123 Main Street
White Plains, New York 10601
914-681-6840
914-287-3309 (FAX)

NewYorkPower Authority

William J. Cahill, Jr.
Chief Nuclear Officer

December 23, 1996
IPN-96-128

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Indian Point 3 Nuclear Power Plant**
Docket No. 50-286
Proposed Technical Specification Changes
Associated With the Upgrade to VANTAGE + Fuel

Reference: Westinghouse Report, "Vantage + Fuel Upgrade - Reload Transition Safety Report for the Indian Point Unit 3 Nuclear Station, Revision 2 ," December 1996.

Dear Sir:

This application for amendment seeks to revise several sections of Appendix A of the Indian Point 3 Technical Specifications to accommodate the transfer from VANTAGE 5 (without intermediate flow mixers) fuel to VANTAGE + fuel. The referenced document evaluates the transition to this new fuel and confirms its acceptable use. A summary of this document is provided in the Safety Evaluation (Attachment II) and a copy of the entire document is provided as Attachment III.

Enclosed for filing is the signed original of a document entitled, "Application for Amendment to Operating License," together with Attachments I and II, comprising a statement of the proposed changes to the Technical Specifications and the associated Safety Evaluation. Attachment IV contains the commitments made by the Authority in this submittal. The changes proposed by this Technical Specification amendment are required to begin refueling for Cycle 10. Therefore, approval of this amendment is requested by April 1, 1997.

In accordance with 10 CFR 50.91, a copy of this application and the associated attachments are being submitted to the designated New York State official.

A001 1/1

If you have any questions, please contact Ms. C. D. Faison.

Very truly yours,

William J. Cahill, Jr.
Chief Nuclear Officer

Attachments: As stated

cc: Regional Administrator
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. F. William Valentino, President
New York State Energy, Research,
and Development Authority
2 Rockefeller Plaza
Albany, NY 12223-1253

Mr. George F. Wunder, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop 14B2
Washington, DC 20555

BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
POWER AUTHORITY OF THE STATE OF NEW YORK	)	Docket No. 50-286
Indian Point 3 Nuclear Power Plant	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the Nuclear Regulatory Commission (NRC), the Power Authority of the State of New York, as holder of Facility Operating License No. DPR-64, hereby applies for an Amendment to the Technical Specifications contained in Appendix A of the license.

This application for amendment seeks to revise several sections of Appendix A of the Indian Point 3 Technical Specifications to accommodate the transfer from VANTAGE 5 (without intermediate flow mixers) fuel to VANTAGE + fuel.

The proposed changes to the Technical Specifications are included as Attachment I to this application. The Safety Evaluation is included as Attachment II.

POWER AUTHORITY OF THE
STATE OF NEW YORK

By
William J. Cahill, Jr.
Chief Nuclear Officer

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and Sworn to before me
this 23rd day of December 1996

Geraldine Strand
Notary Public

GERALDINE STRAND
Notary Public, State of New York
No. 4991272
Qualified in Westchester County
Commission Expires Jan. 27, 19 98

50-286 NYPA INDIAN POINT 3

PROPOSED TECH SPECS CHANGES ASSOCIATED
WITH THE UPGRADE TO VANTAGE + FUEL

REC'D W/LTR DTD 12/23/96....9701020292

ATTACHMENT I TO IPN-96-128

PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH THE UPGRADE TO VANTAGE + FUEL

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

2.0 Safety Limits and Limiting Safety System Settings

2.1 Safety Limits, Reactor Core

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure and coolant temperature during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 for four-loop operation. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System vessel inlet temperature and power level is at any time above the appropriate pressure line.

Basis

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. The safety limits represent a design requirement for establishing the trip setpoints identified in Technical Specification 2.3. Technical Specification 3.1.H, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," provide more restrictive limits to ensure that the safety limits are not exceeded.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant Temperature and Pressure have been related to DNB through correlations which have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: There must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I (normal operation and operational transients) and Condition II (events of moderate frequency) events is greater than or equal to the Design DNBR limit.

In meeting the DNB criterion, uncertainties in operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analyses limit DNBR values is used to offset known DNBR penalites (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of thermal power, Reactor Coolant System pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The calculation of these limits includes:

1. $F_{\Delta H}^{RTP} = F_{\Delta H}^{N}$ limit at Rated Thermal Power (RTP) specified in the COLR.

2. an equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24%, [2]

3. a reactor coolant system total flow rate of greater than or equal to 385,400 gpm as measured at the plant,

4. a reference cosine with a peak of 1.55 for axial power shape.

Figure 2.1-1 includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$$F_{\Delta H}^{N} \leq F_{\Delta H}^{RTP} \ (1 + PF_{\Delta H} \ (1-P))$$

Where P is the fraction of Rated Thermal Power.

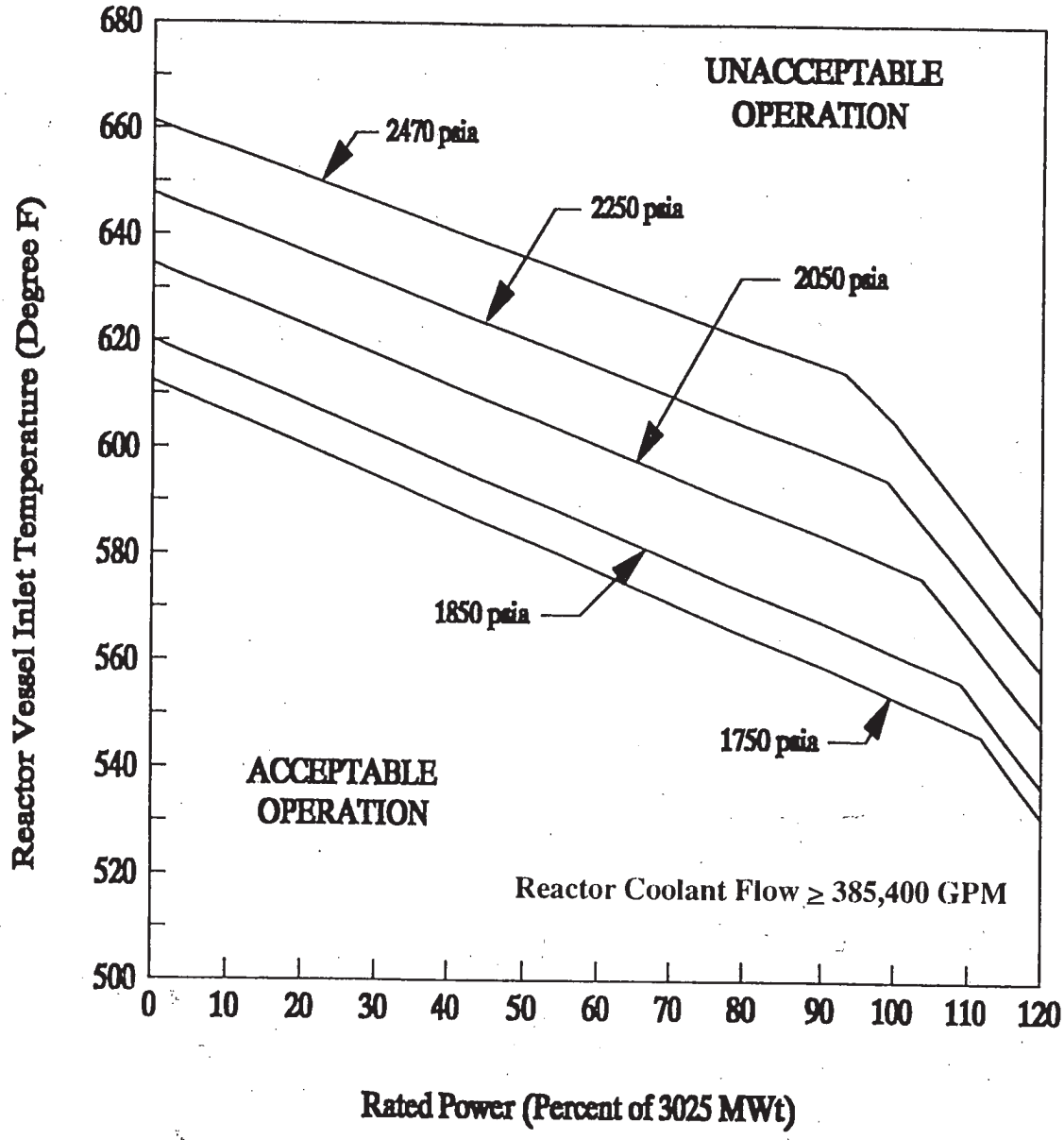
$F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at Rated Thermal Power specified in the COLR, and $PF_{\Delta H}$ is the Power Factor Multiplier specified in the COLR.

When flow or $F_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limits presented. A 2.9% measurement uncertainty on Flow and a 4% measurement uncertainty of $F_{\Delta H}$ have already been included in the above limits.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit (specified in the COLR) assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

REACTOR CORE SAFETY LIMITS

This curve does not provide allowable limits for normal operation.
(See Technical Specification 3.1.H for DNB limits)



100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt

Pressures and temperatures do not include allowance for instrument error.

FIGURE 2.1-1

Amendment No. 86, 170,

ΔT_o $\leq$ Measured full power ΔT for the channel being calibrated, °F

T_{avg} = Average Temperature for the channel being calibrated, °F (input from instrument racks)

T' = Measured full power T_{avg} for the channel being calibrated, °F

P = Pressurizer pressure, psig (input from instrument racks)

P' = 2235 psig (i.e., nominal pressurizer pressure at rated power)

K_1 $\leq$ 1.20

K_2 = 0.0273

K_3 = 0.0013

K_1 is a constant which defines the overtemperature ΔT trip margin during steady state operation if the temperature, pressure, and $f(\Delta I)$ terms are zero.

K_2 is a constant which defines the dependence of the overtemperature ΔT setpoint to T_{avg}.

K_3 is a constant which defines the dependence of the overtemperature ΔT setpoint to pressurizer pressure.

ΔI = $q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power.

$f(\Delta I)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are defined above such that:

(a) for $q_t - q_b$ below 6 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 3.12 percent of rated power.

(5) Overpower ΔT

$$\Delta T \leq \Delta T_o \left(K_4 - K_5 \frac{dTavg}{dt} - K_6(T_{avg} - T')\right)$$

where:

ΔT_o $\leq$ measured full power ΔT for the channel being calibrated, °F

T_{avg} = measured average temperature for the channel being calibrated, °F (input from instrument racks)

T' = measured full power T_{avg} for the channel being calibrated, °F (can be set no higher than 570.3 °F)

K_4 ≤ 1.073

K_5 = 0 for decreasing average temperature

≥ 0.175 sec/°F for increasing average temperature

K_6 = 0 for $T \leq T'$

≥ 0.00134 for $T > T'$

K_4 is a constant which defines the overpower ΔT trip margin during steady state operation if the temperature term is zero.

K_5 is a constant determined by dynamic considerations to compensate for piping delays from the core to the loop temperature detectors; it represents the combination of the equipment static gain setting and the time constant setting.

K_6 is a constant which defines the dependence of the overpower ΔT setpoint to T_{avg}.

$\frac{dTavg}{dt}$ = rate of change of T_{avg}

(6) Low reactor coolant loop flow:

(a) $\geq$ 90% of normal indicated loop flow
(b) Low reactor coolant pump frequency - $\geq$ 57.2 cps

(7) Undervoltage - $\geq$ 70% of normal voltage

Amendment No. 48, 61, 86, 140,

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is backed up by the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. Its setting limit is consistent with the value assumed in the loss of coolant analysis. [4]

The overtemperature Delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3.5 seconds) [5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors [2], is always below the core safety limit as shown on Figure 2.1-1. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced. [6][7] The values of the constants K_1, K_2, and K_3 are determined during the design of the core for operation with all reactor loops in service. The value for K_1 includes an allowance for instrument channel uncertainty, and therefore is a nominal trip setpoint. K_2 and K_3 are analytical limits, and do not require an allowance for instrument channel uncertainty. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and the applicable safety limit DNBR will not be violated.

The overpower Delta-T reactor trip prevents power density anywhere in the core from exceeding 118% of design power density, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation (via the overall gain in the rate controller) for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement. [2] The values of the constants K_4, K_5, and K_6 are determined during the design of the core and the reactor protection system. The value for K_4 includes an allowance for instrument channel uncertainty, and therefore is a nominal trip setpoint. K_5 and K_6 are analytical limits, and do not require an allowance for instrument channel uncertainty.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. Fuel temperature decreases due to cladding creepdown with burnup and consequential reduction of pellet-cladding gap. Thus overpower limits become less restrictive as fuel burnup proceeds.

The T' values represent the measured full power T_{avg} for the overtemperature and overpower Delta-T equations. T' must correspond to the indicated full power T_{avg}, and may only be set as high as 570.3°F if the plant operates at the design full power T_{avg}. Reducing T' for a lower (than design) full power T_{avg} assures that the overtemperature and overpower delta-T setpoint are decreased for any increase in T_{avg} above the indicated loop full power T_{avg}.

pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure and temperature limits are consistent with operation within the nominal operational envelope. A lower pressure will cause the reactor core to approach DNB limits. A higher RCS average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit bounds that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

Applicable Safety Analyses
The requirements of this Specification represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this Specification will result in meeting the applicable DNBR criteria. Changes to the unit that could affect these parameters must be assessed for their effect on the DNBR criteria.

Specification
Specifications 3.1.H.1 and 3.1.H.2 specify limits on the monitored process variables (pressurizer pressure, RCS average temperature, and RCS total flow rate) to ensure that the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The RCS total flow rate limit of 385,400 gpm allows for a measurement uncertainty of 2.9% associated with the performance of Reactor Coolant System Flow Calculation required by Technical Specification 4.3.B. Because the flow instrumentation provides flow indication based on a percentage of full flow, the 385,400 gpm is converted into a percentage of full flow to accomodate the verification that RCS total flow is within limits during channel checks.

The pressurizer pressure limit of 2205 psig allows for measurement uncertainty and instrument error. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit.

The limit on maximum indicated RCS average temperature provides assurance that RCS temperatures are maintained within the normal steady state envelope of operation assumed in the safety analyses performed to support the Vantage + fuel reloads with asymmetric tube

plugging among steam generators. A maximum full power T_{cold} of 547.7°F (including control deadband and measurement uncertainties) was assumed in these safety analyses. A T_{avg} of 578.3°F assures that a T_{cold} of 547.7°F is not exceeded at a measured flow of ≥ 385,400 gpm when considering asymmetric tube plugging among steam generators for DNB considerations. However, T_{avg} will be controlled to a maximum indicated T_{avg} of 571.5°F which assures consistency with analyses for post-LOCA containment integrity.

Applicability

During the POWER OPERATION CONDITION, the limits on pressurizer pressure and RCS coolant average temperature must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. For the same reason, during the POWER OPERATION CONDITION with four reactor coolant pumps running, the limit on RCS flow rate must be maintained. In all other operating conditions, the power level is low enough that DNB is not a concern.

Specification 3.1.H.3 indicates that the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counter productive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in Safety Limit 2.1, "Safety Limits, Reactor Core." Those limits are less restrictive than the limits of this specification but violation of a Safety Limit merits stricter, more severe required action. Should a violation of Specification 3.1.H.1 occur, the operator must check whether or not a Safety Limit has been exceeded.

Actions

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within specification limits, action must be taken to restore the parameter(s).

The 2 hour completion time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience for Westinghouse plants.

If the required action of Specification 3.1.H.4 is not met within the associated completion time, the plant must be brought to a mode in which Specification 3.1.H.1 does not apply. To achieve this status, the plant must be brought to at least the HOT SHUTDOWN CONDITION within 6 hours. The reduced power condition eliminates the potential for violation of the accident analysis bounds. The completion time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

8. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

9. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 145 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 421 hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal absorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications. In the event that more than one region of fuel (72 assemblies) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before the interval of 267 hours has elapsed after shutdown.

10. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of the reactor pressure vessel flange.

11. Hoists or cranes utilized in handling irradiated fuel shall be dead-load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the deadload test and prior to fuel handling. A test of interlocks and overload cutoff devices on the manipulator shall also be performed.

12. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

13. To ensure redundant decay heat removal capability, at least two of the following requirements shall be met:

The waiting time of 267 hours required following plant shutdown before unloading more than one region of fuel from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR. The calculations confirming this are based on an inlet river temperature of 87.8°F, service water flow to the component cooling heat exchangers of 7000 gpm (FSAR) and component cooling flow to the Spent Fuel Pit heat exchanger of 2800 gpm (FSAR).

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite dose to within acceptable limits in the event of a fuel-handling accident. The fuel storage building emergency ventilation system must be operable whenever irradiated fuel is being moved. However, if the irradiated fuel has had a continuous 45 day decay period, the fuel storage building emergency ventilation system is not technically necessary, even though the system is required to be operable during all fuel handling operations. Fuel Storage Building isolation is actuated upon receipt of a signal from the area high activity alarm or by manual operation. The emergency ventilation bypass assembly is manually isolated, using manual isolation devices, prior to movement of any irradiated fuel. This ensures that all air flow is directed through the emergency ventilation HEPA filters and charcoal adsorbers. The ventilation system is tested prior to all fuel handling activities to ensure the proper operation of the filtration system.

When fuel in the reactor is moved before the reactor has been subcritical for at least 421 hours, the limitations on the containment vent and purge system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere.

The limit to have at least two means of decay heat removal operable ensures that a single failure of the operating RHR System will not result in a total loss of decay heat removal capability. With the reactor head removed and 23 feet of water above the vessel flange, a large heat sink is available for core cooling. Thus, in the event of a single component failure, adequate time is provided to initiate diverse methods to cool the core.

The minimum spent fuel pit boron concentration and the restriction of the movement of the spent fuel cask over irradiated fuel were specified in order to minimize the consequences of an unlikely sideways cask drop.

F_Q^E Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

An upper bound envelope of F_Q^{RTP} specified in the COLR times the normalized peaking factor axial dependence of K(Z) specified in the COLR has been determined consistent with Appendix K criteria and is satisfied for OFA transition mixed cores [3] by all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analysis based on this upper bound normalized envelope, K(Z), specified in the COLR demonstrates that the peak clad temperature is below the peak clad temperature limit of 2200°F.[2]

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F_{\Delta H}^N$ there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq F_{\Delta H}^{RTP}/1.04$, where $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^N$ limit at Rated Thermal Power specified in the COLR. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape

(e.g. rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F_{\Delta H}^N$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available. When a measurement of $F_{\Delta H}^N$ is taken, no additional allowances are necessary prior to comparison with the limit of section 3.10.2. A measurement uncertainty of 4% has been allowed for in determination of the design DNBR value.

Measurements of the hot channel factors are required as part of startup physics tests, at least each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design basis including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 12 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

4.13 Containment Vent and Purge System

Applicability

This specification applies to the surveillance requirements of the containment vent and purge system during normal operations and when reactor fuel is anticipated to be moved before the reactor has been subcritical for at least 421 hours.

Objective

To verify the operability of the containment vent and purge system.

Specification

The following surveillance shall be performed as stated.

A. Isolation Valves

1. Each month verify that the containment purge supply and exhaust isolation valves are closed during operation above cold shutdown.

2. At least once per 24 months verify that the mechanical stops on the containment vent isolation valve (PCV-1190, -1191, -1192) actuator is limited to the valve opening angle to 60° (90° = full open).

B. HEPA Filters and Charcoal Absorbers

If fuel movement is to take place before the reactor has been subcritical for at least 421 hours, the containment vent and purge system shall be demonstrated operable as follows:

1. Within 18 months prior to fuel movement and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank within 18 months prior to fuel movement, or (2) after structural maintenance on the HEPA filter or charcoal adsorber housing within 18 months prior to fuel movement, which could effect system operation:

   a. Verify that the charcoal adsorbers remove $\geq$ 99% of halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at the operating flow $\pm$ 10%.

   b. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place while operating the ventilation system at the operating flow rate $\pm$ 10%.

2. Within 18 months prior to fuel movement and after every 720 hours of system operation, subject a representative sample of carbon from the charcoal adsorbers to a laboratory analysis and verify within 31 days a removal efficiency of $\geq$ 90% for radioactive methyl iodine at an operating air flow velocity $\pm$ 20% per test 5.b in Table 2 of Regulatory Guide 1.52, March 1978.

ATTACHMENT II TO IPN-96-128

SAFETY EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH THE UPGRADE TO VANTAGE + FUEL

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

SAFETY EVALUATION
RELATED TO
PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED
WITH THE UPGRADE TO VANTAGE + FUEL

Section I - Description of Changes

This application for amendment seeks to revise several sections of Appendix A of the Indian Point 3 Technical Specifications to accommodate the transfer from VANTAGE 5 (without intermediate flow mixers) fuel to VANTAGE + fuel. The new fuel features associated with this transition are being used by other nuclear plants as documented in Reference 2. Reference 1 evaluates the transition to this new fuel for Indian Point 3 and confirms its acceptable use. A summary of this document is provided below and a copy is attached for your review (Attachment III).

Section II - Evaluation of Changes

Indian Point 3 plans to refuel and operate, commencing with Cycle 10, using upgraded VANTAGE + and PERFORMANCE + Westinghouse fuel features. The VANTAGE + fuel design is based upon a modification of the NRC approved Westinghouse VANTAGE 5 fuel assembly design, for which Indian Point 3 is currently licensed. New product features being incorporated for the first time at Indian Point 3 are ZIRLO™ intermediate flow mixing (IFM) grids, low pressure drop mid-grids, clad and tubing materials, annular axial blankets, and variable pitch plenum springs. Changes to the following Technical Specification parameters are required as a result of the transition to VANTAGE + fuel.

- OTΔT function
- OTΔT constants - K_1, K_2 and K_3
- OPΔT constant - K_6
- RCS flow uncertainty
- core thermal limits
- T' and T_{cold}
- time required for movement of fuel in the reactor prior to subcriticality without operation of the containment building vent and purge system
- heat flux (F_Q) and nuclear enthalpy rise ($F_{\Delta H}$) hot channel factors (the revised values of these parameters will be listed in the core operating limits report (COLR))

Reference 1 evaluated the transfer from VANTAGE 5 fuel to VANTAGE + fuel with regards to mechanical design and compatibility, nuclear design, thermal and hydraulic design, accident evaluation and analysis, and radiological impact. These results are summarized below.

Mechanical Design

Reference 1 evaluates the mechanical design and the compatibility of 15x15 VANTAGE + fuel assemblies with the current 15x15 VANTAGE 5 (without IFMs) during the transition through mixed fuel cores to all VANTAGE + with PERFORMANCE + features fuel assemblies in the

reload cores. The significant new mechanical features of the Cycle 10 VANTAGE + design relative to the current Cycle 9 VANTAGE 5 (without IFMs) design in operation include the following:

- ZIRLO™ low pressure drop mid-grids;
- IFM grids;
- ZIRLO™ guide thimble tubes and ZIRLO™ instrumentation tubes;
- mid-enriched annular pellets in axial blankets;
- assembly dimensional modifications;
- fuel rod design modifications; and
- variable pitch fuel rod plenum spring.

The significant new mechanical features of the PERFORMANCE + design relative to the current VANTAGE 5 design in operation include the following:

- low cobalt top and bottom nozzles;
- coated lower fuel rod; and
- ZIRLO™ mid and IFM grids.

Reference 1 analyzes the effects of these features and concluded that the VANTAGE + fuel assembly with PERFORMANCE + features are mechanically compatible with the VANTAGE 5 assembly, reactor internals interfaces, the fuel handling equipment, and refueling equipment. Evaluation of the 15x15 VANTAGE + with PERFORMANCE + features fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents, verified that the fuel assembly design is structurally acceptable. In addition, an evaluation of the VANTAGE + with PERFORMANCE + features fuel assembly structural integrity, considering the lateral effects of a LOCA and a seismic accident for the condition of the reactor core with and without both upper fuel alignment pins in the peripheral core locations, was performed. The results of this evaluation showed that the fuel assembly has more margin in withstanding the faulted condition transient load than previous Indian Point 3 designs.

Nuclear Design

The key safety parameters evaluated for Indian Point 3 as it transitions to an all 15x15 VANTAGE + core show little change relative to the current design. The changes in values of the key safety parameters are typical of the normal cycle to cycle variations experienced as loading patterns change. The evaluations have shown that the VANTAGE + nuclear design bases are satisfied and that the safety limits characteristic of the VANTAGE 5 (without IFMs) fuel design, which is used as input to the FSAR safety analyses, also apply to the VANTAGE + fuel design. Margin to key safety parameter limits is not reduced by the VANTAGE + fuel design relative to the VANTAGE 5 (without IFMs) designs in similar applications. The changes from the current VANTAGE 5 (without IFMs) fuel core to a core containing the upgraded fuel product will not cause changes to the current Indian Point 3 FSAR nuclear design bases. Nuclear design methodology is not affected by the use of upgraded fuel features. In conclusion, Reference 1 states that changes in the nuclear characteristics due to the transition to VANTAGE + with PERFORMANCE + fuel assembly features will be within the range normally seen from cycle to cycle due to fuel management.

Thermal and Hydraulic Design

The calculational methods employed for the thermal hydraulic analysis of the upgrade to VANTAGE + fuel include use of the Revised Thermal Design Procedure (RTDP) and the THINC-IV-PWR code. Reference 1 states that the thermal hydraulic evaluation of the fuel upgrade for Indian Point 3 has shown that 15x15 VANTAGE 5 (without IFMs) and VANTAGE + fuel assemblies are hydraulically compatible and all current thermal hydraulic design criteria are satisfied.

Non-LOCA Accidents

The Indian Point 3 FSAR includes analyses or evaluations of the non-LOCA accidents. Reference 1 reviewed all of the non-LOCA accidents to address the transition from VANTAGE 5 (without IFMs) fuel to VANTAGE + fuel. The analysis assumptions that were different from those currently used are the RTDP, revised overtemperature delta temperature (OTΔT) and overpower delta temperature reactor trip setpoints (OPΔT), increased rod cluster control assembly (RCCA) scram time, increased $F_{\Delta H}$ and F_Q, and increased uncertainties for the 24 month reload cycles. All of the accidents which are affected by one or more of the VANTAGE + design features or the revised safety analysis assumptions directly associated with the VANTAGE + fuel were reanalyzed. These consist of the following events:

- uncontrolled control rod withdrawal from a subcritical condition;
- uncontrolled control rod assembly withdrawal at power;
- rod assembly misalignment;
- RCCA drop;
- loss of reactor coolant flow;
- loss of external electrical load;
- excessive heat removal due to feedwater system malfunctions;
- excessive load increase incident;
- rupture of a steam pipe (core response); and
- rupture of a control rod drive mechanism housing (RCCA ejection).

Reference 1 concluded that reanalysis shows that the transition from 15x15 VANTAGE 5 (without IFMs) to 15x15 VANTAGE + fuel can be accommodated with margin to the applicable FSAR safety analysis limits. A summary of each of these analyses is presented below.

Uncontrolled Control Rod Withdrawal From a Subcritical Condition

An uncontrolled RCCA withdrawal accident is defined by an addition of reactivity to the reactor core caused by uncontrolled withdrawal of RCCA banks resulting in a power excursion. Reference 1 states that, considering the effects of VANTAGE + fuel, the core and the RCS are not adversely affected by this event since the combination of thermal power and coolant temperature and flow result in a DNBR greater than the limit value. Thus, no fuel or clad damage is predicted as a result of this transient.

Uncontrolled Control Rod Assembly Withdrawal at Power

This event is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of RCCA banks when the core is above the no-load condition. Reference 1 states that the high neutron flux and OTΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates ensuring that the minimum calculated DNBR is always greater than the safety analysis limit value and pressurizer filling does not occur. In addition, peak pressure in the RCS and the secondary steam system do not exceed 110% of their respective design pressures.

Rod Assembly Misalignment

RCCA misalignment accidents include one or more dropped RCCAs within the same group, a dropped RCCA bank, or a statically misaligned RCCA. Reference 1 states that DNB does not occur for the RCCA misalignment incident; thus, there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value.

RCCA Drop

The dropped RCCA accident is initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given bank to drop to the bottom of the core. Reference 1 states that the RCCA drop event was analyzed to show that: 1) the integrity of the core is maintained by the reactor protection system (RPS) as the DNBR remains above the safety analysis limit value; 2) the peak RCS and secondary system pressures remain below the accident analysis pressure limits; and 3) the pressurizer does not reach a water solid condition. Therefore, it is concluded that the insertion of VANTAGE + fuel is acceptable for the RCCA drop event.

Loss of Reactor Coolant Flow

The loss of reactor coolant flow transients encompass the partial loss of forced reactor coolant flow, complete loss of forced reactor coolant flow, reactor coolant pump (RCP) shaft seizure (locked rotor), and the reactor coolant pump shaft break (reverse flow). Reference 1 states that for the partial or complete loss of an RCP event, the DNBR does not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted and all applicable acceptance criteria are met. For the reactor coolant pump shaft seizure or break, Reference 1 states that all safety criteria are satisfied. This demonstrates that the RCS and the core will remain able to provide long term cooling, and off-site doses remain within the guidelines of 10 CFR 100 for the transition from the resident VANTAGE 5 fuel to the VANTAGE + fuel.

Loss of External Electrical Load

The loss of external electrical load and/or turbine trip event is defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip. This event is

analyzed as a turbine trip from full power as this bounds both events. Reference 1 states that analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the plant's design are adequate to limit the maximum pressures to within the design limits. The integrity of the core is maintained by operation of the RPS, ensuring the DNBR is maintained above the safety analysis limit value. Thus, no core safety limit will be violated. In conclusion, the implementation of VANTAGE + fuel design is acceptable for this event.

Excessive Heat Removal Due to Feedwater System Malfunctions

Excessive feedwater additions are postulated to occur from a malfunction of the feedwater control system or an operator error which results in the opening of a feedwater control valve. At initial no load conditions, the maximum reactivity insertion rate that occurs following an excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition. Therefore, the results of this transient are bounded by those discussed previously for rod withdrawal from a subcritical condition.

For the cases of the excessive feedwater addition initiated from full power conditions with and without automatic rod control, Reference 1 states that pressure in the reactor coolant and main steam systems would be maintained below 110% of the design values, fuel cladding integrity would be maintained by ensuring that the minimum DNBR remains above the safety analysis limit value, and the event would not generate a more serious plant condition without other faults occurring independently. In conclusion, the use of VANTAGE + fuel is acceptable for the excessive heat removal due to a feedwater system malfunction event.

Excessive Load Increase Incident

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. Cases are analyzed for manual and automatic reactor control with the beginning of life minimum moderator reactivity feedback and end of life maximum moderator reactivity feedback. For all cases, the plant rapidly reaches a stabilized condition and the DNBR remains above the safety analysis limit DNBR value, thereby precluding fuel or clad damage.

Rupture of a Steam Pipe

A rupture of a steam pipe results in an uncontrolled steam release from a steam generator. Reference 1 contains the results of the reanalysis of this event considering the effects of VANTAGE + fuel. It states that the core analysis performed demonstrated that DNB does not occur. For the limiting steam line break case inside the containment structure, the current licensing basis analysis for containment response remains applicable for the transition to VANTAGE + fuel. Hence, the peak pressure inside the containment following a rupture of a steam pipe is 42.42 psig, which is less than the applicable containment design pressure limit of 47 psig. In conclusion, the analysis and

evaluations for the rupture of a steam pipe events demonstrate that all applicable licensing basis safety analysis criteria are satisfied for the transition from VANTAGE 5 to VANTAGE + fuel.

Rod Ejection

Reference 1 discusses the analysis of four conditions for the rod ejection event, namely at the beginning of life or end of life and at hot zero power or hot full power. Results show that all safety criteria are met for these transients and it is concluded that the RCS and the core will remain able to provide long term cooling, and off-site doses remain within the guidelines of 10 CFR 100. Therefore, the transition to VANTAGE + fuel is acceptable.

Large Break LOCA

An analysis for a large break LOCA was performed considering the VANTAGE + fuel. This analysis included the effects of extended burnup of the VANTAGE + fuel, the optimized fuel rod plenum spring, low cobalt top and bottom nozzles, debris resistant oxide coating, ZIRLO™ guide thimble tubes, ZIRLO™ instrumentation tubes, mid-enriched annular pellets in axial blankets and the protective bottom grid with elongated bottom end plug/eternal grippable top end plug. An additional feature of the 15x15 VANTAGE + fuel, the IFBA, has been specifically analyzed for Indian Point 3. This analysis has determined that the 15x15 VANTAGE + fuel with IFBA proposed for use will result in lower large break LOCA peak clad temperatures (PCTs). The effect of the transition core cycles is conservatively evaluated to result in a maximum change of 50°F in the calculated PCT due to an increase in the hydraulic resistance of the VANTAGE + fuel assemblies. This core penalty can be accommodated by the margin to the 10 CFR 50.46 2200°F limit. In conclusion, results of the analysis for transfer to VANTAGE + fuel show that Indian Point 3 remains in compliance with the requirements of 10 CFR 50.46.

Small Break LOCA

An analysis for a small break LOCA was performed considering the VANTAGE + fuel. This analysis included the effects of extended burnup of the VANTAGE + fuel, optimized fuel rod plenum spring, low cobalt bottom nozzle, debris resistant oxide coating, ZIRLO™ guide thimble tubes, mid-enriched annular pellets in axial blankets, ZIRLO™ instrumentation tubes, and the IFBAs. Hydraulic differences between the VANTAGE 5 (without IFMs) and VANTAGE + fuel assemblies was determined not to be a factor. In conclusion, the results of the small break emergency core cooling system (ECCS) analysis for the transfer to VANTAGE + fuel show that Indian Point 3 remains in compliance with the requirements of 10 CFR 50.46.

Radiological Impact Assessment

The change from VANTAGE 5 fuel to VANTAGE + fuel has been reviewed to determine if there are any changes that would impact the radiological consequences of accidents that have been analyzed as part of the licensing basis. Only two of the changes associated with the VANTAGE + fuel design have been identified as having the potential for impact on the radiological consequences of accidents, namely, the increase in the radial peaking factor and the

increase in fuel burnup limits. Reference 1 documents that the radiological consequences of an accident will not be increased due to these factors.

Section III - No Significant Hazards Evaluation

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously analyzed?

Response:

The probability of occurrence or the consequences of an accident previously evaluated is not significantly increased. The VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble and instrument tubes, IFMs, and LPD mid-grids meet the same fuel assembly and fuel rod design bases as VANTAGE 5 (without IFMs) fuel assemblies in the other fuel regions. In addition, the 10 CFR 50.46 criteria will be applied to the ZIRLO™ clad fuel rods, thimble and instrument tubes, IFM grids, and LPD mid-grids. The use of these fuel assemblies will not result in a change to the proposed Indian Point 3 VANTAGE 5 (without IFMs) transition core design and safety analysis limits. The ZIRLO™ clad material is similar in chemical composition and has similar physical and mechanical properties as that of Zircaloy-4. Thus the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. The ZIRLO™ clad fuel rod improves corrosion resistance and dimensional stability. In addition, the incorporation of LPD mid-grids and IFMs improves dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel assemblies material changes, the radiological consequences of an accident previously evaluated is not significantly increased.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The possibility for a new or different type of accident from any accident previously evaluated is not created, since the VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble and instrument tubes, IFMs, and LPD mid-grids will satisfy the same design bases as that used for VANTAGE 5 (without IFMs) fuel assemblies in the other fuel regions. Since the original design criteria is being met, the ZIRLO™ clad fuel rods, thimble and instrument tubes, IFMs, and LPD mid-grids will not be an initiator for any new accident. All design and performance criteria will continue to be met and no new single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alterations to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The margin of safety is not significantly reduced, since the VANTAGE + fuel assemblies clad fuel rods, thimble and instrument tubes, IFMs, and LPD mid-grids do not change the proposed Indian Point 3 VANTAGE 5 (without IFMs) transition core design and safety analysis limits. The use of these fuel assemblies containing fuel rods, thimble and instrument tubes with ZIRLO™ cladding alloy, IFMs and LPD mid-grids will take into consideration the normal core operating conditions allowed in the Technical Specifications. For the transition core and each future cycle reload core, these fuel assemblies will be specifically evaluated using standard reload design methods and approved fuel rod design models and methods. This will include consideration of the core physics analysis, peaking factors and core average linear heat rate effects. In addition, the 10 CFR 50.46 criteria will be applied each cycle to the ZIRLO™ clad fuel rods, thimble and instrument tubes, IFMs, and LPD mid-grids.

Section IV - Impact of Changes

These changes will not adversely affect the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of these changes: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any technical specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

1. Westinghouse Report, "Vantage + Fuel Upgrade - Reload Transition Safety Report for Indian Point Nuclear Plant Unit 3, Revision 2," December 1996.

2. Westinghouse Report, WCAP-8183, Revision 23, Operational Experience With Westinghouse Cores," January 1996.

ATTACHMENT III TO IPN-96-128

VANTAGE + FUEL UPGRADE
RELOAD TRANSITION SAFETY REPORT FOR INDIAN POINT NUCLEAR PLANT UNIT 3
REVISION 2

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

VANTAGE + Fuel Upgrade
Recommended Licensing Submittal
for
Indian Point Nuclear Plant Unit 3

Indian Point Nuclear Plant Unit 3
VANTAGE + Fuel Upgrade

Table Of Contents

Table Of Contents

Table Of Contents

List Of Tables

List Of Tables

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

List Of Figures

1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

The New York Power Authority plans to refuel and operate the Indian Point Unit 3, commencing with Cycle 10, using upgraded VANTAGE + and PERFORMANCE + Westinghouse fuel features. This report summarizes the safety evaluations that were performed to confirm the acceptable use of these features. Sections 2.0 through 5.0 of this licensing submittal provide the results of the Mechanical, Nuclear, Thermal and Hydraulic, and Accident Evaluations, respectively. Appendix A gives a summary of the Technical Specification changes required and the corresponding change pages. PERFORMANCE + fuel features are addressed on a 50.59 basis.

1.2 Upgraded Fuel Features (VANTAGE + with PERFORMANCE + Features)

Indian Point Unit 3 Cycle 10 and subsequent core loadings will have fuel assemblies that incorporate ZIRLO™ low pressure drop mid-grids, Intermediate Flow Mixing (IFM) grids, fuel clad, guide thimble and instrumentation tubing. Other features include annular axial blankets and variable pitch plenum springs. This upgraded fuel feature is known as VANTAGE + and has been submitted to the NRC in the licensing topical "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 Appendices A through D[1], Appendix E[2], Appendices F and G[3], and associated Addendums 1 through 4[4][5][6][7]. VANTAGE + has received generic NRC approval[8][9][54] for lead rod burnups up to 62,000 MWD/MTU.

The VANTAGE + fuel design is based upon a modification of the NRC-approved Westinghouse VANTAGE 5 fuel assembly design[10], for which Indian Point Unit 3 is currently licensed. VANTAGE 5 and VANTAGE + product features being incorporated for the first time in Indian Point Unit 3 are ZIRLO™ Intermediate Flow Mixing (IFM) grids, low pressure drop mid-grids, clad and tubing materials, and variable pitch plenum springs.

NRC approval for the above VANTAGE 5 features and VANTAGE 5 fuel assembly design was received in a July 1985 SER[10]. Approval for the application of IFBA to non-17x17 arrays was received in March, 1986 in Reference 10, Addendum 1-A.

VANTAGE + will also include the following: modified assembly skeleton dimensions, ZIRLO™ guide thimbles and instrumentation tube, ZIRLO™ fuel rod cladding (Indian Point Unit 3 already licensed for

ZIRLO™ per NRC SER dated May 15, 1992[11]), optimized plenum fuel stack coil spring, and fuel stack height changes, as required. The Indian Point Unit 3 VANTAGE + fuel rod design, for lead rod burnups beyond 62,000 MWD/MTU, is based on the ZIRLO™ fuel performance models given in Reference 1 and as modified in Reference 12 which is currently pending generic NRC approval.

The VANTAGE + fuel assembly includes dimensional changes to accommodate extended burnup (lead rod burnups beyond 62,000 MWD/MTU when licensed by the NRC). The VANTAGE + RTN is identical to that of the VANTAGE 5 RTN.

The VANTAGE + fuel assembly skeleton is identical to the VANTAGE 5 fuel assembly design except for those modifications necessary to accommodate intended fuel operation to higher burnup levels (lead rod burnups beyond 62,000 MWD/MTU when licensed by the NRC). These modifications consist of the use of ZIRLO™ guide thimbles, ZIRLO™ instrumentation tubes, and small skeleton dimensional alterations[1]. Since VANTAGE + fuel is intended to replace either the Westinghouse Standard, Optimized, or VANTAGE 5 fuel designs, the VANTAGE + exterior assembly envelope is equivalent in design dimensions, and the functional interface with the reactor internals is equivalent to those of Westinghouse VANTAGE 5 fuel design for which Indian Point Unit 3 is currently licensed. Also, the VANTAGE + fuel assembly is designed to be mechanically and hydraulically compatible with the Standard, Optimized and VANTAGE 5 designs in full or transition cores, and the same functional requirements and design criteria previously established for the Westinghouse VANTAGE 5 fuel assembly (Reference 10) remain valid for the VANTAGE + fuel assembly.

The use of ZIRLO™ cladding in the VANTAGE + fuel rod represents a modification to the Zircaloy-4 VANTAGE 5 fuel rod for the purpose of supporting operation to higher fuel burnups. Indian Point Unit 3 is licensed for ZIRLO™ fuel rod cladding as stated in the NRC SER dated May 15, 1992[11]. The VANTAGE + fuel rods will contain enriched uranium dioxide fuel pellets, annular axial blanket (mid-enriched uranium dioxide) pellets, and variable pitch plenum springs.

The significant new mechanical features of the PERFORMANCE+ design relative to the current VANTAGE 5 design in operation include the following:

- Low Cobalt Top and Bottom Nozzles
- Coated Lower Fuel Rod
- Protective Bottom Grid
- Longer End Plugs
- ZIRLO™ mid and IFM grids

1.3 Increased Peaking Factors

The future cycles of operations for Indian Point Unit 3 will use increased $F^N_{\Delta H}$ and $F_Q(Z)$ peaking factors. The full power $F^N_{\Delta H}$ peaking factor design limit will increase from the current value of 1.62 to 1.654 for VANTAGE 5 fuel and 1.70 for VANTAGE + fuel. The maximum $F_Q(Z)$ peaking factor limit will increase from the current value of 2.32 to 2.42 and the K(Z) envelope will be modified. These increases will permit more flexibility in developing fuel management schemes (i.e., longer fuel cycles, improvement of fuel economy and neutron utilization).

1.4 Conclusions

The results of the evaluations and analyses described herein lead to the following conclusions:

1. The Westinghouse fuel assemblies containing VANTAGE + with PERFORMANCE + upgraded fuel features for Indian Point Unit 3 are mechanically compatible with the current VANTAGE 5 (w/o IFMs) fuel assemblies, control rods, and reactor internals interfaces.

2. The structural integrity of the 15x15 VANTAGE + with PERFORMANCE + fuel assembly design features for seismic/LOCA loadings has been evaluated for Indian Point Unit 3. Evaluation of the 15x15 VANTAGE + with PERFORMANCE + features fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the fuel assembly design is structurally acceptable. In addition, an evaluation of the VANTAGE + with PERFORMANCE + features fuel assembly structural integrity considering the lateral effects of a LOCA and a seismic accident for the condition of the reactor core with and without both upper fuel alignment pins in the peripheral core locations was performed. The results of this evaluation showed that the fuel assembly has more margin in withstanding the faulted condition transient load than previous Indian Point Unit 3 designs.

3. Changes in the nuclear characteristics due to the transition to VANTAGE + with PERFORMANCE + fuel assembly features will be within the range normally seen from cycle-to-cycle due to fuel management.

4. The reload VANTAGE + with PERFORMANCE + features fuel assemblies are hydraulically compatible with the VANTAGE 5 (w/o IFMs) fuel assembly design currently in use at Indian Point Unit 3.

5. The change in the design full power $F^N_{\Delta H}$ limit from 1.62 to 1.654 for VANTAGE 5 fuel and 1.70 for VANTAGE + fuel (with appropriate treatment of uncertainties) is supported by design basis safety analyses summarized in this report. The corresponding changes to the Technical Specifications are as defined in Appendix A.

6. The change in the maximum $F_Q(Z)$ limit from 2.32 to 2.42 and modification to the K(Z) envelope is supported by licensing basis safety analyses summarized in this report. The corresponding changes to the Technical Specifications are as defined in Appendix A.

7. The core design and safety evaluations documented in this report show the Indian Point Unit 3 core's capability for operating safely at the rated Indian Point Unit 3 design thermal power of 3025 MWt.

8. This report establishes a reference upon which to base Westinghouse reload safety evaluations (RSE) for future reloads with the upgraded fuel features and increased peaking factor limits.

2.0 MECHANICAL DESIGN FEATURES

2.1 Introduction and Summary

This section evaluates the mechanical design and the compatibility of 15x15 VANTAGE + fuel assemblies with Indian Point Unit 3's current 15x15 VANTAGE 5 (w/o IFMs) during the transition through mixed-fuel cores to all VANTAGE + with PERFORMANCE + features fuel assemblies in the reload cores. The VANTAGE + fuel assembly with PERFORMANCE + features has been designed to be compatible with the VANTAGE 5 assembly, reactor internals interfaces, the fuel handling equipment, and refueling equipment. Excluding fuel assembly length and fuel rod length, the VANTAGE + with PERFORMANCE + features design dimensions are essentially equivalent to the current Indian Point Unit 3 VANTAGE 5 (w/o IFMs) assembly design from an exterior assembly envelope and reactor internals interface standpoint. References in this section are made to WCAP-12610-P, "VANTAGE + Fuel Assembly Reference Core Report"[1], and WCAP-10444-P-A, "VANTAGE 5 Fuel Assembly Reference Core Report"[10].

The significant new mechanical features of the Cycle 10 VANTAGE + design relative to the current Cycle 9 VANTAGE 5 (w/o IFMs) design in operation include the following:

- Low Pressure Drop mid-grids
- Intermediate Flow Mixer grids
- ZIRLO™ Guide Thimble Tubes and ZIRLO™ Instrumentation Tubes
- Mid-Enriched Annular Pellets in Axial Blankets
- Assembly Dimensional Modifications
- Fuel Rod Design Modifications
- Variable Pitch Fuel Rod Plenum Spring

The significant new mechanical features of the PERFORMANCE + design relative to the current VANTAGE 5 design in operation include the following:

- Low Cobalt Top and Bottom Nozzles
- Coated Lower Fuel Rod
- Protective Bottom Grid
- Longer End Plugs
- ZIRLO™ mid and IFM grids

Section 2.2 describes and evaluates the differences between the fuel assembly designs.

Based on the evaluation of the VANTAGE + and VANTAGE 5 (w/o IFMs) design differences, it is concluded that the two designs are mechanically compatible with each other. The VANTAGE + fuel rod mechanical design bases remain unchanged from that used for the VANTAGE 5 (w/o IFMs) assemblies in the Cycle 9 core.

Mechanical testing, fuel rod bow and wear, core component evaluation, and an evaluation of fuel assembly integrity during a LOCA and seismic event are discussed in Sections 2.3 through 2.8.

The fuel is designed according to the fuel performance models in References 1, 13, 14 and 15. The fuel is designed to operate so that clad flattening, as analyzed by the Westinghouse model[16] will not occur. The fuel rod internal pressure design bases[1][17] are satisfied for all fuel regions.

2.2 Compatibility of Fuel Assemblies

The VANTAGE + design for Cycle 10 and subsequent cycles incorporates three Intermediate Flow Mixing (IFM) grids and low pressure drop mid-grids. The VANTAGE + assembly has the same cross-sectional envelope as the VANTAGE 5 (w/o IFMs). The grid elevations of the VANTAGE + are slightly different than those of the VANTAGE 5 fuel due to the incorporation of Intermediate Flow Mixer (IFM) grids. However, the two fuel types are hydraulically compatible as discussed in section 4.3; these small mismatches are negligible and well within established limits for acceptable fretting wear.

The VANTAGE + fuel assembly may also incorporate the PERFORMANCE + protective bottom grid to provide additional debris protection. The fuel rod at BOL will be located at 0.065 ± 0.060 inches off the bottom nozzle. The addition of the protective grid features to the fuel assembly has an insignificant impact on the fuel assembly mechanical characteristics.

2.2.1 Fuel Rods

The VANTAGE + fuel rod has the same clad wall thickness and fuel rod diameter as the VANTAGE 5 (w/o IFMs) fuel rod. The Indian Point Unit 3 VANTAGE + fuel also has mid-enriched annular pellets in axial blankets and optimized plenum springs to maximize the available plenum volume for increased burnup.

The VANTAGE + design containing the annular pellets in axial blankets will behave in a similar manner to VANTAGE 5 fuel of similar enrichment and burnup containing solid pellets in axial blankets with regards to xenon stability, load follow capability, peaking factor, rod worths and shutdown margin. Annular pellets in axial blankets reduce neutron leakage and improve fuel utilization.

The key design difference between annular pellets in axial blankets and enriched fuel pellets, aside from U-235 enrichment, is the annulus itself. Annular pellets in axial blankets also have the same chamfer as the enriched fuel pellets, but no dish on the pellet ends. Pellet length-to-diameter ratio is maintained at approximately 1.4; this ratio has been adjusted for the use of an even multiple of pellet lengths to obtain appropriate axial blanket zone length in fabrication. The same fuel stack length as applicable to current fuel design is maintained with the 6" annular axial blanket in the VANTAGE + design.

The relatively low range of linear heat rate which the annular pellet in axial blankets will experience and the modest fraction of the fuel volume which it occupies, assures that its use will not have any significant effect on the limiting fuel temperature or rod internal pressure, other than that due to the additional void volume provided by the axial blanket pellet annulus.

The IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin zirconium diboride (ZrB_2) coating less than 0.001 inch in thickness on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of IFBA rods within an assembly may vary depending on specific application. The ends of the enriched coated pellets and enriched uncoated pellets are dished to allow for greater axial expansion at the pellet centerline and void volume for fission gas release.

The VANTAGE + fuel rods are designed with additional plenum space to accommodate gases released from the fuel. The differential thermal expansion between the cladding and the fuel, and fuel density changes during irradiation are also evaluated to avoid overstressing of the clad or rod welds. Shifting of the fuel within the clad during handling and/or shipping prior to core loading is prevented by the 4g stainless steel optimized plenum spring which bears on the top of the fuel column.

The incorporation of the protective bottom grid features the debris-mitigation bottom end plug and the external grip top end plug in its fuel rod design. The debris-mitigation bottom end plug has a total length of 0.810 inches which is 0.380 inches longer then current design. The external grip top end plug has a total length of 0.450 inches which is 0.093 inches longer than the current design. This increase in the end plugs length will enhance the fuel rod reliability. The debris-mitigation bottom end plug will assure

that the protective bottom grid dimples are contacting the solid end plug and not the cladding. Therefore, debris caught by the DFBN and the protective bottom grid will act against the solid end plug and not the cladding. The external grip top end plug will allow repositioning and simplify reconstituting the fuel rod from the top.

The Indian Point Unit 3 VANTAGE + fuel rod also has the PERFORMANCE + oxide coating at the bottom end of the fuel rod. The extra layer provides additional rod fretting wear protection.

The VANTAGE + fuel rod design bases and evaluation are given in Section 2.0 in Reference 1.

2.2.2 Grid Assemblies

The low-pressure-drop (LPD) mid-grid is only 1.5" high, but provides structural strength comparable to the current 2.25" zircaloy mid-grid which it replaces. The grid has a low-pressure-drop because of its reduced height and because of an innovative grid spring. The grid springs are oriented diagonally, which means they induce less resistance to coolant flow and provide additional strength. In addition, the upstream edges of the grid have been chamfered to further enhance hydraulic conditions.

The small differences between the mechanical characteristics of the fuel assemblies are attributable to the addition of IFM grids. Although these grids are not intended to function as structural components, they provide a slight stiffening effect near the top of the fuel assembly. The presence of the IFM grids also permits additional load sharing by these grids during seismic/LOCA impact which represents a small improvement in the dynamic performance of the fuel assembly.

The IFM grids are located in the three uppermost spans between the LPD mixing vane structural grids and incorporate an identical mixing vane array. Their prime function is mid-span flow mixing in the hottest spans of the fuel assembly. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the IFM can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids and the LPD mixing vane grids, are fabricated from ZIRLO™. This material was selected to take advantage of the material's inherent low neutron capture cross-section and corrosion resistance.

The top and bottom Inconel (non-mixing vane) grids of the VANTAGE + fuel assemblies are similar in design to the Inconel grids of the Cycle 9 VANTAGE 5 (w/o IFMs) fuel assemblies. The five intermediate (mixing vane) Zircaloy grids have the same design as the previous VANTAGE 5 (w/o IFMs) intermediate grids. The Zircaloy grid incorporates the same grid cell support configuration as the Inconel end grid (six support locations per cell, four dimples, and two springs). The Zircaloy interlocking strap joints and grid/sleeve joints are fabricated by laser welding, whereas the Inconel grid joints are brazed.

However, if the protective grid is added the bottom non-mixing vane grid will be slightly modified. The bottom non-mixing vane grid which is used in combination with the protective bottom grid will have 16 of the 20 inserts spot welded to it. The other four inserts will be welded to the protective bottom grid at the four outer most diagonal locations. This is necessary to de-couple the grids to prevent the creation of an excessively stiff composite structure. An analysis has shown that the bottom non-mixing vane grid with 16 inserts attached is an acceptable design.

The purpose of the protective bottom grid is to provide additional debris mitigation protection. The protective bottom grid is located 0.020 $\pm$ 0.015 above the bottom nozzle with its straps subdividing the Debris Filter Bottom Nozzle (DFBN) flow holes to further reduce the amount and size of debris that can enter the fuel bundle. In addition to the enhancement of the debris-mitigation, the protective bottom grid will also aid in reducing grid-to-rod fretting. The protective bottom grid is designed to have its dimples on the full diameter of the solid bottom end plug through EOL.

The protective bottom grid is fabricated with 0.0105 inch thick inner and 0.0205 inch thick outer Inconel 718 straps. The straps are laser welded at the intersects and to the outer grid strap similar to the Zircaloy grids. The protective bottom grid is spot welded to the four outer most diagonal grid inserts, which are not spot welded to the bottom grid, to fix its axial position. The addition of the protective grid will have an insignificant effect on the fuel assembly mechanical characteristics.

The VANTAGE + grid assembly design bases and evaluation are given in Section 2.3.5 in Reference 1.

2.2.3 Guide Thimble and Instrumentation Tubes

To accommodate higher assembly burnup, the overall fuel assembly length has been reduced. VANTAGE + guide thimbles and instrumentation tubes are shorter and are fabricated of ZIRLO™ material which has less growth than Zircaloy-4.

The general design bases for the VANTAGE + guide thimble and instrumentation tubes remain the same as those given in Reference 10 and 18.

2.2.4 Reconstitutable Top Nozzle (RTN) and Holddown Springs

The VANTAGE + RTN will be fabricated of low cobalt 304 stainless steel. The RTN for the VANTAGE + fuel assembly is identical to the VANTAGE 5 (w/o IFMs) design with the exception of the holddown spring. The height of the holddown spring is increased to maintain the required holddown force on the shortened VANTAGE + fuel assembly.

2.2.5 Debris Filter Bottom Nozzle

The bottom nozzle will also be fabricated of low cobalt 304 stainless steel. A modification required for the application of the protective bottom grid is a pitch increase from 0.495 inch to 0.496 inch for the guide thimble and flow holes to match the protective bottom grid cell pitch. The increase in the bottom nozzle guide thimble and flow hole pitch is necessary to assure proper fit-up of the grid inserts (e.g., welded into the protective bottom grid and bottom non-mixing vane grid), guide thimble end plug, and guide thimble screw. Modifying the flow hole pattern to match the protective grid geometry required decreasing the number of flow holes along with a corresponding increase in both the flow hole diameter and flow hole chamfer O.D.. The changes in material and pitch of the guide thimble and flow holes do not impact any design criteria.

2.3 Mechanical Testing

Mechanical testing of components as well as stiffness, impact and lateral vibration testing of comparable 15x15 fuel assembly configurations confirmed that the design changes associated with VANTAGE + do not significantly alter the fuel assembly structural behavior and projected performance as compared with either the Westinghouse VANTAGE 5 (w/o IFMs) or OFAs fuel assembly designs. Furthermore, design changes associated with the 15x15 protective grid package do not significantly influence the VANTAGE + fuel assembly structural characteristics that were determined by the prior mechanical testing. Therefore, the VANTAGE + fuel assembly with protective bottom grid structural behavior and projected performance remain consistent with the respective VANTAGE + fuel assembly design chosen.

2.4 Fuel Rod Performance

Fuel rod performance for all Indian Point Unit 3 fuel is shown to satisfy the NRC Standard Review Plan (SRP) fuel rod design bases on a cycle-by-cycle basis. These same bases are applicable to all fuel rod designs, including the Westinghouse VANTAGE 5 (w/o IFMs) and VANTAGE + fuel designs, with the only difference being that the VANTAGE + fuel is designed to achieve a higher burnup and operate with a higher $F_{\Delta H}$ limit. The design bases for Westinghouse VANTAGE + fuel are discussed in Reference 1.

There is no impact from a fuel rod design standpoint due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles. The mechanical fuel rod design evaluation for each region incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions (e.g., presence of mid-enriched annular pellets in axial blankets or changes in the fuel rod and plenum lengths). Analysis of Integral Fuel Burnable Absorber (IFBA) rods includes any geometry changes necessary to model the presence of the burnable absorber, and conservatively models the gas release from the ZrB_2 coating. Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally evaluated for the mechanical design of a fuel region (for example, a power uprating or an increase in the peaking factors) are addressed for all affected fuel regions as part of the reload safety evaluation process when the plant change is to be implemented.

Fuel rod design evaluations for Indian Point Unit 3 were performed using the NRC approved models in References 1, 13, 14 and 15 to demonstrate that the SRP fuel rod design criteria are satisfied.

2.5 Rod Bow

Intermediate Flow Mixer grids (IFMs) provide additional mid-span radial support to the fuel rods. This has improved rod bow in the upper section of the fuel assembly in other Westinghouse designs. It is predicted that the 15x15 VANTAGE + rod bow magnitudes, like those of the Westinghouse VANTAGE 5 (w/o IFMs) fuel, will be well within the bounds of existing 15x15 OFA assembly rod bow data. The current NRC approved methodology for comparing rod bow for two different fuel assembly designs is given in Reference 19. Based on this approved methodology, a comparison of L^2/I ($I \equiv$ fuel rod bending moment of inertia, $L \equiv$ span length) and the initial rod-to-rod gap for both the OFA, VANTAGE 5 (w/o IFMs) and VANTAGE + designs shows that for a given burnup, the magnitude of

rod bow for the VANTAGE + or VANTAGE 5 (w/o IFMs) design is taken to be the same as that for the OFA assembly design. Additional evaluations related to rod bow penalties are given in Appendices C and D of Reference 19 and Section 4.4 of this submittal.

Rod bow in Indian Point Unit 3 fuel rods containing IFBAs is not expected to differ in magnitude or frequency from that currently observed in Westinghouse OFA or VANTAGE 5 (w/o IFMs) fuel rods under similar operating conditions. No indications of abnormal rod bow have been observed on visual or dimensional inspections performed on the test IFBA rods in the BR-3 test reactor. Rod growth measurements were also within predicted bounds.

2.6 Fuel Rod Wear

Fuel rod wear is dependent on both the support conditions and the flow environment to which the fuel rod is subjected. The overall VANTAGE + fuel assembly pressure drop is compatible with the VANTAGE 5 fuel assembly pressure drop. Long term (1000 hour) wear testing has confirmed that fuel rod wear is acceptable under the cross-flow conditions encountered with current fuel assembly designs interfacing the VANTAGE + fuel design.

2.7 Seismic/LOCA Impact on Fuel Assemblies

The 15x15 VANTAGE + (with PERFORMANCE + features) fuel assembly is slightly stiffer than the VANTAGE 5 (w/o IFMs) design, and the LPD grid is comparable to the VANTAGE 5 grid design. The grid load response is reduced due to more grids to share the faulted condition loads. As a result, the seismic/LOCA load margin is increased for a full VANTAGE + core or a transition core. The results show adequate grid load margin and that the core coolable geometry and control rod insertion requirements are met.

An evaluation of VANTAGE + fuel assembly structural integrity considering the lateral effects of a LOCA and a seismic accident has been performed. The evaluation is based on the reactor core with and without the removal of both upper fuel alignment pins in the peripheral core locations. The VANTAGE + fuel assembly has more margin in withstanding the faulted condition transient load than the OFAs fuel assemblies used in previous Indian Point Unit 3 reloads.

Since the VANTAGE + assembly is structurally equivalent to the VANTAGE 5 (w/o IFMs) fuel used in previous cycles, the evaluation of the VANTAGE + fuel assembly in accordance with

NRC requirements as given in SRP 4.2, Appendix A, shows that the VANTAGE + fuel is structurally acceptable for the Indian Point Unit 3 plant. The grid loads evaluated for the LOCA and seismic events and combined by the SRSS method identified in SRP 4.2 are less than the allowable limit. The same conclusion is true for a transition core composed of both VANTAGE + and VANTAGE 5 (w/o IFMs) (and OFA) assemblies. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections are well within acceptable limits. Thus, the core coolable geometry is maintained. The reactor can be safely shutdown under the combined faulted condition loads.

2.8 Core Components

The VANTAGE + and VANTAGE 5 (w/o IFMs) are designed to be compatible with the core components in the Indian Point Unit 3 Plant. The VANTAGE + and VANTAGE 5 (w/o IFMs) thimble tube provides sufficient clearance for insertion of control rods, Wet Annular Burnable Absorber (WABA) rods, pyrex rods, source rods, and dually compatible thimble plugs to assure the proper operation of these core components. A description and evaluation of the WABA rods are presented in WCAP-10021-P-A, Revision 1, Reference 20.

3.0 NUCLEAR DESIGN

3.1 Introduction and Summary

The effects of using VANTAGE + fuel features on the nuclear design bases and methodologies for Indian Point Unit 3 are evaluated in this section.

The specific values of core safety parameters, e.g., power distributions, peaking factors, rod worths, are primarily loading pattern dependent. The variations in the loading pattern dependent safety parameters are expected to be typical of the normal cycle-to-cycle variations for the standard fuel reloads. The evaluations have shown that the VANTAGE + nuclear design bases are satisfied and that the safety limits characteristic of the VANTAGE 5 (w/o IFMs) fuel design, which is used as input to the FSAR Chapter 14 safety analyses, also apply to the VANTAGE + fuel design. Margin to key safety parameter limits is not reduced by the VANTAGE + fuel design relative to the VANTAGE 5 (w/o IFMs) designs in similar applications. Standard nuclear design analytical models and methods[21][22][23] accurately describe the neutronic behavior of the VANTAGE + fuel design. In addition, the present Indian Point Unit 3 spent fuel pool criticality analysis, Reference 24, is applicable to the VANTAGE + fuel features.

The plant technical specifications core operating limits (COLR) that are established by nuclear design must be reviewed to determine if they remain appropriate or need to be changed to accommodate the transition to a complete VANTAGE + core. The COLR change which impacts the nuclear design is an increase in the peaking factor limits. This change to the COLR is not required in order to transition to 15x15 VANTAGE + fuel. The increased peaking factor limits increase fuel management flexibility (higher discharge burnup, increased low leakage, increased fuel economy and increased nuclear design flexibility).

The changes from the current VANTAGE 5 (w/o IFMs) fuel core to a core containing the upgraded fuel product will not cause changes to the current Indian Point Unit 3 FSAR[18] nuclear design bases. Nuclear design methodology is not affected by the use of upgraded fuel features.

3.2 Design Basis

The specific design bases and their relation to the General Design Criteria (GDC) in 10 CFR 50, Appendix A for the VANTAGE + design are the same as those of the VANTAGE 5 design as stated in

Section 3.2 of Reference 10, except that for the VANTAGE + product, the fuel burnup design is extended to lead rod average discharge burnups beyond 62,000 MWD/MTU (VANTAGE + is currently only licensed to 60,000 MWD/MTU by the NRC).

The effects of extended burnup on nuclear design parameters has been previously discussed in Reference 17, and that discussion is valid for the anticipated VANTAGE + design discharge burnup level. In accordance with the NRC recommendation made in their review of Reference 17, Westinghouse will continue to monitor predicted versus measured physics parameters for extended burnup applications.

3.3 Methodology

No changes to the nuclear design philosophy, methods or models are necessary because of the transition to VANTAGE + fuel. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the FSAR[18] safety evaluation for each reload cycle[21]. These key safety parameters will be evaluated for each Indian Point Unit 3 reload cycle. If one or more of the parameters fall outside the bounds assumed in the safety analysis, the affected transients will be re-evaluated/re-analyzed and the results documented in the RSE for that cycle.

The 0.422 inch diameter fuel rod has had extensive nuclear design and operating experience with the current Indian Point Unit 3 15x15 VANTAGE 5 (w/o IFMs) fuel assembly design. The Zircaloy grid material has also had extensive nuclear design and operating experience with the current 15x15 VANTAGE 5 (w/o IFMs) fuel assembly designs. These changes have a negligible effect on the use of standard nuclear design analytical models and methods to accurately describe the neutronic behavior of the VANTAGE + fuel.

3.4 Design Evaluation - Physics Characteristics and Key Safety Parameters

Three cycles of core models were established to model the transition to a full 15x15 VANTAGE + fueled core. These models incorporate low pressure drop zircaloy grids; Intermediate Flow Mixer (IFMs) grids; ZIRLO™ clad fuel rods, thimble tubes and instrument tubes; annular pellets in axial blankets; assembly dimensional and fuel rod design modifications; reduced length enriched B-10 loading IFBAs; and increased peaking factors (F_Q and $F^N_{\Delta H}$).

Table 3.1 provides the results of the reload transition core analysis compared to the current limits. The changes in fuel design and discharge burnup caused only a small impact on the key safety parameters relative to the current design. The variations in these parameters are typical of the normal cycle-to-cycle variations that occur as fuel loading patterns are changed each cycle. The implementation of the increased peaking factor limits have no impact on the key safety parameter ranges.

3.5 Design Evaluation - Power Distributions and Peaking Factors

The implementation of annular pellets in axial blankets, reduced length enriched B-10 loading IFBAs, and the increased radial and total peaking factor limits will have minor impacts on the core power distributions and peaking factors experienced in Indian Point Unit 3. The use of annular axial blankets, where the top and bottom six inches of the enriched fuel stack are replaced by mid-enriched uranium and the enrichment of the remaining fuel increased slightly, results in higher axial peaking factors. This increase in axial peaking is mitigated by the use of reduced length enriched B-10 loading burnable absorbers. The increased radial peaking factor limit allows the concept of low leakage fuel management to be extended by placing additional burned fuel on the periphery of the core. The reduction in power in the peripheral assemblies is offset by increased power in the remaining assemblies. This increased radial peaking is accommodated by increasing the radial and total peaking factor limits.

Figure 3.1 shows a comparison of the radial peaking factors between the Cycle 9 design (designed to the current radial peaking factor limit, used reduced length enriched B-10 loading burnable absorbers and solid natural axial blankets) and typical core models beginning with Cycle 10 (designed to the higher peaking factor limits and used 6 inch annular axial blankets and reduced length enriched IFBAs). The increased radial peaking factor is expected and results from the reduced power carried by the more highly burned assemblies placed on the core periphery to reduce neutron leakage.

Beyond the power distribution impacts already mentioned, other changes to the core power distributions and peaking factors are the result of the normal cycle-to-cycle variations in core loading patterns. The normal methods of feed enrichment variation and insertion of fresh burnable absorbers will be employed to control peaking factors. Compliance with the peaking factor limits can be assured using these methods.

3.6 Technical Specification Changes Relative to Nuclear Design

The Technical Specification (COLR) change which impacts the nuclear design is an increase in the peaking factor limits.

The following $F_{\Delta H}$ values have been assumed in the reload transition analysis to appropriately bound the transition and full 15x15 VANTAGE + cores:

$$(V5)\ F_{\Delta H} = 1.654[1 + 0.3(1-P)]$$
$$(V+)\ F_{\Delta H} = 1.70[1 + 0.3(1-P)]$$

where P is the fraction of full power. This higher value allows a greater degree of low leakage.

The F_Q limit will be increased to 2.42 using a two segment K(2) curve as shown in Figure 3.2. This provides greater flexibility with regard to accommodating axially heterogeneous cores incorporating such features as annular axial blankets and reduced length enriched B-10 loading burnable absorbers.

3.7 Nuclear Design Evaluation Conclusions

The key safety parameters evaluated for Indian Point Unit 3 as it transitions to an all 15x15 VANTAGE + core show little change relative to the current design. The changes in values of the key safety parameters are typical of the normal cycle-to-cycle variations experienced as loading patterns change.

Power distributions and peaking factors show slight changes as a result of the incorporation of annular pellets in axial blankets, reduced length enriched B-10 loading burnable absorbers, and increased peaking factor limits, in addition to the normal variations experienced with different loading patterns. The usual methods of enrichment and burnable absorber usage will be employed in the transition and full 15x15 VANTAGE + cores to ensure compliance with the Peaking Factor Technical Specifications.

Table 3-1

Range of Key Safety Parameters

Safety Parameter	Current Design Values	RTSR Values
Reactor Core Power (MWt)	3025	3025
Vessel Average Coolant Temp. HFP (°F)*	574.7	574.7
Coolant System Pressure (psia)	2250	2250
Core Average Linear Heat Rate (Kw/ft)	6.24	6.24
Most Positive MTC (pcm/°F)	0	0
Most Positive MDC, Rodded ($\Delta K/g/cm^3$)	0.54	0.54
Doppler Temperature Coefficient (pcm/°F)	-0.9 to -3.2	-0.9 to -3.2
Doppler Only Power Coefficient (pcm/%Power)		
Least Negative, HFP to HZP	-6.40 to -9.55	-6.40 to -9.55
Most Negative, HFP to HZP	-13.35 to -19.40	-13.35 to -19.40
Beta-Effective	0.004 to 0.007	0.004 to 0.007
Normal Operation $F_{\Delta H}$ (with uncertainties)	1.62	1.654 (V5) 1.70 (V+)
F_Q (with uncertainties)	2.32	2.42

* Based on TDF of 80,900 gpm per loop and 24% uniform SGTP.

Figure 3.1

Comparison of Radial Peaking Factors for Cycle 9 and Transition Cycles

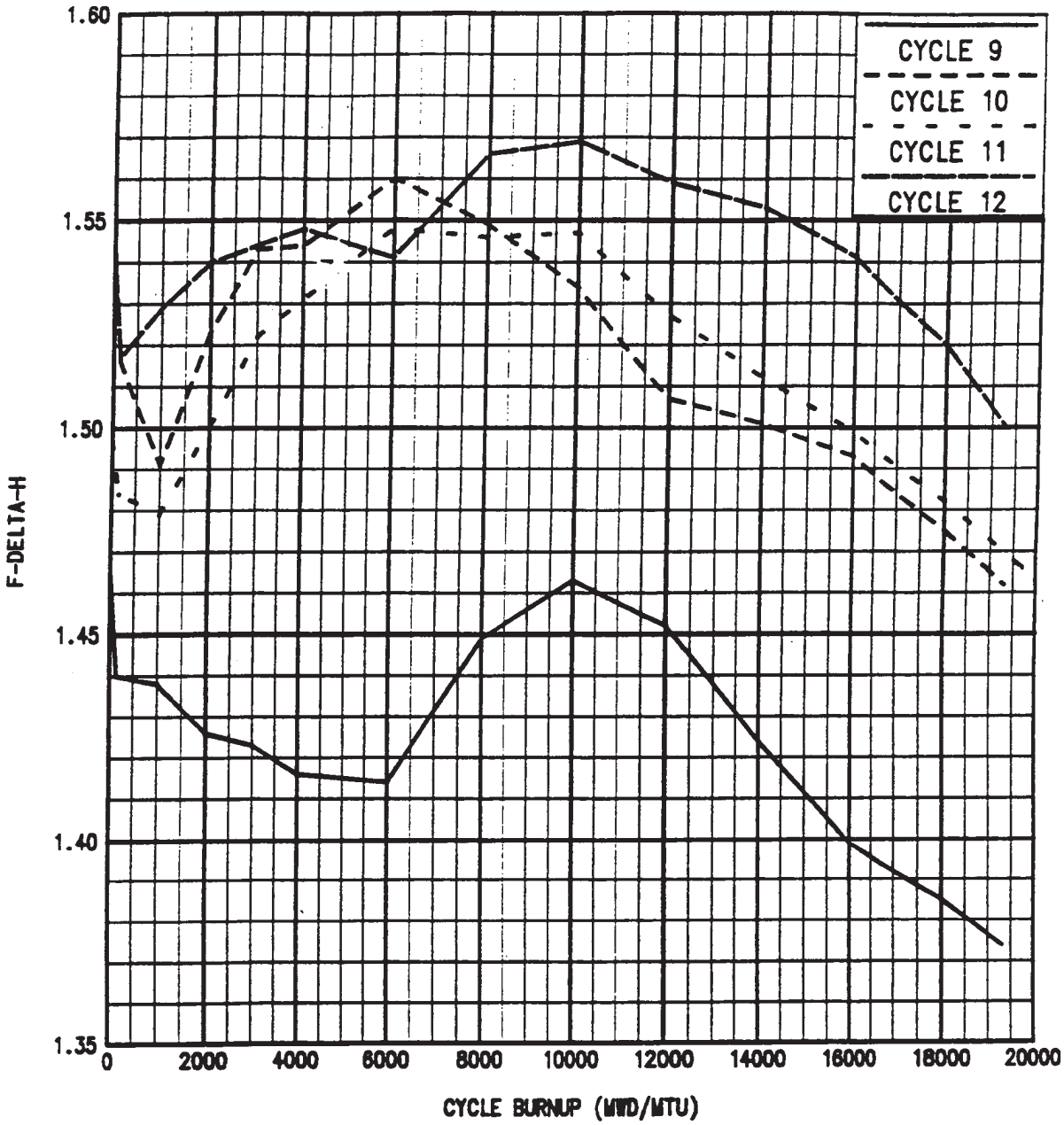
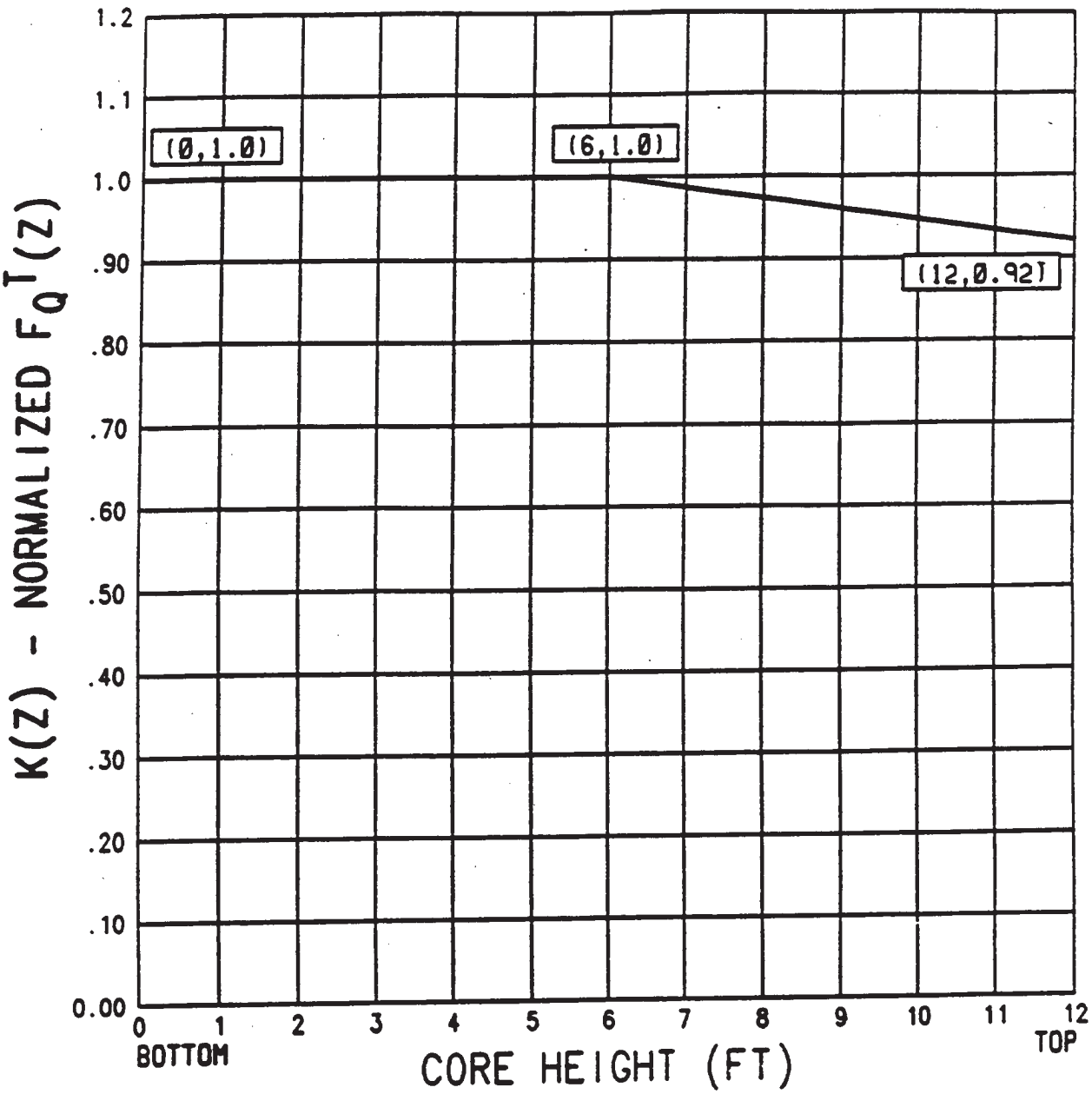


Figure 3.2

K(z) - Normalized $F_Q(Z)$ as a Function of Core Height



4.0 THERMAL AND HYDRAULIC DESIGN

4.1 Introduction and Summary

This section describes the calculational methods used for the thermal-hydraulic analysis, the DNB performance, and the hydraulic compatibility during the transition from a VANTAGE 5 (w/o IFMs) through mixed-fuel cores to an all VANTAGE + core. Based on minimal hardware design differences and prototype hydraulic testing of the fuel assemblies, it is concluded[1] that the VANTAGE 5 (w/o IFMs) and VANTAGE + fuel assembly designs are hydraulically compatible.

The calculational methods employed in the DNBR analyses of the VANTAGE 5 fuel (w/o IFMs) and VANTAGE + fuel are based on the Revised Thermal Design Procedure (RTDP) described in Reference 25. The Improved THINC-IV PWR design modeling method[26] was used. The VANTAGE 5 (w/o IFMs) and VANTAGE + fuel use the WRB-1 DNB correlation[27].

Table 4.1 summarizes the thermal-hydraulic design parameters for the Indian Point Unit 3 used in these analyses. The thermal-hydraulic design criteria remain the same as those presented in the Indian Point Unit 3 FSAR[18] with the exceptions noted in the following sections. All of the current FSAR thermal-hydraulic design criteria are satisfied.

4.2 Methodology

The Improved THINC-IV PWR design modeling method[26] was used for the DNBR analyses of the VANTAGE 5 (w/o IFMs) and VANTAGE + fuel. This modeling scheme improves the accuracy of the analyses by minimizing the inaccuracies which result from the use of the perturbation technique in the solution of the governing equations.

The DNBR analyses of the VANTAGE 5 (w/o IFMs) and VANTAGE + fuel are based on the Revised Thermal Design Procedure (RTDP) methodology described in Reference 25. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Only the random portion of the plant operating parameter uncertainties is included in the statistical combination. Instrumentation bias is treated as direct DNBR penalty. Based on the DNB uncertainty factor, RTDP design limit DNBR values are determined such that there is at least a 95 percent

probability at a 95 percent confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (i.e., Condition I and II events). Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

The primary DNB correlation used in the analysis of the VANTAGE 5 (w/o IFMs) and VANTAGE + fuel is the WRB-1 DNB correlation which is described in the Indian Point Unit 3 FSAR[18]. The W-3 DNB correlation[28][29] is used where the primary correlation is not applicable.

To maintain DNBR margin to offset penalties such as those due to fuel rod bow and transition core, the safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available margin. Table 4-2 summarizes the available DNBR margin and margin allocation for both fuel types during transition core cycles for the Indian Point Units 3. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility.

4.3 Hydraulic Compatibility

The VANTAGE 5 (w/o IFMs) and VANTAGE + design have been shown to be compatible in the VANTAGE + Fuel Assembly Report[1].

4.4 Effects of Fuel Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in Reference 19, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Internal to the fuel rod, the IFBA and fuel pellet designs are not expected to increase the propensity for fuel rods to bow. External to the VANTAGE + fuel rod, the Inconel non-mixing vane and LPD zircaloy mixing vane mid-grids provide fuel rod support. Additional restraint is provided with the Intermediate Flow Mixer (IFM) grids such that the grid-to-grid spacing in DNB limiting spans is approximately 13 inches compared to approximately 26 inches in the VANTAGE 5. Using the rod bow topical report methods and the NRC approved scaling factor (L^2/EI) results in predicted channel closure in the limiting spans of less than 50% closure; therefore, no rod bow DNBR penalty is required in the 13 inch spans in the VANTAGE + safety analyses.

The maximum rod bow penalties accounted for in the design safety analyses are based on an assembly average burnup of 24,000 MWD/MTU. At burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and buildup of fission product inventory. No additional rod bow penalty is required.

4.5 Transition Core DNB Methodology

This section describes the impact on DNB performance when transitioning from VANTAGE 5 (w/o IFMs) fuel to VANTAGE + fuel. The Westinghouse transition core DNB methodology is given in References 30, 31 and 32 and has been approved by the NRC[33][34]. Using this methodology, transition cores are analyzed as if the entire core consisted of one assembly type (full VANTAGE 5 (w/o IFMs) or full VANTAGE +).

The maximum transition core DNBR penalty for VANTAGE 5 was calculated using the methodology described in References 30 and 31. The transition core penalty is a function of the number of VANTAGE + fuel assemblies in the core.

The VANTAGE + transition core DNBR penalty as a function of VANTAGE + fuel assemblies in the core was calculated using the methodology described in Reference 32. The actual VANTAGE + DNBR penalty used for each transition cycle will be dependent on the number of VANTAGE + fuel assemblies in the core. Sufficient DNBR margin is maintained in the VANTAGE + fuel DNBR analyses to completely offset this transition core penalty.

4.6 Fuel Temperature Analysis

The fuel temperatures used in safety analysis calculations for the VANTAGE 5 (w/o IFMs) and VANTAGE + fuel were calculated with the improved fuel performance code[13]. This code was used to perform both design and safety calculations.

4.7 Conclusion

The thermal hydraulic evaluation of the fuel upgrade for Indian Point Unit 3 has shown that 15x15 VANTAGE 5 (w/o IFMs) and VANTAGE + fuel assemblies are hydraulically compatible and all current thermal-hydraulic design criteria are satisfied.

Table 4-1

Indian Point Unit 3 Thermal and Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters*		Design Parameters	
Reactor Core Heat Output, MWt		3,025	
Reactor Core Heat Output, 10^6, BTU/Hr		10,324	
Heat Generated in Fuel, %		97.4	
Core Pressure, Nominal, psia		2270	
Pressurizer Pressure, Nominal, psia		2250	
Radial Power Distribution	VANTAGE 5	1.654[1+0.3(1-P)]	
	VANTAGE +	1.70[1+0.3(1-P)]	
Minimum DNBR at Nominal Conditions**			
Typical Flow Channel	VANTAGE 5	2.29	
	VANTAGE +	2.68	
Thimble (Cold Wall) Flow Channel	VANTAGE 5	2.19	
	VANTAGE +	2.56	
		RTDP	STDP
Design Limit DNBR			
Typical Flow Channel	VANTAGE 5	1.24	1.17
	VANTAGE +	1.23	1.17
Thimble Flow Channel	VANTAGE 5	1.23	1.17
	VANTAGE +	1.22	1.17
DNB Correlation***		WRB-1	WRB-1
HFP Nominal Coolant Conditions			
Vessel Flow Rate (including Bypass)			
Rate, 10^6 lbm/hr		125.46	122.80
GPM		330,800	323,600
Core Flow Rate (excluding Bypass)			
Rate, 10^6 lbm/hr		120.80	116.40
GPM		318,560	306,772

* Based on 24% uniform SGTP level.

** Based on maximum F_Q of 2.42.

*** The W-3 DNB correlation is used where the primary correlation is not applicable.

Table 4-1

Indian Point Unit 3 Thermal and Hydraulic Design Parameters (continued)

HFP Nominal Coolant Conditions*

Core Flow Area, ft^2		51.54
Core Inlet Mass Velocity, 10^6 lbm/hr-ft^2 (Based on MMF)		2.34
Core Inlet Mass Velocity, 10^6 lbm/hr-ft2 (Based on TDF)		2.26
Pressure Drop Across Core,** psi (Based on 371,600 gpm)	VANTAGE 5	25.9 ± 2.6
	VANTAGE +	26.2 ± 2.7

Thermal and Hydraulic Design Parameters	Design Parameters	
	RTDP	STDP
Nominal Vessel/Core Inlet Temperature, °F	543.6	543.0
Vessel Average Temperature, °F	574.7	574.7
Core Average Temperature, °F	577.2	577.8
Vessel Outlet Temperature, °F	605.8	606.4
Core Outlet Temperature, °F	608.0	609.6
Average Temperature Rise in Vessel, °F	62.2	63.4
Average Temperature Rise in Core, °F	64.4	66.6

Heat Transfer

Active Heat Transfer Surface Area, ft^2	52,000
Average Heat Flux, BTU/hr-ft^2	193,000
Average Linear Power, kw/ft	6.25
Peak Linear Power for Normal Operation, kw/ft***	15.13
Temperature Limit for Prevention of Centerline Melt, °F	4700

* Based on 24% uniform SGTP level.

** Based on 0% SGTP level.

*** Based on maximum F_Q of 2.42.

Table 4-2

DNBR Margin Summary

	VANTAGE 5 (w/o IFMs)	VANTAGE +
DNB Correlation	WRB-1	WRB-1
Correlation Limit	1.17	1.17
Design Limit (Typ/Thm)	1.24/1.23	1.23/1.22
Safety Limit (Typ/Thm)	1.45/1.45	1.54/1.54
DNBR Margin	14.5%	20.0%
Available DNBR Margin	5.2%	9.0%

5.0 ACCIDENT EVALUATION AND ANALYSIS

5.1 Non-LOCA Accidents

5.1.1 Introduction

Section 5.1 addresses the effects of the transition from Westinghouse 15x15 VANTAGE 5 fuel to Westinghouse 15x15 VANTAGE + fuel on the Indian Point Unit 3 FSAR Chapter 14 Non-LOCA Accident Analyses. The methods used in the accident analyses and evaluations herein are those described in Reference 21 and are discussed in further detail in Section 5.1.4. These methods are the same as those applied to previous Indian Point Unit 3 fuel reloads.

The Indian Point Unit 3 licensing basis as reported in the FSAR (Reference 18) includes analyses and evaluations for the following non-LOCA accidents:

FSAR Section	Event Title
14.1.1.	Uncontrolled Control Rod Withdrawal from a Subcritical Condition
14.1.2.	Uncontrolled Control Rod Assembly Withdrawal at Power
14.1.3.	Rod Assembly Misalignment
14.1.4.	Rod Cluster Control Assembly (RCCA) Drop
14.1.5.	Chemical and Volume Control System Malfunction
14.1.6.	Loss of Reactor Coolant Flow
14.1.7.	Startup of an Inactive Reactor Coolant Loop
14.1.8.	Loss of External Electrical Load
14.1.9.	Loss of Normal Feedwater
14.1.10.	Excessive Heat Removal Due to Feedwater System Malfunctions
14.1.11.	Excessive Load Increase Incident
14.1.12.	Loss of All AC Power to the Station Auxiliaries
14.1.13.	Startup Accidents Without Reactor Coolant Pump Operation
14.1.14.	Startup Accident With a Full Pressurizer
14.2.5.	Rupture of a Steam Pipe
14.2.6.	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

All of these non-LOCA accidents have been reviewed to address the transition from VANTAGE 5 (w/o IFMs) fuel to VANTAGE + fuel.

The specific design features associated with the VANTAGE + fuel and the revised safety analysis assumptions that were considered in the non-LOCA safety analyses and evaluations are described in Sections 5.1.2 and 5.1.3, respectively.

All of the accidents which are affected by one or more of the VANTAGE + design features or the revised safety analysis assumptions directly associated with the VANTAGE + fuel, as described in Sections 5.1.2 and 5.1.3 of this report, were reanalyzed. These consist of the following events:

Section	Event Title
5.1.6	Uncontrolled Control Rod Withdrawal from a Subcritical Condition
5.1.7	Uncontrolled Control Rod Assembly Withdrawal at Power
5.1.8	Rod Assembly Misalignment
5.1.9	Rod Cluster Control Assembly (RCCA) Drop
5.1.11	Loss of Reactor Coolant Flow
5.1.13	Loss of External Electrical Load
5.1.15	Excessive Heat Removal Due to Feedwater System Malfunctions
5.1.16	Excessive Load Increase Incident
5.1.20	Rupture of a Steam Pipe (Core Response)
5.1.21	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

The remaining FSAR non-LOCA accidents were evaluated for the VANTAGE + transition and the revised safety analysis assumptions directly associated with the VANTAGE + fuel as discussed in the following sections:

Section	Event Title
5.1.10	Chemical and Volume Control System Malfunction
5.1.12	Startup of an Inactive Reactor Coolant Loop
5.1.14	Loss of Normal Feedwater
5.1.17	Loss of All AC Power to the Station Auxiliaries
5.1.18	Startup Accidents Without Reactor Coolant Pump Operation
5.1.19	Startup Accident With a Full Pressurizer
5.1.20	Rupture of a Steam Pipe (Mass & Energy Releases Inside Containment)

To be consistent with the current Indian Point Unit 3 licensing basis accident analyses, the analyses and evaluations performed for the transition to VANTAGE + fuel were also performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq$ 24%; and 2) asymmetric steam generator tube plugging conditions with an average steam

generator tube plugging level of ≤ 24% and with a maximum steam generator tube plugging level of 30% in any one steam generator. The events which include specific analyses to address asymmetric steam generator tube plugging include:

FSAR Section	Event Title
14.1.2	Uncontrolled Control Rod Assembly Withdrawal at Power
14.1.4	Rod Cluster Control Assembly (RCCA) Drop
14.1.6	Loss of Reactor Coolant Flow
14.1.9	Loss of Normal Feedwater
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions
14.1.12	Loss of All AC Power to the Station Auxiliaries
14.2.5	Rupture of a Steam Pipe

Note that asymmetric steam generator tube plugging conditions were originally addressed for Indian Point Unit 3 in Reference 35.

5.1.2 VANTAGE + Design Features

The design features of the VANTAGE + Fuel Design that are considered in the non-LOCA analysis and evaluations include the following features:

- Current VANTAGE 5 (w/o IFMs) Fuel Features
  - Reconstitutable Top Nozzle (RTN)
  - Extended Burnup Fuel Assembly Design
  - Extreme Low Leakage Loading Pattern
  - Enriched Integral Fuel Burnable Absorbers (IFBAs)
  - Debris Filter Bottom Nozzle (DFBN)
  - Axial Blankets
- ZIRLO™ Fuel Cladding
- Low Pressure Drop (LPD) Mid-grids
- Integral Flow Mixing grids (IFMs)
- ZIRLO™ guide thimble & instrument tubes
- Variable Pitch Fuel Rod Plenum Spring
- Mid-enriched Annular Fuel Pellets in Axial Blanket
- Fuel Assembly & Fuel Rod Dimensional Modifications
- Low Cobalt Top and Bottom Nozzles
- Performance Plus Debris Mitigation Features

- Coated Cladding (Pre-oxidized ZIRLO™ cladding)
- Protective Bottom Grid
- Debris-Mitigation Bottom Fuel Rod Endplug
- Gripable Top End Plug

A detailed description of the VANTAGE + design features is contained in Section 2.0. A brief description of each of these and their consideration in the non-LOCA safety analyses follows.

Current VANTAGE 5 (w/o IFMs) Fuel Features

The 15x15 VANTAGE + Fuel Design for Indian Point Unit 3 includes the current VANTAGE 5 Fuel Features listed above. With the exception of the ZIRLO™ fuel cladding and an increase on the Extended Burnup, the effects of each of these features on the non-LOCA safety analyses were evaluated and approved by the NRC as documented in Reference 36.

The use of ZIRLO™ fuel cladding was implemented in the VANTAGE 5 (w/o IFMs) Fuel Design beginning in Cycle 9 (the third and final transition cycle from 15x15 OFA fuel) after issuance of the VANTAGE 5 (w/o IFMs) SER. With respect to the Non-LOCA accident analyses, the use of ZIRLO™ fuel cladding only has an effect on only two analyses; the Locked Rotor/Shaft Break analysis and the RCCA Ejection analysis.

For the Locked Rotor/Shaft Break event, the use of ZIRLO™ fuel cladding results in a very small increase in peak clad temperature (approximately 2 °F). For the RCCA Ejection event, ZIRLO™ fuel cladding results in a negligible benefit in both the fraction of fuel melting at the hot spot and the fuel peak stored energy in the RCCA Ejection analysis.

The effect of ZIRLO™ fuel cladding on the accident analyses for the Locked Rotor/Shaft Break events and the RCCA Ejection event were evaluated and approved by the NRC as documented in Reference 37 and is considered in the safety analyses for these events as described in Sections 5.1.11 and 5.1.21, respectively.

With respect to the Extended Burnup Fuel Assembly Design, the VANTAGE + fuel design includes an increase in the region average discharge burnup from 40,000 + to 50,000 + MWD/MTU. The effects of the increase in extended burnup on the performance of the fuel are included in the safety analyses via the reload safety analysis parameters (e.g., fuel temperatures, peaking factors, etc.) which are taken into

account in the reload design process.

Mid-enriched Annular Fuel Pellets in Axial Blankets and Enriched IFBAs

Axial blankets reduce power at the ends of the rod which increases axial peaking toward the middle of the rod. Used alone, axial blankets reduce DNB margin, but the effect may be offset by the presence of Integral Fuel Burnable Absorbers (IFBAs) which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and the time in core life. The effects on the reload safety analysis parameters due to axial blankets, including annular fuel pellets in the axial blanket, and IFBAs, including enriched IFBAs, are taken into account in the reload design process. The axial and radial power distribution assumptions used in the safety analyses kinetics calculations have been determined to be applicable for evaluating the axial blankets and IFBAs in the Indian Point Unit 3 VANTAGE + Fuel Design.

LPD Mid-grids, IFM, ZIRLO™ guide thimbles & instrument tubes, Low Cobalt Top and Bottom Nozzles and Fuel Assembly & Fuel Rod Dimensional Modifications

The Indian Point Unit 3 15x15 VANTAGE + Fuel Design incorporates the use of ZIRLO™ LPD mid-grids, IFMs, guide thimbles and instrument tubes, and includes minor fuel assembly and fuel rod dimensional modifications to accommodate the Extended Burnup Fuel Assembly Design.

With respect to the non-LOCA accident analyses, the use of ZIRLO™ guide thimbles and instrument tubes, low cobalt top and bottom nozzles and the minor fuel assembly and fuel rod dimensional modifications have no direct effect on the analysis results since the characteristics of these features are not specifically modeled in the transient analyses. Any effects of these items on the performance of the fuel are included in the safety analyses via the reload safety analysis parameters (e.g., fuel temperatures, flow rates, pressure drops) which are taken into account in the reload design process.

The use of the LPD mid-grids in the VANTAGE + Fuel Design are primarily for the purpose of offsetting the increase in flow resistance associated with the introduction of the IFMs. Therefore, with respect to the parameters which affect the non-LOCA accident analyses, the VANTAGE + Fuel Design is hydraulically compatible with the resident VANTAGE 5 (w/o IFMs) Fuel Design and has no direct effect on the non-LOCA safety analyses. Any localized assembly-to-assembly hydraulic differences which may affect the performance of the fuel are included in the safety analyses via the reload safety analysis parameters (e.g., DNBR limits, flow rates, pressure drops) which are taken into account in the reload

design process and documented in the Reload Safety Evaluation (RSE).

With respect to the implementation of IFMs in the VANTAGE + Fuel Design, it should be noted that although the use of IFMs provides a DNB benefit caused by enhanced flow mixing, the benefit is not fully realized in the safety analysis since the resident VANTAGE 5 (w/o IFMs) Fuel Design does not include IFMs and, therefore, is more limiting with respect to DNB. As a result, the Core Thermal Limits and resulting OTΔT and OPΔT reactor trip setpoints (see Section 5.1.3) applicable for the transition to VANTAGE + Fuel are currently based on the DNB performance of the more limiting VANTAGE 5 (w/o IFMs) fuel. Once the transition to a full core of VANTAGE + fuel is completed, the DNB benefit associated with the IFMs can be fully realized. However, this would be addressed at that time by a separate licensing submittal requesting a revision to the appropriate Technical Specifications.

Variable Pitch Fuel Rod Plenum Spring

The optimized fuel rod plenum spring feature of the VANTAGE + Fuel Design is primarily to provide increased fuel rod plenum volume which benefits rod internal pressure concerns. With respect to the non-LOCA accident analyses, the use of the optimized fuel rod plenum spring has no direct effect on the analysis results since the characteristics of this features are not specifically modeled in the transient analyses. Any effects of this spring design on the performance of the fuel is included in the safety analyses via the reload safety analysis parameters (e.g., fuel temperatures) which are taken into account in the reload design process.

Performance Plus Debris Mitigation Features

The 15x15 VANTAGE + Fuel Design for Indian Point Unit 3 may also incorporate the use of Performance Plus Debris Mitigation features listed above.

Pre-oxidized fuel rod cladding has a protective zirconium dioxide coating on the outer surface of the bottom six inches of the fuel rod. The purpose is to provide increased resistance to debris-induced fretting wear, thereby improving fuel reliability. The potential effects of this coating on the non-LOCA safety analyses would result from any changes in the friction induced pressure drop up the fuel rod or any changes to the heat transfer characteristics due to the oxide coating. However, due to the location of the coating, there is no effect on the heat transfer characteristics from the clad outer surface to the coolant. Furthermore, the pre-oxidation coating of the cladding does not result in any direct changes to the fuel assembly hydraulic or mechanical properties. Any effects of these items on the performance of

the fuel are included in the safety analyses via the reload safety analysis parameters (e.g., fuel temperatures, flow rates, pressure drops) which are taken into account in the reload design process.

Like the minor fuel assembly and fuel rod dimensional modifications of the VANTAGE + Fuel design, the Debris-Mitigation Bottom Fuel Rod Endplug and Gripable Top End Plug design features have no direct effect on the analysis results since the characteristics of these features are not specifically modeled in the transient analyses. Any effects of these items on the performance of the fuel are included in the safety analyses via the reload safety analysis parameters which are taken into account in the reload design process.

The Protective Bottom Grid is an inconel grid, similar in design to the IFMs, which is used in the bottom span of the fuel assembly. The purpose of the Protective Bottom Grid is to provide an additional entrance barrier to debris entering the fuel assembly. Since the Protective Bottom Grid is located in close proximity to the bottom nozzle, it is considered as part of the bottom support and has no direct effect on the fuel assembly geometry as it relates to the non-LOCA safety analyses. The effect of the Protective Bottom Grid on the fuel assembly hydraulic characteristics are described in Section 4.0 and have been conservatively included in the thermal and hydraulic performance characteristics used in the non-LOCA safety analyses and evaluations performed for the VANTAGE + fuel.

5.1.3 Revised Safety Analysis Assumptions

Listed below are the analysis assumptions considered in the non-LOCA accident analyses that represent a departure from those currently used for Indian Point Unit 3.

- Revised Thermal Design Procedure (RTDP)
- Revised OTΔT and OPΔT reactor trip setpoints
- Increased RCCA Scram Time
- Increased $F_{\Delta H}$
- Increased F_Q
- Increased Uncertainties for 24 month Reload Cycles

A brief description of each of these assumptions follows.

RTDP

The calculational method utilized to meet the DNB design basis is the Revised Thermal Design Procedure (RTDP), discussed in Reference 25. Uncertainties in plant operating parameters, peaking factors, and the DNB correlation are statistically treated such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than the applicable limits as discussed in Section 4.2. Since the parameter uncertainties are considered in determining the design DNBR value, nominal input parameters without uncertainties are used in the plant safety analyses performed using RTDP. The specific uncertainties and their magnitude are described in further detail in Section 4.0. The use of RTDP is a change from the current licensing basis which uses the Improved Thermal Design Procedure (ITDP).

Revised OTΔT and OPΔT reactor trip setpoints

As a result of implementing the RTDP and the increase in $F_{\Delta H}$, the core thermal safety limits are revised and result in a change to the OTΔT and OPΔT reactor protection trip setpoints. The revised core thermal safety limits are shown in Technical Specification Figure 2.1-1 included in Appendix A and are based on the resident VANTAGE 5 (w/o IFMs) fuel design with a maximum full power $F_{\Delta H}$ of 1.654 (with uncertainties). The corresponding revision to the OTΔT and OPΔT reactor protection trip setpoints are reflected in Technical Specification Section 2.3 also included in Appendix A.

Increased RCCA Scram Time

For the transition to VANTAGE + fuel, the RCCA scram time from the release of the RCCA gripper coil to dashpot has increased from the current value of 2.4 seconds to a value of 2.7 seconds. This RCCA scram time increase is reflected in Technical Specification 3.10-8 included in Appendix A.

This increased scram time of 2.7 seconds has been considered for all of the non-LOCA events and has been explicitly modeled in all the events reanalyzed (see Section 5.5.1) which includes those transients which are sensitive to scram time (i.e., fast transients of short duration).

For events evaluated for the VANTAGE + transition, the change in scram time has no significant effect on the transient results of slow events (e.g., Loss of Normal Feedwater, Loss of All AC Power to the Station Auxiliaries).

Increased $F_{\Delta H}$

The $F_{\Delta H}$ values (with uncertainties) for the VANTAGE 5 (w/o IFMs) and VANTAGE + fuel during the transition cycles are 1.65 and 1.70, respectively. For the VANTAGE 5 (w/o IFMs) fuel, this is an increase from the previous design $F_{\Delta H}$ (with uncertainties) of 1.62.

Revisions to the Core Operating Limits Report (COLR) are required to reflect the use of these two fuel dependent $F_{\Delta H}$ limits.

With the exception of the Startup of an Inactive Reactor Coolant Loop, all of the non-LOCA events which are sensitive to and explicitly consider $F_{\Delta H}$ in the analysis have been reanalyzed and, therefore, appropriately address the effects of this change. The evaluation for the Startup of an Inactive Reactor Coolant Loop is provided in Section 5.1.12.

Increased F_Q

The F_Q (with uncertainties) applicable for the VANTAGE 5 (w/o IFMs) and VANTAGE + fuel during the transition cycles is 2.42. This is an increase from the previous F_Q (with uncertainties) of 2.32. All of the non-LOCA events which explicitly consider F_Q in the analysis have been reanalyzed and, therefore, appropriately address the effect of this change.

Increased Uncertainties for 24 month Reload Cycles

In support of the New York Power Authority program to extend surveillance intervals in support of 24 month operating cycles, changes in various protection system setpoints and initial conditions have been considered to allow for increased uncertainties. Therefore, these changes have been considered in the events reanalyzed in support of the VANTAGE + Fuel features and other VANTAGE + related revisions to the safety analysis assumptions as described in this section.

The uncertainties on initial conditions assumed in the non-LOCA analyses for the VANTAGE + transition are as follows:

	Non-RTDP	RTDP
Reactor Coolant Temperature	± 7 °F	± 4.7 °F*
RCS Pressure	± 60 psi	± 53.3 psi
Power	± 2%	± 2.0%
Pressurizer Level	± 7% level span	± 7.0% level span
Steam Generator Level	± 10% NRS**	± 10.0% NRS**

* Plus a 0.25 °F Bias

** Narrow Range Span

5.1.4 Non-LOCA Safety Evaluation Methodology

The non-LOCA safety evaluation process is described in Reference 21. The process determines if a core configuration is bounded by existing safety analyses in order to confirm that applicable safety criteria are satisfied. The methodology systematically identifies parameter changes on a cycle-by-cycle basis which may invalidate existing safety analysis assumptions and identifies the transients which require reevaluation. This methodology is applicable to the evaluation of VANTAGE + transition and full VANTAGE + cores.

Any required reevaluation identified by the reload methodology is one of two types. If the identified parameter is only "slightly" out of bounds, or the transient is relatively insensitive to that parameter, a simple evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated.

Alternatively, should the deviation be "large" and/or expected to have a significantly or not easily quantifiable effect on the transients, reanalyses are required. The reanalysis approach will utilize the methods which have been used in previous submittals to the NRC. These methods are those which have been presented in FSARs, subsequent submittals to the NRC for a specific plant, reference SARs, or report submittals for NRC approval.

The key safety parameters are documented in Reference 21. Values of these safety parameters which bound both fuel types (VANTAGE 5 (w/o IFMs) and VANTAGE +) were assumed in the safety analyses. For subsequent fuel reloads, the key safety parameters will be evaluated to determine if violations of these values exist. Re-evaluation of the affected transients would take place as described in Reference 21.

5.1.5 Computer Codes Used

Summaries of the principal computer codes used in the transient analyses are given below.

FACTRAN:

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the

time-dependent coolant parameters (pressure, flow, temperature, density). The code uses a fuel model which simultaneously exhibits the following features:

a) A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

b) Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

c) The necessary calculations to handle post-departure from nucleate boiling (DNB) transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel.

FACTRAN is further discussed in Reference 38.

LOFTRAN:

The LOFTRAN program is used for transient response studies of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves are also considered in the program. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on high neutron flux, OTΔT, OPΔT, high and low pressure, low RCS flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The Emergeny Core Cooling System (ECCS), including the accumulators, is also modeled.

LOFTRAN also has the capability of calculating the transient value of DNBR based on input from the DNB core limits. The DNB core limits are the locus of maximum allowable conditions, in terms of inlet temperature as a function of core power at various RCS pressures, corresponding to the applicable safety analysis DNBR limit and are calculated for both the typical and thimble cells.

LOFTRAN is further discussed in Reference 39.

TWINKLE:

The TWINKLE program is a multi-dimensional neutron kinetics code, which was patterned after steady-state codes used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, or three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts various driving functions versus time as input. Examples of these driving functions versus time include inlet temperature, pressure, flow, boron concentration, and control rod motion. Various edits are provided; e.g., channel-wise power, axial offset, enthalpy, volumetric surge, point-wise power and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 40.

THINC:

The THINC-IV computer program is used to perform thermal-hydraulic calculations. The THINC-IV code is used to calculate coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core under all expected operating conditions. The THINC-IV code is described in detail in References 41 and 42, including models and correlations used.

5.1.6 Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

Introduction:

An uncontrolled rod cluster control assembly (RCCA) withdrawal accident is defined by an addition of reactivity to the reactor core caused by uncontrolled withdrawal of RCCA banks resulting in a power excursion. While the occurrence of a transient of this type is highly unlikely, such a transient could be caused by a malfunction of the reactor control system or the control rod drive system, or by operator

error. This could occur with the reactor either subcritical, at hot zero power, or at power. The "at power" case is discussed in Section 5.1.7.

The RCCA drive mechanisms are wired into preselected bank configurations that are not altered during core life. These circuits prevent RCCAs from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled so that no more than two banks can be withdrawn at any time, and then only in the proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type; coil actuation is sequenced to provide variable speed travel. The analysis of the maximum reactivity insertion rate includes the assumption of the simultaneous withdrawal of the two sequential banks having the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very rapid flux increase terminated by the negative Doppler reactivity feedback effect. This self-limitation of the power burst is of primary importance since it limits the power to a tolerable level prior to protective action. Should a continuous RCCA withdrawal occur, and the subsequent source and intermediate range indication and annunciators are ignored, the following protective functions would be available to terminate the transient.

a. Source Range High Neutron Flux Reactor Trip
b. Intermediate Range High Neutron Flux Reactor Trip
c. Power Range High Neutron Flux Reactor Trip (Low Setting)
d. Power Range High Neutron Flux Reactor Trip (High Setting)
e. Power Range Control Rod Stop
f. High Positive Nuclear Flux Rate Reactor Trip

The Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition event is a Condition II event as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants". A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious fault, i.e., a Condition III or IV category event.

The applicable safety analysis licensing basis acceptance criteria for the Condition II Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition event for Indian Point Unit 3 are:

a. Pressure in the reactor coolant and main steam systems should not exceed 110% of their respective design values.

b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit.

c. Fuel centerline temperatures should remain below the minimum temperature at which fuel melting would occur.

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

Method of Analysis and Assumptions:

The analysis of the Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition event is performed in three stages. First, the TWINKLE computer code is used to calculate the core average nuclear power transient, including the various feedback effects, i.e., Doppler and moderator reactivity. Next, the FACTRAN computer code uses the average nuclear power transient calculated by TWINKLE and performs a fuel rod heat transfer calculation to determine the heat flux and fuel temperature transients. Finally, the average heat flux calculated by FACTRAN is used in the THINC computer code for transient DNBR calculations. Refer to Section 5.1.5 for a complete description of these computer codes.

In order to give conservative results for the Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition analysis, the following assumptions are made.

a. Since 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 Doppler reactivity feedback, a conservatively low (absolute magnitude) value for the Doppler power defect is used.

b. The effect of moderator reactivity is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant. However, after the initial neutron flux peak, the moderator temperature reactivity coefficient affects the succeeding rate of power increase. A highly conservative value of the moderator temperature coefficient is assumed in the analysis to yield the maximum peak heat flux.

c. The analysis assumes the reactor to be at hot zero power conditions with a vessel average temperature of 547 °F. This assumption is more conservative than that of a lower initial system temperature (i.e., shutdown conditions). The higher initial system temperature enhances fuel-to-coolant heat transfer and reduces the Doppler power defect, resulting in a higher peak heat flux.

d. Two reactor coolant pumps are assumed to be in operation (Mode 3 Technical Specification allowed operation). This is conservative with respect to the DNB transient.

e. The positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the two sequential RCCA banks having the greatest combined worth at the maximum withdrawal speed.

f. The analysis assumes that the reactor is initially critical at 10^{-9} fraction of nominal power, which is below the power level expected for any shutdown condition. The combination of highest reactivity insertion rate and lowest initial power produces the highest heat flux.

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

Initial Conditions:

The Revised Thermal Design Procedure (RTDP, see Section 5.1.3) is not used in the analysis of this event. Standard Thermal Design Procedure methods (maximum uncertainties in initial conditions) are used instead. Since the event is analyzed from hot zero power, the steady-state uncertainties on core power and RCS average temperature are not considered in defining the initial conditions.

Initial Condition	Value Used in Analysis
Core Power	10^{-9} fraction of nominal
Pressurizer Pressure	2190 psia
Reactor Vessel T_{in}	547.0 °F
Reactor Vessel T_{avg}	547.0 °F
Core Flow	148856 gpm (46% of TDF reflecting 2 RCPs)

Control Systems:

Control systems are assumed to function only if their operation causes more severe accident results. For

the Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition analysis, no control functions are assumed to operate.

Protection Systems:

Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrumentation error, setpoint error, delay for trip signal actuation, and delay for control rod assembly release is taken into account. The analysis assumes a 10% uncertainty in the power range flux trip setpoint (low setting), raising it from the nominal value of 25% nominal to a value of 35% nominal. No credit is taken for the source and intermediate range protection. The reactor trip time delay from reactor trip signal actuation to RCCA release is assumed to be 0.5 seconds.

Reactivity Modeling:

The Uncontrolled RCCA Withdrawal from a Subcritical Condition accident results in a rapid nuclear power excursion which is terminated initially by Doppler reactivity feedback, and ultimately by reactor trip. Reactivity feedback parameters are chosen to yield the most severe power burst. These include a conservatively small (absolute value) Doppler power defect of 955 pcm at full power and a maximum delayed neutron fraction of 0.007. A total trip reactivity of -3% $\Delta k/k$ excluding the highest worth rod is assumed with a scram time of 2.7 seconds from beginning of rod motion until the dashpot is reached.

Heat Transfer Modeling:

A conservatively high fuel rod gap heat transfer coefficient (10,000 Btu/hr-ft^2-°F) and conservatively low hot channel factors (1.0) are assumed for the DNBR evaluation. This maximizes the heat flux during the event, which yields a more severe DNBR transient. For the hot spot fuel temperature calculation, a conservatively low fuel rod gap heat transfer coefficient (500 Btu/hr-ft^2-°F) and conservatively high hot channel factors (6.64) are assumed. This maximizes the fuel and clad temperatures resulting from the nuclear power transient.

Results:

Figure 5.1.6-1 shows the nuclear flux transient for the Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition event. The neutron flux overshoots the full power nominal value for a very short period of time; therefore, the energy release and fuel temperature increase are relatively small. The heat

flux response used in the DNBR evaluation is shown in Figure 5.1.6-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the nominal full power value. Figure 5.1.6-3 shows the transient response of the hot spot fuel centerline, fuel average, and cladding inner temperatures. These temperatures remain below their respective nominal full power values at all times during the event. The minimum DNBR remains above the safety analysis limit at all times.

Table 5.1.6-1 presents the calculated sequence of events. After reactor trip, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal shutdown procedures.

Conclusions:

In the event of an Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition, the core and the RCS are not adversely affected since the combinations of thermal power and coolant temperature and flow result in a DNBR greater than the limit value. Thus, no fuel or clad damage is predicted as a result of this transient.

Table 5.1.6-1

Sequence of Events

Uncontrolled RCCA Bank Withdrawal

From a Subcritical Condition

EVENT	TIME OF EVENT (seconds)
Initiation of Uncontrolled RCCA Withdrawal	0.0
Power Range High Neutron Flux Reactor Trip Setpoint (low setting) Reached (35%)	9.8
Peak Nuclear Power Occurs	9.9
Rods Begin to Fall	10.3
Peak Heat Flux Occurs	11.8
Minimum DNBR Occurs	11.8
Peak Fuel Cladding Inner Temperature Occurs	12.3
Peak Fuel Average Temperature Occurs	12.5
Peak Fuel Centerline Temperature Occurs	13.0

Figure 5.1.6-1

Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

Nuclear Power vs. Time

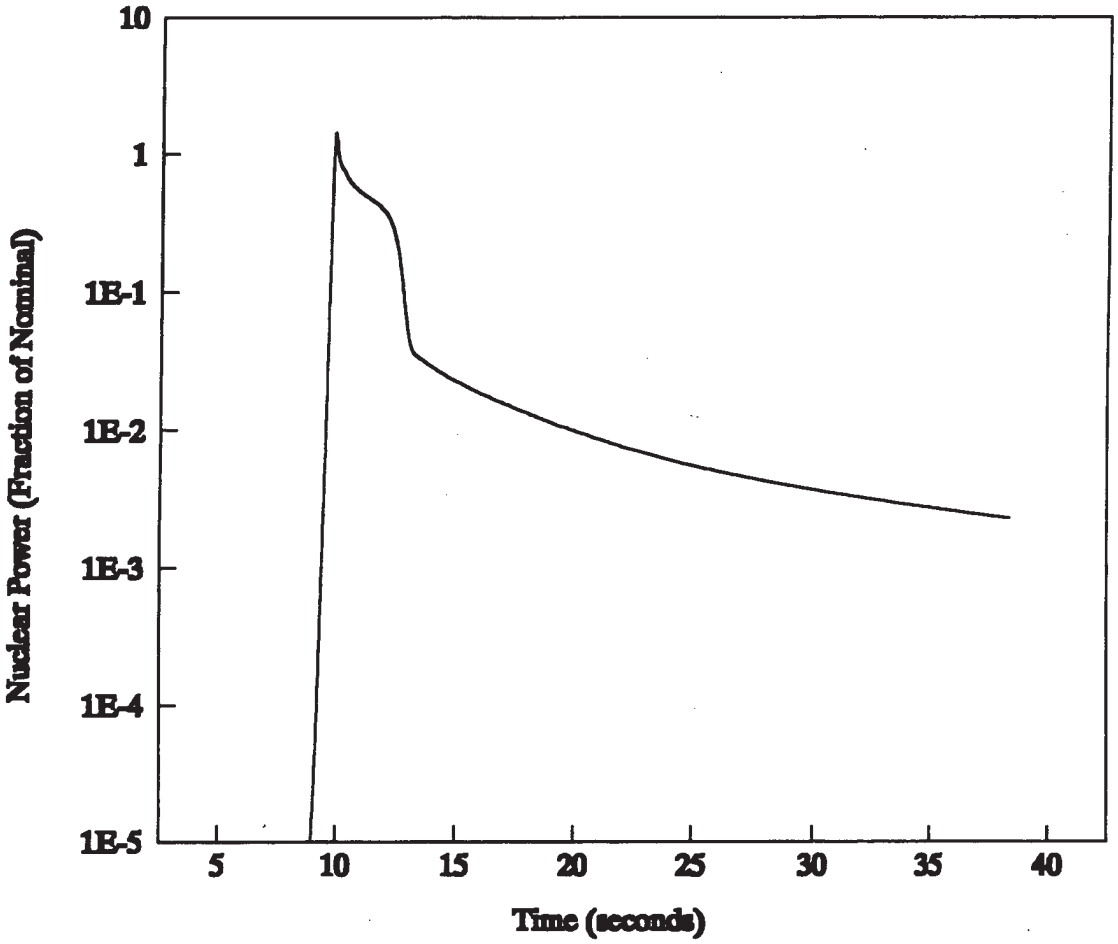


Figure 5.1.6-2

Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

Average Heat Flux vs. Time

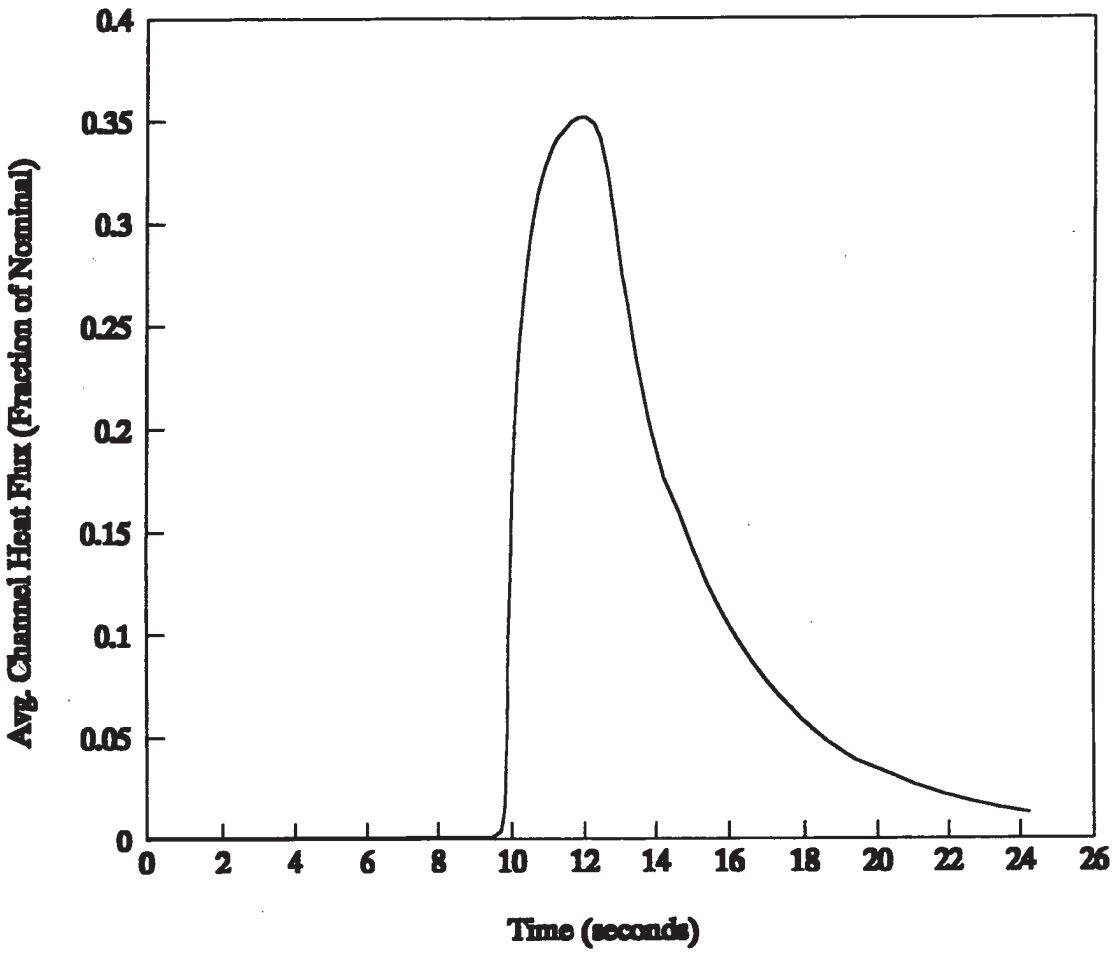
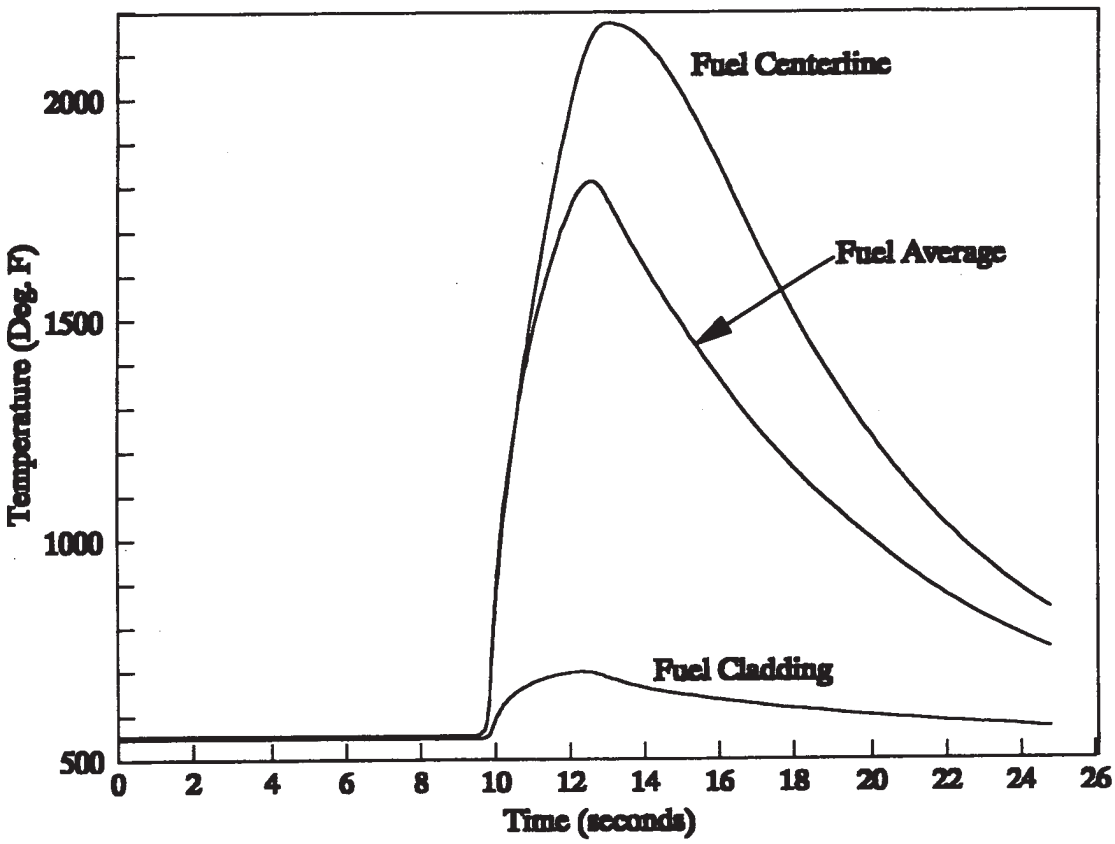


Figure 5.1.6-3

Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

Fuel and Clad Temperatures vs. Time



5.1.7 Uncontrolled RCCA Withdrawal at Power

Introduction:

The "Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power" event is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of RCCA banks when the core is above the no-load condition. The reactivity insertion resulting from the bank (or banks) withdrawal will cause an increase in core nuclear power and subsequent increase in core heat flux. An RCCA withdrawal can occur with the reactor subcritical, at hot zero power, or at power. The uncontrolled RCCA bank at power event is analyzed for Mode 1 (power operation). The uncontrolled RCCA bank withdrawal from subcritical or low power condition is considered as an independent event in FSAR Section 14.1.1 and in Section 5.1.6 of this report.

The event is simulated by modelling a constant rate of reactivity insertion starting at time zero and continuing until a reactor trip occurs. The analysis assumes a spectrum of possible reactivity insertion rates up to a maximum positive reactivity insertion rate greater than that occurring with the simultaneous withdrawal, at maximum speed, of two sequential RCCA banks having the maximum worth. The minimum reactivity insertion rate considered is less than 1 pcm/second.

Unless the transient RCS response to the RCCA withdrawal event is terminated by manual or automatic action, the power mismatch and resultant temperature rise could eventually result in departure from nucleate boiling (DNB) and/or fuel centerline melt. Additionally, the increase in RCS temperature caused by this event will increase the RCS pressure, and if left unchecked, could challenge the integrity of the Reactor Coolant System Pressure Boundary or the Main Steam System Pressure Boundary.

To avert the core damage that might otherwise result from this event, the reactor protection system is designed to automatically terminate any such event before the departure from nucleate boiling ratio (DNBR) falls below the limit value, the fuel rod kw/ft limit is reached, the peak pressures exceed their respective limits, or the pressurizer fills. Depending on the initial power level and the rate of reactivity insertion, the reactor may be tripped and the RCCA withdrawal terminated by any of the following trip signals.

a. power range neutron flux
b. OTΔT
c. OPΔT
d. high pressurizer pressure
e. high pressurizer water level

The uncontrolled RCCA withdrawal at power event is classified as a Condition II event as defined by the ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary PWR Plants." The major hazards associated with an unmitigated and uncontrolled RCCA bank(s) withdrawal at power are the possibility of DNB, filling the pressurizer and an increase in RCS and secondary steam pressures, resulting from the power excursion and subsequent increase in RCS and core temperatures.

The safety analysis criteria for this event are as follows:

1) The pressure in the reactor coolant system and the steam generators should be maintained below 110% of their respective design pressures (i.e., 2748.5 psia and 1208.5 psia, respectively),

2) The critical heat flux and the fuel temperature and clad strain limits should not be exceeded. The peak linear heat generation rate (expressed in kw/ft) should not exceed a value which would cause fuel centerline melting. This is ensured by demonstrating that the minimum DNB ratio does not go below the safety analysis limit values as provided in Table 4-3. Meeting the DNBR limit also ensures that offsite dose requirements of 10 CFR 20 are met.

3) An incident of moderate frequency (Condition II) should not generate a more serious plant condition without other faults occurring independently.

Method of Analysis and Assumptions:

The uncontrolled RCCA withdrawal at power event is analyzed to show that: 1) the integrity of the core is maintained by the reactor protection system as the DNBR remains above the safety analysis limit value; 2) the peak RCS and secondary system pressures remain below the accident analysis pressure limits; and 3) the pressurizer does not reach a water-solid condition and result in water relief through the pressurizer relief and safety valves. Of these, the primary concern for this event is DNB and assuring that the DNBR limit is met.

The RCCA withdrawal at power transient is analyzed with the LOFTRAN computer program (see Section 5.1.5). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator relief and safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

The transient responses for the RCCA bank withdrawal at power event were analyzed for a large number of cases with initial power levels of 100%, 60%, and 10% power. A spectrum of positive reactivity insertion rates from a minimum value (less than 1 pcm/sec) up to a maximum value (100 pcm/sec) greater than that occurring with the simultaneous withdrawal, at maximum speed, of two sequential RCCA banks having the maximum worth were analyzed for each power level. Each combination of power and reactivity insertion rate was analyzed for limiting core reactivity conditions of minimum (BOL) and maximum (EOL) reactivity feedback conditions.

The analyses were performed considering the transition from VANTAGE 5 (w/o IFMs) to VANTAGE + fuel at a nominal core power of 3025 MWt and considered the other design changes associated with the VANTAGE + transition as discussed in Section 5.1.2 and 5.1.3.

The RTDP (see Section 5.1.3) was used in the analysis so the initial conditions for power, RCS pressure, and T_{avg} are at the nominal values. In performing the analysis, the following assumptions are made to assure bounding results are obtained for all possible normal operational conditions:

1) Reactivity Coefficients - Two spectrums are analyzed:

   a. Minimum Reactivity Feedback. A least negative moderator density coefficient of reactivity is assumed, corresponding to beginning of core life core conditions. A conservatively small (absolute magnitude) Doppler power coefficient, variable with core power, was used in the analysis.

   b. Maximum Reactivity Feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

2) A conservatively high setpoint of 118% of full power was assumed for the High Neutron Flux reactor trip. The OTΔT reactor trip function includes all adverse instrumentation and setpoint errors. Delays for trip actuations are assumed to be the maximum values; 0.5 seconds for the High Neutron Flux reactor trip, 2.0 seconds for the OTΔT reactor trip.

3) The trip reactivity is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled.

4) A range of reactivity insertion rates is examined. The maximum positive reactivity

insertion rate is greater than that for the simultaneous withdrawal of the two control banks having the maximum combined worth at maximum speed (100 pcm/sec).

5) Power levels of 10%, 60% and 100% power are considered.

Results:

Figures 5.1.7-1 through 5.1.7-6 show the transient response for a rapid RCCA withdrawal (80 pcm/sec) incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures 5.1.7-7 through 5.1.7-12. Reactor trip on OTΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the safety analysis limit value of 1.54.

Figure 5.1.7-13 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that the high neutron flux and OTΔT reactor trip channels provide protection over the whole range of reactivity insertion rates. The minimum DNBR is never less than the safety analysis limit value of 1.54.

Figures 5.1.7-14 and 5.1.7-15 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents initiating from 60 and 10 percent power levels, respectively, for minimum and maximum reactivity feedback. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the OTΔT trip is effective is increased. In all cases the DNBR remains above the safety analysis limit value of 1.54.

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

Referring to Figure 5.1.7-15, for example, it is noted that:

1) For reactivity insertion rates above ~ 30 pcm/sec reactor trip is initiated by the high neutron flux trip for the minimum 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 significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux. Minimum DNBR during the transient thus decreases with decreasing insertion rate.

2) For reactivity insertion rates below ~ 30 pcm/sec the OTΔT trip terminates the transient. The OTΔT 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.

3) For reactivity insertion rates between ~ 30 pcm/sec and ~ 7 pcm/sec the effectiveness (in terms of increased minimum DNBR) of the OTΔT trip increases due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

4) For reactivity insertion rates less than ~ 7 pcm/sec, 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 sink for the Reactor Coolant System, sharply decreases the rate of increase of Reactor Coolant System average temperature.

For transients initiated from higher power levels (for example, see Figure 5.1.5-11) the effect described in item 4 above, which results in the sharp peak in minimum DNBR at approximately 7 pcm/sec, does not occur since the steam generator safety valves are not actuated prior to trip.

Since 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. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis) and the peak fuel centerline temperature 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 OTΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value and the peak fuel centerline temperature remains below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, the ability of the primary coolant to remove heat from the fuel rod is not reduced.

The calculated sequence of events for an RCCA bank withdrawal from full power for a large and small reactivity insertion rate are shown in Table 5.1.7-1. These sequence of events are for the cases assuming minimum reactivity feedback conditions. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

For all the cases analyzed, the peak pressurizer volume remains below the maximum pressurizer volume and pressurizer filling does not occur. A review of the results shows that the limiting cases with respect to pressurizer filling are those initiated from less than full power and having a slow RCCA withdrawal rate. These cases typically trip on OTΔT (as opposed to high neutron flux). Although a high pressurizer water level reactor trip function exists and is expected to provide a reactor trip (protection against pressurizer filling), no credit for this reactor trip function is taken in the analysis.

Conclusions:

The high neutron flux and OTΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum calculated DNBR is always greater than the safety analysis limit value and pressurizer filling does not occur. In addition, peak pressures in the RCS and secondary steam system do not exceed 110% of their respective design pressures.

Table 5.1.7-1

Time Sequence of Events

for

Uncontrolled RCCA Withdrawal at Full Power

Accident	Event	Time (sec)
Uncontrolled RCCA bank withdrawal at full power and minimum reactivity feedback		
1. Case A	Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (80 pcm/sec)	0
	Power range high neutron flux high trip point reached	1.4
	Rods begin to fall into core	1.9
	Minimum DNBR occurs	3.1
2. Case B	Initiation of uncontrolled RCCA withdrawal at a small reactivity insertion rate (1 pcm/sec)	0
	OTΔT reactor trip signal initiated	71.1
	Rods begin to fall into core	73.1
	Minimum DNBR occurs	73.5

Figure 5.1.7-1

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 80 pcm/sec Withdrawal Rate

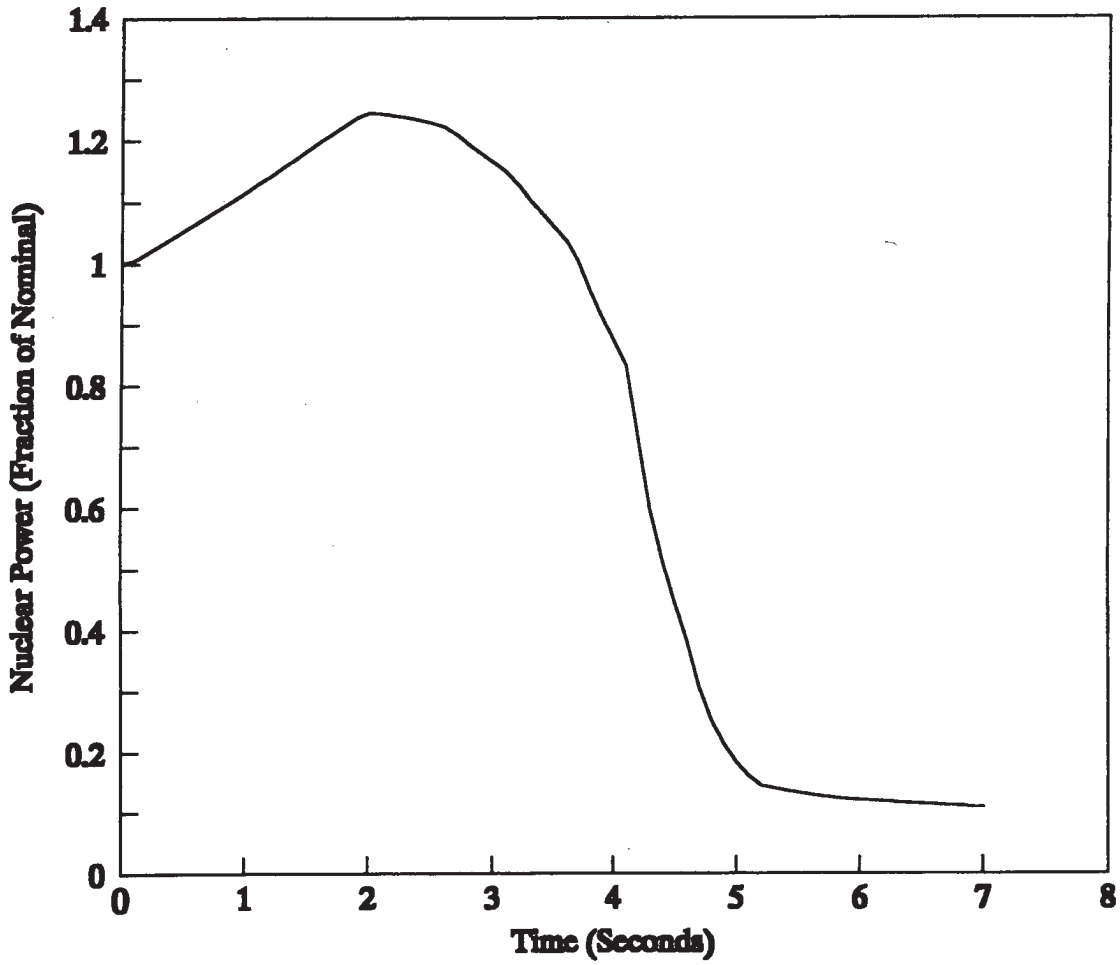


Figure 5.1.7-2

Core Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 80 pcm/sec Withdrawal Rate

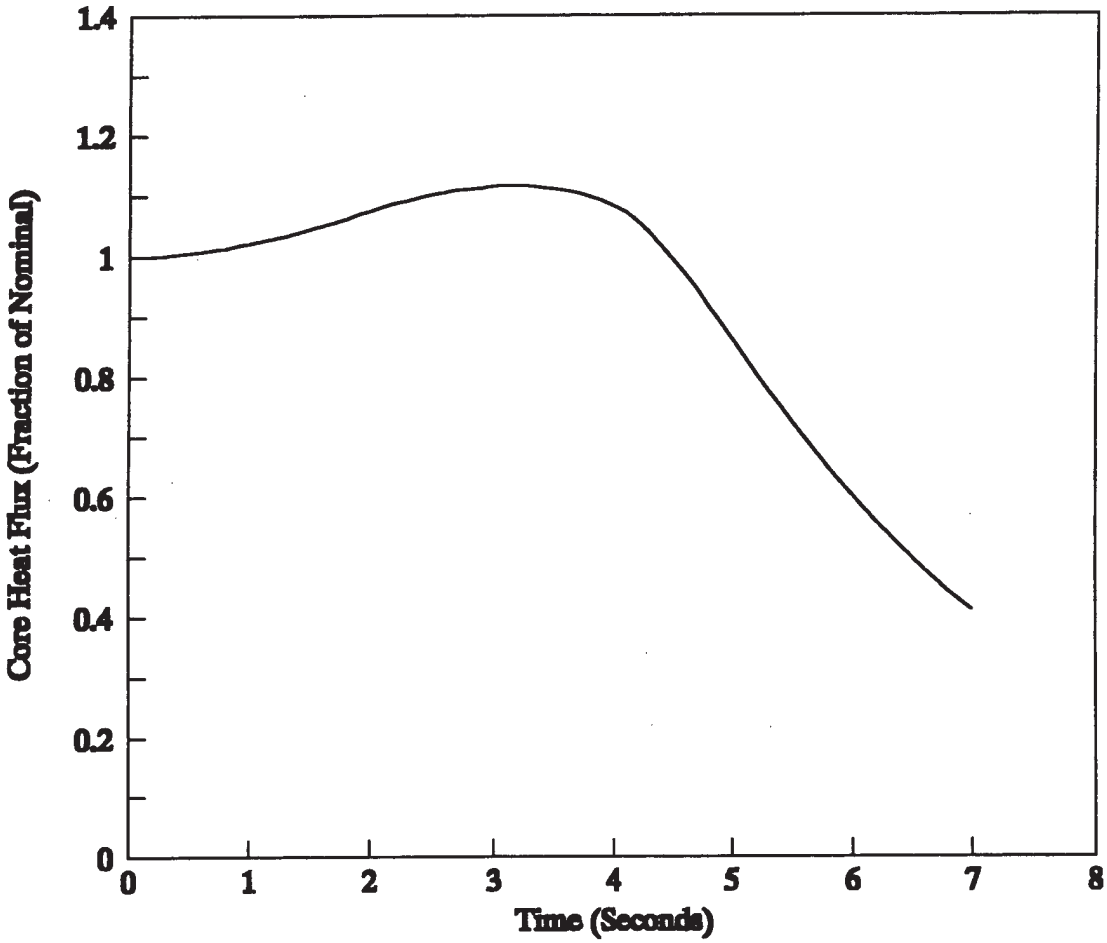


Figure 5.1.7-3

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 80 pcm/sec Withdrawal Rate

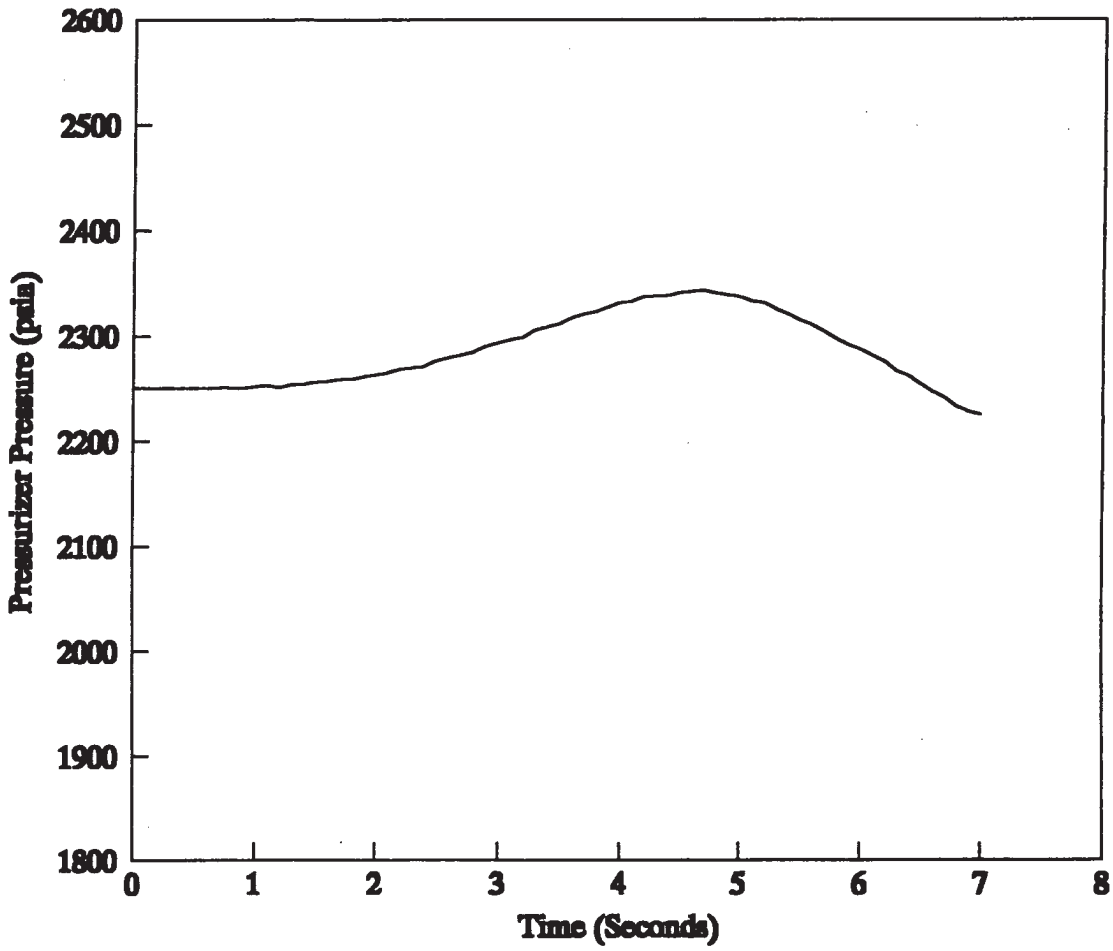


Figure 5.1.7-4

Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 80 pcm/sec Withdrawal Rate

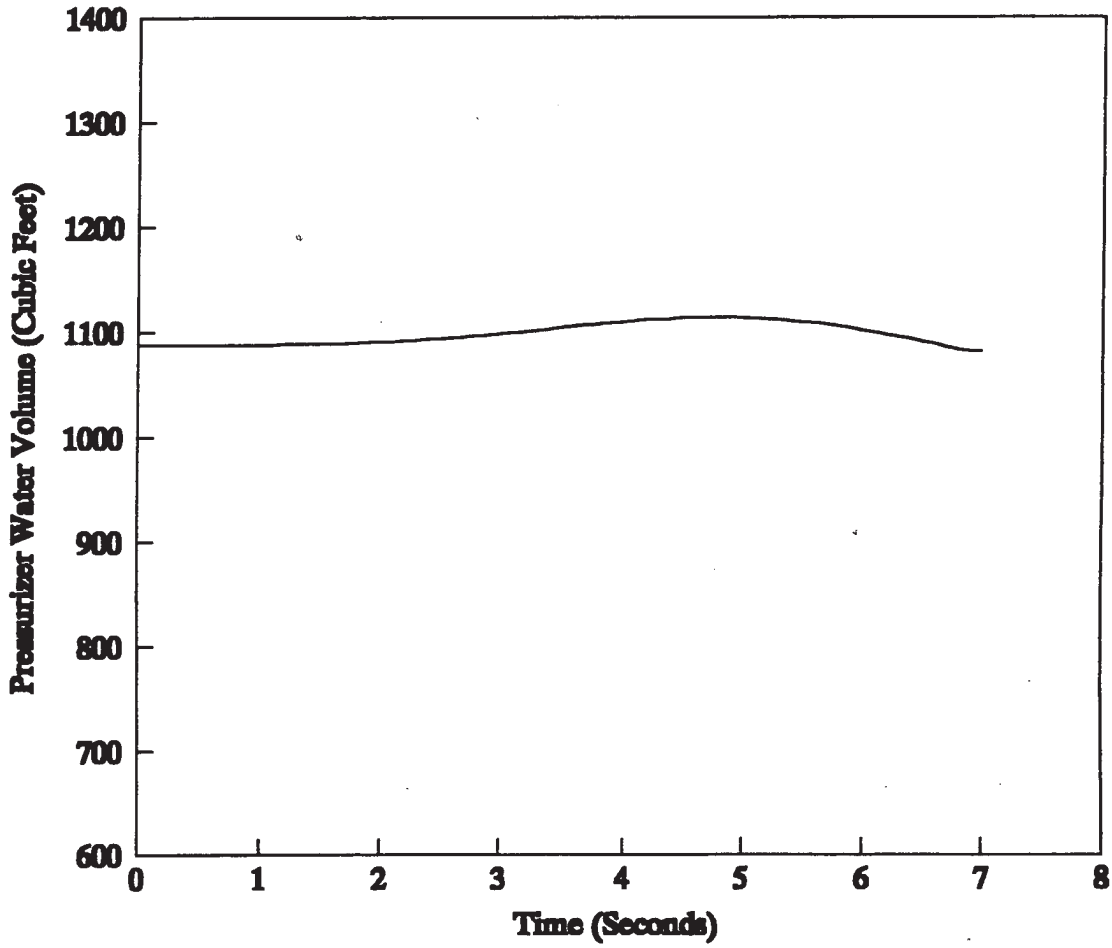


Figure 5.1.7-5

Core Average Water Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 80 pcm/sec Withdrawal Rate

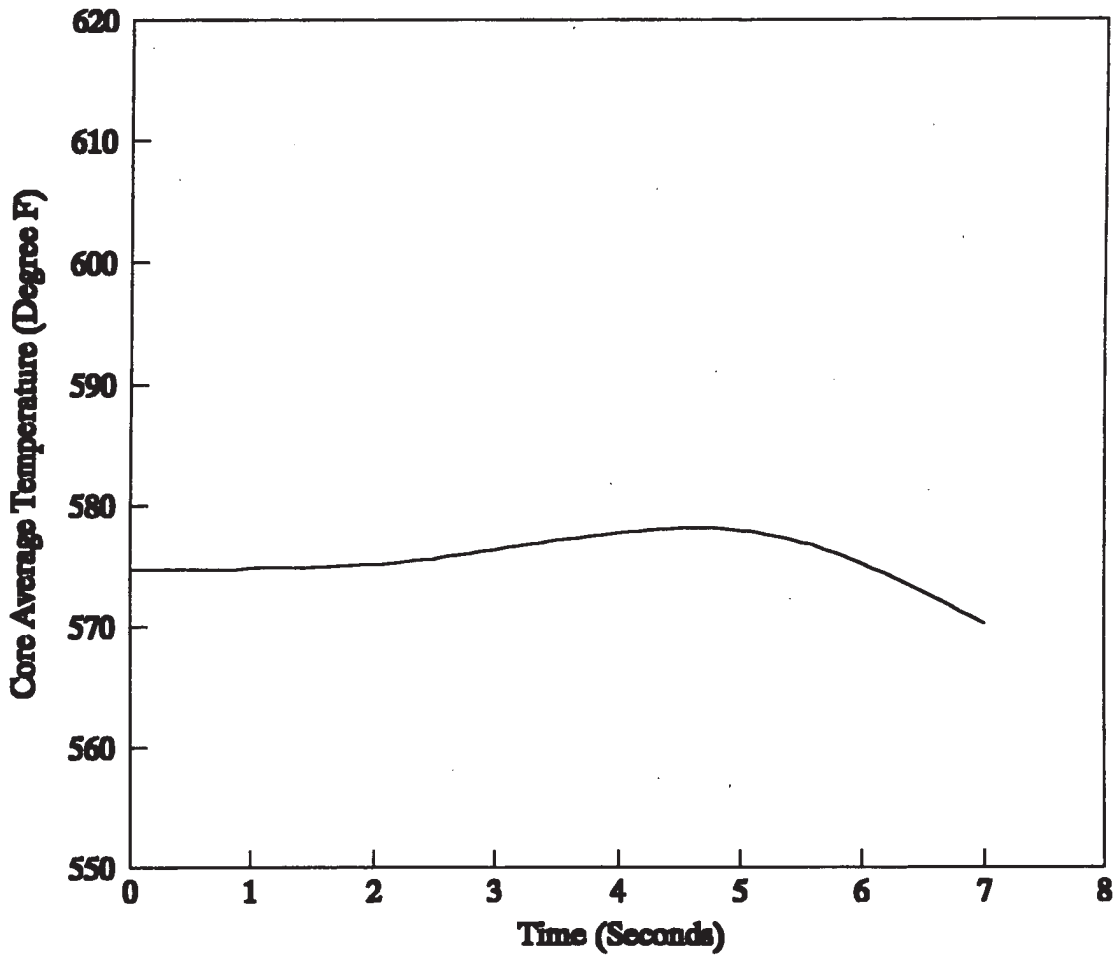


Figure 5.1.7-6

DNB Ratio Versus Time for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 80 pcm/sec Withdrawal Rate

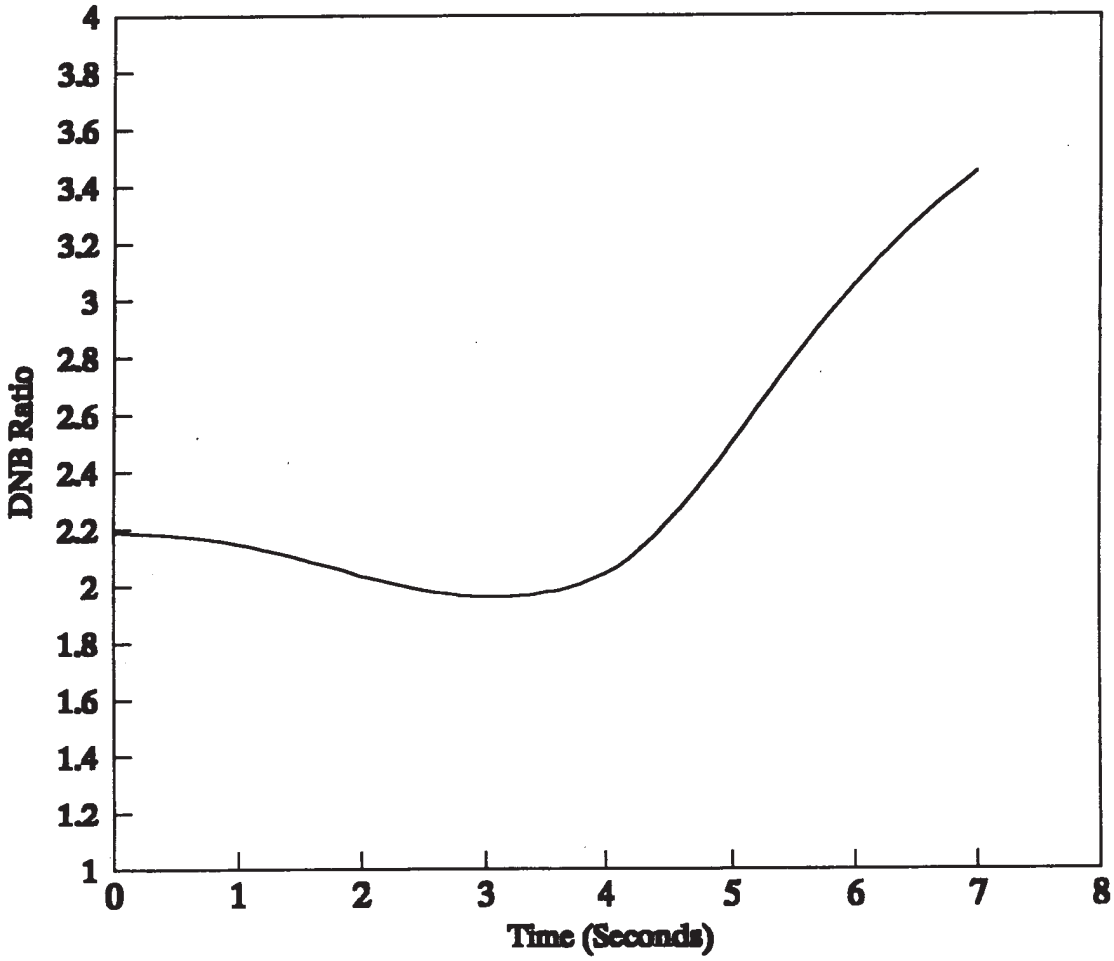


Figure 5.1.7-7

Nuclear Power Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 1 pcm/sec Withdrawal Rate

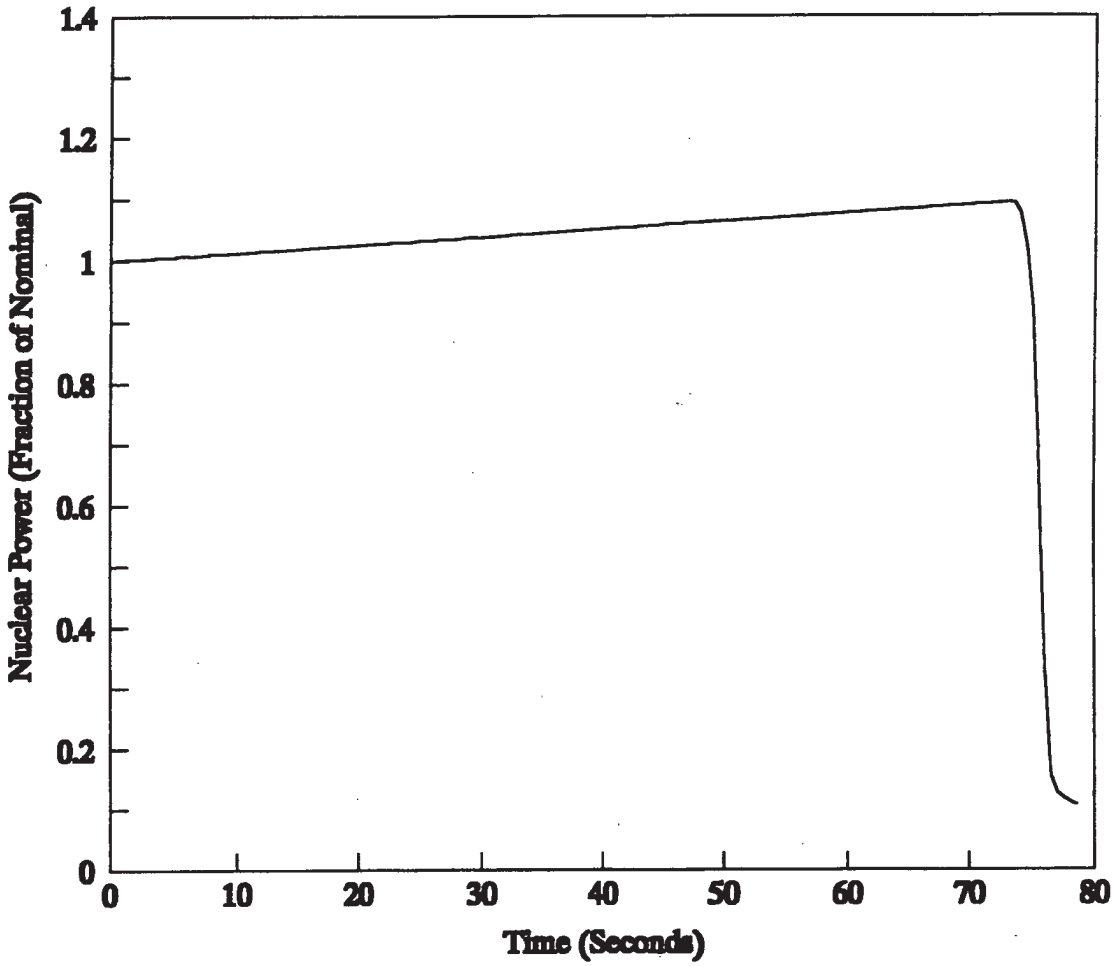


Figure 5.1.7-8

Core Heat Flux Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 1 pcm/sec Withdrawal Rate

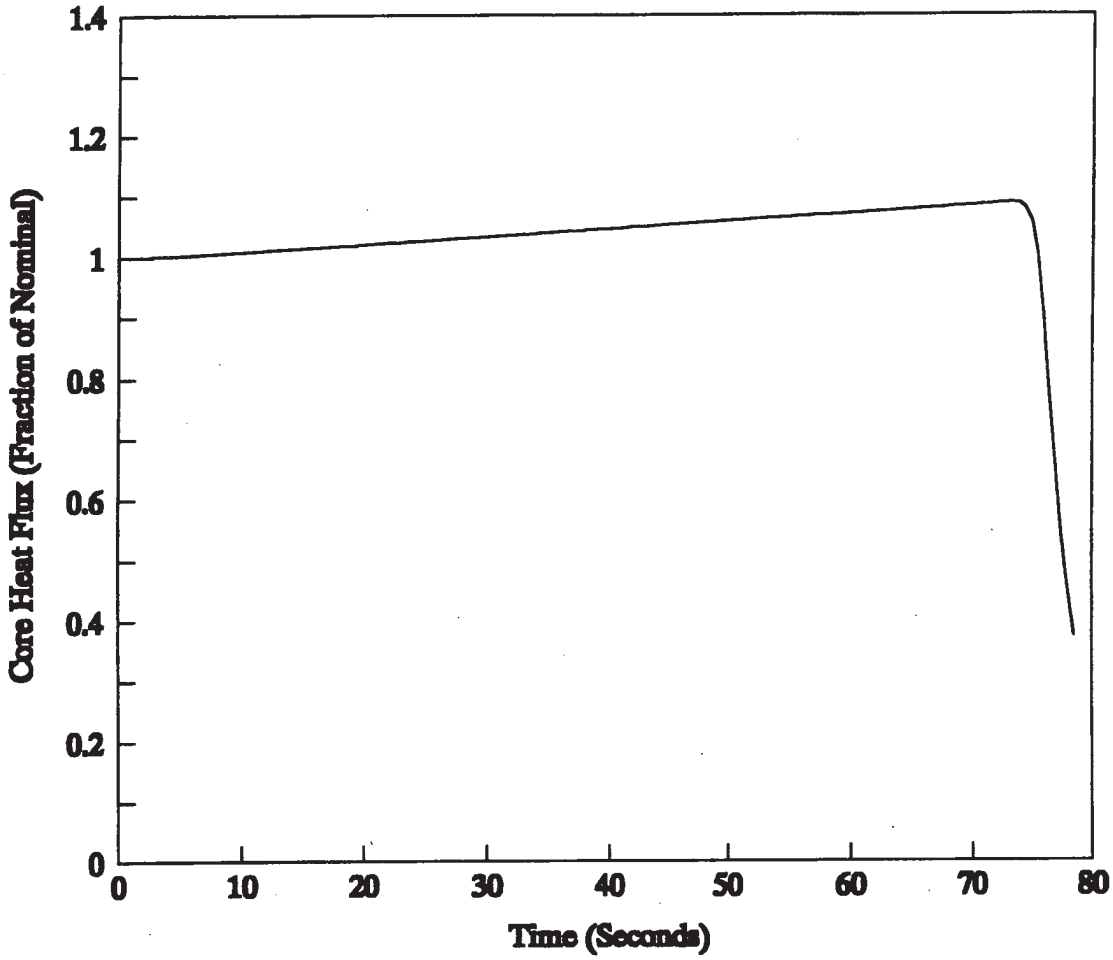


Figure 5.1.7-9

Pressurizer Pressure Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 1 pcm/sec Withdrawal Rate

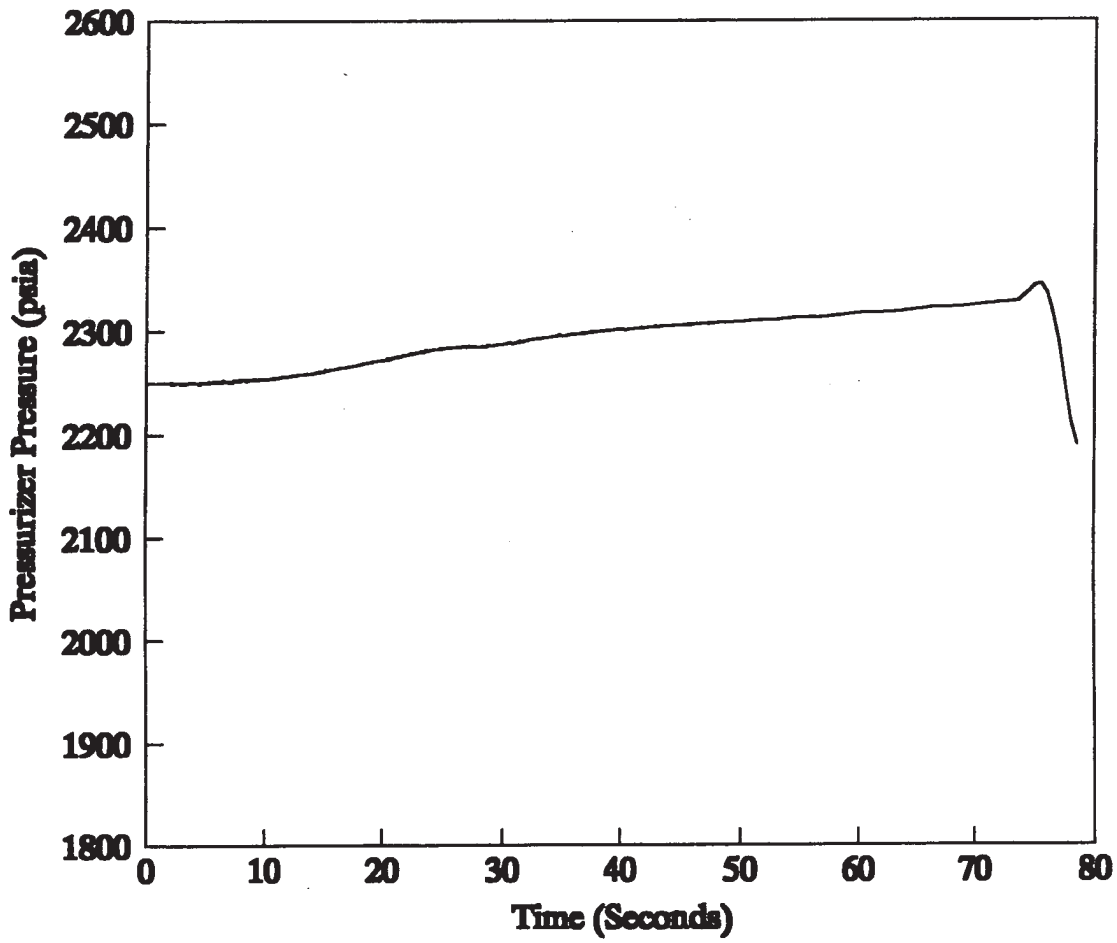


Figure 5.1.7-10

Pressurizer Water Volume Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 1 pcm/sec Withdrawal Rate

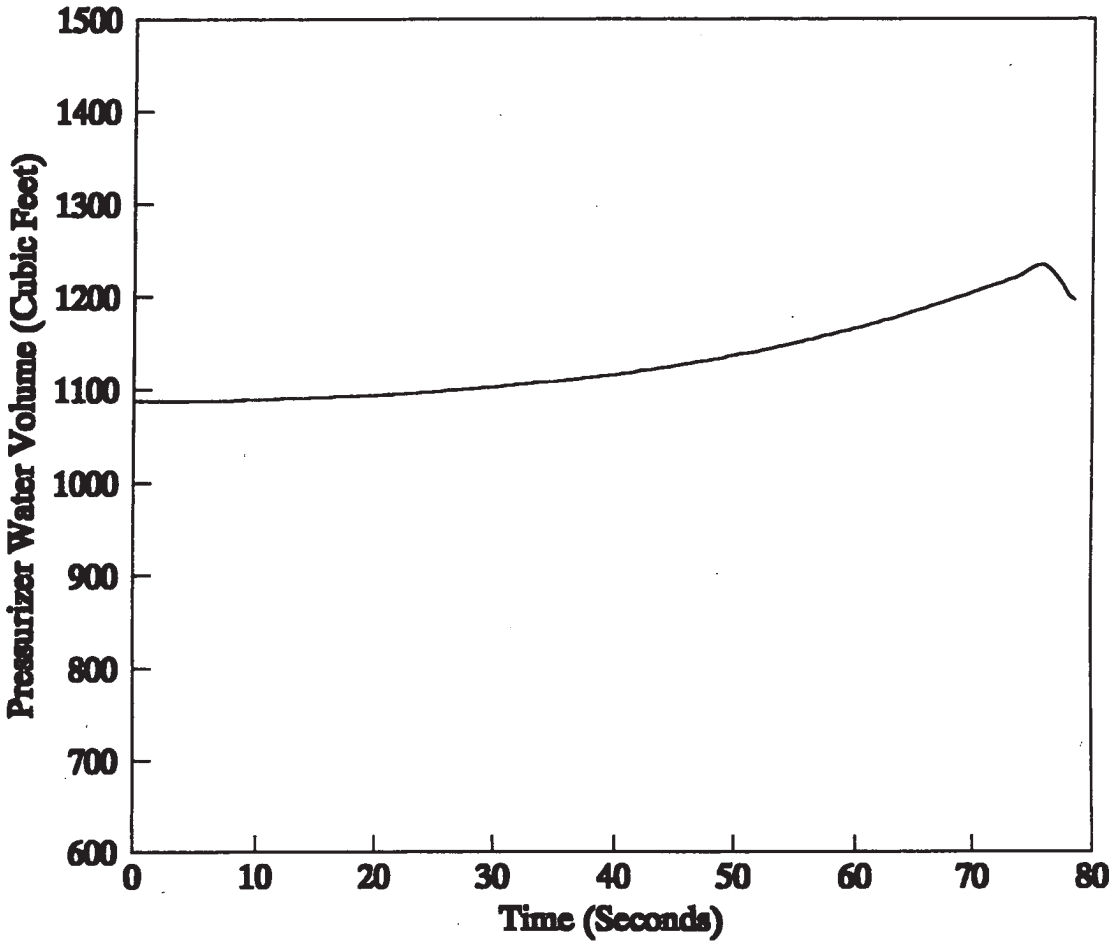


Figure 5.1.7-11

Core Average Water Temperature Transient for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 1 pcm/sec Withdrawal Rate

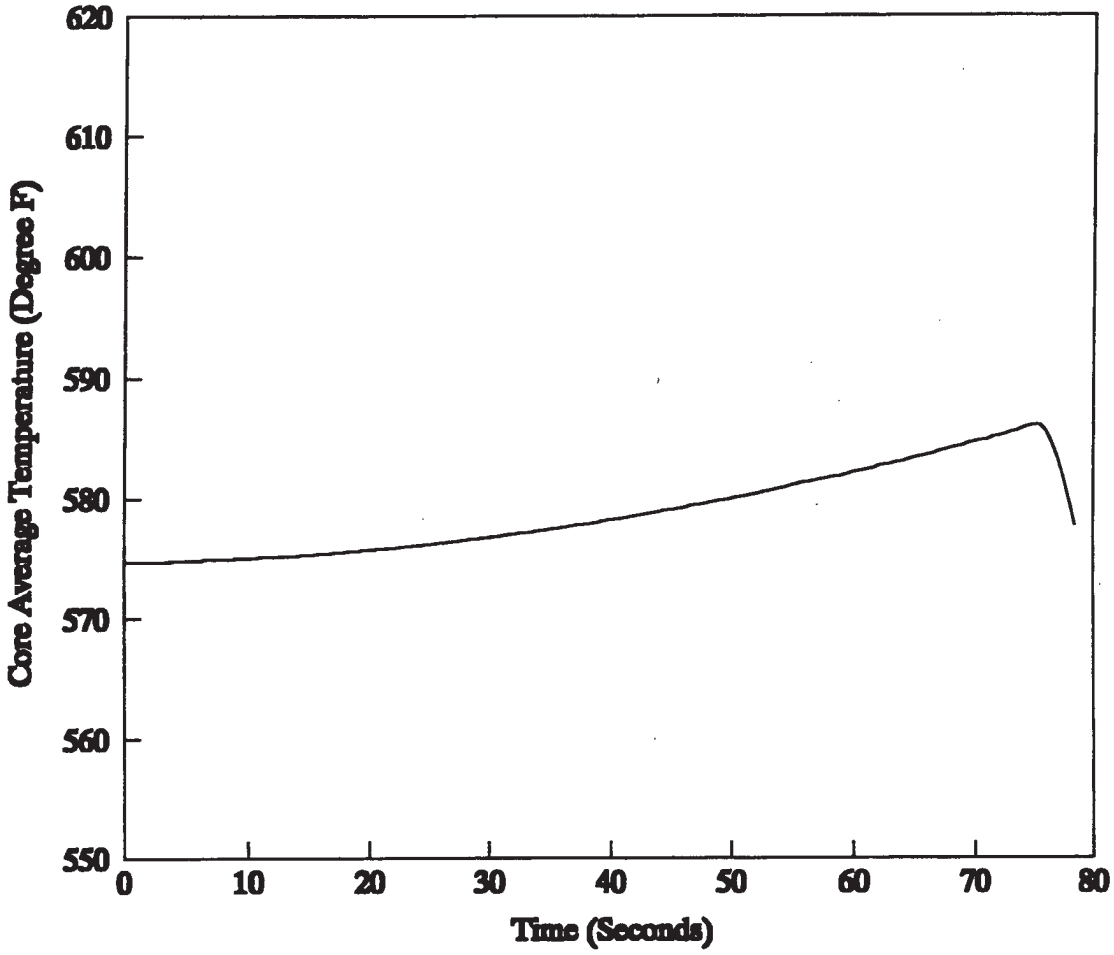


Figure 5.1.7-12

DNB Ratio Versus Time for an Uncontrolled RCCA Bank Withdrawal from Full Power with Minimum Reactivity Feedback, 1 pcm/sec Withdrawal Rate

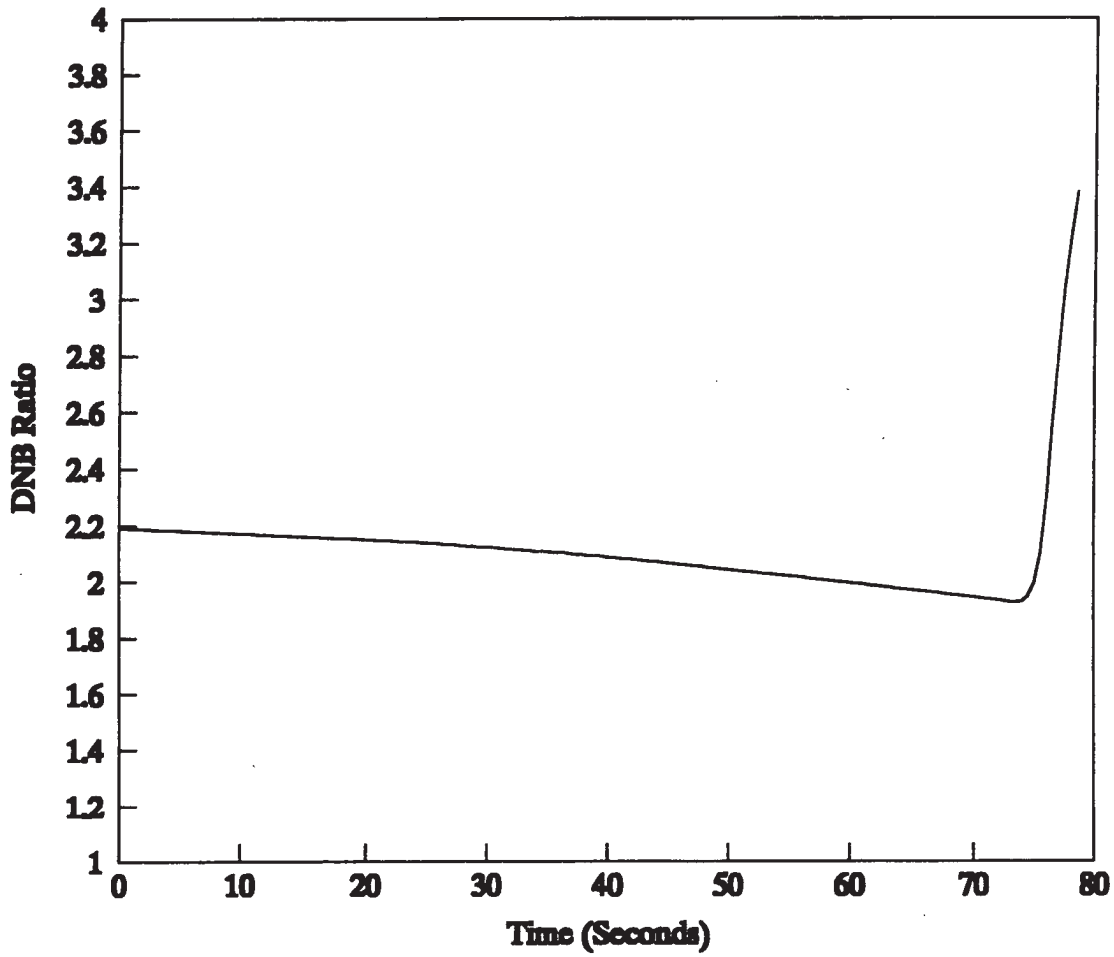


Figure 5.1.7-13

Minimum DNBR Versus Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal from Full Power

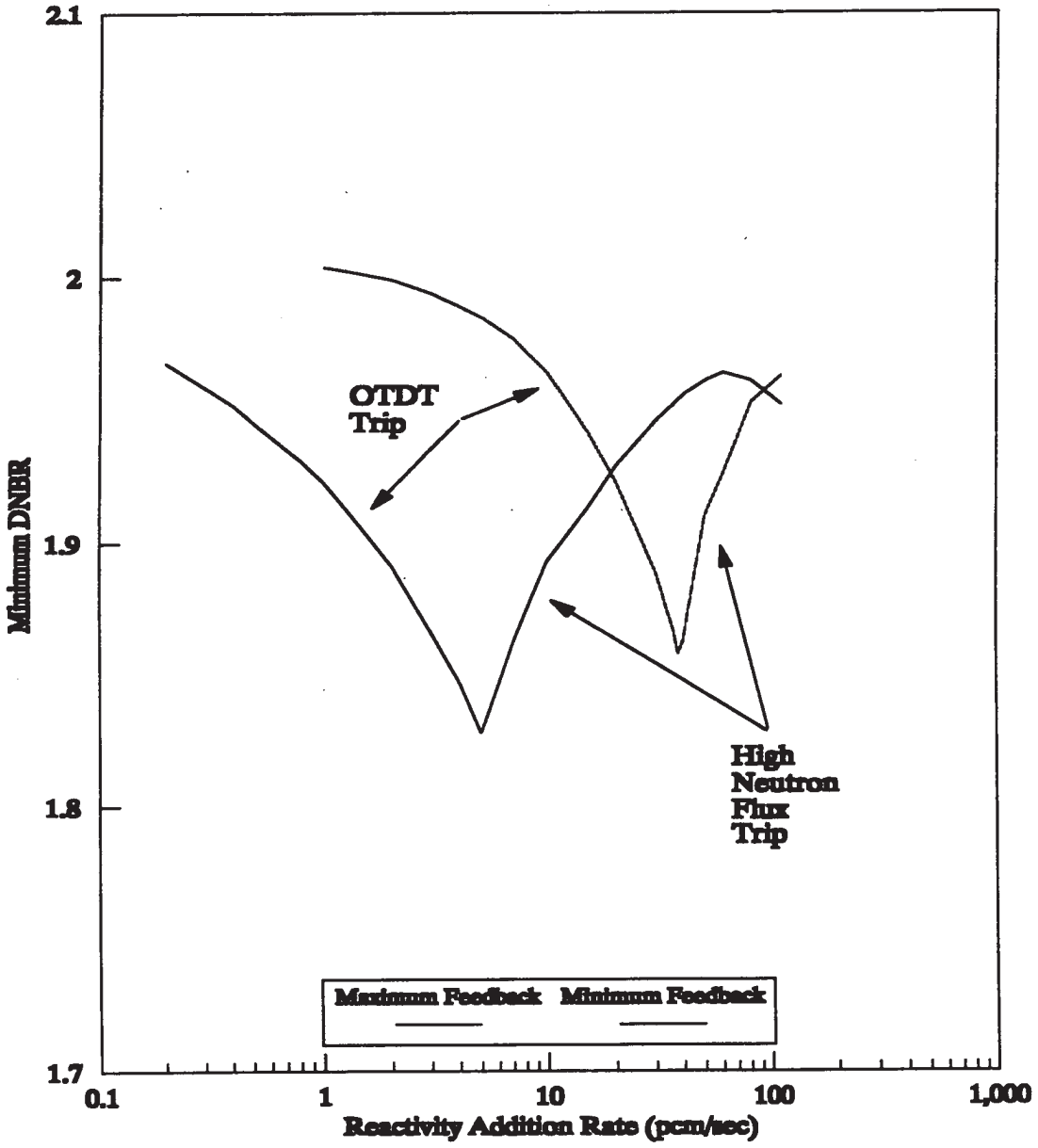


Figure 5.1.7-14

Minimum DNBR Versus Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal from 60% Power

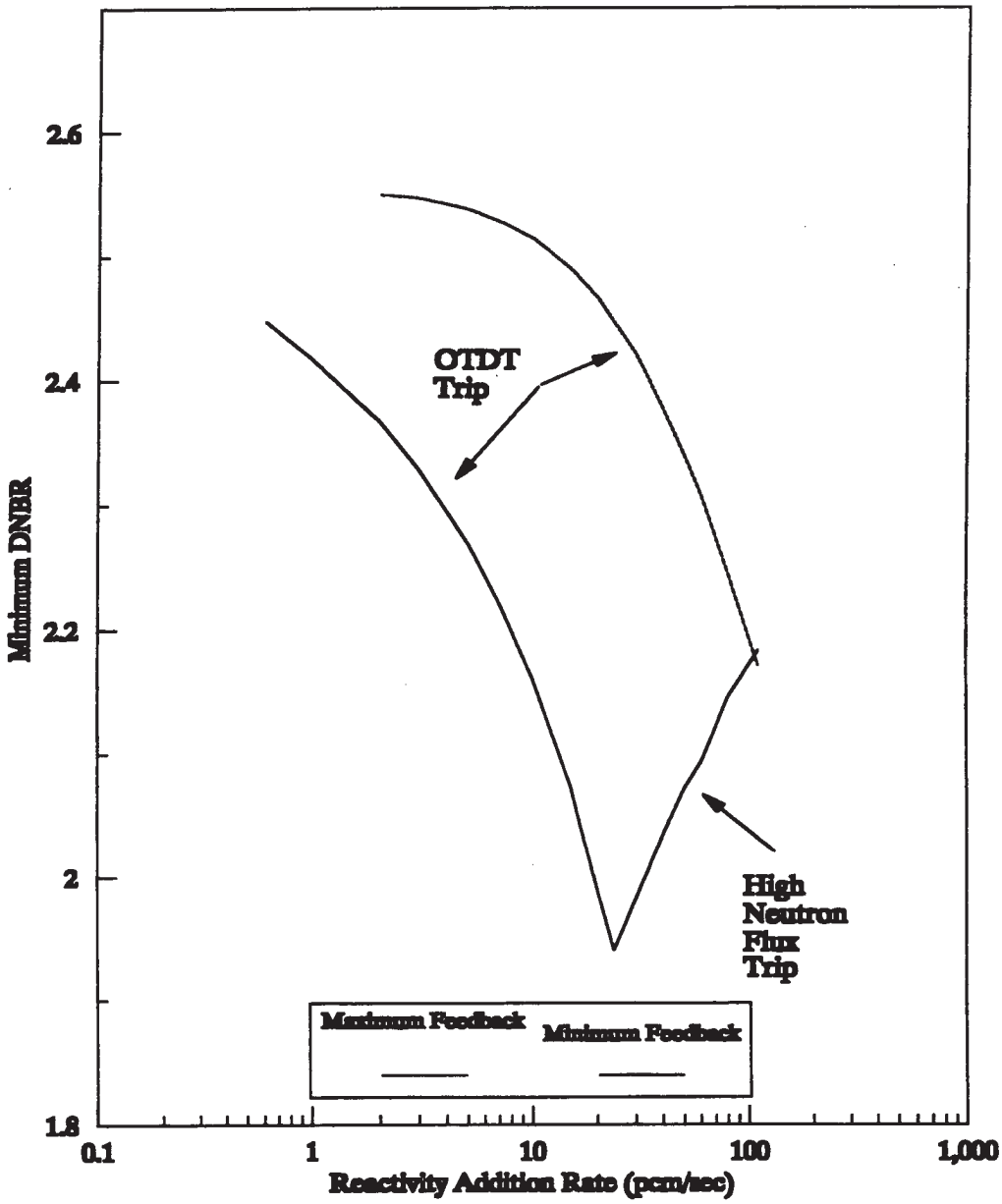
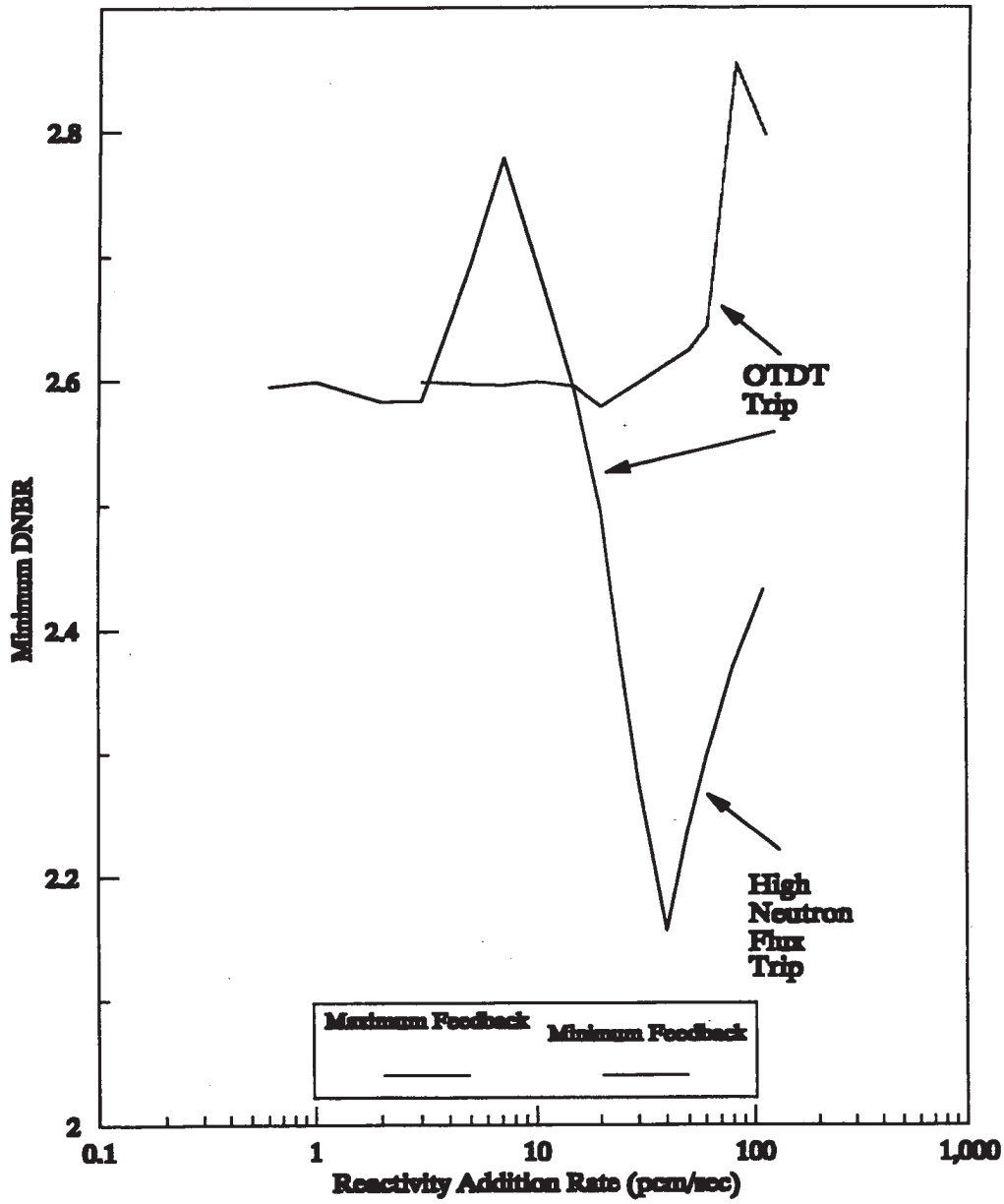


Figure 5.1.7-15

Minimum DNBR Versus Reactivity Insertion Rate for an Uncontrolled RCCA Bank Withdrawal from 10% Power



5.1.8 Rod Assembly Misalignment

Introduction:

The analysis and evaluation herein were performed for the Rod Assembly Misalignment event as described in the FSAR Section 14.1.3 to support the insertion of VANTAGE + Fuel with the design features described in Section 5.1.2. The evaluation also address changes in the safety analysis assumptions associated with the VANTAGE + transition as described in Section 5.1.3.

RCCA misalignment accidents include the following.

a) One or more dropped RCCAs within the same group.

b) A dropped RCCA bank

c) A statically misaligned RCCA

Each RCCA has a position indicator channel which displays the position of the assembly in a display grouping that is convenient to the operator. Fully inserted assemblies are also indicated by a rod at bottom signal which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs move in preselected banks, and the banks always move in the same preselected sequence. Each bank of RCCAs consists of two groups. The rods comprising a group operate in parallel through multiplexing 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 the control rod drive mechanism withdraws the RCCA held by the mechanism. Mechanical failures are in the direction of insertion or immobility.

A dropped RCCA, or RCCA bank is detected by:

a) Sudden drop in the core power level as seen by the nuclear instrumentation system;

b) Asymmetric power distribution as seen on nuclear instrumentation system;

c) Rod at bottom signal;

d) Rod deviation alarm;

e) Rod position indication.

Misaligned RCCAs are detected by:

a) Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

b) Rod deviation alarm;

c) Rod position indicators.

The resolution of the rod position indicator channel is ±5 percent of span (±7.2 in.). Deviation of any RCCA from its group by twice this distance (10 percent of span or 14.4 in.) will 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 10 percent of span. If the rod deviation alarm is not operable, the operator is required to log the RCCA positions in a prescribed time sequence to confirm alignment.

If one or more rod position indicator channels is out of service, the operator must follow detailed operating instructions to ensure the alignment of the nonindicated RCCAs. These operating instructions require selected pairs of core exit thermocouples to be monitored in a prescribed time sequence and following significant motion of the nonindicated assemblies. The operating instructions also call for the use of moveable incore neutron detectors to confirm core exit thermocouple indication of assembly misalignment.

Method of Analysis and Assumptions:

The analysis of one or more Dropped RCCAs or a Dropped RCCA Bank is described in Section 5.1.9 which follows.

For the statically misaligned RCCA, steady-state power distributions are analyzed using appropriate nuclear physics computer codes (see Section 3.3). The peaking factors are then used as input to the THINC code to calculate the DNBR. The analysis examines the case of the worst rod withdrawn from bank D inserted at the insertion limit with the reactor initially at full power. The analysis assumes this incident to occur at beginning of life since this results in the minimum feedback value (least negative) of

the moderator temperature coefficient. This assumption maximizes the power rise and minimizes the tendency of the large (most negative) moderator temperature coefficient to flatten the power distribution.

Results:

The most-severe misalignment situations with respect to DNBR occur at significant power levels. These situations arise from cases in which one RCCA is fully inserted or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the transient approaches the postulated conditions. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the safety analysis limit value.

The insertion limits in the Core Operating Limits Report (COLR) may vary from time to time depending on several limiting criteria. The full-power insertion limits on control bank D must be chosen to be above that position which meets the minimum DNBR and peaking factors. The full- power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with bank D inserted to its full-power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the safety analysis limit value. The analysis of this case assumes that the initial reactor power, pressure, and RCS temperature are at the nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

For RCCA misalignment with one RCCA fully inserted, the DNBR does not fall below the safety analysis limit value. The analysis of this case assumes that initial reactor power, pressure, and RCS temperatures are at the nominal values, with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident; thus, there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the limiting design axial power distribution. The resulting linear heat generation rate is well below that which would cause fuel melting.

After identifying an RCCA group misalignment condition, the operator must take action as required by the plant Technical Specifications and operating instructions.

Conclusions:

For all cases of any RCCA fully inserted, or bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value.

5.1.9 Rod Cluster Control Assembly (RCCA) Drop

Introduction:

The dropped RCCA accident is initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given bank to drop to the bottom of the core. Protection is provided by an automatic turbine runback and an automatic rod withdrawal block. The acceptance criterion for this event is that no fuel failures occur. This is verified by demonstrating that the departure from nucleate boiling ratio (DNBR) remains above the limit value for the plant.

A dropped RCCA or RCCA bank causes an initial reduction in nuclear power which corresponds to the reactivity worth of the RCCA(s). In addition, a turbine runback and automatic rod withdrawal block are actuated. The turbine runback is to a power level determined by the differential oil pressure between the governor and load limiter, and also by the timer on the runback and the rate of runback. The turbine runback is prevented from going below 70% power.

For purposes of the analysis, a single or multiple dropped RCCA occurrence which causes a reduction in core power to a value greater than the turbine power at the runback termination point is called a "dropped rod". The multiple dropped RCCAs may be any number and combination of rods from the same group of a given bank. RCCAs from different groups are not considered since it requires more than one single failure for them to drop. With sufficient reactivity feedback the core power will tend to match the turbine load and the plant will stabilize at the runback setpoint. However, if there is no moderator reactivity feedback, power will stabilize at the level corresponding to that caused by the dropped rod. Primary reactor power will be greater than turbine power, resulting in a heatup of the primary coolant. Depending on the dropped rod worth, increases in the pressurizer pressure and/or steam generator pressure may occur such that the pressurizer relief valves and/or steam generator safety valves open to accommodate the mismatch between reactor and turbine power. If the combined relieving capacity of the turbine, the pressurizer relief valves, and/or the steam generator safety valves is sufficient to match

reactor power, the reactor will reach an equilibrium condition. However, the primary system heatup may be too great in which case an OTΔT reactor trip will terminate the event.

A multiple dropped RCCA occurrence which causes a reduction in core power to a value less than the turbine power at the runback termination point is called a "dropped bank." With maximum reactivity feedback the core power will increase to match turbine load and the plant will stabilize at or near the runback termination setpoint.

Dropped RCCAs can be detected by the excore detectors, core exit thermocouples, rod deviation alarms, and rod position indicators. In addition, the rod-on-bottom alarm will be actuated. These features serve to alert the operator to a dropped RCCA event. The rod-on-bottom signal device provides an indication signal for each RCCA.

In addition to the above indications, single or multiple dropped RCCAs within the same group result in a negative reactivity insertion which may be detected by the power range negative neutron flux rate circuitry. The negative flux rate signal is actuated by sensing a rapid decrease in local neutron flux. A rod drop signal from any rod position indication channel, or from one or more of the four power range channels, initiates a reduction in the turbine load by a preset adjustable amount and blocking of automatic rod withdrawal. The turbine runback is redundantly achieved by acting upon the turbine load limit and/or on the turbine load reference. The rod withdrawal block is redundantly achieved.

The dropped RCCA assemblies and dropped RCCA assembly bank events are classified as a Condition II event as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious fault, i.e., a Condition III or IV category event.

The applicable safety analysis licensing basis acceptance criteria for the Condition II dropped RCCA assemblies and dropped RCCA assembly bank event for Indian Point 3 are:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (i.e., 2748.5 psia and 1208.5 psia, respectively),

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit, and

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

Method of Analysis and Assumptions:

The RCCA Drop event is analyzed to show that: 1) the integrity of the core is maintained by the reactor protection system as the DNBR remains above the safety analysis limit value; 2) the peak RCS and secondary system pressures remain below the accident analysis pressure limits; and 3) the pressurizer does not reach a water-solid condition and result in water relief through the pressurizer relief and safety valves. Of these, the primary concern for this event is DNB and assuring that the DNBR limit is met.

The dropped rod transient is analyzed with the LOFTRAN computer program (see Section 5.1.5). The LOFTRAN computer code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Calculated statepoints and nuclear models form the basis used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code.

The Revised Thermal Design Procedure (RTDP) was used in the analysis so the initial conditions for power, RCS pressure, and T_{avg} are at the nominal values.

The analysis is performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq$ 24%; and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of $\leq$ 24% with a maximum steam generator tube plugging level of 30% in any one steam generator.

In addition, turbine runback was modeled for two different runback conditions. One set of analyses assumed the turbine runback occurred 2 seconds after the rod/bank drop, and the turbine power was runback to 74% power (nominal turbine power level of 70% + 4% uncertainty). The other set of

analyses modeled the turbine runback to 94% power (nominally 90% + 4% uncertainty) after a 15.3 second delay. Both of these turbine runback conditions are consistent with those used in the current licensing basis RCCA Drop analysis.

All cases assumed automatic rod control block initiated by a dropped rod signal (i.e., by a rod-on- bottom signal or change in flux signal). Cases were also analyzed over a range of dropped rod worths.

Cases Analyzed:

For the dropped rod event, cases were analyzed over a range of rod worths beginning with a rod worth of 50 pcm and increased in subsequent cases by 50 pcm until the core heat flux following the dropped rod was calculated to stabilize at or below the turbine runback setpoint used in the analysis. Once a given rod worth value no longer tripped on OTΔT, cases were analyzed every 10 pcm between the last rod worth that tripped and the first rod worth that did not trip. The reactivity feedback assumptions for the dropped rod analysis were:

1) Minimum moderator reactivity feedback corresponding to beginning of core life (0 pcm/°F moderator temperature coefficient).

2) Least negative Doppler temperature coefficient (-0.9 pcm/°F).

For the dropped bank event, cases were analyzed at 200 pcm intervals, starting with the lowest worth that results in core heat flux stabilizing above the turbine runback setpoint. The reactivity feedback assumptions for the dropped bank analysis were:

1) Maximum moderator reactivity feedback corresponding to the end of core life, rodded, core condition (0.54 Δk/gm/cc moderator density coefficient).

2) Most negative Doppler temperature coefficient (-3.2 pcm/°F).

In addition to the analysis of the RCCA Drop modeling turbine runback, dropped rod statepoints were also evaluated to bound possible operation without turbine runback and to address the possibility of a single failure in the rods-on-bottom signal which blocks automatic rod withdrawal

Results:

Figures 5.1.9-1 through 5.1.9-6 show the reactor response and the reactor coolant system response to a dropped rod with a worth of 100 pcm assuming asymmetric steam generator tube plugging and a turbine runback to 74% power. A reactor trip on OTΔT occurs at approximately 70 seconds. The trip is caused by the power mismatch between the reactor and turbine which causes a heatup of the RCS. Figures 5.1.9-7 through 5.1.9-12 show the response to a dropped bank with a worth of 400 pcm assuming asymmetric steam generator tube plugging and a turbine runback to 74% power. Nuclear power and core heat flux stabilize at levels corresponding to the turbine runback power level including the uncertainty. After approximately 100 seconds, all of the plant parameters depicted in Figures 5.1.9-7 through 5.1.9-12 have reached equilibrium values.

The cases assuming uniform steam generator tube plugging have similar results. For all cases analyzed, the DNBR is greater than the applicable safety analysis limit value.

For the cases in which the turbine runs back to 94% power, the asymmetric steam generator tube plugging results are presented. Figures 5.1.9-13 through 5.1.9-18 show the reactor response and the reactor coolant system response to a dropped rod with a worth of 50 pcm. Similar to the case presented for the 74% power turbine runback, a reactor trip on OTΔT occurs at approximately 373 seconds. Figures 5.1.9-19 through 5.1.9-24 show the response to a dropped bank with a worth of 400 pcm. Nuclear power and core heat flux stabilize at levels corresponding to the turbine runback power level of 94% power. After approximately 70 seconds, the plant parameters shown in Figures 5.1.9-19 through 5.1.9-24 have reached equilibrium values.

The 94% power turbine runback cases assuming uniform steam generator tube plugging have results similar to the asymmetric steam generator tube plugging results. For all cases analyzed, the DNBR is greater than the applicable safety analysis limit value.

A typical time sequence of events for this accident are shown on Table 5.1.9-1. This table represents the case with a dropped rod worth of 100 pcm and assuming asymmetric steam generator tube plugging with a turbine runback setpoint of 74% power.

Conclusions:

Based on the DNBR results for all of the cases analyzed, it has been demonstrated that the DNBR criterion is met. This includes evaluations performed to bound possible operation without turbine runback and to address the possibility of a single failure in the rods-on-bottom signal which blocks automatic rod withdrawal. Therefore, it is concluded that dropped RCCAs do not lead to conditions that cause core damage and that all applicable safety criteria is satisfied for this event.

Hence, it is concluded that the insertion of VANTAGE + Fuel and the other design changes associated with the VANTAGE + transition as described in Sections 5.1.2 and 5.1.3 are acceptable for the RCCA Drop events.

Table 5.1.9-1

Typical Sequence of Events

for the

Rod Cluster Control Assembly Drop (100 pcm)

Asymmetric Steam Generator Tube Plugging

Event	Time (Seconds)
Initiation of a rod drop (100 pcm)	10.
Turbine runback (to 74%) initiated	13.
Turbine runback complete	39.
PORVs open	52.
OTΔT reactor trip setpoint reached	69.3
Rods begin to fall	71.3

Figure 5.1.9-1

Rod Cluster Control Assembly Drop Event

Dropped Rod / 100 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Nuclear Power versus Time

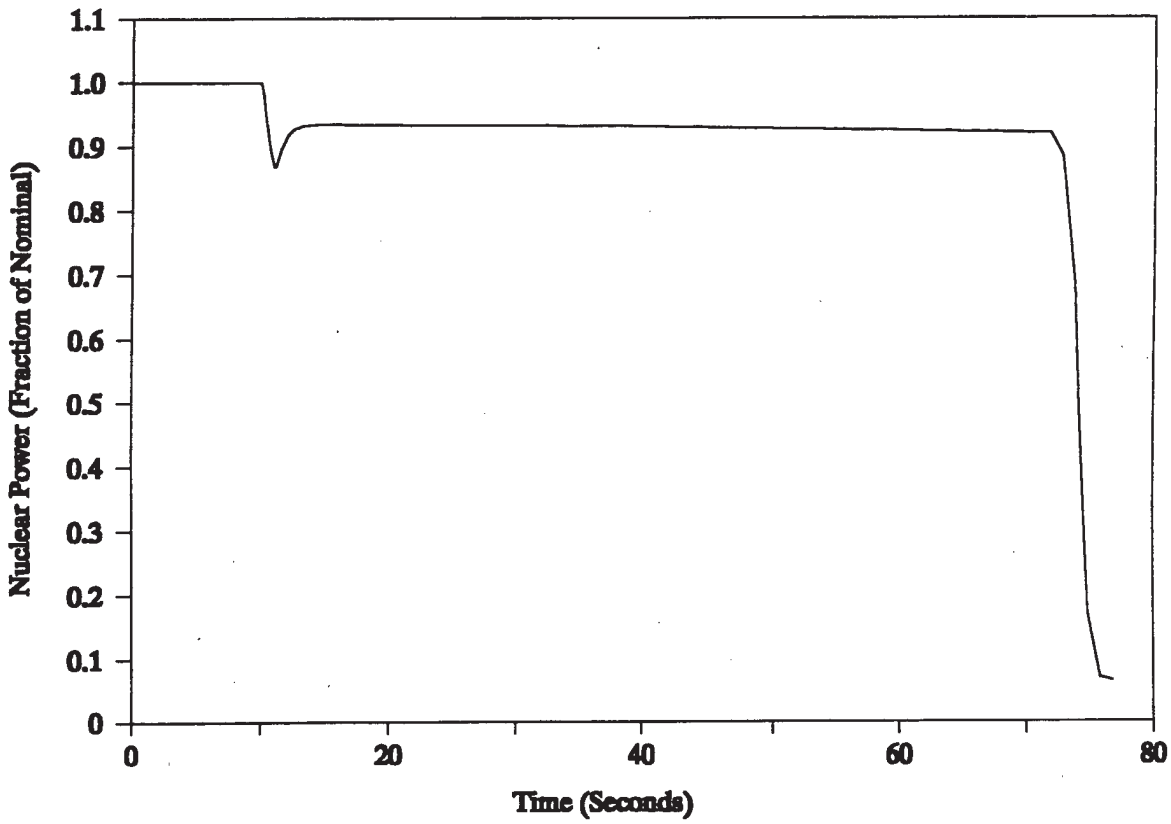


Figure 5.1.9-2

Rod Cluster Control Assembly Drop Event

Dropped Rod / 100 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Core Heat Flux versus Time

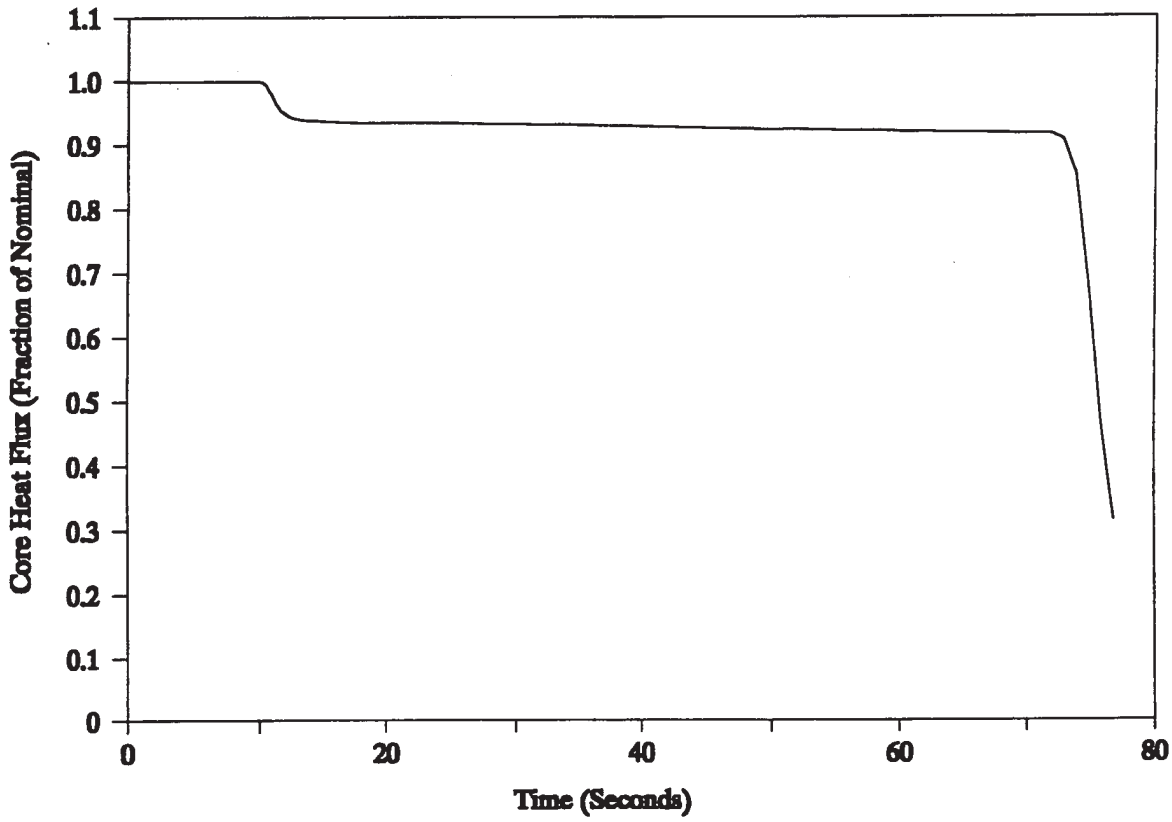


Figure 5.1.9-3

Rod Cluster Control Assembly Drop Event

Dropped Rod / 100 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Vessel Average Temperature versus Time

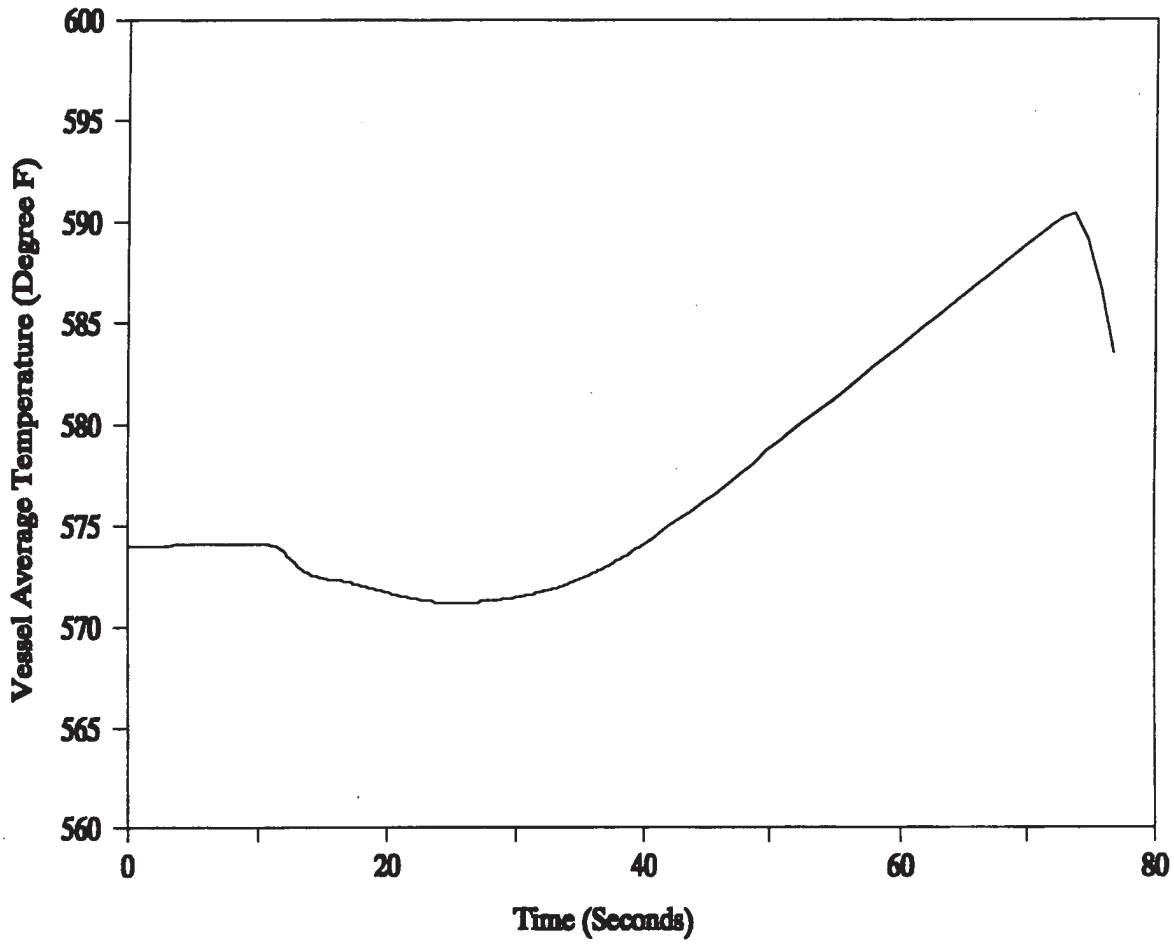


Figure 5.1.9-4

Rod Cluster Control Assembly Drop Event

Dropped Rod / 100 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Inlet Temperature versus Time

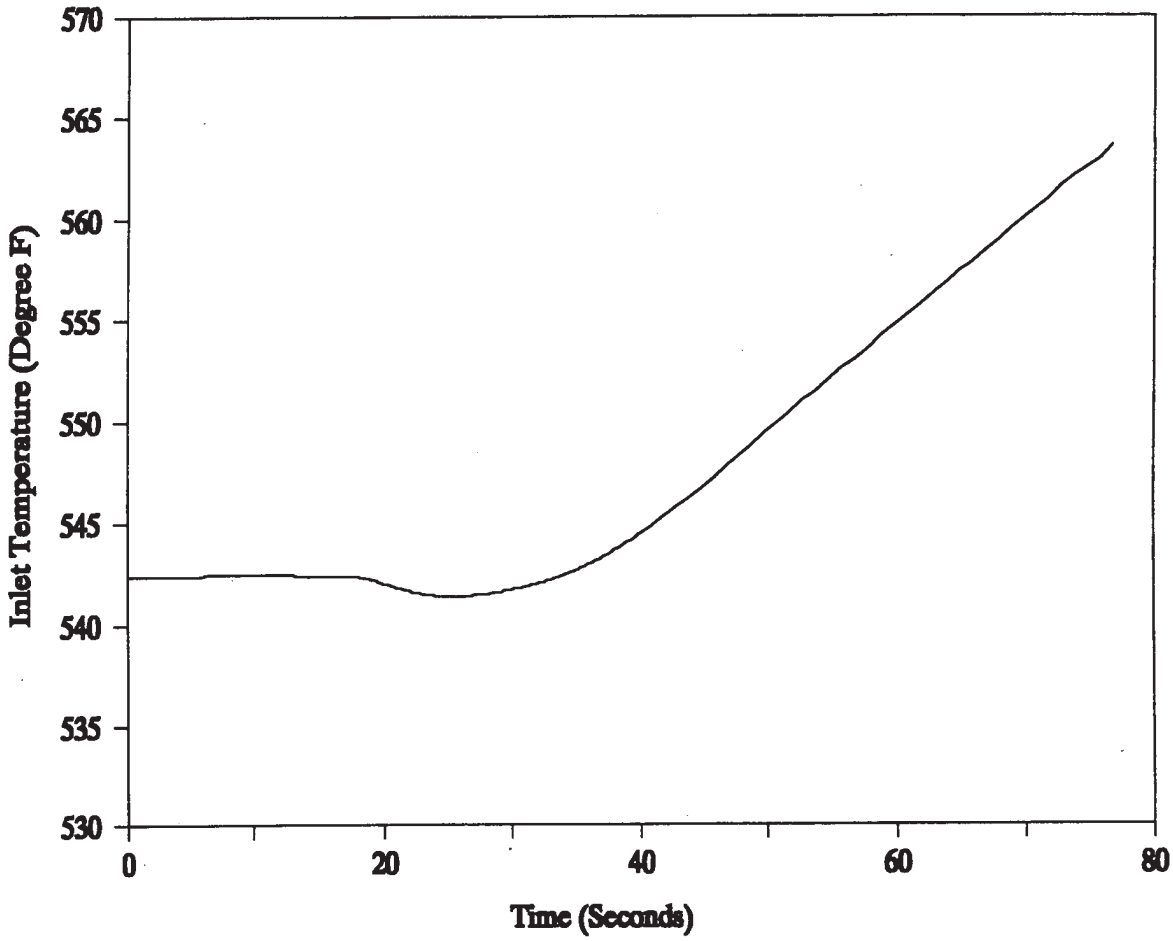


Figure 5.1.9-5

Rod Cluster Control Assembly Drop Event

Dropped Rod / 100 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Pressurizer Pressure versus Time

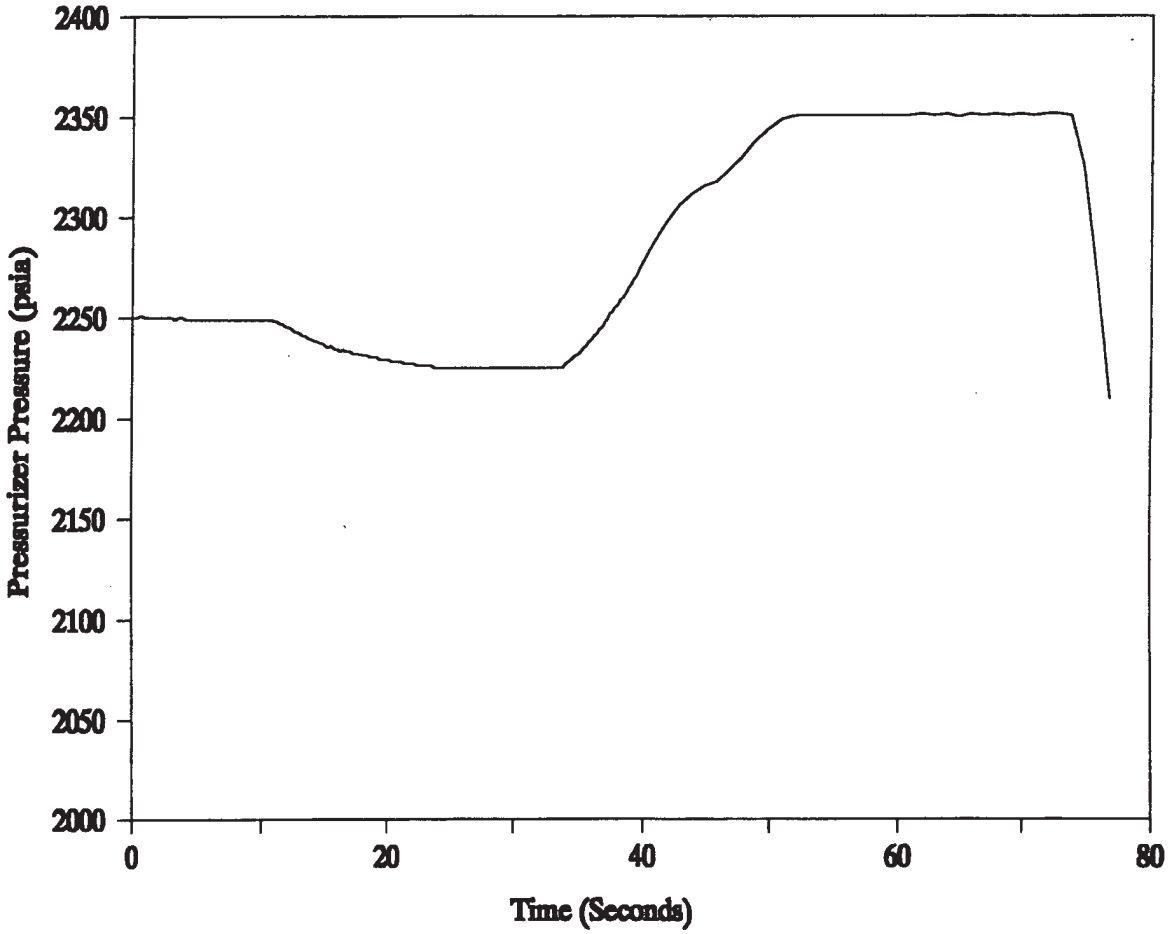


Figure 5.1.9-6

Rod Cluster Control Assembly Drop Event

Dropped Rod / 100 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Steam Flow versus Time

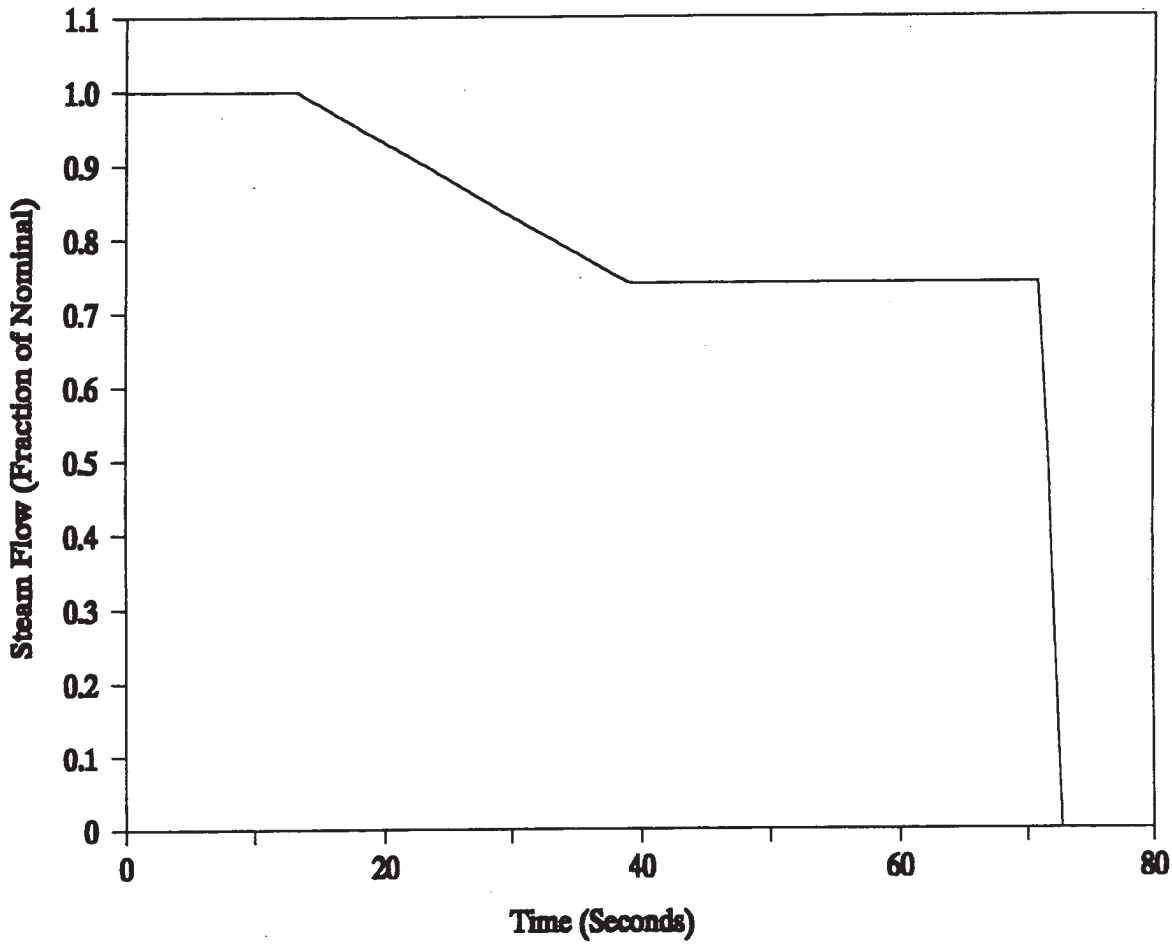


Figure 5.1.9-7

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Nuclear Power versus Time

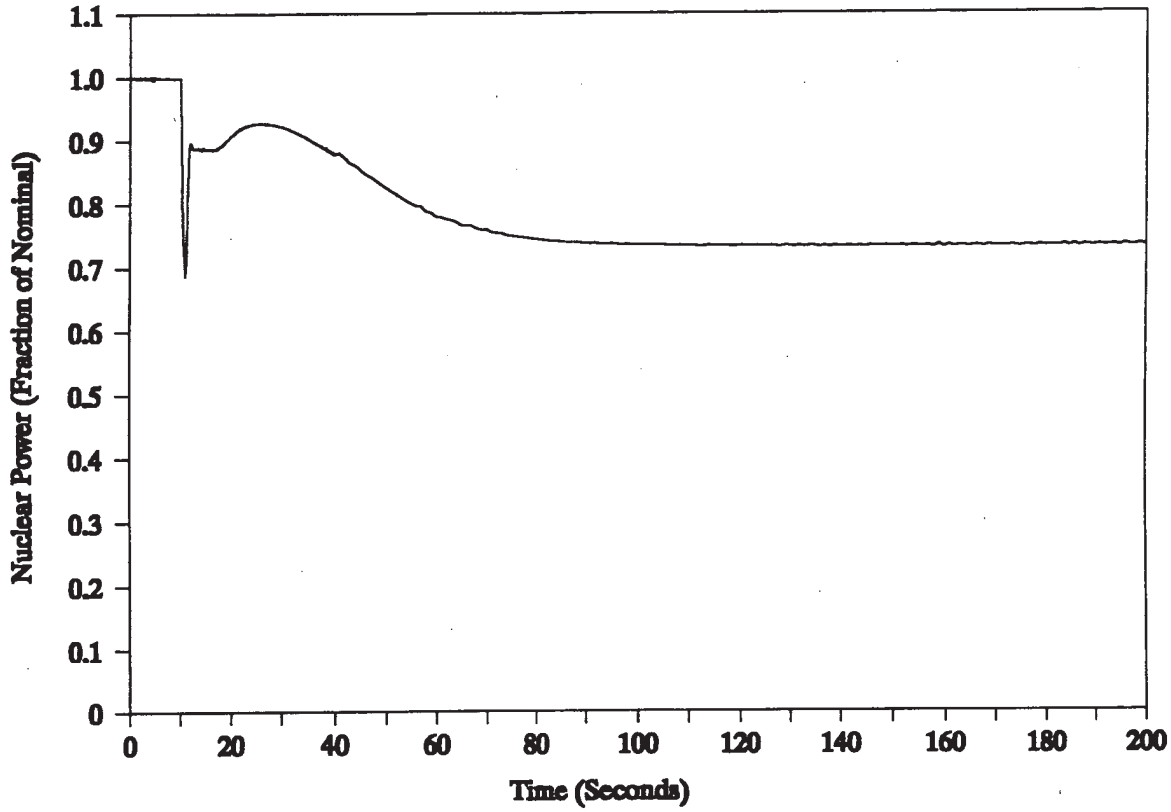


Figure 5.1.9-8

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Core Heat Flux versus Time

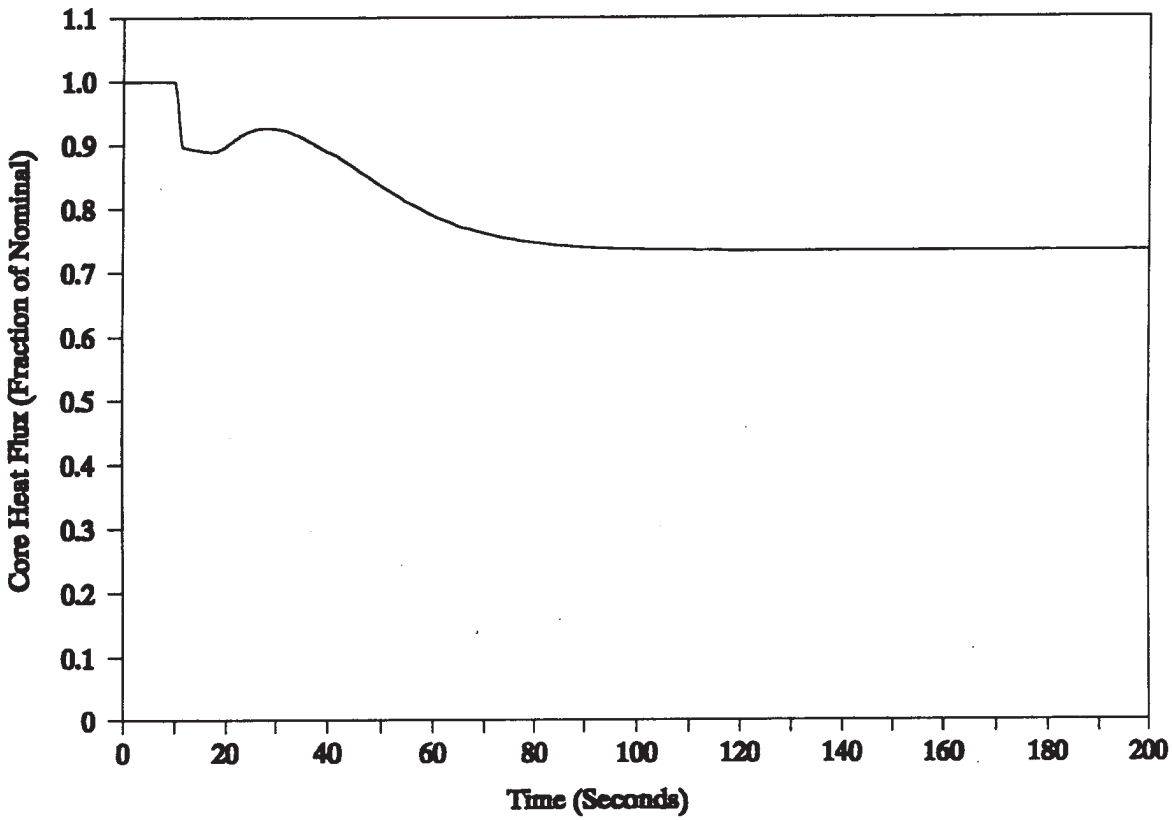


Figure 5.1.9-9

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Vessel Average Temperature versus Time

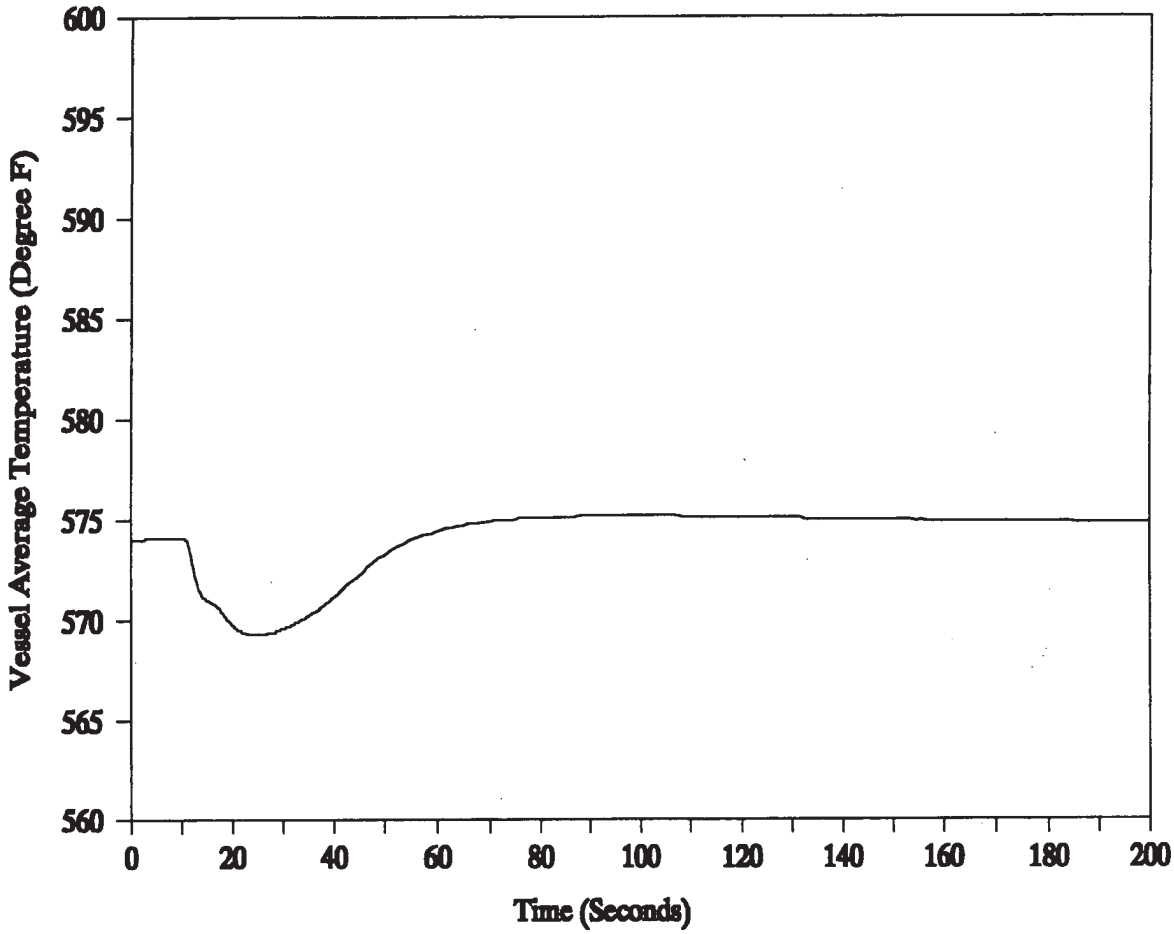


Figure 5.1.9-10

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Inlet Temperature versus Time

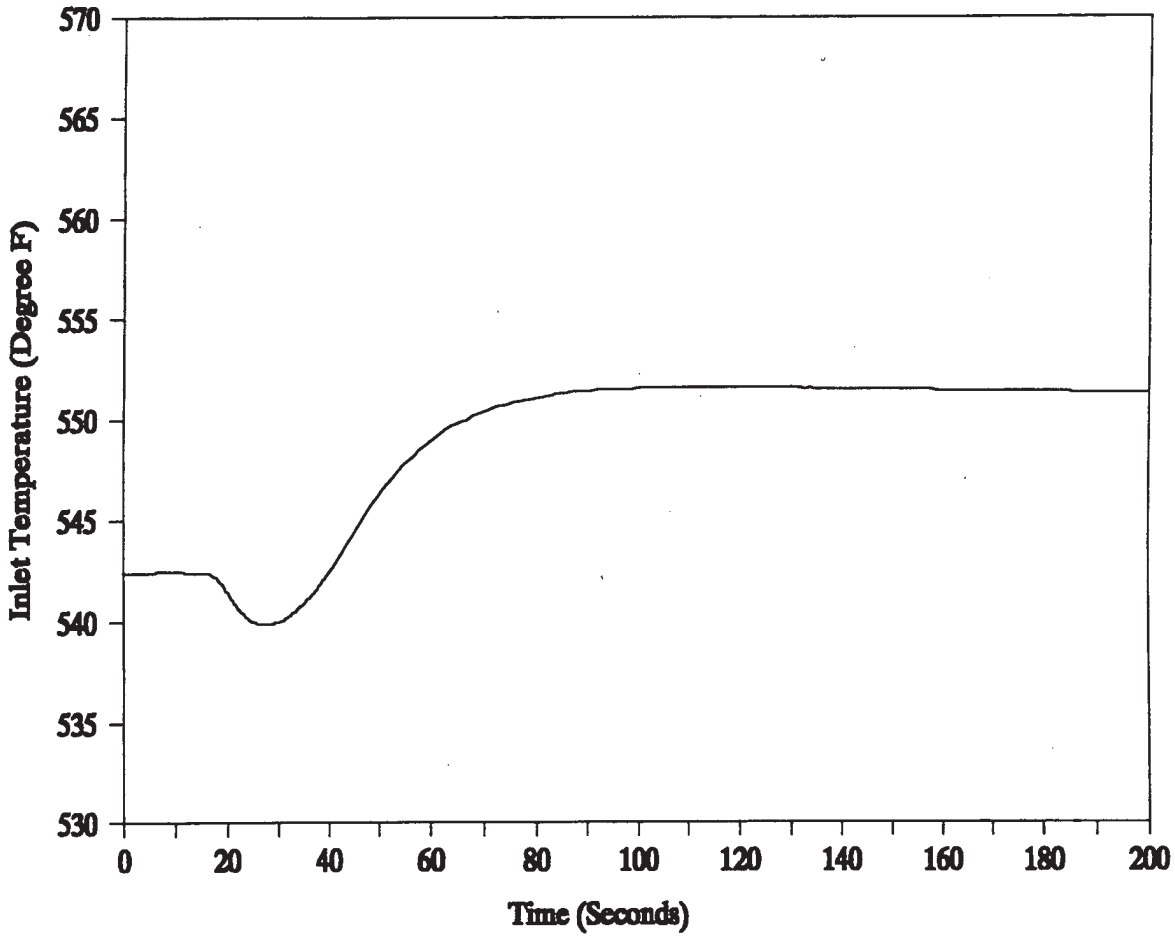


Figure 5.1.9-11

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Pressurizer Pressure versus Time

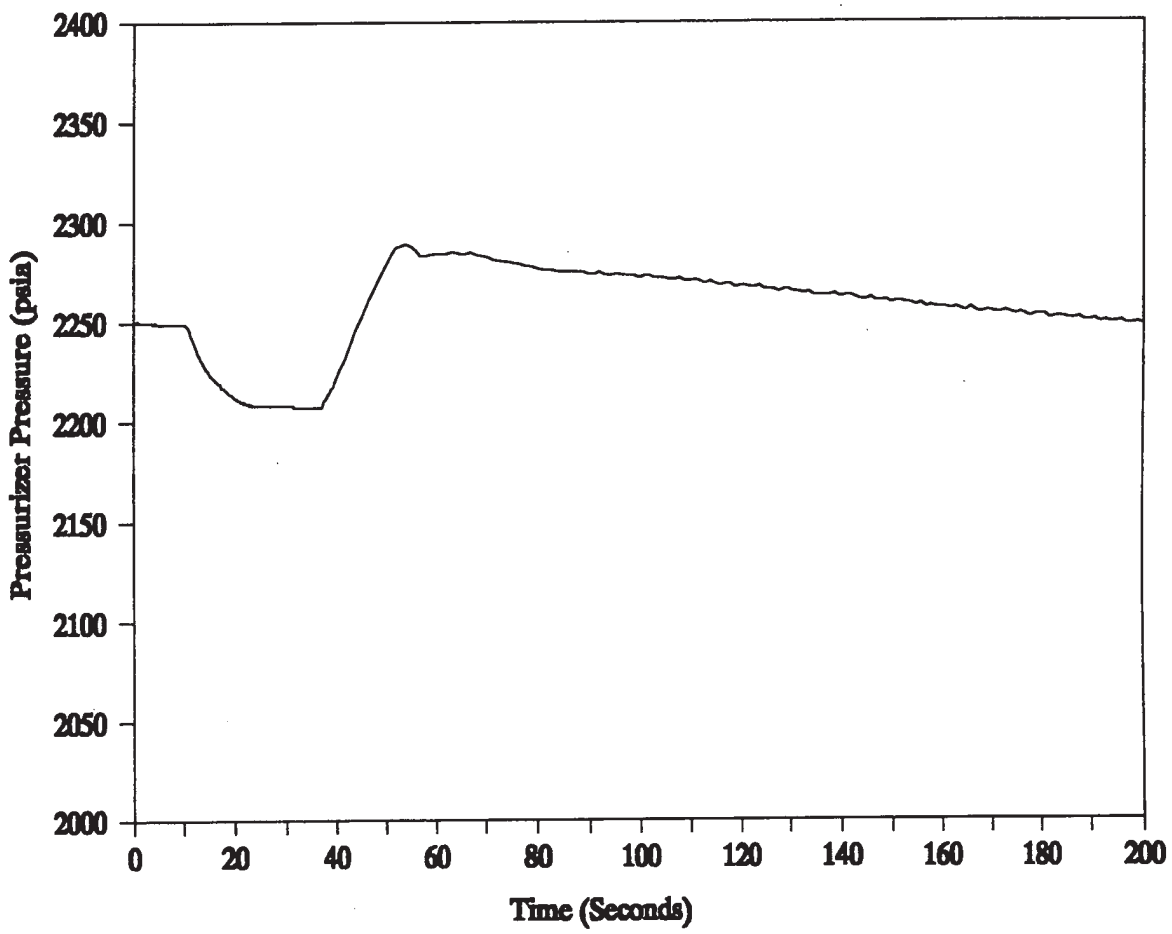


Figure 5.1.9-12

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 74% Power

Asymmetric Steam Generator Tube Plugging

Steam Flow versus Time

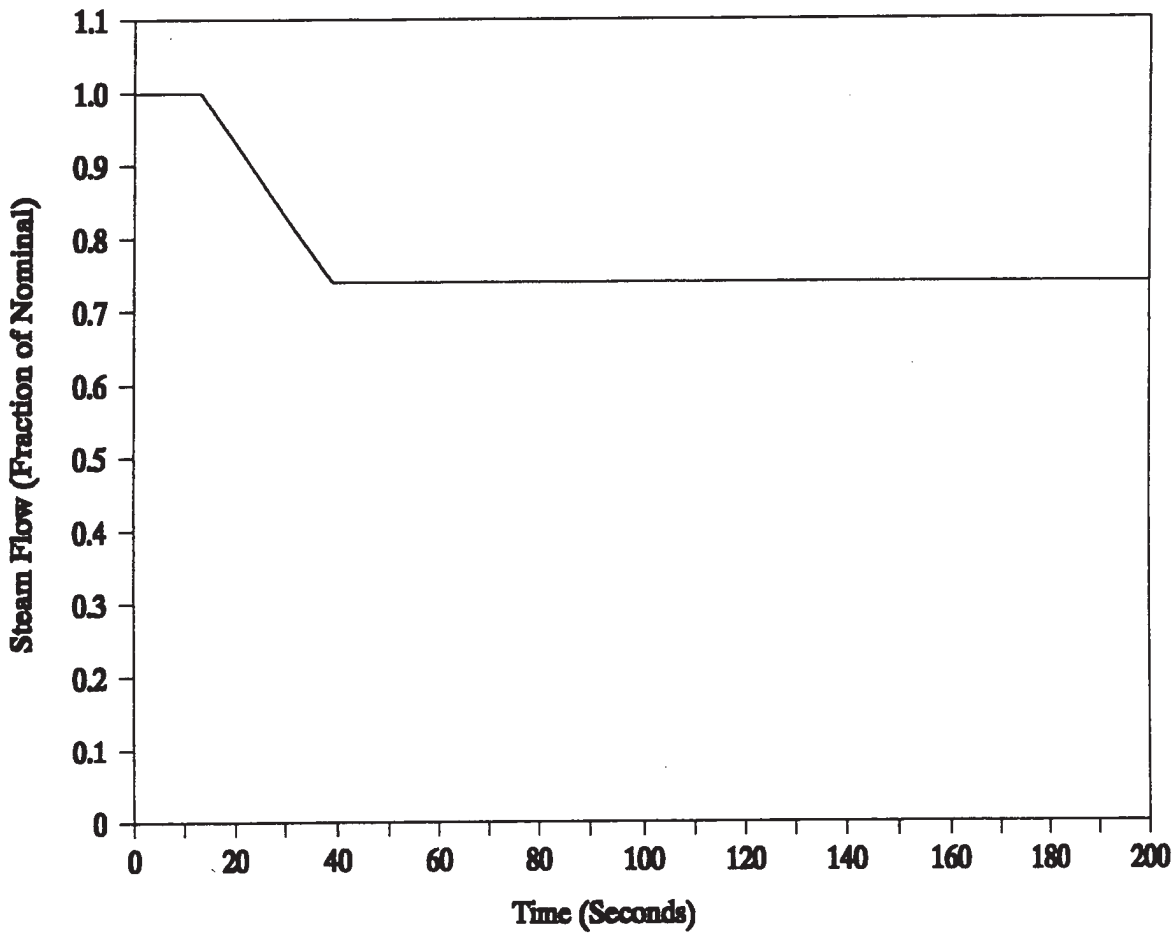


Figure 5.1.9-13

Rod Cluster Control Assembly Drop Event

Dropped Rod / 50 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Nuclear Power versus Time

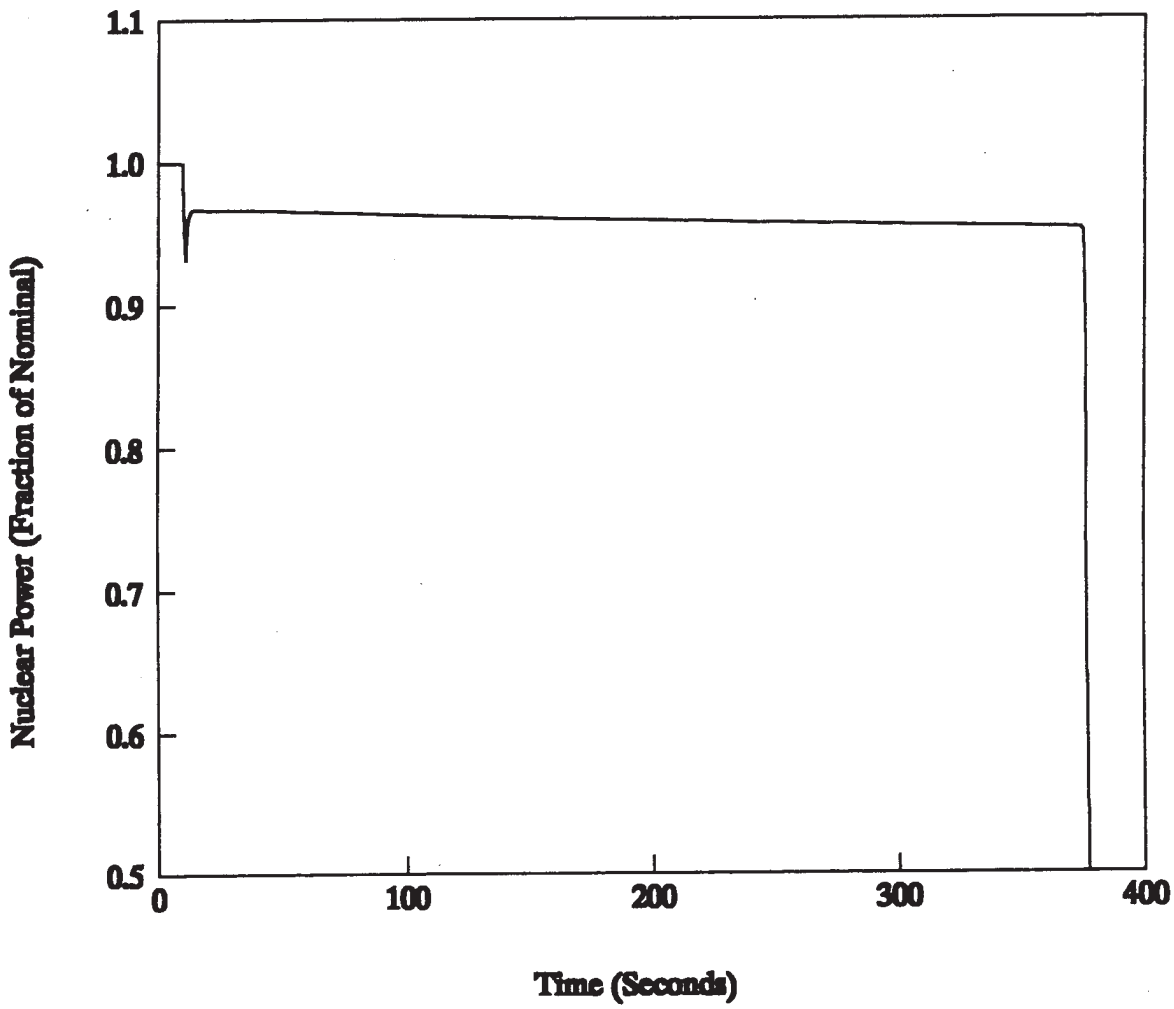


Figure 5.1.9-14

Rod Cluster Control Assembly Drop Event

Dropped Rod / 50 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Core Heat Flux versus Time

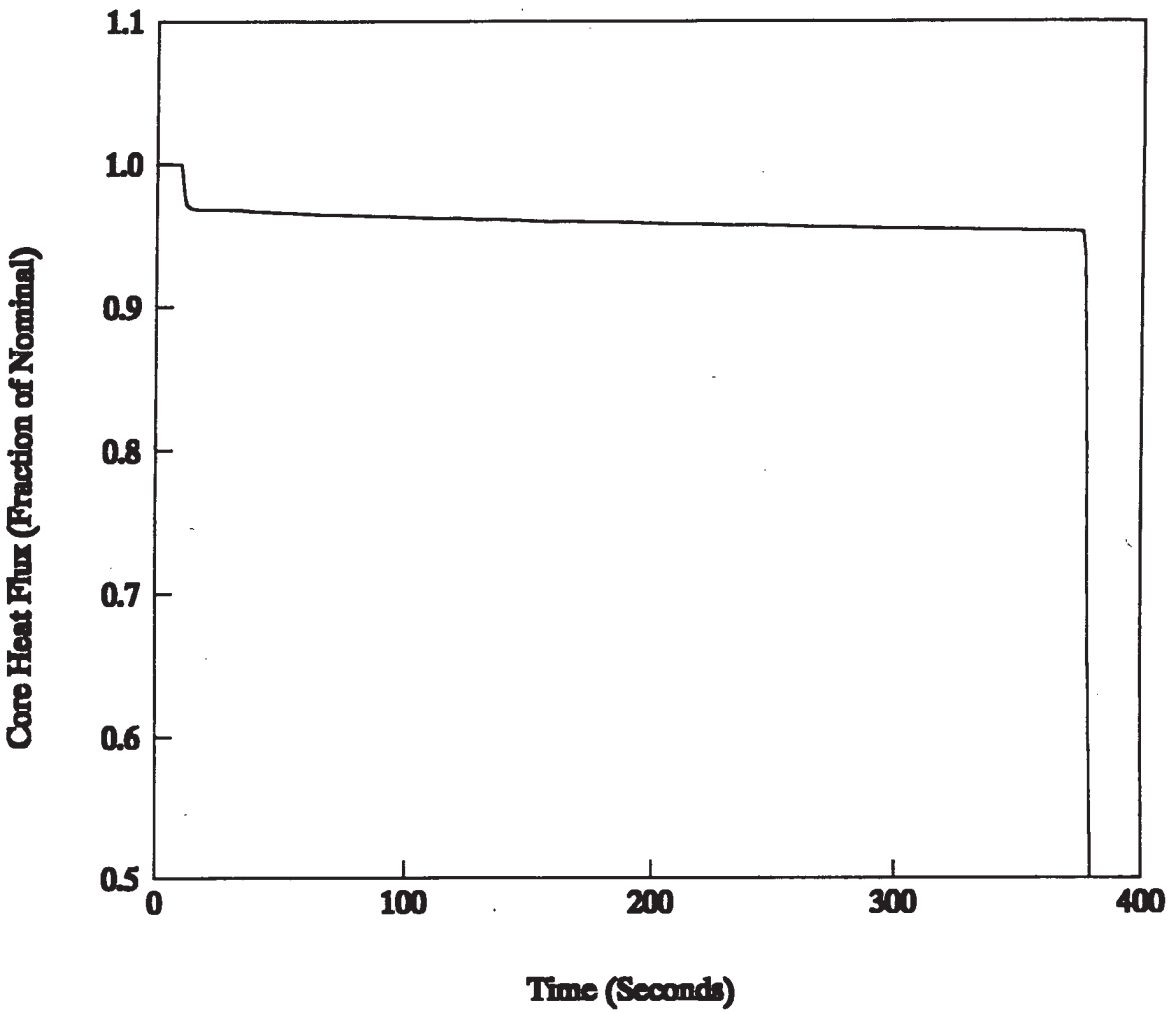


Figure 5.1.9-15

Rod Cluster Control Assembly Drop Event

Dropped Rod / 50 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Vessel Average Temperature versus Time

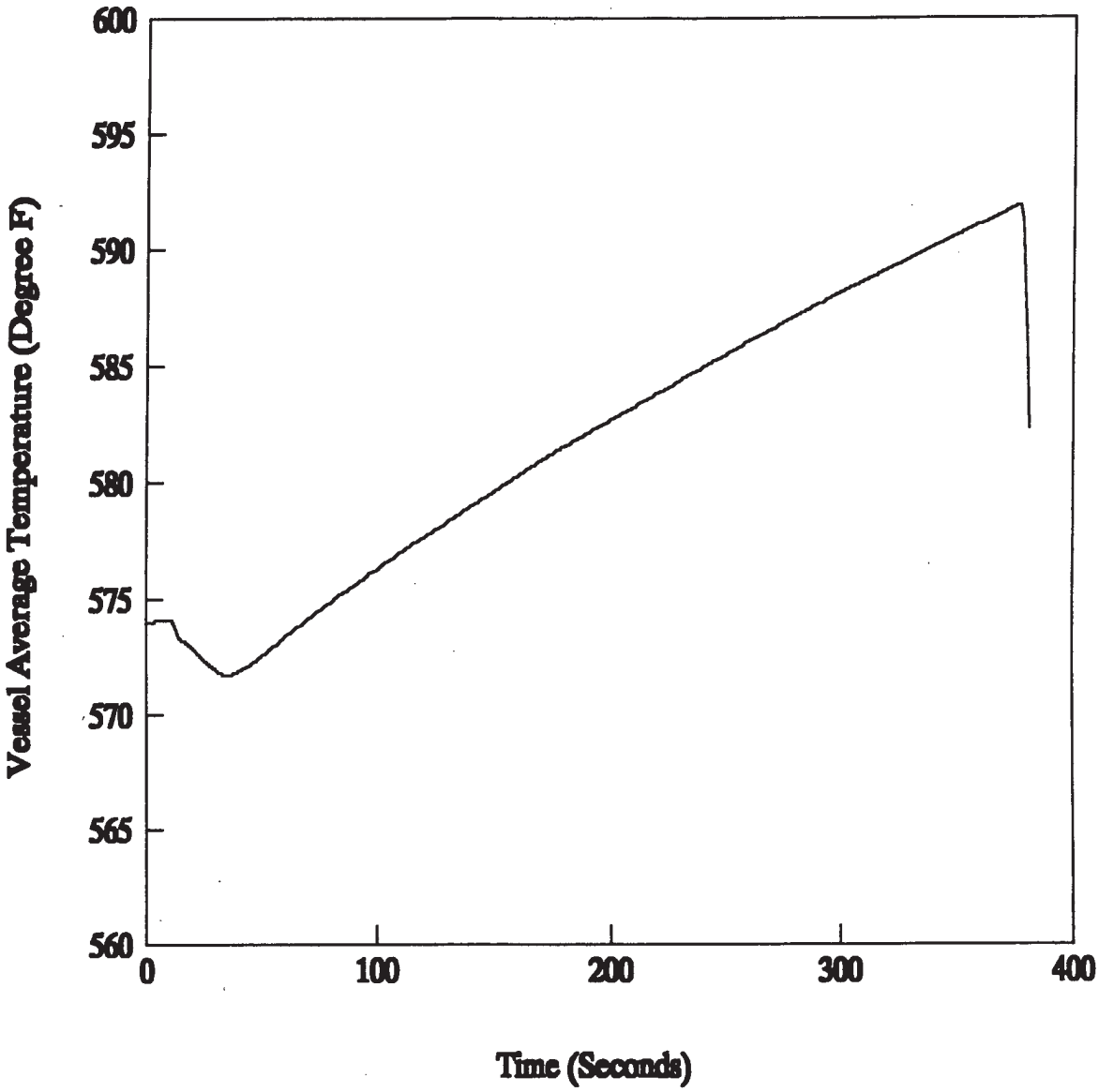


Figure 5.1.9-16

Rod Cluster Control Assembly Drop Event

Dropped Rod / 50 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Inlet Temperature versus Time

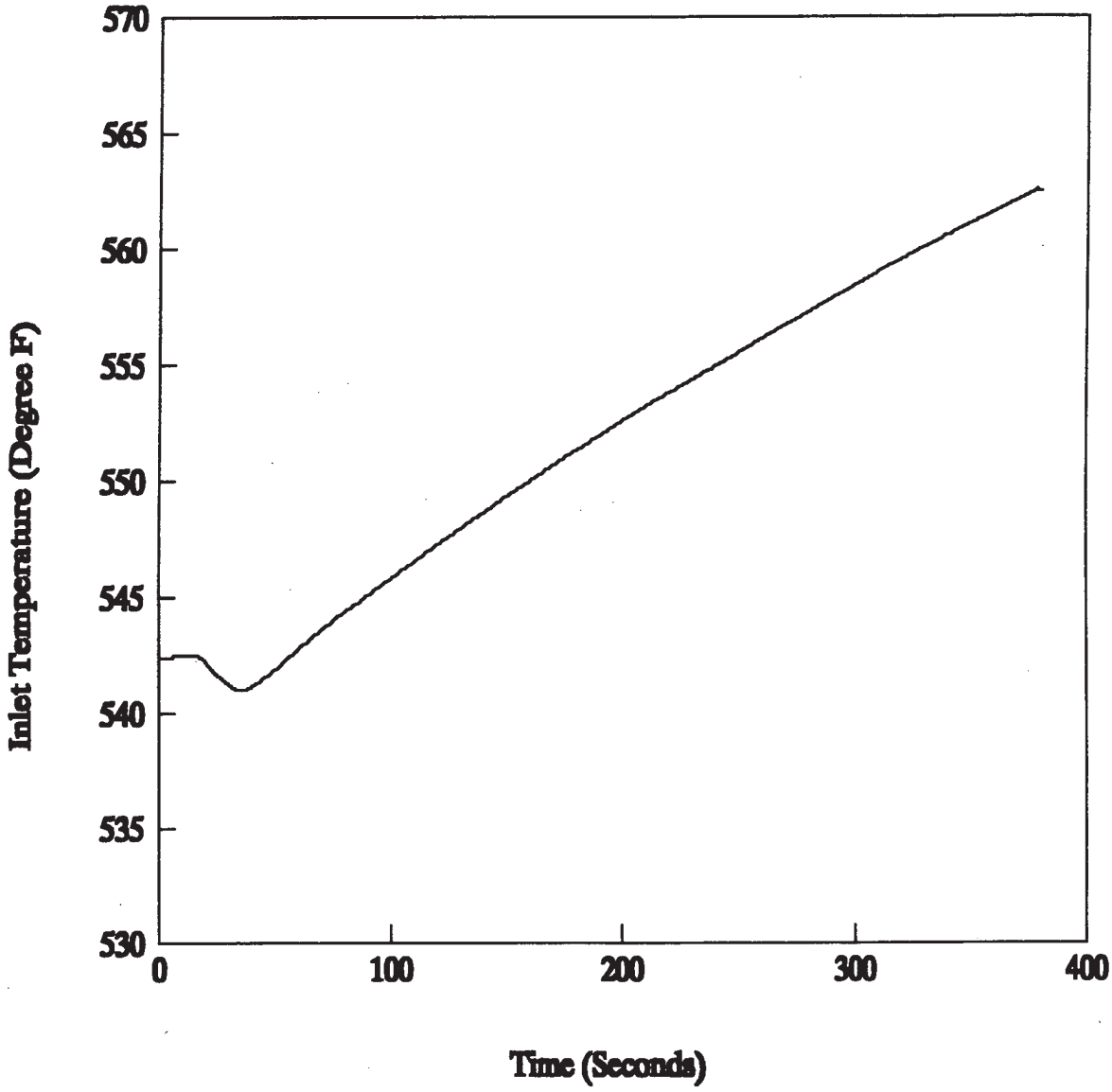


Figure 5.1.9-17

Rod Cluster Control Assembly Drop Event

Dropped Rod / 50 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Pressurizer Pressure versus Time

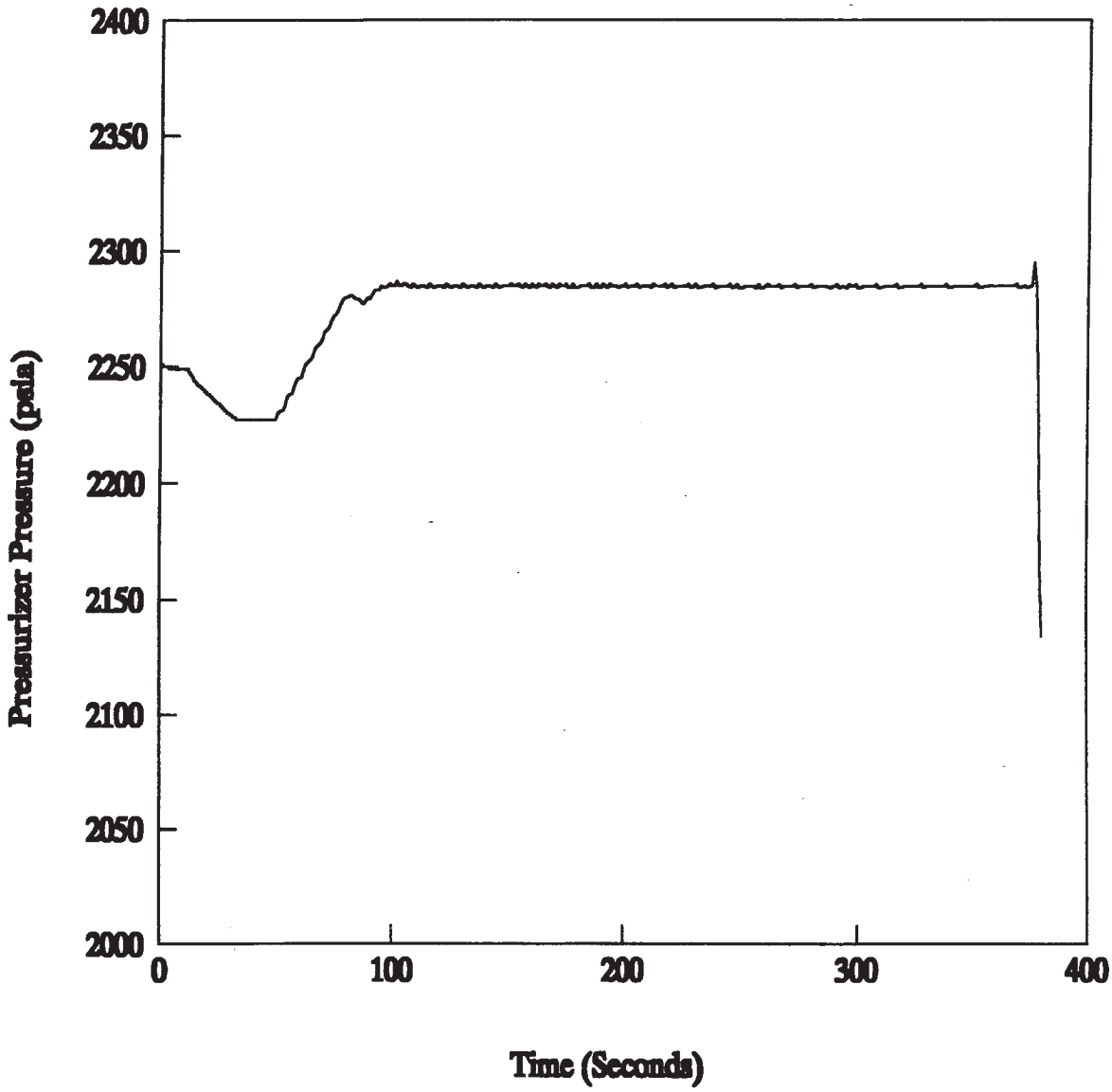


Figure 5.1.9-18

Rod Cluster Control Assembly Drop Event

Dropped Rod / 50 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Steam Flow versus Time

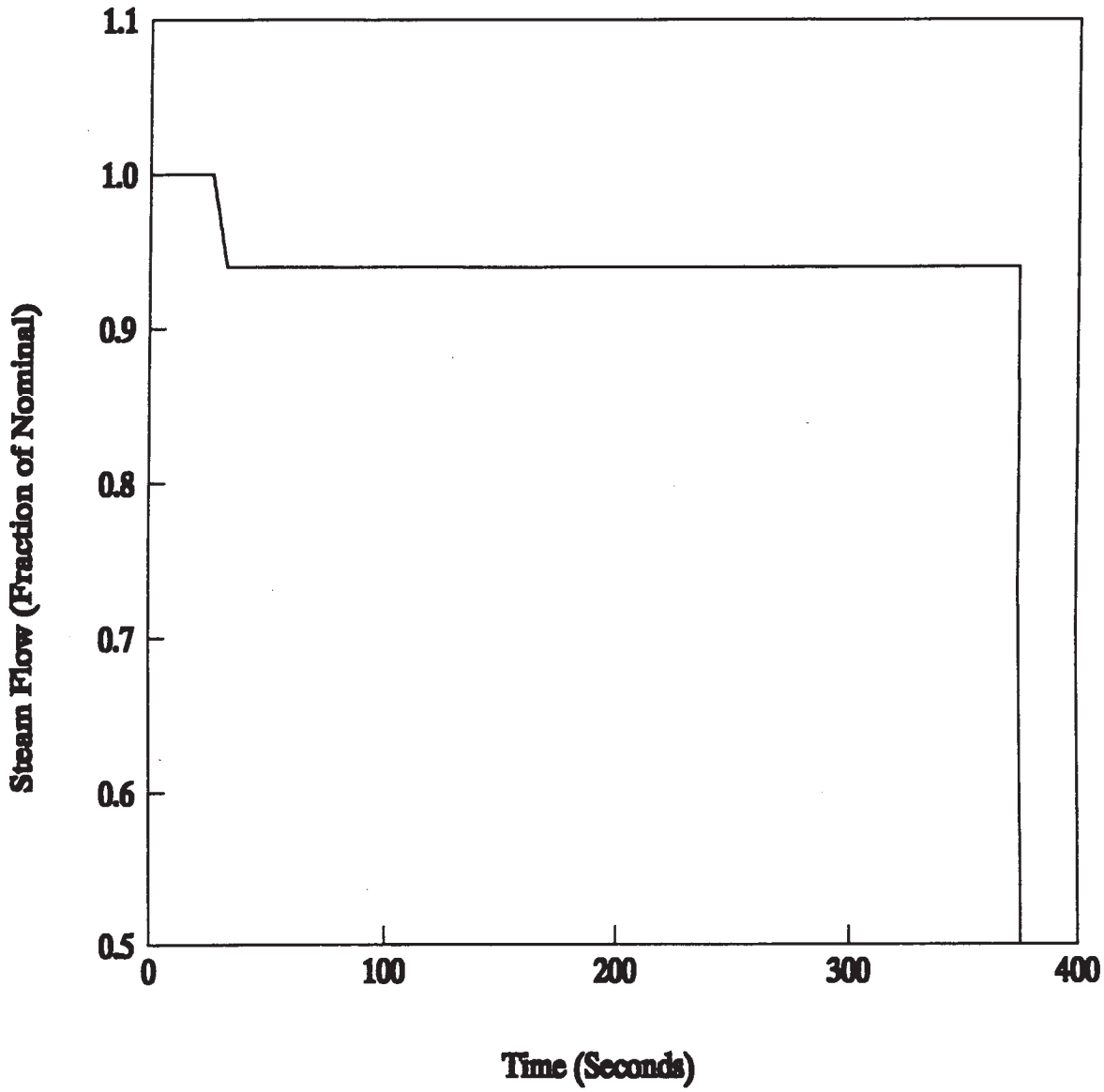


Figure 5.1.9-19

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Nuclear Power versus Time

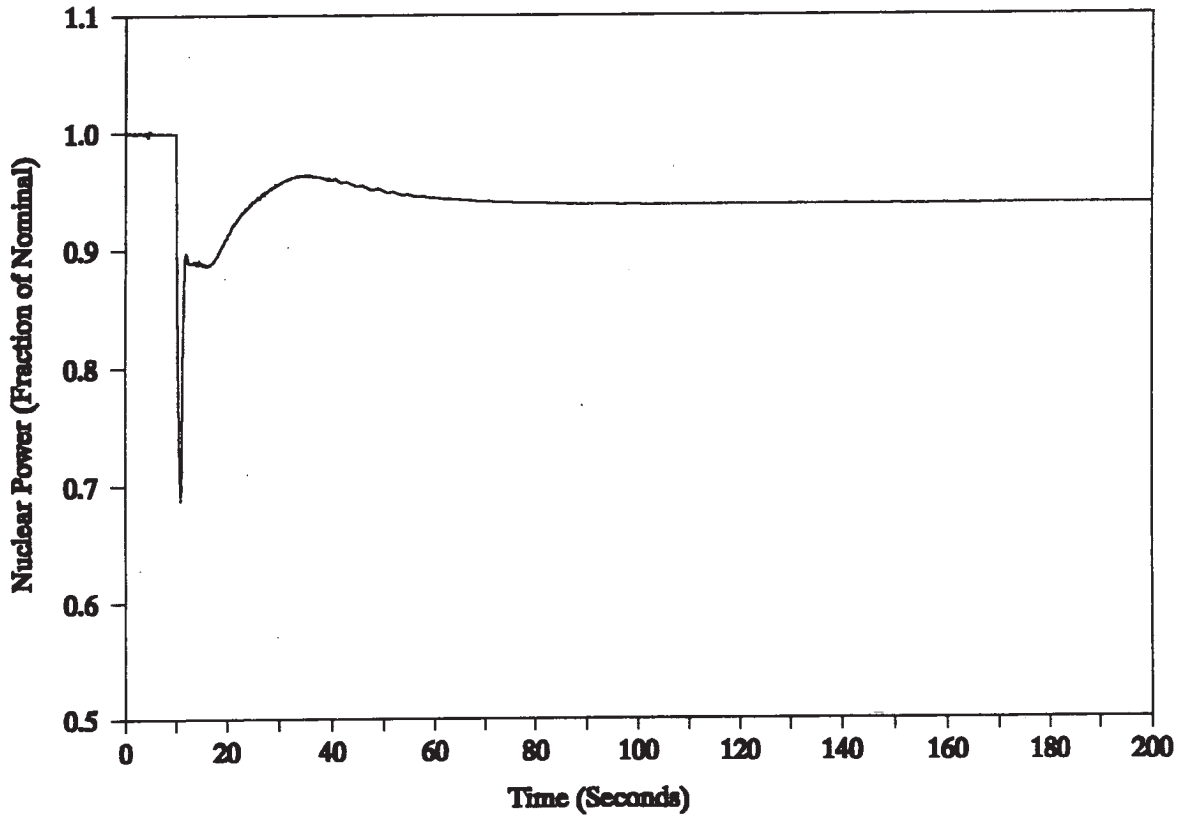


Figure 5.1.9-20

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Core Heat Flux versus Time

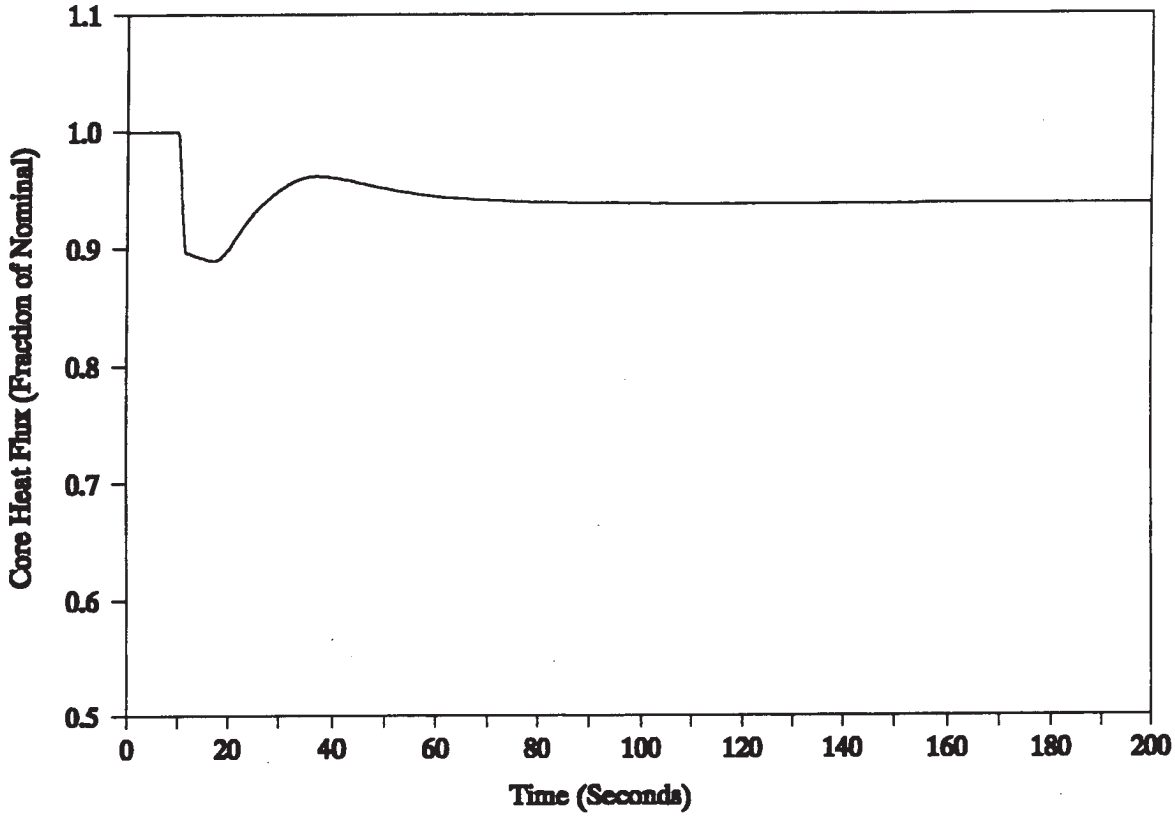


Figure 5.1.9-21

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Vessel Average Temperature versus Time

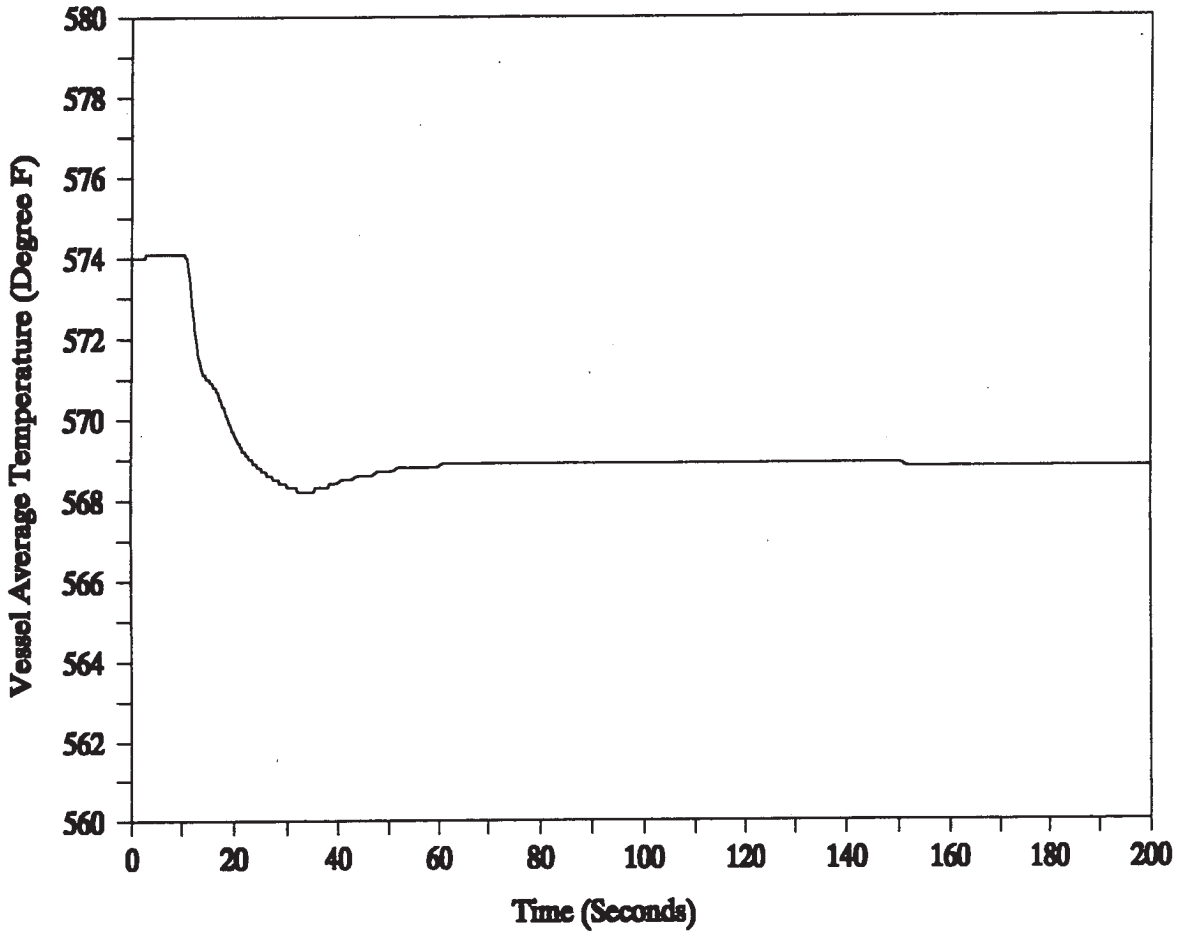


Figure 5.1.9-22

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Inlet Temperature versus Time

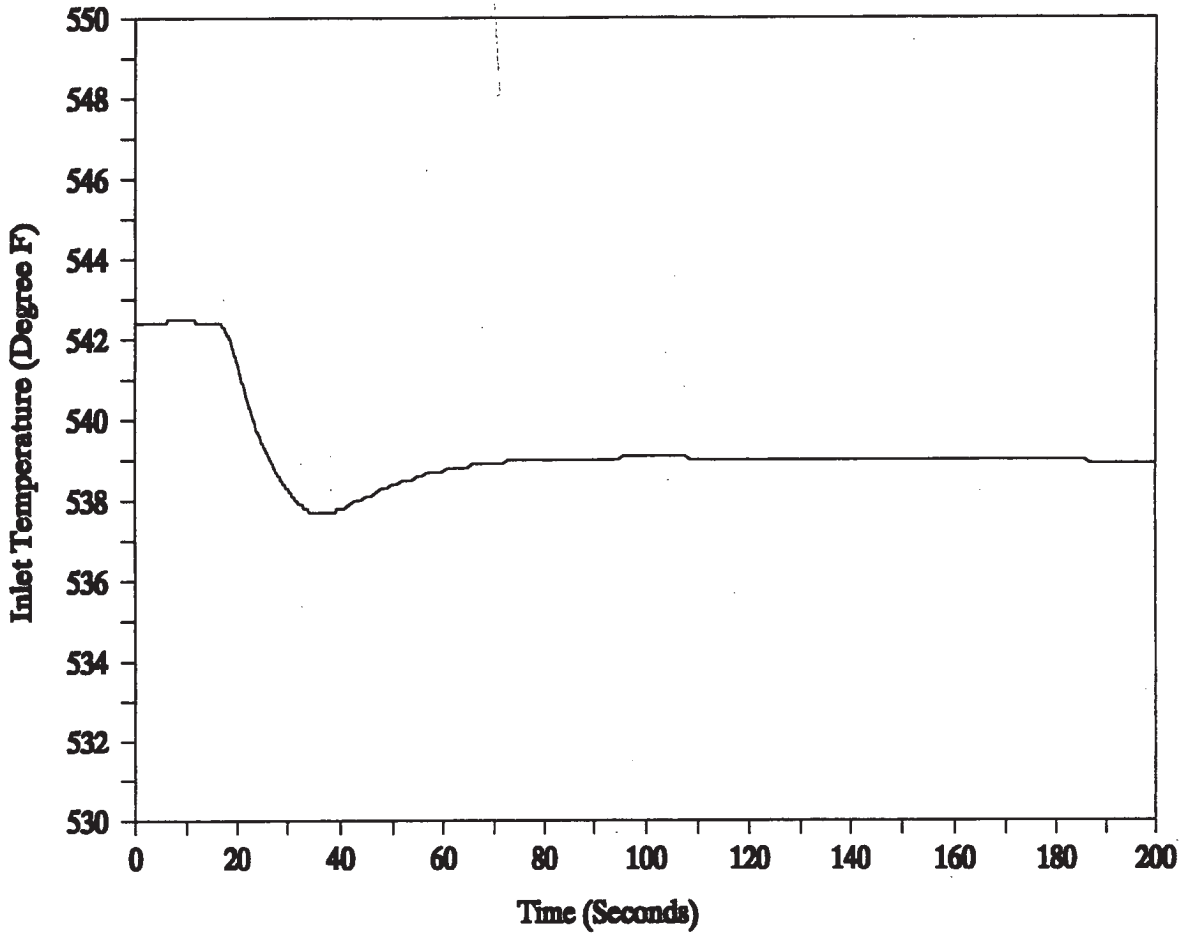


Figure 5.1.9-23

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Pressurizer Pressure versus Time

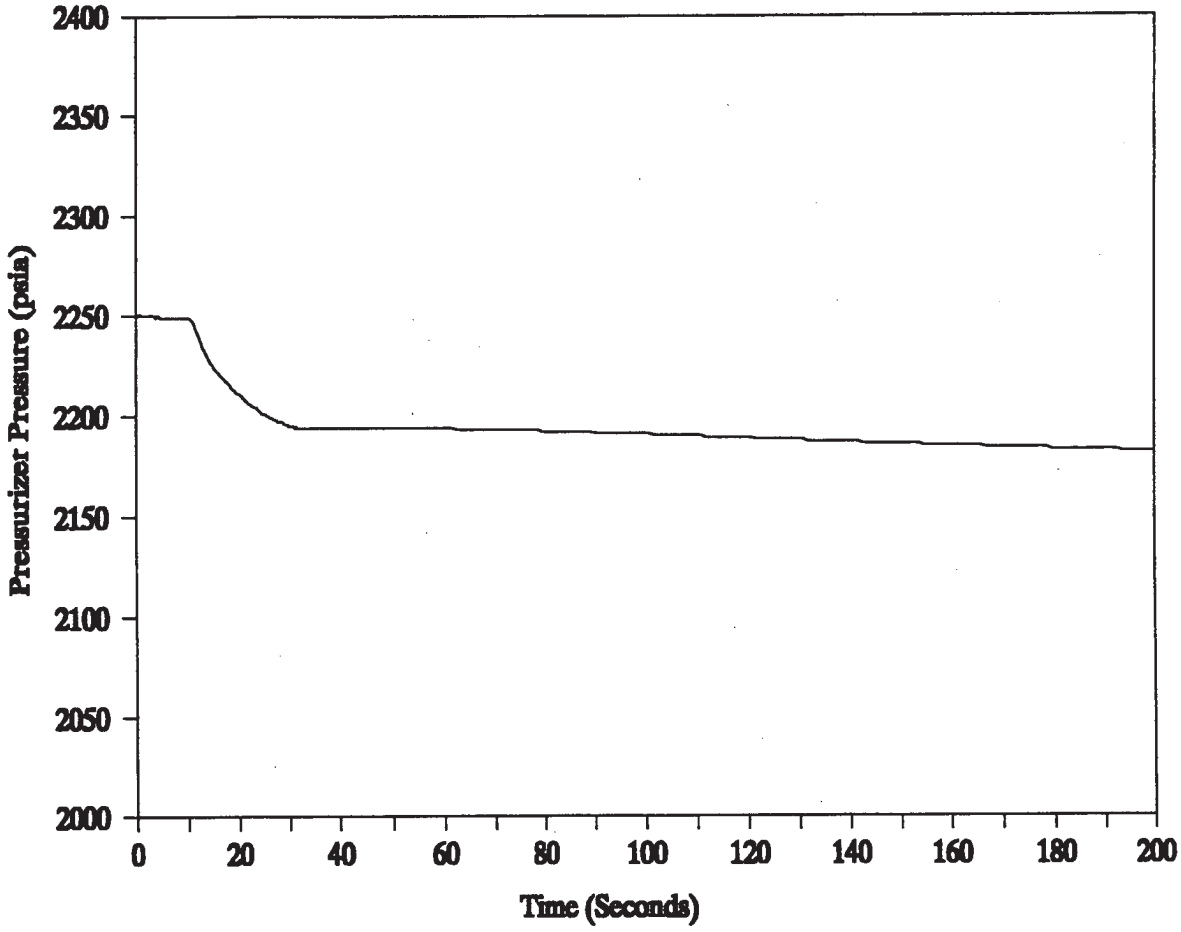


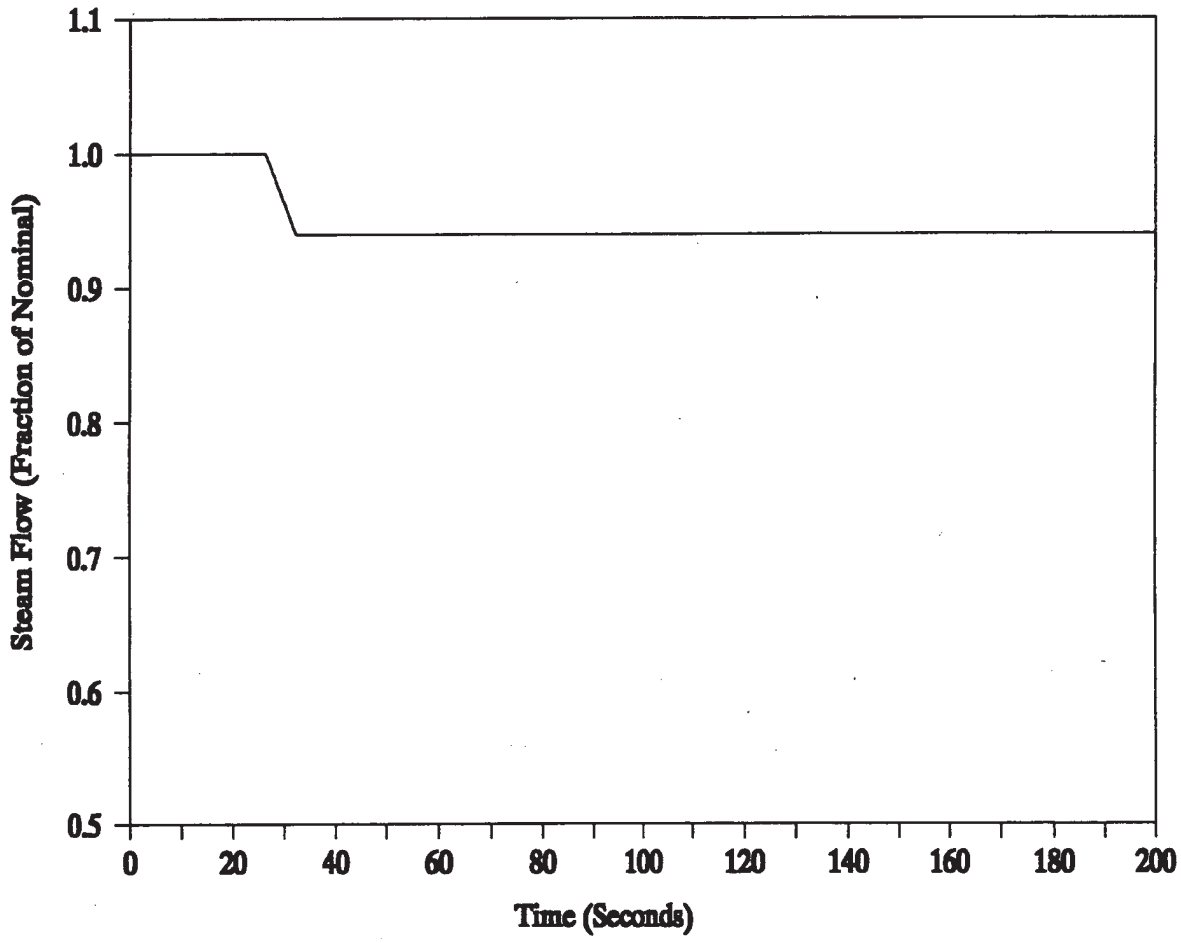
Figure 5.1.9-24

Rod Cluster Control Assembly Drop Event

Dropped Bank / 400 pcm / Turbine Runback to 94% Power

Asymmetric Steam Generator Tube Plugging

Steam Flow versus Time



5.1.10 Chemical and Volume Control System Malfunction

Introduction:

The evaluation herein was performed for the Chemical and Volume Control System Malfunction event as described in the FSAR Section 14.1.5 to support the insertion of VANTAGE + Fuel with the design features described in Section 5.1.2. The evaluation also address changes in the safety analysis assumptions associated with the VANTAGE + transition as described in Section 5.1.3.

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

There is only a single, common source of reactor makeup water to the reactor coolant system from the reactor makeup water system; inadvertent dilution can be readily terminated by isolating this single source. The operation of the reactor makeup water pumps that take suction from this tank provides the only supply of makeup water to the reactor coolant system. In order for makeup water to be added to the reactor coolant system, the charging pumps must be running in addition to the reactor makeup water pumps.

The rate of addition of unborated water makeup to the reactor coolant system is limited to the capacity of the charging pumps when the RCS is at pressure. This addition rate is 262 gpm for all three charging pumps. Normally, only one charging pump is operating while the others are on standby. With the reactor coolant system at atmospheric pressure, the rate of addition of unborated makeup water is limited to 300 gpm by the capacity of the primary makeup water pumps.

The boric acid from the boric acid tank is blended with the reactor makeup water in the blender, and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the reactor makeup control.

Two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, the start-stop switch is moved to start. Omitting either step would prevent dilution. This significantly reduces the possibility of inadvertent dilution.

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

Evaluation:

In all cases, the boron dilution analysis is performed to show that the operator has adequate time to diagnose the event and terminate the source of the dilution prior to reaching core criticality. Moreover, as described in FSAR Section 14.1.5 for the Chemical and Volume Control System Malfunction events, the analyses for these events are independent of fuel type and fuel performance related analysis assumptions.

Therefore, the VANTAGE + Fuel features and other revised safety analysis assumptions as described in Sections 5.1.2 and 5.1.3, respectively, will not have any adverse affect on the existing licensing basis analysis of the Chemical and Volume Control System Malfunction events. Hence, the current licensing basis analyses remain valid for the transition to VANTAGE + Fuel.

Conclusions:

Based on the evaluation herein, it is concluded that the existing licensing basis safety analysis for the Chemical and Volume Control System Malfunction events as discussed in the Evaluation section remain valid for the insertion of VANTAGE + Fuel into Indian Point 3.

Specifically;

a) The existing licensing basis analysis for Refueling shows that more than 30 minutes is required to dilute from a boron concentration of 1900 ppm to a boron concentration of 1330 ppm before the reactor will go critical. This is ample time for the operator to recognize the audible high count-rate signal and isolate the reactor makeup source by closing valves and stopping the reactor makeup water pumps.

b) The existing licensing basis analysis for dilution during Startup, which assumes an initial reactor coolant system boron concentration of 1800 ppm, shows from initiation of the event, there are greater than 15 minutes available for operator action prior to return to criticality at a boron concentration of 1550 ppm.

c) The existing licensing basis analysis for dilution during Power Operation assumes an initial reactor coolant system boron concentration of 1800 ppm and a critical boron concentration of 1450 ppm. The analysis shows the rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator at least 38 minutes prior to criticality. This is sufficient time to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

Moreover, with the reactor in manual control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach the OTΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution event is conservatively estimated at about 2.0 pcm/sec, which is within the range of insertion rates analyzed. Thus, the effects of dilution prior to reactor trip are bounded by the Uncontrolled RCCA Bank Withdrawal at Power analysis as documented in Section 5.1.7. Following reactor trip there are greater than 15 minutes prior to criticality. This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

The aforementioned conclusion is valid for the VANTAGE + Fuel including the design features and associated changes in the safety analysis assumptions as described in Sections 5.1.2 and 5.1.3, respectively.

5.1.11 Loss of Reactor Coolant Flow

The analyses and evaluations documented herein were performed to support the insertion of VANTAGE + Fuel and other design changes associated with the VANTAGE + reload transition as described in Sections 5.1.2 and 5.1.3 and to show these changes are acceptable for the Loss of Reactor Coolant Flow events (presented in FSAR Section 14.1.6):

a) Partial Loss of Forced Reactor Coolant Flow
b) Complete Loss of Forced Reactor Coolant Flow
c) Reactor Coolant Pump Shaft Seizure (Locked Rotor)

d) Reactor Coolant Pump Shaft Break (Reverse Flow)

Partial and Complete Loss of Forced Reactor Coolant Flow

Introduction:

A partial loss of coolant flow can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump supplied by a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

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

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

The partial loss of flow event is classified as an ANS Condition II fault as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" and the complete loss of flow event is classified as a Condition III event. A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious fault, i.e, a Condition III or IV category event. A Condition III occurrence is defined as a infrequent fault and should not propagate to cause a more serious fault, i.e, a Condition IV category event.

The necessary protection against a loss of coolant flow accidents is provided by the following reactor protection signals:

a) Reactor coolant pump power supply undervoltage or underfrequency.

b) Low reactor coolant loop flow.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e, station blackout. This function is blocked below approximately 10 percent power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid.

The reactor trip on low primary coolant flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two-out-of-three low flow signals per reactor coolant loop. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip.

Although the complete loss of flow event is defined as a Condition III event, both the complete and partial loss of flow events are analyzed to Condition II criteria. The applicable safety analysis licensing basis acceptance criteria for Condition II events are:

a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (i.e., 2748.5 psia and 1208.5 psia, respectively),

b) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit, and

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

Method of Analysis and Assumptions:

The loss of one reactor coolant pump with four loops in operation and the loss of all reactor coolant pumps in operation events are analyzed to show that: 1) the integrity of the core is maintained as the DNBR remains above the safety analysis limit value; 2) the peak RCS and secondary system pressures remain below 110% of the design limits; and 3) the pressurizer does not fill. Of these, the primary concern is DNB and assuring that the DNBR limit is met.

The loss of flow events are analyzed with three computer codes. First, the LOFTRAN computer code (see Section 5.1.5) is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN computer code (see Section 5.1.5) is then used to calculate the heat flux transient based on the nuclear power and RCS flow from LOFTRAN. Finally, the THINC computer code (see Section 5.1.5) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and RCS flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

The loss of flow accidents are analyzed with the RTDP as discussed in Section 5.1.3. Initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values.

A conservatively large absolute value of the Doppler-only power coefficient is used. This results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

A conservative trip reactivity of 4% Δk is used and is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time of 2.7 seconds.

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

The analysis is performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq$ 24%; and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of $\leq$ 24% with a maximum steam generator tube plugging level of 30% in any one steam generator.

Results:

Figures 5.1.11-1 through 5.1.11-6 illustrate the transient response for the loss of one reactor coolant pump with four loops in operation. Figure 5.1.11-6 shows that the DNBR always remains above the limit value. This demonstrates the ability of the primary coolant to remove heat from the fuel rods is not greatly reduced.

Figures 5.1.11-7 through 5.1.11-11 illustrate the transient response for the loss of power to all four reactor coolant pumps with four loops in operation. Figure 5.1.11-11 shows that the DNBR always remains above the limit value. This too demonstrates the ability of the primary coolant to remove heat from the fuel rods is not greatly reduced.

The calculated sequence of events for both loss of flow events are shown on Tables 5.1.11-1 and 5.1.11-2.

For the partial loss of flow event, a reactor trip occurs on a low primary reactor coolant flow trip setpoint which is assumed to be 87% of nominal flow. Following reactor trip, the affected reactor coolant pump will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the number of pumps still in operation.

For the complete loss of flow event, a reactor trip occurs on an undervoltage trip setpoint. Following reactor trip, the reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established, as demonstrated in FSAR Section 14.1.6.

With the reactor tripped, a stable plant condition will eventually be attained in both cases. Normal plant shutdown may then proceed.

Conclusions:

The analyses performed has demonstrated that for the loss of a reactor coolant pump event, the DNBR does not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted and all applicable acceptance criteria are met.

Reactor Coolant Pump Shaft Seizure (Locked Rotor):

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

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and

then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon 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, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis.

This event is classified as an ANS Condition IV fault as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." A Condition IV occurrence is defined as a limiting fault. The limits are that the RCS and the core must remain able to provide long term cooling, and offsite dose must remain within the guidelines of 10 CFR 100. The specific (and more restrictive) criteria that Westinghouse uses to ensure that these limits are not violated are as follows:

1) Average fuel pellet enthalpy at the hot spot must be below 200 cal/gm (360 Btu/lbm).

2) Average clad temperature at the hot spot must remain below 2700 °F.

3) Fuel melting will be limited to less than 10% of the fuel volume at the hot spot.

4) The peak reactor coolant pressure must remain less than that which would cause stresses to exceed the Faulted Condition stress limits (approximately 3000 psia).

The necessary protection against a RCP Shaft Seizure accident is provided by the low primary coolant flow reactor trip signal which is actuated in any reactor coolant loop by two out of three low flow signals.

Method of Analysis and Assumptions:

The RCP Shaft Seizure transient is analyzed with two computer codes. First, the LOFTRAN computer code (see Section 5.1.5) is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN computer code (see Section 5.1.5) is then used to calculate

the thermal behavior of the fuel located at the core hot spot based on the nuclear power and RCS flow from LOFTRAN. The FACTRAN computer code includes a film boiling heat transfer coefficient.

At the beginning of the postulated RCP Shaft Seizure accident, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., the maximum guaranteed steady state thermal power, maximum steady state pressure, and maximum steady state coolant average temperature.

The analysis is performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq$ 24%; and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of $\leq$ 24% with a maximum steam generator tube plugging level of 30% in any one steam generator.

A conservatively large absolute value of the Doppler-only power coefficient is used. This results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

A conservative trip reactivity of 4% Δk is used and is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time of 2.7 seconds.

For the peak RCS pressure evaluation, the initial pressure is conservatively estimated as 60 psi above the nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. The RCS pressure response is shown in Figure 5.1.11-13 at the point in the reactor coolant system having the maximum pressure (i.e., at the reactor coolant pump outlet).

For this accident, DNB is assumed to occur in the core, and therefore, an evaluation of the consequences with respect to the fuel rod thermal transients is performed. Results obtained from the analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium water reaction. In the evaluation, the rod power at the hot spot is assumed to be 2.6 times the average rod power (i.e., Fq= 2.6) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature. The program calculates the

film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are based on the LOFTRAN results.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between the pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady-state value consistent with initial fuel temperature to 10,000 BTU/hr-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 is released to the clad at the initiation of the transient.

Zirconium Steam Reaction

The zirconium-steam reaction can become significant above 1,800 °F (clad temperature). The Baker-Just parabolic rate equation is used to define the rate of zirconium-steam reaction. The effect of the zirconium-steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Results:

Figures 5.1.11-12 through 5.1.11-16 illustrate the transient response for the RCP Shaft Seizure event. The peak reactor coolant system pressure is 2569 psia and is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak average clad temperature is 1,818 °F (including a 2 °F penalty due to the use of ZIRLO™ fuel cladding) which is considerably less than the limit of 2,700 °F. The peak fuel enthalpy is 207 BTU/lbm, the fuel melt is 0% and the maximum zirconium-steam reaction at the hot spot is 0.4% by weight.

The sequence of events is given in Table 5.1.11-3. A reactor trip occurs on a low primary reactor coolant flow trip setpoint which is assumed to be 87% of nominal flow.

For the case analyzed for DNB using the Resvised Thermal Design Procedure, the applicable DNB criterion is met. Hence, no "rods-in-DNB" are predicted for the RCP shaft seizure.

Conclusions:

All safety criteria (clad average temperature < 2,700 °F, fuel enthalpy < 360 BTU/lbm, fuel melt < 10%, and Zirc-H_2O reaction < 16%) are satisfied for all cases. This demonstrates that the RCS and the core will remain able to provide long term cooling, and off-site doses remain within the guidelines of 10CFR100 for the transition from the resident VANTAGE 5 fuel to the VANTAGE + fuel, including the design features and related changes in safety analysis assumptions described in Sections 5.1.2 and 5.1.3, respectively.

Reactor Coolant Pump Shaft Break

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. RCS 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. With a failed shaft the pump impeller could conceivably be free to spin in the reverse direction instead of being fixed in position. The effect of such reverse spinning is a slight decrease in the final (steady-stated) core flow.

The analysis presented under the RCP Shaft Seizure section represents the limiting condition, assuming a locked rotor for forward flow but a free-spinning shaft for reverse flow in the affected loop. Therefore, the conclusions for the RCP Shaft Seizure apply to and bound a reactor coolant pump shaft break accident.

Table 5.1.11-1

Sequence of Events

for the

Partial Loss of Flow Event

Event	Time of Event (sec)
Coastdown begins	0.0
Low flow reactor trip	1.8
Rods begin to drop	2.8
Minimum DNBR occurs	4.1

Table 5.1.11-2

Sequence of Events

for the

Complete Loss of Flow Event

Event	Time of Event (sec)
Coastdown begins	0.0
Undervoltage reactor trip	0.0
Rods begin to drop	1.5
Minimum DNBR occurs	3.6

Table 5.1.11-3

Sequence of Events

for the

RCP Shaft Seizure Event

Event	Time of Event (sec)
Rotor on one pump locks	0.0
Low flow reactor trip	0.05
Rods begin to drop	1.05
Maximum RCS pressure occurs	3.2
Maximum clad temperature occurs	3.6

Figure 5.1.11-1
Nuclear Power for Partial Loss of Flow
Four Loops in Operation,
One Pump Coasting Down

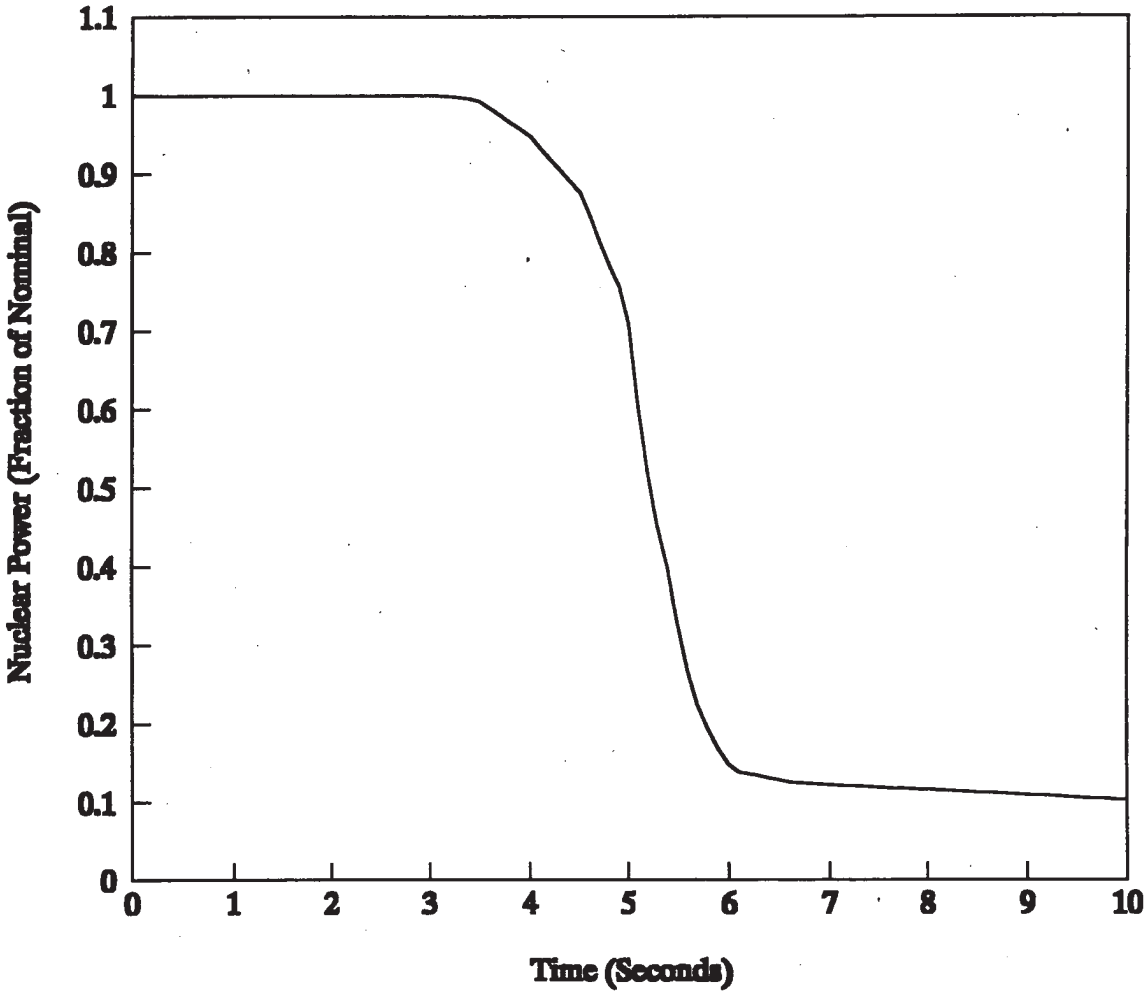


Figure 5.1.11-2
Pressurizer Pressure for Partial Loss of Flow
Four Loops in Operation,
One Pump Coasting Down

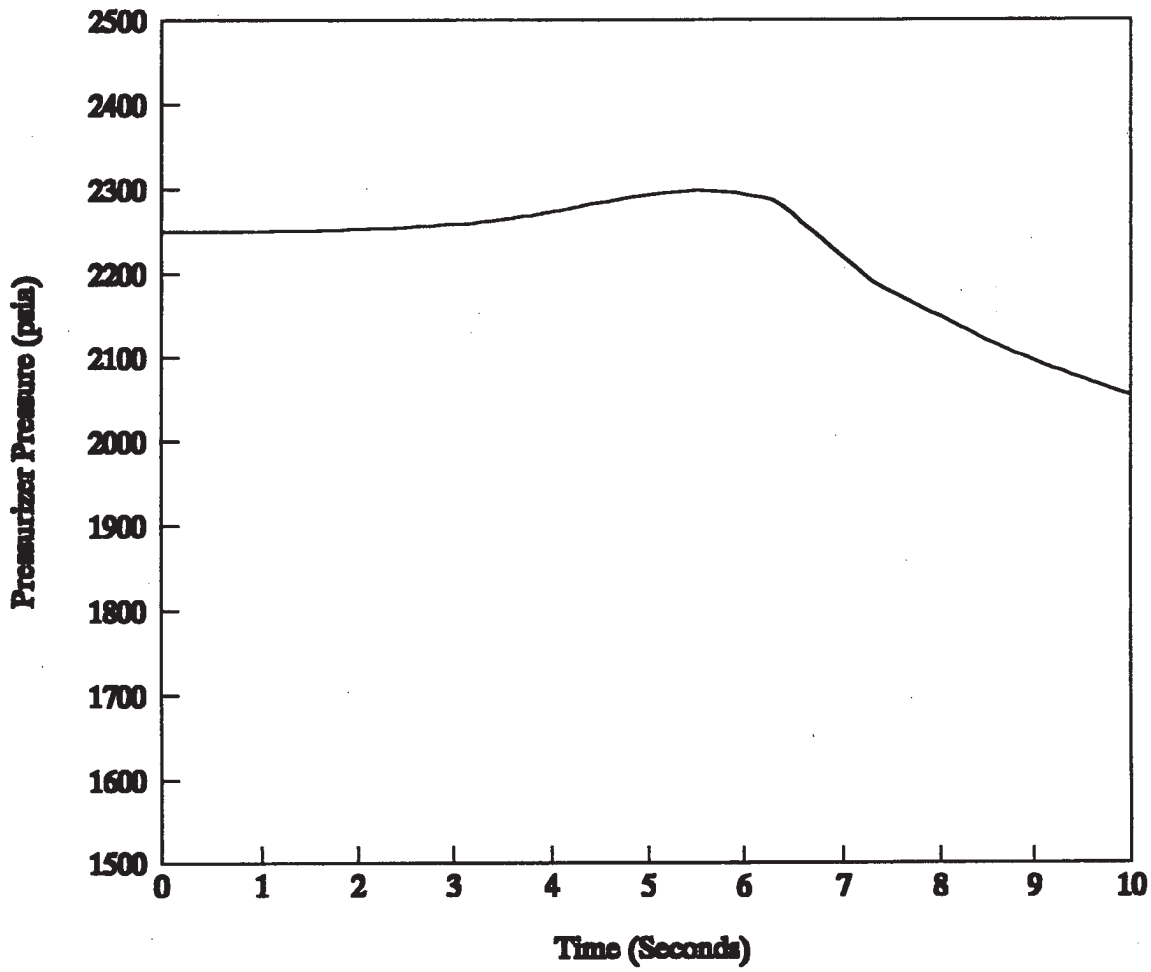


Figure 5.1.11-3
Total RCS Flow for Partial Loss of Flow
Four Loops in Operation,
One Pump Coasting Down

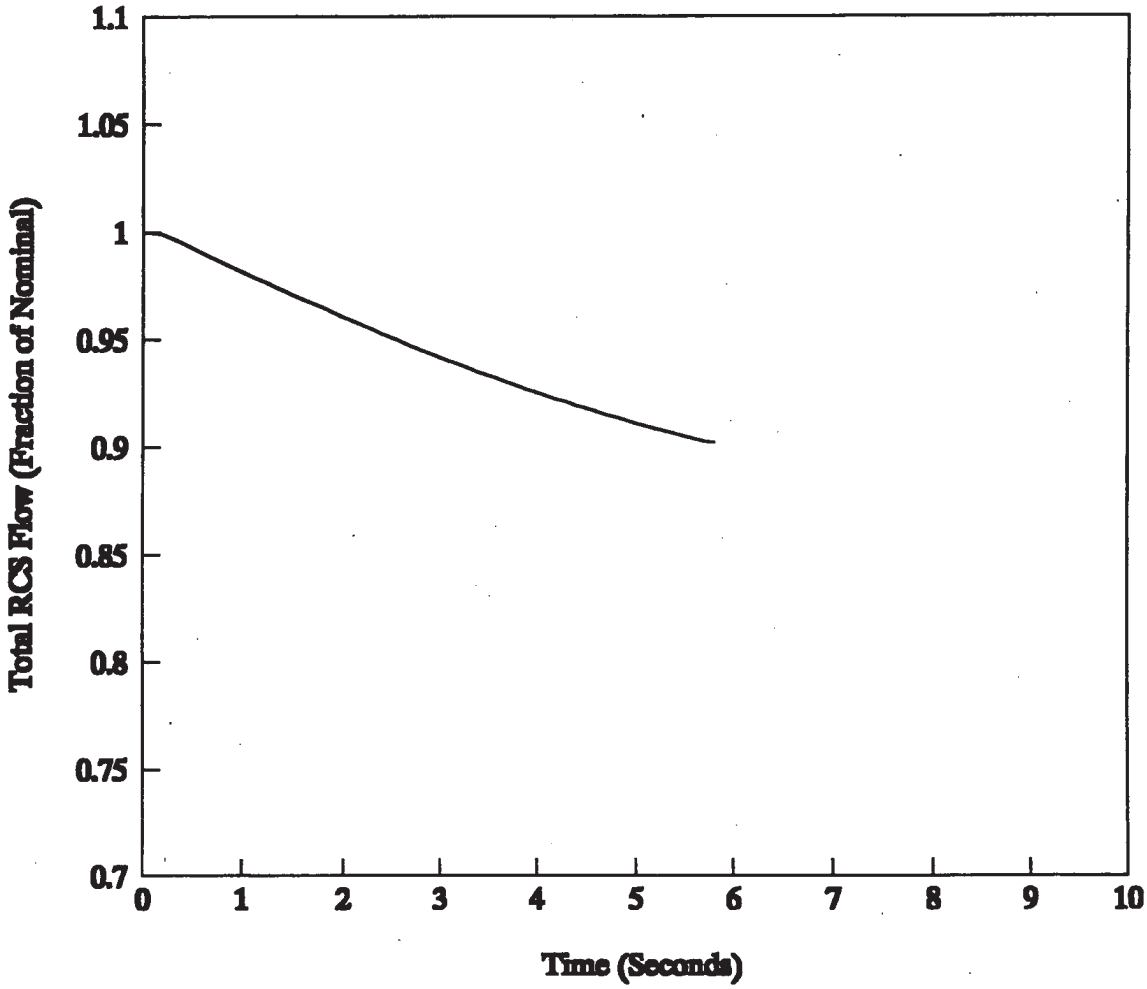


Figure 5.1.11-4
Loop RCS Flow for Partial Loss of Flow
Four Loops in Operation,
One Pump Coasting Down

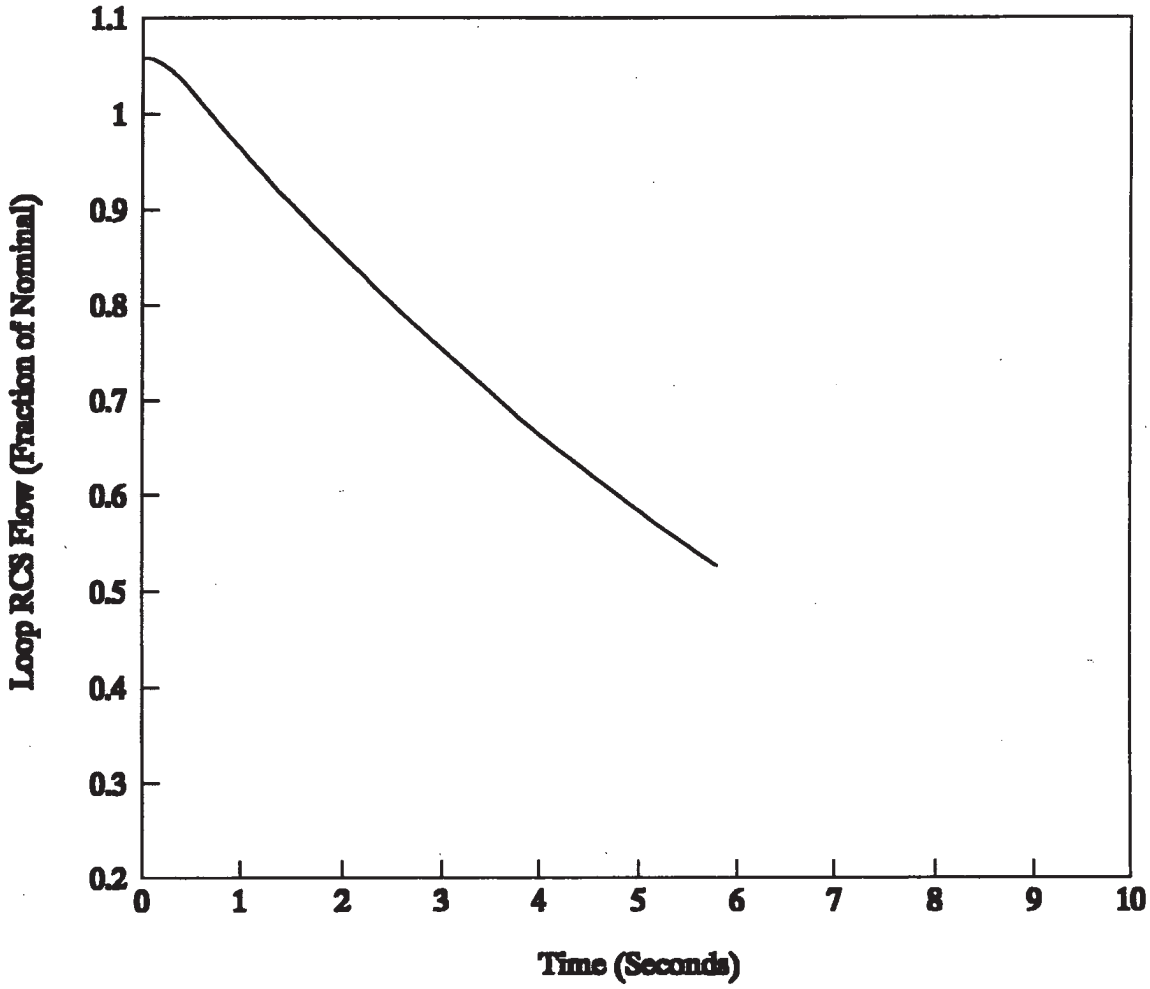


Figure 5.1.11-5
Heat Flux for Partial Loss of Flow
Four Loops in Operation,
One Pump Coasting Down

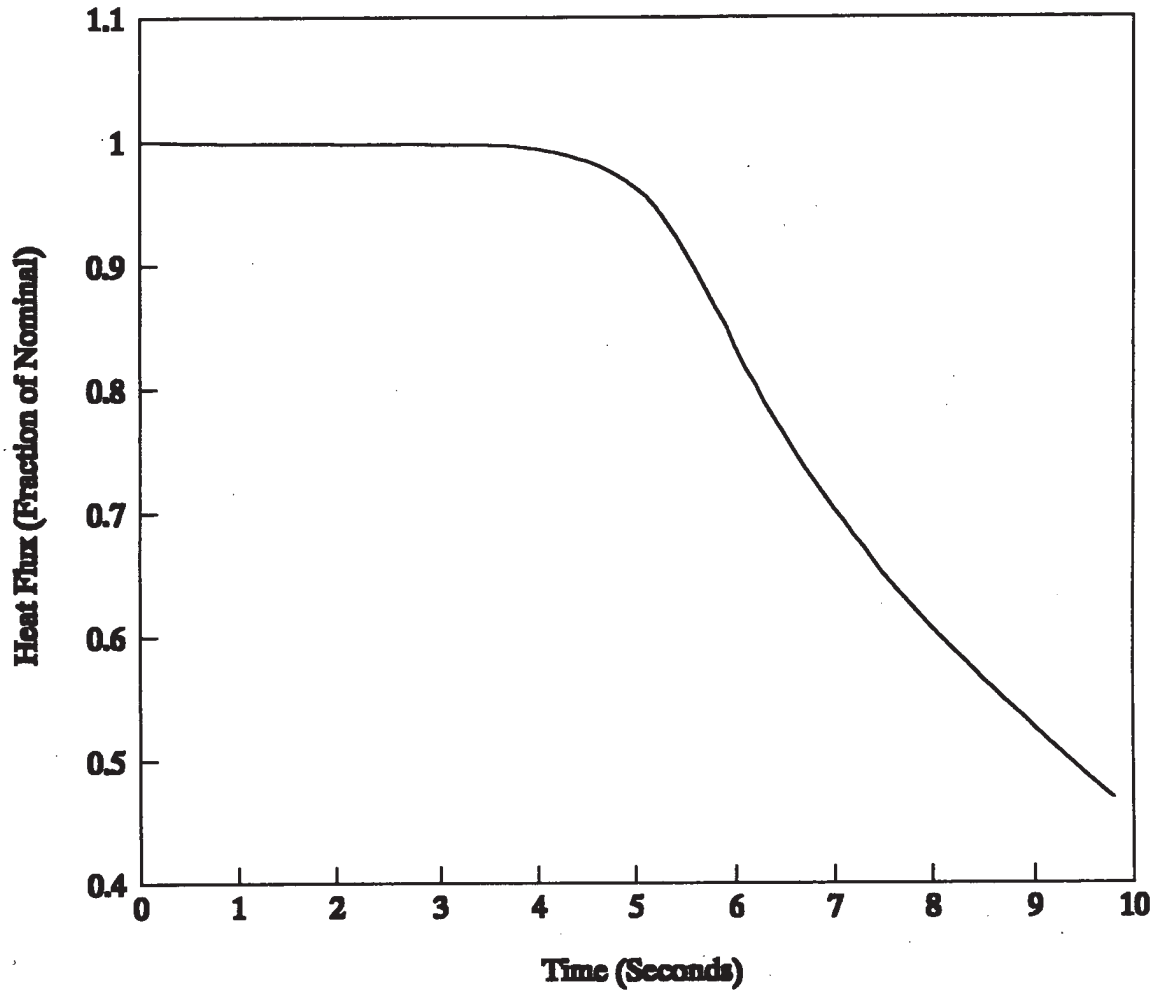


Figure 5.1.11-6
DNBR for Partial Loss of Flow
Four Loops in Operation,
One Pump Coasting Down

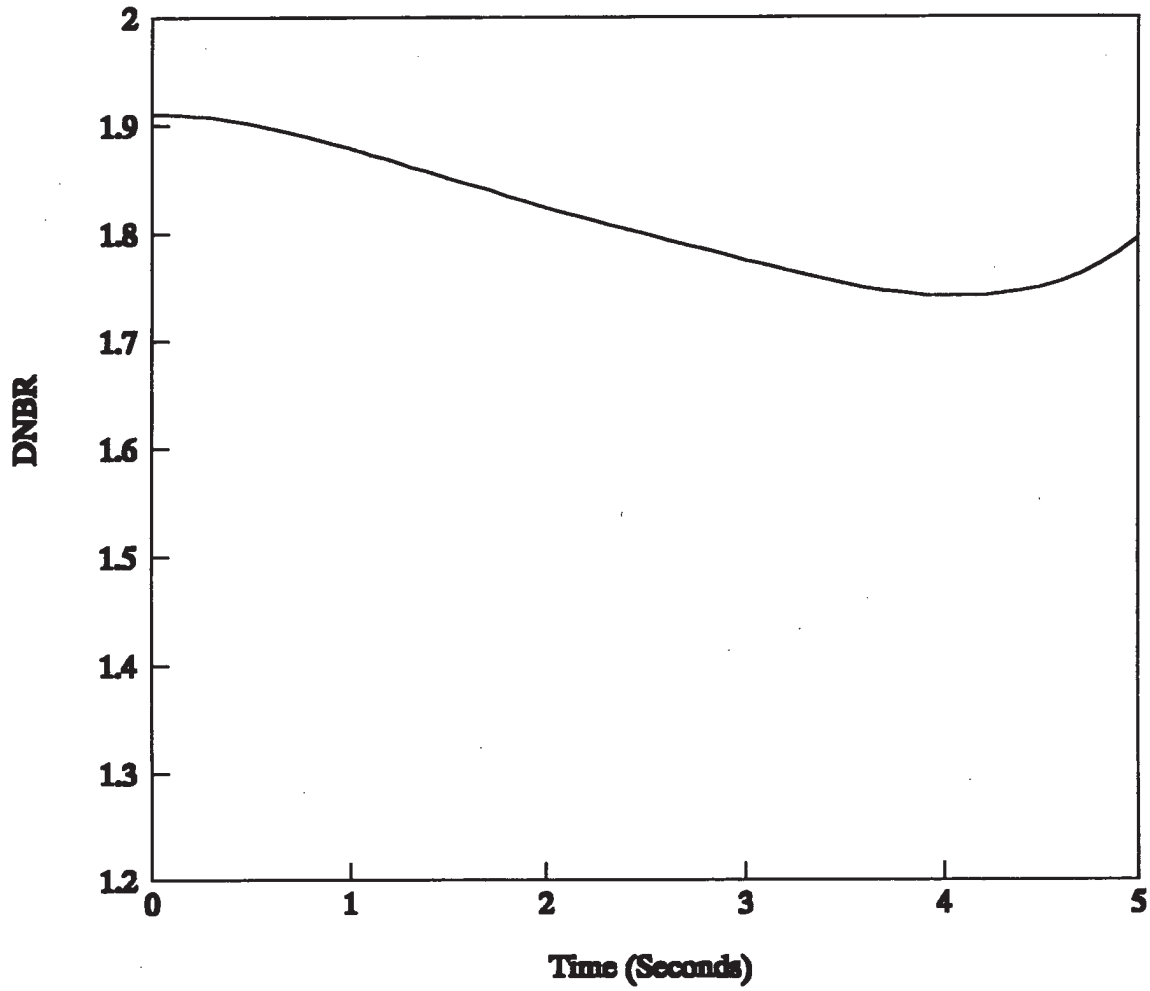


Figure 5.1.11-7
Nuclear Power for Complete Loss of Flow
Four Loops in Operation,
Four Pumps Coasting Down

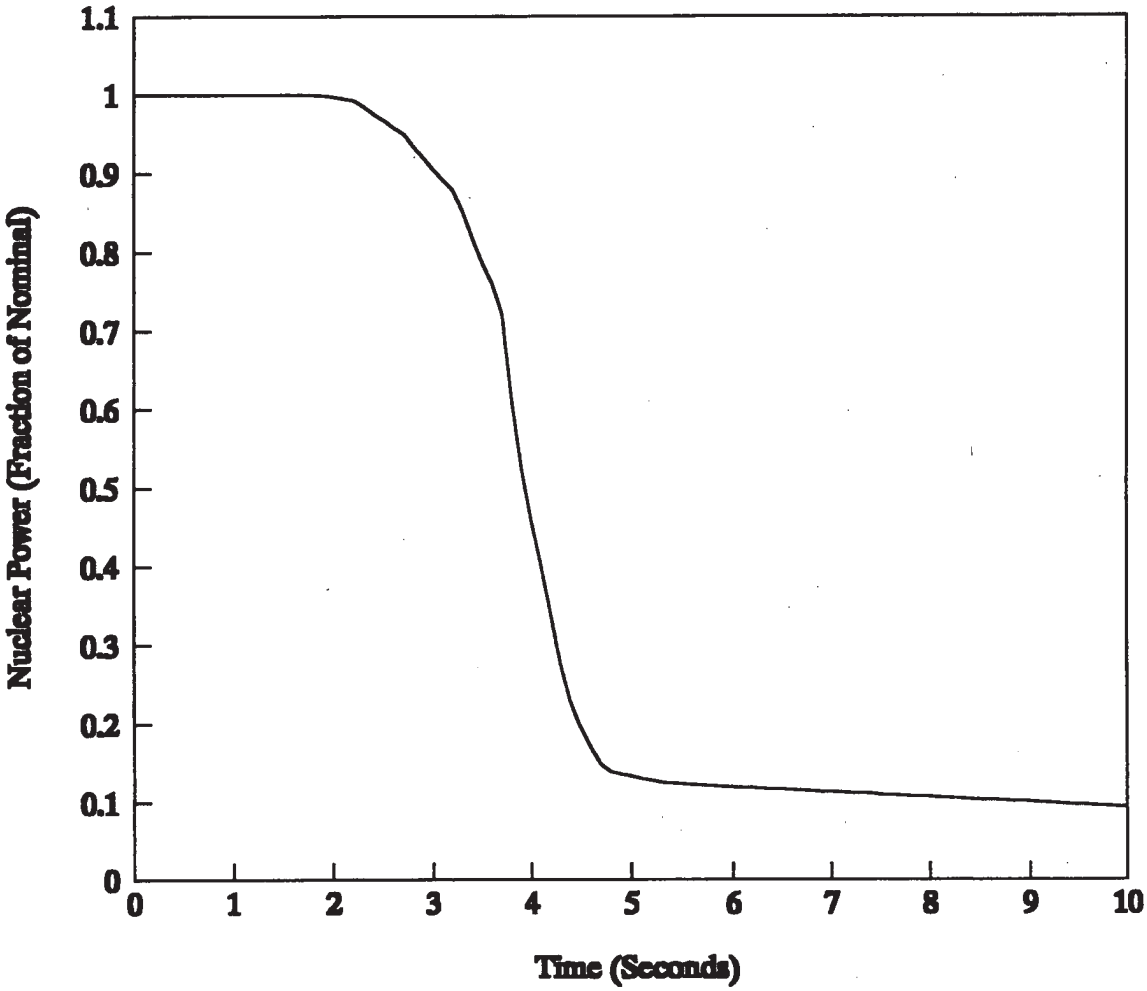


Figure 5.1.11-8
Pressurizer Pressure for Complete Loss of Flow
Four Loops in Operation,
Four Pumps Coasting Down

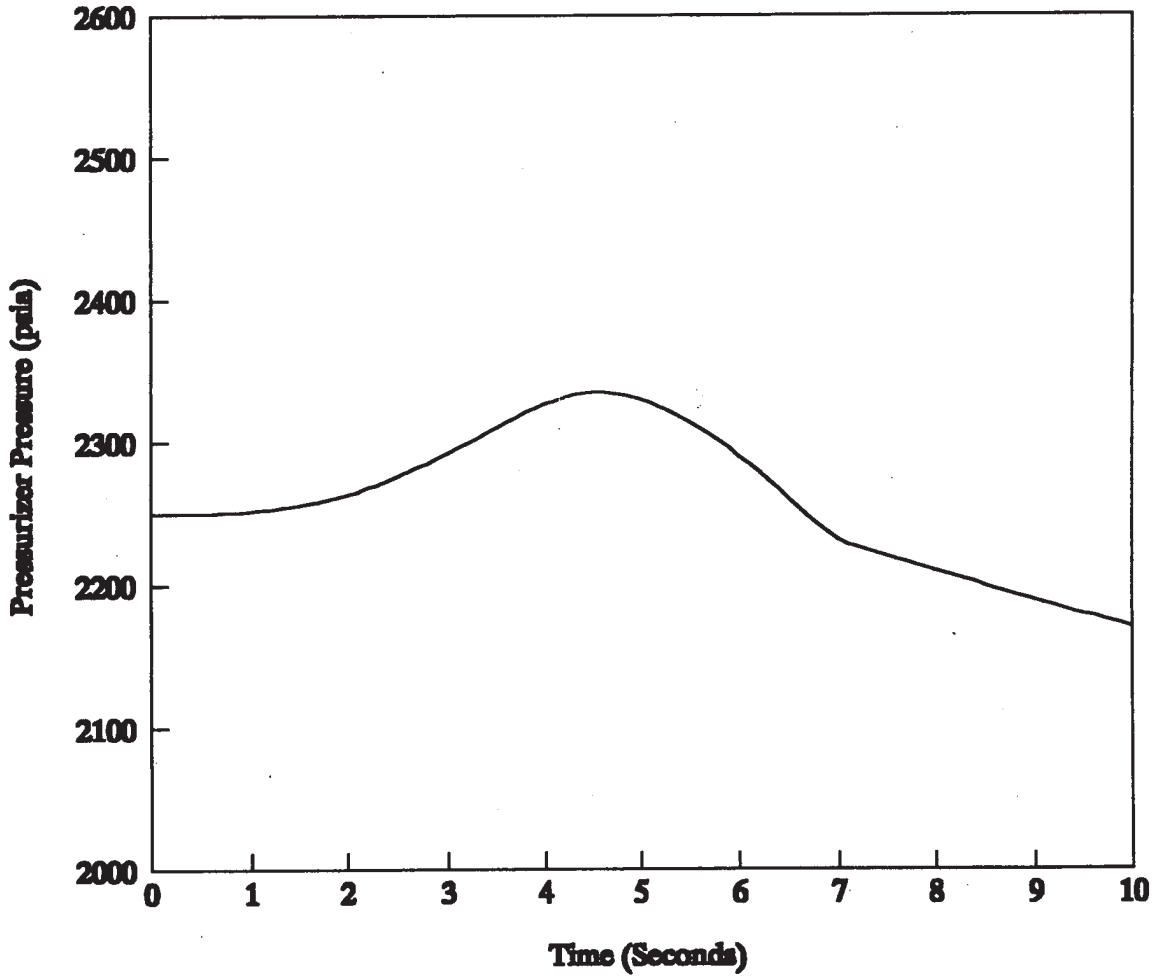


Figure 5.1.11-9
Total RCS Flow for Complete Loss of Flow
Four Loops in Operation,
Four Pumps Coasting Down

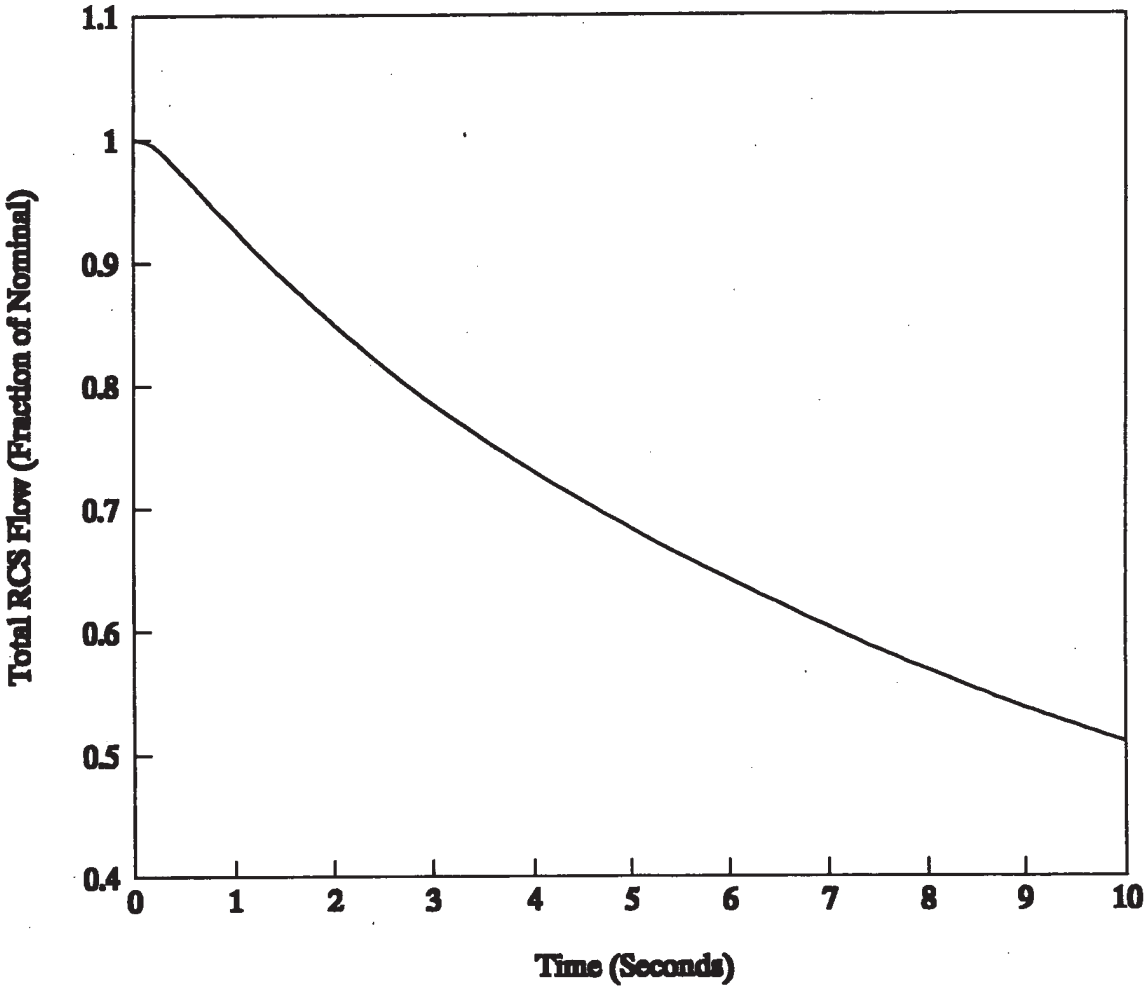


Figure 5.1.11-10
Heat Flux for Complete Loss of Flow
Four Loops in Operation,
Four Pumps Coasting Down

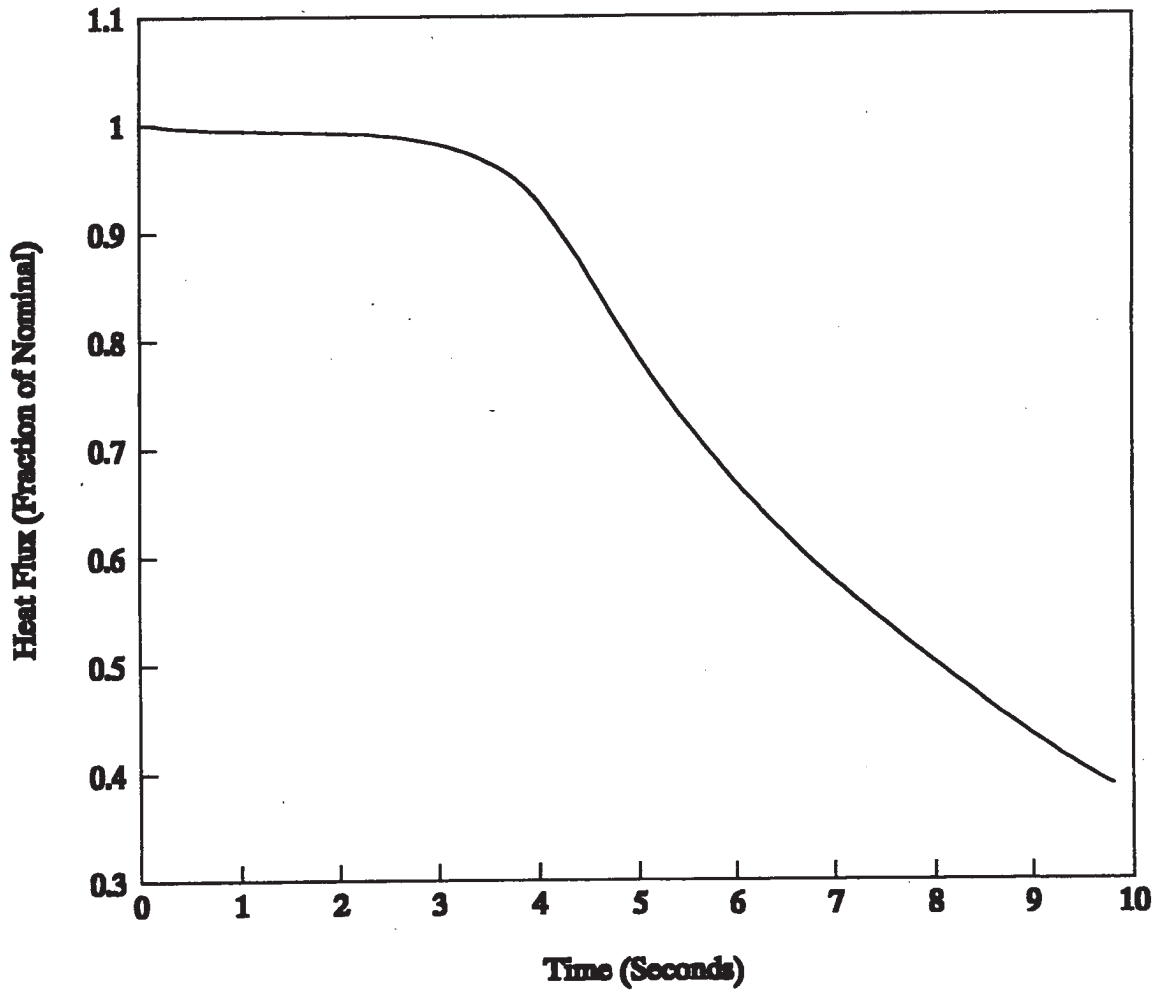


Figure 5.1.11-11
DNBR for Complete Loss of Flow
Four Loops in Operation,
Four Pumps Coasting Down

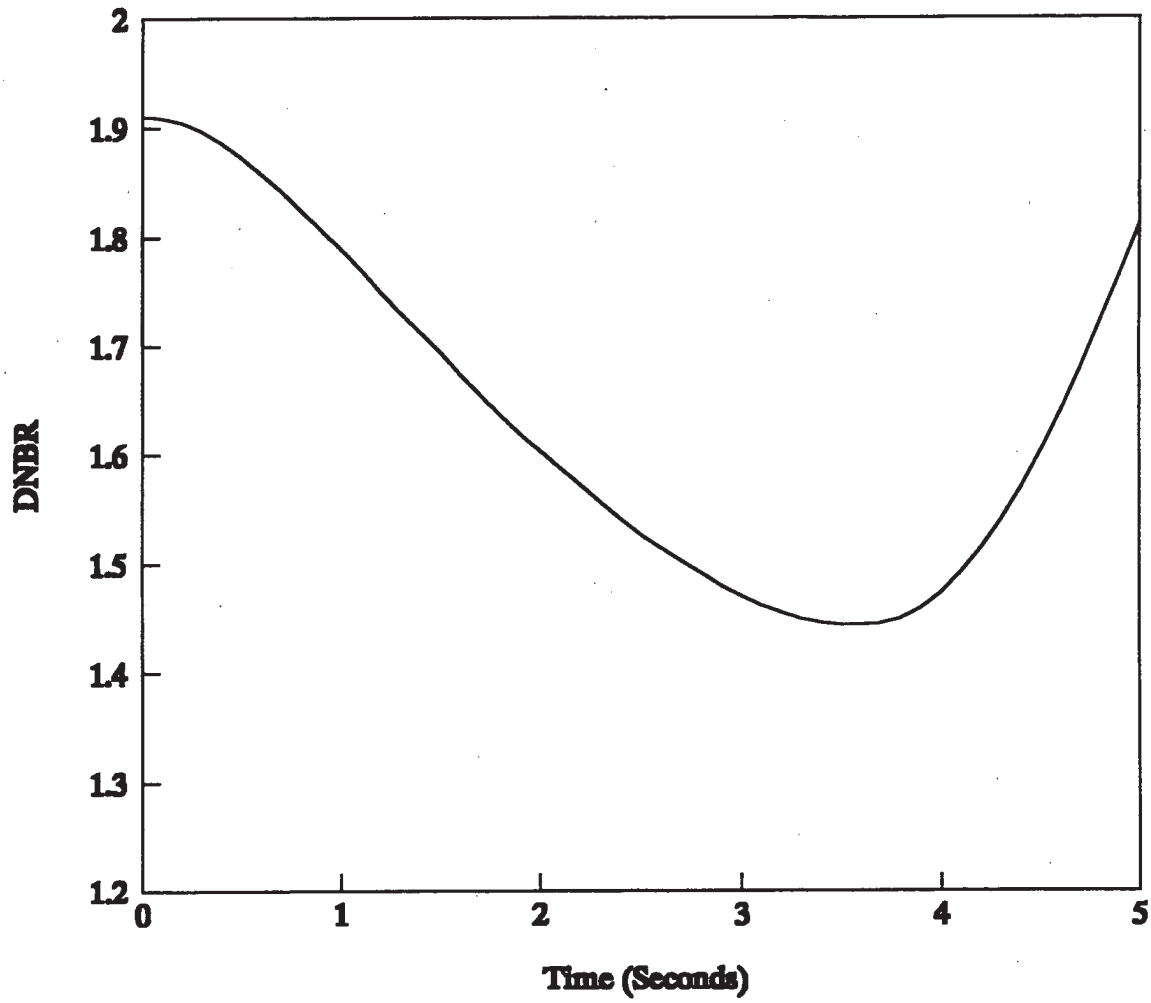


Figure 5.1.11-12
Nuclear Power for RCP Shaft Seizure
Four Loops in Operation

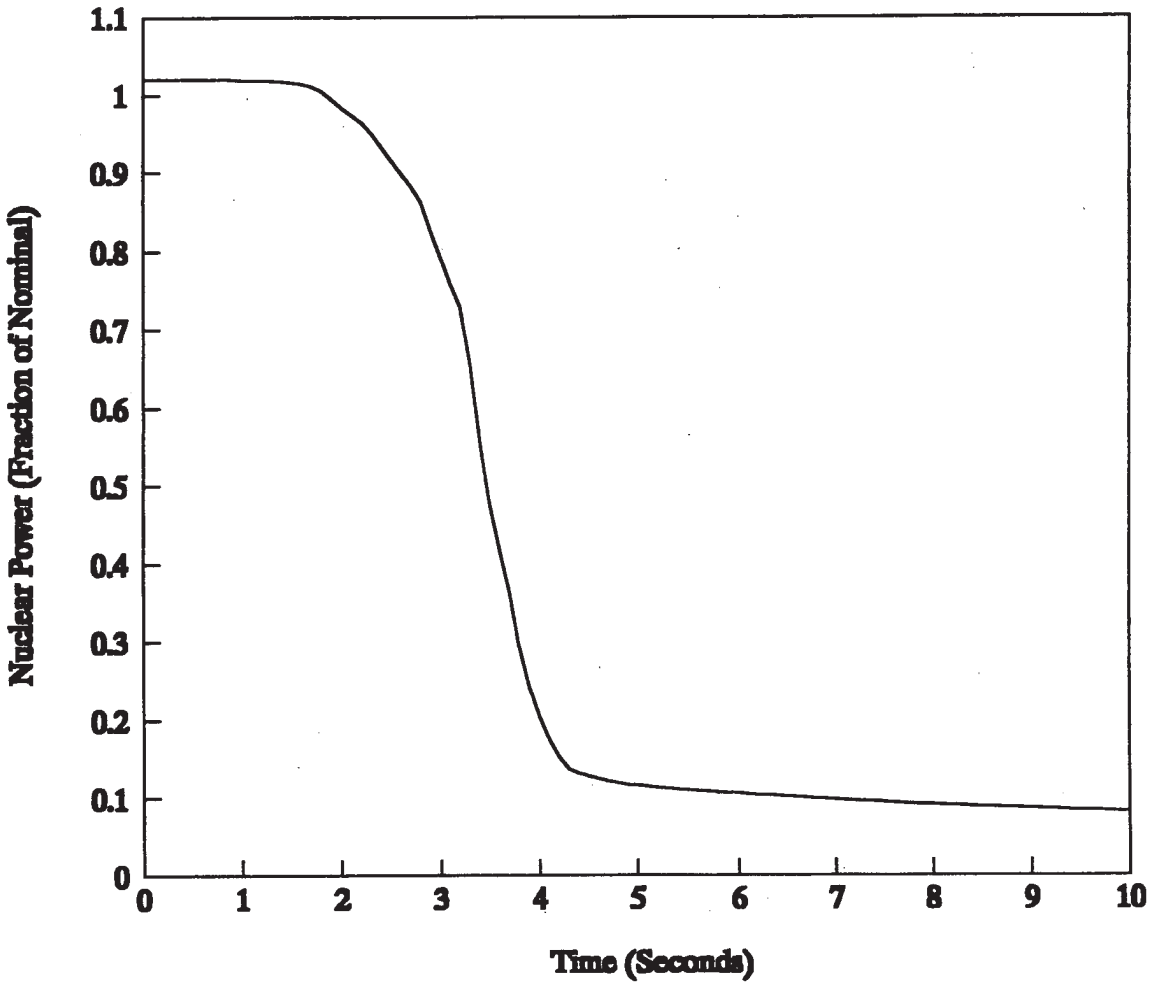


Figure 5.1.11-13
RCS Pressure for RCP Shaft Seizure
Four Loops in Operation

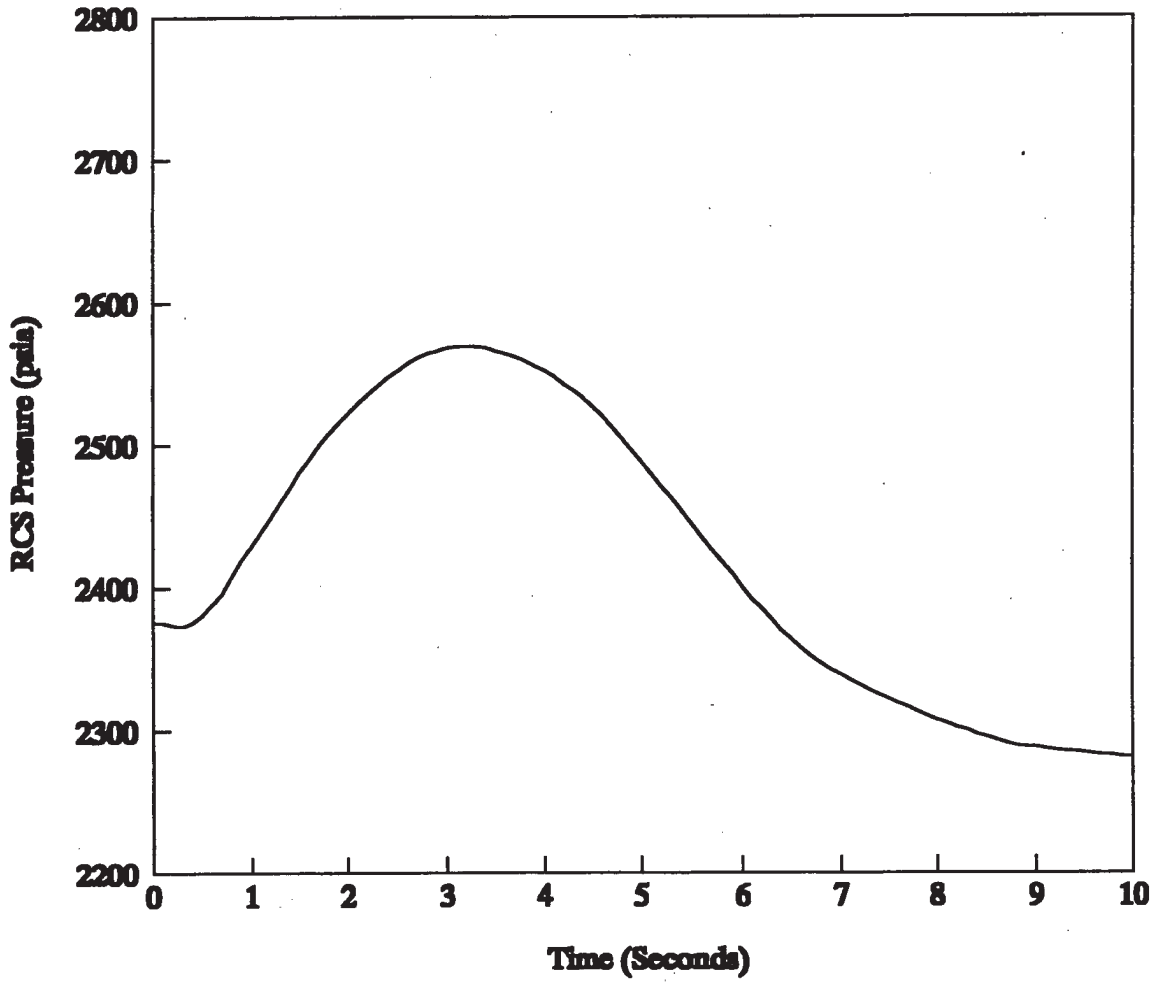


Figure 5.1.11-14
Total RCS Flow for RCP Shaft Seizure
Four Loops in Operation

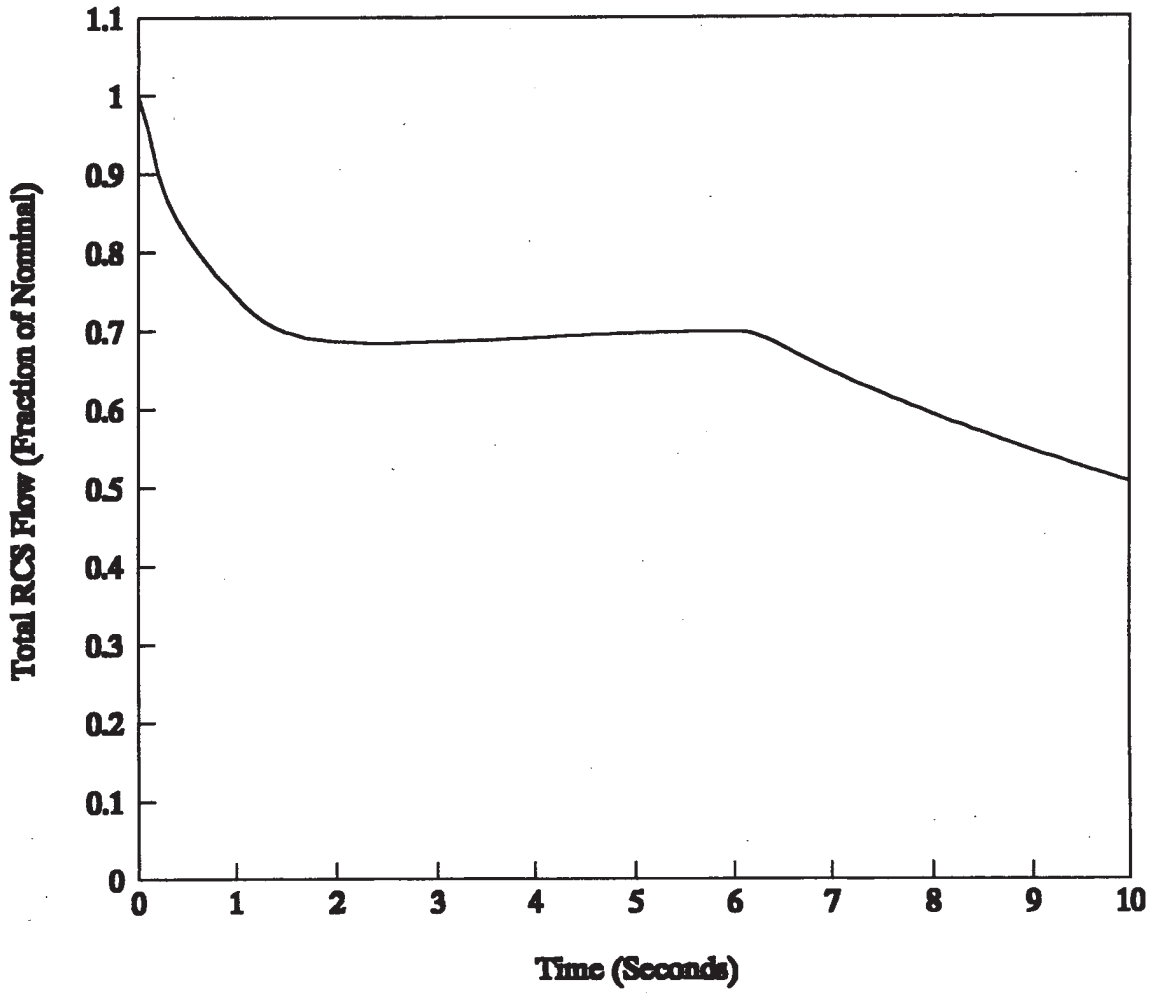


Figure 5.1.11-15
RCS Loop Flow for RCP Shaft Seizure
Four Loops in Operation

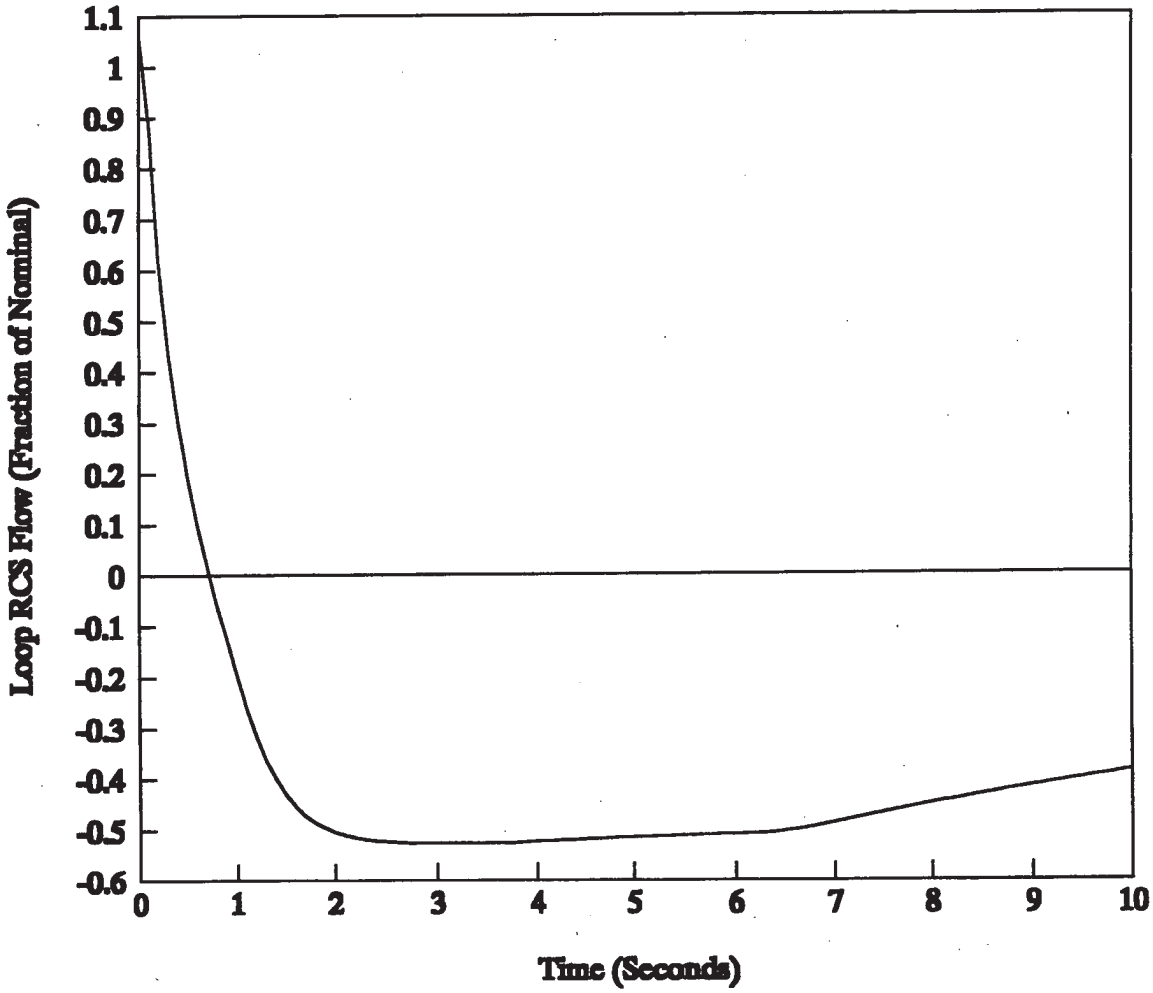
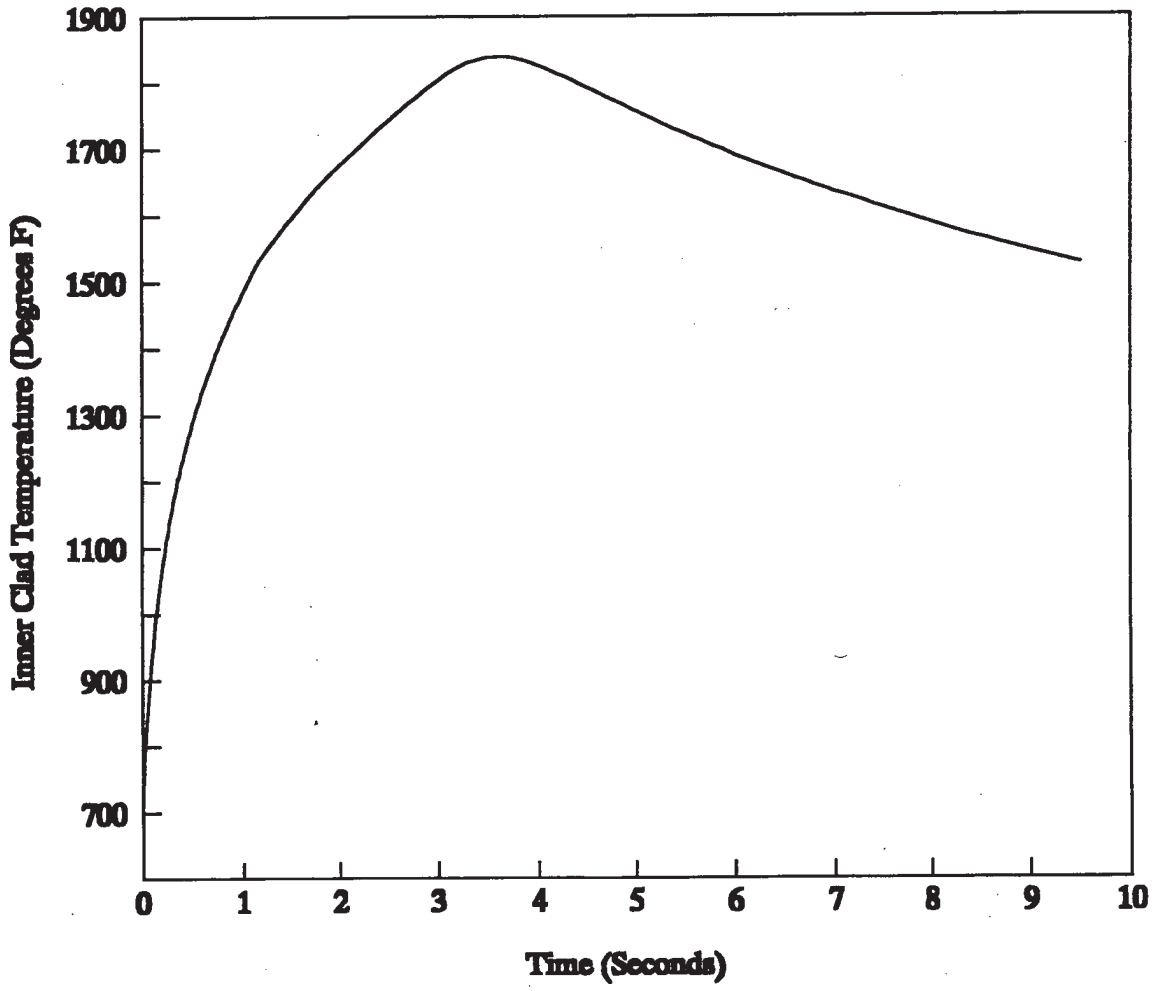


Figure 5.1.11-16
Inner Clad Temperature for RCP Shaft Seizure
Four Loops in Operation



5.1.12 Startup of an Inactive Reactor Coolant Loop

Introduction:

The evaluation herein was performed for the Startup of an Inactive Reactor Coolant Loop event as described in the FSAR Section 14.1.7 to support the insertion of VANTAGE + Fuel with the design features described in Section 5.1.2. The evaluation also addresses changes in the safety analysis assumptions associated with the VANTAGE + transition as described in Section 5.1.3.

As described in the FSAR Section 14.1.7, operation of the plant with an inactive loop causes reversed flow through the inactive loop. This occurs since there are no isolation valves or check valves in the reactor coolant loops. If the reactor was operated at power in this condition, the coolant temperature in the inactive loop would be lower in comparison with the active loops. Subsequent restart of the idle reactor coolant pump, without bringing the loop temperature closer to the average temperature, would result in the injection of cold water into the core. This cooler water causes a rapid reactivity increase.

Evaluation:

The current analysis for this event is the original FSAR licensing basis analysis for Indian Point Unit 3 and is that described in the FSAR. It was originally included in the FSAR licensing basis when operation with a loop out of service was considered.

As described in the FSAR, the analysis conservatively assumes conditions representative of this event with 3 loops operating and assumes a power level equal to 77% of hot full power.

However, the Technical Specifications require that all 4 reactor coolant pumps be operating for reactor power levels above 10% of full power which precludes operation at the initial conditions assumed in the original licensing basis analysis.

For operation at the allowable 10% power level, the difference in coolant temperatures between the active loops and the inactive loop are small since the difference between the average coolant temperature at 10% power and hot zero power is less than 3 °F. This difference (which is less that the uncertainty on T_{avg} at hot full power conditions) will not result in a reactivity increase of the magnitude covered by the existing analysis of this event at conditions representative of 77% of full power.

Conclusions:

Based on the evaluation contained herein, it is concluded that existing FSAR analysis for this event is sufficiently conservative to bound allowable operation with VANTAGE + fuel. As described in the Evaluation section, this determination is based on the Technical Specification limitations placed on the operation of the reactor coolant pumps and the magnitude of conservatism between allowable operation and that assumed in the existing FSAR analysis.

This conclusion is valid for the transition to VANTAGE + fuel, including the associated design features and related changes in safety analysis assumptions described in Sections 5.1.2 and 5.1.3, respectively.

Moreover, since this event was originally included in the FSAR licensing basis when operation with a loop out of service was considered and since the current Indian Point Unit 3 Technical Specifications do not address Three Loop Operation, this event should be removed from current FSAR licensing basis for Indian Point Unit 3.

5.1.13 Loss of External Electrival Load

Introduction:

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

For a turbine trip, the reactor would be tripped directly (unless below the 50% power Permissive 8 setpoint) from a signal derived from the turbine autostop oil pressure and turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, steam generator level would be maintained by the auxiliary feedwater system.

The unit was designed to accept a 59% step loss of load without actuating a reactor trip. The automatic steam dump system, with 40% steam dump capacity to the condenser, was designed to accommodate this load rejection by reducing the severity of the transient imposed upon the RCS. The reactor power is reduced to the new equilibrium power level at a rate consistent with capability of the Rod Control System. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves do not lift for the 50% step loss of load with steam dump.

In the event the steam dump valves fail to open following a large loss of load or in the event of a complete loss of load with steam dump operating, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, the OTΔT signal, the OPΔT signal, or the low low steam generator water level signal. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. However, the pressurizer safety valves and steam generator safety valves are sized to protect the RCS and steam generator against overpressurization for all load losses without assuming the operation of the steam dump system. The steam dump valves will not be opened for load reductions of 10% or less. For larger load reductions they may open. The RCS and main steam supply relieving capacities were designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control and/or steam bypass control systems.

The Loss of Load/Turbine Trip event is classified as an ANS Condition II fault as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious fault, i.e, a Condition III or IV category event.

The applicable safety analysis licensing basis acceptance criteria for Condition II events are:

a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (i.e., 2748.5 psia and 1208.5 psia, respectively),

b) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit, and

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

Method of Analysis and Assumptions:

The loss of load accident is analyzed for the following reasons: 1) to confirm that the pressurizer and steam generator safety valves are adequately sized to prevent overpressurization of the RCS and steam generators, respectively; 2) to form the basis of the required ASME overpressure protection report; and 3) to ensure that the increase in RCS temperature does not result in DNB in the core. The Reactor Protection System is designed to automatically terminate any such transient before the DNBR falls below the limit value.

The total loss of load transients are analyzed with the LOFTRAN computer program (see Section 5.1.5). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator relief and safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. The turbine is assumed to trip without actuating all the turbine stop valve limit switches. This assumption delays reactor trip until conditions in the RCS result in a trip on some other signal. Thus, the analysis assumes a worst case transient and demonstrates the adequacy of the pressure relieving devices and core protection margins.

Major assumptions are summarized below:

1) **Initial Operating Conditions:** The initial reactor power, RCS pressure, and RCS temperatures are assumed at their nominal values consistent with steady state full power operation and the Revised Thermal Design Procedure (RTDP) methodology.

2) **Moderator and Doppler Coefficients of Reactivity:** The turbine trip is analyzed with both maximum and minimum reactivity feedback. The maximum feedback (EOL) cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback (BOL) cases assume a minimum absolute value of the moderator temperature coefficient and the least negative Doppler coefficient.

3) **Reactor Control:** From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

4) **Steam Release:** No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits the secondary steam pressure at the setpoint value. Through maximizing the pressure transient in the main steam system, the saturation temperature in the steam generators is maximized resulting in limiting pressure and temperature conditions in the RCS.

5) **Pressurizer Spray and Power-operated Relief Valves:** Two cases for both BOL and EOL reactivity feedback conditions are analyzed:

   a) Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

   b) No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

6) **Feedwater Flow:** Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur. However, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

7) **Offsite AC Power:** Loss of offsite power is not postulated to occur coincident with the loss of load incident since the resulting DNBR and pressure transients are limiting for the instance when offsite power is available.

Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip. The OTΔT reactor trip and high pressurizer pressure reactor trip are actuated in the analysis.

Results:

The transient responses for a total loss of load from full power operation are shown on Figures 5.1.13-1 through 5.1.13-28 for four cases; two cases for the BOL and two cases for the EOL reactivity feedback conditions.

Figures 5.1.13-1 through 5.1.13-7 show the transient responses for the total loss of steam load at BOL (minimum feedback reactivity coefficients) assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. Following event initiation, the pressurizer pressure and average RCS temperature increase due to the rapidly reduced steam flow and heat removal capacity of the secondary-side. The peak pressurizer pressure and water volume and RCS average temperature are reached shortly after the reactor is tripped by the OTΔT trip function. The DNB ratio decreases initially and then rapidly increases following reactor trip. The minimum DNBR remains well above the safety analysis limit value of 1.54. The pressurizer relief and safety valves are actuated for this case and maintain primary system pressure below 110 percent of the design value. The steam generator safety relief valves open and limit the secondary-side steam pressure increase.

Figures 5.1.13-8 through 5.1.13-14 show the transient responses for the total loss of steam load at EOL conditions (maximum feedback reactivity coefficients). All other plant parameters are the same as in the above case. Nuclear power rises slightly prior to reactor trip by the OTΔT trip function. The DNBR increases throughout the transient and never drops below its initial value. The pressure transient is less severe for this case and does not require actuation of the pressurizer safety valves. The steam generator safety valves limit the peak secondary steam pressure.

The total loss of load event was also analyzed assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray or pressurizer power-operated relief valves. Figures 5.1.13-15 through 5.1.13-21 show the BOL transients without pressure control. The nuclear power remains relatively constant (prior to reactor trip) while pressurizer pressure, pressurizer water volume and RCS average temperature increase. The reactor is tripped on the high pressurizer pressure signal. The nuclear power remains essentially constant at full power until the reactor is tripped. The DNBR does not decrease below initial value throughout the transient. In this case the pressurizer safety valves are actuated and maintain the system pressure below 110 percent of the design value. The steam generator safety valves open and limit the secondary steam pressure.

Figures 5.1.13-22 through 5.1.13-28 show the transients at the EOL with the other assumptions being the same as on Figures 5.1.13-15 through 5.1.13-21. Again, a reactor trip signal is generated by the high pressurizer pressure trip function. The DNBR does not decrease below the initial value throughout the transient and the pressurizer and steam generator safety valves are actuated to limit their respective system pressures.

Table 5.1.13-1 summarizes the sequence of events and limiting conditions for the various transients

considered for the total loss of load cases presented above.

Conclusions:

The results of the analyses show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure-relieving devices incorporated in the plant's design are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system; i.e., the DNBR is maintained above the safety analysis limit value. Thus, no core safety limit will be violated. It is therefore concluded that the implementation of the VANTAGE + fuel design and other related design changes associated with this fuel transition reload are acceptable for this event.

Table 5.1.13-1

Sequence of Events and

Transient Results for the

Loss of External Electrical Load Event

Loss of External Electrical Load Event	With Pressurizer Control		Without Pressurizer Control	
	BOL	EOL	BOL	EOL
Loss of electrical load / turbine trip (sec)	0.0	0.0	0.0	0.0
Reactor trip signal	OTΔT	OTΔT	Hi Prz P	Hi Prz P
Reactor trip setpoint reached (sec)	9.2	9.8	6.7	6.7
Time of rod motion (sec)	11.2	11.8	8.7	8.7
Minimum DNBR	2.176	(a)	(a)	(a)
Minimum DNBR occurs (sec)	12.5	(a)	(a)	(a)
Peak pressurizer pressure (psia)	2523	2404	2551	2537
Peak pressurizer pressure occurs (sec)	13.0	13.5	10.5	9.5
Initiation of steam release from SG safety valves (sec)	15.0	15.5	15.0	15.5
Peak steam generator pressure (psia)	1194	1186	1188	1182
Peak steam generator pressure occurs (sec)	20.5	20.5	18.5	18.0

(a) DNBR does not decrease below its initial value.

Figure 5.1.13-1

Nuclear Power Transient for Loss of External Load,

Minimum Reactivity Feedback,

With Pressure Control

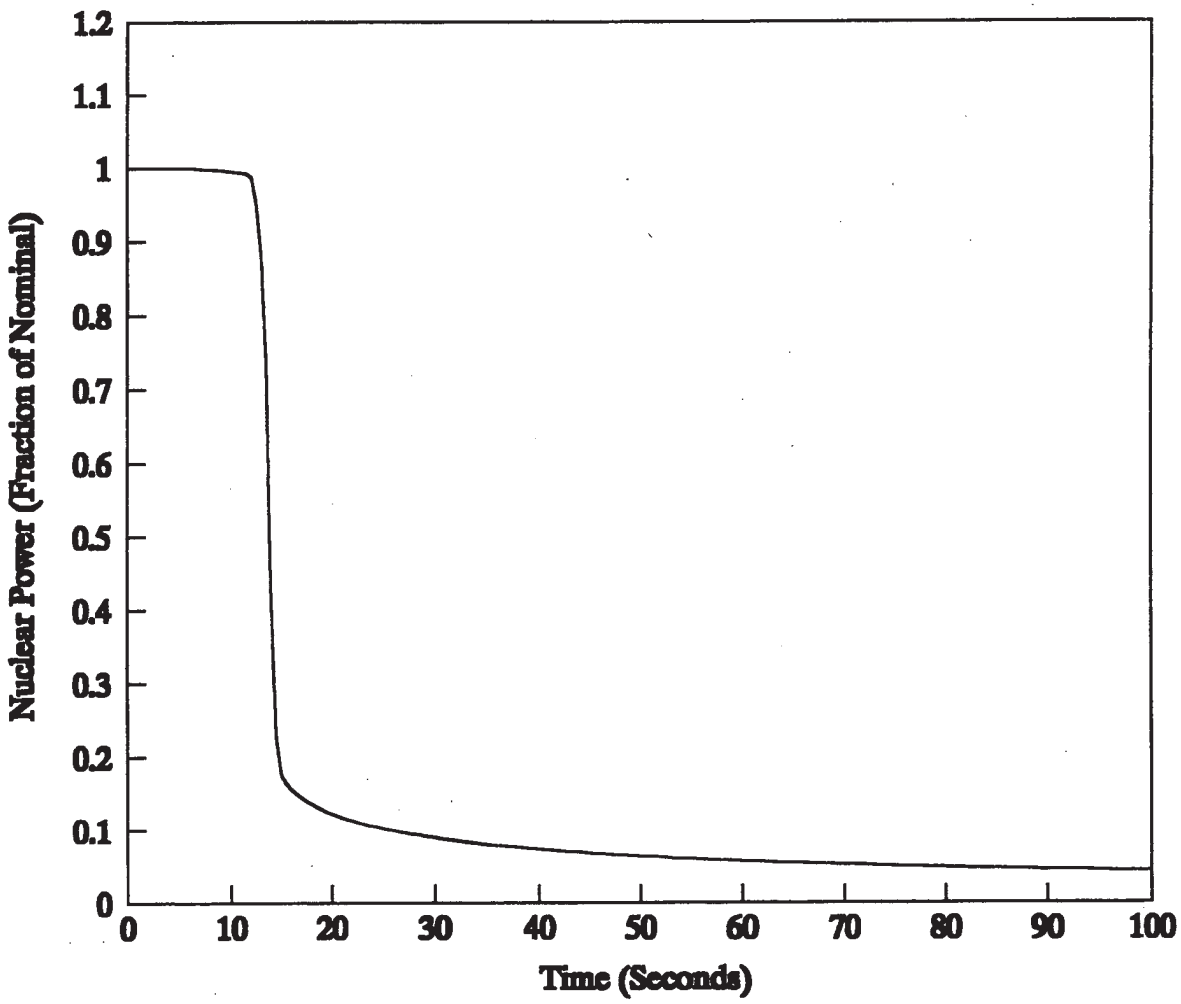


Figure 5.1.13-2

Core Heat Flux Transient for Loss of External Load,

Minimum Reactivity Feedback,

With Pressure Control

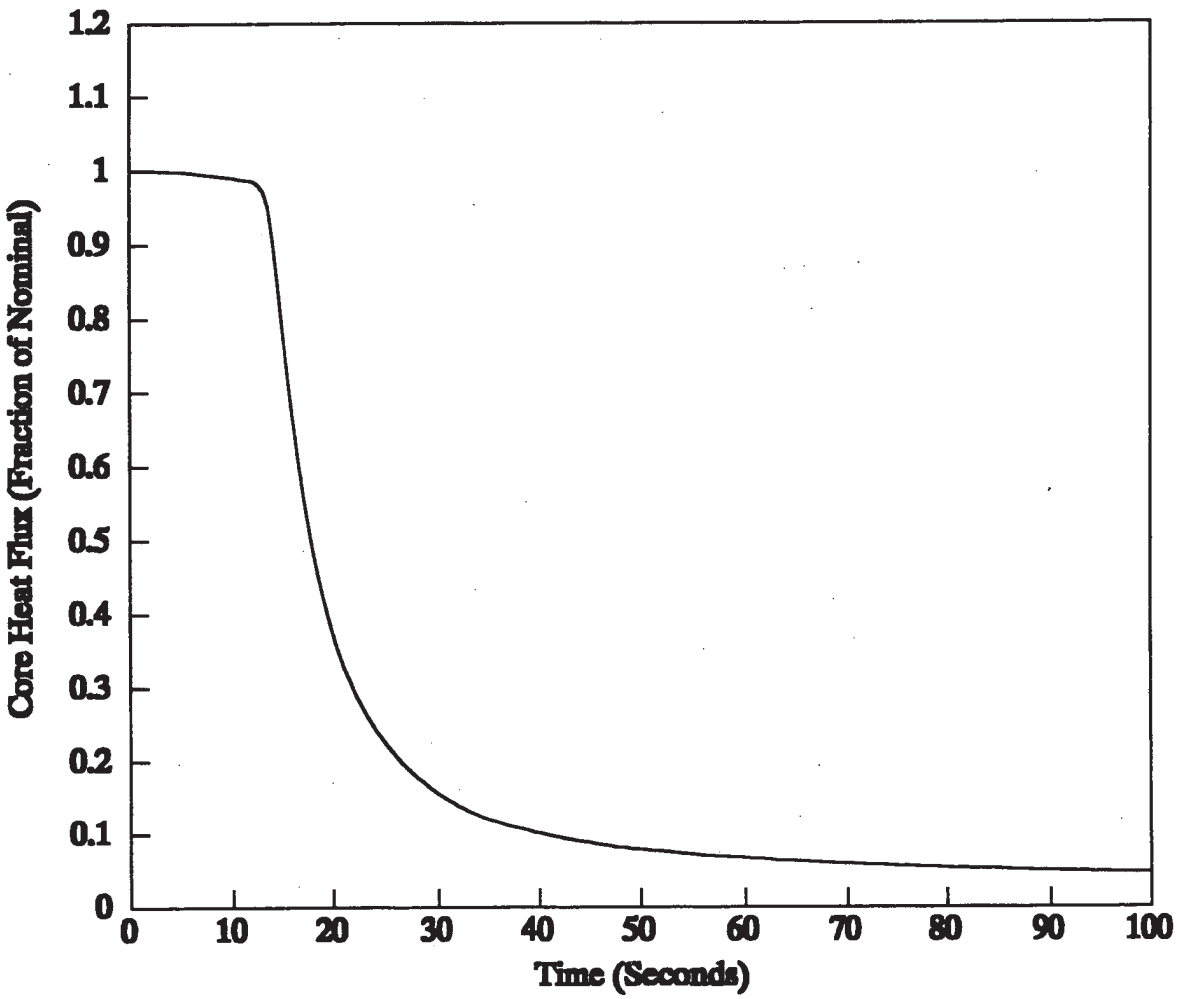


Figure 5.1.13-3

Pressurizer Pressure Transient for Loss of External Load,

Minimum Reactivity Feedback,

With Pressure Control

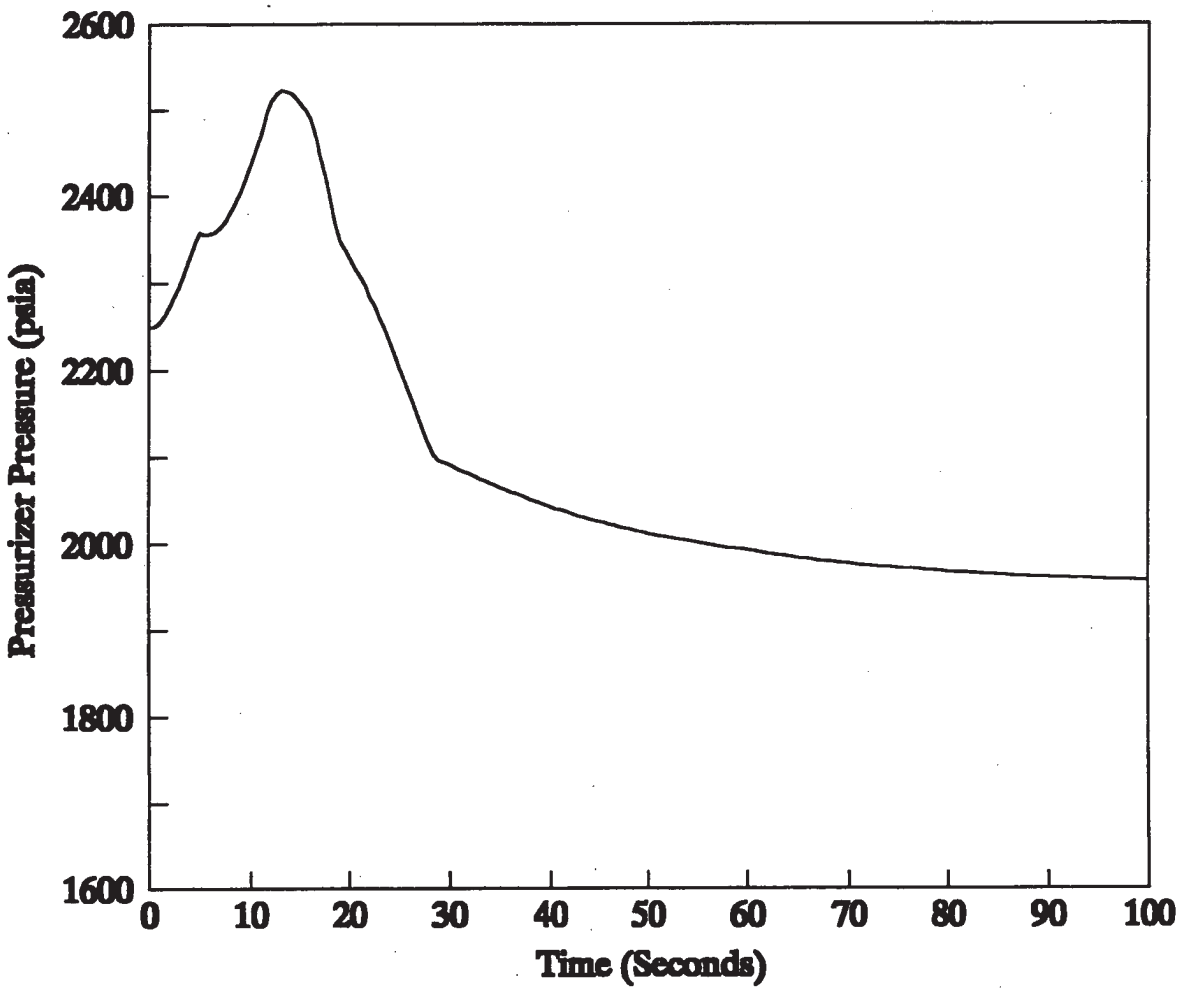


Figure 5.1.13-4

Pressurizer Water Volume Transient for Loss of External Load,

Minimum Reactivity Feedback,

With Pressure Control

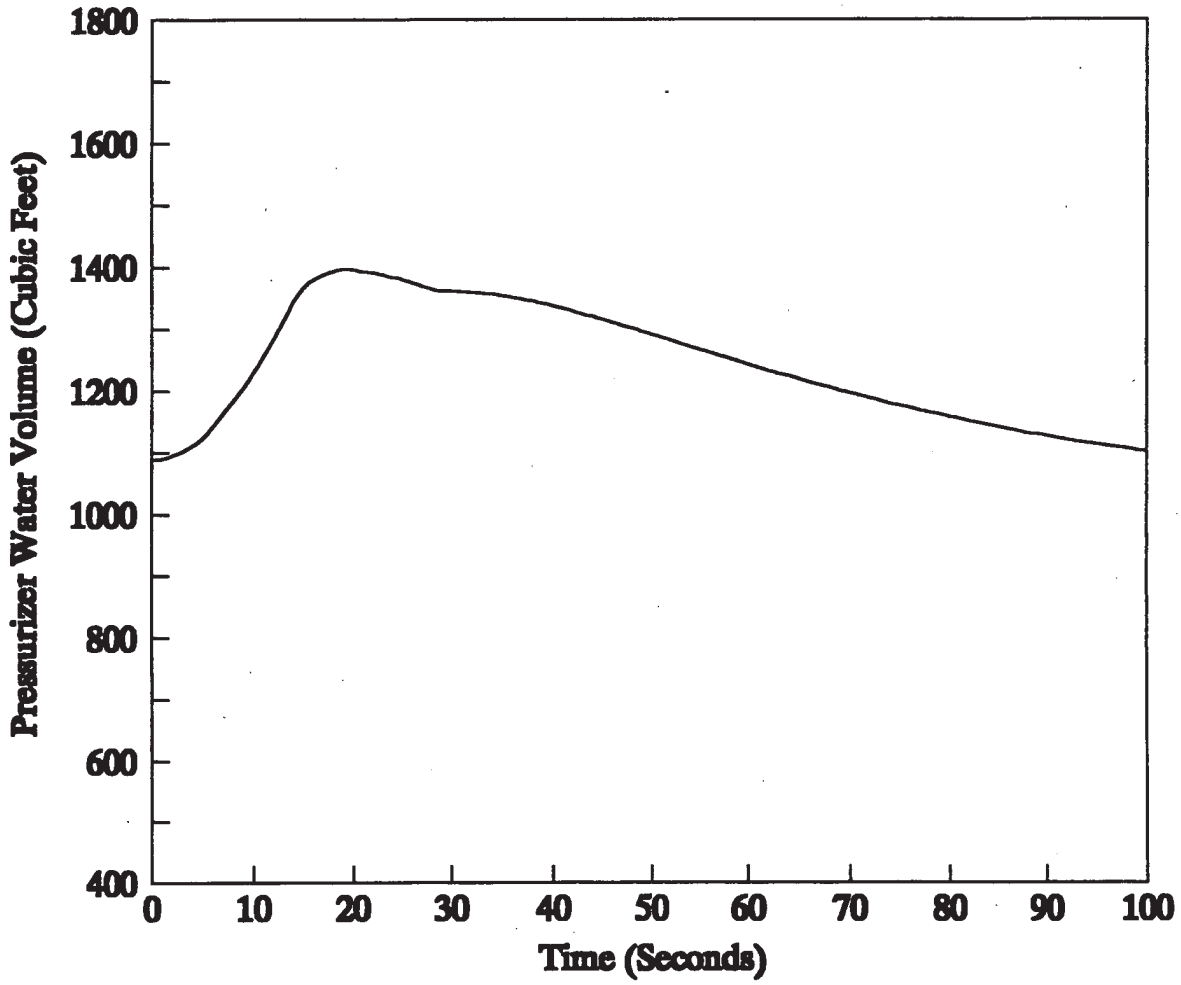


Figure 5.1.13-5

Vesses Average Water Temperature for Loss of External Load,

Minimum Reactivity Feedback,

With Pressure Control

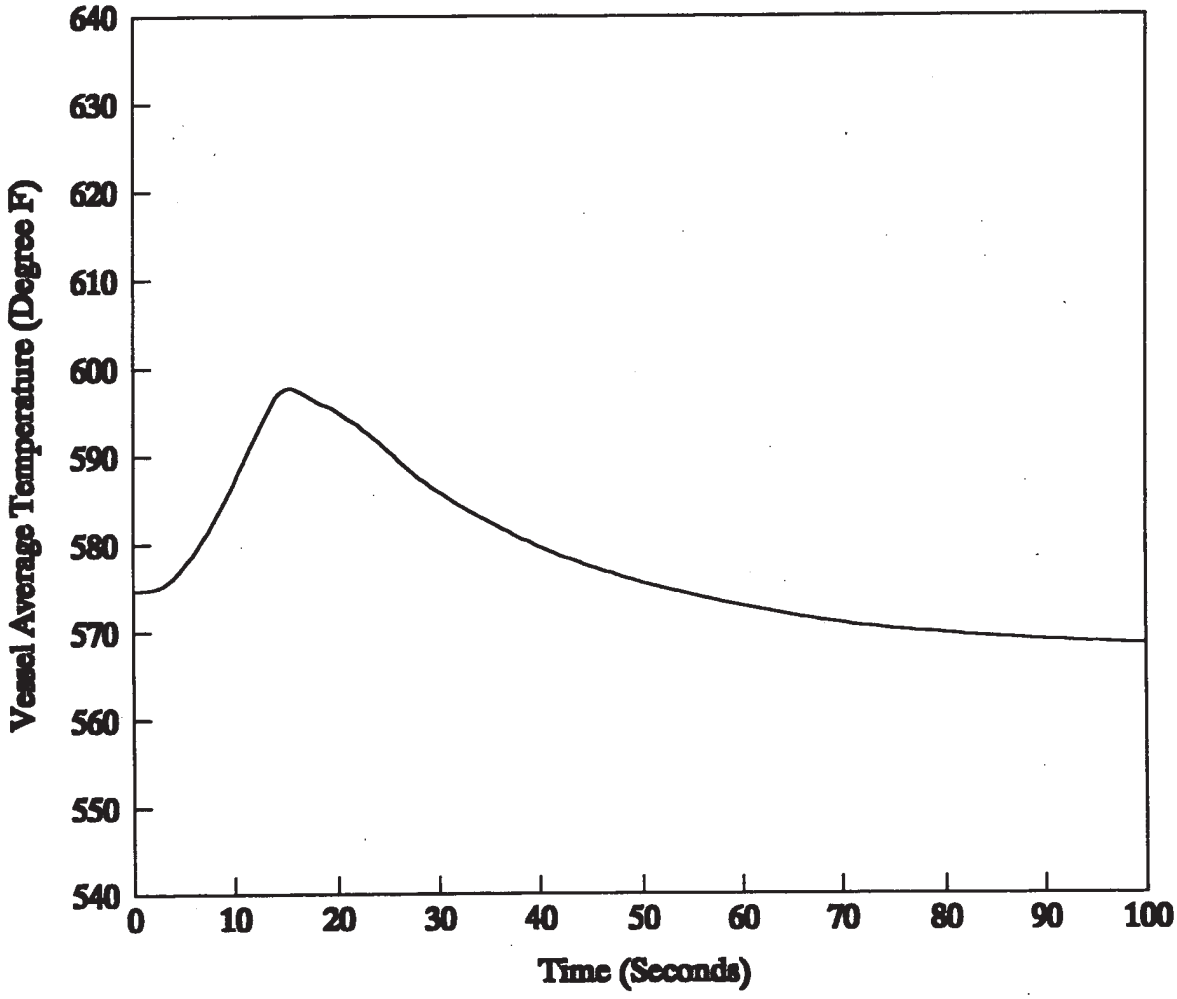


Figure 5.1.13-6

Steam Pressure Transient for Loss of External Load,

Minimum Reactivity Feedback,

With Pressure Control

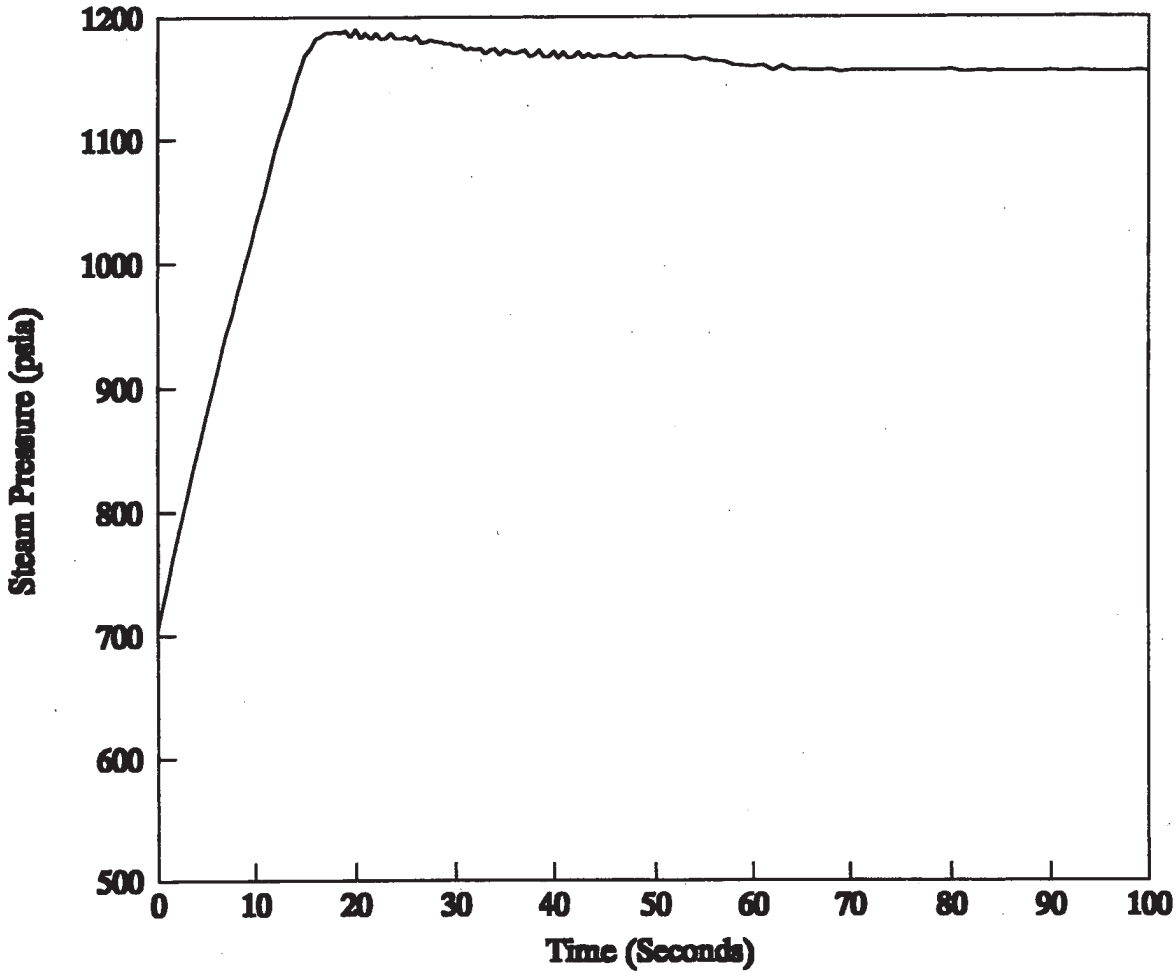


Figure 5.1.13-7

DNB Ratio vs Time for Loss of External Load,

Minimum Reactivity Feedback,

With Pressure Control

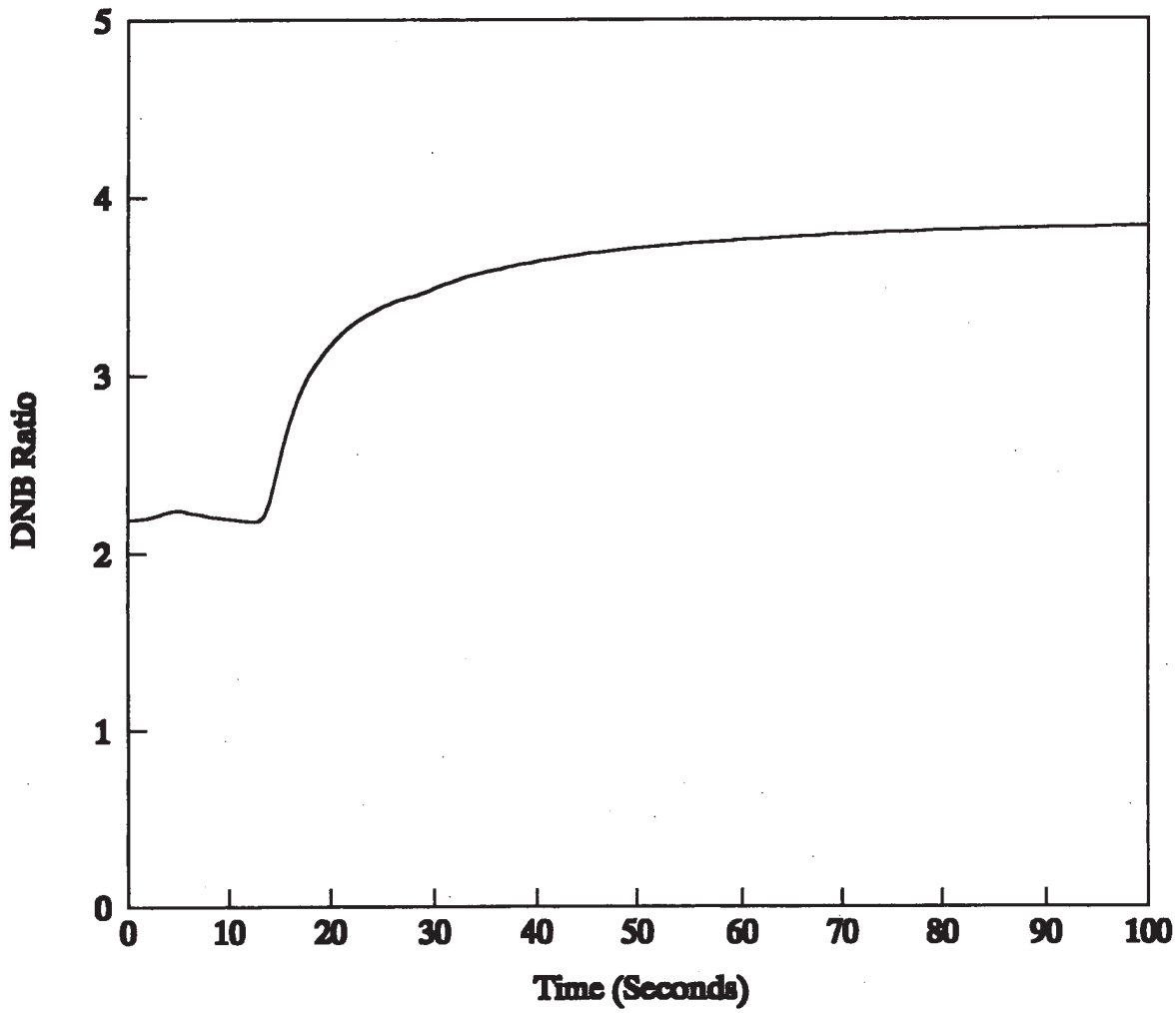


Figure 5.1.13-8

Nuclear Power Transient for Loss of External Load,

Maximum Reactivity Feedback,

With Pressure Control

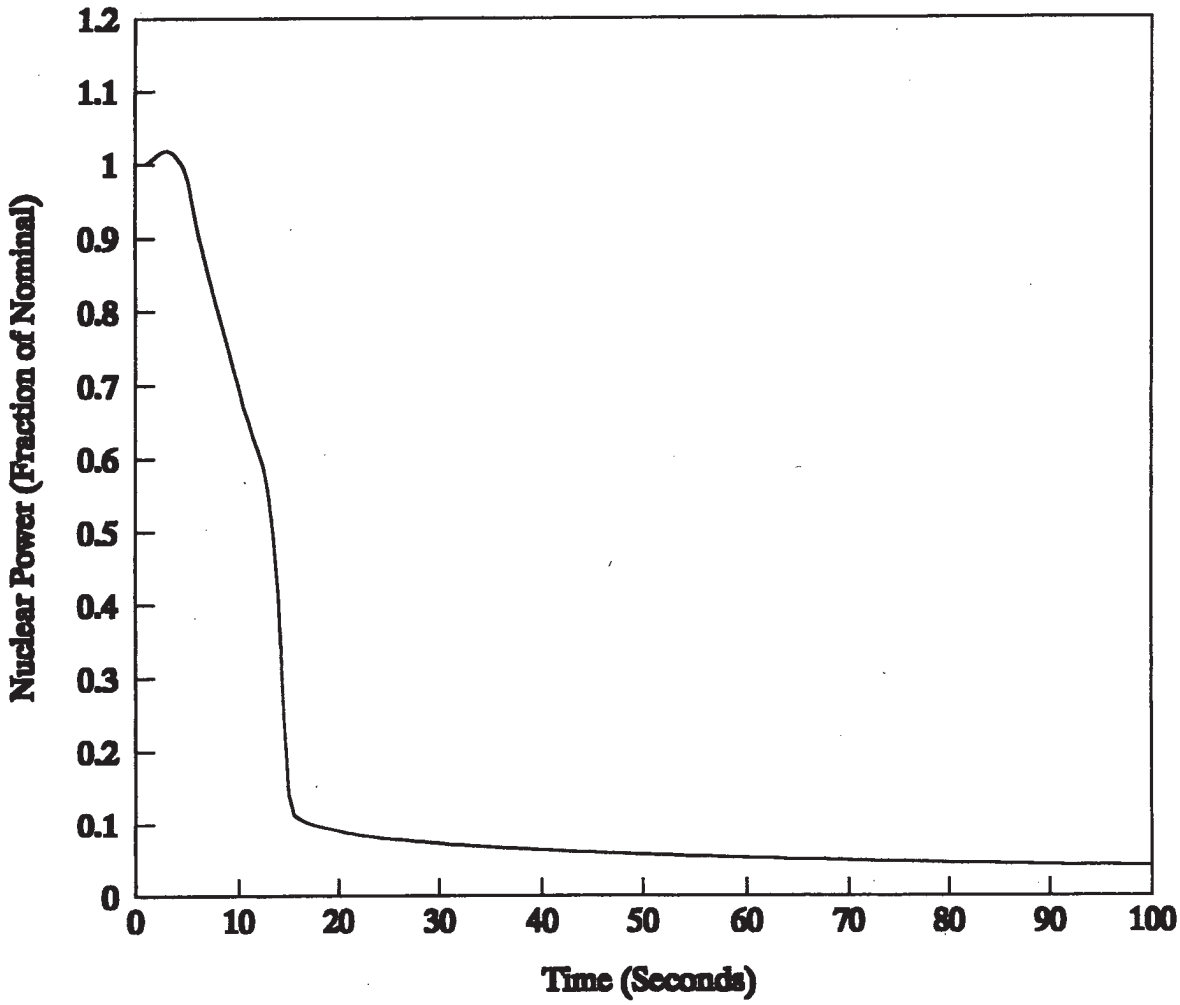


Figure 5.1.13-9

Core Heat Flux Transient for Loss of External Load,

Maximum Reactivity Feedback,

With Pressure Control

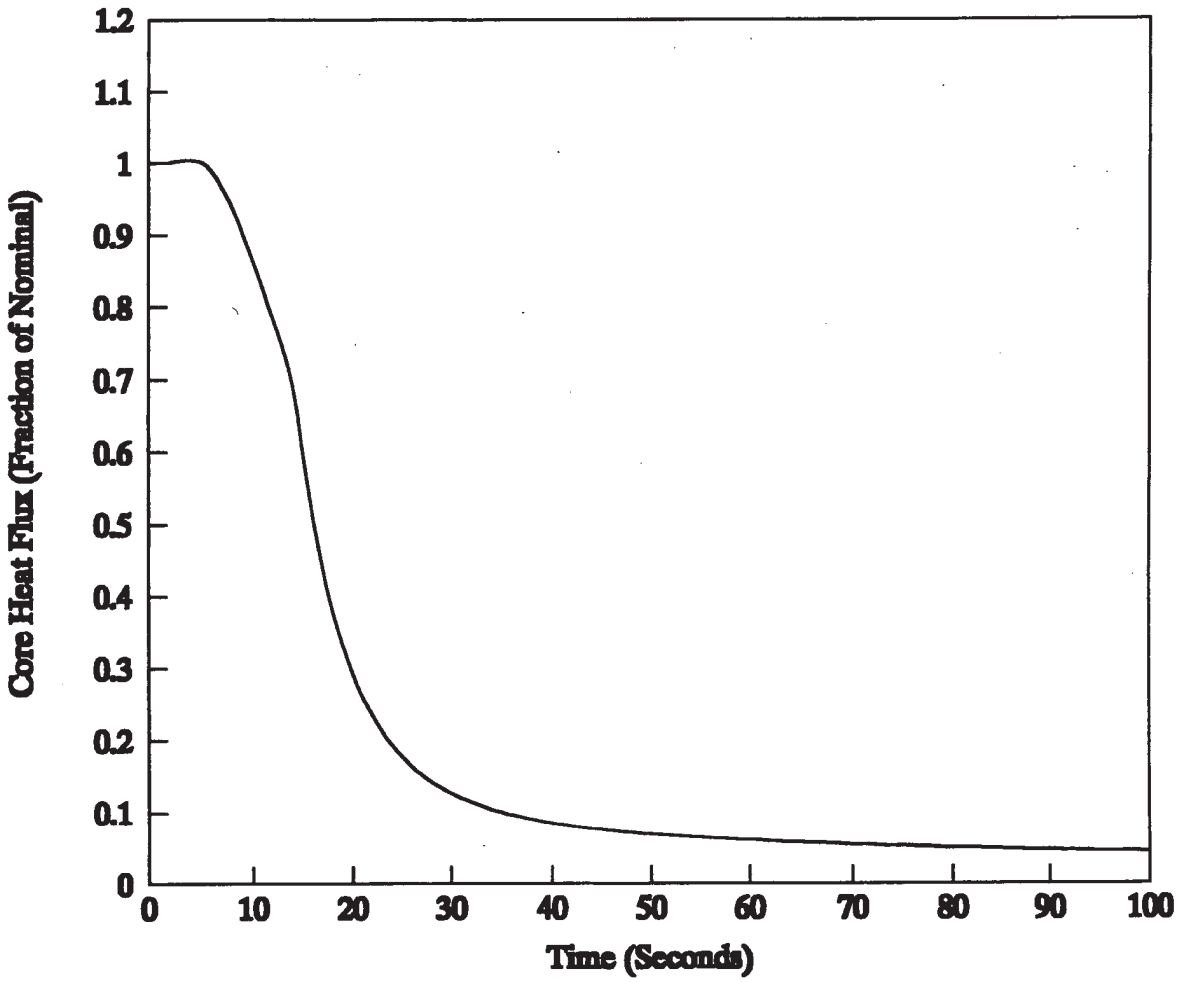


Figure 5.1.13-10

Pressurizer Pressure Transient for Loss of External Load,

Maximum Reactivity Feedback,

With Pressure Control

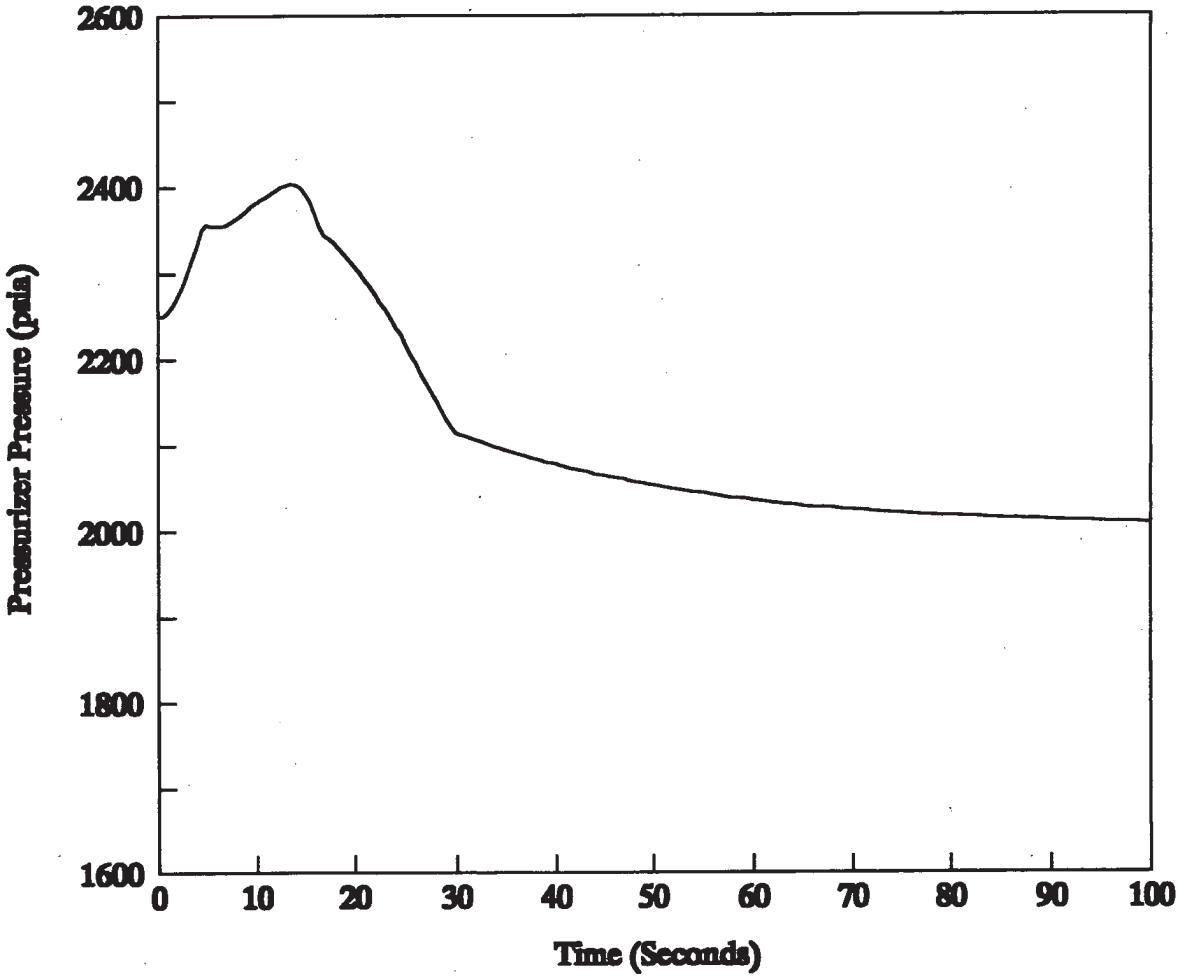


Figure 5.1.13-11

Pressurizer Water Volume Transient for Loss of External Load,

Maximum Reactivity Feedback,

With Pressure Control

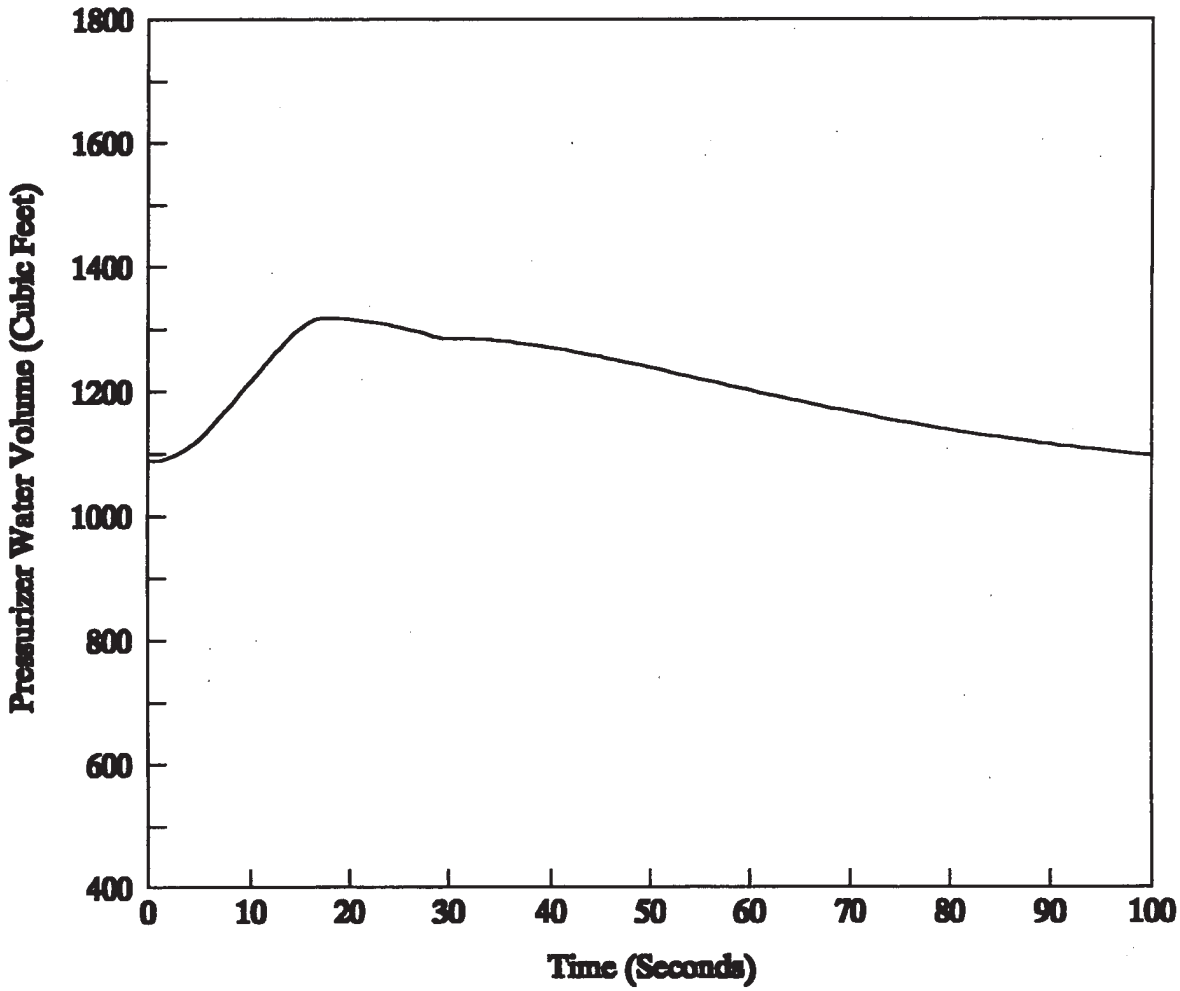


Figure 5.1.13-12

Vessel Average Water Temperature for Loss of External Load,

Maximum Reactivity Feedback,

With Pressure Control

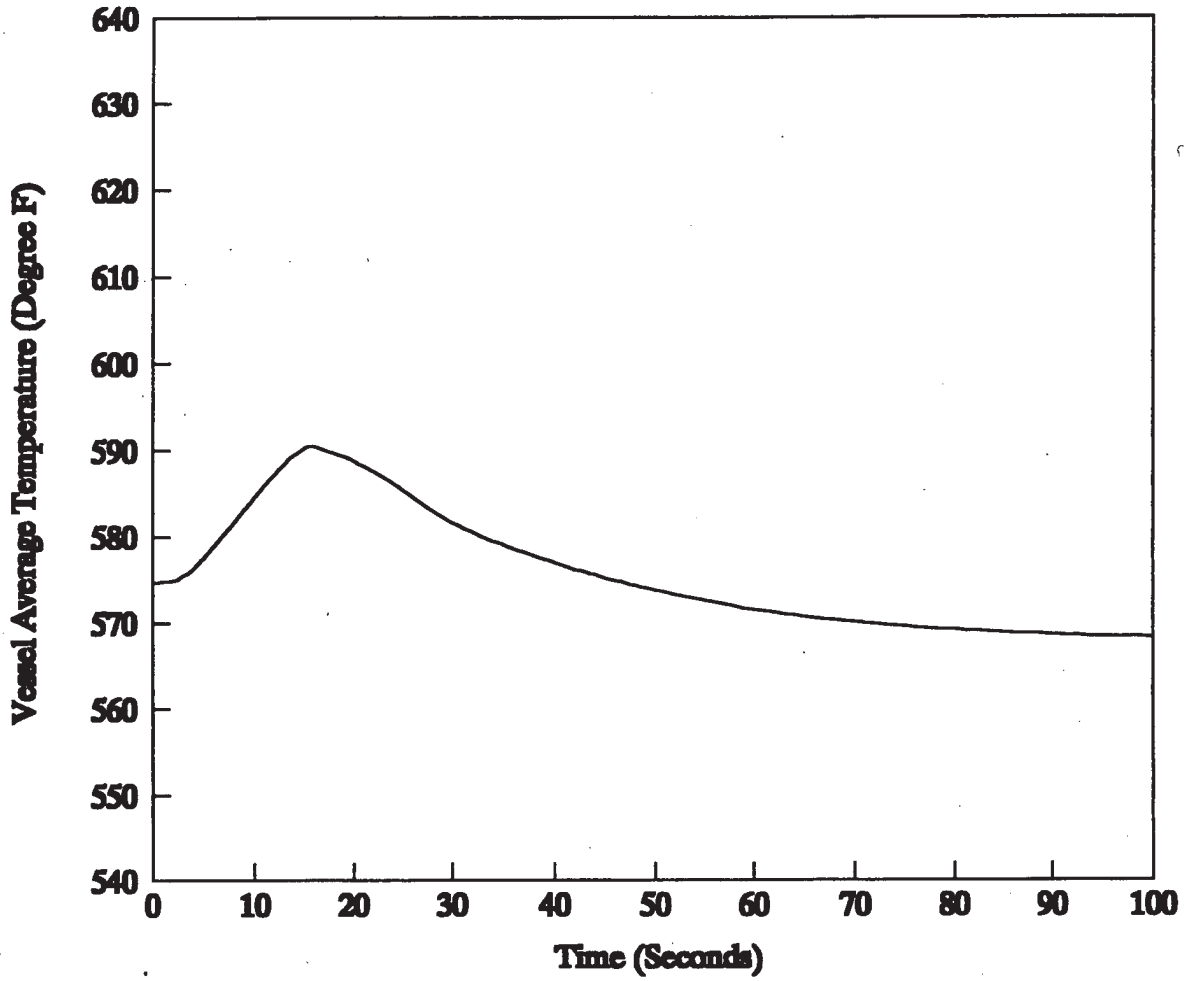


Figure 5.1.13-13

Steam Pressure Transient for Loss of External Load,

Maximum Reactivity Feedback,

With Pressure Control

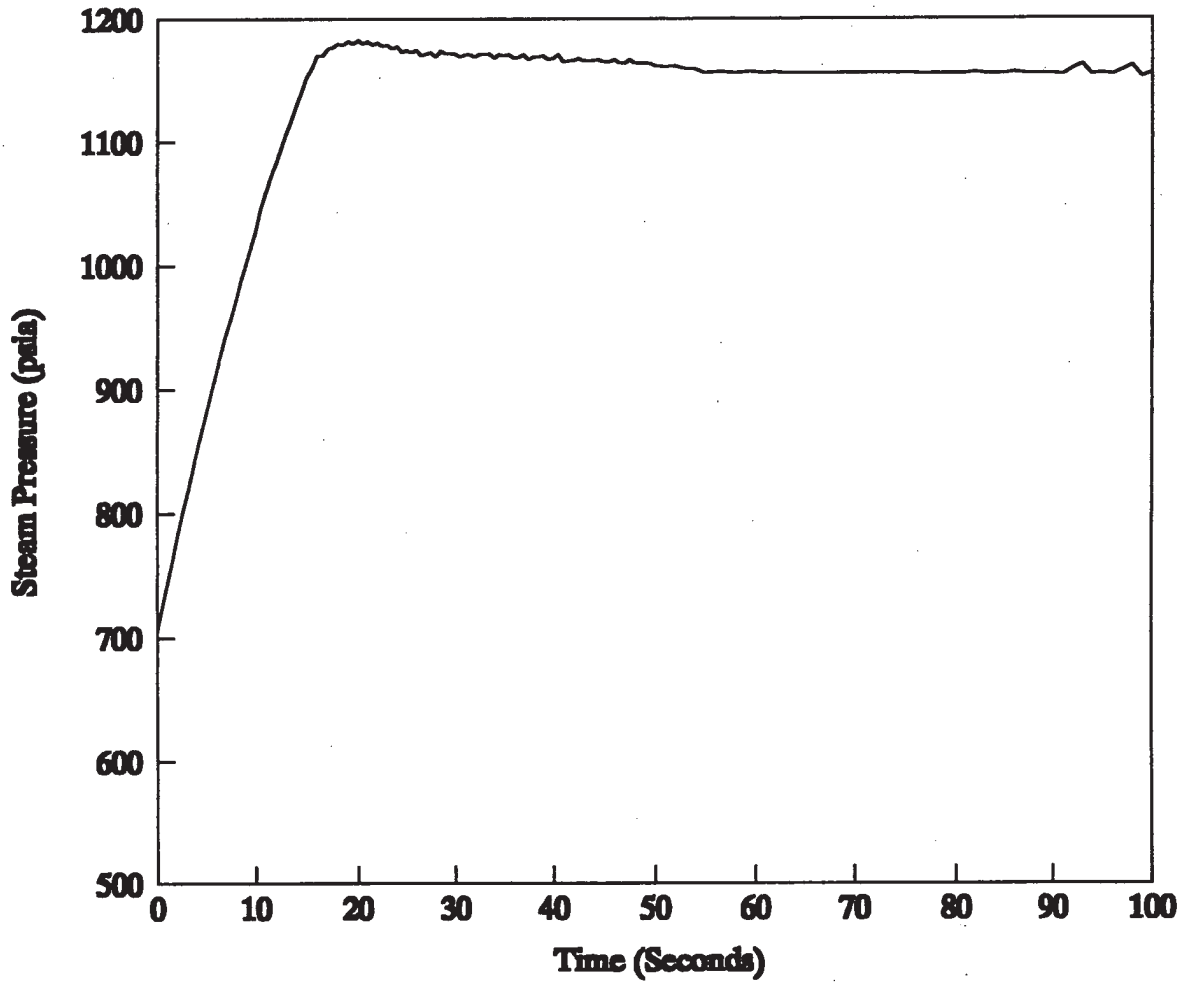


Figure 5.1.13-14

DNB Ratio vs Time for Loss of External Load,

Maximum Reactivity Feedback,

With Pressure Control

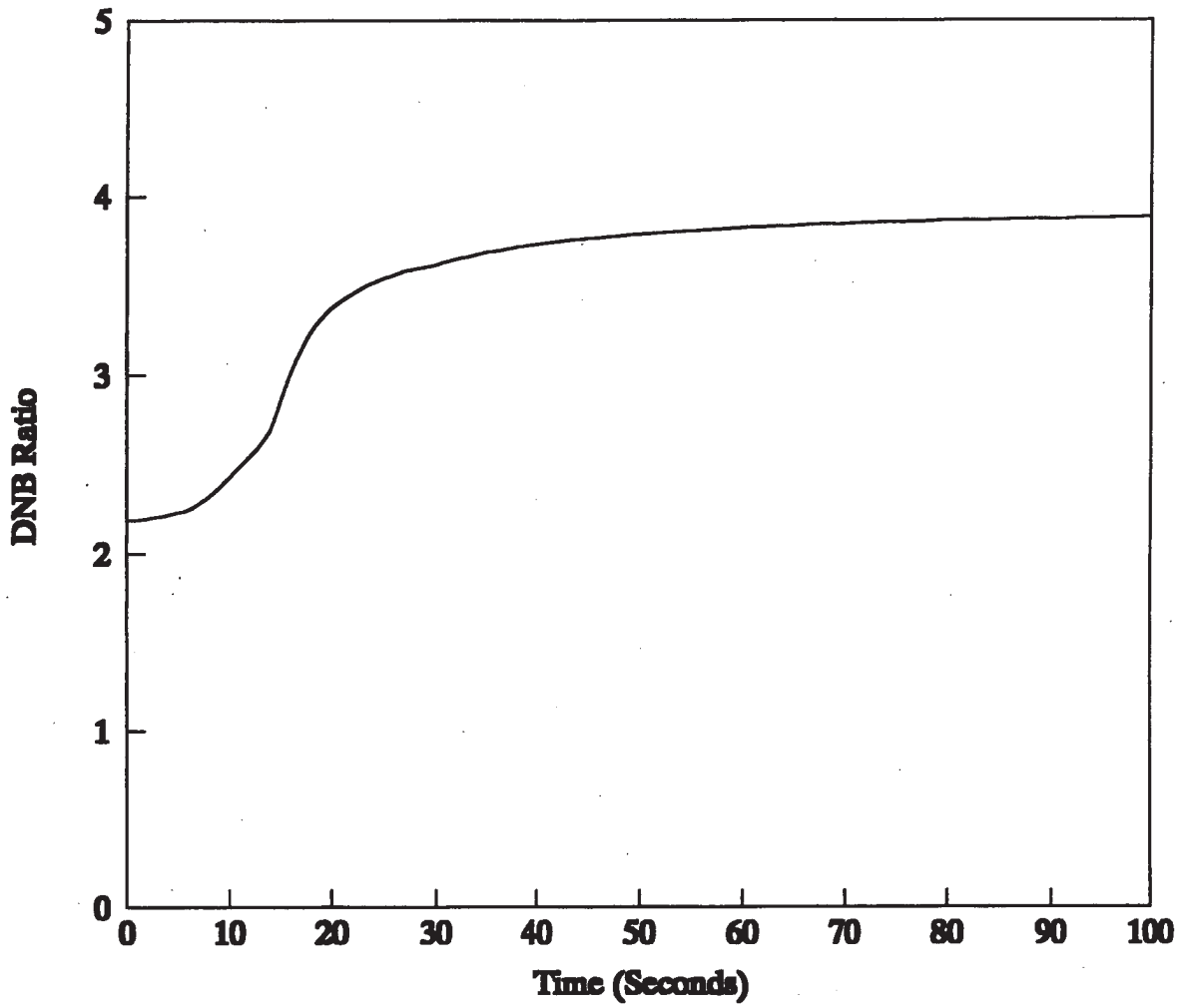


Figure 5.1.13-15

Nuclear Power Transient for Loss of External Load,

Minimum Reactivity Feedback,

Without Pressure Control

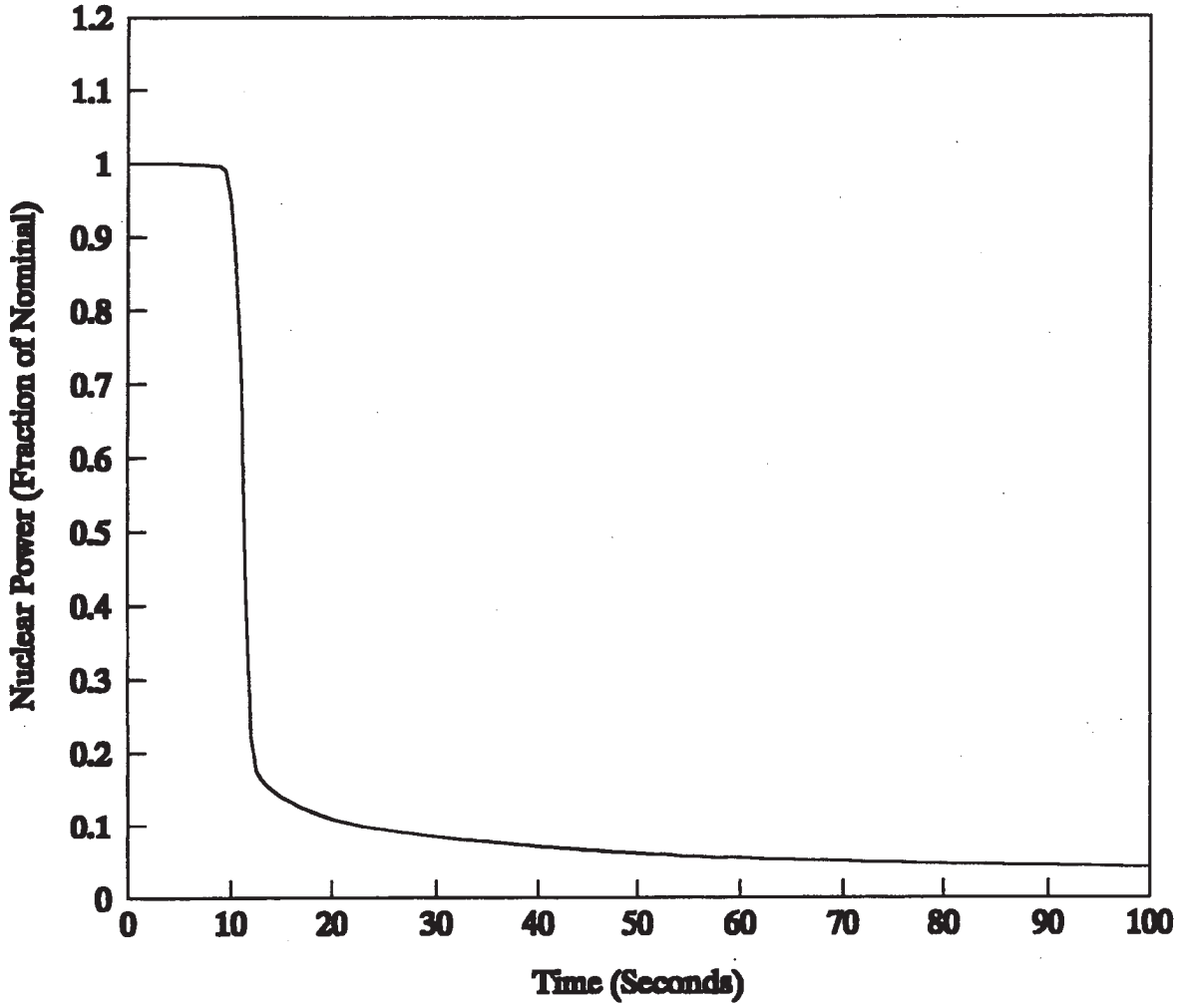


Figure 5.1.13-16

Core Heat Flux Transient for Loss of External Load,

Minimum Reactivity Feedback,

Without Pressure Control

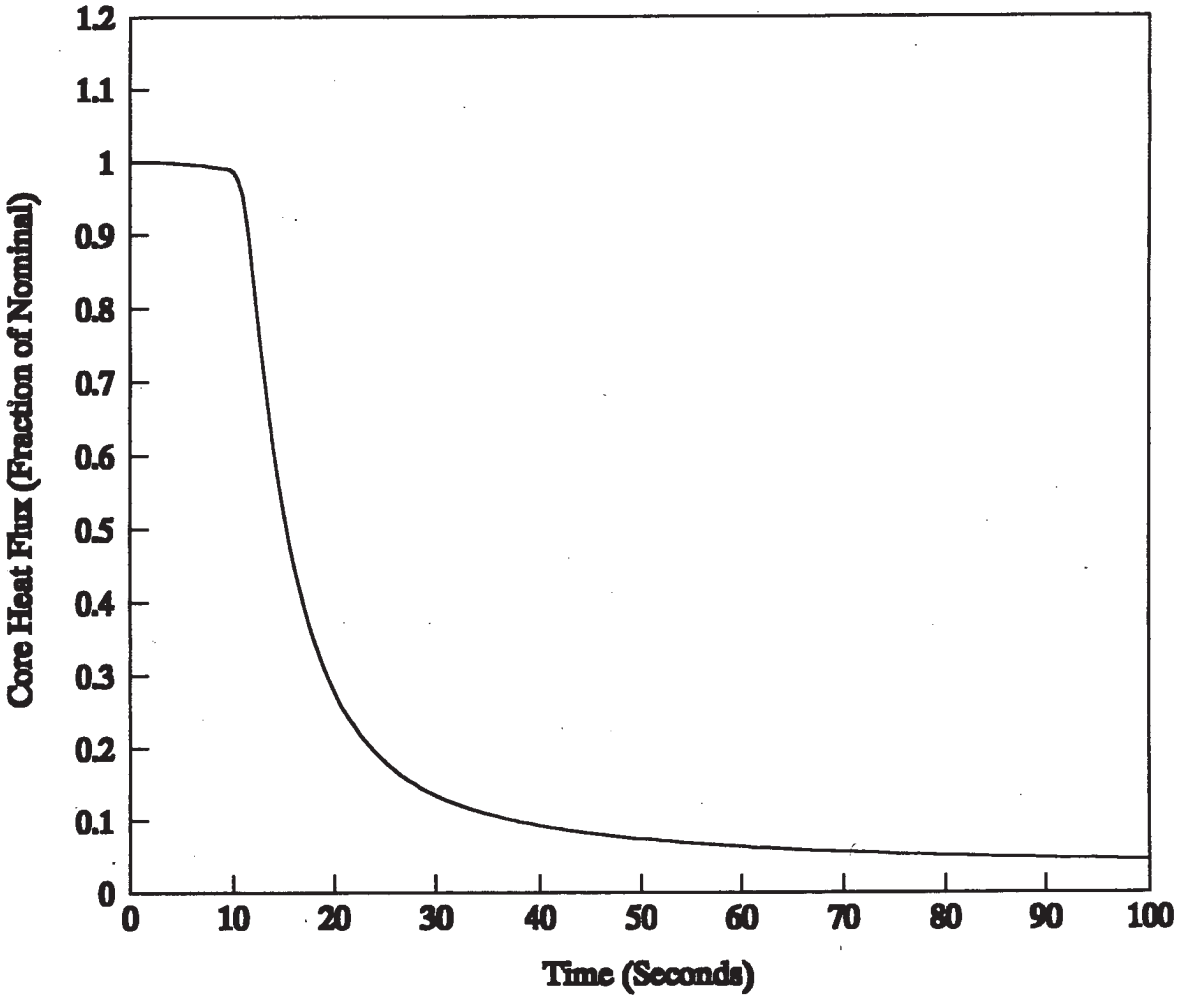


Figure 5.1.13-17

Pressurizer Pressure Transient for Loss of External Load,

Minimum Reactivity Feedback,

Without Pressure Control

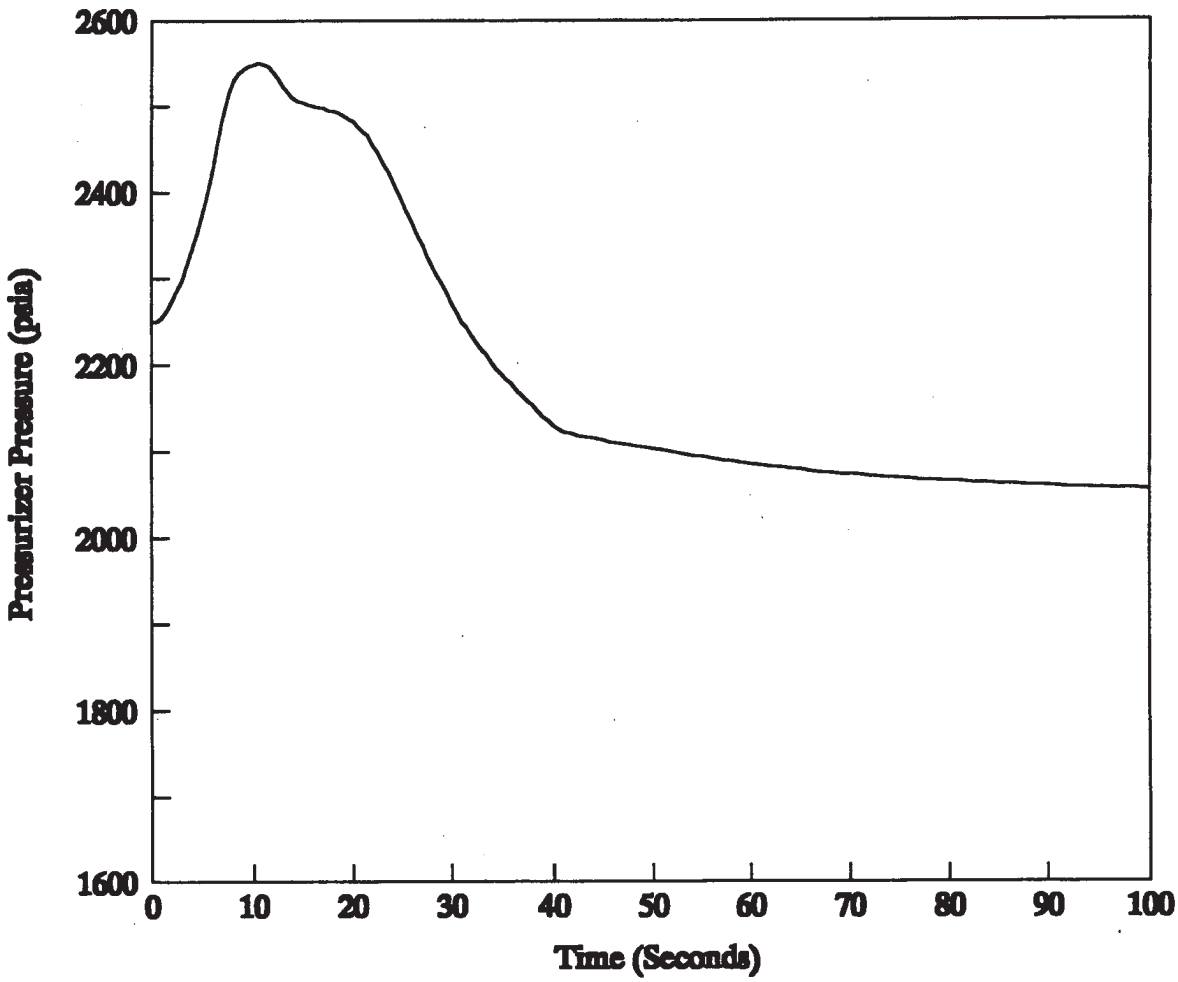


Figure 5.1.13-18

Pressurizer Water Volume Transient for Loss of External Load,

Minimum Reactivity Feedback,

Without Pressure Control

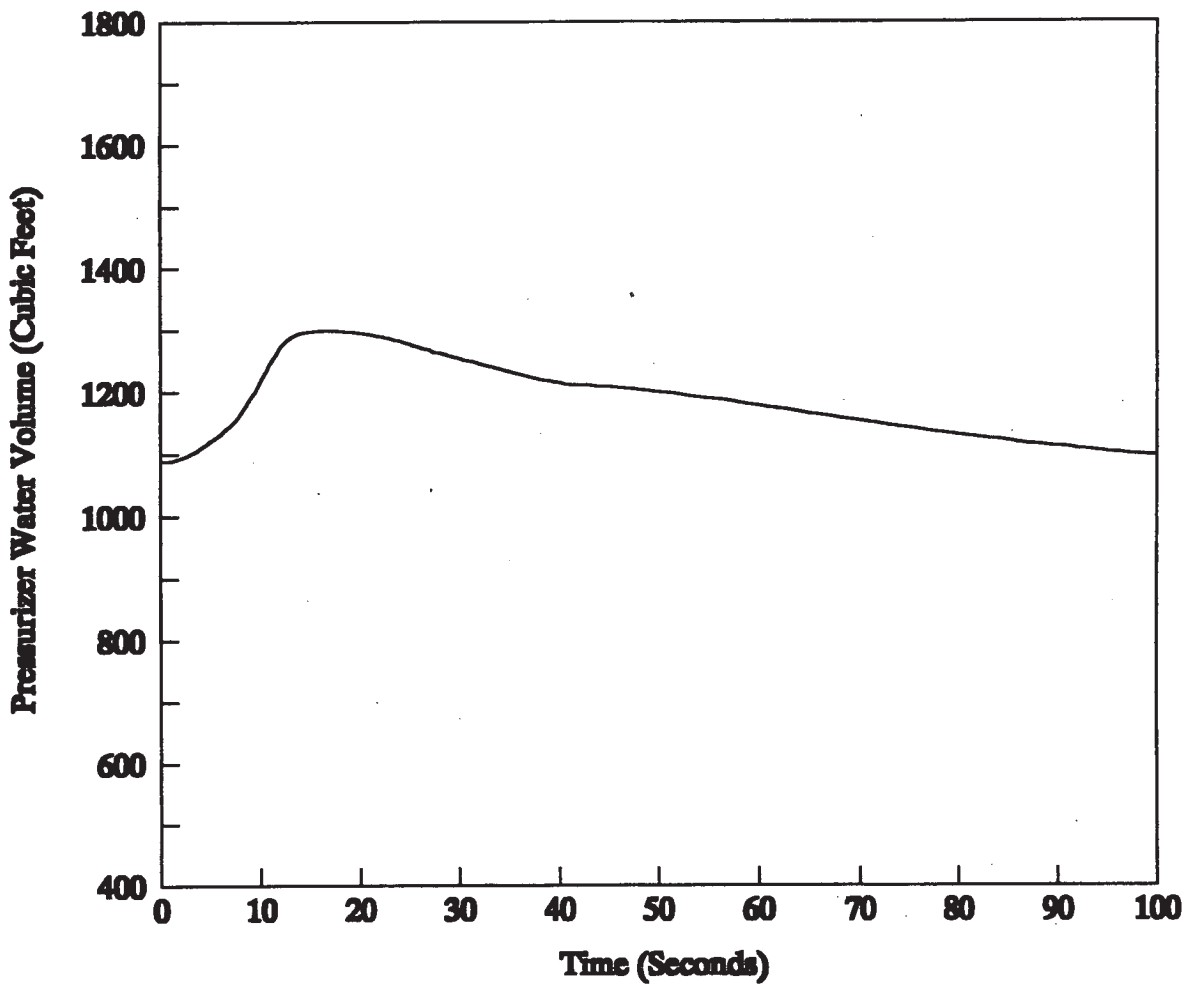


Figure 5.1.13-19

Vessel Average Water Temperature for Loss of External Load,

Minimum Reactivity Feedback,

Without Pressure Control

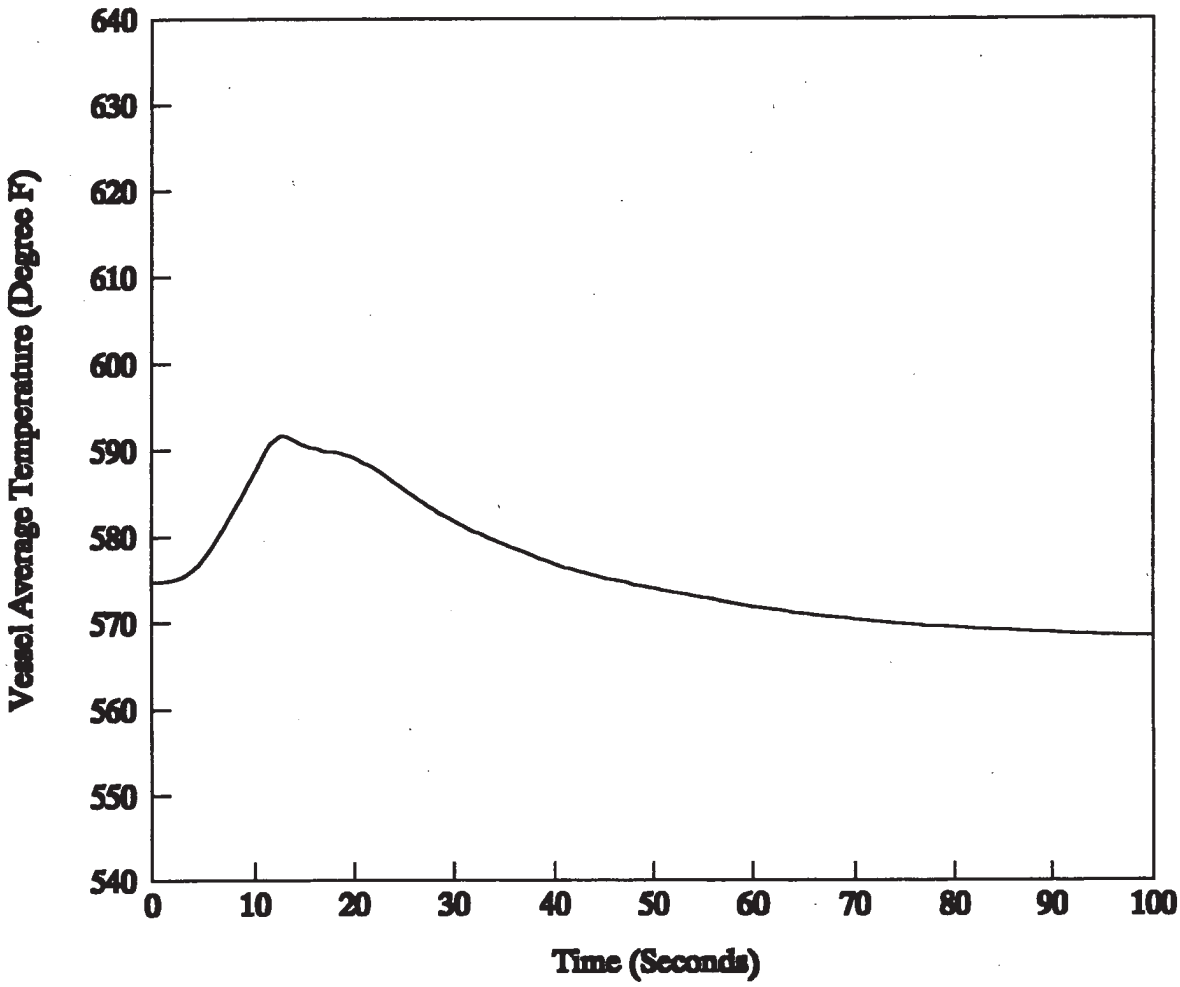


Figure 5.1.13-20

Steam Pressure Transient for Loss of External Load,

Minimum Reactivity Feedback,

Without Pressure Control

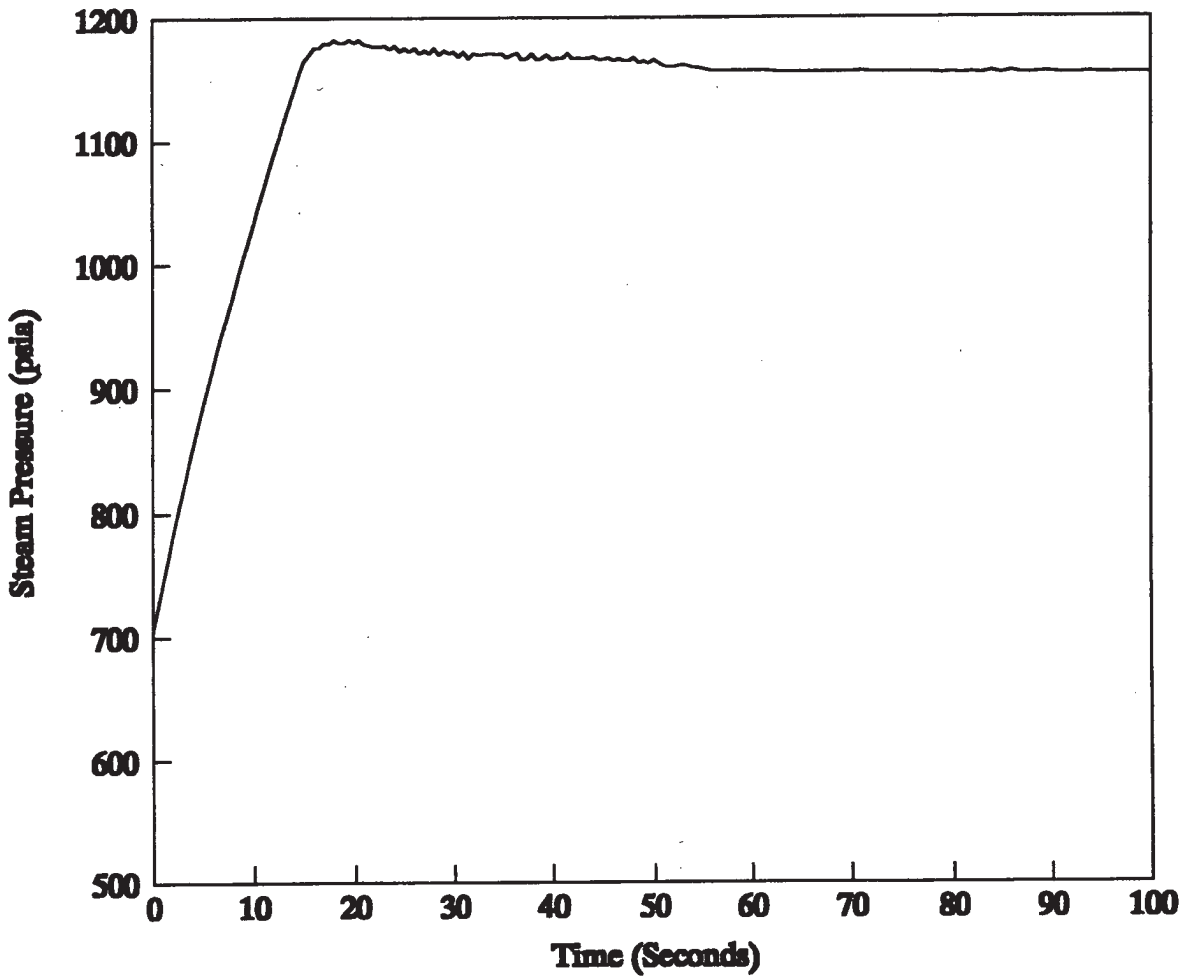


Figure 5.1.13-21

DNB Ratio vs Time for Loss of External Load,

Minimum Reactivity Feedback,

Without Pressure Control

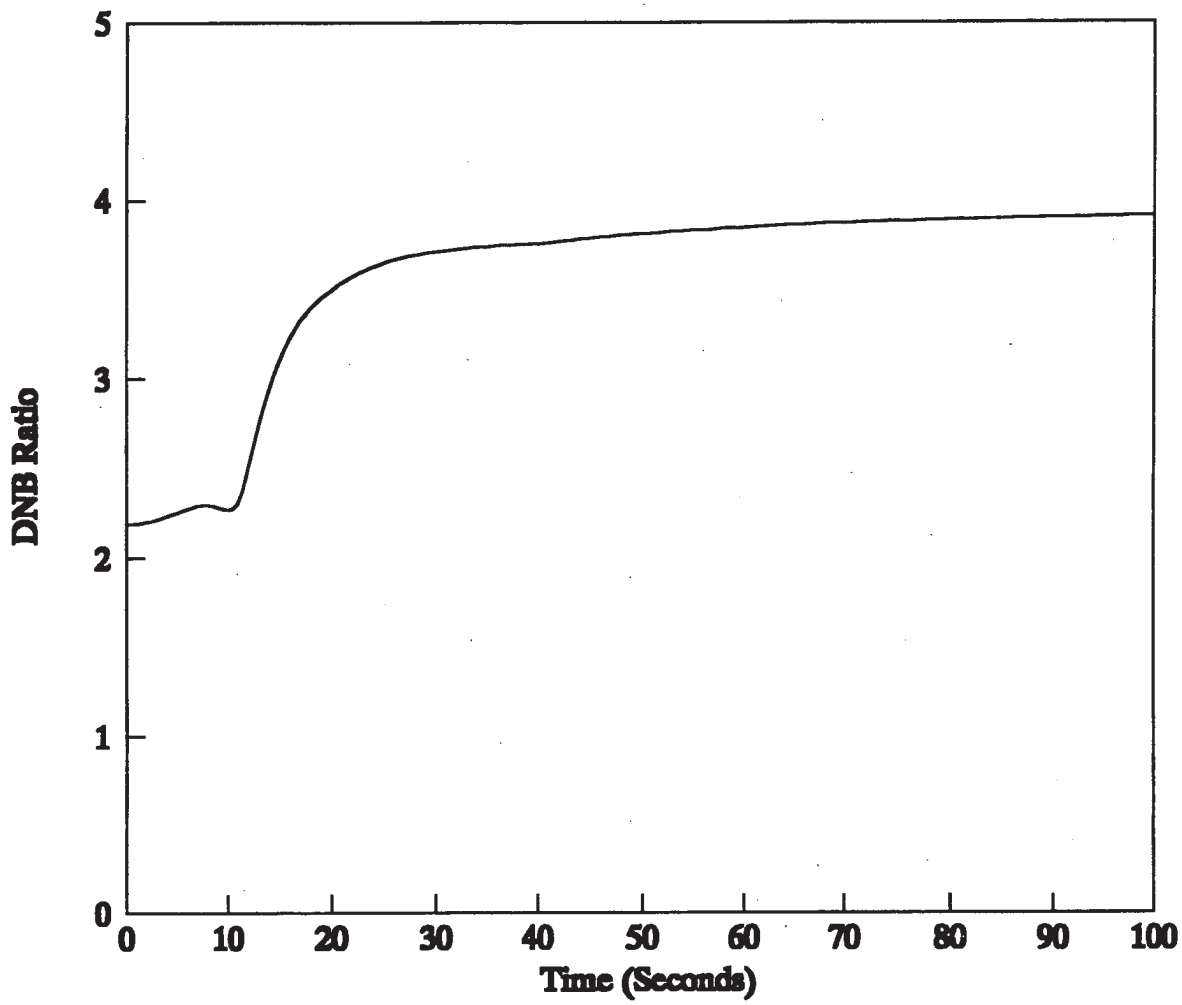


Figure 5.1.13-22

Nuclear Power Transient for Loss of External Load,

Maximum Reactivity Feedback,

Without Pressure Control

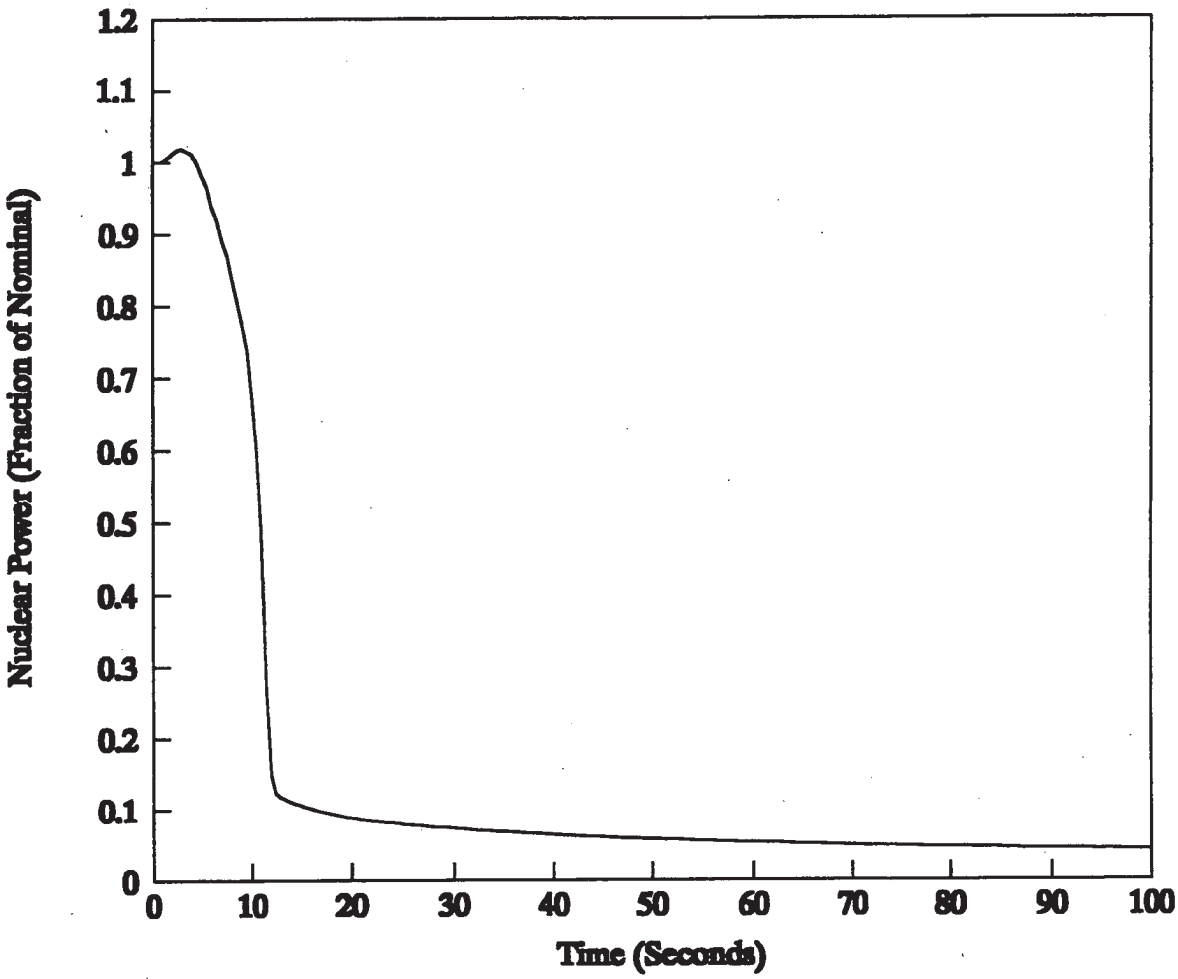


Figure 5.1.13-23

Core Heat Flux Transient for Loss of External Load,

Maximum Reactivity Feedback,

Without Pressure Control

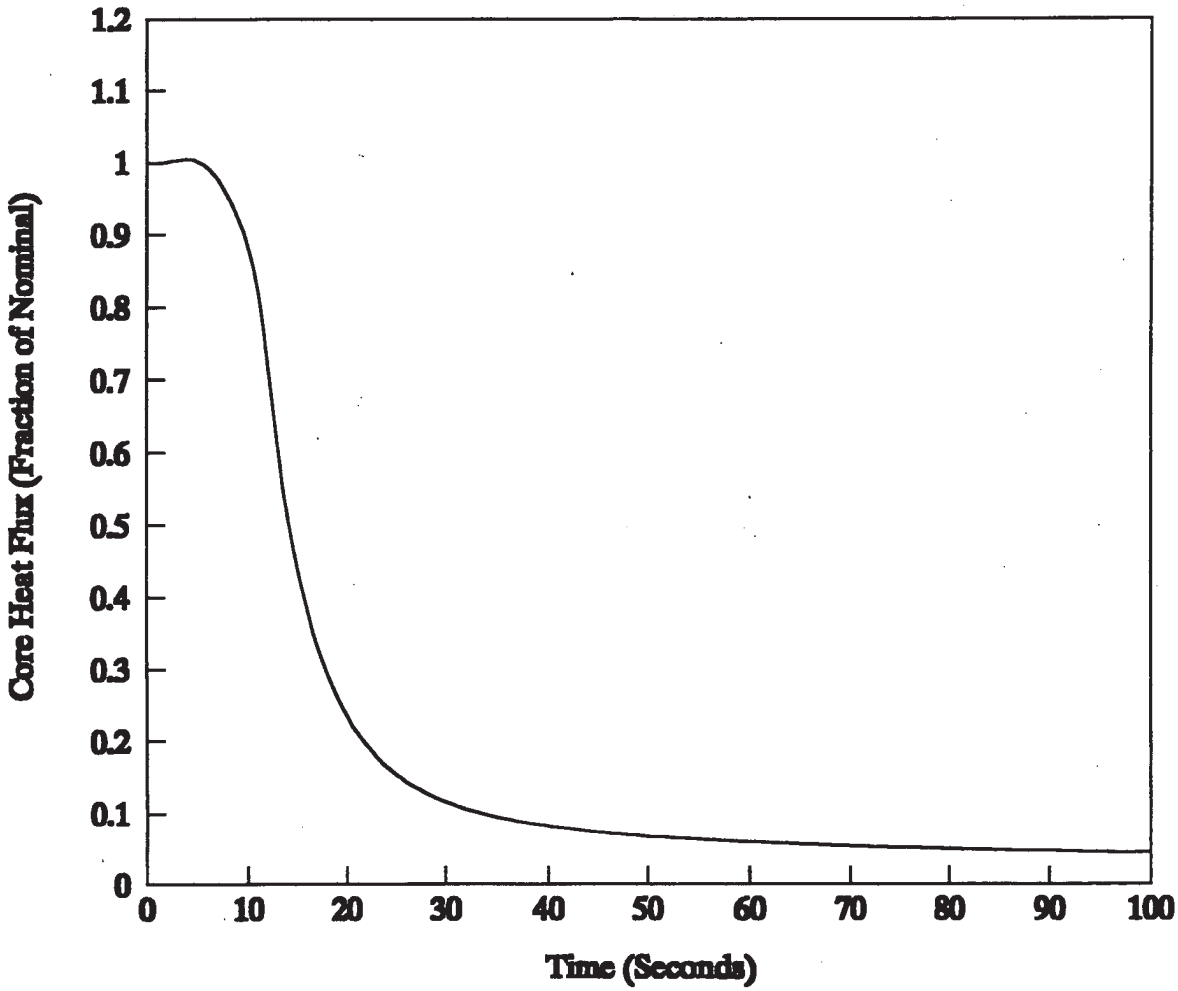


Figure 5.1.13-24

Pressurizer Pressure Transient for Loss of External Load,

Maximum Reactivity Feedback,

Without Pressure Control

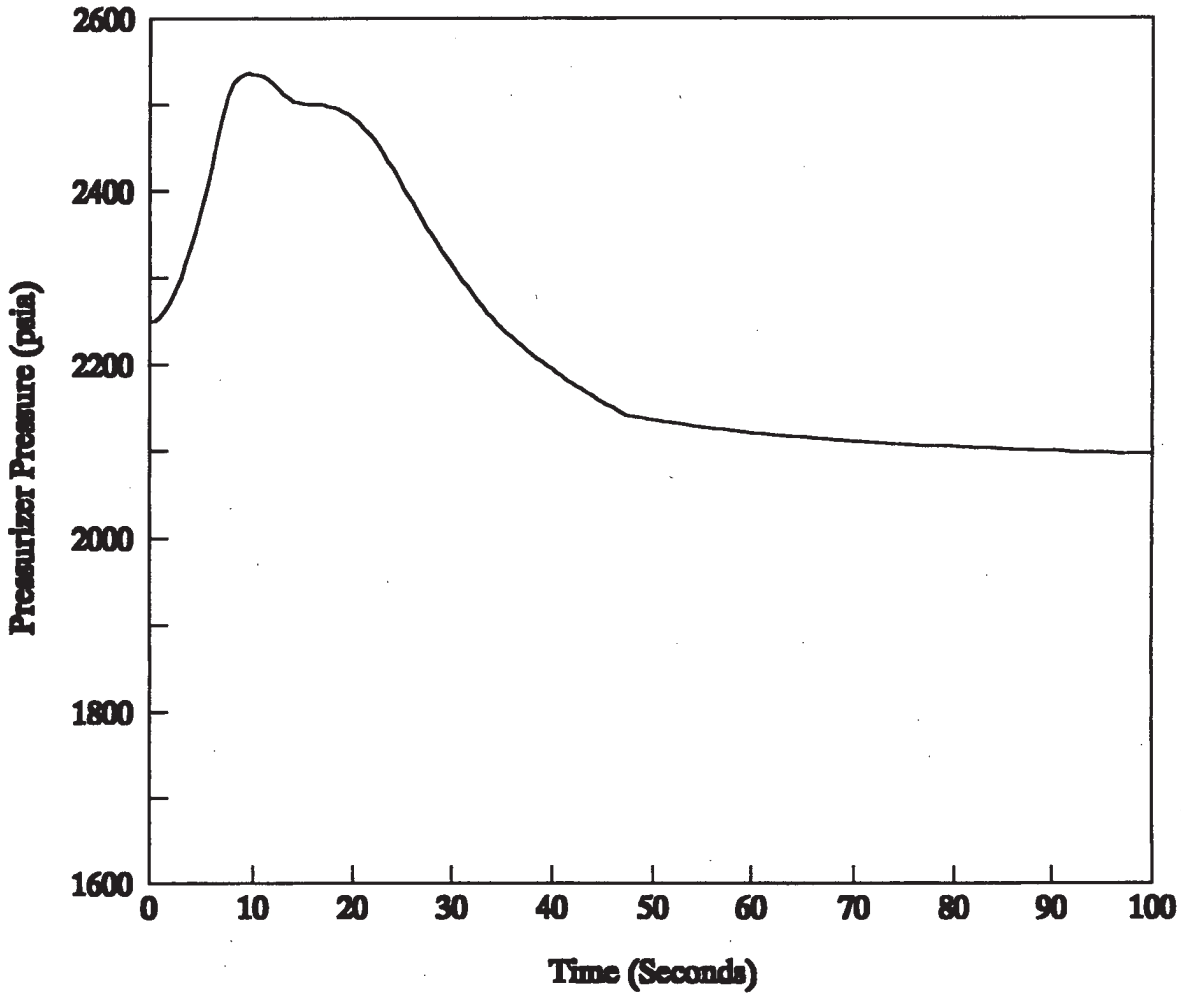


Figure 5.1.13-25

Pressurizer Water Volume Transient for Loss of External Load,

Maximum Reactivity Feedback,

Without Pressure Control

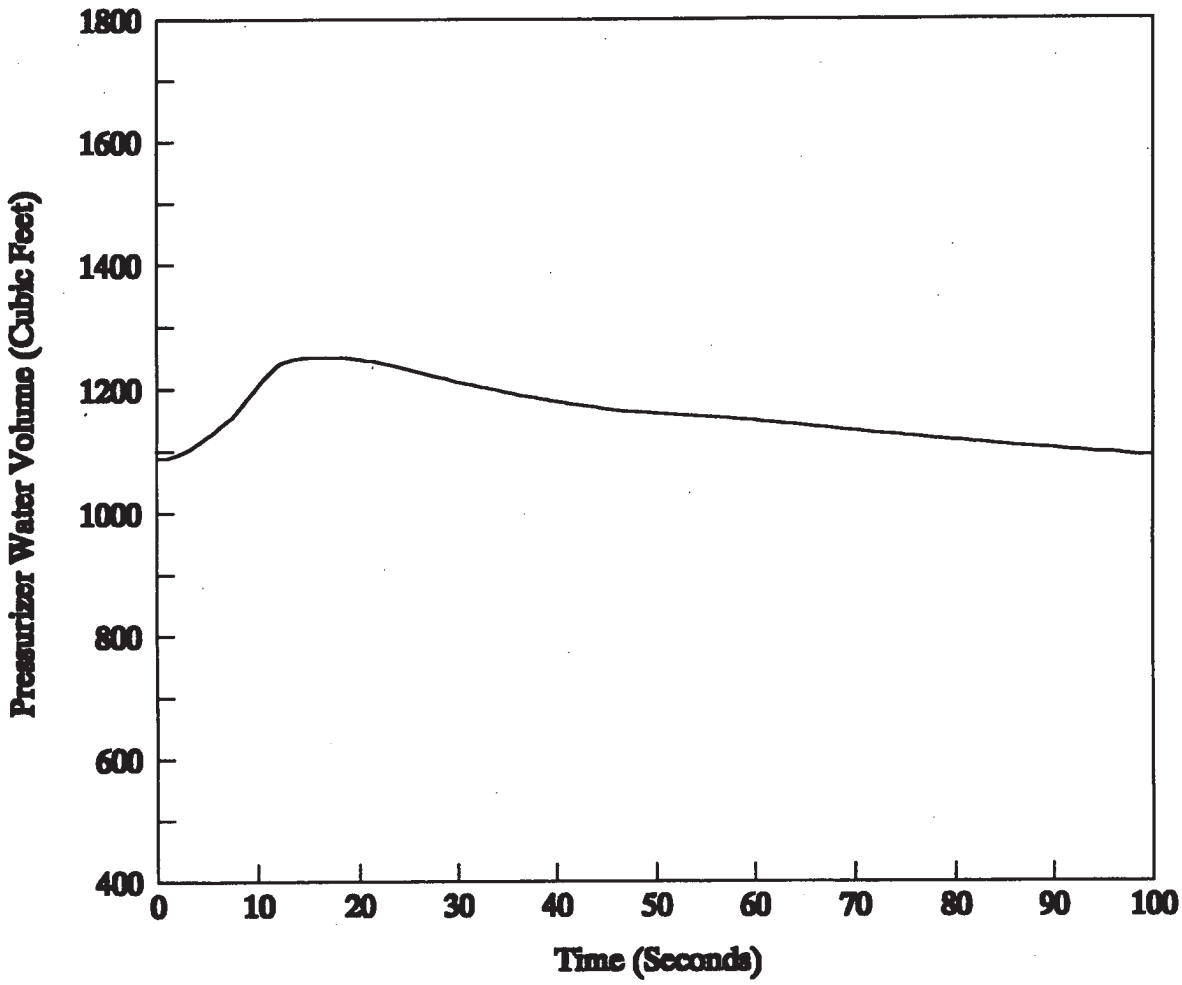


Figure 5.1.13-26

Vessel Average Water Temperature for Loss of External Load,

Maximum Reactivity Feedback,

Without Pressure Control

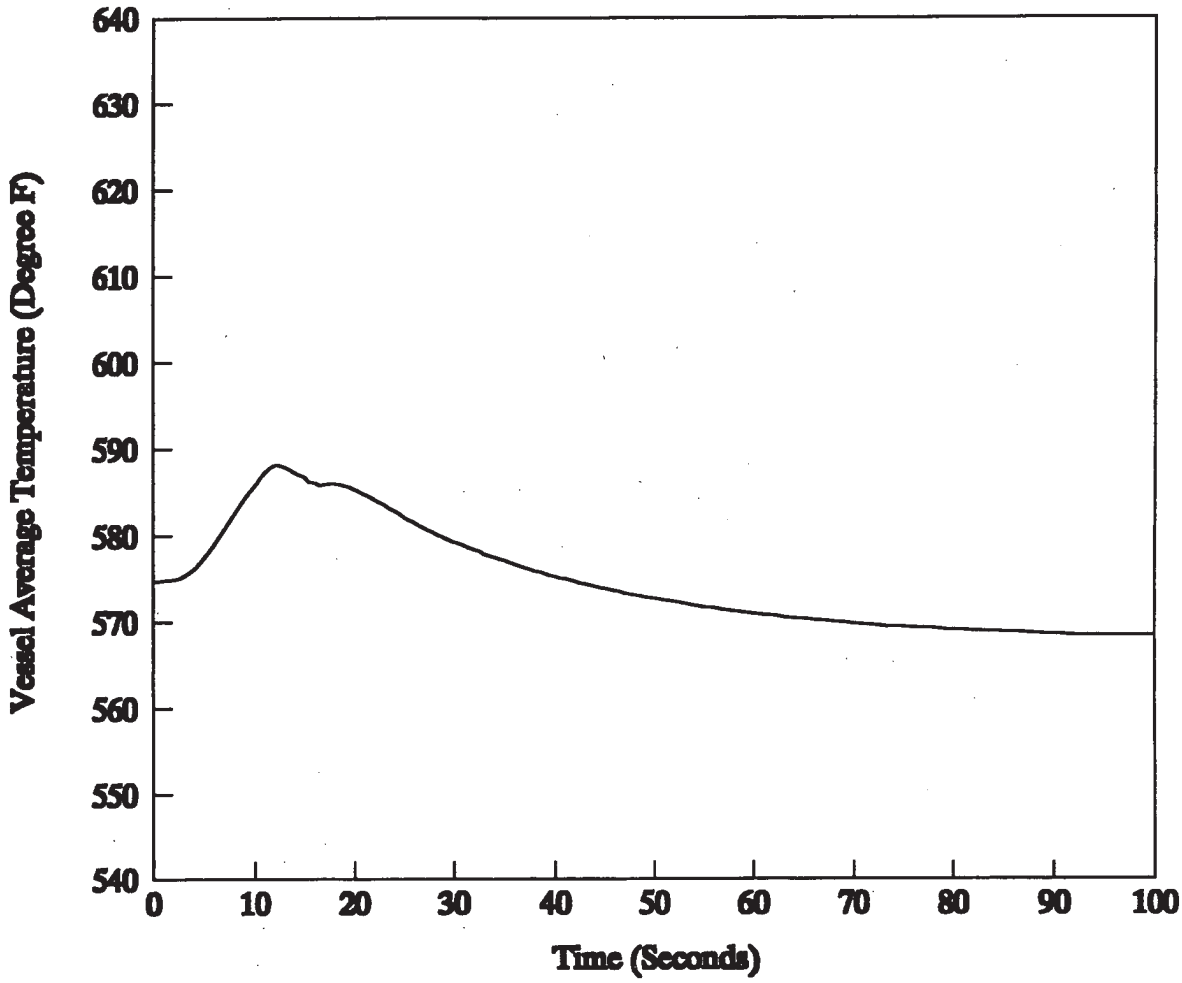


Figure 5.1.13-27

Steam Pressure Transient for Loss of External Load,

Maximum Reactivity Feedback,

Without Pressure Control

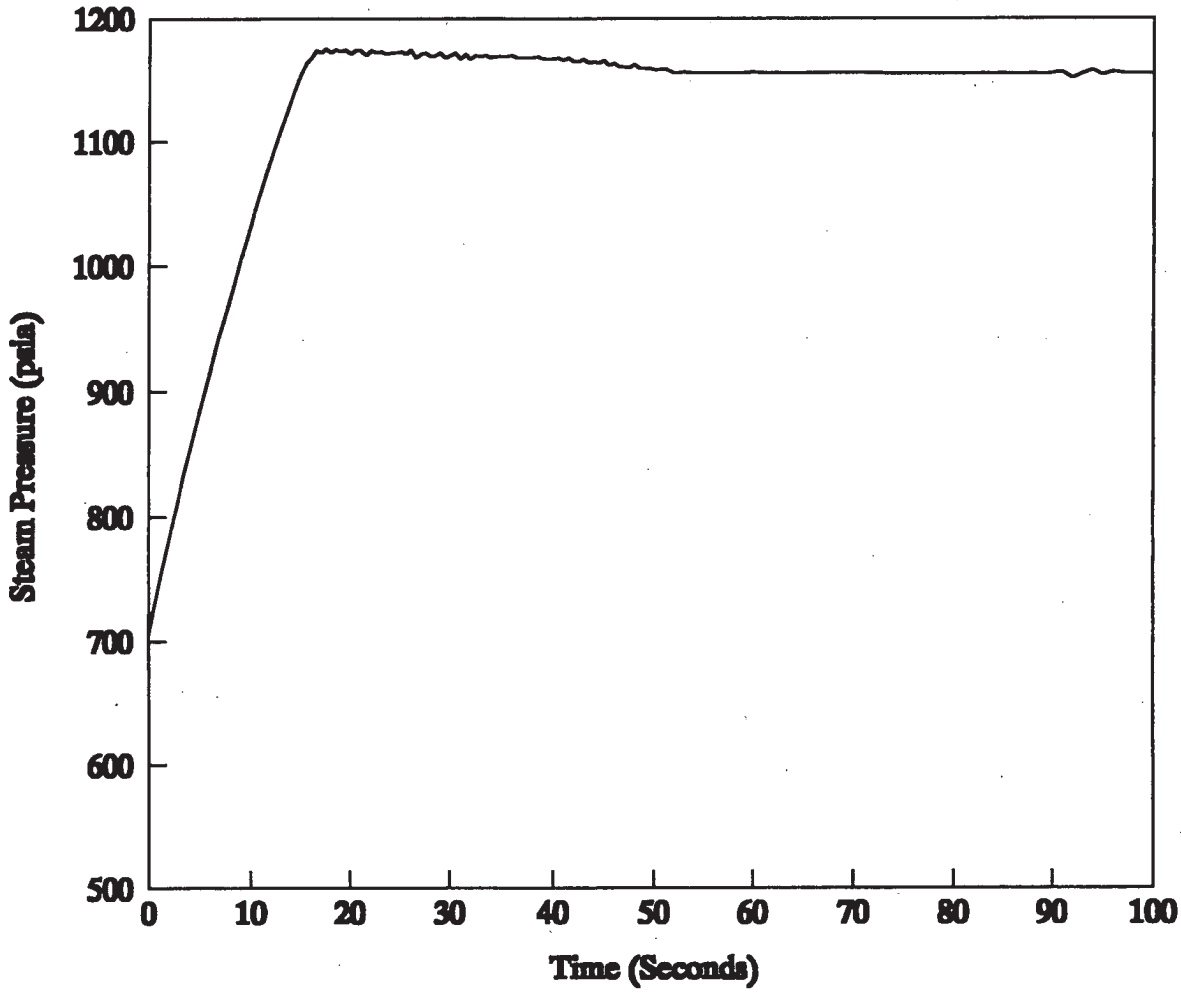
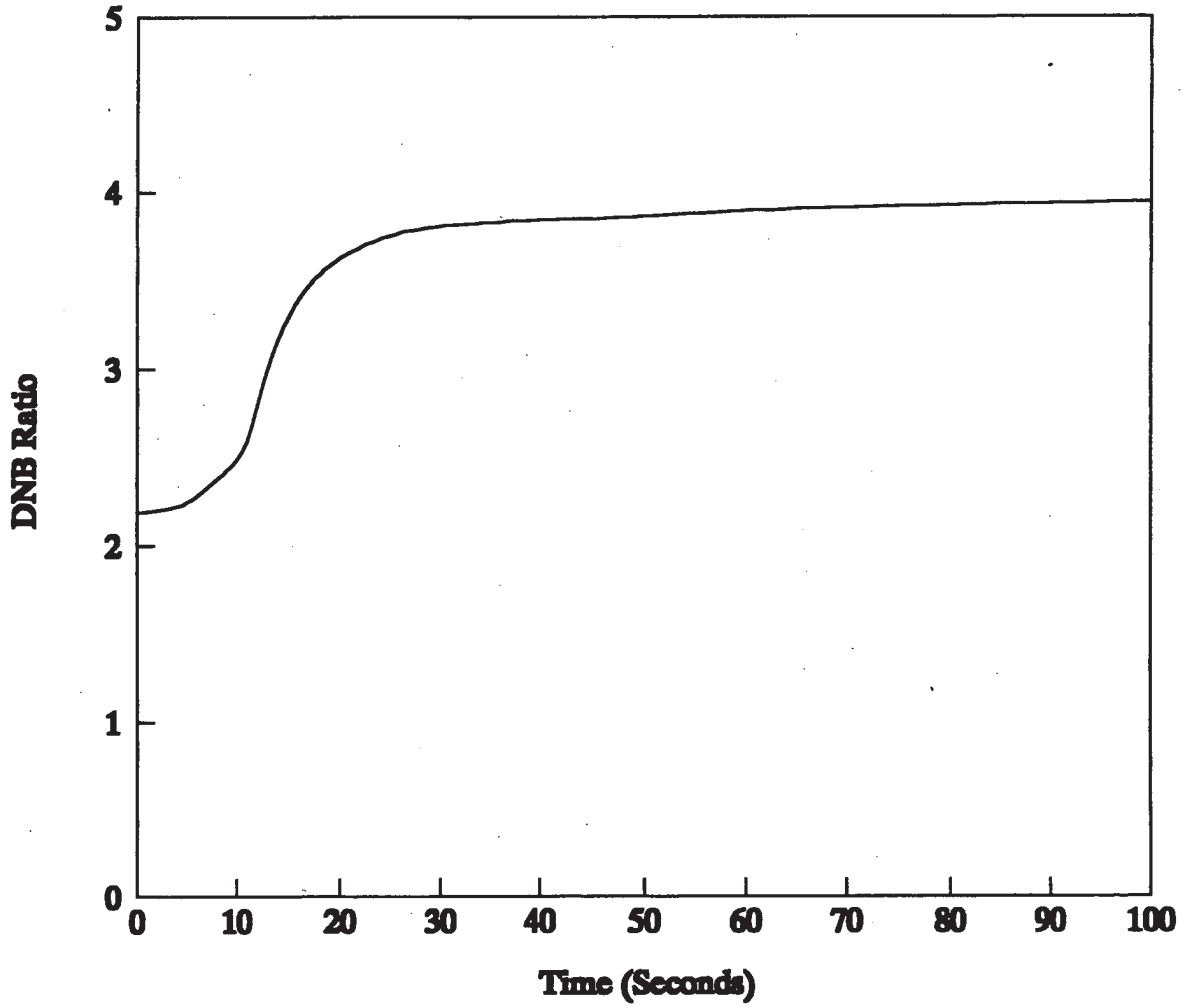


Figure 5.1.13-28

DNB Ratio vs Time for Loss of External Load,

Maximum Reactivity Feedback,

Without Pressure Control



5.1.14 Loss of Normal Feedwater

Introduction:

The evaluation herein was performed for the Loss of Normal Feedwater event as described in the FSAR Section 14.1.9 to support the insertion of VANTAGE + Fuel with the design features described in Section 5.1.2. The evaluation also addresses changes in the safety analysis assumptions associated with the VANTAGE + transition as described in Section 5.1.3.

A loss of normal feedwater (from pump failures or valve malfunctions) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, primary plant damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not furnished, residual heat following reactor trip would heat the primary system water to the point where a water-solid pressurizer condition would occur resulting in subsequent water relief from the pressurizer. A loss of significant water from the reactor coolant system could conceivably lead to core damage. However, a low-low steam generator water level condition generates a reactor trip and acutates auxiliary feedwater ensuring the plant is tripped well before the steam generator heat transfer capacity is significantly reduced. Thus, primary system variables never approach a departure from nucleate boiling condition and long term cooling is assured.

The Loss of Normal Feedwater event is classified as an ANS Condition II fault as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious fault, i.e, a Condition III or IV category event. The applicable Indian Point Unit 3 safety analysis licensing basis acceptance criteria for this Condition II event are:

a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (i.e., 2748.5 psia and 1208.5 psia, respectively);

b) Fuel Cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit; and,

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

To address criterion (c), the following more restrictive criterion has been adopted for ease in interpreting the transient results:

> The pressurizer shall not become water solid as a result of this Condition II transient within 10 minutes; the minimum time required for the operator to identify the event and take corrective action.

The basis for demonstrating that the pressurizer will not become water solid is to preclude the possibility of discharging primary coolant through the pressurizer power operated relief valves and/or the safety relief valves, causing the event to progress from one of moderate frequency to an infrequent fault (i.e., small break LOCA) if the valves should fail to close after discharging water.

Evaluation:

The analysis of this event is performed to show that following a loss of normal feedwater, the auxiliary feedwater system provides sufficient inventory to the secondary-side system to remove stored and residual heat from the reactor core; preventing the pressurizer from reaching a water solid condition and precluding any subsequent water relief through the pressurizer relief or safety valves.

As indicated in the Introduction section, departure from nucleate boiling is not a concern for this event since the plant is tripped well before the steam generator heat transfer capacity is significantly reduced and the primary system variables never approach DNB conditions. Therefore, the VANTAGE + Fuel features and other revised safety analysis assumptions as described in Sections 5.1.2 and 5.1.3, respectively, will not have any adverse effect on the existing licensing basis analysis of the Loss of Normal Feedwater event.

For completeness, it should be noted that the current licensing basis analysis was performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq$ 24%; and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of $\leq$ 24% with a maximum steam generator tube plugging level of 30% in any one steam generator. This is acheived by assuming a uniform 30% steam generator tube plugging level to represent the most conservative assumption in order to depict the reduced

primary-to-secondary heat transfer due to asymmetric steam generator tube plugging.

Conclusions:

Based on the evaluation herein, it is concluded that the current licensing basis safety analysis for the Loss of Normal Feedwater event remains valid for the insertion of VANTAGE + Fuel into Indian Point 3. The safety analysis shows that all applicable Condition II safety criteria are met for this event. Specifically, the minimum auxiliary feedwater capacity of 340 gpm is sufficient to prevent pressurizer from reaching a water-solid condition and precludes any subsequent water relief through the pressurizer relief and safety valves. This also assures that the reactor coolant system is not overpressurized. The current licensing basis analysis also demonstrates that sufficient long-term heat removal capability exists to prevent fuel or clad damage.

The above conclusion is valid for the VANTAGE + Fuel including the design features and associated changes in the safety analysis assumptions as described in Sections 5.1.2 and 5.1.3, respectively.

5.1.15 Excessive Heat Removal due to Feedwater System Malfunctions

Introduction:

Excessive feedwater additions are postulated to occur from a malfunction of the feedwater control system or an operator error which results in the opening of a feedwater control valve. With the reactor at power, this excess feedwater flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater causes a decrease in RCS temperature and a consequential reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous excessive feedwater addition is terminated by the automatic feedwater isolation actuated upon receipt of a steam generator high-high water level signal. The steam generator high-high water level signal also results in a turbine trip and a subsequent reactor trip signal on turbine trip.

Excessive feedwater addition is a means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower and overtemperature protection (OPΔT and OTΔT trips) and high neutron flux trip prevent any power increase that could lead to a DNBR less than the applicable DNBR limit.

The Excessive Heat Removal due to Feedwater System Malfunction event is a Condition II event as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious fault, i.e., a Condition III or IV category event.

The applicable safety analysis licensing basis acceptance criteria for the Condition II Excessive Heat Removal due to Feedwater System Malfunction event for Indian Point Unit 3 are:

a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (i.e., 2748.5 psia and 1208.5 psia, respectively);

b) Fuel Cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit; and,

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

Method of Analysis and Assumptions:

The feedwater system malfunction transient is analyzed using the LOFTRAN computer code as described in Section 5.1.5 to determine the effects of the excessive heat removal on the reactivity insertion rate, RCS pressure, secondary-side pressure, and DNBR, for the primary purpose of assuring the required protection system features are adequate to prevent the applicable safety analysis limits from being exceeded.

The analysis is performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq$ 24%; and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of $\leq$ 24% with a maximum steam generator tube plugging level of 30% in any one steam generator and considers the following four cases of excessive feedwater addition;

1) Accidental opening of one feedwater control valve from full power initial conditions with the reactor in automatic rod control.

2) Accidental opening of one feedwater control valve from full power initial conditions with the reactor in manual rod control.

3) Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming the reactor in automatic rod control.

4) Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions with the reactor in manual rod control.

In all four cases, one feedwater valve is assumed to accidently open fully resulting in the excessive feedwater flow to one steam generator. For the cases analyzed at full power initial conditions, the valve opening is assumed to result in a step increase in feedwater flow to 250% of nominal feedwater flow to one steam generator. For the feedwater control valve accident at zero load conditions, a feedwater valve malfunction is assumed to occur that results in a step increase in flow to one steam generator from zero to the nominal full load feedwater flow rate for one steam generator.

The analysis assumptions are conservatively selected to bound conditions for both 24% uniform and asymmetric steam generator tube plugging levels. For the asymmetric steam generator tube plugging condition, the excessive feedwater flow is assumed to occur in the loop with the least plugged steam generator to maximize the cooldown effects.

Other pertinent analysis assumptions that affect the transient conditions following the postulated feedwater system malfunction are as follow.

Initial Conditions:

Initial conditions consistent with the implementation of the RTDP (see Section 5.1.3) are used in analysis. These include the use of the following nominal initial conditions:

Initial Condition	HFP	HZP
Core Power (MWt)	3025.	30.25
NSSS Power (MWt)	3037.	42.25
Pressurizer Pressure (psia)	2250.	2250.
Reactor Vessel Inlet Temperature (°F)	543.6	547.0
Reactor Vessel Average Temperature (°F)	574.7*	547.0*
Minimum Measured Flow (plant total) (gpm)	330800.	330800.
Core Bypass Flow (fraction)	0.037	0.037

* Based on uniform Steam Generator tube plugging.

Other non-RTDP related initial conditions are:

Initial Condition	HFP	HZP
Pressurizer Level (% level span)	51.3	23.1
Pressurizer Water Volume (ft^3)	921.66	451.53
Steam Generator Level (% NRS)	45.0	45.0
Steam Generator Mass (lbm)	74,014.2*	115,028.0*
Upper Head Temperature (°F)	606.4	547.0
Feedwater Enthalpy (Btu/lbm)	405.2	3.4

* - Based on 90% of SG Mass at initial level

Control Systems:

For the cases analyzed assuming automatic rod control, the rod control system is modeled to maintain the program T_{avg} which is assumed to vary linearly between 547 °F at no-load conditions to 574.7 °F at full power. Since the event is primarily analyzed for DNB (e.g., cooldown events are not limiting with respect to overpressure concerns) using RTDP, no temperature error is assumed on the rod control system. However, the temperature error is statistically considered in establishing the safety analysis DNBR limit. In addition, since the event is primarily analyzed for DNB, portions of the pressurizer pressure control system; including pressurizer power-operated relief valves (PORVs) and pressurizer sprays, are assumed for the purpose of minimizing pressure. The PORVs are conservatively modeled to begin relieving at 2350 psia and be at a full relief capacity of 14.46 ft^3/sec of steam at 2355 psia. The pressurizer sprays are modeled to begin actuating at a pressure 25 psi above the nominal pressure of 2250 psia with a spray rate increasing linearly to a maximum of 84.33 lbm/sec at 75 psi above the nominal pressure.

No other control systems are assumed to operate and none of the control systems which are modeled are assumed for the purpose of mitigating the consequences of this event.

Protection Systems:

For the feedwater system malfunction accident at full power, the feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high water level signal that closes all feedwater control valves and trips the main feedwater pumps. The steam generator high-high water level signal also produces a signal to trip the turbine. In the LOFTRAN analysis, the high-high water level setpoint condition is modeled to occur when the steam generator mass reaches 106,323 lbm corresponding

to a high-high water level trip setpoint of 85% NRS, including uncertainties.

A turbine trip is modeled to occur 5 seconds after the steam generator water level reaches the high-high steam generator water level condition. If a reactor trip has not yet occurred from either a high neutron flux reactor trip signal or an OPΔT reactor trip signal, a reactor trip will occur 4 seconds after the turbine trip (a total of 9 seconds after the high-high steam generator water level setpoint is reached). Should the turbine trip not result in a reactor trip signal, reactor trip would eventually occur on another reactor trip signal (e.g., high neutron flux, low-low steam generator water level).

To determine the maximum reactivity insertion rate that occurs following the feedwater control valve failure the reactor is assumed to be just critical at zero load initial conditions and feedwater isolation, turbine trip, and subsequent reactor trip on turbine trip are not modeled upon reaching a high-high steam generator water level turbine trip setpoint in the zero power cases.

Reactivity Modeling:

The feedwater system malfunction accident results in a cooldown of the primary system due to the excessive feedwater flow. Therefore, reactivity feedback characteristic of end-of-life conditions are assumed in the analysis. These include a conservatively large moderator density coefficient equivalent to a moderator temperature coefficient of -65.8 pcm/°F, a minimum Doppler Power-Only coefficient of -9.55 + 0.00104Q; where Q = power in MWt, a Doppler temperature coefficient of -3.2 pcm/°F, and a maximum β_{eff} of 0.007. In addition, the analysis conservatively assumes no decay heat and radial weighting to the core quadrant with the steam generator receiving the excess feedwater.

A total trip reactivity of -4% ΔK excluding the highest worth rod is conservatively assumed with a scram time of 2.7 seconds from beginning of rod motion until the dashpot is reached.

Heat Transfer Modeling:

Fuel-to-coolant heat transfer coefficients conservatively representing minimum fuel temperature conditions that bound operation with both VANTAGE 5 (w/o IFMs) and VANTAGE + fuel are assumed in the analysis.

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

unaffected steam generators. The primary-to-secondary heat transfer corresponding to no steam generator tube plugging is conservatively assumed to maximize the cooldown associated with this event.

Results:

Zero Power Cases:

In the cases of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rates are less than 75 pcm/sec; which is the reactivity insertion rate analyzed in Section 5.1.6, Uncontrolled Control Rod Withdrawal from a Subcritical Condition. Therefore, the zero power cases are bounded by the detailed analysis presented in Section 5.1.6 for both the automatic and manual rod control cases. It should be noted that if the incident occurs with the reactor just critical at no-load, the reactor may be tripped by the power range neutron flux trip (low setting).

Full Power Cases:

With respect to minimum DNBR, the most limiting full power case is that assuming automatic rod control and asymmetric steam generator tube plugging conditions. For the case initiated from full power conditions assuming automatic rod control and asymmetric steam generator tube plugging, the Nuclear Power, Reactor Vessel Average Temperature, Affected Loop ΔT, Pressurizer Pressure, Steam Generator Pressure, Steam Generator Mass, and DNBR transient results are illustrated in Figure 5.1.15-1 through Figure 5.1.15-7, respectively. Figure 5.1.15-8 through Figure 5.1.15-14 show the equivalent transient conditions for this case with manual rod control. The transient results for the full power cases with and without automatic rod control and modeling uniform steam generator tube plugging are similar to those illustrated in these figures.

For all the full power cases, the steam generator water level rises until the feedwater addition is terminated at 12 seconds after the high-high steam generator water level setpoint (85% narrow range span, including uncertainties) is reached. A turbine trip occurs 5 seconds after reaching the high-high steam generator water level setpoint and a subsequent reactor trip on turbine trip occurs such that rod motion begins 4 seconds after turbine trip. The calculated sequence of events for all the cases analyzed are provided in Tables 5.1.15-1 and 5.1.15-2, respectively, for the uniform and asymmetric steam generator tube plugging conditions.

Since the increase in feedwater flow analyzed for this event occurs in only one loop, it is logical that a feedwater flow increase to the loop with the least plugged steam generator (i.e., 8% tube plugging) for the asymmetric steam generator tube plugging configuration produces the most limiting transient results. However, due to the magnitude of the feedwater flow increase modeled in this analysis (i.e., 250% of nominal to the faulted loop in the full power cases), the effect of differences between uniform and asymmetric steam generator tube plugging modeling are insignificant for this event.

In all cases, the minimum DNBR remains above the applicable safety analysis DNBR limit and the primary and secondary-side maximum pressures are less than 110% of the design values.

Conclusions:

At initial no-load conditions, the maximum reactivity insertion rate that occurs following an excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition. Therefore, the results and conclusions for the Uncontrolled Control Rod Withdrawal from a Subcritical Condition as reported in Section 5.1.6 bound those for the Excessive Heat Removal Due to a Feedwater System Malfunction at no-load conditions.

For the cases of the excessive feedwater addition initiated from full power conditions with and without automatic rod control, the results show that all applicable Condition II acceptance criteria are met for this event.

Hence, it is concluded that the insertion of VANTAGE + Fuel and the other design changes associated with the VANTAGE + transition as described in Sections 5.1.2 and 5.1.3 are acceptable for the Excessive Heat Removal Due to a Feedwater System Malfunction event.

Table 5.1.15-1

Sequence of Events

for the

Feedwater System Malfunction Event at Full Power

24% Uniform Steam Generator Tube Plugging

Feedwater Malfunction at Full Power Event	Time of event, sec With Automatic Rod Control	Time of event, sec Without Automatic Rod Control
Feedwater Flow increases to 250% of Nominal	0.001	0.001
Peak Nuclear Power occurs	14.5	32.5
Minimum DNBR occurs	15.0	30.0
High-High Steam Generator Water Level Setpoint is reached	24.3	24.2
Turbine Trip occurs	29.3	29.2
Rod motion starts	33.3 *	33.2 *
Peak Pressurizer Pressure occurs	36.0	35.5
Feedwater Isolation occurs	36.3	36.2

* Reactor Trip occurs on Turbine Trip

Table 5.1.15-1

(continued)

Sequence of Events

for the

Feedwater System Malfunction Event at Zero Power

24% Uniform Steam Generator Tube Plugging

Feedwater Malfunction at Zero Power Event	Time of event, sec With Automatic Rod Control	Without Automatic Rod Control
Feedwater Flow increases to Nominal full power flow rate	5.0	5.0
Peak Reactivity Insertion Rate occurs	15.0	14.0
Peak Pressurizer Pressure occurs	0.0	37.0
High-High Steam Generator Water Level Setpoint is reached	46.8	46.8

* To determine the Maximum Reactivity Insertion Rate for the HZP Feedwater Malfunction cases, no Turbine Trip or Reactor Trip is modeled after reaching the High-High Steam Generator Water Level Turbine Trip setpoint.

Table 5.1.15-2

Sequence of Events

for the

Feedwater System Malfunction Event at Full Power

Asymmetric Steam Generator Tube Plugging

Feedwater Malfunction at Full Power Event	Time of event, sec With Automatic Rod Control	Time of event, sec Without Automatic Rod Control
Feedwater Flow increases to 250% of Nominal	0.001	0.001
Peak Nuclear Power occurs	14.0	34.5
Minimum DNBR occurs	14.5	32.0
High-High Steam Generator Water Level Setpoint is reached	24.4	26.2
Turbine Trip occurs	31.4	31.2
Rod motion starts	35.4 *	35.2 *
Peak Pressurizer Pressure occurs	38.0	37.0
Feedwater Isolation occurs	38.4	38.2

* Reactor Trip occurs on Turbine Trip

Table 5.1.15-2

(continued)

Sequence of Events

for the

Feedwater System Malfuction Event at Zero Power

Asymmetric Steam Generator Tube Plugging

Feedwater Malfunction at Zero Power Event	Time of event, sec With Automatic Rod Control	Without Automatic Rod Control
Feedwater Flow increases to Nominal full power flow rate	5.0	5.0
Peak Reactivity Insertion Rate occurs	14.5	13.5
Peak Pressurizer Pressure occurs	0.0	39.0
High-High Steam Generator Water Level Setpoint is reached *	46.7	46.7

* To determine the Maximum Reactivity Insertion Rate for the HZP Feedwater Malfunction cases, no Turbine Trip or Reactor Trip is modeled after reaching the High-High Steam Generator Water Level Turbine Trip setpoint.

Figure 5.1.15-1

Feedwater System Malfunction from Hot Full Power

With Automatic Rod Control

Asymmetric Steam Generator Tube Plugging

Nuclear Power versus Time

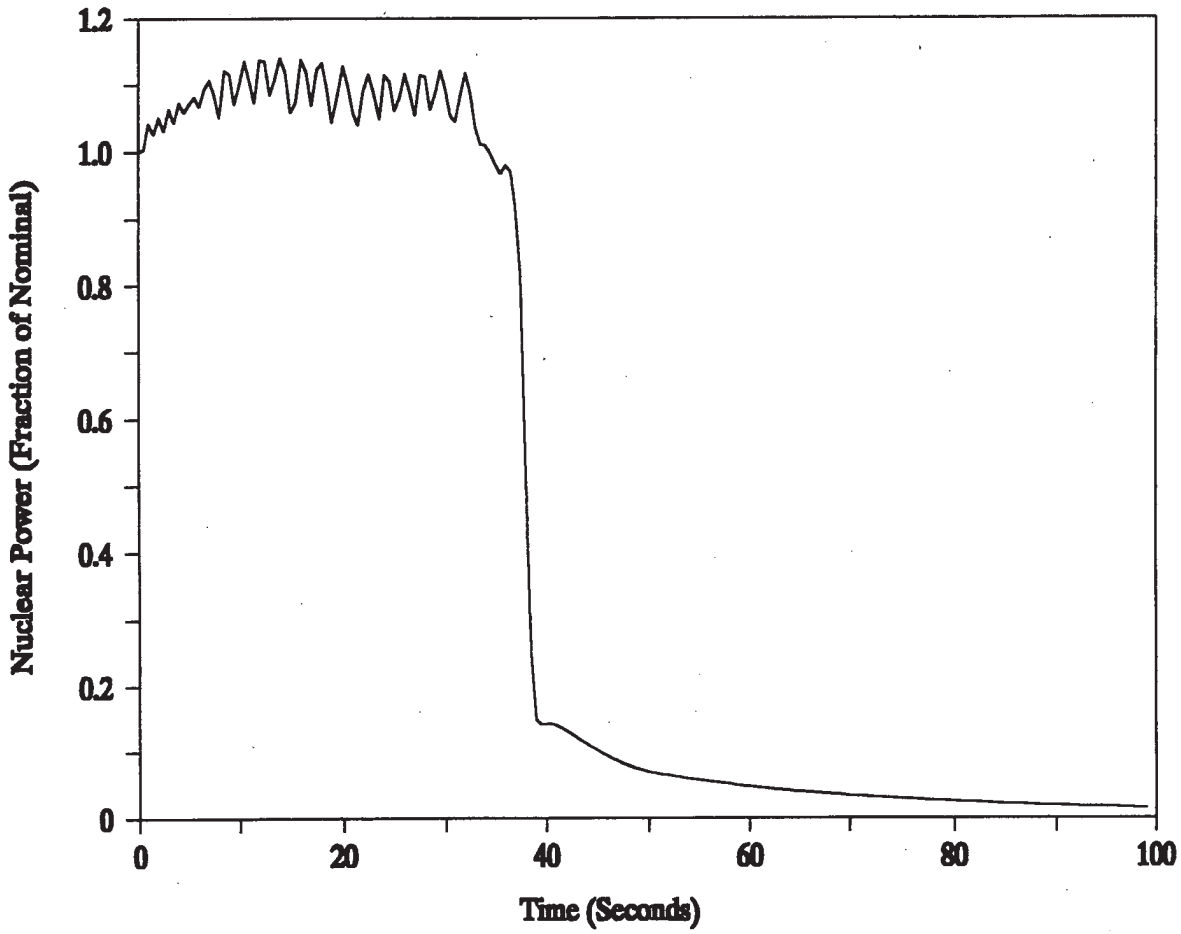


Figure 5.1.15-2

Feedwater System Malfunction from Hot Full Power

With Automatic Rod Control

Asymmetric Steam Generator Tube Plugging

Reactor Vessel Average Temperature versus Time

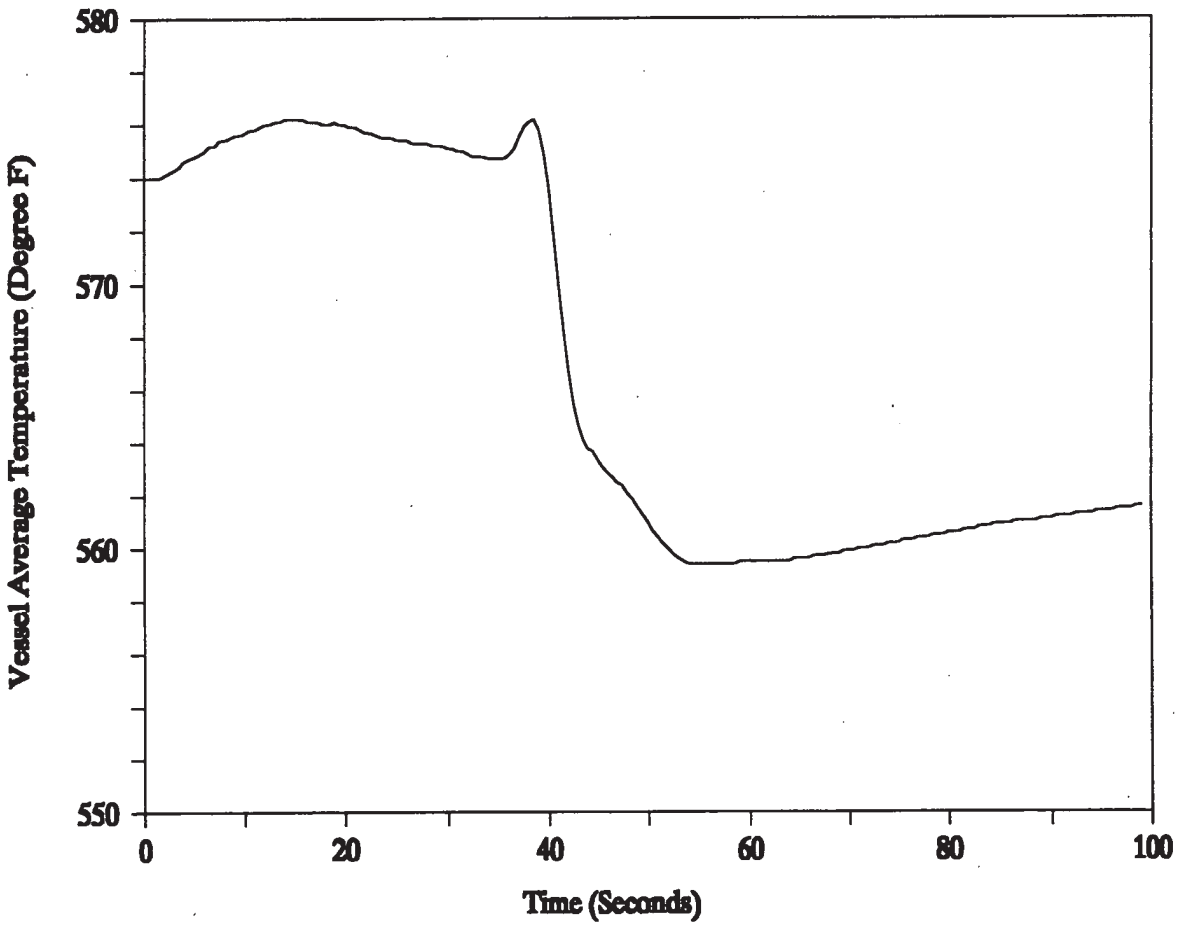


Figure 5.1.15-3

Feedwater System Malfunction from Hot Full Power

With Automatic Rod Control

Asymmetric Steam Generator Tube Plugging

Affected Loop Delta-T versus Time

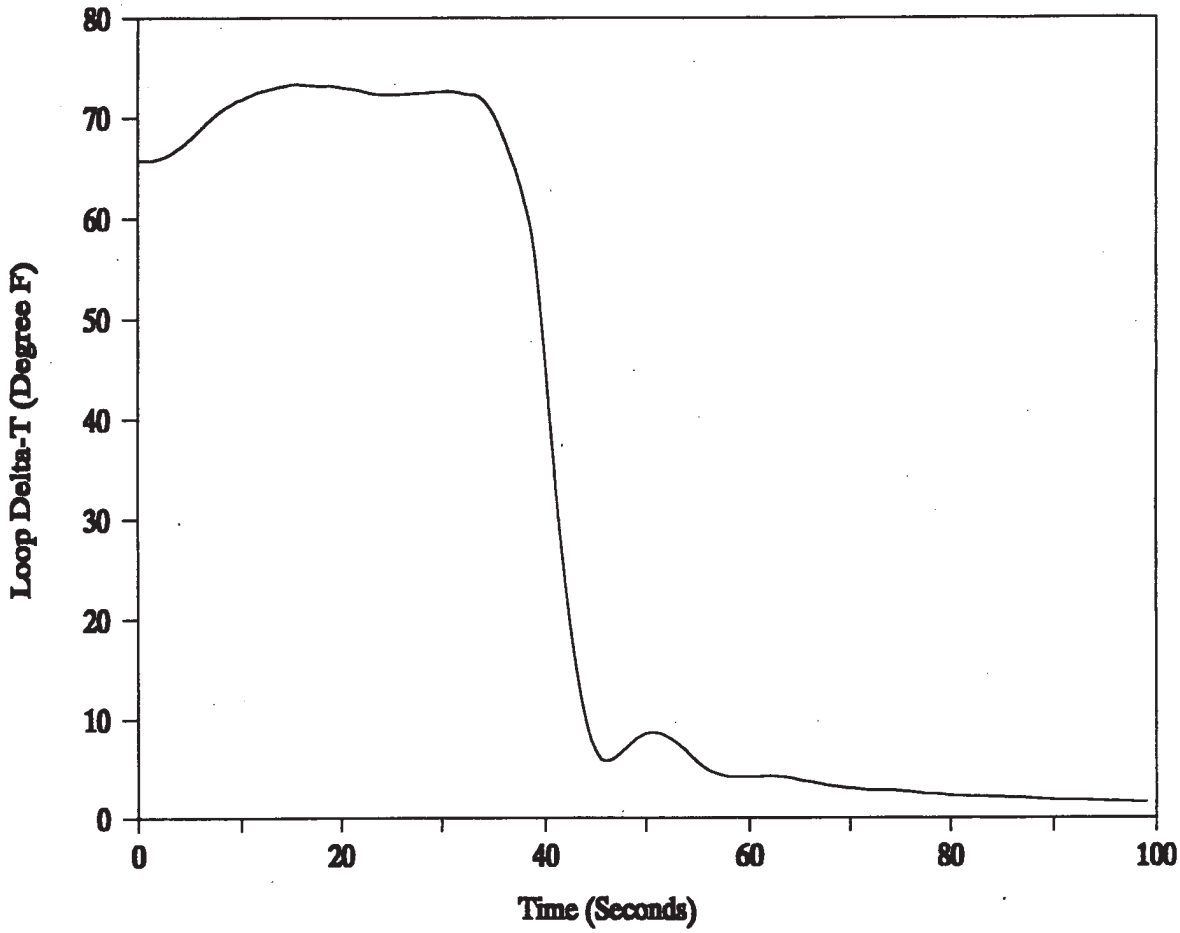


Figure 5.1.15-4

Feedwater System Malfunction from Hot Full Power

With Automatic Rod Control

Asymmetric Steam Generator Tube Plugging

Pressurizer Pressure versus Time

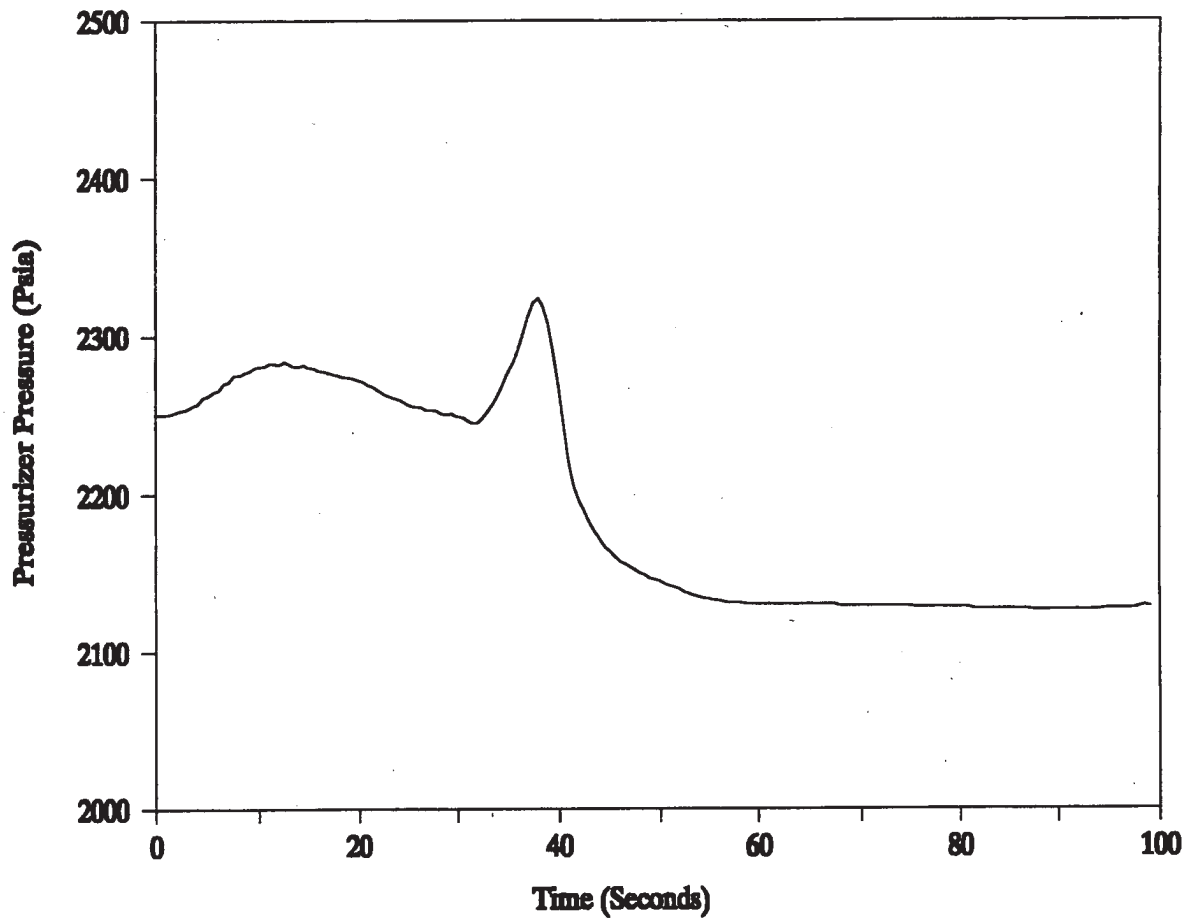


Figure 5.1.15-5

Feedwater System Malfunction from Hot Full Power

With Automatic Rod Control

Asymmetric Steam Generator Tube Plugging

Steam Generator Pressure versus Time

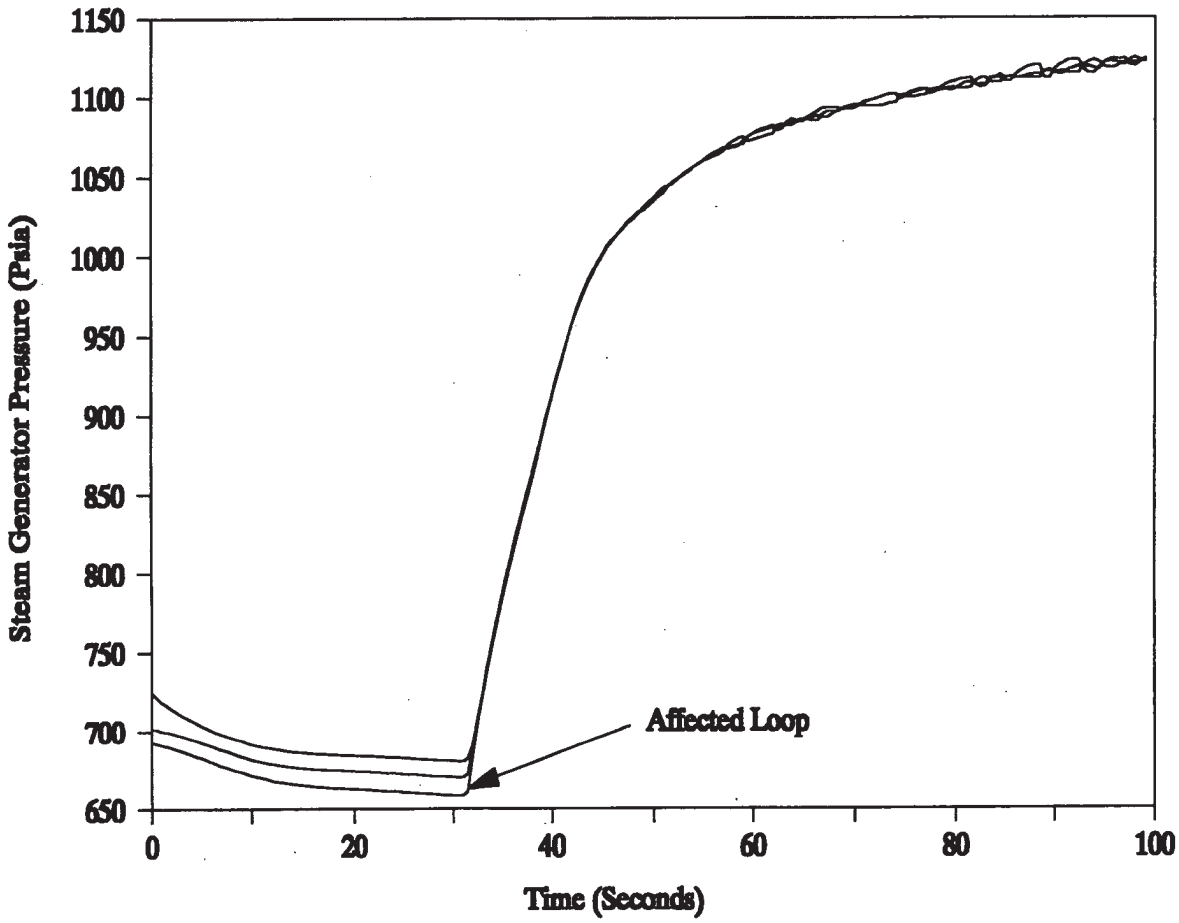


Figure 5.1.15-6

Feedwater System Malfunction from Hot Full Power

With Automatic Rod Control

Asymmetric Steam Generator Tube Plugging

Steam Generator Mass versus Time

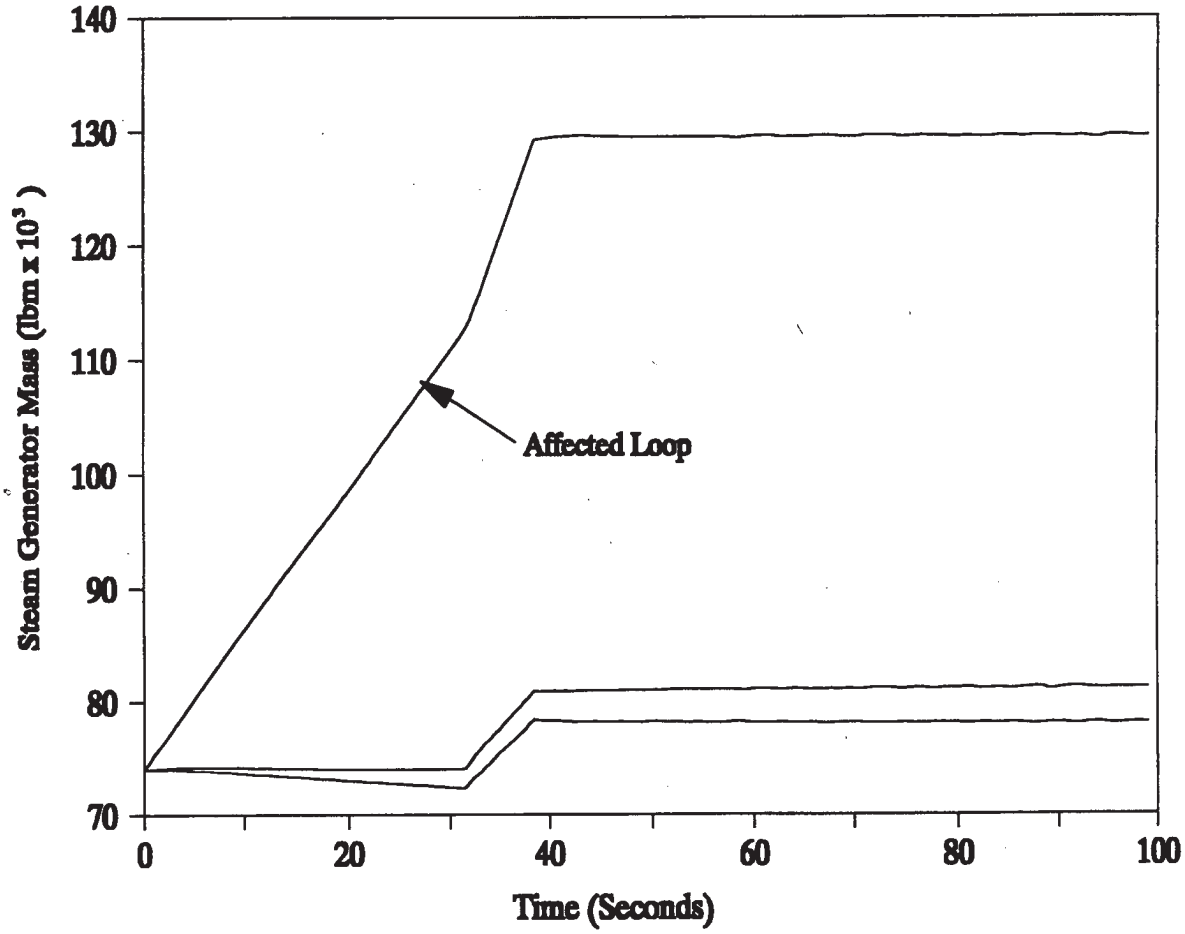


Figure 5.1.15-7

Feedwater System Malfunction from Hot Full Power

With Automatic Rod Control

Asymmetric Steam Generator Tube Plugging

DNBR versus Time

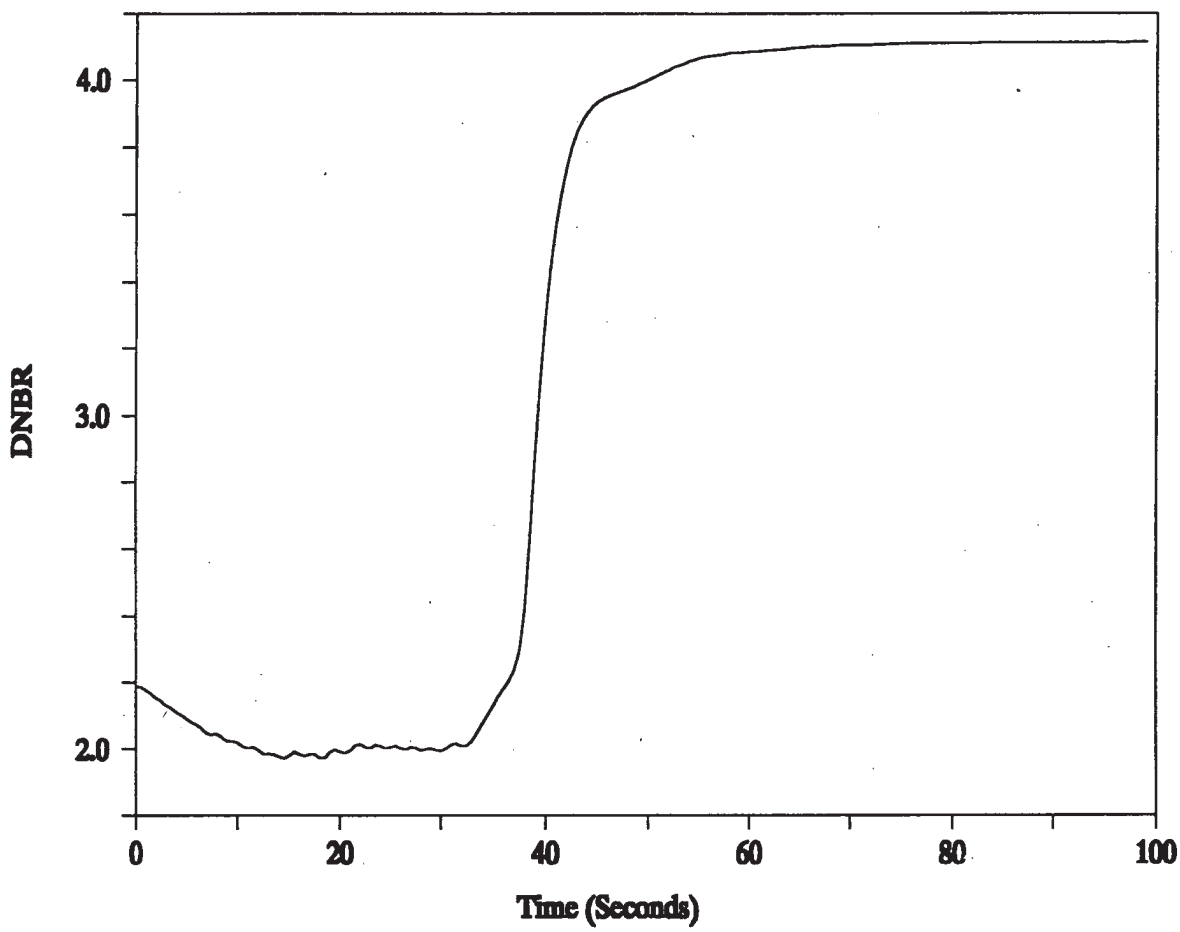


Figure 5.1.15-8

Feedwater System Malfunction from Hot Full Power

Manual Rod Control

Asymmetric Steam Generator Tube Plugging

Nuclear Power versus Time

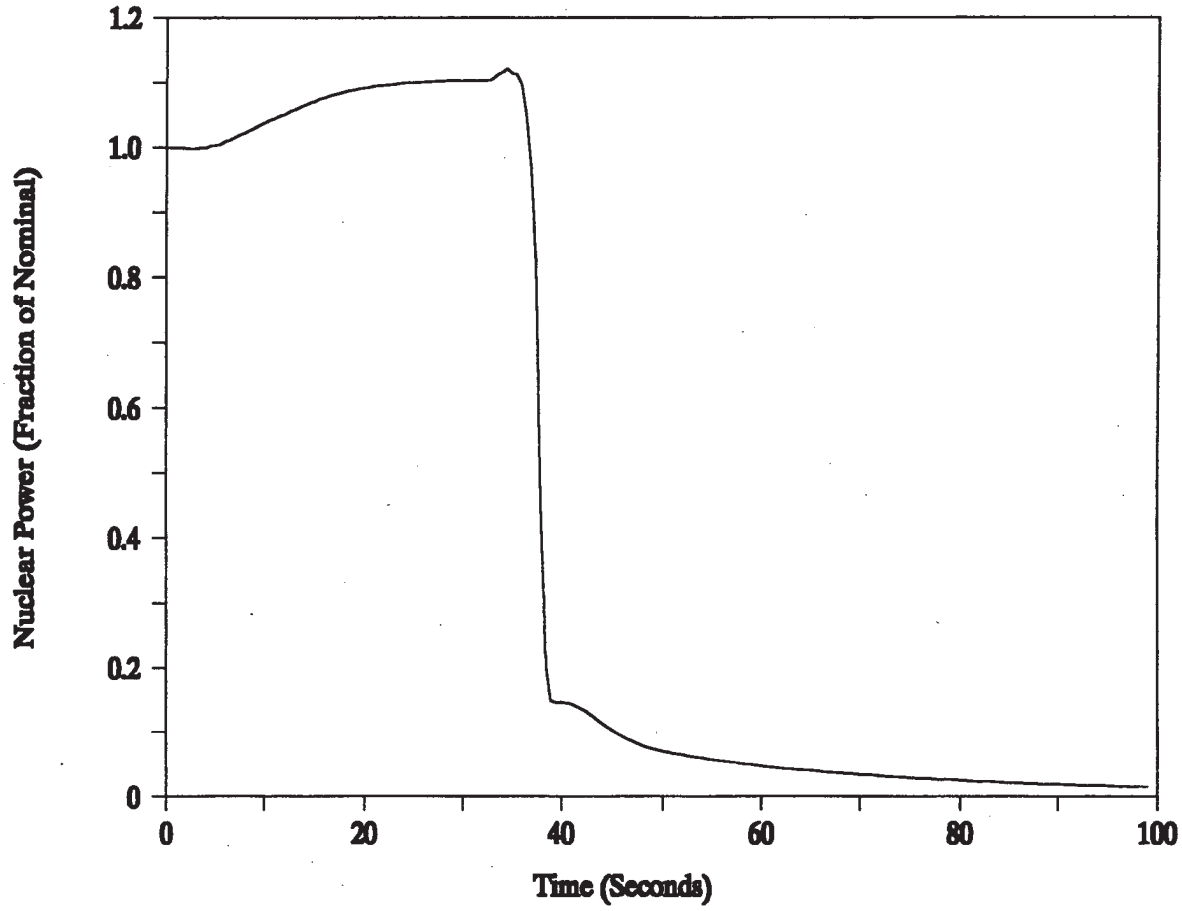


Figure 5.1.15-9

Feedwater System Malfunction from Hot Full Power

Manual Rod Control

Asymmetric Steam Generator Tube Plugging

Reactor Vessel Average Temperature versus Time

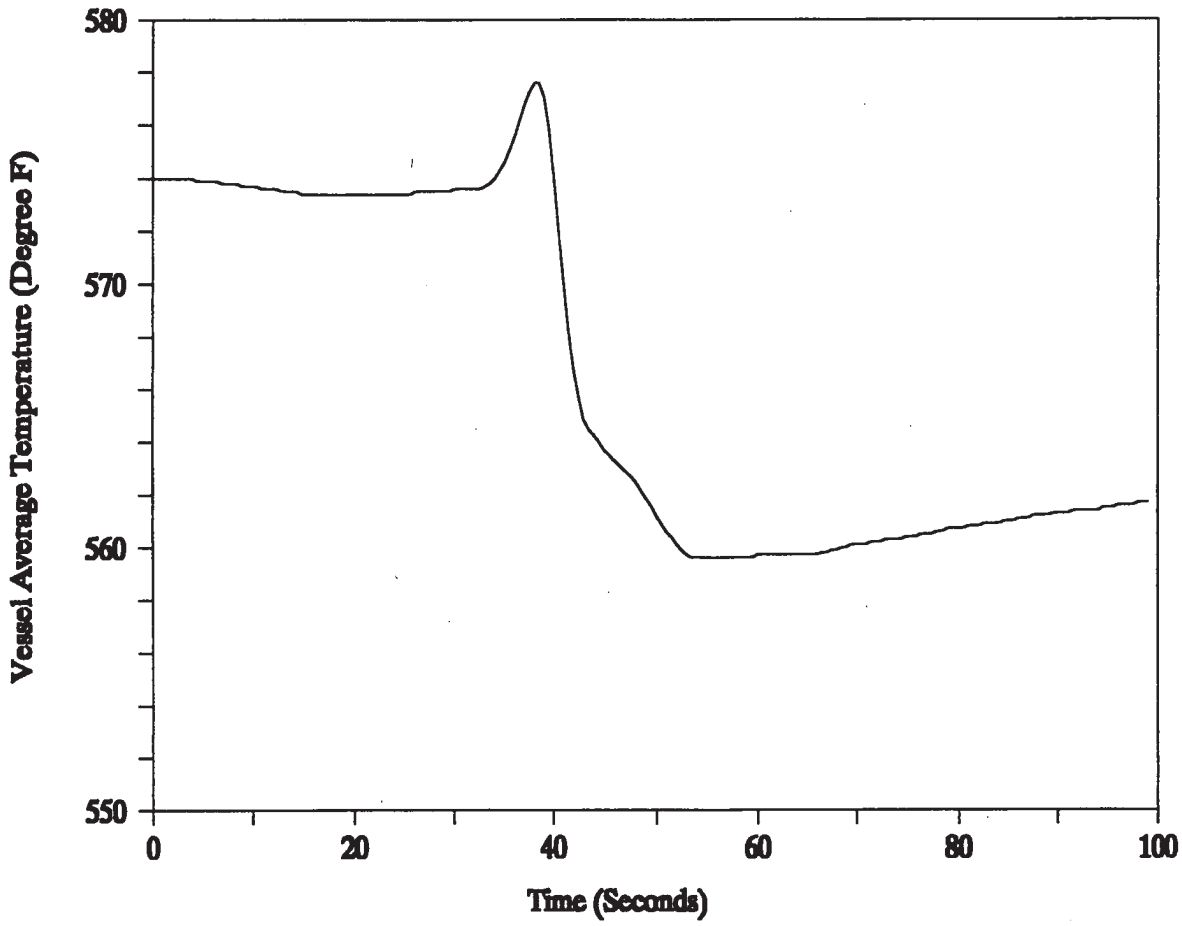


Figure 5.1.15-10

Feedwater System Malfunction from Hot Full Power

Manual Rod Control

Asymmetric Steam Generator Tube Plugging

Affected Loop Delta-T versus Time

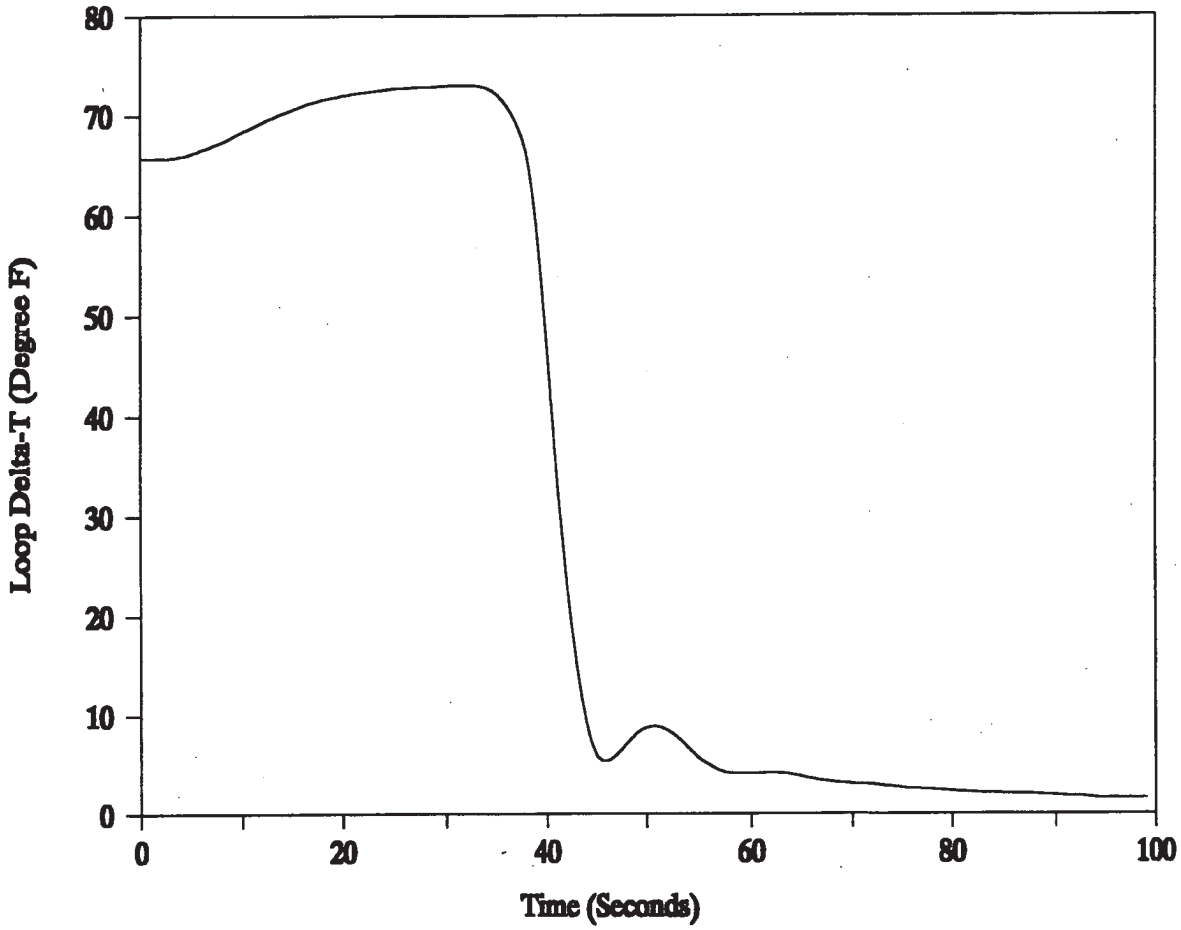


Figure 5.1.15-11

Feedwater System Malfunction from Hot Full Power

Manual Rod Control

Asymmetric Steam Generator Tube Plugging

Pressurizer Pressure versus Time

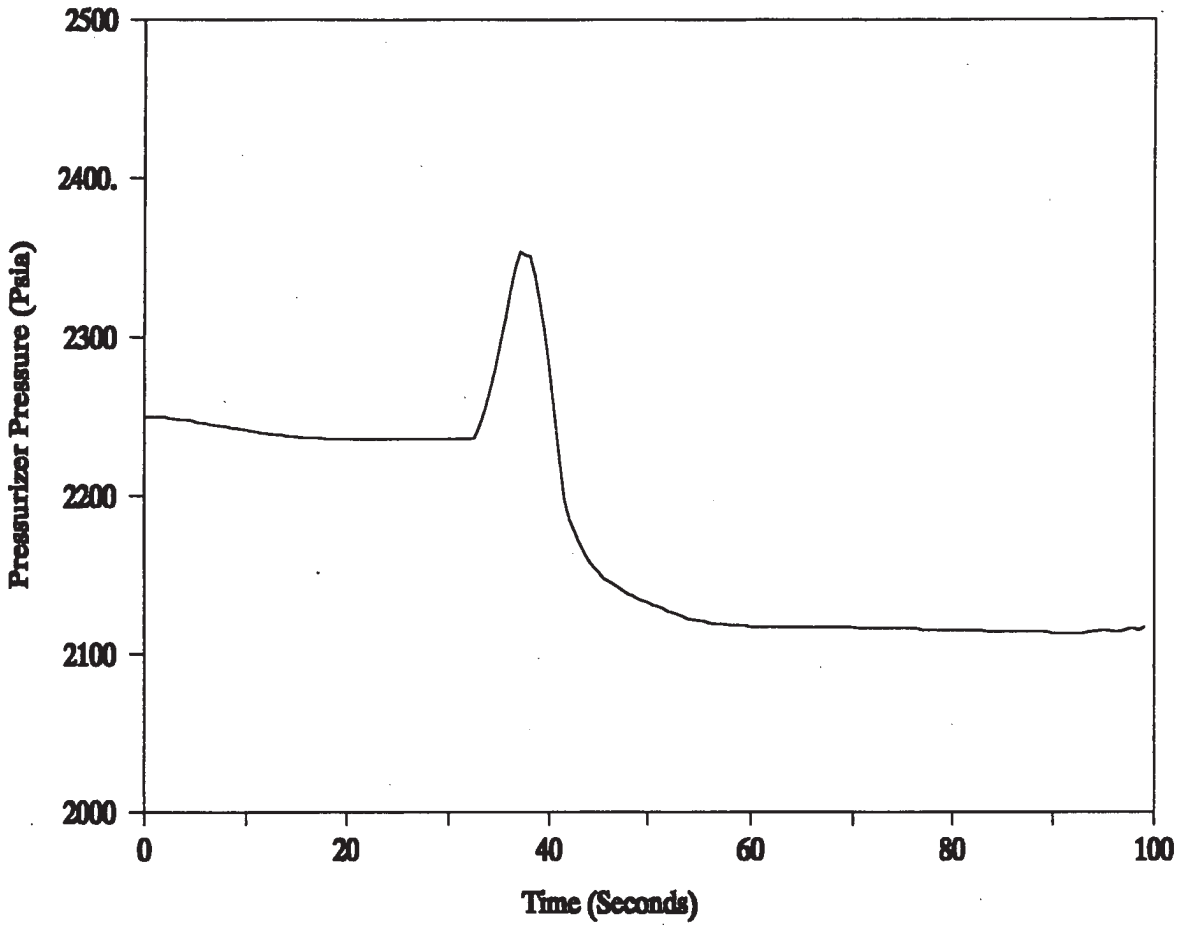


Figure 5.1.15-12

Feedwater System Malfunction from Hot Full Power

Manual Rod Control

Asymmetric Steam Generator Tube Plugging

Steam Generator Pressure versus Time

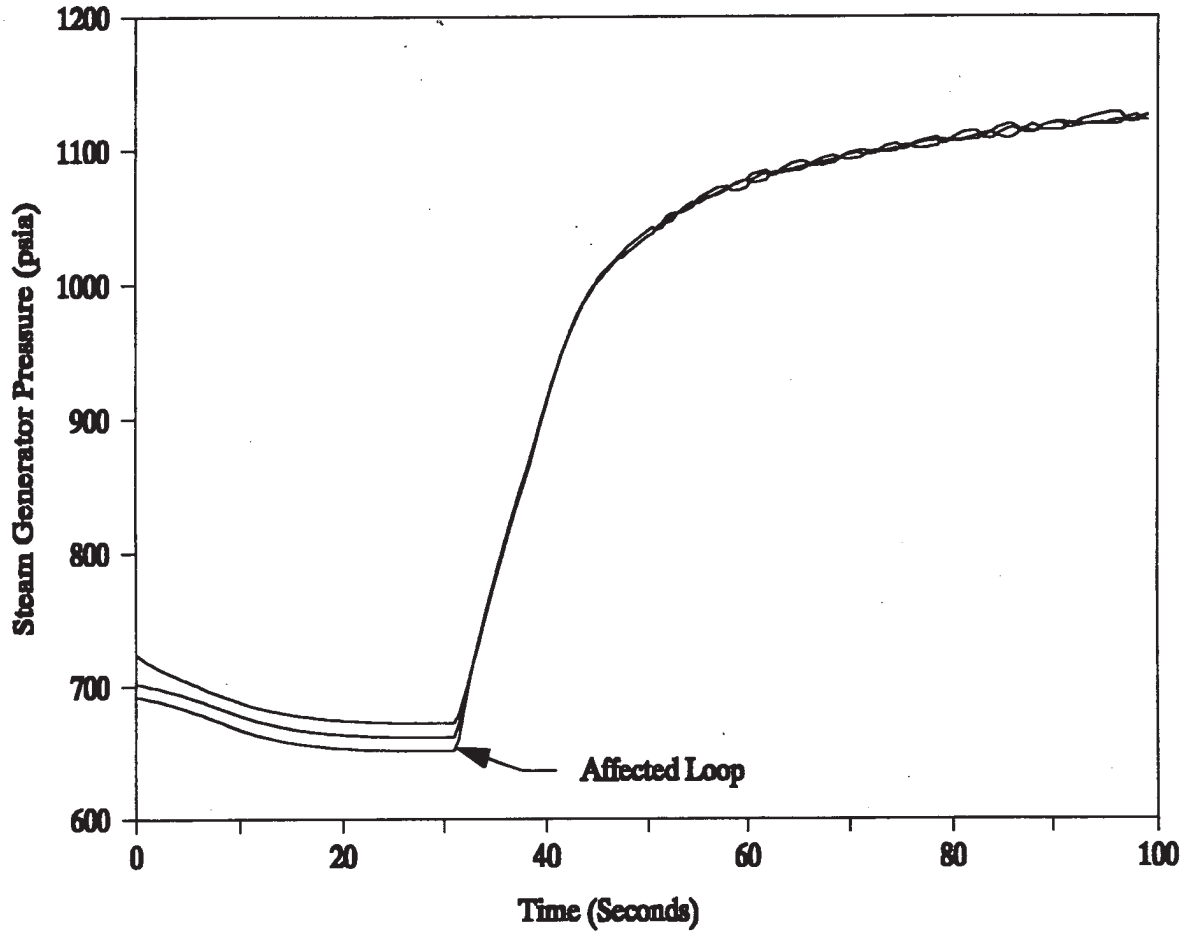


Figure 5.1.15-13

Feedwater System Malfunction from Hot Full Power

Manual Rod Control

Asymmetric Steam Generator Tube Plugging

Steam Generator Mass versus Time

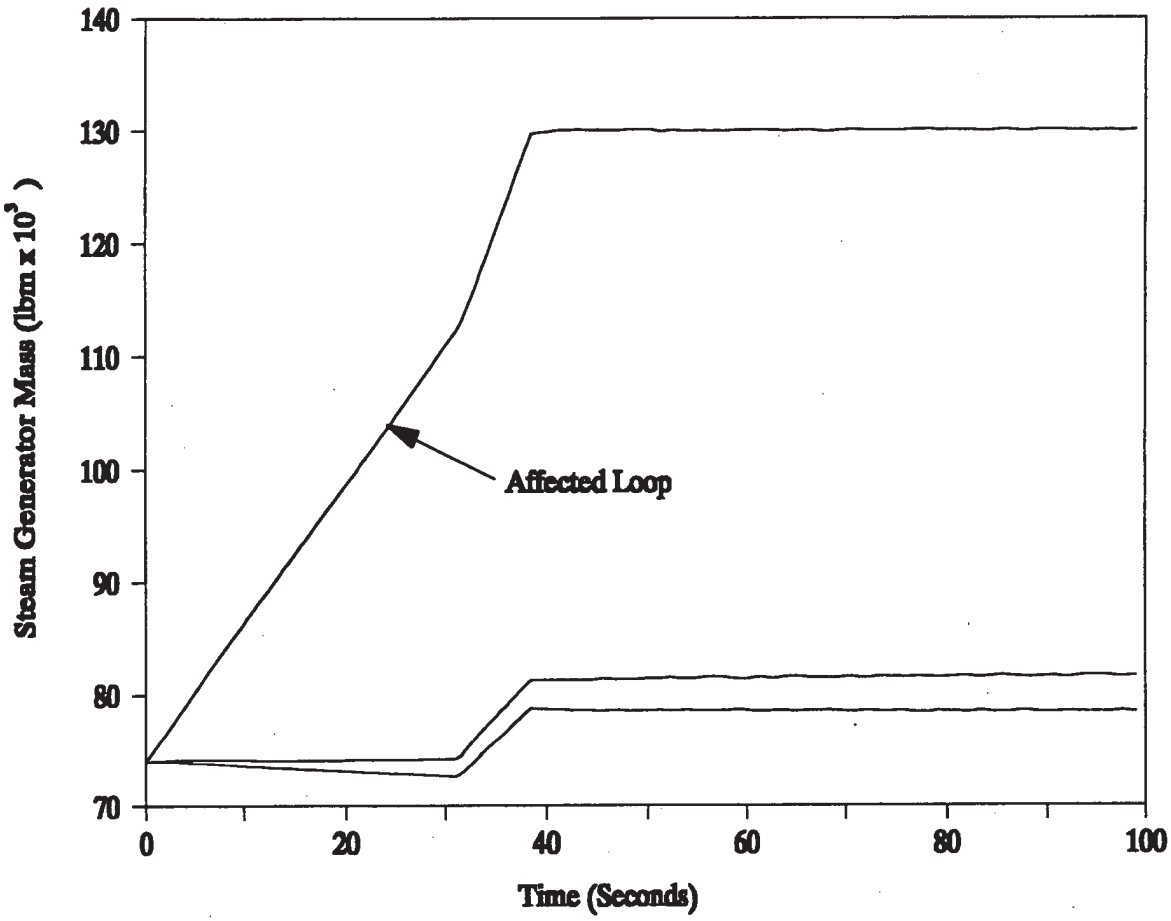


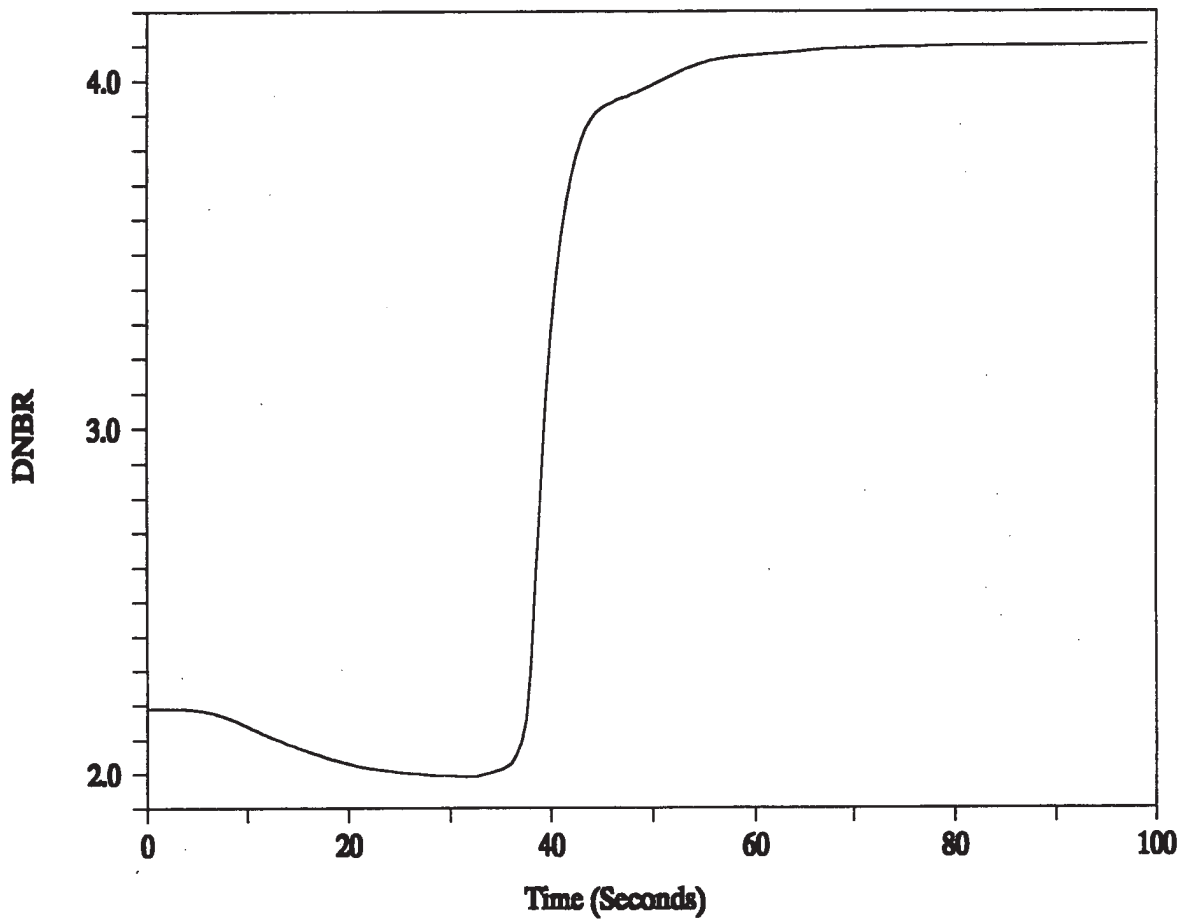
Figure 5.1.15-14

Feedwater System Malfunction from Hot Full Power

Manual Rod Control

Asymmetric Steam Generator Tube Plugging

DNBR versus Time



5.1.16 Excessive Load Increase Incident

Introduction:

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals: i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided that blocks the opening of the valves unless a large turbine load decrease or turbine trip has occurred.

The possible consequence of this accident (assuming no protective functions) is departure from nucleate boiling (DNB) with subsequent fuel damage. Note that the accident, as presently analyzed, is characterized by an approach to protection setpoints without actually reaching the setpoints.

The excessive load increase is classified as a Condition II fault as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressuized Water Reactor Plants." A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious condition, i.e., a Condition III or IV category event.

The applicable safety analysis licensing basis acceptance criteria for the Condition II Excessive Load Increase event for Indian Point Unit 3 are:

a) Pressures in the reactor coolant and main steam systems should be maintained below 110% of the design values (i.e., 2748.5 psia and 1208.5 psia, respectively),

b) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit, and,

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

Method of Analysis and Assumptions:

The excessive load increase event is analyzed to show that: 1) the integrity of the core is maintained without actuation of the reactor protection system as the DNBR remains above the safety analysis limit value; 2) the peak RCS and secondary system pressures remain below 110% of the design limit; and 3) the pressurizer does not fill. Of these, the primary concern is DNB and assuring that the DNBR limit is met.

The excessive load increase transients are analyzed with the LOFTRAN computer program (see Section 5.1.5). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator relief and safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

The analyses were performed considering the transition from VANTAGE 5 to VANTAGE + fuel at a nominal core power of 3025 MWt and the other design changes associated with the VANTAGE + transition as discussed in Section 5.1.2 and 5.1.3.

Four cases are analyzed to demonstrate the plant behavior following a 10% step load increase from rated load. These cases are as follows:

1) Reactor control in manual with beginning-of-life minimum moderator reactivity feedback.

2) Reactor control in manual with end-of-life maximum moderator reactivity feedback.

3) Reactor control in automatic with beginning-of-life minimum moderator reactivity feedback.

4) Reactor control in automatic with end-of-life maximum moderator reactivity feedback.

For the beginning-of-life minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore the least inherent transient response. For the end-of-life maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening (equivalent to 120% turbine load) is assumed, and all cases are analyzed without credit being taken for pressurizer heaters.

This accident is analyzed with the RTDP as discussed in Section 5.1.3. Initial reactor power, RCS pressure and temperature are assumed to be at their nominal values.

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

Automatic rod control is modeled in the analysis to ensure that the worst case is presented. The automatic rod control system is not required or modeled to provide reactor protection.

Results:

Figures 5.1.16-1 through 5.1.16-10 illustrate the transient conditions with the reactor in the manual control mode. As expected, for the beginning-of-life minimum moderator feedback case, there is a slight power increase, and the core average coolant temperature and pressure show a decrease. This results in a DNBR which increases above its initial value throughout the transient. For the end-of-life maximum moderator feedback case in manual control, there is a larger increase in reactor power due to the moderator feedback. A slight reduction in DNBR from its initial value is experienced but the minimum DNBR remains well above the applicable safety analysis limit DNBR value of 1.54.

Figures 5.1.16-11 through 5.1.16-20 illustrate the transient conditions assuming the reactor is in the automatic control mode. Both the beginning-of-life minimum and end-of-life maximum moderator feedback cases show that the core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains well

above the limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for any of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow. The calculated sequence of events for the excessive load increase incident are shown in Table 5.1.16-1.

Since DNB does not occur at any time during the excessive load increase transients, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

Conclusions:

The analysis presented above shows that for a 10% step load increase, the DNBR remains above the safety analysis limit DNBR value, thereby precluding fuel or clad damage. Peak reactor coolant and main steam pressures do not challenge the pressure limits. The plant reaches a stabilized condition rapidly, following the load increase.

Table 5.1.16-1

Sequence of Events
for the
Excessive Load Increase Event

Excessive Load Increase Event	Time of event (sec)			
	With Automatic Rod Control		Without Automatic Rod Control	
Event	BOL	EOL	BOL	EOL
10 percent step load increase occurs	0.01	0.01	0.01	0.01
Peak pressurizer pressure occurs	8.5	13.1	0.4	0.4
Minimum DNBR occurs	146.1	101.1	5.0	56.6
Peak nuclear power occurs	3.0	21.4	221.1	46.4

Figure 5.1.16-1

Nuclear Power Transient for Excessive Load Increase

Without Automatic Rod Control,

Minimum Reactivity Feedback

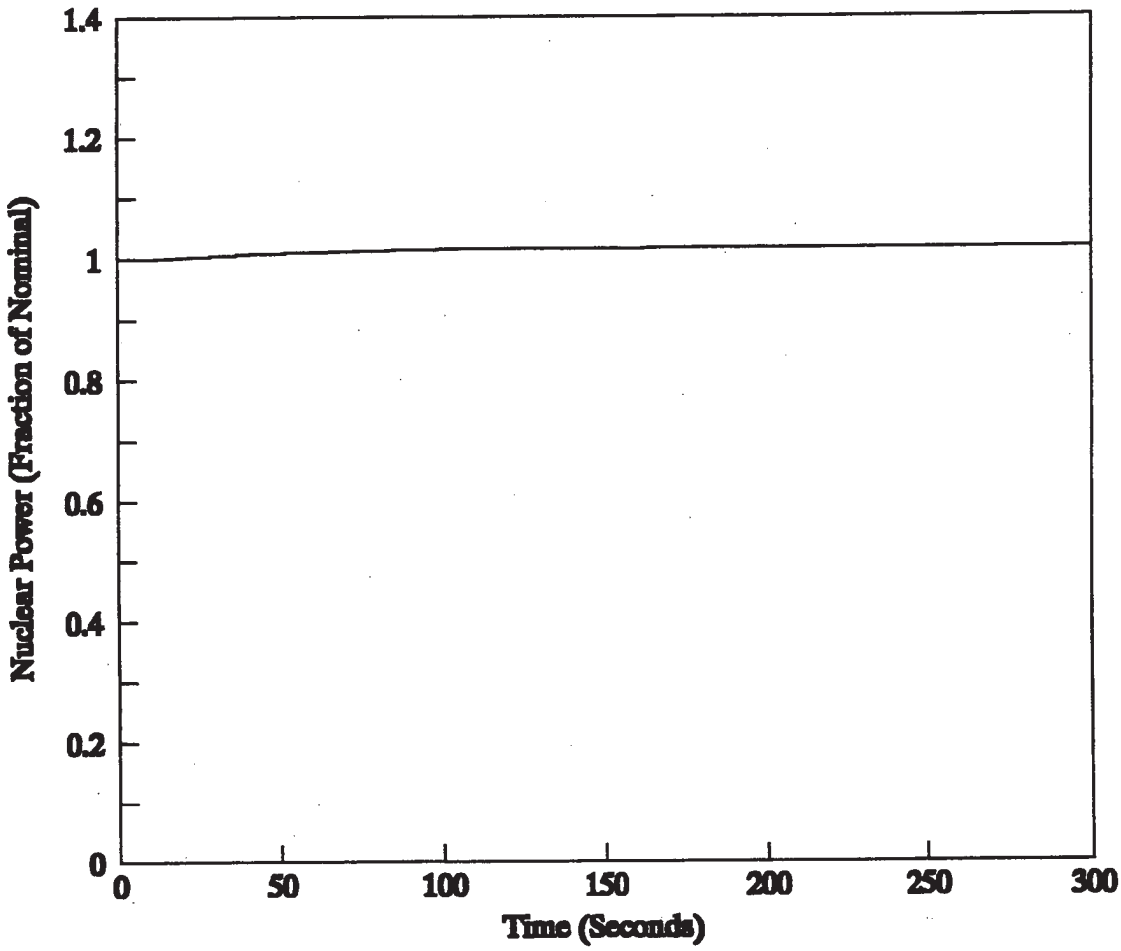


Figure 5.1.16-2

Pressurizer Pressure for Excessive Load Increase

Without Automatic Rod Control,

Minimum Reactivity Feedback

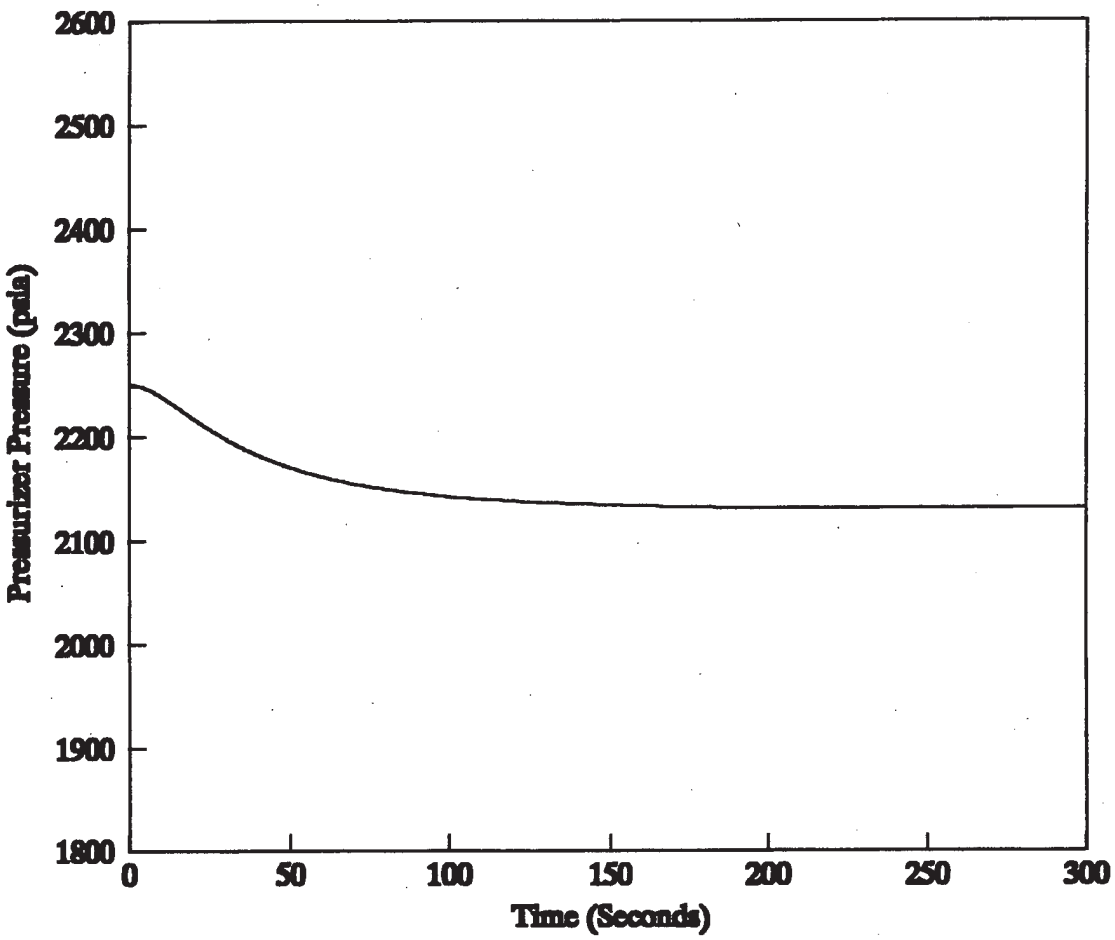


Figure 5.1.16-3

Pressurizer Water Volume Transient for Excessive Load Increase

Without Automatic Rod Control,

Minimum Reactivity Feedback

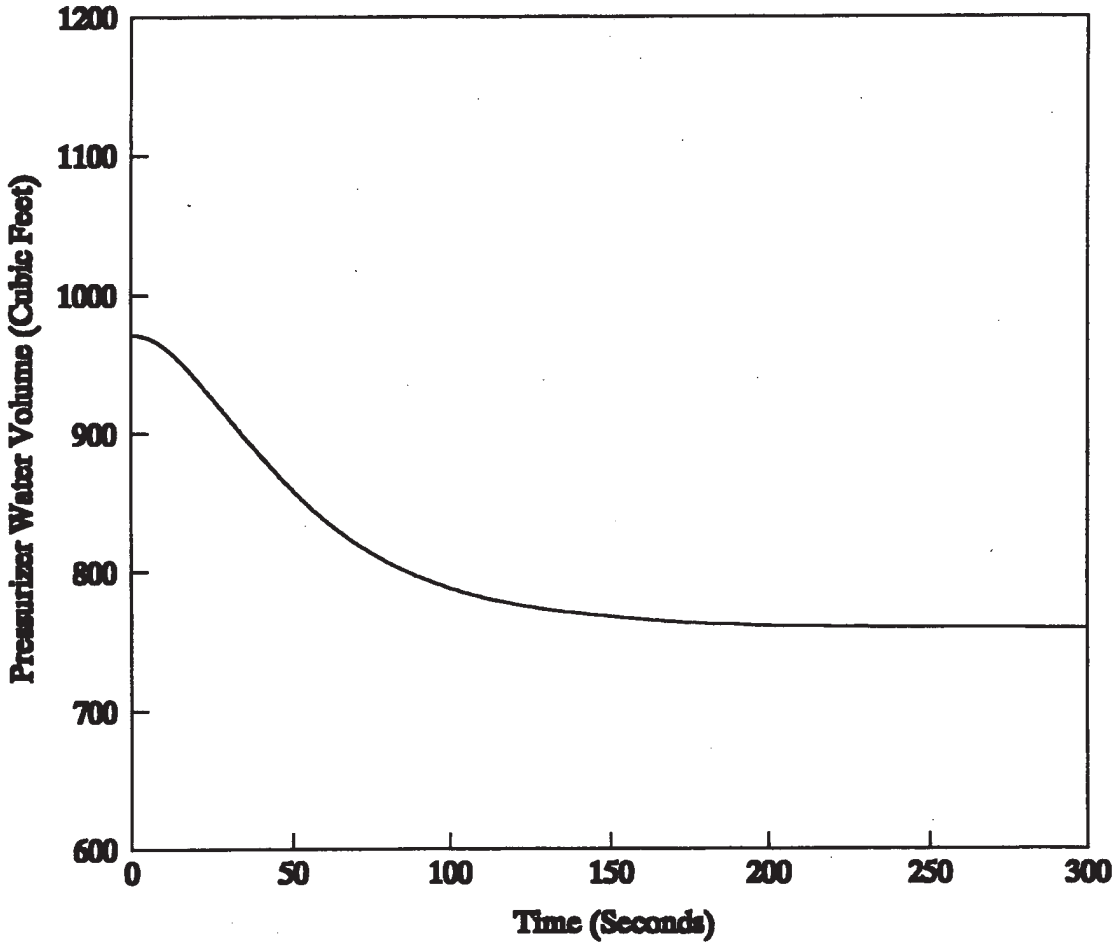


Figure 5.1.16-4

Core Average Temperature Transient for Excessive Load Increase

Without Automatic Rod Control,

Minimum Reactivity Feedback

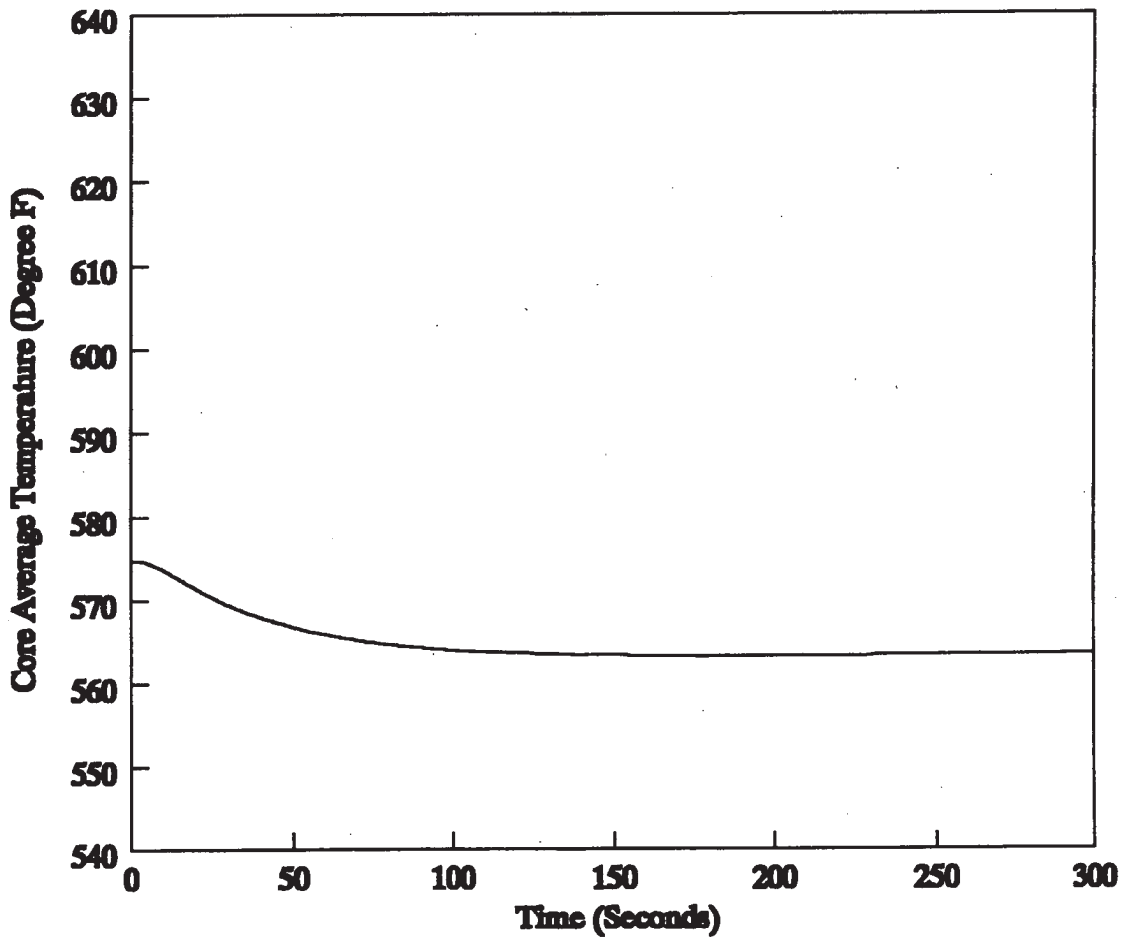


Figure 5.1.16-5

DNBR versus Time for Excessive Load Increase,

Without Automatic Rod Control,

Minimum Reactivity Feedback

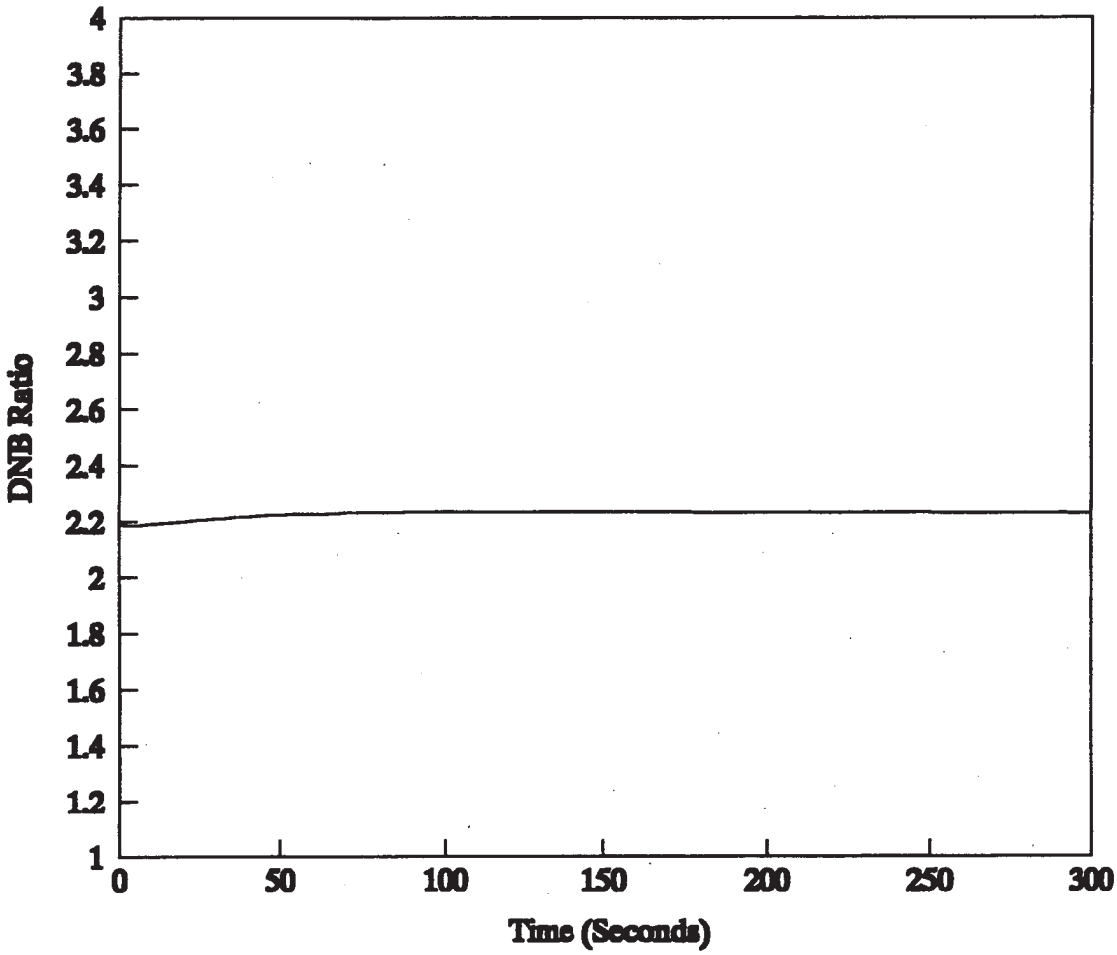


Figure 5.1.16-6

Nuclear Power Transient for Excessive Load Increase

Without Automatic Rod Control,

Maximum Reactivity Feedback

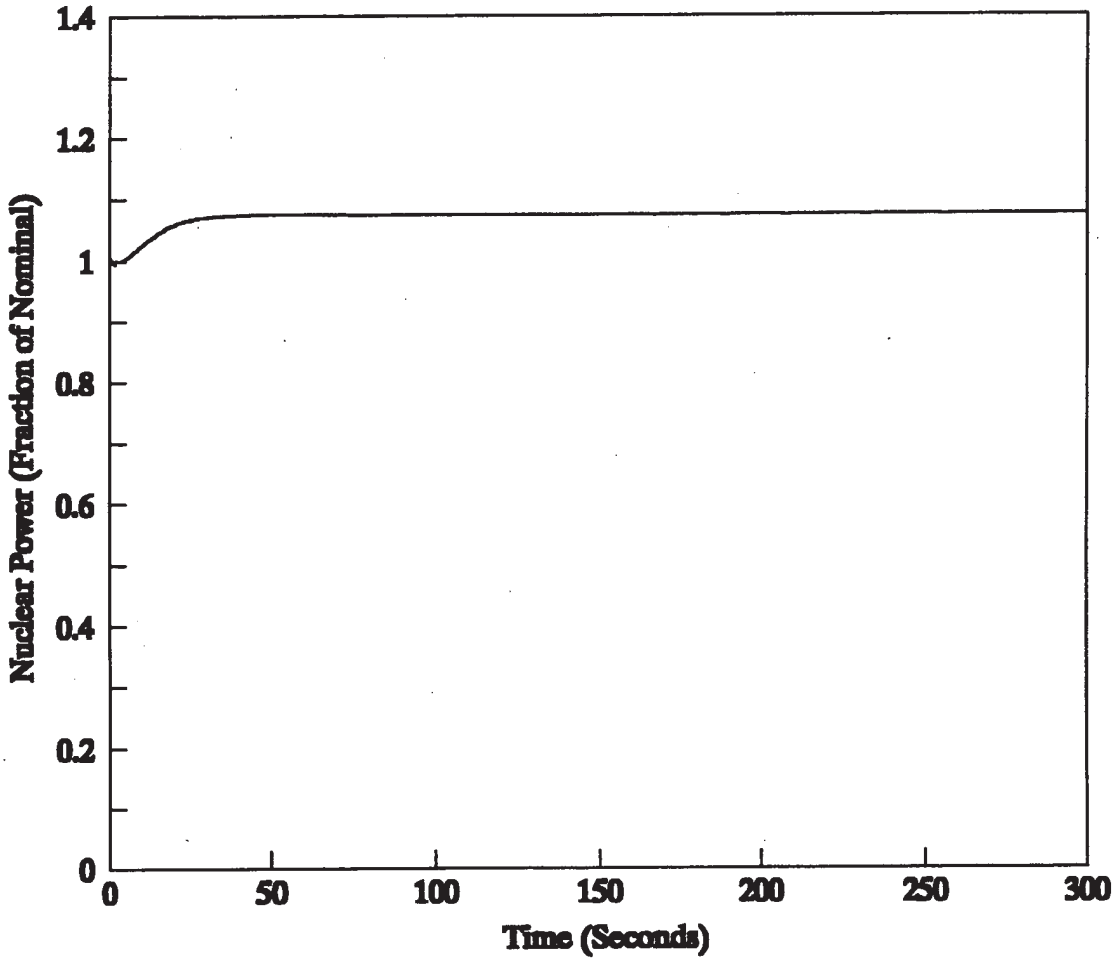


Figure 5.1.16-7

Pressurizer Pressure for Excessive Load Increase

Without Automatic Rod Control,

Maximum Reactivity Feedback

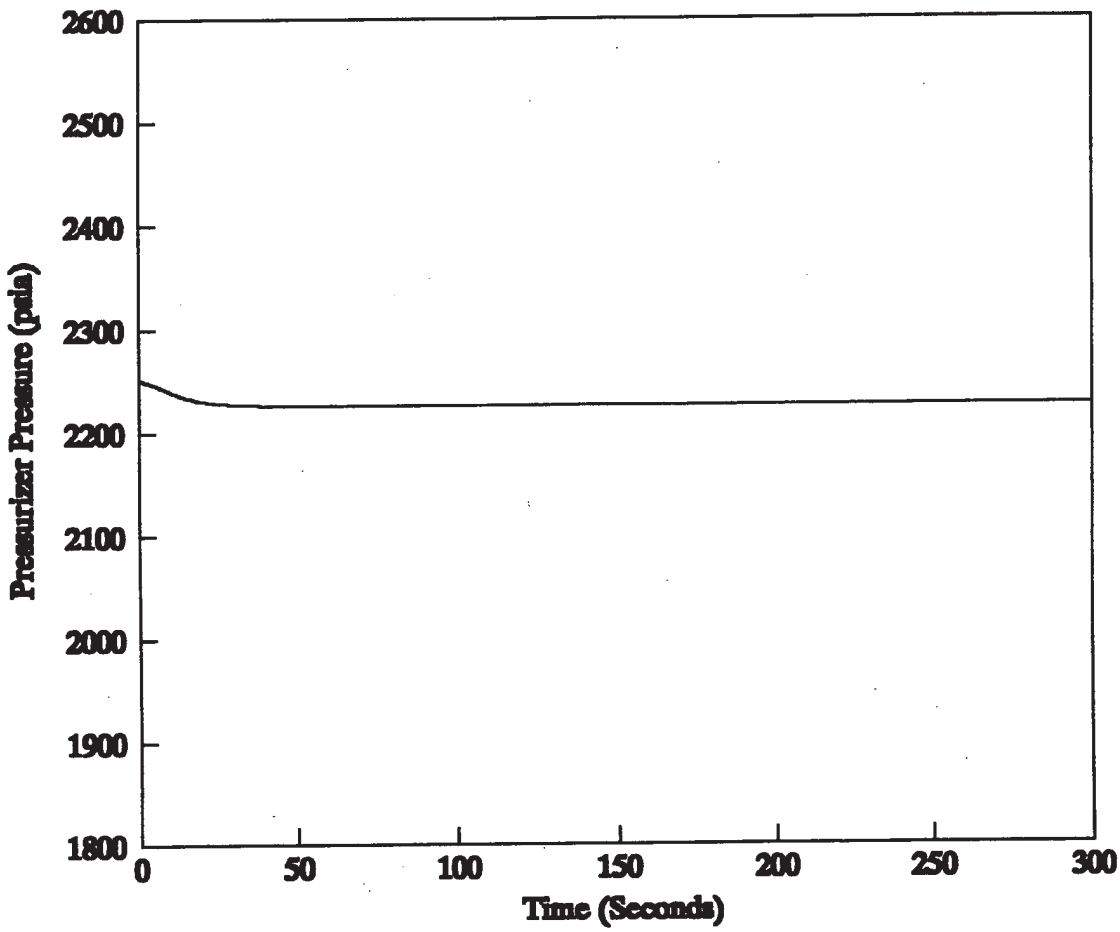


Figure 5.1.16-8

Pressurizer Water Volume Transient for Excessive Load Increase

Without Automatic Rod Control,

Maximum Reactivity Feedback

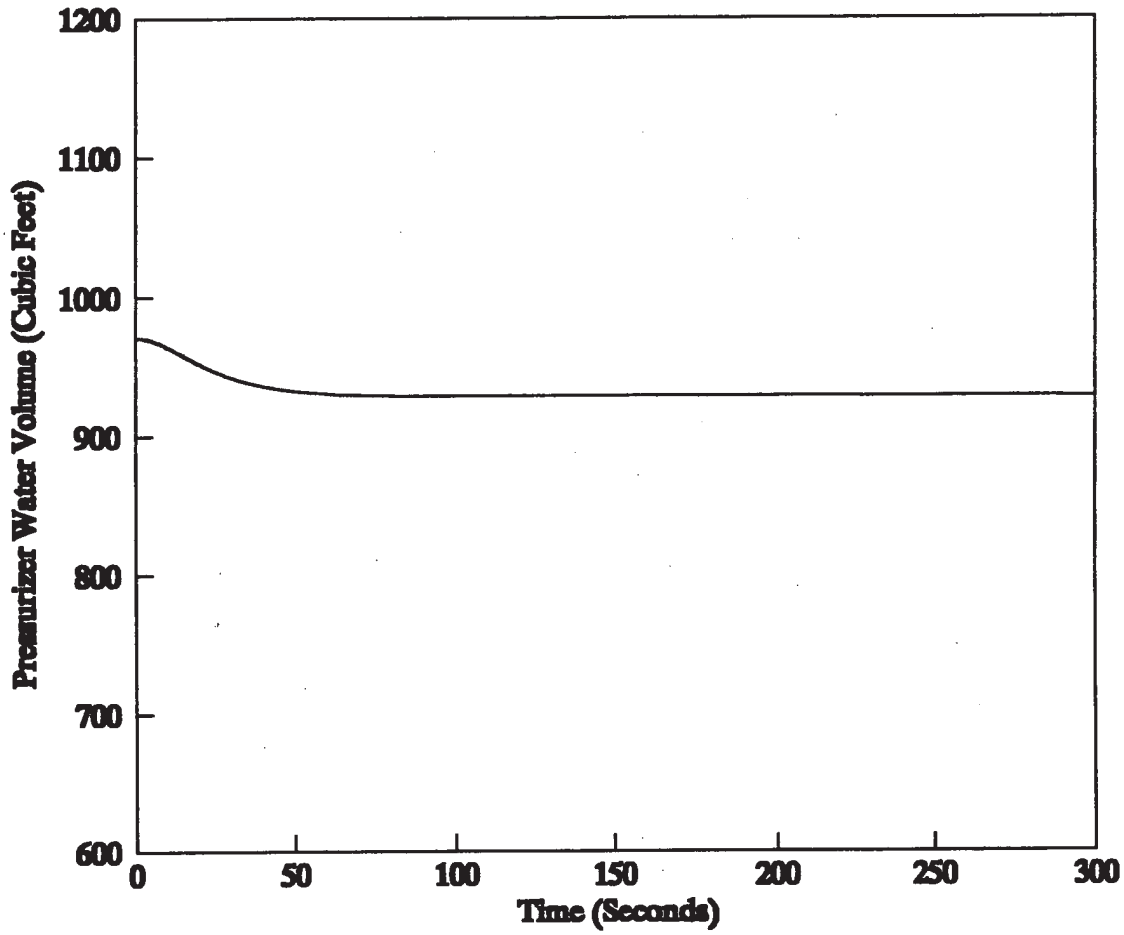


Figure 5.1.16-9

Core Average Temperature Transient for Excessive Load Increase

Without Automatic Rod Control,

Maximum Reactivity Feedback

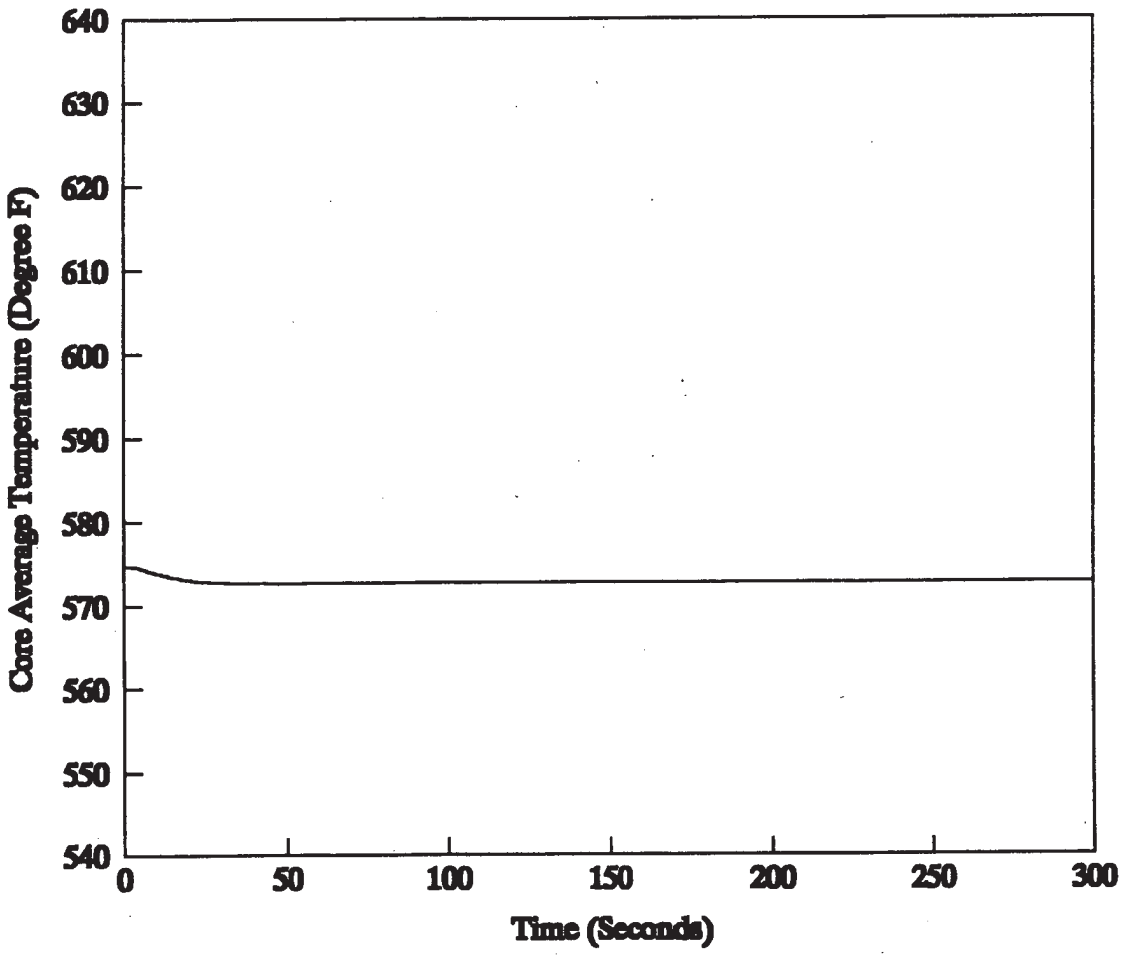


Figure 5.1.16-10

DNBR versus Time for Excessive Load Increase,

Without Automatic Rod Control,

Maximum Reactivity Feedback

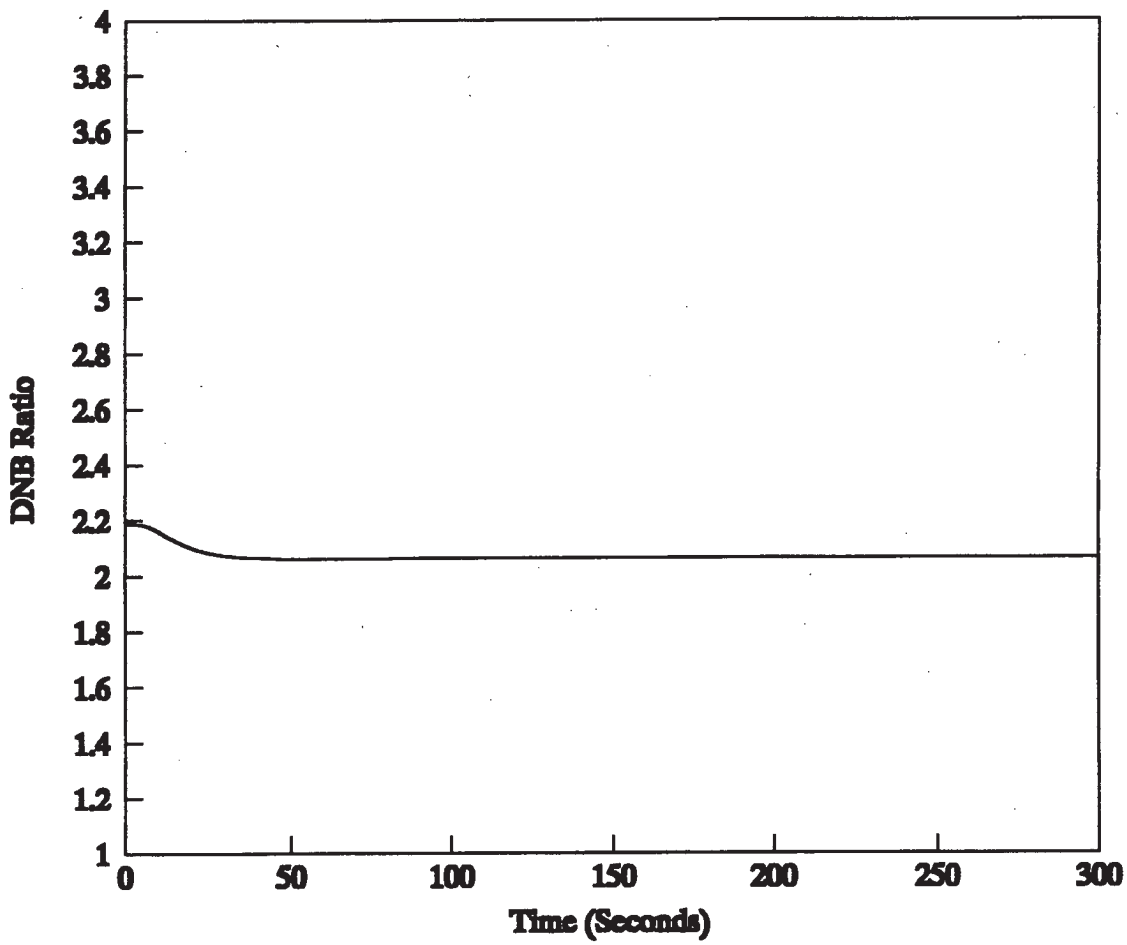


Figure 5.1.16-11

Nuclear Power Transient for Excessive Load Increase

With Automatic Rod Control,

Minimum Reactivity Feedback

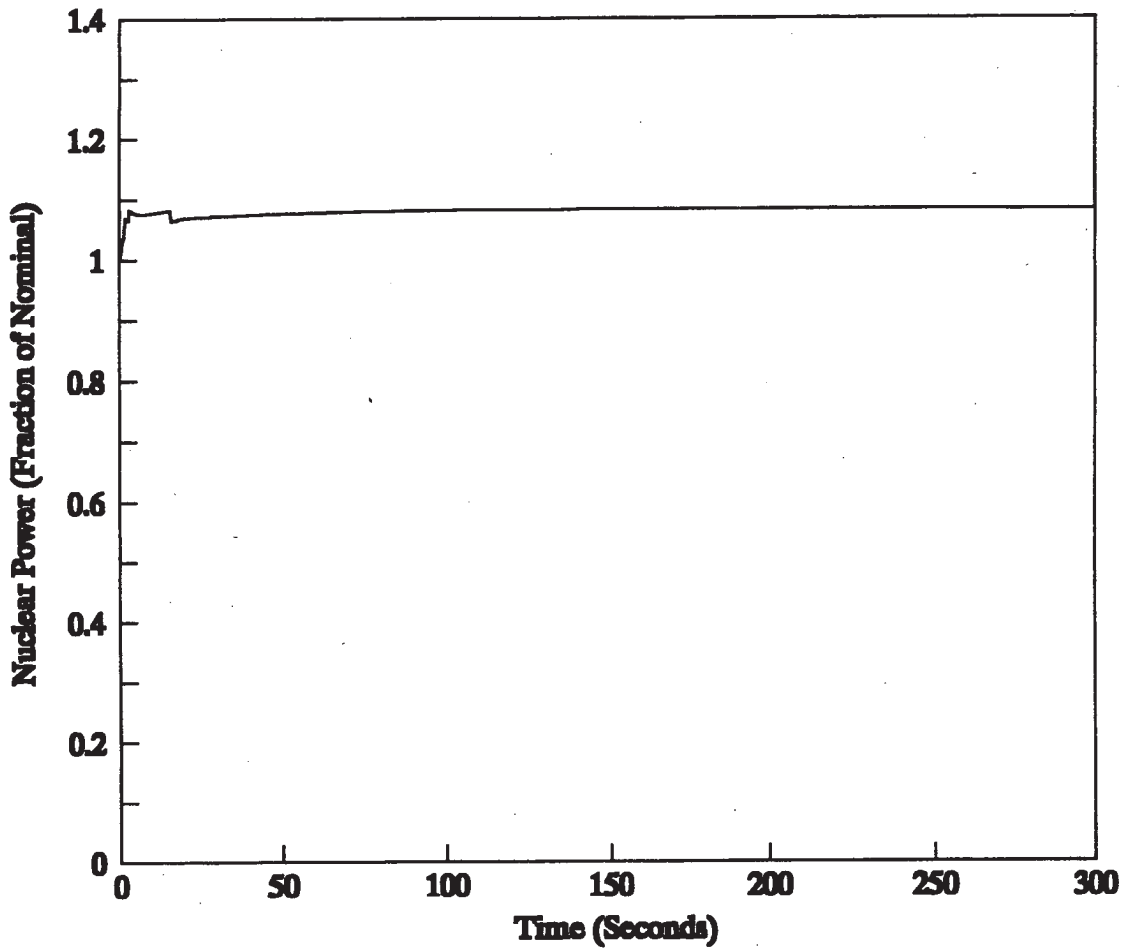


Figure 5.1.16-12

Pressurizer Pressure for Excessive Load Increase

With Automatic Rod Control,

Minimum Reactivity Feedback

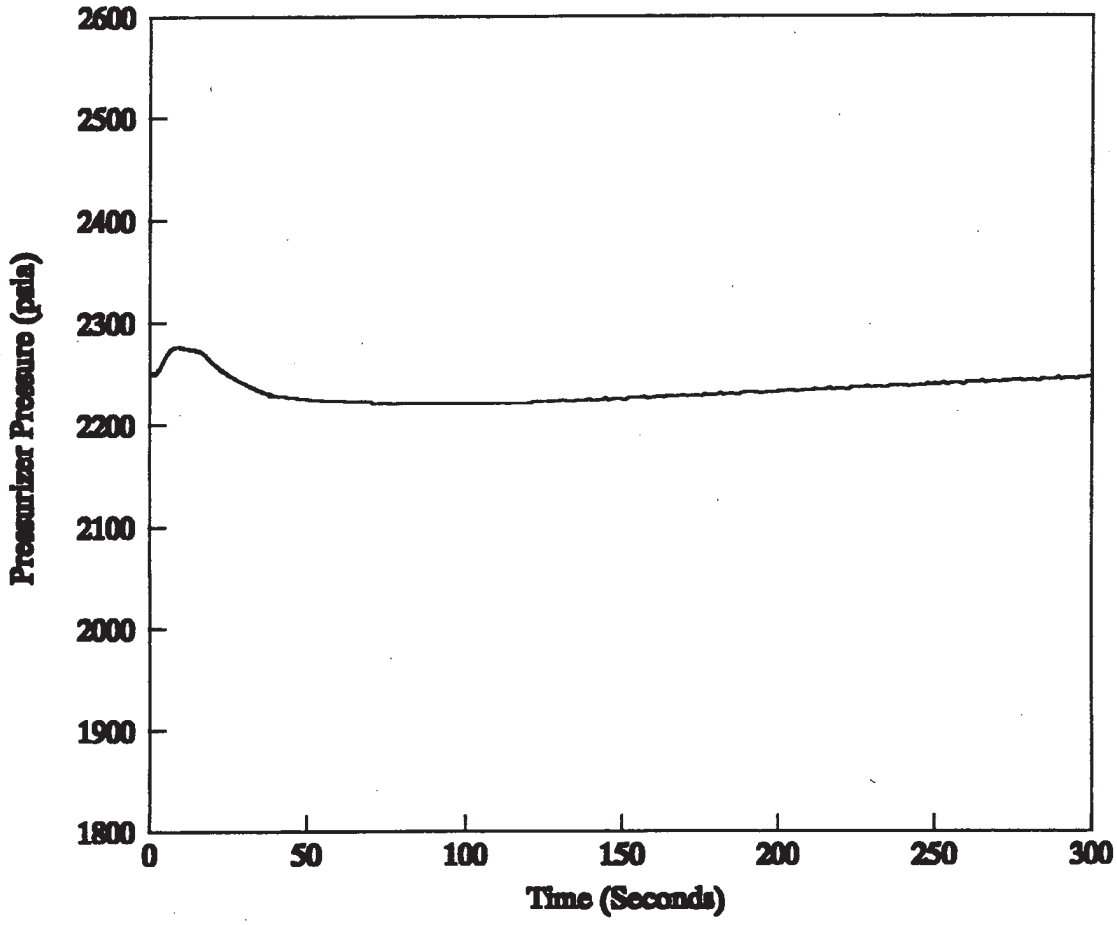


Figure 5.1.16-13

Pressurizer Water Volume Transient for Excessive Load Increase

With Automatic Rod Control,

Minimum Reactivity Feedback

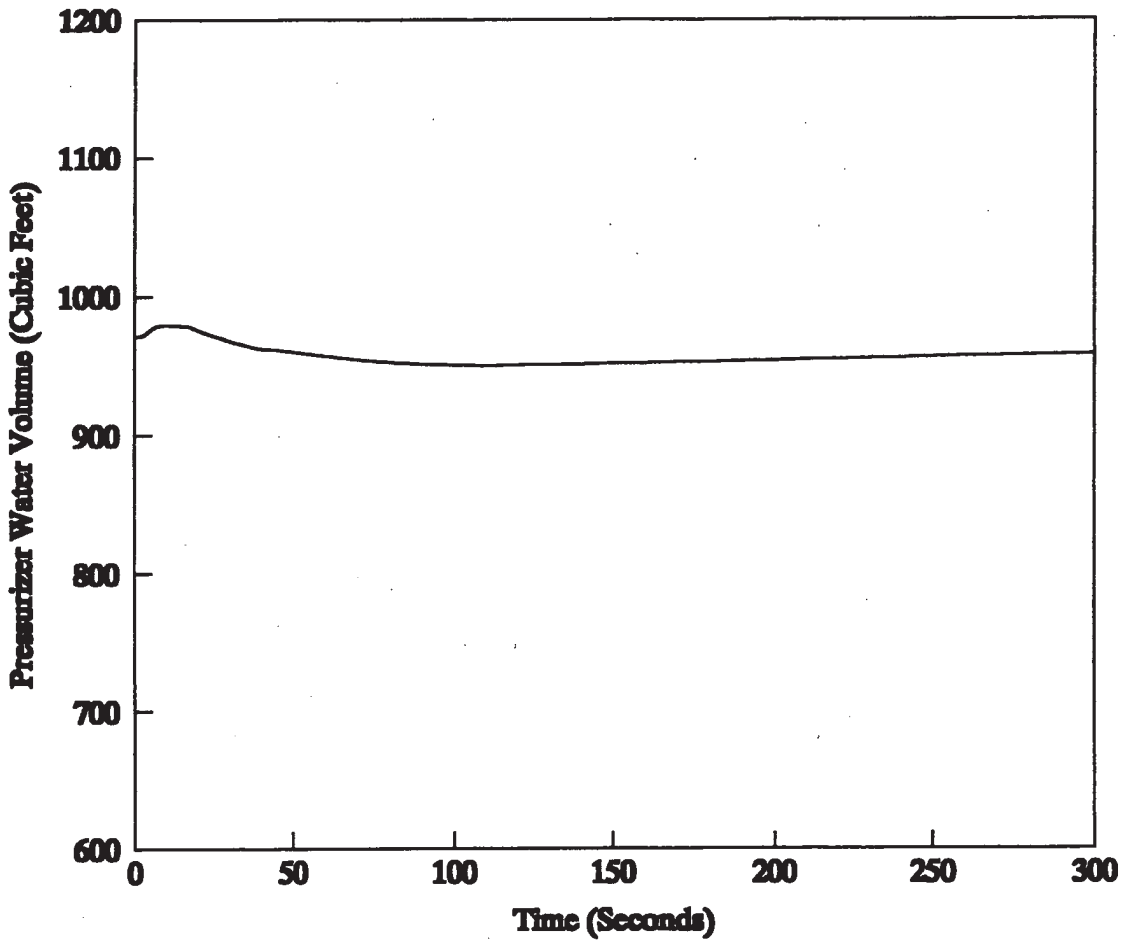


Figure 5.1.16-14

Core Average Temperature Transient for Excessive Load Increase

With Automatic Rod Control,

Minimum Reactivity Feedback

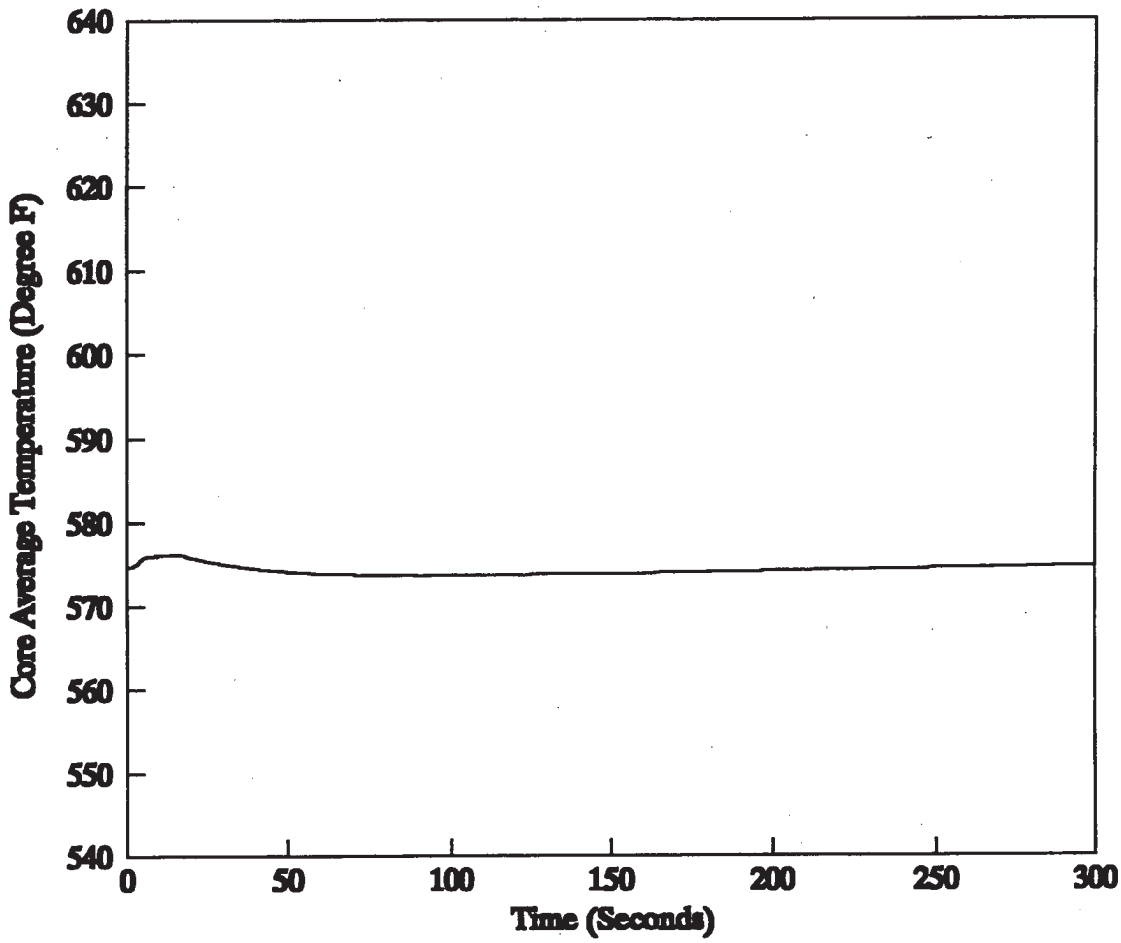


Figure 5.1.16-15

DNBR versus Time for Excessive Load Increase,

With Automatic Rod Control,

Minimum Reactivity Feedback

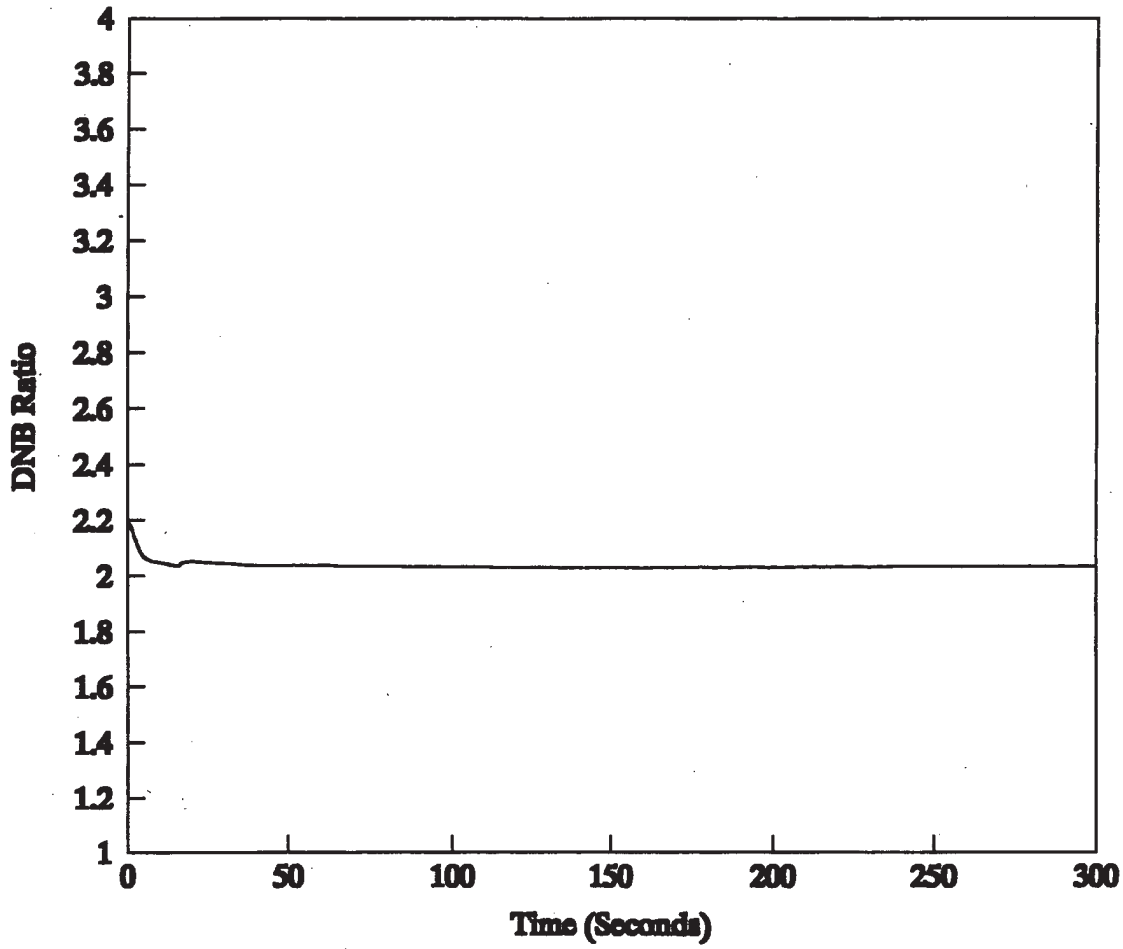


Figure 5.1.16-16

Nuclear Power Transient for Excessive Load Increase

With Automatic Rod Control,

Maximum Reactivity Feedback

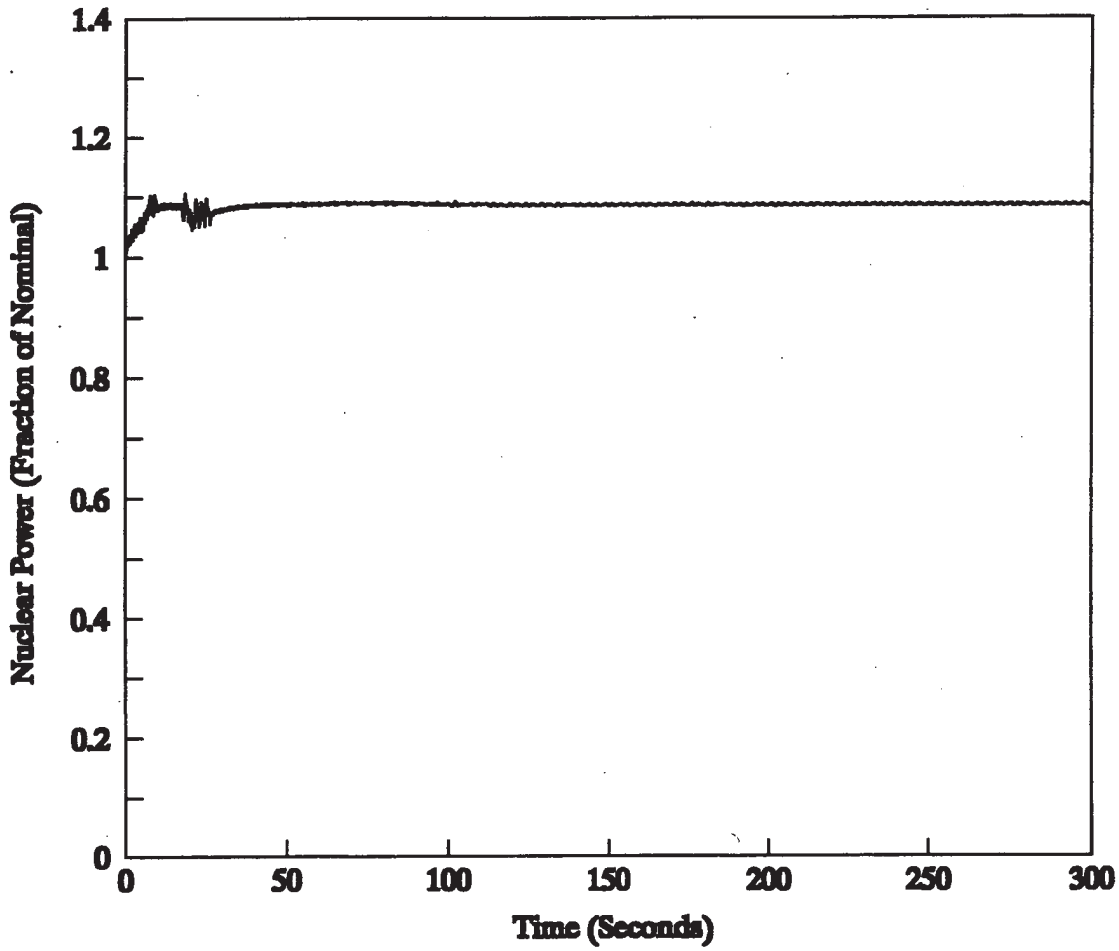


Figure 5.1.16-17

Pressurizer Pressure for Excessive Load Increase

With Automatic Rod Control,

Maximum Reactivity Feedback

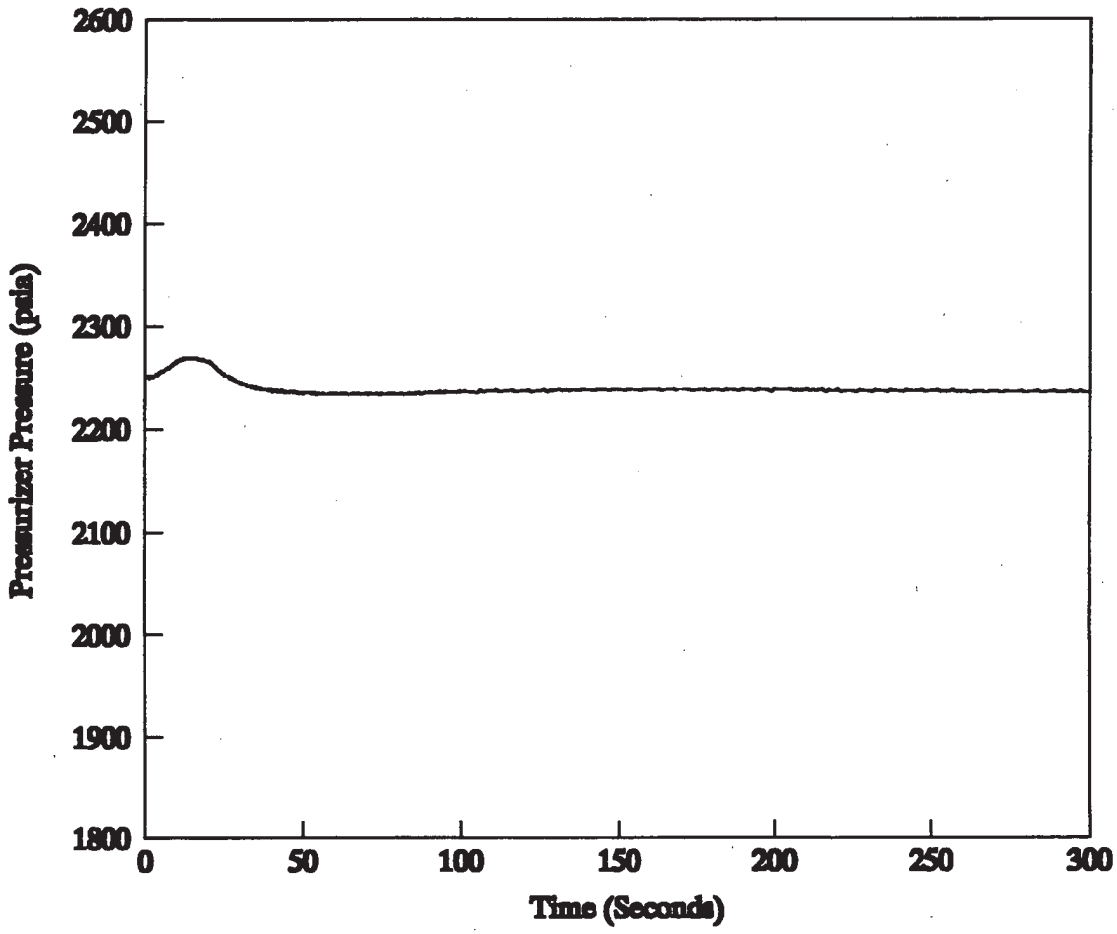


Figure 5.1.16-18

Pressurizer Water Volume Transient for Excessive Load Increase

With Automatic Rod Control,

Maximum Reactivity Feedback

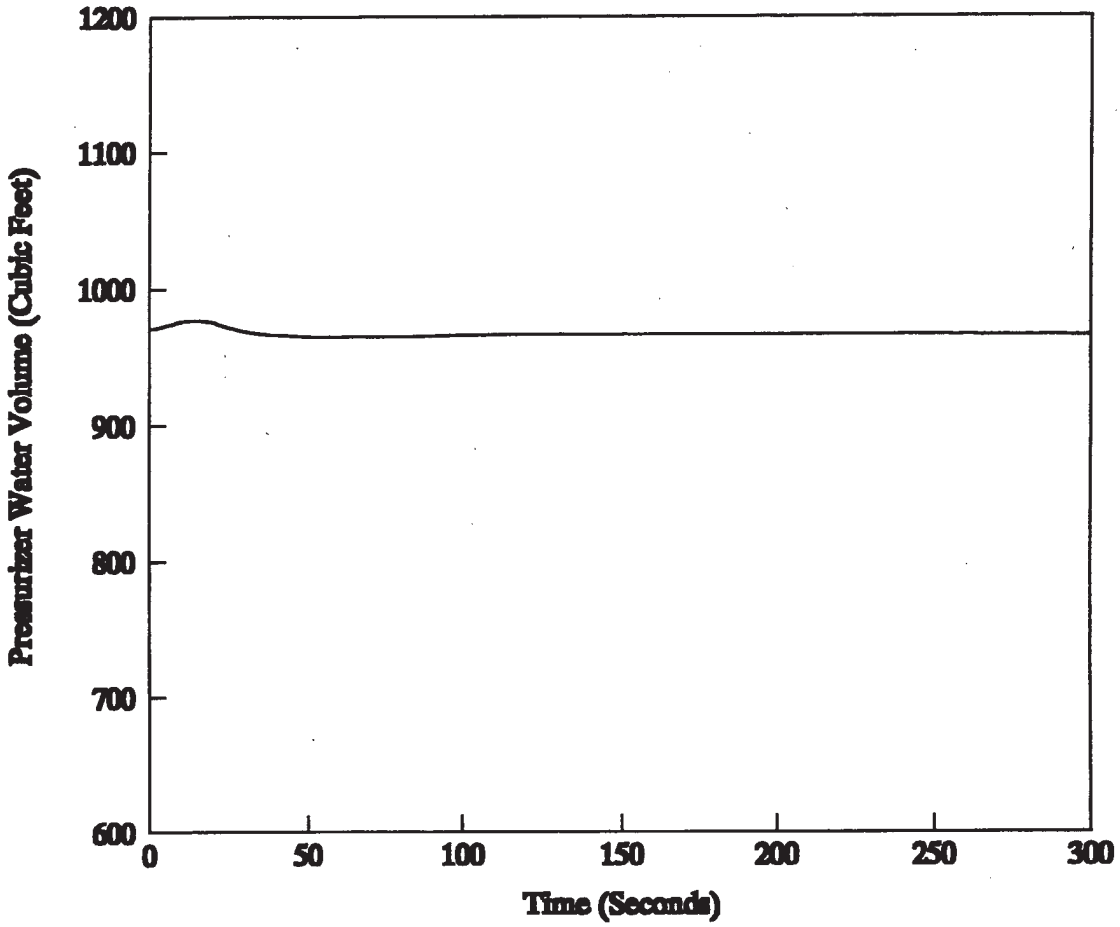


Figure 5.1.16-19

Core Average Temperature Transient for Excessive Load Increase

With Automatic Rod Control,

Maximum Reactivity Feedback

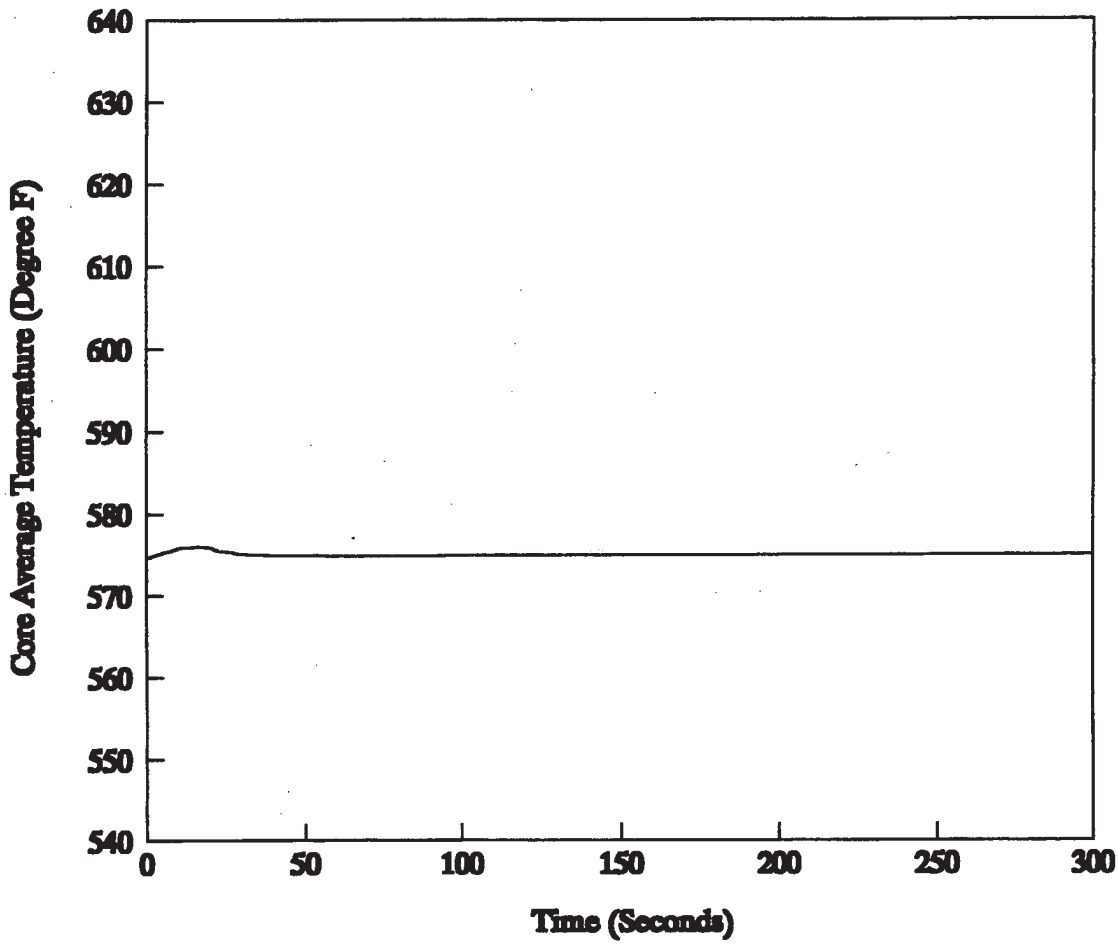
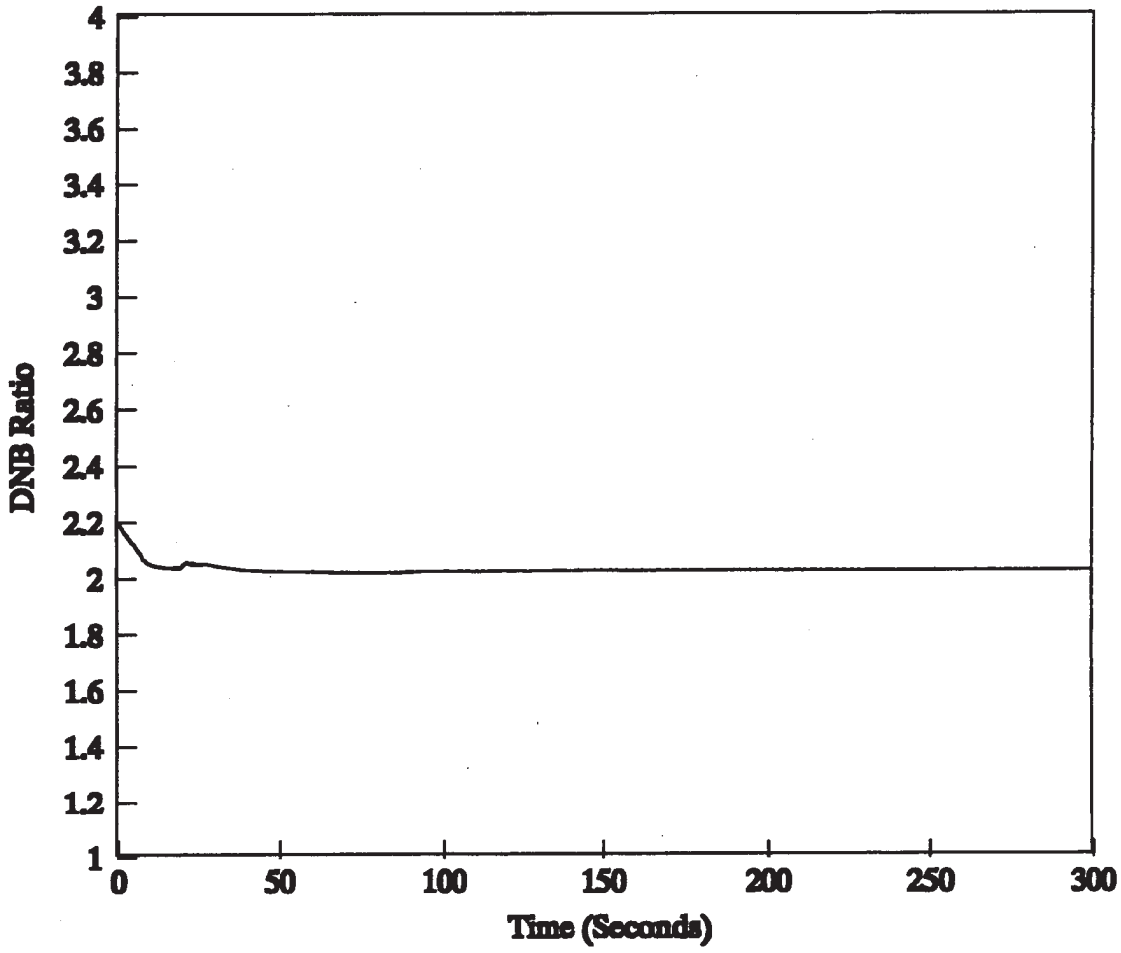


Figure 5.1.16-20

DNBR versus Time for Excessive Load Increase,

With Automatic Rod Control,

Maximum Reactivity Feedback



5.1.17 Loss of All AC Power to the Station Auxiliaries

Introduction:

The evaluation herein was performed for the Loss of All AC Power to the Station Auxiliaries event as described in the FSAR Section 14.1.12 to support the insertion of VANTAGE + Fuel with the design features described in Section 5.1.2. The evaluation also address changes in the safety analysis assumptions associated with the VANTAGE + transition as described in Section 5.1.3.

During a complete loss of offsite power and a turbine trip there will be loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc.

The events following a loss of ac power with turbine and reactor trip are described in the sequence listed below:

1) Plant vital instruments are supplied by emergency power sources.

2) As the steam system pressure rises following the trip, the steam system power-operated relief valves are automatically opened to the atmosphere. Steam bypass to the condenser is not available because of loss of the circulating water pumps. If the steam system power-operated relief valves are not available, the self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

3) As the no-load temperature is approached, the steam system power-operated relief valves (or the self-actuated safety valves, if the power-operated relief valves are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

4) The emergency diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

The auxiliary feedwater system is started automatically, actuated by the low-low steam generator water level signal. The steam-driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The two motor-driven AFW pumps are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

For the analysis of the Loss of All AC power to the Station Auxiliaries event, the worst single failure in the auxiliary feedwater system occurs (turbine-driven pump) and one motor-driven pump is assumed to be unavailable. Therefore, the auxiliary feedwater system is assumed to supply a total of 340 gpm to two steam generators from one motor-driven auxiliary feedwater pump.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. With asymmetric steam generator tube plugging, natural circulation flow rates will be slightly different between loops. However, this difference is neither a large nor a significant effect.

The Loss of All AC Power to the Station Auxiliaries event is classified as an ANS Condition II fault as defined by ANS-051.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." A Condition II occurrence is defined as a fault of moderate frequency, which, at worst, should result in a reactor shutdown with the plant being capable of returning to operation. In addition, a Condition II event should not propagate to cause a more serious fault, i.e, a Condition III or IV category event. The applicable Indian Point Unit 3 safety analysis licensing basis acceptance criteria for this Condition II event are:

a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (i.e., 2748.5 psia and 1208.5 psia, respectively),

b) Fuel Cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit; and,

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

To address criterion (c), the following more restrictive criterion has been adopted for ease in interpreting the transient results:

> The pressurizer shall not become water solid as a result of this Condition II transient within 10 minutes; the minimum time required for the operator to identify the event and take corrective action.

The basis for demonstrating that the pressurizer will not become water solid is to preclude the possibility of discharging primary coolant through the pressurizer power operated relief valves and/or the safety

relief valves, causing the event to progress from one of moderate frequency to an infrequent fault (i.e., small break LOCA) if the valves should fail to close after discharging water.

Evaluation:

As is the case for the Loss of Normal Feedwater event as described in Section 5.1.14, the analysis of the Loss of All AC Power to the Station Auxiliaries event is performed to show that following the loss of normal feedwater with loss of power, the auxiliary feedwater system, powered from the diesel generators, provides sufficient inventory to the secondary-side system to remove stored and residual heat from the reactor core. This prevents the pressurizer from reaching a water solid condition and precludes any subsequent water relief through the pressurizer relief or safety valves.

Like the Loss of Normal Feedwater event, departure from nucleate boiling is not a concern for this event since the plant is tripped well before the steam generator heat transfer capacity is significantly reduced and the primary system variables never approach DNB conditions. Therefore, the VANTAGE + Fuel features and other revised safety analysis assumptions as described in Sections 5.1.2 and 5.1.3, respectively, will not have any adverse affect on the existing licensing basis analysis of the Loss of All AC Power to the Station Auxiliaries event.

For completeness, it should be noted that the current licensing basis analysis was performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq 24\%$; and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of $\leq 24\%$ with a maximum steam generator tube plugging level of 30% in any one steam generator. This is acheived by assuming a uniform 30% steam generator tube plugging level to represent the most conservative assumption in order to depict the reduced primary-to-secondary heat transfer due to asymmetric steam generator tube plugging.

Conclusions:

Based on the evaluation herein, it is concluded that the current licensing basis safety analysis for the Loss of All AC Power to the Station Auxiliaries event remains valid for the insertion of VANTAGE + Fuel into Indian Point 3. The current licensing basis safety analysis shows that all applicable Condition II safety criteria are met for this event.

Specifically, the analysis demonstrates that sufficient long-term heat removal capability exists to prevent

fuel or clad damage and that the minimum auxiliary feedwater capacity of 340 gpm is sufficient to prevent pressurizer from reaching a water-solid condition which precludes any subsequent water relief through the pressurizer relief and safety valves and assures that the reactor coolant system is not overpressurized. The analysis (which assumes a heat transfer coefficient in the steam generator associated with reactor coolant system natural circulation following the reactor coolant pump coastdown) also shows that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

The above conclusion is valid for the VANTAGE + Fuel including the design features and associated changes in the safety analysis assumptions as described in Sections 5.1.2 and 5.1.3, respectively.

5.1.18 Startup Accidents without Reactor Coolant Pump Operation

This accident is discussed in Section 14.1.13 of the FSAR. The FSAR states for this event that the Technical Specifications do not permit the reactor to be critical above 2% rated power unless at least two reactor coolant pumps are in operation (with the exception of test conditions). The design changes associated with the transition to VANTAGE + Fuel as discussed in Sections 5.1.2 and 5.1.3 have no impact on this event as described in FSAR Section 14.1.13.

5.1.19 Startup Accident with a Full Pressurizer

This accident is discussed in Section 14.1.14 of the FSAR. The FSAR states for this event that the Technical Specifications require that the reactor shall be maintained subcritical by at least 1% Δk until normal water level is established in the pressurizer and concludes that the pressurizer will not be solid when criticality is achieved. The design changes associated with the transition to VANTAGE + Fuel as discussed in Sections 5.1.3 and 5.1.4 have no effect on this event as described in FSAR Section 14.1.14.

5.1.20 Rupture of a Steam Pipe

Introduction:

A rupture of a steam pipe results in an uncontrolled steam release from a steam generator. The steam release results in an initial increase in steam flow which decreases during the accident as the steam

pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. In the presence of a negative coolant temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive rod cluster control assembly (RCCA) is assumed to be stuck in its fully withdrawn position, there is an increased possibility that the core will became critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high hot-channel factors which exist when the most reactive RCCA is assumed to be stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances that could lead to power generation following a steam line break, the core is ultimately shut down by the boric acid delivered by the Emergency Core Cooling System.

The analysis of the Rupture of a Steam Pipe event considers both hypothetical and credible steamline breaks. A hypothetical steamline break is defined as the double ended rupture of a main steamline. This event is classified as an ANS Condition IV event, a limiting fault. Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to the public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and Containment.

A credible steamline break is classified as an ANS Condition II event and is defined as the release of steam equivalent to the spurious opening, with failure to close, of the largest of any single steam bypass, relief or safety valve. The applicable Indian Point Unit 3 safety analysis licensing basis acceptance criteria for this Condition II event are:

a) Pressures in the reactor coolant and main steam systems should be maintained below 110% of the design values (2748.5 psia and 1208.5 psia, respectively),

b) Fuel Cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit; and,

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

The purpose of the analysis for the Rupture of a Steam Pipe events is to show that the applicable acceptance criteria for the given ANS Conditions are met for all the cases considered.

The acceptance criteria for hypothetical steamline breaks cases is conservatively demonstrated by showing that the more restrictive Condition II criterion for DNB is met. This ensures that there is no damage to the fuel cladding and no release of fission products from the fuel to the reactor coolant system. The acceptance criterion of no fuel rod failures for credible break case is also demonstrated by showing that no DNB occurs.

The following systems provide the necessary protection against a steam pipe rupture:

1) Safety Injection System actuation from any of the following:

   a) Two-out-of-three low pressurizer pressure signals;

   b) Two-out-of-three high differential pressure signals between steam lines;

   c) High steam flow in two-out-of-four main steam lines (one-out-of-two per line), in coincidence with either low Reactor Coolant System average temperature (two-out-of-four loops) or low main steam line pressure (two-out-of-four);

   d) Two-out-of-three high containment pressure signals;

   e) High-High containment pressure (2 sets of 2/3); or

   f) Manual.

2) The overpower reactor trips (nuclear flux and OPΔT) and the reactor trip occurring in conjunction with receipt of the Safety Injection System.

3) Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

4) Trip of the fast-acting steam line stop valves (designed to close in less than 5 seconds) on:

a) High steam flow in two-out-of-four main steam lines (one-out- of-two per line), in coincidence with either low Reactor Coolant System average temperature (two-out-of-four loops) or low main steam line pressure (two-out-of-four); or

b) High-High containment pressure (2 sets of 2/3)

Each steam line has a fast-closing stop valve with a downstream reverse steam flow check valve. These eight valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, in the case of a break upstream of the stop valve in one main steam line, closure of either the check valve in that line or the stop valves in the other lines will prevent blowdown of the other steam generators. In particular, the arrangement precludes blowdown of more than one steam generator inside the containment and thus prevents structural damage to the containment. In addition, each of the steam generators have integral venturi type flow restrictors located at the steam generator outlet nozzle. These flow restrictors serve to limit the rate of steam release for postulated large steam line breaks inside or outside containment.

Method of Analysis and Assumptions:

The Rupture of a Steam Pipe transients are analyzed to determine: 1) the effects of the excessive cooldown on reactivity, reactor coolant system pressure, and reactor coolant system temperature for DNBR; and 2) the effects on primary-to-secondary heat transfer and secondary-side conditions for mass and energy release rates. The primary purpose of the analysis is to ensure that the required protection system features are adequate to prevent the applicable safety analysis limits from being exceeded.

Specifically, the analysis of a steam pipe rupture is performed to demonstrate that:

1) Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact;

2) Offsite radiation levels during the accident and post-accident control phase are acceptable (Condition IV criterion);

3) No fuel damage will occur for a credible steam line break equivalent to the spurious opening, with failure to close, of the largest of any single steam bypass, relief or safety valve (Condition II criterion); and

4) Energy release to the Containment from the worst hypothetical steam line break does not cause failure of the containment structure.

For items 1 through 3 above, the core heat flux, and the Reactor Coolant System temperature, pressure, and flow transient conditions following a steam pipe rupture are determined using the LOFTRAN computer code described in Section 5.1.5. These transient conditions are then used to determine the thermal and hydraulic behavior of the core following a steam line break using the THINC computer code; a detailed thermal-hydraulic computer code used to determine if DNB occurs for the core conditions computed. The determination of the critical heat flux is based on local coolant conditions.

For item 4, the pressure conditions inside containment resulting from the mass and energy released to containment through the hypothetical steamline rupture are also considered. The mass and energy release rates are determined using the LOFTRAN code. The pressure conditions inside containment are then determined by a separate containment integrity analysis based on the resulting mass and energy release rates.

Core Response Analysis:

The analysis is performed to bound operation with steam generator tube plugging levels up to: 1) a maximum uniform steam generator tube plugging level of $\leq$ 24%; and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of $\leq$ 24% with a maximum steam generator tube plugging level of 30% in any one steam generator and considers the following four cases;

1) Hypothetical Main Steamline Break with offsite power and uniform steam generator tube plugging $\leq$ 24%.

2) Hypothetical Main Steamline Break with a consequential loss of offsite power and uniform steam generator tube plugging $\leq$ 24%.

3) Credible Steamline Break and uniform steam generator tube plugging $\leq$ 24%.

4) Hypothetical Main Steamline Break with offsite power and asymmetric steam generator tube plugging $\leq$ 24%.

Since the hypothetical Main Steamline Break with offsite power is the most limiting case with respect to

DNB, the latter asymmetric steam generator tube plugging configuration is only considered for this case.

In performing the safety analysis for steam generator tube plugging levels $\leq$ 24% (both uniform and asymmetric), the analysis assumptions are conservatively selected to bound conditions within this range. For the asymmetric steam generator tube plugging configuration, the excessive feedwater flow is assumed to occur in the loop with the least plugged steam generator to maximize the cooldown effects.

Other pertinent analysis assumptions that affect the core response steamline break transient conditions are as follows.

Initial conditions:

The plant is assumed to be operating at hot zero power (HZP) with reactor coolant system pressure equal to nominal reactor coolant system pressure of 2250 psia, reactor coolant system flow rate equal to the Thermal Design Flow (TDF) rate of 323600 gpm (total flow) corresponding to a steam generator tube plugging level of $\leq$ 24% (both uniform and asymmetric), reactor coolant system vessel average temperature equal to no-load T_{avg} of 547 °F, and steam generator pressure equal to the no-load pressure of 1000 psia.

In the LOFTRAN model, the HZP initial power level is modeled as 0.01 of the nominal core power level of 3025 MWt. The pressure drops around the reactor coolant system loop reflect reduced flow conditions associated with the 24% steam generator tube plugging level and the reactor coolant system volumes for primary side of the steam generators conservatively reflect the 0% steam generator tube plugging level.

The initial pressurizer water volume is assumed to be 451.83 ft^3. This corresponds to a pressurizer level of 23.1%; the nominal pressurizer water level for zero power conditions.

An initial steam generator mass of 150,000 lbm per steam generator at HZP conditions is conservatively assumed. The conservatively high initial steam generator mass increases the magnitude of the cooldown and, without the isolation of the faulted steam generator, prolongs the duration of the cooldown event.

The initial core boron concentration is assumed to be 0 ppm.

Shutdown margin:

For the HZP initial conditions assumed in the steam line break core response analysis, the reactor is assumed to be tripped when the steam line break event occurs. All the RCCAs are assumed to be inserted with the exception of the highest worth RCCA, which is assumed to be stuck in a fully withdrawn position. With this initial configuration, the reactor is assumed to be subcritical by the minimum required 1.30% Δk amount of shutdown margin. This is the end-of-life design value including design margins at no-load, equilibrium xenon conditions, with the most reactive RCCA stuck in its fully withdrawn position. The actual shutdown capability is expected to be significantly greater. The operation of the RCCA banks during core burnup is restricted in such a way that addition of positive reactivity in a steam break accident will not lead to a more adverse condition than the case analyzed.

Reactivity coefficients:

The negative moderator coefficient of reactivity assumed is that corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod assembly has been included in the overall reactivity balance. The local reactivity feedback is composed of the Doppler reactivity from the high fuel temperatures near the stuck RCCA.

The moderator density coefficients and other physics parameters used in the LOFTRAN point-kinetics model are characteristic of end-of-life conditions and the resulting transient conditions calculated by LOFTRAN are confirmed to be conservative relative to predictions made in confirmatory 3D physics models on a cycle-by-cycle reload basis.

For hypothetical breaks upstream of the flow restrictor, the core properties associated with the sector nearest the faulted steam generator and those associated with the remaining sectors were conservatively combined to obtain average core properties for reactivity feedback calculations. A non-uniform radial weighting factor of 50%, for the sector nearest the faulted steam generator and 16.7% each for the remaining three sectors of the core are assumed for the hypothetical steamline break cases to account for the non-uniform cooldown of the reactor coolant system. These conditions conservatively cause underprediction of the Doppler reactivity feedback in the high power region near the stuck rod. For the power peaking factors, those corresponding to one stuck RCCA and non-uniform core inlet temperatures at end-of-life conditions are assumed. The coldest core inlet temperatures are assumed to occur in the

sector with the stuck RCCA. The power peaking factors account for the effect of the local void in the region of the stuck RCCA during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck RCCA. Since the power peaking factors depend on the core power, operating history, temperature, pressure, and flow, they may differ from cycle to cycle.

To verify the conservatism of the assumptions used in the LOFTRAN point-kinetics reactivity feedback model, the reactivity as well as the power distribution are checked for the limiting statepoints of the cases analyzed. This core analysis considers the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects in the case of the hypothetical breaks.

For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the statepoints. These results verify conservatism; i.e., overprediction of positive reactivity from the cooldown and underprediction of negative reactivity from power generation.

Offsite power:

For the cases analyzed assuming offsite power is available, offsite power is assumed to be available throughout the transient which results in continuous reactor coolant pump operation such that full and constant thermal design flow rate is modeled throughout the event.

For the case analyzed assuming a consequential loss of offsite power, the reactor coolant pumps are assumed to begin a conservative inertial coastdown with the flywheel beginning 3 seconds after event initiation. This results in a reduction of reactor coolant system flow throughout the event.

Feedwater:

To maximize the cooldown following the steam line break event, a full and constant main feedwater flow was conservatively modeled for the hypothetical breaks. Nominal feedwater flow is assumed at the transient initiation and continues until the time of feedwater isolation which occurs after receipt of a safety injection signal. Feedwater isolation is assumed to occur 12 seconds after the safety injection signal is generated. The 12 second delay is a conservatively long time for signal processing, valve realignment,

etc. A conservatively low initial feedwater enthalpy of 70.68 Btu/lbm is assumed for the HZP initial conditions. This corresponds to a feedwater temperature of 100°F. A lower feedwater enthalpy is conservative for steam line break since it increases the magnitude of the cooldown associated with the steam line break event.

Auxiliary Feedwater:

Auxiliary feedwater flow is assumed to start at the transient initiation and continue throughout the transient to maximize the cooldown effects for core response. A flow rate of 1600 gpm is assumed in all cases. This represents two motor driven auxiliary feedwater pumps with design flow rate of 400 gpm each and one turbine driven pump with design flow rate of 800 gpm. For the hypothetical steamline break cases, this total auxiliary feedwater flow is conservatively assumed to be delivered to the faulted steam generator.

For the credible steam line break case, the assumed auxiliary feedwater flow rate and initiation time are the same as in the hypothetical steamline break cases but the 1600 gpm is assumed to be split equally among all four steam generators.

The temperature of the auxiliary feedwater is conservatively assumed to be 32 °F and an auxiliary feedwater purge volume of 1 ft^3 is conservatively modeled.

For both the hypothetical and credible steam line break cases, the auxiliary feedwater flow is not required to mitigate the consequences of the event.

Safety Injection:

In the steamline break analyses, the following assumptions are made regarding the safety injection system:

a) Safety injection flow rates are calculated based on the operation of only one train of safety injection. The failure of the other train is the worst active single failure assumption. In all cases, the safety injection flow rates are calculated based on all cold legs injecting into the reactor coolant system.

b) The refueling water storage tank (RWST) contains borated water with a minimum boron concentration of 2300 ppm and all of the safety injection lines downstream of the RWST, including the boron injection tank (BIT), contain unborated water.

c) A conservatively low enthalpy of 3.05 Btu/lbm for the safety injection fluid in the RWST and the safety injection lines is assumed. A lower enthalpy for the safety injection fluid is conservative since it increases and prolongs the cooldown of the reactor coolant system.

d) A conservative time required to sweep the unborated water from the safety injection piping and BIT before delivering the 2300 ppm borated water from the RWST to the core is modeled.

The sequence of events in the safety injection system are as follows:

a) For the cases where offsite power is assumed, after the generation of the safety injection signal (including conservative delays for the instrumentation, logic, and signal transport), the appropriate valves begin to operate and the high-head safety injection pumps start.

b) Within 12 seconds following (a) above, the valves are assumed to be in their final position to allow full safety injection flow, and the pumps are assumed to be at full speed.

c) In the cases where offsite power is not available, an additional 10 seconds is assumed before (b) above to model the time required to start and load the necessary safety injection equipment onto the diesel generators.

d) For safety injection and steam line isolation signals actuated on high steam flow coincident with either low reactor coolant system average temperature or low steam line pressure, an additional time delay of 6 seconds is assumed after (a) above.

For actuation of the Safety Injection System and closing of the fast-acting steam line stop valves as previously discussed, the following setpoints were assumed in the analysis for the high steam flow coincidence logic.

a) A low steam line pressure setpoint of 435 psia including uncertainties to account for channel errors and adverse environmental errors.

b) A low T_{avg} setpoint of 533 °F including uncertainty to account for channel accuracy.

c) A high steam flow setpoint of 64% of full steam flow including uncertainties for channel errors and adverse environmental errors.

In the Indian Point Unit 3 Technical Specifications, the high steam flow safety injection setpoint is a function defined as follows:

> A Δp corresponding to 40 percent of full steam flow between 0 percent and 20 percent load and then a Δp increasing linearly to a Δp corresponding to 110 percent of full steam flow at full load. A 24 percent uncertainty in steam flow was added to account for channel errors and adverse environmental errors.

For actuation of the Safety Injection System on a low pressurizer pressure signal, the low pressurizer pressure setpoint assumed in the analysis is 1650 psia, including uncertainties. For the credible steam line break case, safety injection flow is assumed to be available 12 seconds after the initiation signal is generated on low pressurizer pressure. The 12 second delay is a conservatively long time for signal processing, valve realignment, etc. Instantaneous full safety injection flow is assumed to occur whenever reactor coolant system pressure falls below safety injection pump head of 1415 psia at any time $\geq$ 12 seconds after the safety injection signal occurs.

The analyses of the hypothetical large steam line break cases assume a failed reverse steam flow check valve in the broken steam line. This conservative assumption results in additional cooldown of the Reactor Coolant System. The additional steam release out the break from the three intact steam generators continues until the main steam stop valves close in the intact steam lines.

Heat Transfer Modeling:

Fuel-to-coolant heat transfer coefficients consistent with limiting end-of-cycle conditions and conservatively representing minimum fuel temperatures that bound operation with both VANTAGE 5 and VANTAGE + fuel are assumed in the analysis.

No credit is taken for heat transfer from the thick metal throughout the reactor coolant system to the coolant.

On the secondary-side, the Westinghouse Model 44F Steam Generators were modeled in the analysis.

Decay Heat:

No credit is taken for decay heat since this would inhibit the cooldown of the reactor coolant system.

Steam Generator Water Entrainment:

Perfect moisture separation in the steam generators is assumed. This assumption leads to conservative results, especially for large breaks, since there would be considerable entrainment of the water in the steam generators following a steamline break. Entrainment of water would reduce the magnitude of the cooldown of the reactor coolant system.

Accident Simulation:

In determining the core power transients which can result from a steam line break, the following steam line break conditions were considered.

a) Complete severance of a main steam line at the exit of the steam generator (down stream of the integral steam flow restrictors) with the plant initially at no-load conditions and all reactor coolant pumps running

b) Case (a) above assuming a loss of offsite power resulting in a coolant pump coastdown 3 seconds following the occurrence of the steam line break.

These hypothetical steam line break cases represent the most severe Condition IV steam line breaks that can be postulated to occur.

A third case, a credible steam line break, was analyzed consistent with the Condition II criterion stated earlier that no fuel damage shall occur as a result of a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam bypass, relief or safety valve.

c) A break equivalent to the release through one steam generator safety valve with offsite power available.

Initial hot shutdown conditions were considered for all of the above cases since this represents the most pessimistic initial conditions for the accident. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection logic when the trip setpoint is reached. Following a trip at power the Reactor Coolant System contains more stored energy than at no load, the average coolant temperature is higher than that at no load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown

caused by the steamline break before the no load conditions of Reactor Coolant System temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertion proceed in the same manner as in the analysis which assumes no load condition at time zero. However, since the initial steam generator water inventory is greatest at no load, the magnitude and duration of the Reactor Coolant System cooldown are less for steam breaks occurring at power.

In computing the steam flow during a steamline break or the inadvertent opening of a steam safety valve, the Moody Curve (Reference 43) for f(L/D) = 0 is used.

The break area assumed for hypothetical breaks downstream of the flow restrictor is 1.4 ft^2 per loop. This is the area of the steamline flow restrictor. All four steam generators are assumed to blow down to atmospheric pressure through their respective flow restrictors until steamline isolation occurs on the intact steam generators. A break area of 0.0945 ft^2 on one loop is assumed for the credible steamline break case.

Containment Response Analysis:

For evaluating containment pressure and temperature response, the hypothetical large steam line breaks are considered in determining the maximum mass and energy releases inside containment. The most limiting case for containment response following a rupture of a steam pipe inside containment is that initiated from 70% power conditions and assuming the failure of a feedwater control valve in the faulted steam generator loop.

For this limiting case, the safety injection actuation signal modeled in the LOFTRAN analysis used to generate the mass and energy release rates is solely based on the High Containment Pressure signal. No other primary or secondary side protection signals are credited in the calculation of the resulting mass and energy release rates.

Therefore, there are no changes associated with the VANTAGE + reload transition which adversely affect the existing mass and energy release rates for this limiting Containment Response case.

Hence, no reanalysis of this portion of the Rupture of a Steam Pipe event is required to support the transition to VANTAGE + fuel.

Results:

Core Power and Reactor Coolant System Transients:

Of the four steamline break cases considered core response, the hypothetical steam pipe rupture with offsite power and considering asymmetric steam generator tube plugging is most limiting.

Figures 5.1.20-1 through 5.1.20-10 show the transient conditions following a complete severance of a main steam line down stream of the integral steam flow restrictors with the plant initially at no-load conditions and all reactor coolant pumps running for the case modeling asymmetric steam generator tube plugging. The break assumed is the largest break that can occur anywhere inside or outside the containment. Offsite power is assumed available such that full reactor coolant flow exists. Since the plant is initially at no-load conditions, the transient shown assumes all RCCAs are inserted at time zero with the exception of the worst stuck RCCA being in a fully withdrawn position.

As shown in Figure 5.1.20-3, the core becomes critical with the rods inserted (with the design shutdown margin and assuming one stuck RCCA) at 22 seconds.

The high steam flow setpoint is reached immediately in all four loops and the low T_{avg} setpoint is reached in at least two loops at 11.5 seconds. After considering appropriate time delays for processing the signal and electronics, at 19.5 seconds the actuation of safety injection and steam line isolation is initiated. At 24.5 seconds, isolation of the intact steam lines via closure of the fast-acting steam line stop valves occurs and main feedwater isolation is complete at 29.5 seconds. At 31.5 seconds, full safety injection flow capability of the available safety injection pumps is reached. After purging unborated water from the safety injection lines down stream of the refueling water storage tank, borated water finally reaches the core at approximately 40 seconds after initiation of the steam line rupture event. The peak core average heat flux for this case is 13.8% of the nominal full power value of 3025 MWt.

The sequence of events for all four cases are summarized in Tables 5.1.20-1 through 5.1.20-4.

Margin to Critical Heat Flux:

Based on the transient conditions for all four cases, a DNBR evaluation was performed using the W-3 DNBR correlation. This evaluation showed that the hypothetical steamline break with offsite power available and asymmetric steam generator tube plugging case is most limiting with respect to minimum

DNBR and that resulting minimum DNBR is greater than the applicable safety analysis DNBR limit. The minimum DNBR for this case (2.2) was reached at 106.3 seconds.

Containment Response:

As described earlier in the Method of Analysis and Assumptions section, no reanalysis of this portion of the Rupture of a Steam Pipe event is required to support the transition to VANTAGE + fuel on the basis that there are no changes associated with the VANTAGE + reload transition which would adversely affect the existing mass and energy release rates for the limiting Containment Response case.

Hence, the existing licensing basis analysis results for this portion of the Rupture of a Steam Pipe events remain applicable for Indian Point Unit 3 with VANTAGE 5 and VANTAGE + fuel.

Conclusions:

Although DNB and possible clad perforation are not precluded in the criteria, the Core Response analysis performed herein for the Rupture of a Steam Pipe events has demonstrated that DNB does not occur. For the limiting steam line break case inside the containment structure, the current licensing basis analysis for containment response remains applicable for the transition to VANTAGE + fuel. Hence, the peak pressure inside the containment following a rupture of a steam pipe is 42.42 psig, which is less than the applicable containment design pressure limit of 47 psig.

Hence, the analysis and evaluations contained herein for the Rupture of a Steam Pipe events demonstrate that all applicable Indian Point Unit 3 licensing basis safety analysis criteria are satisfied for the transition from VANTAGE 5 to VANTAGE + fuel. This conclusion is valid for the transition to VANTAGE + fuel, including the design features and related changes in safety analysis assumptions described in Sections 5.1.2 and 5.1.3, respectively.

Table 5.1.20-1

Sequence of Events

for the

Rupture of a Steam Pipe Events

Hypothetical Steamline Break

With Offsite Power

Uniform Steam Generator Tube Plugging

Event	Time (Seconds)
Steamline rupture occurs	0.0
High Steam Flow setpoint reached in 2 loops	0.04
Low Tavg setpoint reached in 2 loops	11.5
Low Steamline Pressure setpoint reached in 2 loops	12.8
Low Pressurizer Pressure SI Setpoint Reached	15.2
RCS Pressure falls below SI Cut-in Pressure	16.2
Core becomes critical	21.0
Steamline Isolation occurs	24.5
Feedwater Isolation occurs	29.5
Boron reaches core	40.0
Peak Heat Flux occurs	84.3
Minimum RCS Pressure occurs	136.3

Table 5.1.20-2

Sequence of Events

for the

Rupture of a Steam Pipe Events

Hypothetical Streamline Break

With Loss of Offsite Power

Uniform Steam Generator Tube Plugging

Event	Time (Seconds)
Steamline rupture occurs	0.0
High Steam Flow setpoint reached in 2 loops	0.04
Low Steamline Pressure setpoint reached in 2 loops	12.4
Low Tavg setpoint reached in 2 loops	12.7
Low Pressurizer Pressure SI Setpoint Reached	16.6
RCS Pressure falls below SI Cut-in Pressure	18.5
Steamline Isolation occurs	25.4
Core becomes critical	27.0
Feedwater Isolation occurs	30.4
Boron reaches core	52.0
Minimum RCS Pressure occurs	68.3
Peak Heat Flux occurs	170.3

Table 5.1.20-3

Sequence of Events

for the

Rupture of a Steam Pipe Events

Credible Steamline Break

With Offsite Power

Uniform Steam Generator Tube Plugging

Event	Time (Seconds)
Steamline rupture occurs	0.0
Low Pressurizer Pressure SI Setpoint Reached	299.0
Feedwater Isolation occurs	311.0
RCS Pressure falls below SI Cut-in Pressure	337.0
Boron reaches core	390.0
Core becomes critical	438.0
Minimum RCS Pressure occurs	544.8
Peak Heat Flux occurs	600.0

Table 5.1.20-4

Sequence of Events

for the

Rupture of a Steam Pipe Events

Hypothetical Steamline Break

With Offsite Power

Asymmetric Steam Generator Tube Plugging

Event	Time (Seconds)
Steamline rupture occurs	0.0
High Steam Flow setpoint reached in 2 loops	0.04
Low Tavg setpoint reached in 2 loops	11.5
Low Steamline Pressure setpoint reached in 2 loops	12.9
Low Pressurizer Pressure SI Setpoint Reached	15.0
RCS Pressure falls below SI Cut-in Pressure	16.0
Core becomes critical	22.3
Steamline Isolation occurs	24.5
Feedwater Isolation occurs	29.5
Boron reaches core	40.0
Peak Heat Flux occurs	86.3
Minimum RCS Pressure occurs	142.3

Figure 5.1.20-1

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Nuclear Power versus Time

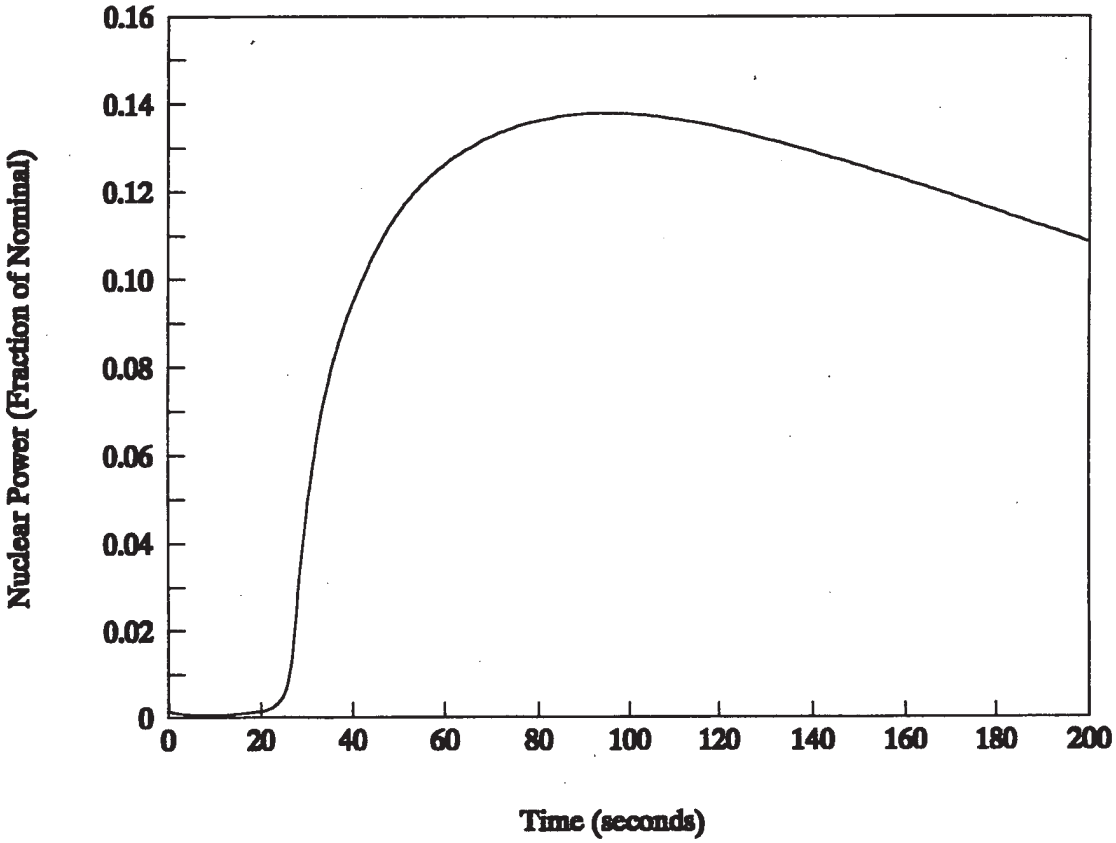


Figure 5.1.20-2

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Core Heat Flux versus Time

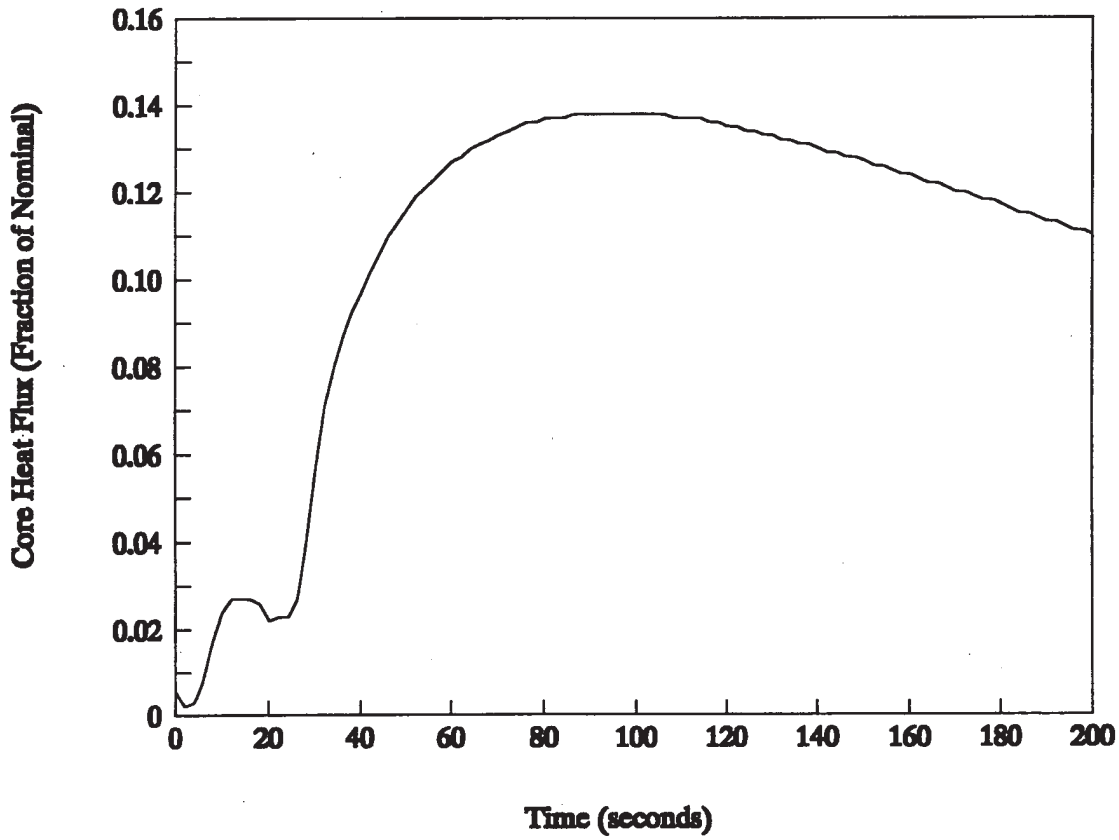


Figure 5.1.20-3

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Reactivity versus Time

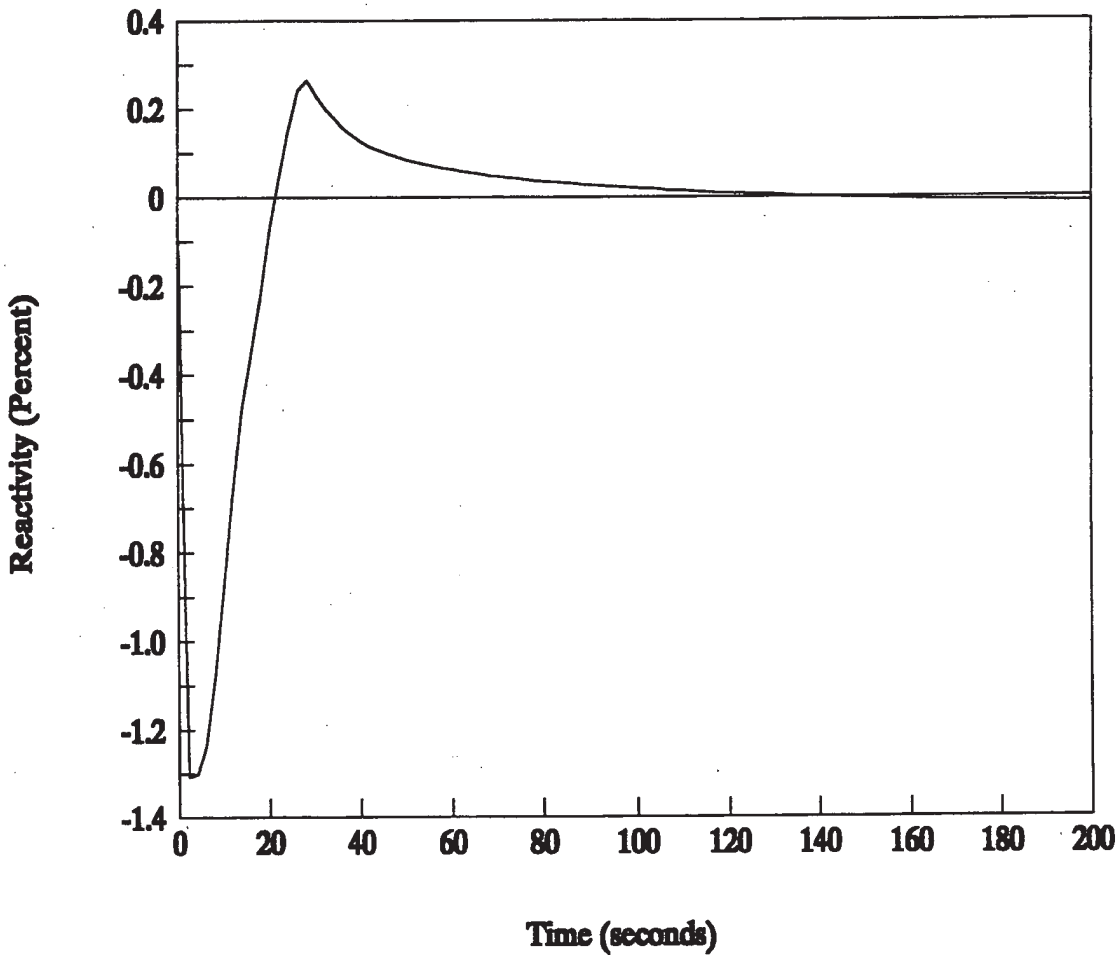


Figure 5.1.20-4

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Reactor Vessel Inlet Temperature versus Time

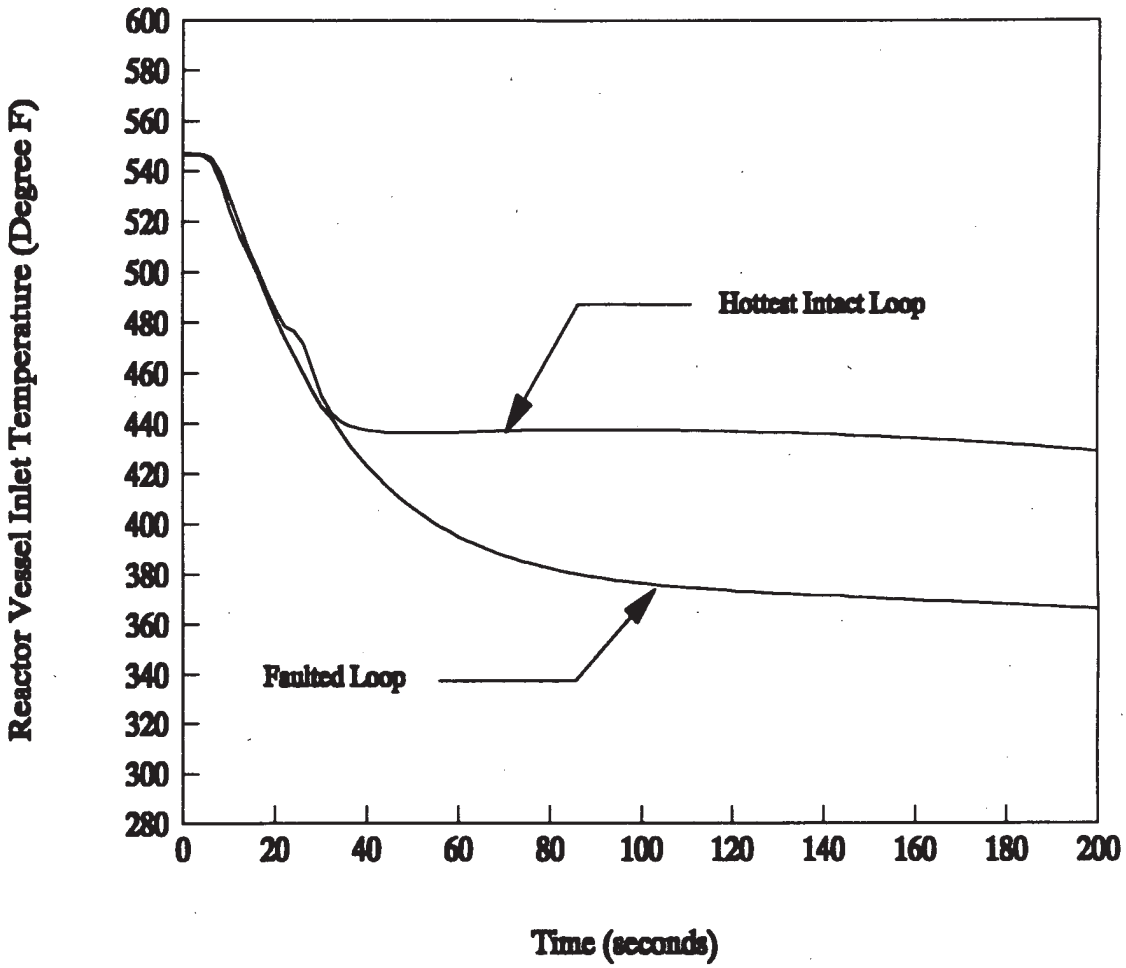


Figure 5.1.20-5

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Reactor Vessel Average Temperature versus Time

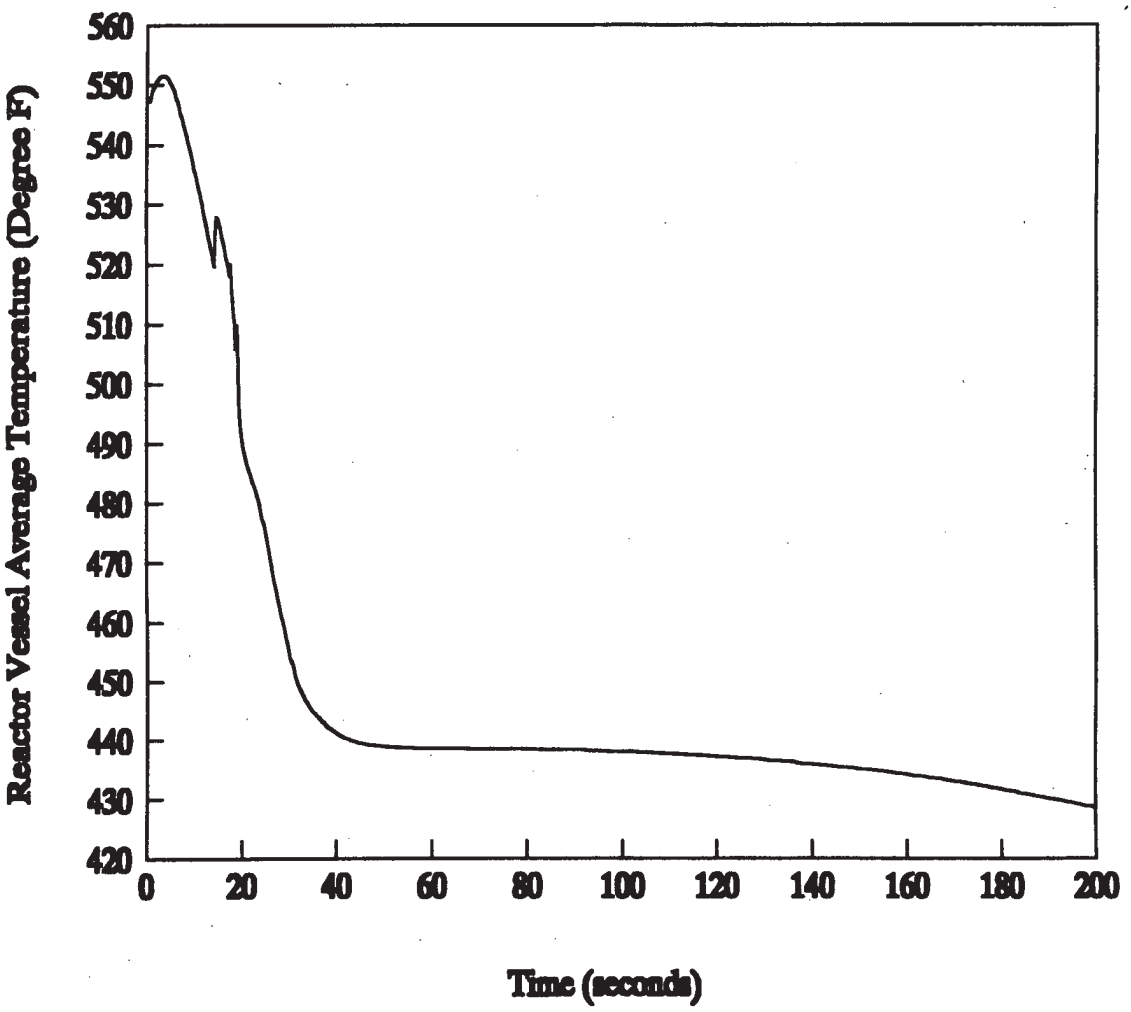


Figure 5.1.20-6

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

RCS Pressure versus Time

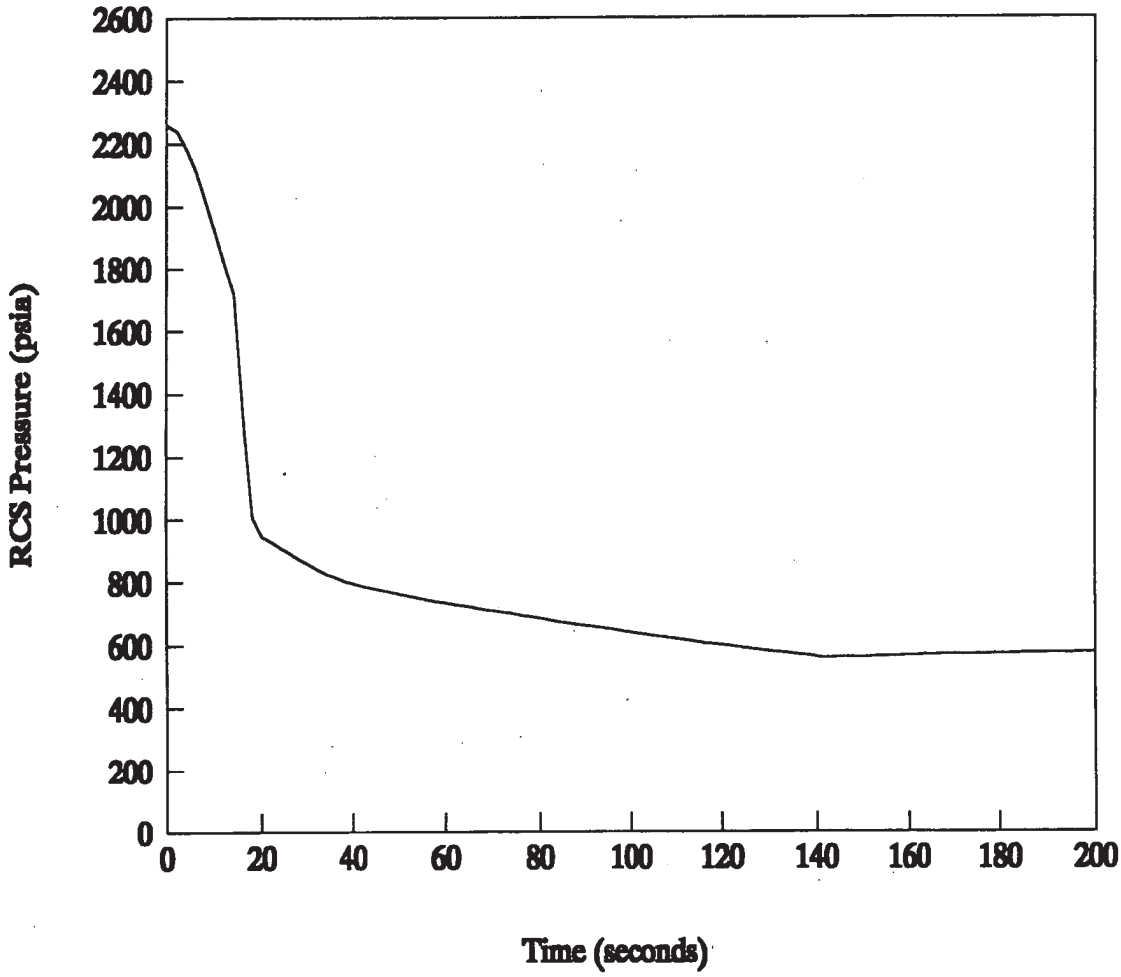


Figure 5.1.20-7

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Core Boron Concentration versus Time

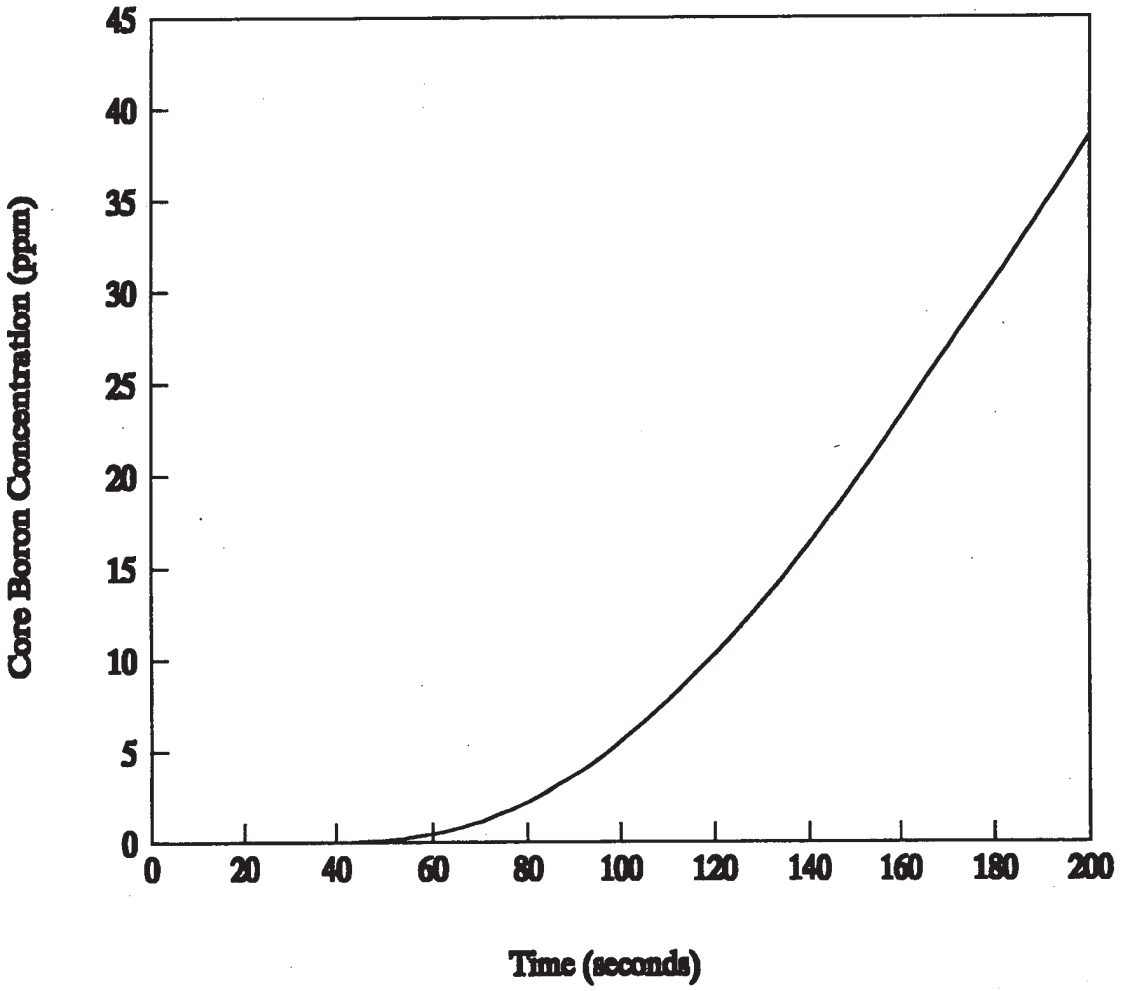


Figure 5.1.20-8

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Steamline Pressure versus Time

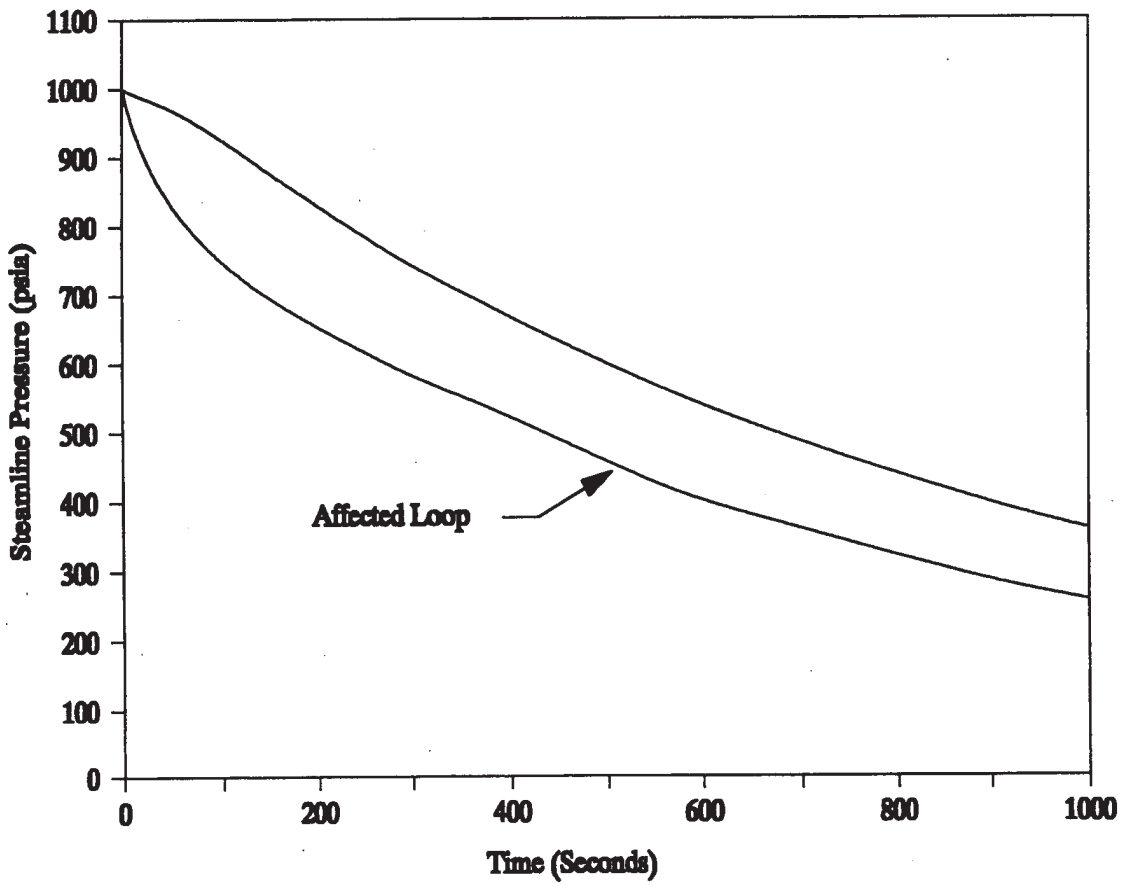


Figure 5.1.20-9

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Steam Generator Mass versus Time

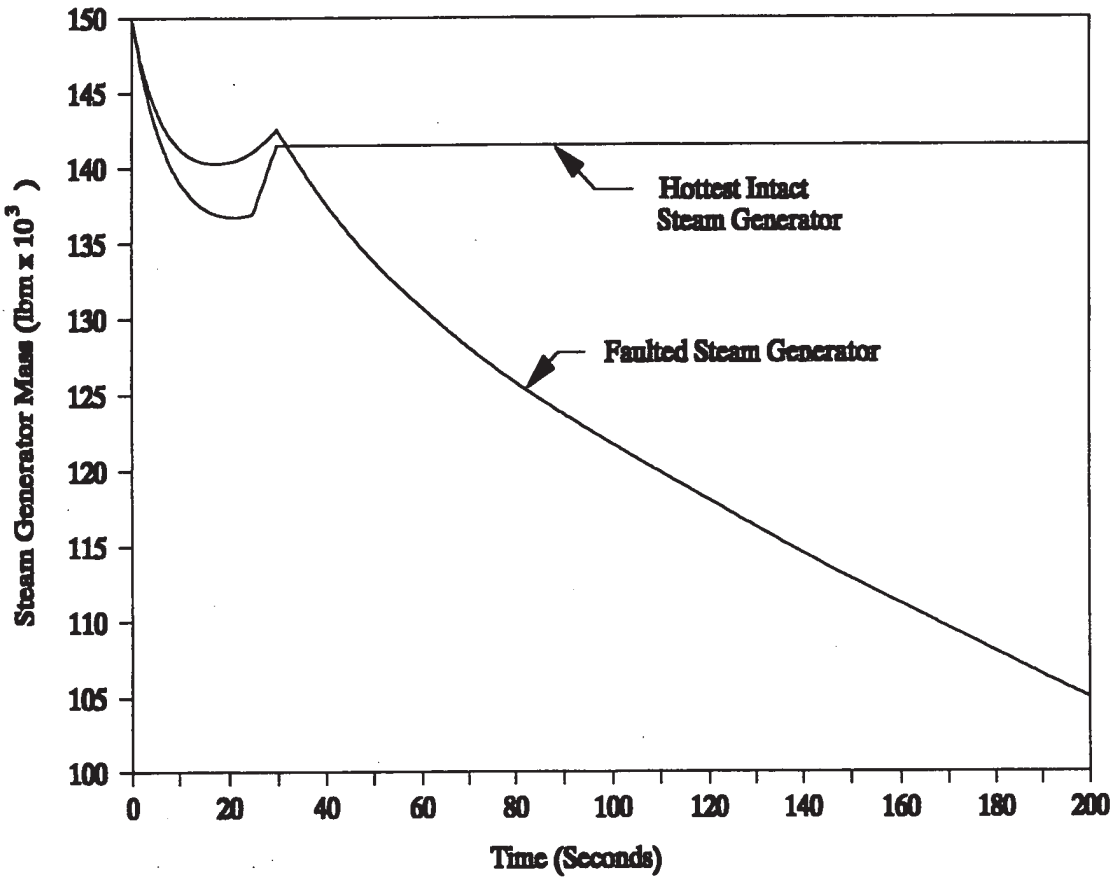


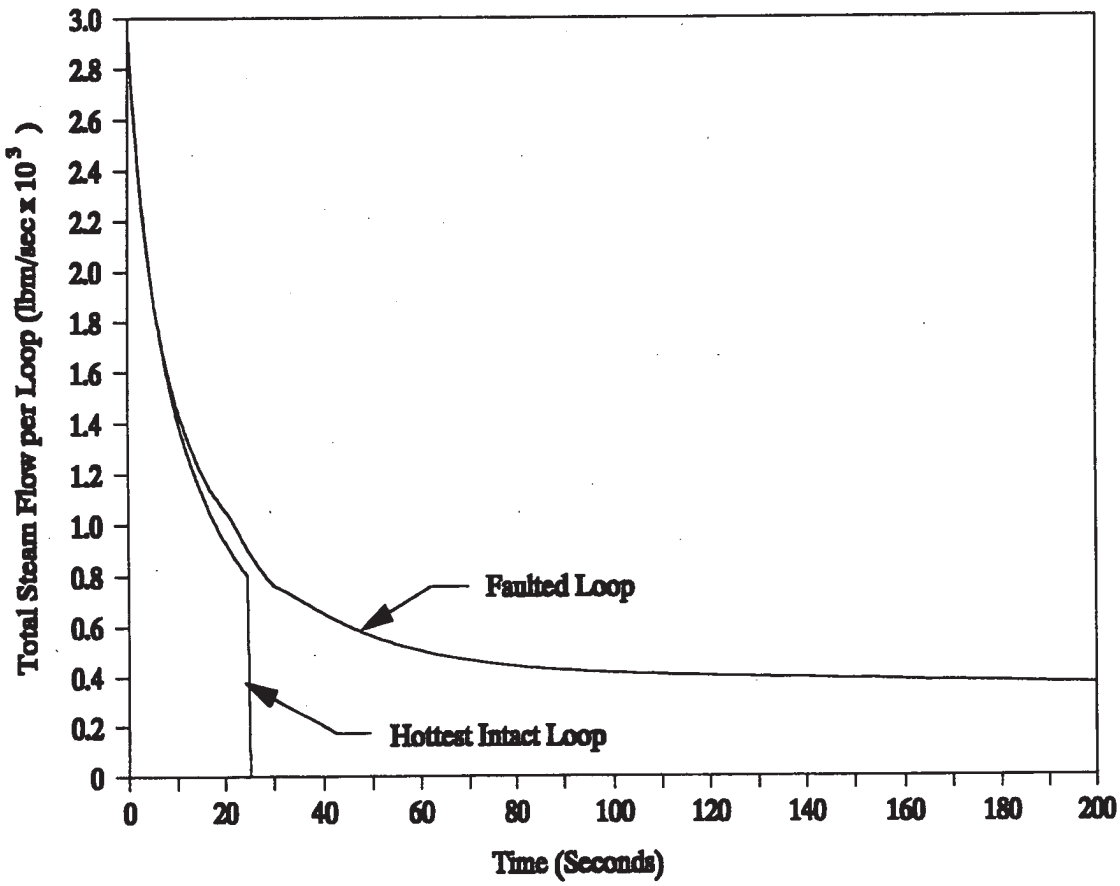
Figure 5.1.20-10

Hypothetical Steamline Break

With Offsite Power Available

Asymmetric Steam Generator Tube Plugging

Steam Flow versus Time



5.1.21 Rod Ejection

Introduction:

Due to the extremely low probability of an RCCA Ejection, this accident is classified as a Condition IV event. The applicable Condition IV criteria are that the RCS and the core must remain able to provide long term cooling, and off-site doses must remain within the guidelines of 10 CFR 100. The specific (and more restrictive) criteria used to ensure that the Condition IV criteria are met are as follows:

1. Average fuel pellet enthalpy at the hot spot must be below 200 cal/gm (360 Btu/lbm).

2. Average clad temperature at the hot spot must remain below 2700 °F.

3. Zirc-H_2O reaction is less than 16%.

4. Fuel melting will be limited to less than 10% of the fuel volume at the hot spot.

5. The peak reactor coolant pressure must remain less than that which would cause stresses to exceed the Faulted Condition stress limits.

Criterion 5 is addressed generically for the RCCA Ejection event for Westinghouse PWRs in Reference 44.

Method of Analysis and Assumptions:

The analysis methods and basis for assumptions used for the RCCA Ejection event are the same as those employed in the current FSAR licensing basis RCCA Ejection analysis as reported in FSAR Section 14.2.6. The RCCA Ejection is transient is simulated using the TWINKLE and FACTRAN computer codes described in Section 5.1.5. Cases are analyzed for four conditions; BOL-HZP, BOL-HFP, EOL-HZP, and EOL-HFP.

The following major assumptions are made in performing the RCCA Ejection analysis:

- Initial Conditions:

	HZP Cases	HFP Cases
Power Level (fraction of nominal)	0	1.02
RCS Pressure (psia)	2190	2190
Vessel Average Temperature (°F)	547.0	581.7
RCS Flow (fraction of TDF)	0.46	1.0

For the zero power cases, RCS flow is conservatively modeled as 46% of Thermal Design Flow (TDF), representing only two reactor coolant pumps in operation. The full power cases assume 100% TDF representing all reactor coolant pumps in operation.

- β_{eff}, the delayed neutron fraction, at BOL is equal to 0.0050 and β_{eff} at EOL is equal to 0.0040.

- Conservative values of trip reactivity are used assuming a stuck rod in addition to the ejected rod. These values are 4% Δk for the full power cases and 2% Δk for zero power cases. Trip reactivity insertion is simulated by dropping a rod of the required worth into the core from the full-out position. The rod drop time assumed is 2.7 seconds.

- For the RCCA Ejection event, protection is provided by a power range High Flux reactor trip. The HFP cases are modeled to trip on a high setpoint of 118% of nominal, including uncertainties. A low setpoint of 35% of nominal, including uncertainties, is modeled for the HZP cases.

- The time delay after the trip setpoint is reached and before the rods start to fall is set to 0.55 seconds. This includes time for processing of the trip signal, opening of the trip breaker, and the releasing of the rods from the coil.

- No control systems are simulated.

- The accident is initiated in the TWINKLE code by linearly changing the initial k_{eff} by an amount equal to the worth of the ejected rod over a 0.1 second time span.

- The following table summarizes the Ejected Rod Worths (%Δk), the transient hot channel factors (F_Q's) and the Doppler Weighting Factors which were used in each of the Rod Ejection cases analyzed herein.

Case	Ejected Rod Worth (%Δk)	F_Q	Doppler Weighting Factor
BOL - HZP	0.65	12.0	2.071
EOL - HZP	0.80	20.0	2.755
BOL - HFP	0.17	6.80	1.300
EOL - HFP	0.20	7.10	1.300

Results:

Figures 5.1.21-1 and 5.1.21-2 illustrate the nuclear power and fuel rod temperature transients for the EOL-HZP case; the case which results in the highest clad average temperature and magnitude of Zirc-H_2O reaction. Figures 5.1.21-3 and 5.1.21-4 illustrate the nuclear power and fuel rod temperature transients for the BOL-HFP case. The latter case results in the maximum fuel enthalpy of the four cases considered and has the highest amount of fuel exceeding the fuel melting temperature (4900 °F at BOL).

A summary of the results are as follow:

Case	Peak Avg Fuel Enthalpy (Btu/lbm)	Clad Avg Temperature (°F)	Fuel Melt (%)	Zirc - H_2O Reaction (%)
BOL-HZP	180.5	1890	0.00	0.330
EOL-HZP	244.7	2484	0.00	1.833
BOL-HFP	330.9	2305	9.87	1.025
EOL-HFP	318.5	2229	9.05	0.866
Limit	< 360.0	< 2700	< 10.0	< 16.0

Conclusions:

The results of the analysis of the RCCA Ejection event described herein show that all safety criteria are met. Specifically; the maximum clad average temperature is less than 2700 °F, maximum fuel enthalpy is less than 360 BTU/lb, fuel melting is less than 10%, and Zirc-H_2O reaction is less than 16%.

The peak reactor coolant pressure, which is addressed generically for the RCCA Ejection event in Reference 44, remains less than that which would cause stresses to exceed the Faulted Condition stress limits.

Based on these results, it is concluded that the RCS and the core will remain able to provide long term cooling, and off-site doses remain within the guidelines of 10 CFR 100. These conclusions are valid for the transition from the resident VANTAGE 5 fuel to VANTAGE + fuel, including the design features and related changes in safety analysis assumptions described in Sections 5.1.2 and 5.1.3, respectively.

Table 5.1.21-1

Sequence of Events

RCCA Ejection

	BOL-HZP	EOL-HZP	BOL-HFP	EOL-HFP
RCCA Ejected	0.0	0.0	0.0	0.0
Reactor Trip Setpoint Reached	0.35	0.18	0.05	0.04
Peak Nuclear Power	0.41	0.21	0.13	0.13
Rods Drop	0.90	0.73	0.60	0.59
Peak Fuel Average Temperature Occurs	2.70	1.84	2.42	2.50
Peak Clad Temperature Occurs	2.57	1.58	2.53	2.61

(All Times in Seconds)

Figure 5.1.21-1

Fuel Rod Temperatures for Rod Ejection

End of Life - Hot Zero Power

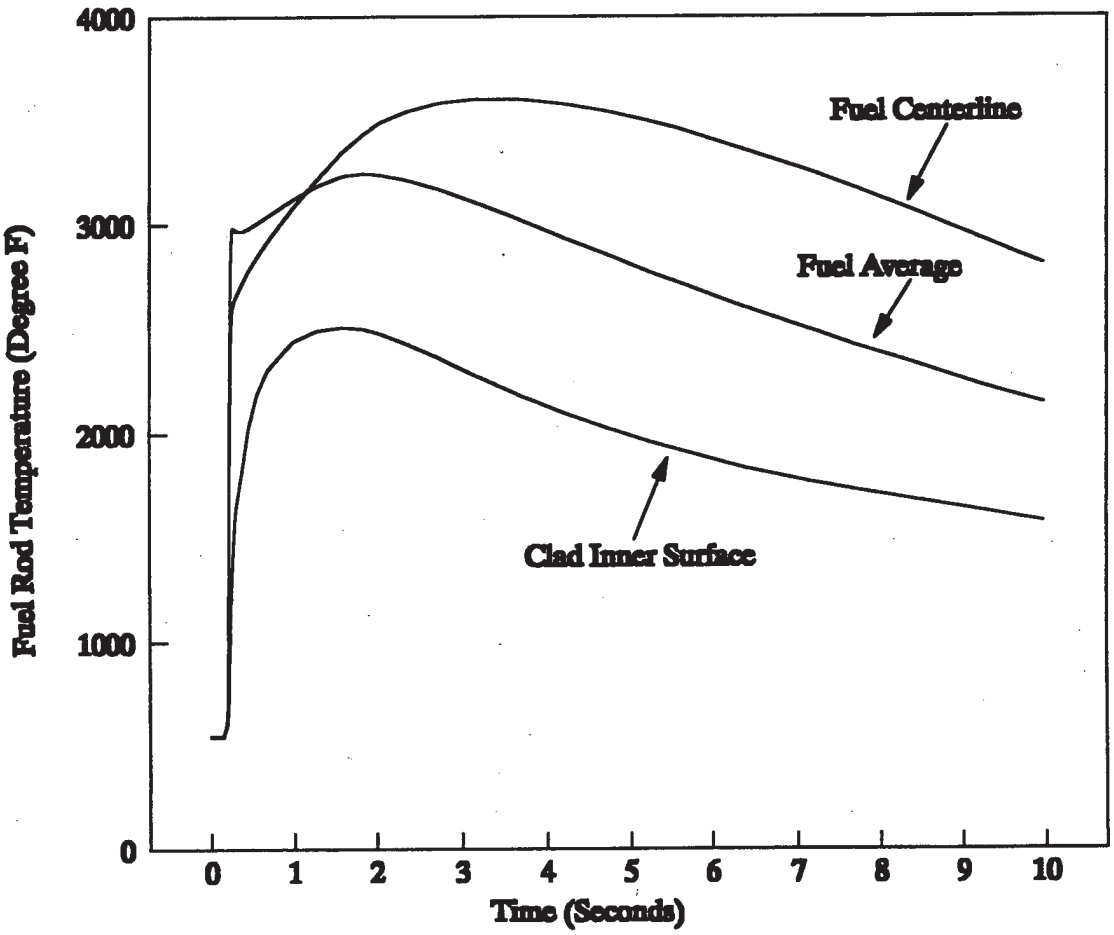


Figure 5.1.21-2

Nuclear Power for Rod Ejection

End of Life - Hot Zero Power

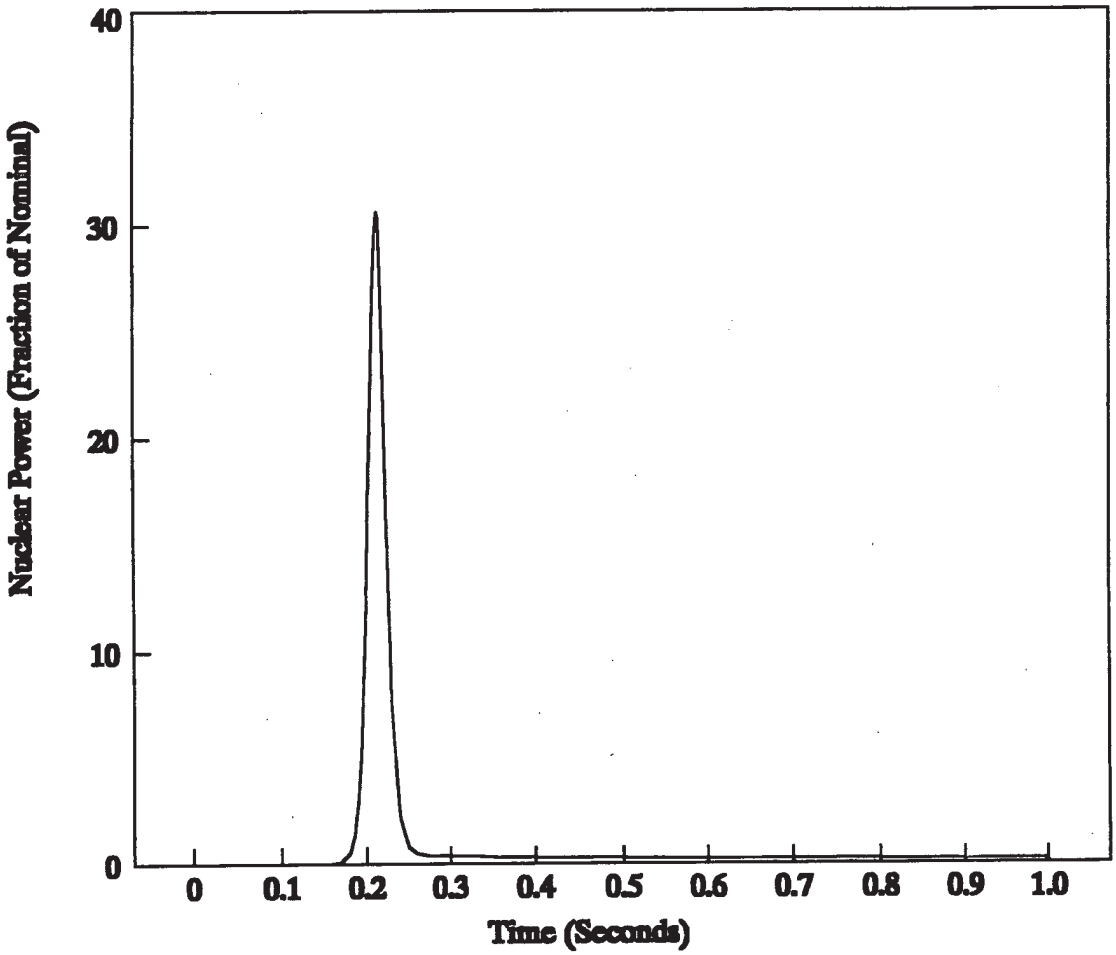


Figure 5.1.21-3

Fuel Rod Temperatures for Rod Ejection

Beginning of Life - Hot Full Power

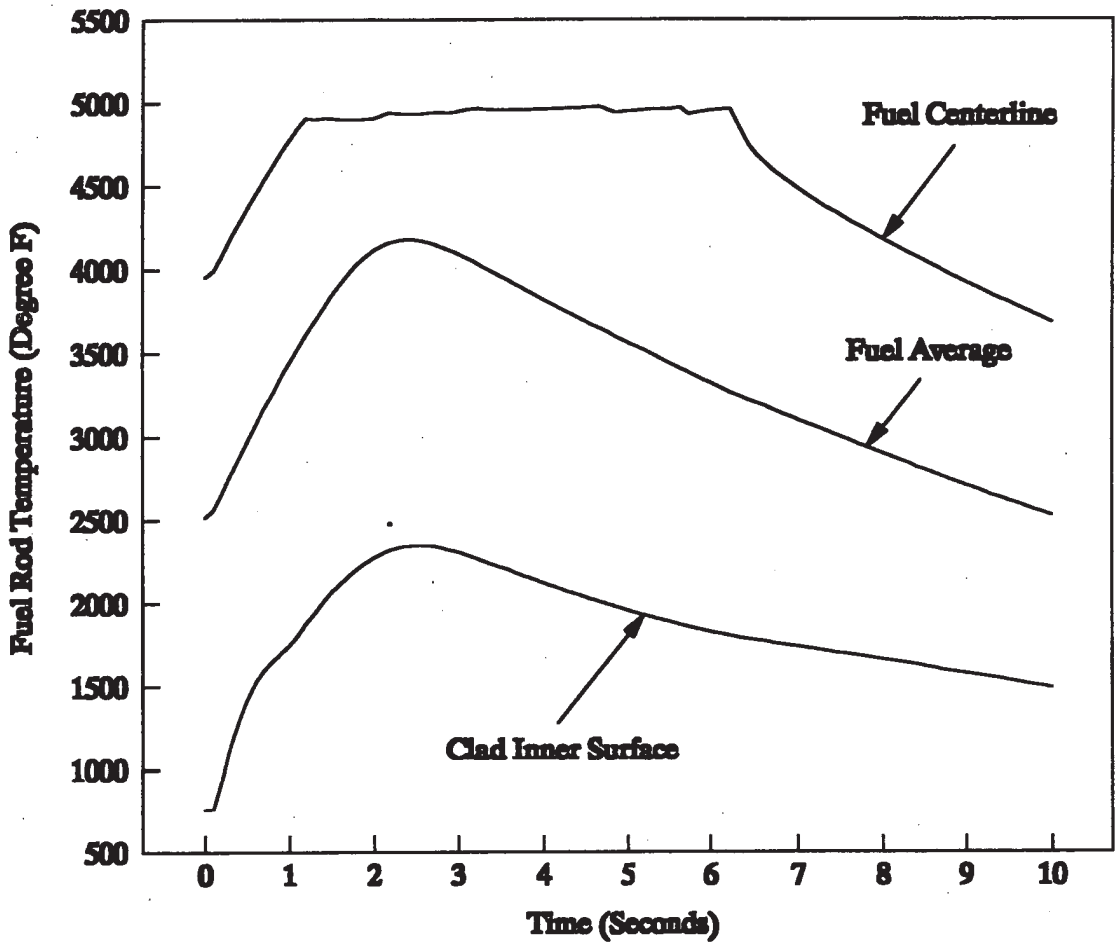
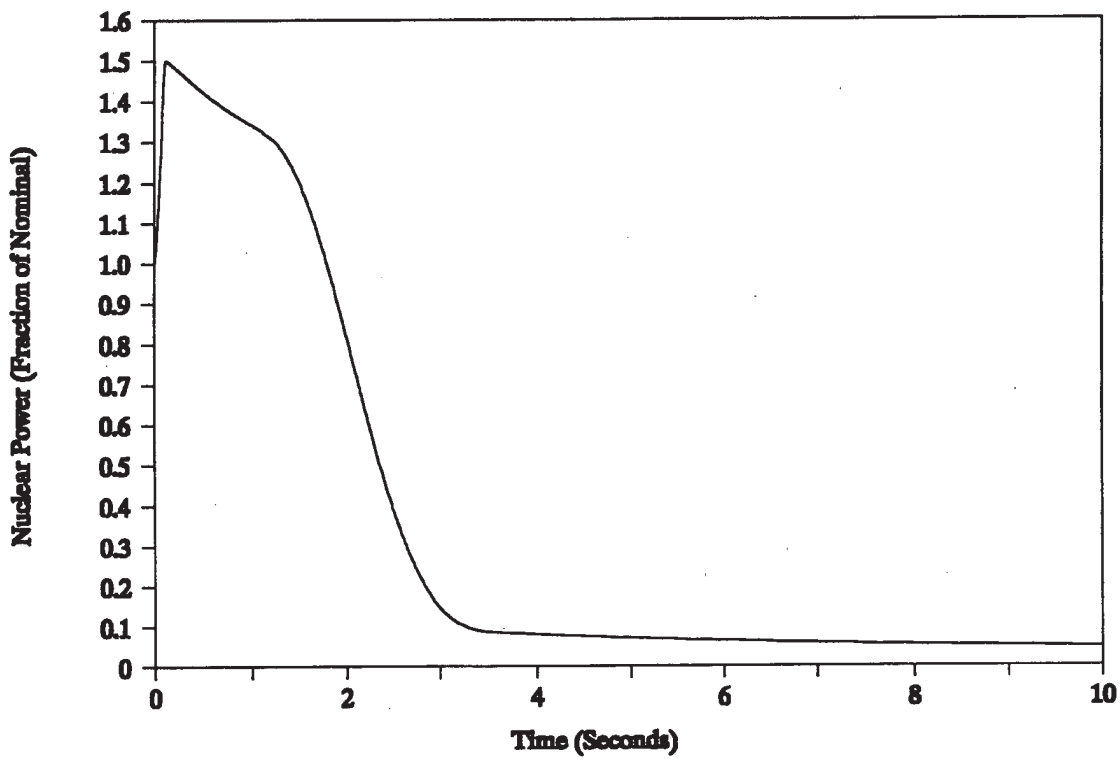


Figure 5.1.21-4

Nuclear Power for Rod Ejection

Beginning of Life - Hot Full Power



5.1.22 Conclusions

The results of the FSAR Chapter 14 Non-LOCA accident reanalyses and evaluations are contained in Sections 5.1.6 through 5.1.21. Based on the plant operating limitations given in 1) the Core Operating Limits Report, 2) the Technical Specifications, and, 3) the proposed Technical Specification changes given in Appendix A of this report, the results show that the transition from 15x15 VANTAGE 5 (w/o IFMs) to 15x15 VANTAGE + fuel, including the design changes discussed in Sections 5.1.2 and 5.1.3, can be accommodated with margin to the applicable FSAR safety analysis limits.

Therefore, it is concluded that Indian Point Unit 3 can safely operate with cores containing a combination of Westinghouse 15x15 VANTAGE 5 and VANTAGE + assemblies at conditions up to the current rated power of 3025 MWt with a maximum vessel average temperature of 574.7 °F with a maximum uniform steam generator tube plugging level of ≤ 24% uniform steam generator tube plugging and 570.5 °F under asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of ≤ 24% and with a maximum steam generator tube plugging level of 30% in any one steam generator. The latter 570.5 °F vessel average temperature corresponding to asymmetric steam generator tube plugging is to assure that the vessel inlet temperature from the hottest loop does not exceed a design value of 543.6 °F.

The above conclusions are valid for all of the VANTAGE + design features and associated changes in the safety analysis assumptions as described in Sections 5.1.2 and 5.1.3 with the exception of the increased uncertainties associated with the New York Power Authority program to extend surveillance intervals in support of 24 month operating cycles as discussed in Section 5.1.3.

The changes associated with the New York Power Authority program to extend surveillance intervals in support of 24 month operating cycles have not been validated for those events identified in Section 5.1.1 which were not reanalyzed in support of the transition to VANTAGE + fuel. As described in Section 5.1.3, only the events specifically requiring reanalysis as a direct result of the VANTAGE + fuel and associated changes to the safety analysis assumptions include the changes in analysis assumptions necessary to support the changes for 24 month cycles.

However, the New York Power Authority extended surveillance interval program is an ongoing effort currently being performed and validation of the acceptability with respect to all the non-LOCA events will be performed prior to such operation.

5.2 LOCA Accidents

5.2.1 Large Break LOCA

5.2.1.1 Description of Analysis Assumptions for 15x15 VANTAGE + Fuel

The Large Break Loss-Of-Coolant Accident (LOCA) analysis for Indian Point Unit 3 applicable to a full core of VANTAGE + fuel was performed using a modified version of the NRC approved 1981 Evaluation Model with BART/BASH (References 45 and 46). Modifications were made to the Large Break Evaluation Model computer codes to represent ZIRLO™ cladding as discussed in Section 5.2.1 of Reference 1. The analysis was performed for a spectrum of Moody discharge coefficients (0.4, 0.6 and 0.8) based on a limiting double-ended guillotine break of the RCS cold leg. Other pertinent analysis assumptions include: licensed power of 3025 MWt, 30% uniform steam generator tube plugging, thermal design flow of 323,600 gpm, total peaking factor of 2.42, and a hot channel enthalpy rise factor of 1.70. The effect of extended burnup of the VANTAGE + fuel on the results of the Large Break LOCA analysis was specifically considered.

The fuel rod diameter is the same for the 15x15 VANTAGE 5 (w/o IFMs) and 15x15 VANTAGE + fuel rods. The fuel assembly grids incorporated into the 15x15 VANTAGE + design include IFMs and LPD mid-grids. These grids were modeled in the analysis, and their effects are reflected in the results. The following additional fuel features have been evaluated for possible effects on the Large Break LOCA analysis and have been determined not to result in changes in the analysis results: the optimized fuel rod plenum spring, low cobalt top and bottom nozzles, debris-resistant oxide coating, ZIRLO™ guide thimble tubes, ZIRLO™ instrumentation tubes, mid-enriched annular pellets in axial blankets and the protective bottom grid with elongated bottom end plug/external grippable top end plug. An additional feature of the 15x15 VANTAGE + fuel, the Integral Fuel Burnable Absorber (IFBA), has been specifically analyzed for Indian Point Unit 3. This analysis has determined that the 15x15 VANTAGE + fuel with IFBA proposed for the use in Indian Point Unit 3 will result in lower Large Break LOCA peak clad temperatures than were calculated for the same fuel without the IFBAs. Thus the results of this analysis bound the 15x15 VANTAGE + fuel with the presence of IFBAs.

The Large Break LOCA analysis performed and documented in the FSAR updates assumed a full core of 15x15 VANTAGE + fuel. When assessing the effect of transition cores on the Large Break LOCA analysis, it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than a complete core of either the 15x15 VANTAGE 5 (w/o IFMs) design or the 15x15 VANTAGE + design. For a given peaking, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution

due to fuel assembly hydraulic resistance mismatch. This hydraulic resistance mismatch may exist only for transition cores and is the only unique difference between a complete core of either fuel type and the transition core. The VANTAGE + fuel design differs hydraulically from the VANTAGE 5 (w/o IFMs) fuel design it replaces. The transition at Indian Point Unit 3 from VANTAGE 5 (w/o IFMs) fuel to VANTAGE + fuel is similar to a transition from OFA fuel to VANTAGE 5 (w/ IFMs) fuel. This is because the addition of the IFM grids increases the hydraulic resistance of the VANTAGE + fuel assembly. However, the difference in the total assembly hydraulic resistance between the two designs is less than 10% (the VANTAGE + design has higher loss coefficient).

Generic analyses have been performed to determine bounding PCT effects for transition cores. These analyses are documented in Reference 10. Based upon the results reported in Reference 10, the increase in hydraulic resistance for the Indian Point Unit 3 VANTAGE + assembly will result in a maximum PCT penalty of 50 °F on the VANTAGE + fuel. Once a full core of VANTAGE + fuel is achieved, the Large Break LOCA analysis will apply without the transition core penalty.

5.2.1.2 Methods of Analysis

The methods of analysis including code descriptions and assumptions are described in detail in the FSAR updates.

5.2.1.3 Results

A more detailed presentation of the analytical results including tabular and plotted results for all analyzed cases is contained in the FSAR updates.

5.2.1.4 Conclusions

For breaks up to and including the doubled ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria of 10 CFR 50.46. That is:

1) The calculated peak fuel element clad temperature does not exceed the requirement of 2200 °F.

2) The amount of fuel element cladding, that reacts chemically with water or steam does not exceed one percent of the total amount of zircaloy in the reactor.

3) The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4) The core remains amenable to cooling during and after the break.

5) The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat from the long-lived radioactivity in the core.

The time sequence of events for all breaks analyzed is shown in Table 14.3-3 contained in the FSAR updates.

The Large Break LOCA analysis for Indian Point Unit 3, utilizing the BASH model, resulted in a peak clad temperature of 1922 °F for the limiting break case (C_D = 0.4 under minimum safeguards assumptions) at a total peaking factor of 2.42. The maximum local metal-water reaction was 3.5 percent, and the total metal-water reaction was less than 1.0 percent for all cases analyzed. The clad temperature turned around at a time when the core geometry is still amenable to cooling. Criterion 5 is addressed separately in a specific evaluation for each reload cycle. An Evaluation of the ZIRLO™ clad fuel rods at extended burnups has concluded that the beginning-of-life remains the most limiting for the Large Break LOCA analysis PCT calculation.

The effects of Containment Pressure Relief, steam generator tube collapse due to combined Safe Shutdown Earthquake (SSE) and LOCA events, and increased T_{avg} uncertainty have been accounted for, resulting in PCT penalties of 1 °F, 20 °F and 3 °F respectively. The addition of these penalties brings the PCT to 1946 °F.

Westinghouse had previously performed an evaluation (Reference 53) to support the removal of upper core plate alignment pins for Indian Point Unit 3. This evaluation was performed to support Cycle 8 operation and was later re-validated to support Cycle 9 operation. Since this evaluation is re-validated on a cycle specific basis, no generic statements can be made here to support VANTAGE + fuel for all future cycles. Rather, the re-validation of the safety evaluation to support Cycle 10 operation will be performed in recognition of the fact that VANTAGE + fuel will be loaded into the Cycle 10 core. The current 5 °F PCT penalty will continue to be tracked on the Indian Point Unit 3 PCT margin assessment form until such time that the re-evaluation is completed.

The effect of the transition core cycles is conservatively evaluated to be at most a 50 °F increase in the calculated PCT. The transition core penalty can be accommodated by the margin to the 10 CFR 50.46 2200 °F limit.

It can be seen from the results of this Large Break ECCS analysis that Indian Point Unit 3 remains in compliance with the requirements of 10 CFR 50.46.

5.2.2 Small Break LOCA

5.2.2.1 Description of Analysis/Assumptions

The Small Break loss-of-coolant accident was analyzed using axial power shapes consistent with the peaking factor limits assumed for the transition and post-transition cores. A modified version of the NRC-approved NOTRUMP Small Break ECCS Evaluation Model (Reference 47) was employed to analyze a spectrum of cold leg break sizes (1.5 in., 2 in. and 3 in. equivalent diameter). Modifications were made to the Small Break LOCA Evaluation Model computer codes to represent ZIRLO™ cladding as described in Reference 3. The FSAR updates contains a full description of the conditions and assumptions utilized for the Small Break LOCA analysis. The effect of extended burnup of the VANTAGE + fuel on the results of the Small Break LOCA analysis was specifically considered.

The fuel rod diameter is the same for the 15x15 VANTAGE 5 (w/o IFMs) and 15x15 VANTAGE + fuel rods. The fuel assembly grids incorporated into the 15x15 VANTAGE + design include IFMs and LPD mid-grids. Westinghouse has evaluated these grids and determined that they will have no effect on the Small Break LOCA analysis. Additional features of the 15x15 VANTAGE + fuel design (optimized fuel rod plenum spring, low cobalt bottom nozzle, debris-resistant oxide coating, ZIRLO™ guide thimble tubes, mid-enriched annular pellets in axial blankets and ZIRLO™ instrumentation tubes) have been evaluated for possible effects on the Small Break LOCA analysis and have been determined not to result in changes in the Small Break LOCA analysis results. An additional feature of the 15x15 VANTAGE + fuel, the Integral Fuel Burnable Absorber (IFBA), has been specifically analyzed for Indian Point Unit 3. This analysis has determined that the 15x15 VANTAGE + fuel with IFBA proposed for the use in Indian Point Unit 3 will result in essentially the same Small Break LOCA calculated peak clad temperatures than have been calculated based on the same fuel without the IFBAs. Thus, the results of this analysis are applicable to the 15x15 VANTAGE + fuel with or without the presence of IFBAs.

The NOTRUMP computer code, described in Reference 48, is used to model the core hydraulics during

a Small Break LOCA event. Since the core flow during the Small Break LOCA transient is relatively slow, flow equilibrium between fuel assemblies is maintained (i.e., no cross flow). Therefore, hydraulic resistance mismatch is not a factor for the Small Break LOCA. Thus, it is not necessary to perform a Small Break evaluation for transition cores, and it is sufficient to reference the Small Break LOCA analysis for the complete core of VANTAGE + fuel as bounding for all transition cycles.

5.2.2.2 Method of Analysis

The methods of analysis including code descriptions and assumptions are described in detail in the FSAR updates.

5.2.2.3 Results

A more detailed presentation of the analytical results including tabular and plotted results for all analyzed cases is contained in the FSAR updates.

5.2.2.4 Conclusions

In the event of a Small Break LOCA the emergency core cooling system provides adequate core protection by satisfying the acceptance criteria of 10 CFR 50.46. That is:

1) The calculated peak fuel element clad temperature does not exceed the requirement of 2200 °F.

2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of zircaloy in the reactor.

3) The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4) The core remains amenable to cooling during and after the break.

5) The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat from the long-lived radioactivity in the core.

The time sequence of events for all breaks analyzed is shown in Table 14.3-8 contained in the FSAR updates.

The Small Break LOCA analysis for Indian Point Unit 3, utilizing the NOTRUMP Evaluation model, resulted in a peak clad temperature of 1476.0 °F for the 2 inch diameter cold leg break. An evaluation performed to address an increase in the uncertainty on T_{avg} yielded a 1 °F PCT penalty. The analysis assumed the limiting Small Break LOCA power shape consistent with a LOCA FQ(z) envelope of 2.42 at the core midplane elevation and 2.24 at the top of the core. The maximum local metal-water reaction was 0.593 percent, and the total metal-water reaction was less than 1.0 percent for all cases analyzed. The clad temperature excursion was calculated to turn around at a time when the core geometry is still coolable. Criterion 5 is addressed separately in a specific evaluation for each reload cycle. An evaluation of the ZIRLO™ clad fuel rods at extended burnups has concluded that the beginning-of-life remains the most limiting for the Small Break LOCA analysis PCT calculation.

It can be seen from the results of this small break ECCS analysis that Indian Point Unit 3 remains in compliance with the requirements of 10 CFR 50.46.

5.2.3 Blowdown Reactor Vessel and Loop Forces

The forces created by a hypothetical break in the RCS piping are principally caused by the motion of the decompression wave through the RCS. The strength of the decompression wave is primarily a function of the assumed break opening time, break area, and RCS operating conditions of power, temperature, and pressure. These parameters will not be affected by the change in fuel at Indian Point Unit 3 from VANTAGE 5 (w/o IFMs) to VANTAGE +. The forcing functions resulting from a LOCA in the vicinity of the core are affected by the core flow area/volume. However, the transition from VANTAGE 5 (w/o IFMs) fuel to VANTAGE + fuel will not result in a change in the core flow area/volume. Thus, the forcing functions resulting from a LOCA will not be increased as a result of the implementation of VANTAGE + fuel.

Forces acting on the RCS loop piping as a result of the postulated LOCA are not influenced by changes in fuel assembly design. Thus, the implementation of VANTAGE + fuel at Indian Point Unit 3 will not result in an increase of the calculated consequences of a postulated LOCA on the reactor vessel internals or RCS loop piping. The current FSAR analysis for forces on the reactor internals and RCS piping resulting from a postulated LOCA are considered to be bounding to the application of VANTAGE + fuel at Indian Point Unit 3.

5.2.4 Post LOCA Long-Term Cooling, Subcriticality Evaluation

The Westinghouse licensing position for satisfying the requirements of 10 CFR 50.46 Paragraph (b)

Item (5) "Long-Term Cooling" is defined in References 49 and 50 and in NRC Technical Bulletin NSID-TB-86-08. The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water alone after a LOCA. Since credit for the control rods is not taken for a Large Break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods out.

Since the use of VANTAGE + grids (including the associated increase in design rod drop time) and thimble plug removal will have no effect on the sources of borated and non-borated water assumed in the long term cooling calculation, it is concluded that there would be no change to the long term cooling capability of the ECCS system. Further, this licensing commitment is verified by Westinghouse on a cycle by cycle basis, ensuring compliance with this requirement independent of this safety evaluation.

Note that the effect of the uncertainties associated with 24 month cycles has not been considered with respect to this analysis. As stated above, the transition to VANTAGE + fuel in itself will have no effect on this analysis. However, the uncertainties associated with 24 month cycle operation may change the results of this analysis. Upon completion of the evaluation supporting 24 month cycle operation, Westinghouse will report the new results of this analysis, if applicable.

5.2.5 Hot Leg Switchover to Prevent Potential Boron Precipitation/Long Term SI Verification

Post-LOCA hot leg recirculation switchover time is determined for the inclusion in emergency procedures to ensure that no boron precipitation occurs in the reactor vessel following a LOCA. This recirculation switchover time is dependent on power level, RCS water volume and boron concentration, RWST water volume and boron concentration, accumulator water volume and boron concentration. The implementation of VANTAGE + fuel will have no effect on the assumptions for the RCS, RWST, and accumulators in the hot leg switchover calculation. Thus, there is no effect on the post-LOCA hot leg switchover time.

Note that the effect of the uncertainties associated with 24 month cycles has not been considered with respect to this analysis. As stated above, the transition to VANTAGE + fuel in itself will have no effect on this analysis. However, the uncertainties associated with 24 month cycle operation may change the results of this analysis. Upon completion of the evaluation supporting 24 month cycle operation, Westinghouse will report the new results of this analysis, if applicable.

5.3 Radiological Impact Assessment

The change from VANTAGE 5 fuel to VANTAGE + fuel for Indian Point Unit 3 has been reviewed to determine if there are any changes that would impact the radiological consequences of accidents that have been analyzed as part of the licensing basis. For non-LOCA accidents there has been no determination of increased fuel damage nor any increase in mass releases that would invalidate current radiological analysis assumptions. The Large Break LOCA radiological consequences are not based on a mechanistic determination of fuel failures and are thus insensitive to fuel design features.

Only two of the changes associated with the VANTAGE + fuel design have been identified as having the potential for impact on the radiological consequences of accidents. One is the increase in the radial peaking factor and the second is the increase in fuel burnup limits.

5.3.1 Radial Peaking Factor

As part of the VANTAGE + fuel design, the radial peaking factor is being increased to 1.70 for the VANTAGE + fuel only. The radial peaking factor is used as radiological consequences analysis input only for the Fuel Handling Accident (FHA) with Regulatory Guide 1.25 stating that a radial peaking factor of 1.65 or greater should be used. Since the fuel design now exceeds the minimum value specified in Regulatory Guide 1.25, the design value was used to reanalyze the Fuel Handling Accident doses.

5.3.2 Extended Fuel Burnup - Core Source Term

With the transition to VANTAGE + fuel, there is an associated increase in the maximum fuel burnup limit. The design burnup for VANTAGE + fuel is a peak fuel pin burnup up to 75,000 MWD/MTU which is an increase over the current peak fuel pin burnup of 62,000 MWD/MTU.

The extension of fuel burnup has been shown to have negligible impact on the core inventory of radioactive isotopes which are of concern in evaluating the radiological consequences of accidents (i.e., the short half-life noble gases and iodines). Compared with fuel operated without extended burnup limits (defined by the NRC as a region average discharge burnup of less than 38,000 MWD/MTU), all the short-lived iodine isotopes are found to decrease slightly as a result of operation to the extended burnup level, and the short-lived noble gases, with the exception of Xe-133 and Xe-135, show a decreased core inventory associated with the extended fuel burnup. The inventory of Kr-85 increases significantly with

extended burnup because of its long half-life but this nuclide is a weak gamma emitter and contributes little to the calculated doses. The core source term impacts are documented by Westinghouse in a topical report evaluating the VANTAGE + fuel design[1].

The impact of the changes in the core source terms defined in Reference 1 on the radiological consequences of accidents is to slightly decrease the thyroid doses and to also slightly decrease the whole body doses for most accidents. For accidents involving releases after a long period of decay (e.g. the latter stages of the LOCA and the Fuel Handling Accident), there is a slight increase in the whole body doses; these increases have been determined to be insignificant, being within the uncertainty of calculational assumptions. Thus the changes in the core source term do not require recalculation of accident doses.

Independent review performed for the NRC reached the conclusion that, for burnups up to 60,000 MWD/MTU, the core inventory of short-lived nuclides does not increase[51]. While the review performed for the NRC does not address burnups beyond 60,000 MWD/MTU for the lead rod, the Westinghouse evaluation[1][54] demonstrates that there is no significant change in the core source term as a result of the increase in burnup up to 75,000 MWD/MTU.

5.3.3 Extended Fuel Burnup - Fission Product Inventory in the Fuel/Clad Gap

Also of concern in evaluating the impact of extended fuel burnup on the radiological consequences of accidents is the fraction of core activity that is assumed to migrate out of the fuel matrix and into the fuel/clad gap region and thus is available for release in the event the cladding is breached. The maximum gap fractions determined for the VANTAGE + fuel with peak fuel pin burnup up to 75,000 MWD/MTU[1] are bounded by the values presented in Table 14.3-12 of the Indian Point Unit 3 FSAR.

In the NRC guidance for calculating Fuel Handling Accident doses (Regulatory Guide 1.25), the gap fraction specified for short-lived nuclides (i.e., nuclides with a half-life of less than a year) is ten percent. As indicated by the gap fractions presented in Reference 1, the Regulatory Guide 1.25 gap fractions remain conservative for the VANTAGE + fuel.

Reference 51 agrees that, with extended fuel burnup, the fuel gap fractions for nuclides of concern for accident doses do not exceed the gap fractions specified in Regulatory Guide 1.25. However,

Reference 51 takes an exception for the Fuel Handling Accident and specifies that the I-131 gap fraction could be as high as 12 percent (an increase of 20 percent over the Regulatory Guide 1.25 value). This increase is inconsistent with the Westinghouse topical report[1] which shows a maximum gap fraction for I-131 of less than two percent.

5.3.4 Reanalysis of the Fuel Handling Accident Doses

The Fuel Handling Accident doses have been reanalyzed to reflect the change in the radial peaking factor discussed in 5.3.1 above and the increase in the I-131 gap fraction identified by Reference 51 (see 5.3.3 above). The full list of assumptions used in the analysis are provided in Table 5.3-1.

The resulting doses are:

Site Boundary thyroid dose	269 rem
Site Boundary whole body dose	0.6 rem
Time at which fuel handling in containment does not require filtered release	421 hours

These doses are within the 10 CFR 100 maximum dose guidelines of 300 rem thyroid and 25 rem whole body. The doses are applicable to the postulated Fuel Handling Accident either inside containment or in the spent fuel pit since there is no credit taken for isolation of the containment nor for the filters in either the containment purge line or the fuel storage building emergency ventilation system. If credit is taken for removal of iodine by the charcoal filters (90 percent efficiency for elemental iodine and 70 percent efficiency for organic iodine), the thyroid dose is reduced to 41 rem.

With the above calculated doses, Technical Specifications 3.8.A.9 and 4.13 will have to be revised to require operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal absorbers at any time there is movement of the fuel inside containment and the reactor has been subcritical for less than 421 hours. This time period ensures that the site boundary thyroid dose for the Fuel Handling Accident postulated to occur inside containment would not exceed 100 rem. The 421 hour value is is an increase from the 365 hours presently in the Technical Specifications.

The above calculated doses use the Reference 51 guidance regarding the I-131 gap fraction (see 5.3.3

above), resulting in a 20 percent increase in the accident thyroid doses over what they would be if calculated using the I-131 gap fraction from Regulatory Guide 1.25. This assumption has been included in the analysis because the NRC has taken the position to include this increase in Fuel Handling Accident thyroid doses when a plant implements extended fuel burnup (defined as being burnup levels exceeding 38,000 MWD/MTU for the discharge batch average burnup). However, the Fuel Handling Accident dose analysis assumes that the subject fuel assembly has been operating at the maximum rod radial peaking factor of 1.70 times the core average power level. The bounding power history curve for the fuel precludes operation of extended burnup fuel at such a high power level. By the time fuel has experienced 50,000 MWD/MTU burnup, the bounding power level is about 1.3 times core average. This reduction in radial peaking factor more than compensates for the 20 percent increase in the I-131 gap fraction identified by Reference 51. At 75,000 MWD/MTU burnup, the bounding power level is below core average power. Thus, if the increase in I-131 gap fraction and the bounding power history curve were both implemented, there would be a net reduction in the calculated doses for extended burnup fuel. The limiting case for the Fuel Handling Accident is when the mishandled assembly is one that has been operating at the maximum power level; this will not be a high burnup assembly.

Table 5.3-1

Assumptions and Parameters for the Fuel Handling Accident Analysis

1. Core inventory at shutdown is as defined by FSAR Table 14.3-12:

I-131	8.07E7 Curies
I-132	1.226E8
I-133	1.811E8
I-134	2.117E8
I-135	1.641E8
Kr-85m	3.57E7
Kr-85	1.74E6
Kr-87	6.85E7
Kr-88	9.74E7
Xe-133m	4.63E7
Xe-133	1.834E8
Xe-135m	2.71E7
Xe-135	4.94E7

2. Number of assemblies in the reactor core = 193.
3. All fuel pins in the dropped fuel assembly are broken.
4. Decay time experienced prior to fuel movement = 145 hrs (consistent with Tech Spec 3.8.A.9).
5. Atmospheric dispersion factor at site boundary = 1.03E-3 sec/m^3 (per page 14.2-7 of the Indian Point Unit 3 FSAR).
6. Operating power in the damaged assembly is 1.70 times core average (this is the design radial peaking factor for the VANTAGE+ fuel).
7. The fission product gap fractions are assumed to be ten percent for all nuclides except Kr-85 which is assumed to be 30 percent and I-131 which is assumed to be 12 percent. This is consistent with Reg. Guide 1.25 as modified by NUREG/CR-5009.
8. The iodine is 99.75 percent elemental and 0.25 percent organic (consistent with Reg. Guide 1.25).
9. There is scrubbing removal of the elemental iodine in the water pool. The decontamination factor achieved by scrubbing is assumed to be 133 (consistent with Reg. Guide 1.25).
10. The breathing rate is 3.47E-4 m^3/sec (consistent with Reg. Guide 1.25).
11. For the accident postulated to occur in the spent fuel pit, no credit is taken for the fact that releases to the environment would pass through charcoal filters.
12. For the accident postulated to occur inside the containment, credit is not taken for isolation of the containment purge.
13. For the accident postulated to occur inside the containment, no credit is taken for the fact that releases to the environment would pass through a charcoal filter.
14. The nuclide decay constants, average gamma energy per disintegration, and thyroid dose conversion factors are those given in FSAR Table 14.3-14.

6.0 ASSESSMENT OF NO UNREVIEWED SAFETY QUESTIONS (10 CFR 50.59 Screening Criteria)

From the evaluation presented in this report, it is concluded that the Indian Point Unit 3 VANTAGE + transition core design does not result in the acceptable safety limits for any accident being exceeded and does not result in any unreviewed safety questions as defined in 10 CFR 50.59. The basis for this determination is delineated below. The VANTAGE + design criteria are referenced throughout this document.

1. Will the probability of an accident previously evaluated in the FSAR be increased?

This RTSR documents that the probability of an accident previously evaluated in the Indian Point Unit 3 FSAR[18] **is not increased.** The VANTAGE + transition core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. Though fuel and core design are not directly related to the probability of any previously evaluated accident, the demonstrated adherence to applicable standards and acceptance criteria precludes new challenges to components and systems that could increase the probability of any previously evaluated accident. Specifically, the mechanical changes as specified in Section 2.0 will not increase the probability of occurrence of an accident previously evaluated in the Indian Point Unit 3 FSAR[18]. The clad integrity is maintained and the structural integrity of the fuel rods, fuel assemblies, and core is not affected. The mechanical features, noted in Section 2.0, have no impact on fuel rod performance or dimensional stability as documented in this RTSR and Reference 11, nor will they cause the core to operate in excess of pertinent design basis operating limits. Therefore, the probability of occurrence of an accident previously evaluated in the FSAR[18] **has not increased.**

2. Will the consequences of an accident previously evaluated in the FSAR be increased?

This RTSR documents that the radiological consequences of an accident previously evaluated in the Indian Point Unit 3 FSAR[18] **is not increased.** The VANTAGE + transition core design does not have a direct role in mitigating the radiological consequences of any accident, and does not affect any of the bases (assumptions, actions, etc) for the current analyses as described in the Indian Point Unit 3 FSAR[18]. The VANTAGE + transition core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. The demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could: a) adversely affect the ability of existing components and systems to mitigate the radiological consequences of any accident and/or; b) adversely affect the integrity of the fuel rod cladding as a fission product barrier (refer to Reference 11 for ZIRLO™ cladding

approval). Furthermore, adherence to applicable standards and criteria ensures that these fission product barriers maintain design margin to safety. Specifically, the mechanical changes as specified in Section 2.0 will not increase the radiological consequences of an accident previously evaluated in the Indian Point Unit 3 FSAR[18]. The mechanical features, noted in Section 2.0, have no impact on chemical, physical or mechanical properties as documented in this RTSR and Reference 11, nor will they cause the core to operate in excess of pertinent design basis operating limits. Thus, clad integrity is maintained. Since the predictions presented in the FSAR[18] are not sensitive to the mechanical changes specified in this report, the radiological consequences of accidents previously evaluated in the Indian Point Unit 3 FSAR[18] **have not increased.**

3. <u>May the possibility of an accident which is different from any already in the FSAR be created?</u>

This RTSR documents that the possibility of an accident which is different from any already in the Indian Point Unit 3 FSAR[18] **is not created.** The VANTAGE + transition core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. The demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. Specifically, the mechanical changes as specified in Section 2.0 will, not create the possibility of an accident of a different type than any previously evaluated in the Indian Point Unit 3 FSAR[18]. The fuel assemblies containing the mechanical features noted in Section 2.0 will satisfy the same design bases[1] as that used for fuel assemblies in the other fuel regions. All design and performance criteria will continue to be met and no new single failure mechanisms have been created as documented in this RTSR and Reference 11, nor will they cause the core to operate in excess of pertinent design basis operating limits. Therefore, the possibility of an accident of a different type than any previously evaluated in the FSAR[18] **has not been created.**

4. <u>Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?</u>

This RTSR documents that the probability of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[18] **is not increased.** The VANTAGE + transition core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. Demonstrated adherence to applicable standards and acceptance criteria precludes new challenges to components and systems that could increase the probability of any previously evaluated malfunction of equipment important to safety. Specifically, the mechanical changes as specified in Section 2.0, in compliance with the methodology established in Reference 1, will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[18]. No new performance requirements are being imposed on any system or component such that any design criteria will be exceeded as documented in this RTSR and Reference 11, nor will they cause the

core to operate in excess of pertinent design basis operating limits. No new modes or new limiting single failures have been created with the mechanical features noted above. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the FSAR[18] **has not increased**.

5. Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

This RTSR documents that the radiological consequences of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[18] **are not increased.** The VANTAGE + transition core design does not have a direct role in mitigating the radiological consequences of any malfunction of equipment important to safety, and does not affect any of the bases (assumptions, actions, etc) for the current analyses as described in the Indian Point Unit 3 FSAR[18]. The VANTAGE + transition core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. The demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could: a) adversely affect the ability of existing components and systems to mitigate the radiological consequences of any accident and/or; b) adversely affect the integrity of the fuel rod cladding as a fission product barrier (refer to Reference 11 for ZIRLO™ cladding approval). Furthermore, adherence to applicable standards and criteria ensures that these fission product barriers maintain design margin of safety. Specifically, the mechanical changes as specified in Section 2.0 will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[18]. The predictions presented in the FSAR[18] are not sensitive to the fuel rod cladding material or other mechanical changes that do not alter the metallurgical composition of the core (refer to Reference 11 for ZIRLO™ cladding approval). The mechanical features mentioned in Section 2.0, do not change the performance requirements on any system or component such that any design criteria will be exceeded as documented in this RTSR and Reference 11, nor will they cause the core to operate in excess of pertinent design basis operating limits. No new modes or new limiting single failures have been created with any of the mechanical features mentioned above. Therefore, the radiological consequences of a malfunction of equipment important to safety previously evaluated in the Indian Point Unit 3 FSAR[18] **have not increased.**

6. May the possibility of a malfunction of equipment important to safety different from any already evaluated in the FSAR be created?

This RTSR documents that the possibility of a malfunction of equipment important to safety different from any already evaluated in the Indian Point Unit 3 FSAR[18] **is not created.** The VANTAGE + transition core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. The demonstrated adherence to these

standards and criteria precludes new challenges to components and systems that could introduce a new type of a malfunction of equipment important to safety. Specifically, the mechanical changes as specified in Section 2.0 will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the Indian Point Unit 3 FSAR[18]. All original design and performance criteria continue to be met, and no new failure modes have been created for any system, component, or piece of equipment. No new single failure mechanisms have been introduced as documented in this RTSR and Reference 11, nor will they cause the core to operate in excess of pertinent design basis operating limits. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR[18] **has not been created.**

7. Will the margin of safety as defined in the BASES to any technical specifications be reduced?

This RTSR documents that the margin of safety as defined in the Bases to any Indian Point Unit 3 Technical Specifications[52] **is not reduced.** The VANTAGE + transition core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. It has been determined that the Indian Point Unit 3 VANTAGE 5 (w/o IFMs) to VANTAGE + fuel transition core design and safety analysis limits remain applicable, and that these limits will be supported by the applicable Indian Point Unit 3 Technical Specification[52] as amended herein. Specifically, the mechanical changes as specified in Section 2.0 will not reduce the margin of safety as defined in the basis for any technical specification. The use of these fuel assemblies will take into consideration cycle transition and reload cores. These fuel assemblies will be specifically evaluated using approved reload design methods[21] and fuel rod design models and methods[1][10][12][13][14][15]. This will include considerations of the core physics analysis peaking factors and core average linear heat rate effects. Therefore, the margin of safety as defined in the Bases to the Indian Point Unit 3 Technical Specifications[52] **has not been reduced.**

Based upon the preceding information, it can be concluded that there are no unreviewed safety questions identified as a result of the Indian Point Unit 3 VANTAGE + transition core design.

7.0 NO SIGNIFICANT HAZARDS CRITERIA EVALUATION

(10 CFR 50.92 Screening Criteria)

The use of the VANTAGE + fuel assemblies; containing fuel rods clad with ZIRLO™, thimble and instrumentation tubes fabricated with ZIRLO™ material, Intermediate Flow Mixer (IFM) grids and Low-Pressure-Drop (LPD) mid-grids; has been determined not to involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is as follows:

1) The probability of occurrence or the consequences of an accident previously evaluated is not significantly increased. The VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble and instrument tubes; IFMs, and LPD mid-grids meet the same fuel assembly and fuel rod design bases as VANTAGE 5 (w/o IFMs) fuel assemblies in the other fuel regions. In addition, the 10 CFR 50.46 criteria will be applied to the ZIRLO™ clad fuel rods, thimble and instrument tubes; IFM grids, and LPD mid-grids. The use of these fuel assemblies will not result in a change to the proposed Indian Point Unit 3 VANTAGE 5 (w/o IFMs) transition core design and safety analysis limits[11]. The ZIRLO™ clad material is similar in chemical composition and has similar physical and mechanical properties as that of Zircaloy-4. Thus the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. The ZIRLO™ clad fuel rod improves corrosion resistance and dimensional stability. In addition, the incorporation of LPD mid-grids and IFMs improves dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel assemblies material changes as specified in this report, the radiological consequences of accidents previously evaluated in the safety analyses remain valid. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2) The possibility for a new or different type of accident from any accident previously evaluated is not created, since the VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble and instrument tubes; IFMs, and LPD mid-grids will satisfy the same design bases[1][10] as that used for VANTAGE 5 (w/o IFMs) fuel assemblies in the other fuel regions. Since the original design criteria is being met, the ZIRLO™ clad fuel rods, thimble and instrument tubes; IFMs, and LPD mid-grids will not be an initiator for any new accident. All design and performance criteria will continue to be met and no new single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alterations to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3) The margin of safety is not significantly reduced, since the VANTAGE + fuel assemblies containing ZIRLO™ clad fuel rods, thimble and instrument tubes; IFMs, and LPD mid-grids do not change the proposed Indian Point Unit 3 VANTAGE 5 (w/o IFMs) transition core design and safety analysis limits[11]. The use of these fuel assemblies containing fuel rods, thimble and instrument tubes with ZIRLO™ cladding alloy; IFMs and LPD mid-grids will take into consideration the normal core operating conditions allowed in the Technical Specifications[52]. For the transition core and each future cycle reload core, these fuel assemblies will be specifically evaluated using standard reload design methods[21] and approved fuel rod design models and methods[1][10][12][13][14][15]. This will include consideration of the core physics analysis, peaking factors and core average linear heat rate effects. In addition, the 10 CFR 50.46 criteria will be applied each cycle to the ZIRLO™ clad fuel rods, thimble and instrument tubes; IFMs, and LPD mid-grids. Analyses or evaluations will be performed each cycle to confirm that 10CFR50.46 will be met. Therefore, the margin of safety as defined in the Bases to the Indian Point Unit 3 Technical Specifications[52] and VANTAGE 5 (w/o IFMs) ZIRLO™ licensing amendment approval[11] is not significantly reduced.

Based upon the preceding information, it has been determined that the proposed change to VANTAGE + fuel assemblies in the Technical Specifications[52] does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change to VANTAGE + fuel meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

8.0 REFERENCES

1. Davidson, S. L., Nuhfer, D. L. (Eds.), "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 and Appendices A through D, June 1990.

2. Burman, D. L., "Appendix E - ZIRLO™ High Temperature Oxidation Tests," WCAP-12610 Appendix E, August 1990.

3. Kachmar, M. P., Iyengar, J., and Shimeck, D. J., "Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications; Appendix G - Accident Evaluations LOCA Plant Specific," WCAP-12610 Appendices F and G, December 1990.

4. Davidson, S. L., Nuhfer, D. L. (Eds.), "Westinghouse Responses to NRC Request for Additional Information on WCAP-12610 VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 Addendum 1, February 1991.

5. Davidson, S. L., Nuhfer, D. L. (Eds.), "Westinghouse Responses to NRC Second Request for Additional Information on WCAP-12610 VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 Addendum 2, February 1991.

6. Davidson, S. L., Nuhfer, D. L. (Eds.), "Westinghouse Responses to NRC Additional Issues on WCAP-12610 VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 Addendum 3, February 1991.

7. Kachmar, M. P., Nissley, M., and Tauche, W., "Additional Information for Appendices F and G of WCAP-12610 Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications; Appendix G - Accident Evaluations LOCA Plant Specific," WCAP-12610 Addendum 4, May 1991.

8. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse): "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report'," July 1, 1991.

9. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse): "Acceptance for Referencing of Topical Report WCAP-12610, Appendices F, 'Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications', and G, 'Appendix G - Accident Evaluations LOCA Plant Specific'," October 9, 1992.

10. Davidson, S. L. (Ed.), et al., "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A and Appendix A, September 1985; Addendum 1-A, March 1986; Addendum 2-A, April 1988.

11. Letter from N. F. Conicella (NRC) to R. E. Beedle (NYPA), "Issuance of Amendment for Indian Point Nuclear Generating Unit No. 3 (TAC NO. M82567)," Docket No. 50-286, May 15, 1992.

12. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "WCAP-12610, Appendix B, 'Extended Burnup Fuel Design Methodology and ZIRLO™ Fuel Performance Models,' Addendum 1 [Proprietary]," ET-NRC-92-3732, August 21, 1992.

13. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

14. Miller, J. V. (Ed.), "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary) and WCAP-8785 (Non-Proprietary), October 1976.

15. Letter from J. F. Stolz (NRC) to T. M. Anderson (Westinghouse), "Safety Evaluation of WCAP-8720," February 9, 1979.

16. George, R. A., Lee, Y. C. and Eng, G. H., "Revised Clad Flattening Model," WCAP-8377, July 1974.

17. Davidson, S. L. (Ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (Proprietary), December 1985.

18. Final Safety Analysis Report - Indian Point Unit 3, Revision 8, Docket Number 50-286, updated July 1991.

19. Skaritka, J. (Ed.), "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1, July 1979.

20. Iorii, J. A. and Petrarca, D. J., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021-P-A, Revision 1, October 1983.

21. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-NP-A, July 1985.

22. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Presurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

23. Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

24. Letter from K. C. Hoskins (Westinghouse) to G. M. Canavan (NYPA), "New York Power Authority, Indian Point Unit 3, Final Criticality Analysis Report," 93IN*-G-0020, April 8, 1993; and Letter from K. C. Hoskins (Westinghouse) to G. M. Canavan (NYPA), "New York Power Authority, Indian Point Unit 3, Change Pages to Final Criticality Analysis Report," 93IN*-G-0030, June 18, 1993

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

26. Friedland, A. J. and Ray, S., "Improved THINC IV Modeling for PWR Core Design," WCAP-12330-P, August 1989.

27. Motley, F. E., et al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A, July 1984.

28. Tong, L. S., "Critical Heat Fluxes in Rod Bundles, Two Phase Flow and Heat Transfer in Rod Bundles," Annual Winter Meeting ASME, November 1968, p. 3146.

29. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Office of Information Services, TID-25887, 1972.

30. Davidson, S. L., Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

31. Letter from E. P. Rahe (Westinghouse) to J. R. Miller (NRC), "WCAP-9500 and WCAP-9401/9402 NRC SER Mixed Core Compatibility Items," NS-EPR-2573, March 19, 1982.

32. Schueren, P. and McAtee, K. R., "Extension of Methodology for Calculating Transition Core DNBR Penalties," WCAP-11837-P-A, January 1990.

33. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), "Supplemental Acceptance No. 2 for Referencing Topical Report WCAP-9500," January 1983.

34. Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-11837," October 1989.

35. Segletes, J. A., et al., "Safety Evaluation for Indian Point Unit 3 With Asymmetric Tube Plugging Among Steam Generators," WCAP-10704, Revision 1, January 1986.

36. Letter from J. D. Neighbors (NRC) to J. C. Brons (NYPA), "Issuance of Amendment (TAC No. 71938)," Docket No. 50-286, June 6, 1989.

37. Letter from N. F. Conicella (NRC) to R. E. Beedle (NYPA), "Issuance of Amendment for Indian Point Nuclear Generating Unit No. 3," TAC No. M82567, May 15, 1994.

38. Hargrove, H. G., "FACTRAN - A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

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

40. Risher, D. H., Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary), WCAP-8028-A (Non-Proprietary), January 1975.

41. Hochreiter, L. E., Chelemer, H., Chu, P. T., "THINC IV, An Improved Program for Thermal Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

42. Hochreiter, L. E., "Application of the THINC IV Program to PWR Design," WCAP-8054, October 1973.

43. Moody, F. S., "Transactions of the ASME, Journal of Heat Transfer," Figure 3, page 134, February 1965.

44. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurizer-Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Rev. 1-A, December 1971.

45. Kabadi, J. N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," WCAP-10266-P-A (Proprietary), March 1987.

46. Young, M., et al., "BART-1A: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561-P-A (Proprietary), March 1984.

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

48. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Non-Proprietary), August 1985.

49. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model Summary," WCAP-8339-NP-A, July, 1974.

50. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model: Supplementary Information," WCAP-8472-NP-A, April, 1975.

51. Baker,D. A., et al., "Assessment of the Use Of Extended Burnup Fuel in Light Water Reactors," NUREG/CR-5009, February 1988.

52. Technical Specifications - Indian Point Unit 3 Nuclear Power Plant, Docket No. 50-286, through Amendment No. 143, January 1994.

53. Letter from J. F. Hofscher, (Westinghouse) to G. Grochowski (NYPA), "New York Power Authority, Indian Point Unit 3, Revised Safety Evaluation Supporting Fuel Assembly Upper Alignment Pin Removal and Straightening," INT-90-647, December 7, 1990.

54. Davidson, S. L., "Westinghouse Fuel Criteria Evaluation Process", WCAP-12488-A, October, 1994.

ATTACHMENT IV TO IPN-96-128

AUTHORITY COMMITMENTS FOR PROPOSED TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH THE UPGRADE TO VANTAGE + FUEL

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

COMMITMENTS ASSOCIATED WITH IPN-96-128

Comm. No.	Commitment Description	Due Date
IPN-96-128-01	Revise COLR to reflect increase in F_Q and $F_{\Delta H}$.	Prior to startup from the 9/10 refueling outage.
IPN-96-128-02	Revise FSAR to incorporate amendment changes.	Next FSAR update which is at least 6 months after NRC approval of this amendment.