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14.0 SAFETY ANALYSES

This section evaluates the safety aspects of Turkey Point Unit 3 and Unit 4 and demonstrates that the units can be operated safely and that exposures from credible accidents do not exceed the guidelines of 10 CFR 50.67. Each unit is designed and licensed at the uprated reactor power level of 2644 MWt and has been analyzed at the conditions associated with that power. The site and engineered safety features are evaluated and presented for both units operating at this rating. This section is divided into three subsections, each dealing with a different behavior category: C26

Core and Coolant Boundary Protection Analysis, Section 14.1

The abnormalities presented in Section 14.1 have no off-site radiation consequences.

Standby Safety Features Analysis, Section 14.2

The accidents presented in Section 14.2 are more severe and may cause release of radioactive material to the environment.

Rupture of a Reactor Coolant Pipe, Section 14.3

The rupture of a reactor coolant pipe, the accident presented in Section 14.3, is the basis for the design of engineered safety features. Even for this accident, the unit design meets the guidelines of 10 CFR 100.

Parameters and assumptions that are common to various accident analyses are described below to avoid repetition in subsequent sections.

Containment Bulk Ambient Temperature

The specific effects of elevated containment bulk temperature above the normal design value of 120°F, with a limit of 125°F for up to two weeks per year, was evaluated with regard to structural integrity, cable ampacities, environmental qualification of equipment, and effect on the conclusions of the accident analysis of this chapter. These effects are discussed in the appropriate sections, and were found to have slight or negligible impact while being well within the existing design limitations of the plant.

Steady State Errors

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

Power (Reactor Core)	$\pm$0.3% for calorimetric error.
Core Inlet Temperature	$\pm 6^\circ$F for deadband and measurement error.
Primary Pressure	$\pm$53 psi for steady state fluctuation and measurement error.

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Hot Channel Factors

Unless otherwise stated in the section describing specific accidents, the hot channel factors used are:

F_Q (heat flux hot channel factor) = 2.40

$F_{\Delta H}$ (enthalpy rise hot channel factor) = 1.65/1.352 (Upgrade Fuel/DRFA Fuel)

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The incore instrumentation system will be available to verify the actual hot channel factors and core power distributions at various times in the core life.

Reactor Trip

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the control rods, which then fall into the core. In order to provide additional assurance of tripping the reactor trip breakers, the reliability is enhanced by using the shunt trip attachments to
open the reactor trip breakers automatically. There are various instrumentation delays associated with each tripping function, including delays in signal actuation, in opening the trip breakers, and in the release of rods by the control rod drive mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. The maximum time delay assumed for each tripping function is as follows:

Tripping Function	Time Delay (Seconds)	Maximum Trip Point Assumed for Analysis
Overpower (nuclear)	0.5	115%
Power Range Flux (low Setting)	0.5	35%
Overtemperature ΔT	2.0*	Variable
Overpower ΔT	2.0*	Variable
High Pressurizer Pressure	2.0	2440 psig
Low Pressurizer Pressure	2.0	1790 psig
High Pressurizer Level	Note 1	100% of pressurizer level span
Low Reactor Coolant Flow		
(from loop flow detectors)	1.0	84.5% loop flow
(from undervoltage)	2.5	Not applicable
(from frequency)	0.6	55 Hz
Turbine Trip	2.0	Not applicable
Low-Low Steam Generator Level		
(Loss of Normal Feedwater)	2.0	4% of narrow range level span
(Feedwater System Pipe Break)	2.0	0% of narrow range level span

NOTE :

* Time delay given only includes channel electronics, trip logic and gripper release. Additional delays in the trip are a 25 second/3 second T_{avg} lead/lag, 2 second filter on the vessel T_{avg} signal, and a 2 second filter on the vessel ΔT signal.

1. Although this function is not explicitly modeled in any non-LOCA transient, it is assumed to be operable in the uncontrolled RCCA bank withdrawal at power event to preclude pressurizer filling.

The negative reactivity insertion following a reactor trip is a function of the acceleration of the control rods and the variation in rod worth as a function of rod position. Control rod positions after trip have been determined experimentally as function of time using an actual prototype assembly under simulated flow conditions. The resulting rod positions were combined with rod worths to define the negative reactivity insertion as a function of time, according to Figure 14-1.

The maximum nuclear overpower trip point assumed for all analyses is 115%. The trips will be calibrated at power such that the calibration error is the calorimetric error of ± 0.3 percent. The design allowance for non-repeatable errors is ±6 percent. Non-repeatable errors include both instrument drift and errors due to process changes such as control rod motion since both are observable as an error between the indicated signal and the known power from calorimetric measurement. In summary, the trip setpoints, established in the Technical Specifications, are less than the trip values assumed in the analyses to ensure that trip occurs within the assumed value when including the design error allowance.

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Positive Moderator Temperature Coefficient Power Operation

Analyses contained in Chapter 14 are based on a most positive moderator temperature coefficient of +7 pcm/°F from 0% to 70% Rated Thermal Power, ramping to 0 pcm/°F at 100% Rated Thermal Power.

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FPL Response to NRC Generic Letter (GL) 93-04

GL 93-04 was issued requesting information pertaining to the single failure of the Rod Control System with respect to the General Design Criteria (GDC) 25 (Draft GDC 31), which requires that acceptable fuel design limits not be exceeded for any single malfunction of the reactivity control system. This GL was in response to the event that occurred at Salem Unit 2 on May 27, 1993, when a withdrawal of a single Rod Cluster Control Assembly (RCCA) occurred when an insert command was given. A Westinghouse Owners Group (WOG) program was initiated to evaluate the event and provide an industry response. The program concluded that the licensing basis continued to be met but recommended revising the Control Rod Drive Mechanism (CRDM) current order timing to enhance the basis for that determination.

PC/M's 94-111 (Unit 3) and 95-087 (Unit 4) implemented this change to the CRDM's. The revised CRDM current order timing ensures that an asymmetric rod withdrawal cannot occur due to a single failure in the rod control circuitry. The effects of the Salem type failure (i.e., simultaneous insert and withdrawal signals) have been altered to ensure more predictable and conservative consequences. Specifically, all rods in a selected group/bank will now insert in the presence of a failure that causes simultaneous insert and withdrawal commands. However, these failure modes have been analyzed (Reference 1) and shown to be bounded by the consequences of other Condition II events already analyzed in the FSAR (specifically, RCCA Misalignment and Dropped Rod events, Section 14.1.4)

<u>Response to NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation"</u>

On October 8, 2009, the NRC issued NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," which noted that irradiation damage and the progressive buildup of fission products in the fuel pellets result in reduced thermal conductivity of the pellets. Data was collected from an instrumented assembly at the Halden ultra-high-burnup experiment during the 1990s, which indicated steady degradation in the thermal conductivity of uranium fuel pellets with increasing exposure. This data indicated a degradation of approximately 5 to 7 percent for every 10 gigawatt-days per metric ton of exposure. The NRC expressed concern that some vendors might still be using codes for safety analyses that do not account for this phenomenon and therefore may produce non-conservative results. As a result of recent information presented to the NRC on December 6, 2011, FPL was asked to address the impact of Thermal Conductivity Degradation (TCD) on the Turkey Point Unit 3 and Unit 4 EPU safety analyses. C26

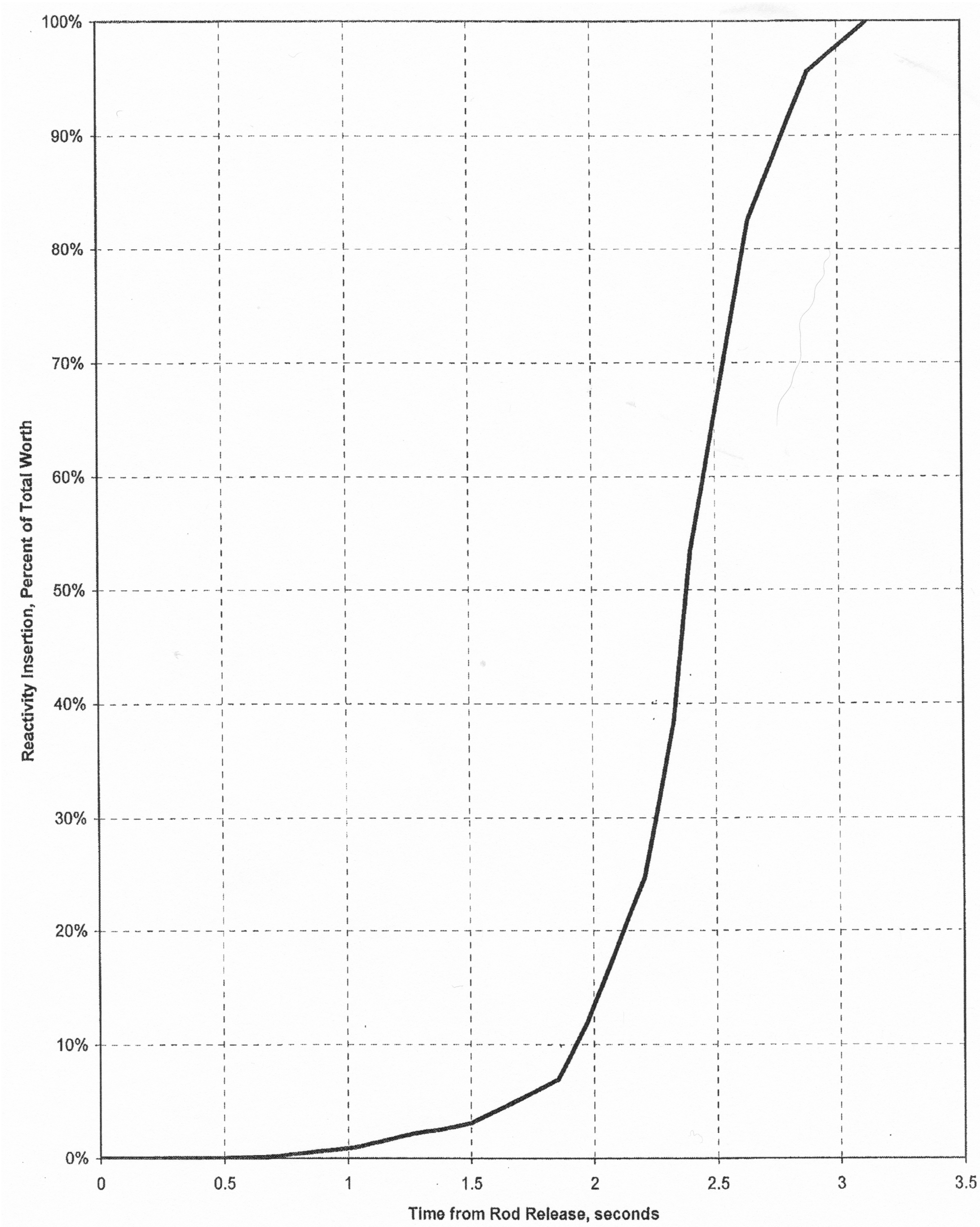
An evaluation of the impact of TCD on the non-LOCA safety analyses performed to support EPU for Turkey Point Units 3 and 4 was performed. This evaluation concluded that the results generated using the RETRAN or LOFTRAN computer codes remain valid for evaluating EPU conditions, even with inclusion of effects associated with TCD. For the non-LOCA analyses performed using the TWINKLE and FACTRAN computers codes (Rod Withdrawal from Subcritical (RWFS) and Rod Ejection) sufficient margins are available in the results to compensate for any reductions in margin associated with consideration of TCD. Explicit calculations were performed for Rod Ejection and are reflected in the results reported in the Rod Ejection section. With regard to the Boron Dilution accident, none of the input parameters used in the analysis (including the critical boron concentrations) have been changed because of TCD; so TCD evaluations were completed for the Locked Rotor Accident. The DNBR and PCT results, including TCD under Fuel Thermal-Hydraulic Design scope, still show adequate margin to the limits. C26

REFERENCES

1. Westinghouse WCAP-13864,"Rod Control System Evaluation Program," Revision 1-A, approved November 10, 1994.

2. Westinghouse WCAP-14276,"Turkey Point Units 3 and 4 - Uprating Licensing Report," Revision 1, December 1995.

3. Westinghouse Letter 96-FP-G-0003,"Turkey Point Unit 4 Cycle 15 Final RSE, Revision 1, Tavg/Power Coastdown," January 17, 1996.



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TURKEY POINT PLANT UNITS 3 & 4

NEGATIVE REACTIVITY INSERTION VERSUS TIME FOR REACTOR TRIP AT EPU CONDITIONS

FIGURE 14-1

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

For the following abnormalities and transients, the reactor control and protection system is relied upon to protect the core and reactor coolant boundary from damage:

a) Uncontrolled RCCA withdrawal from a Subcritical Condition

b) Uncontrolled RCCA Withdrawal, at Power

c) Rod Cluster Control Assembly (RCCA) Drop

d) Chemical and Volume Control System (CVCS) Malfunction[1]

e) Startup of an Inactive Reactor Coolant Loop

f) Excess Feedwater Incident

g) Excessive Load Increase Incident

h) Loss of Reactor Coolant Flow

i) Loss of External Electrical Load

j) Loss of Normal Feedwater

k) Loss of All Normal A-C Power to the Station Auxiliaries

l) Likelihood and Consequences of Turbine Generator Overspeed C26

m) Accidental Depressurization of the Reactor Coolant system C26

The above abnormalities and transients, except for Likelihood and Consequences of Turbine Generator Overspeed and Loss of Reactor Coolant Flow, are classified as Condition II per ANS-51.1/N18.2-1973 (Reference 5). Likelihood and Consequences of Turbine Generator Overspeed is not within the scope of Reference 5. C26

Loss of Reactor Coolant Flow includes: Partial Loss of Reactor Coolant Flow, complete Loss of Reactor Coolant Flow and Reactor Coolant Pump Locked Rotor. Partial Loss of Reactor Coolant Flow is considered Condition II, Complete Loss of Reactor Coolant Flow is considered Condition III, and Reactor Coolant Pump Locked Rotor is considered Condition IV per Reference 5.

All reactor protection criteria are met presupposing the most reactive RCC assembly in its fully withdrawn position. Trip is defined for analytical purposes as the insertion of all full length RCC assemblies except the most reactive assembly which is assumed to remain in the fully withdrawn position. This is to provide margin in shutdown capability against the remote possibility of a stuck RCC assembly condition existing at a time when shutdown is required.

Instrumentation is provided for continuously monitoring all individual RCC assemblies together with their respective group position. This is in the form of a deviation alarm system. If the rod should deviate from its intended position the reactor would then be shut down in an orderly manner and the condition corrected. Such occurrences are expected to be extremely rare based on operation and test experience to date.

In summary, reactor protection is designed to prevent cladding damage in all transients and abnormalities listed above. The most probable modes of failure in each protection channel result in a signal calling for the protective trip. Coincidence of two out of three (or two out of four) signals is required where single channel malfunction could cause spurious trips while at power. A single component or channel failure in the protection system itself coincident with one stuck RCCA is always permissible as a contingent failure and does not cause violation of the protection criteria.

Event classification and acceptance criteria are based on ANS-51.1/18.2-1973 (Reference 5). C26

14.1.1 UNCONTROLLED RCCA WITHDRAWAL FROM A SUBCRITICAL CONDITION

An RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical or at power. The "at power" case is discussed in Section 14.1.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by RCCA withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is with RCCA withdrawal. RCCA motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The rod cluster drive mechanisms are wired into preselected banks, and these bank configurations are not altered during core life. The rods are therefore physically prevented from withdrawing in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two rod banks with the maximum combined worth at maximum speed which is well within the capability of the protection system to prevent core damage.

Should a continuous RCCA withdrawal be initiated and assuming the source and intermediate range indication and annunciators are ignored, the transient will be terminated by the following automatic protective functions.

a) Source range flux level trip - actuated when either of two independent source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above the source range cutoff power level. It is automatically reinstated when both intermediate range channels indicate a flux level below the source range cutoff power level.

b) Intermediate range rod stop - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This rod stop may be manually bypassed when two out of the four power range channels indicate a power level above approximately ten percent power. It is automatically reinstated when three of the four power range channels are below this value.

c) Intermediate range flux level trip - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed, when two of the four power range channels are reading above approximately ten percent power and is automatically reinstated when three of the four channels indicate a power level below this value.

d) Power range flux level trip (low setting) - actuated when two out of the four power range channels indicate a power level above approximately 25 percent. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately ten percent and is automatically reinstated when three of the four channels indicate a power level below this value.

e) Power range flux level trip (high setting) - actuated when two out of the four power range channels indicate a power level above a preset setpoint, usually ≤109 percent of full-power. This trip function is always active.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the initial power burst results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup accident since it limits the power to a tolerable level prior to external control action. After the initial power burst, the nuclear power is momentarily reduced and then if the accident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Termination of the startup accident by the above protection channels prevents core damage. In addition, the reactor trip from high pressurizer pressure serves as backup to terminate the accident before an overpressure condition could occur.

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages. First, a spatial neutron kinetics computer code, TWINKLE (Reference 1), is used to calculate the core average nuclear power transient, including the various core feedback effects, i.e., Doppler and moderator reactivity. Next, the FACTRAN computer code (Reference 2) uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the average heat flux and temperature transients. Finally, the average heat flux calculated by FACTRAN is used in the VIPRE computer code (Reference 3) for DNBR calculations. C26

In order to give conservative results for the uncontrolled RCCA bank withdrawal from subcritical accident analysis, the following assumptions are made concerning the initial reactor conditions:

a) Since the magnitude of the nuclear power peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler Power reactivity coefficient, the least negative design value is used for the uncontrolled RCCA bank withdrawal from subcritical accident analysis.

b) The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and moderator is much longer than the nuclear flux response time constant. However, after the initial nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. Accordingly, the conservative value of 7 pcm/°F is used, since this yields the maximum rate of power increase.

c) The analysis assumes the reactor to be at hot zero power conditions with a nominal temperature of 547°F. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-to-water heat transfer coefficient, a larger specific heat of the water and fuel, and a less-negative (smaller absolute magnitude) Doppler coefficient. The less-negative Doppler coefficient reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel specific heat and larger heat transfer coefficient yields a larger peak heat flux. The analysis assumes the initial effective multiplication factor (K_{eff}) to be 1.0 since this results in the maximum neutron flux peak.

d) Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrumentation error, setpoint error, delay for trip signal activation, and delay for trip signal actuation, and delay for control rod assembly release is taken into account. The analysis assumes a 10 percent uncertainty in the power range flux trip setpoint (low setting), raising it from the nominal value of 25 percent to a value of 35 percent; no credit is taken for the source and intermediate range protection. Figure 14.1.1-1 shows that the rise in nuclear power is so rapid that the effect of error in the trip setpoint on the actual time at which the rods release is negligible. In addition, the total reactor trip reactivity is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position.

e) The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at the maximum speed (45 in/min, which corresponds to 72 steps/min).

f) The DNB analysis assumes the most-limiting axial and radial power shapes possible during the fuel cycle associated with having the two highest combined worth banks in their highest worth position.

g) The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

h) The analysis assumes two reactor coolant pumps (RCPs) to be in operation. This is conservative with respect to the DNB transient.

i) The accident analysis employs the Standard Thermal Design Procedure (STDP) methodology. The use of STDP stipulates that the Reactor Coolant System (RCS) flow rate will be based on a fraction of the Thermal Design Flow for two RCPs operating and that the RCS pressure is at a conservatively low value which accounts for uncertainty due to instrument error. Since the event is analyzed from hot zero power, the steady-state STDP uncertainties on core power and RCS average temperature are not considered in defining the initial conditions.

Results

Figures 14.1.1-1 through 14.1.1-4 show the transient behavior for a reactivity insertion rate of 75 pcm/sec with the accident terminated by reactor trip at 35% of nominal power. The rate is greater than that calculated for the two highest worth sequential control banks with both assumed to be in their highest incremental worth region.

Figure 14.1.1-1 shows the neutron flux transient. The neutron flux overshoots the full power nominal value for a very short period of time; therefore, the energy release and fuel temperature increase are relatively small. The thermal flux response, of interest for the DNB considerations, is shown in Figure 14.1.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux of much less than the nominal full power value. Figures 14.1.1-3 and 14.1.1-4 show the transient response of the hot spot average fuel and cladding inner temperatures, respectively. Note the hot spot average fuel temperature increases, but remains below the full power value. The minimum DNBR remains above the safety analysis limit value at all times.

Table 14.1.1-1 presents the calculated sequence of events. After reactor trip, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal shutdown procedures.

Conclusion

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature result in a minimum DNBR greater than the safety analysis limit value. No damage could occur to the fuel due to low temperatures (<2800°F) if compared to the fuel melting temperature limit (4800°F). Thus, no fuel damage is predicted as a result of this transient.

<u>References</u>

1. Westinghouse WCAP-8028-A, Risher, D. H., Jr., and Barry, R. F., "TWINKLE - a Multi-Dimensional Neutron Kinetics Computer Code," dated January 1975.

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

3. Westinghouse WCAP-15306-NP-A, Sung, Y. X., et al, “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999.

4. Deleted C26

5. American National Standard, ANS-51.1/N18.2-1973, “Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants”, August 6, 1973.

TABLE 14.1.1-1

SEQUENCE OF EVENTS
UNCONTROLLED RCCA WITHDRAWAL FROM SUBCRITICAL ACCIDENT

Event	Time (Sec)
Initiation of Uncontrolled RCCA Withdrawal	0.0
Power Range High Neutron Flux, Low Setpoint Reached	10.34
Peak Nuclear Power Occurs	10.47
Rods Begin to Fall	10.84
Minimum DNBR Occurs	12.31
Peak Average Clad Temperature Occurs	12.74
Peak Average Fuel Temperature Occurs	13.02
Peak Fuel Centerline Temperature Occurs	14.10

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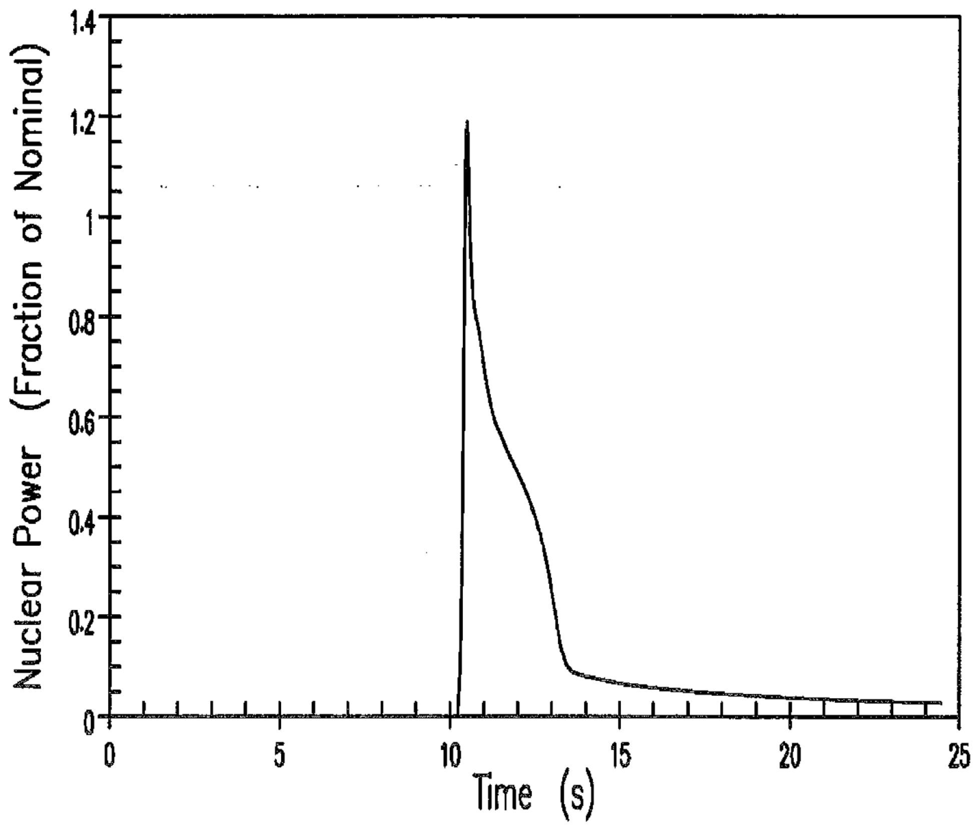

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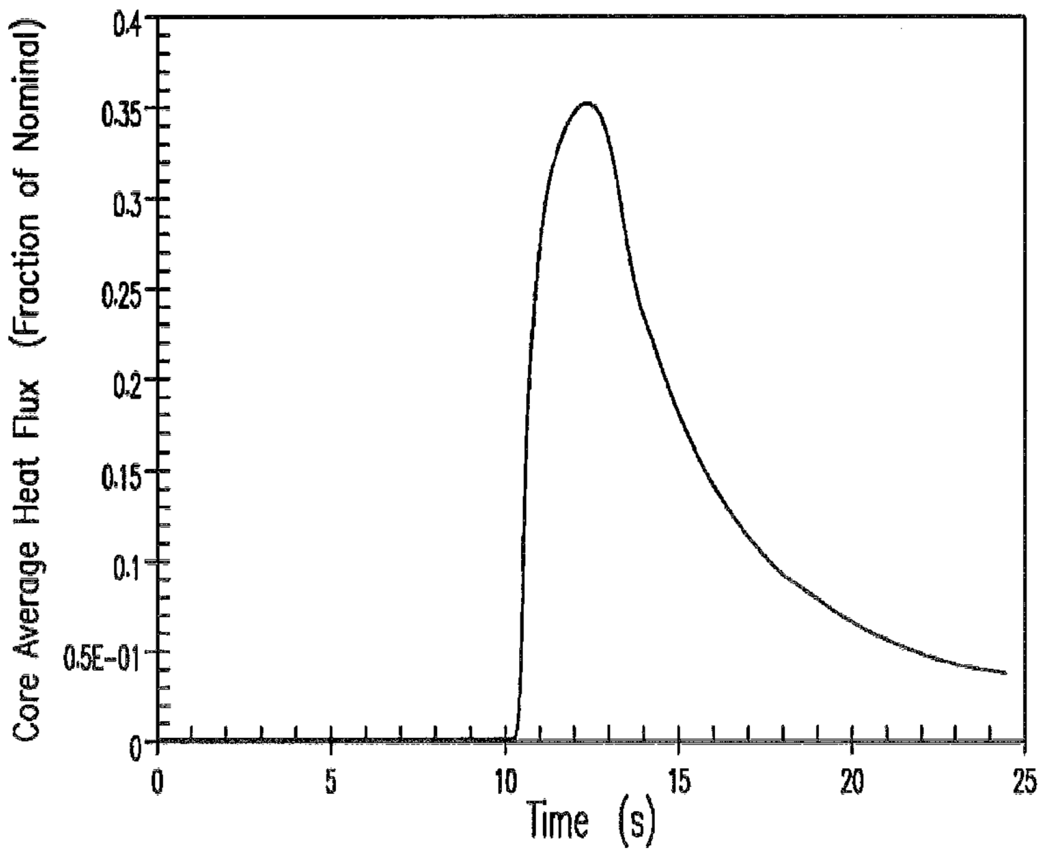
NUCLEAR POWER TRANSIENT DURING

UNCONTROLLED RCCA WITHDRAWAL

FROM SUBRITICAL

FIGURE 14.1.1-1



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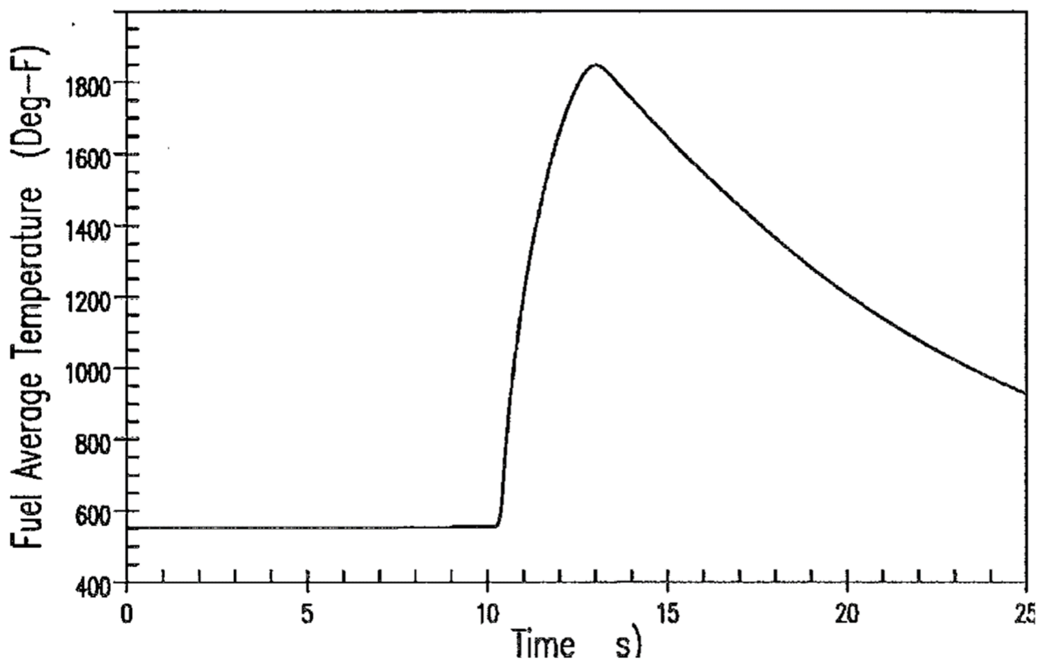
HEAT FLUX TRANSIENT DURING UNCONTROLLED RCCA WITHDRAWAL FROM SUBCRITICAL
FIGURE 14.1.1-2



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HOT SPOT AVERAGE FUEL TEMPERATURE TRANSIENT DURING UNCONTROLLED RCCA WITHDRAWAL FROM SUBCRITICAL

FIGURE 14.1.1-3

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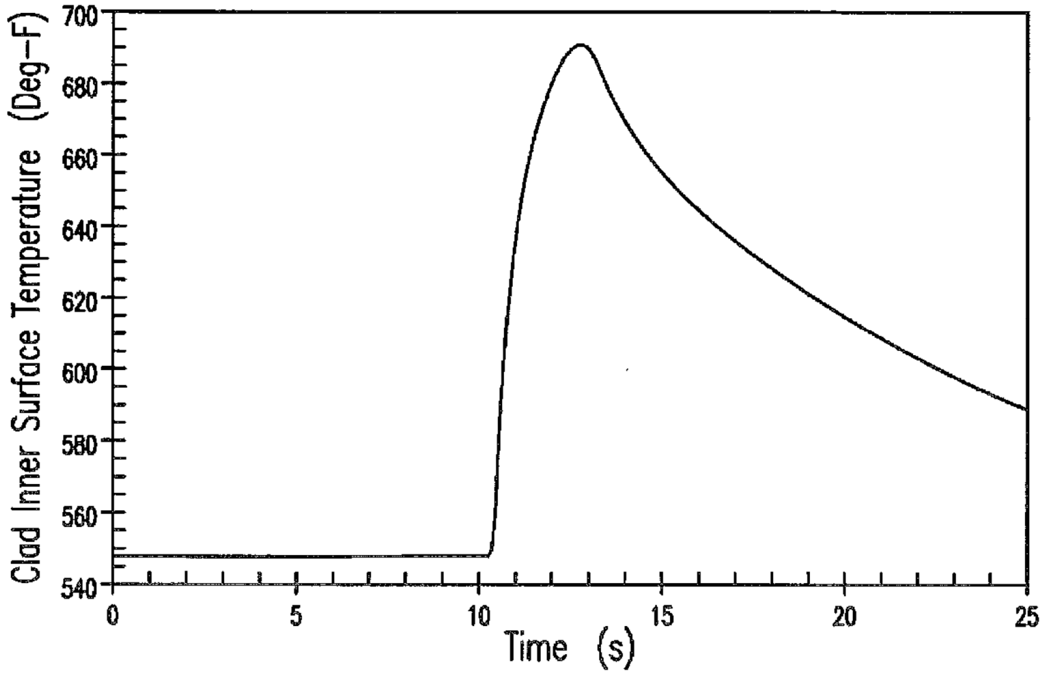

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HOT SPOT CLAD INNER TEMPERATURE TRANSIENT DURING UNCONTROLLED RCCA WITHDRAWAL FROM SUBCRITICAL

FIGURE 14.1.1-4

14.1.2 UNCONTROLLED RCCA WITHDRAWAL AT POWER

An uncontrolled RCCA withdrawal at power results in an increase in 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 reactor coolant temperature. Unless terminated by manual or automatic action, this power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, to avert damage to the fuel cladding, the Reactor Protection System is designed to terminate any such transient before the DNBR falls below the safety analysis limit value or the fuel rod linear heat generation rate (kW/ft) limit is exceeded. C26

The automatic features of the Reactor Protection System which prevent core damage in a rod withdrawal accident at power include the following:

a. Nuclear power range instrumentation actuates a reactor trip on neutron flux if two out of the four channels exceed an overpower setpoint.

b. Reactor trip is actuated if any two out of three ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with power distribution, coolant average temperature and pressurizer pressure to protect against DNB.

c. Reactor trip is actuated if any two out of three ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with coolant average temperature so that the allowable heat generation rate (kW/ft) is not exceeded.

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

e. A high pressurizer water level reactor trip actuates if any two-out-of-three level channels exceed a fixed setpoint.

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

- High neutron flux (one out of four power range).
- Overpower ΔT (two out of three).
- Overtemperature ΔT (two out of three).

Acceptance Criteria

Based on its frequency of occurrence, the uncontrolled RCCA bank withdrawal at power accident is considered a Condition II event as defined by the American Nuclear Society. The following items summarize the acceptance criteria associated with this event (Reference 3):

The critical heat flux should not be exceeded. This is ensured by demonstrating that the minimum DNBR does not go below the limit value at any time during the transient.

Pressure in the reactor coolant and main steam system should be maintained below 110% of the design pressures. With respect to peak pressure, the uncontrolled RCCA bank withdrawal at power accident is bounded by the loss of load/turbine trip analysis.

The automatic features of the Reactor Protection System provide mitigation of the uncontrolled RCCA bank withdrawal at power transient such that the above criteria are satisfied.

Method of Analysis

The purpose of this analysis is to demonstrate the manner in which the high neutron flux and overtemperature ΔT reactor trips function for various reactivity insertion rates (RIR) from different initial conditions. Reactivity coefficients, initial conditions, and effects of control functions govern which protective function occurs first.

Evaluations were performed for the uncontrolled RCCA bank withdrawal at power accident analysis to show that the peak RCS and MSS pressures did not exceed 110% of their respective design pressures. Based on the results of these evaluations, it was determined that a maximum permissible RIR of 28 pcm/second was necessary to obtain acceptable peak RCS and MSS pressures. This maximum RIR will be confirmed on a reload-specific basis for all cycles after implementation of the EPU. C26

The transient is analyzed using the RETRAN code (Reference 1). This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The code computes pertinent plant variables, including temperatures, pressures, power level and an approximation of the departure from nucleate boiling ratio (DNBR).

For an uncontrolled RCCA bank withdrawal at power accident, the following conservative assumptions are made:

a. This accident is analyzed with the Revised Thermal Design Procedure (Reference 2). Therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values and RCS flow is assumed to be the minimum measured flow. Uncertainties in initial conditions are included in the limit DNBR. C26

b. Reactivity coefficients - two cases are analyzed:

   1. Minimum Reactivity Feedback

      A +7 pcm/˚F moderator temperature coefficient (for power levels less than 70% and ramping to 0 pcm/˚F at full power) and a least-negative Doppler only power coefficient form the basis of the beginning-of-life minimum reactivity feedback assumption. C26

   2. Maximum Reactivity Feedback

      A conservatively large positive moderator density coefficient of 0.5 Δk/gm/cc (corresponding to a large negative moderator temperature coefficient) and a most-negative Doppler only power coefficient form the basis of the end-of-life maximum reactivity feedback assumption.

c. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 115 percent of nominal full power. The overtemperature ΔT trip includes all adverse instrumentation and setpoint errors. The delays for the trip actuation for the high neutron flux and overtemperature ΔT reactor trips are assumed to be the maximum values. C26

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

e. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at a conservative speed (45 in/min, which corresponds to 72 steps/min).

f. Power levels of 10%, 60%, 80%, and 100% are considered. C26

Results

Figures 14.1.2-1 and 14.1.2-2 show the transient response for a rapid RCCA bank withdrawal incident (75 pcm/sec) starting from 60% power with minimum feedback. Reactor trip on high neutron flux occurs shortly after the start of the accident. Because of the rapid reactor trip with respect to the thermal time constants of the plant, the changes in T_{avg} and pressure are such that margin in DNB is maintained.

The transient response for a slow RCCA bank withdrawal (1 pcm/sec) from 60% power with minimum feedback is shown in Figures 14.1.2-3 and 14.1.2-4. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature is consequently larger than for rapid RCCA bank withdrawal. Again, the minimum DNBR is greater than the safety analysis limit value.

Figure 14.1.2-5 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 functions (high neutron flux and overtemperature ΔT) provide protection over the whole range of reactivity insertion rates. The minimum DNBR is never less than the safety analysis limit value.

Figures 14.1.2-6, 14.1.2-7, and 14.1.2-8 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents starting at 80%, 60%, and 10% power, respectively. The results are similar to the 100% power case; however, as the initial power decreases, the range over which the overtemperature ΔT trip is effective is increased. The minimum DNBR occurs at 10% power from a reactor trip on high neutron flux. In none of these cases does the DNBR fall below the safety analysis limit value. C28

The calculated sequence of events for this accident is shown on Table 14.1.2-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

In addition, an inadvertent control rod withdrawal at power followed by a failure of the RPS to shut down the reactor is considered an ATWS event. See Section 14.1.15 for applicable discussion.

Conclusions

The high neutron flux and overtemperature ΔT reactor trip functions provide adequate protection over the entire range of possible reactivity insertion rates (i.e., the minimum value of DNBR is always larger than the safety analysis limit value). The RCS and main steam systems are maintained below 110% of the design pressures. Therefore, the results of the analysis show that an uncontrolled RCCA withdrawal at power does not adversely affect the core, the RCS or the main steam system and that all applicable acceptance criteria are met.

REFERENCES

1. Huegel, D. S., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurizer Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999 (Non-Proprietary), May 1999 C26

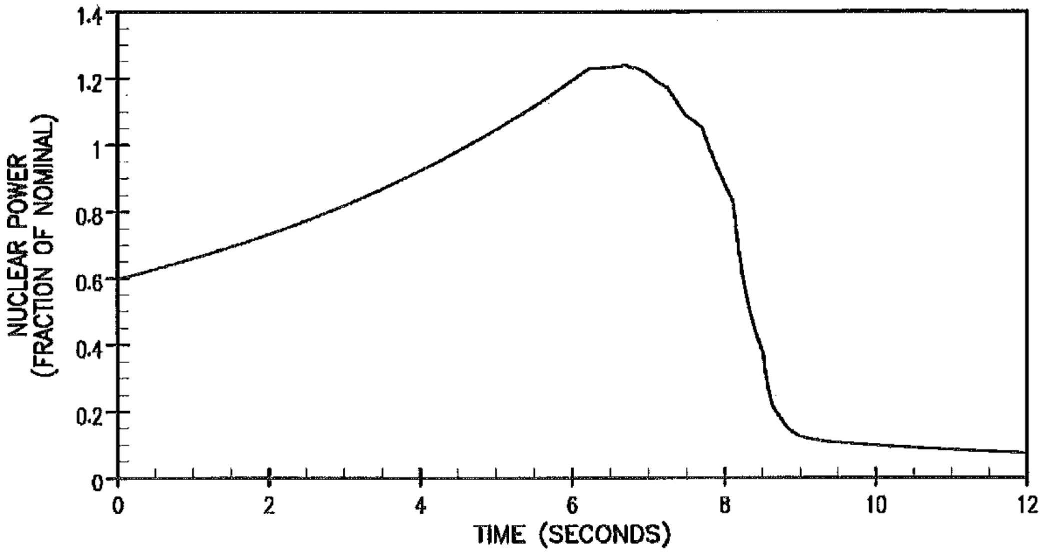
2. Westinghouse WCAP-11397-P-A (Proprietary), Friedland, A. J., and S. Ray, "Revised Thermal Design Procedure," dated April 1989.

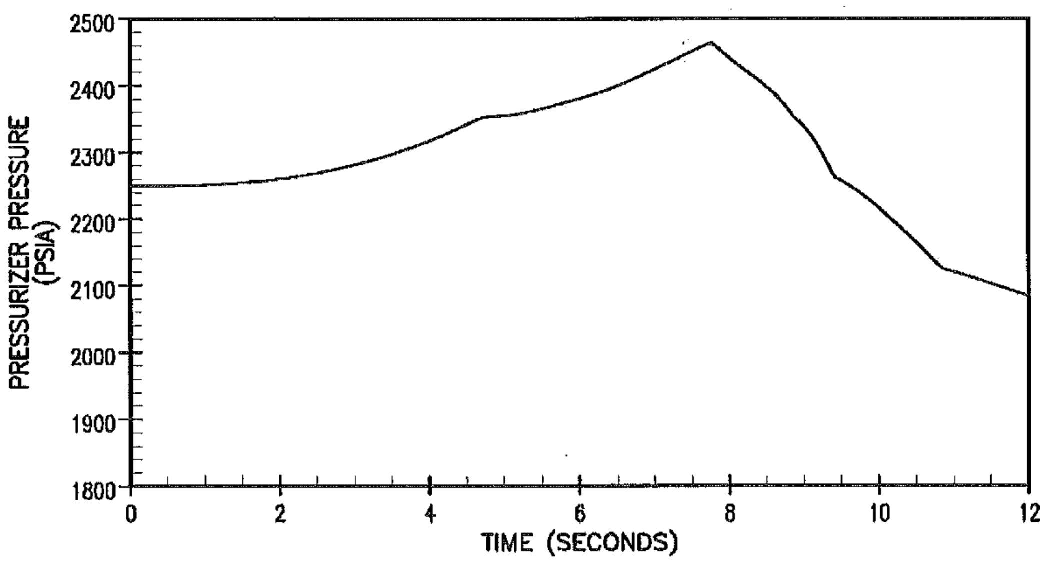
3. Westinghouse WCAP-14291, Volume 1, Section 3.2.2, "Turkey Point Units 3 and 4 Uprating Engineering Report," December 1995.

TABLE 14.1.2-1

SEQUENCE OF EVENTS
UNCONTROLLED RCCA WITHDRAWAL AT POWER ACCIDENT

Case	Event	Time (Sec)
60%Power 75 pcm/sec	Initiation of withdrawal	0.00
	High Neutron Flux Trip Setpoint Reached	5.11
	Rods begin to fall	5.61
	Minimum DNBR reached	7.20
60% Power 1 pcm/sec	Initiation of withdrawal	0.00
	Overtemperature ΔT Trip Setpoint Reached	100.10
	Rods begin to fall	102.10
	Minimum DNBR reached	103.20





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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

UNCONTROLLED RCCA
WITHDRAWAL
60% POWER - MINIMUM FEEDBACK
(75 pcm/sec withdrawal rate)

FIGURE 14.1.2-1

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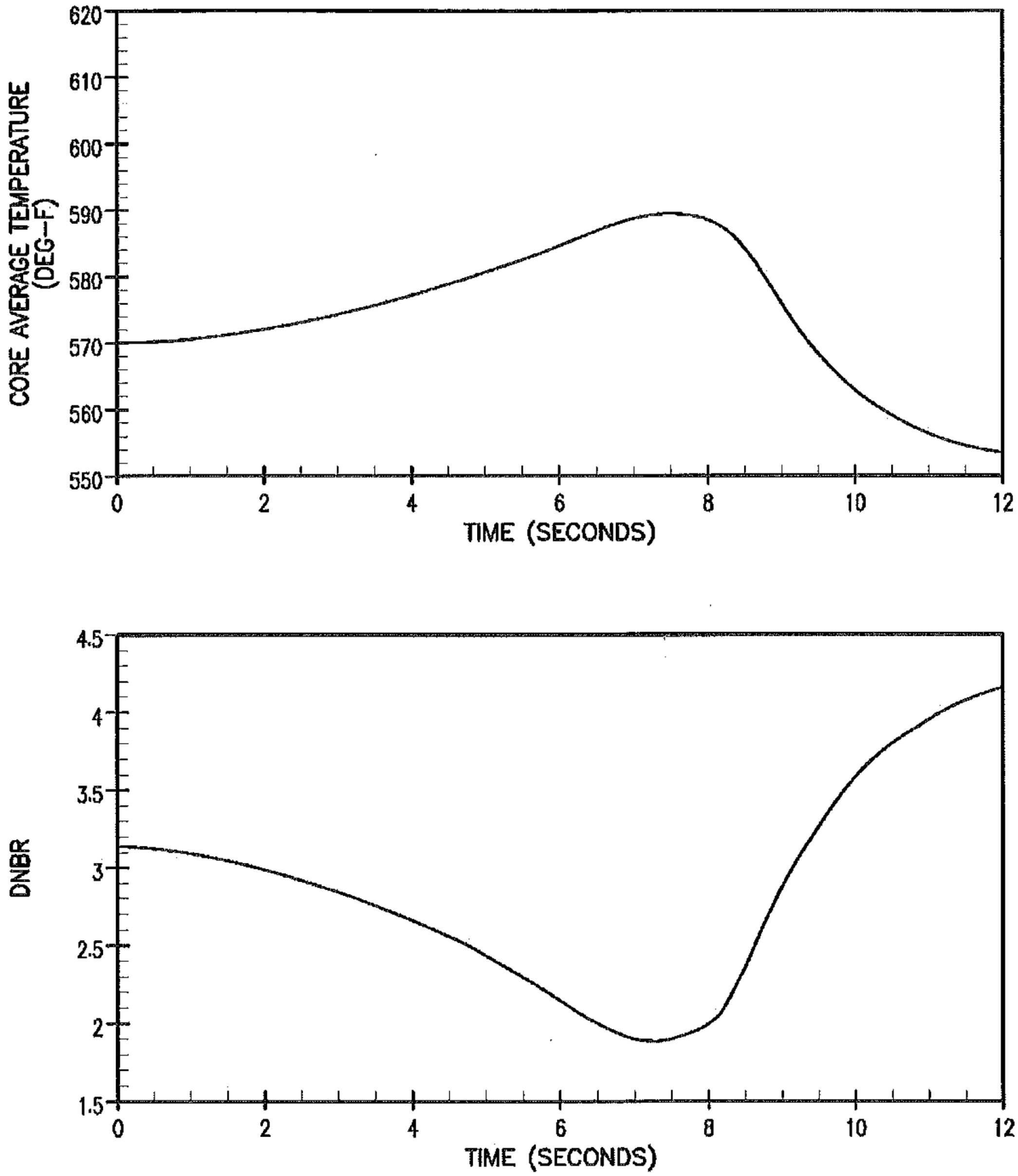

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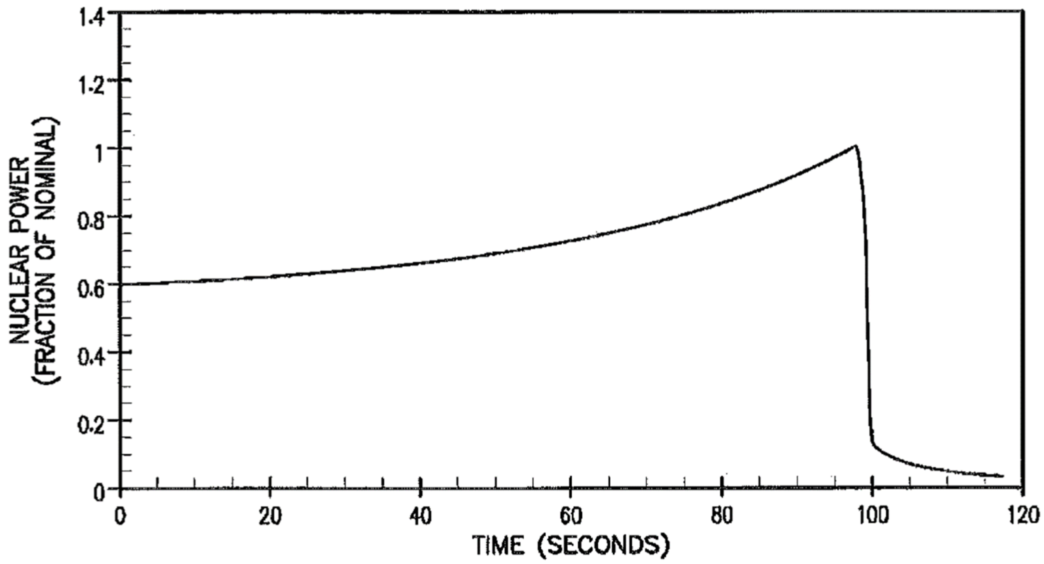
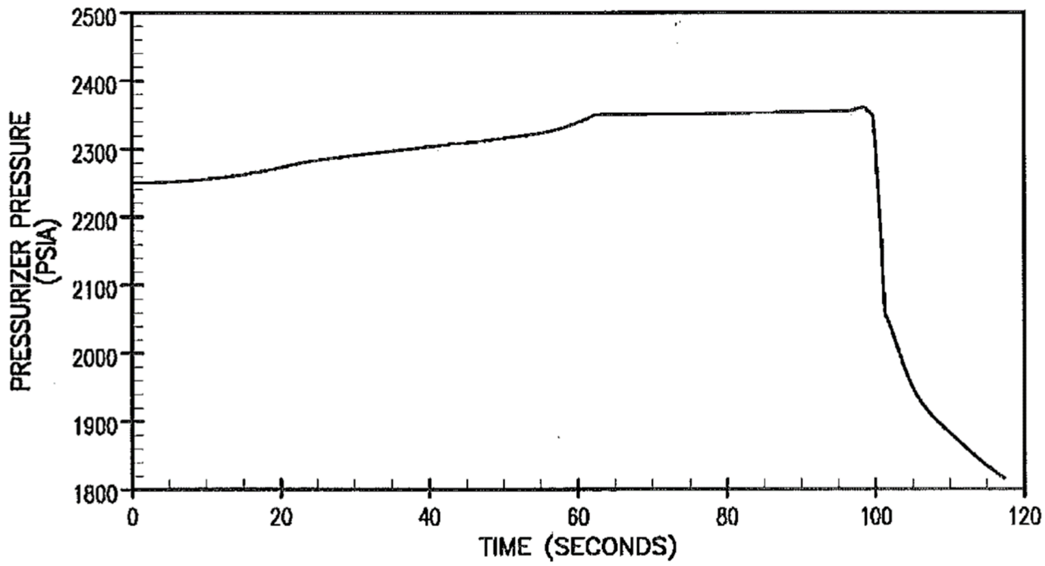
UNCONTROLLED RCCA
WITHDRAWAL
60% POWER - MINIMUM FEEDBACK
(75 pcm/sec withdrawal rate)
FIGURE 14.1.2-2





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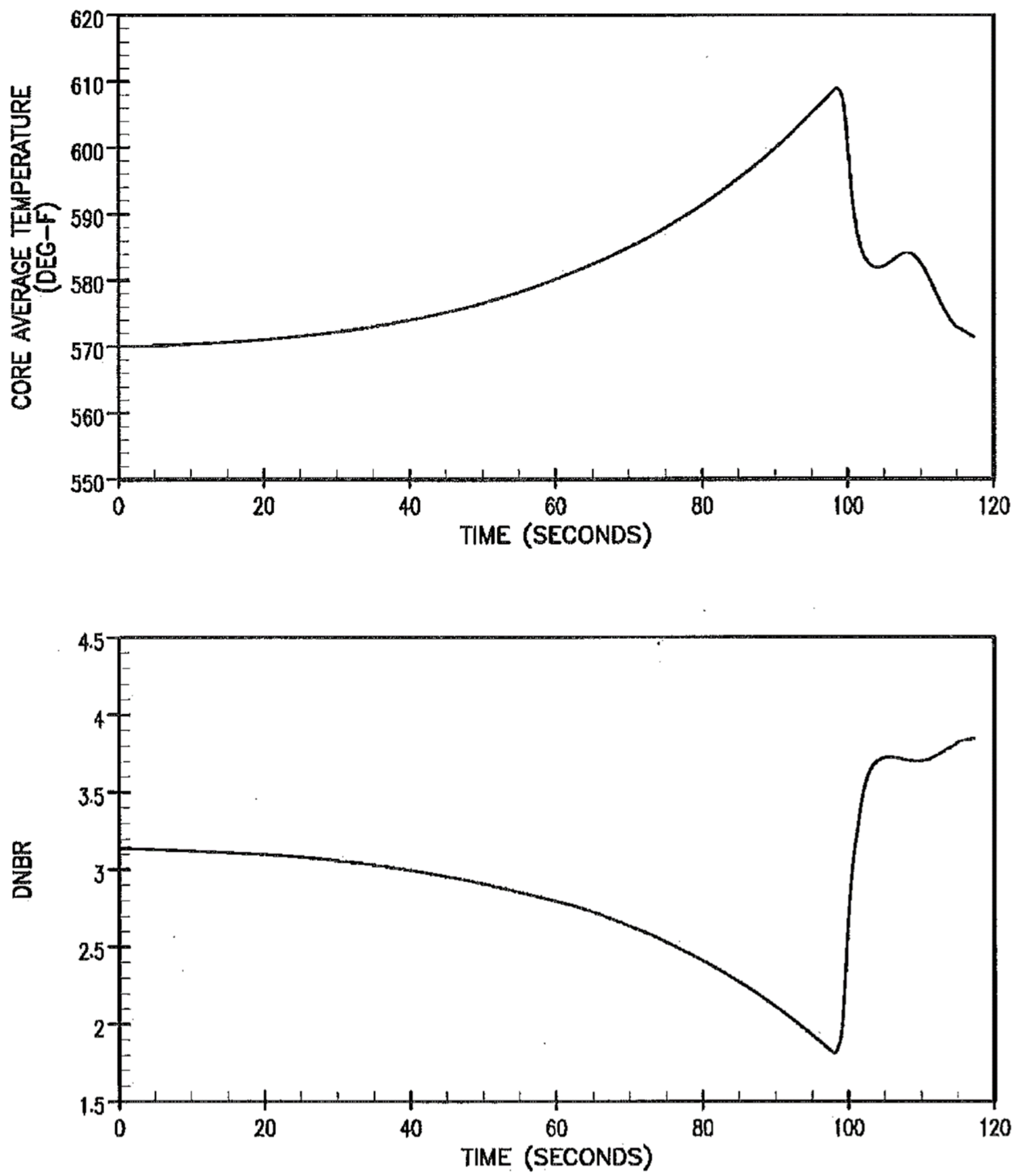
UNCONTROLLED RCCA
WITHDRAWAL
60% POWER - MINIMUM FEEDBACK
(1 pcm/sec withdrawal rate)
FIGURE 14.1.2-3



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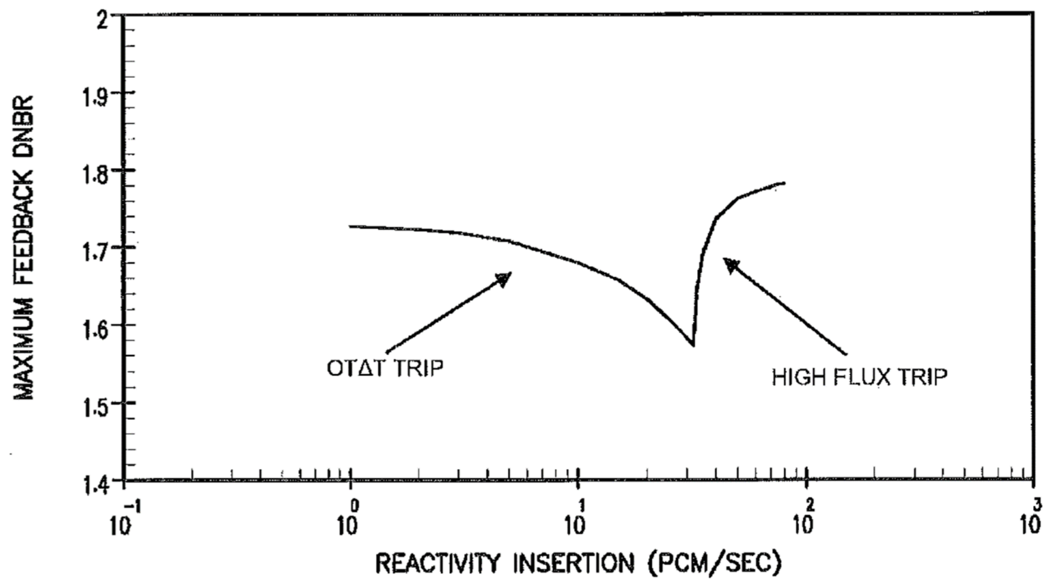
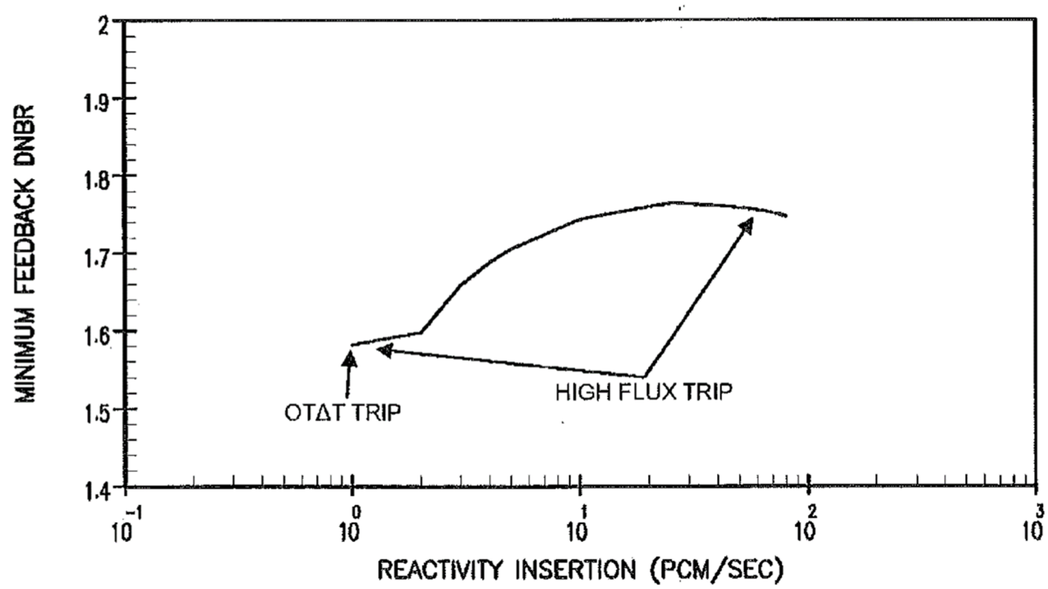
UNCONTROLLED RCCA WITHDRAWAL
60% POWER - MINIMUM FEEDBACK
(1 pcm/sec withdrawal rate)
FIGURE 14.1.2-4





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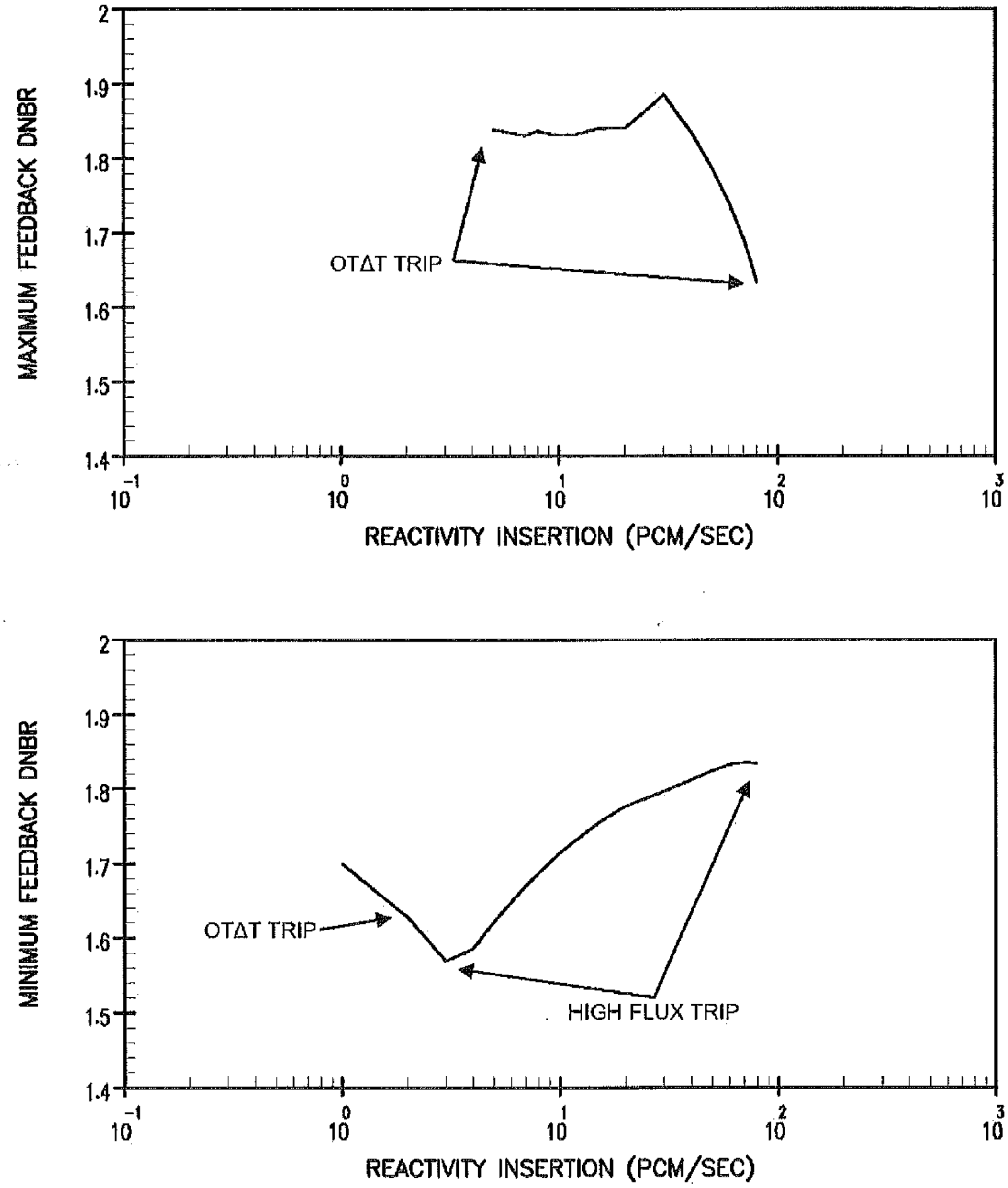
MINIMUM DNBR vs REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL AT 100 PERCENT POWER

FIGURE 14.1.2-5

C26



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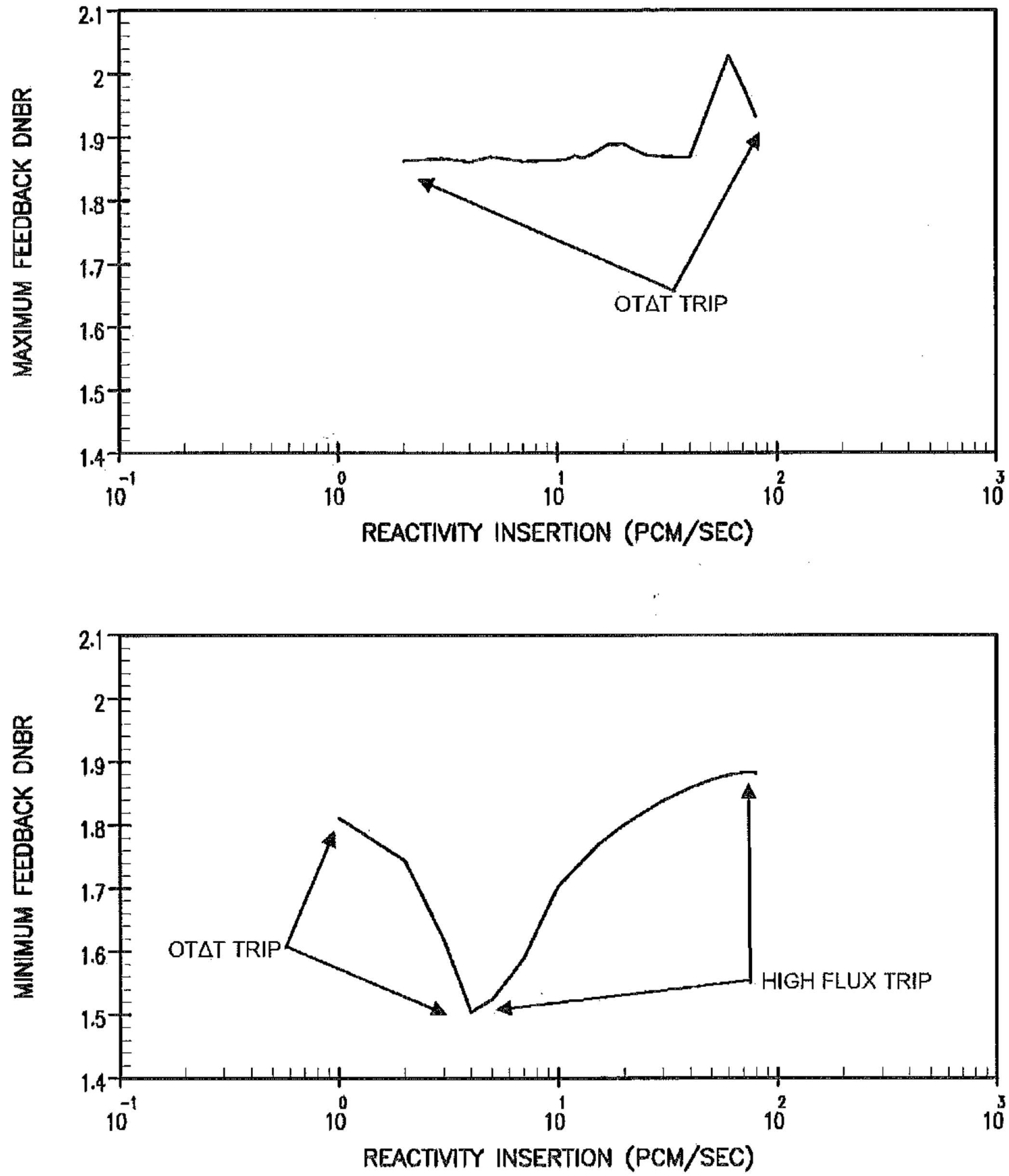
MINIMUM DNBR vs REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL AT 80 PERCENT POWER

FIGURE 14.1.2-6



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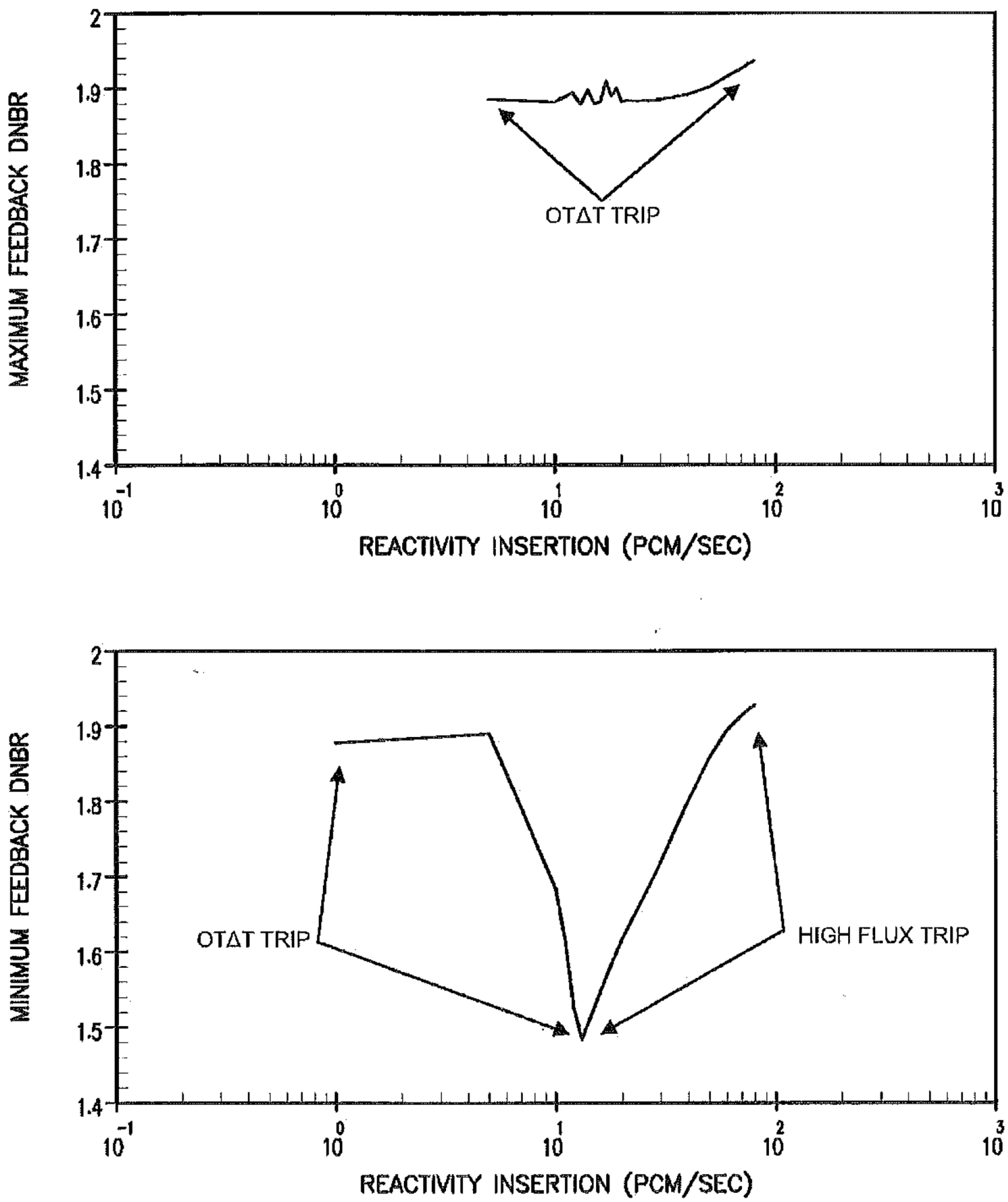
MINIMUM DNBR vs REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL AT 60 PERCENT POWER

FIGURE 14.1.2-7



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MINIMUM DNBR vs REACTIVITY INSERTION RATE FOR ROD WITHDRAWAL AT 10 PERCENT POWER

FIGURE 14.1.2-8

14.1.3 MALPOSITIONING OF THE PART LENGTH RODS

[This Section was deleted in UFSAR Rev. 0]

14.1.4 ROD CLUSTER CONTROL ASSEMBLY (RCCA) DROP

14.1.4.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A dropped RCCA event is a Condition II event that is assumed to be initiated by a single electrical or mechanical failure which causes any number and combination of RCCAs from the same group of a given bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to rapidly decrease. An increase in the hot channel factor may occur due to the skewed power distribution representative of a dropped RCCA configuration. Since this is a Condition II event, it must be shown that the DNB design basis is met for the combination of power, hot channel factor, and other system conditions which exist following the dropped RCCA(s).

If an RCCA drops into the core during power operation, it would be detected by either a rod bottom signal, by an excore detector, or both. The rod bottom signal device provides an indication signal for each RCCA. The other independent indication of a dropped RCCA is obtained by using the excore power range channel signals. This RCCA drop detection circuit is actuated upon sensing a rapid decrease in flux and is designed such that normal load variations do not cause it to be actuated.

Following a dropped RCCA event in manual rod control (or with automatic rod withdrawal defeated), the plant will establish a new equilibrium condition. The equilibrium process is monotonic, in that, there is no power overshoot without control bank withdrawal. The Turkey Point units have deleted the automatic rod withdrawal capability.

14.1.4.2 METHOD OF ANALYSIS

The transient following a dropped RCCA event is determined by a detailed digital simulation of the plant using the LOFTRAN code (Reference 1). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressure spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures and power level. Since LOFTRAN employs a point neutron kinetics model, a dropped RCCA event is modeled as a negative reactivity insertion corresponding to the reactivity worth of the dropped RCCA(s) regardless of the actual configuration of the RCCA(s) that drop. The system transient is calculated by assuming a constant turbine load demand at the initial value (no turbine runback) and no bank withdrawal. A spectrum of dropped RCCA worths from 100 pcm to 1000 pcm was analyzed.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, it is shown that the DNB design basis is met using dropped rod limit lines developed with the VIPRE computer code. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in WCAP-11394 (Reference 2). C26

Results

For a dropped RCCA event, with no automatic rod withdrawal, power may be reestablished by reactivity feedback.

Following a dropped RCCA(s) event, with no automatic rod withdrawal, the plant will establish a new equilibrium condition. Figures 14.1.4-1 and 14.1.4-2 show a typical transient response (specifically for the 100 pcm, 0 pcm/°F case) to a dropped RCCA(s). Uncertainties in the initial conditions are included in the DNB evaluation as described in Reference 2. In all cases, the minimum DNBR remains greater than the limit value.

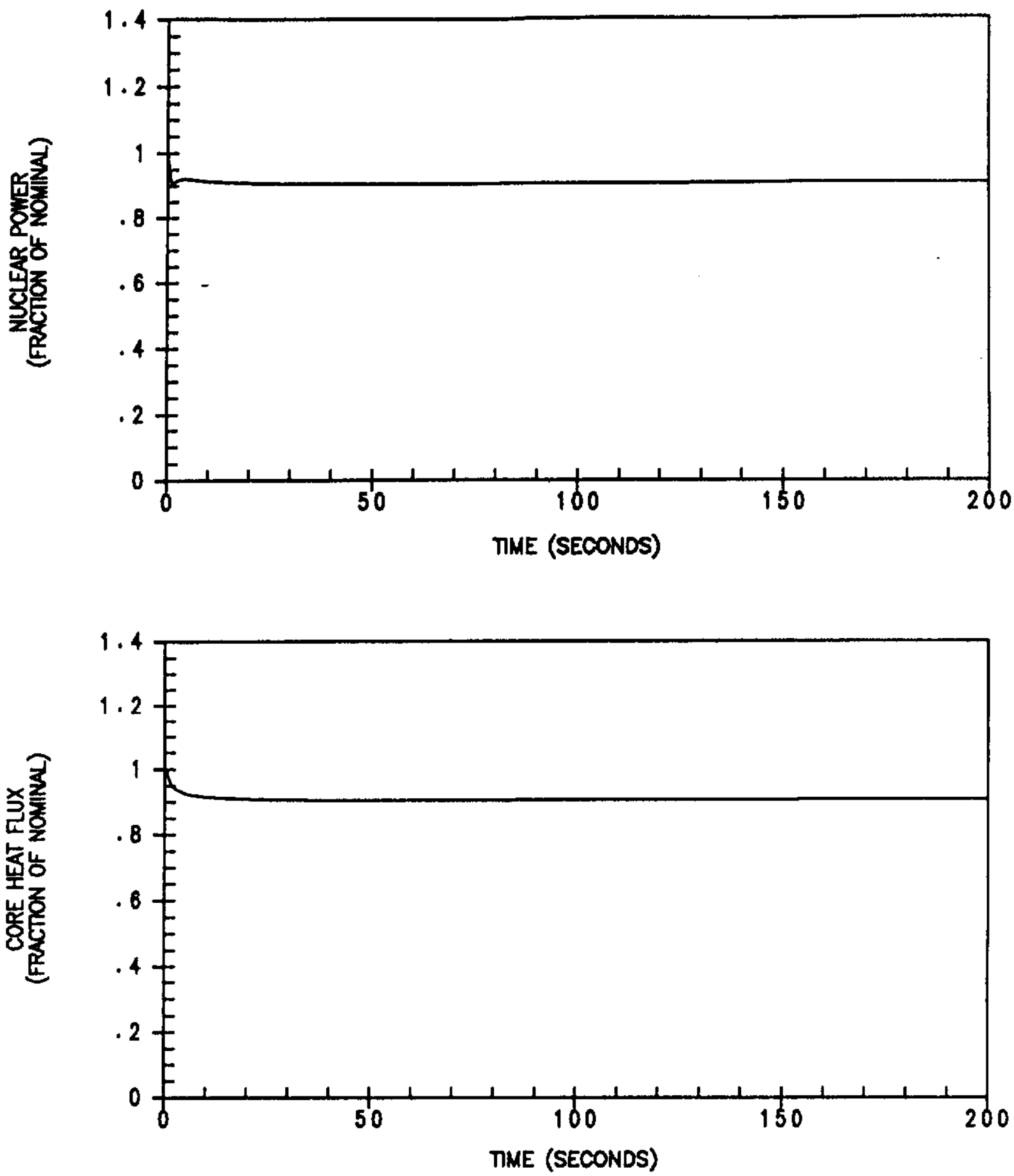
14.1.4.3 CONCLUSIONS

Following a dropped RCCA(s) event, without automatic rod withdrawal, the plant will return to a stabilized condition at less than or equal to the initial power. Results of the analysis show that a dropped RCCA event does not adversely affect the core, since the DNBR remains above the limit value for a range of dropped RCCA worths.

14.1.4.4 REFERENCES

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

2. Westinghouse WCAP-11394 (Proprietary) and WCAP-11395 (Non-Proprietary), Hassler, R. L., et al., "Methodology for the Analysis of the Dropped Rod Event," dated April 1987.



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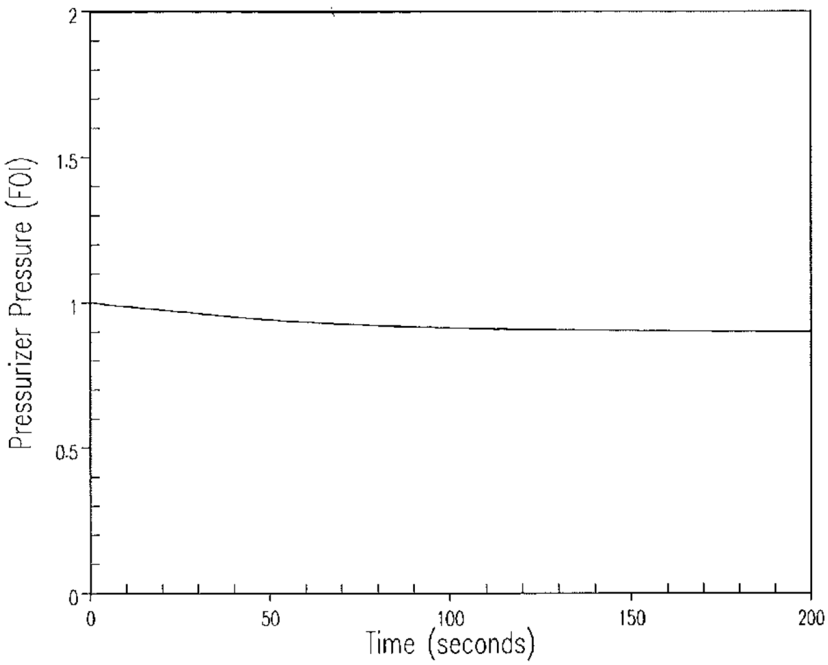
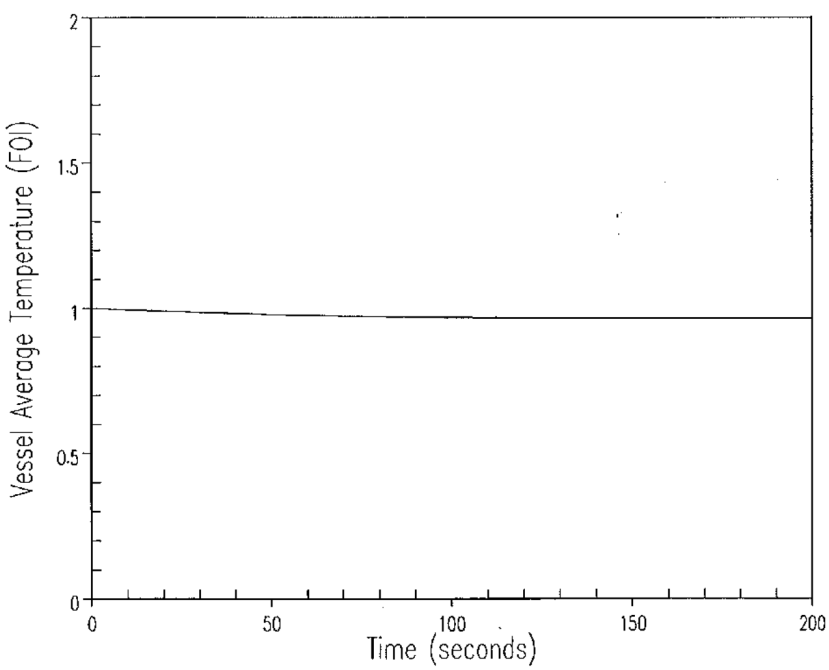
DROPPED RCCA
NUCLEAR POWER AND CORE HEAT FLUX

FIGURE 14.1.4-1





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DROPPED RCCA
PRESSURIZER PRESSURE AND
VESSEL AVERAGE TEMPERATURE
FIGURE 14.1.4-2

14.1.5 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

Reactivity can be added to the core with the Chemical and Volume Control System by feeding primary water makeup into the Reactor Coolant System (RCS) via the reactor makeup control system. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of primary water makeup to that existing in the coolant at the time. The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner. C26

There is only a single, common source of primary water makeup to the RCS from the primary water makeup system, and inadvertent dilution can be readily terminated by isolating this single source. The operation of the primary water makeup pumps which take suction from the primary water storage tank provides the only supply of makeup water to the RCS. In order for makeup water to be added to the RCS, the charging pumps must be running, in addition to the primary water makeup pumps. One of the primary water makeup pumps is operating continuously. C26

The rate of addition of unborated water makeup to the RCS is limited to the charging pump flow required to maintain pressurizer level given maximum letdown flow, maximum reactor coolant pump seal leakoff, and maximum allowable RCS leakage. A conservative dilution flow rate of 252 gpm was applied for Modes 1, 2, and 6, and a conservative dilution flow rate of 150 gpm was applied for Modes 3, 4, and 5. C26

The boric acid from the boric acid tank is blended with the primary water makeup in the blender and the concentration is determined by the preset flow rates of boric acid and primary water makeup on the Reactor Makeup Control. Two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, the RCS Makeup Control Switch must be turned to the start position. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very small.

Information on the status of the primary water makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating status of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

Method of Analysis and Results

Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power operation are considered in this analysis. Table 14.1.5-1 contains the time sequence of events of the boron dilution analysis. Table 14.1.5-2 presents the results of the boron dilution analysis. Also included in this table are pertinent analysis inputs. Perfect mixing is assumed in this analysis. This assumption results in a conservative rate of RCS boron dilution. C26

Dilution During Refueling

During refueling the following conditions exist:

1. One residual heat removal pump is running to provide continuous mixing in the reactor vessel, C26

2. The dilute mode adds water in the Volume Control Tank where the primary water is mixed with letdown before it is pumped back into the system. The alternate dilute mode adds water in the volume control tank and to the charging pump suction header. Either mode can be assumed for the analysis,

3. The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution,

4. The minimum boron concentration of the RCS is 2300 ppm, corresponding to a shutdown of at least 5 percent delta k/k with all control rods in; periodic sampling ensures that this concentration is maintained, and C26

5. Fuel which has been reloaded from the previous cycle provides a sufficient neutron source to assure the excore BF_3 detectors can monitor subcritical multiplication.

A minimum water volume in the RCS of 2951.0 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel to the mid-loop elevation. The maximum dilution flow of 252 gpm at a temperature of 39°F and uniform mixing are also considered. C26

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the containment building and the control room. C26

For dilution during refueling, the boron concentration must be reduced from 2300 ppm to approximately 1600 ppm before the reactor will go critical. This would take at least 31.2 minutes. This is ample time for the operator to recognize the high count rate signal and isolate the primary water makeup source. C26

Dilution During Cold Shutdown

In cold shutdown, the plant is being taken from a long term mode of operation, refueling, to a short term mode of operation, hot shutdown. Typically, the plant is maintained in the cold shutdown mode when reduced RCS inventory is necessary or ambient temperatures are required. The water level can be dropped to the mid-plane of the hot leg for maintenance work that requires the steam generators to be drained. Throughout an operating cycle, the plant may enter cold shutdown if reduced temperatures are required in containment or as the result of a Technical Specification action statement. The plant is maintained in cold shutdown at the beginning of a cycle for startup testing of certain systems. C26

Conditions applied in the cold shutdown boron dilution analysis are:

1. One residual heat removal pump is running to provide continuous mixing in the reactor vessel.

2. A minimum RCS water volume of 2951.0 ft^3, corresponding to a water level at mid-loop elevation.

3. A maximum dilution flow of 150 gpm at a temperature of 39°F.

4. A maximum RCS temperature of 200°F.

5. An initial RCS boron concentration of 1779 ppm, which corresponds to a shutdown margin of 1.77 percent Δk/k. This initial boron concentration represents a minimum concentration that maintains the reactor subcritical by the required shutdown margin, with all RCCAs inserted except for the most reactive RCCA.

6. A conservative, maximum boron concentration at which the reactor will return to critical with all RCCAs inserted except for the most reactive RCCA, at the most reactive cycle burnup time without xenon, is 1600 ppm.

Dilution During Hot Shutdown

In hot shutdown, the plant is being taken from a short-term mode of operation, cold shutdown, to a long term mode of operation, hot standby. Typically, the plant is maintained in the hot shutdown mode to achieve plant heatup before entering hot standby. The plant is maintained in this mode at the beginning of a cycle for startup testing of certain systems. Throughout a cycle, the plant will enter hot shutdown if reduced temperatures are required in containment or as a result of a Technical Specification action statement. In hot shutdown, reactor coolant flow is provided by either the Residual Heat Removal System or a reactor coolant pump. C26

Conditions applied in the hot shutdown boron dilution analysis are:

1. Two reactor coolant flow scenarios were considered to provide continuous mixing in the reactor vessel, one where a reactor coolant pump is running and one where a residual heat removal pump is running. With one reactor coolant pump running, the shutdown margin requirement is 1.0 percent Δk/k, and with one residual heat removal pump running, the shutdown margin requirement is 1.77 percent Δk/k.

2. Whereas having one reactor coolant pump running is adequate to include all three loops (plus the reactor vessel without the upper head) in the active mixing volume, with one residual heat removal loop in operation, the active mixing volume corresponds to the volume of the reactor vessel without the upper head, plus the volume of the Residual Heat Removal System, plus part of one reactor coolant loop (the volume of a cold leg from the Chemical and Volume Control System charging connection to the reactor vessel and the volume of a hot leg from the Residual Heat Removal System connection to the reactor vessel). The active mixing volume with a reactor coolant pump is 6987.0 ft^3, which accounts for a maximum of 10 percent steam generator tube plugging, and the active mixing volume with a residual heat removal pump is 3579.7 ft^3.

3. A maximum dilution flow of 150 gpm at a temperature of 39°F.

4. A maximum RCS temperature of 350°F.

5. An initial RCS boron concentration of 1690 ppm, corresponding to a shutdown margin of 1.0 percent $\Delta k/k$, is applied for the reactor coolant pump case, and an initial RCS boron concentration of 1778 ppm, corresponding to a shutdown margin of 1.77 percent $\Delta k/k$, is applied for the residual heat removal pump case. Each initial boron concentration represents a minimum concentration that maintains the reactor subcritical by the required shutdown margin, with all RCCAs inserted except for the most reactive RCCA.

6. A conservative, maximum boron concentration at which the reactor will return to critical with all RCCAs inserted except for the most reactive RCCA, at the most reactive cycle burnup time without xenon, is 1600 ppm.

Dilution During Hot Standby

In hot standby, the plant is being taken from one short-term mode of operation, hot shutdown, to another, startup. The plant is maintained in hot standby at the beginning of an operating cycle for startup testing of certain systems and to achieve plant heatup before entering startup mode and going critical. During cycle operation, the plant will enter hot standby following a reactor trip or as the result of a Technical Specification action statement. During hot standby operation, rod control is in manual and the rod cluster control assemblies can be either withdrawn to the hot zero power insertion limits, in preparation for entering startup mode, or fully inserted (post-trip). In hot standby, reactor coolant flow is provided by at least one reactor coolant pump. In an effort to balance the heat loss through the RCS and the heat removal of the steam generators, one more of the reactor coolant pumps may be shut off to decrease heat input into the system. In the approach to startup mode, the operator must manually withdraw the control rods and may initiate a limited dilution according to shutdown margin requirements. If the control rods are withdrawn to the hot zero power insertion limits, the dilution scenario is similar to the startup mode analysis where the failure to block the source range trip results in a reactor trip and immediate shutdown of the reactor. The dilution scenario is more limiting if the control rods are not withdrawn and the reactor is shutdown by boron to the Technical Specifications minimum requirement for hot standby. C26

Conditions applied in the hot standby boron dilution analysis are:

1. One reactor coolant pump is running to provide continuous mixing in the reactor vessel.

2. Having one reactor coolant pump running is adequate to include all three loops, plus the reactor vessel without the upper head, in the active mixing volume. The active mixing volume is 6987.0 ft^3, which accounts for a maximum of 10 percent steam generator tube plugging.

3. A maximum dilution flow of 150 gpm at a temperature of 39°F.

4. A maximum RCS temperature of 547°F. C26

5. An initial RCS boron concentration of 1800 ppm, corresponding to 1.0 percent Δk/k shutdown margin, is applied. This initial boron concentration represents a minimum concentration that maintains the reactor subcritical by the required shutdown margin, with all RCCAs inserted except for the most reactive RCCA.

6. A conservative, maximum boron concentration at which the reactor will return to critical with all RCCAs inserted except for the most reactive RCCA, at the most reactive cycle burnup time without xenon, is 1700 ppm.

<u>Dilution During Startup</u>

In this mode, the plant is being taken from one long-term mode of operation, Hot Standby, to another, Power. Typically, the plant is maintained in the Startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. Conditions applied in the startup boron dilution analysis are: C26

1. A maximum dilution flow of 252 gpm at a temperature of 39°F.

2. A minimum RCS water volume of 7619.5 ft^3. This corresponds to the active RCS volume minus the pressurizer and its surge line. C26

3. A maximum RCS average temperature (including uncertainties) of 554.8°F (5 percent power). C26

4. The initial boron concentration is 2000 ppm, corresponding to a critical concentration at the condition of hot zero power, with the RCCAs at the Rod Insertion Limits, and no Xenon.

5. The critical boron concentration following reactor trip is 1800 ppm, corresponding to hot zero power with all RCCAs inserted except the most reactive RCCA, and no Xenon. The 200 ppm change from the initial condition noted above is a conservative minimum value corresponding to a shutdown margin of 1.0 percent $\Delta k/k$. C26

This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods, a process that takes several hours. The Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality, thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source range reactor trip after receiving P-6 from the intermediate range (nominally at 10^5 cps). Too fast a power escalation (due to an unknown dilution) would result in reaching P-6 unexpectedly, leaving insufficient time to manually block the source range reactor trip. Failure to perform this manual action results in a reactor trip and immediate shutdown of the reactor.

However, in the event of an unplanned approach to criticality or dilution during power escalation while in the Startup mode, the plant status is such that minimal impact will result. The plant will slowly escalate in power until the power range high neutron flux low setpoint is reached and a reactor trip occurs. From initiation of the event, there is greater than 15 minutes available for operator action prior to return to criticality.

Dilution at Power

In this mode, the plant may be operated in either automatic or manual rod control. Conditions applied in the at power boron dilution analysis are:

1. A maximum dilution flow of 252 gpm at a temperature of 39°F.

2. A minimum RCS water volume of 7619.5 ft^3. This corresponds to the active RCS volume minus the pressurizer and its surge line.

3. A maximum RCS average temperature (including uncertainties) of 589.0°F (upper end of the full power temperature window).

4. The initial boron concentration is assumed to be 1900 ppm, corresponding to a critical concentration at the condition of hot full power with the RCCAs at the Rod Insertion Limits, and no Xenon. C26

5. The critical boron concentration following reactor trip is 1550 ppm, corresponding to hot zero power with all RCCAs inserted except the most reactive RCCA and no Xenon. The 350 ppm change from the initial condition noted above is a conservative minimum value corresponding to a shutdown margin of 1.0 percent $\Delta k/k$.

With the reactor in automatic rod control, the power and temperature increase from boron dilution results in insertion of the control rods and a decrease in the available shutdown margin. The rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator at least 15 minutes prior to criticality. This is sufficient time to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the Overtemperature ΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution is conservatively estimated to be 2.98 pcm/sec, C26 which is within the range of insertion rates analyzed. Thus, the effects of dilution prior to reactor trip are bounded by the uncontrolled RCCA bank withdrawal at power analysis (Section 14.1.2). Following reactor trip, there is greater than 15 minutes (30.0 minutes calculated) prior to criticality. C26 This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the RCS does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

TABLE 14.1.5-1

SEQUENCE OF EVENTS OF THE BORON DILUTION ANALYSIS

Uncontrolled Boron Dilution		Time (seconds)
1. Dilution During Refueling	Dilution begins	0.0
	Shutdown margin lost (if dilution continues unabated)	>1800.0
2. Dilution During Cold Shutdown	Dilution begins	0
	Shutdown margin lost (if dilution continues unabated)	>900
3. Dilution During Hot Shutdown		
a. With RCP	Dilution begins	0
	Shutdown margin lost (if dilution continues unabated)	>900
b. With RHR	Dilution begins	0
	Shutdown margin lost (if dilution continues unabated)	>900
4. Dilution During Hot Standby	Dilution begins	0
	Shutdown margin lost (if dilution continues unabated)	>900
5. Dilution During Startup	Power range-low setpoint reactor trip due to dilution	0.0 (1)
	Shutdown margin lost (if dilution continues unabated)	>900
6. Dilution During Full-Power Operation		
a. Automatic Reactor Control	Operator receives low-low rod insertion limit alarm due to dilution	0.0
	Shutdown margin lost (if dilution continues unabated)	>900
b. Manual Reactor Control	Reactor trip on OTDT due to dilution	0.0 (1)
	Shutdown margin is lost (if dilution continues unabated)	>900

Notes:

1. Zero time corresponds to time at reactor trip, not start of the dilution event.

TABLE 14.1.5-2

SUMMARY OF BORON DILUTION ANALYSIS RESULTS
AND ANALYSIS INPUTS

C26

Mode of Operation	Dilution Flow Rate (gpm)	Active Volume (cubic feet)	Calculated Time to Criticality (minutes)
Power Operation			
Auto Rod Control	252	7619.5	32.5
Manual Rod Control	252	7619.5	30.0
Startup	252	7619.5	17.7
Hot Standby	150	6987.0	15.01
Hot Shutdown			
With RCP	150	6987.0	17.15
With RHR	150	3579.7	16.94
Cold Shutdown (Mid-loop operation)	150	2951.0	15.02
Refueling) (Mid-loop operation)	252	2951.0	31.2

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OTHER IMPORTANT ANALYSIS INPUTS

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Mode of Operation	Initial Boron Conc. (ppm)	Critical Boron Conc. (ppm)	Average Core Coolant Temp. (°F)
Power Operation			
Auto Rod Control	1900	1550	589.0
Manual Rod Control	1900	1550	589.0
Startup	2000	1800	554.8
Hot Standby	1800	1700	547
Hot Shutdown			
With RCP	1690	1600	350
With RHR	1778	1600	350
Cold Shutdown (Mid-loop operation)	1779	1600	200
Refueling (Mid-loop operation)	2300	1600	140.0

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14.1.6 START-UP OF AN INACTIVE REACTOR COOLANT LOOP

The current Turkey Point plant technical specifications (Reference 1) preclude plant operation with one or more reactor coolant loops inactive. The startup of an inactive reactor loop event was originally included in the Updated FSAR when the potential for operation with a loop out of service was allowed under plant technical specifications. Based on the current plant technical specifications which prohibit plant startup and power operation (Modes 1 and 2) with one or more loops out of service, this event was removed from the Turkey Point licensing basis as part of the plant thermal uprate evaluation (Reference 2).

REFERENCES

1. Turkey Point Technical Specifications, Section 3/4.4.1, "Reactor Coolant Loops and Coolant Circulation," License Amendment No. 137/132, effective August 28, 1991.

2. Westinghouse WCAP-14276 (Non-Proprietary), "Turkey Point Units 3 and 4 – Uprating Licensing Report," Revision 1, dated December 1995. C26

TABLE 14.1.6-1

This table has been deleted.

14.1.7 EXCESS FEEDWATER FLOW AND REDUCTION IN FEEDWATER ENTHALPY INCIDENT

The reduction in feedwater enthalpy is another means of increasing core power above full power. Such increases are attenuated by the thermal capacity in the secondary plant and in the Reactor Coolant System. The overpower-overtemperature protection (high neutron flux, overtemperature ΔT and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than the 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 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. Continuous excessive feedwater addition is prevented by the steam generator high-high level signal.

A second example of excess heat removal by the feedwater system is the transient associated with the accidental opening of the low pressure feedwater heater bypass valve which diverts flow around the low pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost, e.g., during a large sudden load decrease. In the event of accidental opening, there is a sudden reduction in inlet feedwater temperature to the steam generators. The increased subcooling will create the greater load demand on the primary system which can lead to a reactor trip.

<u>Method of Analysis</u>

This accident is analyzed using the RETRAN Code (Reference 1). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level. For the cases analyzed at full-power, the RETRAN code is used to calculate the minimum DNBR. For the cases analyzed at zero-load, the VIPRE code is used to calculate the minimum DNBR. C26 C26

The Reactor Coolant System is analyzed to demonstrate acceptable consequences in the event of a feedwater system malfunction. Feedwater temperature reduction due to low-pressure heater bypass valve actuation in conjunction with an inadvertent trip of the heater drain pump in considered. Additionally, excessive feedwater addition due to a control system malfunction or operator error that allows a feedwater control valve to open fully is considered.

Two excessive feedwater flow cases are analyzed as follows: C26

a. Accidental opening of one feedwater control valve with the reactor just critical at zero-load conditions, assuming a negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. C26

b. Accidental opening of one feedwater control valve with the reactor at full power assuming automatic and manual rod control, also assuming a conservatively large moderator density coefficient characteristic of EOL conditions.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

a. The accident initiated from hot full power is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A (Reference 2). Therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR calculated using the methodology described in Reference 2. C26

b. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 200% of nominal feedwater flow to one steam generator.

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

d. For the zero-load condition, feedwater temperature is at a conservatively low value of 35°F. C26

e. For the full power cases, an initial water level of nominal minus uncertainty in all three steam generators is modeled, while a nominal initial level is modeled for the zero power cases. C26

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

g. The feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high water level signal that closes all feedwater main control and feedwater control-bypass valves, and trips the main feedwater pumps and turbine generator. C26

Note that the steam generator overfill protection function, utilizing the Steam Generator high-high water level, is not part of the Engineered Safety Features Actuation System (ESFAS), but was added to the ESFAS Technical Specification tables without modification of the existing design. This function was specifically developed to meet commitments to the NRC criteria contained in Generic Letter 89-19, dated September 20, 1989. Although the steam generator overfill protection feature uses much of the same instrumentation as the steam generator low-low trip (reactor trip circuitry), portions of the circuitry for steam generator high-high level overfill protection may not meet all the criteria which apply to ESFAS functions. This is because the steam generator high-high level function was not originally designed to be part of the ESFAS system.

h. The 1.0 second time lag in the control logic of the turbine pressure signal to the automatic rod control system is included C26

Normal reactor control systems and engineered safety systems (e.g., Safety Injection) are not required to function. The reactor protection system may actuate to trip the reactor due to an overpower condition or a turbine trip. No single active failure in any system or component required for mitigation will adversely affect the consequences of this event.

Results

Opening of a low-pressure heater bypass valve and trip of the heater drain pumps causes a reduction in the feedwater temperature which increases the thermal load on the primary system. The reduction in the feedwater temperature is less than 60°F, resulting in an increase in the heat load on the primary system of less than 10 percent of full power. The increased thermal load due to the opening of the low-pressure heater bypass valve would result in a transient very similar (but of reduced magnitude) to the Excessive Load Increase incident presented in Section 14.1.8. Thus, the results of this event are bounded by the Excessive Load Increase event and, therefore, not presented here.

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the resulting transient is similar to but less severe than the results of the Hypothetical Steamline Break transient documented in Section 14.2.5. Because the excessive feedwater flow case with the reactor at zero power is bounded by the Steamline Break accident in Section 14.2.5, no transient results are provided in this section. It should be noted that if the incident occurs with the reactor just critical at no-load, the reactor may be tripped by the power range neutron flux trip (low setting). C26

The full-power conditions combined with EOL maximum reactivity feedback yield the largest power increase for this event. Both automatic and manual rod control are assumed at HFP. However, the results of these transients are very similar. The rod control system is not required to function for this event. A turbine trip, which results in a reactor trip, is actuated when the steam generator water level in the affected steam generator reaches the high-high level setpoint.

For all cases of excessive feedwater flow, continuous addition of cold feedwater is prevented by automatic closure of all feedwater control valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and a turbine trip on high-high steam generator water level. In addition, the feedwater pump discharge valves will automatically close upon receipt of the feedwater pump trip signal.

Following turbine trip, the reactor will automatically be tripped, either directly due to the turbine trip or due to one of the reactor trip signals discussed in Section 14.1.10 (Loss of External Electrical Load).

Transient results (see Figures 14.1.7-1 through 14.1.7-3) show the core heat flux, pressurizer pressure, core average temperature, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Steam generator water level rises until the feedwater addition is terminated as a result of the high-high steam generator water level signal. The DNBR does not drop below the limit value at any time.

Since the power level rises during this event, the fuel temperature will also rise until the reactor trip occurs. The core heat flux lags behind the neutron flux due to the fuel rod thermal time constant and, as a result, the peak core heat flux value does not exceed 115% of nominal. Thus, the peak fuel melting temperature will remain well below the fuel melting point. C26

The calculated sequence of events is shown in Table 14.1.7-1. The transient results show that the DNBR does not fall below the limit value at any time during the feedwater flow increase transient; thus, the ability of the primary coolant to remove heat from the fuel rods is not reduced. Therefore, the fuel cladding temperature does not rise significantly above its initial value during the transient.

Conclusion

The decrease in feedwater temperature transient due to an opening of the low-pressure heater bypass valve is less severe than the excessive load increase event (see Section 14.1.8). Based on the results presented in Section 14.1.8, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

For the excessive feedwater addition at power transient, the results show that the DNB ratios encountered are above the limit value; hence, no fuel damage is predicted. The DNB ratios for the rods in manual and automatic cases are almost identical, with the limiting DNBR value obtained for the rods in manual case. Additionally, the results and conclusions of the Steamline Break accident in Section 14.2.5 bound those for the Excessive Heat Removal Due to a Feedwater System Malfunction at no-load conditions. C26

REFERENCES

1. WCAP-14882-P-A (Proprietary), “Retran-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” Huegel, D. S., et al., April 1999. C26

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

3. Deleted C26

TABLE 14.1.7-1

TIME SEQUENCE OF EVENTS
FOR EXCESSIVE FEEDWATER FLOW AT FULL POWER EVENT
WITH AUTOMATIC ROD CONTROL

Event	Time (seconds)
One main feedwater control valve fails fully open	0.0
High-High Steam Generator water level signal generated	38.5
Turbine trip occurs due to High-High Steam Generator water level signal	40.9
Minimum DNBR occurs	42.2
Reactor trip on turbine trip occurs	42.9
Feedwater regulating valves close due to High-High Steam Generator water level signal	68.4

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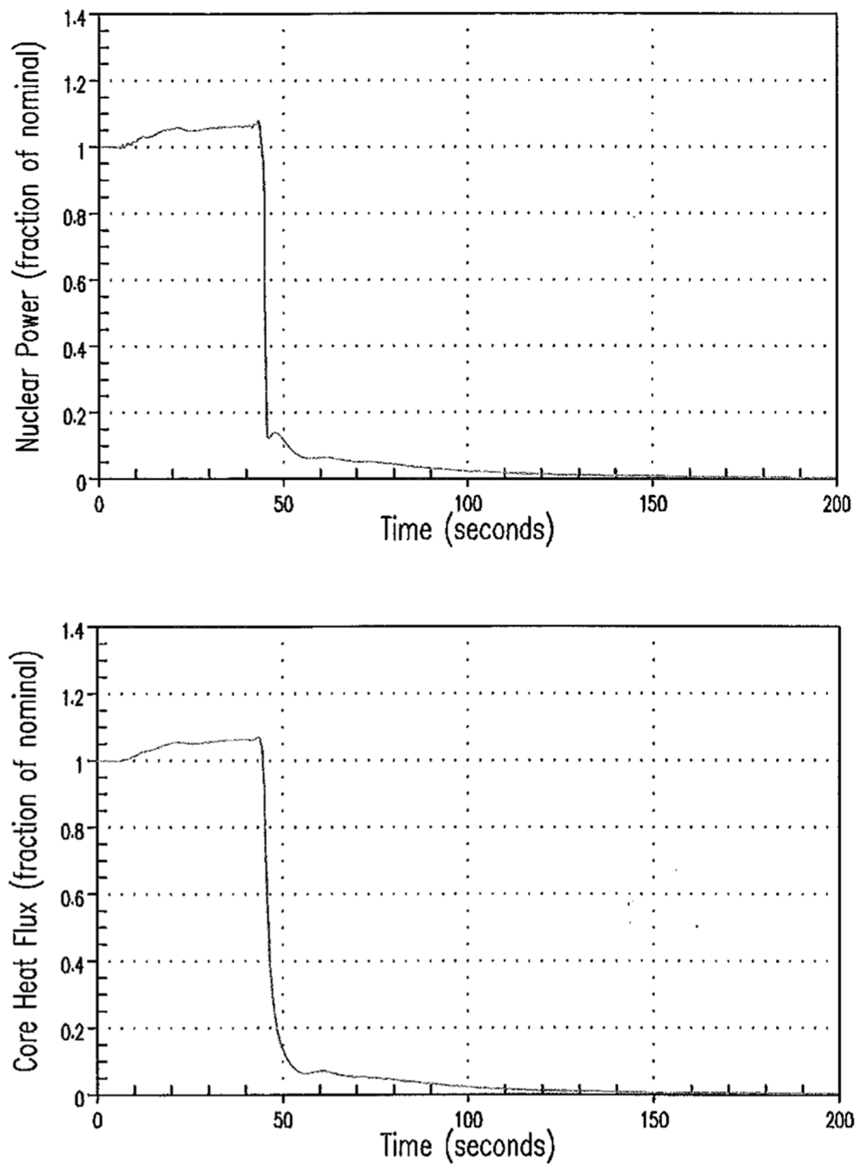

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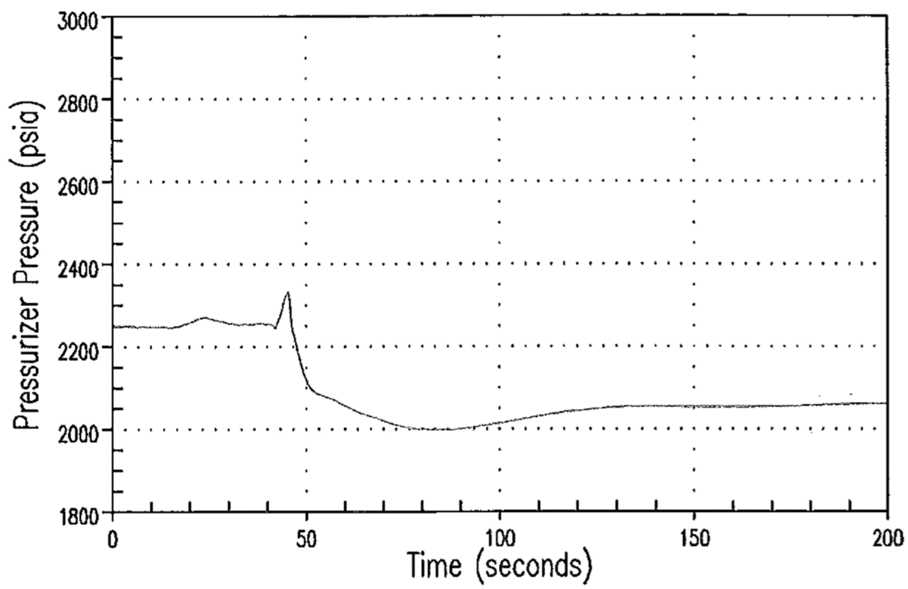
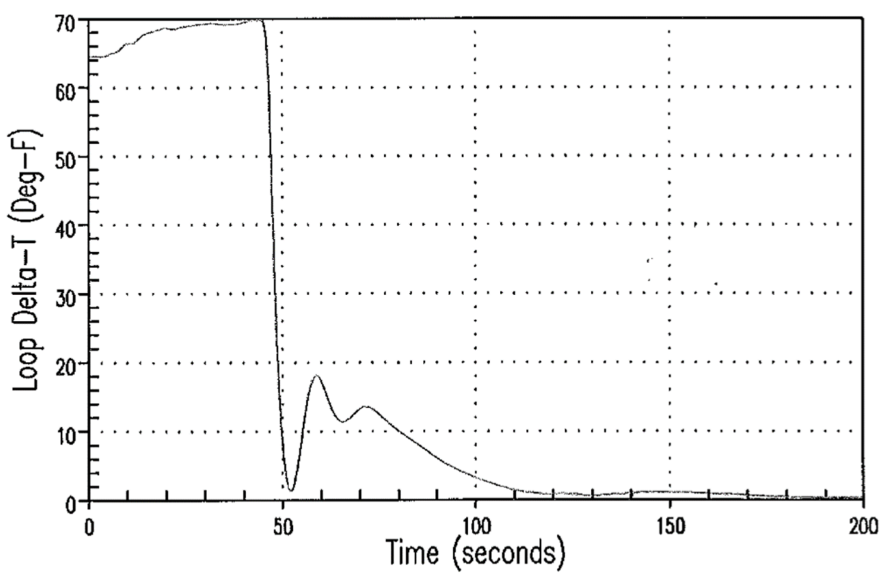
FEEDWATER CONTROL VALVE MALFUNCTION - NUCLEAR POWER AND CORE HEAT FLUX VERSUS TIME

FIGURE 14.1.7-1





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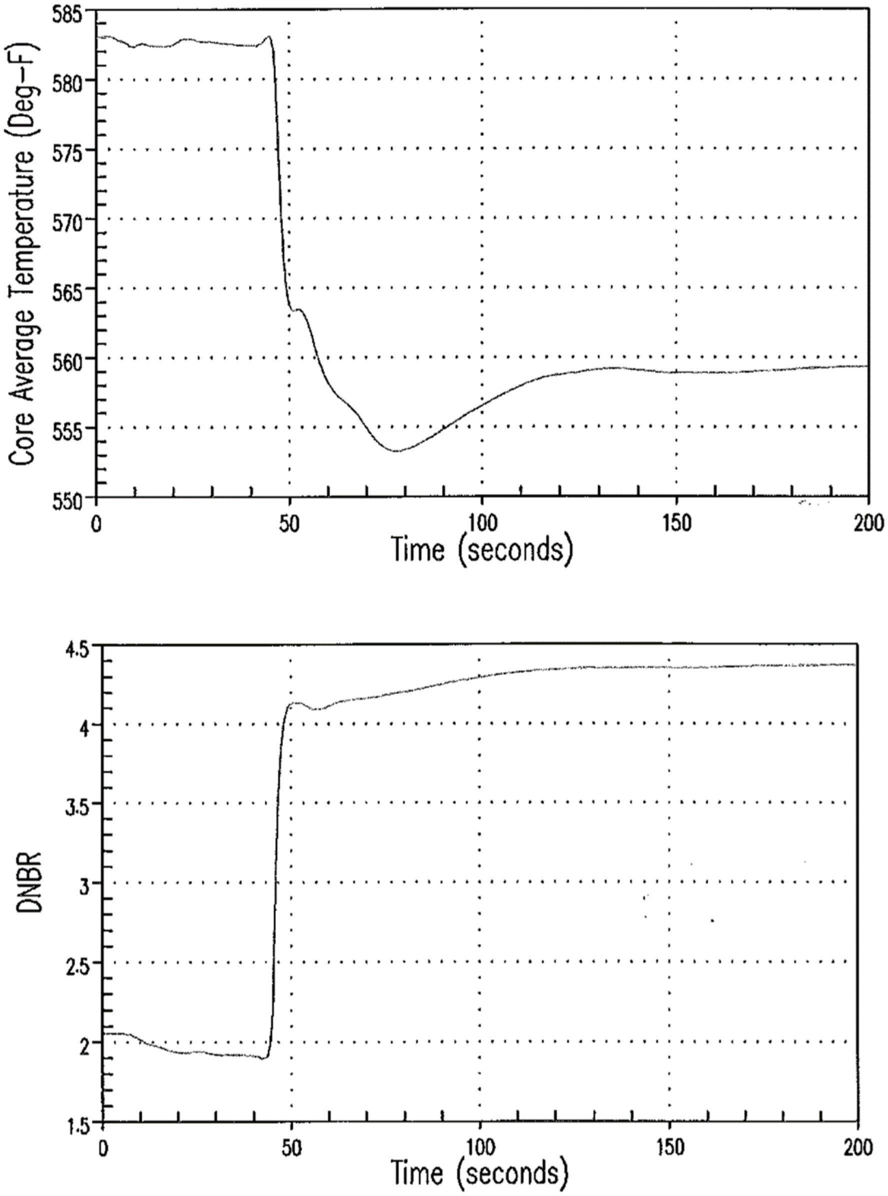
FEEDWATER CONTROL VALVE MALFUNCTION - PRESSURIZER PRESSURE AND LOOP ∆T VERSUS TIME

FIGURE 14.1.7-2



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FEEDWATER CONTROL VALVE MALFUNCTION - CORE AVERAGE TEMPERATURE AND DNBR VS TIME

FIGURE 14.1.7-3

14.1.8 EXCESSIVE LOAD INCREASE INCIDENT

An excessive load increase incident is defined as a rapid increase in steam generator steam flow causing 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 and a 5 percent per minute ramp load increase without a reactor trip in the range of 15 to 100 percent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the protection system. If the load increase exceeds the capability of the reactor control system, the transient is terminated in sufficient time to prevent DNBR from going below the limit value since the core is protected by a combination of the nuclear overpower trip and the overpower-overtemperature trips, as discussed in Section 14.1.7. An excessive load increase incident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction such as steam bypass control or turbine speed control.

The load demand is limited to 100% load by the turbine load limiting software feature in the turbine control system. C28

During power operation, steam bypass to the condenser is controlled by signals of reactor coolant conditions, i.e., abnormally high reactor coolant temperature indicates a need for steam bypass. A single controller malfunction does not cause steam bypass because an interlock is provided which blocks the control signal to the valves unless a sudden large turbine load decrease has occurred. In addition, the reference temperature and loss of load signals are developed by independent sensors.

Regardless of the rate of load increase, the reactor protection system will trip the reactor in time to prevent DNBR from going below the limit value. Increases in steam load to more than design flow are analyzed as steam line ruptures in Section 14.2.5.

Protection against an excessive load increase accident is provided by the following reactor protection system signals.

a. Overtemperature ΔT
b. Power range high neutron flux
c. Low pressurizer pressure
d. Overpower ΔT

Method of Analysis

This accident is analyzed using the RETRAN Code (Reference 1). The code simulates the neutron kinetics, reactor coolant system including natural circulation, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, main steam safety valves, and auxiliary feedwater system. The code computes pertinent plant variables including DNBR, temperatures, pressures, and power level.

Automatic rod withdrawal has been disabled at Turkey Point Units 3 and 4; however, cases have been conservatively analyzed assuming automatic rod withdrawal. Therefore,four cases are analyzed to demonstrate plant behavior following a 10-percent step load increase from rated load. These cases are as follows:

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

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

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

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

For the minimum moderator feedback cases (BOL), the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore, reductions in coolant temperature will have the least impact on core power. For the (EOL) maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A 10-percent step increase in steam demand is assumed, and all cases are studied without credit being taken for pressurizer heaters.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for the cases analyzed. No single active failure will prevent the reactor protection system from performing its intended function.

Results

Figures 14.1.8-1 through 14.1.8-4 illustrate the transient with the reactor in the manual rod control mode. As expected, for the (BOL) minimum moderator feedback case there is a slight power increase, and the average core temperature shows a decrease. This results in a departure from nucleate boiling ratio (DNBR) which increases (after a slight decrease) above its initial value. For the (EOL) maximum moderator feedback, manually controlled case, there is a larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced, but DNBR remains above the limit value.

Figures 14.1.8-5 through 14.1.8-8 illustrate the transient assuming the reactor is in the automatic rod control mode and no reactor trip signals occur. Both the BOL and EOL cases show that core power increases. The BOL case shows the core average temperature oscillates, due to the action of the control rod system, at a slightly higher value from the initial temperature. The EOL case shows that after a slight decrease the core average temperature stabilizes, again due to the action of the rod control system, at a value approximately equal to the initial temperature. For both of these cases, the minimum DNBR remains above the limit value.

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The calculated sequence of events for the excessive load increase incident is shown in Table 14.1.8-1. Note that a reactor trip signal was not generated for any of the four cases.

Conclusions

The analysis presented above shows that for a 10-percent step load increase, the DNBR remains above the limit value. The plant rapidly reaches a stabilized condition following the load increase.

REFERENCES

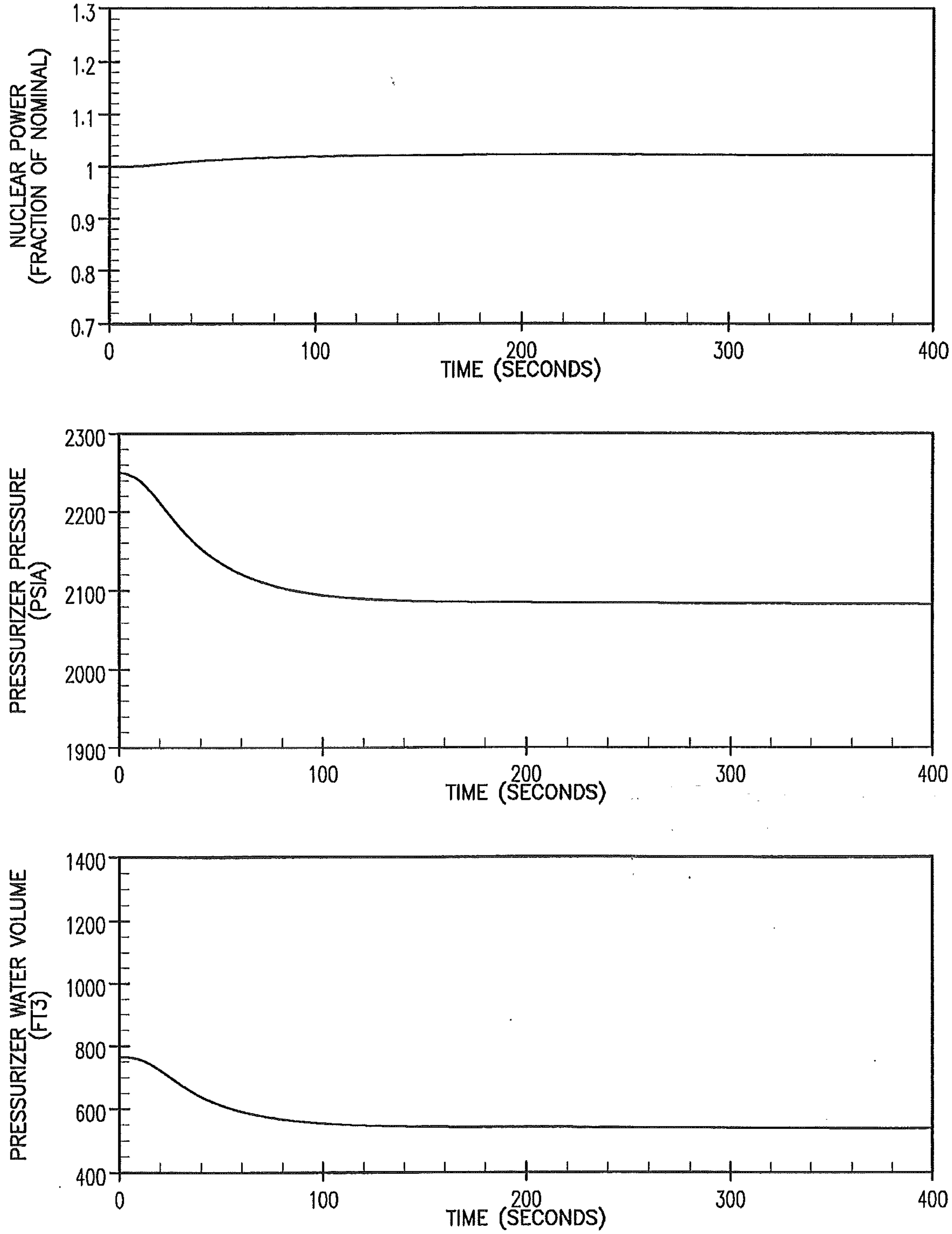
1. WCAP-14882-P-A (Proprietary) and WCAP-15234-A (Non-proprietary), RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, dated April 1999 and May 1999. C26

2. Westinghouse WCAP-11397-P-A, Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," dated April 1989.

TABLE 14.1.8-1

TIME SEQUENCE OF EVENTS
FOR
EXCESSIVE LOAD INCREASE INCIDENT

Case	Event	Time (seconds)	
1. Manual Reactor control (minimum moderator feedback)	10-percent step load increase	0.0	
	Equilibrium conditions reached (approximate time only)	300.0	C26
2. Manual reactor control (maximum moderator feedback)	10-percent step load increase	0.0	
	Equilibrium conditions reached (approximate time only)	300.0	C26
3. Automatic reactor control (minimum moderator feedback)	10-percent step load increase	0.0	
	Equilibrium conditions reached (approximate time only)	300.0	C26
4. Automatic reactor control (maximum moderator feedback)	10-percent step load increase	0.0	
	Equilibrium conditions reached (approximate time only)	300.0	C26



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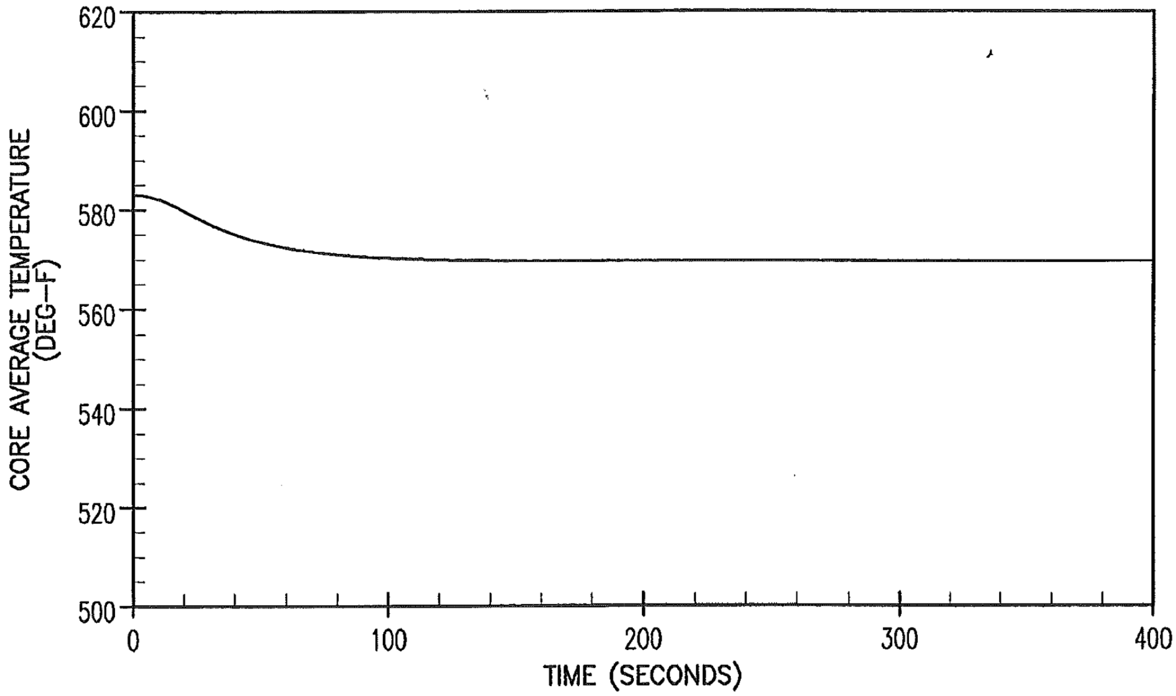
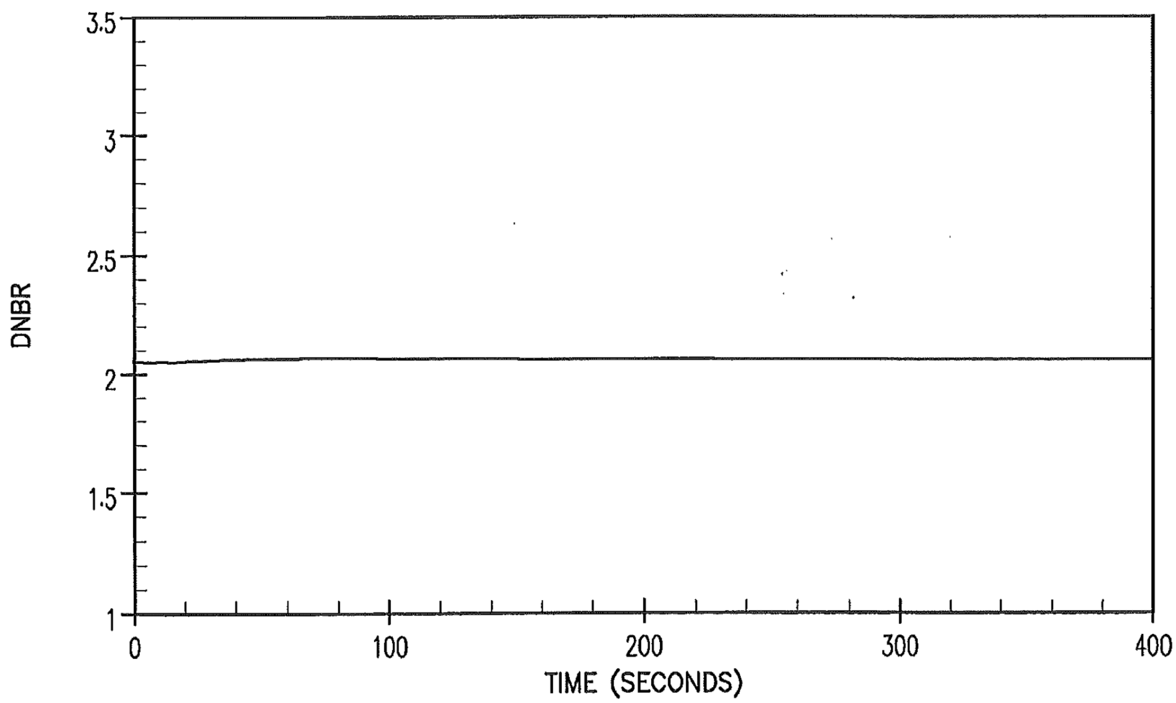
10% STEP LOAD INCREASE
MINIMUM MODERATOR FEEDBACK
MANUAL REACTOR CONTROL
FIGURE 14.1.8-1





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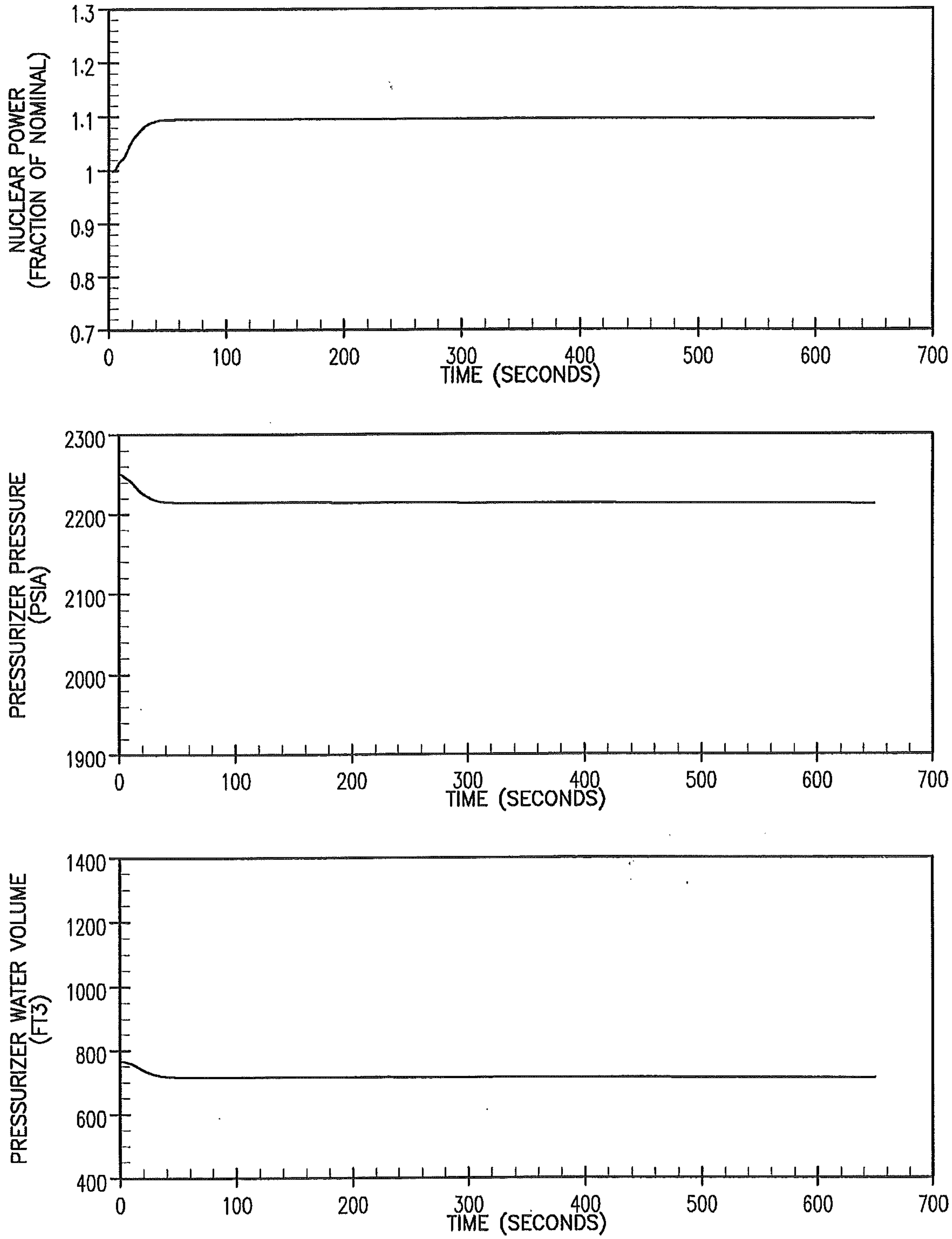
10% STEP LOAD INCREASE
MINIMUM MODERATOR FEEDBACK
MANUAL REACTOR CONTROL
FIGURE 14.1.8-2



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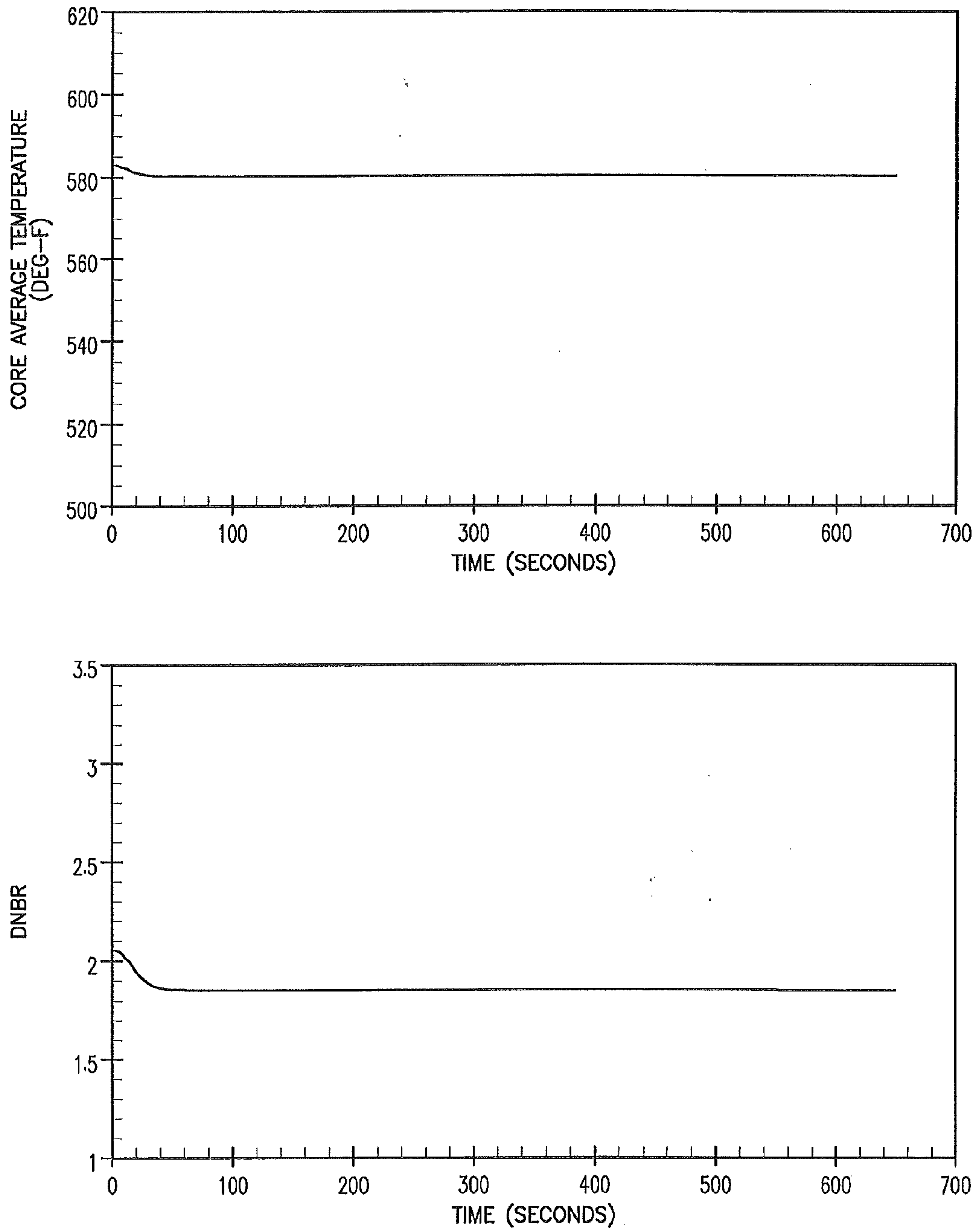
10% STEP LOAD INCREASE
MAXIMUM MODERATOR FEEDBACK
MANUAL REACTOR CONTROL
FIGURE 14.1.8-3



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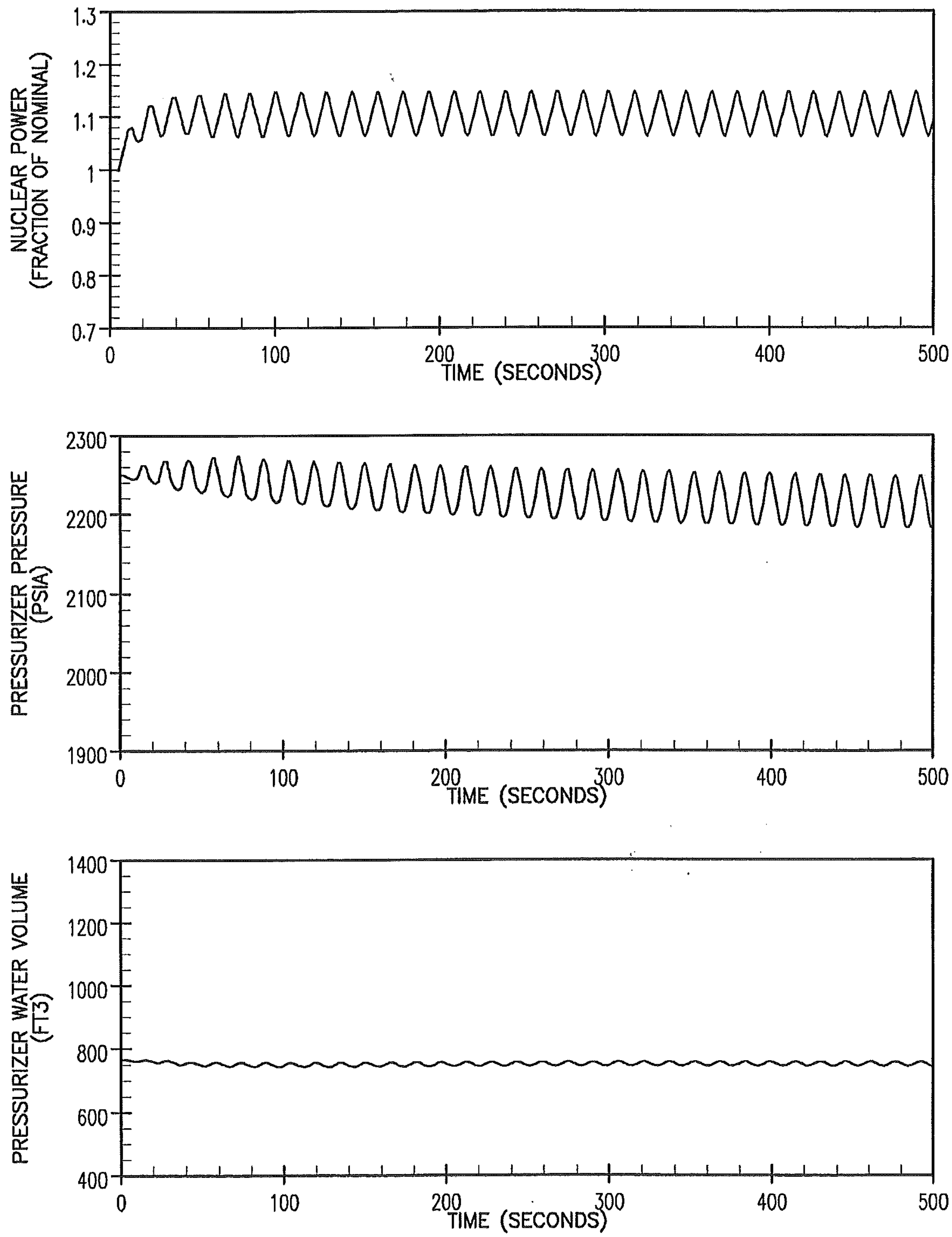
10% STEP LOAD INCREASE
MAXIMUM MODERATOR FEEDBACK
MANUAL REACTOR CONTROL
FIGURE 14.1.8-4



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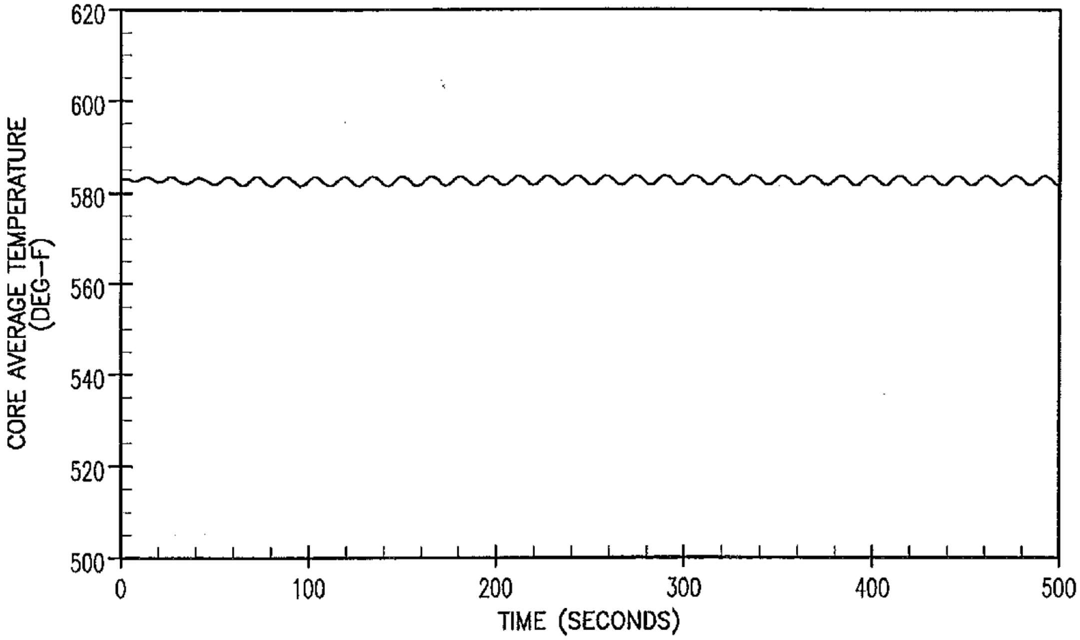
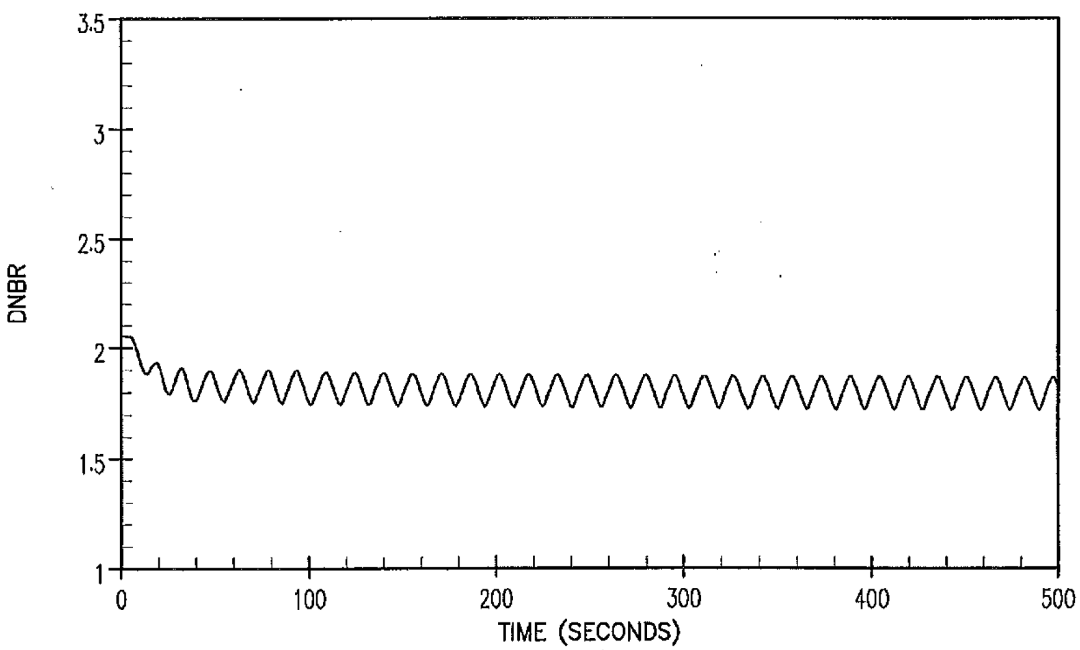
10% STEP LOAD INCREASE
MINIMUM MODERATOR FEEDBACK
AUTOMATIC REACTOR CONTROL
FIGURE 14.1.8-5





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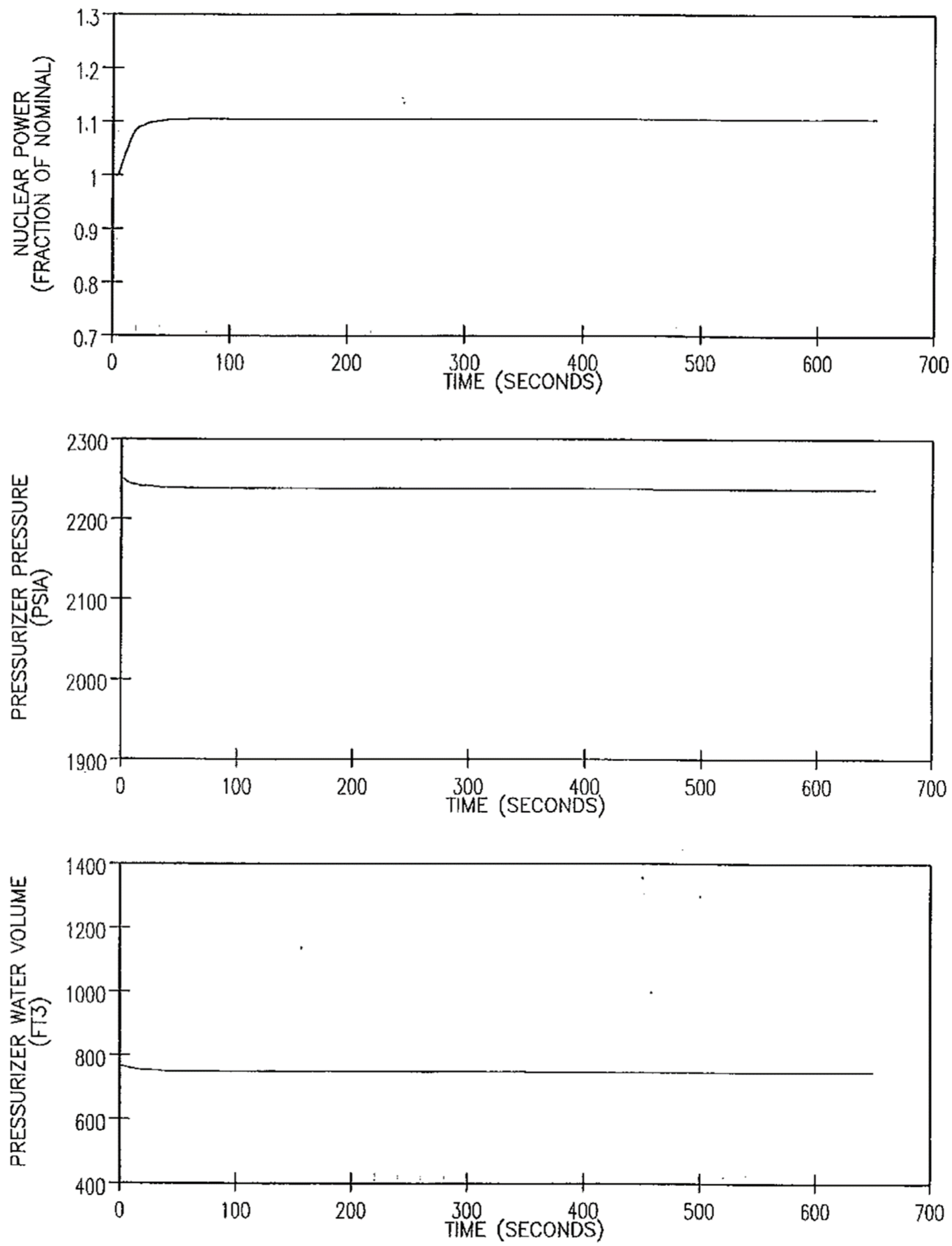
10% STEP LOAD INCREASE
MINIMUM MODERATOR FEEDBACK
AUTOMATIC REACTOR CONTROL
FIGURE 14.1.8-6



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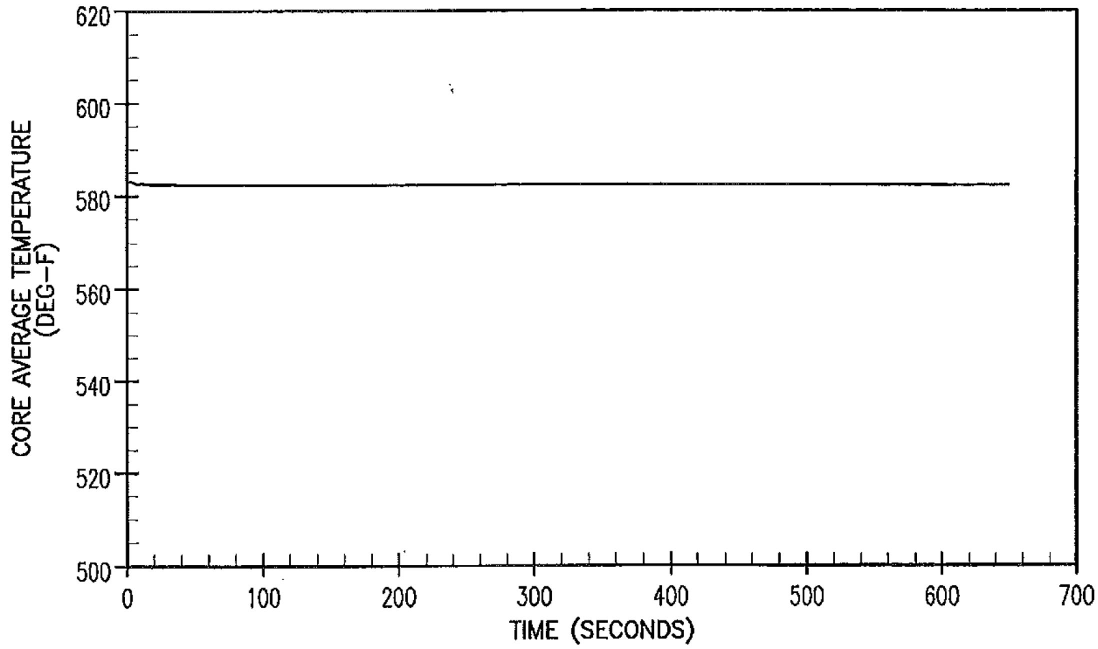
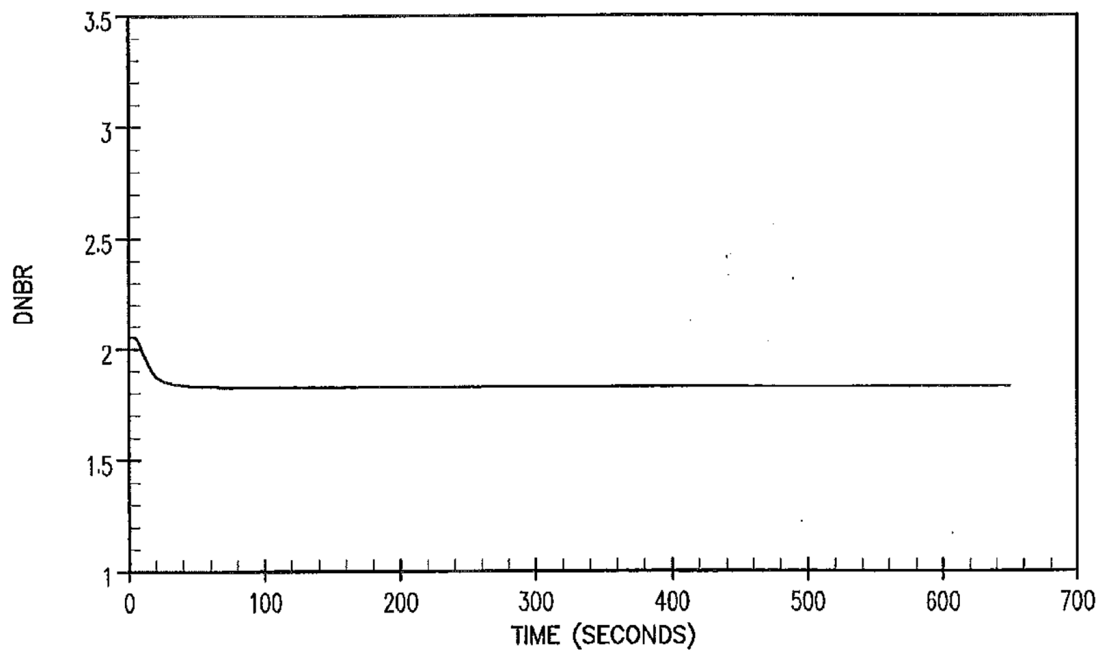
10% STEP LOAD INCREASE
MAXIMUM MODERATOR FEEDBACK
AUTOMATIC REACTOR CONTROL
FIGURE 14.1.8-7





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10% STEP LOAD INCREASE
MAXIMUM MODERATOR FEEDBACK
AUTOMATIC REACTOR CONTROL
FIGURE 14.1.8-8

14.1.9 LOSS OF REACTOR COOLANT FLOW

Flow Coast-Down Accidents

A loss of coolant flow incident can result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly.

Normal power supplies for the pumps are the two buses connected to the generator, one of which supplies power to one of the three pumps and the other of which supplies power to two of the three pumps. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines so that the pumps will continue to provide forced coolant flow to the core.

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

A. Undervoltage or underfrequency on reactor coolant pump power supply buses.
B. Low reactor coolant loop flow.
C. Pump circuit breaker opening.

These trip circuits and their redundancy are further described in Table 7.2-1 Reactor Control and Protection System.

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. This function is blocked below approximately 10 percent power (Permissive P-7). See Table 7.2-2 for a definition of permissive setpoints.

The reactor coolant pump underfrequency function is provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. The underfrequency function will open all reactor coolant pump breakers whenever an underfrequency condition occurs to ensure adequate RCP pump coastdown and to provide breaker open input signals to the pump breaker position reactor trip logic.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect one or two reactor coolant loops. It also serves as a backup to the undervoltage and underfrequency trips for the loss of all three reactor coolant pumps case. This function is generated by two-out-of-three low flow signals per reactor coolant loop. Above Permissive P-8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive P-7) and the power level corresponding to Permissive P-8 (approximately 45% power), low flow in any two loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive P-7. C26

A reactor trip from pump breaker position is provided to implement the underfrequency function and to provide protection against other conditions for which the RCP breakers are designed to trip open. Similar to the low flow trip, above P-8, a breaker open signal from any pump will actuate a reactor trip, and between P-7 and P-8, a breaker open signal from any two pumps will actuate a reactor trip. Reactor trip on reactor coolant pump breakers open is blocked below Permissive P-7. C26

<u>Method of Analysis</u>

The following loss of flow cases have been analyzed:

1. Loss of all three reactor coolant pumps with three loops in operation.

2. Loss of two reactor coolant pumps with three loops in operation.

These transients are analyzed by two digital computer codes. First, the RETRAN code (Reference 1) is used to calculate the loop and core flow transients, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE code (Reference 2) is then used to calculate the heat flux and DNBR transients based on the nuclear power and flow from RETRAN. The DNBR transient presented represents the minimum of the typical and thimble cells. C26 C26

The accidents are analyzed using the Revised Thermal Design Procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the limit departure from nucleate boiling ratio (DNBR) as described in WCAP-11397-P-A (Reference 3).

The core nuclear parameters are used to maximize the energy transfer to the primary coolant during the initial phase of pump coastdown. Since there is an initial heatup due to the reduction in RCS flow, a minimum moderator temperature coefficient of reactivity, most negative Doppler power coefficient, least negative Doppler temperature coefficient and maximum delayed neutron fraction, consistent with beginning of life full-power conditions are used. This results in the maximum core power and hot spot heat flux during the initial part of the transient when the minimum DNBR is reached. The negative reactivity insertion upon reactor trip is based on a 4%Δk trip reactivity from full power. C26

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

Results (Flow Coast-Down)

Figures 14.1.9-1 through 14.1.9-4 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. Figures 14.1.9-5 through 14.1.9-8 show the transient response for the loss of two reactor coolant pumps with three loops initially in operation. The reactor is tripped on a low flow signal. The DNBR-versus-time plots (Figure 14.1.9-4 and 14.1.9-8), representing the limiting cells, show that the DNBR is always greater than the safety analysis limit value of 1.50 or 1.40 for the OFA (optimized fuel assembly) or upgrade fuel assembly, respectively. C26

For the cases analyzed, since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperatures do not increase far above their respective initial values.

The calculated sequence of events for the cases analyzed is shown in Table 14.1.9-1.

Conclusions

The analyses performed have demonstrated that for the above loss of flow incidents, the DNBR does not decrease below the limit value at any time during the transient. Thus, no fuel damage is predicted, and all applicable acceptance criteria are met.

Locked Rotor Accident

A hypothetical transient is analyzed for the postulated instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system is rapidly reduced, leading to a reactor trip on a low-flow signal. C26

Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generator 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 trip). The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generator, causes an insurge into the pressurizer and a 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 is not included in the analysis.

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. Only one analysis has been performed, and it represents the most limiting condition for the locked rotor and pump shaft break accidents.

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 the peak RCS pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the VIPRE code (Reference 2) which uses the core flow and the nuclear power values calculated by RETRAN. The VIPRE code includes a film boiling heat transfer coefficient. C26 C26

One case is analyzed:

1. One locked rotor/shaft break with three loops in operation.

The accident is conservatively evaluated with loss of offsite power coincident with reactor trip. The two unaffected RCPs coast down as a result. C26

Initial Conditions

At the beginning of the postulated locked rotor accident, the plant is assumed to be operating under the most adverse steady-state operating conditions. These include the maximum steady-state power level, pressure, and coolant average temperature. The reactivity coefficients assumed in the analysis include a minimum moderator temperature coefficient of reactivity, most negative Doppler power coefficient, least negative Doppler temperature coefficient and maximum delayed neutron fraction, consistent with beginning of life full-power conditions. For this analysis, the negative reactivity insertion upon trip is based on a 4%Δk trip reactivity from full power. C26

The consequences of a locked rotor (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the locked rotor event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the locked rotor scenario. Only one analysis has been performed, and it represents the most limiting condition for the locked rotor and pump shaft break accidents.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 53 psi above the nominal pressure of 2250 psia to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. The pressure response shown in Figure 14.1.9-10 is at the point in the Reactor Coolant System having the maximum pressure (i.e., the reactor vessel lower plenum). C26 C26

For a conservative analysis of fuel rod behavior, the hot spot evaluation assumes that DNB occurs at the initiation of the transient and continues throughout the event. This assumption reduces heat transfer to the coolant and results in conservatively high hot spot temperatures.

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

<u>Evaluation of the Pressure Transient</u>

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin one second after the flow in the affected loop reaches 84.5 percent of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer power-operated 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 pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are modeled including the effects of the pressurizer safety valve loop seals using WOG methodology (Reference 4). The pressurizer safety valve includes a 2.8% uncertainty (0.8% set pressure shift and a 2.0% set pressure tolerance) over an assumed conservative nominal setpoint of 2480 psia. Additionally, no steam flow is assumed until the valve loop seals are purged. C26

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 2.4 times the value at the initial core power level. C26

Film Boiling Coefficient

The film boiling coefficient is calculated in the VIPRE code using the Bishop-Sandberg-Tong film boiling correlation (Reference 2). The fluid properties are evaluated at the film temperature (average between the 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. C26

For this analysis, the initial value of the pressure is used throughout the transient, since it is more conservative with respect to the clad temperature response. As indicated earlier, DNB was assumed to start at the beginning of the accident. C26

Film 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.

For the initial portion of the transient, a high gap coefficient produces higher clad temperatures, since the heat stored and generated in the fuel redistributes itself in the cooler cladding. This effect is reversed when the clad temperature exceeds the pellet temperature in cases where a zirconium-steam reaction is present. 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 temperatures to 10,000 Btu/hr-ft^2-^{o}F at the initiation of the transient. Thus, the large amount of energy stored in the fuel 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 (Reference 2) shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{dw^2}{dt} = 33.3 \times 10^6 \exp^{(-45{,}500/1.986T)}$$

where,

w = amount reacted, mg/cm^2

t = time, sec

T = temperature, °K

The reaction heat is about 1510 cal/gm

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

Results

The calculated sequence of events is shown in Table 14.1.9-1. The transient results are shown in Figures 14.1.9-9 through 14.1.9-12. The peak Reactor Coolant System pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad temperature is considerably less than 2375°F for ZIRLO and Optimized ZIRLO clad. This conservatively ensures that the core will remain in place and geometrically intact with no loss of core cooling capability. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient. The results of these calculations (peak pressure, peak clad temperature and zirconium-steam reaction) are also summarized in Table 14.1.9-2. The Zirconium water reaction limit is 16 wt. %. C29

Dose Evaluation

The Locked Rotor 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 ADVs and MSSVs. In addition, radioactive iodine contained in the secondary inventory prior to the event is released to the atmosphere as a result of steaming from the steam generators following the accident.

The Locked Rotor dose consequence analysis is consistent with the guidance provided in Appendix G of Reference 6, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1. Regulatory Position 2 – Fuel damage is not predicted for this event. A radiological consequence analysis is not required and this event is bounded by the consequences projected for the main steam line break outside containment. C29

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Other Assumptions C29

1. Reg. Guide 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.
2. Locked Rotor thermal hydraulic analysis demonstrates 0% of fuel rods experience DNB, therefore fuel damage is precluded. C29

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C29

REFERENCES

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

2. WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999. C26

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

4. Westinghouse WCAP-12910, Rev. 1-A, Barrett, G. O., et al.,"Pressurizer Safety Valve Set Pressure Shift," May 1993.

5. Deleted

6. USNRC, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants,” July 2000. C26

7. Code of Federal Regulations, 10CFR50.67, “Accident Source Term,” revised 12/03/02.

TABLE 14.1.9-1

SEQUENCE OF EVENTS
LOSS OF FLOW ACCIDENTS

Case	Event	Time (sec)
Loss of 3 RCP's	Coastdown Begins	0.0
	Low Voltage Trip setpoint Reached	0.0
	Rods Begin to Drop	2.5
	Low Flow Setpoint Reached	2.8
	Minimum DNBR Occurs	4.0
	Maximum RCS Pressure Occurs	4.8
Loss of 2 RCP's	Coastdown Begins	0.0
	Low Flow Setpoint Reached	2.5
	Rods Begin to Drop	3.5
	Minimum DNBR Occurs	4.6
	Maximum RCS Pressure Occurs	5.8
Locked Rotor	Rotor Locks on one RCP	0.0
	Low Flow Setpoint Reached	0.1
	Rods Begin to Drop	1.1
	Maximum Clad Temperature Occurs	4.0
	Maximum RCS Pressure Occurs	4.8

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TABLE 14.1.9-2

SUMMARY OF RESULTS
FOR
THE LOCKED ROTOR TRANSIENT

Criteria	3 Loops Initially Operating One Locked Rotor
Maximum RCS Pressure	2694 psia
Maximum Clad Temperature at Core Hot Spot	1890.1 °F
$Zr-H_2O$ Reaction at Core Hot Spot	0.46 wt. %

C26

TABLE 14.1.9-3

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TABLE 14.1.9-4

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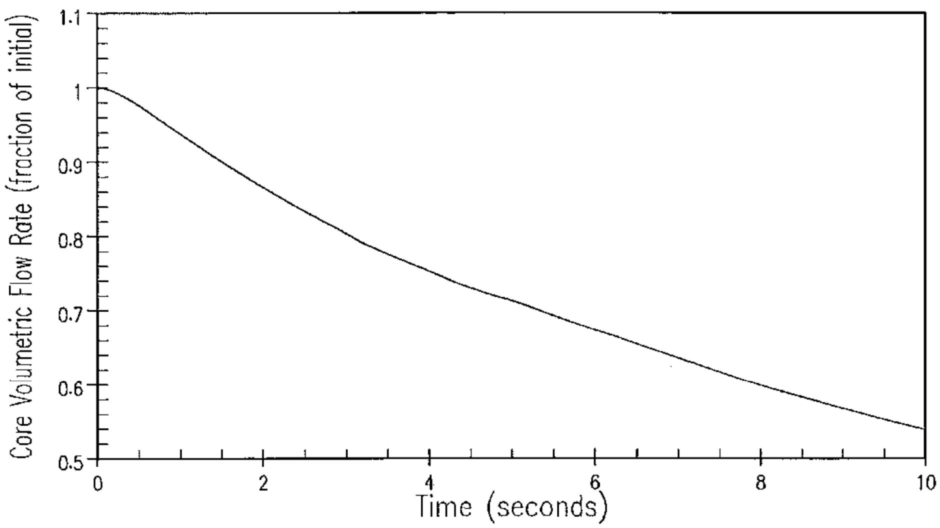
Revised 04/06/2018



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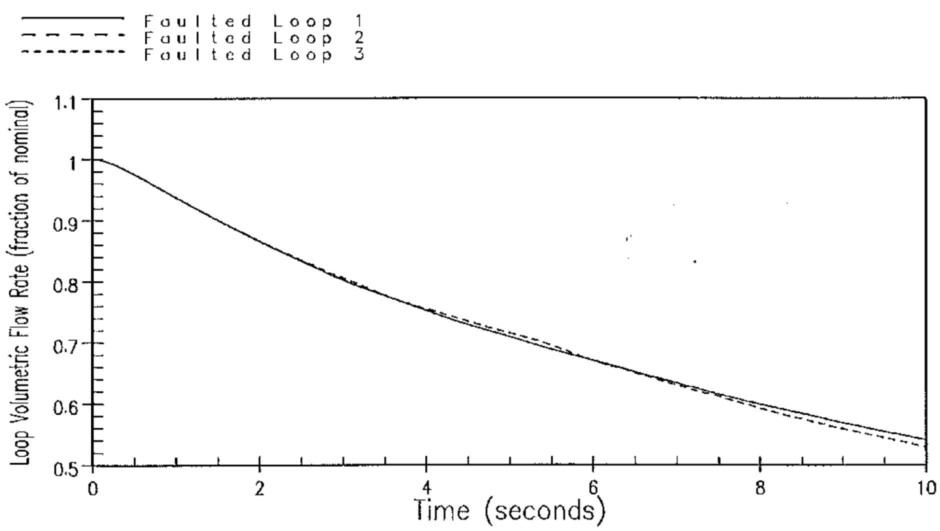
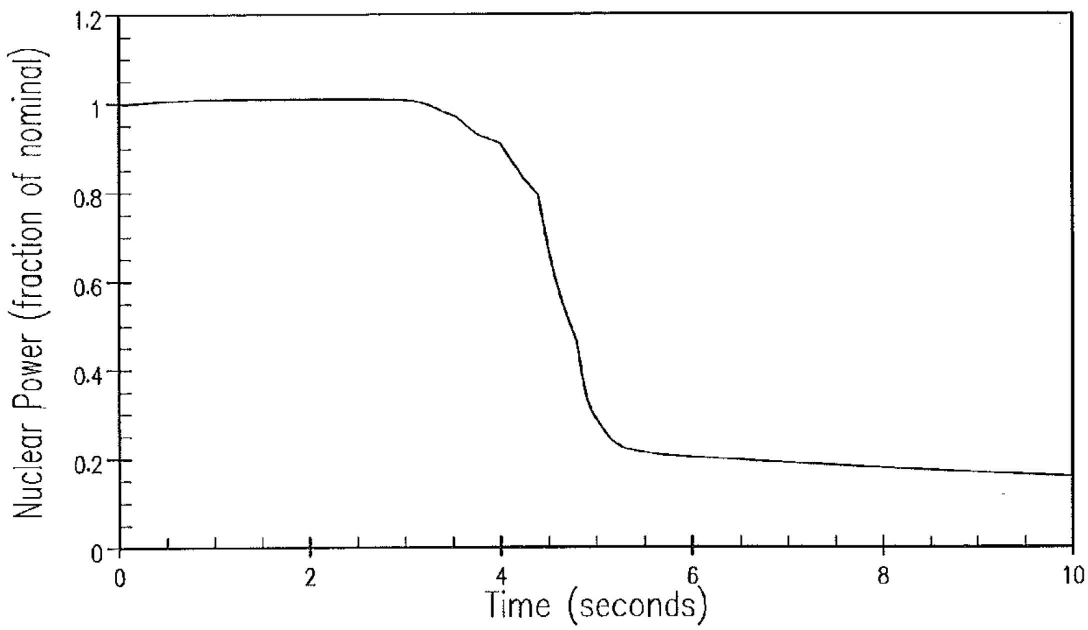


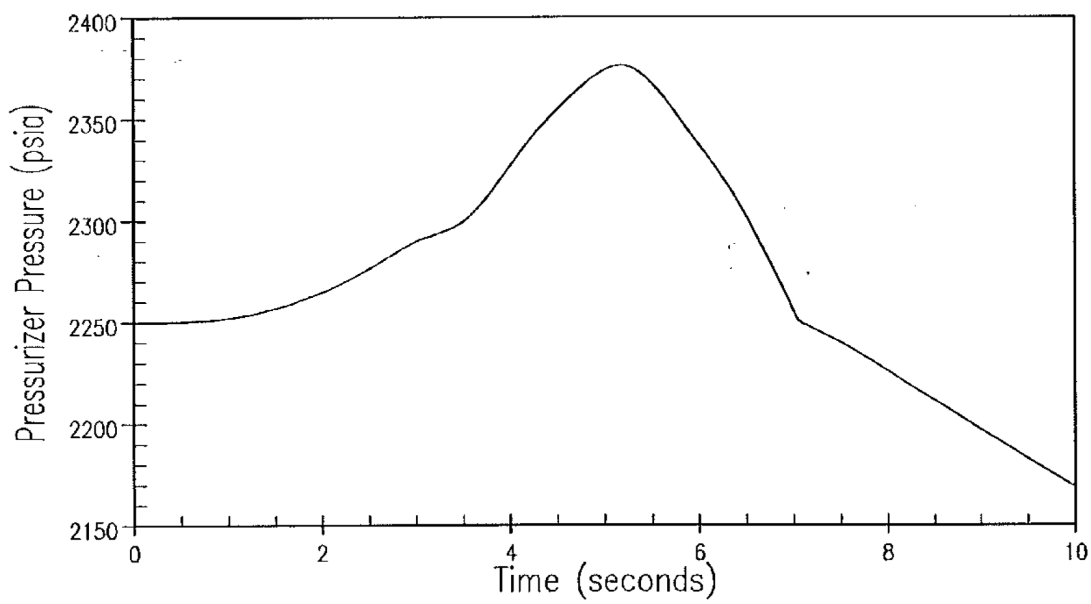
REVISED 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING
ALL LOOPS COASTING DOWN
CORE VOLUMETRIC AND LOOP VOLUMETRIC
FLOW RATE TRANSIENTS

FIGURE 14.1.9-1





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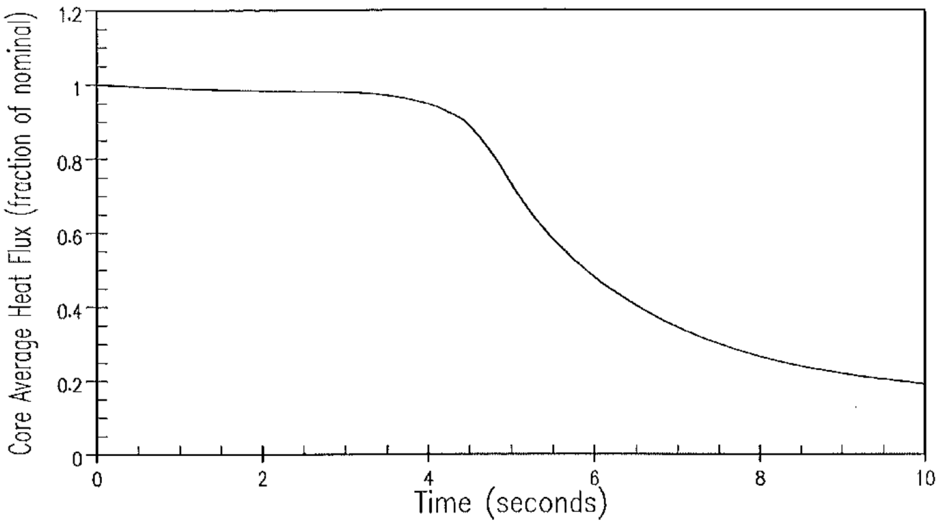
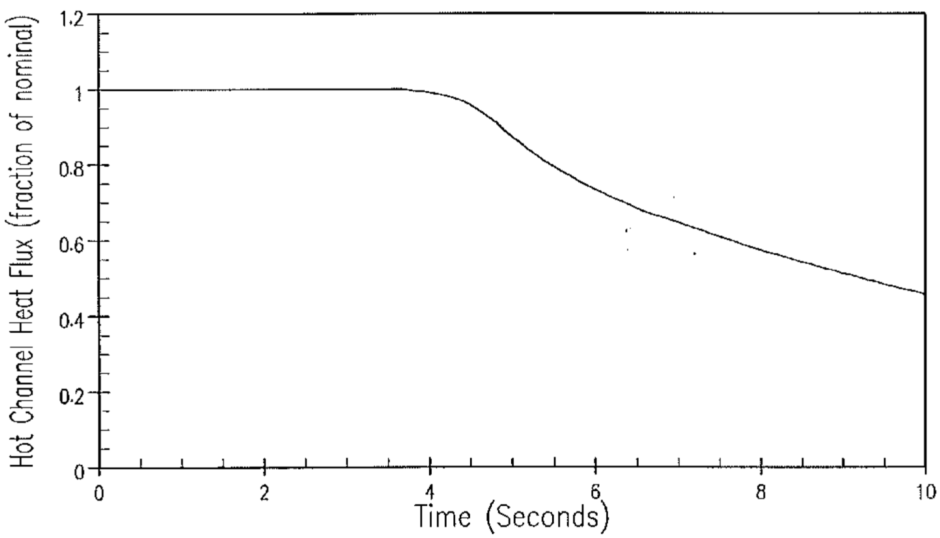
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
ALL LOOPS COASTING DOWN
NUCLEAR POWER AND PRESSURIZER
PRESSURE TRANSIENTS

FIGURE 14.1.9-2





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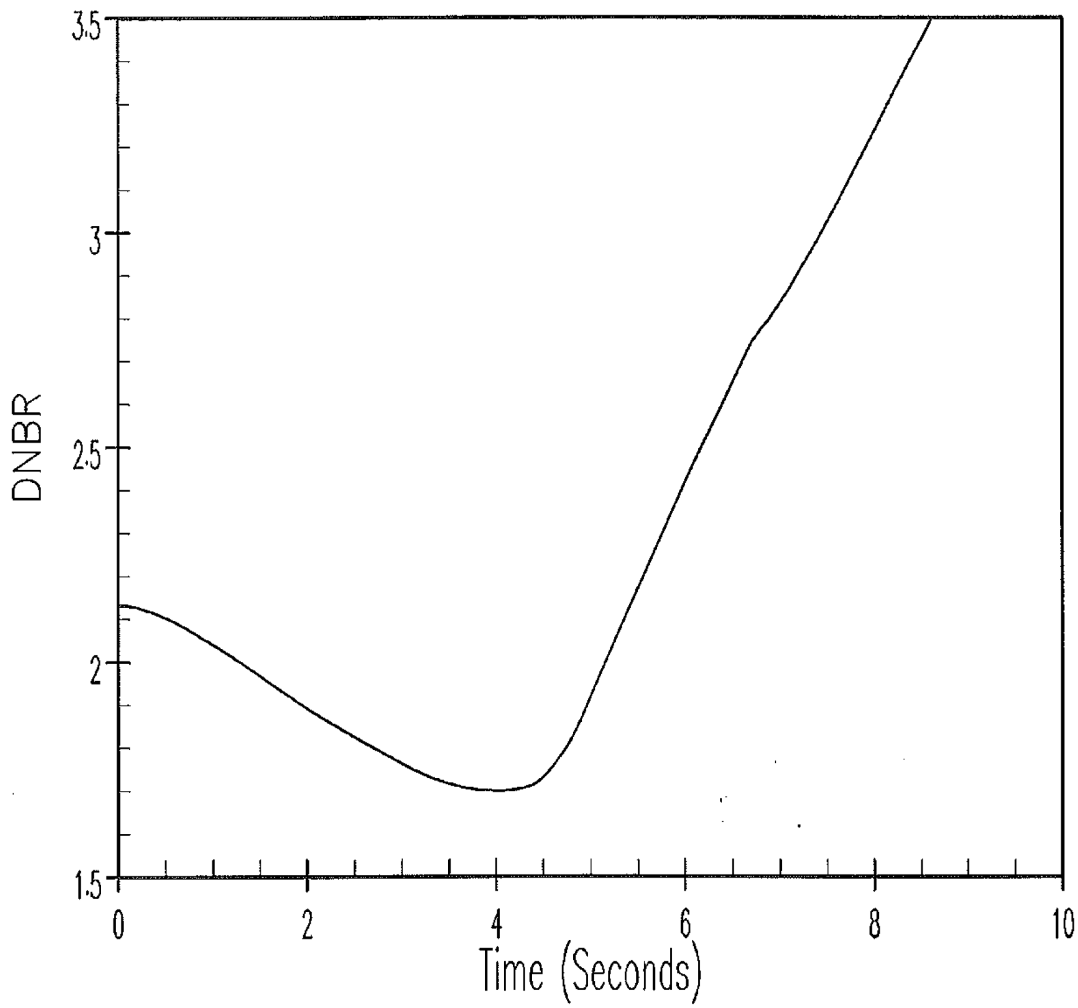
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
ALL LOOPS COASTING DOWN
CORE AVERAGE AND HOT CHANNEL
HEAT FLUX TRANSIENTS
FIGURE 14.1.9-3



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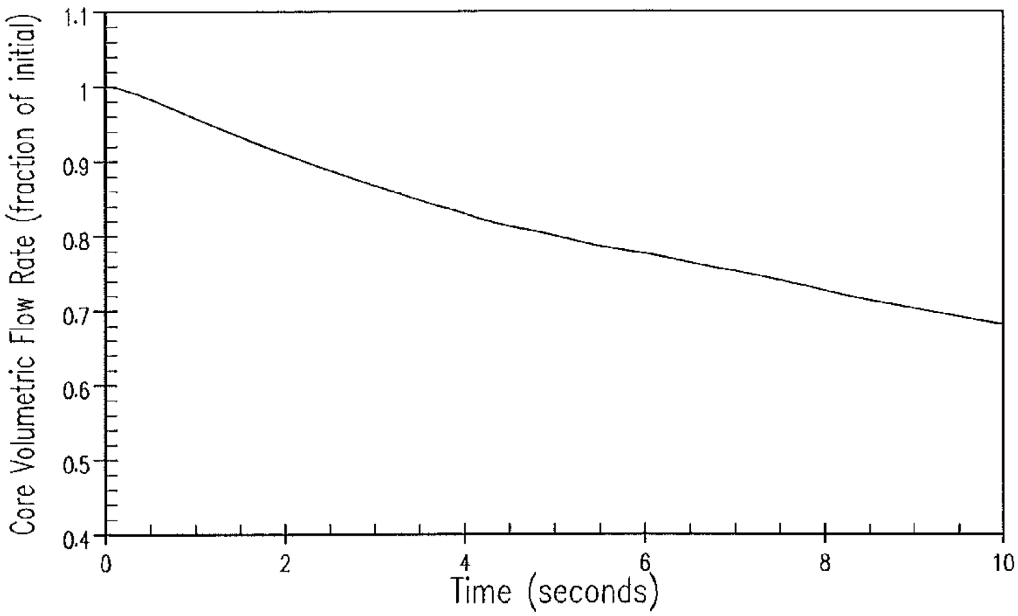
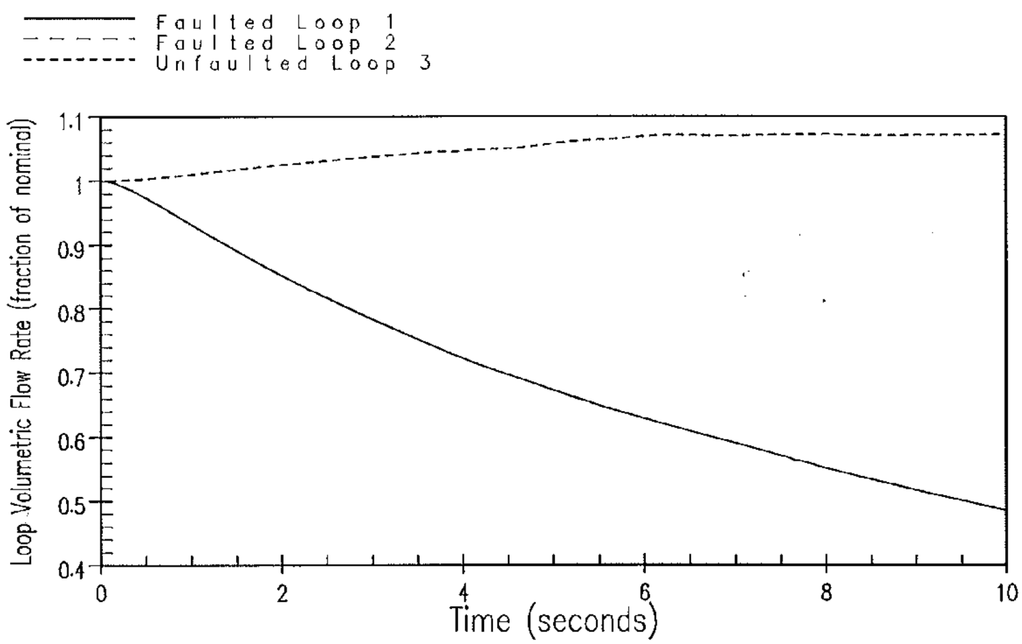
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
ALL LOOPS COASTING DOWN
DNBR TRANSIENT
FIGURE 14.1.9-4





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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
TWO LOOPS COASTING DOWN
CORE VOLUMETRIC AND LOOP VOLUMETRIC
FLOW RATE
FIGURE 14.1.9-5

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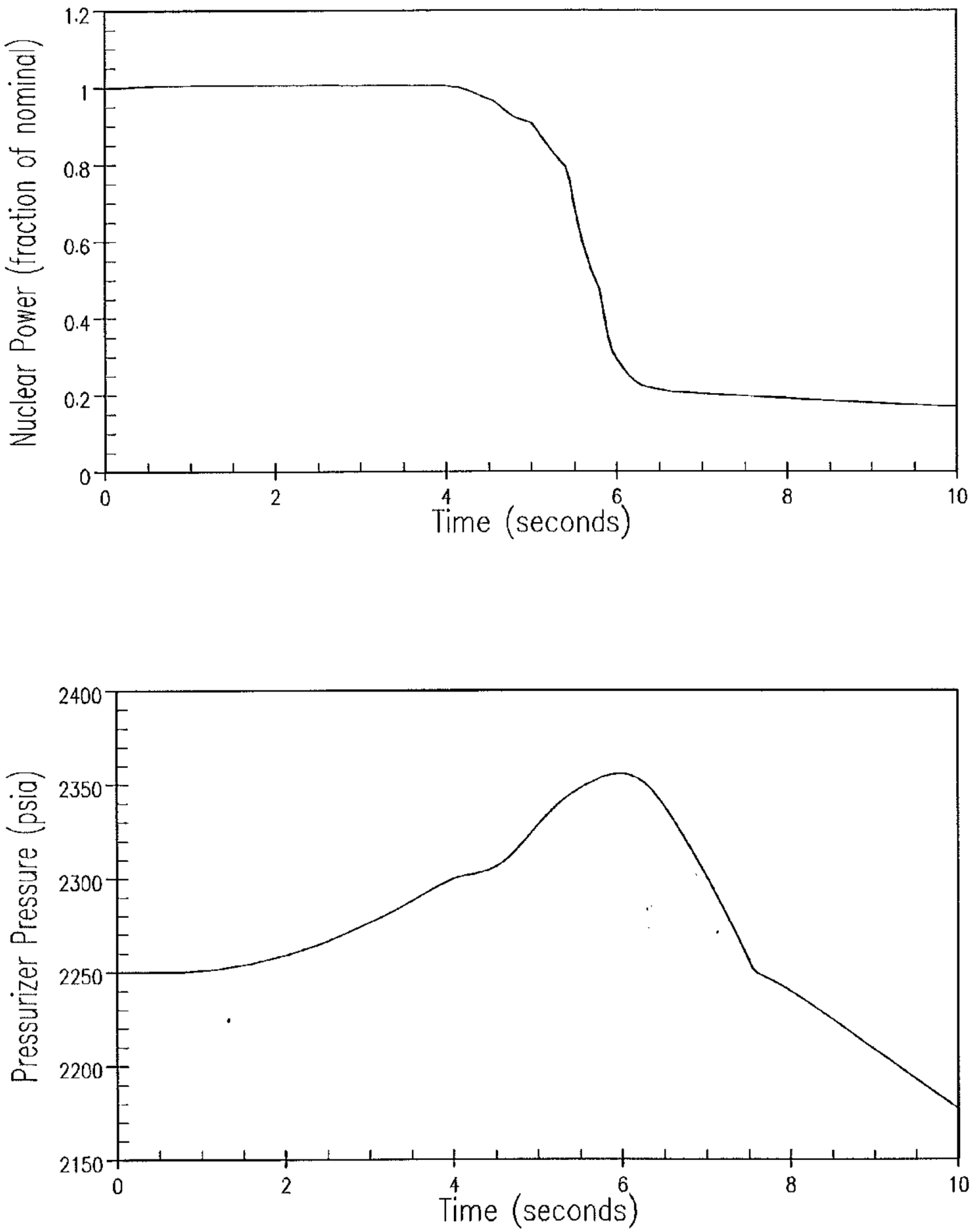

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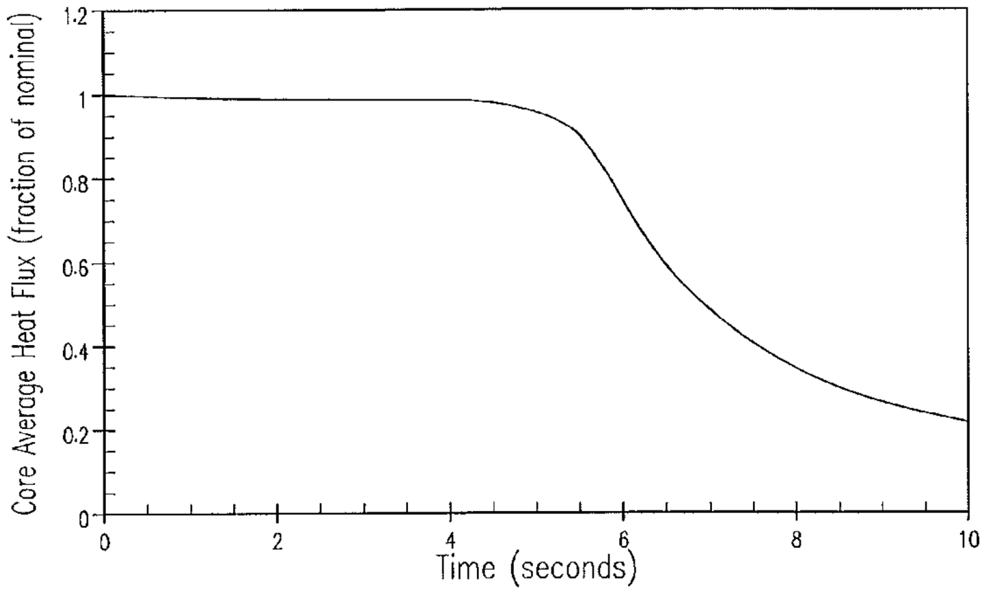
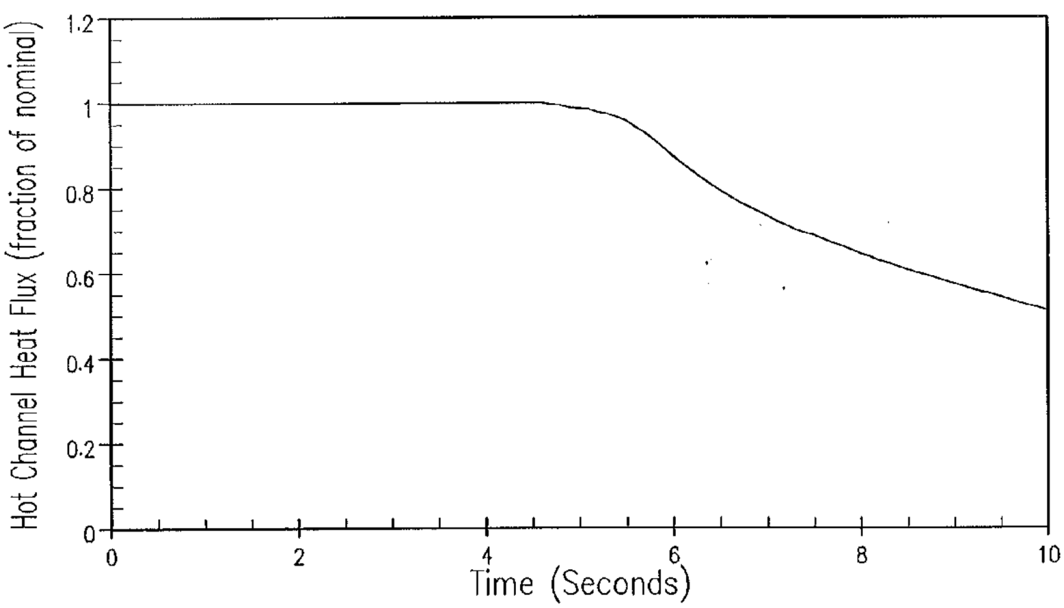
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING
TWO LOOPS COASTING DOWN
NUCLEAR POWER AND PRESSURIZER
PRESSURE TRANSIENTS
FIGURE 14.1.9-6





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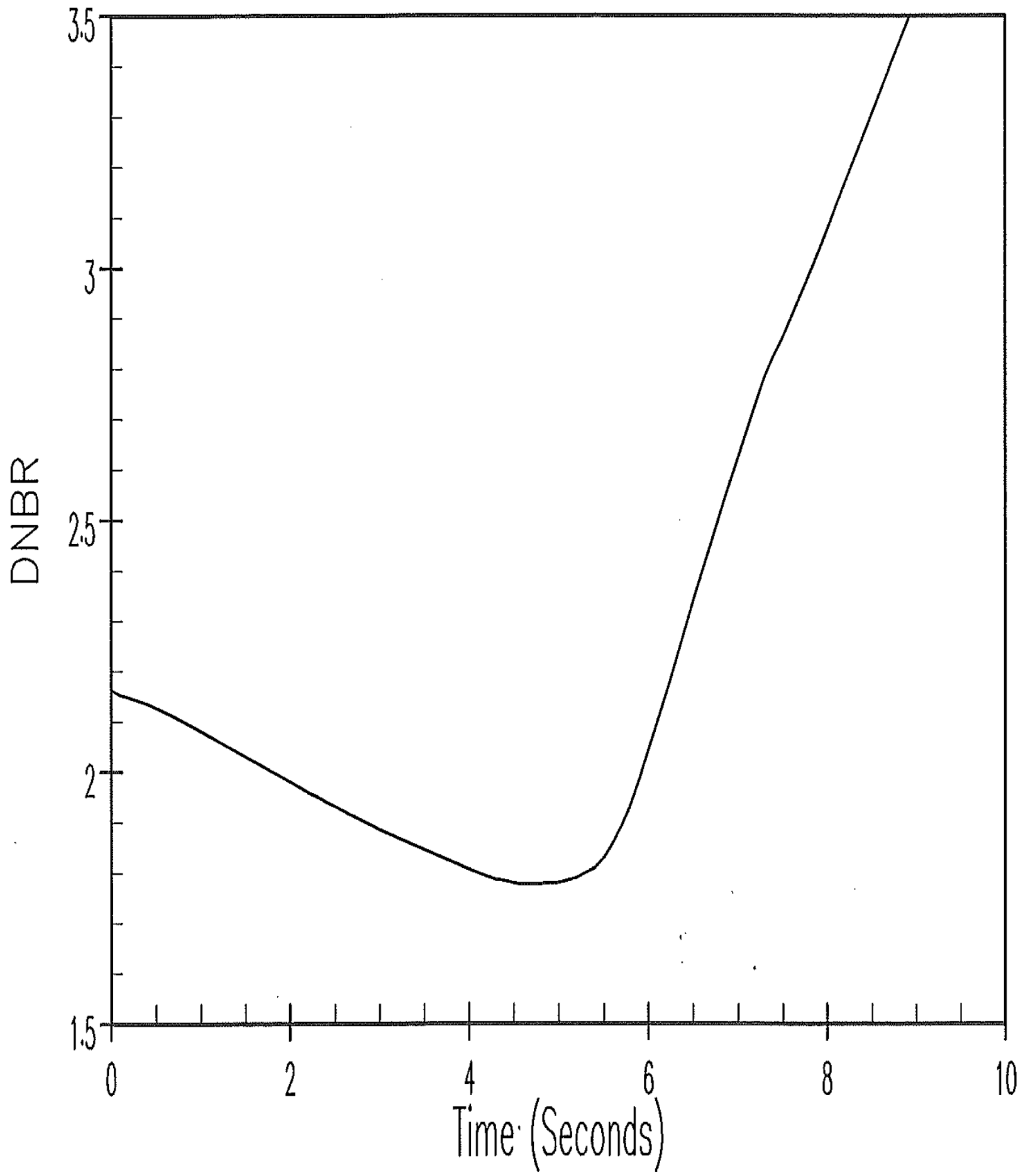
Revised 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
TWO LOOPS COASTING DOWN
CORE AVERAGE AND HOT CHANNEL
HEAT FLUX TRANSIENTS
FIGURE 14.1.9-7



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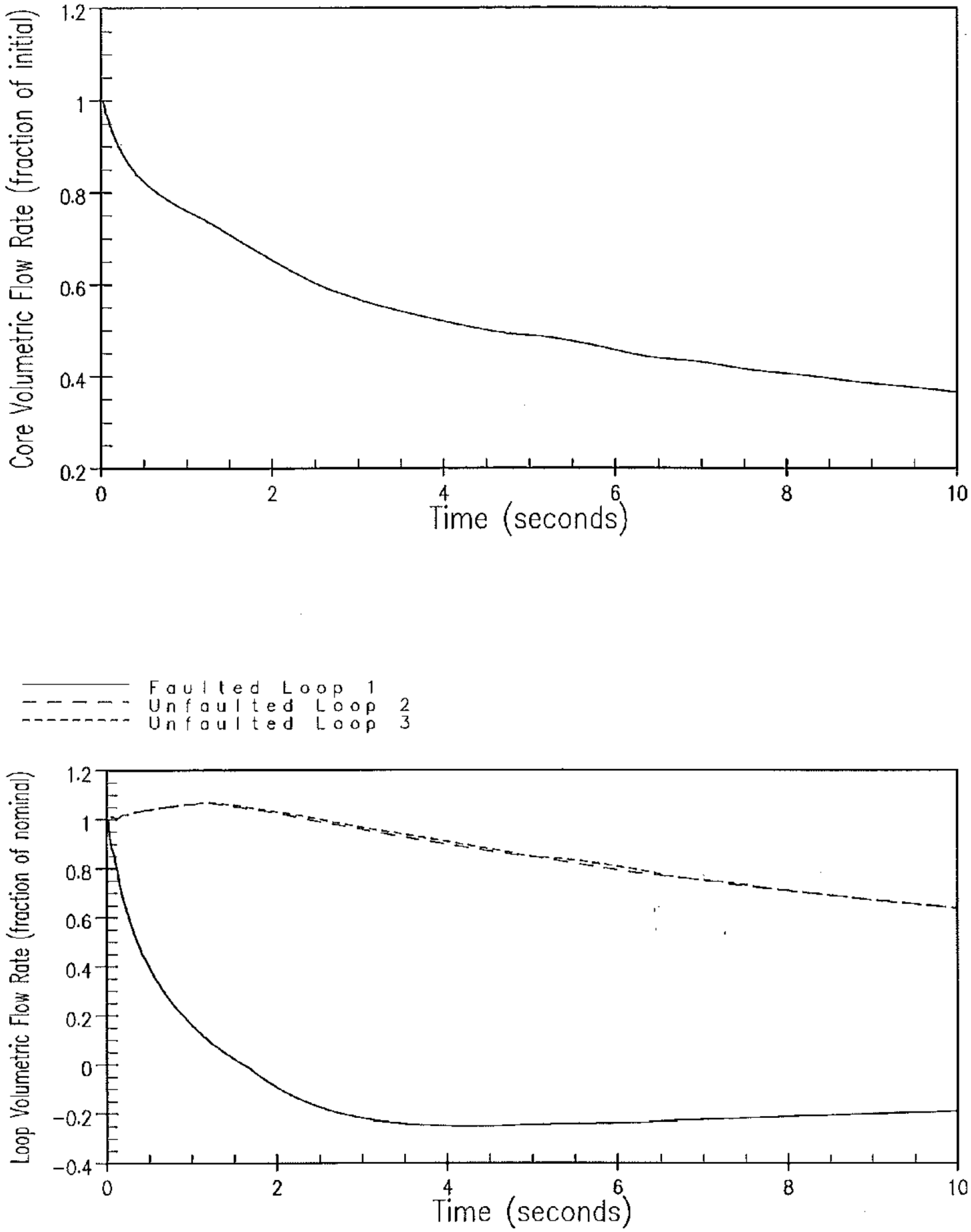
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
TWO LOOPS COASTING DOWN
DNBR TRANSIENT
FIGURE 14.1.9-8



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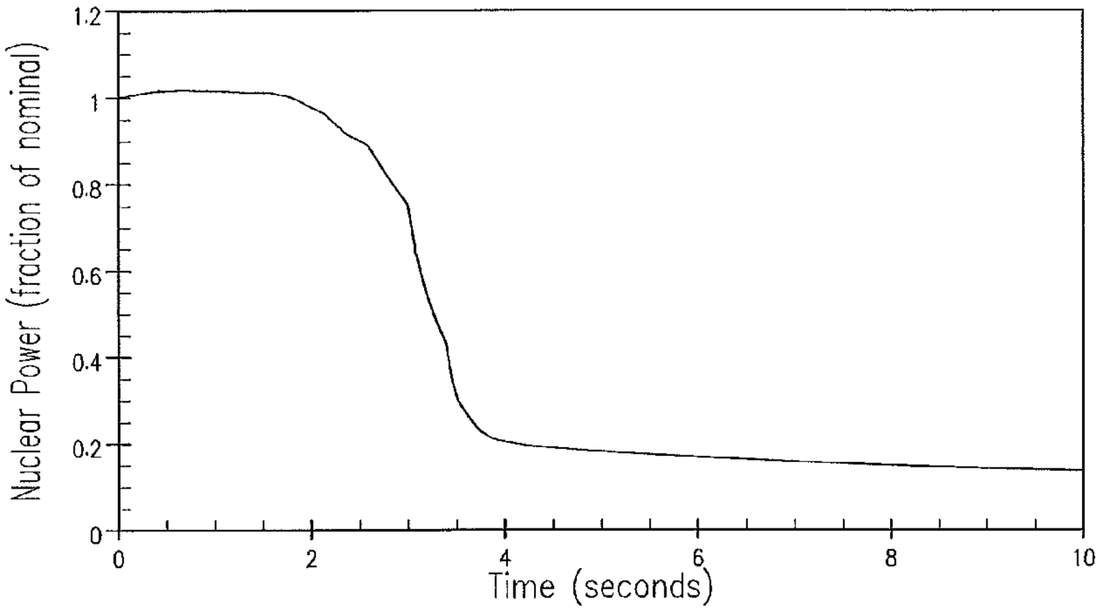
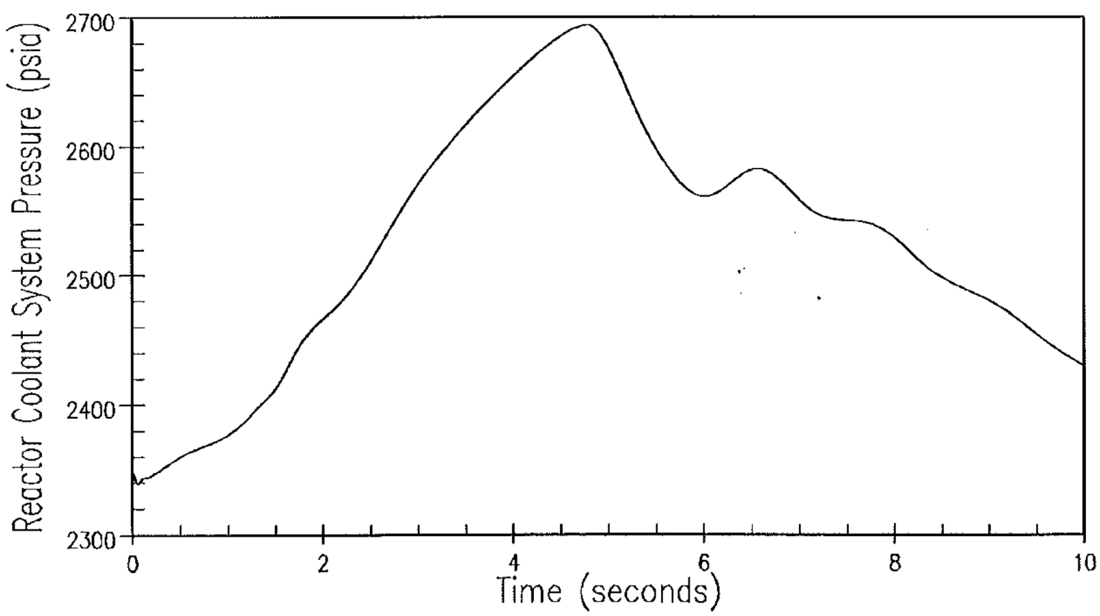
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
ONE LOCKED ROTOR
CORE VOLUMETRIC AND LOOP VOLUMETRIC
FLOW RATE TRANSIENTS
FIGURE 14.1.9-9





C26

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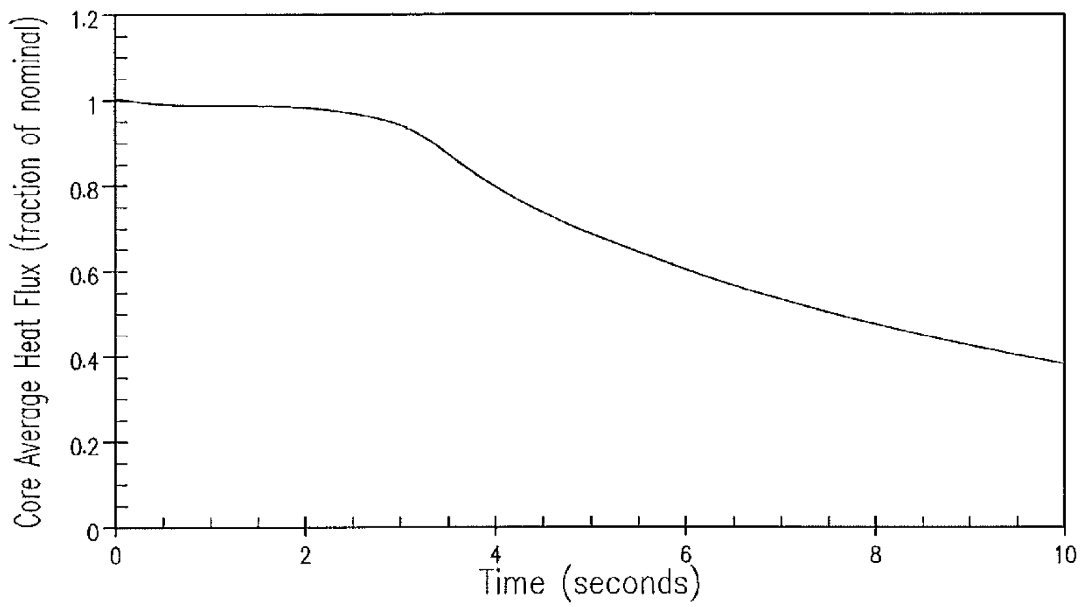
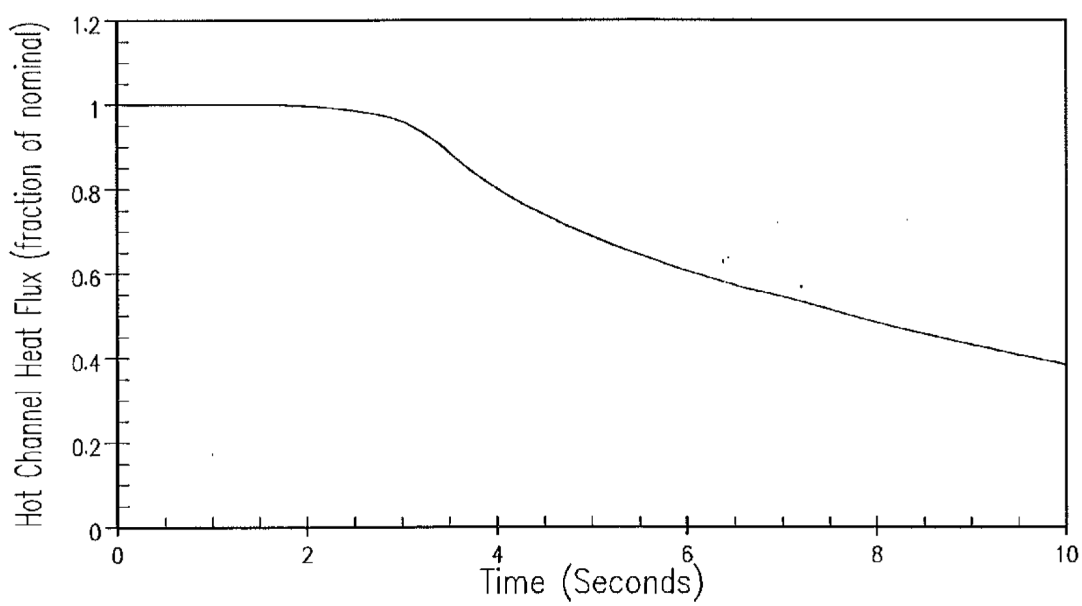
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING
ONE LOCKED ROTOR
NUCLEAR POWER AND REACTOR COOLANT SYSTEM
PRESSURE TRANSIENTS

FIGURE 14.1.9-10





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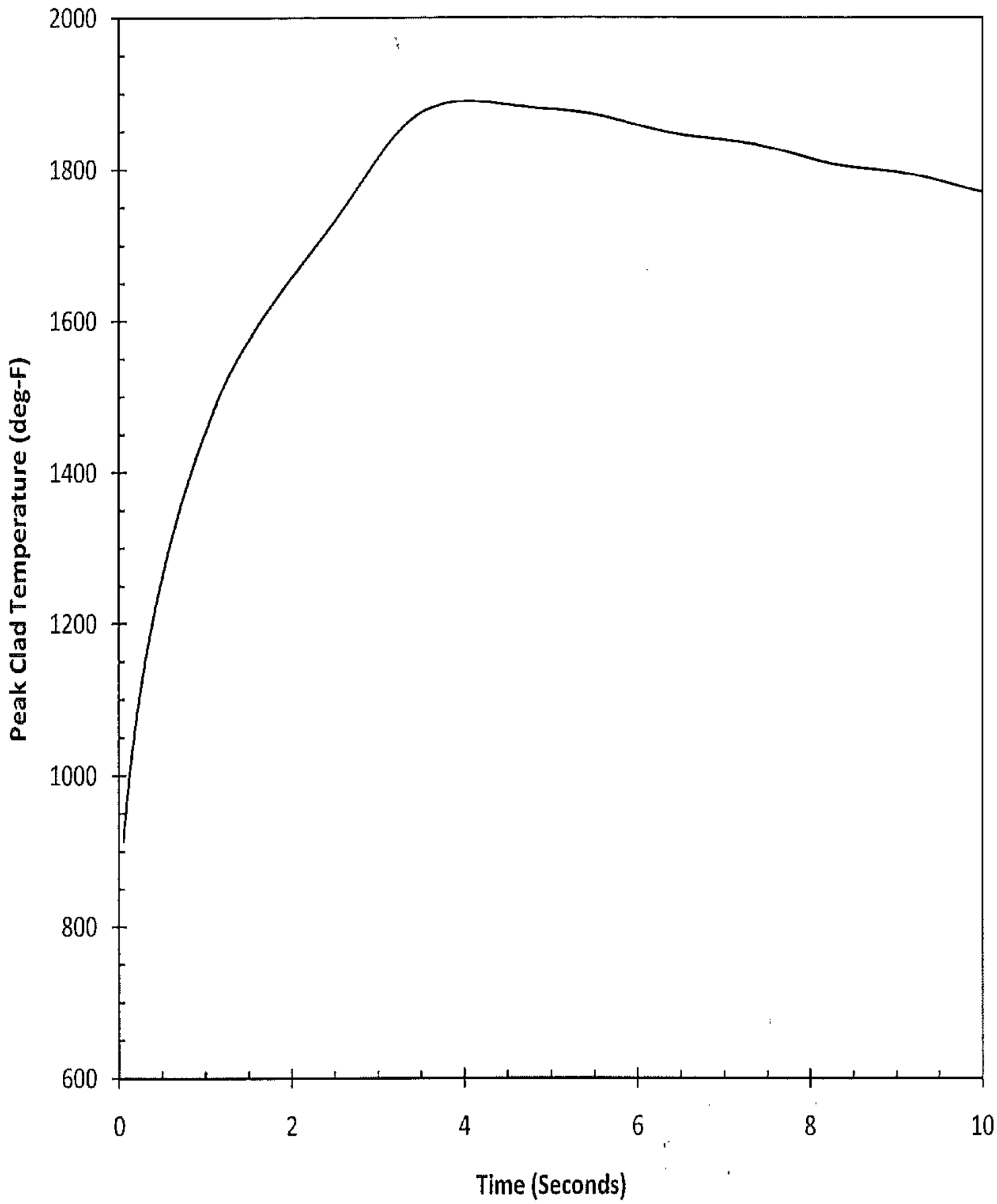
REVISED 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING
ONE LOCKED ROTOR
CORE AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS
FIGURE 14.1.9-11



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REVISED 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ALL LOOPS INITIALLY OPERATING,
ONE LOCKED ROTOR
PEAK CLAD TEMPERATURE
TRANSIENT

FIGURE 14.1.9-12

14.1.10 LOSS OF EXTERNAL ELECTRICAL LOAD

The loss of external electrical load may result from an abnormal increase in network frequency, or an accidental opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid large load reduction by the action of the turbine control. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of all non-emergency AC power is presented in Section 14.1.12.

The unit is designed to accept a 50 percent step loss of load without actuating a reactor trip with all NSSS control systems in automatic (reactor control system, pressurizer pressure and level, steam generator water level control, and steam dumps). Depending on the full power average temperature and steam generator pressure conditions, the automatic turbine bypass system can accommodate approximately 27 to 34 percent design flow to the condenser to accommodate this abnormal load rejection by reducing the transient imposed upon the reactor coolant system (RCS). The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the rod control system. The pressurizer power-operated relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not lift in this case. C26

A loss of external load would normally trip the reactor directly from a signal derived from the turbine emergency trip header pressure (a two out of three signal). Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. C26

In the event the turbine bypass valves fail to open following a large load loss, the main steam safety valves lift and the reactor may be tripped by the high pressurizer pressure signal, high pressurizer level signal or the overtemperature ΔT signal. In the event of feedwater flow also being lost, the reactor may also be tripped by a steam generator low-low water level signal. The steam generator shell side pressure and reactor coolant temperatures increase rapidly. The pressurizer safety valves are sized to protect the RCS against overpressure without taking credit for the turbine bypass system, pressurizer spray, pressurizer power-operated relief valves, automatic RCCA control, or the direct reactor trip on turbine trip. C26

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 main steam safety valves. The pressurizer and main steam safety valves are then able to maintain the RCS and Main Steam System pressures within 110% of the corresponding design pressure without a direct reactor trip on turbine trip action.

The Turkey Point Units 3 and 4 Reactor Protection System and primary and secondary system designs preclude overpressurization without requiring the automatic rod control, pressurizer pressure control, and/or turbine bypass control system.

Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power, without direct reactor trip, primarily to show the adequacy of the pressure-relieving devices, and also to demonstrate core protection margins; i.e., the turbine is assumed to trip without actuating all 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 transient. In addition, no credit is taken for the turbine bypass system. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The turbine trip transients are analyzed by employing the detailed digital computer program RETRAN (Reference 1). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and main steam safety valves. The program computes pertinent plant variables, including temperatures, pressures, DNBR, and power level. C26

Three cases are analyzed for a total loss of load from full power conditions.

1. Minimum DNBR

2. Peak RCS Pressure C26

3. Peak Main Steam System (MSS) Pressure

The major assumptions used in these analyses are summarized below:

A. Initial Operating Conditions

The initial core power, reactor coolant temperature, and reactor coolant pressure are assumed at the most limiting nominal values. The DNBR calculations are performed using the Revised Thermal Design Procedure (Reference 2), in which the uncertainties in the initial conditions are included in the DNBR limit value. For the peak RCS and MSS pressure calculations, uncertainties of 0.3%, 53 psi (including bias), and 6.0°F are applied in the most limiting direction to the initial core power, reactor coolant pressure, and reactor coolant temperature. C26

B. Reactivity Coefficients

The loss of load accident is analyzed with minimum reactivity feedback. These cases assume a moderator temperature coefficient of 0 pcm/°F and the least negative Doppler coefficients.

The loss of load event results in a primary system heatup and therefore is conservatively analyzed with minimum reactivity feedback. C26

C. Reactor Control

From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual rod control.

If the reactor were in automatic rod control, the control rod banks would move prior to trip and reduce the severity of the transient.

D. Pressurizer Spray and Power-Operated Relief Valves

Three cases are analyzed:

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

b. For the peak RCS pressure case, no credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable. C26

c. For the peak MSS pressure case, full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the primary coolant pressure, thereby delaying the time to reactor trip.

The pressurizer safety valve modeling includes the effects of the pressurizer safety valve loop seals. For those cases which are analyzed primarily for DNBR and peak MSS pressure, a -3% tolerance was applied to reduce the setpoint such that the pressurizer safety valves begin to open at 2405.8 psia. For those cases which are analyzed primarily for peak RCS pressure, a +2% tolerance and a +0.8% set pressure shift were applied to increase the set point pressure by a total of 2.8%, such that the pressurizer safety valves begins to open at 2548.7 psia. Additionally, no steam flow is assumed until the valve loop seals are purged. C26 C26

E. 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, however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

F. Reactor Trip

Only the overtemperature ΔT and high pressurizer pressure reactor trips are assumed operable for the purposes of this analysis. No credit is taken for a reactor trip on high pressurizer level, the direct reactor trip on turbine trip, or the low-low steam generator water level reactor trip.

G. Steam Release

No credit is taken for the operation of the steam dump to condenser or atmosphere. This assumption maximizes both primary and secondary pressure. The main steam safety valves are assumed to be fully open at the valve set-pressure plus 3% setpoint tolerance and 5 psi valve accumulation.

H. Pressure Drop in the Main Steam Safety Valves Piping

The pressure drop in the piping between the steam generators and the Main Steam Safety Valves is included (Reference 3).

Results

The transient responses for a total loss of load from 100 percent of full-power operation are shown for three cases. The calculated sequence of events for the accident is shown in Table 14.1.10-1. Figures 14.1.10-1 through 14.1.10-9 show the transient responses for the same cases.

Case 1:

Figures 14.1.10-1 through 14.1.10-3 show the total loss of load accident (Peak RCS Pressure case), assuming the plant to be operating at full power with maximum steam generator tube plugging (10%), and 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. In this case, the pressurizer safety valves are actuated at a conservatively high setpoint.

Case 2:

Figures 14.1.10-4 through 14.1.10-6 show the transient response for the total loss of load (Minimum DNBR case) with minimum reactivity feedback, maximum steam generator tube plugging (10%), and 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 departure from DNBR is well above the limit value. The pressurizer safety valves are actuated at a conservatively low setpoint. C26

Case 3:

Figures 14.1.10-7 through 14.1.10-9 show the transient response for the total loss of load accident (Peak MSS Pressure case), assuming the plant to be initially operating at full power with minimum steam generator tube plugging (0%) and 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 on the overtemperature ΔT trip signal. The pressurizer safety valves are actuated at a conserviatively low setpoint. C26

In addition to the above cases, the loss of external electrical load followed by a failure of the RPS to shut down the reactor is considered an ATWS event. See Section 14.1.15 for applicable discussion.

Conclusions

The analysis indicates that a total loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System and the Steam System. All of the applicable acceptance criteria are met. The minimum DNBR for each case is greater than the safety analysis limit value. The peak primary and secondary pressures remain below 110% of design at all times.

REFERENCES

1. WCAP-14882-P-A (Proprietary) and WCAP-15234-A (Non-Proprietary), Huegel, D. S.,et.al., “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” April 1999. C26

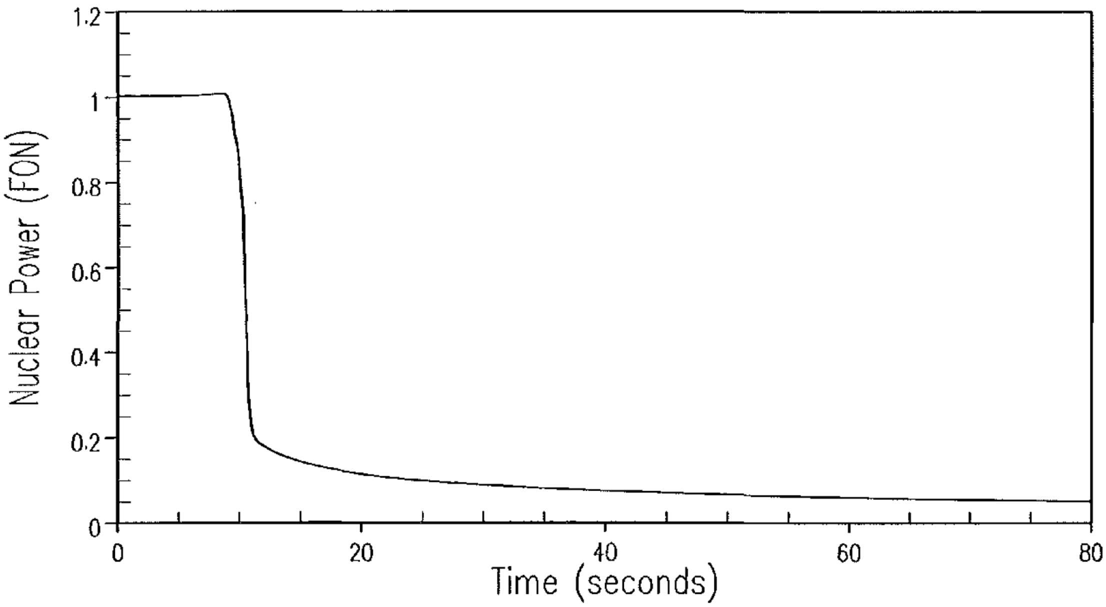
2. Westinghouse WCAP-11397-P-A (Proprietary), Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," April 1989.

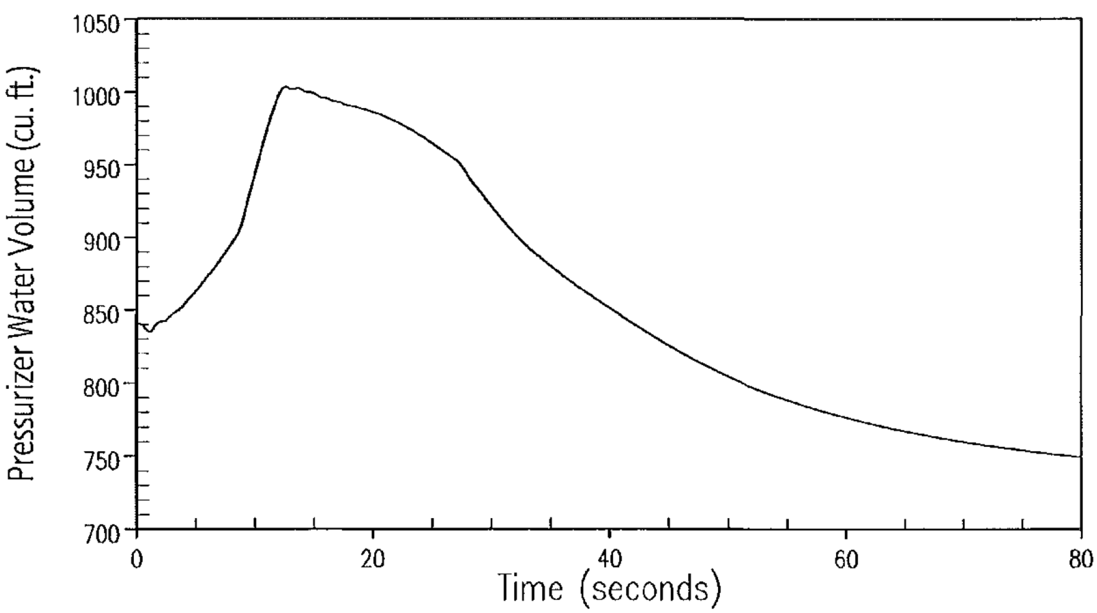
3. "Florida Power & Light Co., Turkey Point Units 3 & 4, UFSAR and AABD Updates Resulting from Reanalysis of Non-LOCA Transients Impacted by ΔP Losses Between the Steam Generator and MSSVs and LOCA Assessment," J. J. Deblasio to J. Perryman, FPLN-97-0108, NSD-SAE-ESI-97-342, June 2, 1997.

TABLE 14.1.10-1

SEQUENCE OF EVENTS - LOSS OF LOAD/TURBINE TRIP ACCIDENTS

Case	Event	Time (sec)
Peak RCS Pressure	Turbine Trip	0.0
	High Pressurizer Pressure setpoint reached	6.3
	Rods begin to drop	8.3
	Peak RCS pressure occurs	8.9
Minimum DNBR	Turbine Trip	0.0
	OTΔT Reactor Trip setpoint reached	14.3 C28
	Minimum DNBR occurs	15.0
	Rods begin to drop	16.3 C28
Peak MSS Pressure	Turbine Trip	0.0
	OTΔT Reactor Trip setpoint reached	12.6
	Rods begin to drop	14.6 C28
	Peak MSS pressure occurs	20.2





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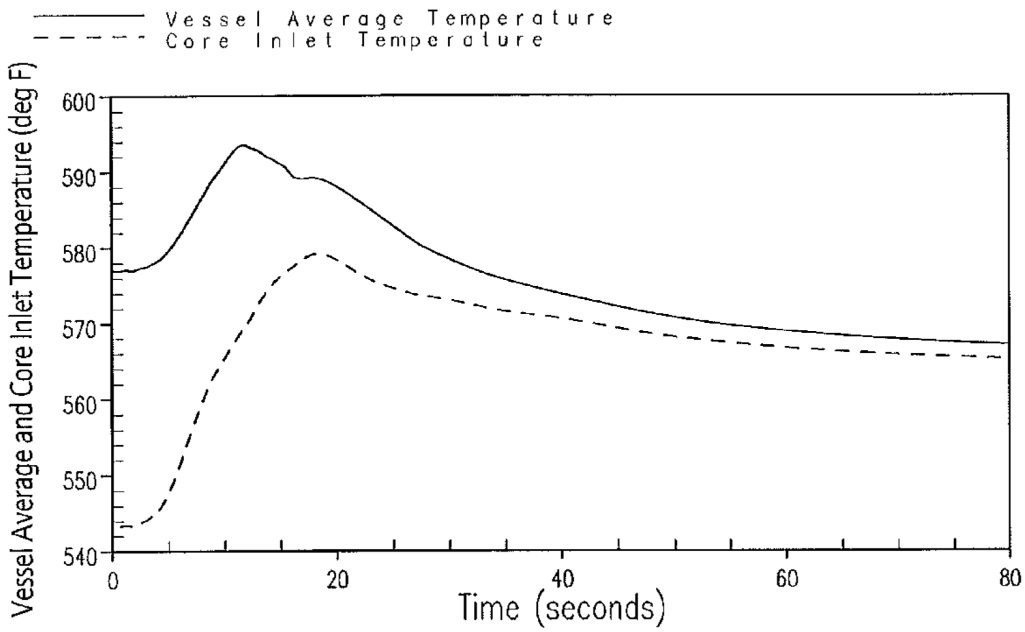
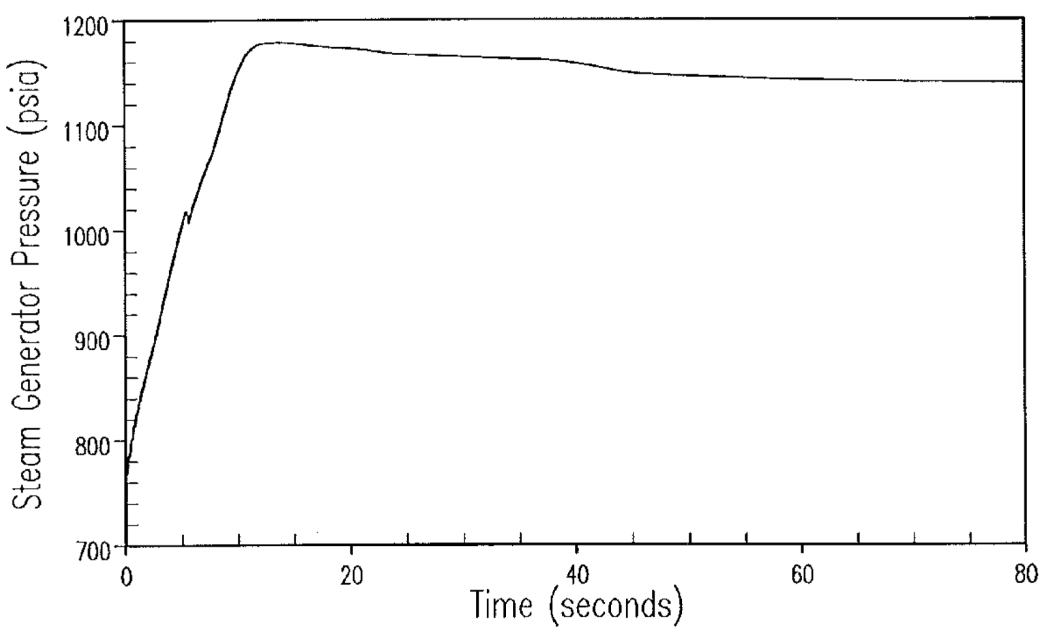
TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD PEAK RCS PRESSURE CASE

FIGURE 14.1.10-1





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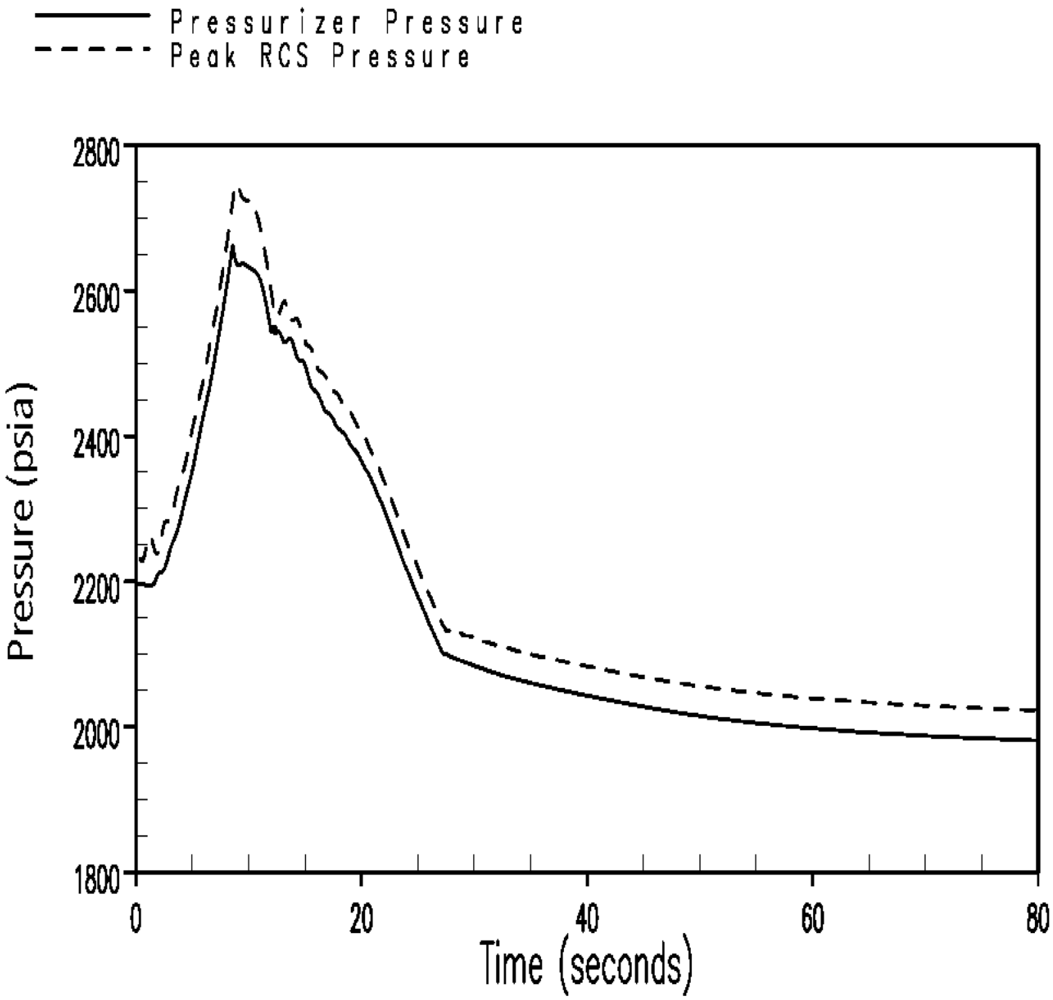
TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD PEAK RCS PRESSURE CASE
FIGURE 14.1.10-2



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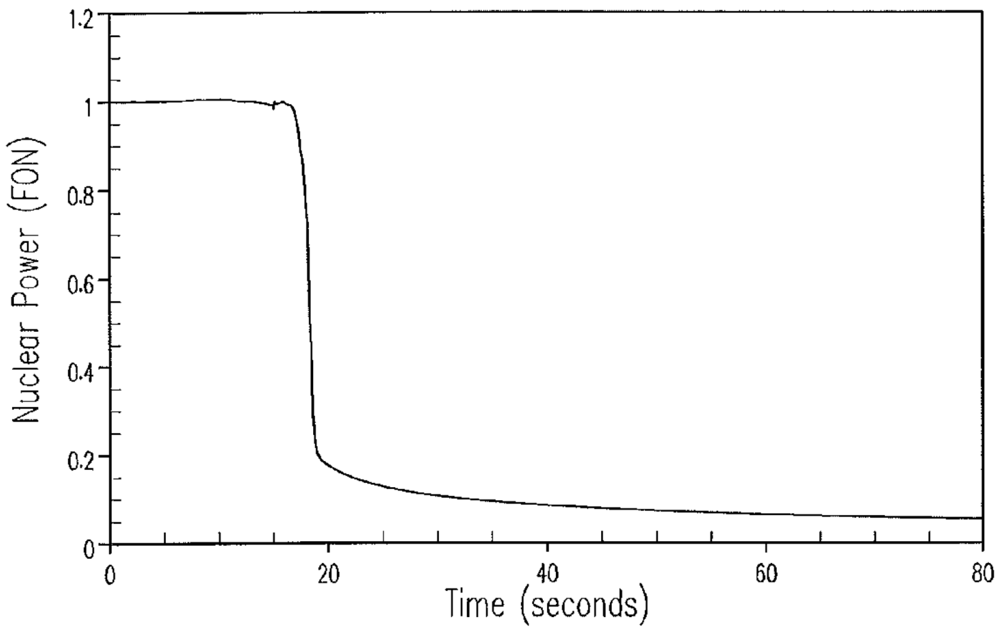
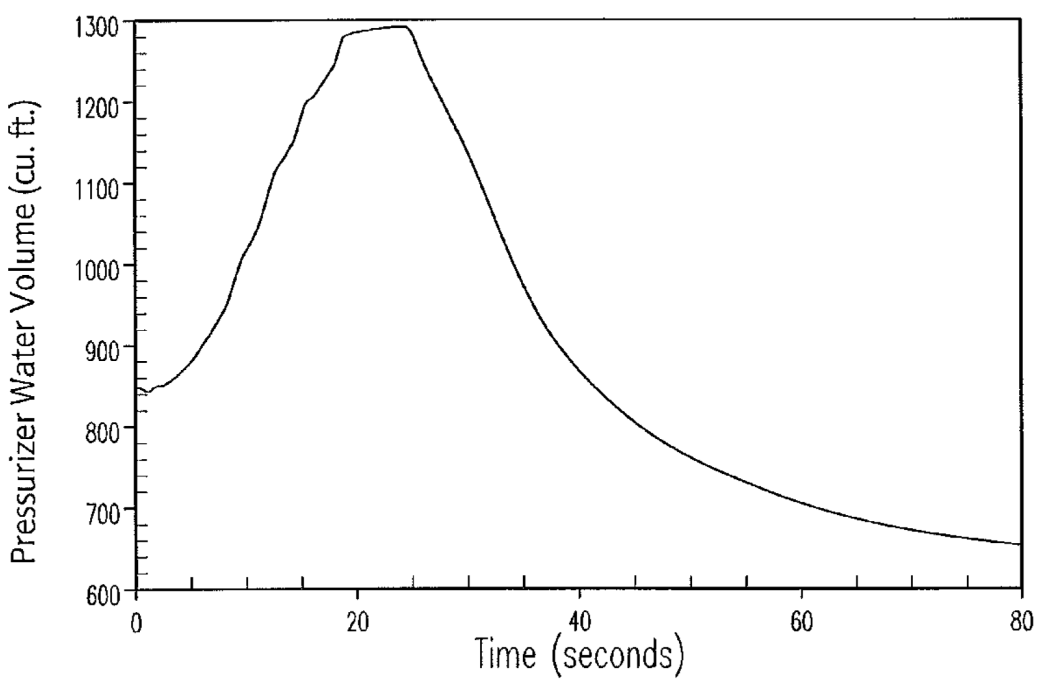
TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD PEAK RCS PRESSURE CASE
FIGURE 14.1.10-3





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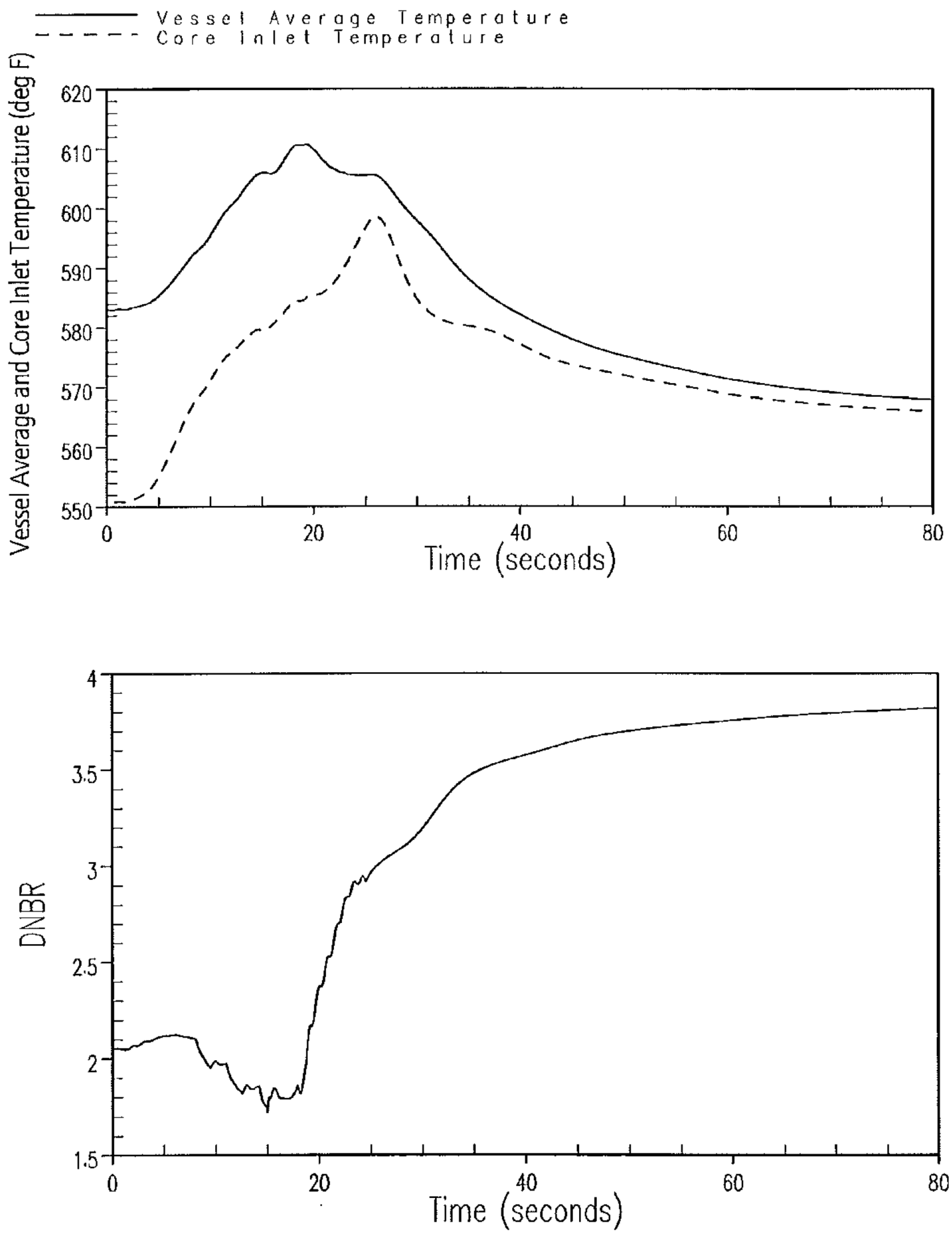
TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD
MINIMUM DNBR CASE
FIGURE 14.1.10-4



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TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD
MINIMUM DNBR CASE
FIGURE 14.1.10-5

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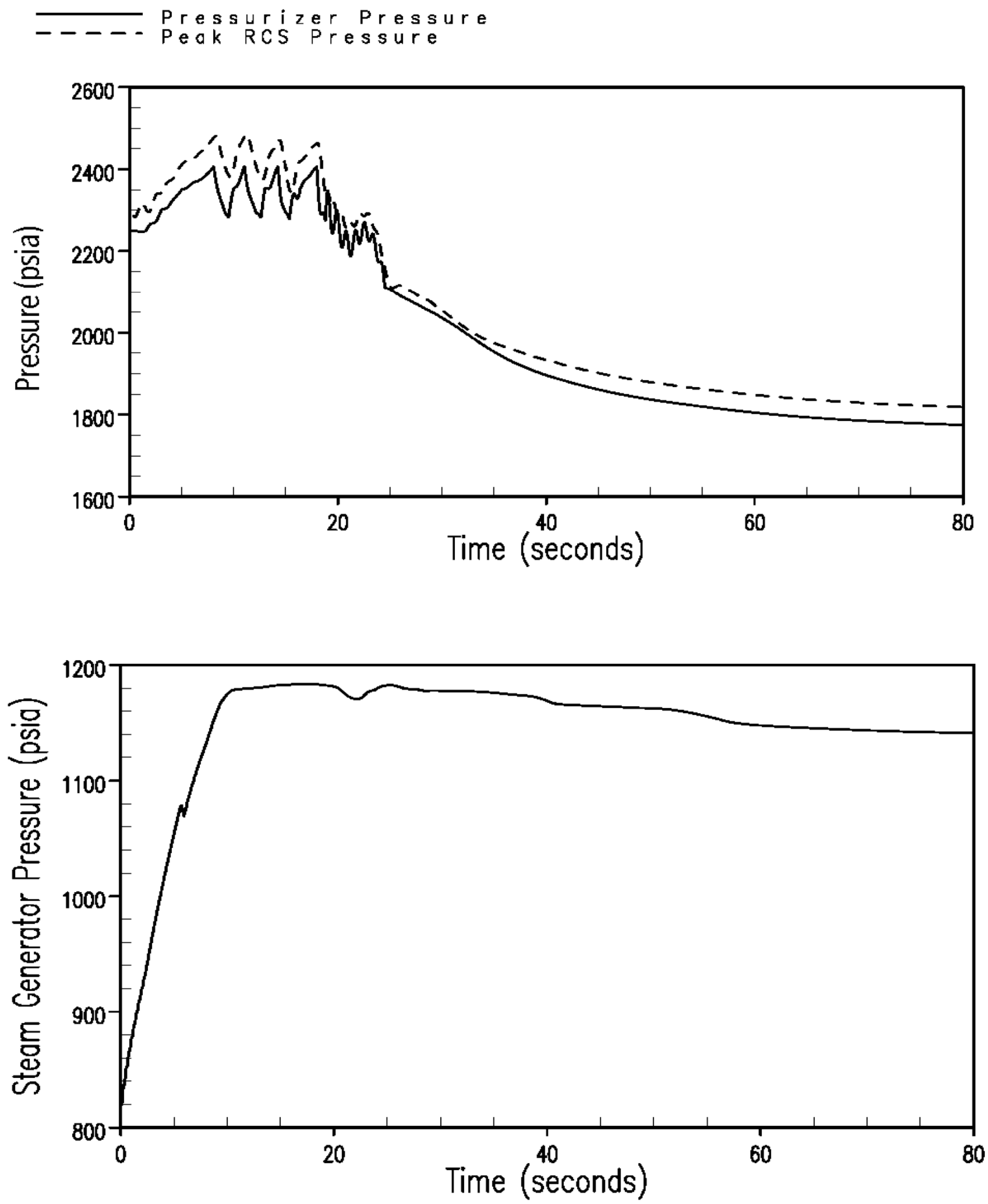

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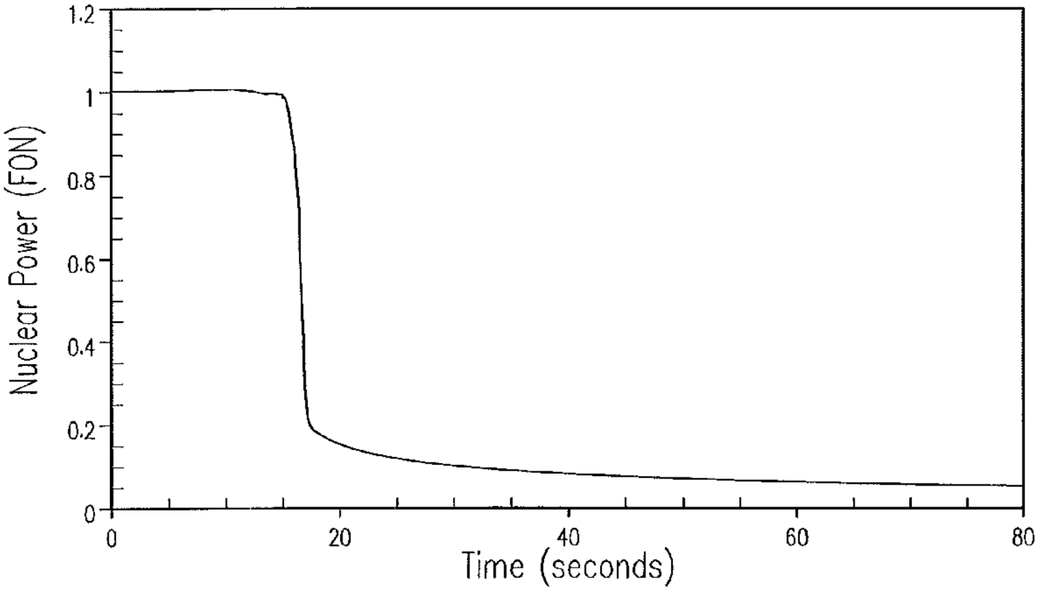
TOTAL LOSS OF EXTERNAL ELECTRICAL
LOAD MINIMUM DNBR CASE

FIGURE 14.1.10-6



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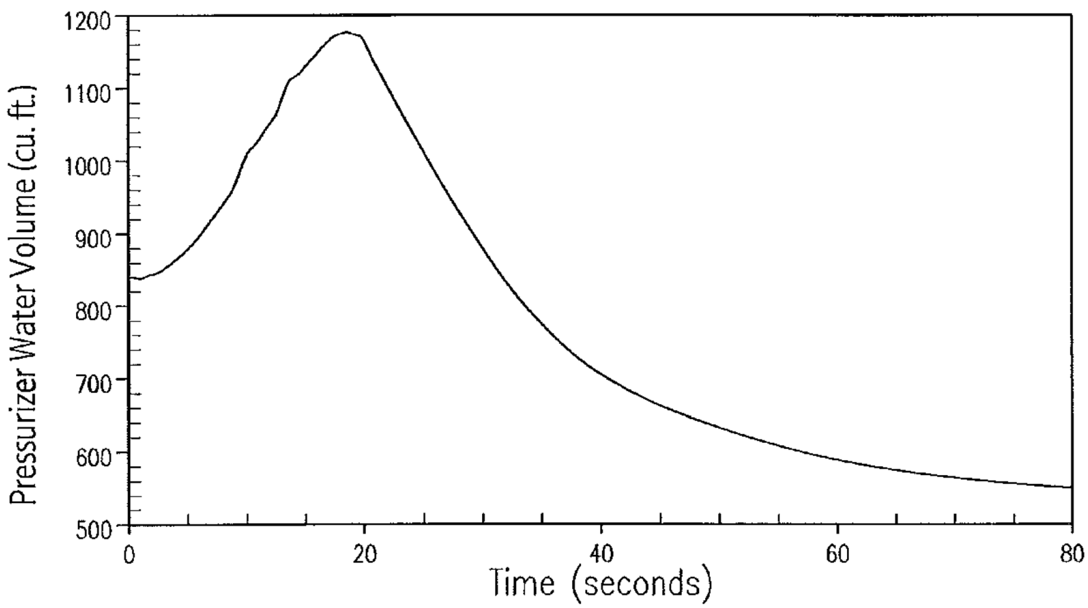

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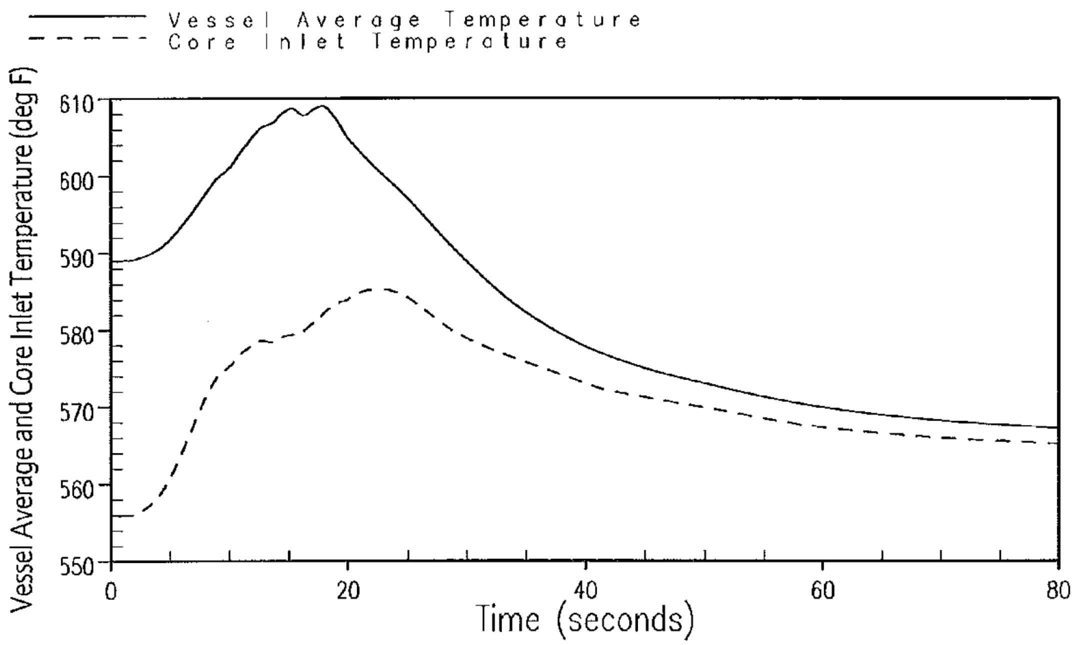
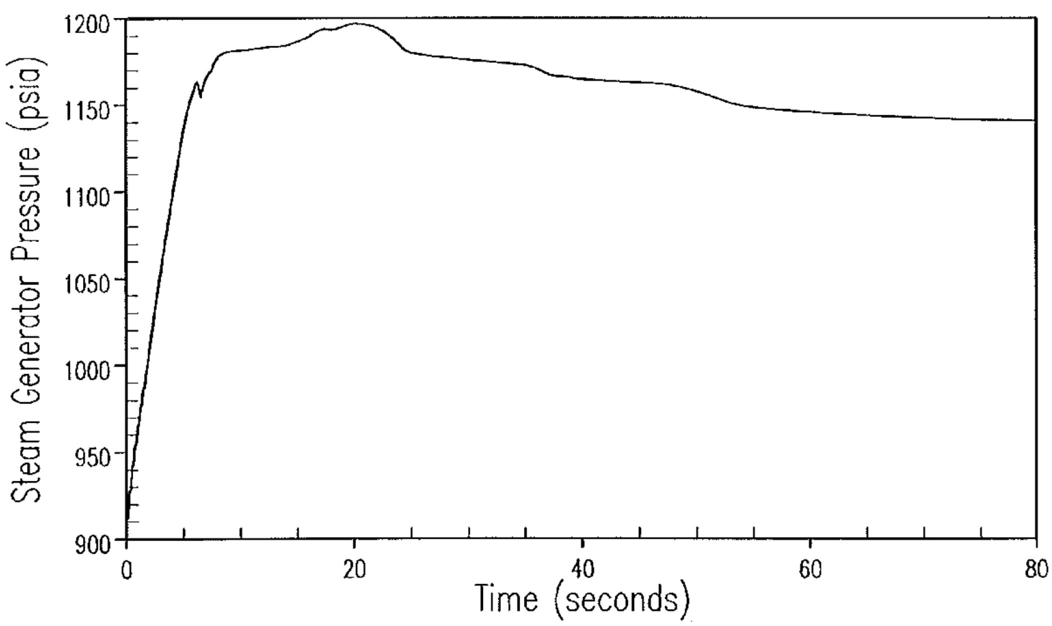
TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD
PEAK MSS PRESSURE CASE
FIGURE 14.1.10-7





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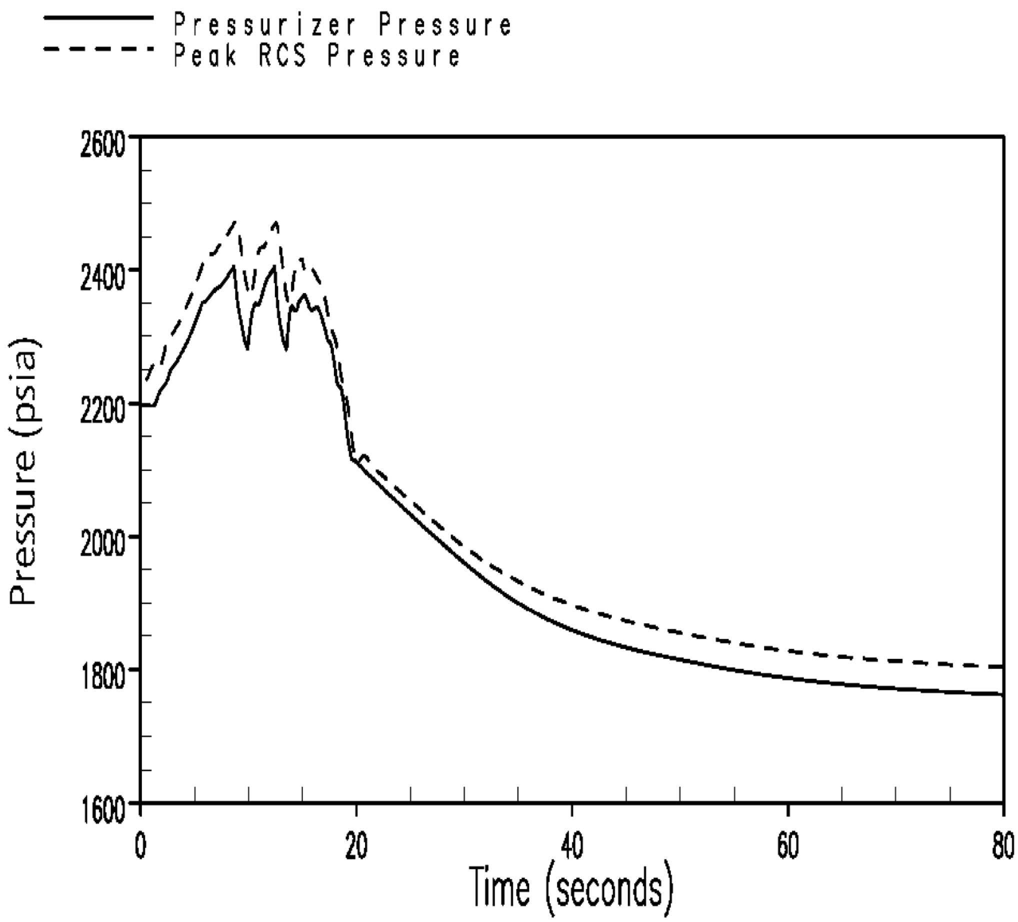
TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD PEAK MSS PRESSURE CASE
FIGURE 14.1.10-8



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TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD
PEAK MSS PRESSURE CASE
FIGURE 14.1.10-9

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TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT PRESSURE CONTROL

MAXIMUM REACTIVITY FEEDBACK

FIGURE 14.1.10-10

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TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT PRESSURE CONTROL

MAXIMUM REACTIVITY FEEDBACK

FIGURE 14.1.10-11

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TOTAL LOSS OF EXTERNAL ELECTRICAL LOAD WITHOUT PRESSURE CONTROL

MAXIMUM REACTIVITY FEEDBACK

FIGURE 14.1.10-12

14.1.11 LOSS OF NORMAL FEEDWATER FLOW

14.1.11.1 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 alternate 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 Reactor Coolant System (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 occur upon loss of normal feedwater (assuming main feedwater pump failures or valve malfunctions):

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

2. As the no load temperature is approached, the atmospheric dump valves (or safety valves, if the atmospheric dump valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

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

1. Reactor trip on:

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

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2. Three turbine-driven auxiliary feedwater pumps (shared by Units 3 & 4) are started on any of the following:

   a. Low-low water level in any steam generator.
   b. Any safety injection signal.
   c. Loss of offsite power (automatic transfer to diesel generators).
   d. Loss of either A or B 4.16 kV bus on either unit.
   e. Trip of all main feedwater pumps in either unit.
   f. Manual actuation.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the Auxiliary Feedwater System is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS, overpressurization of the secondary side, or water relief from the pressurizer and uncovering of the reactor core.

14.1.11.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

Method of Analysis

A detailed analysis using the RETRAN code (Reference 1) is performed in order to obtain the plant transient conditions following a loss of normal feedwater. The analysis addresses the core neutron kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and auxiliary feedwater system. The digital program computes pertinent variables including the pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature. C26

Assumptions made in the analysis are:

1. The plant is initially operating at 100.3 percent of the nominal NSSS power of 2652 MWt which includes a nominal reactor coolant pump heat of 8.0 MWt. The reactor coolant volumetric flow is assumed to remain constant at its Thermal Design value. Although not assumed in the analysis, the reactor coolant pumps may be manually tripped at some later time to reduce the heat addition to the RCS. C26

2. The direction of conservatism for both the initial reactor vessel average coolant temperature and the pressurizer pressure can vary. As such, cases were considered with the initial temperature and pressure uncertainties applied in each direction. The initial average temperature uncertainty was conservatively assumed to be ±6.0°F. The initial pressurizer pressure uncertainty was conservatively assumed to be ±53 psi. The most limiting LONF case was that which modeled the temperature uncertainty subtracted from the high nominal (window) T_{avg} value and the pressure uncertainty added to the nominal value.

3. Reactor trip occurs on steam generator low-low water level at 4.0% of narrow range span.

4. The worst single failure is assumed to occur in the auxiliary feedwater system. This results in the availability of only one auxiliary feedwater pump supplying a minimum of 373 gpm to three steam generators, 95 seconds following a low-low steam generator water level signal.

5. The pressurizer sprays were assumed to be operable. Separate cases were analyzed with the pressurizer PORVs assumed to be operable versus inoperable; the limiting LONF case was one with the PORVs operable. Even if the pressurizer pressure control features (sprays and PORVs) did not operate, the actuation of the PSVs would prevent the RCS pressure from exceeding the RCS pressure limit during this transient. The pressurizer backup heaters were modeled to actuate only on a low pressurizer pressure signal.

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

7. The main steam safety valves are assumed to be fully open at the valve set-pressure plus 3% setpoint tolerance and an additional 5 psi to simulate valve accumulation.

8. The AFW line purge volume is conservatively assumed to be the maximum value for any loop in either unit, which is 173.21 ft^3. An initial maximum AFW enthalpy of 70.9 Btu/lbm is assumed.

9. Core residual heat generation is based on the 1979 version of ANS 5.1 (Reference 2). 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 trip is assumed.

10. The pressure drop in the piping between the steam generators and the main steam safety valves is included.

11. Credit was taken for a portion of the coolant-to-metal heat transfer that would occur during the long-term primary-side heat-up. A RETRAN thick metal mass heat transfer model was developed for use in the LONF and LOAC event analyses using the RETRAN Thick Metal Mass Heat Transfer Model methodology described in Reference 3. C26

Results

Figures 14.1.11-1 and 14.1.11-2 show the transient response of significant plant parameters following a loss of normal feedwater with the assumptions listed in the previous subsection. C26

The calculated sequence of events for this accident is listed in Table 14.1.11-1. Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction, and because steam flow through the safety valves continues to dissipate the stored and generated heat. Ninety-five seconds after the low-low steam generator water level setpoint is reached, delivery of auxiliary feedwater begins, consequently reducing the rate of water level decrease in the steam generators. C26

The capacity of one auxiliary feedwater pump is such that the water level in the steam generators does not recede below the level at which sufficient heat transfer area is available to dissipate core residual heat and reactor coolant pump heat without water relief from the RCS pressurizer relief or safety valves. Figure 14.1.11-1 shows that at no time is there water relief from the pressurizer. If the auxiliary feedwater delivered is greater than that of one AFW pump, or the initial reactor power is less than 100.3% of the NSSS power, or the steam generator water level in one or more steam generators is above the conservatively low 4% narrow range span level assumed for the low-low steam generator setpoint, the results for this transient will be bounded by the analysis presented. C26

In addition, the loss of normal feedwater followed by a failure of the RPS to shut down the reactor is considered an ATWS event. See Section 14.1.15 for applicable discussion. C26

14.1.11.3 CONCLUSIONS

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the main steam system, since the auxiliary feedwater capacity is such that all applicable acceptance criteria are met.

14.1.11.4 REFERENCES

1. WCAP-14882-P-A (Propriatary) and WCAP-15234-A (Non-propriatery), “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” April 1999. C26

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

3. WCAP-14882-S1-P-A (Proprietary) and WCAP-15234-S1-A (Non-Proprietary), “RETRAN-02 Modeling and Qualification For Westinghouse Pressurized Water Reactors Non-LOCA Safety Analyses, Supplement 1 – Thick Metal Mass Heat Transfer Model and NOTRUMP-Based Steam Generator Mass Calculation Method,” October 2005. C26

TABLE 14.1.11-1

SEQUENCE OF EVENTS
FOR
LOSS OF NORMAL FEEDWATER FLOW

Event	Time (sec)
Main feedwater flow stops	0.0
Low-low steam generator water level Reactor trip setpoint reached	29.1
Rods begin to drop	31.1
Flow from one turbine driven auxiliary feedwater pump is started	124.1
Peak water volume in pressurizer occurs (post trip)	1696.5
Core decay and reactor coolant pump heat decreases to auxiliary feedwater heat removal capacity	~1800

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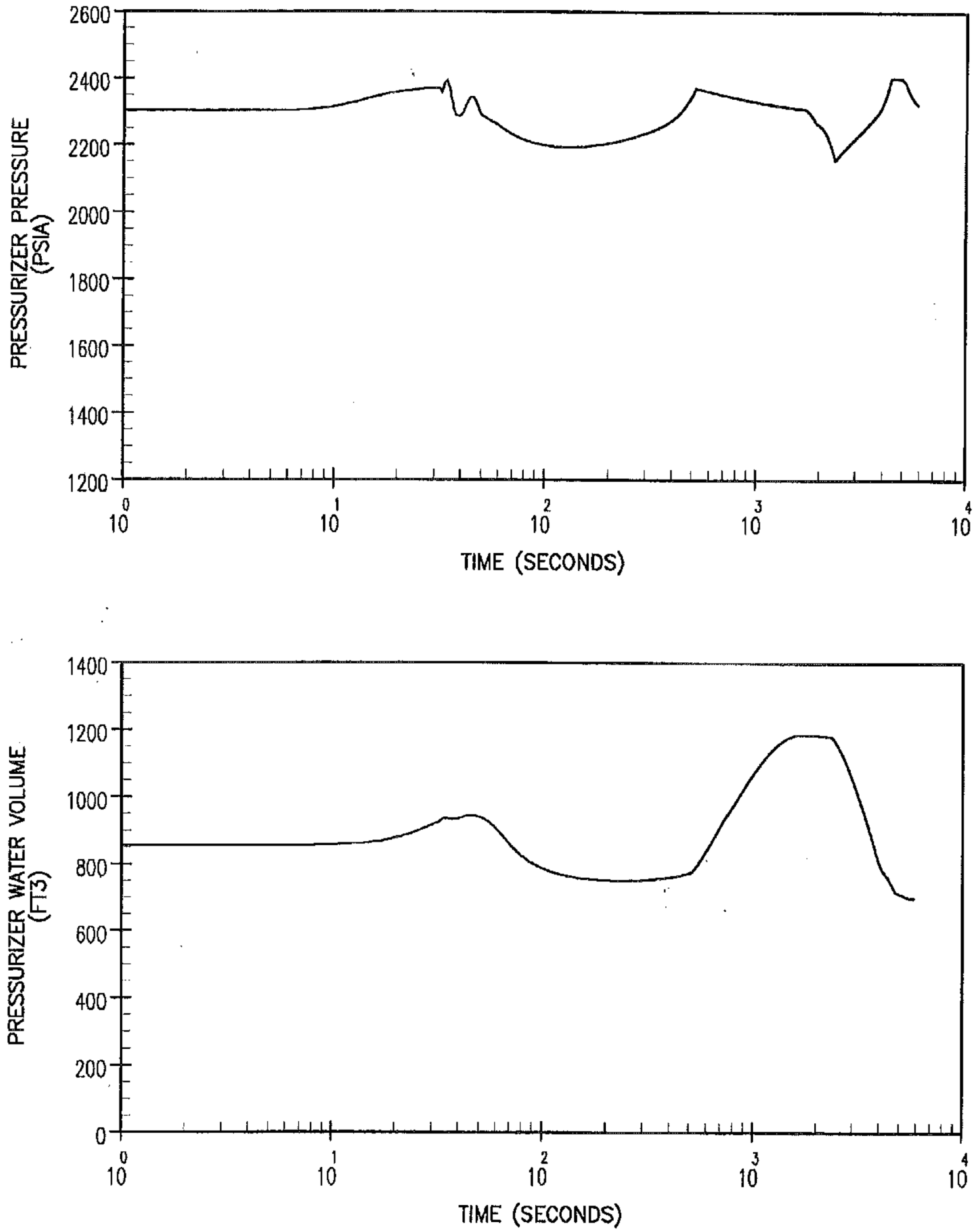

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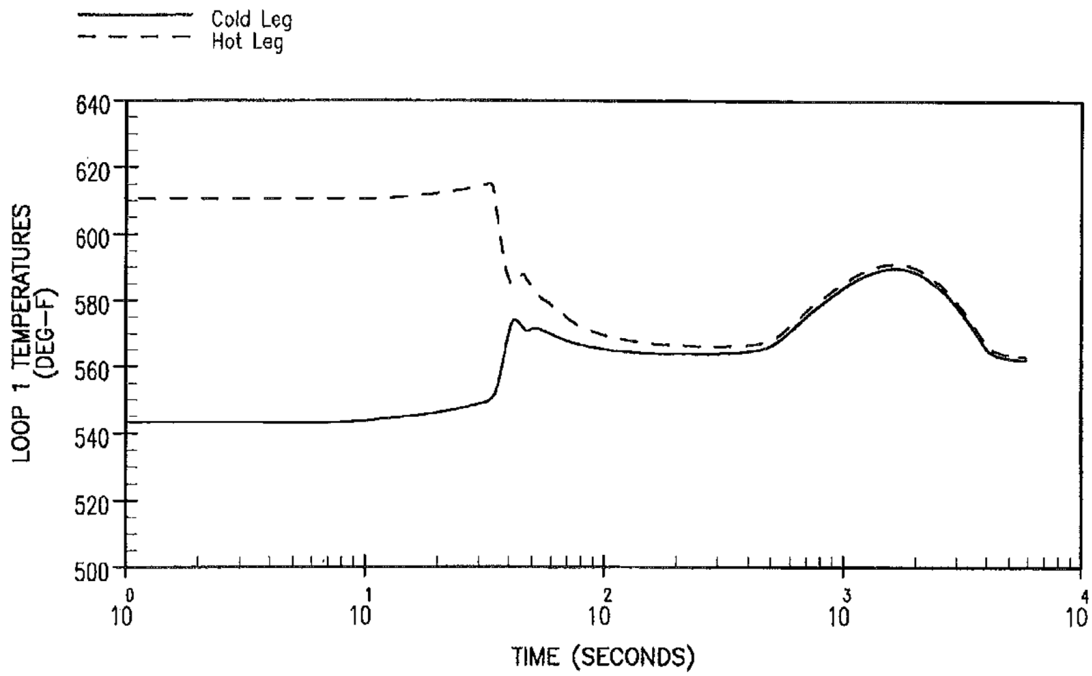
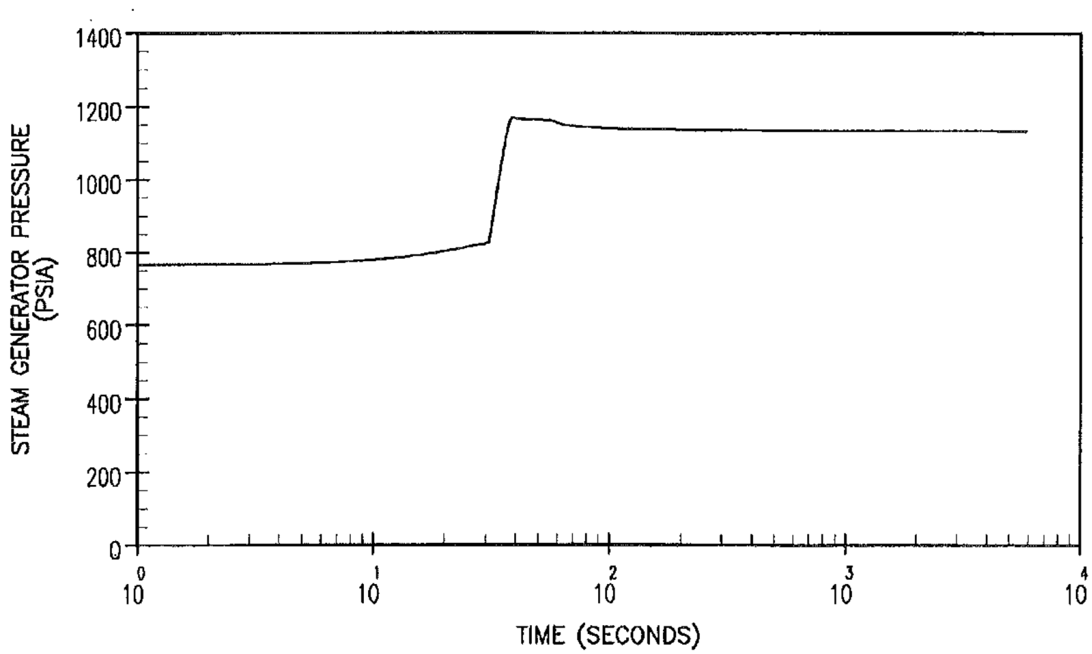
PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR LOSS OF NORMAL FEEDWATER

FIGURE 14.1.11-1





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LOOP TEMPERATURES AND STEAM GENERATOR PRESSURE FOR LOSS OF NORMAL FEEDWATER
FIGURE 14.1.11-2

14.1.12 LOSS OF NON-EMERGENCY A-C POWER TO THE PLANT AUXILIARIES

14.1.12.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

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

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

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

2. As the steam system pressure rises following the trip, the atmospheric dump valves can be opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the dump valves is not available, the main steam safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor. C26

3. As the no load temperature is approached, the atmospheric dump (or safety valves, if the dump valves are not available) is used to dissipate the residual decay heat and to maintain the plant at the hot shutdown condition.

4. Both emergency diesel generators associated with the unit will automatically start following the loss of voltage to the A and B 4160 volt buses of that unit. At the same time, these buses will be isolated from their normal supply and their motor supply and feed breakers will be opened. The breaker from the emergency diesel generator to its associated 4160 volt bus will close energizing the buses. Equipment will be sequentially loaded on to the 4160 volt buses, load centers and motor control centers will be energized as controlled by the load sequencers. All required additional manual loads will be powered by the emergency diesel generators as required by procedures.

The following provides the necessary protection against a loss of AC power:

1. Reactor trip on:

   a. Low-low water level in any steam generator. C26

2. Three turbine-driven auxiliary feedwater pumps (shared by Units 3 & 4) are started on any of the following:

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

   b. Any safety injection signal.

   c. Loss of offsite power (automatic transfer to diesel generators).

   d. Loss of A or B 4.16 kV bus on either unit. C26

   e. Trip of all main feedwater pumps on either unit.

   f. Manual actuation.

The steam driven auxiliary feedwater pumps are started upon the loss of normal feedwater supply. The auxiliary feedwater turbine utilizes steam from the main steam line to drive the auxiliary feedwater pump to deliver water to the steam generators. The pumps take suction directly from the condensate storage tanks for delivery to the steam generators.

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 auxiliary feedwater in the secondary system. An analysis is presented here to show that the natural circulation flow in the RCS following a loss of AC power event is sufficient to remove residual heat from the core.

The analysis shows that following a loss of all AC power to the station auxiliaries, RCS natural circulation and the AFW system are capable of removing the stored and residual heat; consequently, preventing over-pressurization of the RCS, overpressurization of the secondary side, or water relief from the pressurizer and uncovering of the reactor core. The plant is, therefore, able to return to a safe condition.

Turkey Point Units 3 and 4 share common auxiliary feedwater systems. Thus, a loss of non-emergency AC power to the plant auxiliaries could simultaneously affect both units. The auxiliary feedwater system would then be required to provide flow to both units.

The worst single failure in the auxiliary feedwater system could result in availability of only one of the three auxiliary feedwater pumps. Flow from this pump could be as low as 312.4 gpm to one of the units until the operator takes action from the control board to realign the flow split to the units. C26

The analysis is performed for one unit, representing the worst case of the two units.

14.1.12.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

Method of Analysis

A detailed analysis using the RETRAN Code (Reference 1) is performed to obtain the plant transient following a loss of all AC power. The analysis addresses the plant thermal kinetics, RCS including the natural circulation, pressurizer, pressurizer PORVs and sprays, steam generators, main steam safety valves, and the auxiliary feedwater system. The digital program computes pertinent variables including the pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature. C26

The major assumptions used in this analysis are identical to those used in the loss of normal feedwater analysis (Section 14.1.11) with the following exceptions. C26

1. Loss of AC power is conservatively assumed to occur as a result of the reactor trip on low-low SG water level, as this maximizes the time of full-power operation without normal feedwater delivery. C26

2. Power is assumed to be lost to the Reactor Coolant Pump 2 seconds following the start of rod motion. This assumption results in the maximum amount of stored energy in the RCS. C26

3. A heat transfer coefficient in the steam generators associated with RCS natural circulation is assumed following the RCP coastdown.

4. The RCS flow coastdown 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, the as-built pump characteristics and conservative estimates of system pressure losses.

5. The worst single failure assumed to occur is in the AFW system. This results in the availability of only one AFW pump supplying 312.4 gpm to three steam generators 95 seconds following a start signal on low-low steam generator water level. This AFW flow is less than that assumed for a loss of normal feedwater, because Turkey Point Units 3 and 4 have a shared AFW system, and a loss of AC (LOAC) power may occur simultaneously at both units. C26

6. As in the LONF cases (Section 14.1.11), the pressurizer sprays were assumed to be operable. Separate cases were analyzed with the pressurizer Power Operated Relief Valves (PORVs) assumed to be operable versus inoperable; the limiting LOCA case was one with the PORVs inoperable. Even if the pressurizer pressure control features (sprays and PORVs) did not operate, the actuation of the pressurizer Safety Valves would prevent the RCS pressure from exceeding the RCS pressure limit during this transient. The pressurizer backup heaters were modeled to actuate only on a low pressurizer pressure signal. C26

7. As in the LONF cases (Section 14.1.11), credit was taken for a portion of the coolant-to-metal heat transfer that would occur during the long-term primary-side heat-up. A RETRAN thick metal mass heat transfer model was developed for use in the LONF and LOAC event analyses using the RETRAN Thick Metal Mass Heat Transfer Model methodology described in Reference 3.

<u>Results</u>

The transient response of significant plant parameters following a loss of AC power is shown in Figures 14.1.12-1 and 14.1.12-2. The calculated sequence of events for this accident is listed in Table 14.1.12-1. C26

The first few seconds after the loss of power to the reactor coolant pumps, the flow transient closely resembles the complete loss of flow event (Section 14.1.9). In the complete loss of flow event, the RCS flow coastdown is the initiating fault and reactor trip occurs after the flow has already been degraded. In the LOAC event, the flow coastdown occurs after reactor trip, making the DNB consequences much less limiting. Therefore, no DNB calculations are performed for the LOAC event. C26

After the reactor trip, stored and residual heat must be removed to prevent damage to the core and the reactor coolant and main steam systems. The RETRAN code results show that the natural circulation and AFW flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown. C26

The capacity of the turbine-driven AFW pump is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to establish enough natural circulation flow in order to dissipate core residual heat without water release through the RCS relief or safety valves. From Figure 14.1.12-1, it can be seen that at no time is there water relief from the pressurizer.

As shown in Figures 14.1.12-1 and 14.1.12-2, the plant approaches a stabilized condition following reactor trip, pump coastdown, and auxiliary feedwater initiation. C26

In addition, the loss of AC power followed by a failure of the RPS to shut down the reactor is considered an ATWS event. See Section 14.1.15 for applicable discussion. C26

14.1.12.3 CONCLUSIONS

Results of the analysis show that, for the loss of non-emergency AC power to the plant auxiliaries event, all safety criteria are met. The DNBR transient, which is not explicitly analyzed for this event, is bounded by the complete loss of flow event (Section 14.1.9) and remains above the safety analysis limit value. AFW capacity is sufficient to prevent water relief through the pressurizer relief and safety valves; this assures that the RCS is not overpressurized. C26

Analysis of the natural circulation capability of the Reactor Coolant System has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or clad damage.

14.1.12.4 REFERENCES

1. WCAP-14882-P-A (Proprietary) and WCAP-15234-A (Non-Proprietary), “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” April 1999. C26

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

3. WCAP-14882-S1-P-A (Proprietary) and WCAP-15234-S1-A (Non-Proprietary), RETRAN-02 Modeling and Qualification For Westinghouse Pressurized Water Reactors Non-LOCA Safety analyses, Supplement 1 – Thick Metal Mass Heat Transfer Model and NOTRUMP-Based Steam Generator Mass Calculation Method,” October 2005 C26

TABLE 14.1.12-1

SEQUENCE OF EVENTS
FOR
LOSS OF NON-EMERGENCY AC POWER

Event	Time (sec)
Main feedwater flow stops	0.0
Low-low steam generator water level Reactor trip setpoint reached	29.4
Rods begin to drop	31.4
Reactor coolant pumps begin to coastdown	33.4
Flow from one turbine driven auxiliary feedwater pump is started	124.4
Peak water volume in pressurizer occurs	1977.0
Core decay heat decreases to auxiliary feedwater heat removal capacity	~1990

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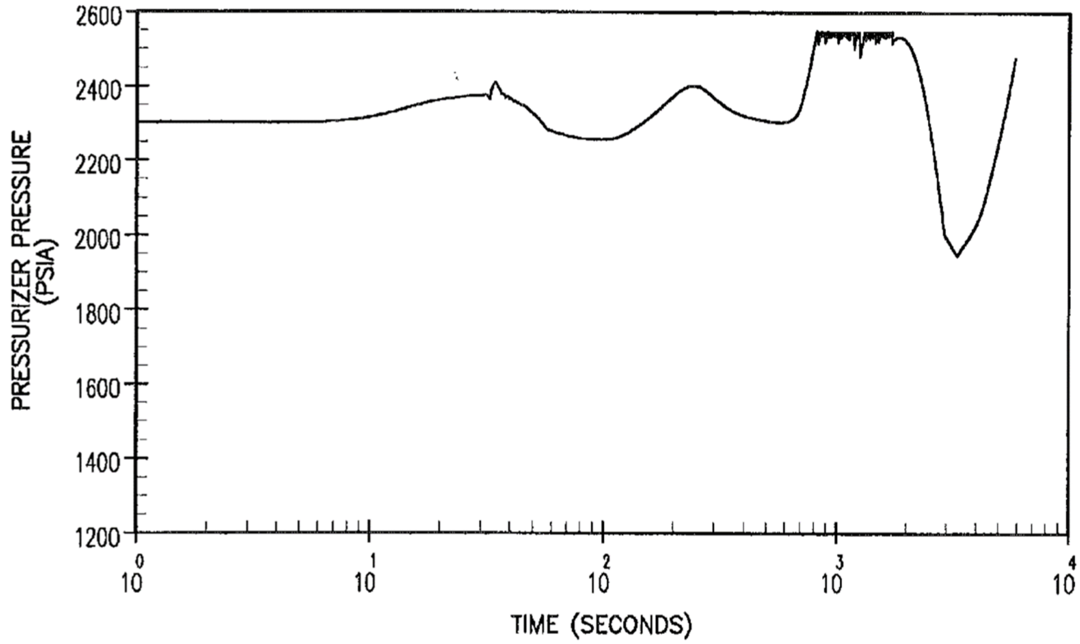

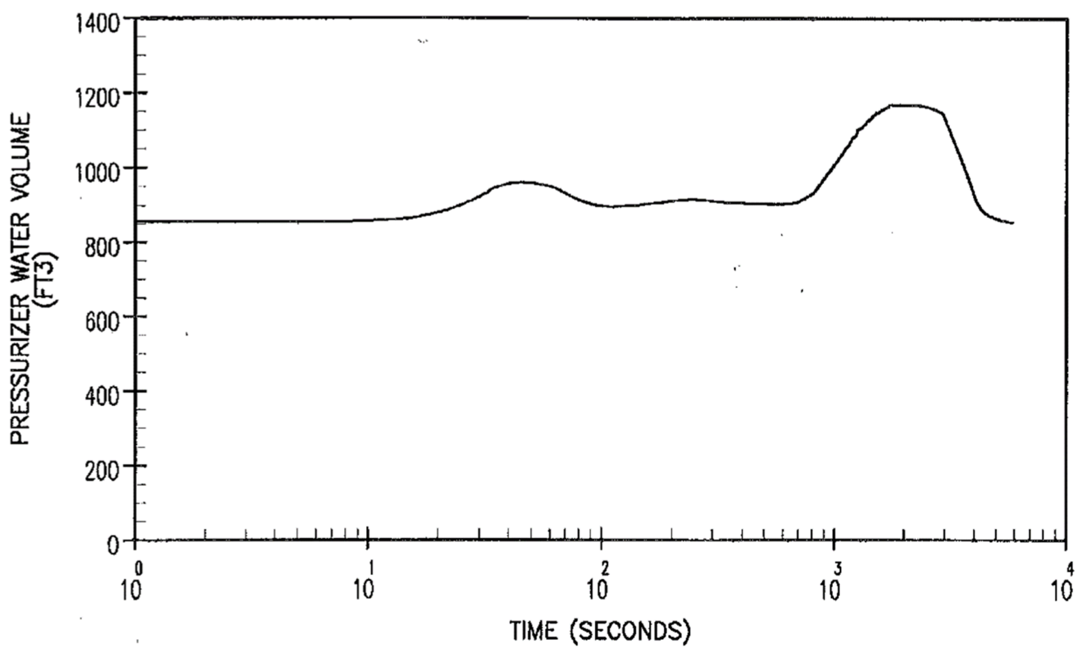

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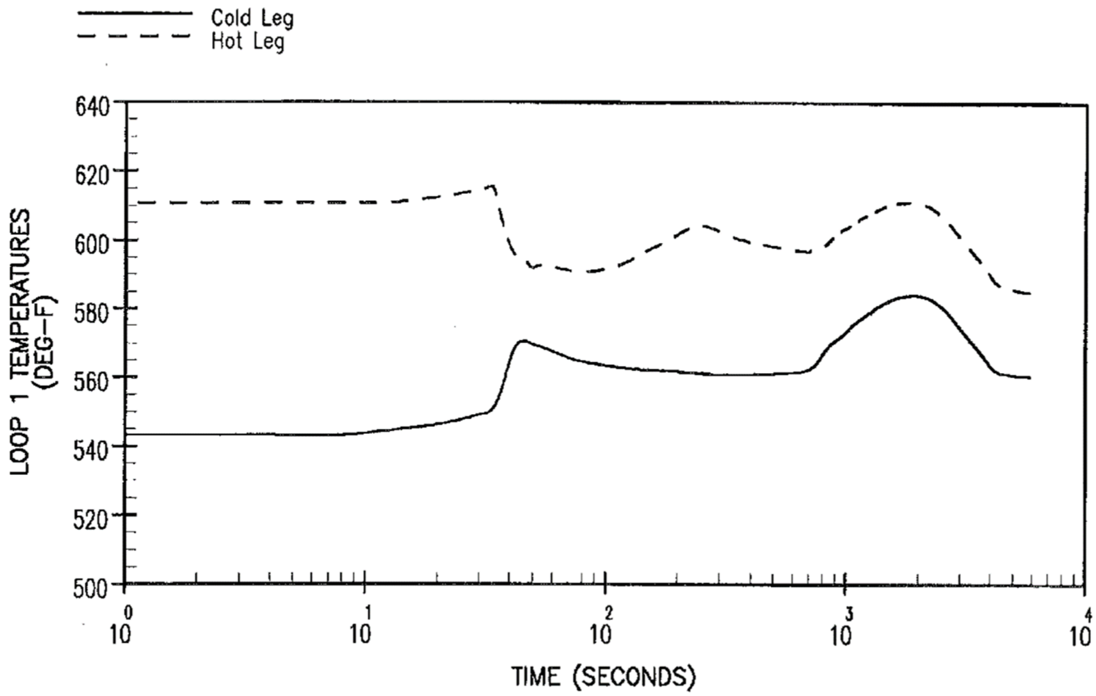
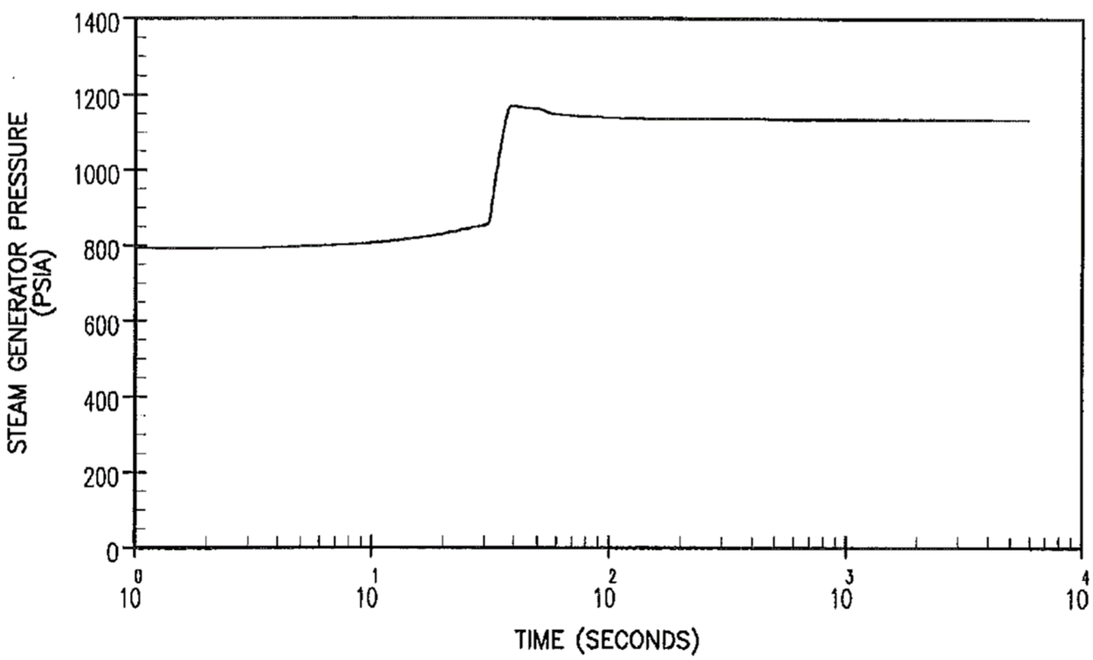
PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR LOSS OF OFFSITE POWER

FIGURE 14.1.12-1





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LOOP TEMPERATURES AND STEAM GENERATOR PRESSURE FOR LOSS OF OFFSITE POWER

FIGURE 14.1.12-2

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14.1.13 TURBINE GENERATOR DESIGN ANALYSIS

Turbine Generator Description

Each turbine generator is a tandem compound four flow machine, with forty five inch last stage blades, which has an operating speed of 1800 rpm.

There is one double flow high pressure cylinder. A sectional view and internal design features are shown in Figures 14.1.13-1 and 14.1.13-2.

There are two double flow low pressure elements. Views of these are shown in Figures 14.1.13-3 and 14.1.13-4.

The turbine mechanical properties are listed in Table 14.1.13-1.

Steam flow of the high pressure turbine is through two main stop control valve assemblies. Each assembly consists of one stop valve with two single seat-type control valves downstream of it, thus providing redundancy in valving.

Exhaust from the high pressure element flows to four moisture separator - reheaters and then to the low pressure elements. In each cross over from a moisture separator - reheater there is a reheat stop valve and an interceptor valve. The stop valves serve as redundant devices to prevent overspeed if the intercepts fail to close when the overspeed trip mechanism operates.

The steam paths described are shown schematically in Figure 10.2-1.

Turbine Generator Speed Control

The turbine generator is a constant speed machine which has its speed controlled by the electrical tie of the generator to the distribution system connected to all generating plants. Output is controlled by the Turbine Digital Control System (TCS), which varies the position of the turbine control valves. The TCS computer has inputs of generator output and turbine inlet pressure which can be used to maintain a constant power output.

In addition to the power related inputs, the TCS continuously monitors turbine speed through speed sensing probes mounted to the shaft. Through the probes the TCS is responsive to speed and acceleration. At 103% of rated speed, it reduces control oil pressure to the Turbine Control Valves and Intercept Valves.

The TCS will also sense a sudden loss of load and closes the control and intercept valves.

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Further, there is an overspeed trip device (the Woodward) which senses separate speed sensors and is monitored by a device independent of the TCS. This overspeed protection is completely independent of the TCS.

In essence there are three levels of speed control:

1. The main electrical tie.

2. The computer using 2 different sets of speed probes.

3. The Woodward using a third set of speed probes.

Energy of Turbine Parts

Modern design, manufacturing and testing practices made the possibility of a major turbine structural failure extremely remote. Disassembled inspection of the turbine ensures that flaws arising during turbine operation are detected and repaired long before they become a potential challenge to turbine structural integrity.

The original low pressure turbine rotors have been replaced with fully integral rotors. The fully integral (FI) rotors have neither discs, nor key-ways to provide areas of stress concentration and stress corrosion cracking previously exhibited by other turbine designs. To ensure that the turbine will not catastrophically fail from stress corrosion cracking, the rotors are inspected at least at the interval recommended by the vendors.

TABLE 14.1.13-1

TURBINE MECHANICAL PROPERTIES(TYPICAL)

P A R T	HP ROTOR	LP ROTOR	HP CASING	LP CASING
Material	26NiCrMoV10-10	3.5 NiCrMoV	CS	ASTM 515-GR65
Tensile Strength, psi, min.	105,878	115,000	70,000	65,000
Yield Strength psi, min.	84,122	100,000	36,000	35,000
Yield Strength, psi, max.	98,626	---	---	---
Elongation in 2", % min.	16	16	22	23
Reduction of area, % min.	50	40	35	---
Impact Strength, Charpy V-Notch ft-lb, min. at room temperature	73.8	40	---	---
50% fracture appearance transition temp, max., °F	32	80	---	---

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STUD MATERIAL

	2 1/2 & Less	Over 2 1/2 to 4	Over 4 to 7
Tensile strength, psi, min.	125,000	115,000	110,000
Yield strength, psi, min.	105,000	95,000	85,000
Elongation in 2", % min.	16	16	16
Reduction in area, % min.	50	50	45

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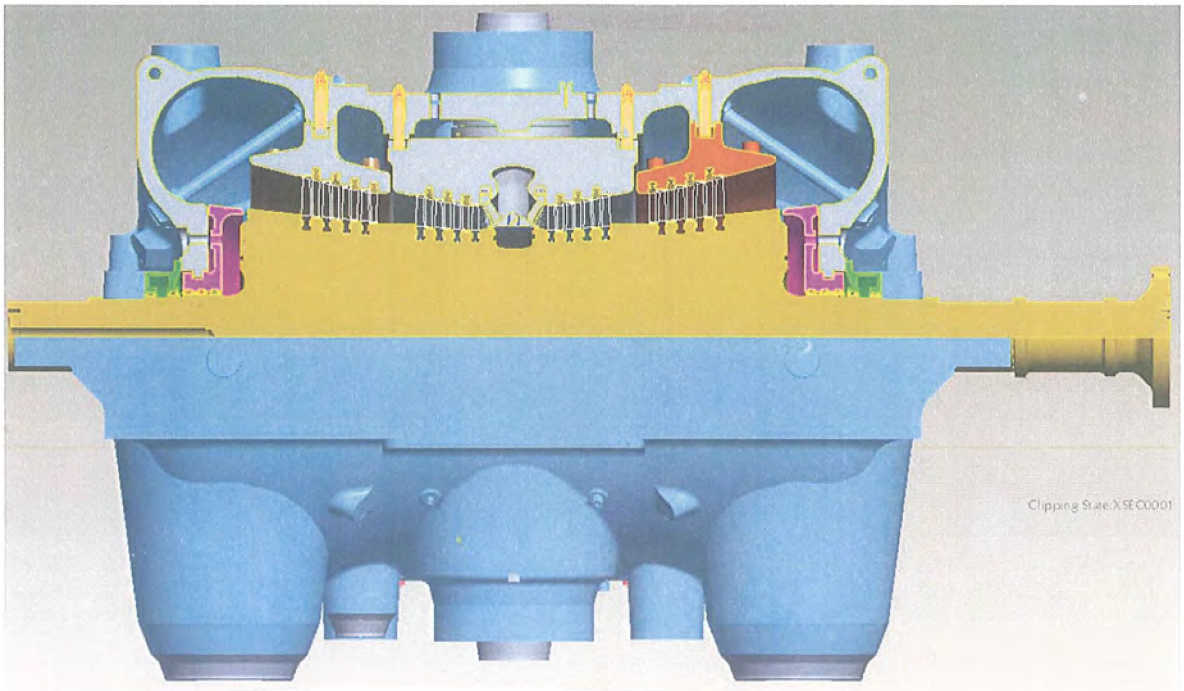

Assembly cross-section of BB96FA HP Turbine

FEATURES:

1. The steam entry in the Inner Casing is by means of four inlets, connected to the outer casing by means of the angle ring connection.
2. The angle ring connection provides steam tight sealing for the main steam while allowing thermal expansion of the inner casing in all directions.
3. Double flow design ensures thrust balance.
4. Dynamic balancing of Rotor before shipment.

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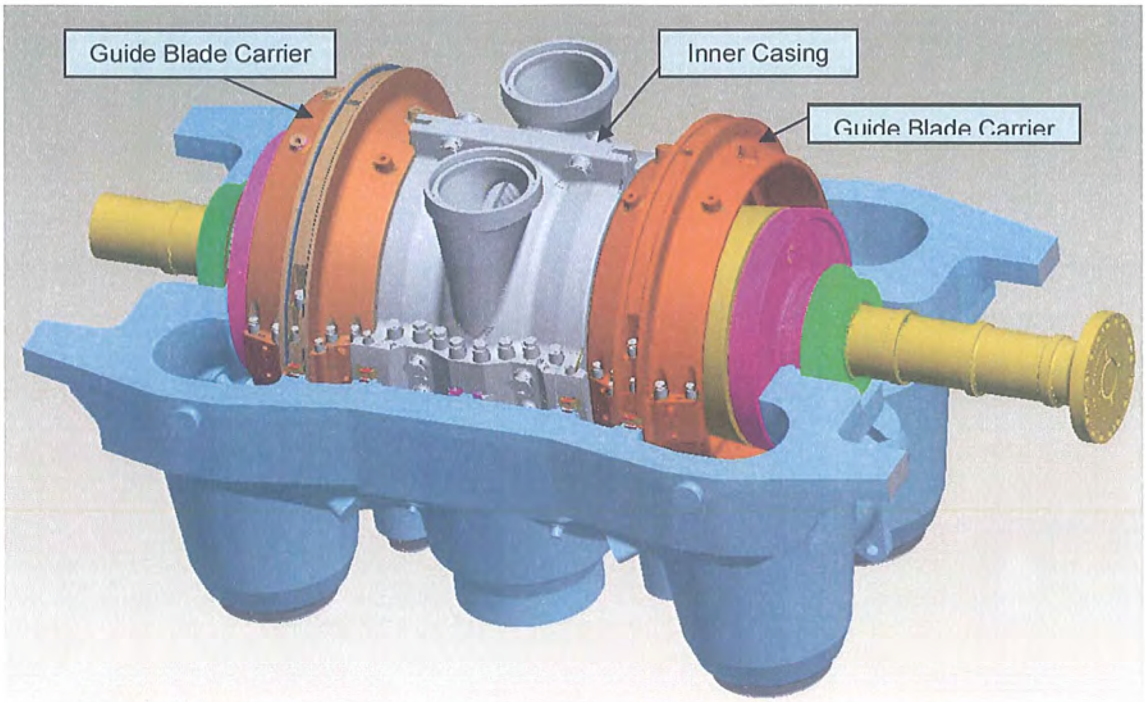
BB96FA HIGH-PRESSURE TURBINE
1800 RPM DOUBLE-FLOW DESIGN
FIGURE 14.1.13-1



BB96FA HP Turbine: Inner Casing & Guide Blade Design Features

FEATURES:

1. The axial fixed point for the Inner Casing is at the center of the HP inlet. The existing locating lugs on the horizontal joint in the outer casing base, used earlier for the fitment of the nozzle chamber to be used as the axial fixed point for the inner casing.

2. The Guide Blade Carrier are axially located by means of circumferential tongue in the outer casing. The Differential pressure across the guide blade carrier keeps the carrier pressed against inlet side of the tongue in the outer casing. This surface acts as the sealing surface between the extraction and exhaust.

3. The Inner casing and Guide Blade Carries are supported in the outer casing by means of support keys and liners. The support keys rest on the outer casing through liners. This support system keeps the Inner Casing/Guide Blade Carries aligned with the outer casing and rotor while allowing for the differential thermal expansion between the various components during operation.

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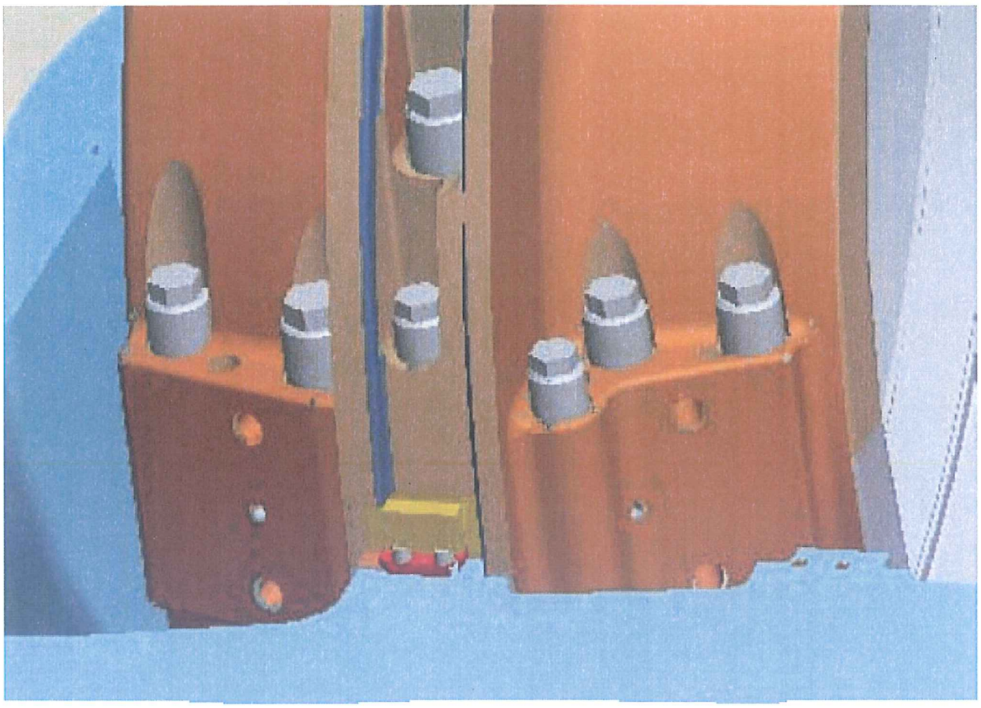
BB96FA HIGH-PRESSURE TURBINE
INNER CASING & GUIDE BLADE CARRIER
ASSEMBLY
FIGURE 14.1.13-2



View of Guide Blade Carrier Assembly in Outer Cylinder

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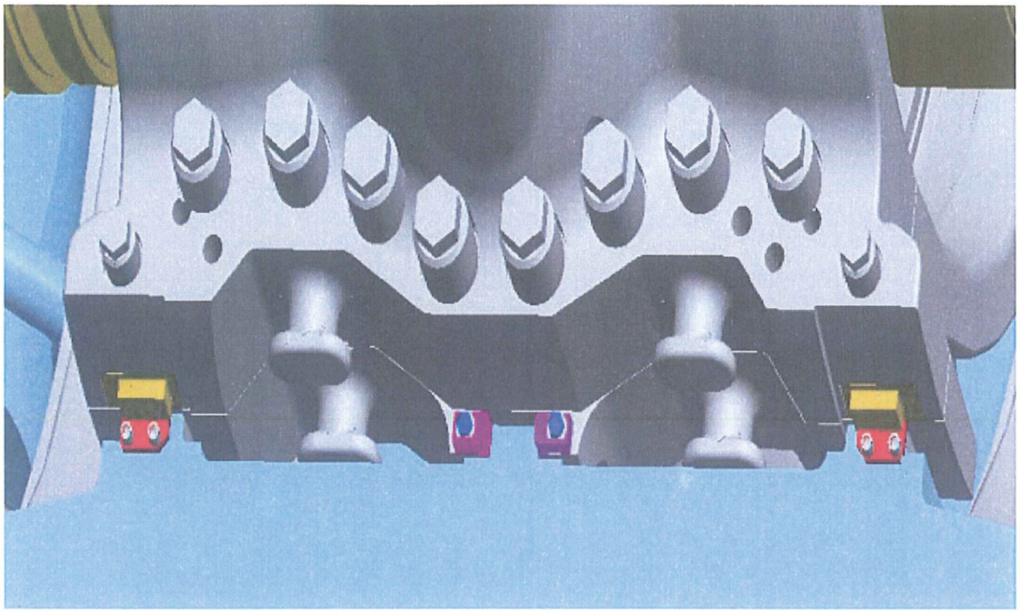
VIEW OF GUIDE BLADE GUIDE BLADE CARRIER ASSEMBLY IN OUTER CYCLINDER

FIGURE 14.1.13-2a



View of Inner Casing Assembly in Outer Cylinder

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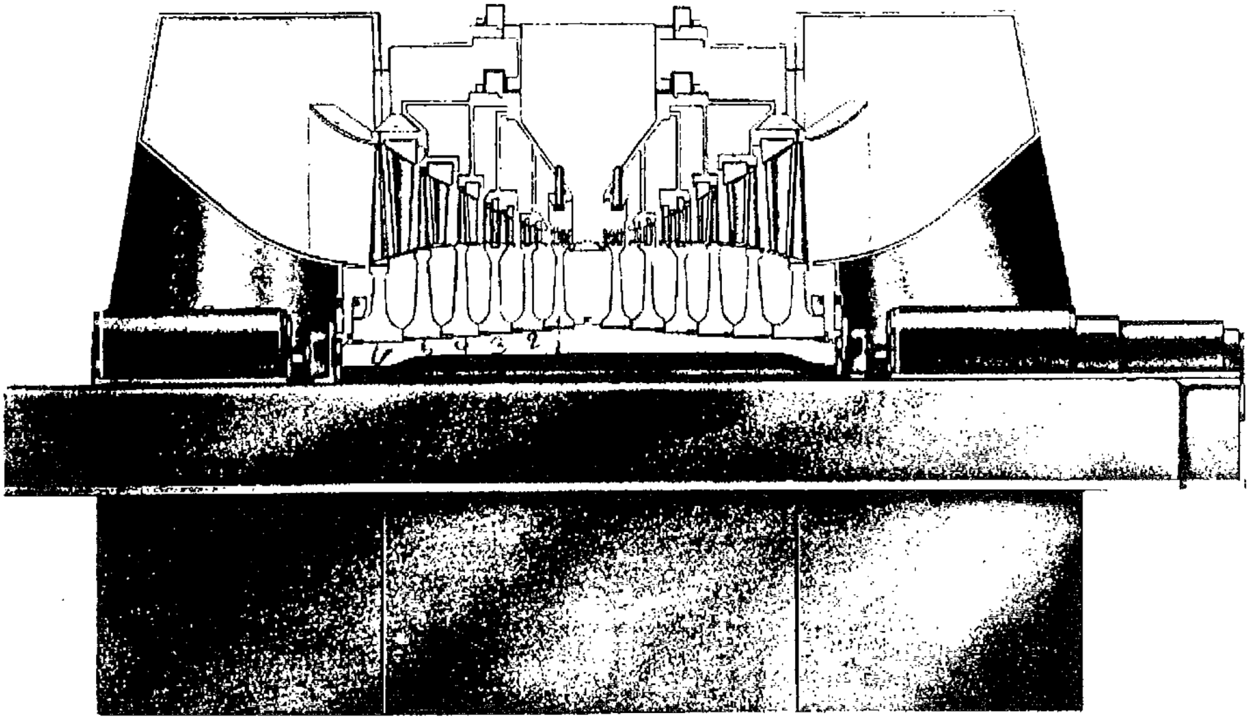
VIEW OF INNER CASING ASSEMBLY IN OUTER CYLINDER

FIGURE 14.1.13-2b

TYPICAL
LOW-PRESSURE ELEMENT
1800-RPM DOUBLE-FLOW DESIGN



FEATURES

1. Blade ring, supported at the horizontal centerline and fixed transversely at the top and bottom by dowel pins, allows freedom of expansion independent of the casing.
2. Entire exhaust casing is at exhaust steam temperature.
3. Exhaust hood of laboratory-proved design minimizes hood loss.
4. Provision for extraction zones with moisture removal.
5. Casing and blade ring of fabricated steel construction.

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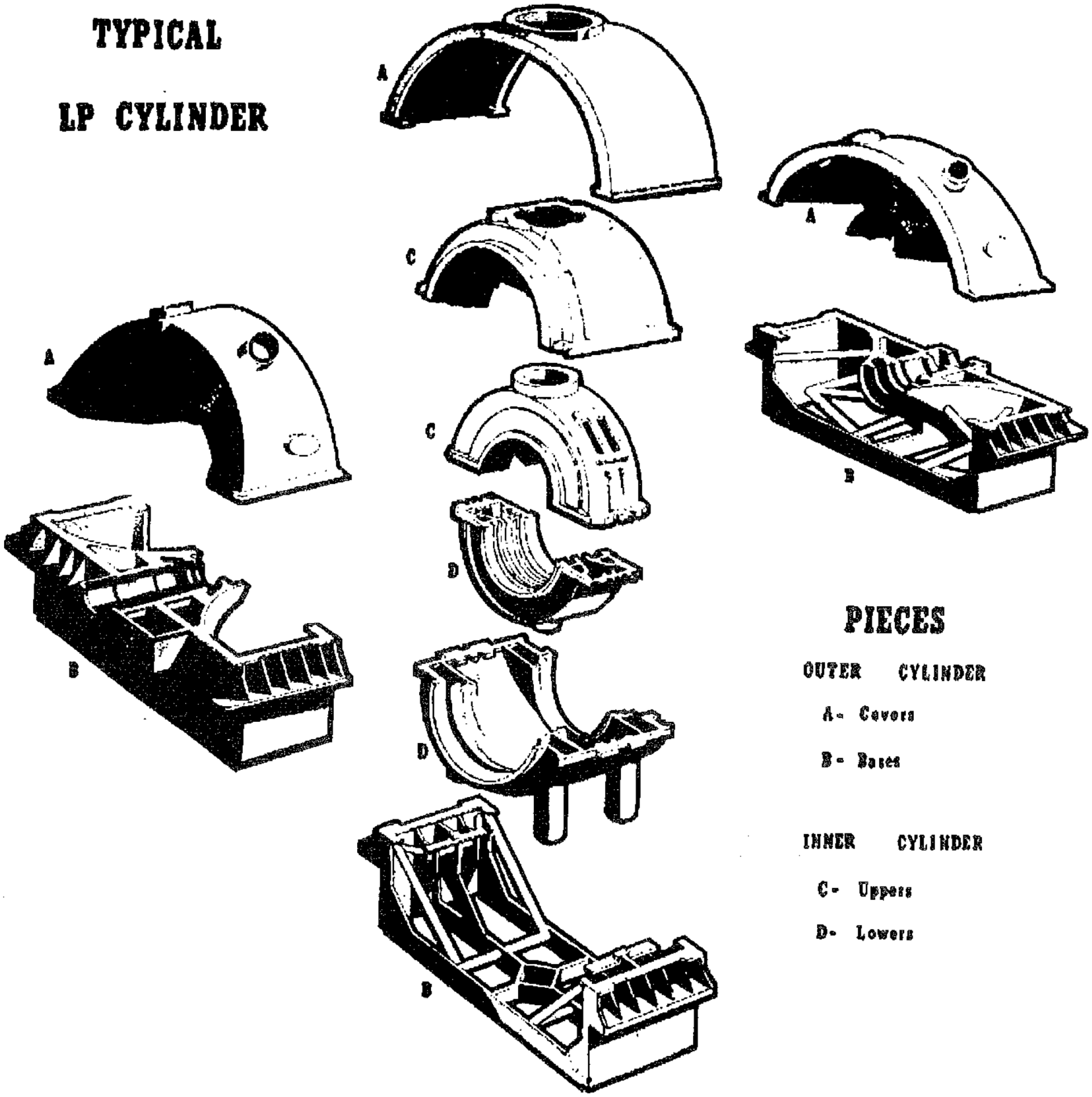
TYPICAL LOW-PRESSURE ELEMENT
1800-RPM DOUBLE-FLOW DESIGN
FIGURE 14.1.13-3



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TYPICAL LP CYLINDER
PIECES
FIGURE 14.1.13-4

14.1.14 Accidental Depressurization of the Reactor Coolant System

An accidental depressurization of the Reactor Coolant System (RCS) could occur as a result of an inadvertent opening of a pressurizer relief, pressurizer safety or pressurizer spray valve. The depressurization resulting from the opening of a relief or safety valve is much more rapid than that which would occur from the accidental opening of a pressurizer spray valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, the most severe core conditions resulting from an accidental depressurization of the RCS are those associated with an inadvertent opening of a pressurizer safety valve. It should be noted that a stuck open pressurizer safety valve is not considered to be a Condition II event, an event of moderate frequency, as would be the case with a control system failure. Nonetheless, the results of this analysis are shown to comply with the acceptance criteria for a Condition II event.

Initially, the event results in a rapidly decreasing RCS pressure, which could reach the hot leg saturation pressure without reactor protection system intervention. If saturated conditions were to be reached, the rate of depressurization would be slowed considerably. However, the pressure continues to decrease throughout the event. The power remains essentially constant throughout the initial stages of the transient. C26

Based on this, the event is primarily analyzed to show that the departure from nucleate boiling design basis is not violated. The reactor may be tripped by the following reactor protection system signals:

- Overtemperature ΔT
- Pressurizer low pressure

Method of Analysis

The accidental depressurization of the RCS transient is analyzed by employing the detailed digital computer code RETRAN (Reference 1). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves. The code computes pertinent plant variables including temperatures, pressures, and power levels.

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

a. The accident is analyzed using the Revised Thermal Design Procedure (Reference 2). Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values, consistent with steady-state full-power operation. Reactor coolant minimum measured flow is modeled. Uncertainties in initial conditions are included in the DNBR safety analysis limit as described in WCAP-11397 (Reference 2).

b. A least negative moderator coefficient of reactivity is assumed. This provides a conservatively low amount of reactivity feedback due to changes in moderator temperature.

c. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. The core power peaking factors are held constant at the design values while, in fact, the void formation and resulting core feedback effects would result in considerable flattening of the power distribution. Although this would significantly increase the calculated DNBR, no credit is taken for this effect.

d. At least negative Doppler coefficient of reactivity is assumed, such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback prior to reactor trip. C26

e. The pressurizer safety valve flowrate is assumed to be 120% of the design capacity of the valve.

Normal reactor control systems are not required to function. Although automatic rod withdrawal has been disabled, the event was conservatively analyzed assuming automatic rod withdrawal. Operation of the rod control system attempts to maintain the full power T_{avg}, which delays reactor trip, thereby resulting in a limiting analysis. 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.

Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 14.1.14-1 and 14.1.14-2. Figure 14.1.14-1 illustrates the nuclear power and pressurizer pressure transients following the accident. Nuclear power increases slowly from the initial value until a reactor trip occurs on overtemperature ΔT. The reactor coolant system average temperature and DNBR transients are given in Figure 14.1.14-2. The DNBR decreases initially, but increases rapidly following the reactor trip. The DNBR remains above the limit value throughout the transient. The calculated sequence of events for this accident is listed in Table 14.1.14-1.

In addition, the stuck open pressurizer safety valve followed by a failure of the RPS to shut down the reactor is considered an ATWS event. See Section 14.1.15 for applicable discussion.

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Conclusions

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

References

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

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

Table 14.1.14-1

Sequence of Events for

Accidental Depressurization

of the Reactor Coolant System

Event	Time (seconds)*
Inadvertent opening of one pressurizer safety valve	100.0
OTΔT reactor trip setpoint reached	137.7
Rod motion begins	139.7
Minimum DNBR occurs	140.5

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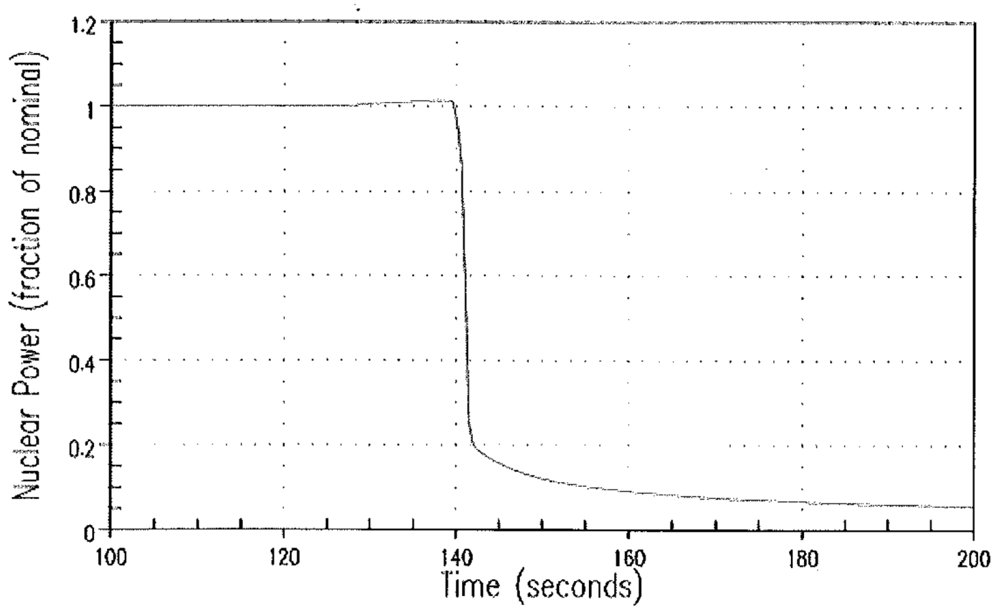
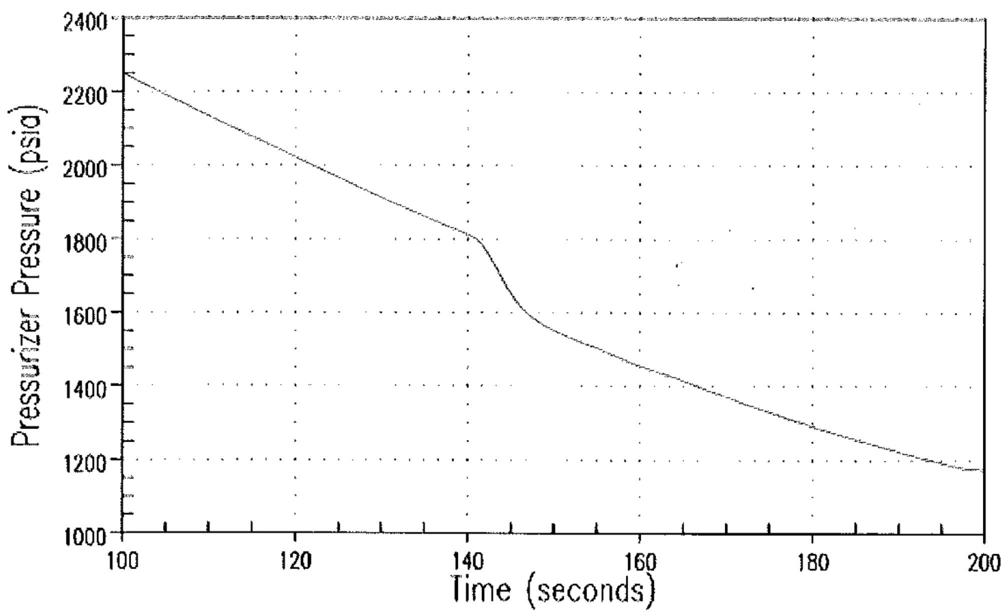
* Times include a 100.0 second steady-state prior to event initiation.





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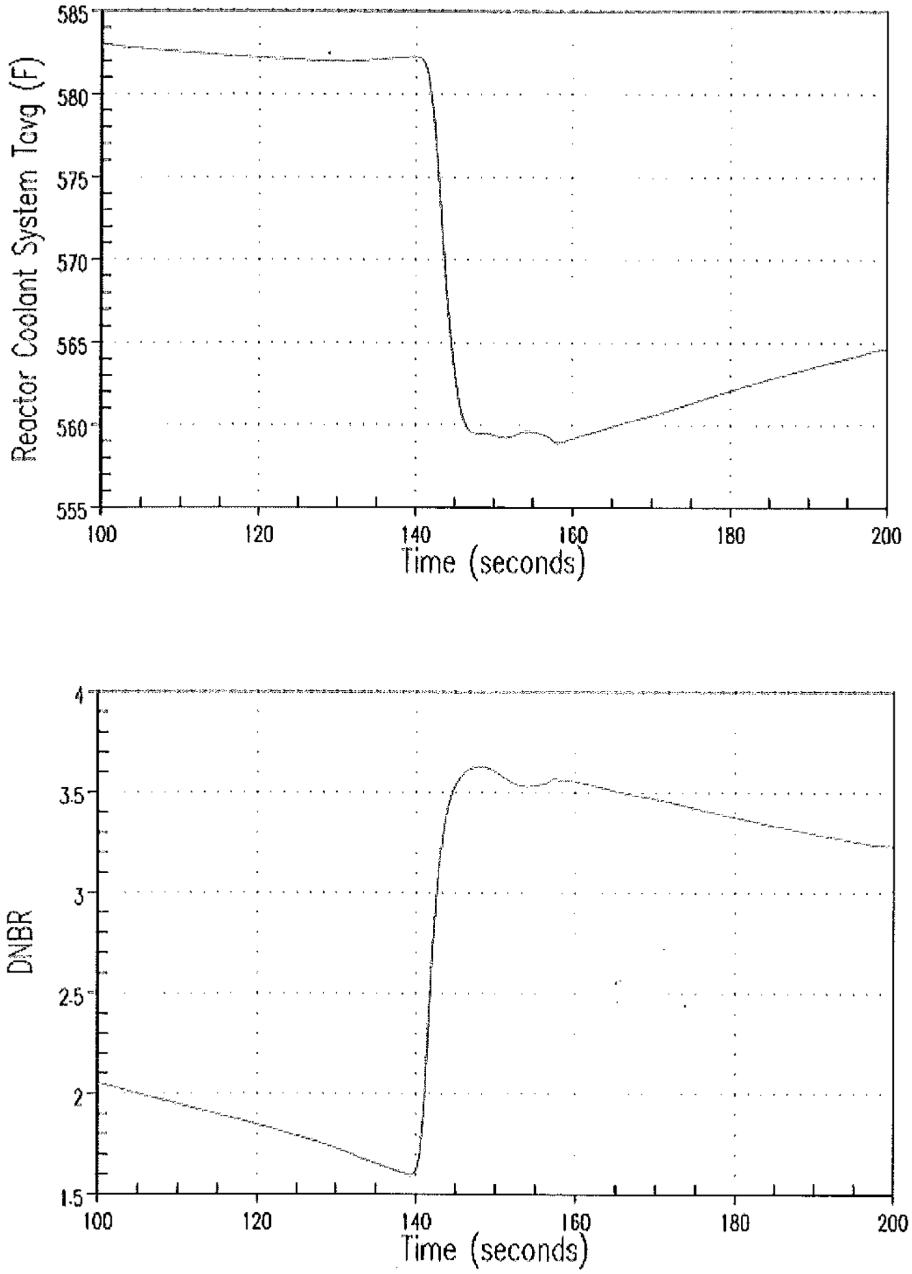
ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM - NUCLEAR POWER AND PRESSURIZER PRESSURE VERSUS TIME

FIGURE 14.1.14-1



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ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM - REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE AND DNBR VERSUS TIME

FIGURE 14.1.14-2

14.1.15 ANTICIPATED TRANSIENTS WITHOUT SCRAM

14.1.15.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

An Anticipated Transient Without Scram (ATWS) event is defined as an anticipated operational occurrence (AOO) followed by the failure of the reactor trip portion of the protection system specified in GDC-20. For Westinghouse pressurized water reactors (PWR), 10 CFR 50.62(c)(1) requires that each plant must have equipment that is diverse from the reactor trip system to automatically initiate the Auxiliary Feedwater (AFW) System and initiate a turbine trip under conditions indicative of an ATWS event. This equipment must perform its function in a reliable manner and be independent of the existing reactor trip system.

As described in UFSAR Section 7.2.2, in order to provide additional assurance of tripping the reactor trip breakers per NRC Generic Letter (GL) 83-28 Item 4.3 (Reference 1), the reliability of the Reactor Protection System (RPS) is enhanced by a design change to also use the shunt trip attachments to open the reactor trip breakers automatically. The automatic shunt trip function is considered safety related. The breaker closing circuit is electrically separated from the tripping circuit and is considered non-safety related. C26

As described in UFSAR Section 7.2.4, 10 CFR 50.62(c)(1) (Reference 2) requires that all PWRs have backup equipment from the sensor output to the final actuation device, that is diverse from the reactor trip system, to automatically initiate the AFW System and turbine trip under conditions indicative of an ATWS event. An ATWS event is an operational transient such as a loss of normal feedwater, loss of load/turbine trip, loss of offsite power, accidental Reactor Coolant System (RCS) depressurization, uncontrolled rod bank withdrawal at power, followed by a failure of the RPS to shut down the reactor. This requirement has been satisfied by the addition of the ATWS Mitigating System Actuating Circuitry (AMSAC) which, in addition to the 10 CFR 50.62 requirements for automatically initiating a turbine trip and the AFW System, initiates a reactor trip by opening the Control Rod Motor-Generator (MG) Set output breakers. AMSAC serves as a non-safety-related backup protective system to RPS by preventing overpressurization of the RCS, providing for conservation of steam generator inventory, and assuring insertion of the control rods following an ATWS event.

The “ATWS Rule” 10 CFR 50.62(c)(1) and AMSAC design are based on WCAP-10858, “AMSAC Generic Design Package.” (Reference 3) The basis for this rule and the AMSAC design are supported by Westinghouse generic analyses documented in NS-TMA-2182 (Reference 6). These analyses were performed based on guidelines published in NUREG-0460 (Reference 7). NS-TMA-2182 describes the methodology employed in the analysis using a multi-loop version of the LOFTRAN code and provides reference analyses for two-loop, three-loop, and four-loop plant designs with several different steam generator models available in plants at that time. NS-TMA-2182 also references WCAP-8330 (Reference 8) and subsequent related documents, which formed the initial Westinghouse submittal to the NRC for ATWS, and which were based on the guidelines set forth in WASH-1270 (Reference 9). NS-TMA-2182 addressed five Condition II ATWS events including loss of load/turbine trip, loss of normal feedwater, loss of offsite power, stuck open pressurizer safety valve, and uncontrolled rod withdrawal at power. Inputs varied with the reference plant designs noting that 3-loop reference plants used AFW full flow of 1400 gpm and a 40% steam dump capacity while the analyses for all plants assumed a conservative delay of 60 seconds in AFW flow following AMSAC initiation. This delay generically included time for emergency diesel generator (EDG) start and load sequencing (for plants with electric motor-driven AFW pumps) and pump acceleration. The ATWS analysis assumed full AFW flow was reached 36 seconds after the actuation signal occurs. The time to purge the feedwater piping of hot water was also considered in the analysis as an additional delay in the delivery of cool AFW flow to the steam generators. C26

The NRC approved the Westinghouse Owners Group (WOG) generic design modification with its issuance of Safety Evaluation Report (SER) on August 10, 1983 (Reference 4). The NRC issued a plant specific SER for GL 83-28 Item 4.3 on December 3, 1984 (Reference 5) and required AMSAC design modifications were installed at Turkey Point in July 1985 for Unit 3 and in June 1986 for Unit 4.

For operation at EPU conditions, the two most limiting RCS overpressure ATWS transients from the Westinghouse generic ATWS analyses, Loss of Normal Feedwater (LONF) and Loss of Load (LOL), were analyzed as discussed below.

14.1.15.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

Method of Analysis

The Loss of Normal Feedwater (LONF) and Loss of Load (LOL) ATWS events were analyzed for the EPU. The following analysis assumptions were used:

- To comply with the ATWS rule, PTN installed an AMSAC system, which in addition to the requirements of 10 CFR 50.62 to automatically initiate the AFW system and trip the turbine, initiates a reactor trip by tripping the MG Set output breakers. The ATWS analyses performed for the EPU conservatively did not credit a reactor trip on the AMSAC actuation output signal to trip the control rod MG set output breakers.
- The nominal and initial conditions reflect PTN with an analyzed NSSS power of 2652 MWt.
- Consistent with the analysis basis for the ATWS Rule (NS-TMA-2182):
  - Thermal design flow (TDF) is assumed.
  - No uncertainties are applied to the initial power, RCS average temperature or RCS pressure (Reference 10). C26
  - Zero percent steam generator tube plugging (SGTP) is assumed. Zero percent SGTP is more limiting (i.e., results in a higher peak RCS pressure based on generic sensitivities) for ATWS events.
  - Control rod insertion was not assumed.
  - A 27 second AMSAC response time to trip the turbine and actuate AFW was assumed. This delay time is added to the time at which the SG water mass reaches a mass equivalent to the water level at the AMSAC low SG water level setpoint of 8.65%. An additional 95 second delay was added to the actuation of AFW to account for the time required to get the AFW pumps up to speed, sensor delays, and logic delays.
- A plant-specific EPU steam dump (turbine bypass) capacity of 32.6% was assumed.
- A plant-specific EPU AFW flow of 780 gpm was assumed.
- The ATWS evaluation for the EPU assumed a PTN-specific MTC of -8 pcm/°F that bounds 95 percent of the cycle. This value is consistent with that assumed in generic ATWS analyses. The ATWS MTC limit is confirmed each cycle as part of the reload process.
- The steam generator parameters and heat transfer characteristics of the Model 44F steam generator were modeled.

The bases for this rule and the AMSAC design are supported by Westinghouse analyses documented in NS-TMA-2182. To remain consistent with the basis of the ATWS Rule and the supporting analyses documented in NS-TMA-2182, the peak RCS pressure reached in the PTN EPU ATWS evaluation should not exceed the ASME B&PV Code, Service Level C stress limit criterion of 3200 psig. This value corresponds to the maximum allowable pressure for the weakest component in the reactor pressure vessel (RPV) (the nozzle safe end). The LONF and LOL ATWS events analyzed for the EPU were performed at a NSSS power level of 2652 MWt with Model 44F steam generators. The LOFTRAN computer code (Reference 11) was used to perform the ATWS analyses for the EPU consistent with the analysis basis for the ATWS Rule. Note that the LONF and LOL events without ATWS are presented in UFSAR Sections 14.1.11 and 14.1.10, respectively, while the other events originally considered by Westinghouse (the loss of AC power, stuck open pressurizer safety valve, and uncontrolled rod withdrawal at power) are presented in UFSAR Sections 14.1.12, 14.1.14, and 14.1.2, respectively.

<u>Results</u>

The results of the ATWS analyses at an NSSS power of 2652 MWt, provided in Table 14.1.15-1, show that the peak RCS pressure obtained in the LONF and LOL ATWS events, 3174.5 psia and 2960.2 psia, respectively, did not exceed the B&PV Code, Service Level C stress limit criterion of 3215 psia (3200 psig). As such, the analytical basis for the ATWS Rule continues to be met for operation at an NSSS power of 2652 MWt. The updated EPU analyses place no restrictions on the existing AMSAC setpoint. C26

Time sequence of events are provided in Tables 14.1.15-2 and 14.1.15-3 for the EPU LONF and LOL ATWS events, respectively. Transient plots for the EPU LONF and LOL ATWS events are provided in Figures 14.1.15-1 through 14.1.15-8.

14.1.15.3 CONCLUSIONS

It has been demonstrated that the AMSAC meets the requirements of 10 CFR 50.62 under EPU conditions. It has been shown that the plant, as a Westinghouse designed PWR, is not required by 10 CFR 50.62 to have a diverse scram system. It has also been demonstrated that the peak primary system pressure following an ATWS event remains below the 3200 psig acceptance limit.

14.1.15.4 REFERENCES

1. NRC Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983.
2. 10 CFR 50.62, "Requirements for Reduction of Risk from ATWS Events for Light Water-Cooled Nuclear Power Plants."
3. WCAP-10858, "AMSAC Generic Design Modification," 1984.
4. NRC Letter to Westinghouse Owners Group (WOG) on WOG Generic Design Modification, August 10, 1983.
5. NRC Letter to FPL, S. A. Varga to J. W. Williams, Reactor Trip System Reliability, Generic Letter (GL) 83-28 Item 4.3, "Automatic Actuation of the Shunt Trip Attachment for Westinghouse Plants," December 3, 1984.
6. NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979. C26
7. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," April 1978.
8. WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis," August 1974.
9. NRC Report WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water Cooled Power Reactors," September 1973.
10. Letter from Roger J. Mattson, Director, Division of Systems Safety Office of Nuclear Reactor Regulation to Thomas M. Anderson, Nuclear Safety Department of Westinghouse Electric Company, February 15, 1979.
11. WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

Table 14.1.15-1
EPU Results

Event	Peak RCS Pressure, psia
Loss of Normal Feedwater	3174.5
Loss of Load	2960.2

Table 14.1.15-2
Time Sequence of Events
Loss of Normal Feedwater ATWS

Event	Time (sec)
FW flow terminated	0.0
AMSAC low SG water level setpoint reached	32.4
Turbine trip occurs	59.4
Peak RCS Pressure (3174.5 psia) reached [versus RCS pressure limit of 3215 psia]	89.2
Full AFW initiated	154.4

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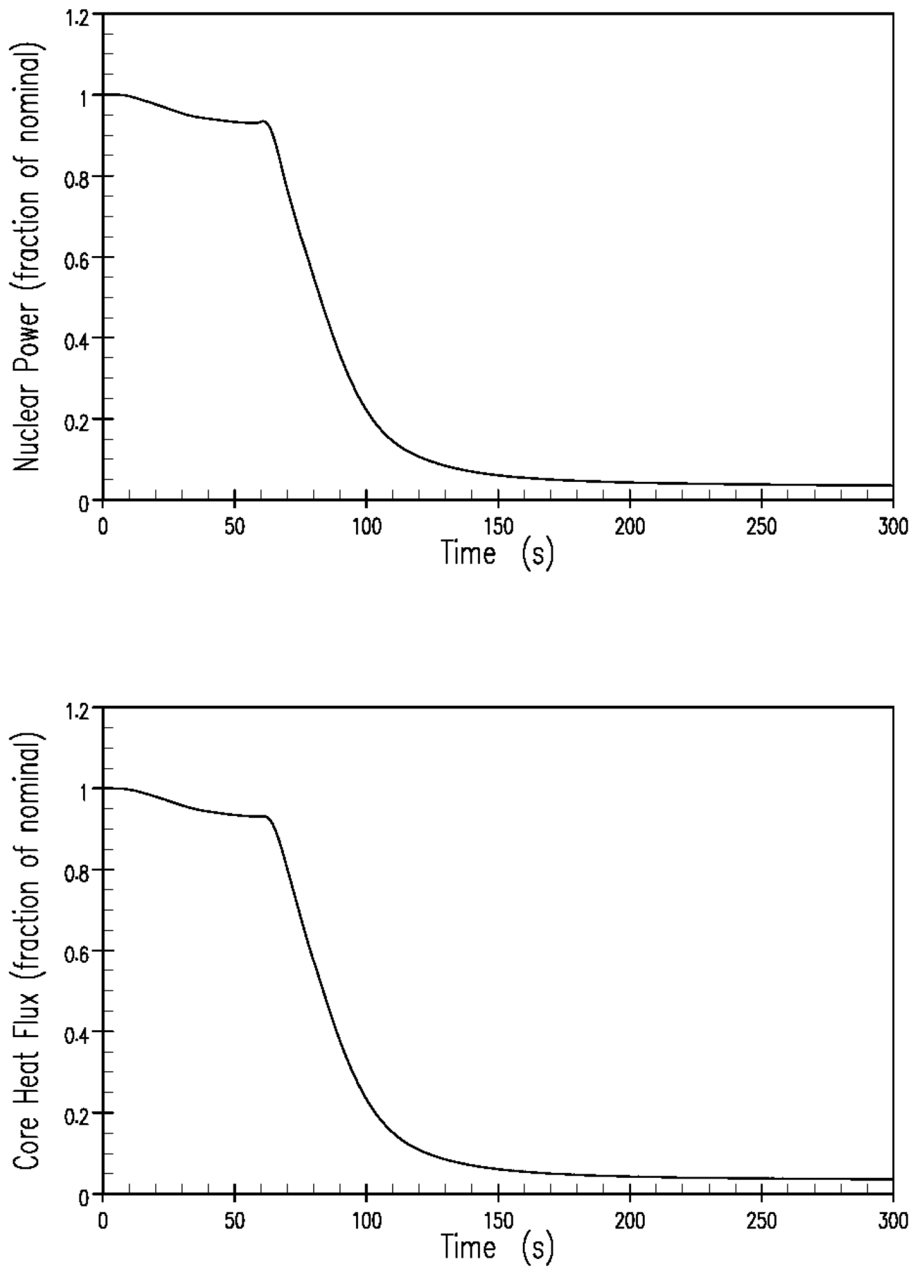
Table 14.1.15-3
Time Sequence of Events
Loss of Load ATWS

Event	Time (sec)
FW flow terminated	0.0
Turbine trip occurs	1.0
AMSAC low SG water level setpoint reached	49.7
Peak RCS Pressure (2960.2 psia) reached [versus RCS pressure limit of 3215 psia]	112.6
Full AFW initiated	171.7



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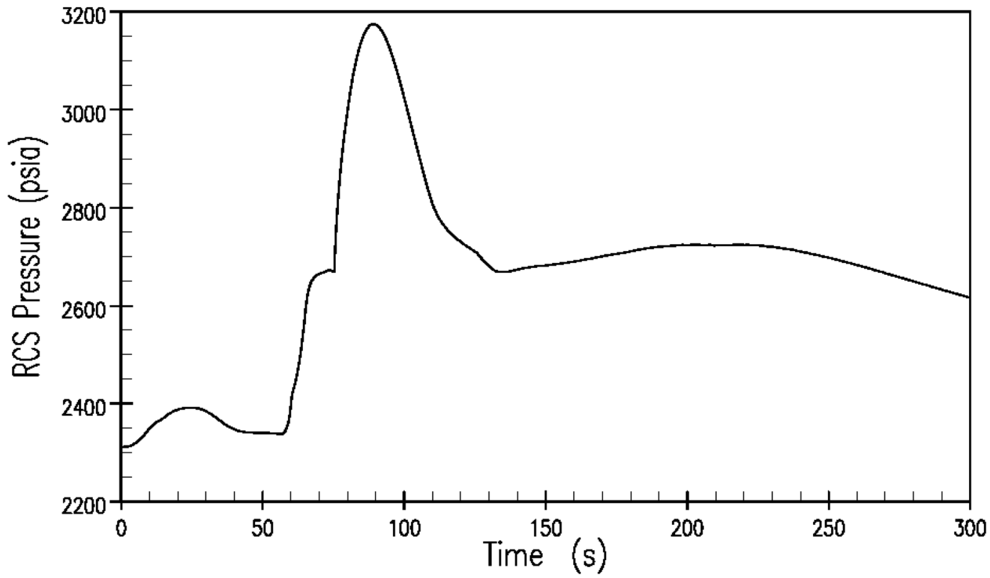
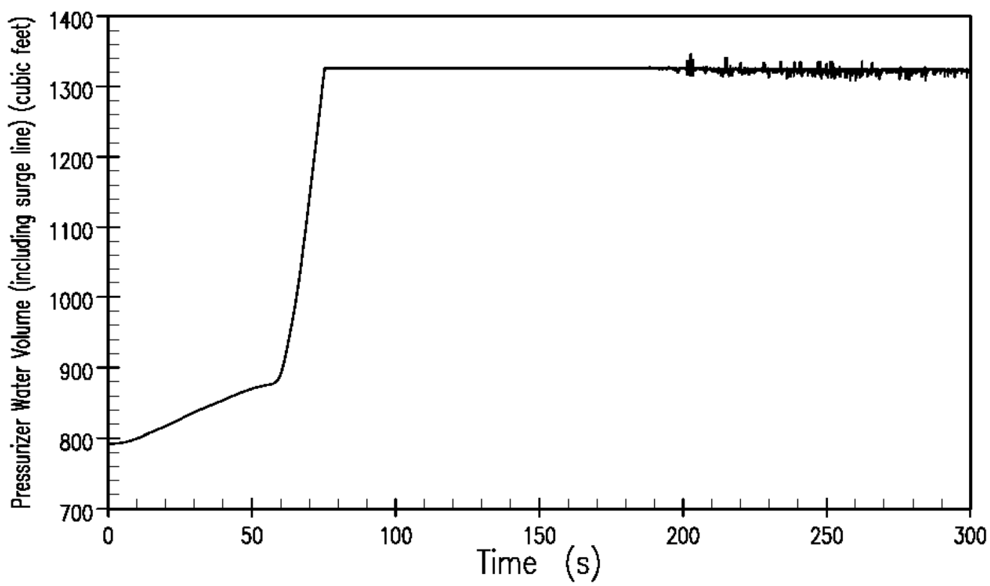
TURKEY POINT PLANT UNITS 3 & 4

NUCLEAR POWER AND CORE HEAT FLUX VERSUS TIME FOR LONF ATWS

FIGURE 14.1.15-1





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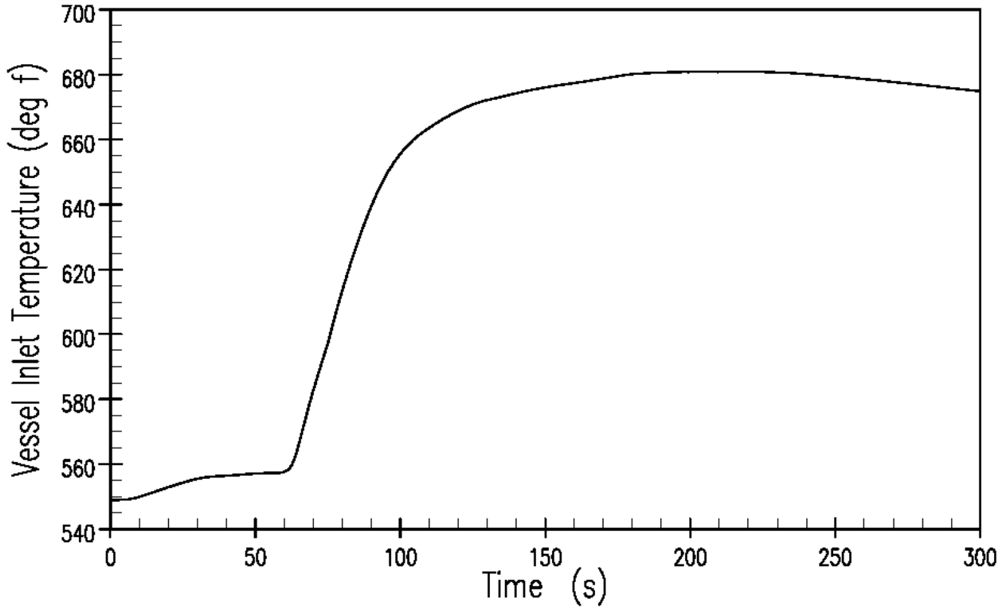
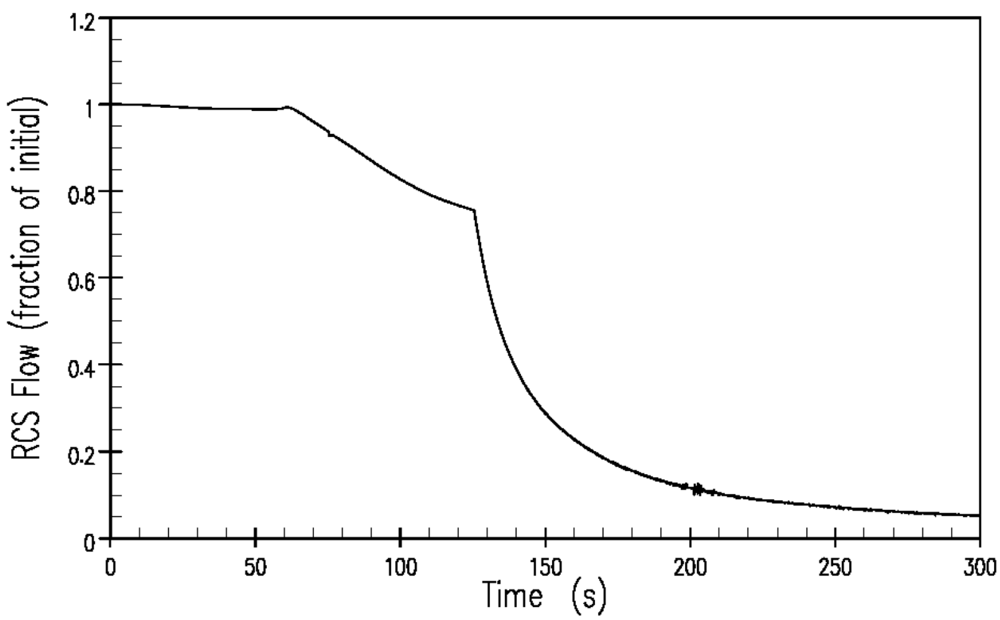
RCS PRESSURE AND PRESSURIZER WATER VOLUME VERSUS TIME FOR LONF ATWS

FIGURE 14.1.15-2





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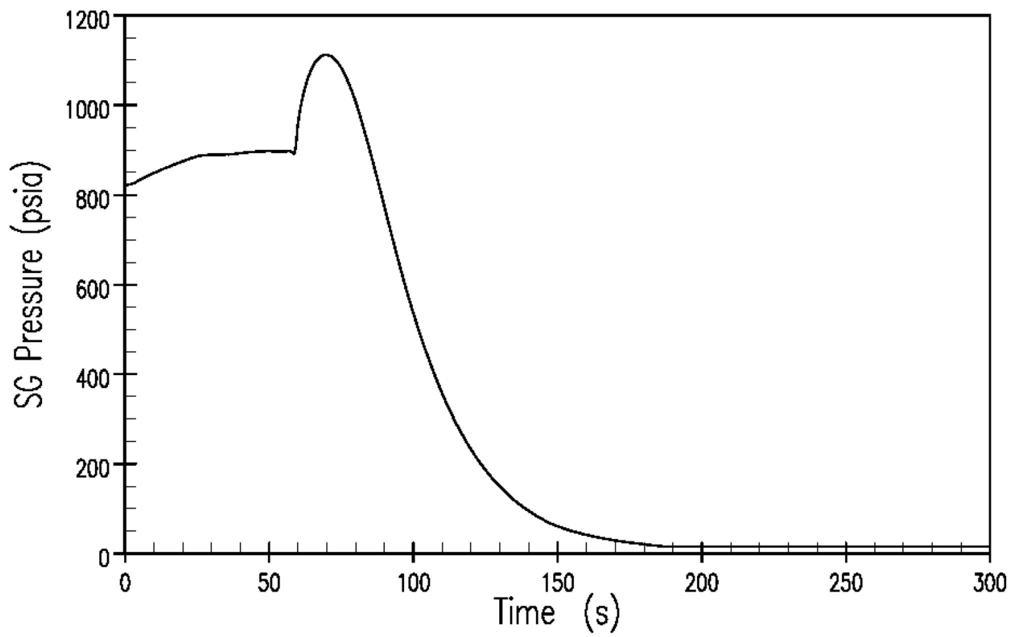
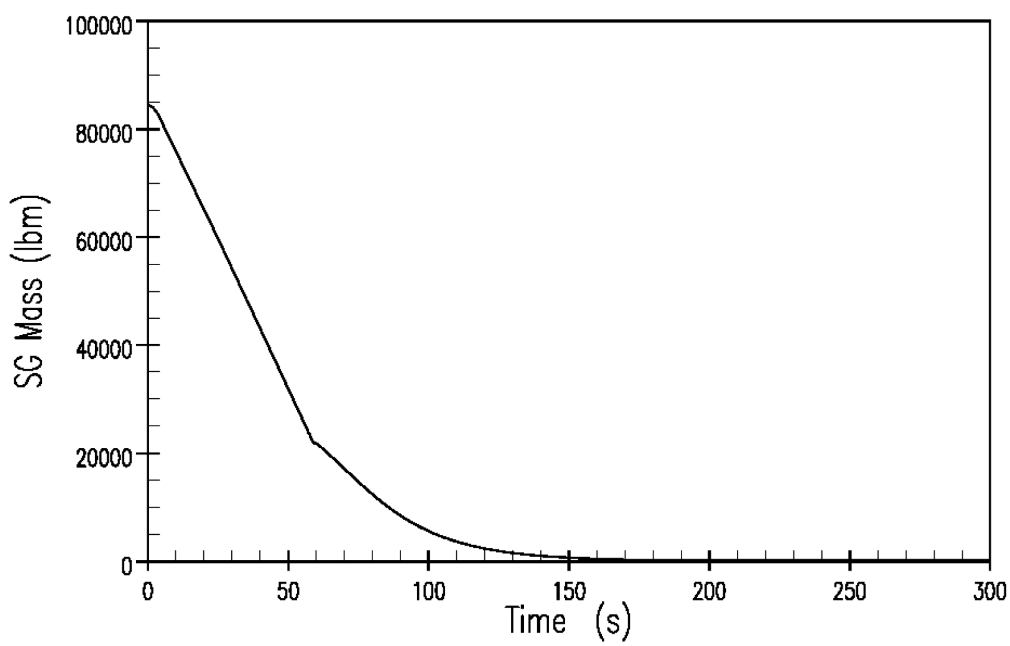
VESSEL INLET TEMPERATURE AND RCS FLOW VERSUS TIME FOR LONF ATWS

FIGURE 14.1.15-3





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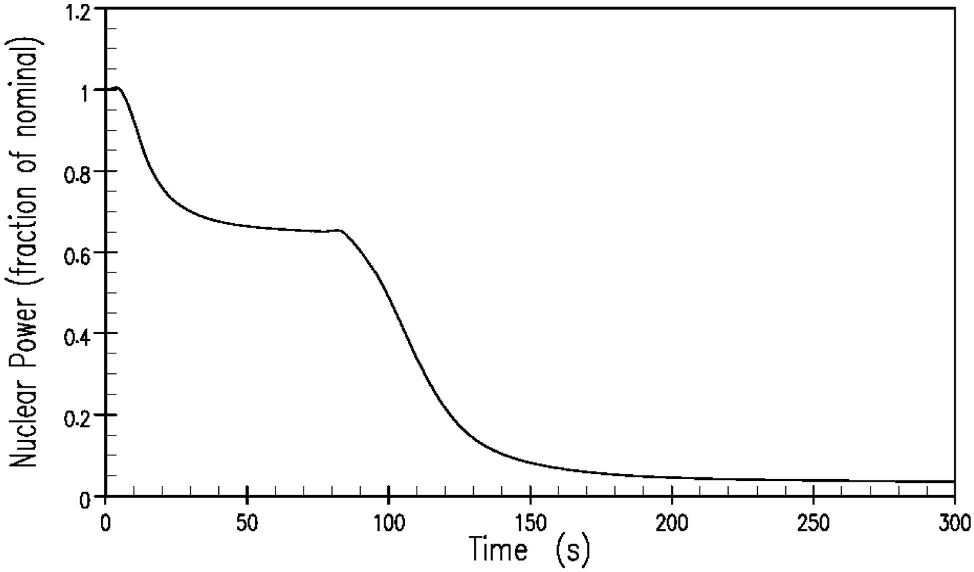
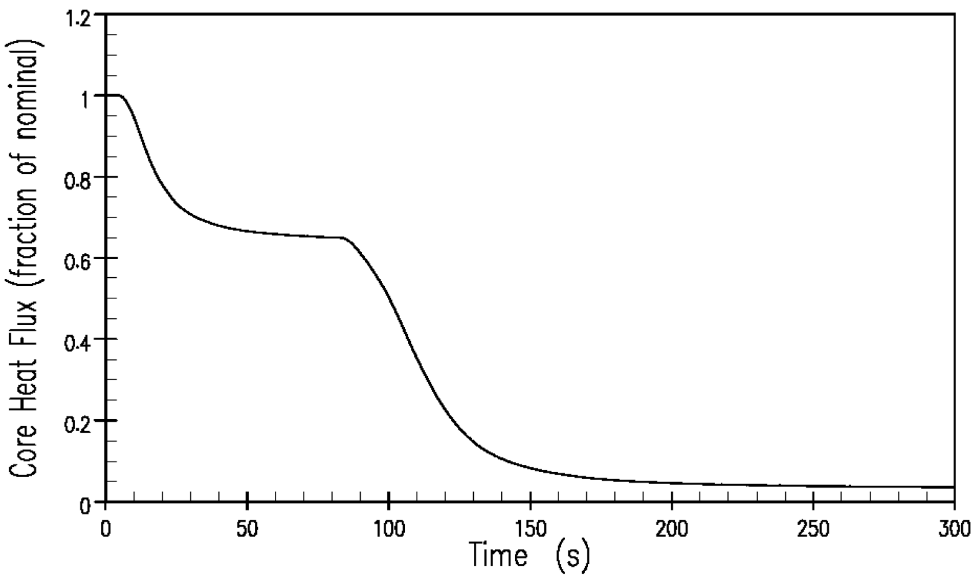
SG PRESSURE AND SG MASS

VERSUS TIME FOR LONF ATWS

FIGURE 14.1.15-4





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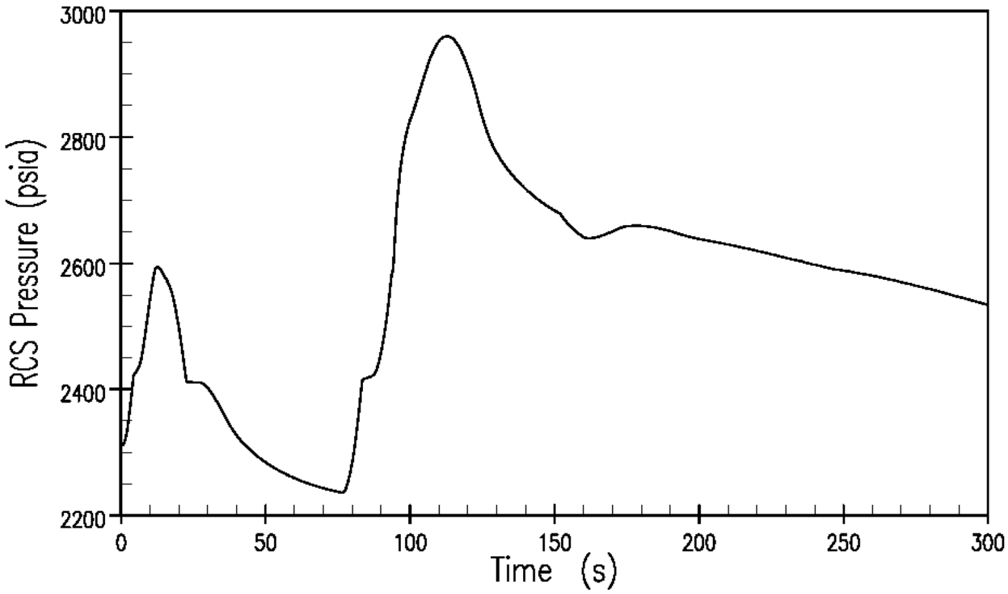
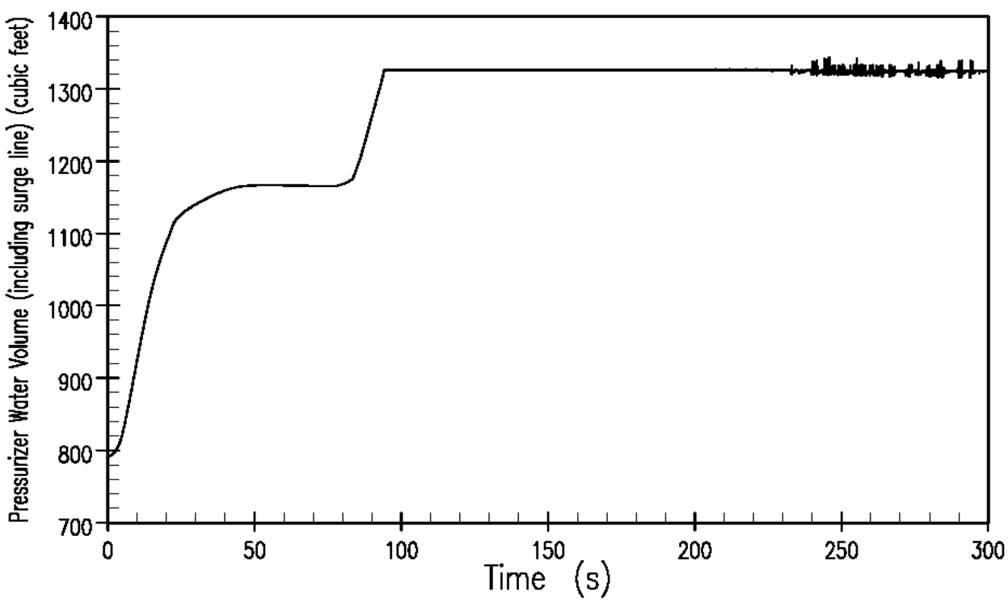
NUCLEAR POWER AND CORE HEAT FLUX VERSUS TIME FOR LOL ATWS

FIGURE 14.1.15-5





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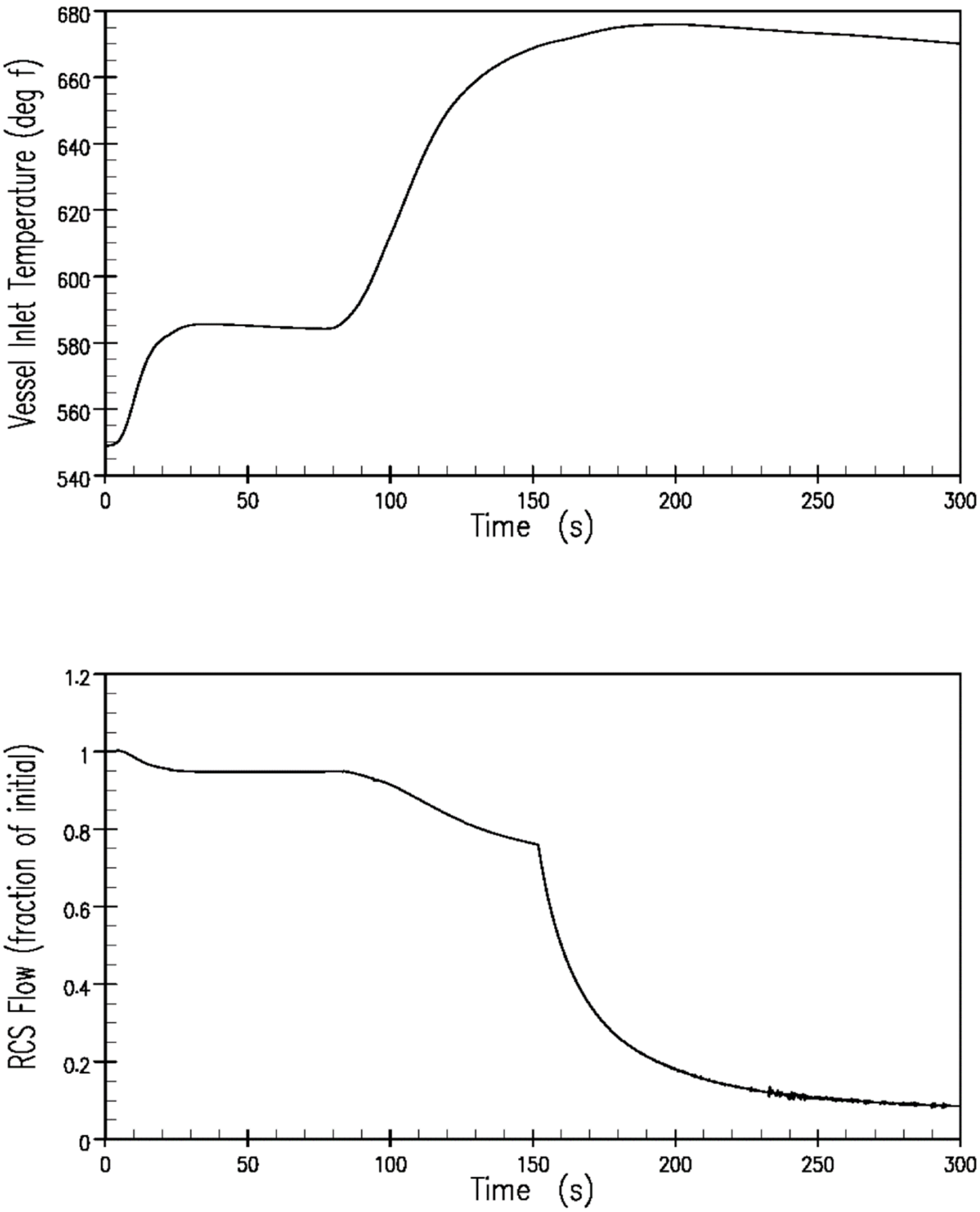
TURKEY POINT PLANT UNITS 3 & 4

RCS PRESSURE AND PRESSURIZER WATER VOLUME VERSUS TIME FOR LOL ATWS

FIGURE 14.1.15-6



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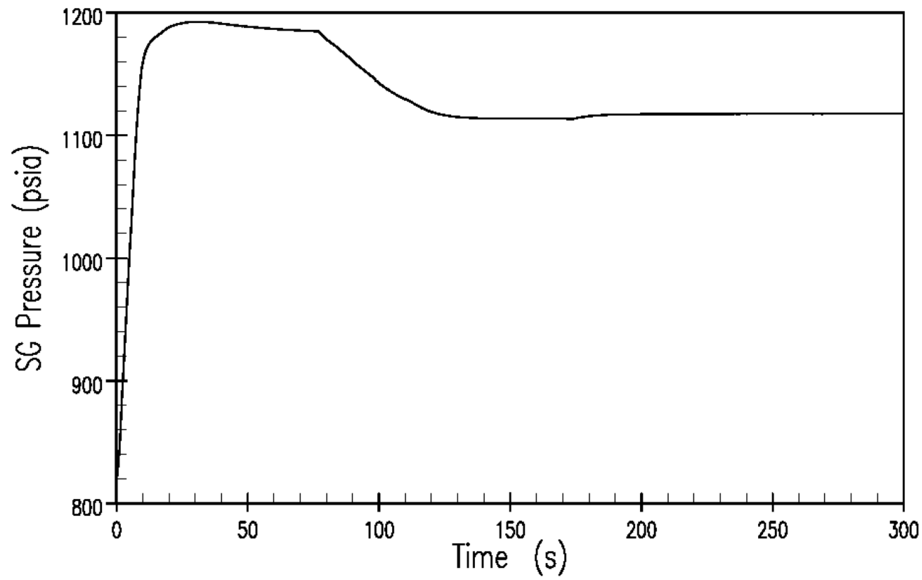
VESSEL INLET TEMPERATURE AND RCS FLOW VERSUS TIME FOR LOL ATWS

FIGURE 14.1.15-7



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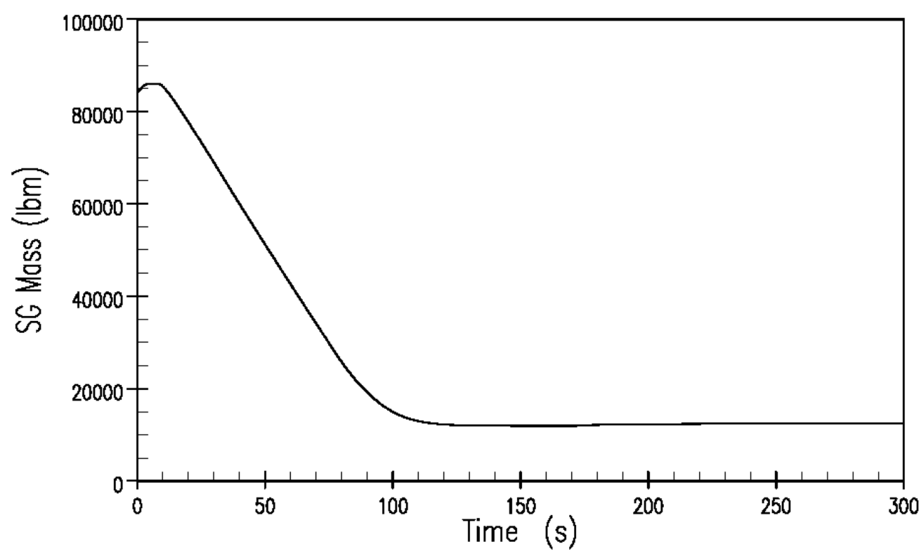

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TURKEY POINT PLANT UNITS 3 & 4

SG PRESSURE AND SG MASS

VERSUS TIME FOR LOL ATWS

FIGURE 14.1.15-8

14.2 STANDBY SAFETY FEATURES ANALYSIS

Adequate provisions have been included in the design of the unit and its standby engineered safety features to limit potential radiation exposure of the public to well below the limits of 10CFR50.67 for situations which have a very low probability of occurrence, but which could conceivably involve uncontrolled releases of radioactive materials to the environment. The postulated scenarios and their ANS-51.1/N18.2-1973 (Reference 23) condition classifications (II, III, IV) which have been considered are: C26 C26

a) Fuel Handling Accidents (Condition IV)
b) Accidental Release of Waste Liquid (Condition III)
c) Accidental Release of Waste Gases (Condition III)
d) Rupture of a Steam Generator Tube (Condition III)
e) Rupture of a Steam Pipe (Condition IV), Stuck Open Safety Valve (Condition II)
f) Rupture of a Control Rod Drive Mechanism Housing - Rod Cluster Control Assembly (RCCA) Ejection (Condition IV)
g) Feedwater System Pipe Break (Condition IV)

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14.2.1 FUEL HANDLING ACCIDENTS

14.2.1.1 Event Description

The following fuel handling accidents are evaluated to ensure that no hazards are created:

a) A fuel assembly is dropped in containment.

b) A spent fuel cask is dropped in the passage between the spent fuel pits of Units 3 & 4 while transferring a fuel element between the spent fuel pits. The consideration of a cask drop accident is historical and is retained as discussed in Section 14.2.1.3.

Causes and Assumptions

The possibility of a fuel handling incident is remote because of the administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. Also, before any refueling operations begin, verification of complete rod cluster control assembly insertion is obtained by tripping each rod individually to obtain indication of rod drop and disengagement from the control rod drive mechanisms. The boron concentration in the coolant is raised to the refueling concentration and verified by sampling. The refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical with all rod cluster assemblies withdrawn. The refueling cavity is filled with water meeting the same boric acid specifications. As the vessel head is raised, a visual check is made to verify that the drive shafts are free in the mechanism housing.

After the vessel head is removed, the rod cluster control drive shafts are removed from their respective assemblies using the auxiliary hoist on the manipulator crane and the drive shaft unlatching tool. A spring scale is used to indicate that the drive shaft is free of the control cluster as the lifting force is applied.

The fuel handling manipulators and hoists are designed so that fuel can only be raised up to positions which provide adequate shield water depth for the safety of operating personnel. This safety feature applies to handling facilities in both the containment and in the spent fuel pit area. In the spent fuel pit, the design of storage racks and manipulation facilities is such that:

- Fuel at rest is positioned by positive restraints in a safe, subcritical, geometrical array, with no credit for boric acid in the water.

- Only one fuel assembly can be handled at a time.

- Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

- Administrative control will be used to prevent the handling of heavy objects such as a spent fuel shipping container, above the fuel racks, until the fuel in the spent fuel pit has decayed for a minimum of 1525 hours.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the refueling cavity or spent fuel pit.

Two Nuclear Instrumentation System source range channels are continuously in operation and provide warning of any approach to criticality during refueling operations. This instrumentation provides a continuous audible signal in the containment, and would annunciate a local horn and a bell and light in the control room if the count rate increased above a preset low level.

Refueling boron concentration is sufficient to maintain the clean, cold, fully loaded core subcritical by at least 5 per cent with all rod cluster control assemblies inserted. At this boron concentration the core would also be subcritical with all control rods withdrawn. The refueling cavity is filled with water meeting the same boric acid specification.

All these safety features make the probability of a fuel handling incident very low. Nevertheless, it is possible that a fuel assembly could be dropped during the handling operations. Therefore, this incident is analyzed both from the standpoint of radiation exposure and accidental criticality.

Special precautions are taken in all fuel handling operations to minimize the possibility of damage to fuel assemblies during transport to and from the spent fuel pit and during installation in the reactor. All irradiated fuel handling operations are conducted under water. The handling tools used in the fuel handling operations are conservatively designed and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies are spaced in a pattern which prevents any possibility of a criticality accident. When the cask area rack is installed, administrative controls are required and put in place to prevent heavy loads from being carried over the cask area rack while fuel assemblies are stored in the rack unless loads are being handled by complete single-failure-proof handling system. In addition, fuel handling equipment design is such that only one fuel assembly can be handled at a given time.

The motions of the cranes which move the fuel assemblies are limited to a low maximum speed. Caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structures in the containment or fuel storage building.

The fuel handling equipment suspends the fuel assembly in the vertical position during fuel movements, except when the fuel is moved through the transport tube.

The design of the fuel assembly is such that the fuel rods are restrained by grid clips which provide a total restraining force of approximately 80 pounds on each fuel rod (Reference 2). If the fuel rods are in contact with the bottom plate of the fuel assembly, any force transmitted to the fuel rods is limited due to the restraining force of the grid clips. The force transmitted to the fuel rods during fuel handling is not of a magnitude great enough to breach the fuel rod cladding. If the fuel rods are not in contact with the bottom plate of the assembly, the rods would have to slide against the 80 pound friction force. This would have the effect of absorbing a shock and thus limit the force on the individual fuel rods.

After the reactor is shut down, the fuel rods contract during the subsequent cooldown and would not be in contact with the bottom plate of the assembly.

Considerable deformation would have to occur before the rod would make contact with the top plate and apply any appreciable load on the fuel rod. Based on the above, it is felt that it is unlikely that any damage would occur to the individual fuel rods during handling. If one assembly is lowered on top of another, no damage to the fuel rods would occur that would breach the integrity of the cladding.

If during handling the fuel assembly strikes against a flat surface, the loads would be distributed across the fuel assemblies and grid clips and essentially no damage would be expected in any fuel rods.

If the fuel assembly were to strike a sharp object, it is possible that the sharp object might damage the fuel rods with which it comes in contact but breaching of the cladding is not expected. It is on this basis that the assumption of the failure of an entire row of fuel rods (15) is a conservative upper limit.

Analyses have been made assuming the extremely remote situations where a fuel assembly is dropped and strikes a flat surface, where one assembly is dropped on another, and where one assembly strikes a sharp object. The analysis of a fuel assembly assumed to be dropped and strikes a flat surface considered the stresses the fuel cladding was subjected to and any possible buckling of the fuel rods between the grid clip supports. The results show that the buckling load at the bottom section of the fuel rod, which would receive the highest loading, is below the critical buckling load and the stresses were relatively low and below the yield stress. For the case where one assembly is dropped on top of another fuel assembly, the loads will be transmitted through the end plates and the RCCA guide tubes of the struck assembly before any of the loads reach the fuel rods.

The end plates and guide thimbles absorb a large portion of the kinetic energy as a result of bending in the lower plate of the falling assembly. Also, energy is absorbed in the struck assembly top end plate before any load can be transmitted to the fuel rods. The results of this analysis indicated that the buckling load on the fuel rods was below the critical buckling loads and the stresses in the cladding were relatively low and below yield.

The refueling operation experience that has been obtained with Westinghouse reactors has verified the fact that no fuel cladding integrity failures are expected to occur during any fuel handling operations.

Although rupture of one complete outer row of fuel rods is considered to be a conservative assumption, the reanalysis of the offsite radiological consequences of a dropped fuel assembly using selected implementation of the Alternative Source Term methodology (Section 14.2.1.2) assumed a case in which all the fuel rods in a single assembly are damaged. C26

14.2.1.2 RADIOLOGICAL DOSE EVALUATION

The fuel handling accident for Turkey Point assumes the failure of one complete fuel assembly; therefore, the fuel handling accident source term is based on a single “bounding” fuel assembly. Per Section 3.1 of Regulatory Guide 1.183, the source term methodology for the Fuel Handling Accident is similar to that used for developing the LOCA source term, except that for DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core , a radial peaking factor of 1.65 is applied in determining the inventory of the damaged rods. C26

The LOCA source term is based on the activity of 157 fuel assemblies and the radial peaking factor is 1.65. Thus, based on the methodology specified in Regulatory Guide 1.183, the fuel handling accident source term is derived by applying a factor of 1.65/157 to the LOCA whole core source term given in Table 14.3.5-7.

For the Fuel Handling and Spent Fuel Cask Drop events, the gap fractions specified in Regulatory Guide 1.183, Position 3.2, Table 3, “Non-LOCA Fraction of Fission Product Inventory in Gap”, are modified to account for high burnup fuel. Footnote 11 on Table 3 establishes burnup limits for the applicability of the gap inventory for non-LOCA events. Consideration is given to fuel with a current burnup greater than 54,000 MWD/MTU, which may have exceeded the 6.3 kw/ft linear heat generation rate during a previous operating cycle. This is done using the guidance of NUREG/CR-5009, which endorses the gap release fractions for fuel handling events outlined in Reg. Guide 1.25, with some modification for higher burnups. Although only a few rods may have exceeded the burnup limits of Table 3 of Reg. Guide 1.183, these values are conservatively applied to the entire fuel assembly.

The FHA dose consequence analysis is consistent with the guidance provided in Regulatory Guide 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 per Section 14.2.1.2.
2. Regulatory Position 1.2 – The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of Regulatory Guide 1.183. The gap activity available for release is modified from that specified by Table 3 of Regulatory Guide 1.183 to account for high burnup fuel. This activity is assumed to be released from the fuel assembly instantaneously. C26
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, a decontamination factor of 285 is applied to the elemental iodine and a decontamination factor of 1 is applied to the organic iodine. As a result, the breakdown of the iodine species above the surface of the water is 57% elemental and 43% organic. Guidance for the use of 285 for the elemental iodine decontamination factor is provided in Reference 22, USNRC Regulatory Issue Summary 2006-04, “Experience with Implementation of Alternate Source Terms.”
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 to the environment over a 2-hour period.
7. Regulatory Position 4.2 – No credit is taken for filtration of the release.

8. Regulatory Position 4.3 – No credit is taken for dilution of the release.

9. Regulatory Position 5.1 – The containment equipment 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 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.

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 flow distribution during this mode is 1000 cfm or unfiltered fresh air plus 100 cfm of unfiltered inleakage. C26

- For the FHA in containment, Control Room isolation occurs on high radiation on the containment radiation monitors. A 30 second delay time is assumed for signal processing and damper closure. For the release from the Fuel Handling Building, the Control Room is assumed to be manually isolated by the operators 30 minutes after the beginning of the event. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 100 cfm of unfiltered in leakage, and 375 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 and 95% for elemental and organic iodine.

The atmospheric dispersion factors (X/Qs) used 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 Appendix 2F. The Control Room atmospheric dispersion factors applied to the FHA in containment are based upon a release from the most limiting containment equipment/personnel hatch. For the Fuel Handling Building release, the most limiting X/Qs correspond to a release from the Unit 4 spent fuel pool.

The EAB and LPZ doses are determined using the X/Q factors for the appropriate time intervals. These X/Q factors are provided in Appendix 2E.

The major assumptions and parameters used in the analysis are itemized in Table 14.2.1-2. Since the conservatively selected assumptions and parameters for a FHA inside containment are identical to those for a FHA in the fuel storage building, the offsite doses and onsite doses are the same regardless of the location of the accident. C26

The dose limits for the FHA using the AST methodology (Reference 4) are 6.3 rem TEDE at the EAB and 5 rem TEDE for control room operators. The TEDE doses for both offsite (EAB) and onsite (CR), due to the FHA using AST methodology, are given in Table 14.2.1-3 and are within these acceptance criteria given above.

14.2.1.3 CASK DROP ACCIDENT

NOTE

In 2011, the spent fuel cask handling crane was upgraded to the single-failure-proof criteria of NUREG-0554, Single-Failure-Proof Cranes for Nuclear Power Plants (Reference 20). The NRC approved the use of the upgraded cask crane as single-failure-proof in License Amendments 243 and 239 (Reference 21). For single-failure-proof lifts, the cask crane must utilize a single-failure-proof handling system. Therefore, the handling system must meet the requirements of NUREG-0612 (Reference 6), Section 5.1.6. Single-failure-proof lifts are those deemed reliable enough such that the consequences of a load drop need not be considered. C26

For spent fuel cask handling, the entire lifting system is single-failure-proof. Therefore, Amendments 243 and 239 authorized the deletion of the cask drop accident from the safety analysis. However, the cask drop accident analysis is retained for historical purposes as it was used to support past spent fuel pool re-racking activities, and to support handling the temporary cask area racks, which may have handling system components that do not meet the criteria of NUREG-0612, Section 5.1.6. The cask drop analysis bounds the drop of a cask area rack.

Sections of the cask drop accident evaluation that follow have been revised to reflect the use of the single-failure-proof crane and the deletion of Technical Specifications 3/4.9.7 and 3/4.9.12, which was authorized by License Amendments 243 and 239. The upgraded single-failure-proof cask crane is authorized to handle spent fuel casks that contain multiple fuel assemblies while the cask drop accident considered the use of a single element cask.

The spent fuel transfer cask will not be moved into the spent fuel pit containing two region density racks until all spent fuel in the pit has decayed for a minimum of 1525 hours. Only a single element cask will be used to transfer one fuel assembly at a time. The radiological effects due to fuel damage resulting from a dropped spent fuel transfer cask during transfer of fuel assemblies between the spent fuel pits have been determined to be lower than those from a design basis fuel handling accident. Also, the spent fuel transfer cask drop while on transit between the Units 3 & 4 spent fuel pits will not damage equipment or structures required for the safe shutdown of Units 3 & 4. An evaluation of the cask drop accident is provided below. Evaluation of the cask drop accident for the uprated power level of 2346 MWt was performed. The assumptions used and the resulting offsite doses for this analysis are presented in Tables 14.2.-4 and 14.2.1-5, respectively (Reference 1).

14.2.1.3.1 Cask Handling Considerations

This section provides information relative to the cask handling crane. It is intended to supplement the information provided in FPL letters to Mr. Goller dated August 23, 1974 and January 10, 1975.

An analysis has been completed of the cask handling crane using the following initial conditions:

1. A single element cask having a weight of 51,200 lbs., (includes the weight of single PWR fuel assembly, water, yoke and crane hook), and a maximum diameter of 47.13 inches. The dry weight of the cask is 45,500 lbs.

2. Cask being lowered at the maximum possible main hoist speed of seven feet per minute.

3. Simultaneous setting of both DC magnetic electric shoe brakes.

4. Each brake has an actual capacity of 219.45 tons.

The results of this analyses are as follows:

	ACTUAL LOAD (Static Plus Dynamic)	Load Required to Initiate Yielding
TRUNNION	28,400 lbs. per Trunnion	119,400 lbs. per Trunnion
YOKE		
Top Section	56,800 lb. total	253,400 lbs.
Lower Section	28,400 lb. per arm	81,100 lb. per arm

HEAD BOLTS Not applicable to cask lift stresses since the trunnions transit their load directly to the cask body and not to the cask head (and, hence, the cask bolts).

The deceleration forces which result from the DC magnetic electric shoe brakes being simultaneously applied while a cask is being lowered at seven feet per minute and imparted to the cask and yoke. These forces are less than the forces required to initiate yielding in the yoke and trunnions. The low deceleration forces are attributable to the ability of the supporting wire ropes to elongate and, hence absorb the kinetic energy of the cask. A conservative length of fifty (50) inches of wire rope was assumed for this calculation. The dynamic and static loads imposed on the wire rope will have no deleterious effect on the rope.

As stated in FPL letter to Mr. Goller of January 10, 1975:

> "A two blocking situation which would result from raising the load too high is prevented by two sets of screw type limit switches. The first set limits the cask height such that the cask will clear the fuel pool wall by approximately six inches. The second set, normally furnished to prevent blocking, will now act as a redundant stop. The operator viewing from the cab, as well as other observers, will add further assurance that this or any other non-routine maneuver will be avoided. "Dead Man" protection is accomplished by spring returns on all the crane's master function switches. This protective function removes power from all drive sources."

There is approximately fifteen (15) feet between the setpoint and the upper limit of hook travel. The fifteen feet allows ample room to accommodate setpoint tolerances. Moreover, ample time is provided for the crane operator to secure power to the crane in the unlikely event that both limit switches failed and, therefore, prevent two-blocking from occurring.

The alleyway (the clear north-south passage that is located just east of the Units 3 & 4 auxiliary building) between Units 3 and 4 above which the cask will be moving during the transfer of fuel elements between the two units, all of the critical piping located beneath it, and the underground cable duct banks have been analyzed for the cask drop. Adequate protection for the areas of interest were provided as required by a combination of the compacted soil, a concrete slab four inches thick, and two pipe trenches (2'0" and 4'9") covered with 1" thick steel plate which provide adequate protection for the piping from a cask drop. The other five foot wide trench was provided with a suitable cover to ensure protection for the piping from a cask drop.

With the precautions mentioned above, we have concluded that the safe shutdown of both reactors will not be precluded in the unlikely event of a spent fuel cask drop during transfer from one unit to the other.

Figure 14.2.1-1 shows the loci of possible impact points in the unlikely event of a cask tip. For this loci of points, we assumed that the cask hit the Spent Fuel Building walls as it was being transferred, the yoke disengaged, and the cask tipped and fell horizontally into the pool.

The re-rack program does not alter the cask handling procedures described in Updated FSAR Section 9.5. The cask handling crane meets the design and operational requirements of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [Reference 6].

In response to a request from the NRC (Reference 14) as part of the cask area rack license amendment process, FPL submitted additional information in Reference 15 regarding a spent fuel cask drop. FPL’s license amendment request (Reference 16) stated that a rack drop accident analysis was not required since it was bounded by the cask drop accident. The Reference 15 FPL response supplements the original cask drop analysis transmitted to the NRC under FPL letter L-76-234 (Reference 17). The supplemental information addresses the degree of compliance of the cask drop analysis to the criteria in Appendix A of NUREG-0612, Control of Heavy Loads at Nuclear Power Plants. This information supports the FPL determination that a drop of the cask area rack is bounded by the cask drop accident analysis. The NRC staff agreed with FPL’s conclusion on the basis of this information as documented in the Safety Evaluation Report for the cask area rack license amendment (Reference 18).

14.2.1.3.2 Radiological Consequences

The information provided in this section is historical. For the calculation of radiological consequences potentially resulting from a cask drop accident, two cases were evaluated regarding the number of fuel assemblies that are assumed to suffer a loss of integrity:

Case 1: The number of assemblies damaged is equal to the number offloaded during a normal refueling plus the remainder of the pool filled with discharged assemblies from previous refuelings.

Case 2: The number of assemblies damaged is equal to a full-core offload plus the remainder of the pool filled with discharged assemblies from previous refuelings.

The model for calculating the thyroid and whole-body site boundary doses incorporated the conservative assumptions specified in Standard Review Plan (SRP) Section 15.7.5 [Reference 7] and Regulatory Guide 1.25 [Reference 8], with the exception that a 1.0 Radial Peaking Factor (RPF) was utilized for Case 2. An RPF of 1.65, as specified in Regulatory Guide 1.25, is intended to represent the highest burnup fuel assembly to which all the impacted fuel assemblies are to be equated. While this value may be appropriate for the analysis of a postulated accident involving a single assembly, it is grossly over conservative when applied to an analysis of a full core whose fuel assemblies have various exposure histories. An RPF of 1.0 has been determined as being more representative for the offload of a full core and has been applied to each assembly in the Case 2 analysis. The use of a 1.0 RPF for the calculation of cask drop radiological consequences has been previously submitted to the NRC for FPL's St. Lucie Unit 1 plant (see Reference 9).

Table 14.2.1-6 lists the thyroid doses for the two cases evaluated. (The whole-body doses are not listed since the thyroid doses are limiting for both cases.) The results of the analysis demonstrate that by requiring the decay time of spent fuel in the pool to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit, the potential offsite doses will be less than the guidelines of SRP Section 15.7.5 should a dropped cask strike the stored fuel assemblies. These doses are well within 10 CFR Part 100 limits. This is conservative since not all spent fuel storage modules located in the pool are susceptible to impact from any single cask drop. Thus, the re-racking does not increase the radiological consequences of a cask drop accident previously evaluated.

14.2.1.3.3 Overhead Cranes

Except for the area described in Section 14.2.1.3.1, the spent fuel cask crane is not capable of traveling over or into the vicinity of the spent fuel pool. A complete cask crane component description, cask handling description, and cask crane design evaluation are provided in Updated FSAR Section 9.5 and were not affected as a result of the re-rack program.

14.2.1.3.4 Construction Accidents

To ensure that potential offsite doses from a construction accident during rack replacement were less than those from a cask drop accident, the re-racking operation took place no sooner than 2150 hours after shutdown for the last batch of spent fuel placed in the SFP. This increased decay time was required, since the water level in the SFP was reduced approximately 8 feet during rack handling operations which resulted in a reduced pool decontamination factor per Regulatory Guide 1.25.

14.2.1.3.5 Acceptability

The accident aspects of review establish acceptability with respect to Sections 14.2.1.3.1 and 14.2.1.3.2 of this report.

Requiring spent fuel decay time to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit will keep potential offsite doses well within 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies.

14.2.1.3.6 Fuel Decay

Prior to cask handling operations, a decay time of 1525 hours for all fuel in the pool is required. Thus with the increased storage capacity, the radiological consequences of a cask drop will be well within the requirements of 10 CFR Part 100.

14.2.1.3.7 Loads Over Spent Fuel

A maximum weight of 2000 pounds may be transported over spent fuel unless the entire handling system meets the single-failure-proof requirements of NUREG-0612, Section 5.1.6.

14.2.1.3.8 Spent Fuel Pool Boiling Analysis for Dose Assessment

During the NRC review of the spent fuel re-rack project design, the NRC requested information related to the consequences of offsite doses calculated as a result of pool boiling. This request was related to an NRC concern over the seismic qualifications of the spent fuel pool make-up sources.

A conservative analysis has been performed to determine the radiological consequences of a postulated spent fuel pool boiling event with makeup from only the Seismic Class I RWST. This analysis is consistent with the methodology and assumptions utilized in a similar pool boiling calculation performed for the Limerick plant (Reference 11). The Limerick pool boiling analysis was reviewed by NRC and found acceptable in Reference 11.

The following assumptions were used for the pool boiling analysis:

a. The saturation noble gas and iodine inventories in the core are based on a power level of 2300 MWt with an initial enrichment of 4.5 weight percent and a discharge burnup of 50,000 MWD/MTU.

b. The SFP cooling systems for both pools (Units 3 and 4) fail simultaneously, one containing a full-core offload of 157 assemblies decayed for 150 hours, and the other pool containing a half-core from the last refueling decayed 15 hours.

c. Pool boiling occurs instantaneously upon loss of SFP cooling; no credit is taken for decay during the pool heatup period.

d. 1% of the fuel rods in the core are defective. The 1/2-core from the last refueling is assumed to contain the defective 1% of the fuel rods from that core.

e. The gap activity consists of 10% of the total noble gases except Kr-85, 30% of the Kr-85 activity, and 10% of the total radioactive iodine contained in the fuel rods.

f. Because of their short decay-times, I and Xe in fuel from past refuelings are negligible.

g. Activity in the SFP water at the initiation of boiling is negligible compared to the activity released from the fuel during pool boiling.

h. The iodine and noble gas leakage rates from the fuel rods are 1.3×10^{-8} sec^{-1} and 6.5×10^{-8} sec^{-1}, respectively (Reference 12). These are the full power design fuel leak rates.

i. Although there is no data to support the phenomena in the SFP boiling scenario, iodine spiking factors of up to 50 are analyzed. In general, spiking has been observed during abrupt temperature and pressure transients associated with startup and shutdown, but such significant spiking effects would not be expected during the gradual temperature change that would be associated with a loss of SFP cooling. Since the temperature of the fuel during boiling is expected to be well below the normal reactor core operating temperature, the use of the full-power leakage rate is considered to be conservative.

j. Activity released from the fuel is uniformly mixed in the SFP water volume.

k. The pool water level is at El. 38' with continuous makeup capability (see the discussion below). This height corresponds to the elevation where the SFP cooling system lower suction line penetrates the SFP wall.

l. The iodine partition factor at the pool surface is 0.1.

m. The activity release rate from both pools is conservatively based on an evaporation (boiloff) rate for a full-core offload.

n. All activity escaping from the pool is instantaneously released at ground level to the atmosphere without filtration or condensation in the ventilation system.

o. The atmospheric dispersion (X/Q) factors for dilution are taken from Section 14.3.5 of the Updated FSAR.

As shown in Table 14.2.1-7, the offsite dose consequences of a postulated pool boiling event are a small fraction of 10CFR100 limits.

14.2.1.3.9 Conclusions

Since the spent fuel cask will not be handled over or in the vicinity of spent fuel except as provided for in Section 14.2.1.3.1, the re-racking does not result in a significant increase in the probability of the cask drop accident previously evaluated in the Turkey Point Updated FSAR or Safety Evaluation Report [Reference 13]. Furthermore, as shown in Section 14.2.1.3.2, by requiring the decay time of spent fuel to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit, the potential offsite doses will be well within 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies. The proposed spent fuel pit modifications will not increase the radiological consequences of a cask drop accident previously evaluated.

14.2.1.4 REFERENCES

1. Westinghouse WCAP-14276,"Florida Power and Light Company Turkey Point Units 3 and 4 Uprating Licensing Report," Revision 1, dated December 1995.

2. S.S. Witter to J.L. Perryman, "Fuel Rod/Grids Spring Loads as Related to Handling and Shipping," 98FP-G-0063, July 7, 1998.

3. Deleted

4. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

5. Turkey Point Plant Units 3 and 4, Technical Specifications, Docket Nos. 50-250 and 50-251, through Amendments 223/218.

6. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, July 1980.

7. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Revision 1, July 1981.

8. Nuclear Regulatory Commission, "Assumption 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," Regulatory Guide 1.25, March 1972.

9. St. Lucie Plant Unit 1, Final Safety Analysis Report, Section 9.1, Docket No. 50-335.

10. Turkey Point Plant Units 3 and 4, Docket Nos. 50-250 and 50-251, License Nos. DPR-3 and DPR-41, Appendix A, Technical Specifications.

11. Limerick Generating Station Units 1 and 2, Final Safety Analysis Report, Vol.10, Section 9.1.

12. Source Term Data for Westinghouse Pressurized Water Reactors, WCAP 8253, July 1975.

13. Turkey Point Plant Units 3 and 4, Safety Evaluation Report supporting Amendments 23 and 22 to License Nos. DPR-31 and DPR-41, respectively, Docket Nos. 50-250 and 50-251.

14. Nuclear Regulatory Commission letter to FPL dated July 18, 2003, Request for Additional Information Regarding Addition of Cask Area Spent Fuel Storage Racks Amendment, Information Request No. 27.

15. FPL letter L-2004-213 to the Nuclear Regulatory Commission dated June 21, 2004, Spent Fuel Pool Cask Area Racks - Response to NRC Request for Additional Information.

16. FPL letter L-2002-214 to the Nuclear Regulatory Commission dated November 26, 2002, Proposed License Amendments – Addition of Cask Area Spent Fuel Storage Racks.

17. FPL letter L-76-234 to the Nuclear Regulatory Commission dated June 23, 1976.

18. Nuclear Regulatory Commission letter to FPL dated November 24, 2004, Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Temporary Spent Fuel Pool Cask Racks (TAC Nos. MB6909 and MB6910).

19. NAI Report NAI-1321-001, Revision 0, "Turkey Point Fuel Handling Accident AST Radiological Analysis with High Burnup Fuel, July 7, 2007.

20. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants", May 1979

21 Turkey Point Unit 3 and 4 - Issuance of Amendments Regarding Technical Specification Changes Related to Movement of Heavy Loads Over Spent Fuel (TAC Nos. ME3379 and ME3380) dated February 25, 2011.

22. USNRC Regulatory Issue Summary 2006-04, “Experience with Implementation of Alternate Source Terms,” March 7, 2006

23. American National Standard, ANS-51.1/N18.2-1973, “Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants”, August 6, 1973. C26

Table 14.2.1-1

Assembly Bounding Source Term for Fuel Handling Accident Analysis

Security-Related Information - Withheld Under 10 CFR 2.390

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Table 14.2.1-2

Major Assumptions and Parameters for Fuel Handling Accident Radiological Dose Analysis

Input/Assumption	Value	
Core Power Level Before Shutdown	2652 MW_{th}	
Discharged Fuel Assembly Burnup	45,000 MWD/MTU	
Fuel Enrichment	3.0 – 5.0 w/o	
Radial Peaking Factor	1.65	
Number of Fuel Assemblies Damaged	1	
Release Fraction from Breached Fuel	**Group**	**Fraction**
	I-131	0.12
	Kr-85	0.30
	Other Noble Gases	0.10
	Other Halogens	0.10
Delay Before Spent Fuel Movement	72 hours	
Release Duration	2 hours	
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%	
Atmospheric Dispersion Factors Offsite Onsite	 Appendix 2E Appendix 2F	
Time of CR Isolation – Containment Release Time of CR Isolation – FHB Release Unfiltered Inleakage	30 seconds - High Containment Radiation 30 minutes from Manual Isolation 100 cfm	
Breathing Rates	Reg. Guide 1.183 Sections 4.1.3 and 4.2.6	
Control Room Occupancy Factor	Reg. Guide 1.183 Section 4.2.6	

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Table 14.2.1-3

Fuel Handling Accident Offsite and Control Room Doses (REM TEDE)

Case	EAB Dose [1] (rem TEDE)	LPZ Dose[2] (rem TEDE)	Control Room Dose[2] (rem TEDE)
FHA in Containment	0.73	0.15	1.22
FHA in Fuel Handling Building	0.73	0.15	3.70
Acceptance Criteria	**6.3[3]**	**6.3[3]**	**5.0[4]**

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Notes:

(1) Worst 2-hour dose
(2) Integrated 30-day
(3) FHA Criteria from Reg. Guide 1.183, Table 6
(4) 10CFR50.67

TABLE 14.2.1-4

ASSUMPTIONS USED
FOR
DROPPED CASK DOSE ANALYSIS

Power	2346 MWt
Radial Peaking Factor	1.0
Damaged Fuel (Base Case)	157 Fuel Assemblies
Fuel Rod Gap Fractions	0.10 for iodines and noble gases, except 0.12 for I-131 and 0.30 for Kr-85
Percent of Gap Activity Released	100%
Pool Decontamination Factors:	
Elemental Iodine	133
Methyl Iodide	1
Noble Gas	1
Iodine Species in Fuel Rod Gap:	
Elemental Iodine	99.75%
Methyl Iodide	0.25%
Minimum Water Depth Above Damaged Assembly	23 feet
Filter Efficiency	No filtration assumed

TABLE 14.2.1-5

DROPPED CASK OFFSITE DOSES
(157 Fuel Assemblies Damaged)

	Exclusion Boundary (EB) (0-2 Hours)	Low Population Zone (LPZ) (0-2 Hours)
Thyroid Dose (rem)	1.77 E+1	1.73 E+0
Whole-Body Dose (rem)	2.42 E-2	2.36 E-3

TABLE 14.2.1-6

SITE BOUNDARY DOSES DUE TO A

SPENT FUEL CASK DROP [1]

Note: The information provided in this table is historical. The updated site boundary doses are provided in Section 14.2.1.3

	Number of Damaged Assemblies Assumed[3]	Decay Time Prior to Cask Handling (hours)	Site Boundary Thyroid Dose (Rem)[2]
Case 1	80 (Refueling)	1475	27
Case 2	Full-Core Offload	1525	27

Notes

1. Based on assumptions specified in Standard Review Plan Section 15.7.5 and Regulatory Guide 1.25 with the exception that a 1.0 Radial Peaking Factor was used for Case 2. See the discussion in Section 14.2.1.3.2.

2. Whole-body doses are not listed since the thyroid doses are limiting for both cases.

3. Remainder of pool filled with discharged assemblies from previous refuelings.

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TABLE 14.2.1-7

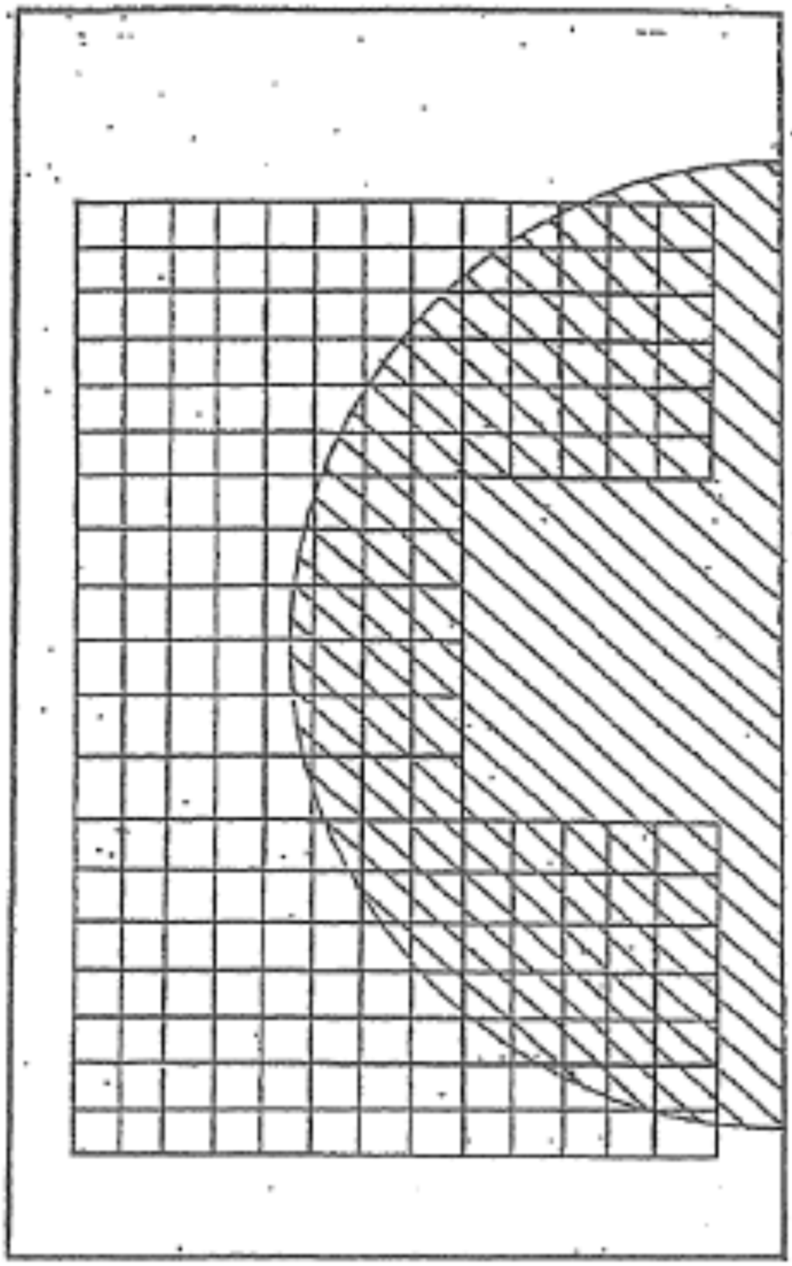
RESULTS OF SPENT FUEL POOL BOILING ANALYSIS

Site Boundary 2-Hour Dose (Rem)

	Spike = 1	Spike = 50
Thyroid	5.7E-3	2.8E-1
Whole-Body	1.3E-5	1.8E-4

LPZ 30-Day Dose (Rem)

	Spike = 1	Spike = 50
Thyroid	1.4E-2	5.6E-1
Whole-Body	2.8E-5	1.8E-4



REVISED 09/29/2005

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

SPENT FUEL PIT

LOCI OF CASK TIP IMPACT POINTS

FIGURE 14.2.1-1

14.2.2 ACCIDENTAL RELEASE-RECYCLE OR WASTE LIQUID

Accidents in the auxiliary building and in the radwaste handling facility building which would result in the release of radioactive liquids are those which may involve the rupture or leaking of system pipe lines or storage tanks. The largest vessels are the three liquid holdup tanks, each sized to hold one and one third of a single unit reactor coolant liquid volume, which are used to process the normal recycle or waste fluids produced. The contents of one tank will be passed through the liquid processing train while another tank is being filled.

All liquid waste components except the reactor coolant drain and the pressurizer relief tanks are located in the auxiliary building and in the radwaste handling building, and any leakage from these components or piping will be collected in the respective building sumps to be pumped back into the liquid waste system.

The gross rupture of these tanks is not considered credible in view of the service conditions. However, the plant design has accommodated tank ruptures as described below.

In the unlikely event of a rupture of a full CVCS holdup tank, all spilled liquid will be contained within the CVCS tank cubicles, and no uncontrolled liquid release will occur. The walls of these cubicles have been coated to the calculated flood height and the floor drains are normally closed. The flooded cubicle can be drained to the auxiliary building waste holdup tank room sump and then pumped to the waste holdup tank. Any liquid remaining in the ruptured holdup tank could be transferred to another holdup tank by means of the recirculation pump.

In the unlikely event of a rupture of a full waste holdup tank all spilled liquid will be contained by the walls surrounding the tank and no uncontrolled liquid release will occur. The enclosure is coated to the calculated flood height. The spilled liquid around the auxiliary building waste holdup tank will drain to the waste holdup tank room sump pumps and can then be discharged to the holdup tanks or to the waste evaporator, through the waste evaporator feed pump.

The holdup tanks are also equipped with safety pressure relief and designed to accept without loss of function the maximum potential seismic forces at the site. Liquids in the Chemical and Volume Control System flowing into and out of these tanks are controlled by manual valve operation and governed by prescribed administrative procedures.

The volume control tank design philosophy is similar in many respects to that applied for the holdup tanks. Level alarms, pressure relief valves and automatic tank isolation and valve control assure that a safe condition is maintained during system operation. Excess letdown flow may be directed to the holdup tanks via the reactor coolant drain tank or via the volume control tank.

Piping external to the containment running between the containment and the auxiliary building area will be in concrete pipe chases.

The effect from these process or waste liquid releases is derived only from the volatilized components. The releases are described and their effects summarized in Section 14.2.3.

The evaluation of the credibility of the accidental release of radioactive fluids above normal concentration (~4 x 10^{-5} uc/cc) from the Waste Disposal System discharge is based upon the following review of waste discharge operating procedure, monitoring function description, monitor failure mode and the consequences of a monitor failure.

The normal procedure for discharging liquid wastes is as follows:

a) A batch of waste is collected in a waste monitor tank or monitor tank for discharge (typical).

b) The particular tank is isolated

c) The tank contents are recirculated to mix the liquid

d) A sample is taken for radiochemical analysis

e) If analysis indicates that release can be made within permissible limits, the quantity of activity to be released is recorded on the basis of the liquid volume in the tank and its activity concentration. If release can not be made within permissible limits, the waste is returned to the waste holdup tank.

f) To release the liquid, the tank to be discharged is lined up; the pump used for discharge is started; valve RCV-018, which trips shut automatically on high radiation signal from the monitor, must be opened manually; the last stop valve in the discharge line (which is normally locked shut) must be unlocked and opened; and finally release flow is throttled via a third valve to the prescribed flow rate.

As the operating procedure indicates, the release of liquid waste is under administrative control. The process radiation monitor RD-18 is provided to maintain surveillance over the release.

The monitor is provided with the following features:

a) A calibration source is provided to permit the operator to check the monitor before discharge by turning a switch in the control room to activate the circuitry.

b) If the monitor falls off scale at any time, an indicator visible to the operator in the control room lights.

c) If the power supply to the monitor fails, a high radiation alarm is annunciated. The trip valve also closes.

d) The trip valve is failed closed, normally closed.

It is concluded that the administrative controls imposed on the operator combined with the safety features built into the equipment provide a high degree of assurance against accidental release of waste liquids.

No credible mechanism exists for accidental release of waste liquids to Biscayne Bay.

14.2.3 ACCIDENTAL RELEASE - WASTE GAS

The leakage of fission products through cladding defects can result in a buildup of radioactive gases in the reactor coolant. Based on experience with other operational, closed cycle, pressurized water reactors, the number of defective fuel elements and the gaseous coolant activity is expected to be low. The shielding and sizing of components such as demineralizers and the waste handling system are based on activity corresponding to 1% defective fuel which is at least an order of magnitude greater than expected. Tanks accumulating significant quantities of radioactive gases during operation are the gas decay tanks, the volume control tank, and the liquid holdup tanks.

The volume control tank accumulates gases over a core cycle by stripping action of the entering spray. Equilibrium gaseous activity for the tank based on operation with 1% defective fuel is tabulated in Table 14.2.3-1. During a refueling shutdown this activity is vented to the waste gas system and stored for decay. Rupture of this tank is assumed to release all of the contained noble gases. Pre-EPU evaluations established that the released activity would be 32,330 curies equivalent Xe-133. The offsite whole body doses due to that volume control tank rupture were 0.038 rem at the exclusion boundary and 0.0036 rem at the low population zone. The offsite whole body doses did not exceed the then current acceptance criteria for dose due to a waste gas system failure. Since the waste gas system failure was identified as the limiting systems failure for a gaseous release, the volume control tank failure was not numerically re-evaluated for the Turkey Point EPU conditions. As was the case for pre-EPU conditions, the waste gas system failure off-site dose results will bound the dose consequence results for the volume control tank failure event. C26

The liquid holdup tanks receive reactor coolant, after passing through demineralizers, during the process of coolant deboration. The liquid is stored and then processed. Each of the three liquid holdup tanks is sized to hold one and one-third of a single unit reactor coolant liquid volume. The contents of one tank are passed through the liquid processing train while another tank is being filled. In analyzing the consequence of rupture of a holdup tank, it is assumed that 100% of the contained noble gas activity is released. This activity is much less than that available for possible release from a waste gas decay tank due to approximately six hours holdup tank filling time during which activity decay occurs and due to the reactor coolant dilution during the letdown operation.

The waste gas decay tanks (WGDT) receive the radioactive gases from the liquids processed by the waste disposal system throughout the operating fuel cycle. The maximum storage of waste gases occurs after a refueling shutdown at which time the gas decay tanks store the radioactive gases stripped from the reactor coolant. The WGDT Dose Equivalent Xe-133 activity based upon the total RCS noble gas activity from operation with 1% fuel defects is 84,274.8 Curies, which bounds the Technical Specification limit of 70,000 Curies Dose Equivalent Xe-133.

Dose Evaluation

Offsite exposure is evaluated for noble gases release generally based on the model described in Section 14.3. The WGDT dose evaluation model incorporates elements of the Alternative Source Term (AST) methodology outlined in Regulatory Guide 1.183 (Reference 1). This reference does not provide specific guidance for the WGDT event, but it does provide general modeling guidance for other accidents. Specific guidance for the WGDT analysis of this event is given in Branch Technical Position 11-5 of the Standard Review Plan (Reference 2). Additional guidance regarding the acceptance criteria for this event is given in Issue #11 of NRC Regulatory Issue Summary 2006-04 (Reference 3). General modeling inputs for the WGDT dose consequence evaluation are provided in Table 14.2.3-2. C26

The WGDT source term provided in Table 14.2.3-3 is based upon the plant operating at a power level of 2652 MW_{th} (2644 + 0.3%) with one percent failed fuel for an extended period of time sufficient to achieve equilibrium radioactive concentrations in the Reactor Coolant System. The entire noble gas inventory of the reactor coolant system is then assumed to be stripped and placed into a single WGDT. The resulting Table 14.2.3-3 WGDT inventory conservatively exceeds the Technical Specification limit for allowable equivalent curies of Xe-133.

The integrated 30 day dose at the EAB for the WGDT event is 0.066 rem TEDE, and the LPZ dose is 0.013 rem TEDE. The off-site doses are both less than the 0.1 rem TEDE limit specified in BTP 11-5 of the Standard Review Plan. It is therefore concluded that an accidental waste gas release would present no adverse effect on safety of the public.

Although BTP 11-5 does not require Control Room dose to be calculated for the WGDT rupture event, the integrated 30 day control Room dose has been evaluated for completeness as 0.34 rem TEDE, within the 5 rem TEDE limit that is applied to Control Room operators for other analyzed events.

14.2.3.1 REFERENCES

1. USNRC, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants”, July 2000.

2. USNRC, NUREG -0800, Standard Review Plan, Branch Technical Position BTP-11-5, “Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure,” Revision 3, March 2007. C26

3. USNRC, Regulatory Issue Summary 2006-04, “Experience with Implementation of Alternative Source Terms”, March 7, 2006.

TABLE 14.2.3-1

VOLUME CONTROL TANK NOBLE GAS ACTIVITY
(Pre-EPU Conditions)

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Security-Related Information - Withheld Under 10 CFR 2.390

TABLE 14.2.3-2

WASTE GAS DECAY TANK (WGDT) FAILURE – INPUTS AND ASSUMPTIONS

Input/Assumption	Value
Core Power Level	2652 MW_{th}
Tank Volume	525 ft^3
Tank leak rate (arbitrarily high)	1E+06 cfm
Time of CR Isolation	Not isolated
Unfiltered Inleakage	100 CFM
Makeup Flow	1000 cfm
Atmospheric Dispersion Factors	
Offsite	Appendix 2E
Onsite	Appendix 2F
Breathing Rates	
Offsite	Reg. Guide 1.183, Section 4.1.3
Control Room	Reg. Guide 1.183, Section 4.2.6
CR Occupancy Factors	Reg. Guide 1.183, Section 4.2.6

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TABLE 14.2.3-3

WGDT DOSE EQUIVALTENT XE-133 DETERMINATION

Security-Related Information - Withheld Under 10 CFR 2.390 C26

14.2.4 STEAM GENERATOR TUBE RUPTURE

The event examined is a complete tube break adjacent to the tube sheet, since a minor leak may not necessitate immediate action depending on the particular circumstances. If a tube breaks, reactor coolant would discharge into the secondary system. Since the reactor coolant is radioactive, methods of operation to limit uncontrolled condensate release have to be considered.

Once the Reactor Coolant System (RCS) pressure is below the steam generator design pressure the ruptured steam generator will be isolated and the possibility of uncontrolled leakage removed.

The following sequence of events is initiated by a tube rupture:

1. Rapidly falling pressure in the pressurizer will initiate a safety injection signal, tripping the unit. The safety injection signal automatically terminates normal feedwater and initiates auxiliary feedwater.

2. The steam generator blowdown monitor and the steam jet air ejector radiation monitor will alarm, indicating the passage of primary fluid into the secondary system.

3. The unit trip will automatically shut off steam flow through the turbine and will open steam bypass valves and bypass steam to the condenser.

4. In the unlikely event of concurrent loss of power, the loss of circulating water through the condenser would eventually result in loss of condenser vacuum and the valves in the turbine bypass lines would automatically close to protect the condenser, thereby causing steam relief to atmosphere.

5. Cooldown procedures are followed which entail condenser relief (if available) or atmospheric relief from the intact steam generators to reduce the reactor coolant temperature.

6. Maximum charging flow may be established prior to SI flow reduction to provide a readily controllable means of maintaining inventory and subcooling.

7. Isolation of the ruptured steam generator is achieved by:

   a. reducing safety injection flow to depressurize the RCS below the ruptured steam generator pressure;

   b. closing the steam line stop valve connected to the affected steam generator (determined by steam generator liquid sample activity monitor); and

   c. turning off the auxiliary feedwater flow to that steam generator.

8. Safety injection flow would be terminated to prevent repressurization of the RCS and reinitiation of break flow while the cooldown is continuing from the intact steam generators.

9. After the residual heat removal system is in operation, the condensate accumulated in the secondary system can be examined. If the radioactivity level is in excess of that allowed, the condensate can be processed through the waste disposal system.

The ruptured unit will be isolated by a steam line isolation valve once the reactor coolant pressure is reduced below the ruptured steam generator pressure. The mass flow into the secondary system and steam relief from the ruptured steam generator will be terminated.

With power available to the circulating water pumps the steam is bypassed to the condenser.

With concurrent loss of power a portion of the reactor coolant system, activity is released to the atmosphere via steam relief from the ruptured steam generator.

Demonstration that the ruptured steam generator does not overfill during the accident has been performed by utilizing an NRC-approved thermal hydraulic analysis code. Reference 2 includes the NRC's approval of the LOFTTR2 computer code that has been used for the Turkey Point overfill analysis. This code simulates the plant response, and models specific operator actions. Thus, a more realistic representation of the break flow during the accident is obtained. The analysis demonstrates that break flow following the complete severance of a steam generator tube is terminated within 53 minutes after initiation of the tube rupture and that overfill of the steam generator does not occur. The analysis also resulted in a steam release that was less than the 30 minute steam release assumed in the dose analysis such that the dose analysis remains bounding for the overfill analysis. C27

14.2.4.1 STEAM GENERATOR TUBE RUPTURE (SGTR) RADIOLOGICAL CONSEQUENCES

The steam generator tube rupture (SGTR) dose consequence analysis for Extended Power Uprate (EPU) conditions is consistent with the guidance provided in Appendix F of Reference 3, "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 Turkey Point SGTR event.

2. Regulatory Position 2 - No fuel damage is postulated to occur for the Turkey Point 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 TS 3.4.8 Figure 3.4-1, 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 TS 3.4.8 value of 0.25 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. C26

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 equal to the value specified by TS 6.8.4.j.b.2, which is 0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions.

8. Regulatory Position 5.2 - The density used in converting primary-to-secondary volumetric leak rates to mass leak rates is 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

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. This is conservatively calculated to occur at 125.4 hours. The release of radioactivity from the unaffected steam generators is assumed to continue until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 hours. Termination of the ruptured steam generator activity release is occurs when the ruptured steam generator is isolated at 30 minutes by operator action. While this isolation terminates releases from the ruptured steam generator, primary-to-secondary leakage continues to provide activity for release from the unaffected steam generators.

10. Regulatory Position 5.4 - The release of fission products from the secondary system is evaluated with the assumption of a loss of offsite power coincident with reactor trip.

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. C26

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 - A portion of the primary-to-secondary ruptured tube flow following the SGTR is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary coolant. The flashed flow is released to the environment with no mitigation. For the unaffected steam generators, flashing is considered immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

    - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the bulk water is credited.

- Appendix E, Regulatory Position 5.5.3 - All of the steam generator leakage and ruptured tube flow that does not immediately 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 radionuclides in the steam generators is limited by the moisture carryover. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the steam generator carryover rate of less than 1%.

C26

- Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery in the unaffected steam generators is postulated for up to 30 minutes following a reactor trip. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

Other assumptions and modeling inputs are described below:

1. RCS and steam generator volumes are assumed to remain constant throughout both the pre-accident and the accident-induced iodine spike SGTR events.

2. During a SGTR event, from the onset of the tube rupture until the time of reactor trip, there will be no actual steam releases through the ADVs or MSSVs. Radionuclides will most likely enter the atmosphere through the condenser Steam Jet Air Ejector (SJAE). Due to relative proximity of the SJAE to the normal Control Room intake, this release-receptor pair produces a more limiting atmospheric dispersion factor than releases from the ADVs/MSSVs to the normal intake. In addition, the pre-trip flashing fraction of the primary-to-secondary leakage is substantially higher than the post-trip flashing fraction. For these reasons, the analysis assumes full rated steam flow from the steam generators prior to reactor trip. In addition to the partition factor of 100 in the steam generator, a partition factor of 100 is also applied to iodines and particulates released from the SJAE prior to the reactor trip. This condenser partition factor is no longer used when the steam release from the ADVs/MSSVs begins.

3. The steam generator and condenser partition factors for iodine and particulates are applied in the analysis by reducing the steam release rate from the steam generator compartment. This methodology conservatively allows the activity, which is not released, to remain in the steam generator compartment and contribute to the radionuclide concentration. C26

4. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the Technical Specification limit of 0.1 µCi/gm Dose Equivalent I-131.

5. Radionuclide concentrations in the secondary side fluid of the steam generators assume that Auxiliary Feedwater is provided to maintain a constant secondary mass during periods of steam release.

6. The steam release rates and ruptured tube flow rates are provided in Table 14.2.4-2.

7. Data used to calculate the iodine equilibrium appearance rate are provided in Table 14.2.4-3.

The major assumptions and parameters used in the analysis are itemized in Table 14.2.4-1. The dose conversion factors, breathing rates, and control room occupancy factors are the same as those given in the LOCA dose analysis in Table 14.3.5-5. The atmospheric dispersion factors used in the SGTR dose calculations are given in Appendices 2E and 2F.

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. Initial radionuclide releases occur through the condenser SJAE until the time of reactor trip; after reactor trip, steam relief is exhausted directly to the atmosphere from the ADVs or MSSVs. This direct steam relief continues until the ruptured steam generator is isolated at 30 minutes.

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the ruptured steam generator relief valves. Table 14.2.4-2 provides the break flow and steam release mass flow rates for the cool down event. The analysis assumes that activity is released as reactor coolant enters the steam generators due to primary-to-secondary leakage. The equilibrium, pre-event source term for this primary RCS activity is presented in Table 14.3.5-8. All noble gases associated with this leakage are assumed to be released directly to the environment. Primary coolant is released into the ruptured steam generator through the ruptured tube and from a fraction of the total proposed allowable primary-to-secondary leakage until the ruptured steam generator is isolated at 30 minutes. Additional activity, based on the primary-to-secondary leakage limits, is released via the unaffected steam generators. All primary-to-secondary leakage is assumed to continue until the temperature of the leakage is less than 212°F, which is conservatively calculated to occur at 125.4 hours. Steam release from the unaffected steam generators is assumed to continue until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 hours. The release of the initial iodine content of the steam generator secondary is also considered. The source term for this activity is presented in Table 14.2.4-4. C26

No fuel melt or clad breach is postulated for the Turkey Point SGTR event. Consistent with Appendix F or Reference 3, 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 Technical Specification transient maximum value of 60 µCi/gm DE I-131. The primary coolant iodine activities for the pre-accident spike case are presented in Table 14.2.4-3.

For the case of the accident-inducted spike, the postulated SGTR event induces an iodine spike. The RCS activity is initially assumed to be the Technical Specification steady-state maximum allowable 0.25 µCi/gm DE I-131. 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. With iodine activity at equilibrium, the iodine release rate is equal to the rate at which iodine is lost due to decay, purification, and primary system leakage. Parameters used in the determination of the iodine equilibrium release rate are provided in Table 14.2.4-3. The iodine activities for the accident-induced (concurrent) iodine spike case are also presented in Table 14.2.4-3. All other release assumptions for this case are identical to those for the pre-accident spike case. C26

For the SGTR 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 flow distribution during this mode is 1000 cfm of unfiltered fresh air plus 100 cfm of unfiltered leakage.

- Control Room is isolated due to a safety infection, which occurs at 291 seconds. A 30-second delay is applied to account for the signal processing, diesel start, and damper closure time. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 100 cfm of unfiltered inleakage, and 375 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 and 95% for elemental and organic iodine.

The atmospheric dispersion factors (X/Qs) used for the Control Room dose evaluation are based on the postulated release locations and the operational mode of the control room ventilation system. These X/Qs are summarized in Appendix 2F. Prior to the time of reactor trip, releases are from the condenser SJAE to the normal intake. Immediately following reactor trip, releases from the steam generators are assumed to occur from the MSSV/ADV which produce the most limiting X/Q. The receptor point shifts to the most limiting emergency intake after control room isolation occurs.
The EAB and LPZ doses are determined using the X/Q factors for the appropriate time intervals. These X/Q factors are provided in Appendix 2E. C26

The offsite and control room dose limits for a steam generator tube rupture are provided in Regulatory Guide 1.183 (Reference 3).

The offsite and control room doses due to the steam generator tube rupture are given in Table 14.2.4-5. The offsite doses due to the steam generator tube rupture are within the acceptance criteria.

Additional (non-DBA) SGTR Event Evaluations C26

The SGTR event is monitored by the main steam line radiation monitor (RAD-6426), which monitors each steam line for high-range noble gas activity (see Subsection 11.2.3). This monitor is kept in service throughout the event. A pre-EPU evaluation of a sample tube failure concurrent with a SGTR was performed (Reference 1). Many hours are available to isolate the monitor without exceeding the offsite dose limits for this event. C26

Additional pre-EPU evaluations were performed for variations of the SGTR event scenario (Reference 1). For reasons to be discussed later in this section, the multiple spontaneous occurrence of gross tube failures in a single incident is not considered credible. In order to perform a rigorous analysis of the flow dynamics of blowdown through multiple tube ruptures one must understand and define mathematically the physical configuration of the ruptures. Because no reasonable mechanism exists for the multiple ruptures, it is instead just as meaningful to analyze the consequences of a pipe rupture, equivalent in terms of discharge rate to various multiples of the single tube rupture discharge rate. C26

Such an analysis reveals that the core cooling system will prevent clad damage for break discharge rates equal to or smaller than that resulting from a broken pipe between 4 inches and 6 inches in diameter. The discharge rates which bracket the onset of clad damage correspond to 18 and 40 times the discharge from a single severed steam generator tube. Actually the ratio would be much larger owing to the fact that the discharge from a tube failure will be limited by the back pressure in the steam generator. Ultimately the tube discharge would terminate when the reactor coolant system and the steam generator reached pressure equilibrium. The operator can initiate cooldown through the unaffected steam generators.

The discharge rate required to lift a secondary safety valve is about 15 times the rate from a single severed tube.

These conclusions are based on single-failure mode performance of the core cooling system. Clad damage is prevented in those cases where the top of the core does not become uncovered.

The discharge rate required to cause the top of the core to become uncovered is 18 to 40 times the rate from a single severed tube.

The incredibility of multiple simultaneous tube failures is supported by the following reasoning:

1. At the maximum operating internal pressure the tube wall sees only about 1530 psi, compared with a calculated bursting pressure in excess of 11,100 psi based on ultimate strength at design temperature (factor of 7.3); and compared with a prefabrication test pressure of 7,000 psi (factor of 4.5).

2. The above margin applies to the longitudinal failure mode, induced by hoop stress. This failure mode is the least likely to cause propagation of failure tube-to-tube. An additional factor of two applies to ultimate pressure strength in the axial direction tending to resist double-ended failure (total factor of 14.6).

3. Failures induced by fretting, corrosion, erosion or fatigue, in addition to being rendered extremely improbable by design, are of such a nature as to produce tell-tale leakage in substantial quantity while ample metal remains to prevent severance of the tube (a small fraction of the original tube wall section, as indicated by the margin derived in 2). Thus it is virtually certain that any incipient failures that would develop to the point of severe leakage requiring a shutdown for repair would happen long before the large safety margin in pressure strength is lost.

14.2.4.2 REFERENCES

1. Westinghouse WCAP-14276,"Florida Power and Light Company Turkey Point Units 3 and 4 Uprating Licensing Report," Revision 1, December 1995.

2. Charles E Rossi, NRC, to Alan E Ladieu, WOG SGTR Subgroup ‘Acceptance for Referencing of Licensing Topical Report WCAP-10698 “SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,” December 1984,’ March 30, 1987.

3. USNRC, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants,” July 2000. C26

TABLE 14.2.4-1

ASSUMPTIONS USED FOR
STEAM GENERATOR TUBE RUPTURE DOSE ANALYSIS

Input/Assumption	Value
Release Inputs:	
Core Power Level	2652 MW_{th}
Initial RCS Equilibrium Activity	0.25 μCi/gm DE I-131, and 447.7 μCi/gm DE Xe-133 (Table 14.3.5-8)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 14.2.4-4)
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Iodine Spike Appearance Rate	335 times
Duration of accident-initiated spike	8 hours
Break Flow Flashing Fraction	Prior to Reactor Trip - 21% Following Reactor Trip – 11%
Time of Reactor Trip	291 seconds
Time to isolate ruptured SG	30 minutes
Steam Generator Tube Leakage Rate	0.2 gpm/SG
Time to establish shutdown cooling and terminate intact steam release	63 hours
Time for RCS to reach 212°F and terminate SG tube leakage	125.4 hours
RCS Mass (minimum)	366,086 lbm
Removal Inputs:	
SG Secondary Side Mass	67,707 lbm per SG
Time to re-cover Intact SG Tubes	30 minutes
Tube Uncovery Flashing Fraction	11%
Secondary Side Partition Coefficients	SG (Flashed tube flow) – none SG (Non-flashed tube flow) – 100 Condenser – 100
Transport Inputs:	
Release Locations(s)	Various – See Appendix 2F
Atmospheric Dispersion Factors	
Offsite	Appendix 2E
Onsite	Appendix 2F
Isolation Signal	Safety Injection
Time of CR Isolation	321 seconds
Unfiltered Inleakage	100 cfm

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TABLE 14.2.4-2

SGTR BREAK FLOW AND STEAM RELEASE RATES [1]

Time (hours)	Break Flow into Ruptured SG (lbm/min)	Steam Release from Ruptured SG (lbm/min)	Steam Release from Unaffected SGs[2] (lbm/min)
0 – 0.0808	6507	64,800	129,600
0.0808– 0.5	4161	3579	4033
0.5 – 2	0	0	4033
2 – 8	0	0	2833
8 – 24	0	0	1525
24 – 63	0	0	1270

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(1) Flowrate is assumed to be constant within the time period
(2) Stored energy above RHR entry conditions is released between 2 and 8 hours

Table 14.2.4-3

SGTR IODINE SPIKE AND APPEARANCE RATE MODELING INPUTS

Pre-Accident (60 µCi/gm D.E. I-131) Iodine Spike RCS Activities

Isotope	Activity (µCi/gm)
Iodine-131	48.1440
Iodine-132	34.1280
Iodine-133	58.3440
Iodine-134	6.3192
Iodine-135	28.8000

Iodine Equilibrium RCS Appearance Rate Analysis Assumptions

Input Assumption	Value
Letdown Flow	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	397,544 lbm
I-131 Decay Constant	6.000E-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}

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Concurrent (335 x) Iodine Spike Appearance Rate In RCS

Isotope	Appearance Rate (Ci/min)	8-hour Production (Ci)
Iodine-131	36.11	17333
Iodine-132	68.23	32750
Iodine-133	51.04	24498
Iodine-134	25.60	12290
Iodine-135	33.84	16241

Table 14.2.4-4

SECONDARY SIDE SOURCE TERM (NON-LOCA)

Nuclide	Activity (μCi/g)
I-131	8.022E-02
I-132	5.688E-02
I-133	9.725E-02
I-134	1.053E-02
I-135	4.801E-02

Note: Activity is equivalent to Technical Specification limit of 0.1 μCi/g dose equivalent I-131

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Table 14.2.4-5

STEAM GENERATOR TUBE RUPTURE OFFSITE DOSES

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
SGTR pre-accident iodine spike	0.67	0.14	3.10
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	0.24	0.052	1.28
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

C26

Notes:

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) Regulatory Guide 1.183, Table 6
(4) 10CFR50.67

14.2.5 RUPTURE OF A STEAM PIPE

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in a pipe line or due to a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive rod control cluster assembly (RCCA) is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining RCCAs inserted. A return to power following a steam pipe rupture is a potential problem only because of the high hot channel factors which may exist when the most reactive RCCA is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following a steam line break, the core is ultimately shut down by the boric acid in the refueling water storage tank.

14.2.5.1 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

14.2.5.1.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

The most severe core conditions for an accidental depressurization of the main steam system result from an inadvertent opening of a single steam dump, relief, or safety valve. However, since the effective steam flow area of these valves is less than a full double-ended rupture of a main steam line, the reactor coolant system cooldown and resulting return to power are much less than the rupture of a main steam line. Therefore, with respect to the analysis criteria, an accidental depressurization of the main steam system is always less-limiting than a main steam line rupture. The analyses performed assuming a rupture of a main steam line are given in Subsection 14.2.5.2. C26

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14.2.5.2 STEAM SYSTEM PIPING FAILURE

14.2.5.2.1 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 RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the safety injection (SI) system.

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

A) Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the Engineered Safety Features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10 CFR 50.67. C26

B) Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, this analysis was performed to determine whether DNB occurs for any rupture assuming the most reactive RCCA stuck in its fully withdrawn position. In addition, although fuel centerline melting is not necessarily unacceptable, this analysis was performed to determine whether the peak linear heat generation rate (expressed in kW/ft) exceeds a value that would cause fuel centerline melt. C26 C26

The major rupture of a steam line is a 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 of the main steam piping, is presented here. C26

The Main Steam Isolation Valve Assembly (MSIVA) consists of the MSIV, the Main Steam Check Valve (MSCV), and the Main Steam Bypass Valve (MSBV). For breaks downstream of the MSIVA, the MSIVs will fully close rapidly following a large break in the steam line, completely terminating the blowdown. For breaks between the steam generator exit and the MSIVA, the passive MSCV will prevent blowdown from the intact steam lines. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the MSIVs fails to close.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator flow restrictor. The effective throat area of the nozzles is about 1.4 square feet, which is considerably less than the C26 main steam pipe and thus the nozzles also serve to limit the maximum steam flow for a break at any location.

14.2.5.2.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

<u>Method of Analysis</u>

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 C26 code (Reference 1) 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, C26 has been used to determine if DNB occurs for the core conditions computed in item A above. (Reference 7) C26

The following conditions are assumed to exist at the time of a main steam line break accident: (see Table 14.3.4.2-1 and 14.3.4.3-1)

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

B) A negative moderator temperature coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The K_{eff} versus temperature at 1050 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 14.2.5-1. C26

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 the 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 RCCA. 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.

C) Minimum capability for injection of boric acid (2400 ppm) solution corresponding to the most restrictive single failure in the SI portion of the Emergency Core Cooling System (ECCS). The ECCS consists of three systems: (1) the passive accumulators; (2) the low head safety injection (residual heat removal) system; and (3) the high head safety injection system. Only the high head safety injection system and the passive accumulators are modeled for the steam line break accident analysis. C26

The modeling of the SI system flow corresponds to that delivered by two high head SI pumps (which considers a single train failure) delivering 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 of the RWST prior to the delivery of boric acid to the reactor coolant loops. C26

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 SI system due to changes in the RCS pressure. The SI system flow calculation includes the line losses in the system as well as the SI pump head curve. Figure 14.2.5-13 provides the relationship between SI flow and RCS pressure.

For the case where offsite power is assumed, the sequence of events in the safety injection system is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the SI pumps start. In 25.1 seconds after the initiation of the event, the valves are assumed to be in their final position and the pumps are assumed to be at full speed. The volume containing the low concentration borated water is swept into core before the 2400 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling. Table 14.2.5-2 provides the specific sequence of events. C26

In the case where offsite power is not available, an additional 22 second delay is assumed to start the diesel generators and to commence loading the necessary safety injection equipment onto them. C26

D) To maximize the primary to secondary heat transfer rate, zero (0 percent) steam generator tube plugging is modeled. C26

E) Since the steam generators are provided with integral flow restrictors with an approximate 1.4 ft^2 throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, would have the same effect on the NSSS as the 1.4 ft^2 break. The following cases have been considered in determining the core power and RCS transients: C26

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

2) Case (1) with loss of offsite power simultaneous with the steam line break. Loss of offsite power results in reactor coolant pump coastdown 3 seconds following the loss of offsite power. C26

F) Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are assumed to occur in the sector with the stuck RCCA. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during the return to power phase following the steam line break. This void in conjunction with the large negative moderator temperature coefficient partially offsets the effect of the stuck RCCA. The power peaking factors depend upon the core power, temperature, pressure, and flow. C26

Both cases above assume initial hot shutdown conditions at time zero. 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 release 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. C26

C26

G) In computing the steam flow during a steam line break, the Moody curve (Reference 2) for fL/D = 0 is used.

H) Feedwater addition aggravates cooldown accidents like the steam line rupture. Therefore, the maximum feedwater flow is assumed. All the main and auxiliary feedwater pumps are assumed to be operating at full capacity when the rupture occurs, even though the plant is assumed to be in a hot standby condition. The maximum auxiliary feedwater flow to the faulted loop is assumed to be 1628.5 gpm. These main feedwater and auxiliary feedwater flow assumptions conservatively maximize the plant cooldown response to a main steam line rupture.

C26

As a result of the Safety Injection signal actuation, the main feedwater pumps trip, the feedwater control valves (FCVs) close, the backup feedwater isolation valves start to close (30 second closure), and the main feedwater pump discharge valves start to close (90 second closure). In the analysis, the FCV in the faulted loop is assumed to fail open, such that the faulted steam generator continues to be fed by the condensate pumps (which do not trip on SI signal actuation) until the backup feedwater isolation valves close. A conservatively high flow rate to the depressurizing steam generator is assumed prior to isolation. C26 C26

For the loss of offsite power case, the condensate pumps would coast down upon losing offsite power, thus ramping the main feedwater flow to zero. However, this coastdown was conservatively not modeled, which maximizes the continued addition of feedwater to the steam generators. Therefore, consistent with the offsite power available case, feedwater isolation occurs in 9 seconds for the unfaulted loops (FCVs) and 30 seconds on the faulted loop (backup feedwater isolation valves). C26

I) The effect of the heat transferred from thick metal in the pressurizer and reactor vessel upper head is not included in the cases analyzed. Studies previously performed have shown that the heat transferred to the coolant from these latent sources is a net benefit in DNB and RCS energy when the effect of the extra heat on reactivity and peak power is considered.

<u>Results</u>

The calculated sequence of events for the cases analyzed is shown in Tables 14.2.5-2 and 14.2.5-3.

The results presented are a conservative indication of the events which would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 14.2.5-5 through 14.2.5-8 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) downstream of the MSIVA at initial no-load condition (Case A). Offsite power is assumed to be available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator after steamline isolation. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by high steam flow coincident with low steam line pressure or low T-avg will trip the reactor. Steam release from more than one steam generator will be prevented by automatic closure of the MSIVs in the steam lines, by high containment pressure signals, or by high steam flow coincident with low steam line pressure. (For a break upstream of the MSIVA, MSIV closure is not required due to the presence of the MSCVs, which prevent blowdown of the unfaulted steam generators. In this case, SI actuation would occur immediately from high differential steam pressure between the faulted steam line and the main steam header. The results would be less severe than those for the cases presented.) C26

As shown in Figure 14.2.5-8, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 2400 ppm enters the RCS. A peak core power well below the nominal full power value is attained. C26

Figures 14.2.5-9 through 14.2.5-12 show the response of the salient parameters for Case B, which corresponds to the case discussed above with additional loss of offsite power at event initiation. The SI system delay time includes 22 seconds to start the diesel generator and load the necessary equipment and 25.1 seconds from the initiation of the event, to start the SI pumps and to open the valves. Criticality is achieved later and the peak core power is much less than in Case A. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. Table 14.2.5-3 provides the specific sequence of events. C26

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 transients is over. In the case of loss of offsite power, this heat is removed to the atmosphere via the steam line safety valves.

Steam System Piping Failure at Full Power

To ensure safe shutdown during Mode 1 operation, the Steam System piping Failure at Full Power event was analyzed at hot full power conditions. For this analysis, initial conditions of core power and pressurizer pressure were assumed to be at their nominal values consistent with steady-state full power operation. Reactor vessel average coolant temperature was assumed to be at its nominal, steady-state, full-power value. Uncertainties in the initial conditions of these parameters as well as a small temperature bias are considered in the DNBR limit rather than explicitly modeled in the transient calculations, consistent with the application of the Revised Thermal Design Procedure (RTDP) methodology. Steam generator water level was assumed to be at its nominal value. Minimum measured reactor coolant flow was modeled according to the RTDP methodology. Zero steam generator tube plugging was assumed to maximize the primary-to-secondary heat transfer, which results in a more severe RCS cooldown transient. C26

The overpower ΔT (OPΔT) and Low Steam Generator Pressure coincident with High Steam Flow – Safety Injection protection functions are relied upon to provide the necessary protection to mitigate the event. The most limiting full power case is typically the largest break that produces a reactor trip on OPΔT. Larger breaks result in a rapid reactor trip as a result of the Low Steam Generator Pressure coincident with High Steam Flow – Safety Injection signal, which terminates the transient before core power increases significantly and therefore results in less limiting conditions. Since PTN has steam exit nozzle flow restrictors which limit the flow area to about 1.388 ft^2, the analysis modeled a spectrum of break sizes up to 1.4 ft^2. The analysis demonstrates that the most limiting break size is 0.65 ft^2 with the reactor trip on OPΔT.

The results of the full-power steam line rupture analysis demonstrate that the DNB design basis is met. In addition, the peak linear heat generation rate (expressed in kW/ft) does not exceed the fuel centerline melt limit. Since this event results in a decrease in both the primary and secondary side pressures, the maximum RCS and Main Steam System pressure criteria are not challenged.

Conclusions

The analysis has shown that the criteria stated earlier are satisfied. Although DNB and possible clad perforation and fuel centerline melting (expressed in kW/ft) following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that the minimum DNBR remains above the limit value and the maximum kW/ft value remains below the kW/ft limit value for any rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

14.2.5.3 CONTAINMENT PRESSURE RESPONSE TO STEAMLINE BREAK

Analyses have been performed for the Main Steam Line Break (MSLB) containment response considering a spectrum of break sizes, power levels, and different single failures. These analyses are described in Section 14.3.4 in detail.

14.2.5.4 DOSE EVALUATION C28

The main steam line break (MSLB) dose consequence analysis for Extended Power Uprate (EPU) conditions is consistent with the guidance provided in Appendix E of Reference 6, "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 Turkey Point MSLB event.

2. Regulatory Position 2 – No fuel damage is postulated to occur for the Turkey Point MSLB event. Therefore, 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 TS 3.4.8 Figure 3.4-1, which is a value of 60.0 μCi/gm DE I-131. 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 proposed TS 3.4.8 value of 0.25 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 equal to the value specified by proposed TS 6.8.4.j.b.2, which is 0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions.

8. Regulatory Position 5.2 - The density used in converting primary-to-secondary volumetric leak rates to mass leak rates is 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions. C26

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. This is conservatively calculated to occur at 125.4 hours. The release of radioactivity from the unaffected steam generators is conservatively assumed to continue until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 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 steam generator, 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, a portion of the leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary coolant immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

12. Regulatory Position 5.5.2 - Any postulated leakage that immediately flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the bulk water is credited.

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 steam generators is limited by the moisture carryover. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the steam generator carryover rate of less than 1%. No reduction in the release is assumed from the faulted steam generator. C26

15. Regulatory Position 5.6 - Steam generator tube bundle uncovery in the intact steam generators is postulated for up to 30 minutes following a reactor trip. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

Other assumptions and modeling inputs are described below:

1. 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 primary-to-secondary leakage or from the safety injection system.

2. 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.

3. The steam mass release rates for the intact steam generators are provided in Table 14.2.5-6.

4. Data used to calculate the iodine equilibrium appearance rate are provided in Table 14.2.5-7.

5. Radionuclide concentrations in the secondary side fluid of the steam generators assume that Auxiliary Feedwater is provided to maintain a constant secondary mass during periods of steam release.

6. Releases from the faulted main steam line are postulated to occur from the main steam line associated with the most limiting atmospheric dispersion factors. Releases from the unaffected steam generators are postulated to occur from the MSSV or ADV with the most limiting atmospheric dispersion factors.

The steam generator partition factor for iodine and particulates is applied in the analysis by reducing the steam release rate from the steam generator compartment. This methodology conservatively allows the activity, which is not released, to remain in the steam generator compartment and contribute to the radionuclide concentration.

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The major assumptions and parameters used in the analysis are itemized in Table 14.2.5-5. The dose conversion factors, breathing rates, and control room occupancy factors are the same as those given in the LOCA dose analysis in Table 14.3.5-5. The atmospheric dispersion factors used in the dose calculations are given in Appendices 2E and 2F.

The analysis assumes that activity is released as reactor coolant enters the steam generators due to primary-to-secondary leakage. The equilibrium, pre-event source term for this RCS activity is presented in Table 14.3.5-8. All noble gases associated with this leakage are assumed to be released directly to the environment. Primary-to-secondary leakage into the faulted steam generator is also assumed to directly enter the atmosphere. Leakage into the unaffected steam generators is partitioned by the secondary fluid and released via the MSSVs and ADVs. All primary-to-secondary leakage is assumed to continue until the primary system is cooled to 212°F at 125.4 hours. The release of the initial iodine content of the steam generator secondary is also considered. The source term for this activity is the same as the SGTR event secondary source, and is presented in Table 14.2.4-4.

Fuel damage is not postulated for the MSLB event. Consistent with Appendix E of reference 6, 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 Specifications. The iodine activities for the pre-accident spike case are the same as the SGTR event, and are presented in Table 14.2.5-7.

For the case of the accident-induced spike, the postulated MSLB event induces an iodine spike. The RCS activity is initially assumed to be the Technical Specification steady-state maximum allowable 0.25 μCi/gm DE I-131. 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. With iodine activity at equilibrium, the iodine release rate is equal to the rate at which iodine is lost due to decay, purification, and primary system leakage. Parameters used in the determination of the iodine equilibrium release rate are the same as the SGTR event and are provided in Table 14.2.5-7. The resulting iodine activities for the MSLB accident-induced (concurrent) iodine spike case are presented in Table 14.2.5-7. C26

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 flow distribution during this mode is 1000 cfm of unfiltered fresh air plus 100 cfm of unfiltered inleakage.

- Control Room is isolated following receipt of a safety injection signal. A 41.5-second delay is applied to account for the signal processing and damper closure time. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 100 cfm of unfiltered inleakage, and 375 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 and 95% for elemental and organic iodine

The atmospheric dispersion factors (*X/Qs*) used 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 Appendix 2F. Releases from the intact steam generators are assumed to occur from the MSSV/ADV which produces the most limiting *X/Q*. Releases from the faulted steam generator are assumed to occur from the location on a steam line closest the in-service intake. C26

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

The offsite and control room dose limits for a steamline break are provided in Regulatory Guide 1.183 (Reference 6).

The offsite and control room doses due to the steamline break are given in Table 14.2.5-8. The offsite doses due to the steamline break are within the acceptance criteria.

14.2.5.5 REFERENCES

1. WCAP-14882-P-A (Proprietary), “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” Huegel, D. S., et al., April 1999. C26

2. Moody, F. J., "Transaction of the ASME, Journal of Heat Transfer," page 134, February 1965.

3. Letter, L-81-211, R. E. Uhrig (FPL) to D. G. Eisenhut (NRC), "NRC IE Bulletin 80-04," May 19, 1981.

4. Westinghouse letter to FPL, FPL-91-651, dated October 21, 1991, Safety Evaluation for Diesel Loading Scheme - Revision 3. Refer to FPL Safety Evaluation JPN-PTN-SEMJ-91-035, Revision 0; transmitted by letter JPN-PTN-91-0784, November 5, 1991.

5. Westinghouse WCAP-14276,"Florida Power and Light Company Turkey Point Units 3 and 4 Uprating Licensing Report," Revision 1, December 1995.

6. USNRC, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Desing Basis Accidents at Nuclear Power Plants,” July 2000.

7. WCAP-14565-P-A, “VIPRE-01 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999. C26

TABLE 14.2.5-1

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TABLE 14.2.5-2

TIME SEQUENCE OF EVENTS
CORE RESPONSE ANALYSIS

Case A - Steam System Piping Failure, With Offsite Power Available

Time (sec)	Event	
t = 0.	A. Reactor at hot zero power. All control rods inserted except most reactive RCCA. Shutdown Margin = 1.77% delta-k/k. Double ended guillotine break located downstream of the Main Steam Isolation Valve Assembly.	
t = 1.1	B. First SIS setpoint is reached - High Steam Flow coincident with Low Steam Pressure.	C26
t = 4.1	C. SIS actuation signal is generated.	C26
t = 13.1	D. Main feedwater flow to unfaulted steam generators terminated by FCV closure 9 seconds after SI actuation signal.	C26
t = 20.0	E. MSIVs are closed.	
t = 25.1	F. Two SI pumps at rated speed 21 seconds after SIS actuation signal is generated.	
t = 35.1	G. Main feedwater flow to faulted steam generator terminated by closure of backup isolation valve 30 seconds after SI actuation signal.	C26
t = 60.0	H. Reactor becomes critical.	
t = 168.3	I. Power reaches maximum level.	

TABLE 14.2.5-3

TIME SEQUENCE OF EVENTS
CORE RESPONSE ANALYSIS

Case B - Steam System Piping Failure, Without Offsite Power Available

Time (sec)	Event
t = 0.	A. Reactor at hot zero power. All control rods inserted except most reactive RCCA. Shutdown Margin = 1.77% delta-k/k. Double ended guillotine break located downstream of the Main Steam Isolation Valve Assembly. Offsite power lost.
t = 1.1	B. First SIS setpoint is reached - High Steam Flow coincident with Low Steam Pressure.
t = 3.0	C. Reactor Coolant Pumps lose power, begin to coast down.
t = 4.1	D. SIS actuation signal is generated.
t = 13.1	E. Main feedwater flow to unfaulted steam generators terminated by FCV closure 9 seconds after SI actuation signal.
t = 20.0	F. MSIVs are closed.
t = 35.1	G. Main feedwater flow to faulted steam generator terminated by closure of backup isolation valve 30 seconds after SI actuation signal.
t = 47.1	H. Two SI pumps at rated speed 43 seconds after SIS actuation signal is generated.
t = 85.8	I. Reactor becomes critical.
t = 433.5	J. Power reaches maximum level.

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TABLE 14.2.5-4

Table deleted in Revision 15

TABLE 14.2.5-5

ASSUMPTIONS USED FOR
STEAM LINE BREAK DOSE ANALYSIS

Input/Assumption	Value
Release Inputs:	
Core Power Level	2652 MW_{th}
Initial RCS Equilibrium Activity	0.25 μCi/gm DE I-131, and 447.7 μCi/gm DE Xe-133 (Table 14.3.5-8)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 14.2.4-4)
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Iodine Spike Appearance Rate	500 times
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	0.2 gpm/SG
Time to establish shutdown cooling and terminate steam release	63 hours
Time for RCS to reach 212°F and terminate SG tube leakage	125.4 hours
RCS Mass (minimum)	366,086 lbm
Removal Inputs:	
SG Secondary Side Mass	Faulted SG – 131,516.5 lbm Intact SGs - 67,707 lbm per SG
Release from Faulted SG	Instantaneous
Time to re-cover Intact SG Tubes	30 minutes
Tube Uncovery Flashing Fraction	11%
Steam Generator Secondary Side Partition Coefficients	Faulted SG – none Intact SGs – 100
Transport Inputs:	
Release Locations(s)	Various – See Appendix 2F
Atmospheric Dispersion Factors Offsite Onsite	 Appendix 2E Appendix 2F
Isolation Signal Time of CR Isolation Unfiltered Inleakage	Safety Injection 41.5 seconds 100 cfm
Breathing Rates Offsite Control Room	 Reg. Guide 1.183, Section 4.1.3 Reg. Guide 1.183, Section 4.2.6
Control Room Occupancy Factors	Reg. Guide 1.183, Section 4.2.6

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TABLE 14.2.5-6

MSLB INTACT SGS STEAM RELEASE RATE*

Time (hours)	Intact SGs Steam Release Rate (lbm/min)
0.0	2622
2.0	2058
3.0	1931
4.0	1814
5.0	1694
8.0	1070
11.0	965
16.0	864
24.0	820
63.0	0.0

* Stored energy above RHR entry conditions is released between 2 and 8 hours

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Table 14.2.5-7

MSLB IODINE SPIKE AND APPEARANCE RATE MODELING INPUTS AND RESULTS

Pre-Accident (60 µCi/gm D.E. I-131) Iodine Spike RCS Activities

Isotope	Activity (µCi/gm)
Iodine-131	48.1440
Iodine-132	34.1280
Iodine-133	58.3440
Iodine-134	6.3192
Iodine-135	28.8000

Iodine Equilibrium RCS Appearance Rate Analysis Assumptions

Input Assumption	Value
Letdown Flow	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	397,544 lbm
I-131 Decay Constant	6.000E-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}

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Concurrent (500 x) Iodine Spike Appearance Rate In RCS

Isotope	Appearance Rate (Ci/min)	8-hour Production (Ci)
Iodine-131	53.90	25870
Iodine-132	101.83	48881
Iodine-133	76.17	36564
Iodine-134	38.22	18343
Iodine-135	50.50	24241

TABLE 14.2.5-8

STEAM LINE BREAK OFFSITE DOSES

Case	EAB Dose[1] (Rem TEDE)	LPZ Dose[2] (rem TEDE)	Control Room Dose[2] (rem TEDE)
MSLB pre-accident iodine spike	0.023	0.018	1.59
Acceptance Criteria (pre-accident iodine spike)	25[3]	25[3]	5[4]
MSLB concurrent iodine spike	0.037	0.032	1.60
Acceptance Criteria (concurrent iodine spike)	2.5[3]	2.5[3]	5[4]

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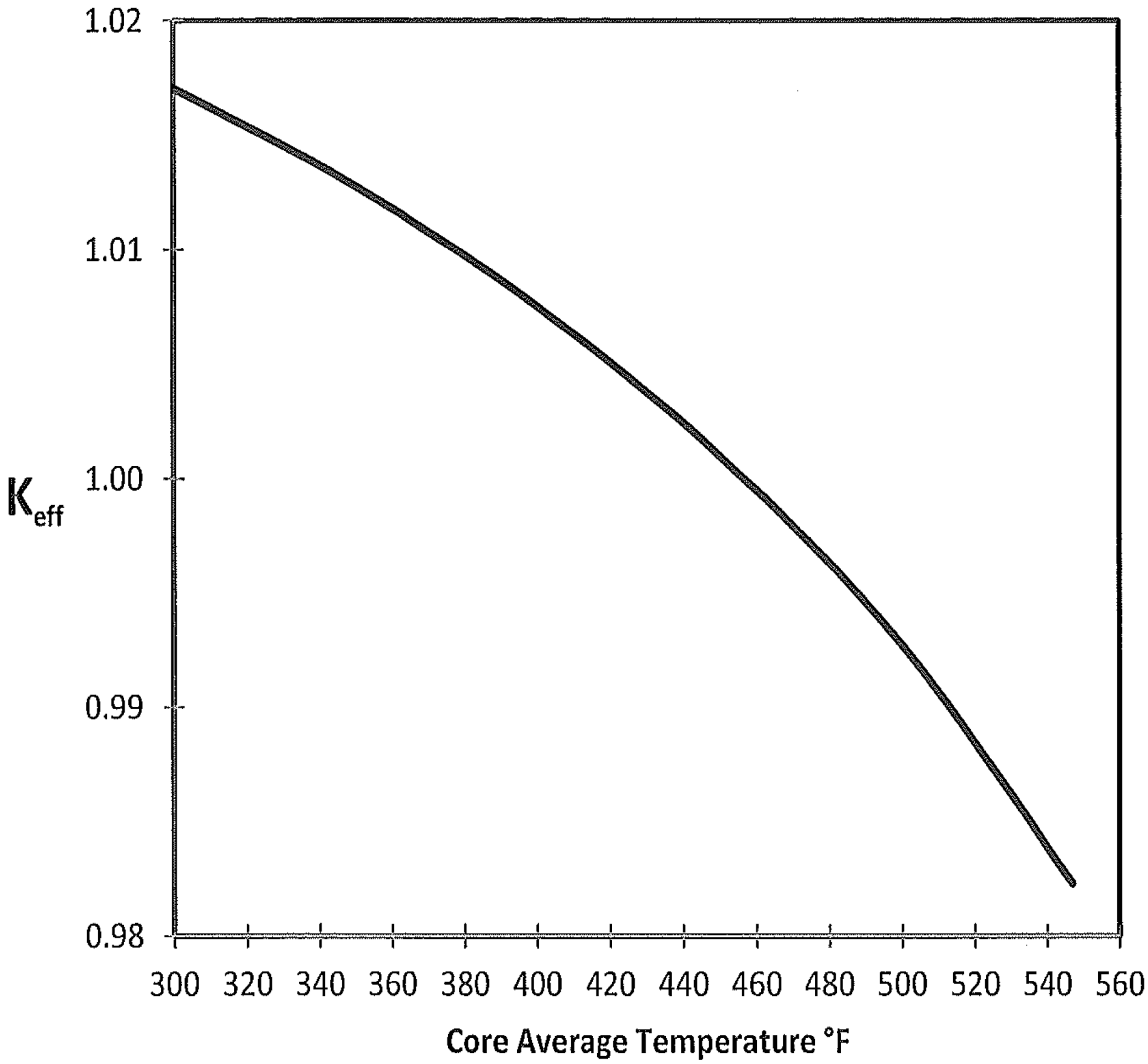
Notes:

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) Regulatory Guide 1.183, Table 6
(4) 10CFR50.67



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K_{eff} VS TEMPERATURE

FIGURE 14.2.5-1

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FAILURE OF STEAM GENERATOR
SAFETY OR RELIEF VALVE

FIGURE 14.2.5-2

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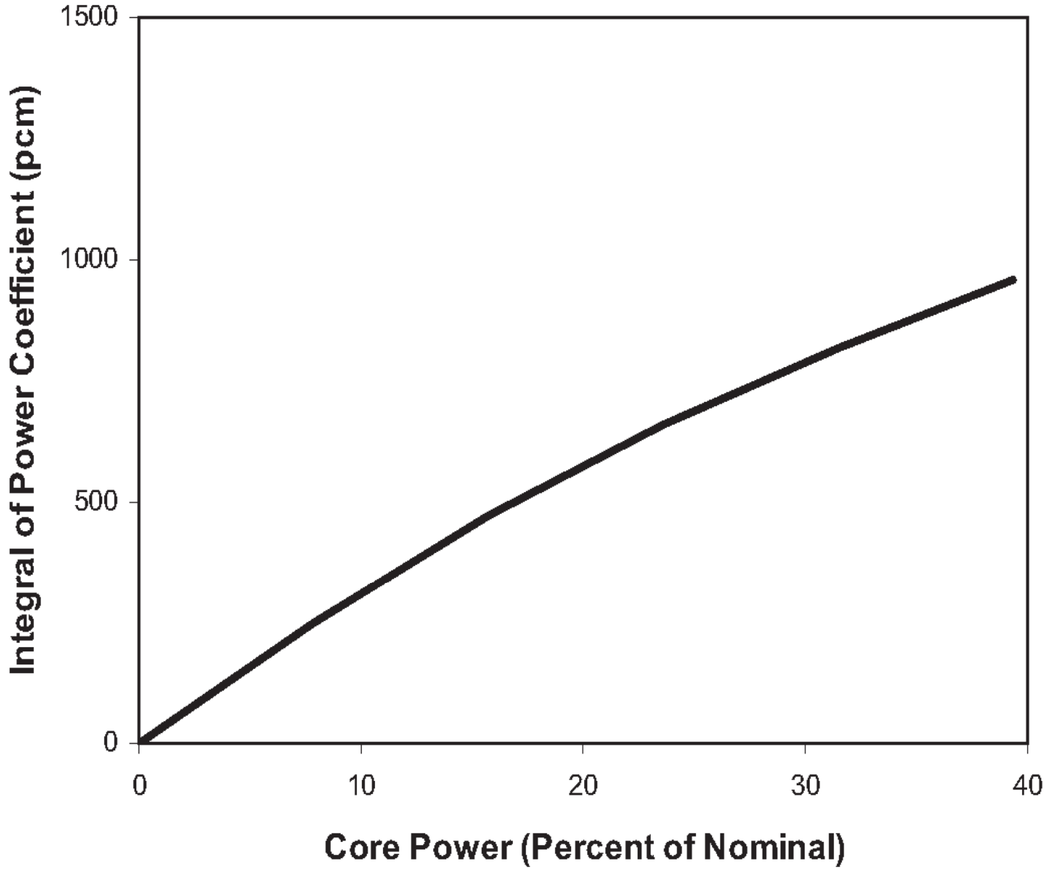
1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE
FIGURE 14.2.5-3



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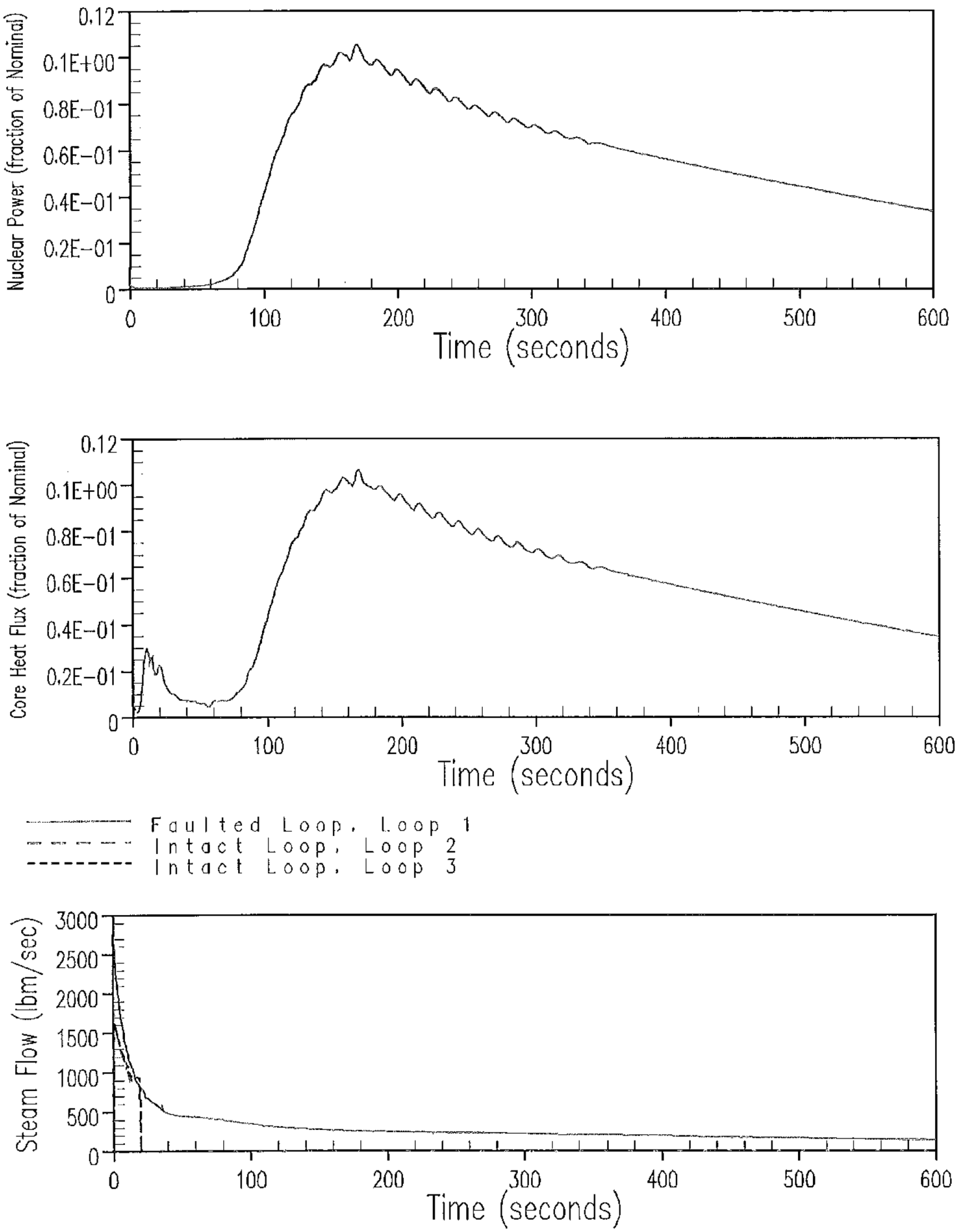
DOPPLER POWER FEEDBACK

FIGURE 14.2.5-4



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1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE
FIGURE 14.2.5-5

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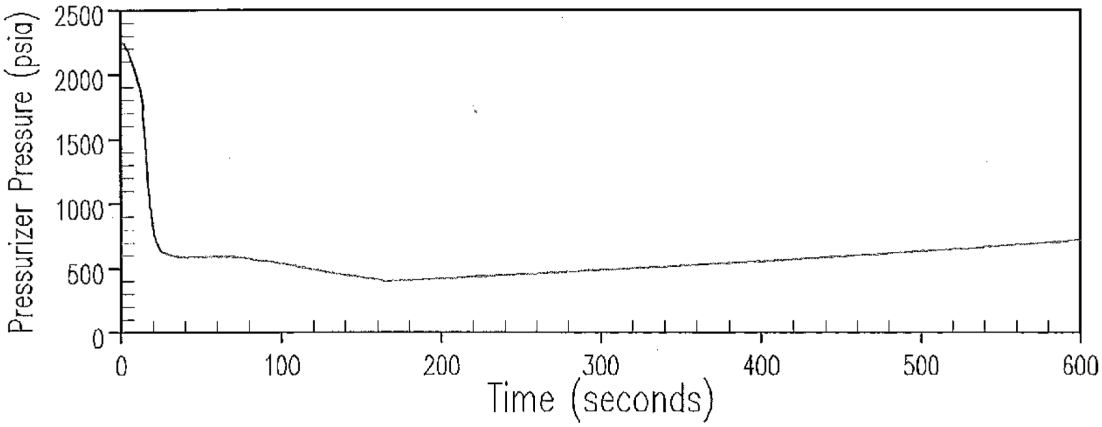

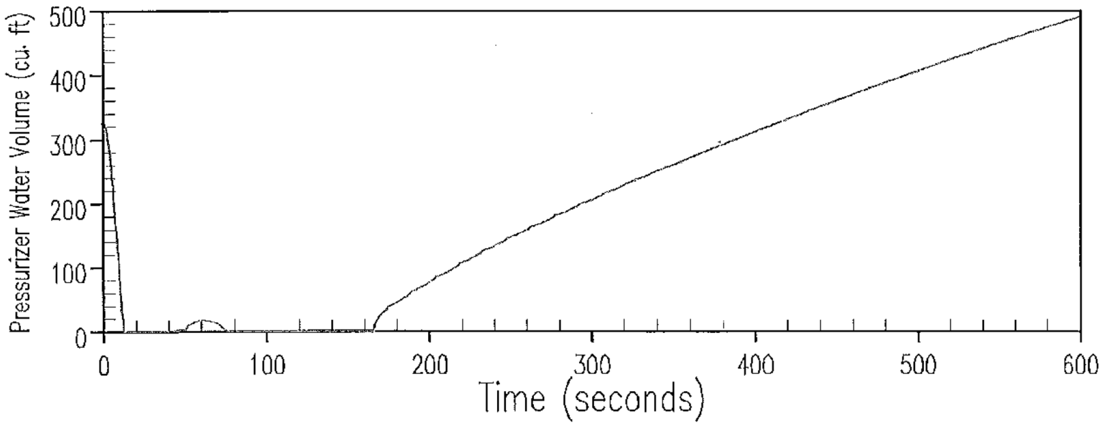

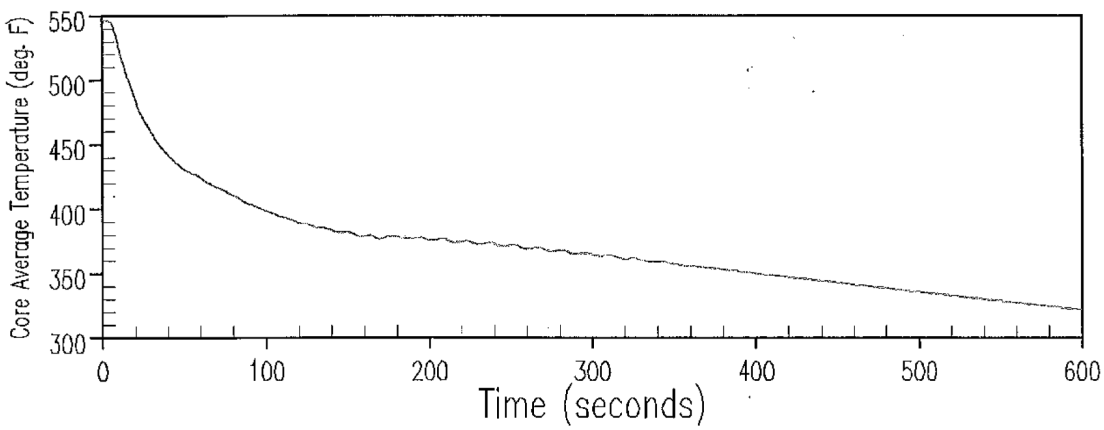

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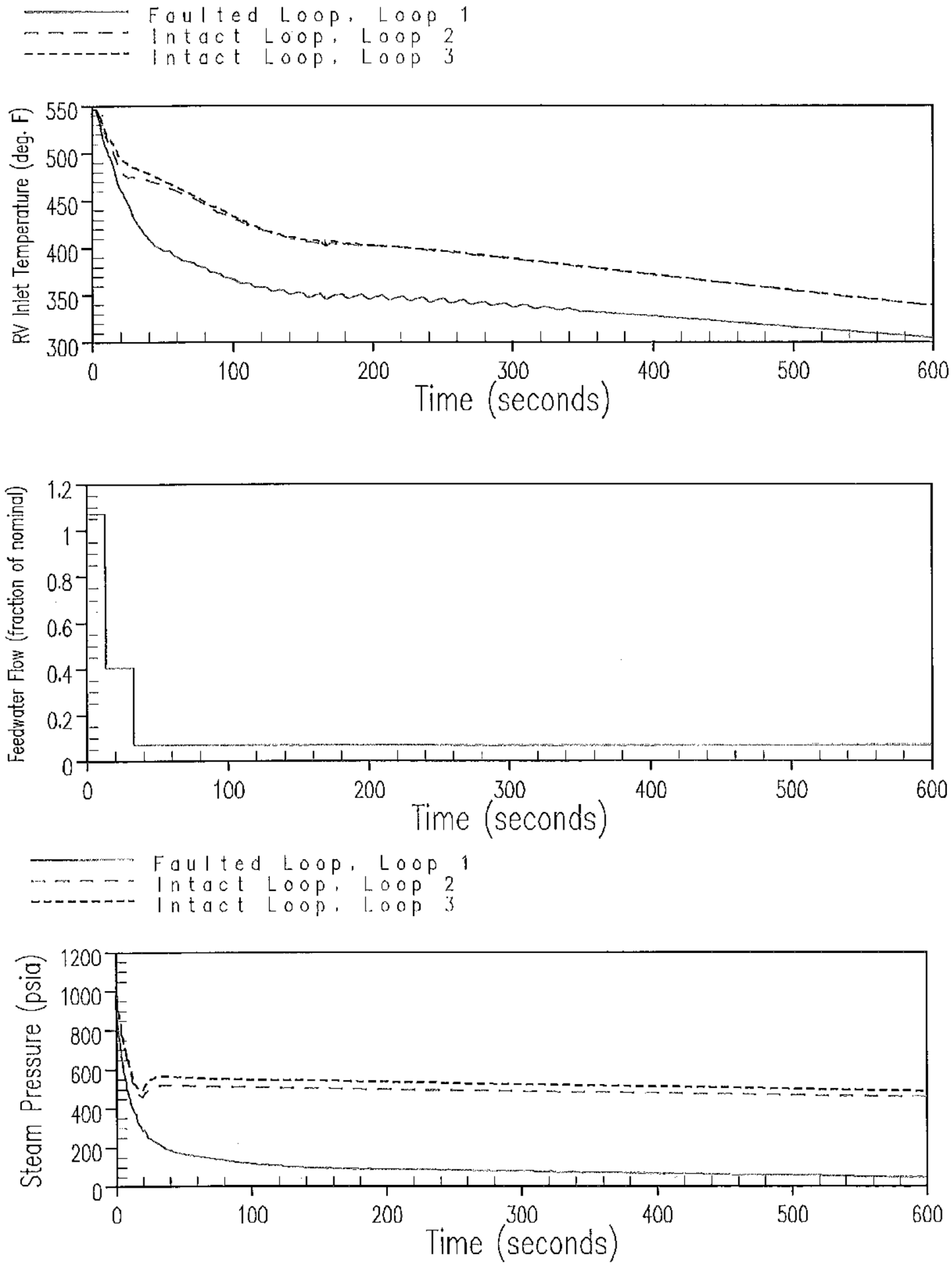
1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE

FIGURE 14.2.5-6



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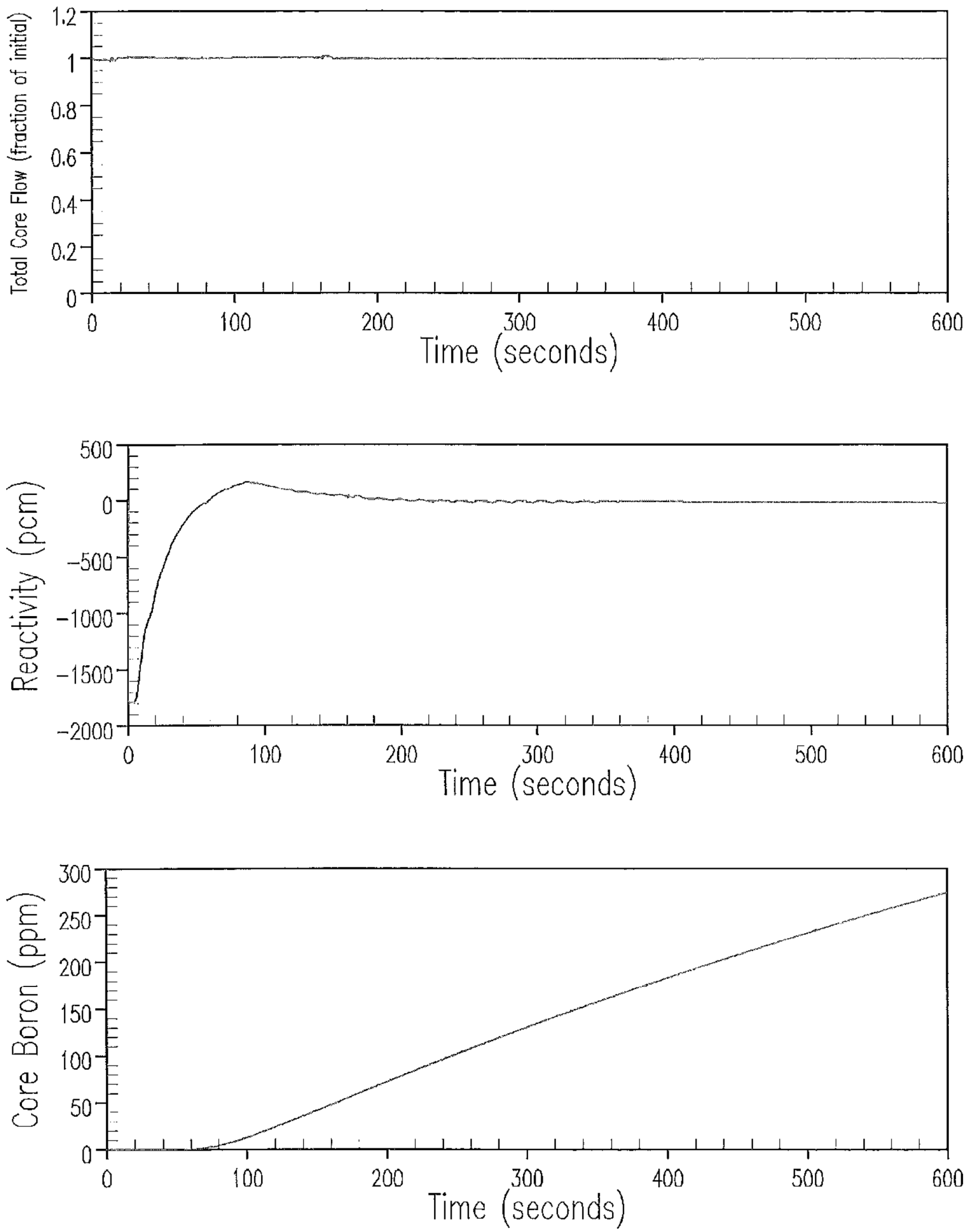
1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE

FIGURE 14.2.5-7



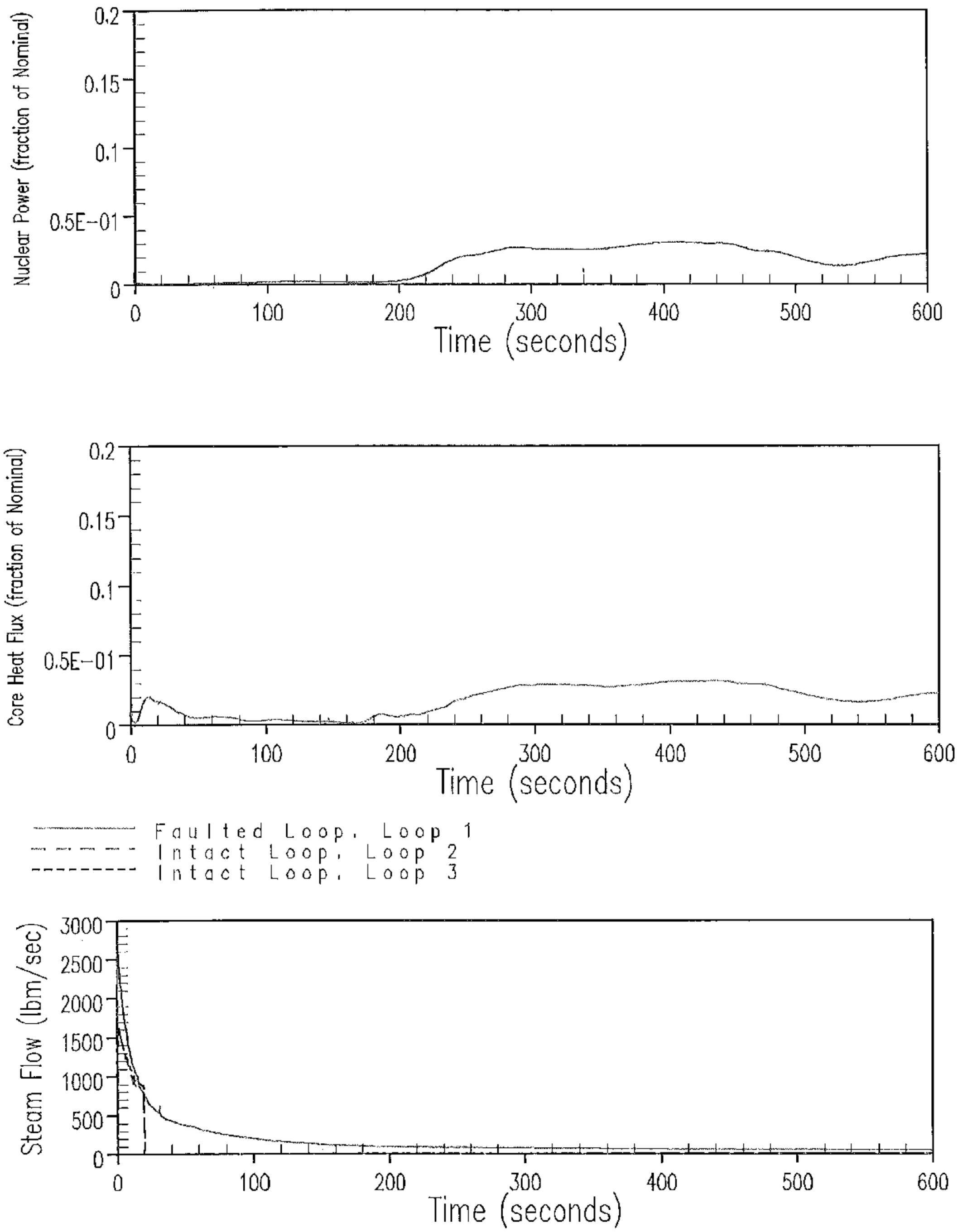
C26

REVISED 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER AVAILABLE

FIGURE 14.2.5-8



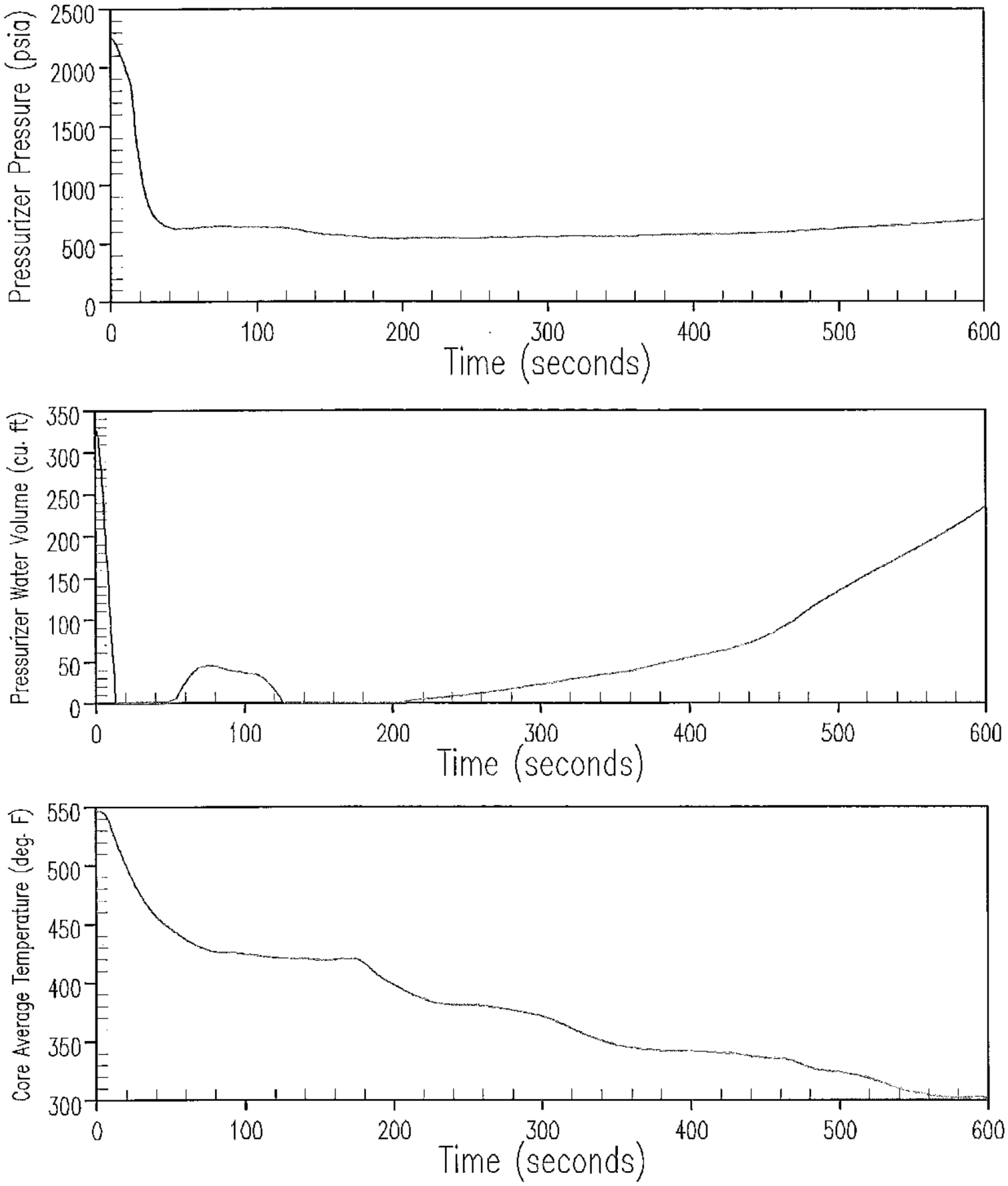
C26

REVISED 04/17/20013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE

FIGURE 14.2.5-9



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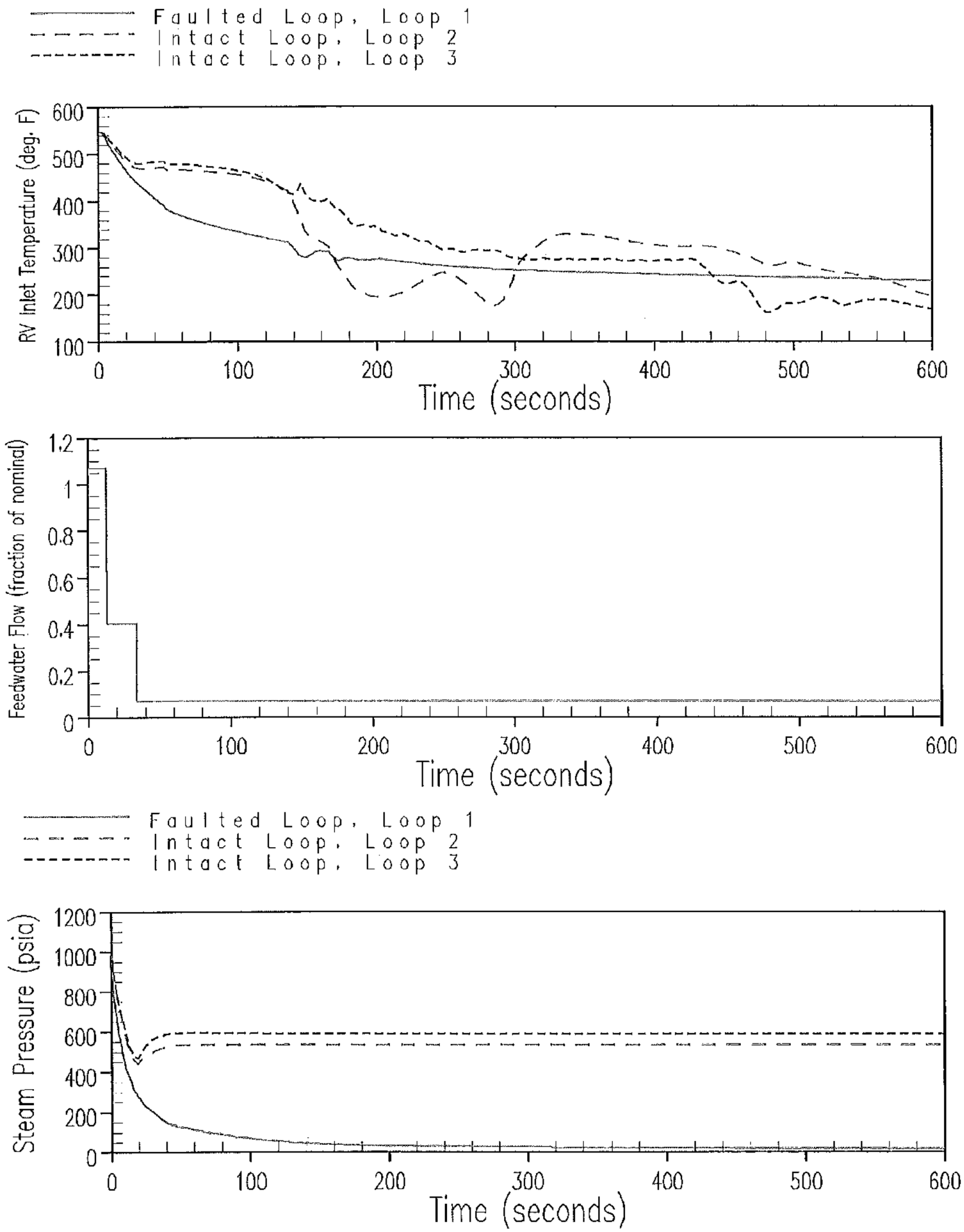
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE

FIGURE 14.2.5-10



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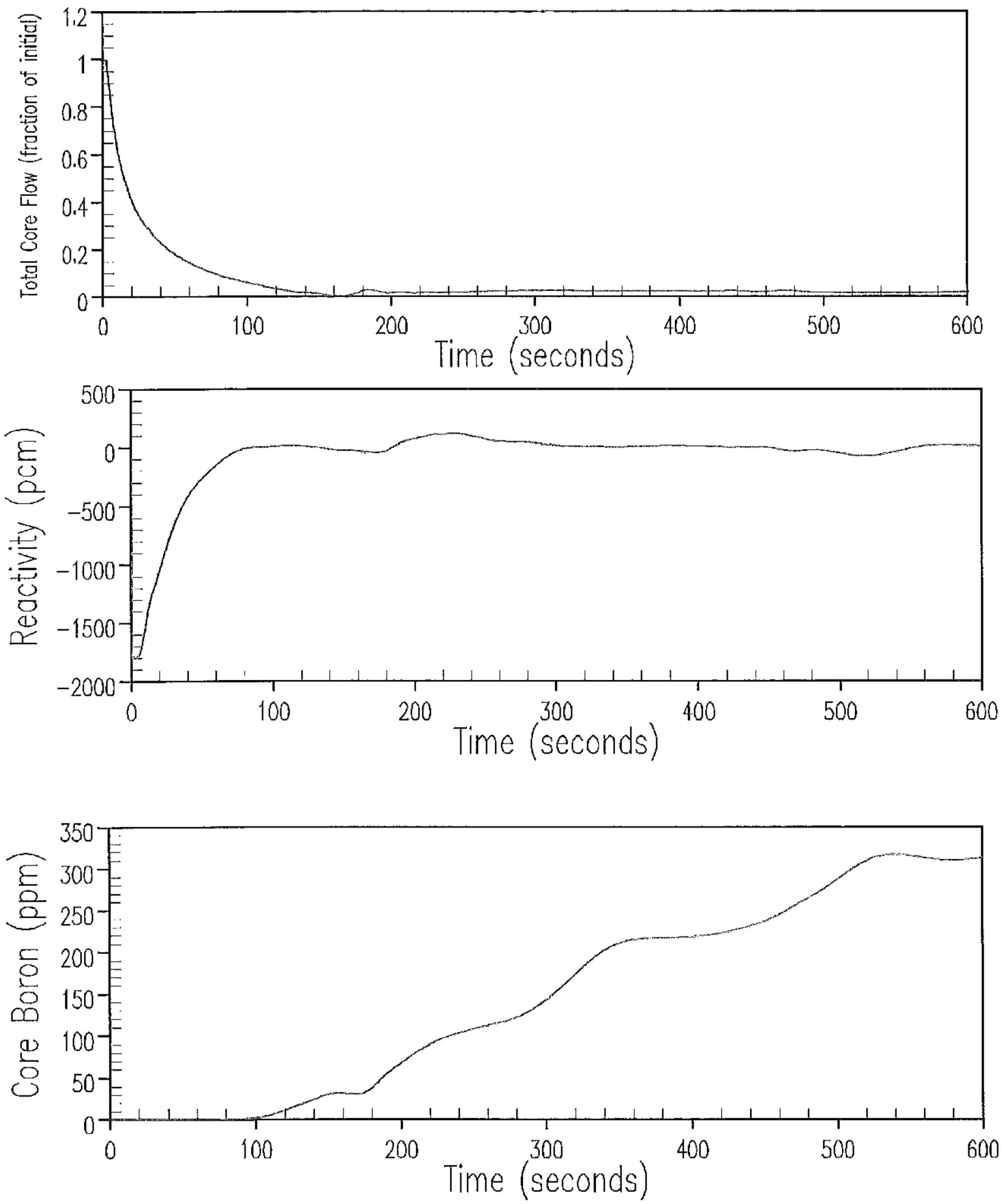
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE

FIGURE 14.2.5-11



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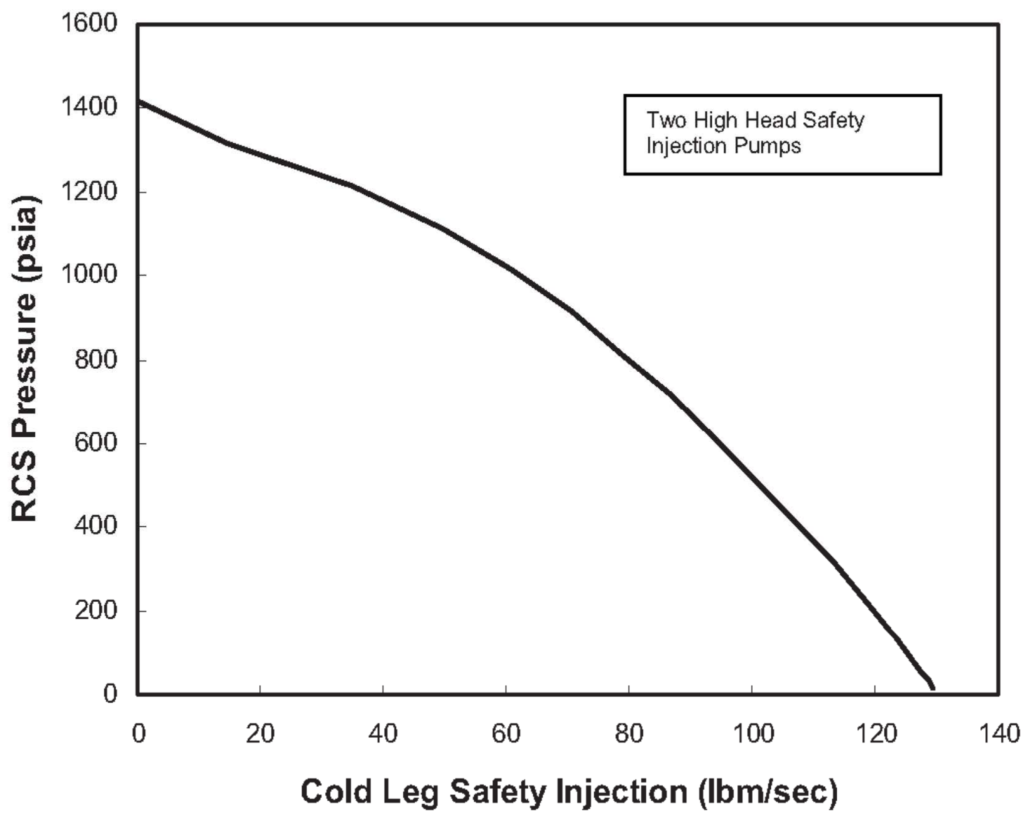
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

1.4 FT^2 STEAM LINE RUPTURE
OFFSITE POWER NOT AVAILABLE

FIGURE 14.2.5-12



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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

STEAM LINE BREAK
SAFETY INJECTION FLOW

FIGURE 14.2.5-13

[DELETED]

REV. 16 (10/99)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

MAIN STEAM LINE BREAK
0% POWER 1.4/2.8 FT^2 MSCV FAILURE
WITH OFFSITE POWER

FIGURE 14.2.5-14

[DELETED]

REV. 16 (10/99)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

MAIN STEAM LINE BREAK
0% POWER 1.4/2.8 FT^2 MSCV FAILURE
WITH OFFSITE POWER

FIGURE 14.2.5-15

[DELETED]

REV. 16 (10/99)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

MAIN STEAM LINE BREAK
0% POWER 1.4/2.8 FT^2 MSCV FAILURE
WITH OFFSITE POWER

FIGURE 14.2.5-16

This figure deleted in Revision 15

REV. 15 (4/98)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

MAIN STEAM LINE BREAK
CONTAINMENT PRESSURE VS TIME
(FINAL CASE)
FIGURE 14.2.5-17

14.2.6 RUPTURE OF A CONTROL ROD MECHANISM HOUSING-RCCA EJECTION

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

a) Each control rod drive mechanism housing is completely assembled and shop-tested at 3450 psig.

b) The mechanism housings are individually tested per ASME code requirements after they are installed on the reactor vessel head to the head adapters, and checked during the pressure test of the completed Reactor Coolant System.

c) Stress levels in the mechanism are not affected by anticipated system transients at power, or by thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

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 that 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 the gross failure of the housing will not occur. The joints between the latch mechanism and the head adapter and between the latch mechanism and the rod travel housing are threaded joints, reinforced using canopy type seal welds.

The operation of a chemical shim plant is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, there are only a few rods in the core at full power. Proper positioning of these rods is monitored by a control room alarm system. There are low and low-low RCCA insertion limit alarms. Operating instructions require boration at the low level alarm and emergency boration at the low-low level alarm. The control rod position monitoring and alarm systems are described in detail in Section 7.3 and in Reference 1.

Due to the extremely low probability of a rod cluster control assembly 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 2). Extensive tests of UO_2 zirconium-clad fuel rods representative of those present 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 exhibited failure as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 3) results which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreased ~10 percent with fuel burnup. The clad failure mechanism appears to be melting for unirradiated (zero burnup) rods and brittle fracture for irradiated rods. The conversion ratio of thermal to mechanical energy is also important. 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.

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

The real physical limits of this accident are that the rod ejection event and any consequential damage to either the core or the Reactor Coolant System must not prevent long-term core cooling, and any offsite dose consequences must be within the guidelines of 10 CFR 50.67 (post-EPU, with Alternative Source Term implementation). More specific and restrictive criteria are applied to ensure fuel dispersal in the coolant, gross lattice distortion or severe shock waves will not occur. In view of the above experimental results, the conclusion of WCAP-7588 Rev I-A (Reference 1), and Reference 4, the limiting criteria are: C26

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A. Average fuel pellet enthalpy at the hot spot must be maintained below 225 cal/gm for unirradiated and 200 cal/gm for irradiated fuel,

B. Peak reactor coolant pressure must be less than that which could cause RCS stresses to exceed the faulted-condition stress limits,

C. Fuel melting is 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 A.

14.2.6.1 METHOD OF ANALYSIS

This section describes the models used and the results obtained. Only the initial few seconds of the power transient are discussed, since the long term considerations are the same as for a loss of coolant accident.

The calculations of the RCCA ejection transient are 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 1. C26

<u>Average Core</u>

The spatial kinetics computer code, TWINKLE (Reference 5), 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. C26

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal heat flux 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 average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors. The hot spot analysis is performed using the detailed fuel and clad transient computer code, FACTRAN (Reference 6). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (see Reference 7) 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.

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 14.2.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

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

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

Power distribution before and after ejection for a "worst case" can be found in Reference 1. During plant startup physics testing, ejected rod worths and power distributions have been 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.

Delayed Neutron Fraction, ß

Calculations of the effective delayed neutron fraction ($ß_{eff}$) typically yield values no less than 0.70 percent at beginning-of-life and 0.50 percent at end-of-life. The ejected rod accident (in the zero power transients) is sensitive to $ß_{eff}$ if the ejected rod worth is equal to or greater than $ß_{eff}$. In order to allow for future cycles, conservative estimates of $ß_{eff}$ of 0.55 percent at beginning of cycle and 0.44 percent at end of cycle are used in the analysis. C26

Reactivity Weighting Factor

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, thus 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.

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 resulting moderator temperature coefficient is at least +7 pcm/°F at the appropriate zero or full power nominal average temperature for the beginning-of-life cases.

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.

Heat Transfer Data

The FACTRAN (Reference 6) code used to determine the hot spot transient uses curves of thermal conductivity versus fuel temperature. During a transient, the peak centerline fuel temperature is independent of the gap conductances during the transient. The cladding temperature is however strongly dependent on the gap conductance and is highest for high gap conductances. For conservatism a high gap heat transfer coefficient value of 10,000 Btu/hr-ft^2-F has been used during transients. This value corresponds to a negligible gap resistance and a further increase would have essentially no effect on the rate of heat transfer. C26

Coolant Mass Flow Rates

When the core is operating at full power, all three reactor coolant pumps will always be operating. However, for zero power conditions, the system may be operating with two pumps. The principal effect of operating at reduced flow is to reduce the film boiling heat transfer coefficient. This results in higher peak cladding temperatures, but does not affect the peak centerline fuel temperature. Reduced flow also lowers the critical heat flux. However, since DNB is always assumed at the hot spot, and since the heat flux rises very rapidly during the transient, this produces only second order changes in the cladding and centerline fuel temperatures. All zero power analyses for both average core and the hot spot have been conducted assuming two pumps in operation.

Trip Reactivity Insertion

The rods were assumed to be released 0.5 seconds after reaching the power range high neutron flux trip setpoint. The delay is constituted of 0.2 seconds for the instrumentation to produce a signal, 0.15 seconds for the trip breaker to open and 0.15 seconds for coil release. In calculating the shape of the insertion versus time curve all the rods are assumed to be dropped as a single bank from the fully withdrawn position. This means that the initial movement is through the low worth region at the extreme top of the core, which results in a conservatively slow reactivity insertion versus time curve.

Fuel Densification Effects

Fuel densification effects on rod ejection are accounted for according to the methods described in Reference 8.

Lattice Deformations

A large temperature gradient exists in the region of the hot spot. Since the fuel rods are free to move in a vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced.

Boiling in the hot spot region will produce a net fluid flow away from that region. However, the fuel heat is released to the water 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 is 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 addressed in the following analyses.

Thermal Conductivity Degradation (TCD)

As described in Section 14.0, TCD effects on rod ejection are explicitly modeled in the analysis. C26

Results

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

A. 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 0.33 percent ΔK and 5.48, respectively. The peak hot spot average fuel pellet enthalpy was 178.3 cal/gm. The peak clad average temperature was 2216°F and the peak fuel centerline temperature was 4990°F. However, fuel melting was within the limiting criterion of 10 percent of the pellet volume at the hot spot. C26

B. 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 0.71 percent ΔK and a hot channel factor of 8.0. The peak hot spot average fuel pellet enthalpy was 87.6 cal/gm. The peak clad average temperature reached 1602°F; the fuel centerline temperature was 2562°F. C26

C. 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 0.30 percent ΔK and 5.52, respectively. The peak hot spot average fuel pellet enthalpy was 170.2 cal/gm. This resulted in a peak clad average temperature of 2070°F and the peak fuel centerline temperature was 4892°F. However, fuel melting was less than the limit of 10 percent of the pellet volume at the hot spot. C26

D. 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 0.84 percent ΔK and 14.3, respectively. The peak hot spot average fuel pellet enthalpy was 138.5 cal/gm. The peak clad average and fuel centerline temperatures were 2364°F and 3691°F, respectfully. C26

A summary of the cases presented above is given in Table 14.2.6-1. 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 cases) are presented in Figures 14.2.6-1 and 14.2.6-2, and a time sequence of events is given in Table 14.2.6-2. C26

14.2.6.2 FISSION PRODUCT RELEASE

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

With the implementation of Alternative Source Term Regulatory Guide 1.183 (Reference 10), conservative bounding fuel failure assumption values will be used in dose consequence analysis to bound any predicted actual fuel failures that may result from this event. C26

14.2.6.3 PRESSURE SURGE

A detailed calculation of the pressure surge for an ejected worth of one dollar at beginning-of-life, hot full power, indicates that the peak pressure does not exceed that which would cause reactor pressure vessel stress to exceed the faulted condition stress limits (Reference 1). 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.

14.2.6.4 DOSE EVALUATION

This event consists of the ejection of a single RCCA and is described as the Rod Ejection event in Appendix H of the Regulatory Guide 1.183 (Reference 10). In accordance with the guidance in Reference 10, two RCCA Ejection cases are considered. The first case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogenously 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. C26

The RCCA Ejection dose consequence analysis for Extended Power Uprate (EPU) conditions is consistent with the guidance provided in Appendix H of Reference 10, "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 Regulatory Guide 1.183, Regulatory Position 3.1 and is provided in Table 14.3.5-7. The inventory provided in Table 14.3.5-7 is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The release fractions provided in Regulatory Guide 1.183, Table 3, are adjusted to comply with the specific Regulatory Guide 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. C26

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 steam generators.

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. Containment sump pH is controlled to 7.0 or higher.

5. Regulatory Position 5 - The chemical form of radioiodine released from the steam generators 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. Containment sprays are not credited in the mitigation of this event.

7. Regulatory Position 6.2 - The containment is assumed to leak at the proposed Technical Specification maximum allowable rate of 0.20% of containment air weight for the first 24 hours and at 50% of this maximum allowable rate (0.10% of containment air weight) for the remainder of the event.

8. Regulatory Position 7.1 - The primary-to-secondary leak rate is equal to the Technical specification maximum allowable values which are 0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions. C26

9. Regulatory Position 7.2 -. The density used in converting primary-to-secondary volumetric leak rates to mass leak rates is 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

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 - 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 – A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary coolant immediately following plant trip when tube uncovery is postulated. The flashed leakage is assumed to be released to the environment with no mitigation. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

    - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the bulk water is credited.

- Appendix E, Regulatory Position 5.5.3 - All of the steam generator leakage flow that does not immediately 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 radionuclides in the steam generators is limited by the moisture carryover. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the steam generator moisture carryover rate of less than 1%.

- Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery in the unaffected steam generators is postulated for up to 30 minutes following a reactor trip. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator. C26

Other Assumptions and modeling inputs are described below:

1. This analysis assumed that the equilibrium specific activity on the secondary side of the steam generators is equal to the Technical Specification limit of 0.1 μCi/gm Dose Equivalent I-131.

2. The steam mass release rates for the secondary release are provided in Table 14.2.6-4.

3. It is assumed that 0.25% of the fuel is assumed to experience melting and 10% of the fuel is breached due to DNB.

4. Radionuclide concentrations in the secondary side fluid of the steam generators assume that Auxiliary Feedwater is provided to maintain a constant secondary mass during periods of steam release.

As a result of the rod ejection accident less than 10% of the fuel rods in the core are predicted to undergo DNB. In determining the offsite doses following rod ejection accident, it is conservatively assumed that 10% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released to the RCS. A small fraction (i.e., 0.25%) of the fuel in the core is also conservatively assumed to melt as a result of the rod ejection accident. The gap and release fractions from all of the damaged fuel correspond to the requirements set out in Regulatory Position 1 of Appendix H to Regulatory Guide 1.183.

For the containment release case, 100% of the failed fuel gap activity is released instantaneously to the containment. Natural deposition of the released activity inside of containment is credited. Radionuclide removal by the Emergency Containment Filters and containment spray is not credited. The containment is assumed to leak at the Technical Specification maximum allowable rate of 0.20% of containment air weight for the first 24 hours and 0.10% of containment air weight for the remainder of the event.

For the secondary release case, primary coolant activity consists of the failed fuel gap activity fraction of the core inventory, and is released into the steam generators by leakage across the steam generator tubes. The core source term used as the basis for this activity is presented in Table 14.3.5-7. Core activities are then increased by the radial peaking factor of 1.65 for this event and then reduced to gap activities and releases by applying appropriate fractions. All noble gases associated with this leakage are assumed to be released directly to the environment. Secondary activity is then released to the atmosphere via steaming from the MSSVs/ADVs until the RHR system is capable of removing decay heat and for providing for any further cooldown. The release of the initial iodine content of the steam generator secondary coolant is also considered. The source term for this secondary side activity is presented in Table 14.2.4-4. C26

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 flow distribution during this mode is 1000 cfm of unfiltered fresh air plus 100 cfm of unfiltered inleakage.

- For the secondary release, the Control Room is isolated on a high radiation reading at the normal intake monitors. A 60 second delay is applied to account for the time to reach the setpoint, signal processing, and damper closure time. For the containment release, Control Room isolation occurs on high radiation on the containment radiation monitors. The 60 second delay time is conservatively applied to this release model. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 100 cfm of unfiltered inleakage, and 375 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 and 95% for elemental and organic iodine.

The atmospheric dispersion factors (X/Q) used for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These X/Q valves are summarized in Appendix 2F. Releases from the steam generators are assumed to occur from the MSSV/ADV which produces the most limiting X/Q. Atmospheric dispersion factors for the containment release correspond to the nearest containment penetration. C26

The EAB and LPZ doses are determined using the X/Q factors for the appropriate time intervals. These X/Q factors are provided in Appendix 2E.

The secondary release scenario credits control room isolation from a high radiation signal on the control room intake monitor. The Technical Specification setpoint for this instrument is 2 mR/hr. In the RCCA Ejection analysis, an analytical setpoint of 5 mR/hr was used to account for measurement and test uncertainties and to apply additional conservatism. For the design basis fuel failure and core melt fractions, the calculated exposure rate at the detector exceeded the analytical setpoint by approximately 35%. It was recognized that with only 35% margin, a scenario could be postulated with fuel failure fractions less than the design values in which the analytical setpoint would not be reached and a delayed manual isolation must be assumed. While the offsite dose consequences would be lower in such a scenario, the relative impact of lower fuel failure fractions with a longer control room isolation time was not immediately obvious. Therefore, an additional case was performed which combined the reduced source term with a 30-minute control room isolation time.

The major assumption and parameters used in the analysis are itemized in Table 14.2.6-3. The dose conversion factors, breathing rates, and control room occupancy factors are the same as those given in the LOCA dose analysis in Table 14.3.5-5. The atmospheric dispersion factors used in the dose calculations are given in Appendices 2E and 2F.

The offsite and control room dose limits for a rod ejection accident are provided in Regulatory Guide 1.183 (Reference 10). C26

The offsite and control room doses due to the rod ejection accident are given in Table 14.2.6-5. The offsite doses due to the rod ejection accident do not exceed the acceptance criteria.

The dose consequence to the control room is acceptable under conditions where the control room ventilation automatically isolates, as well as conditions where the control room ventilation system must be manually isolated by operator action.

14.2.6.5 CONCLUSIONS

Despite the conservative assumptions, 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 demonstrate that the fission product release as a result of fuel rods entering DNB is limited to less than 10 percent of the fuel rods in the core.

14.2.6.6 REFERENCES

1. Risher, D. H., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors using Special Kinetics Methods," WCAP-7588, Rev 1-A, January 1975. C26

2. Taxebius, T. G., ed., "Annual Report - Spert Project, October 1968 - September 1969," IN-1370 Idaho Nuclear Corporation, June 1970.

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

4. Letter from W. J. Johnson of Westinghouse Electric Corporation to Mr. R. C. Jones of the Nuclear Regulatory Commission, Letter Number NS-NRC-89-3466, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," October 23, 1989.

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7. Bishop, A. A., Sandberg, R. O. and Tong, L. S., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux", ASME 65-HT-31, August 1965.

8. "Fuel Densification Turkey Point Unit No. 3," WCAP-8074, February 1973.

9. S.S. Witter to J.L. Perryman “Turkey Point Units 3 & 4-ZIRLO Safety Assessment Revision 1,” 98FP-G-0054, June 9, 1998.

10. USNRC, Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants,” July 2000. C26

TABLE 14.2.6-1

RESULTS OF THE
ROD CLUSTER CONTROL ASSEMBLY (RCCA)
EJECTION ACCIDENT ANALYSIS

	Beginning of Cycle	Beginning of Cycle	End of Cycle	End of Cycle
Power level, percent	100.3	0	100.3	0
Ejected rod worth, percent ΔK	0.33	0.71	0.30	0.84
Delayed neutron fraction, percent	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.29	1.60	1.30	2.28
Trip reactivity percent ΔK	4.0	2.0	4.0	2.0
Hot Channel Factor before rod ejection	2.40	---	2.40	---
Hot Channel Factor after rod ejection	5.48	8.0	5.52	14.3
Number of operational pumps	3	2	3	2
Max fuel pellet average temperature, °F	4076	2222	3920	3296
Max fuel centerline temperature, °F	4990	2562	4892	3691
Max clad average temperature, °F	2216	1602	2070	2364
Max fuel stored energy, cal/gm	178.3	87.6	170.2	138.5
Fuel melt in hot pellet, percent	8.23	0	8.44	0

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TABLE 14.2.6-2

SEQUENCE OF EVENTS
RCCA EJECTION ACCIDENT

CASE	EVENT	TIME (SEC)
BOL, full power	Initiation of Rod Ejection	0.00
	Power Range High Neutron Flux Setpoint Reached	0.04
	Peak Nuclear Power Occurs	0.13
	Rods Begin to Fall	0.54
	Peak Clad Temperature Occurs	2.24
	Peak Heat Flux Occurs	2.25
	Peak Fuel Center Temperature Occurs	5.52
EOL, zero power	Initiation of Rod Ejection	0.00
	Power Range High Neutron Flux Setpoint Reached	0.17
	Peak Nuclear Power Occurs	0.21
	Rods Begin to Fall	0.67
	Peak Clad Temperature Occurs	1.37
	Peak Heat Flux Occurs	1.38
	Peak Fuel Center Temperature Occurs	3.49

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TABLE 14.2.6-3

ASSUMPTIONS USED
FOR
ROD EJECTION ACCIDENT DOSE ANALYSIS

Input/Assumption	Value
Release Inputs:	
Core Power Level	2652 MW_{th}
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 –5.0 wt. %
Radial Peaking Factor	1.65
Percent of Core in DNB	
Design Basis	10%
Manual CR Isolation Case	6.22%
Percent of Core with Centerline Melt	
Design Basis	0.25%
Manual CR Isolation Case	0.16%
Gap Release Fraction	Reg. Guide 1.183, Appendix H, Position 1
Core Fission Product Inventory	Table 14.3.5-7
Initial Secondary Side Equilibrium Iodine Activity	0.1 µCi/gm DE I-131 (Table 14.2.4-4)
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
Removal Inputs:	
Secondary Side Partition Coefficients	SG (Flashed leakage) – none SG (Non-flashed leakage) – 100
Steam Generator Tube Leakage Rate	0.2 gpm/SG
Time to establish shutdown cooling and terminate steam release	63 hours
Time to re-cover SG Tubes	30 minutes
Tube Uncovery Flashing Fraction	11%
RCS Mass (minimum)	366,086 lbm
SG Secondary Side Mass	67,707 lbm per SG
Chemical Form of Iodine Released to Containment	Particulate – 95% Elemental – 4.85% Organic – 0.15%
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97 % Organic – 3%
Transport Inputs:	
Atmospheric Dispersion Factors	
Offsite	Appendix 2E
Onsite	Appendix 2F
Time of CR Isolation – Containment Release	30 seconds - High Containment Radiation
Time of CR Isolation – Secondary (Automatic)	60 seconds - High Radiation on CR Intake Monitors
Time of CR Isolation – Secondary (Manual)	30 Minutes - Manual Isolation
Unfiltered Inleakage	100 cfm

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ASSUMPTIONS USED
FOR
ROD EJECTION ACCIDENT DOSE ANALYSIS

Input/Assumption	Value
Transport Inputs (Continued):	
Breathing Rates	Reg. Guide 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	Reg. Guide 1.183 Section 4.2.6
Containment Release Inputs:	
Containment Volume	1.60E+06 ft^3
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.20% (by weight)/day 0.10% (by weight)/day
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 5.58 hr^{-1} Organic Iodine – None

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TABLE 14.2.6-4

ROD EJECTION INTACT STEAM GENERATOR STEAM RELEASE RATE*

Time (hours)	Intact SGs Steam Release Rate (lbm/min)
0.0	2598
2.0	2143
3.0	2016
4.0	1900
5.0	1779
8.0	2598
11.0	965
16.0	864
24.0	820
63.0	0.0

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* Stored energy above RHR entry conditions is released between 2 and 8 hours

TABLE 14.2.6-5

ROD EJECTION ACCIDENT DOSES

Case	EAB Dose[1] (rem TEDE)	LPZ Dose[2] (rem TEDE)	Control Room Dose[2] (rem TEDE)
RCCA Ejection – Containment Release	0.70	0.29	2.07
RCCA Ejection – Secondary Release (Automatic CR Isolation)	0.49	0.43	1.18
RCCA Ejection – Secondary Release (Manual CR Isolation)	0.29	0.26	3.44
Acceptance Criteria (Concurrent iodine spike)	6.3 [3]	6.3 [3]	5 [4]

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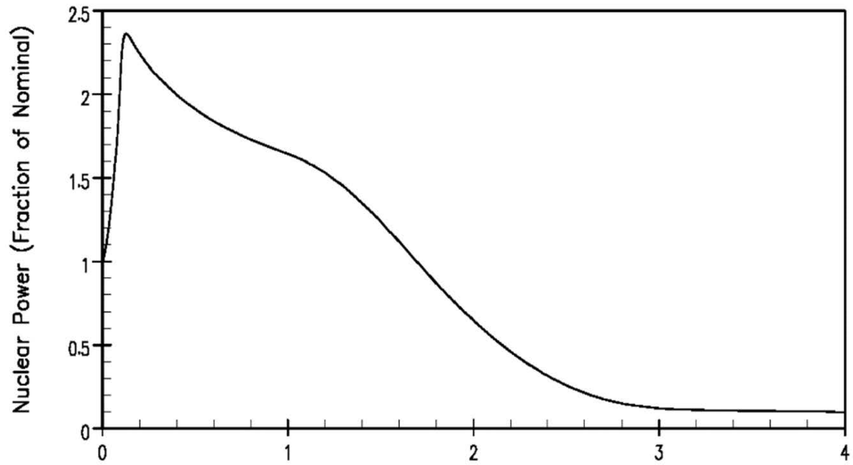
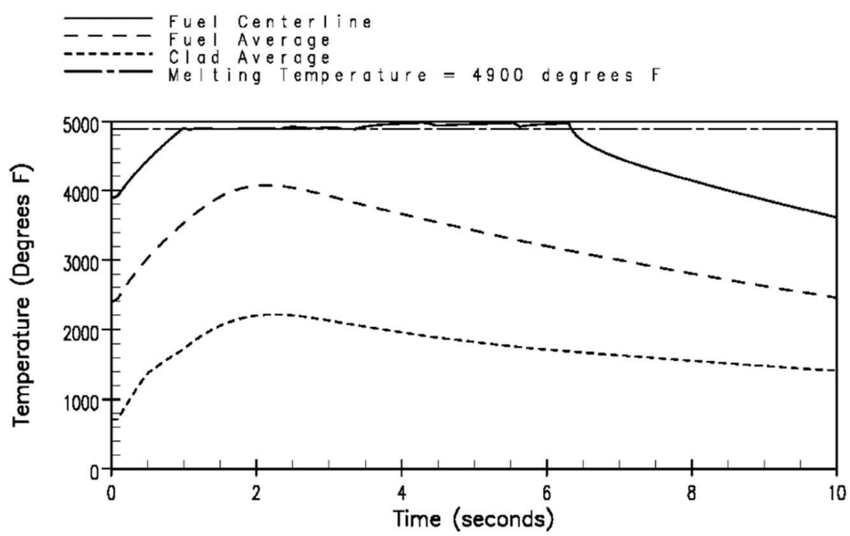
Notes:

[1] Worst 2-hour dose
[2] Integrated 30-day dose
[3] Reg. Guide 1.183, Table 6
[4] 10CFR50.67





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RCCA EJECTION TRANSIENT
BEGINNING OF LIFE
FULL POWER
FIGURE 14.2.6-1

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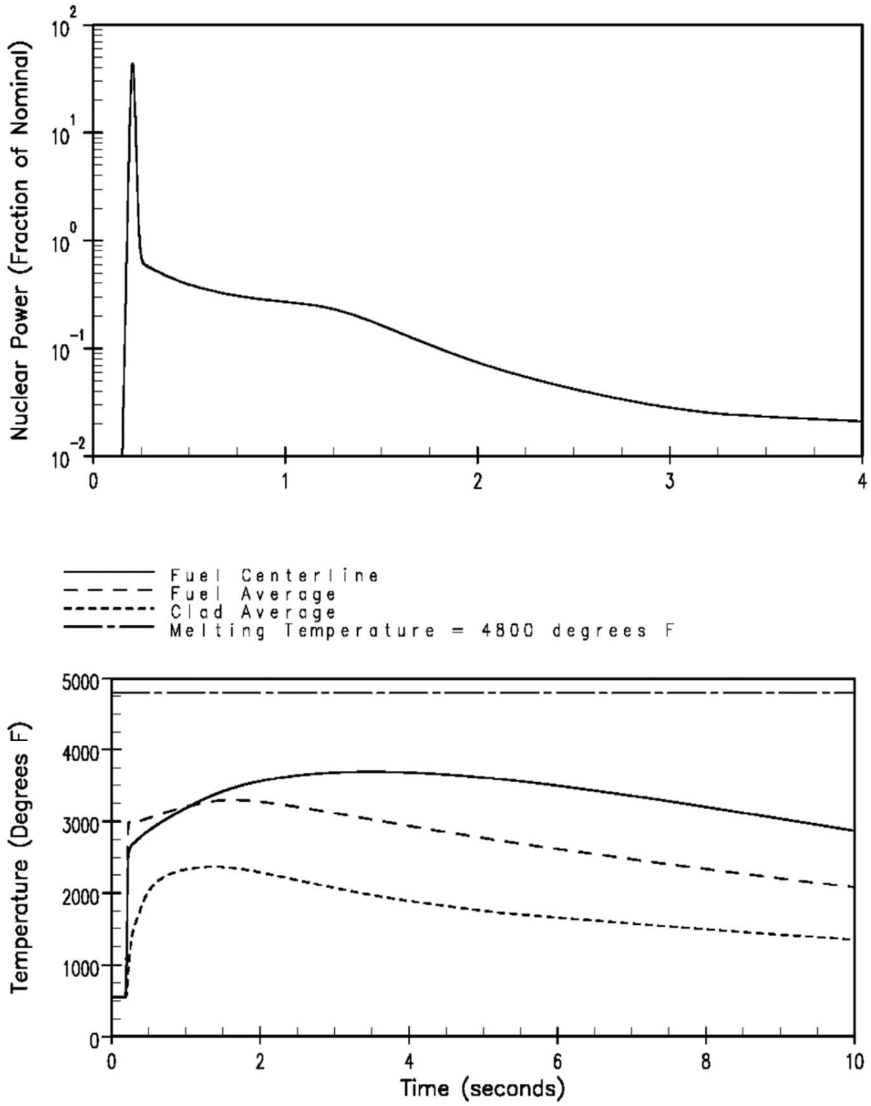

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RCCA EJECTION TRANSIENT
BEGINNING OF LIFE
ZERO POWER
FIGURE 14.2.6-2

14.2.7 FEEDWATER SYSTEM PIPE BREAK

As discussed in Attachment 1 of Reference 1, the Turkey Point analysis of the feedwater system pipe break event is not considered to be a typical design basis safety analysis. Rather, the analysis, originally performed to support the extended power uprate project, is intended to provide reasonable assurance that the event consequences would not present a safety concern. In support of this intent, the key analysis inputs were made consistent with those that are typically limiting for similar plants with feedring-type steam generators.

The evaluation presented below is distinct from a typical design basis Feedwater Line Break (FWLB) safety analysis, which would normally consider numerous input scenarios to determine the limiting case. Using input assumptions that are typical of the limiting analyzed case, the evaluation provides reasonable assurance that the consequences of a FWLB event do not present a safety concern for operation at EPU conditions.

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14.2.7.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A feedwater line break (FWLB) event is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the steam generators. If the break were to occur in the main feedwater line between the check valve and the steam generator, fluid from the steam generator would be discharged through the break. Furthermore, with steam generators, because the auxiliary feedwater piping connects to the main feedwater line, a break between a main feedwater line check valve and the corresponding steam generator could preclude the subsequent addition of auxiliary feedwater to that steam generator. In contrast, if a break occurs upstream of a feedwater line check valve, the transient would progress like a loss of normal feedwater event (Section 14.1.11), where there is no sudden loss of steam generator water inventory. Based on the size of the break and the plant operating conditions at the time the break occurs, a FWLB could cause either a cooldown, via excessive energy discharge through the break, or a heatup of the reactor coolant system. As the consequences of a reactor coolant system cooldown resulting from a feedwater line break are bounded by the cooldown consequences of a steam system pipe rupture (Section 14.2.5) where the higher enthalpy of steam translates into a greater heat transfer between the primary and secondary systems, and thus a greater cooldown, the feedwater system pipe break event is analyzed only with respect to reactor coolant system heatup.

When the subcooled feedwater flow to the steam generators is reduced by a FWLB, the long-term ability of the secondary system to remove heat from the reactor coolant system is diminished. The feedwater flow reduction can cause the reactor coolant system temperatures to increase prior to reactor trip. Additionally, for the FWLB event located in the main feedwater line between the check valve and the steam generator, the loss of steam generator fluid inventory through the break will reduce the heat sink volume available for decay heat removal following reactor trip.

The size of the break and the functionality of the main feedwater control system are two important factors during a FWLB event. Some breaks may be small enough such that a properly functioning main feedwater control system will be able to completely make up for the resultant inventory loss. In contrast, a larger FWLB can cause a sizeable blowdown (inventory loss) that prevents the main feedwater control system from being able to supply enough feedwater to maintain shell-side fluid inventory in the steam generators. This leads to a low-low steam generator water level reactor trip and auxiliary feedwater actuation. Another important factor during a FWLB transient is the shell-side fluid inventory in the unfaulted steam generators at the time of reactor trip. It is conservative if this fluid inventory is minimized because it minimizes the heat removal capacity of the steam generators, which maximizes the reactor coolant system heatup. It is also conservative if the initial fluid inventory in the faulted steam generator is maximized because this will delay a reactor trip on low-low steam generator level. C26

Following a FWLB, there is a rapid decrease in the steam generator inventory, a fast increase in the average reactor coolant temperature, a surge of water into the pressurizer with a resultant pressure increase in the reactor coolant system, as well as a pressure increase in the main steam system. When the steam generator water level reaches the low-low reactor protection setpoint, a reactor trip occurs and the auxiliary feedwater system is actuated. A subsequent turbine trip further reduces the heat removal ability of the steam generators, and the steam generator pressure in each of the unfaulted loops increases rapidly. Ignoring the non-safety-related, secondary-side power-operated relief valves and atmospheric steam dump valves, the pressure would reach the setpoint of the first (lowest setpoint) main steam safety valve of each unfaulted loop and remain there until the reactor coolant system heatup ceases, i.e., until the heat removal ability of the steam generators being fed auxiliary feedwater is sufficient to remove the decay heat generated in the core (also known as the time of event turnaround).

During the heatup period after reactor trip, the pressurizer pressure increases to, and is maintained near, the setpoint of the pressurizer power-operated relief valves. At event turnaround, the reactor coolant system temperature and pressure, and the pressurizer water level begin to decrease. Subsequently, the plant operators can follow the applicable emergency operating procedures to first bring the plant to a stabilized temperature condition using the pressurizer power-operated relief valves and steam generator atmospheric dump valves (if offsite power is available), and then eventually to a cold shutdown condition.

Unless the effects of the FWLB and subsequent steam generator water level reduction are counteracted by manual or automatic action, the rise in reactor coolant temperature could eventually result in a loss of subcooled margin in the reactor coolant system hot or cold legs, and/or a challenge to the integrity of the reactor coolant system and main steam system pressure boundaries.

The following provide the necessary protection against a feedwater system pipe break: C26

1. Reactor trip on:
   a. High pressurizer pressure.
   b. High pressurizer water level.
   c. Overtemperature ΔT.
   d. Low-low water level in any steam generator.
   e. Safety injection signal on:
      i. High containment pressure.
      ii. Low pressurizer pressure.
      iii. High differential pressure between the steam line header and any steam line.
      iv. High steam line flow coincident with either low steam generator pressure or low vessel average temperature.
   f. Low pressurizer pressure
   g. Steam flow-feedwater flow mismatch coincident with low water level in any steam generator.

2. Three turbine-driven auxiliary feedwater pumps (shared by Units 3 & 4) are started on any of the following:

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

   b. Any safety injection signal.

   c. Loss of offsite power (automatic transfer to diesel generators).

   d. Loss of either A or B 4.16 kV bus on either unit.

   e. Trip of all main feedwater pumps in either unit.

   f. Manual actuation.

3. The main steam safety valves open to provide an additional heat sink and protection against secondary side overpressure.

4. Operator action may be needed to isolate the auxiliary feedwater flow from the break and redirect that flow to the unfaulted loops.

The analysis of the FWLB event is intended to show that the core will remain covered with water by demonstrating that there is no boiling in the reactor coolant loops, thus confirming the adequacy of the auxiliary feedwater system for removing the stored and residual heat. C26

14.2.7.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

Method of Analysis

As discussed in Attachment 1 of Reference 1, a representative FWLB case, where key analysis inputs were made consistent with those that are typically limiting for similar plants with feedring-type steam generators, was analyzed. Key inputs include modeling the largest break possible (a double-ended rupture) for the Turkey Point steam generator model and modeling minimum reactivity feedback parameters.

A detailed analysis using the RETRAN code (Reference 2) is performed in order to obtain the plant transient conditions following a FWLB. The analysis addresses the core neutron kinetics, reactor coolant system including natural circulation, pressurizer, pressurizer power-operated relief valves and sprays, steam generators, main steam safety valves, and auxiliary feedwater system. The code computes pertinent variables, including the pressurizer pressure, pressurizer water level, steam generator mass, and reactor coolant average temperature.

The major inputs and assumptions used in the analysis are described as follows:

1. The analysis modeled an initial NSSS power of 100.3 percent of the nominal NSSS power of 2652 MWt; this nominal value consists of a core power of 2644 MWt and a reactor coolant pump heat of 8 MWt. The 0.3 percent above nominal accounts for the power uncertainty.

2. An initial reactor vessel average coolant temperature of 589°F, which accounts for an uncertainty of +6.0°F, was applied.

3. An initial pressurizer pressure of 2197 psia, which accounts for an uncertainty of -53 psi, was applied.

4. An initial pressurizer water level of 67.1 percent span, which accounts for an uncertainty of +7.1 percent span, was applied.

5. The initial main feedwater enthalpy was set to a value corresponding to the maximum full power main feedwater temperature of 440°F. All main feedwater flow is terminated coincident with the time of the break to simulate it spilling out through the break. C26

6. An initial water level of 56 percent narrow range span, which accounts for an uncertainty of +6 percent narrow range span, was applied for the faulted loop steam generator, and an initial water level of 38 percent narrow range span, which accounts for an uncertainty of -12 percent narrow range span, was applied for each of the unfaulted loop steam generators. The level in the faulted steam generator is maximized so as to maximize the time it takes to reach the low-low steam generator water level setpoint, and the level in each of the unfaulted steam generators is minimized so as to minimize the water inventory available for primary-to-secondary heat transfer.

7. The reactor coolant pumps were modeled to coast down following a reactor trip, as a result of a loss of offsite power. Prior to reactor coolant pump coastdown, a constant reactor coolant volumetric flow equal to the thermal design flow value (260,700 gpm) was modeled. A loss of offsite power that affects both units was assumed because it poses the greatest challenge on the auxiliary feedwater system to provide auxiliary feedwater flow to the unfaulted loops.

8. Reactor trip and auxiliary feedwater actuation occur on steam generator low-low water level at 0 percent narrow range span.

9. The auxiliary feedwater flow modeled corresponds to one auxiliary feedwater pump providing flow to both units. Only one auxiliary feedwater pump is credited because of single failure and pump availability considerations, and the auxiliary feedwater flow is divided among both units because the assumed loss of offsite power affects both units. The unit with the FWLB requires auxiliary feedwater flow to address the effects of the break and loss of offsite power, and the other unit requires auxiliary feedwater flow to address the effects of the loss of offsite power.

   The analysis models a minimum auxiliary feedwater flow of 96 gpm to each of the two unfaulted steam generators following a 115 second delay after a low-low steam generator level setpoint is reached. This auxiliary feedwater flow rate continues up to 10 minutes after the auxiliary feedwater initiation signal. The remaining auxiliary feedwater flow to the faulted unit is assumed to be lost through the break, resulting in no auxiliary feedwater flow to the faulted steam generator. After 10 minutes, it is assumed that plant operators have isolated the auxiliary feedwater flow from the break and redirected some of that flow to the two unfaulted steam generators. With the flow redirection, the auxiliary feedwater flow rate to each unfaulted steam generator is increased to 124 gpm. The auxiliary feedwater flow conditions were based on a maximum temperature of 106°F. C26

10. As the shutoff head for the safety injection system is below the setpoint for the pressurizer power-operated relief valves, these valves were assumed to be available and were modeled as being operable for maintaining reactor coolant pressure, which is conservative for the FWLB analysis in which bulk boiling in the reactor coolant system is a concern. The pressurizer sprays and heaters were not modeled.

11. Secondary system steam relief is achieved through the self-actuated main steam safety valves, which were modeled with 3.0 percent tolerance plus 5 psi to simulate valve accumulation. Note that steam relief would normally be provided by the steam generator atmospheric dump valves or condenser dump valves for most feedwater system pipe break events. However, the condenser dump valves and the atmospheric dump valves were conservatively assumed to be unavailable.

12. Minimum reactivity feedback was modeled.

13. A maximum break flow area of 0.89 ft^2 was modeled, which corresponds to the maximum effective flow area for a double-ended rupture of a main feedwater line.

14. Credit was taken for the main steam line check valves to prevent steam flow from the unfaulted steam generators into the faulted steam generator.

15. Credit was taken for a portion of the coolant-to-metal heat transfer that would occur during the long-term primary-side heatup. A RETRAN thick metal mass heat transfer model was developed using the methodology described in Reference 3.

16. Core residual heat generation was based on the 1979 version of ANS 5.1 (Reference 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 trip was assumed.

17. The maximum steam generator tube plugging level of 10 percent was applied.

18. Although safety injection was modeled, the shutoff head for the safety injection system, 1400 psig, is well below the minimum reactor coolant system pressure expected during a FWLB transient, and thus no safety injection flow is expected.

Results C26

Figures 14.2.7-1 through 14.2.7-5 show the significant plant parameters following a FWLB with the analytical inputs and assumptions identified above. The calculated sequence of events for this transient is presented in Table 14.2.7-1. Note that a steady-state run of 100 seconds preceded the initiation of the transient. The results of the FWLB analysis showed that no bulk boiling occurred in the primary coolant system prior to the time that the heat removal capability of the steam generators being fed auxiliary feedwater exceeded NSSS residual heat generation.

Another case was considered to address the possibility that the harsh environment resulting from a FWLB prevents the low-low steam generator level function, which was credited in the first case, from generating a reactor trip signal. For this case, it was determined that the water vapor entering the containment from the FWLB would cause the containment pressure to increase to the high pressure setpoint for initiating a safety injection signal, which would actuate a reactor trip that is several seconds later than was calculated crediting the low-low steam generator level function. The safety injection signal also actuated auxiliary feedwater, which occurs later than in the case that credited the low-low steam generator level function. Despite the delayed reactor trip and delayed auxiliary feedwater, the results showed no bulk boiling in the primary coolant system prior to the time that the heat removal capability of the steam generators being fed auxiliary feedwater exceeded NSSS residual heat generation.

14.2.7.3 CONCLUSIONS

Based on the analysis results, saturated conditions are not reached in the hot and cold legs during a FWLB transient, which conservatively demonstrates that the core remains covered. Thus, it is concluded that the available auxiliary feedwater capacity is adequate for long-term decay heat removal, and the applicable acceptance criteria for the feedwater system pipe break analysis are met.

14.2.7.4 REFERENCES

1. FPL Letter L-2011-438, "Turkey Point Units 3 and 4 Docket Nos. 50-250 and 50-251 Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205," October 15, 2011. C26

2. WCAP-14882-P-A (Proprietary) and WCAP-15234-A (Non-Proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999 and May 1999.

3. WCAP-14882-S1-P-A (Proprietary) and WCAP-15234-S1-A (Non-proprietary), "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses, Supplement 1 – Thick Metal Mass Heat Transfer Model and NOTRUMP – Based Steam Generator Mass Calculation Method," October 2005.

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

TABLE 14.2.7-1

SEQUENCE OF EVENTS
FOR
FEEDWATER SYSTEM PIPE BREAK

Event	Time (seconds)
Feedwater system pipe break	100.0
Low-low steam generator water level setpoint reached	105.4
Rods begin to drop	107.4
Turbine trip occurs	107.9
Reactor coolant pumps trip	109.4
Auxiliary feedwater flow is initiated to two unfaulted steam generators (96 gpm per steam generator)	220.4
Auxiliary feedwater flow is increased to two unfaulted steam generators (124 gpm per steam generator) via operator action	705.4
Final reactor coolant system cooldown begins (time of event turnaround)	3600

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Security-Related Information - Withheld Under 10 CFR 2.390

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NUCLEAR POWER AND CORE AVERAGE HEAT FLUX TRANSIENTS FOR FEEDWATER PIPE BREAK

FIGURE 14.2.7-1

Security-Related Information - Withheld Under 10 CFR 2.390

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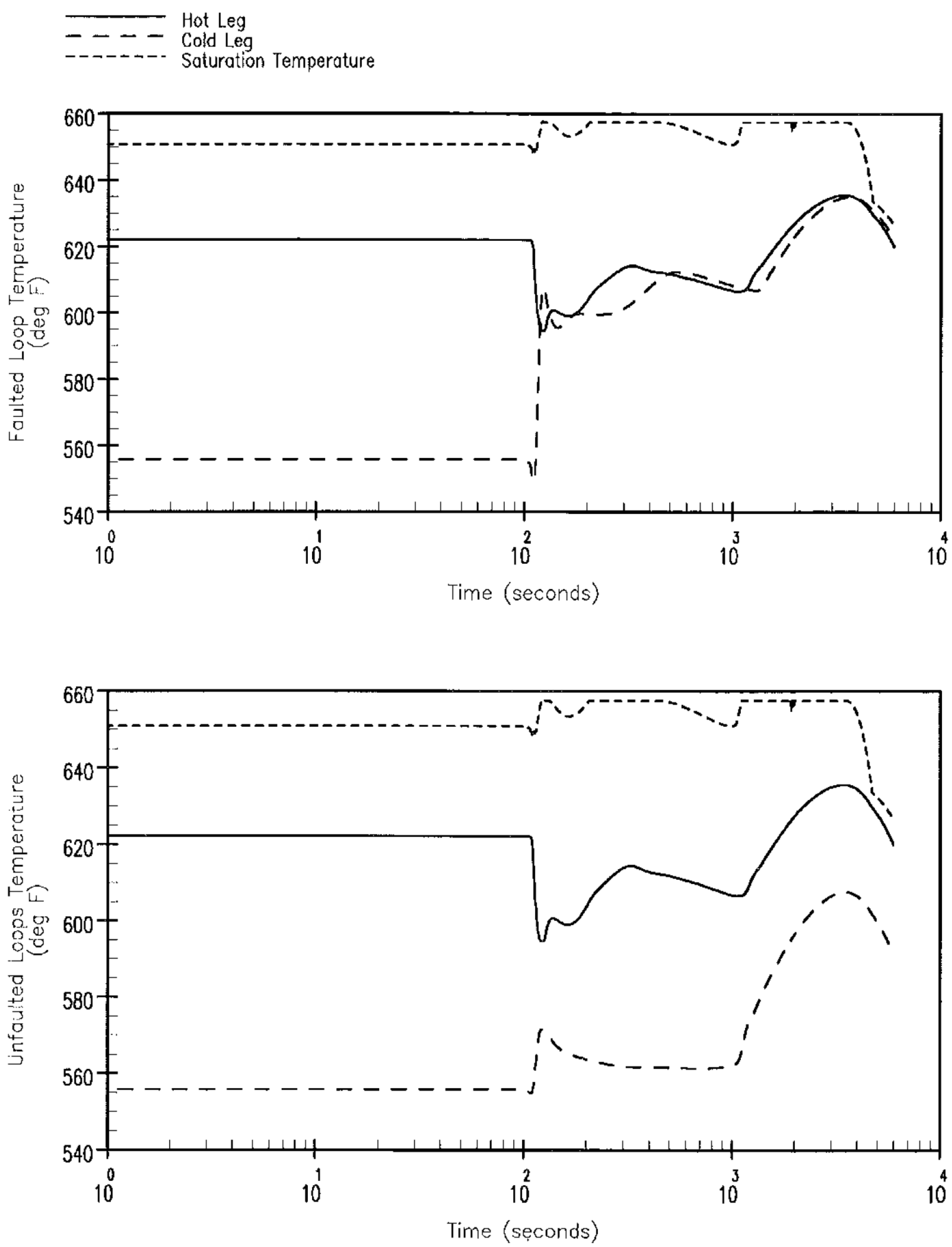
PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR FEEDWATER PIPE BREAK

FIGURE 14.2.7-2



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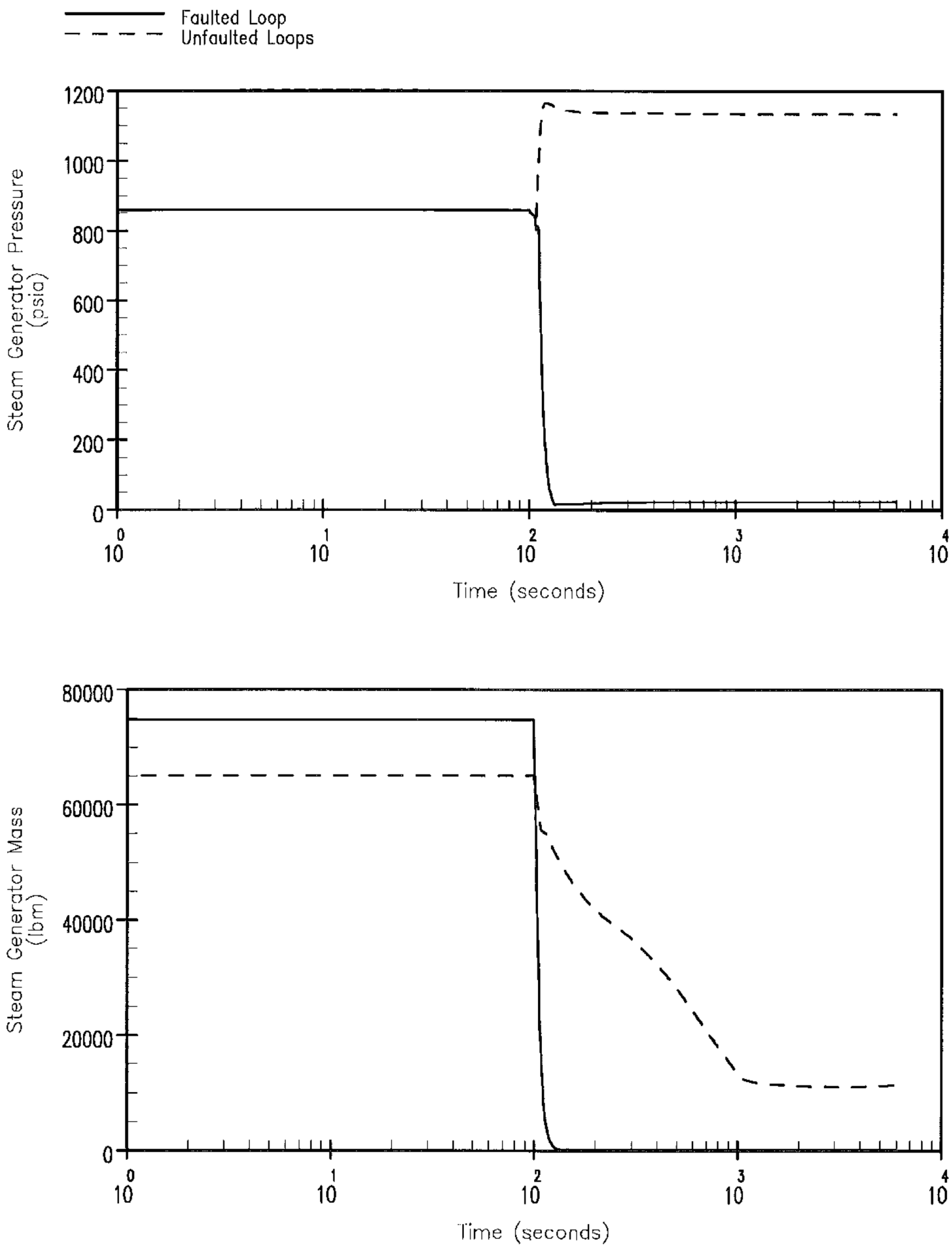
REACTOR COOLANT LOOP TEMPERATURE TRANSIENTS FOR FEEDWATER PIPE BREAK

FIGURE 14.2.7-3



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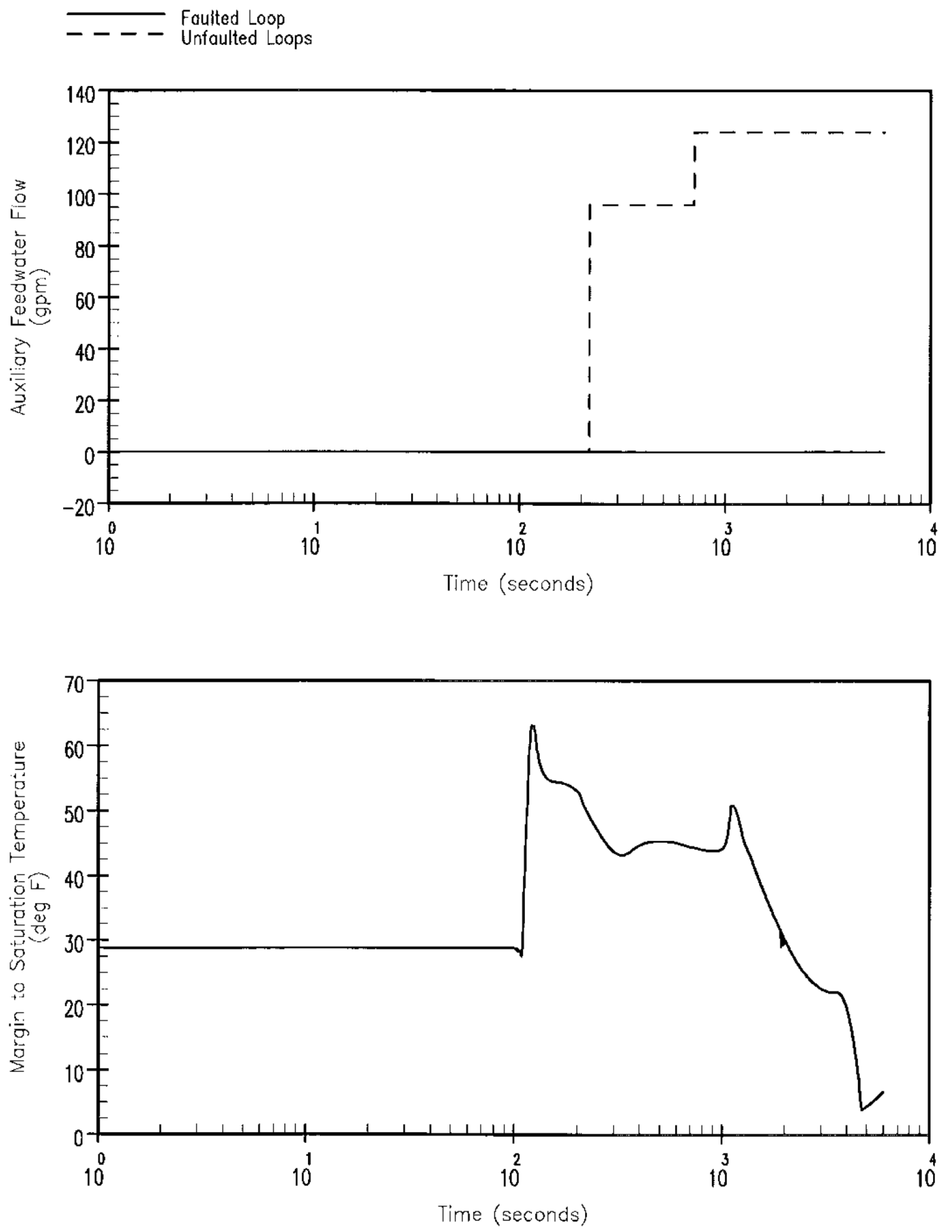
STEAM GENERATOR PRESSURE
AND MASS TRANSIENTS FOR
FEEDWATER PIPE BREAK

FIGURE 14.2.7-4



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AUXILIARY FEEDWATER FLOW AND
RCS SATURATION MARGIN FOR
FEEDWATER PIPE BREAK

FIGURE 14.2.7-5

14.3 REACTOR COOLANT SYSTEM PIPE RUPTURE

A comprehensive safety analysis of postulated pipe ruptures within the Reactor Coolant System (RCS) boundary has been performed. This analysis has included cases of the Loss of Coolant Accident (LOCA) resulting from a broad spectrum of small and large pipe ruptures including the Maximum Hypothetical Accident (MHA) case of the double ended break of the largest RCS pipe. Per ANS-51.1/N18.2-1973 (Reference 14), Small and Large Break LOCA events are classified as Condition III and Condition IV events, respectively. Offsite dose consequences are bounded by the Large Break LOCA event, therefore, Small and Large Break LOCA events are classified as Condition IV. C26

The objective of the analysis has been to determine the condition of the RCS, core, and containment in the event of a postulated LOCA, and to determine that the various Emergency Core Cooling Systems (ECCS) have the capability to control each LOCA, including the MHA.

14.3.1 GENERAL

A LOCA would result from a rupture of the Reactor Coolant System (RCS) or of any line connected to that system up to the first closed valve. The charging pumps have the capability to make up for leakage resulting from ruptures of a small cross section, thus permitting an orderly shutdown. The coolant released would remain in the containment.

For a postulated large break, reactor trip is initiated when the pressurizer low pressure set point is reached while the Safety Injection System (SIS) signal is actuated by pressurizer low pressure. The reactor trip and SIS actuation are also initiated by a high containment pressure signal. The consequences of the accident are limited in two ways:

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

b) Injection of borated water ensures sufficient flooding of the core to prevent excessive temperatures.

Before the reactor trip occurs, the reactor is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After reactor trip and turbine trip, core heat, heat from hot internals and the vessel is transferred to the RCS fluid, and then to the secondary system. The secondary system pressure increases and steam dump may occur. Make-up to the secondary side is automatically provided by the auxiliary feedwater pumps. The SIS signal stops normal feedwater flow by closing the feedwater regulating valves and initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure. When the RCS pressure falls below 600 psia, the accumulators begin to inject borated water. The reactor coolant pumps are assumed to be tripped at the initialization of the accident and effects of pump coastdown are included in the blowdown analyses.

<u>Performance Criteria for Emergency Core Cooling System</u>

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

The ECCS is designed to limit the cladding temperature to 2200°F in accordance with 10CFR50.46. In addition, the core metal-water reaction is limited to less than 1% of the available Zircaloy, and the oxidation to less than 17% of the cladding thickness.

14.3.2 THERMAL ANALYSIS

The analysis specified by 10 CFR 50.46,"Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors," (Reference 1) is presented in this section. The results of the loss of coolant accident analyses are summarized in Tables 14.3.2.1-3 and 14.3.2.2-2 and show compliance with the Acceptance Criteria.

The potential for adverse boric acid concentration occurring in the reactor vessel during the long term recirculation phase following LOCA has been analyzed. The analysis showed sufficient decay heat removal and that there is an effective active dilution mechanism to halt and reverse the concentration of boric acid in the core prior to the solubility limit being reached, thereby maintaining long term cooling. C26

The boundary considered for loss of coolant accidents as related to connecting piping is defined in Section 4.1.

The method of analysis to determine peak cladding temperature is divided into two types of analysis: (1) large break LOCA; and (2) small break LOCA. The method of analysis for large and small break LOCA is described below and results are given.

14.3.2.1 Best-Estimate Large-Break Loss of Coolant Accident (BE-LBLOCA) Analysis C26

Should a major break occur, depressurization of the RCS would result in a pressure decrease in the pressurizer. The reactor trip signal would subsequently occur when the pressurizer low pressure trip setpoint is reached. An SI signal is generated when the appropriate setpoint (high containment pressure or low pressurizer pressure) is reached. These countermeasures will limit the consequences of the accident in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to the residual level corresponding to fission product decay heat. Insertion of control rods to shut down the reactor is neglected in the initial phase of the large break analysis*. An average RCS/sump mixed boron concentration is calculated to ensure that the post-LOCA core remains subcritical.

B. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

In the current best-estimate analysis, the most limiting large break LOCA single failure is assumed to be the loss of one High Head Safety Injection (HHSI) pump and one Low Head Safety Injection (LHSI) pump due to the failure of a single emergency diesel generator. While this is maintained for Turkey Point Units 3 and 4, the cross-tie between units causes an additional HHSI pump to be credited, giving 2 HHSI pumps and 1 LHSI pump total.

For the large break analysis, one ECCS train, including two HHSI pumps and one RHR (low-head) pump, starts and delivers flow through the injection lines (one for each loop) with one branch injection line spilling to the containment backpressure. Both emergency diesel generators (EDG) are assumed to start in the modeling of the containment fan coolers and spray pumps. Modeling full containment heat removal systems operation is required by Reg. Guide 1.157 and is conservative for the large break LOCA.

Prior to the accident, the RCS is assumed to be operating normally at full power. A large cold leg break is assumed to open nearly instantaneously in one of the main coolant pipes. Calculations, where the location and size of the break have been varied, indicate that a break in the cold leg between the pump and the vessel leads to the most severe transient. For this break, the LOCA transient can be conveniently divided into a number of time periods in which specific phenomena are occurring. The description of these time periods is described in Section 14.3.2.1.4.

* As documented in Engineering Evaluation: PTN-ENG-SEFJ-02-016, control rods are assumed to be inserted at the time of hot leg switchover during the recovery phase from a cold leg large break LOCA. Control rod insertability is assumed in order to address recriticality concerns, due to excessive sump dilution, at the time of hot leg switchover. This assumption only applies to cold leg large break LOCAs, not to hot leg large break LOCAs for which the control rods are ignored for the entire duration of the accident.

14.3.2.1.1 General

When the Final Acceptance Criteria (FAC) governing the loss-of-coolant accident (LOCA) for Light Water Reactors was issued in Appendix K of 10 CFR 50.46 (Reference 1), both the Nuclear Regulatory Commission (NRC) and the industry recognized that the stipulations of Appendix K were highly conservative. That is, using the then accepted analysis methods, the performance of the Emergency Core Cooling system (ECCS) would be conservatively underestimated, resulting in predicted Peak Clad Temperatures (PCT) much higher than expected. At that time, however, the degree of conservatism in the analysis could not be quantified. As a result, the NRC began a large-scale confirmatory research program with the following objectives:

1) Identify, through separate effects and integral effects experiments, the degree of conservatism in those models permitted in the Appendix K rule. In this fashion, those areas in which a purposely prescriptive approach was used in the Appendix K rule could be quantified with additional data so that a less prescriptive future approach might be allowed. C26

2) Develop improved thermal-hydraulic computer codes and models so that more accurate and realistic accident analysis calculations could be performed. The purpose of this research was to develop an accurate predictive capability so that the uncertainties in the ECCS performance and the degree of conservatism with respect to the Appendix K limits could be quantified.

Since that time, the NRC and the nuclear industry have sponsored reactor safety research programs directed at meeting the above two objectives. The overall results have quantified the conservatism in the Appendix K rule for LOCA analyses and confirmed that some relaxation of the rule can be made without a loss of safety to the public. It was also found that some plants were being restricted in operating flexibility by the overly conservative Appendix K requirements. In recognition of the Appendix K conservatism that was being quantified by the research programs, the NRC adopted an interim approach for evaluation methods. This interim approach is described in Reference 2. The Reference 2 approach retained those features of Appendix K that were legal requirements, but permitted applicants to use best-estimate thermal-hydraulic models in their ECCS evaluation model. Thus, Reference 2 represented an important step in basing licensing decisions on realistic calculations, as opposed to those calculations prescribed by Appendix K.

In 1998, the NRC Staff amended the requirements of 10 CFR 50.46 and Appendix K, “ECCS Evaluation Models”, to permit the use of a realistic evaluation model to analyze the performance of the ECCS during a hypothetical LOCA. This decision was based on an improved understanding of LOCA thermal-hydraulic phenomena gained by extensive research programs. 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 LOCA 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 criteria of 10 CFR 50.46. Further guidance for the use of best-estimate codes is provided in Reference 3.

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 4). This method outlined an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifying the uncertainties in a LOCA analysis. C26

The LOCA evaluation methodology for three- and four-loop Pressurized Water Reactor (PWR) plants based on the revised 10 CFR 50.46 rules was developed by Westinghouse with the support of EPRI and Consolidated Edison, and has been approved by the NRC (Reference 5).

More recently, Westinghouse developed an alternative uncertainty methodology called ASTRUM, which stands for Automated Statistical Treatment of Uncertainty Method (Reference 6). This method is still based on the CQD methodology (Reference 5) and follows the steps in the CSAU methodology (Reference 4). However, the uncertainty analysis (Element 3 in the CSAU) is replaced by a technique based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in Reference 6.

The three 10 CFR 50.46 criteria (peak clad temperature, maximum local oxidation (MLO), and core-wide oxidation) are satisfied by running a sufficient number of WCOBRA/TRAC calculations (sample size). In particular, the statistical theory predicts that 124 calculations are required to simultaneously bound the 95^{th} percentile values of three parameters with a 95 percent confidence level.

This analysis is in accordance with the applicability limits and usage conditions defined in Section 13-3 of Reference 6, as applicable to the ASTRUM methodology. Section 13-3 of Reference 6 was found to acceptably disposition each of the identified conditions and limitations related to WCOBRA/TRAC and the CQD uncertainty approach per Section 4.0 of the ASTRUM Final Safety Evaluation Report appended to this topical report.

The Turkey Point Units 3 and 4 ASTRUM LBLOCA uses a plant-specific adaptation of the ASTRUM methodology that includes a more refined downcomer model, explicit modeling of fuel thermal conductivity degradation (TCD), as well as a larger sampling range for rod internal pressure (RIP) uncertainty.

Preliminary results with a less refined downcomer model for a prior analysis with the as-approved ASTRUM method were observed to yield unexpected results which were attributed to overly conservative aspects of the model. Consequently, an adaptation of ASTRUM was developed to better model the downcomer region by increasing the number of circumferential noding stacks by a factor of three. For Turkey Point Units 3 and 4, nine downcomer stacks were modeled instead of the usual three. The detailed radial noding of the vessel wall remains unchanged from the approved ASTRUM Evaluation Model and therefore does not alter the historically approved method for addressing downcomer boiling during reflood. This finer nodalization has been assessed against experimental data, as described in Reference 9; the results of these assessments confirmed the continuing applicability of the Reference 6 uncertainty methodology for the Turkey Point Units 3 and 4 plant specific adaptation. C26

Explicit modeling of TCD in the fuel performance code leads directly to increased fuel temperatures as well as other fuel performance related effects beyond beginning-of-life. This will tend to increase the stored energy at the beginning of the simulated large-break LOCA event, which leads to an increase in PCT if there is no provision to credit off-setting effects. In order to mitigate the impact of increased effect of pellet TCD with burnup, the analysis also credited peaking factor burndown (summarized in Table 14.3.2.1-8) to address fuel in its second cycle of irradiation. Analysis of fuel in its second cycle of irradiation is beyond the first cycle considered in the approved ASTRUM methodology, but was considered in the analysis when explicitly modeling TCD to demonstrate that analyzing the hot rod and hot assembly in the first cycle of operation is still bounding with respect to PCT and MLO. Physically, accounting for TCD leads to an increase in fuel temperature as the fuel is burned, while accounting for peaking factor burndown leads to a reduction in fuel temperature as the fuel is burned.

The compensating nature of these phenomena is considered in order to appropriately capture the effect of TCD in the LBLOCA analysis. In addition, a different fuel thermal conductivity model in WCOBRA/TRAC and HOTSPOT was used to more accurately model the fuel temperature profile when accounting for TCD.

In addition to the standard uncertainty calculations, the Turkey Point Units 3 and 4 LBLOCA analysis sampled a larger rod internal pressure (RIP) uncertainty than originally included in the ASTRUM methodology (Reference 6). It was discovered that the as-approved sampling range did not bound the plant-specific RIP uncertainties for Turkey Point. Therefore, the approved sampling range was expanded to bound the Turkey Point Units 3 and 4 plant specific data.

14.3.2.1.2 Method of Analysis

The methods used in the application of WCOBRA/TRAC to the large break LOCA with ASTRUM are described in Reference 5 and Reference 6. 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. WCOBRA/TRAC MOD7A was used for the execution of ASTRUM for Turkey Point Units 3 and 4. C26

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 completed and detailed simulation of a PWR. This best-estimate computer code contains the following features:

1) Ability to model transient three-dimensional flows in different geometries inside the vessel

2) Ability to model thermal and mechanical non-equilibrium between phases

3) Ability to mechanistically represent interfacial heat, mass, and momentum transfer in different flow regimes

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

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 proceeds continuously, using the same computer code (WCOBRA/TRAC) and the same modeling assumptions. Containment pressure is modeled with the BREAK component using a time dependent pressure table. Containment pressure is calculated using the COCO code (Reference 7) and mass and energy releases from the WCOBRA/TRAC calculation.

The final step of the best-estimate methodology, in which all uncertainties of the LOCA parameters are accounted for to estimate a PCT, MLO, and Core-Wide Oxidation (CWO) at 95-percent probability, is described in the following sections.

1) Plant Model Development:

   In this step, a WCOBRA/TRAC model of the plant 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 ensure 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 difference, such as in the upper plenum of the reactor vessel or the ECCS injection configuration. C26

2) Determination of Plant Operating Conditions:

   In this step, the expected or desired operating range of the plant to which the analysis applies is established. The parameters considered are based on a “key LOCA parameters” list that was developed as part of the methodology. A set of these parameters, at mostly nominal values, is chosen for input as initial conditions to the plant model. A transient is run utilizing these parameters and is known as the “initial transient”. Next, several confirmatory runs are made, which vary a subset of the key LOCA parameters over their expected operating range in one-at-a-time sensitivities. Because certain parameters are not included in the uncertainty analysis, these parameters are set at their bounding condition. This analysis is commonly referred to as the confirmatory analysis. The most limiting input conditions, based on these confirmatory runs, are then combined into the model that will represent the limiting state for the plant, which is the starting point for the assessment of uncertainties.

3) Assessment of Uncertainty:

The ASTRUM methodology is based on order statistics. The technical basis of the order statistics is described in Section 11 of Reference 6. The determination of the PCT uncertainty, MLO uncertainty, and CWO uncertainty relies on a statistical sampling technique. According to the statistical theory, 124 WCOBRA/TRAC calculations are necessary to assess against the three 10 CFR 50.46 criteria (PCT, MLO, CWO).

The uncertainty contributors are sampled randomly from their respective distributions for each of the WCOBRA/TRAC calculations. The list of uncertainty parameters, which are randomly sampled for each time in the cycle, break type (split or double-ended guillotine), and break size are also sampled as uncertainty contributors within the ASTRUM methodology.

Results from the 124 calculations are tallied by ranking the PCT from highest to lowest. A similar procedure is repeated for MLO and CWO. The highest rank of PCT, MLO, and CWO will bound 95 percent of their respective populations with 95-percent confidence level. C26

4) The 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 or may be narrower for some parameters to gain additional margin.

14.3.2.1.3 Analysis Assumptions

The expected PCT and its uncertainty developed are valid for a range of plant operating conditions. The range of variation of the operating parameters has been accounted for in the uncertainty evaluation. Table 14.3.2.1-1 summarizes the operating ranges as defined for the proposed operating conditions which are supported by the Best-Estimate LBLOCA analysis for Turkey Point Units 3 and 4. Tables 14.3.2.1-2 and 14.3.2.1-3 summarize the LBLOCA containment data used for calculating containment pressure (for both units). Table 14.3.2.1-8 summarizes the peaking factor margin and burndown credits supported by the BE LBLOCA analysis for Turkey Point Units 3 and 4. If operation is maintained within these ranges, the LBLOCA results developed in this report using WCOBRA/TRAC are considered to be valid. Note that some of these parameters vary over their range during normal operation (accumulator temperature) and other ranges are fixed for a given operational condition (T_{avg}).

14.3.2.1.4 Design Basis Accident

The Turkey Point Units 3 and 4 PCT-limiting transient is a double-ended cold leg guillotine break which analyzes conditions that fall within those listed in Table 14.3.2.1-1. Traditionally, cold leg breaks have been limiting for large break LOCA. This location is the one where flow stagnation in the core appears most likely to occur. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (Reference 5).

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. Specific important transient phenomena and heat transfer regimes are discussed below, with the transient results shown in Figures 14.3.2.1-1 to 14.3.2.1-11. The PCT-limiting case was chosen to show a conservative representation of the response to a large break LOCA.

1) Critical Heat Flux (CHF) Phase: C26

   Immediately following the cold leg rupture, the break discharge rate is subcooled and high break flow rates are observed (Figure 14.3.2.1-2). The regions of the reactor coolant system (RCS) with the highest initial temperatures (cord, upper plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses, and the fuel rods begin to go through departure from nucleate boiling (DNB). The fuel cladding rapidly heats up (Figure 14.3.2.1-1) while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash (Figures 14.3.2.1-6 and 14.3.2.1-11). The mixture swells and intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

2) Upward Core Flow Phase:

   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, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. Figure 14.3.2.1-3 shows the void fraction for two intact loop pumps and the broken loop pump. The figure shows that the intact loops remain in single-phase liquid flow for several seconds, resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase, and the pump head degrades.

3) Downward Core Flow Phase:

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, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top third of core vapor flow (Figure 14.3.2.1-4) best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure (Figure 14.3.2.1-5) the accumulators begin to inject cold borated water into the intact cold legs (Figure 14.3.2.1-8). During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow, and consequently the downward core flow, is reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS WATER (Figure 14.3.2.1-7).

4) Refill Period: C26

As the refill period begins, the core enters a period of heatup and the vessel begins to fill with ECCS water (Figures 14.3.2.1-8, 14.3.2.1-9). This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

5) Reflood Period:

During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system re-pressurization and the lower core region begins to quench (Figure 14.3.2.1-10). During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow is oscillatory as cold water periodically re-wets and quenches the hot fuel cladding, which generates steam and causes system re-pressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break.

This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force. From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain a full downcomer and complete the reflood period. As the quench front progresses up the core, the PCT location moves higher into the top core region. As the vessel continues to fill, the PCT location is cooled and the early reflood period is terminated.

A second cladding heatup transient may occur due to boiling in the downcomer. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel and vessel metal, reduces the subcooling of ECCS water in the lower plenum and downcomer. The saturation temperature is dictated by the containment pressure. If the liquid temperature in the downcomer reaches saturation, subsequent heat transfer from the vessel and other structures will cause boiling and level swell in the downcomer. The downcomer liquid will spill out of the broken cold leg and reduce the driving head, which can reduce the reflood rate, causing a late reflood heatup at the upper core elevations. Figure 14.3.2.1-11 shows a slight reduction in downcomer level. However, as seen in Figure 14.3.2.1-1, the cladding had already begun to quench and a late reflood heatup did not occur in the limiting PCT transient. C26

14.3.2.1.5 Additional Evaluations

An evaluation of IFBA fuel, including the effects of pellet TCD, shows that IFBA fuel is limiting. The AOR PCT and MLO results in Tables 14.3.2.1-4 and 14.3.2.1-5 reflect the results with the effects of IFBA taken into account.

In addition to the analyses presented in this section, evaluations may be performed as needed to address computer code errors and emergent issues, or to support plant changes. The issue or changes are evaluated, and the impact on the PCT is determined. The resultant increase or decrease in PCT is applied to the analysis of record PCT. The PCT, including all penalties and benefits, is presented in Table 14.3.2.1-4 for the Turkey Point large break LOCA. The current PCT is demonstrated to be less than the 10 CFR 50.46(b) requirement of 2200°F.

The transition from 15x15 Debris Resistant Fuel Assembly (DRFA) fuel TO 15x15 Upgrade Fuel has been evaluated. The new Upgrade fuel has different thermal-hydraulic characteristics than the current DRFA fuel, which will affect the LBLOCA transient behavior. The PCT impact of those differences has been evaluated for the Turkey Point Units 3 and 4 fuel transition at the Reference Transient conditions. The conclusion of the transition core evaluation was that Turkey Point Units 3 and 4 remain in compliance with the requirements of 10 CFR 50.46. A 12°F PCT transition core penalty is assessed only for cycles containing both DRFA fuel and Upgrade fuel.

The effects of Optimized ZIRLO™ cladding on the BE LBLOCA analysis described herein have been considered. It has been concluded that The LOCA ZIRLO® models are acceptable for application to Optimized ZIRLO™ cladding in ECCS performance analyses. Therefore, the use of Optimized ZIRLO™ cladding is deemed acceptable for Turkey Point Units 3 and 4. No PCT penalty will be required for the Turkey Point Units 3 and 4 BE LBLOCA analysis with 15x15 Upgrade Fuel when Optimized ZIRLO™ is implemented.

In addition, 10 CFR 50.46 requires that licensees assess and report the effect of changes from, or errors in, the evaluation model used in the large break LOCA analysis. These reports constitute addenda to the analysis of record provided in the UFSAR until the overall changes become significant as defined by 10 CFR 50.46. If the assessed changes or errors in the evaluation model result in significant changes in calculated PCT, a schedule for formal reanalysis or other action as needed to show compliance will be addressed in the report to the NRC. C26

Finally, the criteria of 10 CFR 50.46 requires that holders and users of the evaluation models establish a number of definitions and processes for assessing changes in the models or their use. Westinghouse, in consultation with the PWR Owners Group (PWROG), has developed an approach for compliance with the reporting requirements. FPL provides the NRC with annual and 30-day reports, as applicable, for Turkey Point Units 3 and 4.

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14.3.2.1.6 Conclusions

It must be demonstrated that there is a high level of probability that the limits set forth in 10 CFR 50.46 are met. The demonstration that these limits are met is as follows:

(b)(1) The limiting PCT corresponds to a bounding estimate of the 95^{TH} percentile PCT at the 95-percent confidence level. Since the resulting PCT for the limiting case is 2152°F, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(1), i.e., “Peak Clad Temperature less than 2200°F”, is demonstrated for Turkey Point Units 3 and 4. The results are shown in Table 14.3.2.1-5.

(b)(2) The maximum cladding oxidation corresponds to a bounding estimate of the 95^{th} percentile MLO at the 95-percent confidence level. Since the resulting MLO for the limiting case is 10.46 percent for Turkey Point Units 3 and 4, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2), i.e., “Maximum Local Oxidation of the cladding less than 17 percent” is demonstrated. The results are shown in Table 14.3.2.1-5. C26

(b)(3) The limiting core-wide oxidation corresponds to a bounding estimate of the 95^{th} percentile CWO at the 95-percent confidence level. The limiting Hot Assembly Rod (HAR) total maximum oxidation is 0.40 percent for Turkey Point Units 3 and 4. A detailed CWO calculation takes advantage of the core power census that includes many lower power assemblies. Because there is significant margin to the regulatory limit, the CWO value can be conservatively chosen as that calculated for the limiting HAR. A detailed CWO calculation is therefore not needed because the outcome will always be less than the HAR value. Since the resulting HAR is less than 1.0 percent, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(3), i.e., “Core-Wide Oxidation less than 1 percent”, is demonstrated. The results are shown in Table 14.3.2.1-5.

(b)(4) 10 CFR 50.46 acceptance criterion (b)(4) requires that the calculated changes in core geometry are such that the core remains amenable to cooling. This criterion has historically been satisfied by adherence to criteria (b)(1) and (b)(2), and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. It has been demonstrated that the PCT and maximum cladding oxidation limits remain in effect for Best-Estimate LOCA applications. The approved methodology (Reference 5) specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the core periphery (i.e., at least 1 face on the baffle). This conclusion is based on taking credit for the low power generation in the peripheral assemblies, and the observation that any flow redistribution which may occur would tend to benefit the inboard assemblies. For Turkey Point Units 3 and 4, grid crushing has been predicted to occur only on the core periphery, and low power generation has been confirmed for all core peripheral assemblies. The actions, automatic or manual, that are currently in place at Turkey point Units 3 and 4 to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (Reference 6). Therefore, acceptance criterion (b)(4) is satisfied. C26

(b)(5) 10 CFR 50.46 acceptance criterion (b)(5) requires that long-term core cooling be provided following the successful initial operation of the ECCS. Long-term cooling is dependent on the demonstration of continued delivery of cooling water to the core. The actions, automatic or manual, that are currently in place at Turkey Point Units 3 and 4 to maintain long-term cooling remain unchanged with the application of the ASTRUM methodology (Reference 6). Therefore, acceptance criterion (b)(5) is satisfied.

Based on the ASTRUM Analysis results (Tables 14.3.2.1-4 and 14.3.2.1-5), it is concluded that Turkey Point Units 3 and 4 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

14.3.2.1.7 References

1. 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors,” January 4, 1974.

2. SECY-83-472, Information Report from W. J. Dircks to the Commissioners, “Emergency Core Cooling System Analysis Methods,” November 17, 1983.

3. Regulatory Guide 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” USNRC, May 1989.

4. NUREG/CR-5249, “Qualifying Reactor Safety Margins: Application of Code Scaling Applicability and Uncertainty (CSAU) Evaluation Methodology to a Large Break Loss-of-Coolant-Accident,” B. Boyack, et. al., 1989.

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8. WCAP-13451, “Westinghouse Methodology for Implementation of 10 CFR 50.46 Reporting,” October 1992.

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14.3.2.2 SMALL BREAK LOCA (SMALL RUPTURED PIPES OR CRACKS IN LARGE PIPES) WHICH ACTUATE THE EMERGENCY CORE COOLING SYSTEM

This section presents the results of the 2009 small break loss-of-coolant accident (LOCA) analysis of record performed to support the Turkey Point Units 3 and 4 Extended Power Uprate (EPU) in conformance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50. (Reference 1) C26 C26

14.3.2.2.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A LOCA is defined as a rupture of the reactor coolant system (RCS) piping or of any line connected to that system. A small break, as considered in this section, is defined as a rupture of the RCS piping with a cross sectional area of less than 1.0 ft^2, in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure.

The most limiting single active failure assumed for a small break LOCA is that of an emergency power train failure which results in the loss of one complete train of Emergency Core Cooling System (ECCS) components. In addition, a loss-of-offsite power (LOOP) is assumed to occur coincident with reactor trip. This means that credit may be taken for at most two high head safety injection (SI) pumps and one low head (RHR) pump. These C26 countermeasures limit the consequences of the small break LOCA accident in two ways:

1. Control rod insertion and borated injection (SI) supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

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

Prior to break initiation, the plant is assumed to be in a full power (100.3%) equilibrium condition, i.e., the heat generated in the core is being C26 removed via the secondary system. Other initial plant conditions assumed in the analysis are given in Table 14.3.2.2-1. Subsequent to the break opening, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction and is a function of the relative temperatures of the primary and secondary.

In the case of continuous heat addition to the secondary during a period of quasi-equilibrium, an increase in the secondary system pressure results in steam relief via the steam generator safety valves. Makeup flow to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal main feedwater by closing the feedwater isolation valves and initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps. The heat transferred to the secondary side of the steam generator aids in the reduction of the RCS pressure. C26

During the earlier part of the small break transient (prior to the assumed loss-of-offsite power coincident with reactor trip), the loss of flow through the break is not sufficient enough to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, a partial period of core uncovery occurs. Ultimately, the small break transient analysis is terminated when the ECCS flow provided to the RCS exceeds or is in equilibrium with the break flow rate. C26

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For the break sizes where the RCS depressurizes to approximately 575 psig, the accumulators begin to inject borated water into the reactor coolant loops.

14.3.2.2.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

Small Break LOCA Evaluation Model C26

For small breaks (less than 1.0 ft^2) the NOTRUMP evaluation model (NOTRUMP-EM) (References 9, 10 and 11) was employed to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of the fluid flow through the break. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features, including the calculation of thermal non-equilibrium in all fluid volumes flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. Also, safety injection into the faulted loop is modeled using the COSI condensation model (Reference 11). The NOTRUMP small-break LOCA Emergency Core Cooling System (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs, and to address NRC concerns expressed in NUREG-0611 (Reference 12). C26 C26 C26

The reactor coolant system model is nodalized into volumes interconnected by flow paths. The faulted loop is modeled explicitly, while the two intact loops are lumped into one intact loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multinode capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, provides a proper calculation of the behavior of the loop seal during a loss-of-coolant accident. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations. A more detailed description of the NOTRUMP code, its models, and the associated small break evaluation model is provided in References 10 and 11.

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The fuel rod heat up calculations are performed with the small break LOCA version of LOCTA-IV code (Reference 13) which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height from the NOTRUMP hydraulic calculations as input. For all computations where core uncover occurs, the NOTRUMP and LOCTA runs are terminated after the time the core mixture level reaches the top of the core following uncovery.

A schematic representation of the computer code interface is given in Figure 14.3.2.2-1.

Input Parameters and Initial Conditions

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Significant input parameters are given in Table 14.3.2.2-1.

The SBLOCA analysis assumes that reactor trip occurs coincident with the LOOP, which results in the following: (a) Reactor Coolant Pump (RCP) trip and coastdown and (b) Steam Dump System being inoperable.

The SBLOCA analysis assumes that coolable core geometry is maintained as long as grid crush remains in peripheral assembly locations for homogenous cores of 15x15 Upgrade fuel.

The SBLOCA analysis is performed with a high nominal vessel average temperature (T_{AVG}) value of 583°F. The high T_{AVG} value chosen is evaluated to be applicable over the range of nominal vessel average temperature values (570.0°F to 583.0°F)

After the small break LOCA is initiated, reactor trip is calculated to occur due to a low pressurizer pressure reactor trip signal at 1805 psia (including uncertainties). Following the reactor trip signal, the safety injection signal is calculated to actuate due to a low pressurizer pressure safety injection signal of 1615 psia (including uncertainties). A rod drop time of 2.0 seconds was assumed in addition to a 2.2 second signal processing delay time, resulting in a total delay time of 4.4 seconds from the time of the low pressurizer pressure reactor trip signal (1805 psia) to full rod insertion. The onset of full safety injection flow was assumed to be delayed 45 seconds following the occurrence of the injection signal to account for emergency diesel generator startup and emergency power bus loading in the case of a loss-of-offsite-power coincident with a LOCA.

The safety injection systems consist of accumulator tanks pressurized with Nitrogen gas, and pumped injection systems. The small break LOCA analysis assumed an accumulator water volume of 892 ft^3 with a cover gas pressure of 575 psig. The minimum pumped safety injection capability assumed for the analysis is from at most two HHSI and one RHR pump. For break sizes less C26
than the accumulator line diameter (i.e., break sizes less than 8.75-inches), the faulted loop safety injection flow is assumed to “spill” to RCS pressure. The flow assumed for these break sizes is from two HHSI pumps for both the injection and cold leg recirculation phases of the transient.

For the accumulator line break (8.75 inch equivalent diameter), the faulted loop safety injection flow is assumed to “spill” to containment back pressure (0 psig). Safety injection during the injection phase of the transient is from 2 HHSI pumps and one RHR pump and flow from only one HHSI pump is assumed during the cold leg recirculation phase of the transient.

The hot rod axial power shape used to perform the small break LOCA analysis is given in Figure 14.3.2.2-2. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core. Such a distribution is limiting for small-break LOCAs, because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The small break LOCA analysis assumes full power operation of the core until the control rods are completely inserted.

Results - Limiting Break Case

This section presents the results of the limiting small break LOCA analysis, as determined by the highest calculated peak cladding temperature (PCT). A break spectrum considering 1.5 inch, 2 inch, 3 inch, 4 inch, 6 inch, and 8.75 inch (accumulator line) breaks was performed. The limiting break for the Turkey Point Units was found to be a 4 inch diameter break with a peak cladding temperature of 1231°F (Table 14.3.2.2-2). A summary of the key transient event times is given in Table 14.3.2.2-3.

A summary of the transient response for the limiting 4 inch break case is shown in Figures 14.3.2.2-6 through 14.3.2.2-14. The following transient parameters are presented for this break size.

- Reactor Coolant System Pressure
- Core Mixture Level
- Cladding Temperature Transient at PCT Elevation
- Total Reactor Coolant System Mass
- Core Exit Vapor Temperature
- Vapor Mass Flow Rate Out of the Top of the Core
- Total Break Flow and Safety Injection Flow
- Cladding Surface Heat Transfer Coefficient at PCT Elevation
- Fluid Temperature at PCT Elevation

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From the cladding temperature transient for the 4 inch break given in Figure 14.3.2.2-8, it is seen that the peak cladding temperature (PCT) occurs near the time when the core is most deeply uncovered (Figure 14.3.2.2-7) and the top of the core is being cooled by steam. This time is characterized by the highest vapor superheating above the mixture level (refer to Figure 14.3.2.2-11). In addition, Figures 14.3.2.2-13 and 14.3.2.2-14 provide the hot rod surface heat transfer coefficient at the hot spot and fluid temperature at the hot spot, respectively.

A comparison of the pumped safety injection system flow to the total break mass flow rate at the end of the transient (Figure 14.3.2.2-11), shows that at the time the transient was terminated, the safety injection flow rate that was delivered to the RCS exceeds the mass flow rate out the break. In addition, the core mixture level has recovered to the top of the core (Figure 14.3.2.2-7).

Additional Break Cases

To ensure that the 4 inch diameter break was indeed the most limiting, calculations were also performed with break equivalent diameters of 1.5, 2, 3, 6, and 8.75 inches. The results of each of these cases are given in Tables 14.3.2.2-2 and 14.3.2.2-3.

Plots of the following parameters for each non-limiting break size are given in Figures 14.3.2.2-15 through 14.3.2.2-17 for the 1.5 inch break, Figures 14.3.2.2-18 through 14.3.2.2-21 for the 2 inch break, Figures 14.3.2.2-22 through 14.3.2.2-25 for the 3 inch break, Figures 14.3.2.2-26 through 14.3.2.2-29 for the 6-inch break, and Figures 14.3.2.2-30 through 14.3.2.2-32 for the 8.75 inch accumulator line break. Fuel rod heat up calculations were not performed for the 1.5 inch or 8.75 inch break sizes because core uncovery for these breaks was either minimal or did not occur. The following transient parameters are presented for the non-limiting breaks: C26

- Reactor Coolant System Pressure
- Core Mixture Level
- Core Exit Vapor Temperature
- Cladding Temperature Transient at PCT Elevation (2 inch, and 6 inch break sizes only)

As seen in Table 14.3.2.2-2, the peak cladding temperature for each of these non-limiting break sizes was calculated to be less than that for the 4 inch break case.

The small break LOCA analysis presented herein shows that the available flow from the high head safety injection and RHR pumps along with the accumulators provide sufficient core heat removal and that the 10 CFR 50.46 acceptance criteria (Reference 1) are met. The peak cladding temperature is less than 2200°F; the maximum local oxidation is less than 17 percent; the core wide hydrogen generation is less than 1 percent: the core geometry remains amenable
to cooling; and the core temperature is maintained at an acceptably low value by the time the transient is terminated for all cases examined herein.

14.3.2.2.3 CONCLUSIONS - SMALL BREAK LOCA ANALYSIS

For small breaks in the reactor coolant system pipe up to a cross sectional area of less than 1.0 ft^2, the Emergency Core Cooling System will meet the Acceptance Criteria presented to 10 CFR 50.46. That is:

1. The calculated peak fuel cladding temperature provides for a substantial margin to the requirement of 2200°F.

2. The amount of fuel cladding that reacts chemically with the water or steam does not exceed 1% of the hypothetical amount that would be generated if all the zirconium metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

3. The localized cladding oxidation limit of 17% is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the LOCA.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat produced by the long-lived radioactivity remaining in the core.

14.3.2.3 POST-LOCA LONG TERM COOLING (AFTER A SMALL BREAK OR LARGE BREAK LOCA)

This section presents the results of the post-LOCA long term cooling analysis of record performed to support the Turkey Point Unit 3 and Unit 4 uprating in conformance with 10 CFR 50.46 and Appendix K to 10 CFR 50.

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14.3.2.3.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A LOCA is defined as a rupture of the reactor coolant system (RCS) piping or of any line connected to that system. Post-LOCA long term cooling covers the entire spectrum of break sizes and begins upon the transfer to sump recirculation.

The most limiting single active failure is assumed. Generally, this is an assumed loss of one complete train of Emergency Core Cooling System (ECCS) components. In addition, a loss-of-offsite power (LOOP) is assumed to occur coincident with the reactor trip. Due to the single failures assumed and the structure of the emergency operating procedures, this means that credit may be taken for at most two high head safety injection (HHSI) pumps and one low head residual heat removal (RHR) pump.

Post-LOCA long term cooling consists of three distinct areas: Subcriticality, decay heat removal, and boric acid precipitation control. These countermeasures limit the consequences of the LOCA in the following ways:

1. Borated safety injection originally provided by the refueling water storage tank (RWST) and accumulators is re-circulated through the reactor coolant system (RCS) via the safety injection (SI) pumps. During post-LOCA, the core is maintained in a shutdown state by borated water.

2. After the RWST deliverable volume has been drained, the ECCS pumps enter into recirculation mode where borated water is drawn from the containment sump and is cooled in the residual heat removal heat exchangers. Therefore, post-LOCA long term cooling decay heat removal is maintained by the ECCS. C26

3. A time to initiate an active dilution mechanism to halt and reverse the concentration of boric acid in the core prior to reaching the solubility limit is determined. Hence, post-LOCA boric acid precipitation control is maintained in the long term.

The post-LOCA long term cooling scenario begins at the transfer to sump recirculation. The initial LOCA transients have been terminated and the RCS is assumed to be in a quasi-steady pool boiling state.

14.3.2.3.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

<u>Method of Analysis</u>

A post-LOCA Subcriticality boron limit was developed for the uprated conditions. The post-LOCA Subcriticality analysis is used to demonstrate that the core will remain subcritical post-LOCA provided that the cycle-specific maximum critical boron concentration remains below the post-LOCA sump boron concentration limit. The core will remain subcritical and the only heat generation will be that due to the remaining long-lived radioactivity.

Turkey Point Unit 3 and Unit 4 utilize safety injection (SI) during recirculation to remove decay heat and provide boric acid precipitation control. The boric acid precipitation control analysis determines the latest acceptable time to initiate an active dilution mechanism to halt and reverse the concentration of boric acid in the core. The decay heat removal analysis determines the earliest acceptable time to initiate hot leg switchover while still meeting minimum flow requirements in order to prevent core uncovery and to effectively remove decay heat. Due to the unique features of the Turkey Point Unit 3 and Unit 4 ECCS design, continuous cycling between hot leg and cold leg injection is required until boiling in the core has been terminated.

Input Parameters and Initial Conditions

The sump boron concentration model used to show that Subcriticality is maintained in the long term is based on the following assumptions: C26

- The calculation of the sump mixed mean boron concentration assumes minimum mass and minimum boron concentrations for significant boron sources and maximum mass and minimum boron concentrations for significant dilution sources.

- Boron is mixed uniformly in the sump. The post-LOCA sump inventory is made up of constituents that are equally likely to return to the containment sump; that is selective holdup in containment is neglected.

- The sump mixed mean boron concentration is calculated as a function of the pre-trip RCS conditions.

- Core reactivity is evaluated assuming all-rods-out (ARO) and no Xenon (NOXE). However, boric acid accumulation in the reactor vessel could dilute the containment sump and introduce positive reactivity at the onset of hot leg recirculation. Generic Westinghouse assessment of this condition (Reference 2) has shown that negative reactivity from control rod insertion and post-accident xenon buildup would adequately offset the boron dilution effect, so no separate analysis has been performed for subcriticality with diluted sump boron conditions.

The boric acid precipitation and decay heat models meet NRC guidance relative to the interim methodology (Reference 1) and is based on the following assumptions:

- The boric acid concentration in the core region was computed over time with consideration of the effect of core voiding on liquid volumes.

- The boric acid concentration limit is the experimentally determined boric acid solubility limit of 29.27 weight percent. For large breaks, the effect of containment on RCS pressure above atmospheric pressure is not credited and the boric acid solubility limit at the boiling point of saturated boric acid solution is assumed. For breaks where RCS depressurization is not complete (where the RCS might remain at elevated pressures), the solubility limit associated with the saturation temperature of water at the associated elevated pressure is not credited. Credit for the solubility limit at elevated pressures is taken when analyzing the maximum allowable cooldown rate.

- The liquid mixing volume used in the calculation includes 50 percent of the lower plenum. C26

- The effect of containment sump pH additives on increasing the boric acid solubility limit is not credited.

- The boric acid concentration of the containment sump during recirculation is calculated mixed mean boric acid concentration that assumes maximum mass and maximum boron concentrations for significant boron sources, and minimum mass and maximum boron concentrations for significant dilution sources.

- NRC requirements pertaining to the decay heat generation rate for both boric acid accumulations and decay heat removal (1971 ANS Standard for an infinite operating time with 20 percent uncertainty) is considered when performing the boric acid precipitation calculations. The assumed core power includes a multiplier to address uncertainty as identified by Section 1.A of 10 CFR 50, Appendix K.

- ECCS recirculation flows are evaluated by comparing the limiting single-failure minimum safety injection pump flows to the flows necessary to dilute the core and replace boil-off, thus keeping the core quenched and amenable to cooling.

14.3.2.3.3 RESULTS – POST-LOCA LONG TERM COOLING ANALYSIS

The results of the long-term cooling Subcriticality analysis are confirmed as acceptable on a cycle-specific basis as part of the Westinghouse Reload Safety Evaluation (RSE) Methodology.

The results of the long-term cooling boric acid precipitation control and decay heat removal analysis include operator action times to both halt and reverse the concentration of boric acid and to ensure adequate removal of decay heat. Cold leg recirculation provided by 2 HHSI pumps is initiated at 45 minutes after the event. Hot leg recirculation is initiated at 5.5 hrs and completed by 6.5 hrs after the event. ECCS shall be cycled back to the cold legs by 17 hours into the event with all subsequent cycling occurring on 16 hour intervals until boiling in the core is terminated.

14.3.2.3.4 CONCLUSIONS – POST-LOCA LONG TERM COOLING ANALYSIS

For the full spectrum of breaks in the RCS and after any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptable low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core. The results of the long-term cooling Subcriticality analysis are confirmed as acceptable on a cycle-specific basis as part of the Westinghouse Reload Safety Evaluation (RSE) Methodology. C26

<u>Supplemental Evaluations</u>

The USNRC had several particular areas of interest relative to the post-LOCA long-term cooling analysis that are summarized here:

1. Condensation Efficiency: The post-LOCA boric acid precipitation analysis model assumes that vapor generated in the core returns to the sump as unborated liquid (i.e., 100% condensation efficiency). This assumption is based on the limited capacity of the containment atmosphere to hold water vapor mass relative to the total water mass in the containment sump. Results of this assumption are a steady dilution of the containment sump. As demonstrated to the NRC, a relaxation of the condensation efficiency assumption relative to the most conservative value (i.e., 0% condensation efficiency) is within the conservatism of the calculation and there is still margin to the solubility limit at atmospheric conditions at the Hot Leg Switchover (HLSO) time of 5.5 hours.

2. Lower Plenum Mixing/Transport: The mixing volume in the boric acid precipitation calculation includes 50% of the lower plenum. This is based upon testing observations at the BACCHUS test facility. Mixing between the core region (212°F) and lower plenum region (150°F) was observed to initiate when the boric acid gradient between the regions is roughly 8.5 percent. Once the inception criterion is met, mixing and transport between the core and lower plenum region continues through the rest of the transient.

3. SBLOCA Concerns: The maximum allowable cooldown rate of the RCS is 100°F/hr. If adequate depressurization does not occur early in the transient and the system pressure remains at or above 120 psia, boric acid precipitation in the event of a rapid cooldown due to a late initiation of HLSO is shown not to occur. It was demonstrated that the RCS did not cool down and depressurize faster than the hot leg injection was capable of diluting the core and mitigating boric acid precipitation.

14.3.2.4 REFERENCES C26

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Table 14.3.2.1-1

Plant Operating Range
Analyzed by the Best-Estimate Large-Break LOCA Analysis for
Turkey Point Units 3 and 4

	Parameter	As-Analyzed Value or Range
1.0	**Plant Physical Description**	
	a) Dimensions	Nominal
	b) Pressurizer location	On an intact loop
	c) Hot assembly location	Anywhere in core(1)
	d) Hot assembly type(2)	15x15 Upgrade Fuel design
	e) Steam generator tube plugging level	≤ 5%
	f) Fuel assembly type(2)	15x15 Upgrade Fuel with ZIRLO® or Optimized ZIRLO™ cladding, non-IFBA or IFBA, IFMs
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Core power	≤ 100% of 2652 MWt
	b) Peak heat flux hot channel factor (F_Q)(2)	Table 14.3.2.1-8
	c) Peak hot rod enthalpy rise hot channel factor ($F_{\Delta H}$)(2)	Table 14.3.2.1-8
	d) Hot assembly radial peaking factor (P_{HA})(2)	$F\Delta H/1.04$
	e) Hot assembly heat flux hot channel factor (F_{QHA})	$F_Q/1.04$
	f) Axial power distribution (P_{BOT}, P_{MID})(2)	Figure 14.3.2.1-12
	g) Low power region relative power (P_{LOW})(2)	$0.2 \leq P_{LOW} \leq 0.8$
	h) Hot assembly burnup	≤ 75,000 MWD/MTU, lead rod(1), (4)
	i) MTC	≤ 0 at hot full power (HFP)
	j) Typical cycle length	18 months
	k) Minimum core average burnup(2)	≥ 10,000 MWD/MTU
	l) Maximum steady state depletion, F_Q(2)	Table 14.3.2.1-8
	2.2 Fluid Conditions	
	a) T_{AVG}	577 – 6°F ≤ T_{AVG} ≤ 583 + 6°F
	b) Pressurizer pressure	2250 – 53 psia ≤ P_{RCS} ≤ 2250 + 53 psia
	c) Loop flow	TDF ≥ 86,900 gpm/loop(6)
	d) Upper head design	T_{HOT}
	e) Pressurizer level	607.15 to 765.78 ft^3
	f) Accumulator temperature	85°F ≤ T_{ACC} ≤ 126°F
	g) Accumulator pressure	589.7 psia ≤ P_{ACC} ≤ 714.7 psia

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Plant Operating Range
Analyzed by the Best-Estimate Large-Break LOCA Analysis for
Turkey Point Units 3 and 4

Parameter		As-Analyzed Value or Range
	h) Accumulator liquid volume	$872\ ft^3 \leq V_{ACC} \leq 920\ ft^3$
	i) Accumulator fL/D	5.477(3)
	j) Minimum accumulator boron	≥ 2300 ppm
3.0	**Accident Boundary Conditions**	
	a) Minimum safety injection flow	Table 14.3.2.1-6
	b) Safety injection temperature	34°F ≤ SI Temp ≤ 105°F
	c) Safety injection delay	HHSI: 17 seconds with offsite power or 35 seconds with LOOP LHSI: 23 seconds with offsite power or 35 seconds with LOOP
	d) Containment modeling	See Figure 14.3.2.1-13 and raw data in Tables 14.3.2.1-2 and 14.3.2.1-3
	e) Minimum containment air partial pressure	See Table 14.3.2.1-2
	f) Containment spray initiation delay	See Table 14.3.2.1-2
	g) Recirculation spray initiation delay	Not modeled
	h) Single failure	ECCS: Loss of 1 HHSI and 1 LHSI(5)

Notes:

1. Core peripheral locations will not physically be the lead power assembly.
2. In the Westinghouse Reload Safety Analysis Checklist (RSAC) process, this parameter is identified as a key safety analysis parameter that could be impacted by a fuel reload.
3. fL/D based on average L/D of 405.67.
4. The fuel temperature and rod internal pressure data is only provided up to 62,000 MWD/MTU. In addition, the hot assembly/ hot rod will not have a burnup this high in ASTRUM analyses.
5. The ECCS single failure criteria is considered to be the standard assumption of loss of 1 HHSI and 1 LHSI. However, the Turkey Point model credits an additional HHSI line (2 HHSI and 1 LHSI total) to account for the cross-tie between Turkey Points Units 3 and 4.
6. TDF of 86,900gpm, consistent with the Performance Capability Working Group (PCWG) and other Safety Analysis groups, is supported for all SGTP levels from 0 to 10%. C27

Table 14.3.2.1-2

Large-Break LOCA Containment Data
Used for Calculation of Containment Pressure for
Turkey Point Units 3 and 4

Containment Net Free Volume	1,600,000 ft^3
Initial Conditions	
Minimum air initial containment partial pressure at full power operation	13.26 psia[(1)]
Minimum steam initial containment partial pressure at full power operation	0.69 psia
Minimum initial containment temperature at full power operation	90.0°F
RWST temperature	39°F
Temperature outside containment	39°F
Initial spray temperature	39°F
Spray System	
Number of containment spray pumps operating	2
Post-accident containment spray system initiation delay	11.0 sec
Maximum spray system flow from all containment spray pumps	3520 gal/min.
Fan Coolers	
Maximum number of containment fan coolers in operation	2
Post-accident fan cooler system initiation delay	11.0 sec
Fan Cooler Heat Removal Rate	See Table 14.3.2.1-7
Recirculation Spray	Not Modeled

Note:

1. Reduced from 14.7 psia to account for containment purge.

C26

Table 14.3.2.1-3

Large-Break LOCA Containment Structural Data
Used for Calculation of Containment Pressure for
Turkey Point Units 3 and 4[1]

Security-Related Information - Withheld Under 10 CFR 2.390

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Table 14.3.2.1-3

Large-Break LOCA Containment Structural Data Used for Calculation of Containment Pressure for Turkey Point Units 3 and 4[(1)]

Security-Related Information - Withheld Under 10 CFR 2.390

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Table 14.3.2.1-3

Large-Break LOCA Containment Structural Data
Used for Calculation of Containment Pressure for
Turkey Point Units 3 and 4(1)

Wall	Material Type	Area (ft^2)	Thickness (ft)
48	S-Steel	4.09	0.15
49	S-Steel	41.44	0.2671
50	Paint	510.99	0.000583
	C-Steel	510.99	0.0165
51	Paint	230.01	0.000583
	C-Steel	230.01	0.025
52	Paint	182.92	0.000583
	C-Steel	182.92	0.10
53	S-Steel	5713.78	0.0057
54	S-Steel	28.80	0.0375
55	C-Steel	11773.20	0.0154
56	C-Steel	3259.09	0.0243
57	Paint	802.88	0.000583
	C-Steel	802.88	0.7209
58	S-Steel	28.61	0.45

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Note:

1. All values provided in this table correspond with the analyzed values. Revised heat sink data was provided per customer letter EPU-PTN-12-0021. It was determined that the existing WCOBRA/TRAC containment pressure transient used in the Turkey Point Units 3 and 4 ASTRUM analysis remains conservative/applicable and there is no impact on PCT, MLO, and CWO results.

Table 14.3.2.1-4

Peak Clad Temperature (PCT) Including All Penalties and Benefits,
Best-Estimate Large Break LOCA (BE LBLOCA)
For Turkey Point Units 3 and 4

PCT for Analysis-of-Record (AOR)	**2152 °F**
Transition Core Evaluation	12 °F [(1)]
BE LBLOCA PCT for Comparison to 10 CFR 50.46 Requirements	2164 °F

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Note:

1. This PCT penalty is only applicable during a transition core configuration.

Table 14.3.2.1-5

Turkey Point Units 3 and 4
Best-Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	Criteria
95/95 PCT[1] (°F)	2152	< 2,200
95/95 MLO[2] (%)	10.46	< 17
95/95 CWO[3] (%)	0.40	< 1

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1 Peak Cladding Temperature
2 Maximum Local Oxidation
3 Core-Wide Oxidation

Table 14.3.2.1-6

Safety Injection Flow Rates
Used in Best-Estimate Large-Break LOCA Analysis for
Turkey Point Units 3 and 4

RCS Pressure (psia)	Total HHSI – 2 pumps (gpm)	Total LHSI – 1 pump (gpm)
14.7	617	1914
19.7	615.25	1876
24.7	613.5	1836
29.7	611.75	1370
34.7	610	942
39.7	608	540
44.7	606	155
46.7	605.2	7
47.7	604.8	0
54.7	602	0
74.7	595	0
94.7	587	0
114.7	579	0
134.7	572	0
154.7	564	0
174.7	556	0
194.7	548	0
214.7	540	0
314.7	500	0
364.7	0	0

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Table 14.3.2.1-7
Fan Cooler Performance Data
Used in Best-Estimate Large Break LOCA Analysis for
Turkey Point Units 3 and 4

Containment Temperature (°F)	Heat Removal Rate (MBTU/hr)
110	0.0
120	1.554
140	3.770
160	6.906
180	11.221
200	17.353
220	25.352
240	36.492
260	50.880
283	73.125

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Note :

A conservative 100°F minimum Component Cooling Water (CCW) temperature was assumed.

Table 14.3.2.1-8

Summary of Peaking Factor Margin and Burndown
Supported by Best-Estimate Large Break LOCA Analysis for
Turkey Point Units 3 & 4

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Hot Rod Burnup (GWD/MTU)	**FdH (with uncertainties)**	**FQ Transient (with uncertainties)**	**FQ Steady-state (without uncertainties)**
0	1.6	2.3	1.9
30	1.6	2.3	1.9
49	1.33	1.84	1.52
65	1.33	1.84	1.52

Table 14.3.2.2-1

Input Parameters Used in the Small Break LOCA Analysis

Parameter	Value
100% Licensed Core Power (MWt)(1)	2644
Peak Linear Power (kW/ft)	16.18
Total Peaking Factor [F_Q]	2.40
Axial Peaking Factor [F_Z]	1.45
Hot Channel Enthalpy Rise Factor [$F_{\Delta H}$]	1.65
Hot Assembly Peaking Factor [PHA]	1.515
Axial Power Shape(2)	Figure 14.3.2.2-2
Fuel Type(3)	15x15 Upgrade Fuel
Accumulator Water Volume (ft^3)	892
Accumulator Tank Volume (ft^3)	1200
Accumulator Gas Pressure, Minimum (including uncertainties), (psia)	589.7
Pumped Safety Injection Flow	Figure 14.3.2.2-3 Figure 14.3.2.2-4 Figure 14.3.2.2-5
Steam Generator Tube Plugging Level (%)(4)	10
Thermal Design Flow (gpm/loop)	86,900
Nominal Vessel Average Temperature (°F)	583.0
Reactor Coolant Pressure (including uncertainties), (psia)	2303
AFW Flow (minimum) per Steam Generator (gpm/SG)	66.7

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Notes:

1. Analyzed core power of 2652 MWt includes an additional 0.3% to account for calorimetric uncertainty. Reactor coolant pump heat is not modeled in the small break LOCA analysis.

2. This represents a power shape corresponding to a one-line segment peaking factor envelope, K(z), based on FQ = 2.40.

3. Analysis performed considering ZIRLO® cladding; evaluation performed addressing Optimized ZIRLO™ cladding.

4. Maximum plugging in any one or all steam generators

Table 14.3.2.2-2

Small Break LOCA Analysis Fuel Cladding Results

Results (1)	**1.5-Inch**	**2-Inch**	**3-Inch**	**4-Inch**	**6-Inch**	**8.75-Inch**
PCT, °F	N/A (2)	1003	1086	1231	658	N/A (2)
PCT Time, sec		2430.4	1297.9	752.5	348.9	
PCT Elevation, ft		11.00	11.25	11.00	10.75	
Maximum ZrO_2, %		0.02	0.04	0.07	0.00	
Maximum ZrO_2 Elevation, ft		11.00	11.25	11.00	10.75	
Average ZrO_2, %		0.00	0.01	0.01	0.00	

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Notes:

1. Neither the hot rod nor the hot assembly average rod burst during the fuel rod heat-up calculations.
2. The core does not uncover or uncovers for an insignificantly short time; therefore fuel rod heat-up calculations are not warranted for these break sizes.

Table 14.3.2.2-3

Small Break LOCA Analysis Time Sequence of Events

Event (sec)	1.5-Inch[3]	2-Inch	3-Inch	4-Inch	6-Inch	8.75-Inch[3]
Transient Initiated	0	0	0	0	0	0
Reactor Trip Signal	205.9	38.9	16.1	9.5	6.0	5.0
Safety Injection Signal	221.8	56.6	26.6	19.1	13.8	8.3
Safety Injection Begins[1]	266.8	101.6	71.6	64.1	58.8	53.3
Loop Seal Clearing Occurs[2]	1582	879	401	204	50	16
Top of Core Uncovered	N/A	1398	575	435	312	N/A
Accumulator Injection Begins	N/A[4]	N/A[4]	1285	599	262	126
Top of Core Recovered	N/A	3239	2170	1266	364	N/A
RWST Low-Low-Level	4006.4	3902.1	3693.9	3592.7	3518.3	2172.1

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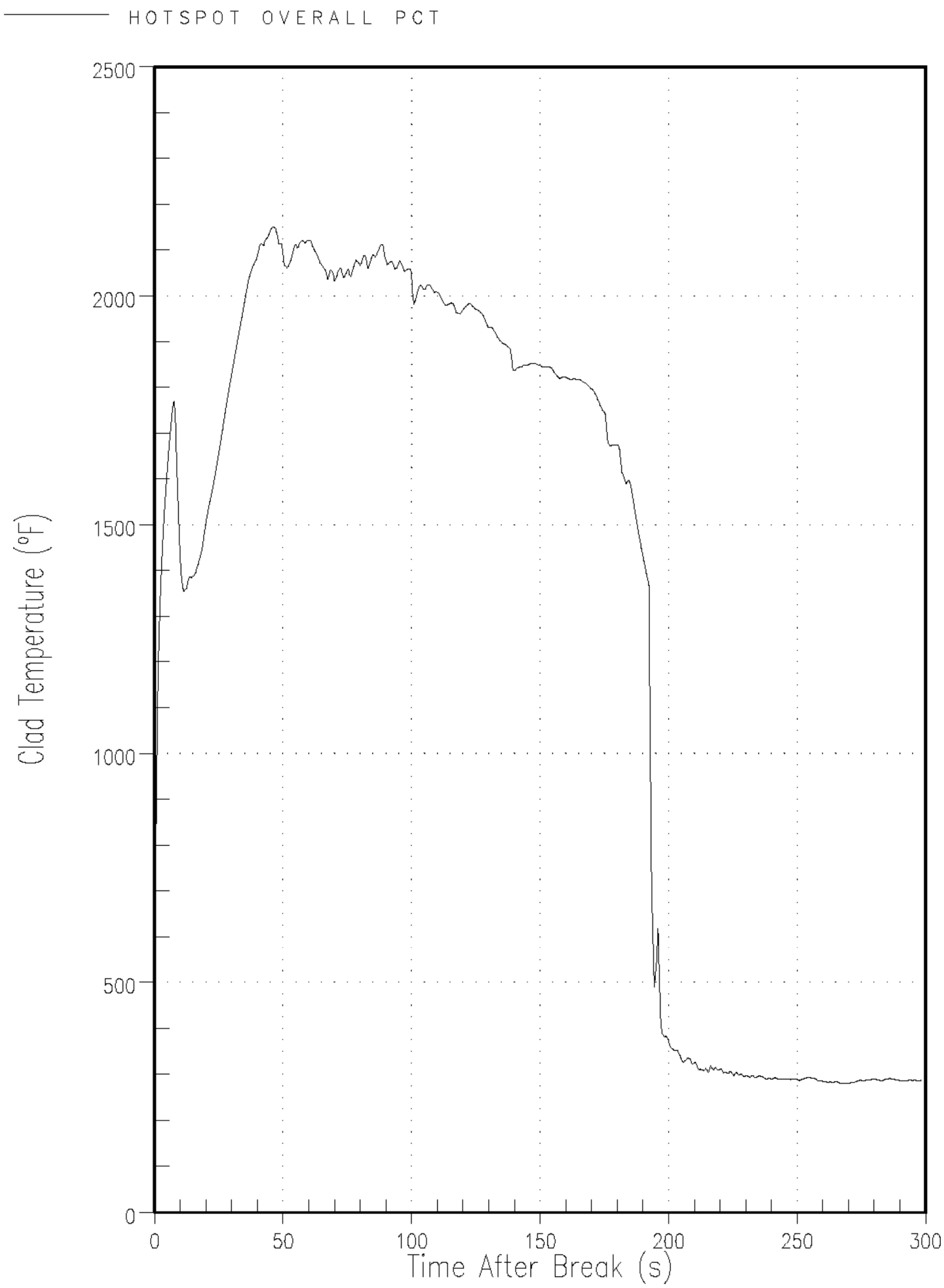
Notes:

1. Safety injection begins 45 seconds after the SI signal is generated.
2. Loop seal clearing is considered to occur when the faulted loop loop seal vapor flow rate is sustained above 1 lbm/sec and the mixture level is at or below the top of the loop seal.
3. There is either no core uncovery or minimal core uncovery for these break sizes.
4. Accumulator injection does not occur for these break sizes.



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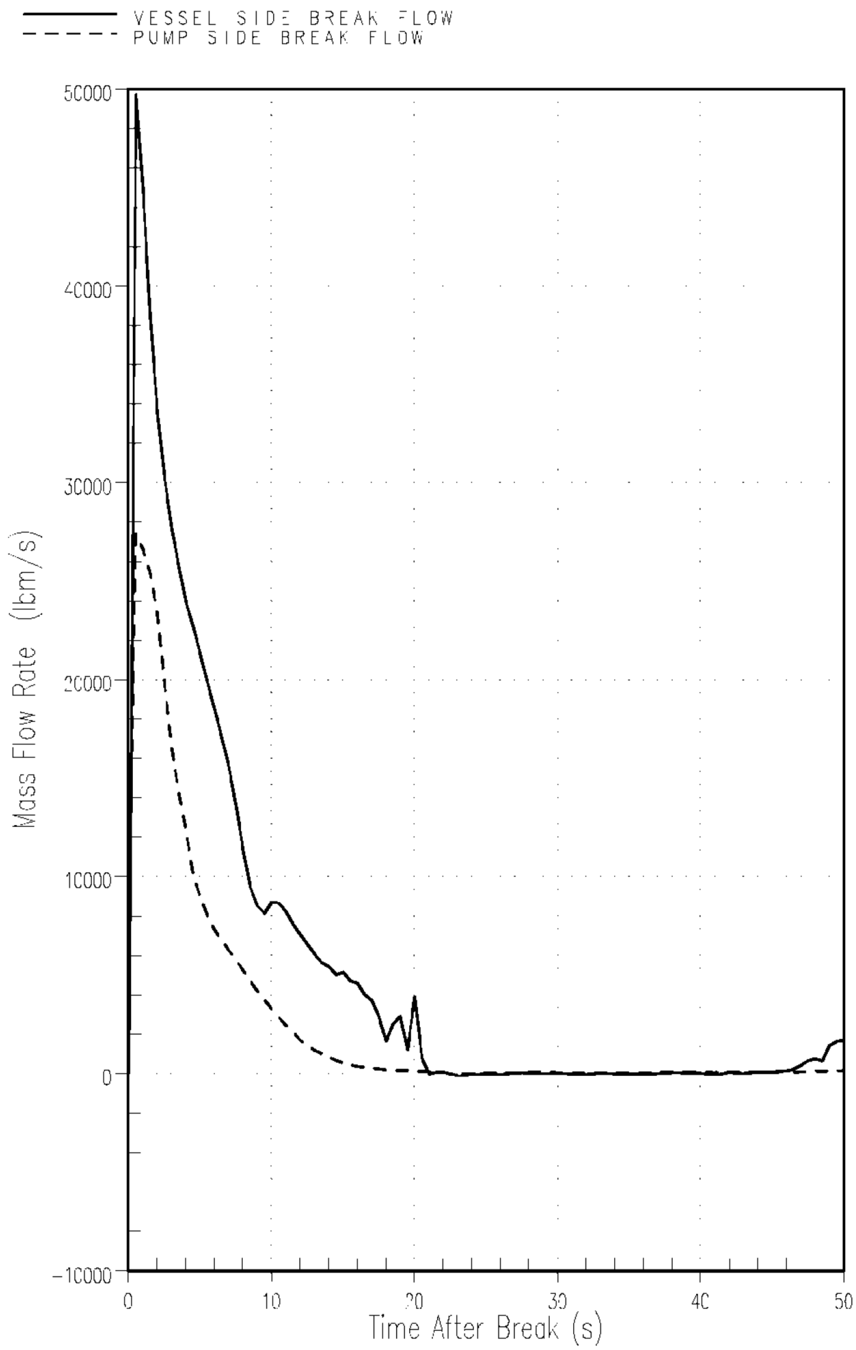
LIMITING PCT CASE HOTSPOT PCT

FIGURE 14.3.2.1-1



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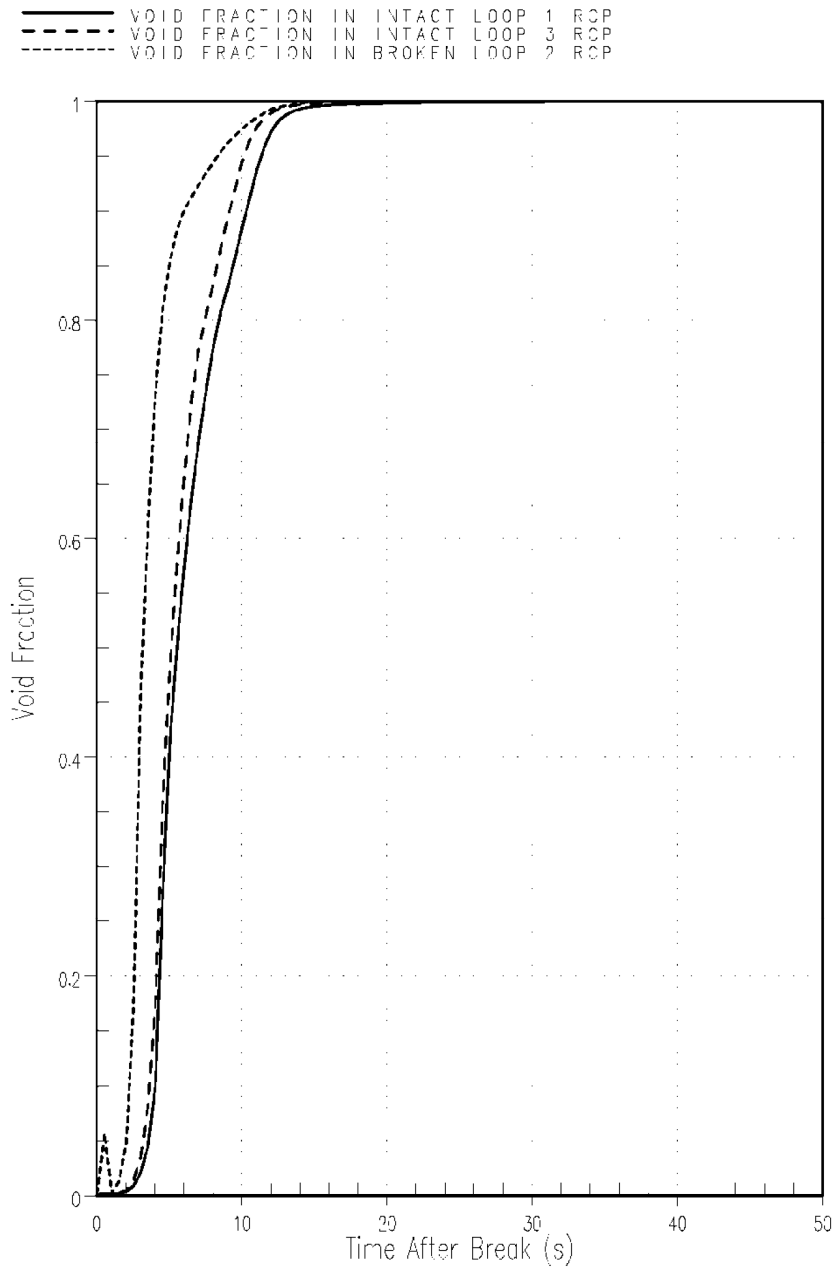
LIMITING PCT CASE VESSEL
AND PUMP SIDE BREAK FLOW

FIGURE 14.3.2.1-2



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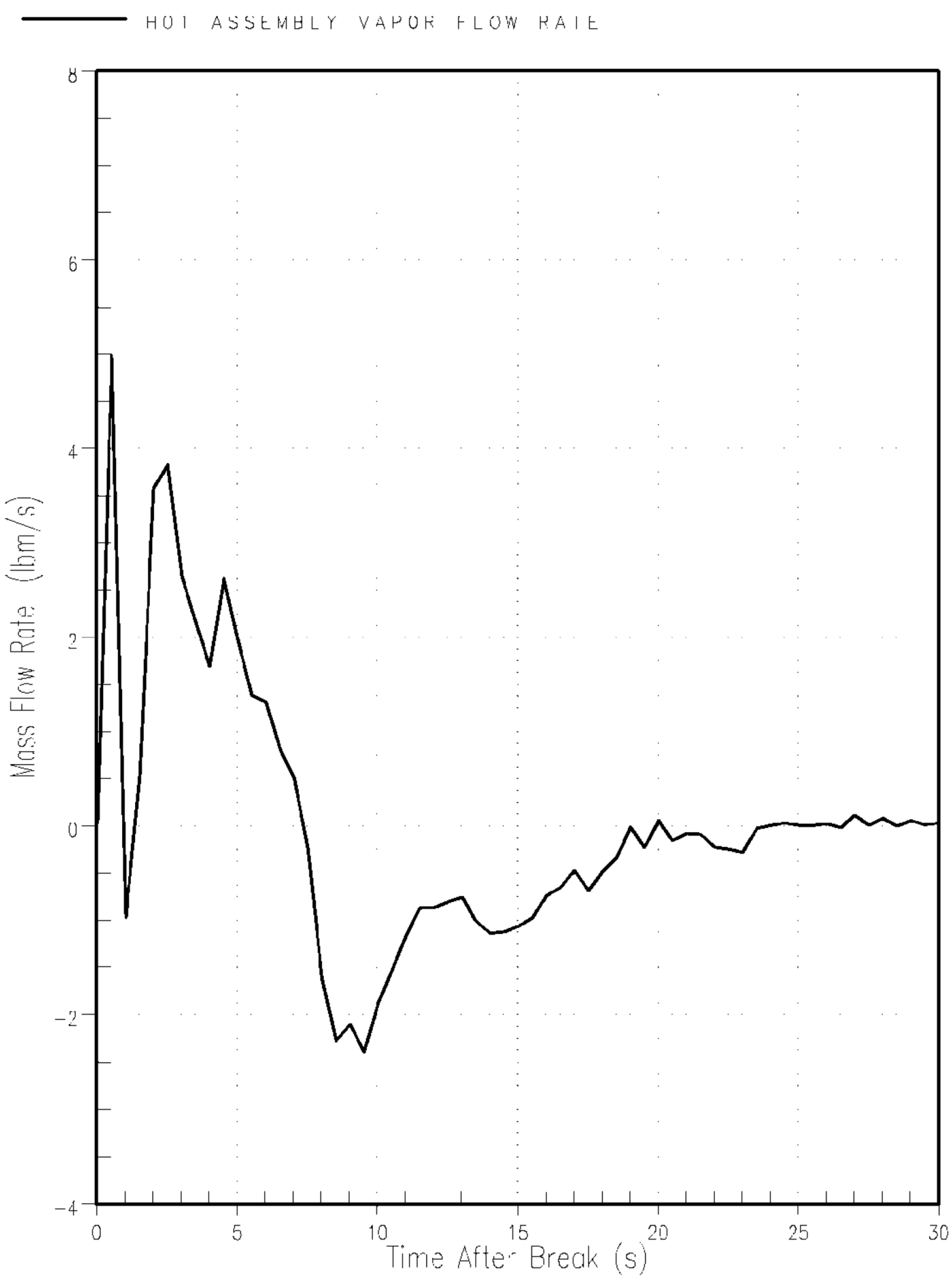
LIMITING PCT CASE BROKEN
AND INTACT LOOP RCP VOID FRACTION

FIGURE 14.3.2.1-3



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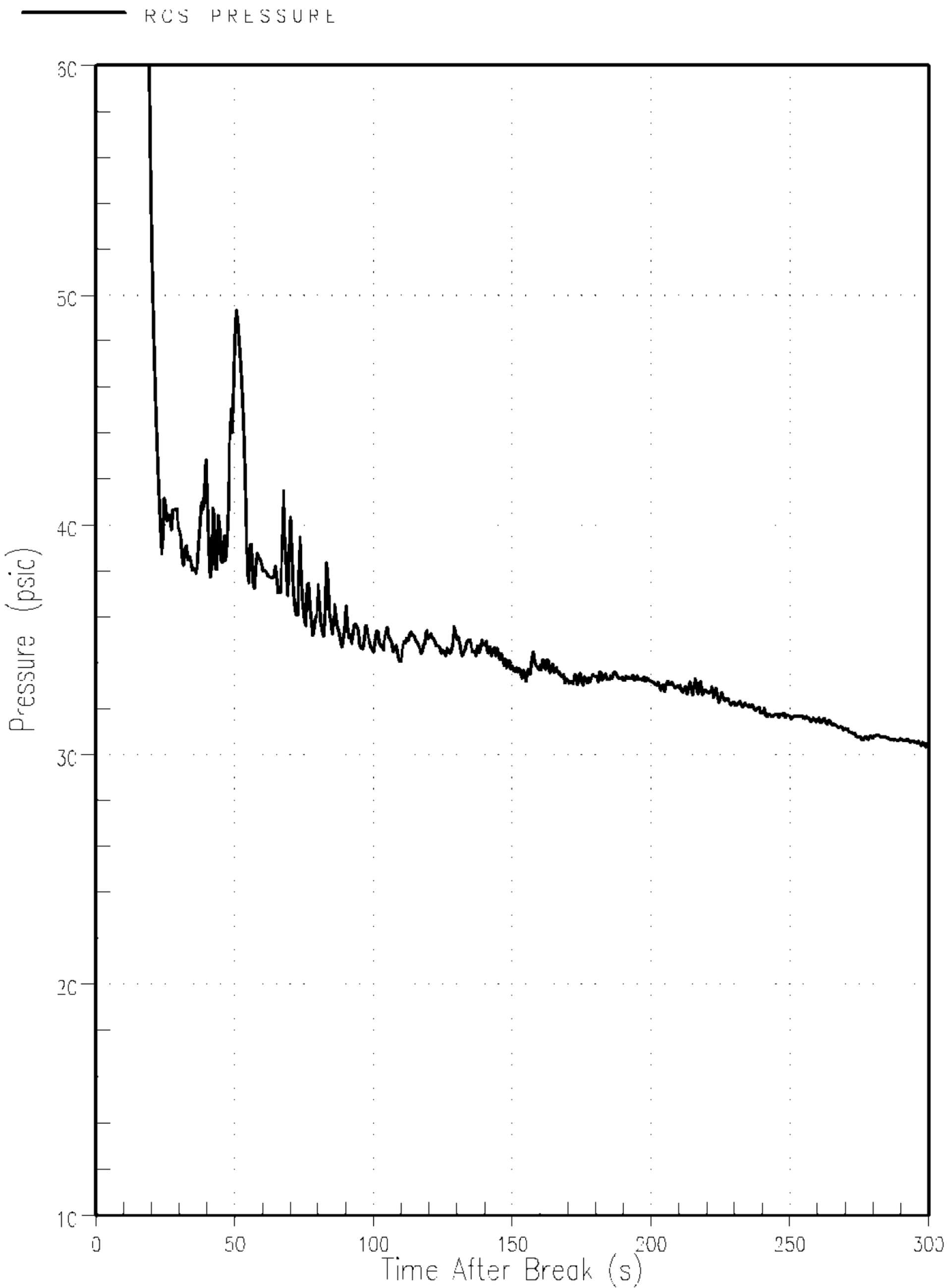
LIMITING PCT CASE HOT ASSEMBLY VAPOR FLOW IN THE TOP THIRD OF THE CORE

FIGURE 14.3.2.1-4



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LIMITING PCT CASE RCS PRESSURE

FIGURE 14.3.2.1-5

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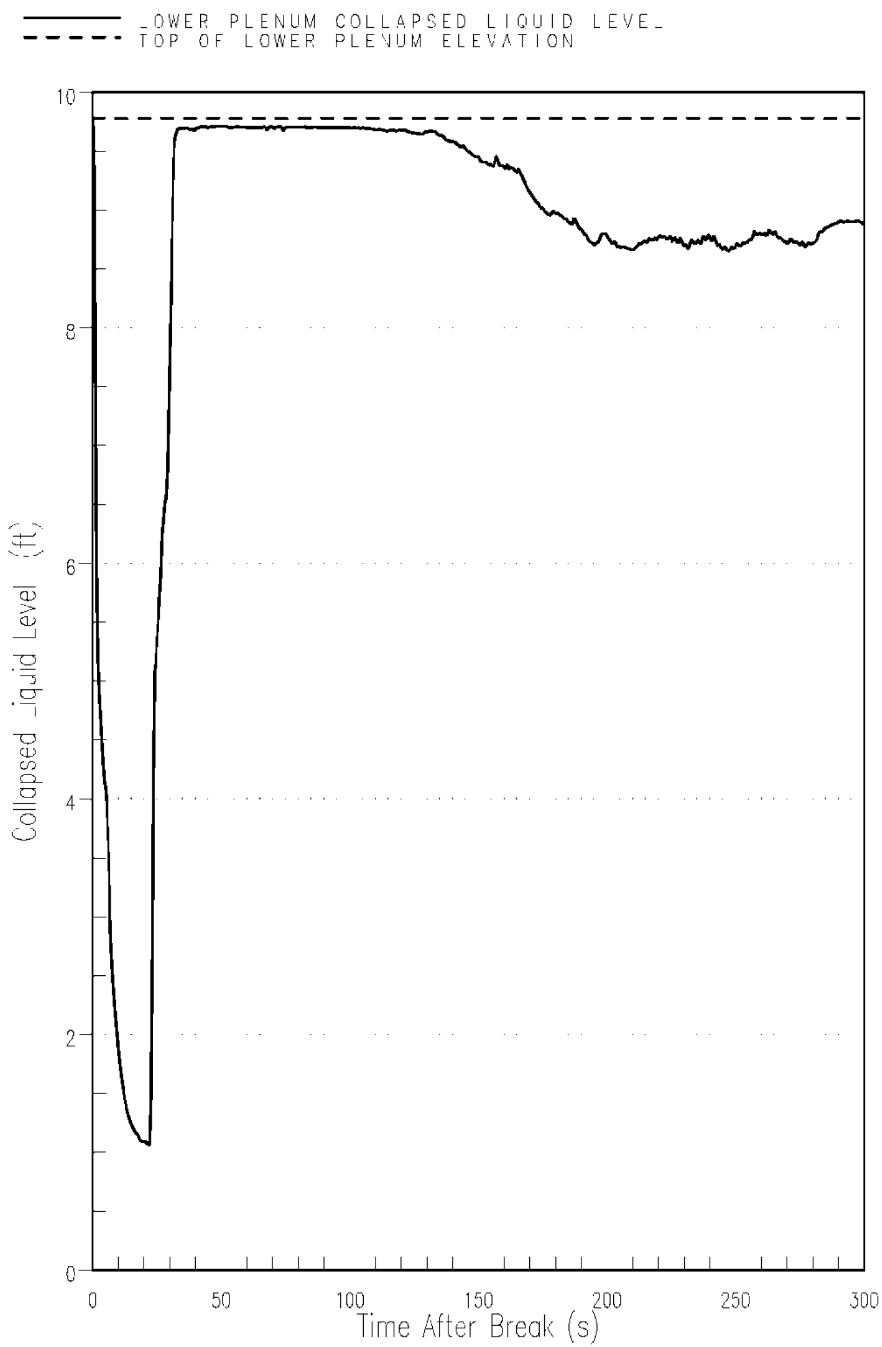

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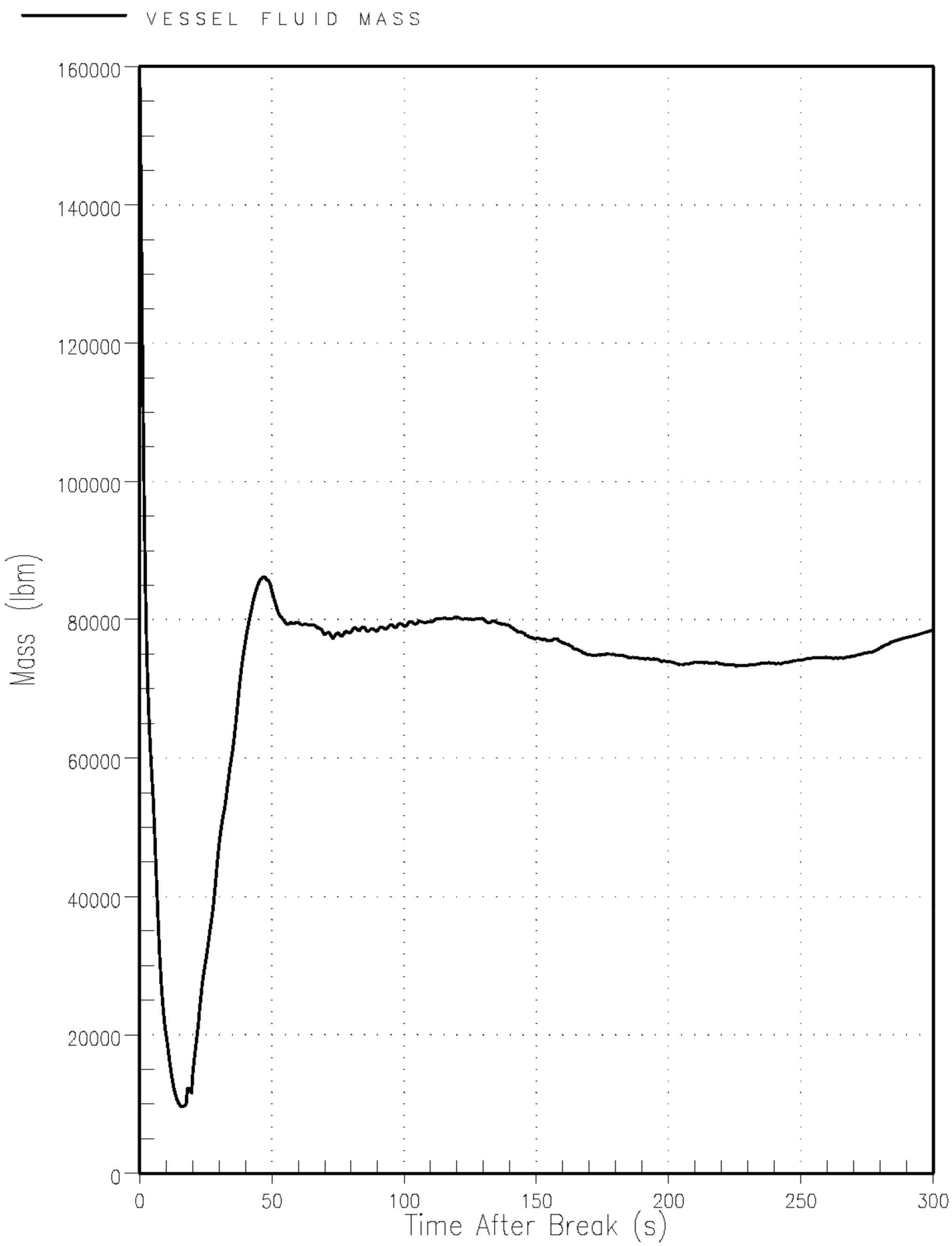
LIMITING PCT CASE LOWER PLENUM COLLAPSED LIQUID LEVEL

FIGURE 14.3.2.1-6



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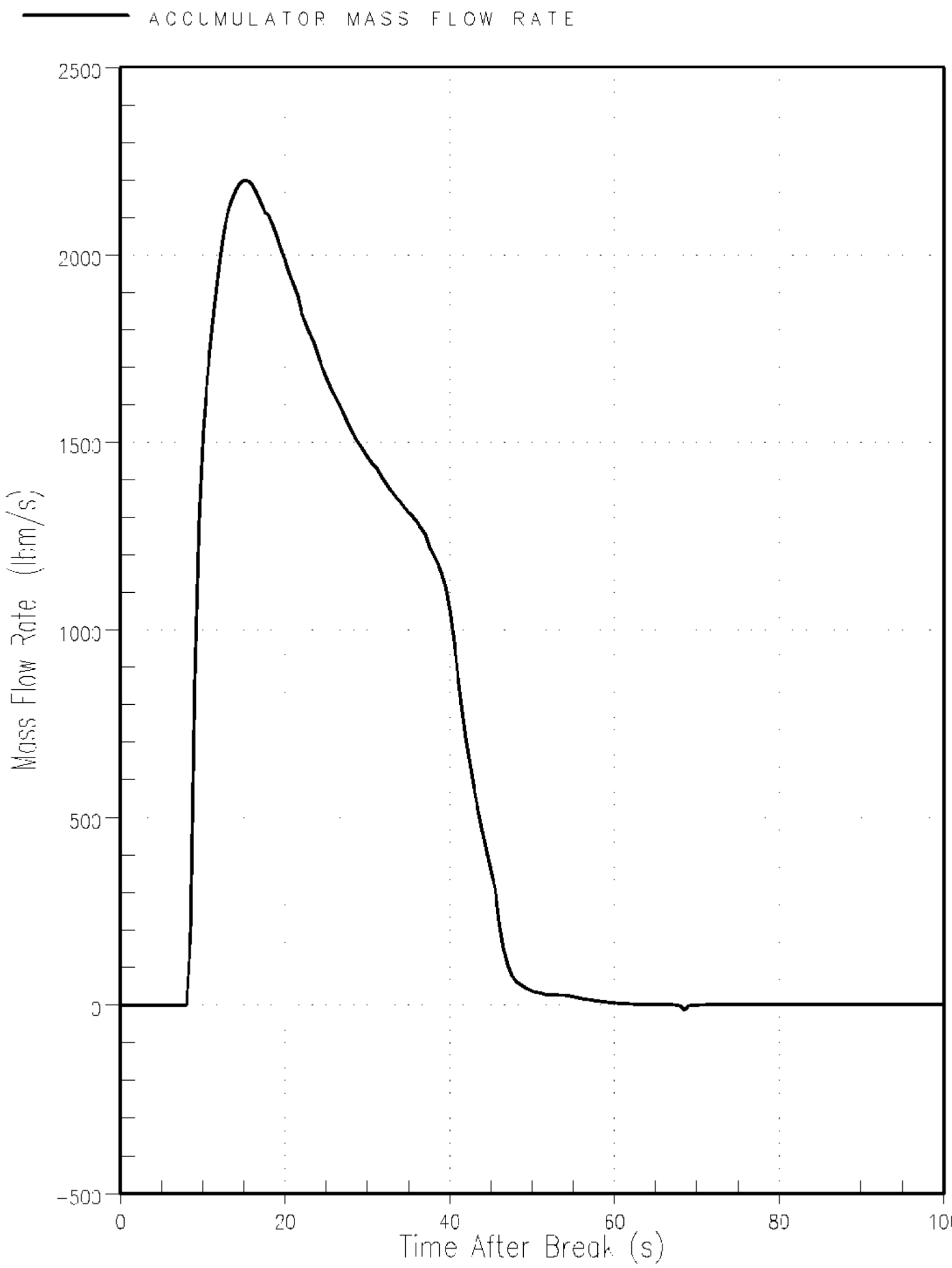
LIMITING PCT CASE VESSEL FLUID MASS

FIGURE 14.3.2.1-7



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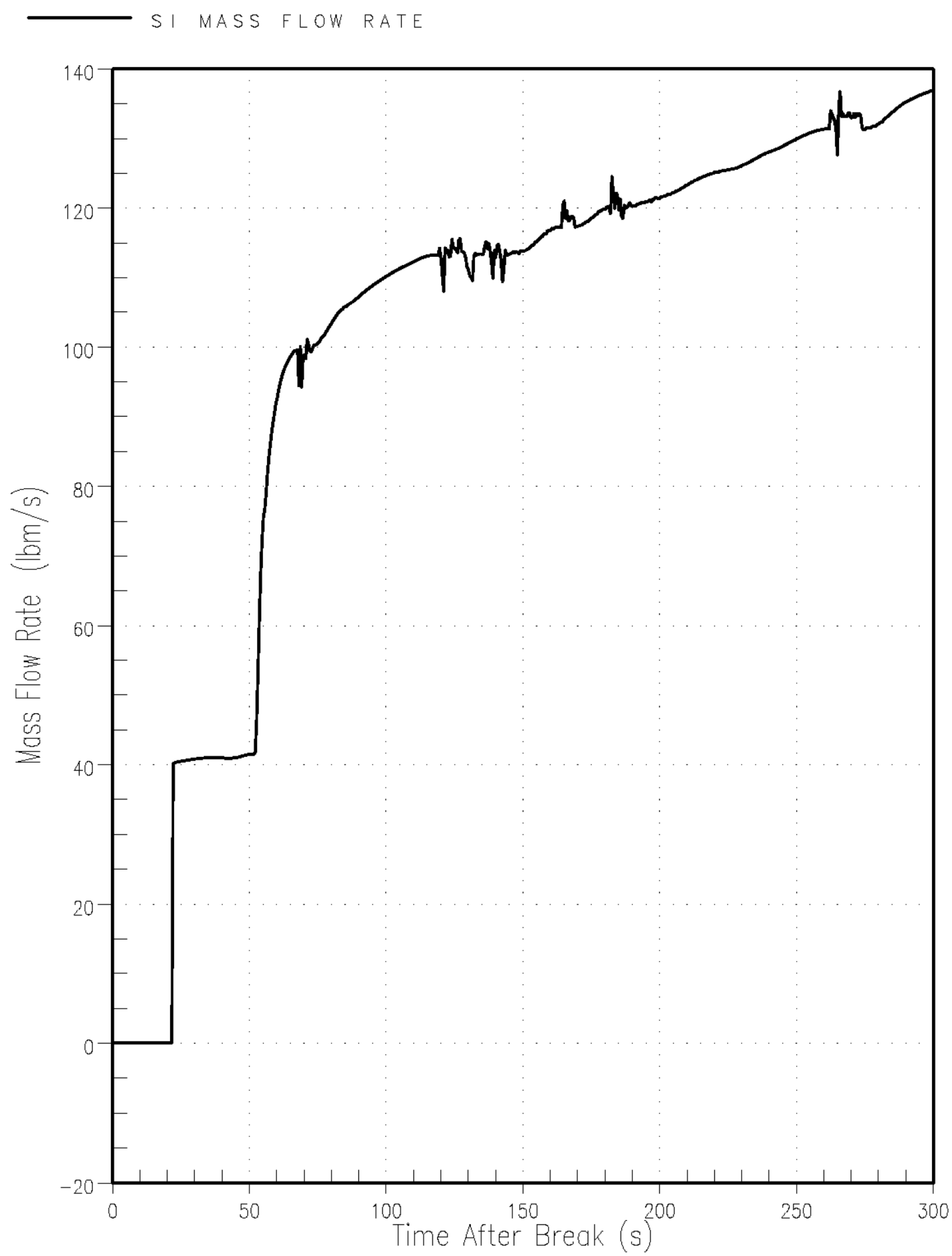
TURKEY POINT PLANT UNITS 3 & 4

LIMITING PCT CASE

ACCUMULATOR FLOW RATE

FIGURE 14.3.2.1-8



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LIMITING PCT CASE

SAFETY INJECTION FLOW RATE

FIGURE 14.3.2.1-9

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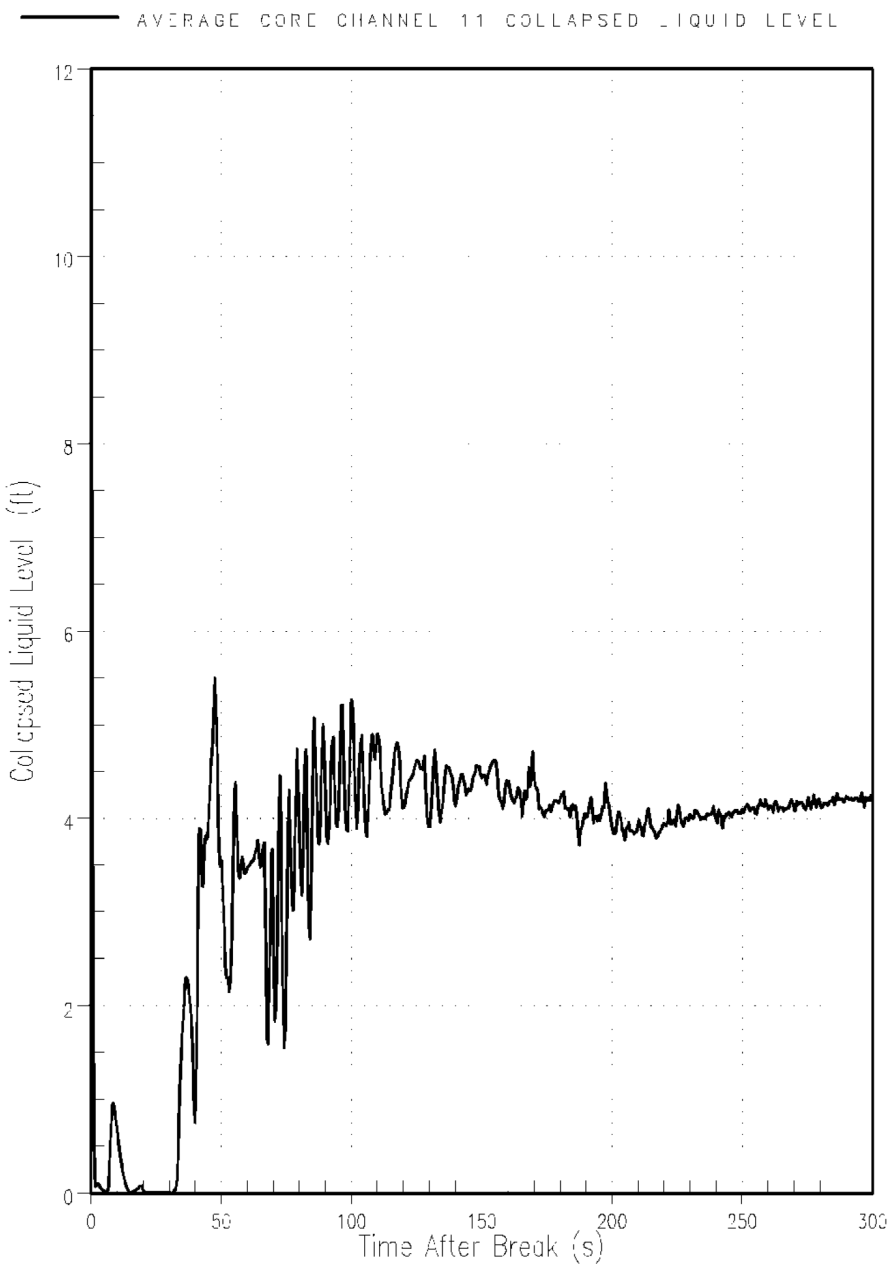

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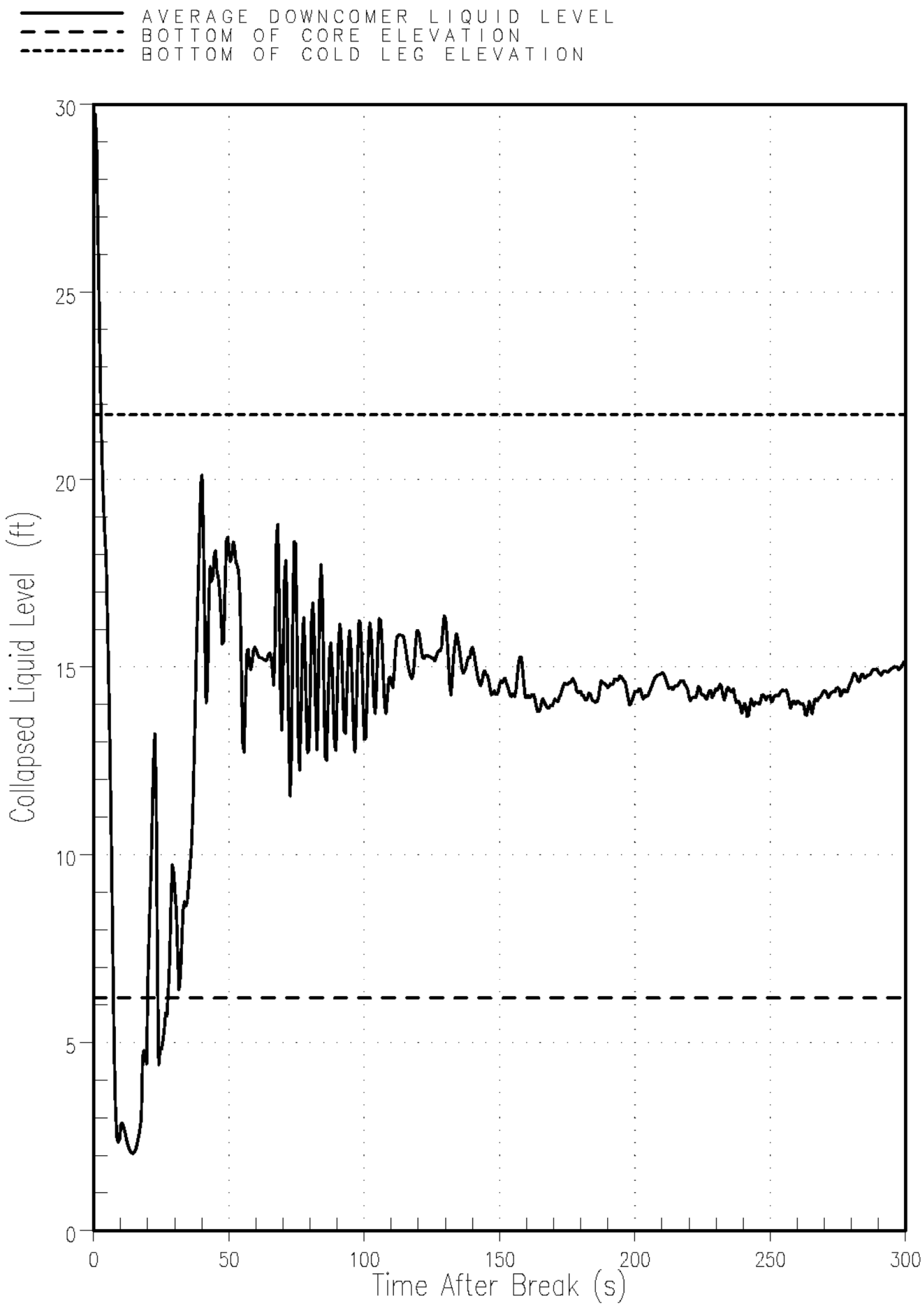
LIMITING PCT CASE
CORE AVERAGE CHANNEL
COLLAPSED LIQUID LEVEL

FIGURE 14.3.2.1-10



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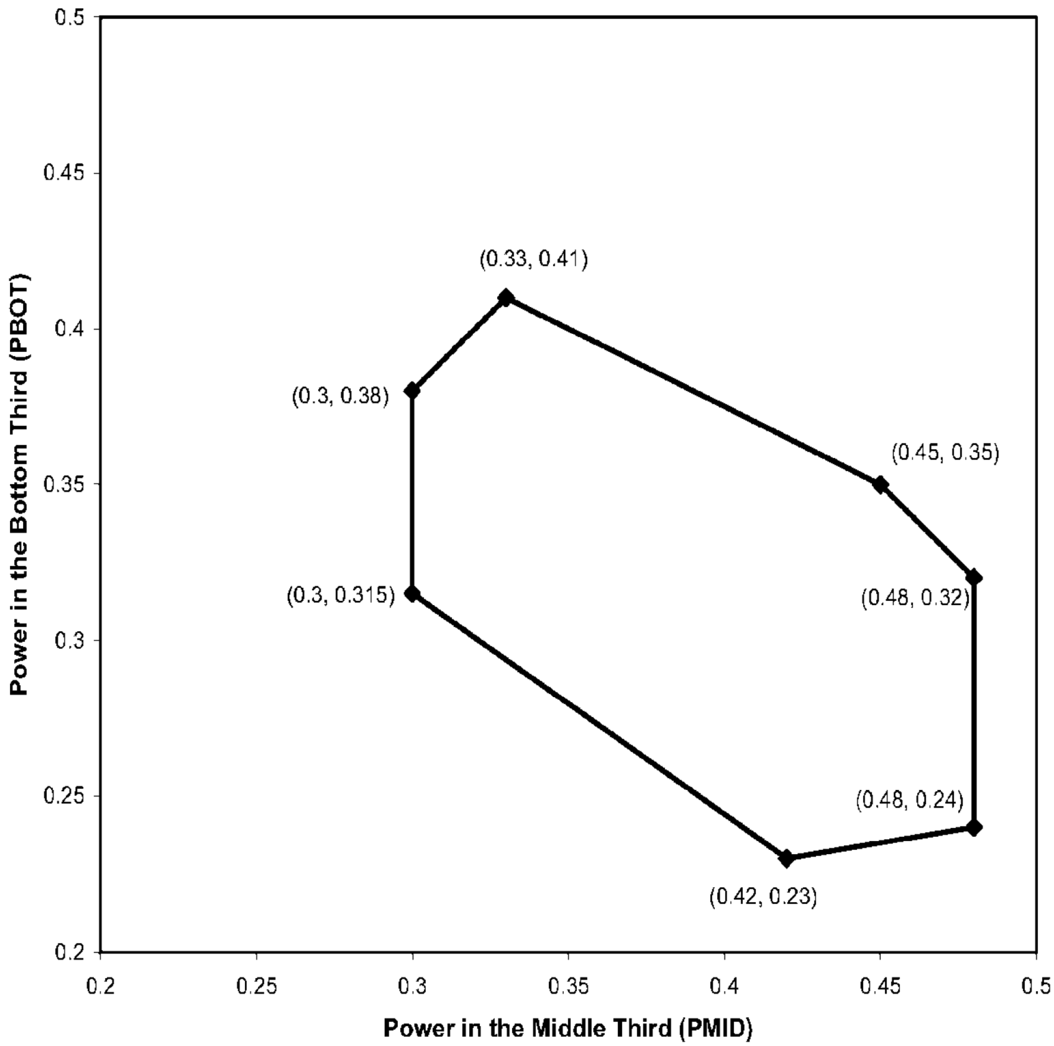
LIMITING PCT CASE
DOWNCOMER COLLAPSED LIQUID LEVEL

FIGURE 14.3.2.1-11



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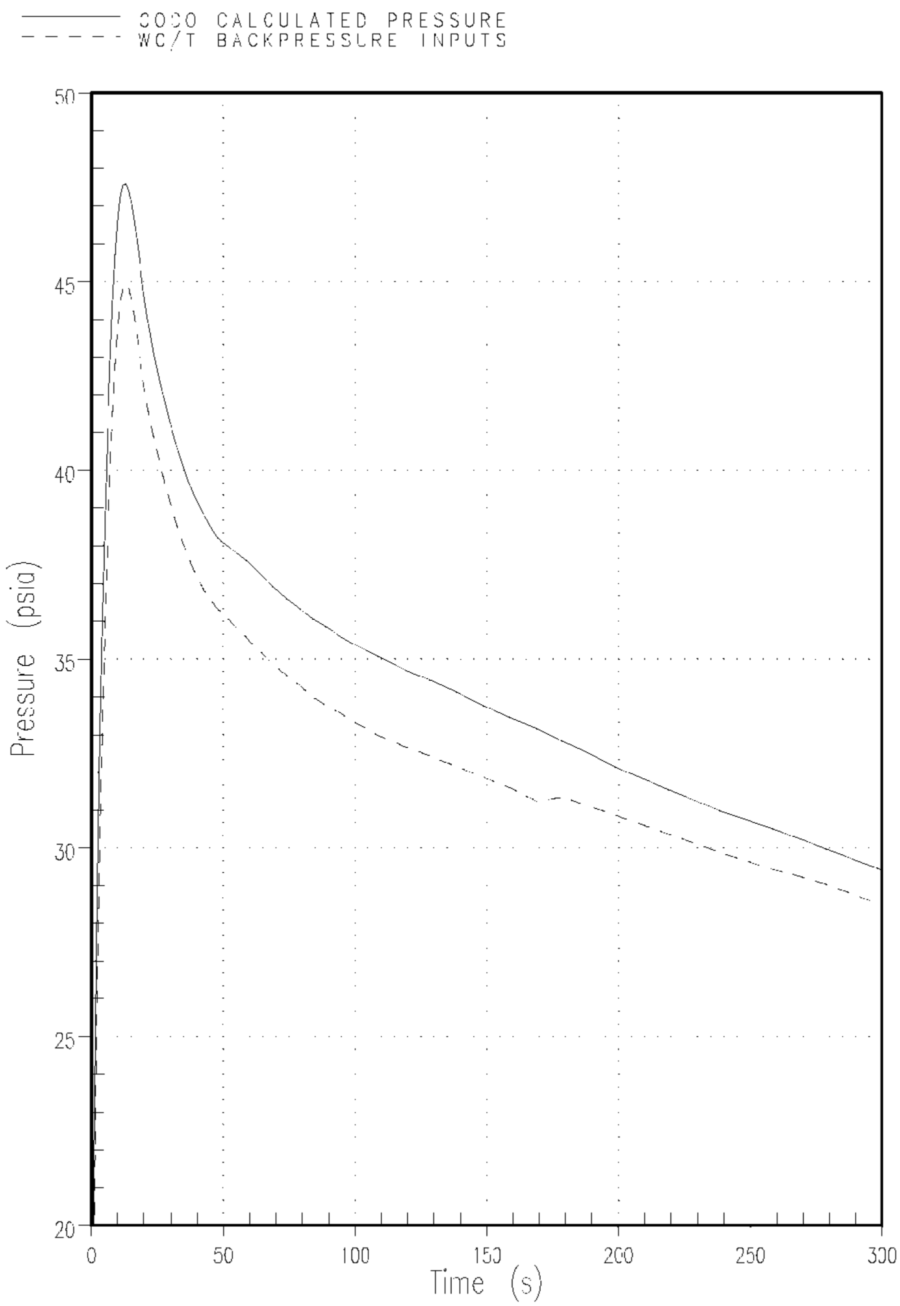
BE-LOCA ANALYSIS AXIAL POWER SHAPE OPERATING SPACE ENVELOPE

FIGURE 14.3.2.1-12



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LOWER BOUND CONTAINMENT PRESSURE

FIGURE 14.3.2.1-13

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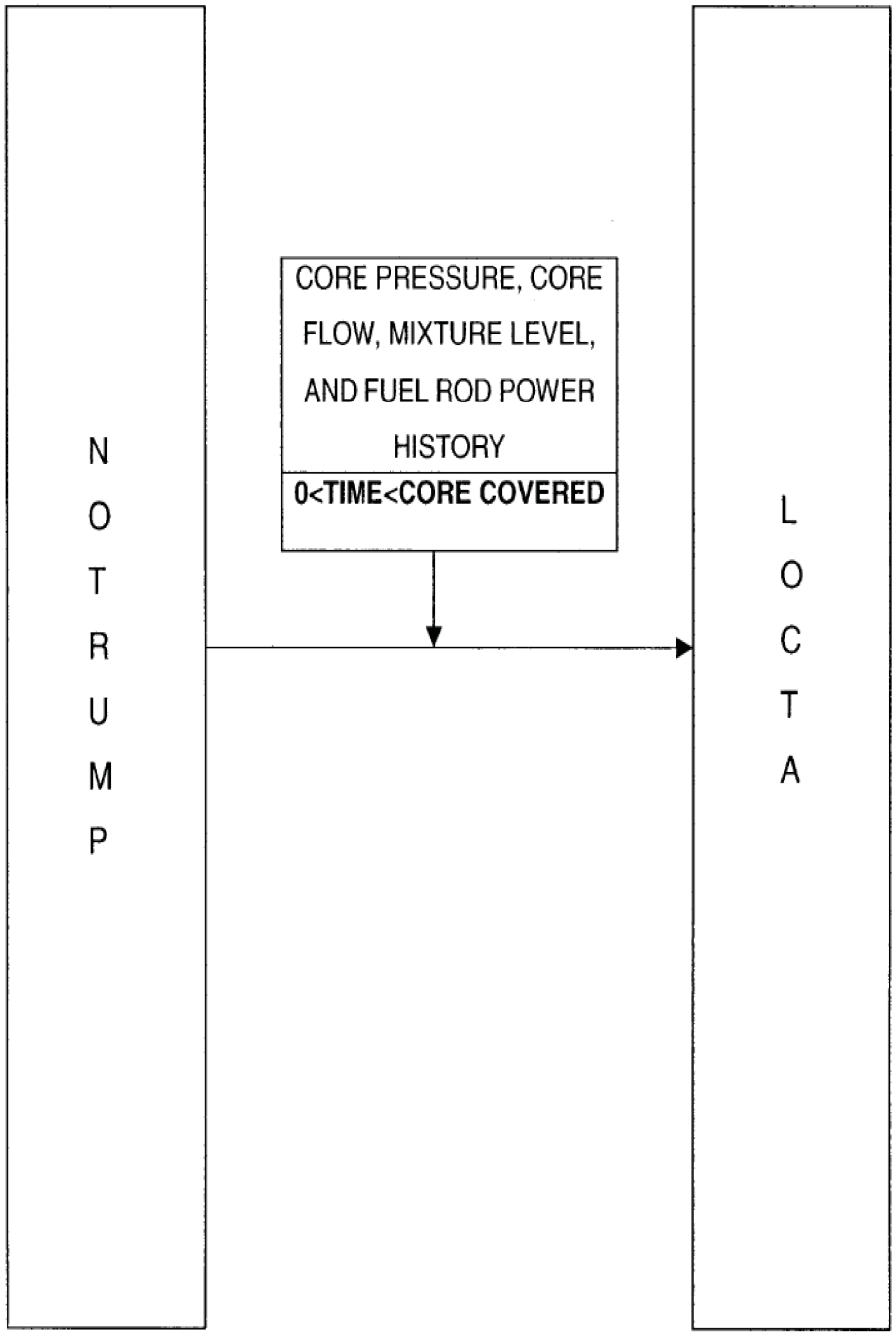
CONTAINMENT BACKPRESSURE
FOR THE NOMINAL SPLIT BREAK
TRANSIENT

FIGURE 14.3.2.1-14



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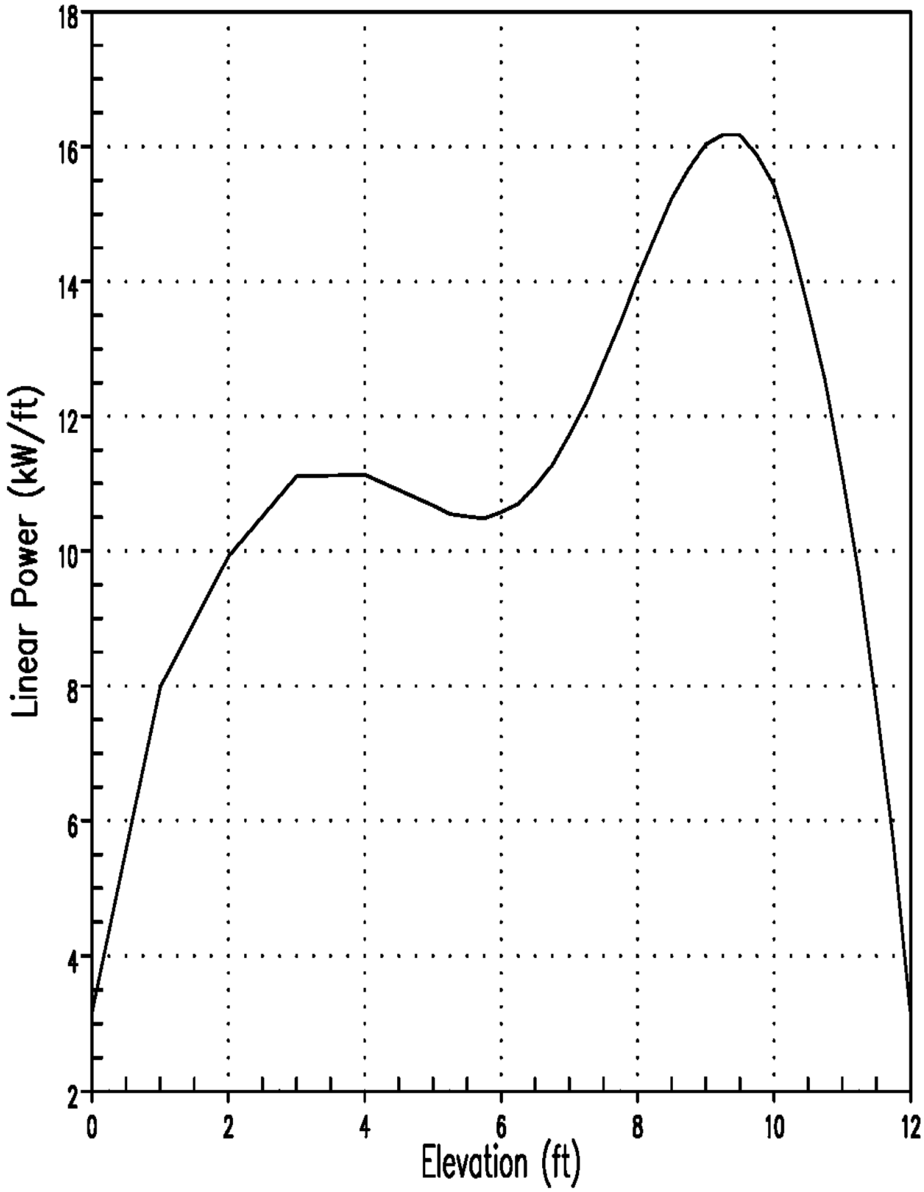
CODE INTERFACE DESCRIPTION FOR THE
SMALL BREAK LOCA MODEL

FIGURE 14.3.2.2-1



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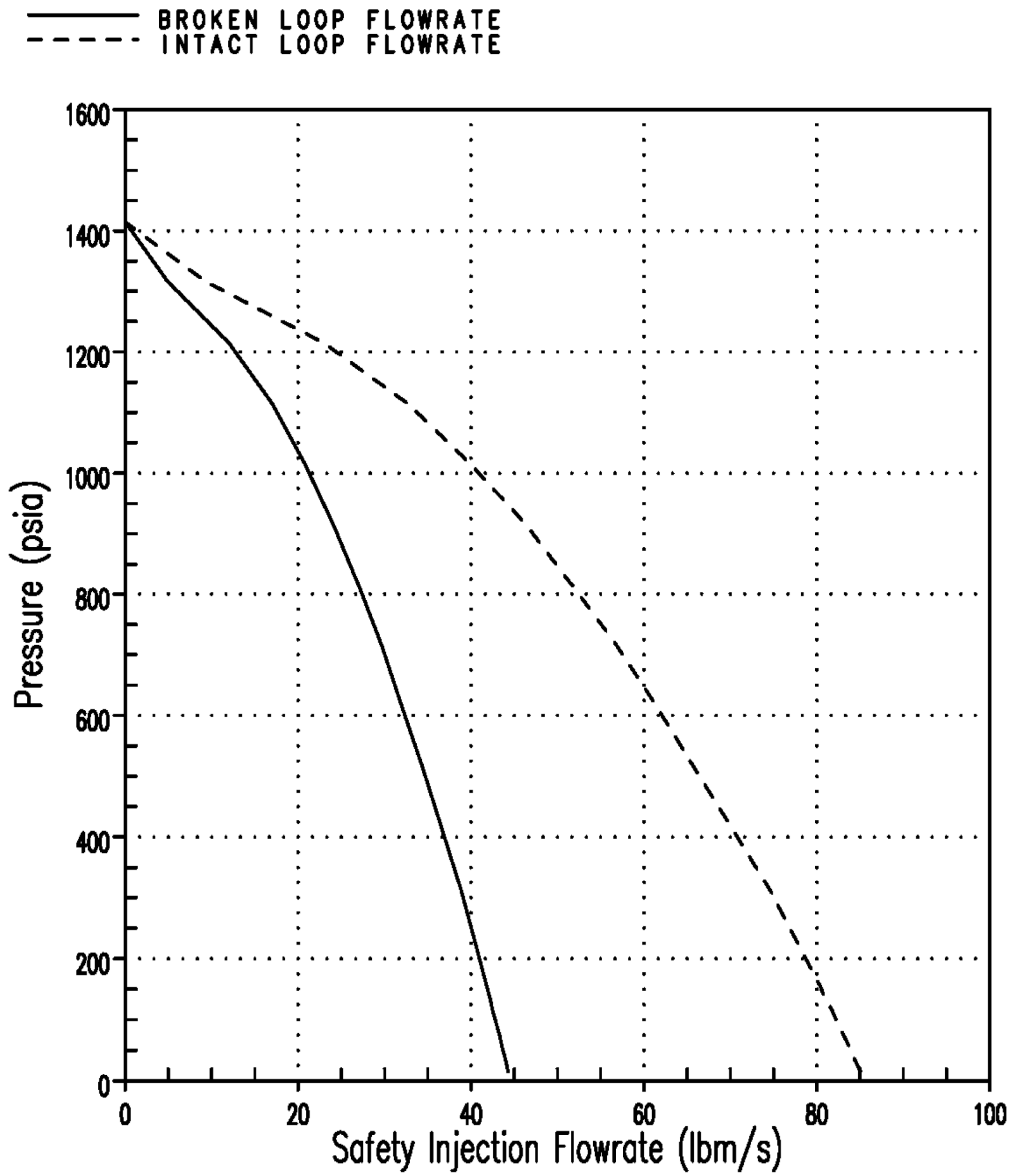
SMALL BREAK LOCA

HOT ROD AXIAL POWER SHAPE

FIGURE 14.3.2.2-2



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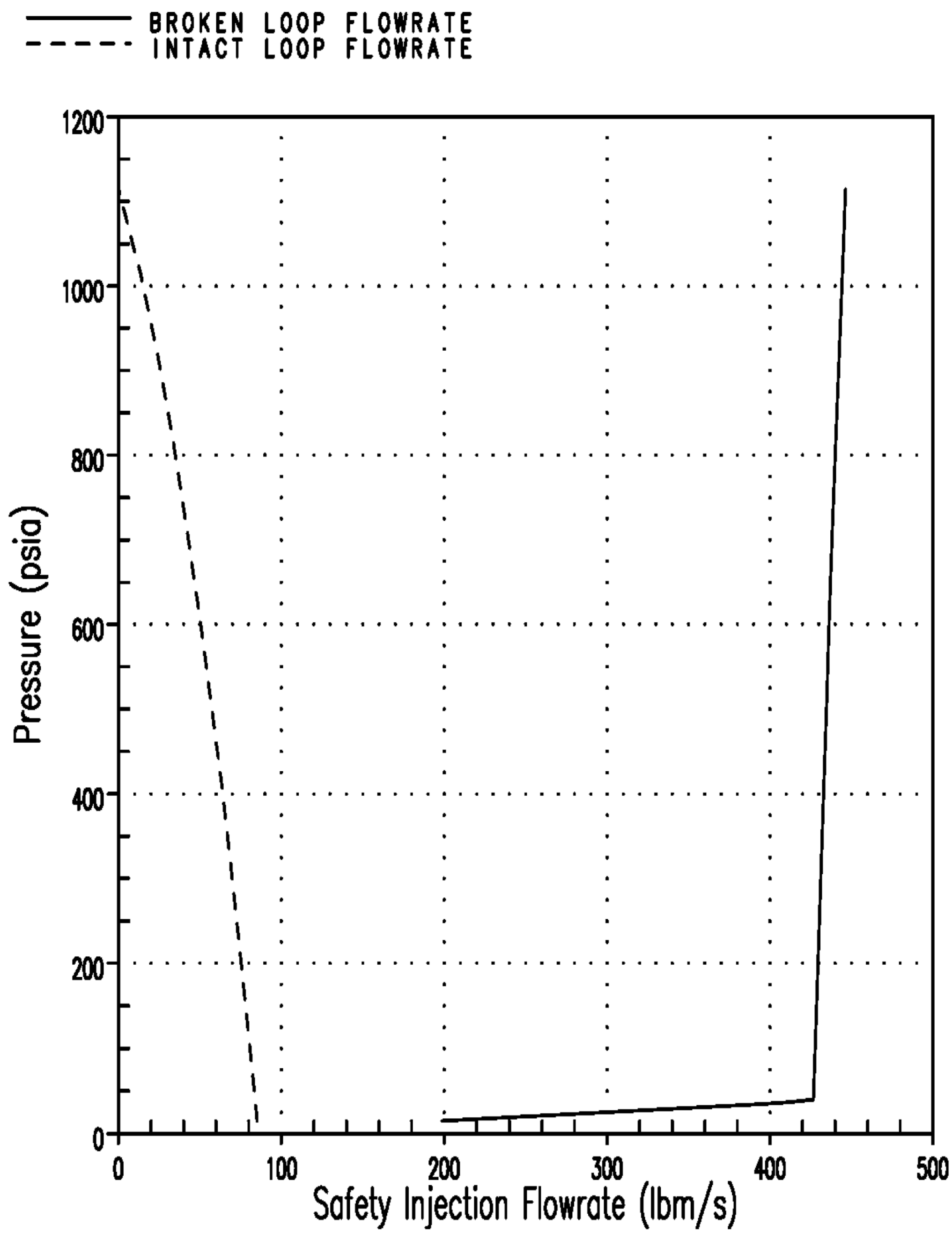
SMALL BREAK LOCA SI FLOW RATE: FAULTED LOOP SPILLS TO RCS PRESSURE (TWO HHSI PUMPS, INJECTION AND COLD LEG RECIRCULATION PHASES)

FIGURE 14.3.2.2-3



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SMALL BREAK LOCA SI FLOW RATE: FAULTED LOOP SPILLS TO CONTAINMENT PRESSURE (0 PSIG) (TWO HHSI PUMPS, INJECTION PHASE)

FIGURE 14.3.2.2-4

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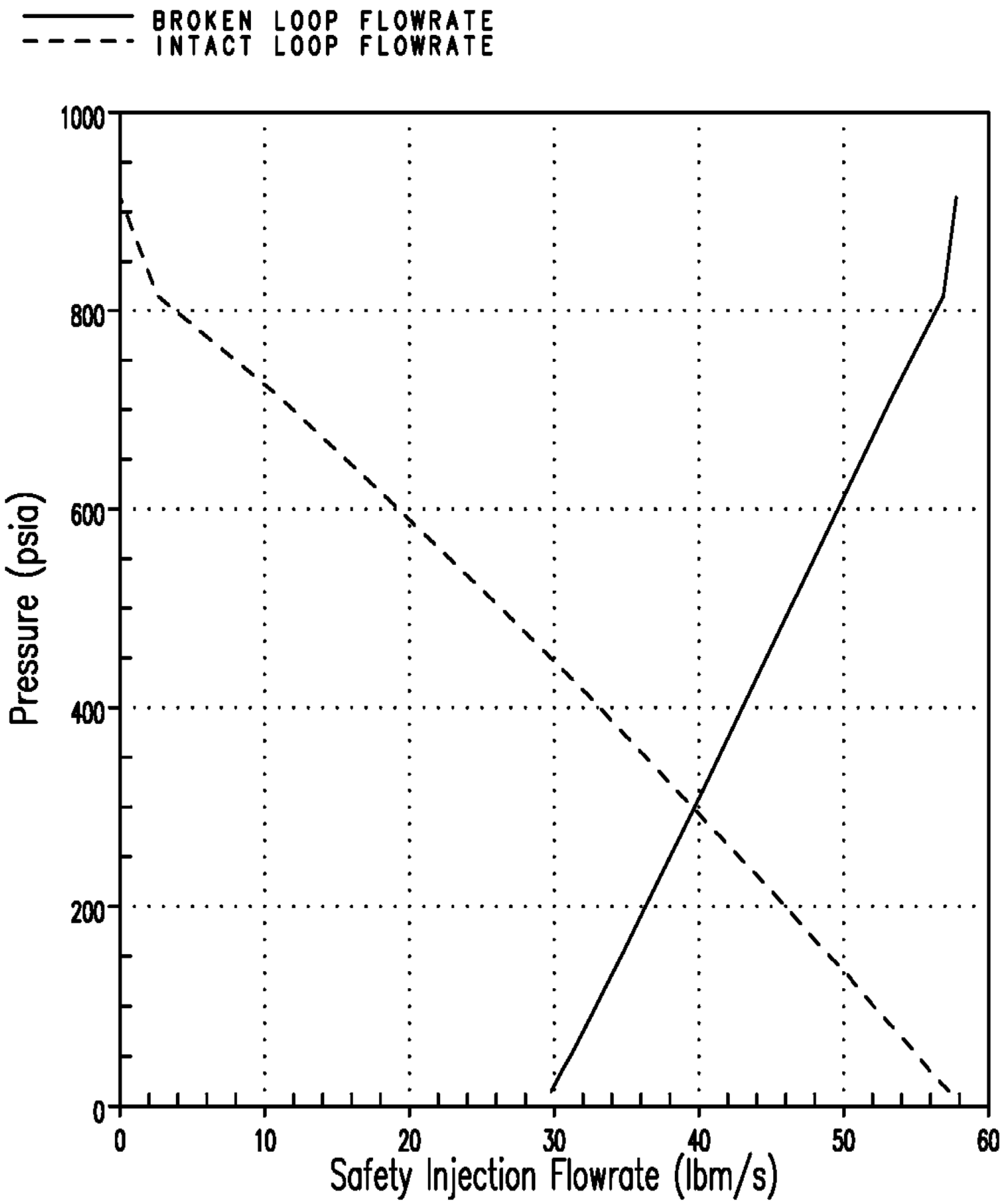

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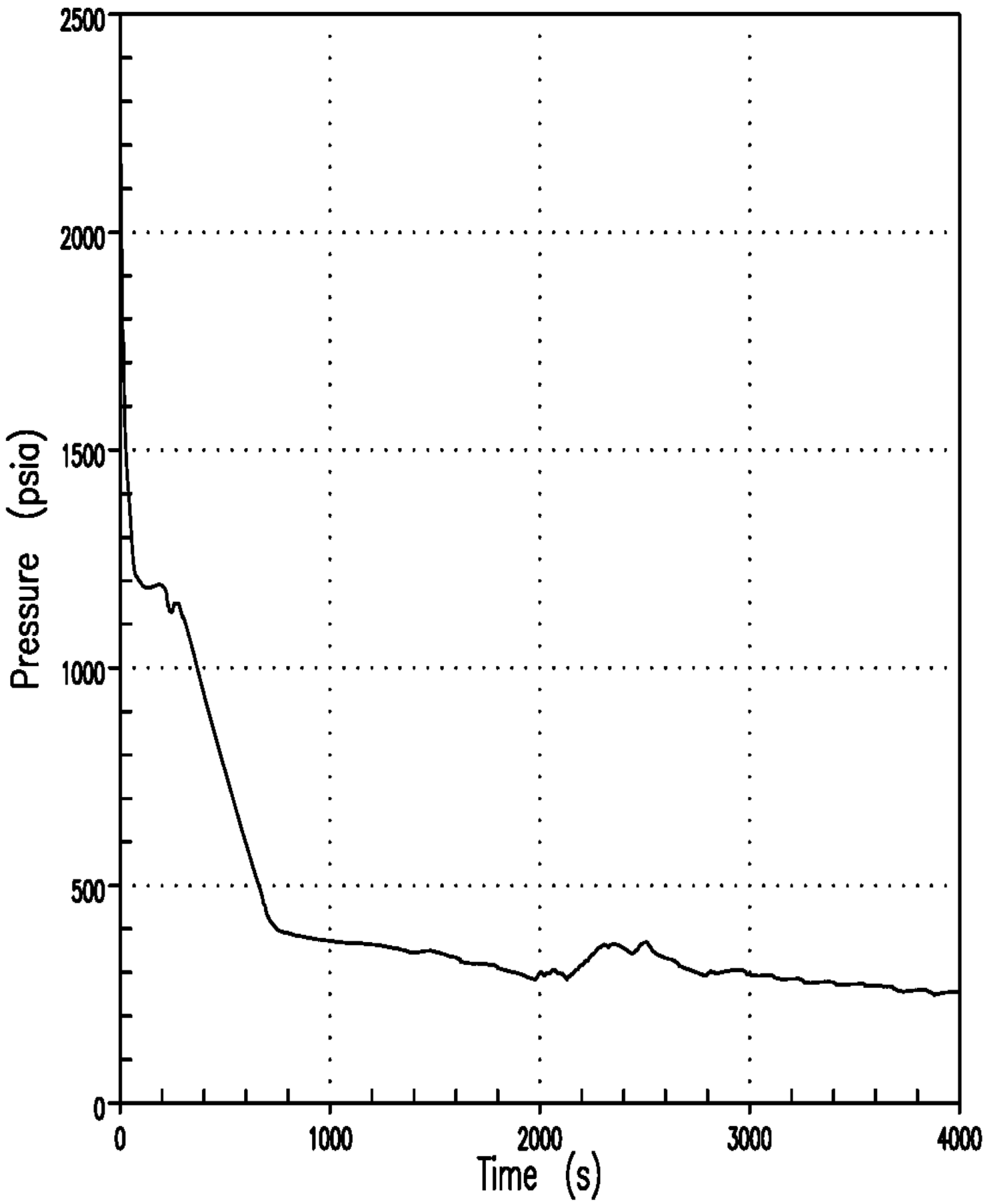
SMALL BREAK LOCA SI FLOW RATE: FAULTED LOOP SPILLS TO CONTAINMENT PRESSURE (0 PSIG) (ONE HHSI PUMP, COLD LEG RECIRCULATION PHASE)

FIGURE 14.3.2.2-5



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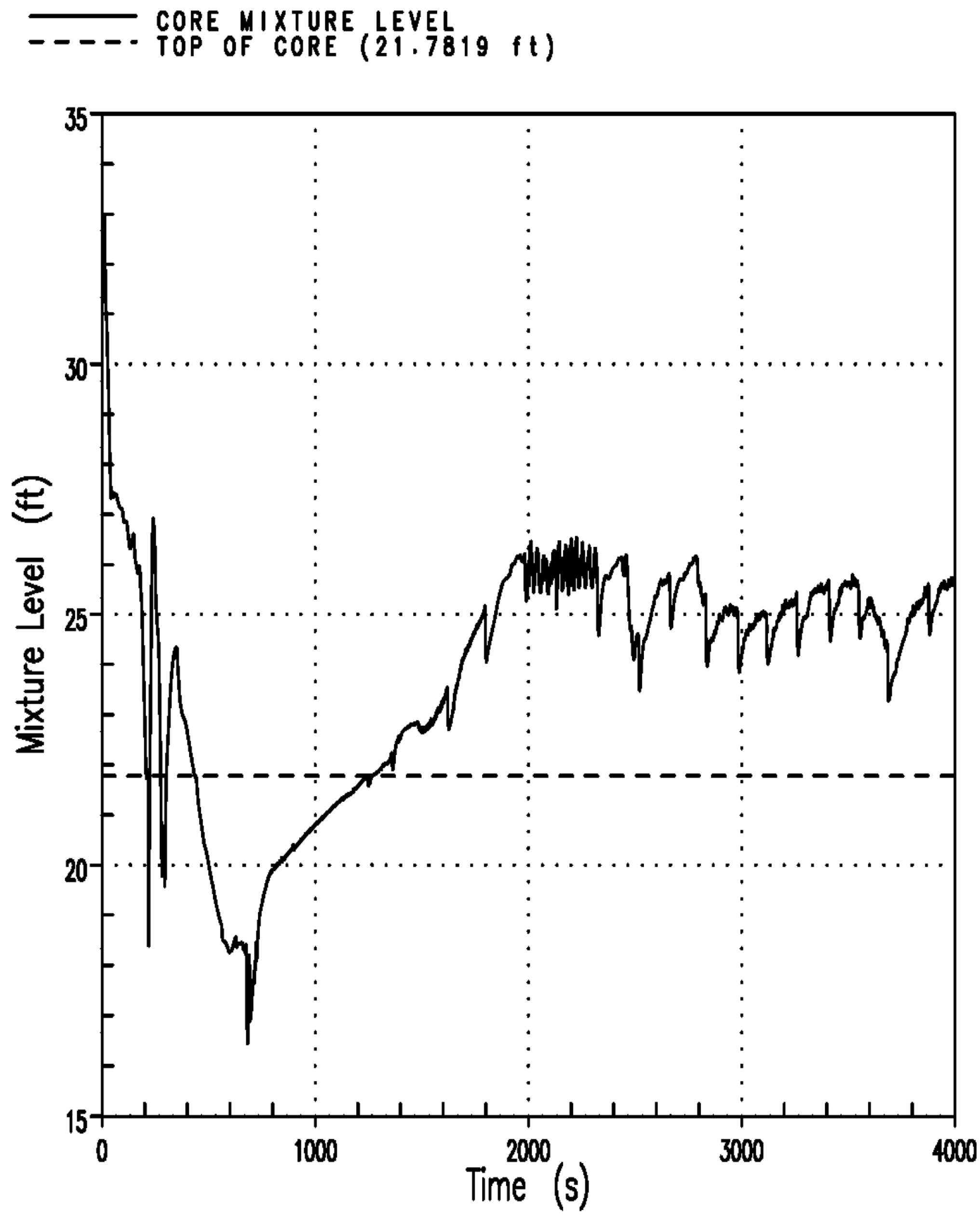
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

REACTOR COOLANT SYSTEM PRESSURE

4-INCH BREAK

FIGURE 14.3.2.2-6



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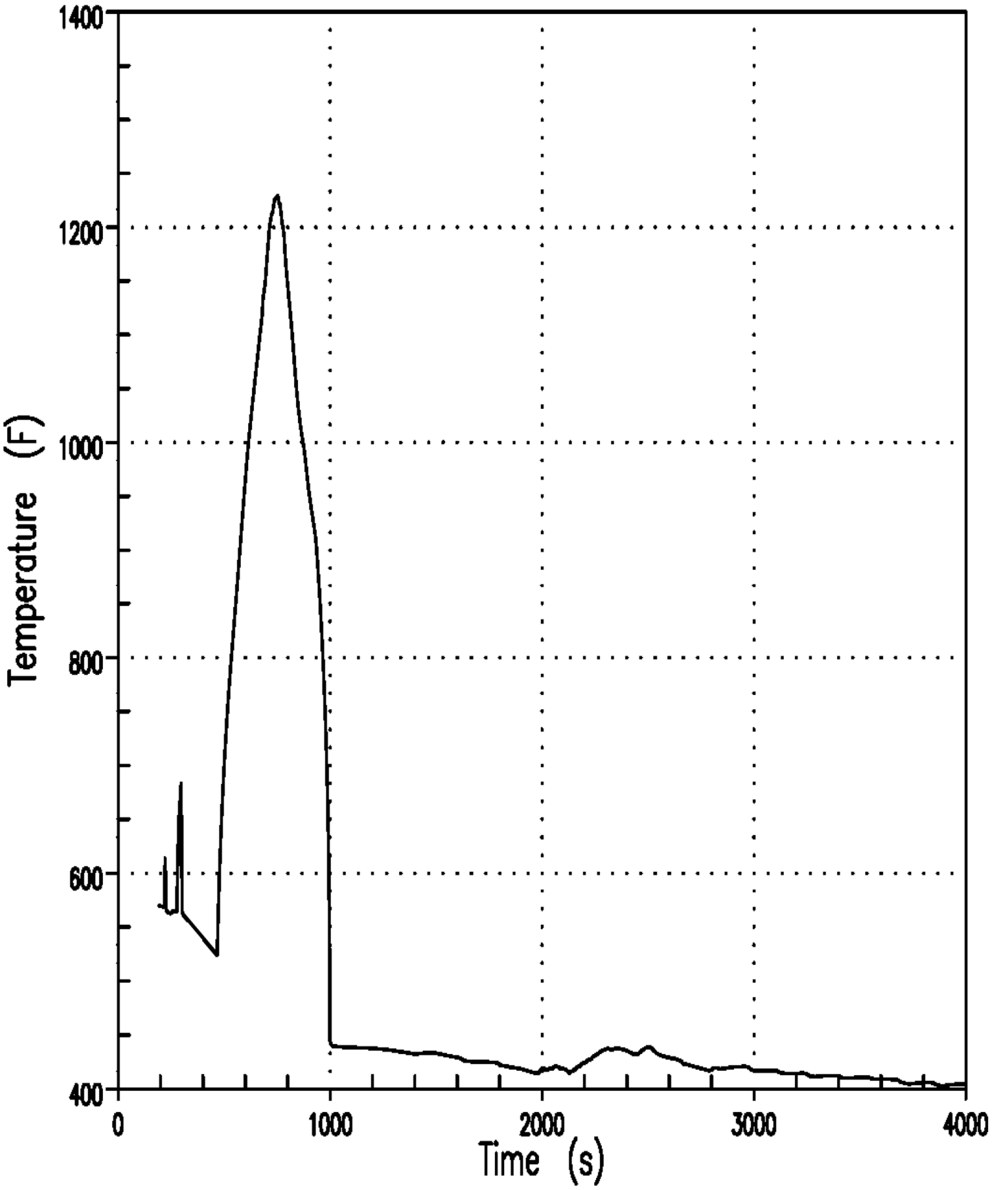
SMALL BREAK LOCA

CORE MIXTURE LEVEL

4-INCH BREAK

FIGURE 14.3.2.2-7



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TURKEY POINT PLANT UNITS 3 & 4

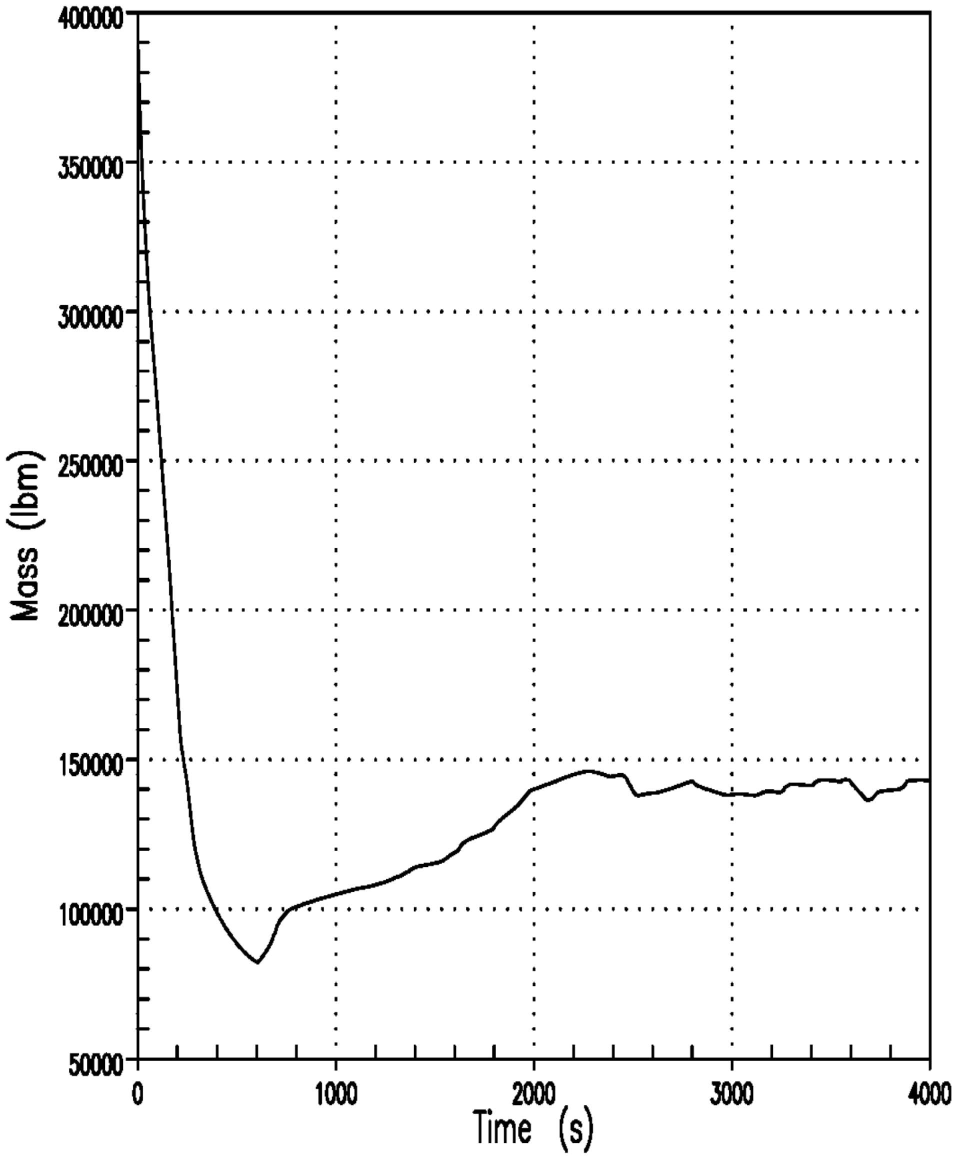
SMALL BREAK LOCA

CLADDING TEMPERATURE TRANSIENT AT

PCT ELEVATION

4-INCH BREAK

FIGURE 14.3.2.2-8



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TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

TOTAL REACTOR COOLANT SYSTEM MASS

4-INCH BREAK

FIGURE 14.3.2.2-9

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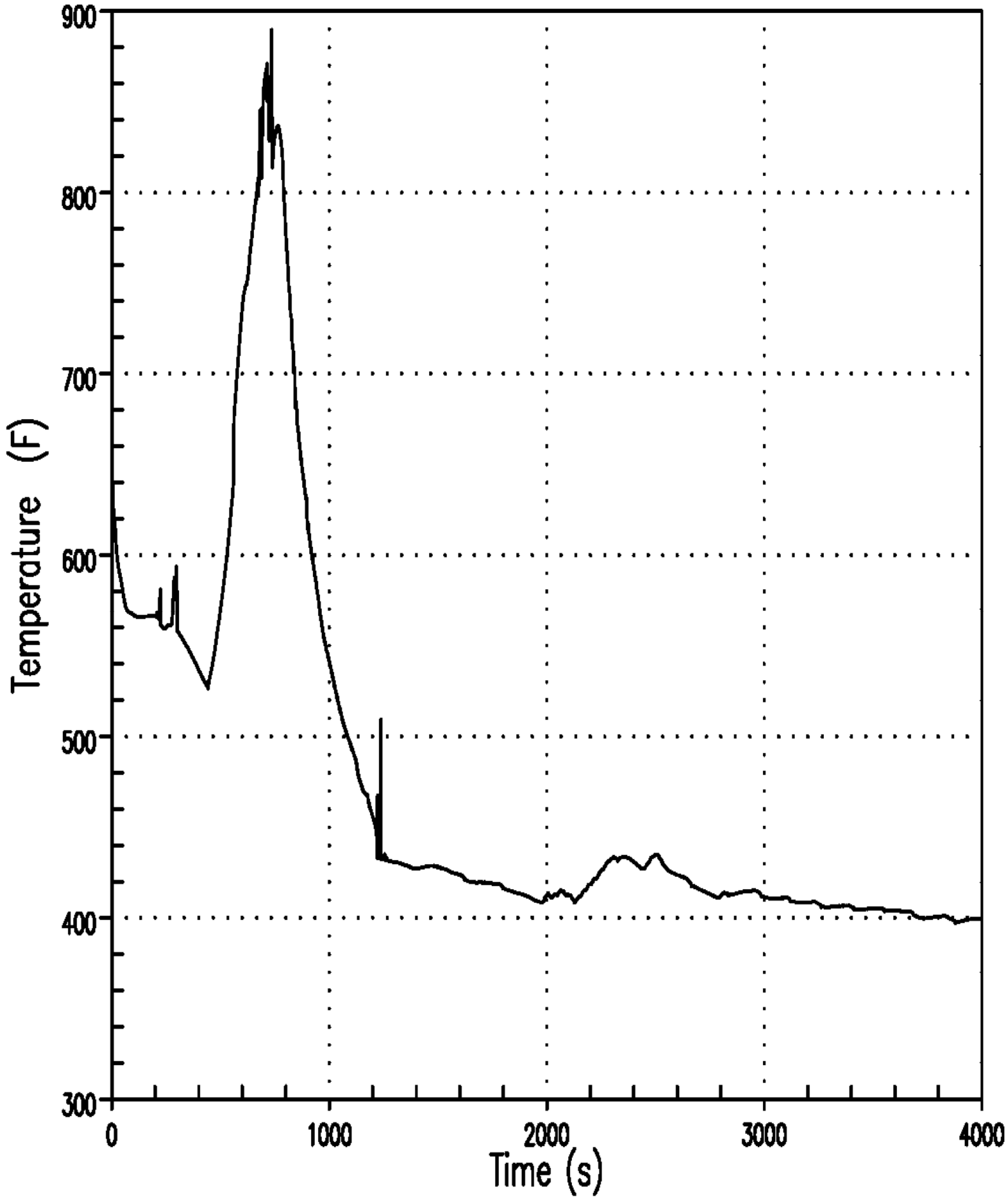

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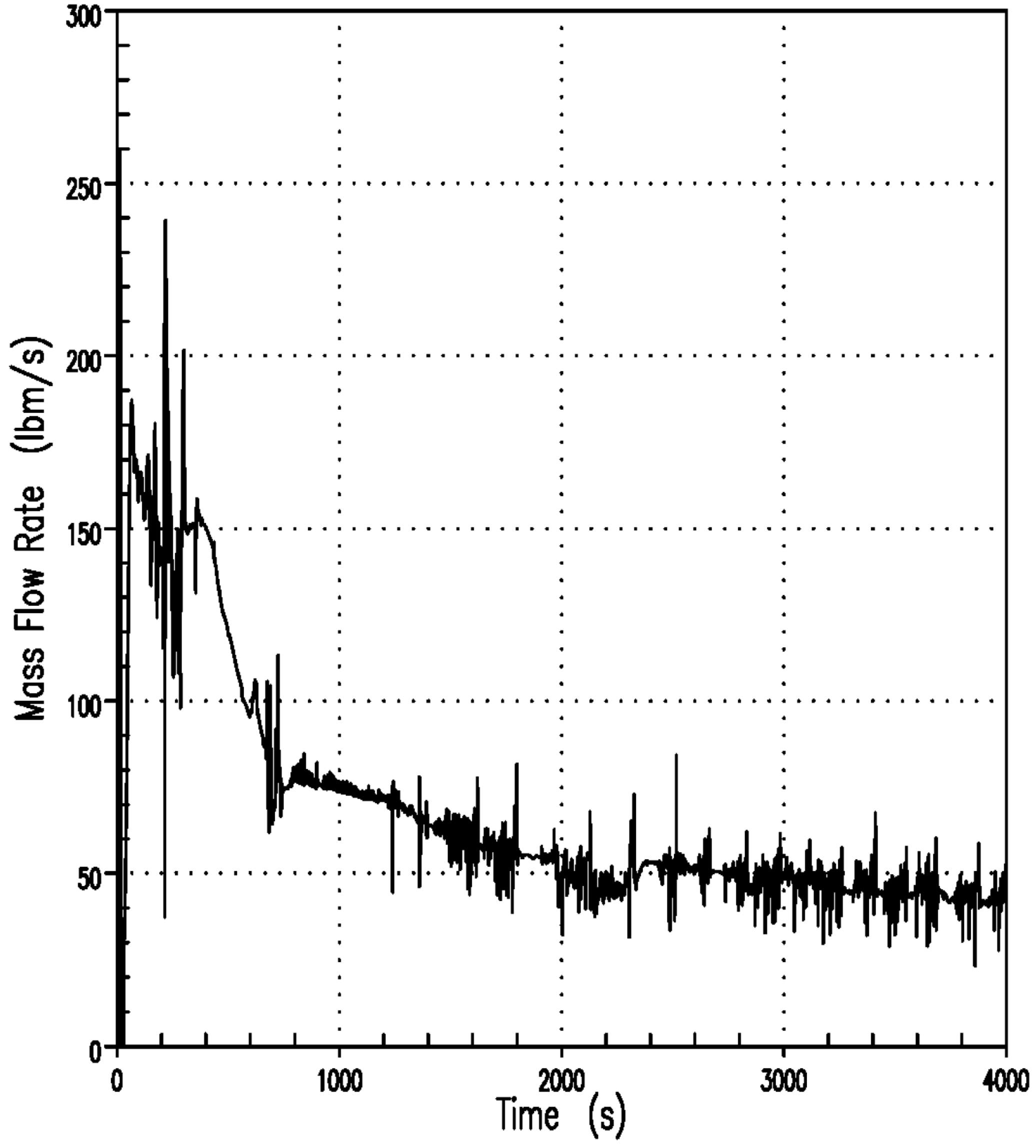
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE EXIT VAPOR TEMPERATURE

4-INCH BREAK

FIGURE 14.3.2.2-10



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TURKEY POINT PLANT UNITS 3 & 4

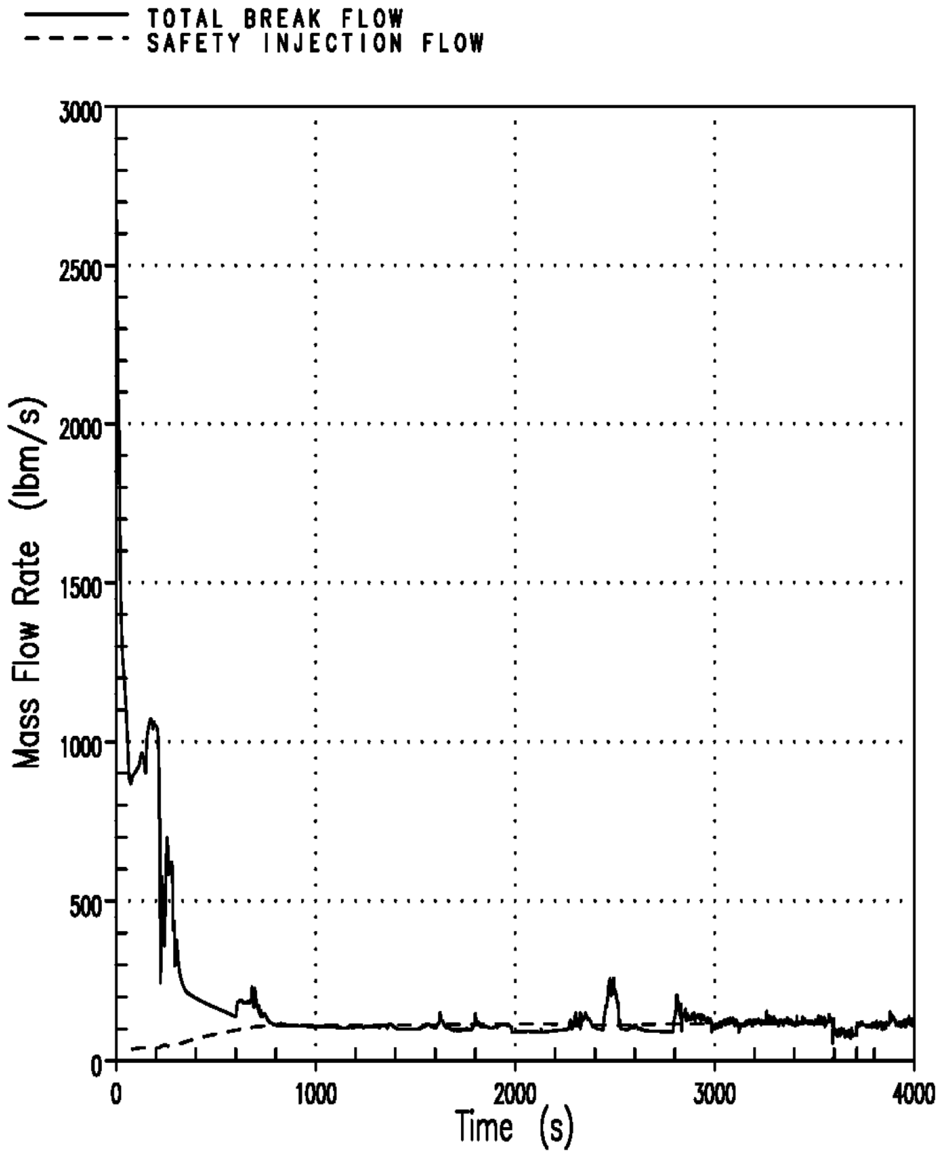
SMALL BREAK LOCA

VAPOR MASS FLOW RATE OUT OF THE

TOP OF THE CORE

4-INCH BREAK

FIGURE 14.3.2.2-11



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TURKEY POINT PLANT UNITS 3 & 4

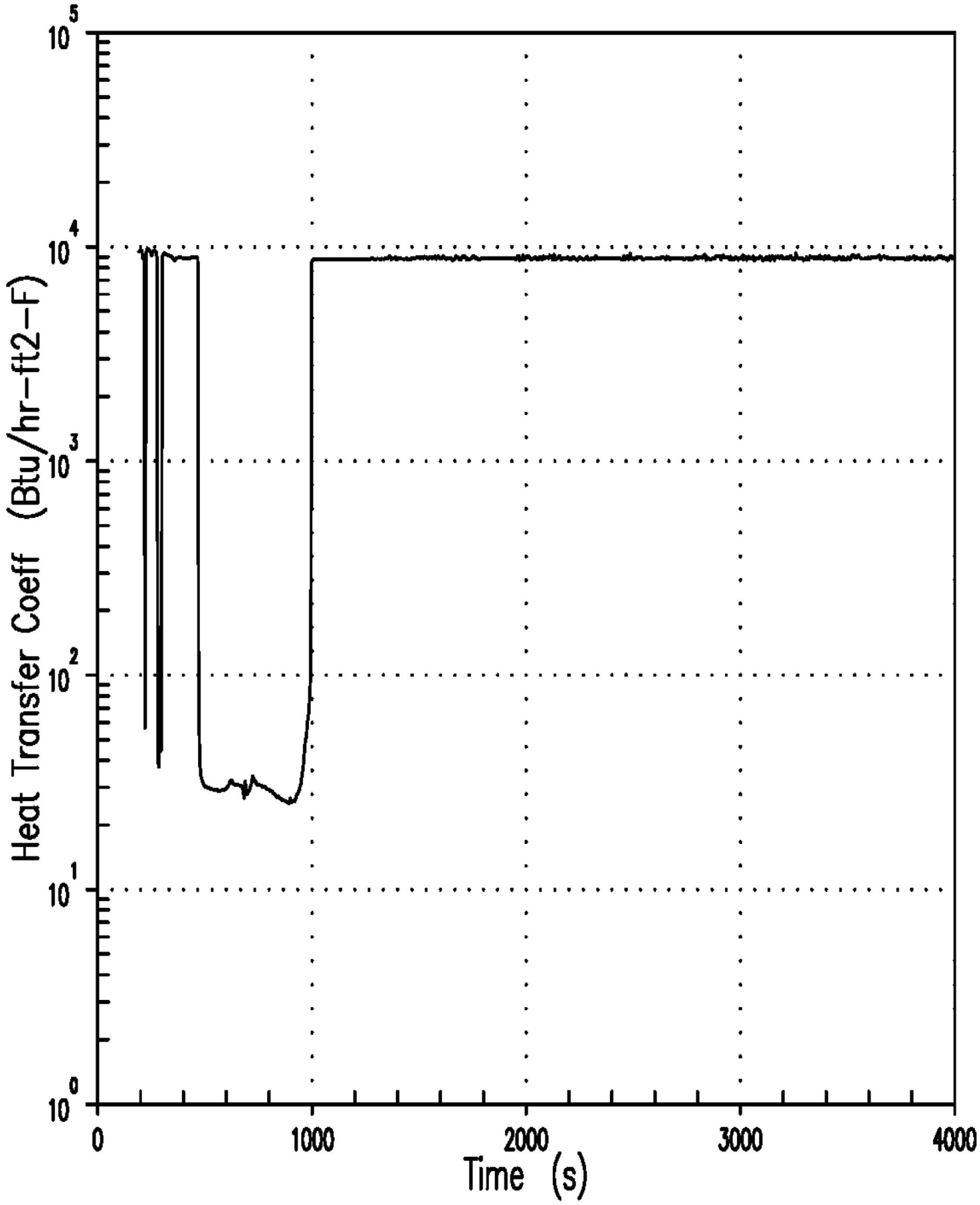
SMALL BREAK LOCA

TOTAL BREAK FLOW AND SAFETY

INJECTION FLOW

4-INCH BREAK

FIGURE 14.3.2.2-12



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TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CLADDING SURFACE HEAT TRANSFER

COEFFICIENT AT PCT ELEVATION

4-INCH BREAK

FIGURE 14.3.2.2-13

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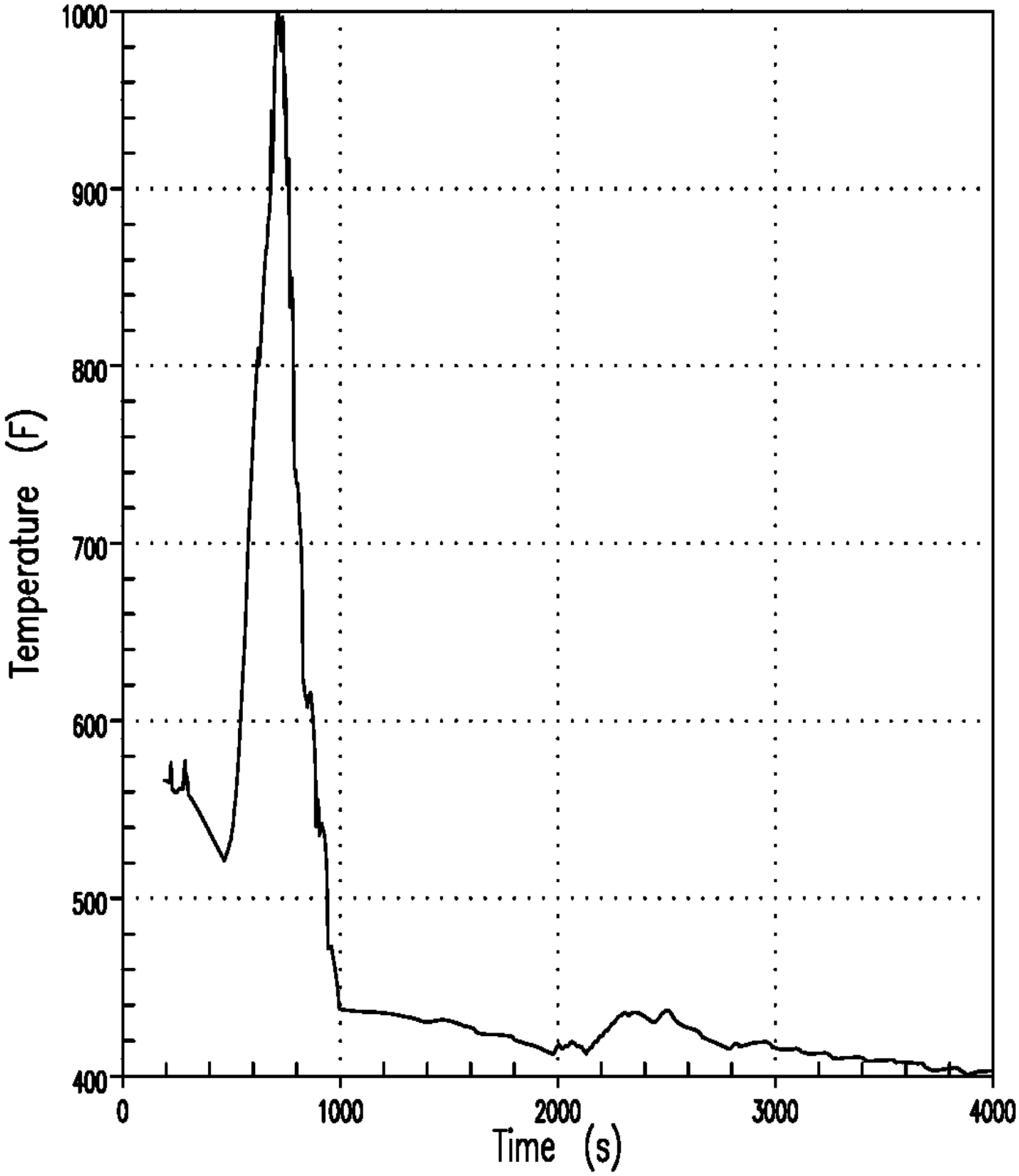

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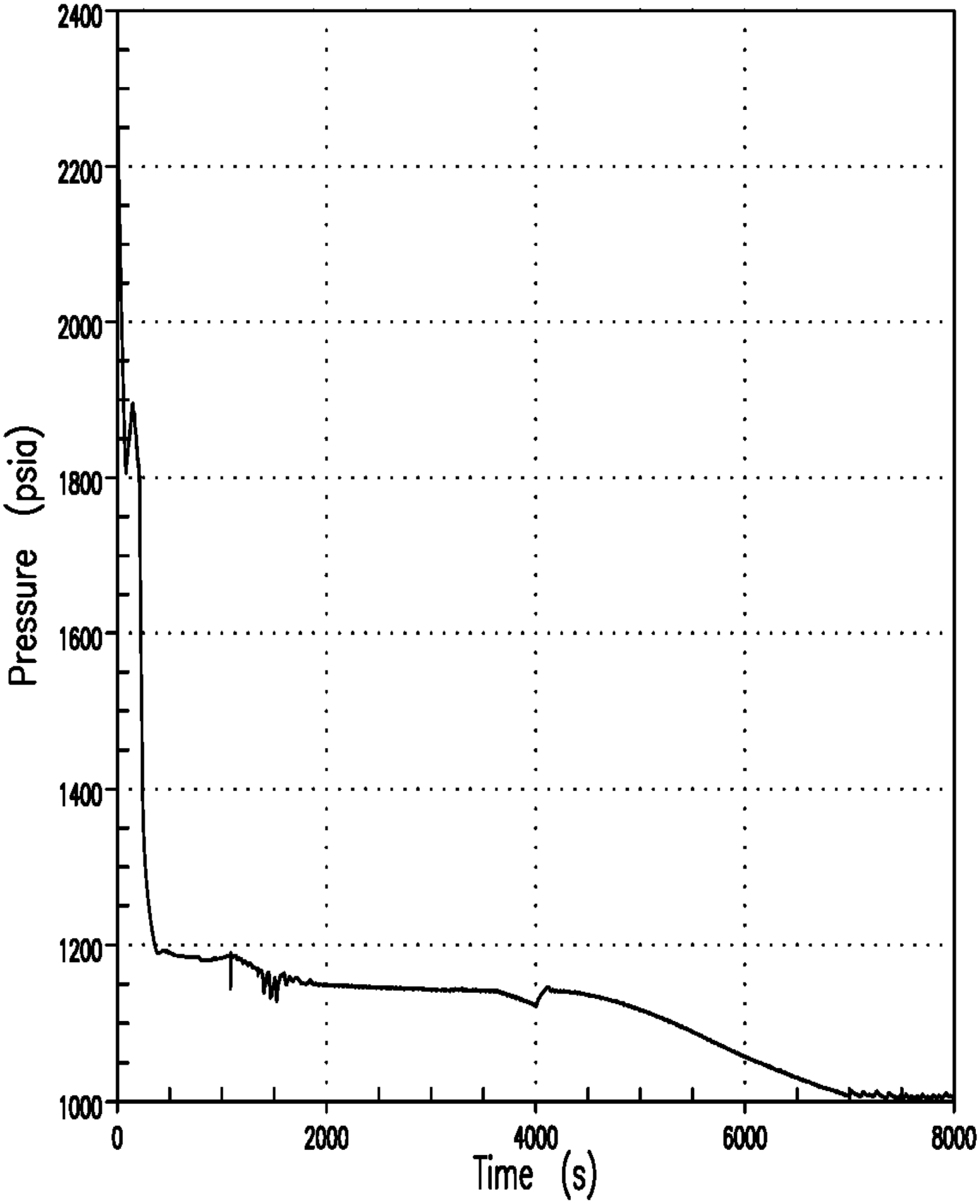
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

FLUID TEMPERATURE AT PCT ELEVATION

4-INCH BREAK

FIGURE 14.3.2.2-14



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TURKEY POINT PLANT UNITS 3 & 4

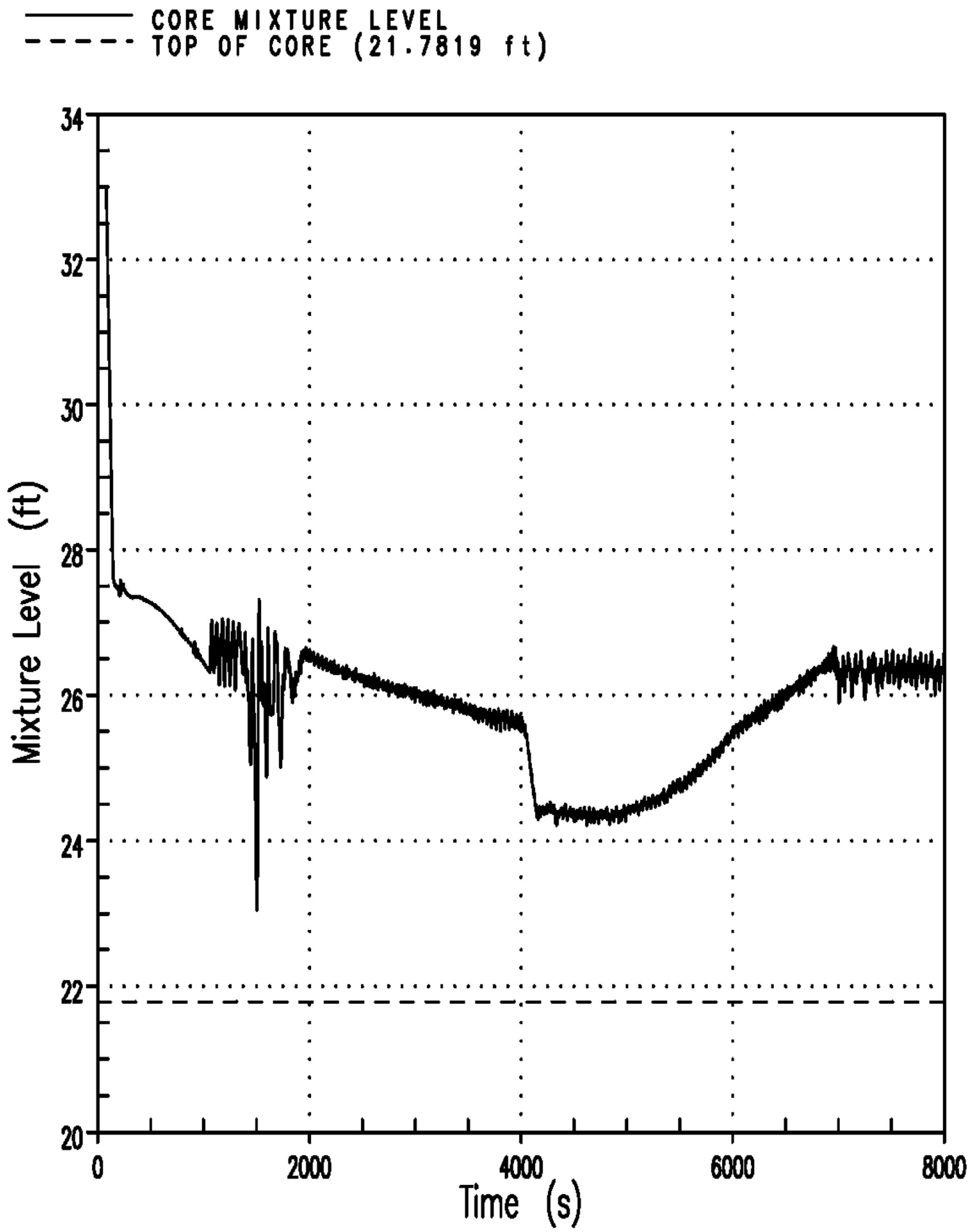
SMALL BREAK LOCA

REACTOR COOLANT SYSTEM PRESSURE

1.5-INCH BREAK

FIGURE 14.3.2.2-15

C26



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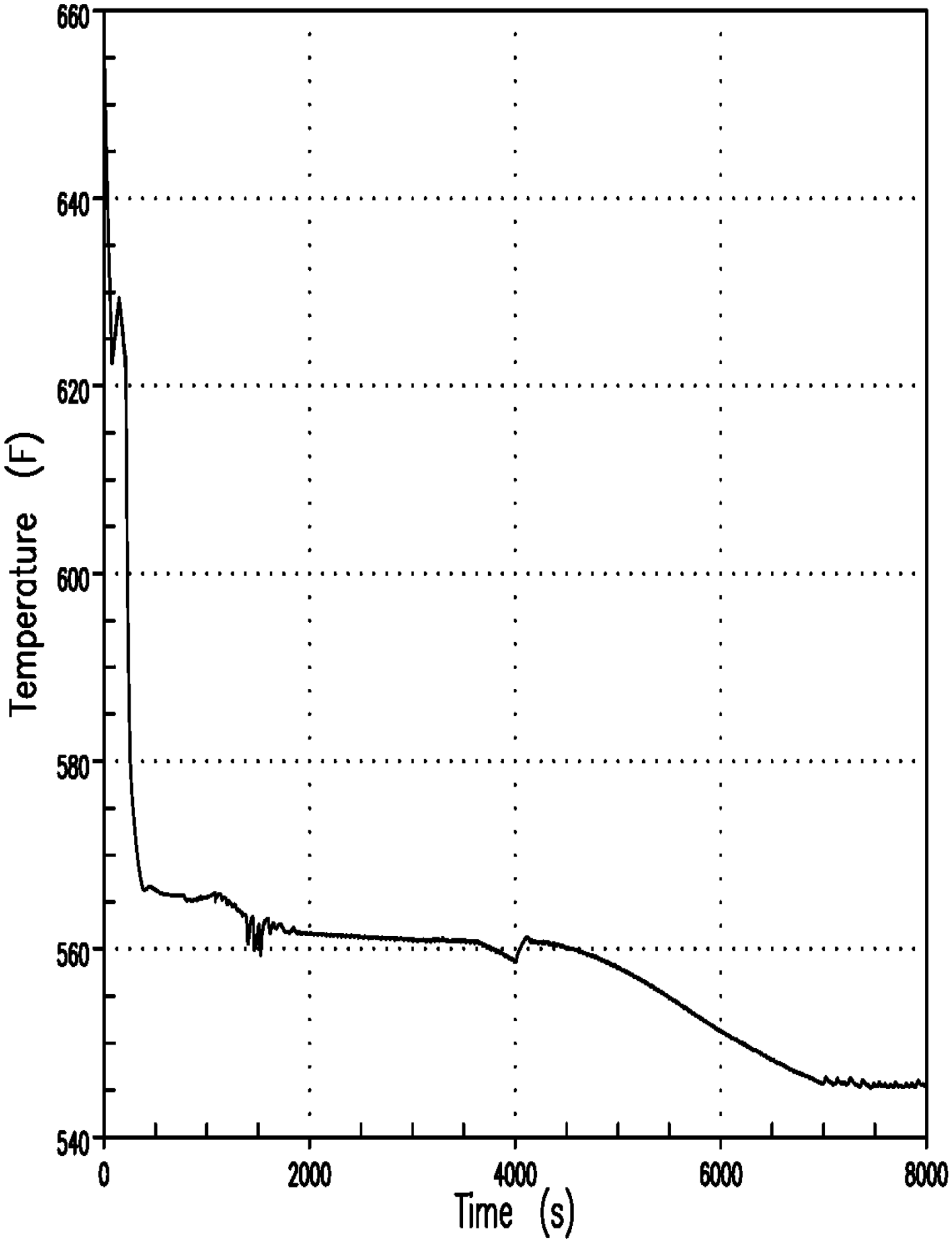
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE MIXTURE LEVEL

1.5-INCH BREAK

FIGURE 14.3.2.2-16



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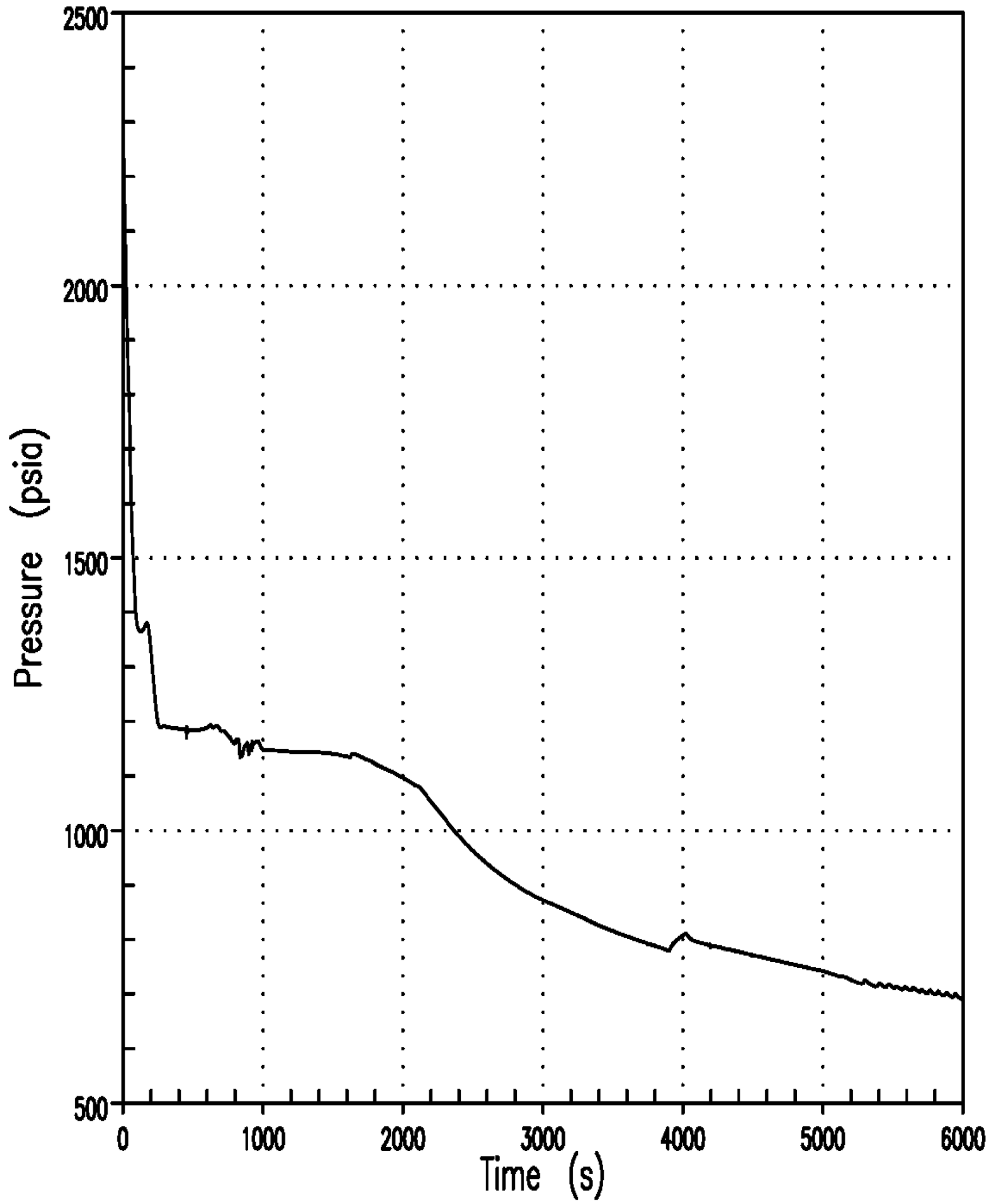
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE EXIT VAPOR TEMPERATURE

1.5-INCH BREAK

FIGURE 14.3.2.2-17



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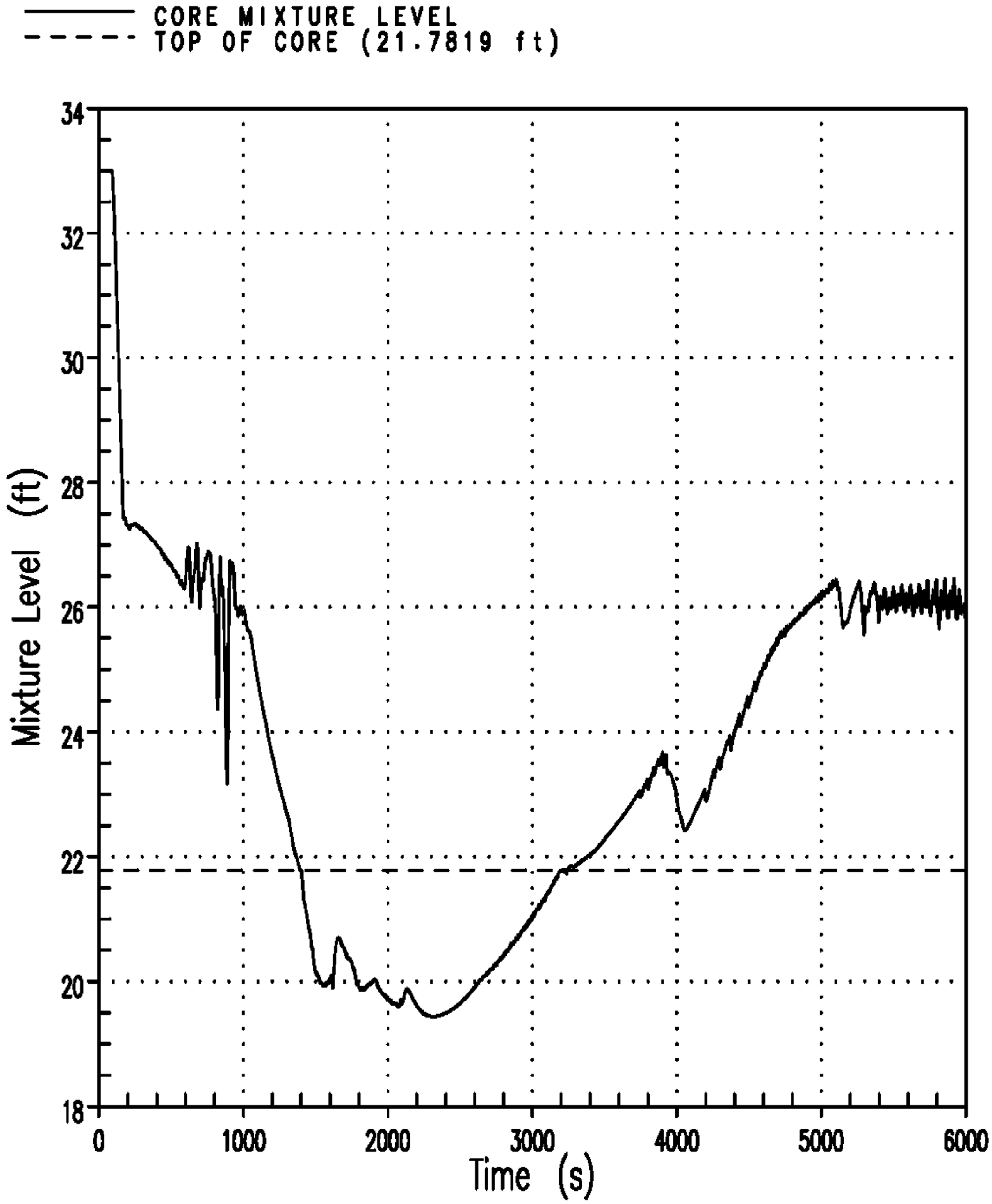
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

REACTOR COOLANT SYSTEM PRESSURE

2-INCH BREAK

FIGURE 14.3.2.2-18



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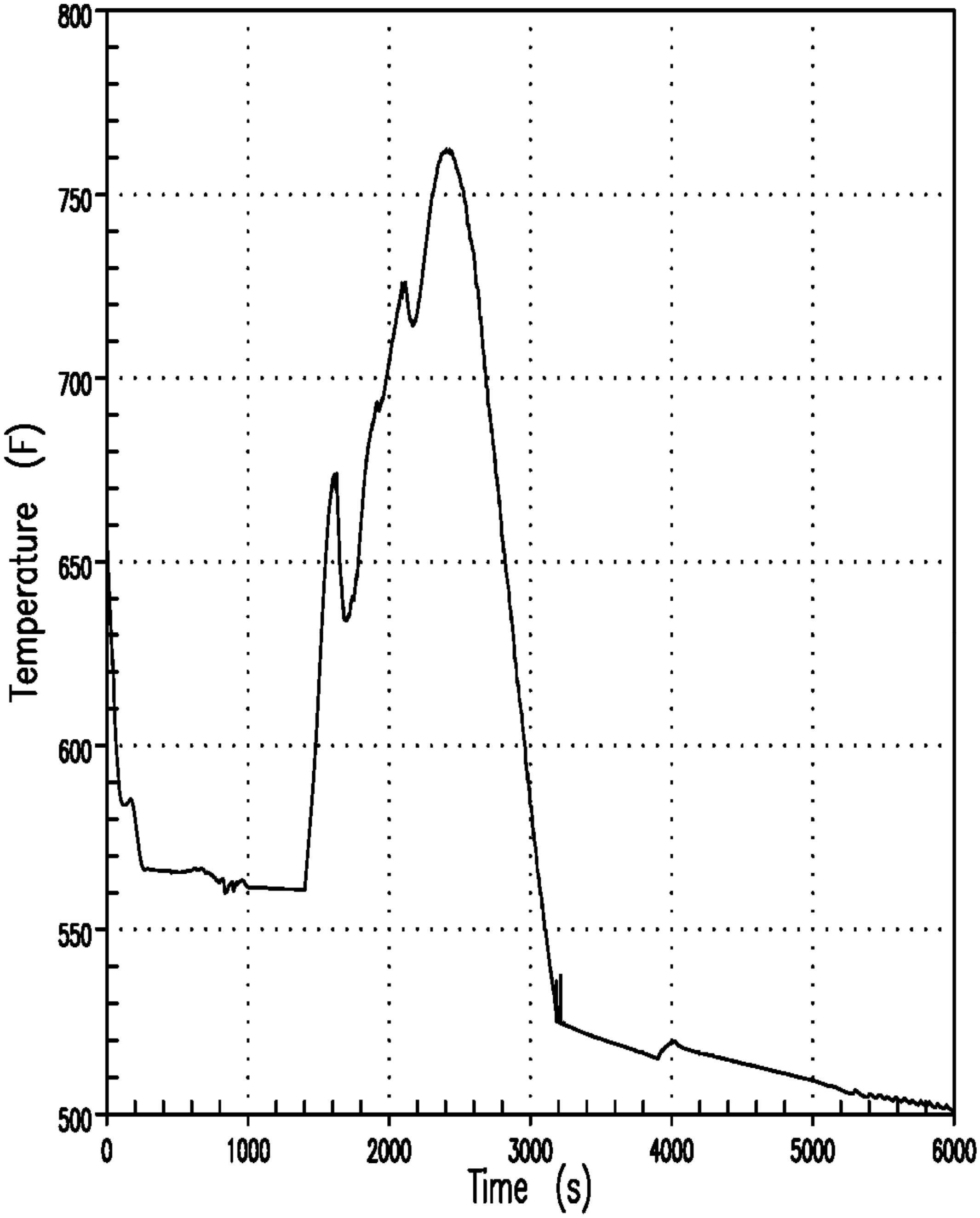
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE MIXTURE LEVEL

2-INCH BREAK

FIGURE 14.3.2.2-19



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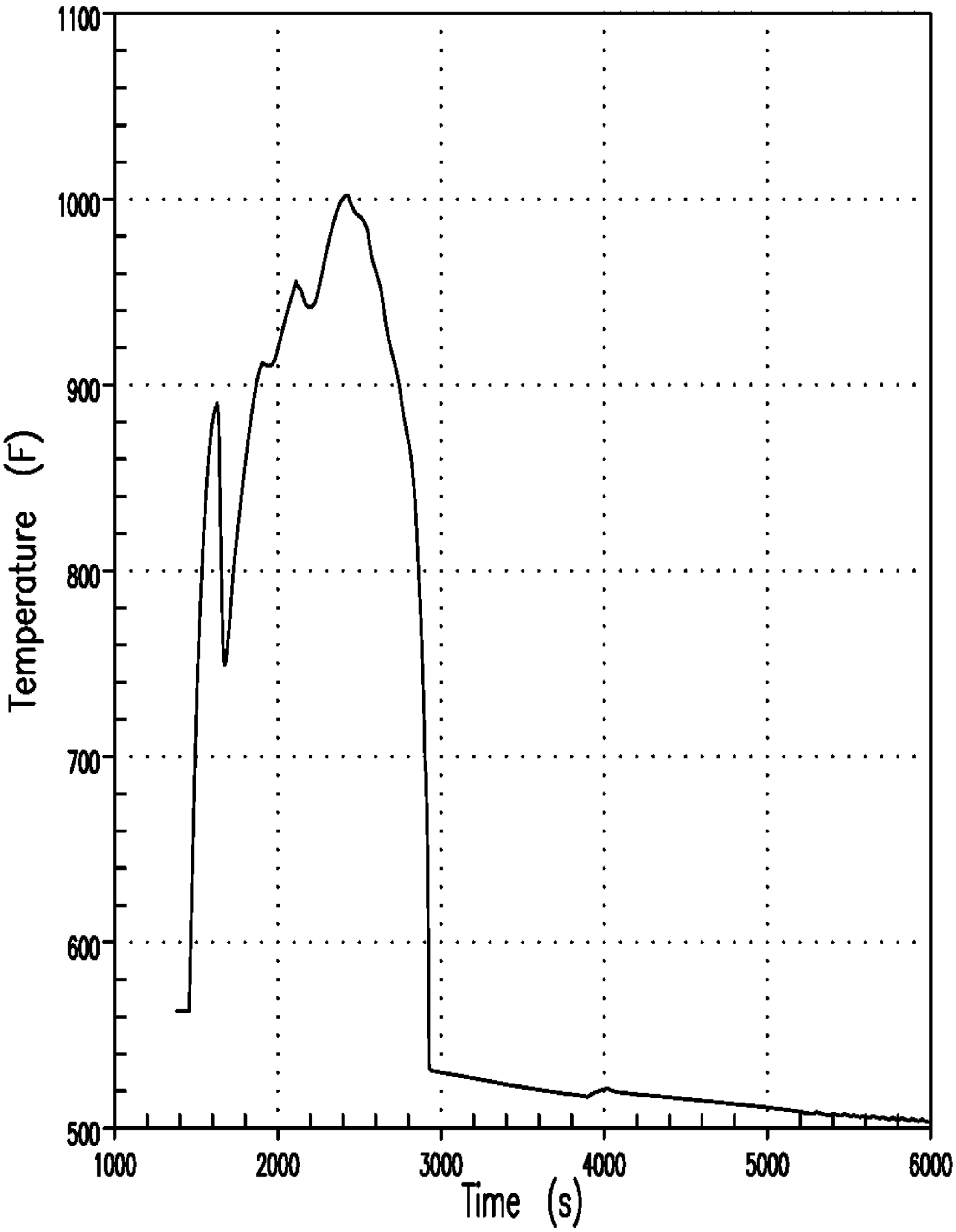
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE VAPOR TEMPERATURE

2-INCH BREAK

FIGURE 14.3.2.2-20



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TURKEY POINT PLANT UNITS 3 & 4

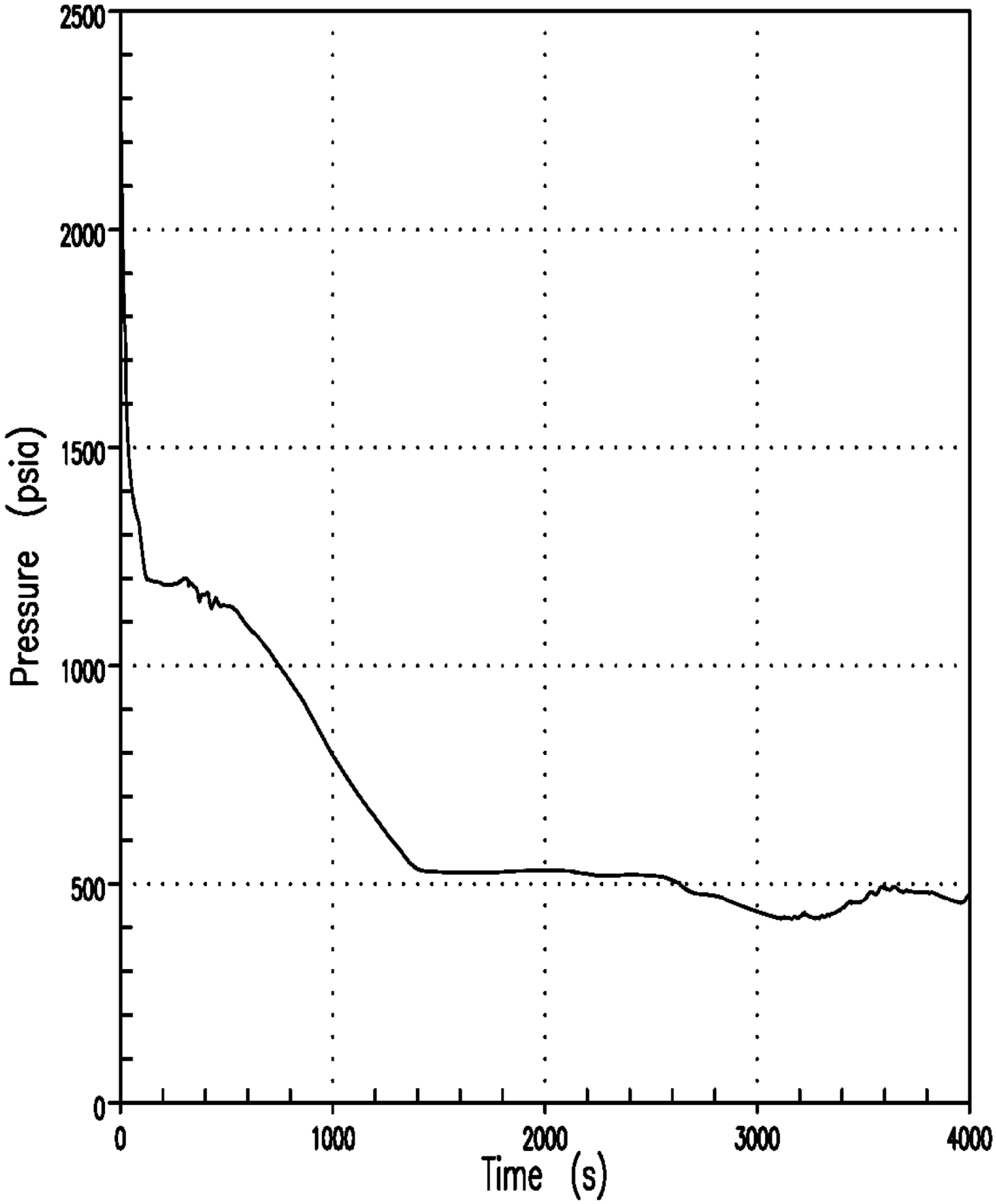
SMALL BREAK LOCA

CLADDING TEMPERATURE TRANSIENT AT

PCT ELEVATION

2-INCH BREAK

FIGURE 14.3.2.2-21



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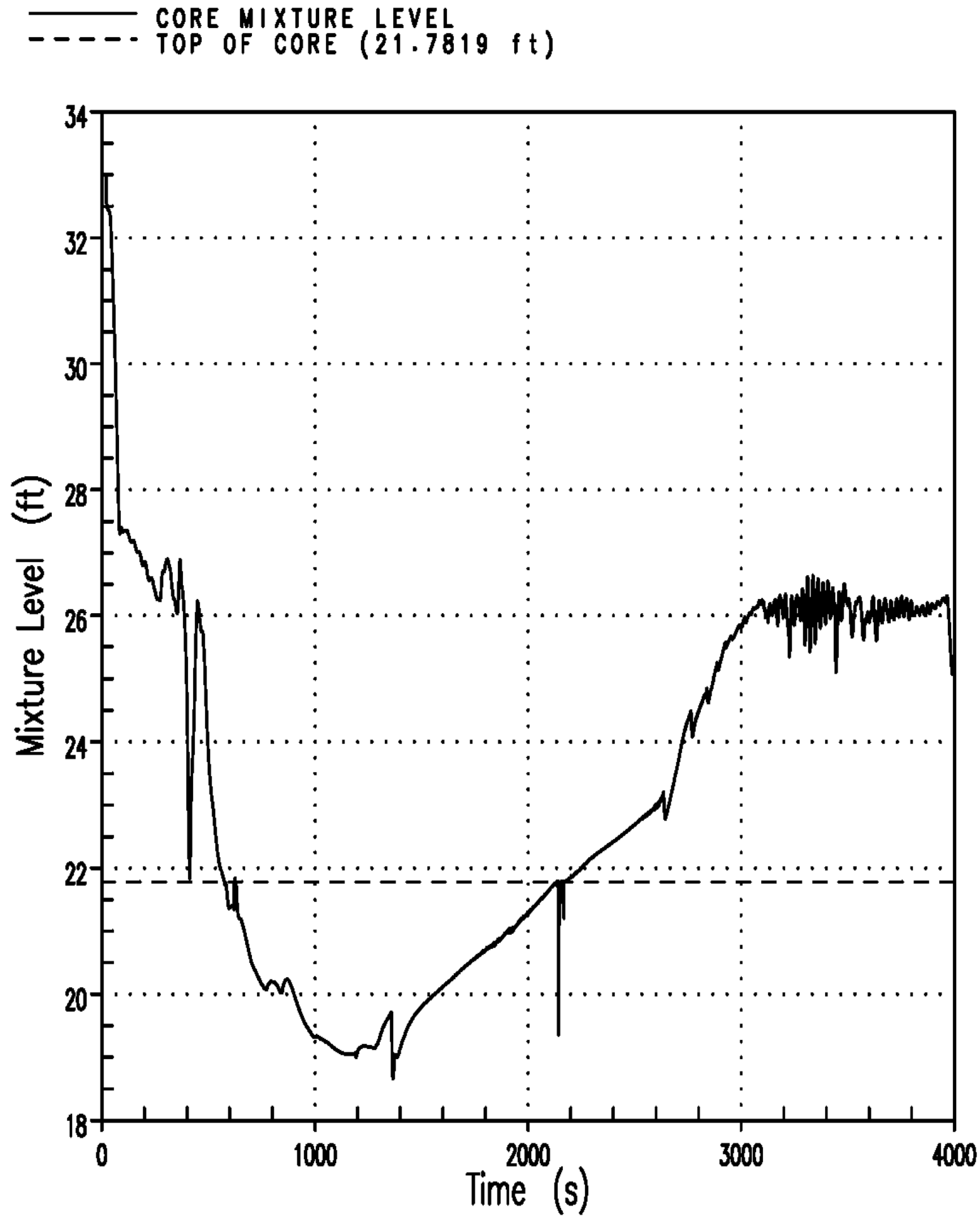
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA
REACTOR COOLANT SYSTEM PRESSURE
3-INCH BREAK

FIGURE 14.3.2.2-22



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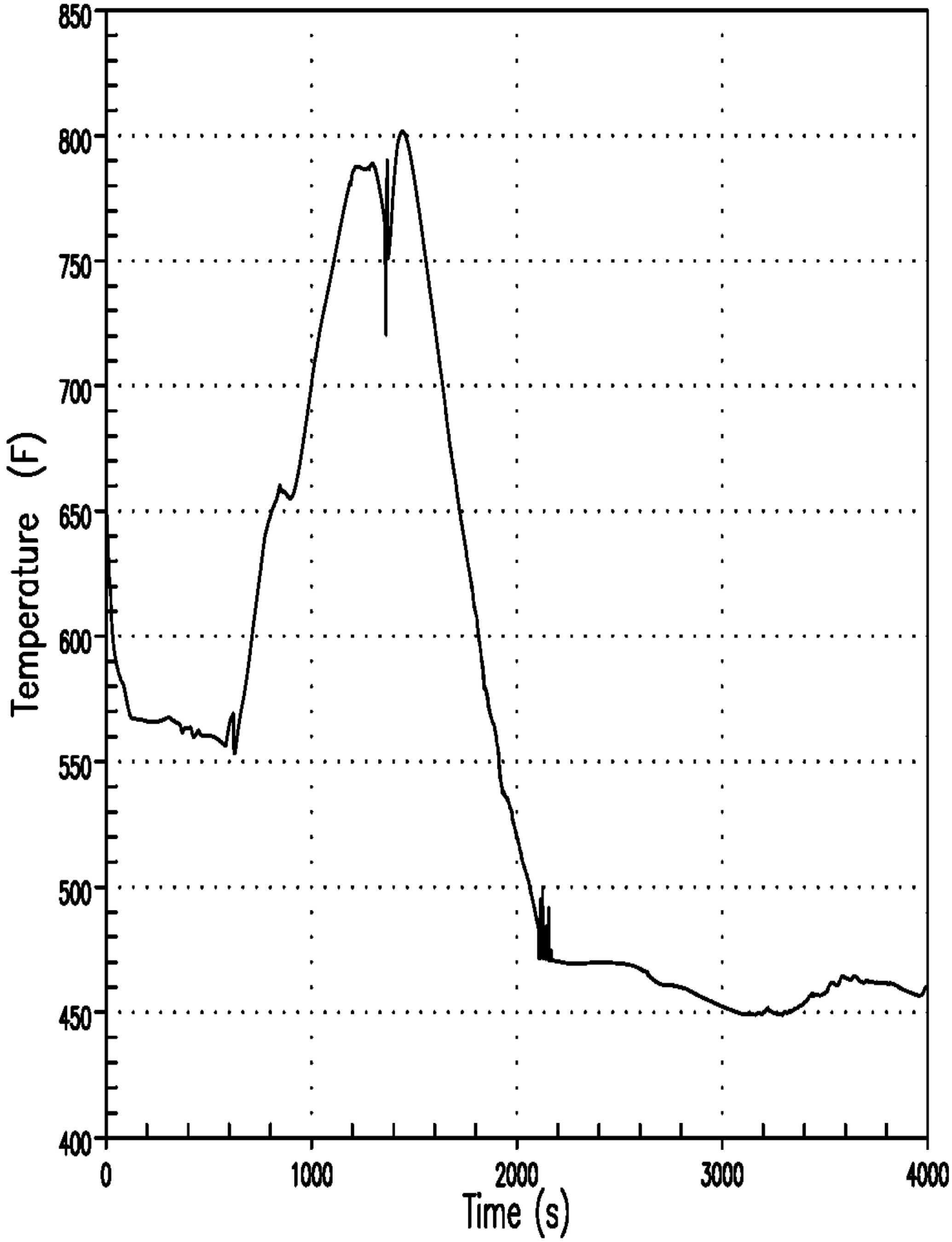
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE MIXTURE LEVEL

3-INCH BREAK

FIGURE 14.3.2.2-23



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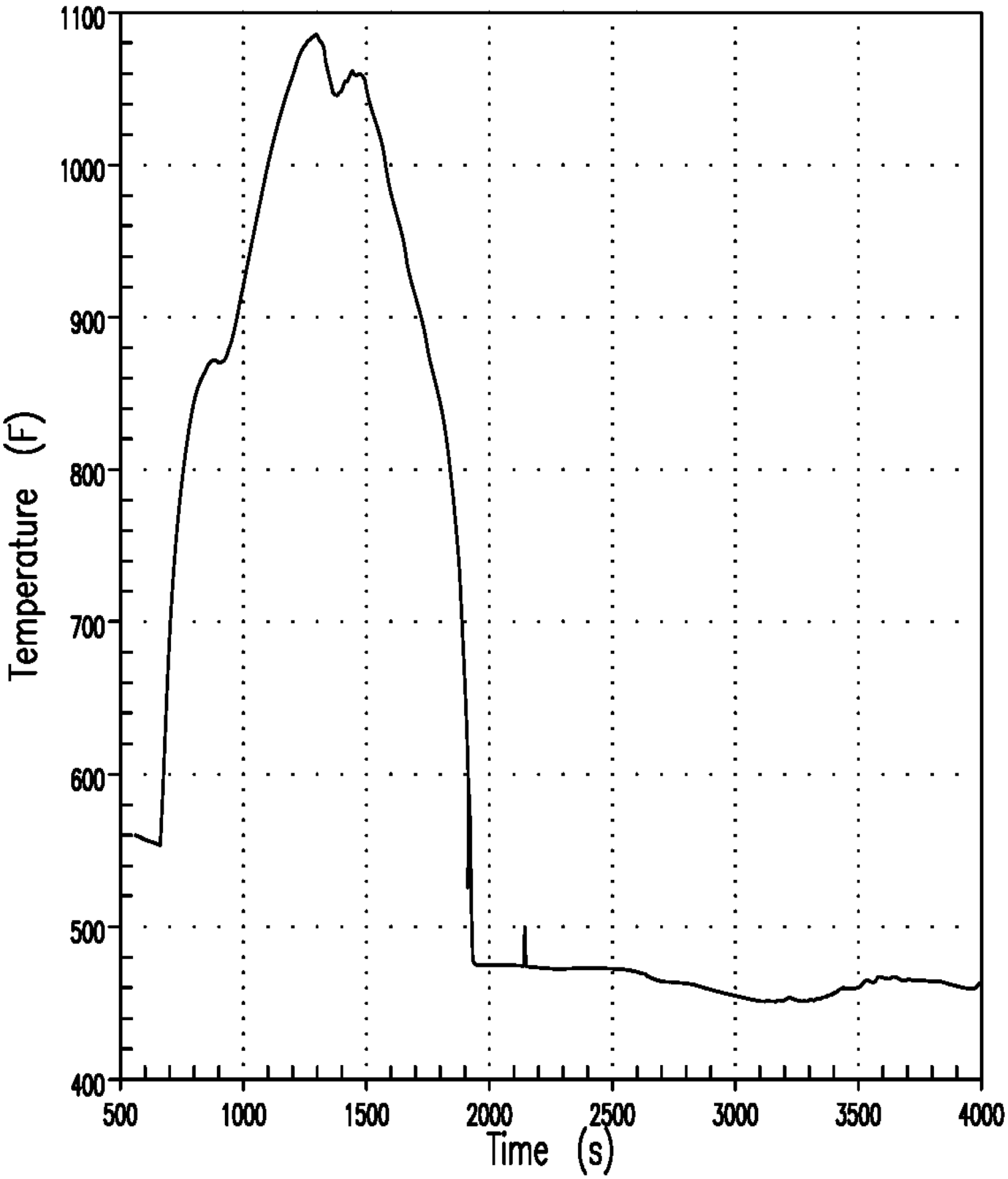
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE EXIT VAPOR TEMPERATURE

3-INCH BREAK

FIGURE 14.3.2.2-24



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TURKEY POINT PLANT UNITS 3 & 4

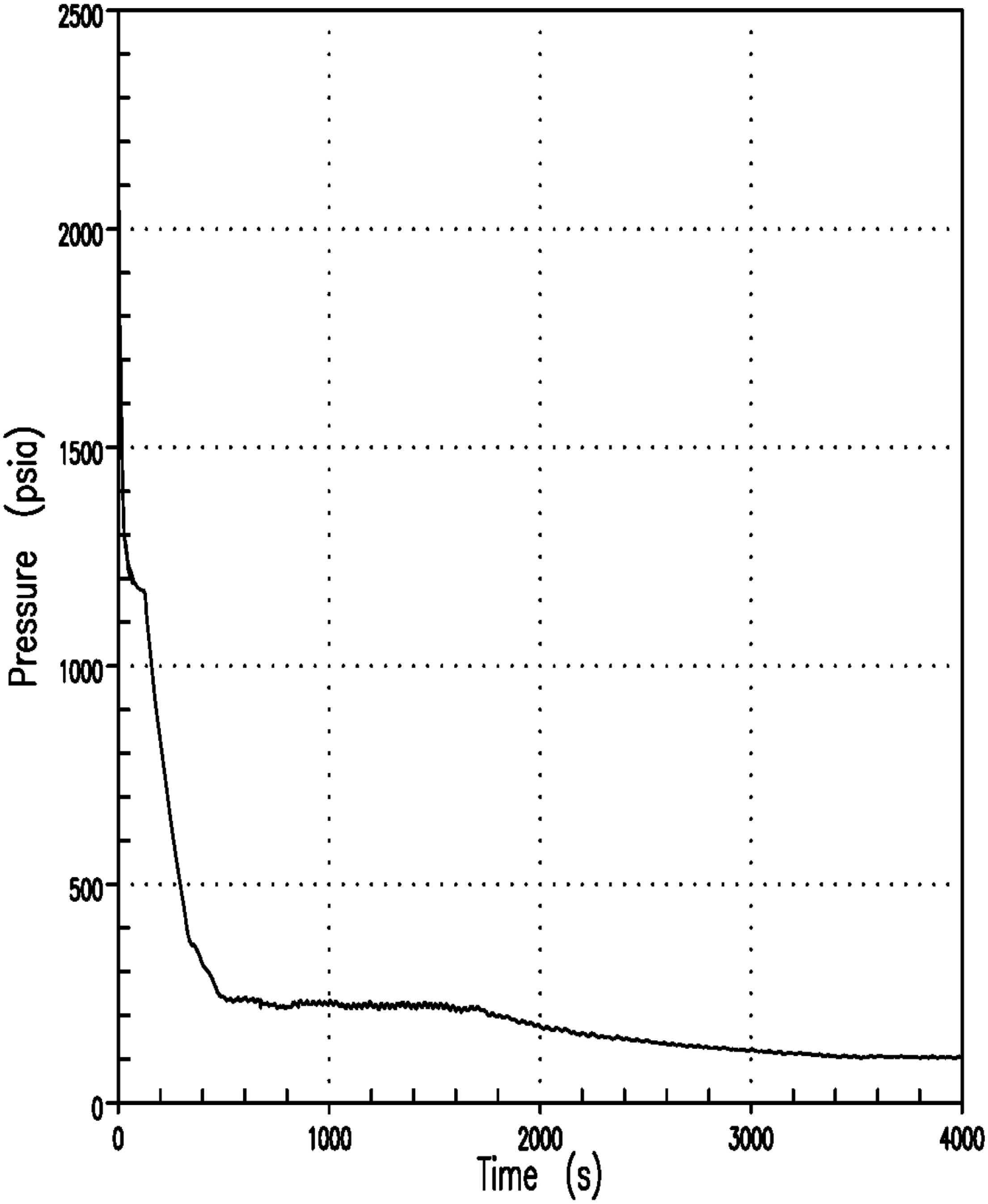
SMALL BREAK LOCA

CLADDING TEMPERATURE TRANSIENT AT

PCT ELEVATION

3-INCH BREAK

FIGURE 14.3.2.2-25



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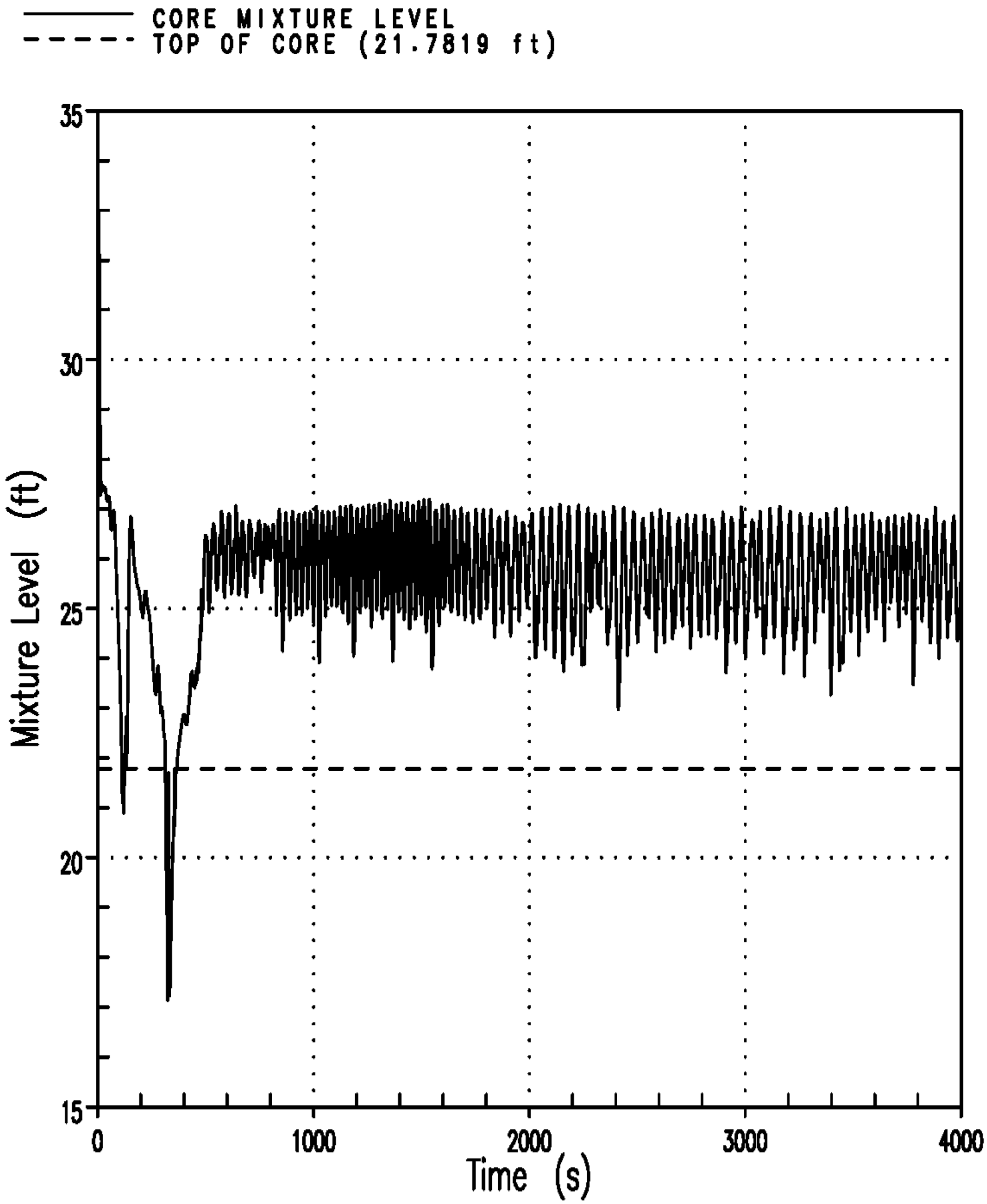
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

REACTOR COOLANT SYSTEM PRESSURE

6-INCH BREAK

FIGURE 14.3.2.2-26



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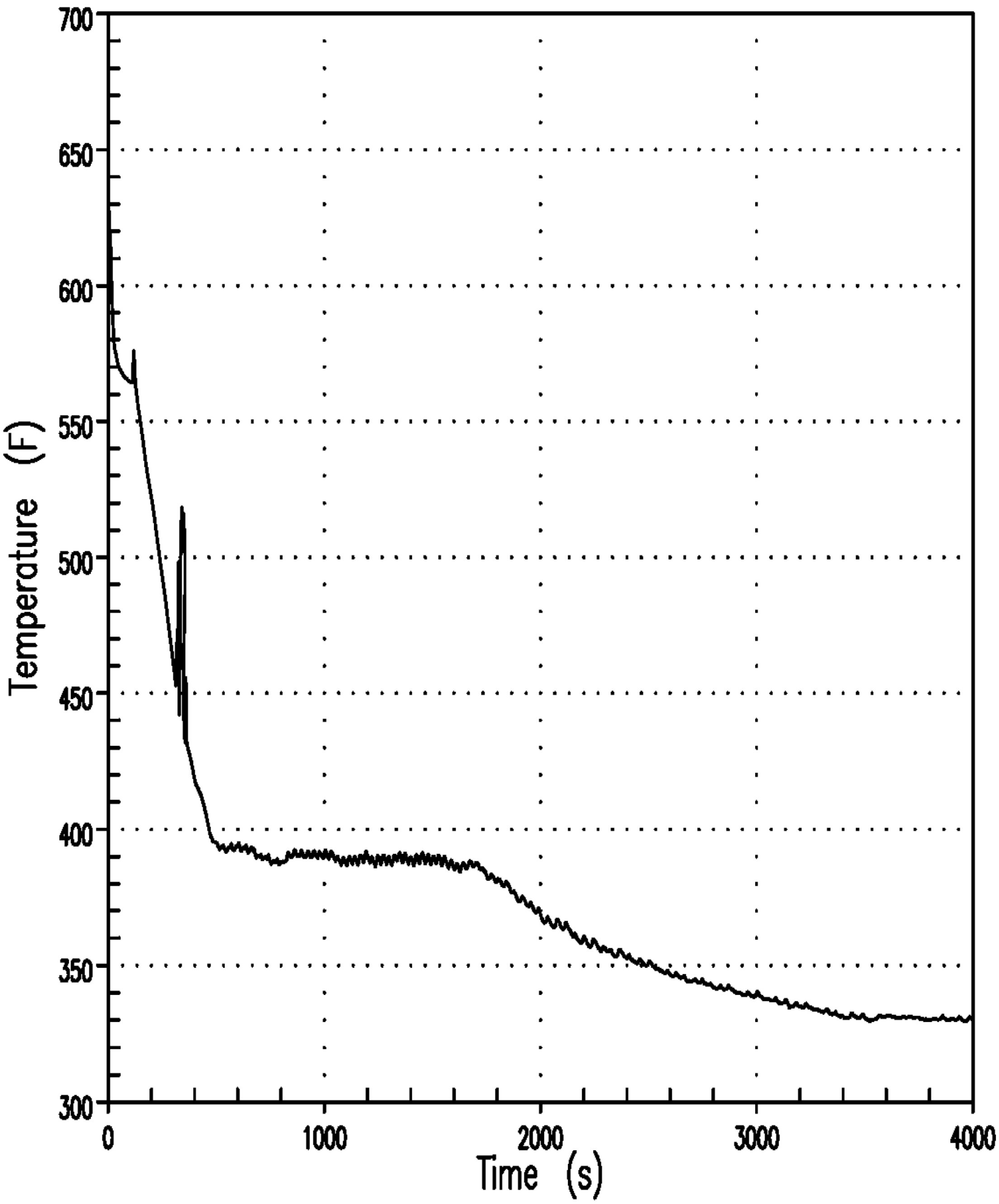
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE MIXTURE LEVEL

6-INCH BREAK

FIGURE 14.3.2.2-27



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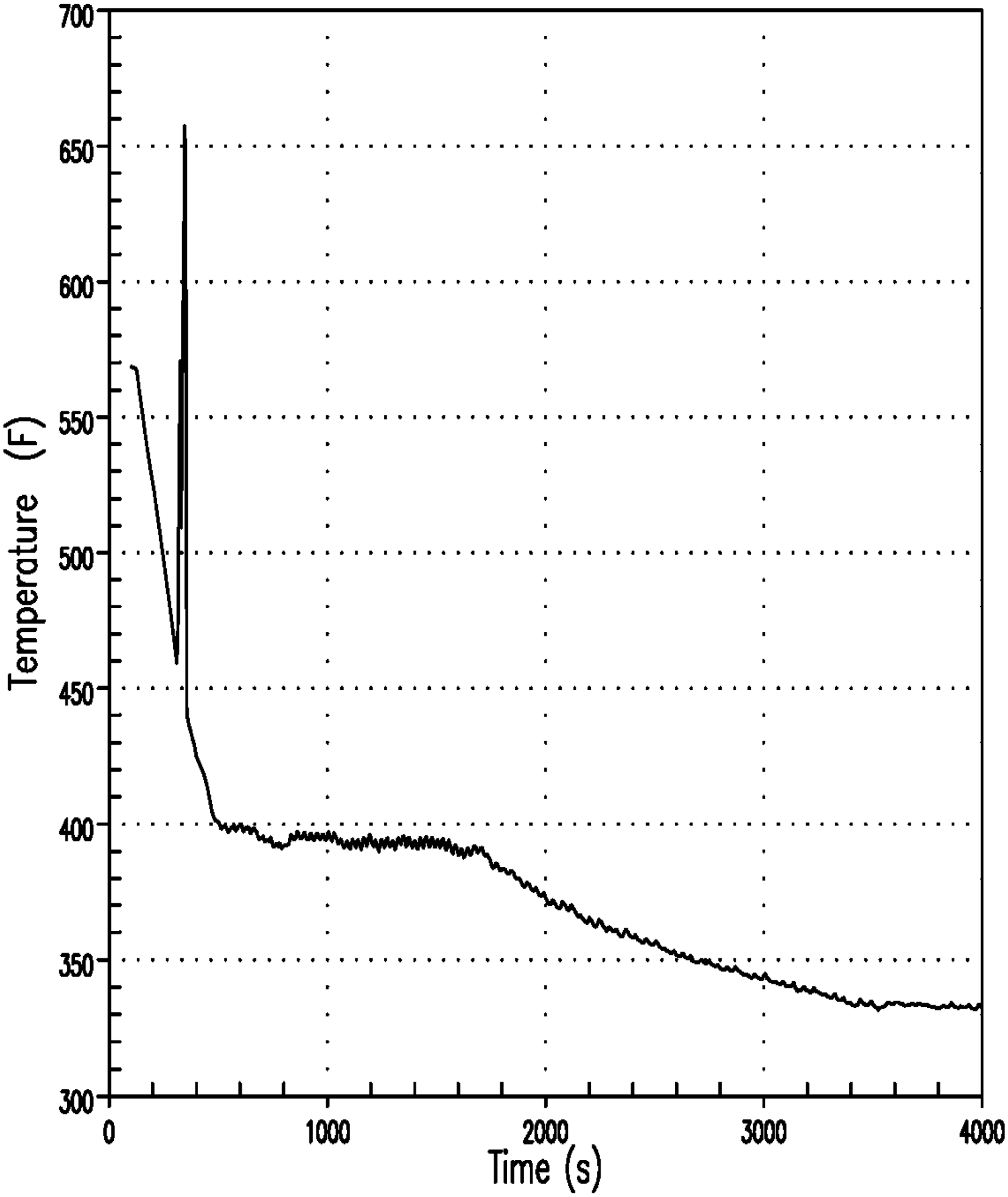
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE EXIT VAPOR TEMPERATURE

6-INCH BREAK

FIGURE 14.3.2.2-28



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TURKEY POINT PLANT UNITS 3 & 4

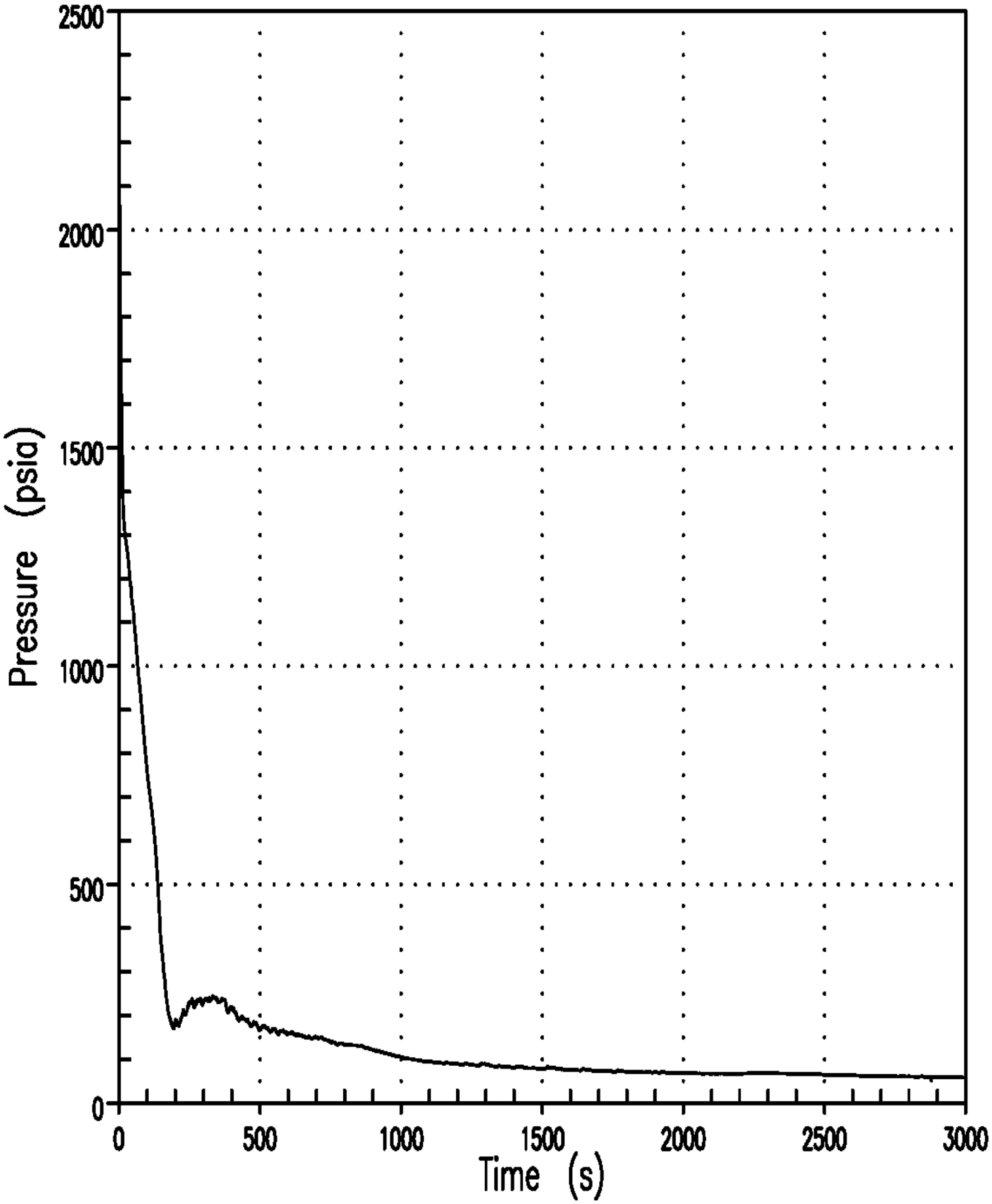
SMALL BREAK LOCA

CLADDING TEMPERATURE TRANSIENT AT

PCT ELEVATION

6-INCH BREAK

FIGURE 14.3.2.2-29



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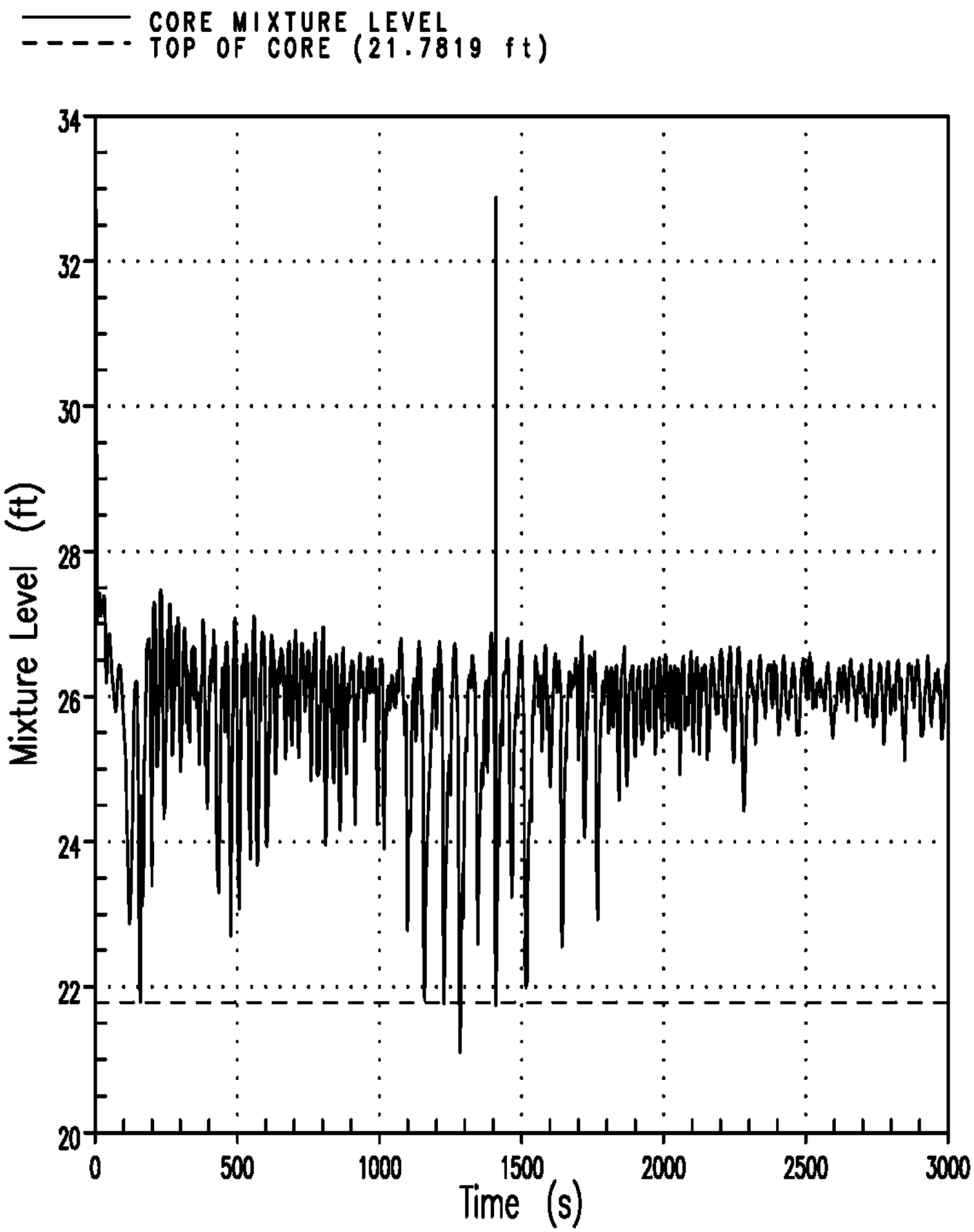
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

REACTOR COOLANT SYSTEM PRESSURE

8.75-INCH BREAK

FIGURE 14.3.2.2-30



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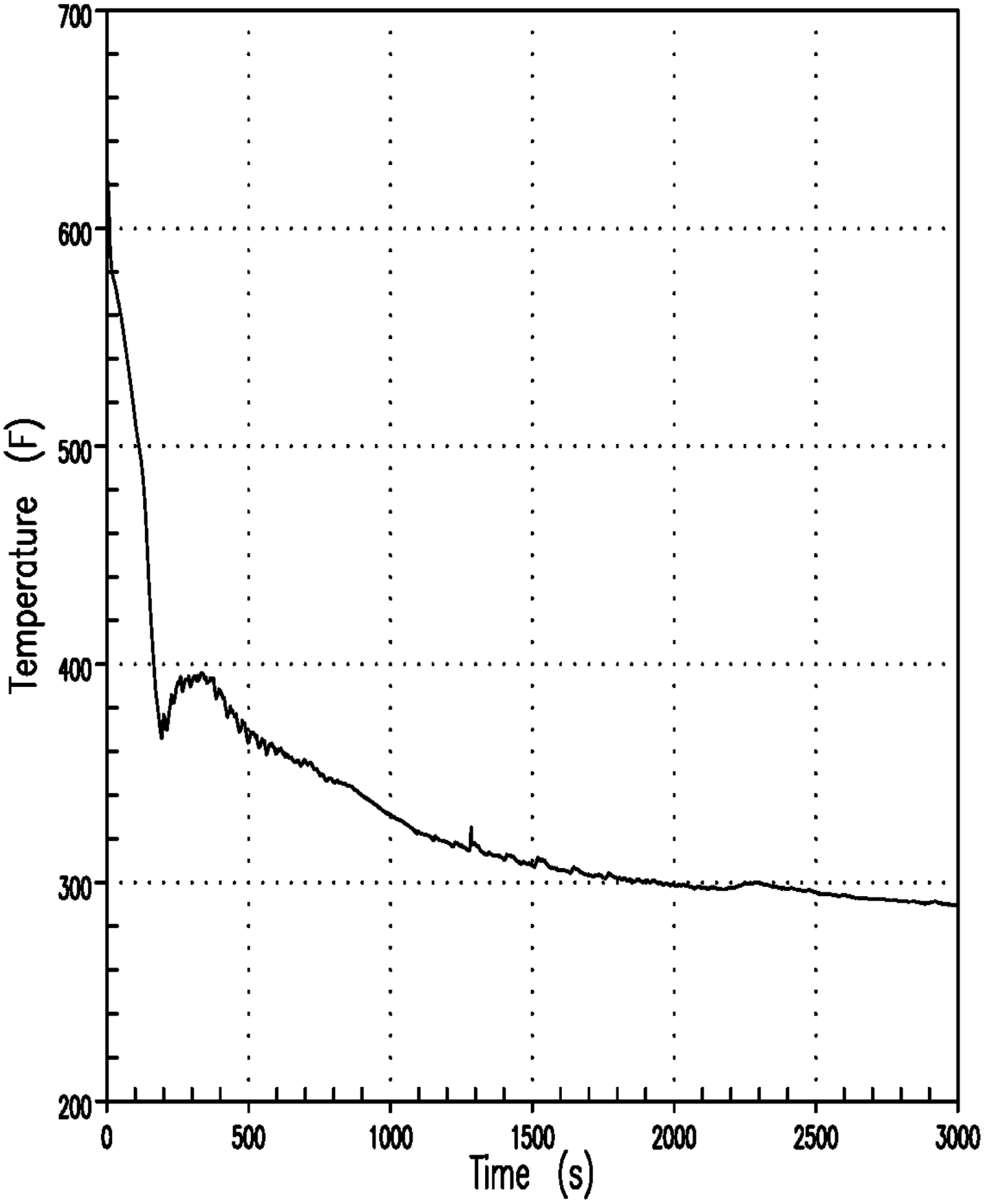
TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE MIXTURE LEVEL

8.75-INCH BREAK

FIGURE 14.3.2.2-31



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TURKEY POINT PLANT UNITS 3 & 4

SMALL BREAK LOCA

CORE EXIT VAPOR TEMPERATURE

8.75-INCH BREAK

FIGURE 14.3.2.2-32

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14.3.3 CORE AND INTERNALS INTEGRITY ANALYSIS

Internals Evaluation

The forces exerted on reactor internals and core, following a loss-of-coolant accident, are computed by employing the MULTIFLEX 3.0 digital computer program developed for the space-time-dependent analysis of multi-loop PWR plants.

Design Criteria

The criteria for acceptability are that the core should be coolable and intact following a pipe rupture up to and including a double ended rupture of the Reactor Coolant System. This implies that core cooling and adequate core shutdown must be assured. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts.

Critical Internals

Upper Barrel

The upper barrel deformation has the following limits:

> To assure reactor trip and to avoid disturbing the RCC guide structure, the barrel should not interfere with any guide tubes. This condition requires a stability check to assure that the barrel will not buckle under the accident loads.

RCC Guide Tubes

The RCC guide tubes in the upper core support package have the following allowable limits. Tests on guide tubes show that when the transverse deflection of the guide tube becomes significant, the cross section of the RCC guide tube changes. An allowable transient maximum transverse deflection of 1.0 inch has been established for the blowdown accident. Beam deflections above these limits produce cross section changes with increasing delay in scram time until the control rod will not scram due to interference between the rods and the guide. The no loss of function limit is established as 1.75 inches. With a maximum transient transverse deflection of 1.75 inches, the cross section distortion will not exceed 0.072", after load removal. This cross section distortion allows control rod insertion. For a maximum transient transverse deflection of 1.0 inch, a cross section distortion not in excess of 0.035" is anticipated.

Fuel Assemblies

The limitations for this case are related to the stability of the thimbles at the upper end. During the accident, the fuel assembly will have a vertical displacement and could touch the upper package subjecting the components to dynamic stresses.

The upper end of the thimbles shall not experience stresses above the buckling compressive stresses because any buckling of the upper end of the thimbles will distort the guide line and could affect the fall of the control rod.

Upper Package

The maximum allowable local deformation of the upper core plate where a guide tube is located is 0.100 inch. This deformation will cause the plate to contact the guide tube since the clearance between plate and guide tube is 0.100 inch. This limit will prevent the guide tubes from being put in compression. In order to maintain the straightness of the guide tube a maximum allowable total deflection of 1" for the upper support plate and deep beam has been established. The corresponding no loss of function deflection is above 2".

Allowable Stress Criteria

The allowable stress criteria fall into two categories dependent upon the nature of the stress state: membrane or bending. A direct state of stress (membrane) has a uniform stress distribution over the cross section. The allowable (maximum) membrane or direct stress is taken to be equal to the stress corresponding to 0.2 of the uniform material strain or the yield strength, whichever is higher. For unirradiated 304 stainless steel at operating temperature the stress corresponding to 20% of the uniform strain is:

$$(S_m)_{allowable} = 39{,}500 \text{ psi}$$

For irradiated materials, the limit stress is higher.

For a bending state of stress, the strain is linearly distributed over a cross-section. The average strain value is, therefore, one half of the outer fiber strain where the stress is a maximum. Thus, by requiring the average strain to satisfy an allowable criterion similar to that for the direct state of stress, the outer fiber strain may be 0.4 times the uniform strain. The maximum allowable outer fiber bending stress is then taken to be equal to the stress corresponding to 40% of the uniform strain or the yield strength, whichever is higher. For unirradiated 304 stainless steel at operating temperature, we obtain from the stress strain curve:

$$(S_b)_{allowable} = 50{,}000 \text{ psi}$$

For combinations of membrane and bending stresses, the maximum allowable stress is taken to be equal to the stress corresponding to the maximum outer fiber strain not in excess of 40% uniform strain and average strain not in excess of 20% uniform strain.

In comparing this criterion with the concept of fully plastic moment, the shape factors for rectangular cross section in <u>Resistance of Materials</u>, by Seely and Smith (Wiley, 1956) p. 232 is:

$$\frac{\sigma_2}{\sigma_1} = 1.5$$

where σ_1 = maximum allowable stress for pure axial tension.

σ_2 = fictitious outer fiber stress assuming linear stress distribution in the cross section, under the fully plastic moment.

For the faulted condition, the ratio adopted is 50,000/39,500 = 1.25 which is less than the real shape factor of 1.5.

The reference made to corresponding strains when the allowable stresses are selected (0.2 ϵ_u and 0.4 ϵ_u) is directed primarily to show margins.

Blowdown and Force Analysis

Blowdown Model

The MULTIFLEX 3.0 (Reference 4), which is an enhancement and extension of MULTIFLEX 1.0 (Reference 1), NRC reviewed and approved computer code, was employed to generate the blowdown thermal-hydraulic transient in the primary reactor coolant system due to a postulated pipe rupture, or Loss-Of-Coolant-Accident (LOCA) in both the reactor coolant system hot and cold legs. The computer program considers subcooled, transition, and two-phase (saturated) blowdown regimes, employing the method of characteristics to solve the conservation laws, assuming one dimensional flow and a homogeneous liquid-vapor mixture. With its ability to model flow branches and a large number of nodes, MULTIFLEX 3.0 has the required flexibility to represent various flow passages within the primary reactor coolant system. The reactor coolant system is divided into subregions in which the fluid flows along longitudinal axes. While each subregion is regarded as an equivalent pipe, a complex network of these equivalent pipes is used to represent the entire primary RCS.

A coupled fluid-structure interaction is incorporated into the analysis by accounting for the deflection of the constraining boundaries, which are represented by separate spring-mass oscillator systems. The reactor core barrel is modeled as an equivalent beam with the structural properties of the core barrel in a plane parallel to the broken inlet nozzle. Horizontally, the barrel is divided into ten segments, with each segment consisting of three walls. Mass and stiffness matrices that are then calculated by applying the spatial pressure variation to the wall area at each of the elevations representative of the ten mass points of the beam model. The resultant core barrel motion is then translated into an equivalent change in flow area in each downcomer annulus flow channel. At every time increment, MULTIFLEX 3.0 iterates between the hydraulic and structural subroutines for each location confined by a flexible wall.

Because of the applicability of leak-before-break licensing to the Turkey Point units, large double ended guillotine (DEG) breaks are excluded from the design basis and only limiting auxiliary line breaks are considered. For the Turkey Point units, the limiting auxiliary line breaks are the residual heat removal line break on the hot leg and the accumulator line break on the cold leg. Postulated pressurizer surge auxiliary line breaks are bounded by the residual heat removal line break.

As per Reference 3, leak-before-break has been applied to support taking credit for the control rods at the time of hot leg switchover (HLSO) during recovery from cold leg large break LOCAs. This assumption is required to address the potential for core recriticality, due to sump dilution, at the time of HLSO. Sump dilution does not occur for hot leg large break LOCAs and credit for the control rods during post-LOCA recovery is not required for these types of breaks.

Horizontal Force Model

MULTIFLEX 3.0 evaluates the pressure and velocity transients for a maximum of 2000 locations throughout the system. These pressure and velocity transients are stored as a computer file and are made available to the program FORCE2 (Reference 1. Appendix B) which utilizes a detailed geometric description in evaluating the vertical loading on the reactor internals.

Each reactor component for which force calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated summing the effects of:

1. The pressure differential across the element.

2. Flow stagnation on, and unrecovered orifice losses across the element.

3. Friction losses along the element.

Input to the code, in addition to the MULTIFLEX 3.0 calculated blowdown pressure and velocity transients, includes the effective area of each element on which acts the vertical force due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which the shear forces act.

The horizontal forces on the vessel wall, core barrel, and thermal shield are computed using the LATFORC code (Reference 1 and Appendix A).

During blowdown, significant asymmetrical loadings on the reactor vessel internals can be generated as a result of variation of the fluid pressure distribution in the downcomer annulus region. To determine these horizontal forces, LATFORC utilizes MULTIFLEX 3.0 generated field pressures, together with geometric vessel information (component radial and axial lengths).

In LATFORC, the downcomer annulus is subdivided into cylindrical segments, formed by dividing this region into circumferential and axial zones. The X (or Y) component of the hydraulic force acting on each segment is determined by multiplying the mean pressure acting over the segment by the X (or Y) projected segment area. In LATFORC, the X-axis coincides with the axis of the broken loop's inlet nozzle and the positive direction is directed away from this nozzle.

Structural Model and Method of Analysis

The mathematical model of the reactor equipment system model (RESM) is a three-dimensional nonlinear finite element model which represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The RESM three-dimensional nonlinear finite element model is shown in Figure 14.3.3-2. The entire integrated head assembly/CRDM model is incorporated into the RESM as a super element substructure. There are three concentric structural submodels connected by nonlinear impact elements and stiffness matrices.

The first submodel represents the reactor vessel and associated components. The reactor vessel is restrained by reactor vessel supports located beneath the nozzles and by the attached primary coolant piping. The reactor vessel supports, attached piping, and vessel nozzles are represented by stiffness matrices. The second submodel represents the reactor core barrel, thermal shield, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations. The third and innermost submodel represents the upper support plate, guide tubes, upper and lower support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and second by stiffness matrices and nonlinear elements. Fluid-structure interaction is included in the reactor pressure vessel model for seismic evaluation. The horizontal fluid-structure interaction is significant in the annulus between the core barrel and reactor vessel (the downcomer). Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel and nodes on the reactor vessel. The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel and core barrel. The off-diagonal terms reflect the fact that all the water mass does not participate when there is no relative motion of the vessel and core barrel.

The matrices are a function of the properties of two cylinders with a fluid in the cylindrical annulus, specifically; inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the core barrel allows inclusion of radii variations along the core barrel height and approximates the effects of core barrel beam deformation. In the finite element approach, the structure is divided into a finite number of members or elements. Nodal displacements and impact forces are stored for post-processing.

For LOCA excitation, time-history forcing functions are obtained from the LATFORC and FORCE2 computer codes described in the previous section. These codes calculate the transient forces on the reactor internals during blowdown using transient pressures and fluid velocities. For the blowdown analysis the forcing functions are applied directly to the various internal masses. For the earthquake analysis of the reactor internals, the forcing function, which is simulated earthquake response, is applied to the multi-mass system at the ground connections (the reactor vessel). Therefore, the external excitation is transmitted to the internals through the springs at the ground connections.

Results

LOCA and seismic dynamic analyses for the reactor pressure vessel and internals system were completed in Reference 2. Results show the internals are adequate to withstand blowdown and seismic forces.

Analysis of Effects of Loss of Coolant and Safety Injection on the Reactor Vessel and Internals

The following information was provided as part of the orginal plant licensing process and is considered historical in nature.

The analysis of the effects of injecting safety injection water into the reactor coolant system following a postulated loss of coolant accident are being incorporated into a WCAP report to be submitted to the AEC.

For the reactor vessel, three modes of failure are considered including the ductile mode, brittle mode and fatigue mode.

a) Ductile Mode - the failure criterion used for this evaluation is that there shall be no gross yielding across the vessel wall using the material yield stress specified in Section III of the ASME Boiler and Pressure Vessel Code. The combined pressure and thermal stresses during injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at the times during the safety injection transient.

The results of the analyses showed that local yielding may occur in approximately the inner 12 per cent of the base metal and in the cladding.

b) Brittle Mode - the possibility of a brittle fracture of the irradiated core region has been considered from both a transition temperature approach and a fracture mechanics approach.

The failure criteria used for the transition temperature evaluation is that a local flaw cannot propagate beyond any given point where the applied stress will remain below the critical propagation stress at the applicable temperature at that point.

The results of the transition temperature analysis showed that the stress-temperature condition in the outer 65 per cent of the base metal wall thickness remains in the crack arrest region at all times during the safety injection transient. Therefore, if a defect were present in the most detrimental location and orientation (i.e., a crack on the inside surface and circumferentially directed) it could not propagate any farther than approximately 35 per cent of the wall thickness, even considering the worst case assumptions used in this analysis.

The results of the fracture mechanics analysis, considering the effects of water temperature, heat transfer coefficients and fracture toughness of the material as a function of time, temperature and irradiation will be included in the report. Both a local crack effect and a continuous crack effect have been considered with the latter requiring the use of a rigorous finite element axisymmetric code.

c) Fatigue Mode - the failure criterion used for the failure analysis was the one presented in Section III of the ASME Boiler and Pressure Vessel Code. In this method the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the code allowable usage factor of one. The results of this analysis showed that the combined usage factor never exceeded 0.2, even after assuming that the safety injection transient occurred at the end of plant life.

In order to promote a fatigue failure during the safety injection transient at the end of plant life, it has been estimated that a wall temperature of approximately 1100°F is needed at the most critical area of the vessel (instrumentation tube welds in the bottom head).

The design basis of the Safety Injection System ensures that the maximum cladding temperature does not exceed the melting temperature of the cladding. This is achieved by prompt recovery of the core through flooding, with the passive accumulator and the injection systems. Under these conditions, a vessel temperature of 1100°F is not considered a credible possibility and the evaluation of the vessel under such elevated temperatures is for a hypothetical case.

For the ductile failure mode, such hypothetical rise in the wall temperature would increase the depth of local yielding in the vessel wall.

The results of these analyses show that the integrity of the reactor vessel is never violated.

The safety injection nozzles have been designed to withstand ten postulated safety injection transients without failure. This design and associated analytical evaluation were made in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The maximum calculated pressure plus thermal stress in the safety injection nozzle during the safety injection transient was calculated to be approximately 50,900 psi. This value compares favorably with the code allowable stress of 80,000 psi.

These ten safety injection transients are considered along with all the other design transients for the vessel in the fatigue analysis of the nozzles. This analysis showed the usage factor for the safety injection nozzles was 0.47 which is well below the code allowable value of 1.0.

The safety injection nozzles are not in the highly irradiated region of the vessel and thus they are considered ductile during the safety injection transient.

The effect of the safety injection water on the fuel assembly grid springs has been evaluated and due to the fact tht the springs have a large surface area to volume ratio, being in the form of thin strips, and are expected to follow the coolant temperature transient with very little lag, no thermal shock is expected and the core cooling is not compromised.

Evaluations of the core barrel and thermal shield have also shown that core cooling is not jeopardized under the postulated accident conditions.

An analysis has been made of the thermal stresses in the core support components. Analysis shows that the highest thermal stress case occurs in the core barrel. The barrel is affected by the cold water in the downcomer and the somewhat hotter water in the compartments between barrel and baffle, producing a thermal gradient across barrel wall. The lower support structure is cooled more uniformly because of the large and numerous flow holes and consequently thermal stresses are lower.

The method used to obtain the maximum barrel stresses is as follows:

1) temperature distribution across the barrel wall is computed as a function of time taking into consideration water temperatures and film coefficients.

2) assuming that the obtained thermal gradients are axisymetrically distributed, which is conservative for stresses, maximum thermal stresses are computed in the barrel considered as an infinite cylinder.

3) thermal stresses are added to primary stresses including seismic in order to obtain the maximum stress state of the barrel.

Results of studies performed for different conditions show that maximum thermal stresses in the barrel wall are below the allowable criteria given for design by Section III of the ASME Code.

REFERENCES

1. Takeuchi, K. et.al., "MULTIFLEX - A Fortran-IV Computer Program for Analyzing Thermal-Hydraulic- Structure System Dynamics," WCAP-8708-P-A, September 1977.

2. Westinghouse Technical Report WCAP-17152-P. “Turkey Point Units 3 and 4, Extended Power Uprate (EPU) Engineering Report,” August 2012. C26

3. FPL Engineering Evaluation, PTN-ENG-SEFJ-02-016, “Potential Core Recriticality During Hot Leg Switchover Following a LOCA.”

4. Takeuchi, K. et.al., "MULTIFLEX 3.0, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics Advanced Beam Model,” WCAP-9735, Revision 2, WCAP-9736 (Non-proprietary), February, 1999. C26

TABLE 14.3.3-1

DELETED C25

Security-Related Information - Withheld Under 10 CFR 2.390

REV. 14 (2/97)

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REACTOR VESSEL INTERNALS

FIGURE 14.3.3-1

Security-Related Information - Withheld Under 10 CFR 2.390

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TURKEY POINT PLANT UNITS 3 & 4

RESM NODES AND CONNECTIVITY

FIGURE 14.3.3-2

14.3.4 CONTAINMENT INTEGRITY EVALUATION

Method of Analysis

The containment system is designed such that for all break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary system pipe, the containment peak pressure is below the design pressure with margin. This section details the mass and energy releases and resulting containment response subsequent to a hypothetical loss of coolant accident (LOCA) or a main steamline break (MSLB). Containment Integrity is classified as Condition IV per ANS-51.1/N18.2-1973 (Reference 16). C26

14.3.4.1 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOCAs

14.3.4.1.1 INTRODUCTION

The purpose of this analysis was to calculate the long-term Loss-of-Coolant Accident (LOCA) mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture and double-ended hot leg (DEHL) rupture break cases with the uprated conditions for the Turkey Point Units 3 and 4 Extended Power Uprate (EPU) conditions. C26

The uncontrolled release of pressurized high temperature reactor coolant, termed a LOCA, will result in release of steam and water into the containment. This, in turn, will result in an increase in the containment pressure and temperature. The mass and energy release rates described in this section form the basis of further computations to evaluate the structural integrity of the containment following a postulated accident (see Section 14.3.4.3).

14.3.4.1.2 INPUT PARAMETERS AND ASSUMPTIONS

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Some of the most-critical items are the reactor coolant system (RCS) initial conditions, core decay heat, safety injection flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed below. Tables 14.3.4.1-1 and 14.3.4.1-2 present key data assumed in the analysis.

For the long-term mass and energy release calculations, operating temperatures which bound the highest average coolant temperature range were used as bounding analysis conditions. The modeled core rated power of 2652 MWt, which contains an adjustment for calorimetric error, was the basis in the analysis. C26

The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance to account for instrument error and deadband is reflected in the initial RCS temperatures. The initial reactor coolant system (RCS) pressure in this analysis is based on a nominal value of 2250 psia plus an allowance which accounts for the measurement uncertainty on pressurizer pressure. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

C26

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty) is modeled.

Regarding safety injection flow, the mass and energy calculation considered configurations/failures to conservatively bound respective alignments. A spectrum of cases included:

1. Diesel Failure Case (2 HHSI, 1 RHR, & 1 CS Pump)
2. Containment Spray Pump Failure Case (4 HHSI, 2 RHR, & 1 CS Pump) C26

The following assumptions were employed to assure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment.

1. Maximum expected operating temperature of the reactor coolant system (100% full-power conditions).

2. An allowance in temperature for instrument error and dead-band (+5.7°F). C26

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty).

4. Analyzed core power of 2652 MWt (includes allowance for calorimetric error). C26

5. Conservative coefficient of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer).

6. Allowance in core stored energy for effect of fuel densification.

7. A total uncertainty for fuel temperature calculation based on a statistical combination of effects and dependent upon fuel type, power level, and burnup. C26

8. An allowance for RCS initial pressure uncertainty (+37.7 psi).

9. A maximum containment backpressure equal to design pressure.

10. Steam generator tube plugging leveling (0% uniform) C26

    - Maximizes reactor coolant volume and fluid release.

    - Maximizes heat transfer area across the SG tubes.

    - Reduces coolant loop resistance, which reduces the Δp upstream of the break and increases break flow.

Thus, based on the previously discussed conditions and assumptions, a bounding analysis of Turkey Point Units 3 and 4 is made for the release of mass and energy from the RCS in the event of a LOCA at 2652 MWt (including uncertainty). C26

14.3.4.1.3 DESCRIPTION OF ANALYSES

The evaluation model used for the long-term LOCA mass and energy release calculations was the March 1979 model described in Reference 1. This evaluation model has been reviewed and approved generically by the NRC. This model is consistent with previous Turkey Point approved analyses and is not a change in methodology. C26

This section presents the long-term LOCA mass and energy releases that were generated in support of the Turkey Point Units 3 and 4 thermal uprating program. These mass and energy releases are then subsequently used in the containment integrity analysis presented in Section 14.3.4.3.

14.3.4.1.4 LOCA MASS AND ENERGY RELEASE PHASES

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA mass and energy analysis, is typically divided into four phases.

1. Blowdown - the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium state.
2. Refill - the period of time when the lower plenum is being filled by accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

14.3.4.1.5 COMPUTER CODES

The Reference 1 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, and FROTH. These codes were used to calculate the long-term LOCA mass and energy releases for Turkey Point Units 3 and 4.

The SATAN VI code calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation, including pressure, enthalpy, density, mass and energy flowrates, and energy transfer between primary and secondary systems as a function of time.

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the Emergency Core Cooling refills the reactor vessel and provides cooling to the core. The most-important feature is the steam/water mixing model (refer to Section 14.3.4.1.8.2).

FROTH models the post-reflood portion of the transient. The FROTH code is used for the steam generator heat addition calculation from the broken and intact loop steam generators.

14.3.4.1.6 BREAK SIZE AND LOCATION

Generic studies have been performed with respect to the effect of postulated break size on the LOCA mass and energy releases. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator).
2. Cold leg (between pump and vessel).
3. Pump suction (between steam generator and pump).

The break locations analyzed for this program are the double-ended pump suction (DEPS) rupture (10.5 ft^2), and the double-ended hot leg (DEHL) rupture (9.2 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion on each break location. C26

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the steam generator secondary is minimal because the majority of the fluid which exits the core bypasses the steam generators venting directly to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the mass and energy releases for the hot leg break blowdown phase are calculated and presented in this section.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the cold leg break is not included in the scope of this uprating.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to containment.

14.3.4.1.7 APPLICATION OF SINGLE-FAILURE CRITERION

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. Modeling a loss of offsite power results in extended delays to the start of various equipment (pumps, fan coolers, etc.) which is conservative in the analysis, but the primary impact of considering the loss of offsite power is the ability to postulate a loss of a diesel generator as the single failure. Assuming the loss of a diesel generator as a single failure corresponds to the loss of one safety injection train and the containment safeguards components on the faulted diesel. This minimizes safety injection flow and containment cooling resulting in a maximum calculated containment pressure. This is not an issue for the blowdown period which is limited by the DEHL break. C26

Two cases have been analyzed for the effects of a single failure. The first case postulated the single failure as the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train. The second case is the assumed failure of a containment spray pump. As compared to the first case, the SI flow would be greater and the time of RWST depletion would be earlier. The analysis of the cases described provides confidence that the effect of credible single failures is bounded. C26 C26

14.3.4.1.8 MASS AND ENERGY RELEASE DATA

14.3.4.1.8.1 BLOWDOWN MASS AND ENERGY RELEASE DATA

A version of the SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in Reference 1.

Table 14.3.4.1-3 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break; break path 2 refers to the mass and energy exiting from the steam generator side of the break.

Table 14.3.4.1-6 presents the calculated mass and energy releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the steam generator side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

14.3.4.1.8.2 REFLOOD MASS AND ENERGY RELEASE DATA

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped safety injection and accumulators, reactor coolant pump performance, and steam generator release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; i.e., the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and emergency core cooling injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation. Even though the Reference 1 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 1). This assumption is justified and supported by test data, and is summarized as follows. C26

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3-scale tests (Reference 3), which are the largest scale data available and thus most-clearly simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3-scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flowrates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3-scale steam/water mixing data.

Additionally, the following justification is also noted. The post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double ended rupture break. For this break, there are two flowpaths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECCS injection nozzle. A description of the test and test results is contained in References 1 and 3.

Table 14.3.4.1-7 presents the calculated mass and energy release for the reflood phase of the pump suction double-ended rupture with a single limiting failure of a diesel generator. This failure case was the most limiting for the LOCA containment integrity analysis (see Section 14.3.4.3) for the post-blowdown phase. The spray-pump failure scenario was analyzed, but since the diesel failure is the most limiting it will be presented. C26

The transients of the principal parameters during reflood are given in Table 14.3.4.1-8 for the DEPS diesel failure case.

14.3.4.1.8.3 POST-REFLOOD MASS AND ENERGY RELEASE DATA

The FROTH code (Reference 4) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture level present in the steam generator tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop steam generators. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the steam generators contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs and becomes superheated as it passes through the steam generator. Once the broken loop cools, the break flow becomes two phase.

Once the broken loop cools, the break flow becomes two phase. The methodology for the use of this model is described in Reference 1. The mass and energy release rates are calculated by FROTH until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Table 14.3.4.1-9 presents the two-phase post-reflood (FROTH) mass and energy release data for the DEPS diesel-failure case.

14.3.4.1.8.4 DECAY HEAT MODEL

A plant-specific decay heat curve following American Nuclear Society Standard 5.1 was used in the LOCA M&E release model for Turkey Point Units 3 and 4 for the determination of decay heat energy. This standard was balloted by the Nuclear Power Plant Standards Committee in October 1978 and subsequently approved. The official standard (Reference 5) was issued in August 1979. The curve is based off Turkey Point specific information based upon the fuel and operating conditions. The fuel related parameters will be tracked on a cycle to cycle basis via the reload safety analysis checklist process. C26

Significant assumptions in the generation of the decay heat curve for use in design basis containment integrity LOCA analyses include:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10, of Reference 5. C26

5. The fuel has been assumed to be at full power for 10^8 seconds.

6. The number of atoms of U-239 produced per second has been assumed to be equal to 70% of the fission rate.

7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

8. Two-sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Based upon NRC staff review, Safety Evaluation Report (SER) of the March 1979 evaluation model, use of the ANS Standard-5.1, November 1979 decay heat model was approved for the calculation of mass and energy releases to the containment following a loss-of-coolant accident.

14.3.4.1.8.5 STEAM GENERATOR EQUILIBRATION AND DEPRESSURIZATION

Steam generator (SG) equilibration and depressurization is the process by which secondary side energy is removed from the SG in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is Tsat at the containment design pressure. After the FROTH calculations, SG secondary energy is removed based on first and second stage rates. The first stage rate is applied until the SG reaches Tsat at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second stage rate is used until the final depressurization, when the secondary reaches the reference temperature of Tsat at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop steam generators are calculated separately.

During the FROTH calculations, SG heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. SG energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time.

At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds (i.e., 14.7 psia and 212°F).

14.3.4.1.8.6 SOURCES OF MASS AND ENERGY

The sources of mass and energy considered in the LOCA mass and energy release analysis are given in Table 14.3.4.1-10. These sources are the reactor coolant system, accumulators, and pumped safety injection. C26

The sources include: C26

1. Reactor Coolant System Water

2. Accumulator Water

3. Pumped Injection Water

4. Decay Heat

5. Core Stored Energy

6. Reactor Coolant System Metal - Primary Metal (includes SG tubes)

7. Steam Generator Metal (includes transition cone, shell, wrapper, and other internals)

8. Steam Generator Secondary Energy (includes fluid mass and steam mass)

9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

Energy Reference Points

1. Available Energy: 212°F; 14.7 psia
2. Total Energy Content: 32°F; 14.7 psia

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of broken loop steam generator equilibration to pressure setpoint
6. Time of intact loop steam generator equilibration to pressure setpoint
7. Time of full depressurization (3600 seconds)

The energy release from the metal-water reaction rate is considered as part of the WCAP-10325-P-A (Reference 1) methodology. Based on the way that the energy in the fuel is conservatively released to the vessel fluid, the fuel cladding temperature does not increase to the point where the metal-water reaction is significant. This is in contrast to the 10 CFR 50.46 analyses, which are biased to calculate high fuel rod cladding temperatures. For the LOCA mass and energy release calculation, the energy created by the metal-water reaction value is small and is not explicitly provided in the energy balance tables. The energy that is determined is part of the mass and energy releases and is therefore already included in the overall mass and energy releases for Turkey Point Units 3 and 4.

14.3.4.1.9 CONCLUSIONS

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied. Any other conclusions cannot be drawn from the generation of mass and energy releases directly since the releases are inputs to the containment integrity analyses. The containment response must be performed. See Section 14.3.4.3 for the LOCA containment integrity conclusions.

14.3.4.2 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED SECONDARY SYSTEM PIPE RUPTURES INSIDE CONTAINMENT

14.3.4.2.1 INTRODUCTION

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steamline rupture is dependent upon the many possible configurations of the plant steam system and containment designs as well as the plant operating conditions and the size of the rupture. These variations make a reasonable determination of the single absolute worst case for both containment pressure and temperature evaluations following a steamline break difficult. The analysis considers a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size in determining the containment response to a secondary system pipe rupture.

14.3.4.2.2 INPUT PARAMETERS AND ASSUMPTIONS

The postulated break area can have competing effects on blowdown results. The large break areas maximize the rate of mass and energy release. The small split break can result in higher integrated mass and energy releases due to a longer period before a protection actuation signal is received. C26

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steamline, spectrums of both variables have been evaluated. At plant power levels of 100%, 70%, 30% and 0% of nominal full-load power, two break types have been defined. These break areas are defined as the following. C26

1. A full double-ended rupture (DER) downstream of the flow restrictor in one steamline. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. No water entrainment is credited in the break effluent.

2. A small split rupture that will neither generate a steamline isolation signal from the Westinghouse Engineered Safety Features nor result in water entrainment in the break effluent.

The analysis examined a full spectrum of cases, varying both single failures and power levels at EPU conditions. The results of the analysis are presented in Section 14.3.4.2.4 for Turkey Point Units 3 and 4. Initial containment conditions for the limiting case were assumed to be +1.4 psig and 130°F based on the allowable technical specification value plus instrument uncertainty. The split steamline break was modeled assuming isolation is accomplished by the main steam isolation valve in each intact steamline. The important plant conditions and features that were assumed are discussed in the following paragraphs. C28

14.3.4.2.2.1 INITIAL POWER LEVEL

Steamline breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power levels will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the steam generators, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during "at-power" operation may be greater than for breaks occurring with the plant in a hot-shutdown condition. Additionally, steam pressure and the dynamic conditions in the steam generators change with increasing power and have a significant influence on the rate of blowdown.

Because of the opposing effects (mass versus energy release) of changing power level on steamline break releases, no single power level can be singled out as a worst case initial condition for a steamline break event. Therefore, several different power levels spanning from full- to zero-power conditions have been investigated for Turkey Point Units 3 and 4. C28

In general, the plant initial conditions are assumed to be at the nominal value corresponding to the initial power plus appropriate uncertainties. Table 14.3.4.2-1 identifies the values assumed for RCS pressure, RCS vessel average temperature, pressurizer water volume, steam generator water level, and feedwater enthalpy corresponding to the limiting steamline break case analyzed. C26

14.3.4.2.2.2 SINGLE-FAILURE ASSUMPTIONS C26

Four single failures were examined for this analysis. Each single failure either increases the mass and energy (M&E) released out of the break or decreases the containment heat removal. An overview of each single failure is summarized below.

Auxiliary Feedwater Control Valve Failure

The Auxiliary feedwater (AFW) modeling credits a control valve that limits the maximum auxiliary feedwater flowrate for a defined duration. The single failure of this control valve allows a higher AFW flowrate for the duration of the transient.

Feedwater Control Valve (FCV) Failure

The main feedwater system contains two feedwater isolation valves (FIV) in series on each loop specific feedline section. If the FCV is postulated to fail open, the back-up valve, the FIV, is credited to close. The back-up valve has a slower closure time than the main FCV, and additional pumped main feedwater may enter the SG. The FIV is further away from the SG than the FCV, and there is a larger unisolable feedline volume that will allow more feedwater to flash into the faulted SG as it depressurizes below the saturation pressure of the feedwater. C26

Steamline Check Valve Failure

The steamline check valve is credited to function when another single failure is postulated. When the check valve is postulated as the single failure the check valve on the faulted loop does not close. The steamline blowdown will include steam mass from the steamline header/intact SGs until isolation of the break is credited due to the closure of the MSIVs on the intact SGs.

Containment Safeguards Failure

The containment safeguards failure is the loss of safeguards train due to the breakdown of an EDG sequencer. It is postulated that the EDG sequencer could fail with or without the presence of offsite power. For MSLB, this would result in the loss of both a CS pump and an ECC. All M&E data produced uses safety injection flows that assume the loss of one train of safeguards; therefore the M&E input is appropriate for the containment safeguards failure.

14.3.4.2.2.3 MAIN FEEDWATER SYSTEM

Main feedwater flow was conservatively modeled by assuming an increase in feedwater flow in response to increases in steam flow following initiation of the steamline break. This maximizes the total mass addition prior to feedwater isolation. The feedwater control valves are the primary method of isolating the main feedwater, with a total delay of 9 seconds following a safety injection signal. The feedwater isolation valves can be credited for secondary feedwater isolation, with a total delay of 30 seconds. The main feedwater bypass line also contains tandem primary and secondary isolation valves with the same isolation times.

Following feedwater isolation, as the SG pressure decreases, some of the fluid in the feedwater lines downstream of the isolation valve may flash to steam if the feedwater temperature exceeds the saturation pressure. This unisolable feedwater line volume is an additional source of high-energy fluid that was assumed to be discharged out of the break. The unisolable volume in the feedwater lines is maximized for the faulted loop and minimized for the intact loops.

The following piping volumes available for steam flashing were assumed in the analysis.

- Volume from SG nozzle to FCV (faulted loop) - 178.31 ft^3
- Volume from SG nozzle to FCV (intact loops) - 123.5 ft^3 and 100.32 ft^3

14.3.4.2.2.4 AUXILIARY FEEDWATER SYSTEM

Generally, within the first minute following a steamline break, the AFW will be initiated on any one of several protection system signals. Addition of AFW to the SG will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The AFW FCVs are set to supply a fixed flow to each SG regardless of the back pressure in the SG. A higher AFW flowrate to the faulted loop SG is conservative for the steamline break (SLB) event. Conversely, a lower AFW flowrate is conservative for the intact loop SGs. The intact loops SGs receive 280 gpm/SG for the entire duration of the SLB event. The faulted loop assumes that the control valves over shoot the programmed setpoint during the initial AFW startup. For a duration between 65.0 seconds and 209.9 seconds, depending on SG pressure, the AFW flowrates are higher because of the FCV response. After the defined duration, the flowrate to the faulted loop returns to 294 gpm. Operator action is credited to isolate AFW flow after 600 seconds. C28 C28

14.3.4.2.2.5 STEAM GENERATOR FLUID MASS

Maximum initial SG masses in the SGs were used in the analyzed cases. The use of high initial SG masses maximizes the SG inventory available for release to containment. The initial masses were calculated as the mass corresponding to the programmed level +6% narrow range span.

All SG fluid masses are calculated corresponding to 0% tube plugging which is conservative with respect to the RCS cooldown through the faulted loop SG resulting from the steamline break.

14.3.4.2.2.6 STEAM GENERATOR REVERSE HEAT TRANSFER

Once the steamline isolation is complete, those SGs in the intact steam loops become sources of energy which can be transferred to the SG with the broken line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the SG tubes drops below the temperature of the secondary fluid in the intact SG resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the SG with the broken steamline. The effects of reverse SG heat transfer are included in the results.

14.3.4.2.2.7 BREAK FLOW MODEL

Piping discharge resistances were not included in the calculation of the releases resulting from the steamline ruptures (Moody Curve for an f(L/D) = 0 was used). Saturated dry steam is assumed to exit the break. C26

14.3.4.2.2.8 CORE DECAY HEAT

Core decay heat generation assumed is based on the 1979 ANS Decay Heat + 2σ model (Reference 5).

14.3.4.2.2.9 STEAMLINE VOLUME BLOWDOWN

The contribution to the mass and energy releases from the secondary plant steam piping was included in the mass and energy release calculations. The flowrate was determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. For the full spectrum of steamline break cases analyzed at EPU conditions, the unisolable steamline mass is included in the mass exiting the break from the time of steamline isolation until the unisolable mass is completely released to containment. C26

14.3.4.2.2.10 MAIN STEAMLINE ISOLATION

The postulated single failure for the limiting case is the failure to close the Main Steam Check Valve (MSCV) in the faulted loop. In this instance, MSIV closure in the intact loops is required to terminate the blowdown. A delay time of 7 seconds was assumed (2-second signal processing plus 5-second valve closure) with full steam flow assumed through the valve during the valve stroke. The assumption of full steam flow for this time conservatively accounts for the effects of the unisolable steamline volume which would be released following closure of the MSIVs. For single failures other than the MSCV, the check valve is credited to prevent reverse blowdown from the intact SGs. C26 C26

14.3.4.2.2.11 REACTOR COOLANT SYSTEM METAL HEAT CAPACITY

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps. As this occurs, the heat stored in the metal is available to be transferred to the SG with the broken line. Stored metal heat does not have a major impact on the calculated mass and energy releases. The effects of this RCS metal heat are included in the results using conservative thick metal masses and heat transfer coefficients.

14.3.4.2.2.12 ROD CONTROL

The rod control system was assumed to be in manual operation for the steamline break analyses.

14.3.4.2.2.13 PROTECTION SYSTEM ACTUATIONS

The protection systems available to mitigate the effects of a MSLB accident inside containment include reactor trip, safety injection, steamline isolation, feedwater isolation, emergency fan coolers, and containment spray. The first protection system signal actuated for the limiting case was High Containment Pressure (2-of-3 signals) which initiated safety injection; the safety injection signal produced a reactor trip signal. Feedwater isolation occurred as a result of the safety injection signal. Finally, steamline isolation occurred via a High-High containment pressure signal. C26 C26

14.3.4.2.2.14 SAFETY INJECTION SYSTEM

Minimum safety injection (SI) system flowrates corresponding to the failure of one SIS train (2-of-4 pumps) were assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow was assumed to be 23 seconds for this analysis. C26

14.3.4.2.2.15 CORE REACTIVITY COEFFICIENTS

Conservative core reactivity coefficients corresponding to end-of-cycle conditions, including HZP stuck-rod moderator density coefficients, were used to maximize the reactivity feedback effects resulting from the steamline break. Use of maximum reactivity feedback results in higher power generation if the reactor returns critical, thus maximizing heat transfer to the secondary side of the SGs.

14.3.4.2.3 DESCRIPTION OF ANALYSIS

The break flows and enthalpies of the steam release through the steamline break are analyzed with the RETRAN (Reference 7) computer code. Blowdown mass and energy releases determined using RETRAN include the effects of core power generation, main and auxiliary feedwater additions, engineered safeguards systems, reactor coolant system thick metal heat storage, and reverse steam generator heat transfer. The cases considered in the analysis all assumed the availability of offsite power. Specifically, this means no credit was taken for tripping the reactor coolant pumps in the steamline break mass and energy release calculation which would significantly reduce the blowdown rate.

The Turkey Point NSSS at EPU conditions is analyzed using RETRAN to determine the transient steam mass and energy releases inside containment following a steamline break event. The tables of mass and energy releases are used as input conditions to the analysis of the containment response as discussed in Section 14.3.4.3.

The single most limiting case analyzed with respect to peak containment pressure, is a 1.19 ft^2 split break at hot-zero-power (HZP) conditions. C28

The split steamline break event was modeled taking credit only for MSIV closure on the intact loops for steamline isolation.

14.3.4.2.4 RESULTS

The mass and energy release rates at the EPU conditions were developed to determine the containment pressure response for the limiting steamline break case noted in Section 14.3.4.2.3. The mass and energy releases from the 1.19 ft^2 split break at HZP conditions with a single failure of a MSCV resulted in the highest containment pressure. The steam mass and energy releases discussed in this section provide the basis for the containment response described in Section 14.3.4.3 of this report. Table 14.3.4.3-6 provides the sequence of events for the limiting steamline break inside containment.

14.3.4.2.5 CONCLUSIONS

The mass and energy releases from the steamline break case, resulting in the limiting containment pressure response, have been analyzed at the EPU power conditions. The assumptions delineated in Section 14.3.4.2.2 have been included in the steamline break analysis such that the applicable acceptance criteria are met. The steam mass and energy releases discussed in this section provide the basis for the containment response described in Section 14.3.4.3.

14.3.4.3 CONTAINMENT RESPONSE

14.3.4.3.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a high-energy line break inside containment. The impact of MSLB or LOCA mass and energy releases on the containment pressure is addressed to assure that the containment pressure remains below its design pressure at the uprated 2652 MWt core power conditions including uncertainties.

14.3.4.3.2 INPUT PARAMETERS AND ASSUMPTIONS

An analysis of containment response to the rupture of the RCS or main steamline must start with knowledge of the initial conditions in the containment. The pressure, temperature, and humidity of the containment atmosphere prior to the postulated accident are specified in the analysis.

Also, values for the initial temperature of the component cooling water (CCW) and temperature of the intake cooling water (ICW) and refueling water storage tank (RWST) solution are assumed, along with the initial water inventory of the RWST. All of these values are chosen conservatively, as shown in Table 14.3.4.3-1.

The major modeling input parameters and assumptions that are used in the Turkey Point containment evaluation model for the LOCA and steamline break events are identified in this section. The assumed initial conditions and input assumptions associated with the containment volume, containment fan coolers, and containment sprays are listed in Tables 14.3.4.3-1, 14.3.4.3-4, and 14.3.4.3-5. The containment structural heat sink input is provided in Table 14.3.4.3-2a for LOCA and Table 14.3.4.3-2b for steamline break. The corresponding material properties are listed in Table 14.3.3-3. C28

The following are the major assumptions made in the analysis:

- The mass and energy released to the containment are described in Section 14.3.4.1 for LOCA and Section 14.3.4.2 for steamline break.

- Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.

- Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

- For the blowdown portion of the analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the LOCA analysis, steam and water releases are input separately.

- The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

Noding Structure

The Turkey Point GOTHIC containment evaluation model for the LOCA and steam line break events consist of one lumped-parameter control volume representing containment. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release and sump recirculation. Injection of accumulator nitrogen during a LOCA event is modeled as a boundary condition. The recirculation system model uses GOTHIC component models for the residual heat removal and component cooling water heat exchangers and pumps. Recirculation flow from the sump is modeled as a boundary condition.

Volume Input

GOTHIC requires the volume, height, diameter, and elevation input values for each node. The containment is modeled as a single control volume in the containment model. The minimum free volume of 1,460,000 ft^3 was used.

Flow Paths

Flow boundary conditions linked to functions that define the releases model the break flow to the containment. The boundary conditions are connected to the containment control volume via flow paths. The injection spray is modeled as a boundary condition connected to the containment control volume via a flow path.

The flow rates through the flow paths are specified by boundary conditions, so the purpose of the flow path is to direct the flow to the proper control volume. Standard values are used for the area, hydraulic diameter, friction length, and inertia length of the flow path.

Heat Sinks

The heat sinks in the containment are modeled as GOTHIC thermal conductors. The heat sink data is based on conservatively low surface areas and is summarized in Table 14.3.4.3-2a (LOCA) and Table 14.3.4.3-2b (steamline break). A thin air gap is assumed to exist between the steel and concrete for steel-jacketed heat sinks. A gap conductivity of 0.0174 Btu/hr-ft-°F is assumed between steel and concrete. The thermophysical properties for the heat sink materials are summarized in Table 14.3.4.3-3. C28

Heat and Mass Transfer Correlations

GOTHIC has several heat transfer coefficient options that can be used for containment analyses. For the Turkey Point GOTHIC model, the direct heat transfer coefficient set is used with the diffusion layer model mass transfer correlation for the heat sinks inside containment. This correlation does not require the user to specify a revaporization input value. C28

The direct heat transfer coefficient set is used for the heat sinks representing floors, ceilings, and walls. The submerged conductors are essentially insulated from the vapor after the pool develops. Insulated surfaces are modeled with no heat loss (0.0 Btu/hr-ft^2/°F).

Containment Fan Coolers

The Containment Fan Coolers are modeled in GOTHIC as a cooler/heater component in the containment volume. They are initiated on a "Hi" containment pressure signal. The heat removal rate for one fan cooler is defined by a function in GOTHIC. Multipliers are used to define the amount of operational fan coolers. See Table 14.3.4.3-1 and Table 14.3.4.3-5 for the fan cooler parameters and heat removal capability assumed for the containment response analyses.

Sump Recirculation

The residual heat removal heat exchanger cools the water from the containment sump. The residual heat removal system injects the cooled water into the reactor coolant system to cool the core. The residual heat removal heat exchanger is cooled with component cooling water and intake cooling water provides the ultimate heat sink, cooling the component cooling water heat exchangers. The GOTHIC heat exchanger model has been benchmarked against vendor specification sheets to ensure it adequately predicts active heat removal.

Mass and Energy Release

The LOCA and steamline break mass and energy release methodologies generate releases from both sides of the break, and are, therefore, input to the GOTHIC containment model via two flow boundary conditions. The LOCA mass and energy releases are documented in Section 14.3.4.1. The steamline break mass and energy releases are documented in Section 14.3.4.2. The break mass and enthalpy are linked to the boundary conditions as external functions defined by control variables. During blowdown, the liquid portion of the break flow is released as drops with an assumed diameter of 100 microns (0.00394 inches).

The LOCA mass and energy releases (see Section 14.3.4.1), from the boundary conditions are analyzed for Turkey Point Units 3 and 4 to the end of the injection period for the double-ended pump suction break minimum safeguard case (the double-ended hot leg mass and energy releases are only analyzed to the end of blowdown). At the start of recirculation, the LOCA mass and energy release to the containment is assumed to be from steam generated by decay heat. The long-term mass and energy release calculations are performed through user defined functions by GOTHIC. These input functions are used to incorporate the sump water cooling in the long term and are consistent with the Westinghouse methodology previously approved by the NRC (Reference 1). A flow boundary condition is defined to provide the long-term decay heat boil-off mass and energy release to containment. The mass flow rate and enthalpy of the flow is calculated using GOTHIC control variables.

The ANS Standard 5.1 (Reference 5) decay heat model ($+2\sigma$ uncertainty) is used to calculate the long-term boil-off from the core. All of the decay heat is assumed to produce steam from the recirculated emergency core cooling system water. The remainder of the water is returned to the sump region of the containment control volume. These assumptions are consistent with the long-term LOCA mass and energy methodology documented in Reference 1.

Containment Spray System

Containment spray is modeled with one boundary condition for the injection phase and two coupled boundary conditions for the recirculation phase (LOCA only since the steamline break transient is completed prior to the start of recirculation). Turkey Point has two trains of containment safeguards available with one spray pump per train. Containment spray is actuated on the “Hi-Hi” containment pressure setpoint. C28

The sprays begin injecting water from the refueling water storage tank after a delay for pump start-up and diesel start-up, if there is a loss of offsite power. The containment spray flow varies according to containment pressure and can be found in Table 14.3.4.3-4. The spray flow rate is modeled in GOTHIC as a control variable. Other containment spray parameters are detailed in Table 14.3.4.3-1. The spray droplet size is 700 microns and is assumed to be homogeneous and well-mixed.

<u>Accumulator Nitrogen Gas Modeling</u> C26

The accumulator nitrogen gas release is modeled with a flow boundary condition in the LOCA containment model only. The nitrogen release rate was conservatively calculated by maximizing the mass available to be injected. The nitrogen gas release rate was used as input for the GOTHIC function, as a specified rate over a fixed time period. Nitrogen gas is released at a rate of 160.1 lbm/second; beginning at 55 seconds (average accumulator tank water volume empty time) and ending at 75 seconds.

14.3.4.3.3 DESCRIPTION OF ANALYSIS

The containment integrity analyses are performed to demonstrate the acceptability of the containment heat removal system to mitigate the consequences of a LOCA or steamline break inside containment.

Calculation of the containment response following a postulated LOCA or steamline break is analyzed by use of the GOTHIC computer code. The GOTHIC technical manual (Reference 14) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC qualifications report (Reference 15) provides a comparison of the solver results with both analytical solutions and experimental data. C26

14.3.4.3.3.1 LOCA CONTAINMENT INTEGRITY

The purpose of this analysis was to calculate the increase in containment building temperature and pressure associated with the LOCA mass and energy release. The mass and energy releases for the DEPS and DEHL break scenarios were used in this analysis to assess the containment response. The post-LOCA containment temperature and pressure response for these break scenarios was determined for an initial containment pressure of 1.4 psig. C26

The analysis cases are intended to confirm that the peak containment pressure following a postulated accident will not exceed the containment design pressure of 55 psig, assuming that the initial pressure is at the maximum value allowed by plant technical specifications.

The sequence of events for each of the limiting LOCA cases is shown in Tables 14.3.4.3-8 and 14.3.4.3-9.

14.3.4.3.3.2 MSLB CONTAINMENT INTEGRITY

The MSLB mass and energy releases that were assumed for the 1.19 ft^2 split break at Hot Zero Power (HZP), as discussed in Section 14.3.4.2, were used to analyze the containment response. The failure of a MSCV was the limiting single failure for MSLB containment integrity. Since the failure was postulated to occur in the secondary steam system safety equipment, all of the containment heat removal equipment was assumed to be operational. This case was analyzed to the time of steam generator dryout. The sequence of events for this case is shown in Table 14.3.4.3-6. C28

14.3.4.3.4 RESULTS

The results of the transient analysis of the containment at an initial pressure of 1.4 psig for the LOCA cases are shown in Figures 14.3.4.3-3 through 14.3.4.3-6. The containment response to the DEHL blowdown is presented in Figures 14.3.4.3-3 and 14.3.4.3-4. The results of the long term DEPS transient with only one ECC operating initially and a second ECC manually actuated at 60 seconds are presented in Figures 14.3.4.3-5 and Figure 14.3.4.3-6. The containment pressure transient for the 1.19 ft^2 Split break SLB at 0% power with a MSCV failure is shown in Figure 14.3.4.3-7. All of these cases show that the containment pressure will remain below the design pressure of 55 psig. C28

In each case, the containment pressure is shown to peak and begin to decrease due to heat absorption by the containment internal structures, prior to active heat removal via operation of the safeguards equipment. After the peak pressure is attained, the operation of the safeguards system reduced the containment pressure. For the LOCA, at 24 hours following the accident, the containment pressure has been reduced to a value well below 50 percent of the peak calculated value. The containment integrity results are shown in Table 14.3.4.3-10 for LOCA and the MSLB ruptures.

14.3.4.3.5 CONCLUSIONS

The containment integrity analyses have been performed for the Extended Power Uprate program at Turkey Point Units 3 & 4. The analyses included both long-term MSLB and LOCA transients. As described in the results Section 14.3.4.3.4, all cases resulted in a peak containment pressure that was less than 55 psig. In addition, all long-term cases were well below 50% of the peak value within 24 hours. Based on these results, all applicable acceptance criteria have been met and Turkey Point Units 3 & 4 are safe to operate at 2652 MWt (core).

14.3.4.4 CONTAINMENT COMPARTMENTS

The compartments within the containment which enclose or surround the various portions of the reactor coolant system consists of a reactor cavity and three steam generator enclosures.

The compartments pressure buildup following LOCA is calculated by the use of Bechtel proprietary computer program COPRA. This program calculates the mass and energy balance of the two-phase mixture as it discharges into the compartment and leaves through openings into the main containment atmosphere. This calculation does not account for heat sinks or engineered safeguards system as their influence is negligible for such short time transient. In all blowdown cases, the largest possible reactor coolant pipe rupture that could occur within the compartments was assumed. The reactor cavity free volume was taken as 9350 ft^3 and the main containment $1.55x10^6$ ft^3. The initial containment condition was assumed to be 120°F(1) and 14.7 psia.

The reactor cavity has four different types of openings for pressure relief and flow expansion into the main containment atmosphere. However, the cavity blowdown is conservatively assumed to be able to vent to the main containment only through three, these are: (1) The annular clearances around the reactor coolant pipe penetrations (2) The annular space between concrete surface and the reactor vessel flange, and (3) The pipe chase connected with the reactor cavity.

It is assumed that the plugs for nozzle weld inspection remain in place and do not provide additional vent area.

NOTE:

1. Refer to Reference 12 and FSAR Section 14.0 for discussion of effects of operation with elevated normal bulk containment temperatures up to 125°F for short periods of time.

Both double-ended and slot type of reactor coolant pipe ruptures have been postulated. For the double-ended break at the reactor nozzle, the lateral separation of the ruptured pipe end is restricted by the size of the pipe sleeve through the reactor shield walls. The re-straining effect allows an opening area from the primary system of about 0.5 ft^2 in each direction. On the other hand, slot type failure having a maximum failure length of two times the inside pipe diameter gives an opening area of about 4.75 $ft.^2$ (equivalent to the cross-sectional flow area of the pipe.). This break produces the higher differential pressure across the cavity wall. The coolant released into the annulus splits into two paths, one leading into the reactor cavity, and the other leading into the secondary compartment.

The steam generator compartments are vented through the baffle wall geometry of the secondary shield walls.

Per Reference 13, Leak-Before-Break (LBB) Technology can be applied to the calculations of the short term mass and energy releases. Under LBB, the most limiting break would be a double-ended rupture of one of the largest RCS loop branch lines (i.e., the pressurizer surge line, the accumulator/SI line, or the RHR suction line). The mass and energy released from these breaks are bounded by the current design basis ruptures discussed above.

14.3.4.5 REFERENCES

1. "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Nonproprietary).

2. Not Used

3. EPRI 294-2, “Mixing of Emergency Core Cooling Water with Steam; 1/3 Scale Test and Summary,” (WCAP-8423), Final Report June 1975.

4. "Westinghouse Mass and Energy Release Data For Containment Design," WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A (Nonproprietary).

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

6. Deleted C28

7. WCAP-14882-P-A, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” April 1999.

8. Not Used

9. Not Used

10. Not Used

11. Not Used

12. Safety Evaluation JPN-PTN-SENJ-88-052,"Safey Evaluation for Containment Bulk Amendment Temperatures," Revision 3, dated April 13, 1989.

13. Letter, G. E. Edison (NRC) to W. F. Conway (FPL), "NRC Generic Letter 84-04, Asymmetric LOCA Loads for Turkey Point Units 3 and 4", dated November 28, 1988.

REFERENCES (cont’d)

14. NAI 8907-06, Rev. 16, “GOTHIC Containment Analysis Package Technical Manual, Version 7.2a (QA)”, January 2006.

15. NAI 8907-09, Rev. 9, “GOTHIC Containment Analysis Package Qualification Report, Version 7.2a (QA)”, January 2006. C26

16. American National Standard, ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", August 6, 1973.

TABLE 14.3.4.1-1

SYSTEM PARAMETERS
INITIAL CONDITIONS

PARAMETERS	VALUE
Core Thermal Power (MWt), including uncertainty	2652
Reactor Coolant System Total Flowrate (lbm/sec)	27,306
Vessel Outlet Temperature (°F) (1)	621.1
Core Inlet Temperature (°F) (1)	553.3
Vessel Average Temperature (°F) (1)	587.2
Initial Steam Generator Steam Pressure (psia)	853.0
Steam Generator Design	Model 44F
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm) (4)	97,500
Assumed Maximum Containment Backpressure (psia)	69.7
Accumulator	
Water Volume (ft^3) (2)	920.7
N_2 Cover Gas Pressure (psia) (3)	714.7
Temperature (°F)	130.0
Safety Injection Delay (sec)	35.0

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NOTE:

1. Analysis value includes an additional +5.7°F allowance for instrument error and dead-band.
2. Includes accumulator line volume
3. N_2 cover gas pressure includes uncertainty of +25 psi.
4. The steam generator mass is taken to be 110% of the mass at the nominal steam generator level. No level uncertainty is considered.

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TABLE 14.3.4.1-2

SAFETY INJECTION FLOW
DIESEL FAILURE (SINGLE TRAIN)

INJECTION MODE (REFLOOD PHASE)

RCS Pressure (psig)	Total Flow (gpm)
0	3971
20	3714
40	3433
60	3124
80	2760
100	2277
120	1497
140	886
200	862

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INJECTION MODE (POST-REFLOOD PHASE)

RCS Pressure (psig)	Total Flow (gpm)
55	918

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COLD LEG RECIRCULATION MODE

RCS Pressure (psig)	Total Flow (gpm)
0	970

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TABLE 14.3.4.1-3

DOUBLE-ENDED HOT LEG BREAK BLOWDOWN
MASS AND ENERGY RELEASES
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Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
0.000	0.0	0.0	0.0	0.0
0.001	43532.1	27945.9	43531.1	27944.5
0.002	46215.1	29666.7	45950.5	29491.4
0.102	40394.2	26269.7	26769.4	17144.1
0.201	35142.9	22908.8	22982.1	14620.9
0.302	34442.0	22388.9	20508.3	12854.4
0.402	33494.4	21750.2	19289.6	11876.5
0.502	33139.7	21513.1	18510.5	11201.5
0.602	33080.1	21488.1	17967.8	10709.9
0.702	32636.2	21247.0	17526.5	10311.1
0.801	32091.1	20962.2	17245.8	10031.6
0.902	31743.1	20840.9	16945.8	9760.6
1.002	31375.2	20731.8	16738.8	9558.5
1.102	30832.3	20504.1	16648.7	9434.2
1.202	30320.0	20294.7	16630.8	9359.3
1,302	29789.5	20077.1	16676.4	9326.6
1.402	29161.0	19796.5	16765.1	9323.6
1.501	28425.1	19440.1	16879.7	9340.4
1.601	27653.2	19053.0	17003.5	9367.5
1.701	26897.4	18671.3	17129.6	9400.9
1.802	26148.5	18285.5	17249.3	9435.6
1.902	25393.7	17879.4	17354.9	9467.3
2.002	24633.0	17451.0	17444.3	9494.0
2.101	23890.4	17013.0	17514.1	9513.7
2.202	23198.8	16596.5	17566.3	9526.4
2.302	22567.1	16207.0	17599.5	9531.3
2.401	21943.1	15801.3	17615.2	9528.5
2.501	21373.5	15418.8	17618.2	9520.7
2.602	20841.8	15045.2	17607.5	9507.0
2.702	20382.4	14709.8	17584.6	9488.3
2.801	19974.7	14400.5	17549.7	9464.4
2.902	19605.6	14106.0	17501.6	9434.6
3.001	19292.9	13845.8	17442.6	9400.0
3.102	19009.3	13598.2	17372.1	9360.2
3.202	18765.1	13375.3	17290.9	9315.6
3.301	18549.7	13171.0	17200.0	9266.7

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TABLE 14.3.4.1-3

DOUBLE-ENDED HOT LEG BREAK BLOWDOWN
MASS AND ENERGY RELEASES
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Time Seconds	Break Path No. 1[(1)]		Break Path No. 2[(2)]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
3.402	18358.1	12982.9	17098.7	9213.2
3.501	18193.9	12816.6	16990.2	9156.6
3.602	18063.0	12675.7	16871.8	9095.7
3.702	17989.2	12581.6	16746.5	9031.9
3.802	17956.3	12520.6	16611.7	8963.9
3.901	17952.7	12475.0	16471.8	8894.0
4.002	17984.8	12440.6	16322.6	8820.2
4.201	18178.8	12426.7	15989.2	8655.7
4.402	18453.3	12435.9	15579.7	8452.4
4.600	18723.4	12448.2	15149.8	8240.1
4.801	18980.7	12473.9	14738.9	8040.4
5.003	19236.5	12512.5	14323.4	7837.9
5.201	19634.2	12626.8	13838.2	7594.3
5.400	14850.4	10453.1	13304.8	7322.0
5.601	14715.4	10422.7	12786.2	7056.2
5.801	14452.6	10260.5	12269.8	6789.5
6.000	14320.1	10142.8	11767.9	6528.6
6.201	14275.7	10094.6	11287.7	6277.6
6.400	14366.9	10050.9	10825.3	6034.2
6.601	14516.8	10060.5	10376.3	5796.8
6.802	14315.7	9862.4	9953.0	5572.5
7.001	14506.9	9867.0	9566.2	5367.3
7.201	14641.3	9850.9	9199.3	5172.6
7.401	14732.7	9817.3	8857.0	4991.1
7.601	14778.3	9764.5	8529.0	4817.1
7.801	14788.6	9693.3	8212.4	4649.4
8.003	14723.7	9585.9	7906.6	4488.0
8.202	14549.0	9427.4	7612.2	4333.1
8.402	14249.5	9210.3	7323.3	4181.8
8.601	13878.9	8964.5	7043.2	4036.1
8.801	13510.5	8728.5	6770.6	3895.3
9.002	13162.0	8509.7	6508.3	3760.9
9.202	12813.6	8294.9	6256.8	3633.0
9.402	12430.0	8062.5	6016.5	3511.8
9.601	11997.7	7805.6	5781.9	3394.0
9.800	11527.5	7534.2	5544.7	3275.1
10.001	11032.3	7256.1	5292.2	3149.0

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TABLE 14.3.4.1-3

DOUBLE-ENDED HOT LEG BREAK BLOWDOWN
MASS AND ENERGY RELEASES
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Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
10.200	10543.7	6992.2	5029.6	3019.9
10.402	10056.2	6738.5	4755.3	2888.0
10.404	10051.5	6736.0	4752.6	2886.7
10.406	10047.4	6734.2	4750.3	2885.8
10.602	9585.4	6500.1	4481.5	2759.2
10.802	9131.7	6273.3	4215.4	2635.5
11.002	8698.5	6057.2	3962.6	2517.3
11.200	8287.9	5850.8	3732.8	2407.2
11.401	7891.8	5651.3	3526.4	2303.8
11.600	7496.2	5449.8	3351.7	2211.4
11.800	7077.3	5236.3	3206.0	2129.5
12.000	6620.5	5010.2	3086.3	2058.0
12.200	6110.3	4769.8	2995.3	1999.7
12.400	5581.6	4562.1	2920.3	1945.6
12.600	5071.1	4346.4	2843.3	1889.4
12.800	4582.9	4129.2	2762.6	1837.3
13.001	4107.4	3920.5	2675.1	1788.3
13.200	3636.1	3705.3	2582.6	1740.6
13.401	3158.5	3414.7	2487.5	1692.9
13.601	2892.7	3205.2	2395.2	1647.3
13.800	2790.2	3035.0	2302.4	1602.4
14.001	2713.5	2884.9	2211.4	1560.8
14.200	2246.9	2485.7	2120.8	1522.0
14.400	1824.2	2104.3	2029.5	1484.7
14.600	1538.4	1822.0	1938.7	1448.3
14.800	1348.5	1637.3	1849.5	1412.5
15.001	1216.5	1497.7	1756.8	1374.5
15.201	1153.4	1433.1	1638.5	1328.4
15.400	1122.5	1409.6	1507.3	1312.4
15.600	1076.8	1355.5	1333.8	1314.7
15.801	1063.8	1343.9	1116.5	1256.8
16.001	1048.7	1320.3	1020.9	1179.4
16.201	1004.7	1267.9	966.6	1135.0
16.400	901.1	1141.6	851.9	1033.1
16.600	864.3	1098.8	718.1	881.7
16.800	818.0	1041.9	608.9	750.7
17.000	777.4	991.7	527.1	651.1

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TABLE 14.3.4.1-3

DOUBLE-ENDED HOT LEG BREAK BLOWDOWN
MASS AND ENERGY RELEASES
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Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
17.201	743.4	948.9	445.4	551.4
17.401	711.3	908.2	385.3	478.2
17.600	666.6	852.7	357.6	444.6
17.800	736.4	911.2	352.3	438.5
18.001	748.1	925.8	348.6	434.1
18.201	793.8	966.2	363.2	452.6
18.400	755.4	937.0	370.8	462.1
18.601	757.0	934.6	372.3	464.1
18.801	764.8	936.6	367.4	458.2
19.000	720.2	896.8	367.4	458.5
19.201	765.6	935.4	358.6	447.6
19.401	760.6	924.0	350.6	437.9
19.600	740.8	908.8	361.8	452.1
19.800	795.7	914.3	362.6	453.3
20.000	867.7	878.3	369.4	461.9
20.200	822.7	879.3	363.1	454.1
20.400	766.0	831.5	345.8	432.7
20.600	510.2	643.6	328.7	411.5
20.800	407.0	515.6	171.9	215.4
21.000	368.7	459.5	0.0	0.0
21.200	428.8	493.1	177.2	224.5
21.400	590.8	400.2	232.9	293.9
21.600	877.2	478.6	237.7	298.9
21.800	911.7	509.3	211.9	266.7
22.001	517.0	409.0	173.3	218.3
22.201	332.5	352.7	145.8	184.3
22.401	235.3	290.2	135.7	172.0
22.601	186.2	237.7	127.2	161.4
22.801	175.0	222.9	133.6	169.6
23.001	143.7	183.6	109.1	138.6
23.200	332.1	408.3	75.9	96.8
23.400	421.5	480.5	99.2	126.4
23.601	452.2	297.7	177.1	224.7
23.801	603.9	333.9	74.0	93.8
24.001	810.9	430.7	85.7	109.7
24.201	862.1	505.0	115.2	147.2
24.400	796.6	549.6	180.1	228.5

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TABLE 14.3.4.1-3

DOUBLE-ENDED HOT LEG BREAK BLOWDOWN
MASS AND ENERGY RELEASES

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
24.601	771.2	571.8	226.3	285.9
24.801	710.5	539.0	202.2	255.4
25.000	604.2	503.5	156.2	197.6
25.200	368.2	374.5	79.6	101.3
25.400	219.8	254.9	82.7	106.2
25.600	0.0	0.0	0.0	0.0

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Notes:

Path 1: M&E exiting from the reactor vessel side of the break.

Path 2: M&E exiting from the steam generator side of the break.

TABLE 14.3.4.1-4

DOUBLE-ENDED HOT LEG BREAK
MASS AND ENERGY BALANCE

Time (seconds)		0.00	25.60	25.60
Mass (thousand lbm)				
Initial Mass	RCS and Accumulator	563.15	563.15	563.15
Added Mass	Pumped Injection	0.00	0.00	0.00
	Total Added	0.00	0.00	0.00
****** Total Available ******		***563.15***	***563.15***	***563.15***
Distribution	Reactor Coolant	392.60	85.73	118.82
	Accumulator	170.56	97.79	64.70
	Total Contents	563.15	183.53	183.53
Effluent	Break Flow	0.00	379.61	379.61
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	379.61	379.61
****** Total Accountable ******		***563.15***	***563.13***	***563.13***
Energy (million Btu)				
Initial Energy	RCS, Accumulator, SG	604.39	604.39	604.39
Added Energy	Pumped Injection	0.00	0.00	0.00
	Decay Heat	0.00	5.95	5.95
	Heat from Secondary	0.00	3.15	3.15
	Total Added	0.00	9.10	9.10
****** Total Available ******		***604.39***	***613.49***	***613.49***
Distribution	Reactor Coolant	232.20	21.01	24.18
	Accumulator	16.63	9.53	6.37
	Core Stored	22.86	7.59	7.59
	Primary Metal	128.78	119.19	119.19
	Secondary Metal	38.31	36.84	36.84
	SG	165.62	169.32	169.32
	Total Contents	604.39	363.49	363.49
Effluent	Break Flow	0.00	249.51	249.51
	ECCS Spill	0.00	0.00	0.00
	Total Effluent	0.00	249.51	249.51
****** Total Accountable ******		***604.39***	***613.00***	***613.00***

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TABLE 14.3.4.1-5

DOUBLE-ENDED HOT LEG BREAK
MASS BALANCE

DELETED

TABLE 14.3.4.1-6

DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASE
PAGE 1 OF 5

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
0.000	0.0	0.0	0.0	0.0
0.001	82893.5	45363.9	40200.6	21957.5
0.002	41410.8	22619.3	41056.1	22423.4
0.1	40943.0	22450.9	20320.5	11086.0
0.2	41741.4	23079.1	23081.4	12606.7
0.3	42703.6	23869.4	24455.5	13365.7
0.4	43677.4	24724.9	24343.8	13310.7
0.5	44333.7	25436.2	23678.1	12953.4
0.6	44336.0	25760.6	23108.0	12649.1
0.7	43367.0	25482.1	22836.6	12508.1
0.8	41959.8	24907.3	22741.5	12461.7
0.9	40691.5	24383.3	22636.0	12408.1
1.0	39601.9	23948.7	22495.2	12333.8
1.1	38585.4	23545.2	22329.9	12245.1
1.2	37590.6	23144.8	22156.3	12151.1
1.3	36676.9	22772.2	21983.5	12056.8
1.4	35852.7	22437.2	21809.8	11961.5
1.5	35093.5	22137.4	21645.7	11871.1
1.6	34329.3	21847.6	21498.6	11790.1
1.7	33519.2	21549.1	21362.2	11714.8
1.8	32440.3	21098.5	21211.7	11631.6
1.9	31207.5	20551.2	21033.0	11532.4
2.0	29658.3	19779.7	20650.6	11320.5
2.1	27953.3	18865.8	20153.8	11045.9
2.2	25863.1	17624.3	19710.6	10801.5
2.3	23653.7	16210.1	19271.2	10559.4

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TABLE 14.3.4.1-6
DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASE

Time Seconds	Break Path No. 1[(1)]		Break Path No. 2[(2)]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
2.4	21388.8	14689.0	18789.4	10294.5
2.5	19977.1	13741.4	18384.3	10072.2
2.6	19829.0	13660.2	18014.5	9869.9
2.7	18940.4	13059.6	17675.8	9685.0
2.8	17989.2	12443.5	17396.0	9532.8
2.9	16930.9	11759.2	17005.1	9319.0
3.0	15752.8	10990.1	16642.3	9121.2
3.1	14731.3	10327.1	16359.6	8968.1
3.2	13905.6	9790.1	16103.1	8829.5
3.3	13273.9	9379.8	15869.6	8703.5
3.4	12792.7	9069.1	15635.8	8577.1
3.5	12397.6	8813.6	15410.8	8455.6
3.6	12073.4	8606.9	15212.6	8348.9
3.7	11800.0	8436.8	15044.8	8259.0
3.8	11538.1	8275.3	14888.2	8175.1
3.9	11269.6	8110.1	14735.6	8093.2
4.0	11006.0	7950.7	14590.6	8015.4
4.2	10530.4	7672.7	14305.9	7862.6
4.4	10082.0	7405.1	14254.2	7840.2
4.6	9741.4	7200.6	14877.5	8188.1
4.8	9551.8	7075.5	15040.0	8280.6
5.0	9456.1	6996.9	15165.4	8353.2
5.2	9355.9	6927.6	15009.7	8269.0
5.4	9233.8	6852.4	14844.4	8179.9
5.6	9328.0	6930.7	14725.9	8116.3
5.8	9741.9	7363.8	14641.4	8070.8
6.0	8637.1	7342.1	14443.3	7960.3

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TABLE 14.3.4.1-6
DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASE

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
6.2	7508.8	6841.2	14190.6	7819.0
6.4	7404.0	6587.1	13944.6	7681.9
6.6	7563.0	6445.0	13731.3	7563.3
6.8	7777.2	6364.6	13539.1	7455.3
7.0	7948.8	6300.3	13333.8	7338.7
7.2	8039.9	6265.0	13127.8	7221.4
7.4	7991.6	6204.2	12919.3	7103.1
7.6	7853.8	6118.4	12697.2	6978.1
7.8	7677.3	6023.6	12475.7	6854.6
8.0	7480.8	5923.4	12253.8	6731.4
8.2	7275.9	5818.5	12031.8	6608.5
8.4	7071.6	5709.1	11806.3	6483.7
8.6	6880.4	5600.4	11583.3	6360.6
8.8	6699.2	5489.7	11356.9	6235.6
9.0	6528.0	5377.1	11137.4	6114.7
9.2	6374.1	5275.8	10927.5	5999.3
9.4	6218.6	5165.7	10703.5	5876.4
9.6	6077.1	5057.6	10498.2	5764.1
9.8	5937.8	4947.5	10290.2	5650.4
10.0	5786.1	4826.6	10063.5	5526.1
10.2	5627.8	4694.1	9838.3	5402.9
10.4	5479.6	4555.0	9658.3	5296.1
10.4	5478.7	4554.0	9657.3	5295.4
10.4	5477.7	4553.1	9656.3	5294.7
10.6	5333.1	4404.6	9460.1	5155.4
10.8	5171.2	4238.1	9290.6	5005.7
11.0	4994.9	4056.1	9160.7	4858.2

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TABLE 14.3.4.1-6
DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASE
PAGE 4 OF 5

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
11.2	4837.0	3883.3	9103.0	4740.5
11.4	4732.0	3743.6	9143.8	4681.0
11.6	4652.9	3629.3	9051.4	4568.6
11.8	4582.2	3535.7	8971.8	4473.9
12.0	4506.5	3455.5	8876.5	4379.7
12.2	4425.1	3384.4	8728.9	4267.2
12.4	4338.9	3320.1	8615.6	4176.9
12.6	4250.4	3262.5	8498.0	4089.7
12.8	4157.9	3209.4	8352.4	3989.6
13.0	4060.3	3160.0	8239.0	3903.7
13.2	3958.8	3115.7	8132.4	3821.0
13.4	3852.0	3075.5	7938.2	3698.4
13.6	3740.5	3042.2	7816.7	3610.0
13.8	3623.6	3012.2	7726.1	3536.8
14.0	3502.4	2985.6	7572.3	3436.9
14.2	3373.9	2965.0	7330.9	3299.9
14.4	3239.1	2950.5	6904.7	3082.1
14.6	3100.5	2944.7	6504.7	2877.8
14.8	2941.7	2934.6	6327.1	2771.7
15.0	2691.8	2859.4	6000.4	2600.3
15.2	2412.2	2756.2	5418.4	2319.1
15.4	2201.2	2645.1	5092.6	2142.5
15.6	1956.1	2397.1	5145.2	2110.0
15.8	1755.7	2165.6	5560.3	2210.6
16.0	1584.9	1962.0	6003.9	2320.9
16.2	1434.0	1779.9	5875.2	2226.8
16.4	1300.5	1617.7	5266.8	1971.0

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TABLE 14.3.4.1-6
DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASE

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
16.6	1186.0	1477.9	4659.5	1722.9
16.8	1064.8	1329.6	4213.7	1534.0
17.0	942.9	1179.1	3897.7	1391.6
17.2	843.4	1055.9	3588.8	1253.3
17.4	750.2	940.2	3263.7	1114.2
17.6	662.1	830.5	2960.7	988.5
17.8	581.5	730.0	2709.7	885.0
18.0	493.8	620.3	2438.3	779.6
18.2	409.1	514.3	2148.8	673.2
18.4	332.0	417.6	1833.9	563.8
18.6	259.5	326.8	1485.2	449.2
18.8	192.1	242.1	1101.0	328.6
19.0	128.0	161.6	682.1	201.8
19.2	80.0	101.2	260.9	77.0
19.4	30.7	39.0	0.0	0.0
19.6	0.0	0.0	0.0	0.0

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Note:

Path 1: M&E exiting from the steam generator side of the break.

Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

TABLE 14.3.4.1-7
DOUBLE-ENDED PUMP SUCTION BREAK
REFLOOD MASS AND ENERGY RELEASE
MINIMUM SI

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
19.6	0.0	0.0	0.0	0.0
20.2	0.0	0.0	0.0	0.0
20.3	0.0	0.0	0.0	0.0
20.5	0.0	0.0	0.0	0.0
20.6	0.0	0.0	0.0	0.0
20.7	0.0	0.0	0.0	0.0
20.8	26.2	30.9	0.0	0.0
20.9	25.4	30.0	0.0	0.0
21.0	12.0	14.2	0.0	0.0
21.1	7.5	8.8	0.0	0.0
21.2	10.8	12.8	0.0	0.0
21.3	19.3	22.7	0.0	0.0
21.4	24.3	28.7	0.0	0.0
21.5	28.9	34.1	0.0	0.0
21.6	34.2	40.3	0.0	0.0
21.7	38.1	44.9	0.0	0.0
21.8	41.5	48.9	0.0	0.0
21.9	44.6	52.6	0.0	0.0
22.0	47.5	56.1	0.0	0.0
22.1	51.1	60.4	0.0	0.0
22.2	53.0	62.6	0.0	0.0
22.4	56.3	66.5	0.0	0.0
22.5	59.6	70.3	0.0	0.0
22.6	61.9	73.0	0.0	0.0
22.7	64.8	76.6	0.0	0.0

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TABLE 14.3.4.1-7
DOUBLE-ENDED PUMP SUCTION BREAK
REFLOOD MASS AND ENERGY RELEASE
MINIMUM SI
PAGE 2 OF 6

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
23.7	85.1	100.5	0.0	0.0
24.7	101.8	120.2	0.0	0.0
25.7	116.1	137.1	0.0	0.0
26.7	128.8	152.1	0.0	0.0
27.4	137.4	162.3	0.0	0.0
27.7	140.1	165.5	0.0	0.0
28.7	150.7	178.1	0.0	0.0
29.7	220.6	260.9	1872.3	304.8
30.8	265.0	313.5	2532.2	423.3
31.8	264.9	313.4	2530.3	427.9
32.8	261.0	308.7	2481.1	422.9
33.3	258.9	306.3	2455.0	420.0
33.8	256.8	303.9	2428.9	417.2
34.8	252.9	299.1	2377.6	411.4
35.8	249.0	294.5	2327.4	405.7
36.8	245.3	290.1	2278.5	400.2
37.8	241.7	285.9	2230.8	394.7
38.8	238.3	281.8	2184.4	389.3
39.8	257.2	304.3	2484.5	406.2
39.9	256.9	303.9	2480.2	405.6
40.8	253.9	300.4	2442.1	401.0
41.8	250.7	296.6	2400.8	395.9
42.8	247.6	292.9	2360.5	390.9
43.8	244.7	289.4	2321.2	386.1
44.8	241.8	286.0	2282.8	381.3
45.8	239.0	282.6	2245.4	376.7
46.8	236.3	279.4	2208.8	372.1

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TABLE 14.3.4.1-7
DOUBLE-ENDED PUMP SUCTION BREAK
REFLOOD MASS AND ENERGY RELEASE
MINIMUM SI

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
47.1	235.5	278.5	2198.0	370.8
47.8	233.6	276.3	2173.1	367.7
48.8	231.1	273.3	2138.2	363.3
49.8	228.6	270.3	2104.0	359.0
50.8	226.2	267.5	2070.6	354.8
51.8	223.9	264.7	2037.8	350.7
52.8	221.6	262.0	2005.7	346.7
53.8	219.4	259.4	1974.3	342.7
54.8	217.2	256.8	1943.5	338.9
55.0	214.8	254.0	1890.4	334.2
55.9	300.3	355.5	272.3	167.4
56.9	299.0	354.0	271.6	166.6
57.9	294.8	348.9	269.7	164.0
58.9	290.6	343.9	267.7	161.5
59.9	286.4	339.0	265.8	158.9
60.9	282.3	334.1	263.9	156.4
61.9	278.1	329.1	262.1	153.9
62.9	274.0	324.2	260.2	151.4
63.9	270.2	319.7	258.5	149.1
64.9	266.4	315.2	256.8	146.9
65.9	262.6	310.7	255.1	144.6
66.9	259.0	306.4	253.4	142.4
67.9	255.3	302.0	251.8	140.3
68.9	251.7	297.7	250.2	138.1
69.9	248.1	293.5	248.6	136.1
70.1	247.4	292.7	248.3	135.6
70.9	244.7	289.4	247.0	134.0

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TABLE 14.3.4.1-7
DOUBLE-ENDED PUMP SUCTION BREAK
REFLOOD MASS AND ENERGY RELEASE
MINIMUM SI

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
71.9	241.2	285.3	245.5	132.0
72.9	237.8	281.3	244.0	130.0
73.9	234.5	277.3	242.5	128.1
74.9	231.2	273.4	241.1	126.2
75.9	227.9	269.5	239.7	124.3
76.9	224.7	265.7	238.3	122.5
77.9	221.5	261.9	236.9	120.7
78.9	218.4	258.2	235.5	118.9
79.9	215.3	254.6	234.2	117.2
80.9	212.3	251.0	232.9	115.5
82.9	206.4	244.0	230.4	112.2
84.9	200.7	237.2	227.9	109.0
86.9	195.2	230.7	225.6	106.0
88.2	191.7	226.6	224.1	104.1
88.9	189.9	224.5	223.3	103.2
90.9	184.9	218.5	221.2	100.4
92.9	180.0	212.7	219.2	97.8
94.9	175.4	207.2	217.2	95.4
96.9	171.0	202.0	215.4	93.0
98.9	166.8	197.0	213.7	90.8
100.9	162.8	192.3	212.0	88.8
102.9	159.0	187.8	210.5	86.8
104.9	155.4	183.6	209.0	85.0
106.9	152.0	179.6	207.7	83.3
108.9	148.8	175.8	206.4	81.7
110.3	146.7	173.3	205.6	80.6
110.9	145.8	172.3	205.2	80.2

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TABLE 14.3.4.1-7
DOUBLE-ENDED PUMP SUCTION BREAK
REFLOOD MASS AND ENERGY RELEASE
MINIMUM SI
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Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
112.9	143.0	169.0	204.1	78.8
114.9	140.4	165.9	203.1	77.5
116.9	138.0	163.0	202.1	76.3
118.9	135.7	160.3	201.3	75.1
120.9	133.6	157.8	200.4	74.1
122.9	131.6	155.5	199.7	73.2
124.9	129.8	153.4	199.0	72.3
126.9	128.2	151.4	198.4	71.5
128.9	126.6	149.6	197.8	70.7
130.9	125.2	147.9	197.2	70.1
132.9	124.0	146.4	196.7	69.4
134.9	122.8	145.0	196.3	68.9
136.7	121.8	143.9	195.9	68.4
136.9	121.7	143.8	195.9	68.4
138.9	120.8	142.6	195.5	67.9
140.9	119.9	141.6	195.2	67.5
142.9	119.1	140.7	194.9	67.1
144.9	118.5	139.9	194.6	66.8
146.9	117.9	139.2	194.4	66.5
148.9	117.4	138.6	194.2	66.2
150.9	116.9	138.1	194.0	66.0
152.9	116.5	137.6	193.9	65.8
154.9	116.2	137.2	193.7	65.7
156.9	115.9	136.9	193.6	65.5
158.9	115.7	136.6	193.5	65.4
160.9	115.5	136.4	193.4	65.3
162.9	115.3	136.2	193.3	65.2

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TABLE 14.3.4.1-7
DOUBLE-ENDED PUMP SUCTION BREAK
REFLOOD MASS AND ENERGY RELEASE
MINIMUM SI
PAGE 6 OF 6

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
164.9	115.2	136.0	193.3	65.1
165.9	115.1	135.9	193.3	65.1
166.9	115.1	135.9	193.2	65.0
168.9	115.0	135.8	193.2	65.0
170.9	115.0	135.8	193.2	65.0
172.9	115.0	135.8	193.2	64.9
174.9	115.0	135.8	193.1	64.9
176.9	115.0	135.8	193.1	64.9
178.9	115.0	135.9	193.1	64.9
180.9	115.1	135.9	193.2	64.9
182.9	115.2	136.0	193.2	64.9
184.9	115.3	136.2	193.2	65.0
186.9	115.4	136.3	193.2	65.0
188.9	115.5	136.4	193.2	65.0
190.9	115.7	136.6	193.3	65.1
192.9	115.8	136.8	193.3	65.1
194.9	116.0	137.0	193.4	65.2
196.8	116.1	137.2	193.4	65.2

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Notes:

Path 1: M&E exiting from the steam generator side of the break.
Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

TABLE 14.3.4.1-8

DOUBLE-ENDED PUMP SUCTION

MINIMUM SAFETY INJECTION

PRINCIPAL PARAMETERS DURING REFLOOD

PAGE 1 OF 2

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Down-Comer Height ft	Flow Fraction	Total lbm/sec	Injection Accumulator lbm/sec	SI Spill lbm/sec	Enthalpy Btu/lbm
19.6	176.2	0.000	0.000	0.00	0.00	0.333	0.000	0.0	0.0	0.00
20.5	174.6	20.304	0.000	0.74	0.95	0.000	4358.000	4358.0	0.0	99.79
20.7	174.0	19.252	0.000	1.08	0.96	0.000	4328.600	4328.6	0.0	99.79
21.1	173.9	1.640	0.092	1.31	1.50	0.240	4256.900	4256.9	0.0	99.79
21.5	174.1	2.572	0.161	1.38	2.33	0.335	4194.700	4194.7	0.0	99.79
22.4	174.4	2.293	0.291	1.51	3.81	0.401	4082.900	4082.9	0.0	99.79
23.7	174.9	2.231	0.431	1.66	6.16	0.422	3918.400	3918.4	0.0	99.79
27.4	176.6	2.482	0.601	2.00	12.36	0.440	3528.400	3528.4	0.0	99.79
30.8	178.4	3.353	0.670	2.29	15.61	0.563	3103.100	3103.1	0.0	99.79
32.8	179.6	3.233	0.688	2.46	15.62	0.559	2955.400	2955.4	0.0	99.79
33.3	179.9	3.202	0.691	2.50	15.62	0.557	2922.600	2922.6	0.0	99.79
38.8	183.5	2.959	0.711	2.92	15.62	0.540	2601.400	2601.4	0.0	99.79
39.8	184.2	3.049	0.714	3.00	15.62	0.563	2926.000	2500.4	0.0	95.17
39.9	184.3	3.045	0.714	3.01	15.62	0.563	2921.000	2495.3	0.0	95.16
47.1	189.3	2.837	0.722	3.50	15.62	0.548	2598.800	2169.2	0.0	94.53
55.0	194.9	2.657	0.725	4.00	15.62	0.529	2256.900	1824.2	0.0	93.70
55.9	195.6	3.263	0.733	4.06	15.54	0.601	415.900	0.0	0.0	68.00

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TABLE 14.3.4.1-8
DOUBLE-ENDED PUMP SUCTION
MINIMUM SAFETY INJECTION
PRINCIPAL PARAMETERS DURING REFLOOD

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Down-Comer Height ft	Flow Fraction	Total lbm/sec	Injection Accumulator lbm/sec	SI Spill lbm/sec	Enthalpy Btu/lbm
62.9	201.6	2.991	0.733	4.55	14.51	0.595	420.200	0.0	0.0	68.00
70.1	208.1	2.735	0.732	5.00	13.68	0.589	424.800	0.0	0.0	68.00
78.9	216.4	2.460	0.731	5.52	12.91	0.579	429.200	0.0	0.0	68.00
88.2	225.0	2.210	0.729	6.00	12.37	0.567	432.700	0.0	0.0	68.00
98.9	233.8	1.978	0.727	6.51	12.03	0.553	435.300	0.0	0.0	68.00
110.3	241.6	1.793	0.725	7.00	11.92	0.537	437.200	0.0	0.0	68.00
124.9	249.8	1.637	0.725	7.57	12.05	0.521	438.600	0.0	0.0	68.00
136.7	255.5	1.561	0.726	8.00	12.30	0.513	439.300	0.0	0.0	68.00
152.9	262.3	1.506	0.729	8.57	12.75	0.506	439.700	0.0	0.0	68.00
165.9	267.1	1.485	0.733	9.00	13.17	0.505	439.800	0.0	0.0	68.00
182.9	272.6	1.475	0.738	9.56	13.73	0.505	439.800	0.0	0.0	68.00
194.9	276.1	1.473	0.742	9.94	14.14	0.506	439.800	0.0	0.0	68.00
196.8	276.6	1.473	0.743	10.00	14.20	0.506	439.800	0.0	0.0	68.00

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TABLE 14.3.4.1-9

DOUBLE-ENDED PUMP SUCTION BREAK

POST–REFLOOD MASS AND ENERGY RELEASE

MINIMUM SI

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Time Seconds	Break Path No. 1[(1)]		Break Path No. 2[(2)]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
196.8	156.2	194.3	286.0	97.1
201.8	152.0	189.0	290.2	95.6
206.8	155.3	193.2	286.9	96.9
211.8	154.6	192.3	287.6	96.8
216.8	154.5	192.2	287.7	96.6
221.8	153.8	191.3	288.4	96.6
226.8	153.7	191.2	288.5	96.3
231.8	153.0	190.3	289.2	96.3
236.8	152.9	190.2	289.3	96.1
241.8	152.1	189.2	290.1	96.1
246.8	152.0	189.1	290.2	95.8
251.8	151.2	188.1	291.0	95.8
256.8	151.0	187.9	291.2	95.6
261.8	150.2	186.9	292.0	95.6
266.8	150.0	186.7	292.2	95.4
271.8	149.2	185.6	293.0	95.4
276.8	149.0	185.4	293.2	95.2
281.8	148.2	184.3	294.0	95.1
286.8	147.9	184.0	294.3	95.0
291.8	147.6	183.7	294.6	94.8
296.8	146.8	182.6	295.4	94.8
301.8	146.5	182.2	295.7	94.6
306.8	146.1	181.8	296.1	94.4
311.8	145.2	180.6	297.0	94.4

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TABLE 14.3.4.1-9

DOUBLE-ENDED PUMP SUCTION BREAK

POST–REFLOOD MASS AND ENERGY RELEASE

MINIMUM SI

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
316.8	144.9	180.2	297.4	94.3
321.8	144.5	179.7	297.7	94.1
326.8	144.0	179.2	298.2	94.0
331.8	143.1	178.0	299.2	94.0
336.8	146.4	182.1	295.8	95.1
341.8	145.8	181.4	296.4	95.0
346.8	145.2	180.6	297.0	94.9
351.8	144.6	179.9	297.6	94.8
356.8	144.4	179.7	297.8	94.5
361.8	143.7	178.8	298.5	94.5
366.8	143.0	177.8	299.2	94.4
371.8	142.7	177.5	299.5	94.2
376.8	141.8	176.4	300.4	94.1
381.8	141.4	175.9	300.8	94.0
386.8	140.9	175.3	301.3	93.8
391.8	140.4	174.7	301.8	93.7
396.8	139.8	173.9	302.4	93.6
401.8	139.1	173.1	303.1	93.5
406.8	138.4	172.2	303.8	93.4
411.8	138.1	171.8	304.1	93.2
416.8	137.2	170.7	305.0	93.2
421.8	136.6	170.0	305.6	93.0
426.8	139.9	174.0	302.4	94.0
431.8	139.5	173.6	302.7	93.8
436.8	138.9	172.8	303.3	93.7
441.8	138.4	172.2	303.8	93.6

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TABLE 14.3.4.1-9

DOUBLE-ENDED PUMP SUCTION BREAK

POST–REFLOOD MASS AND ENERGY RELEASE

MINIMUM SI

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
446.8	137.5	171.0	304.7	93.5
451.8	137.2	170.6	305.1	93.3
456.8	136.5	169.8	305.7	93.2
461.8	135.6	168.7	306.6	93.1
466.8	135.2	168.2	307.0	92.9
471.8	134.3	167.1	307.9	92.8
476.8	133.7	166.3	308.6	92.7
481.8	136.8	170.1	305.5	93.7
685.2	136.8	170.1	305.5	93.7
685.3	70.8	87.3	371.4	108.0
686.8	70.8	87.3	371.5	107.9
1029.0	70.8	87.3	371.5	107.9
1029.1	64.0	73.6	378.2	31.0
1894.9	55.1	63.4	387.1	32.6
1895.0	55.1	63.4	71.5	11.2
3600.0	46.7	53.8	79.9	12.7
3600.1	34.5	39.7	92.1	6.3
4742.2	31.4	36.2	95.2	6.5

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Notes:

Path 1: M&E exiting from the steam generator side of the break.
Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

TABLE 14.3.4.1-10

DOUBLE-ENDED PUMP SUCTION

MASS AND ENEGY BALANCE

PAGE 1 OF 2

Time (seconds)		0.00	19.60	19.60	196.79	685.34	1028.97	3600.00
Mass (thousand lbm)								
Initial Mass	Reactor Coolant System and Accumulator	563.04	563.04	563.04	563.04	563.04	563.04	563.04
Added Mass	Pumped Injection	0.00	0.00	0.00	68.65	284.69	436.65	1035.42
	Total Added	0.00	0.00	0.00	68.65	284.69	436.65	1035.42
*****Total Available*****		**563.04**	**563.04**	**563.04**	**631.69**	**847.73**	**999.69**	**1598.46**
Distribution	Reactor Coolant	392.60	29.55	62.57	114.23	114.23	114.23	114.23
	Accumulator	170.44	133.20	100.18	0.00	0.00	0.00	0.00
	Total Contents	563.04	162.75	162.75	114.23	114.23	114.23	114.23
Effluent	Break Flow	0.00	400.28	400.28	517,46	733.50	885.45	1484.24
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	400.28	400.28	517.46	733.50	885.45	1484.24
*****Total Accountable*****		**563.04**	**563.03**	**563.03**	**631.68**	**847.72**	**999.68**	**1598.46**
Energy (million Btu)								
Initial Energy	Reactor Coolant System, Accumulator, SG	584.02	584.02	584.02	584.02	584.02	584.02	584.02
Added Energy	Pumped Injection	0.00	0.00	0.00	4.67	19.36	29.69	70.41
	Decay Heat	0.00	4.73	4.73	20.47	53.07	72.57	184.33

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TABLE 14.3.4.1-10
DOUBLE-ENDED PUMP SUCTION
MASS AND ENERGY BALANCE
PAGE 2 OF 2

	Heat from Secondary	0.00	3.32	3.32	3.32	3.32	3.32	3.32
	Total Added	0.00	8.06	8.06	28.47	75.75	105.59	258.06
*****Total Available*****		**584.02**	**592.08**	**592.08**	**612.49**	**659.77**	**689.61**	**842.09**
Distribution	Reactor Coolant	232.20	7.73	11.03	30.64	30.64	30.64	30.64
	Accumulator	17.01	13.29	10.00	0.00	0.00	0.00	0.00
	Core Stored	22.86	12.75	12.75	4.03	3.99	3.83	2.68
	Primary Metal	113.18	107.76	107.76	92.62	63.50	54.96	37.84
	Secondary Metal	38.31	37.72	37.72	34.44	25.17	20.52	13.98
	SG	160.47	168.91	168.91	149.95	104.76	85.33	57.44
	Total Contents	584.02	348.17	348.17	311.67	228.06	195.28	142.58
Effluent	Break Flow	0.00	243.85	243.85	300.63	431.51	499.60	706.77
	ECCS Spill	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	Total Effluent	0.00	243.85	243.85	300.63	431.51	499.60	706.77
*****Total Accountable*****		**584.02**	**592.02**	**592.02**	**612.29**	**659.58**	**694.88**	**849.35**

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TABLE 14.3.4.1-11

DOUBLE-ENDED PUMP SUCTION BREAK

WITH DIESEL FAILURE – ENERGY BALANCE

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TABLE 14.3.4.2-1

NOMINAL PLANT PARAMETERS AND
INITIAL CONDITION ASSUMPTIONS [(1)]

MAIN STEAM LINE BREAK - MASS AND ENERGY RELEASES

NSSS Power, Mwt	2652
Core Power, Mwt	2644
Reactor Coolant Pump Heat, Mwt	8
Reactor Coolant Flow (total), gpm	260,700
Pressurizer Pressure, psia	2250
Core Bypass, %	6.3
Reactor Coolant Temperatures, °F	
Core Outlet	620.8
Vessel Outlet	616.8
Core Average	587.1
Vessel Average	583.0
Vessel/Core Inlet	549.2
Steam Generator	
Steam Temperature, °F	524.6
Steam Pressure, psia	845
Steam Flow (total), 10^6 lbm/hr	11.70
Feedwater Temperature, °F	440
Zero-Load Temperature, °F	547

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INITIAL CONDITIONS POWER LEVEL (%)

Parameter	Full	70	30	0
RCS Average Temperature	589[(1)]	578.20	563.80	549.00
RCS Flowrate (gpm)	260,700	260,700	260,700	260,700
RCS Pressure (psia)	2,250	2,250	2,250	2,250
Pressurizer Water Level (% span)	60.00	48.66	33.54	22.20
Feedwater Enthalpy (Btu/lbm)	419.40	385.59	312.59	70.68
SG Water Level (% span)	56.00	56.00	56.00	56.00

Note:

1. Noted values correspond to plant conditions defined by 0% steam generator tube plugging and the high end of the RCS T-avg window.

TABLE 14.3.4.3-1

CONTAINMENT ANALYSIS PARAMETERS

ICW temperature (°F)[Containment Integrity][1]	100
Refueling water temperature (°F)	100
RWST minimum water deliverable volume (gal)	247,738
Initial containment temperature (°F)	130
Initial containment pressure (psia)	16.1
Initial relative humidity (%)	30
Net free volume (ft^3)	1.46×10^6

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Emergency Containment Coolers

Total	3
Analysis maximum	2
Analysis minimum	(LOCA) 2
	(Steam Line Break) 1
Setpoint (psig)	6.0
Delay time (sec)	
Without Offsite Power	50.0
With Offsite Power	35.0

Containment Spray Pumps

Total	2
Analysis maximum	2
Analysis minimum	1
Setpoint (psig)	25.0
Delay time (sec)	
Without Offsite Power	60.0
With Offsite Power	45.0

1. ICW temperatures up to 104 degrees F are allowed if supported by the CCW heat exchanger performance monitoring program.

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TABLE 14.3.4.3-2a

LOCA CONTAINMENT HEAT SINK DATA

GOTHIC Heat Sink Number (Description - Orientation)	Area (ft^2)	Material	Thickness (in)
1	10377.15	Admiralty	0.052
2	133257.26	Aluminum	0.008
3	0.001	Aluminum	0.001
4	0.001	Aluminum	0.001
5	125060.96	Carbon Steel	0.075
		Paint Type 1	0.012
6	58675.32	Carbon Steel	0.151
		Paint Type 1	0.012
7	21609.99	Carbon Steel	0.256
		Paint Type 1	0.012
8	11164.83	Carbon Steel	0.347
		Paint Type 1	0.012
9	10624.54	Carbon Steel	0.460
		Paint Type 1	0.012
10	2889.88	Carbon Steel	0.520
		Paint Type 1	0.012
11	1237.66	Carbon Steel	0.630
		Paint Type 1	0.012
12	487.90	Carbon Steel	0.752
		Paint Type 1	0.012
13	1288.42	Carbon Steel	0.886
		Paint Type 1	0.012
14	2708.34	Carbon Steel	0.977
		Paint Type 1	0.012
15	2820.86	Carbon Steel	1.391
		Paint Type 1	0.012
16	3538.57	Carbon Steel	1.656
		Paint Type 1	0.012
17	3080.73	Carbon Steel	2.675
		Paint Type 1	0.012
18	17406.40	Stainless Steel	0.065
19	1152.60	Stainless Steel	0.154
20	3268.80	Stainless Steel	0.255
21	665.35	Stainless Steel	0.367
22	1118.14	Stainless Steel	0.494
23	69.38	Stainless Steel	0.514
24	869.26	Stainless Steel	0.626
25	15.61	Stainless Steel	0.814
26	305.38	Stainless Steel	0.998
27	3.41	Stainless Steel	1.500
28	34.54	Stainless Steel	2.671
29 (Pooling - Horizontal)	425.83	Carbon Steel	0.165
		Paint Type 1	0.012

TABLE 14.3.4.3-2a

LOCA CONTAINMENT HEAT SINK DATA

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GOTHIC Heat Sink Number (Description - Orientation)	Area (ft^2)	Material	Thickness (in)
30 (Pooling - Horizontal)	191.68	Carbon Steel	0.250
		Paint Type 1	0.012
31 (Pooling - Horizontal)	152.43	Carbon Steel	1.000
		Paint Type 1	0.012
32 (Pooling - Horizontal)	4761.49	Stainless Steel	0.057
33 (Pooling - Horizontal)	24.00	Stainless Steel	0.375
34 (Conduit)	9811.00	Galvanized Steel	0.154
35 (Conduit)	2715.91	Galvanized Steel	0.243
36	669.07	Carbon Steel	7.209
		Paint Type 1	0.012
37	23.84	Stainless Steel	4.500
38 (Unlined Concrete - Vertical/Sump)	1850.50	Concrete	6.000
		Paint Type 2	0.049
39 (Unlined Concrete - Vertical/Sump)	7848.53	Concrete	12.000
		Paint Type 2	0.049
40 (Unlined Concrete – Vertical)	27461.89	Concrete	12.000
		Paint Type 2	0.049
41 (Unlined Concrete - Horizontal/Up)	6079.92	Concrete	12.000
		Paint Type 2	0.064
42 (Unlined Concrete - Horizontal/Down)	5742.24	Concrete	12.000
		Paint Type 2	0.064
43 (Containment Dome – Horizontal/Down)	16921.00	Paint Type 1	0.012
		Carbon Steel	0.250
		Gap	0.021
		Concrete	12.000
44 (Containment Cylinder Liner – Vertical/Sump)	45292.66	Paint Type 1	0.012
		Carbon Steel	0.250
		Gap	0.021
		Concrete	12.000
45 (Containment Cylinder – Vertical/Sump)	2037.62	Paint Type 1	0.012
		Carbon Steel	0.5000
		Gap	0.021
		Concrete	12.000
46 (Refueling Cavity - Horizontal/Up)	881.91	Stainless Steel	0.120
		Gap	0.021
		Concrete	12.000
47 (Refueling Cavity Horizontal/Up)	66.29	Stainless Steel	0.375
		Gap	0.021
		Concrete	12.000
48 (Refueling Cavity – Vertical)	5514.10	Stainless Steel	0.0625
		Gap	0.021
		Concrete	12.000

TABLE 14.3.4.3-2a

LOCA CONTAINMENT HEAT SINK DATA

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GOTHIC Heat Sink Number (Description - Orientation)	Area (ft^2)	Material	Thickness (in)
49 (Refueling Cavity – Vertical)	816.67	Stainless Steel	0.188
		Gap	0.021
		Concrete	12.000
50 (Reactor Cavity Wall – Vertical/Sump)	2615.08	Stainless Steel	0.250
		Gap	0.021
		Concrete	12.000
51 (Reactor Cavity Wall – Vertical/Sump)	107.65	Stainless Steel	0.310
		Gap	0.021
		Concrete	12.000
52 (Containment Floor – Horizontal/Up)	7698.81	Paint Type 2	0.064
		Concrete	6.000
53 (Containment Floor Liner – Horizontal/Up)	473.00	Paint Type 2	0.064
		Carbon Steel	0.500
		Gap	0.021
		Concrete	6.000
54 (Lined Concrete Liner – Vertical/Sump)	231.00	Paint Type 2	0.012
		Carbon Steel	0.500
		Gap	0.021
		Concrete	12.000
55 (Lined Concrete – Horizontal/Down)	174.00	Paint Type 1	0.012
		Carbon Steel	0.500
		Gap	0.021
		Concrete	12.000
56 (RX Sump Floor Unlined Concrete – Horizontal/Up)	674.50	Paint Type 2	0.064
		Concrete	6.000
57 (RX Sump Floor Lined Concrete – Horizontal/Up)	16.00	Stainless Steel	0.250
		Gap	0.021
		Concrete	6.000

TABLE 14.3.4.3-2b

STEAMLINE BREAK CONTAINMENT HEAT SINK DATA

PAGE 1 OF 3

GOTHIC Heat Sink Number (Description - Orientation)	Area (ft^2)	Material	Thickness (in)
1	6092.94	Admiralty	0.046
2	0.001	Aluminum	0.0001
3	200.00	Aluminum	0.125
4	0.001	Aluminum	0.0001
5	126312.16	Carbon Steel	0.075
		Paint Type 1	0.012
6	66539.16	Carbon Steel	0.146
		Paint Type 1	0.012
7	21499.95	Carbon Steel	0.257
		Paint Type 1	0.012
8	12033.48	Carbon Steel	0.346
		Paint Type 1	0.012
9	11202.97	Carbon Steel	0.460
		Paint Type 1	0.012
10	3092.10	Carbon Steel	0.519
		Paint Type 1	0.012
11	1237.66	Carbon Steel	0.630
		Paint Type 1	0.012
12	529.79	Carbon Steel	0.747
		Paint Type 1	0.012
13	1288.42	Carbon Steel	0.886
		Paint Type 1	0.012
14	2714.70	Carbon Steel	0.977
		Paint Type 1	0.012
15	2820.86	Carbon Steel	1.391
		Paint Type 1	0.012
16	3552.93	Carbon Steel	1.657
		Paint Type 1	0.012
17	3097.22	Carbon Steel	2.675
		Paint Type 1	0.012
18	19199.88	Stainless Steel	0.067
19	1152.60	Stainless Steel	0.154
20	2202.18	Stainless Steel	0.258
21	665.35	Stainless Steel	0.367
22	883.98	Stainless Steel	0.493
23	69.38	Stainless Steel	0.514
24	869.26	Stainless Steel	0.626
25	15.61	Stainless Steel	0.814
26	305.38	Stainless Steel	0.998
27	3.41	Stainless Steel	1.500
28	34.54	Stainless Steel	2.671
29 (Pooling - Horizontal)	1045.97	Carbon Steel	0.174
		Paint Type 1	0.012
30 (Pooling - Horizontal)	191.68	Carbon Steel	0.250
		Paint Type 1	0.012

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TABLE 14.3.4.3-2b

STEAMLINE BREAK CONTAINMENT HEAT SINK DATA

GOTHIC Heat Sink Number (Description - Orientation)	Area (ft^2)	Material	Thickness (in)
31 (Pooling - Horizontal)	152.43	Carbon Steel	1.000
		Paint Type 1	0.012
32 (Pooling - Horizontal)	4761.49	Stainless Steel	0.057
33 (Pooling - Horizontal)	24.00	Stainless Steel	0.375
34 (Conduit)	9811.00	Galvanized Steel	0.154
35 (Conduit)	2715.91	Galvanized Steel	0.243
36	669.07	Carbon Steel	7.209
		Paint Type 1	0.012
37	23.84	Stainless Steel	4.500
38 (Unlined Concrete - Vertical/Sump)	1850.50	Paint Type 2	0.049
		Concrete	6.000
39 (Unlined Concrete - Vertical/Sump)	7848.53	Paint Type 2	0.049
		Concrete	12.000
40 (Unlined Concrete - Vertical)	27461.89	Paint Type 2	0.049
		Concrete	12.000
41 (Unlined Concrete - Horizontal/Up)	6079.92	Paint Type 2	0.064
		Concrete	12.000
42 (Unlined Concrete - Horizontal/Down)	5742.24	Paint Type 2	0.064
		Concrete	12.000
43 (Containment Dome - Horizontal/Down)	16921.00	Paint Type 1	0.012
		Carbon Steel	0.250
		Air Gap	0.021
		Concrete	12.000
44 (Containment Cylinder Liner - Vertical/Sump)	45292.66	Paint Type 1	0.012
		Carbon Steel	0.250
		Air Gap	0.021
		Concrete	12.000
45 (Containment Cylinder - Vertical/Sump)	2037.62	Paint Type 1	0.012
		Carbon Steel	0.500
		Air Gap	0.021
		Concrete	12.000
46 (Refueling Cavity - Horizontal/Up)	881.93	Stainless Steel	0.120
		Air Gap	0.021
		Concrete	12.000
47 (Refueling Cavity - Horizontal/Up)	66.29	Stainless Steel	0.375
		Air Gap	0.021
		Concrete	12.000
48 (Refueling Cavity - Vertical)	5514.10	Stainless Steel	0.0625
		Air Gap	0.021
		Concrete	12.000
49 (Refueling Cavity - Vertical)	816.67	Stainless Steel	0.188
		Air Gap	0.021
		Concrete	12.000
50 (Reactor Cavity Wall - Vertical/Sump)	2615.08	Stainless Steel	0.250
		Air Gap	0.021
		Concrete	12.000

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TABLE 14.3.4.3-2b

STEAMLINE BREAK CONTAINMENT HEAT SINK DATA

GOTHIC Heat Sink Number (Description - Orientation)	Area (ft^2)	Material	Thickness (in)
51 (Reactor Cavity Wall - Vertical/Sump)	107.65	Stainless Steel	0.310
		Air Gap	0.021
		Concrete	12.000
52 (Containment Floor - Horizontal/Up)	7698.81	Paint Type 2	0.064
		Concrete	6.000
53 (Containment Floor Liner - Horizontal/Up)	473.00	Paint Type 2	0.064
		Carbon Steel	0.500
		Air Gap	0.021
		Concrete	6.000
54 (Lined Concrete Liner - Vertical/Sump)	231.00	Paint Type 2	0.012
		Carbon Steel	0.500
		Air Gap	0.021
		Concrete	12.000
55 (Lined Concrete - Horizontal/Down)	174.00	Paint Type 1	0.012
		Carbon Steel	0.500
		Air Gap	0.021
		Concrete	12.000
56 (RX Sump Floor Unlined Concrete - Horizontal/Up)	674.50	Paint Type 2	0.064
		Concrete	6.000
57 (RX Sump Floor Lined Concrete - Horizontal/Up)	16.00	Stainless Steel	0.250
		Air Gap	0.021
		Concrete	6.000
58	170576.07	Copper	0.008
59	6479.62	Copper/Nickel	0.061

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TABLE 14.3.4.3-3

THERMAL PROPERTIES OF CONTAINMENT HEAT SINK

Material Type	Temperature (°F)	Thermal Conductivity (Btu/hr-ft-°F)	Vol. Heat Cap. (Btu/ft^3-°F)
Admiralty	68	63.6	48.3
Aluminum Brass	100	58.0	46.8
Aluminum	120	84.17	38.64
Carbon Steel	70	27.3	51.5
	100	27.6	53.1
	150	27.8	55.2
	200	27.8	57.1
	250	27.6	58.6
	300	27.3	60.0
	350	26.9	61.1
Stainless Steel	77	8.56	57.5
	212	9.14	59.7
	392	10.24	62.9
Concrete	120	0.8	28.8
Air Gap	120	0.0174	0.0145
Paint Type 1 (Carboline 890/890)	122	0.138	10.55
	212	0.138	11.66
	302	0.138	11.22
Paint Type 2 (Carboline 2011S/890)	122	0.126	24.08
	212	0.126	28.29
	302	0.126	14.93
Copper(1)	50	226	51.34
	500	226	51.34
90/10 Copper/Nickel(1)	50	29	49.77
	500	29	49.77

NOTE: 1. This material type is only utilized in the MSLB analysis

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TABLE 14.3.4.3-4

CONTAINMENT SPRAY PERFORMANCE

Containment Pressure (psig)	Injection			Recirculation (1 Pump)	
	Above Low Level RWST Setpoint		Between Low Level RWST Setpoint and Low Low Level RWST Setpoint		
	1 Pump (gpm)	2 Pumps (gpm)	1 Pump (gpm)	Cold Leg (gpm)	Hot Leg (gpm)
0	1520	3010	1507	1736	1760
10	1480	2931	1467	1704	1731
20	1439	2850	1426	1673	1699
30	1397	2766	1383	1641	1667
40	1355	2680	1340	1608	1635
50	1309	2588	1293	1575	1602
55	1285	2540	1269	1557	1586

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TABLE 14.3.4.3-5

EMERGENCY CONTAINMENT COOLER PERFORMANCE

CONTAINMENT INTEGRITY ANALYSES

(Btu/sec/ECC)

(Based on 2000 gpm CCW Flow/ECC and 25,000 CFM Steam-Air Flow)

		Containment Temperature (°F)									
		120	140	160	180	200	220	240	260	280	300
Component Cooling Water Temp. (°F)	95	319.7	898	1726	2852	4504	6652	9599	13505	18320	25209
	110	222.4	806	1635	2780	4406	6550	9485	13294	18164	24972.7
	115	117	703	1532	2716	4295	6433	9360	13077	18064	24717
	120	0	589	1421	2585	4181	6311	9168	12921	17900	24450
	125	---	464	1298	2468	4059	6177	9021	12752	17464	24214
	130	---	325	1162	2302	3917	6030	8860	12577	17253	23705
	135	---	170	1012	2171	3767	5871	8704	12368	17036	23402
	140	---	0	848	2016	3603	5702	8518	12196	16797	23107
	145	---	---	664	1840	3422	5516	8251	11865	16541	22740
	150	---	---	464	1649	3230	5310	7954	11618	16082	22357
	170	---	---	---	636.4	2188.4	4227.2	6762.0	10291.1	14651.9	20622.4
	210	---	---	---	---	---	1022.3	3373.6	6597.6	10588.4	16012.1

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TABLE 14.3.4.3-6

1.19 FT^2 MSLB HOT ZERO POWER WITH MSCV FAILURE
SEQUENCE OF EVENTS

Time (sec)	Event Description
0.0	Main Steamline break occurs.
5.0	Hi-1 Containment pressure setpoint reached.
7.0	Rod motion occurs (Hi-1 actuates SI which actuates Reactor Trip).
14.0	Main feedwater isolation occurs
31.92	Safety injection flow starts
40.0	Hi-2 containment pressure reached
39.6	Emergency Containment Coolers actuate
47.1	Steamline isolation occurs via hi-2 containment pressure
52.1	Peak containment vapor temperature occurs (304.9)
83.1	Containment sprays actuate
312.2	Peak containment pressure occurs (53.53 psig) and peak structural temperature (279.4°F) occur
653	Mass and energy releases terminate

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TABLE 14.3.4.3-7

DELETED

TABLE 14.3.4.3-8

DOUBLE-ENDED PUMP SUCTION BREAK

SEQUENCE OF EVENTS

Time (Sec)	Event Description
0.0	Break Occurs and Loss of Offsite Power is Assumed
0.5	Containment Fan Cooler Actuation Pressure Setpoint (6.0 psig; Analysis Value) Reached
2.5	Compensated Pressurizer Pressure for Reactor Trip (1881.5 psia) Reached and Turbine Trip Occurs
3.5	Containment Spray Actuation Pressure Setpoint (25.0 psig; Analysis Value) Reached
3.8	Low Pressurizer Pressure Safety Injection Setpoint (1615 psia) Reached (Safety Injection begins Coincident with Low Pressurizer Pressure Safety Injection Setpoint)
10.1	Broke Loop Accumulator Begins Injecting Water
10.2	Intact Loop Accumulator Begins Injecting Water
12.9	Feedwater Isolation Valves Closed (assumes 6 second valve ramp closure)
19.6	End of Blowdown Phase
38.9	Pumped Safety Injection Begins (Includes 35 second Diesel Delay)
50.6	Containment Fan Cooler Begins Heat Removal (Includes 50 second Delay)
54.9	Broken Loop Accumulator Water Injection Ends
55.2	Intact Loop Accumulator Water Injection Ends
60.0	Swing Fan Cooler Begins Heat Removal
63.6	Containment Spray Flow begins
196.8	End of Reflood
486.8	Broken Loop Steam Generator Equilibrium at containment Design Pressure (55.0 psig)
682.2	Containment Peak Pressure (53.85 psig) Occurs
692.2	Containment Peak Structural Temperature (273.5°F) Occurs
685.3	Broken Loop Steam Generator Equilibrium at Containment Pressure of 52.3 psig
1005	Intact Loop Steam Generator Equilibrium at containment Design Pressure (55.0 psig)
1029	Intact Loop Steam Generator Equilibrium at Containment Pressure of 51.3 psig
3600	Broken and Intact Loop Steam Generator Full Equilibrium (0 psig and 212°F)
4742.2	Switchover to Recirculation Begins
5042.2	Recirculation Sprays Initiated (Injection Spray Termination Plus 300 Second Delay)
8.426×10^6	Containment Temperature 122°F Reached
10.0×10^6	Transient Modeling Terminated

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TABLE 14.3.4.3-9

DOUBLE-ENDED HOT LEG

SEQUENCE OF EVENTS

Time (Sec)	Event Description
0.00	Break Occurs and Loss of Offsite Power is Assumed
0.39	Containment Fan Cooler Actuation Pressure Setpoint (6.0 psig; Analysis Value) Reached
2.28	Compensated Pressurizer Pressure for Reactor Trip (1881.5 psia) Reached and Turbine Trip Occurs
3.56	Containment Spray Actuation Pressure Setpoint (25.0 psig; Analysis Value) Reached
3.63	Low Pressurizer Pressure Safety Injection Setpoint (1615 psia) Reached (Safety Injection begins Coincident with Low Pressurizer Pressure Safety Injection Setpoint)
8.94	Broken Loop Accumulator Begins Injecting Water
9.03	Intact Loop Accumulator Begins Injecting Water
12.9	Feedwater Isolation Valves Closed (assumes 6 second valve ramp closure)
20.51	Peak Pressure (52.9 psig) and Peak Temperature Occur (278.7°F)
25.60	End of Blowdown Phase
25.60	Transient Modeling Terminated

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TABLE 14.3.4.3-10

CONTAINMENT INTEGRITY RESULTS

LOCA
(Loss of Offsite Power Assumed)

FAILURE SCENARIO	INITIAL CONT. PRESS (psig)	PEAK PRESS (psig)	TIME OF PEAK (sec)	PEAK TEMP (°F)	TIME OF PEAK (sec)	PRESS AT 24 HRS (psig)
DEPS w/Diesel, 1 ECC, 2nd ECC at 24 hrs; w/Continued Recirc Spray	1.4	53.85	682.2	273.5	692.2	14.4
DEHL	1.4	52.9	20.51	278.7	20.51	---

MSLB
(Offsite Power Available)

FAILURE SCENARIO	INITIAL CONT. PRESS (psig)	PEAK PRESS (psig)	TIME OF PEAK (sec)	PEAK STRUCTURAL TEMP (°F)	TIME OF PEAK (sec)
1.19 ft^2 SPLIT BREAK at HZP	1.4	53.53	312.2	279.4	312.2

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FCP 08-0007

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTAINMENT PRESSURE
DEPS : DIESEL FAILURE CASE WITH
1 CSS AND 2 ECCs AT Pcont = 0.3 PSIG

FIGURE 14.3.4.3-1

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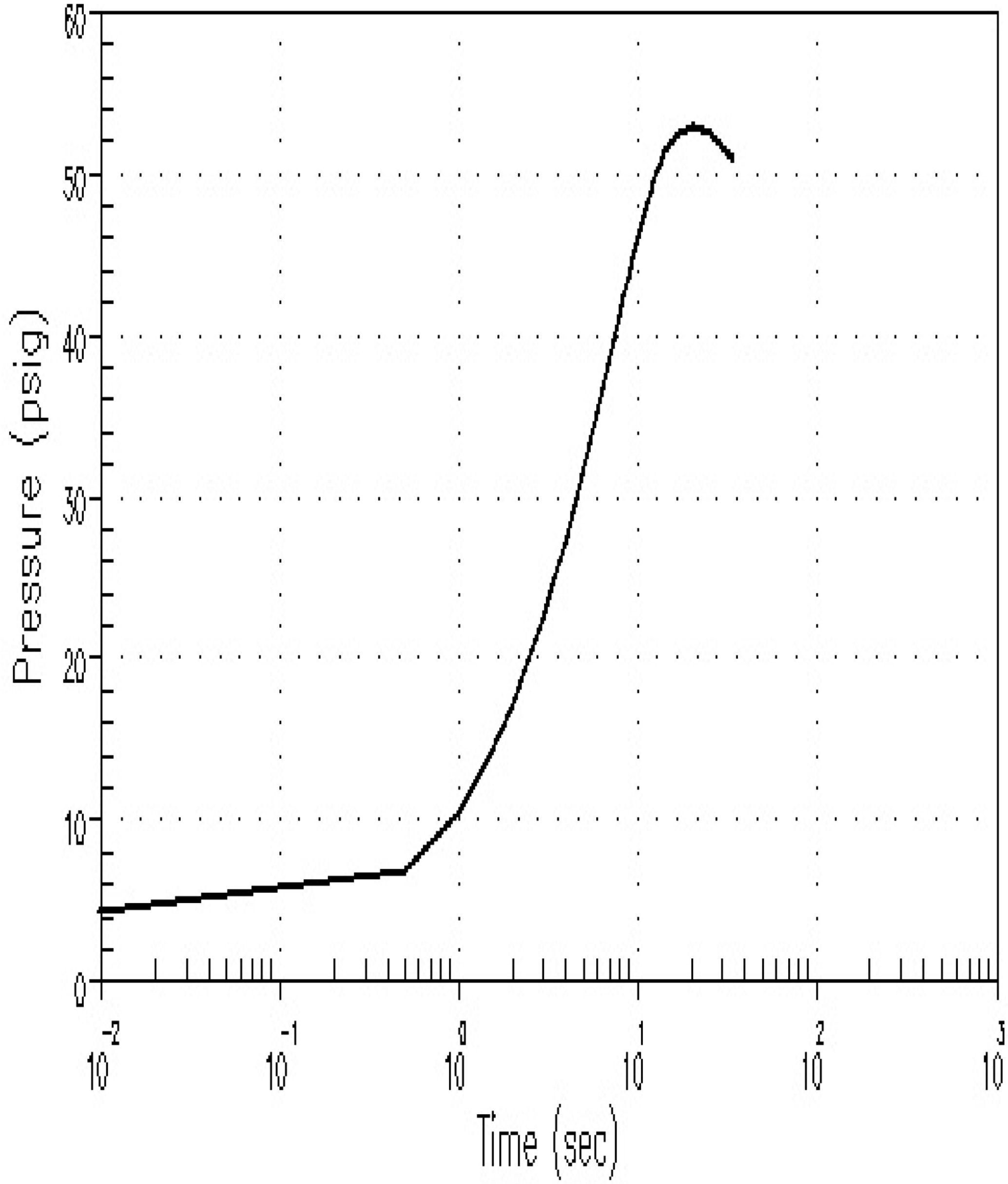
FCP 08-0007

C23

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTAINMENT STEAM TEMPERATURE
DEPS : DIESEL FAILURE CASE WITH
1 CSS AND 2 ECCs AT Pcont = 0.3 PSIG

FIGURE 14.3.4.3-2



REVISED 04/17/2013

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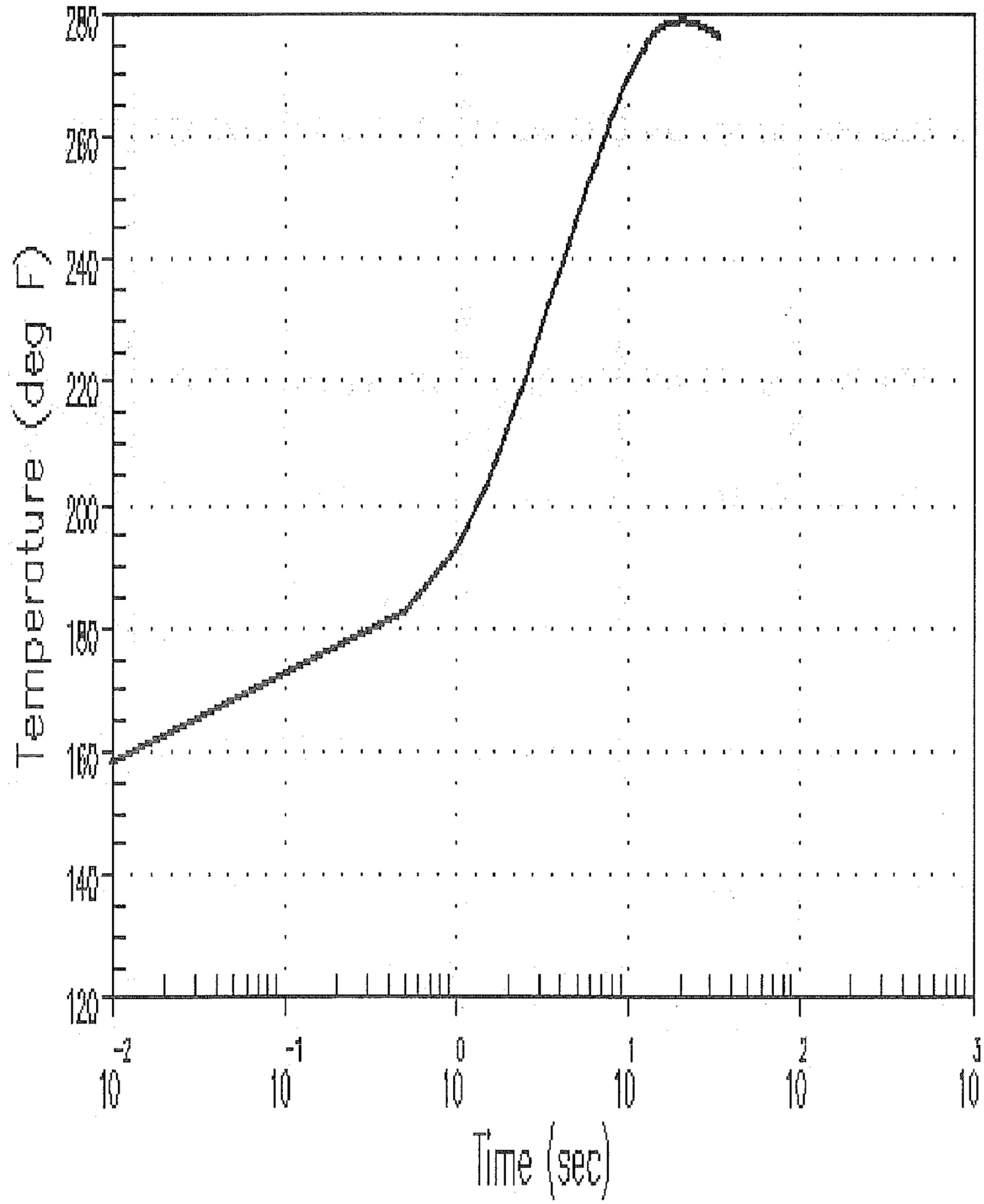
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTAINMENT PRESSURE
DOUBLE –ENDED HOT LEG BREAK
(MINIMUM SAFEGUARDS)

FIGURE 14.3.4.3-3



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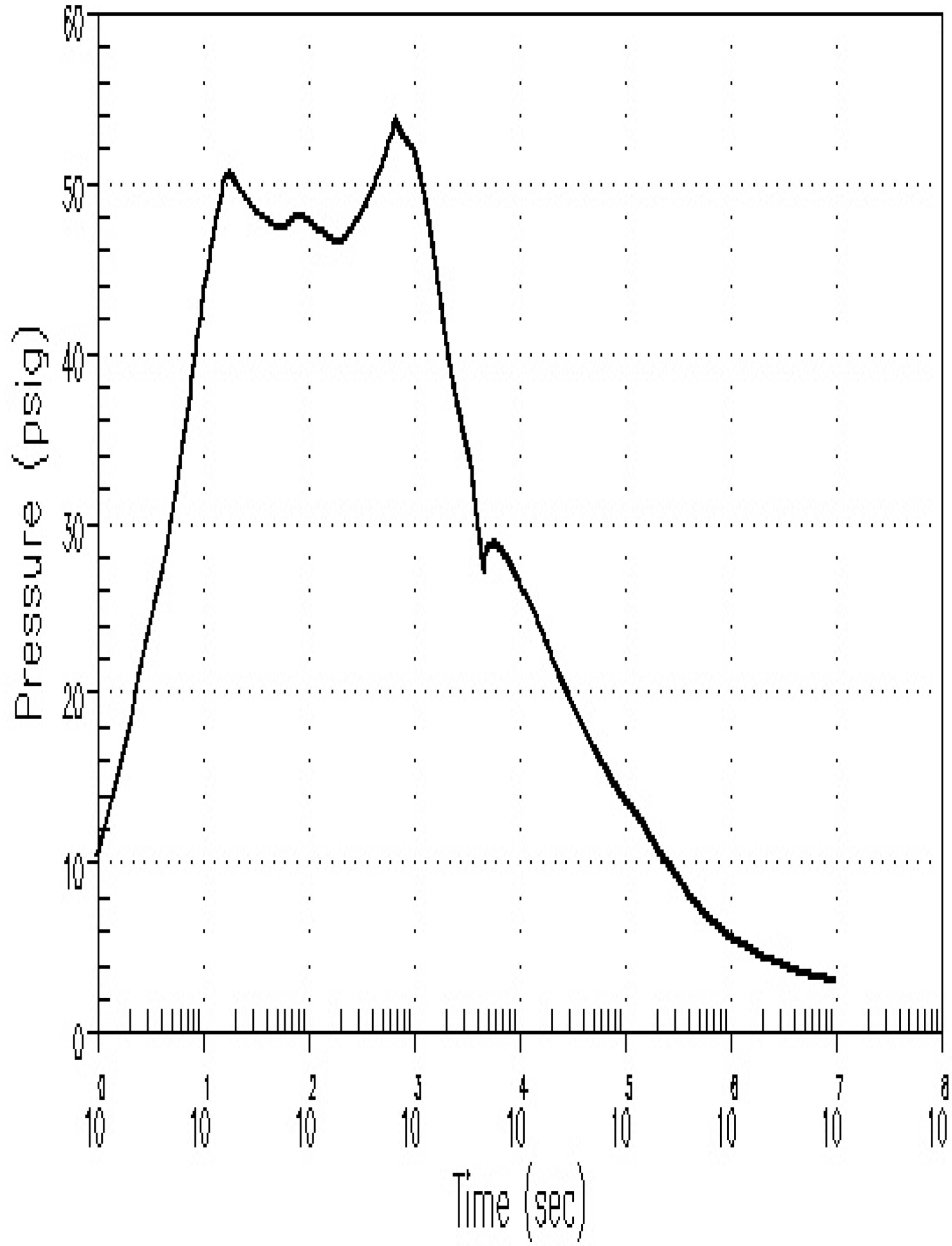
REVISED 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTAINMENT TEMPERATURE
DOUBLE –ENDED HOT LEG BREAK
(MINIMUM SAFEGUARDS)

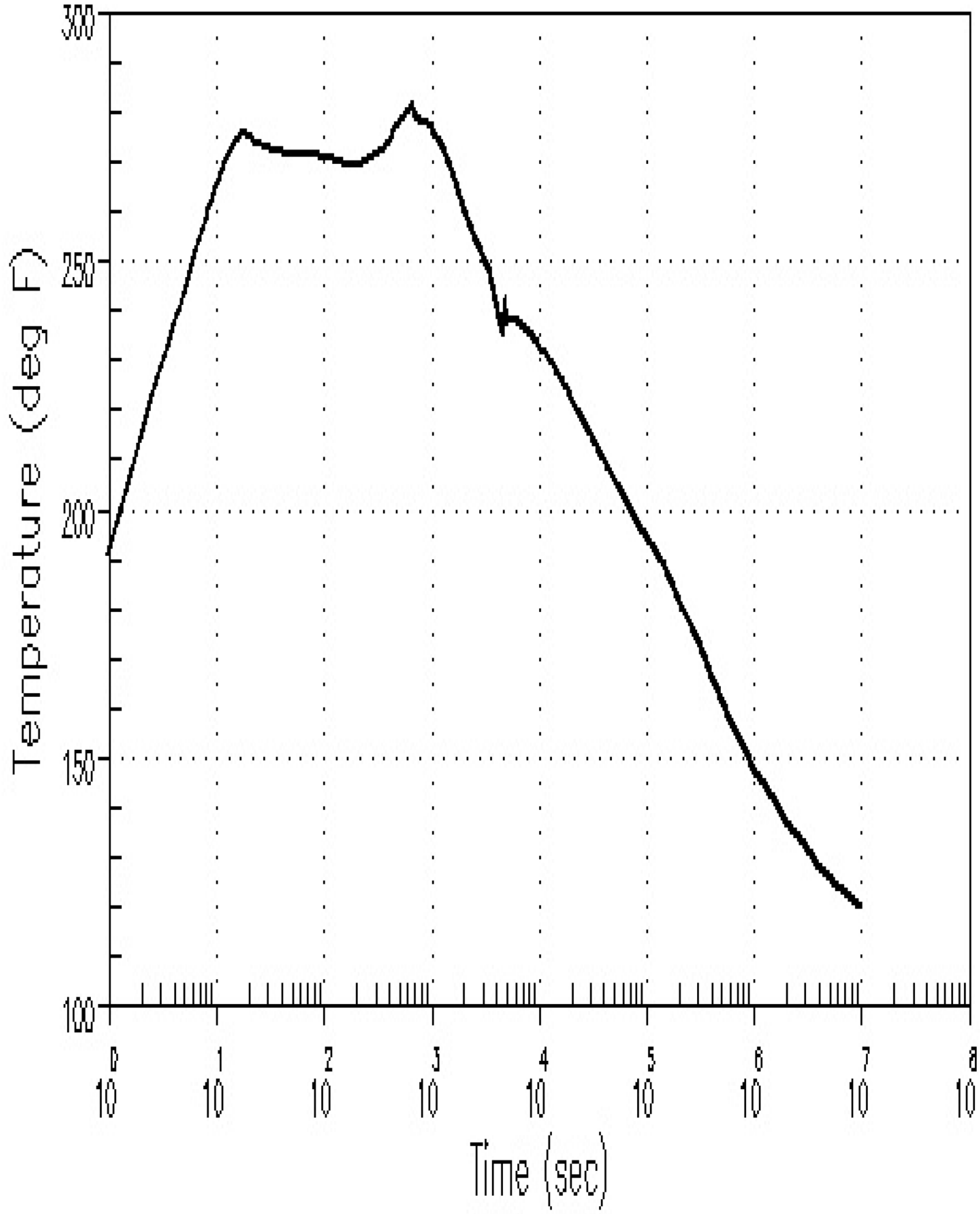
FIGURE 14.3.4.3-4



REVISED 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTAINMENT PRESSURE
DOUBLE-ENDED PUMP SUCTION
BREAK (MINIMUM SAFEGUARDS)
FIGURE 14.3.4.3-5



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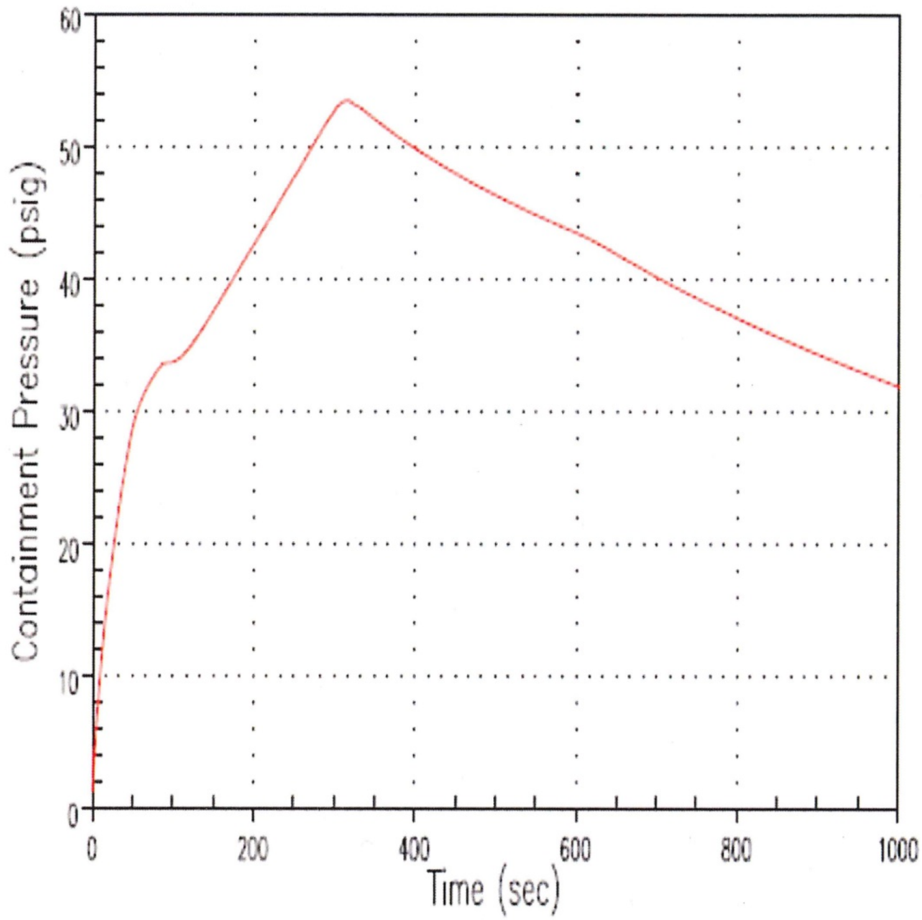
REVISED 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTAINMENT TEMPERATURE
DOUBLE-ENDED PUMP SUCTION
BREAK
(MINIMUM SAFEGUARDS)
FIGURE 14.3.4.3-6



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Revised 07/28/2016

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTAINMENT PRESSURE 1.19 FT^2 HZP STEAMLINE BREAK, MSCV FAILURE SPLIT BREAK, 2 ECCs AND CCSs

FIGURE 14.3.4.3-7

14.3.5 ENVIRONMENTAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

14.3.5.1 ANALYSIS

The original licensing basis LOCA dose analysis can be found in its entirety in Appendix 14F. Also refer to Appendix 14F for the discussion of the atmospheric dispersion model, whole body dose computations, and the radiological assessment of containment purging. This section describes the LOCA dose analysis performed as part of the Extended Power Uprate (EPU) Project. C26

A large pipe rupture in the reactor coolant system (RCS) is assumed to occur. As a result of the accident, it is assumed that core damage occurs and iodine and noble gas activity is released to the containment atmosphere. A portion of this activity is released via the containment purge system, which is open when the accident occurs and activity is released to the atmosphere through this path until the containment purge system is isolated. Also, once Engineered Safety Features (ESF) recirculation is established, leakage from ESF equipment outside containment releases activity to the outside environment.

The LOCA dose consequence analysis is consistent with the guidance provided in Appendix A of Reference 1, "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 Reg. Guide 1.183, Regulatory Position 3.1. A conservative power level is used which exceeds 102% of the rated core thermal power. The resulting core source term is provided in Table 14.3.5-7. 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 the Regulatory Guide 1.183.
2. Regulatory Position 2 – The sump pH is controlled at a value greater than 7.0 based on the addition of sodium tetraborate decahydrate baskets. 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. C26

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 5.58 hr^{-1}. This removal is credited in both the sprayed and unsprayed regions of containment.

   A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region of containment as well as in the sprayed region when sprays are not operating.

   No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3 – A single train of containment spray provides coverage to 34.5% of the containment volume. Therefore, the Turkey Point containment building atmosphere is not considered to be a single, well-mixed volume. The containment is divided into three regions: sprayed and unsprayed region above the operating deck and an unsprayed region below the operating deck. The mixing rates for the containment sprayed and unsprayed regions are based on a GOTHIC analysis which produced results consistent with NUREG/CR-4102 (Reference 9). C26

   The GOTHIC analysis utilized for Turkey Point to demonstrate the level of spray induced mixing in containment included both subdivided and lumped parameter models. The detailed subdivided models were used to calculate flow patterns produced by the containment sprays and the emergency containment coolers. Gas concentrations from the subdivided models were compared with concentrations in the lumped parameter model and used to determine equivalent mixing flow rates for the lumped model.

   Based on the results of this analysis, the AST dose calculations were conducted using a three volume model similar to the lumped parameter GOTHIC model. The AST model includes separate volumes representing the unsprayed lower, unsprayed upper and sprayed upper regions of containment. Mixing flow rates up to 375,000 cfm between lower and upper unsprayed regions and 990,000 cfm between upper sprayed and unsprayed regions conservatively cover the possible combinations of sprays and emergency fans that may be available during an accident scenario.

The method used in the Turkey Point AST LOCA analysis for determining the time period required to reach an elemental iodine decontamination factor (DF) of 200 was based on a containment atmosphere peak iodine concentration equal to 40 percent of the core iodine inventory per Table 2 of the Regulatory Guide 1.183.

The SRP requires that the elemental iodine spray removal coefficient should be set to zero when a DF of 200 is reached for elemental iodine. In addition, the particulate spray removal coefficient should be reduced by a factor of 10 when a DF of 50 is reached for the aerosols.

As discussed in the SRP, the iodine DF is a function of the effective iodine partition coefficient between the sump and containment atmosphere. Thus, the loss of iodine due to other mechanisms (containment leakage, surface deposition, etc.), would not be included in the determination of the time required to reach a DF of 200. In addition, since the iodine in the containment atmosphere and sump are decaying at the same rate, decay should not be included in determining the time to reach a DF of 200. Additional RADTRAD-NAI cases were performed for determining the time to reach a DF of 200. C26

The first RADTRAD-NAI case was used to determine the peak containment atmosphere elemental iodine concentration and amount of aerosol in the containment atmosphere. This case included:

- No containment spray
- No elemental iodine surface deposition
- No aerosol surface deposition
- No decay
- No containment leakage

The second RADTRAD-NAI case determined the time required to reach a DF of 200 based on the peak elemental iodine concentration from the first RADTRAD-NAI case. The second RADTRAD-NAI case included:

- Containment sprays actuated at 0.018 hours
- No surface deposition
- No decay
- No containment leakage

Due to the high mixing rate between the containment regions, the activity in all three containment regions was considered. The second RADTRAD-NAI case showed that a DF of 200 for elemental iodine was reached at a time greater than 2.305 hours.

A separate RADTRAD-NAI case was then used to determine the time required to reach a DF of 50 for aerosols based on the peak aerosol mass from the first RADTRAD-NAI case. This RADTRAD-NAI case included:

- Containment sprays actuated at 0.018 hours
- Aerosol surface deposition credited
- No decay
- No containment leakage

Due to the high mixing rate between the containment regions, the activity in all three containment regions is considered. The third RADTRAD-NAI case showed that a DF of 50 was reached at a time greater than 3.06 hours.

Containment spray flow is assumed to be stopped for a period of five minutes to allow for manual re-alignment of the pump suction from the RWST to the recirculation sump. Termination of spray flow is considered in the determination of the iodine decontamination factors and is reflected in the mixing rates between the containment regions. C26

6. Regulatory Position 3.4 – Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems is not credited in this analysis.
7. Regulatory Position 3.5 – This position relates to suppression pool scrubbing in BWRs, which is not applicable to Turkey Point.
8. Regulatory Position 3.6 – This position relates to activity retention in ice condensers, which is not applicable to Turkey Point.
9. Regulatory Position 3.7 – A containment leak rate of 0.20% per day of the containment air is assumed for the first 24 hours based on Technical Specification leak rate limits. After 24 hours, the containment leak rate is reduced to 0.10% per day of the containment air. The containment leakage was applied to all three containment regions.

10. Regulatory Position 3.8 – routine containment purge is considered in this analysis. The purge release evaluation assumes that 100% of the radionuclide inventory of the RCS is released instantaneously at the beginning of the event. The containment purge flow is 700 cfm and is isolated after 8 seconds, which is before the onset of the gap release phase. No filters are credited.

11. Regulatory Position 4.1 through 4.6 provide guidance for the evaluation of the transport, reduction, and release of radioactive material through dual containment structure. These positions are not applicable to Turkey Point.

12. 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.

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13. Regulatory Position 5.2 – Leakage from the ESF system is taken as two times the value from Table 6.2-12. ECCS leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continue for the 30-day duration. Backleakage to the Refueling Water Storage Tank is also considered separately as 0.1 gph, which exceeds two times the expected leakage through the two sets of isolation valves between the RWST and recirculation flow. Back leakage to the RWST is assumed to begin at the start of recirculation and continue for the remainder of the 30-day duration.

14. Regulatory Position 5.3 – with the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

15. Regulatory Position 5.4 – A flashing fraction of 9.2% was calculated based on a conservative maximum sump liquid temperature and containment design pressure. However, consistent with Regulatory Position 5.5, the flashing fraction for ECCS leakage is assumed to be 10%. 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 to the RWST tank.

16. Regulatory Position 5.5 – The amount of iodine that becomes airborne is conservatively assumed to be 10% of the total iodine activity in the leaked fluid for the ECCS leakage entering the Reactor Auxiliary Building. For the ECCS leakage back to the RWST, the sump and pH history and temperature are used to evaluate the amount of iodine that enters the RWST air space.

17. Regulatory Position 5.6 – For ECCS leakage into the auxiliary building, the form of the released iodine is 97% elemental and 3% organic. No credit for ESF filtration of the ECCS leakage nor holdup or dilution in the auxiliary building is taken. For releases from the RWST, 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 dilution of activity in the RWST.

18. Regulatory Position 6 – This position relates to MSSV leakage in BWRs, which is not applicable to Turkey Point.

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19. 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.

The EPU Power level of 2644 MW_{th}, with 0.3% calorimetric uncertainty, or 2652 MW_{th} is used in the analysis. The source term represents end of cycle conditions assuming enveloping initial fuel enrichment and an average core burnup of 45,000 MWD/MTU. The resulting design basis radiological dose analysis whole core radionuclide inventory is provided in Table 14.3.5-7.

For ECCS leakage and containment purge dose analyses, the RCS radionuclide inventory is required as an initial input into the release calculations. The primary coolant source term for Turkey Point is calculated based upon maximum equilibrium concentrations from operation at 2652 MW_{th} with small defects in 1 percent of the fuel rod cladding. The equilibrium iodine activities were then adjusted to achieve the Technical Specification limit of 0.25 μCi/gm dose equivalent I-131 (DE I-131). The non-iodine inventory was adjusted to remain below the pre-EPU inventory limit of 100/E-Bar. The noble gas radionuclides were adjusted to the Technical Specification limit of 447.7 μCi/gm dose equivalent Xe-133 (DE Xe-133).

The non-iodine activities were determined by first developing a list of isotopes which satisfied the radionuclide requirements specified in Table 5 of Reg. Guide 1.183. Iodine nuclides and isotopes with half lives less than 30 days were deleted from this list in accordance with the pre-EPU Technical Specification definition of E-Bar. Equilibrium RCS activities based upon 1% fuel defects were combined with corrosion product activities from ANSI/ANS-18.1-1999 (Reference 8) for Cr-51, Fe-55, F-59, and Mn-54 to calculate a total RCS specific activity. This value was found to be slightly greater than the pre-EPU Technical Specification limit of 100/E-bar. The activity for each isotope was then adjusted by a constant factor such that the sum of the adjusted activities was equal to the pre-EPU 100/E-bar limit. The determination of the DE I-131 adjustment applied to all non-noble gas radionuclides is presented in Table 14.3.5-9.

A value for DE Xe-133 was calculated using the Technical Specification definition and the equilibrium noble gas activities based upon 1% fuel defects. This value was corrected using the adjustment factor needed to achieve a total specific activity equal to the pre-EPU limit of 100/E-bar described above. The determination of the adjusted DE Xe-133 inventory of 447.7 μCi/gm is presented in Table 14.3.5-9. C26

The resulting design basis (adjusted) primary coolant source term activities for the RCS inventory of dose significant isotopes is presented in Table 14.3.5-8.

Both offsite and control room doses are determined. The dose evaluation includes not only determining doses due to containment leakage but also doses due to an open containment purge system, ECCS leakage to the auxiliary building, and ECCS leakage to the RWST. The total control room operator doses include a shine dose component that considers shine from the containment, the environment outside the control room, as well as the control room filtration equipment sources.

Environmental Consequences of a LOCA is classified as Condition IV per ANS-51.1/N18.2-1973 (Reference 10).

Containment Leakage

The Technical Specification design basis containment leak rate of 0.20% by weight of containment air is used for the initial 24 hours. Thereafter, the containment leak rate is assumed to be one-half the design value, or 0.1% per day, in accordance with Regulatory Position 3.7 of Regulatory Guide 1.183 (Reference 1).

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 (Reference 3) as 5.96 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 is credited in the sprayed region. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 534,442 ft^3 of the total 1,55E6 ft^3 containment volume which ranges from a minimum of 1.45E6 ft^3 to a maximum of 1.60E6 ft^3. Therefore, the Turkey Point containment building atmosphere is not considered to be a single, well-mixed volume. The containment is divided into three regions: sprayed and unsprayed regions above the operating deck and an unsprayed region below the operating deck. The mixing rates between the regions are based on a separate sensitivity study evaluating various combinations of containment fans and sprays to produce the most conservative mixing rates. The final conservative mixing rates are 1,300,000 cfm between the upper sprayed and upper unsprayed containment regions above the operating deck and 355,000 cfm between the lower unsprayed region below the operating deck and the upper unsprayed region above the operating deck. C26

According to SRP 6.5.2, the effectiveness of elemental iodine removal by the containment sprays is presumed to end when the decontamination factor (DF) reaches a maximum value of 200. The maximum initial airborne elemental iodine concentration is based on the release of 40 percent of the core iodine inventory. With the elemental iodine spray removal rate set to the SRP limit of 20 hr^{-1}, the decontamination factor for elemental iodine reaches 200 at just over 2.25 hours.

The spray aerosol removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated aerosol iodine removal rate of 6.09 hr^{-1}, the time for containment spray to produce an aerosol decontamination factor of 50 is calculated to be greater than 3.06 hours.

The major assumptions and parameters used to determine the doses as the result of containment leakage are given in Table 14.3.5-1 C26

Containment Purge

The containment purge system is assumed to be open at the time the accident occurs. However, the large break LOCA results in a containment isolation signal, which automatically closes the containment purge system isolation valve. Although the valve closure time is approximately 5 seconds, a closure time of 8 seconds is used in this analysis to account for time for signal generation.

The time at which fuel clad damage would be initiated (i.e., the hot rod burst time) following the accident is well after 8 seconds. Thus, the activity release to containment for this case is limited to the RCS activity prior to the large break LOCA which results in a pre-accident iodine spike. C26
The iodine release from the RCS to the containment is assumed to be 100% elemental iodine. Since only HEPA filters (which remove particulate iodine) exist in the containment purge system, it is assumed that the iodine release from the plant stack is unfiltered. C26

The containment purge system flowrate is limited to 7000 cfm when the system is open during power operation of the plant.

The RCS noble gas activity prior to the LOCA is based on a 1.0% fuel defect level.

The major assumptions and parameters used to determine the doses as the result of containment purge are given in Table 14.3.5-2.

ECCS Leakage to the Auxiliary Building

The ECCS leakage to the auxiliary building is 4,650 cc/hr based upon two times the current licensing basis value of 2,325 cc/hr. The leakage is assumed to start at 15 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that 10% of the total iodine is released from the leaked liquid. The form of the released iodine is 97% elemental and 3% organic. C26

Since the auxiliary building ventilation system cannot be assured to be in operation for the duration of this event, the release point may occur from any penetration in the building. Examination of the layout of the facility indicates that the assumption that the release occurs from the RWST vent location would conservatively bound the expected atmospheric dispersion/transport mechanism for this release.

The major assumptions and parameters used to determine the doses as the result of ECCS leakage to the auxiliary building are given in Table 14.3.5-3.

<u>ECCS Leakage to the RWST</u>

The ECCS backleakage to the RWST is assumed to be 0.1 gph based upon doubling of the expected total seat leakage through both sets of motor operated valves which isolate the recirculation flow from the RWST. Leakage is assumed to start at 15 minutes into the event when recirculation begins and to continue throughout the 30-day period.

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The time-dependent concentration of the total iodine in the RWST, including stable iodine, was determined from the tank liquid volume and leak rate. A constant value of 1.0E-07 gm-atom/liter, which bounds the maximum concentration determined by a detailed time dependent analysis, is applied in the dose consequence analysis. Application of NUREG/CR-5950 (Reference 2) methodology to the RWST iodine and pH conditions resulted in an elemental iodine fraction of 0.1058, which was then used to calculate the iodine release rate from the RWST.

The elemental iodine in the liquid leaked into 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/CR-5950. 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 of the RWST. The average daily temperature swing of 10.1°F is applied for every 24-hour period for 30 days and no credit is taken for cooling of the tank contents via conduction. The iodine release is implemented via an adjustment to the vapor flow rate from the RWST. This adjustment accounts for the time-dependent relationship between the elemental iodine concentration in the RWST vapor space with respect to the sump iodine concentration. The average adjusted RWST vapor release rate is then applied to the entire iodine inventory in the containment sump.

This same approach is used with the organic iodine. An organic iodine fraction of 0.0015 is assumed in combination with a partition coefficient of 1.0. The particulate portion of the leakage is assumed to be retained in the liquid phase of the RWST. Therefore, the total iodine flow from the RWST represents the sum of the elemental and organic concentrations in the RWST vapor space.

The major assumptions and parameters used to determine the doses as the result of ECCS leakage to the RWST are given in Table 14.3.5-3.

<u>Shine Dose to Control Room Operators</u>

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 from ECCS leakage. C26

2. External radioactive plume shine contribution from the containment and ECCS 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.

Each component of the shine dose is evaluated based on the time dependent source terms in the appropriate locations (containment, auxiliary building, Control Room filters) determined by the dose analysis computer code. These source terms are used in a shielding analysis computer code (Reference 4) model of source, distance and shielding to determine the shine dose for operators throughout the 30 day duration of the dose analysis event.

<u>Control Room Parameters</u>

The doses to personnel in the control room are determined for each of the activity release paths discussed above. The control room volume is 47,786 ft^3, the filtered makeup flow is 525 cfm, the filtered recirculation flow is 375 cfm, and the unfiltered inleakage flow is 100 cfm. The control room filter removal efficiency is 95% for all chemical forms of iodine.

The control room dose analysis assumptions and parameters are given in Table 14.3.5-4. The dose conversion factors, occupancy factors, breathing rates and atmospheric dispersion factors used in the dose calculations are given in Table 14.3.5-5. Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12, respectively (References 6 and 7),

Acceptance Criteria

The offsite doses must meet the guidelines of Regulatory Guide 1.183 (Reference 1) and 10 CFR 50.67 (Reference 5), or 25 rem TEDE for the initial 2 hour period following the accident at the exclusion boundary (EB) and for the duration of the accident at the low population zone (LPZ). The dose criteria for control room personnel following the accident is 5 rem TEDE.

14.3.5.2 RESULTS

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The total offsite and control room doses are given in Table 14.3.5-6, along with the doses due to the activity release from all release paths. The total offsite doses and the total control room doses due to the large break LOCA meet the acceptance criteria.

14.3.5.3 REFERENCES

1. USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000.

2. USNRC, NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992.

3. USNRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 6.5.2.

4. MicroShield Version 5 "User's Manual" and "Verification & Validation Report, Rev. 5," Grove Engineering, both dated October 1996.

5. US Code of Federal Regulations, 10 CFR 50.67, and "Accident Source Term."

6. Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

7. Federal Guidance Report No. 12 (FGR 12), “External Exposure to Radionuclides in Air, Water, and Soil,” 1993.

8. ANSI/ANS-18.1-1999, “Radioactive Source Term for Normal Operation of Light Water Reactors,” approved September 21, 1999; including Errata dated December 1, 2005. C26

9. USNRC NUREG/CR-4102, “Air Currents Driven by Sprays in Reactor Containment Buildings”.

10. American National Standard, ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", August 6, 1973.

TABLE 14.3.5-1

ASSUMPTIONS USED FOR
LARGE BREAK LOCA DOSE ANALYSIS CONTAINMENT LEAKAGE

Input/Assumption	Value
Release Inputs:	
Core Power Level	2652 MW_{th}
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Initial RCS Equilibrium Activity	0.25 µCi/gm DE I-131, and 447.7 µCi/gm DE Xe-133 (Table 14.3.5-8)
RCS Mass (maximum)	397,544 lbm
Containment Free Volume	1.45E6 ft3 – 1.60E6 ft3
Containment Leakage Rate	
0 to 24 hours	0.20% (by weight)/day
After 24 hours	0.10% (by weight)/day
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	Reg. Guide 1.183, Sections 3.1, 3.2, and Table 2
Removal Inputs:	
Containment Aerosol/Particulate Natural Deposition (only credited in unsprayed regions)	0.1/hour
Surface Area for Wall Deposition	537,903 ft^2
Containment Elemental Iodine Wall Deposition	5.96/hour
Containment Sprayed Region Volume	534,442 ft^3
Spray Fall Height	70 feet
Volumetric Spray Flow Rate	2.827 ft^3/sec
Containment Upper Unsprayed Region Volume	643,864 ft^3
Containment Lower Unsprayed Region Volume (below operating deck)	371,694 ft^3
Flowrate Between Sprayed and Upper Unsprayed Volumes	1,300,000 cfm
Flowrate Between Upper Unsprayed and Lower Unsprayed Volumes	355,000 cfm
Spray Removal Rates:	
Elemental Iodine	20 hr^{-1}
Time to reach DF of 200	2.25 hours
Aerosols	6.09 hr^{-1} (reduced to 0.609 at 3.028 hours)
Time to reach DF of 50	Greater than 3.027 hours
Spray Initiation Time	63.8 seconds
CR HVAC Isolation Signal	High Containment Radiation
Time of CR Isolation	30 seconds
Unfiltered Inleakage	100 cfm
Containment Purge Filtration	0%
Transport Inputs:	
Release Location	Nearest containment penetration to CR ventilation intakes
Atmospheric Dispersion Factors	
Offsite	Appendix 2E
Onsite	Appendix 2F

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TABLE 14.3.5-2

ASSUMPTIONS USED
FOR
LARGE BREAK LOCA DOSE ANALYSIS CONTAINMENT PURGE

Input/Assumption	Value
Release Inputs:	
Core Power Level	2652 MW_{th}
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Initial RCS Equilibrium Activity	0.25 μCi/gm DE I-131, and 447.7 μCi/gm DE Xe-133 (Table 14.3.5-8)
RCS Mass (maximum)	397,544 lbm
Containment Free Volume	1.45E6 ft^3
Containment Leakage Rate	
0 to 24 hours	0.20% (by weight)/day
After 24 hours	0.10% (by weight)/day
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	Reg. Guide 1.183, Sections 3.1, 3.2, and Table 2
Containment Purge Release (Unfiltered)	7,000 cfm for 8 seconds C26
Removal Inputs:	
CR HVAC Isolation Signal	High Containment Radiation
Time of CR Isolation	30 seconds
Unfiltered Inleakage	100 cfm
Containment Purge Filtration	0%
Transport Inputs:	
Release Location	Plant Stack to CR ventilation intakes
Atmospheric Dispersion Factors	
Offsite	Appendix 2E
Onsite	Appendix 2F

TABLE 14.3.5-3

ASSUMPTIONS USED FOR
LARGE BREAK LOCA DOSE ANALYSIS
ECCS LEAKAGE TO AUXILIARY BUILDING AND TO RWST

Input/Assumption	Value
Release Inputs:	
Core Power Level	2652 MW_{th}
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Initial RCS Equilibrium Activity	0.25 μCi/gm DE I-131, and 447.7 μCi/gm DE Xe-133 (Table 14.3.5-8)
RCS Mass (maximum)	397,544 lbm
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	Reg. Guide 1.183, Sections 3.1, 3.2, and Table 2
Removal Inputs:	
<u>ECCS Systems Leakage (from 15 minutes to 30 days)</u>	
Sump Volume (minimum)	239,000 gallons (31,949.5 ft^3)
ECCS Leakage (2 times allowed value)	4650 cc/hr
Flashing Fraction	0.10
Chemical form of the iodine released from the ECCS leakage	97% elemental, 3% organic
No filtration or credit for building dilution, released directly to the environment	
<u>RWST Back-leakage</u>	
Sump Volume (minimum)	239,000 gallons (31,949.5 ft^3)
ECCS Leakage to RWST (2 times allowed value)	0.1 gph
Flashing Fraction	0% based on temperature of fluid reaching the RWST. Elemental iodine is released into tank space based upon partition factor.
RWST liquid/vapor elemental iodine partition factor	41.18
CR HVAC Isolation Signal	High Containment Radiation
Time of CR Isolation	30 seconds
Unfiltered Inleakage	100 CFM
Containment Purge Filtration	0%
Transport Inputs:	
ECCS Release Location	RWST Vent to CR ventilation intakes
RWST Backleakage Release Location	RWST Vent to CR ventilation intakes
Atmospheric Dispersion Factors	
Offsite	Appendix 2E
Onsite	Appendix 2F

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TABLE 14.3.5-4

ASSUMPTIONS USED FOR
LARGE BREAK LOCA DOSE ANALYSIS
CONTROL ROOM MODEL

Parameter	Value
Control Room Volume	47,786 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Ricirculation flow Rate	0 cfm
Unfiltered Make-up Flow Rate	1000 cfm
Unfiltered Inleakage	100 cfm
Emergency Operation	
Recirculation Mode:	
Filtered Make-up Flow Rate	525 cfm
Filtered Ricirculation flow Rate	375 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage	100 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%

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TABLE 14.3.5-5

DOSE CONVERSION FACTORS
BREATHING RATES AND
ATMOSPHERIC DISPERSION FACTORS

Dose Conversion Factors for Dose Analysis Events

Dose conversion factors for inhalation and submersion are from Federal Guidance reports (FGR) Nos. 11 and 12, respectively

Breathing Rates & Occupancy Factors for Dose Analysis Events

Reg. Guide 1.183, Sections 4.1.3 and 4.2.6

Time (hours)	EAB & LPZ Breathing Rate (m^3/sec)	Control Room Breathing Rate (m^3/sec)	Control Room Occupancy Factor
0.0	3.5E-4	3.5E-4	1.0
8.0	1.8E-4	3.5E-4	0.6
24.0	2.3E-4	3.5E-4	0.4
720.0	2.3E-4	3.5E-4	0.4

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Offsite Atmospheric Dispersion (X/Q) Factors for Dose Analysis Events

See Appendix 2E

Time Period	EAB X/Q (sec/m^3)	LPZ X/Q (sec/m^3)
0-2 hours	1.37E-04*	2.73E-05
0-8 hours	7.89E-05	1.23E-05
8-24 hours	6.00E-05	8.24E-06
1-4 days	3.30E-05	3.46E-06
4-30 days	1.40E-05	9.95E-07

*With the exception of the WGDT Rupture, only the 0-2 hour EAB X/Q is used in the event analyses.

Onsite Atmospheric Dispersion (X/Q) Factors for Dose Analysis Events

See Appendix 2F

TABLE 14.3.5-6

LARGE BREAK LOCA OFFSITE AND CONTROL ROOM DOSES

Dose contribution	EAB Dose[1] (rem TEDE)	LPZ Dose[2] (rem TEDE)	Control Room Dose[2] (rem TEDE)
Containment Purge	4.2151E-03	8.4187E-04	6.1532E-02
Containment Leakage	4.9006E+00	1.1302E+00	2.3399E+00
ECCS Leakage	7.1444E-02	8.5838E-02	5.0252E-01
RWST Leakage	2.1021E-04	5.4898E-04	4.3934E-03
Shine Dose			0.728
Total	4.98	1.22	3.64
Acceptance Criteria	25[3]	25[3]	5[4]

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Notes:

(1) Worst 2-hour dose

(2) Integrated 30-day dose

(3) Regulatory Guide 1.183, Table 6

(4) 10CFR50.67

TABLE 14.3.5-7

DESIGN BASIS WHOLE CORE RADIONUCLIDE INVENTORY SOURCE TERM

Security-Related Information - Withheld Under 10 CFR 2.390

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TABLE 14.3.5-8

DESIGN BASIS PRIMARY COOLANT (RCS) RADIONUCLIDE INVENTORY SOURCE TERM
(1% FUEL DEFECT PLUS DEI-131 AND DEXe-133 ADJUSTMENTS)

Security-Related Information - Withheld Under 10 CFR 2.390

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TABLE 14.3.5-9

DESIGN BASIS PRIMARY COOLANT (RCS) ADJUSTMENTS

IODINE RADIONUCLIDE INVENTORY ADJUSTMENT

Isotope	Equilibrium RCS Activity (μCi/gm)	DCF_i	RCS Activity x DCF_i	Adjusted RCS Activity (μCi/gm)	Adjusted RCS Activity x DCF
I-131	3.609E+00	8.890E-09	3.208E-08	2.006E-01	1.783E-09
I-132	2.559E+00	1.030E-10	2.636E-10	1.422E-01	1.465E-11
I-133	4.375E+00	1.580E-09	6.913E-09	2.431E-01	3.841E-10
I-134	4.738E-01	3.550E-11	1.682E-11	2.633E-02	9.347E-13
I-135	2.160E+00	3.320E-10	7.171E-10	1.200E-01	3.985E-11
Total			3.999E-08		2.223E-09

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Raw DE I-131 4.499 Adj DE I-131 0.25

NOBLE GAS RADIONUCLIDE INVENTORY ADJUSTMENT

Isotope	Equilibrium Activity (μCi/gm)	Effective DCF_i (Sv/Bq)	Activity x DCF
Kr-85m	1.188E+00	7.480E-15	8.889E-15
Kr-85	3.486E+01	1.190E-16	4.149E-15
Kr-87	7.157E-01	4.120E-14	2.949E-14
Kr-88	2.113E+00	1.020E-13	2.156E-13
Xe-131m	2.849E+00	3.890E-16	1.108E-15
Xe-133m	3.275E+00	1.370E-15	4.487E-15
Xe-133	2.360E+02	1.560E-15	3.681E-13
Xe-135m	4.570E-01	2.040E-14	9.324E-15
Xe-135	5.080E+00	1.190E-14	6.045E-14
Xe-138	4.766E-01	5.770E-14	2.750E-14
Total			7.290E-13

De Xe-133 = 7.290E-13 / 1.560E-15 = 467.3

14.3.6 HYDROGEN CONCENTRATION CONTROL

Sources and Characteristics of Hydrogen

As described in the Turkey Point Updated FSAR Section 9.12, Turkey Point Units 3 and 4 received an exemption from the hydrogen control requirements of 10 CFR 50.44 and 10 CFR 50, Appendix A, General Design Criteria 41, 42, and 43, in December 2001 (Reference 16). The following discussion of hydrogen C26
generation pertains to plant operation following the thermal uprate of 1995 and is presented for historical context only, not intended for update in future FSAR revisions.

For several months following a maximum hypothetical accident there would be gradual rise in hydrogen concentration in the reactor containment. Hydrogen is generated by radiolysis of the reactor coolant, by the zirconium-water reaction and by chemical reaction of materials in the post-accident Containment environment. The hydrogen concentration could potentially increase to levels where a flammable recombination reaction with oxygen occurs (i.e., a hydrogen burn) releasing additional energy within the containment. The resultant rise in temperature and pressure would not be expected to affect the containment vapor barrier integrity nor the health and safety of the public.

The following factors ensure that sufficient safety margin will exist following a hydrogen burn during design basis and severe accidents to preclude loss of the containment function.

The containment pressure will be reduced to approximately 3 psig due to operation of the containment heat removal systems when the hydrogen concentration is predicted to reach the flammability limit. Thus, significant pressure margin will exist inside containment to accommodate hydrogen burns.

- Deflagration would be the most likely mode of hydrogen combustion in the containment building. This combustion mechanism is the least damaging and does not produce any dynamic or impulsive loads on the containment structure.
- Hydrogen concentrations above the flammability limit will not last very long without being ignited due to the large number of random ignition sources inside containment. Common sources of random ignition inside containment include sparks from electrical equipment and small static electric discharges.

The lower flammability limit for hydrogen in saturated air at room temperature and atmospheric pressure is 4.1 volume percent (v/o)(References 1, 2, and 3). The propagation characteristics in the flammability range up to about 18 v/o lie within subsonic velocities. The subsonic combustion waves or deflagrations result in quasistatic (nearly steady state) loads on the containment. Flame propagation occurs only in the upward direction up to 6 v/o concentration because the rate of convective rise is greater than its rate of propagation. Up to 9 v/o concentration both horizontal and upward propagation occurs. From 9 to 18 v/o the rate of flame propagation increases rapidly in all directions. Detonation occurs at concentrations above 18 v/o. The combustion waves associated with detonations travel supersonically and produce dynamic or impulsive loads on containment in addition to quasistatic loads.

Not all of the hydrogen burns when ignition occurs in concentrations under 10 v/o. At about 5.6 v/o, only 50% of the hydrogen initially present recombines. Sparks from electrical equipment, small static electric discharges, or hot surfaces can cause ignition. Hydrogen ignites also without a spark or other external energy supply when the temperature is sufficiently high (Reference 4). This spontaneous ignition temperature varies with emission velocity and steam content, and occurs conservatively, at $1256^{\circ}F$ for low velocities and high vapor concentrations, down to as low as about $968^{\circ}F$ where a hydrogen jet impinges on a solid object at high velocity.

A. Radiolysis of Water

Following the postulated accident, a potentially major source of hydrogen production would result from the decomposition of water by radiolysis. Such decomposition of water is caused by the complex interaction of ionizing radiation and water or dilute aqueous solutions. The initial products of radiolysis are generally believed to be the hydrated electron $e^-_{(aq)}$, the OH^- radical, and H_3O^+ and are formed along the path of energy absorption. These initial products next either react with one another or other constituents of the solution. These subsequent reactions occur, with different rate constants, to form hydrogen, hydrogen peroxide, and oxygen in addition to other products. These subsequent reactions are also responsible for a certain amount of recombination which can occur. The essential net result is the generation of oxygen and hydrogen gases unless the solutions contain material which reacts with them.

In a closed system, the net rate of decomposition of water eventually becomes zero. The exact equilibrium concentrations, however, depend upon a number of factors such as water purity, the amount of hydrogen, hydrogen peroxide or oxygen in the solution. It is important to note, however, that the equilibrium concentration is strongly affected by the loss from the system of gaseous reaction products. Since the situation at Turkey Point limits this loss, the calculation of the equilibrium value is conservative.

Following the accident, it is not possible to determine the degree to which gaseous reaction products are lost from the water since for some period of time following the accident the emergency core cooling water may be at or near saturation enthalpy. Furthermore, the coolant is not pure water but contains boric acid, materials added for pH control and various corrosion and fission products.

The rate of hydrogen production is customarily expressed in terms of G values. Primary or direct yields of a species are indicated by a subscript, e.g., G_{H2}, and the net production considering secondary reactions is indicated by a parenthetical notation e.g., $G(H_2)$. For pure water, there appears to be sufficient evidence that the maximum rate of production of the species, H_2, as a result of beta and gamma radiation is 0.44 molecules of hydrogen per 100 electron volts absorbed, or $G_{H2} = 0.44$. For pure water or dilute solutions which do not contain reactive solutes the maximum net yield of hydrogen is equal to the initial direct yield when no recombination occurs, hence $G_{H2} = G(H_2)$.

Westinghouse studies of radiolysis in dynamic systems (Reference 5) show 0.44 molecules per 100 ev to be a maximum yield for high solution flow rates through a gamma radiation field. Work by ORNL (References 6 and 7), Zittel (Reference 8), and Allen (Reference 9) confirm this value.

A value of $G(H_2) = 0.44$ is a representative maximum value to describe the net hydrogen yield immediately following the loss-of-coolant accident. This value would be expected to decrease somewhat as coolant temperature decreases and, hence, gas solubility increases resulting in increasing recombination within the liquid.

The energy source of radiolysis derives from the decay of fission products originally located within the fuel rods. Following a large loss-of-coolant accident, some cladding damage is expected and consequently a fraction of the more volatile fission products contained in the fuel rod gas gap would be released and be distributed throughout the water and atmosphere within the containment.

To be consistent with the general approach used to evaluate the offsite effects of a major accident with a nuclear reactor, the released fission products are grouped into three broad categories, viz, the halogens, the noble gases, and solids.

It is worth noting at this point that the hydrogen yield from a given amount of any fission product is greater if that fission product is dissolved or suspended in the coolant than if it remained within the fuel rod. This is because essentially all the beta energy and all but a few percent of the gamma energy is absorbed within the fuel rod. Therefore, to be conservative, the assumptions regarding fission product release are the same as is used for reactor siting purposes as described in TID-14844 (Reference 10). These assumptions are:

a. 100% release of noble gases.
b. 50% release of halogens.
c. 1% release of "solids".

The total radiolytic hydrogen produced is the sum of that produced by fission products retained in the core and that produced by fission products released from the core but which remain with the coolant. Since energy is produced from these two sources at different rates, the hydrogen production from these sources are determined separately.

1. In-Core Contribution

The in-core contribution is determined from the fission product decay energy, based on the assumption that 7.4% of the gamma energy is absorbed by the solution in the region of the core. It is assumed that the noble gases escape to the containment vapor space.

The $G(H_2)$ value described above, 0.44 molecules per 100 ev, is utilized in the analysis.

2. Out-of-Core Contribution

In the case of the out-of-core contribution to radiolysis, the total decay energy from the released fission products, both beta and gamma, is assumed to be fully absorbed in the solution.

The depth of the sump solution inhibits the ready diffusion of hydrogen from solution; this retention of hydrogen in solution will have a significant effect in reducing the hydrogen yields to the containment atmosphere. The buildup of hydrogen in solution will enhance the back reaction to form water and lower the net hydrogen yields in the same manner as a reduction in the gas to liquid volume ratio will reduce the yield. Based on the work of Bell (Reference 11), a value of 0.30 molecules per 100 ev has been used for the net G value when computing the net production of hydrogen in the sump water.

3. Total Radiolytic Hydrogen

The amount of hydrogen produced in-core and out-of-core and the total radiolytic production are included in Figure 14.3.6-1. It was assumed that the reactor had been operating at 2346 Mw(t) and that just prior to the loss-of-coolant accident the containment temperature was 130°F at 1 atmosphere pressure. The containment atmosphere temperature at the time of the accident affects the initial amount of air with which the hydrogen will be mixed and decreases with increasing initial temperature; hence, the value selected is based on the highest expected normal containment temperature during operation.

B. Zirconium - Water Reaction

Zirconium will react with steam given the proper conditions according to the following reaction:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

The reaction rate becomes significant at a temperature of 1800°F and increases rapidly with increasing temperature. Thus, the hydrogen would be formed in an environment at a temperature considerably higher than that required for ignition. However, the action of the emergency core cooling systems will limit temperatures attained by the reactor core following a loss-of-coolant such that only a small fraction of the Zirconium in the core will react. Calculations indicate that a small fraction of 1% of the Zirconium in the core will react. Because of the temperature distribution across the core, the highest local fraction reached will be less than 1% while some parts of the core will not experience any reaction.

The reactor core contains approximately 36,800 lb of Zircaloy; 36,300 lb is cladding which is potentially subject to the high temperature required for significant reaction.

For conservatism, the amount of Zirconium reacted is assumed to be 5% or 1840 lbs. This reaction is assumed to occur essentially instantaneously.

The hydrogen discharge issuing from a reactor coolant pipe rupture would be impinging on solid objects at high velocity which spontaneous ignition temperature was earlier stated to be approximately 968°F. Thus, in order to prevent ignition as hydrogen flows from the break, it would be necessary to cool it by at least 544°F, or more likely, by as much as 832°F.

Calculations have shown that the heat loss from the hydrogen stream to the reactor coolant structure will not reduce the temperature below the spontaneous ignition temperature along the direct flow path to the rupture location. Cooling by mixing with saturated vapor does not appear likely considering that the zirconium - water reaction model assumes the availability and consumption of steam to sustain the reaction.

C. Corrosion of Metals

The problem of corrosion of metals has received a great deal of study and has been found to be a very complex subject. Although it is generally believed that corrosion is basically an electrochemical process, there are questions of protective films, polarization, oxidation, concentration cells and electrode potentials which confuse the issue so that practical solutions to corrosion problems are largely empirical. The fact that corrosion studies are slanted to the protection of the metal makes it difficult to apply the available information on corrosion to the problem which concerns us here, i.e., the generation of hydrogen within the containment after a loss-of-coolant accident.

To better understand the complexities of the corrosion problem, a brief review of the sequence of events following the postulated MHA is presented. On the initiation of the break, the reactor cooling system water will spurt out, partly flashing into steam, and impinge on any equipment in its path. The water will flow down all paths available to it and collect in the bottom of the containment. The composition of the solution collecting in the containment bottom initially will have the same composition that it had in the reactor coolant system when the reactor was operating at power except to be somewhat concentrated because of the flashing to steam. At the beginning of life, this composition could be as high as 1250 ppm of boron as boric acid with the pH adjusted by the addition of a chemical, such as lithium hydroxide. At the end of life, the boron concentration in the primary coolant will be essentially zero, and there may be a very small amount of lithium present for pH adjustment

A few seconds after the break, boron will start to be injected into the reactor coolant system. The system contains 1950 to 2050 ppm boric acid. The solution from the accumulators and the refueling water storage tank will fill the reactor coolant system as far as possible with the remainder spilling and running into the containment bottom. Accordingly, the solution discharging onto the equipment within the containment at the break may start out as a neutral or slightly alkaline solution and then will become more acid as the blowdown proceeds.

Approximately 30-60 seconds after the break, the containment spray system will start to operate, spraying water from the refueling water storage tank into the top of the containment. Accordingly, all of the components and structures in the containment will be drenched by this boric acid spray. The water in the bottom of the containment will be a solution which probably will be somewhat alkaline and will become more acidic as the spray continues until a maximum of approximately 300,000 gallons of solution have been used.

The spray introduced into the containment will rapidly come to temperature equilibrium with the air-steam atmosphere. The temperature of the containment atmosphere will reach 270°F approximately ten seconds after the break and fall slowly (Reference 12). This figure is for the minimum safeguards operating. After a period of time, the pumps' suction will be switched from the refueling water storage tanks to the containment sump. The volume of solution in the sump will be approximately 300,000 gallons when the reactor coolant system has been refilled. Depending upon the location of the break, some portion of the approximately 65,000 gallons of reactor coolant volume will be added to the 300,000 gallons in the sump.

Once the recirculation mode is started, the composition of the solution sprayed in the containment and that in the sump will be the same, except that the spray liquid will contain appreciable amounts of dissolved oxygen due to the exposure to the containment atmosphere. Within the primary system and in portions of the sump the dissolved oxygen may be consumed in the corrosion reactions.

Within containment there are a variety of metals and zinc-rich coatings which can potentially be important sources of hydrogen generation during the post-accident period. The most important materials include galvanized coatings, non-coated and coated zinc primers, and exposed aluminum metal. The total inventory of these types of materials which have the potential to generate hydrogen by chemical reaction in the post-accident containment environment were calculated (Reference 15). Hydrogen production was then calculated for these inventories of zinc and aluminum within containment.

For zinc and aluminum, the reactions of concern are the following:

$$Zn+H_2O \rightarrow ZnO+H_2$$

and,

$$2Al+3H_2O \rightarrow Al_2O_3+3H_2$$

Corrosion rates for galvanized metal, coated steel, and aluminum are based on industry data or national laboratory experiments to evaluate corrosion rates in a post-LOCA environment. Corrosion rates for aluminum metal are taken from R. C. Burchell and D.D. Whyte (Reference 13), based on an anticipated pH of 7.

In the case of the zinc material in containment, a distinction was made between zinc metal and zinc primer with epoxy topcoat. The latter was based on NUREG/CR-3803 (Reference 14). This approach acknowledges the fact that the qualified coating will remain intact, at least for some period of time following the accident. According to Reference 14, failure of the phenolic topcoat in the vapor/spray mode occurs "via a cracking of the phenolic, but with no delamination." The tests described in Reference 14 show that "the cracked phenolic in these cases did not become detached from the substrate, but remained bonded to the primer." It is reasonable to make use of a decreased corrosion rate in consideration of this fact.

D. Total Hydrogen Generation

The total hydrogen generated from the radiolysis of water, the zirconium-water reaction, and metal corrosion are given in Figure 14.3.6-1. The zirconium-water contribution assumes a 5% reaction takes place immediately following the MHA, while the contributions from radiolysis and corrosion are time dependent.

E. Dispersion of Local H_2 Concentrations

Hydrogen mixing within the containment is accomplished by the Containment Emergency Cooling System fans and the Containment Spray System. These systems and the internal structures of the containment are designed to maintain a well-mixed containment atmosphere, and to prevent hydrogen pocketing.

The safety equipment for containment air mixing (containment spray and containment emergency cooler fans) start on automatic signals following a LOCA. Operation of the containment spray provides thermodynamic mixing of the contents of the containment atmosphere. Operation of the emergency containment cooler fans provides mechanical mixing of the atmosphere contents. The time required to process one containment volume for Turkey Point Units 3 and 4 is approximately 30 minutes with two fan coolers operating at their rated capacity.

Control of Post-Accident Combustible Gases

Turkey Point Units 3 and 4 received an exemption from the hydrogen control requirements of 10 CFR 50.44 and 10 CFR 50, Appendix A, General Design Criteria 41, 42, and 43, in December 2001 (Reference 16). The exemption was based in part on recent industry studies (References 17, 18, and 19) which concluded that large dry containment building designs such as those at Turkey Point, have a very low risk of failure from hydrogen combustion during design basis accidents.

The exemption also considered the impact of hydrogen combustion during severe accidents. Severe accidents can result in large quantities of hydrogen being released over short periods of time. Reference 16 acknowledged that the hydrogen control systems necessitated by 10 CFR 50.44 would likely be overwhelmed under severe accident conditions in which there is a significant amount of core damage. Thus, operation of such systems would provide no benefit in limiting the effects of hydrogen combustion, and hence are not needed for severe accident mitigation.

The Turkey Point Individual Plant Examination (IPE) concluded that the containment would remain intact for severe accidents without operation of hydrogen control systems, as long as the containment heat removal systems (Containment Emergency Cooling and Containment Spray) remained operable to reduce the containment temperature between burns.

Reference 18 indicates that full core meltdown accidents in which both in-vessel and ex-vessel hydrogen generation occur can potentially challenge the performance of large, dry containment designs. Hydrogen control during these events can be accomplished by either venting the containment, or inerting the containment atmosphere.

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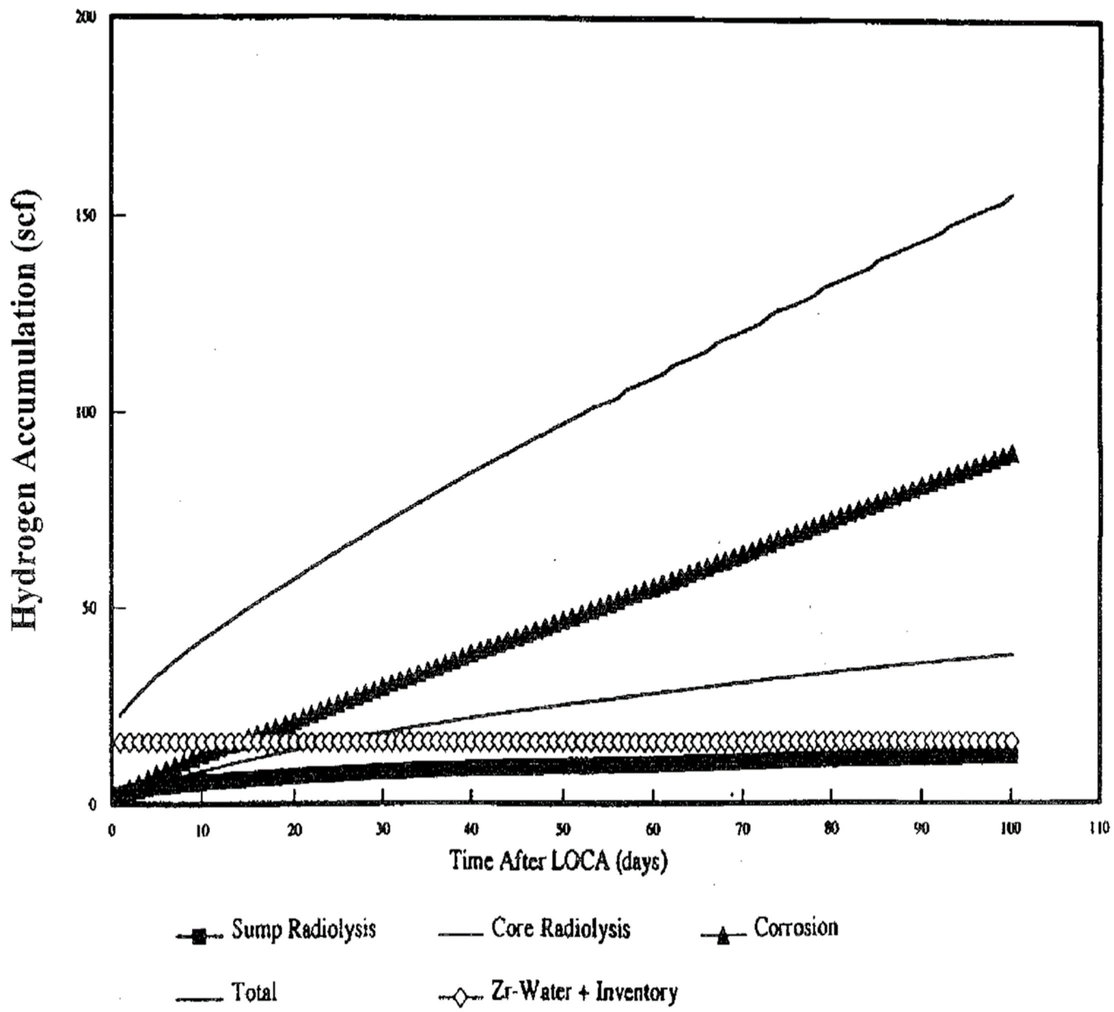

REV. 14 (2/97)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

HYDROGEN ACCUMULATION -
NO RECOMBINER
5% ZIRCONIUM-WATER REACTION
FIGURE 14.3.6-1