15.4 Reactivity and Power Distribution Anomalies

A number of faults are postulated that result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. These events are discussed in this section. Analyses are presented for the most limiting of these events.

The following incidents are discussed in this section:

A. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition

B. Uncontrolled RCCA bank withdrawal at power

C. RCCA misalignment

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

E. A malfunction or failure of the flow controller in a boiling water reactor recirculation loop that results in an increased reactor coolant flow rate (not applicable to AP600)

F. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant

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

H. Spectrum of RCCA ejection accidents

Items A, B, D, and F above are Condition II events, item G is a Condition III event, and item H is a Condition IV event. Item C includes both Conditions II and III events.

The applicable transients in this section have been analyzed. It has been determined that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing as discussed in subsection 15.4.8.

Radiological consequences are reported only for the limiting case.

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

15.4.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCAs which results in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor subcritical, at hot zero power, or at power. The at-power case is discussed in subsection 15.4.2.

Although the reactor is normally brought to power from a subcritical condition by RCCA withdrawal, initial startup procedures with a clean core use boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see subsection 15.4.6).

The RCCA drive mechanisms are grouped into preselected bank configurations. These groups prevent the RCCAs from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks are withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are the magnetic latch type, and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed is that occurring with the simultaneous withdrawal of the combination of two sequential RCCA banks having the maximum combined worth at maximum speed.

This event is a Condition II event (a fault of moderate frequency) as defined in subsection 15.0.1.

The neutron flux response to a continuous reactivity insertion is characterized by a fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion limits the power during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient is terminated by the following automatic features of the protection and safety monitoring system:

- Source range high neutron flux reactor trip

  This trip function is actuated when two out of four independent source range channels indicate a neutron flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the coincident two out of four intermediate range channels indicate a flux level below a specified level.

- Intermediate range high neutron flux reactor trip

  This trip function is actuated when two out of four independent, intermediate range channels indicate a flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after two out of four power range channels are reading above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (low setting)

  This trip function is actuated when two out of four power range channels indicate a power level above approximately 25 percent of full power. It may be manually bypassed when two out of four power range channels indicate a power level above approximately 10 percent of full power. It is automatically reinstated when the coincident two out of four channels indicate a power level below this value.

- Power range high neutron flux reactor trip (high setting)

  This trip function is actuated when two out of four power range channels indicate a power level above a preset setpoint. It is always active.

- High nuclear flux rate reactor trip

  This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above a preset setpoint. The trip may be manually bypassed after the coincident two out of four nuclear power range channels are manually reset.

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

15.4.1.2 Analysis of Effects and Consequences

15.4.1.2.1 Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation; then, an average core heat transfer calculation; and finally, the departure from nucleate boiling ratio (DNBR) calculation. In the first stage, the average core nuclear calculation is performed using spatial neutron kinetics methods, using the code TWINKLE (Reference 1), to determine the average power generation with time, including the various total core feedback effects (doppler reactivity and moderator reactivity).

In the second stage, the average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN (Reference 2). In the final stage, the average heat flux is used in THINC (described in Section 4.4) for the transient DNBR calculation.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. The following assumptions are made to give conservative results for a startup accident:

- Because the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values, as a function of power, are used (see Table 15.0-2).

- Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. After the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value is used in the analysis to yield the maximum peak heat flux (see Table 15.0-2).

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

- Reactor trip is assumed to be initiated by the power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10-percent uncertainty increase is assumed for the power range flux trip setpoint, raising it to 35 percent from the nominal value of 25 percent.

  Because the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See subsection 15.0.5 for RCCA insertion characteristics.

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

- The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high-worth position, are assumed in the departure from nucleate boiling (DNB) analysis.

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

- Three reactor coolant pumps are assumed to be in operation.
- Pressurizer pressure is assumed to be 50 psi below nominal for steady-state fluctuations and measurement uncertainties.

Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or components adversely affect the consequences of the accident. A loss of offsite power as a consequence of a turbine trip disrupting the grid is not considered because the accident is initiated from a subcritical condition where the plant is not providing power to the grid.

15.4.1.2.2 Results

Figures 15.4.1-1 through 15.4.1-3 show the transient behavior for the uncontrolled RCCA bank withdrawal from subcritical incident. The accident is terminated by reactor trip at 35 percent of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest-worth sequential rod cluster control banks, both assumed to be in their highest incremental worth region.

Figure 15.4.1-1 shows the average neutron flux transient. The energy release and the fuel temperature increases are relatively small. The heat flux response (of interest for DNB considerations) is also shown in Figure 15.4.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full-power nominal value. There is margin to DNB during the transient because the rod surface heat flux remains below the critical heat flux value, and there is a high degree of subcooling at all times in the core. Figure 15.4.1-3 shows the response of the average fuel and cladding temperatures. The minimum DNBR at all times remains above the design limit value (see Section 4.4). Downstream of the mixing vane grids, the WRB-2 correlation is applied and the minimum calculated value is 1.86. Because this transient has a peaked-to-the-bottom axial power shape associated with it, the DNBR is also calculated in the first grid span, which is downstream from a nonmixing vain grid, where the WRB-2 correlation is not applicable. In the first grid span, the W-3 correlation is applied and the minimum calculated DNBR is 1.58.

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

15.4.1.3 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected because the combination of thermal power and the coolant temperature results in a DNBR greater than the safety analysis limit value. Thus, no fuel or cladding damage is predicted as a result of DNB.

15.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Because the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, to avert damage to the fuel cladding, the protection and safety monitoring system is designed to terminate any such transient before the DNBR falls below the design limit (see Section 4.4).

This event is a Condition II incident (a fault of moderate frequency) as defined in subsection 15.0.1.

The automatic features of the protection and safety monitoring system that prevent core damage following the postulated accident include the following:

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

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

- Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to prevent the allowable linear heat generation rate (kW/ft) from being exceeded.

- A high pressurizer pressure reactor trip is actuated from any two out of four pressure channels when a set pressure is exceeded. This set pressure is less than the set pressure for the pressurizer safety valves.

- A high pressurizer water level reactor trip is actuated from any two out of four level channels that exceed the setpoint when the reactor power is above approximately 10 percent (permissive-P10).

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

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

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of reactor coolant system conditions is described in Chapter 7.

Figure 15.0.3-1 presents allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include adverse instrumentation and setpoint uncertainties so that under nominal conditions, a trip occurs well within the area bounded by these lines.

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips:

- High neutron flux (fixed setpoint)
- High pressurizer pressure (fixed setpoint)
- Low pressurizer pressure (fixed setpoint)
- Overpower and overtemperature ΔT (variable setpoints)

For meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, an analysis has been performed to evaluate the effects produced by a possible consequential loss of offsite power during the RCCA withdrawal at-power event. In addressing the loss of offsite power issue, the minimum DNBR cases with full reactor coolant system flow are analyzed assuming that the turbine trip in parallel with the reactor trip causes a subsequent loss of offsite power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

15.4.2.2 Analysis of Effects and Consequences

15.4.2.2.1 Method of Analysis

This transient is primarily analyzed by the LOFTRAN (Reference 3) code. This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generators, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. The core limits as illustrated in Figure 15.0.3-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

For that portion of the RCCA withdrawal at-power analysis that includes a primary coolant flow coastdown caused by the consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analysis. First, the LOFTRAN code is used to predict the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient. The FACTRAN code (Reference 2) is then used to calculate the heat flux based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (see Section 4.4) is used to calculate the DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. In performing a conservative analysis for an uncontrolled RCCA bank withdrawal at-power accident, the following assumptions are made:

- The nominal initial conditions are assumed in accordance with the revised thermal design procedure. Uncertainties in the initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 9).

- Two sets of reactivity coefficients are considered:

  Minimum reactivity feedback – A least-negative moderator temperature coefficient of reactivity is assumed, corresponding to the beginning of core life. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed (see Figure 15.0.4-1).

  Maximum reactivity feedback – A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed (see Figure 15.0.4-1).

- The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include adverse instrumentation and setpoint uncertainties; the delays for trip actuation are assumed to be the maximum values.

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

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

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

In addressing the loss of offsite power issue, the minimum DNBR cases with full reactor coolant system flow are analyzed assuming that the turbine trip in parallel with the reactor trip causes a subsequent loss of offsite power, as described in subsection 15.0.14, the loss of offsite power is modeled to occur 3.0 seconds after the turbine trip. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in these systems or equipment adversely affects the consequences of the accident. A discussion of anticipated transients without scram considerations is presented in Section 15.8.

15.4.2.2.2 Results

Figures 15.4.2-1 through 15.4.2-6 show the transient response for a representative rapid RCCA withdrawal incident starting from full power with offsite power available throughout the transient. Reactor trip on high neutron flux occurs shortly after the start of the accident. Because this is rapid with respect to the thermal time constants of the plant, small changes in temperature and pressure result, and the minimum DNBR is greater than the design limit described in Section 4.4.

The transient response for a representative slow RCCA withdrawal from full power, with offsite power available throughout the transient, is shown in Figures 15.4.2-7 through 15.4.2-12. Reactor trip on overtemperature ΔT occurs after a longer period. The rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. The minimum DNBR is greater than the design limit value described in Section 4.4.

Figure 15.4.2-13 shows the minimum DNBR as a function of reactivity insertion rate from initial full-power operation for minimum and maximum reactivity feedback with offsite power available throughout the transient. Two reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is never less than the design limit value described in Section 4.4.

Figures 15.4.2-14 and 15.4.2-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents for minimum and maximum reactivity feedback, with offsite power available throughout the transient, starting at 60-percent and 10-percent power, respectively. The results are similar to the 100-percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the design limit described in Section 4.4.

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

Referring to Figure 15.4.2-14, for example, it is noted that:

A. For high reactivity insertion rates (between 10 pcm/s and 110 pcm/s), reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. Reactor trip is initiated by high neutron flux for reactivity insertion rates between approximately 85 pcm/s and 110 pcm/s for the maximum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to a significant increase in heat flux or water temperature with resultant high minimum DNBRs during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures remain more nearly in equilibrium with the neutron flux. Thus, minimum DNBR during the transient decreases with decreasing insertion rate.

B. The overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured reactor coolant system average temperature and pressure. This trip circuit is described in Chapter 7. The average temperature contribution to the circuit is lead-lag compensated to decrease the effect of the thermal capacity of the reactor coolant system in response to power increases.

C. For reactivity insertion rates less than 30 pcm/s for the minimum feedback cases, the rise in reactor coolant system pressure is sufficiently high that the pressurizer safety valve setpoint is reached prior to reactor trip. Opening of this valve limits the rise in reactor coolant pressure as the temperature continues to rise. Because the overtemperature ΔT reactor trip setpoint is based on both temperature and pressure, limiting the reactor coolant pressure by opening the pressurizer safety valve brings about the overtemperature ΔT earlier than if the valve remains closed. For this reason, the overtemperature ΔT setpoint initiates reactor trip at reactivity insertion rates of approximately 7 pcm/s for the minimum feedback cases and as high as 75 pcm/s for the maximum feedback cases.

D. For reactivity insertion rates less than approximately 5 pcm/s for the minimum feedback cases and less than approximately 50 pcm/s for maximum feedback cases, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load of the reactor coolant system, sharply decreases the rate of increase of reactor coolant system average temperature. This decrease in the rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation. This causes the overtemperature ΔT setpoint to be reached later, with resulting lower minimum DNBRs.

The delay in overtemperature ΔT reactor trip due to the opening of the steam generator safety valves outweighs the effect of the opening of the pressurizer safety valve (as described in item C) and causes the high neutron flux setpoint to initiate reactor trip for reactivity insertion rates less than approximately 4 pcm/s for cases of minimum feedback and less than approximately 40 pcm/s for cases of maximum feedback. At these slow insertion rates, a sharp decrease in minimum DNBR occurs.

E. With further decrease in reactivity insertion rate, the overtemperature ΔT and high neutron flux trips become equally effective in terminating the transient. For reactivity insertion rates less than approximately 1 pcm/s for minimum feedback cases and insertion rates less than approximately 15 pcm/s for maximum feedback cases, the overtemperature ΔT trip predominates and the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNB) because for these lower reactivity insertion rates, the power increase is slower, the rate of rise of average coolant temperature is slower, and the system lags and delays become less significant.

For transients initiated from full power (see Figure 15.4.2-13), the competing effects due to the opening of the pressurizer safety valve and steam generator safety valves described in items C and D are demonstrated only for the maximum feedback cases. Both the overtemperature ΔT and high neutron flux trips are equally effective in terminating the transient for insertion rates between approximately 10 pcm/s and 20 pcm/s. The effect of the opening of the steam generator safety valves is demonstrated for the reactivity insertion rate of 13 pcm/s where the sharp peak in minimum DNBR occurs.

Transients initiated from 10-percent power (see Figure 15.4.2-15) exhibit the same trends in minimum DNBR, but the results are bounded by those from the transients initiated from higher power levels.

Figures 15.4.2-13, 15.4.2-14, and 15.4.2-15 illustrate minimum DNBR calculated for minimum and maximum reactivity feedback.

Because the RCCA withdrawal at-power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant and the core heat flux lags behind the neutron flux response. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel temperature still remains below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the overtemperature ΔT reactor trip before DNB occurs. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak cladding centerline temperature remains below the fuel melting temperature.

The reactor is tripped fast enough during the RCCA bank withdrawal at-power transient that the ability of the primary coolant to remove heat from the fuel rods is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

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

For the analysis performed modeling a loss of offsite power and the subsequent reactor coolant pump coastdown, the results show that the minimum DNBR is predicted to occur during the time period of the RCCA withdrawal at-power event prior to the time the flow coastdown begins. Therefore, the minimum DNB ratios provided in Figures 15.4.2-6 and 15.4.2-12 through 15.4.2-15 are bounding. The reason for this is that because the loss of offsite power is delayed for 3.0 seconds after the turbine trip signal, the RCCAs are inserted well into the core before the reactor coolant system flow coastdown begins. The resulting power reduction compensates for the reduced flow encountered once ac power to the reactor coolant pumps is lost.

15.4.2.3 Conclusions

The high neutron flux and overtemperature ΔT trip functions provide adequate protection over the entire range of possible reactivity insertion rates. The DNB design basis, as defined in Section 4.4, is met for all cases.

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

15.4.3.1 Identification of Causes and Accident Description

RCCA misoperation accidents include:

- One or more dropped RCCAs within the same group
- Statically misaligned RCCA
- Withdrawal of a single RCCA

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

RCCAs are moved in preselected banks, and the banks are moved in a preselected sequence. Each bank of RCCAs is divided into one or two groups of four or five RCCAs each. The rods comprising a group operate in parallel through a multiplexing system, using thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Because the stationary gripper, movable gripper, and lift coils associated with the RCCAs of a rod group are driven in parallel, any single failure which causes rod withdrawal affects the entire group. A single electrical or mechanical failure in the plant control system could, at most, result in dropping one or more RCCAs within the same group. Mechanical failures can cause either RCCA insertion or immobility, but not RCCA withdrawal.

The dropped RCCAs, dropped RCCA bank, and statically misaligned RCCA events are Condition II incidents (incidents of moderate frequency) as defined in subsection 15.0.1. The single RCCA withdrawal event is a Condition III incident, as discussed below.

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

The event is classified as a Condition III incident consistent with the philosophy and format of American National Standards Institute, ANSI N18.2. By definition, "Condition III occurrences include incidents, any one of which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged . . ." (Reference 10).

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

A dropped RCCA or RCCA bank may be detected by one or more of the following:

- Sudden drop in the core power level as seen by the nuclear instrumentation system
- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod at bottom signal
- Rod deviation alarm
- Rod position indication

Misaligned RCCAs are detected by one or more of the following:

- Asymmetric power distribution as seen by the incore or excore neutron detectors or core exit thermocouples, through online core monitoring
- Rod deviation alarm
- Rod position indicators

The resolution of the rod position indicator channel is ±5 percent span (±7.5 inches). A deviation of any RCCA from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span. If the rod deviation alarm is not operable, the operator takes action as required by the Technical Specifications.

If one or more of the rod position indicator channels is out of service, operating instructions are followed to verify the alignment of the nonindicated RCCAs. The operator also takes action as required by the Technical Specifications.

In the extremely unlikely event of multiple electrical failures that result in single RCCA withdrawal, rod deviation and rod control urgent failure are both displayed on the plant annunciator, and the rod position indicators indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, results in activation of the same alarm and the same visual indication. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the RCCA. Automatic protection for this event is provided by the overtemperature ΔT reactor trip. The Condition III Standard Review Plan Section 15.4.3 evaluation criteria are met; however, due to the increase in local power density, the limits in Figure 15.0.3-1 may be exceeded.

Plant systems and equipment available to mitigate the effects of the various control rod misoperations are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

15.4.3.2 Analysis of Effects and Consequences

15.4.3.2.1 Dropped RCCAs, Dropped RCCA Bank, and Statically Misaligned RCCA

15.4.3.2.1.1 Method of Analysis

- One or more dropped RCCAs from the same group

  A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the core radial power distribution. Depending on the worth and position of the dropped rods, this may cause the allowable design power peaking factors to be exceeded. Following the drop, the reduced core power and continued steam demand to the turbine causes the reactor coolant temperature to decrease. In the manual control mode, the plant will establish a new equilibrium condition. The new equilibrium condition is reached through reactivity feedback. In the presence of a negative moderator temperature coefficient, the reactor power rises monotonically back to the initial power level at a reduced inlet temperature with no power overshoot. The absence of any power overshoot establishes the automatic operating mode as a limiting case. If the reactor coolant system temperature reduction is very large, the turbine power may not be able to be maintained due to the reduction in the secondary-side steam pressure and the volumetric flow limit of the turbine system. In this case, the equilibrium power level is less than the initial power. In the automatic control mode, the plant control system detects the drop in core power and initiates withdrawal of a control bank. Power overshoot may occur, after which the control system will insert the control bank and return the plant to the initial power level. The magnitude of the power overshoot is a function of the plant control system characteristics, core reactivity coefficients, the dropped rod worth, and the available control bank worth.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code (Reference 3). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, pressurizer spray, steam generator and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures and power level.

Steady-state nuclear models using the computer codes described in Table 4.1-2 are used to obtain a hot channel factor consistent with the primary system transient conditions and reactor power. By combining the transient primary conditions with the hot channel factor from the nuclear analysis, the departure from nucleate boiling design basis is shown to be met using the THINC code.

- Statically misaligned RCCA

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

15.4.3.2.1.2 Results

- One or more dropped RCCAs

Figures 15.4.3-1 through 15.4.3-4 show the typical transient response of the reactor to a dropped rod (or rods) in automatic control. The nuclear power and heat flux drop to a minimum value and recover under the influence of both rod withdrawal and thermal feedback. The prompt decrease in power is governed by the dropped rod worth because the plant control system does not respond during the short rod drop time period. The plant control system detects the reduction in core power and initiates control bank withdrawal to restore the primary side power. Power overshoot occurs after which the core power is restored to the initial power level.

The primary system conditions are combined with the hot channel factors from the nuclear analysis for the DNB evaluation. Uncertainties in the initial conditions are included in the DNB evaluation as discussed in subsection 15.0.3.2. The calculated minimum DNBR for the limiting case for any single or multiple rod drop from the same group was 1.97. This is greater than the design limit value described in Section 4.4.

The analysis described previously includes consideration of drops of the RCCA groups which can be selected for insertion as part of the rapid power reduction system. This system is provided to allow the reactor to ride out a complete loss of load from full power without a reactor trip and is described in subsection 7.7.1.10. If these RCCAs are inadvertently dropped (in the absence of a loss-of-load signal), the transient behavior is the same as for the RCCA drop described. The evaluation showed that the DNBR remains above the design limit value as a result of the inadvertent actuation of the rapid power reduction system.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the dropped RCCA event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR occurs before the reactor coolant pumps begin to coast down.

- Statically misaligned RCCA

  The most severe misalignment situations with respect to DNBR arise from cases in which one RCCA is fully inserted, or where the mechanical shim or axial offset rod banks are inserted up to their insertion limit with one RCCA fully withdrawn while the reactor is at full power. Multiple independent alarms, including a bank insertion limit or rod deviation alarm, alert the operator well before the postulated conditions are approached.

  For RCCA misalignments in which the mechanical shim or axial offset banks are inserted to their respective insertion limits, with any one RCCA fully withdrawn, the DNBR remains above the safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and reactor coolant system temperature are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA. Uncertainties in the initial conditions are included in the DNB evaluation as described in subsection 15.0.3.2.

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

  Following the identification of an RCCA group misalignment condition by the operator, the operator takes action as required by the plant Technical Specifications and operating instructions.

15.4.3.2.2 Single Rod Cluster Control Assembly Withdrawal

15.4.3.2.2.1 Method of Analysis

Power distributions within the core are calculated using the computer codes described in Table 4.1-2. The peaking factors are then used by THINC to calculate the DNBR for the event. The case of the worst rod withdrawn from the mechanical shim or axial offset bank inserted at the insertion limit, with the reactor initially at full power, is analyzed. This incident is assumed to occur at beginning of life because this results in the minimum value of moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution.

15.4.3.2.2.2 Results

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

A. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature and an increase in the local hot channel factor in the area of the withdrawing RCCA. In the overall system response, this case is similar to those presented in subsection 15.4.2. The increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum DNBR from falling below the safety analysis limit value. Evaluation of this case at the power and coolant conditions at which the overtemperature ΔT trip is expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety analysis limit value is 5 percent. The limiting dose evaluation is bounded by the locked rotor results presented in subsection 15.3.3.

B. If the reactor is in the automatic control mode, the multiple failures that result in the withdrawal of a single RCCA result in the immobility of the other RCCAs in the controlling bank. The transient then proceeds in the same manner as case A.

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

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the single RCCA withdrawal event. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit value (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.3.3 Conclusions

For cases of dropped RCCAs or dropped banks, including inadvertent drops of the RCCAs in those groups selected to be inserted as part of the rapid power reduction system, it is shown that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For cases of any one RCCA fully inserted, or the mechanical shim or axial offset banks inserted to their rod insertion limits with any single RCCA in one of those banks fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value.

For the case of the accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with the mechanical shim or axial offset banks at their insertion limits, an upper bound of the number of fuel rods experiencing DNB is 5 percent of the total fuel rods in the core.

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

15.4.4.1 Identification of Causes and Accident Description

If the plant is operating with one reactor coolant pump out of service, the affected reactor coolant loop flow rate is less than half of its nominal value. If the reactor is operating at power, the steam generator in the inactive pump loop removes less than half the total power.

Starting an idle reactor coolant pump results in an increase in the injection of cold water into the core, which causes a reactivity insertion and subsequent power increase.

The incident is a Condition II event (a fault of moderate frequency), as defined in subsection 15.0.1.

If the startup of an inactive reactor coolant pump accident occurs, the transient is terminated automatically by a reactor trip on power range high neutron flux setpoint or low flow (P-8 interlock).

15.4.4.2 Analysis of Effects and Consequences

15.4.4.2.1 Method of Analysis

This transient is analyzed using three digital computer codes. The LOFTRAN code (Reference 3) is used to calculate the core flow, nuclear power, and core pressure and temperature transients following the startup of an idle pump. FACTRAN (Reference 2) is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC code (Section 4.4) is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and on heat flux as calculated by FACTRAN.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. To obtain conservative bounding results for the startup of an inactive pump, the following assumptions are made:

- Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to DNB exist. For this analysis, a conservative value of 70-percent nominal power is assumed. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive pump cold leg temperature.

- Following startup of the idle pump, the inactive pump loop flow accelerates to its nominal full-flow value. For this analysis, it is conservatively assumed that the flow rate acceleration occurs in 4 seconds (a linear ramp).

- A conservatively large negative moderator temperature coefficient is assumed.

- A least negative Doppler-only power coefficient is assumed (see Figure 15.0.4-1).

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

- The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power.

Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0-6. No single active failure in these systems or equipment adversely affects the consequences of the accident.

15.4.4.2.2 Results

The results following the startup of an idle pump are shown in Figures 15.4.4-1 through 15.4.4-6.

As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the limit. (See Section 4.4 for a description of the DNBR design basis.)

Reactivity addition for the inactive pump startup is due to the decrease in core inlet water temperature. During the transient, this decrease is due both to the increase in reactor coolant flow and to the colder water entering the core from the cold leg (colder temperature side before the start of the transient) of the previously inactive pump loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown in Figure 15.4.4-1.

The calculated sequence of events for this accident is shown in Table 15.4-1. The transient results illustrated in Figures 15.4.4-1 through 15.4.4-6 indicate that a stabilized plant condition, with the reactor tripped, is rapidly approached.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for this event. The reactor coolant system conditions at the time the reactor coolant pumps would begin to coast down as a result of an assumed loss of offsite power were compared to those at the start of the complete loss of reactor coolant flow analysis presented in subsection 15.3.1. All reactor coolant system conditions (core heat flux, pressure, inlet temperature, and flow) at the time the reactor coolant pumps begin coasting down are the same as or more favorable for DNBR compared to the initial conditions in the complete loss of reactor coolant system flow. In addition, the reactor is tripped and the power trending downward before the reactor coolant pumps begin coasting down. While in the complete loss of reactor coolant system flow transient, reactor trip does not occur until after the underspeed trip setpoint is reached.

15.4.4.3 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the DNB limit, so the DNB design basis as described in Section 4.4 is met.

A startup of an inactive pump with a consequential loss of offsite power is less limiting than the complete loss of reactor coolant system flow presented in subsection 15.3.1, and the DNBR design basis is met.

15.4.5 A Malfunction or Failure of the Flow Controller in a Boiling Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

This subsection is not applicable to the AP600.

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

15.4.6.1 Identification of Causes and Accident Description

One of the two principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system into the reactor coolant system through the reactor makeup portion of the chemical and volume control system. Boron dilution with these systems is manually initiated under strict administrative controls requiring close operator surveillance. Procedures limit the rate and duration of the dilution. A boric acid blend system is available to allow the operator to match the makeup water boron concentration to that of the reactor coolant system during normal charging.

An inadvertent boron dilution is caused by the failure of the demineralized water transfer and storage system or chemical and volume control system, either by controller operator or mechanical failure. The chemical and volume control system and demineralized water transfer and storage system are designed to limit, even under various postulated failure modes, the potential rate of dilution to values that, with indication by alarms and instrumentation, allow sufficient time for automatic or operator response to terminate the dilution.

An inadvertent dilution from the demineralized water transfer and storage system through the chemical and volume control system may be terminated by isolating the makeup pump suction line to the demineralized water transfer and storage system storage tank. Flow from the demineralized water transfer and storage system, which is the source of unborated water, may be terminated by closing isolation valves in the chemical and volume control system. Lost shutdown margin may be regained by opening the isolation valve to the boric acid tank and thus allow the addition of borated water (greater than 4000 ppm) to the reactor coolant system.

Generally, to dilute, the operator performs two actions:

- Switch control of the makeup from the automatic makeup mode to the dilute mode.
- Start the chemical and volume control system makeup pumps.

Failure to carry out either of those actions prevents initiation of dilution. Because the AP600 chemical and volume control system makeup pumps do not run continuously (they are expected to be operated once per day to make up for reactor coolant system leakage), a makeup pump is started when the volume control system is placed into dilute mode.

The status of the reactor coolant system makeup is available to the operator by the following:

- Indication of the boric acid and blended flow rates

- Chemical and volume control system makeup pumps status

- Deviation alarms, if the boric acid or blended flow rates deviate by more than the specified tolerance from the preset values

- Source range neutron flux – when reactor is subcritical

  - High flux at shutdown alarm
  - Indicated source range neutron flux count rates
  - Audible source range neutron flux count rate
  - Source range neutron flux – multiplication alarm

- When the reactor is critical

  - Axial flux difference alarm (reactor power ≥50 percent rated thermal power)
  - Control rod insertion limit low and low-low alarms
  - Overtemperature ΔT alarm (at power)
  - Overtemperature ΔT reactor trip
  - Power range neutron flux – high, both high and low setpoint reactor trips.

This event is a Condition II incident (a fault of moderate frequency), as defined in subsection 15.0.1.

15.4.6.2 Analysis of Effects and Consequences

Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis. Conservative values for necessary parameters are used (high reactor coolant system critical boron concentrations, high boron worths, minimum shutdown margins, and lower-than-actual reactor coolant system volumes). These assumptions (see Table 15.4-2) result in conservative determinations of the time available for operator or automatic system response after detection of a dilution transient in progress.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, a loss of offsite power is considered for the boron dilution case initiated from the power mode of operation (Mode 1) with the reactor in manual control. This is the analyzed Mode 1 boron dilution case that produces a reactor and turbine trip. The loss of offsite power is assumed to occur as a direct result of a turbine trip that would disrupt the grid and produce a consequential loss of offsite ac power. As discussed in subsection 15.0.14, that scenario can occur only with the plant at power and connected to the grid. Therefore, only a boron dilution case initiated from full power will address the consequential loss of offsite power.

15.4.6.2.1 Dilution During Refueling (Mode 6)

An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent dilution is prevented by administrative controls, which isolate the reactor coolant system from the potential source of unborated water by locking closed specified valves in the chemical and volume control system during refueling operations. These valves block the flow paths that allow unborated makeup water to reach the reactor coolant system. Makeup which is required during refueling uses water supplied from the boric acid tank (which contains borated water).

15.4.6.2.2 Dilution During Cold Shutdown (Mode 5)

The following conditions are assumed for inadvertent boron dilution while in this operating mode:

- A dilution flow of 200 gpm of unborated water exists.

- A volume of 2245 ft^3 is a conservative estimate of the minimum active reactor coolant system volume corresponding to the water level at mid-loop in the vessel while on normal residual heat removal. The assumed active volume does not include the volume of the reactor vessel upper head region.

- Control rods are fully inserted, which is the normal condition in cold shutdown and a critical boron concentration of 1315 ppm. This is a conservative boron concentration with control rods inserted and allows for the most reactive rod to be stuck in the fully withdrawn position.

- The shutdown margin is equal to 1.6-percent $\Delta k/k$, the minimum value required by the Technical Specifications for the cold shutdown mode. Combined with the preceding, this gives a shutdown boron concentration of 1489 ppm.

- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that when in Mode 5, at least one reactor coolant pump shall be operable, which provides sufficient flow through the system to maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in section 15.4.6.2.1.

In the event of an inadvertent boron dilution transient during cold shutdown, the source range nuclear instrumentation detects an increase of 60 percent of the neutron flux by comparing the current source range flux to that of about 50 minutes earlier. Upon detection of the flux increase, an alarm is sounded for the operator, and valves are actuated to terminate the dilution automatically.

Upon any reactor trip signal, source range flux multiplication signal, low input voltage to the Class 1E dc and uninterruptable power supply system battery chargers, or safety injection signal, a safety function automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. The suction lines for the chemical and volume control system pumps are automatically realigned to draw borated (greater than 4000 ppm) water from the chemical and volume control system boric acid tank. The realignment of the chemical and volume control system valves to terminate the dilution is a safety-related function. The realignment of pump suction to the boric acid tank is a nonsafety-related operation. The chemical and volume control system pumps are nonsafety-related, so their operation is not credited in the analysis. The analysis does consider the initial portion of this boration phase by treating it as a continuing dilution until any unborated water in the chemical and volume control system lines is purged.

The automatic protective actions initiate about 8.9 minutes after the start of dilution. These automatic actions minimize the approach to criticality and maintain the plant in a subcritical condition. After the automatic protection functions take place, the operator may take action to restore the Technical Specification shutdown margin.

15.4.6.2.3 Dilution During Safe Shutdown (Mode 4)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 200 gpm of unborated water exists.

- Reactor coolant system water volume is 2601 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system while on normal residual heat removal.

- All control rods are fully inserted, except the most reactive rod which is assumed stuck in the fully withdrawn position, and a conservative critical boron concentration of 1303 ppm.

- The shutdown margin is equal to 1.6-percent Δk/k, the minimum value required by the Technical Specifications for the hot shutdown mode. This gives a shutdown boron concentration of 1482 ppm.

- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that when in Mode 5, at least one reactor coolant pump shall be operable, which provides sufficient flow through the system to maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in section 15.4.6.2.1.

In the event of an inadvertent boron dilution transient during hot shutdown, the source range nuclear instrumentation detects an increase of 60 percent of the neutron flux, automatically initiates valve movement to terminate the dilution, and sounds an alarm.

As in Mode 5, the safety analysis considers the potential penalty of the subsequent nonsafety-related boration function by accounting for the purge volume associated with the chemical and volume control system piping. The protective actions initiate about 8.9 minutes after start of dilution. No operator action is required to terminate this transient.

15.4.6.2.4 Dilution During Hot Standby (Mode 3)

The following conditions are assumed for an inadvertent boron dilution while in this mode:

- A dilution flow of 200 gpm of unborated water exists.

- The reactor coolant system volume is 5737 ft^3. This is a conservative estimate of the minimum active volume of the reactor coolant system with the reactor coolant system filled and vented and one reactor coolant pump running.

- Critical boron concentration is 246 ppm. This is a conservative boron concentration assuming control rods are fully inserted minus the most reactive rod, which is assumed stuck in the fully withdrawn position.

- The shutdown margin is equal to 1.6-percent Δk/k, the minimum value required by the Technical Specifications for the hot standby mode. This gives a shutdown boron concentration of 426 ppm.

- The reactor coolant system dilution volume is considered well-mixed. The Technical Specifications require that when in Mode 5, at least one reactor coolant pump shall be operable, which provides sufficient flow through the system to maintain the system well-mixed. If a reactor coolant pump is not operating, the demineralized water isolation valves are closed and an uncontrolled boron dilution transient cannot occur, as discussed in section 15.4.6.2.1.

In the event of an inadvertent boron dilution transient in hot standby, the source range nuclear instrumentation detects an increase of 60 percent of the neutron flux, automatically initiates valve movement to terminate the dilution, and sounds an alarm.

As in the analyses for Modes 4 and 5, the only consideration of the boration function in safety analysis is to account for the additional dilution effect due to the purge volume associated with the chemical and volume control system piping. Protective actions initiate about 116 minutes after start of dilution. No operator action is required to terminate this transient.

15.4.6.2.5 Dilution During Startup (Mode 2)

The plant is in the startup mode only for startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. Normal actions taken to change power level, either up or down, require operator actuation. The Technical Specifications require an available shutdown margin of 1.6-percent $\Delta k/k$ and four reactor coolant pumps operating. Other conditions assumed are the following:

- There is a dilution flow of 200 gpm of unborated water.
- Minimum reactor coolant system water volume is 6579 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.
- An initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits, is 784 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 517 ppm. Full rod insertion, minus the most reactive stuck rod, occurs because of reactor trip.

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

Upon any reactor trip signal, source range flux multiplication signal, low input voltage to the Class 1E dc and uninterruptable power supply system battery chargers, or a safety injection signal, a safety-related function automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

Because the realignment of the suction for the chemical and volume control system pumps to the boric acid tank is a nonsafety-related operation, the only consideration given to the reboration phase of the event in the safety analysis is the unborated chemical and volume control system purge volume.

After reactor trip, the dilution would have to continue for approximately 93.2 minutes to overcome the available shutdown margin. Even assuming that the nonsafety-related boration operation does not occur, the unborated water that may remain in the purge volume of the chemical and volume control system is not sufficient to return the reactor to criticality. Therefore, the automatic termination of the dilution flow from the demineralized water transfer and storage system prevents a post-trip return to criticality.

15.4.6.2.6 Dilution During Full Power Operation (Mode 1)

The plant may be operated at power two ways: automatic T_{avg}/rod control and under operator control. The Technical Specifications require an available shutdown margin of 1.6-percent $\Delta k/k$ and four reactor coolant pumps operating. With the plant at power and the reactor coolant system at pressure, the dilution rate is limited by the capacity of the chemical and volume control system makeup pumps. The analysis is performed assuming two chemical and volume control system pumps are in operation, even though normal operation is with one pump. Conditions assumed for a dilution in this mode are the following:

- There is a dilution flow of 200 gpm of unborated water.

- Minimum reactor coolant system water volume is 6579 ft^3. This is a very conservative estimate of the active reactor coolant system volume, minus the pressurizer volume.

- An initial maximum critical boron concentration, corresponding to the rods inserted to the insertion limits, is 784 ppm. The minimum change in boron concentration from this initial condition to a hot zero power critical condition with all rods inserted is 324 ppm. Full rod insertion, minus the most reactive stuck rod, occurs due to reactor trip.

With the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise causes the reactor to reach the overtemperature ΔT trip setpoint resulting in a reactor trip. Upon any reactor trip signal, a safety-related function automatically isolates the unborated water from the demineralized water transfer and storage system and thereby terminates the dilution. Additionally, the suction lines for the chemical and volume control system pumps are automatically realigned to draw borated water from the chemical and volume control system boric acid tank.

Because the realignment of the suction for the chemical and volume control system pumps to the boric acid tank is a nonsafety-related operation, the only consideration given to the reboration phase of the event in the safety analysis is the unborated purge volume.

After reactor trip, the dilution would have to continue for at least 57.2 minutes to overcome the available shutdown margin. The unborated water that may remain in the purge volume of the chemical and volume control system does not return the reactor to criticality. Therefore, the automatic termination of the dilution flow from the demineralized water transfer and storage system precludes a post-trip return to criticality.

Should a consequential loss of offsite power occur after reactor and turbine trip, it does not alter the fact that the dilution event has been terminated by automatic protection features. As indicated previously, the reactor trip signal that occurs in parallel with the turbine trip will actuate a safety-related function that automatically isolates the unborated water from the demineralized water system and thereby terminates the dilution. A subsequent loss of offsite power will cause the chemical and volume control system pumps to shut down. Should power and chemical and volume control system flow be restored, the unborated water that may remain in the purge volume of the chemical and volume control system will still not return the reactor to criticality.

The boron dilution transient in this case is essentially the equivalent to an uncontrolled rod withdrawal at power. The maximum reactivity insertion rate for a boron dilution transient is conservatively estimated to be 1.6 pcm per second and is within the range of insertion rates analyzed for uncontrolled rod withdrawal at power. Before reaching the overtemperature ΔT reactor trip, the operator receives an alarm on overtemperature ΔT and an overtemperature ΔT turbine runback.

With the reactor in automatic rod control, the pressurizer level controller limits the dilution flow rate to the maximum letdown rate. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller throttles charging flow down to match letdown rate. For the safety analysis, a conservative dilution flow rate of 200 gpm is assumed. With the reactor in automatic rod control, a boron dilution results in a power and temperature increase in such a way that the rod controller attempts to compensate by slow insertion of the control rods. This action by the controller results in at least three alarms to the operator:

A. Rod insertion limit – low level alarm
B. Rod insertion limit – low-low level alarm if insertion continues
C. Axial flux difference alarm (ΔI outside of the target band)

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. The operator has at least 61.64 minutes from the rod insertion limit low-low alarm until shutdown margin is lost at beginning of cycle. The time is significantly longer at end of cycle because of the low initial boron concentration.

Because the analysis for the boron dilution event with the reactor in automatic rod control does not predict a reactor and turbine trip, considering the consequential loss of offsite power for this case is not needed.

The preceding results demonstrate that in all modes of operation, an inadvertent boron dilution is prevented or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition.

15.4.6.3 Conclusions

Inadvertent boron dilution events are prevented during refueling and automatically terminated during cold shutdown, safe shutdown, and hot standby modes. Inadvertent boron dilution events during startup or power operation, if not detected and terminated by the operators, result in an automatic reactor trip. Following reactor trip, automatic termination of the dilution occurs and post-trip return to criticality is prevented.

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

15.4.7.1 Identification of Causes and Accident Description

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

An error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes more peaked than those calculated with the correct enrichments. A 5-percent uncertainty margin is included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The online core monitoring system is used to verify power shapes at the start of life and is capable of revealing fuel assembly enrichment errors or loading errors that cause power shapes to be peaked in excess of the design value. Power-distribution-related measurements are incorporated into the evaluation of calculated power distribution information using the incore instrumentation processing algorithms contained within the online monitoring system. The processing algorithms contained within the online monitoring system are functionally identical to those historically used for the evaluation of power distributions measurements in Westinghouse pressurized water reactors.

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

The power distortion due to a combination of misplaced fuel assemblies could significantly increase peaking factors and is readily observable with the online core monitoring system. The fixed incore instrumentation within the instrumented fuel assembly locations is augmented with core exit thermocouples. There is a high probability that these thermocouples would also indicate any abnormally high coolant temperature rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

This event is a Condition III incident (an infrequent fault) as defined in subsection 15.0.1.

15.4.7.2 Analysis of Effects and Consequences

15.4.7.2.1 Method of Analysis

Steady-state power distributions in the x-y plane of the core are calculated at 30-percent rated thermal power using the three-dimensional nodal code ANC (Reference 7). Representative power distributions in the x-y plane for a correctly loaded core are described in Chapter 4.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown in the incore detector locations. (See Figures 15.4.7-1 through 15.4.7-4.)

15.4.7.2.2 Results

The following core loading error cases are analyzed:

Case A:

Case in which a Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered is the interchange of two assemblies near the periphery of the core (see Figure 15.4.7-1).

Case B:

Case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. For the particular case considered, the interchange is assumed to take place close to the core center and with burnable poison rods located in the correct Region 2 position, but in a Region 1 assembly mistakenly loaded in the Region 2 position (see Figure 15.4.7-2).

Case C:

Enrichment error – Case in which a Region 2 fuel assembly is loaded in the core central position (see Figure 15.4.7-3).

Case D:

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (see Figure 15.4.7-4).

15.4.7.3 Conclusions

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

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

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

15.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

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

15.4.8.1.1 Design Precautions and Protection

15.4.8.1.1.1 Mechanical Design

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

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

- The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head. The housings are checked during the hydrotest of the completed reactor coolant system.

- Stress levels in the mechanism are not affected by anticipated system transients at power or by the thermal movement of the coolant loops. Moments induced by the safe shutdown earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class 1 components.

- The latch mechanism housing and rod travel housing are each a single length of forged stainless steel. This material exhibits excellent notch toughness at 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 confidence that gross failure of the housing does not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds, which are subject to periodic inspections.

15.4.8.1.1.2 Nuclear Design

If a rupture of an RCCA drive mechanism housing is postulated, the operation using chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the power control (or mechanical shim) RCCAs inserted only far enough to permit load follow. The axial offset RCCAs are positioned so that the targeted axial offset can be met throughout core life. Reactivity changes caused by core depletion and xenon transients are normally compensated for by boron changes and the mechanical shim banks, respectively. Further, the location and grouping of the power control and axial offset RCCAs are selected with consideration for an RCCA ejection accident. Therefore, should an RCCA be ejected from its normal position during full-power operation, a less severe reactivity excursion than analyzed is expected.

It may occasionally be desirable to operate with larger than normal insertions. For this reason, a power control and axial offset rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit provides adequate shutdown capability and an acceptable power distribution. The position of the RCCAs is continuously indicated in the main control room. An alarm occurs if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. Operating instructions require boration at the low level alarm and emergency boration at the low-low level alarm.

15.4.8.1.1.3 Reactor Protection

The reactor protection in the event of a rod ejection accident is described in WCAP-7588, Revision 1A (Reference 4). The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

15.4.8.1.1.4 Effects on Adjacent Housings

Failures of an RCCA mechanism housing, due to either longitudinal or circumferential cracking, does not cause damage to adjacent housings. The control rod drive mechanism is described in subsection 3.9.4.1.1.

15.4.8.1.1.5 Effects of Rod Travel Housing Longitudinal Failures

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

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

15.4.8.1.1.6 Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing occurs, the broken-off section of the housing is ejected vertically because the driving force is vertical and the position indicator coil assembly and the drive shaft tend to guide the broken-off piece upward during its travel. Travel is limited by the missile shield and thereby limits the projectile acceleration. When the projectile reaches the missile shield, it partially penetrates the shield dissipating its kinetic energy. The water jet from the break continues to push the broken-off piece against the missile shield.

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

15.4.8.1.1.7 Consequences

The probability of damage to an adjacent housing is considered remote. If damage is postulated, it is not expected to lead to a more severe transient because RCCAs are inserted in the core in symmetric patterns and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal. This is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

15.4.8.1.1.8 Summary

Failure of a control rod housing, due either to longitudinal or circumferential cracking, does not cause damage to adjacent housings that increase the severity of the initial accident.

15.4.8.1.2 Limiting Criteria

This event is a Condition IV incident (ANSI N18.2). See subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project (Reference 5). Extensive tests of uranium dioxide (UO_2) zirconium-clad fuel rods representative of those in pressurized water reactor cores such as AP600 have demonstrated failure thresholds in the range of 240 to 257 cal/g. Other rods of a slightly different design have exhibited failure as low as 225 cal/g. These results differ significantly from the TREAT (Reference 6) results, which indicated a failure threshold of 280 cal/g. Limited results indicate that this threshold decreases by about 10 percent with fuel burnup. The cladding failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods.

Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/g for unirradiated rods and 200 cal/g for irradiated rods. Catastrophic failure (large fuel dispersal, large pressure rise), even for irradiated rods, did not occur below 300 cal/g.

Regulatory Guide 1.77 criteria are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- Average fuel pellet enthalpy at the hot spot is below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.
- Peak reactor coolant pressure is less than that which could cause stresses to exceed the "Service Limit C" as defined in the ASME code.
- Fuel melting is limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of the first criterion.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

The calculation of the RCCA ejection transients is performed in two stages: first, an average core channel calculation and then, a hot region calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity). Enthalpy and temperature transients at the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.

A discussion of the method of analysis appears in WCAP-7588, Revision 1A (Reference 4).

Average Core Analysis

The spatial kinetics computer code TWINKLE (Reference 1) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in 1, 2, or 3 spatial dimensions (rectangular coordinates) for 6 delayed neutron groups and up to 2000 spatial points. The computer code includes a multiregion, transient fuel-clad-coolant heat transfer model for the calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one-dimensional axial kinetics code because it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. Because the radial dimension is missing, it is necessary to use conservative methods (described as follows) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in subsection 15.0.11.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal value multiplied by the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. The assumption is made that the hot spots before and after ejection are coincident. This is conservative because the peak after ejection occurs in or adjacent to the assembly with the ejected rod, and before ejection, the power in this region is depressed.

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

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB and the Bishop-Sandburg-Tong correlation (Reference 8) to determine the film boiling coefficient after DNB. The Bishop-Sandburg-Tong correlation is conservatively used, assuming zero-bulk fluid quality. The DNBR is not calculated. Instead, the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient is calculated by the code. It is adjusted to force the full power, steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in subsection 15.0.11.

System Overpressure Analysis

There is little likelihood of fuel dispersal into the coolant. The pressure surge may be calculated on the basis of conventional heat transfer from the fuel and prompt heat absorption by the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC (Section 4.4) calculation is performed to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

15.4.8.2.1 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. Table 15.4-3 presents the important parameters used in this analysis.

15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

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

Appropriate safety analysis margins are added to the ejected rod worth and hot channel factors to account for calculational uncertainties, including an allowance for nuclear peaking due to densification.

Power distributions before and after ejection for a worst case can be found in WCAP-7588, Revision 1A (Reference 4). During plant startup physics testing, rod worths and power distributions have been measured in the zero-power configuration and compared to values used in the analysis. The ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

15.4.8.2.1.2 Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. This means that the reactivity feedback is larger than that indicated by a simple single channel analysis.

Physics calculations are carried out for temperature changes with a flat temperature distribution and with a large number of axial and radial temperature distributions. Reactivity changes are compared, and effective reactivity feedback weighting factors are shown to be conservative. These weighting factors take the form of multipliers that, when applied to single-channel feedbacks, correct them to effective whole-core feedbacks for the appropriate flux shape.

In this analysis, because a one-dimensional (axial) spatial kinetics method is used, axial reactivity weighting is not necessary if the initial condition matches the ejected rod configuration. In addition, no reactivity weighting is applied to the moderator feedback.

A conservative radial reactivity weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time, accounting for the missing spatial dimension. These reactivity weighting factors are shown to be conservative compared to three-dimensional analysis (Reference 5).

15.4.8.2.1.3 Moderator and Doppler Coefficients

The critical boron concentrations at the beginning of cycle and end of cycle are adjusted in the nuclear code to obtain moderator density coefficient curves that are conservative compared to actual design conditions for the plant. No weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one-dimensional, steady-state computer code with a Doppler weighting factor of one. The Doppler defect used is given in subsection 15.0.4. The Doppler weighting factor increases under accident conditions.

15.4.8.2.1.4 Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70 percent at beginning of cycle and 0.50 percent at end of cycle for the first cycle. The accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff} as in zero-power transients. To allow for future cycles, pessimistic estimates of β_{eff} of 0.55 percent at beginning of cycle and 0.44 percent at end of cycle are used in the analysis.

15.4.8.2.1.5 Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-3 and includes the effect of one stuck RCCA. These values are reduced by the ejected rod reactivity. The shutdown reactivity is simulated by dropping a rod of the required worth into the core. The start of rod motion occurs 0.5 second after the high neutron flux trip setpoint is reached. This delay is assumed to consist of 0.2 second for the instrument channel to produce a signal, 0.15 second for the trip breakers to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.4 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power accidents.

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

15.4.8.2.1.6 Reactor Protection

As discussed in subsection 15.4.8.1.1.3, reactor protection for a rod ejection is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are part of the protection and safety monitoring system. No single failure of the protection and safety monitoring system negates the protection functions required for the rod ejection accident or adversely affects the consequences of the accident.

15.4.8.2.1.7 Results

Because the control rod insertion limits for the AP600 are multidimensional, a significant number of rodded configurations are evaluated to determine the most limiting cases, (that is, those cases that produced the least amount of margin to the Standard Review Plan Section 15.4.8 evaluation acceptance criteria). The hot zero power cases and hot full power cases assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event. The limiting RCCA ejection cases, for both the beginning and end of cycle at zero and full power, are presented next.

- Beginning of cycle, full power

  The limiting ejected rod worth and hot channel factor are conservatively assumed to be 0.38-percent Δk and 7.0, respectively. The peak hot spot cladding average temperature is 2597°F. The peak hot spot fuel temperature reaches melting at 4900°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.

- Beginning of cycle, zero power

  For this condition, the limiting ejected rod worth and hot channel factor are conservatively assumed to be 0.744-percent Δk and 13.0, respectively. The peak hot spot cladding average temperature is 2795°F, and the peak hot spot fuel temperature is 3881°F.

- End of cycle, full power

  The ejected rod worth and hot channel factor are conservatively assumed to be 0.33-percent Δk and 9.0, respectively. The peak hot spot cladding average temperature is 2523°F. The peak hot spot fuel temperature reaches melting at 4800°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.

- End of cycle, zero power

  The ejected rod worth and hot channel factor for this case are conservatively assumed to be 0.83-percent Δk and 16.0, respectively. The peak hot spot cladding average temperature is 2901°F, and the peak hot spot fuel temperature is 3862°F.

A summary of the preceding cases is given in Table 15.4-3. The nuclear power and fuel and cladding temperature transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-4.

The calculated sequence of events for the limiting case rod ejection accidents, as shown in Figures 15.4.8-1 through 15.4.8-4, is presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents (LOCAs) are discussed in subsection 15.6.5. Following the RCCA ejection, the plant response is the same as a LOCA.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the enthalpy and temperature transients resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak fuel and cladding temperatures occur before the reactor coolant pumps begin to coast down.

15.4.8.2.1.8 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 15 percent of the rods are assumed to enter DNB based on a detailed three-dimensional THINC analysis (Reference 4). Although limited (less than 10 percent) fuel melting at the hot spot is allowed for the full-power cases, in practice, melting is not expected because the analysis conservatively assumes that the hot spots before and after ejection are coincident.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the calculation of the number of rods assumed to enter DNB for the RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the minimum DNBR, for rods where the DNBR did not fall below the design limit (see Section 4.4) in the cases described, occurs before the reactor coolant pumps begin to coast down.

15.4.8.2.1.9 Pressure Surge

A calculation of the pressure surge for an ejection worth of about one dollar at beginning of cycle, hot full power, demonstrates that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. Because the severity of the analysis does not exceed the worst-case analysis, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the pressure surge transient resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak system pressure occurs before the reactor coolant pumps begin to coast down.

15.4.8.2.1.10 Lattice Deformations

A large temperature gradient exists in the region of the hot spot. Because the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion, tending to bow the midpoint of the rods toward the hotter side of the rod.

Calculations indicate that this bowing results in a negative reactivity effect at the hot spot because the core is undermoderated, and bowing tends to increase the undermoderation at the hot spot. In practice, no significant bowing is anticipated because the structural rigidity of the core is sufficient to withstand the forces produced.

Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that crossflow is sufficient to produce lattice deformation. Even if massive and rapid boiling, sufficient to distort the lattices, is hypothetically postulated, the large void fraction in the hot spot region produces a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. The net effect is therefore a negative feedback.

In conclusion, no credible mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

15.4.8.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes result in a buildup of activity in the secondary coolant.

As a result of the accident, 15 percent of the fuel rods are assumed to be damaged (see subsection 15.4.8.2.1.8) such that the activity contained in the fuel-cladding gap is released to the reactor coolant. In addition, a small fraction of fuel is assumed to melt and release core inventory to the reactor coolant.

Activity released to the containment via the spill from the reactor vessel head is assumed to be available for release to the environment because of containment leakage. Activity carried over to the secondary side due to primary-to-secondary leakage is available for release to the environment through the steam line safety or power-operated relief valves.

15.4.8.3.1 Source Term

The significant radionuclide releases due to the rod ejection accident are the iodines, cesiums, and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The initial reactor coolant noble gas concentrations are assumed to be those associated with the design fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG-1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fraction is increased to 3.6 percent of the inventory to address concerns identified in NUREG-1465 regarding the applicability of the 3-percent gap fraction to high burnup fuel (that is, fuel with burnup in excess of 40-gigawatt days per metric ton of uranium).

Even though no fuel centerline melting is expected, a conservative upper limit for fuel melting was determined to be 0.375 percent of the core based on the following assumptions:

1. No more than 50 percent of the rods experiencing clad damage will experience centerline melting. (Based on 15 percent of rods failing, this is 7.5 percent of the core.)

2. Due to the power distribution within the core, no more than 50 percent of the axial length of the affected fuel rods will experience melting. (This reduces the equivalent number of rods experiencing melting to 3.75 percent of the core.)

3. Of rods experiencing centerline melting, only a conservative maximum of the innermost 10 percent of the fuel volume will actually melt. (Based on 3.75 percent of the rods experiencing melting, the resulting fraction of the core experiencing melting is 0.375 percent.)

All of the noble gases and half of the iodines and cesiums are assumed to be released from the melted fuel.

The initial secondary coolant activity is assumed to be 0.04 μCi/g dose equivalent I-131. This is 10 percent of the design basis primary coolant activity.

15.4.8.3.2 Release Pathways

There are three components to the accident releases:

- The activity initially in the secondary coolant is available for release as long as steam releases continue.

- The reactor coolant leaking into the steam generators is assumed to mix with the secondary coolant. The activity from the primary coolant mixes with the secondary coolant and, as steam is released, a portion of the iodine and cesium in the coolant is released. The fraction of activity released is defined by the partition coefficient assumed for the steam generator. The noble gas activity entering the secondary side is released to the environment. These releases are terminated when the steam releases stop.

- The activity from the reactor coolant system and the core is released to the containment atmosphere and is available for leakage to the environment through the assumed design basis containment leakage.

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion by ground deposition during transport offsite.

15.4.8.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.4.8.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.4-4.

15.4.8.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of conservatisms:

- Although fuel damage is assumed to occur as a result of the accident, no fuel damage is anticipated.

- The reactor coolant activities are based on an assumed fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less than this (see Section 11.1).

- The leakage of reactor coolant into the secondary system, at 1000 gallons per day, is conservative. The leakage is normally a small fraction of this.

- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

- The leakage from containment is assumed to continue for a full 30 days. It is expected that containment pressure is reduced to the point that leakage is negligible before this time.

15.4.8.3.6 Doses

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be 2 rem at the site boundary and 1 rem at the low population zone outer boundary. These doses are well within the proposed dose guideline of 25 rem total effective dose equivalent identified in 10 CFR Part 50.34. The phrase "well within" is taken as being 25 percent or less.

At the time the rod ejection accident occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because the pool boiling would not occur until after the first 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE, and when this is added to the dose calculated for the rod ejection accident, the resulting total dose remains less than 1.0 rem TEDE.

15.4.9 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.4.10 References

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Table 15.4-1 (Sheet 1 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	Power range high neutron flux (low setting) setpoint reached	10.3
	Peak nuclear power occurs	10.5
	Rods begin to fall into core	10.8
	Minimum DNBR occurs	12.6
	Peak average clad temperature occurs	12.8
	Peak heat flux occurs	12.8
	Peak average fuel temperature occurs	13.0
One or more dropped RCCAs	Rods drop	0.0
	Control system initiates control bank withdrawal	0.4
	Peak nuclear power occurs	224.0
	Minimum DNBR occurs	228.0

Table 15.4-1 (Sheet 2 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
Uncontrolled RCCA bank withdrawal at power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high-reactivity insertion rate (75 pcm/s)	0.0
	Power range high neutron flux high trip point reached	4.9
	Rods begin to fall into core	5.8
	Minimum DNBR occurs	6.3
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/s)	0.0
	Overtemperature ΔT setpoint reached	718.7
	Rods begin to fall into core	720.7
	Minimum DNBR occurs	721.1
Startup of inactive reactor coolant loop at an incorrect temperature	Initiation of pump startup	0.0
	Power reached P8 trip setpoint	0.9
	Rods begin to drop	1.8
	Minimum DNBR occurs	3.2

Table 15.4-1 (Sheet 3 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the rector coolant		
1. Dilution during startup	Power range – low setpoint reactor trip due to dilution	0.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	170
2. Dilution during full-power operation		
a. Automatic reactor control	Operator receives low-low rod insertion limit alarm due to dilution	0.0
	Shutdown margin lost	3698
b. Manual reactor control	Initiate dilution	0.0
	Reactor trip on overtemperature ΔT due to dilution	265.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	435.0
RCCA ejection accident		
1. End of cycle, full power	Initiation of rod ejection	0.00
	Power range high neutron flux (high setting) setpoint reached	0.03
	Peak nuclear power occurs	0.14
	Rods begin to fall into core	0.53
	Peak fuel average temperature occurs	2.32
	Peak cladding temperature occurs	2.39
	Peak heat flux occurs	2.40

Table 15.4-1 (Sheet 4 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
2. Beginning of cycle, zero power	Initiation of rod ejection	0.00
	Power range high neutron flux (low setting) setpoint reached	0.32
	Peak nuclear power occurs	0.39
	Rods begin to fall into core	0.82
	Peak cladding temperature occurs	2.58
	Peak heat flux occurs	2.61
	Peak fuel average temperature occurs	2.61
3. Beginning of cycle, full power	Initiation of rod ejection	0.00
	Power range high neutron flux (high setting) setpoint reached	0.03
	Peak nuclear power occurs	0.14
	Rods begin to fall into core	0.53
	Peak fuel average temperature occurs	2.22
	Peak cladding temperature occurs	2.31
	Peak heat flux occurs	2.31

Table 15.4-1 (Sheet 5 of 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
4. End of cycle, zero power	Initiation of rod ejection	0.00
	Power range high neutron flux (low setting) setpoint reached	0.20
	Peak nuclear power occurs	0.24
	Rods begin to fall into core	0.70
	Peak cladding temperature occurs	1.81
	Peak heat flux occurs	1.81
	Peak fuel average temperature occurs	2.00

Table 15.4-2

PARAMETERS

Assumed Dilution Flowrates

Mode	Flow Rate (gal/min)
1 through 5	200

Volumes

Mode	Volume (ft^3)	Volume (gal)
1 and 2	6579	49,211
3	5737	42,913
4	2601	19,455
5	2245	16,793

Table 15.4-3

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	HZP[1] Beginning	HFP[2] Beginning	HZP End	HFP End
Power level (%)	0	102	0	102
Ejected rod worth (%Δk)	0.744	0.38	0.83	0.33
Delayed neutron fraction (%)	0.55	0.55	0.44	0.44
Feedback reactivity weighting	2.071	1.30	2.32	1.60
Trip reactivity (%Δk)	2.0	4.0	2.0	4.0
F_q before rod ejection	–	2.756	–	2.756
F_q after rod ejection	13.0	7.0	16.0	9.0
Number of operational pumps	2	4	2	4
Maximum fuel pellet average temperature (°F)	3470	4138	3508	4050
Maximum fuel center temperature (°F)	3881	4956	3862	4861
Maximum cladding average temperature (°F)	2795	2597	2901	2523
Maximum fuel stored energy (cal/g)	147	182	149	177
Percent of fuel melted at hot spot	0	<10	0	<10

Notes:

1. HZP – Hot zero power
2. HFP – Hot full power

Table 15.4-4 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 24 μCi/g of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine activity	0.04 μCi/g dose equivalent I-131 (10% of design basis reactor coolant concentrations listed in Table 11.1-2)
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.15
– Fission product gap fractions	0.036
Core melting	
– Fraction of core melting	0.00375
– Fraction of activity released	
Iodines and cesiums	0.5
Noble gases	1.0
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3 in Appendix 15A
Nuclide data	See Table 15A-4 in Appendix 15A
Reactor coolant mass (lb)	3.6 E+05

Note:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity assumed to be released from damaged fuel, it is not significant.

Table 15.4-4 (Sheet 2 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	260[a]
– Secondary coolant mass (lb)	2.45E+05
– Duration of steam release from secondary system (sec)	1400
– Steam released from secondary system (lb)	2.45 E+05
– Partition coefficient in steam generators	0.01
Containment leakage release path	
– Containment leak rate (% per day)	0.10
– Airborne activity removal coefficients (hr^{-1})	
Elemental iodine	2.0[b]
Organic iodine	0
Particulate iodine or cesium	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	2.6

Notes:

a. Equivalent to 1000 gpd at 561.5°F and 2250 psia
b. From Appendix 15B

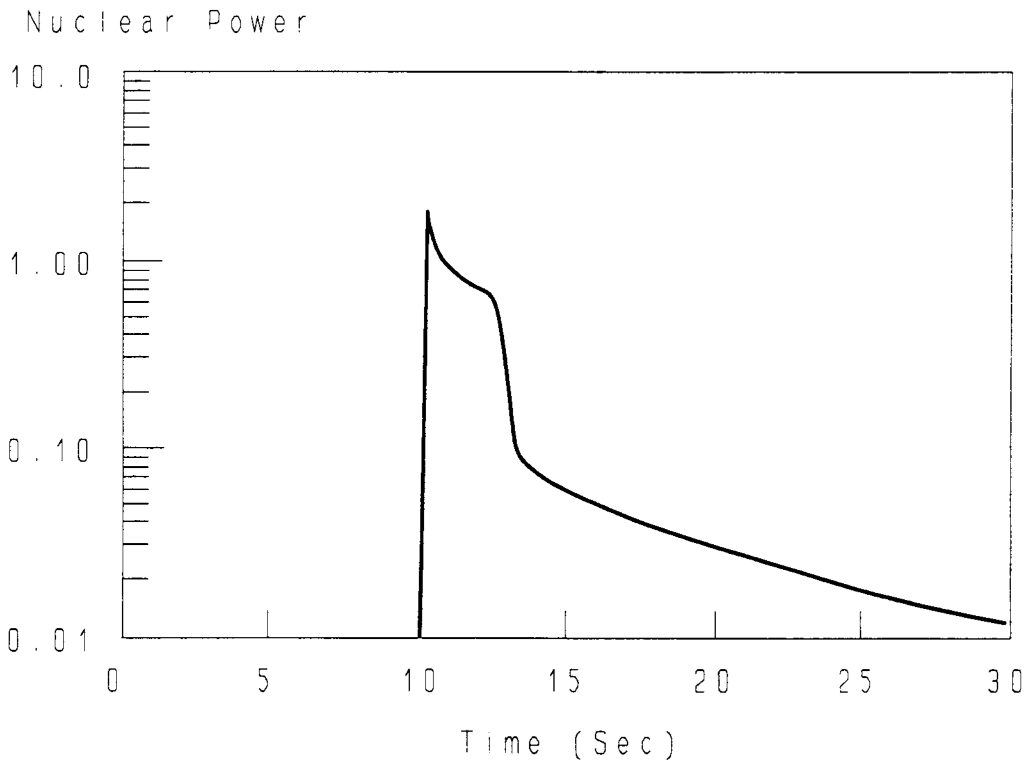

Figure 15.4.1-1

RCCA Withdrawal from Subcritical Nuclear Power

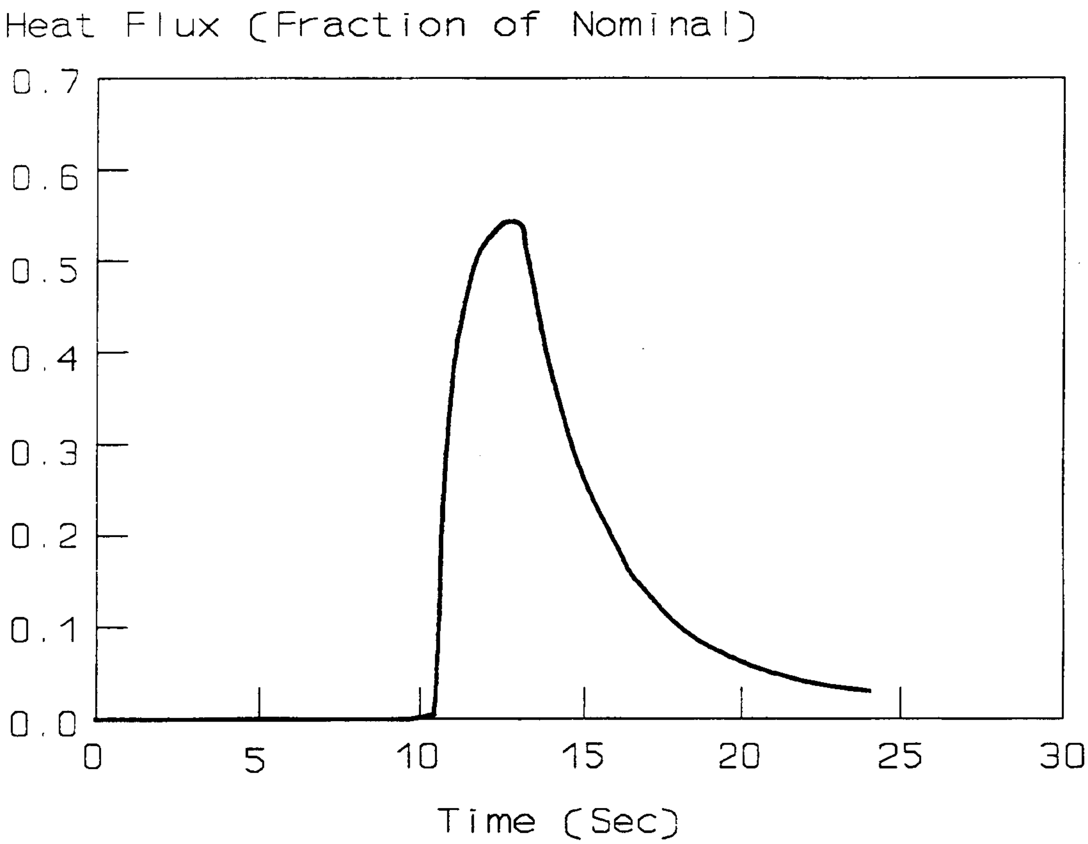

Figure 15.4.1-2

RCCA Withdrawal from Subcritical Thermal Flux

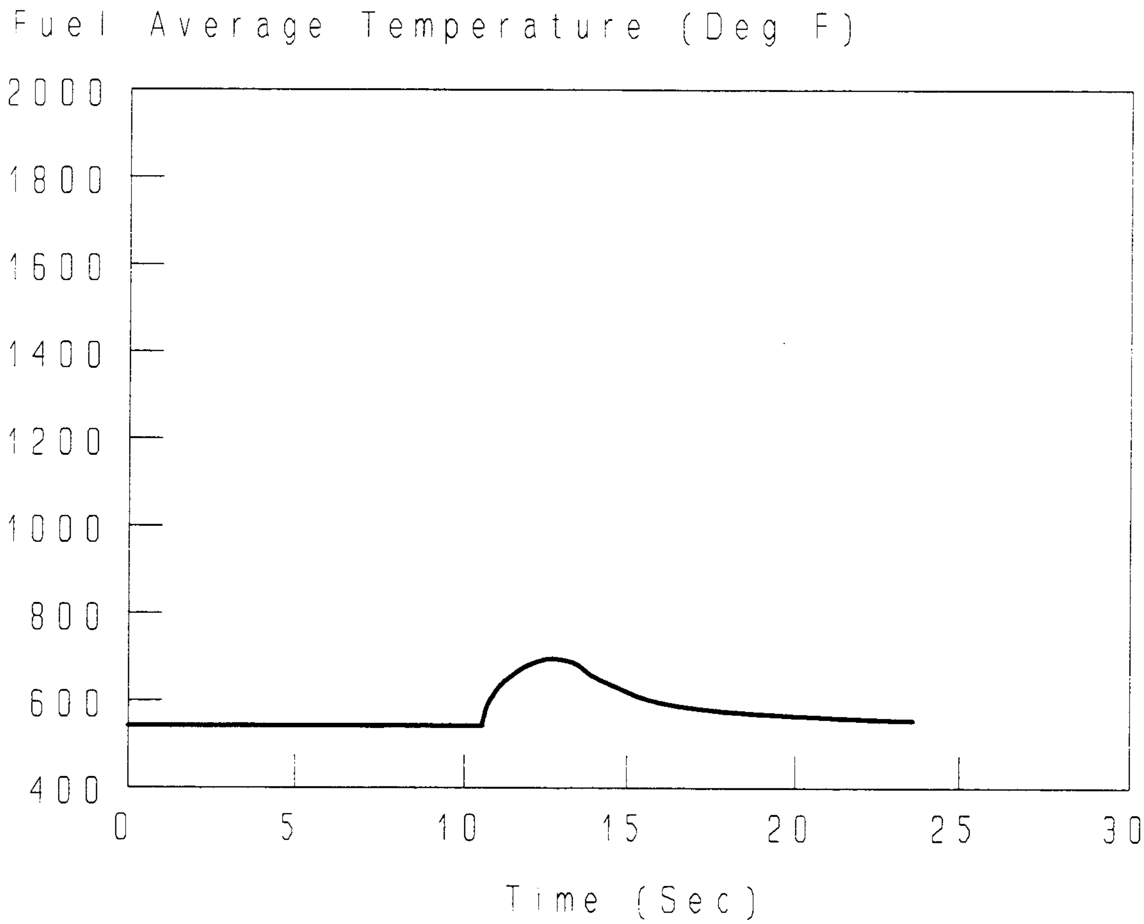

Figure 15.4.1-3 (Sheet 1 of 2)

RCCA Withdrawal from Subcritical Fuel Average Temperature

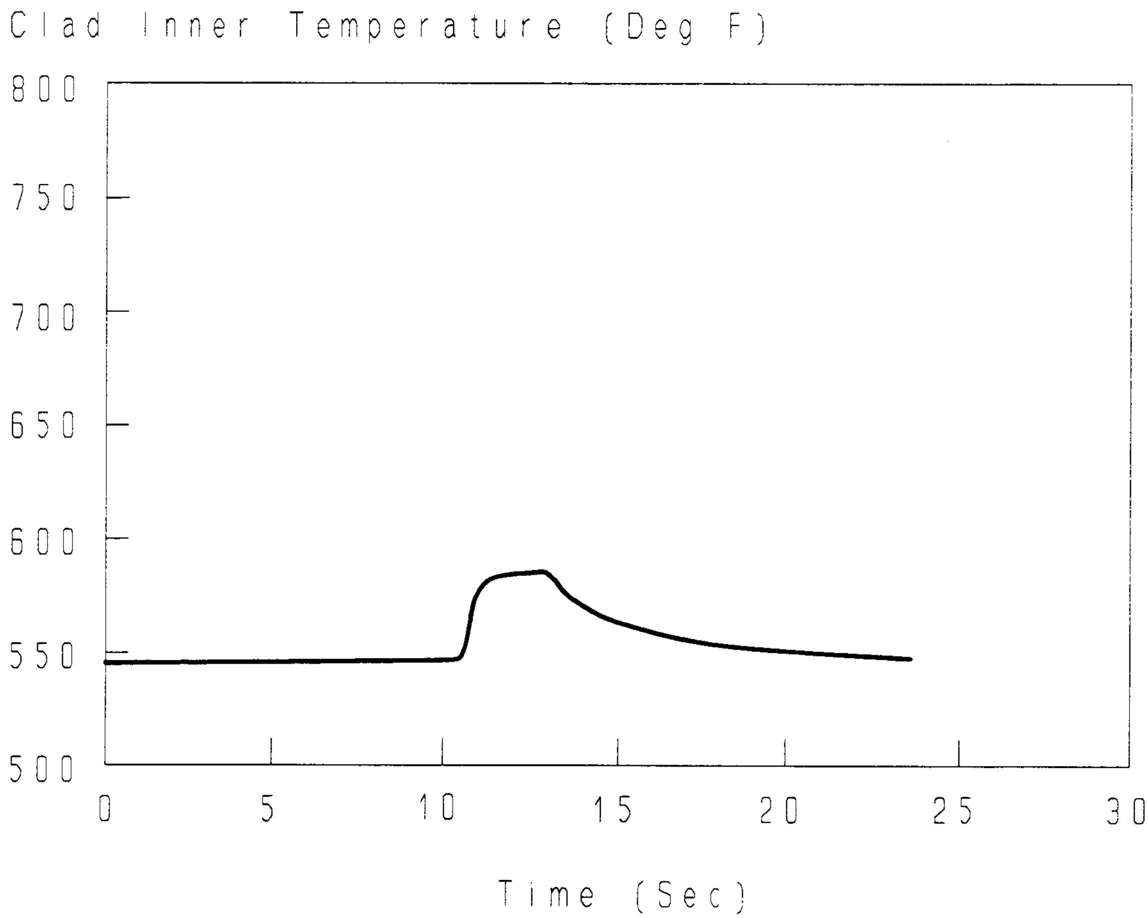

Figure 15.4.1-3 (Sheet 2 of 2)

RCCA Withdrawal from Subcritical Cladding Inner Temperature

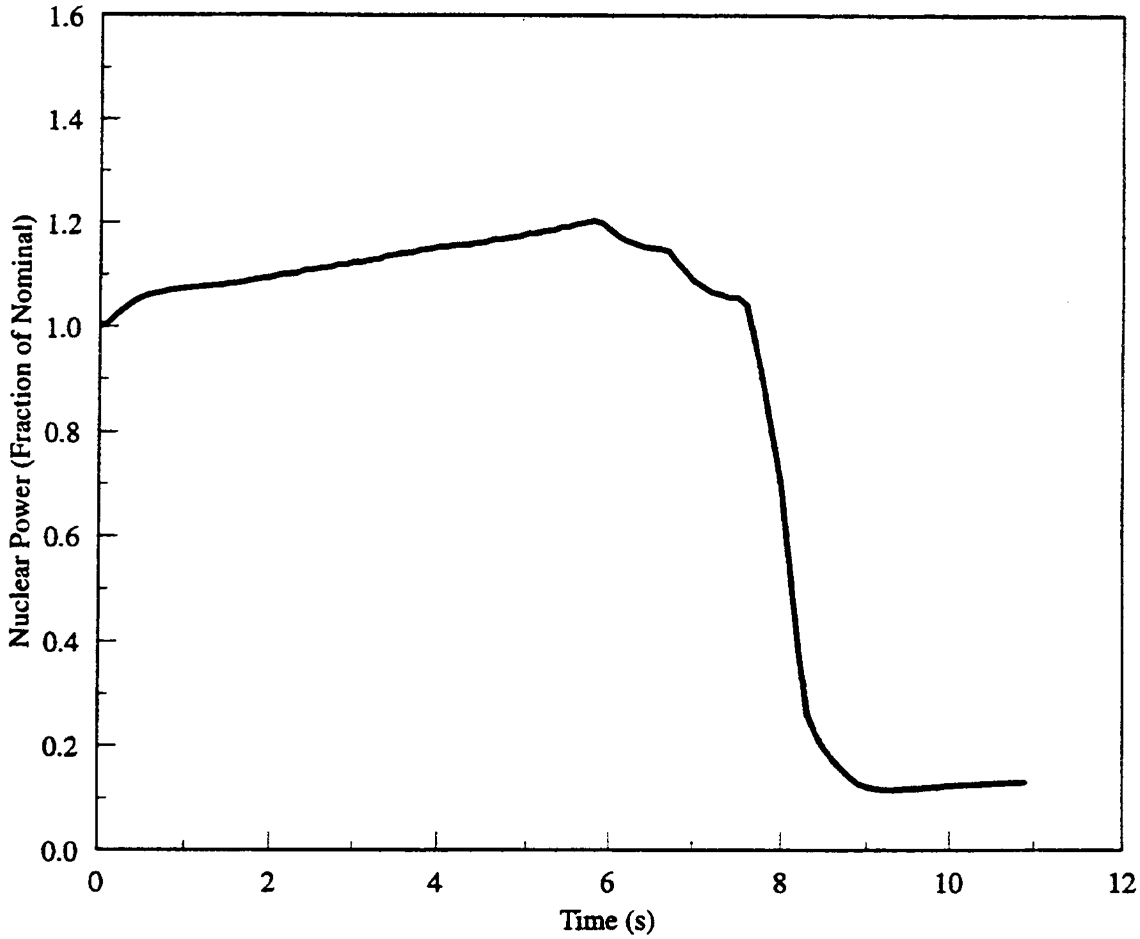

Figure 15.4.2-1

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Insertion Rate)

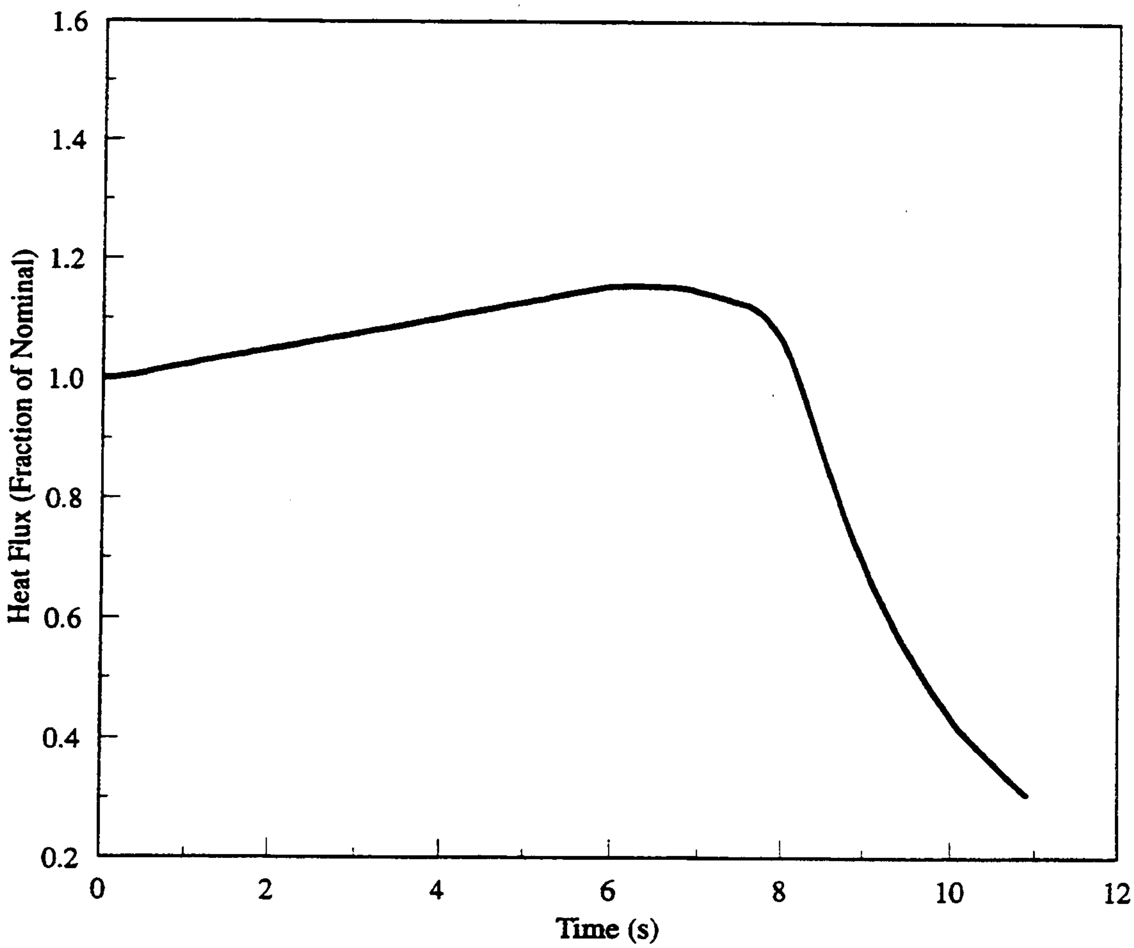

Figure 15.4.2-2

Thermal Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Insertion Rate)

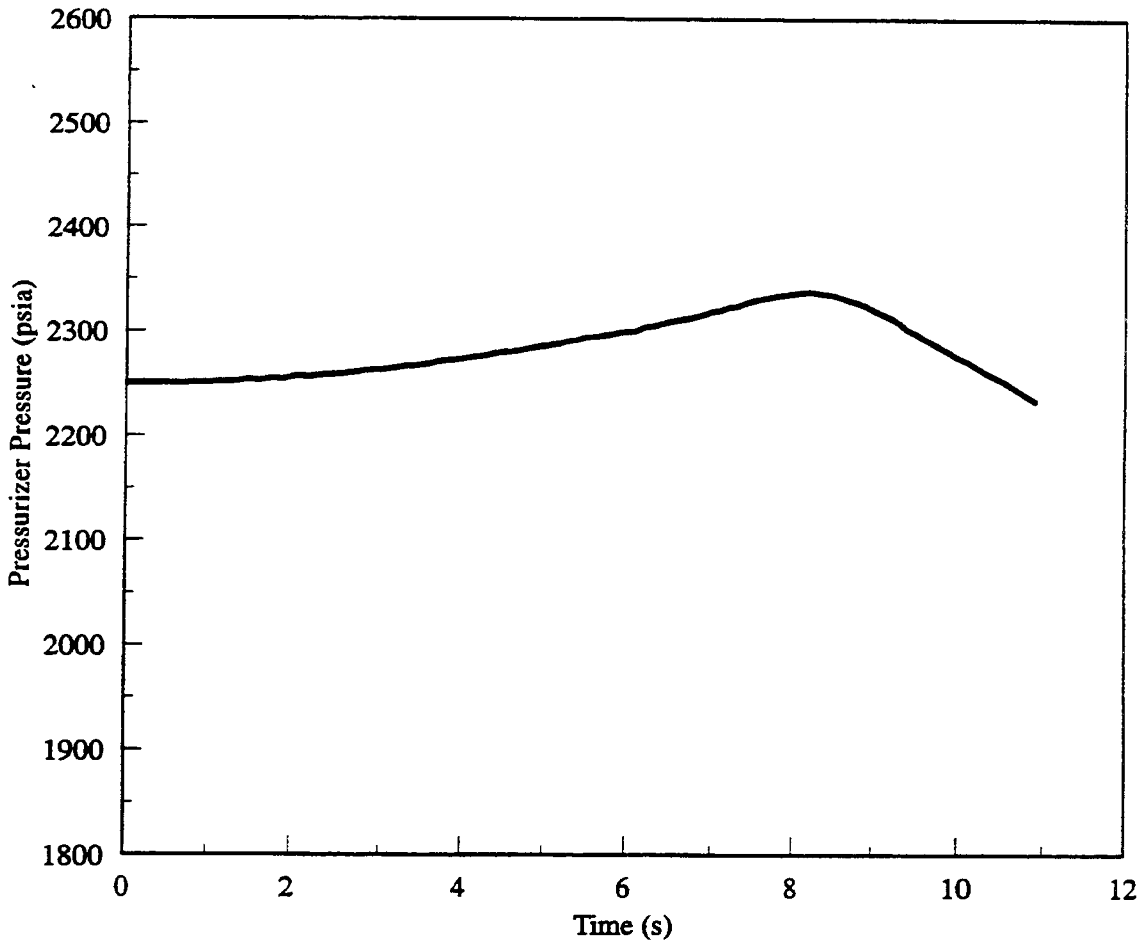

Figure 15.4.2-3

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Insertion Rate)

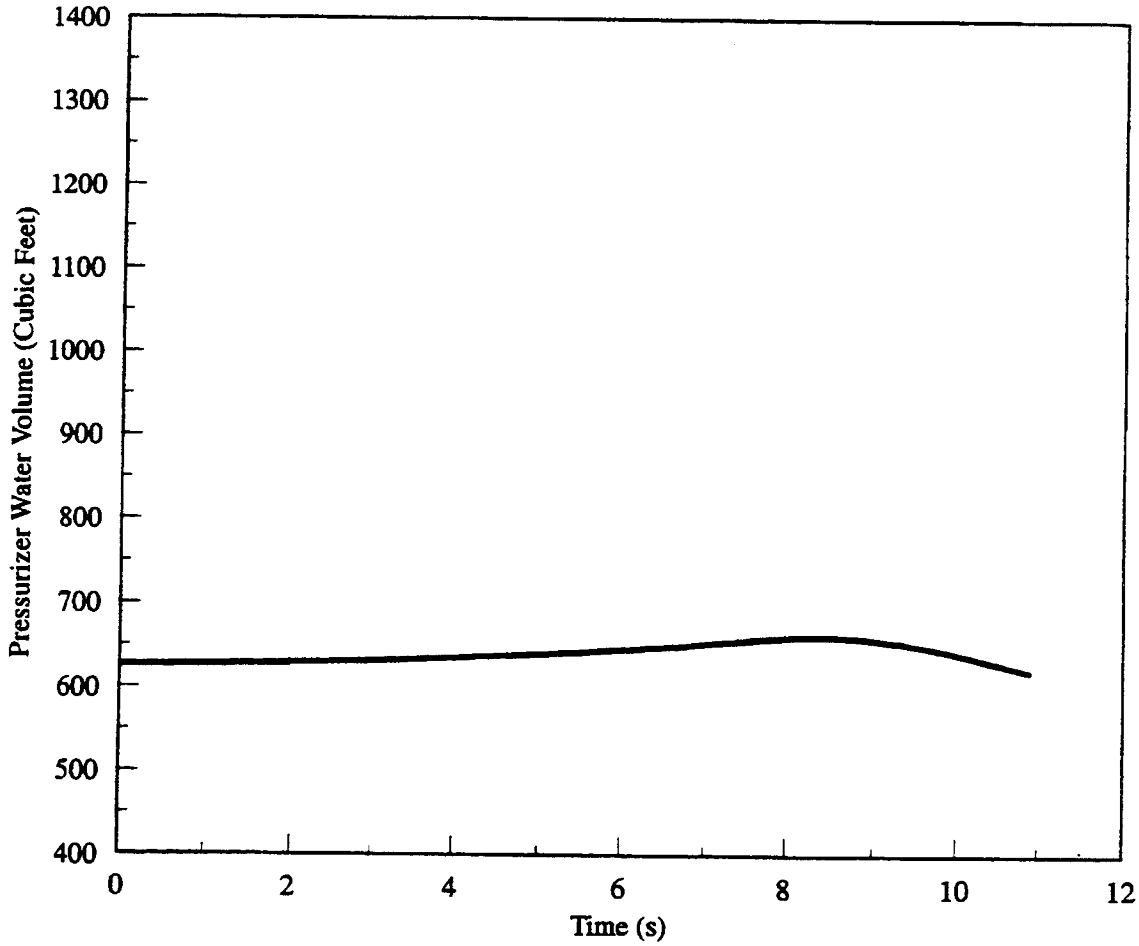

Figure 15.4.2-4

Pressurizer Water Volume for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Insertion Rate)

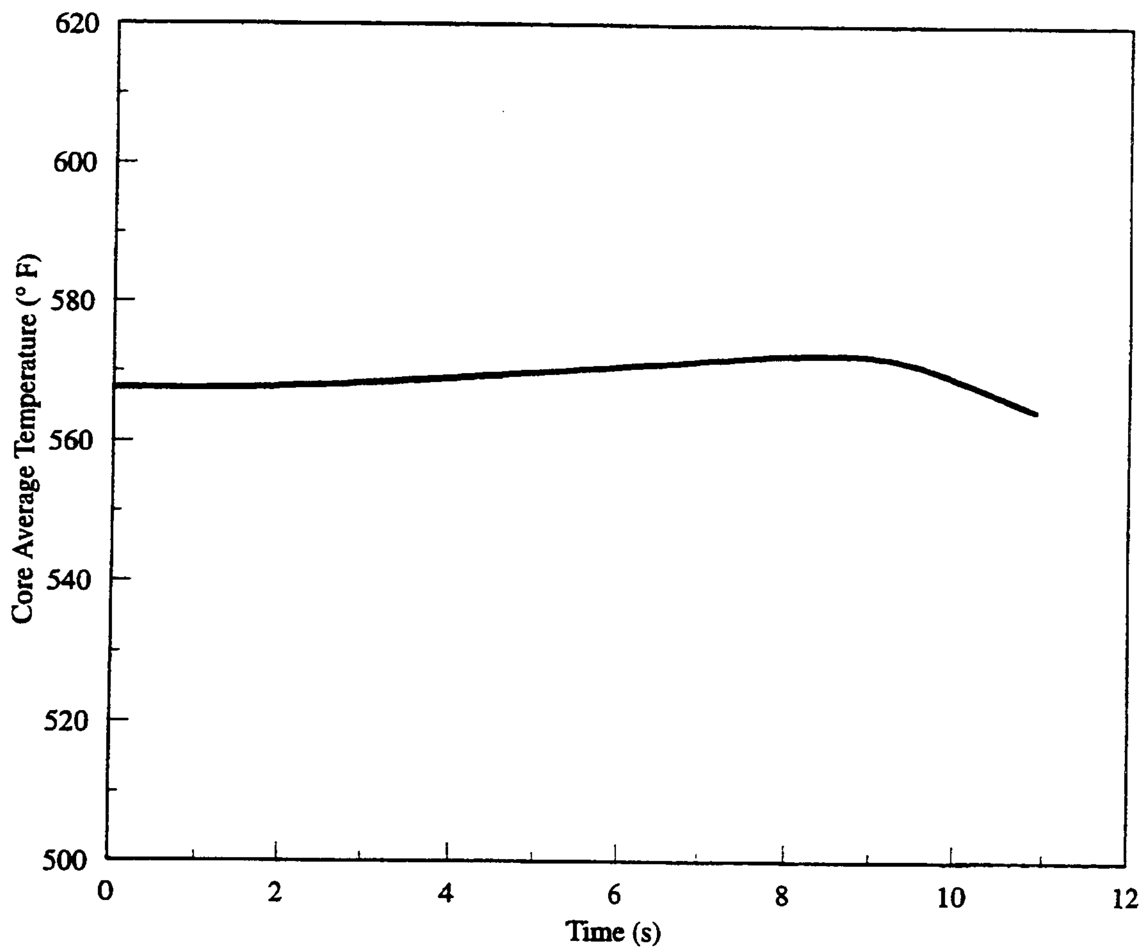

Figure 15.4.2-5

Core Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Insertion Rate)

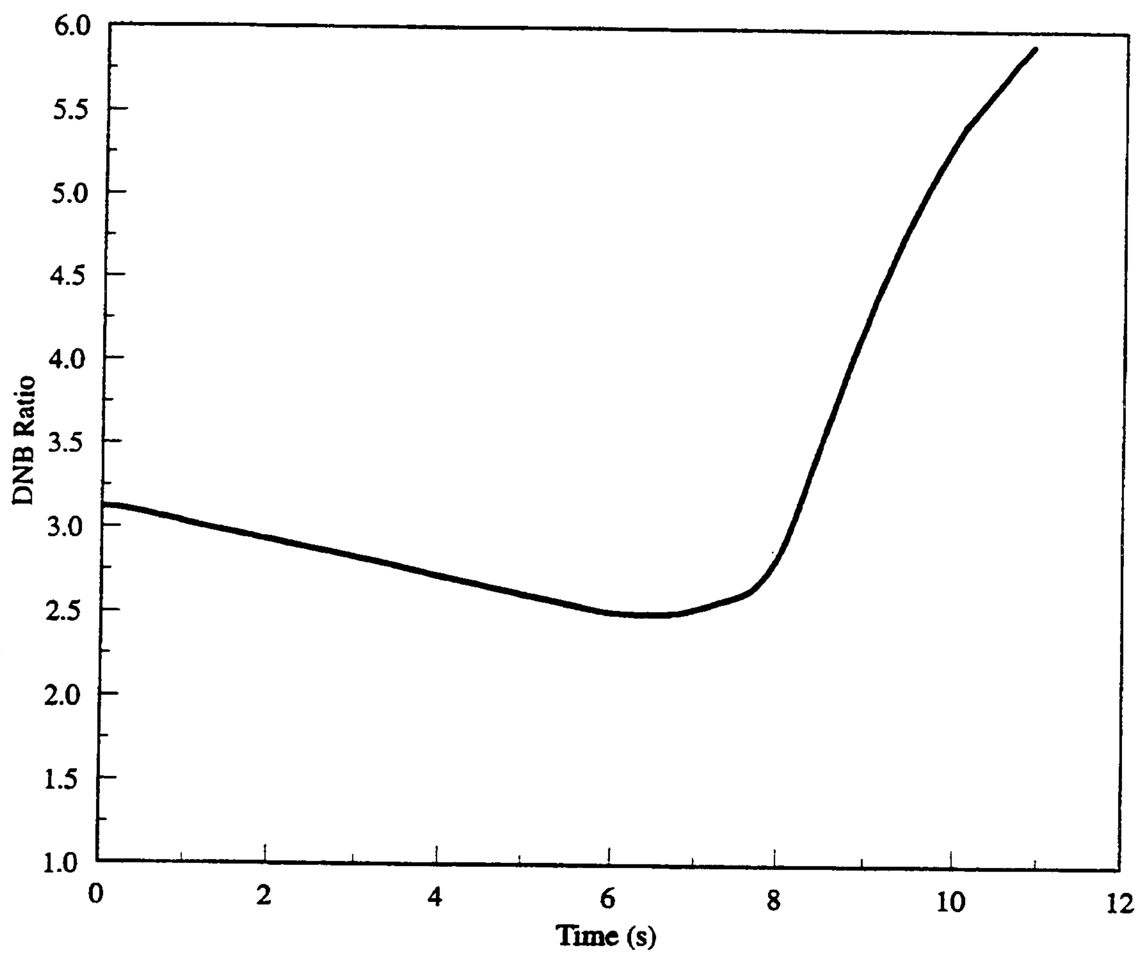

Figure 15.4.2-6

DNBR Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (75 pcm/s Insertion Rate)

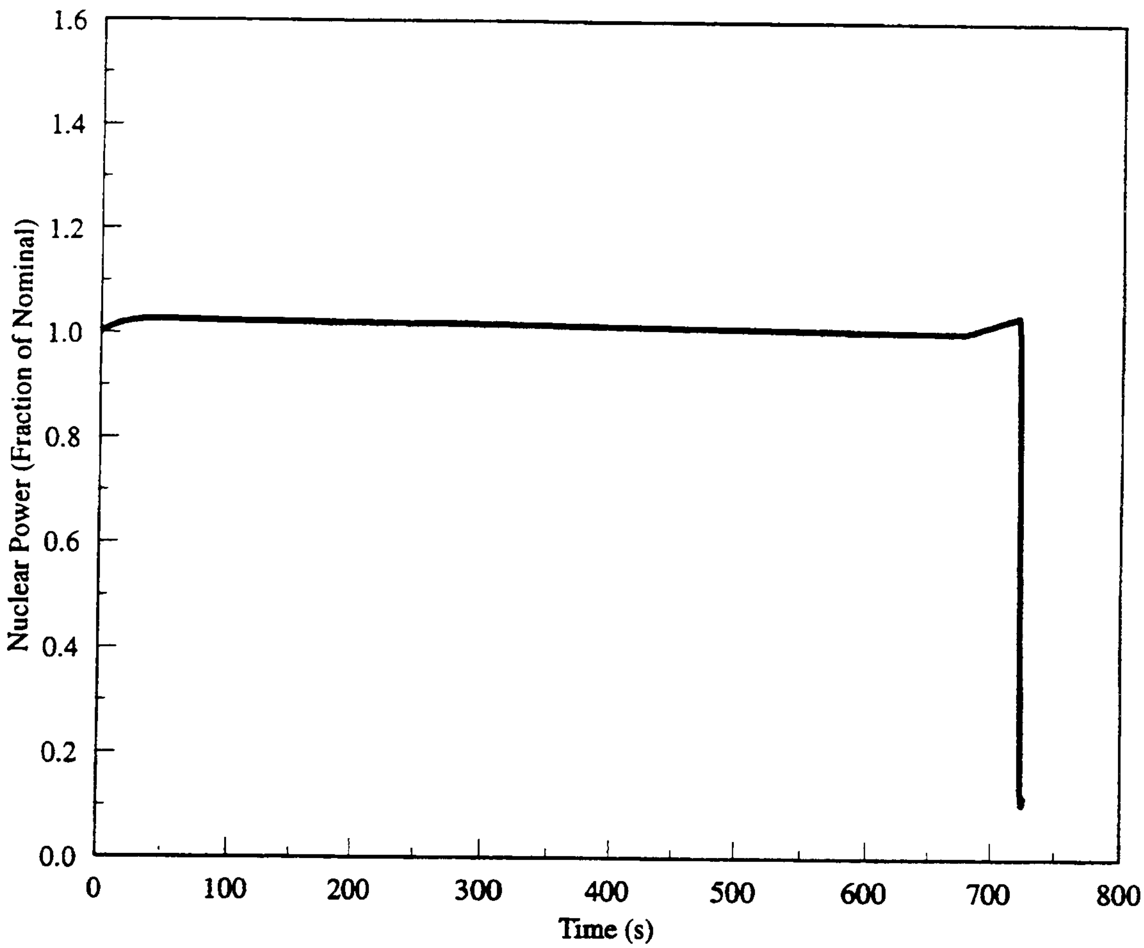

Figure 15.4.2-7

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Insertion Rate)

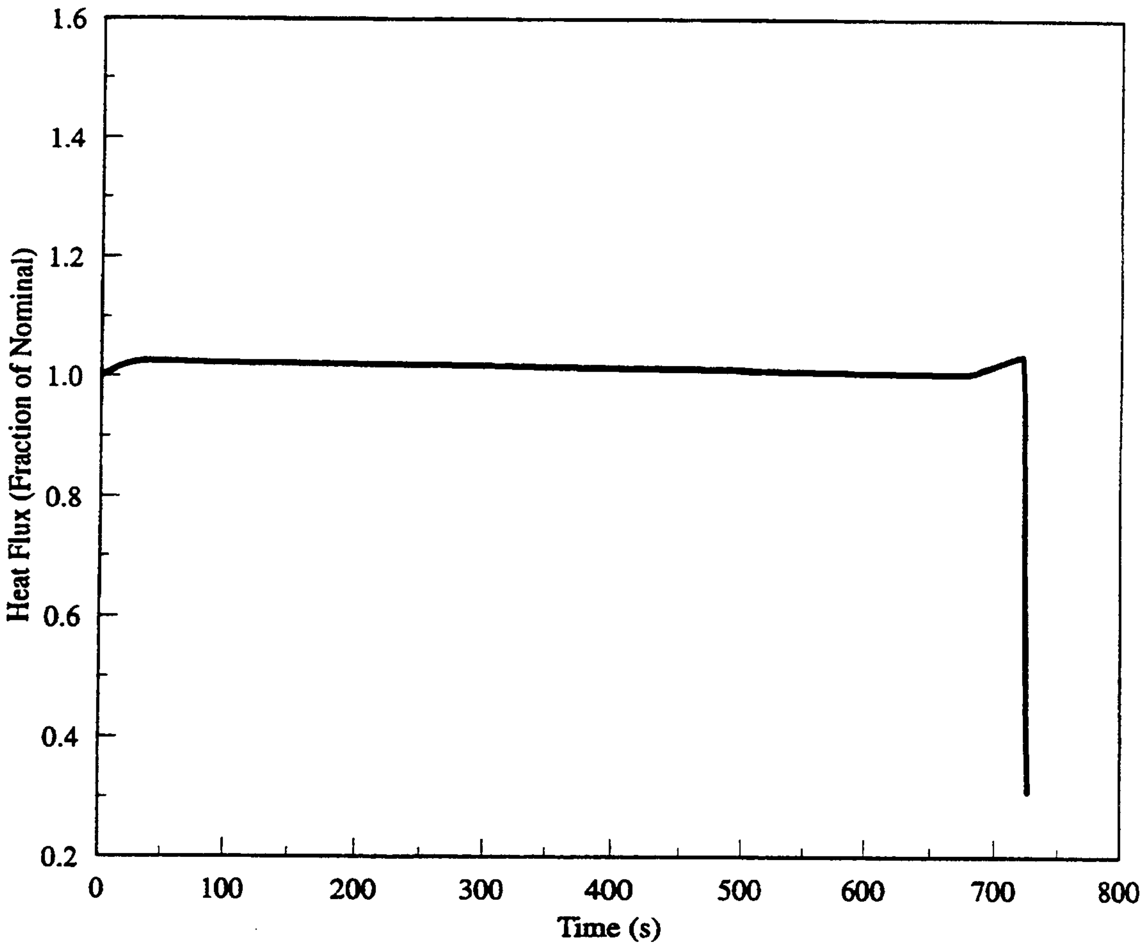

Figure 15.4.2-8

Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Insertion Rate)

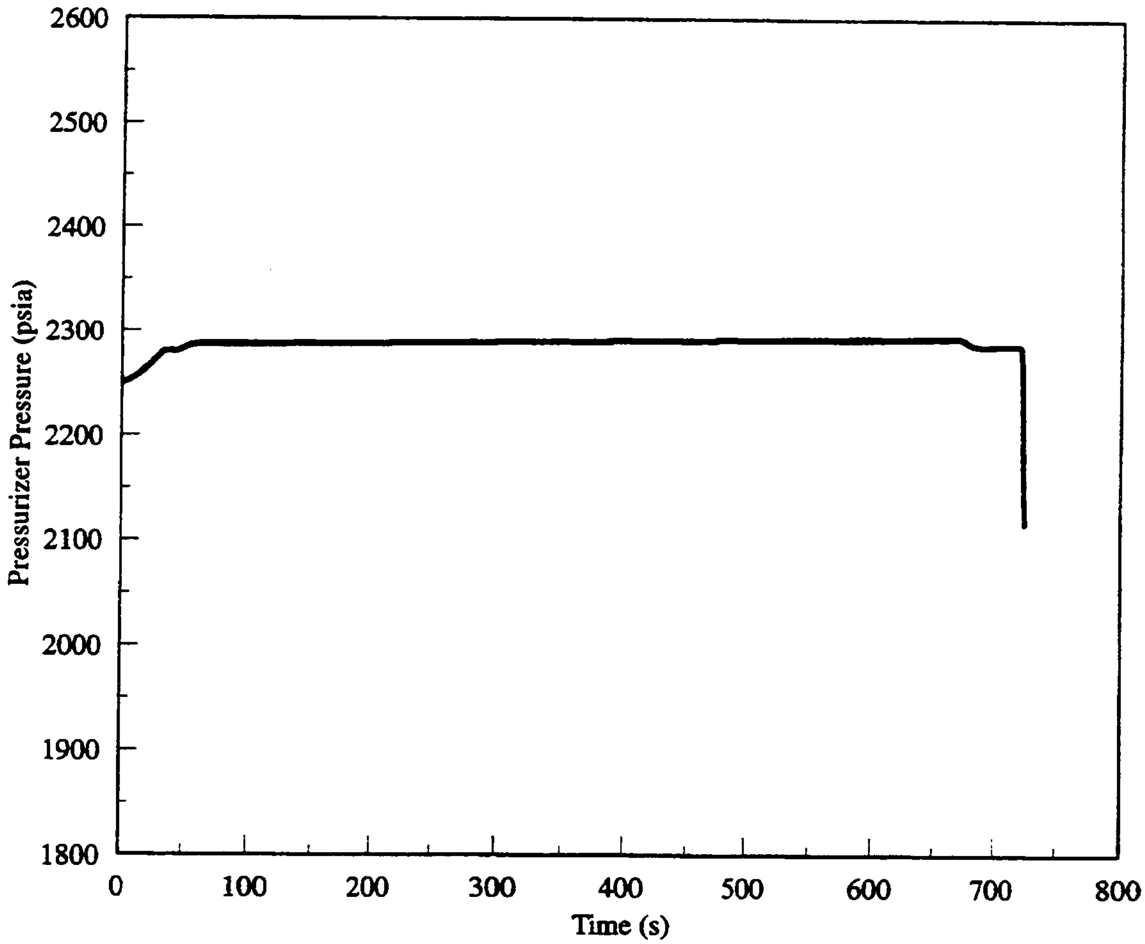

Figure 15.4.2-9

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Insertion Rate)

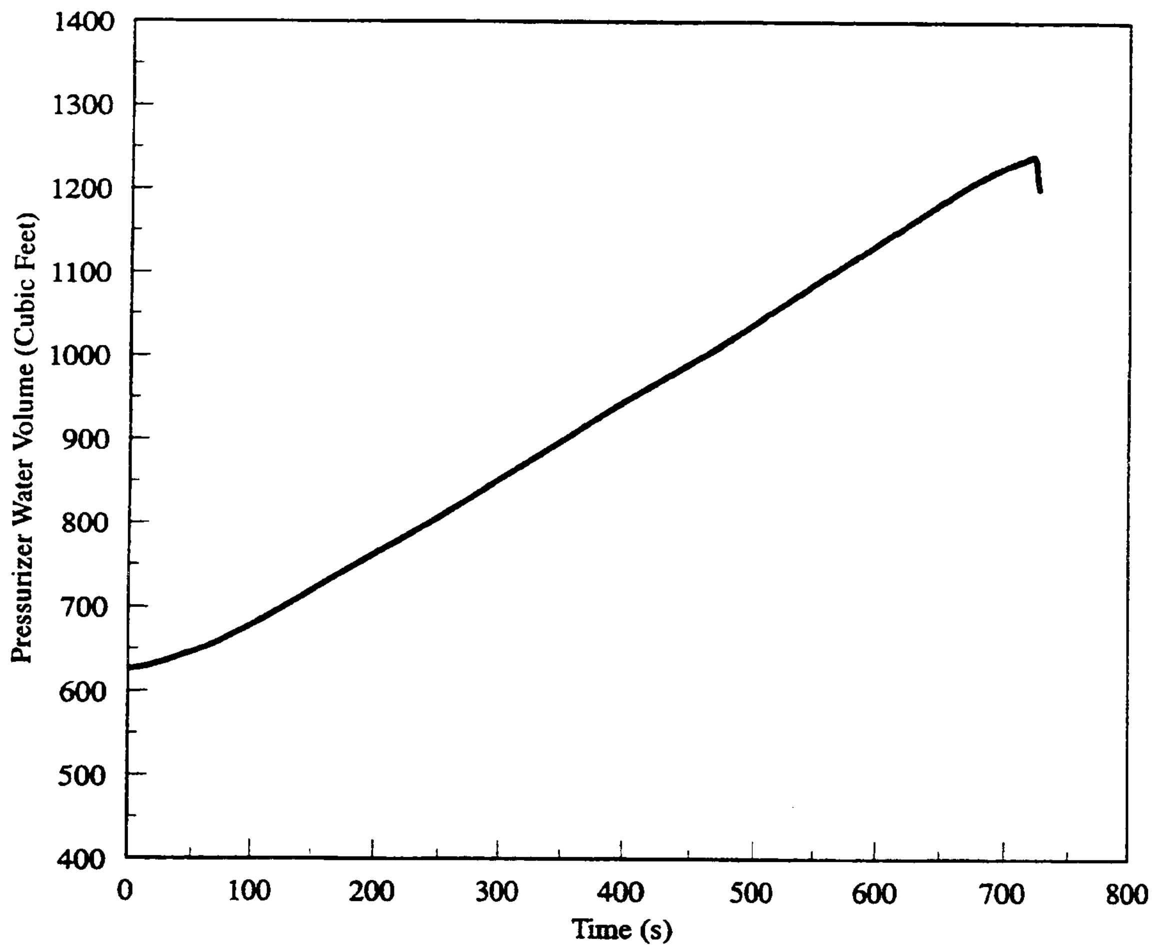

Figure 15.4.2-10

Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Insertion Rate)

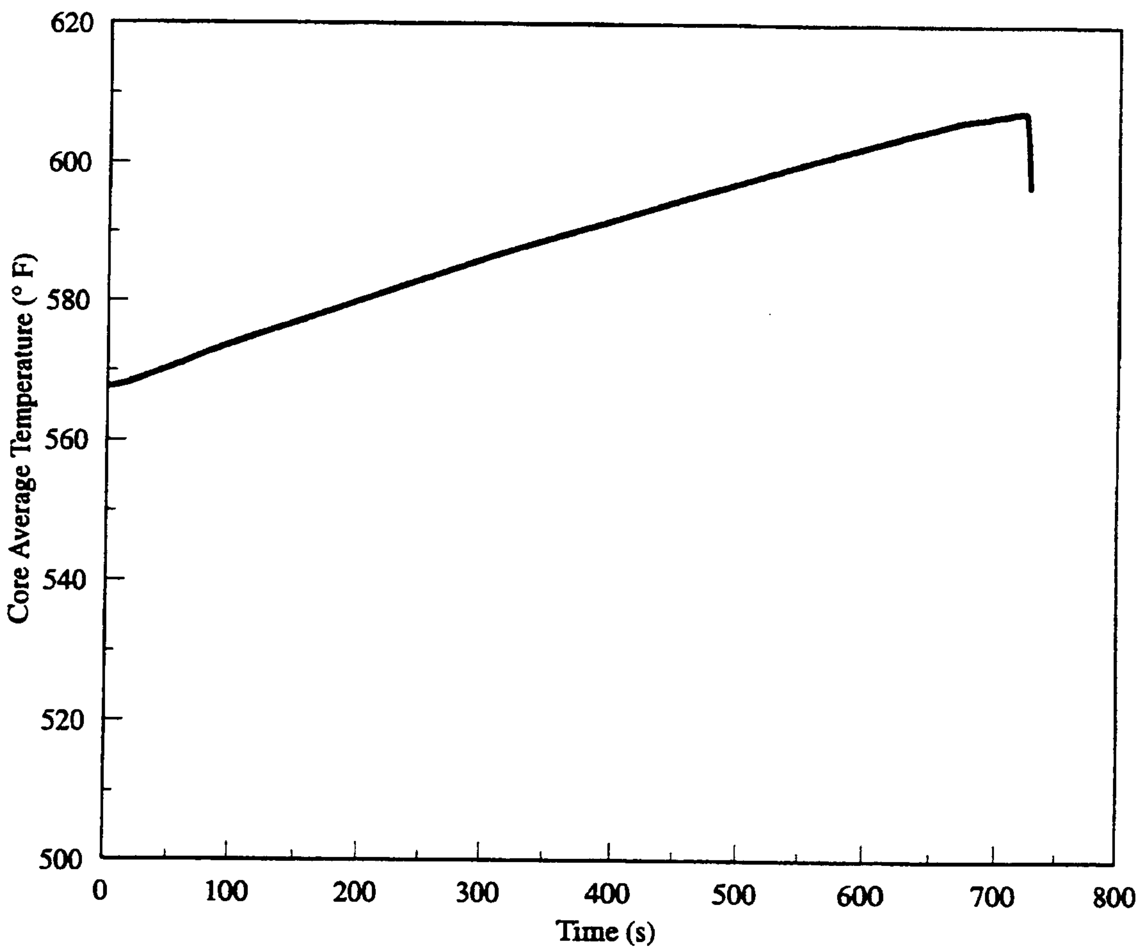

Figure 15.4.2-11

Core Average Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Insertion Rate)

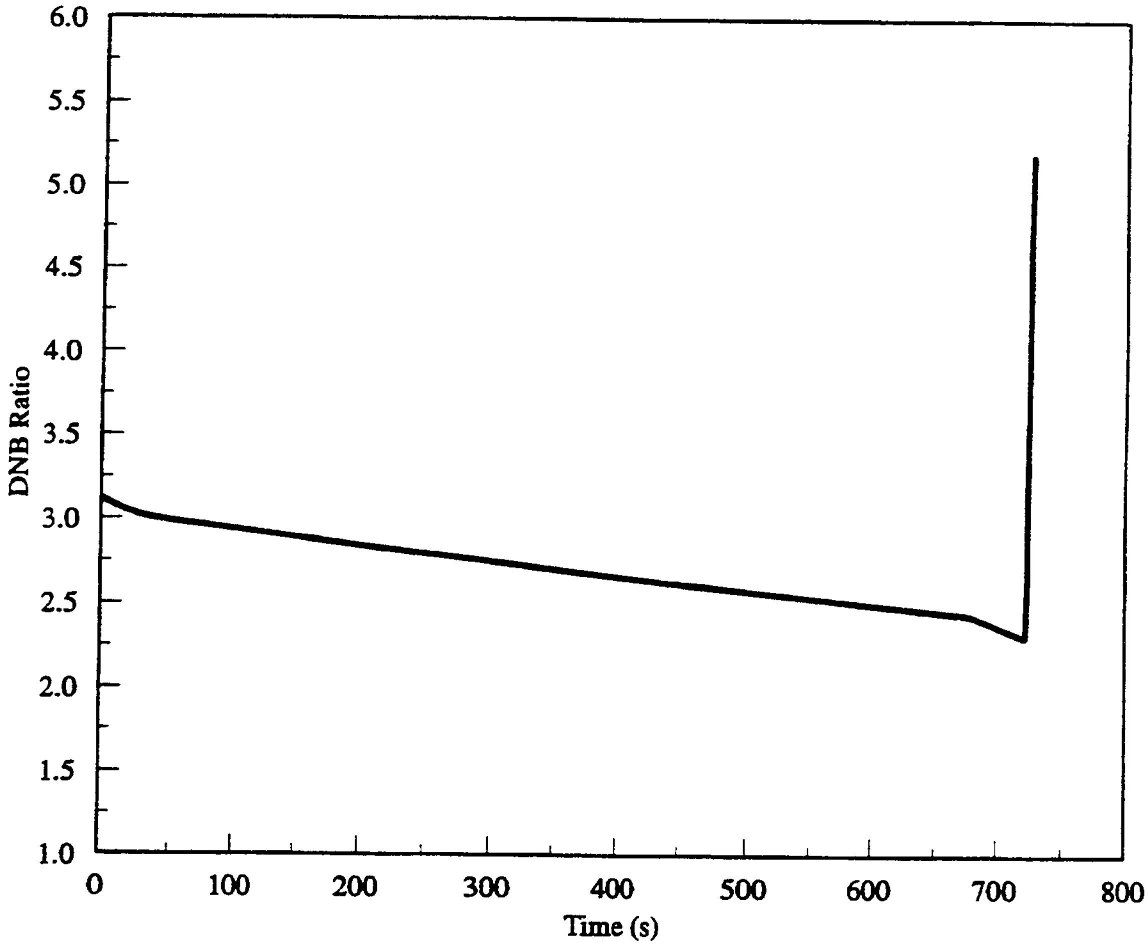

Figure 15.4.2-12

DNBR Ratio Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power With Maximum Reactivity Feedback (3 pcm/s Insertion Rate)

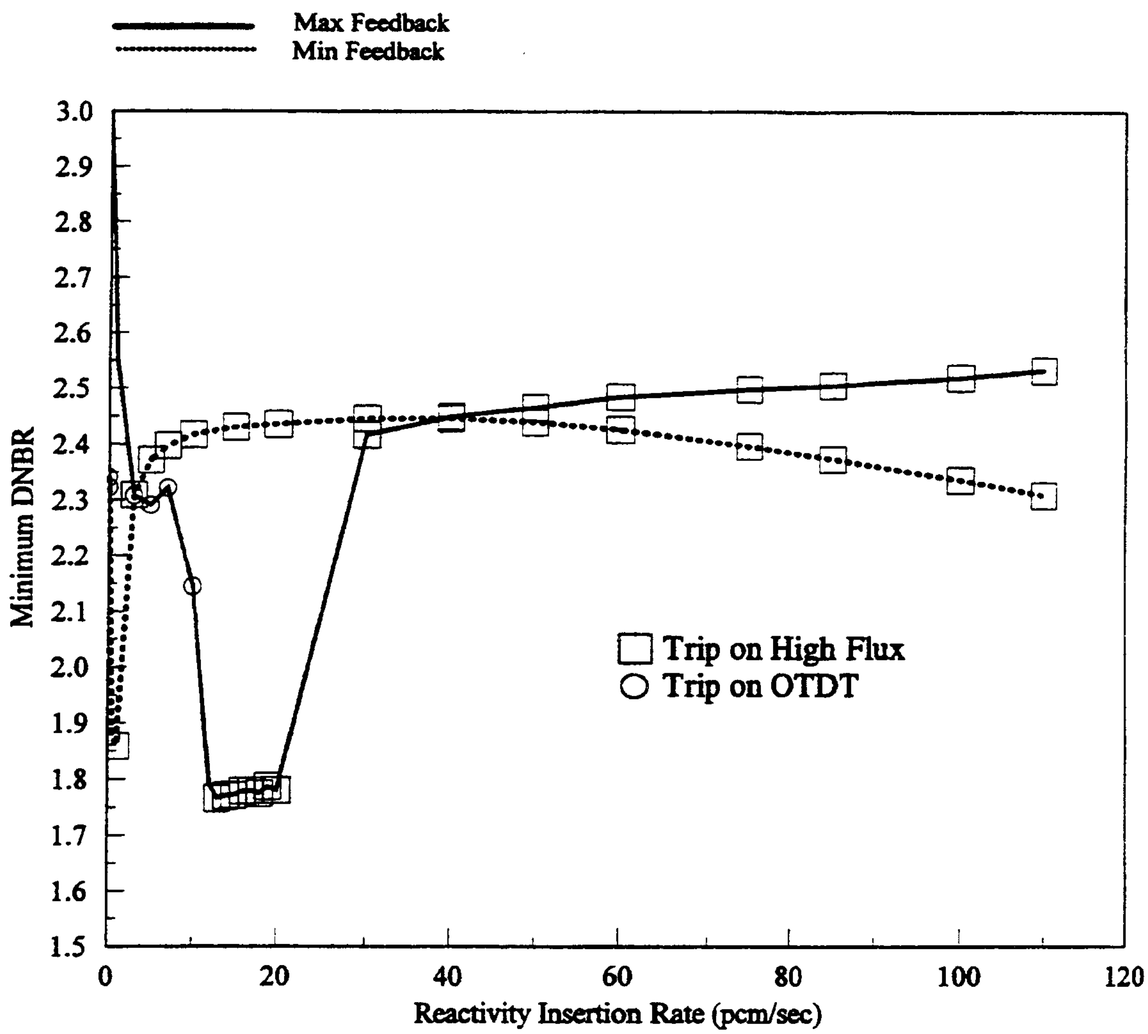

Figure 15.4.2-13

Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 100-Percent Power

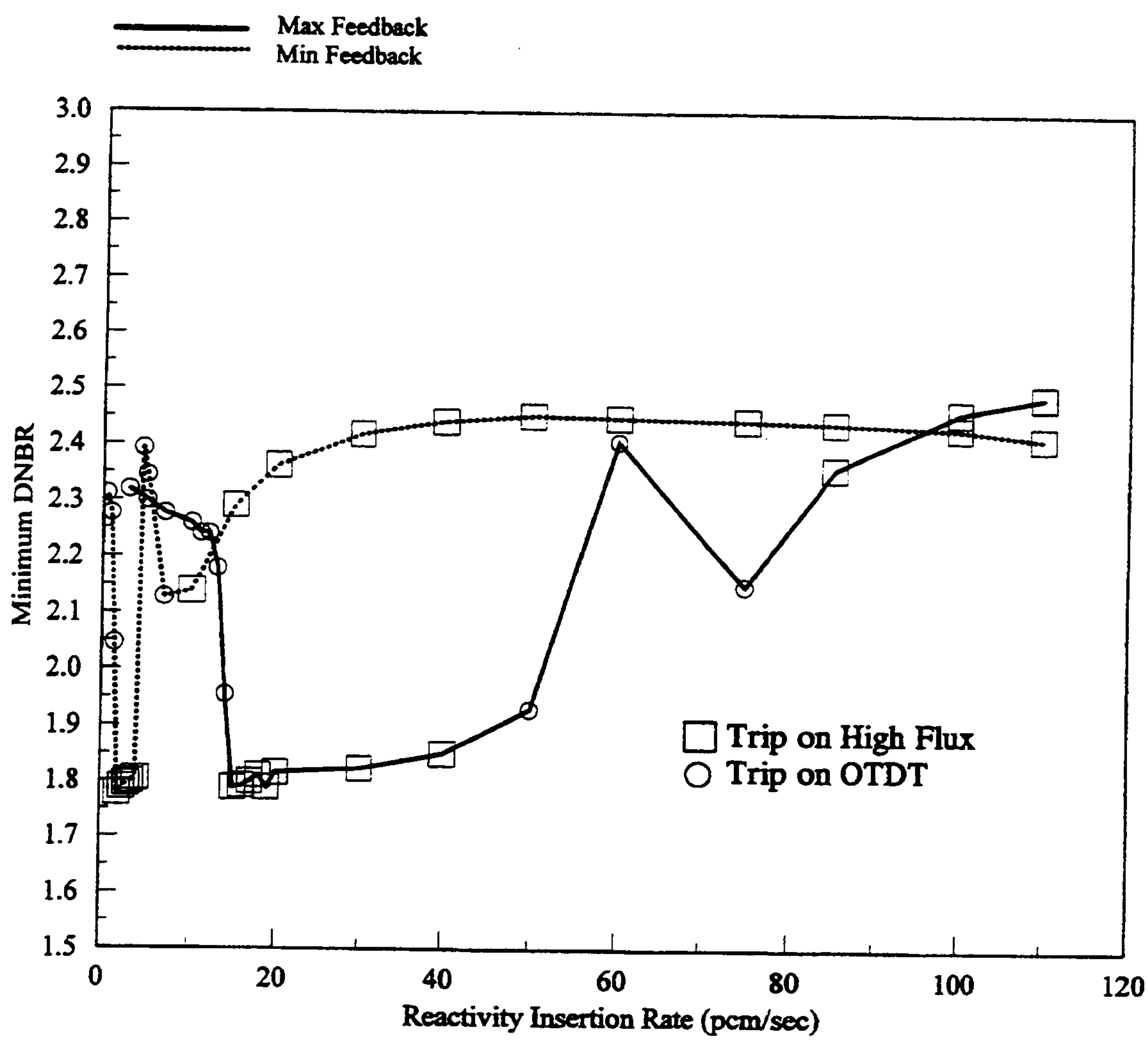

Figure 15.4.2-14

Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 60-Percent Power

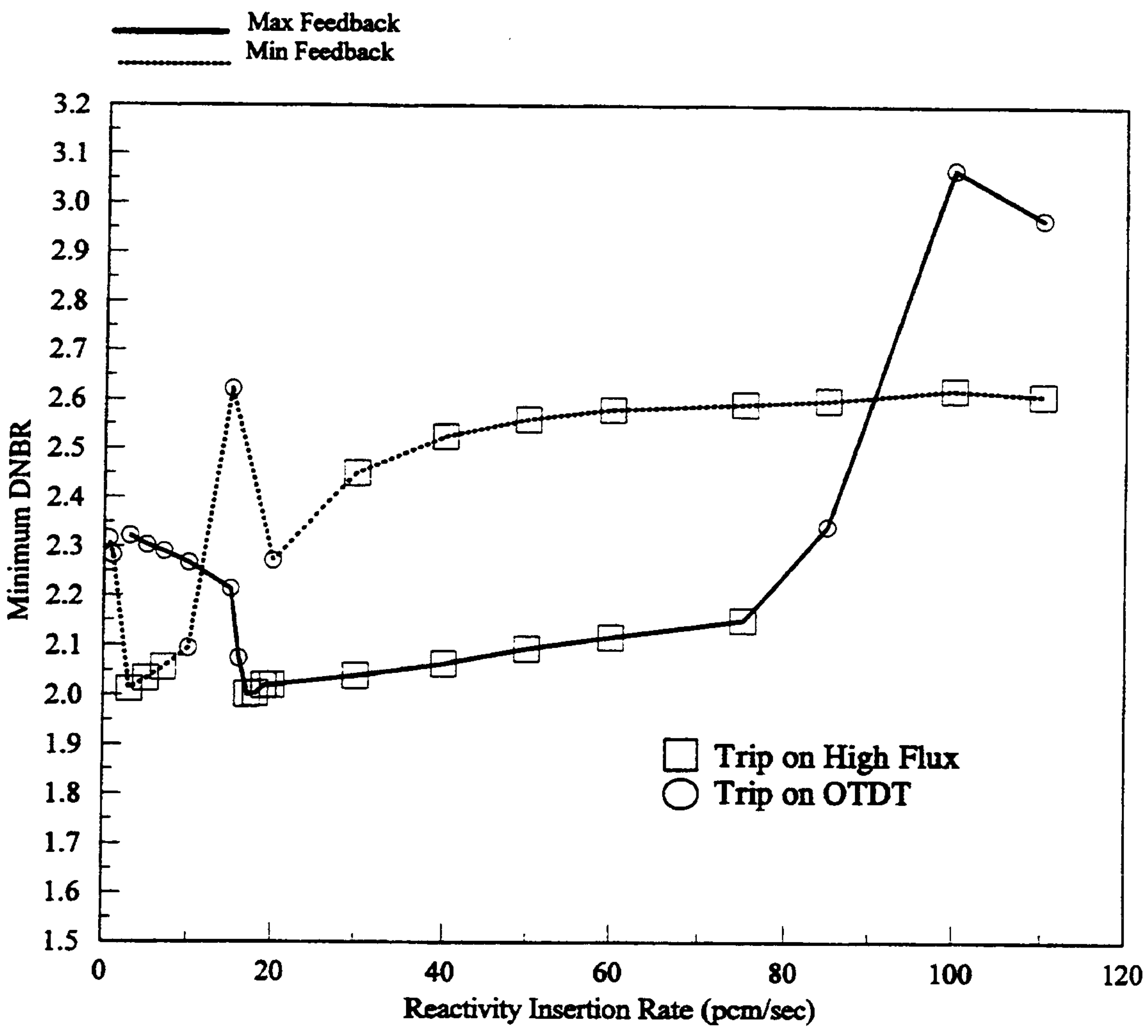

Figure 15.4.2-15

Minimum DNBR Versus Reactivity Insertion Rate for Rod Withdrawal at 10-Percent Power

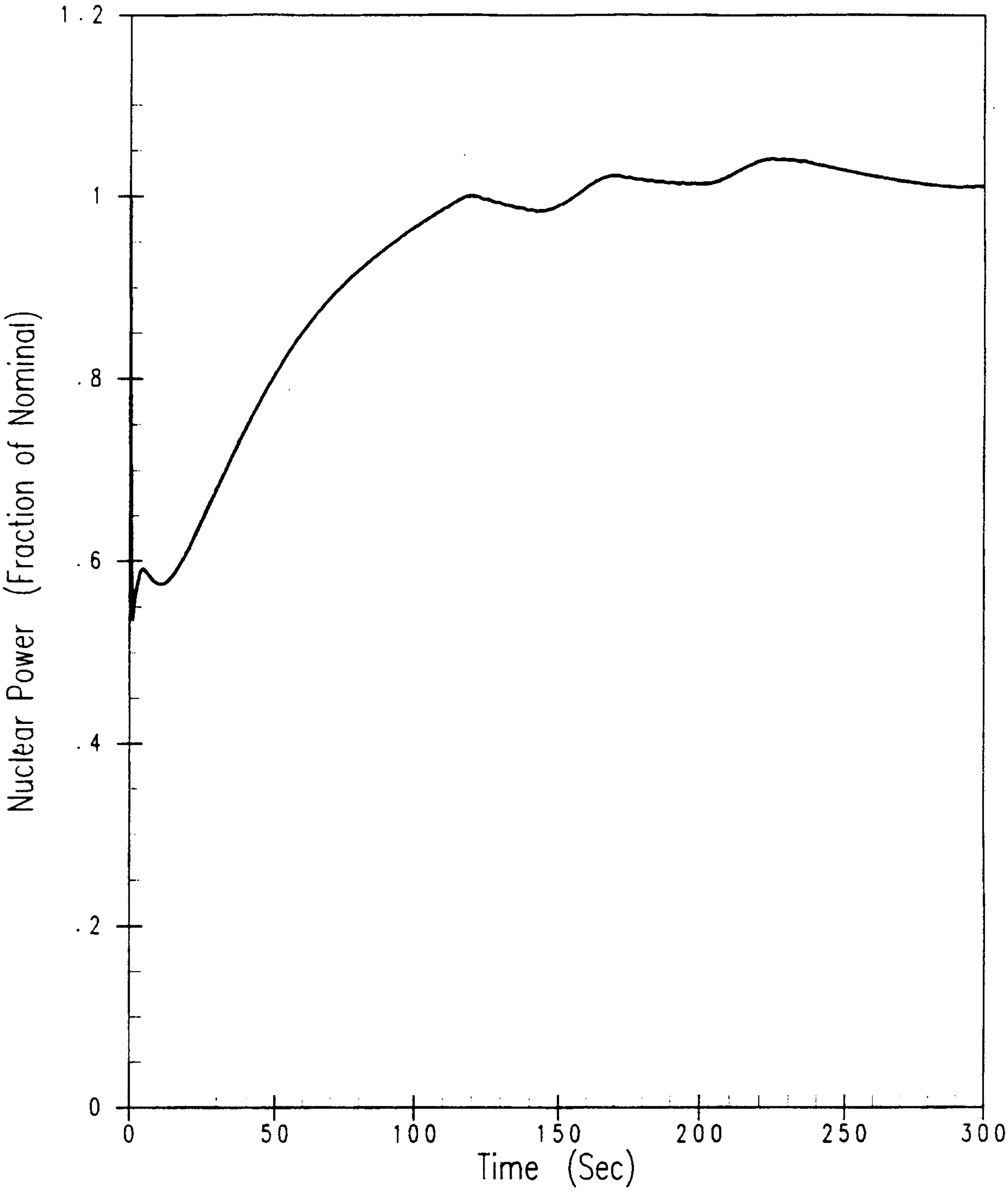

Figure 15.4.3-1

Nuclear Power Transients for Dropped RCCA

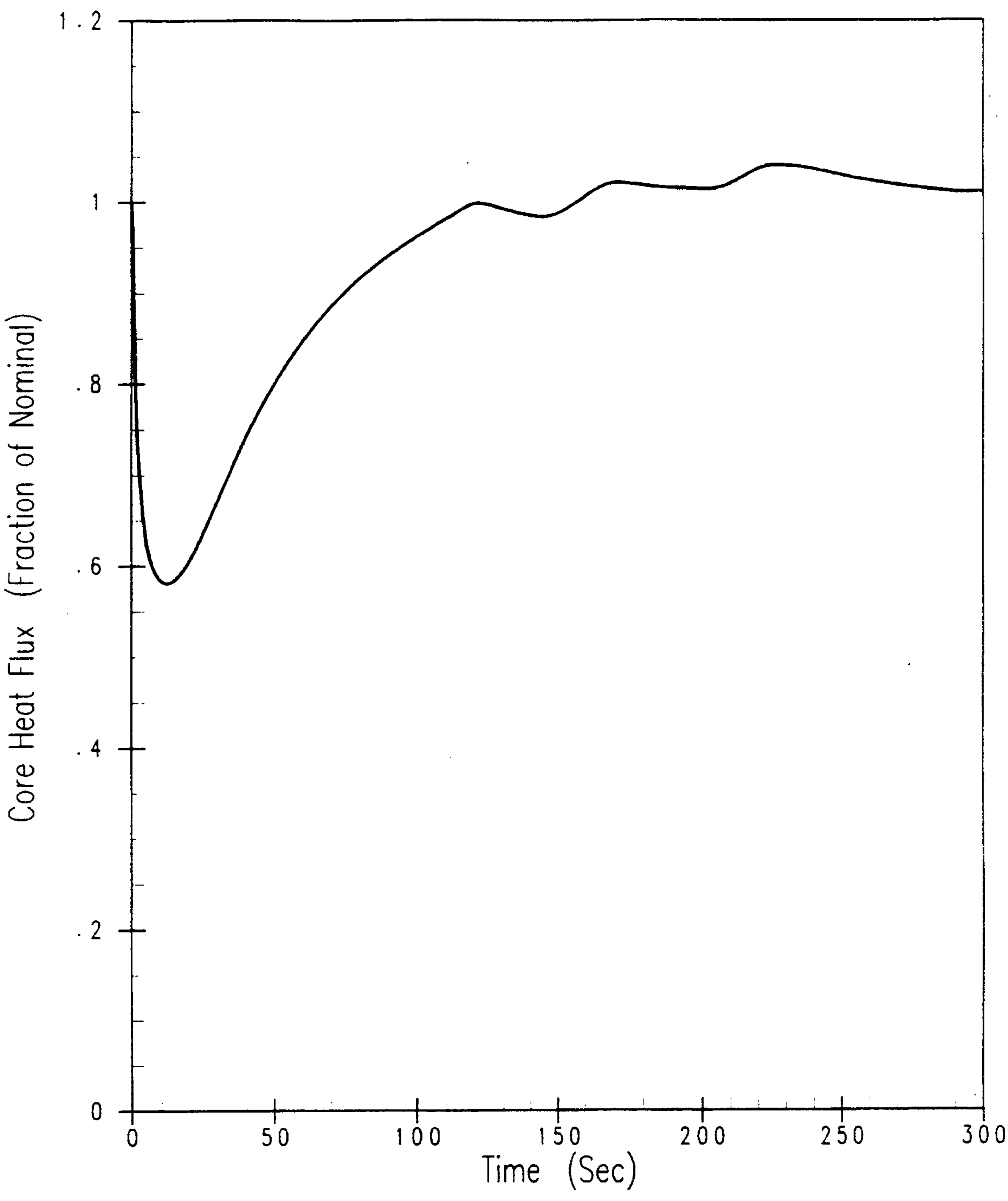

Figure 15.4.3-2

Core Heat Flux Transients for Dropped RCCA

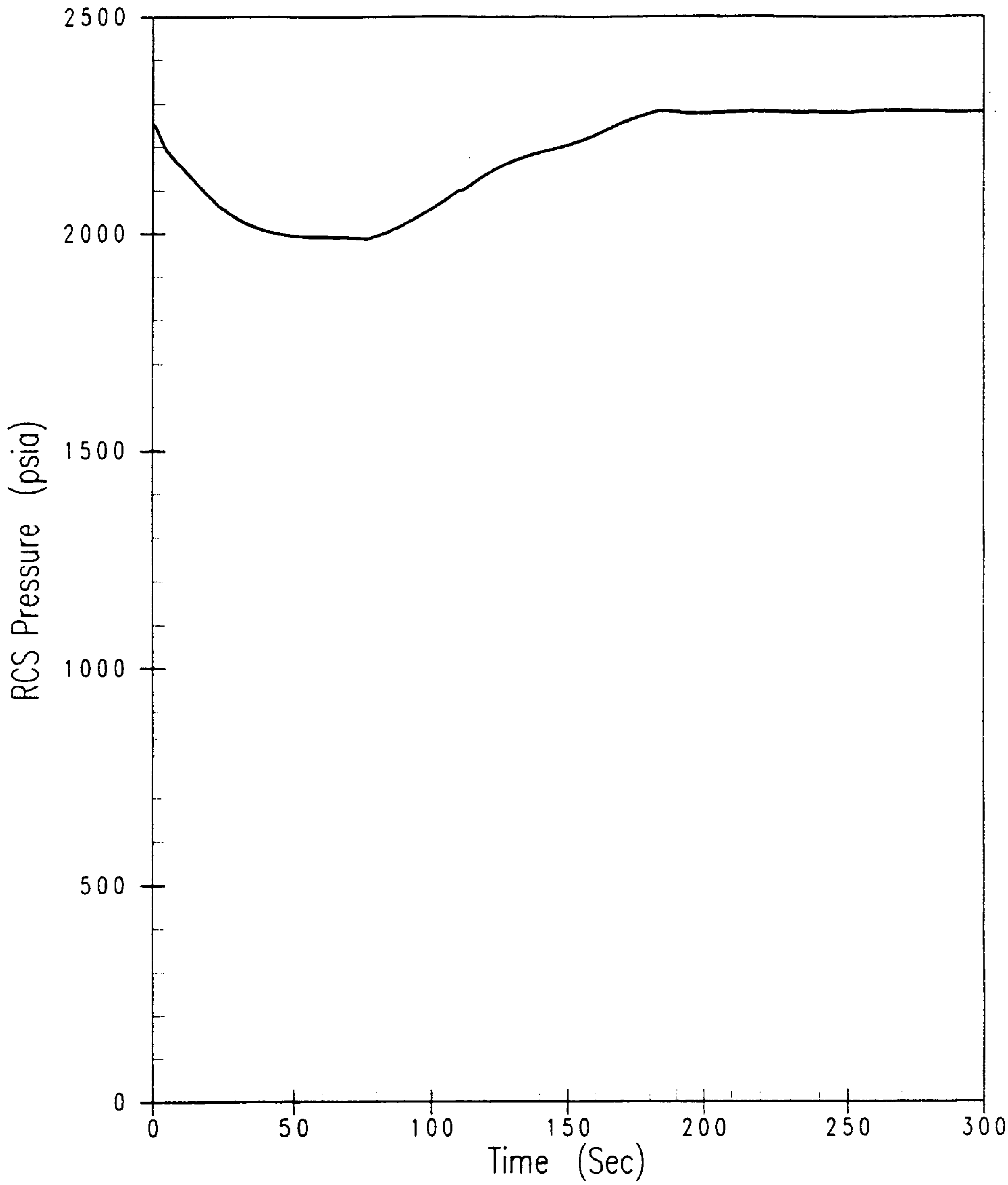

Figure 15.4.3-3

Reactor Coolant System Pressure Transient for Dropped RCCA

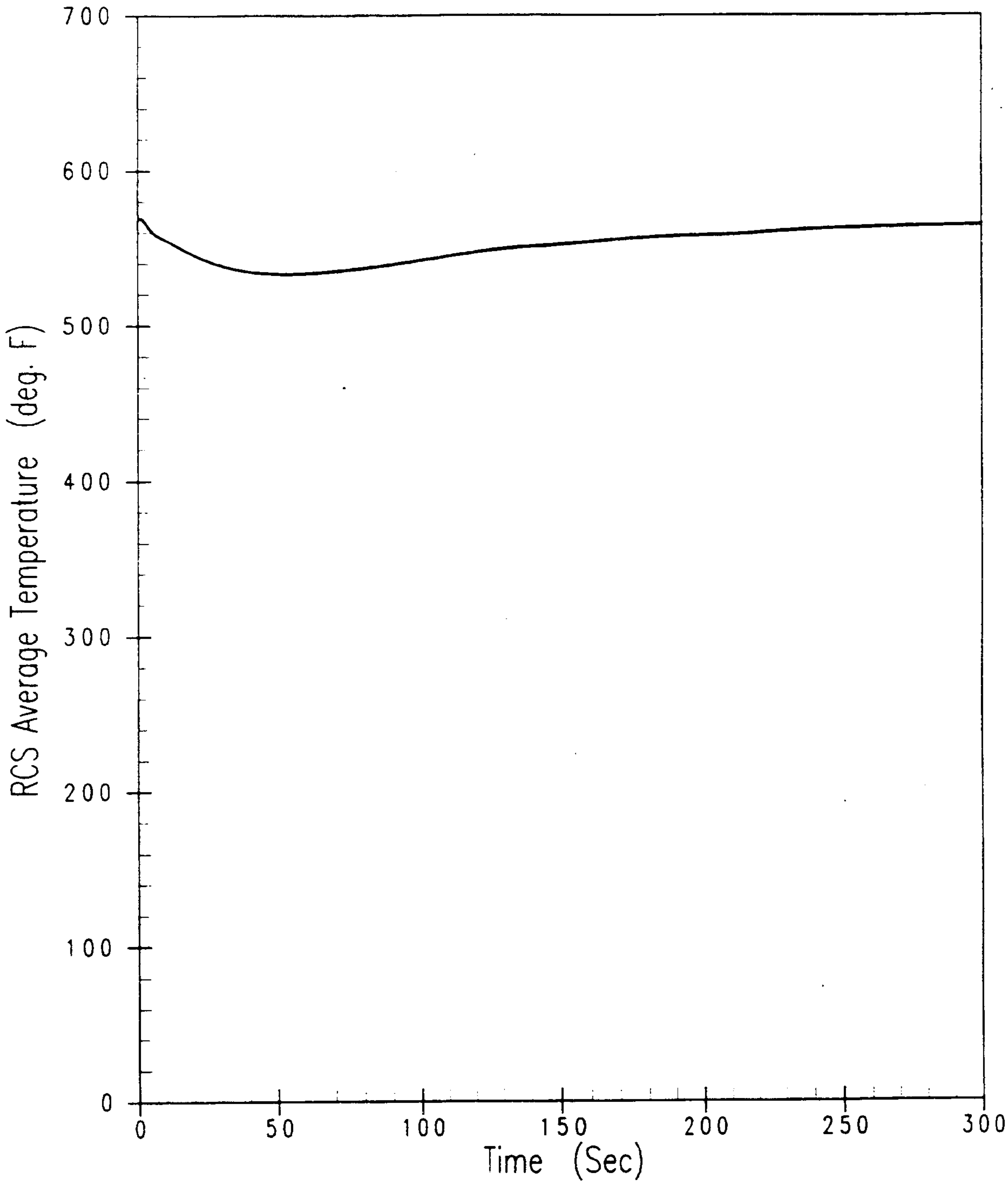

Figure 15.4.3-4

Core Average Temperature Transient for Dropped RCCA

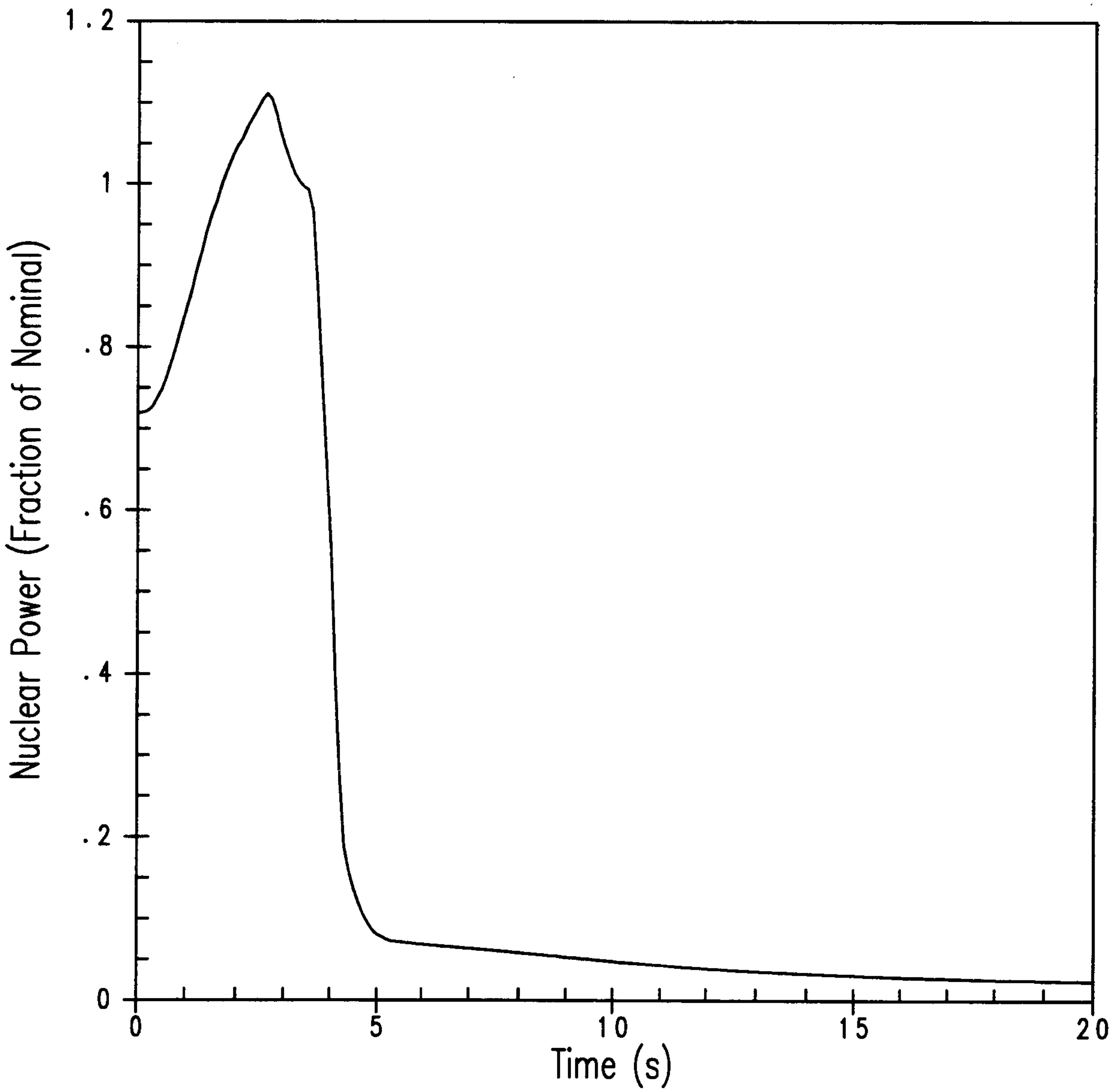

Figure 15.4.4-1

Nuclear Power Transient for an Improper Startup of an Inactive Reactor Coolant Pump

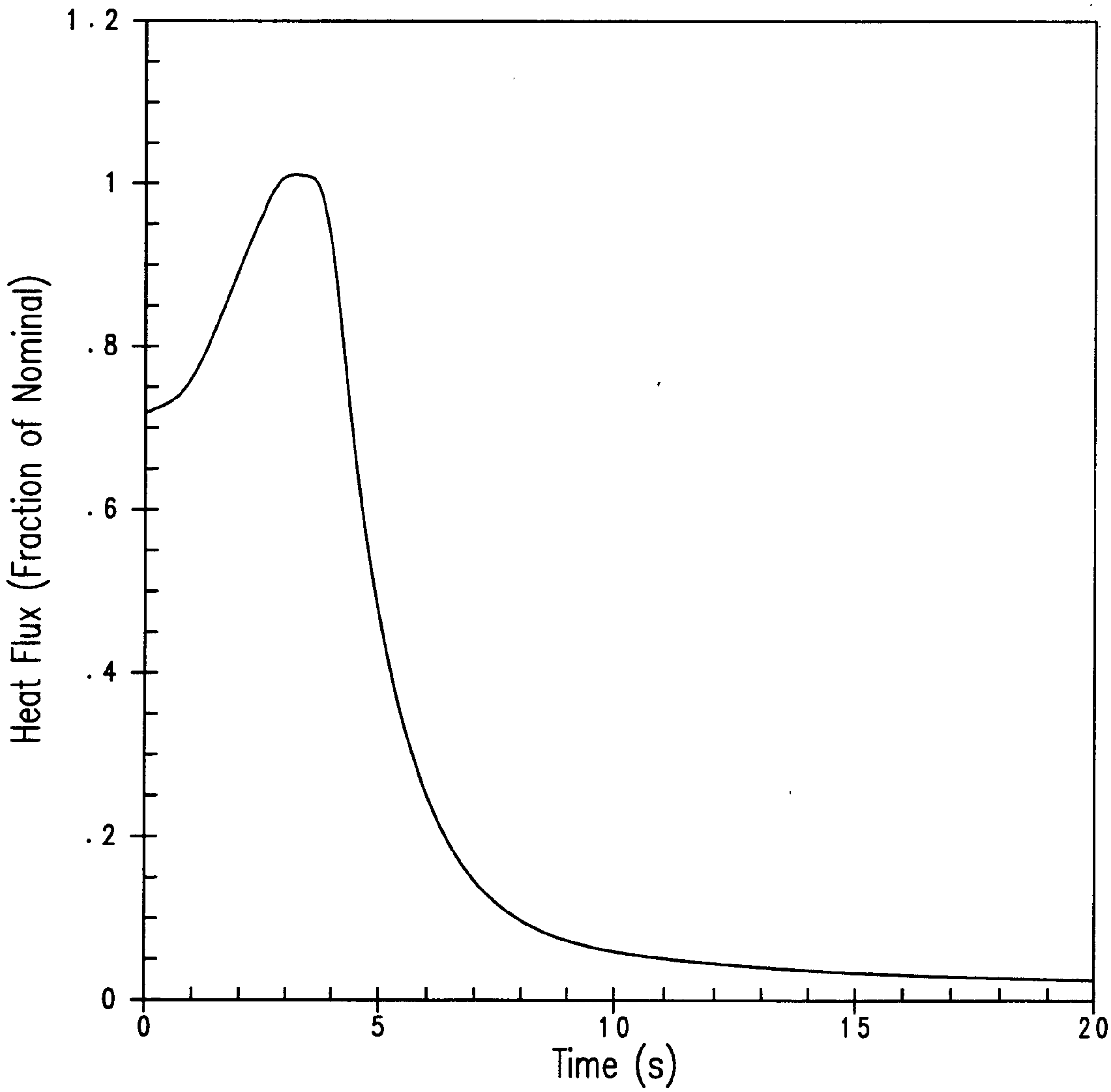

Figure 15.4.4-2

Heat Flux Transient for an Improper Startup of an Inactive Reactor Coolant Pump

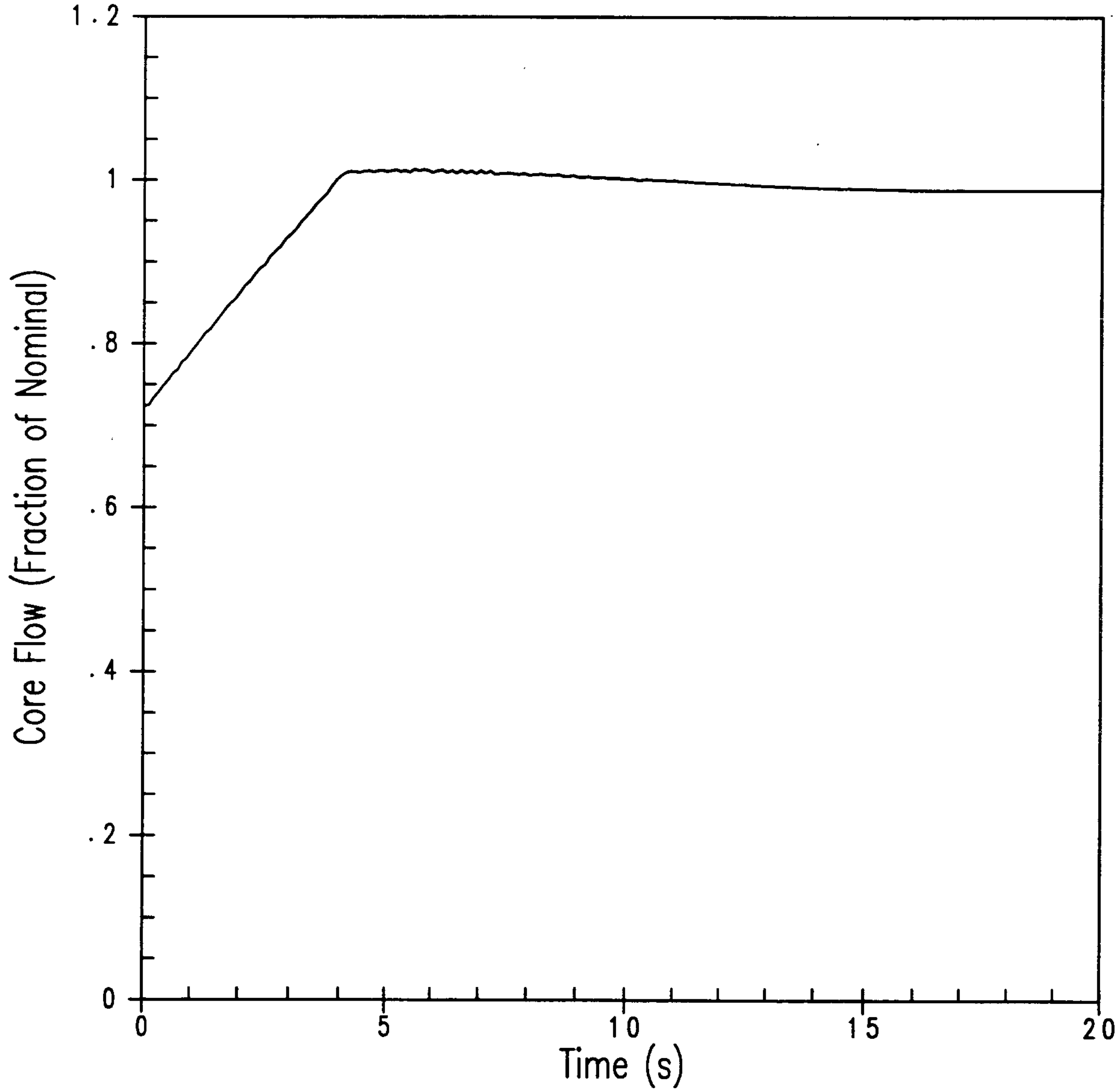

Figure 15.4.4-3

Core Flow Transient for an Improper Startup of an Inactive Reactor Coolant Pump

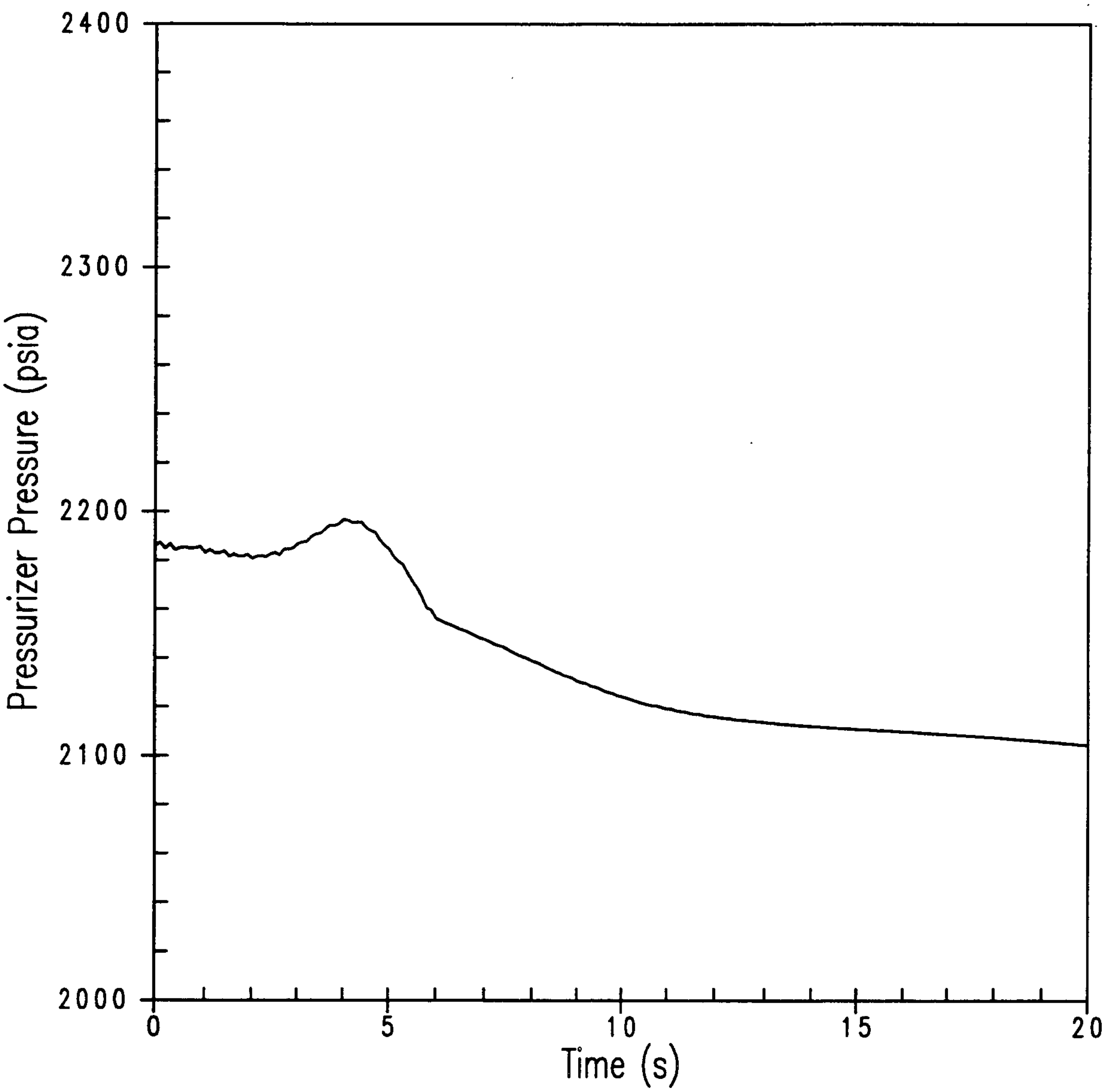

Figure 15.4.4-4

Pressurizer Pressure Transient for an Improper Startup of an Inactive Reactor Coolant Pump

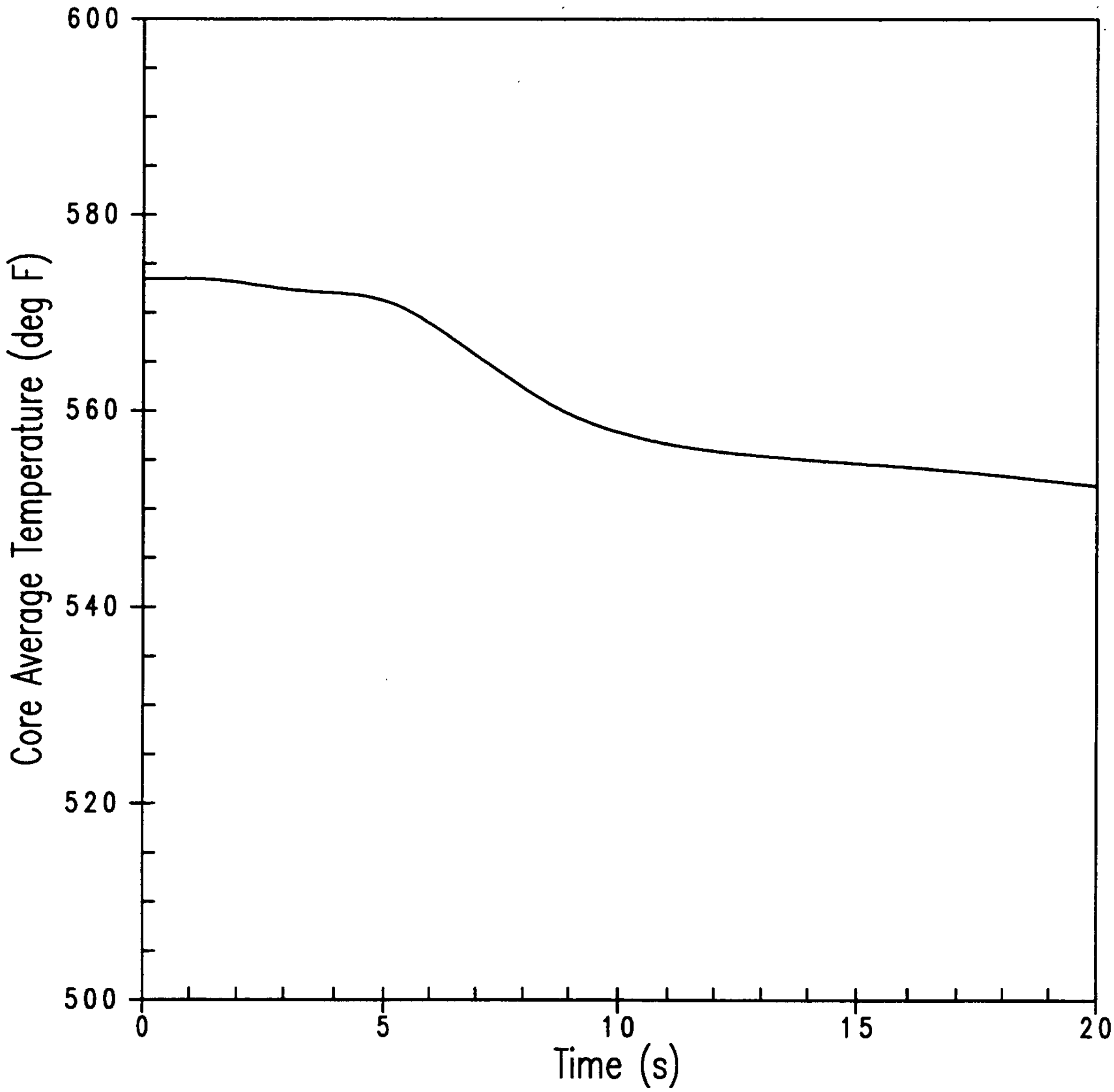

Figure 15.4.4-5

Core Average Transient for an Improper Startup of an Inactive Reactor Coolant Pump

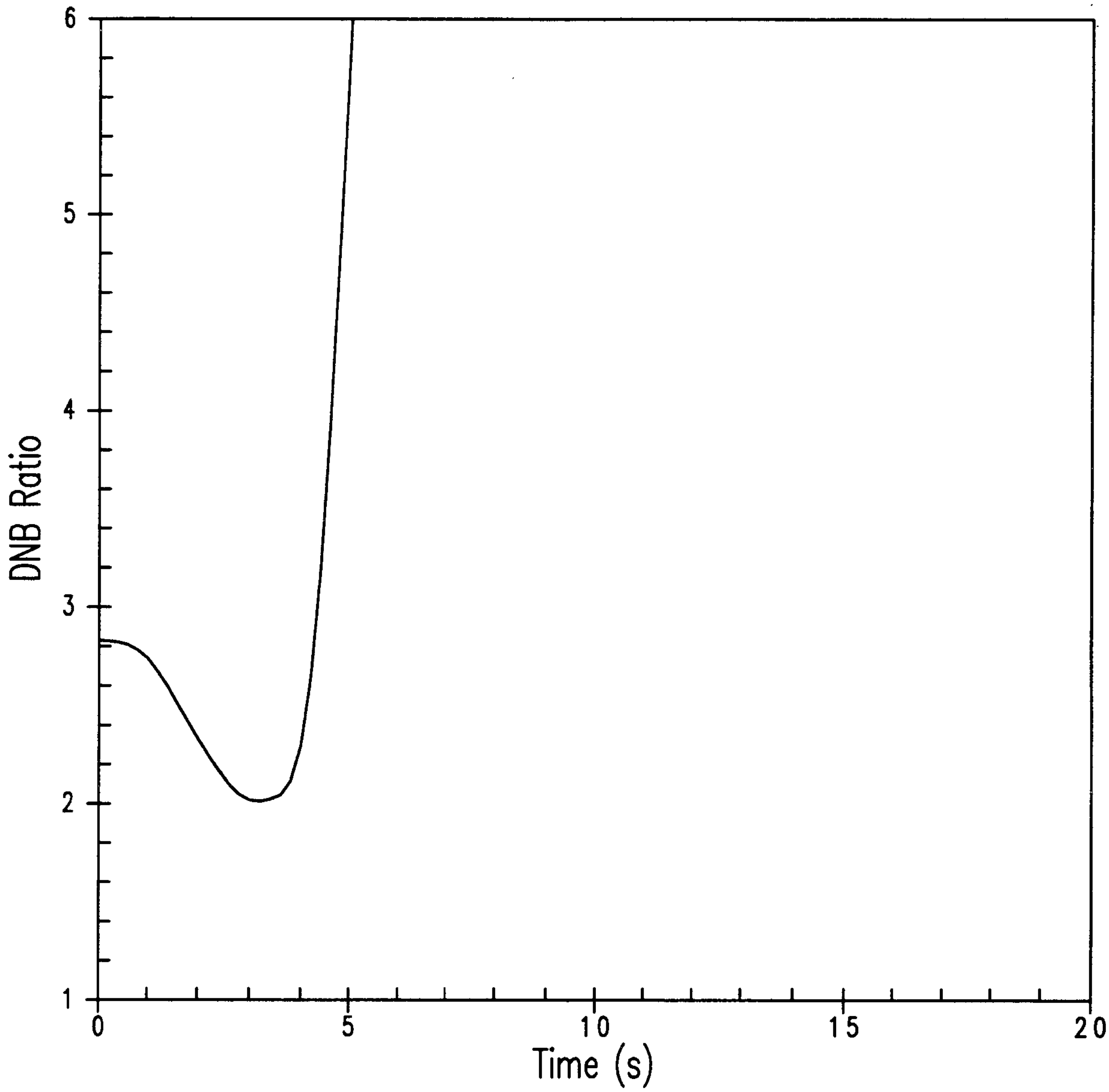

Figure 15.4.4-6

DNBR Transient for an Improper Startup of an Inactive Reactor Coolant Pump

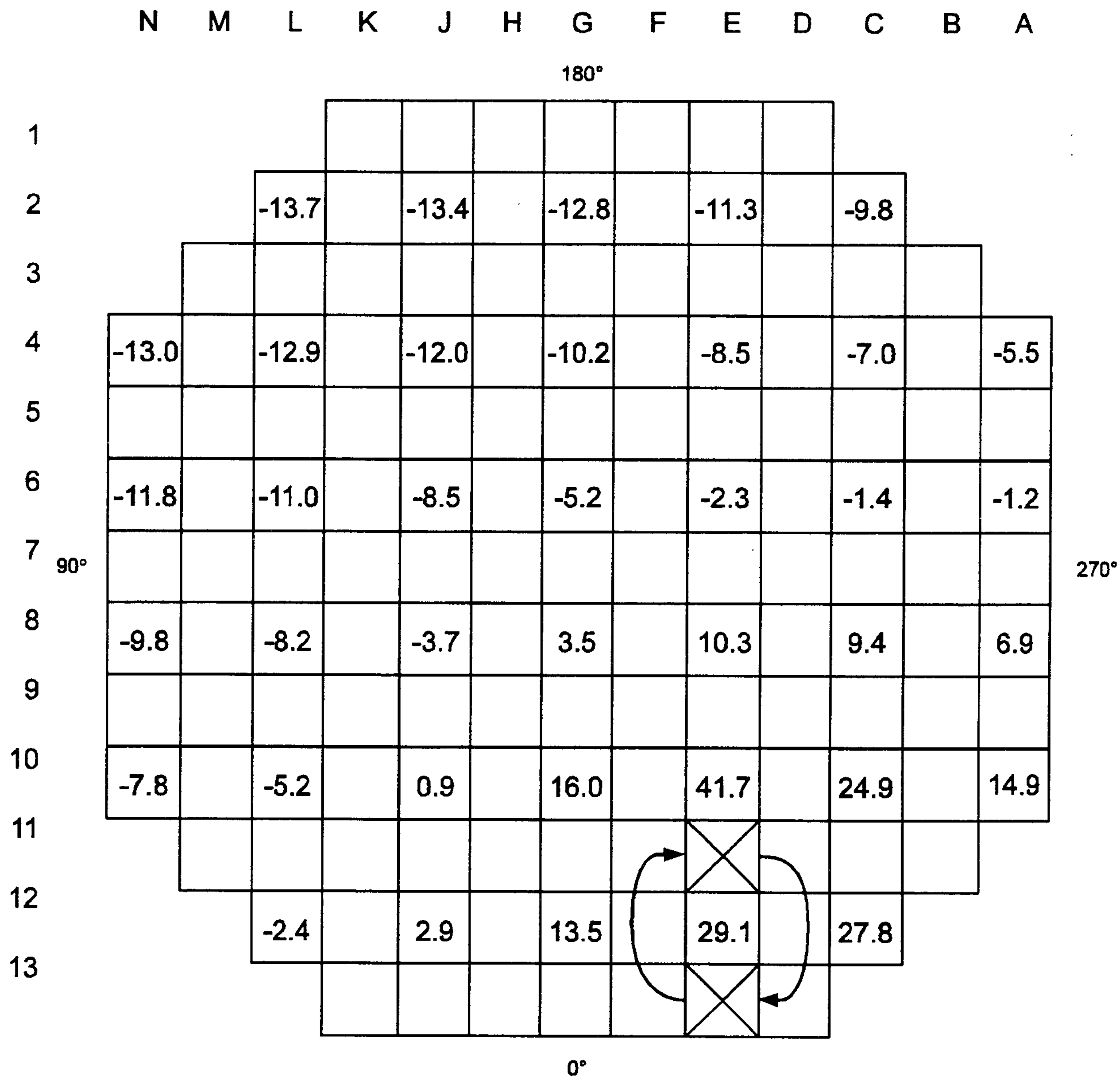

Figure 15.4.7-1

Representative Percent Change in Local Assembly Average Power for Interchange Between Region 1 and Region 3 Assembly

180°

	N	M	L	K	J	H	G	F	E	D	C	B	A
1													
2			-3.8		-4.2		-4.5		-4.6		-4.1		
3													
4	-3.0		-3.3		-3.8		-4.7		-4.6		-3.6		-2.8
5													
6	-2.2		-2.2		-2.1		-5.3		-5.2		-1.8		-1.4
7													
8	-1.2		-0.7		1.5		9.4		9.8		2.3		0.9
9													
10	-0.4		0.5		2.8		7.6		7.8		4.0		2.3
11													
12			1.1		2.4		4.1		4.6		4.0		
13													

90° (left) 270° (right)

0°

Figure 15.4.7-2

Representative Percent Change in Local Assembly Average Power for Interchange Between Region 1 and Region 2 Assembly with the BP Rods Transferred to Region 1 Assembly

180°

	N	M	L	K	J	H	G	F	E	D	C	B	A
1													
2			-5.3		-4.7		-4.3		-4.8		-5.4		
3													
4	-5.4		-4.0		-0.2		3.0		-0.3		-4.2		-5.8
5													
6	-5.1		-2.1		8.9		24.9		8.8		-2.3		-5.3
7													
8	-5.1		-2.1		8.9		24.8		8.8		-2.2		-5.3
9													
10	-5.4		-4.0		-0.1		3.1		-0.2		-4.0		-5.6
11													
12			-5.2		-4.6		-4.3		-4.7		-5.3		
13													

90° (left) 270° (right)

0°

Figure 15.4.7-3

Representative Percent Change in Local Assembly Average Power for Enrichment Error (Region 2 Assembly Loaded into Core Central Position)

180°

	N	M	L	K	J	H	G	F	E	D	C	B	A
1													
2			-5.3		-9.2		-12.8		-15.2		-16.5		
3													
4	5.7		0.3		-5.0		-10.3		-14.3		-16.4		-17.3
5													
6	15.5		12.3		3.0		-6.0		-12.1		-15.7		-16.7
7													
8	35.2		42.1		15.5		-1.4		-9.8		-14.3		-15.8
9			X										
10	44.0		46.7		19.5		0.9		-8.5		-12.8		-14.7
11													
12			25.8		11.9		0.2		-7.3		-11.1		
13													

90° (left side, row 7) — 270° (right side, row 7)

0°

Figure 15.4.7-4

Representative Percent Change in Local Assembly Average Power for Loading Region 2 Assembly into Region 1 Position Near Core Periphery

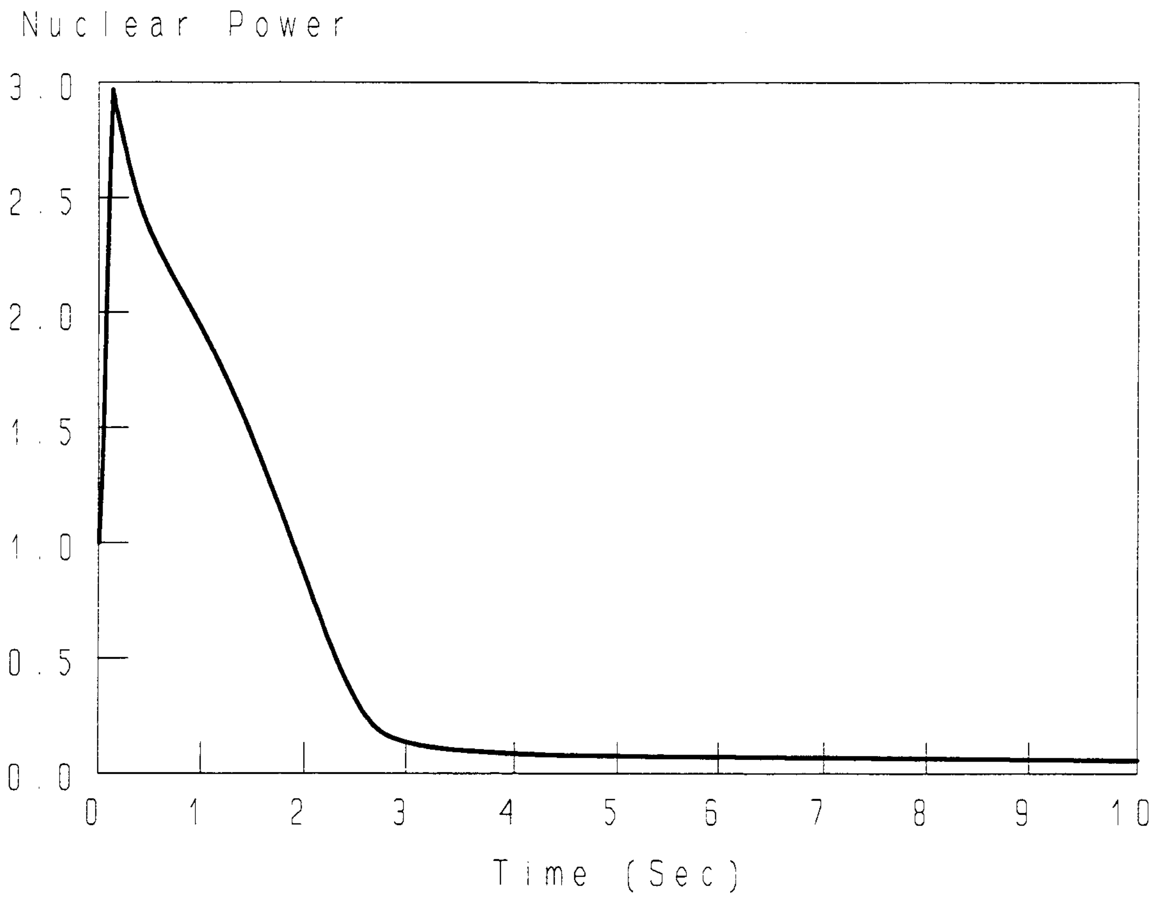

Figure 15.4.8-1

Nuclear Power Transient Versus Time at Beginning of Life, Full Power

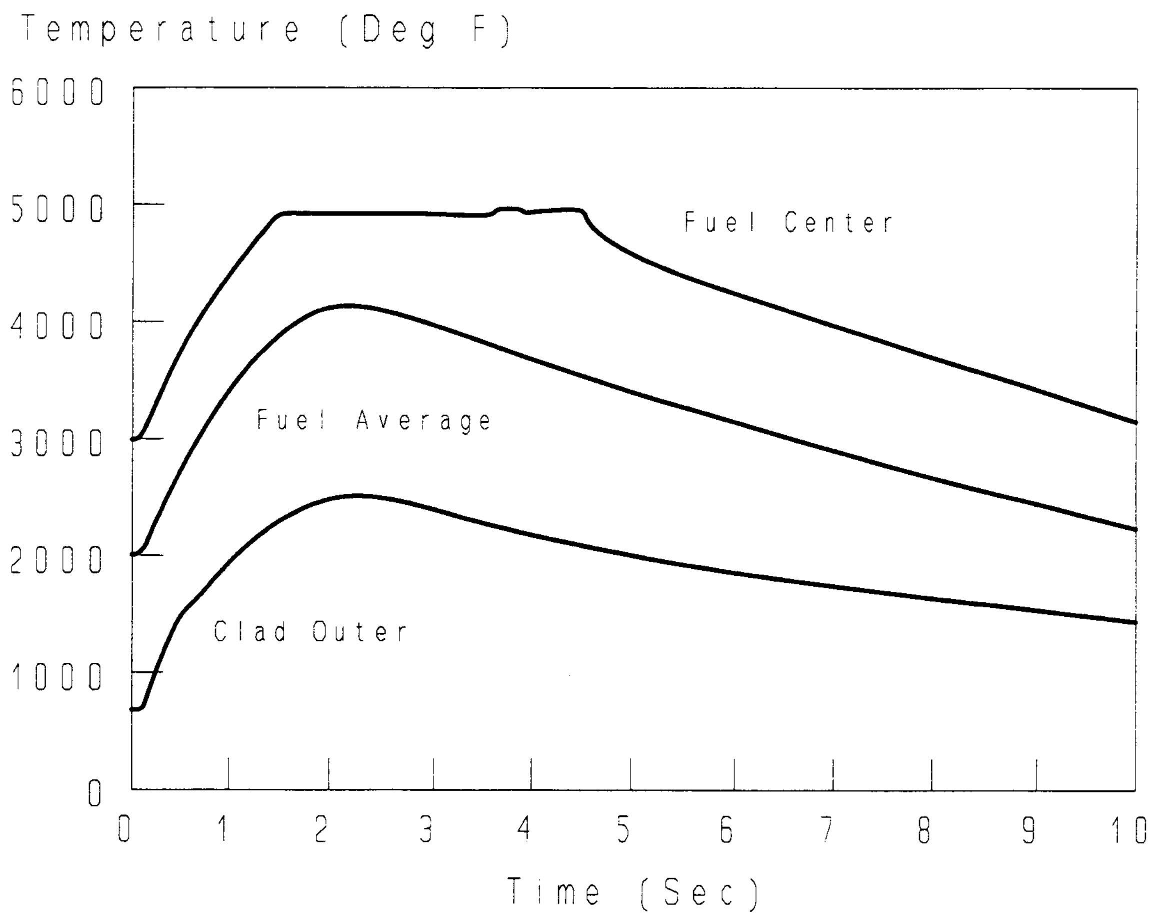

Figure 15.4.8-2

Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature Versus Time at Beginning of Life, Full Power

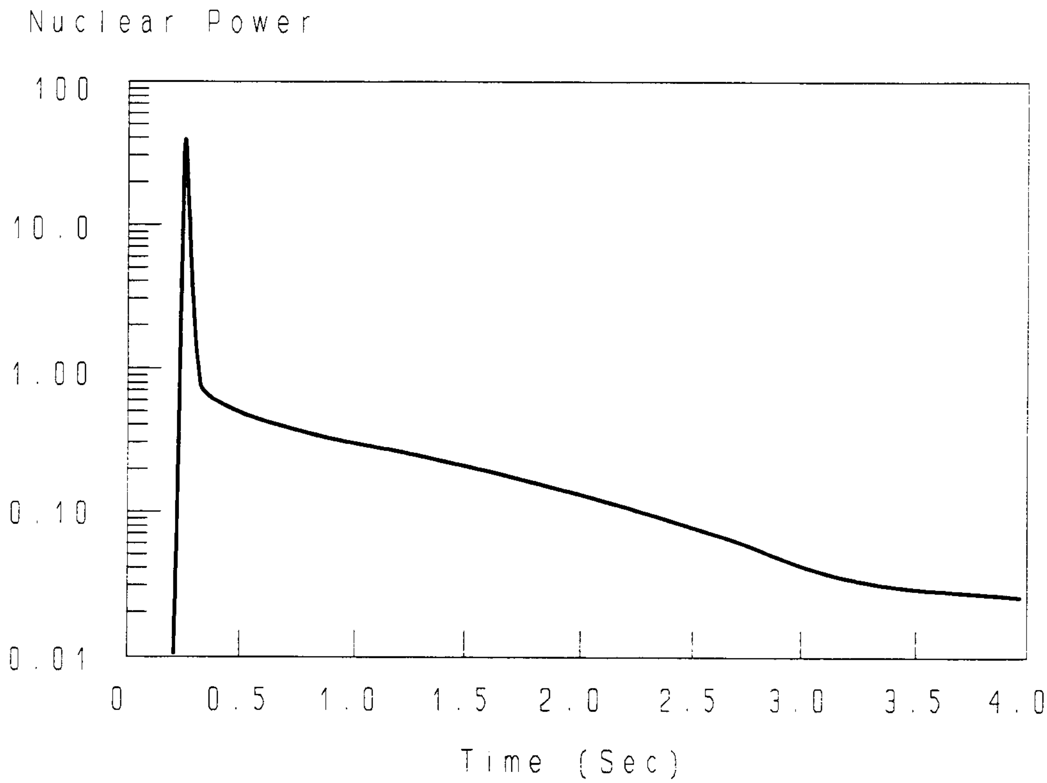

Figure 15.4.8-3

Nuclear Power Transient Versus Time at End of Life, Zero Power

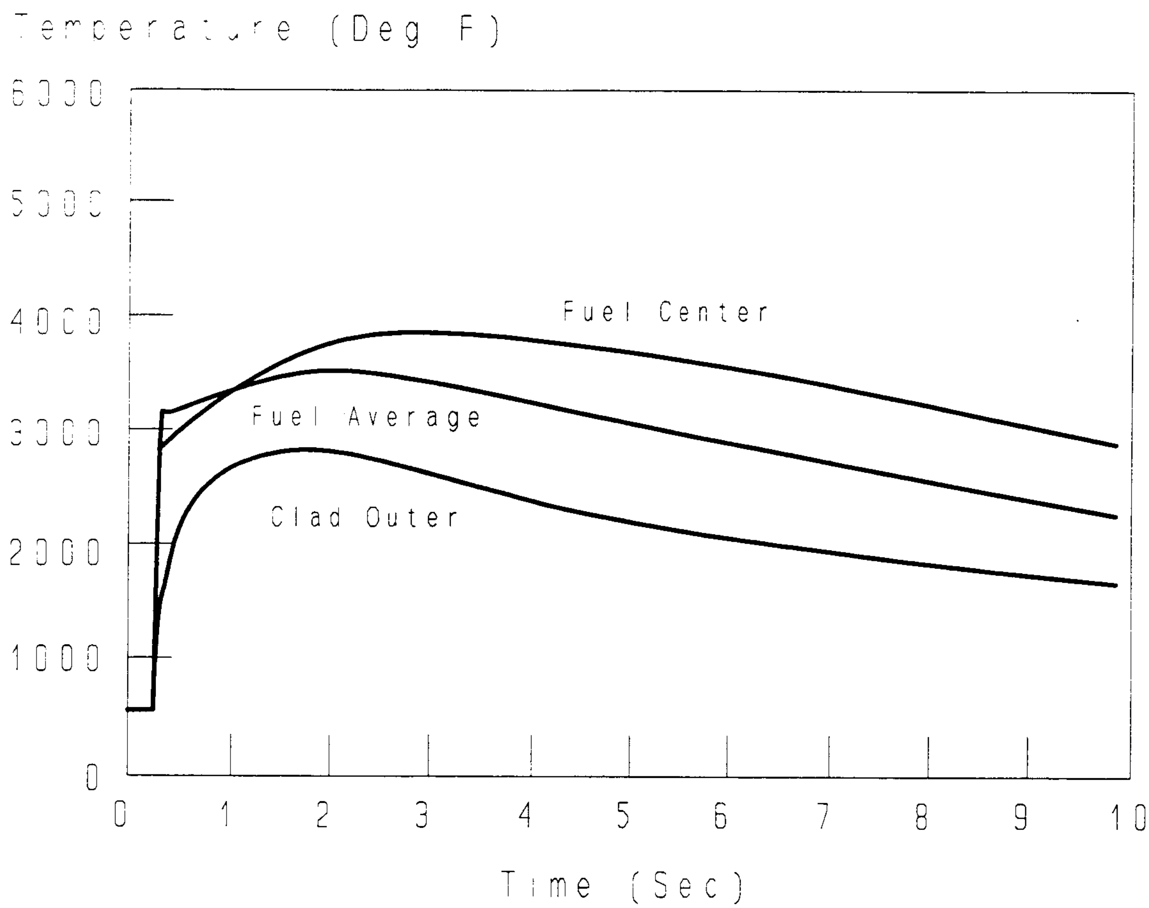

Figure 15.4.8-4

Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature Versus Time at End of Life, Zero Power