Conclusions

The analysis performed has demonstrated that for the partial loss-of-coolant event, the DNBR does not decrease below the limit value at any time during the transient. Therefore, no fuel or cladding damage is predicted and all applicable acceptance criteria are met.

Complete Loss of Forced Reactor Coolant Flow

Accident Description

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

Normal power for the RCPs is supplied through buses from a transformer connected to the generator and the offsite power system. Each pump is on a separate bus. When a generator trip occurs, the buses continue to be supplied from external power lines and the pumps continue to supply coolant flow to the core.

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

- Reactor coolant pump power supply undervoltage reactor trip
- Low reactor coolant loop flow reactor trip
- Pump circuit breaker opening, (RCP supply underfrequency opens pump circuit breaker, which trips the reactor).

The reactor trip on RCP undervoltage is provided to protect against conditions that can cause a loss of voltage to all RCPs; that is, station blackout. This function is blocked below approximately 10-percent NIS and 10-percent turbine power (Permissive 7).

The reactor trip on low primary coolant flow is provided to protect against loss-of-flow conditions that affect one or both reactor coolant loops. This function is generated by two-out-of-three low flow signals per reactor coolant loop. Above 10-percent NIS power (Permissive 8), low flow in either loop will actuate a reactor trip. Above 10-percent NIS and 10-percent turbine power (Permissive 7), low flow in both loops will actuate a reactor trip.

The reactor trip on RCP underfrequency (pump circuit breaker opening) is available to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. However, the analysis conservatively assumes that this function is not available to provide a reactor trip. Therefore, the low primary coolant flow reactor trip function is assumed to provide primary protection against an underfrequency event.

This event is conservatively analyzed to the following acceptance criteria:

- Pressure in the RCS and MSS should be maintained below 110 percent of the design values.

- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the limit value.

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

Method of Analysis

The complete loss-of-flow transient is analyzed as a loss of both RCPs with both loops in operation. The event is analyzed to show that the integrity of the core is maintained as the DNBR remains above the safety analysis limit value. The loss-of-flow events do result in an increase in RCS and MSS pressures, but these pressure increases are generally not severe enough to challenge the integrity of the RCS and MSS. Since the maximum RCS and MSS pressures do not exceed 110 percent of their respective design pressures for the loss-of-load event, it is concluded that the maximum RCS and MSS pressures will also remain below 110 percent of their respective design pressures for the loss-of-flow events.

Two cases are analyzed:

- Complete loss-of-flow transient due to a loss of power to both pumps
- Complete loss-of-flow transient due to an underfrequency condition

The underfrequency case represents the worst credible coolant flow loss. For this case, flow decreases due to a constant frequency decay rate of 5 Hz/s. Reactor trip is then caused by a low-flow signal.

The transients are analyzed with two computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code is then used to calculate the heat flux and DNBR transients based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for the fuel.

This event is analyzed with RTDP. Initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values. Minimum measured flow is also assumed. A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive MTC limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

A limiting EOC DNB axial power shape is assumed in VIPRE for the calculation of DNBR. This shape provides the most limiting minimum DNBR for the loss-of-flow events.

A conservatively low trip reactivity value (3.5-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event.

This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (1.8 seconds to dashpot). The trip reactivity versus rod position curve is confirmed to be valid as part of the RSAC verification process.

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

A maximum, uniform, SGTP level of 10 percent was assumed in the RETRAN analysis. Reactor coolant system loop flow asymmetry due to a loop-to-loop SGTP imbalance does not need to be considered for transients in which both reactor coolant pumps experience a coastdown.

Results

Figures 5.1.8-9 through 5.1.8-16 illustrate the transient response for the complete loss of flow associated with a loss of power to both RCPs with both loops in operation. The minimum DNBR is 1.386/1.386 (thimble/typical) which occurred at 4.0 seconds (DNBR limit: 1.34/1.34 (thimble/typical)).

Figures 5.1.8-17 through 5.1.8-24 illustrate the transient response for the complete loss-of-flow (underfrequency) case. Both RCPs decelerate at a constant rate until a reactor trip on low flow is initiated. The minimum DNBR is 1.423/1.420 (thimble/typical), which occurred at 4.15 seconds (DNBR limit: 1.34/1.34 (thimble/typical)). These results are based on a cycle-specific worst-power shape that was utilized to obtain margin between the safety analysis DNBR limit and the design DNBR limit.

The calculated sequence of events for both complete loss-of-flow cases are shown on Table 5.1.8-2. Following reactor trip, the RCPs will continue to coast down, and natural circulation flow will eventually be established. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

Conclusions

The analysis performed has demonstrated that for the complete loss-of-flow event, the DNBR does not decrease below the limit value at any time during the transient. Therefore, no fuel or cladding damage is predicted and all applicable acceptance criteria are met.

Locked-Rotor Accident

Accident Description

The postulated locked-rotor accident is an instantaneous seizure of an RCP rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low-flow signal. The consequences of a postulated pump shaft break accident are similar to the locked-rotor event. With a broken shaft, the impeller is free to spin, as opposed to it being fixed in position during the locked-rotor event. Therefore, the initial rate of reduction in core flow is greater during a locked-rotor event than in a pump shaft break event because the fixed shaft causes greater resistance than a free-spinning impeller early in the transient, when flow through the affected loop is in the positive direction. As the transient continues, the flow direction through the affected loop is reversed. If the impeller is able to spin free, the flow to the core will be less than that available with a fixed-shaft during periods of reverse flow in the affected loop. Because peak pressure, cladding temperature, and DNB occur very early in the transient, the reduction in core flow during the period of forward flow in the affected loop dominates the severity of the results. Consequently, the bounding results for the locked-rotor transients also are applicable to the RCP shaft break.

After the locked rotor, reactor trip is initiated on an RCS low-flow signal. At the time of reactor trip, the unaffected RCP is assumed to lose power and coast down freely.

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

The locked-rotor event is analyzed to the following criteria:

- Pressure in the RCS should be maintained below the designated limit (see below).

- Coolable core geometry is ensured by showing that the peak cladding temperature and maximum oxidation level for the hot spot are below 2700°F and 16.0 percent by weight, respectively.

- Activity release is such that the calculated doses meet 10 CFR Part 100 guidelines.

For KNPP, the locked-rotor RCS pressure limit is equal to 110 percent of the design value, or 2750 psia. For the secondary side, the locked-rotor pressure limit is also assumed to be equal to 110 percent of design pressure, or 1210 psia. Since the loss-of-load analysis bounds the locked rotor, a specific MSS overpressurization analysis is not performed.

A hot-spot evaluation is performed to calculate the peak cladding temperature and maximum oxidation level. Finally, a calculation of the "rods-in-DNB" is performed for input to the radiological dose analysis.

<u>Method of Analysis</u>

The locked-rotor transient is analyzed with three computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary-system pressure and temperature transients. The FACTRAN computer code is then used to calculate the thermal behavior of the fuel located at the core hot spot based on the nuclear power and RCS flow from RETRAN. The FACTRAN computer code includes a film boiling heat transfer coefficient. Finally, the VIPRE code is used to calculate the rods-in-DNB using the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN.

For the case analyzed to determine the maximum RCS pressure and peak cladding temperature, the plant is assumed to be in operation under the most adverse steady-state operating conditions; that is, a maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. The case analyzed to determine the rods-in-DNB utilizes the RTDP methodology. Initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values. Minimum measured flow is also assumed.

A maximum, uniform, SGTP level of 10 percent was assumed in the RETRAN analysis. However, a core flow reduction of 1.1 percent, which addresses the potential reactor coolant flow asymmetry associated with a maximum loop-to-loop SGTP imbalance of 10 percent, was applied.

A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive MTC limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the peak RCS pressures and hot-spot results are reached.

A conservatively low trip reactivity value (3.5-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (1.8 seconds from dashpot). The trip reactivity versus rod position curve is confirmed to be valid as part of the RSAC verification process.

A loss-of-offsite-power is assumed with the unaffected RCP losing power instantaneously at reactor trip.

For the peak RCS pressure evaluation, the initial pressure is conservatively estimated as 50.1 psi above the nominal pressure (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 peak RCS pressure occurs in the lower plenum of the vessel. The pressure transient in the lower plenum is shown in Figure 5.1.8-30.

For this accident, an evaluation of the consequences with respect to the fuel rod thermal transient is performed. The evaluation incorporates the assumption of rods going into DNB as a conservative initial condition to determine the cladding temperature and zirconium water reaction resulting from the locked rotor. Results obtained from the analysis of this hot-spot condition represent the upper limit with respect to cladding temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.5 times the average rod power (that is, FQ = 2.5) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature. The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and RCS flow rate as a function of time are based on the RETRAN results.

Fuel Cladding Gap Coefficient

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

Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above 1800°F (cladding temperature). The Baker-Just parabolic rate equation is used to define the rate of zirconium-steam reaction. The effect of the zirconium-steam reaction is included in the calculation of the hot-spot cladding temperature transient.

Results

Figures 5.1.8-25 through 5.1.8-33 illustrate the transient response for the locked-rotor event (peak RCS pressure/peak cladding temperature case). The peak RCS pressure is 2683 psia and is less than the acceptance criterion of 2750 psia. Also, the peak cladding temperature is 1900°F, which is considerably less than the limit of 2700°F. The zirconium-steam reaction at the hotspot is 0.61 percent by weight, which meets the criterion of less than 16-percent zirconium-steam water reaction. For the radiological dose evaluation, the total percentage of fuel rods calculated to experience DNB is less than 50 percent (rods-in-DNB case). The sequence of events for the peak RCS pressure/peak cladding temperature case is given in Table 5.1.8-3. This transient trips on a low primary reactor coolant flow trip setpoint, which is assumed to be 86.5 percent.

Conclusions

The analysis performed has demonstrated that for the locked-rotor event, the RCS pressure remains below 110 percent of the design pressure and the hot-spot cladding temperature and oxidation levels remain below the limit values. Therefore, all applicable acceptance criteria are met. In addition, the total percentage of rods calculated to experience DNB is less than 50 percent.

Table 5.1.8-1 Sequence of Events – Partial Loss of Reactor Coolant Flow	
Event	**Time (seconds)**
One Operating RCP Loses Power and Begins Coasting Down	0.0
Low Flow Reactor Trip Setpoint is Reached	1.62
Rods Begin to Drop	2.37
Minimum DNBR Occurs	3.50

Table 5.1.8-2 Sequence of Events – Complete Loss of Reactor Coolant Flow	
Complete Loss of Flow	
Event	**Time (seconds)**
All Operating RCPs Lose Power and Coastdown Begins	0.0
Low Flow Reactor Trip Setpoint is Reached	1.82
Rods Begin to Drop	2.57
Minimum DNBR Occurs	4.00
Complete Loss of Flow - Underfrequency	
Event	**Time (seconds)**
Frequency Decay Begins and All Operating RCPs Begin to Decelerate	0.0
Low Flow Reactor Trip Setpoint is Reached	1.88
Rods Begin to Drop	2.63
Minimum DNBR Occurs	4.15

Table 5.1.8-3 Sequence of Events – Reactor Coolant Pump Locked Rotor	
Event	**Time (seconds)**
Rotor on One Pump Locks	0.00
Low Flow Reactor Trip Setpoint Reached	0.05
Rods Begin to Drop	0.80
Loss-of-Offsite-Power (remaining active pump begins to coastdown)	0.80
Maximum RCS Pressure Occurs	4.50
Maximum Cladding Temperature Occurs	5.00

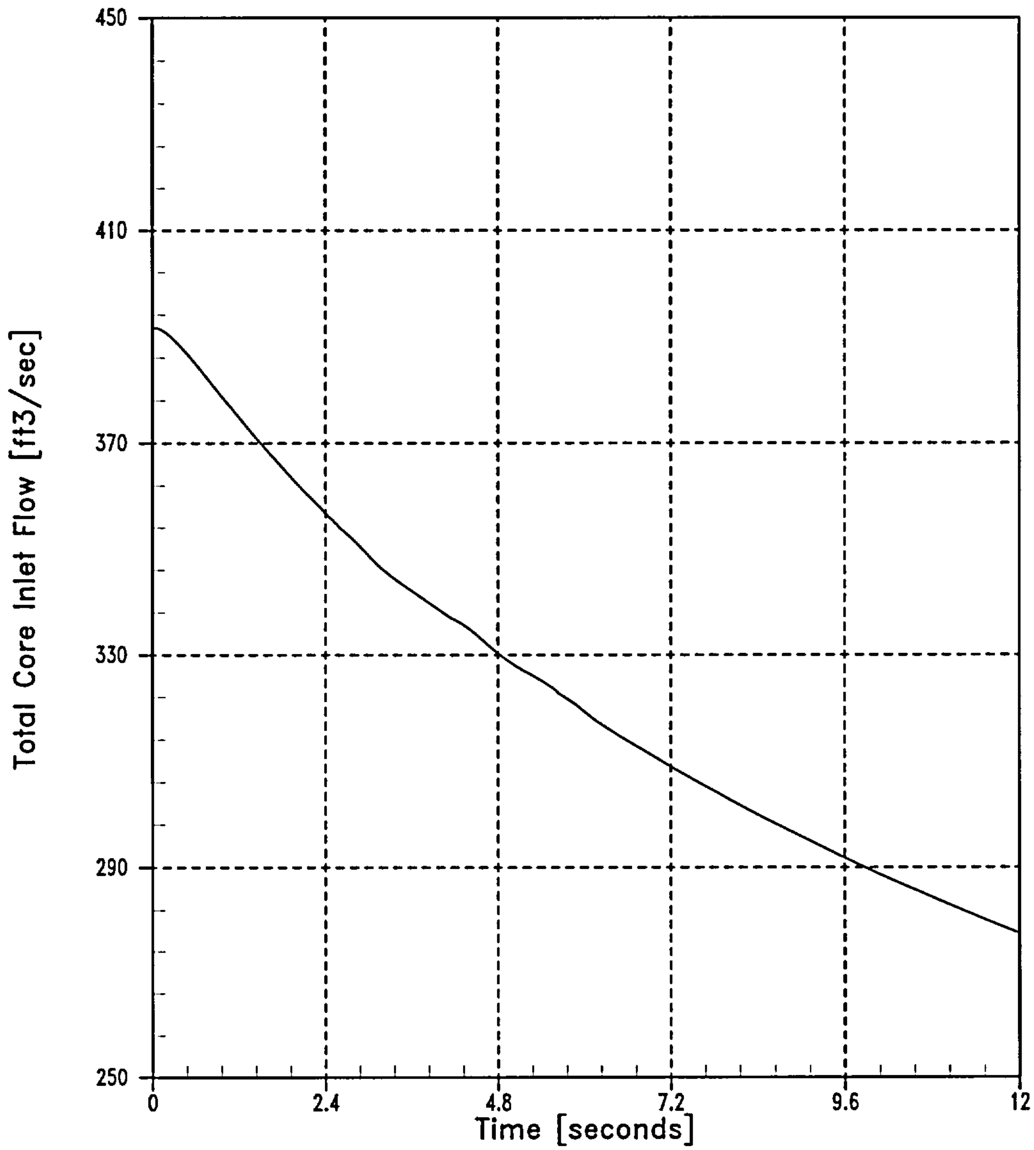

Figure 5.1.8-1 Total Core Inlet Flow versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

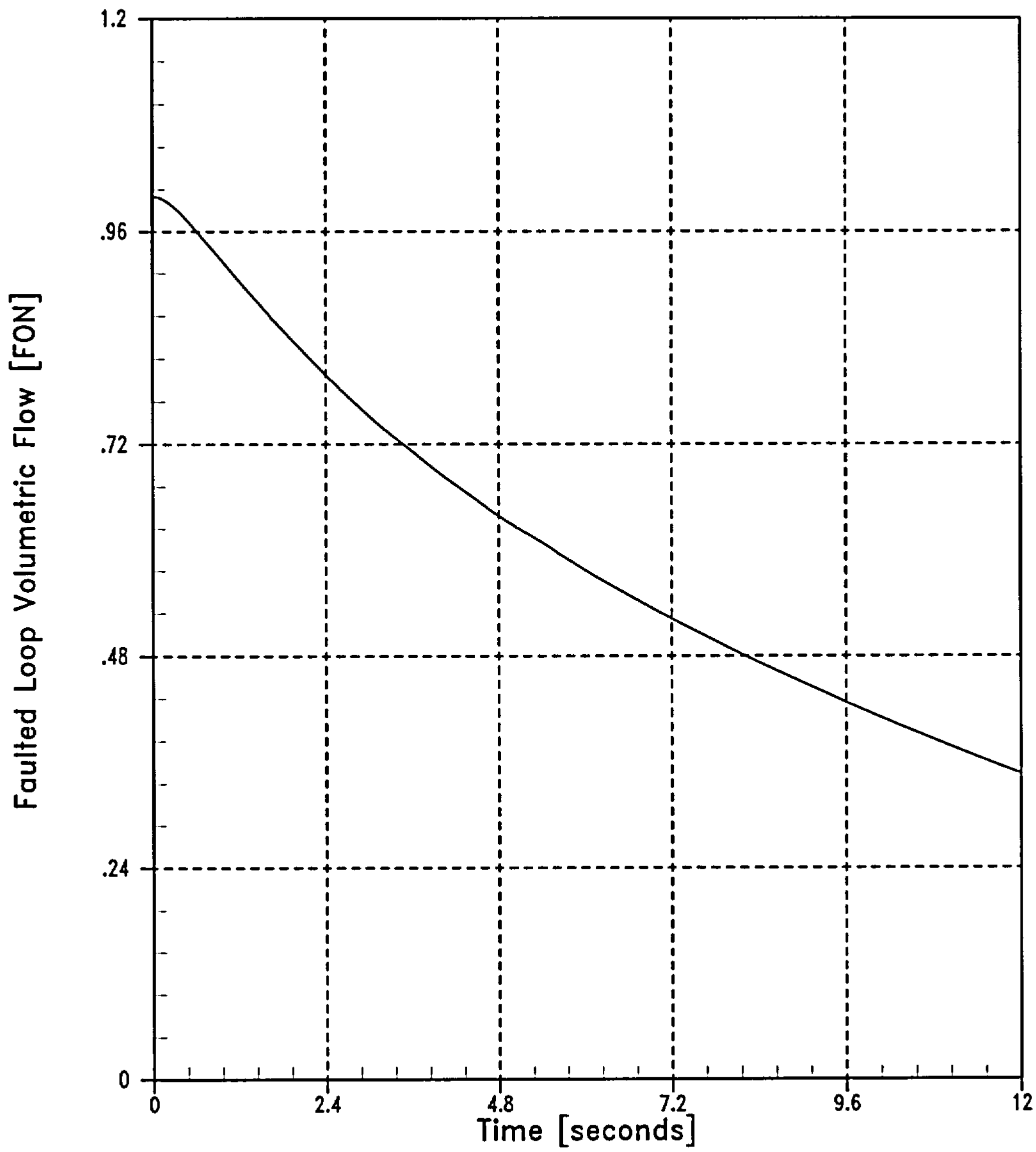

Figure 5.1.8-2 RCS Faulted Loop Flow versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

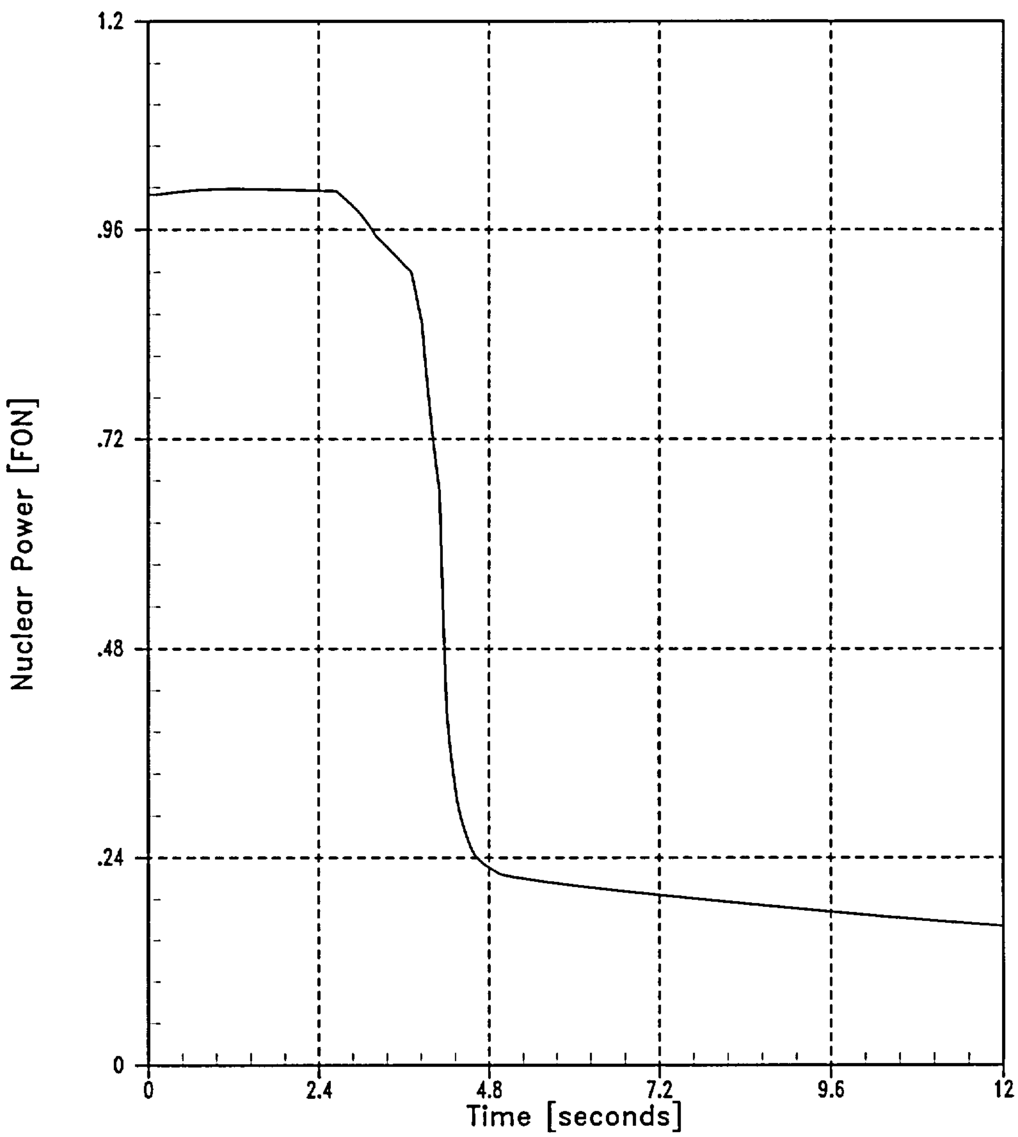

Figure 5.1.8-3 Nuclear Power versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

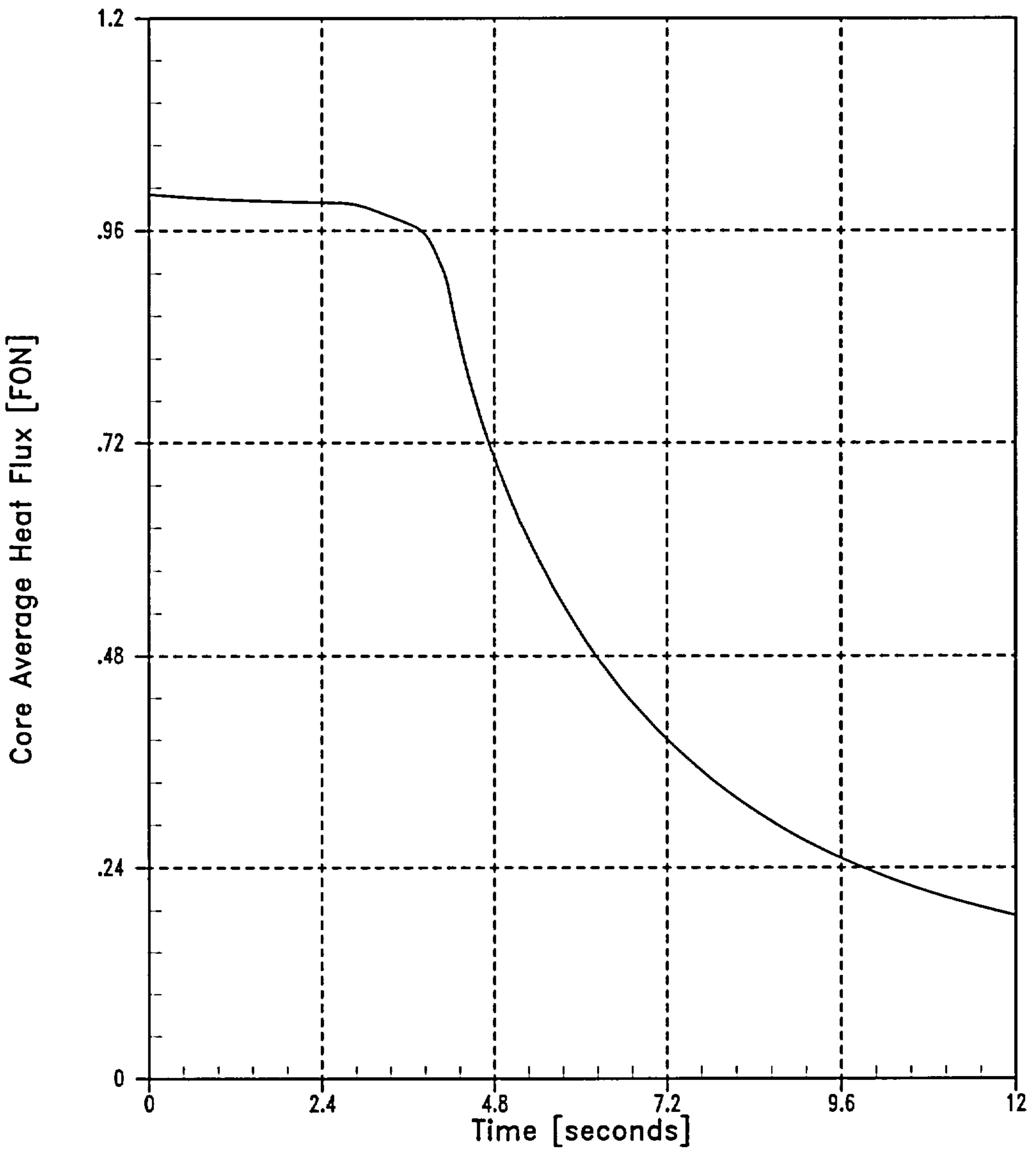

Figure 5.1.8-4 Core Average Heat Flux versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

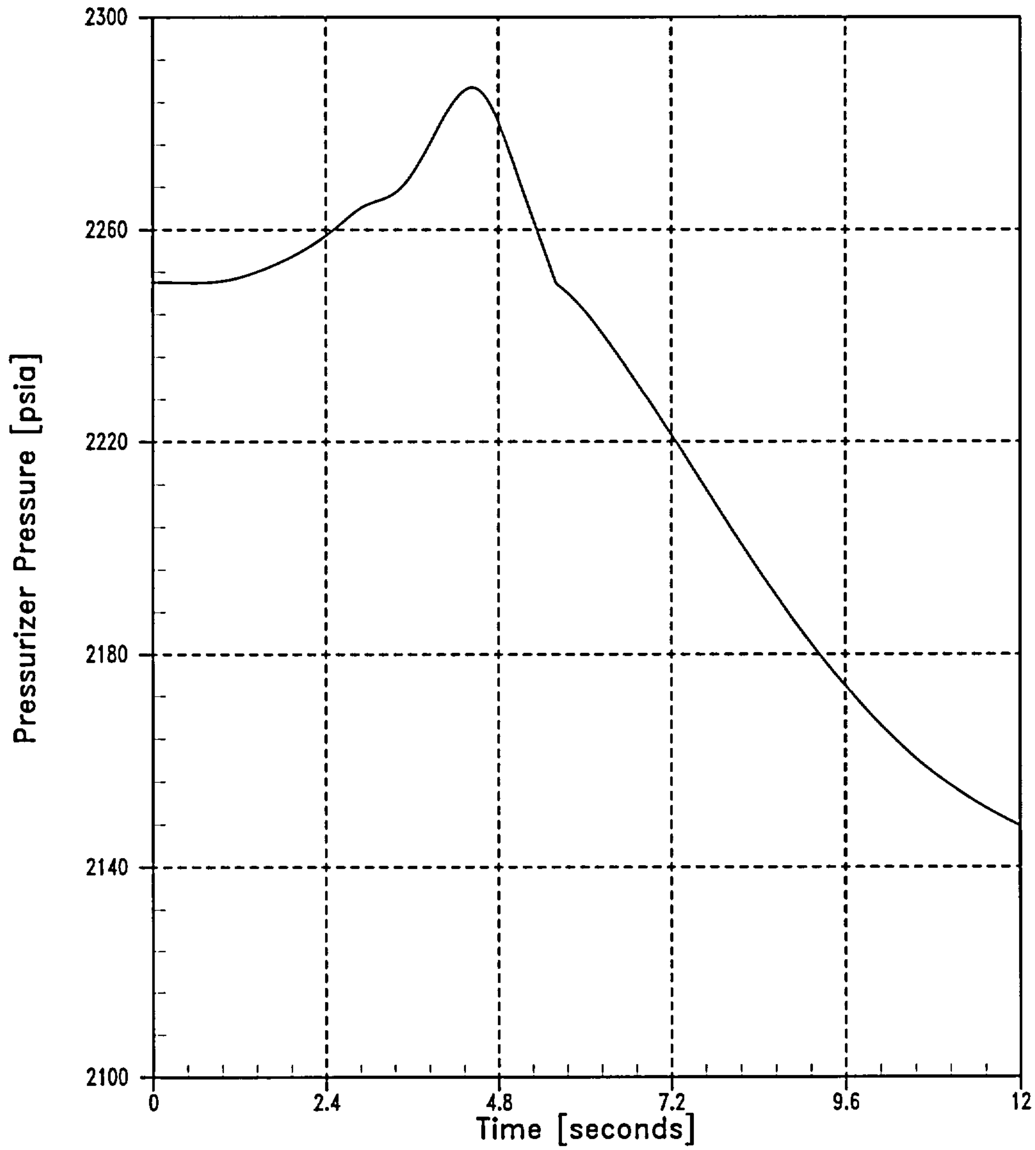

Figure 5.1.8-5 Pressurizer Pressure versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

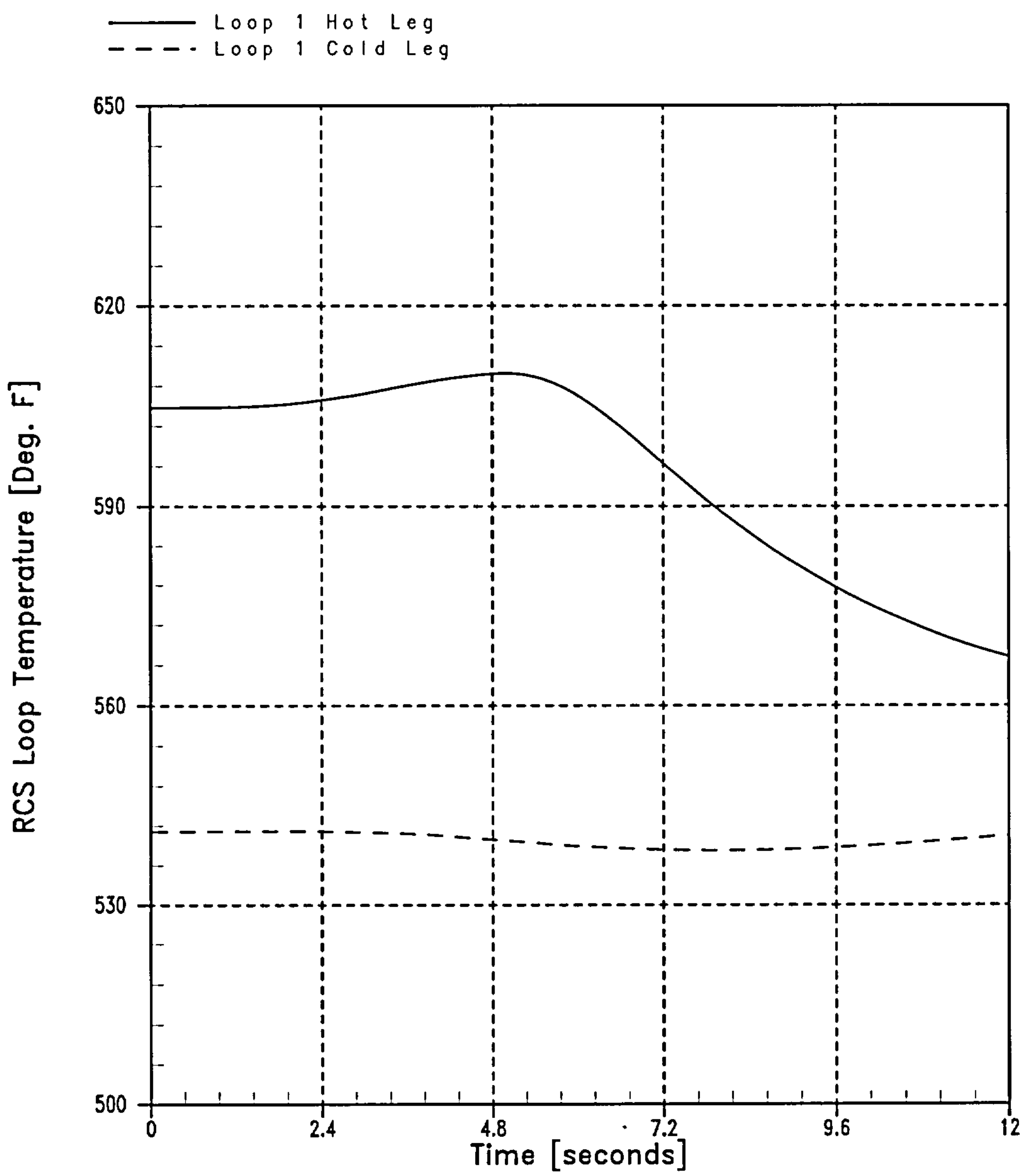

Figure 5.1.8-6 RCS Faulted Loop Temperature versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

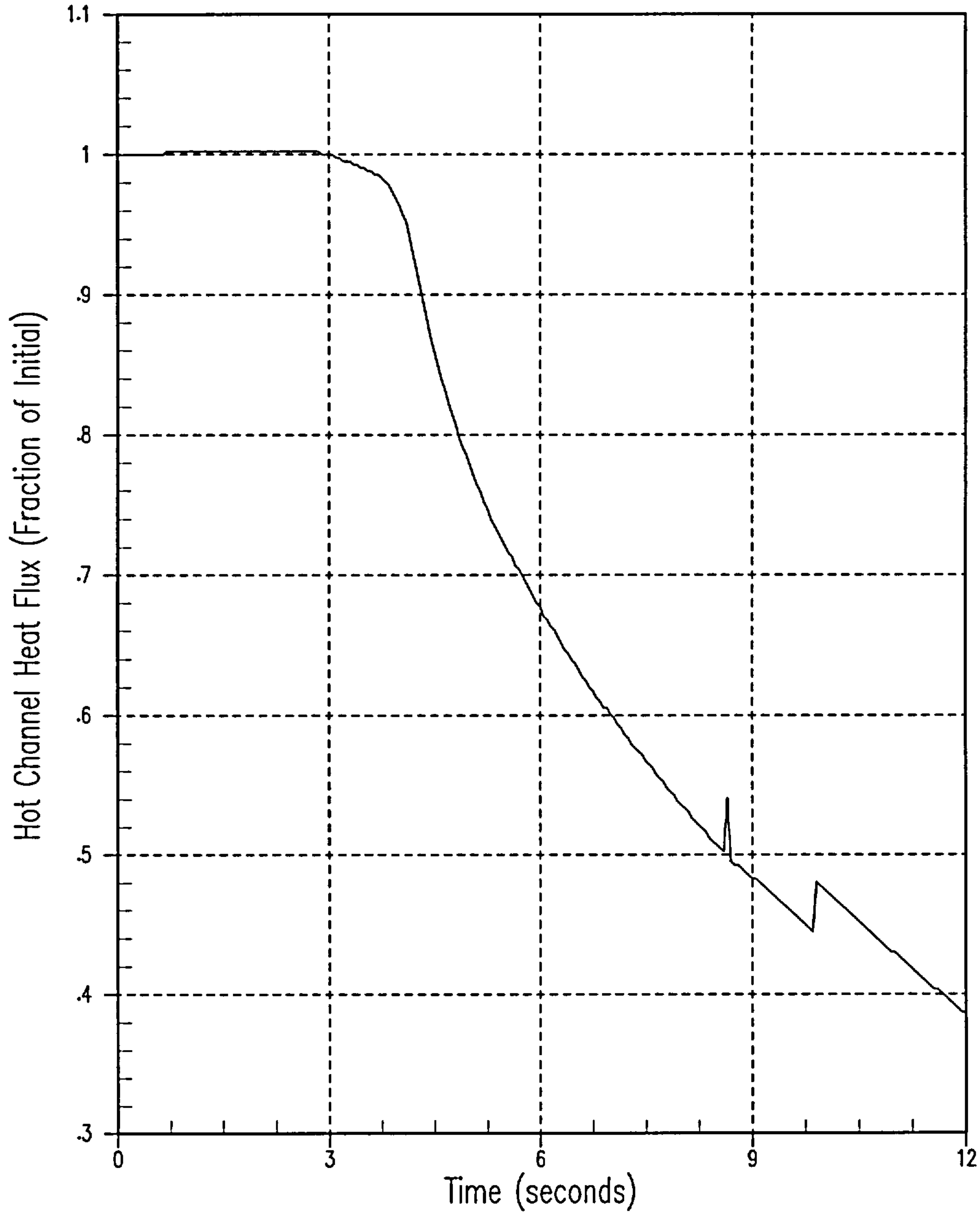

Figure 5.1.8-7 Hot Channel Heat Flux versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

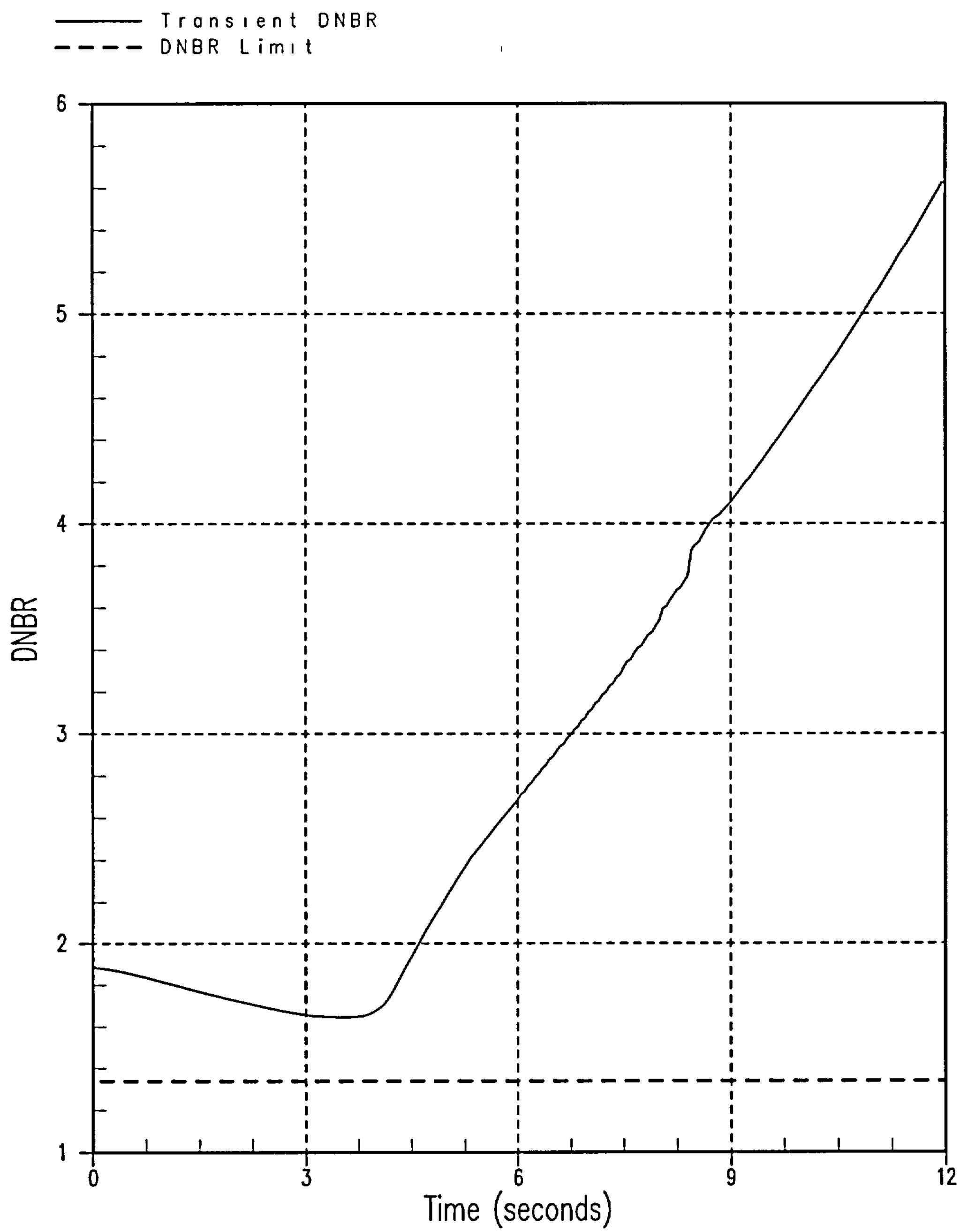

Figure 5.1.8-8 DNBR versus Time – Partial Loss of Flow, One Pump Coasting Down (PLOF)

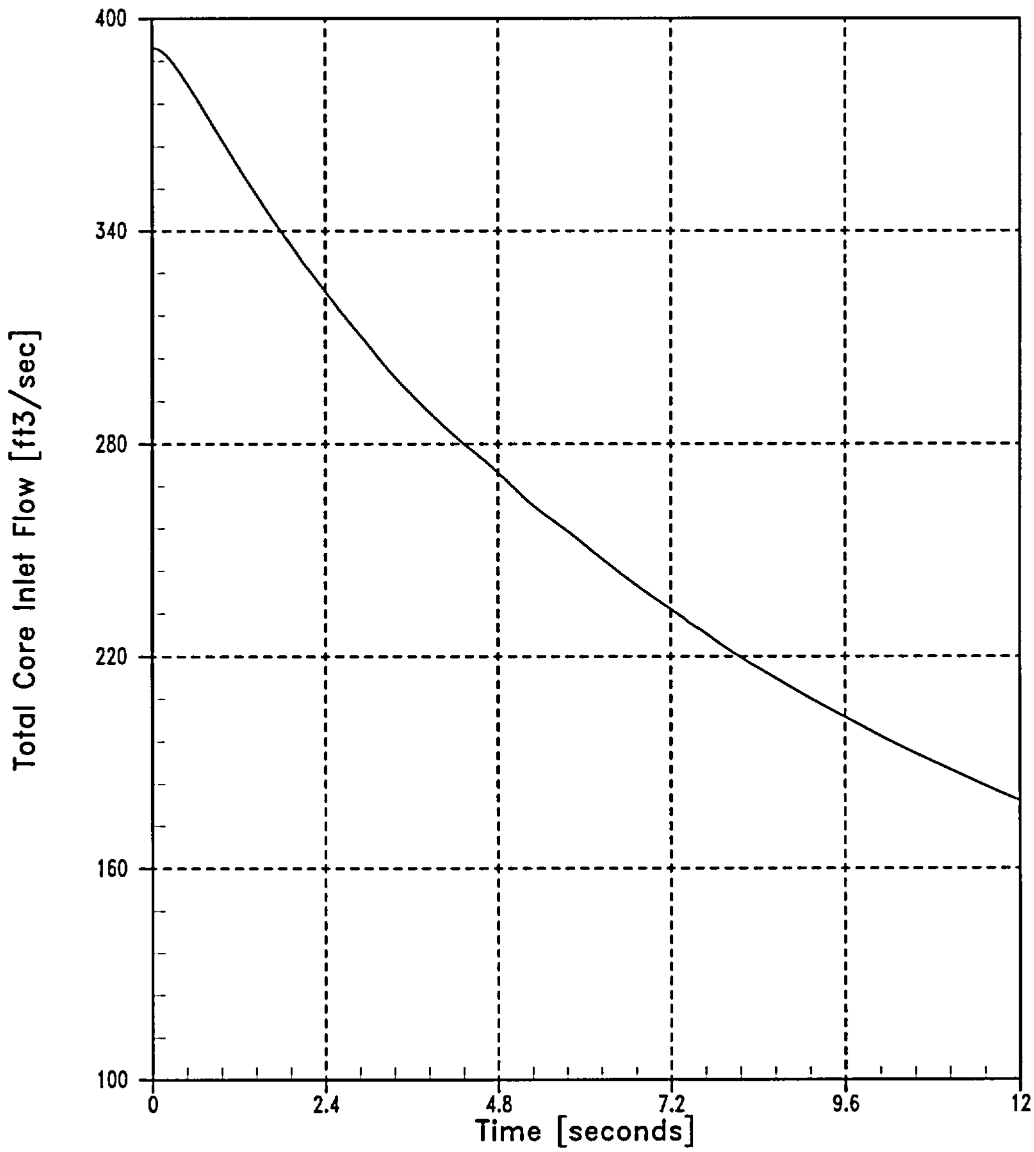

Figure 5.1.8-9 Total Core Inlet Flow versus Time – Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

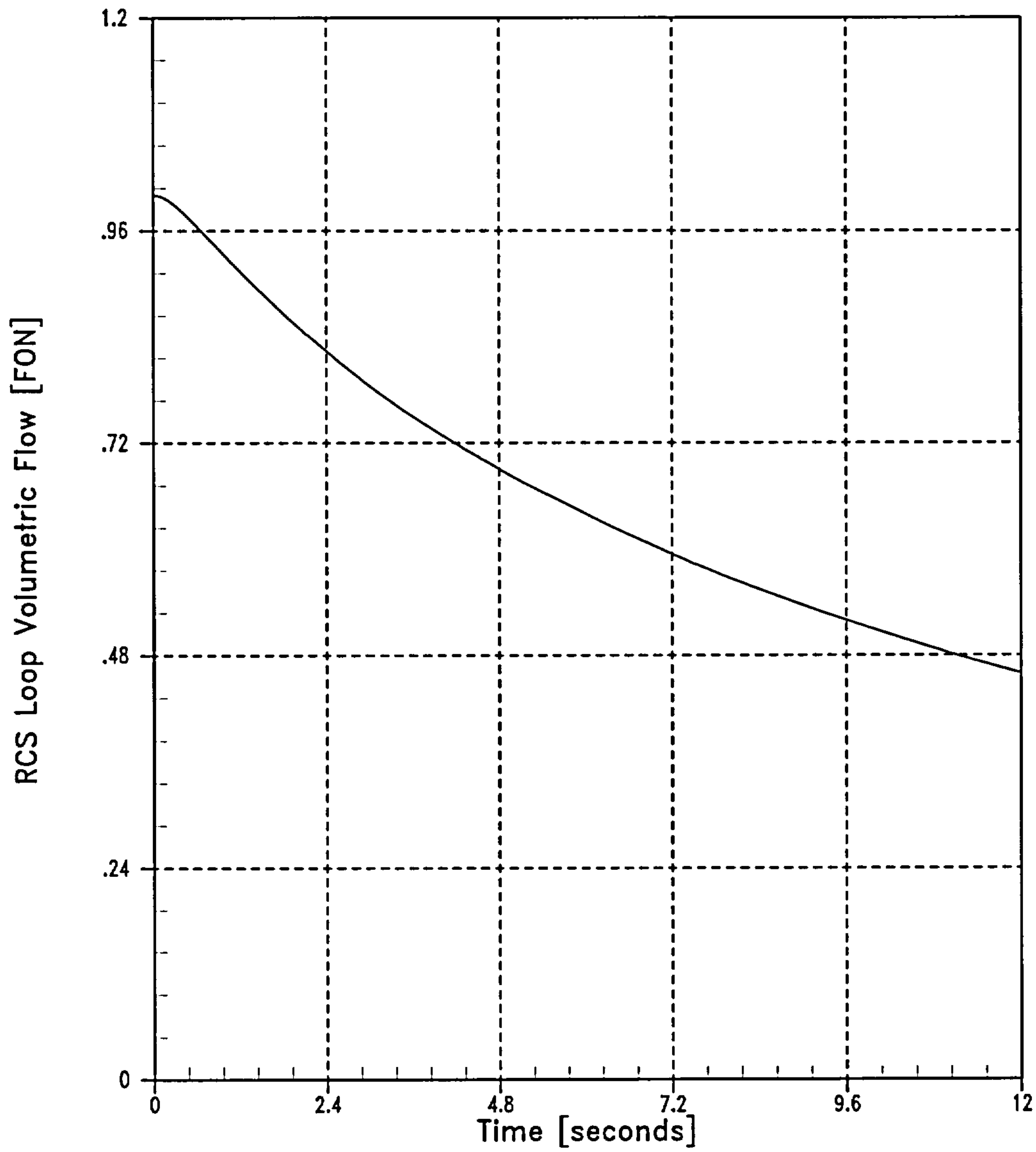

Figure 5.1.8-10 RCS Loop Flow versus Time – Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

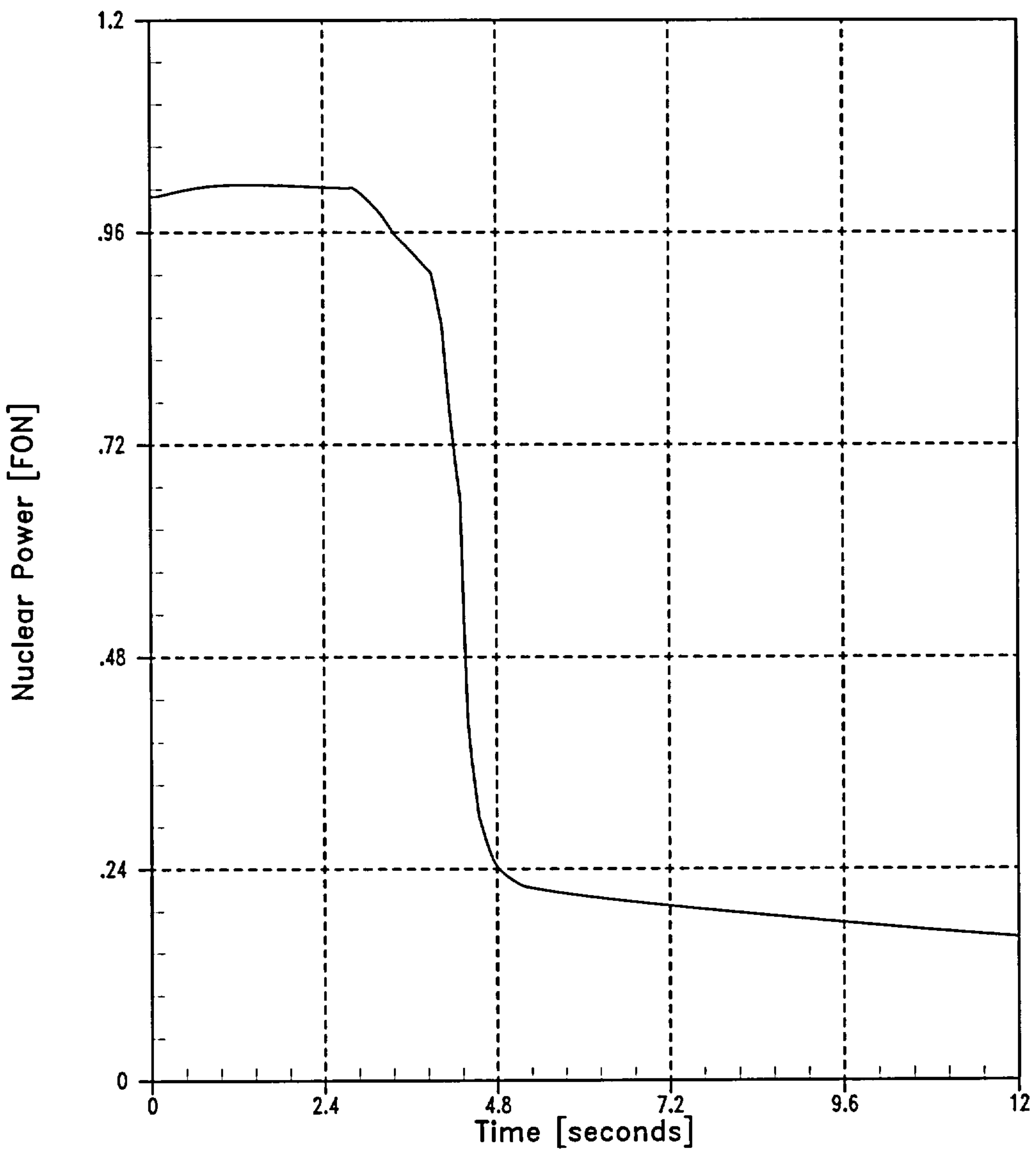

Figure 5.1.8-11 Nuclear Power versus Time – Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

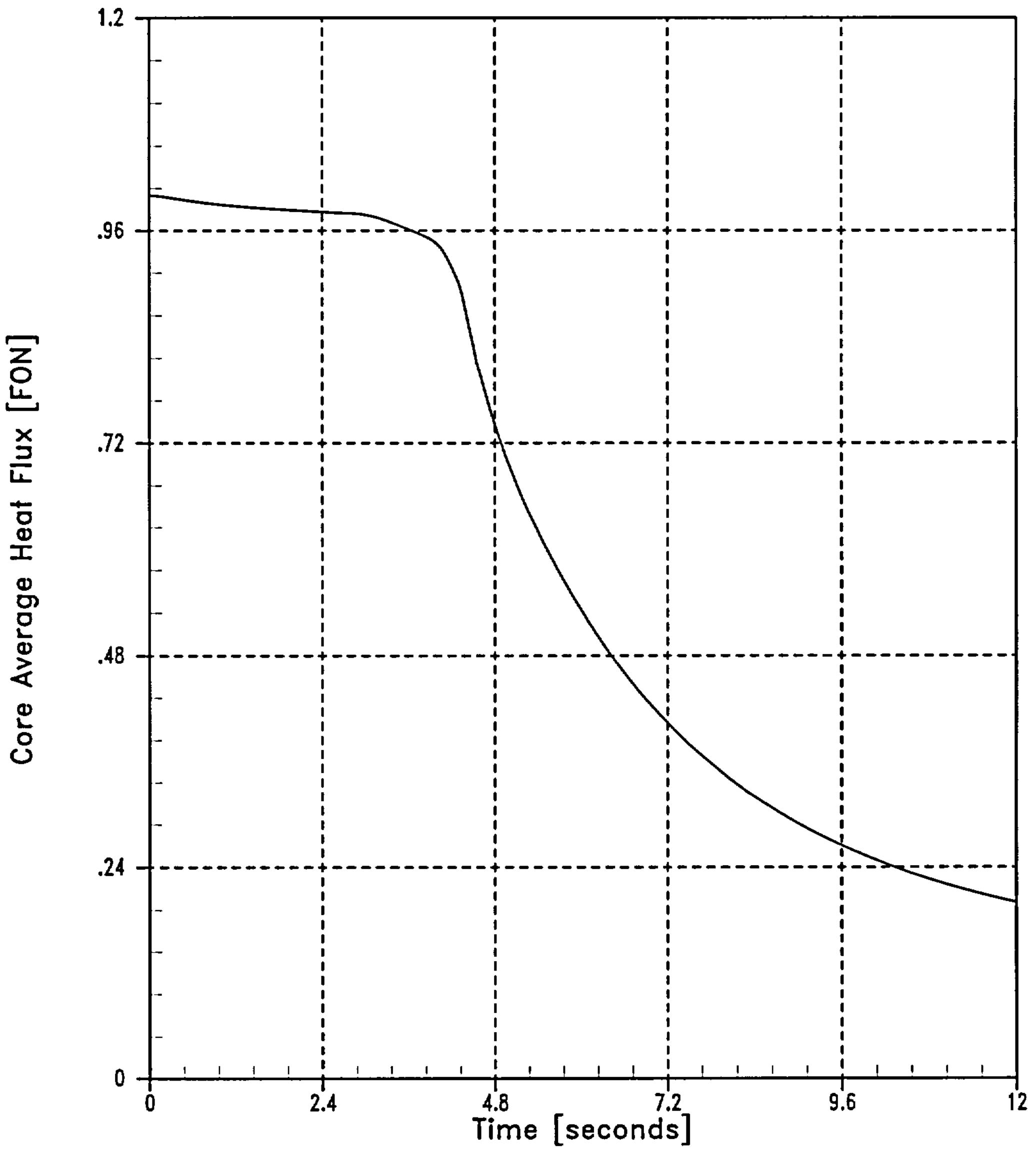

Figure 5.1.8-12 Core Average Heat Flux versus Time – Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

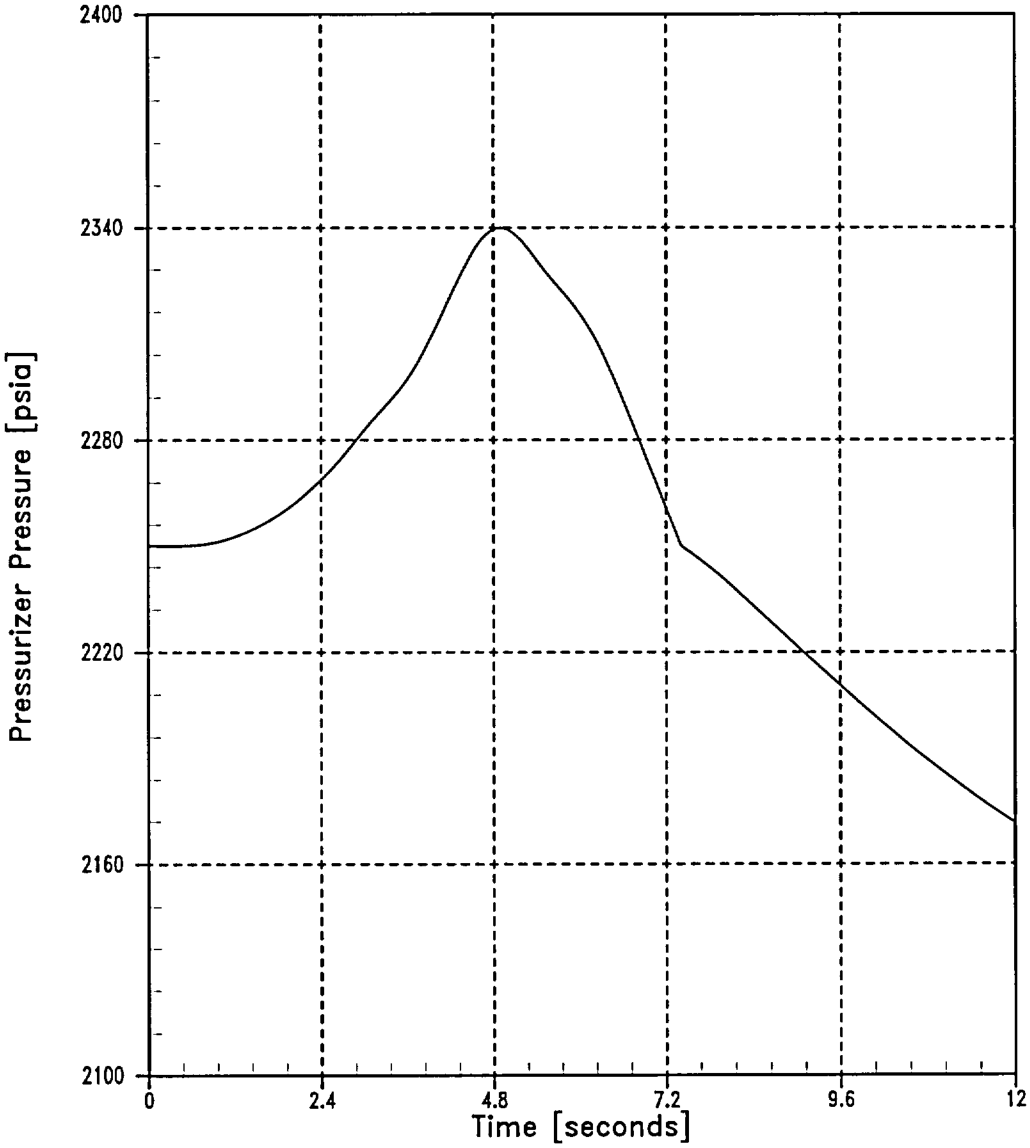

Figure 5.1.8-13 Pressurizer Pressure versus Time – Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

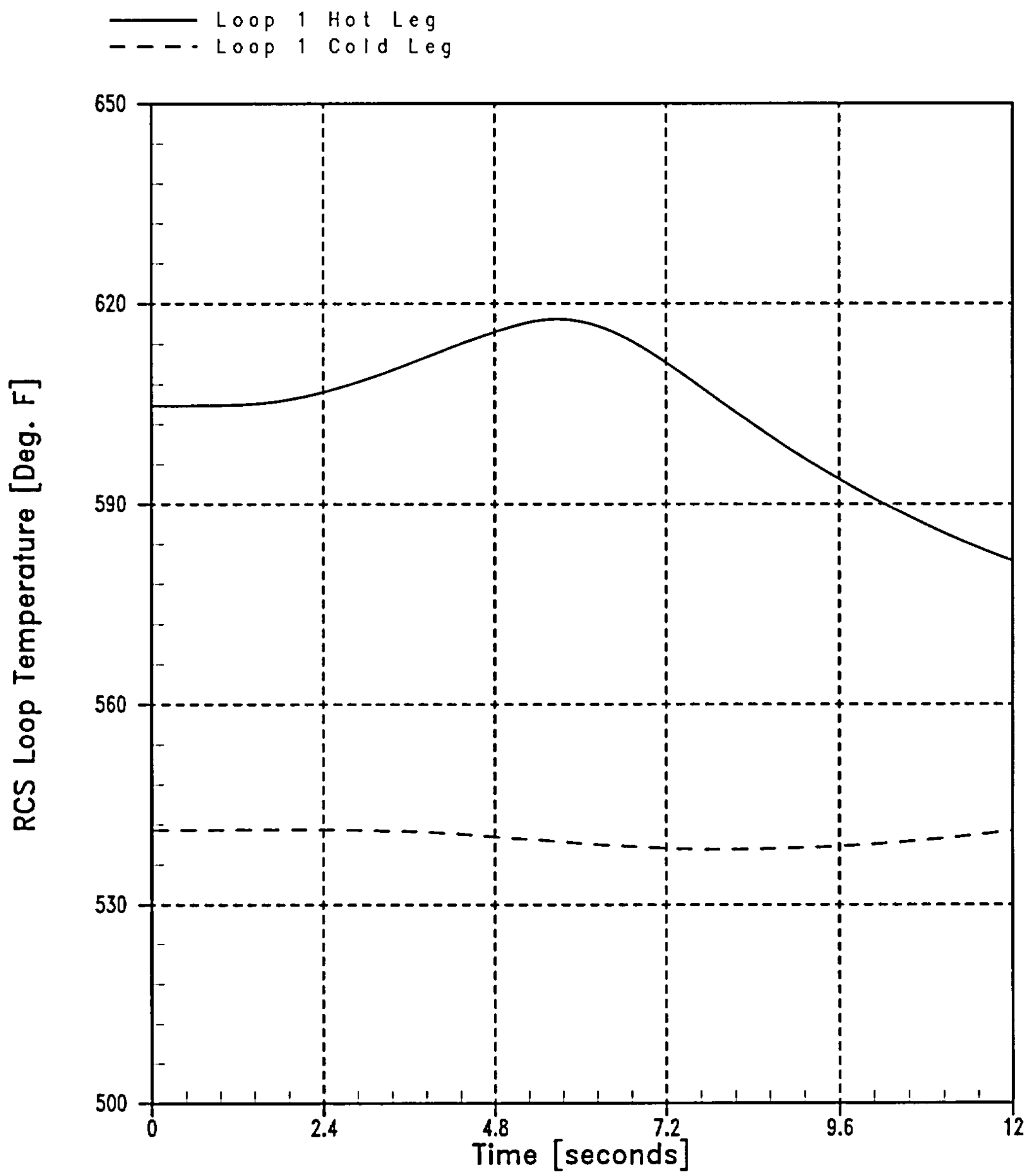

Figure 5.1.8-14 RCS Faulted Loop Temperature versus Time – Complete Loss of Flow – Two Pumps Coasting Down (CLOF)

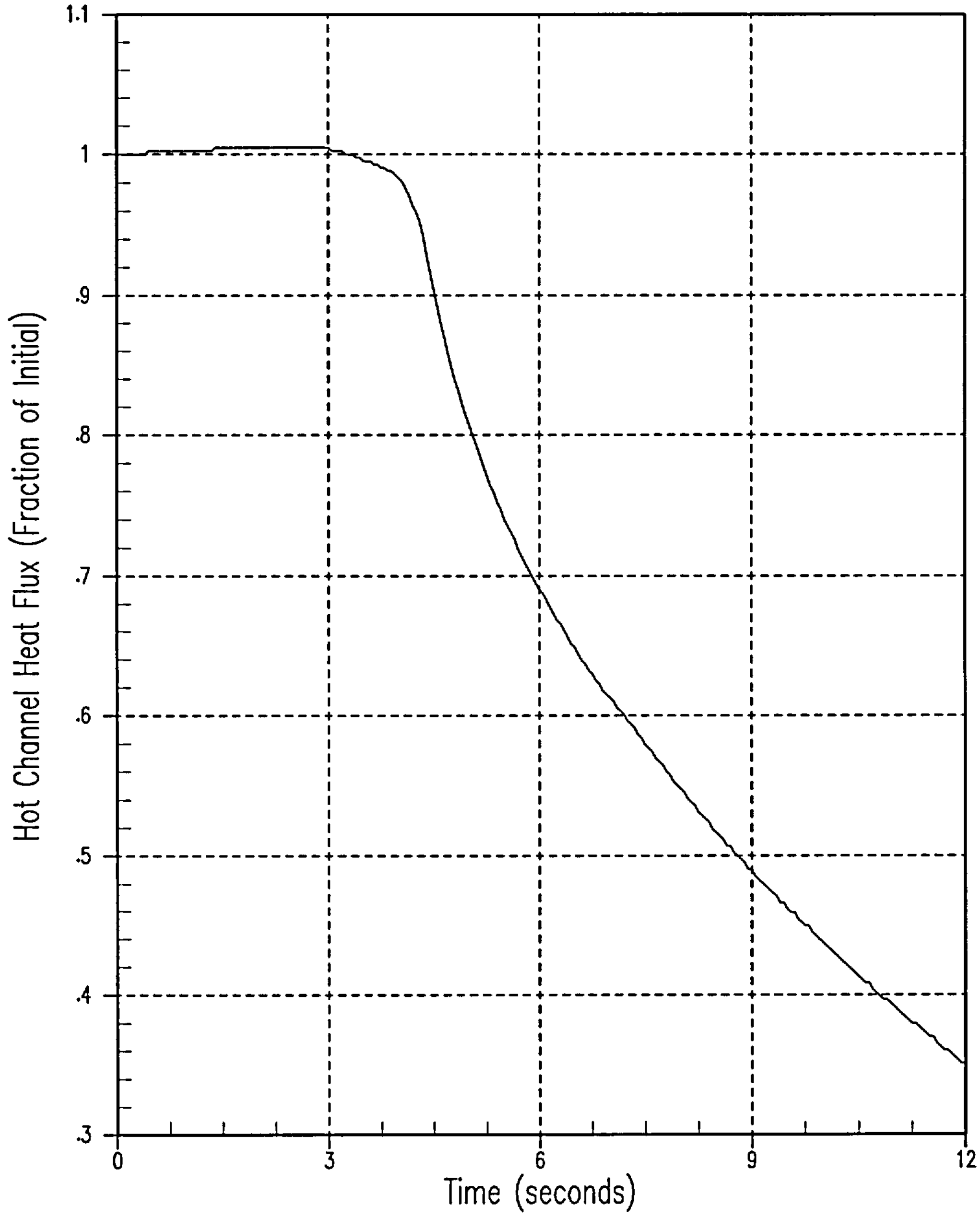

Figure 5.1.8-15 Hot Channel Heat Flux versus Time – Complete Loss of Flow, Two Pumps Coasting Down (CLOF)

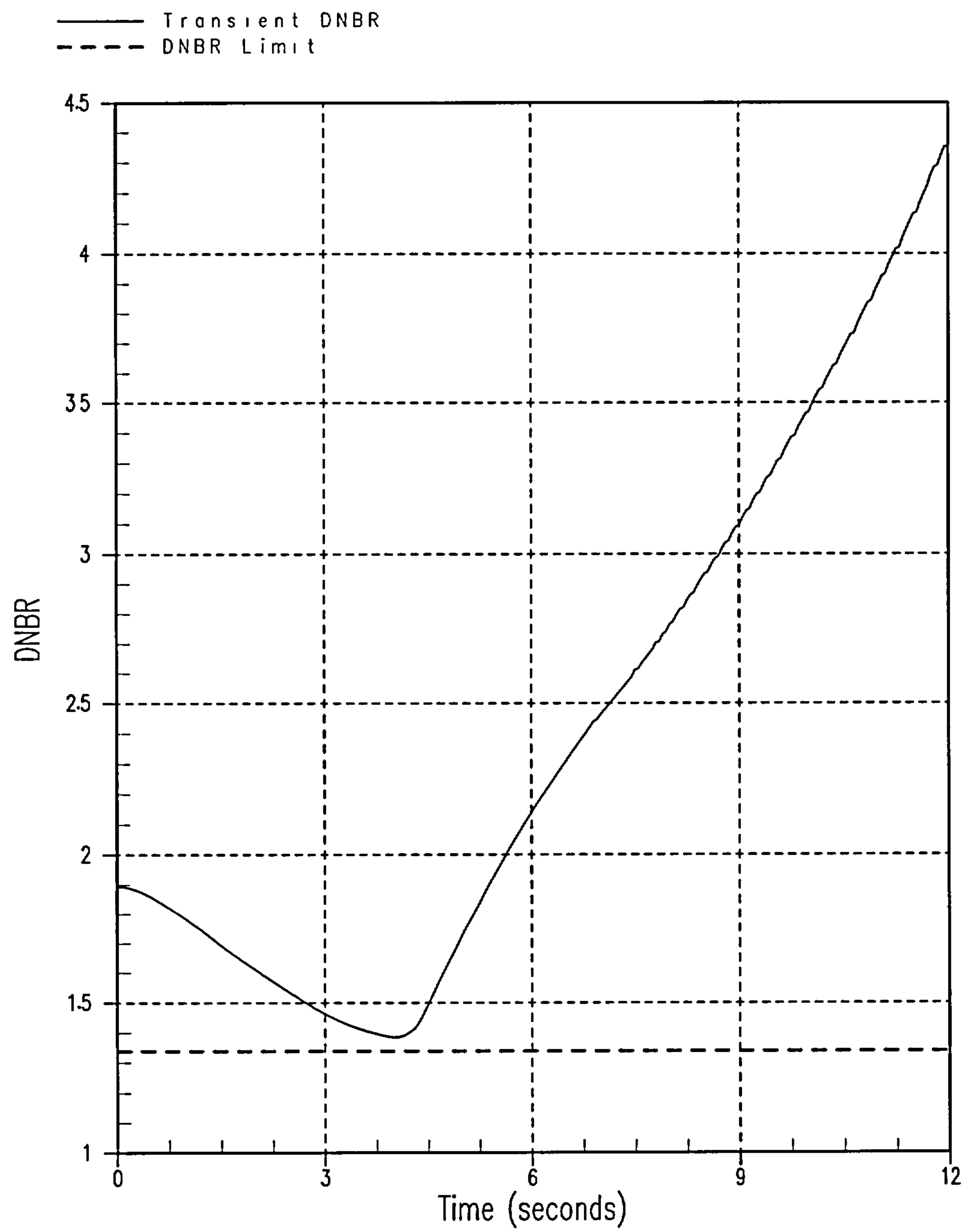

Figure 5.1.8-16 DNBR versus Time – Complete Loss of Flow, Two Pumps Coasting Down (CLOF)

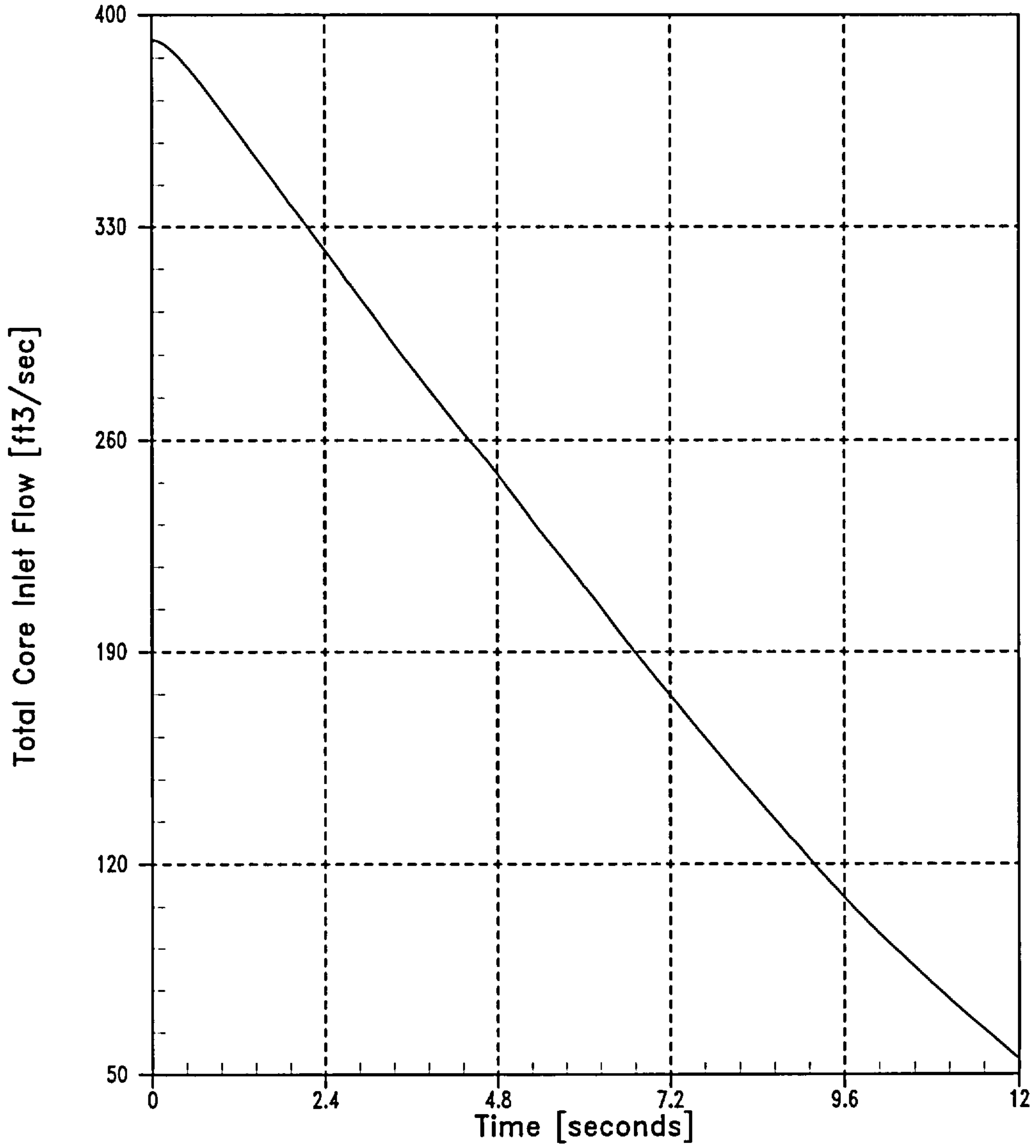

Figure 5.1.8-17 Total Core Inlet Flow versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

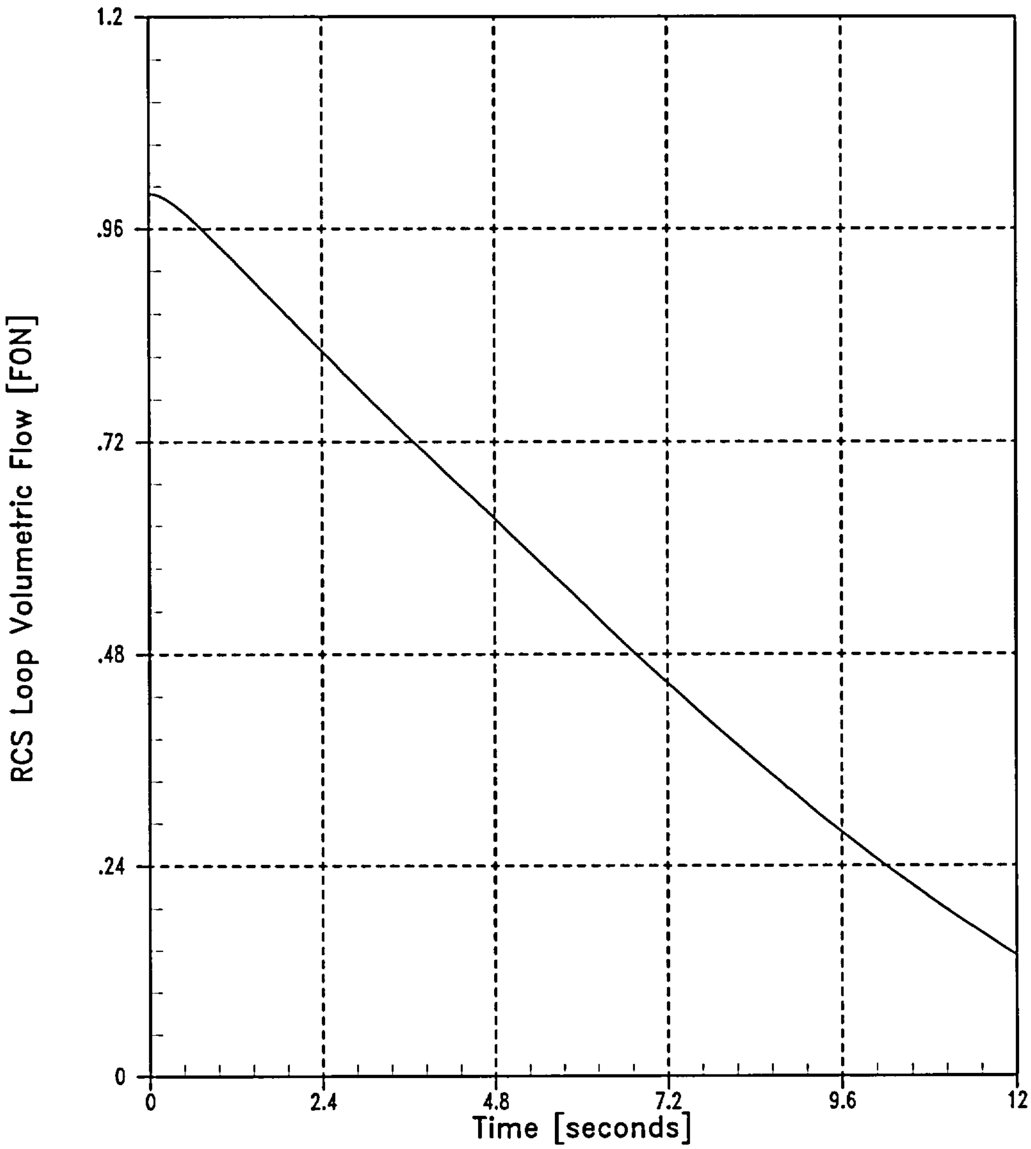

Figure 5.1.8-18 RCS Loop Flow versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

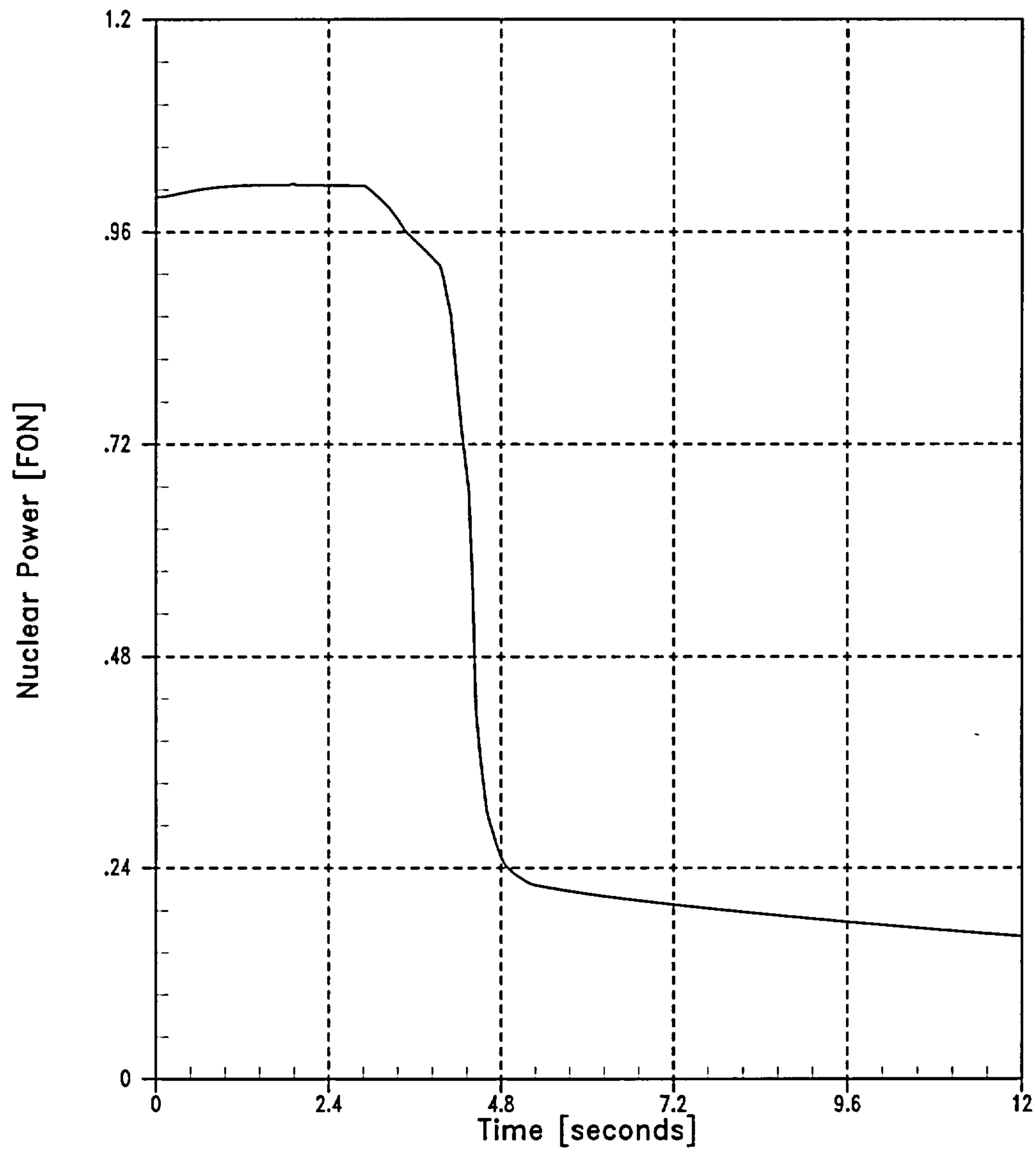

Figure 5.1.8-19 Nuclear Power versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

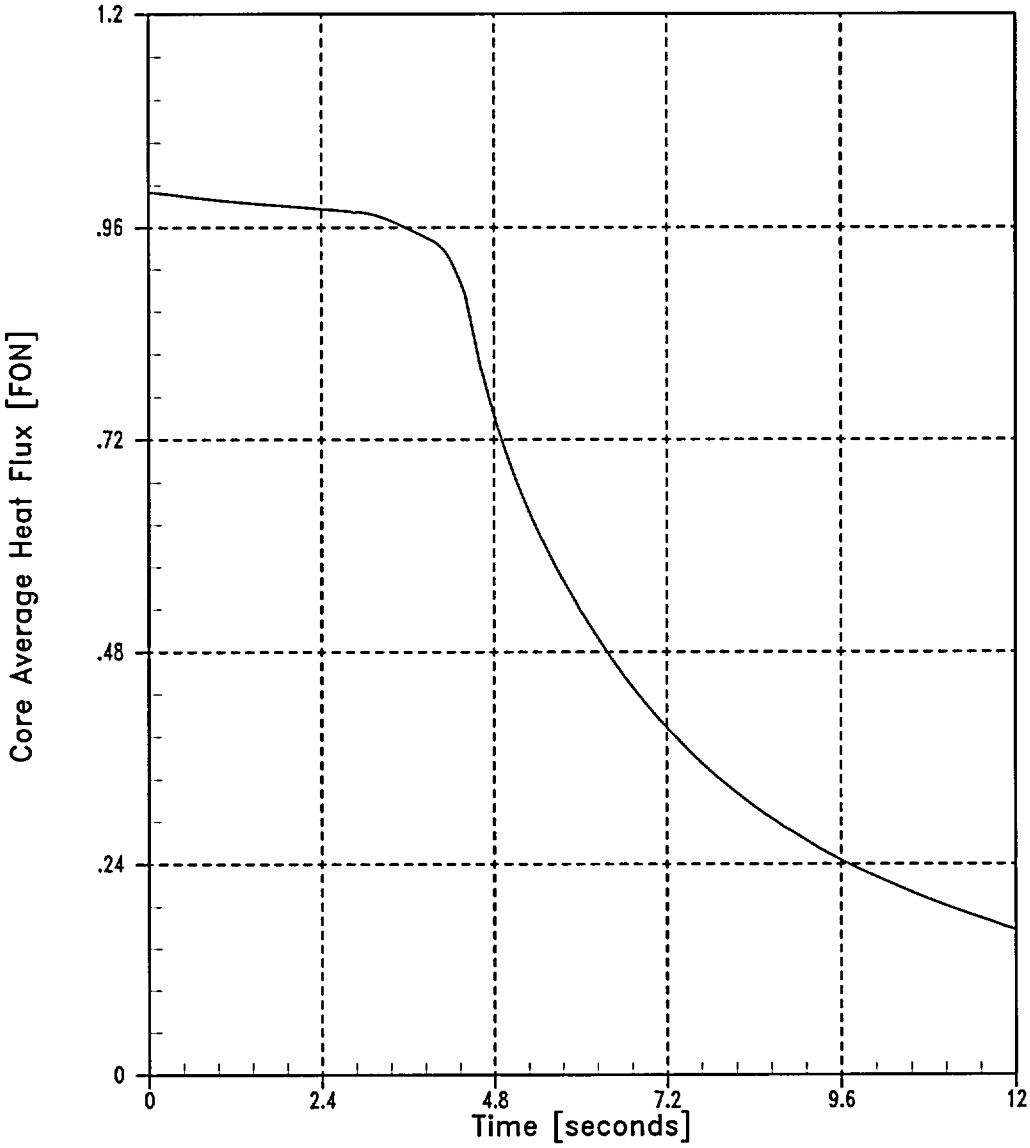

Figure 5.1.8-20 Core Average Heat Flux versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

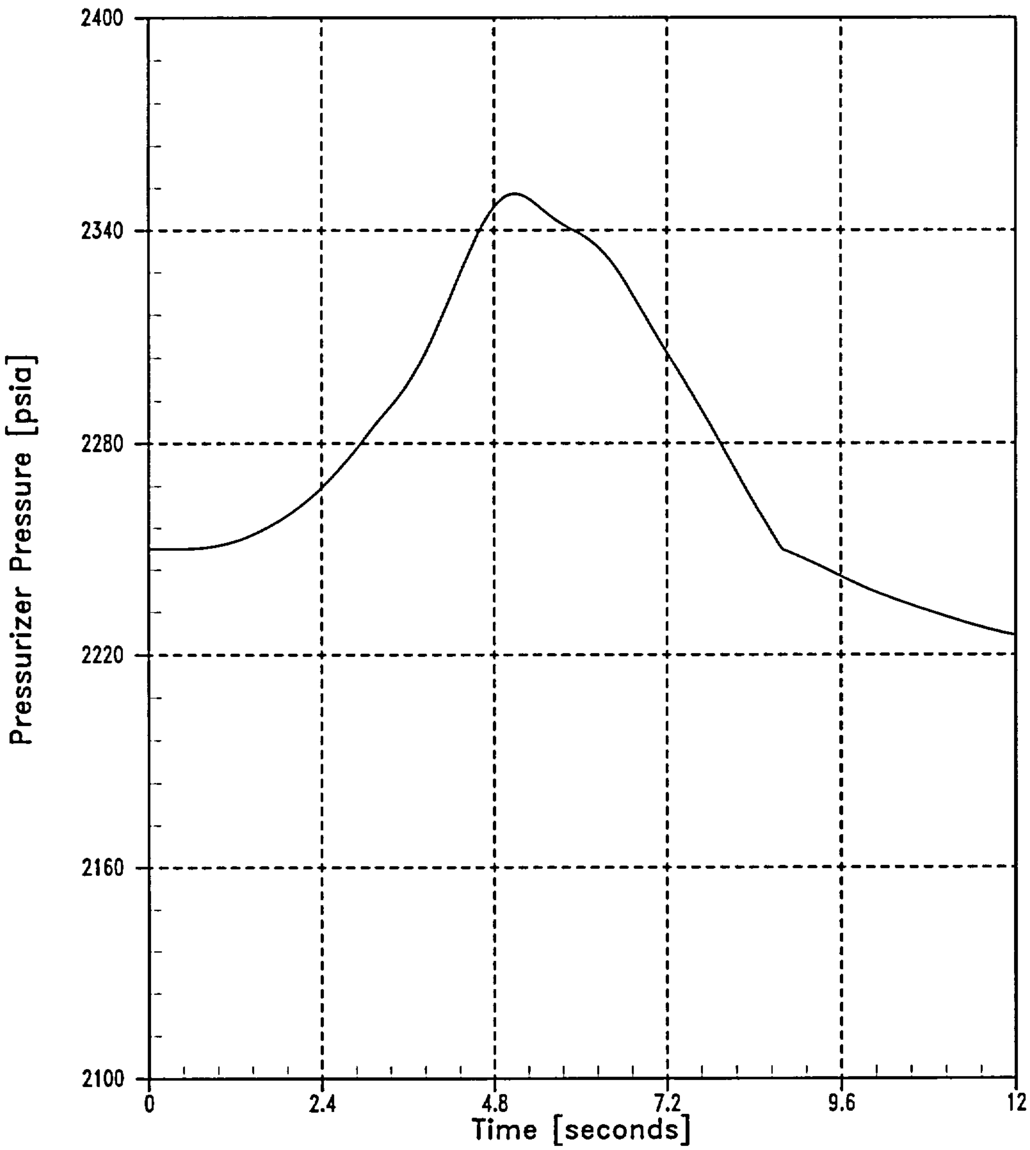

Figure 5.1.8-21 Pressurizer Pressure versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

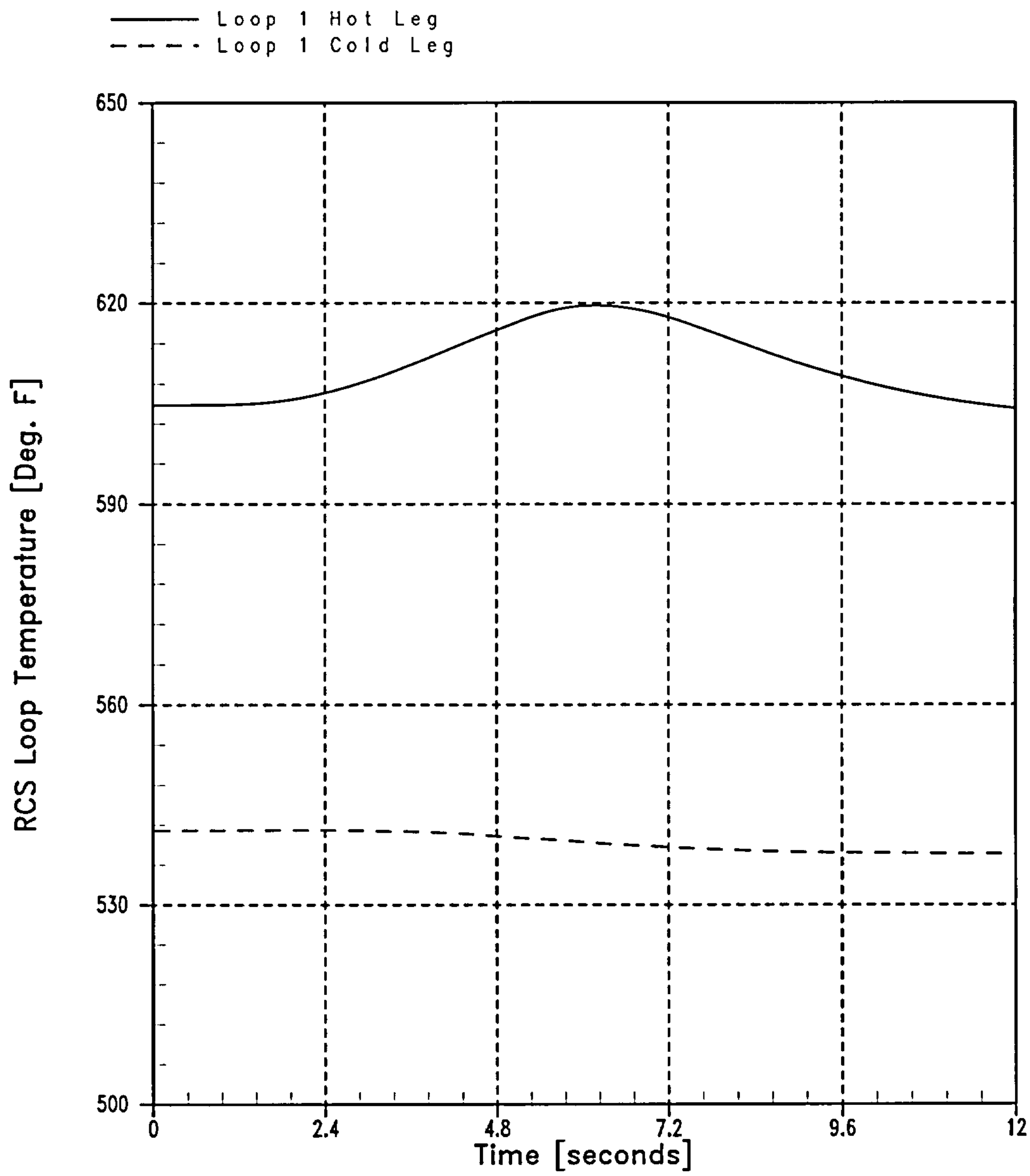

Figure 5.1.8-22 RCS Loop Temperature versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

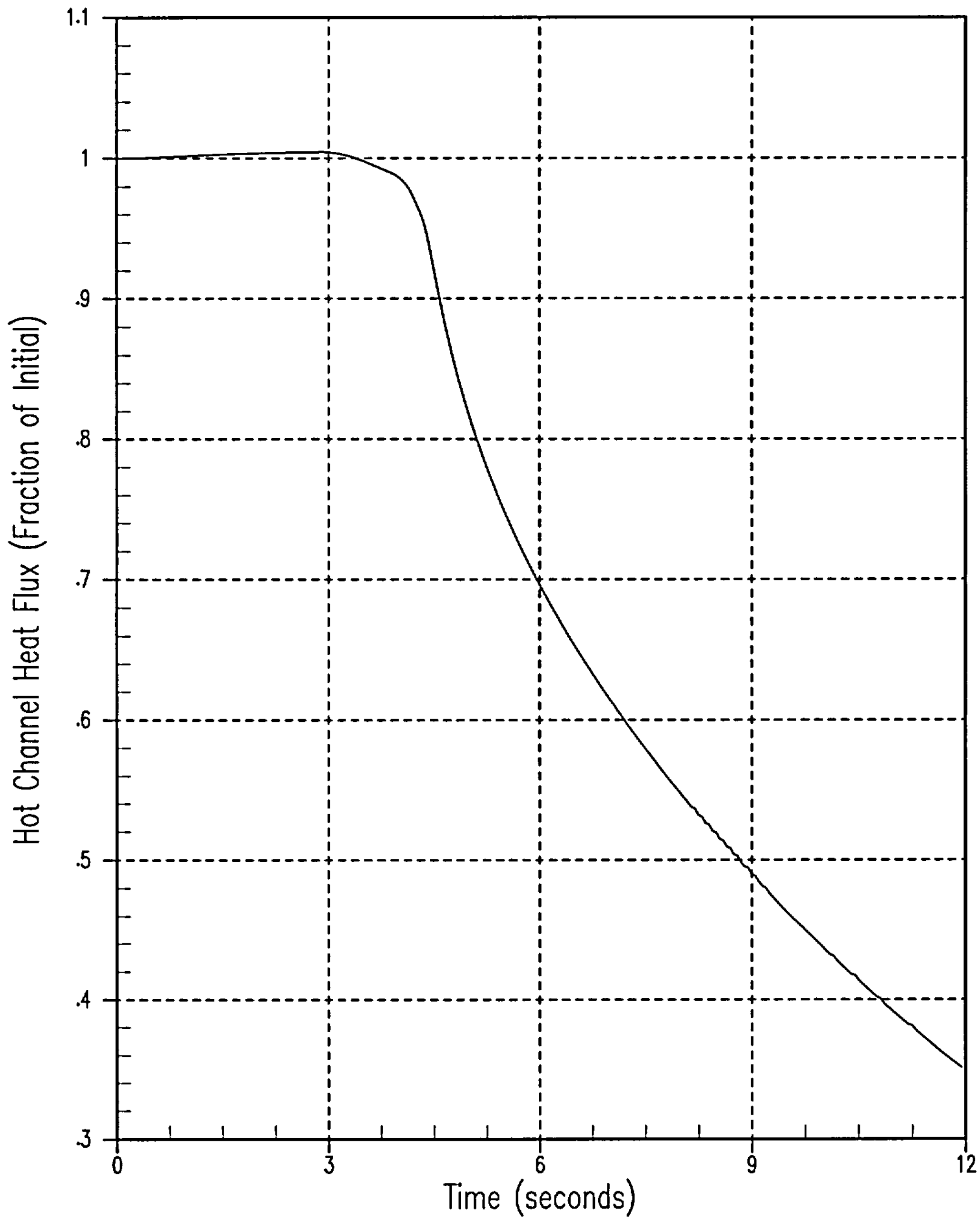

Figure 5.1.8-23 Hot Channel Heat Flux versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

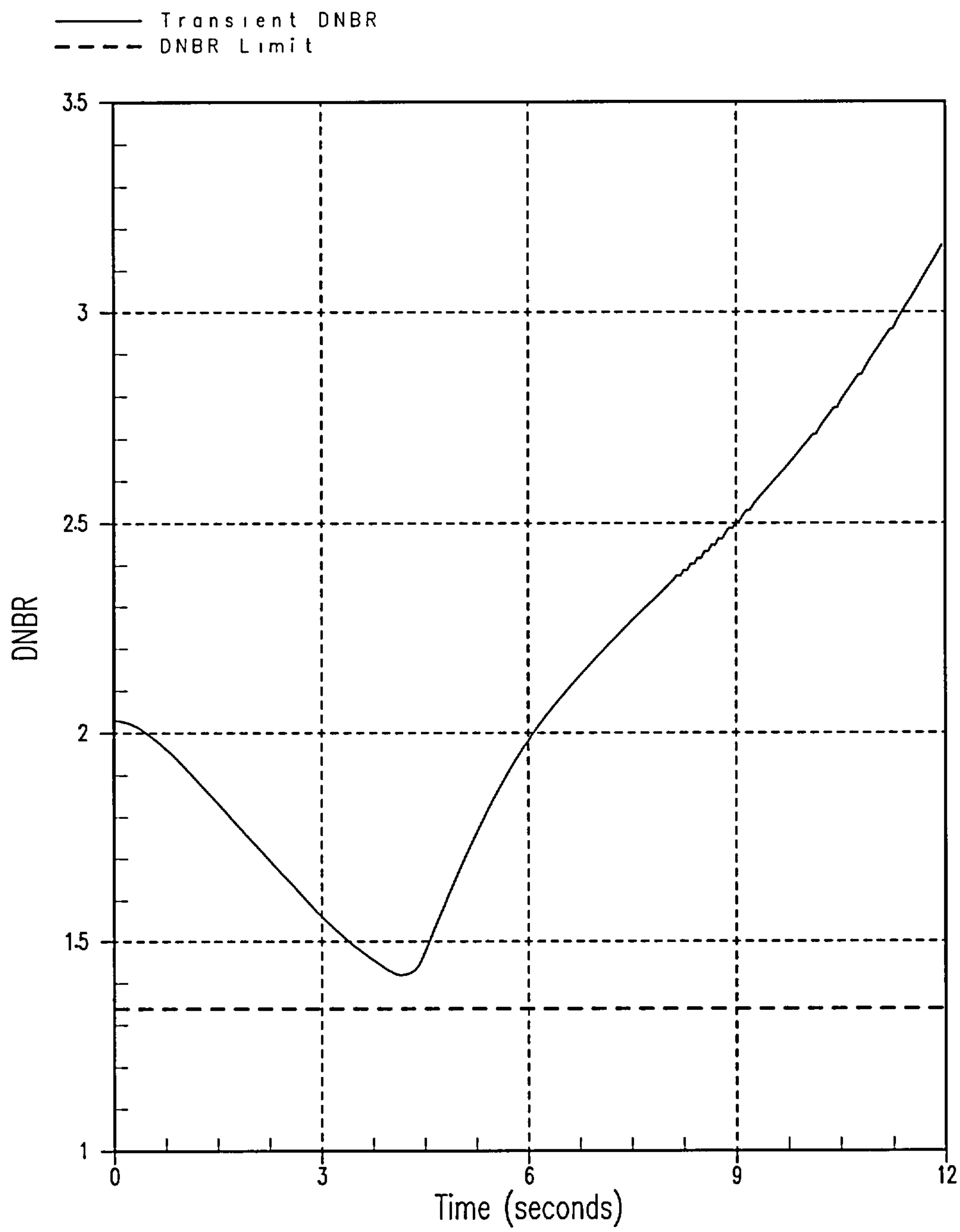

Figure 5.1.8-24 DNBR versus Time – Complete Loss of Flow – Frequency Decay in Two Pumps (CLOF-UF)

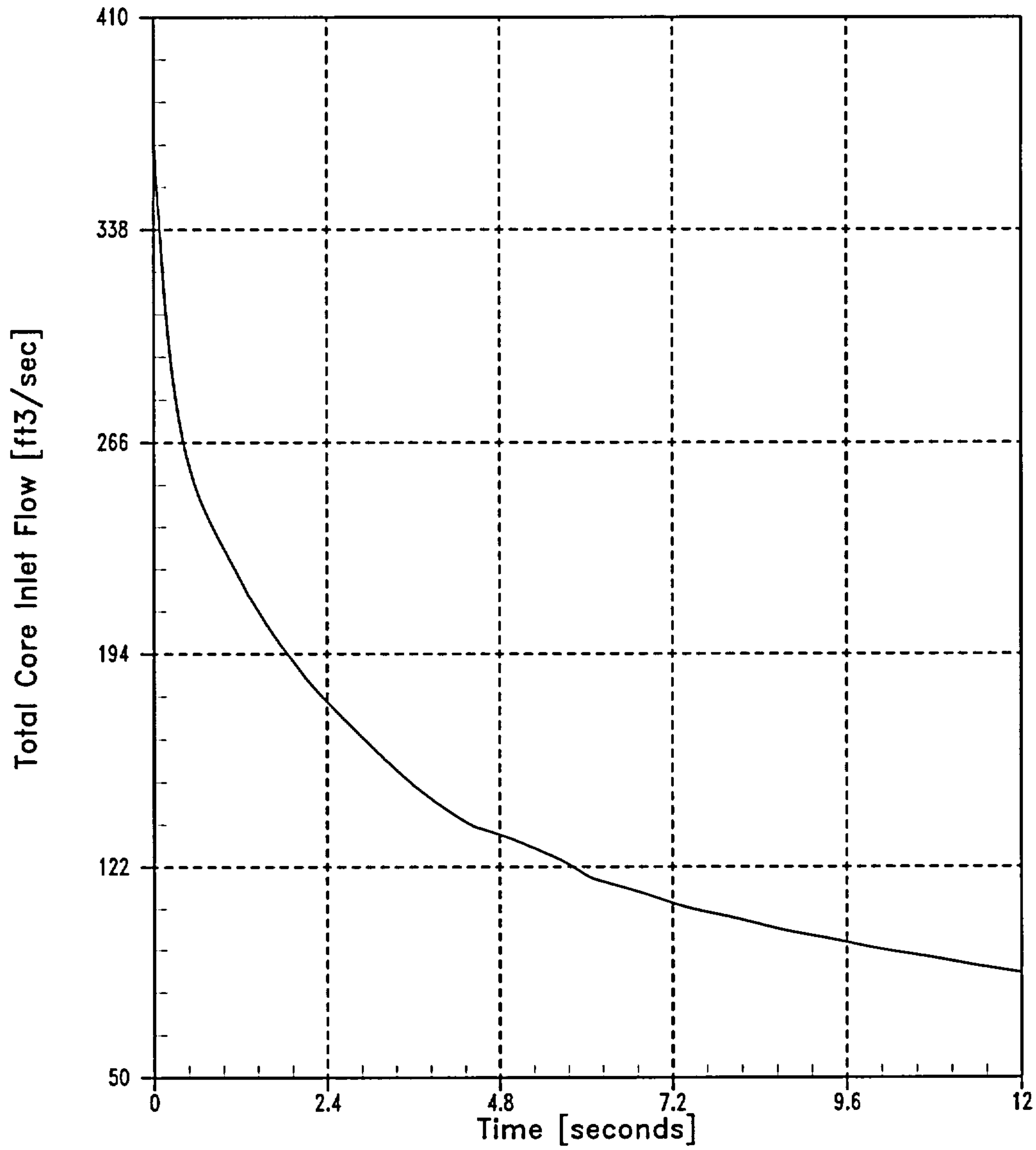

Figure 5.1.8-25 Total Core Inlet Flow versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

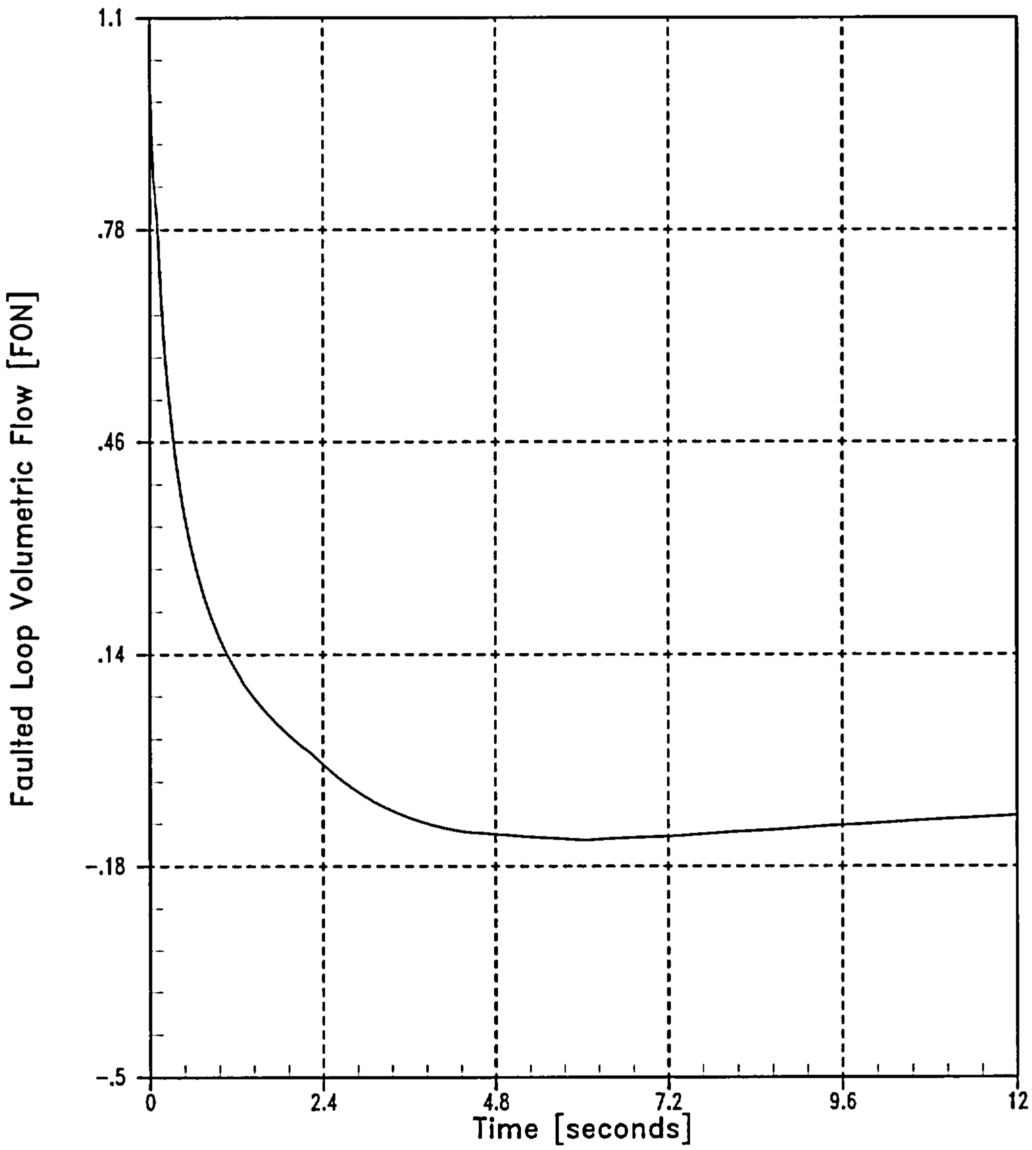

Figure 5.1.8-26 RCS Loop Flow versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

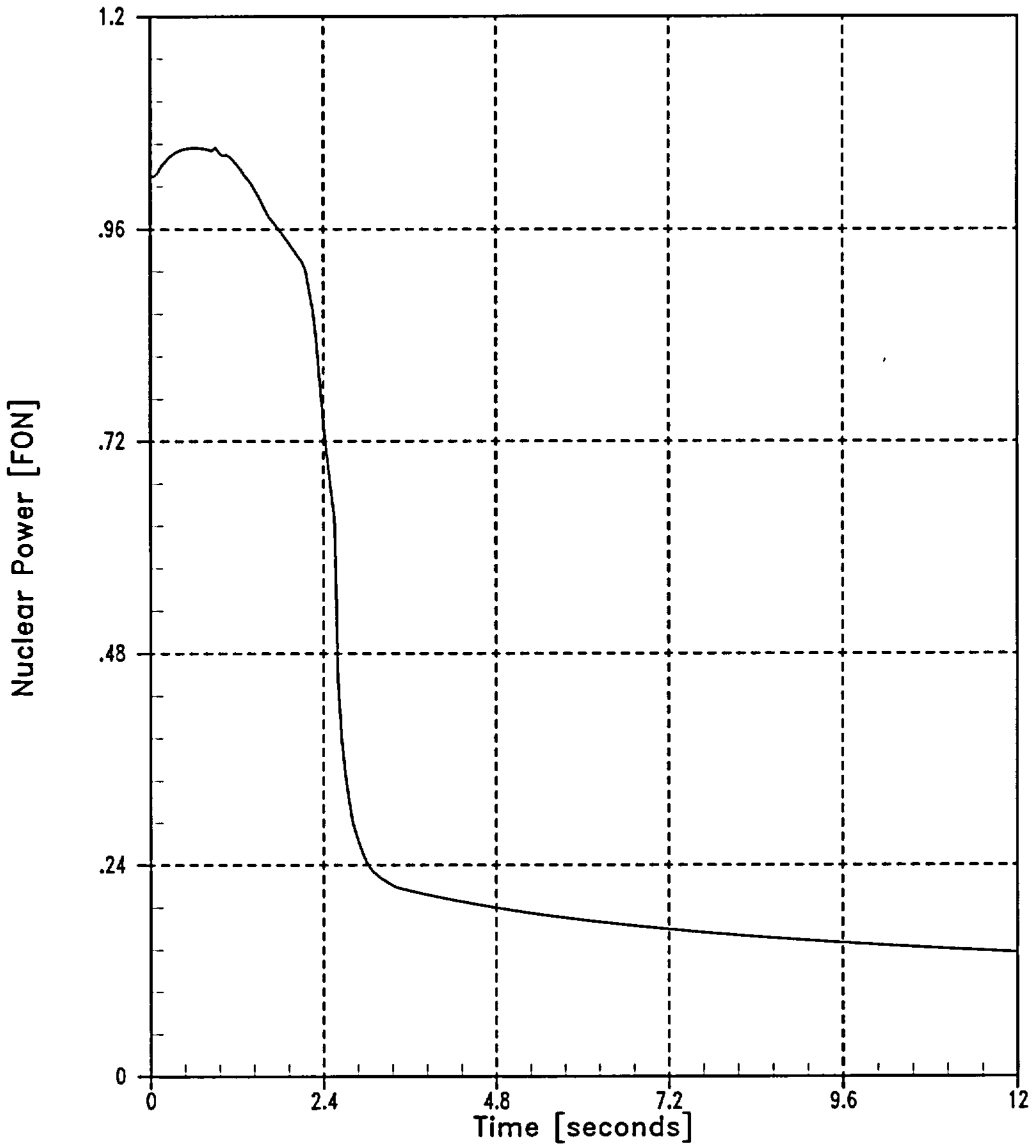

Figure 5.1.8-27 Nuclear Power versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

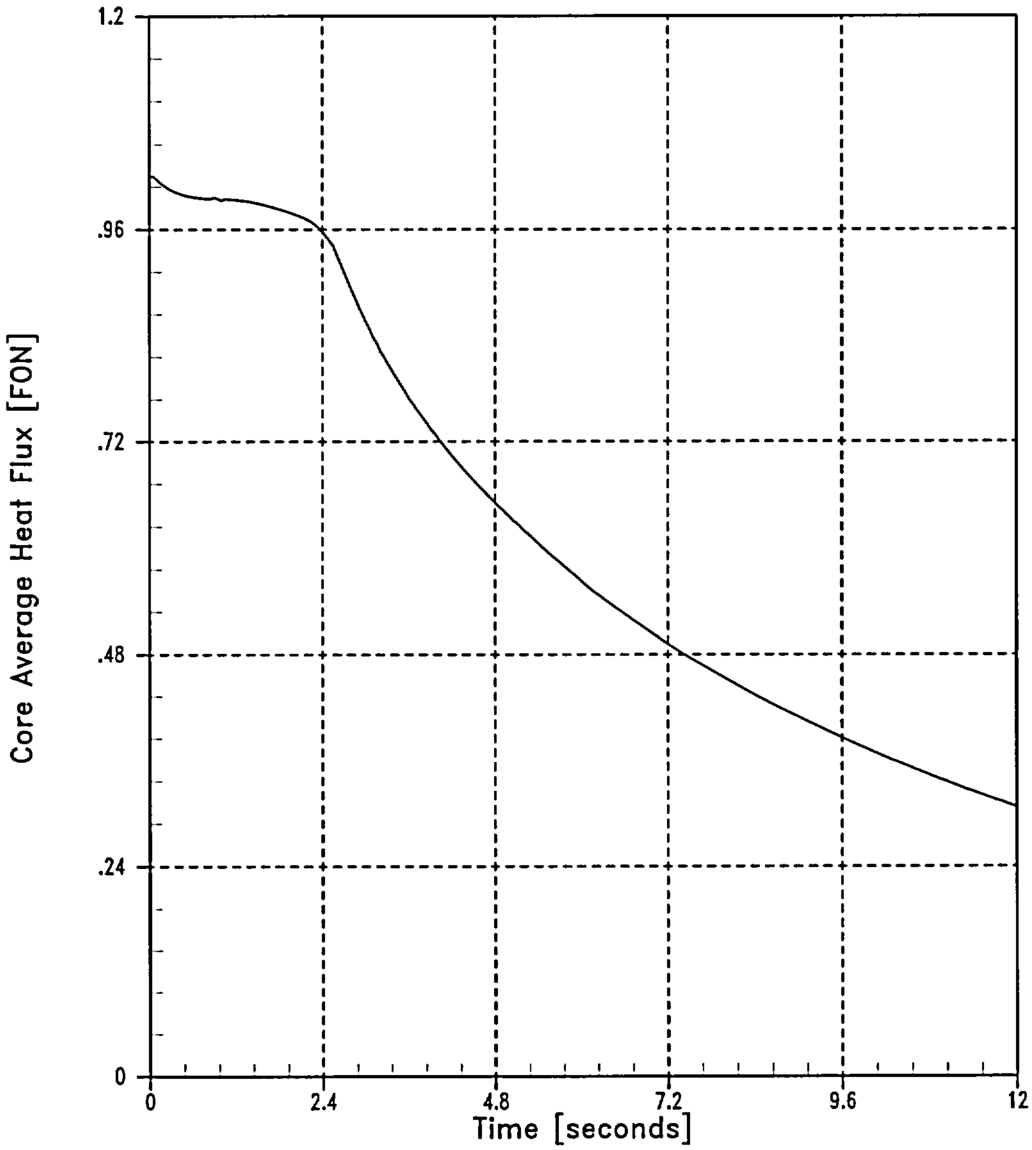

Figure 5.1.8-28 Core Average Heat Flux versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

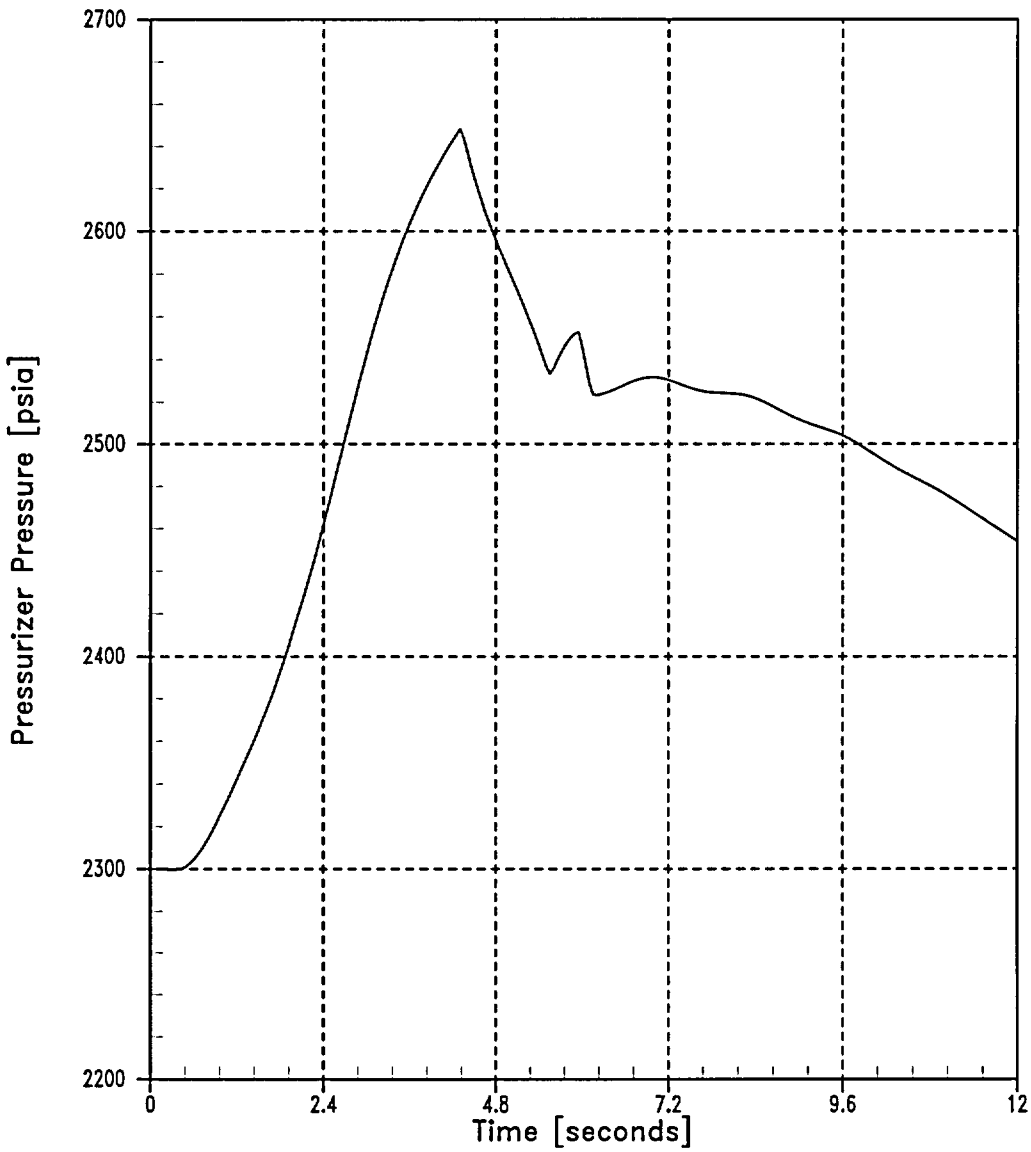

Figure 5.1.8-29 Pressurizer Pressure versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

Figure 5.1.8-30 Vessel Lower Plenum Pressure versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

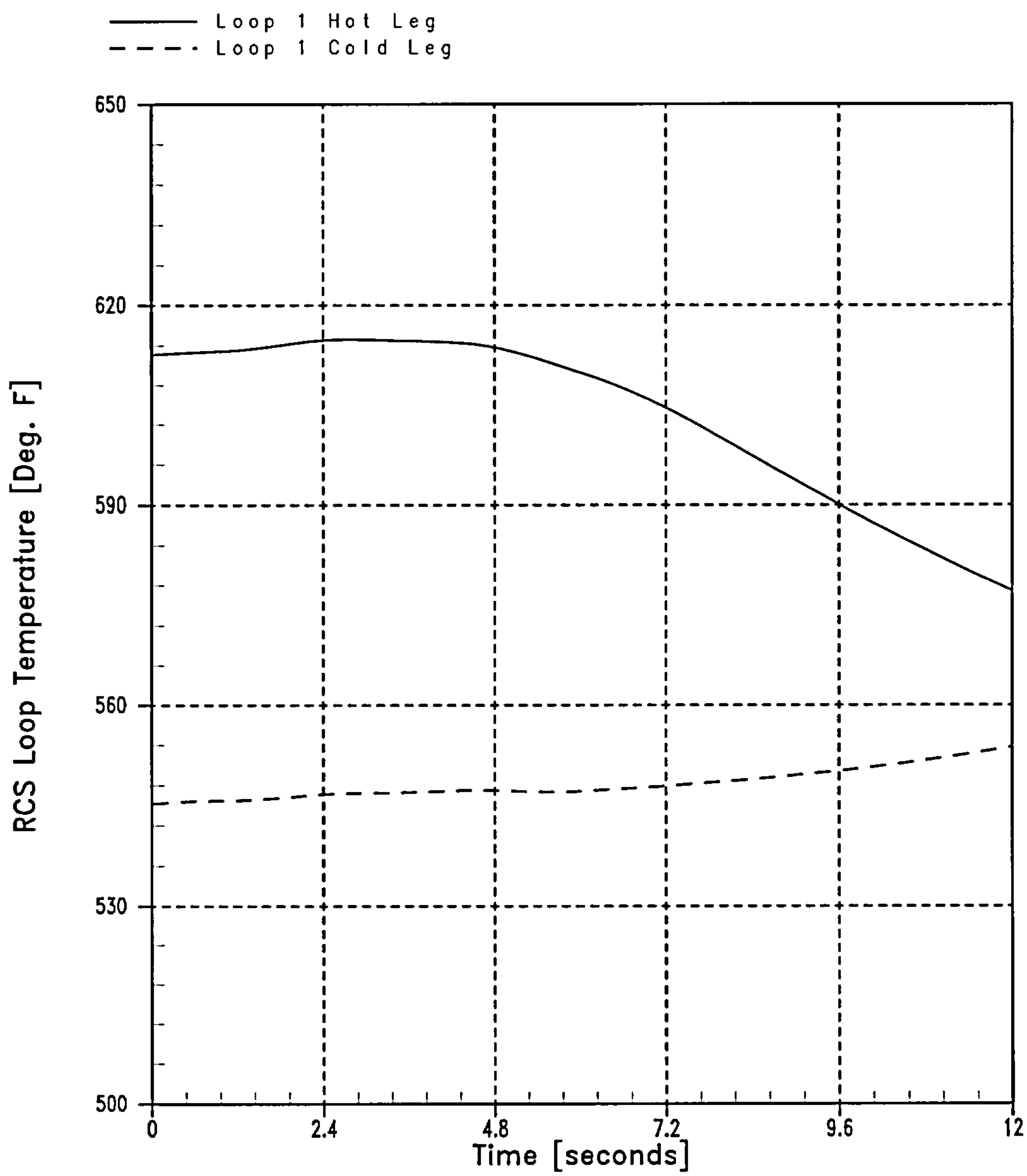

Figure 5.1.8-31 RCS Loop Temperature versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

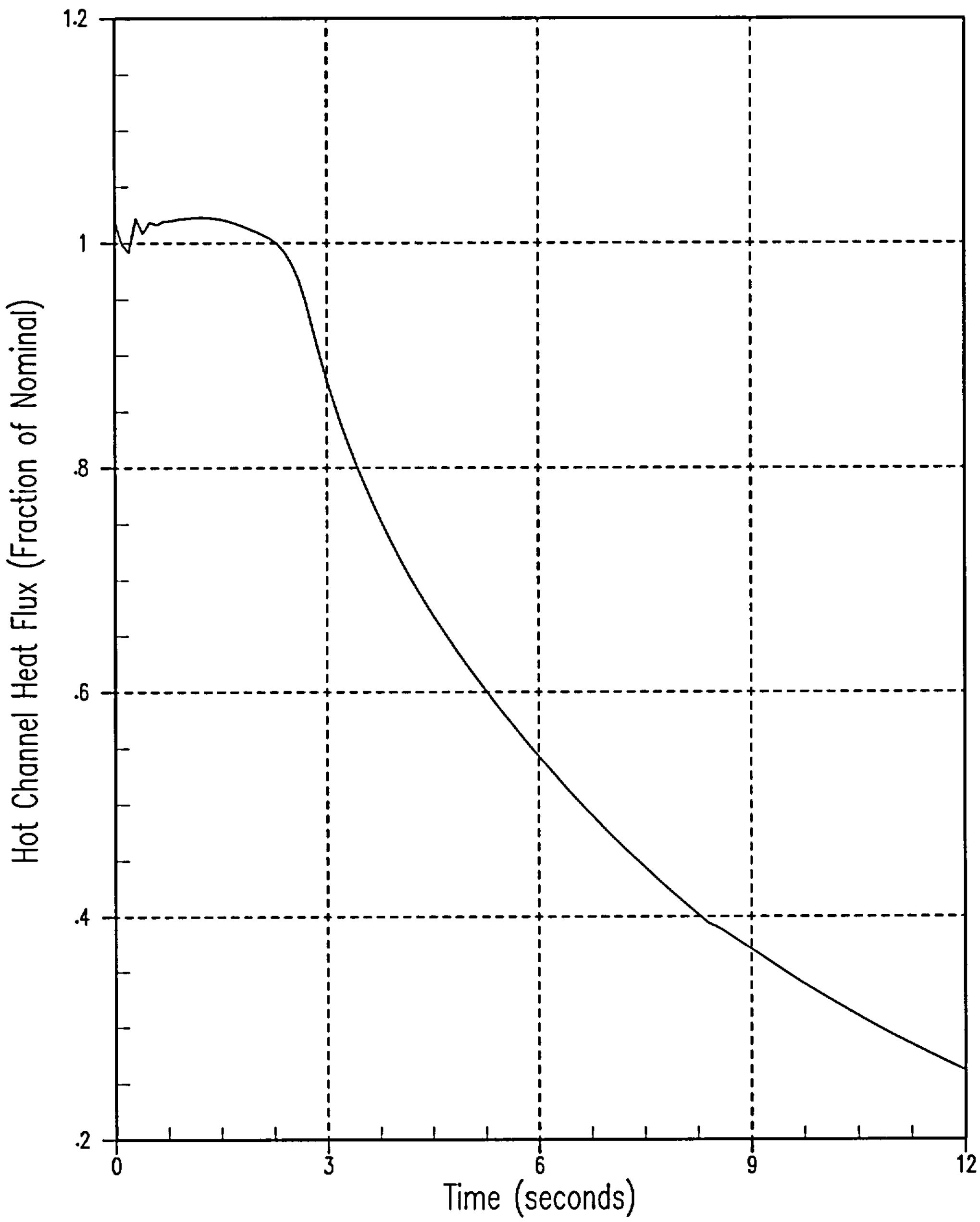

Figure 5.1.8-32 Hot Channel Heat Flux versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

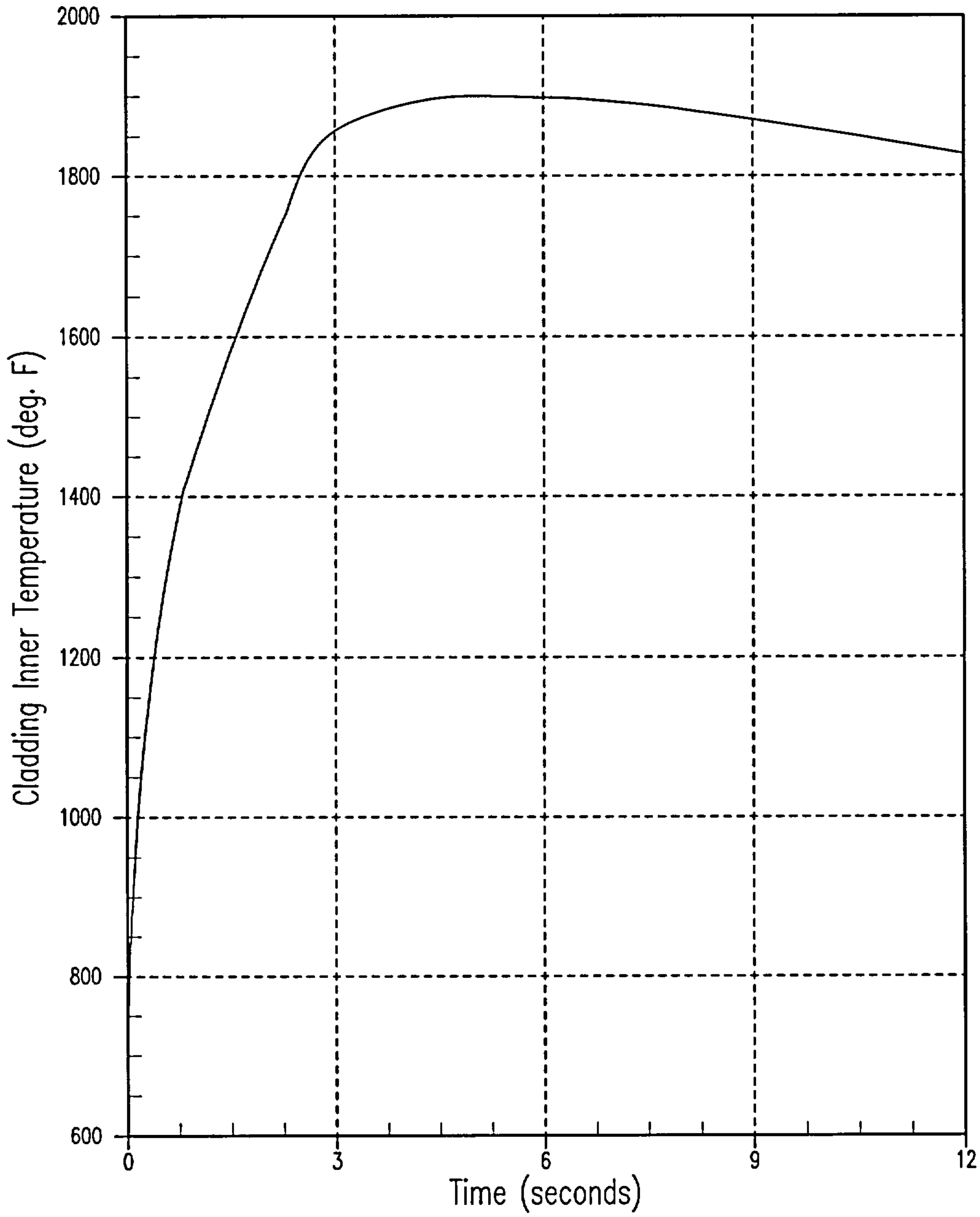

Figure 5.1.8-33 Hot-Spot Cladding Inner Temperature versus Time – Locked Rotor/Shaft Break – RCS Pressure/Peak Cladding Temperature Case

5.1.9 Loss of External Electrical Load (USAR Section 14.1.9)

Accident Description

The loss-of-external-electrical-load event is defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip. This anticipated transient is analyzed as a turbine trip from full power because it bounds both events—the loss of external electrical load and turbine trip. The turbine-trip event is more severe than the total loss-of-external-electrical-load event since it results in a more rapid reduction in steam flow.

For a turbine trip, the reactor would be tripped directly (unless below approximately 10-percent power) from a signal derived from either the turbine auto-stop oil pressure or a closure of the turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressures do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, steam generator level would be maintained by the auxiliary feedwater (AFW) system.

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

In the event of a large loss of load in which the steam dump valves fail to open or a complete loss of load with the steam dump operating, the MSSVs may lift and the reactor may be tripped by any of the following signals: high pressurizer pressure, high pressurizer water level, OTΔT and OPΔT, or lo-lo steam generator water level. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. However, the PSVs and MSSVs are sized to protect the RCS and steam generators against overpressure for all load losses without assuming the operation of the steam dump system. The steam dump valves will not be opened for load reductions of 10 percent or less, but may open for larger load reductions. The RCS and MSS steam relieving capacities were designed to ensure safety of the unit without requiring automatic rod control, pressurizer pressure control, steam bypass control systems, or a reactor trip on turbine trip.

Method of Analysis

The loss-of-load transients are analyzed using the RETRAN computer code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and MSSVs. The code computes pertinent plant variables including temperatures, pressures, and power levels.

The loss-of-load accident is analyzed for the following:

- To confirm that the PSVs and MSSVs are adequately sized to prevent overpressurization of the primary RCS and MSS, respectively

- To ensure that the increase in RCS temperature does not result in a DNB in the core

The RPS is designed to automatically terminate any such transient before the DNBR falls below the limit value.

In this analysis, the behavior of the unit is evaluated for a complete loss-of-steam load from full power with no credit taken for a direct reactor trip on turbine trip. This assumption will delay reactor trip until conditions in the RCS cause a trip on some other signal. Therefore, the analysis assumes a worst-case transient and demonstrates the adequacy of the pressure-relieving devices and plant-specific RPS setpoints assumed in the analysis for this event.

Of the three cases analyzed, one is performed to address DNB concerns, one ensures that the peak primary RCS pressure remains below the design limit (2750 psia), and the final case confirms that the peak MSS pressure remains below 110 percent of the steam generator shell design pressure (1210 psia). The major assumptions for these cases are summarized as follows:

a. For the case analyzed to demonstrate that the core thermal limits are adequately protected (BOC reactivity feedback conditions with automatic pressurizer pressure control), the loss-of-load accident is analyzed using the RTDP (Reference 5-1). For this case, initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at the nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in determining the DNBR limit value (Reference 5-1). For the case analyzed to demonstrate the adequacy of the primary pressure-relieving devices (BOC reactivity feedback conditions without automatic pressurizer pressure control), the loss-of-load accident is analyzed using the STDP. For this case, initial core power and reactor coolant temperature are assumed at the maximum values consistent with steady-state full-power operation, including allowances for calibration and instrument errors. Initial pressurizer pressure is assumed at the minimum value for this case, since it delays reactor trip on high pressurizer pressure and results in more severe primary-side temperature and pressure transients. This results in the maximum power difference for the loss of load. Similar to the primary RCS overpressurization case, the MSS overpressurization case is analyzed assuming the STDP assumptions with respect to initial conditions and uncertainties and also assumes BOC reactivity feedback conditions. However, the MSS overpressurization case differs from the primary RCS overpressurization case in that automatic pressurizer pressure control is assumed in order to delay reactor trip.

b. The loss-of-load event results in a primary-system heatup and, therefore, is conservatively analyzed assuming minimum reactivity feedback consistent with BOC conditions. This includes assuming an MTC value consistent with BOC HFP conditions (that is, zero MTC) and a least negative DPC. Maximum feedback (EOC) cases that were previously considered in the USAR are no longer analyzed since they have been determined (as part of the Westinghouse methodology for the analysis of this event) to be non-limiting with respect to the minimum DNBR, peak primary RCS pressure, and peak MSS pressure.

c. It is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

d. No credit is taken for the operation of the steam dump system or steam generator power-operated relief valve (PORVs). The steam generator pressure rises to the safety valve setpoints, where steam release through the MSSV limits the secondary-side steam pressure to the setpoint values. The MSSV was explicitly modeled in the loss-of-load licensing basis analysis assuming a +1.0-percent tolerance with a 5 psi pop to full open. The MSSV model also assumed a 40 psi pressure drop from the steam generator exit to the MSSV inlet in determining the opening setpoints and an additional 10 psi pressure drop at full-open and full-flow conditions. Justification for the use of the 5 psi pop instead of modeling accumulation is based on test data documented in WCAP-10105 (Reference 5-13) and WCAP-12910 (Reference 5-2). Note that by maximizing the pressure transient in the MSS, the saturation temperature in the steam generators is maximized, resulting in limiting pressure and temperature conditions in the RCS.

e. Three cases are analyzed:

   1. For the case analyzed for DNB, automatic pressurizer pressure control is assumed. Therefore, full credit is taken for the effect of the pressurizer spray and PORVs in reducing or limiting the primary coolant pressure. Safety valves are also available and are modeled assuming a -1-percent setpoint tolerance.

   2. For the case analyzed for primary RCS overpressure concerns, it is assumed that automatic pressurizer pressure control is not available. Therefore, no credit is taken for the effect of the pressurizer spray or PORVs in reducing or limiting the primary coolant pressure. Safety valves are assumed operable, but are modeled assuming a +1-percent setpoint tolerance. The effects of the PSV loop seals are also conservatively modeled in the analysis.

   3. For the case analyzed for MSS overpressure concerns, it is assumed that automatic pressurizer pressure control is available. Credit is taken for the effect of the pressurizer spray and PORVs in reducing or limiting the primary coolant pressure, therefore conservatively delaying the actuation of the RPS until an OTΔT reactor trip signal is generated. Delaying the reactor trip ensures that the energy input to the secondary system, and subsequently the MSS pressure, is maximized.

f. Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for AFW flow since a stabilized plant condition will be reached before AFW initiation is normally assumed to occur for full-power cases. However, the AFW pumps would be expected to start on a trip of the main feedwater pumps. The AFW flow would remove core decay heat following plant stabilization.

g. The analysis is performed for operation with 422V+ fuel and a maximum SGTP level (uniform) for KNPP of ≤ 10 percent.

h. A maximum SGTP level of 10 percent is modeled. SGTP imbalances do not adversely affect this transient.

Results

The transient responses for a total loss of load from full-power operation are shown in Figures 5.1.9-1 through 5.1.9-16 for the 3 cases assuming BOC reactivity feedback conditions with and without automatic pressurizer pressure control (pressurizer spray and PORVs).

Figures 5.1.9-1 through 5.1.9-6 show the transient responses for the total loss of steam load at BOC (minimum feedback reactivity coefficients) assuming full credit for the pressurizer spray and PORVs to calculate the transient DNBR response. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped by the OTΔT trip function. The DNBR initially increases slightly, then decreases until the reactor trip is tripped. Finally, following reactor trip, it increases rapidly. The minimum DNBR remains well above the safety analysis limit value. The MSSVs actuate to limit the MSS pressure below 110 percent of the steam generator shell design pressure. Table 5.1.9-1 summarizes the sequence of events and limiting conditions for this case.

The total loss-of-load event was also analyzed assuming the plant to be initially operating at full power at BOC with no credit taken for the pressurizer spray or PORVs to maximize the primary RCS pressure response. Figures 5.1.9-7 through 5.1.9-11 show the transients for this case. The neutron flux remains relatively constant prior to reactor trip, while pressurizer pressure, pressurizer water volume, and RCS average temperature increase due to the sudden reduction in primary to secondary heat transfer. The reactor is tripped on the high pressurizer pressure trip signal. In this case, the PSVs are actuated and maintain the primary RCS pressure below 110 percent of the design value. The MSSVs actuate to limit the MSS pressure below 110 percent of the steam generator shell design pressure. Table 5.1.9-2 summarizes the sequence of events and limiting conditions for this case.

Figures 5.1.9-12 through 5.1.9-16 show the transient responses for the total loss of steam load at BOC (minimum feedback reactivity coefficients) assuming full credit for the pressurizer spray and PORVs to maximize the MSS pressure response. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped by the OTΔT trip function. The MSS pressure increases, resulting in the actuation of the first three MSSVs, and then decreases rapidly following reactor trip. The MSSVs actuate to limit the MSS pressure below 110 percent of the steam generator shell design pressure. Table 5.1.9-3 summarizes the sequence of events and limiting conditions for this case.

Conclusions

The results of the analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the primary RCS or MSS. Pressure-relieving devices that have been incorporated into the plant design are adequate to limit the maximum pressures to within the safety analysis limits; that is, 2750 psia for the primary RCS and 1210 psia for the MSS.

The integrity of the core is maintained by operation of the RPS; that is, the minimum DNBR is maintained above the safety analysis limit value of 1.34. Therefore, no core safety limit will be violated as a result of implementing the FU/PU.

Table 5.1.9-1 Sequence of Events and Transient Results – Loss of External Electrical Load – with Pressurizer Pressure Control (for Minimum DNB)	
Event	**Time (seconds)**
Turbine Trip	0.0
Reactor Trip on OTΔT	11.9
Rod Motion Begins	13.9
Time of Minimum DNBR	14.9
Time of Peak MSS Pressure	21.1
Minimum DNBR Value	1.74
DNBR Limit	1.34
Peak MSS Pressure	1194 psia
MSS Pressure Limit	1210 psia

Table 5.1.9-2 Sequence of Events and Transient Results – Loss of External Electrical Load – without Pressurizer Pressure Control (for Primary RCS Overpressure)	
Event	**Time (seconds)**
Turbine Trip	0.0
Reactor Trip on High Pressurizer Pressure	7.9
Rod Motion Begins	8.9
Time of Peak RCS Pressure	11.1
Time of Peak MSS Pressure	16.9
Peak RCS Pressure	2697 psia
RCS Pressure Limit	2750 psia
Peak MSS Pressure	1182 psia
MSS Pressure Limit	1210 psia

Table 5.1.9-3 Sequence of Events and Transient Results – Loss of External Electrical Load – with Pressurizer Pressure Control (for MSS Overpressure)

Event	Time (seconds)
Turbine Trip	0.0
Reactor Trip on OTΔT	10.2
Rod Motion Begins	12.2
Time of Peak RCS Pressure	11.1
Time of Peak MSS Pressure	17.6
Peak RCS Pressure	2432 psia
RCS Pressure Limit	2750 psia
Peak MSS Pressure	1202 psia
MSS Pressure Limit	1210 psia

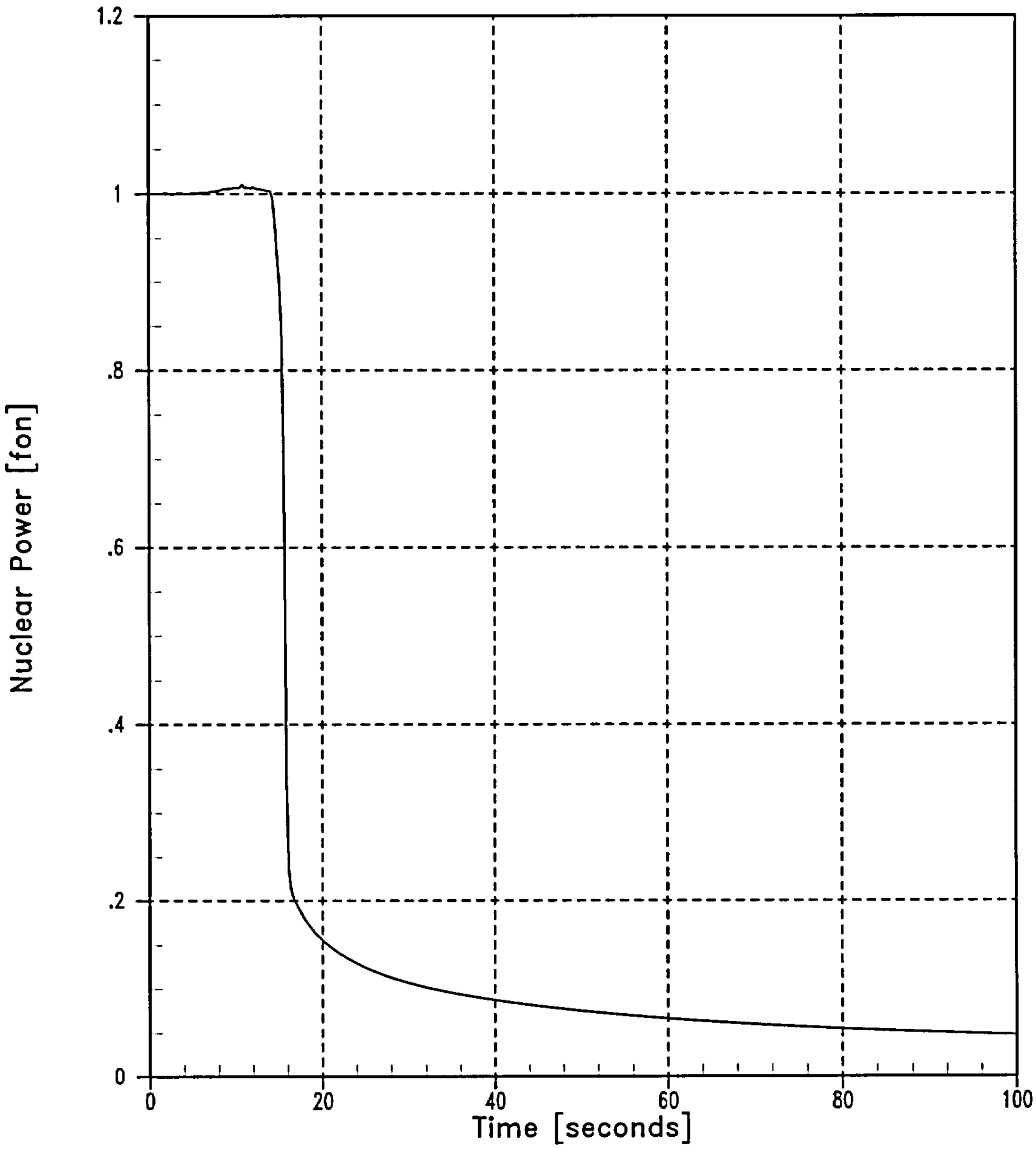

Figure 5.1.9-1 Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Nuclear Power versus Time

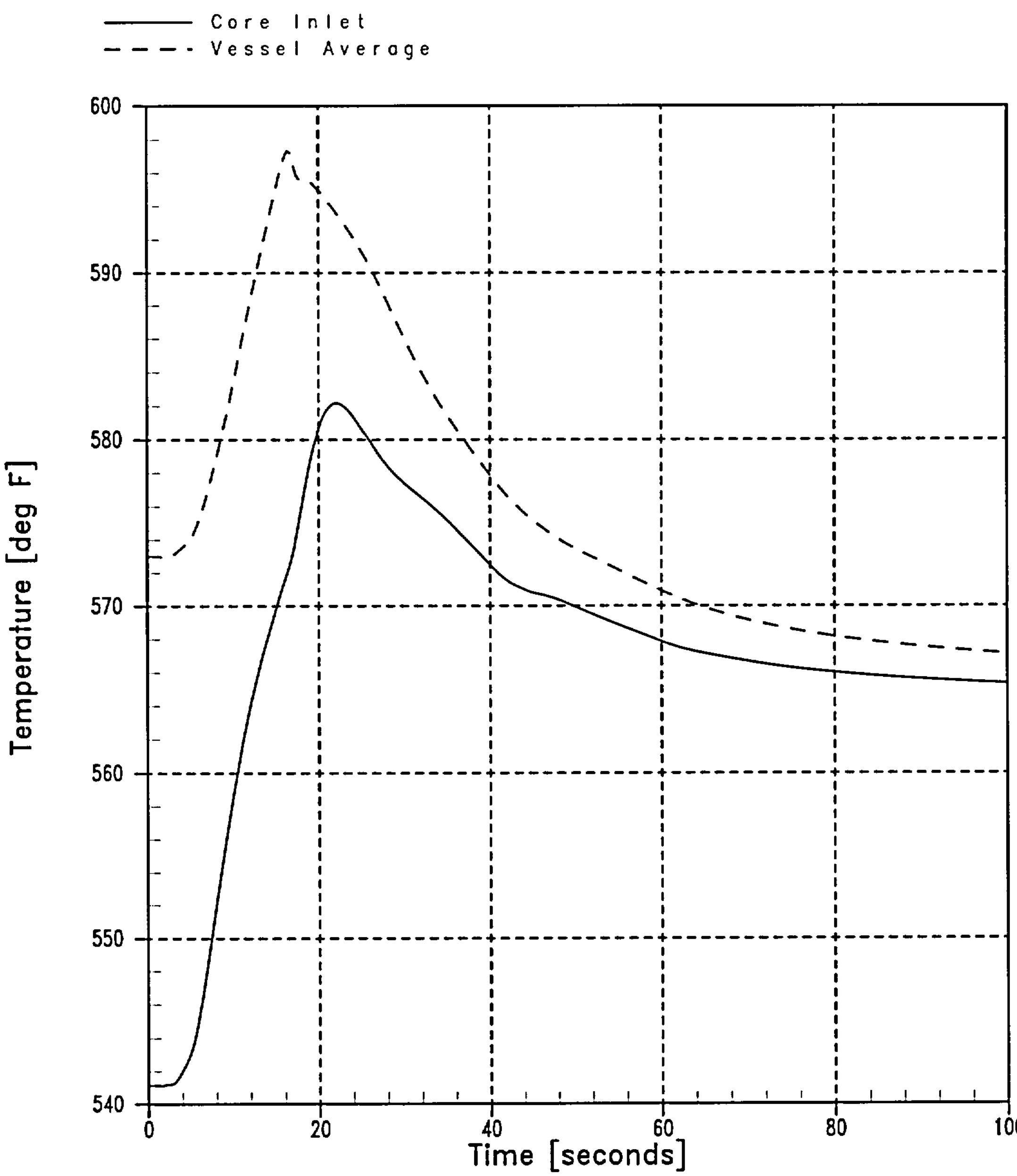

Figure 5.1.9-2 Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Vessel Average and Core Inlet Temperatures versus Time

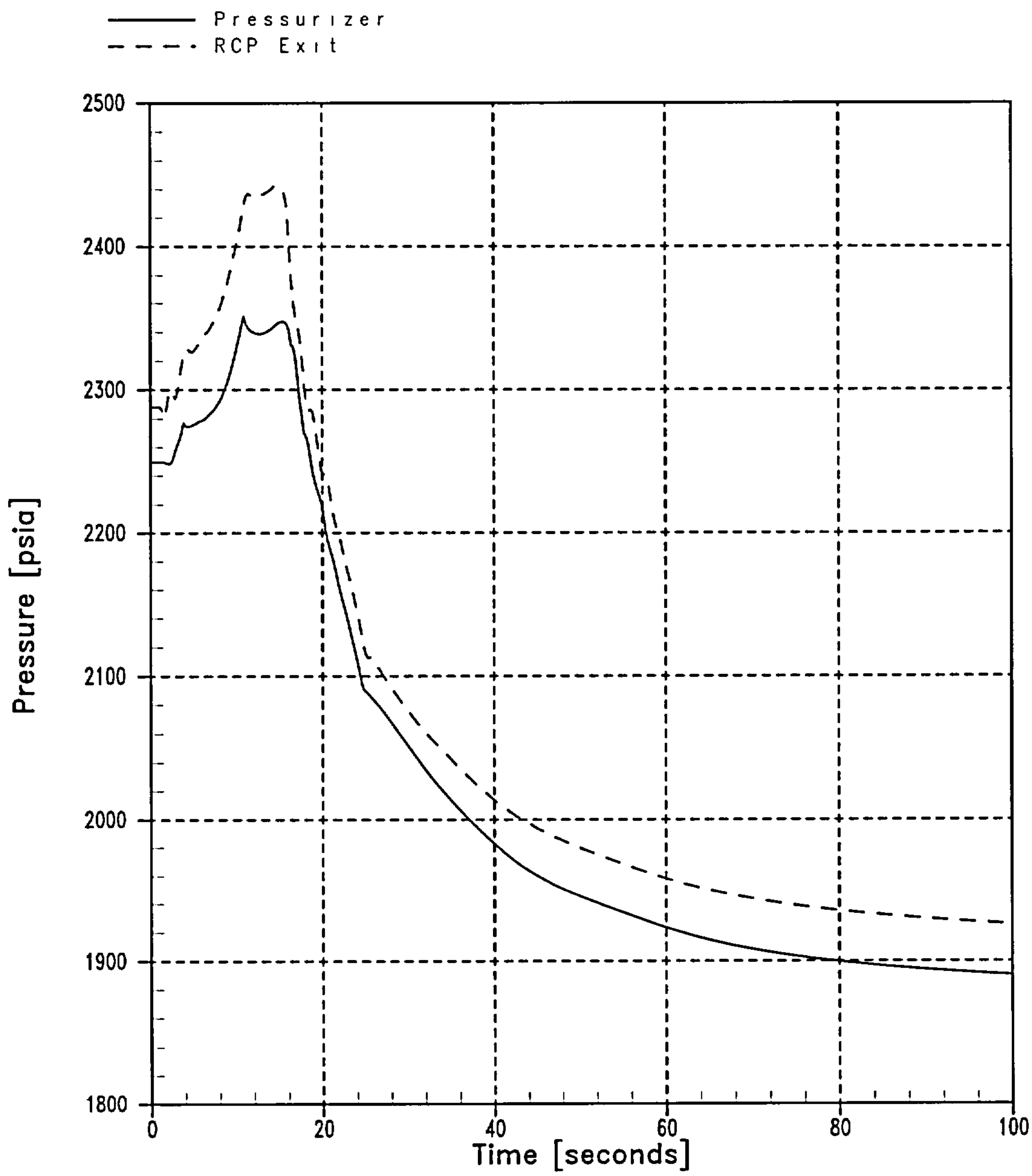

Figure 5.1.9-3 Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Pressurizer and RCP Exit Pressures versus Time

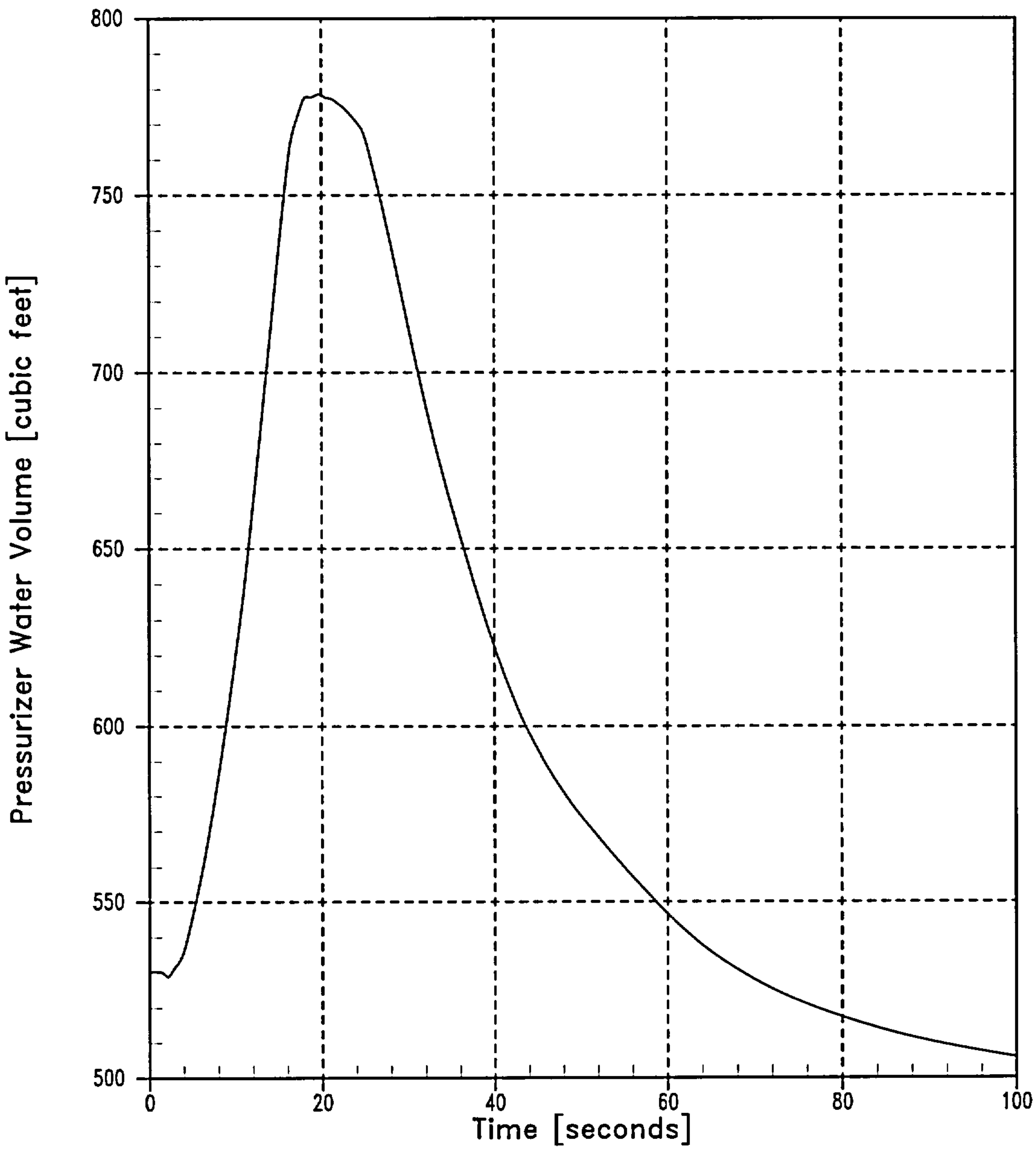

Figure 5.1.9-4 Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Pressurizer Water Volume versus Time

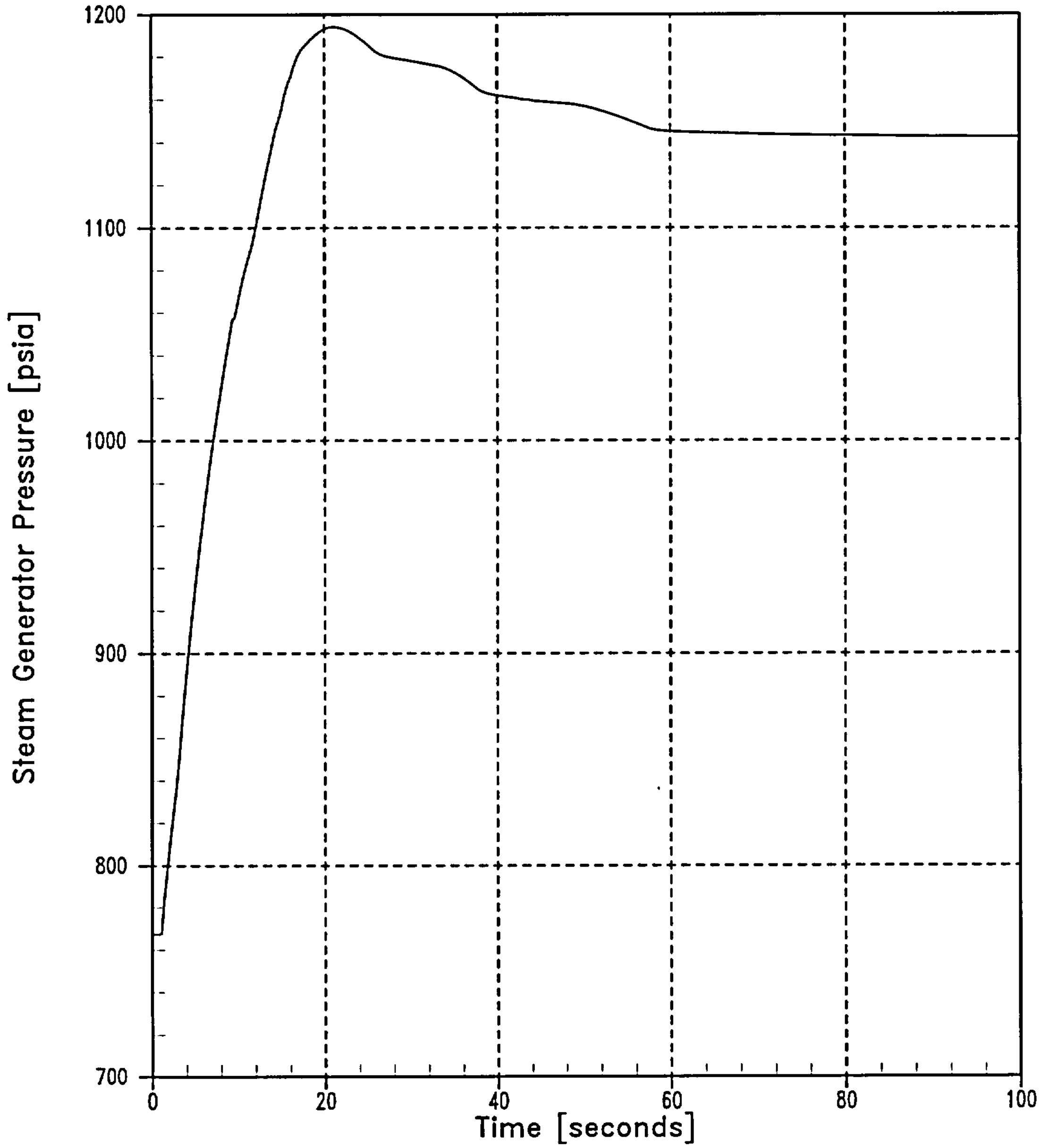

Figure 5.1.9-5 Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – Steam Generator Pressure versus Time

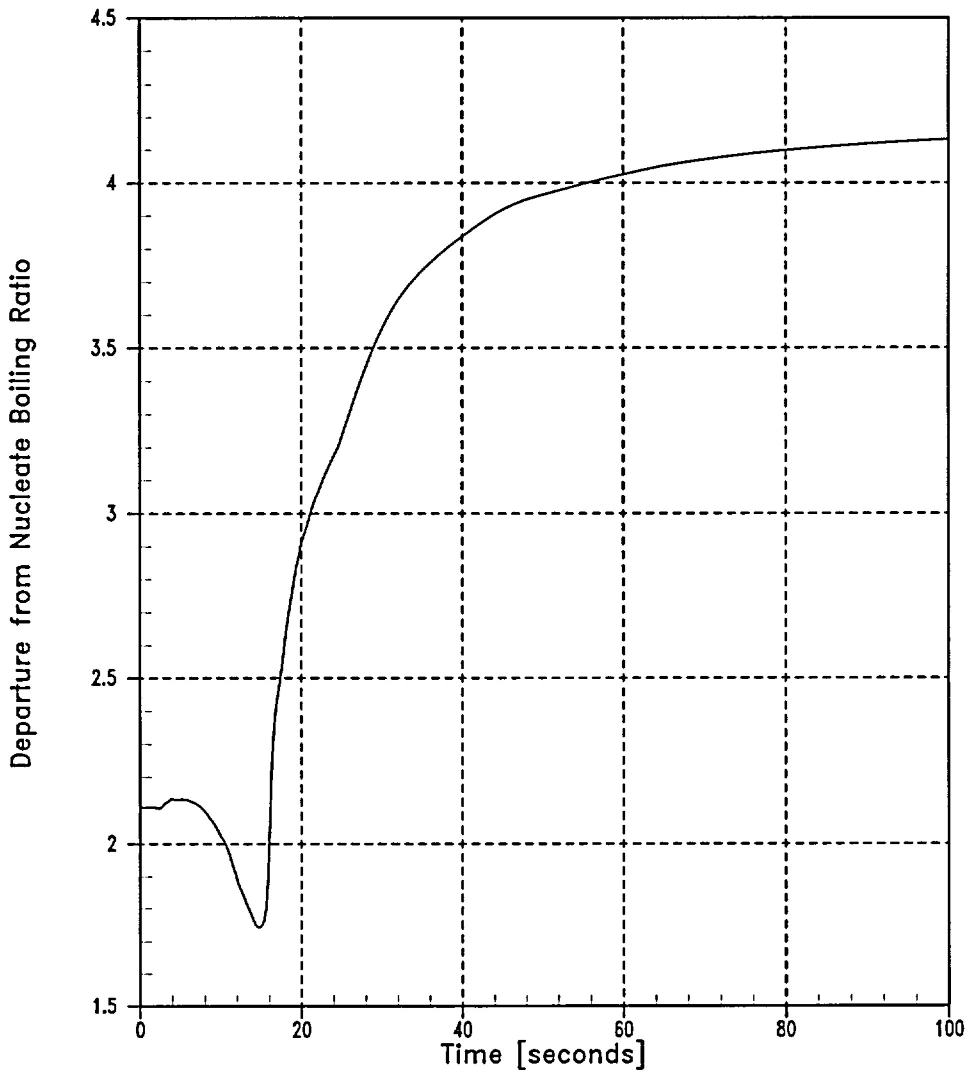

Figure 5.1.9-6 Loss of External Electrical Load with Automatic Pressure Control (DNB Case) – DNBR versus Time

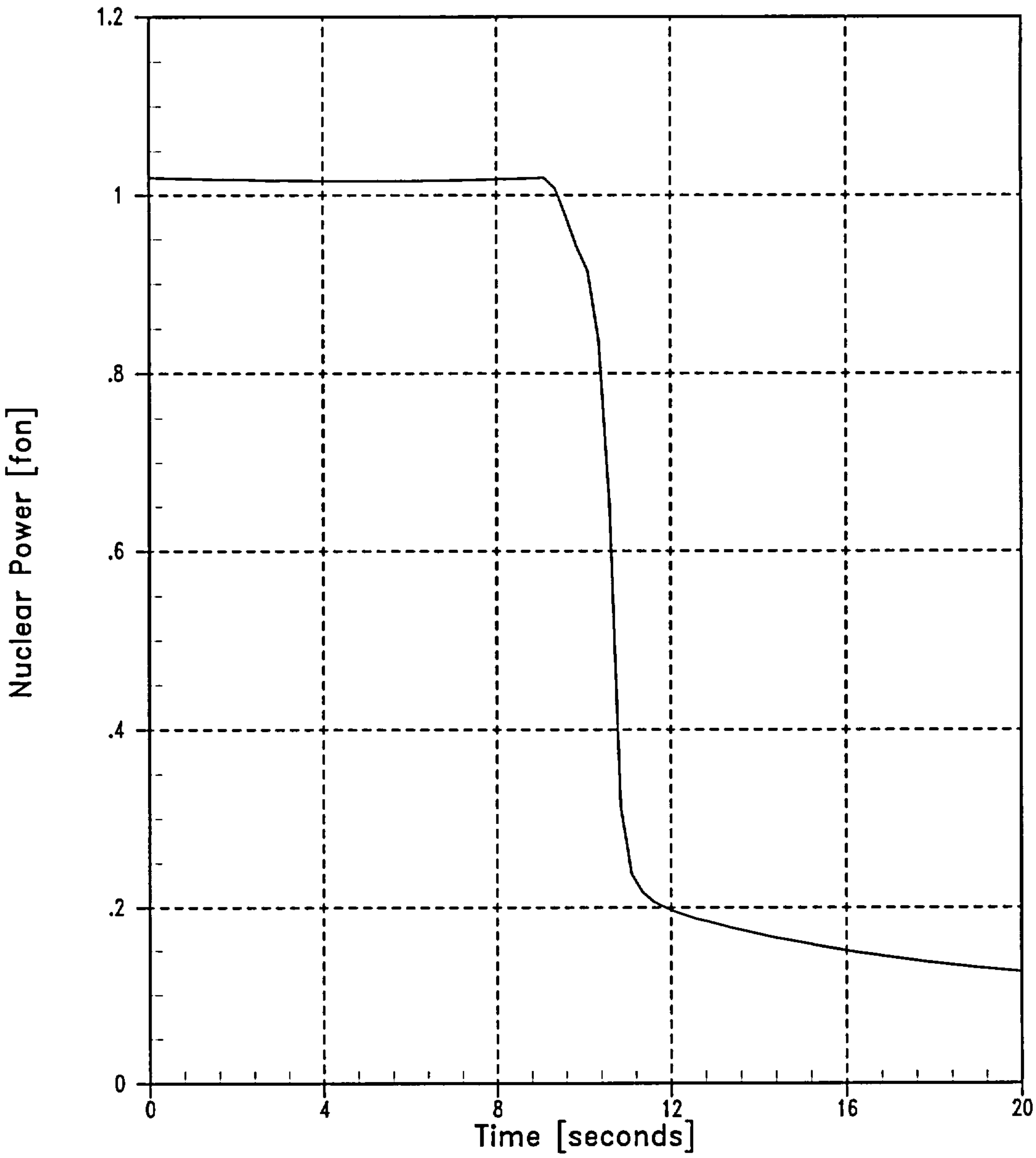

Figure 5.1.9-7 Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Nuclear Power versus Time

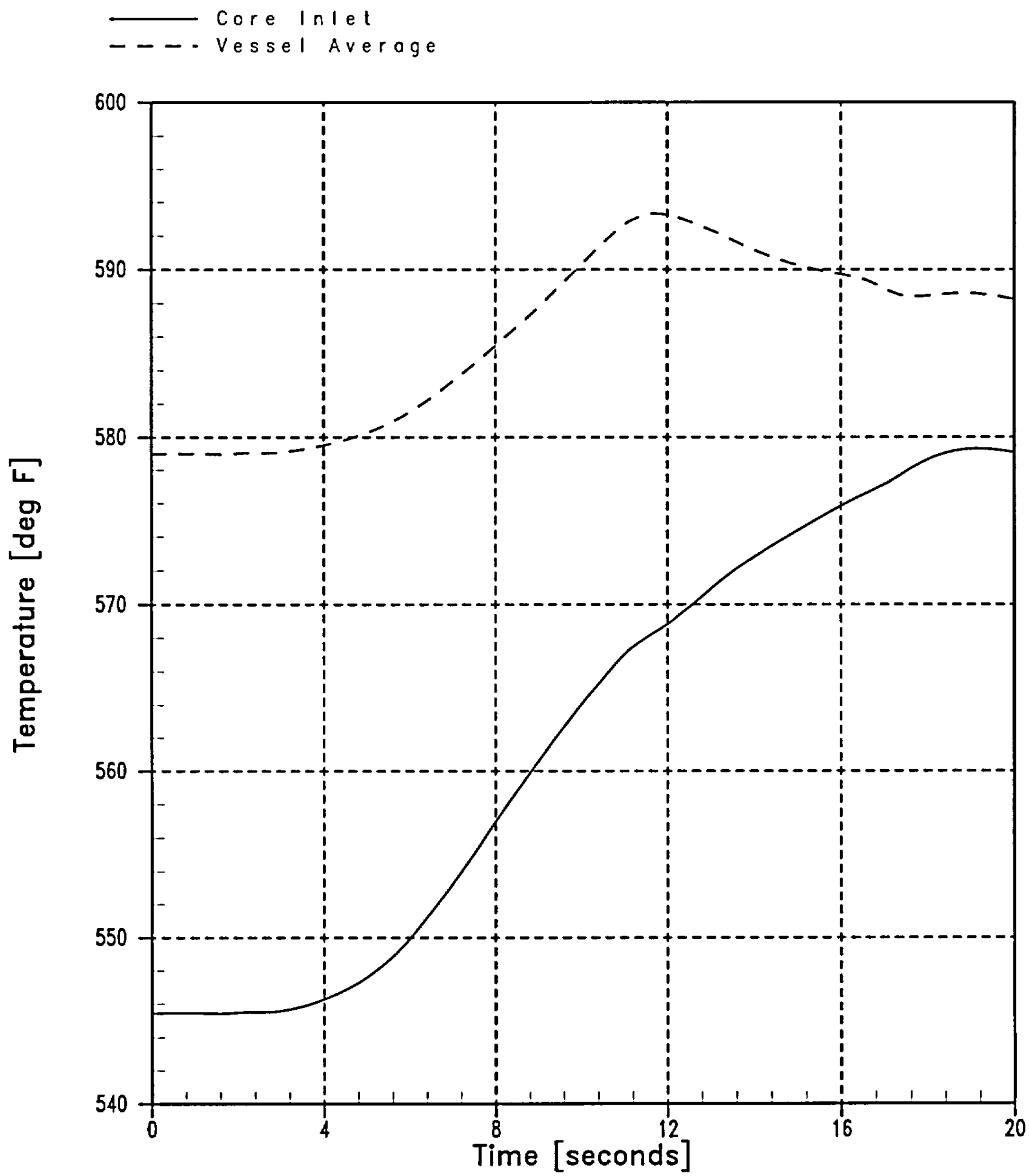

Figure 5.1.9-8 Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Vessel Average and Core Inlet Temperatures versus Time

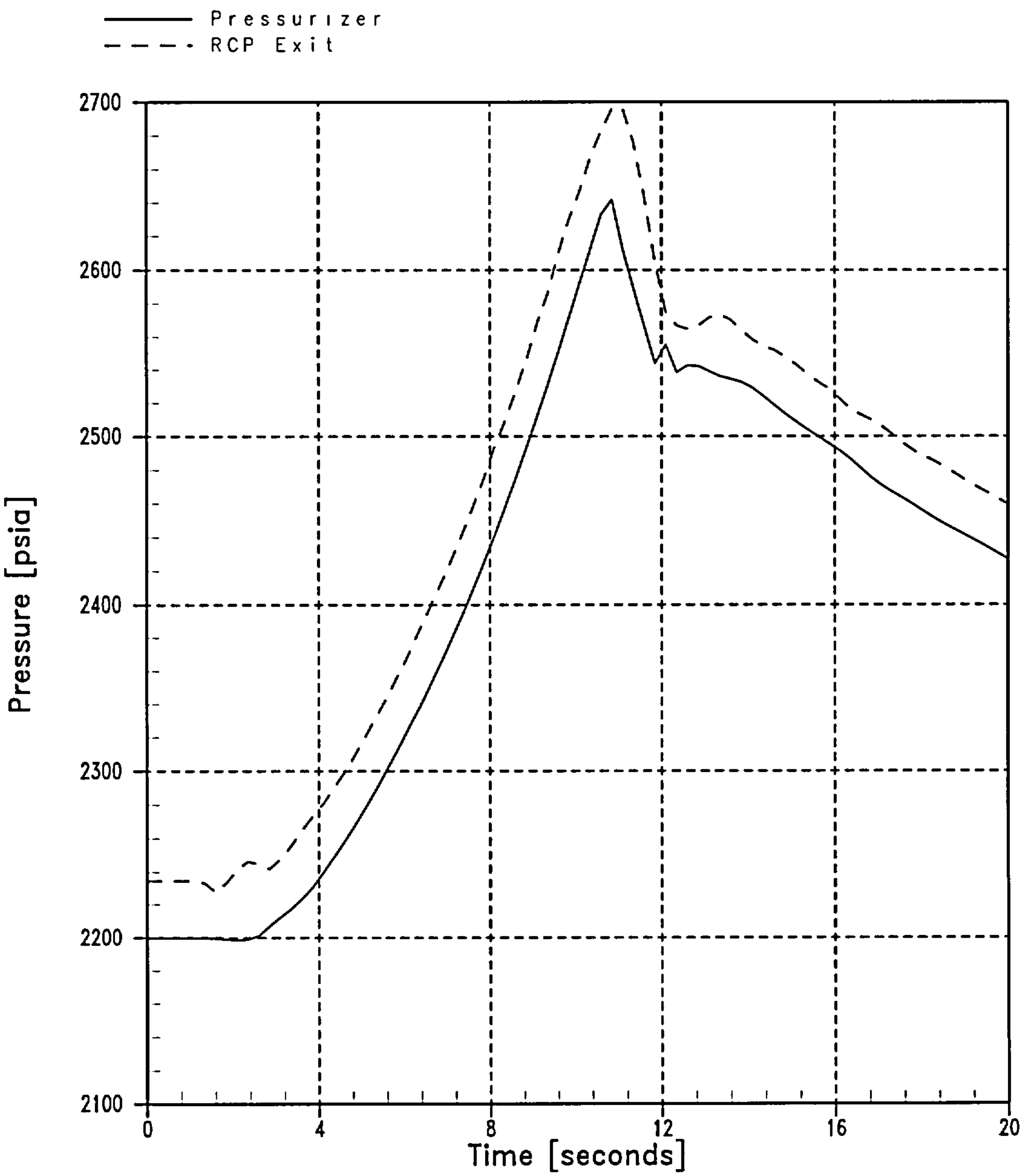

Figure 5.1.9-9 Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Pressurizer and RCP Exit Pressures versus Time

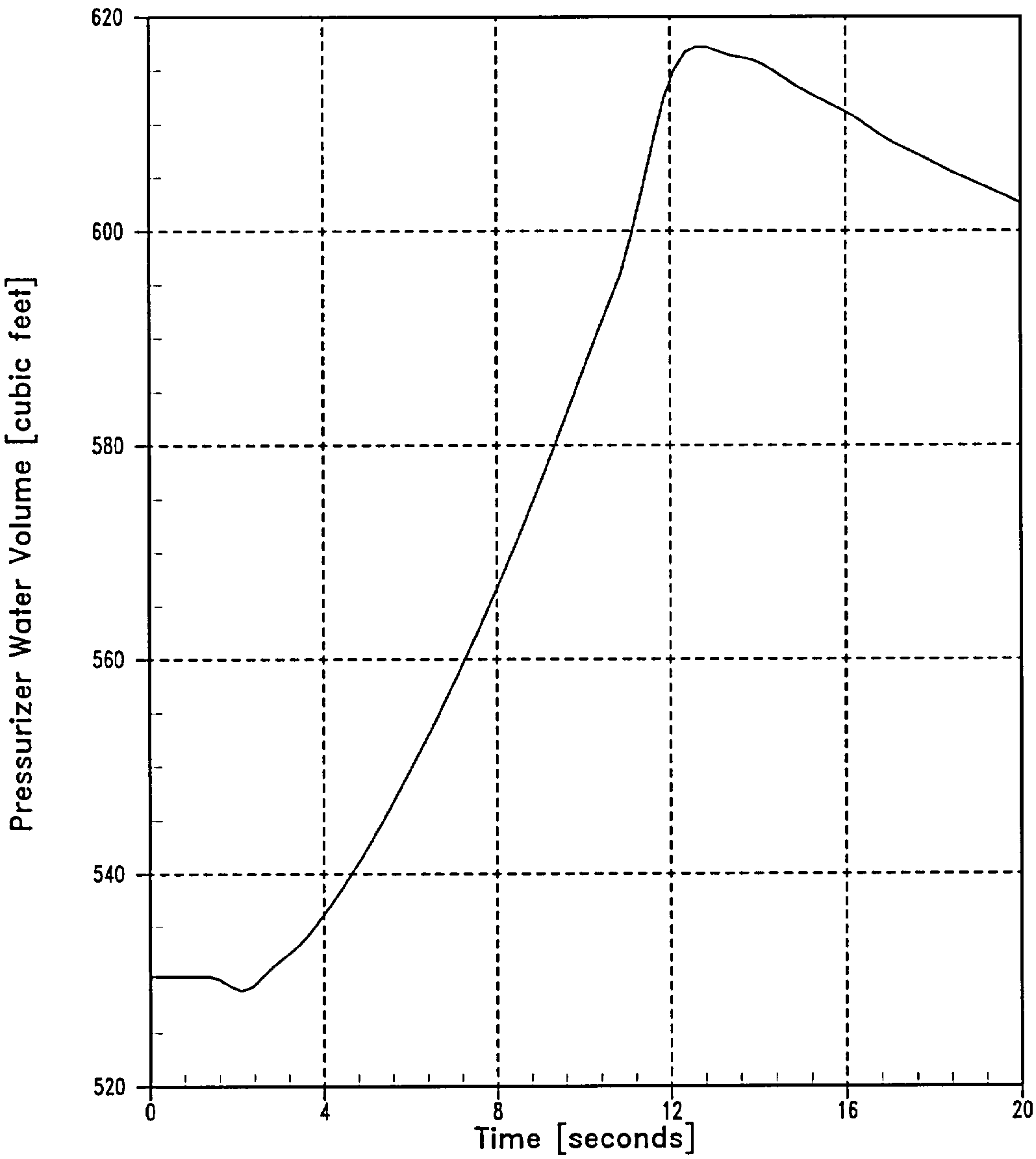

Figure 5.1.9-10 Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Pressurizer Water Volume versus Time

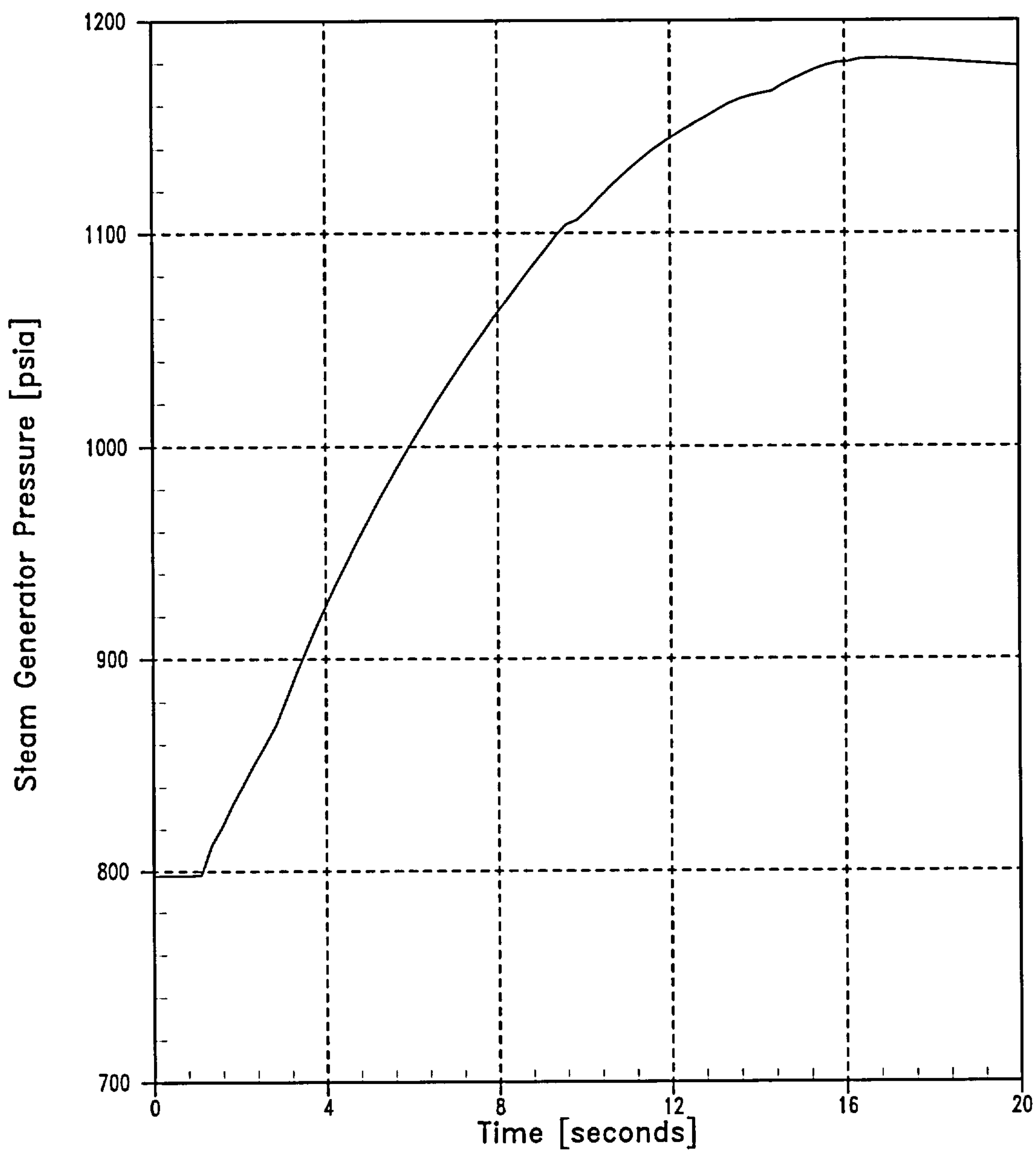

Figure 5.1.9-11 Loss of External Electrical Load Without Automatic Pressure Control (RCS Overpressure Case) – Steam Generator Pressure versus Time

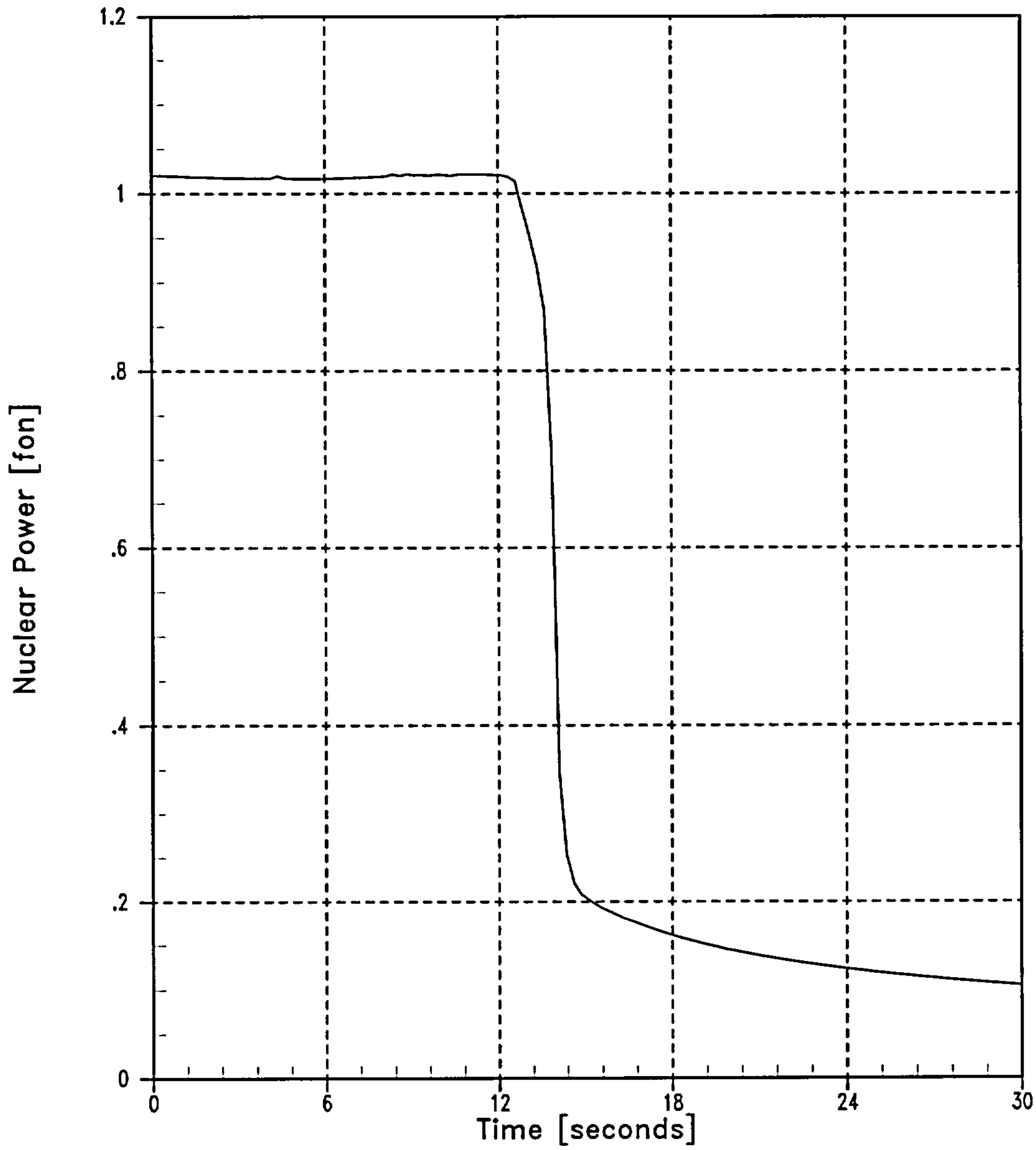

Figure 5.1.9-12 Loss of External Electrical Load with Automatic Pressure Control (MSS Overpressure Case) – Nuclear Power versus Time

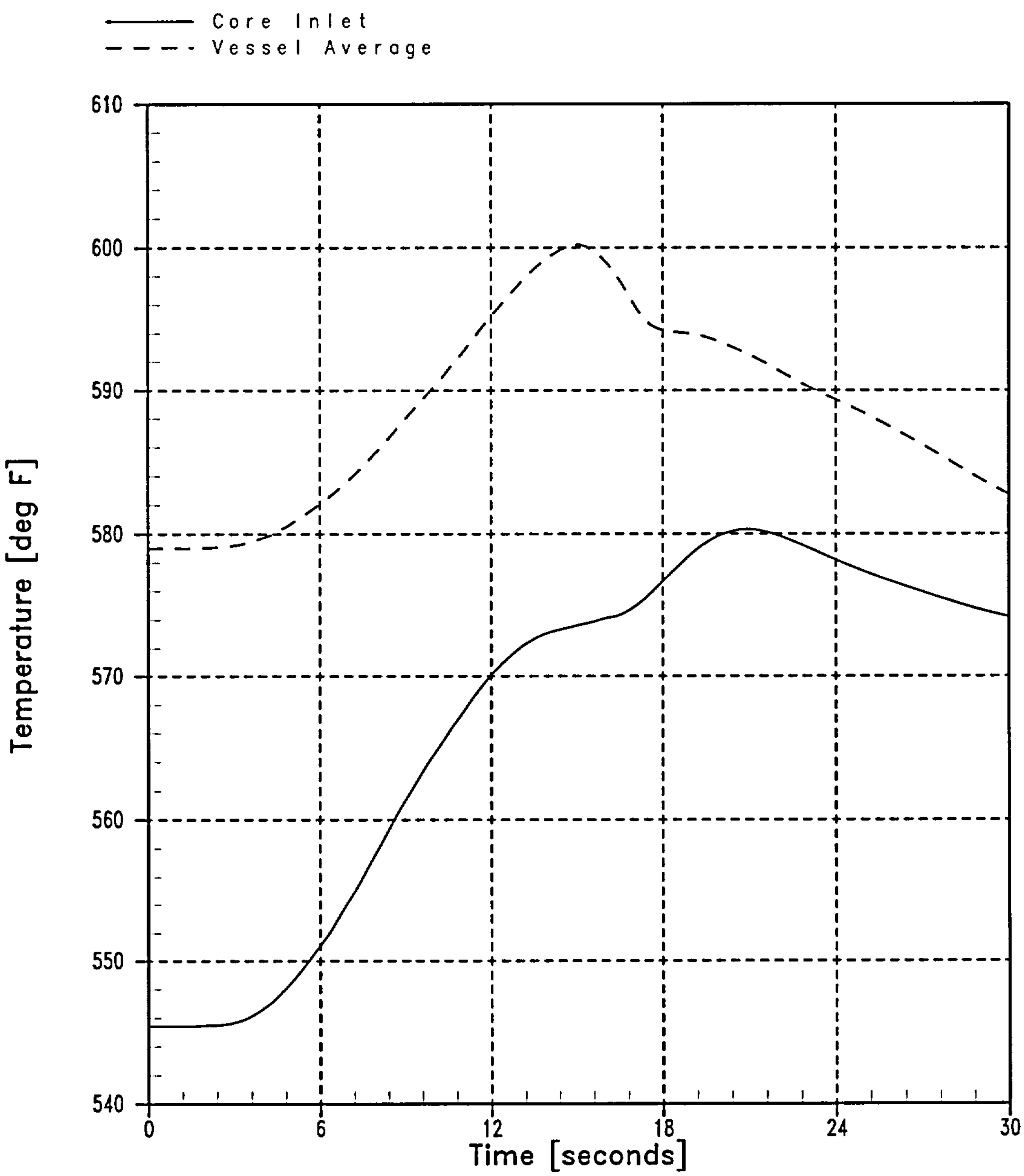

Figure 5.1.9-13 Loss of External Electrical Load with Automatic Pressure Control (MSS Overpressure Case) – Vessel Average and Core Inlet Temperatures versus Time

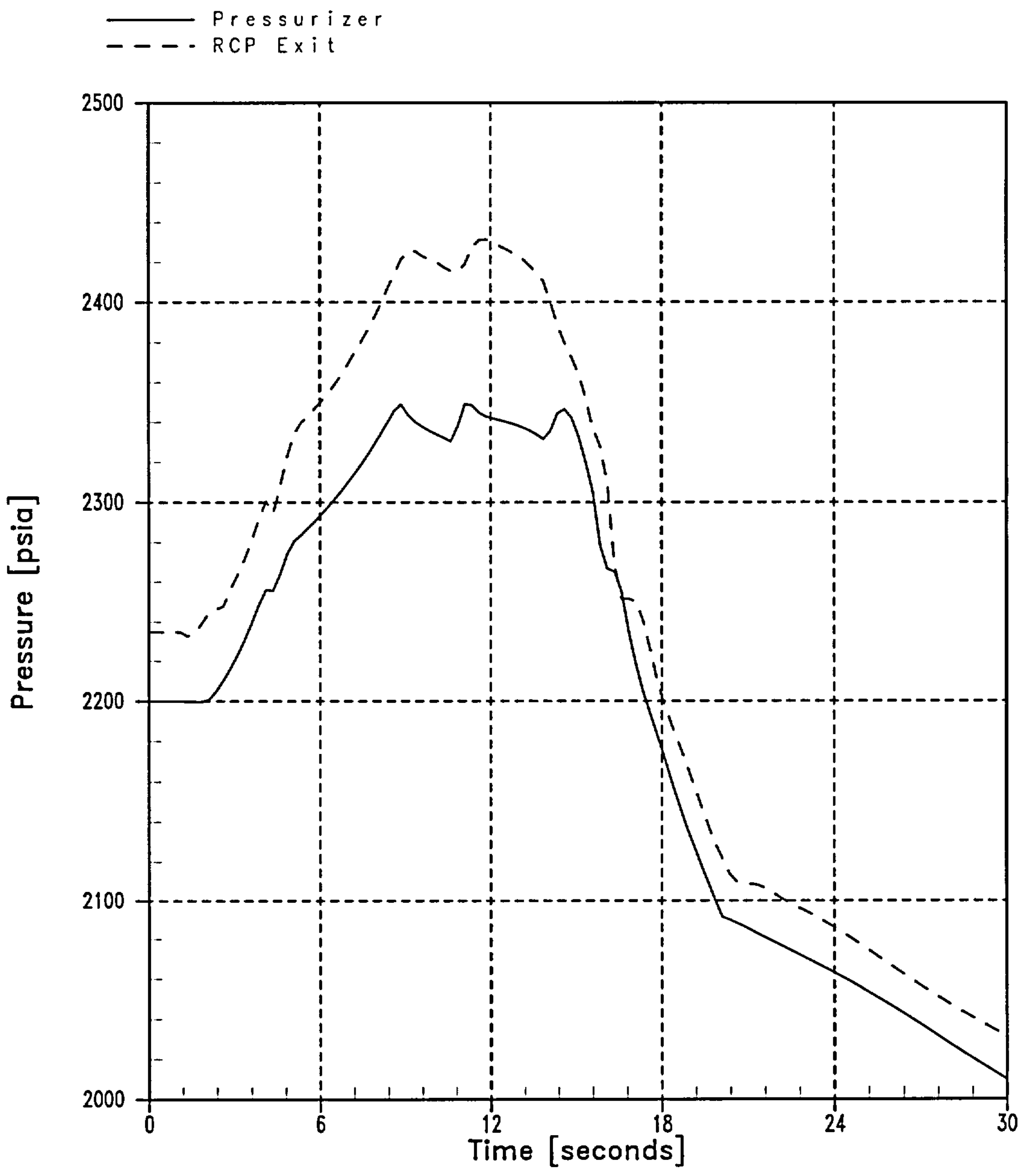

Figure 5.1.9-14 Loss of External Electrical Load with Automatic Pressure Control (MSS Overpressure Case) – Pressurizer and RCP Exit Pressures versus Time

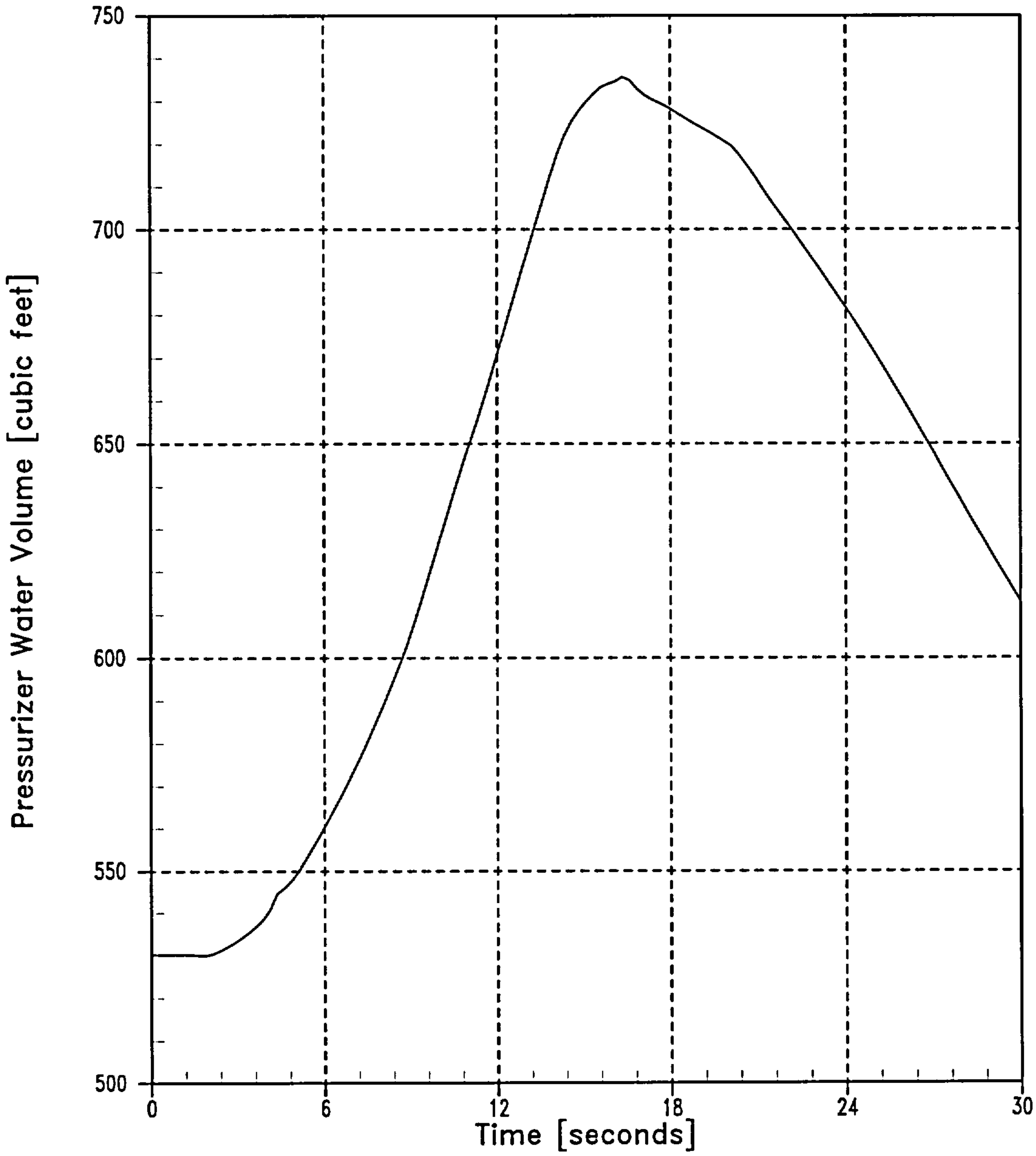

Figure 5.1.9-15 Loss of External Electrical Load with Automatic Pressure Control (MSS Overpressure Case) – Pressurizer Water Volume versus Time

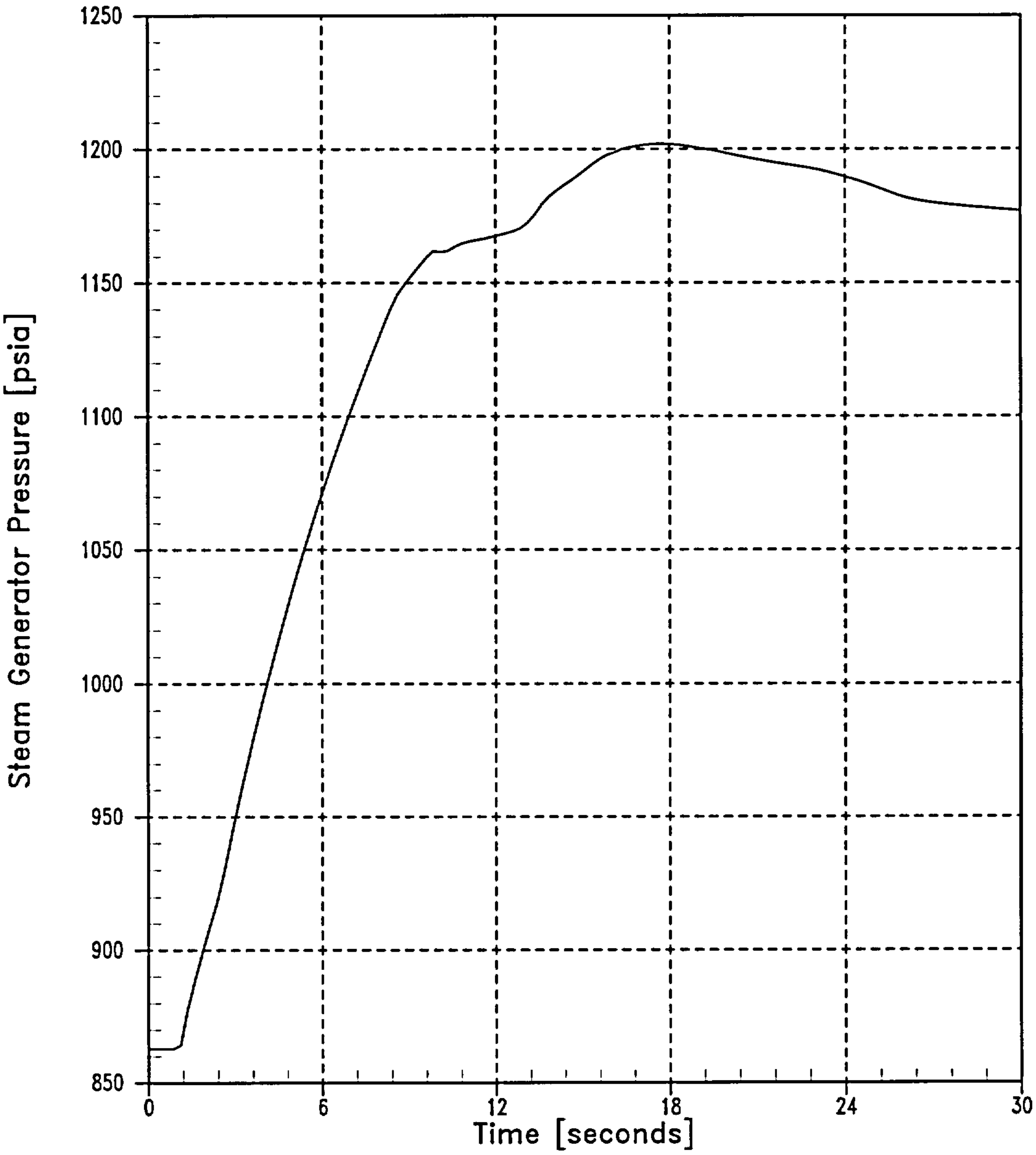

Figure 5.1.9-16 Loss of External Electrical Load with Automatic Pressure Control (MSS Overpressure Case) – Steam Generator Pressure versus Time

5.1.10 Loss of Normal Feedwater (USAR Section 14.1.10)

The Loss of Normal Feedwater analysis has been retracted.

5.1.11 Loss of AC Power to the Plant Auxiliaries (USAR Section 14.1.12)

Accident Description

A complete loss of non-emergency AC power results in the loss of all power to the plant auxiliaries; such as the RCPs or condensate pumps. 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.

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

- Plant vital instruments are supplied from emergency power sources
- Steam dump to the condenser and steam generator PORVs are unavailable. Therefore, the MSSVs lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat.
- As the no-load temperature is approached, the steam generator PORVs (or the safety valves, if the PORVs are not available) are used to dissipate the residual decay heat and maintain the plant at the hot shutdown condition.
- The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The AFW system is started automatically, as discussed in the loss-of-normal-feedwater analysis (Section 5.1.10). The TDAFWP utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven AFW pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

Upon the loss of power to the RCPs, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. Following the RCP coastdown caused by the loss of AC power, the natural circulation capability of the RCS removes residual and decay heat from the core, aided by the AFW in the secondary system.

Method of Analysis

The loss-of-all-AC-power-to-the-station-auxiliaries transient is analyzed using the RETRAN computer code. The code simulates the neutron kinetics, RCS including natural circulation, pressurizer, pressurizer relief and safety valves, pressurizer heaters, pressurizer spray, steam generators, feedwater system, and MSSVs. The code computes pertinent plant variables including steam generator mass, pressurizer water volume, and reactor coolant average temperature.

Major assumptions made in the loss of all auxiliary AC power analysis are the following:

a. The plant is initially operating at 102 percent of the 1780 MWt.

b. Reactor trip occurs on steam generator lo-lo level at 0 percent of narrow range span (NRS). Turbine trip occurs coincident with reactor trip.

c. A conservative core residual heat generation based on ANS 5.1-1979 decay heat plus 2 sigma is assumed (Reference 5-3).

d. The amount of heat transfer assumed to occur in the steam generators following the RCP coastdown is based on RCS natural circulation conditions.

e. One minute after the lo-lo steam generator water level setpoint is reached, the AFW system provides 176 gpm of flow split equally between the two steam generators (equal split is the limiting case). The AFW flow assumption is conservative with respect to the worst-case scenario for available AFW flow during a loss-of-all-auxiliary-AC-power event, as the TDAFWP (single failure) and the second MDAFWP are assumed to be unavailable. The AFW enthalpy is assumed to be 90.8 BTU/lbm (120°F and 1100 psia).

f. Secondary-system steam relief is achieved through the MSSVs, which include a +2-percent setpoint tolerance, a 5 psi ramp for the valve to pop open, and a pressure difference from the steam generator to the safety valves of approximately 42 psi. Steam relief through the steam generator PORVs or condenser dump valves is assumed unavailable.

g. The initial reactor coolant average temperature is assumed to be 6°F lower than the nominal value of 573.0°F because this results in a greater expansion of the RCS water during the transient, thus, resulting in a higher pressurizer water level.

h. The initial pressurizer pressure is assumed to be 50 psi above its nominal value.

i. Nominal reactor control systems are not assumed to function. However, the pressurizer PORVs, pressurizer heaters, and pressurizer spray are assumed to operate normally. This assumption results in a conservative transient with respect to the peak pressurizer water level. If these control systems did not operate, the pressurizer safety valves would maintain peak RCS pressure around the actuation setpoint throughout the transient.

The assumptions used in the analysis are similar to the loss of normal feedwater (Section 5.1.10) except that power is assumed to be lost to the reactor coolant pumps due to the reactor trip.

Results

Figures 5.1.11-1 through 5.1.11-6 show the significant plant responses following a loss-of-all-AC-power-to-the-station-auxiliaries event. The calculated sequence of events is listed in Table 5.1.11-1.

The first few seconds after the loss of power to the RCPs will closely resemble the simulation of the complete loss-of-flow accident (USAR Section 14.1.8), where core damage due to rapidly increasing core temperature is prevented by promptly tripping the reactor.

After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core. The peak pressurizer water volume is 698 ft^3, which is less than the limit of 1010.10 ft^3. The maximum steam generator pressure calculated was less than 110 percent of the design pressure of 1085 psig. Also, the analysis shows that the RCS overpressurization limit is not challenged during this transient. However, note that the pressurizer PORVs are assumed to be operable so as to maximize the potential for pressurizer filling. This event is bounded by the loss of external electrical load (Section 5.1.9) with respect to peak RCS and MSS pressures.

The LOFTRAN code results show that the reactor coolant natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

Conclusions

The results of the analysis show that a loss of all AC power to the station auxiliaries does not adversely affect the core, the RCS, or the MSS. The AFW capacity is sufficient to dissipate core residual heat. Consequently, reactor coolant is not relieved through the pressurizer relief or safety valves.

Table 5.1.11-1 Sequence of Events – Loss of AC Power to the Plant Auxiliaries	
Event	**Time (seconds)**
Main Feedwater Flow Stops	20
Lo-Lo Steam Generator Water Level Trip Setpoint Reached	54.7
Rods Begin to Drop	56.2
RCPs Begin to Coast Down	58.2
Two Steam Generators Begin to Receive Auxiliary Feedwater from One Motor-Driven AFW Pump	116.2
Peak Water Level in the Pressurizer Occurs	4235
Core Heat Decreases to Auxiliary Feedwater Heat Removal Capacity	~4300
Peak Pressurizer Water Volume	698 ft^3
Pressurizer Water Volume Limit	1010.1 ft^3

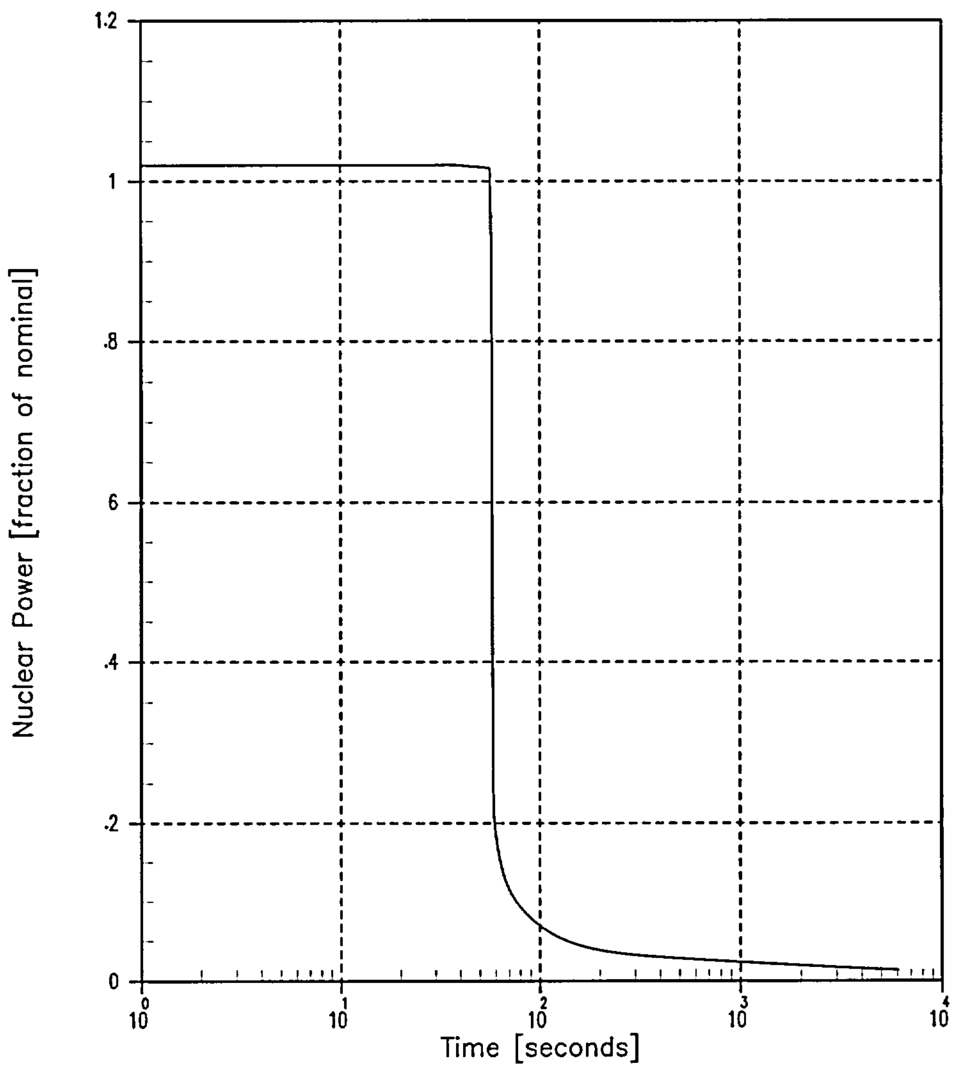

Figure 5.1.11-1 Loss of AC Power to the Plant Auxiliaries* – Nuclear Power

*Non-emergency AC power to station auxiliaries is lost following reactor trip.

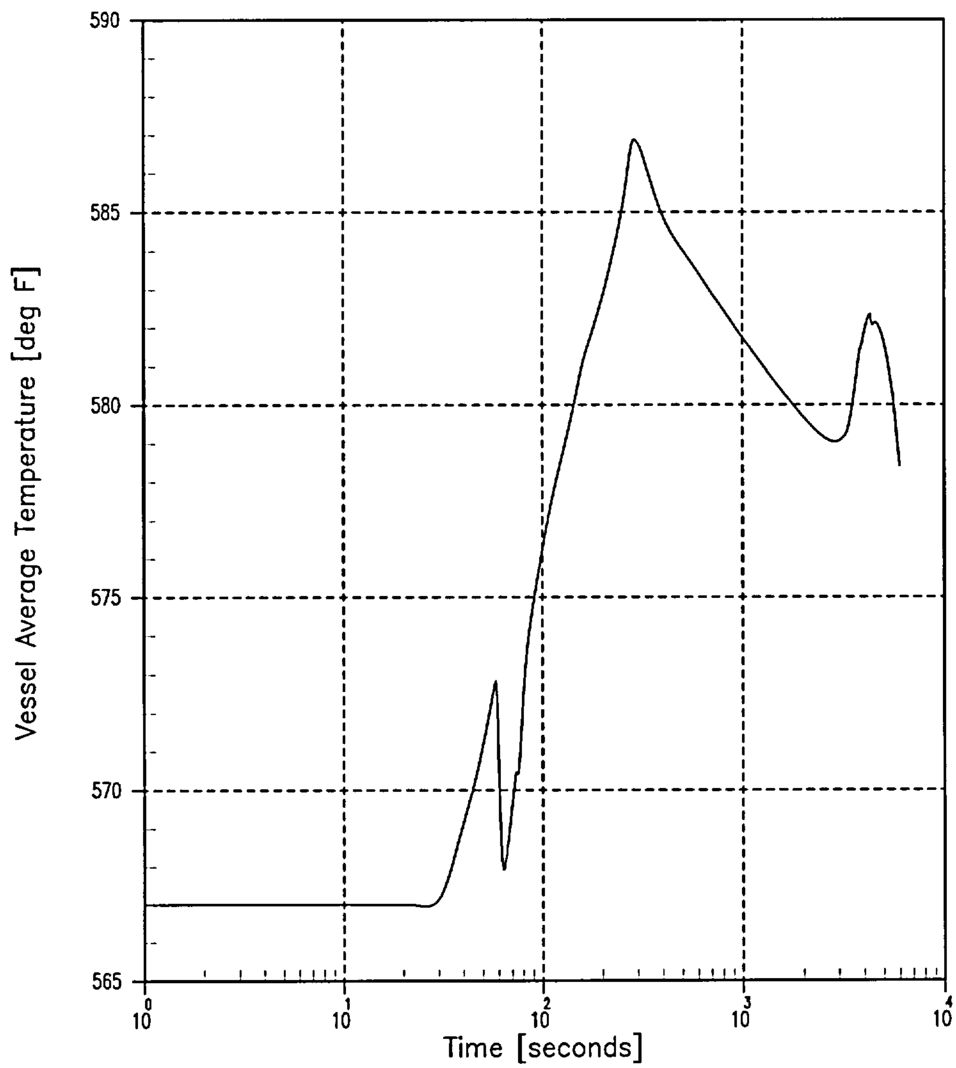

Figure 5.1.11-2 Loss of AC Power to the Plant Auxiliaries* – Vessel Average Temperature

*Non-emergency AC power to station auxiliaries is lost following reactor trip.

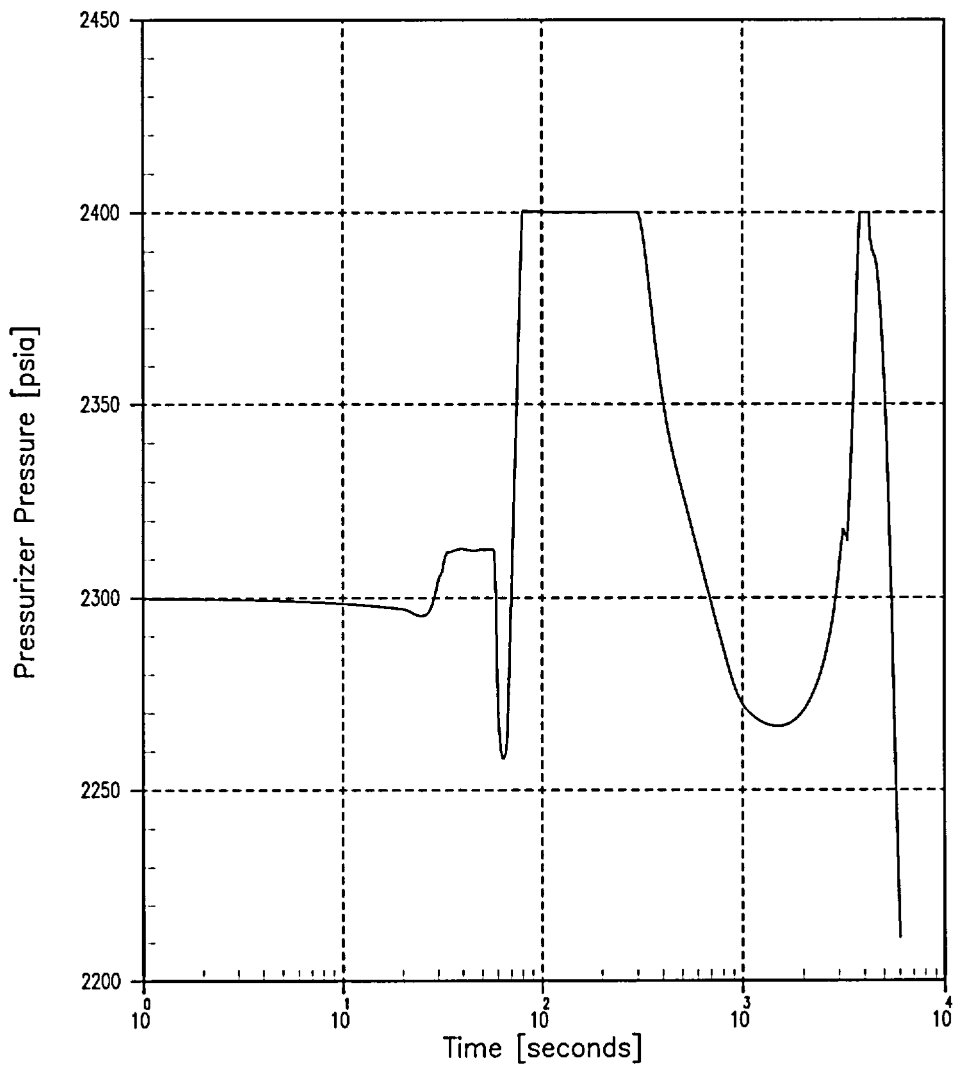

Figure 5.1.11-3 Loss of AC Power to the Plant Auxiliaries* – Pressurizer Pressure

*Non-emergency AC power to station auxiliaries is lost following reactor trip.

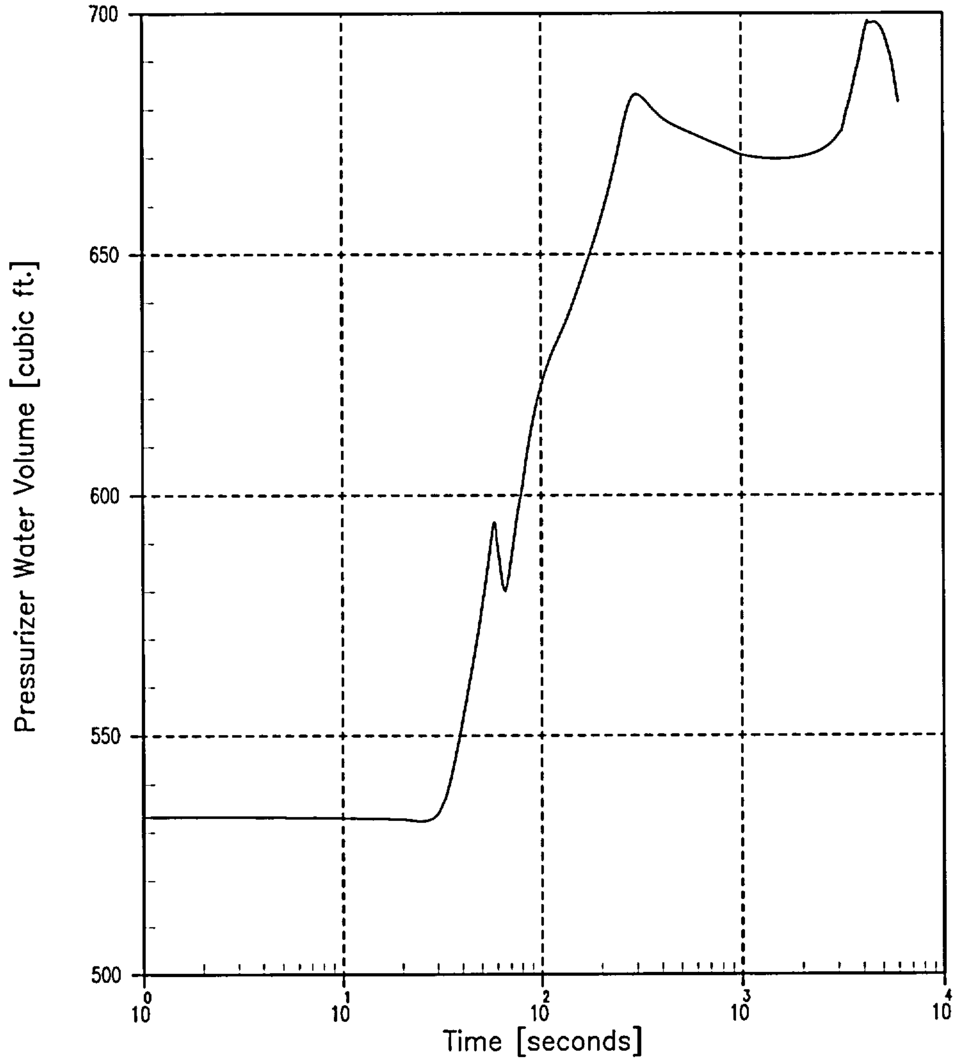

Figure 5.1.11-4 Loss of AC Power to the Plant Auxiliaries* – Pressurizer Water Volume

*Non-emergency AC power to station auxiliaries is lost following reactor trip.

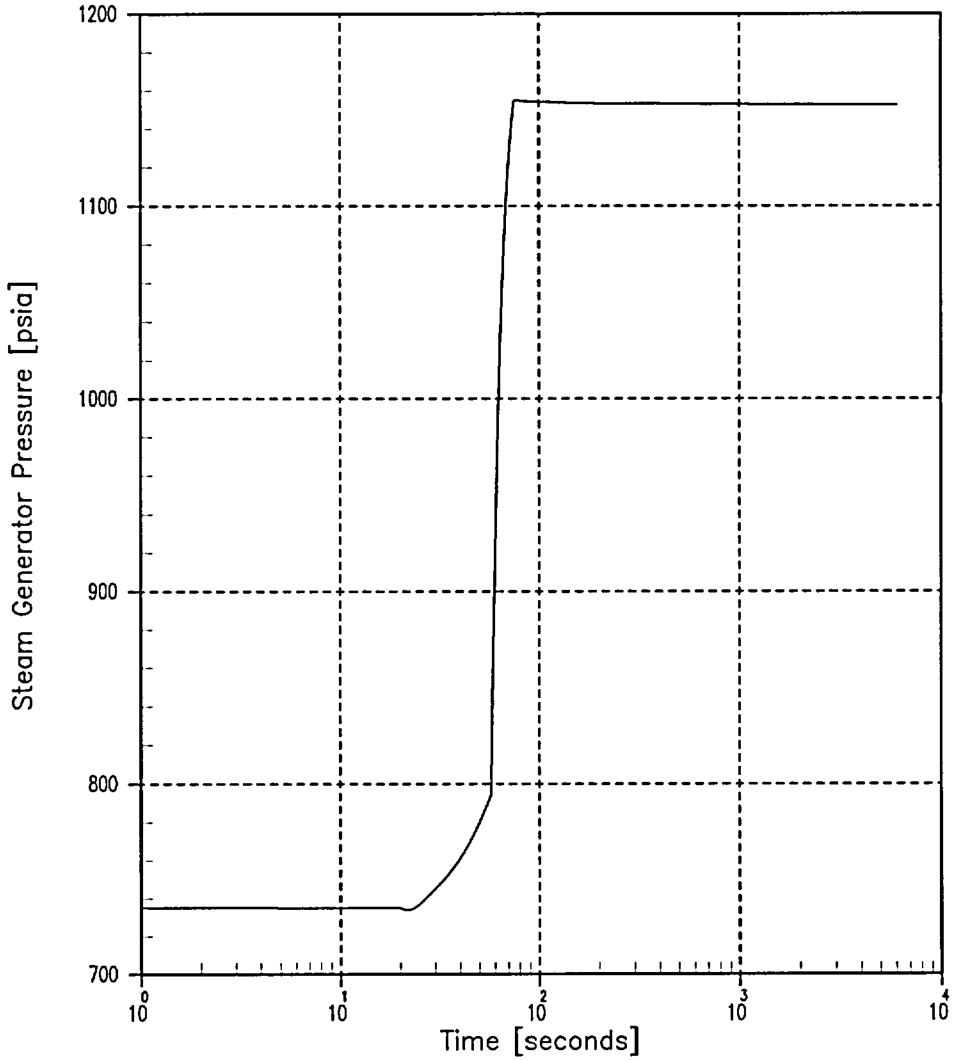

Figure 5.1.11-5 Loss of AC Power to the Plant Auxiliaries* – Steam Generator Pressure

*Non-emergency AC power to station auxiliaries is lost following reactor trip.

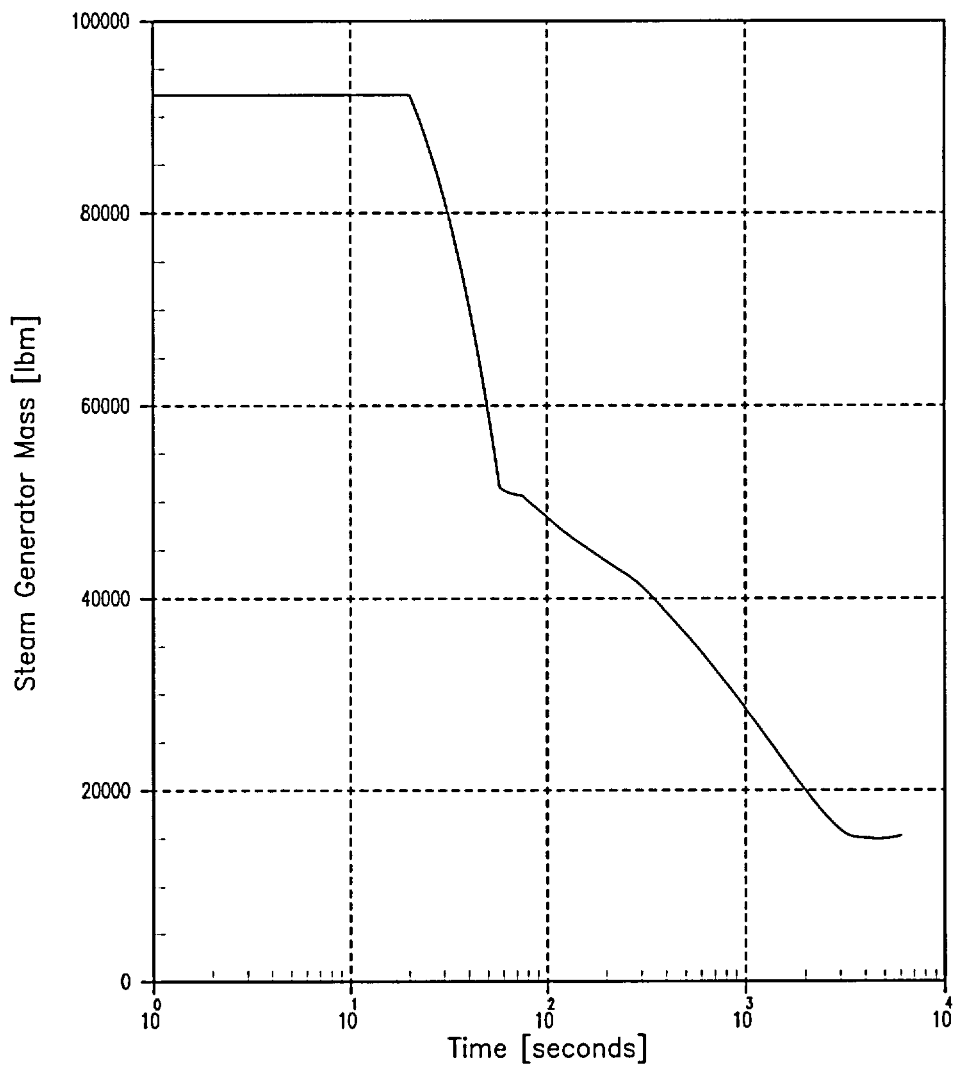

Figure 5.1.11-6 Loss of AC Power to the Plant Auxiliaries* – Steam Generator Mass

*Non-emergency AC power to station auxiliaries is lost following reactor trip.

5.1.12 Steam Line Break (USAR Section 14.2.5)

Accident Description

A steam line break transient would result in an uncontrolled increase in steam flow release from the steam generators, with the flow decreasing as the steam pressure drops. This steam flow release increases the heat removal from the RCS, which decreases the RCS temperature and pressure. With the existence of a negative MTC, the RCS cooldown results in a positive reactivity insertion, and consequently a reduction of the core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, the possibility is increased that the core will become critical and return to power. A return to power following a steam line break is a concern with the high-power peaking factors that may exist when the most reactive RCCA is stuck in its fully withdrawn position. Following a steam line break, the core is ultimately shut down by the boric acid injected into the RCS by the emergency core cooling system (safety injection).

The steam line break analysis discussed herein was performed to demonstrate that there is no consequential damage to the primary system and that the core remains in place and intact. This analysis is known as the steam line break core response analysis. Assuming the most reactive RCCA is stuck in its fully withdrawn position, and applying the most limiting single failure of one safety injection train, steam line break core response cases were examined with and without offsite power available. Although DNB and fuel cladding damage are not necessarily unacceptable consequences of a steam line break transient, the analysis described herein demonstrates that there is no consequential damage to the primary system, and that the core remains in place and intact, by showing that the DNB design basis is satisfied following a steam line break.

The systems and components that provide the necessary protection against a steam line break are listed as follows.

- Safety injection system actuation by any of the following:
  - Two-out-of-three pressurizer pressure channels with low signals
  - Two-out-of-three steam line pressure channels on either loop with lo-lo signals
  - Two-out-of-three containment pressure channels with high signals

- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring from the receipt of the safety injection signal.

- Redundant isolation of the main feedwater lines; sustained high feedwater flow would cause additional cooldown. In addition to normal control action that isolates main feedwater following a reactor trip, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

- Closure of the main steam isolation valves (MSIVs). These valves are designed to close within five seconds after receipt of any of the following:
  - A safety injection signal coincident with one-out-of-two steam flow channels on Loop A with a hi-hi signal (isolates Loop A)
  - A safety injection signal coincident with one-out-of-two steam flow channels on Loop B with a hi-hi signal (isolates Loop B)
  - A safety injection signal coincident with one-out-of-two steam flow channels on Loop A with a high signal AND two-out-of-four T_{avg} channels with lo-lo signals (isolates Loop A)
  - An safety injection signal coincident with one-out-of-two steam flow channels on Loop B with a high signal AND two-out-of-four T_{avg} channels with lo-lo signals (isolates Loop B)
  - Two-out-of-three containment pressure channels with hi-hi signals

The MSS conducts steam in 30-inch piping from each of the two steam generators within the reactor containment, through a swing-disc type isolation valve (MSIV) and a swing-disc type non-return check valve to the turbine stop and control valves. The isolation and non-return check valves are located outside of the containment, and an equalizing line near the turbine interconnects the two steam lines. The non-return check valves prevent reverse flow of steam. Therefore, if a break occurs between a non-return check valve and a steam generator, only the affected steam generator would blow down. The steam generator blowdown from a steam line break located downstream of a non-return check valve would be terminated upon closure of both MSIVs.

Each main steam line contains a 16-inch diameter venturi-type flow restrictor located upstream of the MSIV and inside containment. These flow restrictors are used to measure the steam flow from each steam generator. Additional flow restrictors that are an integral part of the steam generator outlet nozzles serve to limit the steam release rate during a steam line break transient. The nozzle flow restrictors limit the effective maximum steam line break size to 1.4 ft^2 per steam generator.

Method of Analysis

The analysis of the steam line break transient has been performed to demonstrate that the DNB design basis is satisfied. This is accomplished by showing that the calculated minimum DNBR is greater than the safety analysis limit DNBR of 1.472 (W-3 low pressure DNB correlation limit). The overall analysis process is described as follows.

Using the RETRAN code (Reference 5-5), transient values of key plant parameters identified as statepoints (core average heat flux, core pressure, core inlet temperature, RCS flow rate, and core boron concentration) were calculated first. Next, the advanced nodal code (ANC) core design code (Reference 5-7) was used to:

- Evaluate the nuclear response to the RCS cooldown so as to justify the RETRAN transient prediction of the average core power/reactivity

- Determine the peaking factors associated with the return to power in the region of the stuck RCCA

Finally, using the RETRAN-calculated statepoints and the ANC-calculated peaking factors, the detailed thermal and hydraulic computer code VIPRE (Reference 5-9) was used to calculate the minimum DNBR based on the W-3 DNB correlation.

The following assumptions were made in the analysis of the main steam line break:

a. A hypothetical double-ended rupture (DER) of a main steam line was postulated at HZP/hot shutdown conditions. The maximum break size is effectively limited to the flow area of the steam generator outlet nozzle flow restrictors (1.4 ft^2 per steam generator). The assumed conditions correspond to a subcritical reactor, an initial vessel average temperature at the no-load value of 547°F, and no core decay heat. These conditions are conservative for a steam line break transient because the resultant RCS cooldown does not have to remove any latent heat. Also, the steam generator water inventory is greatest at no-load conditions, which increases the capability for cooling the RCS.

b. Two DER cases were considered: one with offsite power and one with a loss-of-offsite-power. The difference being that both RCPs begin coasting down three seconds after the steam line break initiation for the case without offsite power. Note that steam line break transients associated with the inadvertent opening of a steam dump or relief valve were not analyzed because the resultant RCS cooldown, and thus the minimum DNBR, would be less limiting compared to the DER cases.

c. Perfect moisture separation within the steam generators was conservatively assumed.

d. An end-of-life shutdown margin of 1.3-percent Δk/k corresponding to no-load, equilibrium xenon conditions, with the most reactive RCCA stuck in its fully withdrawn position was assumed. The stuck RCCA was assumed to be in the core location exposed to the greatest cooldown; that is, related to the faulted loop. The reactivity feedback model included a positive moderator density coefficient (MDC) corresponding to an end-of-life rodded core with the most reactive RCCA in its fully withdrawn position. The variation of the MDC due to changes in temperature and pressure was accounted for in the model.
Figure 5.1.12-1 presents the k_{eff} versus temperature relationship at 1050 psia corresponding to the assumed negative MTC plus the Doppler temperature feedback effect.

   The reactivity and power predicted by RETRAN were compared to those predicted by the ANC core design code. The ANC core analysis considered the following:

   - Doppler reactivity feedback from the high fuel temperature near the stuck RCCA
   - Moderator feedback from the high water enthalpy near the stuck RCCA
   - Power redistribution effects
   - Non-uniform core inlet temperature effects

   The ANC core analysis confirmed that the RETRAN-predicted reactivity is acceptable.

e. Assuming no frictional losses, the Moody critical flow curve was applied to conservatively maximize the break flow rate.

f. The non-return check valves were neglected to conservatively allow blowdown from both steam generators up to the time of MSIV closure. This assumption was made along with not crediting containment protection signals, to assure that any postulated break location or single failure assumption, is bounded by a single analysis.

g. The closure of the MSIV of the intact/unfaulted loop was conservatively modeled to be complete at 7.6 seconds after receipt of a safety injection signal due to the coincidence of a hi-hi steam flow rate (~200 percent of nominal full-power steam flow) signal and a lo-lo steam line pressure (495 psia) signal from the same loop.

h. The safety injection pumps were assumed to provide flow to the RCS at 25 seconds after receipt of a safety injection signal for the case with offsite power available, and at 30 seconds after a safety injection signal for the case without offsite power available. These delays account for signal processing and pump startup delays, and, as applicable, diesel generator startup time.

i. The minimum capability for the injection of highly concentrated boric acid solution, corresponding to the most restrictive single active failure in the SIS, was assumed. The assumed safety injection flow (see Figure 5.1.12-2) corresponds to the operation of one high-head safety injection pump. Boric acid solution from the refueling water storage tank (RWST), with a minimum concentration of 2400 ppm and a minimum temperature of 40°F, was the assumed source of the safety injection flow. The safety injection lines downstream of the RWST were assumed to initially contain unborated water to conservatively maximize the time it takes to deliver the highly concentrated RWST boric acid solution to the reactor coolant loops.

j. The safety injection accumulator tanks (one per loop) provide a passive injection of up to 2500 ft^3 of borated water into the RCS. The accumulators were assumed to have a minimum boron concentration of 1850 ppm, a minimum temperature of 40°F, and an initial gas pressure of 714.7 psia.

k. Main feedwater flow equal to the nominal (100-percent power) value was assumed to initiate coincident with the postulated break, and was maintained until feedwater isolation occurs. The feedwater isolation was assumed to be complete at 85.7 seconds after the steam line pressure in the faulted loop reaches the lo-lo setpoint signal that generates the safety injection signal.

l. A minimum SGTP level of 0 percent was assumed to maximize the cooldown of the RCS.

m. Maximum (1200 gpm) auxiliary feedwater at a minimum temperature of 35°F was assumed to initiate coincident with the postulated break to maximize the cooldown of the RCS.

Results

The results of the statepoint evaluation demonstrate that both cases analyzed meet the applicable DNBR acceptance criterion. The most limiting case is the case in which offsite power was assumed to be available. The time sequence of events for each case is presented in Table 5.1.12-1.

Double-Ended Rupture With Offsite Power Available

Figures 5.1.12-3 through 5.1.12-10 show the steam pressure, steam flow, pressurizer pressure, pressurizer water volume, reactor vessel inlet temperature, core heat flux, core boron concentration, and core reactivity following a double-ended rupture of a main steam line at initial no-load conditions with offsite power available (full reactor coolant flow). The effective break size was limited to 1.4 ft^2 per steam generator by the flow area of the steam generator outlet nozzles, and both steam generators were assumed to discharge through the break until steam line isolation had occurred. It is important to note that at approximately 102 seconds the faulted loop (Loop 1) break (outlet nozzle) mass flow rate spikes (see Figure 5.1.12-4) as a result of the upper steam generator node becoming water-solid. This spike occurs after the peak heat flux is reached and does not invalidate the results.

Double-Ended Rupture Without Offsite Power Available

Figures 5.1.12-11 through 5.1.12-18 show the steam pressure, steam flow, pressurizer pressure, pressurizer water volume, reactor vessel inlet temperature, core heat flux, core boron concentration, and core reactivity following a double-ended rupture of a main steam line at initial no-load conditions with a loss-of-offsite-power (RCPs begin coasting down three seconds after break initiation). The effective break size was limited to 1.4 ft^2 per steam generator by the flow area of the steam generator outlet nozzles, and both steam generators were assumed to discharge through the break until steam line isolation had occurred.

Conclusions

The main steam line break transient was conservatively analyzed with respect to the reactor core response. Key analysis assumptions were made to conservatively maximize the cooldown of the RCS, so as to maximize the positive reactivity insertion, and thus maximize the peak return to power. Other key assumptions include: end-of-life shutdown margin with the most-reactive RCCA stuck in its fully withdrawn position, maximum delays in actuating engineered safeguard features such as safety injection, main steam isolation and feedwater isolation, and minimum safety injection flow with a minimum boron concentration.

A DNBR statepoint analysis was performed for two DER cases: one with offsite power and one with a loss-of-offsite power. The case with offsite power available—that is, the case with full reactor coolant flow—was found to be the limiting case. The minimum DNBR for each case was determined to be greater than the DNBR safety analysis limit, and thus the DNBR design basis is met.

Table 5.1.12-1 Steam Line Break Analysis Assumptions and Sequence of Events		
	Double-Ended Rupture with Offsite Power	**Double-Ended Rupture Without Offsite Power**
Steam Generator Model	54F	54F
Loss-of-Offsite Power	No	Yes
Time of Main Steam Line Rupture, seconds	0.01	0.01
Time Maximum AFW (600 gpm per loop) Initiated, seconds	0.01	0.01
Time Unfaulted Loop Steam Flow Reaches Hi-Hi Setpoint (~200% of Nominal), seconds	0.71	0.71
Time Steam Pressure Reaches Lo-Lo Setpoint (495 psia)		
- Faulted Loop, seconds	1.44	1.44
- Unfaulted Loop, seconds	2.01	2.01
Time of SI Signal Actuation Due to Coincidence of Hi-Hi Steam Flow and Lo-Lo Steam Pressure, seconds	2.72	2.72
Time of RCP Trip (Loss-of-Offsite-Power), seconds	N/A	3.00
Time of Steam Line Isolation (MSIV Closure) Due to SI Signal Actuation, seconds	10.22	10.22
Time Core Returns to Criticality, seconds	22.75	28.25
Time SI Pump Reaches Full Speed, seconds	27.72	32.72
Time Accumulator Tanks Begin Injecting into RCS, seconds	53.25	79.75
Time of Peak Heat Flux, seconds	56.50	132.75
Time of Minimum DNBR, seconds	56.25	~132.75
Time of Feedwater Isolation (Main Feedwater Isolation Valve Closure) Due to SI Signal Actuation, seconds	87.82	87.82
Peak Heat Flux, fraction of nominal	0.288	0.096
Minimum DNBR	2.29	Bounded by other case

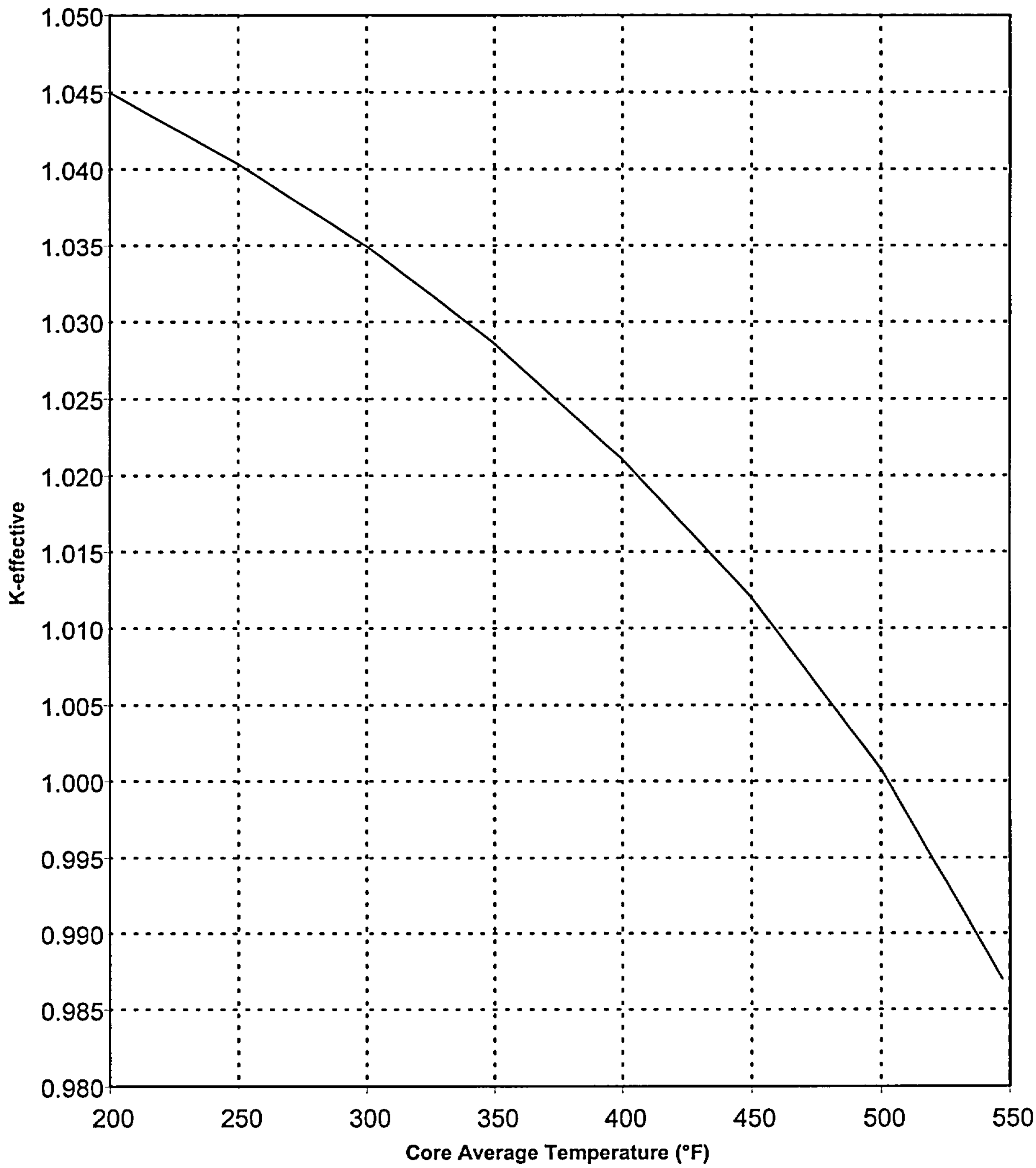

Figure 5.1.12-1 Variation of K_{eff} with Core Temperature

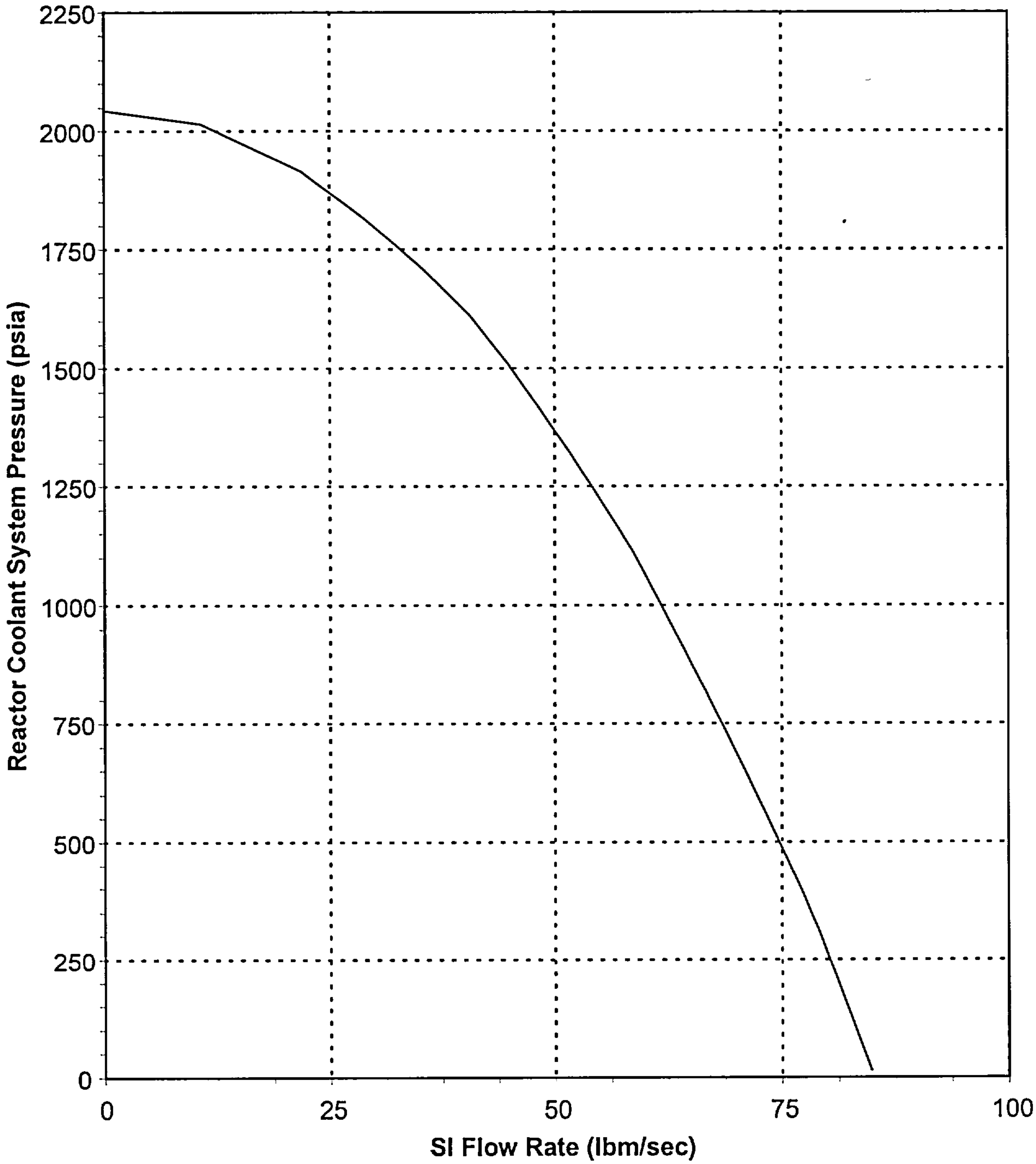

Figure 5.1.12-2 Safety Injection Curve

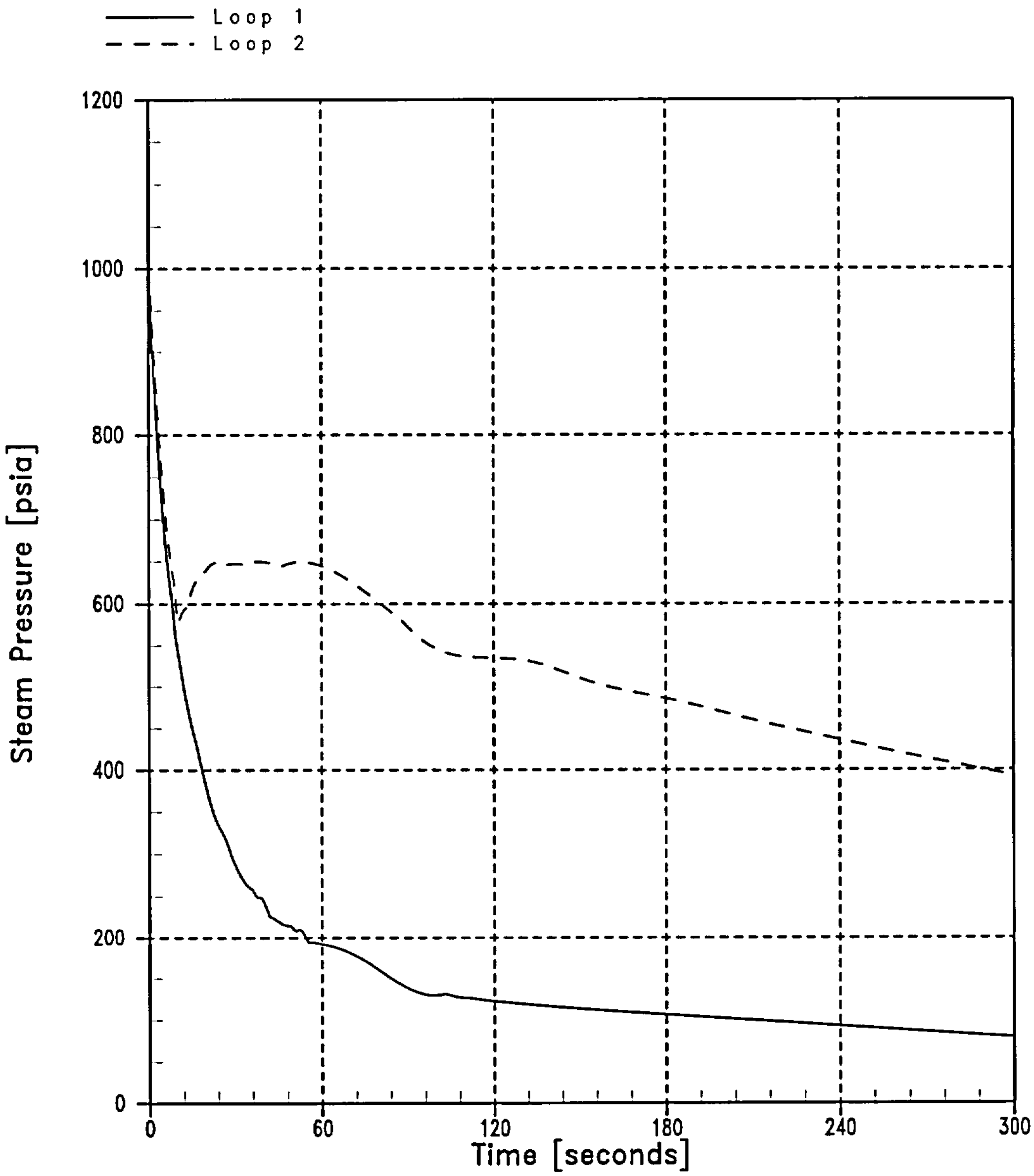

Figure 5.1.12-3 Main Steam Line Break with Offsite Power – Steam Generator Steam Pressure versus Time

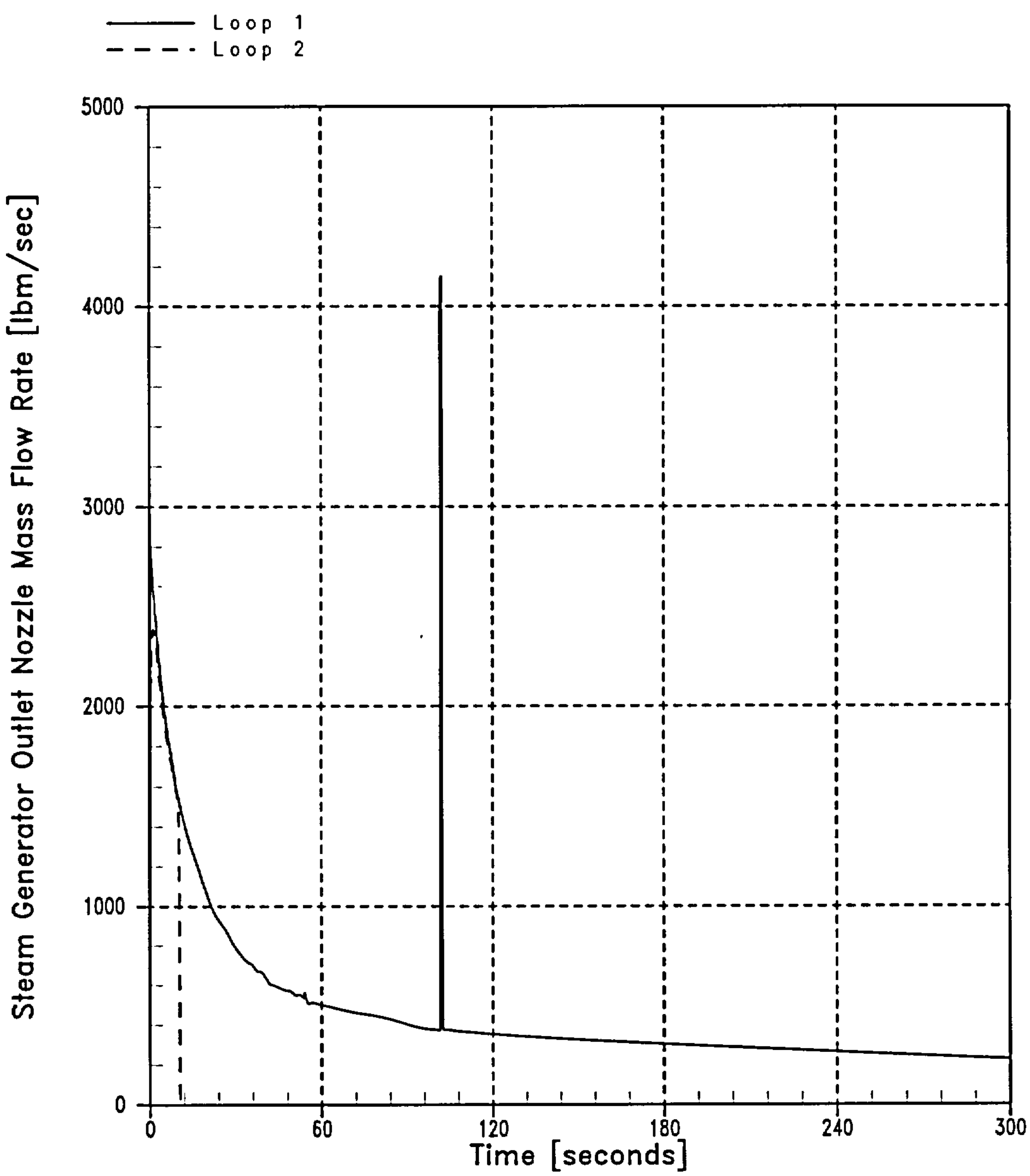

Figure 5.1.12-4 Main Steam Line Break with Offsite Power – Steam Generator Outlet Nozzle Mass Flow Rate versus Time

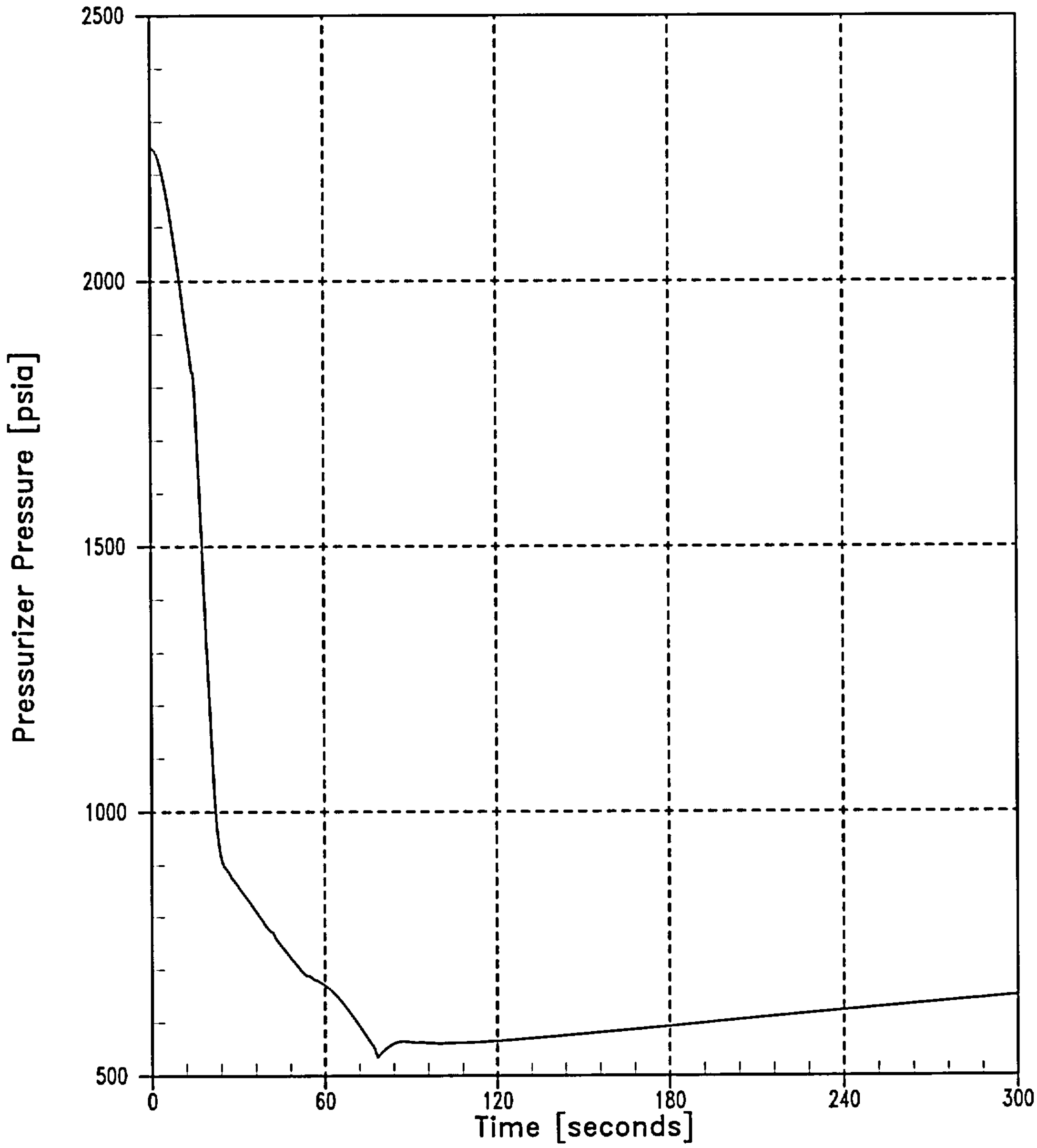

Figure 5.1.12-5 Main Steam Line Break with Offsite Power – Pressurizer Pressure versus Time

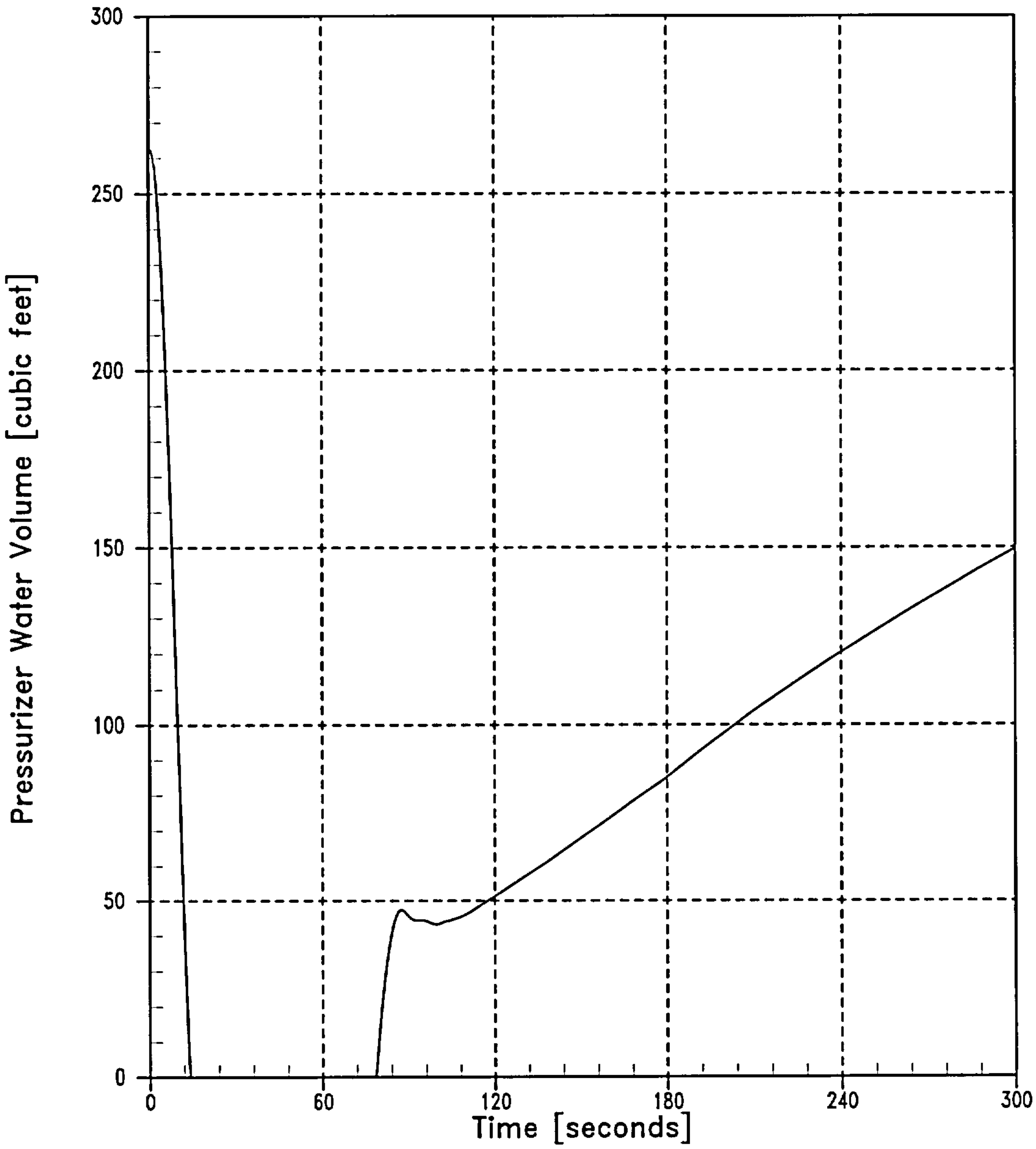

Figure 5.1.12-6 Main Steam Line Break with Offsite Power – Pressurizer Water Volume versus Time

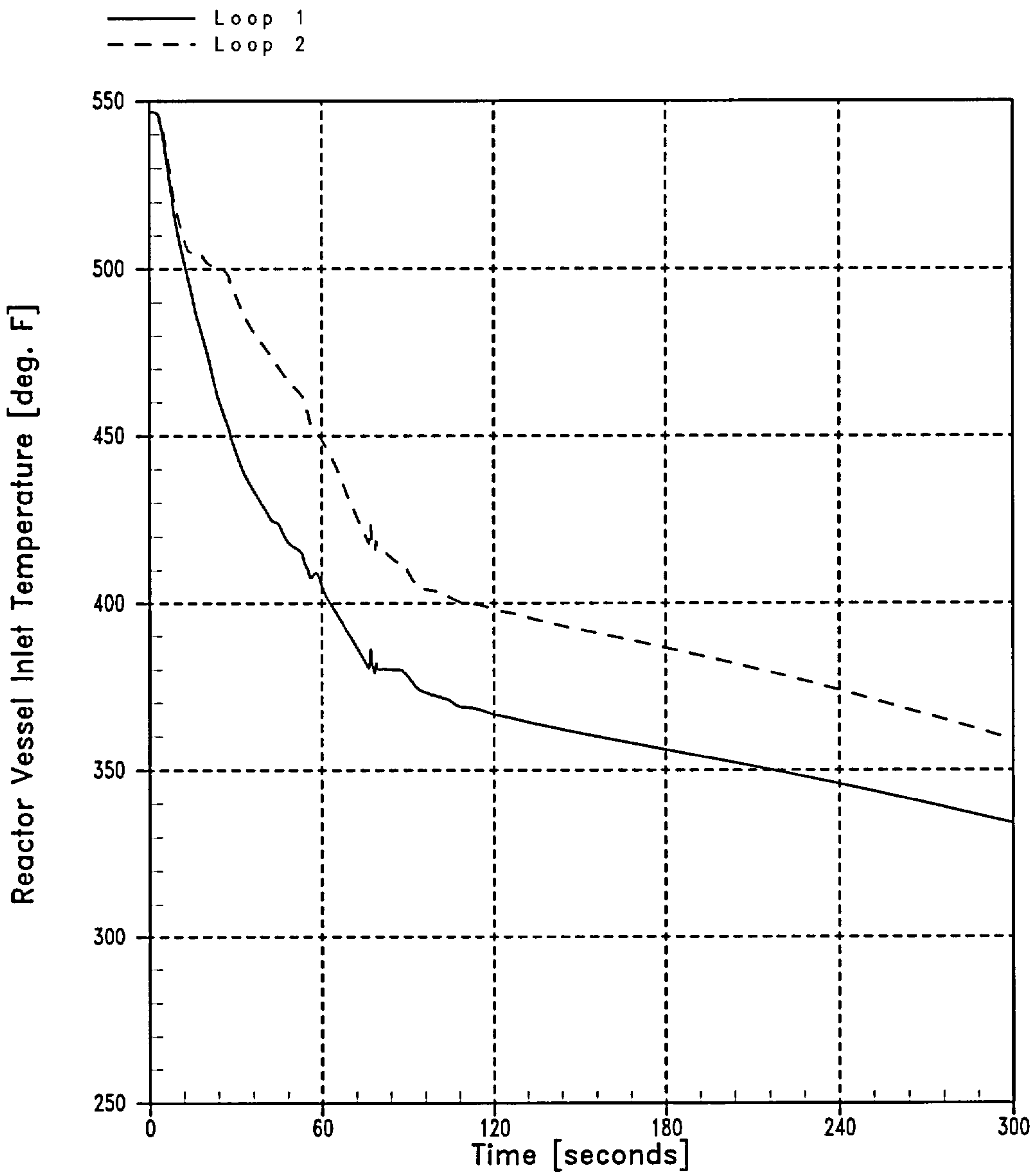

Figure 5.1.12-7 Main Steam Line Break with Offsite Power – Reactor Vessel Inlet Temperature versus Time

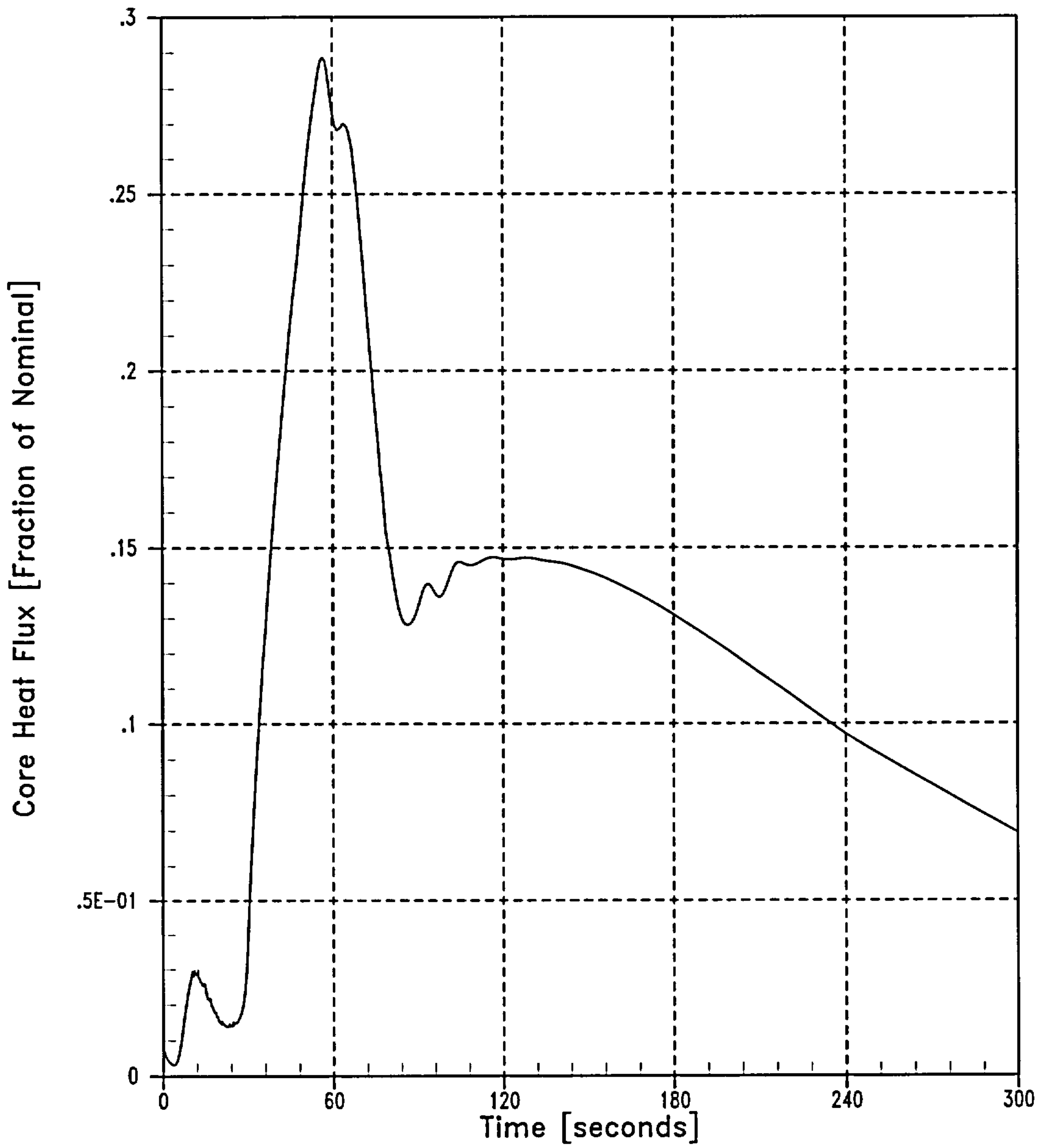

Figure 5.1.12-8 Main Steam Line Break with Offsite Power – Core Heat Flux versus Time

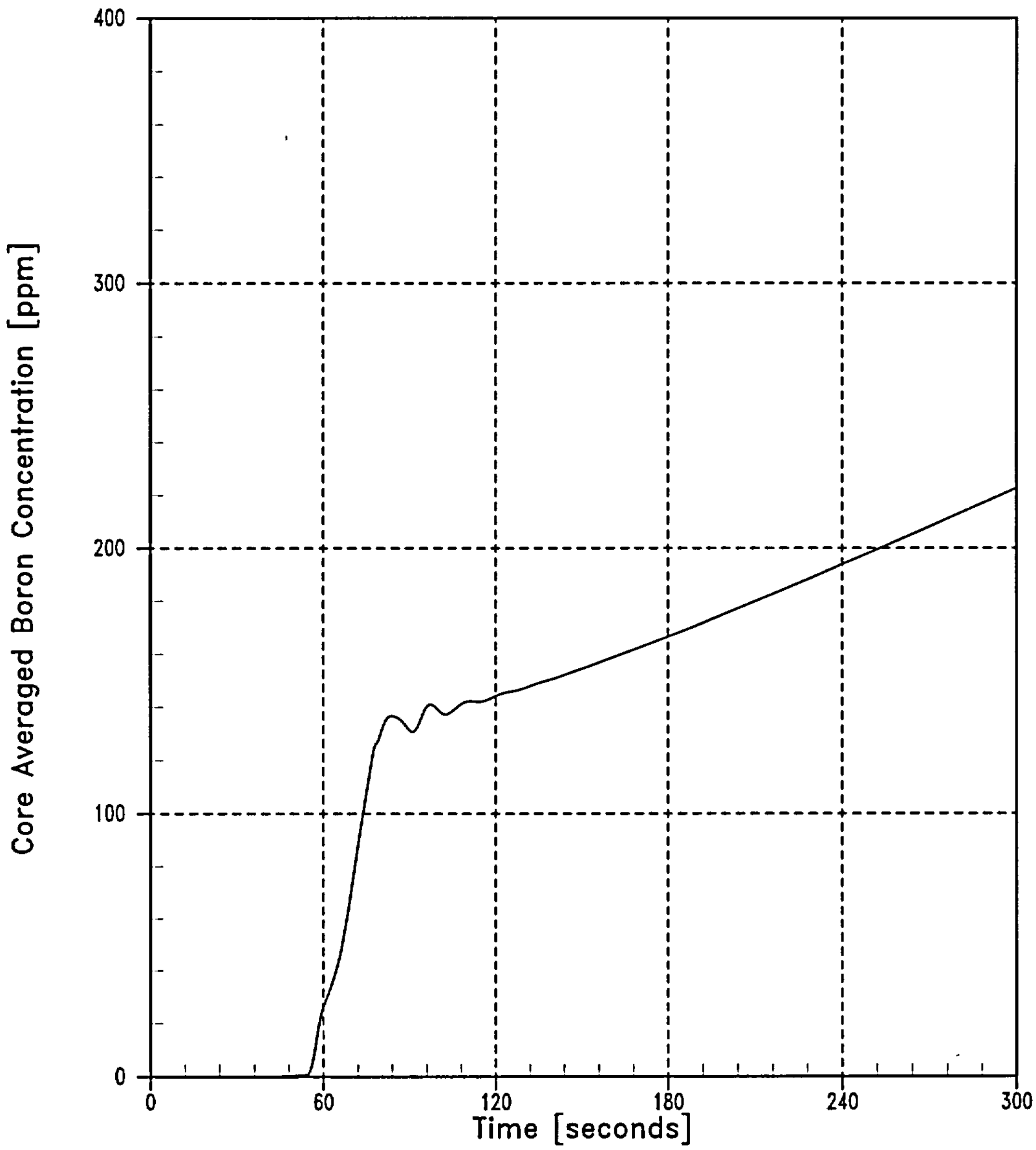

Figure 5.1.12-9 Main Steam Line Break with Offsite Power – Core Averaged Boron Concentration versus Time

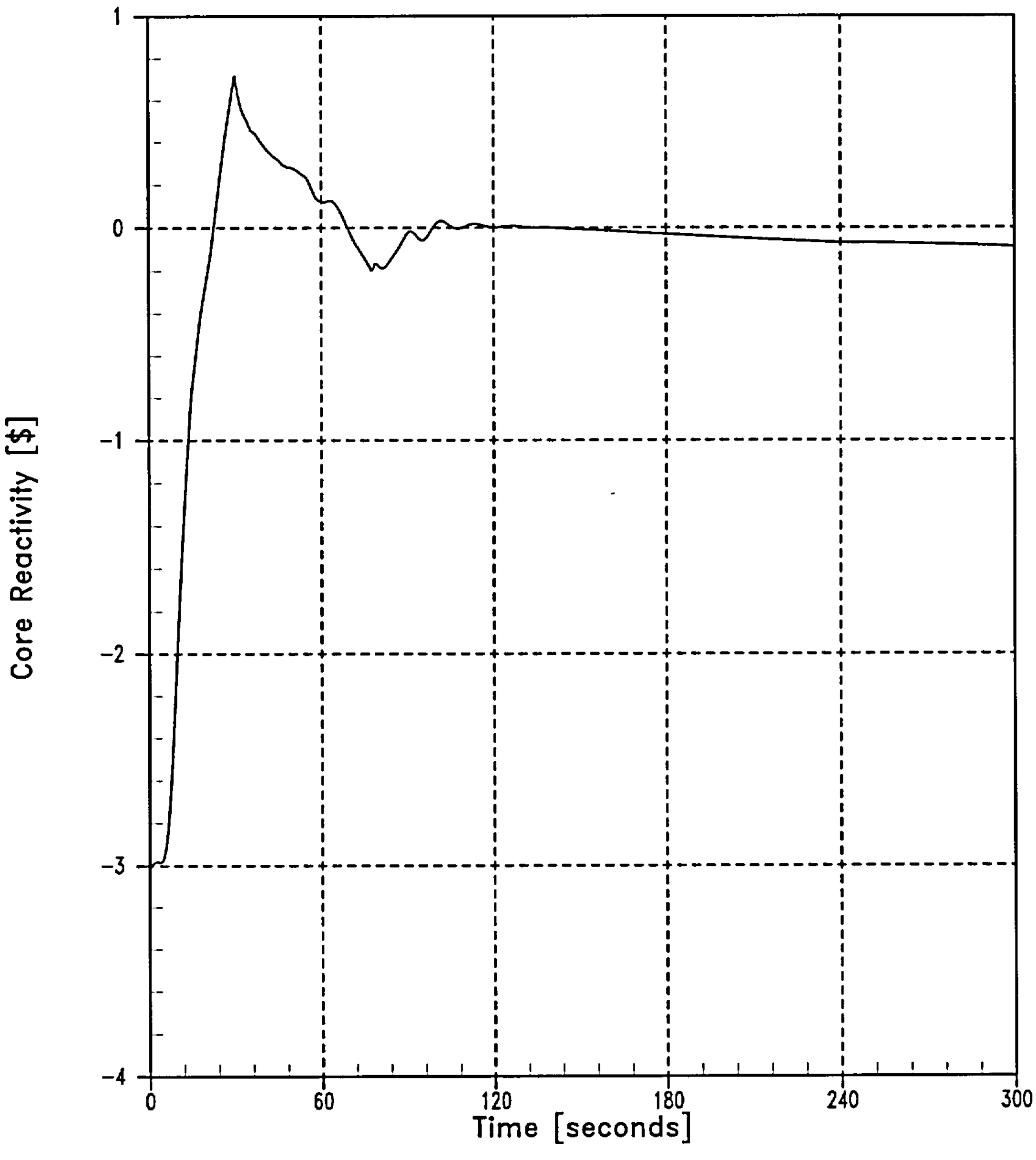

Figure 5.1.12-10 Main Steam Line Break with Offsite Power – Reactivity versus Time

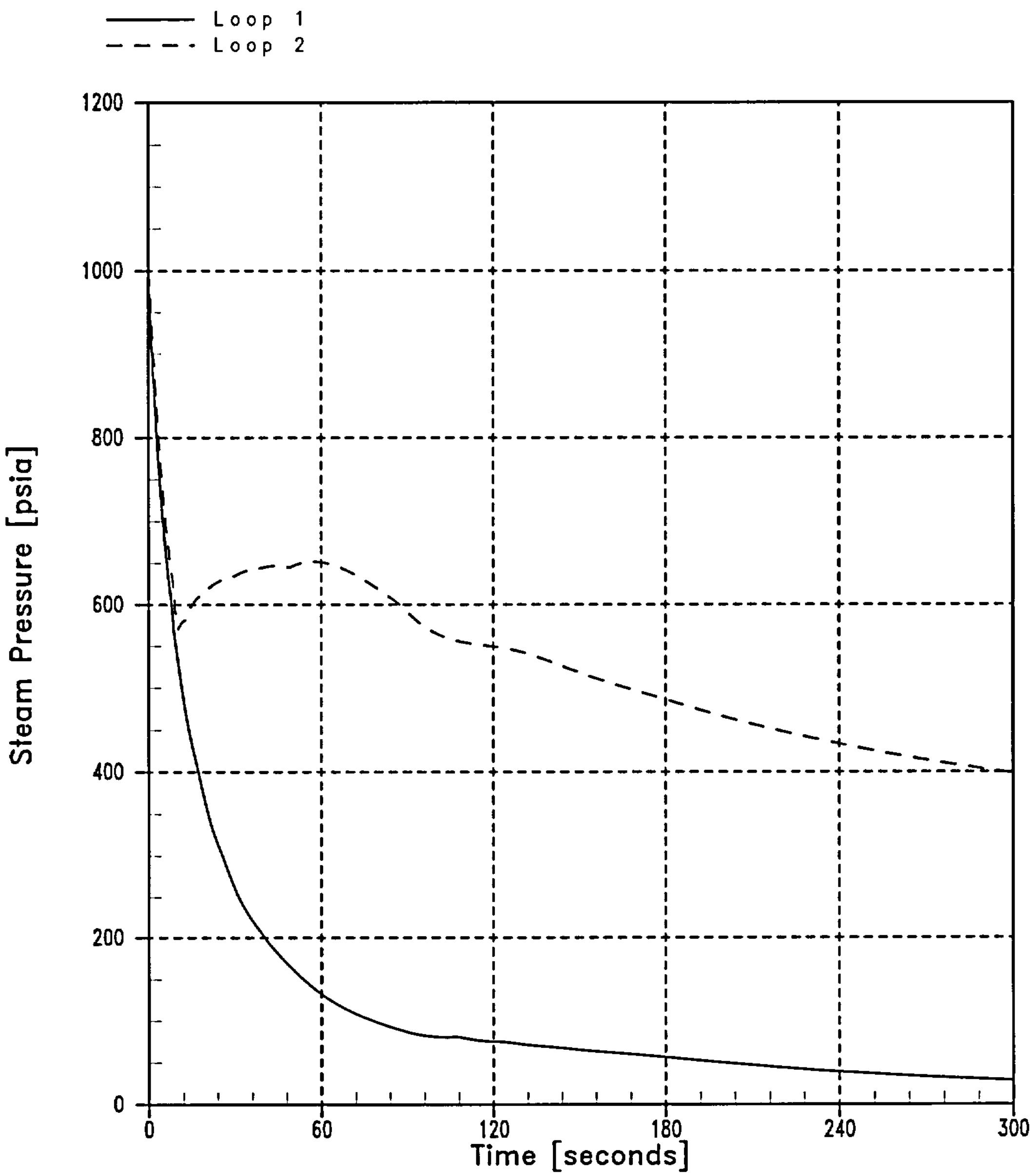

Figure 5.1.12-11 Main Steam Line Break Without Offsite Power – Steam Pressure versus Time

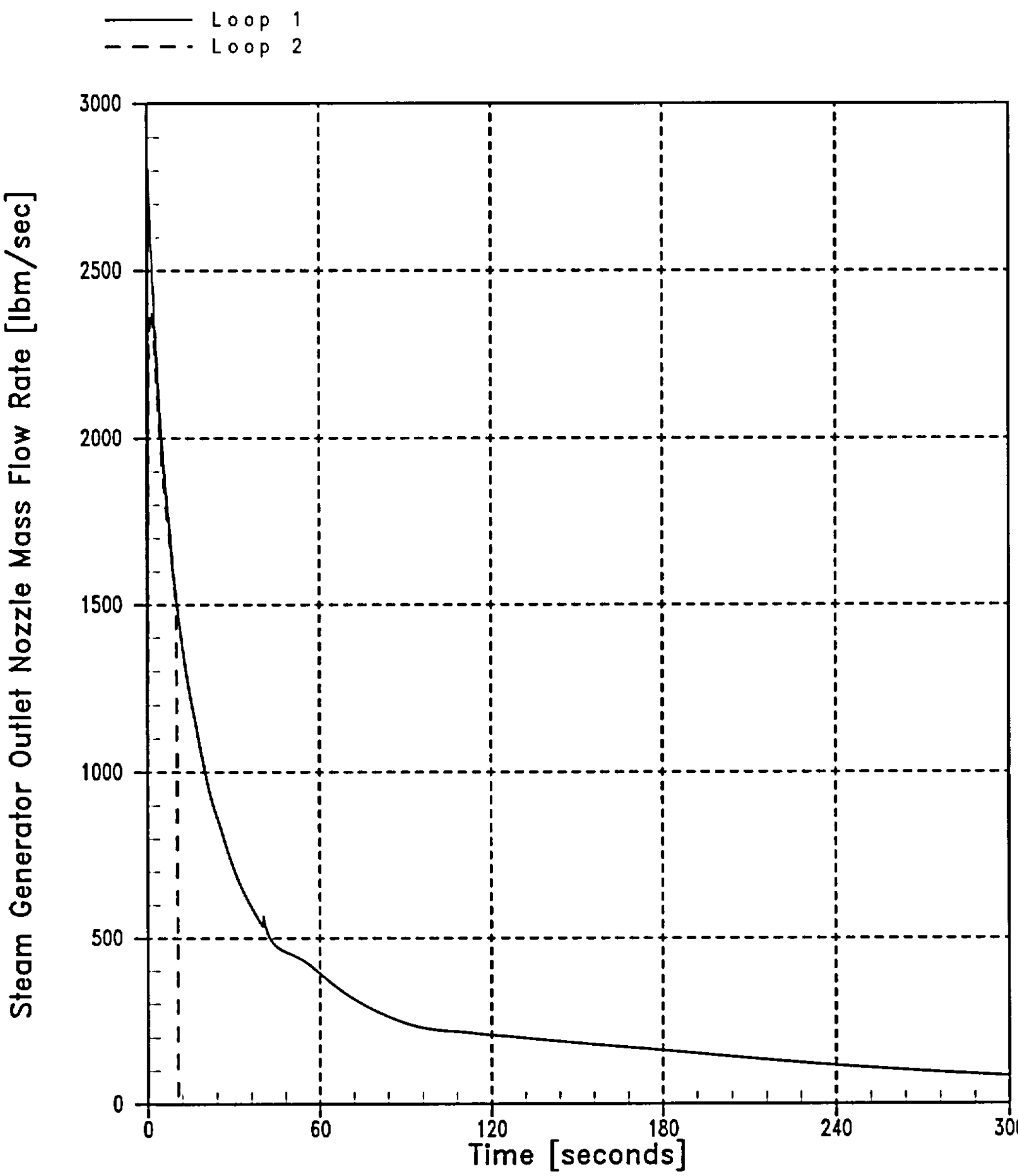

Figure 5.1.12-12 Main Steam Line Break Without Offsite Power – Steam Generator Outlet Nozzle Mass Flow Rate versus Time

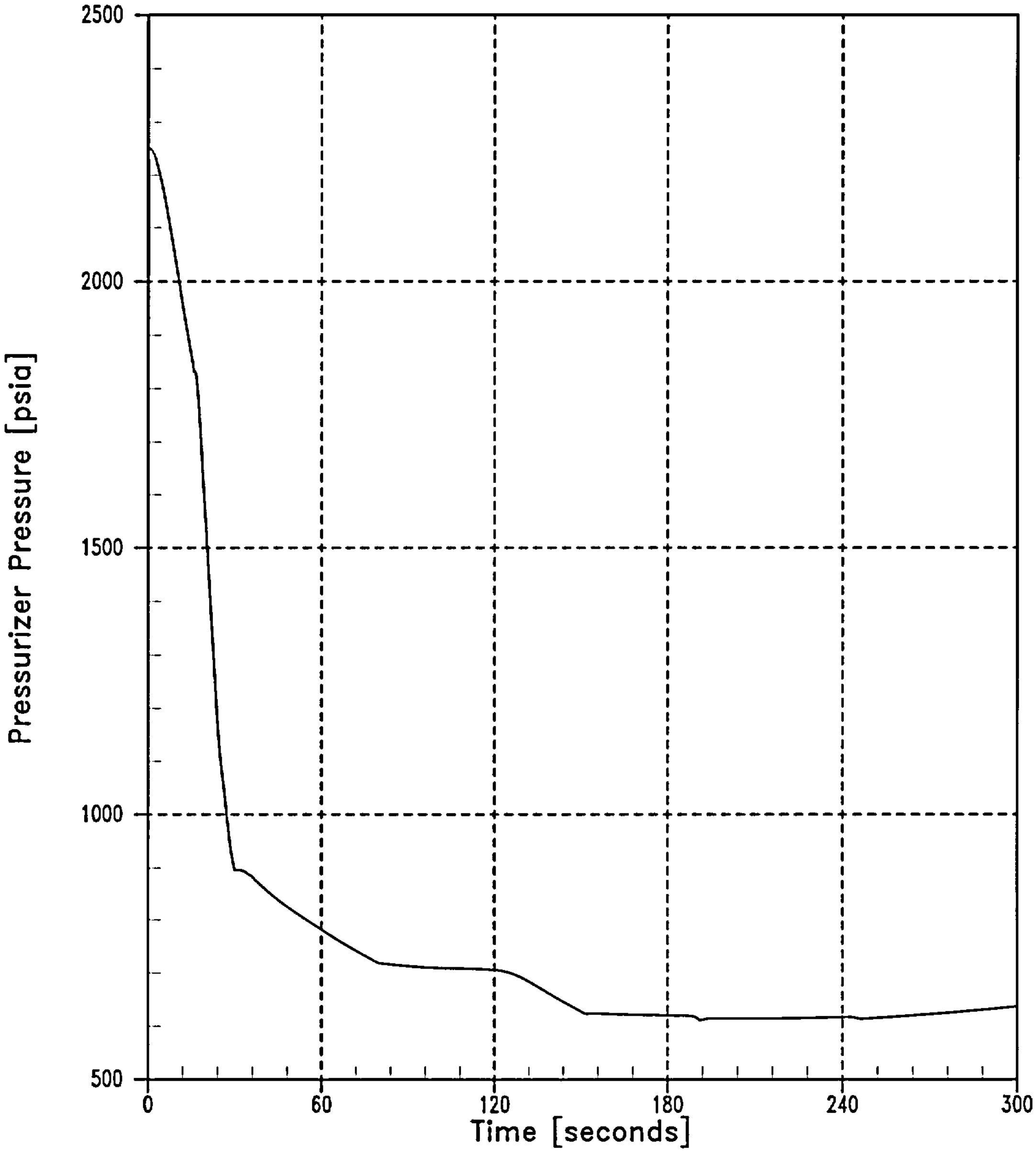

Figure 5.1.12-13 Main Steam Line Break Without Offsite Power – Pressurizer Pressure versus Time

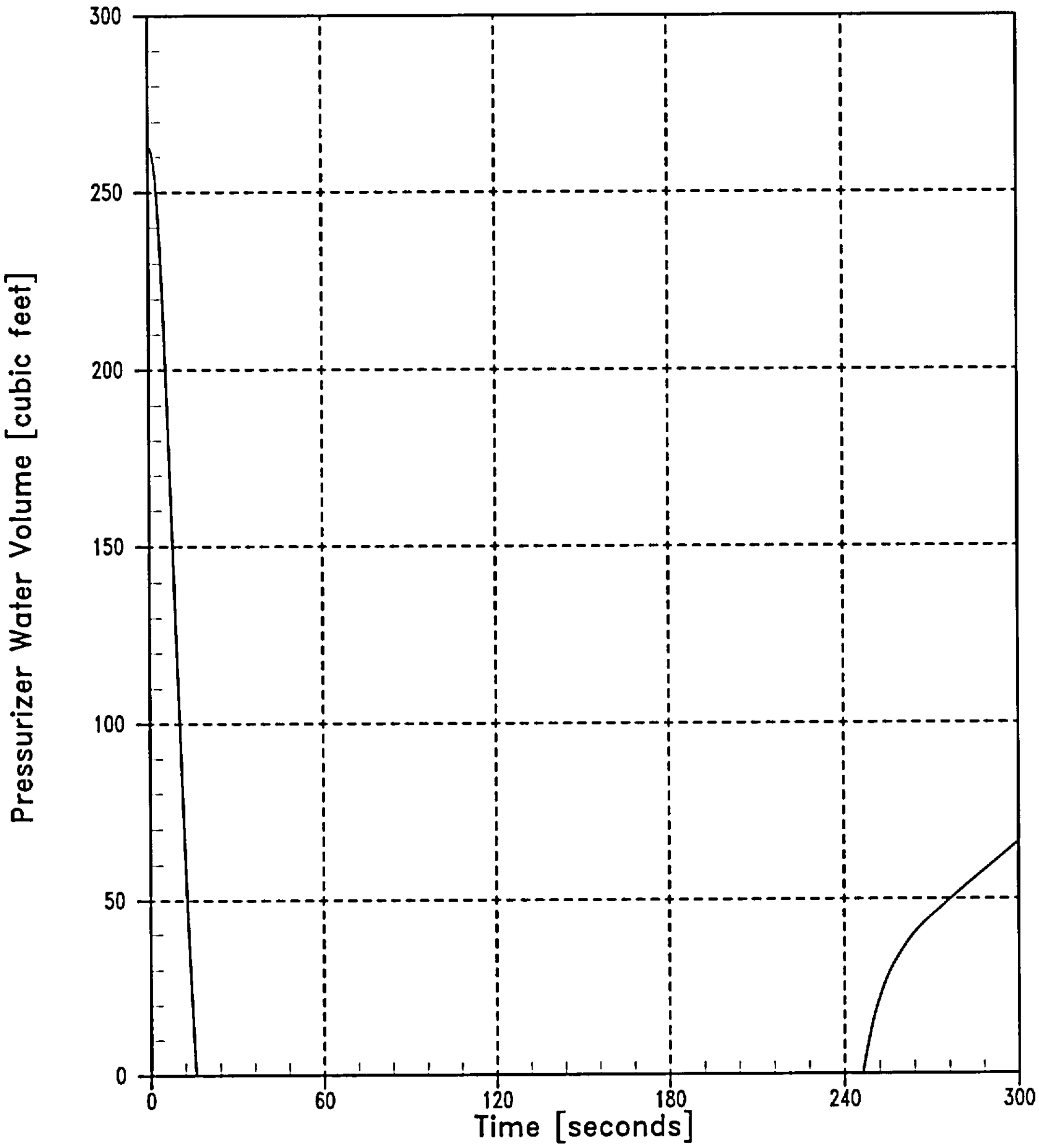

Figure 5.1.12-14 Main Steam Line Break Without Offsite Power – Pressurizer Water Volume versus Time

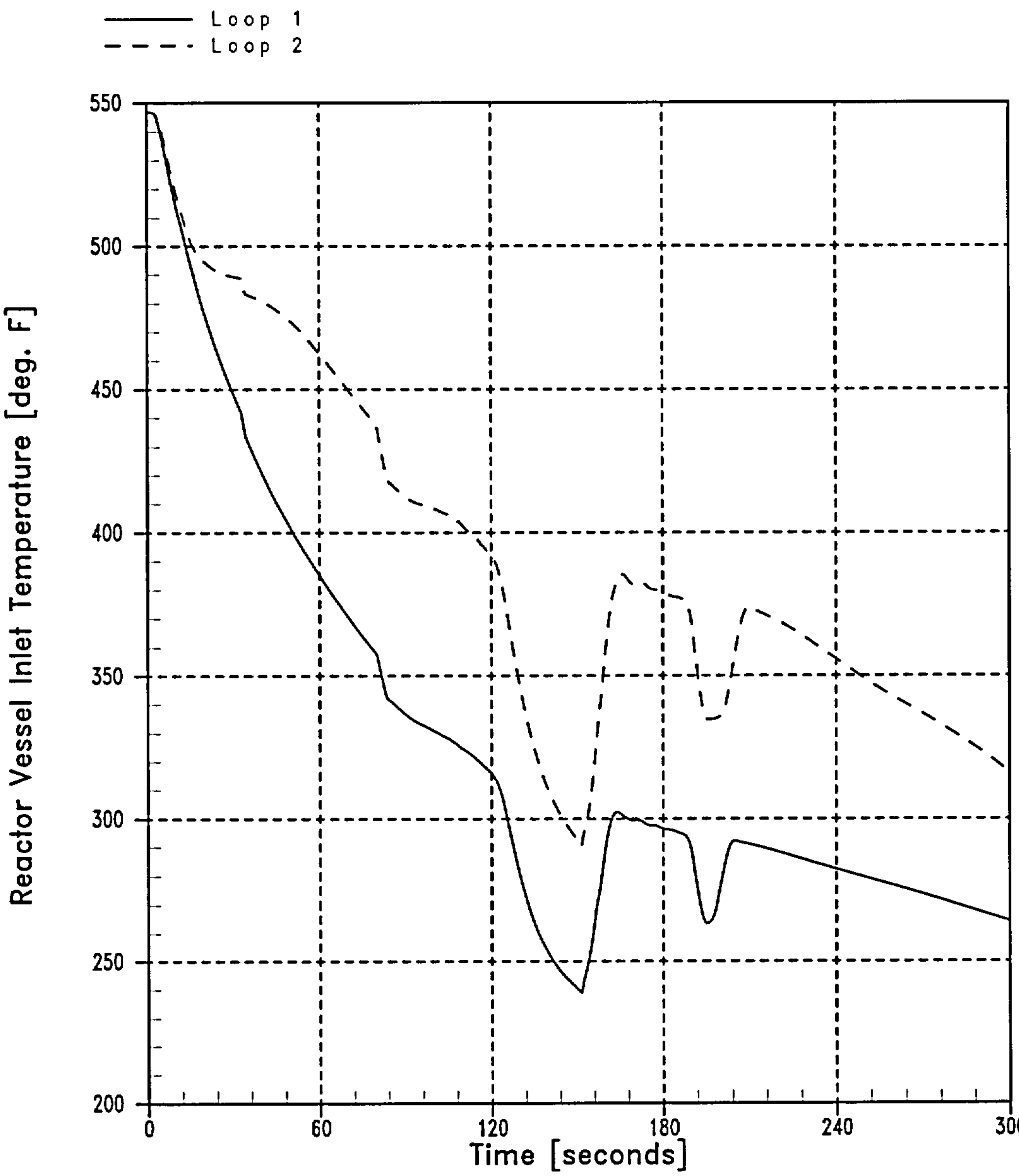

Figure 5.1.12-15 Main Steam Line Break Without Offsite Power – Reactor Vessel Inlet Temperature versus Time

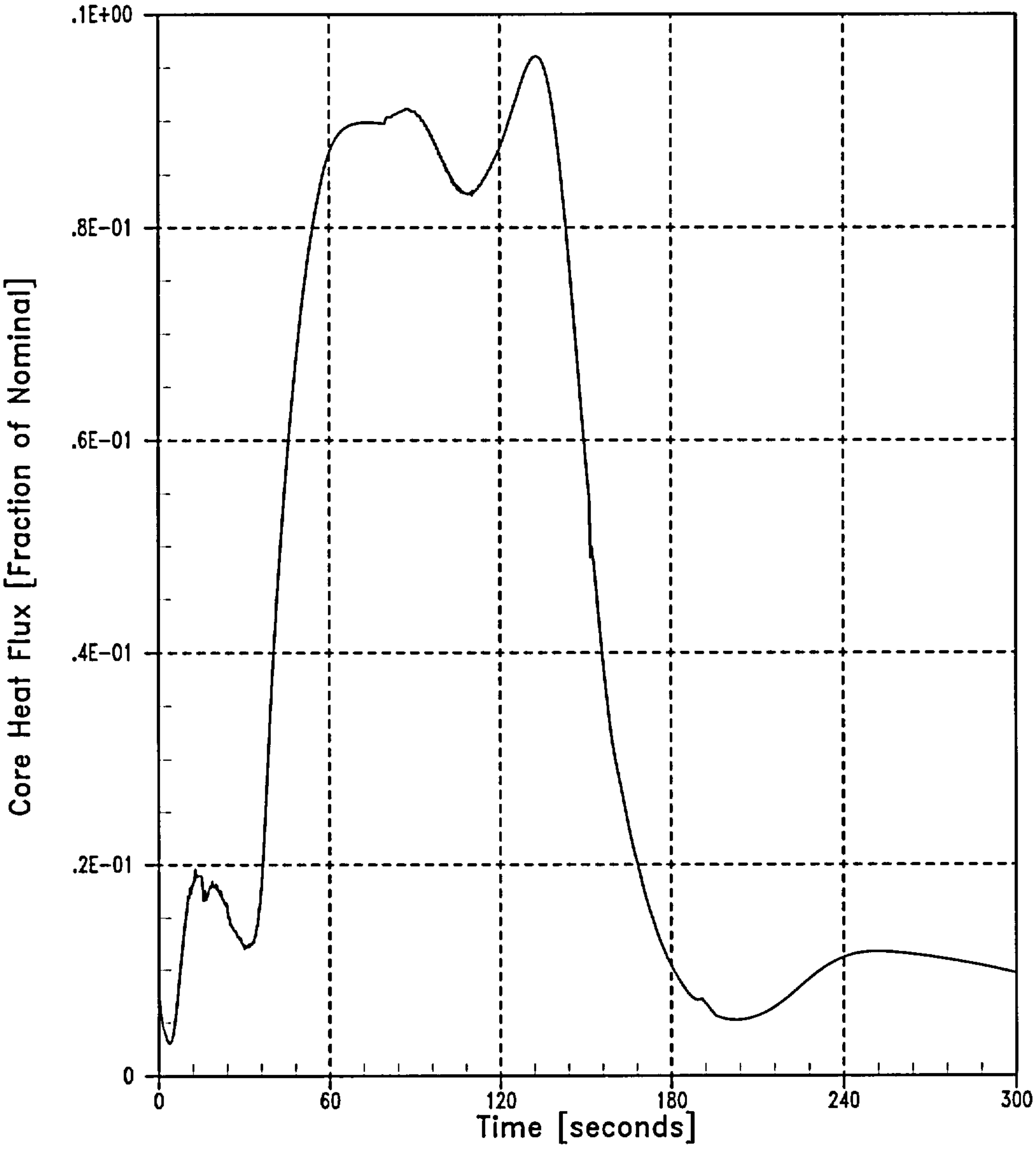

Figure 5.1.12-16 Main Steam Line Break Without Offsite Power – Core Heat Flux versus Time

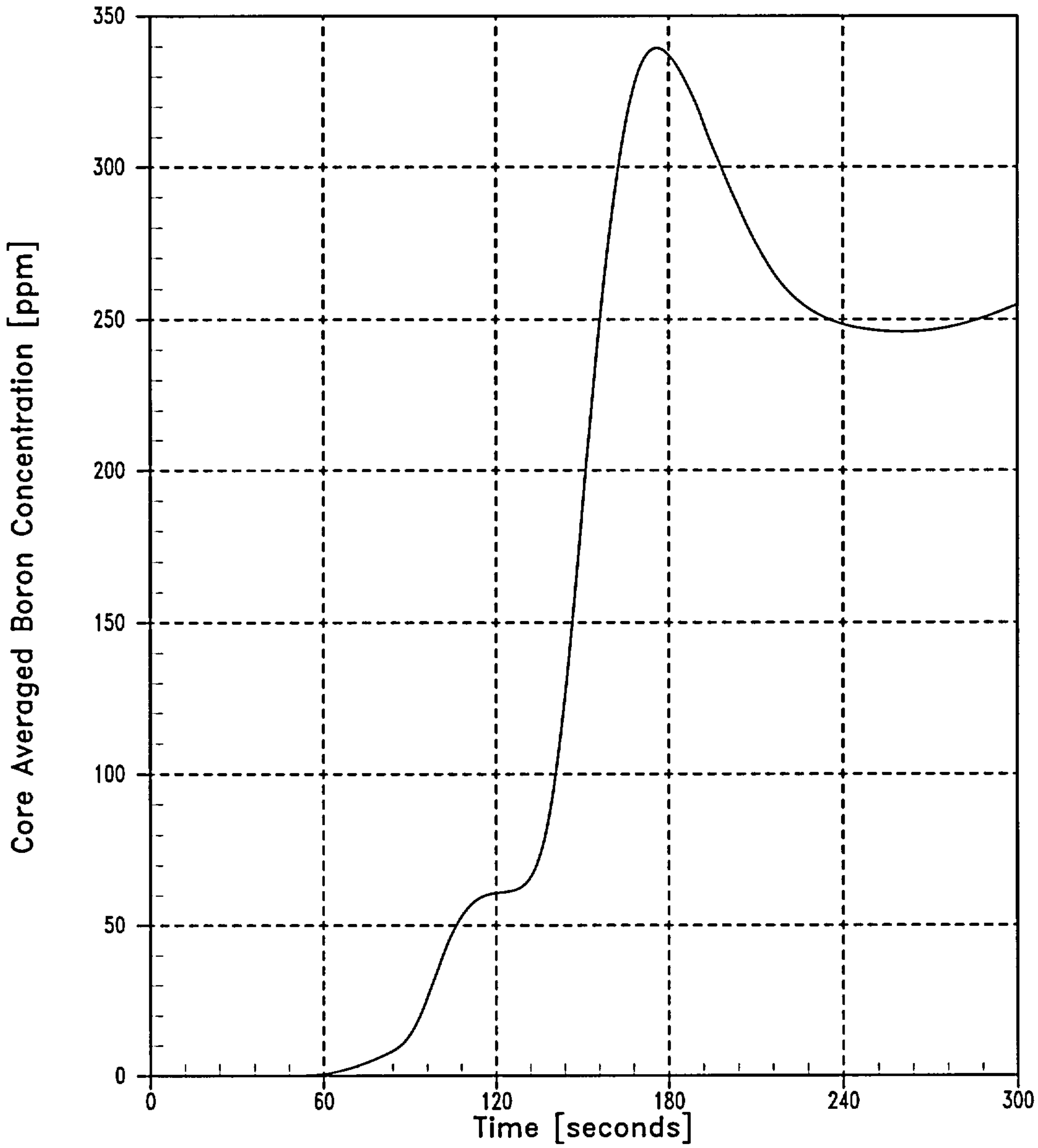

Figure 5.1.12-17 Main Steam Line Break Without Offsite Power – Core Averaged Boron Concentration versus Time

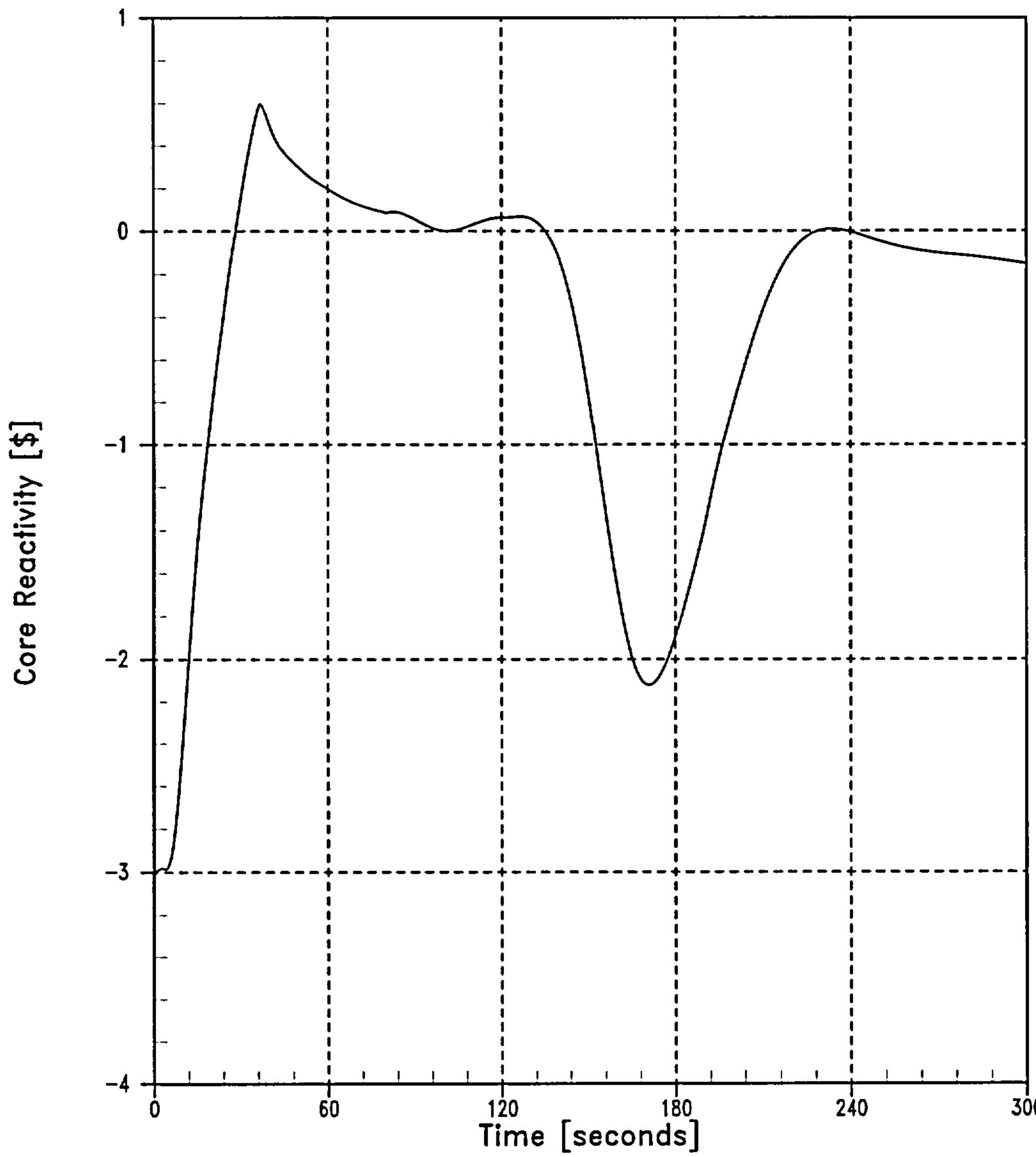

Figure 5.1.12-18 Main Steam Line Break Without Offsite Power – Reactivity versus Time

5.1.13 Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) (USAR Section 14.2.6)

Accident Description

This accident is the result of the extremely unlikely mechanical failure of a control rod drive mechanism pressure housing such that the RCS pressure would eject the RCCA and drive shaft. The consequences of this mechanical failure, in addition to being a minor LOCA, may also be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

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

The mechanical design is discussed in Section 3 of the USAR. A failure of the full-length 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:

- Each control rod drive mechanism housing is completely assembled and shop-tested at 4100 psi.

- The mechanism housings are individually hydrotested as they are installed on the reactor vessel head to the head adapters, and checked during the hydrotest of the completed RCS.

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

- 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 are encountered.

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

Even if a rupture of the control rod mechanism housing is postulated, the operation of a chemical shim plant is such that the severity of an ejected rod is inherently limited. In general, the reactor is operated with control rods inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and groupings of control rod banks are selected during the core nuclear design to lessen the severity of an ejected control rod assembly. Therefore, should an RCCA be ejected from the reactor vessel during normal operation, there probably would be no reactivity excursion since most of the control rods are fully withdrawn from the core, or a minor reactivity excursion if an inserted RCCA is ejected from its normal position.

However, it may occasionally be desirable to operate with larger control rod insertions. For this reason, rod insertion limits are defined in the Technical Specifications as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. There are low and lo-lo level insertion monitors with visual and audio signals. Operating instructions require boration when receiving either alarm.

If an RCCA ejection accident were to occur, a fuel rod thermal transient that could cause a DNB may occur together with limited fuel damage. The amount of fuel damage that can result from such an accident will be governed mainly by the worth of the ejected RCCA and the power distribution attained with the remaining control rod pattern. The transient is limited by the Doppler reactivity effects of the increase in fuel temperature and is terminated by reactor trip actuated by neutron flux signals. It is terminated before conditions are reached that can result in damage to the reactor coolant pressure boundary, or significant disturbances in the core, its support structures or other reactor pressure vessel internals that would impair the capability to cool the core.

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 DPC. This self limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should an RCCA ejection accident occur, the following automatic features of the RPS are available to terminate the transient:

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

- The intermediate-range high neutron flux reactor trip is actuated when either of two independent intermediate-range channels indicates a flux level above a pre-selected manually adjustable setpoint. This trip function may be manually bypassed when two-out-of-four power-range channels give readings above approximately 10 percent of full power and is automatically reinstated when three-out-of-four channels indicate a power below this value.

- The power-range high neutron flux reactor trip (low setting) is actuated when two-out-of-four power-range channels indicate a power level above approximately 25 percent of full power. This trip function may be manually bypassed when two-out-of-four power-range channels indicate a power level above approximately 10 percent of full power and is automatically reinstated when three-out-of-four channels indicate a power level below this value.

- The power-range high neutron flux reactor trip (high setting) is actuated when two-out-of-four power-range channels indicate a power level above a preset setpoint (typically, 109-percent power). This trip function is always active when the reactor is at power.

- The high nuclear flux rate reactor trip is actuated when the positive rate of change of neutron flux on two-out-of-four nuclear power-range channels indicates a rate above the preset setpoint. This trip function is always active.

The ultimate acceptance criteria for this event is that any consequential damage to either the core or the RCS must not prevent long-term core cooling, and that any offsite dose consequences must be within the guidelines of 10 CFR 100. To demonstrate compliance with these requirements, it is sufficient to show that the RCS pressure boundary remains intact, and that no fuel dispersal in the coolant, gross lattice distortions, or severe shock waves will occur in the core. Therefore, the following acceptance criteria are applied to the RCCA ejection accident:

- Maximum average fuel pellet enthalpy at the hot spot must remain below 200 cal/g (360 Btu/lbm).

- Peak RCS pressure must remain below that which would cause the stresses in the RCS to exceed the faulted condition stress limits.

- Maximum fuel melting must be limited to the innermost 10 percent of the fuel pellet at the hot spot, independent of the above pellet enthalpy limit.

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages: a neutron kinetic analysis and a hot-spot fuel heat transfer analysis. The spatial neutron kinetics code TWINKLE (Reference 5-8) is used in a 1-D axial kinetics model to calculate the core nuclear power including the various total core feedback effects; that is, Doppler reactivity and moderator reactivity. The average core nuclear power is multiplied by the post-ejection hot-channel factor, and the fuel enthalpy and temperature transients at the hot spot are calculated with the detailed fuel and cladding transient heat transfer computer code, FACTRAN (Reference 5-4). The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. Additional details of the methodology are provided in WCAP-7588 (Reference 5-14).

The overpressurization of the RCS and number of rods in DNB, as a result of a postulated ejected rod, have both been analyzed on a generic basis for Westinghouse PWRs as detailed in Reference 5-14.

If the safety limits for fuel damage are not exceeded, there is little likelihood of fuel dispersal into the coolant or a sudden pressure increase from thermal-to-kinetic energy conversion. The pressure surge for this analysis can, therefore, be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, HFP, indicates that the peak pressure does not exceed that which would cause stresses in the RCS to exceed their faulted condition stress limits. Since the severity of the KNPP analysis does not exceed this worst-case analysis, the RCCA ejection accident will not result in an excessive pressure rise or further damage to the RCS.

Reference 5-14 also documents a detailed three-dimensional THINC-III calculation, which demonstrates an upper limit to the number of rods-in-DNB for the RCCA ejection accident as 10 percent. Since the severity of the KNPP analysis does not exceed this worst-case analysis, the maximum number of rods in DNB following an RCCA ejection will be less than 10 percent, which is well within the 15 percent used in the radiological dose evaluation. The most limiting break size resulting from an RCCA ejection will not be sufficient to uncover the core or cause DNB at any later time. Since the maximum number of fuel rods experiencing DNB is limited to 15 percent, the fission product release will not exceed that associated with the guidelines of 10 CFR 100.

In calculating the nuclear power and hot-spot fuel rod transients following RCCA ejection, the following conservative assumptions are made:

a. The RTDP is not used for the RCCA ejection analysis. Instead, the STDP (maximum uncertainties in initial conditions) is employed. The analysis assumes uncertainties of 2.0 percent in nominal core power, 6.0°F in nominal vessel T_{avg}, and 50 psi in nominal pressurizer pressure. An additional 0.1-psi uncertainty has been determined to be negligible.

b. A minimum value for the delayed neutron fraction for BOC and EOC conditions is assumed, which increases the rate at which the nuclear power increases following RCCA ejection.

c. A minimum value of the Doppler power defect is assumed, which conservatively results in the maximum amount of energy deposited in the fuel following RCCA ejection. A minimum value of the moderator feedback is also assumed. A positive MTC is assumed for the BOC, zero-power case.

d. Maximum values of ejected RCCA worth and post-ejection total hot-channel factors are assumed for all cases considered. These parameters are calculated using standard nuclear design codes for the maximum allowed bank insertion at a given power level as determined by the rod insertion limits. No credit is taken for the flux flattening effects of reactivity feedback.

e. The start of rod motion occurs 0.65 seconds after the high neutron flux trip point is reached.

The analysis is performed to bound operation with Westinghouse 422V+ fuel and a maximum loop-to-loop SGTP imbalance of 10 percent.

Results

Figures 5.1.13-1 through 5.1.13-8 present the nuclear power and hot-spot fuel rod thermal transients for the RCCA ejection cases analyzed. The transient results of the analysis are summarized in Table 5.1.13-1. A time sequence of events is provided in Table 5.1.13-2. For all cases, the maximum fuel pellet enthalpy remained below 200 cal/g. For the HFP cases, the peak hot-spot fuel centerline temperature reached the fuel melting temperature (4900°F at BOC and 4800°F at EOC). However, melting was restricted to less than 10 percent of the pellet. For the HZP cases, no fuel melting was predicted.

Conclusions

The analysis performed has demonstrated that, for the RCCA ejection event, the fuel thermal criteria are not exceeded. In addition, the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits. Also, the upper limit to the number of rods-in-DNB is 15 percent, which will not result in fission product releases in excess of that associated with the requirements of 10 CFR 100. Consequently, all applicable acceptance criteria are met.

Table 5.1.13-1 Assumptions and Results – RCCA Ejection		
Beginning of Cycle	**Full Power**	**Zero Power**
Initial Power Level, %	102	0
Ejected RCCA Worth, % Δk	0.380	0.770
Delayed Neutron Fraction	0.0049	0 0049
Doppler Power Defect, % Δk	1.000	1.000
Feedback Reactivity Weighting	1.139	2.008
Trip Reactivity, % Δk	3.5	1.0
F_Q Before Ejection	2.5	N/A
F_Q After Ejection	4.2	11.0
Number of RCPs Operating	2	1
Maximum Fuel Pellet Enthalpy, cal/g	167.4	144.9
Maximum Fuel Melted, %	2.17	None
End of Cycle	**Full Power**	**Zero Power**
Initial Power Level, %	102	0
Ejected RCCA Worth, % Δk	0.370	0.930
Delayed Neutron Fraction	0.0043	0.0043
Doppler Power Defect, % Δk	0.900	0.900
Feedback Reactivity Weighting	1.316	2.144
Trip Reactivity, % Δk	3.5	1.0
F_Q Before Ejection	2.5	N/A
F_Q After Ejection	5.69	13.0
Number of RCPs Operating	2	1
Maximum Fuel Pellet Enthalpy, cal/g	170.3	161.6
Maximum Fuel Melted, %	5.89	None

Table 5.1.13-2 Sequence of Events – RCCA Ejection	
Beginning of Cycle - Hot Zero Power	**Time (seconds)**
RCCA Ejection Occurs	0.000
High Neutron Flux Setpoint (Low Setting) is Reached	0.208
Peak Nuclear Power Occurs	0.252
Rods Begin to Fall Into the Core	0.858
Peak Cladding Average Temperature Occurs	2.134
Peak Heat Flux Occurs	2.150
Peak Fuel Average Temperature Occurs	2.273
Beginning of Cycle - Hot Full Power	**Time (seconds)**
RCCA Ejection Occurs	0.000
High Neutron Flux Setpoint (High Setting) is Reached	0.030
Peak Nuclear Power Occurs	0.135
Rods Begin to Fall Into the Core	0.680
Peak Fuel Average Temperature Occurs	1.904
Peak Cladding Average Temperature Occurs	2.024
Peak Heat Flux Occurs	2.040

Table 5.1.13-2 Sequence of Events – RCCA Ejection (cont.)	
End of Cycle - Hot Zero Power	**Time (seconds)**
RCCA Ejection Occurs	0.000
High Neutron Flux Setpoint (Low Setting) is Reached	0.147
Peak Nuclear Power Occurs	0.176
Rods Begin to Fall Into the Core	0.797
Peak Cladding Average Temperature Occurs	1.592
Peak Heat Flux Occurs	1.596
Peak Fuel Average Temperature Occurs	1.827
End of Cycle - Hot Full Power	**Time (seconds)**
RCCA Ejection Occurs	0.000
High Neutron Flux Setpoint (High Setting) is Reached	0.024
Peak Nuclear Power Occurs	0.129
Rods Begin to Fall Into the Core	0.674
Peak Fuel Average Temperature Occurs	1.902
Peak Cladding Average Temperature Occurs	2.035
Peak Heat Flux Occurs	2.050

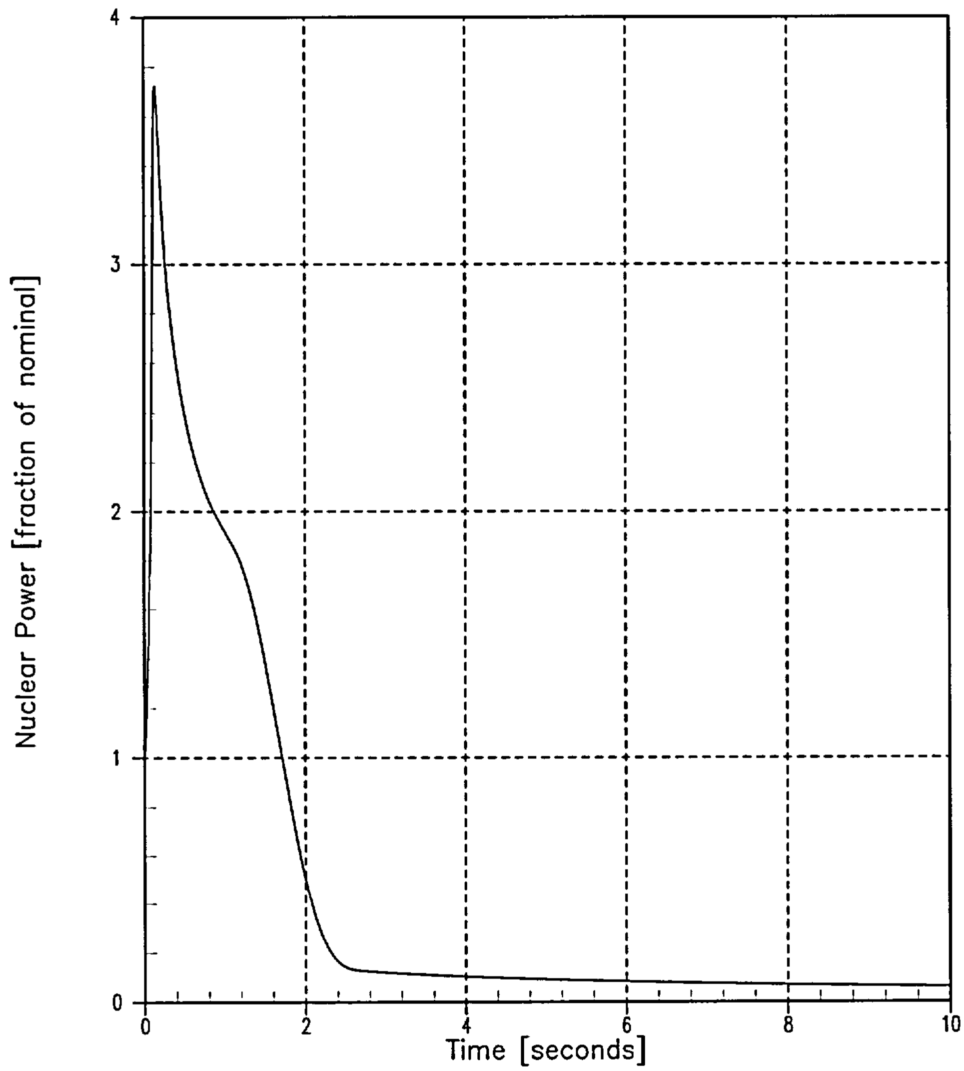

Figure 5.1.13-1 RCCA Ejection Accident from Full Power Beginning of Cycle – Reactor Power versus Time

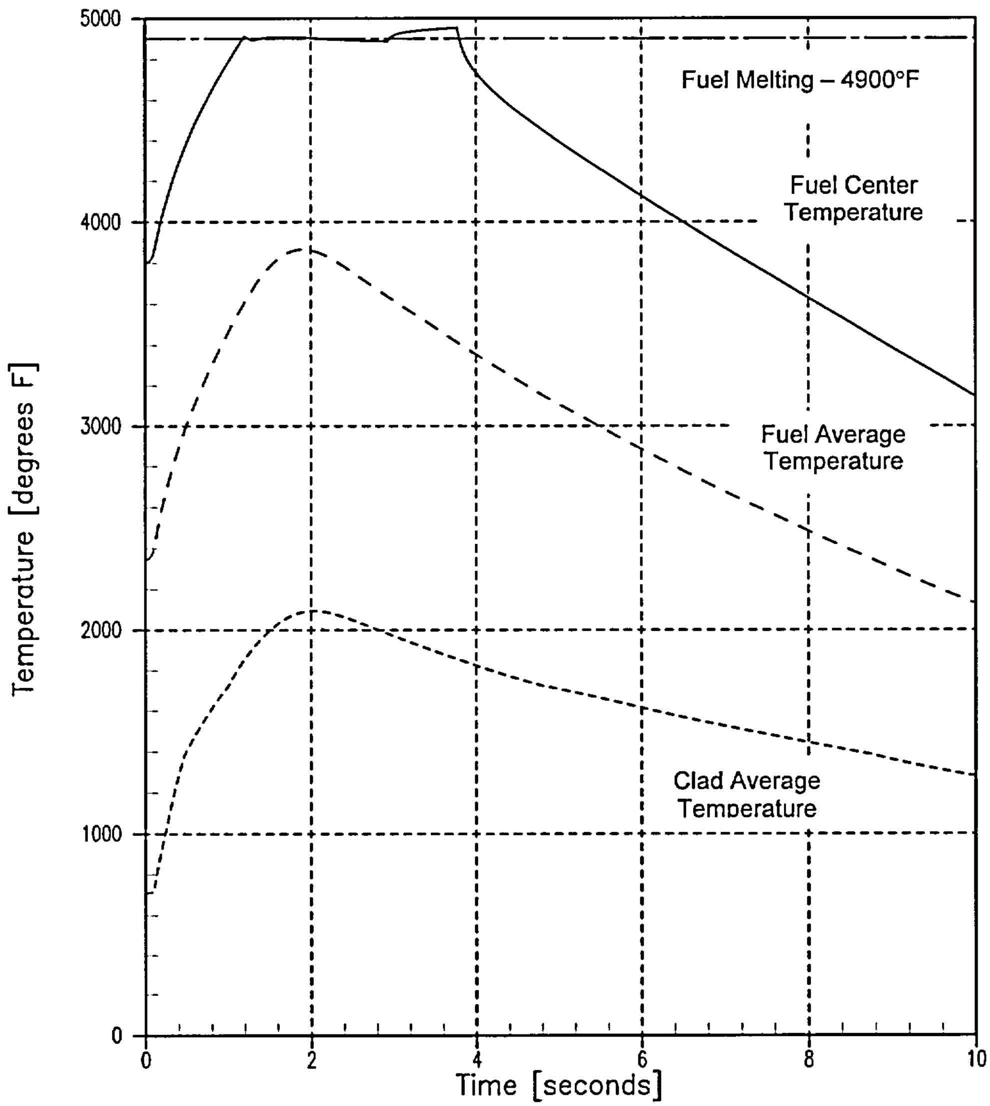

Figure 5.1.13-2 RCCA Ejection Accident from Full Power Beginning of Cycle – Fuel and Cladding Temperatures versus Time

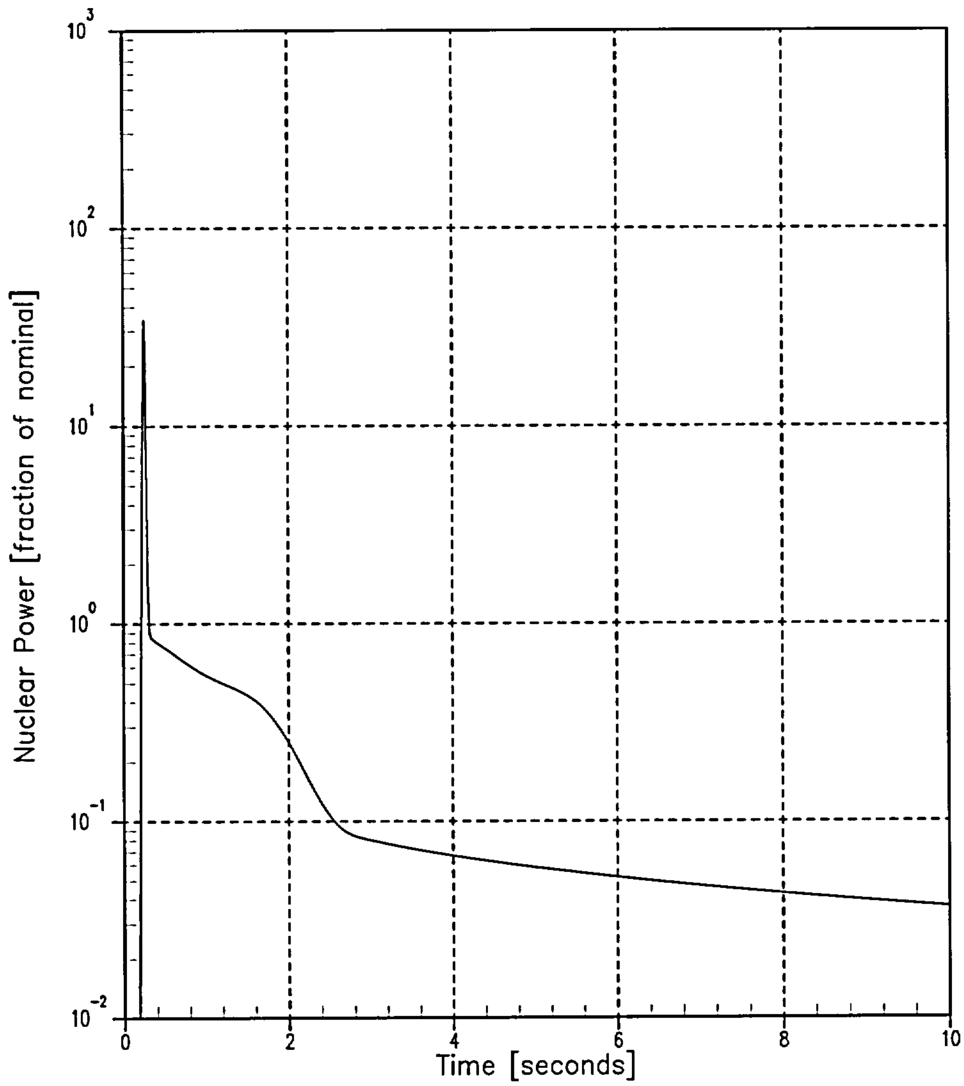

Figure 5.1.13-3 RCCA Ejection Accident from Zero Power Beginning of Cycle – Reactor Power versus Time

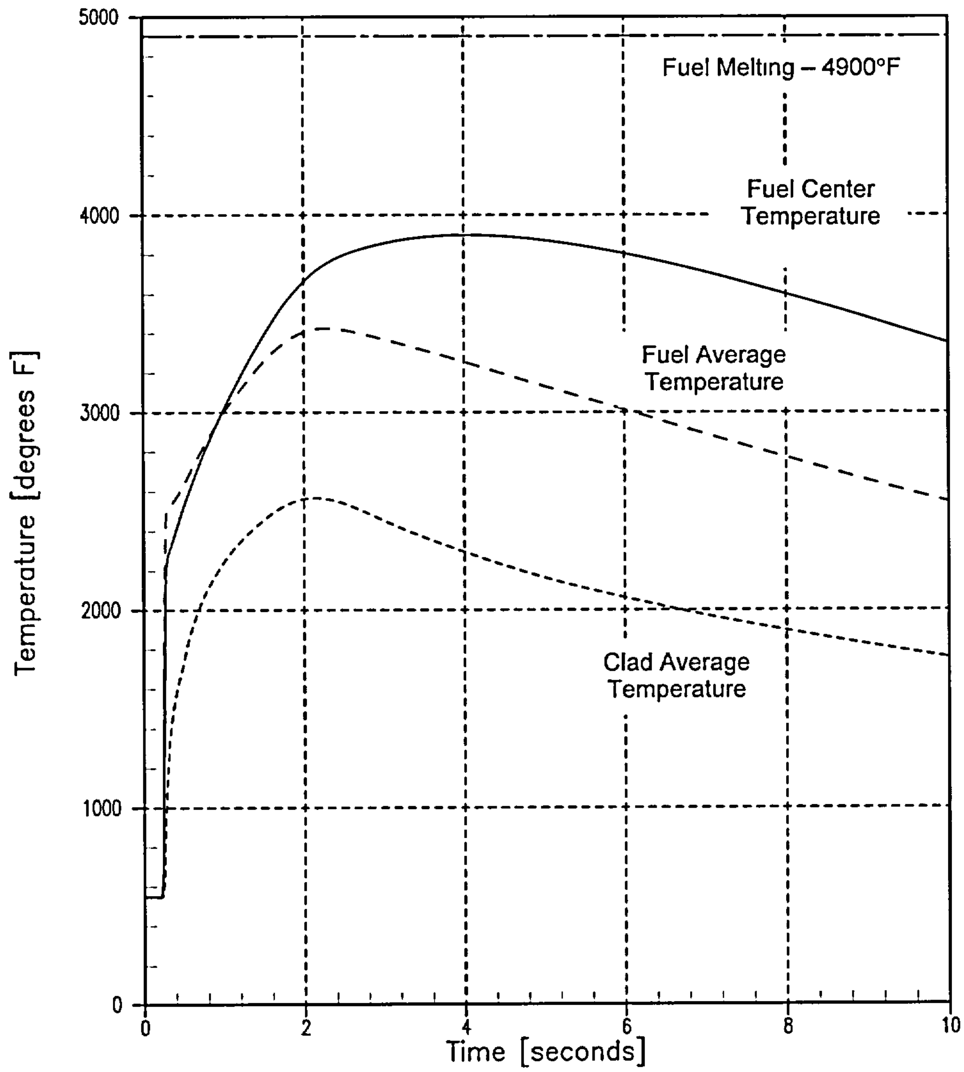

Figure 5.1.13-4 RCCA Ejection Accident from Zero Power Beginning of Cycle – Fuel and Cladding Temperatures versus Time

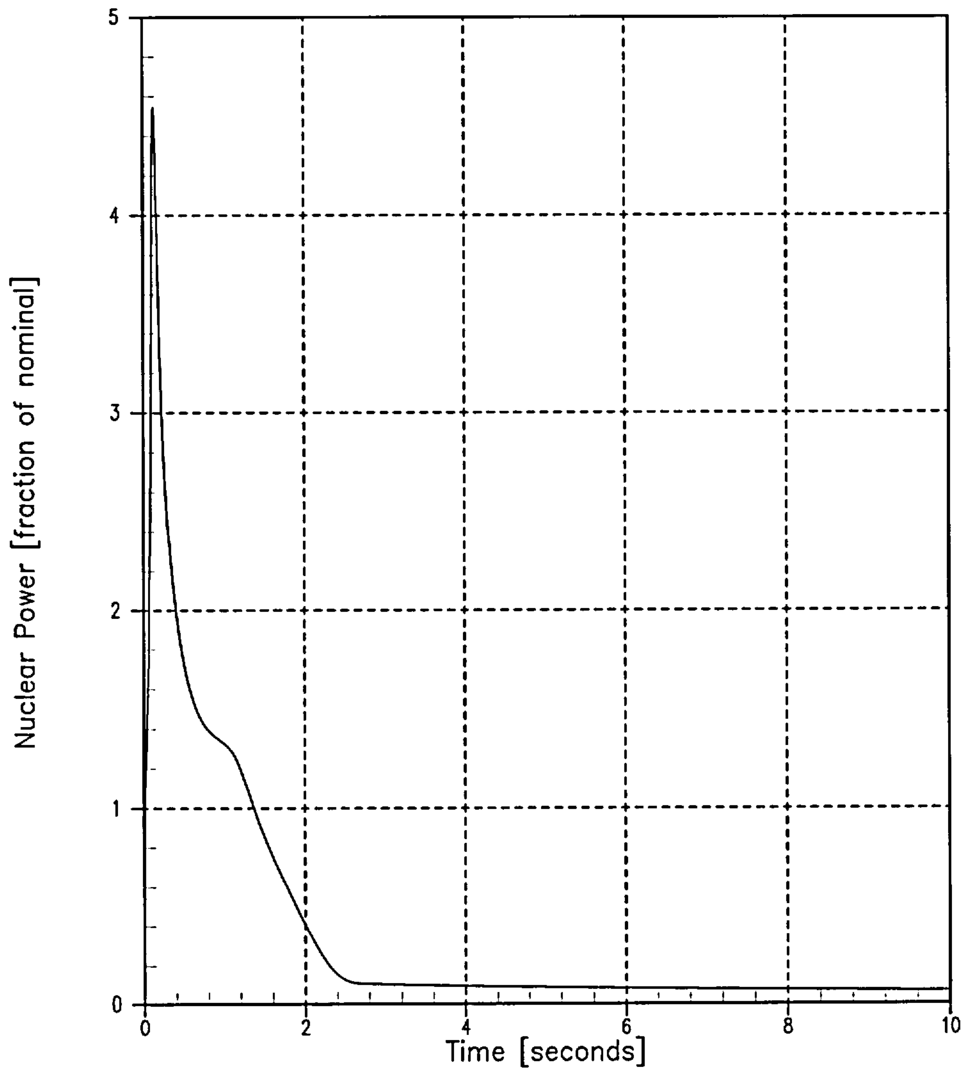

Figure 5.1.13-5 RCCA Ejection Accident from Full Power End of Cycle – Reactor Power versus Time

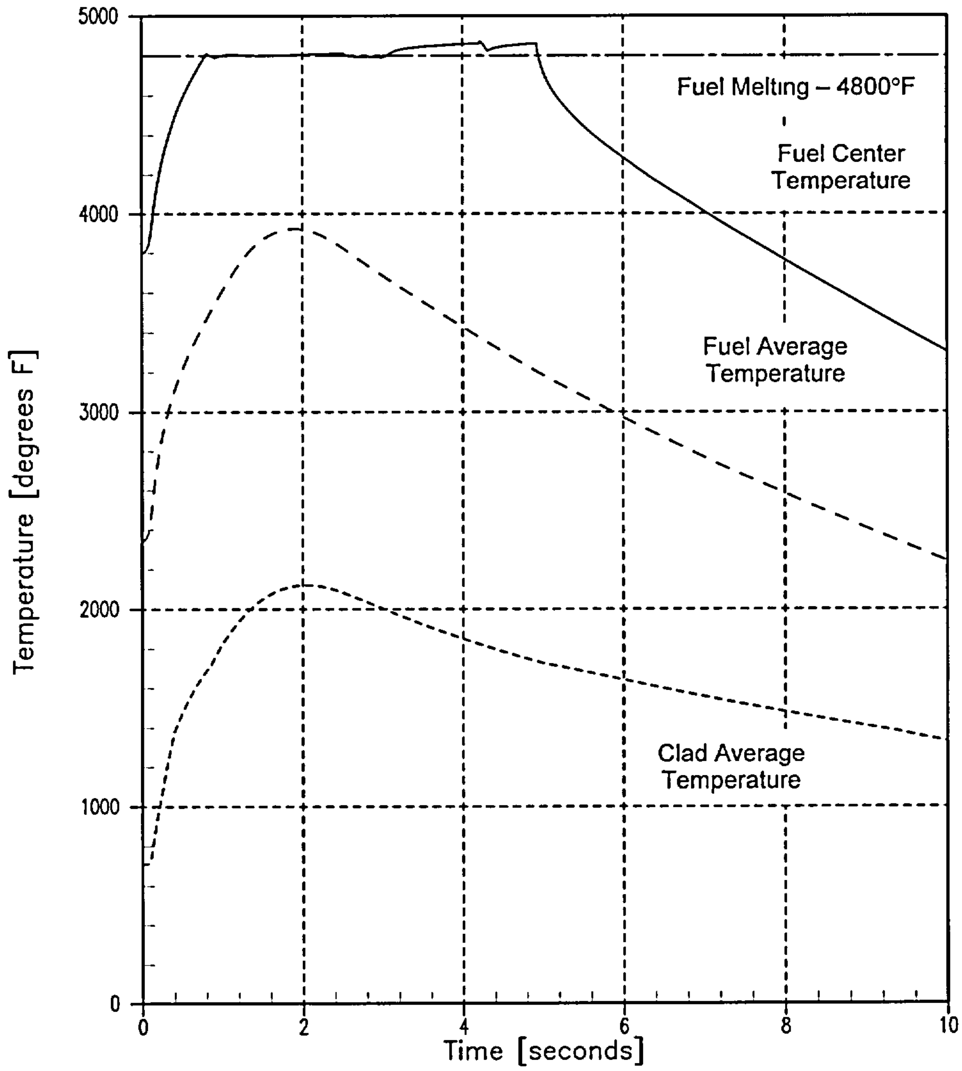

Figure 5.1.13-6 RCCA Ejection Accident from Full Power End of Cycle – Fuel and Cladding Temperatures versus Time

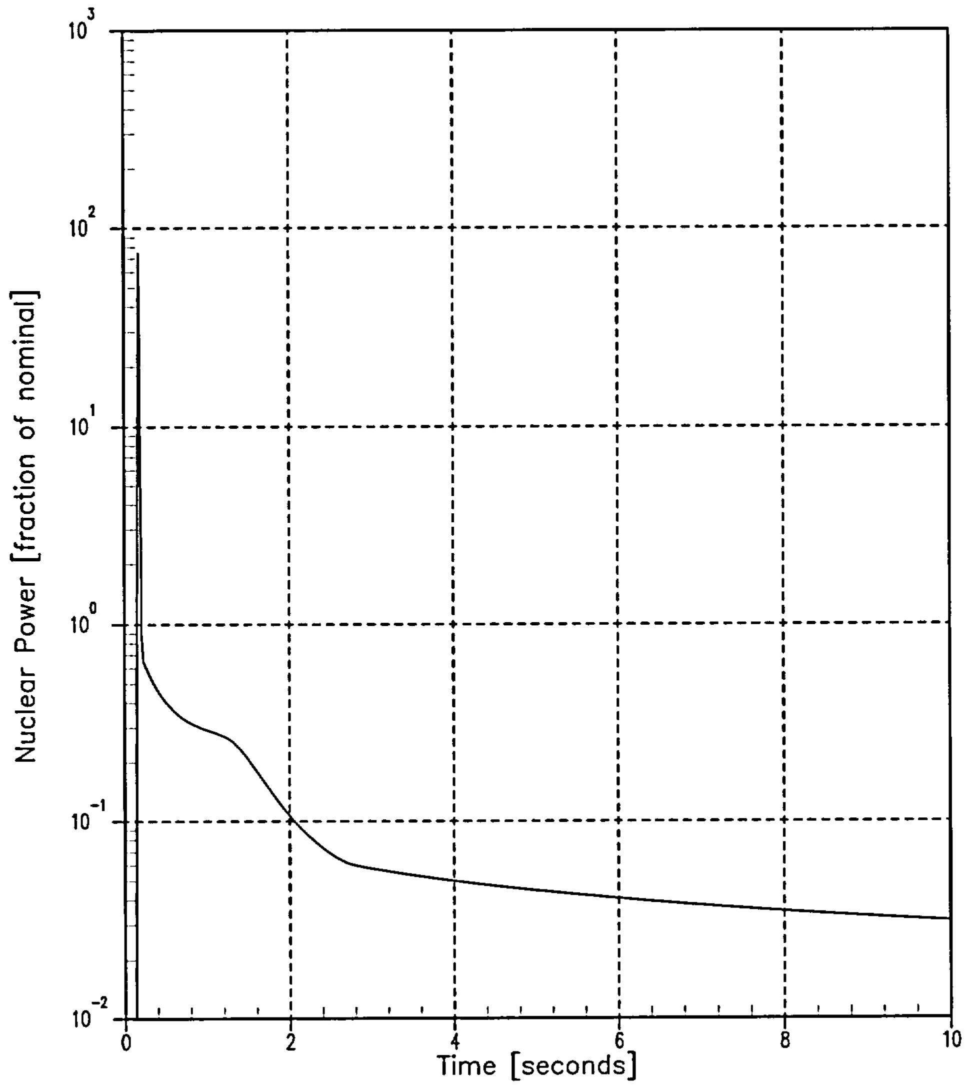

Figure 5.1.13-7 RCCA Ejection Accident from Zero Power End of Cycle – Reactor Power versus Time

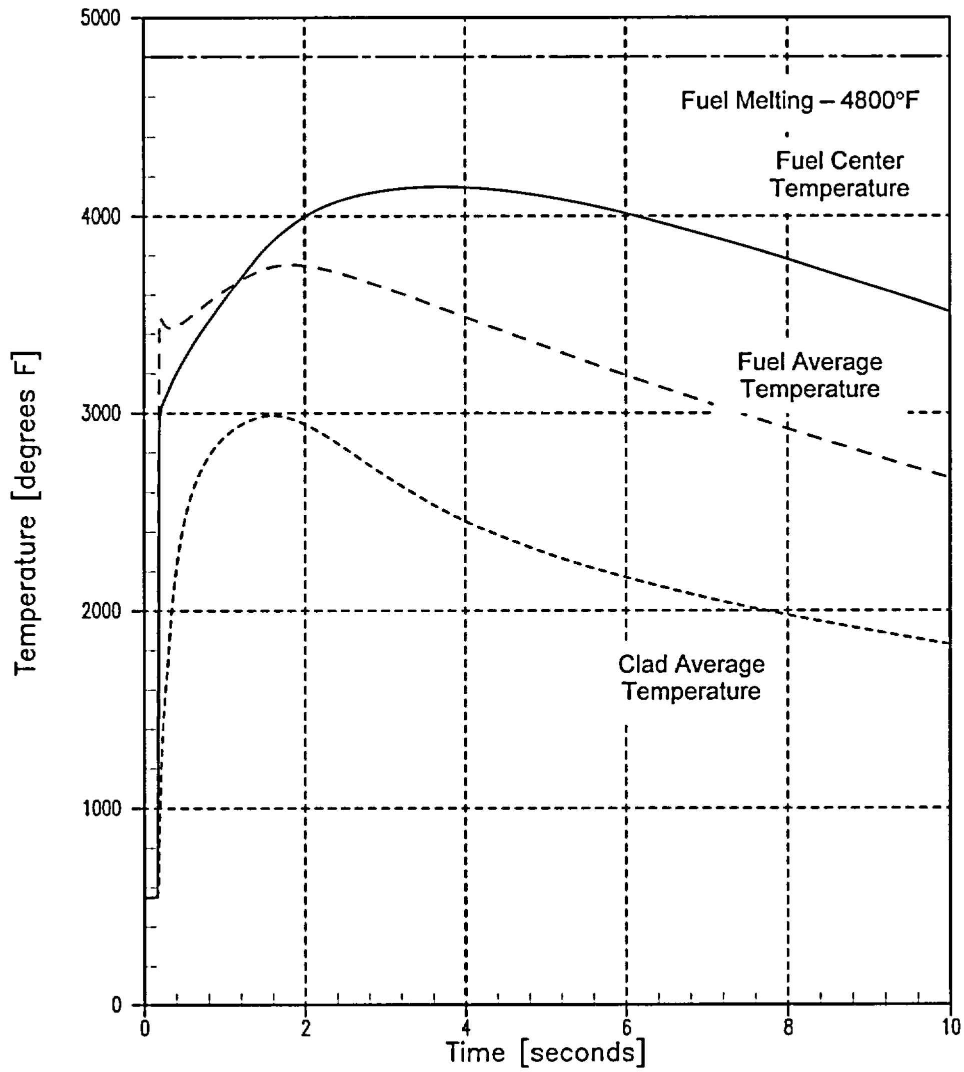

Figure 5.1.13-8 RCCA Ejection Accident from Zero Power End of Cycle – Fuel and Cladding Temperatures versus Time

5.1.14 Anticipated Transients Without Scram (USAR Section 14.1.11)

For Westinghouse-designed PWRs, the implementation of AMSAC is a requirement of the Final ATWS Rule, 10 CFR 50.62(b) (Reference 5-15). AMSAC has been installed at KNPP, and therefore, the requirements of 10CFR50.62(b) have been satisfied. After implementation of the fuel transition, it is assumed that the AMSAC will continue to be operable at KNPP in compliance with the requirements of the Final ATWS Rule. The current AMSAC design for KNPP is based on the Logic 1 generic AMSAC design for Westinghouse PWRs (AMSAC actuation on low steam generator water level) as described in WCAP-10858P-A, Revision 1 (Reference 5-16). An exception to the generic design is that AMSAC is armed at all power levels (the "c-20 permissive" signal is not used). The logic of AMSAC is to trip the turbine and start all three auxiliary feedwater pumps when lo-lo steam generator water level signals are present on 3 of 4 channels for a specified time period. As a supplement to AMSAC, a diverse scram system (DSS) has been installed at KNPP. Initiated on a signal from the AMSAC system, the DSS de-energizes the rod drive motor generator set exciter field, which interrupts power to the control rod grippers, allowing the control rods to free fall into the core, ending the ATWS event. As identified in the KNPP USAR, the U.S. NRC has approved the implementations of the AMSAC and DSS at KNPP. For the proposed fuel transition, it is assumed that the Nuclear Management Company will maintain and operate the AMSAC and DSS consistent with their designs and as approved by the U.S. NRC. Therefore, no specific evaluation of AMSAC or plant-specific, ATWS-related analyses are considered necessary to support operation of KNPP for the fuel transition.

5.5 REFERENCES

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5-6 Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

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5-10 "Technical Specifications – Kewaunee Nuclear Power Plant," Docket No. 50-305, Amendment 159, October 2001.

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5-14 D. H. Risher, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January 1975.

5-15 ATWS Final Rule – Code of Federal Regulations 10 CFR 50.62 and Supplementary Information Package, "Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

5-16 Adler, M. R., "AMSAC Generic Design Package," WCAP-10858P-A, Revision 1

ATTACHMENT C

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

Standard Technical Specification to KNPP TS Matrix

Abbreviated Standard TS to KNPP TS
Actions and Completion Times Matrix
RAI Question Attachment 2 Number 3a

Item		STS Action	STS Time	RTSR KNPP Action	RTSR KNPP TS Time
	FQC(Z) (KNPP FQN)	Reduce RTP 1% for each 1% exceeded	15 Min	Reduce RTP 1% for each 1% exceeded	15 Min
		Reduce NI Setpoint >/= 1% for each 1% > limit	72 hours	Reduce NI Setpoint and OPDT by at least 1% for each 1% > limit	72 hours
		Reduce OPDT Setpoint >/= 1% for each 1% > limit	72 hours		
		Verify FQC(Z) and FQW(Z) within limits	prior to increasing power	Correct & Verify within limit	prior to increasing power
		Required Actions & Completion Times Not Met be in MODE 2	6 hours	none (reduce RTP to < 5%)	none (6 hours)
		SR Verify within limits	Once after each refueling prior to exceeding 75% and within [12] hours of reaching equilibrium condition 10% greater than the last verification and 31 EFPD thereafter	SR Verify within limits	Following initial loading and EFPM interval thereafter

Abbreviated Standard TS to KNPP TS
Actions and Completion Times Matrix
RAI Question Attachment 2 Number 3a

Item		STS Action	STS Time	RTSR KNPP Action	RTSR KNPP TS Time
FQN(Z)		Reduce AFD Limits >/= 1% for every 1% limits exceeded	4 hours	Reduce AFD Limit 1% for every 1% limits exceeded, and	4 hours
		Reduce NI Setpoint >/= 1% for every 1% AFD limits reduced	72 hours	reduce NI and OPDT setpoints >/= 1% for each 1% AFD limit is reduced, and	72 hours
		Reduce OPDT Setpoint >/= 1% for every 1% AFD limits reduced	72 hours		
		Verify FQC(Z) and FQW(Z) within limits	prior to increasing power	Verify HCF limits satisfied	prior to exceeding AFD thermal power limit
		Required Actions & Completion Times Not Met be in MODE 2	6 hours	none (reduce RTP to < 5%)	none (6 hours)
	FQW(Z) (KNPP FQEQ)	SR Verify within limits	Once after each refueling prior to exceeding 75% and within [12] hours of reaching equilibrium condition 10% greater than the last verification and 31 EFPD thereafter	SR Verify within limits	During comparison of incore to AFD, or Once per EFPM whichever occurs first Upon achieving Equilibrium after reaching a RPT > 10% higher than last measurement
		If FWQ(Z) measurements indicate maximum over z [FC Q(Z) / K(Z)] has increased since the previous evaluation of FCQ(Z)·		If Peak Pin power increased by 2% or more,	
		Increase FW Q(Z) by the appropriate factor and reverify FW Q(Z) is within limits or		increase penalty factor or	
		Repeat SR 3 2 1 2 once per 7 EFPD until two successive flux maps indicate maximum over z [FCQ(Z) / K(Z)] has not increased	7EFPD	measure FQEQ using incore	7 EFPD
				SR Verify within limits for central 80% of core	

Abbreviated Standard TS to KNPP TS
Actions and Completion Times Matrix
RAI Question Attachment 2 Number 3a

Item		STS Action	STS Time	RTSR KNPP Action	RTSR KNPP TS Time
FNDH		Restore to within limits or	4 hours	Restore to within limits or	4 hours
		reduce RTP to < 50%, and	4 hours	reduce RTP to < 50%, and	4 hours
		Reduce NI setpoint to </= 55%, and	72 hours	Reduce NI setpoint to </= 55%, and	72 hours
		verify within limits, and	24 hours	verify within limits, or	24 hours
		verify within limits	prior to exceeding 50%, 75%, and 24 hours after >/= 95%	verify within limits	prior to exceeding 50%, 75%, and 24 hours after >/= 95%
		Required Actions & Completion Times Not Met be in MODE 2	6 hours	reduce RTP to < 5%	2 hours (6 hours)
		SR Verify within limits	prior to exceeding 75% RTP and 31 EFPD thereafter	SR Verify within limits	Following Initial loading and EFPM interval thereafter

Abbreviated Standard TS to KNPP TS
Actions and Completion Times Matrix
RAI Question Attachment 2 Number 3a

Item		STS Action	STS Time	RTSR KNPP Action	RTSR KNPP TS Time
AFD				restore to within limits, or (delete)	15 minutes (delete)
		reduce RTP to < 50%	30 minutes	reduce RTP to < 50%, and	30 minutes
				Reduce NI setpoint to </= 55%, and (delete)	72 hours (delete)
				alarms inoperable verify AFD within limits (delete, TSTF 110 R2)	within one hour and hourly thereafter (delete)
		SR Verify within limits	7 days	none (SR Verify within limits)	none (weekly)

Abbreviated Standard TS to KNPP TS
Actions and Completion Times Matrix
RAI Question Attachment 2 Number 3a

Item		STS Action	STS Time	RTSR KNPP Action	RTSR KNPP TS Time
QPTR		Reduce RTP >/= 3% from RTP for each 1% QPTR 1 00, and	2 hours after each QPTR determination	> 1.02 eliminate tilt or restrict max RTP 2% for every 1% tilt > 1 00, or	2 hours
		verify within limit, and	once per 12 hours	if tilt not eliminated reduce RTP to </= 50%, and	24 hours
		verify FQN and FNDH within limits, and	24 hours after reaching EQ and once per 7 days	> 1 09 and rod misaligned restrict max RTP by 2% for every 1% > 1.00, and	
		reevaluate SA and confirm results remain valid, and	prior to increasing power	eliminate tilt, or	12 hours
		normalize NI's to restore QPTR to within limit, and	prior to increasing power	bring reactor to </= 30 MWe	
		verify FQN and FNDH within limits, or	24 hours after reaching EQ not to exceed 48 hours	> 1 09 and no rod misaligned restrict max RTP to </= 5%	immediately
		Reduce RTP to </= 50%	4 hours		
		SR Verify within limits by calculation	7 days		
		SR Verify within limit using movable incore	12 hours if one or more NI inoperable		
QPTR Monitor				individual upper and lower excore detector caliubrated outputs and the quadrant tilt shal be logged	once per shift, or load change > 10%, or after > 24 steps of control rod motion
DNBR Parameters	Core Average Temperature	Restore to within limits or	2 hours	Restore to within limits or	2 hours
	RCS Pressure	Be in mode 2	6 hours	reduce power to < 5%	additional 6 hours
	Reactor Coolant Flow				

ATTACHMENT D

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

Loss Of Normal Feedwater (LONF)
Notice of Analysis Retraction

Reload Transition Safety Report (RTSR)
Loss Of Normal Feedwater (LONF)
Notice of Analysis Retraction

The Loss of Normal Feedwater (LONF) safety analysis documented in the fuel transition licensee amendment request (LAR) 187 is being retracted and will be superceded by an updated LONF analysis. The updated LONF analysis will be provided with the stretch uprate LAR.

The retraction of the LONF safety analysis is the result of an analysis assumption that is necessary to achieve acceptable analysis results in the Westinghouse safety analysis methodology for the LONF in the fuel transition LAR. The assumption is the crediting of manual operator action to trip the reactor coolant pumps at 15 minutes into the LONF event. The LAR LONF pre-analysis demonstrated the heat input from the reactor coolant pumps late in the transient (>15 minutes) caused the pressurizer to "go solid" (completely fill with water). The LONF acceptance criteria that the pressurizer must not "go solid" would have been violated. Tripping the reactor coolant pumps at 15 minutes into the transient reduced the heat input into the reactor coolant system and enabled an acceptable pressurizer response and therefore this scenario was documented in the LAR.

Following discussions with plant operations staff it was decided that the crediting of manual operator action in this Condition II transient event is not acceptable. As a result an alternate solution that can achieve an acceptable pressurizer response in the LONF event is being pursued. The LONF safety analysis will be presented in the stretch uprate LAR slated to be submitted March of 2003.

The replacement steam generator LONF safety analysis is bounding for operation in Cycle 26 up to the time that the stretch power uprate is implemented. The LONF safety analysis is driven by decay heat and the decay heat model is independent of fuel design (see response to RAI Attachment 3, #47). Furthermore, the Westinghouse 422V+ fuel and the Framatome ANP Heavy fuel are of very similar design and are mechanically, thermal-hydraulically and neutronically compatible. In addition, the LONF Nuclear Steam Supply System (NSSS) models utilize only a limited amount of detail in the fuel-related input assumptions (e.g., fuel and cladding dimensions, cladding material, fuel temperatures, core bypass flow) and the balance of plant models do not utilize any fuel-related input assumptions. Finally, the accident progression for the LONF event and the non-fuel-related acceptance criteria parameters (e.g., RCS pressure, MSS pressure, pressurizer level) are not sensitive to the fuel-related input assumptions. For these reasons, the results of the replacement steam generator LONF analyses are applicable to cores in transition to and operation with 422V+ fuel. Therefore the replacement steam generator LONF safety analysis is valid for Cycle 26 operation up to the time of the implementation of the stretch power uprate.

ATTACHMENT E

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

TS Strike-out Pages

TS 2.3-2
TS 2.3-3
TS 3.10-1 through TS 3.10-10
TS 6.9-5 through 6.9-7
TS Table 4.1-1 (Page 7 of 7)

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \, [\, K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f (\Delta I) \,]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, % ~~RATED POWER~~
T	=	~~Reference~~ Average Temperature, ~~at RATED POWER, °~~°F
T'	≤	[*]°F
P	=	Pressurizer pressure, psig
P'	=	[*] psig
K_1	=	[*]
K_2	=	[*]
K_3	=	[*]
τ_1	=	[*] sec.
τ_2	=	[*] sec.
$f(\Delta I)$	=	An even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, f (ΔI) = 0.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, % ~~RATED POWER~~

T = Average Temperature, °F

T' $\leq$ [*]°F

K_4 $\leq$ [*]

K_5 $\geq$ [*] for increasing T; [*] for decreasing T

K_6 $\geq$ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

$f(\Delta I)$ = 0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop $\geq$ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint $\geq$ 55.0 Hz

2. Low voltage setpoint $\geq$ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level $\geq$ 5% of narrow range instrument span.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure: 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the SHUTDOWN MARGIN shall be at least that as specified in the COLR

b. Power Distribution Limits

1. At all times, except during Low Power Physics Tests, the hot channel factors defined in the basis must meet the following limits:

   A. $F_Q^N(Z)$ Limits shall be as specified in the COLR.
   B. $F_{\Delta H}^N$ Limits shall be as specified in the COLR.

2. If $F_{\Delta H}^N$ not within limits~~exceeds its limit~~:

   A. Perform the following:

      i. Within 4 hours either, restore $F_{\Delta H}^N$ to within its limit or reduce thermal power to less than 50% of RATED POWER

      ii. ~~and r~~Reduce the Power Range Neutron Flux-High Trip Setpoint to ≤ 55% of RATED POWER within ~~the next~~ 72 hours.

      iii. Verify $F_{\Delta H}^N$ within limits within 24 hours

   B. If the actions of TS 3.10.b.2.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power increases~~operation~~ may proceed provided that $F_{\Delta H}^{N}$ is demonstrated, through incore flux mapping, to be within its limits prior to exceeding the following thermal power levels:

i. ~~A nominal~~ 50% of RATED POWER,
ii. ~~A nominal~~ 75% of RATED POWER, and
iii. Within 24 hours of attaining ≥ 95% of RATED POWER

3. If the $F_Q^N(Z)$ equilibrium relationship is not within ~~exceeds~~ its limit:

A. Reduce the thermal power ≥~~at least~~ 1% RATED POWER for each 1% the $F_Q^N(Z)$ equilibrium relationship exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower Δ-T Trip Setpoints within ~~the next~~ 72 hours by ≥~~at least~~ 1% for each 1% $F_Q^N(Z)$ equilibrium relationship exceeds its limit.

B. If the actions of TS 3.10.b.3.A are not completed within the specified time, then reduce thermal power to ≤ 5% of RATED POWER within the next 6 hours.

C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $F_Q^{EQ}(Z)$ transient relationships are within limits~~Identify and correct the cause of the out of limit condition~~ prior to increasing thermal power above the reduced thermal power limit required by action A, above.~~ Thermal power may be increased provided $F_Q^N(Z)$ is demonstrated, through incore flux mapping to be within its limit.~~

~~2.If, for any measured hot channel factor, the relationships of $F_Q^N(Z)$ and $F_{\Delta H}N$ specified in the COLR are not true, then reactor power shall be reduced by a fractional amount of the design power to a value for which the relationships are true, and the high neutron flux trip setpoint shall be reduced by the same fractional amount. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, then the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.~~

~~3.~~4. ~~Following initial loading and at regular effective full-power monthly intervals thereafter, p~~Power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied. (Note: time requirements may be extended by 25%)

A. For $F_Q^N(Z)$ equilibrium relationship, once after each refueling prior to thermal power exceeding 75% of RATED POWER; and once within 12 hours after achieving equilibrium conditions, after exceeding, by ≥ 10% of RATED POWER, the thermal power at which the $F_Q^N(Z)$ equilibrium relationship was last verified; and 31 effective full power days thereafter.

B. For $F_{\Delta H}^{N}$, following each refueling prior to exceeding 75% RATED POWER and 31 effective full power days thereafter.

5. ~~4.~~The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core as specified in the COLR.

6. ~~5.~~Power distribution maps using the movable detector system shall be made to confirm the relationship of $F_Q^{EQ}(Z)$ specified in the COLR according to the following schedules with allowances for a 25% grace period:

A. Once after each refueling prior to exceeding 75% RATED POWER and every 31 effective full power days thereafter~~During the target flux difference determination or once per effective full-power monthly interval, whichever occurs first~~.

B. Once within 12 hours of ~~Upon~~ achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.~~5~~6.A.

C. If a power distribution map measurement indicates that the $F_Q^{EQ}(Z)$ transient relationship's margin to the limit, as specified in the COLR, has decreased since the previous evaluation~~an increase in peak pin power,~~ $\cancel{F_Q^N}$-$\cancel{F_{\Delta H}N}$~~, of 2% or more, due to exposure, when compared to the last power distribution map~~, then either of the following actions shall be taken:

i. $F_Q^{EQ}(Z)$ transient relationship shall be increased by the penalty factor specified in the COLR for comparison to the transient limit as specified in the COLR and reverified within the transient limit~~relationship of~~ $\cancel{F_Q^{EQ}(Z)}$ ~~specified in the COLR~~, or

ii. Repeat the determination of the $F_Q^{EQ}(Z)$ transient relationship~~shall be measured by power distribution maps using the incore movable detector system at least~~ once every seven effective full-power days until either i. above is met, or two successive maps indicate that the $F_Q^{EQ}(Z)$ transient relationship's margin to the transient limit has not decreased~~a power distribution map indicates that the peak pin power,~~ $\cancel{F_Q^N}$-$\cancel{F_{\Delta H}^N}$~~, is not increasing with exposure when compared to the last power distribution map~~.

~~6.~~7. If, for a measured F_Q^{EQ}, the transient relationships of $F_Q^{EQ}(Z)$ specified in the COLR are not within limits~~satisfied and the relationships of~~ $\cancel{F_Q^N(Z)}$ ~~and~~ $\cancel{F_{\Delta H}^N}$ ~~specified in the COLR are satisfied~~, then ~~within 12 hours~~ take ~~one of~~ the following actions:

A. Reduce the axial flux difference limits ≥~~at least~~ 1% for each 1% the $F_Q^{EQ}(Z)$ transient relationship exceeds its limit within 4 hours after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower ΔT Trip Setpoints within ~~the~~ 72 hours by ≥~~at least~~ 1% for each 1% that the maximum allowable power of the axial flux difference limits is reduced.

B. If the actions of TS 3.10.b.7.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $F_Q^{EQ}(Z)$ transient relationships are within limits prior to increasing thermal power above the reduced thermal power limit required by action A. above~~Confirm that the hot channel factor limits of TS 3.10.b.1~~

~~are satisfied prior to increasing thermal power above the maximimmaximum allowable power of the axial flux difference limits.~~

8. Axial Flux Difference

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

   i. If the axial flux difference is not within limits, ~~within 15 minutes restore to within limits. If this action and associated completion time is not met,~~ reduce thermal power to less than 50% RATED POWER within 30 minutes.

~~A. Take corrective actions to improve the power distribution and upon achieving equilibrium conditions measure the target flux difference and verify that the relationships of $F_Q^{EQ}(Z)$ specified in the COLR are satisfied,~~

~~OR~~

~~B. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the left hand sides of the relationships of $F_Q^{EQ}(Z)$ specified in the COLR exceed the limits specified in the right hand sides. Reactor power may subsequently be increased provided that a power distribution map verifies that the relationships of $F_Q^{EQ}(Z)$ specified in the COLR are satisfied with at least 1% of margin for each percent of power level to be increased.~~

~~7. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per full-power month.~~

~~8. The indicated axial flux difference shall be considered outside of the limits of TS 3.10.b.9 through TS 3.10.b.12 when more than one of the OPERABLE excore channels are indicating the axial flux difference to be outside a limit.~~

~~9. Except during physics tests, during excore detector calibration and except as modified by TS 3.10.b.10 through TS 3.10.b.12, the indicated axial flux difference shall be maintained within the target band about the target flux difference as specified in the COLR.~~

~~11. At power levels > 50% and ≤ 90% of rated power:~~

~~A. The indicated axial flux difference may deviate from the target band, specified in the COLR, for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed the outer envelope specified in the COLR. If the cumulative time exceeds one hour, then the reactor power shall be reduced to ≤50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.~~

~~If the indicated axial flux difference exceeds the outer envelope specified in the COLR, then the reactor power shall be reduced to ≤50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤55% of rated power.~~

~~B. A power increase to a level > 90% of rated power is contingent upon the indicated axial flux difference being within its target band.~~

~~12. At a power level no greater than 50% of rated power:~~

~~A. The indicated axial flux difference may deviate from its target band.~~

~~B. A power increase to a level > 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) of the preceding 24-hour period.~~

~~One-half of the time the indicated axial flux difference is out of its target band, up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level ≤90% of rated power.~~

~~13. Alarms shall normally be used to indicate nonconformance with the flux difference requirement of TS 3.10.b.10 or the flux difference time requirement of TS 3.10.b.11.A. If the alarms are temporarily out of service, then the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.~~

c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio > 1.02, one of the following actions shall be taken within two hours:

   A. Eliminate the tilt.

   B. Restrict maximum core power level 2% for every 1% of indicated power tilt ratio > 1.0.

2. If the tilt condition is not eliminated after 24 hours, then reduce power to 50% or lower.

3. Except for Low Power Physics Tests, if the indicated quadrant tilt is > 1.09 and there is simultaneous indication of a misaligned rod:

   A. Restrict maximum core power level by 2% of rated values for every 1% of indicated power tilt ratio > 1.0.

   B. If the tilt condition is not eliminated within 12 hours, then the reactor shall be brought to a minimum load condition (≤30 Mwe).

4. If the indicated quadrant tilt is > 1.09 and there is no simultaneous indication of rod misalignment, then the reactor shall immediately be brought to a no load condition (≤5% reactor power).

d. Rod Insertion Limits

1. The shutdown rods shall be withdrawn to within the limits, as specified in the COLR, when the reactor is critical or approaching criticality.

2. The control banks shall be limited in physical insertion; insertion limits are specified in the COLR. If any one of the control bank insertion limits is not met:

   A. Within one hour, initiate boration to restore control bank insertion to within the limits specified in the COLR, and

B. Restore control bank insertion to within the limits specified in the COLR within two hours of exceeding the insertion limits.

C. If any one of the conditions of TS 3.10.d.2.A or TS 3.10.d.2.B cannot be met, then within one hour action shall be initiated to:

- Achieve HOT STANDBY within 6 hours
- Achieve HOT SHUTDOWN within the following 6 hours

3. Insertion limit does not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin, as specified in the COLR, must be maintained except for the Low Power Physics Test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one high worth rod inserted.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In TS 3.10.e.1 and TS 3.10.e.2, the magnitude, in steps, of an indicated rod misalignment may be determined by comparison of the respective bank demand step counter to the analog individual rod position indicator, the rod position as noted on the plant process computer, or through the conditioning module output voltage via a correlation of rod position vs. voltage. Rod misalignment limitations do not apply during physics testing.

1. When reactor power is ≥85% of rating, the rod cluster control assembly shall be maintained within ± 12 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 12 steps when reactor power is ≥85%, then the rod will be realigned or the core power peaking factors shall be determined within four hours, and TS 3.10.b applied. If peaking factors are not determined within four hours, the reactor power shall be reduced to < 85% of rating.

2. When reactor power is < 85% but ≥ 50% of rating, the rod cluster control assemblies shall be maintained within ± 24 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 24 steps when reactor power is < 85% but ≥ 50%, the rod will be realigned or the core power peaking factors shall be determined within four hours, and TS 3.10.b applied. If the peaking factors are not determined within four hours, the reactor power shall be reduced to < 50% of rating.

3. And, in addition to TS 3.10.e.1 and TS 3.10.e.2, if the misaligned rod cluster control assembly is not realigned within eight hours, the rod shall be declared inoperable.

~~3.~~

f. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service, then:

   A. For operation between 50% and 100% of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) at least once per eight hours, or subsequent to rod motion exceeding a total displacement of 24 steps, whichever occurs first.

   B. During operation < 50% of rating, no special monitoring is required.

2. Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3. If a rod cluster control assembly having a rod position indicator channel out of service is found to be misaligned from TS 3.10.f.1.A, then TS 3.10.e will be applied.

g. Inoperable Rod Limitations

1. An inoperable rod is a rod which does not trip or which is declared inoperable under TS 3.10.e or TS 3.10.h.

2. Not more than one inoperable full length rod shall be allowed at any time.

3. If reactor operation is continued with one inoperable full length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is made OPERABLE earlier. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

h. Rod Drop Time

At OPERATING temperature and full flow, the drop time of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If drop time is > 1.8 seconds, the rod shall be declared inoperable.

i. Rod Position Deviation Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged at least once per eight hours after a load change > 10% of rated power or after > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change > 10% of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

k. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant total flow rate shall be ≥ ~~93,000~~178,000 gallons per minute average ~~per loop~~ and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, ~~between 70%~~at or above 90% ~~and 95%~~ power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in two hours or less to within limits or reduce power to < 5% of thermal rated power within an additional six hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

(3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

(10) WCAP-8745-P-A, “Design Basis for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function”, dated September 1986.

~~(11) WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985. (W Proprietary)~~

~~(12) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.~~

~~(13) WCAP-8385, “Power Distribution Control and Load Following Procedures-Topical Report,” September 1974. (Westinghouse Proprietary)~~

~~(14) WCAP-12610-P-A, “VANTAGE+ Fuel Assembly Reference Core Report,” April 1995. (Westinghouse Proprietary)~~

~~(15) WCAP-11397-P-A, "Revised Thermal Design Procedure, "April 1989.~~

C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

(1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
43. AFW Pump Low Discharge Pressure Trip	Not Applicable	Each refueling cycle	Each refueling cycle	
44. Axial Flux Difference (AFD)	Weekly			Verify AFD within limits for each OPERABLE excore channel

ATTACHMENT F

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

TS Clean Pages

TS 2.3-2
TS 2.3-3
TS 3.10-1 through TS 3.10-7
TS 6.9-4 through 6.9-6
TS Table 4.1-1 (Page 7 of 7)

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \le \Delta T_0 \; [\, K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f(\Delta I) \,]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, %
T	=	Average Temperature, °F
T'	≤	[*]°F
P	=	Pressurizer pressure, psig
P'	=	[*] psig
K_1	=	[*]
K_2	=	[*]
K_3	=	[*]
τ_1	=	[*] sec.
τ_2	=	[*] sec.
$f(\Delta I)$	=	An even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, f (ΔI) = 0.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, %

T = Average Temperature, °F

T' $\leq$ [*]°F

K_4 $\leq$ [*]

K_5 $\geq$ [*] for increasing T; [*] for decreasing T

K_6 $\geq$ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

$f(\Delta I)$ = 0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop $\geq$ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint $\geq$ 55.0 Hz

2. Low voltage setpoint $\geq$ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level $\geq$ 5% of narrow range instrument span.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure: 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the SHUTDOWN MARGIN shall be at least that as specified in the COLR

b. Power Distribution Limits

1. At all times, except during Low Power Physics Tests, the hot channel factors defined in the basis must meet the following limits:

   A. $F_Q^N(Z)$ Limits shall be as specified in the COLR.
   B. $F_{\Delta H}^N$ Limits shall be as specified in the COLR.

2. If $F_{\Delta H}^N$ not within limits:

   A. Perform the following:

      i. Within 4 hours either, restore $F_{\Delta H}^N$ to within its limit or reduce thermal power to less than 50% of RATED POWER

      ii. Reduce the Power Range Neutron Flux-High Trip Setpoint to ≤ 55% of RATED POWER within 72 hours.

      iii. Verify $F_{\Delta H}^N$ within limits within 24 hours

   B. If the actions of TS 3.10.b.2.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power increases may proceed provided that $F_{\Delta H}^{N}$ is demonstrated, through incore flux mapping, to be within its limits prior to exceeding the following thermal power levels:

i. 50% of RATED POWER,
ii. 75% of RATED POWER, and
iii. Within 24 hours of attaining ≥ 95% of RATED POWER

3. If the $F_Q^N(Z)$ equilibrium relationship is not within its limit:

A. Reduce the thermal power ≥ 1% RATED POWER for each 1% the $F_Q^N(Z)$ equilibrium relationship exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% for each 1% $F_Q^N(Z)$ equilibrium relationship exceeds its limit.

B. If the actions of TS 3.10.b.3.A are not completed within the specified time, then reduce thermal power to ≤ 5% of RATED POWER within the next 6 hours.

C. Verify the $F_Q^N(Z)$ equilibrium relationship and the $F_Q^{EQ}(Z)$ transient relationships are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

4. Power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied. (Note: time requirements may be extended by 25%)

A. For $F_Q^N(Z)$ equilibrium relationship, once after each refueling prior to thermal power exceeding 75% of RATED POWER; and once within 12 hours after achieving equilibrium conditions, after exceeding, by ≥ 10% of RATED POWER, the thermal power at which the $F_Q^N(Z)$ equilibrium relationship was last verified; and 31 effective full power days thereafter.

B. For $F_{\Delta H}^{N}$, following each refueling prior to exceeding 75% RATED POWER and 31 effective full power days thereafter.

5. The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core as specified in the COLR.

6. Power distribution maps using the movable detector system shall be made to confirm the relationship of $F_Q^{EQ}(Z)$ specified in the COLR according to the following schedules with allowances for a 25% grace period:

A. Once after each refueling prior to exceeding 75% RATED POWER and every 31 effective full power days thereafter.

B. Once within 12 hours of achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.6.A.

C. If a power distribution map measurement indicates that the $F_Q^{EQ}(Z)$ transient relationship's margin to the limit, as specified in the COLR, has decreased since the previous evaluation, then either of the following actions shall be taken:

i. $F_Q^{EQ}(Z)$ transient relationship shall be increased by the penalty factor specified in the COLR for comparison to the transient limit as specified in the COLR and reverified within the transient limit, or

ii. Repeat the determination of the $F_Q^{EQ}(Z)$ transient relationship once every seven effective full-power days until either i. above is met, or two successive maps indicate that the $F_Q^{EQ}(Z)$ transient relationship's margin to the transient limit has not decreased.

7. If, for a measured F_Q^{EQ}, the transient relationships of $F_Q^{EQ}(Z)$ specified in the COLR are not within limits, then take the following actions:

A. Reduce the axial flux difference limits ≥ 1% for each 1% the F_Q^{EQ} (Z) transient relationship exceeds its limit within 4 hours after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower ΔT Trip Setpoints within 72 hours by ≥ 1% for each 1% that the maximum allowable power of the axial flux difference limits is reduced.

B. If the actions of TS 3.10.b.7.A are not completed within the specified time, then reduce thermal power to ≤ 5% of rated power within the next 6 hours.

C. Verify the $F_Q^{N}(Z)$ equilibrium relationship and the $F_Q^{EQ}(Z)$ transient relationships are within limits prior to increasing thermal power above the reduced thermal power limit required by action A, above.

8. Axial Flux Difference

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

i. If the axial flux difference is not within limits, reduce thermal power to less than 50% RATED POWER within 30 minutes.

c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio > 1.02, one of the following actions shall be taken within two hours:

    A. Eliminate the tilt.

    B. Restrict maximum core power level 2% for every 1% of indicated power tilt ratio > 1.0.

2. If the tilt condition is not eliminated after 24 hours, then reduce power to 50% or lower.

3. Except for Low Power Physics Tests, if the indicated quadrant tilt is > 1.09 and there is simultaneous indication of a misaligned rod:

    A. Restrict maximum core power level by 2% of rated values for every 1% of indicated power tilt ratio > 1.0.

    B. If the tilt condition is not eliminated within 12 hours, then the reactor shall be brought to a minimum load condition (≤30 Mwe).

4. If the indicated quadrant tilt is > 1.09 and there is no simultaneous indication of rod misalignment, then the reactor shall immediately be brought to a no load condition (≤5% reactor power).

d. Rod Insertion Limits

1. The shutdown rods shall be withdrawn to within the limits, as specified in the COLR, when the reactor is critical or approaching criticality.

2. The control banks shall be limited in physical insertion; insertion limits are specified in the COLR. If any one of the control bank insertion limits is not met:

    A. Within one hour, initiate boration to restore control bank insertion to within the limits specified in the COLR, and

    B. Restore control bank insertion to within the limits specified in the COLR within two hours of exceeding the insertion limits.

    C. If any one of the conditions of TS 3.10.d.2.A or TS 3.10.d.2.B cannot be met, then within one hour action shall be initiated to:

        - Achieve HOT STANDBY within 6 hours
        - Achieve HOT SHUTDOWN within the following 6 hours

3. Insertion limit does not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin, as specified in the COLR, must be maintained except for the Low Power Physics Test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one high worth rod inserted.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In TS 3.10.e.1 and TS 3.10.e.2, the magnitude, in steps, of an indicated rod misalignment may be determined by comparison of the respective bank demand step counter to the analog individual rod position indicator, the rod position as noted on the plant process computer, or through the conditioning module output voltage via a correlation of rod position vs. voltage. Rod misalignment limitations do not apply during physics testing.

1. When reactor power is ≥85% of rating, the rod cluster control assembly shall be maintained within ± 12 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 12 steps when reactor power is ≥85%, then the rod will be realigned or the core power peaking factors shall be determined within four hours, and TS 3.10.b applied. If peaking factors are not determined within four hours, the reactor power shall be reduced to < 85% of rating.

2. When reactor power is < 85% but ≥ 50% of rating, the rod cluster control assemblies shall be maintained within ± 24 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 24 steps when reactor power is < 85% but ≥ 50%, the rod will be realigned or the core power peaking factors shall be determined within four hours, and TS 3.10.b applied. If the peaking factors are not determined within four hours, the reactor power shall be reduced to < 50% of rating.

3. And, in addition to TS 3.10.e.1 and TS 3.10.e.2, if the misaligned rod cluster control assembly is not realigned within eight hours, the rod shall be declared inoperable.

f. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service, then:

   A. For operation between 50% and 100% of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) at least once per eight hours, or subsequent to rod motion exceeding a total displacement of 24 steps, whichever occurs first.

   B. During operation < 50% of rating, no special monitoring is required.

2. Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3. If a rod cluster control assembly having a rod position indicator channel out of service is found to be misaligned from TS 3.10.f.1.A, then TS 3.10.e will be applied.

g. Inoperable Rod Limitations

1. An inoperable rod is a rod which does not trip or which is declared inoperable under TS 3.10.e or TS 3.10.h.

2. Not more than one inoperable full length rod shall be allowed at any time.

3. If reactor operation is continued with one inoperable full length rod, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days unless the rod is made OPERABLE earlier. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

h. Rod Drop Time

At OPERATING temperature and full flow, the drop time of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry. If drop time is > 1.8 seconds, the rod shall be declared inoperable.

i. Rod Position Deviation Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged at least once per eight hours after a load change > 10% of rated power or after > 24 steps of control rod motion.

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change > 10% of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

k. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant total flow rate shall be ≥ 178,000 gallons per minute average and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, at or above 90% power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in two hours or less to within limits or reduce power to < 5% of thermal rated power within an additional six hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

(3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) XN-NF-82-06 (P)(A) Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Exxon Nuclear Company, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

(10) WCAP-8745-P-A, “Design Basis for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function”, dated September 1986.

(11) WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985. (W Proprietary)

(12) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(13) WCAP-8385, “Power Distribution Control and Load Following Procedures-Topical Report,” September 1974. (Westinghouse Proprietary)

(14) WCAP-12610-P-A, “VANTAGE+ Fuel Assembly Reference Core Report,” April 1995. (Westinghouse Proprietary)

(15) WCAP-11397-P-A, "Revised Thermal Design Procedure, "April 1989.

C. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

D. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

b. Unique Reporting Requirements

1. Annual Radiological Environmental Monitoring Report

A. Routine Radiological Environmental Monitoring Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the OFF-SITE DOSE CALCULATION MANUAL (ODCM) and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

2. Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit for the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the PCP, and in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

3. Special Reports

A. Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

(1) Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
43. AFW Pump Low Discharge Pressure Trip	Not Applicable	Each refueling cycle	Each refueling cycle	
44. Axial Flux Difference (AFD)	Weekly			Verify AFD within limits for each OPERABLE excore channel

ATTACHMENT G

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

<u>License Amendment Request 187a</u>

TS Basis Strike-out Pages

TS B2.1-1 through TS B2.1-2
TS B3.4-1 through TS B3.4-4

BASIS - Safety Limits-Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNBR is equal to the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below these lines.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

<u>Two departure from nucleate boiling ratio (DNBR) correlations are used in the generation and validation of the safety limit curves: the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are 1.14 for the HTP DNBR correlation, and 1.17 for the WRB-1 DNBR correlation.</u>~~Three departure from nucleate boiling ratio (DNBR) correlations used in the safety analyses: the WRB-1 DNBR correlation, the high thermal performance (HTP) DNBR correlation and the W-3 DNBR correlation. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The W-3 correlation is used when the coolant conditions are outside the range of the WRB-1 correlation or for the analysis of non-HTP FRA-ANP fuel designs and for all fuel designs at low pressure and temperature conditions (e.g., the conditions~~

~~analyzed during a main steam line break accident). DNBR correlations have been qualified and approved for application to Kewaunee. The DNBR limits are 1.14 for the HTP correlation, 1.17 for the WRB-1 correlation, and 1.30 for the W-3 correlation.~~

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure. ~~The maximum full-power steam flow at 1721 1780 MWt is 7,449,000 7,760,000 lbs./hr.~~ This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-heat-sink event. ~~Therefore, the main steam safety valves will be able to relieve the total maximum steam flow if necessary.~~

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System and supplying AFW to separate steam generators. The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following:

1. Throttling the discharge valves on the motor-driven AFW pumps

2. Closing one or both of the cross-connect flow valves

3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on analyses performed by WPSC and the Westinghouse Electric Corporation. These analyses conservatively assumed the plant was at 100% initial power and demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident with no steam generator dryout or core damage. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER to provide further margin in the analysis.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump

2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink

3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses to fulfill the above functions. Since the AFW System is a safety features system, the backup pump is provided. This redundant motor-driven capability is also supplemented by the turbine-driven pump.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. However, analyses from full power demonstrate that initiation of flow can be delayed for at least 10 minutes with no steam generator dryout or core damage. The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

Analyses by WPSC and the Westinghouse Electric Corporation show that AFW-2A and AFW-2B may be in the throttled or closed position, or the AFW pump control switches located in the control room may be in the "pull out" position without a compromise to safety. This does not constitute a condition of inoperability as listed in TS 3.4.b.1 or TS 3.4.b.2. The analysis shows that diverse automatic reactor trips ensure a plant trip before any core damage or system overpressure occurs and that at least 10 minutes are available for the operators to manually initiate auxiliary feedwater flow (start AFW pumps or fully open AFW-2A and AFW-2B) for any credible accident from an initial power of 100%.

The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications.

For all other design basis accidents, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Although analyses have demonstrated that operation with the cross-connect valves closed is acceptable, the TS restrict operation with the valves closed to <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

Condensate Storage Tank (TS 3.4.c)

The specified minimum water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [1]

[1] USAR Section 8.2.4

Secondary Activity Limits (TS 3.4.d)

The maximum dose ~~to the thyroid and whole body~~ that an individual may receive following an accident is specified in GDC 19 and 10 CFR ~~100~~ 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR ~~100~~50.67 limits.

The secondary side of the steam generator's activity is limited to ≤ 0.1 μCi/~~cc~~gram DOSE EQUIVALENT I-131 to ensure the ~~thyroid~~ dose does not exceed the GDC-19 and 10 CFR ~~100~~ 50.67 guidelines. The applicable accidents identified in the USAR[2] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/~~cc~~ gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site ~~thyroid dose~~ doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR ~~100~~ 50.67 limits.

[2] USAR Section 14.0

ATTACHMENT H

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

TS Basis Clean Pages

TS B2.1-1
TS B3.4-1 through TS B3.4-4

BASIS - Safety Limits-Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNBR is equal to the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below these lines.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

Two departure from nucleate boiling ratio (DNBR) correlations are used in the generation and validation of the safety limit curves: the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are 1.14 for the HTP DNBR correlation, and 1.17 for the WRB-1 DNBR correlation.

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure. This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-heat-sink event.

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System and supplying AFW to separate steam generators. The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following:

1. Throttling the discharge valves on the motor-driven AFW pumps
2. Closing one or both of the cross-connect flow valves
3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on analyses performed by WPSC and the Westinghouse Electric Corporation. These analyses conservatively assumed the plant was at 100% initial power and demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident with no steam generator dryout or core damage. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER to provide further margin in the analysis.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump
2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink
3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses to fulfill the above functions. Since the AFW System is a safety features system, the backup pump is provided. This redundant motor-driven capability is also supplemented by the turbine-driven pump.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. However, analyses from full power demonstrate that initiation of flow can be delayed for at least 10 minutes with no steam generator dryout or core damage. The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

Analyses by WPSC and the Westinghouse Electric Corporation show that AFW-2A and AFW-2B may be in the throttled or closed position, or the AFW pump control switches located in the control room may be in the "pull out" position without a compromise to safety. This does not constitute a condition of inoperability as listed in TS 3.4.b.1 or TS 3.4.b.2. The analysis shows that diverse automatic reactor trips ensure a plant trip before any core damage or system overpressure occurs and that at least 10 minutes are available for the operators to manually initiate auxiliary feedwater flow (start AFW pumps or fully open AFW-2A and AFW-2B) for any credible accident from an initial power of 100%.

The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications.

For all other design basis accidents, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Although analyses have demonstrated that operation with the cross-connect valves closed is acceptable, the TS restrict operation with the valves closed to <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

Condensate Storage Tank (TS 3.4.c)

The specified minimum water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [1]

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[1] USAR Section 8.2.4

Secondary Activity Limits (TS 3.4.d)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The secondary side of the steam generator's activity is limited to ≤ 0.1 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[2] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

(2) USAR Section 14.0

ATTACHMENT I

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

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COLR Pages

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Page 6
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2.6 Nuclear Heat Flux Hot Channel Factor ($F_Q^N(Z)$)

2.6.1 $F_Q^N(Z)$ Limits for ~~FRA-ANP~~ Fuel

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.35)/P \times K(Z)$ for $P > 0.5$ **[~~FRA-ANP~~ Hvy]**

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.70) \times K(Z)$ for $P \leq 0.5$ **[~~FRA-ANP~~ Hvy]**

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z)$ for $P > 0.5$ **[~~FRA-ANP~~ Std]**

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.56) \times K(Z)$ for $P \leq 0.5$ **[~~FRA-ANP~~ Std]**

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.50)/P \times K(Z)$ for $P > 0.5$ **[422 V+]**

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (5.00) \times K(Z)$ for $P \leq 0.5$ **[422 V+]**

where:

- P — is the fraction of full power at which the core is OPERATING
- K(Z) — is the function given in Figure 3
- Z — is the core height location for the F_Q of interest

2.6.2 The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core for ~~FRA-ANP~~ fuel:

$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times \cancel{V}W(Z) \leq (2.35)/P \times K(Z)$ **[~~FRA-ANP~~ Hvy]**

$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times \cancel{V}W(Z) \leq (2.28)/P \times K(Z)$ **[~~FRA-ANP~~ Std]**

$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times W(Z) \leq (2.5)/ P \times K(Z)$ **[422 V +]**

where:

- P — is the fraction of full power at which the core is OPERATING
- ~~V~~W(Z) — is defined in COLR Figure 5
- $F_Q^{EQ}(Z)$ — is a measured F_Q distribution obtained during the target flux determination

2.6.3 The penalty factor for TS 3.10.b.5.C.i shall be 2%.

2.9 Overtemperature ΔT Setpoint

Overtemperature ΔT setpoint parameter values:

ΔT_0	=	Indicated ΔT at RATED POWER, % RATED POWER
T	=	reference Average temperature at RATED POWER, °F
T'	≤~~=~~	~~567.3~~573.0 °F
P	=	Pressurizer Pressure, psig
P′	=	2235 psig
K_1	=	~~1.11~~1.20
K_2	=	~~0.0090~~0.015/°F
K_3	=	~~0.000566~~0.00072/psig
τ_1	=	30 seconds
τ_2	=	4 seconds
f(ΔI)	=	An even function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that

(a) For $q_t - q_b$ within -~~12~~22, +~~9~~12 %, f(ΔI) = 0

(b) For each percent that the magnitude of $q_t - q_b$ exceeds +~~9~~12 % the ΔT trip setpoint shall be automatically reduced by an equivalent of ~~2.5~~0.96 % of RATED POWER.

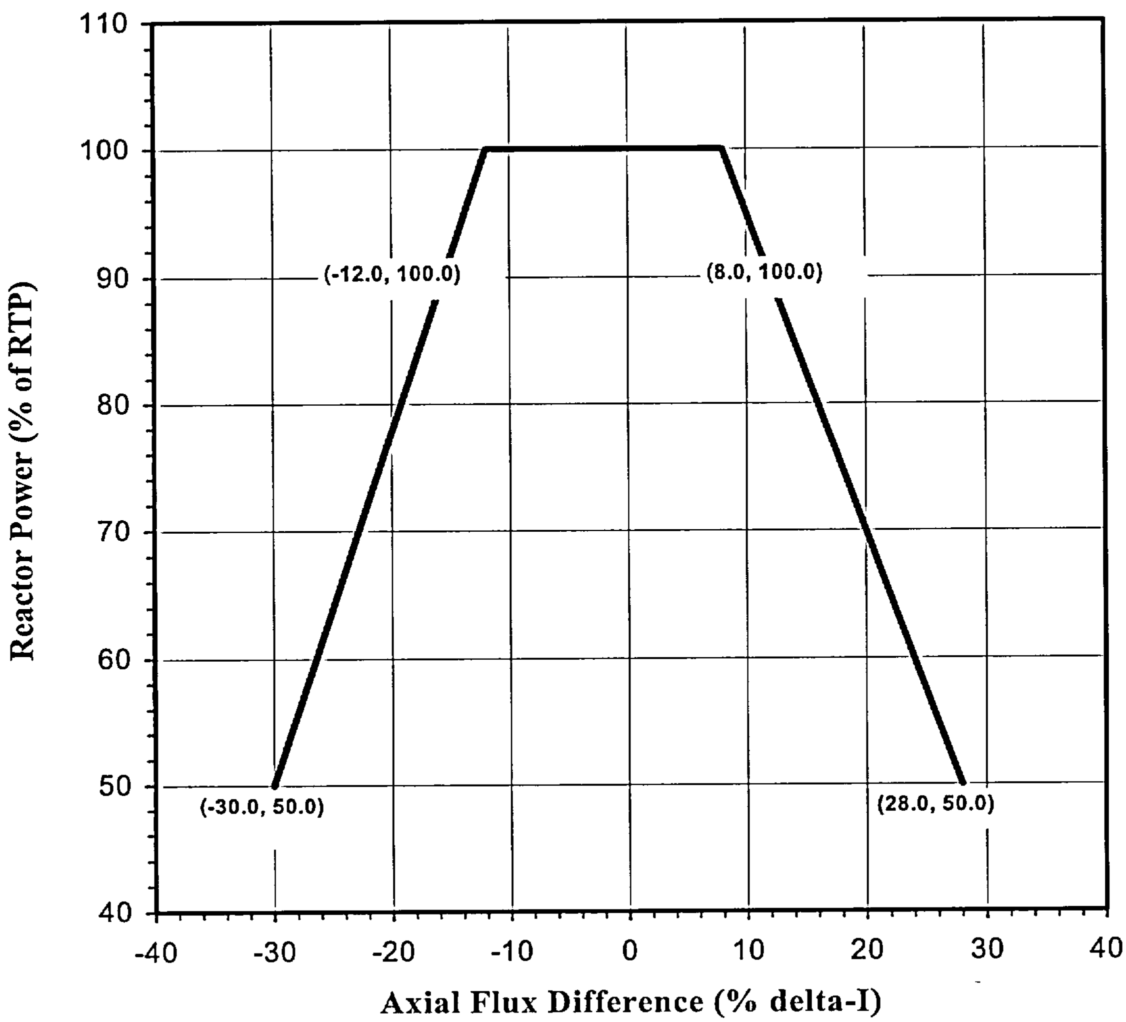
(c) For each percent that the magnitude of $q_t - q_b$ exceed -~~12~~22 % the ΔT trip setpoint shall be automatically reduced by an equivalent of ~~1.5~~0.86 % of RATED POWER.

2.10 Overpower ΔT Setpoint

Overpower ΔT setpoint parameter values:

ΔT_0	=	Indicated ΔT at RATED POWER, % RATED POWER
T	=	reference Average temperature at RATED POWER, °F
T'	≤~~=~~	~~567.3~~573.0 °F
K_4	~~=~~≤	~~1.10~~1.095
K_5	~~=~~≥	~~0.0275~~0.0275/°F for increasing T; 0 for decreasing T
K_6	~~=~~≥	~~0.002~~0.00103/°F for T > T′ ; 0 for T < T′
τ_3	=	10 seconds
f(ΔI)	=	~~Same as in 2.9~~0 for all ΔI

Figure 6
Axial Flux Difference (Typical)



2.6 Nuclear Heat Flux Hot Channel Factor ($F_Q^N(Z)$)

2.6.1 $F_Q^N(Z)$ Limits for Fuel

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.35)/P \times K(Z)$ for $P > 0.5$ **[FRA-ANP Hvy]**

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (4.70) \times K(Z)$ for $P \leq 0.5$ **[FRA-ANP Hvy]**

$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.28)/P \times K(Z)$ for $P > 0.5$ **[FRA-ANP Std]**

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$F_Q^N(Z) \times 1.03 \times 1.05 \leq (2.50)/P \times K(Z)$ for $P > 0.5$ **[422 V+]**

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where:

- P is the fraction of full power at which the core is OPERATING
- K(Z) is the function given in Figure 3
- Z is the core height location for the F_Q of interest

2.6.2 The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the following relationship for the central axial 80% of the core for fuel:

$F_Q^{EQ}(Z) \times 1.03 \times 1.05 \times W(Z) \leq (2.35)/P \times K(Z)$ **[FRA-ANP Hvy]**

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where:

- P is the fraction of full power at which the core is OPERATING
- W(Z) is defined in COLR Figure 5
- $F_Q^{EQ}(Z)$ is a measured F_Q distribution obtained during the target flux determination

2.6.3 The penalty factor for TS 3.10.b.5.C.i shall be 2%.

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P	=	Pressurizer Pressure, psig
P′	=	2235 psig
K_1	=	1.20
K_2	=	0.015/°F
K_3	=	0.00072/psig
τ_1	=	30 seconds
τ_2	=	4 seconds
$f(\Delta I)$	=	An even function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that

(a) For $q_t - q_b$ within -22, +12 %, $f(\Delta I) = 0$

(b) For each percent that the magnitude of $q_t - q_b$ exceeds +12 % the ΔT trip setpoint shall be automatically reduced by an equivalent of 0.96 % of RATED POWER.

(c) For each percent that the magnitude of $q_t - q_b$ exceed -22 % the ΔT trip setpoint shall be automatically reduced by an equivalent of 0.86 % of RATED POWER.

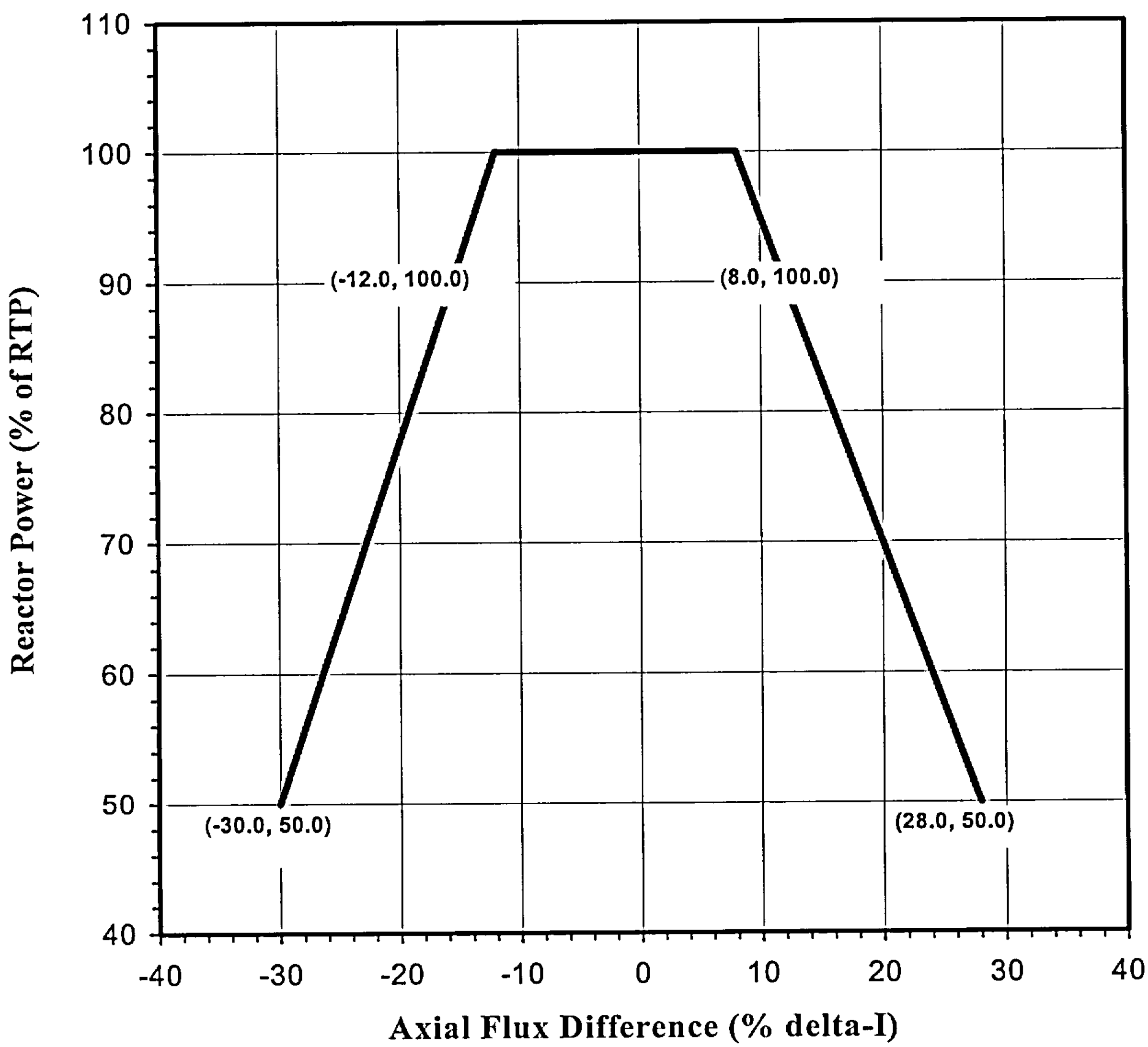
2.10 Overpower ΔT Setpoint

Overpower ΔT setpoint parameter values:

ΔT_0	=	Indicated ΔT at RATED POWER, % RATED POWER
T	=	reference Average temperature at RATED POWER, °F
T'	≤	573.0 °F
K_4	≤	1.095
K_5	≥	0.0275/°F for increasing T; 0 for decreasing T
K_6	≥	0.00103/°F for T > T′ ; 0 for T < T′
τ_3	=	10 seconds
$f(\Delta I)$	=	0 for all ΔI

Figure 6

Axial Flux Difference (Typical)



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