15.3 Decrease in Reactor Coolant System Flow Rate

A number of faults that could result in a decrease in the reactor coolant system flow rate are postulated. These events are discussed in this section. Detailed analyses are presented for the most limiting of the following reactor coolant system flow decrease events:

- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (locked rotor)
- Reactor coolant pump shaft break

The first event is a Condition II event, the second is a Condition III event, and the last two are Condition IV events.

The four limiting flow rate decrease events described above are analyzed in this section. The most severe radiological consequences result from the reactor coolant pump shaft seizure accident discussed in subsection 15.3.3. Doses are reported only for that case.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or an electrical failure of a reactor coolant pump or from a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the event, the immediate effect of the loss of coolant flow is a rapid increase in the coolant temperature.

Normal power for the pumps is supplied through two buses connected to the generator. When a generator trip occurs, the buses are supplied from offsite power. The pumps continue to operate.

A partial loss of coolant flow is classified as a Condition II incident (a fault of moderate frequency), as defined in subsection 15.0.1.

Protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated by two-out-of-four low-flow signals in any reactor coolant loop. Above permissive P8, low flow in any one cold leg actuates a reactor trip (see Section 7.2). Between approximately 10-percent power (permissive P10) and the power level corresponding to permissive P8, low flow in any two cold legs actuates a reactor trip.

As specified in GDC 17 of 10 CFR Part 50, Appendix A, the effects of a loss of offsite power are considered in evaluating partial loss of forced reactor coolant flow transients. As discussed in subsection 15.0.14, the loss of offsite power is considered to be a potential consequence of the event due to disruption of the electrical grid following a turbine trip during the event. A delay of 3 seconds is assumed between the turbine trip and the loss of offsite power. The primary effect of the loss of offsite power is to cause the remaining operating reactor coolant pumps to coast down.

15.3.1.2 Analysis of Effects and Consequences

15.3.1.2.1 Method of Analysis

This transient is analyzed using three computer codes. First, the LOFTRAN code (Reference 1) is used to calculate the core flow during the transient, the time of reactor trip based on the input loop flows, the nuclear power transient, and the primary system pressure and temperature transients as predicted from the loss of two reactor coolant pumps. The FACTRAN code (Reference 2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the WESTAR code (see Section 4.4) is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical cell or the thimble cell.

15.3.1.2.2 Initial Conditions

Initial reactor power, pressure, and reactor coolant system temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the DNBR limit, as described in WCAP-11397-P-A (Reference 5).

Plant characteristics and initial conditions assumed in this analysis are further discussed in subsection 15.0.3.

15.3.1.2.3 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Figure 15.0.4-1). This is equivalent to a total integrated Doppler reactivity from 0- to 100-percent power of 0.0118 Δk.

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

For these analyses, a curve of trip reactivity versus time based on a 2.4-second rod cluster control assembly insertion time to the dashpot is used (see subsection 15.0.5).

15.3.1.2.4 Flow Coastdown

A conservative flow coastdown is used to simulate the transient. The flow coastdown is calculated based on reactor coolant system pressure losses and reactor coolant pump characteristics. Reactor coolant fluid momentum is neglected to obtain a conservative calculation.

Plant systems and equipment necessary 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 equipment adversely affects the consequences of the accident.

15.3.1.2.5 Results

Figures 15.3.1-1 through 15.3.1-6 show the transient response for the loss of two reactor coolant pumps with offsite power available. Figure 15.3.1-6 shows the DNBR to be always greater than the design limit value as defined in Section 4.4.

The plant is tripped by the low-flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rods is not greatly reduced. The average fuel and cladding temperatures do not increase significantly above their initial values.

The calculated sequence of events for the case analyzed is shown in Table 15.3-1. The affected reactor coolant pumps continue to coast down, and the core flow reaches a new equilibrium value.

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

In the event that a loss of offsite power occurs as a consequence of a turbine trip during a partial loss of reactor coolant flow, the DNB design basis continues to be met. The loss of offsite power causes the remaining two operating reactor coolant pumps to coast down.

At the time when the remaining two operating reactor coolant pumps start coasting down, reactor trip has already been initiated, core heat flux has started decreasing, and DNBR is increasing. DNBR continues to increase as the remaining two reactor coolant pumps coast down because the core heat flux has decreased and is continuing to decrease rapidly. The minimum DNB ratio occurs at the same time for cases with and without offsite power available.

15.3.1.3 Conclusions

The analysis shows that, for the partial loss of reactor coolant flow, the DNBR does not decrease below the design basis value at any time during the transient. The DNBR design basis is described in Section 4.4. The applicable Standard Review Plan, subsection 15.3.1 Reference 4), evaluation criteria are met.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description

A complete loss of flow accident may result from a simultaneous loss of electrical supplies to the reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature.

Electric power for the reactor coolant pumps is supplied through buses, two connected to the generator. When a generator trip occurs, the buses receive power from external power lines and the pumps continue to supply coolant flow to the core.

A complete loss of flow accident is a Condition III event (an infrequent fault), as defined in subsection 15.0.1. The following signals provide protection against this event:

- Reactor coolant pump underspeed
- Low reactor coolant loop flow

The reactor trip on reactor coolant pump underspeed protects against conditions that can cause a loss of voltage to the reactor coolant pumps. This function is blocked below approximately 10-percent power (permissive P10).

The reactor trip on reactor coolant pump underspeed is also provided to trip the reactor for an underfrequency condition resulting from frequency disturbances on the power grid. If the maximum grid frequency decay rate is less than approximately 5 hertz per second, this trip protects the core from underfrequency events. WCAP-8424, Revision 1 (Reference 3), provides analyses of grid frequency disturbances and the resulting protection requirements that are applicable to the AP600.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions that affect only one or two reactor coolant loop cold legs. This function is generated by two-out-of-four low-flow signals per reactor coolant loop cold leg. Above permissive P8, low flow in any loop actuates a reactor trip. Between approximately 10-percent power (permissive P10) and the power level corresponding to permissive P8, low flow in any two reactor coolant loop cold legs actuates a reactor trip. If the maximum grid frequency decay rate is less than approximately 2.5 hertz per second, this trip function also protects the core from this underfrequency event. This effect is described in WCAP-8424, Revision 1 (Reference 3).

15.3.2.2 Analysis of Effects and Consequences

15.3.2.2.1 Method of Analysis

The complete loss of flow transient is analyzed for a loss of power to four reactor coolant pumps.

For the case analyzed with a complete loss of voltage, followed by the reactor coolant pumps coasting down, the method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in subsection 15.3.1, with one exception. Following the loss of power supply to all pumps at power, a reactor trip is actuated by the reactor coolant pump underspeed trip.

A loss of forced primary coolant flow can result from a reduction in the reactor coolant pump motor supply frequency. The results of the complete loss of voltage, followed by the reactor coolant pump coasting down, bound the complete loss of flow initiated by a frequency decay of up to 5 hertz per second. Therefore, only the results of the complete loss of voltage case are presented in subsection 15.3.2.2.2.

15.3.2.2.2 Results

Figures 15.3.2-1 through 15.3.2-6 show the transient response for the complete loss of voltage to all four reactor coolant pumps. The reactor is assumed to trip on the reactor coolant pump underspeed signal. Figure 15.3.2-6 shows that the DNBR is always greater than the design limit value defined in Section 4.4.

The calculated sequences of events for the cases analyzed are shown in Table 15.3-1. The reactor coolant pumps continue to coast down, and natural circulation flow is established, as demonstrated in subsection 15.2.6. With the reactor tripped, a stable plant condition is attained. Normal plant shutdown may then proceed.

15.3.2.3 Conclusions

The analysis demonstrates that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the design basis limit value at any time during the transient. The design basis for the DNBR is described in Section 4.4. The applicable Standard Review Plan, subsection 15.3.1 (Reference 4), evaluation criteria are met.

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low-flow signal.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant temperature to increase and expand. At the same time, heat transfer to the shell side of the steam generators is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. (Turbine steam flow is reduced to 0 upon plant trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

This event is classified as a Condition IV incident (a limiting fault), as defined in subsection 15.0.1.

15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Method of Analysis

Two digital computer codes are used to analyze this transient. The LOFTRAN code (Reference 1) calculates the resulting core flow transient following the pump seizure and the nuclear power following reactor trip. This code is also used to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated by using the FACTRAN code (Reference 2). This code uses the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes a film-boiling heat transfer coefficient.

At the beginning of the postulated locked rotor accident (at the time the shaft in one of the reactor coolant pumps is assumed to seize), the plant is assumed to be in operation under the most adverse steady-state operating conditions, that is, maximum steady-state thermal power, maximum steady-state pressure, and maximum steady-state coolant average temperature. Plant characteristics and initial conditions are further discussed in subsection 15.0.3. The accident is evaluated for both cases with and without offsite power available. For the case without offsite power available, power is lost to the unaffected pumps at the time of reactor trip.

For the peak pressure evaluation, the initial pressure is conservatively estimated as 50 psi above nominal pressure (2250 psia), which allows for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure.

15.3.3.2.2 Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin 1.45 seconds after the flow in the affected loop reaches the reactor trip setpoint. No credit is taken for the pressure-reducing effect of the pressurizer spray, steam dump, or controlled feedwater flow after plant trip. Although these operations are expected to result in a lower peak reactor coolant system pressure, an additional conservatism is provided by ignoring their effect.

The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.

15.3.3.2.3 Evaluation of Departure from Nucleate Boiling in the Core During the Accident

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

In the evaluation, the rod power at the hot spot is conservatively assumed to be 2.6 times the average rod power (that is, $F_Q = 2.6$) at the initial core power level.

15.3.3.2.4 Film-Boiling Coefficient

The film-boiling coefficient is calculated in the FACTRAN code (Reference 2) using the Bishop-Sandberg-Tong film-boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step, based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

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

15.3.3.2.5 Fuel Cladding Gap Coefficient

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

15.3.3.2.6 Zirconium-Steam Reaction

The zirconium-steam reaction can become significant above a cladding temperature of 1800°F. The Baker-Just parabolic rate equation is used to define the rate of the zirconium-steam reaction:

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

where:

w = amount reacted (mg/cm^2)

t = time (s)
T = temperature (Kelvin)

The reaction heat is 1510 cal/g.

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

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 equipment adversely affects the consequences of the accident.

15.3.3.2.7 Results

Figures 15.3.3-1 through 15.3.3-7 show the transient results for one locked rotor with four reactor coolant pumps in operation without offsite power available. The without-offsite-power case bounds the results for the case with offsite power. The results of these calculations are also summarized in Table 15.3-2. The peak reactor coolant system pressure reached during the transient is less than that which causes stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also, the peak cladding surface temperature is considerably less than 2700°F. The cladding temperature is conservatively calculated, assuming that DNB occurs at the initiation of the transient. These results represent the most limiting conditions with respect to the locked rotor event or the pump shaft break.

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

15.3.3.3 Radiological Consequences

The evaluation of the radiological consequences of a postulated locked reactor coolant pump rotor accident assumes that the reactor has been 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 have resulted in a buildup of activity in the secondary coolant.

As a result of the accident, it is determined that no more than 18 percent of the fuel rods are damaged such that the activity contained in the fuel-cladding gap is released to the reactor coolant. Activity carried over to the secondary side because of primary-to-secondary leakage is available for release to the environment via the steam line safety valves or the power-operated relief valves.

15.3.3.3.1 Source Term

The significant radionuclide releases due to the locked rotor 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 basis fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of the gap inventory of fission products from the portion of the core assumed to fail because of the accident.

Based on NUREG-1465 (Reference 6), 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).

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.

15.3.3.3.2 Release Pathways

There are two 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 iodine and cesium from the primary coolant mix with the secondary coolant. 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.

Credit is taken for the 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.

15.3.3.3.3 Dose Calculation Models

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

15.3.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.3-3.

15.3.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant 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 a 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.

15.3.3.3.6 Doses

Using the assumptions from Table 15.3-3, the calculated total effective dose equivalent (TEDE) doses are determined to be 1.0 rem at the site boundary and 0.5 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is identified as 10 percent or less consistent with the Standard Review Plan (Reference 4).

At the time the locked reactor coolant pump rotor event 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 locked rotor event, the resulting total dose remains less than 0.5 rem TEDE.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip occurs on a low-flow signal in the affected loop.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced because: 1) the reduced flow results in a decreased tube-side film coefficient, and 2) the reactor coolant in the tubes cools down while the shell-side temperature increases. (Turbine steam flow is reduced to 0 upon plant trip.) The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

This event is classified as a Condition IV incident (limiting fault), as defined in subsection 15.0.1.

15.3.4.2 Conclusion

With a failed shaft, the impeller could be free to spin in a reverse direction as opposed to being fixed in position as assumed in the locked rotor analysis. However, the net effect on core flow is negligible, resulting in only a slight decrease in the end point (steady-state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient. The calculated results presented for the locked rotor analysis bound the reactor coolant pump shaft break event.

15.3.5 Combined License Information

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

15.3.6 References

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

2. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908 (Proprietary) and WCAP-7337 (Nonproprietary), June 1975.

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

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

5. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Nonproprietary), April 1989.

6. Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

Table 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

Accident	Event	Time (seconds)
Partial loss of forced reactor coolant flow		
– Loss of two pumps with four pumps running	Coastdown begins	0.00
	Low-flow reactor trip	0.56
	Rods begin to drop	2.01
	Minimum DNBR occurs	3.90
Complete loss of forced reactor coolant		
– Loss of four pumps with four pumps running	Operating pumps lose power and begin coasting down	0.00
	Reactor coolant pump underspeed trip point reached	0.32
	Rods begin to drop	1.09
	Minimum DNBR occurs	2.80
Reactor coolant pump shaft seizure (locked rotor)		
– One locked rotor with four pumps running with offsite power available	Rotor on one pump locks	0.00
	Low-flow trip point reached	0.01
	Rods begin to drop	1.46
	Maximum cladding temperature occurs	2.20
	Maximum reactor coolant system pressure occurs	4.20
– One locked rotor with four pumps running without offsite power available	Rotor on one pump locks	0.00
	Low-flow trip point reached	0.01
	Rods begin to drop, loss of offsite power occurs	1.46
	Maximum cladding temperature occurs	3.10
	Maximum reactor coolant system pressure occurs	4.10

Table 15.3-2

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS (FOUR REACTOR COOLANT PUMPS OPERATING INITIALLY)

	Without Offsite Power Available
Maximum reactor coolant system pressure (psia)	2657
Maximum cladding temperature, core hot spot (°F)	1817
Zr-H_2O reaction, core hot spot (percentage by weight)	0.46

Table 15.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR 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	0.04 μCi/g dose equivalent activity I-131 (10% of design basis reactor coolant concentrations listed in Table 11.1-2)
Fraction of fuel rods assumed to fail	0.18
Core activity	See Table 15A-3
Fission product gap fractions	0.036
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95
Reactor coolant mass (lb)	3.39 E+05
Secondary coolant mass (lb)	2.15 E+05
Condenser	Not available
Duration of accident (hr)	8
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	260[b]
Steam released (lb)	
– 0-2 hours	5.75 E+05
– 2-8 hours	1.04 E+06
Partition coefficient in steam generators for iodine and cesium	0.01

<u>Note:</u>

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

b. Equivalent to 1000 gpd at 561.5°F and 2250 psia

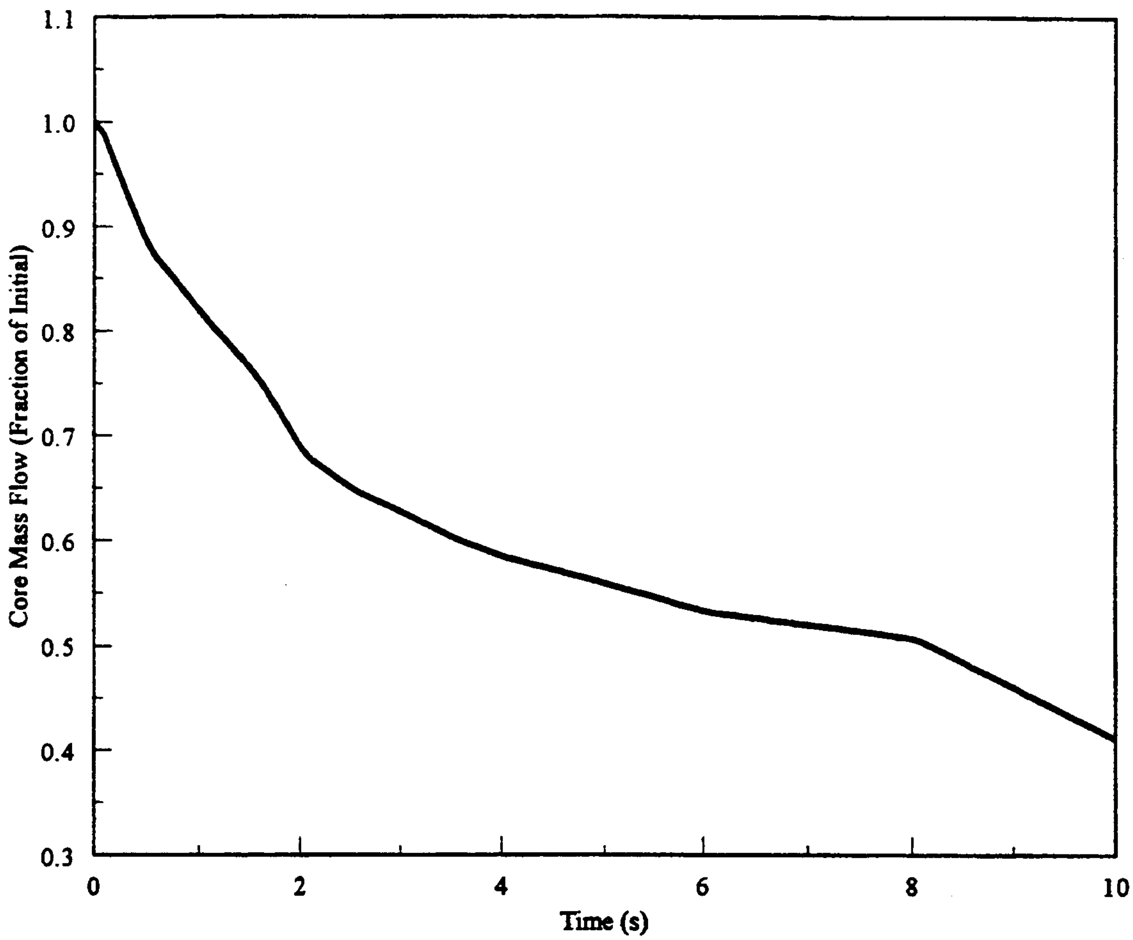

Figure 15.3.1-1

Flow Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

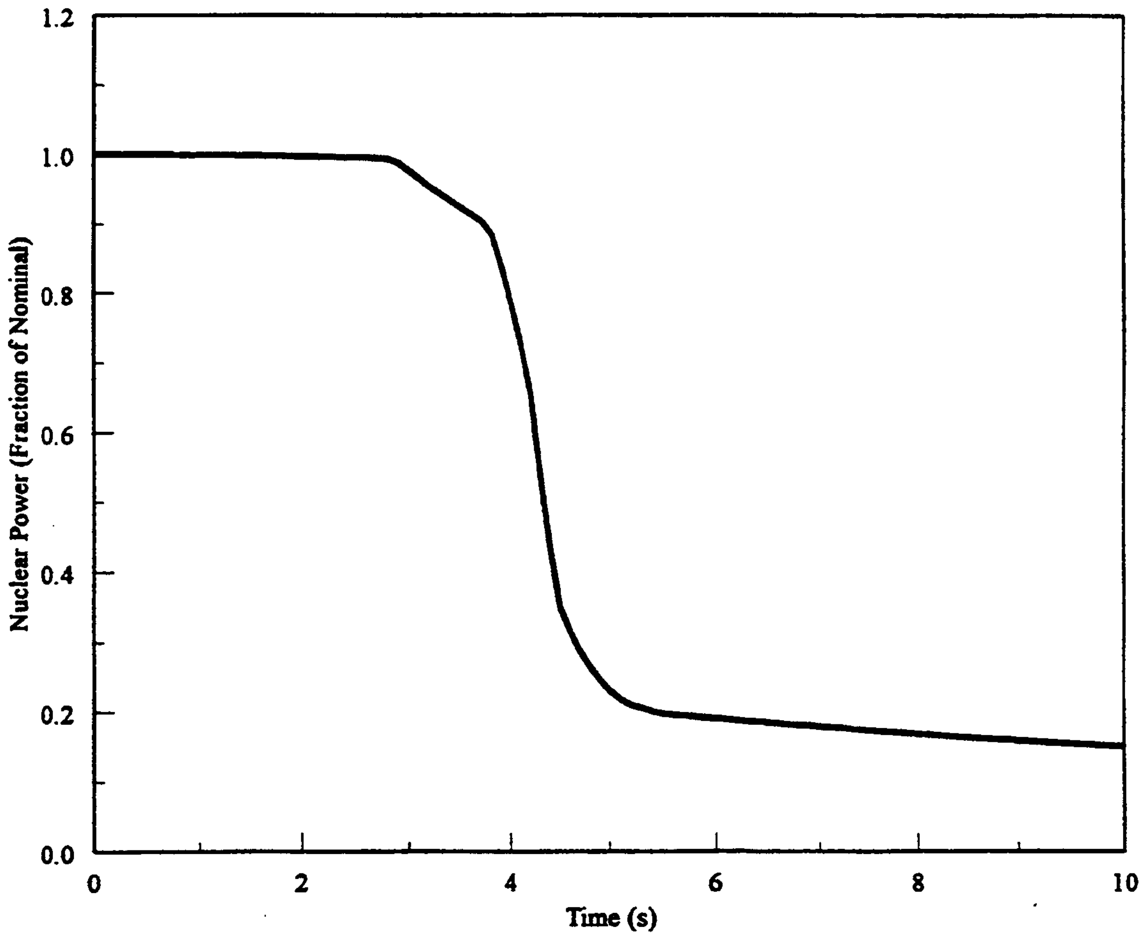

Figure 15.3.1-2

Nuclear Power Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

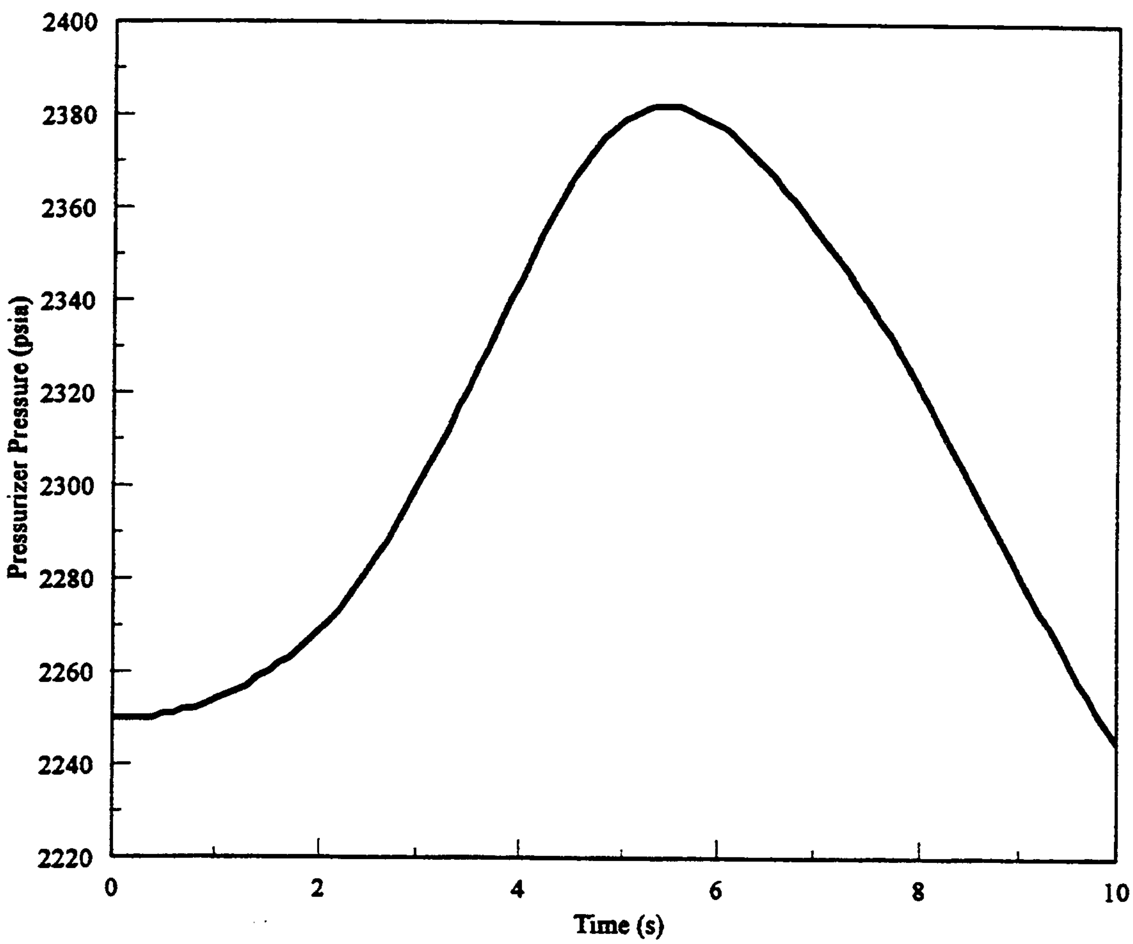

Figure 15.3.1-3

Pressurizer Pressure Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

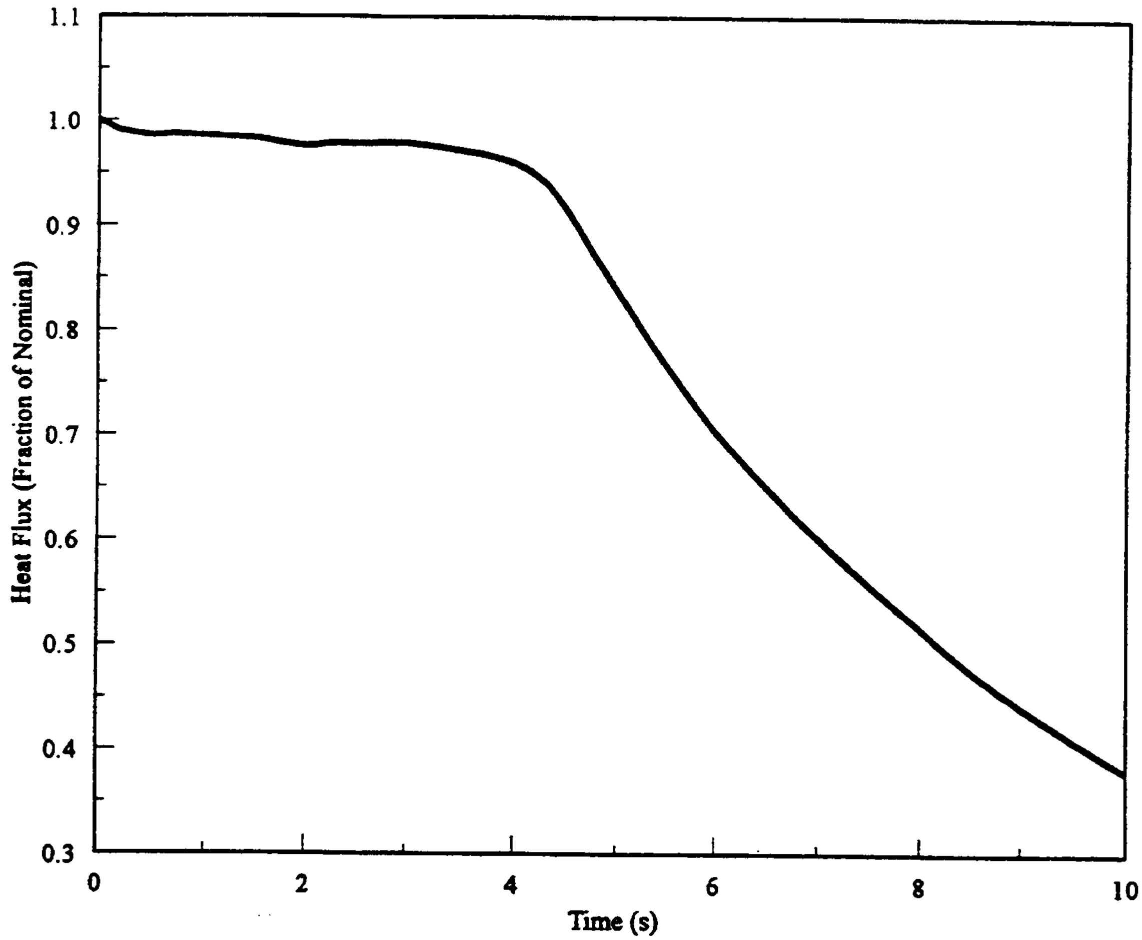

Figure 15.3.1-4

Average Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

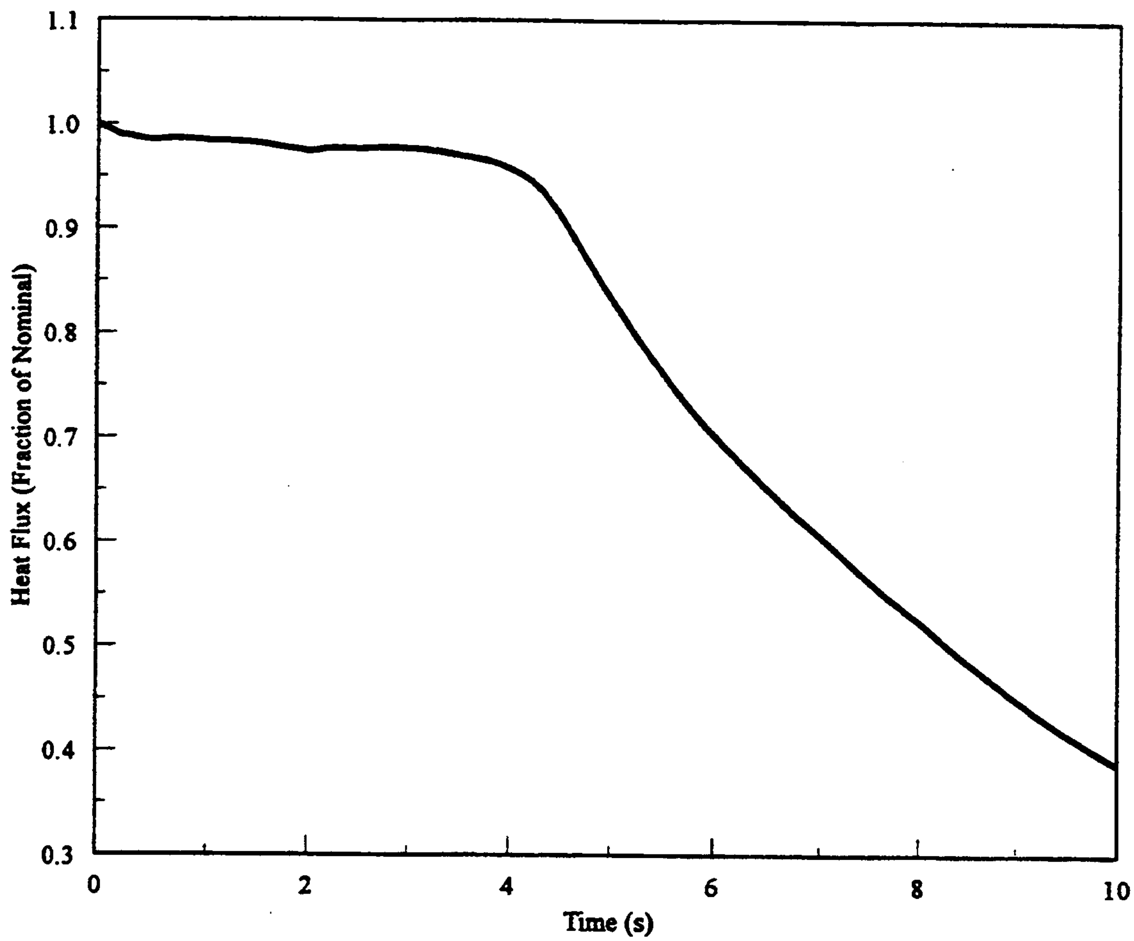

Figure 15.3.1-5

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

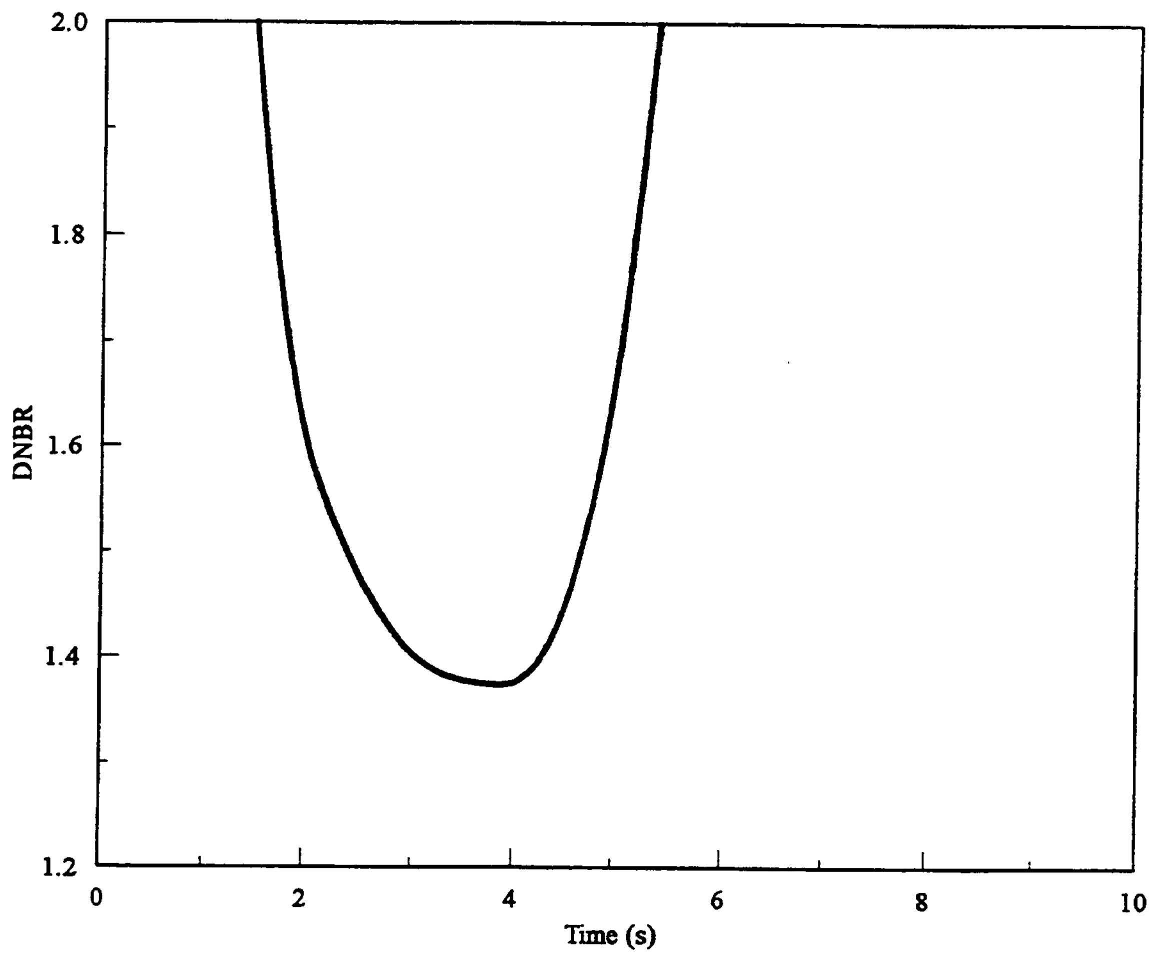

Figure 15.3.1-6

DNB Transient for Four Cold Legs in Operation, Two Pumps Coasting Down

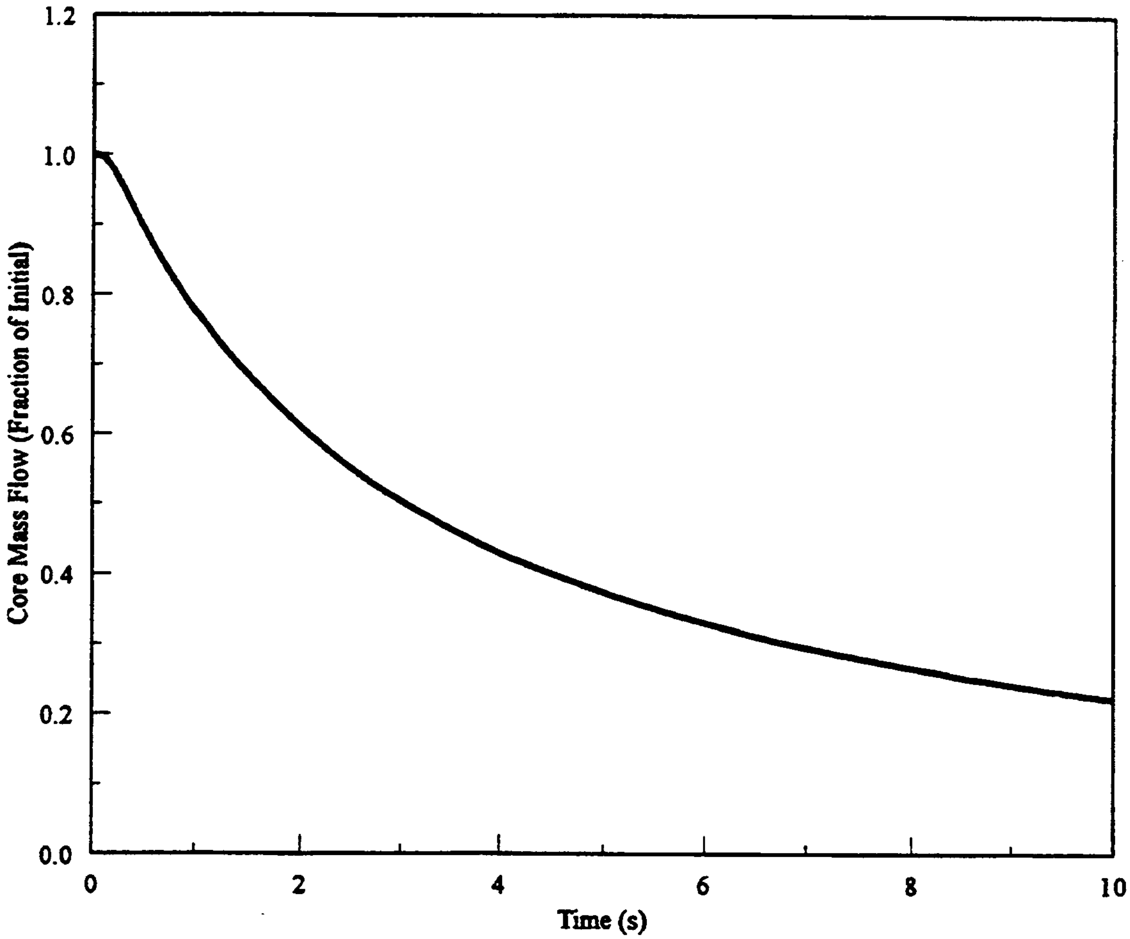

Figure 15.3.2-1

Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

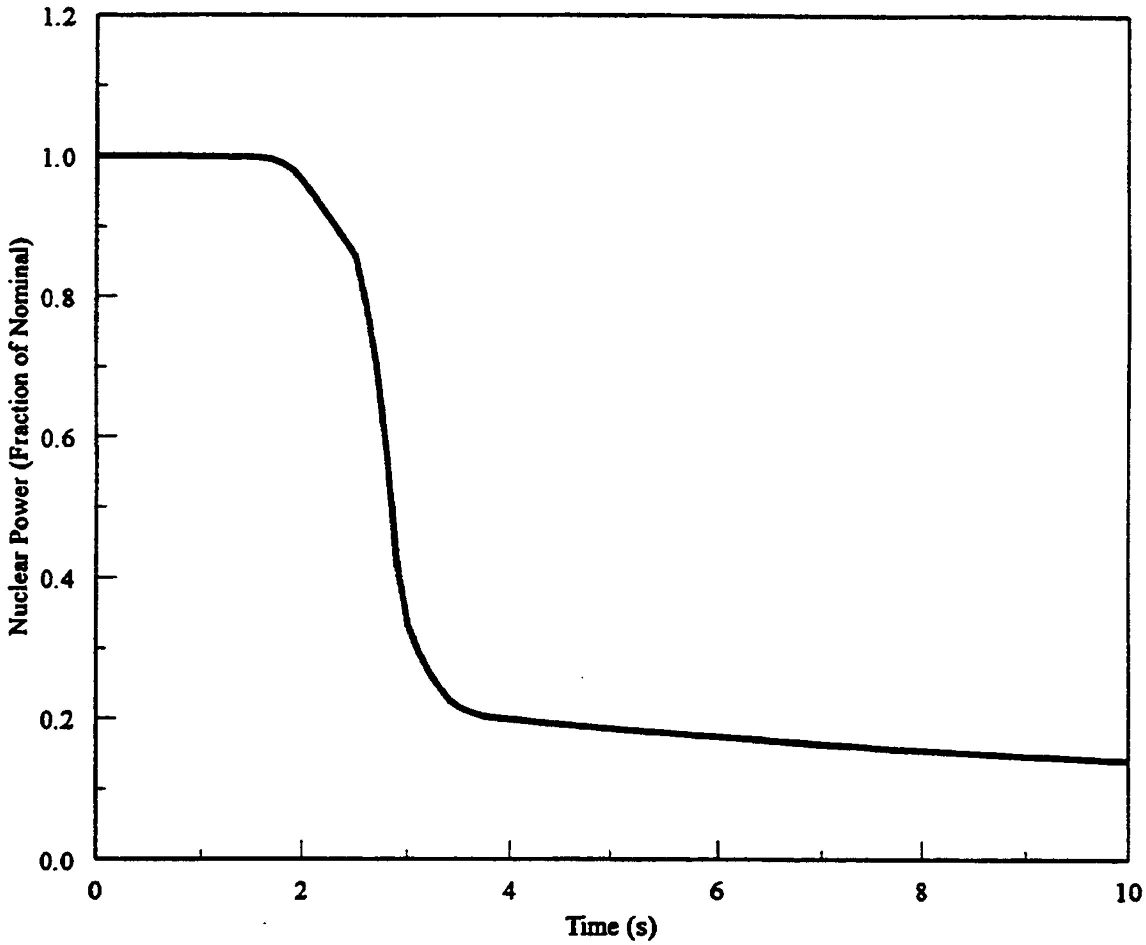

Figure 15.3.2-2

Nuclear Power Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

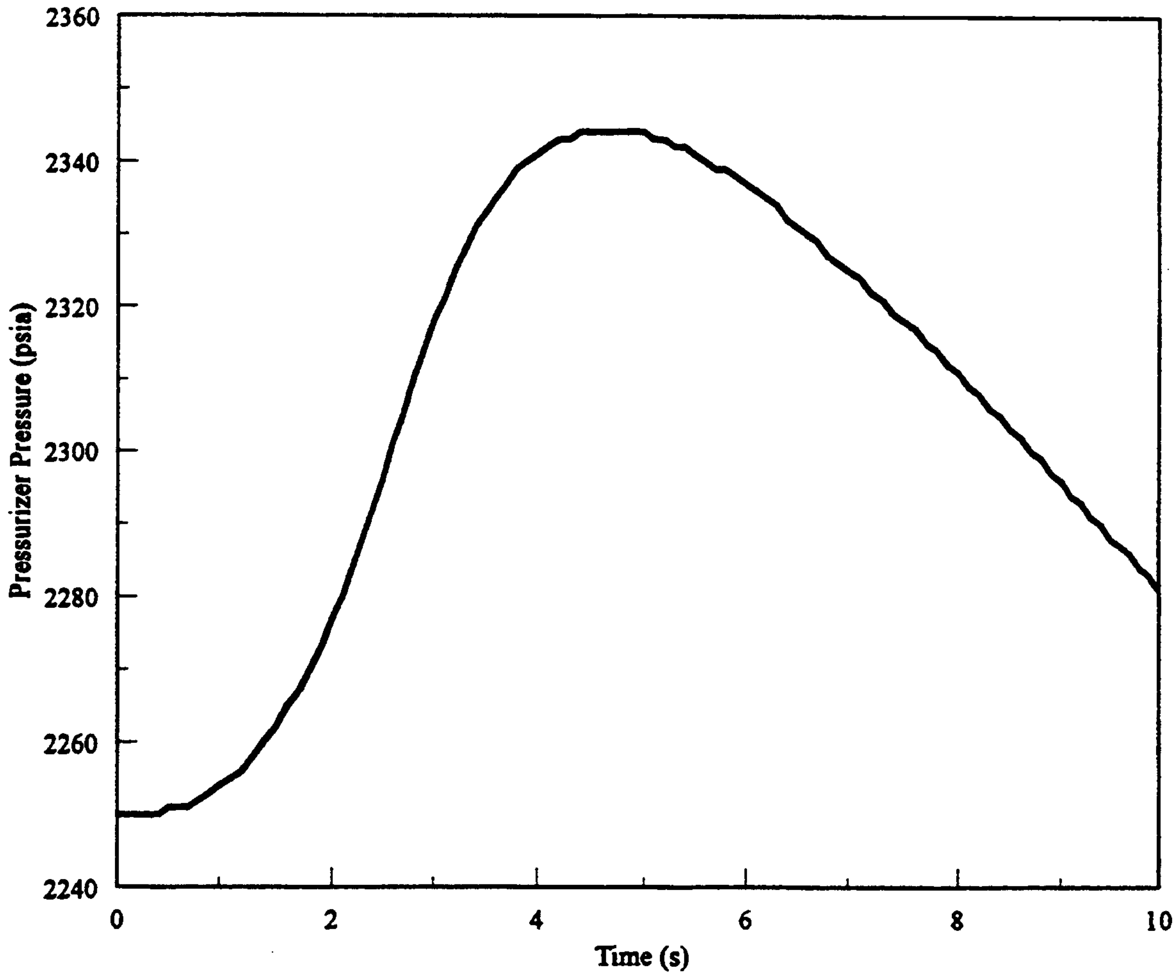

Figure 15.3.2-3

Pressurizer Pressure Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

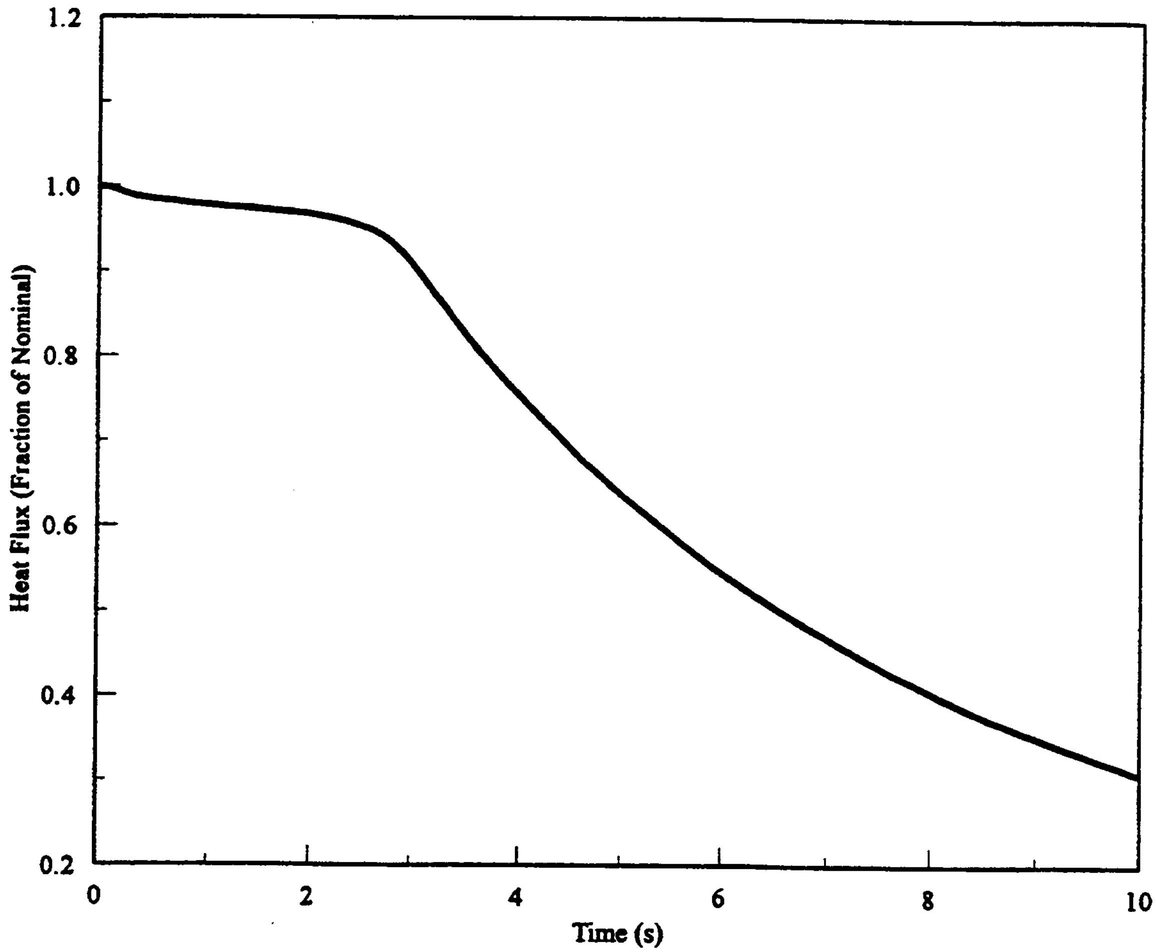

Figure 15.3.2-4

Average Channel Heat Flux Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

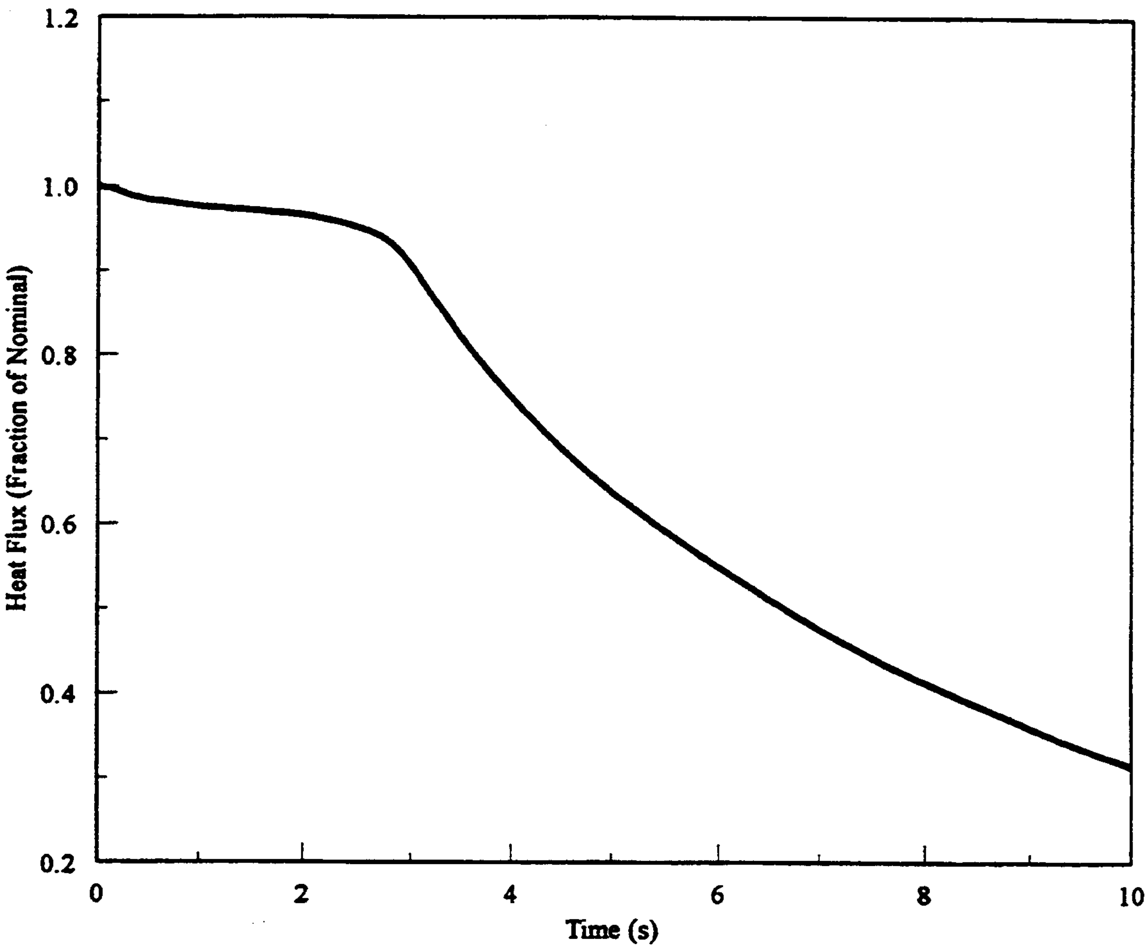

Figure 15.3.2-5

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

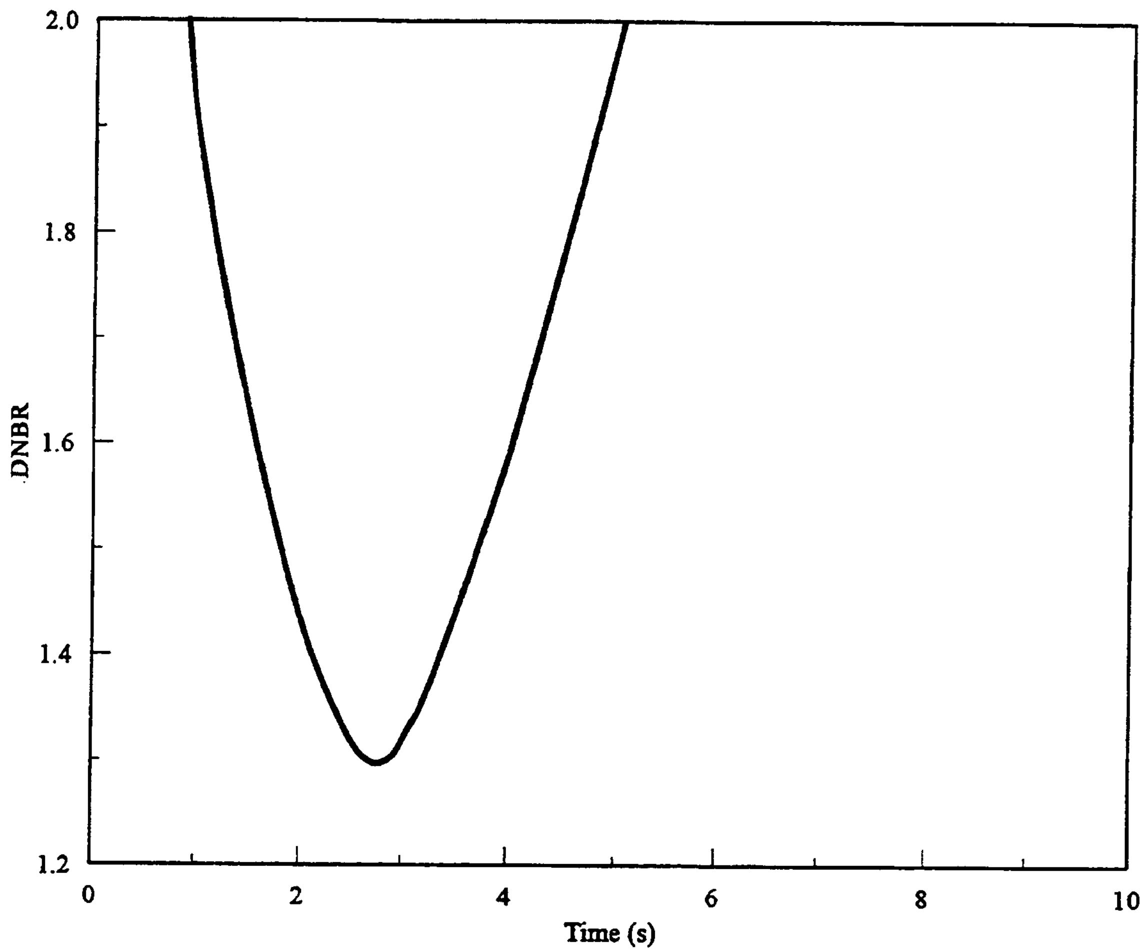

Figure 15.3.2-6

DNBR Transient for Four Cold Legs in Operation, Four Pumps Coasting Down

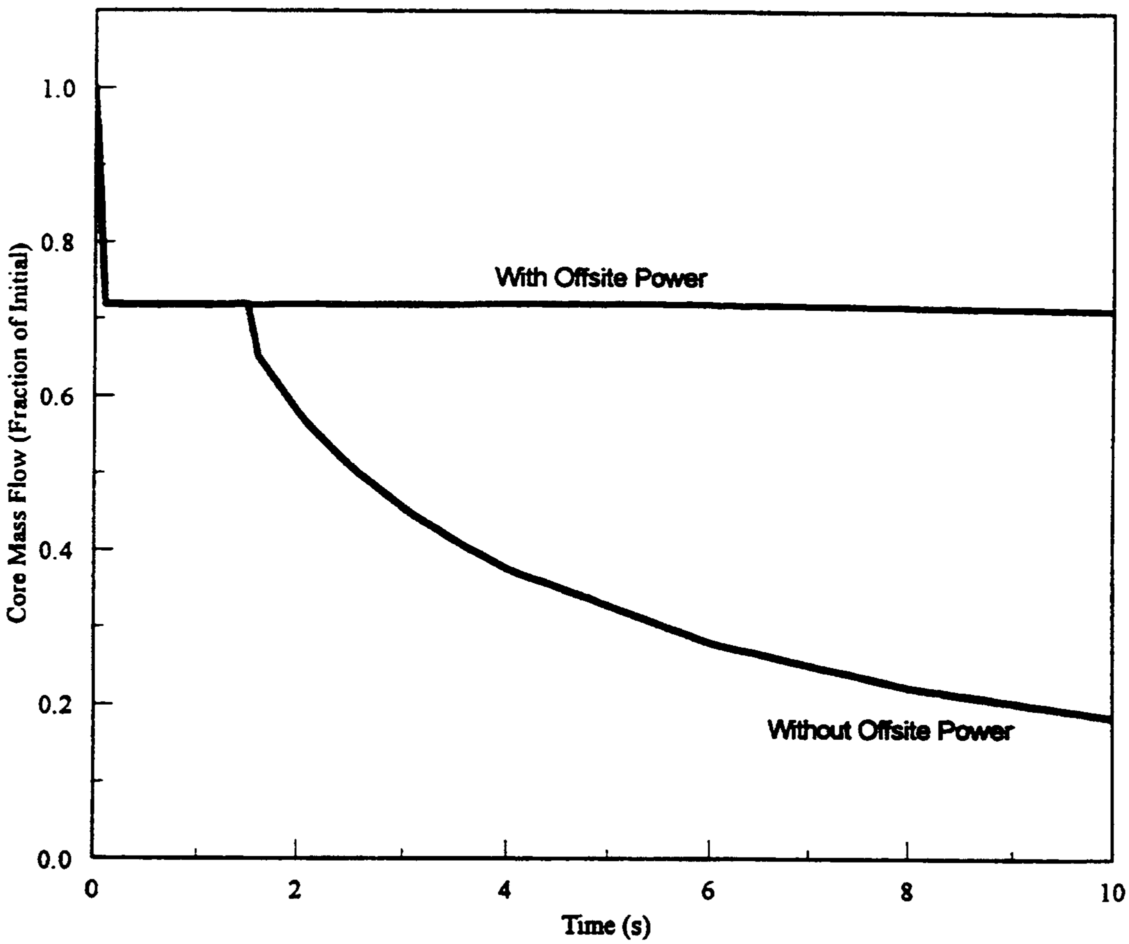

Figure 15.3.3-1

Flow Transient for Four Cold Legs in Operation, One Locked Rotor

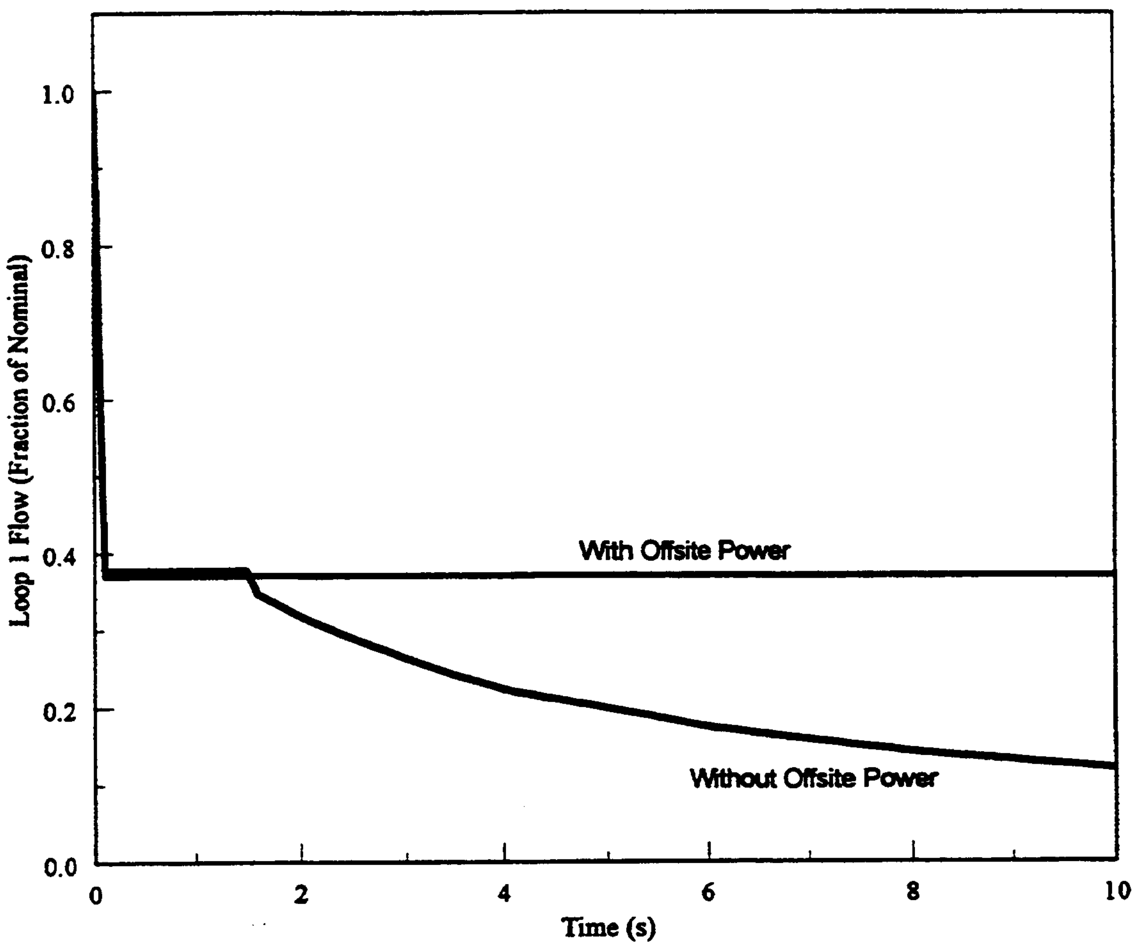

Figure 15.3.3-2

Flow Transient for Four Cold Legs in Operation, One Locked Rotor

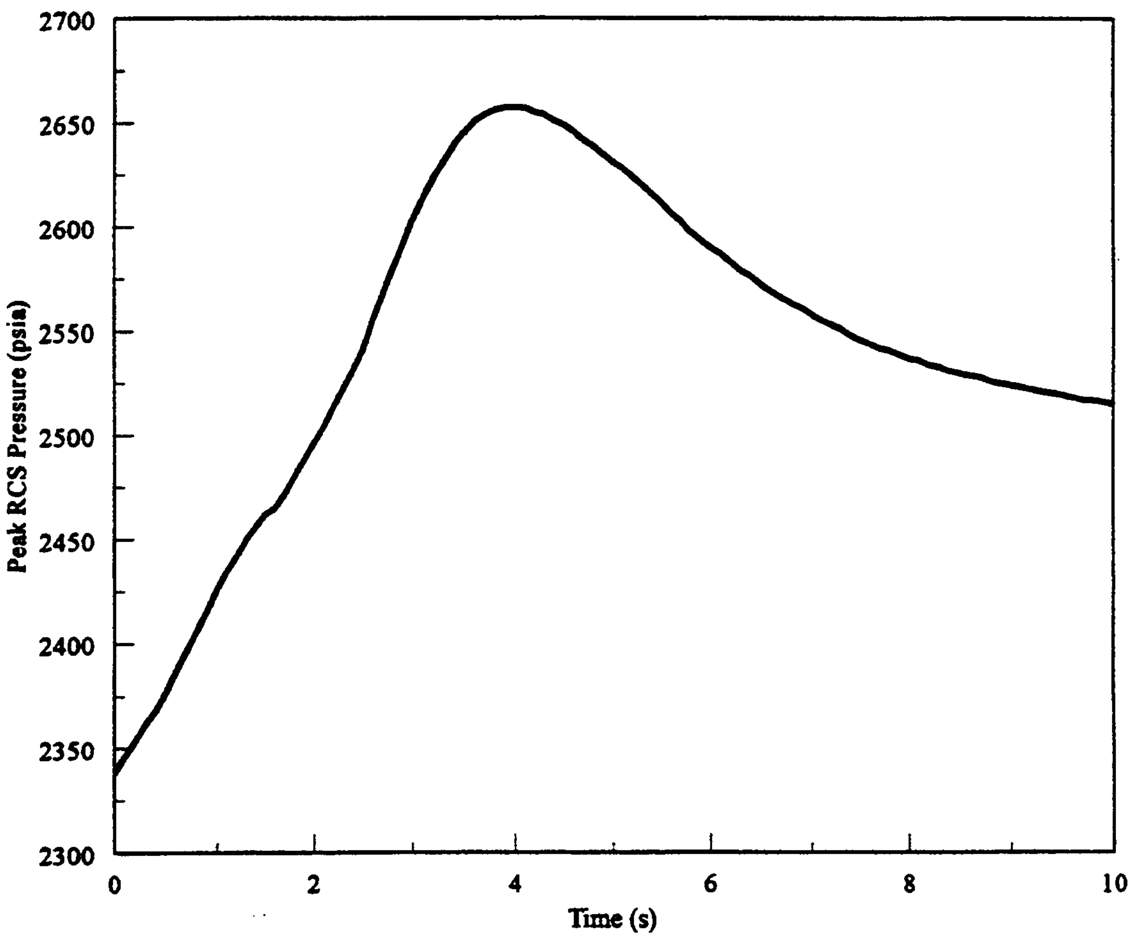

Figure 15.3.3-3

Peak Reactor Coolant Pressure for Four Cold Legs in Operation, One Locked Rotor

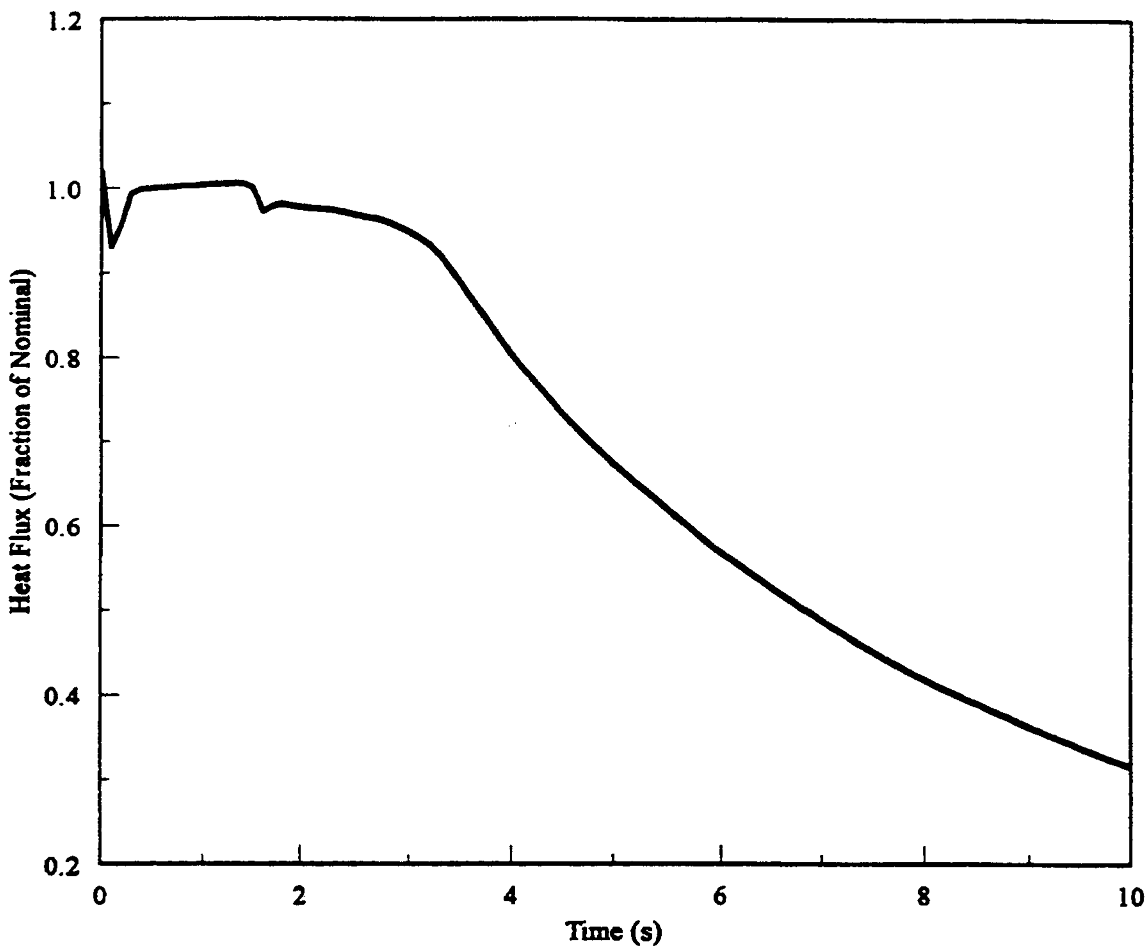

Figure 15.3.3-4

Average Channel Heat Flux Transient for Four Cold Legs in Operation, One Locked Rotor

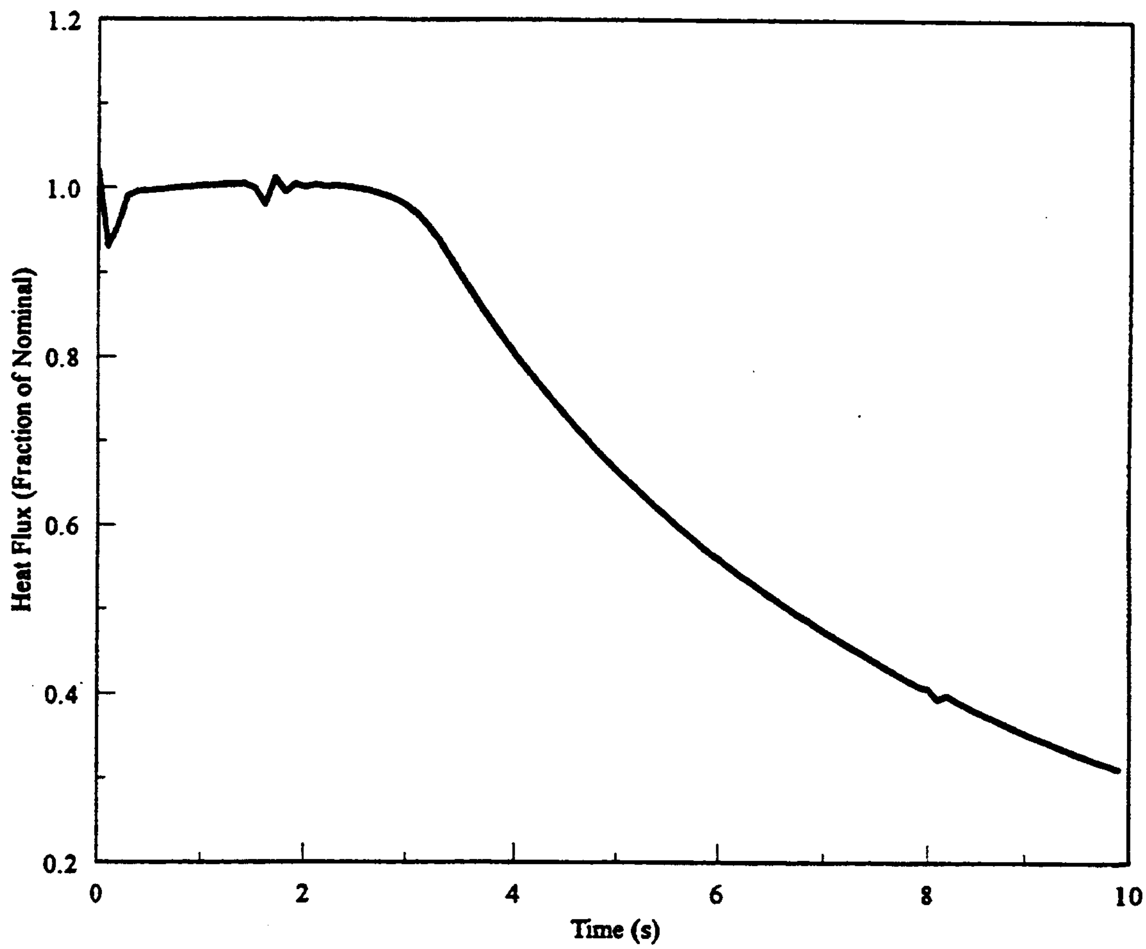

Figure 15.3.3-5

Hot Channel Heat Flux Transient for Four Cold Legs in Operation, One Locked Rotor

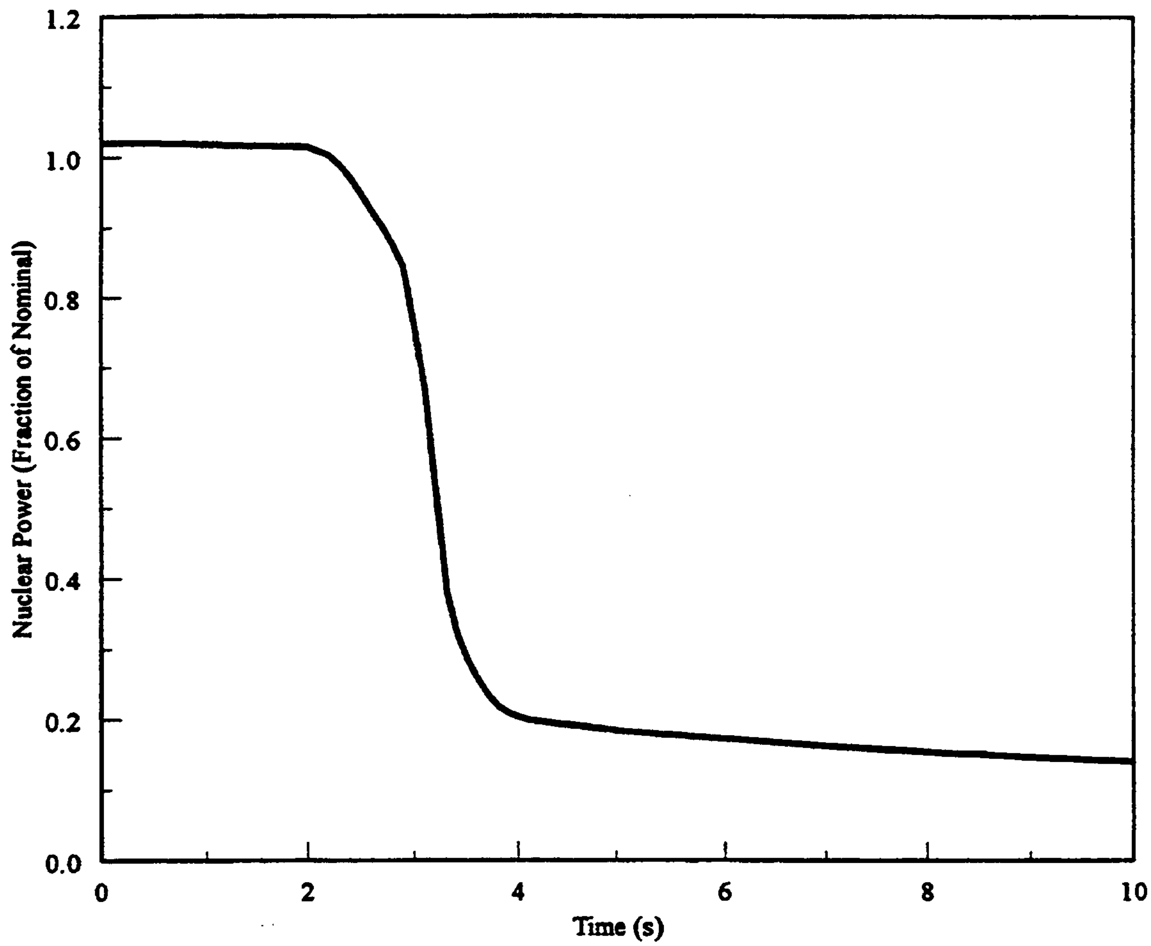

Figure 15.3.3-6

Nuclear Power Transient for Four Cold Legs in Operation, One Locked Rotor

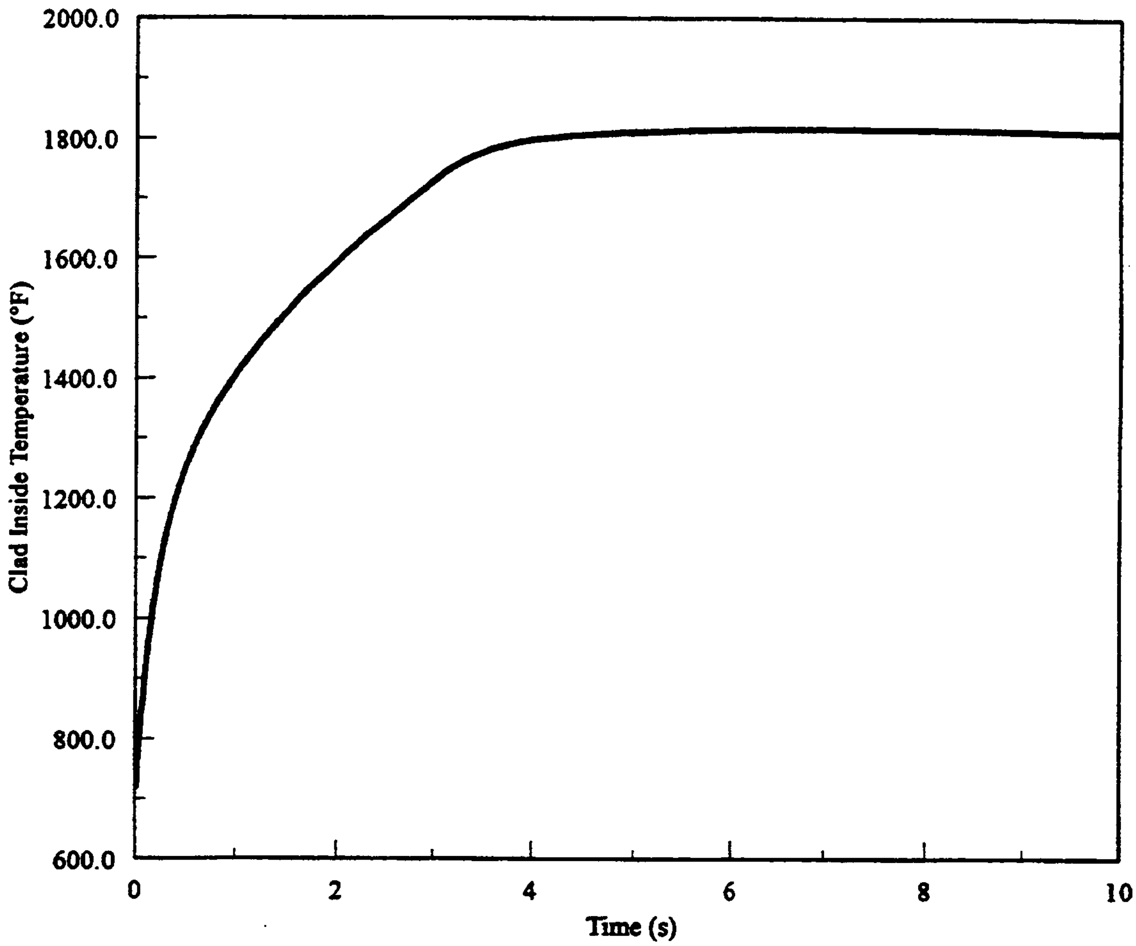

Figure 15.3.3-7

Cladding Inside Temperature Transient for Four Cold Legs in Operation, One Locked Rotor