Westinghouse Electric Company
Nuclear Power Plants
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

Document Control Desk
U S Nuclear Regulatory Commission
Two White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Direct tel: 412-374-2035
Direct fax: 724-940-8505
e-mail: ziesinrf@westinghouse.com

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Subject: CI-SRP-7.A-SRSB-03, Markup of DCD Revision 18, Chapter 15-Section 15.4.2

Westinghouse is submitting a markup of the AP1000® Design Control Document (DCD) in support of the AP1000® Design Certification Amendment Application (Docket No. 52-006). The information included in the response is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

Enclosure 1 provides a markup of Section 15.4.2 of the DCD which has been modified to satisfy the requirements identified in CI-SRP-7.A-SRSB-03.

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000® Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

R. F. Ziesing

R. F. Ziesing
Director, U.S. Licensing

/Enclosure

1. CI-SRP-7.A-SRSB-03, Markup of DCD Revision 18, Chapter 15

D063
NRO

cc:	P. Clark	-	U.S. NRC	1E
	E. McKenna	-	U.S. NRC	1E
	P. Buckberg	-	U.S. NRC	1E
	S. K. Mitra	-	U.S. NRC	1E
	T. Spink	-	TVA	1E
	P. Hastings	-	Duke Power	1E
	R. Kitchen	-	Progress Energy	1E
	A. Monroe	-	SCANA	1E
	P. Jacobs	-	Florida Power & Light	1E
	C. Pierce	-	Southern Company	1E
	E. Schmiech	-	Westinghouse	1E
	G. Zinke	-	NuStart/Entergy	1E
	R. Grumbir	-	NuStart	1E
	S. Ritterbusch	-	Westinghouse	1E
	J. DeBlasio	-	Westinghouse	1E
	J. Monahan		Westinghouse	1E

ENCLOSURE 1

CI-SRP-7.A-SRSB-03, Markup of DCD Revision 18, Chapter 15

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 AP1000)

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.

  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 VIPRE-01 (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.

- Four 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 affects 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).

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 (PMS) 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 PMS 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 divisions exceed an overpower setpoint. In particular, the power range neutron flux instrumentation provides the following reactor trip functions:

  1. Reactor trip on high power range neutron flux (high setpoint)
  2. Reactor trip on high power range positive neutron flux rate

  The latter trip protects the core when a sudden abnormal increase in power is detected in the power range neutron flux channel in two out of four PMS divisions. It provides protection against reactivity insertion rates accidents at mid and low power, and it is always active.

- Reactor trip is actuated if any two out of four ΔT power divisions 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 power divisions 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 divisions 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 divisions 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 and Reference 13.

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)

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, the effects of a possible consequential loss of offsite power during the RCCA bank withdrawal at-power event have been evaluated to not adversely impact the analysis results. This conclusion is based on a review of the time sequence associated with a consequential loss of offsite power in comparison to the reactor shutdown time for an uncontrolled RCCA bank withdrawal at-power event. The primary effect of the loss of offsite power is to cause the reactor coolant pumps (RCPs) to coast down. The PMS includes a 5.0 second minimum delay between the reactor trip and the turbine trip. In addition, a 3.0 second delay between the turbine trip and the loss of offsite power is assumed, consistent with Section 15.1.3 of NUREG-1793. Considering these delays between the time of the reactor trip and RCP coastdown due to the loss of offsite power, it is clear that the plant shutdown sequence will have passed the critical point and the control rods will have been completely inserted before the RCPs begin to coast down. Therefore, the consequential loss of offsite power does not adversely impact this uncontrolled RCCA bank withdrawal at-power analysis because the plant will be shut down well before the RCPs begin to coast down.

15.4.2.2 Analysis of Effects and Consequences

15.4.2.2.1 Method of Analysis

This transient is analyzed using the LOFTRAN (References 3 and 11) 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 to define the inputs to LOFTRAN that determine the minimum DNBR during the transient.

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.

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 lost as a consequence of turbine trip. 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 DNB design basis described in Section 4.4 is met.

The transient response for a representative slow RCCA withdrawal from full power, with offsite power lost as a consequence of turbine trip, 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 DNB design basis described in Section 4.4 is met.

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. Minimum DNBR, occurs immediately after rod motion. Two reactor trip functions provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT functions. The DNB design basis described in Section 4.4 is met.

Figures 15.4.2-14 and 15.4.2-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA bank withdrawal incidents for minimum and maximum reactivity feedback, starting at 60-percent and 10-percent power, respectively. Minimum DNBR occurs immediately after rod motion and before the loss of offsite power. 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 and for the maximum feedback cases the transient is always terminated by the overtemperature ΔT reactor trip. The DNB design basis described in Section 4.4 is met.

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 PMS action in initiating a reactor trip.

Referring to Figure 15.4.2-14, for example, it is noted that for transients initiated from 60-percent power:

A. For high reactivity insertion rates above 14 pcm/s, reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. Reactor trip is initiated by overtemperature ΔT for the whole range of reactivity insertion rates for the maximum reactivity feedback cases. For minimum reactivity feedback cases, the neutron flux level in the core rises rapidly for the higher reactivity 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 two out of four ΔT power divisions exceed an overtemperature ΔT setpoint. This trip circuit is described in Chapter 7 and Reference 13. The T_{COLD} and T_{HOT} signals, which are inputs to the overtemperature ΔT setpoint calculation, are lead-lag compensated to account for the inherent thermal and transport delays in the reactor coolant system in response to power increases.

C. For reactivity insertion rates less than approximately 40 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 14 pcm/s and below for the minimum feedback cases. For the maximum feedback case, the pressurizer safety valves open prior to reactor trip for reactivity insertion rates as high as 110 pcm/s.

D. For the minimum feedback case, at reactivity insertion rates less than approximately 14 pcm/s 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.

E. For reactivity insertion rates less than approximately 3 pcm/s for the minimum feedback cases and less than approximately 70 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 on 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.

As described in item D above, at lower reactivity insertion rates the overtemperature ΔT trip predominates and the effectiveness of the overtemperature ΔT trip increases (in terms of increased minimum DNBR) 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.

Steam generator safety valves never open before the reactor trip for transients initiated at full power. So there are not the competing effects due to the opening of the pressurizer safety valve and steam generator safety valves described in items C and E. Hence, for both the minimum and maximum feedback cases, the local minimum in the DNBR curve due to the steam generator safety valves opening is not present.

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 bank 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 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.

As discussed previously in subsection 15.4.2.1, even if a consequential loss of offsite power and the subsequent RCP coastdown were to be explicitly modeled, the minimum DNBR would be predicted to occur during the time period of the RCCA bank withdrawal at-power event prior to the time the flow coastdown begins. Therefore, the minimum DNBRs calculated in the analysis are bounding.

15.4.2.3 Conclusions

The power range neutron flux instrumentation 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.

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Table 15.4-1 (Sheet 1 of 3)

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	10.4
	Peak nuclear power occurs	10.6
	Rods begin to fall into core	11.3
	Peak heat flux occurs	12.7
	Minimum DNBR occurs	12.7
	Peak average clad temperature occurs	13.3
	Peak average fuel temperature occurs	13.4
One or more dropped RCCAs	Rods drop	0.0
	Control system initiates control bank withdrawal	0.4
	Peak nuclear power occurs	21.7
	Peak core heat flux occurs	24.2
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	6.6
	Rods begin to fall into core	7.5
	Minimum DNBR occurs	7.7
	Loss of ac power occurs	15.2
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (3 pcm/s)	0.0
	Overtemperature ΔT setpoint reached	524.4
	Rods begin to fall into core	526.4
	Minimum DNBR occurs	526.7
	Loss of ac power occurs	534.1

Table 15.4-1 (Sheet 2 of 3)

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	215.0
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	19,680
b. Manual reactor control	Initiate dilution	0.0
	Reactor trip on overtemperature ΔT due to dilution	180.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	395.0
RCCA ejection accident		
1. 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	0.93
	Peak cladding temperature occurs	2.36
	Peak heat flux occurs	2.37
	Peak fuel center temperature occurs	4.54

Table 15.4-1 (Sheet 3 of 3)

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	0.37
	Peak nuclear power occurs	0.44
	Rods begin to fall into core	1.27
	Peak heat flux occurs	1.53
	Peak cladding temperature occurs	2.55
	Peak fuel center temperature occurs	3.32
3. End of cycle, full power	Initiation of rod ejection	0.00
	Power range high neutron flux (high setting) setpoint reached	0.035
	Peak nuclear power occurs	0.14
	Rods begin to fall into core	0.94
	Peak cladding temperature occurs	2.36
	Peak heat flux occurs	2.37
	Peak fuel center temperature occurs	4.34
4. End of cycle, zero power	Initiation of rod ejection	0.00
	Power range high neutron flux (low setting) setpoint reached	0.23
	Peak nuclear power occurs	0.27
	Rods begin to fall into core	1.13
	Peak cladding temperature occurs	1.83
	Peak heat flux occurs	1.85
	Peak fuel center temperature occurs	2.94

Table 15.4-2

PARAMETERS

Assumed Dilution Flow Rates	
Mode	**Flow Rate (gal/min)**
3 through 5	175
1 through 2	200

Volume		
Mode	**Volume (ft^3)**	**Volume (gal)**
1 and 2	8126	60,786
3	7539.8	56,401
4	7539.8	56,401
5	2592.2	19,391

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[3]	0	102[3]
Ejected rod worth (%Δk)	0.65	0.37	0.75	0.30
Delayed neutron fraction (%)	0.49	0.49	0.44	0.44
Feedback reactivity weighting	2.155	1.22	2.9	1.35
Trip reactivity (%Δk)	2.0	4.0	2.0	4.0
F_q before rod ejection	–	2.6	–	2.6
F_q after rod ejection	12.0	4.9	19.6	6.0
Number of operational pumps	2	4	2	4
Maximum fuel pellet average temperature (°F)	2573	4118	2848	3926
Maximum fuel center temperature (°F)	3018	4974	3263	4871
Maximum cladding average temperature (°F)	1907	2265	2122	2151
Maximum fuel stored energy (cal/g)	104	181	117	170
Percent of fuel melted at hot spot	0	<10	0	<10

Notes:
1. HZP – Hot zero power
2. HFP – Hot full power
3. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.

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 60 μCi/g of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in failed/melted fuel)	1.65
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fission product gap fractions	
Iodines and noble gases	0.1
Alkali metals	0.12
Core melting	
– Fraction of core melting	0.0025
– Fraction of activity released	
Iodines and alkali metals	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)	3.7 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)	104.3[(a)]
– Leak flashing fraction	0.04[(b)]
– Secondary coolant mass (lb)	6.06 E+05
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05
– Partition coefficient in steam generators	
• Iodine	0.01
• Alkali metals	0.001
Containment leakage release path	
– Containment leak rate (% per day)	
• 0-24 hr	0.10
• >24 hr	0.05
– Airborne activity removal coefficients (hr^{-1})	
• Elemental iodine	1.7[(c)]
• Organic iodine	0
• Particulate iodine or alkali metals	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	3.1

Notes:

a. Equivalent to 300 gpd cooled liquid at 62.4 lb/ft^3.

b. No credit for iodine partitioning is taken for flashed leakage.

c. From Appendix 15B.

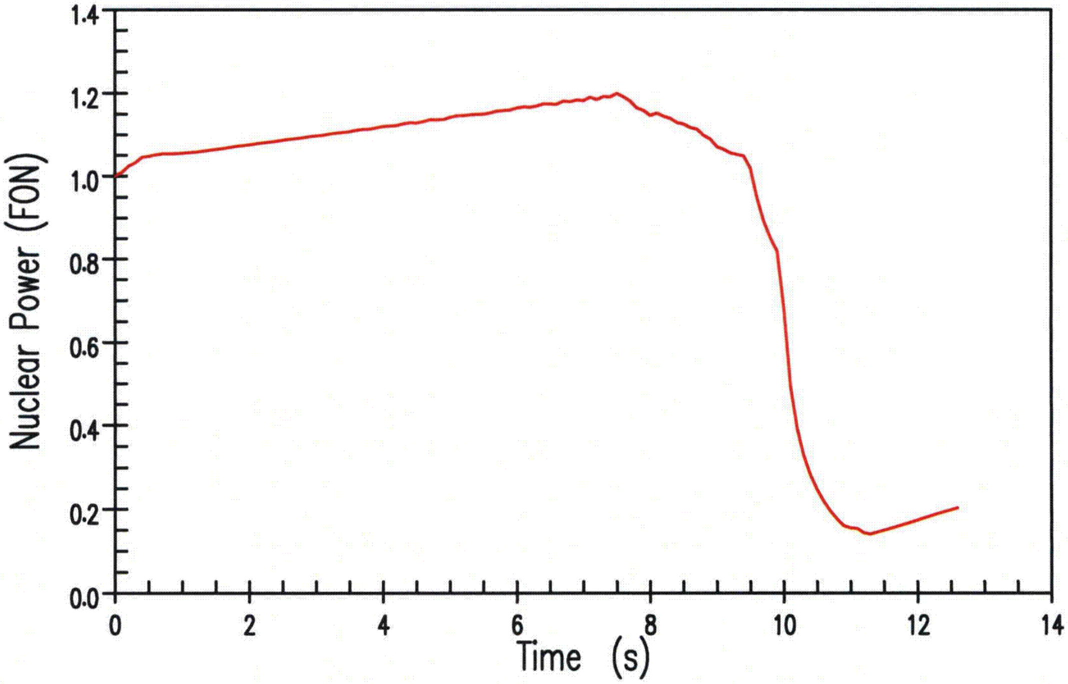

Figure 15.4.2-1

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

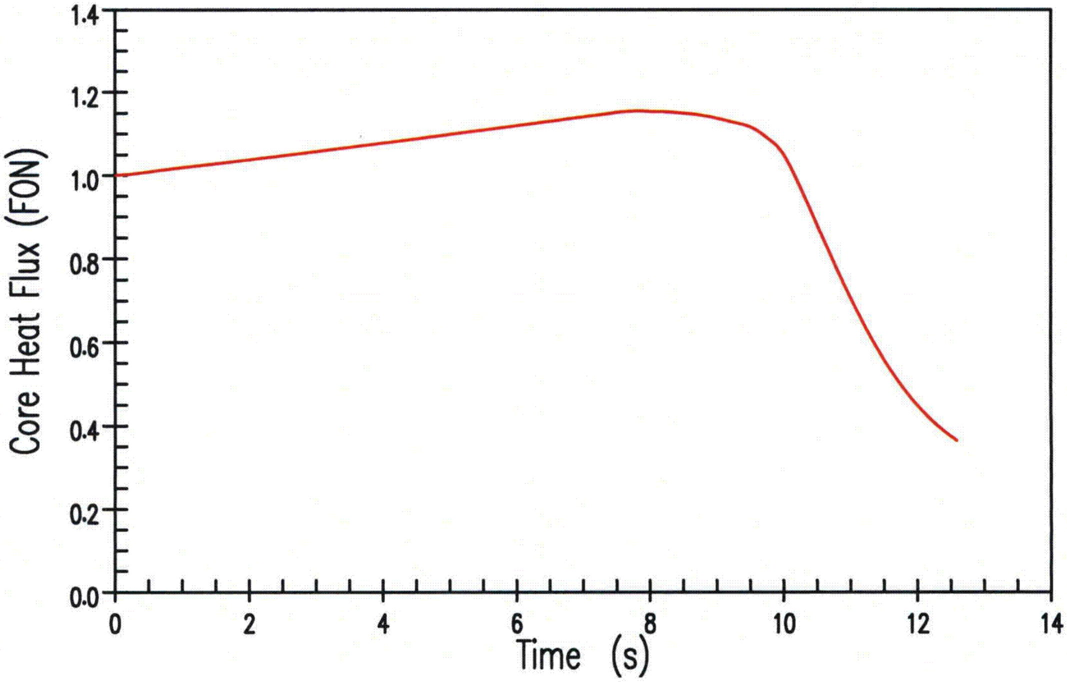

Figure 15.4.2-2

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

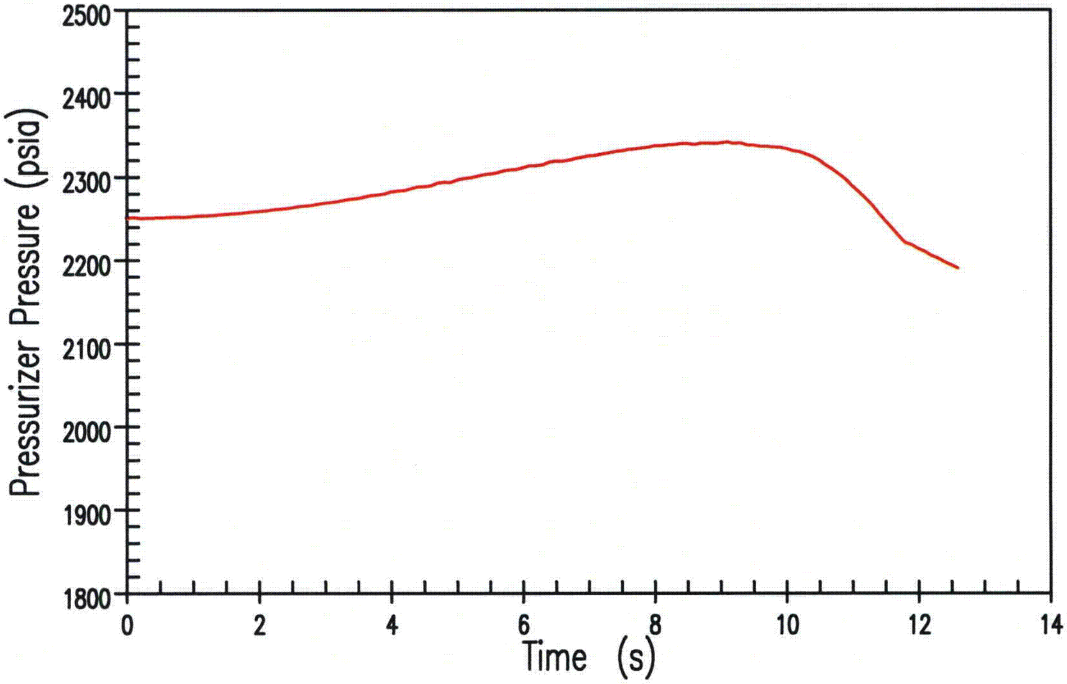

Figure 15.4.2-3

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

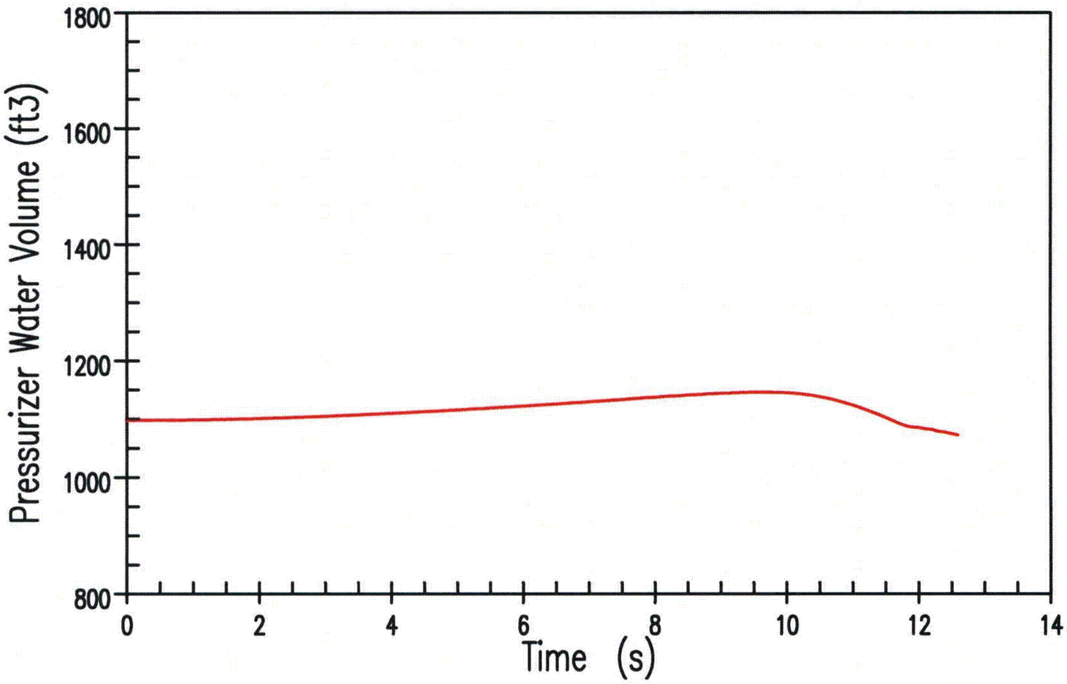

Figure 15.4.2-4

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

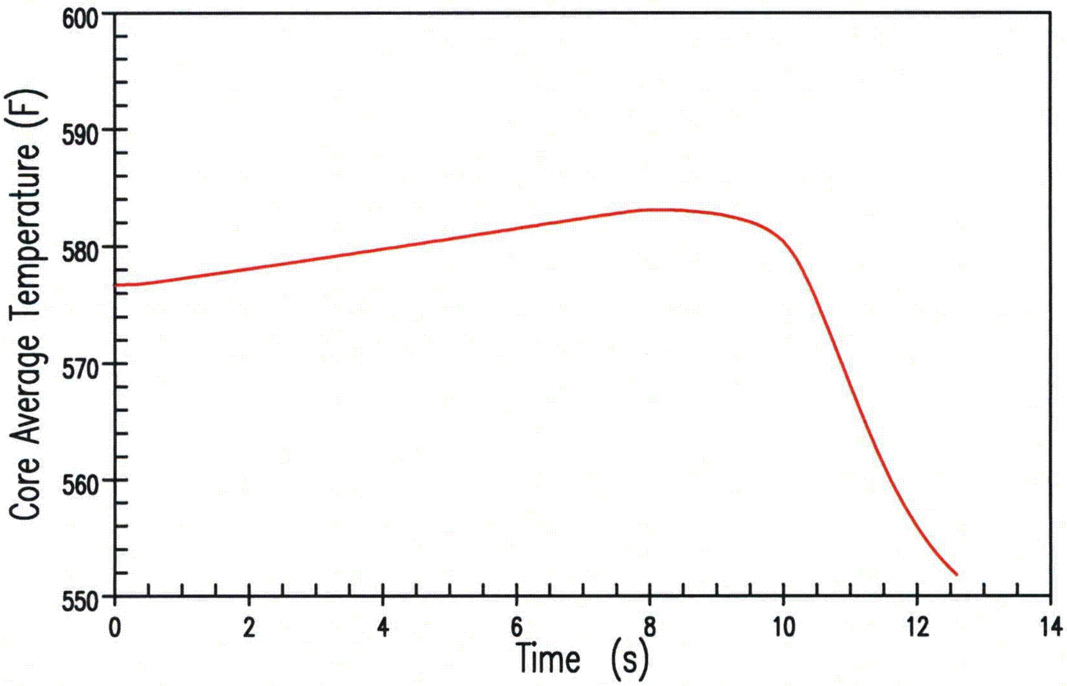

Figure 15.4.2-5

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

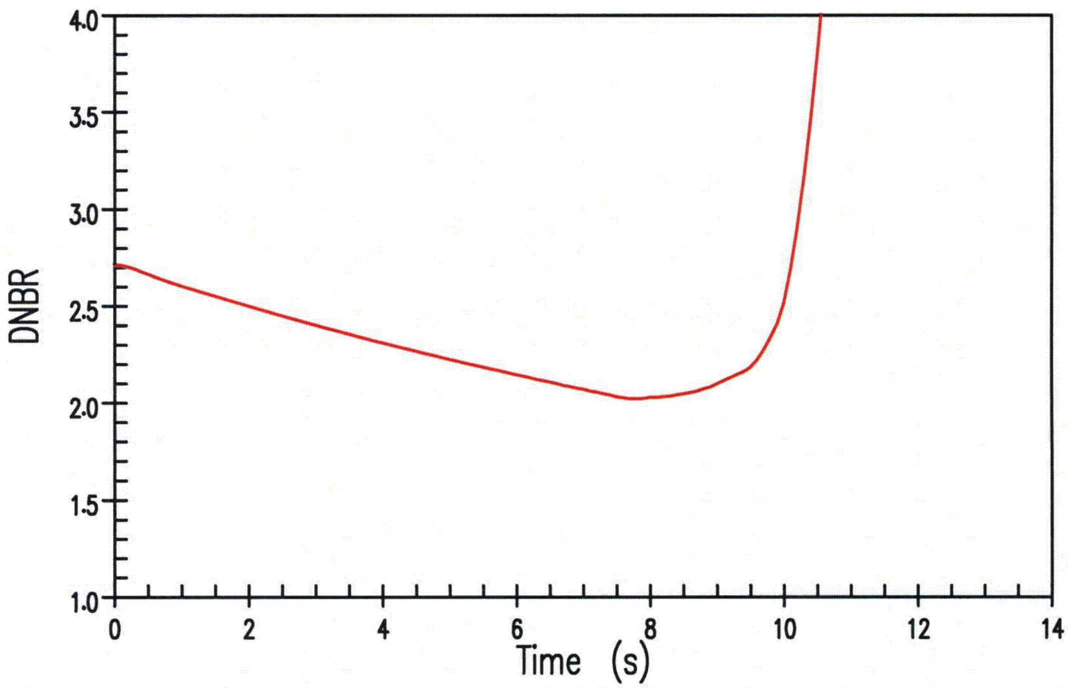

Figure 15.4.2-6

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

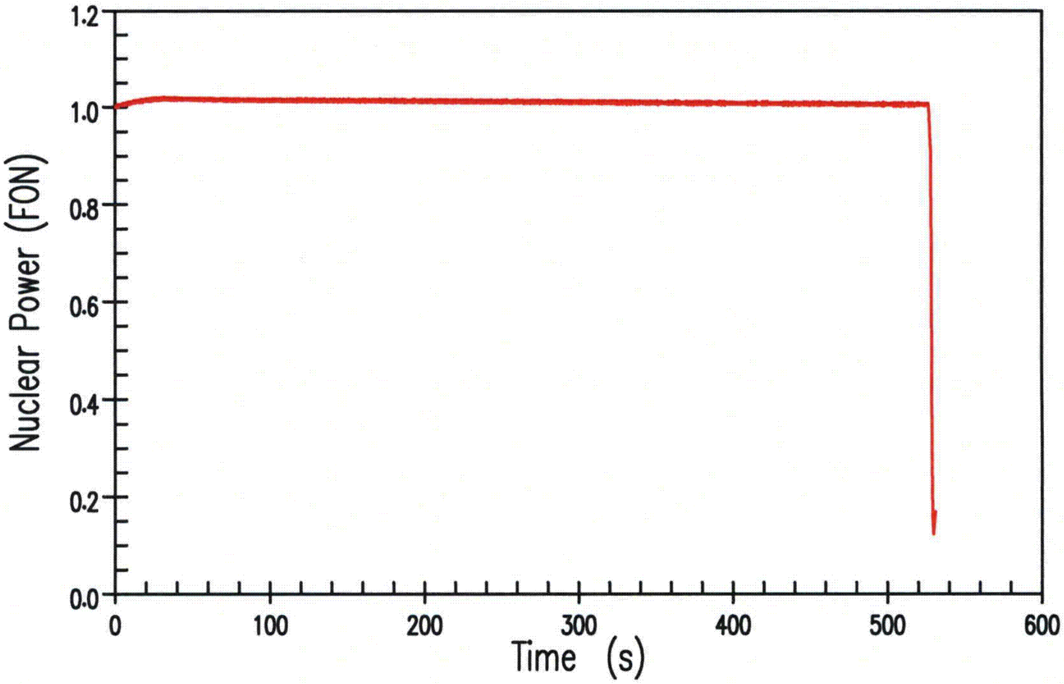

Figure 15.4.2-7

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

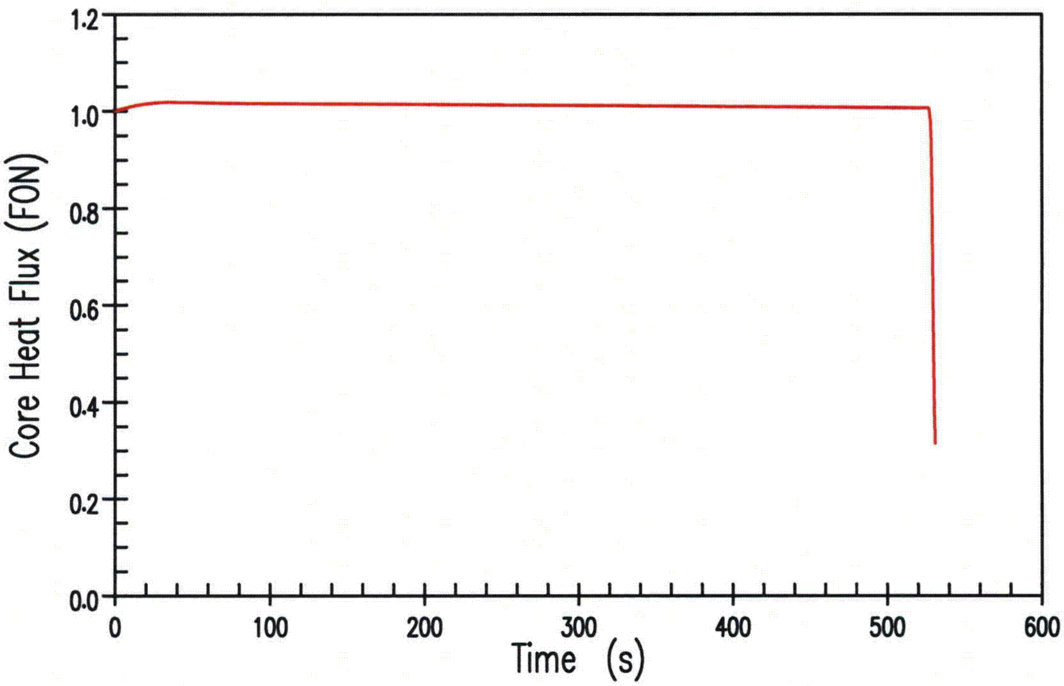

Figure 15.4.2-8

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

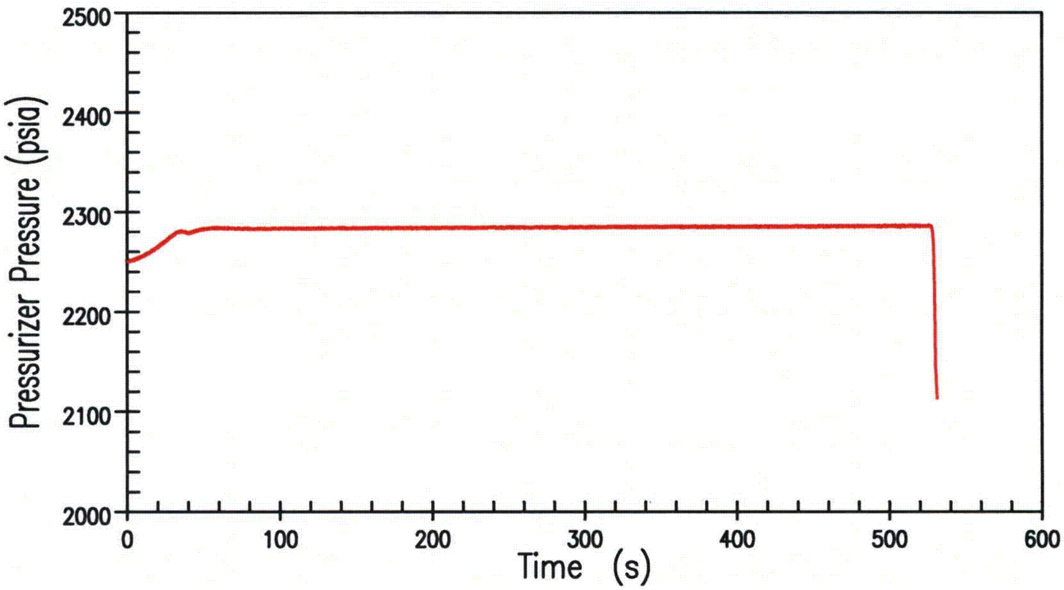

Figure 15.4.2-9

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

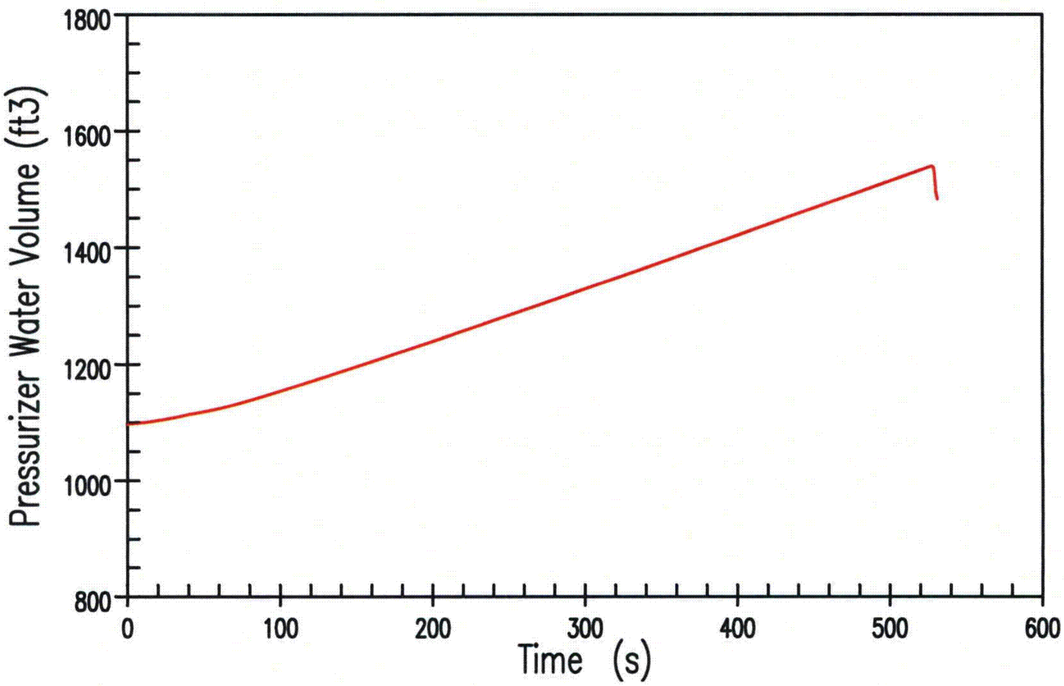

Figure 15.4.2-10

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

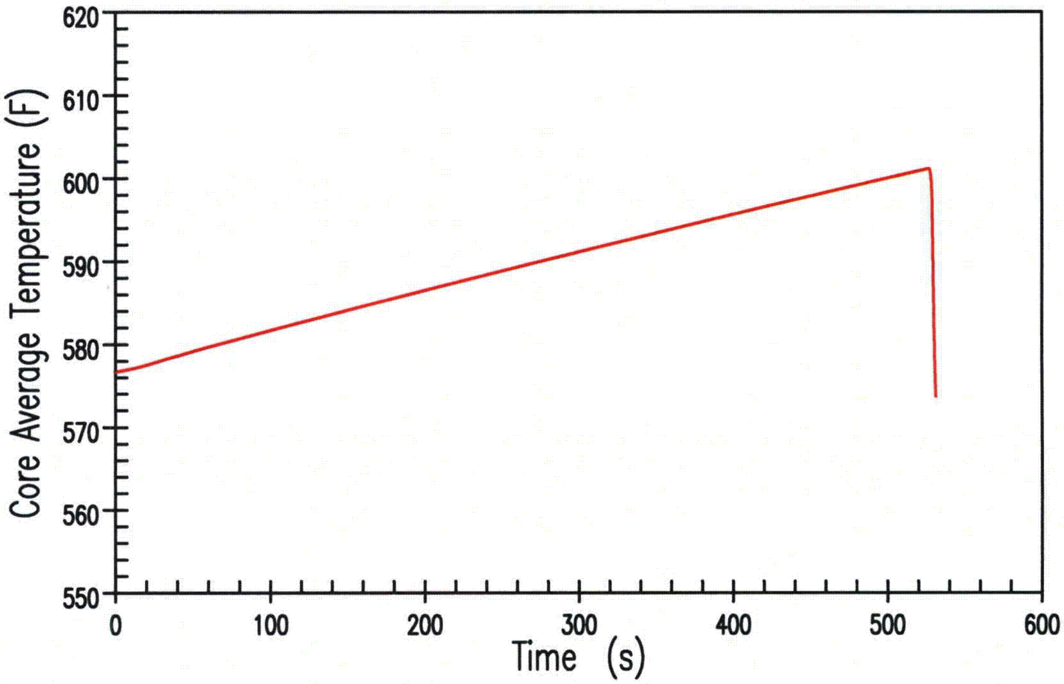

Figure 15.4.2-11

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

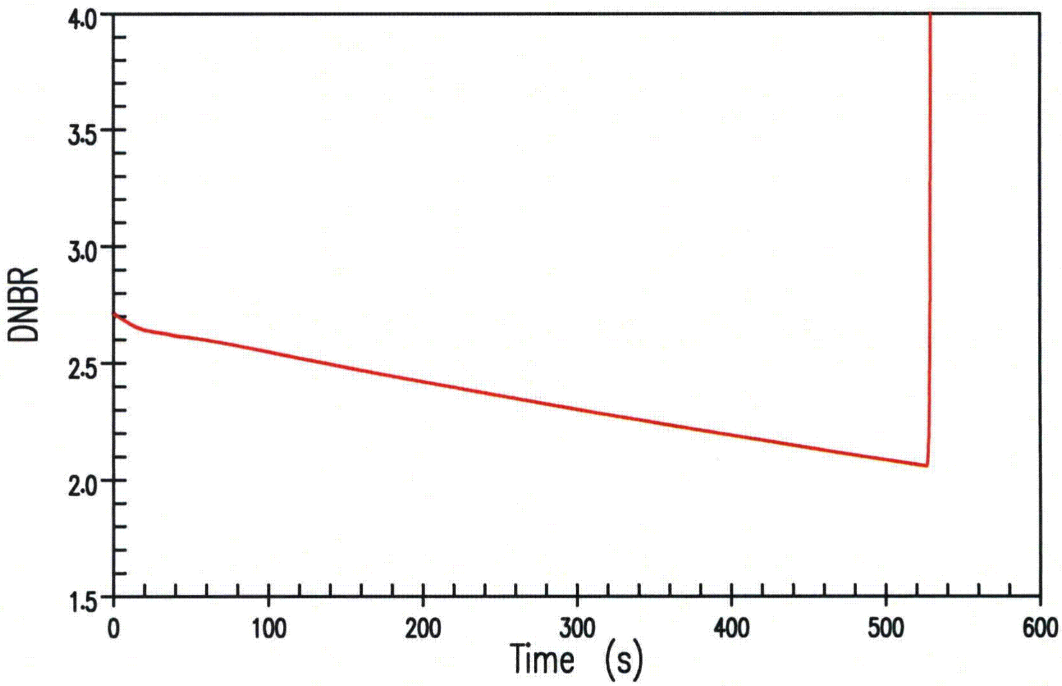

Figure 15.4.2-12

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

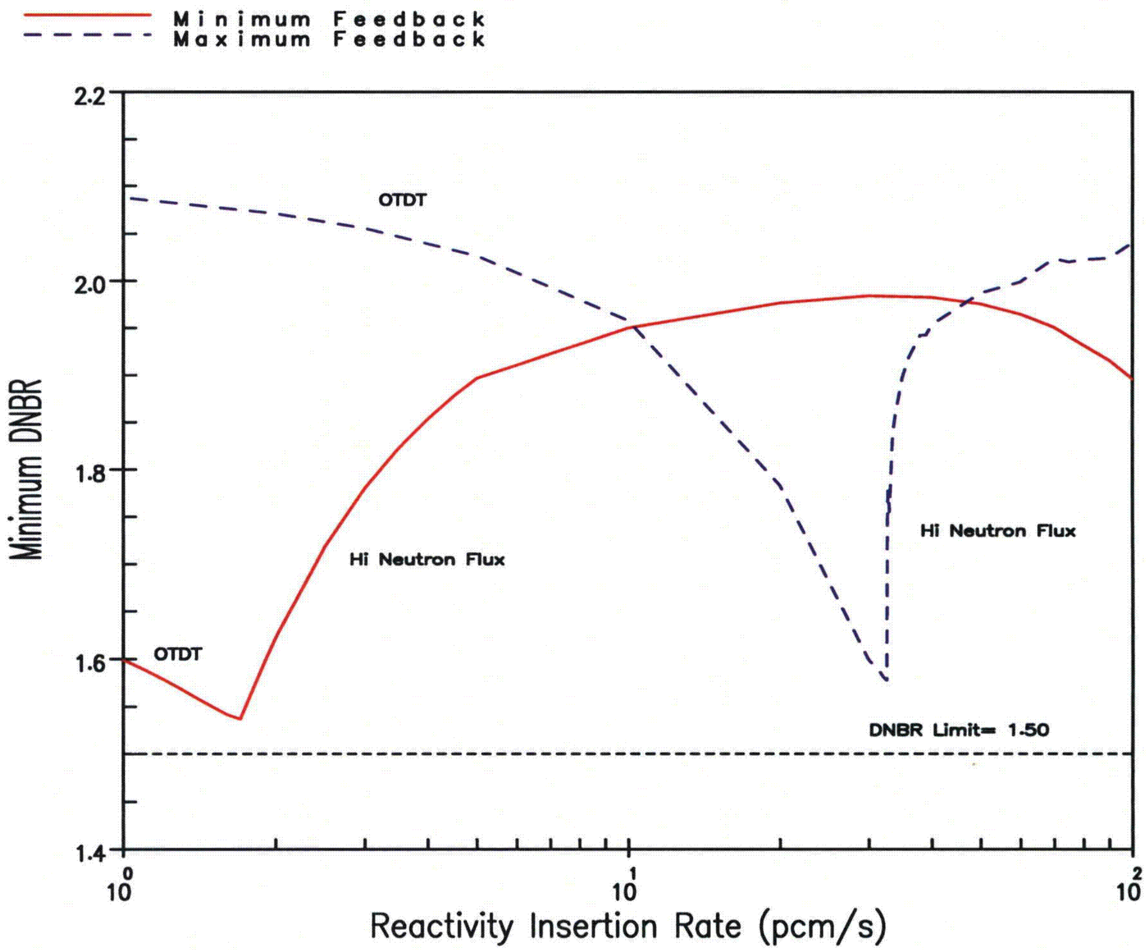

Figure 15.4.2-13

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

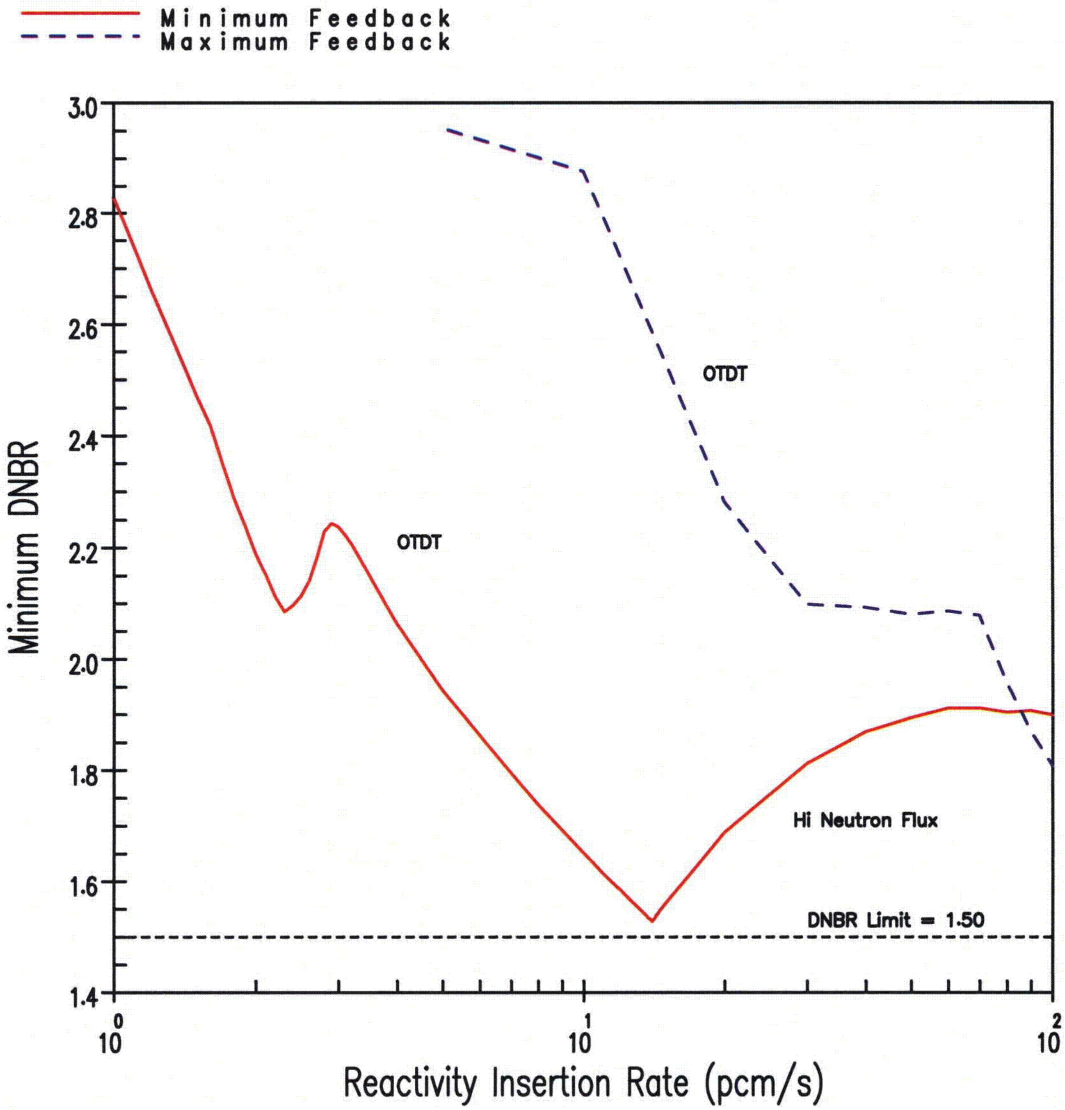

Figure 15.4.2-14

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

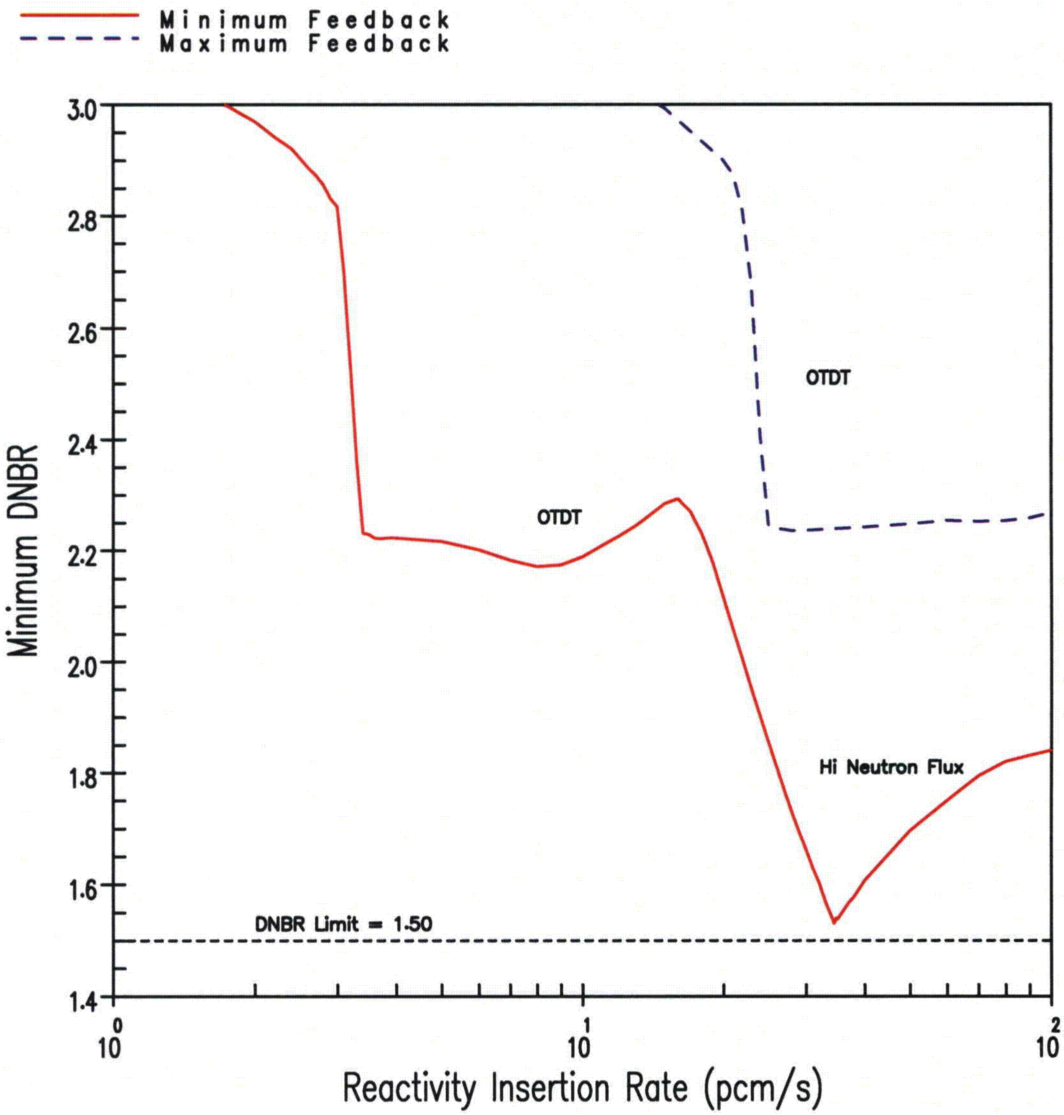

Figure 15.4.2-15

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

Figures 15.4.2-16 and 15.4.2-17 not used.