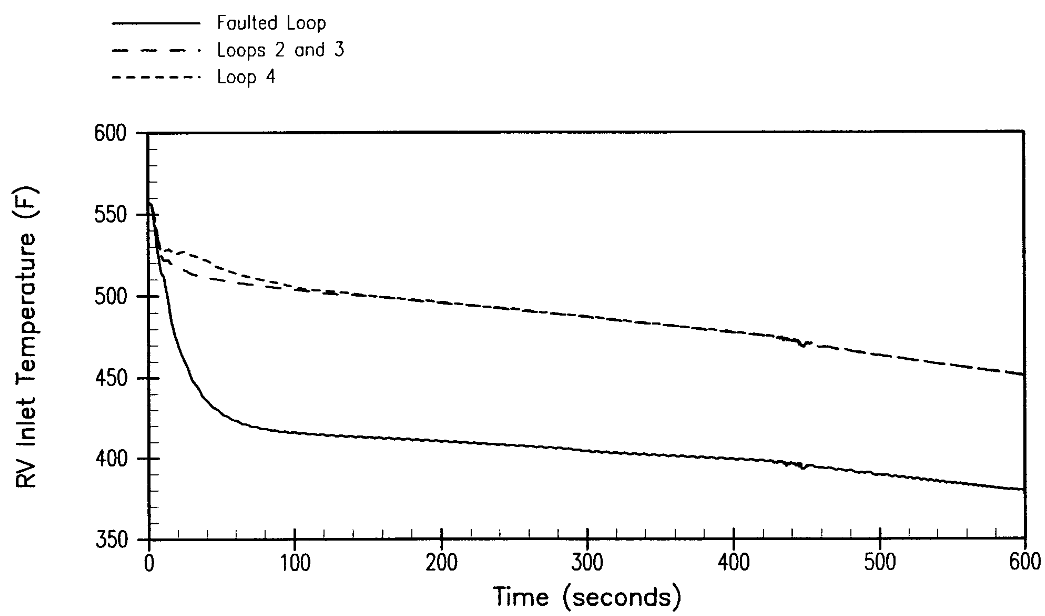


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure and Pressurizer Water Volume Transients for a Steam Line Rupture at Zero Power with Offsite Power Available	
		Figure 15.1-6 Sh. 3 of 6

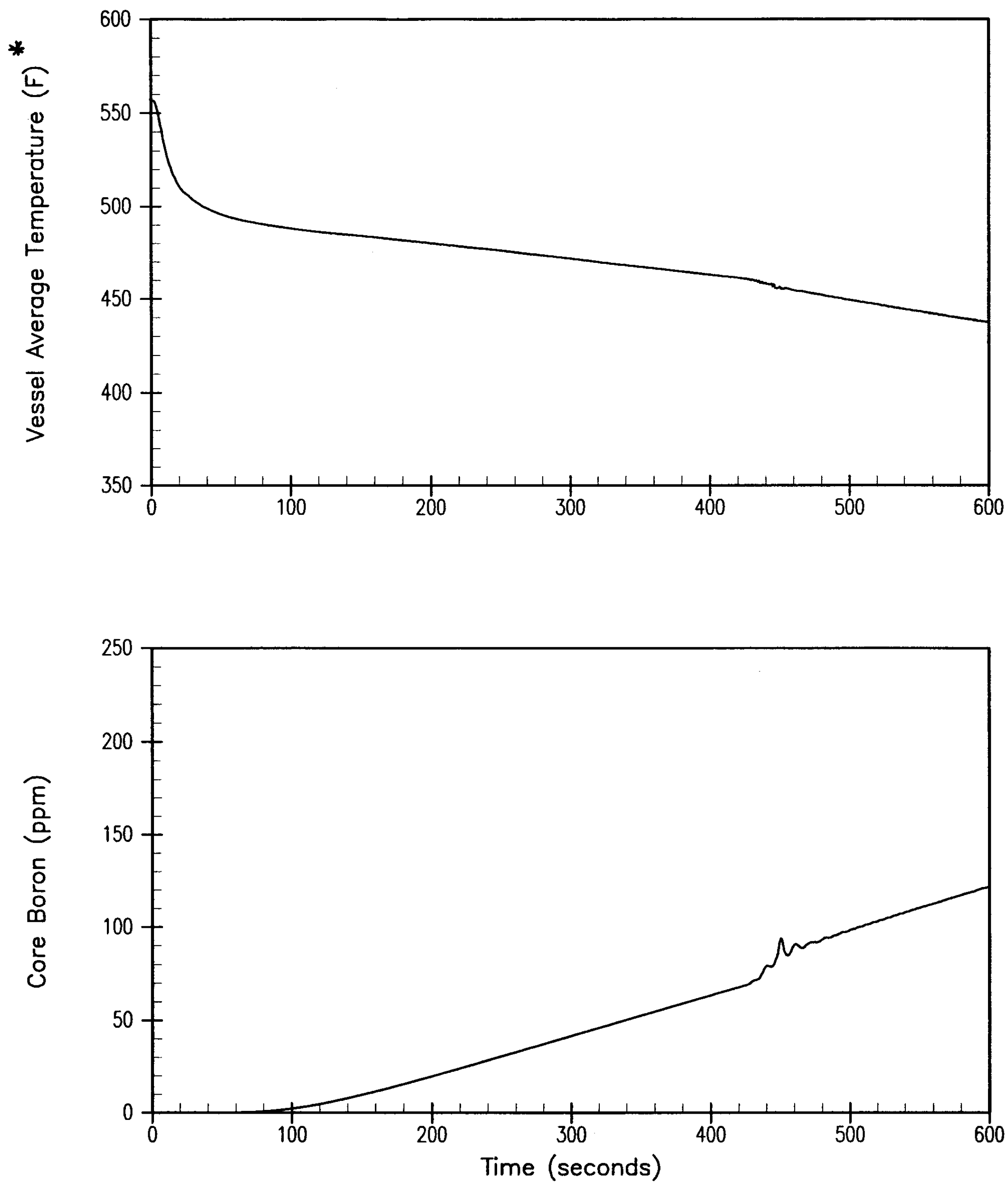


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater Flow and Steam Flow Transients for a Steam Line Rupture at Zero Power with Offsite Power Available
	Figure 15.1-6 Sh. 4 of 6





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



*Average of Loop Average Temperatures

Figure 15.1-7
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Figure 15.1-8
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Figure 15.1-9
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Figure 15.1-10
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Figure 15.1-11
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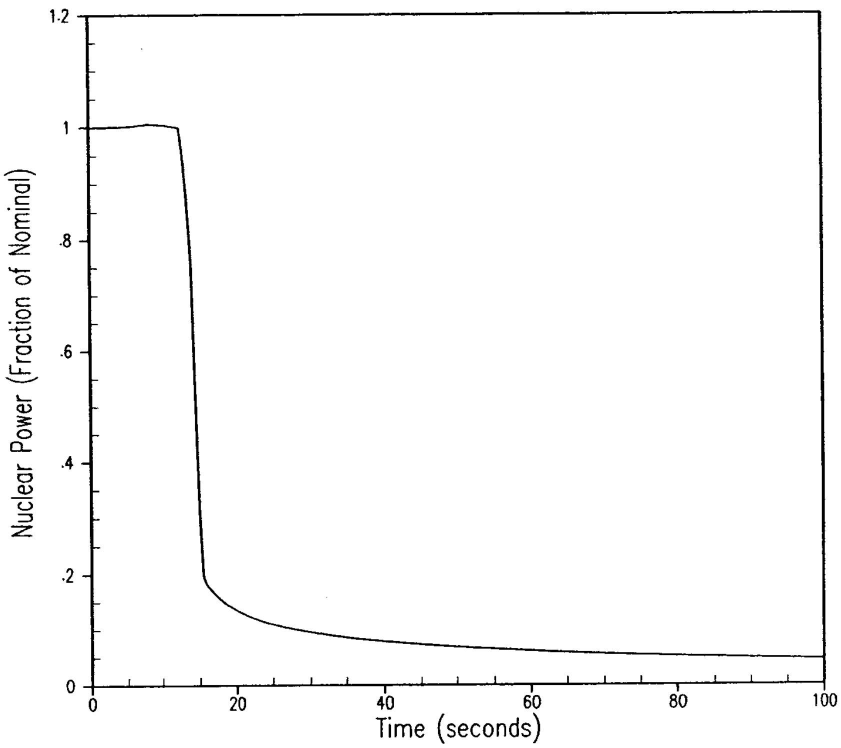
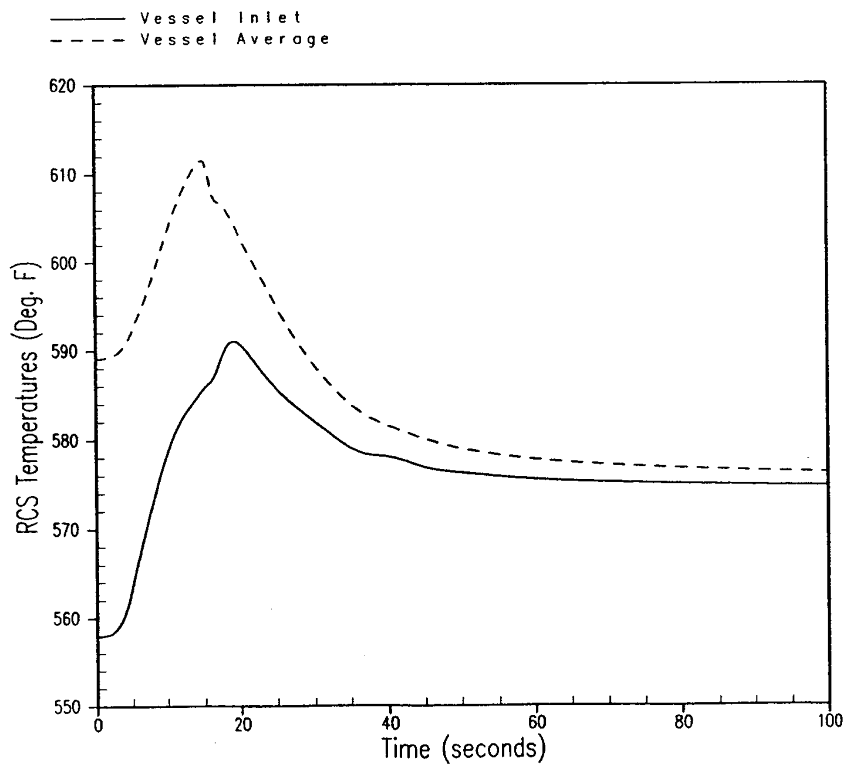
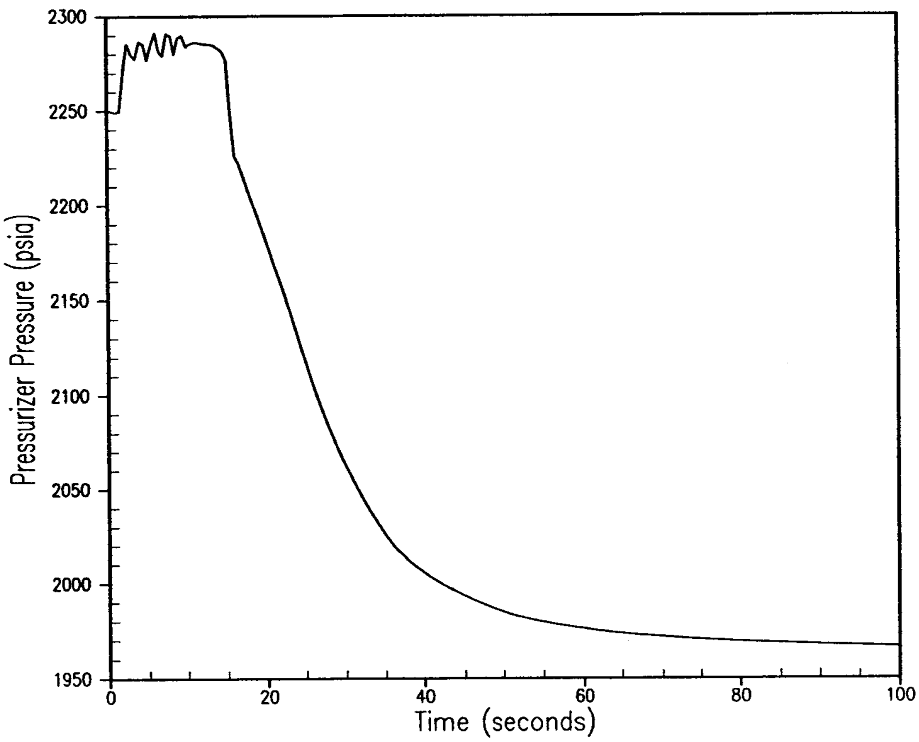
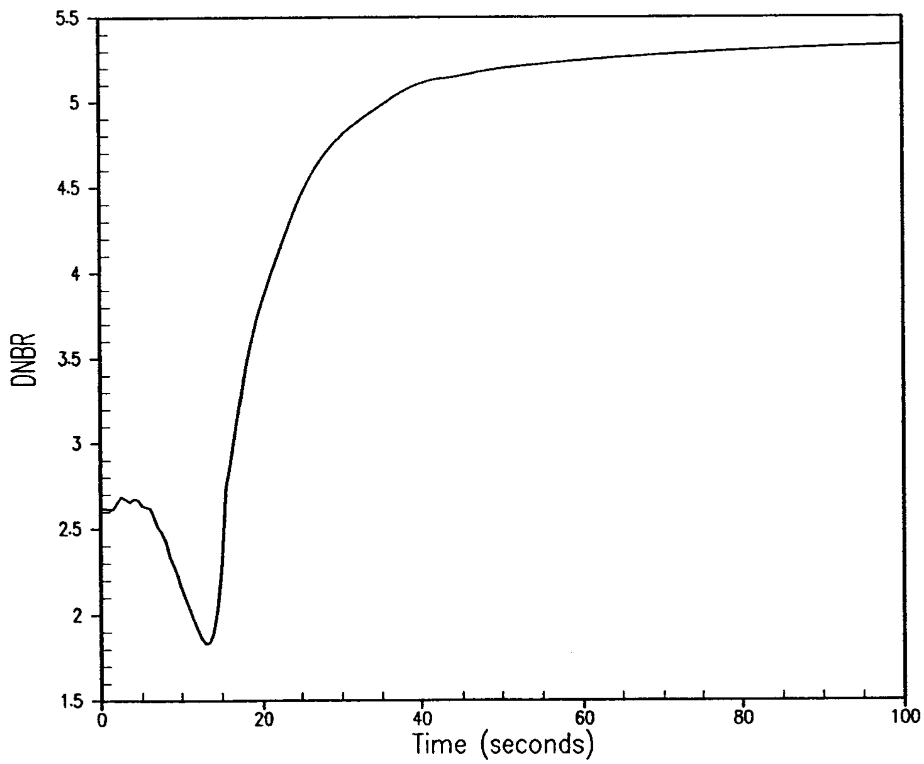


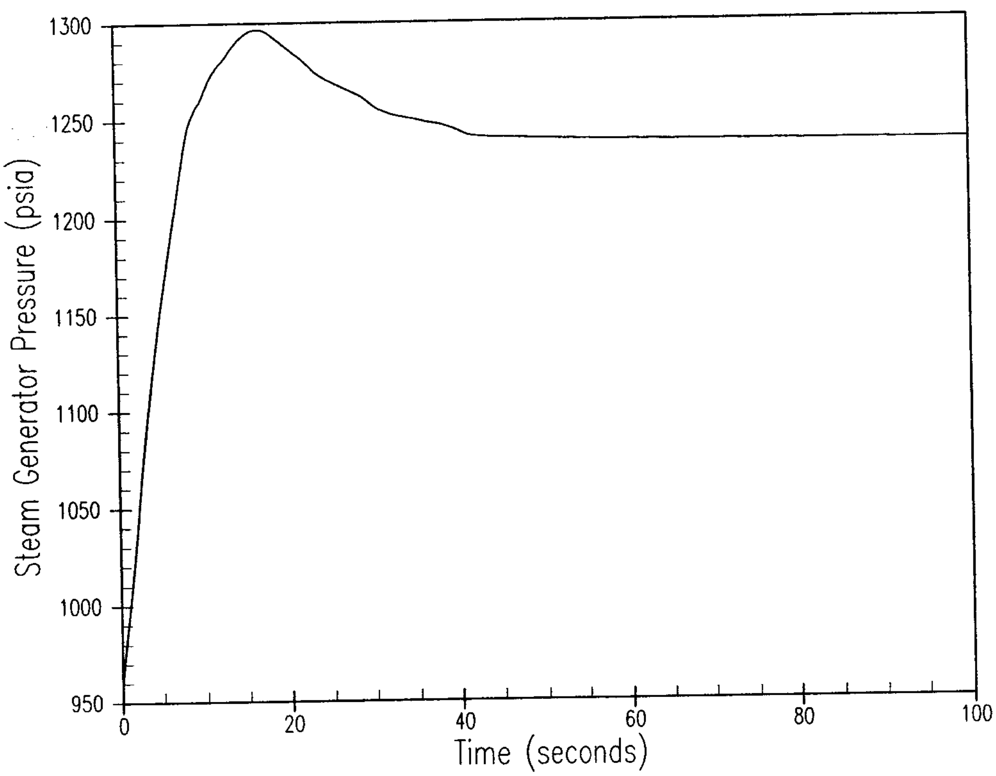


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Temperature Transients for a Turbine Trip Event With Pressure Control	
		Figure 15.2-1 Sh. 1 of 5

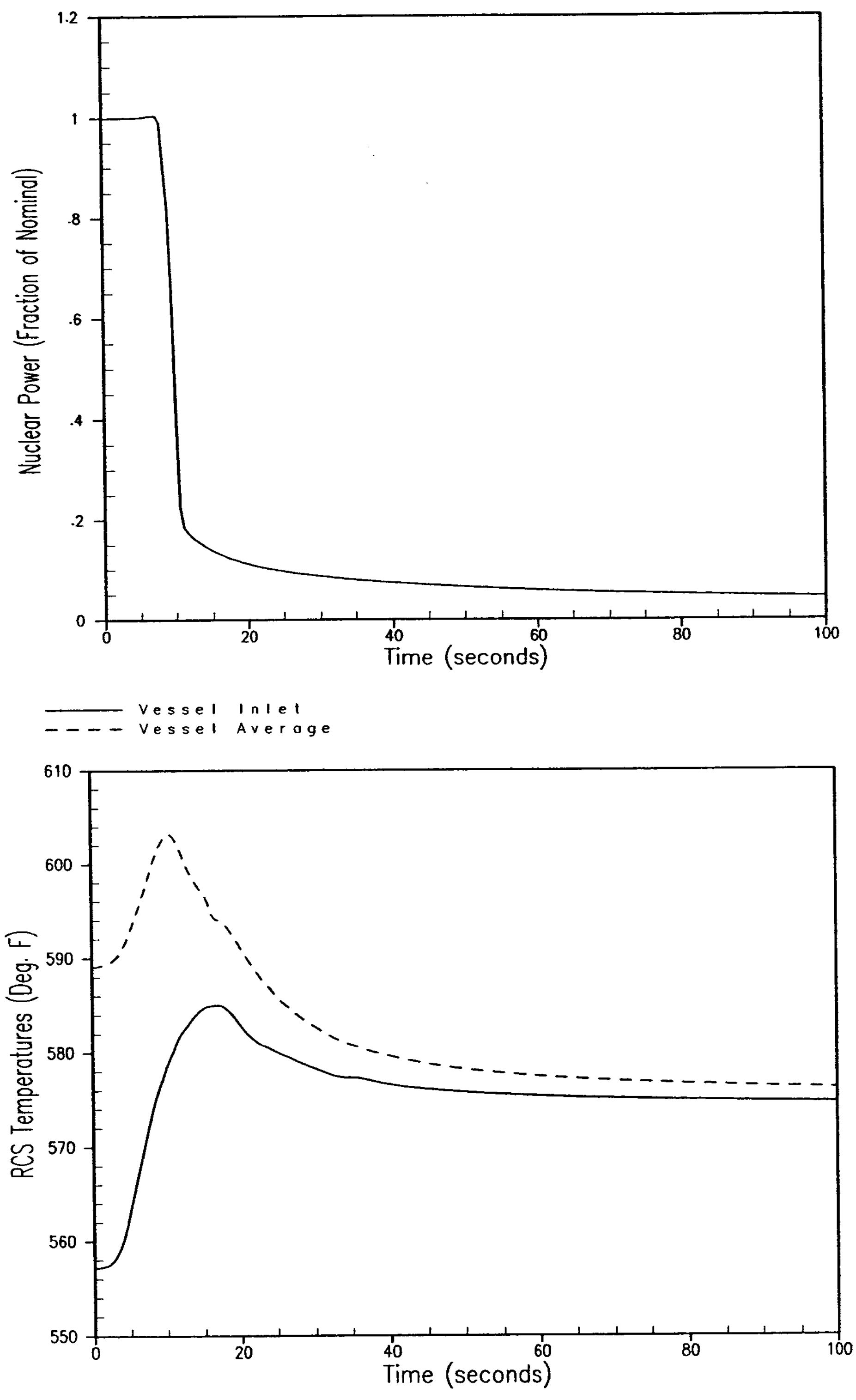




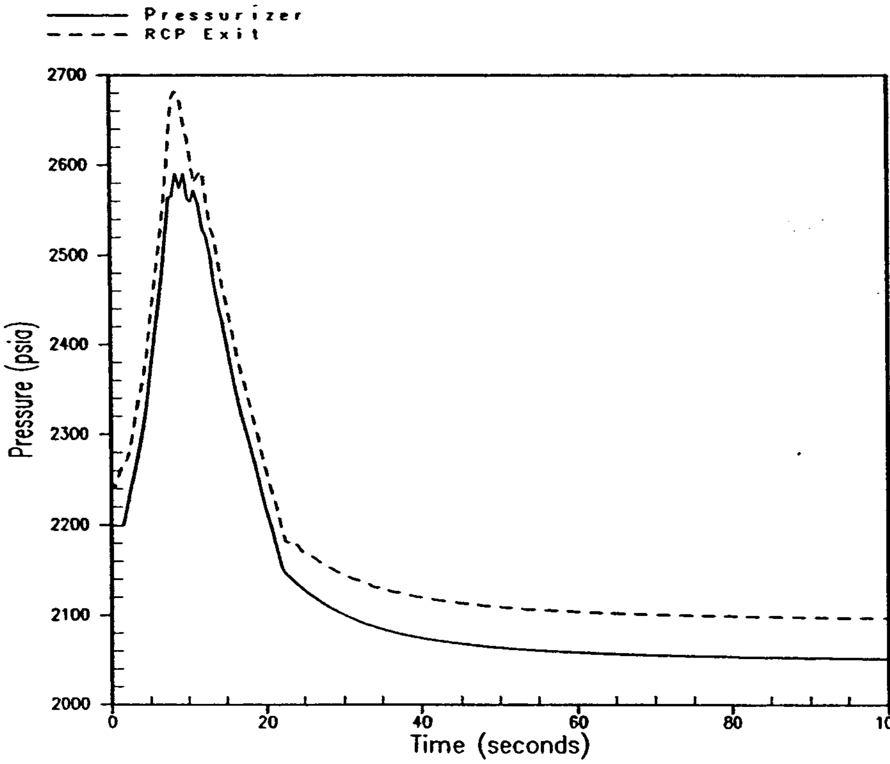
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Presssurizer Pressure and DNBR Transients for a Turbine Trip Event With Pressure Control	
		Figure 15.2-1 Sh. 2 of 5

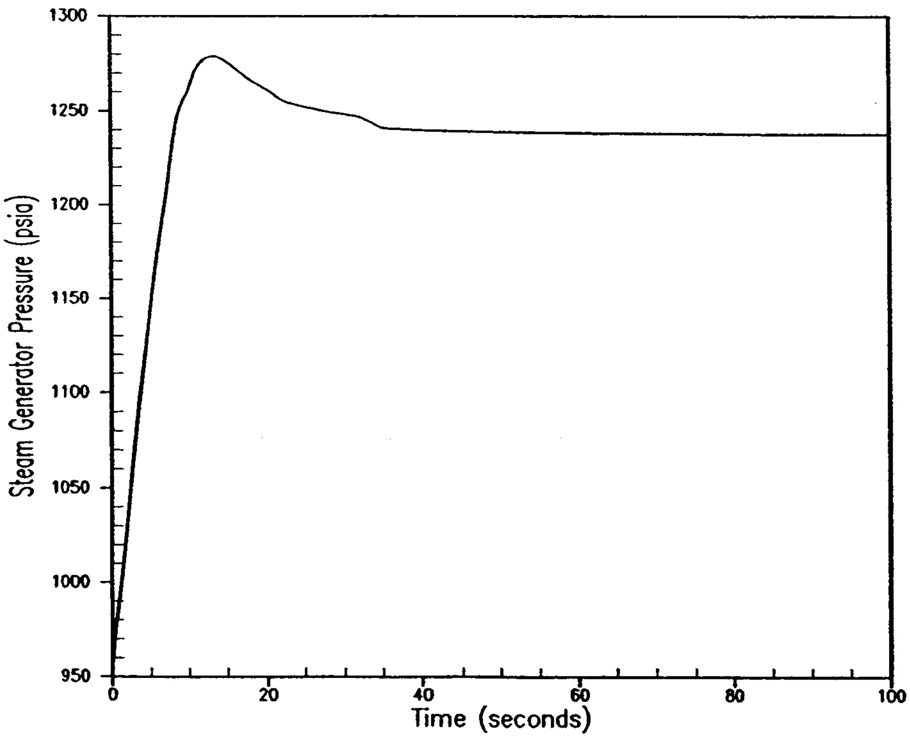


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and DNBR Transients for a Turbine Trip Event With Pressure Control	
		Figure 15.2-1 Sh. 3 of 5



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Temperature Transients for a Turbine Trip Event Without Pressure Control
	Figure 15.2-1 Sh. 4 of 5





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Temperature Transients for a Turbine Trip Event Without Pressure Control	
		Figure 15.2-1 Sh. 5 of 5

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FIGURE 15.2-2

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SEABROOK UPDATED FSAR

FIGURE 15.2-3

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FIGURE 15.2-4

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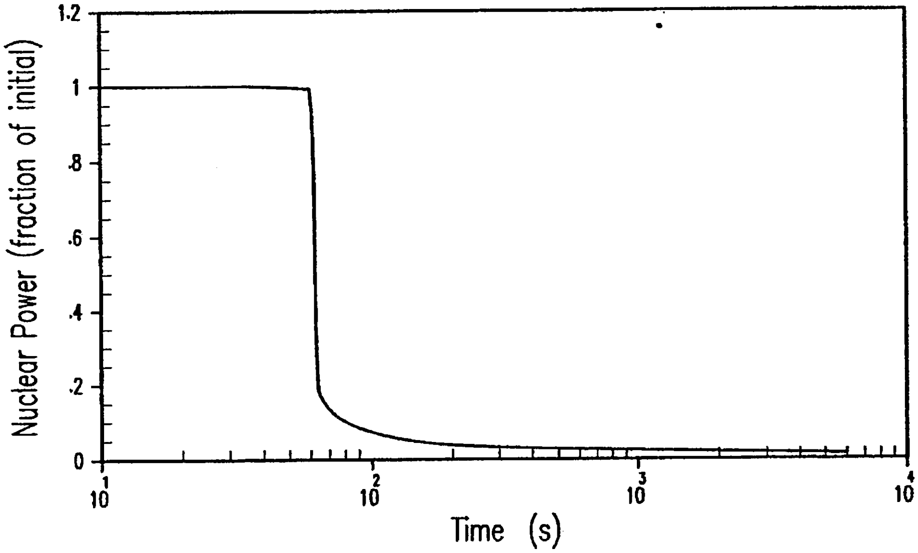
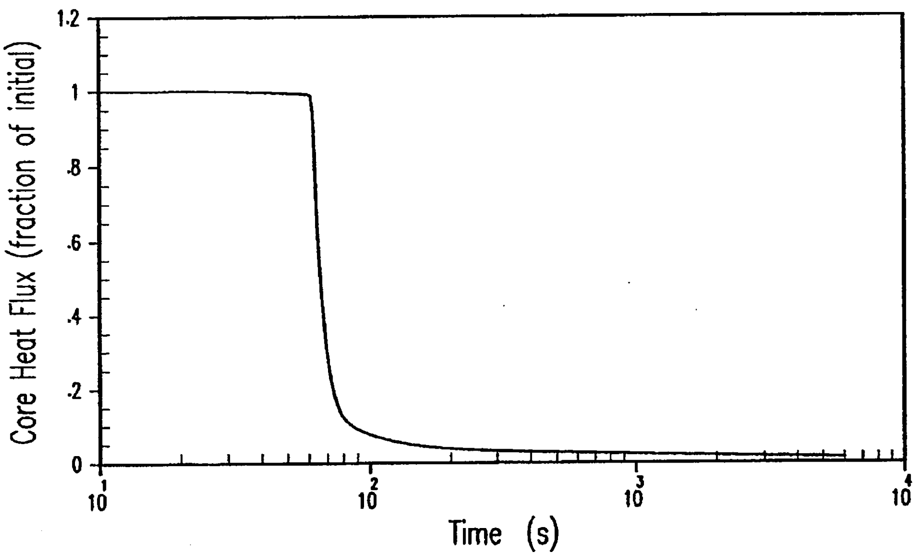
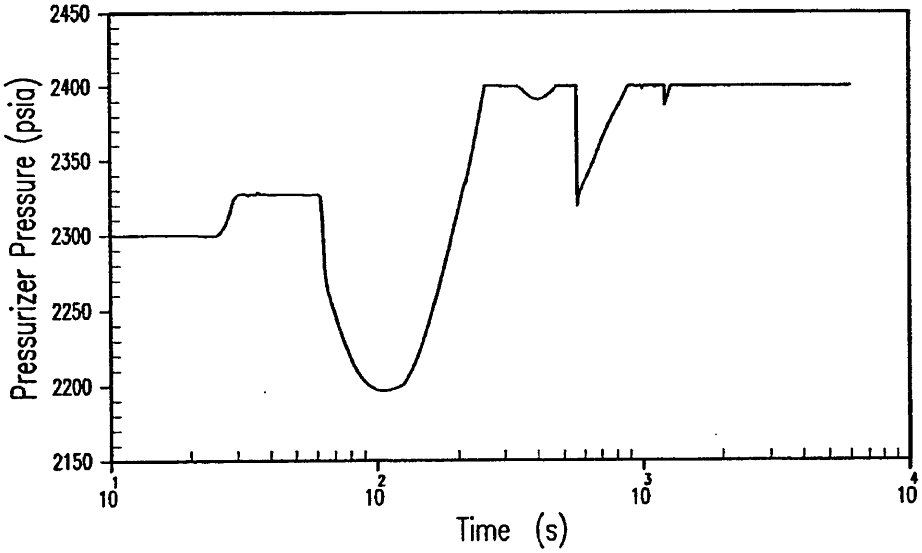
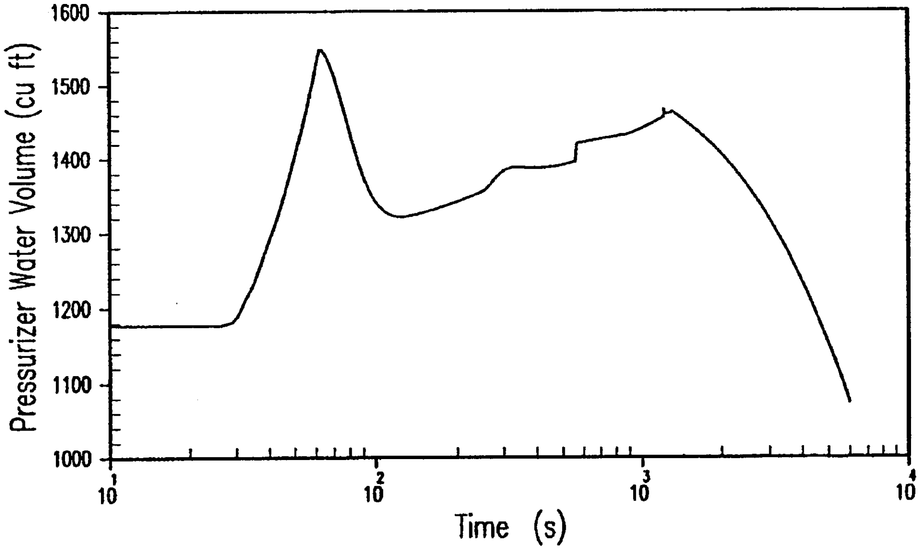
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	DELETED	
		Figure 15-2-4



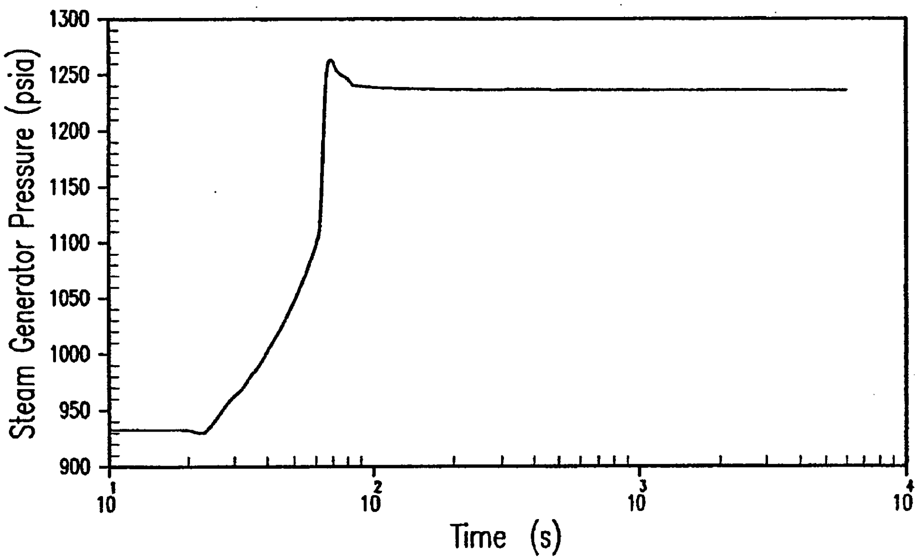


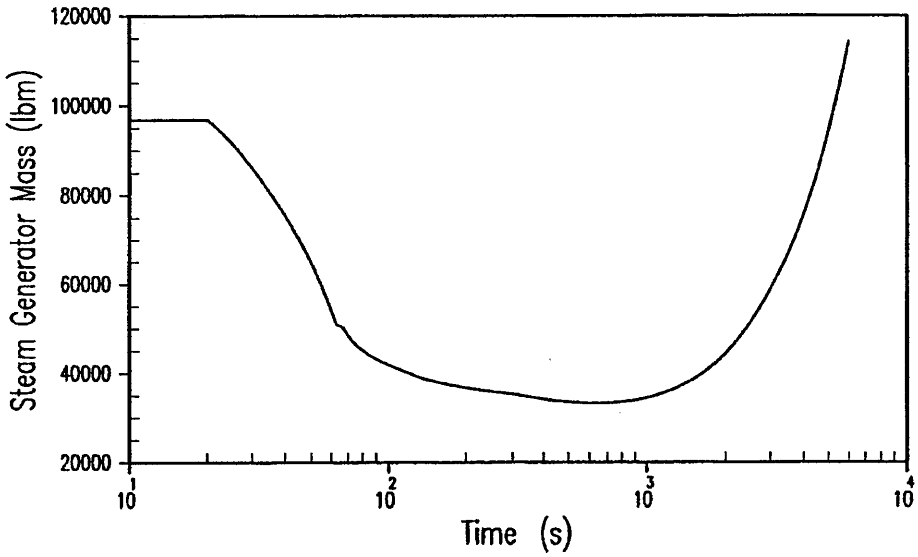
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Loss of Non-Emergency AC to the Station Auxiliaries	
		Figure 15.2-5 Sh. 1 of 4



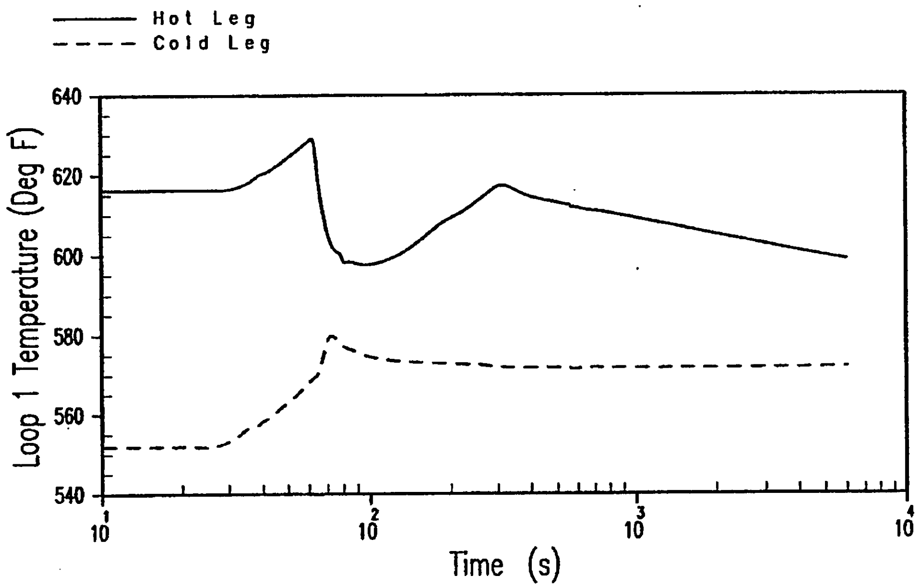


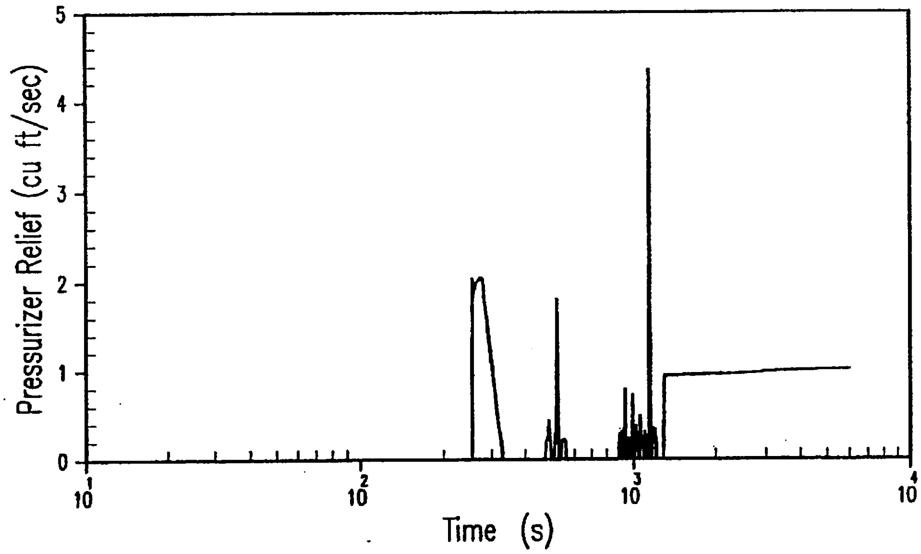
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for a Loss of Non-Emergency AC to the Station Auxiliaries	
		Figure 15.2-5 Sh. 2 of 4



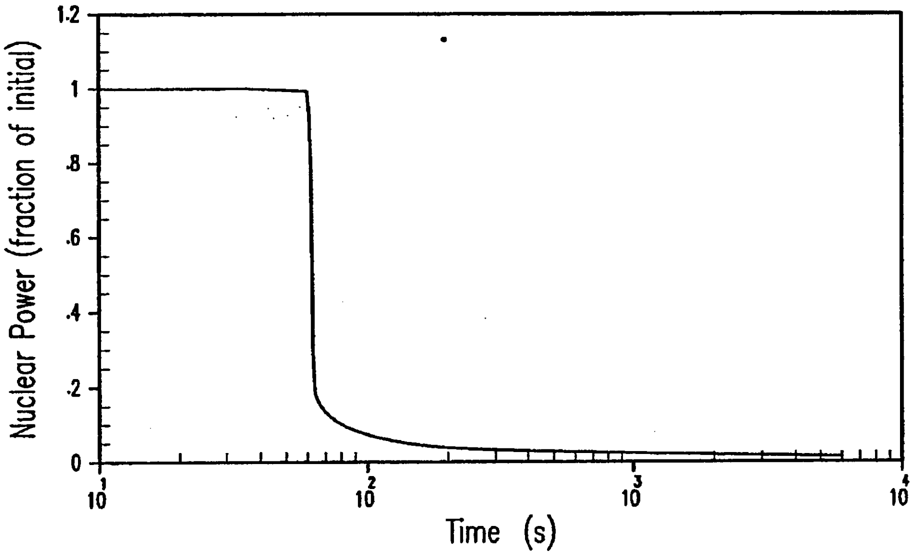


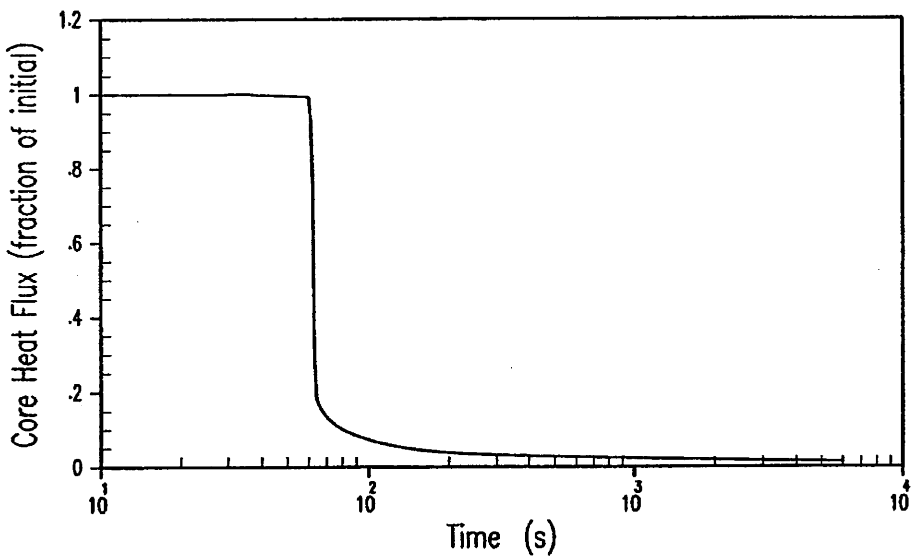
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	SG Pressure and SG Mass Transients for a Loss of Non-Emergency AC to the Station Auxiliaries	
		Figure 15.2-5 Sh. 3 of 4



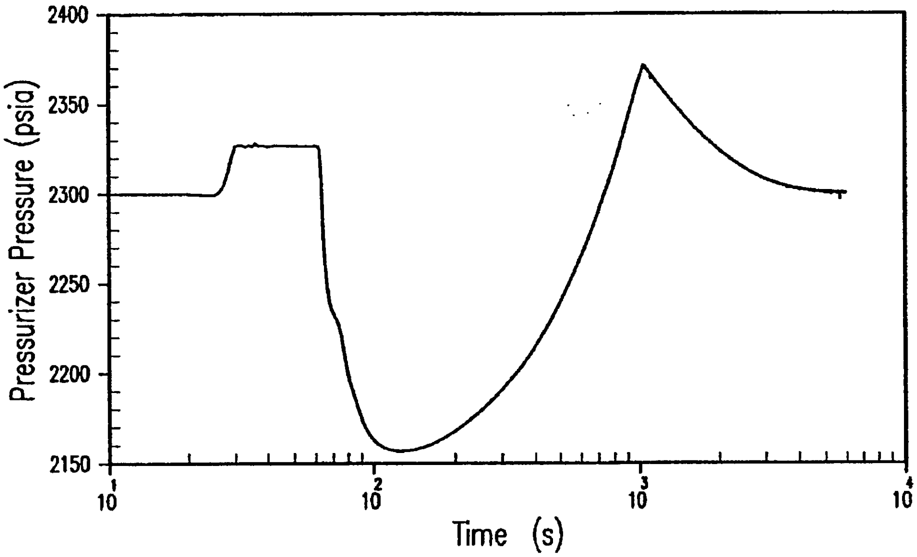


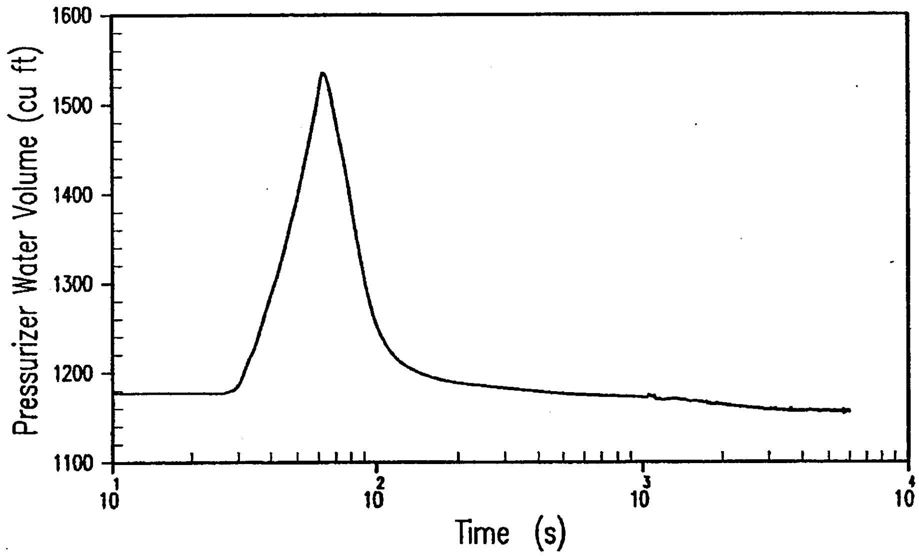
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loop 1 Temperature and Pressurizer Relief Transients for a Loss of Non-Emergency AC to the Station Auxiliaries	
	Figure	15.2-5 Sh. 4 of 4



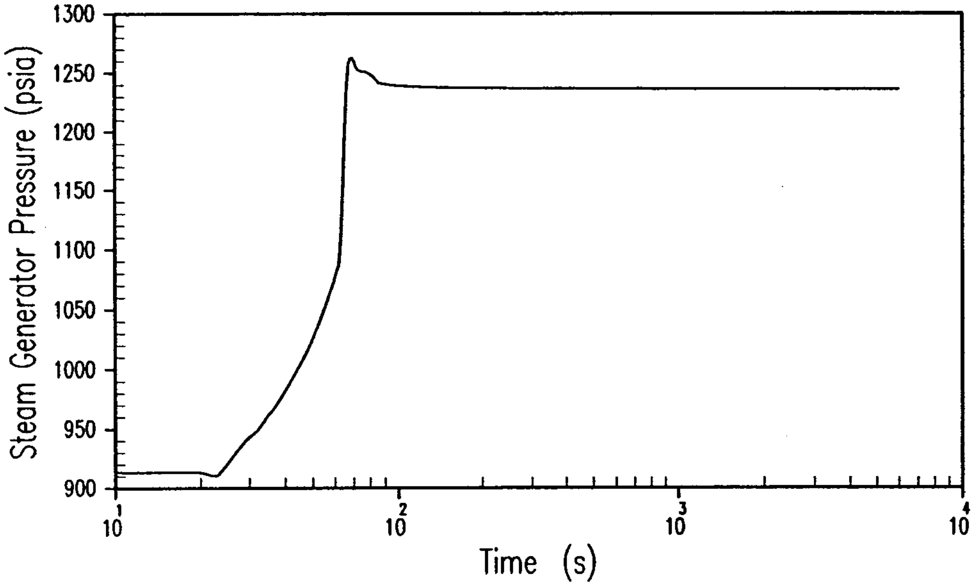


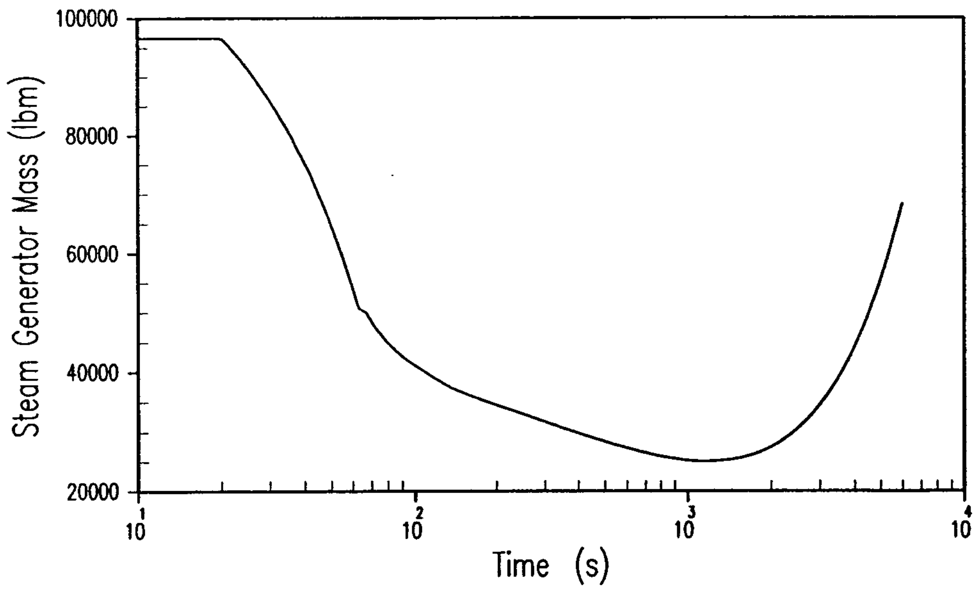
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Loss of Normal Feedwater
	Figure 15.2-6 Sh. 1 of 4

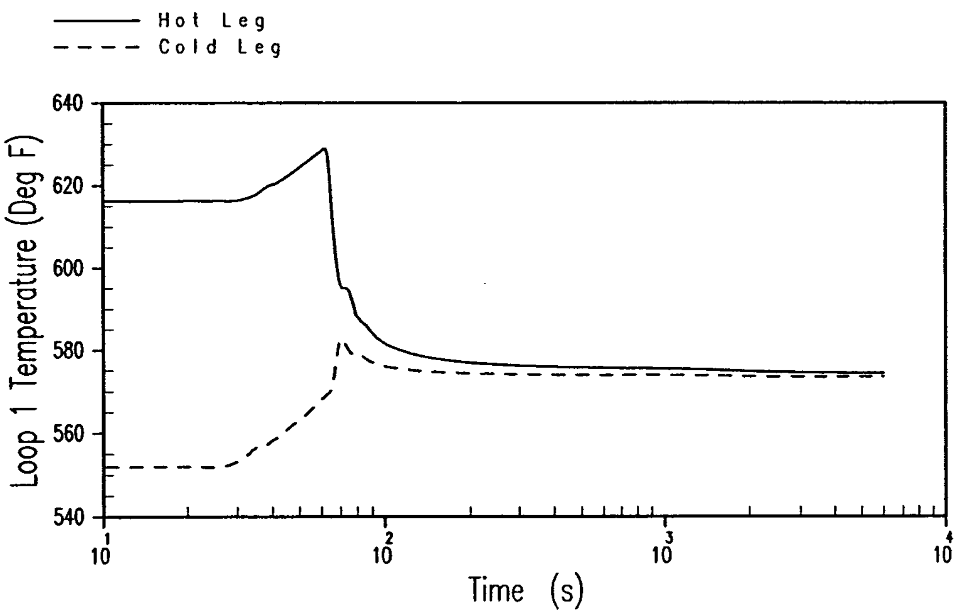


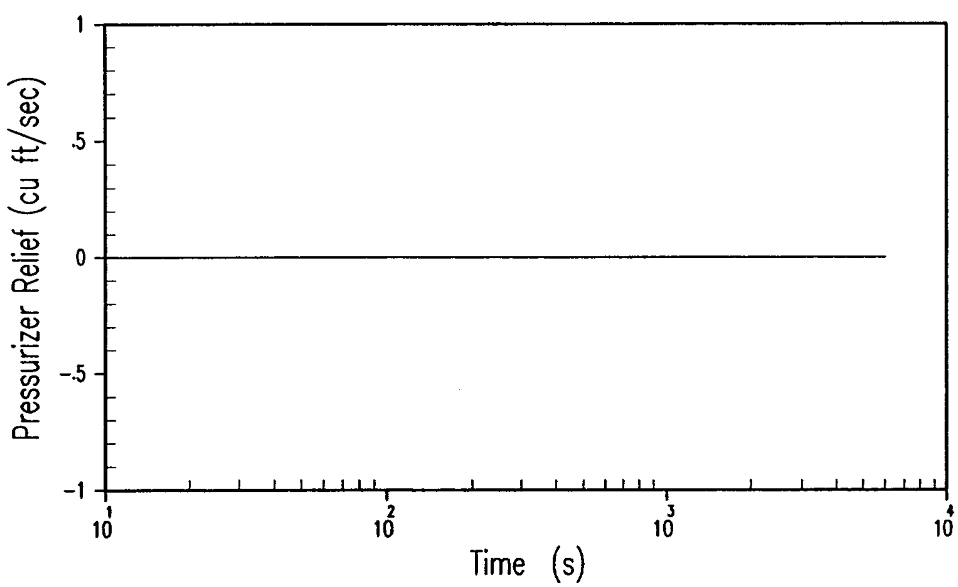


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for a Loss of Normal Feedwater	
		Figure 15.2-6 Sh. 2 of 4

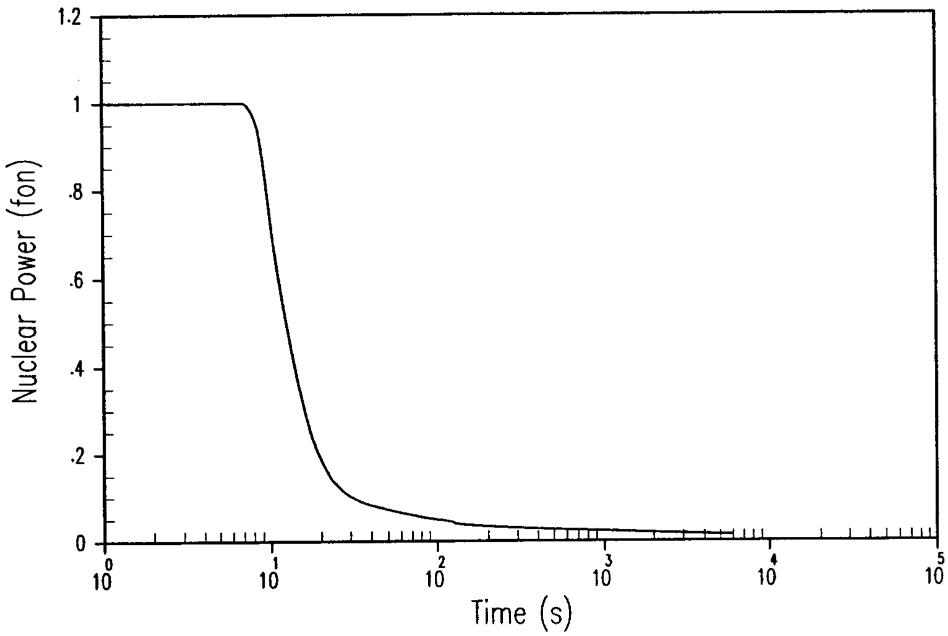

1300
1250
1200
1150
1100
1050
1000
950
900
Steam Generator Pressure (psia)
10^1
10^2
10^3
10^4
Time (s)

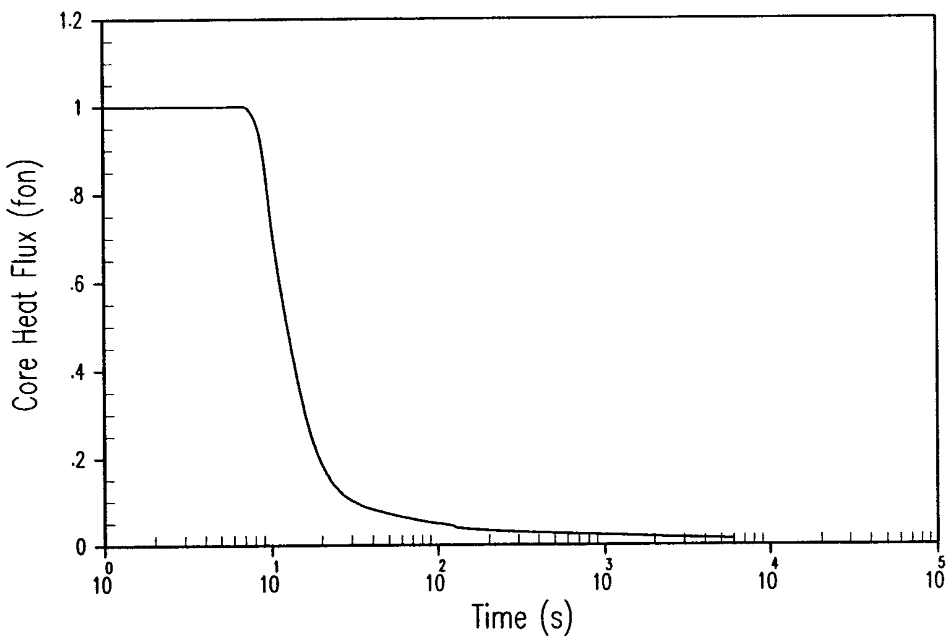

100000
80000
60000
40000
20000
Steam Generator Mass (lbm)
10^1
10^2
10^3
10^4
Time (s)



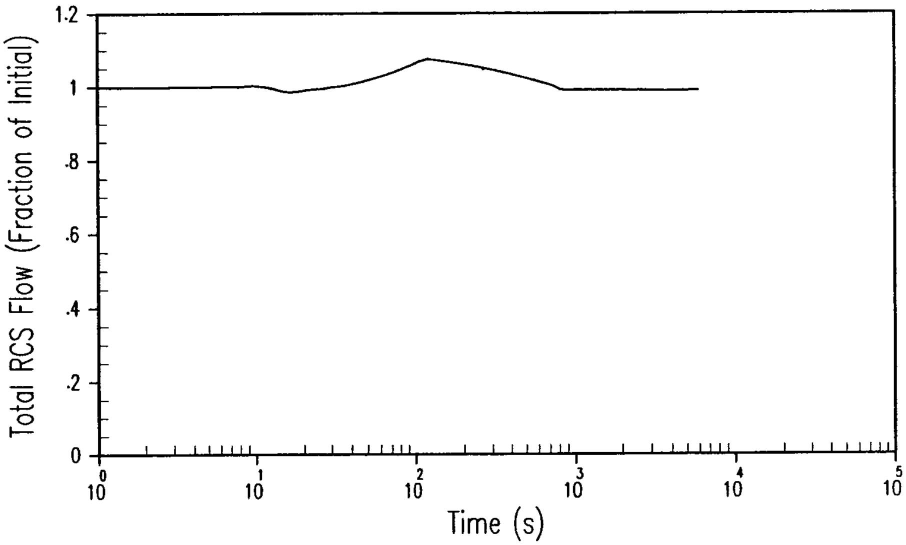


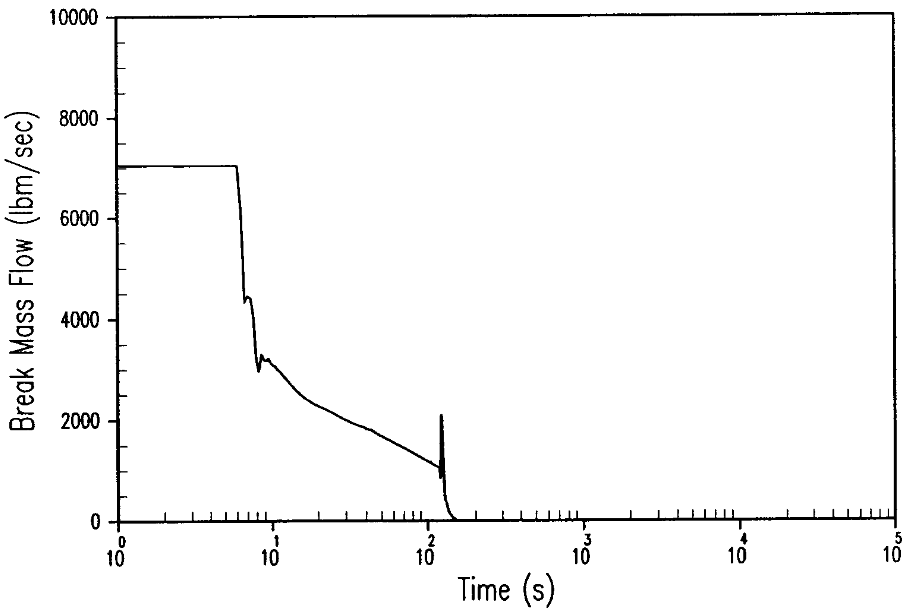
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loop 1 Temperature and Pressurizer Relief Transients for a Loss of Normal Feedwater	
		Figure 15.2-6 Sh. 4 of 4



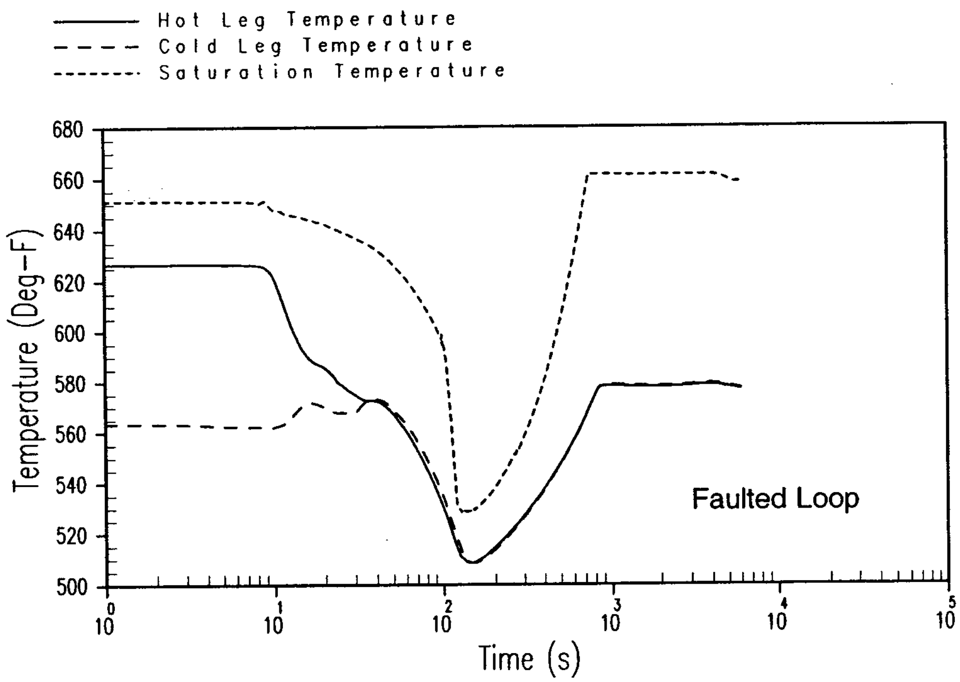


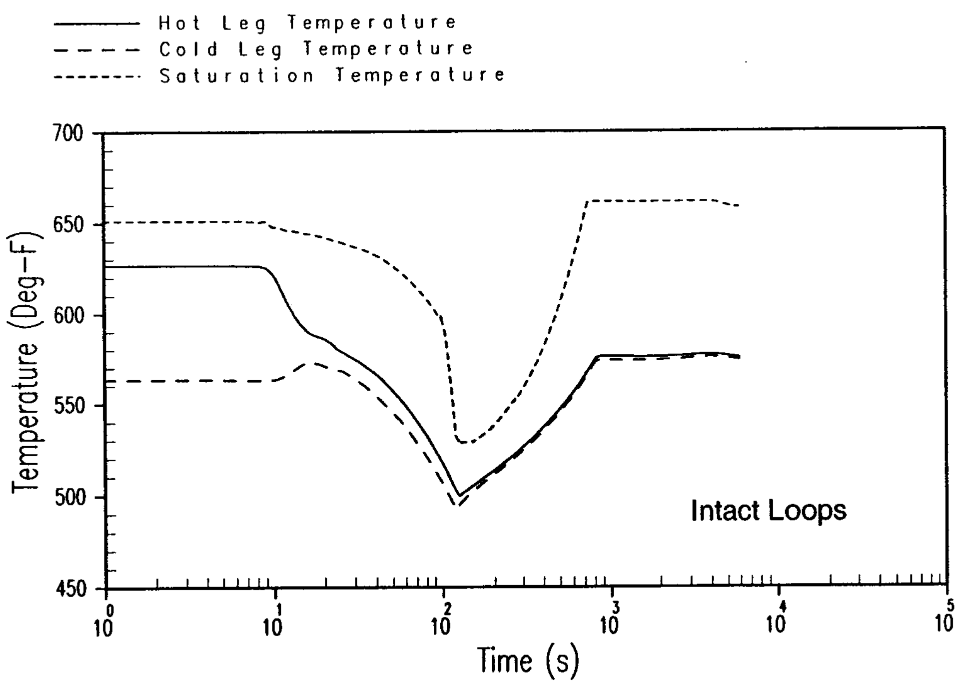
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Heat Flux Transients for a Feedwater System Pipe Break	
		Figure 15.2-7 Sh. 1 of 5



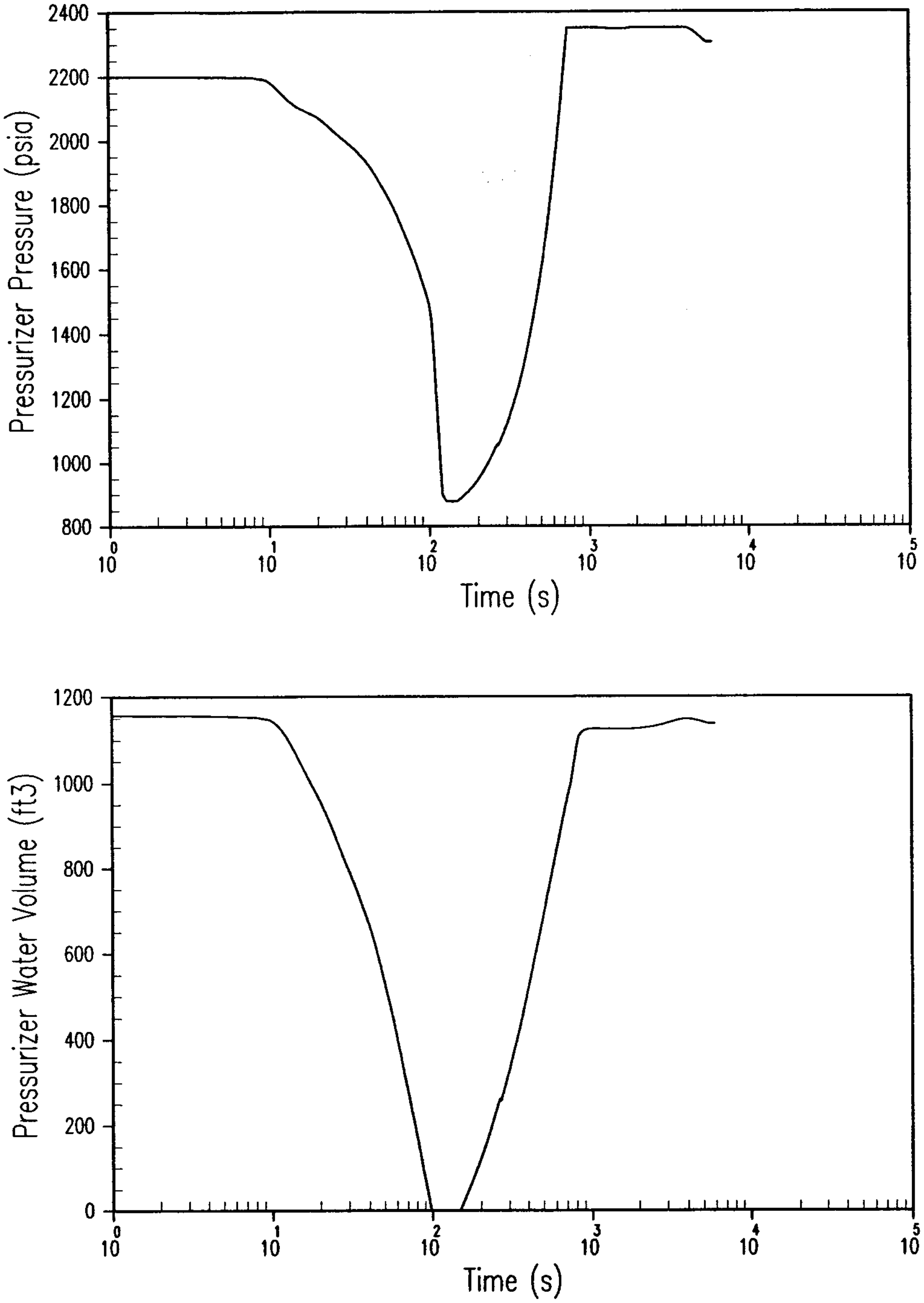


Seabrook Station Updated Final Safety Analysis Report	RCS Flow and FLB Flow Transients for a Feedwater System Pipe Break
	Figure 15.2-7 Sh. 2 of 5

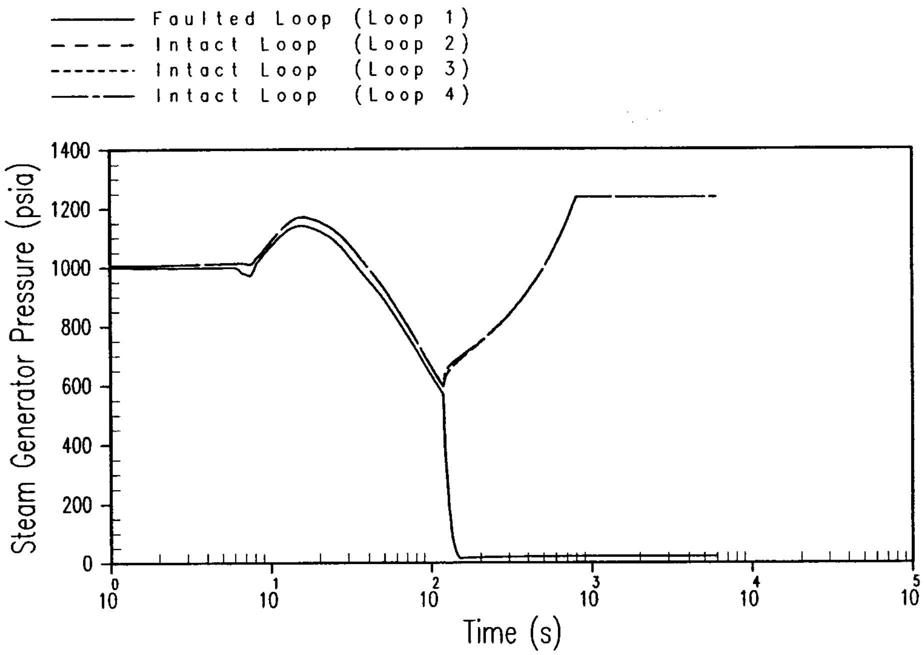


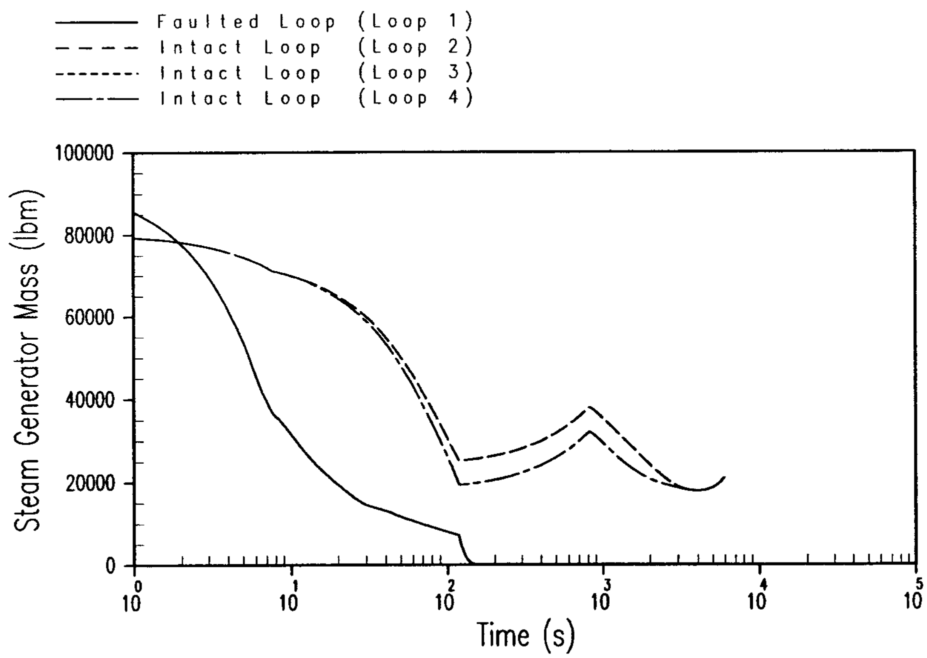


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Loop 1 (faulted) and Loop 2 (unfaulted) Temperature Transients for a Feedwater System Pipe Break	
		Figure 15.2-7 Sh. 3 of 5



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for a Feedwater System Pipe Break	
		Figure 15.2-7 Sh. 4 of 5





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Pressure and Steam Generator Mass Transients for a Feedwater System Pipe Break	
		Figure 15.2-7 Sh. 5 of 5

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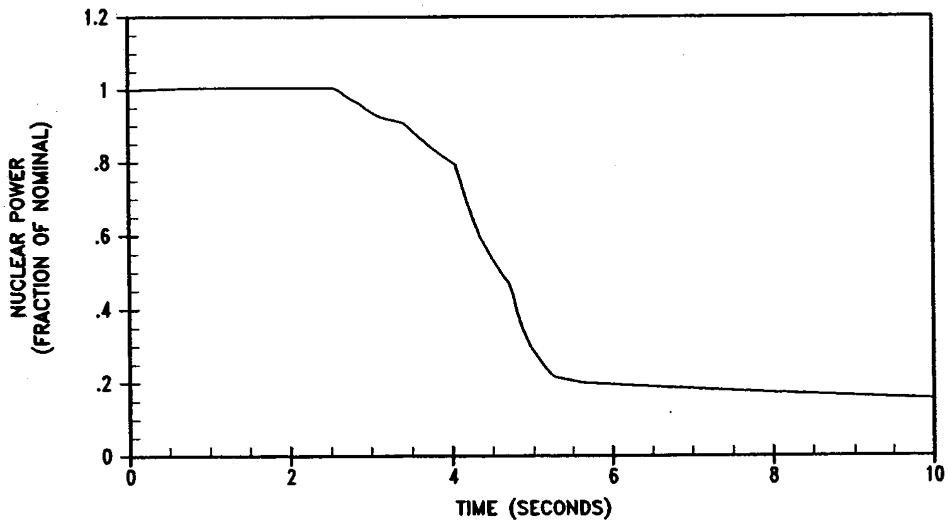
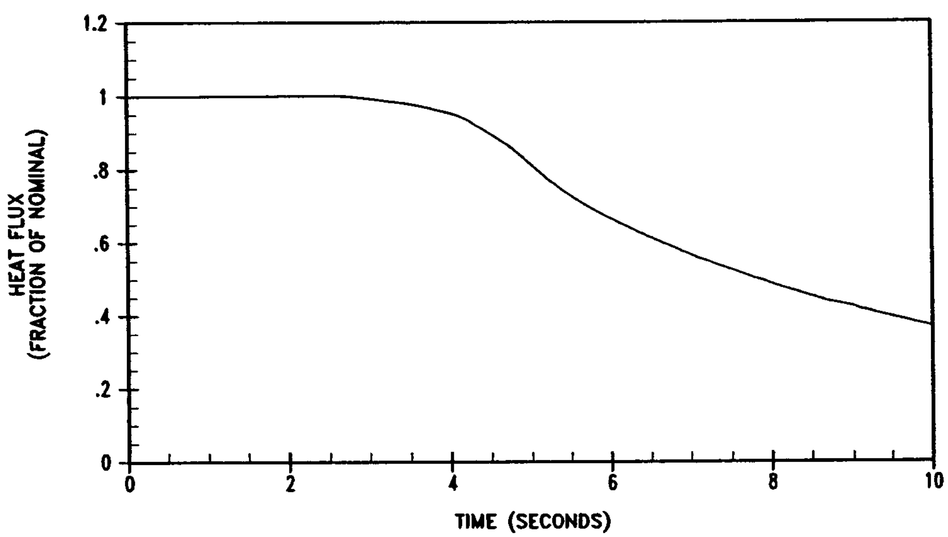
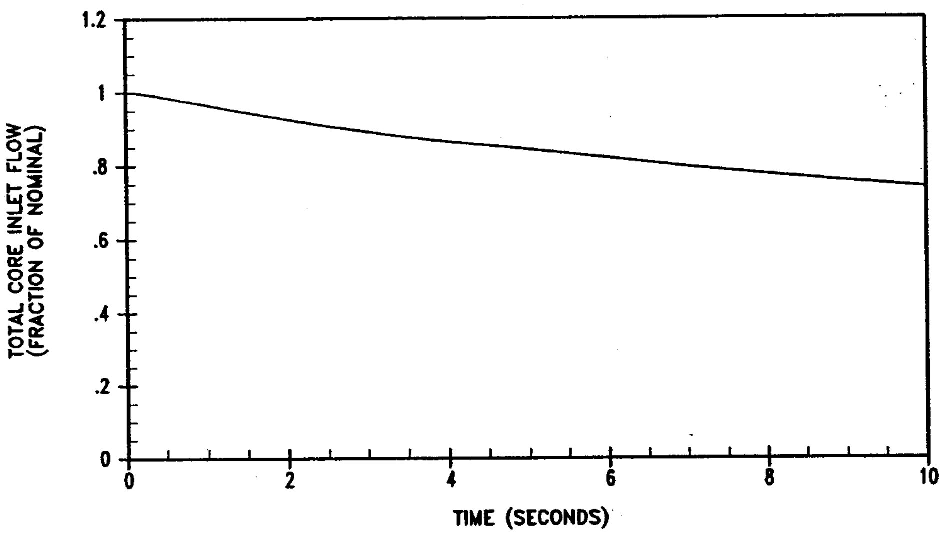
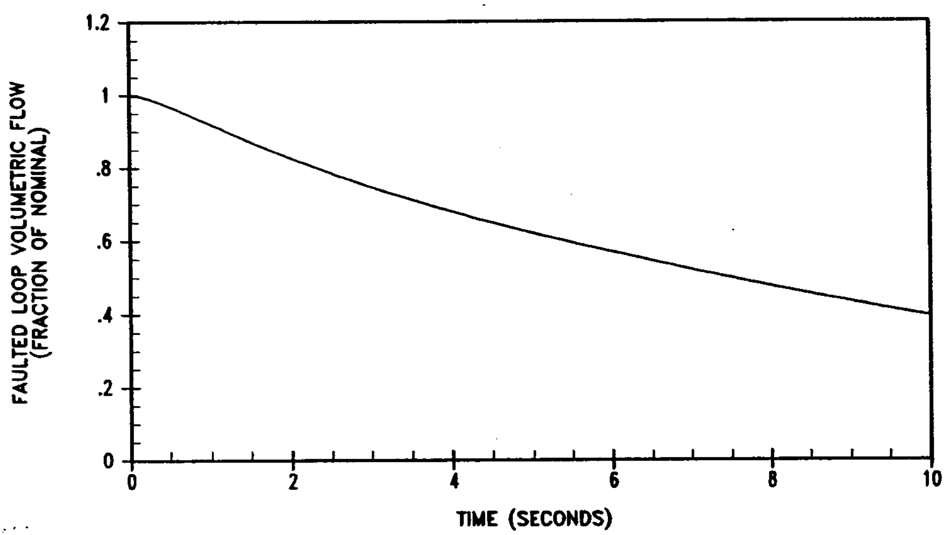


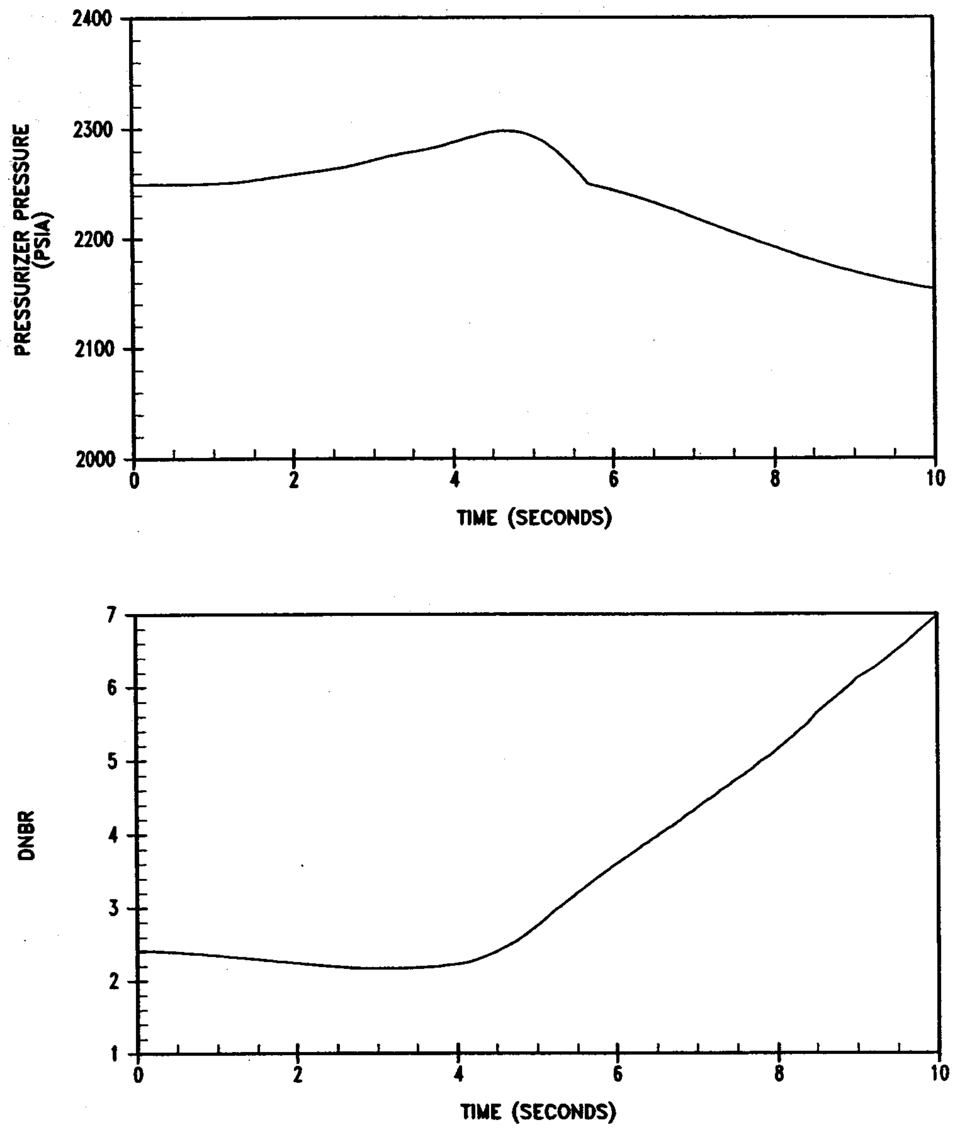


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Heat Flux Transients for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
		Figure 15.3-1 Sh. 1 of 2

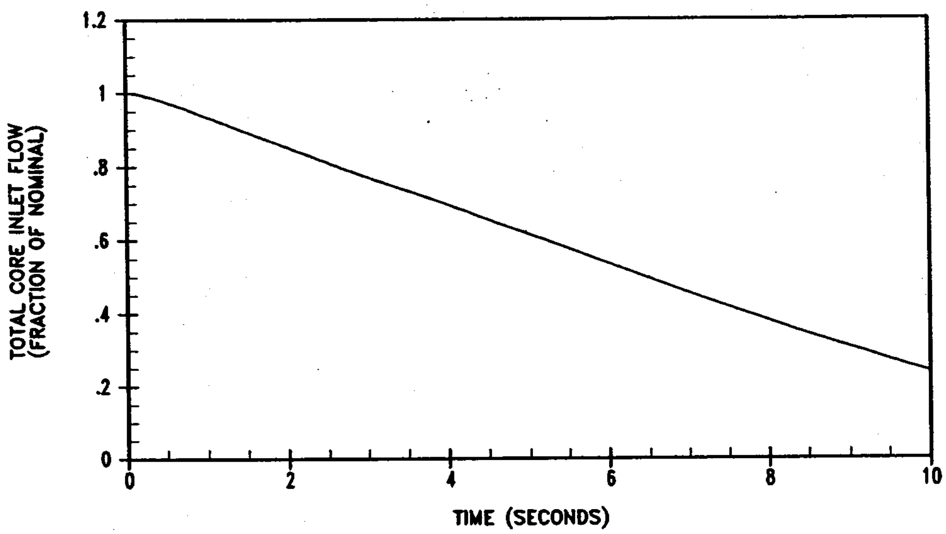


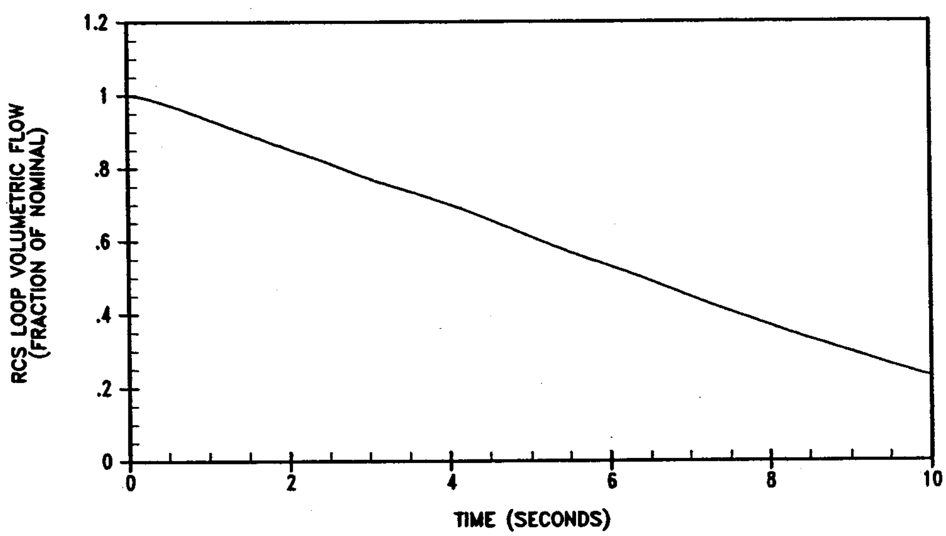


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Flow and Faulted Loop Flow Transients for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
		Figure 15.3-1 Sh. 2 of 2

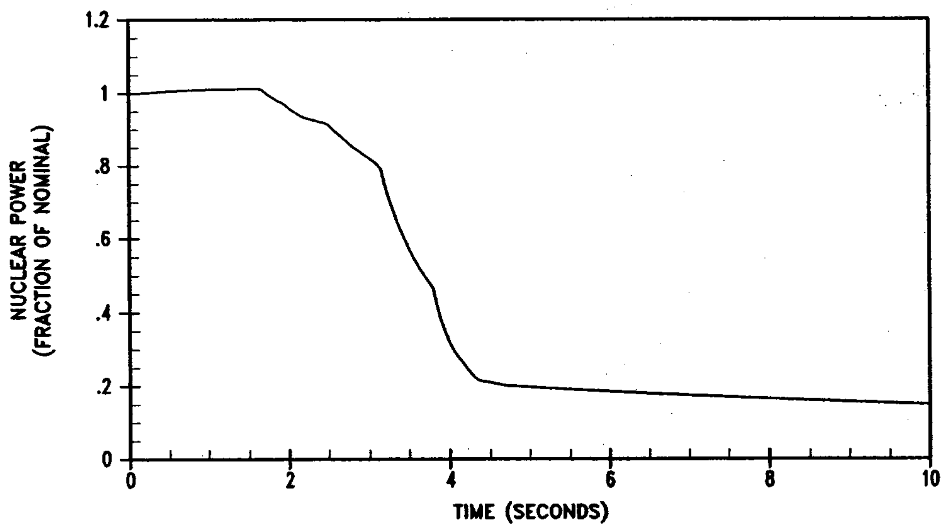


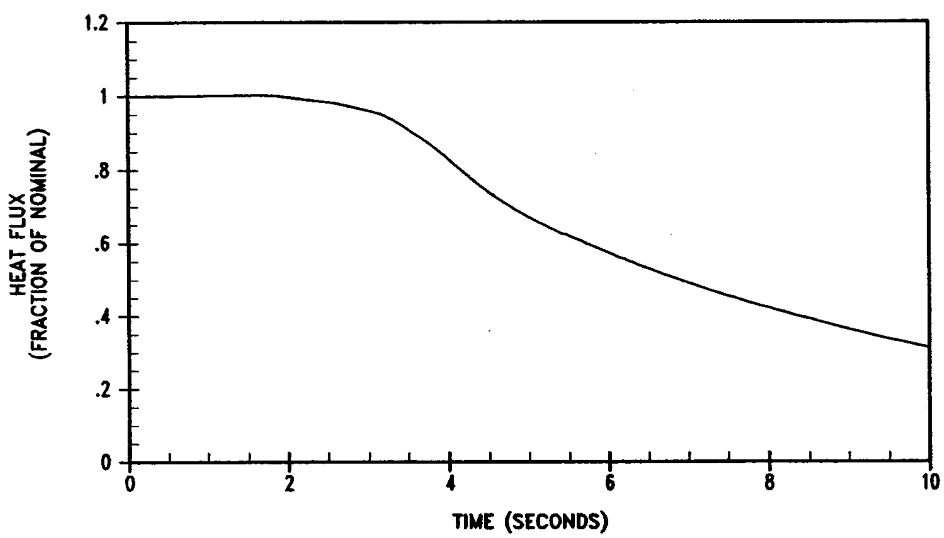
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant Pressure and DNBR Transients for a Partial Loss of Forced Reactor Coolant Flow (4 loops in operation, 2 RCPs coasting down)	
		Figure 15.3-2



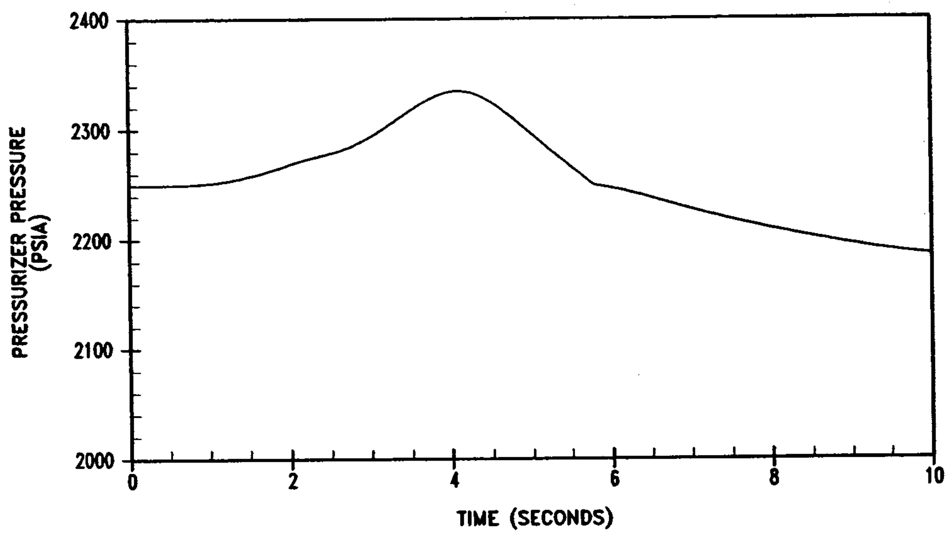


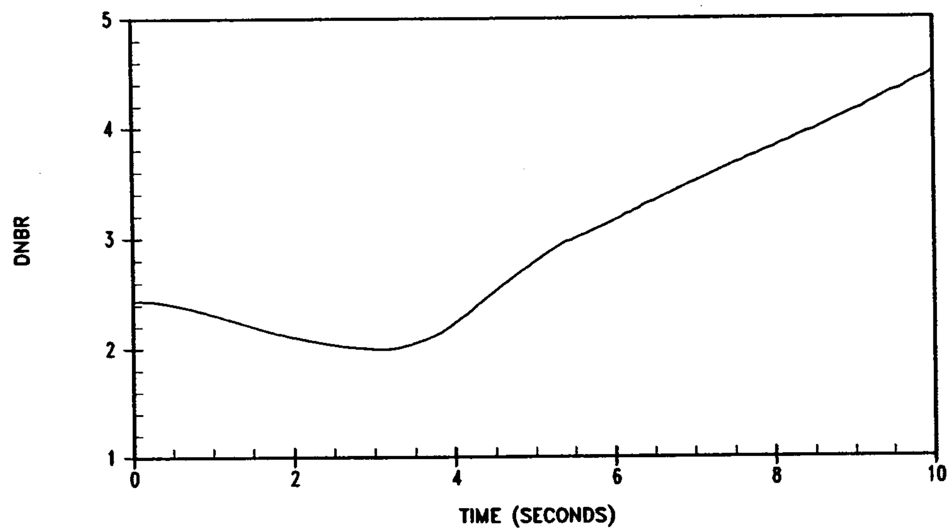
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Flow and Faulted Loop Flow Transients for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
		Figure 15.3-3



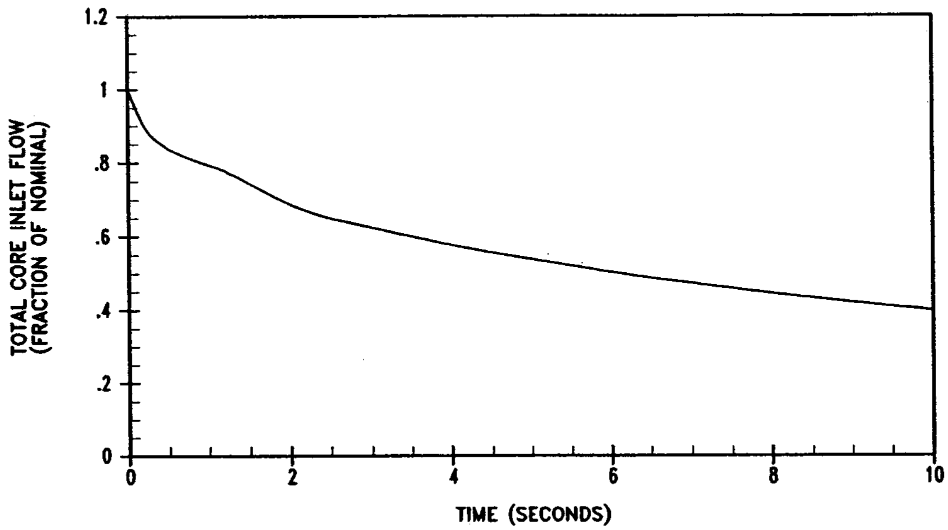


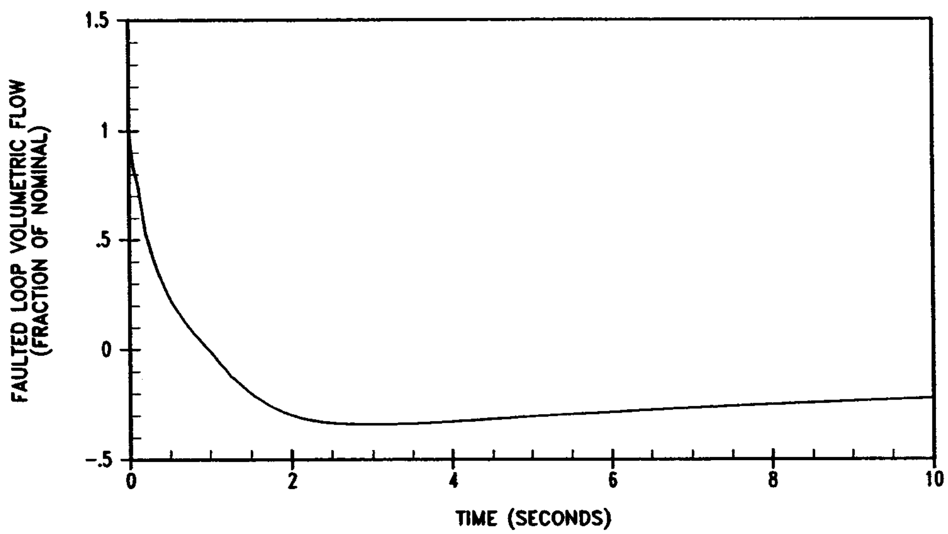
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Heat Flux Transients for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
		Figure 15.3-4



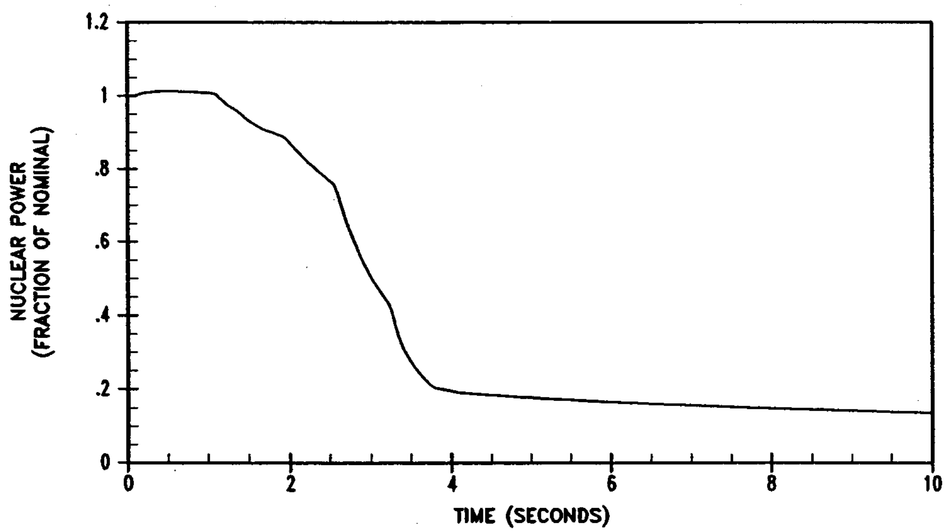


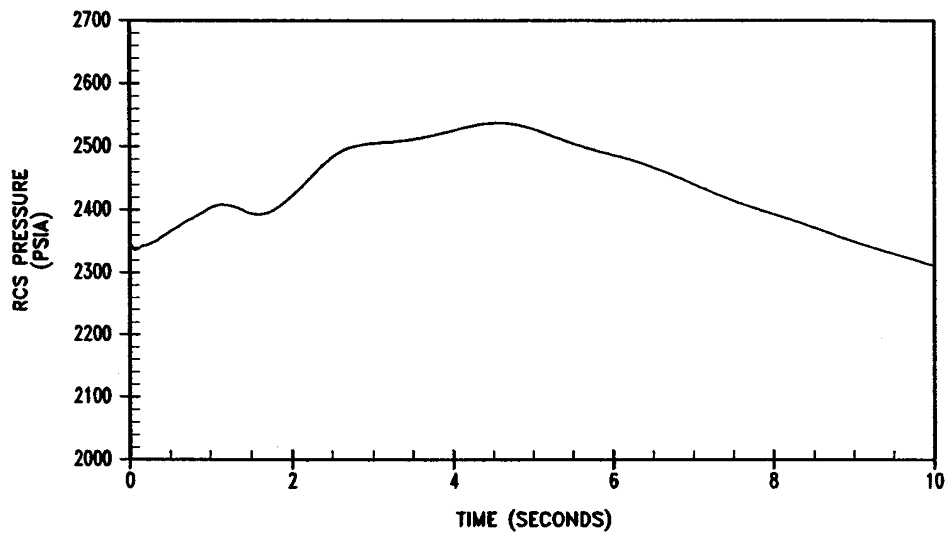
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant Pressure and DNBR Transients for a Complete Loss of Forced Reactor Coolant Flow (4 loops in operation, 4 RCPs coasting down)	
		Figure 15.3-5



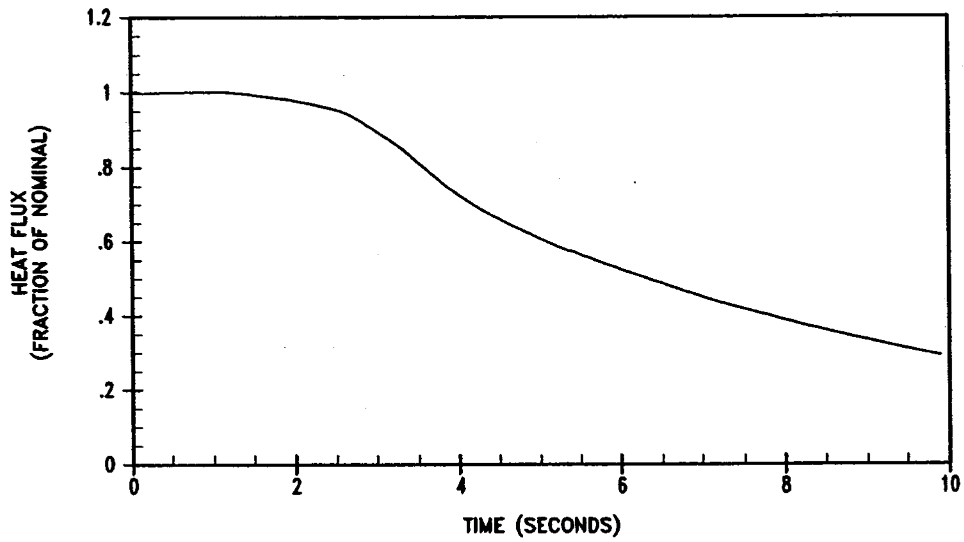


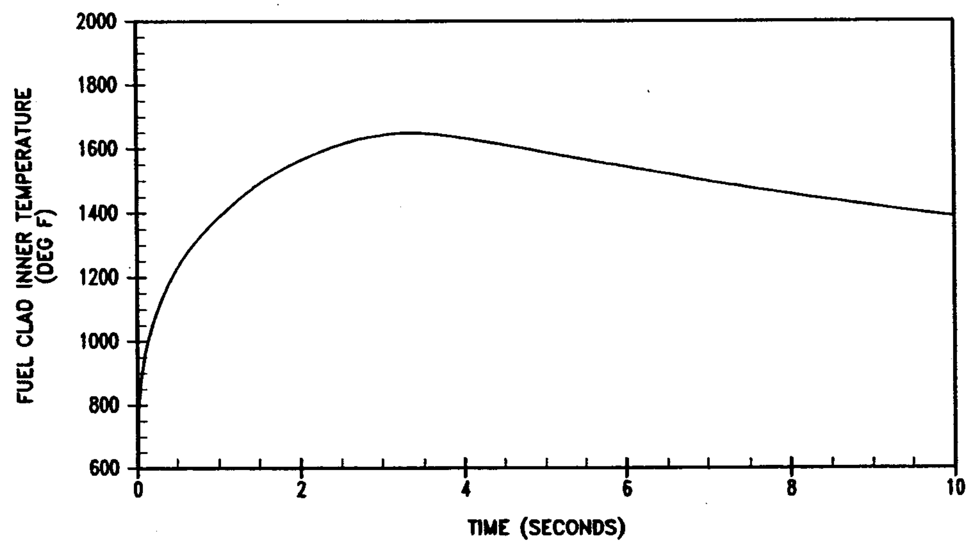
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Flow and Faulted Loop Flow Transients for a Locked Rotor	
		Figure 15.3-6 Sh. 1 of 3





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and RCS Pressure Transients for a Locked Rotor	
		Figure 15.3-6 Sh. 2 of 3





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Heat Flux and Clad Inner Temperature Transient for a Locked Rotor	
		Figure 15.3-6 Sh. 3 of 3

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FIGURE 15.3-7

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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	DELETED	
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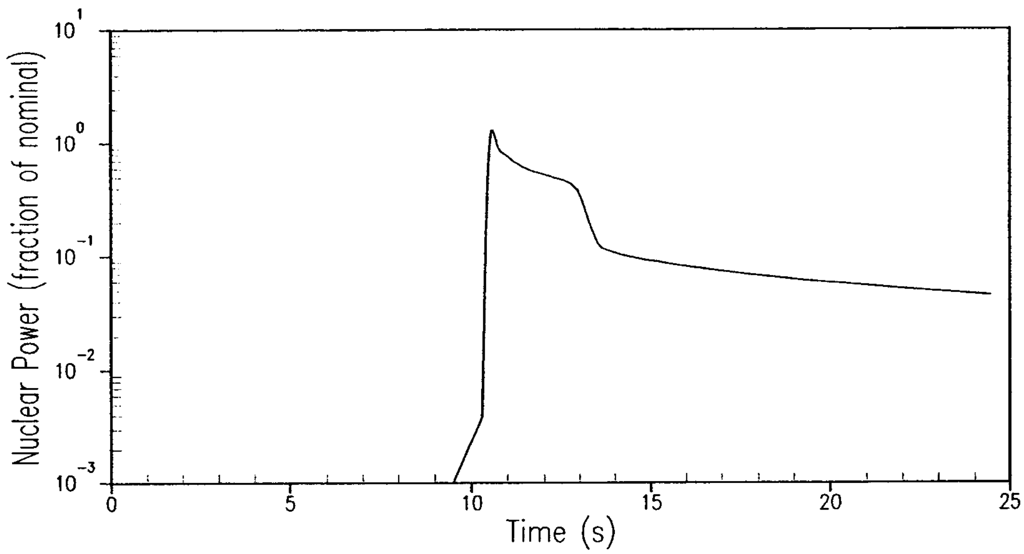
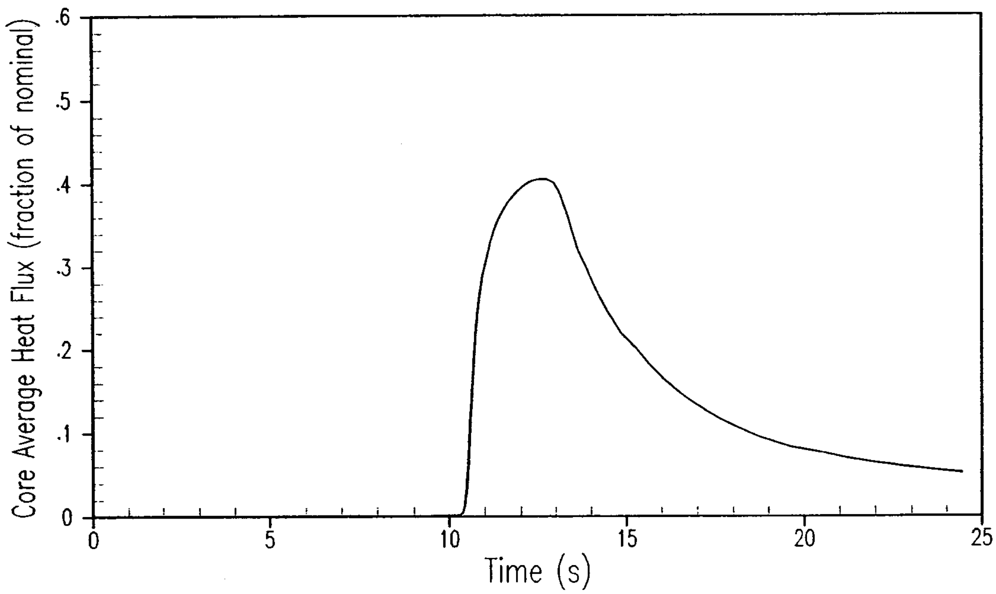
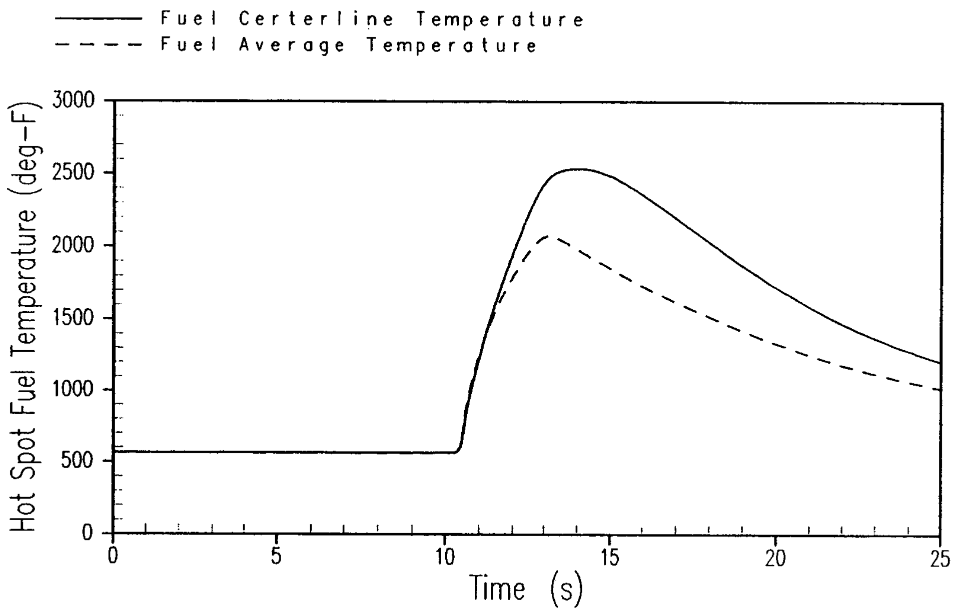


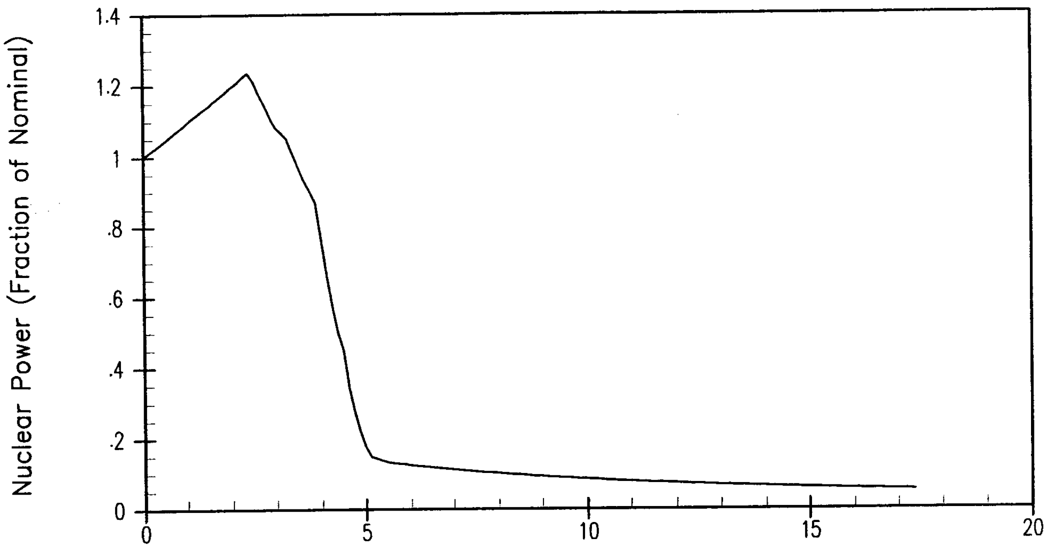
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power Transient for Uncontrolled Rod Withdrawal from a Subcritical Condition	
		Figure 15.4-1

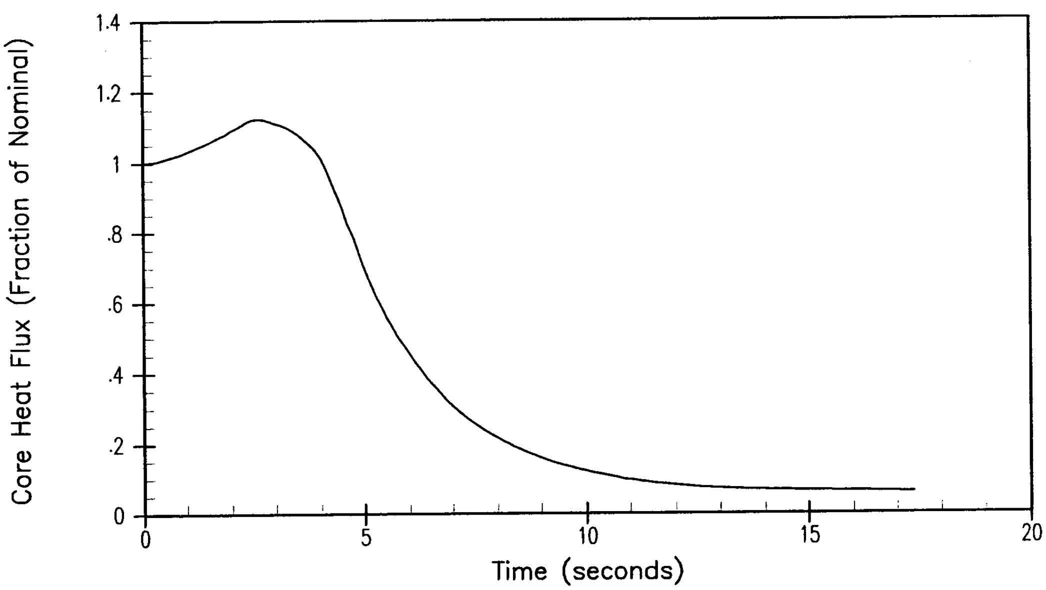


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Heat Flux Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition	
		Figure 15.4-2

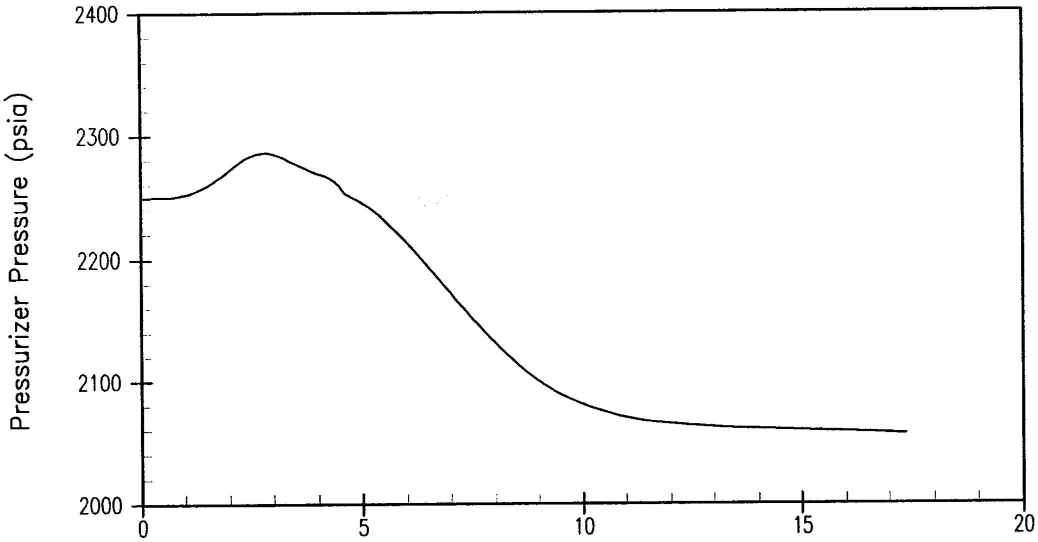


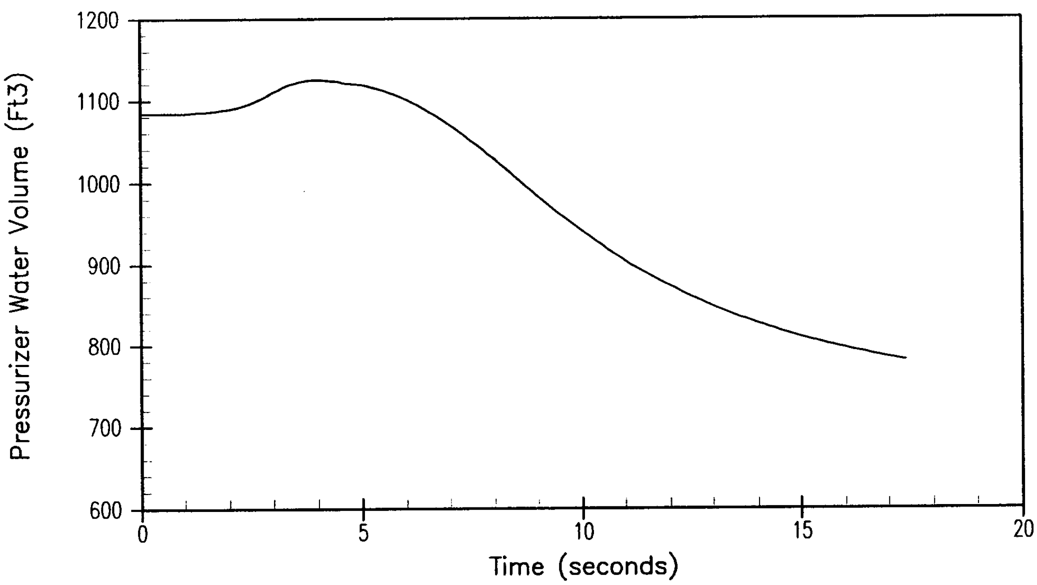
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline and Fuel Average Hot Spot Temperature Transients for Uncontrolled Rod Withdrawal from a Subcritical Condition
	Figure 15.4-3



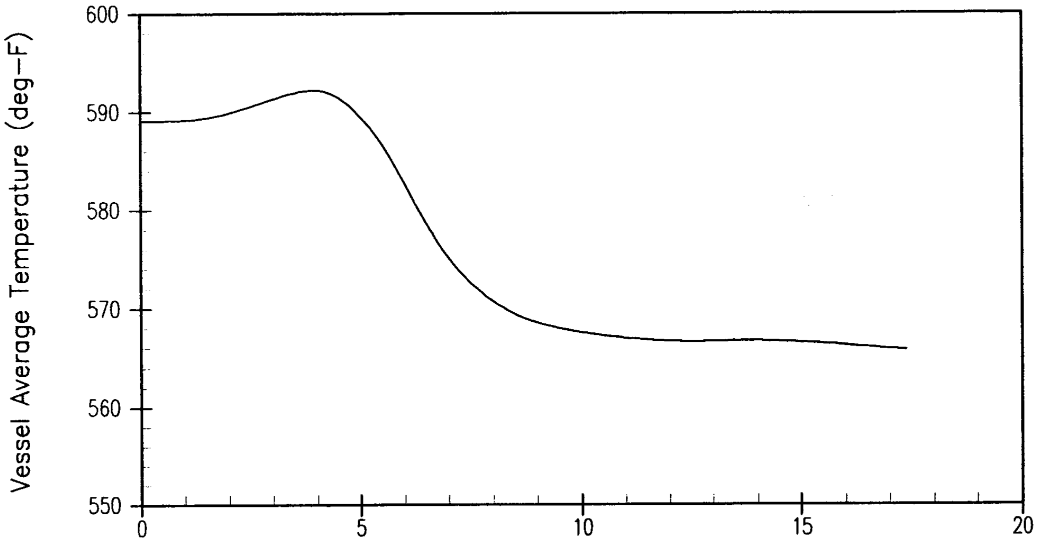


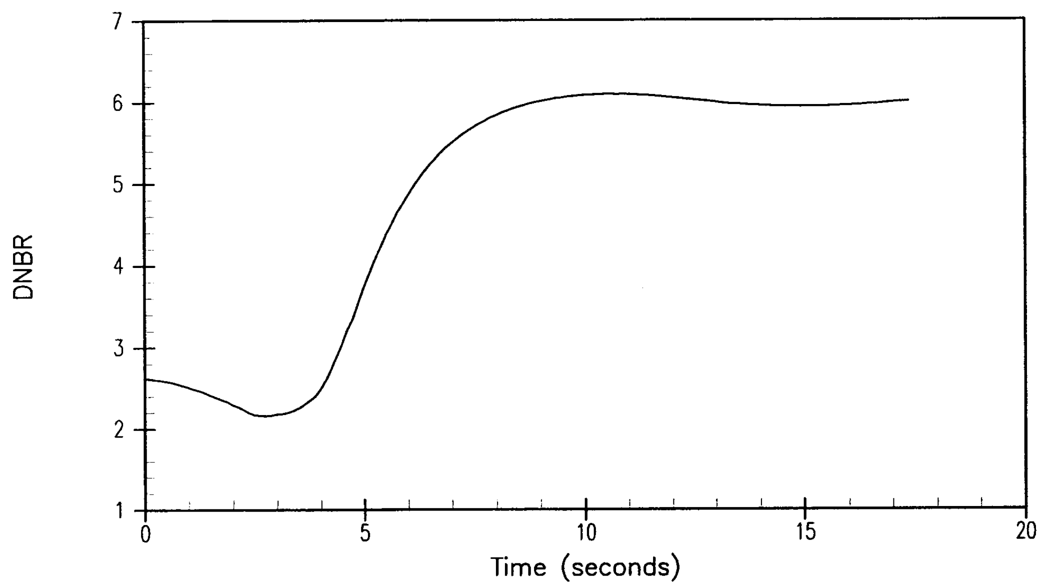
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Heat Flux Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
		Figure 15.4-4 Sh. 1 of 3



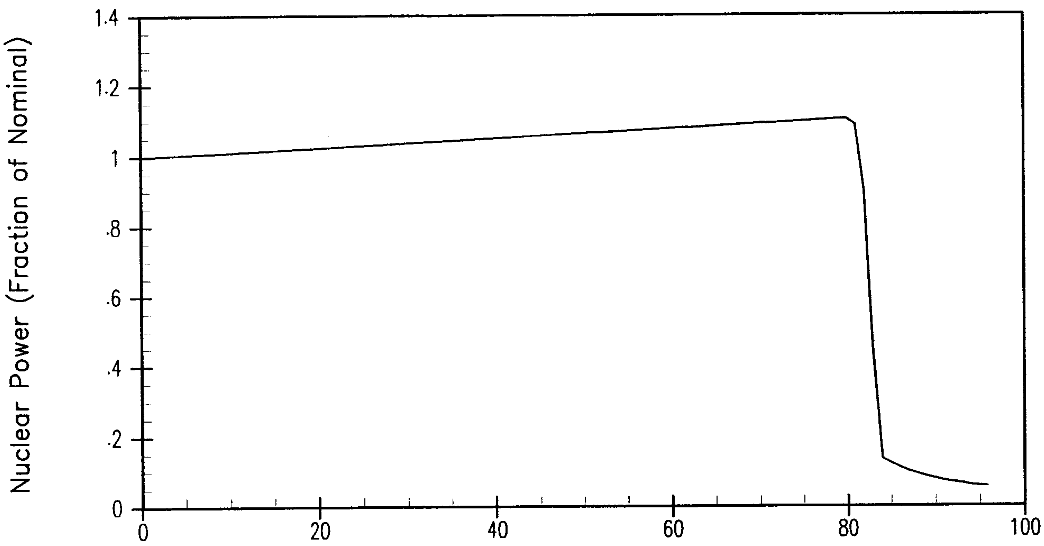


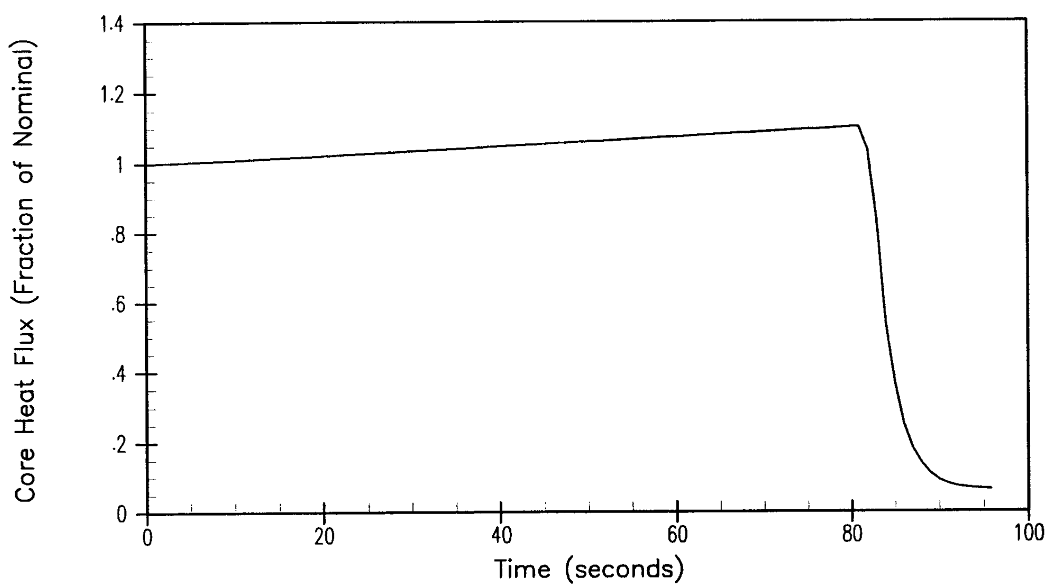
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
		Figure 15.4-4 Sh. 2 of 3



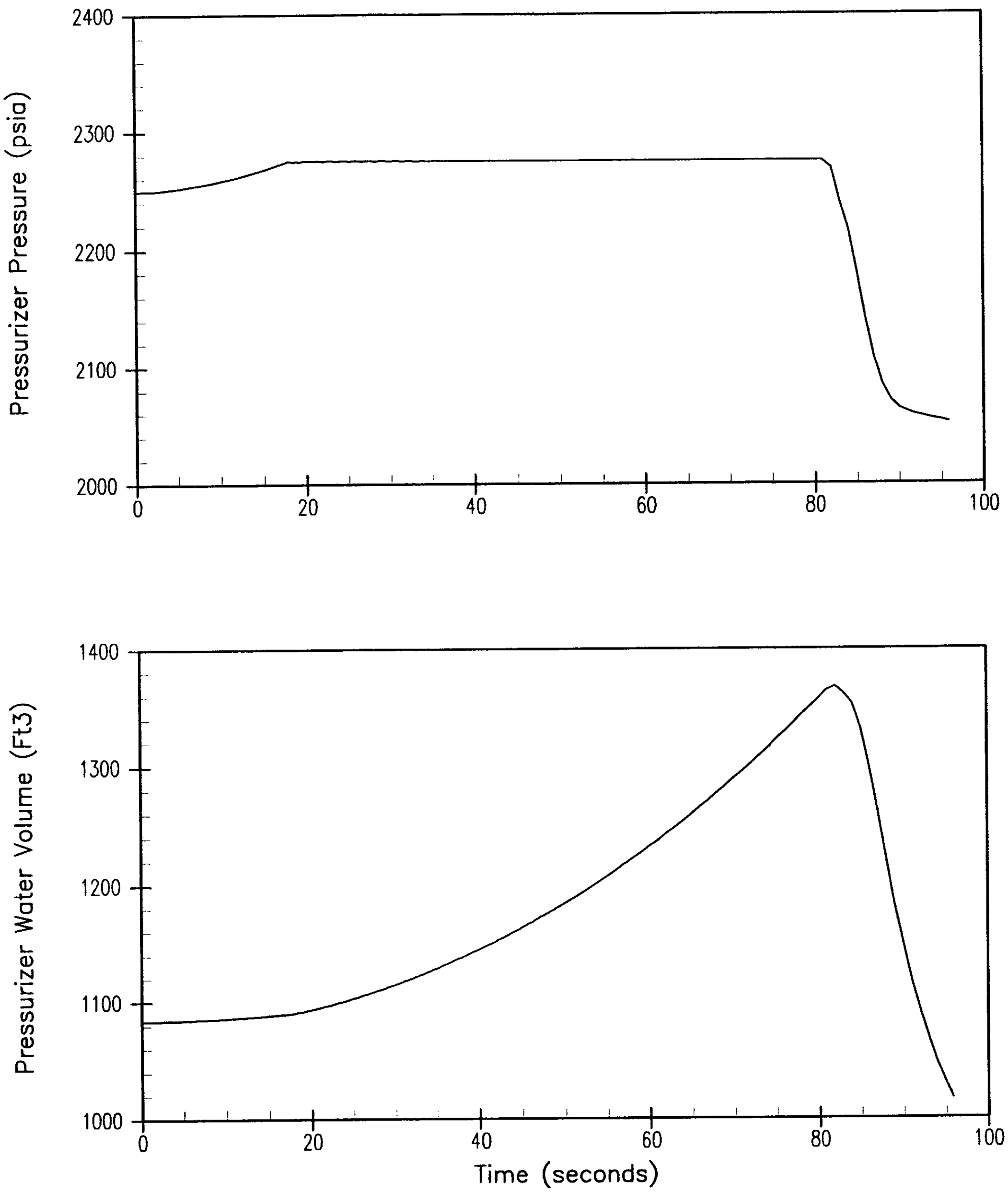


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature and DNBR Transients for an Uncontrolled RCCA Bank Withdrawal of 75 pcm/sec at 100% Power with Minimum Feedback	
		Figure 15.4-4 Sh. 3 of 3

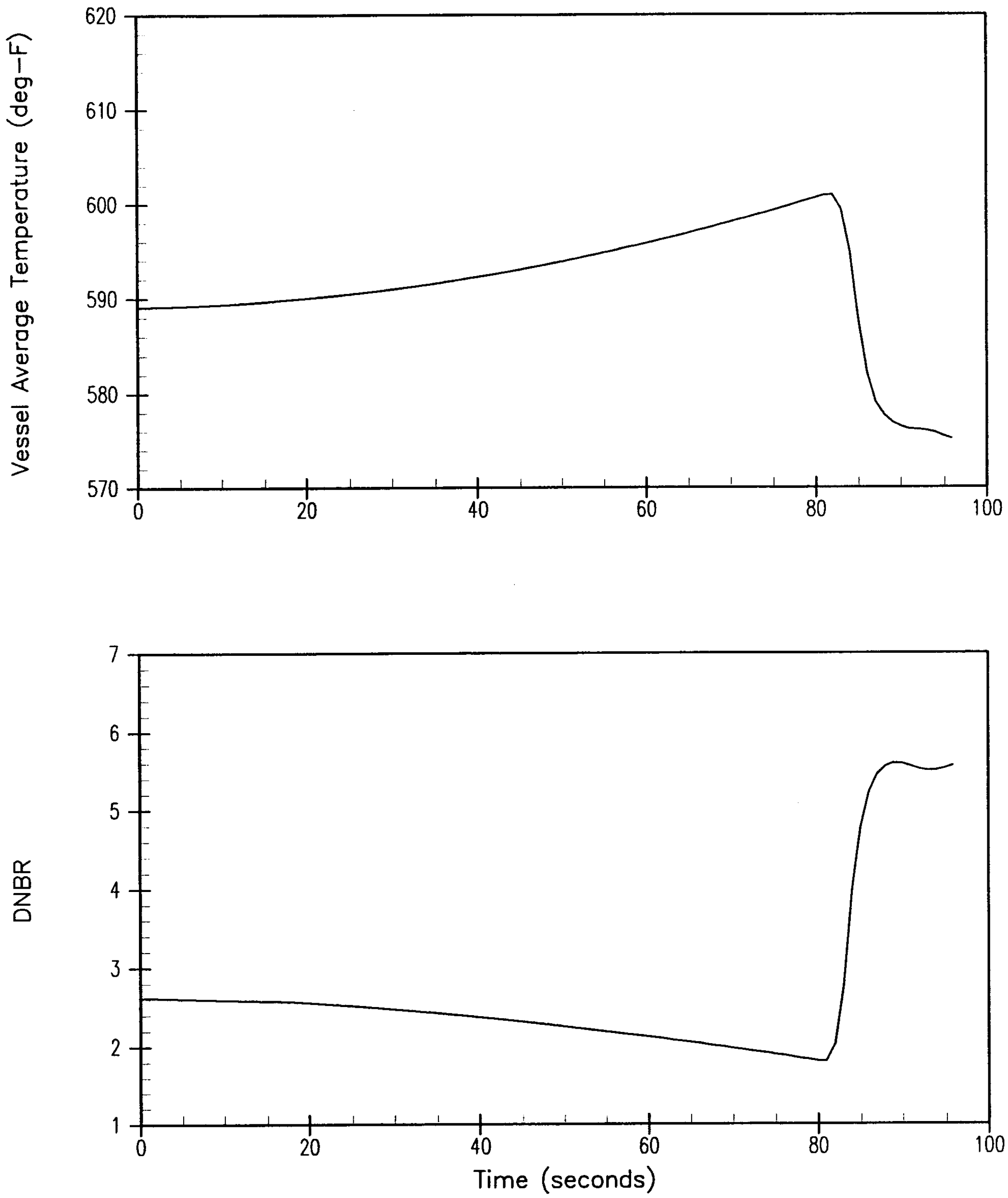




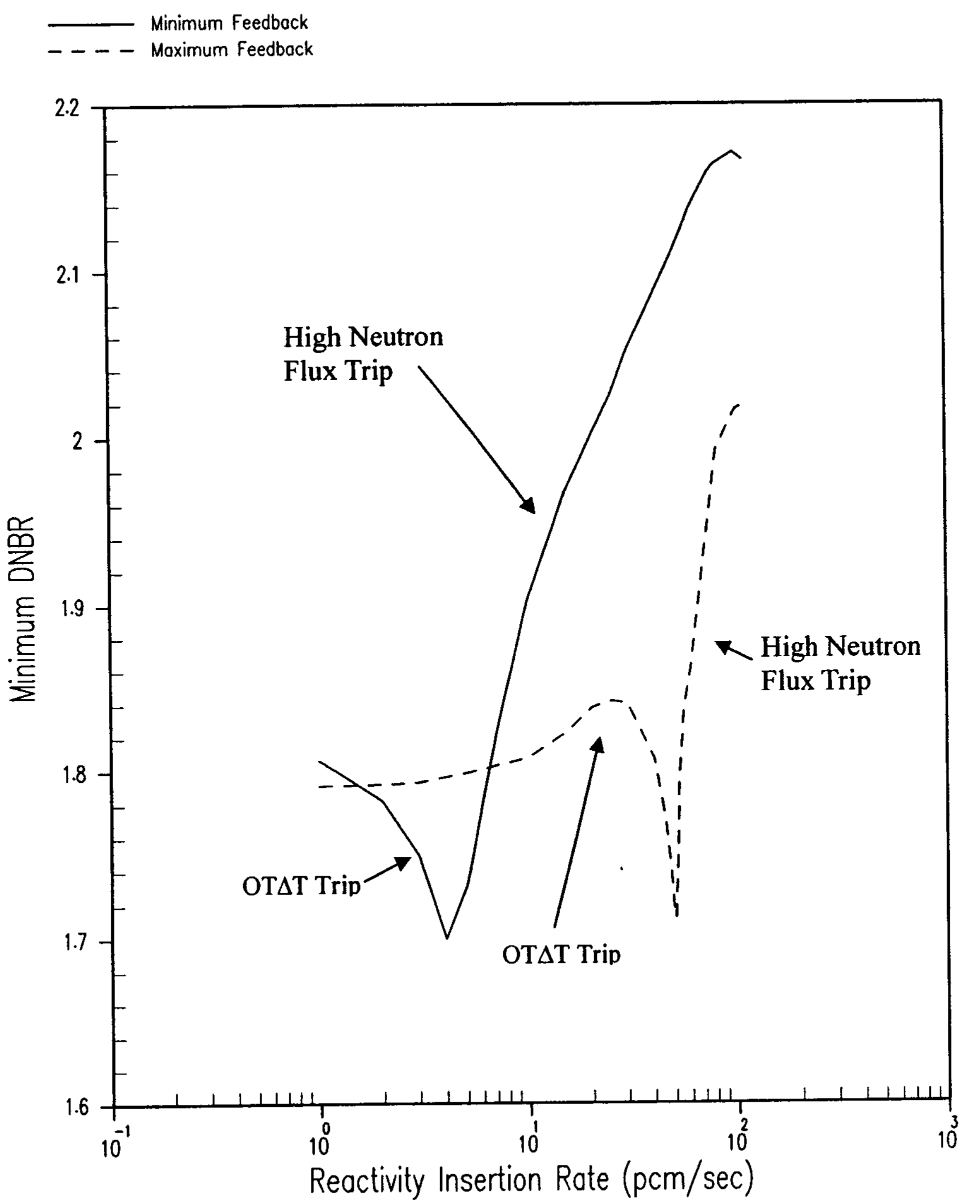
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Heat Flux Transients for an Uncontrolled RCCA Bank Withdrawal of 1 pcm/sec at 100% Power with Minimum Feedback	
		Figure 15.4-5 Sh. 1 of 3



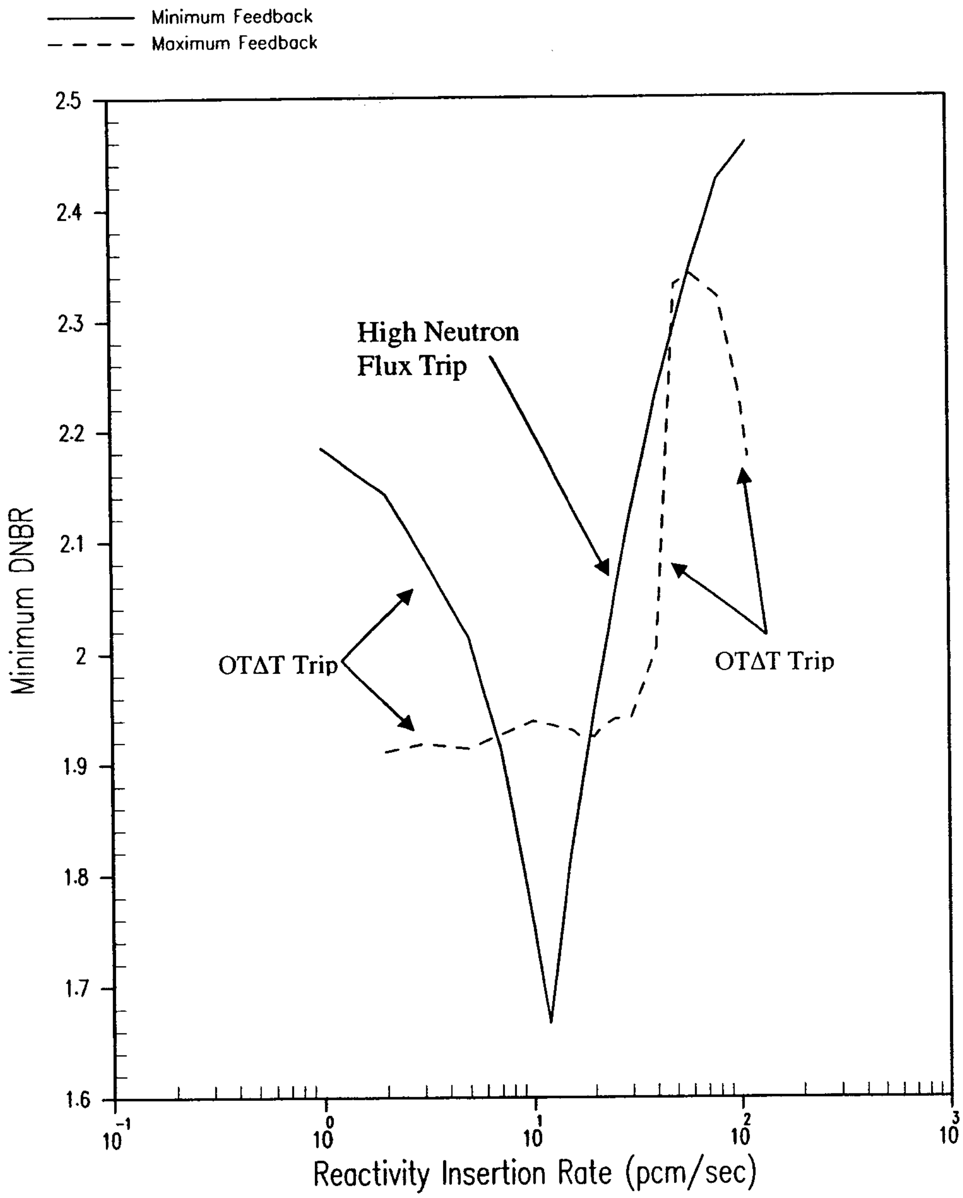
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for an Uncontrolled RCCA Bank Withdrawal of 1 pcm/sec at 100% Power with Minimum Feedback	
		Figure 15.4-5 Sh. 2 of 3



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Temperature and DNBR Transients for an Uncontrolled RCCA Bank Withdrawal of 1 pcm/sec at 100% Power with Minimum Feedback	
		Figure 15.4-5 Sh. 3 of 3



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Minimum DNBR vs. Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal at 100% Power	
		Figure 15.4-6



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Minimum DNBR vs. Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal at 60% Power	
		Figure 15.4-7

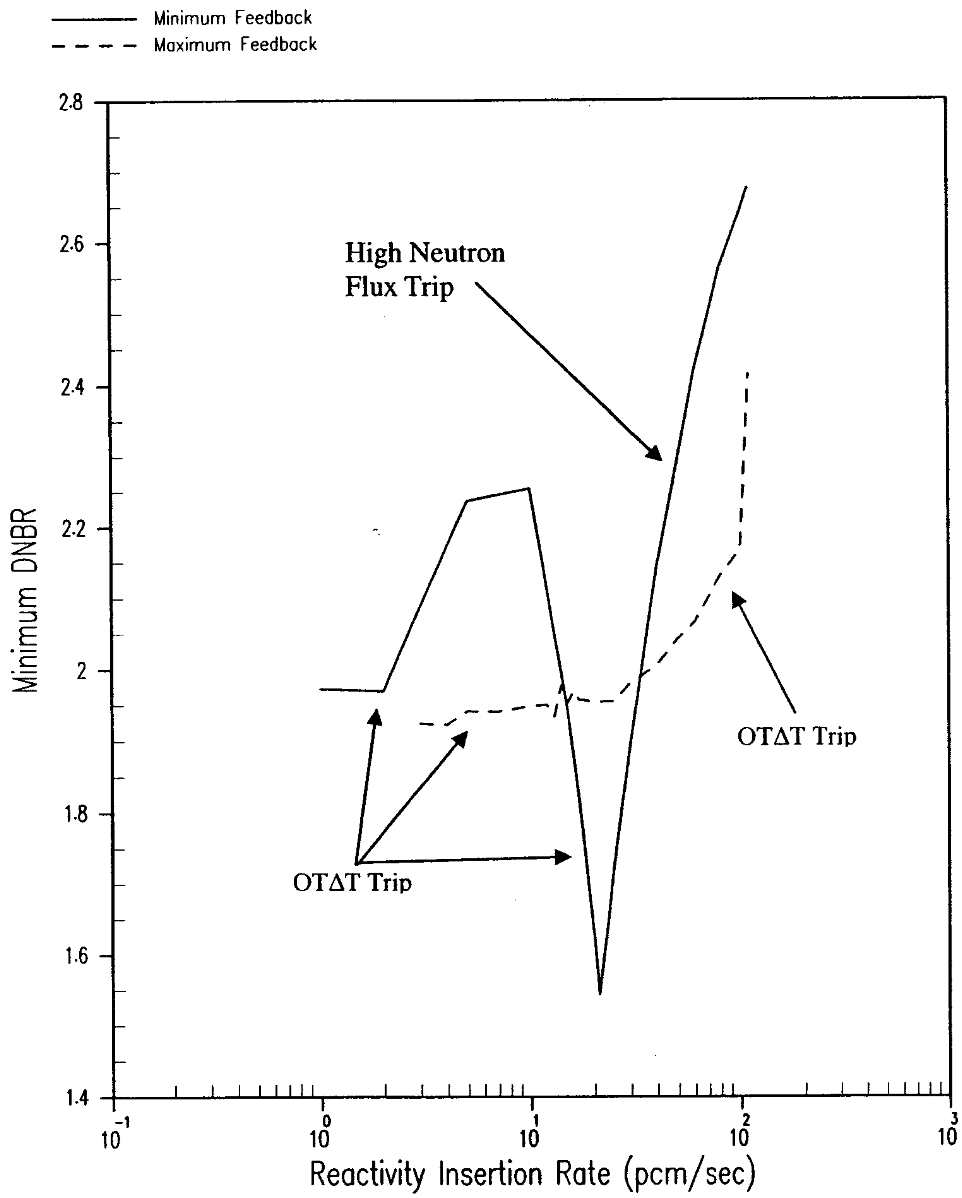


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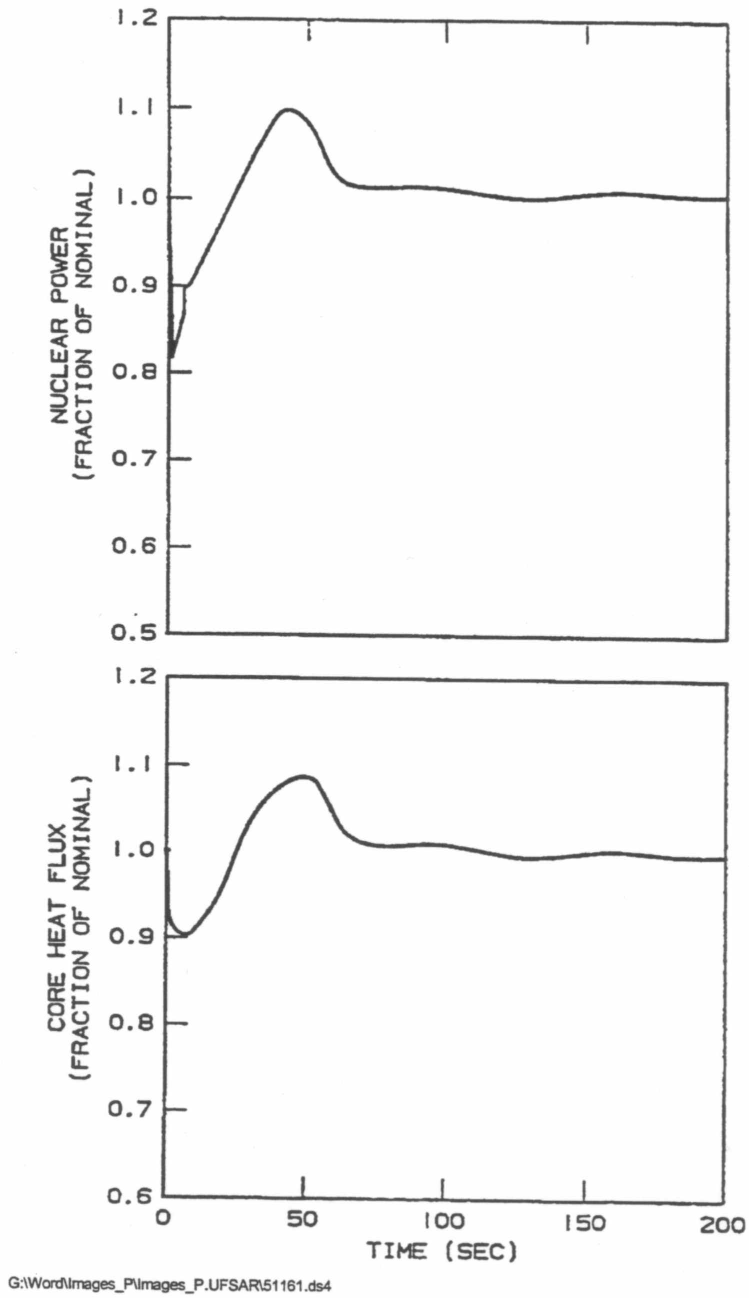
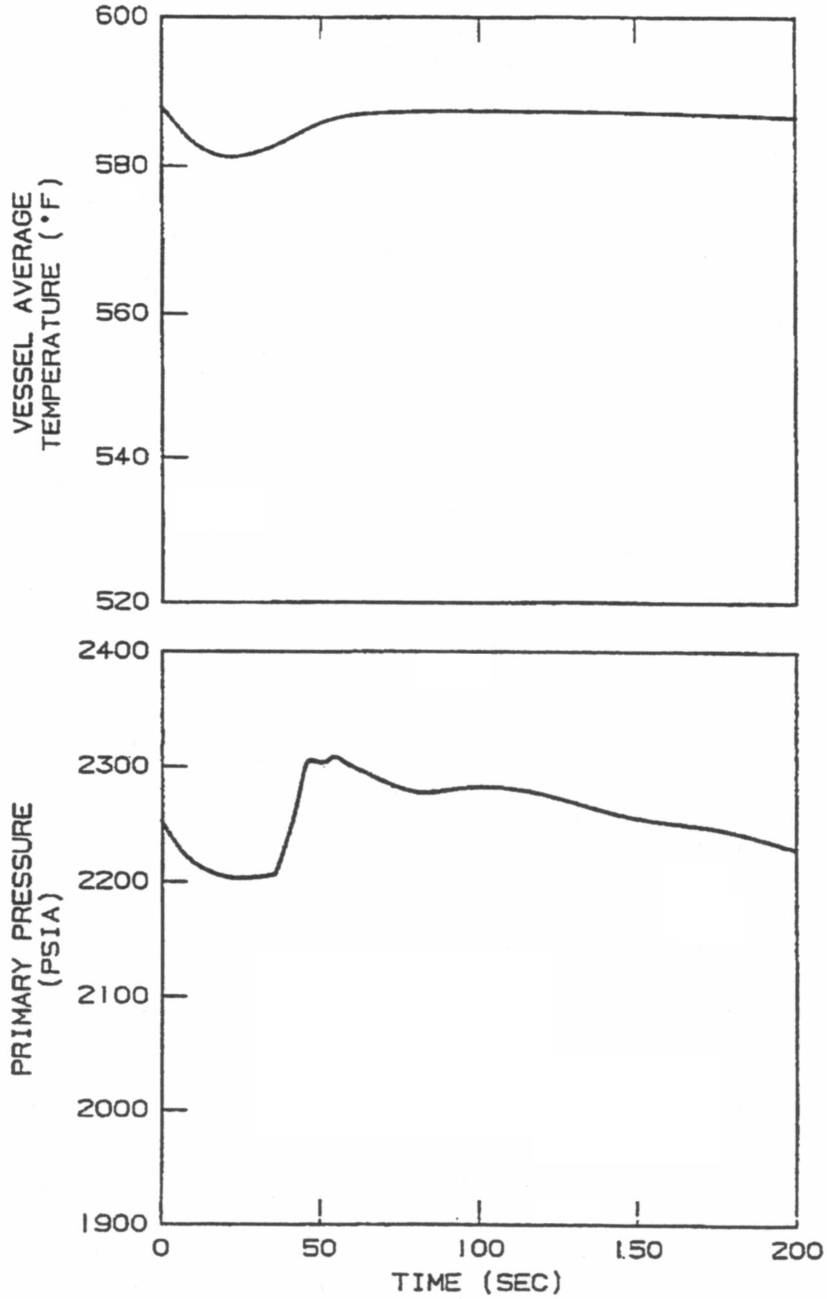
		Figure 15.4-8



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Heat Flux Transients for a Dropped RCCA	
		Figure 15-4-9 Sh. 1 of 2



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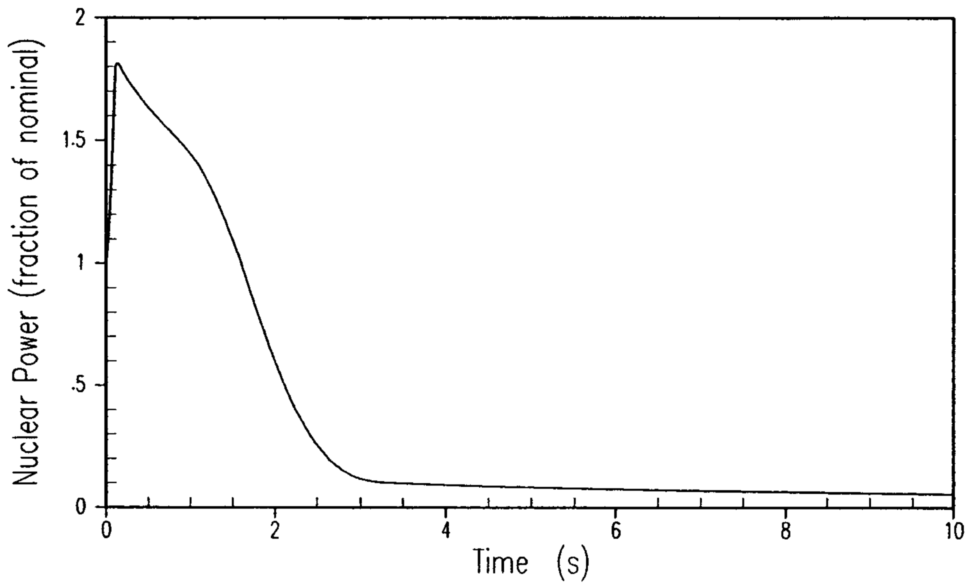
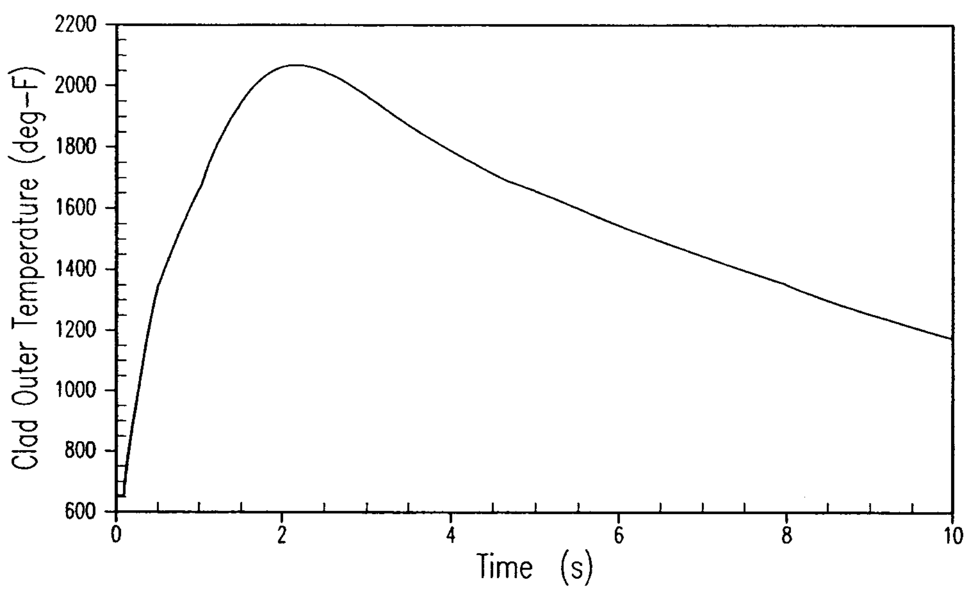
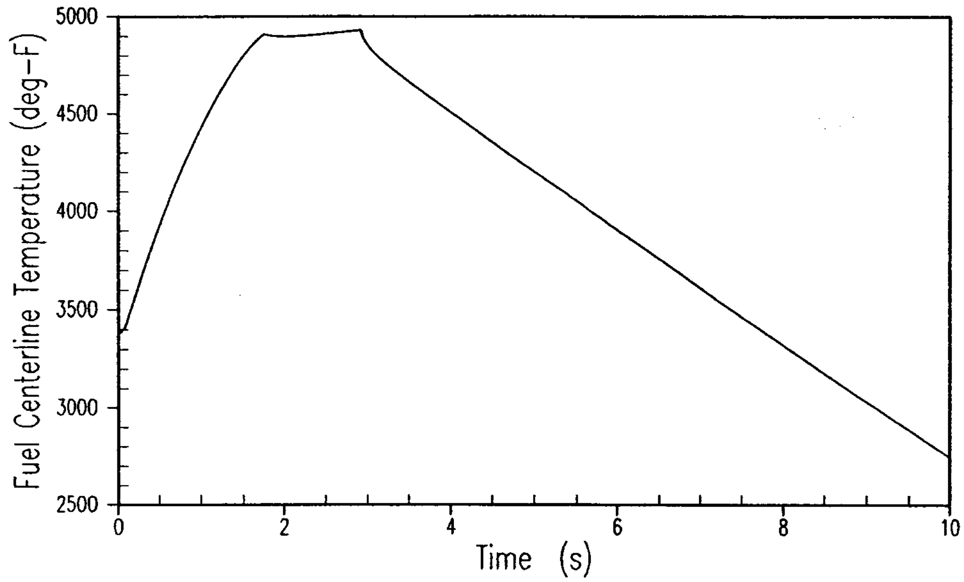
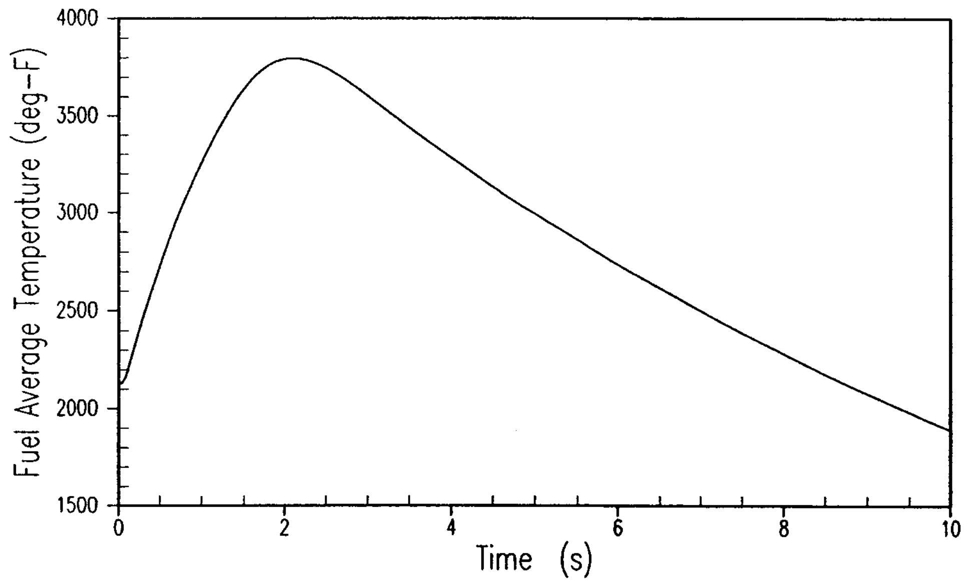
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Average Coolant Temperature and Pressurizer Pressure for a Dropped RCCA	
		Figure 15-4-9 Sh. 2 of 2



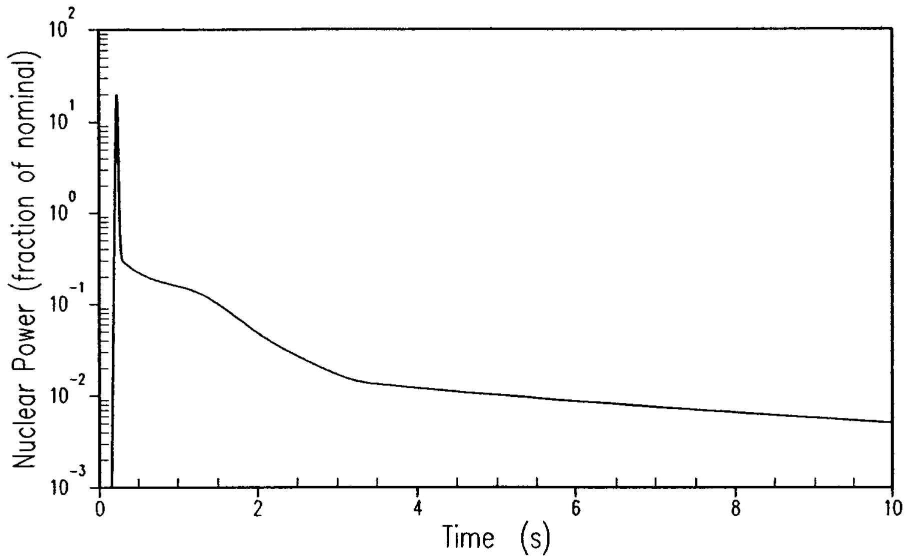


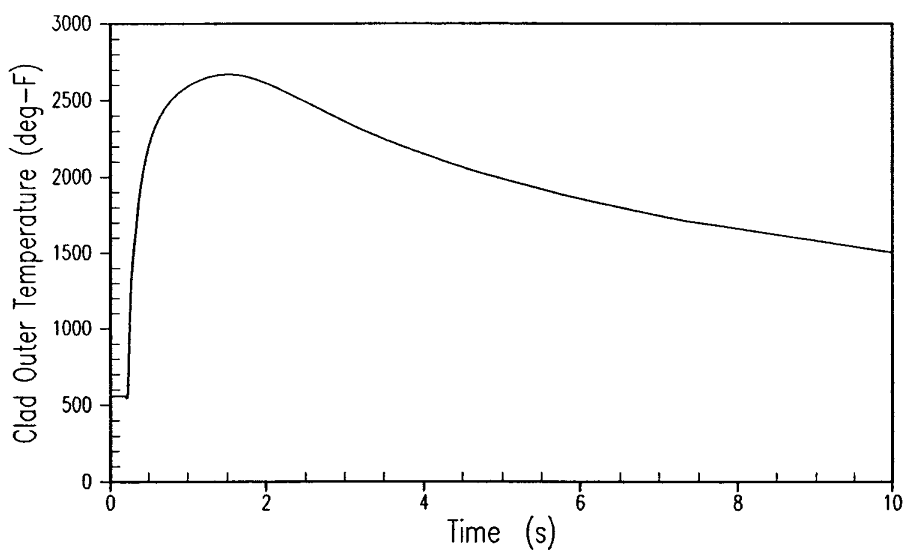
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Clad Outer Temperature Transients for an Uncontrolled RCCA Ejection at BOL HFP	
		Figure 15.4-10 Sh. 1 of 2



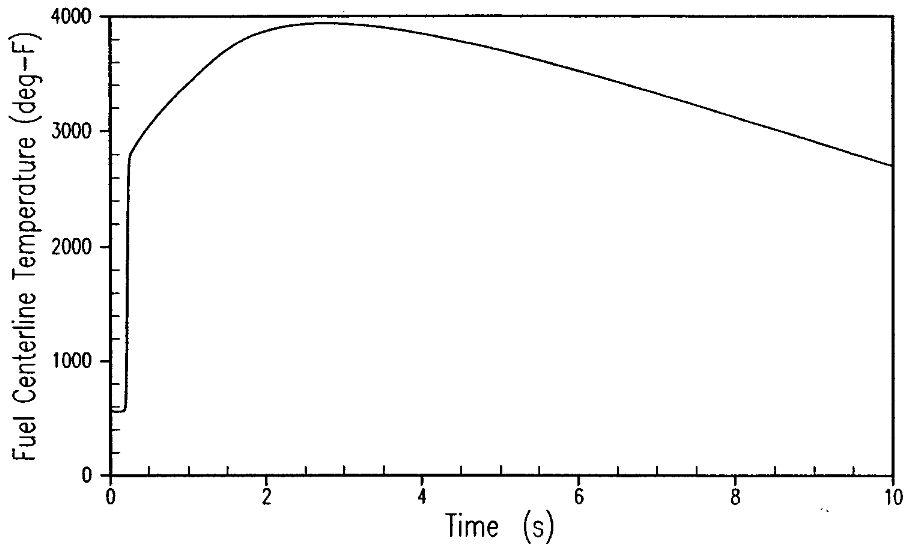


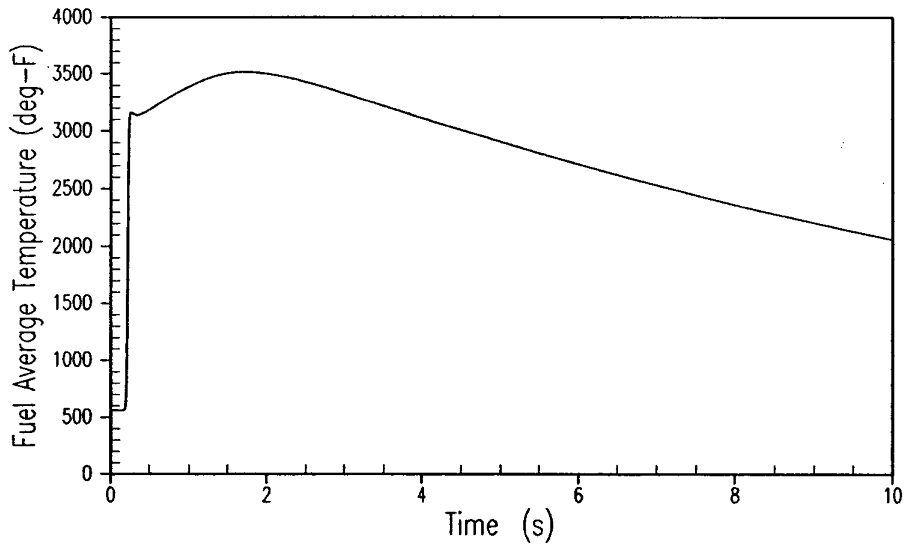
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline Temperature and Fuel Average Temperature Transients for an Uncontrolled RCCA Ejection at BOL HFP	
		Figure 15.4-10 Sh. 2 of 2





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Clad Outer Temperature Transients for an Uncontrolled RCCA Ejection at EOL HZP
	Figure 15.4-11 Sh. 1 of 2





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Fuel Centerline and Fuel Average Temperature Transients for an Uncontrolled RCCA Ejection at COL HZP	
		Figure 15.4-11 Sh. 2 of 2

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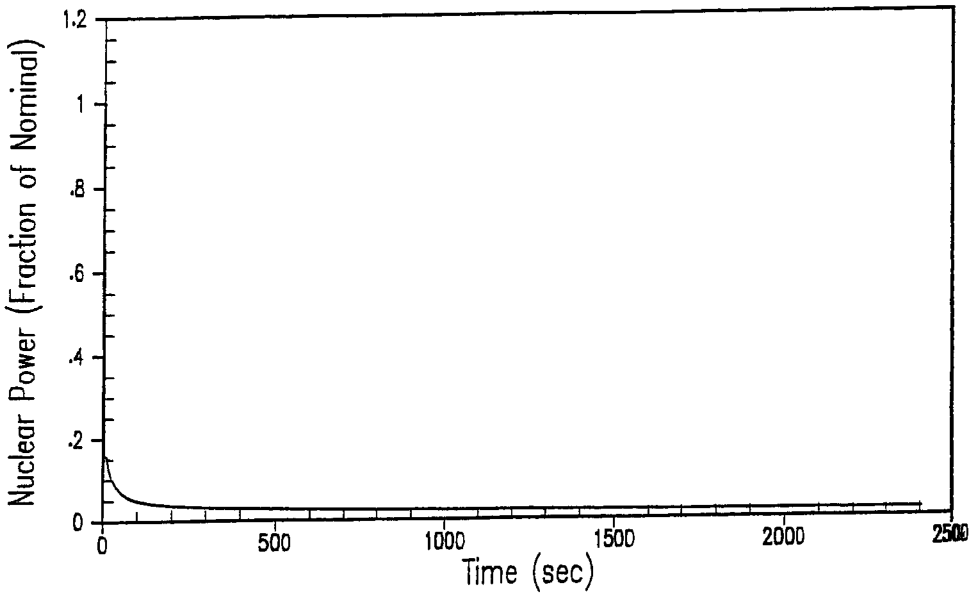

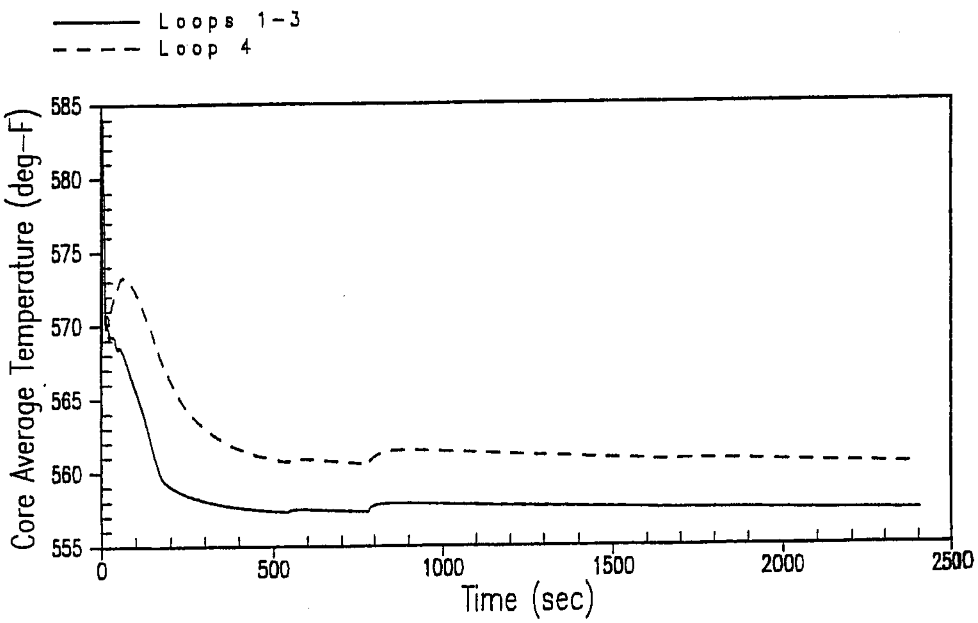

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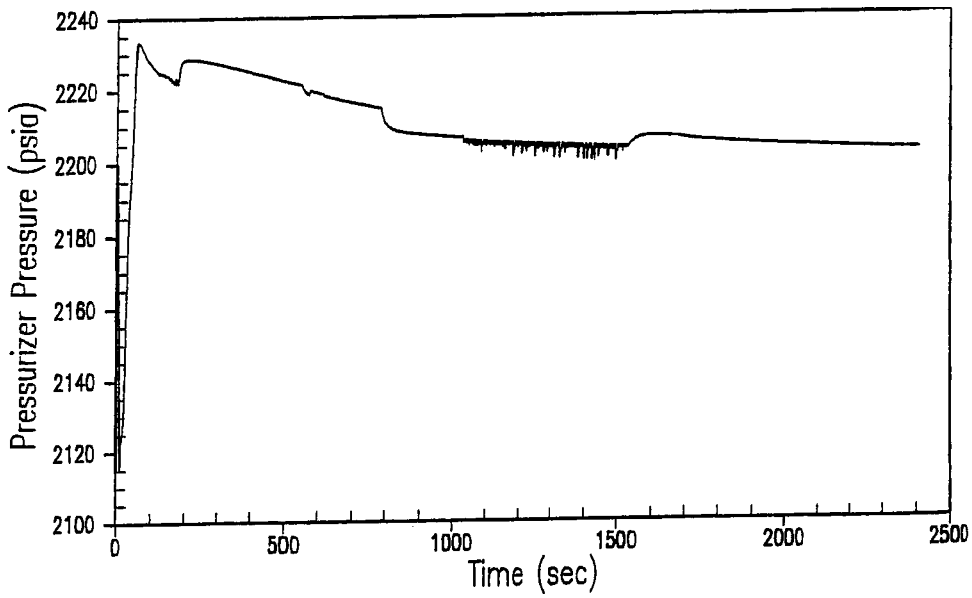
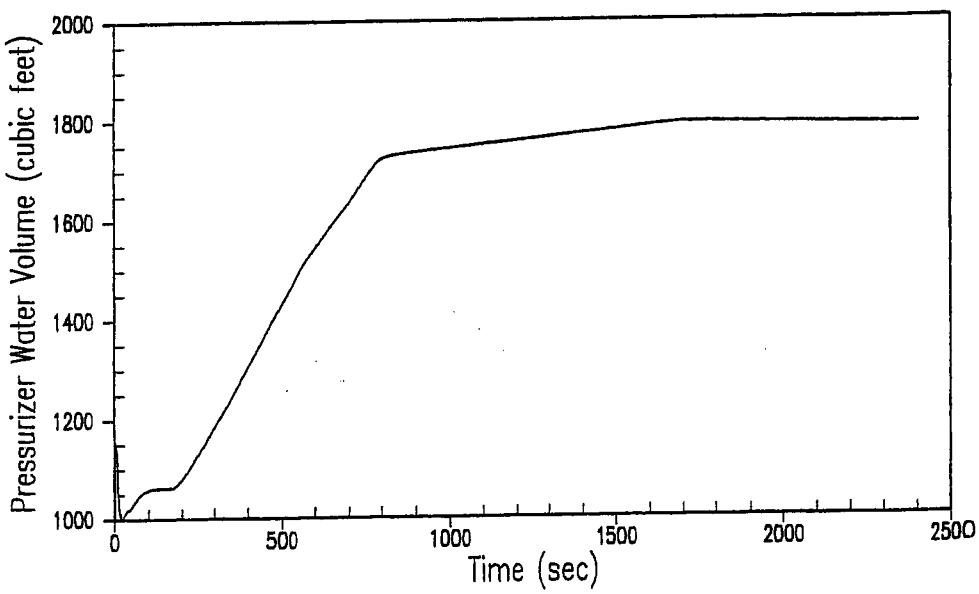
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Vessel Average Temperature Transients for an Inadvertent ECCS Actuation During Power Operation	
		Figure 15.5-1 Sh. 1 of 4





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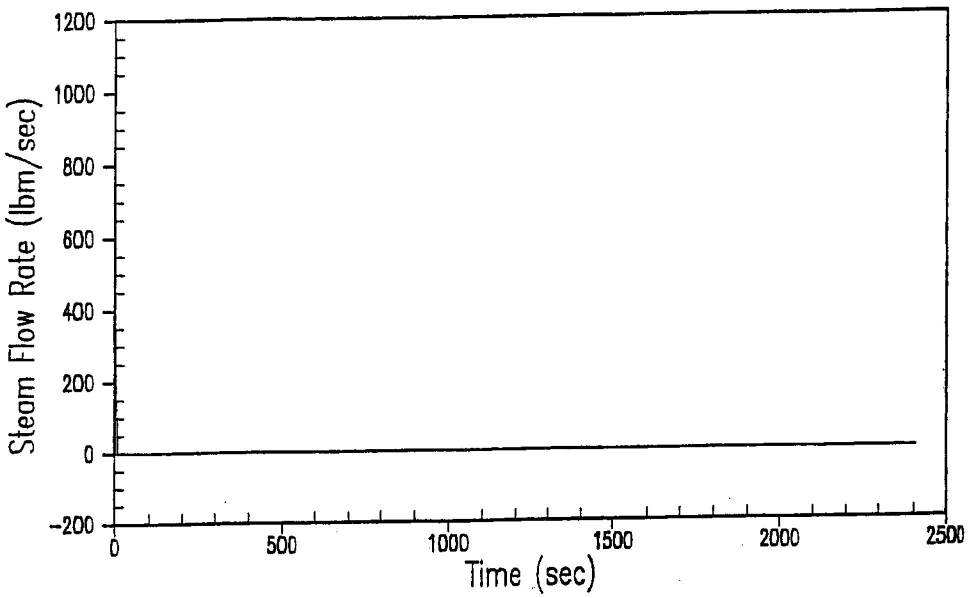
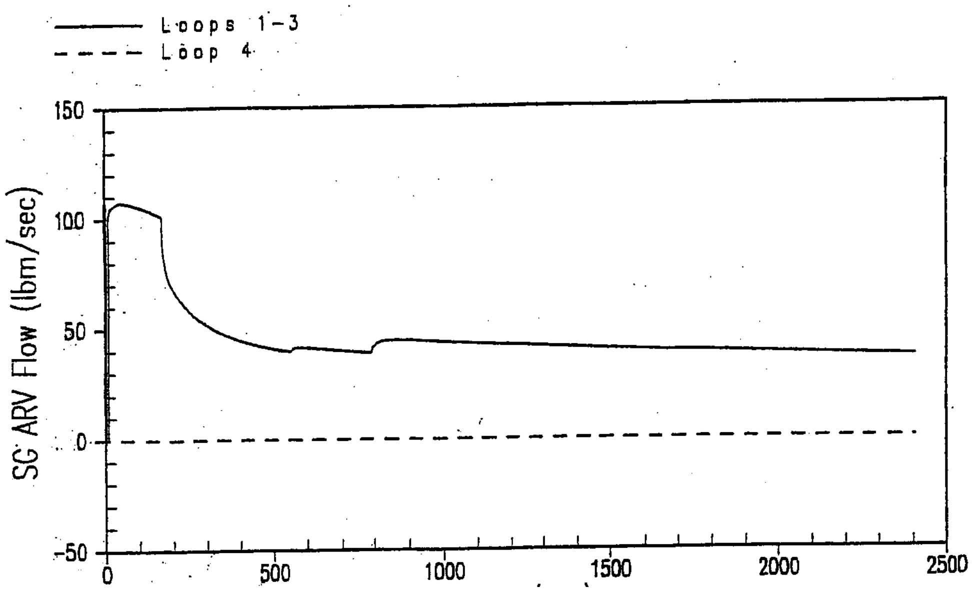
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Pressurizer Water Volume Transients for an Inadvertent ECCS Actuation During Power Operation
	Figure 15.5-1 Sh. 2 of 4





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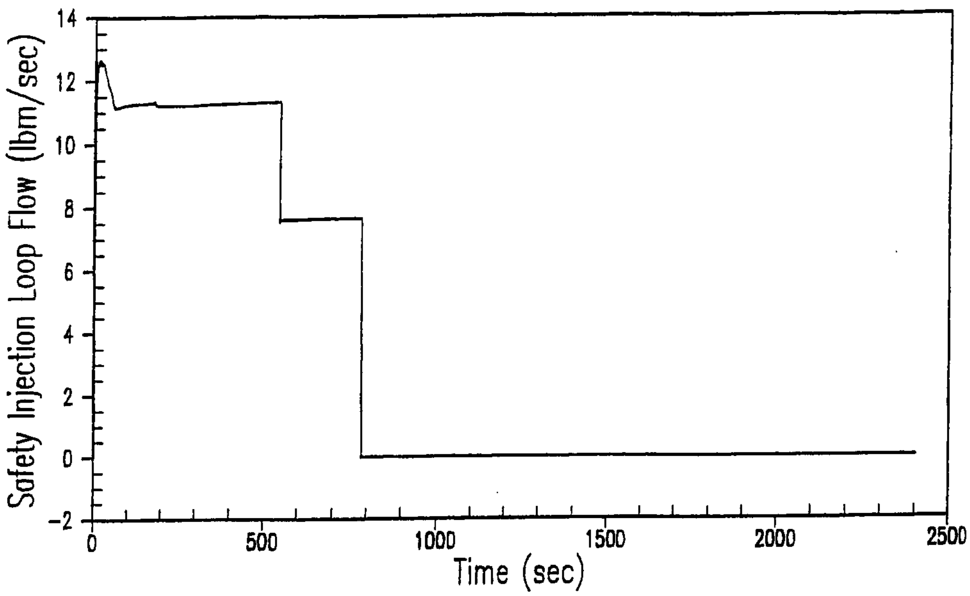
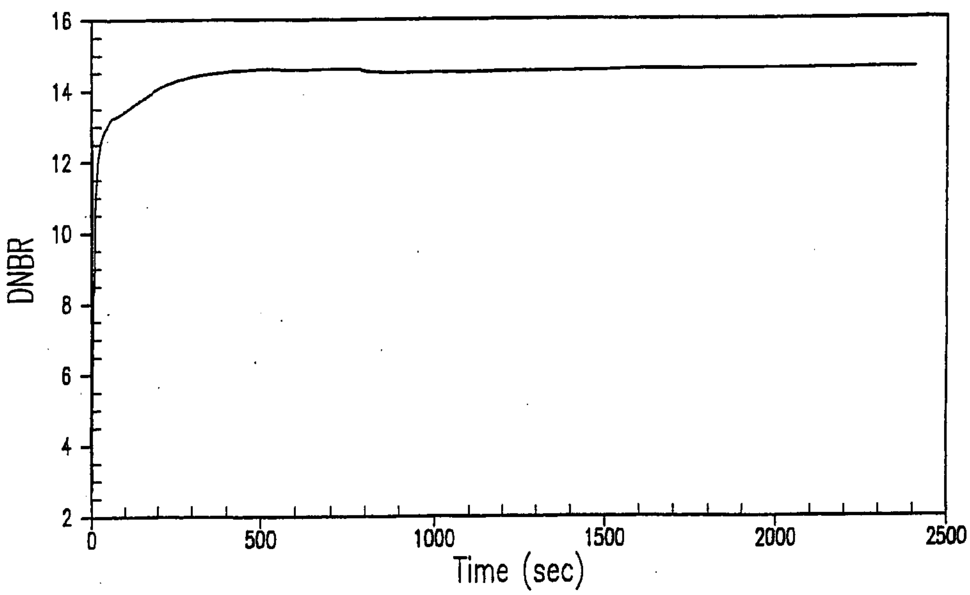
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Main Steam Flow and Steam Generator Atmospheric Relief Valve Flow Transients for an Inadvertent ECCS Actuation During Power Operation	
		Figure 15.5-1 Sh. 3 of 4





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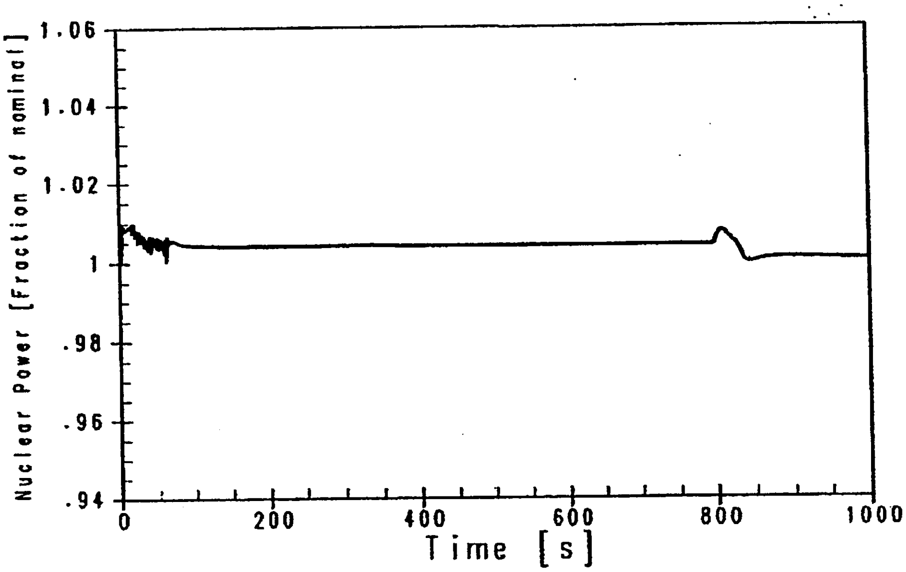
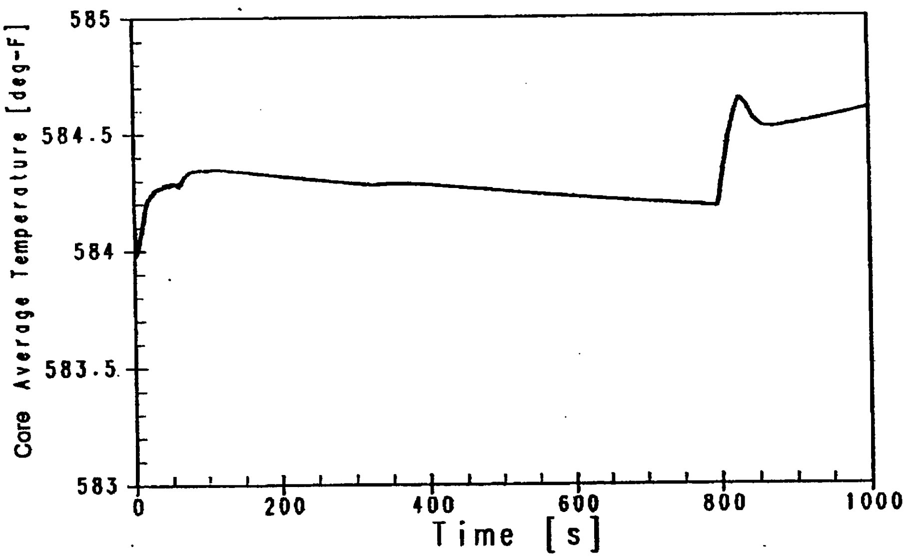
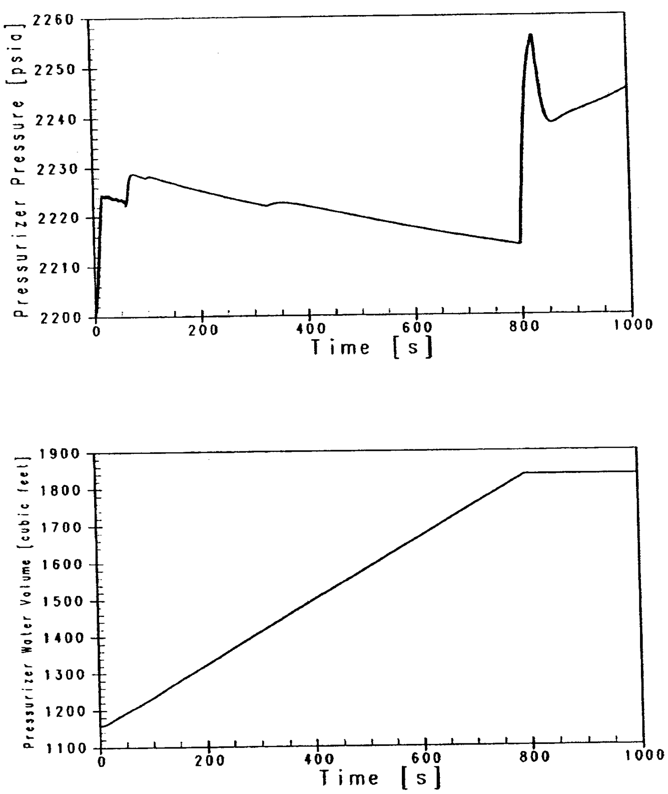
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Safety Injection Flow and DNBR Transient for an Inadvertent ECCS Actuation During Power Operation	
		Figure 15.5-1 Sh. 4 of 4





SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and Core Average Temperature Transients for a CVCS Malfunction	
		Figure 15.5-2 Sh. 1 of 2



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurized Pressure and Pressurizer Water Volume Transients for a CVCS Malfunction	
		Figure 15.5-2 Sh. 2 of 2

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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT		
		Figure 15.5-3

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FIGURE 15.5-5

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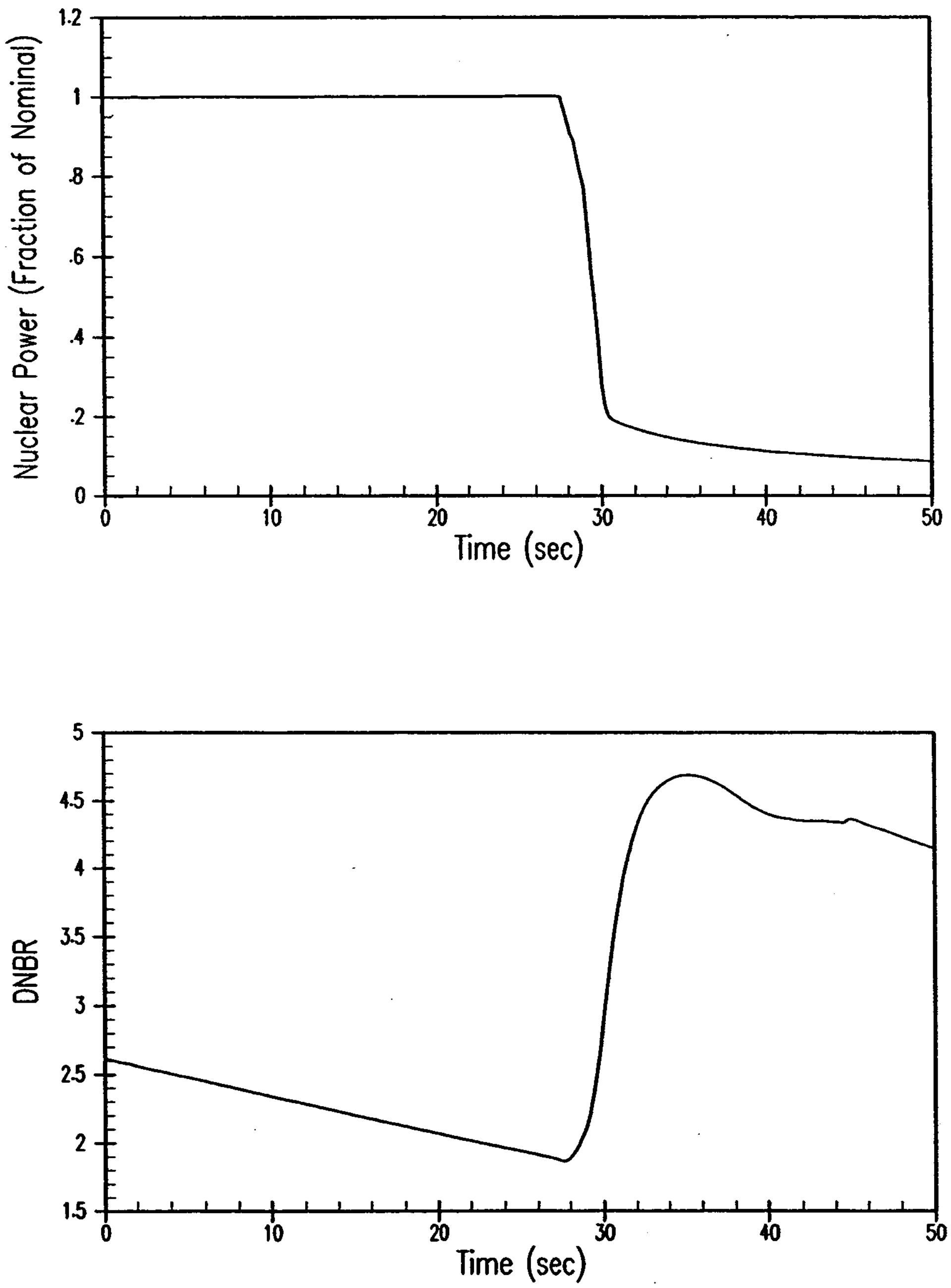
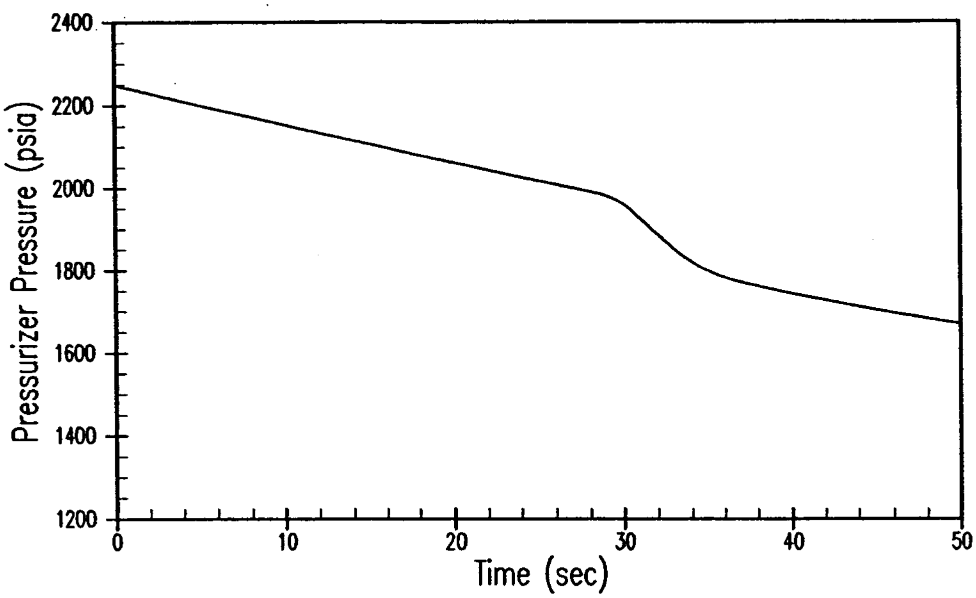
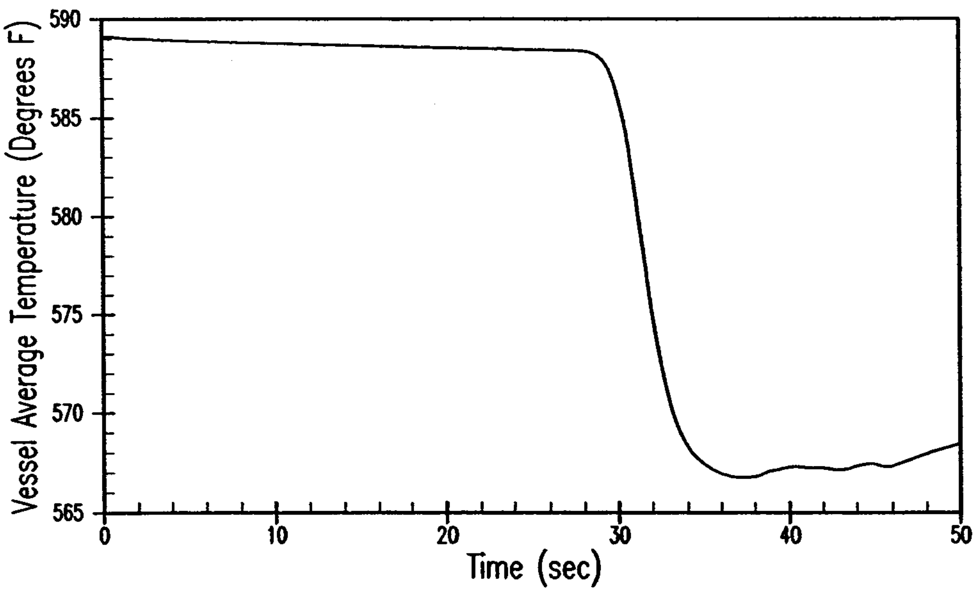
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	DELETED	
		Figure 15-5-5



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nuclear Power and DNBR Transients for an Inadvertent Opening of a Pressurizer Safety Valve	
		Figure 15.6-1

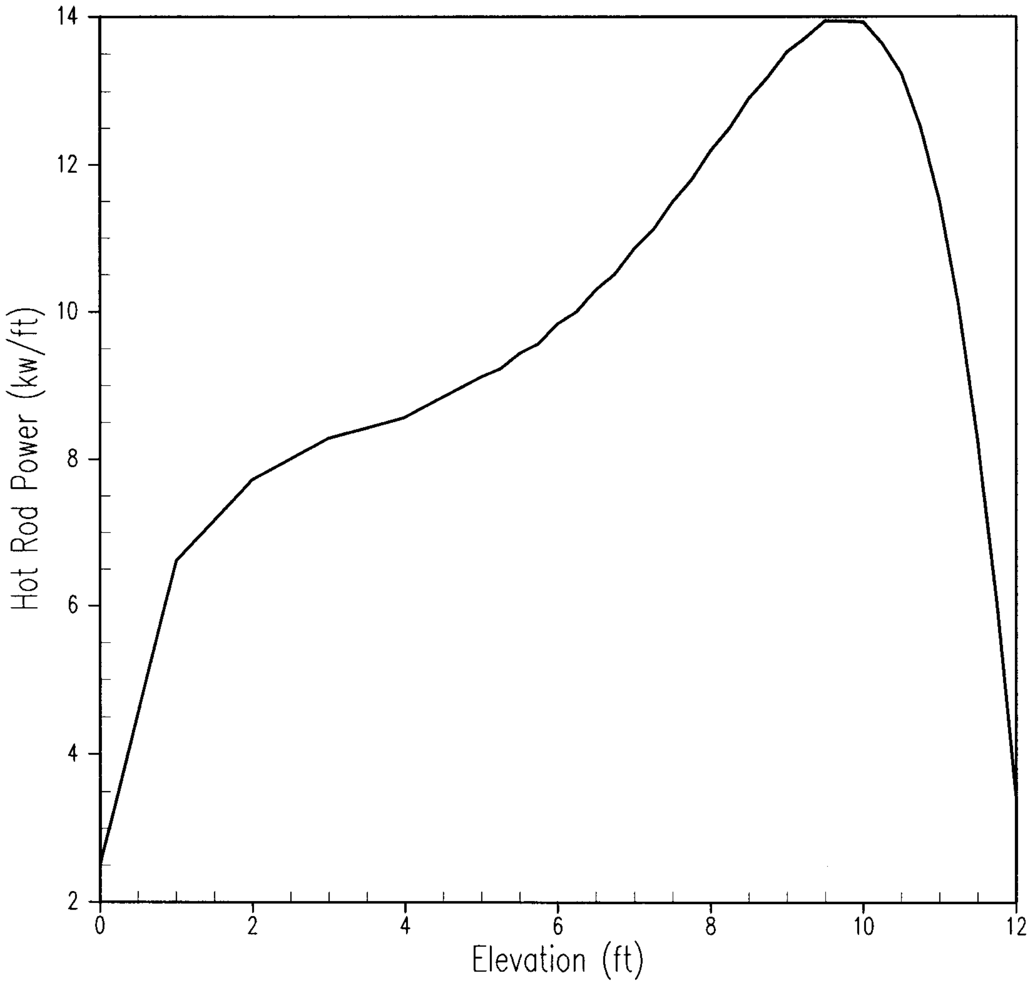




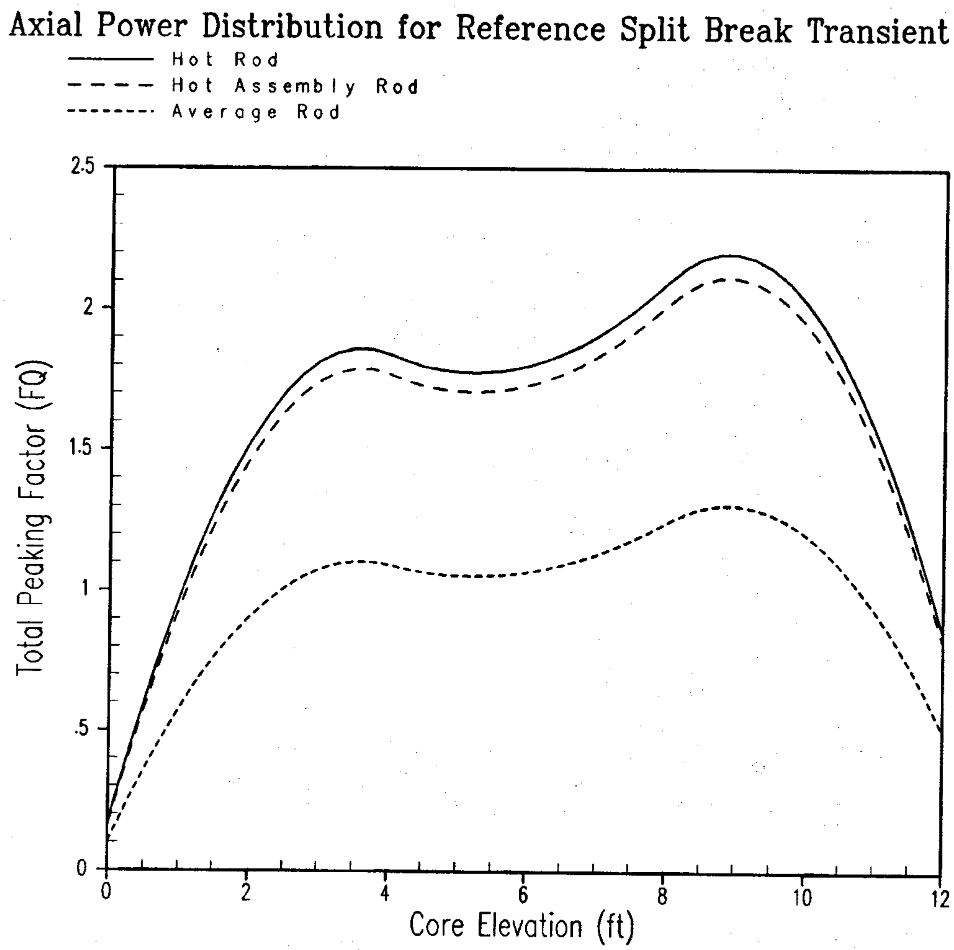
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure and Vessel Average Temperature Transients for an Inadvertent Opening of a Pressurizer Safety Valve	
		Figure 15.6-2

BLOWDOWN	0 second	BREAK OCCURS
		PUMP SI SIGNAL (LOW PRESSURIZER PRESSURE)
		ACCUMULATOR INJECTION BEGINS
	~28 seconds	END OF BLOWDOWN
REFILL	~28 – 33 seconds	BOTTOM OF CORE RECOVERY
REFLOOD		PUMPED ECCS INJECTION BEGINS
		CONTAINMENT SPRAY STARTS
		ACCUMULATOR EMPTIES
	~33 - 200 seconds	CORE QUENCHED
LONG TERM CORE COOLING	24 hours	SWITCH TO COLD LEG RECIRCULATION ON RWST LOW LEVEL ALARM
		SWITCH TO HOTLEG/COLD/LEG RECIRCULATION

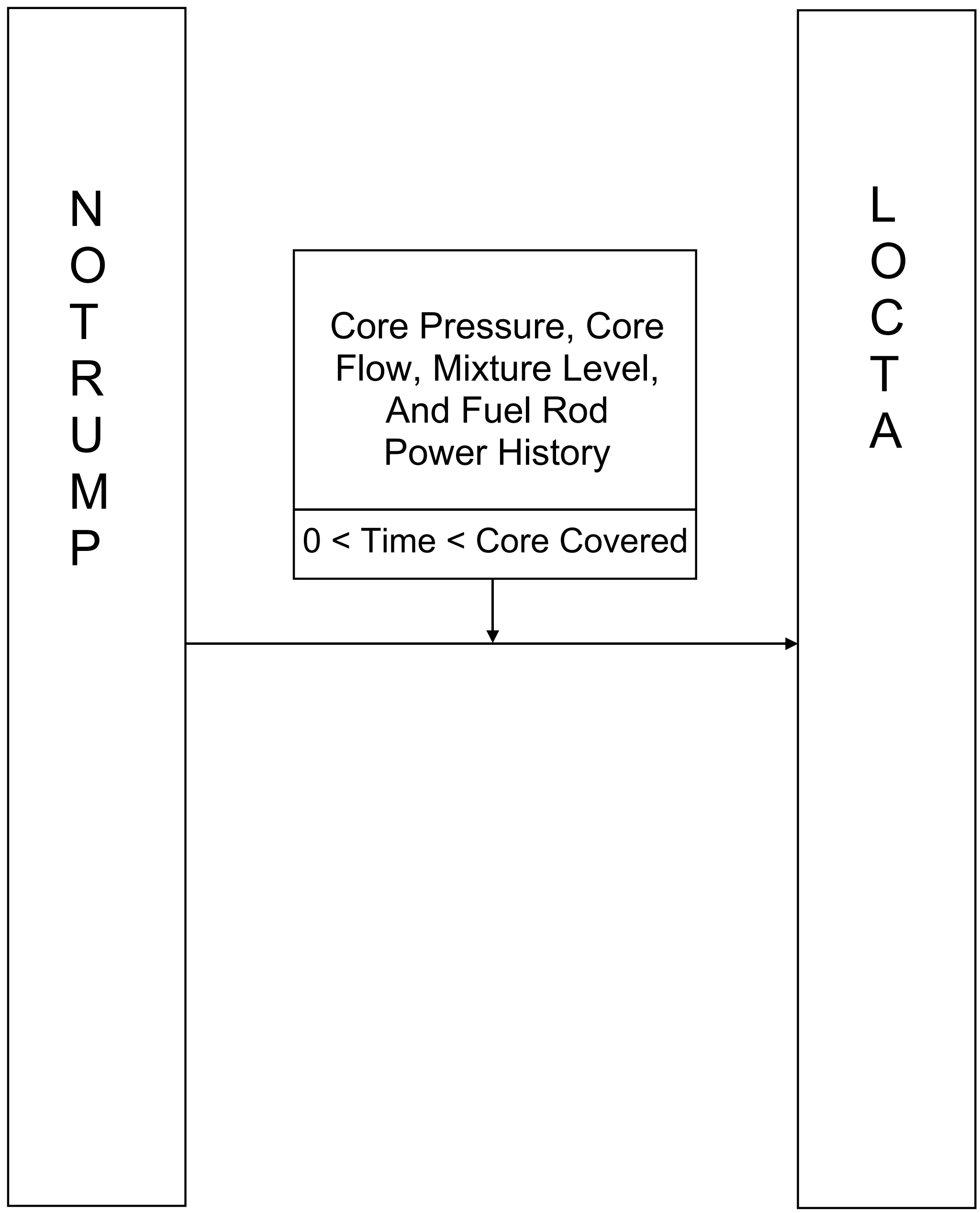
Figure 15.6-3: Typical Time Sequence of Events for the Seabrook Station Best-Estimate Large Break LOCA Analysis



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Small Break LOCA Hot Rod Power Shape	
		Figure 15.6-4



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reference Transient Axial Power Distribution for Seabrook Station BE LBLOCA	
		Figure 15.6-5



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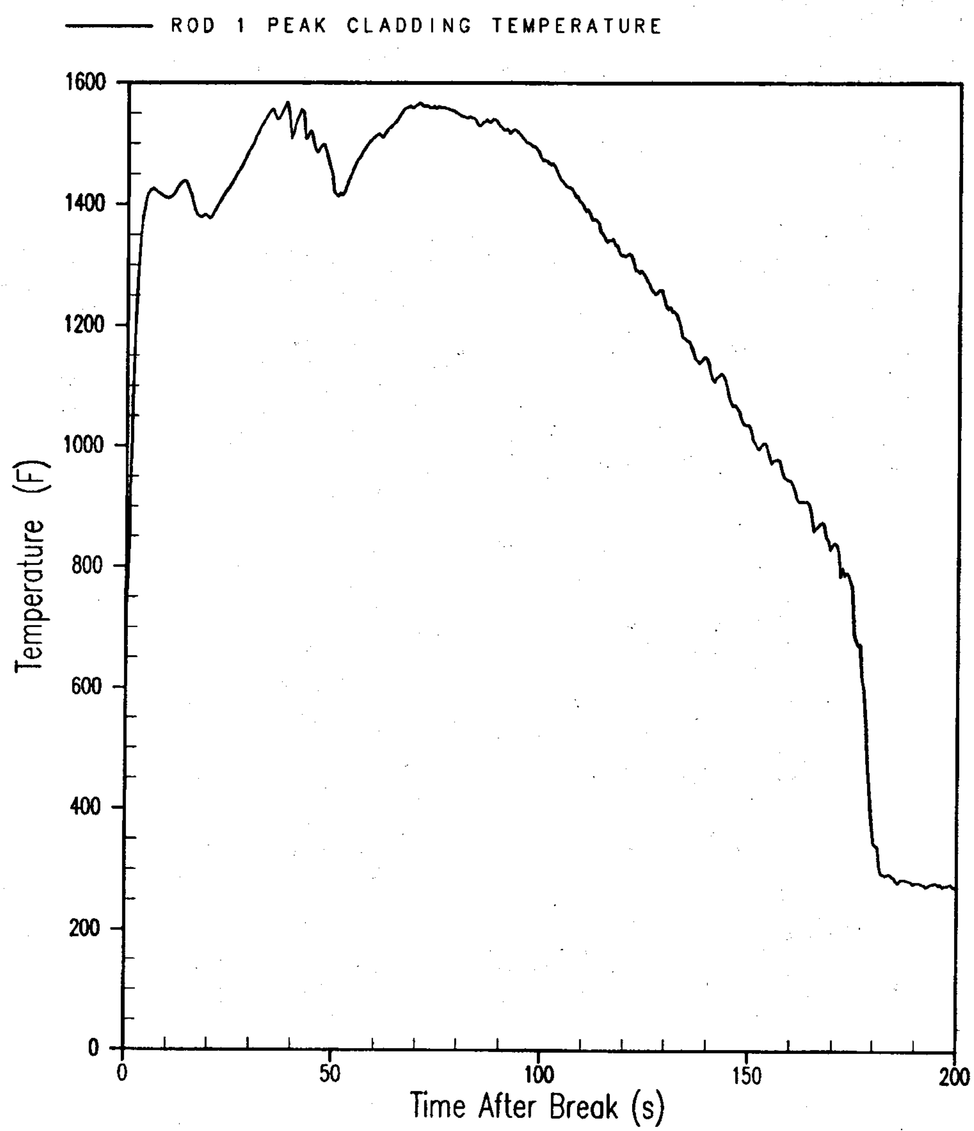
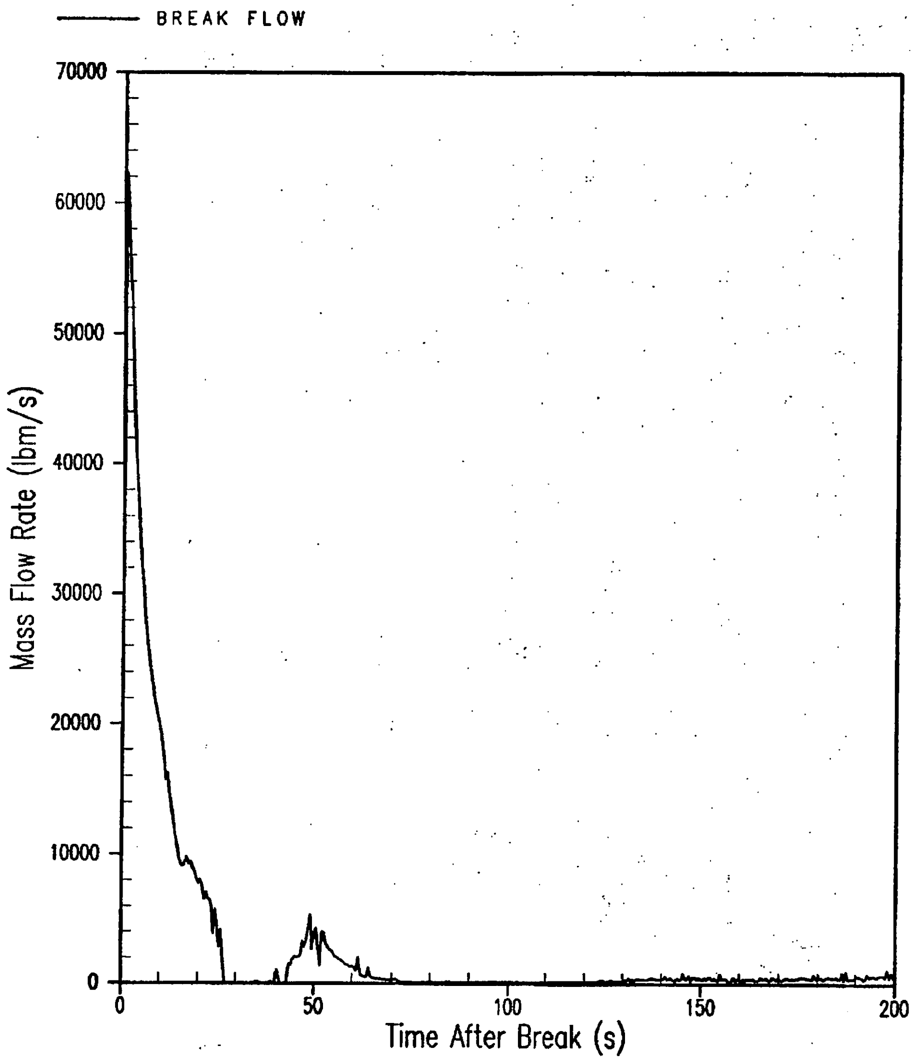
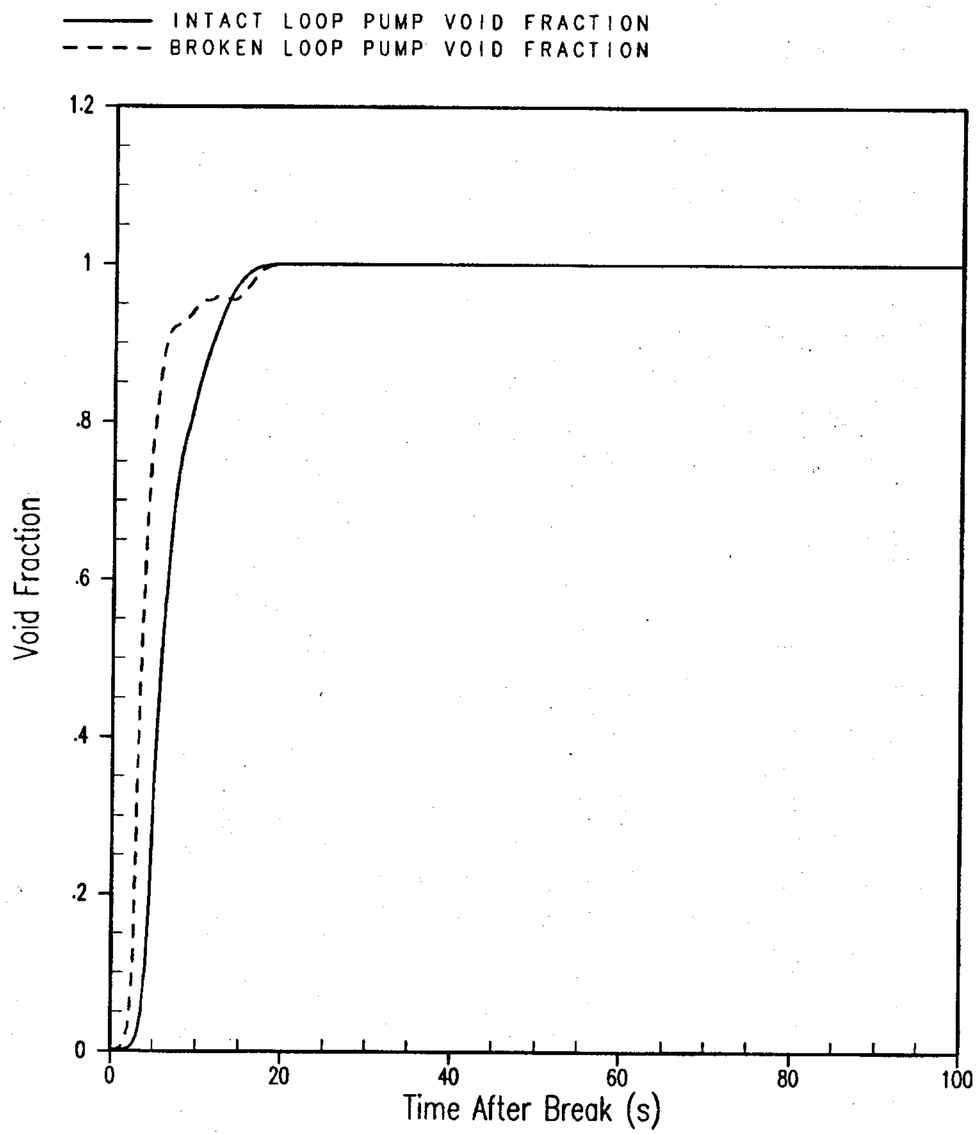
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Small Break LOCA Code Interface
	Figure 15-6-6



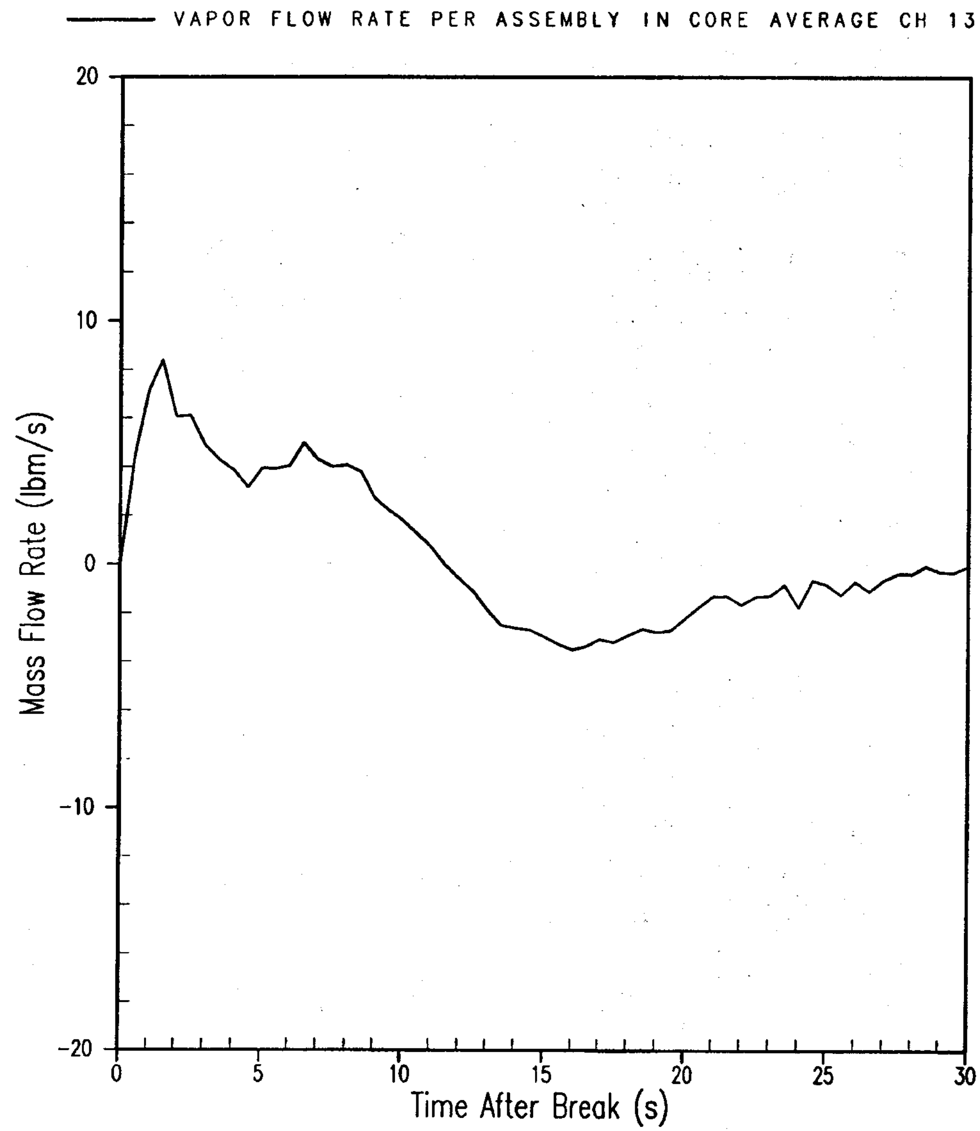
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Cladding Temperature for Reference Split Transient BE LBLOCA	
		Figure 15.6-7


BREAK FLOW
70000
60000
50000
40000
30000
20000
10000
0
Mass Flow Rate (lbm/s)
0
50
100
150
200
Time After Break (s)



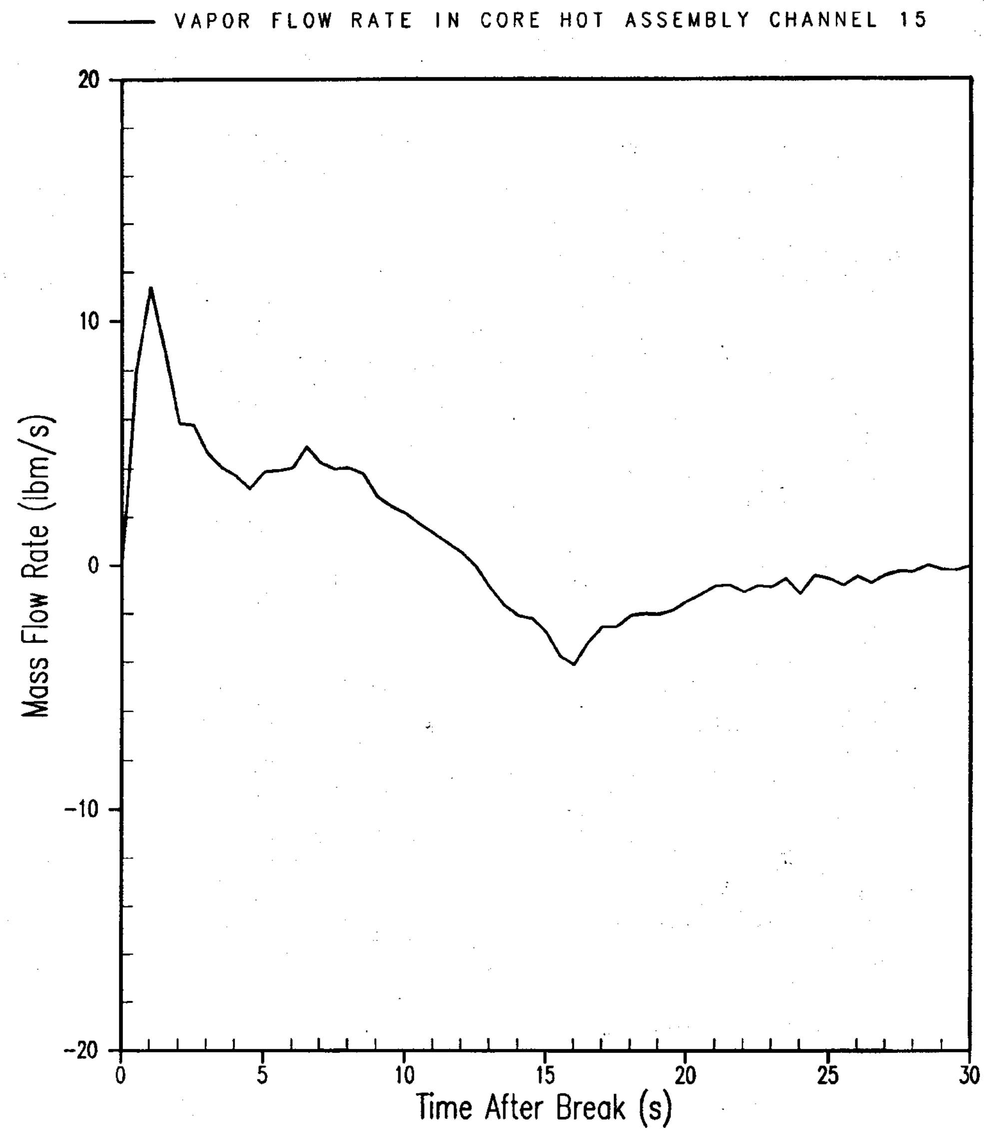
Void Fraction at the Intact and Broken Loop Pump Inlet for Reference Split Transient – BE LBLOCA

Figure 15.6-9

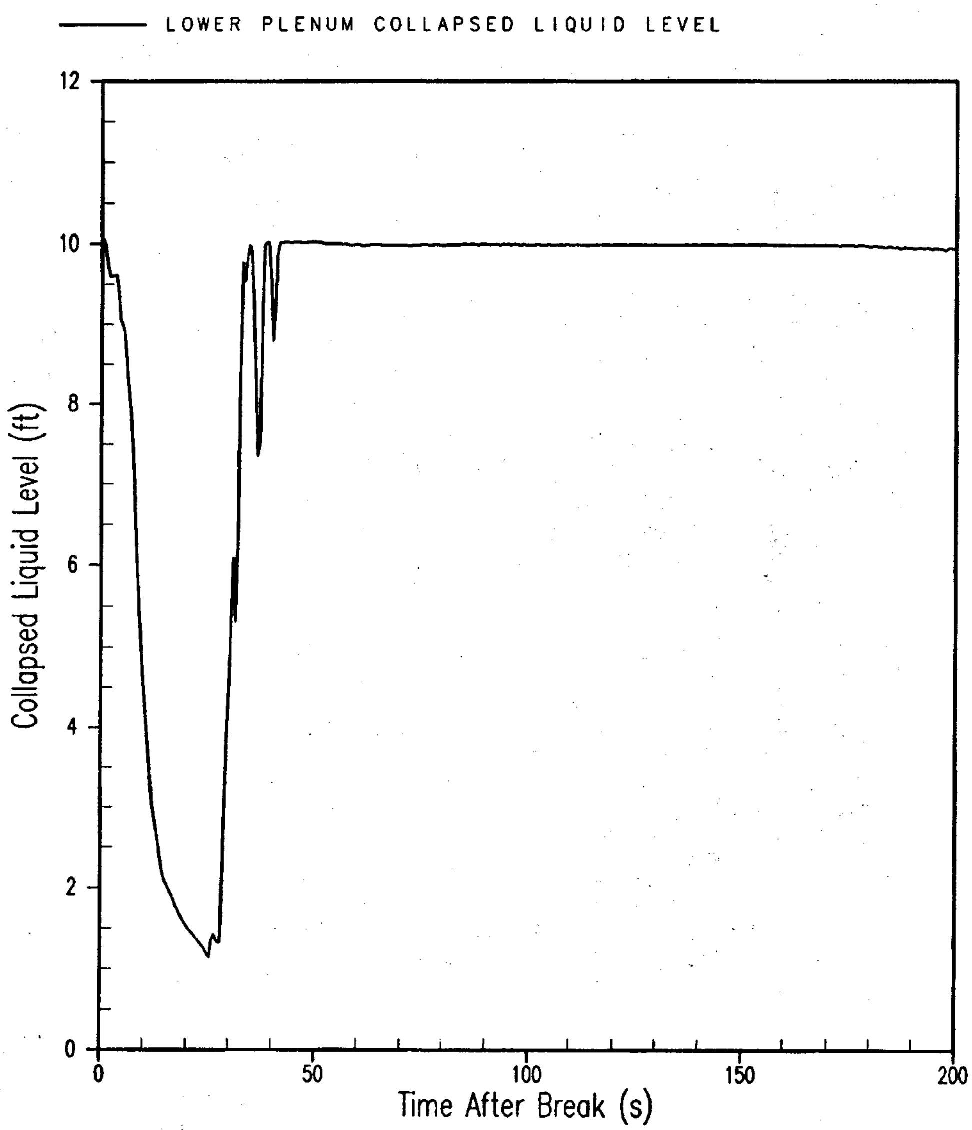


Vapor Flow Rate at Midcore in Channel 13 During Blowdown for Reference Split Transient – BE LBLOCA

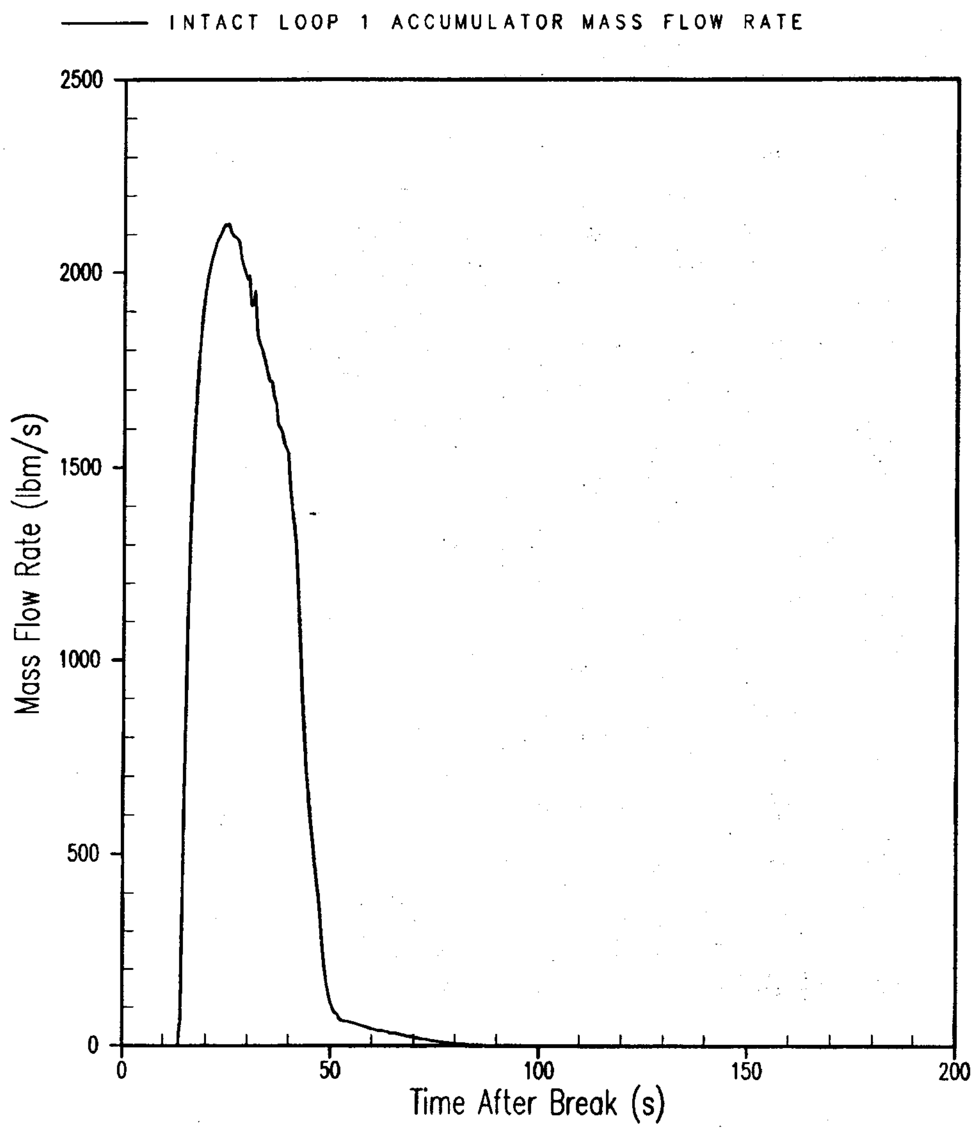
Figure 15.6-10



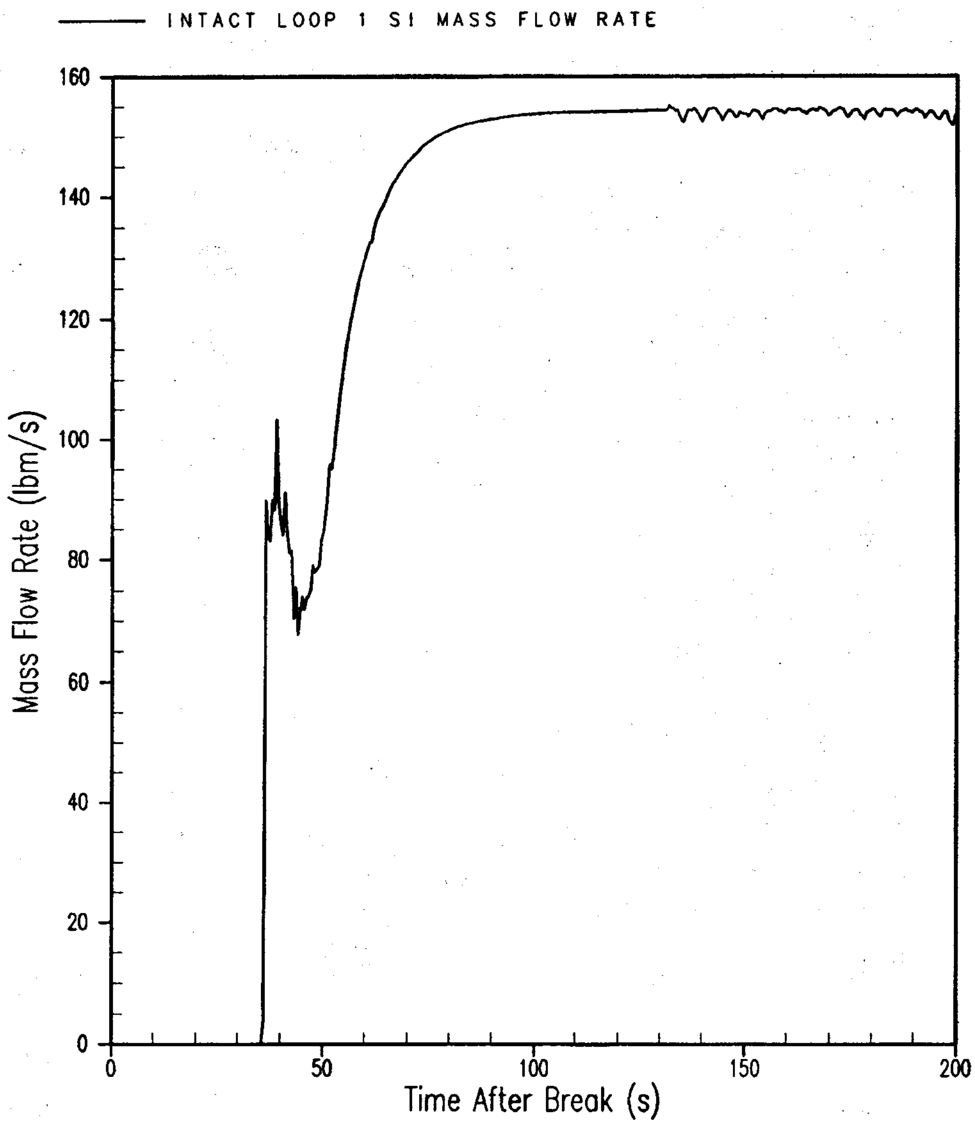
VAPOR FLOW RATE IN CORE HOT ASSEMBLY CHANNEL 15
20
10
0
-10
-20
Mass Flow Rate (lbm/s)
0
5
10
15
20
25
30
Time After Break (s)



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Collapsed Liquid Level in Lower Plenum for Reference Split Transient – BE LBLOCA
	Figure 15.6-12

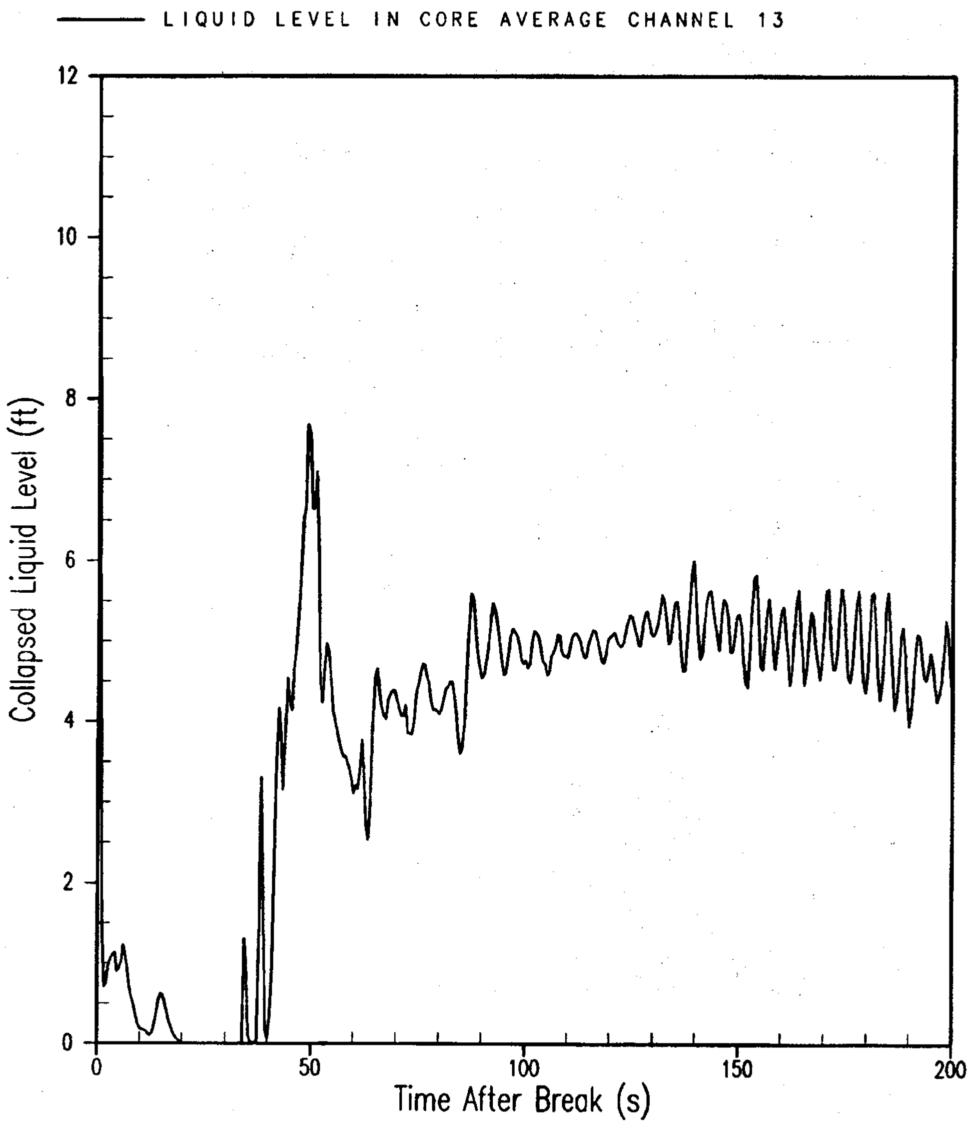


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Accumulator Mass Flow Rate for Reference Split Transient – BE LBLOCA	
		Figure 15.6-13

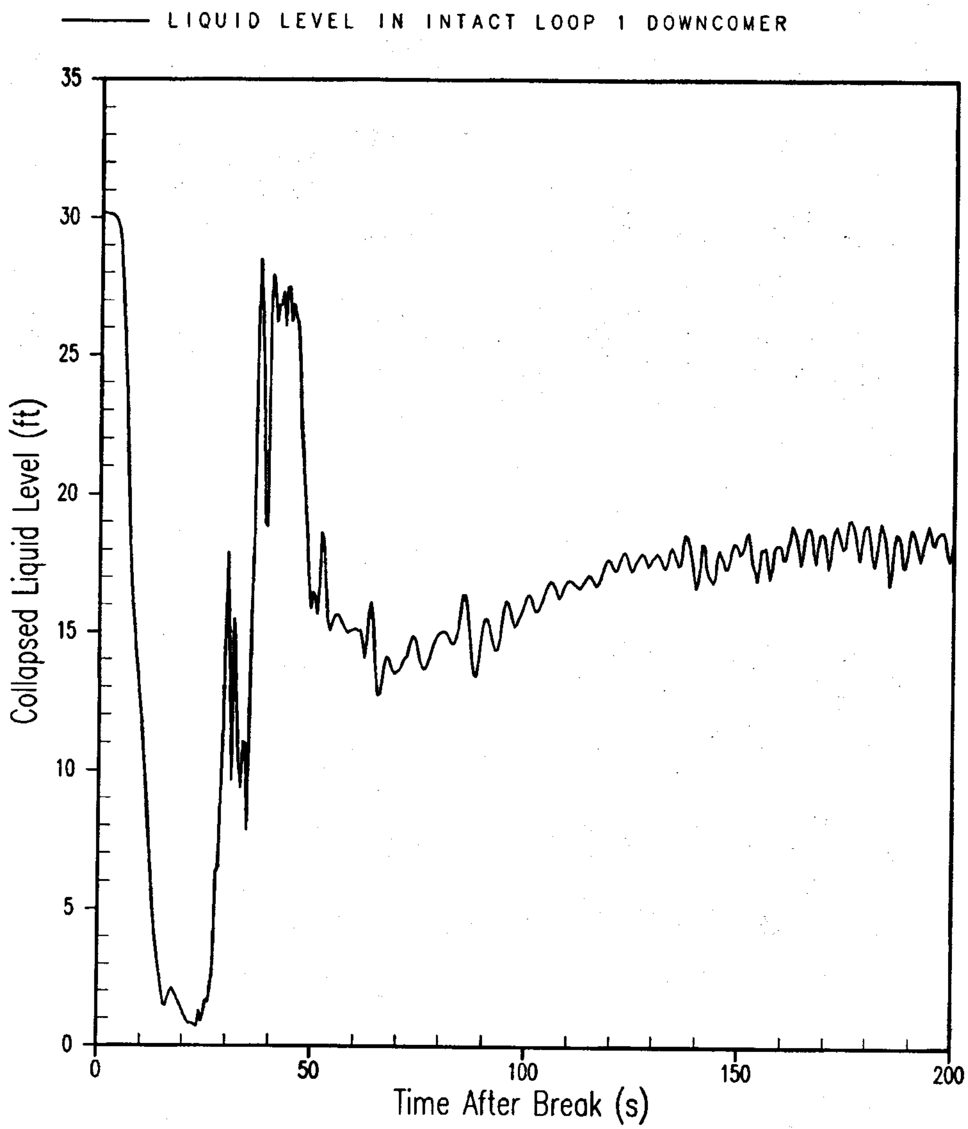


SI Mass Flow Rate for Reference Split Transient – BE LBLOCA

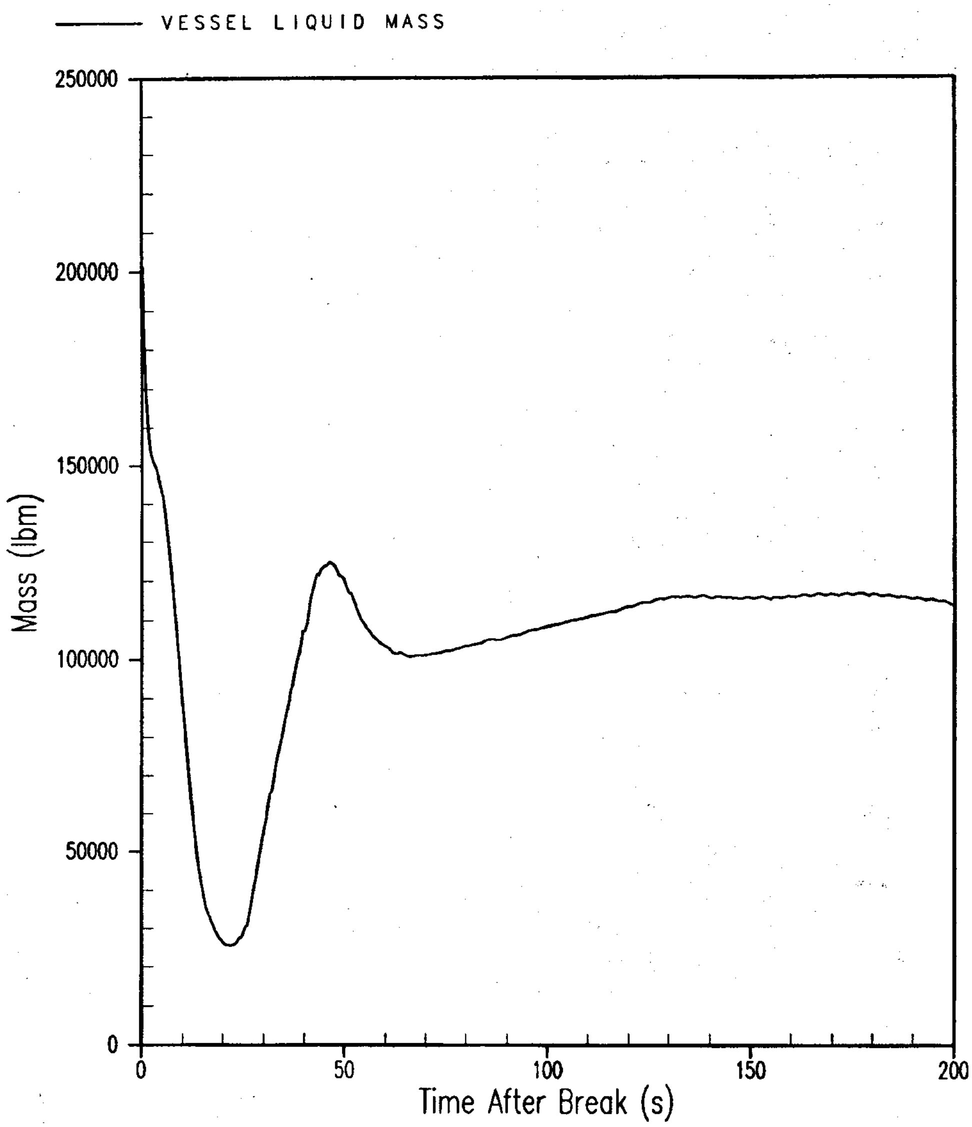
Figure 15.6-14



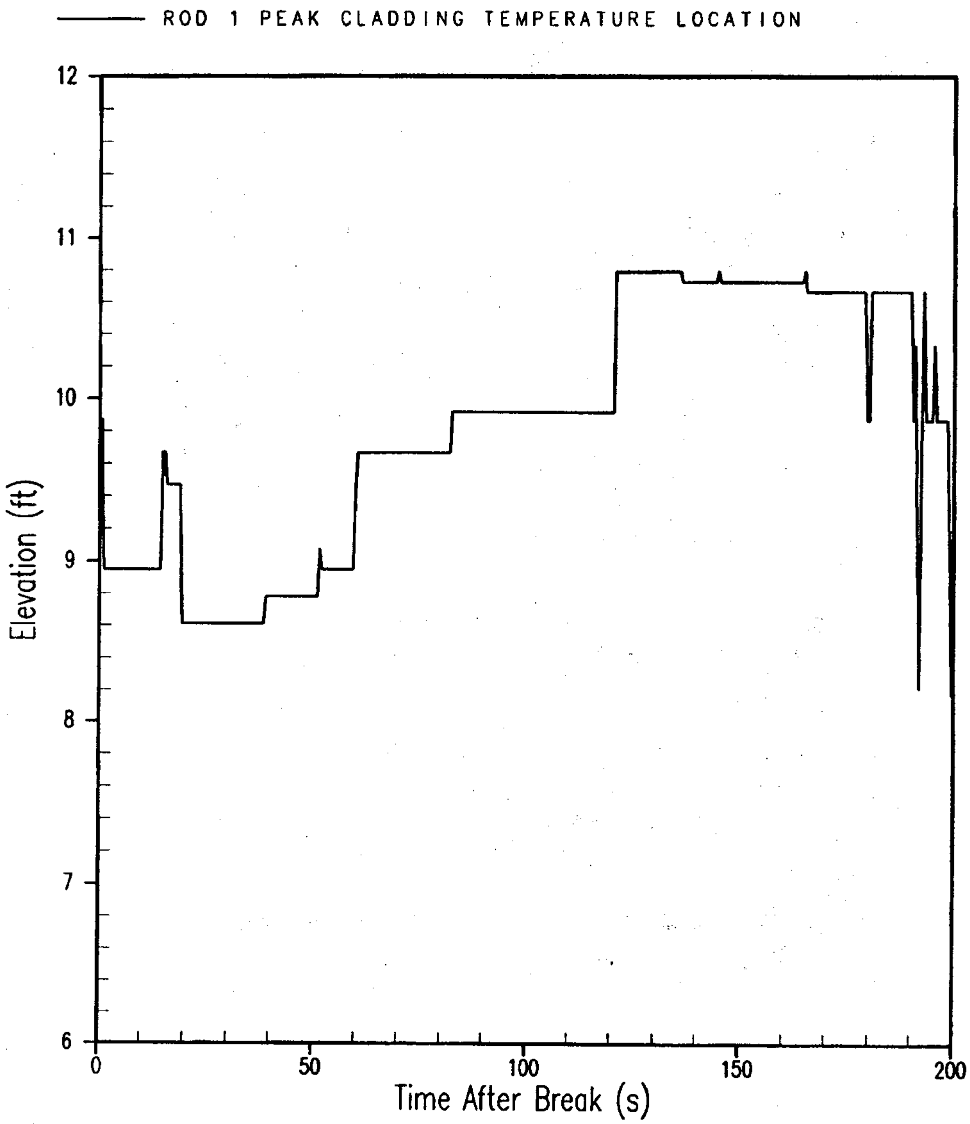
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Collapsed Liquid Level in Core for Reference Split Transient – BE LBLOCA	
		Figure 15.6-15



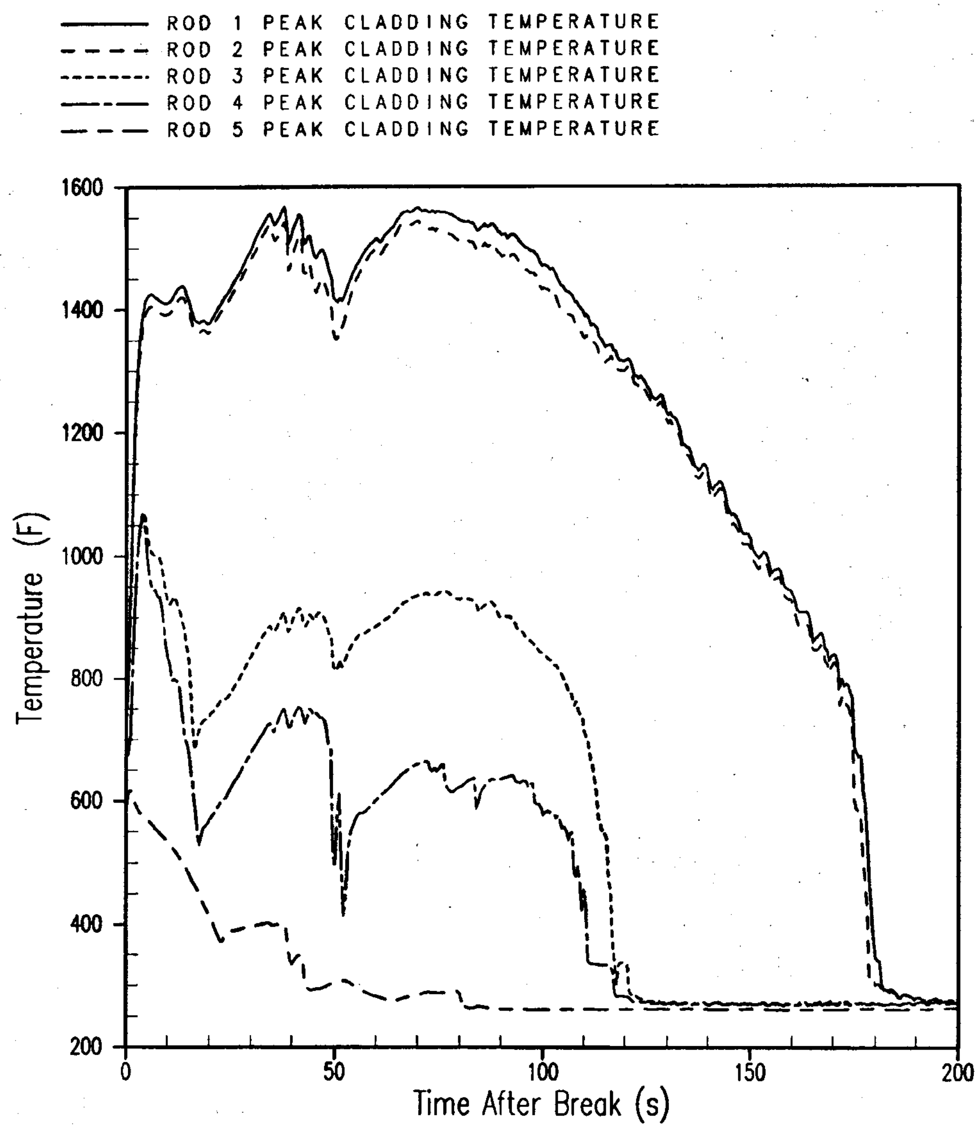
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Collapsed Liquid Level in Downcomer for Reference Split Transient BE LBLOCA	
		Figure 15.6-16



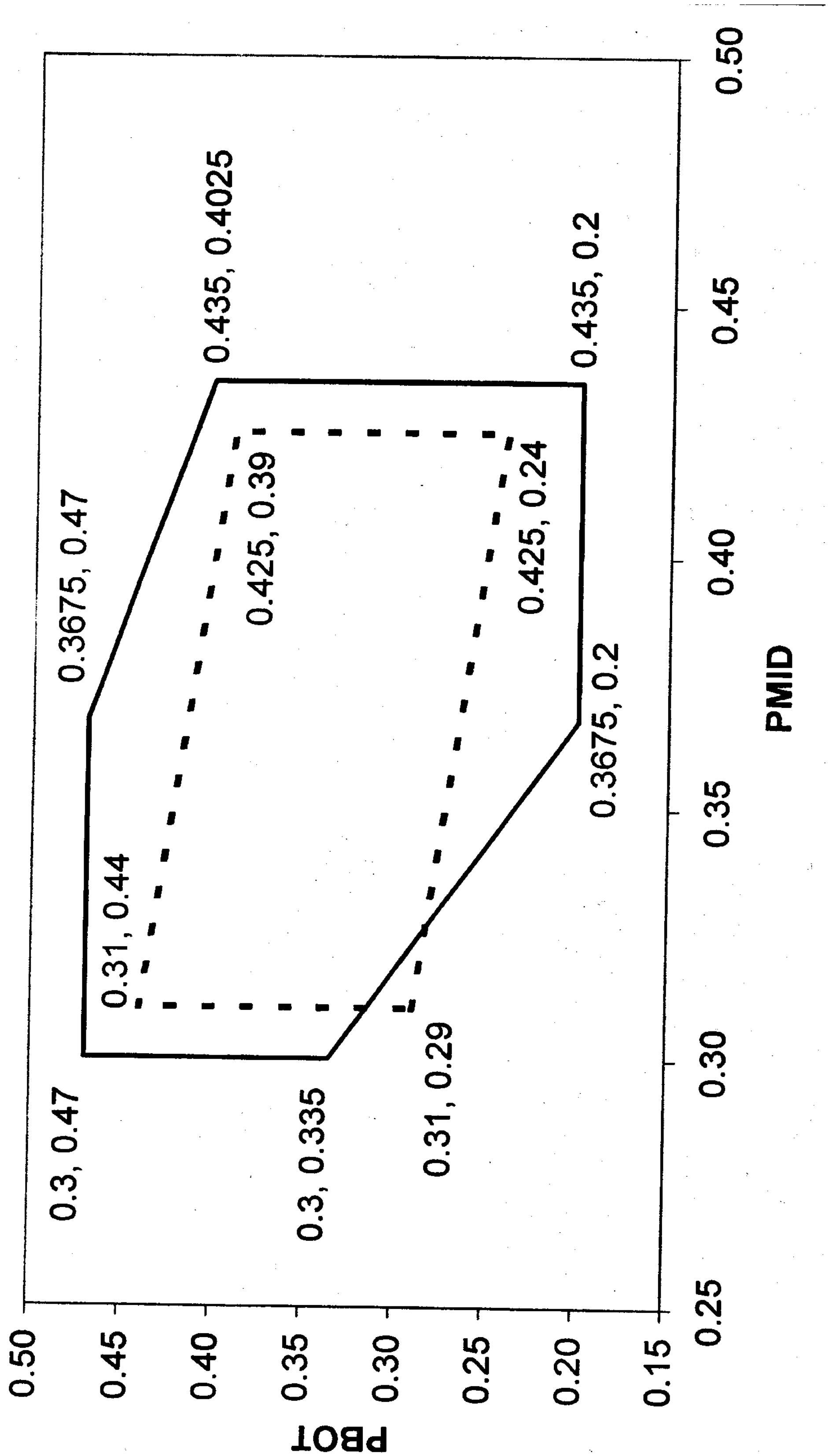
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Vessel Fluid Mass for Reference Split Transient – BE LBLOCA	
		Figure 15.6-17



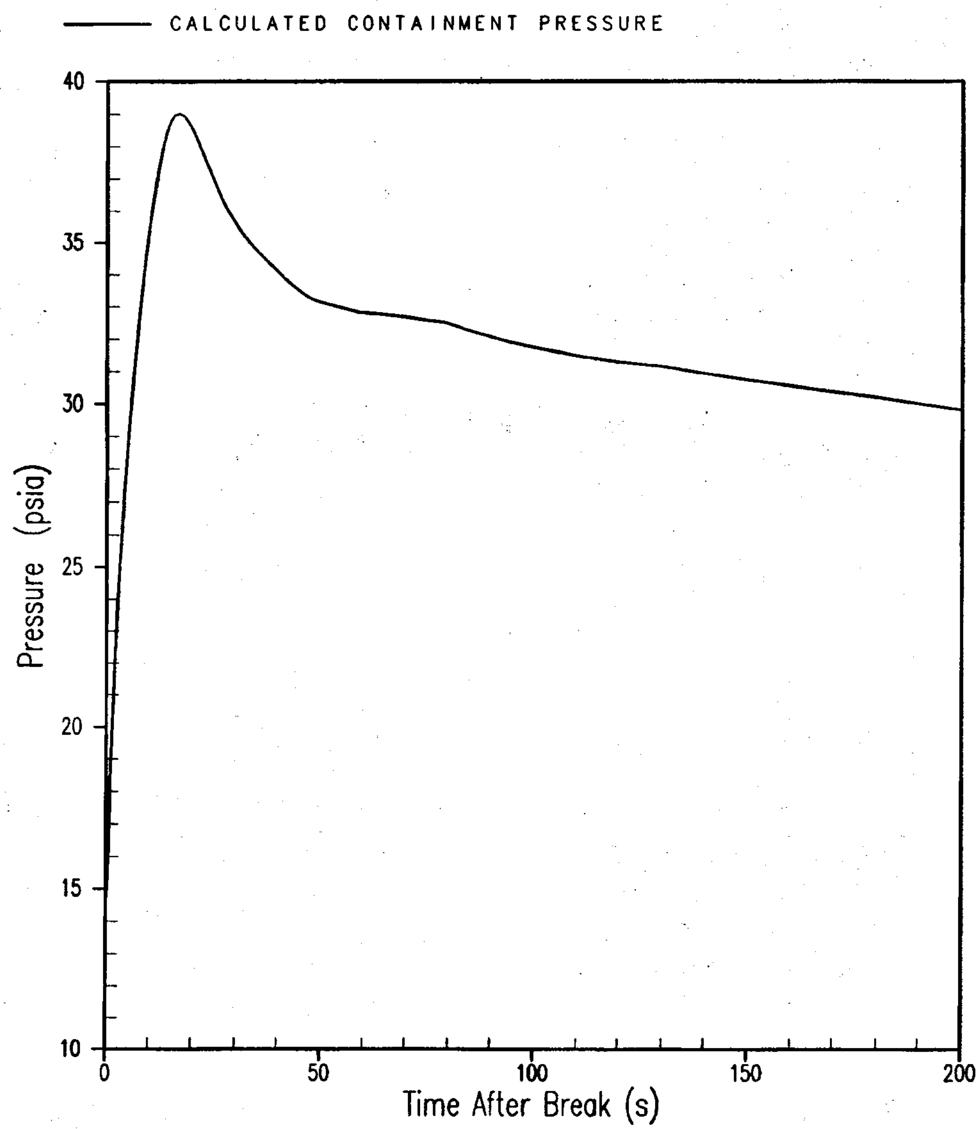
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Cladding Temperature Location for Reference Split Transient – BE LBLOCA	
		Figure 15.6-18



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Cladding Temperature Comparison for Five Rods for Reference Split Transient – BE LBLOCA	
		Figure 15.6-19



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	PBOT/PMID Sampling Limits (Plant Operating range indicated by dashed line; WC/T Range indicated by solid line)	
		Figure 15.6-20



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Calculated Containment Pressure for Seabrook Station – BE LBLOCA	
		Figure 15.6-21

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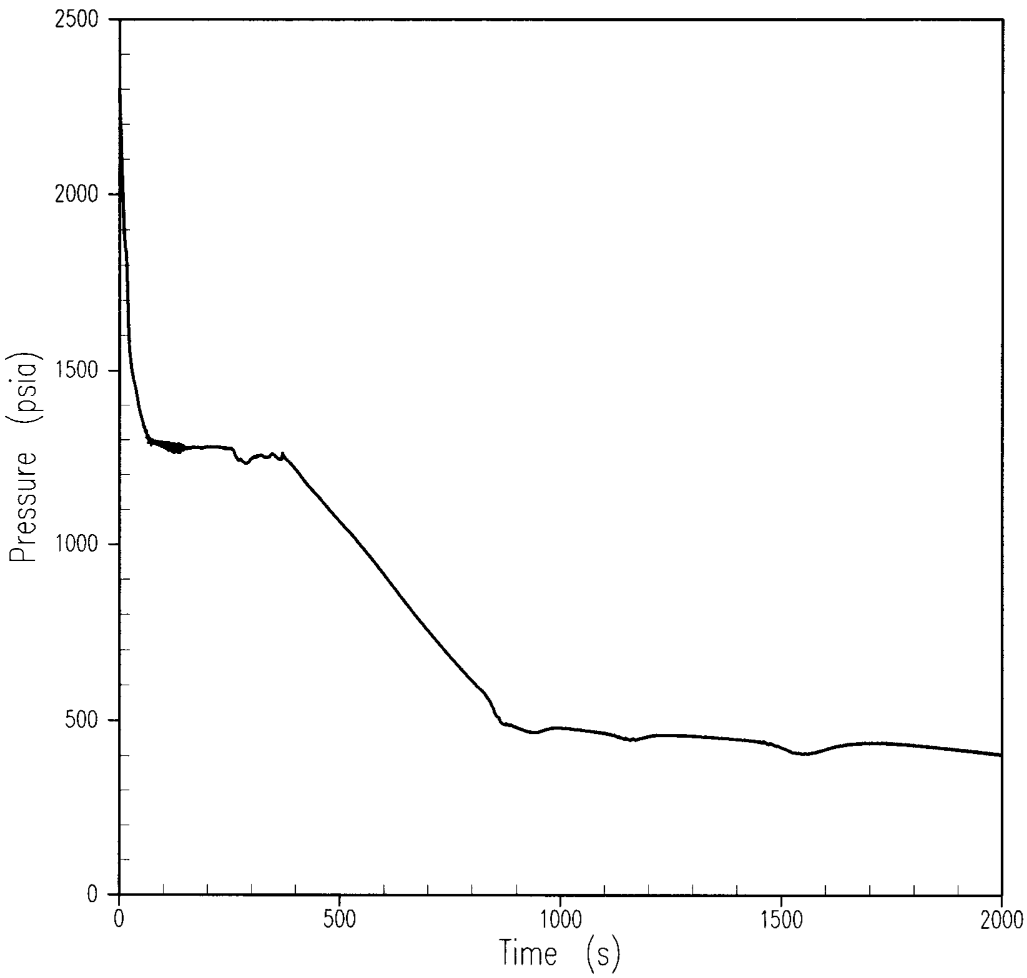
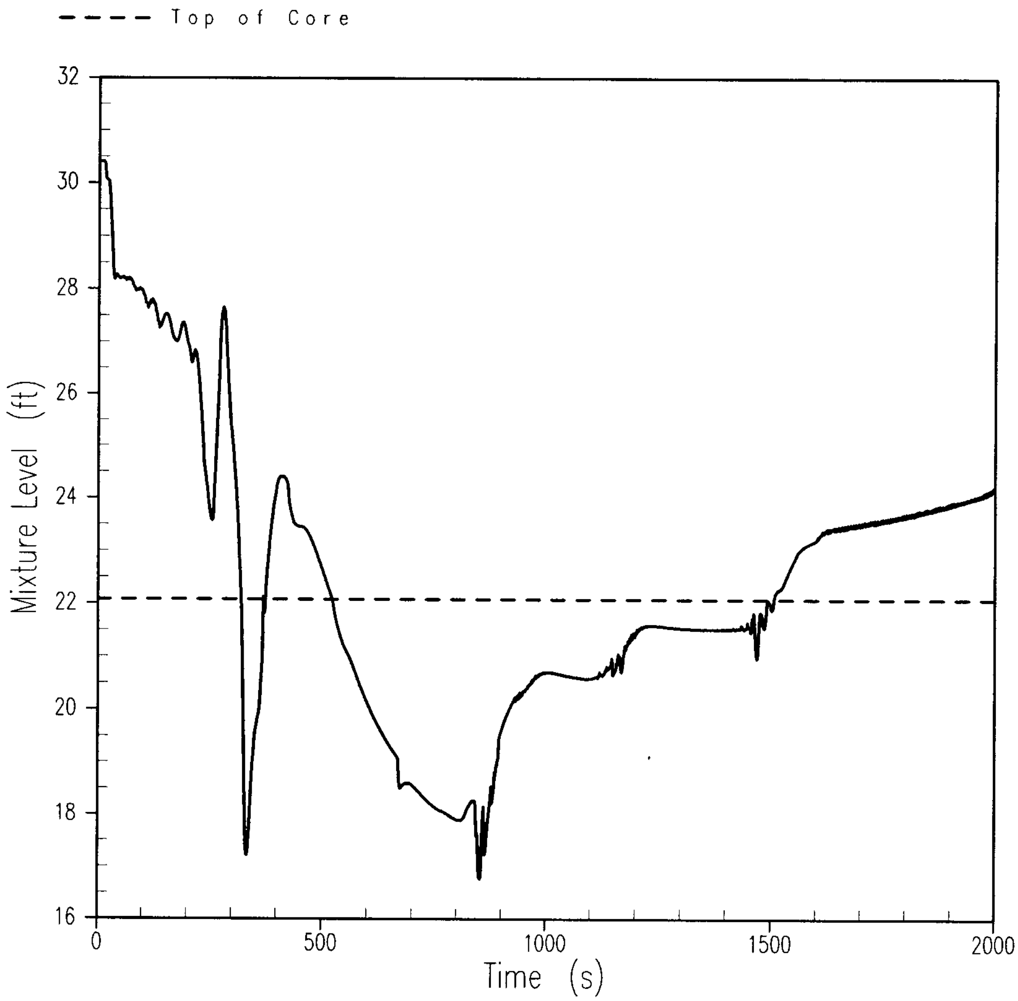
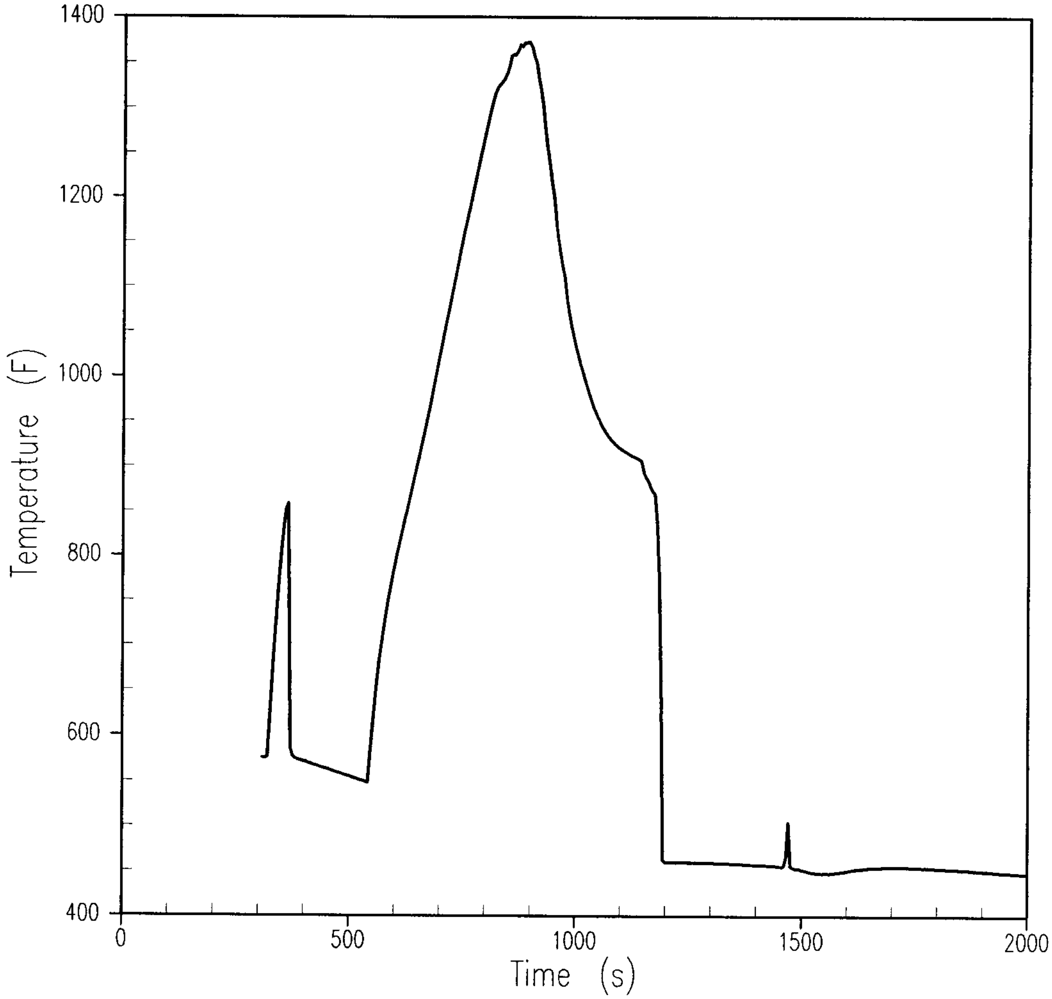


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure (4 Inch Break)
	Figure 15.6-34

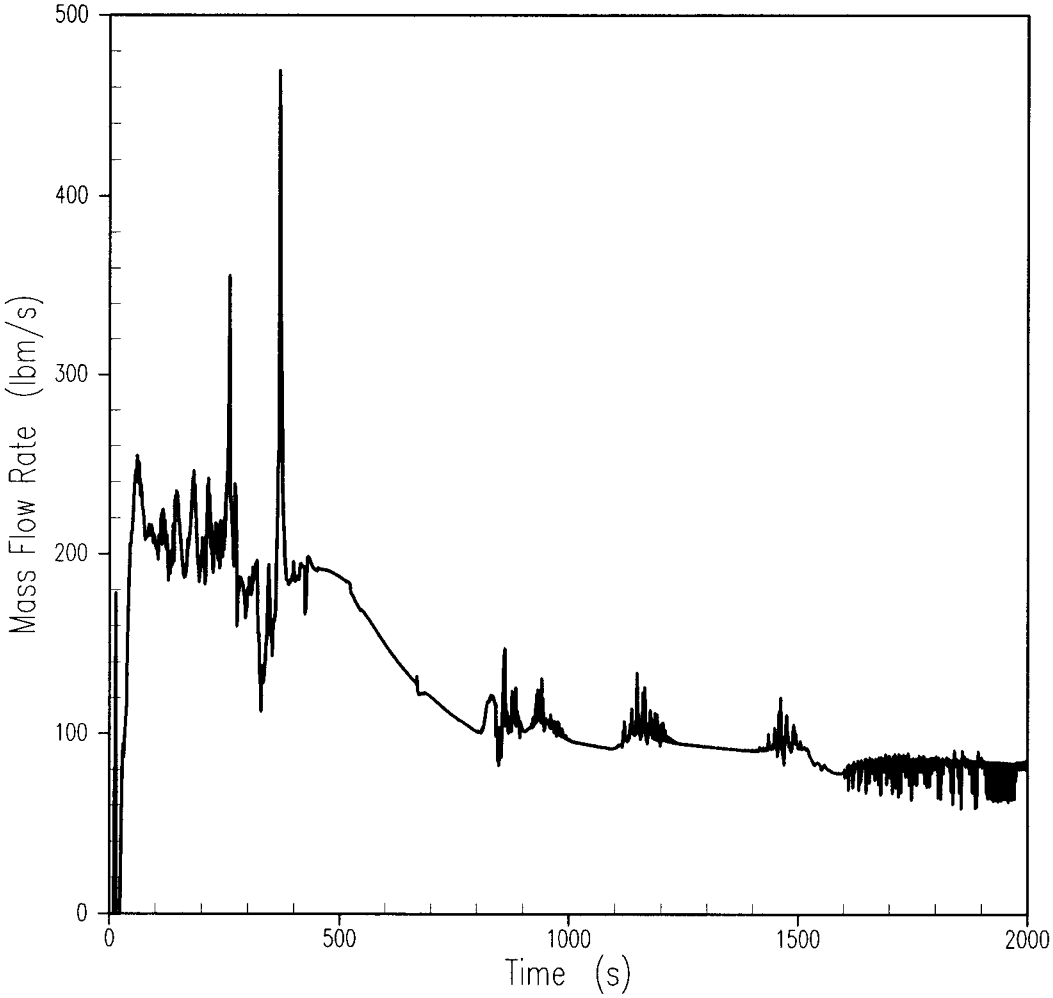


Top of Core
Mixture Level (ft)
32
30
28
26
24
22
20
18
16
0
500
1000
1500
2000
Time (s)

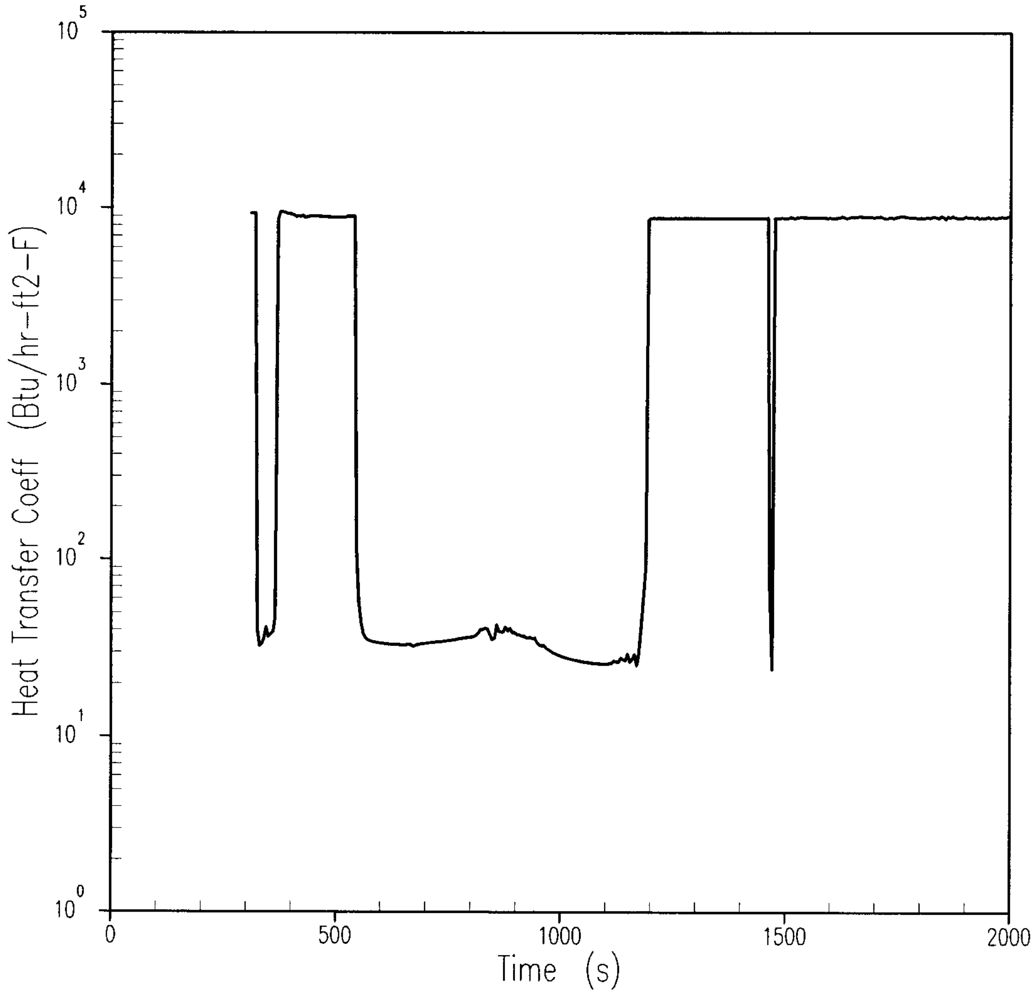


Peak Clad Temperature (4 Inch Break)

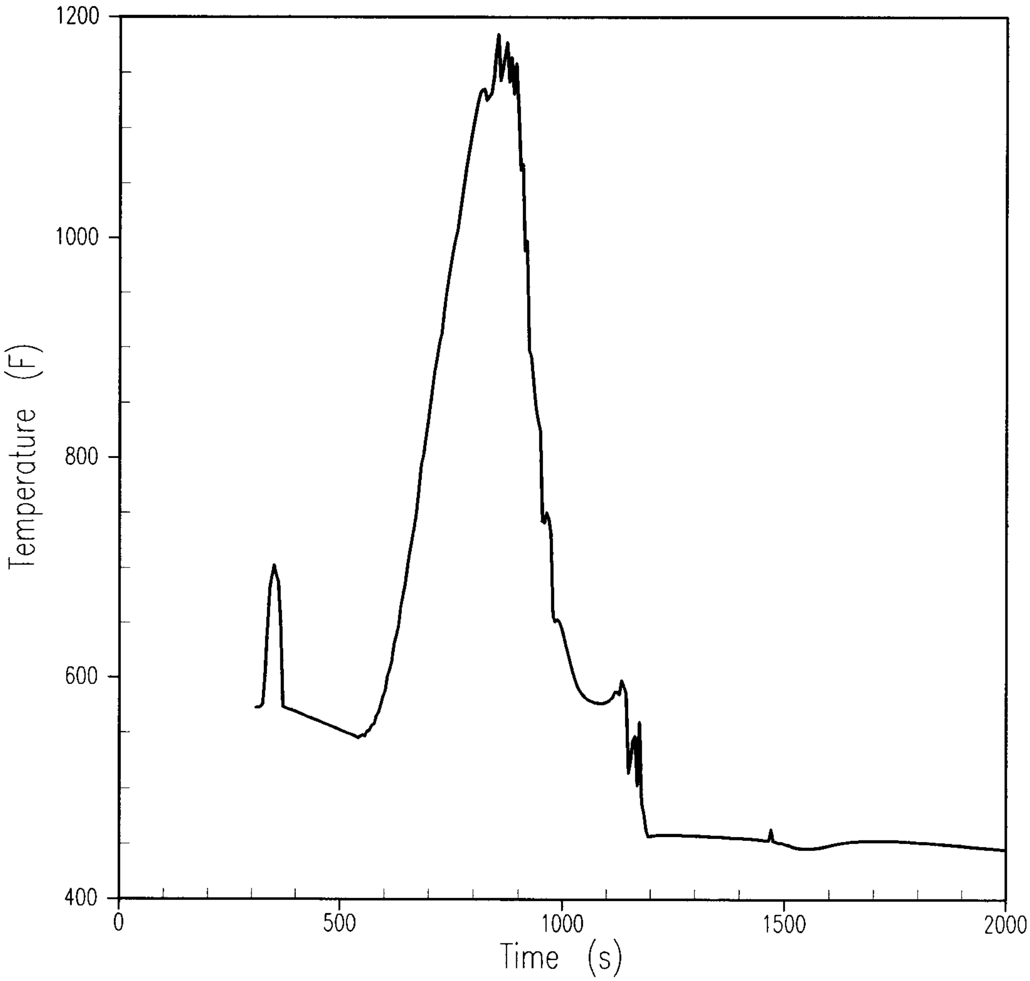
Figure 15.6-36



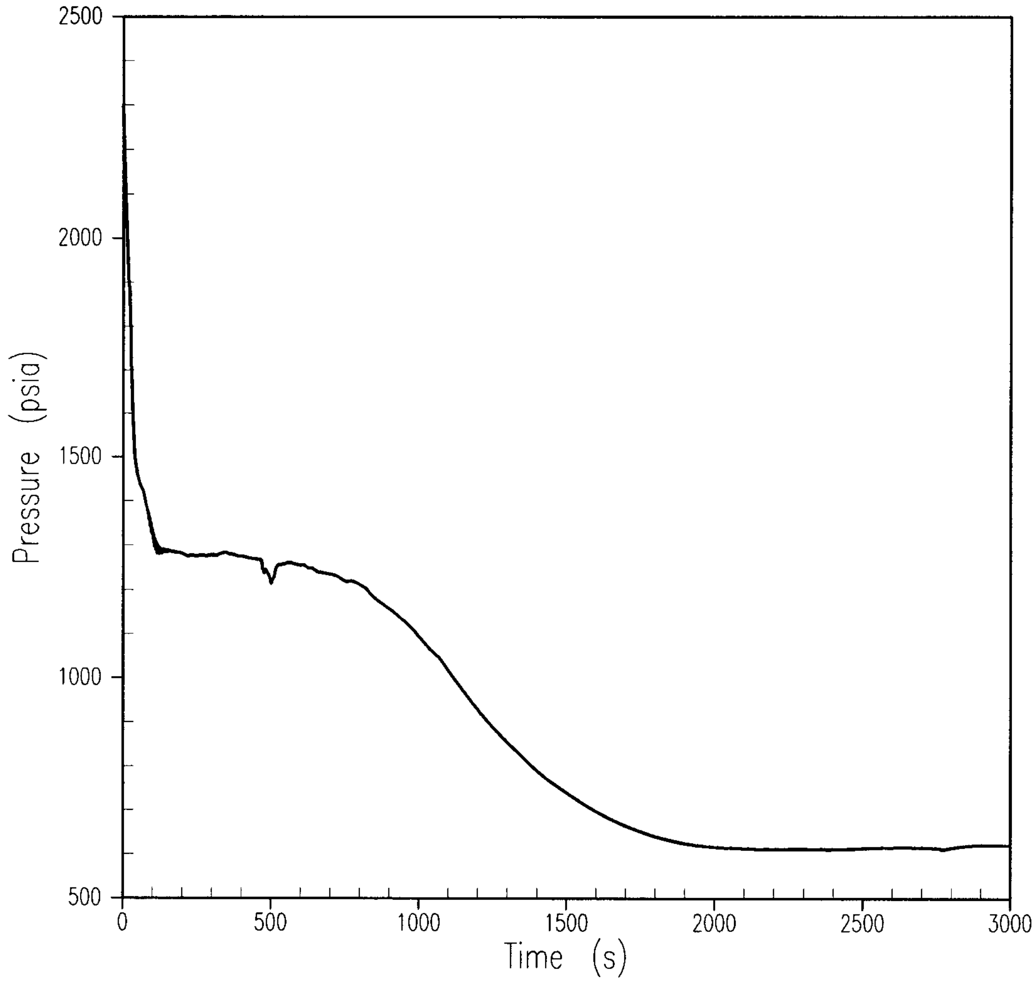
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Exit Steam Mass Flow Rate (4 Inch Break)	
		Figure 15.6-37



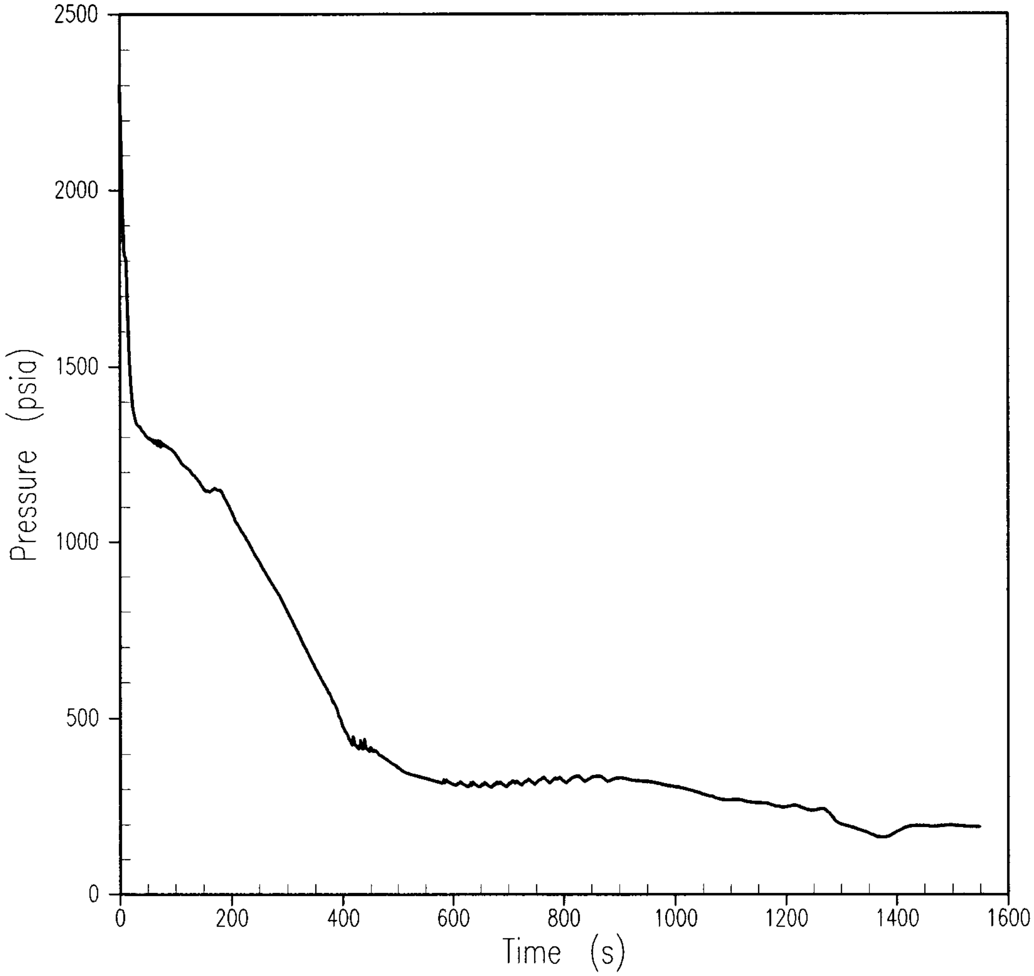
Heat Transfer Coeff (Btu/hr-ft2-F)
Time (s)



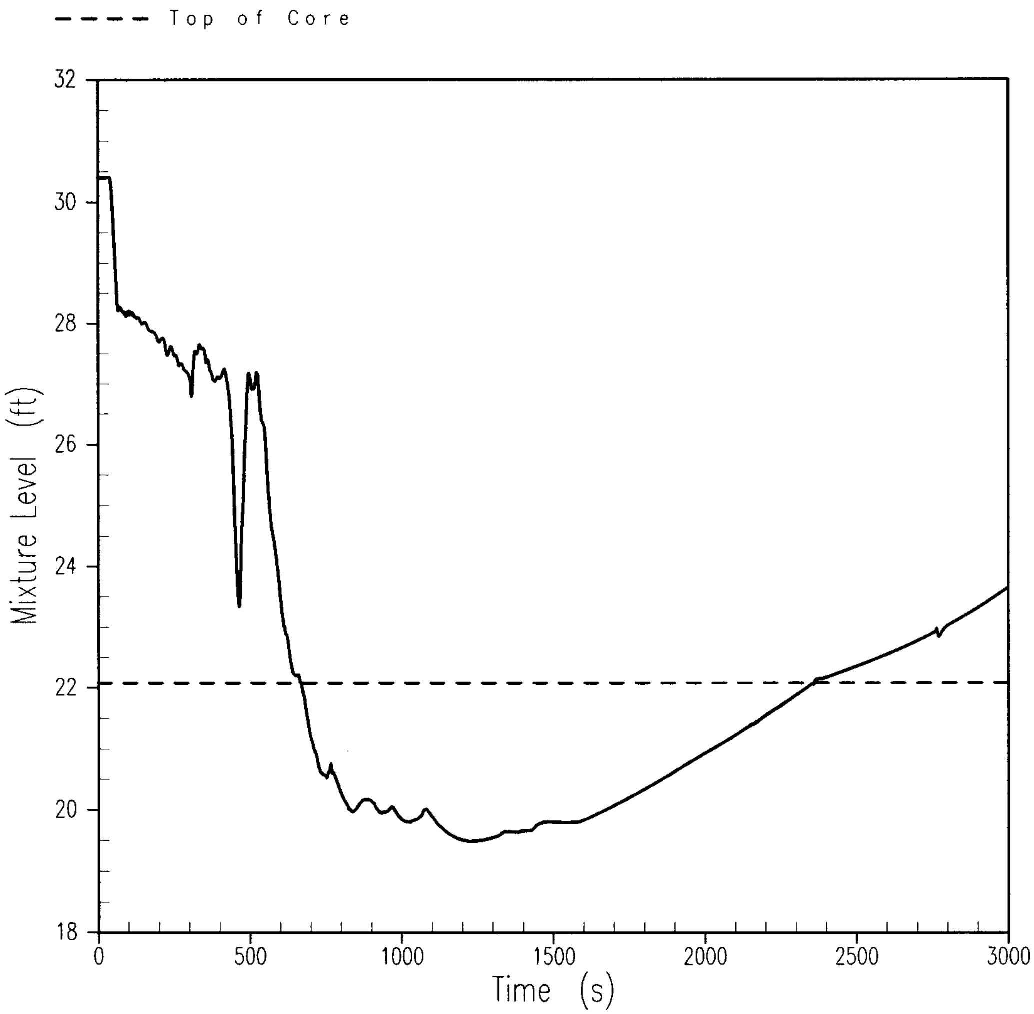
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Clad Hot Spot Fluid Temperature (4 Inch Break)	
		Figure 15.6-39



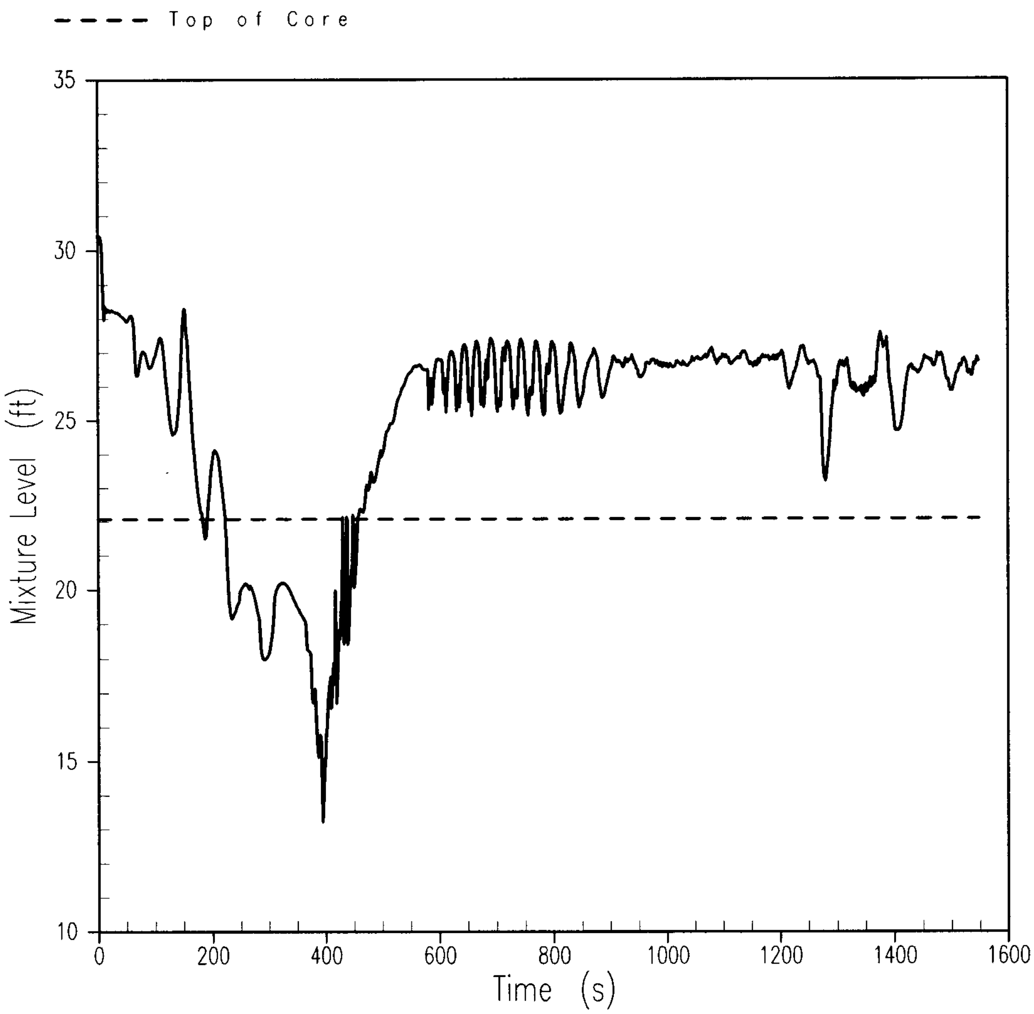
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure (3 Inch Break)
	Figure 15.6-40



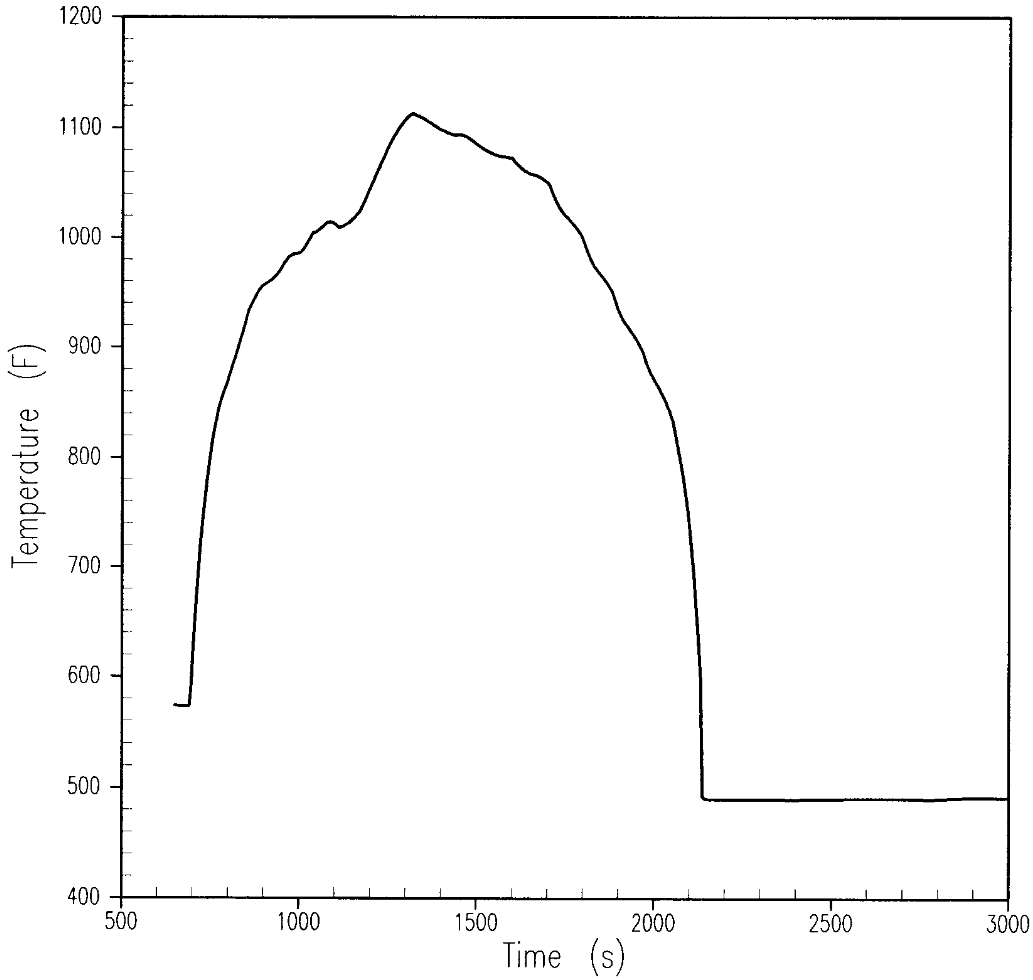
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	RCS Pressure (6 Inch Break)	
		Figure 15.6-41



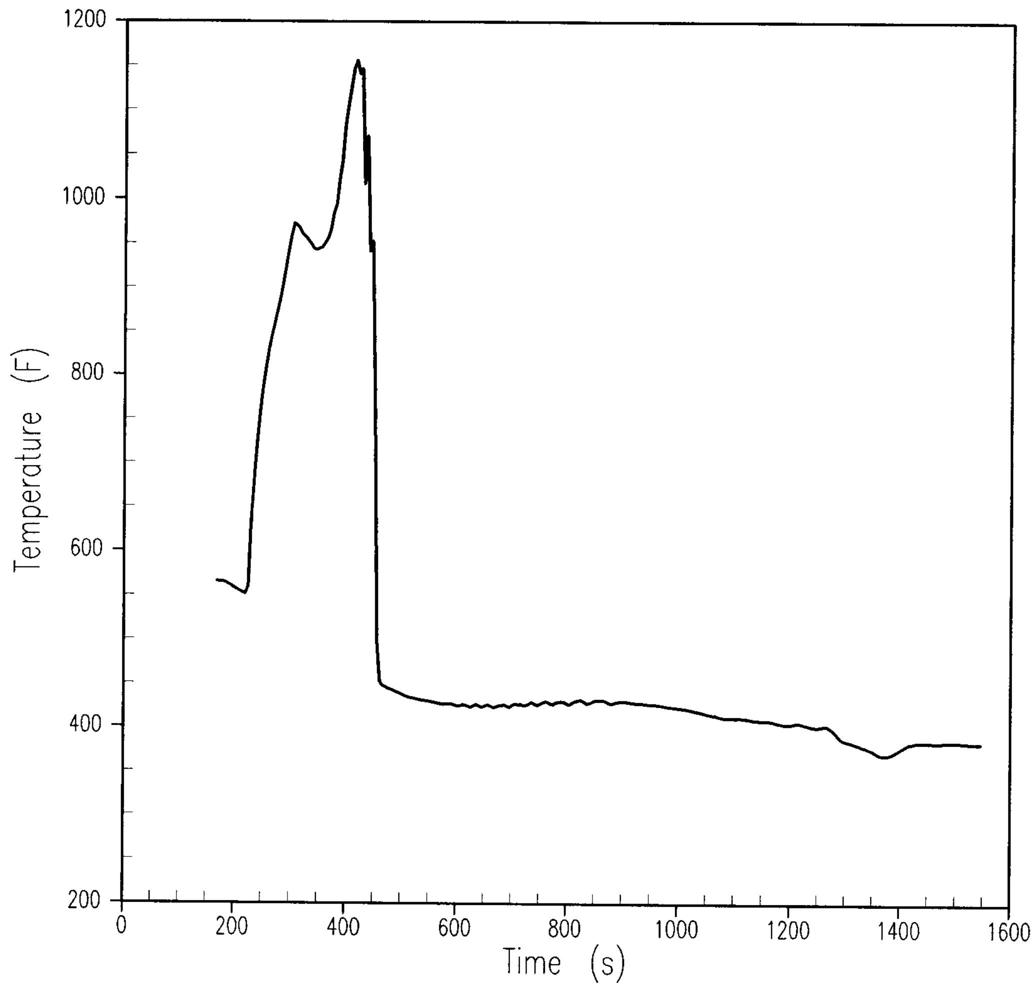
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Mixture Level (3 Inch Break)	
		Figure 15.6-42



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Mixture Level (6 Inch Break)	
		Figure 15.6-43



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Clad Temperature (3 Inch Break)	
		Figure 15.6-44



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Peak Clad Temperature (6 Inch Break)	
		Figure 15.6-45

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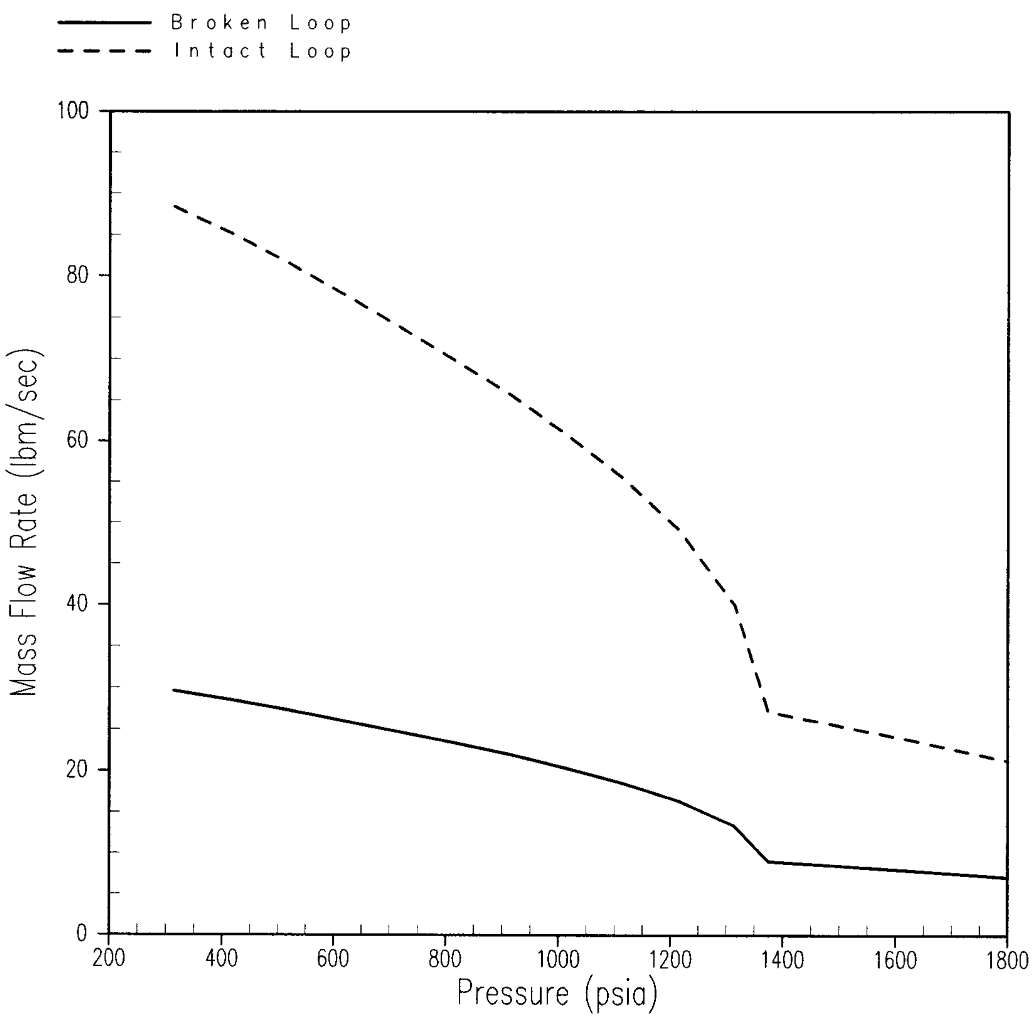
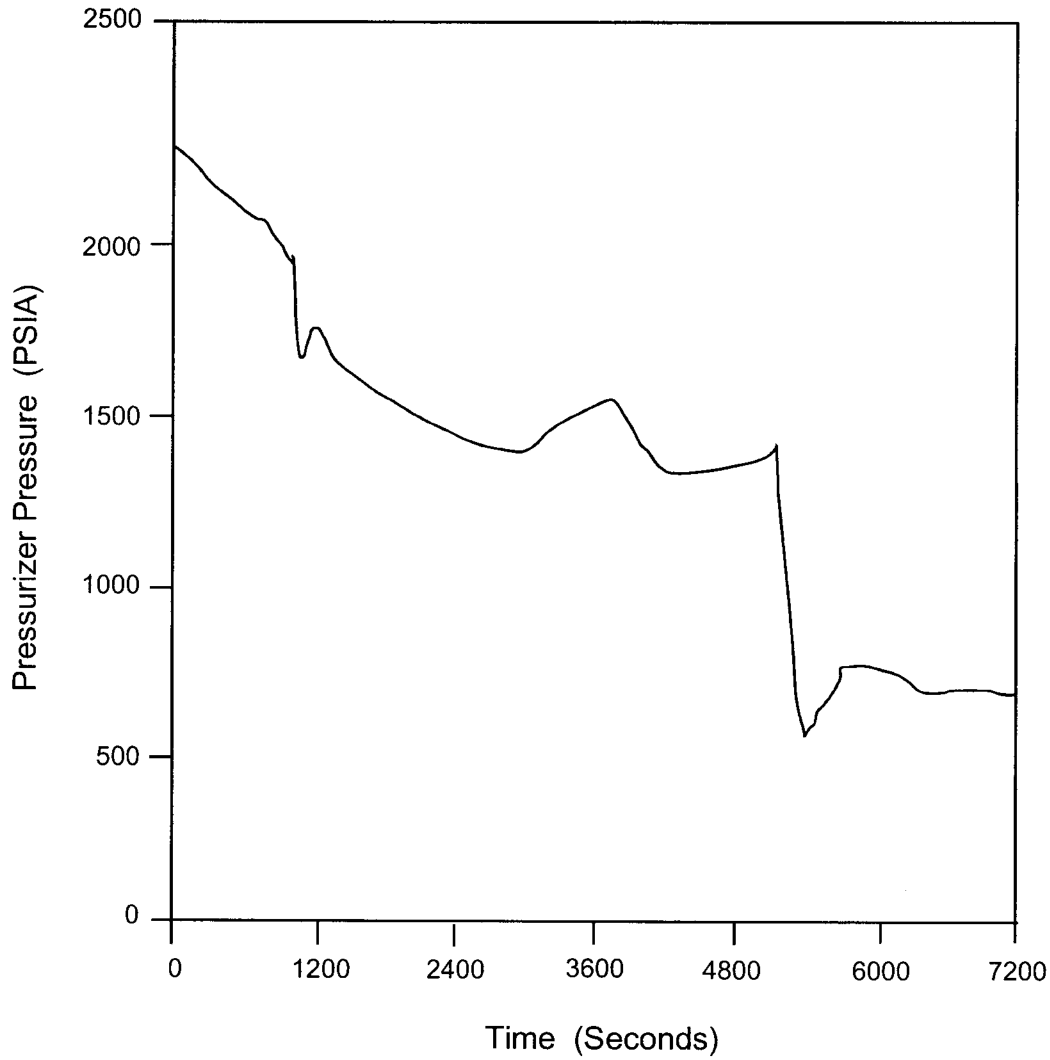
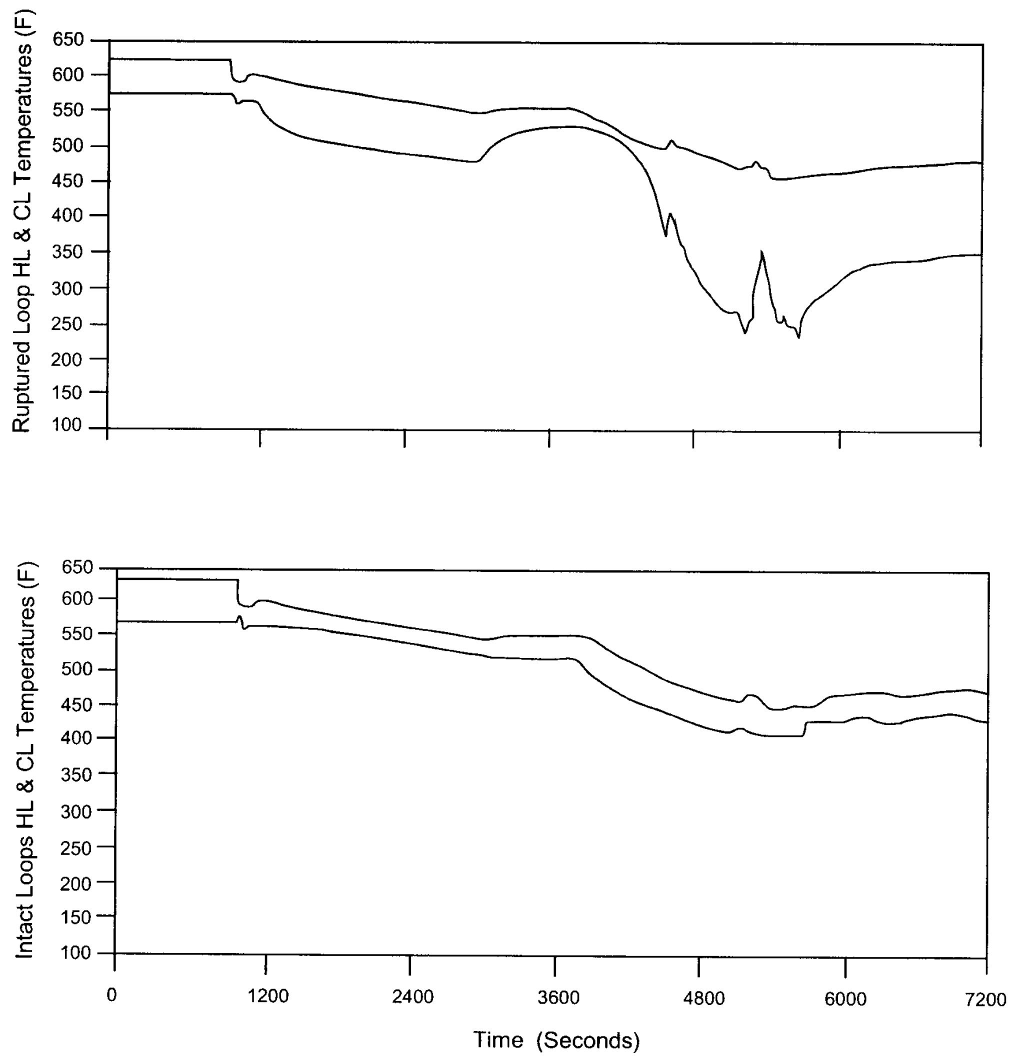


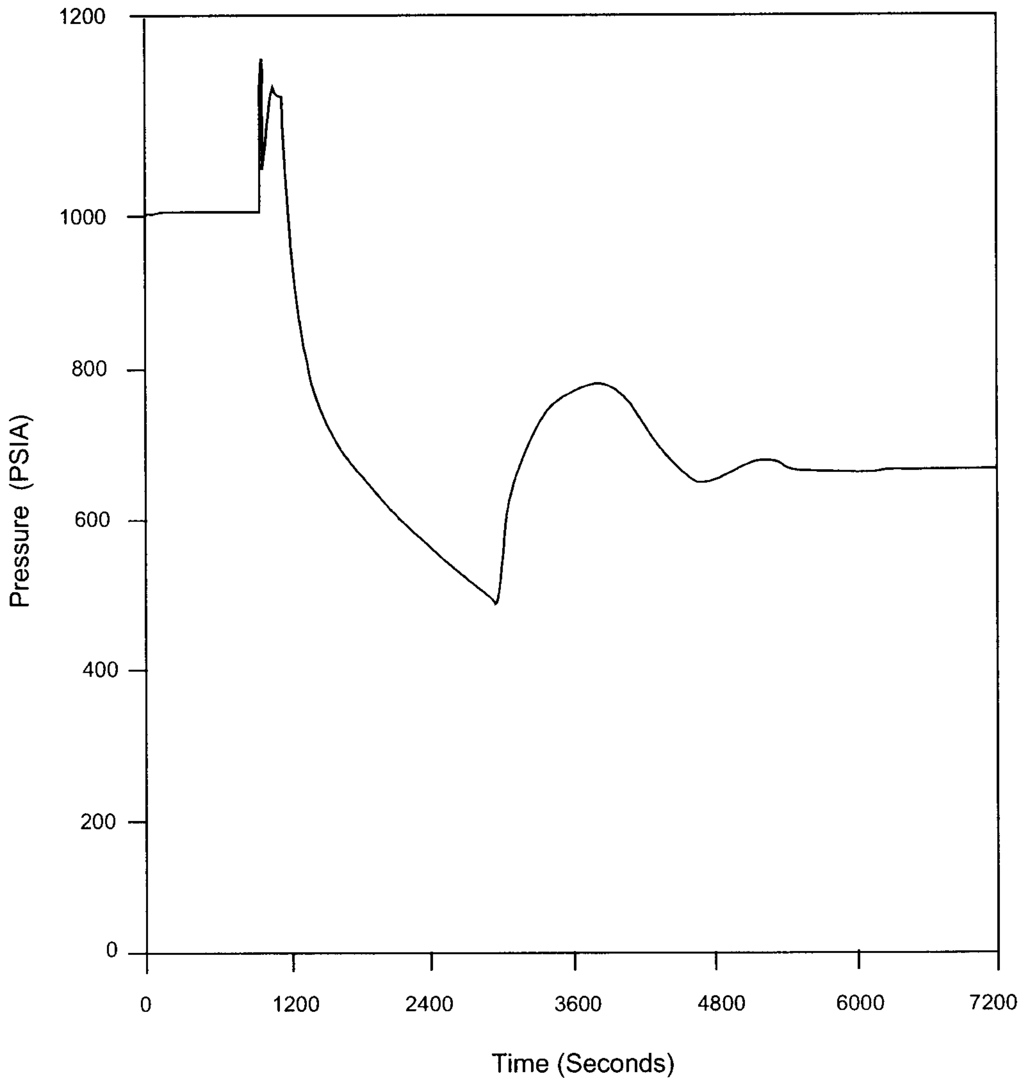
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pumped Safety Injection Flow vs. RCS Pressure	
		Figure 15.6-49



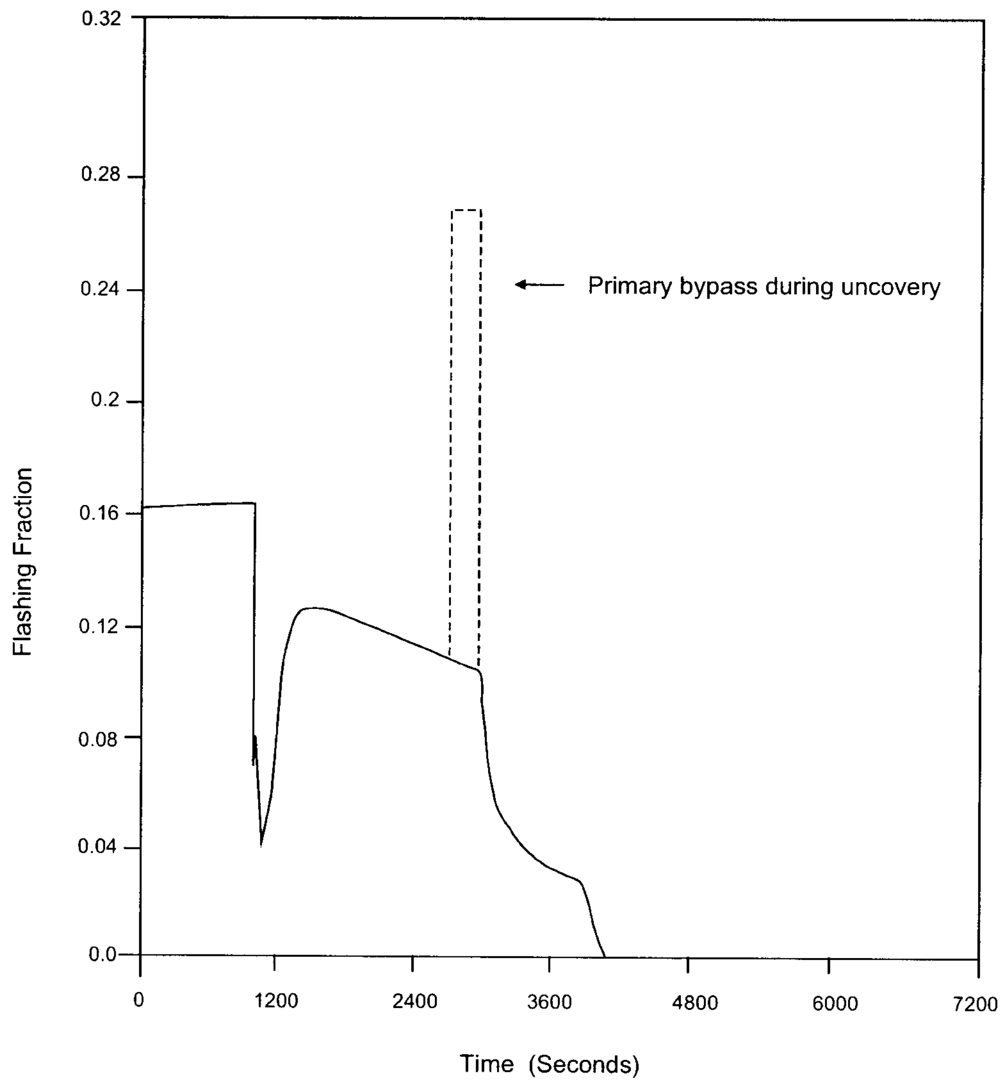
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Pressurizer Pressure as the Result of Steam Generator Tube Rupture	
		Figure 15.6-50



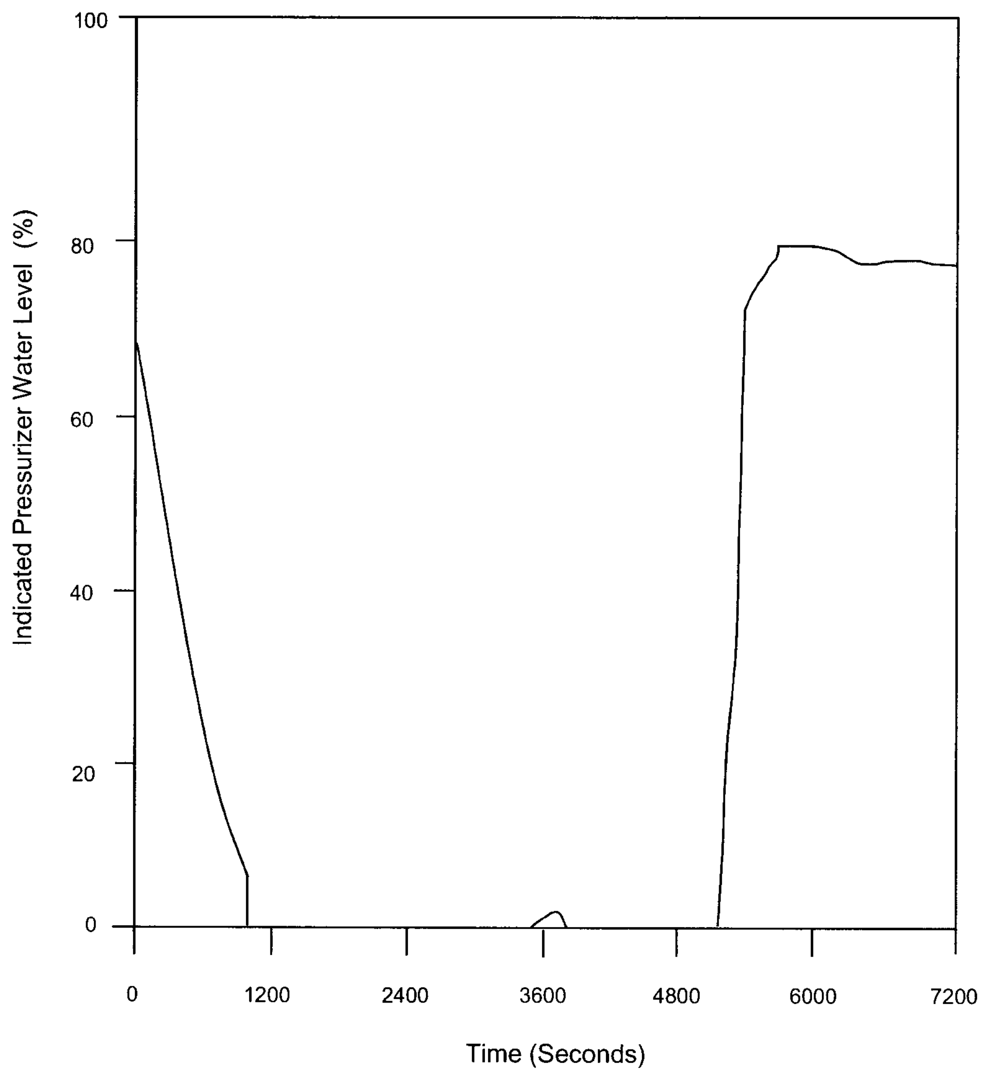
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Reactor Coolant System Temperature as the Result of Steam Generator Tube Rupture	
		Figure 15.6-51



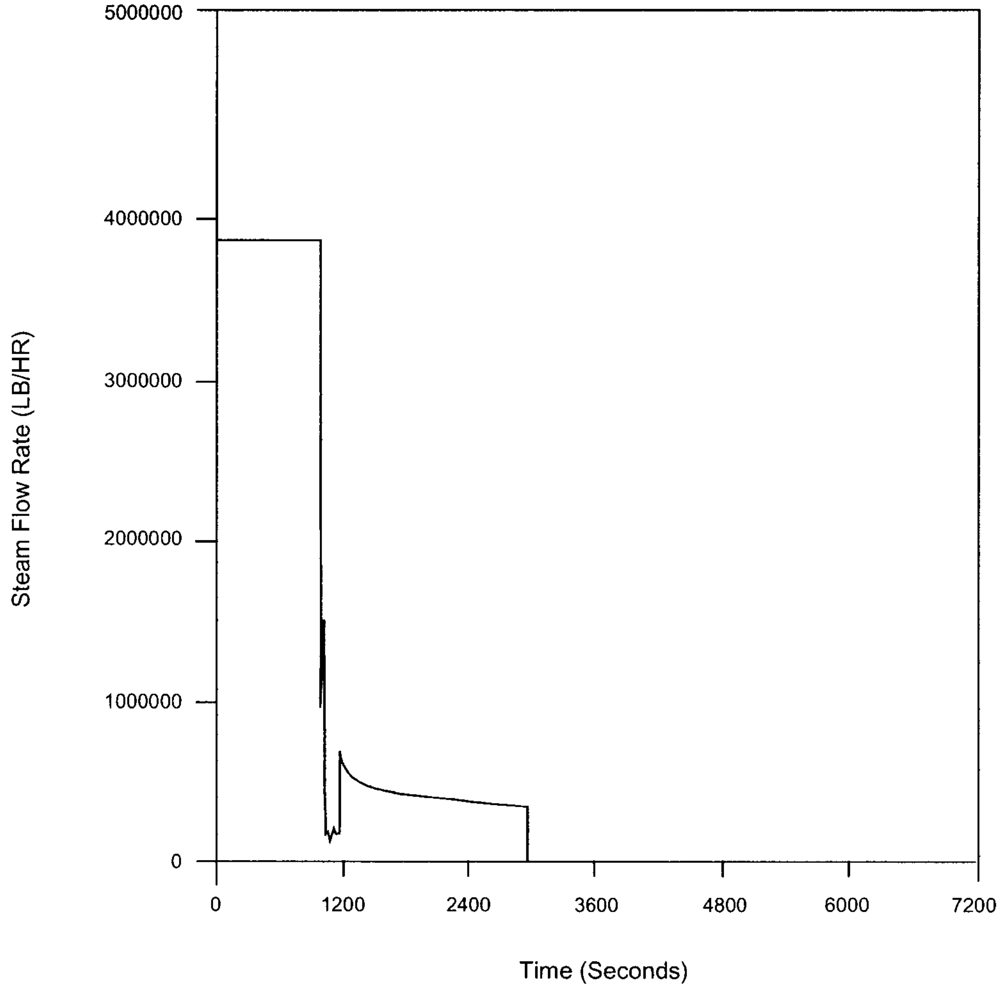
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Faulted Steam Generator Pressure as the Result of Steam Generator Tube Rupture	
		Figure 15.6-52



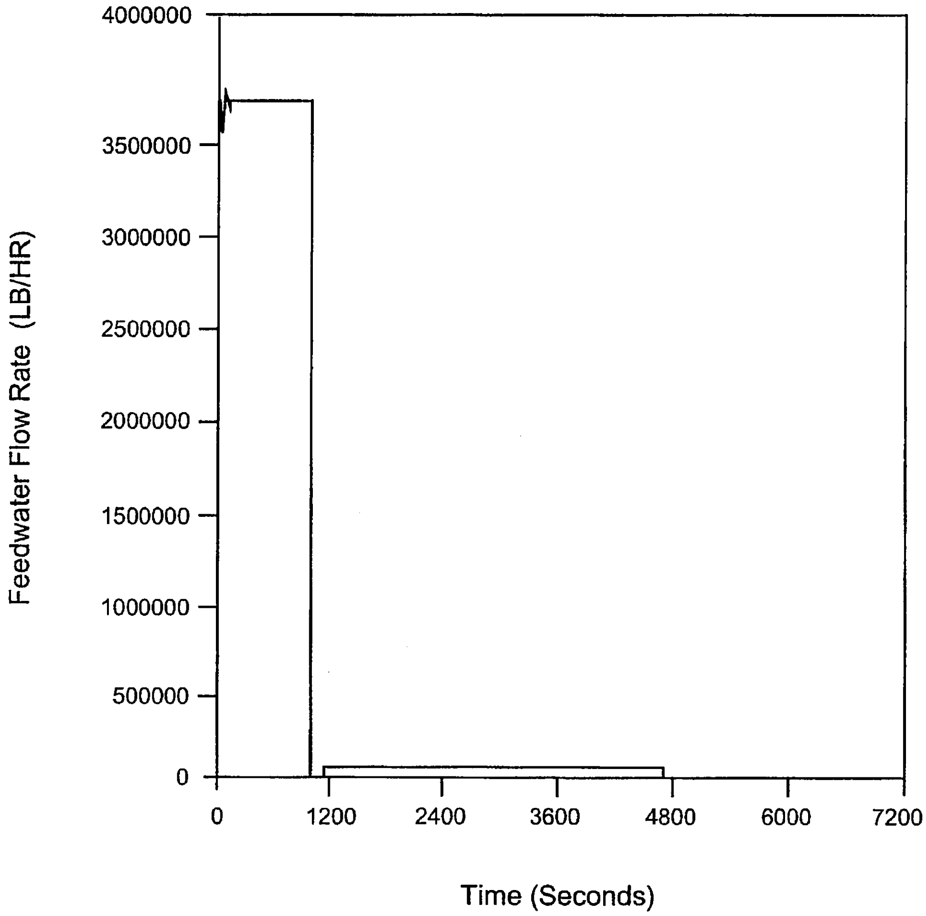
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Primary Coolant Flashing in Faulted Steam Generator as the Result of Steam Generator Tube Rupture
	Figure 15.6-53



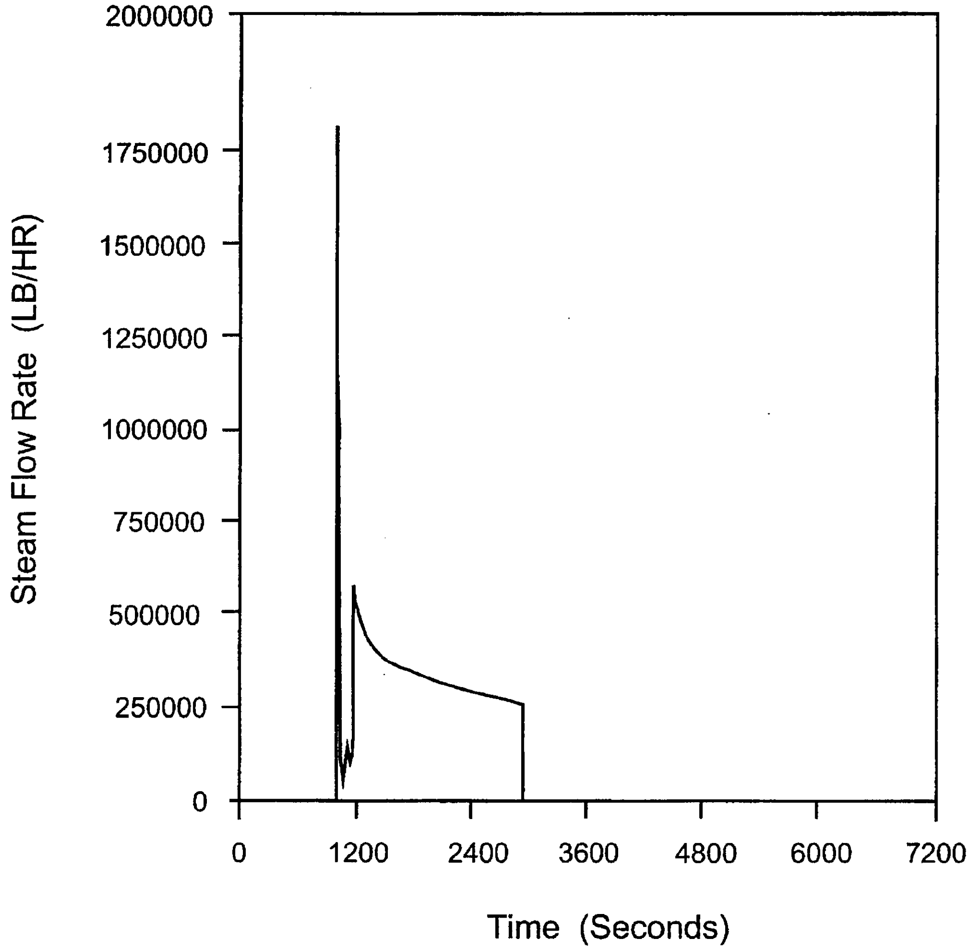
100
80
60
40
20
0
Indicated Pressurizer Water Level (%)
0
1200
2400
3600
4800
6000
7200
Time (Seconds)



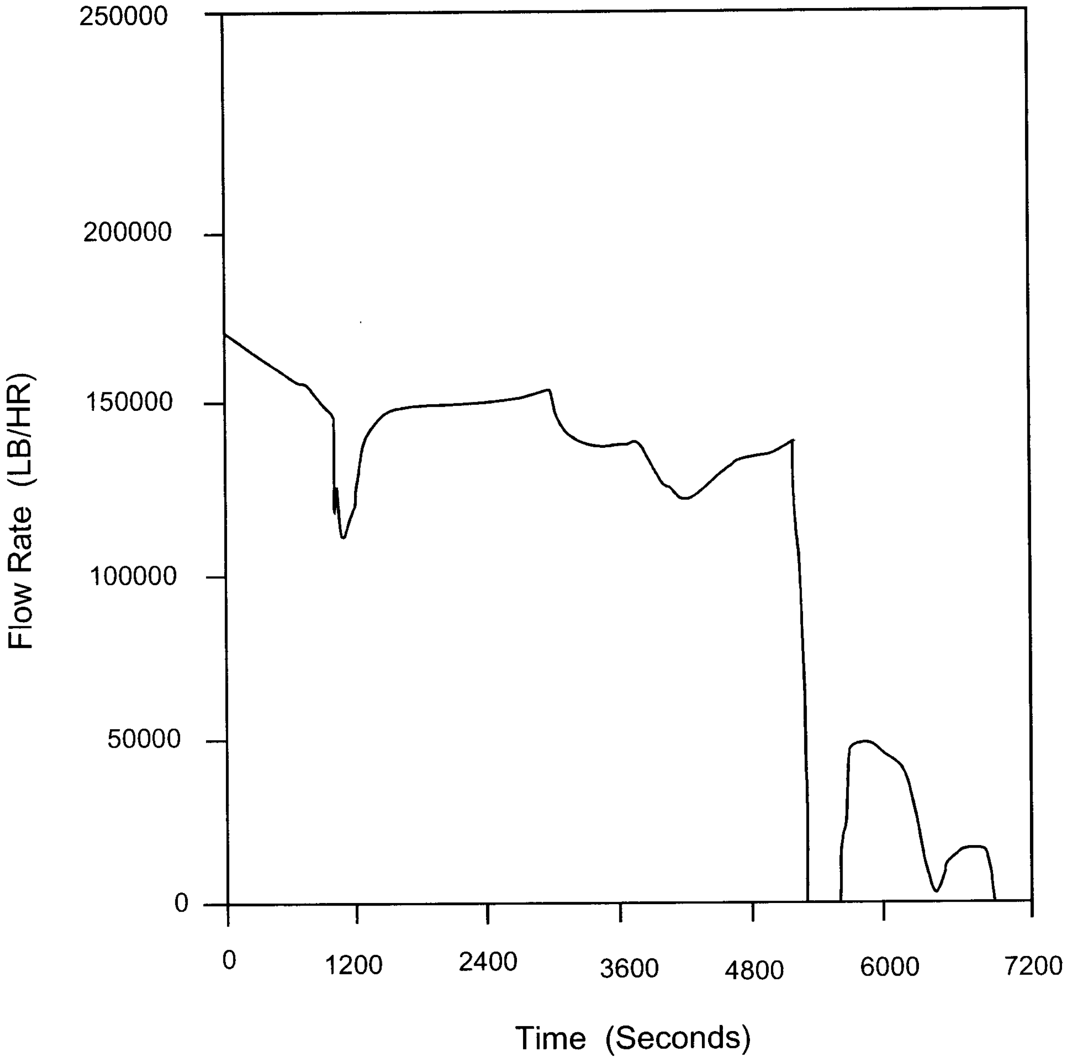
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Faulted Steam Generator Steam Flow Rate as the Result of Steam Generator Tube Rupture
	Figure 15.6-55



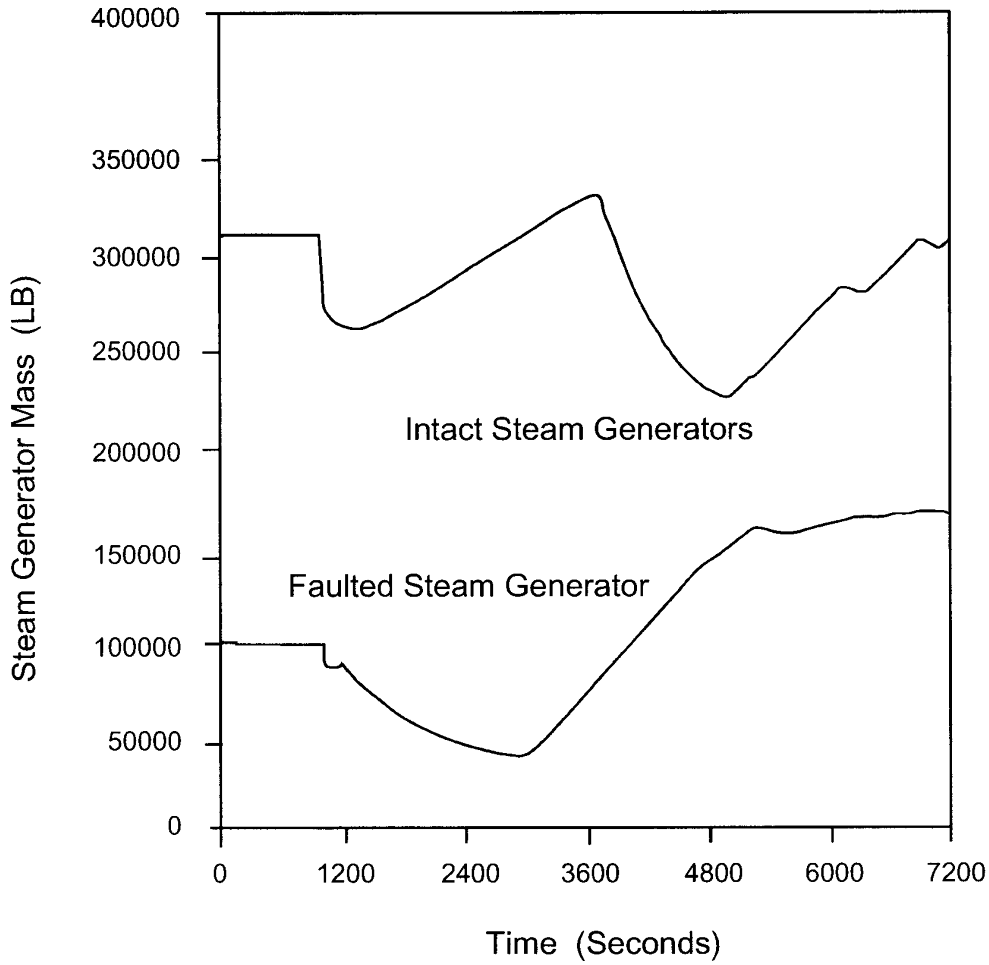
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Feedwater Flow Rate to Faulted Steam Generator as the Result of Steam Generator Tube Rupture	
		Figure 15.6-56



2000000
1750000
1500000
1250000
1000000
750000
500000
250000
0
Steam Flow Rate (LB/HR)
0
1200
2400
3600
4800
6000
7200
Time (Seconds)

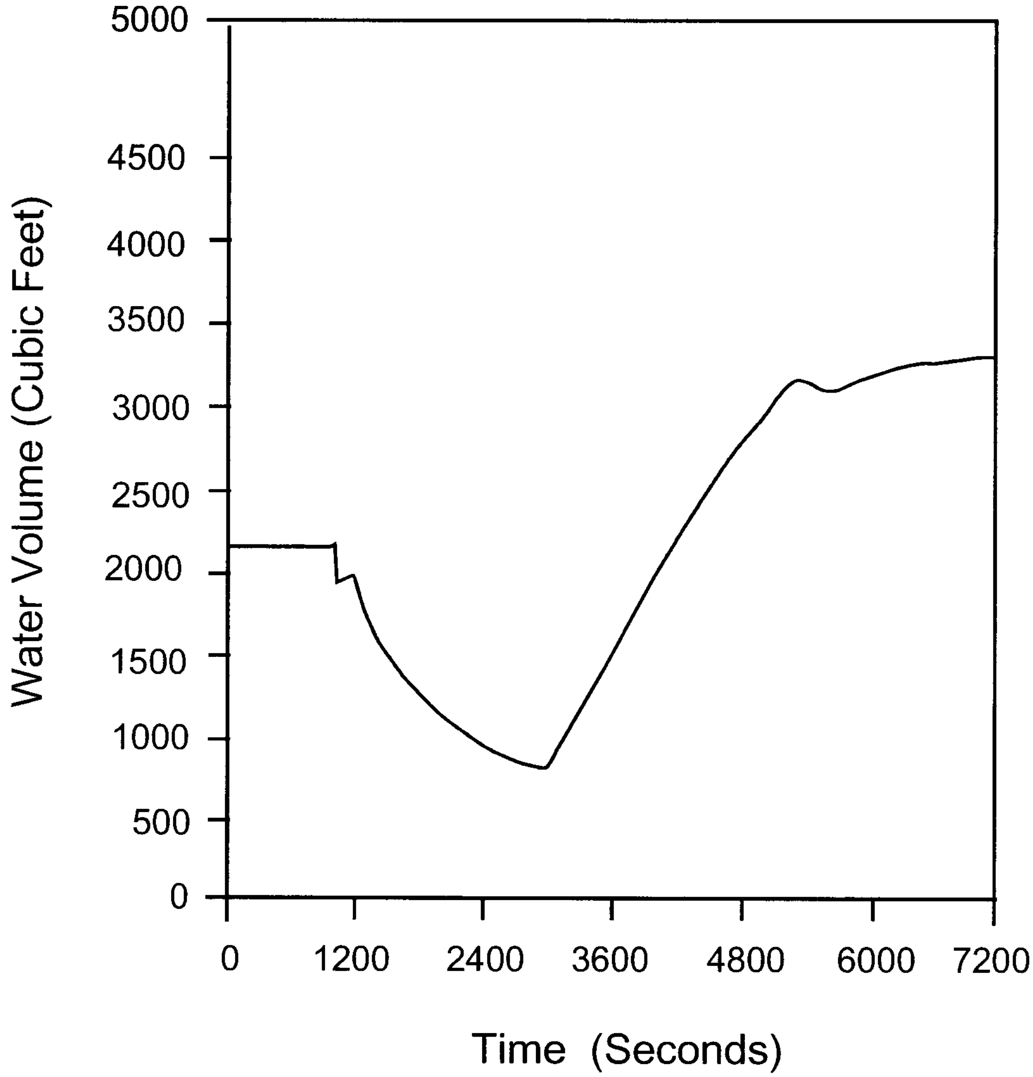


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Faulted Steam Generator Break Flow Rate to the Result of Steam Generator Tube Rupture
	Figure 15.6-58

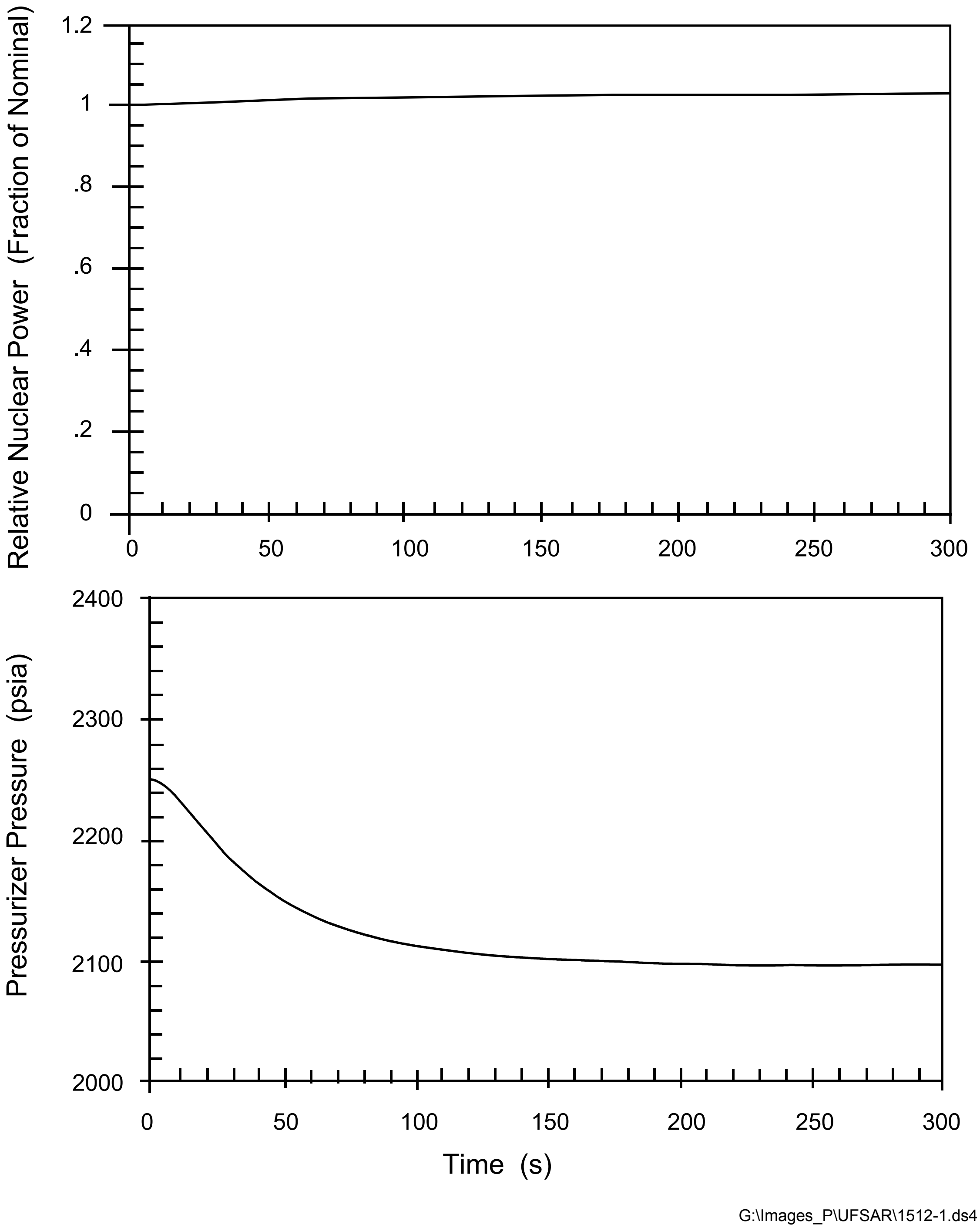


Steam Generator Mass as the Result of Steam Generator Tube Rupture

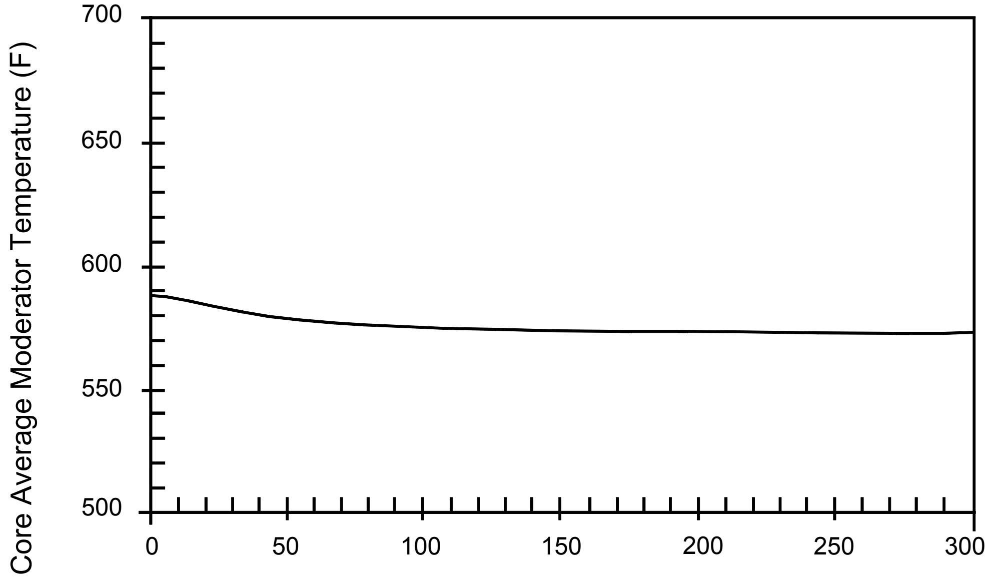
Figure 15.6-59

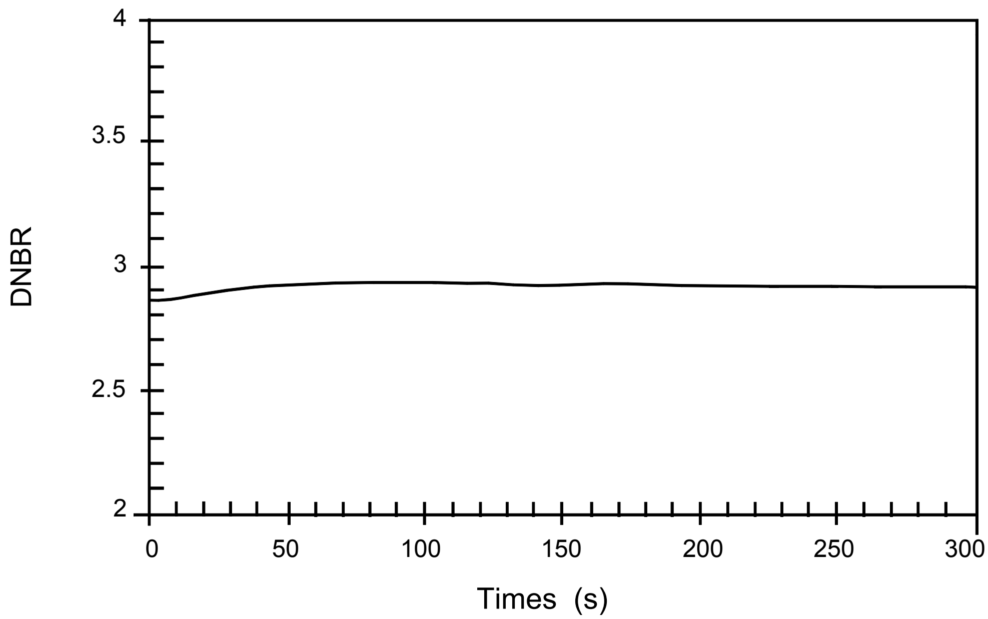


5000
4500
4000
3500
3000
2500
2000
1500
1000
500
0
Water Volume (Cubic Feet)
0
1200
2400
3600
4800
6000
7200
Time (Seconds)



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Relative Power and Pressurizer Pressure Transients for an Excessive Load Increase Event (manual rod control; minimum reactivity coefficients)	
		Figure 15-1-2 Sh. 1 of 2





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SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Core Average Moderator Temperature and DNBR Transients for an Excessive Load Increase Event (manual rod control; minimum reactivity coefficients)	
		Figure 15-1-2 Sh. 2 of 2