Public Service
Electric and Gas
Company

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MAY 10 1996

LR-N96114
LCR S94-41

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

**REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
MARGIN RECOVERY PROGRAM
SALEM GENERATING STATION NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311**

Gentlemen:

In accordance with 10CFR50.90, Public Service Electric & Gas Company (PSE&G) requests a revision to the Technical Specifications for Salem Generating Station Unit Nos. 1 and 2. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed Technical Specification (TS) changes contained herein represent changes to the following Sections: 1.0 Definitions, 2.0 Safety Limits and Limiting Safety System Settings, 3/4 1.0 Reactivity Control Systems, 3/4 2.0 Power Distribution Limits, 5.0 Design Features, and 6.0 Administrative Controls. These changes and those requested and approved in TS Amendment 154/135, dated August 22, 1995, constitute the Fuel Upgrade Margin Recovery Program. The previous amendment approved the use of Vantage+ fuel.

This submittal includes the Margin Recovery portion of the program. These changes incorporate the results of the revised safety analyses (Margin Recovery) and the establishment of a Core Operating Limits Report (COLR). The NRC provided guidance for establishing a COLR to control cycle specific TS limits in Generic Letter 88-16 Removal of Cycle Specific Parameter Limits for Technical Specifications, dated October 4, 1988.

A001/1

PSE&G requests these changes to support increased steam generator tube plugging, improved fuel reliability, reduced fuel costs, longer fuel cycles, reduced spent fuel storage, and enhanced reactor safety. Steam generator tube inspections, being performed during the current Unit 1 refueling outage (1R12), have identified a large increase in the number of indications of potential defects. PSE&G is evaluating these indications at this time. Based on the conclusions of the evaluations which may require increased steam generator tube plugging, PSE&G requests approval of these TS changes to support startup of Salem Unit 1.

The proposed changes have been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and PSE&G has concluded that this request involves no significant hazards considerations.

The basis for the requested change is provided in Attachment 1. A 10CFR50.92 evaluation with a determination of no significant hazards consideration is provided in Attachment 2. Attachment 3 describes the results of the safety evaluations and analyses which support implementation of the Margin Recovery Program. The marked-up TS pages affected by the proposed changes are provided in Attachment 4.

Upon NRC approval of this proposed change, PSE&G requests that the amendment be made effective on the date of issuance, but allow an implementation period associated with a refueling outage to allow for the implementation of a cycle specific COLR. For Salem Unit 1, PSE&G requests that implementation be prior to startup (entry into Mode 2) from the current refueling outage, current schedule is undefined. For Salem Unit 2, PSE&G requests that implementation be extended to startup (entry into Mode 2) from the next refueling outage, currently scheduled for Spring, 1997.

Should you have any questions regarding this request, we will be pleased to discuss them with you.

Sincerely,

Louis F. Storz

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STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

L. F. Storz, being duly sworn according to law deposes and says:

I am Senior Vice President - Nuclear Operations of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning Salem Generating Station, Units 1 and 2, are true to the best of my knowledge, information and belief.

Louis F. Storz

Subscribed and Sworn to before me
this 10th day of May, 1996

Kimberly Jo Brown
Notary Public of New Jersey

My Commission expires on KIMBERLY JO BROWN NOTARY PUBLIC OF NEW JERSEY My Commission Expires April 21, 1998

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Attachment 1

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SALEM GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
CHANGE TO TECHNICAL SPECIFICATIONS
MARGIN RECOVERY PROGRAM

BASIS FOR REQUESTED CHANGE

REQUESTED CHANGE AND PURPOSE

The changes identified in Attachment 4, will implement the revisions to the Technical Specifications (TS) necessary to support the Margin Recovery Program (MRP). These changes support increased steam generator tube plugging, improved fuel reliability, reduced fuel costs, longer fuel cycles, reduced spent fuel storage, and enhance reactor safety. These changes include the following:

Core Operating Limits Report (COLR) - Cycle specific parameters are being relocated from the TS Limiting Conditions for Operation (LCO's) into an administratively controlled plant document referred to as the COLR. The NRC provided guidance for establishment of a COLR in Generic Letter (GL) 88-16 - Removal of Cycle-Specific Parameter Limits for Technical Specifications, dated October 4, 1988. The COLR will be updated and submitted to the NRC with each fuel cycle. A sample COLR is provided in Enclosure 1 to Attachment 4. Cycle specific limits for the following LCO's are being relocated from the TS to the COLR:

- Moderator Temperature Coefficient
- Rod Insertion Limits
- Axial Flux Difference
- Heat Flux Hot Channel Factor F_Q (Z)
- Nuclear Enthalpy Hot Channel Factor $F^N_{\Delta H}$

TS 6.9, Administrative Controls - Reporting Requirements, is being revised to replace the administrative requirements for the Radial Peaking Factor Limit Report with the controls for the COLR.

Cross references to TS whose limits are moving to the COLR are being revised to refer to the COLR.

Three-Loop Operation - Changes are being made to eliminate reference to three-loop operation since it is not presently permitted nor analyzed.

Reactor Coolant System (RCS) Flow - The design RCS flow used to calculate the low flow trip setpoint is being reduced.

Reactor Core Safety Limits and Overtemperature Delta T (OTΔT)/ Overpower Delta T(OPΔT) Trip Functions - The equations for calculating the OTΔT and OPΔT trip setpoints are being revised. Figure 2.2-1 is being revised to reflect new core safety limits defining acceptable operation.

Safety Limit Bases - The Bases are being revised to discuss the measurement uncertainties relative to calculating Departure from Nucleate Boiling Ratio (DNBR), to clarify the relationship between control rod position and the FΔH limits, and to refer to the COLR.

Shutdown Margin - The minimum required shutdown margin in Modes 1 through 4 is being changed from 1.6% Δk/k to 1.3% Δk/k.

Departure From Nucleate Boiling (DNB) Parameters - The DNB parameters of RCS T_{avg}, pressurizer, and RCS flow, are being revised to include a change to the RCS flow measurement uncertainty.

Editorial Changes - Although they are not directly related to MRP, they are included herein for clarification:

- Specification 2.1.2 Reactor Coolant System Pressure is moved to page 2.3.

- The reference in specification 3.1.1.4 (3.1.1.3 for Unit 2) Moderator Temperature Coefficient, Action a.1, is being corrected from "3.1.3.6" to "3.1.3.5."

- Index pages IV and XI for Unit 2 only are being changed to reference Nuclear Enthalpy Hot Channel Factor.

- On Table 2.2-1 Reactor Trip System Instrumentation Trip Setpoints, T′ in Note 1 for Unit 2 only and T″ in Note 2 are being changed from "Reference T_{avg}" to "Indicated T_{avg}".

- The Bases are being changed to refer to the "DNB design criterion" rather than a specific numerical value for minimum DNBR.

BACKGROUND

PSE&G has been working with Westinghouse Corporation to provide the capability for more efficient core designs and also increase fuel reliability, decrease RCS coolant activity level, and improve operating margins via revised safety analyses for Salem Generating Station (SGS). The program that was developed to accomplish this, termed Fuel Upgrade Margin Recovery Program (FUMRP), consists of two related parts. The Fuel Upgrade portion of the program pertaining to the use of ZIRLO-clad, VANTAGE+ fuel has been approved for use at SGS Units 1 and 2 via License Amendments 154/135, dated August 22, 1995. This upgraded fuel design, in conjunction with the MRP changes proposed herein, comprise the two elements of the FUMRP.

The revised safety analyses, which include the Westinghouse WCAP for Revised Thermal Design Procedure (RTDP) discussed later, result in additional DNBR margin. A portion of this margin will be utilized to increase the allowable $F_{\Delta H}$ peaking factor limit which will permit a larger number of burned fuel assemblies to be used on the core periphery (thereby reducing neutron leakage and improving uranium utilization). Higher peaking factor limits can also reduce the number of burnable absorbers required. The higher allowable $F_{\Delta H}$ and F_Q peaking factors would facilitate the design of lower leakage cycle loading patterns and will contribute to reduced fuel costs, higher capacity factors, increased operational flexibility, increased reactor vessel lifetime, and decreased spent fuel storage/disposal.

Steam generator tube inspections being performed during the current Unit No. 1 refueling outage (1R12) have identified a large increase in the number of indications of potential defects. PSE&G is evaluating these indications at this time. Based on the conclusions of the evaluations, increased tube plugging may be required. The MRP includes the analyses necessary to support a reduced RCS flow and RCS volume and an increased steam generator tube plugging limit of 20% average and 25% in any steam generator.

JUSTIFICATION OF REQUESTED CHANGES

SUPPORTING ANALYSES

All of the licensing basis safety analyses that are affected by the MRP have been evaluated. Attachment 3 provides details of the evaluations and analyses that were performed to confirm the acceptability of the MRP. The evaluations of the safety analyses that support the MRP were performed by Westinghouse under the cognizance of PSE&G.

The licensing basis safety analyses described in Attachment 3 incorporate several improved analysis methodologies which are briefly described below:

NOTRUMP Small Break Loss of Coolant Accident (LOCA) Methodology: The MRP includes a complete spectrum Small Break LOCA analysis utilizing the Emergency Core Cooling System (ECCS) NOTRUMP model. This methodology was submitted to the NRC on April 2, 1993 and approved via NRC letter dated August 25, 1993. This code has been used to demonstrate SGS compliance with the ECCS performance requirements in accordance with 10CFR50.46 for plant conditions consistent with implementation of the MRP.

Nuclear Steam Supply System (NSSS) Components Structural Analysis Using Leak-Before-Break Methodology: NSSS components structural analyses were performed to support plant operating flexibility (increased allowable steam generator tube plugging, reduced allowable RCS flow, and increased allowable RCS T_{avg} range). Consistent with current industry practices, these structural analyses utilized the Leak-Before-Break methodology which eliminated the requirement to analyze the instantaneous double-ended guillotine break as applied to RCS Primary loop piping. This methodology was submitted to the NRC and a safety evaluation was received on May 25, 1994.

Revised Thermal Design Procedure (RTDP): RTDP is being applied to transient analyses involving calculations of DNBR. Prior SGS analyses used the Standard Thermal Design Procedure. In the RTDP, uncertainties in initial conditions, peaking factors, and DNB correlations are statistically combined to define the Design Limit DNBR. Since the Design Limit DNBR accounts for these uncertainties, the initial nominal condition of the operating parameters and minimum measured RCS flow are used in the DNBR analysis. The RTDP report is included as Enclosure A in Attachment 3.

LOCA Containment Response Analysis: This analysis incorporates a new methodology for the mass and energy release model, and is included as Enclosure B to Attachment 3.

CONCLUSIONS

The proposed changes to the TS identified to support the MRP incorporate the revised safety analyses which were performed using the above listed methodologies. In addition, these changes include the establishment of a COLR to control cycle specific limits, consistent with GL 88-16. These changes and analyses have been reviewed and evaluated by PSE&G and found to be acceptable.

SALEM GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
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CHANGE TO TECHNICAL SPECIFICATIONS
MARGIN RECOVERY PROGRAM

10CFR50.92 EVALUATION

Public Service Electric & Gas Company (PSE&G) has concluded that the proposed changes to the Salem Generating Station Unit 1 and 2 Technical Specifications (TS) do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGE

The proposed TS changes associated with the Margin Recovery Program (MRP) revise sections 2.0 Safety Limits and Limiting Safety System Settings, 3/4 1.0 Reactivity Control Systems, 3/4 2.0 Power Distribution Limits, 5.0 Design Features, and 6.0 Administrative Controls. These changes incorporate the results of the revised safety analyses. In addition, these changes include the establishment of a Core Operating Limits Report to control cycle specific TS limits consistent with Generic Letter 88-16 Removal of Cycle Specific Parameter Limits for Technical Specifications, dated October 4, 1988.

BASIS

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The accidents potentially affected by the parameters and assumptions associated with the MRP have been evaluated/ analyzed and all design standards and applicable safety criteria are met. The consideration of these changes does not result in a situation where the design, material, or construction standards that were applicable prior to the change have been altered. Therefore, the changes occurring with the MRP will not result in any additional challenges to plant equipment that could increase the probability of any previously evaluated accident.

The changes associated with the MRP do not affect plant systems such that their function in the control of radiological consequences is adversely affected. The safety evaluation documents that the design standards and applicable safety criteria limits continue to be met and therefore fission barrier integrity is not challenged. The MRP changes have been shown not to adversely affect the response of the plant to postulated accident scenarios. In all cases, the calculated doses are within the regulatory criteria and therefore do not constitute an increase in consequences. These changes will, therefore, not affect the mitigation of the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR).

Based on the above, it is concluded that the probability or consequences of an accident previously evaluated is not significantly increased by the proposed changes.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The possibility for a new or difference type of accident from any accident previously evaluated is not created since the changes associated with the MRP do not result in a change to the design basis of any plant component or system. The evaluation of the effects of the MRP changes shows that all design standards and applicable safety criteria limits are met. These changes therefore do not cause the initiation of a new accident nor create any new failure mechanisms. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The MRP changes will not result in more adverse conditions and will not result in any increase in the challenges to safety systems.

Therefore, the consideration of the MRP as described in the safety evaluation does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

   The margin of safety is maintained by assuring compliance with acceptance limits reviewed and approved by the NRC. Since all of the appropriate acceptance criteria for the various analyses and evaluations have been met, by definition there has not been a reduction in any margin of safety.

   Therefore, the margin of safety as defined in the Bases to the Salem Unit 1 and 2 Technical Specifications has not been significantly reduced.

CONCLUSION

Based on the above, PSE&G has determined that the proposed changes do not involve a significant hazards consideration.

ATTACHMENT 3

SUPPORTING FUMRP ANALYSES/EVALUATIONS

1.0 Introduction and Summary
2.0 Nuclear Design
3.0 Thermal and Hydraulic Design
4.0 Accident Analysis

Enclosure A RTDP WCAP 13651 (Proprietary), 13652 (Non-Propritary)

Enclosure B LOCA Contianment WCAP 13839

ATTACHMENT 3
SUPPORTING FUMRP ANALYSES/EVALUATIONS

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ATTACHMENT 3
SUPPORTING FUMRP ANALYSES/EVALUATIONS

1.0 INTRODUCTION AND SUMMARY

1.1 Introduction

The Salem Nuclear Generating Station Unit 1 and Unit 2 proposes to implement the Technical Specification changes associated with the Margin Recovery Program (MRP) starting with Salem Unit 1 Cycle 14 and Salem Unit 2 Cycle 11, respectively. This document summarizes the safety evaluations/analyses that were performed to confirm the acceptable implementation of the MRP. Sections 2.0 through 4.0 of this document provide the results of the Nuclear, Thermal and Hydraulic, and Accident Evaluations, respectively.

1.2 Margin Recovery

The Margin Recovery Program supports the following major changes:

- 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 1.55 to 1.65. The maximum $F_Q(Z)$ peaking factor limit will increase from the current value of 2.32 to 2.40 and the K(Z) envelope will be modified. These increases will permit more flexibility in developing fuel management strategies (i. e., longer fuel cycles, improvement of fuel economy and neutron utilization).

- A decrease in shutdown margin from 1.6% Δk to 1.3% Δk.

- Reduction of the Reactor Coolant System (RCS) Thermal Design Flow (TDF) from 87,300 gpm/loop to 82,500 gpm/loop.

- Operation of the units at any RCS average temperature (T_{avg}) within the range of 566.0°F to 577.9°F.

- Increased Steam Generator Tube Plugging to an average of 20% and peak level of 25%.

- Implementation of the Revised Thermal Design Procedure (WCAP-13651).

The accident analyses and components/systems analyses have incorporated the input data and the operating parameters which encompass the new operating conditions of the Margin Recovery Program. The NSSS components structural integrity analyses were performed to bound the current Salem plant NSSS power level of 3423 MWt (reactor power limit of 3411 MWt).

1.3 Conclusions

The results of evaluation/analysis described herein lead to the following conclusions:

1. The change in the design full power $F^N_{\Delta H}$ limit from 1.55 to 1.65 (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.

   The change in the maximum $F_Q(Z)$ limit from 2.32 to 2.40 and modification to the K(Z) envelope is supported by design basis safety analyses summarized in this report. The corresponding changes to the Technical Specifications are as defined in Appendix A.

2. The reduction in shutdown margin is supported by the design basis safety analyses.

3. The core design and safety results documented in this report show the core's capability for operating safely with the Margin Recovery parameters at the rated Salem Unit 1 and Unit 2 design thermal power.

4. This submittal establishes a reference upon which to base reload safety evaluations for future reloads.

2.0 NUCLEAR DESIGN

2.1 Introduction and Summary

The effects of the Margin Recovery Program and associated Technical Specification changes on the nuclear design bases and methodologies for Salem Nuclear Generating Station Unit 1 and Unit 2 have been evaluated.

The plant technical specifications that are established by nuclear design have been reviewed. The technical specification changes which impact the nuclear design are the increase in the peaking factor limits and shutdown margin. The increased peaking factor limits and reduced shutdown margin requirements increase fuel management flexibility (lower leakage, increased fuel economy and increased nuclear design flexibility). In summary, the Technical Specification changes associated with the Margin Recovery Program will not cause changes to the current Salem Unit 1 and Unit 2 UFSAR nuclear design bases. Nuclear design methodology is not affected by the Margin Recovery Program.

2.2 Methodology

No changes to the nuclear design philosophy, methods or models are necessary because of the MRP Technical Specification changes. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the FSAR safety evaluation for each reload cycle. These key safety parameters will be evaluated for each Salem Unit 1 and Unit 2 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.

2.3 Design Evaluation - Power Distributions and Peaking Factors

The implementation of increased radial and total peaking factor limits will have minor impacts on the core power distributions and peaking factors experienced in Salem Unit 1 and Unit 2. 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.

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 Technical Specifications can be assured using these methods.

2.4 Nuclear Design Evaluation Conclusions

Power distributions may show slight changes as a result of the 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 to ensure compliance with the Peaking Factor Technical Specifications.

TABLE 2-1
Range of Key Safety Parameters

Safety Parameter	Current Design Values	New Design Values
Reactor Core Power (MWt)	3411	3411
Vessel Average Coolant Temp. HFP (°F)	577.9	577.9 to 566.0
Coolant System Pressure (psia)	2250	2250
Core Average Linear Heat Rate (Kw/ft)	5.43	5.43
Most Positive MTC (pcm/°F)	0	0
Most Positive MDC ($\Delta K/g/cm^3$)	0.43	0.52
Doppler Temperature Coefficient (pcm/°F)	-2.90 to -1.0	-3.50 to -0.91
Doppler Only Power Coefficient (pcm/% Power)		
Least Negative, HFP to HZP	-10.18 to -6.68	-9.30 to -6.05
Most Negative, HFP to HZP	-19.4 to -12.6	-23.0 to 14.0
Beta-Effective	0.0044 to 0.0075	0.0040 to 0.0075
Normal Operation $F_{\Delta H}$ (with uncertainties)	1.55	1.65
Shutdown Margin (%$\Delta\rho$)[1]	1.60	1.30
Normal Operation F_Q	2.32	2.40

1 See Section 5.1.0.1

3.0 THERMAL AND HYDRAULIC DESIGN

3.1 Introduction and Summary

This section describes the calculational methods used for the thermal-hydraulic analysis and DNB analysis. Table 3-1 summarizes the thermal-hydraulic parameters for Salem Unit 1 and Unit 2 that were used in this analysis. The thermal-hydraulic criteria and methods remain the same as those presented in the Salem Unit 1 and Unit 2 FSAR with the exceptions noted in the following sections. All of the current FSAR thermal-hydraulic design criteria are satisfied.

3.2 Methodology

The DNB analysis of the core incorporates the Revised Thermal Design Procedure (RTDP, WCAP-13651) and an Improved THINC-IV Model. The W-3 correlation and STDP are still used when conditions are outside the range of the WRB-1 correlation and the RTDP.

The WRB-1 DNB correlation is based entirely on rod bundle data and takes credit for the significant improvements in the accuracy of the critical heat flux predictions over previous DNB correlations. With RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor which is used to define the design limit DNBR that satisfies the DNB design criterion. The criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95% (at 95% confidence level) for any Condition I or II event. 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 uncertainties included in the combined peaking factor uncertainty are the nuclear enthalpy rise hot channel factor, ($F^N_{\Delta H}$); the enthalpy rise engineering hot channel factor, ($F^E_{\Delta H}$); uncertainties in the THINC-IV and transient codes; and uncertainties, based on surveillance data, associated with vessel coolant flow, core power, coolant temperature, system pressure, and effective core flow fraction (i.e., bypass flow). The increase in DNB margin is realized when nominal values of the peaking and hot channel factors are used in the DNB safety analyses.

Instrumentation uncertainties are documented in the Salem RTDP Instrument Uncertainty Methodology Report. The following instrumentation uncertainties (which bound the values from the above report) were used in determining the DNBR design limits:

Power	±1.6%
RCS Flow	±3.5%
Pressure	±32 psi
Inlet Temperature	±3.5 °F

For use in the DNB safety analyses, the DNBR limit is conservatively increased to provide DNB margin to offset the effect of rod bow, transition core and any other DNB penalties that may occur, and to provide flexibility in design and operation of the plant. The safety analysis limit DNBR values of 1.34 for typical cells and 1.33 for thimble cells are employed in the analysis.

Table 3-2 summarizes the available DNBR margin for Salem Nuclear Generating Station Unit 1 and Unit 2.

3.3 Conclusion

The thermal hydraulic evaluation for the Margin Recovery Program for Salem Unit 1 and Unit 2 has shown that the DNB margin gained through use of the RTDP methodology with the WRB-1 correlation is sufficient to allow an increase in the full power $F^N_{\Delta H}$ from 1.55 to 1.65. All current thermal-hydraulic design criteria are satisfied.

TABLE 3-1
Salem Unit 1 and Unit 2
Thermal and Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters (using RTDP)	Design Parameters
Reactor Core Heat Output, Mwt	3,411
Reactor Core Heat Output, 10^6, BTU/Hr	11,642
Heat Generated in Fuel, %	97.4
Core Pressure, Nominal, psia	2265
Pressurizer Pressure, Nominal, psia	2250
Nuclear Enthalpy Hot Channel Factor*	1.65[1+0.3(1-P)], where $P = \frac{\text{Thermal Power}}{\text{Rated T.P.}}$
HFP Nominal Coolant Conditions	
Vessel Thermal Design Flow (TDF)	
Rate (including Bypass), 10^6 lbm/hr	125.2
GPM	330,000
Core Flow Rate**	
(excluding Bypass, based on TDF)	116.2
10^6 lbm/hr	306,240
GPM	51.3
Core Flow Area, ft^2	
Core Inlet Mass Velocity, 10^6 lbm/hr-ft^2 (Based on TDF)	2.27

* Includes 4% measurement uncertainty.
** Based on design bypass flow of 7.2%

TABLE 3-1
Salem Unit 1 and Unit 2
Thermal and Hydraulic Design Parameters (Continued)

Thermal and Hydraulic Design Parameters (based on TDF)	Design Parameters
Nominal Vessel/Core Inlet Temperature, °F	543.2
Vessel Average Temperature, °F	577.9
Core Average Temperature, °F	582.3***
Vessel Outlet Temperature, °F	612.6
Average Temperature Rise in Vessel, °F	69.4
Average Temperature Rise in Core, °F	74.2
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	59,742
Average Heat Flux, BTU/hr-ft^2	189,800
Average Linear Power, kw/ft	5.45
Peak Linear Power for Normal Operation, ****kw/ft	13.08
Peak Linear Power for Prevention of Centerline Melt, kw/ft	22.5
Pressure Drop Across Core, psi	22.2

*** Based on average enthalpy in core.
**** Based on maximum F_Q of 2.40.

TABLE 3-2
DNBR Margin Summary*

DNB Correlation		WRB-1
DNBR Correlation Limit		1.17
DNBR Design Limit	(TYPICAL)	1.24
	(THIMBLE)	1.24
DNBR Safety Limit	(TYPICAL)	1.34
	(THIMBLE)	1.33
Rod Bow DNBR Penalty		1.3%
Transition Core DNBR Penalty		0%

* Steamline break is analyzed using the W-3 correlation with STDP. The correlation limit DNBR is 1.45 in the range of 500 to 1000 psia and the safety limit DNBR is 1.667.

Rod withdrawal from subcritical and feedwater malfunction are analyzed using the W-3 correlation with STDP below the bottom MV grid. The correlation limit DNBR is 1.30 above 1000 psia and the safety limit DNBR is 1.376 which covers the correlation limit plus rod bow penalty. WRB-1 with STDP is used for rod withdrawal from subcritical above the bottom MV grid.

4.0 ACCIDENT ANALYSIS

This Reload Transition Safety Report (RTSR) is provided to support implementation of the Margin Recovery Program (MRP). The safety analyses support the Margin Recovery Program features described in Section 4.1.0.2. Justification for the proposed MRP Technical Specification changes is summarized for the non-LOCA and LOCA design basis calculations in Sections 4.1 and 4.2, respectively.

4.1.0 NON-LOCA ACCIDENTS

This section summarizes the non-LOCA analyses and evaluations performed to support MRP implementation at Salem Units 1 and 2.

4.1.0.1 Peaking Factors and Shutdown Margin

The MRP analyses account for an increased enthalpy hot channel factor (radial peaking, $F^N_{\Delta H}$) of 1.65 and heat flux hot channel factor (total peaking, F_Q) of 2.40. $F^N_{\Delta H}$ plays an important role in transients that are departure from nucleate boiling (DNB)-limited. Since $F^N_{\Delta H}$ increases with decreasing power level (due to rod insertion), all transients that may be DNB-limited are assumed to begin with an $F^N_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications (Tech Specs). F_Q is important for transients that may be overpower-limited. F_Q may increase with decreasing power level such that the full-power hot spot heat flux is not exceeded (i.e., F_Qx Power equals the design hot spot heat flux). Consequently, all non-LOCA transients for this RTSR that may be overpower-limited assumed an initial hot full power F_Q of 2.40.

The analyses sensitive to minimum shutdown margin (SDM) assumed 1.3% Dk.

4.1.0.2 Margin Recovery Program Features

The Margin Recovery Program features which were evaluated include:

a. A reactor coolant average temperature range from 566.0°F to 577.9°F

b. An NSSS power level of 3423 MWt (no uprated power levels)

c. Reactor coolant system (RCS) thermal design flow of 82500 gpm/loop

d. A 20% average steam generator tube plugging (SGTP), not to exceed 25% in any steam generator

For most accidents which are DNB-limited, nominal values of the initial conditions are assumed. The uncertainty allowances on power, temperature, pressure, and RCS flow are included on a statistical basis and are included in the limit DNBR value by using the Revised Thermal Design Procedure (RTDP). For accident analyses which are not DNB limited, or for which RTDP is not employed, the initial conditions are obtained by applying the maximum steady-state errors to rated values (Standard Thermal Design Procedure - STDP).

The following steady-state errors are considered in the STDP analyses:

a. For core power, a ±2% allowance for calorimetric error is conservatively applied in the non-LOCA accident analyses;

b. The average RCS temperature allowance for dead band and system measurement error is ±5°F, and

c. The pressurizer pressure allowance is ±50 psi for steady-state fluctuations and measurement errors.

Accidents employing RTDP assume a minimum measured flow (MMF), while others assume the thermal design flow (TDF). In addition to being the flow used in the DNB analysis for RTDP methodology, the MMF is bounded by the Tech Specs minimum flow measurement requirement. A MMF of 341000 gpm total includes allowance for plant flow measurement uncertainty (including a conservative flow allowance to bound the potential effects of cold leg streaming).

4.1.0.3 Other Major Items

a. The non-LOCA MRP analyses apply for either the analog or digital feedwater control system.

b. Since the Salem units are not licensed for N-1 loop operation, the non-LOCA portion of the MRP addressed only operation with all four RCS loops operating.

c. ANS 5.1-1979 Residual Decay Heat is assumed (plus 2 Sigma). The fission product contribution to decay heat assumed in the non-LOCA analyses is the decay heat model increased by two standard deviations for conservatism.

4.1.0.4 Overtemperature-and Overpower-DT

The overtemperature - and overpower-DT (OT/OPDT) setpoints were recalculated for the FU/MRP based on the most conservative core limits. The most conservative core limits were based on RTDP safety limits as described in RTSR Section 4.0. The core limits used to calculate the OT/OPDT setpoints are provided in Figure 4.1-1 and in Tech Spec Figure 2.1-1. The UFSAR events that rely on OT/OPDT for protection were reanalyzed to reflect the setpoint changes, as provided in the revised Tech Specs. It has been confirmed that these OT/OPDT setpoints protect the core safety limits as shown in Figure 4.1-2.

4.1.0.5 RCCA Reactivity Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCAs and the variation in rod worth as a function of rod position. With respect to the accident analyses, the critical parameter is the time from beginning of RCCA insertion to dashpot entry, or approximately 85% of the RCCA travel. For the accident analyses, the insertion time from fully withdrawn to dashpot entry remains at the Tech Spec limit of 2.7 seconds from the beginning of stationary gripper voltage decay.

The UFSAR contains three figures relating to RCCA drop time and reactivity worth. The rod drop time remains at 2.7 seconds from fully withdrawn to the dashpot. The RCCA position (percent insertion) versus the time from release is presented in Figure 4.1-3. This figure has not changed from the current UFSAR. The normalized reactivity worth assumed in the MRP safety analyses is shown in Figures 4.1-4 and 4.1-5, which present the worth versus rod insertion and time from release, respectively.

For analyses requiring the use of a dimensional diffusion theory code, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the RCCA position versus time of Figure 4.1-3 is used.

4.1.0.6 Reactivity Coefficients

The transient response of the RCS is dependent on reactivity feedback effects, in particular the MTC and the Doppler power coefficient (DPC). Depending upon event specific characteristics, conservatism dictates use of either large or small reactivity coefficient values. Justification for the use of the reactivity coefficient values is treated on an event-specific basis.

4.1.0.7 Non-LOCA Events Evaluated or Analyzed

The effect of MRP implementation on each of the UFSAR transients listed on Table 4.1-1 were evaluated or analyzed. These transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria continue to be met.

Figure 4.1-1

Reactor Core Safety Limit
Four Loops In Operation

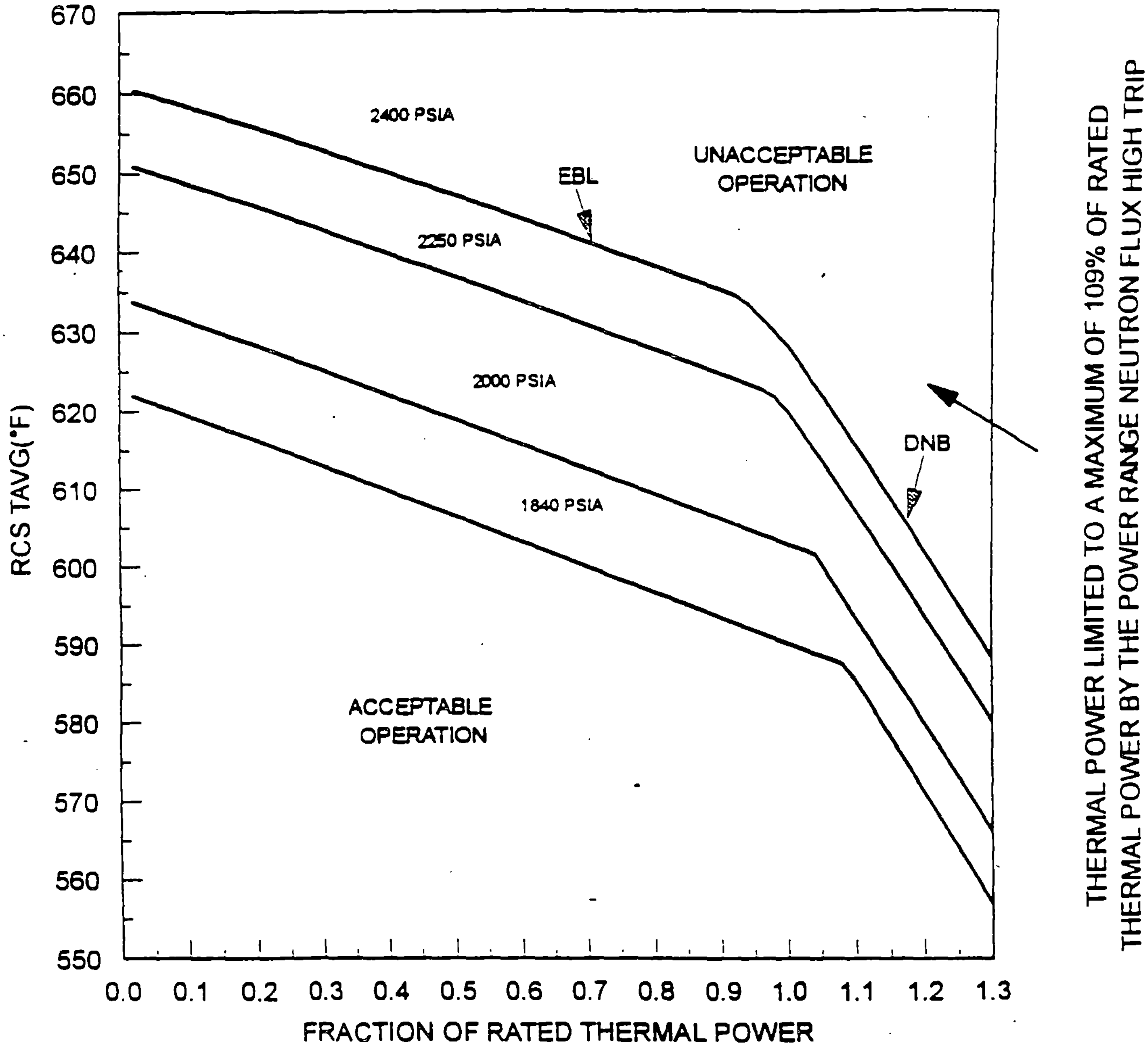


Figure 4.1-2

Illustration of Overtemperature and Overpower ΔT Protection

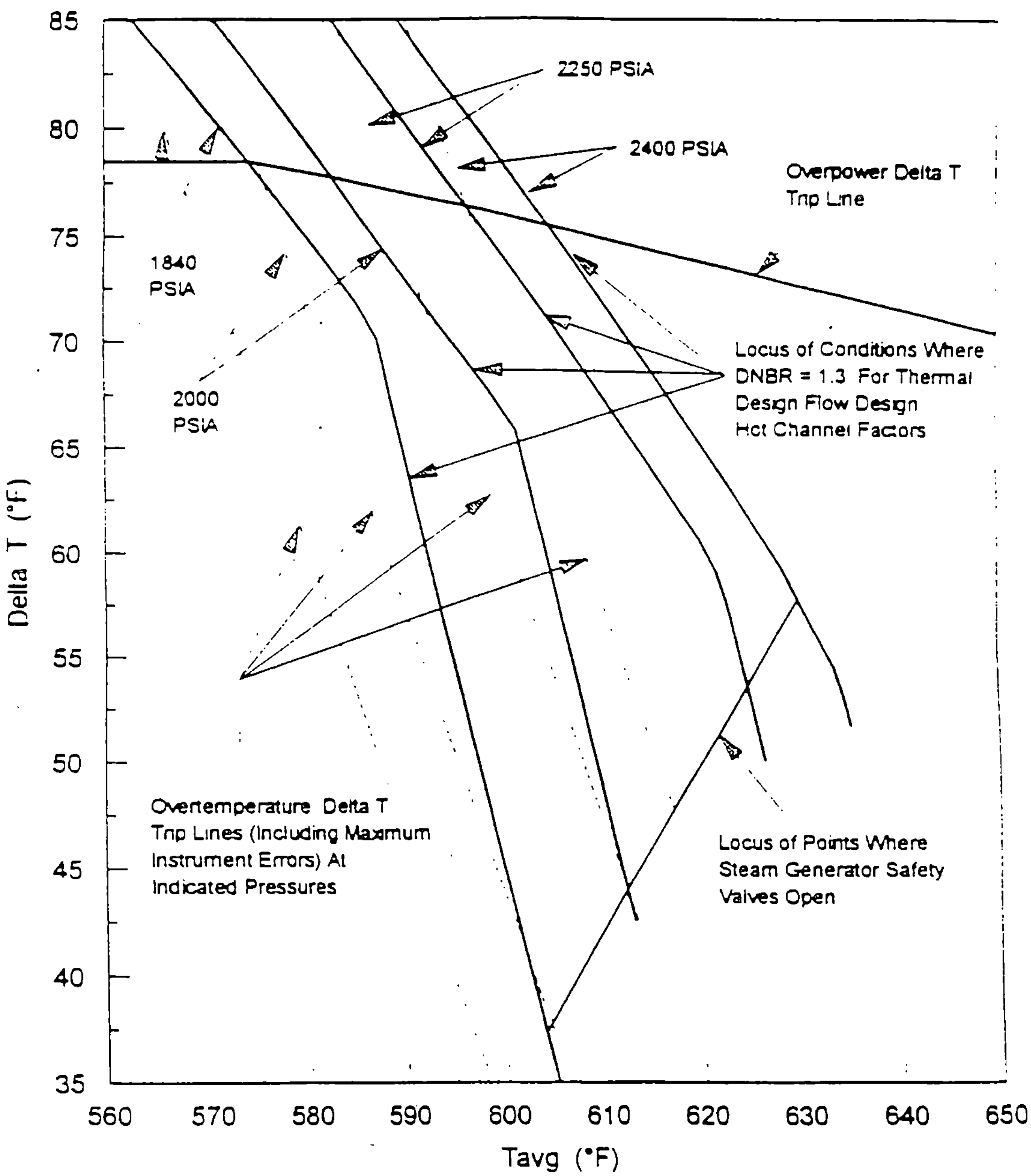


Figure 4.1-3

RCCA Position (Insertion) vs
Time From Release

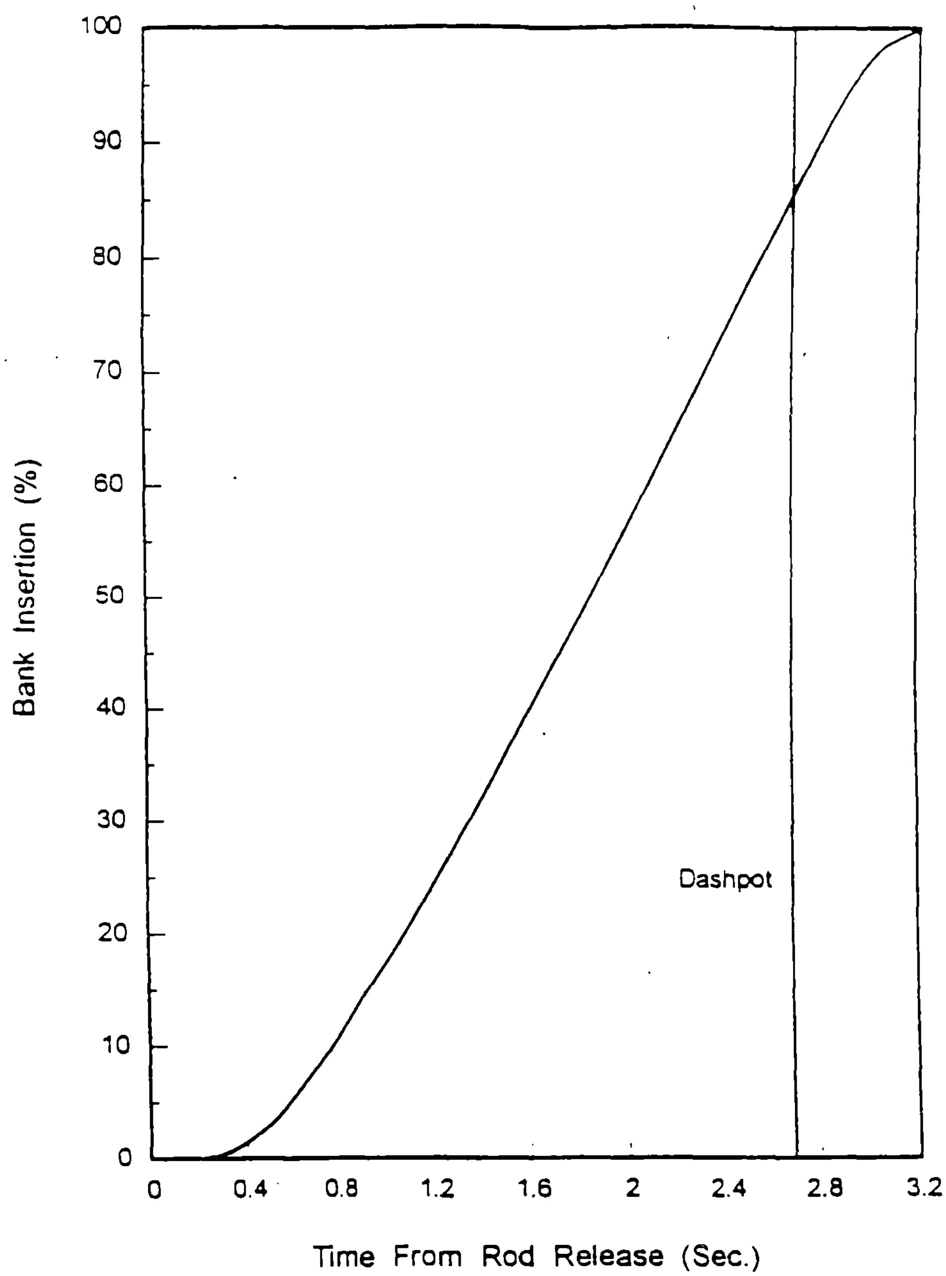


Figure 4.1-4

Normalized RCCA Reactivity Worth vs
Percent Insertion

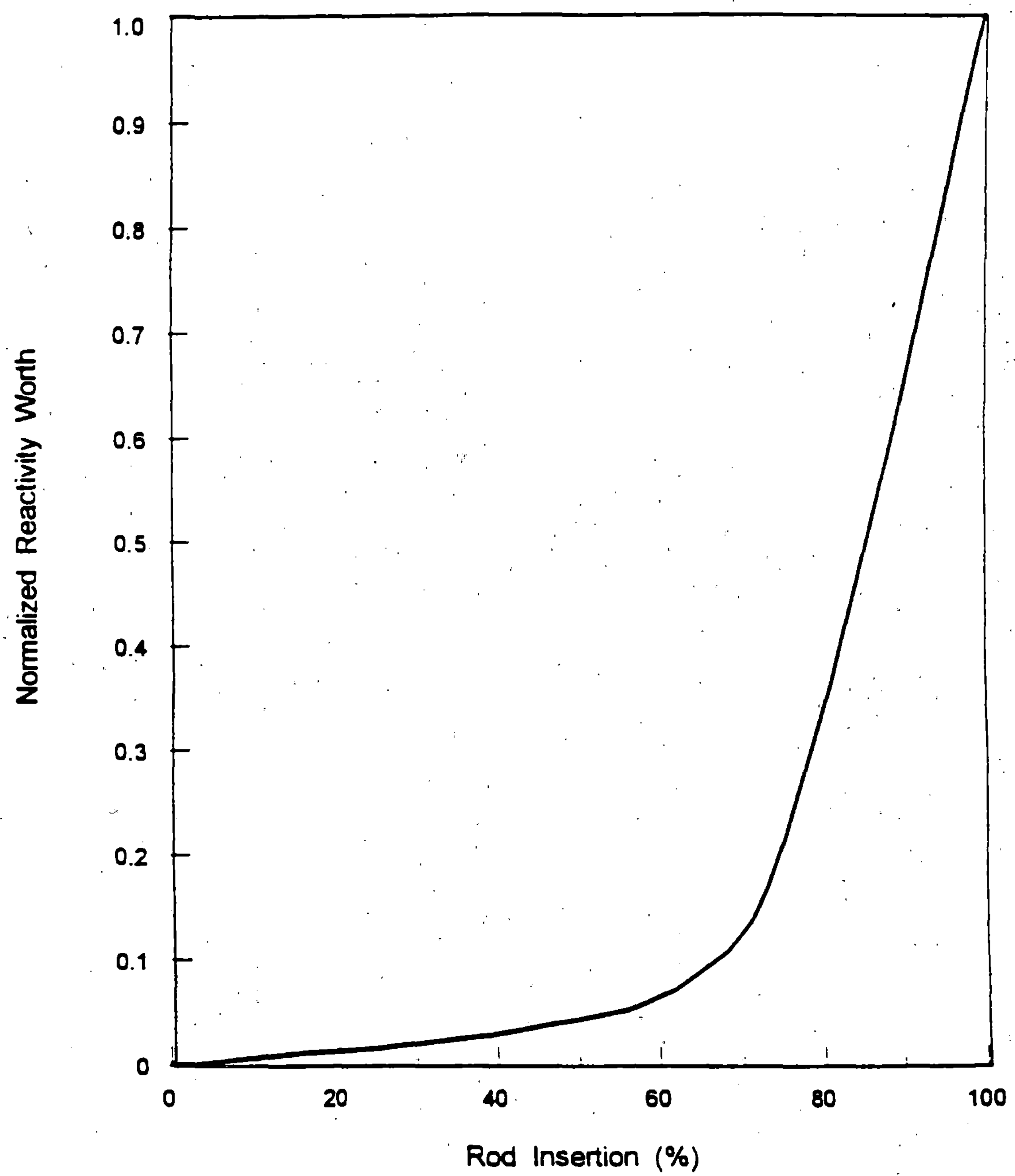
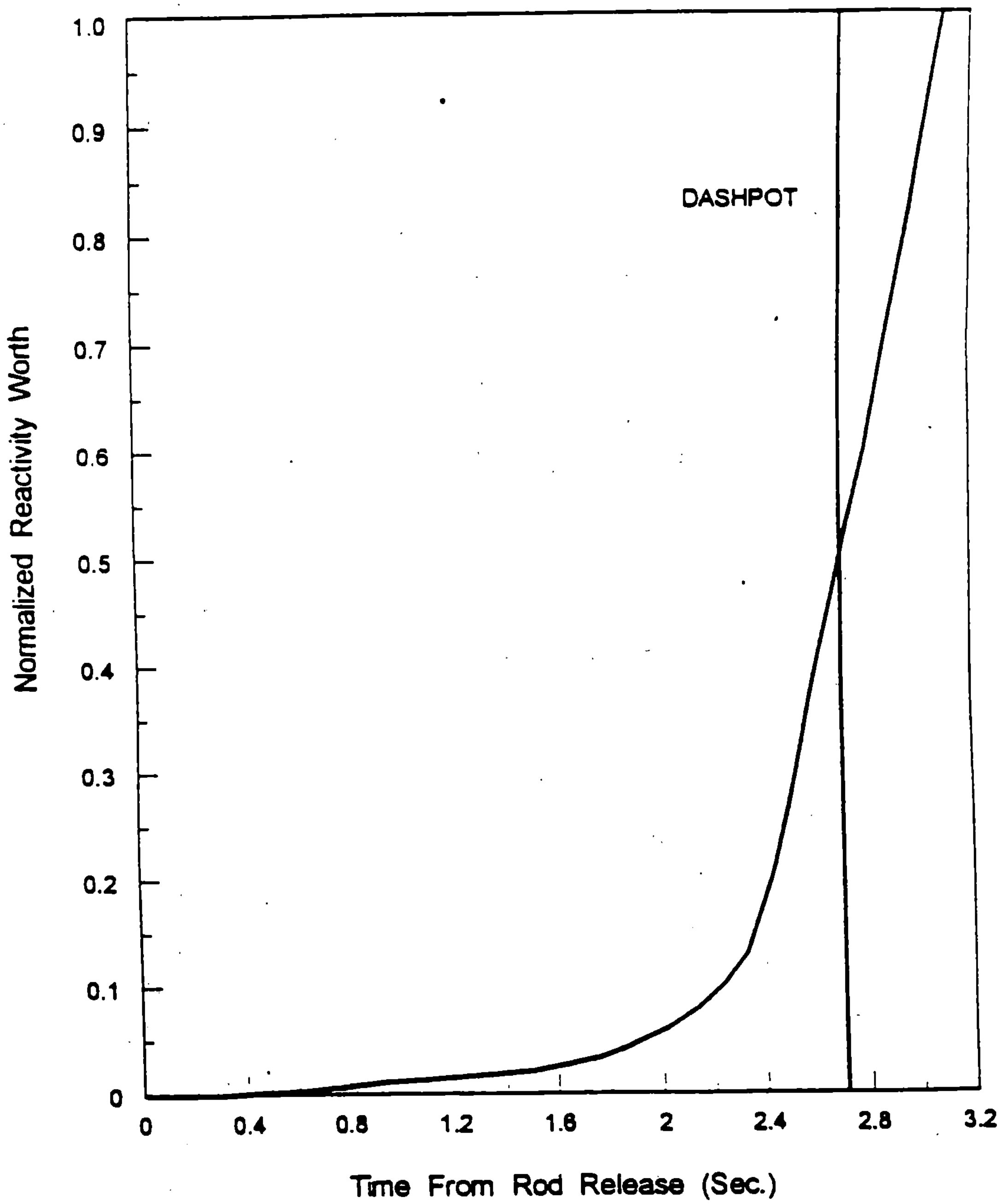


Figure 4.1-5

Normalized RCCA Bank Reactivity Worth vs
Time After Trip



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4.1.1 Uncontrolled RCCA Bank Withdrawal From a Subcritical Condition (UFSAR 15.2.1)

<u>Accident Description:</u>

A Condition II event, an RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks resulting in a power excursion. This Condition II transient can occur with the reactor either subcritical, at HZP, or at power. The "at power" case is discussed in Section 4.1.2.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous control rod assembly withdrawal event occur, the following automatic features of the reactor protection system are available to terminate the transient.

a. The source range high neutron flux reactor trip.

b. The intermediate range high neutron flux reactor trip.

c. The power range high neutron flux reactor trip (low setting).

d. The power range high neutron flux reactor trip (high setting).

e. The high nuclear flux rate reactor trip.

In addition, control rod stops on high intermediate range flux and high power-range flux signals serve to cease rod withdrawal and prevent the need to actuate the intermediate-range flux reactor trip and the power-range flux reactor trip, respectively.

Method of Analysis:

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages. First, a spatial neutron kinetics computer code, TWINKLE, is used to calculate the core average nuclear power transient, including the various core feedback effects, i.e., Doppler and moderator reactivity. FACTRAN then uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the average heat flux and temperature transients. Finally, the average heat flux calculated by FACTRAN is used in THINC for DNBR calculations.

In order to give conservative results for a startup accident, the following assumptions are made:

a. A conservatively low (absolute magnitude) value for the Doppler power defect is used.

b. The effective MTC used in the RWFS event analysis bounds the least negative MTC allowed for Salem Units 1 and 2.

c. The analysis assumes the reactor to be at a HZP nominal temperature of 547°F. This assumption is more conservative than that of a lower initial system temperature (i.e., shutdown conditions).

d. Reactor trip is assumed to be initiated by power-range high neutron flux (low setting). In addition, the total reactor trip reactivity is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position.

e. The maximum positive reactivity insertion rate assumed bounds that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at a conservative speed (45 in./min, which corresponds to 72 steps/min).

f. 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 high worth position.

g. The analysis employs the STDP methodology.

Results:

Figures 4.1.1-1 and 4.1.1-2 show the transient behavior for the indicated reactivity insertion rate. Figure 4.1.1-1 shows the neutron flux transient. 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, of interest for the DNB considerations, is also shown in Figure 4.1.1-1. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux of much less than the nominal full power value. Figure 4.1.1-2 shows the transient response of the hot spot fuel temperatures. Table 4.1.1-1 presents the calculated sequence of events. For this event the minimum DNBR remains above the safety analysis limit value at all times.

Conclusions:

In the event of an RCCA bank withdrawal accident from a subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature result in a DNBR greater than the safety analysis limit value.

Table 4.1.1-1

Sequence of Events
Uncontrolled RCCA Withdrawal From a Subcritical Condition

Event	Time (seconds)
Initiation of Uncontrolled RCCA Bank Withdrawal	0.0
Power Rnage High Neutron Flux Low Setpoint Reached	10.4
Peak Nuclear Power Occurs	10.6
Rod Motion Begins	10.9
Peak Heat Flux Occurs	12.7
Minimum DNBR Occurs	12.7
Peak Clad Temperature Occurs	13.5
Peak Fuel Average Temperature Occurs	13.7
Peak Fuel Centerline Temperature Occurs	14.2

Figure 4.1.1-1

Uncontrolled Rod Cluster Control Assembly
Bank Withdrawal From Subcritical

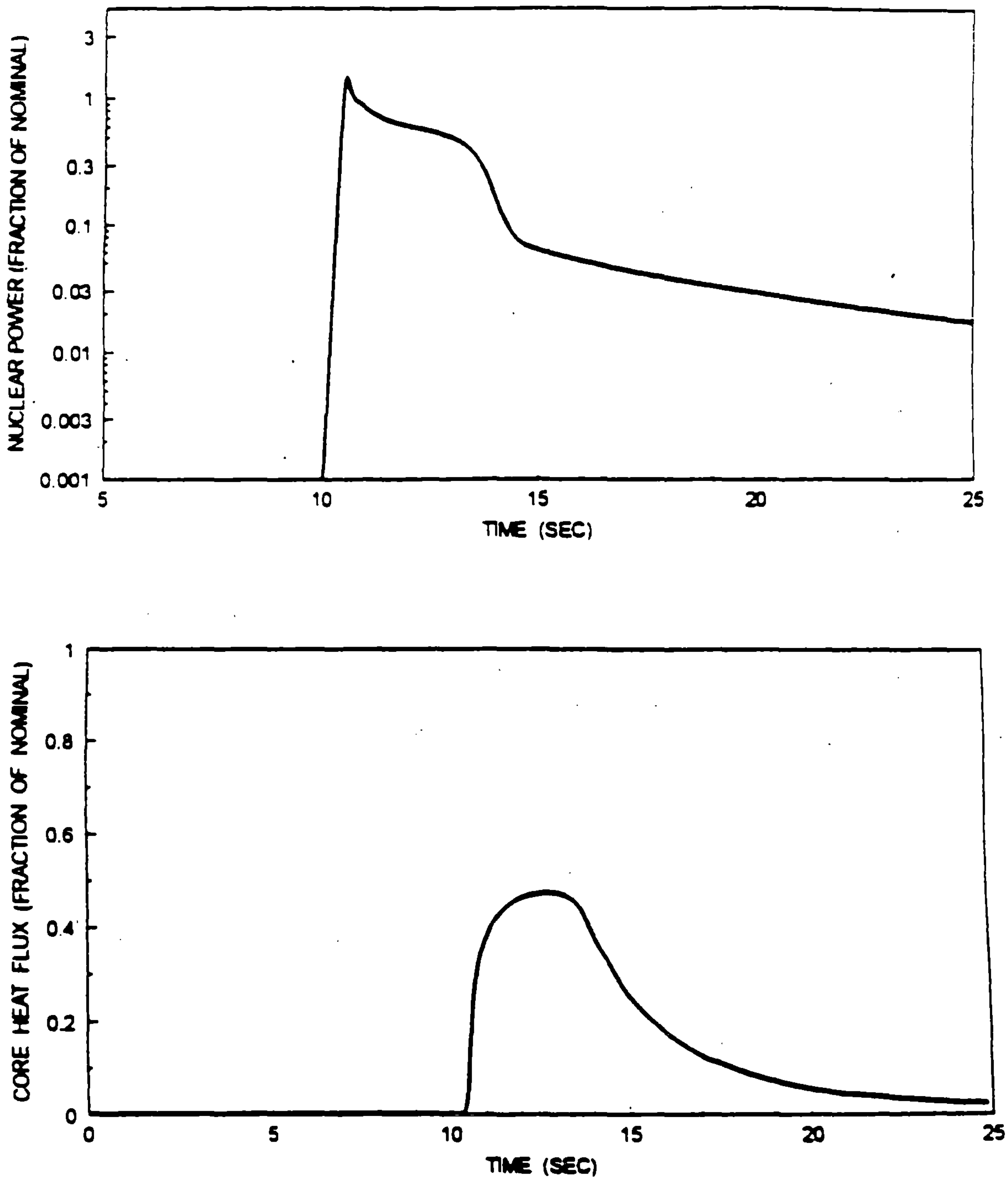
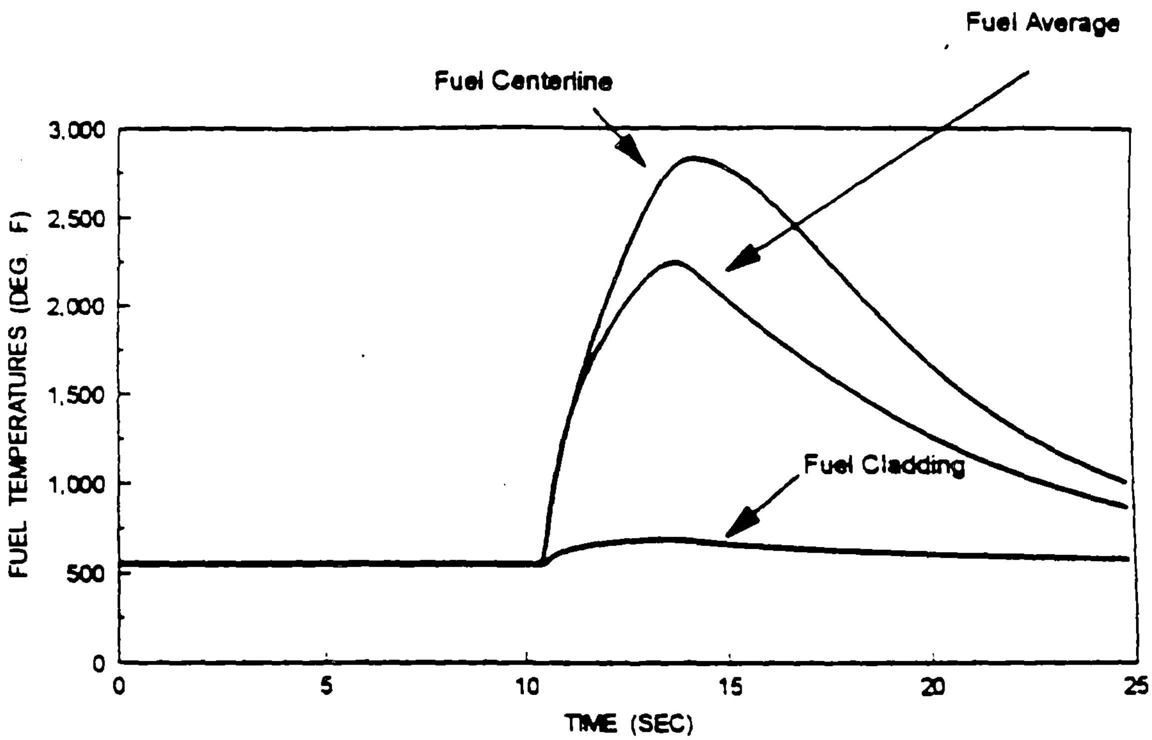


Figure 4.1.1-2

Uncontrolled Rod Cluster Control Assembly
Bank Withdrawal From Subcritical



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4.1.2 Uncontrolled RCCA Bank Withdrawal at Power (UFSAR 15.2.2)

Accident Description:

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 withdrawal will cause an increase in core nuclear power and subsequent increase in core heat flux. A RCCA bank withdrawal can occur with the reactor subcritical, at HZP, or at power. The uncontrolled RCCA bank at power event is analyzed for Mode 1 (power operation). The uncontrolled RCCA bank withdrawal from a subcritical or low power condition is considered in Section 4.1.1.

To prevent the core damage which might otherwise result from this event, the RPS is designed to automatically terminate any such event before the 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. overtemperature DT

c. overpower DT

d. high pressurizer pressure

e. high pressurizer water level

Method of Analysis:

The RCCA bank withdrawal at power transient is analyzed with the LOFTRAN computer program for a spectrum of positive reactivity insertion rates. Since the RTDP was used in the analysis, 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 ensure bounding results are obtained for all possible normal operating conditions.

a. Two reactivity coefficient conditions are analyzed:

   1. Minimum Reactivity Feedback. A least positive moderator density coefficient of reactivity is assumed, corresponding to beginning of cycle life (BOL) core conditions. A conservatively small (absolute magnitude) Doppler Power Coefficient (DPC), variable with core power, was used in the analysis.

   2. Maximum Reactivity Feedback. A conservatively large positive moderator density coefficient and a large (absolute magnitude) DPC are assumed.

b. A conservatively high setpoint was assumed for the high neutron flux reactor trip. The OTDT reactor trip function includes all adverse instrumentation and setpoint errors. Delays for trip actuations bound those values allowed by the Salem Unit 1 and 2 Technical Specifications.

c. 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.

d. 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.

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

Results:

Figures 4.1.2-1 and 4.1.2-2 show the transient response for a rapid RCCA withdrawal (75 pcm/sec) incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident.

The transient response for a slow RCCA bank withdrawal from full power is shown in Figures 4.1.2-3 and 4.1.2-4. Reactor trip on OTDT occurs after a longer period and the rise in temperature is consequently larger than for rapid RCCA bank withdrawal.

Figure 4.1.2-5 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 OTDT reactor trip channels provide protection over the whole range of reactivity insertion rates.

Figures 4.1.2-6 and 4.1.2-7 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents initiating from 60% and 10% power levels, respectively, for minimum and maximum reactivity feedback. The results are similar to the 100% power case, except as the initial power is decreased, the range over which the OTDT trip is effective is increased.

The shape of the curves of minimum DNBR versus reactivity insertion rate in the reference figures is due to reactor core and coolant system transient response and to protection system action in initiating a reactor trip. In all cases margin to DNB is maintained as the DNBR calculated meets the safety analysis DNBR limit.

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 4.1.2-1.

Conclusions:

In the event of an uncontrolled RCCA bank withdrawal at power, the high neutron flux, high pressurizer pressure, and OTDT trip channels provide adequate protection over the entire range of possible reactivity insertion rates. The 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 4.1.2-1

Sequence of Events
Uncontrolled RCCA Withdrawal at Power

High Reactivity Insertion Rate

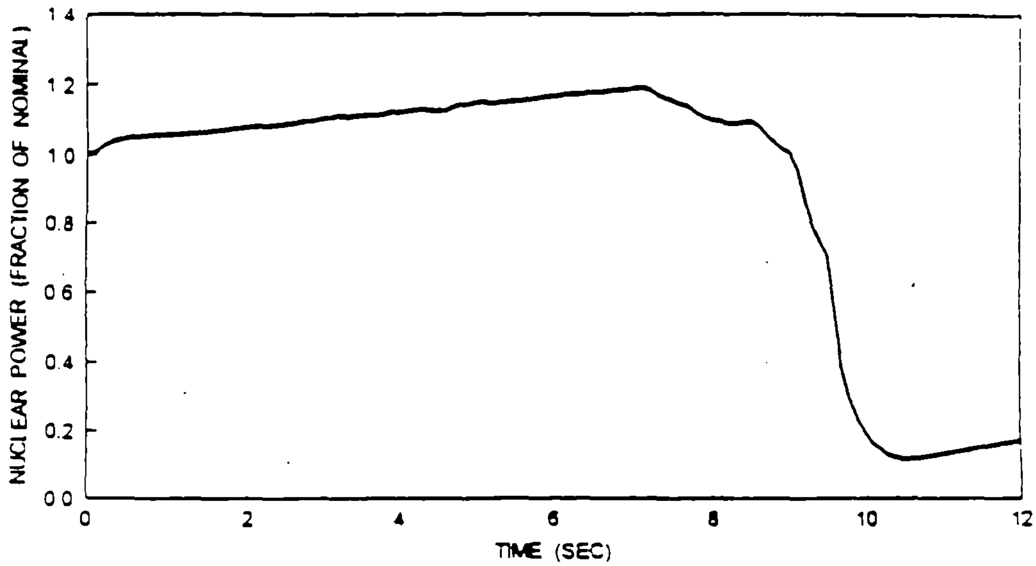
Event	Time (seconds)
Initiation of uncontrolled RCCA withdrawal at a high reactivity insertion rate (75 pcm/sec)	0
Power-range high neutron flux high trip point reached	6.6
Rod motion begins	7.1
Minimum DNBR occurs	7.4

Low Reactivity Insertion Rate

Event	Time (seconds)
Initiation of uncontrolled RCCA withdrawal at a low reactivity insertion rate (3 pcm/sec)	0
Overtemperature ΔT reactor trip signal initiated	472.3
Rod motion begins	473.8
Minimum DNBR occurs	474.1

Figure 4.1.2-1

Uncontrolled RCCA Bank Withdrawal At Power
(75 pcm/sec - Full Power) High Neutron Flux Trip
(Maximum Feedback)



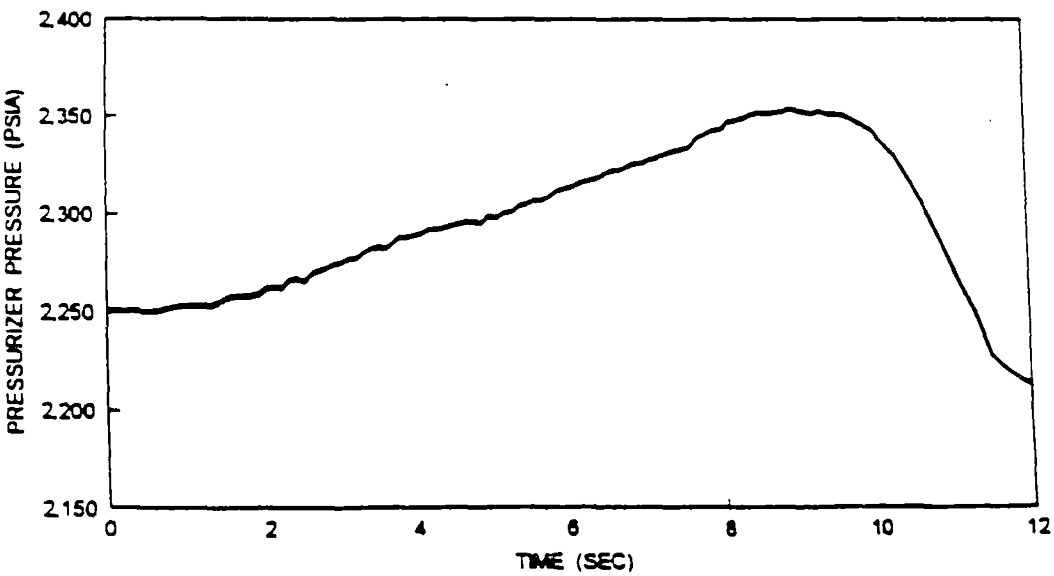
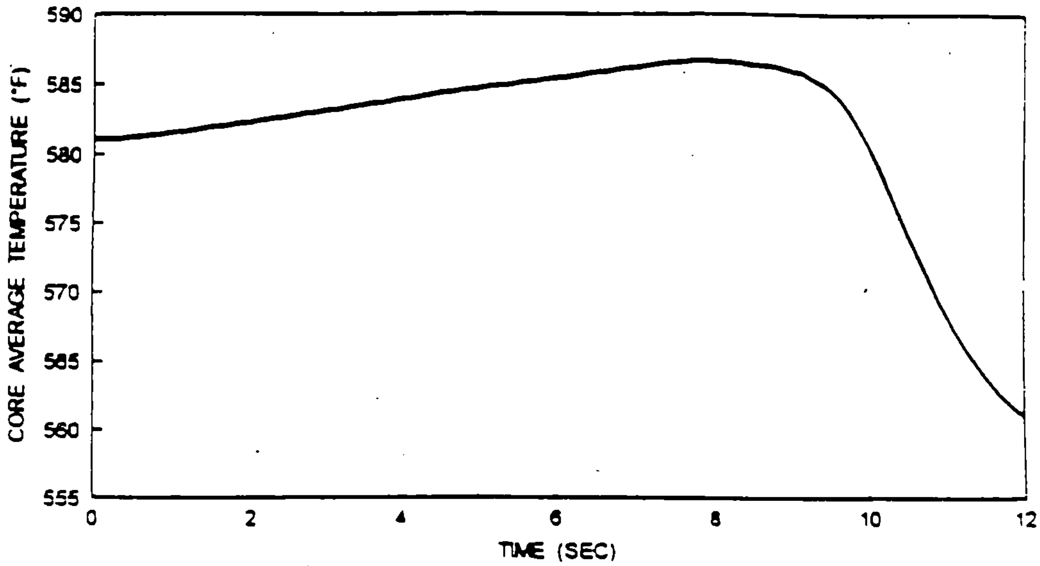


Figure 4.1.2-2

Uncontrolled RCCA Bank Withdrawal At Power
(75 pcm/sec - Full Power) High Neutron Flux Trip
(Maximum Feedback)



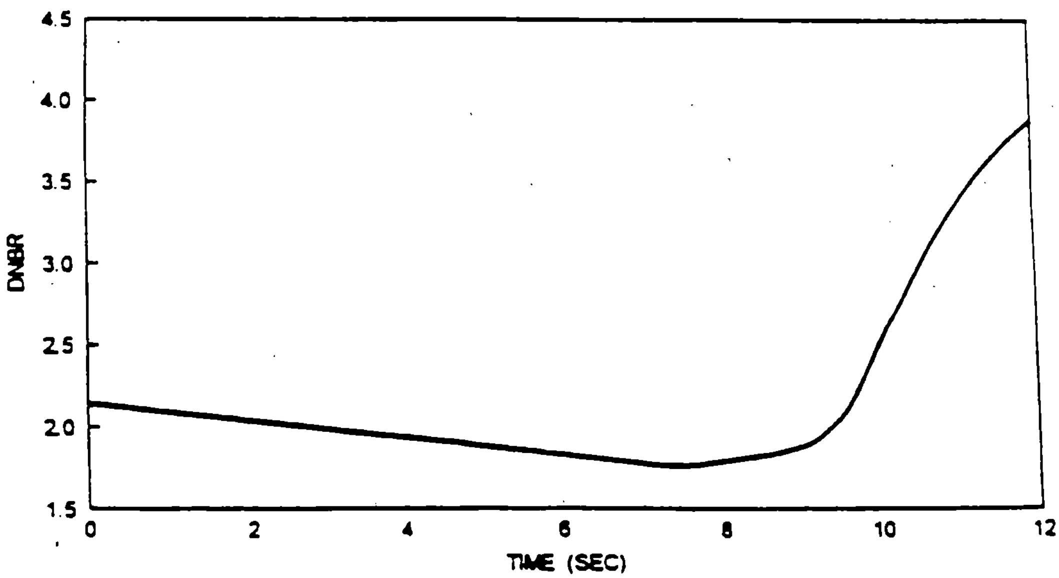
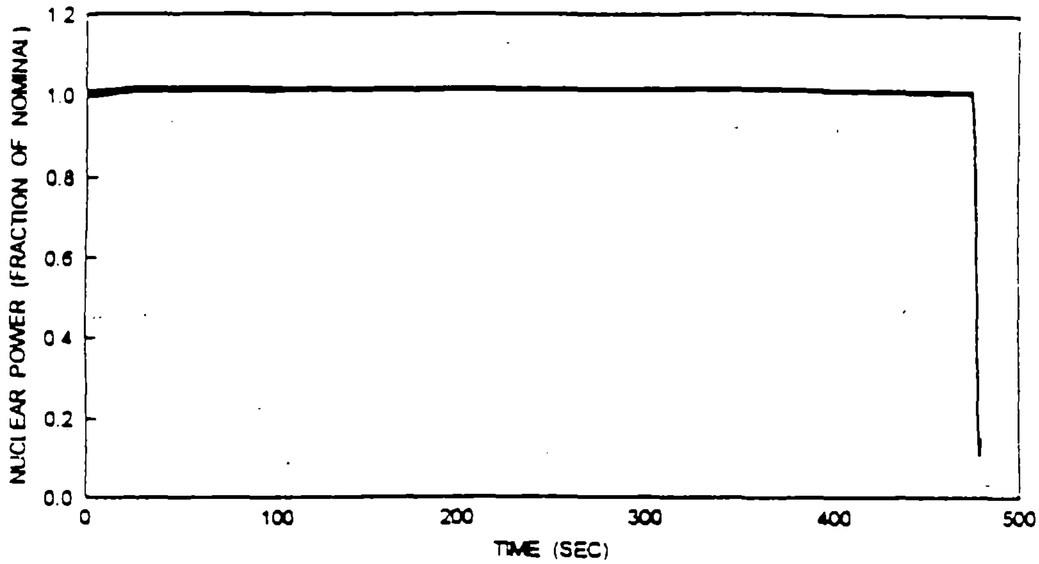


Figure 4.1.2-3

Uncontrolled RCCA Bank Withdrawal At Power
(3 pcm/sec - Full Power) Overtemperature ΔT Trip
(Maximum Feedback)



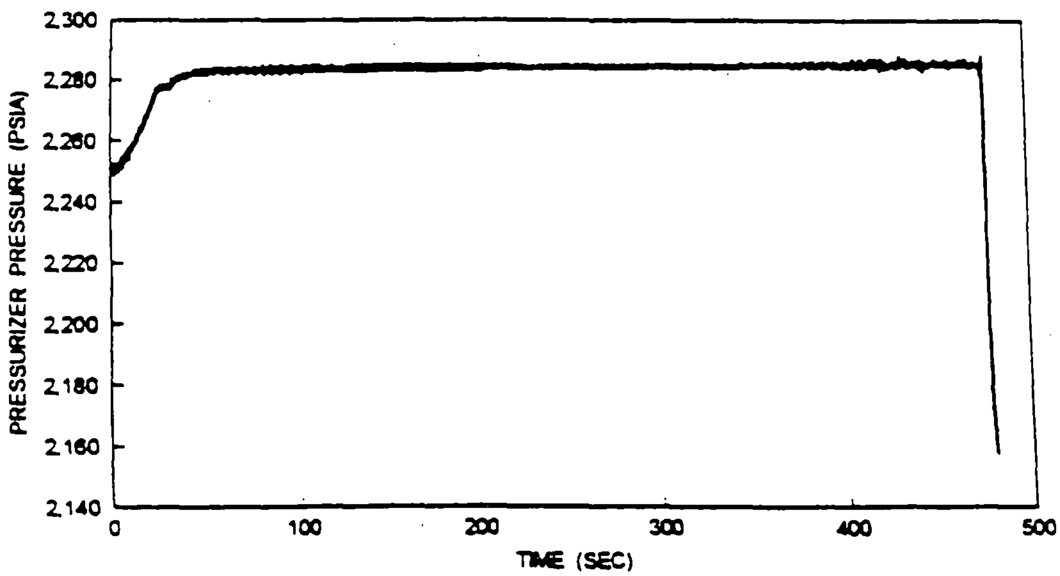


Figure 4.1.2-4

Uncontrolled RCCA Bank Withdrawal At Power
(3pcm/sec - Full Power) Overtemperature ΔT Trip
(Maximum Feedback)

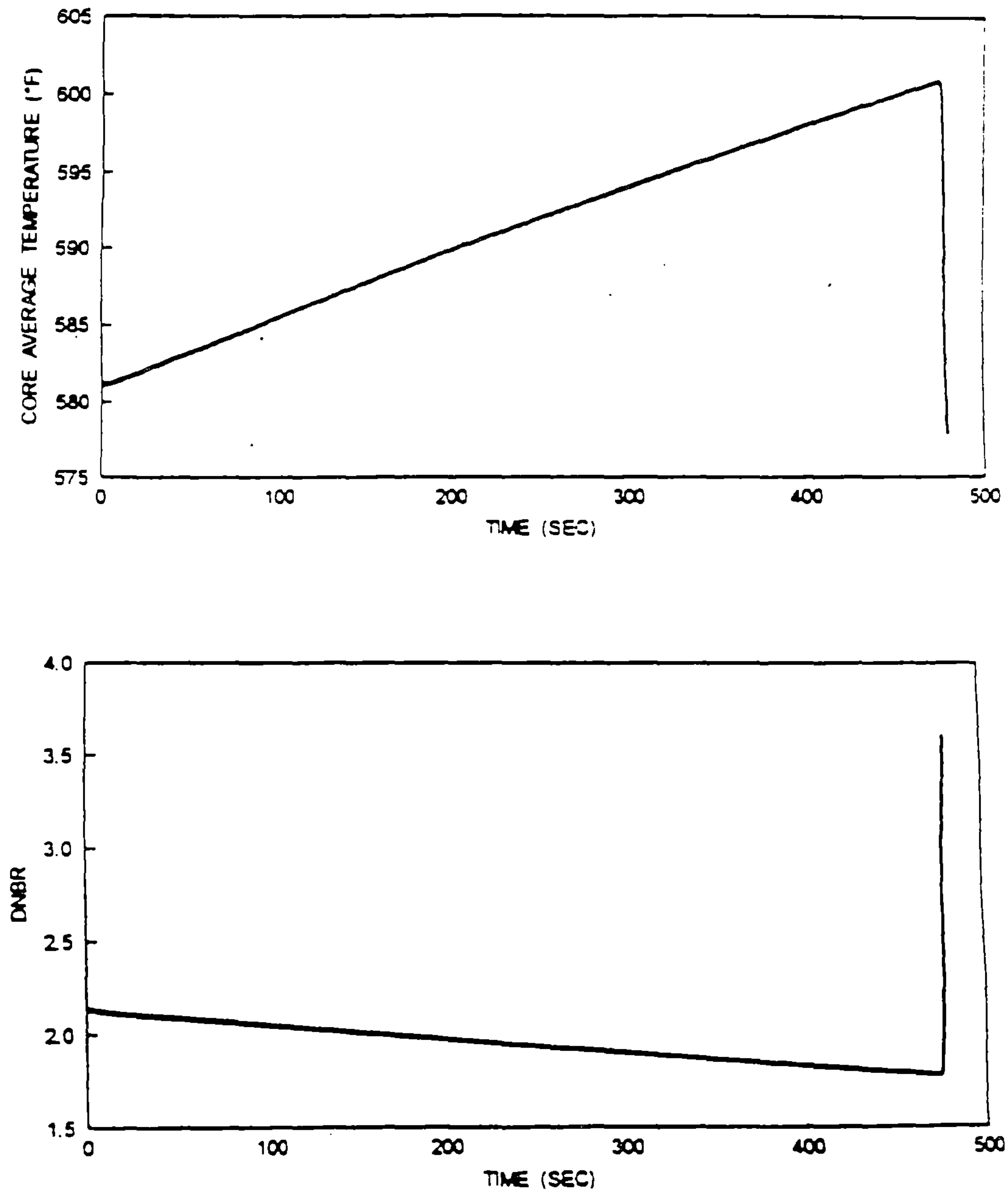


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Figure 4.1.2-5

Uncontrolled RCCA Bank Withdrawal At Power
(Full Power)

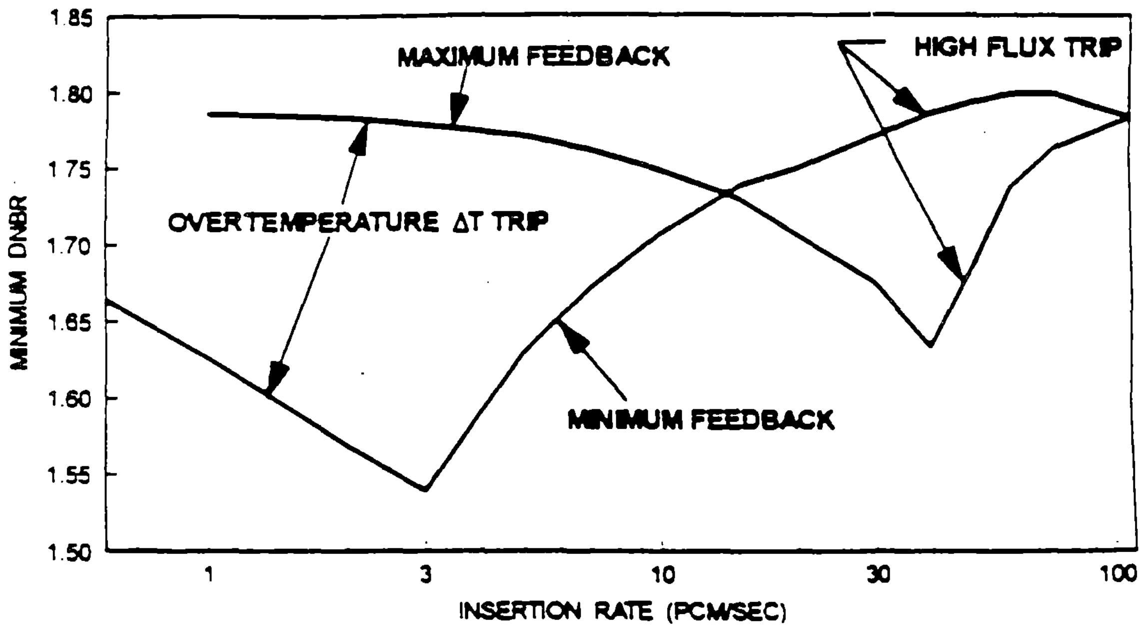

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Figure 4.1.2-6

Uncontrolled RCCA Bank Withdrawal At Power
(60% Power)

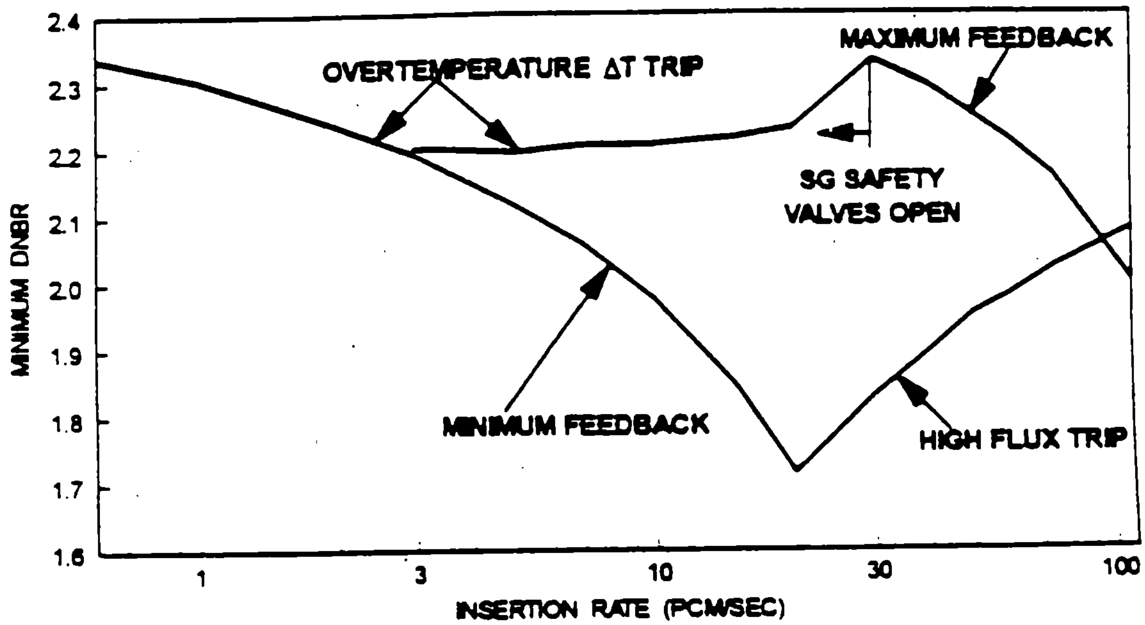

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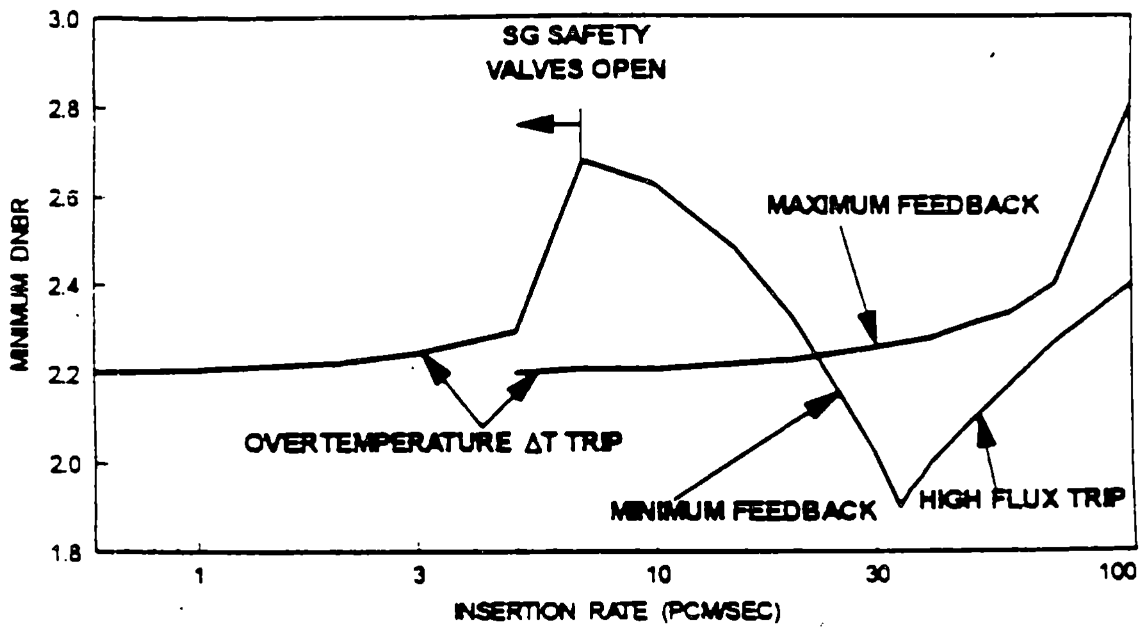
Figure 4.1.2-7

Uncontrolled RCCS Bank Withdrawal At Power
(10% Power)



4.1.3 Rod Cluster Control Assembly Misalignment (UFSAR 15.2.3)

Accident Description:

The Condition II rod cluster control assembly (RCCA) misalignment accidents include:

a. A dropped full-length assembly;

b. a dropped full-length assembly bank; and

c. statically misaligned assembly.

A dropped RCCA or RCCA banks are detected by:

a. Sudden drop in the core power level as seen by the Nuclear Instrumentation System;

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

c. Rod bottom light(s);

d. Rod deviation alarm; and

e. Rod position indication.

Misaligned RCCAs are detected by:

a. Technical Specification surveillances;

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

c. Rod deviation alarm; and

d. Rod position indicators.

Method of Analysis:

One or More Dropped RCCAs from the Same Group

The LOFTRAN computer code calculates the transient system response for the evaluation of the dropped RCCA event. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and main steam safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Transient RCS statepoints (temperature, pressure and power) are calculated by LOFTRAN. Nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient analysis and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code. Note that the analysis does not take credit for the power-range negative flux rate reactor trip.

Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core.

Statically Misaligned RCCA (Single RCCA)

Steady-state power distributions are analyzed using appropriate nuclear physics computer codes. The peaking factors are then used as input to the THINC code to calculate the DNBR. The analysis examines the following cases: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out, all with the reactor initially at full power. The analysis assumes this incident to occur at BOL since this results in the least-negative value of the MTC. This assumption minimizes the tendency of the MTC to flatten the power distribution.

Statically Misaligned RCCAs

Steady-state power distributions and peaking factors are used to calculate the DNBR. The analysis examines the following cases: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out, all with the reactor initially at full power. The analysis assumes this incident to occur at BOL since this results in the least-negative value of the MTC. This assumption minimizes the tendency of the MTC to flatten the power distribution.

Results:

One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period, since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will re-establish power.

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. Without control system interaction, a new equilibrium is achieved at a reduced power level and reduced primary temperature. Thus, the automatic rod control mode of operation is the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 4.1.3-1 and 4.1.3-2 show a typical transient response to a dropped RCCA (or RCCAs) in the automatic rod control mode. In all cases, the minimum DNBR remains above the limit value.

Dropped RCCA Bank

A dropped RCCA bank results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed as described previously in the "One or More Dropped RCCAs" section; however, the return to power will be less due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric.

Statically Misaligned RCCA (Single RCCA)

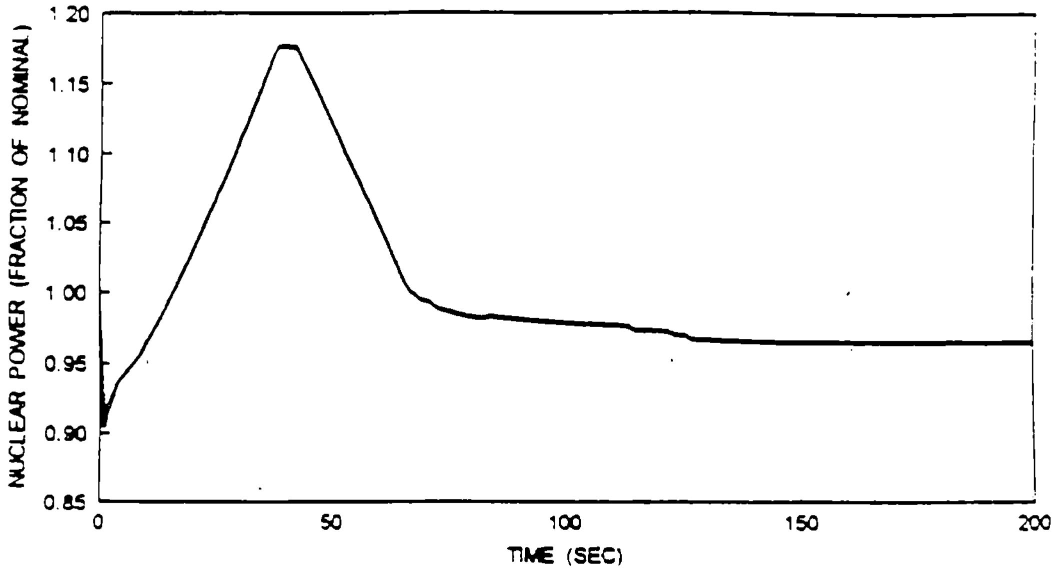
The most-severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted with either all rods out or bank D at its insertion limit, or where bank D is inserted to its insertion limit with one RCCA fully withdrawn. Multiple independent 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 or inserted without the DNBR falling below the limit value.

Conclusions:

For cases of dropped RCCAs or dropped banks encompassing all possible dropped rod worths, the DNBR remains greater than the limit value; therefore, the DNB design criterion is met. For all cases of any single or multiple RCCAs fully inserted, or bank D inserted to its rod insertion limits with any single or multiple RCCAs in that bank fully withdrawn (static misalignment), the DNBR remains greater than the limit value.

Figure 4.1.3-1

Transient Response to Dropped
Rod Cluster Control Assembly



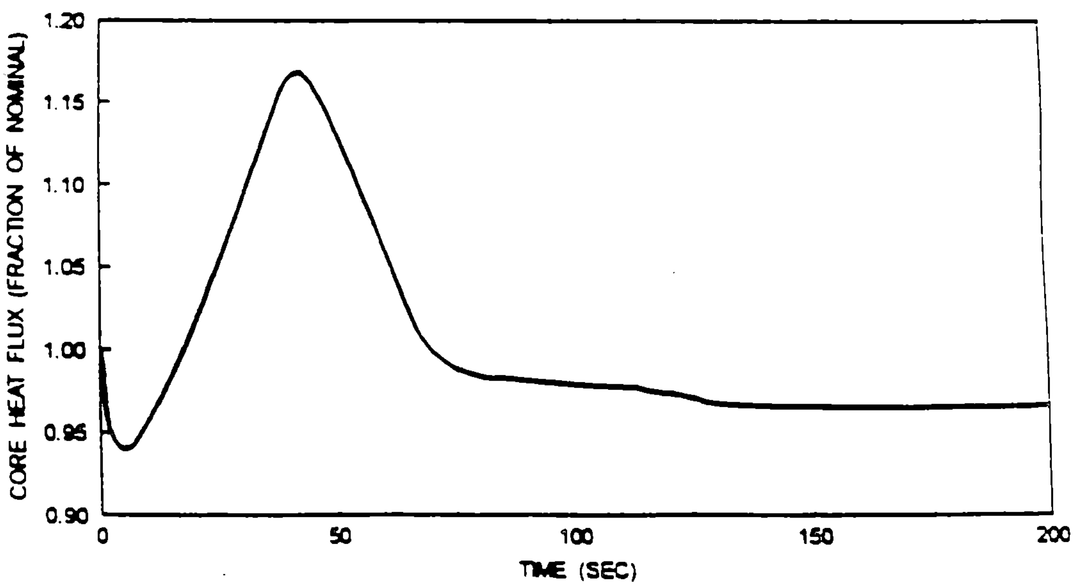
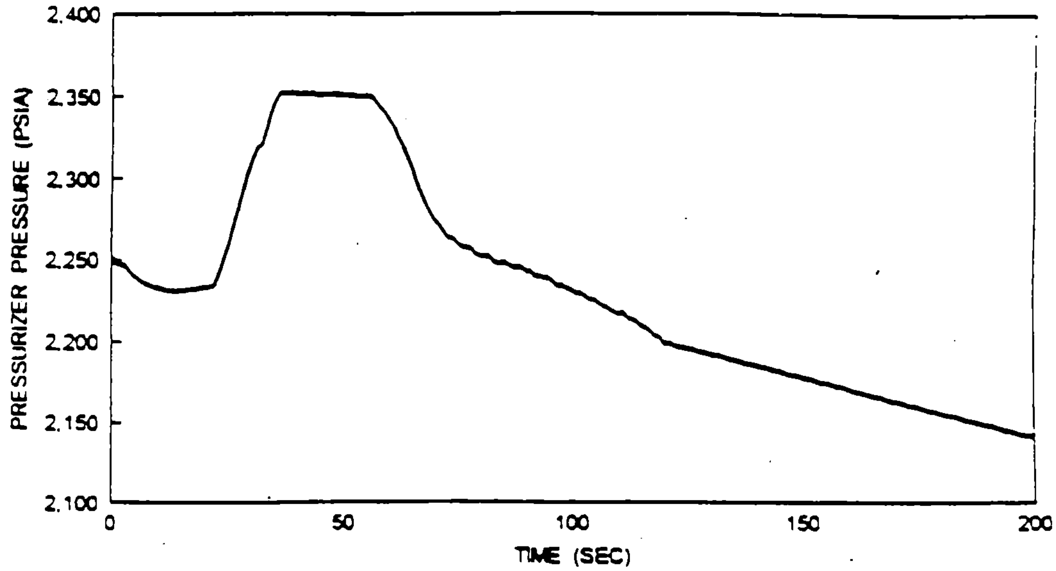
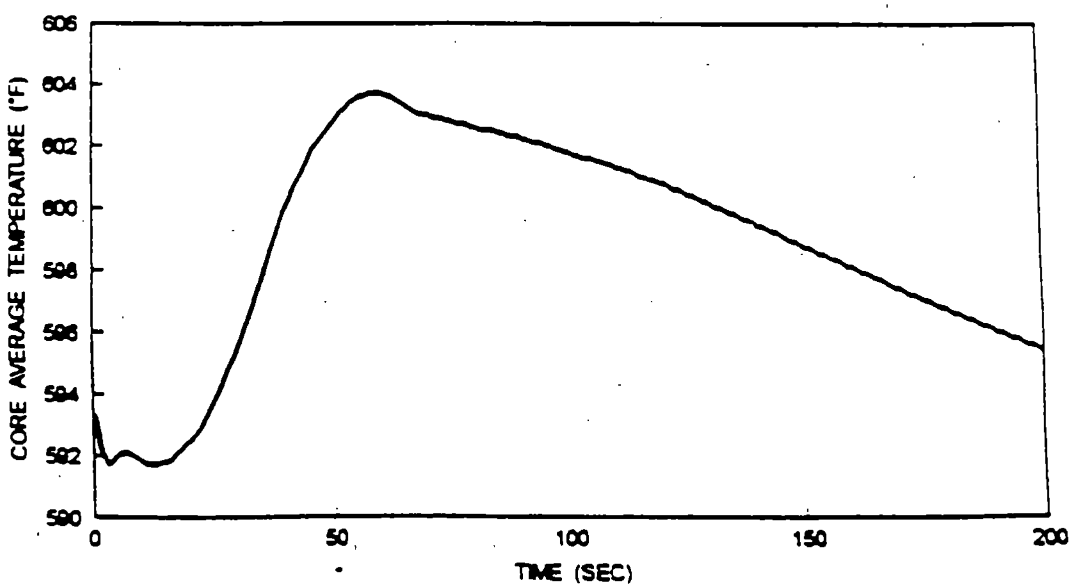


Figure 4.1.3-2

Transient Response to Dropped
Rod Cluster Control Assembly





4.1.4 Uncontrolled Boron Dilution (UFSAR 15.2.4)

Accident Description:

Reactivity can be added to the core by feeding primary-grade water into the RCS via the reactor makeup portion of the chemical and volume control system. Boron dilution is a manual operation under strict administrative controls, with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. 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.

Method of Analysis:

Dilution During Refueling (Mode 6)

Administrative controls help preclude an uncontrolled boron dilution accident from occurring by isolating the reactor makeup water system from the RCS during refueling. Dilution flow is the maximum capacity of the primary makeup water pumps, 300 gpm.

Dilution During Startup (Mode 2)

During startup rod control is assumed to be in manual. All normal actions required to change power level require operator initiation.

Conditions assumed for the analysis are:

Dilution flow is the maximum capacity of the primary makeup water pumps, 300 gpm.

A minimum RCS water volume corresponding to the active RCS volume (e.g., not including the pressurizer volume) and accounting for 20% steam generator (SG) tube plugging is assumed.

Dilution at Power

With the unit at power and the RCS at pressure, the dilution rate is limited by the capacity of the charging pumps. Dilution flow is 236 gpm. The reactivity rate used in the following evaluation is approximately 1 pcm/sec.

A minimum RCS water volume corresponding to the active RCS volume (e.g., not including the pressurizer volume) and accounts for 20% SG tube plugging.

The at power event is analyzed for manual and automatic rod control.

Results:

Dilution During Refueling

For dilution during refueling, the minimum time required for the SDM to be lost and the reactor to become critical is 30 minutes.

For Dilution During Startup

For dilution during startup, the minimum time required for the SDM to be lost and the reactor to become critical is 19 minutes.

For Dilution at Power

With the reactor in automatic control at full power, the power and temperature increase from a boron dilution results in the insertion of the rod cluster control assemblies (RCCA) and decrease in shutdown margin (SDM). Continuation of the dilution and RCCA insertion would cause the assemblies to reach the minimum limit of the rod insertion monitor. Before reaching this point, however, two alarms are actuated to warn the operator of the accident condition. The first of these, the low insertion limit alarm, alerts the operator to initiate normal boration. The other, the low-low insertion limit alarm, alerts the operator to follow emergency boration procedures.

The low alarm is set sufficiently above the low-low alarm to allow normal boration without the need for emergency procedures. If dilution continues after reaching the low-low alarm, it takes approximately 18.3 minutes before SDM is lost due to the dilution.

With the reactor in manual control, if no operator action is taken, the power and temperature rise causes the reactor to reach the OTDT trip setpoint. The boron dilution accident in this case is essentially identical to a rod cluster control assembly withdrawal accident at power. Prior to the OTDT trip, an OTDT alarm and turbine runback would be actuated. There is time available (approximately 16.3 minutes) after a reator trip for the operator to determine the cause of the dilution, isolate the source, and initiate reboration before a return to criticality would occur.

The time sequence of events is provided in Table 4.1.4-1.

Conclusions:

The boron dilution analyses have shown acceptable results given the MRP implementation.

Table 4.1.4-1

Sequence of Events
Uncontrolled Boron Dilution

	Event	Time (seconds)
Mode 6	Dilution begins	0
	Operator isolates source of dilution; minimum margin to criticality occurs	1800
Mode 2	Dilution begins	0
	Operator isolates source of dilution; minimum margin to criticality occurs	1140
Mode 1 Automatic reactor control	Dilution begins	0
	1.3% shutdown margin lost	1098
Mode 1 Manual reactor control	Dilution begins	0
	OTΔT reactor trip signal reached	120
	Rod motion begins	122
	1.3% shutdown is lost if dilution continues after trip	1098

4.1.5 Loss of Forced Reactor Coolant Flow

The Loss of Reactor Coolant Flow events consist of the following transients:

4.1.5.1 Partial Loss of Reactor Coolant Flow (UFSAR 15.2.5)
4.1.5.2 Complete Loss of Reactor Coolant Flow (UFSAR 15.3.4)
4.1.5.3 Single Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break (UFSAR 15.4.5)

4.1.5.1 Partial Loss of Forced Reactor Coolant Flow (UFSAR 15.2.5)

Accident Description:

A Condition II event, a partial loss of flow accident can result from a mechanical or electrical failure in an RCP, or from a fault in the power supply to the RCP. If the reactor is at power at the time of the accident, the immediate effect of a loss of 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.

The necessary protection against a partial loss of coolant flow 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. Above Permissive 8, low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip.

Method of Analysis:

The loss of two reactor coolant pumps with four loops in operation is analyzed to show that the integrity of the core is maintained as the DNBR remains above the safety analysis limit value.

The partial loss of flow (PLOF) event is analyzed with three computer codes. First, the LOFTRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN computer code is then used to calculate the heat flux transient based on the nuclear power and RCS flow from LOFTRAN. Finally, the THINC computer code 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.

This event is analyzed with the RTDP. Initial reactor power, pressurizer pressure and RCS temperature are assumed to be at their nominal values. A conservatively large absolute value of the DPC 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 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.

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 £ 20%, and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of £ 20% and a maximum steam generator tube plugging level of 25% in any steam generator.

<u>Results:</u>

Figures 4.1.5.1-1 and 4.1.5.1-2 illustrate the transient response for the PLOF event. Figure 4.1.5.1-2 shows that the DNBR always remains above the limit value. The PLOF minimum DNBR is greater than the more limiting DNBR calculated for the complete loss of flow underfrequency event (Section 4.1.5.2).

The calculated sequence of events table is shown on Table 4.1.5-1.

Conclusions:

The analysis performed has demonstrated that for the partial loss of flow event, the DNBR does not decrease below the safety analysis limit value at any time during the transient.

Table 4.1.5-1

Sequence of Events
Partial Loss of Forced Reactor Coolant Flow

Event	Time (seconds)
Coastdown begins	0.0
Low flow reactor trip	1.6
Rod motion begins	2.6
Minimum DNBR occurs	3.9
Maximum RCS Pressure occurs	4.7

Figure 4.1.5.1-1

All Loops Operating
Two Loops Coasting Down (PLOF)

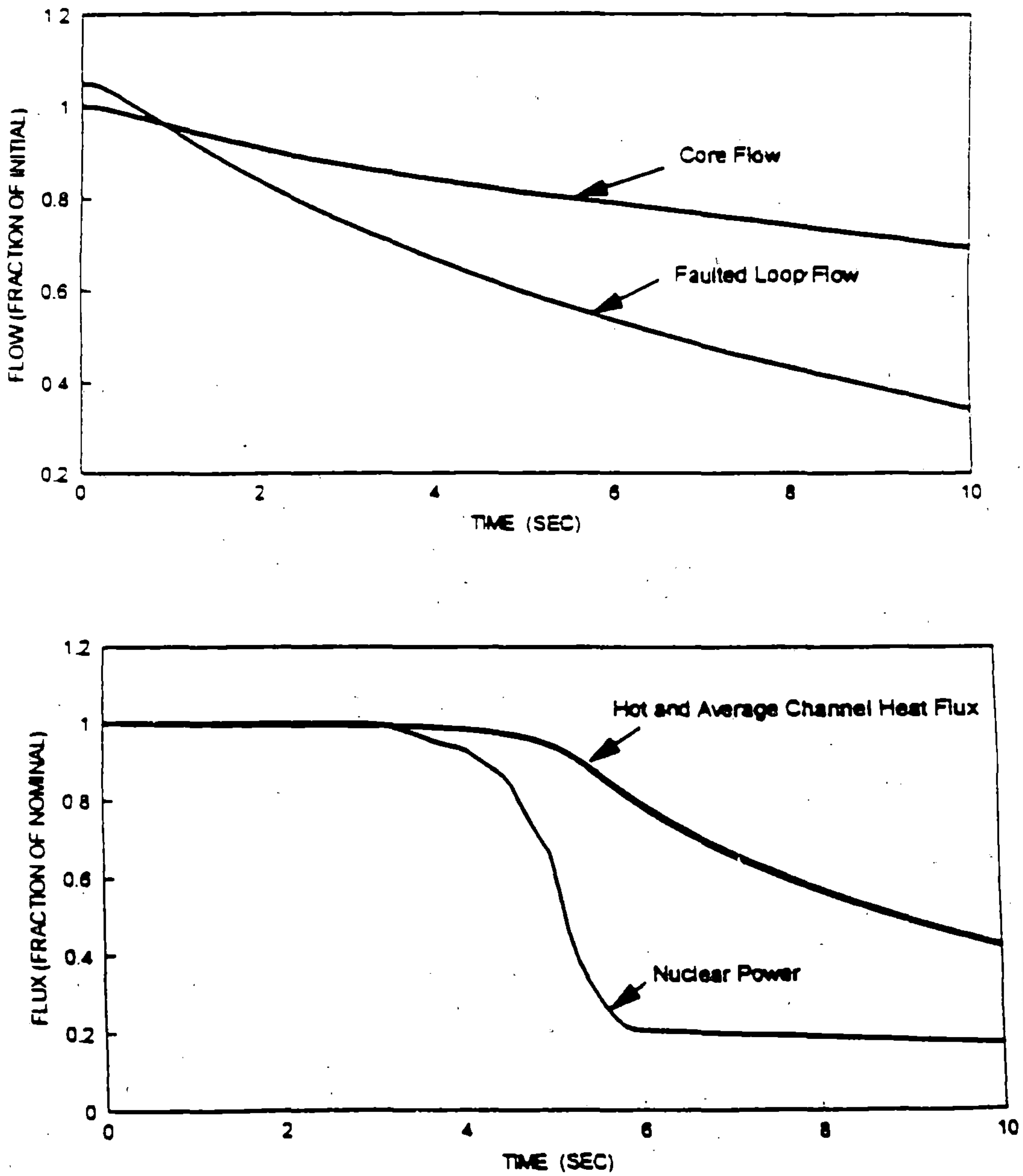
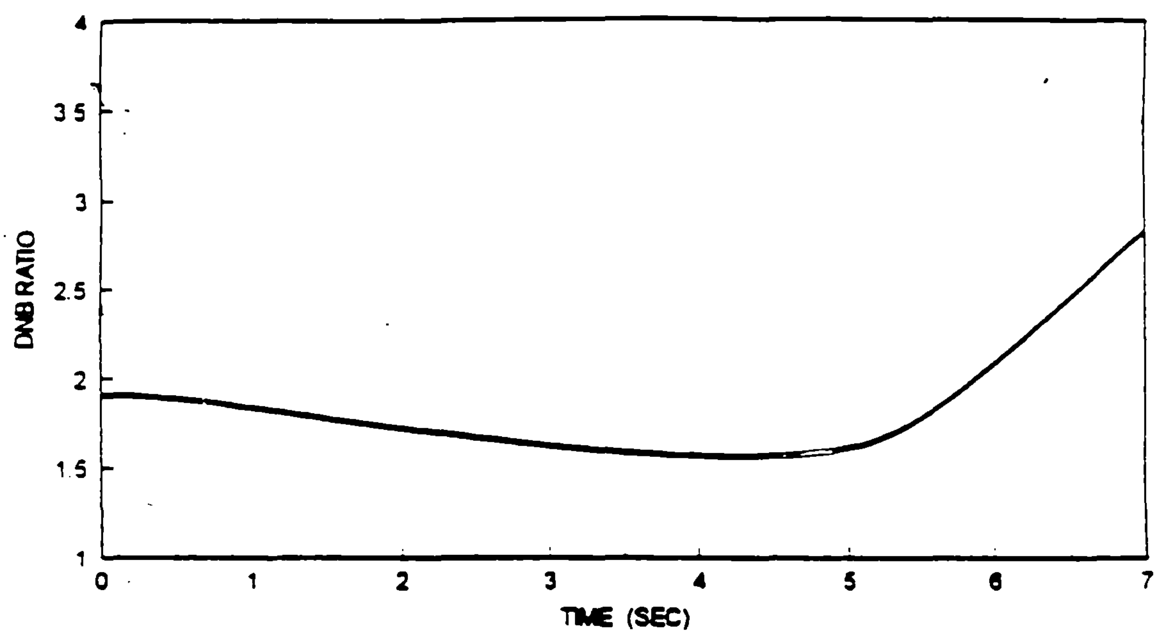


Figure 4.1.5.1-2

All Loops Operating
Two Loops Coasting Down (PLOF)



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4.1.5.2 Complete Loss of Forced Reactor Coolant Flow (UFSAR 15.3.4)

Accident Description:

A Condition III event, 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 a complete loss of flow (CLOF) is a rapid increase in the coolant temperature.

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

a. Reactor coolant pump power supply undervoltage or underfrequency;

b. Low reactor coolant loop flow; and

c. Pump circuit breaker opening.

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% 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 one or more reactor coolant loops. The reactor trip from pump break position is provided as an anticipatory signal which serves as a backup to the low flow signals.

Although the CLOF is defined as a Condition III event, the event is conservatively analyzed to Condition II criteria.

Method of Analysis:

The complete loss of flow transient is a loss of all four reactor coolant pumps with four loops in operation. The following two cases are analyzed:

a. Complete loss of flow transient due to an undervoltage condition; and

b. Complete loss of flow transient due to an underfrequency condition.

The transient is analyzed with three computer codes. First, the LOFTRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN computer code is then used to calculate the heat flux transient based on the nuclear power and RCS flow from LOFTRAN. Finally, the THINC computer code 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.

This event is analyzed with the RTDP. A conservative trip reactivity of 4% 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.

Results:

Figures 4.1.5.2-1 and 4.1.5.2-2 illustrate the transient response for the CLOF (undervoltage) for a loss of power to all four reactor coolant pumps. Figure 4.1.5.2-2 shows that the DNBR always remains above the safety analysis limit value. The undervoltage minimum DNBR is greater than the more limiting underfrequency event DNBR.

Figures 4.1.5.2-3 and 4.1.5.2-4 illustrate the transient response for the CLOF (underfrequency) with a frequency decay of all four reactor coolant pumps. Figure 4.1.5.2-4 shows that the DNBR always remains above the limit value.

The calculated sequence of events for both CLOF cases (undervoltage and underfrequency) are shown on Table 4.1.5-2.

Conclusions:

The analysis performed has demonstrated that for the complete loss of flow event, the DNBR does not decrease below the limit value at any time during the transient.

Table 4.1.5-2

Sequence of Events
Complete Loss of Forced Reactor Coolant Flow

Undervoltage

Event	Time (seconds)
All operating RCPs lose power and costdown begins	0.0
Undervoltage reactor trip	0.0
Rod motion begins	1.5
Minimum DNBR occurs	3.4
Maximum RCS Pressure occurs	4.4

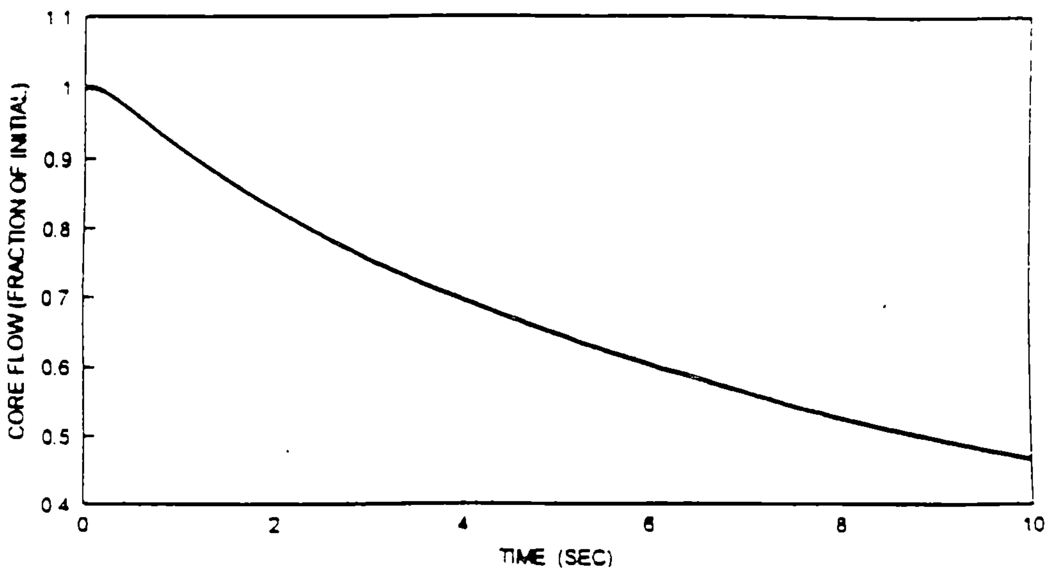
Underfrequencey

Event	Time (seconds)
Frequency decay begins and RCS flowis reduced	0.0
Rod motion begins due to Underfrequency Reactor Trip	1.8[1]
Minimum DNBR occurs	3.9
Maximum RCS Pressure occurs	4.8

1 Assumes a trip setpoint of 53.9 Hz, a frequency decay of 5 Hz/sec, and a delay of 0.6 sec.

Figure 4.1.5.2-1

All Loops Operating
All Loops Costing Down
(CLOF Undervoltage)



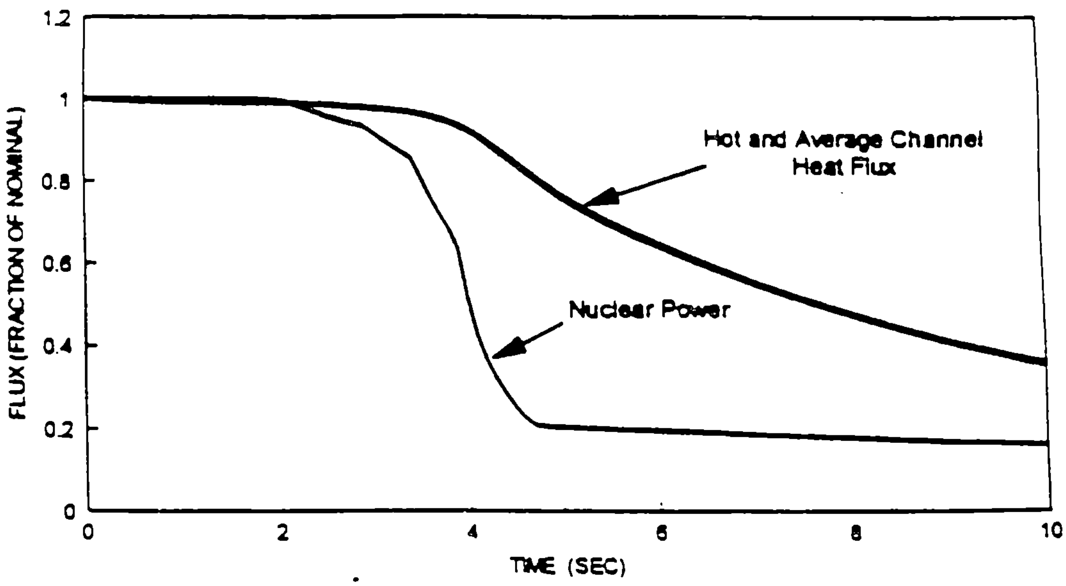


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Figure 4.1.5.2-2

All Loops Operating
All Loops Coasting Down
(CLOF Undervoltage)

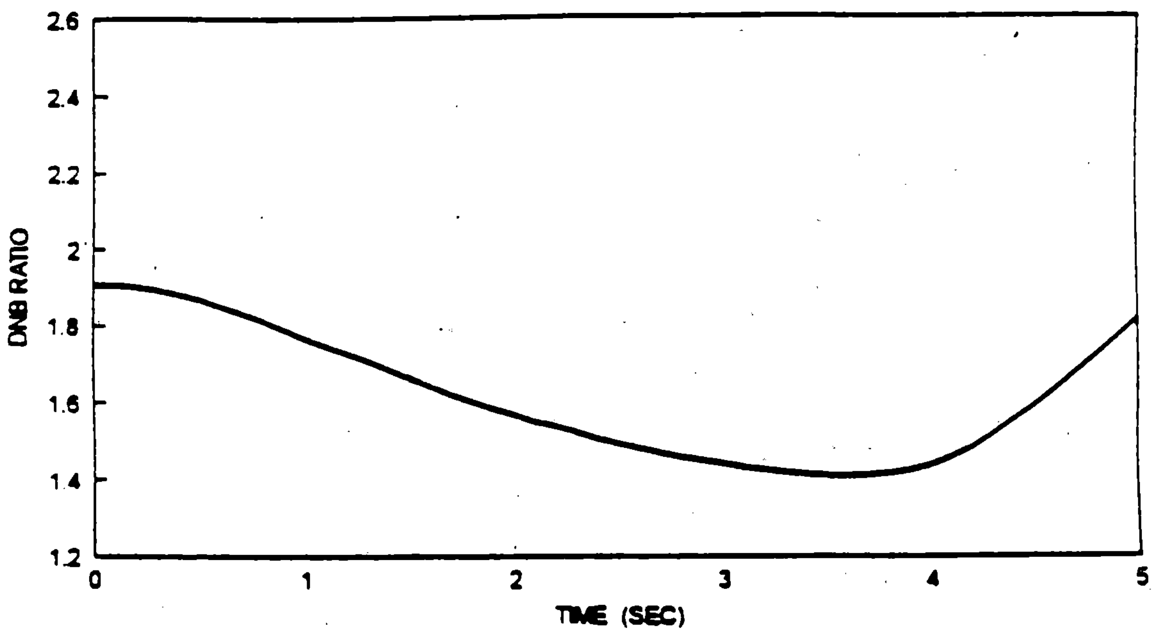


Figure 4.1.5.2-3

All Loops Operating
All Loops Coasting Down
(CLOF underfrequency)

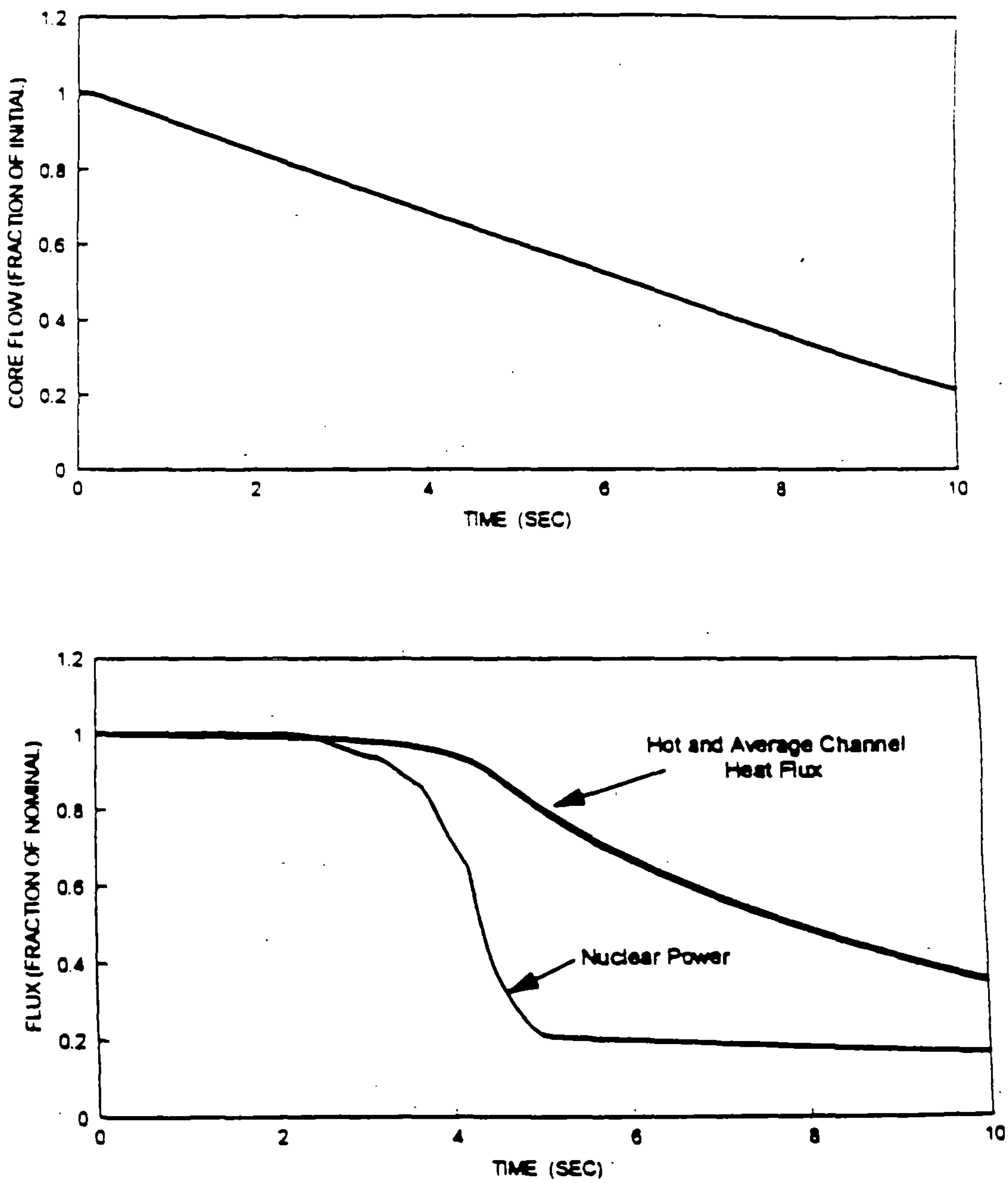
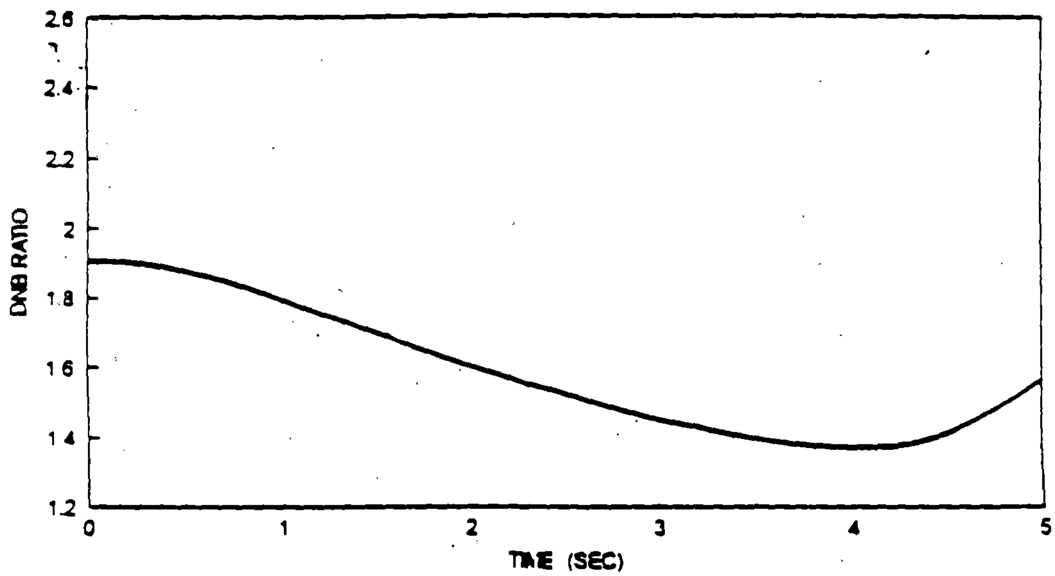


Figure 4.1.5.2-4

All Loops Operating
All Loops Coasting Down
(CLOF Underfrequency)



4.1.5.3 Single Reactor Coolant Pump Locked Rotor and Reactor Coolant Pump Shaft Break (UFSAR 15.4.5)

Accident Description:

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

Method of Analysis:

The RCS pressurization part of the RCP Locked Rotor transient is analyzed with two computer codes. First, the LOFTRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN computer code is then used to calculate the thermal behavior of the fuel located at the core hot spot based on the nuclear power and RCS flow from LOFTRAN.

At the beginning of the postulated RCP Locked Rotor accident, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., a maximum 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 £ 20%, and 2) asymmetric steam generator tube plugging conditions with an average steam generator tube plugging level of £ 20% and a maximum steam generator tube plugging level of 25% in any one steam generator.

A conservatively large absolute value of the DPC 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 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.

For the peak RCS pressure evaluation, the initial pressure is conservatively estimated as 50 psi above the nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. The peak RCS pressure occurs at the pump outlet. The pressure transient at the pump outlet is shown in Figure 4.1.5.3-3.

For this accident, DNB is assumed to occur in the core, therefore an evaluation of the consequences with respect to the fuel rod thermal transients is performed. Two DNB-related analyses are performed. The first incorporates the assumption of rods going into DNB as a conservative initial condition to determine the clad temperature and zirconium water reaction.

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 3.0 times the average rod power (i.e., $F_Q = 3.0$) at the initial core power level. The F_Q of 3.0, which includes an allowance for nuclear power peaking due to fuel densification, was conservatively used for the LR analysis and bounds the MRP value of $F_Q = 2.4$.

The second DNB related analysis is performed to determine the percentage of rods, if any, is expected to be in DNB during the transient. Analyses to determine this percentage for the locked rotor and shaft break accidents use three digital computer codes. In addition to the LOFTRAN and FACTRAN codes, the THINC code is used to calculate DNBR during the transient, based on flow calculated by LOFTRAN and heat flux calculated by FACTRAN. This second analysis is analyzed with the RTDP.

Results:

Figures 4.1.5.3-1 through 4.1.5.3-3 illustrate the transient response for the RCP Locked Rotor event. The peak RCS pressure is less than that which would cause stresses to exceed the faulted condition stress limits. The zirconium-steam reaction at the hot spot meets the criterion of less than 16% zirconium-steam water reaction. Less than 5% of the total fuel rods experience DNB. The sequence of events is given in Table 4.1.5-3.

Conclusions:

In the event of a Locked Rotor or Shaft Break, all safety criteria are satisfied. This demonstrates that the RCS and the core will remain able to provide long term cooling given the MRP implementation.

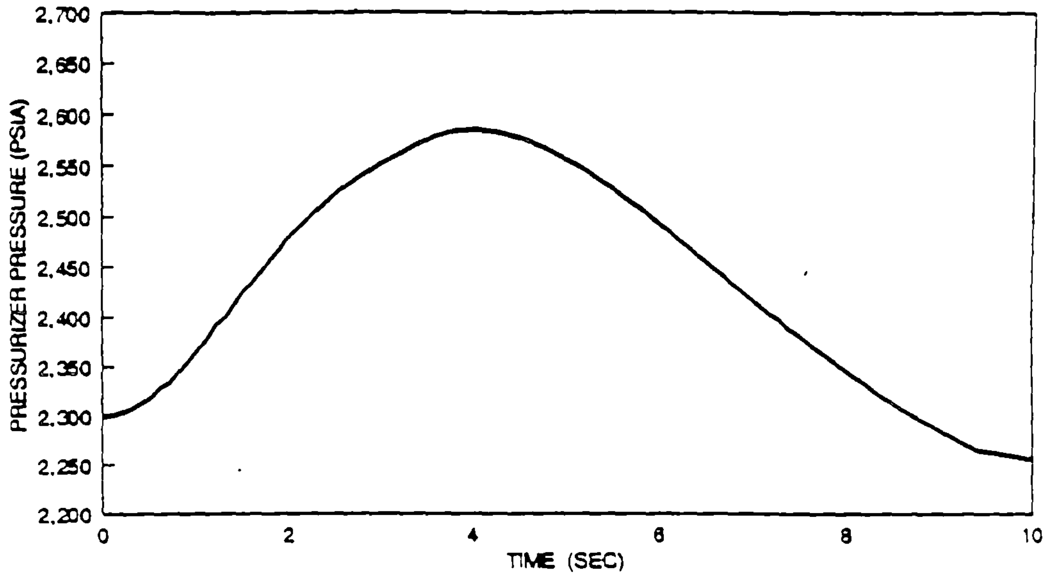
Table 4.1.5-3

Sequence of Events
Reactor Coolant Pump Locked Rotor/Shaft Break

Event	Time (seconds)
Rotor on one pump locks	0.00
Low flow reactor trip	0.03
Rod motion begins	1.03
Reactor Coolant Pumps Coastdown	2.53
Maximum clad temperature occurs	3.7
Maximum RCS pressure occurs	3.8

Figure 4.1.5.3-1

All Loops Operating
One Locked Rotor (LR)



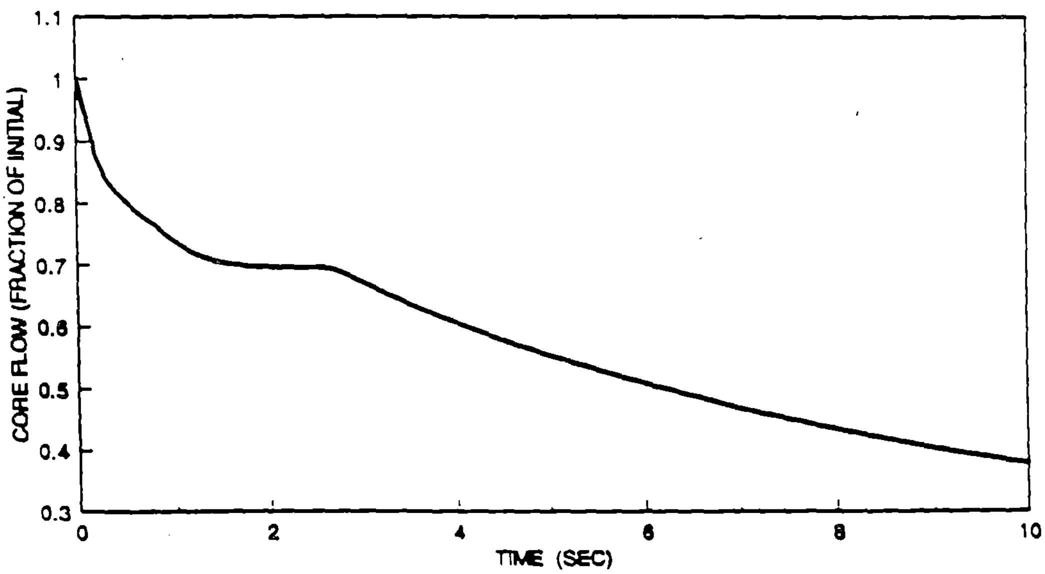
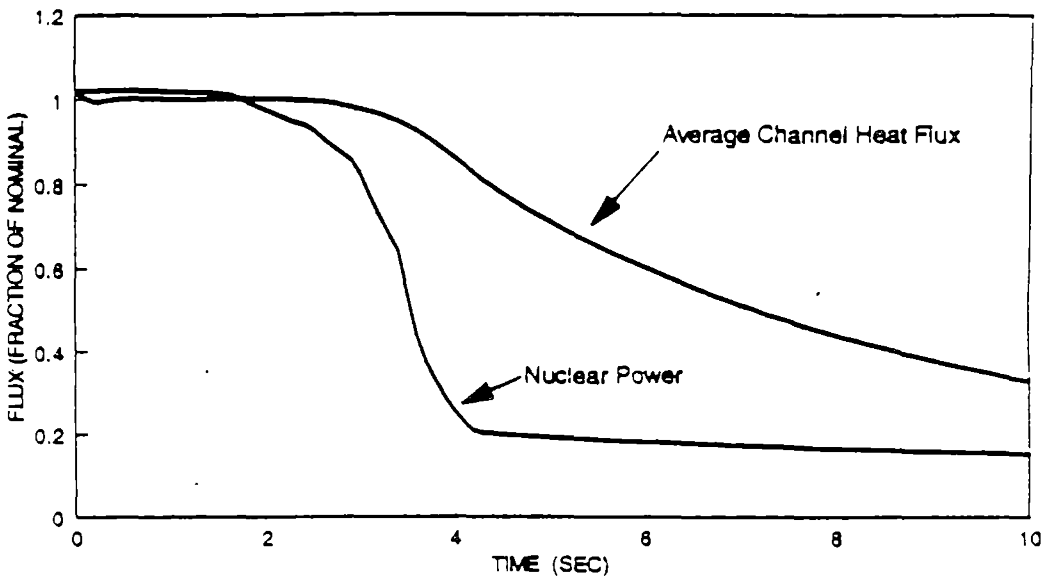


Figure 4.1.5.3-2

All Loops Operating
One Locked Rotor (LR)



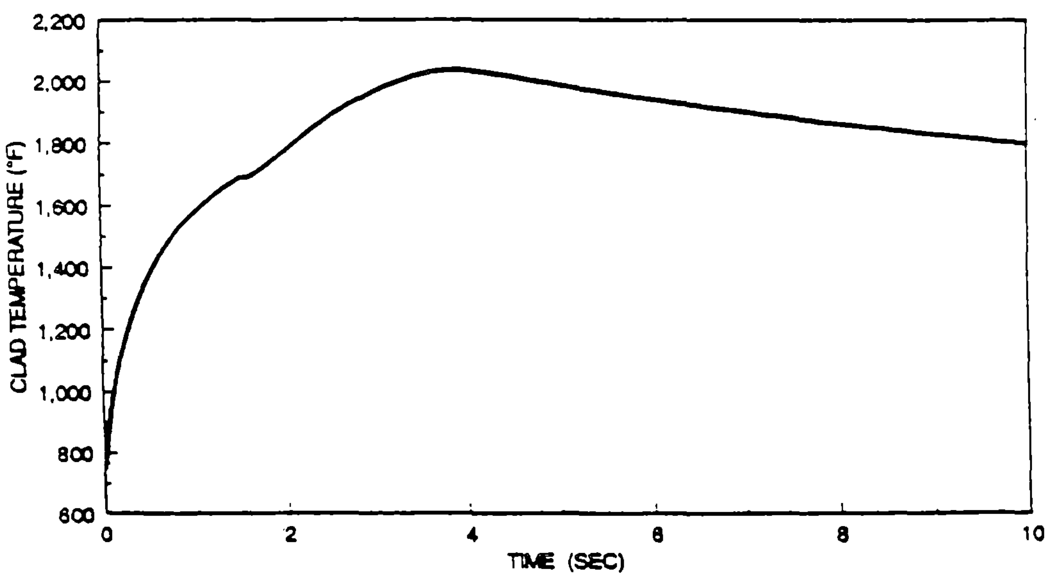
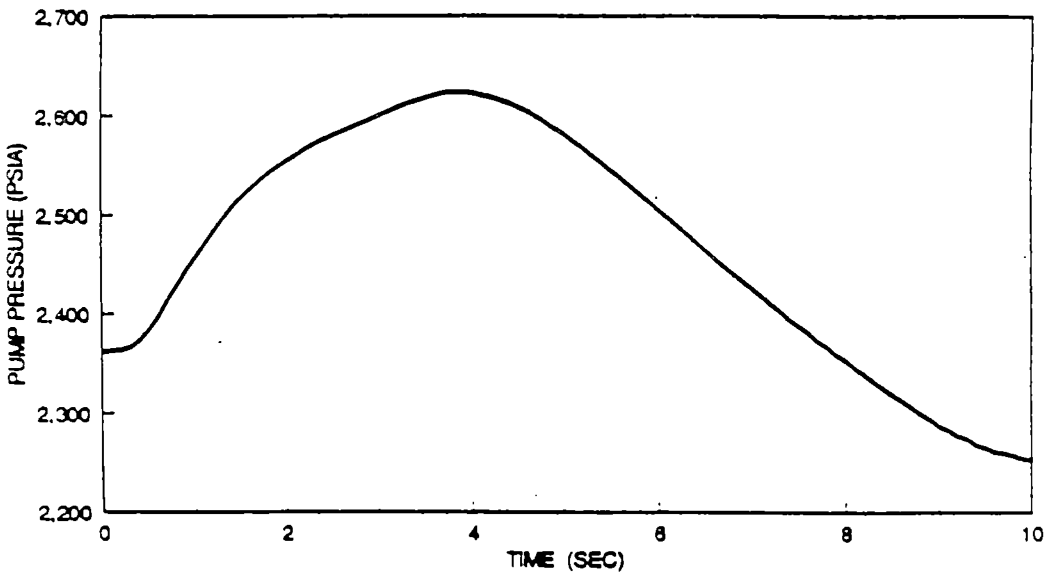


Figure 4.1.5.3-3

All Loops Operating
One Locked Rotor (LR)



4.1.7 Loss of External Electrical Load and/or Turbine Trip (UFSAR 15.2.7)

Accident Description:

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 Condition II 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.

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 valves operating, the main steam safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, the OTDT signal, or the low-low steam generator water level signal. The steam generator shell-side pressure and the reactor coolant pressure will increase rapidly. However, the RCS and MSS relieving capacities were designed to ensure safety of the unit without requiring the automatic rod control, pressurizer pressure control, steam bypass control systems, or reactor trip on turbine trip.

Method of Analysis:

The loss of load accident is analyzed to show the following: (1) the peak primary and secondary side pressures remain below 110% of their respective design pressures and (2) the DNBR remains above the safety analysis DNBR limit.

The total loss of load transients are analyzed with the LOFTRAN computer program. 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 major assumptions are summarized below:

a. Two cases for both BOL and EOL reactivity feedback conditions are analyzed:

   1. For cases with automatic pressurizer pressure control, full credit is taken for the effect of pressurizer spray and PORVs in reducing or limiting the coolant pressure.

   2. For cases without automatic pressurizer pressure control, no credit is taken for the effect of pressurizer spray and PORVs in reducing or limiting the coolant pressure.

b. For the cases analyzed to demonstrate that the core protection margins are maintained (BOL and EOL with automatic pressurizer pressure control), the Loss of Load accident is analyzed using the RTDP. For these cases, initial core power, reactor coolant temperature, and reactor coolant pressure are assumed to be at their nominal values consistent with steady-state full power operation.

   For the cases analyzed to demonstrate the adequacy of the pressure relieving devices (BOL and EOL without automatic pressurizer pressure control), the Loss of Load accident is analyzed using the STDP. For these cases, initial core power and reactor coolant temperature are assumed at their maximum values consistent with steady-state full power operation including allowances for calibration and instrument errors.

c. The loss of load event is analyzed with both maximum and minimum reactivity feedback. The maximum feedback (EOL) cases assume a large negative MTC and the most negative DPC. The minimum feedback (BOL) cases assume a zero MTC and a least negative DPC.

d. From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual rod control.

e. No credit is taken for the operation of the steam dump system or steam generator PORVs.

f. Main feedwater flow to the steam generators is conservatively assumed to be lost at the time of turbine trip.

Reactor trip is actuated by the first RPS trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip.

Results:

The transient responses for a total loss of load from full power operation are shown on Figures 4.1.7-1 through 4.1.7-12 for four cases; BOL reactivity feedback conditions with and without pressurizer spray and pressurizer PORVs, and EOL reactivity feedback conditions with and without pressurizer spray and pressurizer PORVs. The cases without pressurizer spray and pressurizer PORVs are analyzed to demonstrate the adequacy of the pressure relieving devices; the cases with pressurizer spray and pressurizer PORVs are analyzed to verify core protection margin.

Figures 4.1.7-1 through 4.1.7-3 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 PORVs. Following event initiation, the DNBR initially increases slightly, then decreases slightly, and finally, following reactor trip, increases rapidly. The minimum DNBR remains well above the safety analysis limit value.

Figures 4.1.7-4 through 4.1.7-6 show the transient responses for the total loss of steam load at EOL conditions (maximum feedback reactivity coefficients) assuming full credit for the pressurizer spray and pressurizer PORVs. The DNBR increases throughout the transient and never drops below its initial value.

Figures 4.1.7-7 through 4.1.7-9 show the BOL transients without pressure control. The neutron flux 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 neutron flux remains essentially constant at full power until the reactor is tripped. In this case the RCS and main steam system pressures remain below 110% of their design values. The pressurizer and main steam safety valves are actuated to limit their respective system pressures.

Figures 4.1.7-10 through 4.1.7-12 show the transients at the EOL with the other assumptions being the same as those assumed for the transients detailed on Figures 4.1.7-7 through 4.1.7-9. Again, a reactor trip signal is generated by the high pressurizer pressure trip function and the pressures remain below their respective safety analysis limits. The pressurizer and main steam safety valves are actuated to limit the RCS and main steam system pressures.

Table 4.1.7-1 summarizes the sequence of events and limiting conditions for the BOL and EOL cases without pressurizer sprays and pressurizer PORVs.

Conclusions:

The results of the analyses show that the plant design is such that a total loss of external electrical load/turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the MSS. Pressure-relieving devices incorporated in the plant design are adequate to limit the maximum pressures to within the safety analysis limits. The integrity of the core is maintained by operation of the RPS; i.e., the DNBR is maintained above the safety analysis limit value.

Table 4.1.7-1

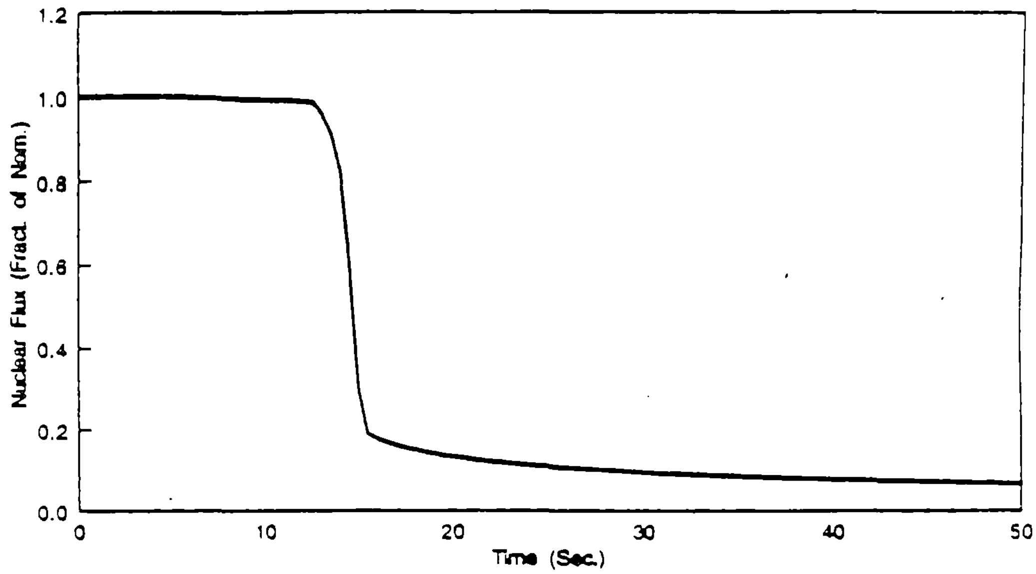
Sequence of Events and Transient Results
Loss of External Electrical Load

Without Pressurizer Pressure Control

Event	Time (seconds) BOL	EOL
Loss of load turbine trip	0.0	0.0
Reactor trip on high pressurizer pressure	6.2	6.3
Rod motion begins	8.2	8.3
Peak pressurizer pressure occurs	10.0	10.0
Initiation of steam release from main steam safety valves	10.5	10.5

Figure 4.1.7-1

Loss of Load With Automatic Pressure Control
Minimum Feedback (BOL)



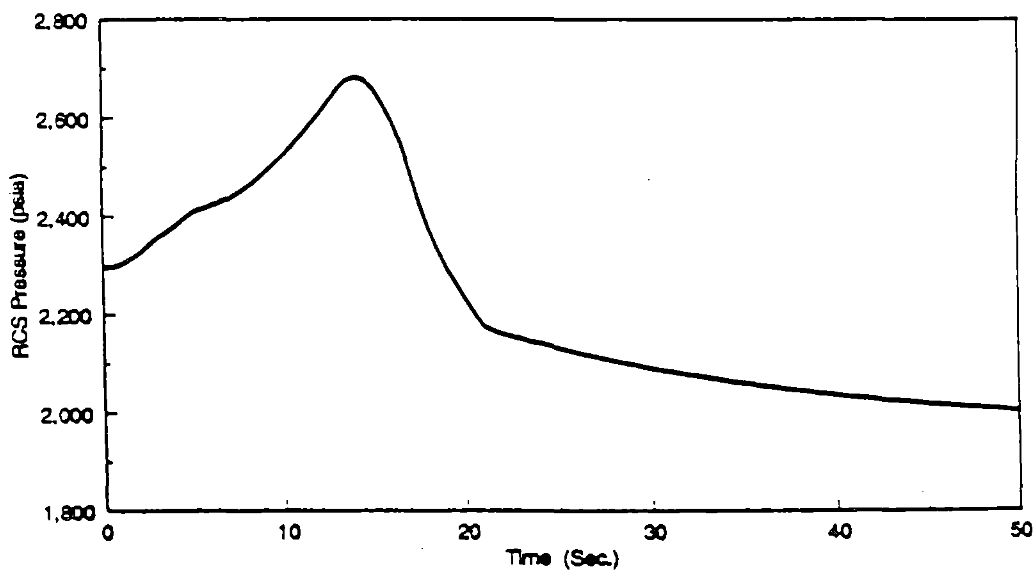
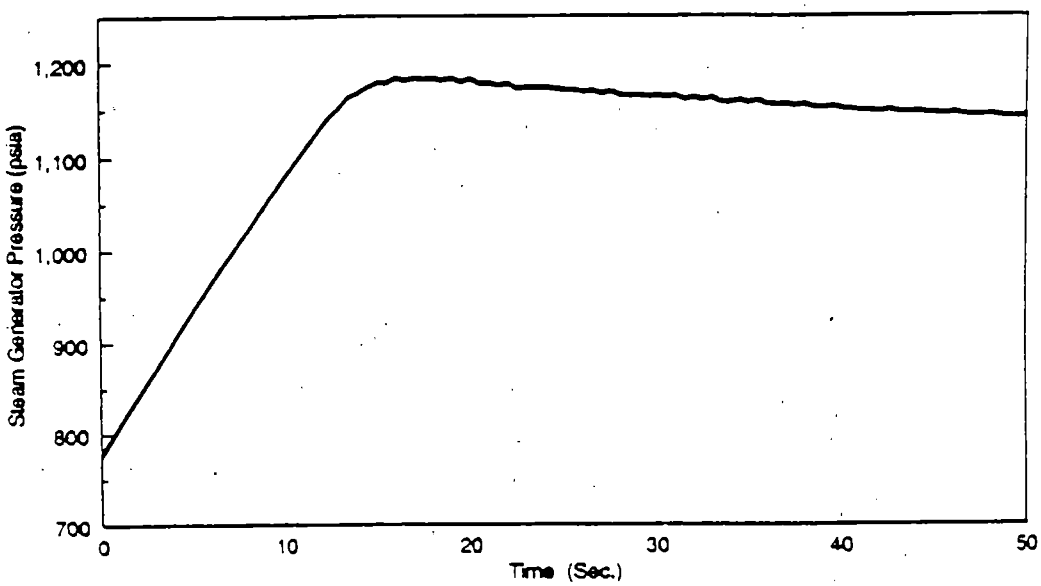


Figure 4.1.7-2

Loss of Load With Automatic Pressure Control
Minimum Feedback (BOL)



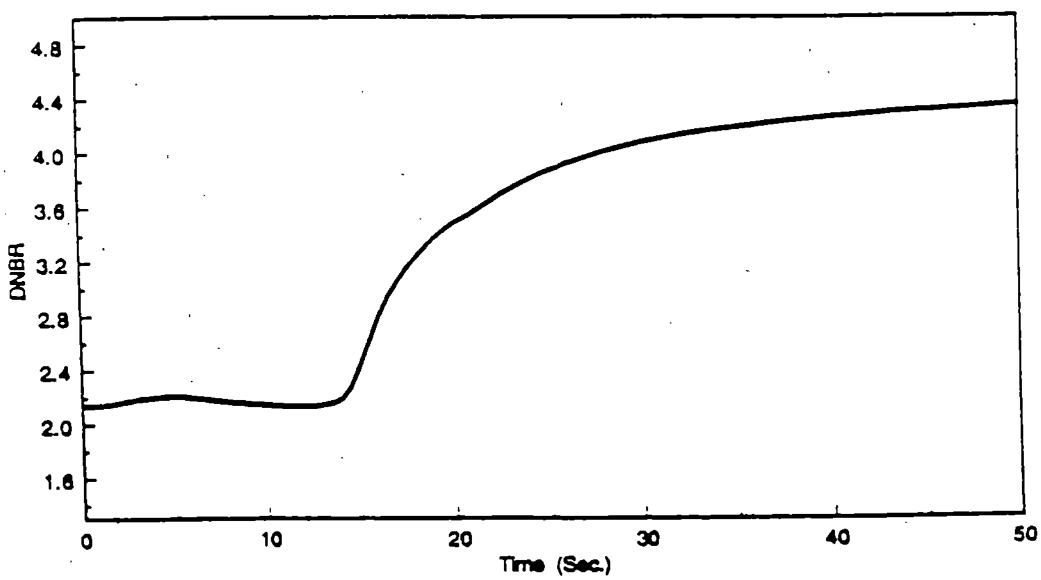
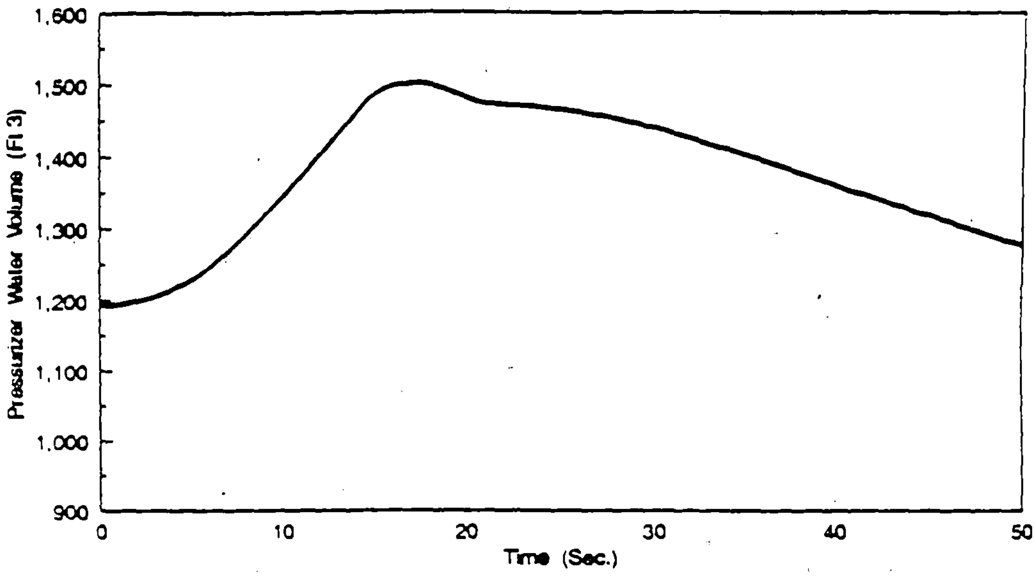


Figure 4.1.7-3

Loss of Load With Automatic Pressure Control
Minimum Feedback (BOL)



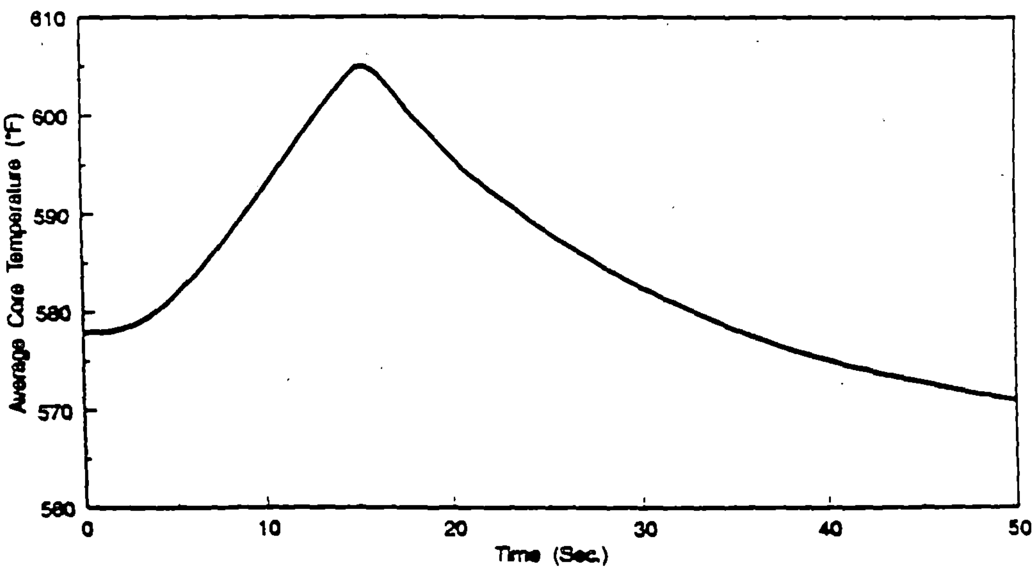
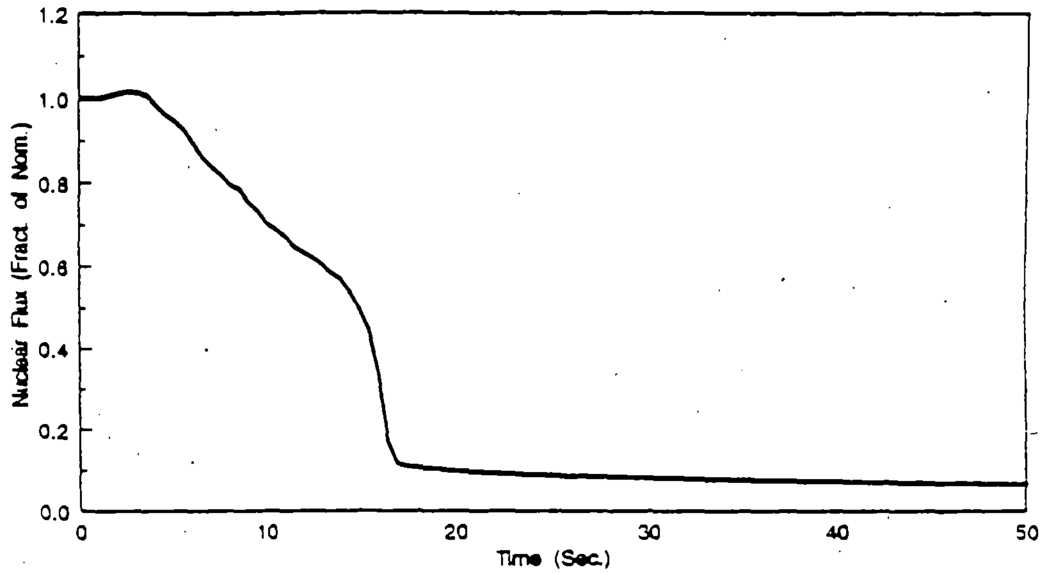


Figure 4.1.7-4

Loss of Load With Automatic Pressure Control
Maximum Feedback (EOL)



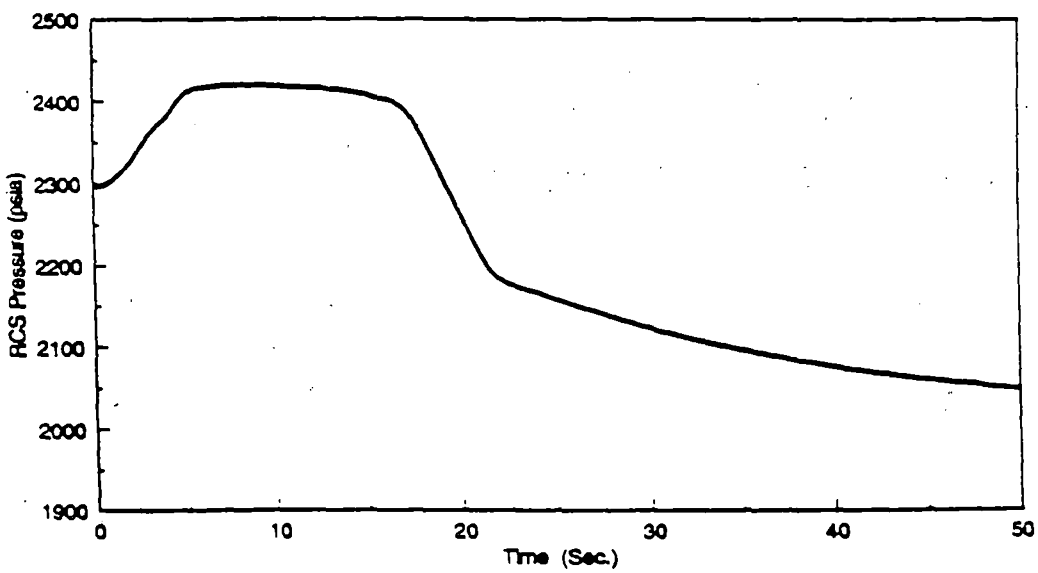
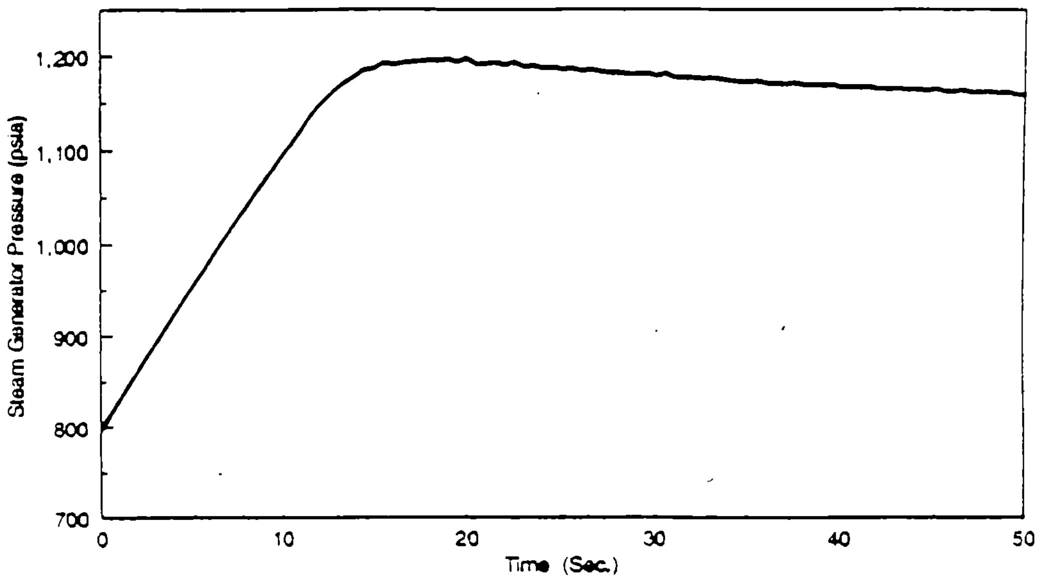


Figure 4.1.7-5

Loss of Load With Automatic Pressure Control
Maximum Feedback (EOL)



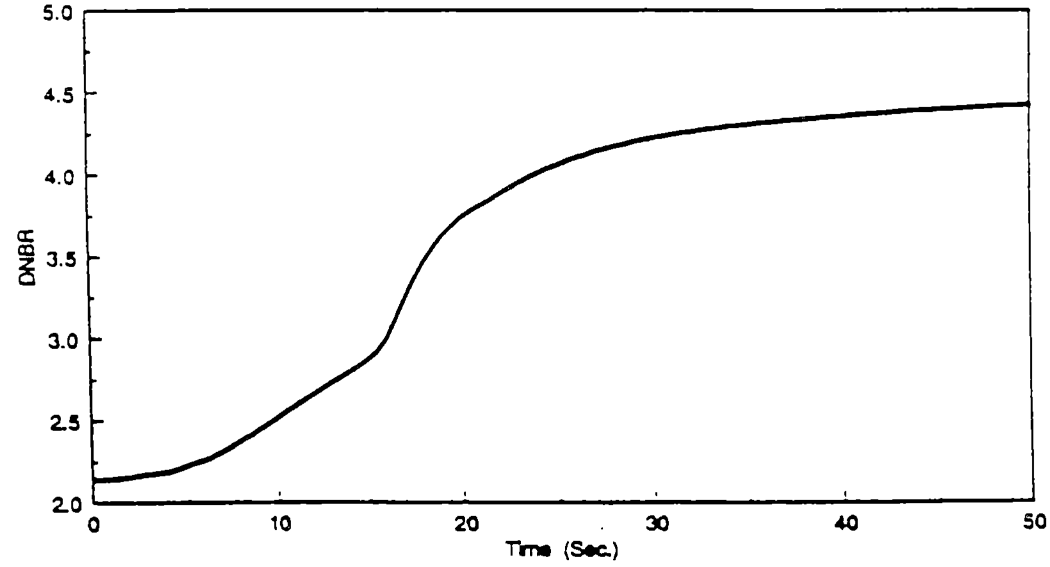


Figure 4.1.7-6

Loss of Load With Automatic Pressure Control
Maximum Feedback (EOL)

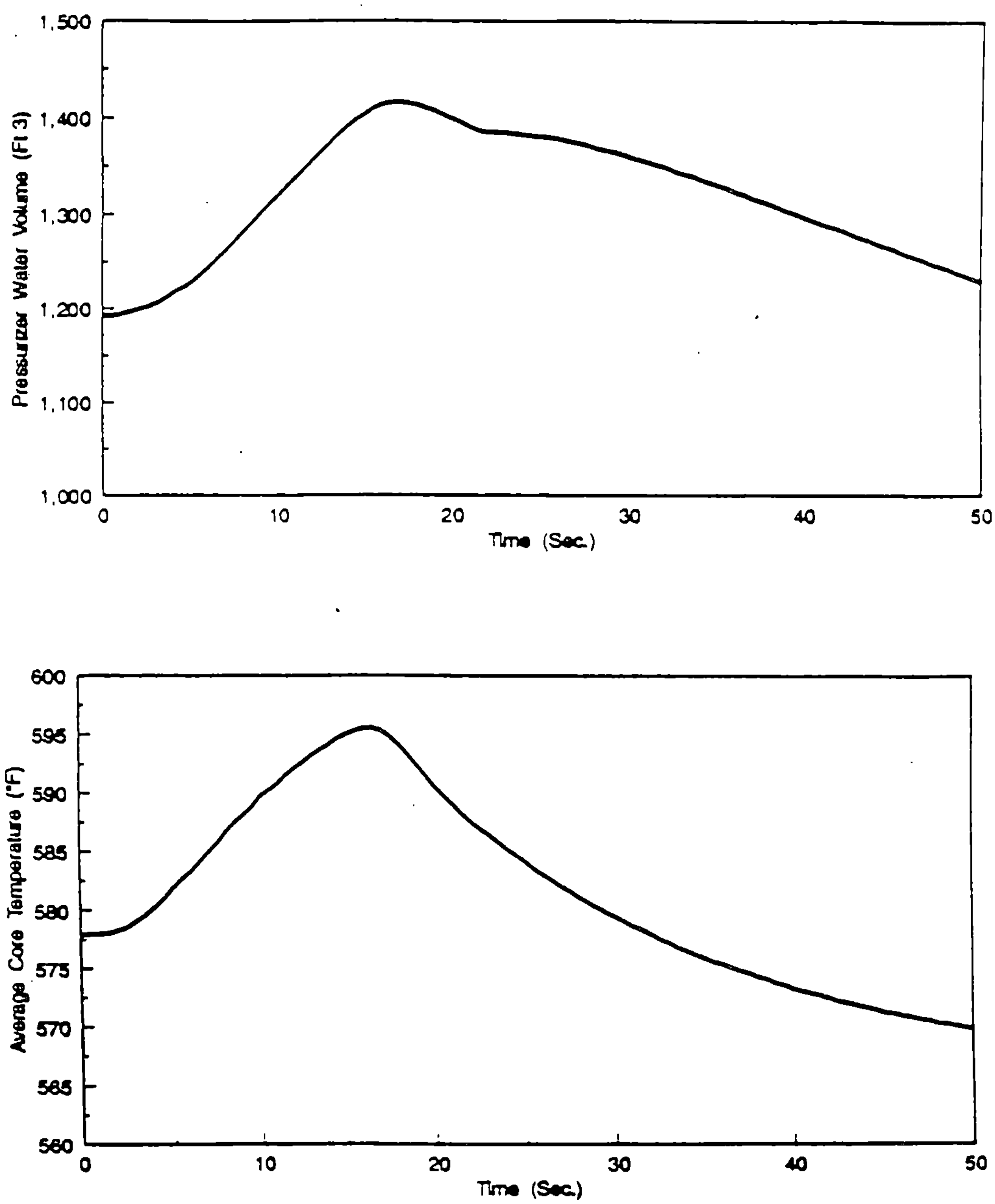


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Figure 4.1.7-7

Loss of Load Without Automatic Pressure Control
Minimum Feedback (BOL)

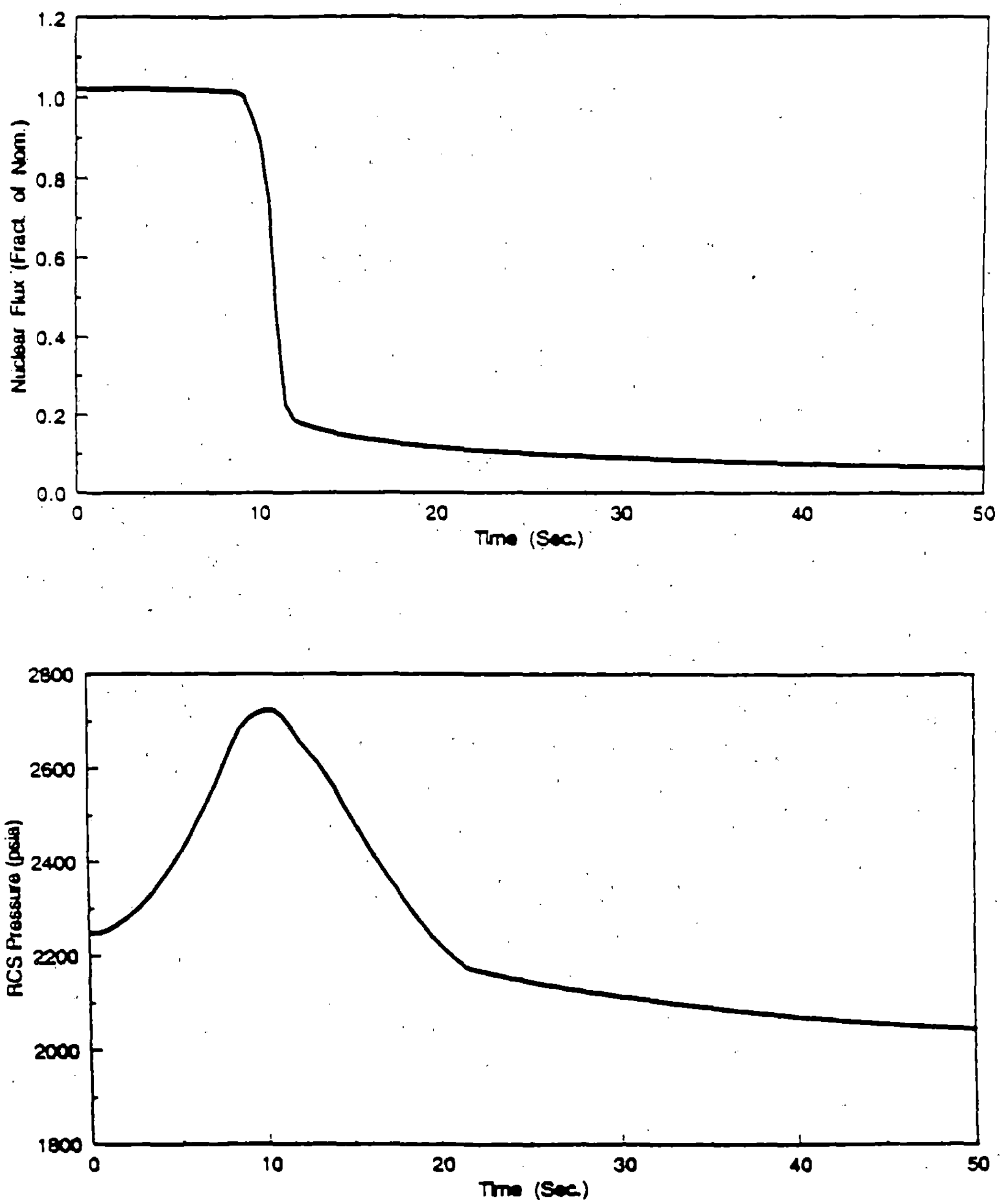
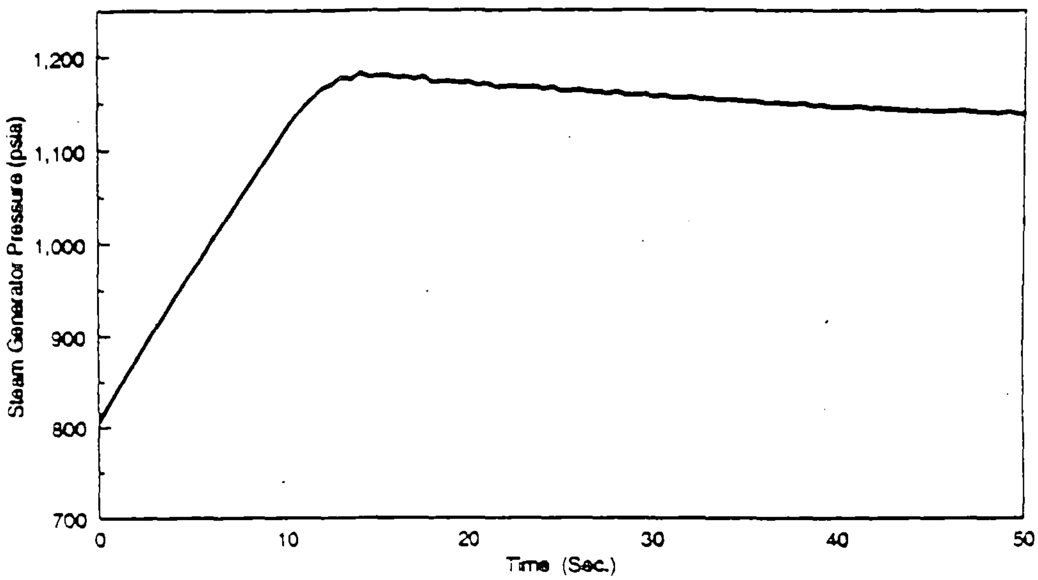


Figure 4.1.7-8

Loss of Load Without Automatic Pressure Control
Minimum Feedback (BOL)



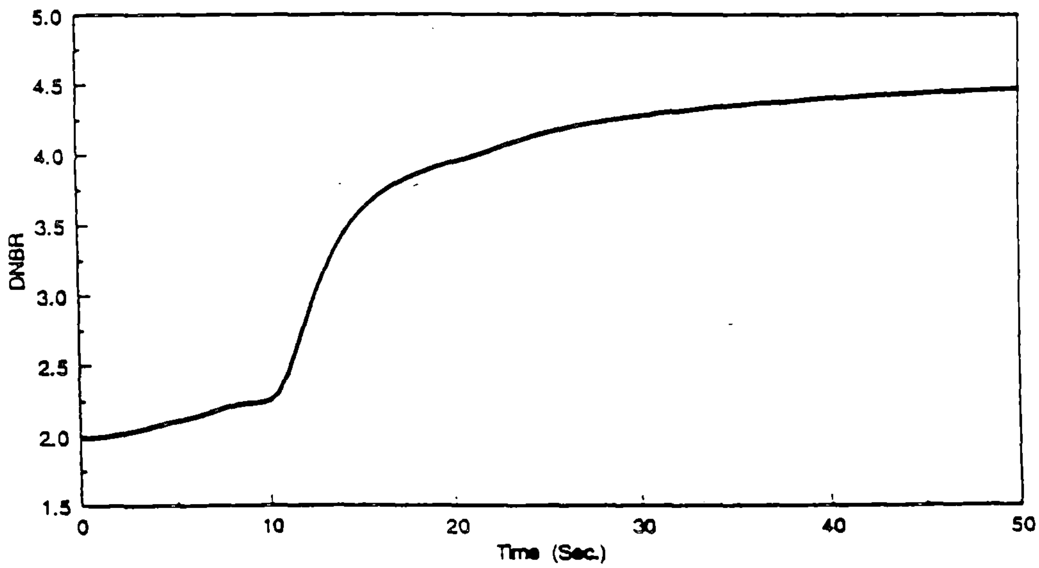
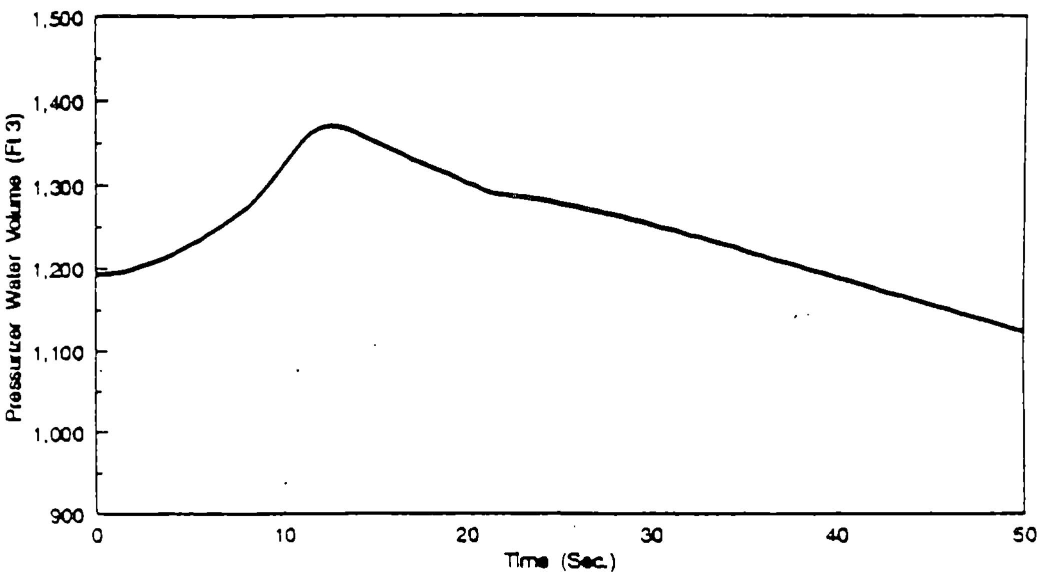


Figure 4.1.7-9

Loss of Load Without Automatic Pressure Control
Minimum Feedback (BOL)



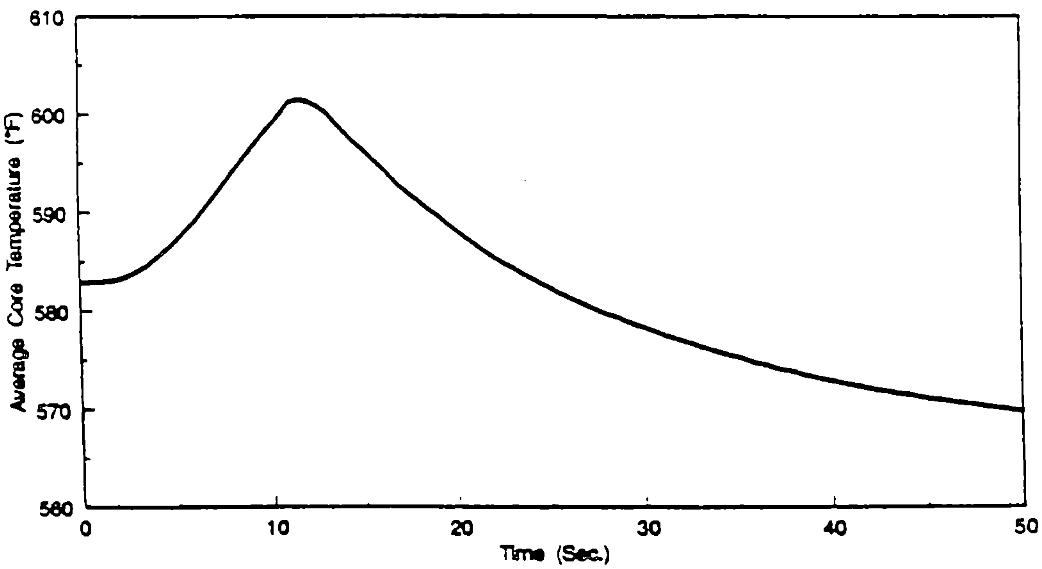


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Figure 4.1.7-10

Loss of Load Without Automatic Pressure Control
Maximum Feedback (EOL)

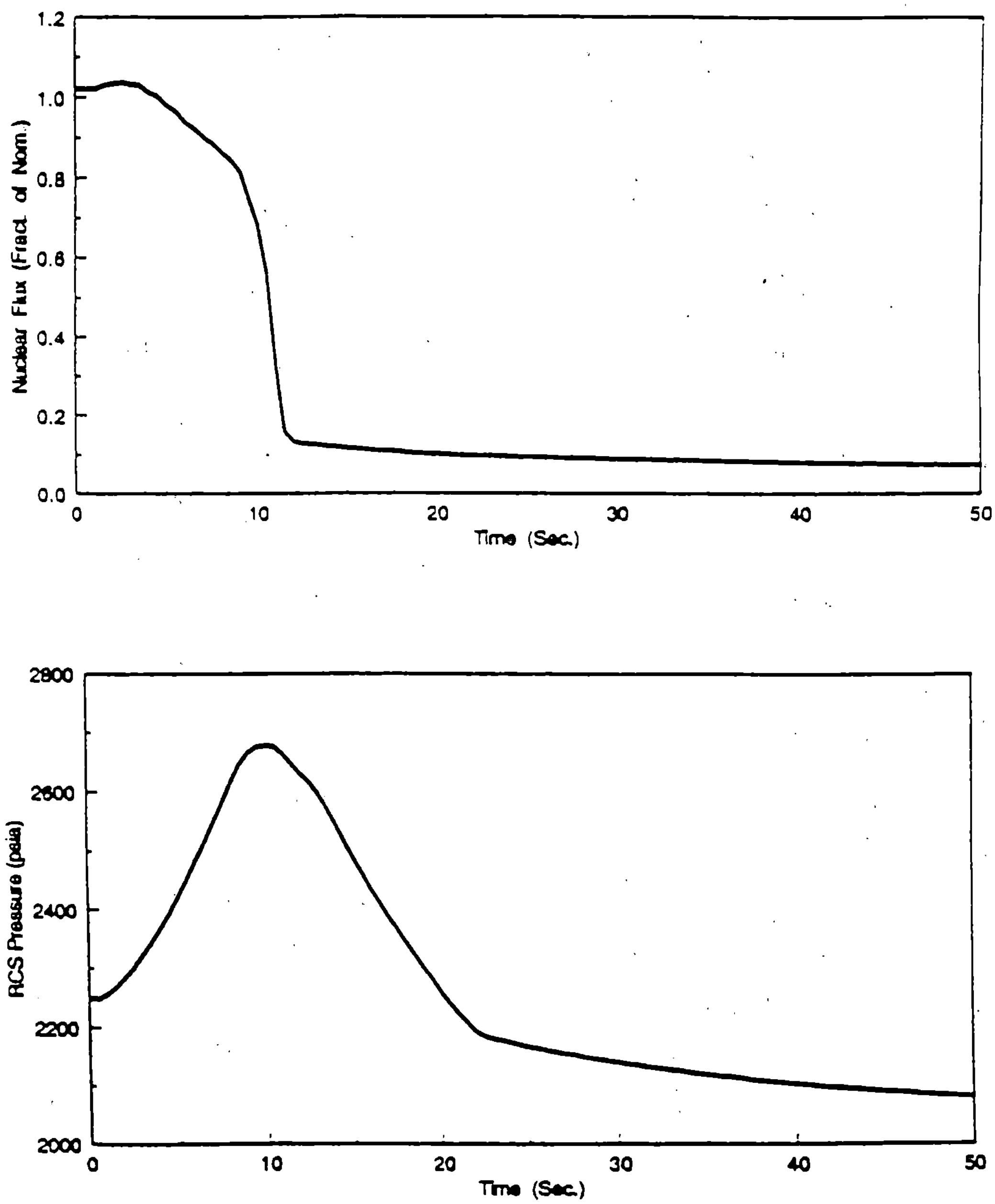
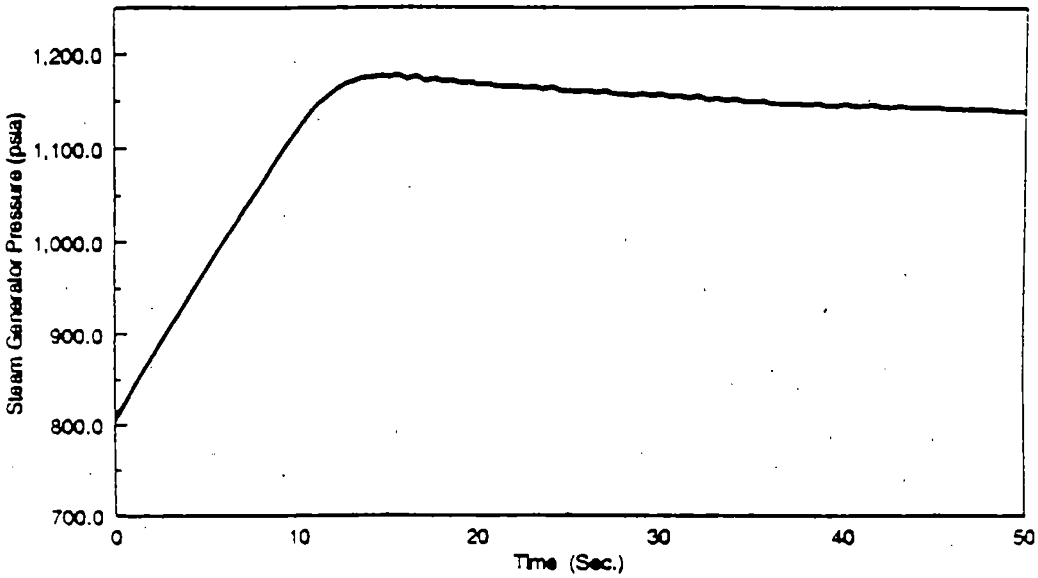


Figure 4.1.7-11

Loss of Load Without Automatic Pressure Control
Maximum Feedback (EOL)



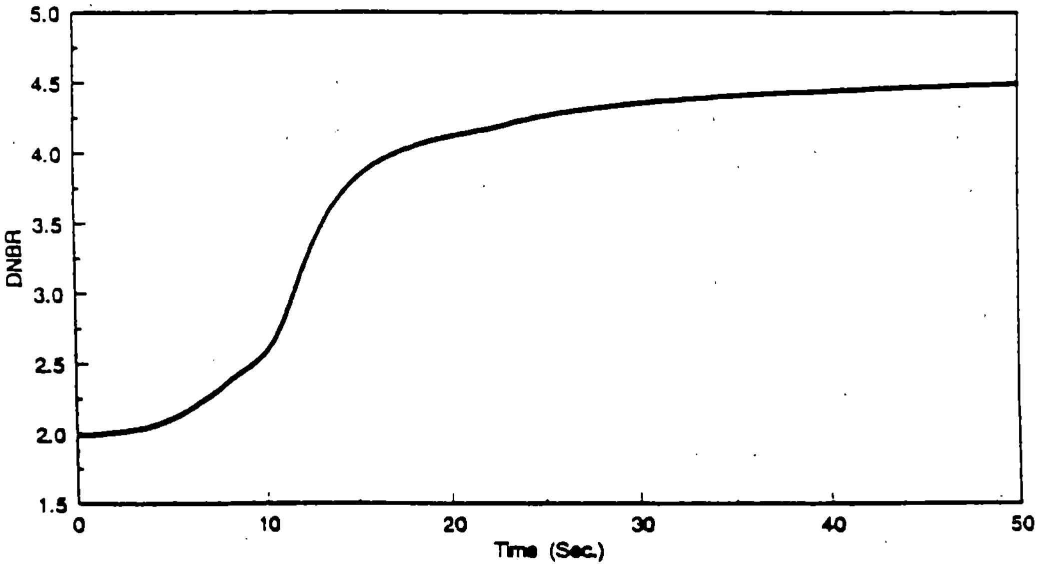
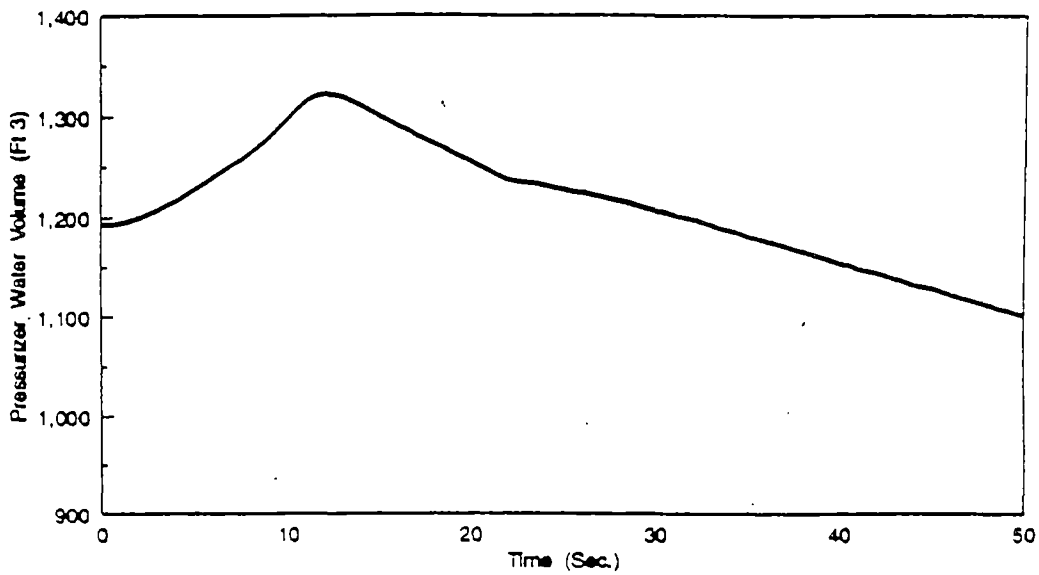
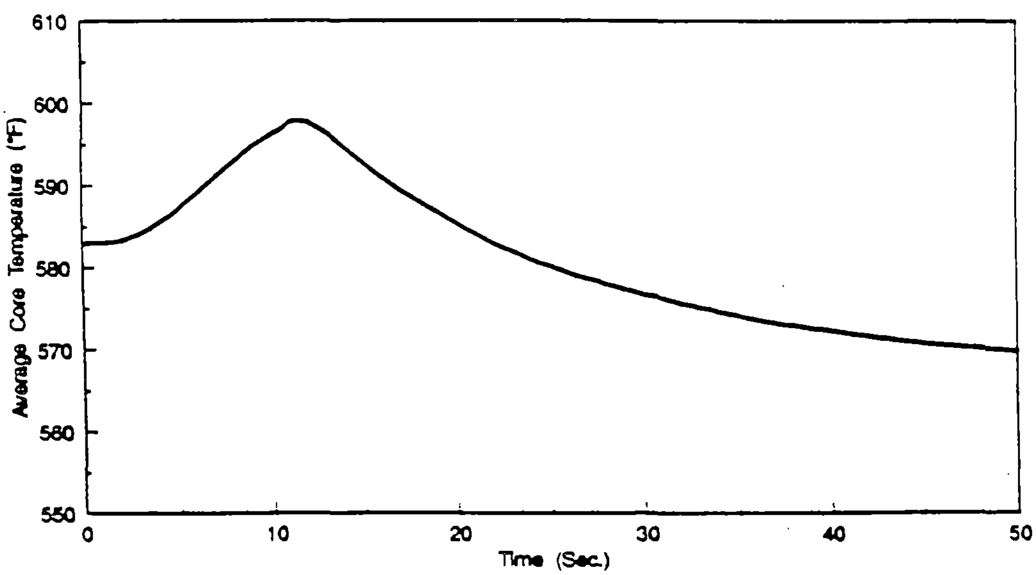


Figure 4.1.7-12

Loss of Load Without Automatic Pressure Control
Maximum Feedback (EOL)





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4.1.8 Loss of Normal Feedwater (UFSAR 15.2.8)

Accident Description:

A loss of normal feedwater (from pump failures, valve malfunctions or loss of offsite ac power) is a Condition II event which results in a reduction of the secondary system's ability to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss of heat sink. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a departure from nucleate boiling condition.

The analysis shows that following a loss of normal feedwater, the AFW system is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or pressurizer filling.

Method of Analysis:

A detailed analysis using the LOFTRAN code is performed to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators, and feedwater system. The digital program computes pertinent variables, including the steam generator mass, pressurizer water volume and reactor coolant average temperature.

The major assumptions are summarized below.

a. Reactor trip occurs on steam generator low-low water level.

b. The plant is initially operating at 102% of the NSSS power rating.

c. A conservative core residual heat generation is assumed, based on ANS 5.1-1979 decay heat (plus 2 Sigma).

d. The most severe single failure in the AFW system is assumed to occur (failure of the turbine driven AFW pump). For additional conservatism, only one motor-driven AFW pump is assumed available to deliver AFW flow (one minute after initiation of low-low SG level trip).

e. AFW is delivered to two of four steam generators.

f. Secondary system steam relief is achieved through the main steam safety valves. The steam generator PORVs and turbine bypass valves are assumed unavailable.

g. The initial reactor coolant average temperature is 5°F higher than the nominal value since this results in a greater expansion of RCS water during the transient and in a higher pressurizer water level.

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

i. Normal reactor control systems are not required to function. However, the pressurizer PORVs and pressurizer spray system are assumed to operate normally. This results in a conservative transient with respect to peak pressurizer water level. If these control systems did not operate the pressurizer safety valves would maintain peak RCS pressure near or below their actuation setpoint throughout the transient.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the RPS and engineered safeguards features (ESF) (e.g., the AFW system) to remove long-term decay heat and prevent pressurizer filling. As such, the assumptions used in the analysis are designed to minimize the energy removal capability of the system and maximize the possibility of filling the pressurizer by maximizing the coolant system expansion.

Results:

Figures 4.1.8-1 and 4.1.8-2 show the significant plant parameter transients following a loss of normal feedwater. The calculated sequence of events is listed in Table 4.1.8-1.

Following the reactor and turbine trip from full load, the water level in the steam generators falls because of the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. In one minute following the initiation of the low-low level trip, the AFW pumps are automatically started, reducing the rate of water level decrease.

The capacity of the AFW system is such that the water level in the steam generators being fed does not fall below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat. Figure 4.1.8-2 shows that at no time is the pressurizer water solid. Plant procedures may be followed to further cool down the plant. The maximum RCS and steam generator pressures are below the limit values.

Conclusions:

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the MSS given the MRP implementation. The AFW capacity is sufficient to dissipate core residual heat and prevent the pressurizer from filling.

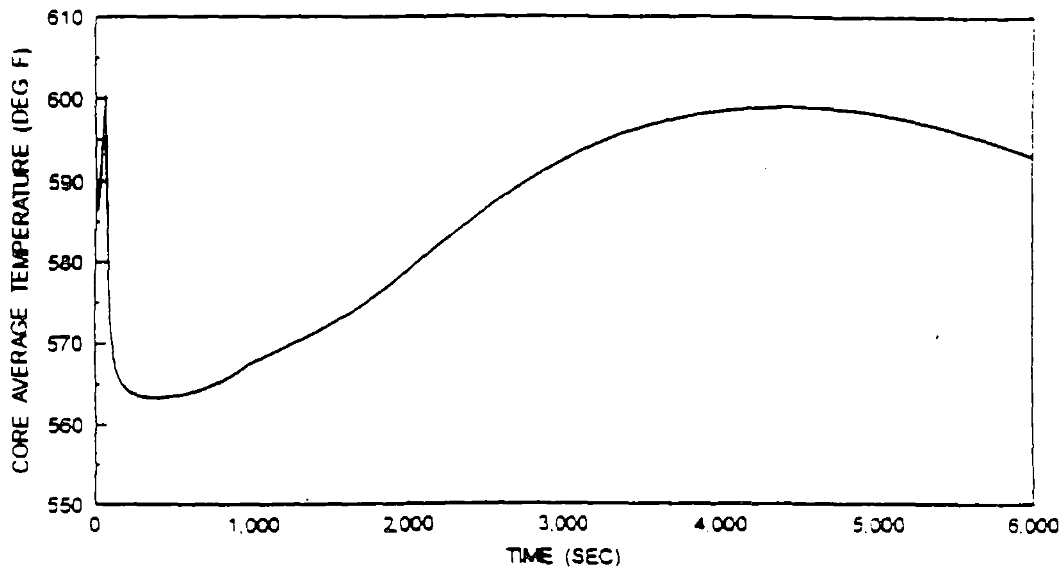
Table 4.1.8-1

Sequence of Events
Loss of Normal Feedwater

Event	Time (seconds)
Main feedwater flow stops	10
Pressurizer relief valves open	47
Reactor trip on low-low steam generator water level	61
Rod motion begins	63
Pressurizer relief valve sclose and peak pressurizer water level occurs (first peak limiting)	67
Main steam safety valves open	68
Two steam generators receive AFW flow from one motor-driven AFW pump	121

Figure 4.1.8-1

Loss of Normal Feedwater



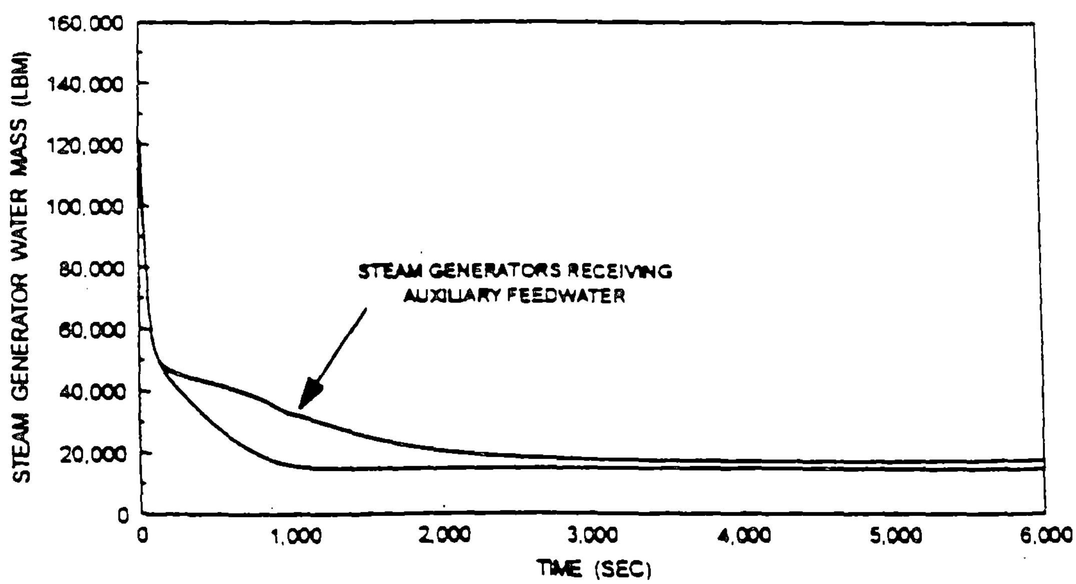
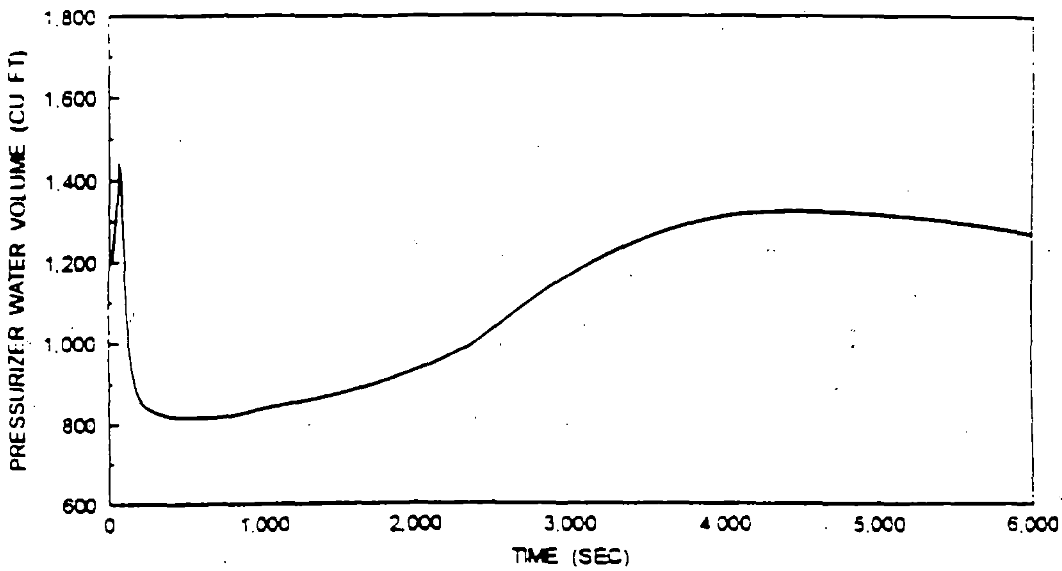
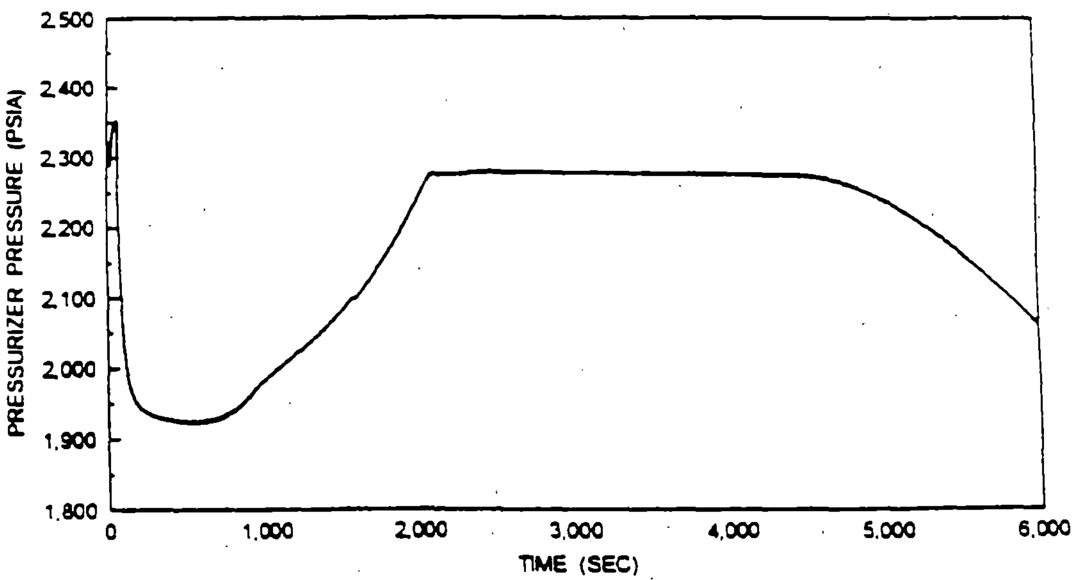


Figure 4.1.8-2

Loss of Normal Feedwater





4.1.9 Loss of Offsite Power to the Station Auxiliaries (UFSAR 15.2.9)

Accident Description:

A complete loss of non-emergency ac power is a Condition II event which may result in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite ac distribution system.

Method of Analysis:

A detailed analysis using the LOFTRAN code is performed in order to obtain the plant transient following a station blackout event. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables, including the steam generator mass, pressurizer water volume, and reactor coolant average temperature.

Major assumptions made in the station blackout analysis are:

a. The plant is initially operating at 102% of the NSSS power rating.

b. A conservative core residual heat generation based on ANS 5.1-1979 decay heat (plus 2 Sigma).

c. A heat transfer coefficient in the steam generator associated with RCS natural circulation, following the reactor coolant pump coastdown.

d. Reactor trip occurs on steam generator low-low level.

e. The most severe single failure in the AFW system is assumed to occur (failure of the turbine driven AFW pump). For additional conservatism, only one motor-driven AFW pump is assumed available to deliver AFW flow (one minute after initiation of low-low SG level trip).

f. AFW flow is delivered to two of four steam generators.

g. Secondary system steam relief is achieved through the main steam safety valves.

The assumptions used in the analysis are similar to the loss of normal feedwater (Section 4.1.8) except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip.

Results:

The transient response of the RCS following a loss of ac power is shown in Figures 4.1.9-1 and 4.1.9-2. The calculated sequence of events for this event is listed in Table 4.1.9-1. The first few seconds after the loss of power to the reactor coolant pumps will closely resemble the simulation of the complete loss of flow accident (UFSAR Section 15.3.4), where core damage due to rapidly increasing core temperature is prevented by promptly tripping the reactor.

After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core. The maximum RCS and the steam generator pressures are below the limit values.

The LOFTRAN code results show that the reactor coolant natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown. The natural circulation flow as a function of reactor power is provided in Table 4.1.9-2.

Conclusions:

Given the MRP implementation, a loss of offsite power to the station auxiliaries does not cause any adverse condition in the core since it does not result in water relief from the pressurizer relief or safety valves.

Table 4.1.9-1

Sequence of Events
Loss of Offsite Power

Event	Time (seconds)
Main feedwater flow stops	10
Reactor trip on low-low steam generator water level	60
Rod motion begins	62
Pressurizer relief valves open	63
Pressurizer relief valves close	65
Main steam safety valves open	68
Two steam generators receive AFW flow from one motor-driven AFW pump	120
Peak pressurizer water level (second peak limiting)	2248

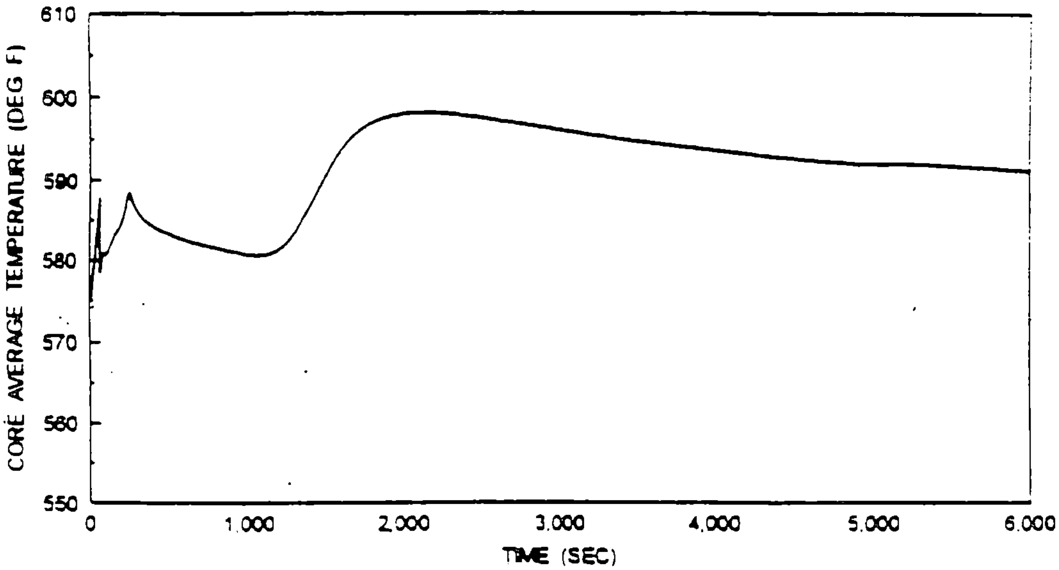
Table 4.1.9-2

Reactor Coolant Natural Circulation Flow

Percent of Nominal Power	Percent of Nominal Flow
1.0	3.07
1.5	3.52
2.0	3.89
2.5	4.22
3.0	4.49
3.5	4.74
4.0	4.96

Figure 4.1.9-1

Loss of Offsite Power to the
Station Auxiliaries (Station Blackout)



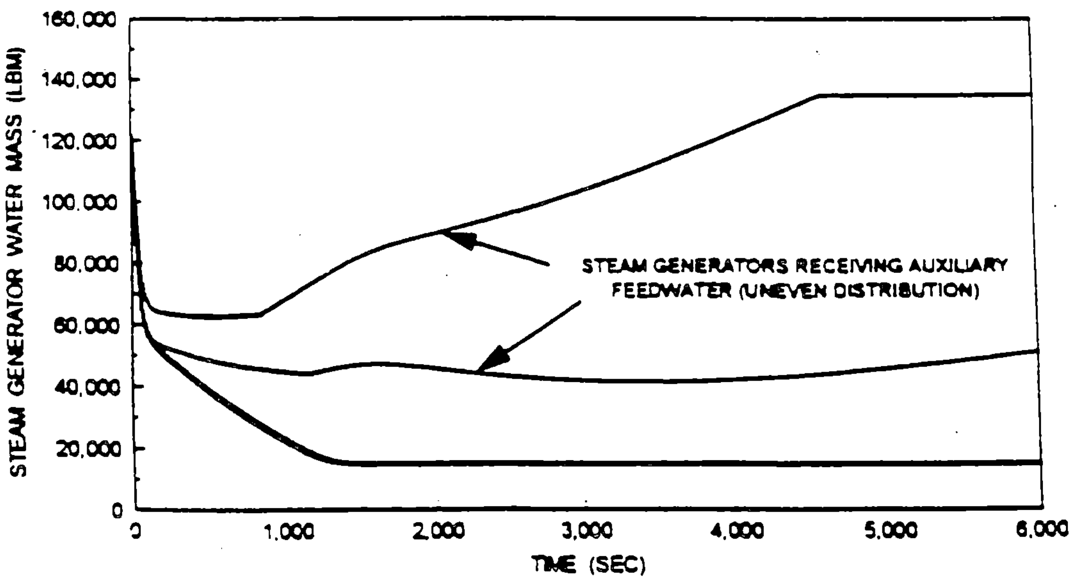
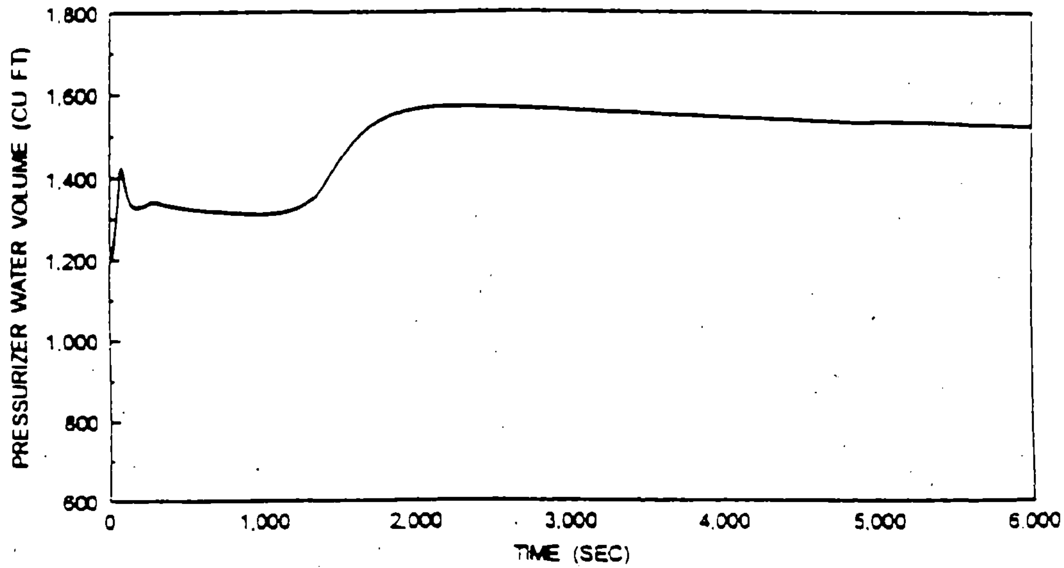
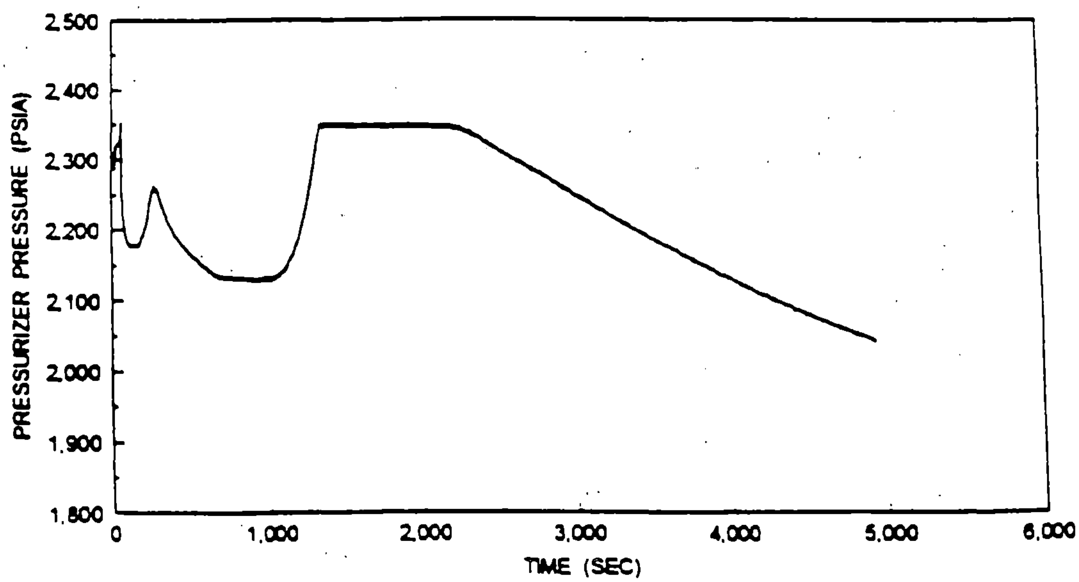


Figure 4.1.9-2

Loss of Offsite Power to the
Station Auxiliaries (Station Blackout)





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4.1.10 Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR 15.2.10)

Accident Description:

Reductions in feedwater temperature or excessive feedwater flow additions are means of increasing core power above full power. The overpower/overtemperature protection (high neutron flux, OTDT, and OPDT trips) prevent any power increase that could lead to a DNBR that is less than the safety analysis limit value.

An example of excessive feedwater flow would be a full opening of one or more feedwater control valves (FCVs) due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generators. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative MTC. Continuous excessive feedwater flow addition is prevented by the steam generator high-high level trip, which closes all feedwater control and isolation valves, trips the main feedwater pump, and trips the turbine.

A second example of excess heat removal is the transient associated with the accidental opening of the low pressure feedwater heater bypass valve which diverts flow around the low pressure feedwater heaters. At power, this increased subcooling will create a greater load demand on the RCS.

Both of these feedwater malfunction events are classified as Condition II events.

Method of Analysis:

The excessive heat removal due to a feedwater system malfunction transient is analyzed with the LOFTRAN computer code. The LOFTRAN code simulates a multi-loop system, neutron kinetics, the pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and main steam safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. For the zero power cases only, the THINC code is used to calculate DNBR during the transient.

The system is analyzed to demonstrate acceptable results in the event of a feedwater system malfunction. Feedwater temperature reduction due to low-pressure heater bypass valve actuation in conjunction with an inadvertent trip of the heater drain pump is considered. Excessive feedwater flow addition due to a control system malfunction or operator error that allows one or more feedwater control valves (FCVs) and feedwater control bypass valves (FCBVs) to open fully is considered.

Eight excessive feedwater flow cases are analyzed, four single loop cases and four multiple loop cases. All eight cases are analyzed at EOL (maximum reactivity feedback) conditions. The following cases are each analyzed as a single loop and multiple loop case:

1. Zero Power, Manual Rod Control Case

2. Zero Power, Automatic Rod Control Case

3. Full Power, Manual Rod Control Case

4. Full Power, Automatic Rod Control Case

The feedwater system malfunction cases are performed using the following assumptions:

a. The analyses conservatively assume the steam generator PORVs fail full open simultaneous with the FCVs and FCBVs. This is bounding for the Advanced Digital Feedwater Control System (ADFCS) as well as the current system. The ADFCS design has the SG PORVs, FCVs, and FCBVs on the same digital processing unit (DPU). With two control systems on the same DPU, it is conservatively postulated that one or more SG PORVs could be open at the same time one or more FCVs are open.

b. For the zero load condition, feedwater temperature is assumed to be 32°F.

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

d. No credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

e. The feedwater flow resulting from a fully open FCV and FCBV is terminated by the steam generator high-high water level signal that closes all main feedwater control and bypass valves and trips the main feedwater pumps and turbine.

Results:

Opening of a low pressure feedwater heater bypass valve and trip of the heater drain pumps causes a reduction in the feedwater temperature which increases the thermal load on the primary system. This effect is less limiting than the 10% excessive load increase evaluated in Section 4.1.11. Thus, the results of this event are bounded by the Excessive Load Increase event and, therefore, not presented here.

In the case of the accidental full opening of one or more feedwater control and bypass control valves with the reactor at zero power, the maximum reactivity insertion rate is conservatively calculated. A DNB analysis was performed to demonstrate that the DNB design basis is met. The results of the DNB analysis show that the DNBR remains above the safety analysis limit value. It should be noted that if the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting).

For the full power excessive feedwater flow cases, the single loop manual rod control case with one FCV and one FBCV failure and the multi-loop automatic rod control case with four FCV and four FBCV failures, result in the closest approach to the safety analysis limit DNBR.

For all cases of excessive feedwater flow, a high high steam generator water level signal closes the feedwater control valves, closes the feedwater bypass valves, trips the feedwater pumps, and causes a turbine trip.

Transient results for both the full power single loop manual rod control case and the full power multi-loop automatic rod control case are shown in Figures 4.1.10-1 through 4.1.10-6. These figures show the core heat flux, pressurizer pressure, core average temperature, and DNBRs, as well as the increase in nuclear power and loop DT associated with the increased thermal load on the reactor.

The sequence of events for the single loop and multi-loop cases are shown in Table 4.1.10-1.

Conclusions:

The decrease in the feedwater temperature transient due to an opening of the low-pressure feedwater heater bypass valve is less severe than the Excessive Load Increase event (Section 4.1.11). Based on the results presented in that section, the applicable acceptance criteria for the decrease in feedwater temperature event have been met.

For the excessive feedwater flow at full power transient, the results show that the DNBRs encountered are above the safety analysis limit value. Additionally, an analysis at hot zero power demonstrates that the minimum DNBR remains above the safety analysis limit for a maximum reactivity insertion rate conservatively bounding an excessive feedwater addition at no-load conditions.

Table 4.1.10-1

Sequence of Events
Feedwater System Malfunction

Excessive feedwater at full power (single loop)

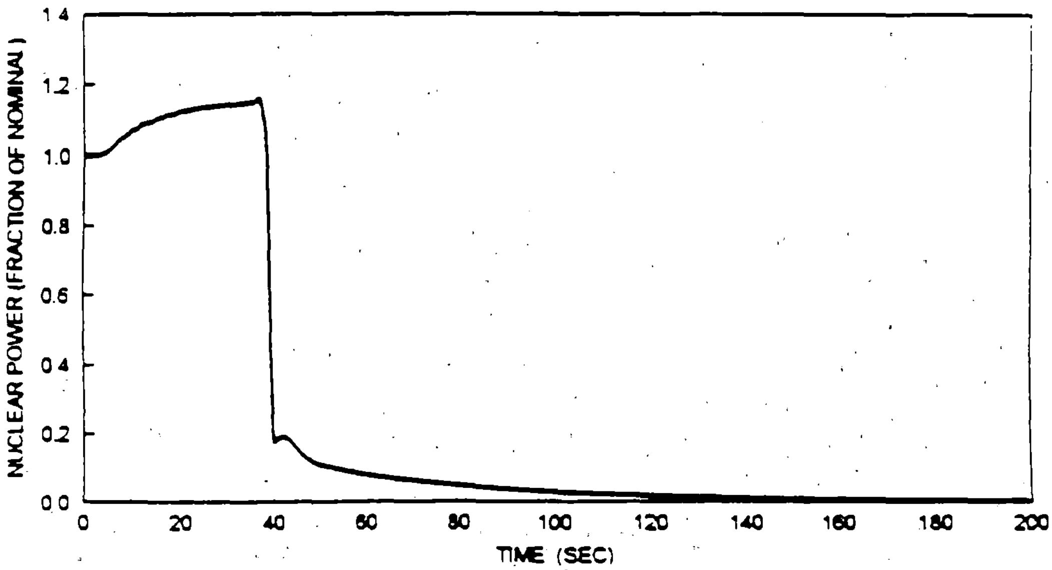
Event	Time (seconds)
One FCV and one FBCV fail fully open	0.0
High-high steam generator water level signal reached	32.0
Turbine trip occurs	34.5
Minimum DNBR occurs	35.0
Rod motion begins	36.5
Feedwater flow isolated due to high-high steam generator water level	64.0

Excessive feedwater at full power (multi-loop)

Event	Time (seconds)
Four FCV and four FBCV fail fully open	0.0
Minimum DNBR occurs	44.0
High-high steam generator water levels signal reached	119.7
Turbine trip occurs	122.2
Rod motion begins	124.2
Feedwater flow isolated due to high-high steam generator water level	151.7

Figure 4.1.10-1

Feedwater Malfunction Single Loop
Manual Rod Control



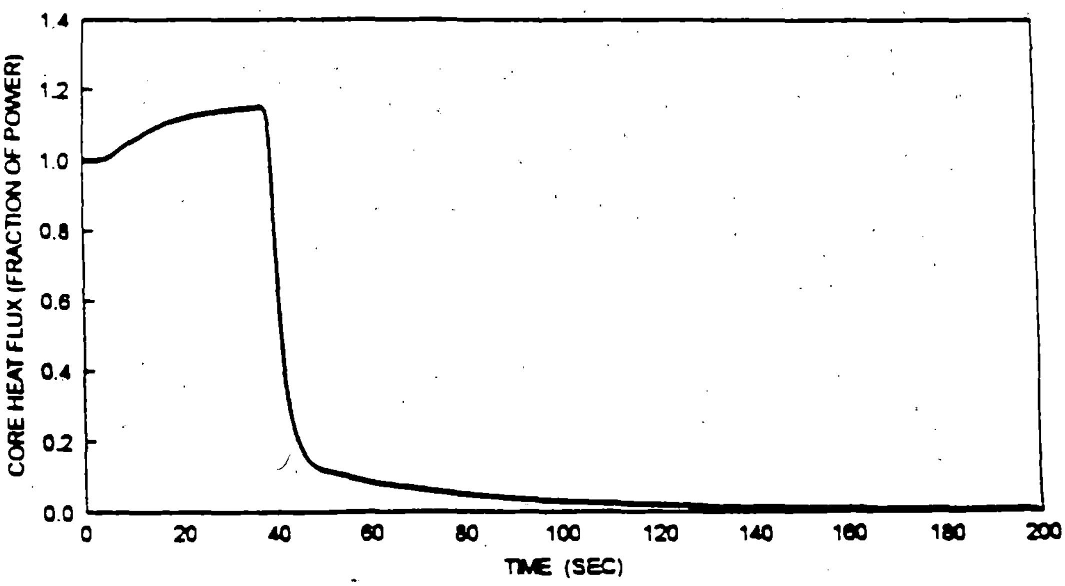


Figure 4.1.10-2

Feedwater Malfunction Single Loop
Manual Rod Control

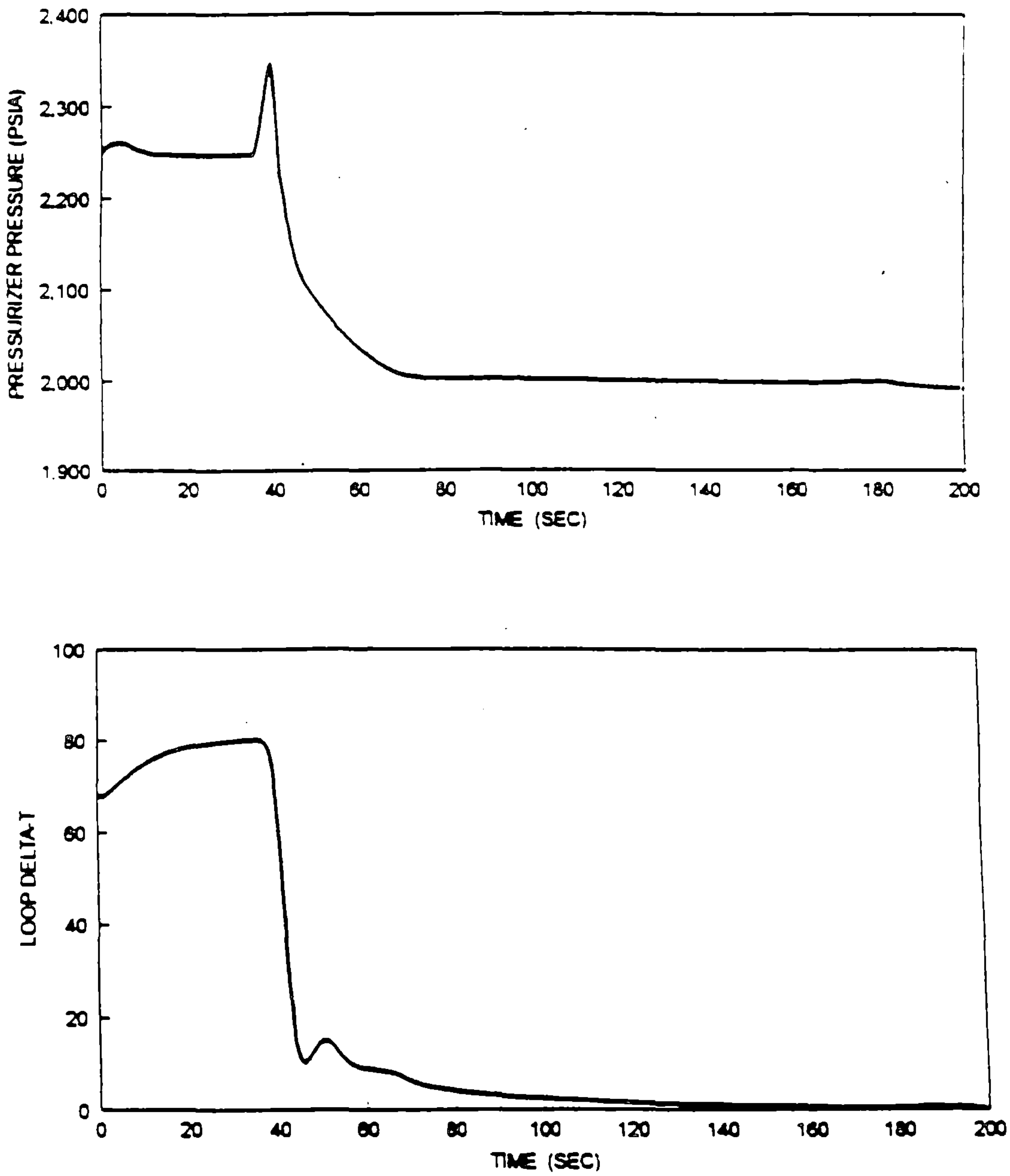


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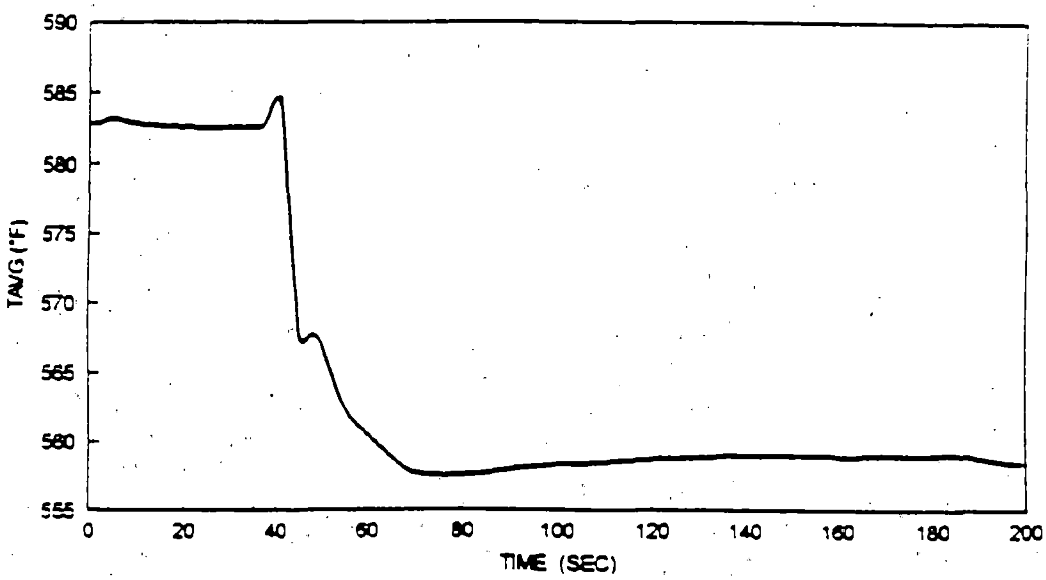
Figure 4.1.10-3

Feedwater Malfunction Single Loop
Manual Rod Control



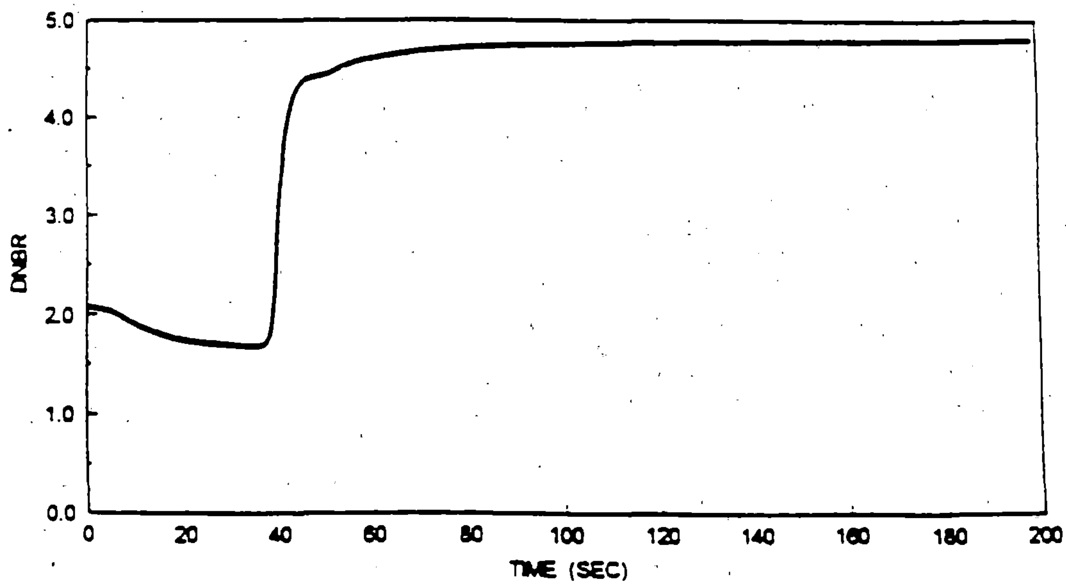
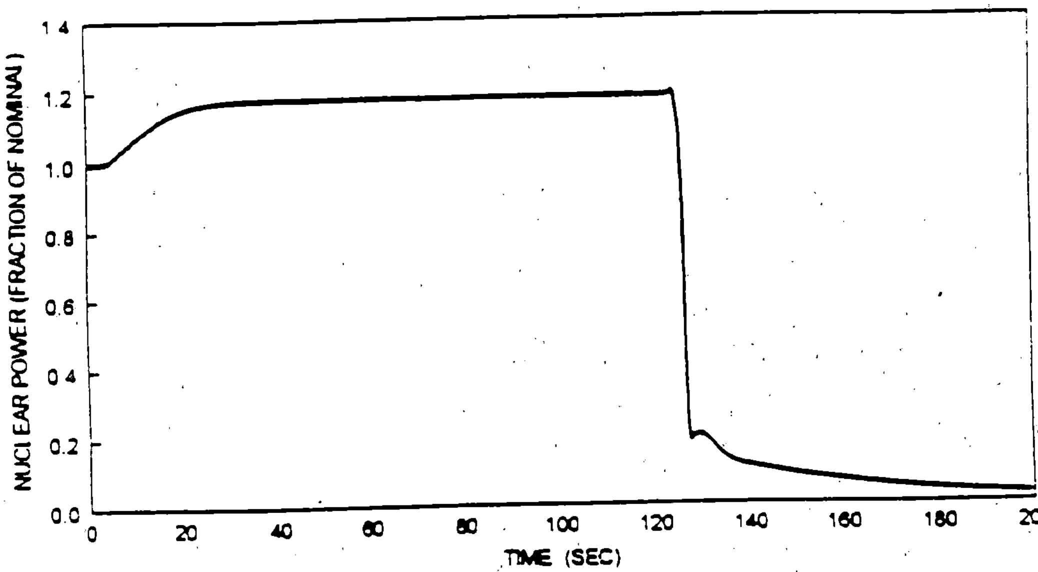


Figure 4.1.10-4

Feedwater Malfunction Multi-Loop
Automatic Rod Control



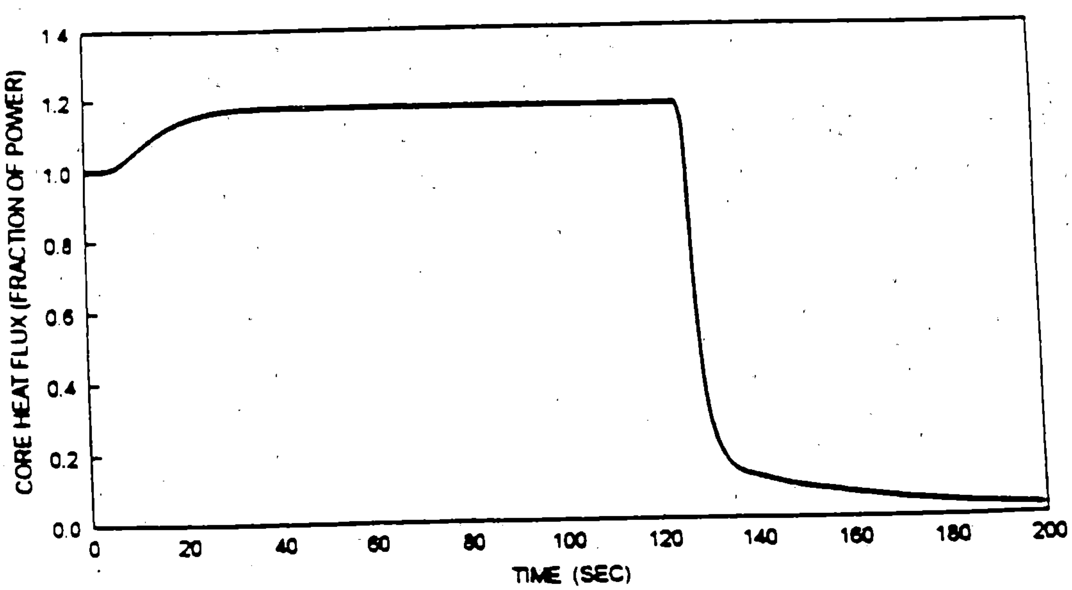


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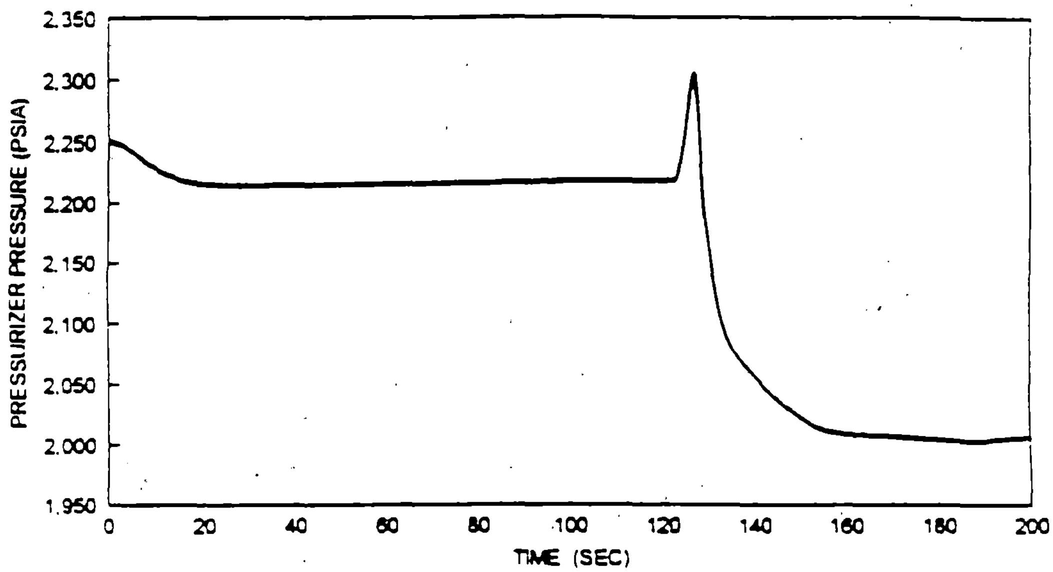
Figure 4.1.10-5

Feedwater Malfunction Multi-Loop
Automatic Rod Control



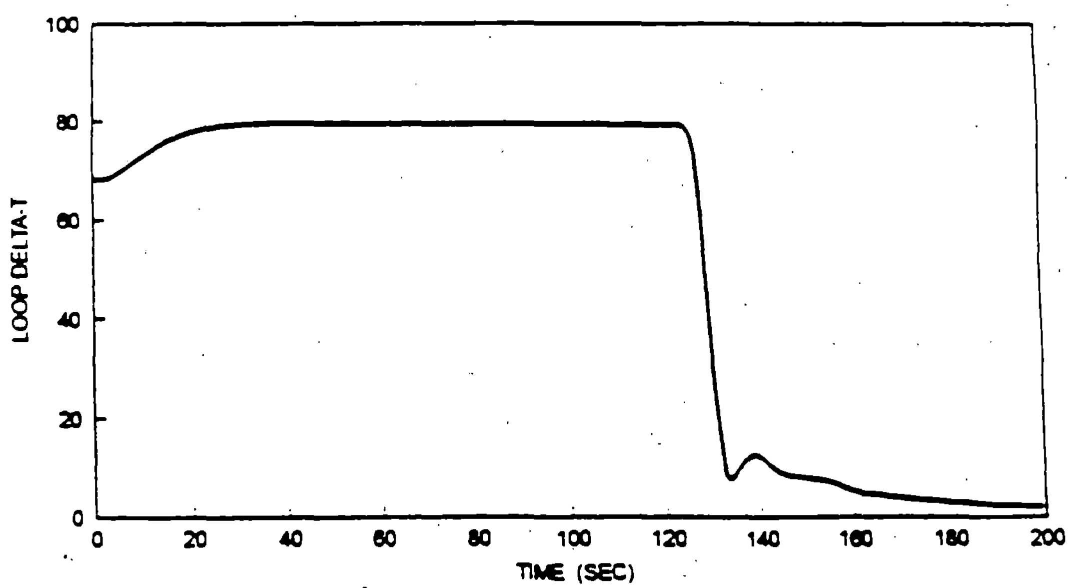


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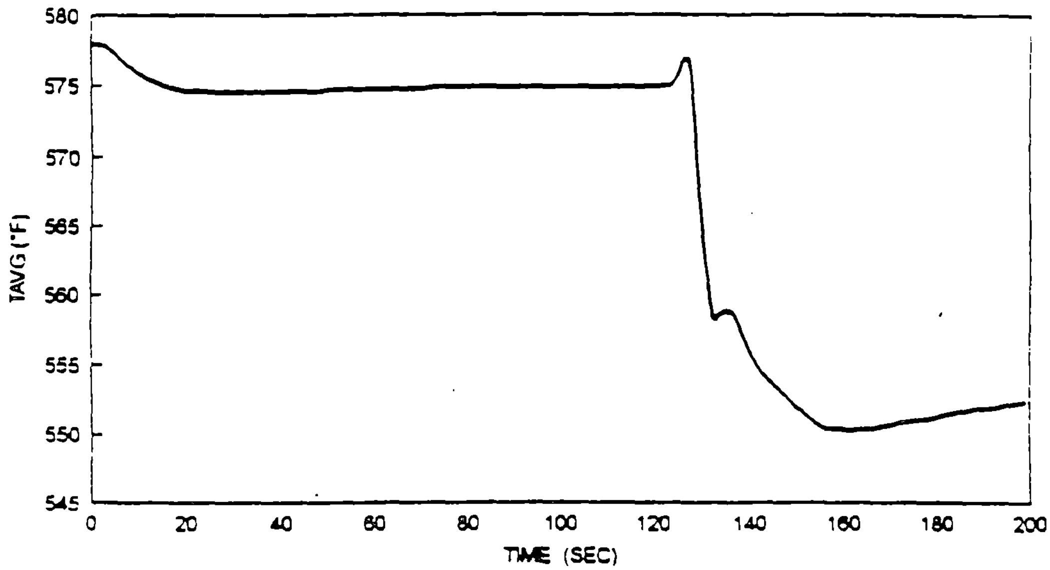
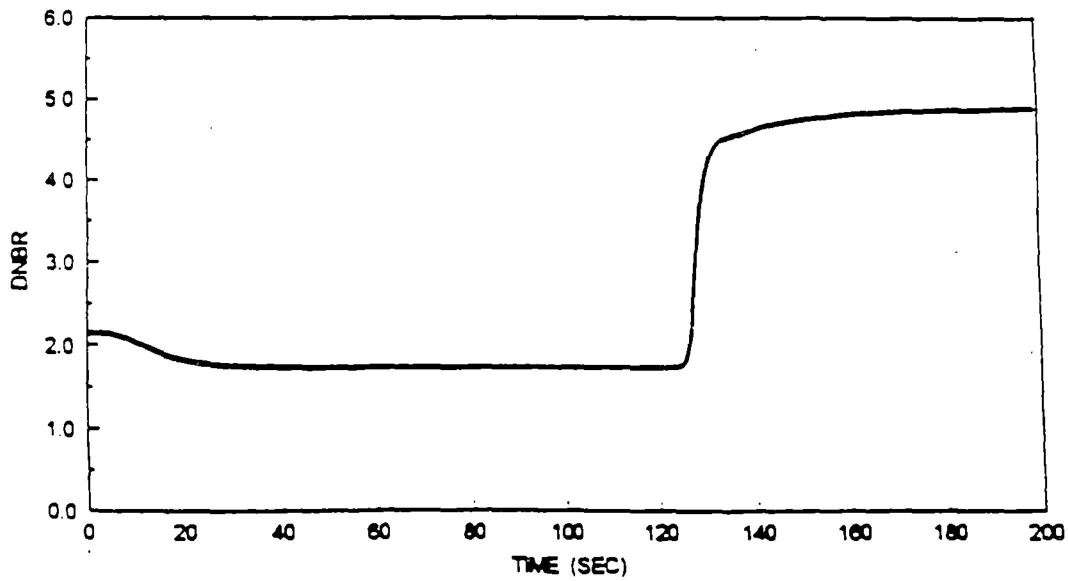
Figure 4.1.10-6

Feedwater Malfunction Multi-Loop
Automatic Rod Control





4.1.11 Excessive Load Increase (UFSAR 15.2.11)

Accident Description:

An excessive load increase incident is defined as a Condition II event resulting from 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 RPS.

Conclusions:

The excessive load increase has been reviewed for the impact of the proposed Technical Specification changes and is not significantly impacted by those changes.

4.1.12 Accidental Depressurization of the Reactor Coolant System (UFSAR 15.2.12)

Accident Description:

The most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, this Condition II event results in a rapidly decreasing RCS pressure which could reach the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease would be to decrease power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature until reactor trip occurs. The pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip. The reactor may be tripped by either of the following RPS signals: (1) OTDT or (2) pressurizer low pressure.

Method of Analysis:

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and main steam safety valves. The code computes pertinent plant variables, including temperatures, pressures, and power level.

In calculating the DNBR, the following conservative assumptions are made:

a. The accident is analyzed using the RTDP. Initial core power, reactor coolant average temperature, and RCS pressure are assumed to be at their nominal values consistent with steady-state full-power operation.

b. A MTC of zero is assumed in this analysis. Thus, no credit is taken for any reactivity feedback from the moderator density change.

c. A high (absolute value) DPC is assumed such that the resultant amount of positive feedback is conservatively high in order to slow the power decrease due to rod insertion following reactor trip.

It should also be noted that, in the analysis, power peaking factors are kept constant at the design values, while, in fact, the core feedback effects would result in considerable flattening of the power distribution. This could increase the calculated DNBR; however, no credit is taken for this effect.

Results:

Figure 4.1.12-1 illustrates the nuclear power and flux transients following the accident. The pressurizer pressure and volume results are given in Figure 4.1.12-2. The resulting DNBR never goes below the safety analysis limit value, as shown in Figure 4.1.12-3. The RCS average temperature transient is also shown in Figure 4.1.12-3. The calculated sequence of events is listed in Table 4.1.12-1.

Conclusions:

The pressurizer low pressure and the OTDT reactor protection trip functions provide adequate protection against this accident. The minimum DNBR remains in excess of the safety analysis limit value.

Table 4.1.12-1

Sequence of Events
Accidental Depressurization of the Reactor Coolant System

Event	Time (seconds)
Inadvertent opening of one pressurizer safety valve	0.0
Reactor trip setpoint reached for overtemperature ΔT	35.0
Rod motion on reactor trip signal	36.5
Minimum DNBR occurs	37.0

Figure 4.1.12-1

Accidental RCS Depressurization

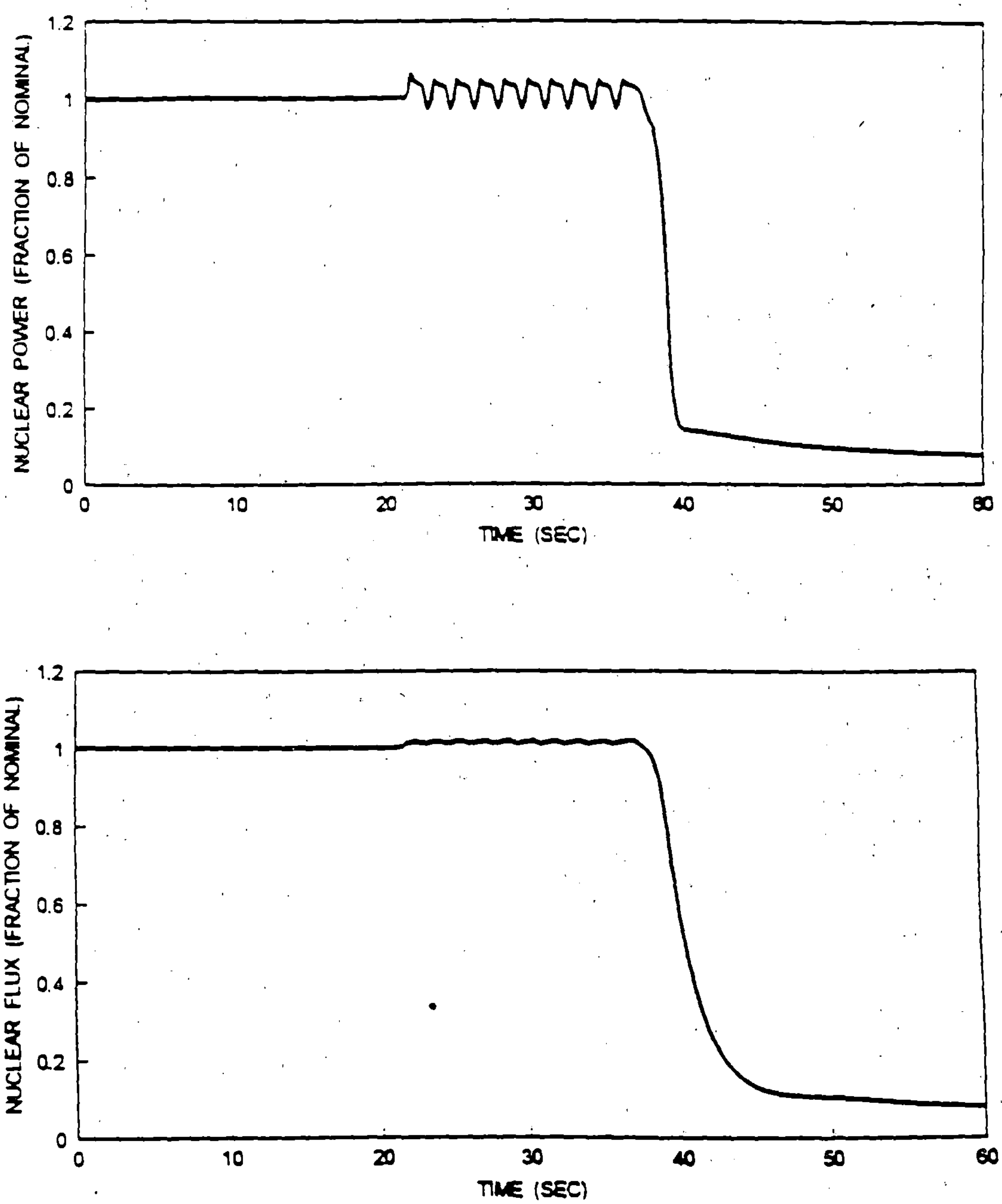
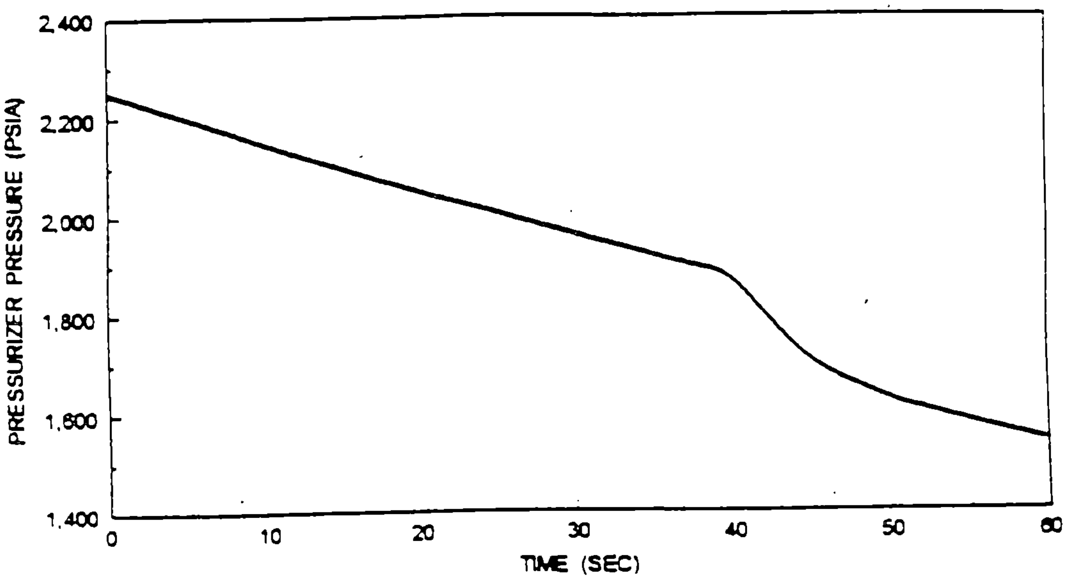


Figure 4.1.12.2

Accidental RCS Depressurization



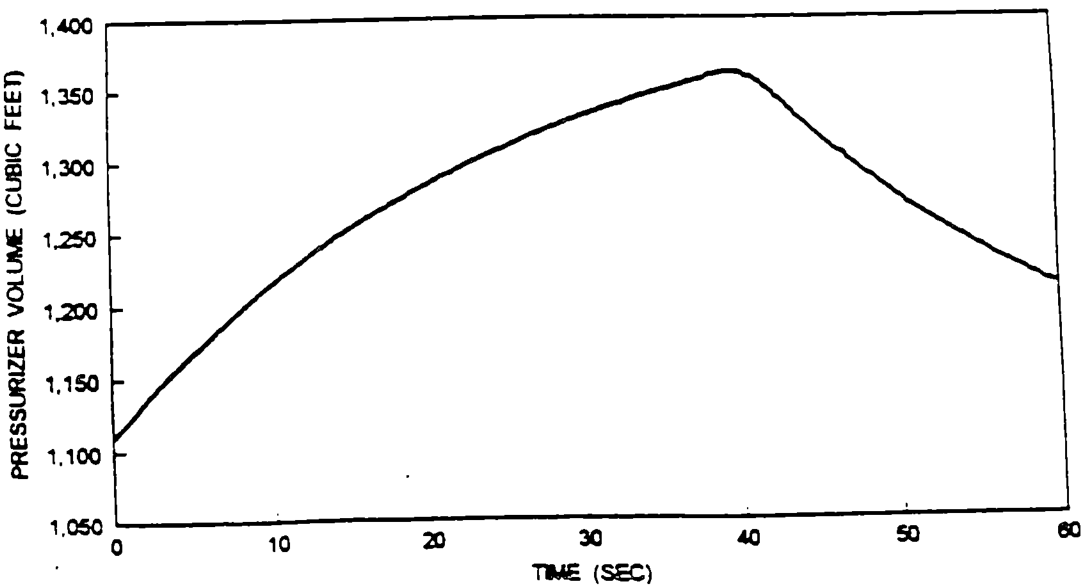
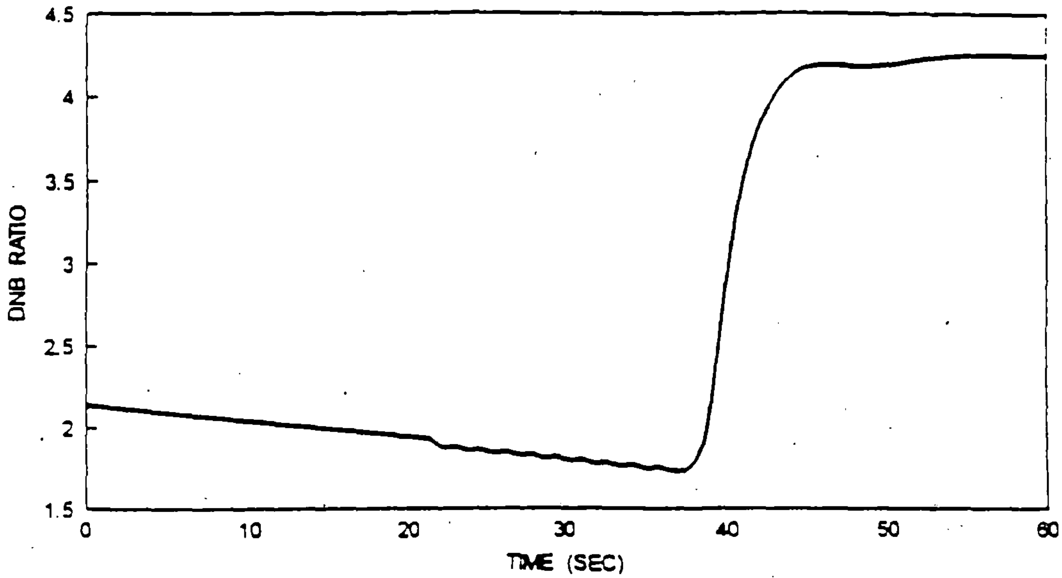
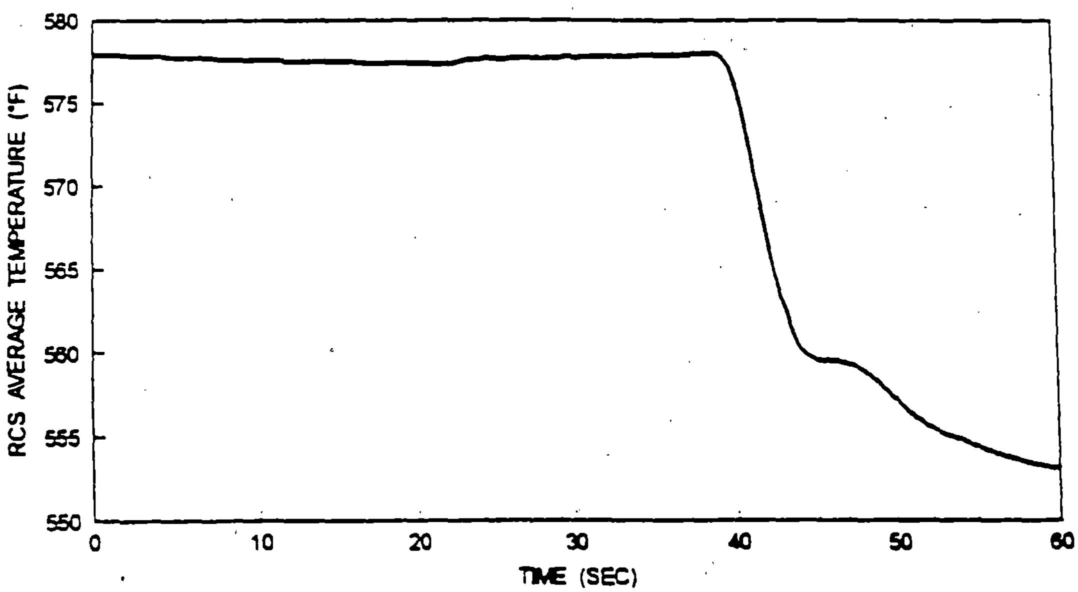


Figure 4.1.12-3

Accidental RCS Depressurization





4.1.13 Main Steam System Failures

The main steam system failure events consist of the following transients:

4.1.13.1 Accidental Depressurization of the Main Steam System (UFSAR 15.2.13)
4.1.13.2 Minor Secondary System Pipe Breaks (UFSAR 15.3.2)
4.1.13.3 Major Secondary System Pipe Breaks (UFSAR 15.4.2)

4.1.13.1 Accidental Depressurization of the Main Steam System (UFSAR 15.2.13)

Accident Description:

This Condition II transient reviews the most severe core conditions resulting from an accidental depressurization of the MSS associated with an inadvertent opening of a single steam dump, main steam relief or main steam safety valve.

The analysis is performed to demonstrate that the DNBR safety analysis limit is not violated for a steam release equivalent to the spurious opening (with failure to close) of the largest of any single steam dump, main steam relief, or main steam safety valve.

Method of Analysis:

The following analyses of a secondary system steam release are performed for this section:

a. A full plant digital computer simulation using LOFTRAN to determine RCS temperature and pressure during the cooldown.

b. An analysis to confirm that there is no DNB.

The following conditions are assumed to exist at the time of a secondary system steam release:

a. End of Life (EOL) shutdown margin at no load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position.

b. The negative moderator density coefficient corresponds to the EOL rodded core with the most reactive RCCA in the fully withdrawn position and includes variation of the coefficient with temperature and pressure. The k_{eff} versus temperature corresponding to the negative MTC used plus the Doppler temperature effect is shown on Figure 4.1.13.1-1.

c. Minimum capability for injection of boric acid solution corresponds to the most restrictive single failure in the SIS. The safety injection flow is provided by one charging pump delivering its full contents to the cold leg header, as shown on Figure 4.1.13.1-2. No credit is taken for the low concentration boric acid which must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) prior to the delivery of 2,300 ppm boric acid to the reactor coolant pumps (RCP). The boron injection tank (BIT) concentration was assumed to be 0 ppm.

d. The case studied is an initial total steam flow of 305 lbs/second at 1,000 psia from one steam generator with offsite power available. Initial hot shutdown conditions at time zero are assumed.

e. The Moody Curve for f(L/D) = 0 is used in computing the steam flow.

f. Perfect moisture separation in the steam generator is conservatively assumed.

Results:

Figures 4.1.13.1-3 through 4.1.13.1-5 show the transients resulting from a steam release of 305 lbs/second at 1,000 psia. In this case, safety injection is initiated automatically by low pressurizer pressure. The minimum DNBR is above the safety analysis limit.

The transient is conservative with respect to the cooldown since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators.

Table 4.1.13.1-1 provides the time sequence of events for the uniform and nonuniform MSS depressurization event.

Conclusions:

Given an accidental depressurization of the MSS, the acceptance criteria are met. With MRP implementation, the DNB transient is bounded by the main steamline rupture presented in Section 4.1.13.2.

Table 4.1.13.1-1

Sequence of Events
Accidental Depressurization of the Main Steam System

Nonuniform Depressurization

Event	Time (seconds)
Inadvertent opening of one main steam safety or relief valve	0
SIS actuated on high steamline differential pressure	78.2
Feedwater isolation occurs	88.2
Pressurizer empties	196.0
Boron reaches reactor coolant system loops	282.7

Uniform Depressurization

Event	Time (seconds)
Inadvertent opening of one main steam safety or relief valve	0
Pressurizer empties	189.4
SIS actuated on low pressurizer pressure	204.4
Feedwater isolation occurs	214.4
Boron reaches reactor coolant system loops	359.2

Figure 4.1.13.1-1

Variation of K_{eff} With
Core Temperature

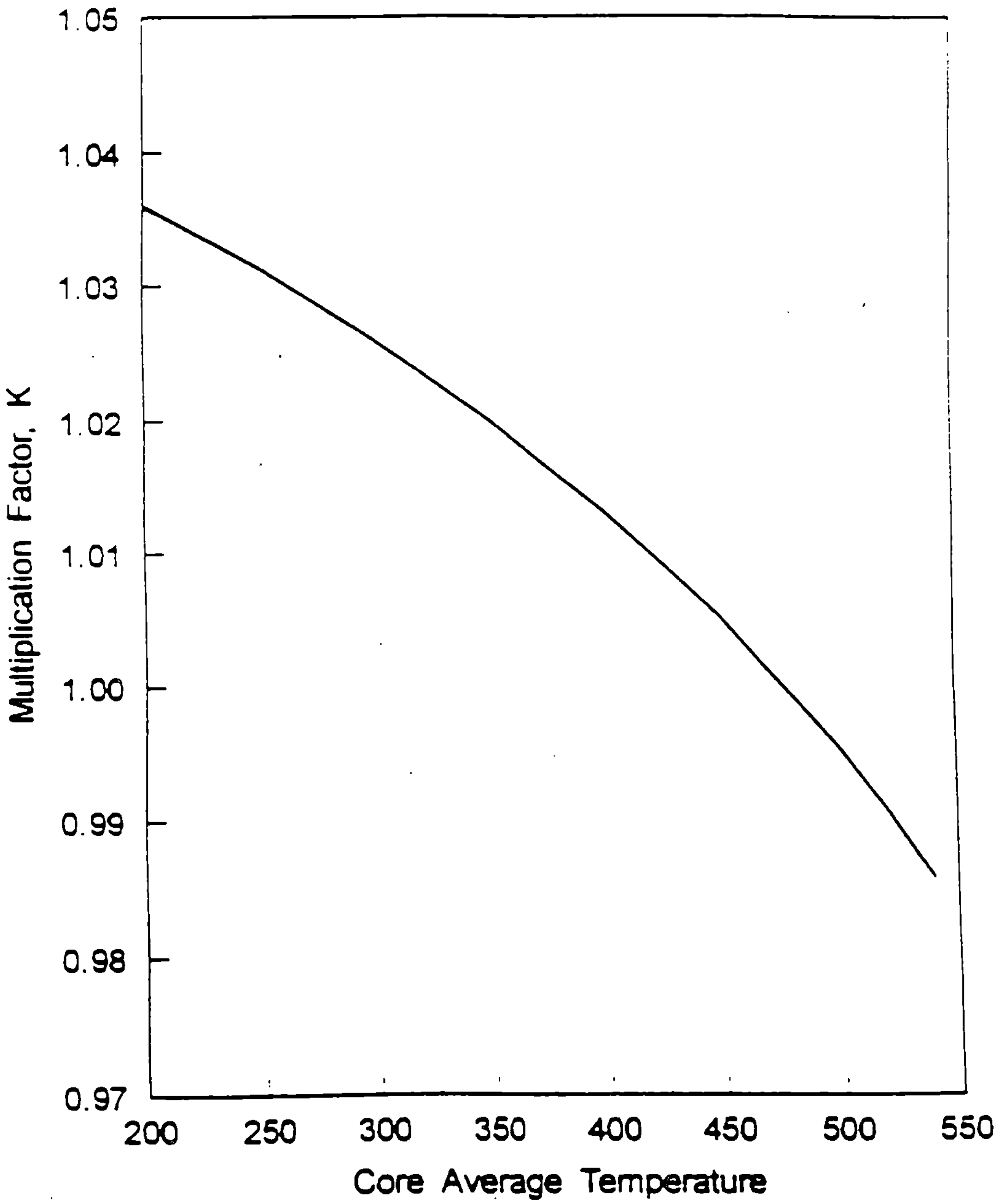


Figure 4.1.13.1-2

Safety Injection Curve

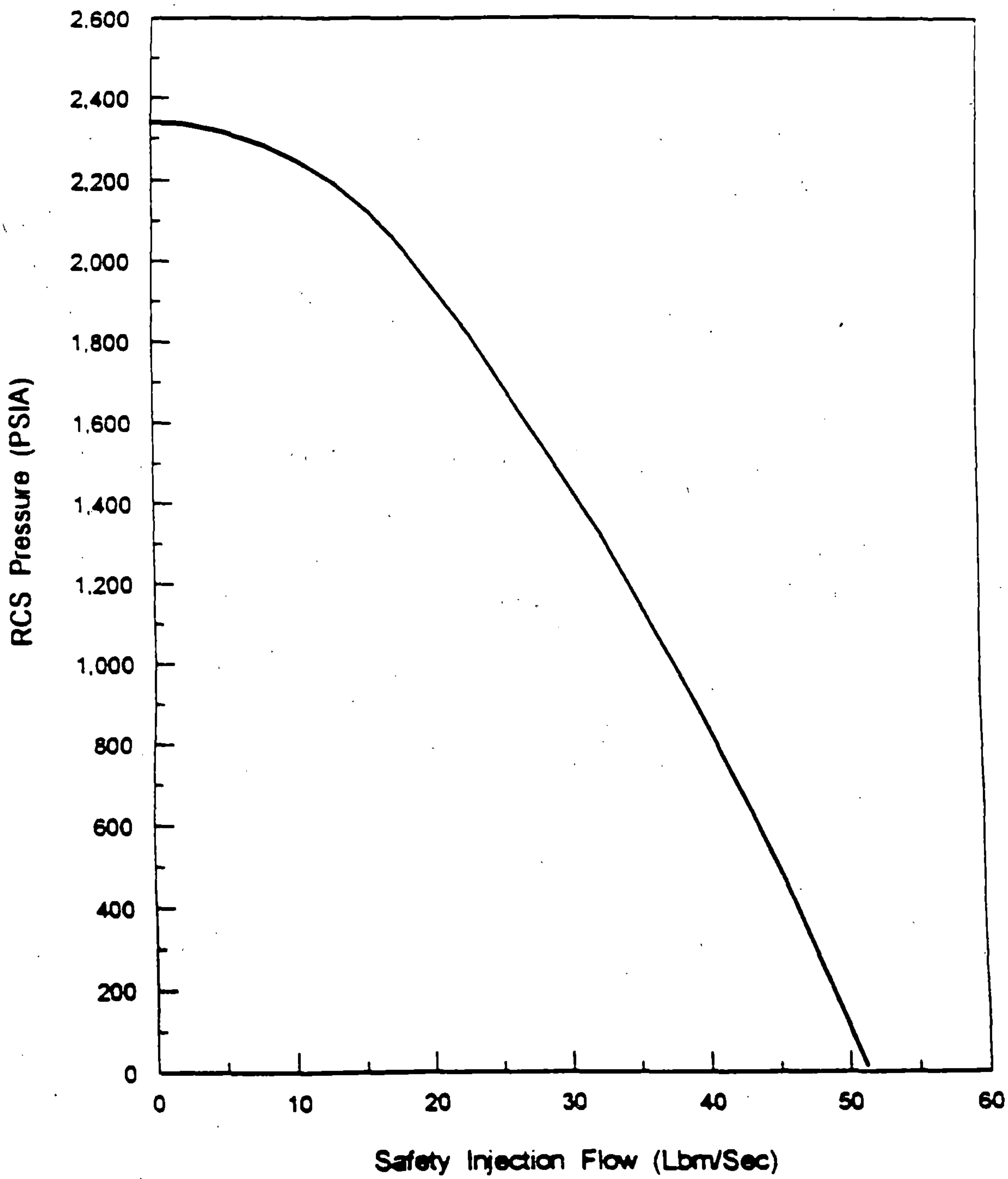


Figure 4.1.13.1-3

Transient Response to Steamline Break
Equivalent to 305 lb/sec at 1,000 psia
with Offsite Power Available

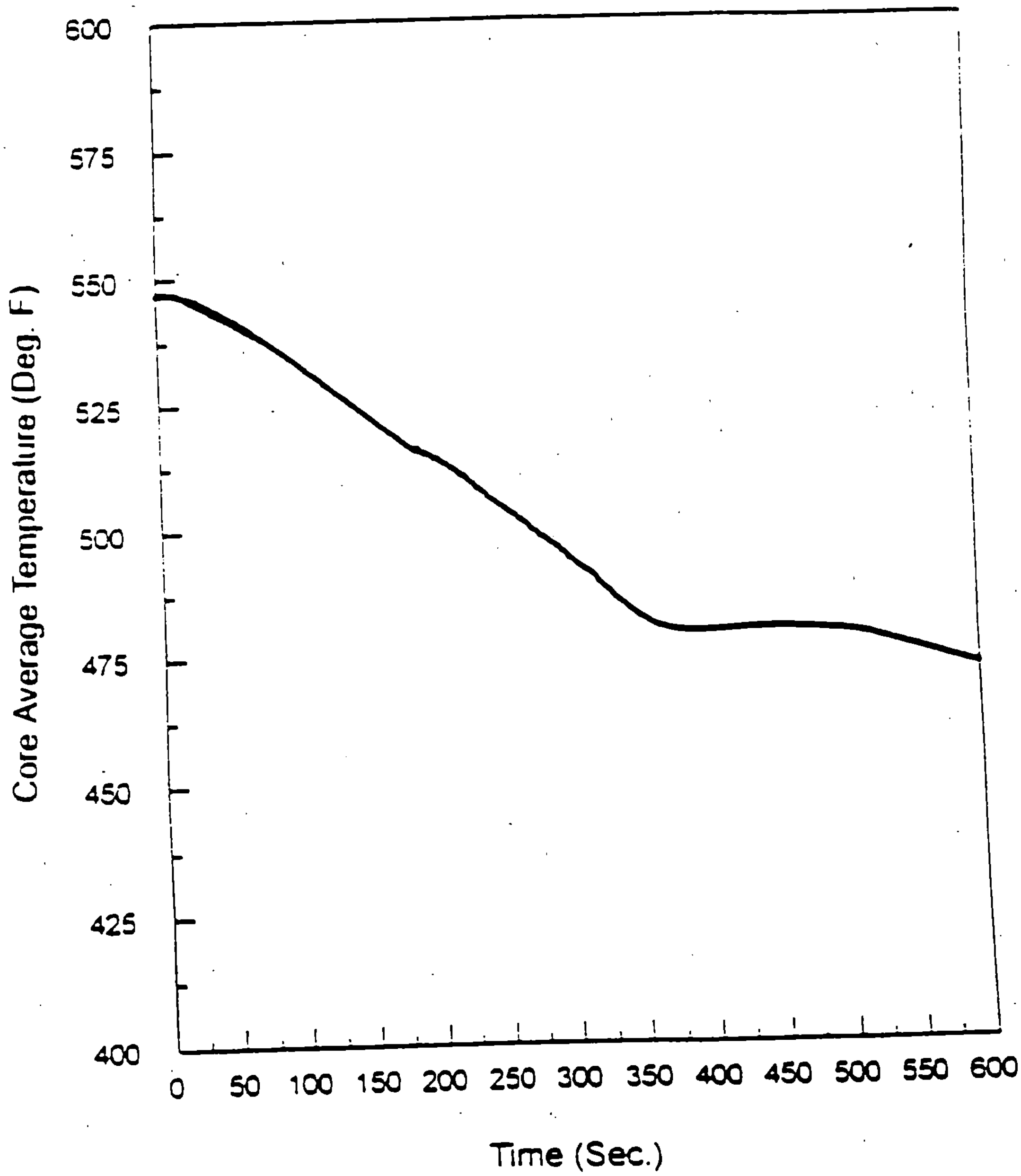


Figure 4.1.13.1-4

Transient Response to Streamline Break
Equivalent to 305 lb/sec at 1,000 psia
With Offsite Power Available

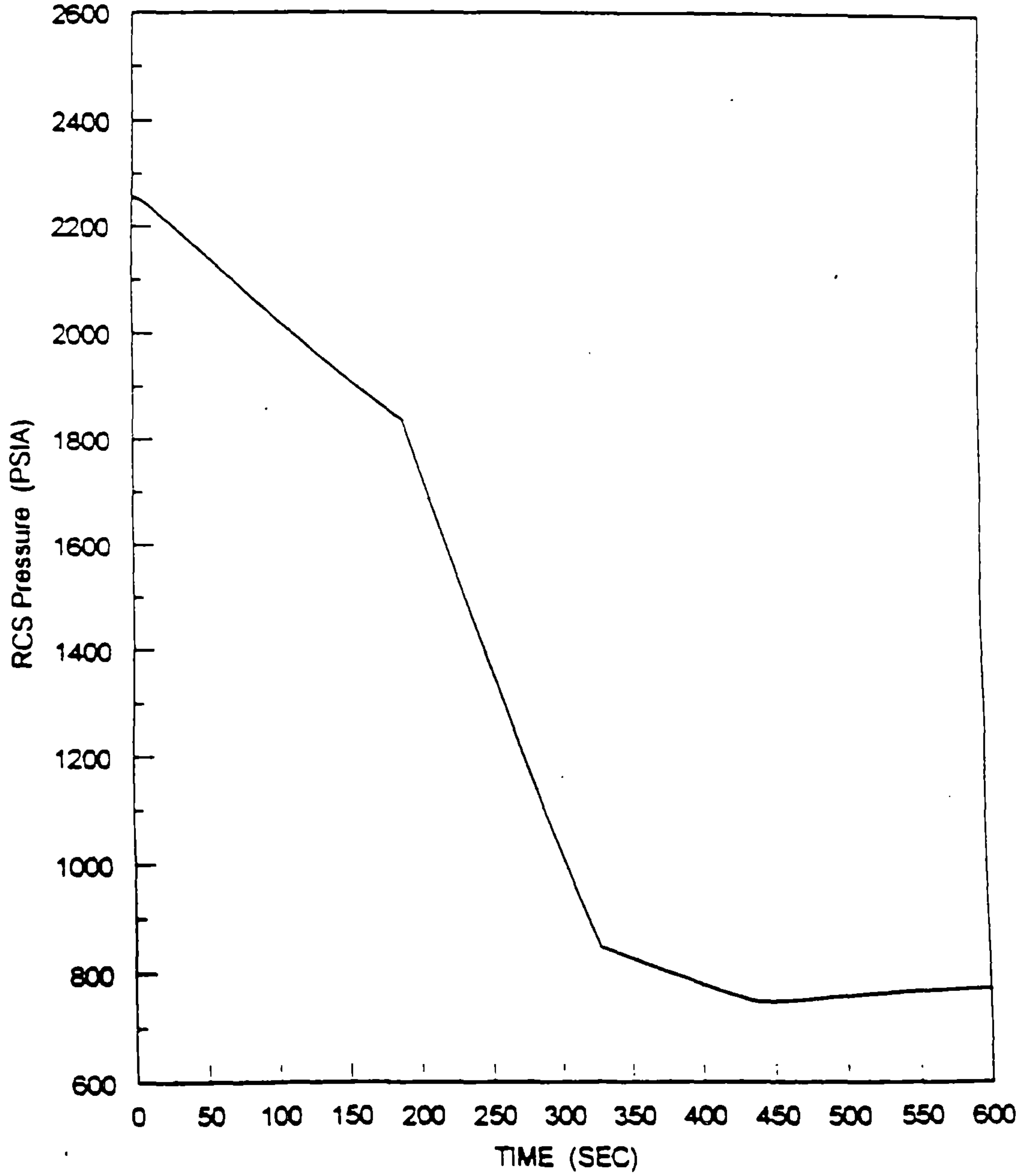
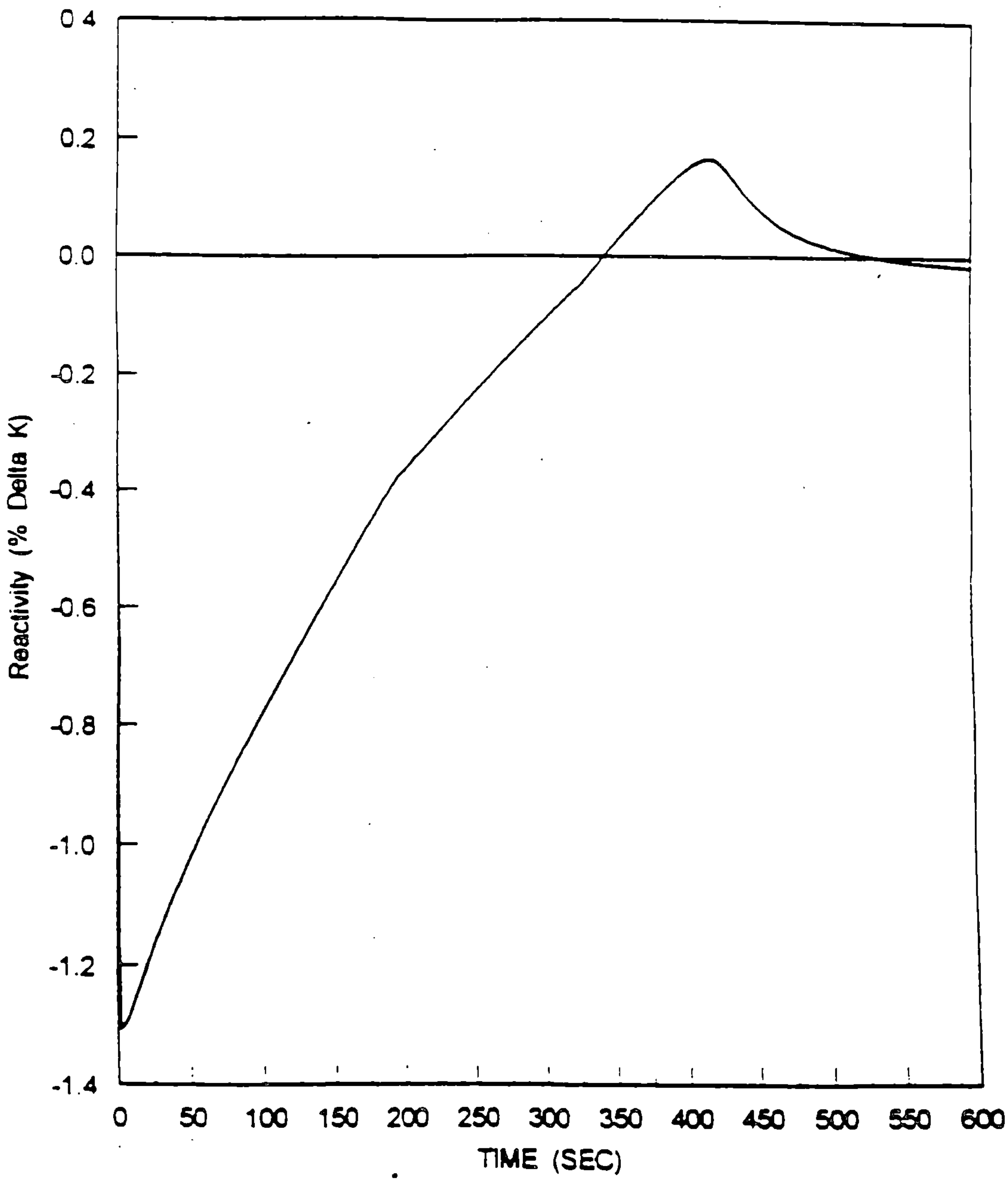


Figure 4.1.13.1-5

Transient Response to Streamline Break
Equivalent to 305 lb/sec at 1,000 psia
With Offsite Power Available



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4.1.13.2 Minor Secondary System Pipe Breaks (UFSAR 15.3.2)

Accident Description:

This section includes ruptures of secondary system lines which would result in steam release rates equivalent to a six inch diameter break or smaller. This accident is classified as a Condition III event.

Method of Analysis:

Minor secondary system pipe breaks are bounded by the results of the major secondary system pipe rupture presented in Section 4.1.13.3 which are conservatively analyzed to meet Condition II acceptance criteria. Therefore, separate analyses for minor secondary system pipe breaks are not required.

Results/Conclusions:

The analysis presented in Section 4.1.13.3 for major secondary system pipe breaks bounds the consequences of a minor secondary system pipe break. Given MRP implementation, results of a minor secondary system pipe break are acceptable since the calculated DNBR would be greater than the safety limit met for more severe major secondary system pipe breaks.

4.1.13.3 Major Secondary System Pipe Rupture (UFSAR 15.4.2)

Accident Description:

The steam release arising from a rupture of a main steam pipe would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. This is classified as a Condition IV event. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative MTC, the cooldown results in a reduction of core SDM. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power.

A return to power following a steam pipe rupture is a potential concern mainly because of the high power peaking factors which exist assuming the most reactive RCCA is stuck in its fully withdrawn position. The core is ultimately shutdown by the boric acid injection delivered by the SIS.

The analysis of a main steam pipe rupture is performed to demonstrate that the following criteria are satisfied:

a. There is no damage to the primary system and the core remains intact.

b. Energy releases to containment from the worst steam pipe break do not cause failure of the containment structure.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis shows that no DNB occurs for any rupture assuming the most reactive RCCA is stuck in its fully withdrawn position. Steamline break mass and energy releases used to demonstrate containment integrity are discussed in Section 4.1.18.

<u>Method of Analysis:</u>

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux, RCS temperature, and pressure resulting from the cooldown following a steam line break. These are determined by using the LOFTRAN code.

2. The DNBR for the core conditions computed by LOFTRAN is determined using the THINC code.

The following conditions are assumed to exist at the time of a main steamline break:

a. End of Life (EOL) shutdown margin at no load, equilibrium xenon conditions, with the most reactive RCCA stuck in its fully withdrawn position is assumed.

b. The negative moderator density coefficient corresponds to the EOL rodded core with the most reactive RCCA in the fully withdrawn position and includes variation of the coefficient with temperature and pressure. The k_{eff} versus temperature at 1,000 psi corresponding to the negative MTC used plus the Doppler temperature effect is shown on Figure 4.1.13.1-1. The variation of reactivity with power at a constant core average temperature is shown on Figure 4.1.13.3-1.

To verify the conservatism of this analysis, the reactivity and power distribution were checked. These core analyses consider the following:

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

For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included.

c. Minimum capability for injection of boric acid solution of 2,300 ppm corresponds to the most restrictive single failure in the SIS. The safety injection flow, as shown on Figure 4.1.13.1-2, is provided by one charging pump delivering its full contents to the cold leg header. Low concentration boric acid must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) prior to the delivery of boric acid to the reactor coolant loops. This effect is considered by assuming that the lines contain unborated water. The boron injection tank (BIT) concentration was assumed to be zero ppm.

d. Four combinations of break sizes and initial plant conditions have been considered in determining the core power and RCS transients. These cases are:

   A. Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle, with the plant initially at no-load conditions. Offsite power is assumed to be available such that full reactor coolant flow exists.

   B. Complete severance of a pipe inside the containment, at the outlet of the steam generator, with the plant initially at no-load conditions. Offsite power is assumed to be available such that full reactor coolant flow exists.

   C. Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle, with the plant initially at no-load conditions. A loss of offsite power is assumed simultaneous with safety injection signal initiation resulting in reactor coolant pump coastdown.

   D. Complete severance of a pipe inside the containment, at the outlet of the steam generator, with the plant initially at no-load conditions. A loss of offsite power is assumed simultaneous with safety injection signal initiation resulting in reactor coolant pump coastdown.

e. Power peaking factors corresponding to one stuck RCCA and non-uniform core inlet coolant temperatures are determined at EOL. The power peaking factors are different for each case studied since they depend on the core power, temperature, pressure, and flow. All of the cases studied assume initial hot shutdown conditions at time zero since this represents the most limiting initial condition.

f. The Moody Curve for f(L/D) = 0 is used in computing the steam flow.

g. Assuming perfect moisture separation in the steam generator leads to conservative results since considerable water would actually be discharged.

h. Minimum shutdown margin of 1.3% Dk/k.

Results:

The analyses showed that the previous steamline break analyses would not be significantly impacted by the MRP implementation and all cases provided acceptable results. The previously limiting case (Case B) remains limiting and bounds the results of the other steamline break and MSS depressurization cases.

The time sequence of events is presented in Table 4.1.13.3-1. It should be noted that only one steam generator blows down completely following this steam line break event.

Case A

Figures 4.1.13.3-2 through 4.1.13.3-4 show the core average temperature, RCS pressure, total steam flow, core heat flux, reactivity, and core boron following a main steam pipe rupture downstream of the flow measuring nozzle at initial no-load conditions.

As shown on Table 4.1.13.3-1, the core attains criticality with the RCCAs inserted (with the design SDM and assuming one stuck RCCA) before the 2,300 ppm boron solution enters the RCS from the SIS. A peak core power below the nominal full power value is attained.

Case B

Figures 4.1.13.3-5 through 4.1.13.3-7 show the core average temperature, RCS pressure, total steam flow, core heat flux, reactivity, and core boron transients. The sequence of events shown in Table 4.1.13.3-1 is similar to that for Case A except that criticality is obtained earlier due to a more rapid cooldown, a higher peak core average power is attained, and the accumulators are actuated.

Cases C and D

Figures 4.1.13.3-8 through 4.1.13.3-13 show the RCS transient and core heat flux for Cases C and D which assume a loss of offsite power at the time the safety injection signal is generated. In each case, criticality is achieved later and the core increase is slower than in the similar cases (Cases A and B) with offsite power assumed available. For both Cases C and D, the peak core power remains well below the full power value.

Conclusions:

A DNB analysis was performed for the limiting major secondary system pipe break. Case B, complete severance of a pipe at the outlet of the steam generator with offsite power available, was determined to be limiting with respect to minimum margin to DNB. The minimum DNBR remains above the safety limit. This case bounds the other steamline break core response results.

Table 4.1.13.3-1

Sequence of Events
Major Secondary System Pipe Rupture

Case	Event	Time (seconds)
A	Steam line ruptures	0
	SIS actuated on high steam flow coincident with low steam pressure	0.8
	Feedwater Isolation	10.8
	Pressurizer empties	12.6
	Steamline Isolation	12.8
	Criticality attained	26.2
	2300 ppm boron solution reaches reactor coolant loops	128.4
B	Steam line ruptures	0
	SIS actuated on hihg steamline differential pressure	1.2
	Feedwate rIsolation	11.2
	Pressurizer empties	13.8
	Steamline Isolation	14.0
	Criticality attained	18.2
	2300 ppm boron solution reaches reactor coolant loops	128.0
	Accumulators actuated	145.0
C	Steam line ruptures	0
	SIS actuated on high steam flow coincident with low steam pressure	0.8
	Feedwater Isolation	10.8
	Steamline Isolation	12.8
	Pressurizer empties	13.6
	Criticality attained	30.2
	2300 ppm boron solution reaches reactor coolant loops	134.0
D	Steam line ruptures	0
	SIS actuated on high steamline differential pressure	1.2
	Feedwater Isolation	11.2
	Steamline Isolation	14.0
	Pressurizer empties	15.4
	Criticality attained	24.2
	2300 ppm boron solution reaches reactor coolant loops	135.2

Figure 4.1.13.3-1

Variation of Reactivity with Power
At Constant Core Average Temperature

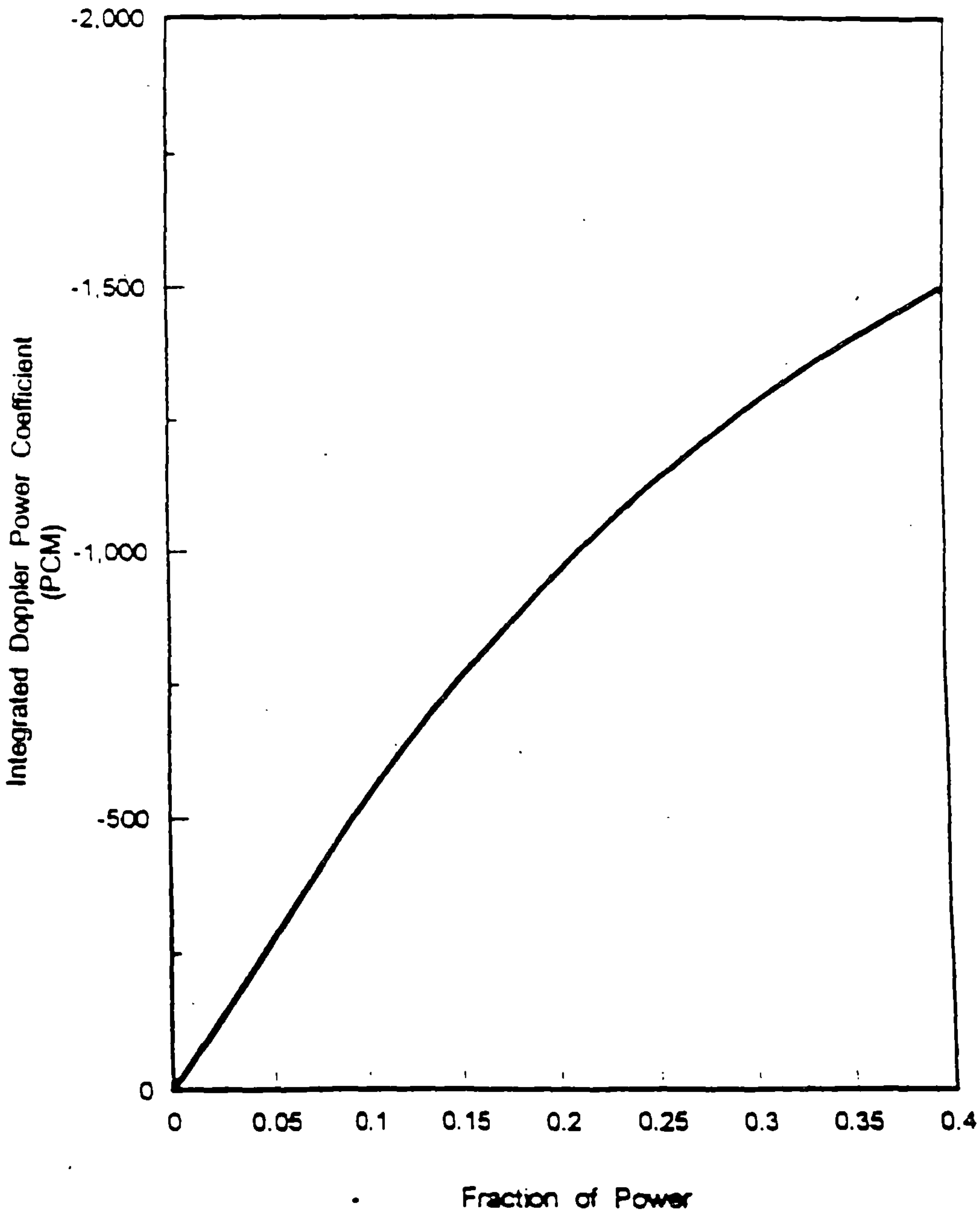
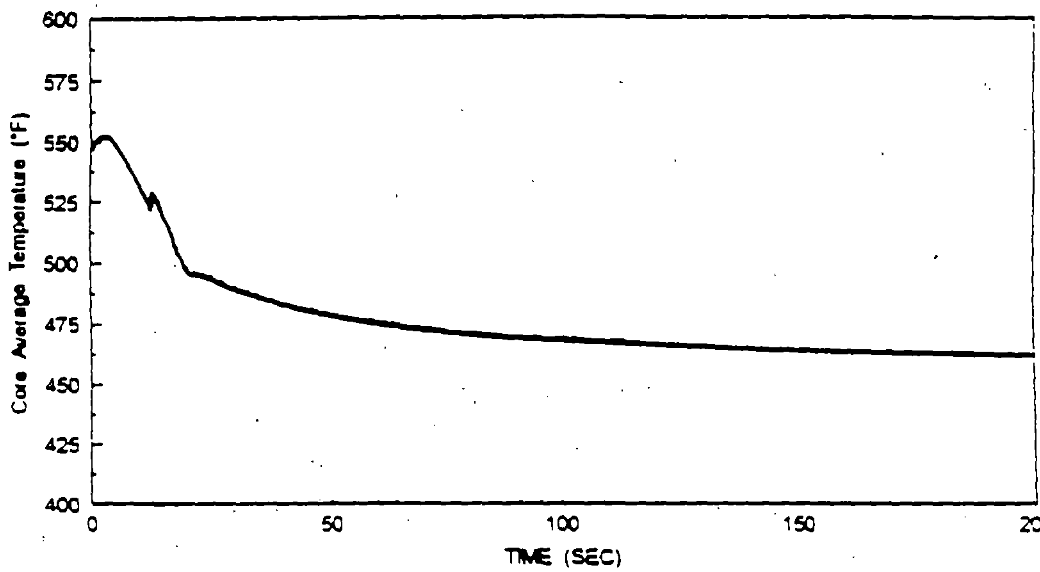


Figure 4.1.13.3-2

Transient Response to Steamline Break
Downstream of Flow Measuring Nozzle with
Safety Injection and Offsite Power
(Case A)



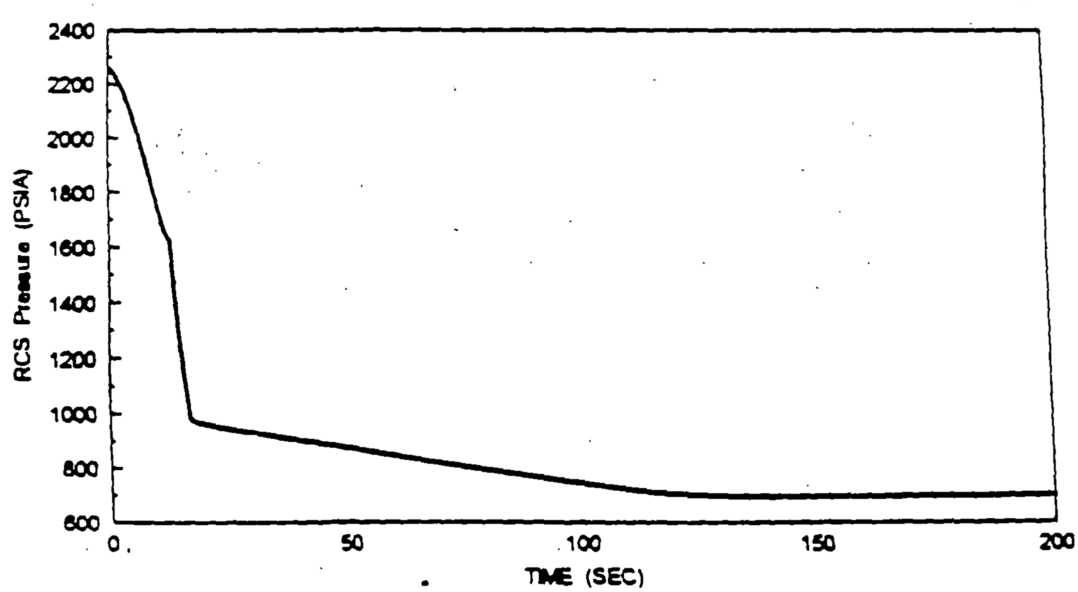


Figure 4.1.13.3-3

Transient Response to Steamline Break Downstream of Flow Measuring Nozzle with Safety Injection and Offsite Power (Case A)

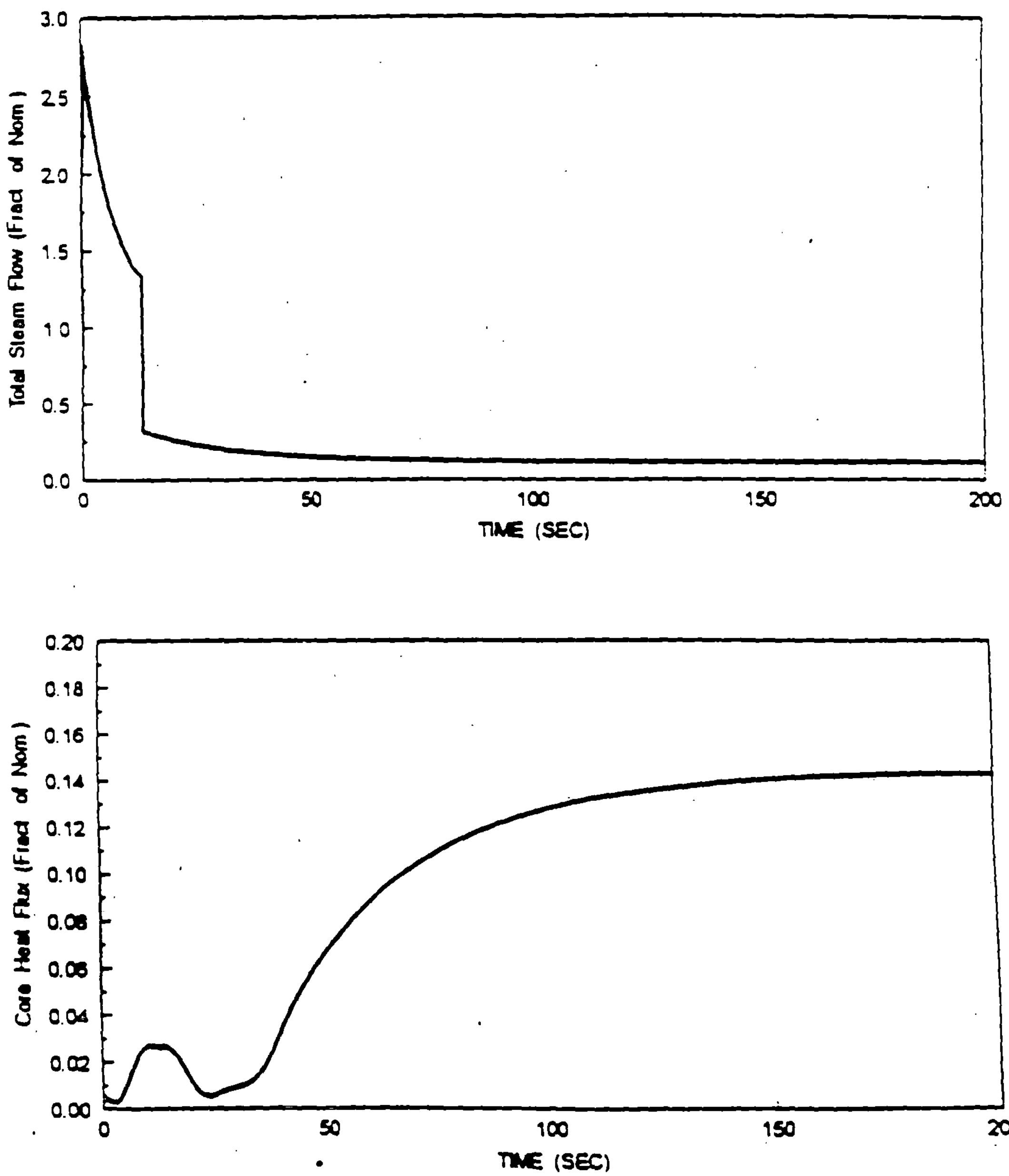


Figure 4.1.13.3-4

Transient Response to Steamline Break
Downstream of Flow Measuring Nozzle with
Safety Injection and Offsite Power
(Case A)

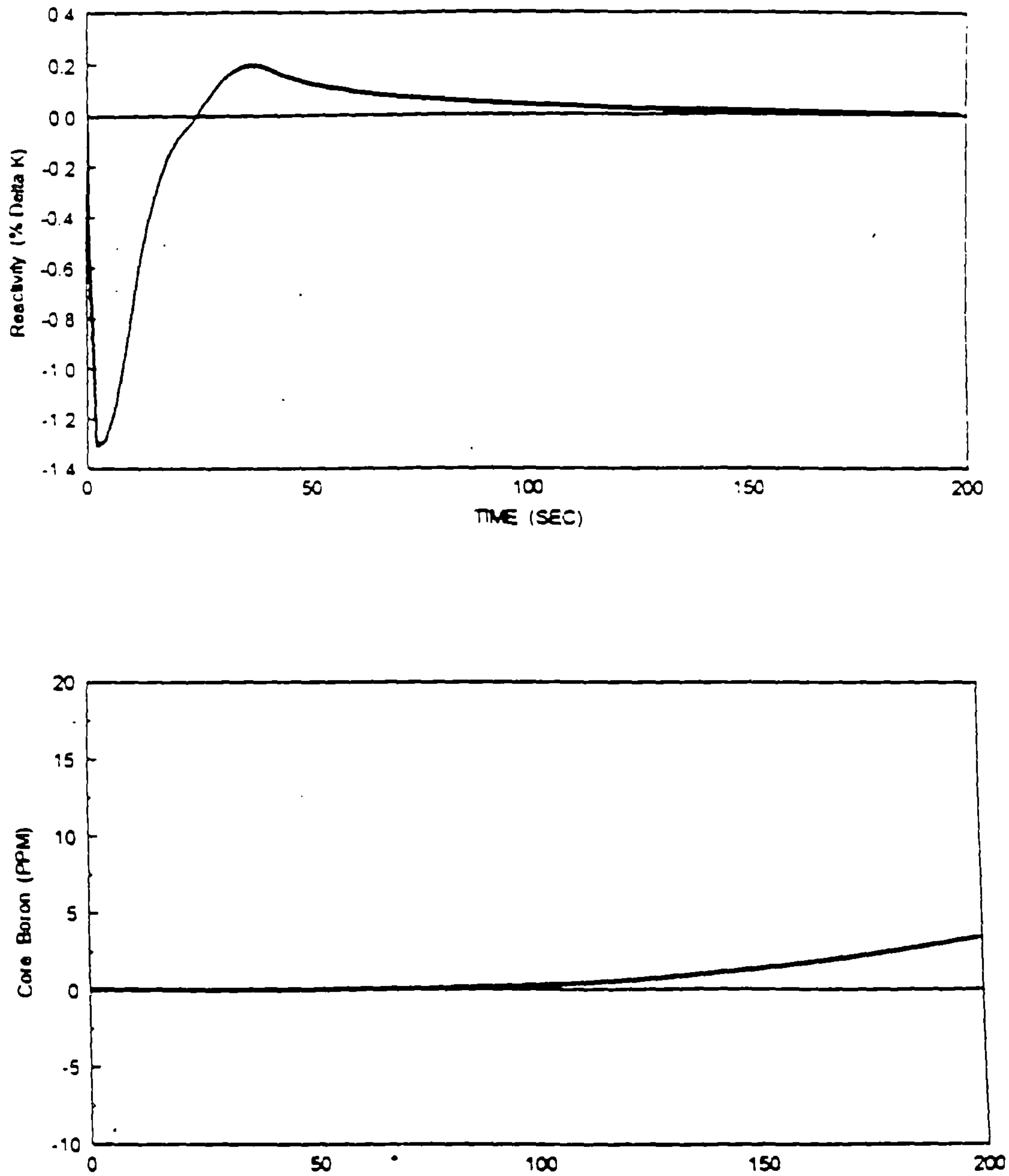
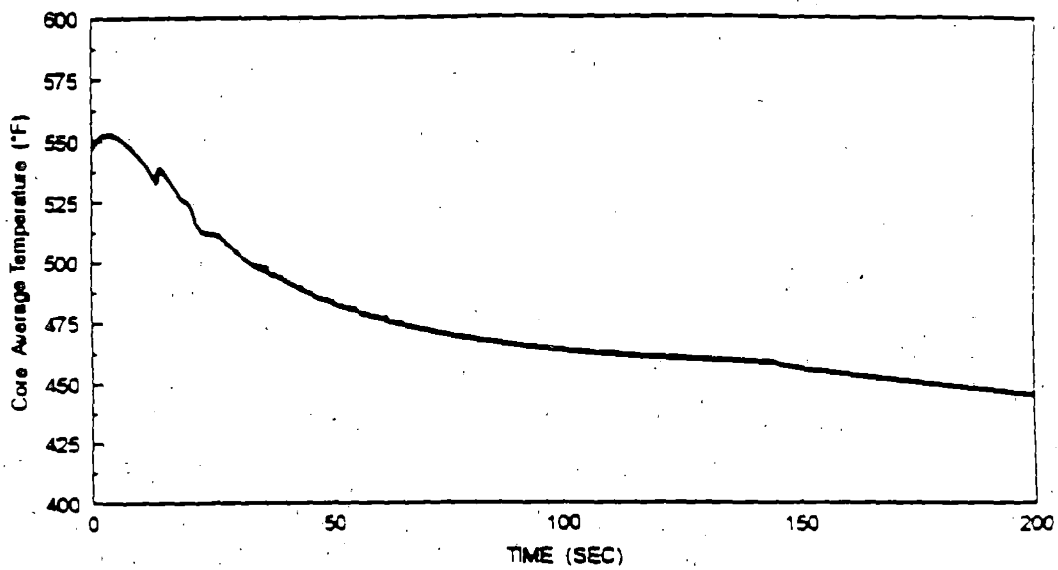


Figure 4.1.13.3-5

Transient Response to Steamline Break at Exit of Steam Generator with Safety Injection and Offsite Power (Case B)



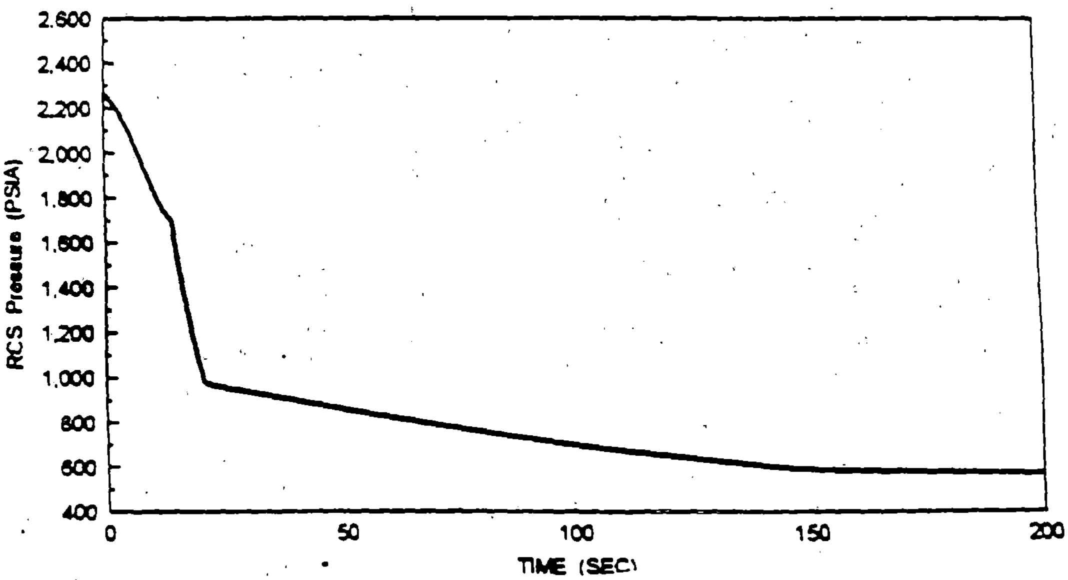
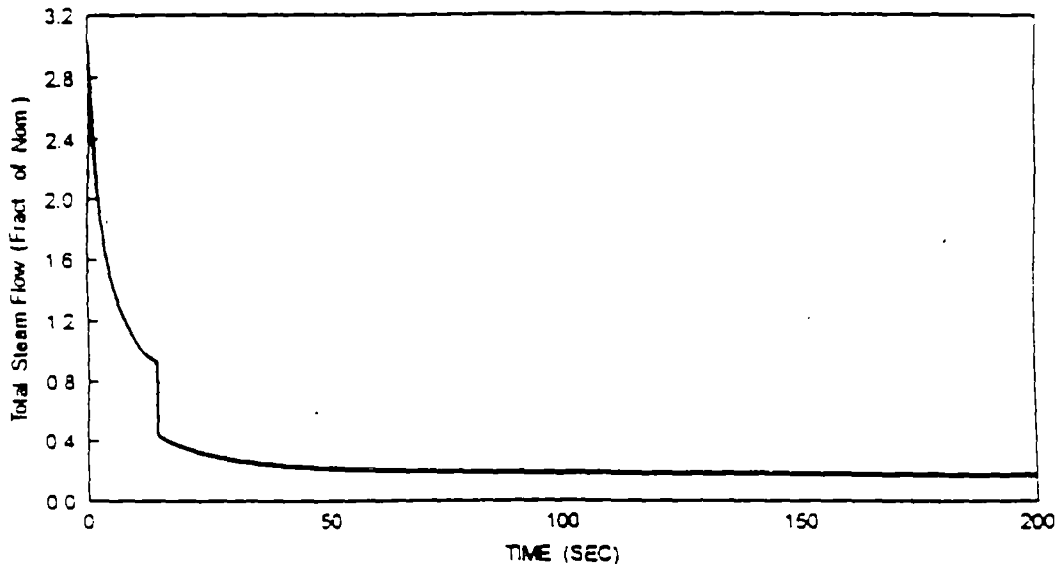


Figure 4.1.13.3-6

Transient Response to Steamline Break
at Exit of Steam Generator with
Safety Injection and Offsite Power
(Case B)



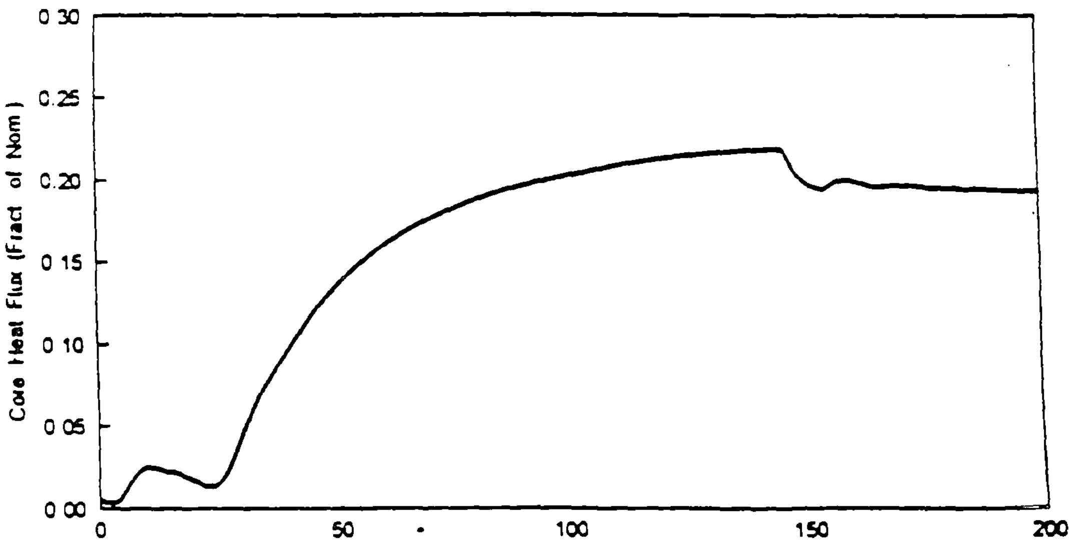


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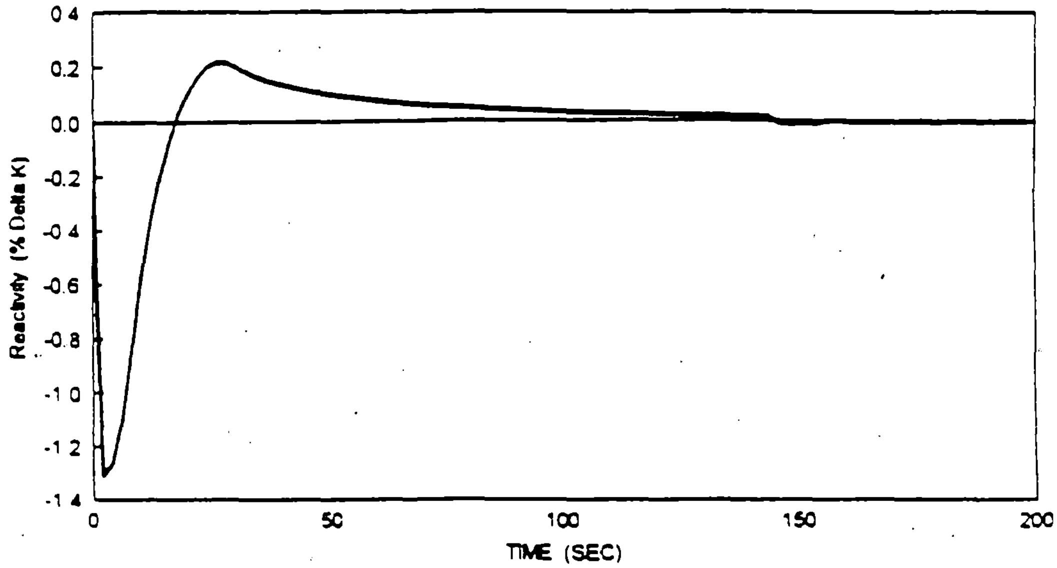
Figure 4.1.13.3-7

Transient Response to Steamline Break at Exit of Steam Generator with Safety Injection and Offsite Power (Case B)



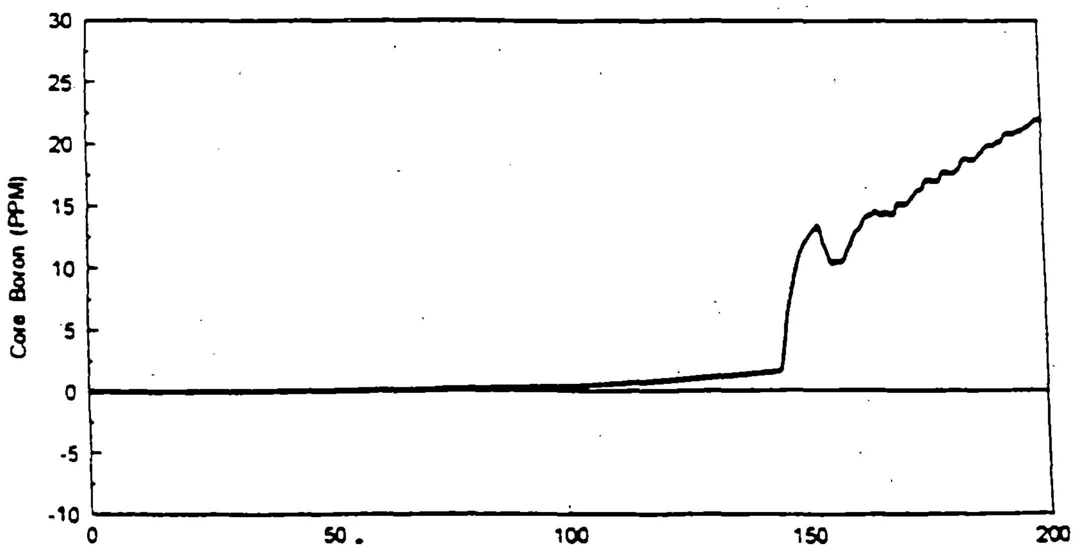
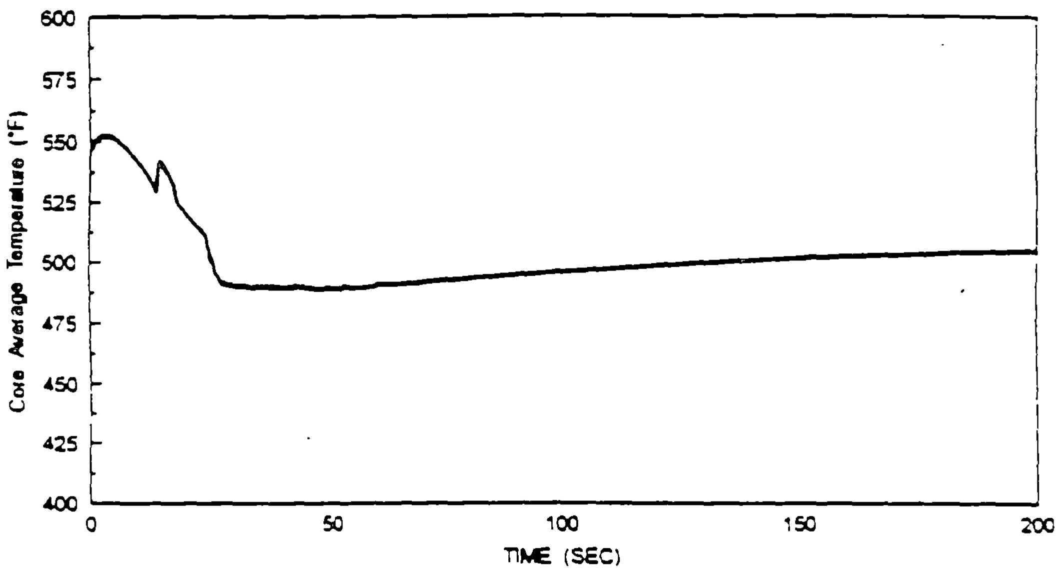


Figure 4.1.13.3-8

Transient Response to Steamline Break Downstream of Flow Measuring Nozzle with Safety Injection, Without Offsite Power (Case C)



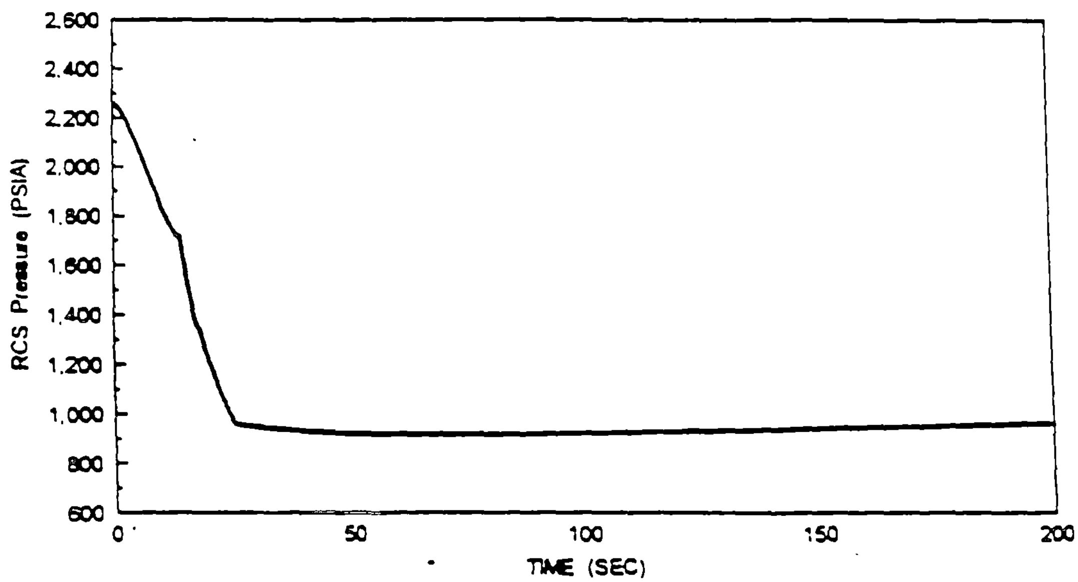


Figure 4.1.13.3-9

Transient Response to Steamline Break Downstream of Flow Measuring Nozzle with Safety Injection, Without Offsite Power (Case C)

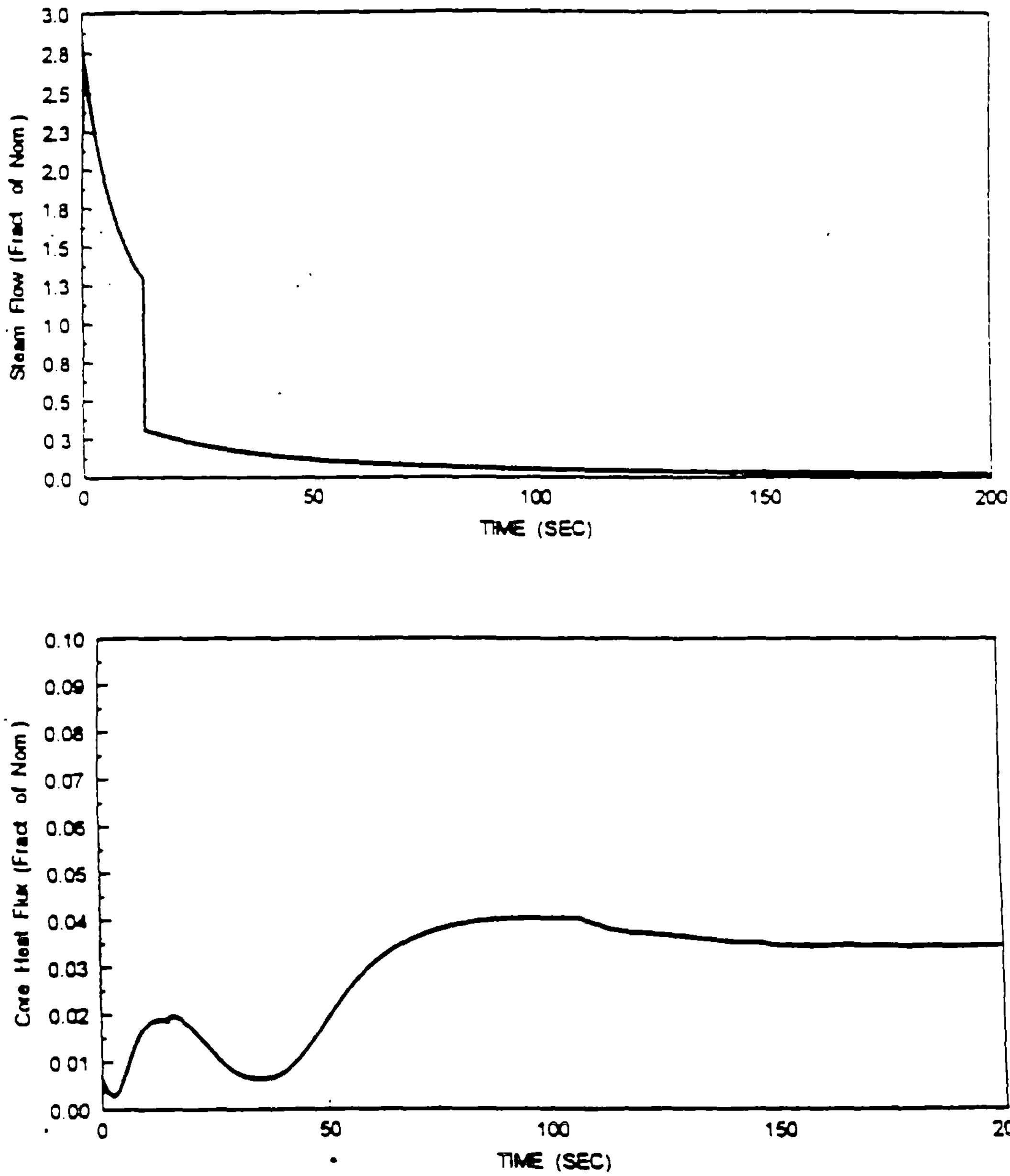


Figure 4.1.13.3-10

Transient Response to Steamline Break Downstream of Flow Measuring Nozzle with Safety Injection, Without Offsite Power (Case C)

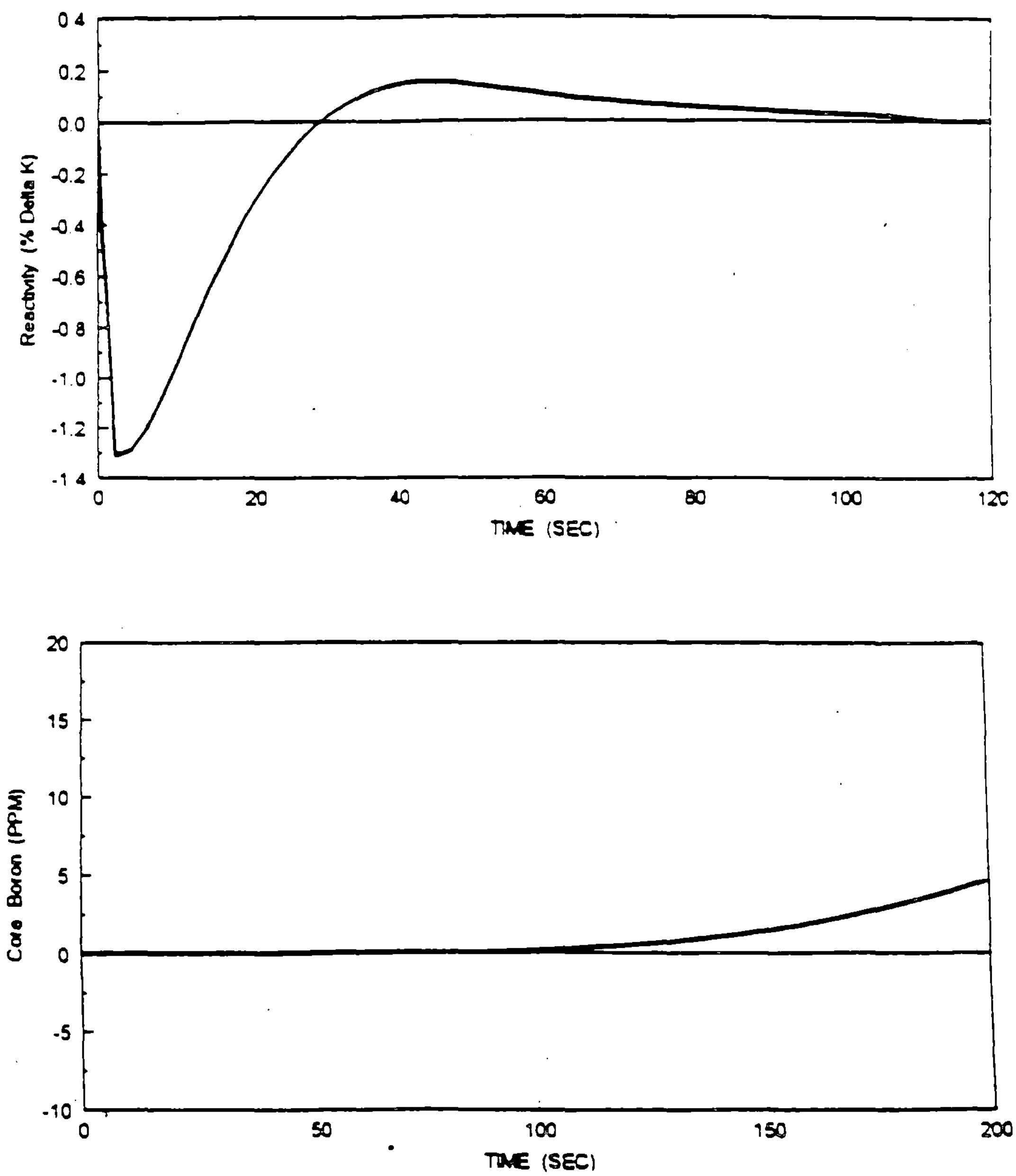
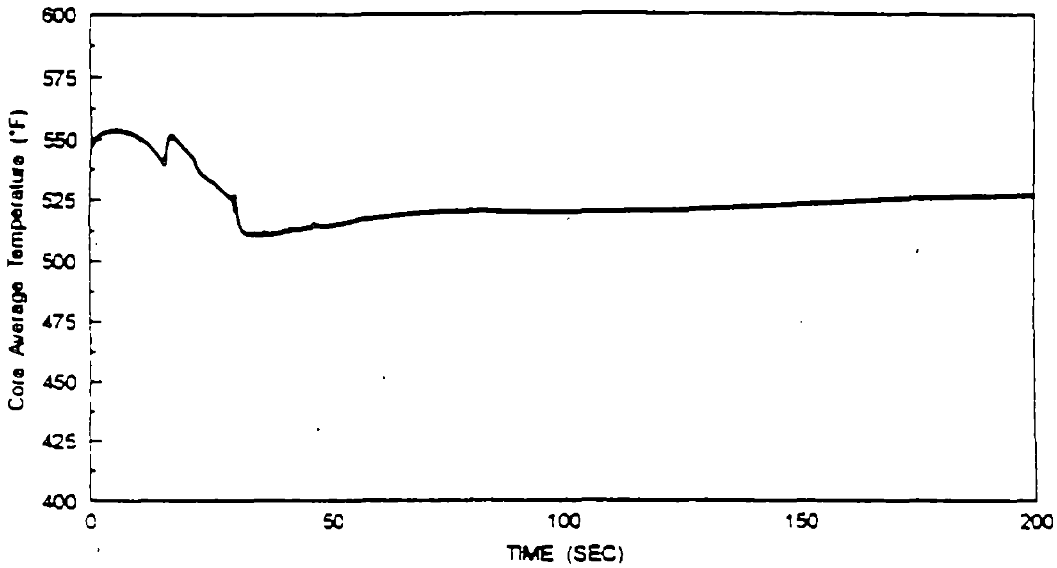


Figure 4.1.13.3-11

Transient Response to Steamline Break at Exit of Steam Generator with Safety Injection, Without Offsite Power (Case D)



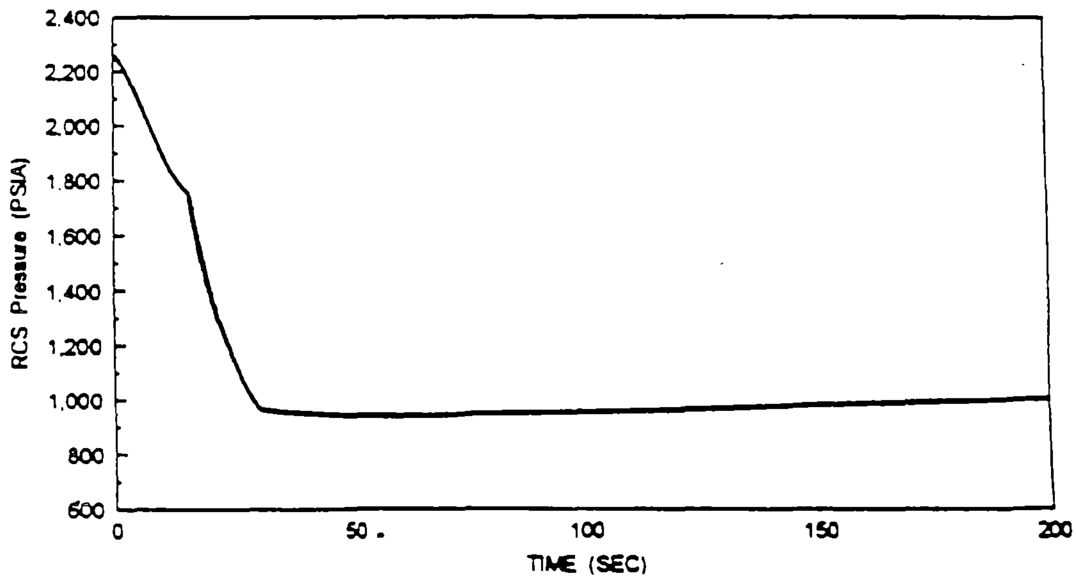


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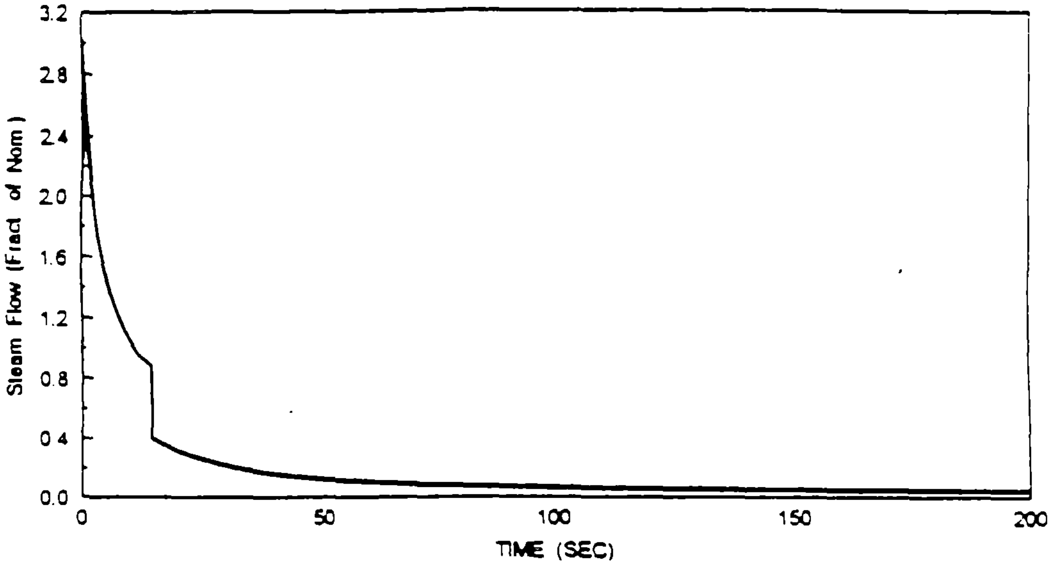
Figure 4.1.13.3-12

Transient Response to Steamline Break
at Exit of Steam Generator with
Safety Injection, Without Offsite Power
(Case D)



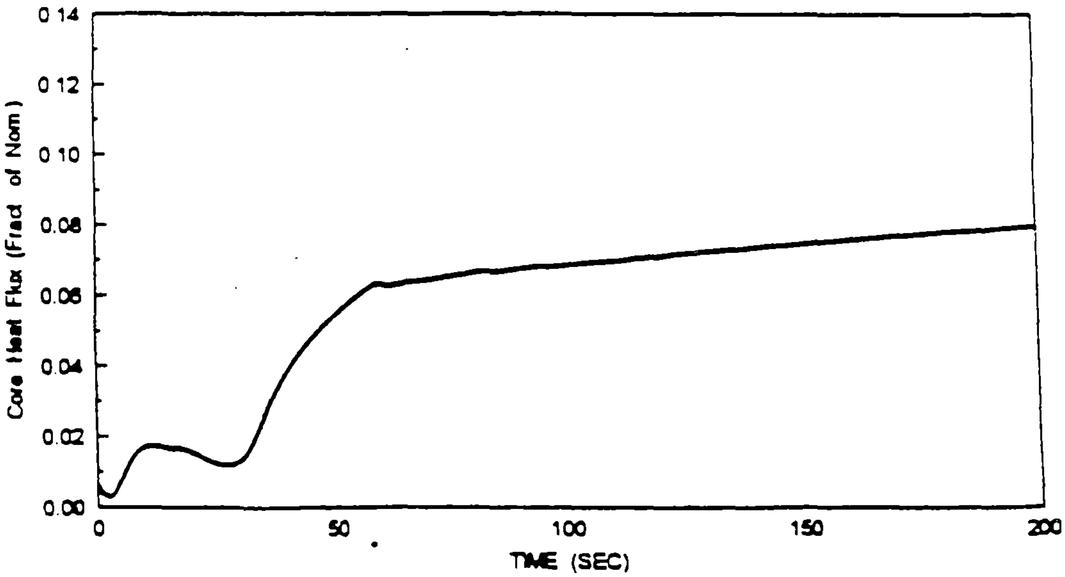
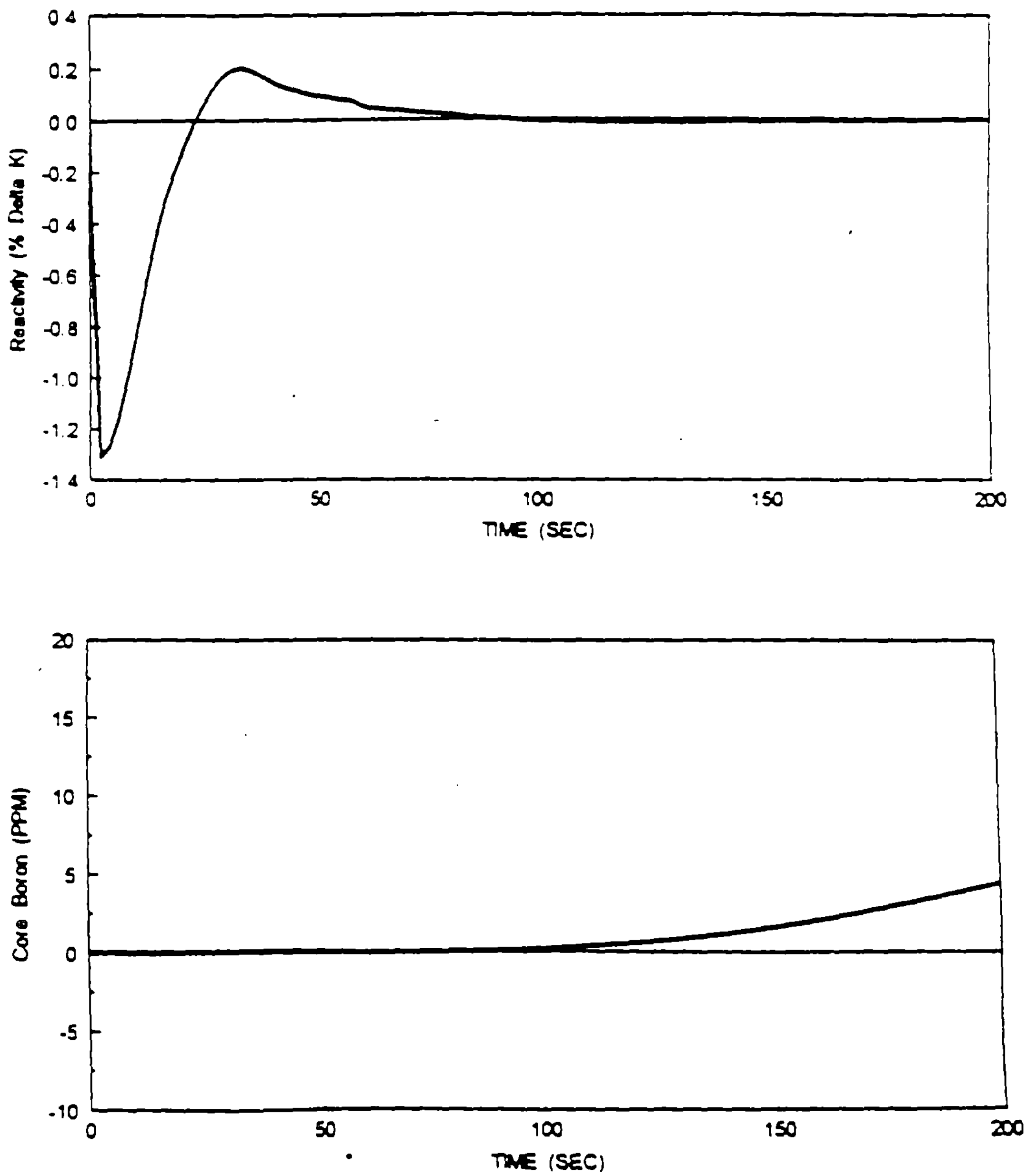


Figure 4.1.13.3-13

Transient Response to Steamline Break
at Exit of Steam Generator with
Sfaety Injection, Without Offsite Power
(Case D)



4.1.14 Spurious Operation of the Safety Injection System at Power (UFSAR 15.2.14)

Accident Description:

The Spurious Operation of the SIS at Power accident occurs as a result of an inadvertent or spurious actuation of the Emergency Core Cooling System (ECCS) which may be caused by either operator error or a false electrical actuating signal. Since the pressurizer water volume increases when the ECCS is inadvertently actuated, operator action is eventually required to terminate the safety injection flow and recover from the event. The Spurious SIS, a Condition II event, has been reviewed for the impact of the proposed Technical Specification changes and is not significantly impacted by those changes.

4.1.15 Single Rod Cluster Control Assembly Withdrawal at Full Power (UFSAR 15.3.5)

Accident Description:

This Condition III event is the unlikely occurrence of a failure which result in continuous withdrawal of a single RCCA, it is not possible in all cases to provide assurance of automatic reactor trip such that core safety limits are not violated. Withdrawal of a single RCCA results in both a positive reactivity insertion and an increase in local power density in the core area "covered" by the RCCA.

Method of Analysis:

Power distributions within the core are calculated using the appropriate computer codes. The peaking factors are then used by THINC to calculate the minimum DNBR for the event. The limiting single RCCA withdrawal was determined to be the worst (most-reactive) rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power. This event was analyzed at BOL and assumes the least negative value for the moderator temperature coefficient. This maximizes the power rise and minimizes the tendency of the increased moderator temperature to flatten the power distribution.

Results:

If the reactor is in the manual control mode, continuous withdrawal of a single RCCA will result in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the failed RCCAs. In terms of the overall system response, this case is similar to those presented in Section 4.1.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCAs, automatic reactor trip may not occur quickly enough to prevent the minimum core DNBR from falling below the limit value. Evaluation of these cases, at the power and coolant conditions at which the OTDT trip would be expected to trip the plant, shows that an upper limit for the number of rods with a DNBR less than the limit value is 5%.

If the reactor is in automatic control mode, withdrawal of an RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as described above. A trip will ultimately ensue, although not quickly enough in all cases to prevent a minimum DNBR in the core less than the limit value.

Conclusions:

In the event of a single RCCA withdrawal, the number of fuel rods experiencing DNBR was less than the limit value, which is 5% of the total fuel rods in the core. Consequently, acceptable results were obtained when assuming MRP implementation.

4.1.16 Major Rupture of a Main Feedwater Line (UFSAR 15.4.3)

Accident Description:

A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. This event is a considered to be a Condition IV event. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of AFW to the affected steam generator. A break upstream of the feedwater line check valve would affect the nuclear steam supply system only as a loss of feedwater. (This case is covered by the evaluation in Section 4.1.8).

Depending on the size of the break and the plant operating conditions at the time of the break, the break could cause either a cooldown or a heatup of the RCS. RCS cooldown is caused by excessive energy discharge through the break. Potential cooldown resulting from a secondary pipe rupture is evaluated in Section 4.1.13. In this section, only the RCS heatup effects are evaluated.

Method of Analysis:

The feedwater line break cases are analyzed with and without offsite power available. The breaks analyzed assume a double ended rupture of the feedwater piping at full power. Major assumptions are as follows:

a. The plant is initially at 102% of NSSS power.

b. Conservative initial RCS temperature and pressurizer pressure values are assumed.

c. Main feedwater flow to all of the steam generators is assumed to be lost at the time the break occurs.

d. Reactor trip is assumed to be initiated when the low-low level trip setpoint is reached in the faulted steam generator.

e. Conservative core residual heat generation is assumed based on long term operation at the initial power level preceding the trip.

No reactor control systems are assumed to function, except for the pressurizer PORVs. The RPS is required to function following a feedwater line rupture as analyzed here. No single active failure prevents operation of this system. The only ESFs assumed to function are the AFW system and the SIS.

Following the trip of the reactor coolant pumps for the feedwater line rupture without offsite power, there is a flow coastdown until flow in the loops reaches the natural circulation value. The natural circulation capability of the RCS is shown in Section 5.1.9 for the loss of AC power transient. Its capability is sufficient to remove decay heat following reactor trip. Pump coastdown characteristics are demonstrated in Section 4.1.5.

Results:

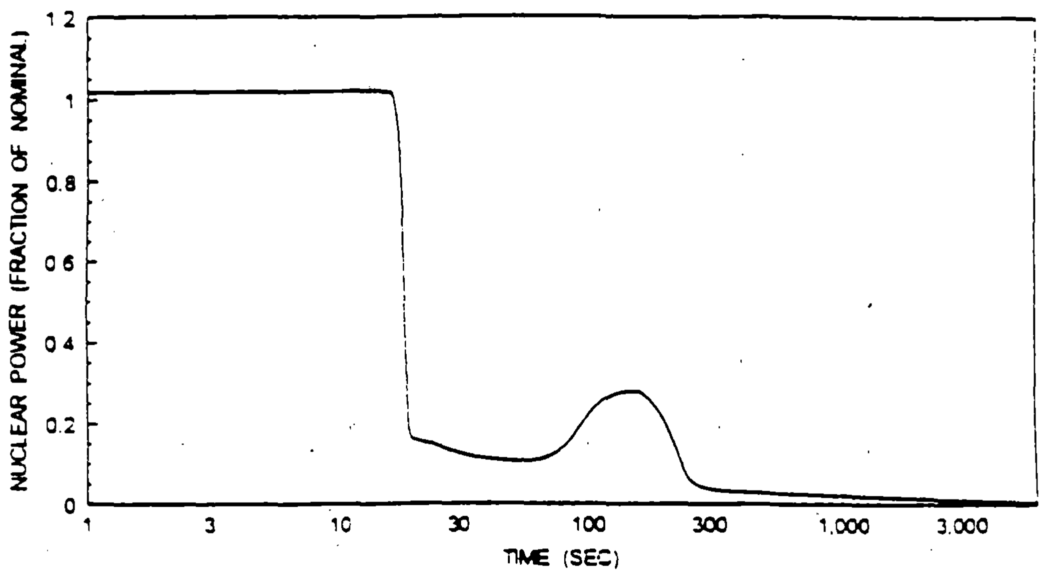
Calculated plant parameters following a major feedwater line rupture are shown in Figures 4.1.16-1 through 4.1.16-6. Results for the limiting case which assumes available offsite power are shown in Figures 4.1.16-1 through 4.1.16-3. Results for the case without offsite power are presented in Figures 4.1.16-4 through 4.1.16-6.

Conclusions:

The results of the analyses show that for the postulated feedwater line rupture, the assumed AFW system capacity is adequate to remove decay heat, to prevent overpressurization of the RCS and MSS, and to prevent uncovering the reactor core (demonstrated by no bulk boiling in the RCS) when assuming implementation of the Margin Recovery Program.

Figure 4.1.16-1

Major Rupture of a Main Feedwater Pipe
With Offsite Power



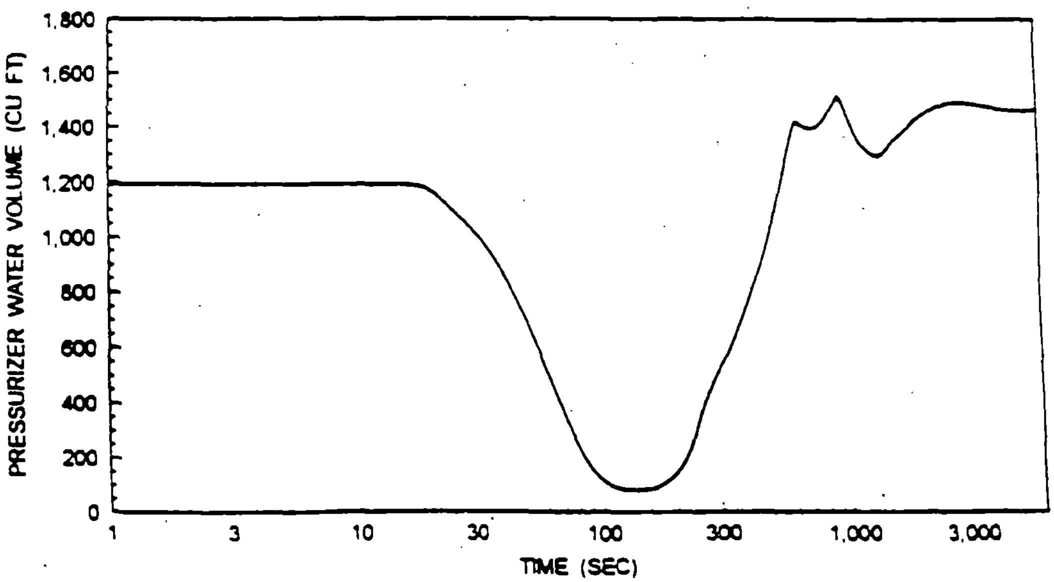
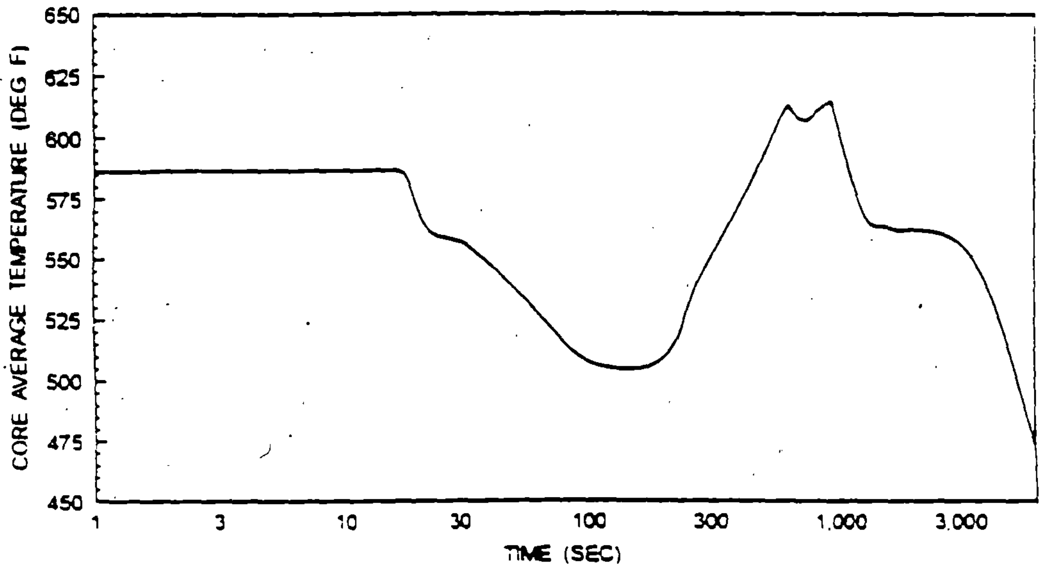


Figure 4.1.16-2

Major Rupture of a Main Feedwater Pipe
With Offsite Power



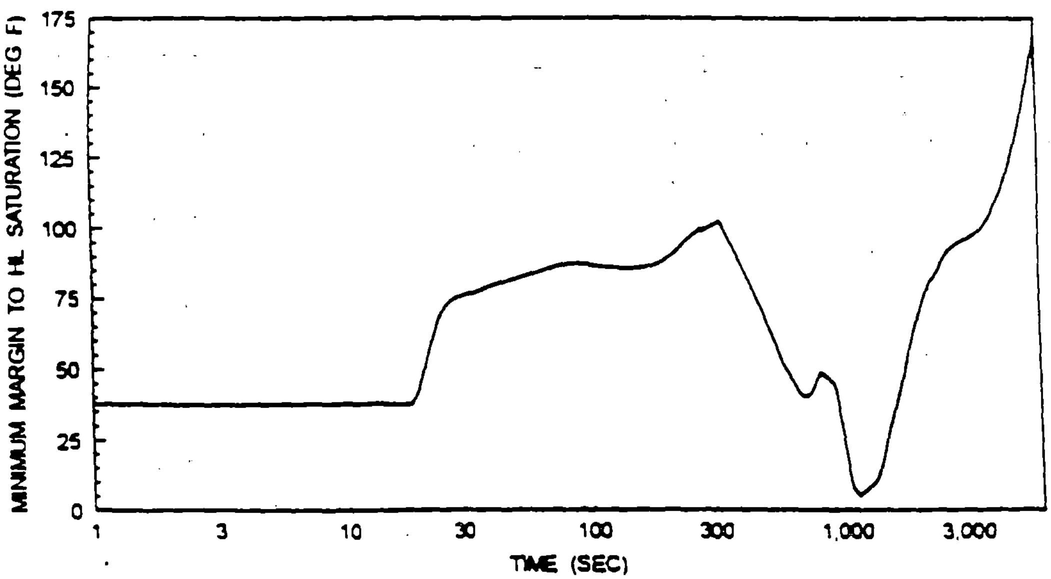


Figure 4.1.16-3

Major Rupture of a Main Feedwater Pipe
With Offsite Power

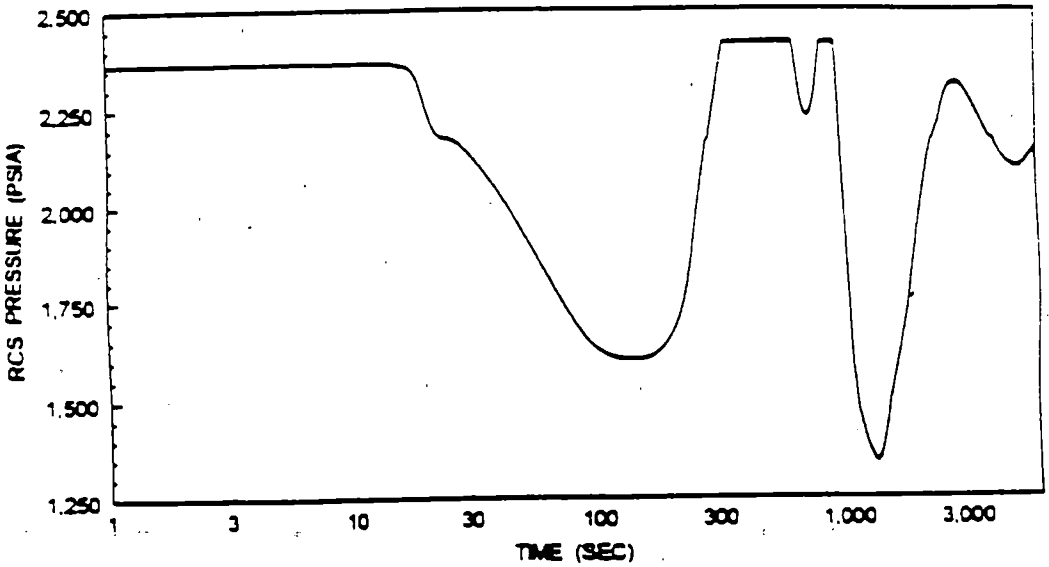
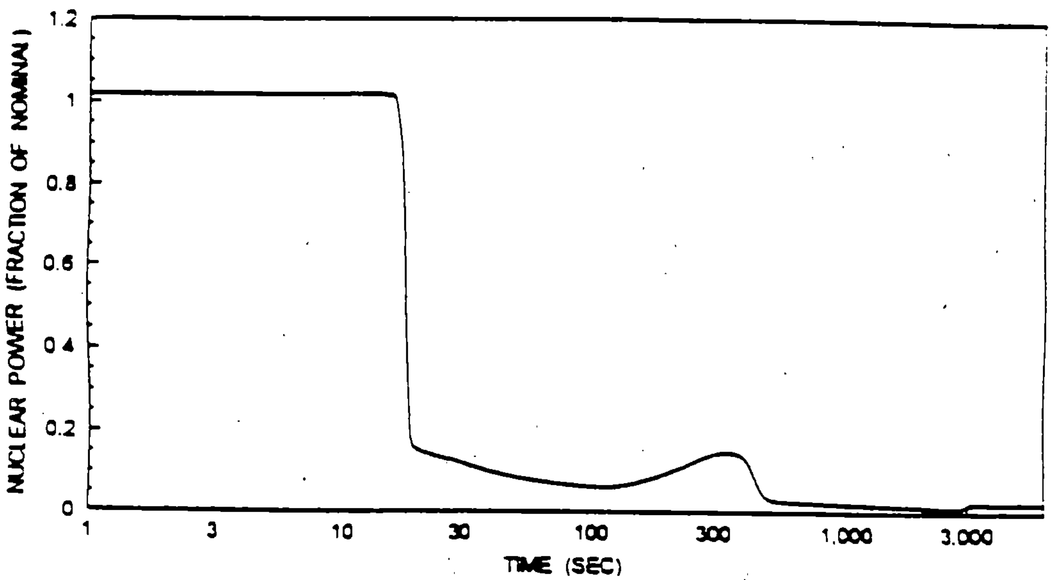


Figure 4.1.16-4

Major Rupture of a Main Feedwater Pipe
Without Offsite Power



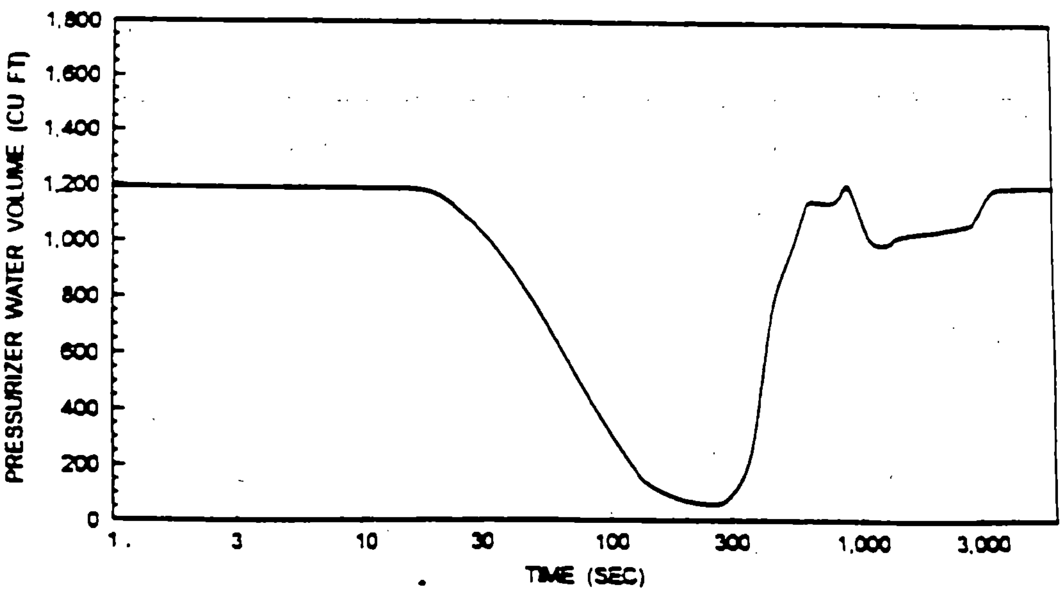


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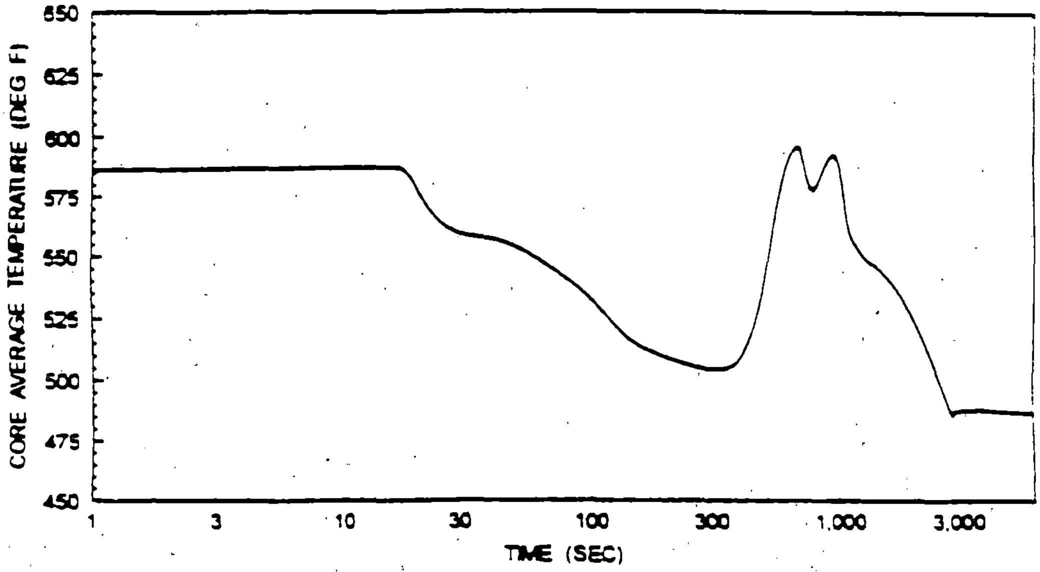
Figure 4.1.16-5

Major Rupture of a Main Feedwater Pipe
Without Offsite Power



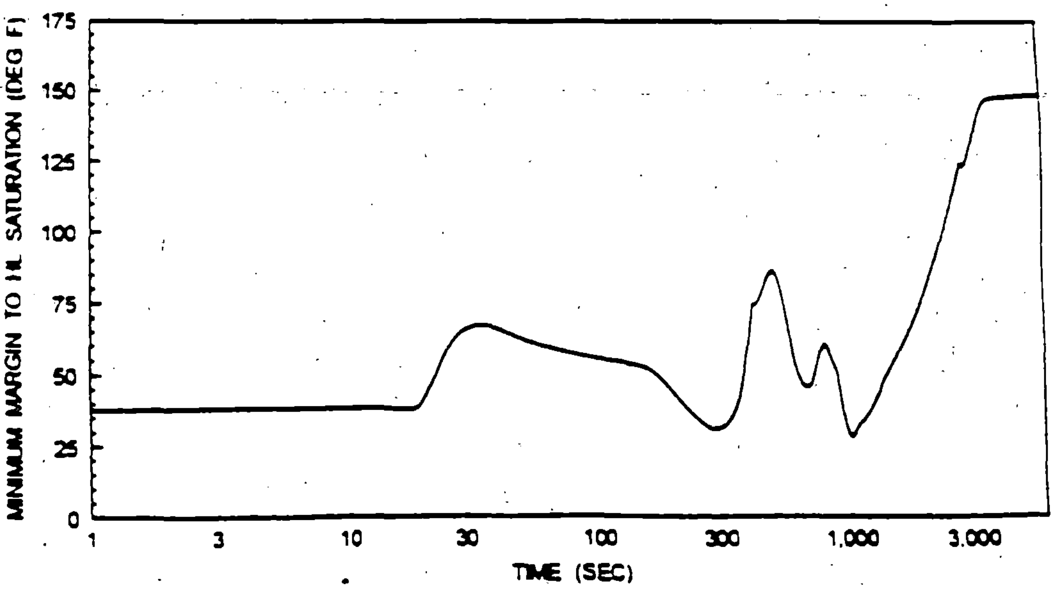
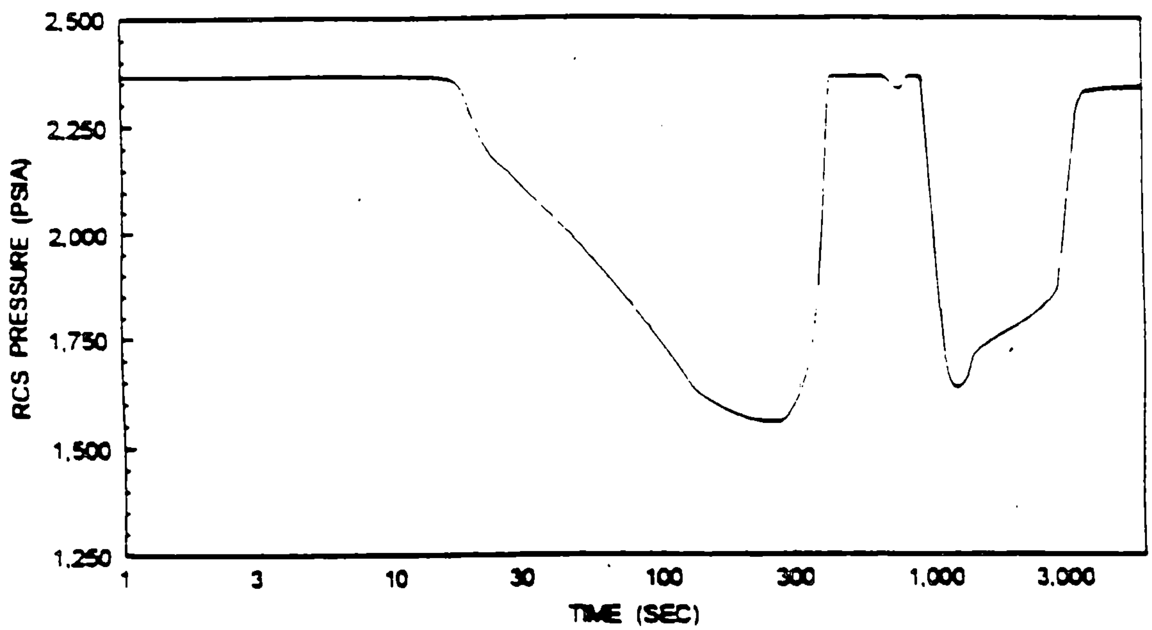


Figure 4.1.16-6

Major Rupture of a Main Feedwater Pipe
Without Offsite Power



4.1.17 Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) (UFSAR 15.4.7)

Accident Description:

This accident is the result of the assumed mechanical failure of a control rod drive mechanism pressure housing such that the RCS pressure would eject the control rod cluster and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Should a RCCA Ejection accident occur, the following automatic features of the RPS are available to terminate the transient:

a. the source range high neutron flux reactor trip

b. the intermediate-range high neutron flux reactor trip

c. the power-range high neutron flux reactor trip (low setting)

d. the power-range high neutron flux reactor trip (high setting)

e. the high nuclear flux rate reactor trip

Due to the extremely low probability of an RCCA Ejection accident, this event is classified as an ANS Condition IV event (Limiting Fault). The following acceptance criteria are applied to the RCCA Ejection accident:

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

b. Peak RCS pressure must remain below that which would cause the stresses in the RCS to exceed the Faulted Condition stress limits.

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

Method of Analysis:

The calculation is divided into two parts: a neutron kinetic analysis and a hot spot fuel heat transfer analysis. The spatial neutron kinetics code TWINKLE is used to calculate the core nuclear power including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average core nuclear power is multiplied by the post-ejection hot channel factor, and the fuel enthalpy and temperature transients at the hot spot are calculated with the detailed fuel and cladding transient heat transfer computer code, FACTRAN.

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

a. The RTDP is not used for the RCCA Ejection analysis. Instead, the STDP (maximum uncertainties in initial conditions) is employed.

b. Minimum values of the delayed neutron fraction are assumed.

c. Least negative values of the Doppler power defect are assumed.

d. Maximum values of ejected RCCA worth and post-ejection total hot channel factors are assumed for all cases considered. No credit is taken for the flux flattening effects of reactivity feedback.

Results:

Figures 4.1.17-1 through 4.1.17-4 illustrate the nuclear power and hot spot fuel rod thermal transients following RCCA Ejection. A time sequence of events is provided in Table 4.1.17-1. For all cases, the maximum fuel pellet enthalpy remained below 200 cal/g. For the Full Power cases, the peak hot spot fuel centerline temperature reached the fuel melting temperature (4,900°F at BOL and 4,800°F at EOL), however melting was restricted to less than 10% of the pellet. For the Zero Power cases, the peak hot spot fuel centerline temperature remained below the fuel melting temperature at all times.

Conclusions:

Even on a conservative basis, the analysis indicates that the fuel thermal limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant, gross lattice distortions, or severe shock waves that could result in an uncoolable core geometry. The upper limit to the number of rods-in-DNB is 10%, which will not result in fission product releases in excess of that associated with the requirements of 10 CFR 100.

Table 4.1.17-1

Sequence of Events
RCCA Ejection

Beginning of Cycle	Full Power	Zero Power
Catastrophic Control Rod Drive	0.0	0.0
Mechanism Housing Failure Occurs		
RCCA is fully ejected from core	0.1	0.1
High nuclear flux reactor trip setpoint reached	0.05	0.25
Peak nuclear power occurs	0.13	0.30
Rod motion begins	0.55	0.75
Maximum fuel pellet enthalpy occurs	2.36	2.6
Peak clad temperature occurs	2.48	2.55
Maximum fuel melt occurs	2.82	N/A

Table 4.1.17-1 (Continued)

Sequence of Events
RCCA Ejection

End of Cycle	Full Power	Zero Power
Catastrophic Control Rod Drive	0.0	0.0
Mechanism Housing Failure Occurs RCCA is fully ejected from core	0.1	0.1
High nuclear flux reactor trip setpoint reached	0.04	0.17
Peak nuclear power occurs	0.13	0.20
Rod motion begins	0.54	0.67
Maximum fuel pellet enthalpy occurs	2.42	1.98
Peak clad temperature occurs	2.50	1.79
Maximum fuel melt occurs	2.65	N/A

Figure 4.1.17-1

RCCA Ejection Accident From Full Power
Beginning of Cycle

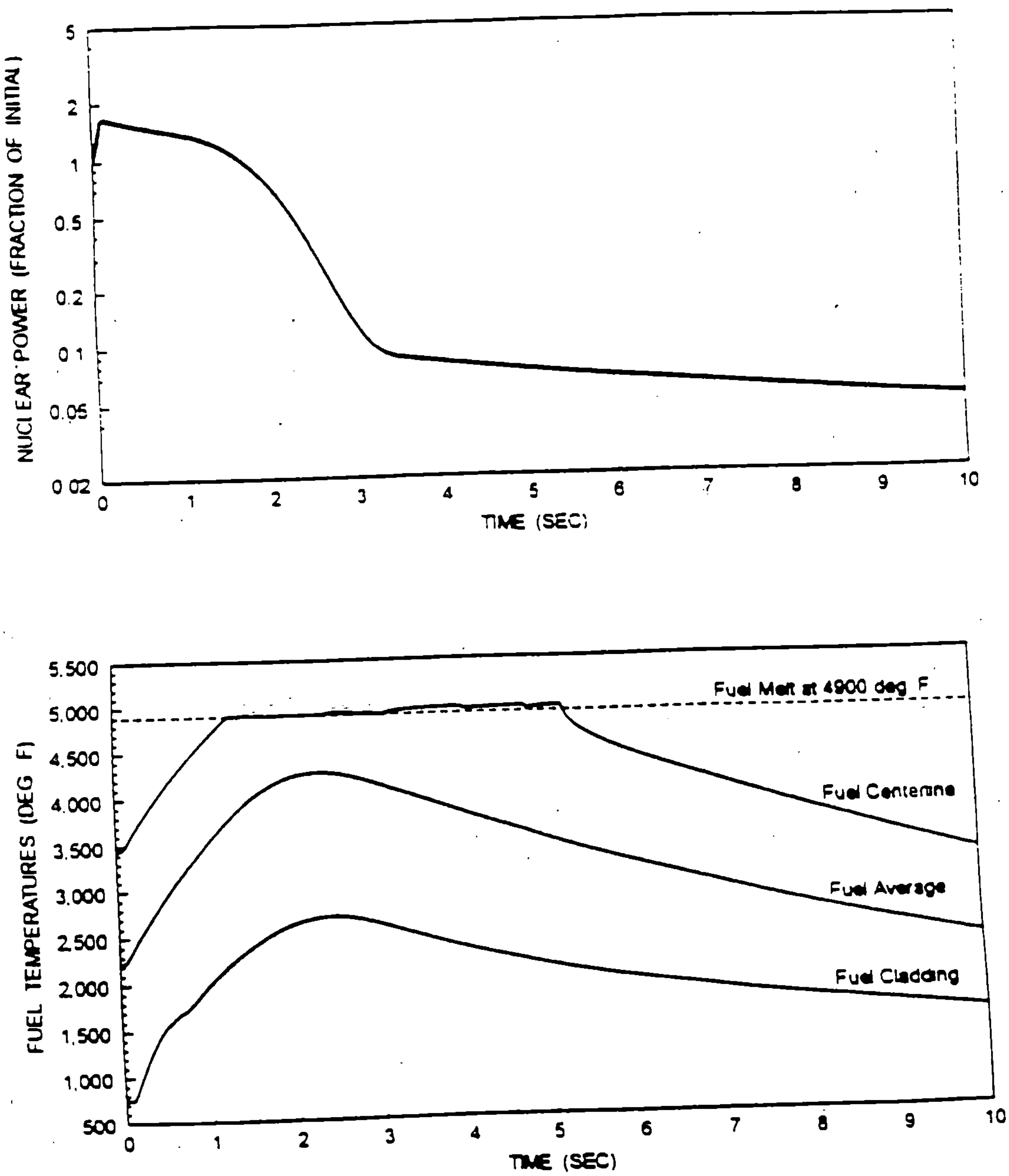


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Figure 4.1.17-2

RCCA Ejection Accident From Zero Power
Beginning of Cycle

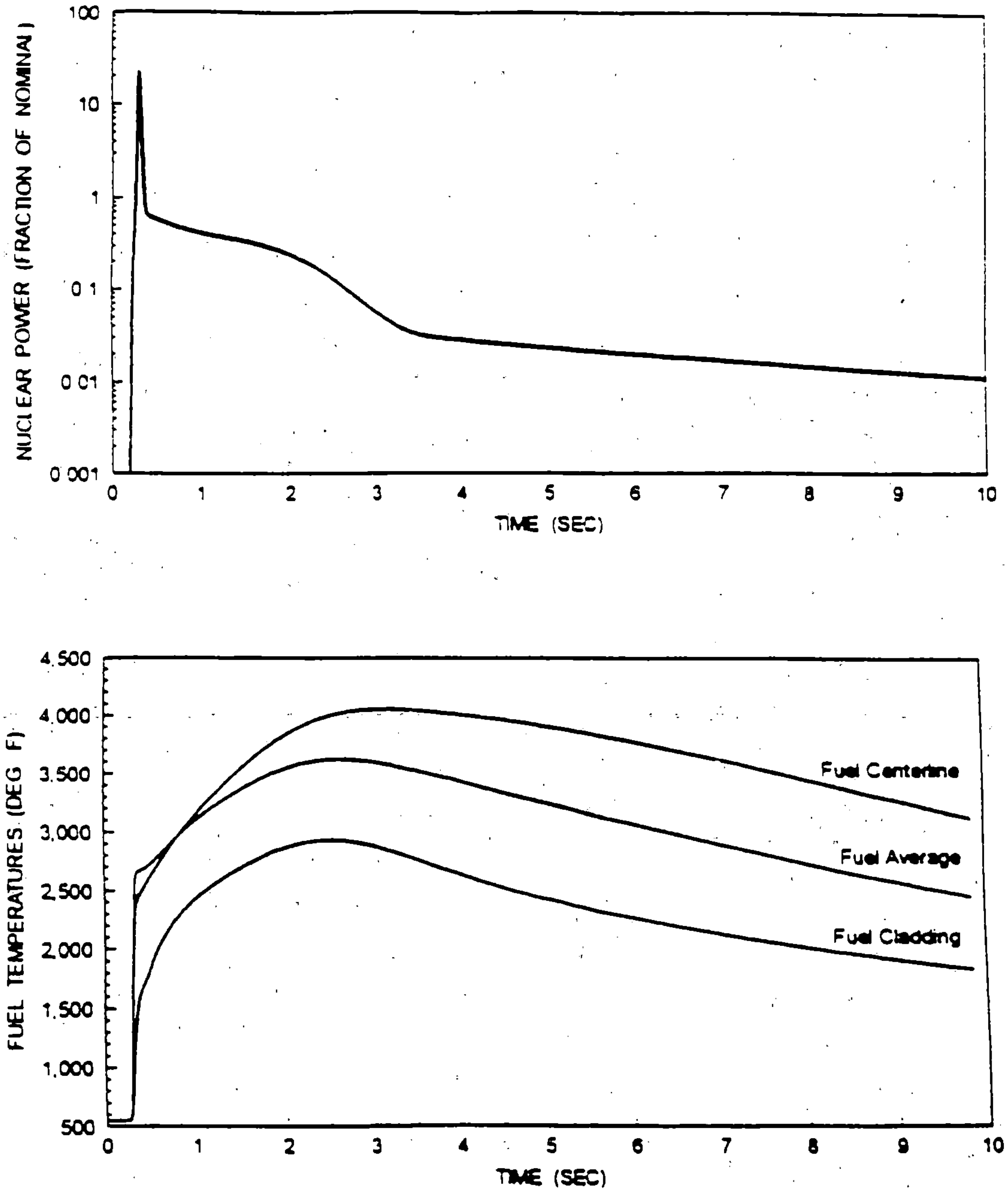
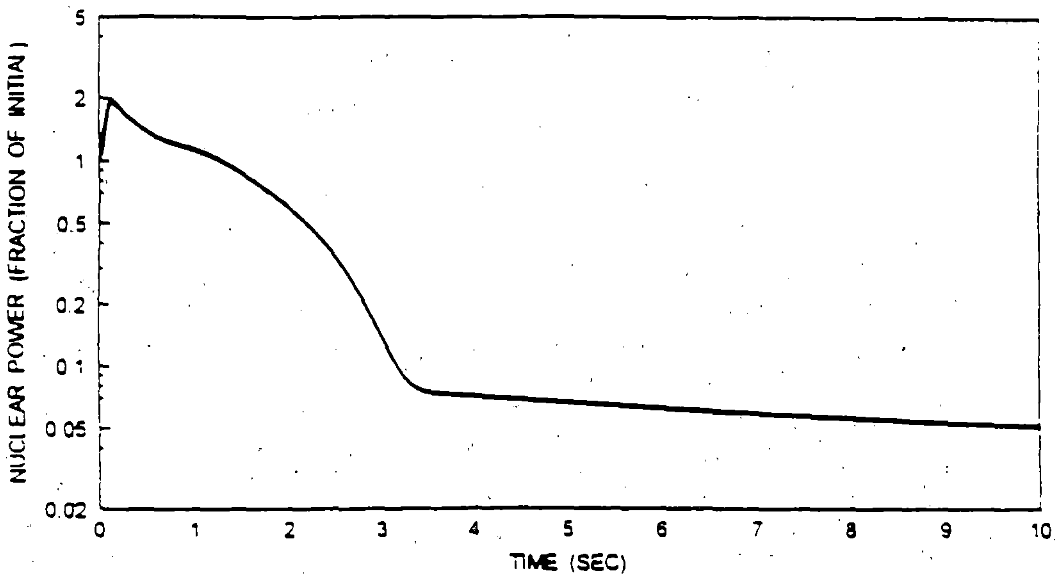


Figure 4.1.17-3

RCCA Ejection Accident From Full Power
End of Cycle



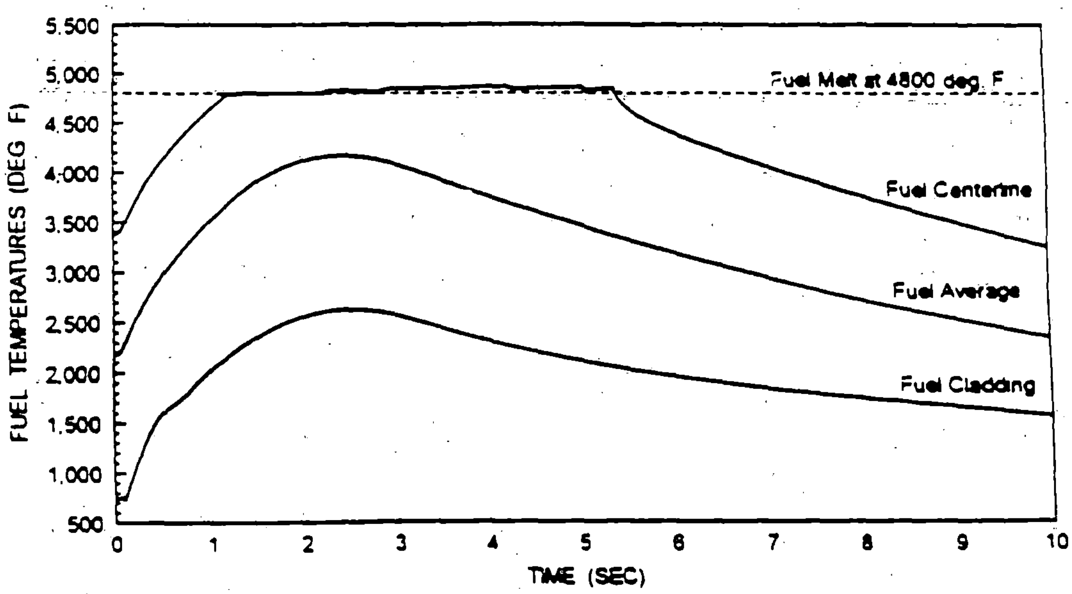


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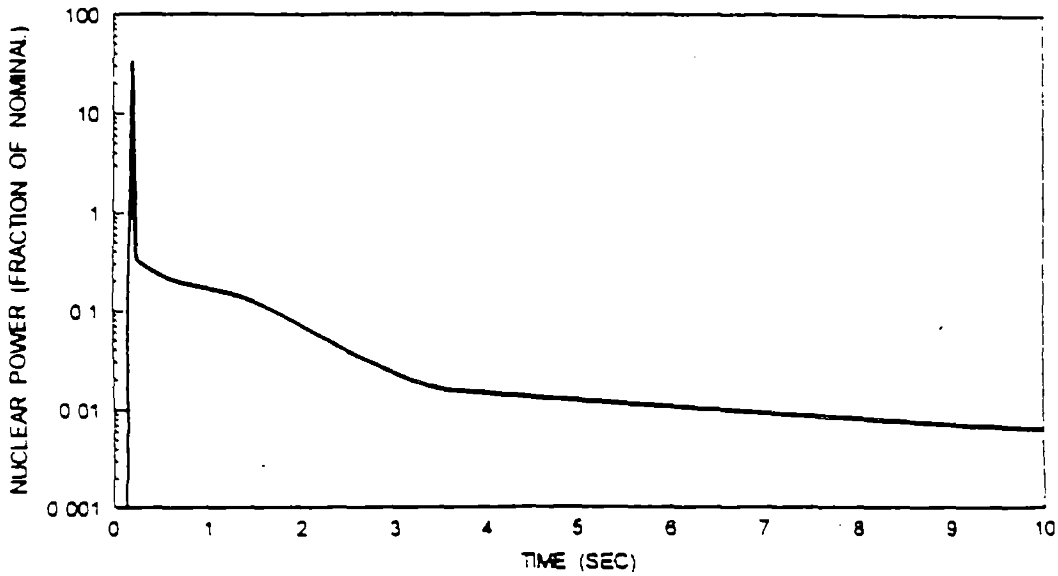
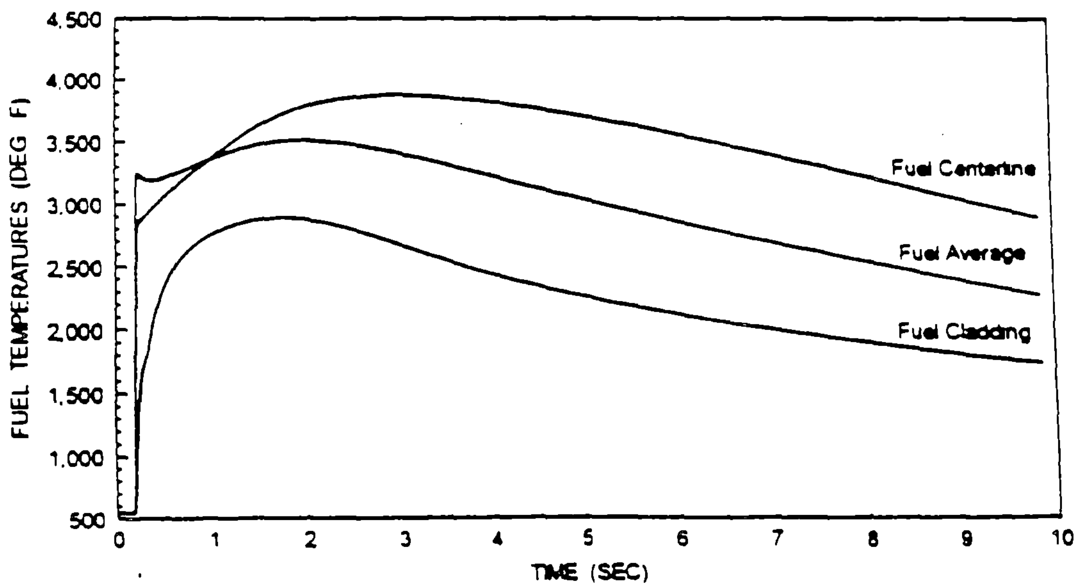
Figure 4.1.17-4

RCCA Ejection Accident From Zero Power
End of Cycle





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4.1.18 Mass and Energy Releases to Containment Following a Steamline Rupture (UFSAR 15.4.8.2)

Accident Description:

Steamline ruptures occurring inside a reactor containment structure may result in significant releases of high energy fluid to the containment environment. These mass and energy releases inside containment can result in increased containment temperature and pressure. Thus, it is demonstrated that the containment pressure and temperature conditions resulting from steamline ruptures remain acceptable given the Technical Specification changes associated with the Margin Recovery Program.

The safety features which provide the necessary protection to limit the mass and energy releases to containment are reactor trip, safety injection, feedline isolation, and steamline isolation. Reactor trip may be provided during a steamline break from OPDT, high neutron flux, safety injection (from any source), low pressurizer pressure, or high containment pressure. A safety injection signal (which will also isolate main feedwater) can be generated on any one of the following functions:

a. Low Steamline Pressure with High Steamline Flow

b. Low-Low RCS T_{avg} with High Steamline Flow

c. High Steamline Differential Pressure

d. Low Pressurizer Pressure

e. High Containment Pressure

Steamline isolation can be generated on any one of the following functions:

a. Low Steamline Pressure with High Steamline Flow

b. Low-Low RCS T_{avg} with High Steamline Flow

c. High-High Containment Pressure

Method of Analysis:

A complete analysis of main steamline breaks inside containment has been performed using the LOFTRAN code and the Westinghouse containment computer code, COCO. All blowdown calculations with the LOFTRAN code were done assuming the RCPs were running (i.e., offsite power available), because this increases the primary to secondary heat transfer. Although this assumption is inconsistent with the delay times assumed in containment fan cooler and spray initiations, where loss of offsite power is assumed, the combined effect of these assumptions provides extra conservatism in the calculated containment conditions.

Several failures can be postulated which would impair the performance of various steamline break protection systems and therefore would change the net energy releases from a ruptured line. Four different single failures were considered for each break condition resulting in a limiting transient. These were: 1) failure of a main feed regulating valve; 2) failure of a main steam isolation valve; 3) failure of the AFW runout protection equipment; and 4) failure of a containment safeguards train. Details about each of the single failures and their major assumptions follow.

Feedwater Flow

There are two valves in each main feedwater line which serve to isolate main feedwater flow following a steamline break, the main feedwater regulator valve and the feedwater isolation valve. Additionally, the main feedwater pumps receive a trip signal following a steamline break. Thus, the worst failure in this system is a failure of the main feedwater regulator valve to close. This failure results in additional time during which feedwater from the Condensate Feed System may be added to the faulted steam generator. Also, since the feedwater isolation valve is upstream of the regulator valve, failure of the regulator valve results in additional feedline volume which is not isolated from the faulted steam generator. Thus, water in this portion of the lines can flash and enter into the faulted steam generator.

Main Steam Isolation

Since all main steam isolation valves are assumed to isolate rapidly, failure of one of these valves affects only the volume of the main steam and turbine steam piping which cannot be isolated from the pipe rupture.

Steam contained in the unisolatable portions of the steamlines and turbine plant was considered in the containment analyses in two ways. For the large double-ended ruptures (DER), steam in the unisolatable steamlines is released to containment as part of the reverse flow. The flow is held constant at this rate for a time period sufficient to purge the entire unisolated portion of the steamlines. Enthalpy of the flow is also held constant at the initial steam enthalpy. Following this period of constant flow representing purging of the steamlines, flow from the intact steam generators, as calculated by LOFTRAN, is added to the containment and continues until steamline isolation is complete.

When considering split ruptures, steam in the steamlines is included in the analysis by adding the total mass in the lines to the initial mass of steam in the faulted steam generator. This is necessary because, unlike DERs, the total break area of a split is unchanged by steamline isolation; only the source of the blowdown effluent is changed. Thus, steam flow from the piping in the intact loops is indistinguishable from steam leaving the faulted steam generator. However, by adding the water mass in the piping to the faulted steam generator mass and by having dry steam blowdowns, the steamline inventory is included in the total blowdown.

Auxiliary Feedwater Flow

The mass addition to the faulted steam generator from the AFW System was conservatively determined by using the following assumptions:

a. The entire AFW System was assumed to be actuated at the time of the break and instantaneously pumping at its maximum capacity.

b. The flows to the faulted steam generator were conservatively modelled based on the faulted and intact steam generator pressures. The effect of flow limiting devices was considered.

c. The flow to the faulted steam generator from the AFW system was assumed to exist from the time of rupture until realignment of the system was complete.

d. The failure of the AFW runout control was considered as one of the single failures. Failure of runout control results in significantly higher AFW flow to the faulted steam generator and lower flows to the intact steam generators.

The AFW System is assumed to be manually realigned by the operator 10 minutes into the transient. Therefore, the analysis assumes a conservatively high AFW flow to the depressurizing faulted steam generator for a full 10 minutes.

Heat Sinks

The worst effect of a containment safeguards failure is the loss of a spray pump which reduces containment spray flow by 50%. In all analyses, conservative times are assumed for initiation of containment sprays and fan coolers. These times are based on the assumption of a loss of offsite power, and the delays are consistent with Tech Spec limits. The delay time for spray delivery includes the time required for the spray pumps to reach full speed and the time required to fill the spray headers and piping.

The saturation temperature corresponding to the partial pressure of the vapor in the containment is conservatively assumed for the temperature in the calculation of condensing heat transfer to the passive heat sinks. This temperature is also conservatively assumed for the calculation of heat removal by the containment fan coolers. The conservatively assumed fan cooler heat removal rate as a function of containment temperature is presented on Figure 4.1.18-1.

Results:

A total of 80 different blowdowns covering four power levels and fourteen different break sizes were evaluated. The fourteen break sizes considered at each power level were a 4.6 ft^2 full DER with entrainment, a 1.4 ft^2 full DER with entrainment, a small DER with an area just larger than that at which entrainment occurs, a small DER with an area just smaller than that at which entrainment occurs, and the largest split rupture that will neither result in generation of a Steamline Isolation signal from the primary plant protection system equipment, nor result in entrainment. In the analysis of the third, fourth, and fifth (split) break, reactor trip, feedline isolation, and steamline isolation are generated by high containment pressure signals. The containment responses resulting from the mass and energy releases are plotted in Figures 4.1.18-2 through 4.1.18-7.

Figures 4.1.18-2 and 4.1.18-3 display the pressure and temperature transients for the large DER case producing the highest containment pressure of those analyzed. This case is the 4.6 ft^2 DER at 30% power with a feedwater control (regulator) valve failure. Shown in Figures 4.1.18-4 and 4.1.18-5 are the pressure and temperature transients for the small break case producing the highest containment pressure of the small breaks analyzed. This case is the 0.944 split break at 30% power with a containment safeguards train failure. Of all cases analyzed, the highest containment atmosphere steam temperature was produced by the 0.6 ft^2 small DER with entrainment at 102% power with a main steam isolation valve failure. The pressure and temperature transients for this case are shown in Figures 4.1.18-6 and 4.1.18-7, respectively.

Conclusions:

The results of the cases analyzed for Mass and Energy Releases to Containment event demonstrate that the containment design pressure limit is not exceeded given the MRP implementation. In addition, the containment temperatures are acceptable with respect to Equipment Qualification.

Figure 4.1.18-1

Fan Cooler Heat Removal Rate

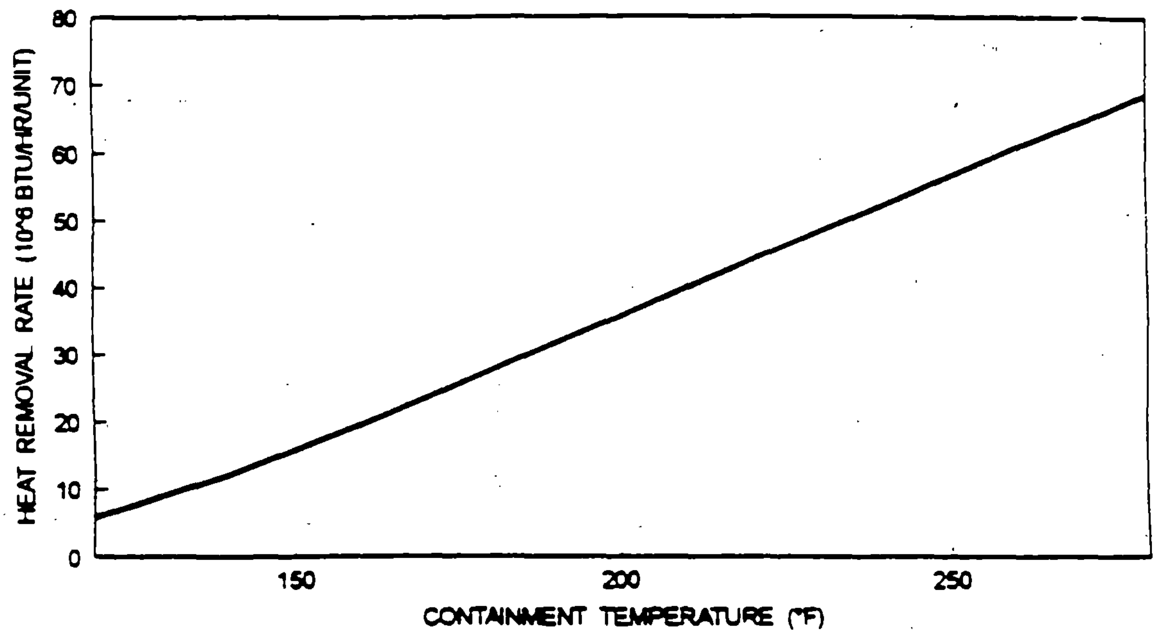

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Figure 4.1.18-2

MSLB Containment Pressure Transient
4.6 Ft^2 DER - 30% Power
Feedwater Control Valve Failure

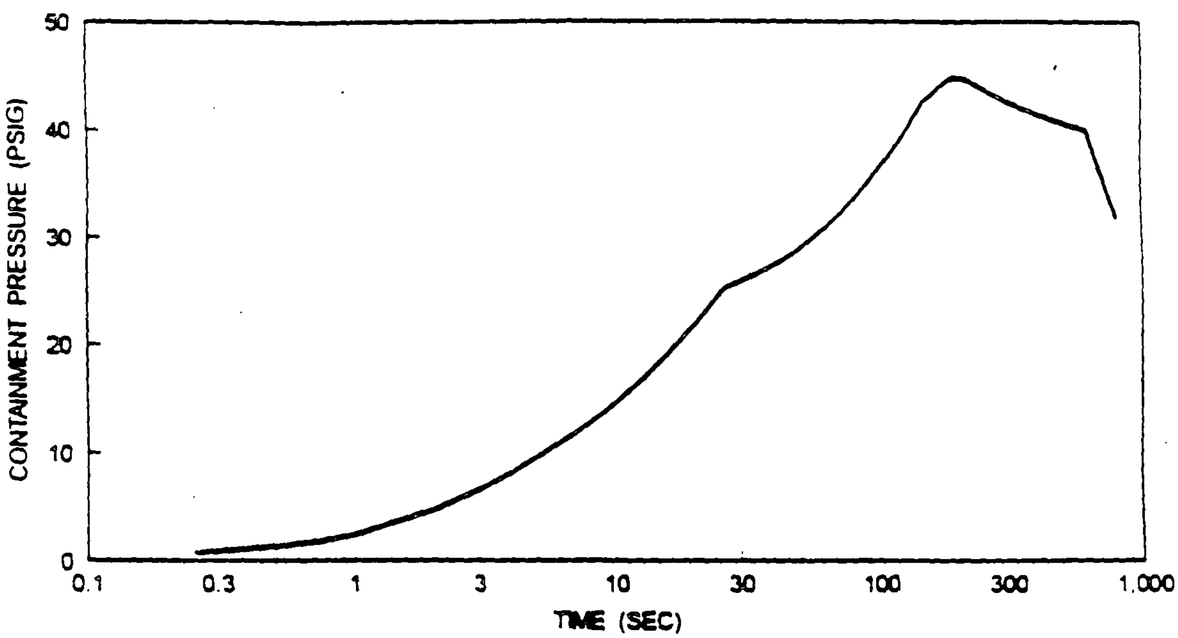

Figure 4.1.18-3

MSLB Containment Temperature Transient
4.6 Ft^2 DER - 30% Power
Feedwater Control Valve Failure

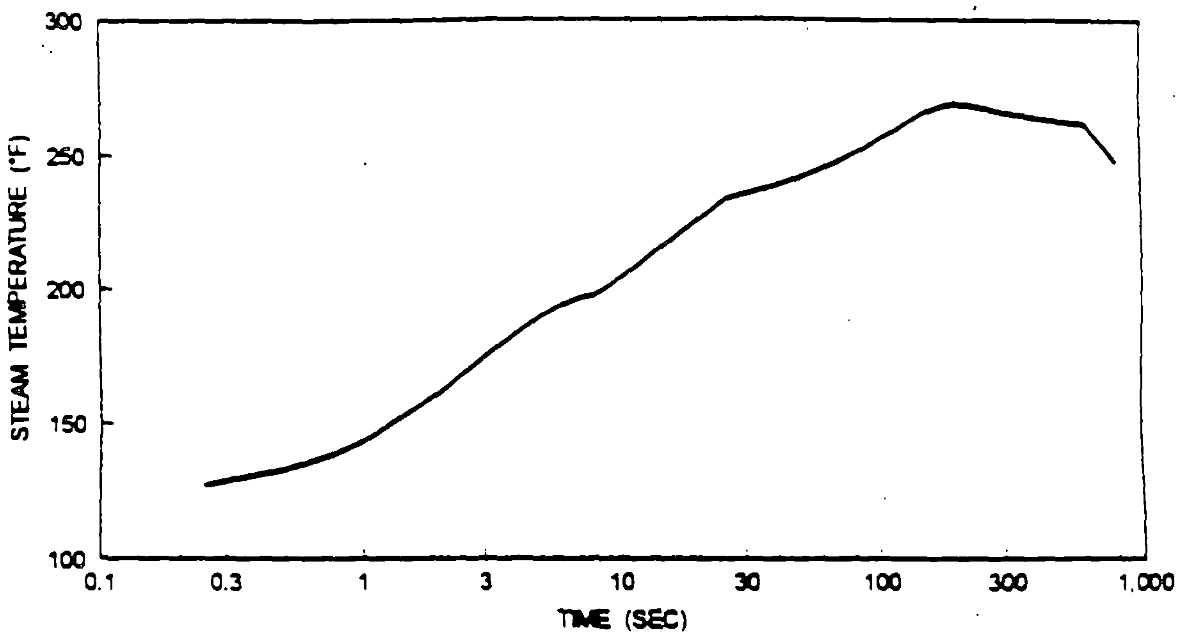

Figure 4.1.18-4

MSLB Containment Pressure Transient
0.944 Ft^2 Split Break - 30% Power
Containment Safeguards Train Failure

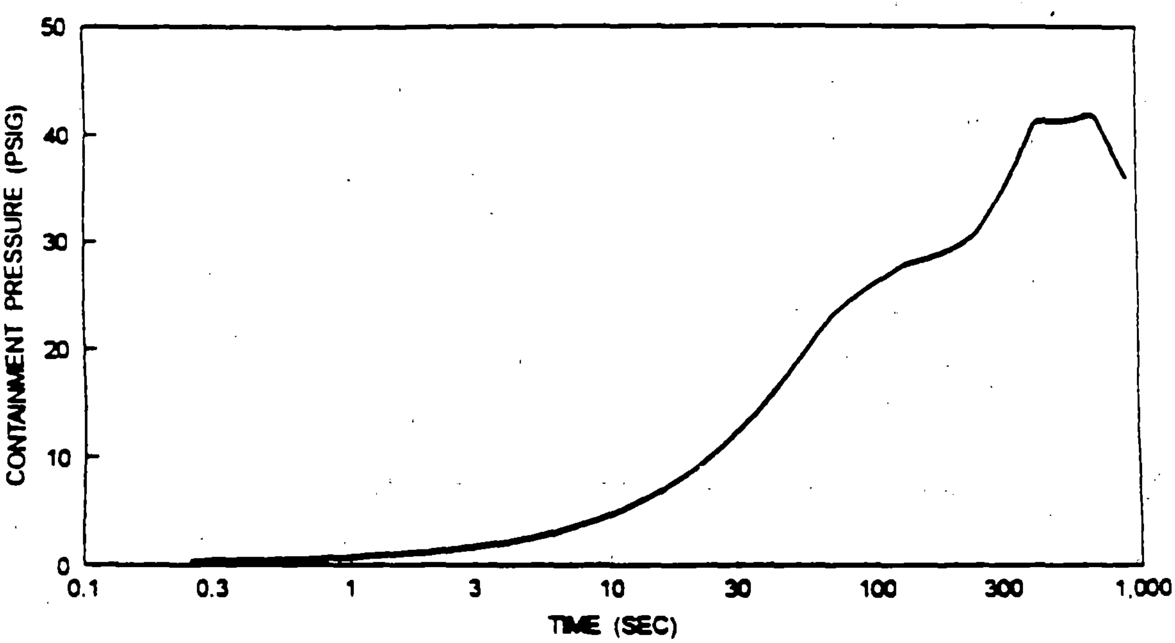

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Figure 4.1.18-5

MSLB Containment Temperature Transient
0.944 Ft2 Split Break - 30% Power
Containment Safeguards Train Failure

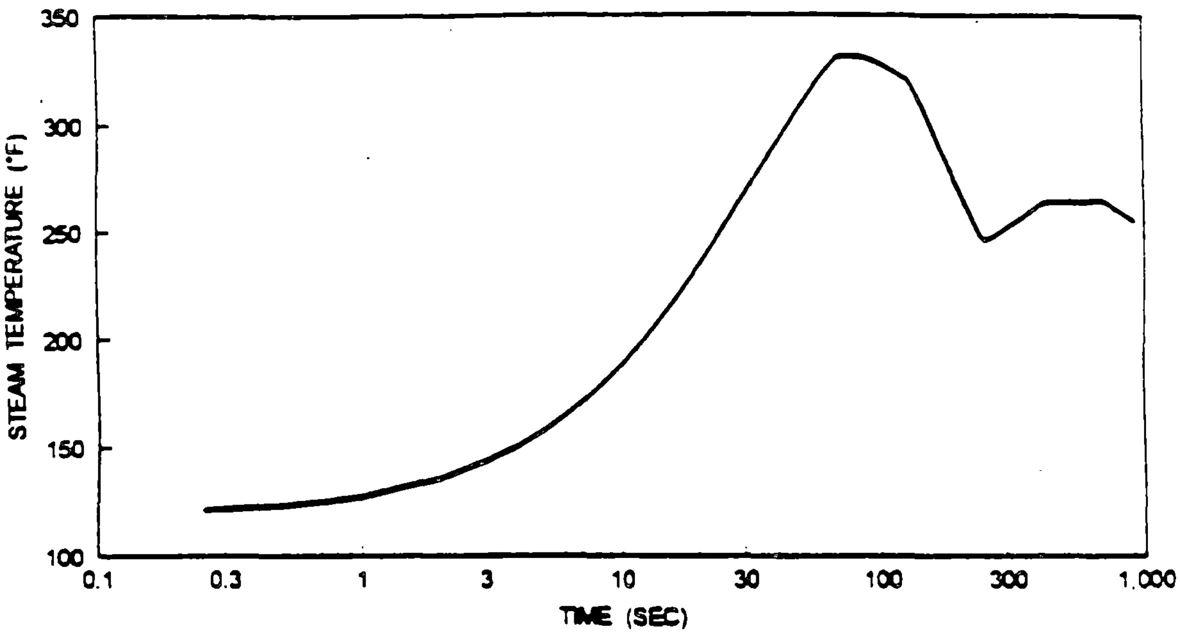

Figure 4.1.18-6

MSLB Containment Pressure Transient
0.6 Ft^2 Small DER - 102% Power
Main Steam Isolation Valve Failure

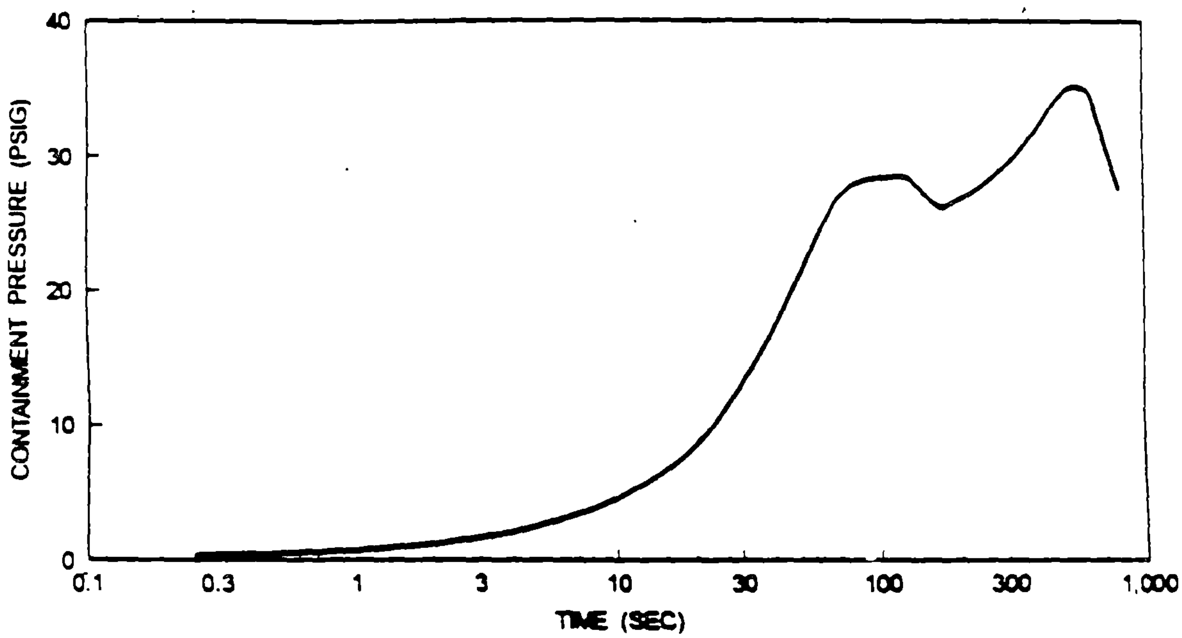

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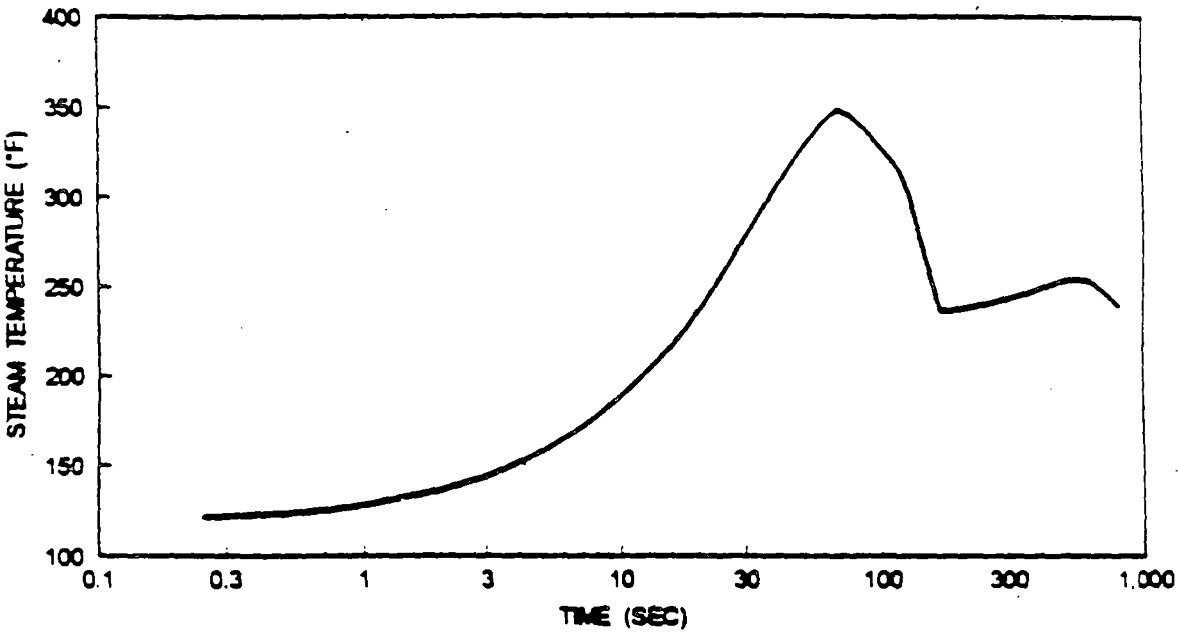
Figure 4.1.18-7

MSLB Containment Temperature Transient
0.6 Ft^2 Small DER - 102% Power
Main Steam Isolation Valve Failure



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4.2 LOCA Accidents

This section summarizes the LOCA related reanalyses and evaluations performed for the Margin Recovery Program (MRP).

4.2.1 Large Break LOCA (UFSAR Section 15.4.1)

4.2.1.1 Description of Analysis Assumptions

The Large Break Loss-Of-Coolant Accident (LOCA) analysis for Salem Unit 1 and 2 applicable for the MRP was performed using a modified version of the NRC approved 1981 Evaluation Model with BASH. The important analysis assumptions include: licensed core power of 3411 MWt, 25% uniform steam generator tube plugging (SGTP), T_{avg} operating window of 566°F to 580°F, thermal design flow of 82,500 gpm/loop, maximum peaking factor $F_Q(Z)$ of 2.40, and a hot channel enthalpy rise factor $F^N_{\Delta H}$ of 1.65.

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. The spectrum was performed assuming T_{avg} was at the high end of the operating window, minimum safeguards safety injection flow was available. The 0.4 Moody discharge coefficient was determined to be the limiting discharge coefficient. Cases assuming T_{avg} operation at the low end of the operating window and maximum safeguards safety injection flow were then performed at the limiting Moody discharge coefficient. These cases confirmed that operation at the high end of the T_{avg} operating window and minimum safeguards safety injection flow was limiting.

4.2.1.2 Methods of Analysis

The Large Break LOCA analysis was performed using the 1981 Evaluation Model with BASH methodology and computer codes. These documents describe the major phenomena modeled, the interface between the computer codes, and the features of the codes which ensure compliance with the requirements defined in Appendix K to 10 CFR 50.

The SATAN-VI, WREFLOOD, COCO, and LOCBART codes are also used in the LOCA analysis. These codes are used to assess the core heat transfer characteristics and to determine if the core remains amenable to cooling throughout the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD and BASH computer codes are used to calculate this transient during the refill and reflood phases of the accident. The COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCBART computer code is used to compute the core fluid and heat transfer conditions and the fuel cladding thermal transient of the hot assembly, including the hot rod, during the three phases.

Several additional modifications have been made to the codes used in this analysis. Miscellaneous minor LOCBART error corrections have been made. These include pellet/clad contact and clad thinning models which were included in the updated code version used in this analysis. These errors were deemed to have negligible effect on the transient for this analysis. Various discretionary changes to input/output format and inclusion of code diagnostics are also contained in the LOCBART version used. These changes do not affect the results. The version of the BASH code used was modified to create a plot tape in the standard plotting code format and to correct a problem with a library compatibility which previously prevented code restarts. There are no effects on the calculated results from these changes.

4.2.1.3 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 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 Large Break LOCA analysis for Salem Unit 1 and 2, utilizing the BASH model, resulted in a peak cladding temperature of 2020°F for the limiting break case (C_D = 0.4 under minimum safeguards safety injection flow and maximum operating RCS T_{avg} assumptions). The maximum local metal-water reaction was 6.3 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. The results of this Large Break ECCS analysis have shown that Salem Unit 1 and 2 remains in compliance with the requirements of 10 CFR 50.46.

4.2.2 Small Break LOCA (UFSAR Section 15.3.1)

4.2.2.1 Description of Analysis Assumptions

The Small Break Loss-Of-Coolant Accident (LOCA) analysis for Salem Unit 1 and 2 which incorporated the MRP was formally submitted as WCAP-13657 for NRC review and approval. Per written correspondence dated August 25, 1993, the NRC concluded that the NOTRUMP code can be used to demonstrate compliance with the requirements in 10CFR 50.46 for Salem Units 1 and 2. In addition, it was recognized that the evaluations described in the submitted WCAP were performed in support of the MRP.

4.2.3 Blowdown Reactor Vessel and Loop Forces (UFSAR Section 3.9.1.5)

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. Some of the assumptions which were considered were: 25% uniform SGTP, a T_{avg} operating window of 566°F to 580°F, thermal design flow of 82,500 gpm/loop, maximum peaking factor $F_Q(Z)$ of 2.40, and a hot channel enthalpy rise factor $F^N_{\Delta H}$ of 1.65. The forcing functions were generated primarily to support the reduced thermal design flow and reduced temperature. In order to compensate for the effects of the reduced temperature on the forces, credit for Leak-Before-Break (LBB, WCAP-13659 and 13660 SER dated 5/25/94) was used to allow consideration of branch line breaks only. Therefore the forcing functions generated were based on breaks of the accumulator line and the pressurizer surge line, which have smaller areas than postulated breaks in the main RCS loop piping.

Forces acting on the RCS loop piping as a result of the hypothesized LOCA are not influenced by the changes in the MRP. Thus, the MRP will not result in an increase of the calculated consequences of a hypothesized LOCA on the RCS loop piping. The current FSAR analysis for forces on RCS piping resulting from a hypothesized LOCA are considered to be bounding for the MRP at Salem Units 1 and 2.

4.2.4 Post LOCA Long-Term Cooling, Subcriticality Evaluation (related to UFSAR Section 15.4.1)

The Westinghouse licensing position for satisfying the requirements of 10 CFR Part 50.46 (b)(5) "Long-Term Cooling" is defined in WCAP-8339-NP-A, WCAP-8472-NP-A, and Technical Bulletin NSID-TB-86-08. The 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 LBLOCA, 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.

A reduced thermal design flow of 82,500 gpm/loop, 25% uniform SGTP, maximum peaking factor $F_Q(Z)$ of 2.40, and a hot channel enthalpy rise factor $F^N_{\Delta H}$ of 1.65 have a negligible effect on the sources of borated and non-borated water assumed in the long term cooling calculation. However, the minimum temperature associated with a T_{avg} operating window of 566°F to 580°F will result in a small increase in RCS mass which can impact the source of water with a relatively low boron concentration. Also the minimum available RWST volume including uncertainties which impacts the borated water assumption was considered. These effects were evaluated to determine the impact on the long term cooling capability of the ECCS system, and it was determined that adequate margin currently exists.

4.2.5 Hot Leg Switchover to Prevent Potential Boron Precipitation (UFSAR Sections 15.4.1 and 6.3.2)

Post-LOCA hot leg recirculation time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This recirculation time is dependent on power level, and the RCS, RWST, and accumulator water volumes and boron concentrations. The MRP parameters have a negligible effect on the assumptions for the RCS, RWST, and the accumulators in the hot leg switchover calculation.

However, the T_{avg} operating window of 566°F to 580°F can result in a small reduction in the RCS mass, which can affect the post-LOCA hot leg switchover time.

An evaluation was performed for the reduced RCS mass which showed that there was no significant change to the post-LOCA hot leg switchover time. Therefore, the current hot leg switchover time remains applicable. The cold leg and hot leg recirculation flows are not impacted by the increased peaking factors and margin recovery effects. Therefore Long Term Core cooling is maintained.

4.3 Steam Generator Tube Rupture (UFSAR Section 15.4.4)

A Radiological Dose Analysis has been performed for the Salem Units 1 and 2 Steam Generator Tube Rupture (SGTR) accident. The plant parameters considered for the analysis include the current licensed power level of 3423 MWt with a T_{avg} temperature range from 566°F to 577.9°F and a maximum steam generator tube plugging level of 25%, along with an associated thermal design flow range from 82,500 to 87,300 gpm per loop and a steam pressure range from 677 to 828 psia.

The SGTR accident analysis for Salem was performed to evaluate the radiological consequences due to the event. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The primary-to-secondary break flow following a SGTR results in depressurization of the RCS, which leads to automatic reactor trip and SI actuation. A loss of offsite power is assumed to occur at reactor trip, and the steam generator pressure increases rapidly after reactor trip, resulting in steam release to the atmosphere through the steam generator safety and/or power-operated relief valves. Thus, a SGTR accident results in the transfer of radioactive coolant to the secondary system and subsequent release of activity to the atmosphere. The SGTR analysis in the Salem UFSAR indicates that the offsite radiation doses due to a SGTR will be less than the 10 CFR 100 guidelines.

The major factors that affect the extent of the radioactive release and the resultant offsite radiation doses for a SGTR are the amount of fuel defects (level of reactor coolant contamination), the primary to secondary mass transfer through the ruptured tube, and the steam released from the faulted steam generator to the atmosphere. The SGTR analysis consists of a thermal and hydraulic analysis to determine the primary to secondary break flow and the steam released to the atmosphere, and a radiological consequences analysis to calculate the offsite radiation doses resulting from the event.

4.3.1 SGTR Analysis Assumptions and Methodology

The SGTR thermal and hydraulic analysis was performed using the methodology and assumptions which were used for the Salem UFSAR SGTR analysis. The SGTR accident is a double-ended rupture of a single steam generator tube. The loss of reactor coolant via the ruptured tube leads to RCS depressurization. Reactor trip and safety injection actuation are assumed to occur simultaneously when the pressurizer pressure decreases to the low pressure safety injection setpoint. Following SI actuation, the break flow rate is assumed to equilibrate at the pressure where the safety injection flow rate is balanced by the outgoing break flow rate. This resultant equilibrium break flow rate is assumed to persist until 50 minutes, at which time it is assumed that the operator actions to terminate the break flow are completed. The break flow rates prior to and following reactor trip and SI actuation are based on the pressure differentials for the two periods and are used to determine the total primary-to-secondary break flow.

Since a loss of off-site power is assumed to occur at the time of reactor trip, the condenser steam dump system would not be operable. Thus, the steam generator pressure increases rapidly following reactor trip, and steam is relieved through the steam generator safety and/or power-operated relief valves to dissipate the plant residual heat and the core decay heat.

For the SGTR analysis, it was assumed that the steam generators are maintained at the lowest safety valve pressure following reactor trip and SI actuation. A mass and energy balance for the primary and secondary systems was utilized to calculate the steam released via the safety valves on the faulted and intact steam generators to 32 hours, plant cooldown to the RHR operating conditions is assumed to be performed by releasing steam from the intact steam generators. After 32 hours, the steam release is assumed to be terminated and the RHR System is used to remove decay heat and to continue the cooldown to cold shutdown. A mass and energy balance for the primary and secondary systems was used to calculate the steam releases and feedwater flows for the three intact steam generators.

The results of the SGTR analysis are bounding for operation of Salem Units 1 and 2 within the range of parameters considered.

4.3.2 SGTR Dose Analysis Results

The results of the Salem SGTR analyses for the MRP are summarized below. The results of the SGTR thermal and hydraulic analysis were used to calculate the offsite radiation doses at the site boundary for a 2 hour exposure and at the low population zone for the 32 hour duration of the release.

The offsite doses were calculated for both a pre-accident iodine spike and an accident initiated iodine spike. The calculations are based on the Technical Specification reactor coolant activity and an assumed total primary to secondary leakage rate of 1.0 gpm to all steam generators prior to the accident. This represents a change from the Salem UFSAR since the SGTR offsite doses were previously calculated based on 1% defective fuel without assuming any iodine spiking, and as a function of primary to secondary leakage rates from 0 - 10 gpm. The results of the offsite dose analysis are compared below with the acceptance criteria.

4.3.3 SGTR Analysis Conclusions

The results of the revised analysis are either less than or greater than those of the current UFSAR analysis, depending upon which cases are compared. Because the radiological basis for the current calculation has been upgraded to meet more current NRC requirements (Standard Review Plan, NUREG-0800), the new and old analyses are not directly comparable. In addition the new analysis accommodates a longer operator action time of 50 minutes. For example, the UFSAR presents the offsite doses as a function of primary-to-secondary leak rate, which is varied from 1 to 10 gpm. The current analysis only considers a 1 gpm leak rate, which is equal to the Technical Specification LCO. The UFSAR analysis utilizes primary coolant iodine activity based on 1% fuel defects, while the current analysis is based on pre-accident and accident initiated iodine spikes, which are more conservative than the assumption of 1% defects. Regardless, the calculated doses for an SGTR with both pre-accident and accident initiated iodine spikes are well below the appropriate NRC acceptance criteria.

4.3.4 Radiological Consequences of a Tube Rupture

	V+	Current FSAR	Acceptance Criteria[1]
Thyroid Dose with Pre-Accident Iodine Spike			
Site Boundary (0-2 hours)	23.6	6.62	10 CFR 100
Low Population Zone (0-32 hours)	2.2	0.18[**,3]	10 CFR 100
Thyroid Dose with Accident Initiated Spike			
Site Boundary (0-2 hours)	3.6	n/a	30 rem
Low Population Zone (0-32 hours)	0.6	n/a	30 rem
Whole - body y			10 CFR 100
Site Boundary (0-2 hours)	1.2E-1	1.9E-1	10 CFR 100
Low Population Zone (0-32 hours)	1.0E-2	1.3E-2[***]	10 CFR 100

4.4 Containment Analysis

Containment Integrity Analyses are performed to ensure that the pressure inside containment will remain below the containment building design pressure if a Loss-of-Coolant Accident (LOCA) should occur during plant operation. The analysis ensures that the containment heat removal capability is sufficient to remove the maximum possible discharge of mass and energy to containment without exceeding the containment design pressure. Short-term LOCA analyses are conducted to determine the ability of containment sub-compartments to withstand the high pressure pulse associated with the rupture of a high energy pipe.

The purpose of this discussion is to review the evaluation conducted to determine if the LOCA mass and energy releases and the resulting containment response from the Containment Margin Program can be shown to bound the Margin Recovery Program (MRP). From a short-term LOCA perspective, an evaluation was conducted that compared the current releases with the MRP conditions and the recently evaluated Rerating Conditions.

1 SRP Section 15.6.3

2 Without iodine spike

3 0-8 hour dose at the LPZ

4.4.1 Evaluation of Plant Changes on LOCA Containment Integrity

Purpose:

The purpose of this evaluation was to estimate the effect of the above plant changes on the LOCA mass and energy releases and the resulting containment response.

Steam Generator Tube Plugging:

The Containment Margin Program and the Rerating Study used a SGTP of 0%, which is conservative for containment integrity analysis. A 0% Steam Generator Tube Plugging level:

- Maximizes reactor coolant volume
- Maximizes heat transfer area across the SG tubes
- Lower resistance in loop, therefore increased break flow, lower DP up-stream of break

The effects of asymmetric tube plugging on the double-ended pump suction (DEPS) case are bounded by the assumption of no tube plugging. This is due to the effects described above as well as the insensitivity of total energy released to tube plugging levels. Therefore, the mass and energy release and containment response is bounded by the Containment Margin Program and Rerating Study.

RCS Pressure Uncertainty and RCS T_{avg} Range:

Long-term LOCA mass and energy release analyses are bounded by high pressure and low temperature. The new RCS pressure uncertainty and T_{avg} range result in a slight difference in values from the Containment Margin program values. The difference is offset by margin; thus, the analysis remains bounding for long-term LOCA mass and energy.

An evaluation was conducted to determine if the current short-term releases bound the new conditions. A short-term sub-compartment evaluation was conducted for Rerating Conditions to determine the effect of proposed rerating on structural integrity. The Rerating Study concluded that the current sub-compartment analysis results remain bounding for the proposed rerating of 3600 MWt. However, rerating of the Salem Units did not occur. Since Salem has whip restraints, using double-ended data is conservative and RCS loop breaks (except for the reactor cavity) are less than a single-ended break area. The increase in releases for hot leg breaks was analyzed to be less than 17%. However, assuming a hot leg break size of not larger than single-ended, a benefit of at least 58% was documented. For cold leg breaks, an increase in releases of 15% was calculated. However, the benefit, for cold leg breaks, associated with assuming no break larger than single-ended is at least 23%. Therefore, the current short-term analysis will remain bounding for the margin recovery program.

RCS Thermal Design Flow:

The effects of thermal design flow are propagated in containment integrity analysis through RCS temperature and pressure. For long-term LOCA, containment temperature and pressure peaks after blowdown, where the effect of thermal design flow is negligible. Therefore, the Containment Margin Program analysis remains bounding. For short-term LOCA, there is no significant impact of thermal design flow on the calculations. Thus, the current analysis remains bounding.

4.4.2 Accumulator Operating Conditions

Historically, nominal accumulator pressure and water volume have been used in calculation of containment peak pressure for long-term LOCA analysis. However, analyses have shown that minimum accumulator pressure and maximum accumulator volume are more conservative from containment integrity analysis standpoint. Still, the amount of margin obtained by using nominal accumulator values is small compared to the inherent conservatism in the containment analysis.

4.4.3 Results

The evaluation has shown that changes in plant parameters described previously have negligible effect on mass and energy releases.

For long-term, the current releases remain bounding. For short-term LOCA, the current releases remain bounding when whip restraints and/or Leak-Before-Break (LBB) technology are taken into account. These assumptions reduce the possible break area, and are a benefit for mass and energy releases.

4.5 Component Evaluations

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code provides criteria and requirements for the evaluation of stress levels in the pressure boundary for design, normal operating, and accident conditions. The margin of safety provided by use of the design pressure as a basis for pressure limits is provided by the inherent safety factors in the criteria and requirements of the ASME Code. In 10 CFR 50.55a, the Nuclear Regulatory Commission (NRC) defines, for design purposes, the applicable ASME Code Edition for Class 1 components for plants whose construction permits were issued prior to May 14, 1984 to be the Code Edition defined in the construction permit. The applicable Editions of the Code and subsequent Addenda are defined in Table 4.2-9 of the Salem UFSAR.

4.5.1 Reactor Vessel

Evaluations were performed for the various regions of the Salem reactor vessels to determine the stress and fatigue usage effects of Nuclear Steam Supply (NSSS) operation at the Margin Recovery Program (MRP) conditions throughout the current plant operating license. The evaluations assess the effects of the revised design transients, operating parameters, and reactor vessel/internals interface loads on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions. Where appropriate, stress and fatigue evaluations were performed for the MRP conditions. Results of these evaluations were compared to the applicable Code Sections limits. The conclusion of the evaluations are that the Salem reactor vessels can be operated for the remainder of their 40 year life over the range of applicable T_{hot} and T_{cold} values.

For all evaluated components, the stress intensity range remains below the Code limit of $3S_m$ with the exception of the Control Rod Drive Mechanism (CRDM) housings. Therefore, the acceptability of operation at the MRP conditions was justified by simplified elastic-plastic analysis.

4.5.2 Reactor Internals

Since the operating parameters for the MRP differ from the original design, the reactor vessel system/fuel interface was thoroughly addressed in order to assure compatibility and structural integrity of the core during operation. In addition, thermal-hydraulic analyses are required to verify that existing core bypass flow limits are not exceeded and to develop pressure drops and upper head temperatures for input to Appendix K (Emergency Core Cooling System), non-LOCA accident analyses, and NSSS performance evaluations. The subject areas most likely to be affected by changes in system operating conditions are:

1) Reactor internals system thermal/hydraulic performance

2) Rod control cluster assembly (RCCA) scram performance, and

3) Reactor internals system structural response and integrity

4.5.2.1 System Pressure Losses

Total coolant pressure drops across the reactor internals were evaluated for the current plant configuration. Two cases were evaluated for the effect on the reactor vessel/internals/fuel pressure drops. Case I included a core inlet temperature of 529.0°F, 25% peak steam generator tube plugging (SGTP), and a thermal design flow of 82,500 gpm/loop. Case II included a core inlet temperature of 543.2°F, 25% peak SGTP, and a thermal design flow of 82,500 gpm/loop. For the worst condition, Case I, the total reactor internal pressure drop will decrease compared to the present condition.

4.5.2.2 Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Analyses were performed to estimate core bypass flow values to ensure that the design bypass flow limit for the plant is not exceeded.

Bypass flow is composed of leakage through the baffle joints and into the core prior to the flow entering the bottom of the core; diverted flow into the vessel closure head region for cooling purposes; leakage from the inlet/downcomer region directly into the outlet nozzle; leakage through the baffle plate cavity gap at the periphery of the core; and fuel assembly thimble tube leakage. The sum of these flows was determined to be below the design limit of 7.2%.

4.5.2.3 Hydraulic Lift Force Analysis

An evaluation was performed to estimate the hydraulic lift forces on the various reactor internal components for the MRP. This analysis is required to determine if the reactor internals remain seated and stable. Hydraulic lift forces were calculated based on a mechanical design flow rate of 99,600 gpm/loop. The overall effect of the MRP upon reactor internal hydraulic forces is negligible compared to the original analyzed condition.

4.5.2.4 RCCA Scram Performance Evaluation

The RCCA drop time and the corresponding normalized RCCA position versus time curve were evaluated for the MRP conditions and found to be acceptable.

4.5.2.5 Structural Evaluation

Structural evaluations were performed to demonstrate that the structural integrity of the reactor components is not adversely affected by the change in RCS conditions and transients, or by the change on reactor thermal/hydraulic or structural performance. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth which must be accounted for in the design and analysis of the various components. Evaluations were performed for the critical reactor internal components which indicated that the structural integrity of the reactor internals is maintained for the proposed MRP conditions.

4.5.2.6 Mechanical System Evaluations

Evaluations of the critical reactor internal components were performed which indicated that the MRP conditions will not adversely impact the response of the reactor internals systems and components to Seismic/LOCA excitations and flow induced vibrations.

4.5.3 Steam Generators

The Salem Units 1 and 2 steam generators were evaluated with respect to structural integrity, thermal hydraulic performance, U-bend vibration, and U-bend wear.

4.5.3.1 Structural Evaluation

Structural evaluations of the critical components of the Model 51 steam generators were performed to justify operation at MRP conditions. The critical components considered included the tubesheet and shell junctions, the divider plate, the steam generator tubes, the tube/tubesheet weld, the nozzles, and the shell including the upper shell penetrations. The MRP conditions, in particular the reduction in vessel outlet temperature, cause a reduction in the secondary side temperature and pressure, which results in a higher pressure differential across the primary-to-secondary boundary.

The results of the evaluations showed that the maximum stress intensities remain within the limits for all conditions analyzed. Fatigue usage factors remained below the ASME Code allowable limit of 1.0 for all components, with the exception of the manway bolts. The bolts were not originally qualified for 40 year operation at the current conditions, and the same is true for the MRP conditions.

4.5.3.2 Thermal Hydraulic Evaluation

The thermal hydraulic characteristics, such as circulation ratio, hydrodynamic stability, and secondary mass of the Salem steam generators were shown to be acceptable for the range of conditions which bound the conditions of the MRP.

4.5.3.3 U-Bend Vibration

The potential for vibration of the small radius U-bends due to fluid elastic instability at the MRP operating conditions was evaluated. The evaluation identified all tubes which might be sufficiently affected by vibration such that preventative action would be required. For each of these tubes a minimum operating steam pressure was determined which would permit the tube to remain in service for the remainder of the operating plant license.

4.5.3.4 U-Bend Wear

Tubes in the U-bend region of steam generator tube bundles have shown some degree of wear at the intersections with the anti-vibration bars (AVBs). Rather than corrosion, AVB indications have been confirmed as wear caused by fluid elastic vibration. Estimates for increased wear at the AVB intersections, which could result from the MRP conditions, were developed. The proportion of tubes which become unstable as a result of the changed operating conditions in relation to the tubes which are already unstable was determined. The evaluation performed showed that a very small number of tubes (<5 tubes per steam generator) might be affected by long term operation at the MRP conditions. Therefore, the increase in wear resulting from the revised operating conditions is not considered significant.

4.5.4 Pressurizer

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and pressure and to keep the reactor coolant system at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow to or outflow from the pressurizer, as required.

The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool water spray into the steam space at the top of the pressurizer.

The pressurizer components were evaluated for the conditions associated with the MRP. The results showed that all pressurizer components meet the ASME Code requirements. The evaluations showed that there is no significant effect on the component stresses and fatigue analysis, except for the surge nozzle. The surge nozzle stress analysis was revised for the revised thermal stratification pipe loads, and was determined to be acceptable.

4.5.5 Reactor Coolant Pumps (RCP)

The revised conditions associated with the MRP were reviewed, and it was determined that the RCP and RCP motor integrity and functions remain acceptable under these new conditions.

The RCP structural integrity and RCP coastdown was determined to be acceptable for the MRP conditions. Pump pressure boundary components are not adversely affected.

The revised pump hot and cold horsepower projections are 6320 and 7970 hp, respectively. Calculated values of winding temperatures rise remain within the allowable limits. Therefore, continued operation of the motors at the MRP conditions is acceptable.

The worst case starting scenario was determined for maximum reverse flow under cold loop conditions with 80% of rated voltage. The revised parameters represent a marginal increase over the current loading, and therefore, should not have a significant impact on motor starting performance.

4.5.6 Control Rod Drive Mechanism

An evaluation was performed to determine the effects of the MRP on the Model L106A CRDMs at Salem Units 1 and 2. The evaluation determined that the current analysis of the CRDMs remains bounding for the conditions associated with the MRP.

4.5.7 Reactor Coolant Piping and Supports

An evaluation was performed to determine the impact on the design basis analysis for the reactor coolant loop piping and the primary equipment supports from operating at the MRP conditions which incorporate a range of Reactor Coolant System (RCS) temperature, a reduced Thermal Design Flow (TDF), 20% average and 25% peak Steam Generator Tube Plugging (SGTP) levels.

The focus of this evaluation centered on the variation of the operating temperatures in the system. The proposed operating temperature for the hot leg will range from 601.3°F to 616.3°F compared to the original design temperature of 610.8°F. This range considered the envelope of all the conditions presented in this program. The proposed operating temperatures for the cold leg ranged from 529°F to 543.7°F compared to the original design temperature of 545°F.

Three types of analyses were directly or indirectly influenced by the changes in the operating temperature. The thermal, seismic and LOCA analysis were evaluated for the potential impact of these modified temperatures. For the conditions defined for the Salem MRP the existing analysis results have been reconciled and continued to be applicable for both the old and new plant conditions. The pressurizer thermal stratification analysis was reconciled to the specified conditions without changes to the usage factor and the surge nozzle loadings.

4.5.8 Auxiliary Equipment

4.5.8.1 Auxiliary Heat Exchanger/Tanks

The regenerative heat exchanger, residual heat exchanger, seal water heat exchanger, excess letdown heat exchanger, non-regenerative heat exchanger, component cooling water heat exchanger, sample heat exchanger, and spent fuel pit heat exchanger were evaluated for the conditions associated with the MRP. The evaluation concluded that there is no adverse impact on the auxiliary heat exchangers. In addition the auxiliary tanks are not impacted by the MRP.

4.5.8.2 Auxiliary Valves

The original design and qualification requirements of the auxiliary valves at Salem Units 1 and 2 were evaluated. The transients resulting from the MRP are bounded by the original design transients, and therefore, will not adversely impact the auxiliary valves.

4.5.8.3 Auxiliary Pumps

The auxiliary pumps, including positive displacement pump, safety injection pump, residual heat removal pump, CVCS charging pump, component cooling water pump, containment spray pump, spent fuel pit pump, boric acid transfer pump, chemical drain tank pump, holdup tank recirc. pump, and RCS drain tank pump were evaluated for the MRP conditions. The specifications require the pumps to be qualified for pressure and temperature transients, or, if the equipment was not expected to be significantly affected by the transients, it was designed for maximum steady state pressures and temperatures only. The evaluation concluded that the MRP will not affect the qualification of the auxiliary pumps.

4.5.9 Fluid and Auxiliary Systems Evaluations

In general, the direct consequences of the MRP on the Salem NSSS fluid systems are changes in RCS operating temperatures (T_{hot}, T_{cold} and T_{avg}) and reductions in RCS volume, SG thermal performance (e.g., SG outlet pressure) and RCS loop flows. The scope of this section applies to the following NSSS and auxiliary fluid systems:

- Reactor Coolant System (RCS)
- Chemical and Volume Control System (CVCS)
- Safety Injection System (SIS)
- Residual Heat Removal System (RHRS)
- Spent Fuel Pit Cooling System (SFPCS)
- Component Cooling Water System (CCWS)
- Waste Disposal System (WDS)
- Containment Spray System (CSS)
- Sampling System (SS)

Each of the systems was reviewed to determine the effect of implementation of the MRP operating conditions. Where the revised conditions were outside of the original design parameters, the system was further assessed to evaluate the acceptability of the new operating conditions.

4.5.9.1 Reactor Coolant System

The Salem Unit 1 and 2 RCS is comprised of four parallel heat transfer loops which are connected to a single reactor vessel which houses the reactor core. Each heat transfer loop is comprised of a SG and a RCP. To allow RCS pressure to be maintained and controlled during both normal and abnormal plant operating conditions, a pressurizer vessel is provided. A surge line is provided at the bottom of the pressurizer which is also connected to one RCS heat transfer loop. RCS pressure control is provided by use of pressurizer electric heaters, a pressurizer vessel spray subsystem and pressurizer Power Operated Relief Valves (PORVs).

System overpressure protection is provided via three safety relief valves which are mounted at the top of the pressurizer vessel. Discharge from the pressurizer PORVs and/or safeties is directed to a Pressurizer Relief Tank (PRT).

The direct consequence of additional SG tube plugging is a reduction in overall RCS volume. For this project, a range of RCS liquid volume and corresponding mass was calculated based on the project variations in RCS operating parameters and existing values for component nominal volumes. In general, a reduction in RCS power volume/mass associated with SG tube plugging has no direct impact on system operation. The impact of this reduction on plant safety analyses are discussed in other sections of this report.

The RCS temperature allowable operating ranges were compared to assumptions utilized to generate auxiliary equipment design transients. The comparison showed that the existing transients are either unchanged or are still bounding at the specified RCS operating conditions.

The results of the various evaluations showed that the revised RCS operating conditions remain within the system design and performance requirements without any limitations.

4.5.9.2 Chemical and Volume Control System

The Salem Unit 1 and 2 CVCS is comprised of a letdown and charging subsystem which provides support services to the RCS during plant operations. During normal plant operations, the CVCS is used to maintain pressurizer level, provide purification and dilution of the RCS, provide seal injection flow to support RCP operation and provide means for RCS chemical addition (boron, lithium, hydrogen, etc.). The CVCS is provided with three parallel system pumps (two centrifugal charging and one positive displacement) capable of taking suction from the Volume Control Tank (VCT), the Refueling Water Storage Tank (RWST) and the emergency boration flow path (boric acid tanks/transfer pumps).

The direct consequence of higher RCS operating temperatures is a higher interface temperature with the CVCS (i.e., letdown temperature). With the nonregenerative and/or excess letdown heat exchangers in service, a higher inlet temperature has the potential to increase heat exchanger duty and/or reduce heat exchanger thermal performance (e.g., outlet temperature). The expected worst-case (highest) letdown temperature (@RCS Tcold conditions) under the revised RCS operating conditions is maintained at or below the subject heat exchanger design inlet temperatures. As such, the regenerative and excess letdown heat exchangers heat load, process outlet temperature and service side outlet temperature would not increase. Since the nonregenerative (letdown) and seal water heat exchangers receive flow from these heat exchangers, they would also not be impacted by the revised RCS operating temperatures.

Likewise, the direct consequence of lower RCS operating temperatures is a lower interface temperature with the CVCS (i.e., letdown temperature). From a heat exchanger thermal performance (heat load) perspective, this is generally a non-limiting condition. With respect to the RCS, the letdown fluid temperature sets the temperature of the charging flow returned to the RCS (a lower letdown temperature will result in a lower charging return temperature). The overall reduction in charging flow return temperature is evaluated to have an insignificant impact on RCS normal operations due to 1) the relatively large flow difference between the RCS loop (82,500 gpm) and the charging returned flow (<100 gpm) and 2) RCS loop Tcold is measured downstream of the charging return connection.

The direct consequence of additional SG tube plugging is a reduction in overall RCS volume/mass. In general, reduced RCS volume/mass associated with SG tube plugging is conservative with respect to CVCS chemical addition operations, and as such, is evaluated to be acceptable. The CVCS design basis is confirmed as part of the reload process. Thus, the reduction in shutdown margin and its effect are also confirmed on a reload basis.

The results of the various evaluations showed that the revised RCS operating conditions remain within the system design and performance requirements without any limitations.

4.5.9.3 Safety Injection System

The Salem Unit 1 and 2 SIS is comprised of multiple subsystems which provide emergency core cooling following primary and secondary side design basis events. The revised RCS operating conditions have no direct impact on the performance capability of the SIS.

4.5.9.4 Residual Heat Removal System

The Salem Unit 1 and 2 RHRS is comprised of two parallel cooling trains. Each train contains a centrifugal pump and a shell and tube heat exchanger. To support plant heat-up and cool down, cold shut down and refueling modes of operation, the system takes suction from one of the RCS hot legs and can return the flow to all four RCS cold leg loops. System heat removal and pump cooling is provided via the CCWS.

The RHRS is aligned for operation only during Hot Shutdown, Cold Shutdown, and Refueling conditions. The operating conditions for the RHRS are not impacted by the full power conditions for the MRP. There is a minor effect of reduced RCS volume inventories at the higher steam generator tube plugging levels in terms of sensible heat load, but this is conservative with respect to RHRS performance.

4.5.9.5 Spent Fuel Pool Cooling System

The Salem Unit 1 and 2 SFPCS is comprised of a single cooling train which contains two parallel centrifugal cooling pumps. The system cooling pumps recirculate Spent Fuel Pit (SFP) fluid through a shell and tube heat exchanger where SFP heat can be transferred to the CCWS. The SFPCS has no direct connection with the RCS. Since the decay heat from the spent fuel will remain the same, the performance capability of the SFPCS is not impacted.

4.5.9.6 Component Cooling Water System

The Salem Unit 1 and 2 CCWS is comprised of two parallel cooling loops which is serviced by three parallel centrifugal cooling pumps. Each cooling loop is provided with a heat exchanger where waste heat can be transferred to the Service Water System (SWS). The direct consequence of higher RCS operating temperatures is a higher letdown temperature. The interface with the CCWS is at the excess letdown heat exchanger, letdown (or non-regenerative) heat exchanger, and seal water heat exchanger. The expected letdown temperature (at RCS T_{cold} conditions) under the MRP operating conditions is maintained at or below the subject heat exchanger design inlet temperatures. As such, the non-regenerative, excess letdown, and seal water heat exchangers heat load, process outlet temperature, and service side outlet temperature will not increase.

4.5.9.7 Waste Disposal System

The Salem Unit 1 and 2 WDS is comprised of separate gaseous and liquid waste processing subsystems. Plant operation with reduced RCS volumes (due to SG tube plugging) could potentially reduce total liquid waste volumes/increase waste activity levels originating from the RCS and its connected auxiliary systems. This is judged to have an insignificant impact on system operation. Operations with the revised RCS operating conditions have no direct impact on WDS performance capability.

4.5.9.8 Containment Spray System

The Salem Unit 1 and 2 CSS is comprised of two separate trains which can provide post-accident containment cooling, sump pH adjustment and sump iodine retention. Each CSS train contains a centrifugal pump which takes suction from the RWST and delivers its flow to two CSS ring headers located in the upper portion of the containment building. Both CSS trains utilize an eductor device to meter in a small amount of concentrated NaOH solution from a common Spray Additive Tank (SAT). The concentrated NaOH mixes with the borated RWST water and raises the pH of the sprayed solution to an alkaline condition.

The revised RCS operating conditions have no direct impact on the performance capability of the CSS.

4.5.9.9 Sampling System

The Salem Unit 1 and 2 SS is comprised of various flow paths which provide means for samples from the RCS and selected auxiliary systems to be drawn and cooled for analyses. The CCWS is used to provide cooling to each of the sample coolers.

The direct consequence of higher RCS operating temperatures is a higher interface temperature with the SS when a fluid sample is drawn. In general, a higher fluid inlet temperature has the potential to increase sample heat exchanger duty and/or reduce heat exchanger thermal performance (e.g., outlet temperature). The expected worst-case (highest) sample temperature (@RCS Thot conditions) under the revised Margin Recovery RCS operating conditions is maintained at or below the subject heat exchanger design inlet temperatures. As such, sample heat exchanger heat load, process outlet temperature and service side outlet temperature (or design flow requirements) would not increase. In general, lower operating temperatures represents a non limiting condition from the heat exchanger thermal performance perspective. The results of the various evaluations showed that the revised RCS operating conditions remain within the system design and performance requirements without any limitations.

4.5.10 Conclusion

NSSS components were evaluated and results compared to the allowable stress and fatigue limits defined by the ASME Code Editions to which the components were originally designed and evaluated. In almost all cases, the MRP conditions and transient loadings resulted in stresses and fatigue usage factors below the Code allowable limits. The exceptions were justified by simplified elastic plastic analysis. Therefore, it has been determined that the NSSS components will not be adversely affected by the MRP. Moreover, the MRP will not adversely impact the NSSS and auxiliary fluid systems.

Document Control Desk
Attachment 4

LR 96114
LCR S94-41

SALEM GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
CHANGE TO TECHNICAL SPECIFICATIONS
MARGIN RECOVERY PROGRAM

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. DPR-70 (Salem Unit No. 1) are affected by this change request:

Technical Specification		Page
Index		I - II
1.0	Definitions	1-2
2.0	Safety Limits and Limiting Safety System Settings	2-1 - 2-3 2-5 2-8 - 2-9
B2.0	Bases	B2-1 - B2-2 B2-5 - B2-6
3/4.1	Reactivity Control System	3/4 1-1 - 3/4 1-2 3/4 1-5 - 3/4 1-5a 3/4 1-18 3/4 1-23 - 3/4 1-25
3/4.2	Power Distribution Limits	3/4 2-1 - 3/4 2-2 3/4 2-4 - 3/4 2-9 3/4 2-14
B3/4	Bases	B3/4 1-1 - B3/4 1-3 B3/4 2-1 - B3/4 2-5 B3/4 4-1
5.0	Design Features	5-5
6.0	Admin Controls	6-24

DEFINITIONS

Core Operating Limits Report

SECTION | PAGE

1.0 DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

1.8 NOT USED

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

Insert A

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

INSERT A

1.9a

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure~~s~~ 2.1-1 ~~and 2.1-2~~ for 4 ~~and 3~~ loop operation, ~~respectively~~.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

Move to 2-3

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

INSERT
B
HERE

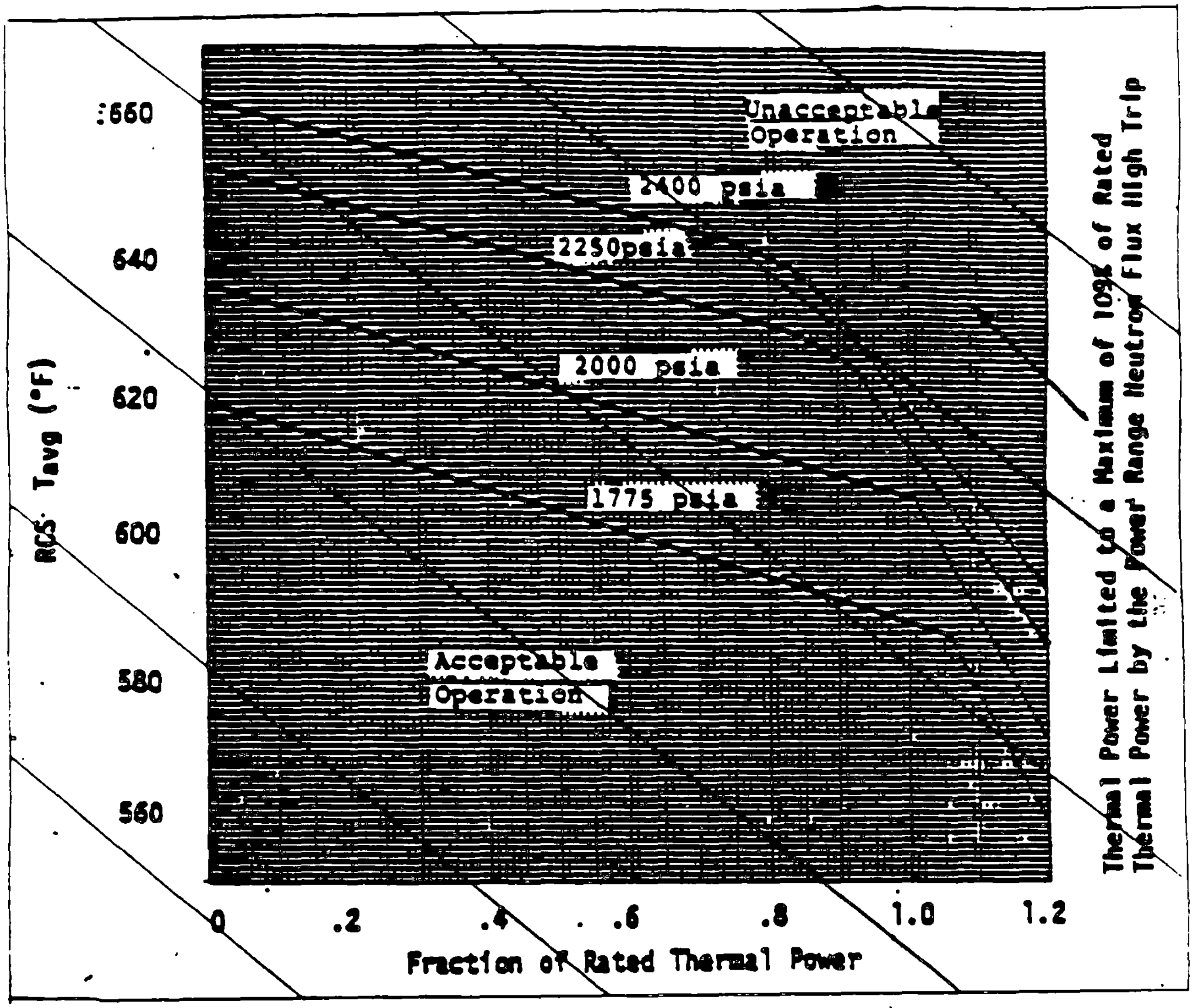
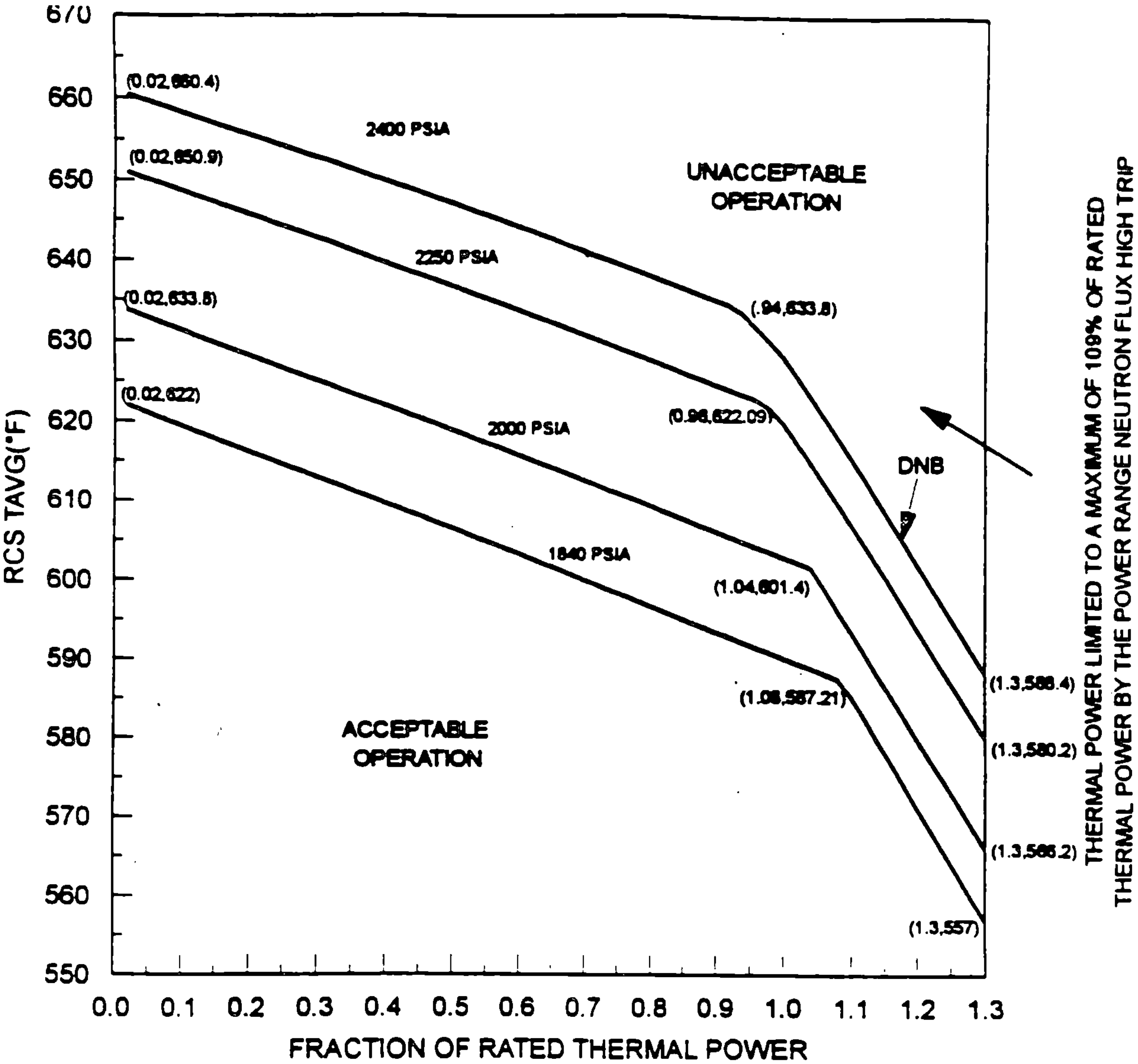


FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

INSERT B



(0.02,660.4)
2400 PSIA
(0.02,650.9)
UNACCEPTABLE OPERATION
2250 PSIA
(0.02,633.8)
(.94,633.8)
(0.02,622)
(0.96,622.09)
2000 PSIA
DNB
1840 PSIA
(1.04,601.4)
(1.08,587.21)
ACCEPTABLE OPERATION
(1.3,588.4)
(1.3,580.2)
(1.3,566.2)
(1.3,557)
RCS TAVG(°F)
670
660
650
640
630
620
610
600
590
580
570
560
550
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
FRACTION OF RATED THERMAL POWER
THERMAL POWER LIMITED TO A MAXIMUM OF 109% OF RATED THERMAL POWER BY THE POWER RANGE NEUTRON FLUX HIGH TRIP

Reactor Coolant System Pressure
From Pg 2.1

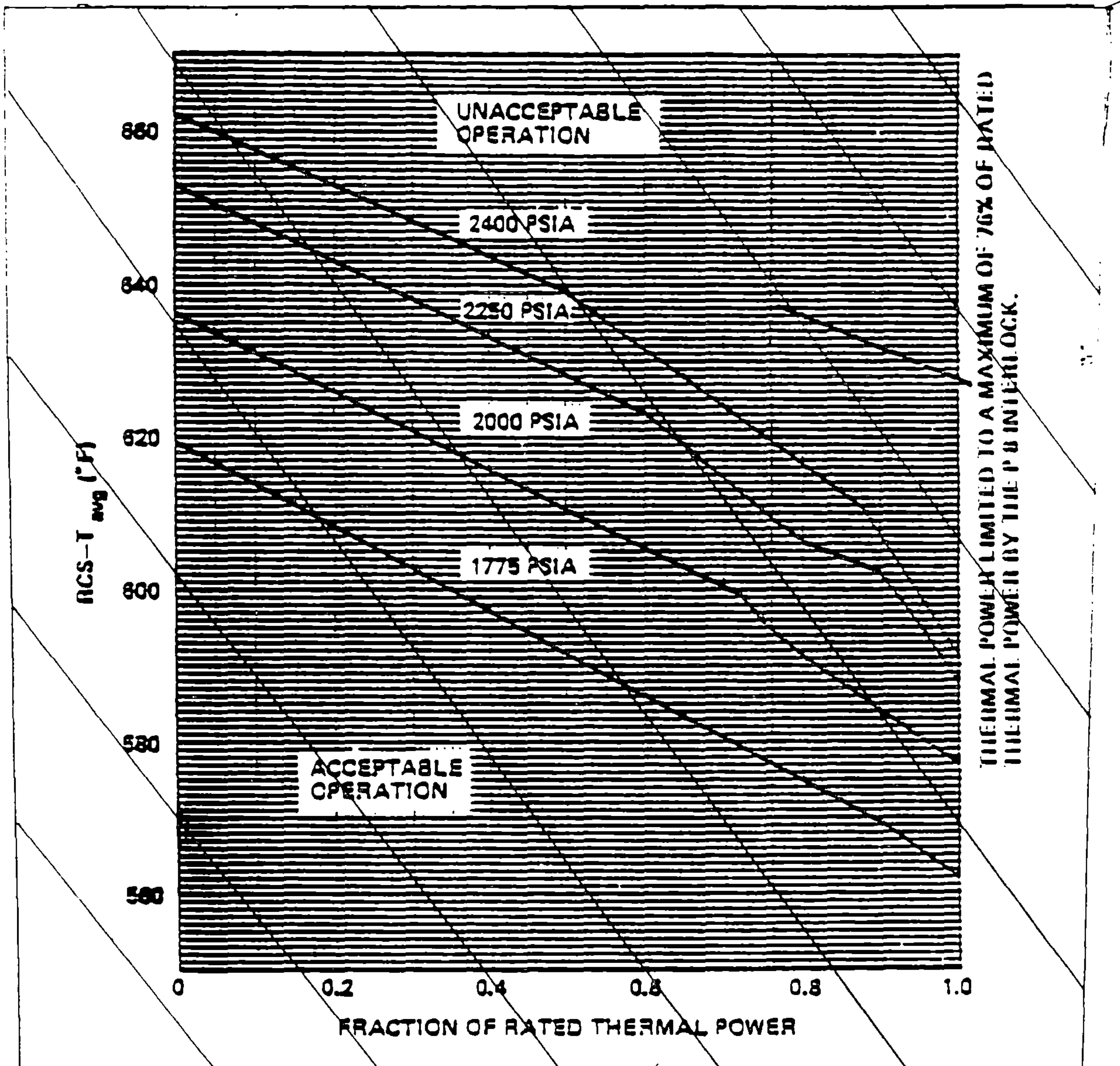


FIGURE 2.1-2 REACTOR CORE SAFETY LIMIT – THREE LOOPS IN OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is ~~87,300~~ gpm per loop.

82,500

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops	~~Operation with 3 Loops~~
K_1 = ~~1.164~~ 1.22	~~K_1 = 1.05~~
K_2 = ~~0.01434~~ 0.02037	~~K_2 = 0.01434~~
K_3 = ~~0.00073~~ 0.001020	~~K_3 = 0.00073~~

and f_1 (ΔI) is 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 such that:

(i) for $q_t - q_b$ between -23 percent and ~~+10~~ +13 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -23 percent, the ΔT trip setpoint shall be automatically reduced by 1.26 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds ~~+10~~ +13 percent, the ΔT trip setpoint shall be automatically reduced by ~~1.34~~ 2.63 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T'') - f_2(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

Indicated

T'' = ~~Reference~~ T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

K_4 = ~~1.089~~ 1.09

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

0.00149

K_6 = ~~0.00115~~/°F for $T > T''$; K_6 = 0 for $T \leq T''$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator, Sec^{-1}.

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.1 ~~3.1~~ percent.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.1 ~~3.0~~ percent.

BASES

2.1.1 REACTOR CORE

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.

correlations which have

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 ~~the W-3, R-Grid correlation for standard (LOPAR) fuel assemblies and WRB-1 correlation for Vantage 5H fuel assemblies. The DNB correlation has~~ 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 percent probability that the minimum DNBR of the limiting rod during condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3 R-Grid correlation). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR correlation limit (1.17 for the WRB-1 or 1.30 for the W-3 R-Grid).~~

The curves of Figure~~s~~ 2.1-1 ~~and 2.1-2~~ shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The DNB design basis is as follows: uncertainties in the WRB-1 and WRB-2 correlations, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

BASES

$F^{RTP}_{\Delta H}$

The curves are based on an enthalpy hot channel factor, ~~$F^N_{\Delta H}$, of 1.55~~ and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

~~$F^N_{\Delta H} = 1.55\ [1 + 0.3(1-P)]$~~

~~where P is the fraction of RATED THERMAL POWER~~

INSERT C HERE

rod positions From

These limiting heat flux conditions are to higher than those calculated for the range of all control rods FULLY WITHDRAWN the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature 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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 1955 Edition while the valves are designed to ANSI B 16.5, MSS-SP-66-1964, or ASME Section III-1968, which permit maximum transient pressures of up to 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

INSERT C

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}\ [1.0 + PF_{\Delta H}(1.0 - P)]$$

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR, and P is $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

has not been evaluated and is not permitted

Operation with a reactor coolant loop out of service below the 4 loop P-8 set point does not require reactor protection system set point modification because the P-8 set point and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature ΔT set point. Three loop operation above the 4 loop P-8 set point ~~is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature ΔT channels and raising the P-8 set point to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.~~

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients ~~and anticipated transients when 3 loops are in operation and the Overtemperature ΔT trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients with 3 loops in operation~~.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ ~~1.6%~~ 1.3% Δk/k ~~###~~.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN < ~~1.6%~~ 1.3% Δk/k ~~###~~, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ ~~1.6%~~ 1.3% Δk/k ~~###~~:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits ~~of~~ in the COLR per Specification 3.1.3.5.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits ~~of~~ in the COLR per specification 3.1.3.5.

*See Special Test Exception 3.10.1

#With K_{eff} ≥ 1.0

##With K_{eff} < 1.0

~~### 1.85% delta k/k during Cycle 11 of operation.~~

SURVEILLANCE REQUIREMENTS (Continued)

in the COLR per

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit ~~of~~ Specification 3.1.3.5.

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,

2. Control rod position,

3. Reactor coolant system average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

INSERT D HERE

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

~~a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.~~

~~b. Less negative than -4.4 x 10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~

Beginning of Cycle Life (BOL) Limit

APPLICABILITY: ~~Specification 3.1.1.4.a~~ - MODES 1 and 2* only#
~~Specification 3.1.1.4.b~~ - MODES 1, 2 and 3 only#

End of Cycle Life (EOL) Limit

ACTION:

a. With the MTC more positive than the BOL limit ~~of 3.1.1.4.a, above,~~ specified in the COLR, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than ~~0 delta k/k/°F~~ the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits in the COLR per ~~of~~ Specification 3.1.3.~~6~~ 5

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit ~~of 3.1.1.4.b, above,~~ specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

INSERT D

within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less positive than or equal to 0 $\Delta k/k/°F$.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit ~~of Specification 3.1.1.4.a, above,~~ prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

The 300 ppm surveillance limit specified in the COLR

b. The MTC shall be measured at any THERMAL POWER and compared to ~~-3.7 x 10 delta k/k/°F~~ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~-3.7 x 10 delta k/k/°F~~, the MTC shall be remeasured, and compared to the EOL MTC limit ~~of specification 3.1.1.4.b,~~ at least once per 14 EFPD during the remainder of the fuel cycle.

specified in the COLR,

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or mis-aligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or mis-aligned from its group step counter demand position by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits ~~of Figures 3.1-1 and 3.1-2;~~ in the COLR per Specification 3.1.3.5. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

POSITION INDICATION SYSTEM SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as ~~shown in Figures 3.1-1 and 3.1-2.~~ specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1*, and 2*#

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the ~~above figures, or~~ insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours by use of the group demand counters and verified by the analog rod position indicators** except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours**.

*See Special Test Exceptions 3.10.2 and 3.10.3

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

#With K_{eff} greater than or equal to 1.0

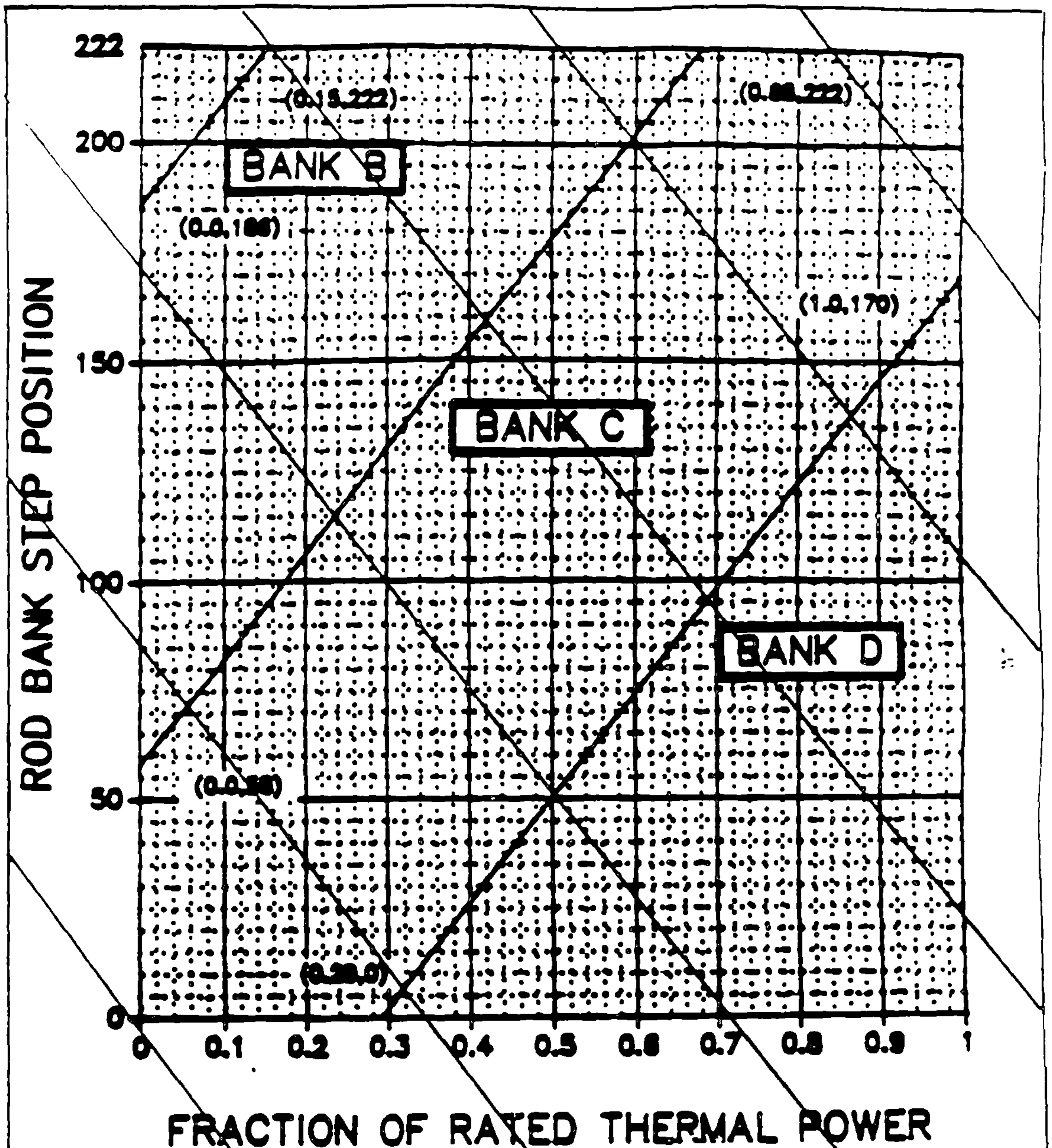


Figure 3.1-1
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

FIGURE 3.1-2

INTENTIONALLY LEFT BLANK PENDING
COMMISSION APPROVAL OF THREE LOOP OPERATION

FIGURE 3.1-2 ROD BANK INSERTION LIMITS VERSUS THERMAL POWER FOR THREE LOOP OPERATION

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

the

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within ~~+6, -9%~~ target band ~~(flux difference units)~~ about the target flux difference.

as specified in the CORE OPERATING LIMITS REPORT (COLR)

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ~~above~~ limits and with THERMAL POWER:

1. Above 90% of RATED THERMAL POWER, within 15 minutes:

a) Either restore the indicated AFD to within the target band limits, or

b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. Between 50% and 90% of RATED THERMAL POWER:

specified in the COLR

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the [redacted] limits for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits ~~shown on Figure 3.2-1~~. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to <55% of RATED THERMAL POWER within the next 4 hours.

b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits ~~of Figure 3.2-1~~. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ~~above~~ limits, and ACTION 2.a) 1), above has been satisfied.

Specified in the COLR

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ~~above~~ limits for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

of the target band

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside ~~the limits of Specification 3.2.1~~. Penalty deviation outside ~~of the limits~~ shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels below 50% of RATED THERMAL POWER.

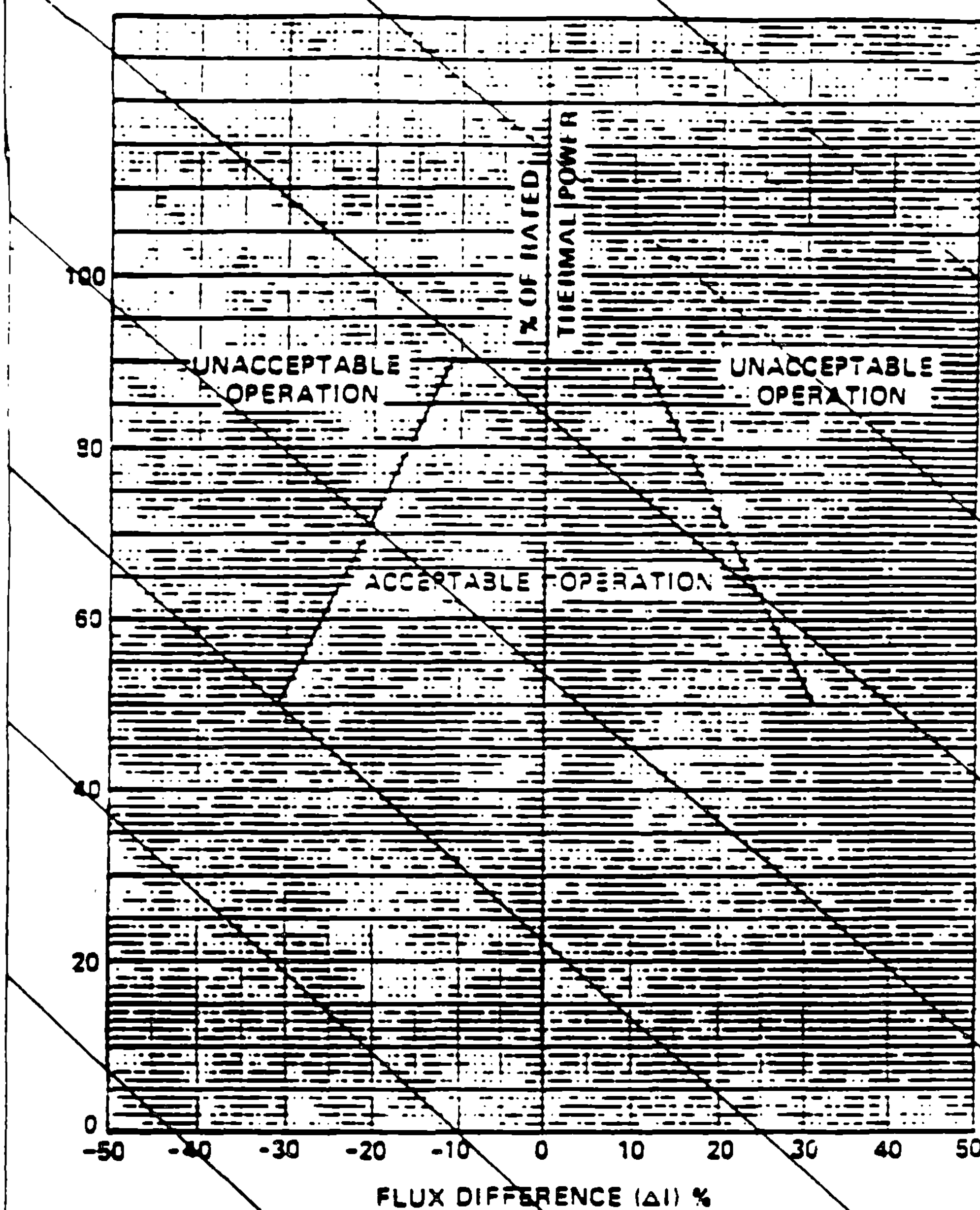


FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

INSERT E HERE

~~$F_Q(Z) \leq [\frac{2.32}{P}]\ [K(Z)]$ for $P > 0.5$~~

~~$F_Q(Z) \leq [(4.64)]\ [K(Z)]$ for $P \leq 0.5$~~

~~where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$~~

~~and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.~~

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

INSERT E

$$F_Q(z) \leq \frac{F_Q^{RTP}}{P} * K(z) \text{ for } P > 0.5, \text{ and}$$

$$F_Q(z) \leq \frac{F_Q^{RTP}}{0.5} * K(z) \text{ for } P \leq 0.5,$$

Where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}, \text{ and}$$

K(z) = the normalized $F_Q(z)$ as a function of core height as specified in the COLR.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f below, and

2. The relationship:

PF_{xy}

$$F_{xy}^{L} = F_{xy}^{RTP}\ [1+\text{[redacted]}(1-P)]$$

where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP}, PF_{xy} is the power factor multiplier for F_{xy} in the COLR, and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} :

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

b) At least once per 31 EFPD, wh chever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in ~~a Radial Peaking Factor Limit Report~~ the COLR per specification 6.9.1.9.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15% inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2% inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

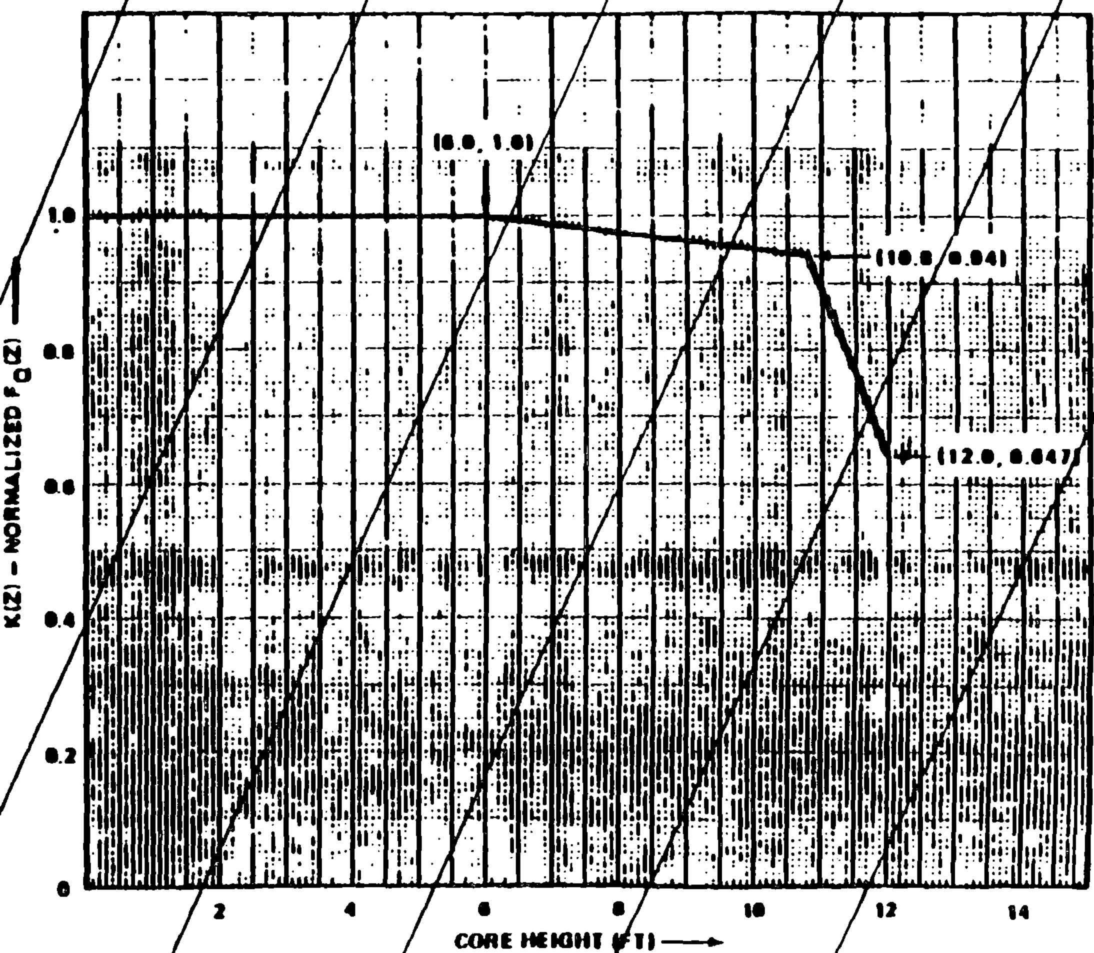


FIGURE 3.2-2

K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$\cancel{F^N_{\Delta H} \le 1.55\,[1.0+0.3(1-P)]}$$

~~where: P = $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$~~

INSERT
C
HERE

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

INSERT C

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}\ [1.0 + PF_{\Delta H}(1.0 - P)]$$

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR, and P is $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 4 Loops In Operation	LIMITS: 3 Loops In Operation
Reactor Coolant System T_{avg}	582.9°F ≤ ~~582°F~~	~~≤ 572°F~~
Pressurizer Pressure	2200 ≥ ~~2220~~ psia*	~~≥ 2220 psia*~~
Reactor Coolant System Flow	~~344,000~~ 341,000 ≥ ~~357,200~~ gpm#	~~≥ 284,500 gpm#~~

*Limit not applicable during either THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

#Includes a ~~2.2%~~ 2.4% flow measurement uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS Tavg. The most restrictive condition occurs at EOL, with Tavg at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, 1.3% a minimum SHUTDOWN MARGIN of ~~1.6%~~ $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With Tavg $\leq$ 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the accident and transient analyses.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analysis to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR analysis to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR analysis into the limiting MTC value. ~~of -4.4×10^{-4} delta k/k/°F. The MTC value of -3.7×10^{-4} delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value -4.4×10^{-4} delta k/k/°F~~.

End of Cycle Life (EOL)

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains with its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

The 300 ppm surveillance limit MTC value represents a conservative value at a core condition of 300 ppm equilibrium boron concentration that is obtained by correcting the limiting EOL MTC for burnup and boron concentration.

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature ≥ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. 1.3% The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of ~~1.6%~~ delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The analysis assumes that the most reactive control rod is not inserted into the core. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires borated water from a boric acid tank in accordance with TS Figure 3.1-2, and additional makeup from either: (1) the second boric acid tank and/or batching, or (2) a maximum of 41,800 gallons of 2,300 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 73,800 gallons of 2,300 ppm borated water is required. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 350,000 gallons during operations in MODES 1, 2, 3 and 4.

The boric acid tanks, pumps, valves, and piping contain a boric acid solution concentration of between 3.75% and 4.0% by weight. To ensure that the boric acid remains in solution, the tank fluid temperature and the process pipe wall temperatures are monitored to ensure a temperature of 63°F, or above is maintained. The tank fluid and pipe wall temperatures are monitored in the main control room. A 5°F margin is provided to ensure the boron will not precipitate out.

Should ambient temperature decrease below 63°F, the boric acid tank heaters, in conjunction with boric acid pump recirculation, are capable of maintaining the boric acid in the tank and in the pump at or above 63°F. A small amount of boric acid in the flow path between the boric acid recirculation line and the suction line to the charging pump will precipitate out, but it will not cause flow blockage even with temperatures below 50°F.

With the RCS temperature below 350°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

3/4.2 POWER DISTRIBUTION LIMITS

meeting the DNB design Criterion

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) ~~maintaining the minimum DNBR in the core ≥ 1.30~~ during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ 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.

$F_{xy}(Z)$ Radial Peaking Factor is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of ~~[illegible]~~ times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

in the COLR per

~~required by~~
Specification 3.2.1

POWER DISTRIBUTION LIMITS

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ~~+6, -9%~~ target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits ~~of Figure 3.2-1~~ specified in the COLR while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD are derived from the plant nuclear instrumentation system through the AFD Monitor Alarm. A control room recorder continuously displays the auctioneered high flux difference and the target band limits as a function of power level. An alarm is received any time the auctioneered high flux difference exceeds the target band limits. Time outside the target band is graphically presented on the strip chart.

Figure B 3/4 2-1 shows a typical monthly target band.

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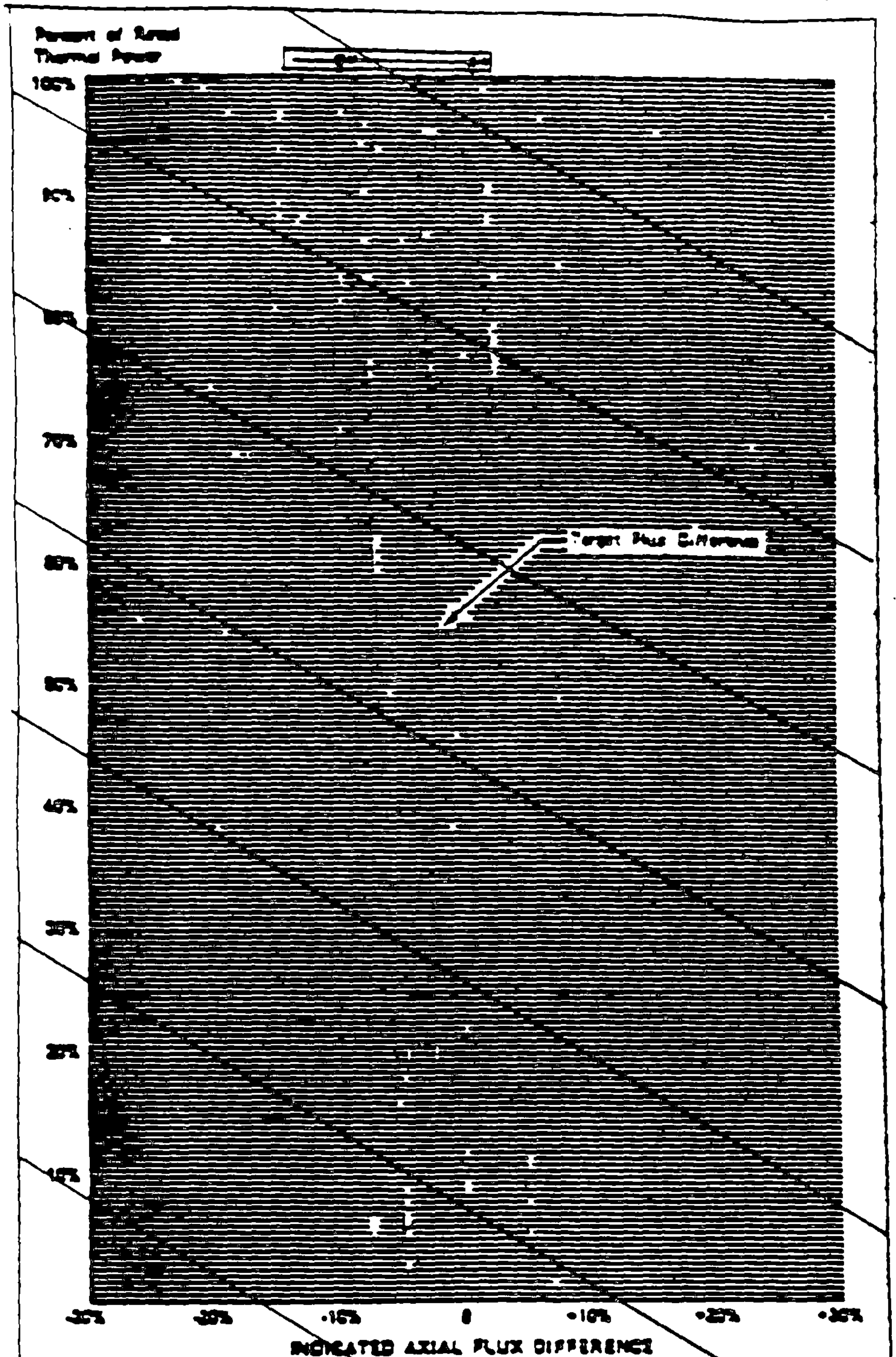

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

INFORMATION ONLY *

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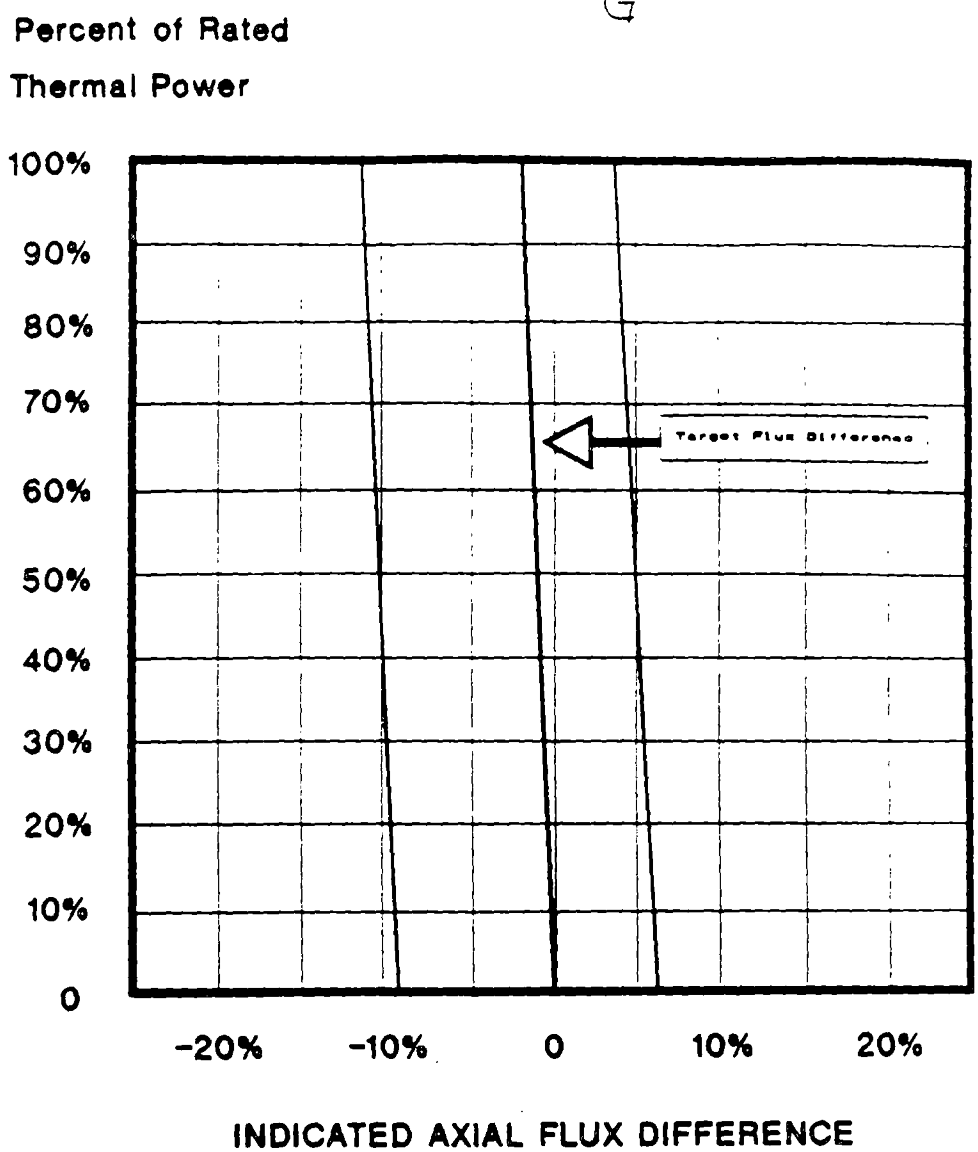

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

* REFER TO COLR FIGURE 2 FOR ACTUAL LIMITS

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL AND RADIAL PEAKING FACTORS-

$F_Q(Z)$, $F^N_{\Delta H}$ and $F_{xy}(Z)$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F^N_{\Delta H} \le$ ~~1.55~~ $F^{RTP}_{\Delta H}$/1.08. Where $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMIT REPORT (COLR). The 8% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

The radial peaking factor $F_{xy}(z)$ is measured periodically to provide assurance that the hot channel factor, $F_Q(z)$, remains within its limit. The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}), as provided in the ~~Radial Peaking Factor Limit Report~~ COLR per specification 6.9.1.9, was determined from expected power control manuevers over the full range of burnup conditions in the core.

3/4 2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and ~~maintain DNBR above 1.30~~ meet the DNB design criterion during all normal operations and anticipated transients. In MODES 1 and 2 with less than all coolant loops in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal for removing decay heat; but, single failure considerations require all loops be in operation whenever the rod control system is energized and at least one loop be in operation when the rod control system is deenergized.

In MODE 4, a single reactor coolant loop or RHR loop provides sufficient heat removal for removing decay heat; but, single failure considerations require that at least 2 loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two RHR loops be OPERABLE.

In MODE 5, single failure considerations require that two RHR loops be OPERABLE. The provisions of Sections 3.4.1.4 and 3.9.8.2 [paragraph (b) of footnote (*)] which permit one service water header to be out of service, are based on the following:

1. The period of time during which plant operations rely upon the provisions of this footnote shall be limited to a cumulative 45 days for any single outage, and

2. The Gas Turbine shall be operable, as a backup to the diesel generators, in the event of a loss of offsite power, to supply the applicable loads. The basis for OPERABILITY is one successful startup of the Gas Turbine no more than 14 days prior to the beginning of the Unit outage.

The operation of one Reactor Coolant Pump or one RHR Pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during Boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with Boron concentration reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 312°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer (thereby providing a volume into which the primary coolant can expand, or (2) by restricting the starting of Reactor Coolant Pumps to those times when secondary water temperature in each steam generator is less than 50°F above each of the RCS cold leg temperatures.

a. In accordance with the code requirements specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is ~~12,811 ± 100~~ 12,446 ± 426 cubic feet at a nominal Tavg of ~~576.7~~ 573.0 °F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The new fuel storage racks are designed and shall be maintained with:

a. A maximum Keff equivalent of 0.95 with the storage racks flooded with unborated water.

b. A nominal 21.0 inch center-to-center distance between fuel assemblies.

c. A maximum unirradiated fuel assembly enrichment of 4.5 w/o U-235.

5.6.1.2 The spent fuel storage racks are designed and shall be maintained with:

a. A maximum Keff equivalent of 0.95 with the storage racks filled with unborated water.

b. A nominal 10.5 inch center-to-center distance between fuel assemblies stored in Region 1 (flux trap type) racks.

c. A nominal 9.05 inch center-to-center distance between fuel assemblies stored in Region 2 (non-flux trap) racks.

d. Fuel assemblies stored in Region 1 racks shall meet one of the following storage constraints.

1. Unirradiated fuel assemblies with a maximum enrichment of 4.25 w/o U-235 have unrestricted storage.

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list of descriptions of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

INSERT H HERE

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.9 The F_{xy} limits for Rated Thermal Power (F_{xy}^{RTP}) for all core planes containing bank "D" control rods and all unrodded core planes, and the plot of predicted heat flux hot channel factor times relative power ($F_Q^T * P_{REL}$) vs. Axial Core Height with the limit envelope shall be provided to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector. The Report shall be provided to the Commission upon issuance.

In addition, in the event that the limit should change, requiring a new submittal or submittal of an amended Peaking Factor Limit Report, it will be submitted upon issuance.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

INSERT H

6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.4,

2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a. Approved by Safety Evaluation dated May 28, 1985.

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, September 1974 (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference. Approved by Safety Evaluation dated January 31, 1978.

3. WCAP-10054-P-A, Rev. 1, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, August 1985 (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

4. WCAP-10266-P-A, Rev. 2, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, Rev. 2. March 1987 (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

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

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

Document Control Desk
Attachment 4

LR 96114
LCR S94-41

The following Technical Specifications for Facility Operating License No. DPR-75 (Salem Unit No. 2) are affected by this change request:

Technical Specification		Page
Index		I - II IV XI
1.0	Definitions	1-2
2.0	Safety Limits and Limiting Safety System Settings	2-1 - 2-3 2-5 2-7 - 2-9
B2.0	Bases	B2-1 B2-5 - B2-6
3/4.1	Reactivity Control System	3/4 1-1 - 3/4 1-2 3/4 1-4 - 3/4 1-5 3/4 1-13 3/4 1-20 - 3/4 1-22
3/4.2	Power Distribution Limits	3/4 2-1 - 3/4 2-2 3/4 2-4 - 3/4 2-9 3/4 2-17
B3/4	Bases	B3/4 1-1 - B3/4 1-3 B3/4 2-1 - B3/4 2-5 B3/4 4-1
5.0	Design Features	5-4
6.0	Admin Controls	6-24

DEFINITIONS

Core Operating Limits Report

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR

INDEX

BASES

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

1.8 NOT USED

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

INSERT A

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

INSERT A

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure~~s~~ 2.1-1 ~~and 2.1-2~~ for 4 ~~and 3~~ loop operation~~, respectively~~.

APPLICABILITY: MODES 1 and 2.

ACTION:

Move to p. 2-3

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

INSERT
B
HERE

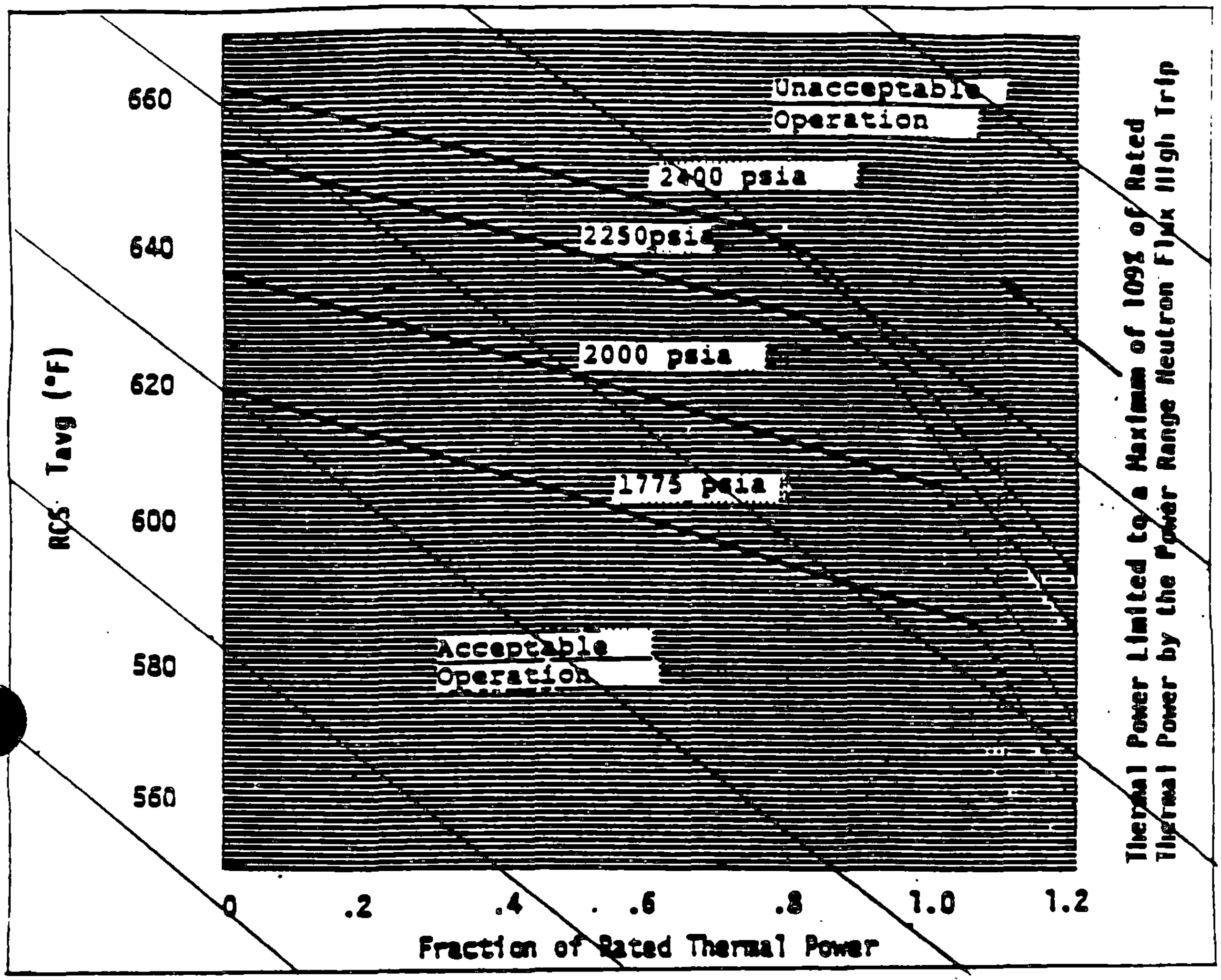
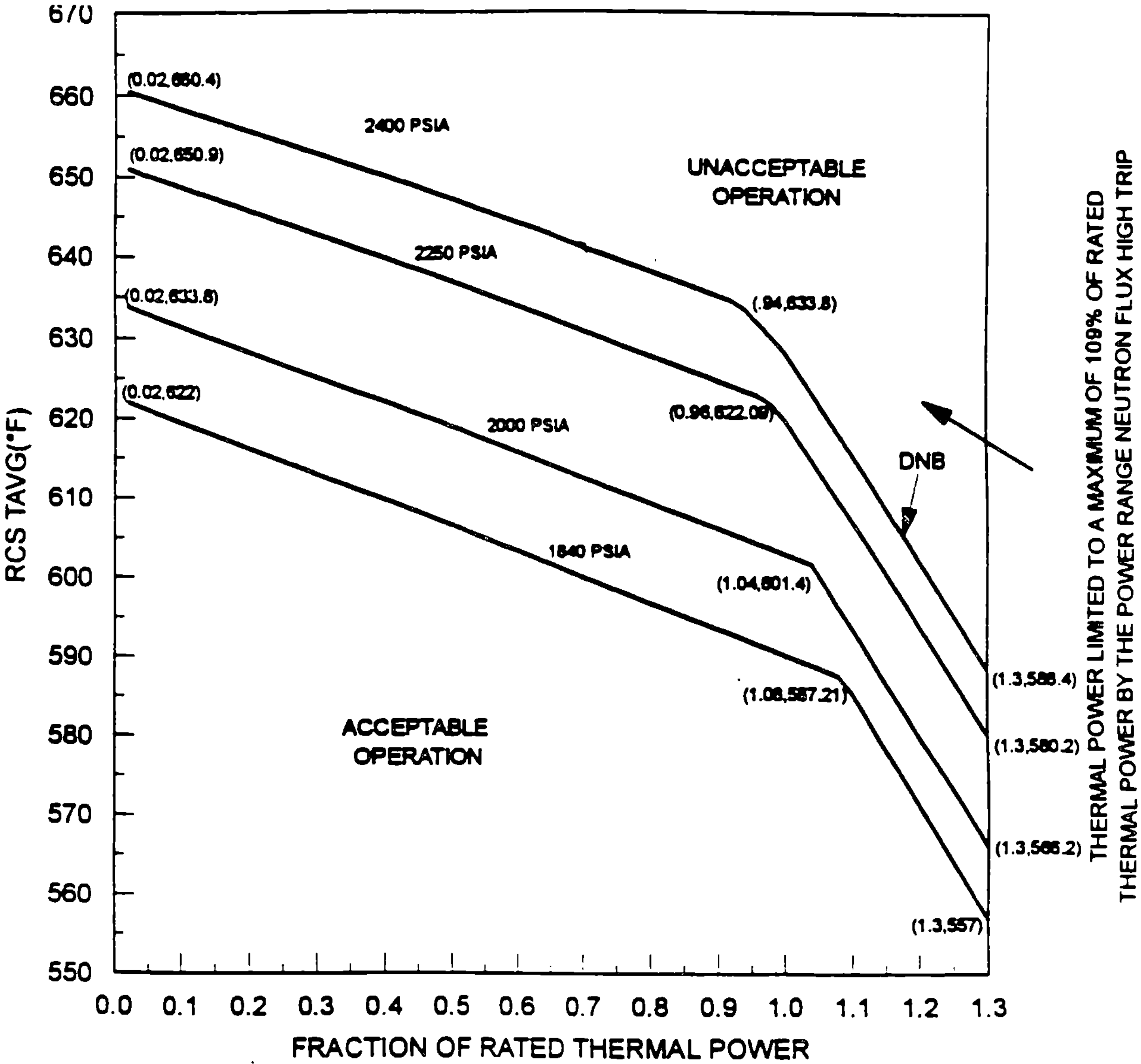


FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

INSERT B



RCS TAVG(°F)
670
660
650
640
630
620
610
600
590
580
570
560
550
(0.02,660.4)
2400 PSIA
(0.02,650.9)
2250 PSIA
(0.02,633.8)
2000 PSIA
(0.02,622)
1840 PSIA
UNACCEPTABLE OPERATION
(.94,633.8)
(0.96,622.09)
(1.04,601.4)
(1.08,587.21)
DNB
(1.3,588.4)
(1.3,580.2)
(1.3,566.2)
(1.3,557)
ACCEPTABLE OPERATION
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
FRACTION OF RATED THERMAL POWER
THERMAL POWER LIMITED TO A MAXIMUM OF 109% OF RATED THERMAL POWER BY THE POWER RANGE NEUTRON FLUX HIGH TRIP

FIGURE 2.1-2 REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

FIGURE 2.1-2 INTENTIONALLY LEFT BLANK PENDING

COMMISSION APPROVAL OF THREE LOOP OPERATION

REACTOR COOLANT SYSTEM PRESSURE

FROM Pg 2-1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not applicable	Not applicable
2. Power Range, Neutron Flux	Low setpoint - ≤ 25 % of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERM 'L POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is ~~86,430~~ gpm per loop.

82,500

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \frac{1+\tau_1 S}{1+\tau_2 S}(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

Indicated

T' = ~~Reference~~ T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

P = Pressurizer pressure, psig

P' = 2235 psig (Indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 30 secs, τ_2 = 4 secs.

S = Laplace transform operator, Sec^{-1}.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops

K_1 = ~~1.164~~ ← $\boxed{1.22}$

K_2 = ~~0.01434~~ ← $\boxed{0.02037}$

K_3 = ~~0.00073~~ ← $\boxed{0.001020}$

and f_1 (ΔI) is 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 such that:

(i) for $q_t - q_b$ between -23 percent and ~~+10~~ $\boxed{+13}$ percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -23 percent, the ΔT trip setpoint shall be automatically reduced by 1.26 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~+10~~ $\boxed{+13}$ percent, the ΔT trip setpoint shall be automatically reduced by ~~1.34~~ $\boxed{2.63}$ percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T - T") - f_2(\Delta I)]$

where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T"$ = Indicated ~~Reference~~ T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

K_4 = ~~1.088~~ 1.09

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00149 ~~0.00115~~/°F for $T > T"$; $K_6 = 0$ for $T \leq T"$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator, Sec^{-1}.

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.1 ~~3.1~~ percent.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.1 ~~3.0~~ percent.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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.

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 ~~the W-3, R-Grid correlation for standard (LOPAR) fuel assemblies and WRB-1 correlations for Vantage 5H fuel assemblies. The DNB correlation has~~ 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.

correlations which have

~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 or W-3, R-Grid correlation). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR correlation limit (1.17 for the WRB-1 or 1.30 for the W-3 R-Grid).~~

INSERT C HERE

The curves of Figure~~s~~ 2.1-1 ~~and 2.1-2~~ show s the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

$F^{RTP}_{\Delta H}$

The curves are based on an enthalpy hot channel factor, ~~$F^N_{\Delta H}$ of 1.55~~ and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

~~$F^N_{\Delta H} \leq 1.55 \, [1.0+0.3(1-P)]$~~

~~where P is the fraction of RATED THERMAL POWER~~

INSERT D HERE

These limiting heat flux conditions are higher than those calculated for the range of all control rod~~s~~ positions from FULLY WITHDRAWN to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power

INSERT C

The DNB design basis is as follows: uncertainties in the WRB-1 and WRB-2 correlations, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered stastically such that there is at least a 95 percent probability with 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

INSERT D

$$F^{N}_{\Delta H} = F^{RTP}_{\Delta H} \; [1.0 + PF_{\Delta H} \; (1.0 - P)]$$

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the COre Operating Limits Report (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^{N}_{\Delta H}$ specified in the COLR, and P is $\dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

LIMITING SAFETY SYSTEM SETTINGS

BASES

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature delta T setpoint. Three loop operation above the 4 loop P-8 setpoint has not been evaluated and is not permitted ~~is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature delta T channels and raising the P-8 setpoint to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level~~.

Overpower Delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients. ~~and anticipated transients when 3 loops are in operation and the Overtemperature delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients with 3 loops in operation.~~

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~1.6%~~ 1.3% delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than ~~1.6%~~ 1.3% delta k/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1.6%~~ 1.3% delta k/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits ~~of~~ in the COLR per Specification 3.1.3.5.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits ~~of~~ in the COLR per Specification 3.1.3.5.

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit ~~of~~ Specification 3.1.3.5. in the COLR per

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

INSERT E HERE

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

~~a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.~~

~~b. Less negative than -4.4 x 10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~

Beginning of Cycle Life (BOL) Limit

APPLICABILITY: ~~Specification 3.1.1.3.a~~ - MODES 1 and 2* only#
~~Specification 3.1.1.3.b~~ - MODES 1, 2 and 3 only#

End of Cycle Life (EOL) Limit

ACTION:

a. With the MTC more positive than the BOL limit ~~of 3.1.1.3.a. above,~~ specified in the COLR operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than ~~0 delta k/k/°F~~ the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of in the COLR per Specification 3.1.3.~~6~~ 5

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the EOL limit ~~of 3.1.1.3.b. above,~~ specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With Keff greater than or equal to 1.0

#See Special Test Exception 3.10.3

INSERT E

within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less positive than or equal to 0 $\Delta k/k/°F$.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit ~~of Specification 3.1.1.3.a, above~~ prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

The 300 ppm surveillance limit specified in the COLR

b. The MTC shall be measured at any THERMAL POWER and compared to ~~-3.7×10^{-4} delta k/k/°F~~ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~-3.7×10^{-4} delta k/k/°F~~, the MTC shall be remeasured, and compared to the EOL MTC limit ~~of specification 3.1.1.3.b~~ at least once per 14 EFPD during the remainder of the fuel cycle.

specified in the COLR

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or mis-aligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or mis-aligned from its group step counter demand position by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

in the COLR per

2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of ~~Figures 3.1-1 and 3.1-2~~; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation, or

of Specification 3.1.3.5

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

POSITION INDICATION SYSTEM SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as shown in ~~Figures 3.1-1 and 3.1-2.~~ specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1*, and 2*#

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the ~~above figures, or~~ insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours by use of the group demand counters and verified by the analog rod position indicators** except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours**.

*See Special Test Exceptions 3.10.2 and 3.10.3

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

#With Keff greater than or equal to 1.0

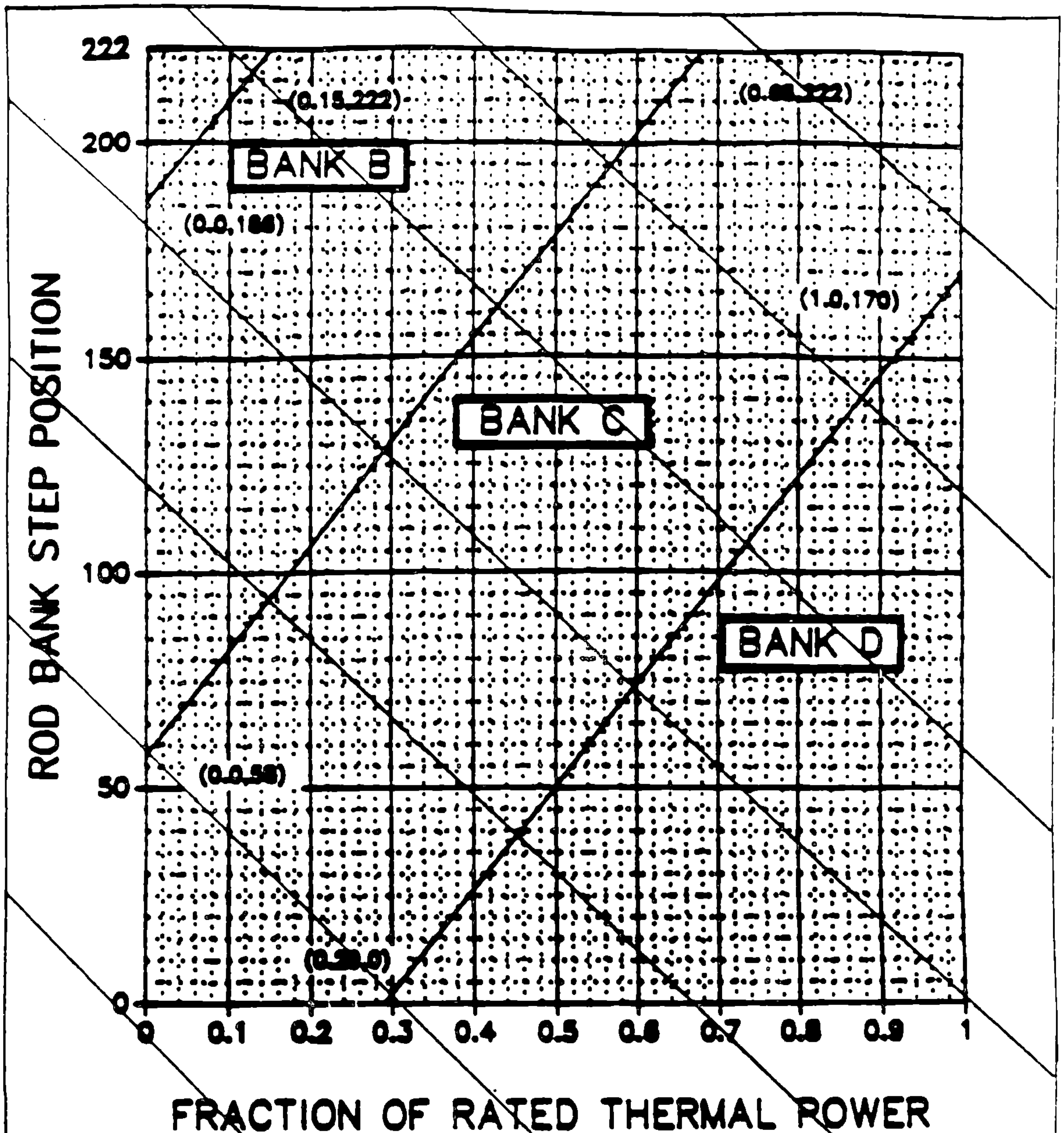


Figure 3.1-1
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

FIGURE 3.1-2

INTENTIONALLY LEFT BLANK PENDING
COMMISSION APPROVAL OF THREE LOOP OPERATION

FIGURE 3.1-2 ROD BANK INSERTION LIMITS VERSUS THERMAL POWER FOR THREE LOOP OPERATION

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

the

3.2.1 The indicated AXIAL FLUX DIFFERENCE shall be maintained within ~~± +6, -9%~~ target band ~~(flux difference units)~~ about the target flux difference.

as specified in the CORE OPERATING LIMITS REPORT (COLR)

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ~~+6, -9%~~ target band about the target flux difference and with THERMAL POWER:

1. Above 90% of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. Between 50% and 90% of RATED THERMAL POWER:

as specified in the COLR

   a) POWER OPERATION may continue provided:

      1) The ~~indicated~~ AFD has not been outside of the ~~+6, -9%~~ target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits ~~shown on Figure 3.2-1~~. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits ~~of Figure 3.2-1~~. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ~~+6, -9%~~ target band as specified in the COLR and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ~~+6, -9%~~ target band as specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours. Power increases above 50% of RATED THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ~~+6, -9%~~ target band when at least 2 or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ~~+6, -9%~~ target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

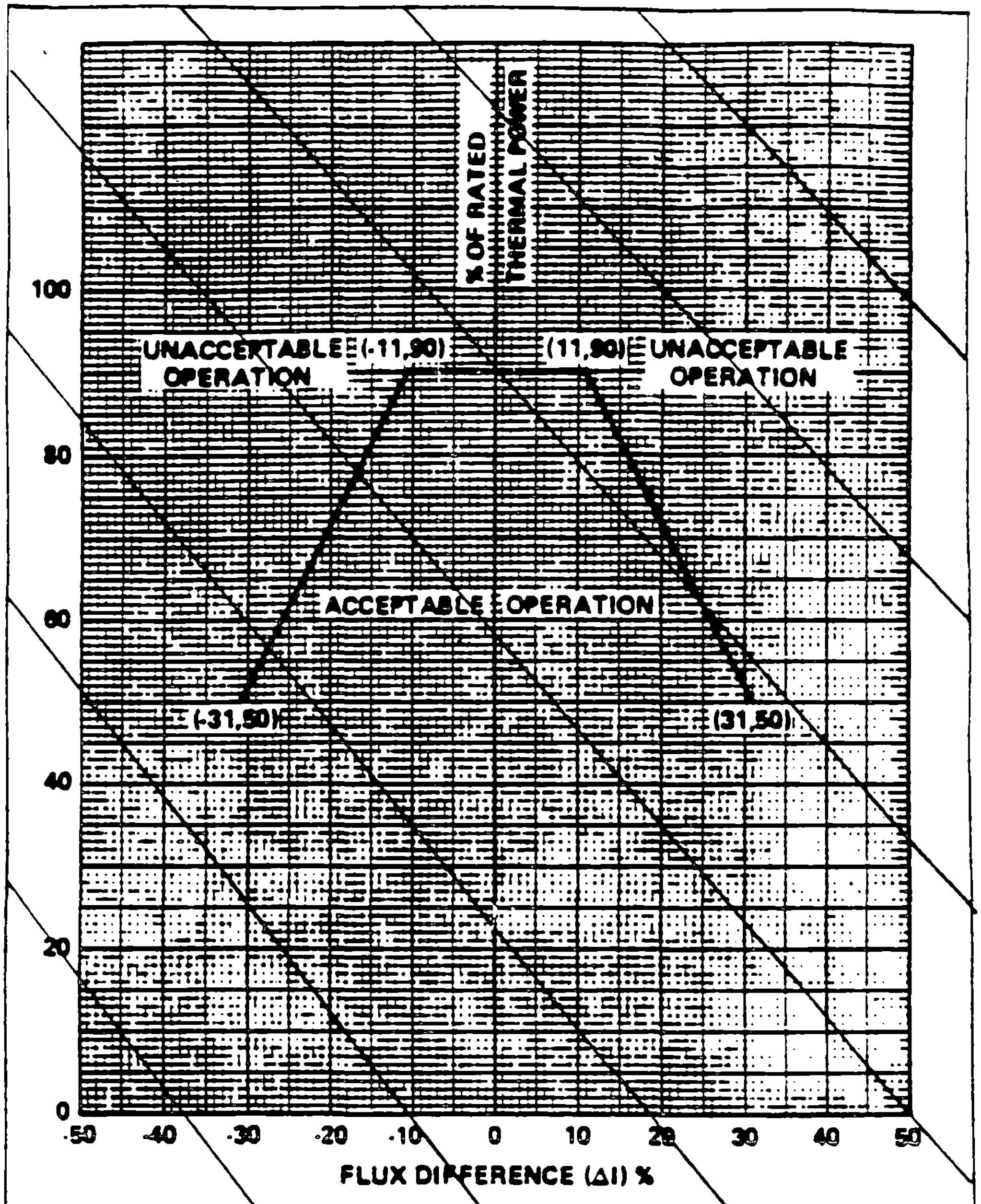


FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

INSERT F HERE

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

~~3.2.2 $F_Q(Z)$ shall be limited by the following relationships:~~

~~$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)]$ for $P > 0.5$~~

~~$F_Q(Z) \leq [(4.64)] [K(Z)]$ for $P \leq 0.5$~~

~~where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$~~

~~and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.~~

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

INSERT F

$$F_Q(z) \le \frac{F_Q^{RTP}}{P} * K(z) \text{ for } P > 0.5, \text{ and}$$

$$F_Q(z) \le \frac{F_Q^{RTP}}{0.5} * K(z) \text{ for } P \le 0.5,$$

Where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}, \text{ and}$$

K(z) = the normalized $F_Q(z)$ as a function of core height as specified in the COLR.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e. and f., below, and

2. The relationship:

PF$_{xy}$

$$F_{xy}^L = F_{xy}^{RTP} \; [1 + \boxed{0.2}(1-P)]$$

PFxy is the power factor multiplier for Fxy in the COLR

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in ~~a Radial Peaking Factor Limit Report~~ per specification 6.9.1.9.

the COLR

f. The F_{xy} limits of e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0% to 15%, inclusive.
2. Upper core region from 85% to 100%, inclusive.
3. Grid plane regions at 17.8% ± 2%, 32.1% ± 2%, 46.4% ± 2%, 60.6% ± 2% and 74.9% ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

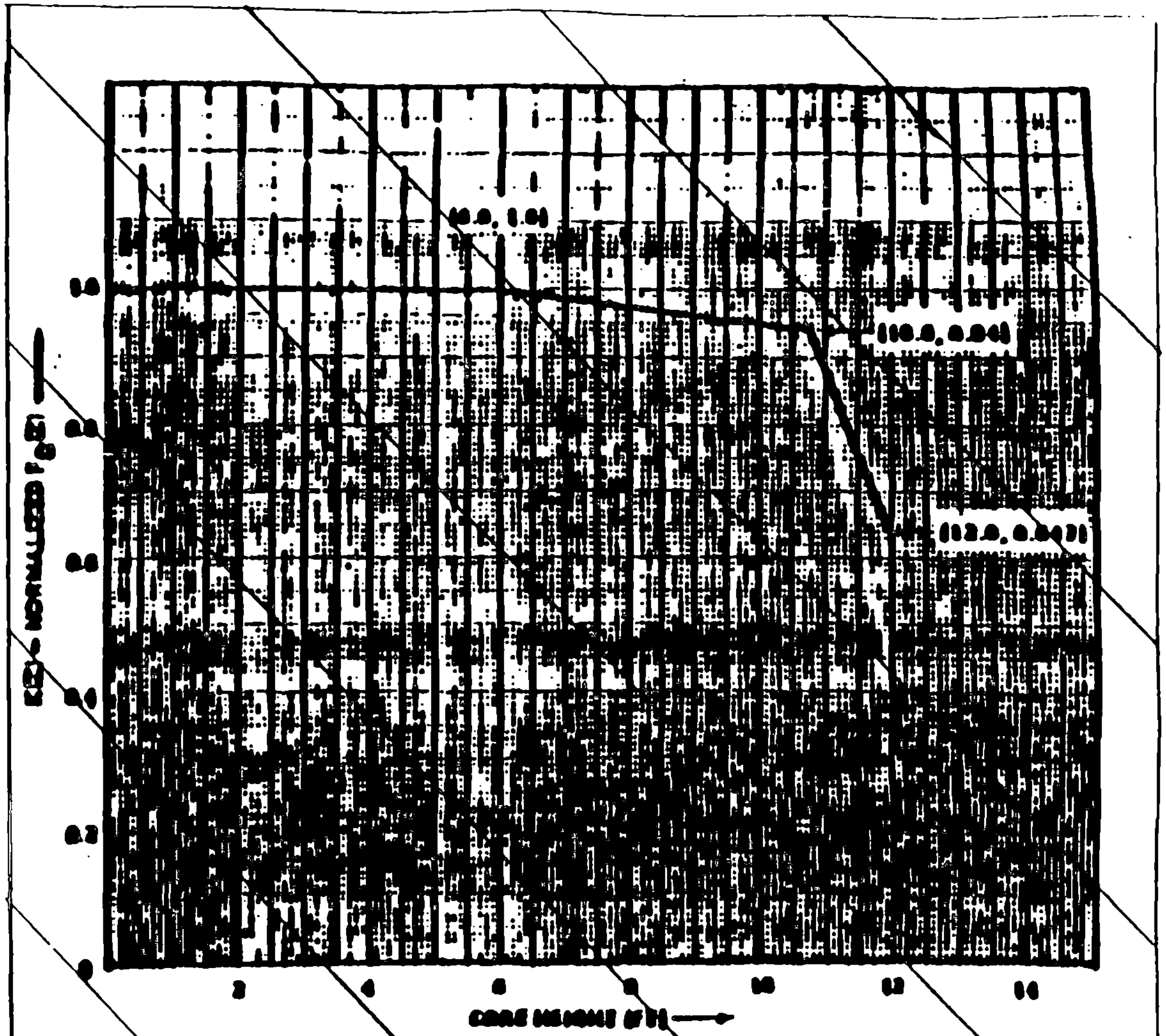


FIGURE 3.2-3

K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$\cancel{F^N_{\Delta H} \leq 1.55 \, [1.0 + 0.3 \, (1.0 - P)]}$$

~~where: P = THERMAL POWER / RATED THERMAL POWER~~

INSERT D HERE

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

b. Demonstrate thru in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b. above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

INSERT D

$$F^N_{\Delta H} = F^{RTP}_{\Delta H} \; [1.0 + PF_{\Delta H}(1.0 - P)]$$

Where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR).

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR, and P is $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
	4 Loops in Operation
Reactor Coolant System T_{avg}	≤ ~~582°F~~ 582.9°F
Pressurizer Pressure	2200 ≥ ~~2220~~ psia*
Reactor Coolant System Total Flow Rate	≥ ~~353,700~~ gpm# ~~344,000~~ 341,000

---

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

\# Includes a ~~2.2%~~ 2.4% flow uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3% ~~1.6%~~ $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the accident and transient analyses.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

END OF CYCLE LIFE (EOL)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analysis to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR analysis to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed between EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in normal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR analysis into the limiting MTC value ~~of -4.4 x 10^-4 delta k/k/°F. The MTC value of -3.7 x 10^-4 delta k/k/°F represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value -4.4 x 10^-4 delta k/k/°F.~~

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains with its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

The 300 ppm surveillance limit MTC value represents a conservative value at a core condition of 300 ppm equilibrium boron concentration that is obtained by correcting the limiting EOL MTC for burnup and boron concentration.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature ≥ 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of ~~1.6%~~ 1.3% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability (minimum boration volume) requirement is established to conservatively bound expected operating conditions throughout core operating life. The analysis assumes that the most reactive control rod is not inserted into the core. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires borated water from a boric acid tank in accordance with TS Figure 3.1-2, and additional makeup from either: (1) the second boric acid tank and/or batching, or (2) a maximum of 41,800 gallons of 2,300 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 73,800 gallons of 2,300 ppm borated water is required. However, to be consistent with the ECCS requirements, the RWST is required to have a minimum contained volume of 350,000 gallons during operations in MODES 1, 2, 3 and 4.

The boric acid tanks, pumps, valves, and piping contain a boric acid solution concentration of between 3.75% and 4% by weight. To ensure that the boric acid remains in solution, the tank fluid temperature and the process pipe wall temperatures are monitored to ensure a temperature of 63°F. or above is maintained. The tank fluid and pipe wall temperatures are monitored in the main control room. A 5°F margin is provided to ensure the boron will not precipitate out.

Should ambient temperature decrease below 63°F, the boric acid tank heaters, in conjunction with boric acid pump recirculation, are capable of maintaining the boric acid in the tank and in the pump at or about 63°F. A small amount of boric acid in the flowpath between the boric acid recirculation line and the suction line to the charging pump will precipitate out, but it will not cause flow blockage even with temperatures below 50°F.

With the RCS temperature below 350°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE OPERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

MEETING THE DNB DESIGN CRITERIA

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) ~~maintaining the minimum DNBR in the core greater than or equal to 1.30~~ during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ 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.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope ~~of 2.32~~ times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions with the part length control rods withdrawn from the core. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

in the COLR per Specification 3.2.1

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ~~+6, -9%~~ target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits ~~of Figure 3.2-1~~ while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

specified in the COLR

Provisions for monitoring the AFD are derived from the plant nuclear instrumentation system through the AFD Monitor Alarm. A control room recorder continuously displays the auctioneered high flux difference and the target band limits as a function of power level. An alarm is received any time the auctioneered high flux difference exceeds the target band limits. Time outside the target band is graphically presented on the strip chart.

Figure B 3/4 2-1 shows a typical monthly target band.

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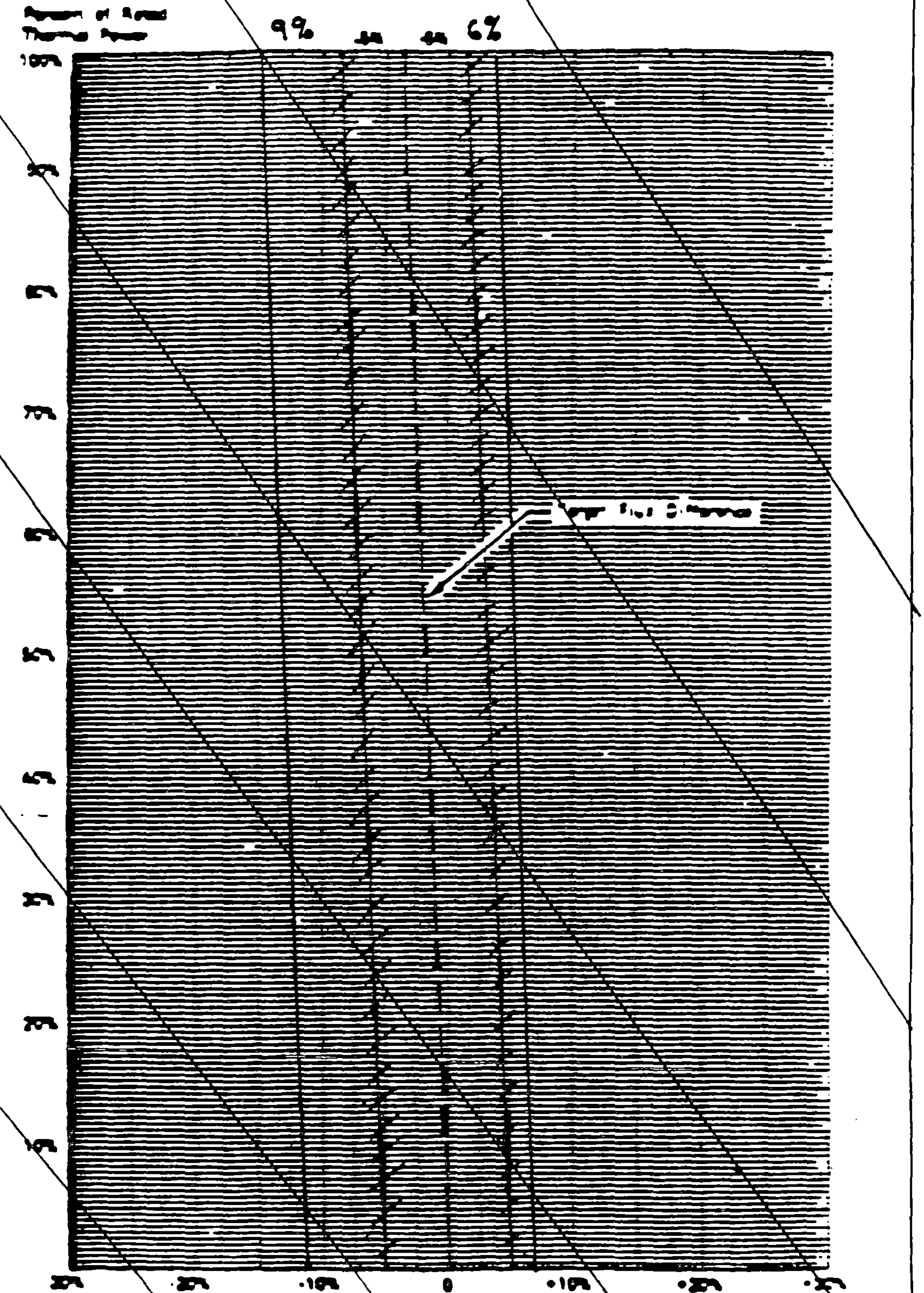

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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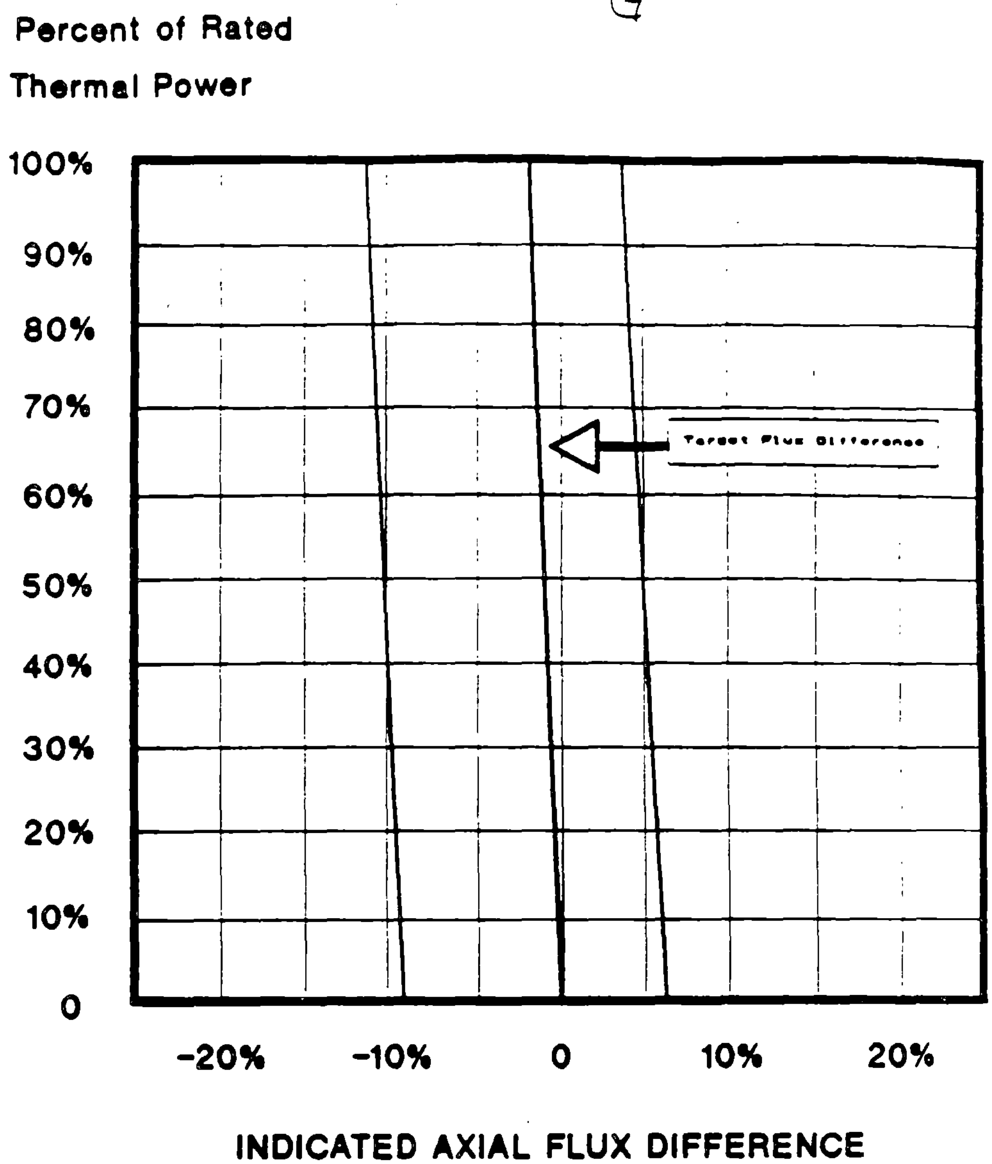


Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

* REFER TO COLR FIGURE 2 FOR ACTUAL LIMITS

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL AND RADIAL PEAKING FACTORS - $F_Q(Z)$ AND $F_{\Delta H}^N$

The limits on heat flux and nuclear enthalpy hot channel factors and RCS flow rate ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F_{\Delta H}^N$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F_{\Delta H}^N$ will be maintained within its limits provided conditions a thru d above, are maintained.

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 incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

When $F_{\Delta H}^N$ is measured, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system. The specified limit for $F_{\Delta H}^N$ also contains an 8% allowance for uncertainties which mean that normal operation will result in $F_{\Delta H}^N \leq$ ~~1.55~~ $\boxed{F_{\Delta H}^{RPT}}$/1.08. The 8% allowance is based on the following considerations:

Where $F_{\Delta H}^{RPT}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMIT REPORT (COLR).

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL AND RADIAL PEAKING FACTORS - $F_Q(Z)$ AND $F^N_{\Delta H}$ (Continued)

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q.

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics test can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less rapidly available.

The radial peaking factor $F_{xy}(Z)$ is measured periodically to provide assurance that the hot channel factor $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RPT}_{xy}), as provided in the ~~Radial Peaking Factor Limit report~~ COLR per specification 6.9.1.9, was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

meet the DNB design criteria

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, ~~and maintain DNBR above 1.30~~ during all normal operations and anticipated transients. In MODES 1 and 2 with less than all coolant loops in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal for removing decay heat; but, single failure considerations require all loops be in operation whenever the rod control system is energized and at least one loop be in operation when the rod control system is deenergized.

In MODE 4, a single reactor coolant loop or RHR loop provides sufficient heat removal for removing decay heat; but, single failure considerations require that at least 2 loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires that two RHR loops be OPERABLE.

In MODE 5, single failure considerations require that two RHR loops be OPERABLE. The provisions of Sections 3.4.1.4 and 3.9.8.2 [paragraph (b) of footnote (*)] which permit one service water header to be out of service, are based on the following:

1. The period of time during which plant operations rely upon the provisions of this footnote shall be limited to a cumulative 45 days for any single outage, and

2. The Gas Turbine shall be operable, as a backup to the diesel generators, in the event of a loss of offsite power, to supply the applicable loads. The basis for OPERABILITY is one successful startup of the Gas Turbine no more than 14 days prior to the beginning of the Unit outage.

The operation of one Reactor Coolant Pump or one RHR Pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during Boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with Boron concentration reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 312°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer (thereby providing a volume into which the primary coolant can expand, or (2) by restricting the starting of Reactor Coolant Pumps to those times when secondary water temperature in each steam generator is less than 50°F above each of the RCS cold leg temperatures.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 47 psig and an air temperature of 271°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain at least 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN FEATURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirement specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is ~~12,811 ± 100~~ cubic feet at a nominal T_{avg} of ~~581.0~~°F.

12,446 ± 426

573.0

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list of descriptions of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

INSERT H HERE

RADIAL PEAKING FACTOR LIMIT REPORTS

6.9.1.9 The F_{xy} limits for Rated Thermal Power (F_{xy}^{RTP}) for all core planes containing bank "D" control rods and all unrodded core planes, and the plot of predicted heat flux hot channel factor times relative power ($F_Q^T * P_{REL}$) vs. Axial Core Height with the limit envelope shall be provided to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector. The Report shall be provided to the Commission upon issuance.

In addition, in the event that the limit should change, requiring a new submittal or submittal of an amended Peaking Factor Limit Report, it will be submitted upon issuance.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

INSERT H

6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a. Approved by Safety Evaluation dated May 28, 1985.

2. WCAP-8385, Power Distribution Control and Load Following Procedures - Topical Report, September 1974 (W Proprietary) Methodology for Specification 3/4.2.1 Axial Flux Difference. Approved by Safety Evaluation dated January 31, 1978.

3. WCAP-10054-P-A, Rev. 1, Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code, August 1985 (W Proprietary), Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved for Salem by NRC letter dated August 25, 1993.

4. WCAP-10266-P-A, Rev. 2, The 1981 Version of Westinghouse Evaluation Model Using BASH Code, Rev. 2. March 1987 (W Proprietary) Methodology for Specification 3/4.2.2 Heat Flux Hot Channel Factor. Approved by Safety Evaluation dated November 13, 1986.

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

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

Enclosure 1

SAMPLE COLR

SALEM UNIT 1, CYCLE 13

AND

SALEM UNIT 2, CYCLE 9

SAMPLE

SALEM GENERATING STATION UNIT 1 CYCLE 13

CORE OPERATING LIMITS REPORT

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Salem Unit 1 Cycle 13 has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The Technical Specifications affected by this report are listed below:

3/4.1.1.4	Moderator Temperature Coefficient
3/4.1.3.5	Control Rod Insertion Limits
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor
3/4.2.3	Nuclear Enthalpy Hot Channel Factor

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.9.

2.1 Moderator Temperature Coefficient (Specification 3/4.1.1.4)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less positive than 0 $\Delta k/k°F$.

The EOL/ARO/HZP-MTC shall be less negative than -4.7×10^{-4} $\Delta k/k°F$.

2.1.2 The MTC Surveillance limit is:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to:

-4.0×10^{-4} $\Delta k/k°F$.

where:

BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero THERMAL POWER
EOL stands for End of Cycle Life
RTP stands for RATED THERMAL POWER

2.2 Control Rod Insertion Limits (Specification 3/4.1.3.5)

2.1.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.3 Axial Flux Difference (Specification 3/4.2.1) {CAOC methodology)

2.3.1 The AXIAL FLUX DIFFERENCE (AFD) target band is +6%, -9%.

2.3.2 The AFD Acceptable Operation Limits are provided in Figure 2.

2.4 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3/4.2.2) {F_{xy} methodology)

$$F_Q(z) \leq \frac{F_Q^{RTP}}{P} * K(z) \text{ for } P > 0.5 \text{ and}$$

$$F_Q(z) \leq \frac{F_Q^{RTP}}{0.5} * K(z) \text{ for } P \leq 0.5.$$

Where: $P = \dfrac{THERMAL\ POWER}{RATED\ THERMAL\ POWER}$

2.4.1 $F_Q^{RTP} = 2.40$

2.4.2 K(Z) is provided in Figure 3.

$F_{xy}^{L} = F_{xy}^{RTP}\ [1.0 + PF_{xy}\ (1.0 - P)]$

2.4.3 Where: F_{XY}^{RTP} = ____[1] for the unrodded core planes
= ____[1] for the core plan containing Bank D control rods

$PF_{xy} = 0.3$

---

[1] Value to be determined during the RSE process

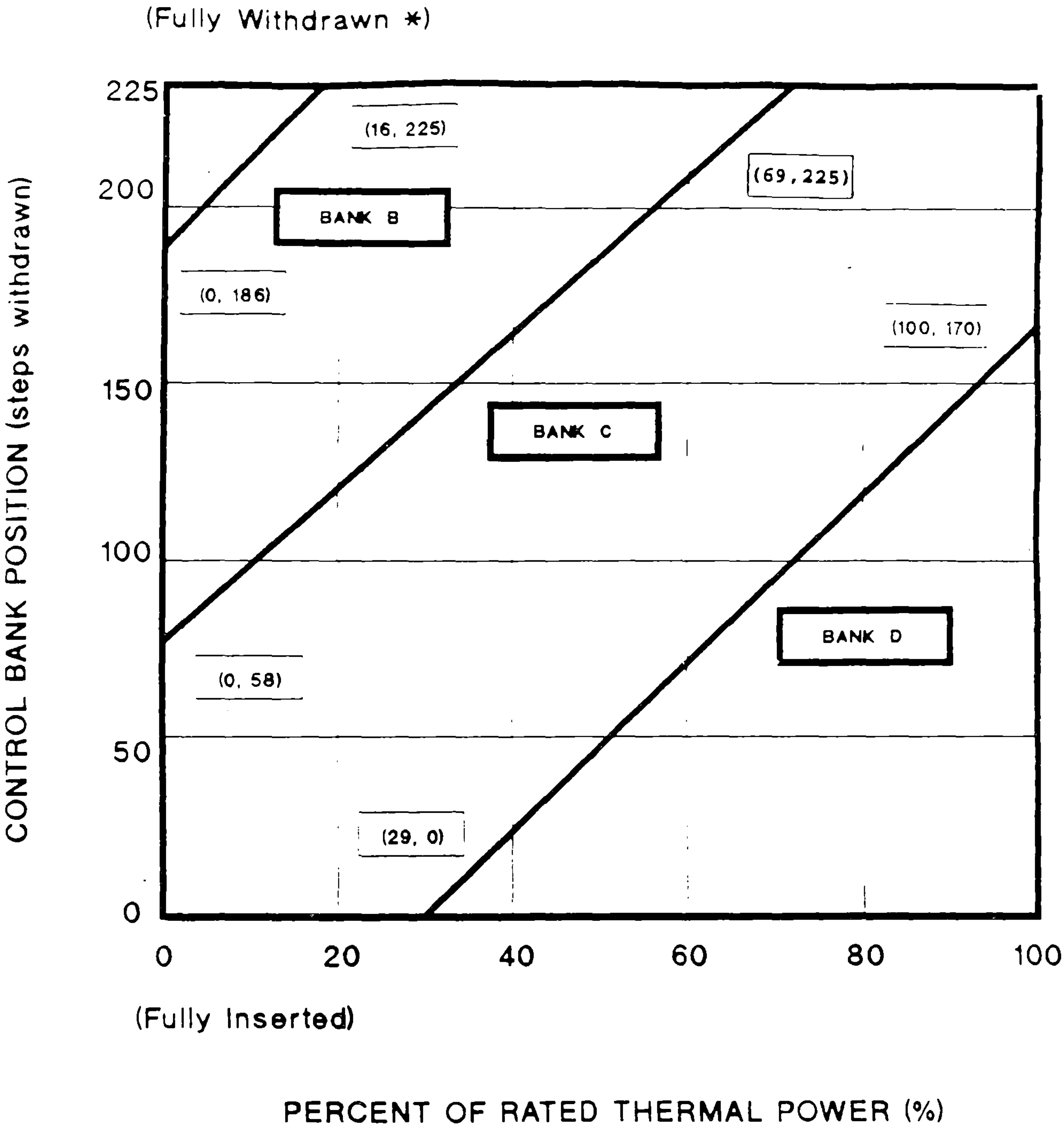
2.5 Nuclear Enthalpy Rise Hot Channel Factor - $F^{N}_{\Delta H}$ (Specification 3/4.2.3)

$$F^{N}_{\Delta H} = F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$$

Where:

2.5.1 $F^{RTP}_{\Delta H} = 1.65$

2.5.2 $PF_{\Delta H} = 0.3$



* Fully withdrawn for the current cycle shall be the condition where control rods are at a position of 225 steps withdrawn. Withdrawal to 228 steps is permitted during rod drop time measurements and rod position indicator calibration.

FIGURE 1

ROD BANK INSERTION LIMIT VERSUS THERMAL POWER

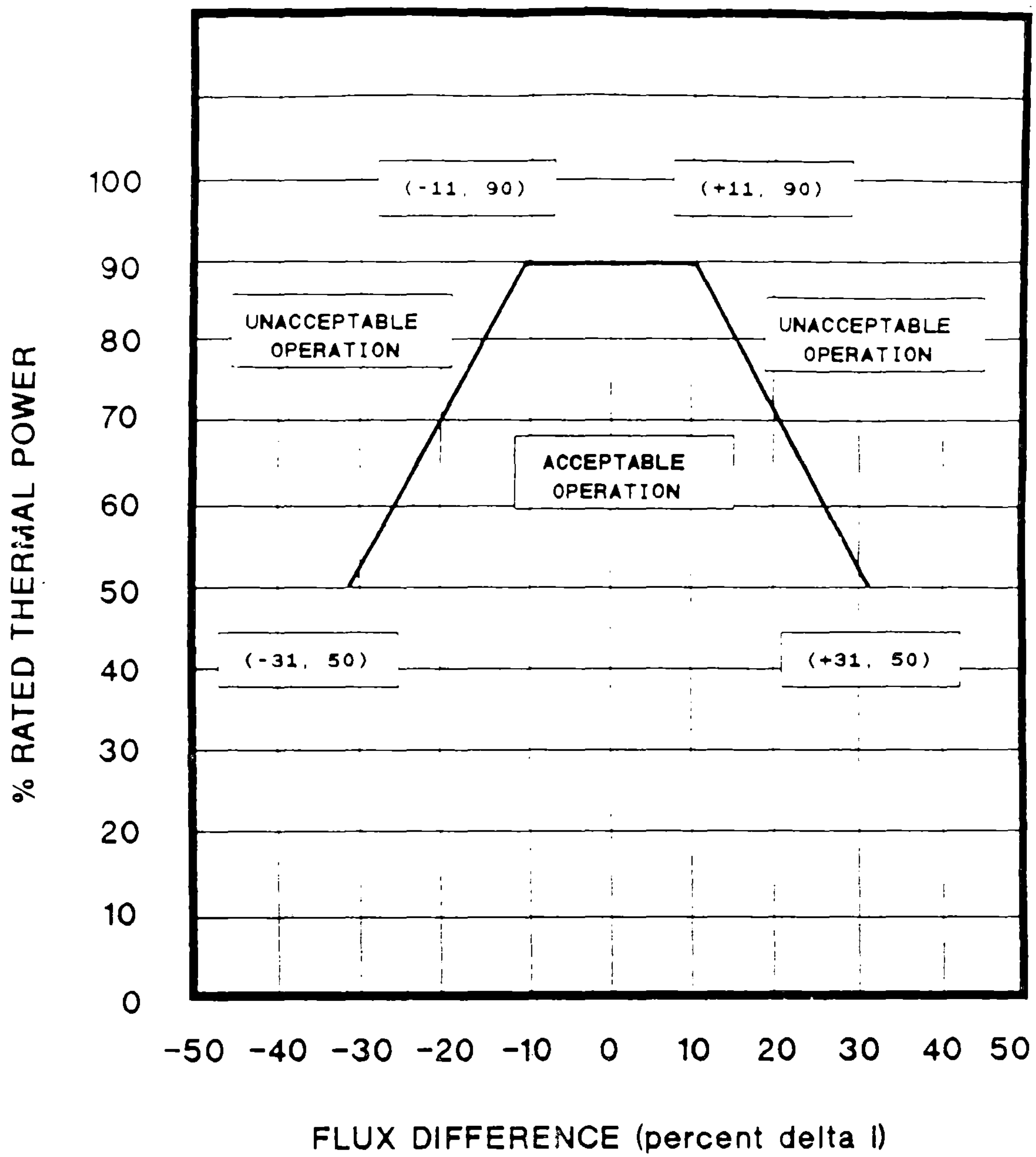


FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

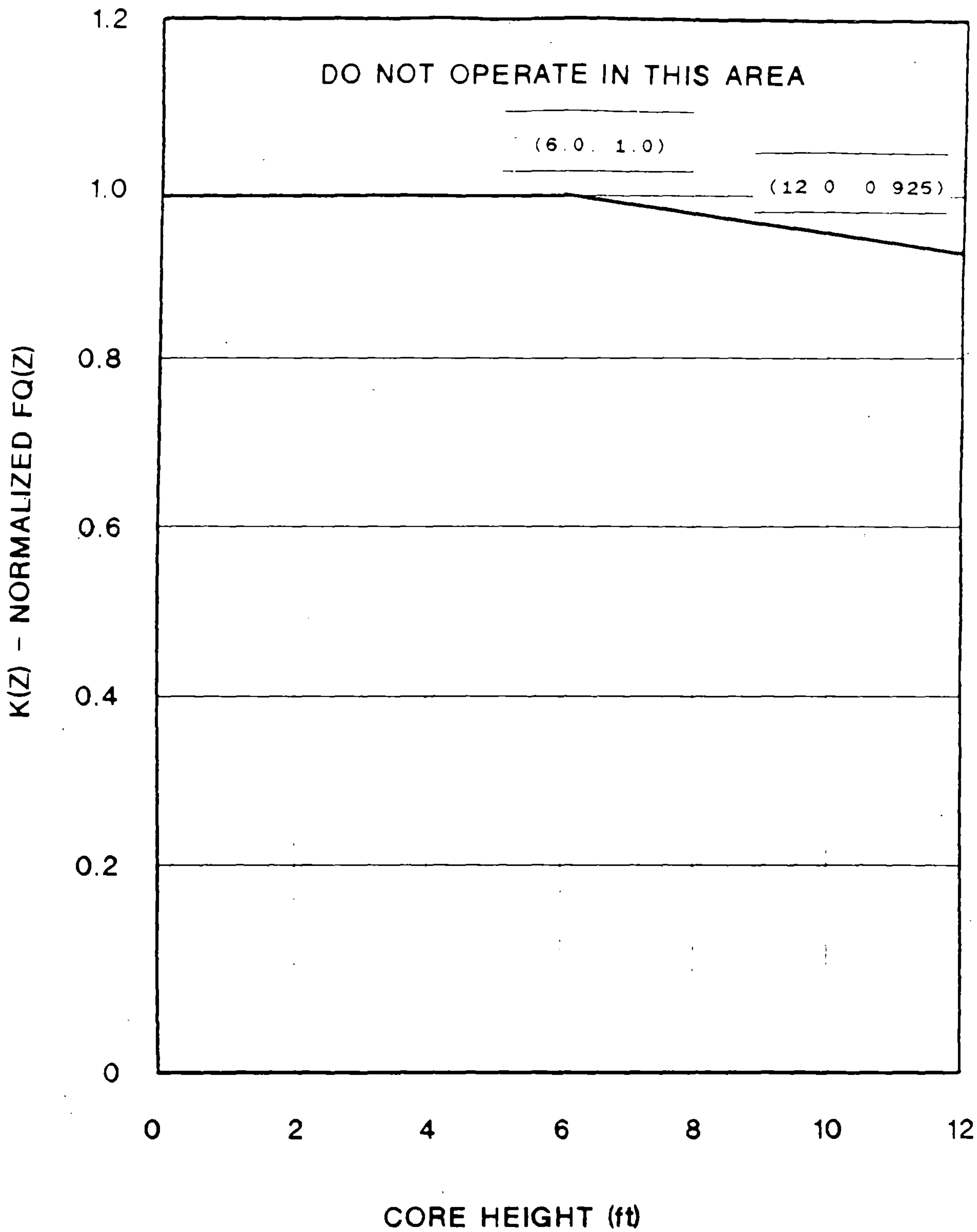


FIGURE 3

K(Z) – NORMALIZED FQ(Z) AS A FUNCTION OF CORE HEIGHT

SAMPLE

SALEM GENERATING STATION UNIT 2 CYCLE 9

CORE OPERATING LIMITS REPORT

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Salem Unit 2 Cycle 9 has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The Technical Specifications affected by this report are listed below:

3/4.1.1.3	Moderator Temperature Coefficient
3/4.1.3.5	Control Rod Insertion Limits
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor
3/4.2.3	Nuclear Enthalpy Hot Channel Factor

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using the NRC-approved methodologies specified in Technical Specification 6.9.1.9.

2.1 Moderator Temperature Coefficient (Specification 3/4.1.1.3)

2.1.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP-MTC shall be less positive than 0 $\Delta k/k°F$.

The EOL/ARO/HZP-MTC shall be less negative than $-4.7 \times 10^{-4}\ \Delta k/k°F$.

2.1.2 The MTC Surveillance limit is:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to:

$-4.0 \times 10^{-4}\ \Delta k/k°F$.

where:

BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero THERMAL POWER
EOL stands for End of Cycle Life
RTP stands for RATED THERMAL POWER

2.2 Control Rod Insertion Limits (Specification 3/4.1.3.5)

2.1.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.3 Axial Flux Difference (Specification 3/4.2.1)
{CAOC methodology)

2.3.1 The AXIAL FLUX DIFFERENCE (AFD) target band is +6%, -9%.

2.3.2 The AFD Acceptable Operation Limits are provided in Figure 2.

2.4 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3/4.2.2)
{F_{xy} methodology)

$$F_Q(z) \leq \frac{F_Q^{RTP}}{P} * K(z) \text{ for } P > 0.5 \text{ and}$$

$$F_Q(z) \leq \frac{F_Q^{RTP}}{0.5} * K(z) \text{ for } P \leq 0.5.$$

Where: $P = \dfrac{\mathit{THERMAL\ POWER}}{\mathit{RATED\ THERMAL\ POWER}}$

2.4.1 $F_Q^{RTP} = 2.40$

2.4.2 K(Z) is provided in Figure 3.

$$F_{xy}^{L} = F_{xy}^{RTP} \; [1.0 + PF_{xy} \; (1.0 - P)]$$

2.4.3 Where: F_{XY}^{RTP} = ____[1] for the unrodded core planes
= ____[1] for the core plan containing Bank D control rods

$PF_{xy} = 0.3$

[1] Value to be determined during the RSE process

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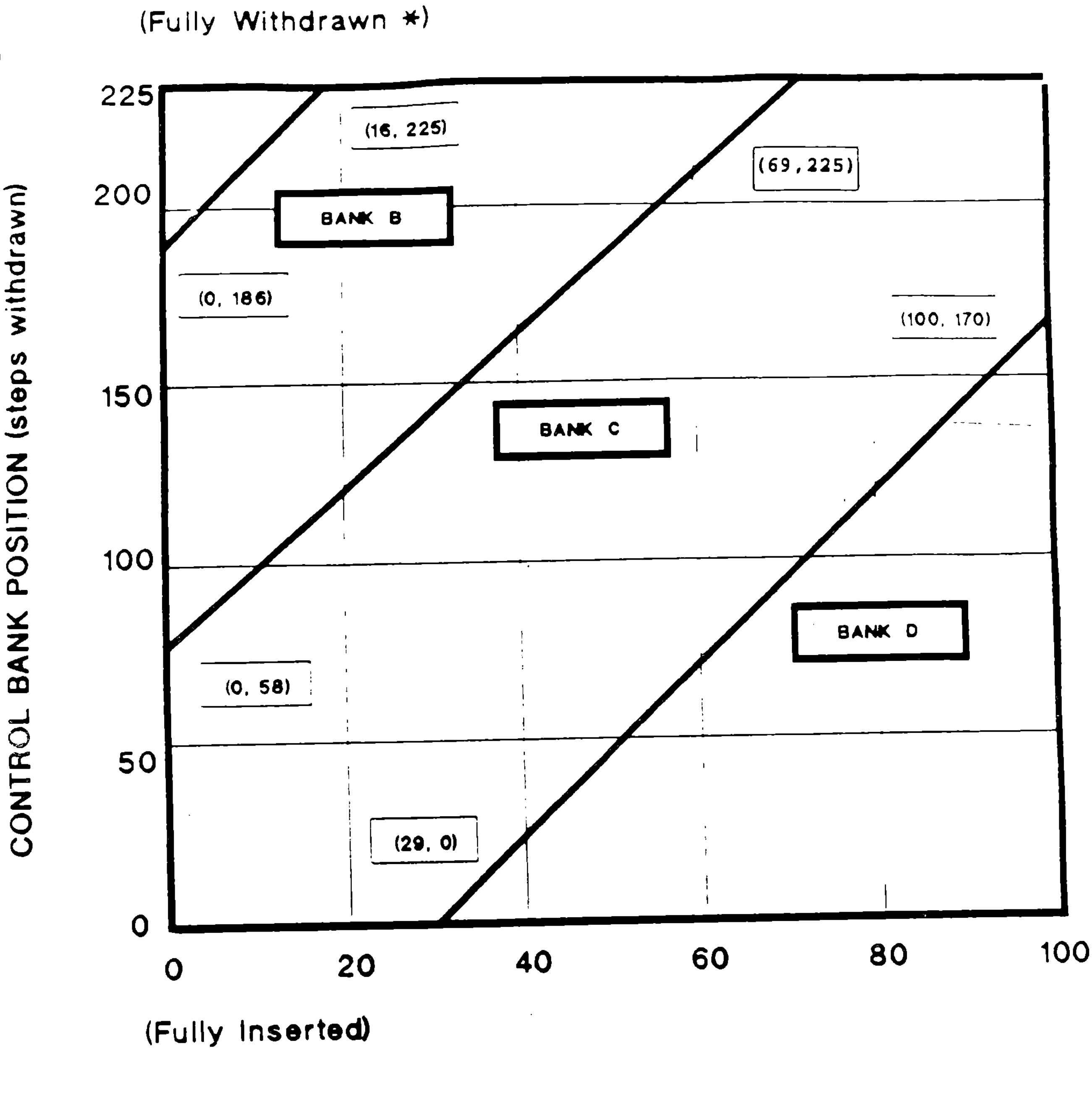
2.5 Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ (Specification 3/4.2.3)

$$F^N_{\Delta H} = F^{RTP}_{\Delta H} [1.0 + PF_{\Delta H} (1.0 - P)]$$

Where:

2.5.1 $F^{RTP}_{\Delta H} = 1.65$

2.5.2 $PF_{\Delta H} = 0.3$



* Fully withdrawn for the current cycle shall be the condition where control rods are at a position of 225 steps withdrawn. Withdrawal to 228 steps is permitted during rod drop time measurements and rod position indicator calibration.

FIGURE 1

ROD BANK INSERTION LIMIT VERSUS THERMAL POWER

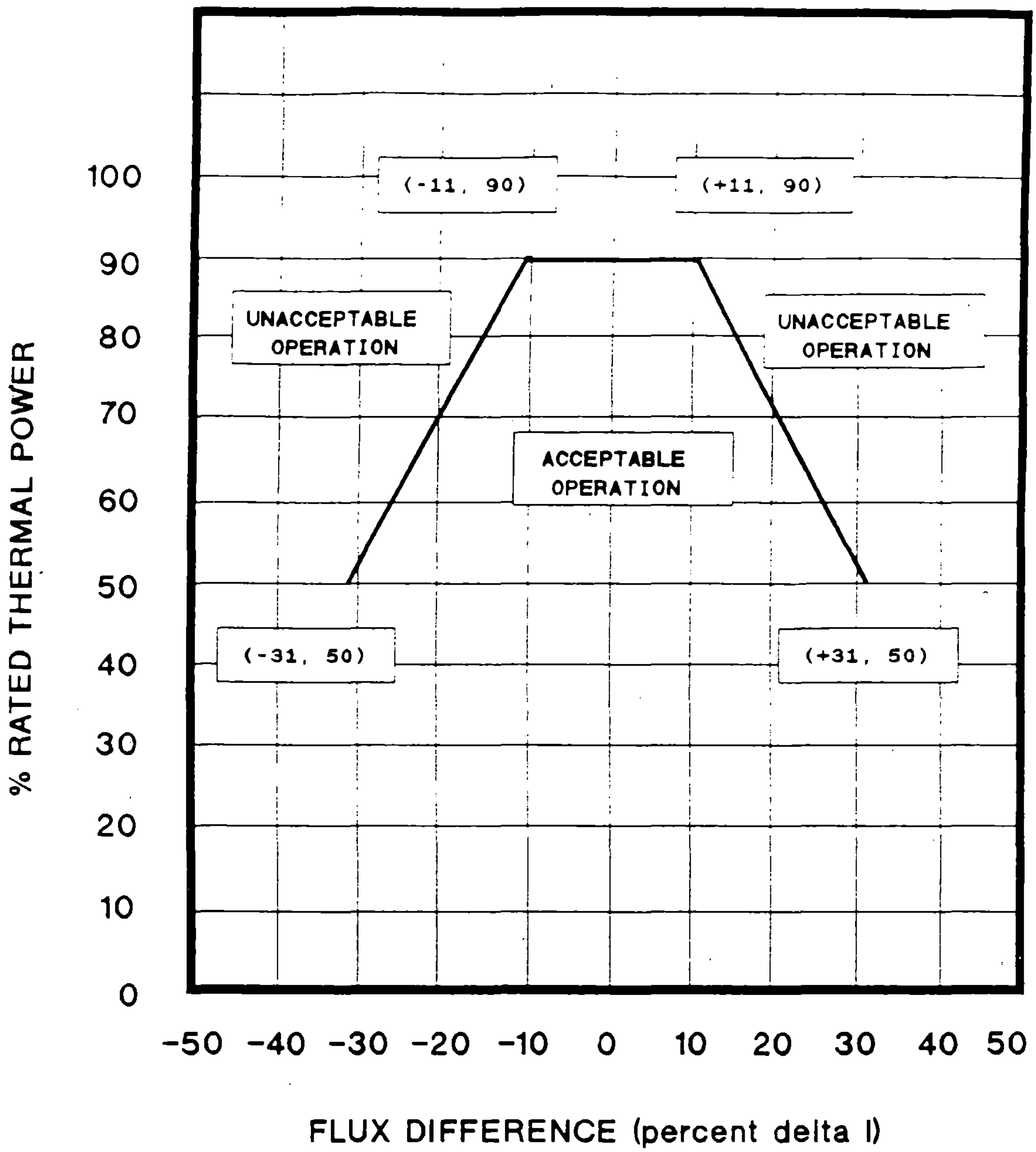


FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF

RATED THERMAL POWER

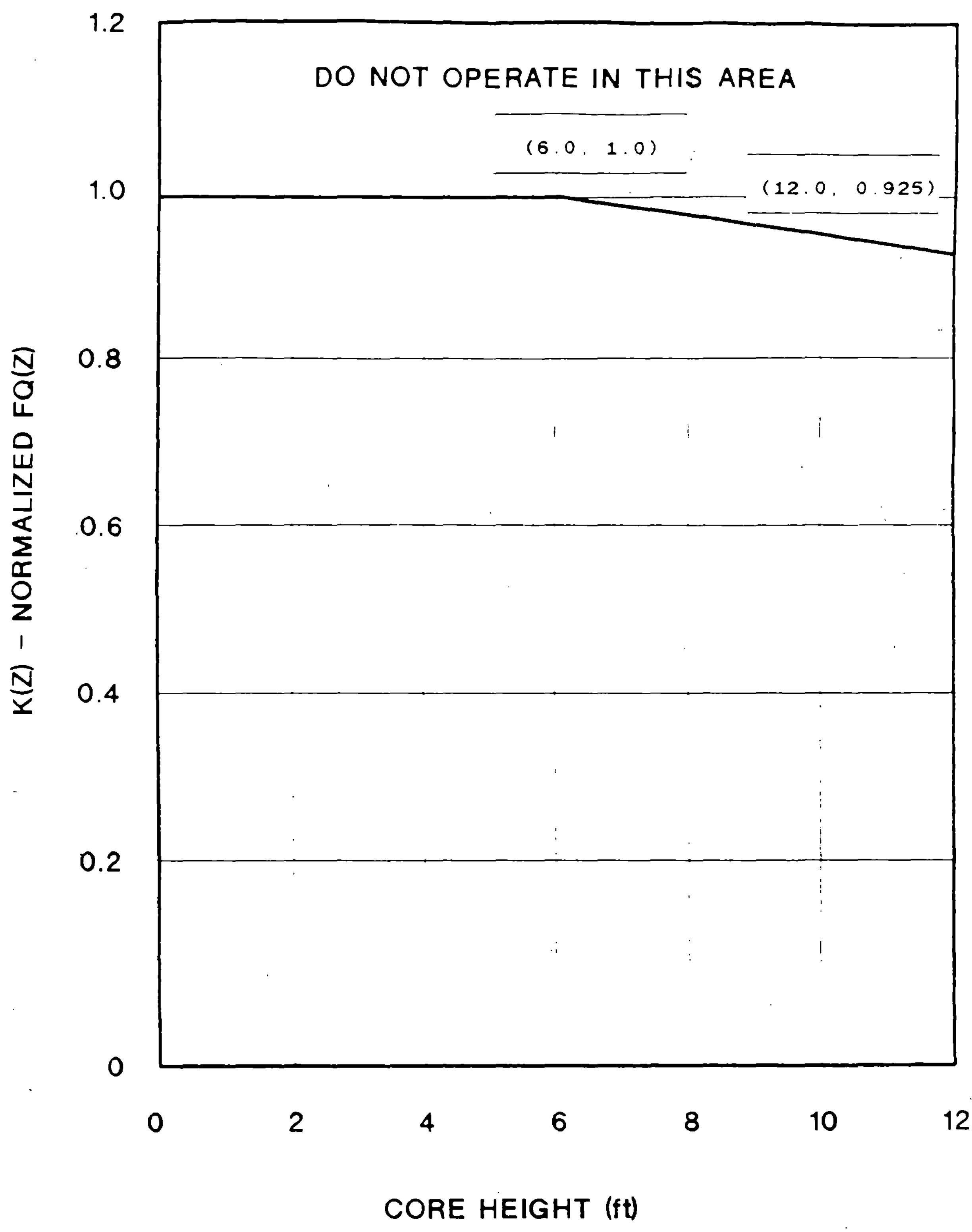


FIGURE 3

K(Z) – NORMALIZED FQ(Z) AS A FUNCTION OF CORE HEIGHT