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 WRB-1 correlation and the W-3 correlation for conditions outside the range of WRB-1 correlation. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is ~~indicative of the margin to DNB~~. R142

DESIGN DNBR LIMITS

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 correlation in this application). 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 limit. DESIGN R142

THE DESIGN

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, specified in the Core Operating Limit Report (COLR) 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: R159

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}[1+ PF_{\Delta H}(1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, R159

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

DESIGN DNBR

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis ~~(1.38)~~ and the ~~WRB-1 correlation~~ limit ~~(1.17)~~ to completely offset the rod bow penalty. R142

The applicable value of rod bow penalty is referenced in the FSAR. R159

Margin in excess of the rod bow penalty is available for plant design flexibility. R142

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor, W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function is specified in the COLR. R159

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 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.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R142

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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 WRB-1 correlation and the W-3 correlation for conditions outside the range of WRB-1 correlation. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. R130

The DNB design basis is as follows: DESIGN DNBR LIMIT. 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 correlation in this application). 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 DESIGN DNBR limit. R130

THE DESIGN The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. R104 R130

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, specified in the Core Operating Limit Report (COLR) 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: R146 R21

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}[1+ PF_{\Delta H} (1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR. R146

DESIGN DNBR

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis ~~(1.38)~~ and the ~~WRB-1 correlation~~ limit ~~(1.17)~~ to completely offset the rod bow penalty. R130

The applicable value of rod bow penalty is referenced in the FSAR. R146

Margin in excess of the rod bow penalty is available for plant design flexibility. R130

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function is specified in the COLR. R21 R146

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. R21

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.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R130

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. R21

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-14)

DESCRIPTION AND JUSTIFICATION FOR

DEPARTURE FROM NUCLEATE BOILING RATIO (DNBR) LIMITS BASES CHANGE

Description of Change

TVA proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 technical specifications (TSs) Bases Sections 2.1.1, 3/4.2.2, and 3/4.2.3. Bases Section 2.1.1 will have the reference to the departure from nucleate boiling (DNB) correlation limit replaced with the design departure from nucleate boiling ratio (DNBR) limit that will be calculated using the new Mini-Revised Thermal Design Procedure (Mini-RTDP). Additionally, the Bases Section 3/4.2.3 will delete the DNBR value of 1.38 used in the safety analyses and the WRB-1 correlation limit of 1.17. This will be replaced with a reference to the design DNBR limit.

Reason for Change

SQN is requesting this change as a result of using the Mini-RTDP to determine the DNBR margin as identified by Safety Evaluation Check List 91-451, Revision 2. This is a new methodology being applied at SQN. The standard thermal design procedure, which has been used to show acceptable conformance to DNBR limits for Updated Final Safety Analysis Report, Chapter 15 accident analysis for the current and previous SQN cycles, will be replaced with the Westinghouse Electric Corporation Mini-RTDP starting with the Cycle 7 core design. The Mini-RTDP method generates additional DNBR margin through a statistical combination of uncertainties. Use of this method provides sufficient DNBR margin to offset the DNBR margin utilized when the core-peaking factors are increased.

For future reload packages, SQN may elect to use a different staff-approved thermal design procedure. The proposed TS wording was established to provide that flexibility.

Justification for Change

The Mini-RTDP was reviewed and approved by NRC, and a staff evaluation was issued in 1989. This method conservatively satisfies the design criterion that protects against DNB in a pressurized water reactor core while providing additional DNBR margin. A description of the Mini-RTDP, along with the specific values of the design DNBR limit, the safety analysis DNBR limit, and the DNBR correlation limit, is provided in the marked up Updated Final Safety Analysis Report (UFSAR), Section 4.4.1.1, in the attached safety evaluation check list. There is no value in maintaining the specific limit values in the bases as they are design related in nature and are best reflected in the UFSAR. In addition, any changes would be addressed in each unit's reload analysis and a 10 CFR 50.59 review performed.

Additionally, similar changes were approved by NRC on May 1, 1992, for Beaver Valley Power Station Unit 2. Note that, although the Unit 2 analysis for the present Cycle 6 operation does not use the Mini-RTDP, the proposed wording for the basis is still valid.

ATTACHMENT TO ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY

SEQUOYAH UNITS 1 AND 2

INCREASE FdH TO 1.62 AND INCREASE Fq TO 2.4 WITH

Mini-RTDP FINAL SAFETY EVALUATION

(SECL-91-451, REVISION 2)

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

Mr. J. T. Robert
Nuclear Fuel Economics
Tennessee Valley Authority
1101 Market Street
BR 6N 60A
Chattanooga, Tennessee 37402-2810

92TV*-G-0073
ET-NSL-OPL-I-92-500
October 27, 1992

Ref. 1) 92TV*-G-0070

Tennessee Valley Authority
Sequoyah Units 1 and 2
Increase FdH to 1.62 and Increase Fq to 2.4 with
Mini RTDP Final Safety Evaluation
(SECL-91-451, Revision 2)

Dear Mr. Robert:

The revised safety evaluation, SECL-91-451, Revision 2, addressing increasing FdH from 1.55 to 1.62 and increasing Fq from 2.32 to 2.4 with mini RTDP is attached. This revision incorporates TVA's comments, and addresses a mixed core that includes once burned, 350 psi backfill pressure standard fuel with inconel grids.

Except for the NOTRUMP Bessel Function Potential Issue (PI-92-006), the status of the dispositioning of the LOCA-related potential issues transmitted via Reference 1 continues to be applicable. The NOTRUMP Bessel Function potential issue has been resolved and information regarding this resolution is attached. The effects of this resolution is to increase the Peak Clad Temperature (PCT) 11°F for the Small Break LOCA. This is shown on the PCT Rackup in the Safety Evaluation (Page 40).

If you have any questions do not hesitate to contact us.

Very truly yours,

N.R. Metcalf

N. R. Metcalf
Project Engineer
Mktg. & Customer Projects

LVTomasic/cld
Attachment

cc: L. Evans - (W) Chattanooga Sales

C765:LVT/100292

TVA SEQUOYAH

NOTRUMP BESSEL FUNCTION ERROR

Resolution

INTRODUCTION

Westinghouse has completed its evaluation of an issue affecting the NOTRUMP small break LOCA Evaluation Model. This information is being provided to allow affected utilities to assess individual reporting requirements which may exist due to changes in Peak Cladding Temperature (PCT) in their small break LOCA analysis results.

BACKGROUND

During a recently completed effort, anomalous behavior was noted in the NOTRUMP runs. This behavior was eventually traced to an error in SUBROUTINE BESSJ0 which calculates Bessel Function values used during the transient solution. During the time before this error was corrected, convergence anomalies were observed in NOTRUMP. It has been determined that this error was introduced in Cycle 23 of the NOTRUMP code and that only analyses performed with this version of the code are affected. Subsequent reruns with a corrected version of NOTRUMP (cycle 24) showed that the convergence abnormalities were indeed the result of the Bessel Function error, and that the standard convergence criteria used for Evaluation Model calculations continue to be valid when the corrected code is used.

EFFECTS OF ISSUE

The effect of this issue on Sequoyah Unit 1 (TVA) has been determined by a plant specific calculation to be a change of +11°F. This result should be evaluated to determine reporting requirements under 10 CFR 50.46.

SECL-91-451, Rev. 2
Customer Reference No(s).
N/A
Westinghouse Reference No(s).
N/A

WESTINGHOUSE NUCLEAR SAFETY
SAFETY EVALUATION CHECK LIST (SECL)

1) NUCLEAR PLANT(S): Sequoyah Units 1 and 2

2) SUBJECT (TITLE): Increased F Delta H from 1.55 to 1.62, Increased FQ From 2.32 to 2.4, With Mini RTDP

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59(b) has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A - 10CFR50.59(a)(1)

3.1) Yes X No __ A change to the plant as described in the FSAR?
3.2) Yes __ No X A change to procedures as described in the FSAR?
3.3) Yes __ No X A test or experiment not described in the FSAR?
3.4) Yes X No __ A change to the plant technical specifications? (See Note on Page 2.)

4) CHECK LIST - PART B - 10CFR50.59(a)(2) (Justification for Part B answers must be included on page 2.)

4.1) Yes __ No X Will the probability of an accident previously evaluated in the FSAR be increased?
4.2) Yes __ No X Will the consequences of an accident previously evaluated in the FSAR be increased?
4.3) Yes __ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?
4.4) Yes __ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.5) Yes __ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.6) Yes __ No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
4.7) Yes __ No X Will the margin of safety as described in the bases to any technical specification be reduced?

NOTES:

If the answer to any of the above questions is unknown, indicate under 5.) REMARKS and explain below.

If the answer to any of the above questions in Part A (3.4) or Part B cannot be answered in the negative, based on written safety evaluation, the change review would require an application for license amendment as required by 10CFR50.59(c) and submitted to the NRC pursuant to 10CFR50.90.

5) REMARKS:

Note for Part A item 3.4, F-DELTA-H is in the COLR for both units. Technical Specification bases changes are provided for Units 1 and 2.

FOR FSAR UPDATE

Section:______ Pages:______ Tables:_______ Figures:_______

Reason for / Description of Change:

Marked-up affected FSAR and Technical Specification and COLR marked pages are enclosed

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

SAFETY EVALUATION APPROVAL LADDER:

Nuclear Safety Preparer:	D. Cecchett D. E. Cecchett	Date: 10-30-92
Nuclear Safety Reviewer:	L. V. Tomasic L. V. Tomasic	Date: 10.30.92
Coordinated with Engineers:	J. Doman (Signature on file)	Date:
Coordinated with Engineers:	R. Anderson (Signature on file)	Date:
Coordinated with Engineers:	F. Baskerville (Signature on file)	Date:

Table of Contents

1.0 Introduction

Revision 2 addresses a mixed core of the following fuel types: 1) once burned, 350 psi backfill pressure standard (inconel grids); 2) fresh, 275 psi backfill pressure V-5H (zirc grids); 3) fresh, 100 psi or greater backfill pressure IFBA (zirc grids).

Prior revisions addressed a mixed cores identical to the one described above except that it did not consider the once burned, 350 psi back pressure standard fuel with inconel grids. TVA's comments were duscussed and incorporated in Revision 1.

The above revisions do not alter the discussions, bases, or conclusions of the original safety evaluation and do not represent an unreviewed safety question.

The original issue of this safety evaluaton addressed the impact of increasing F_{DH} from 1.55 to 1.62 and increasing F_Q from 2.32 to 2.40 using the Mini Revised Thermal Design Procedure (Mini RTDP) on the UFSAR Chapter 4, Thermal and Hydraulic Analysis, Chapter 6 and 15 LOCA and non-LOCA accident analyses for Sequoyah Units 1 and 2.

2.0 Non-LOCA Evaluation

This section summarizes the non-LOCA reanalyses and evaluations performed for the Sequoyah Unit 1 and 2 increased F_{DH}, increased F_Q, and Mini RTDP implementation. The increase in the design limit value of the nuclear enthalpy rise hot channel factor, F_{DH}, is from 1.55 to 1.62. The increase in the design limit value for the nuclear heat flux hot channel factor, F_Q, is from 2.32 to 2.40.

2.1 The Effects of an Increase F_{DH}

In general, an increase in F_{DH} results in a decrease in Departure from Nucleate Boiling Ratio (DNBR) for a given set of thermal-hydraulic conditions. On this basis it would be expected that all transients for which DNBR is calculated would be affected. However, the margins obtained through the use of the WRB-1 DNB correlation allow for the increased peaking factor without changing the core thermal limits. Therefore, only those transients which explicitly incorporate a value of F_{DH} in the calculation of the thermal-hydraulic conditions existing at the time of minimum DNBR require reanalysis. These are the Partial Loss of Flow, Complete Loss of Flow (including Reactor Coolant Pump (RCP) Underfrequency), RCP Locked Rotor and Startup of an Inactive Loop at an Incorrect Temperature.

The following non-LOCA accident analyses were not reanalyzed because F_{DH} is not an explicit analysis assumption:

UFSAR Chapter	
15.2.4	Uncontrolled Boron Dilution
15.2.8	Loss of Normal Feedwater
15.2.9	Loss of Offsite Power (Station Blackout)
15.2.10	Excessive Heat Removal Due to Feedwater System Malfunctions
15.2.11	Excessive Load Increase
15.2.12	Accidental Depressurization of the Reactor Coolant System
15.2.13	Accidental Depressurization of the Main Steam System
15.2.14	Spurious Operation of the Safety Injection System at Power
15.3.3	Inadvertent Loading of a Fuel Assembly into an Improper Position

The following non-LOCA accident analyses were not reanalyzed because the increase in F_{DH} does not change the transient conditions and sufficient DNBR margin exists to maintain the same DNBR limit used in the current licensing-basis safety analyses:

UFSAR Chapter	
15.2.1	RCCA Withdrawal from Subcritical
15.2.2	RCCA Withdrawal at Power
15.2.3	Rod Cluster Control Assembly Misalignment
15.2.7	Loss of External Load and/or Turbine Trip
15.3.2	Minor Secondary System Pipe Breaks
15.3.6	Single RCCA Withdrawal at Full Power
15.4.2	Major Secondary System Pipe Rupture
15.4.6	Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

A summary of the non-LOCA design basis calculations that were performed for Sequoyah Units 1 and 2 at an increased F_{DH} of 1.62 follows.

2.1.1 Partial and Complete Loss of Forced Reactor Coolant Flow (UFSAR 15.2.5 & 15.3.4)

The Partial Loss of Flow accident is an ANS Condition II event. The Complete Loss of Flow accident is an ANS Condition III event. Both of these transients have been reanalyzed in support of N-loop operation with the increased F_{DH}.

The Partial Loss of Flow transient assumes the coastdown of two RCPs during 4-loop, full-power operation while Complete Loss of Flow transient assumes the coastdown of 4 RCPs. The analyses have incorporated the increased design F_{DH} of 1.62 in the determination of the thermal-hydraulic conditions existing at the time of minimum DNBR.

The results of these two transients are shown in Figures 3.1-1 through 3.1-4 and 3.1-5 through 3.1-8, respectively. The flow coastdown transients are shown in Figures 3.1-1 and 3.1-5. For both transients, the FACTRAN code [1] is used to calculate the heat flux transient based upon nuclear power and flow from LOFTRAN [2]. The Partial Loss of Flow transient is terminated by a low Reactor Coolant System (RCS) loop flow reactor trip. The Complete Loss of Flow transient is terminated by reactor trip on reactor coolant pump undervoltage. In both cases, the DNBR safety analysis limit is not violated at the design F_{DH} of 1.62. Therefore, the safety analysis DNBR limits are met and the conclusions of the UFSAR remain valid.

The complete loss of forced reactor coolant flow from a pump frequency decay in all four RCPs was also reanalyzed for the increased F_{DH}. The transient assumptions for this complete loss of flow case are identical to the complete loss of flow case above except for the flow coastdown. The Underfrequency analysis assumed a constant frequency decay rate of 5.0 Hz/second. The transient is terminated by reactor trip on RCP underfrequency. The transient results indicate that the safety analysis DNBR limit is not violated for an F_{DH} design limit of 1.62. Therefore, the safety analysis DNBR acceptance criterion is met for this loss of flow event. It is determined that the underfrequency event is the limiting loss of flow case for these analyzed conditions.

The recommended UFSAR markups for the Partial and Complete Loss of Forced Reactor Coolant Flow accidents are included in the Appendix.

2.1.2 Single Reactor Coolant Pump Locked Rotor - Rods in DNB (UFSAR 15.4.4)

Reactor Coolant Pump Locked Rotor - Rods in DNB is analyzed to determine the percentage of fuel rods in the core that experience DNB during the accident. The Locked Rotor accident is postulated as an instantaneous seizure of one reactor coolant pump rotor at full-power conditions with all four loops in operation. Flow through the reactor coolant pump is rapidly reduced leading to an initiation of a reactor trip on a low flow signal.

The Locked Rotor - Rods in DNB transient was reanalyzed to incorporate a full-power F_{DH} design limit of 1.62. The FACTRAN code [1] is used to calculate the heat flux transient based upon nuclear power and flow from LOFTRAN [2]. Enough DNBR margin is available to maintain the number of rods in DNB to less than 10%. Therefore, for an increased F_{DH}, the Locked Rotor - Rods in DNB analysis adheres to safety analysis limits and is bounded by previous radiological dose release analyses.

2.1.3 Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature (UFSAR 15.2.6)

The Startup of an Inactive Loop transient is an ANS Condition II event analyzed to demonstrate that the DNB design basis is met. The transient has been reanalyzed incorporating the increased F_{DH} consistent with a full-power design limit of 1.62. The FACTRAN code [1] is used to calculate the heat flux transient based upon nuclear power and flow from LOFTRAN [2]. The transient results are shown in Figure 3.1-9 through Figure 3.1-12. The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors. The DNBR safety analysis limit is not violated at the increased F_{DH} of 1.62. Therefore, the safety analysis DNBR limit is met and the conclusions of the UFSAR remain valid.

UFSAR markups for the Startup of an Inactive Loop event analysis are included in the Appendix.

2.2 The Effects of an Increase F_Q

The peaking factor F_Q is the ratio of maximum local or "hot spot" rod power to average rod power. The full-power design limit F_Q is explicitly assumed in two UFSAR non-LOCA events.

UFSAR Chapter

15.4.4 Single Reactor Coolant Pump Locked Rotor - Fuel/Clad Temperature
15.4.6 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

These two events are adversely affected by an increase in the full-power design limit F_Q and require evaluation to ensure that cladding integrity and fuel melting at the "hot spot" are maintained within the applicable safety analysis limits.

2.2.1 Single Reactor Coolant Pump Locked Rotor - Fuel/Clad Temperature (UFSAR 15.4.4)

The Locked Rotor event is classified as a Condition IV event. The UFSAR analysis for the Locked Rotor fuel/clad temperature transient was conservatively analyzed with a 3.0 F_Q including allowances for calculational uncertainty and nuclear power peaking due to densification. The results of the current UFSAR Locked Rotor analysis show that the maximum clad temperature at the core hot spot is 2026°F. This is less than the limit of 2700°F. The amount of Zr-water reaction is small, calculated to be 0.70% by weight. Because these UFSAR fuel and clad temperature transient results were analyzed with a 3.0 F_Q, the increase in F_Q to 2.40 is bounded and the applicable fuel/clad temperature safety criteria continue to be met. The UFSAR conclusions for the Locked Rotor analysis continue to remain valid.

2.2.2 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (UFSAR 15.4.6)

The Rod Cluster Control Assembly (RCCA) Ejection event is classified as a Condition IV event for which conservative criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

2. Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits.

3. Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion (1) above.

To conservatively bound the F_Q increase to 2.40, the UFSAR BOL and EOL full-power RCCA Ejection cases were reanalyzed with a 2.625 F_Q including allowances for calculational uncertainty and nuclear power peaking due to densification. The FACTRAN code [1] was used to calculate the fuel and clad transient based upon nuclear power from the TWINKLE code [3]. A detailed discussion of the method of analysis can be found in Reference 4.

The nuclear power transient and hot spot fuel, average fuel, and clad temperature versus time for the EOL full-power case (which is the limiting case in terms of the fuel melt criterion) are presented in Figures 3.2-1 and 3.2-2. A summary of parameters used in both of the full-power rod ejection cases and the results of these cases, are presented in Table 3.2-1. The analysis results demonstrate that the applicable UFSAR fuel melting and stored energy limits are not exceeded for the increased F_Q. Therefore, the UFSAR conclusions for the RCCA Ejection analysis remain valid.

UFSAR markups for the RCCA Ejection analysis are included in the Appendix.

2.3 Utilization of the Mini Revised Thermal Design Procedure

The Mini Revised Thermal Design Procedure (Mini RTDP) is described in Reference 5. Mini RTDP is similar to the current fixed value DNB design basis methodology in that initial condition assumptions for power, flow, temperature, pressure, and bypass flow are assumed to be at their extreme values when used in the plant transient analyses. The Mini RTDP differs from the fixed value DNB design basis methodology in that it statistically combines peaking factor uncertainties, et. al, with the DNB correlation uncertainty. The statistical convolution of uncertainties results in a net increase in DNBR margin for any event which uses the Mini RTDP.

As Mini RTDP results in a net increase in DNBR margin, its implementation does not adversely affect any of the UFSAR events. Therefore, the UFSAR conclusions remain valid with the implementation of the Mini RTDP.

2.4 Non-LOCA Conclusions

An increased F_{DH} from 1.55 to 1.62 and an increased F_Q from 2.32 to 2.40 has been evaluated utilizing Mini RTDP to determine the effects on Sequoyah Units 1 and 2 Chapter 6 and Chapter 15 accident analyses. The following non-LOCA transients were reanalyzed for the increased F_{DH} and F_Q: Partial and Complete Loss of Flow (including RCP Underfrequency), RCP Locked Rotor - Rods in DNB, Startup of an Inactive Loop, and full-power RCCA Ejection.

As previously demonstrated in this safety evaluation, all applicable acceptance criteria for these events have been satisfied and the conclusions presented in the UFSAR still remain valid.

The increased F_{DH} and F_Q will have no impact on the remaining non-LOCA transients because sufficient DNBR margin is used to maintain the safety analysis DNBR limits. Thus the proposed increase in F_{DH} to 1.62 and increase in F_Q to 2.40 does not constitute an unreviewed safety question and the non-LOCA accident analyses, as presented in this report, support the proposed change.

2.5 Non-LOCA References

1) Hargrove, H. G., "FACTRAN - A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

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

3) D. H. Risher, Jr., R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary), WCAP-8028-A (Non-Proprietary), January 1975.

4) D. H. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January, 1975.

5) S. Ray, "Mini Revised Thermal Design Procedure (Mini RTDP)," WCAP-12429-A, October 1989.

Table 3.2-1
Summary of Rod Ejection Analysis Results

Parameter	BOL, HFP	EOL, HFP
Total Core Peaking Factor	7.11	7.88
Ejected Rod Worth, (pcm)	200	210
Maximum Fuel Pellet Average Temperature, (°F)	4121	4056
Maximum Fuel Pellet Centerline Temperature, (°F)	4971	4879
Maximum Clad Average Temperature, (°F)	2319	2267
Maximum Fuel Enthalpy, (cal/gm)	181	177
Maximum Fuel Centerline Melt, (%)	7.0	8.7

Figure 3.1-1
Flow Transients for Partial Loss of Flow,
All Loops Operating, Two Loops Coasting Down

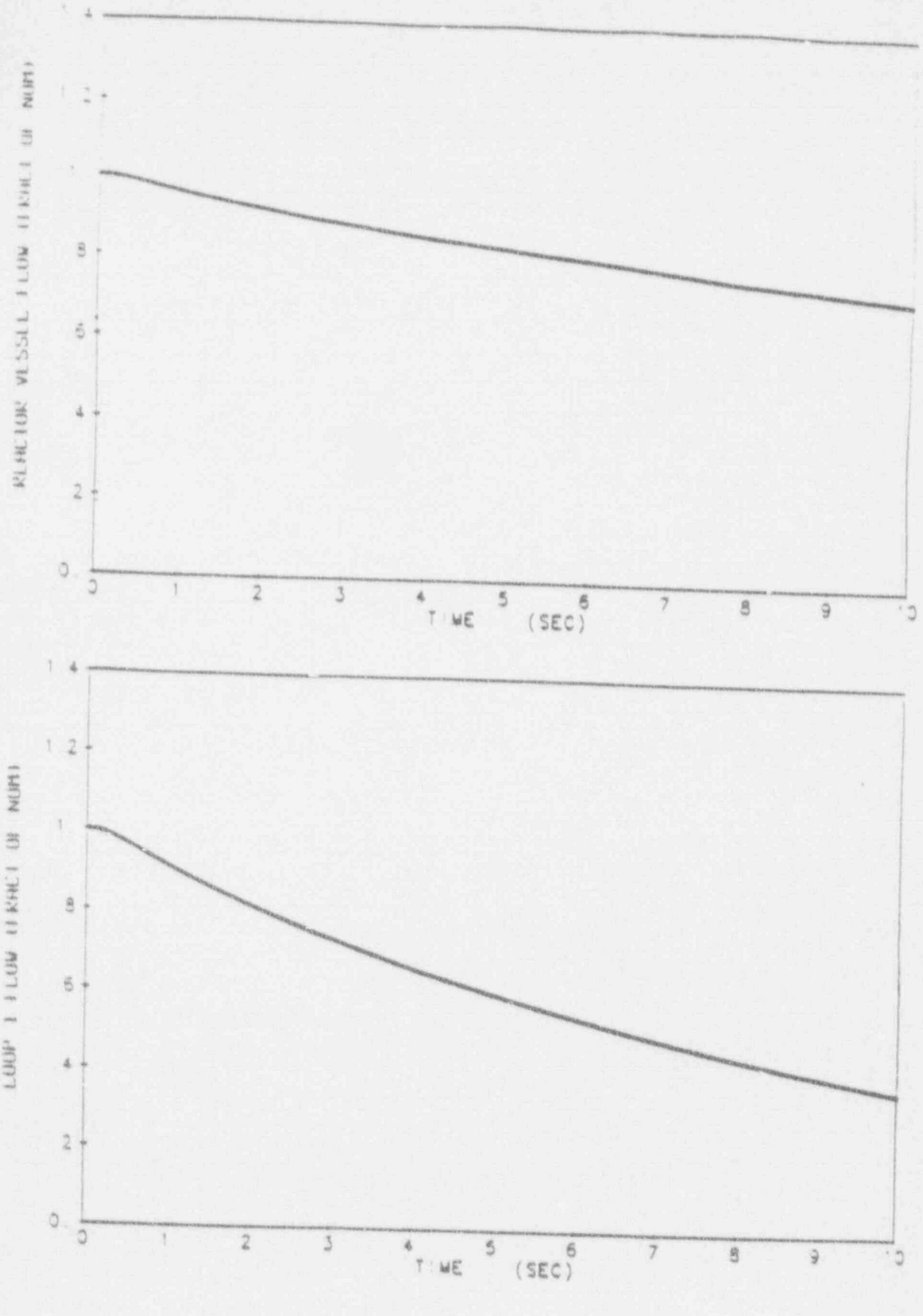

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Figure 3.1-2
Nuclear Power and Pressurizer Pressure for Partial Loss of Flow,
All Loops Operating, Two Loops Coasting Down

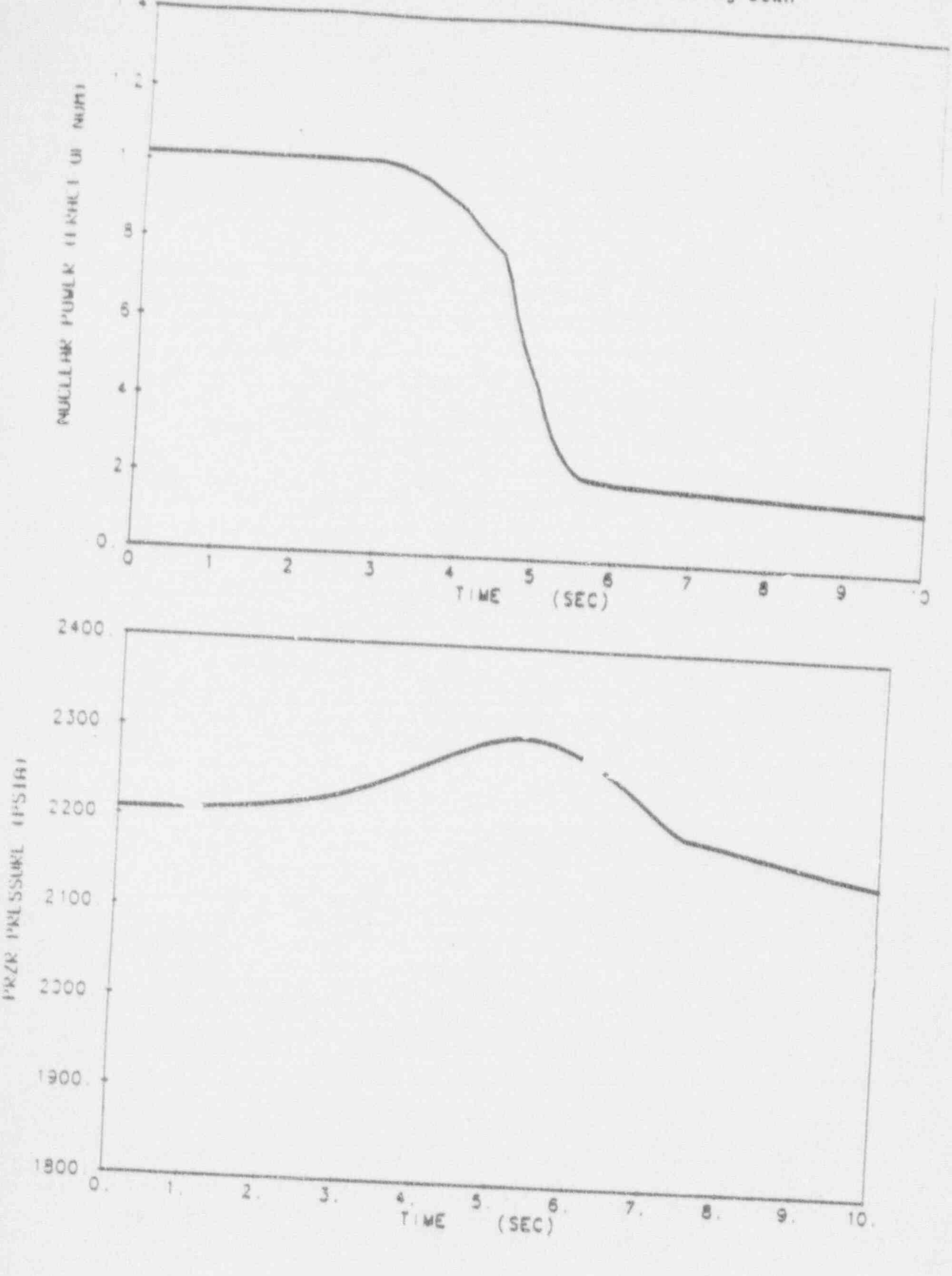
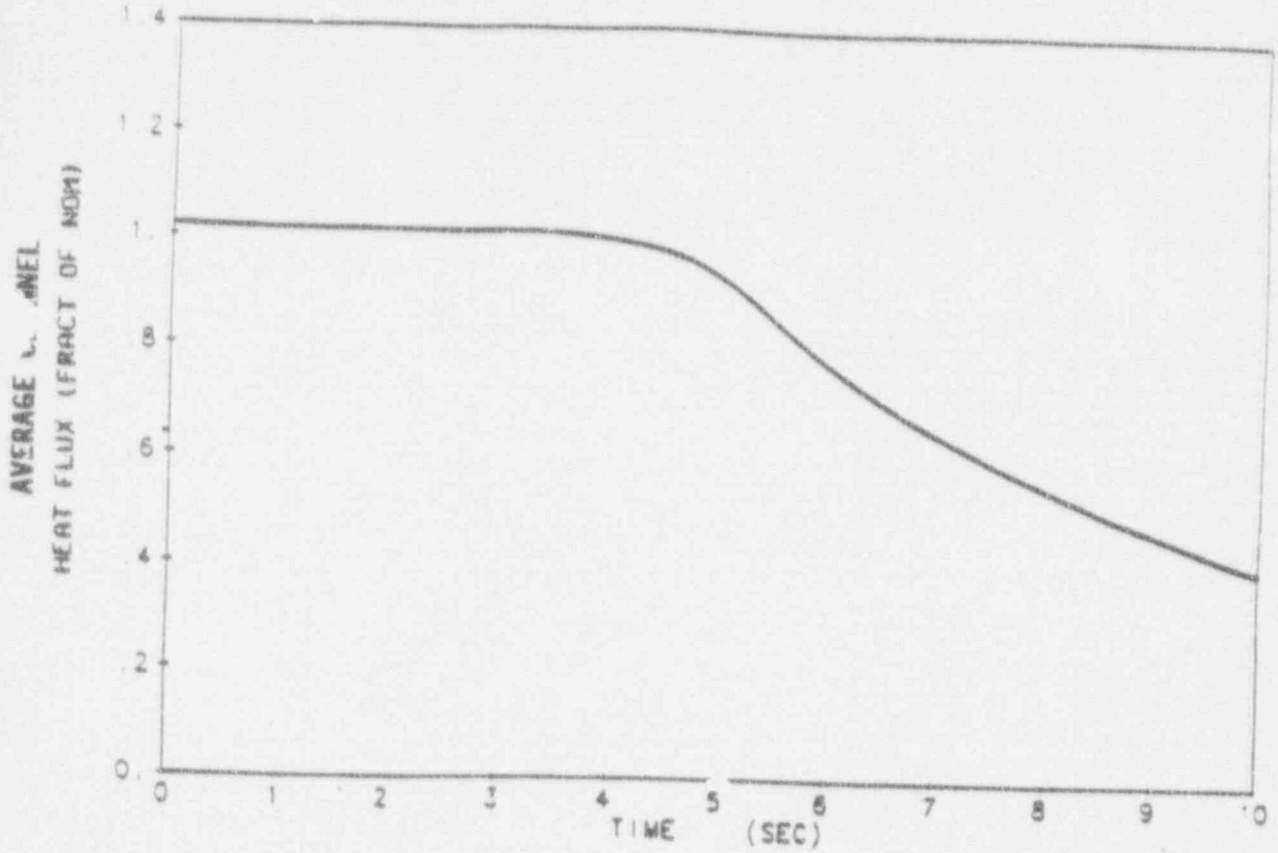


Figure 3.1-3
Average and Hot Channel Heat Flux Transient for Partial Loss of Flow,
All Loops Operating, Two Loops Coasting Down



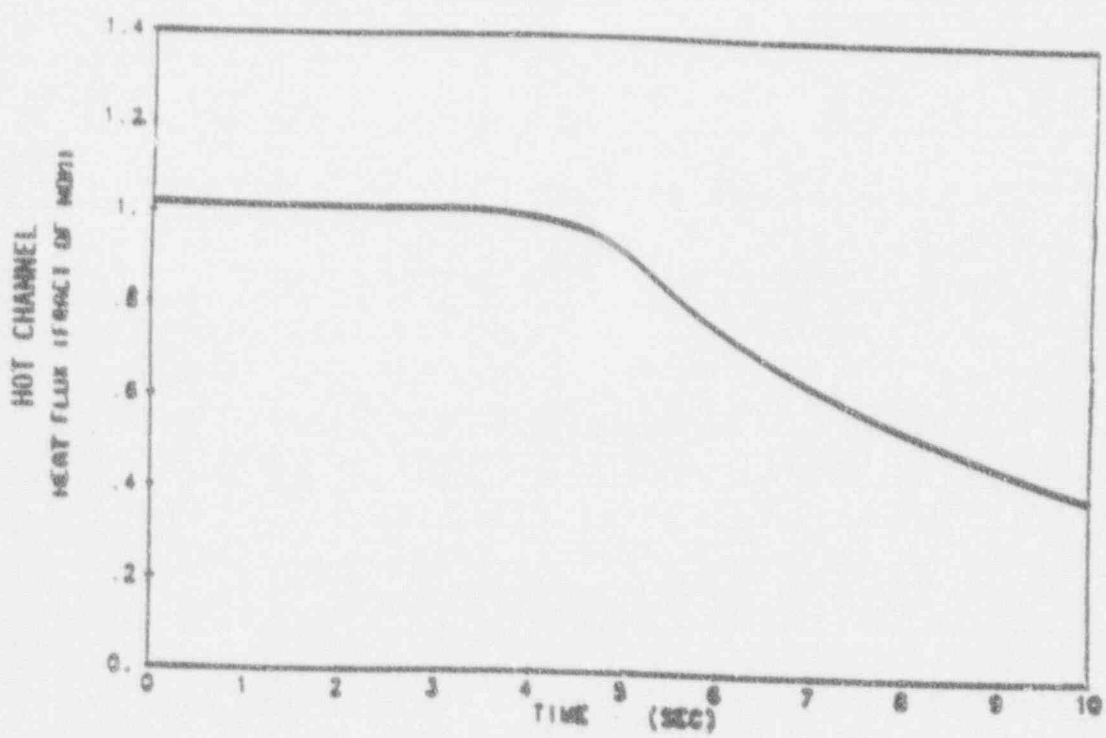
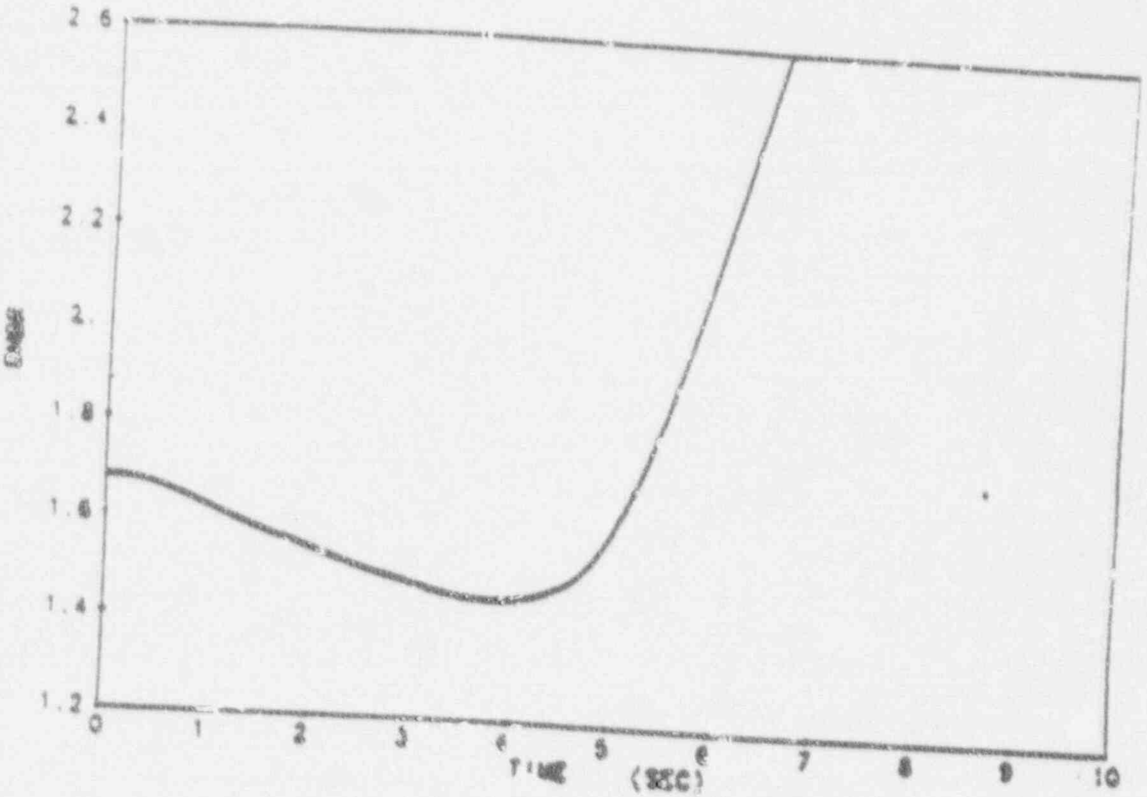


Figure 3.1-4
DNBR versus Time for Partial Loss of Flow,
All Loops Operating, Two Loops Coasting Down



1.5

Figure 3.1-5
Reactor Vessel Flow versus Time for All Loops Operating,
All Loops Coasting Down, Complete Loss of Flow

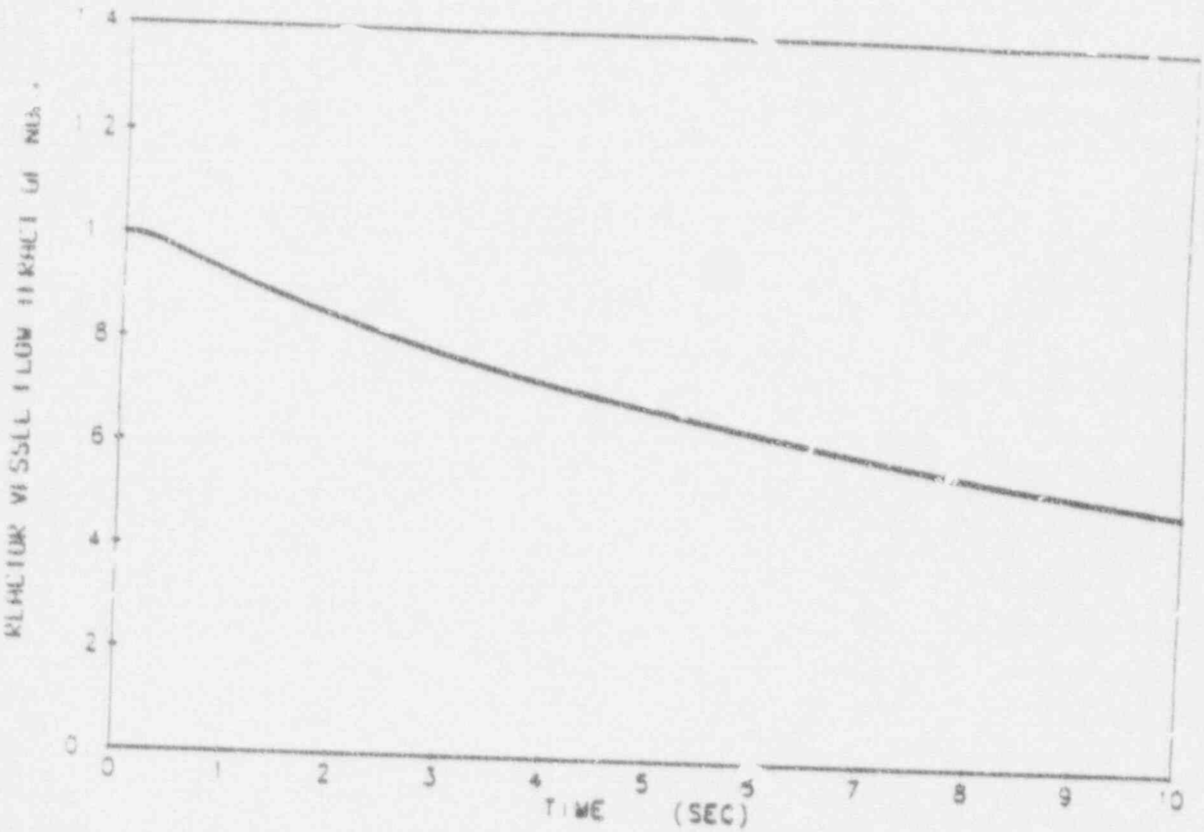


Figure 3.1-6
Nuclear Power Transient and Pressurizer Pressure Transient for All Loops Operating, All Loops Coasting Down, Complete Loss of Flow

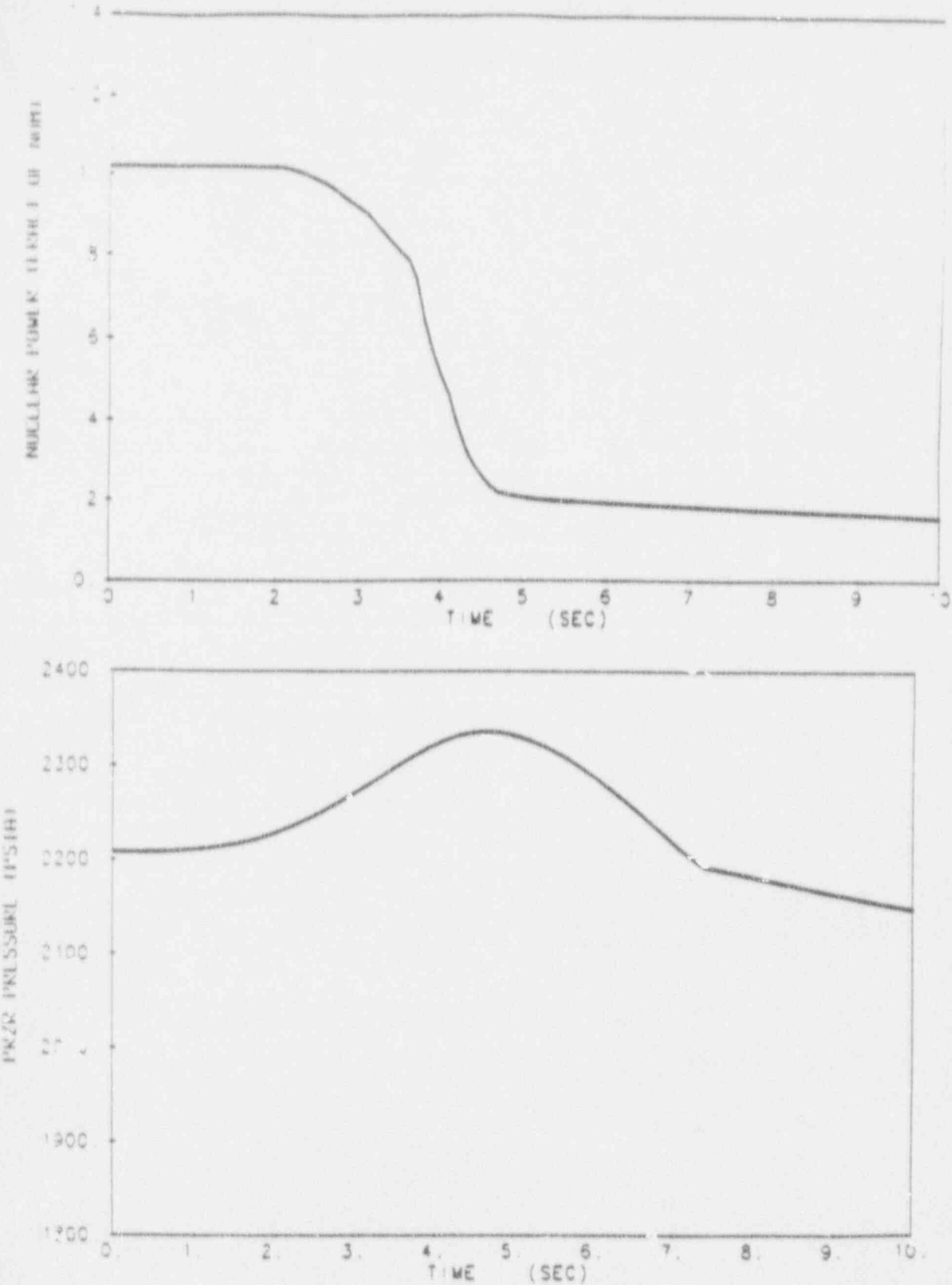


Figure 3.1-7
Average and Hot Channel Heat Flux Transients for All Loops Operating, All Loops Coasting Down, Complete Loss of Flow

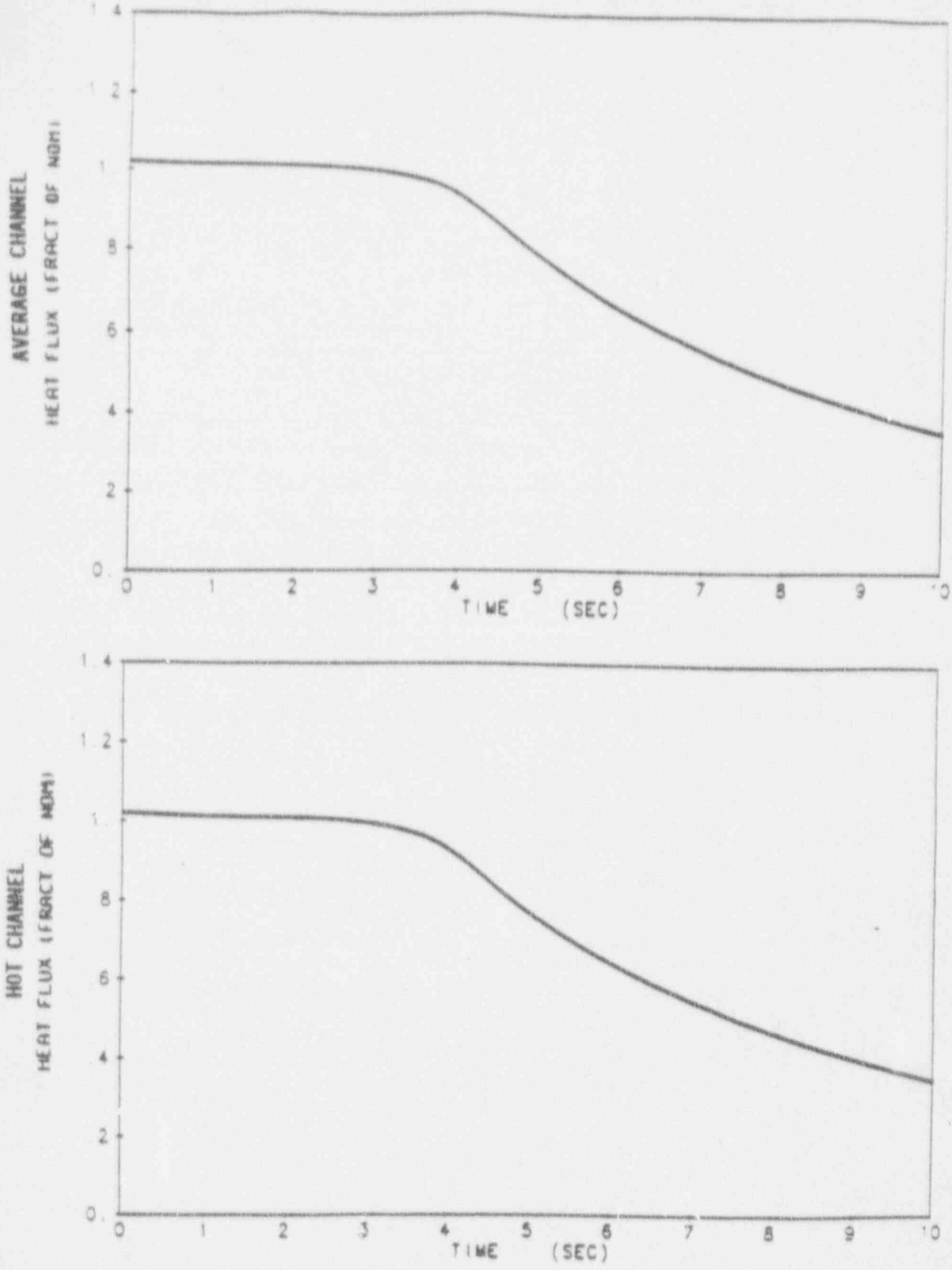


Figure 3.1-8
DNBR versus Time for All Loops Operating,
All Loops Coasting Down, Complete Loss of Flow

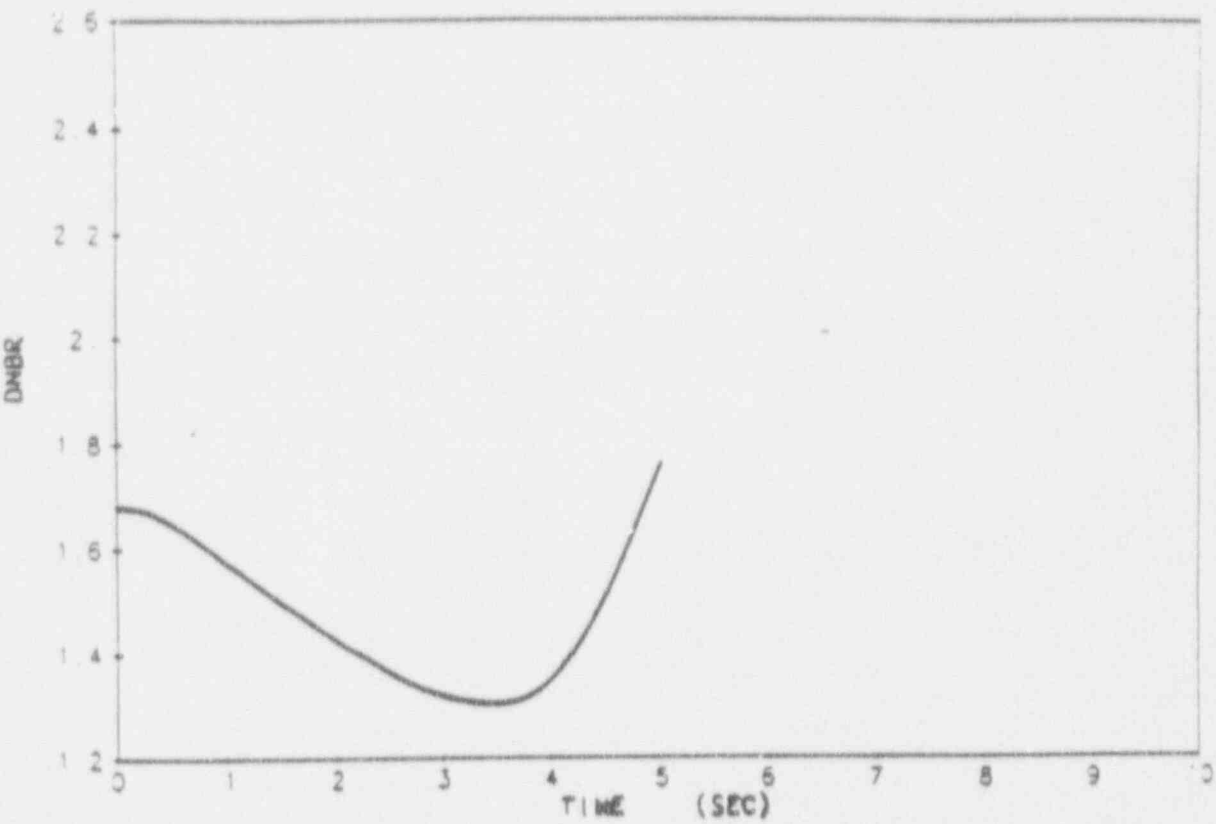


Figure 3.1-9
Nuclear Power versus Time,
Startup of an Inactive Reactor Coolant Pump

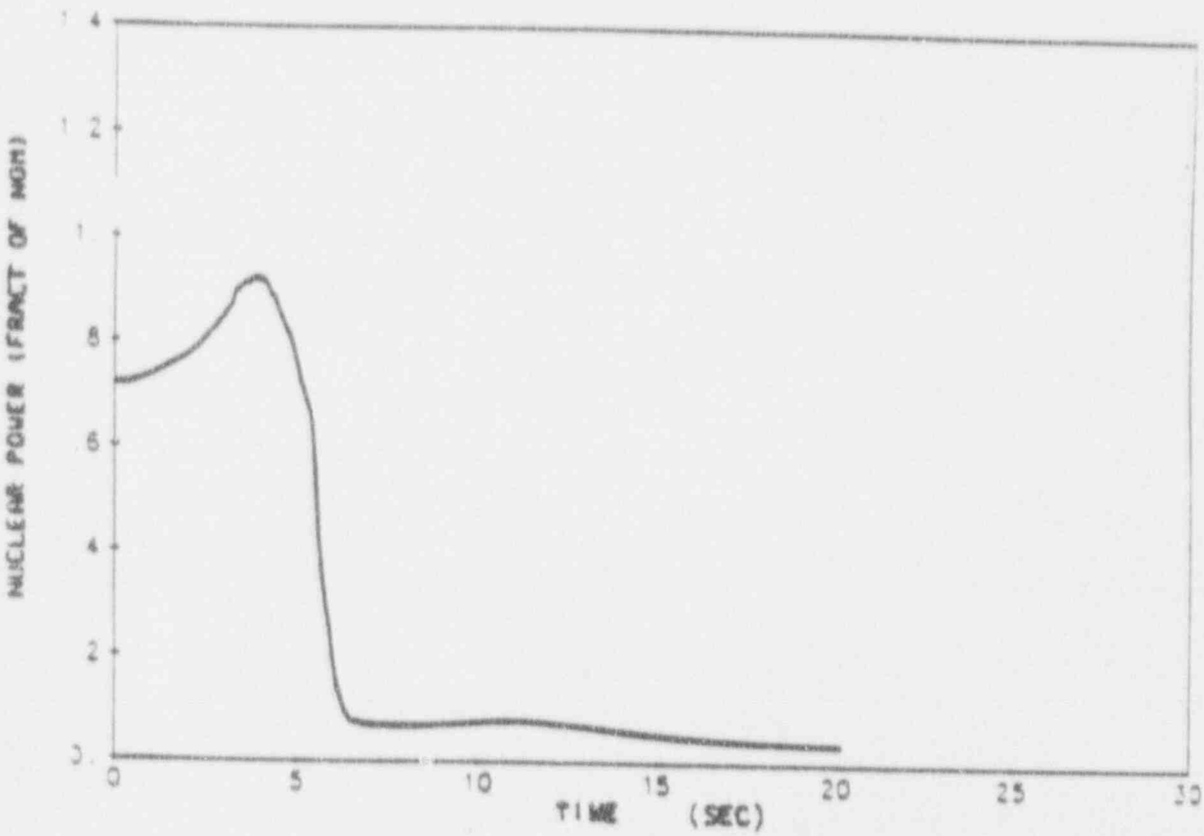


Figure 3.1-10
Core Heat Flux versus Time,
Startup of an Inactive Reactor Coolant Pump

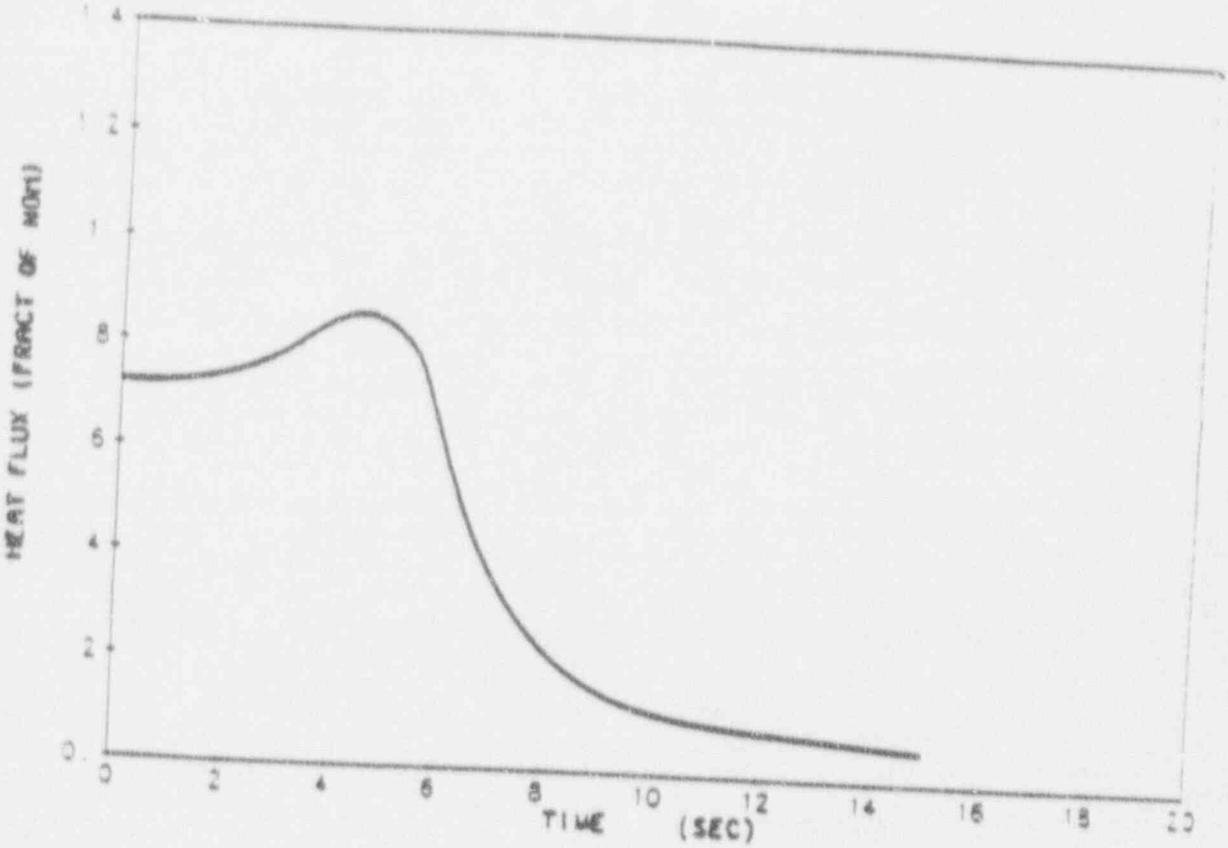


Figure 3.1-11
Core Average Temperature versus Time,
Startup of an Inactive Reactor Coolant Pump

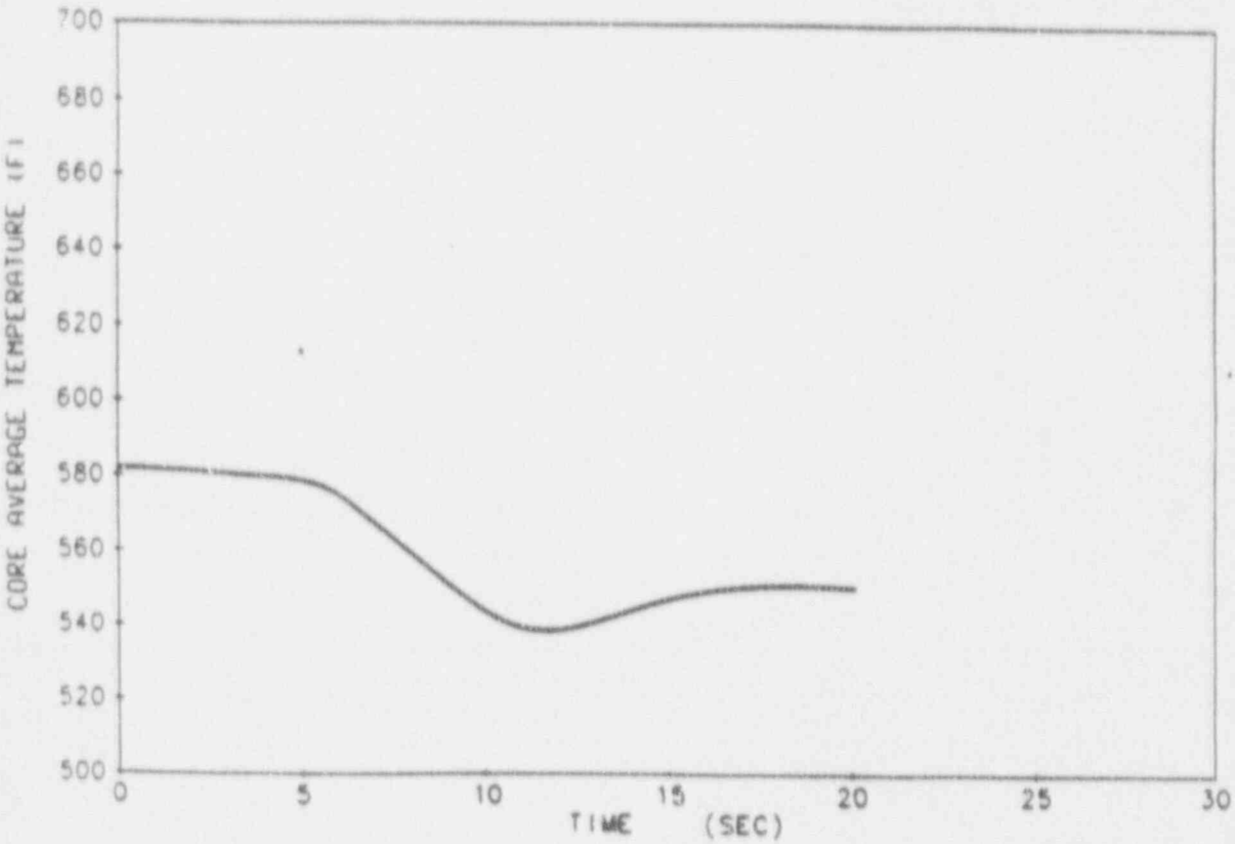


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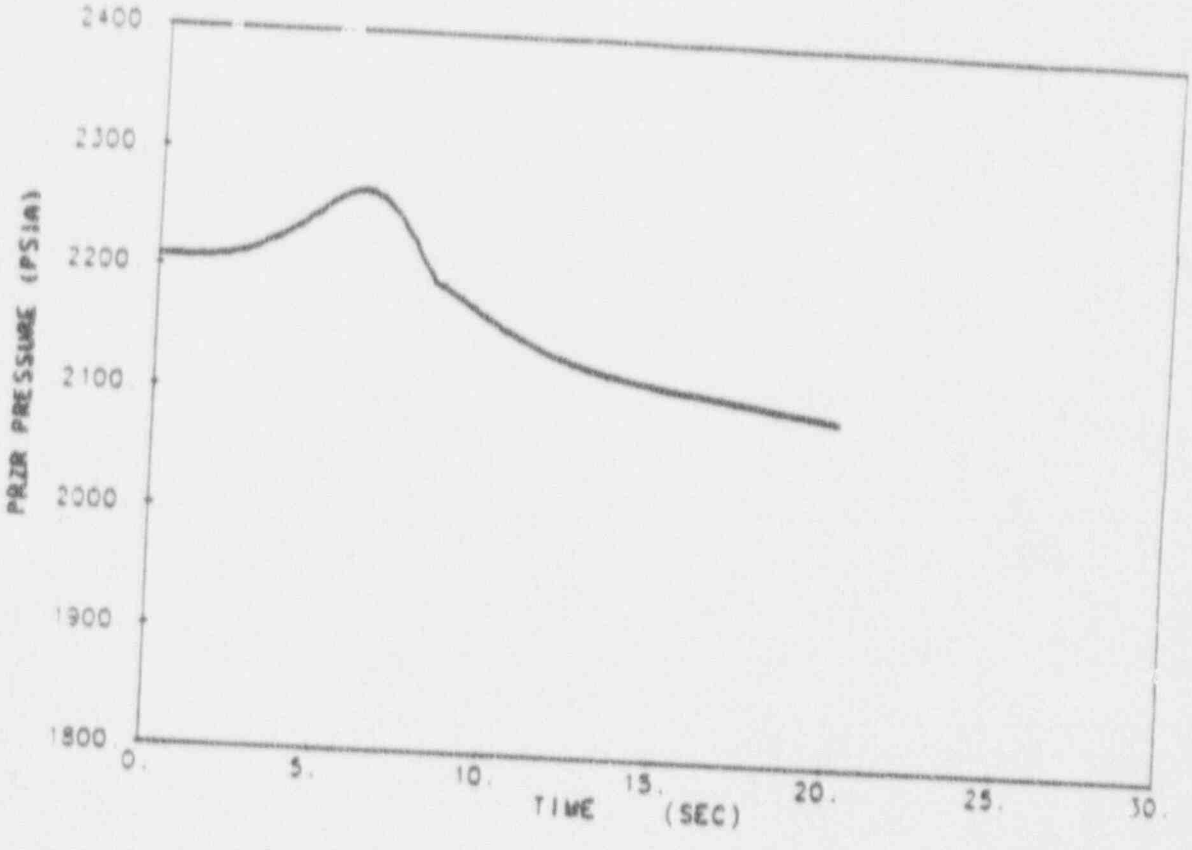
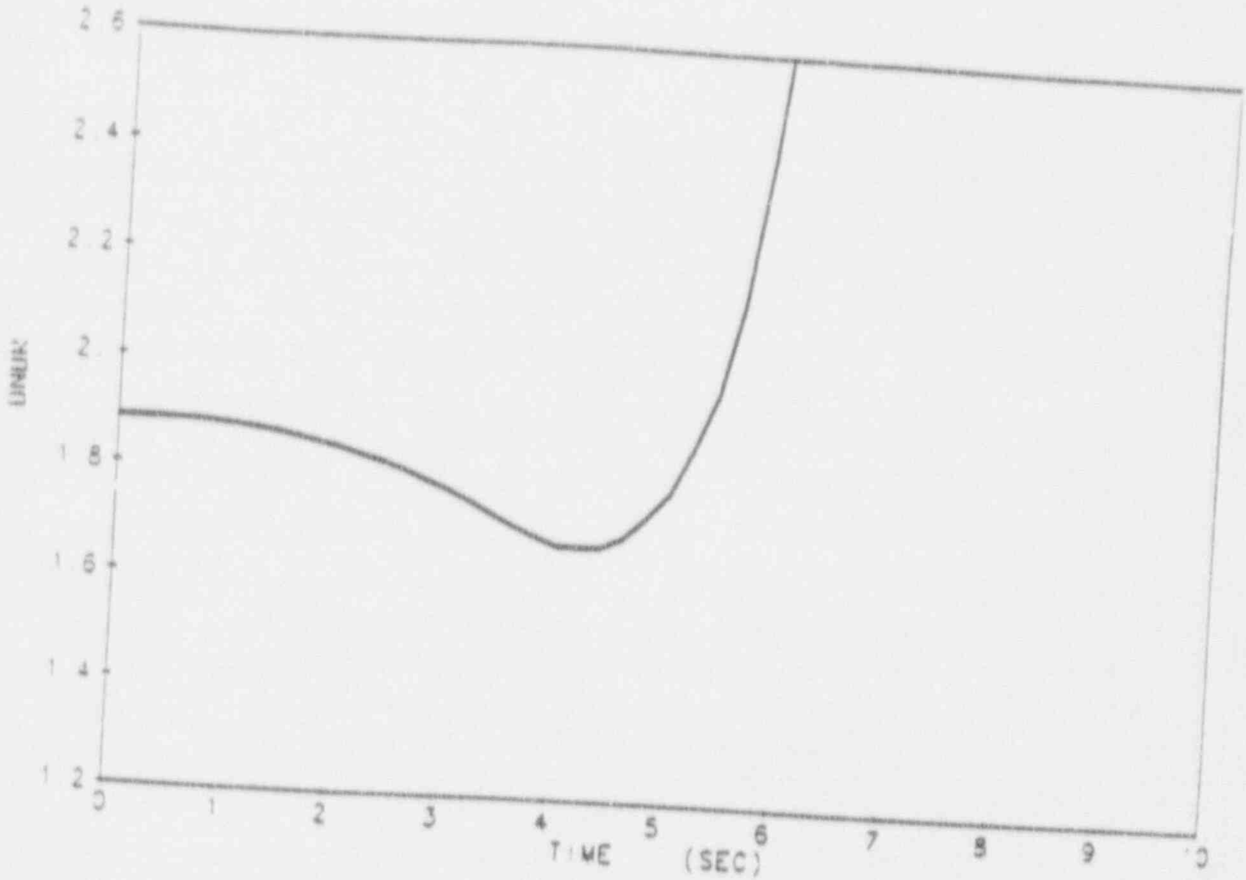
Figure 3.1-12
Pressurizer Pressure versus Time,
Startup of an Inactive Reactor Coolant Pump



Versus Time for Startup of an Inactive Reactor Coolant Loop



2[4]

Figure 3.2-1
Nuclear Power Transient, HFP EOL Rod Ejection Accident

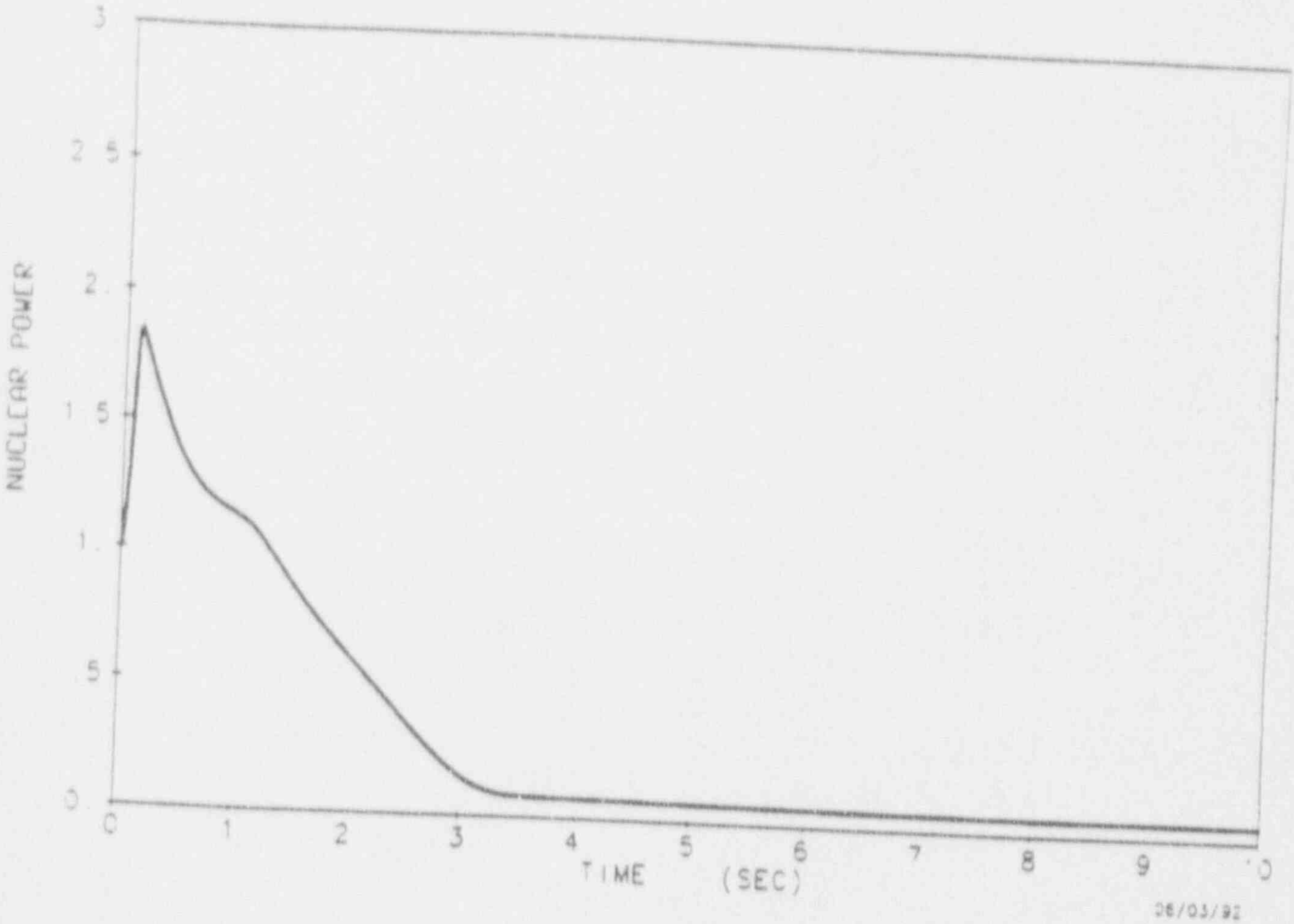
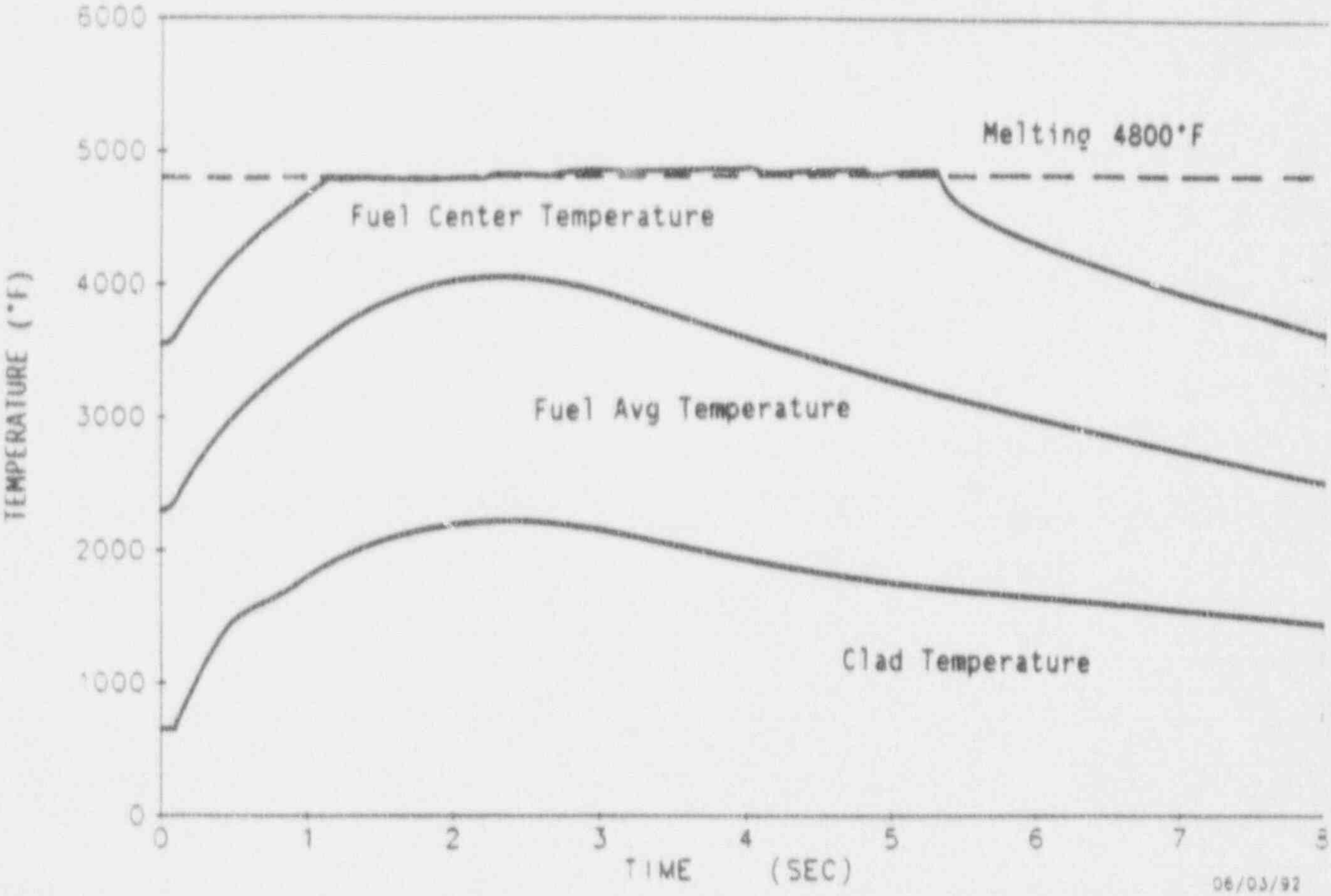


Figure 3.2-2
Hot Spot Fuel, Average Fuel, and Clad Temperature versus Time,
EOL HFP Rod Ejection Accident



3.0 LOCA Evaluation

Revision 2 addresses a mixed core of the following fuel types: 1) once burned, 350 psi backfill pressure standard (inconel grids); 2) fresh, 275 psi backfill pressure V-5H (zirc grids); 3) fresh, 100 psi or greater backfill pressure IFBA. Similar fuel with higher burn ups would be bounded by these fuel types. The following LOCA-related accidents will be considered: large break and small break LOCA; reactor vessel and loop blowdown forces; hot leg switchover to preclude boron precipitation; and post-LOCA long term core cooling minimum flow and subcriticality. It must be shown that the higher peaking factors will not increase the probability of occurrence or the consequences of any previously analyzed accident. And that they will not lead to the possibility of an accident different from any previously analyzed.

The preceding revision addressed the increase in peaking factors for a mixed core condition identical to the one described above--except that it did not consider the once burned, standard fuel. Because of its inconel grids which generally result in approximately a 100°F penalty in the large break LOCA analysis, the standard fuel is likely to bound all other fuel types.

3.1 Large Break LOCA Analysis - FSAR CHAPTER 15.4.1

A large rupture of the reactor coolant system (RCS) piping is a hypothetical event postulated to demonstrate that the calculated performance of the emergency core cooling system is adequate to mitigate the consequences of such a scenario. The effect of increased core peaking factors during a hypothetical large rupture of the RCS piping is examined to ensure that the bases and assumptions of the calculation remain valid, and that a conservative approach will yield values that conform to 10CFR 50.46 standards. Following a large rupture of the cold leg RCS piping, the RCS depressurizes in approximately 30 seconds to a pressure nearly equal to the containment pressure. During this time the core flow reverses and the core is cooled by a two-phase mixture flowing down through the core, up the downcomer and out the break. When the reverse core flow period ends, end of bypass occurs, and the lower plenum can begin filling with cold safety injection water. After the lower plenum fills, and the bottom of the core is reached, the process of reflooding the core begins. The peak cladding temperature (PCT) for large break LOCAs occurs during the reflooding portion of the transient at elevations near or above the mid-plane of the core (6 feet) for the Westinghouse ECCS Evaluation Models. This is due in part to the fact that the chopped cosine power distribution has been demonstrated to be limiting (see Addendum 1 to Reference 1).

A large break LOCA analysis for Sequoyah Nuclear Plant Unit 1 was performed using the 1981 Evaluation Model with BASH (Reference 1). The analysis assumed a reactor power level of 102% of 3411 Mwt, $Fq=2.40$, $F\Delta h=1.62$, and uniform 10% steam generator tube plugging. The DECLG limiting discharge coefficient of $CD=0.6$ was analyzed. The PCT calculated in this analysis was 2069°F for a mixed core in which the V-5H fuel is bounding (that is, no standard fuel is present) (Reference 6).

With a mixed core including the once burned, standard fuel, the limiting PCT has been determined to be 2169°F. With respect to the 2069° F PCT of the current licensing basis analysis, this represents a 100°F analysis penalty due to the inconel grids. When compared to the current licensing basis PCT of 2013°F, this figure also represents a penalty of approximately 160°F for the increase in Fq and FΔh. Until such time as the standard fuel is removed from the core the limiting PCT will remain at the figure of 2169. Conformance to 10CFR 50.46 limits is documented in the margin utilization documentation provided.

3.2 Small Break LOCA Analysis - FSAR Chapter 15.3.1

A small rupture of the RCS piping is postulated to demonstrate that the calculated performance of the ECCS design is adequate to meet the requirements of 10CFR 50.46 for these more realistic LOCAs. In the Westinghouse small break LOCA ECCS Evaluation Model (Reference 2), the ECCS performance is a function of the break size, core power level and operational performance. For the Westinghouse small break LOCA ECCS Evaluation Model, the primary system response to a small rupture of the RCS piping is typically a rapid depressurization to a pressure equal to the hot leg saturation pressure. Usually, the break energy removal capability in conjunction with the secondary heat removal capability exceeds the decay heat production, and the RCS will depressurize to a pressure slightly above the secondary pressure. This ensures that the steam generator secondary sides continue to remove decay heat, producing a condition of quasi-equilibrium pressure at which the primary system tends to stabilize prior to the venting of steam through the broken leg loop seal. Following the venting of this steam, core boil-off may continue, possibly exceeding the safety injection mass flow rate and resulting in a boil-off core uncovery transient. The depth and duration of uncovery can be influenced by several parameters (e.g., initial power level, break size, safety injection flowrates, etc.). Reference 3 contains the results of several analyses for typical break sizes, power levels and system capabilities in Westinghouse PWRs for the Westinghouse small break LOCA ECCS Evaluation Model.

The current licensing basis small break LOCA analysis for Sequoyah Units 1 and 2 was performed using the 1985 NOTRUMP model (Reference 2). The analysis assumed a reactor power level of 102% of 3411 Mwt, Fq=2.7, FΔh=1.7, and uniform 15% steam generator tube plugging. The analysis determined the limiting break size to be a 3 inch diameter cold leg break. Note that this analysis bounds any mixed core at peaking factors of Fq=2.4, FΔh=1.62. Conformance to 10CFR 50.46 standards is maintained as indicated in the margin utilization sheet provided.

In addition to this safety evaluation, this rackup also reflects the resolution of the NOTRUMP Bessel Function Potential Issue (PI-92-006) which results in an increase of 11° to PCT for the Small Break LOCA.

3.3 Blowdown Reactor Vessel and Loop Forces - FSAR Chapter 3.9

The blowdown hydraulic forcing functions resulting from a loss of coolant accident are considered in Section 3.9.1.5 (Analysis Methods Under LOCA Loadings), and Section 3.9.3.5 (Blowdown Forces Due to Cold and Hot Leg Break) of Volume 4 of the Sequoyah Units 1 and 2 FSAR. Neither the mixed core condition described above nor the increased peaking factors will have a significant effect on the LOCA blowdown hydraulic loads or on the results of the LOCA hydraulic forces calculations.

3.4 Post LOCA Longterm Core Cooling Subcriticality Requirement - FSAR CHAPTER 15.4.1

The Westinghouse licensing position for satisfying the requirements of 10CFR Part 50 Section 50.46 Paragraph (b) Item (5) "Long Term cooling" is defined in WCAP-8339 (Reference 4, pp. 4-22). The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water residing in the sump following a LOCA (Reference 4). Since credit for the control rods is not taken for large break LOCA, the borated ECCS water provided by the RWST and Accumulators must have a concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical assuming all control rods out (ARO). This requirement is not affected by the increase in core peaking factors or the particular type of mixed core.

3.5 Hot Leg Switchover To Prevent Potential Boron Precipitation - FSAR Chapter 6.3.3.2

During a large break LOCA the plant switches to cold leg recirculation after the RWST switchover setpoint has been reached. If the break is in the cold leg there is a concern that the cold leg injection water will fail to establish flow through the core. Safety injection entering the broken loop will spill out the break, while SI entering the intact cold legs will circulate around the downcomer and out the break. With no flow path established through the core the fluid in the core remains stagnant. As steam is produced in the core from decay heat, the boron associated with the steam will remain in the vessel. Thus, as water is boiled off with no circulation present in the core, the boric acid concentration increases. The boron concentration in the vessel will increase until the solubility limit of the boric acid solution is reached, at which time boron will begin to precipitate. As the boron precipitates, it may plate out on the fuel rods, which would adversely affect their heat transfer characteristics.

The purpose of the hot leg recirculation switchover time analysis is to provide the time at which hot leg recirculation must be established to prevent boron precipitation in the core. Neither an upgrade in core peaking factors nor mixed core as described above will affect this calculation. The current time for hot leg switchover remains applicable.

3.6 LOCA Conclusion

The evaluation performed for the LOCA-related accidents applies to a mixed core design in which once burned, 350 psi fill pressure, standard fuel has been established as the bounding fuel type. The LOCA-related accidents within the scope of Safeguards Analysis have been examined to determine whether or not conformance to 10CFR 50.46 criteria can be demonstrated with the mixed core design at increased core peaking factors of $F\Delta h = 1.62$ and $Fq = 2.40$. As detailed in the preceding discussion it has been concluded by Westimghouse that this change will not increase the consequences of any previously analyzed accident. The included margin utilization documentation demonstrates conformance to 10CFR 50.46 standards for the mixed core design at increased peaking factors.

3.7 LOCA References

1. WCAP-10266-P-A Rev. 2, with Addenda, Besspiata, J. J., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March, 1987.

2. WCAP-10054-P-A (Proprietary), WCAP-10081 (Non-Proprietary), Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August, 1985.

3. WCAP-11145-P-A, WCAP-11372 (Non-Proprietary), Rupprecht, S. D., et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code", October, 1986.

4. Westinghouse Technical Bulletin NSID-TB-86-08, "Post-LOCA Long-Term Cooling: Boron Requirements," October 31, 1986.

5. WCAP-8301, Bordelon, F. M., et al., "Locta-IV Program: Loss-of-Coolant Transient Analysis," June, 1974.

6. TVA-92-021, 01/31/92, "Large Break LOCA Analysis Worst Case Break with BASH Final Analysis Results."

4.0 Thermal Hydraulic Design Evaluation

The proposed change to the Sequoyah Units 1 and 2 Core Operating Limits Report (COLR) Section 2.6 which impacts the DNBR calculations is the value of $F^N_{\Delta H}$ enthalpy rise hot channel factor, determined from the following equation:

$$F^N_{\Delta H} = 1.62\ [(1.0 + 0.3\ (1.0 - P))]$$

where P = Thermal Power / Rated Power

$F^N_{\Delta H}$ = Measured value of FΔH obtained by using the moveable incore detectors to obtain a power distribution map with approximate uncertainties.

The measured radial peaking factor limit increase from 1.55 to 1.62 has a direct impact on DNBR calculations. The thermal-hydraulic design method for Sequoyah Units 1 and 2 is the mini-Revised Thermal Design Procedure (Mini-RTDP), Reference 1, which replaces the previous Standard Design Procedure. The mini-RTDP approach is a statistical approach which combines the uncertainties on the nuclear, thermal and the fuel fabrication parameters with the uncertainties on THINC-IV and the transient codes. The resulting overall DNBR uncertainty is then combined statistically with the DNB correlation uncertainty to define the design limit DNBR. The implementation of the mini-RTDP procedure generates additional DNBR margin to offset the 4.5% increase in FΔH. Therefore, the current core limits in the Technical Specification, Figure 2.1-1, and the DNBR limits for the FSAR Reference 2 analysis remain valid for Sequoyah Units 1 and 2.

The limit on the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, will take the following form in Section 2.6 of the Core Operating Limits Report (COLR):

$F^N_{\Delta H} \leq 1.62\ [(1 + 0.3(1\text{-}P))]$
where P = Thermal Power / Rated Thermal Power, and

$F^N_{\Delta H}$ = Measured valued of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map with appropriate uncertainties.

The increase in the measured radial peaking factor limit will allow additional flexibility for fuel management and for determining core loading patterns.

Cycle specific reload core analysis performed in accordance with the methodology described in References 1 and 2 demonstrates that the new radial peaking factor limit is met. No changes to the current methodology are required as a result of this change.

4.1 T/H Conclusion

In summary, the effect of increasing $F^N_{\Delta H}$ from 1.55 to 1.62 at rated thermal power has been accommodated in the safety analyses by generating additional DNBR margin by utilizing the mini-RTDP procedure. The current DNB-related safety limits, including the core limits in Figure 2.1-1 of Sequoyah Units 1 and 2 Technical Specifications, remain valid.

4.2 T/H References

1. S. Ray, "Mini Revised Thermal Design Procedure (Mini-RTDP)," WCAP-12178-P-A, October 1989.

2. Sequoyah Nuclear Plant Final Safety Analysis Report, USNRC Docket No. 50-327/328.

4.3 F_Q Increase

The limit on the heat flux hot channel factor, $F_Q(z)$, will take the following form in the Section 2.5 of the COLR:

$F_Q(z) \leq (2.40/P) * (K(z))$ for $P > 0.5$, and

$F_Q(z) \leq (4.80) * (K(z))$ for $P \leq 0.5$

where P = Thermal Power / Rated Thermal Power, and

K(z) = the function obtained from Figure 3 of the COLR for a given core height location.

The increased total peaking factor limit will allow additional flexibility in fuel management and core operation as well as accommodate the increased radial peaking factor limit. The increased radial peaking factor discussed above will result in increases in the total peaking factor, $F_Q(z)$, experienced in the core.

Many cycles of cores representative of the Sequoyah Units indicate that the new increased $F_Q(z)$ limit will be met for Sequoyah reload cores operating with the increased radial peaking factor $F_{\Delta H}^N$ limit. Actual Sequoyah reload cores will employ the usual methods of enrichment variation and burnable absorber usage to ensure compliance with the new COLR peaking factor limits. Cycle specific reload core analysis performed in accordance with the methodology described in References 1 and 2 will demonstrate that the new total peaking factor limits will be met.

4.4 FQ Conclusion

The increase in nuclear enthalpy rise hot channel factor and heat flux hot channel factor will be met on a cycle specific basis. No changes to the current methodology will be required. The increased total peaking factor limit will have no impact on other key safety parameters used as input to the FSAR Chapter 15 accident analyses.

4.5 FQ References

1. Sequoyah Nuclear Plant Final Safety Analysis Report - USNRC Docket No. 50-327 and 50-328
2. Davidson, S. L. (Ed.), et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

5.0 Sequoyah Units 1 and 2 Technical Specification Basis Changes for Mini-RTDP

The following Basis changes in the Sequoyah Units 1 and 2 Technical Specifications reflect the implementation of the Mini-RTDP method:

5.1 Section 2.1 SAFETY LIMITS (page B 2-1)

Reference to the correlation limit is replaced with the design limit which is calculated using the Mini-RTDP procedure.

5.1.1 Basis for the change:

The DNB design basis is such that there is at least a 95 percent probability that DNB will not occur on the limiting fuel rod during normal operation, operational transients, and any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95 percent confidence level. This criterion is met by limiting the minimum DNBR to a design limit DNBR. The value of the design limit DNBR depends on the thermal design method selected.

To produce margin to offset penalties such as those due to rod bow and transition core, and for core design flexibilty, the design limit DNBR values are increased to values designated as the safety analysis DNBR limit. The safety analysis DNBR limits are used when performing the Thermal Hydraulics and reactor safety analysis.

In the Standard Thermal Design Procedure (STDP) the design limit DNBR is set equal to the correlation DNBR limit. Then, all design parameters are treated in a conservative way from a DNBR standpoint; that is, uncertainties are added to all parameters to give the lowest minimum DNBR.

The Mini Revise Thermal Design Procedure (Mini-RTDP), which as reviewed and approved by NRC, and a staff evaluation [1] was issued in 1989 conservatively satisfies the design criterion that protects against DNB in a PWR core, while providing DNBR margin. In the mini-RTDP, the uncertainties on nuclear and thermal parameters, the fuel fabrication parameters are statistically combined with the uncertainties on THINC-IV and the transient code.

The resulting value is then combined statistically with the DNB correlation uncertainty to define the design limit DNBR. The rest of the parameters such as; reactor power, flow, temperature, pressure and bypass flow are excluded from the statistical combination process. The appropriate dervative plant initial condition assumptions are used for these parameters. Therefore, the design limit DNBR defined by mini-RTDP is different from the DNBR limit of the correlation used.

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1. S.Ray, "Mini Revised Thermal Design Procedure (mini-RTDP)", WCAP-112178-P-A, October 1989.

5.2 Section 3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS (page B 3/4 2-4)

The following sentence:

Margin has been retained between the DNBR value used in the safety analysis (1.38) and the WRB-1 correlation limit (1.17) to complety offset the rod bow penalty.

Is replaced by:

Margin has been retained between the DNBR value used in the safety analysis and the design DNBR limit to completly offset the rod bow penalty.

5.2.1 Bases for change:

The argument presented to support the change in Section 2.1 applies.

6.0 Assessment of Unreviewed Safety Questions

1. Will the probability of an accident previously evaluated in the SAR be increased?

No. As addressed in this safety evaluation, all non-LOCA transients affected by an increased F_{DH} and F_Q were reanalyzed utilizing the NRC approved Mini RTDP (WCAP-12178-P-A) or evaluated and found to adhere to the safety analysis acceptance criteria. The increased F_{DH} and F_Q have no impact on the remaining non-LOCA transients. Therefore, the probability of an accident occurring that is already evaluated in the SAR will not increase.

Since the total core peaking factors affect only 1/4 ft. length of one fuel rod, the hot channel peaking factors affect one fuel rod in rod heat up models, and all analysis acceptance criteria continue to be met, these changes will not increase the probability of the occurrence of the LOCA.

2. Will the consequences of an accident previously evaluated in the SAR be increased?

No. Per the discussion presented in the Evaluation section, all the applicable non-LOCA acceptance criteria are still met for the transients evaluated and for the events reanalyzed. Additionally, no new limiting single failure is introduced by the proposed change. Therefore, there is no potential for an increase in the consequences of an accident previously evaluated in the SAR.

The increase in core peaking factors would not adversely affect the safeguards systems actuations or the accident mitigation capabilities important to LOCA events. This conclusion is based on the fact that the following countermeasures intended to limit the consequences of a LOCA (as described in the Sequoyah Nuclear Plant FSAR) would not be compromised.

a. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

b. Injection of borated water provides heat transfer from core and prevents excessive clad temperature.
Therefore, the consequences of a LOCA will not be increased.

3. May the possibility of an accident which is different than any already evaluated in the SAR be created?

No. Increasing the F_{DH} and F_Q does not introduce a new accident initiator mechanism. Thus, no new accident will be created.

4. Will the probability of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

   No. Increasing the F_{DH} to 1.62 and increasing the F_Q to 2.40 will not adversely affect the operation of the Reactor Protection System, any of the protection setpoints, or any other device required for accident mitigation.

   Increased core peaking factors will not increase the probability of the malfunction of any equipment important to safety as concerns the LOCA.

5. Will the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?

   No. As discussed in the responses to questions 2 and 4, there is no possibility of increasing the consequences of a malfunction of equipment for an increase in F_{DH} and F_Q as defined in the attached safety evaluation.

   Increased core peaking factors will not increase the consequences of the malfunction of any equipment important to safety as concerns the LOCA.

6. May the possibility of a malfunction of equipment important to safety different than already evaluated in the SAR be created?

   No. As discussed in question 4, an increase in F_{DH} and an increase in F_Q will not impact any other equipment important to safety.

7. Will the margin of safety as described in the bases to any technical specification be reduced?

   No. As discussed in the safety evaluation, the proposed increase in F_{DH} and F_Q will not invalidate any of the non-LOCA conclusions presented in the UFSAR accident analyses. Thus, there is no reduction in the margin of safety.
   Increased core peaking factors will not create the possibility of a malfunction of equipment important to safety as concerns the LOCA different than that already evaluated in the FSAR.

   Since the calculated PCT of 2069°F is within the limit of 2200°F set by 10 CFR 50.46, an increase in core peaking factors does not reduce this margin of safety as defined in the bases to any technical specifications.

7.0 Conclusions

The preceding evaluation demonstrates that all safety analyses acceptance criteria have been met and supports the Tech Spec and COLR changes provided.

8.0 LOCA PCT Rack-Ups

STANDARDIZED SUMMARY FORMAT SHEET FOR
REPORTING OF 10CFR50.46 MARGIN UTILIZATION
LARGE BREAK LOCA

PLANT NAME: SEQUOYAH PLANT UNIT 1
UTILITY NAME: TENNESSEE VALLEY AUTHORITY

A. ANALYSIS OF RECORD — PCT = 2169 °F

Comments: Evaluation Model: BASH, FQT = 2.40, FΔH = 1.62, SGTP = 10 %,
Other: once burnt, standard fuel, 350 psi fill pressure

B. PRIOR LOCA MODEL ASSESSMENTS - 1991 — ΔPCT = + 0°F
(Permanent Assessment of PCT Margin - Letter #: TVA-91-181)
1. SG-TUBE SEISMIC/LOCA ASSUMPTION — ΔPCT = + 20°F

C. CURRENT LOCA MODEL ASSESSMENTS - 02/1992
(Permanent Assessment of PCT Margin - Letter #: ______)

D. 10CFR50.59 SAFETY EVALUATIONS
(Permanent Assessment of PCT Margin) — ΔPCT = + 0°F

E. CURRENT LOCA MODEL ISSUES (Temporary Use of PCT Margin):
1. LB-LOCA POWER DISTRIBUTION ASSUMPTION — ΔPCT = NOTE 1
2. CORE AVERAGE ZIRC-WATER REACTION — ΔPCT = NOTE 2
3. BOL IFBA IMPACT ON SAFETY ANALYSIS — ΔPCT = NOTE 3

F. OTHER LOCA RELATED MARGIN ALLOCATION (Temporary Use of PCT Margin):
1. ECCS FLOW INCONSISTENCIES (1989) — ΔPCT = NOTE 4
2. ECCS FLOW MEASUREMENT INACCURACY (1990) — ΔPCT = NOTE 5
3. COLD LEG STREAMING TEMPERATURE GRADIENT — ΔPCT = + 10°F

G. OTHER MARGIN ALLOCATIONS (Temporary Use of PCT Margin):
1. ANALYSIS MARGINS USED: ______ — ΔPCT = + 0°F
2. PLANT MARGINS USED: 5% SGTP (SECL-88-417 Rev. 1) — ΔPCT = - 20°F
3. FUEL MARGINS USED: ______ — ΔPCT = + 0°F

LICENSING BASIS PCT + MARGIN ALLOCATION — PCT = 2179 °F

Notes:

1. 0°F PCT Margin allocated to date on the basis of the core design axial offset.
2. An additional 0.7% of Zr-H_2O margin allocated for all plants for reasonable assurance of safe operation within the licensing basis.
   No ECCS Analysis PCT margin is allocated for this issue on the basis that, more likely than not, the issue will be resolved without any change to the ECCS analysis results or the ECCS Evaluation Model.
3. No margin allocated to date because of high burn-up currently on the Cycle 6 IFBA fuel.
4. The ECCS Analysis of record has addressed this issue be modelling safety injection pump line flow imbalances as specified by TVA.
5. No ECCS Analysis PCT margin is allocated for this issue; no specific safety evaluation has been performed.

STANDARDIZED SUMMARY FORMAT SHEET FOR
REPORTING OF 10CFR50.46 MARGIN UTILIZATION
LARGE BREAK LOCA

PLANT NAME: SEQUOYAH PLANT UNIT 2
UTILITY NAME: TENNESSEE VALLEY AUTHORITY

A. ANALYSIS OF RECORD — PCT = 2069 °F

Comments: Evaluation Model: BASH, FQT = 2.40, FΔH = 1.62, SGTP = 10 %,
Other: fresh, generic, V-5H fuel, 275 psi fill pressure

B. PRIOR LOCA MODEL ASSESSMENTS - 1991 — ΔPCT = + 0°F
(Permanent Assessment of PCT Margin - Letter #: TVA-91-181)
1. SG-TUBE SEISMIC/LOCA ASSUMPTION — ΔPCT = + 20°F

C. CURRENT LOCA MODEL ASSESSMENTS - 02/1992
(Permanent Assessment of PCT Margin - Letter #:)
1. LB-LOCA BURST & BLOCKAGE ASSUMPTION — ΔPCT = + 0°F

D. 10CFR50.59 SAFETY EVALUATIONS
(Permanent Assessment of PCT Margin)
1. Letter: SECL-90-537 Issue: Loose Fuel Parts — ΔPCT = + 45°F

E. CURRENT LOCA MODEL ISSUES (Temporary Use of PCT Margin):
1. LB-LOCA POWER DISTRIBUTION ASSUMPTION — ΔPCT = NOTE 1
2. CORE AVERAGE ZIRC-WATER REACTION — ΔPCT = NOTE 2

F. OTHER LOCA RELATED MARGIN ALLOCATION (Temporary Use of PCT Margin):
1. ECCS FLOW INCONSISTENCIES (1989) — ΔPCT = NOTE 3
2. ECCS FLOW MEASUREMENT INACCURACY (1990) — ΔPCT = NOTE 4
3. COLD LEG STREAMING TEMPERATURE GRADIENT — ΔPCT = + 10°F

G. OTHER MARGIN ALLOCATIONS (Temporary Use of PCT Margin):
1. ANALYSIS MARGINS USED: — ΔPCT = + 0°F
2. PLANT MARGINS USED: 5% SGTP (SECL-88-417 rev. 1) — ΔPCT = - 20°F
3. FUEL MARGINS USED: — ΔPCT = + 0°F

LICENSING BASIS PCT + MARGIN ALLOCATION — PCT = 2124 °F

Notes:

1. 0°F PCT Margin allocated on basis of the core design axial offset.
2. An additional 0.7% of $Zr-H_2O$ margin allocated for all plants for reasonable assurance of safe operation within the licensing basis.
   No ECCS Analysis PCT margin is allocated for this issue on the basis that, more likely than not, the issue will be resolved without any change to the ECCS analysis results or the ECCS Evaluation Model.
3. The ECCS Analysis of record has addressed this issue be modelling safety injection pump line flow imbalances as specified by TVA.
4. No ECCS Analysis PCT margin is allocated for this issue; no specific safety evaluation has been performed.

STANDARDIZED SUMMARY FORMAT SHEET FOR
REPORTING OF 10CFR50.46 MARGIN UTILIZATION
SMALL BREAK LOCA

PLANT NAME: SEQUOYAH PLANT UNIT 2 (BOUNDS UNIT 1)
UTILITY NAME: TENNESSEE VALLEY AUTHORITY

A. ANALYSIS OF RECORD — PCT = 2105 °F

Comments: Evaluation Model: NOTRUMP, FQT = 2.70, FΔH = 1.70, SGTP = 15 %,

B. PRIOR LOCA MODEL ASSESSMENTS - 1991 — ΔPCT = + 0°F
(Permanent Assessment of PCT Margin - Letter #:____________)

C. CURRENT LOCA MODEL ASSESSMENTS - 02/1992
(Temporary Assessment of PCT Margin - Letter #:____________)
1. SB-LOCA ROD INTERNAL PRESSURE ASSUMPTION — ΔPCT = + 0°F
2. SB-LOCA BURST AND BLOCKAGE — ΔPCT = + 26°F (NOTE 5)
3. SECONDARY SIDE MODELING IN SB/INPUT CORRECTIONS — ΔPCT = - 147°F
4. SB-LOCA NOTRUMP BESSEL FUNCTION — ΔPCT = + 11°F

D. 10CFR50.59 SAFETY EVALUATIONS
(Permanent Assessment of PCT Margin)
1. Letter: SECL-90-537 Issue: Loose Fuel Parts — ΔPCT = + 37°F

E. CURRENT LOCA MODEL ISSUES (Temporary Use of PCT Margin):
1. CORE AVERAGE ZIRC-WATER REACTION — ΔPCT = NOTE 1
2. RCCA INSERTION ASSUMPTION IN SB-LOCA — ΔPCT = NOTE 2

F. OTHER LOCA RELATED MARGIN ALLOCATION (Temporary Use of PCT Margin):
1. ECCS FLOW INCONSISTENCIES (1989) — ΔPCT = NOTE 3
2. ECCS FLOW MEASUREMENT INACCURACY (1990) — ΔPCT = NOTE 4
3. COLD LEG STREAMING TEMPERATURE GRADIENT — ΔPCT = + 2°F

G. OTHER MARGIN ALLOCATIONS (Temporary Use of PCT Margin):
1. ANALYSIS MARGINS USED:____________ — ΔPCT = + 0°F
2. PLANT MARGINS USED:____________ — ΔPCT = + 0°F
3. FUEL MARGINS USED:____________ — ΔPCT = + 0°F

LICENSING BASIS PCT + MARGIN ALLOCATION — PCT = 2034 °F

Notes:

1. An additional 0.7% of $Zr\text{-}H_2O$ margin allocated for all plants for reasonable assurance of safe operation within the licensing basis.
2. No ECCS Analysis PCT margin is allocated for this issue on the basis that, more likely than not, the issue will be resolved without any change to the ECCS analysis results or the ECCS Evaluation Model.
3. The ECCS Analysis of record has addressed this issue be modelling safety injection pump line flow imbalances as specified by TVA.
4. No ECCS Analysis PCT margin is allocated for this issue; no specific safety evaluation has been performed.
5. PCT margin allocated on basis of the core design axial offset.

9.0 List of UFSAR, Tech Spec and COLR Mark-Ups

9.1 UFSAR Markups

9.1.1 List of Figures

9.1.2 For the Startup of an Inactive Reactor Coolant Pump the following changes were made: Reference 3 updated, Table 15.2-1 (sheet 3), and Figures 15.2.6-1 through 15.2.6-4 were replaced.

9.1.3 For the Complete Loss of Flow event the following changes were made: Text markups, Reference 8 updated, Table 15.3.4-1 revised, and Figures 15.3.4-1 through 15.3.4-3 were replaced.

9.1.4 For the Partial Loss of Flow event the following changes were made: Text changes, Reference 3 update, Table 15.2-1 (sheet 2) revised, and Figures 15.2.5-1 through 15.2.5-3 were replaced.

9.1.5 For the RCCA Ejection event the following changes were made: Text changes, References 25 and 29 updated, Tables 15.4.1-12 (sheet 3) and 15.4.6-1 were revised, and Figures 15.4.6-1 through 15.4.6-2 were replaced.

9.1.6 For the " Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident) " event, changes were made to the text of section 15.4.1, Tables 15.4-1-1, 15.4.1-3, 15.4.1-6 and 15.4.1-7 and Figures 15.4.1-1 through 15.4.1-20.

9.1.7 For the Thermal and Hydraulic Design, changes were made to :

Tables 4.1-1 (sheet 1), 4.3.2-2 (sheet 1) and 4.4.2-1 (sheets 1 and 2)

Pages 4.3-18, 4.4-1, 4.4-1a, 4.4-8,- 9, -24, -31, -38, -39.

9.2 Affected Technical Specification and COLR Technical Specification

Technical Specifications

Page B 2-1

Page B3/4 2-4

COLR
(Unit-1)

Page 2 of 10
Page 3 of 10

COLR
(Unit- 2)

Page 2 of 11
Page 3 of 11

10.0 Appendix: UFSAR Markups, Tech spec Markups, COLR Markups

10.1 UFSAR Markups

10.1.1 Non LOCA

LIST OF FIGURES (Continued)

UFSAR Inserts - LIST OF FIGURES

LIST OF FIGURES (Continued)

Replace with Insert C

8

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UFSAR Inserts - LIST OF FIGURES

Insert C:

LIST OF FIGURES (Continued)

Replace with Insert D:

8

K9

UFSAR Inserts - LIST OF FIGURES

Insert D:

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (See Table 15.1.2-2). The total integrated Doppler reactivity from 0 to 100% power is assumed to be 0.016 Δk. The lowest absolute magnitude of the moderator temperature coefficient (0.0 Δk/°F) is assumed since this results in the maximum hot-spot heat flux during the initial part of the transient when the minimum DNBR is reached.

Flow Coastdown

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

NOT REVIEWED

Results

6 ok   6 ok

The calculated sequence of events is shown on Table 15.2-1 for the cases analyzed. Figures 15.2.~~5~~-1 through 15.2.~~5~~-3 show the loop coastdowns, the core flow coastdowns, the nuclear power coastdowns and the average and hot channel heat flux coastdowns for each of the cases. The minimum DNBR for each of the cases is not less than the safety analysis limit.

15.2.5.3 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit at any time during the transient. Thus there will be no cladding damage and no release of fission products to the Reactor Coolant System.

15.2.6 Startup Of An Inactive Reactor Coolant Loop

15.2.6.1 Identification of Causes and Accident Description

If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the unit be brought to a load of less than 25% of full power prior to staring a pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without

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bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core which causes a rapid reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (a fault of moderate frequency).

Should the startup of an inactive reactor coolant pump accident occur, the transient will be terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three loop operation.

15.2.6.2 Analysis of Effects and Consequences

Method of Analysis

This transient is analyzed by three digital computer codes. The LOFTRAN Code (4) is used to calculate the loop and core flow, nuclear power and core pressure and temperature transients following the startup of an idle pump. FACTRAN (3) is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC Code (see Section 4.4) is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

Plant characteristics and initial conditions are discussed in Section 15.1.2. In order to obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

1. Initial conditions of maximum core power and reactor coolant average temperatures and minimum reactor coolant pressure resulting in minimum initial margin to DNB. These values are consistent with the maximum steady state power level allowed with three loops in operation. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

2. Following initiation of startup of the idle pump, flow in the inactive loop reverses and accelerates to its nominal full flow value in approximatley 7 seconds.

3. A conservatively large moderator density coefficient (see Section 15.1.6).

4. A conservatively small (absolute value) negative Doppler power coefficient (see Section 15.1.6).

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

6. The reactor trip is assumed to occur on low coolant loop flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

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Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.2.6-1 through 15.2.6-4. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average water temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit. See Section 4.4 for a description of the DNBR design basis.

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient, this decrease is due both to the increase in reactor coolant flow and, as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.2.6-2.

The calculated sequence of events for this accident is shown on Table 15.2-1. The transient results illustrated in Figures 15.2.6-1 through 15.2.6-4 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.2.6.3 Conclusions

The transient results show that the core is not adversely affected, i.e., there is considerable margin to the DNB safety analysis limit.

NOT Reviewed

~~15.2.7 Loss Of External Electrical Load And/Or Turbine Trip~~

~~15.2.7.1 Identification of Causes and Accident Description~~

~~Major load loss on the plant can result from loss of external electrical load or from a turbine trip. For either case off-site power is available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of all AC power (station blackout) is analyzed in Subsection 15.2.9. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the steam generator shell. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise, which in turn causes coolant expansion, pressurizer insurge, and RCS pressure rise.~~

~~For a turbine trip, the reactor would be tripped directly (unless below approximately 50% power) from a signal derived from the turbine autostop oil pressure (Westinghouse Turbine) and turbine stop valves. The turbine stop valves close on loss of autostop oil pressure actuated by one of a number of possible turbine trip signals. Turbine-trip initiation signals include:~~

~~1. Generator Trip~~

2. Low Condenser Vacuum

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6. Turbine Load - Turbine load was assumed constant until the electro- hydraulic governor drives the throttle valve wide open. Then turbine load drops as steam pressure drops.

7. Reactor Trip - Reactor Trip was initiated by low pressurizer pressure assumed at a conservatively low value of 1775 psia.

Results

The transient response is shown in Figures 15.2.14-1 and 15.2.14- 2. Nuclear power starts decreasing immediately due to boron injection but steam flow does not decrease until 15 seconds into the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The low pressure trip set point is reached at 64 seconds and rods start moving into the core at 66 seconds.

Not Reviewed

After trip, pressures and temperatures slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat.

15.2.14.3 Conclusions

Results of the analysis show that spurious safety injection with or without immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System.

DNB ratio is never less than the initial value. Thus there will be no cladding damage and no release of fission products to the reactor coolant system.

If the reactor does not trip immediately, the low pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident.

15.2.15 References

1. W. C. Gangloff, "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486, May 1971.

2. D. H. Risher, Jr., R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary), WCAP-8028-A (Non-Proprietary), January 1975. B

Replace with Insert A

3. ~~C. Hunin, "FACTRAN, A Fortran Code for Thermal Transients in UO_2 Fuel Rod," WCAP-7908, June 1972.~~

4. Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non_Proprietary), April 1984. B

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FSAR 15.2.6 - Startup of an Inactive Reactor Coolant Loop

Inserts for reanalysis due to increased $F_{\Delta H}$

Insert A : Change Reference 3 to the following:

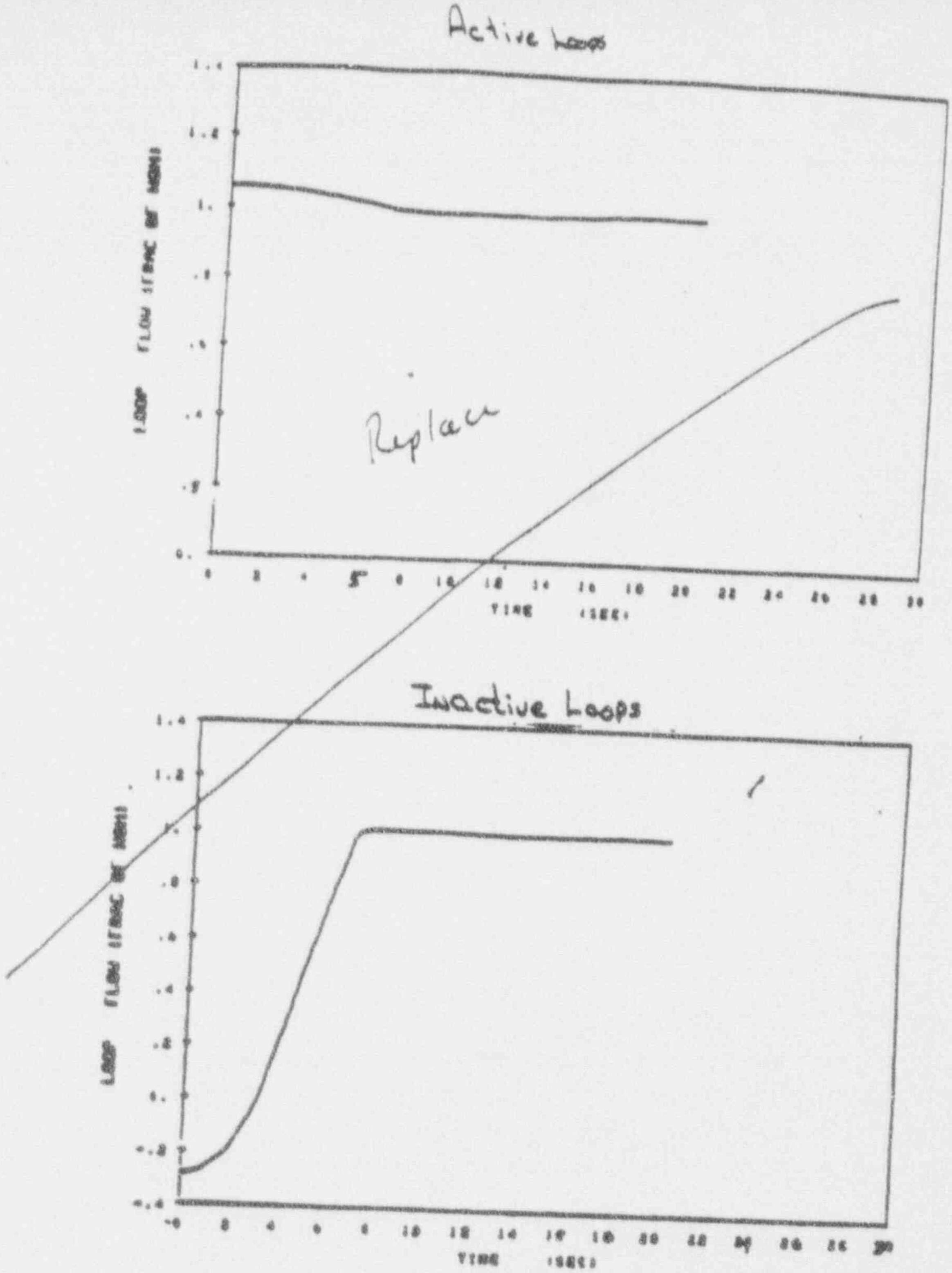
3. Hargrove, H. G., "FACTRAN - A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

TABLE 15.2-1 (Sheet 3)
(Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION II EVENTS

Accident	Event	Time (Sec.)
Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	Initiation of pump startup	0
	Power reaches P-8 trip setpoint	2.8
	Rods begin to drop	3.3
	Minimum DNBR occurs	4.0
Loss of External Electrical Load		
1. With pressurizer Control (BOL)	Loss of electrical load	0
	Initiation of steam release from steam generator safety valves	8.0
	Over temperature ΔT Reactor Trip Setpoint reached	8.9
	Rods begin to drop	10.9
	Minimum DNBR occurs	11.5
	Peak pressurizer pressure occurs	12.0

NOT REVIEWED



SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive Reactor Coolant Pump

FIGURE 15.2.6-1

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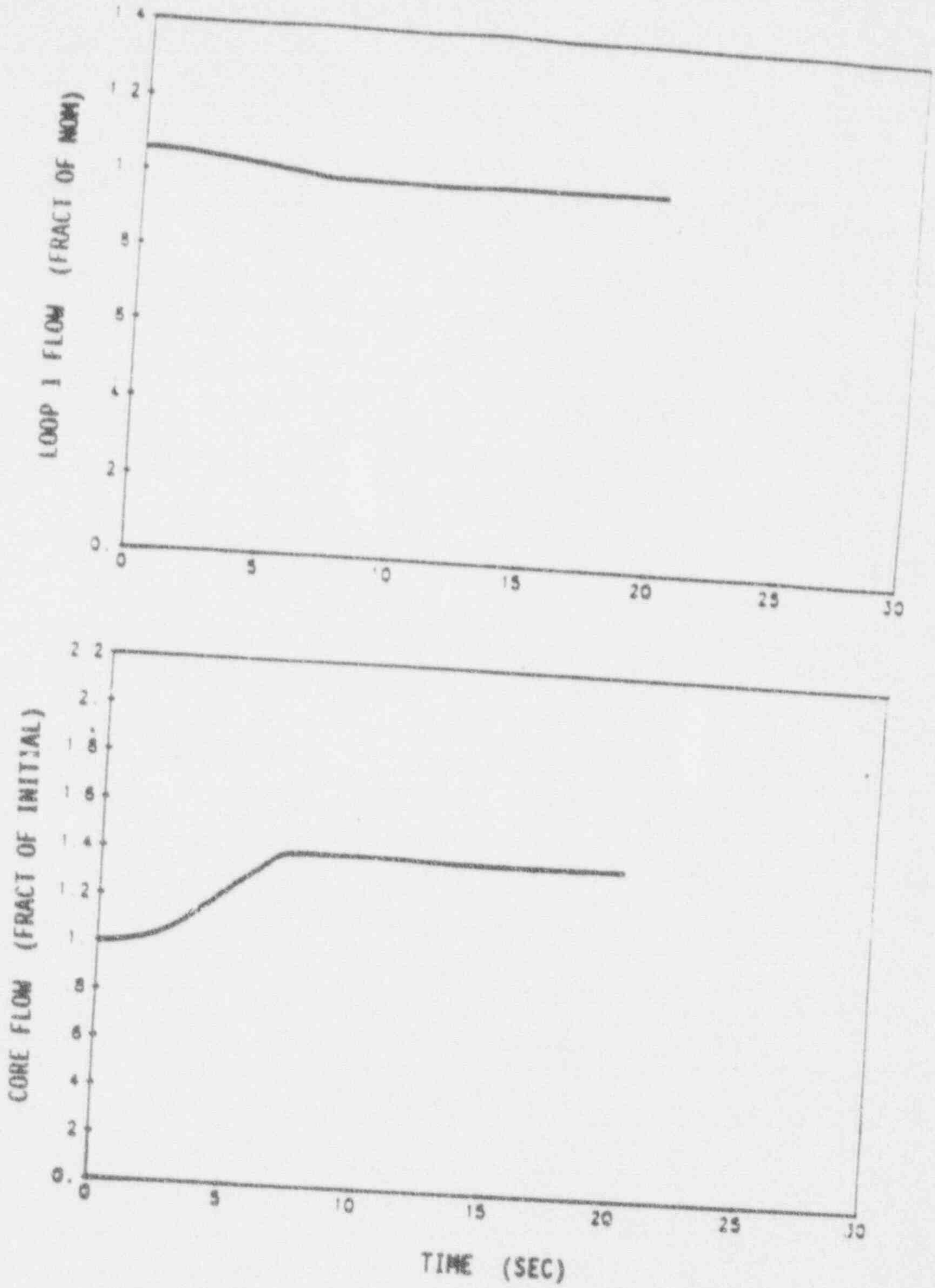
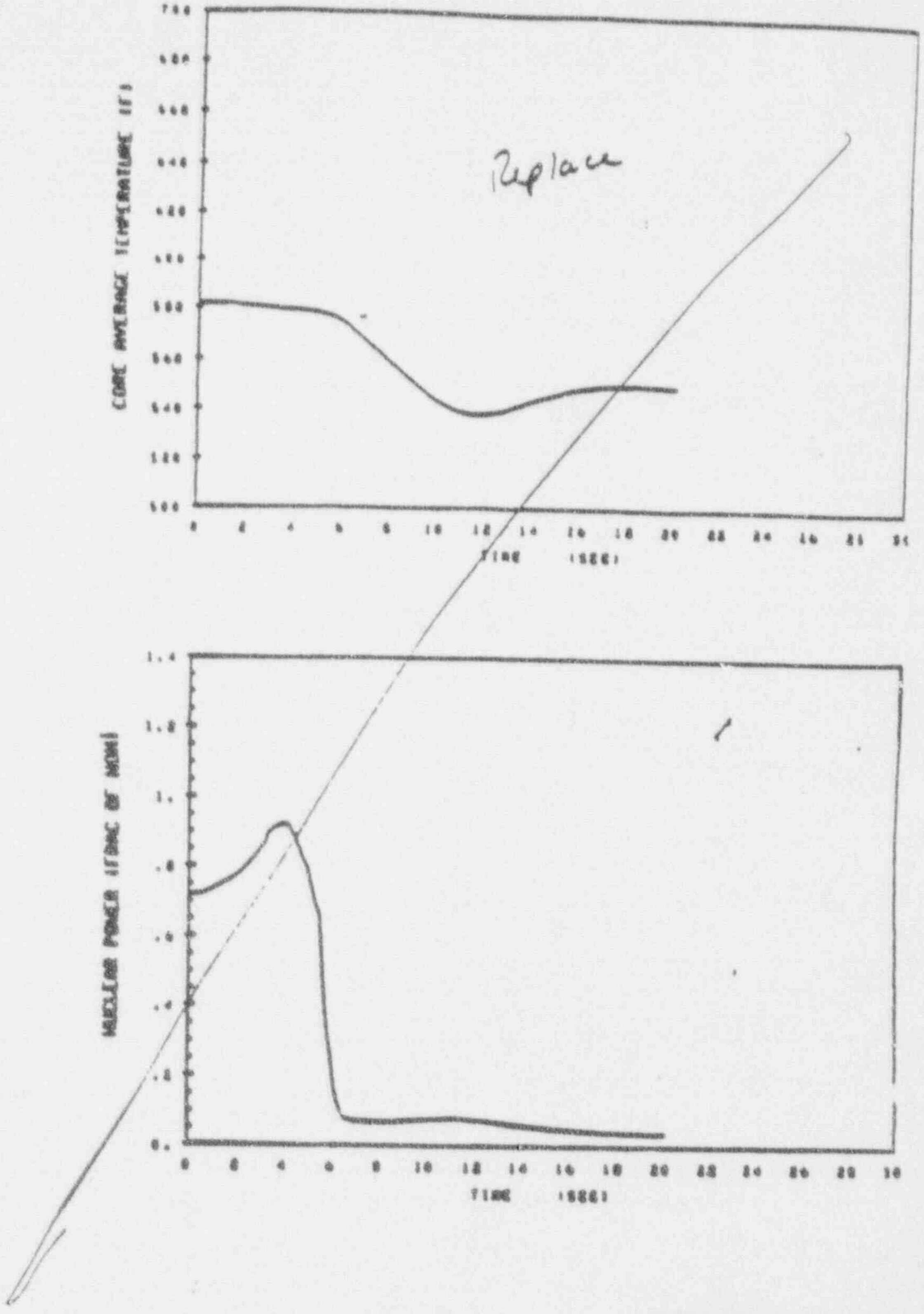


SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive Reactor Coolant Pump
Active Loop Flow vs. Time;
Core Flow vs. Time

Figure 15.2.6-1



SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive
Reactor Coolant Pump

FIGURE 15.2.6-2

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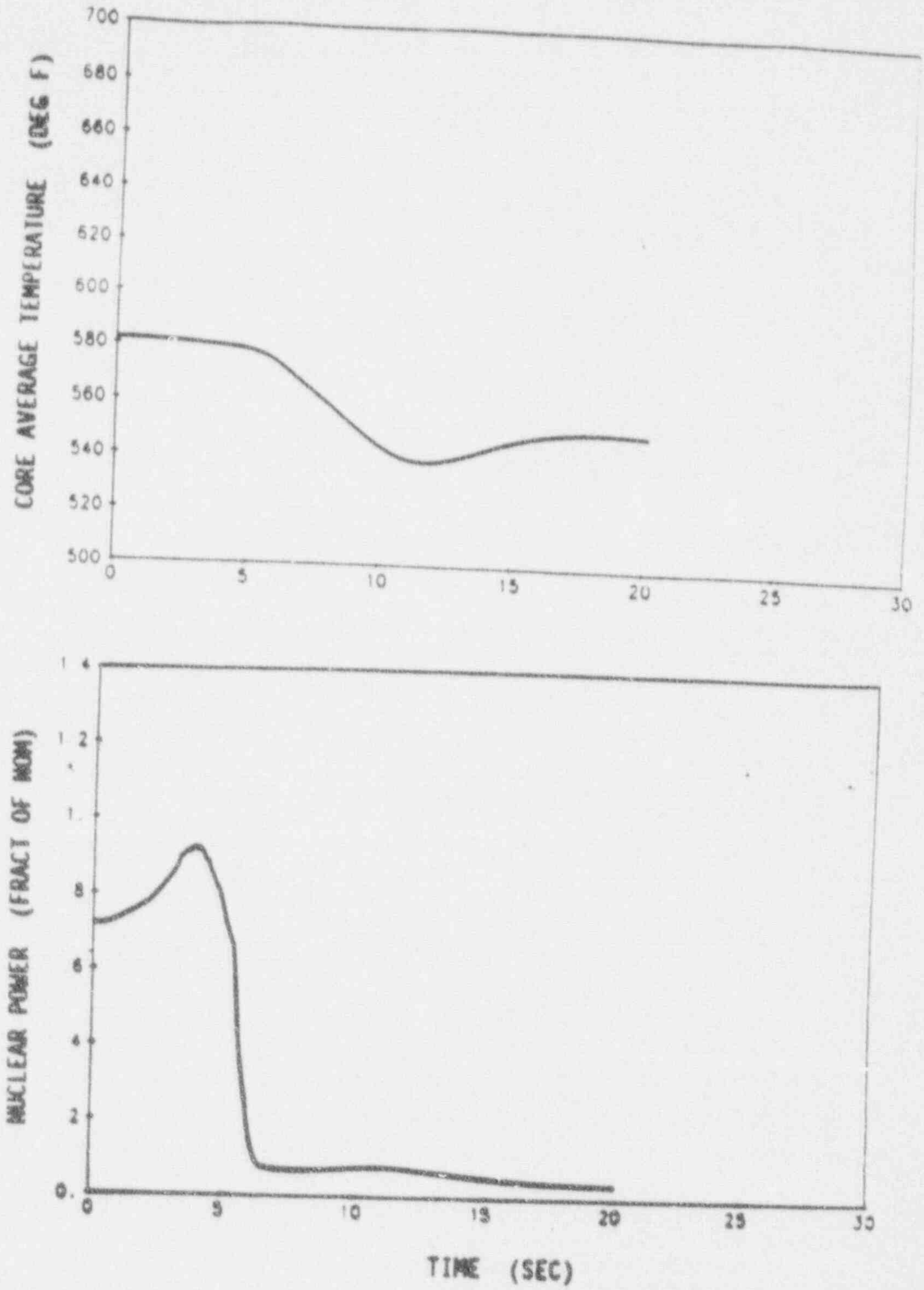

SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive Reactor Coolant Pump
Core Average Temperature vs. Time;
Nuclear Power vs. Time

Figure 15.2.6-2

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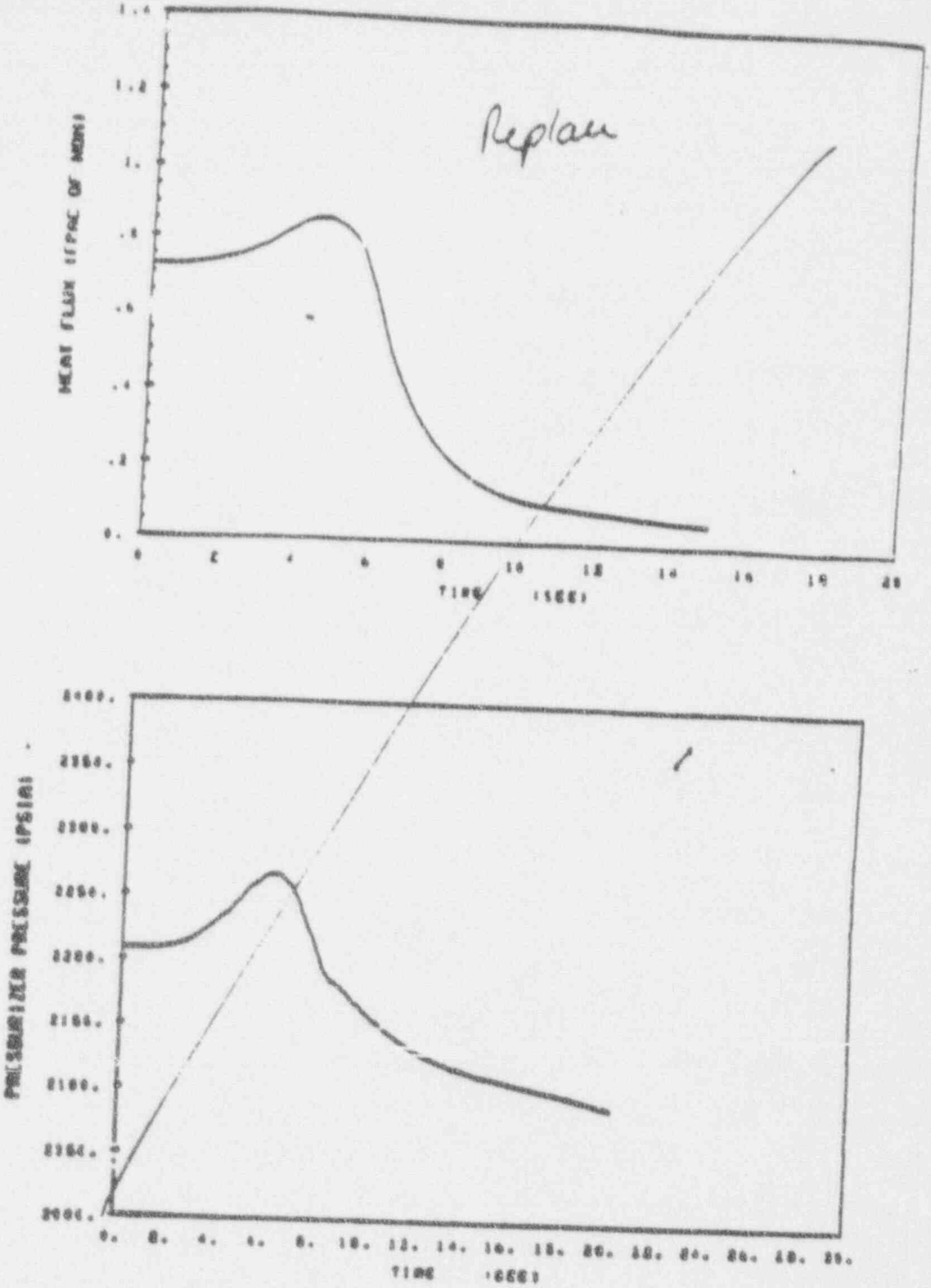
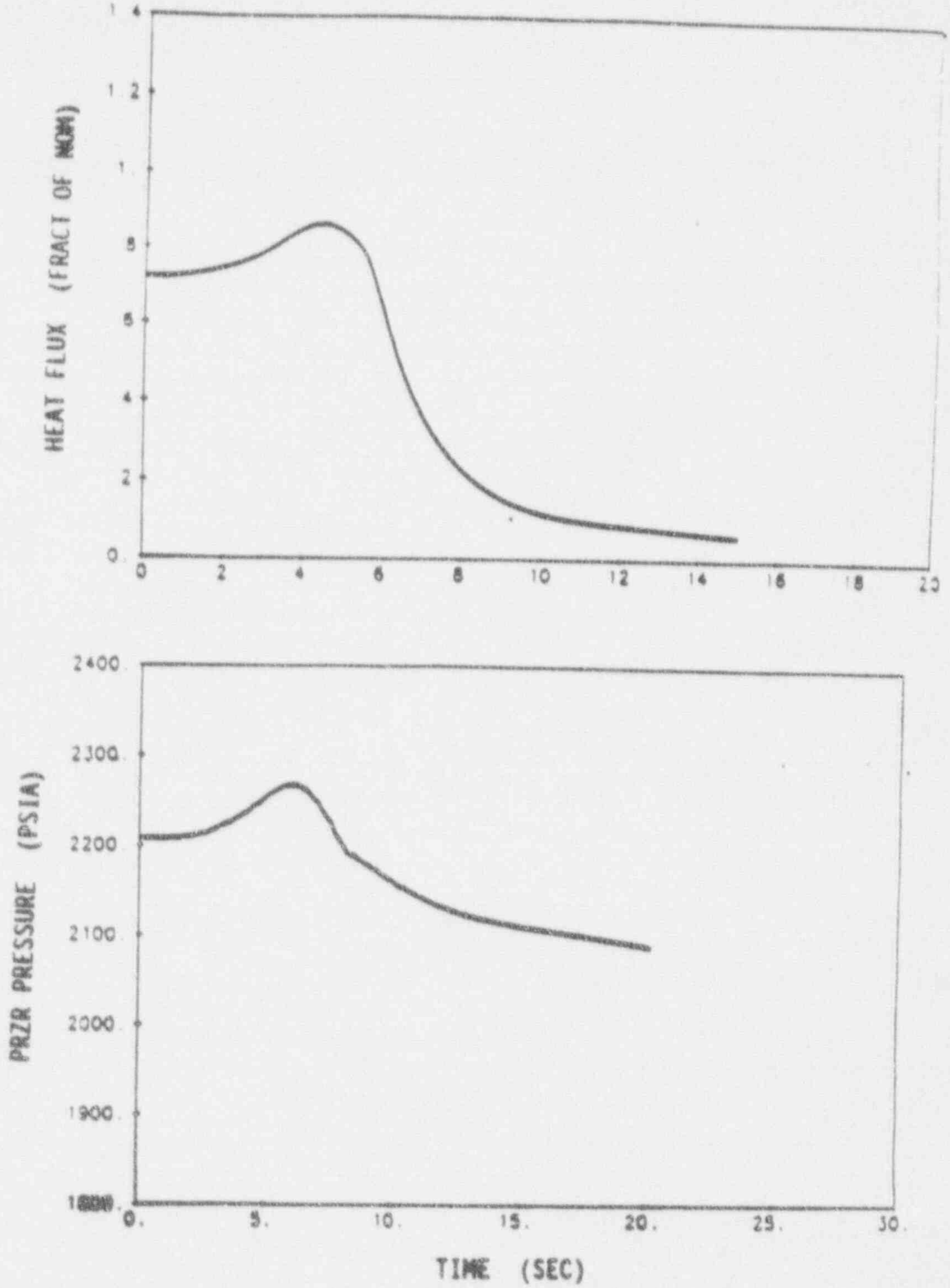


SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive Reactor Coolant Pump

FIGURE 15.2.6-3



SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive Reactor Coolant Pump
Heat Flux vs. Time;
Pressurizer Pressure vs. Time

Figure 15.2.6-3

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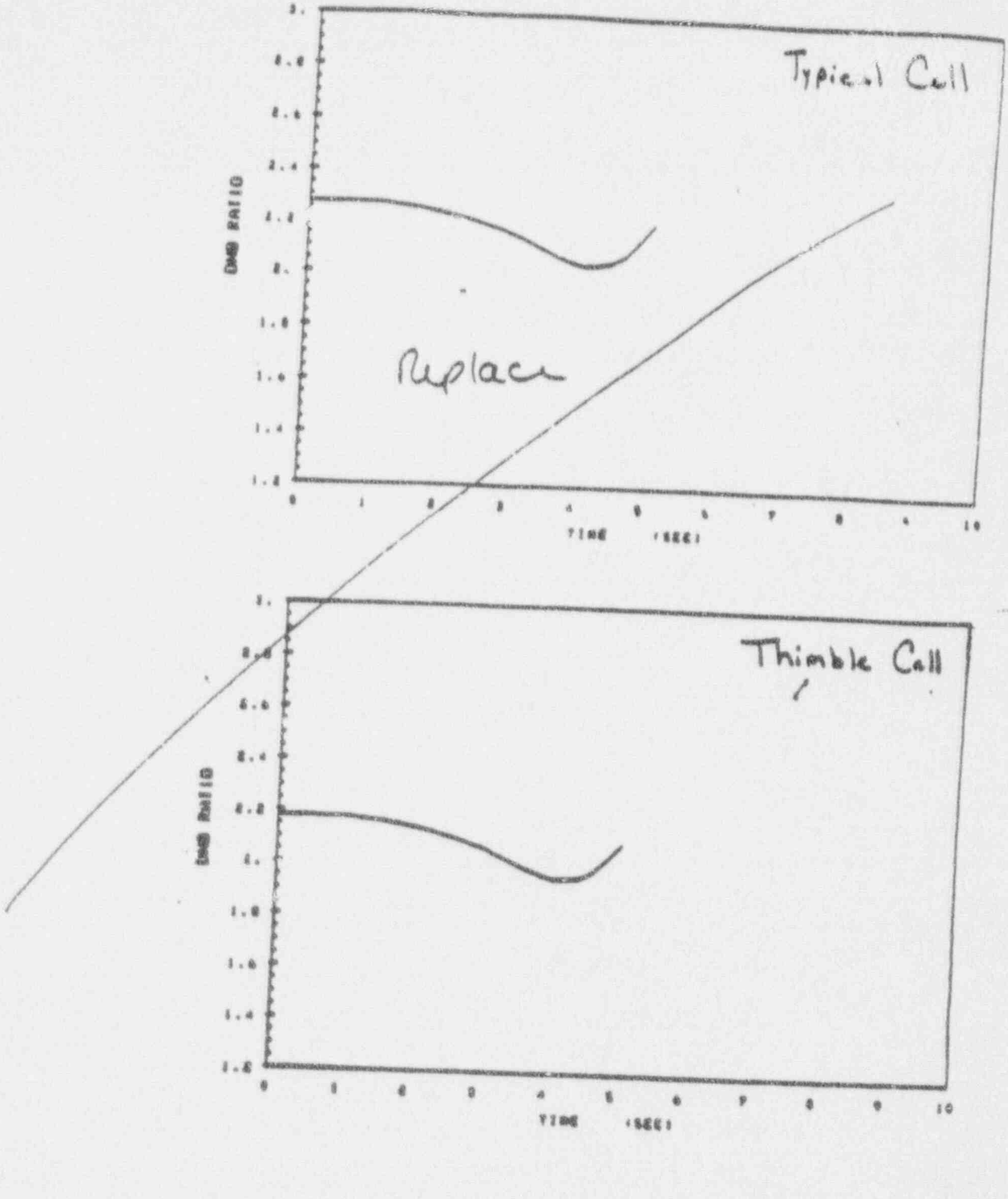
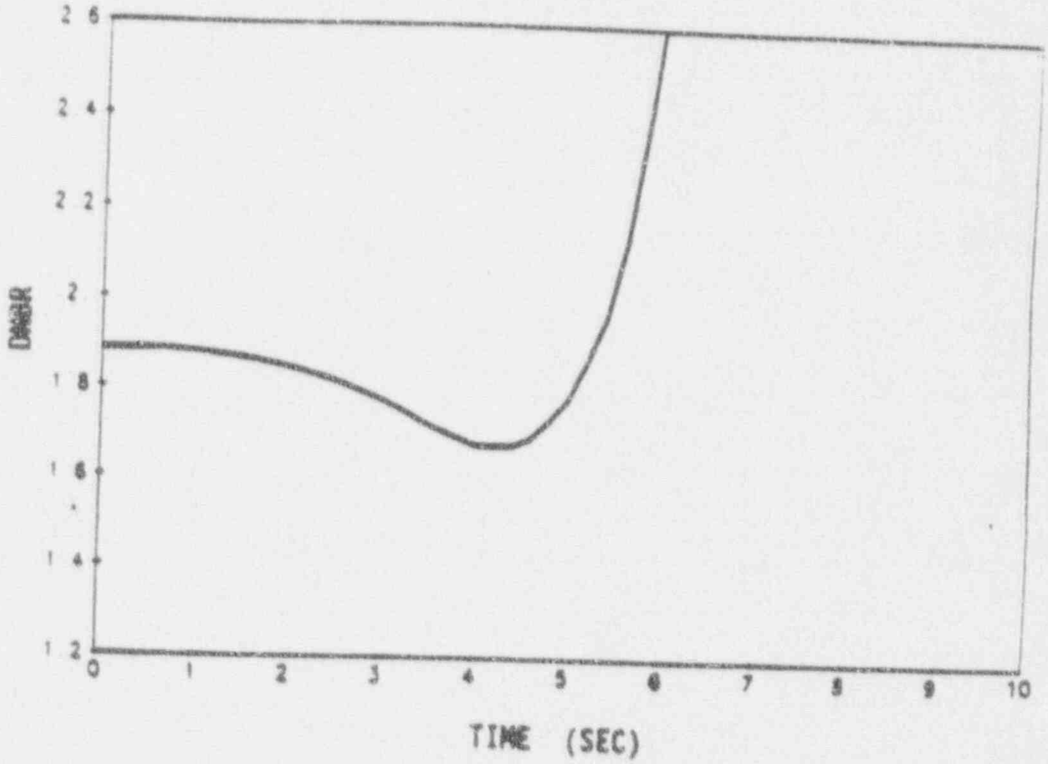


SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive Reactor Coolant Pump

FIGURE 15.2.6-4



SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Startup of an Inactive Reactor Coolant Pump
DNBR vs. Time

Figure 15.2.6-4

15.3.4 Complete Loss of Forced Reactor Coolant Flow

15.3.4.1 Identification of Causes and Accident Description

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

1. Undervoltage or underfrequency on reactor coolant pump power supply busses.

2. Low reactor coolant loop flow.

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

The reactor trip on reactor coolant pump underfrequency is provided to open the reactor coolant pump breakers and trip the reactor for an underfrequency condition, resulting from frequency disturbances on the major power grid. The trip disengages the reactor coolant pumps from the power grid so that the pumps' kinetic energy is available for full coastdown.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. It also serves as a backup to the undervoltage and underfrequency trips. This function is generated by two out of thre low flow signals per reactor coolant loop. Above approximately 35 percent power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10 percent power and 35 percent power (Permissive 7 and Permissive 8), low flow in any two loops will actuate a reactor trip.

Normal power for the reactor coolant pumps is supplied through busses from a transformer connected to the generator. Each pump is on a separate bus. When generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip, where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

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15.3.4.2 Analysis of Effects and Consequences

Method of Analysis

The complete loss of flow transient has been analyzed for a loss of four pumps with four loops in operation.

The transient is analyzed by three digital computer codes. First, the LOFTRAN code (Reference ~~8~~ 7) 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 code (Reference ~~8~~) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The WRB-1 correlation is used for DNBR calculation. The DNBR transients presented represent the minimum of the typical or thimble cell.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2, except that following the loss of supply to all pumps at power, a reactor trip is actuated by either bus undervoltage or bus underfrequency. An additional exception is 1.5°F which was added to the initial average temperature for conservatism. 8

Insert B

~~The calculated sequence of events is shown on Table 15.3.4-1 for the cases analyzed. Figures 15.3.4-1 through 15.3.4-3 show the core flows, heat fluxes (average and hot channel), and DNB ratios as a function of time for each of the cases. The Reactor is assumed to trip on the undervoltage signal. The DNBR curves for each of the cases is not less than the safety analysis limit.~~ 8 8

15.3.4.3 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit during the transient and thus there is no clad damage or release of fission products to the Reactor Coolant System. 8

Analysis has shown that for frequency decay rates less than ~~5.6~~ 5.0 Hz/second, no reactor coolant pump trip is necessary. A grid analysis was provided for the Sequoyah Nuclear Plant which determined that, for the worst case, the grid decay rate is less than 5.0 Hz/second.

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2. If the reactor is in automatic control mode, withdrawal of a single rod cluster control assembly will result in the immobility of the other rod cluster control assemblies in the controlling bank. The transient will then proceed in the same manner as Case 1 described above. For such cases as above a trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNB ratio in the core of less than the safety analysis limit.

15.3.6.3 Conclusions

Not Reviewed

For the case of one rod cluster control assembly fully withdrawn, with the reactor in the automatic or the manual control mode and initially operating at full power with Bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNBR 1.3 is 5 percent of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before DNB could occur. For case 2 discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard.

15.3.7 References

1. Meyer, P. E., "NOTRUMP, A Nodal Transient Small Break and General Network Code", WCAP-10080-A, August 1985.

2. Lee, H., Rupprecht, S. D., Schwartz, W. R., Tauche, W. D., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10081-A, August 1985.

3. W. A. Bezella, C. L. Caso, A. C. Spencer, "LOCTRA-R2 Program Loss-of-Coolant Transient Analysis," WCAP 7835, January 1972.

4. S. Altomare and R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758, September 1971.

5. R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP 3269-26, September 1963.

6. F. M. Bordelon, "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP 7969, September 1972.

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

Replace with Insert A

~~8. C. Hunin, "FACTRAN, A Fortran-IV Code for Thermal Transients in UO_2 Fuel Rods," WCAP 7908, June 1972.~~

9. F. M. Bordelon, et. al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305. (Non-Proprietary), WCAP-8301 (Proprietary) June, 1974.

8

67

FSAR 15.3.4 - Complete Loss of Forced Reactor Coolant Flow

Inserts for reanalysis due to increased $F_{\Delta H}$

Insert A : Change Reference 8 to the following:

8. Hargrove, H. G., "FACTRAN - A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

Insert B:

The calculated sequence of events is shown in Table 15.3.4-1 for the limiting case analyzed. This case corresponds to a reactor trip occurring on a bus underfrequency condition. Figures 15.3.4-1 through 15.3.4-3 show the resulting transient conditions for the Complete Loss of Flow analysis. Included in these figures are total RCS flow, average and hot channel heat flux, nuclear power, and DNBR, each as a function of time. The minimum DNBR is not less than the safety analysis limit.

TABLE 15.3.4-1

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Accident	Event	Time (sec.)
Complete Loss of Forced Reactor Coolant Flow		
All loops operating, all pumps coasting down	Coastdown begins	0
	Rod Motion begins	1.5
	Minimum DNBR occurs	3.~~7~~5

Replace with the Following

TABLE 15.3.4-1

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

Accident	Event	Time (sec.)
Complete Loss of Forced Reactor Coolant Flow		
All loops operating, all pumps coasting down	All operating pumps begin to coastdown at 5 Hz/sec frequency decay rate	0.0
	Reactor underfrequency trip setpoint reached	0.84
	Rods begin to drop	1.44
	Minimum DNBR occurs	3.6

REPLACE WITH THE FOLLOWING

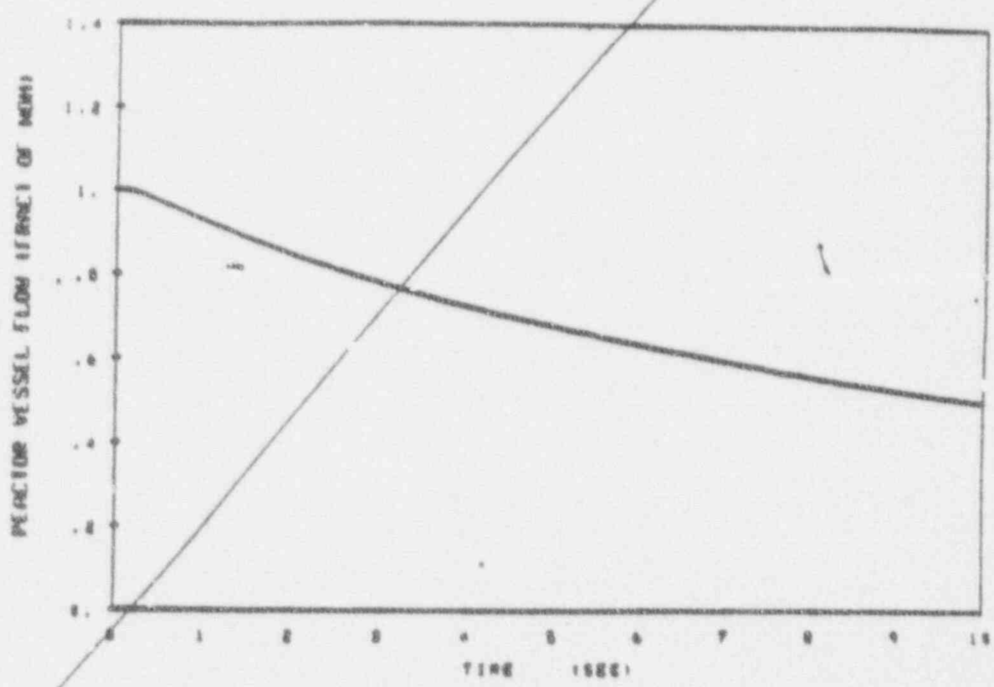
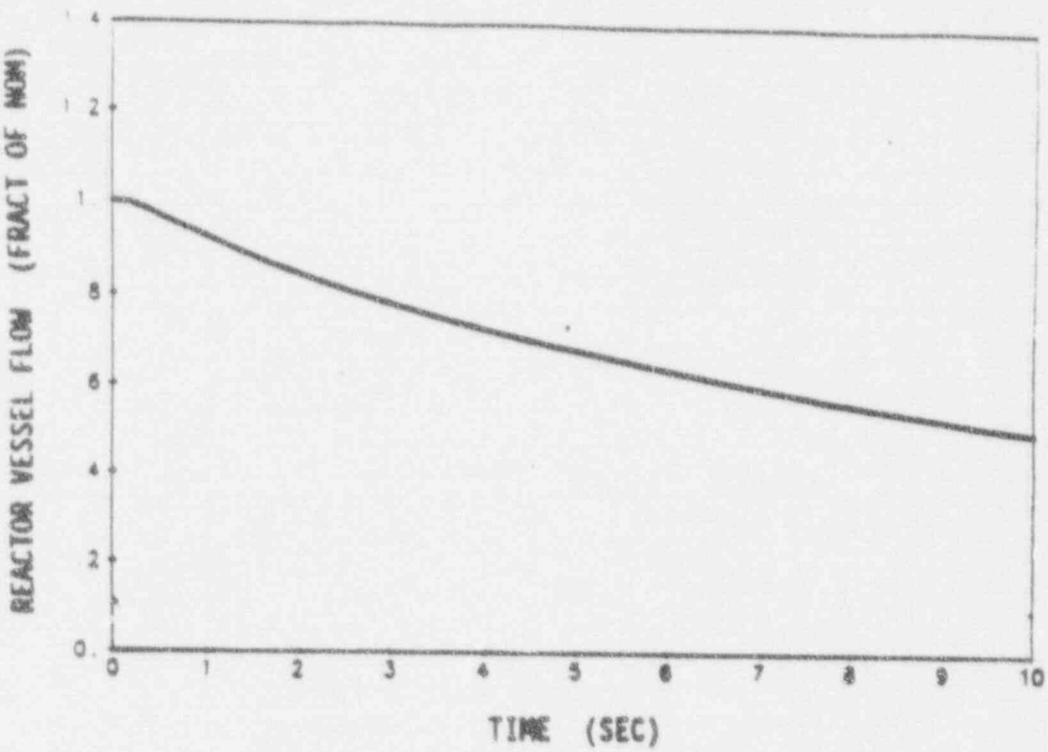


FIGURE 15.3.4-1

Complete Loss of Forced Reactor Coolant Flow
Core and Loop Flow versus Time



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Complete Loss of Forced Reactor Coolant Flow
Reactor Vessel Flow vs. Time

Figure 15.3.4-1

REPLACE WITH THE FOLLOWING

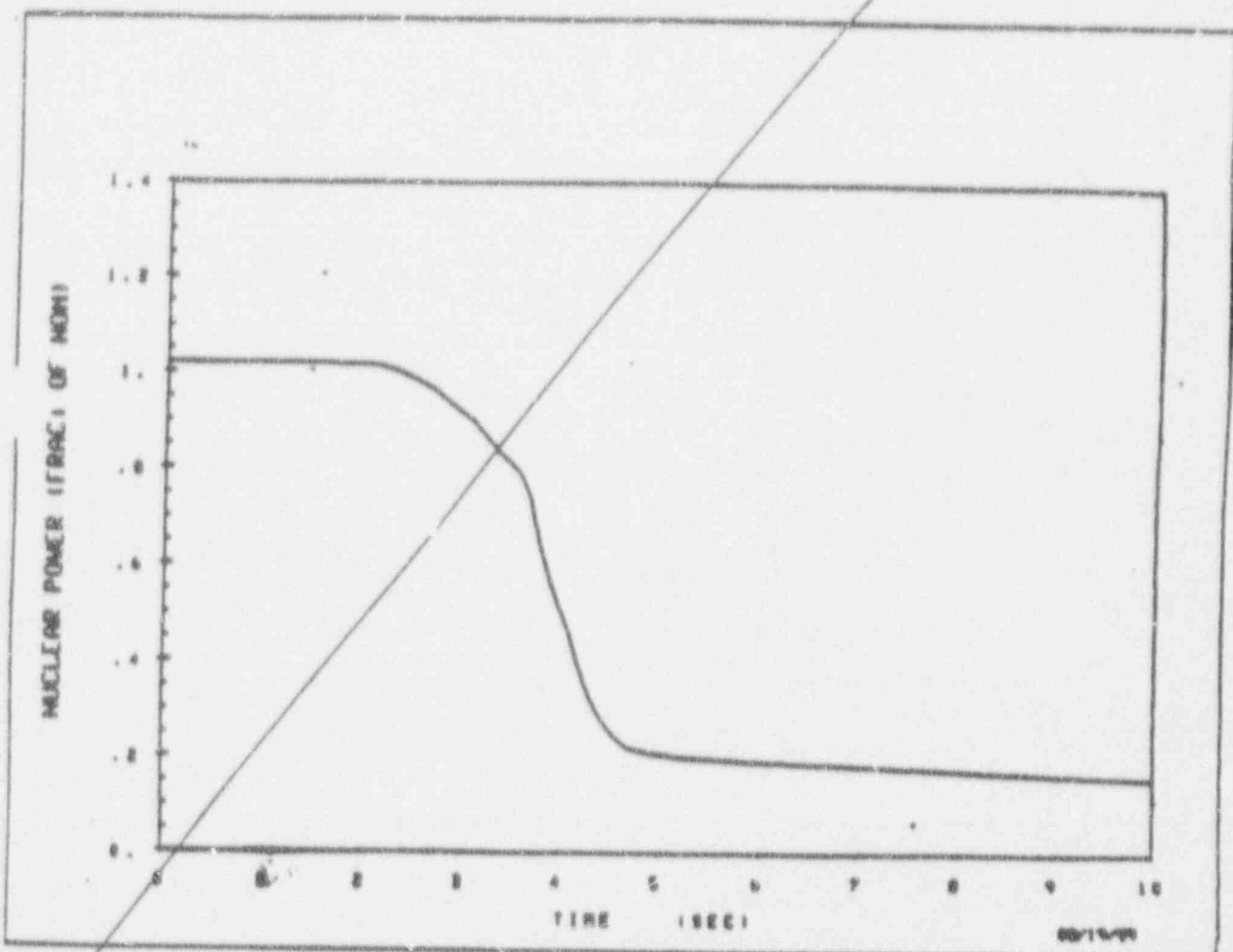
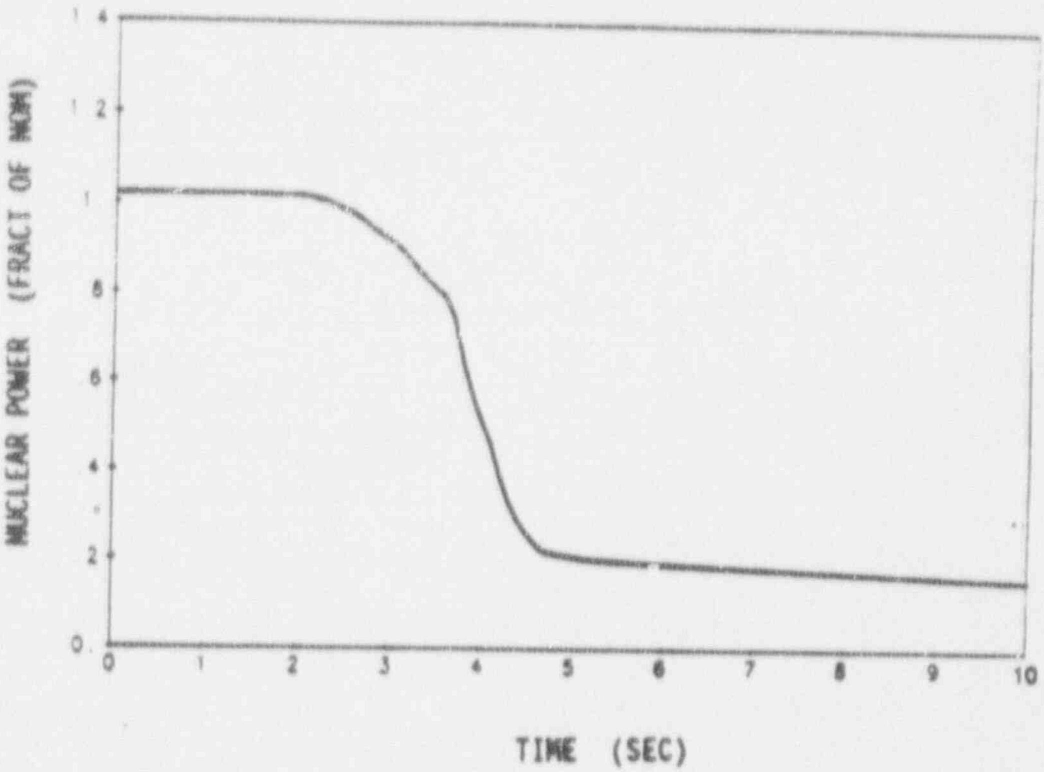


FIGURE 15.3.4-2a



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Complete Loss of Forced Reactor Coolant Flow
Nuclear Power vs. Time

Figure 15.3.4-2a

REPLACE WITH THE FOLLOWING

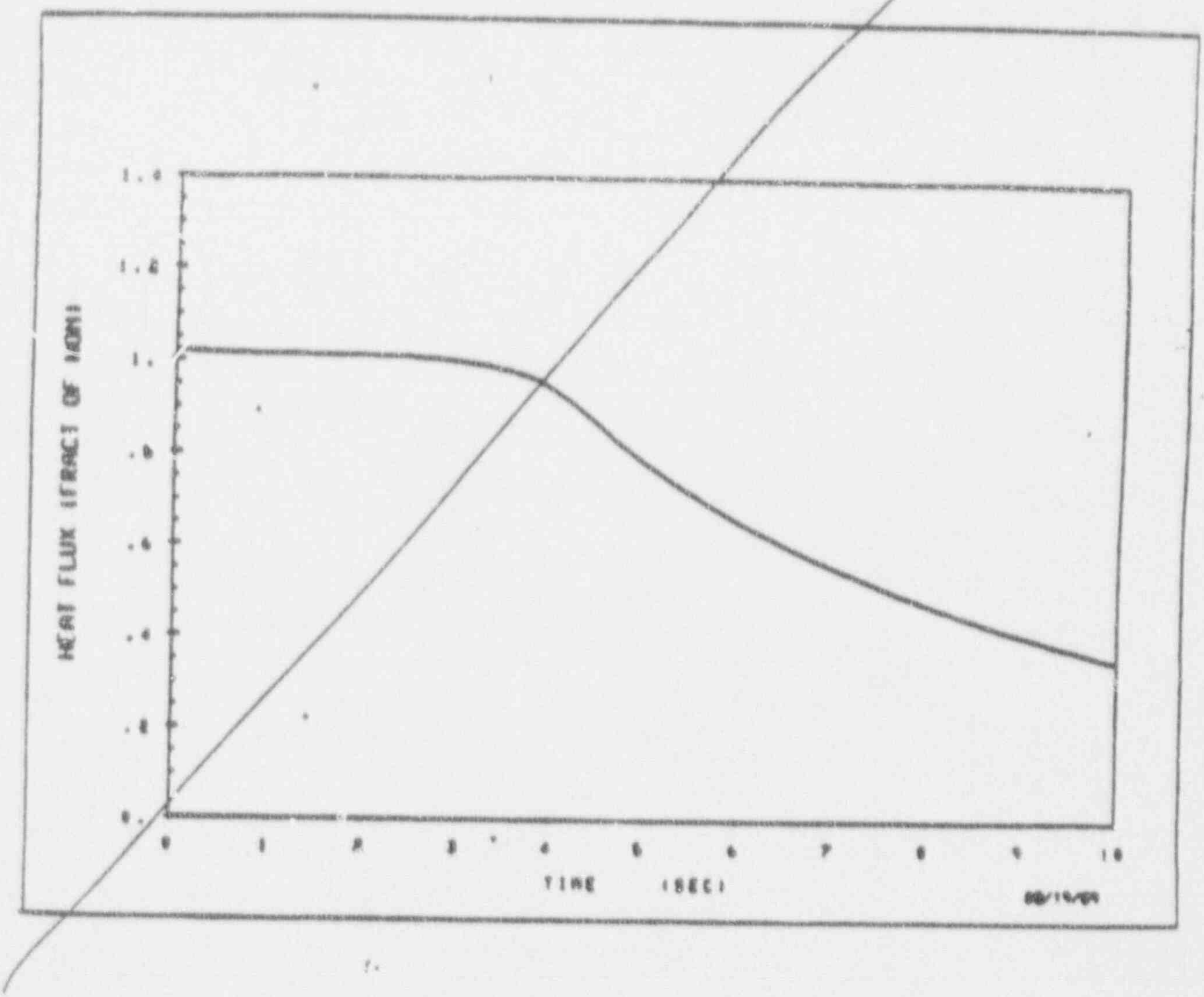
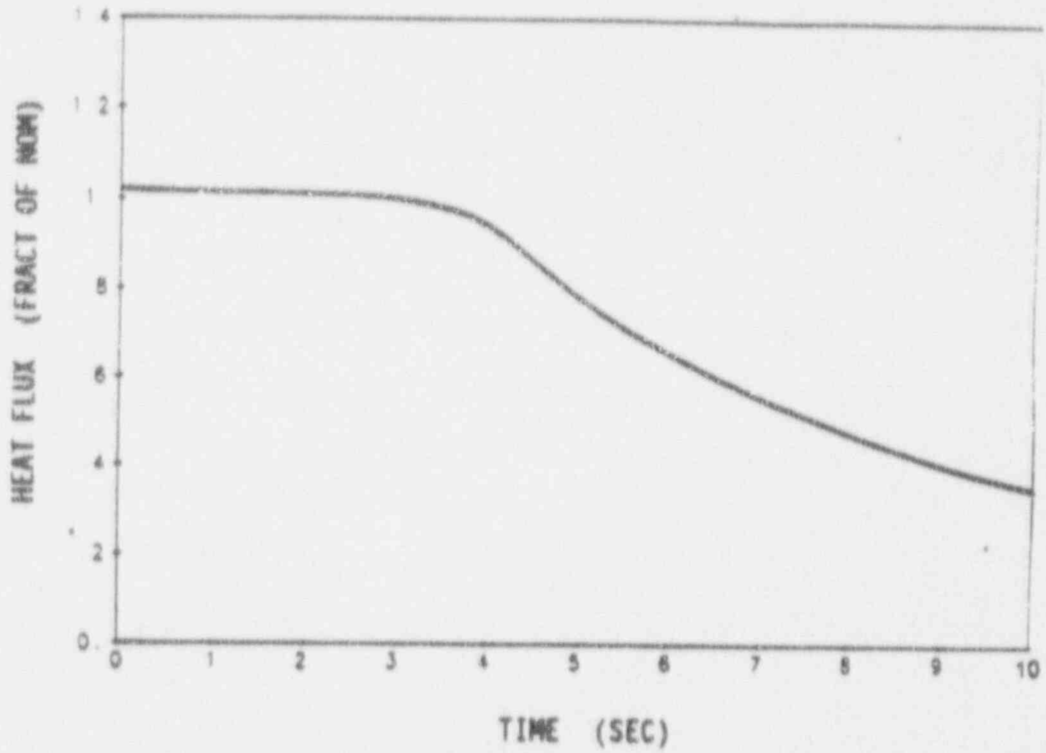


FIGURE 15.3.4-2b



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Complete Loss of Forced Reactor Coolant Flow
Heat Flux vs. Time (Average Channel)

Figure 15.3.4-2b

REPLACE WITH THE FOLLOWING

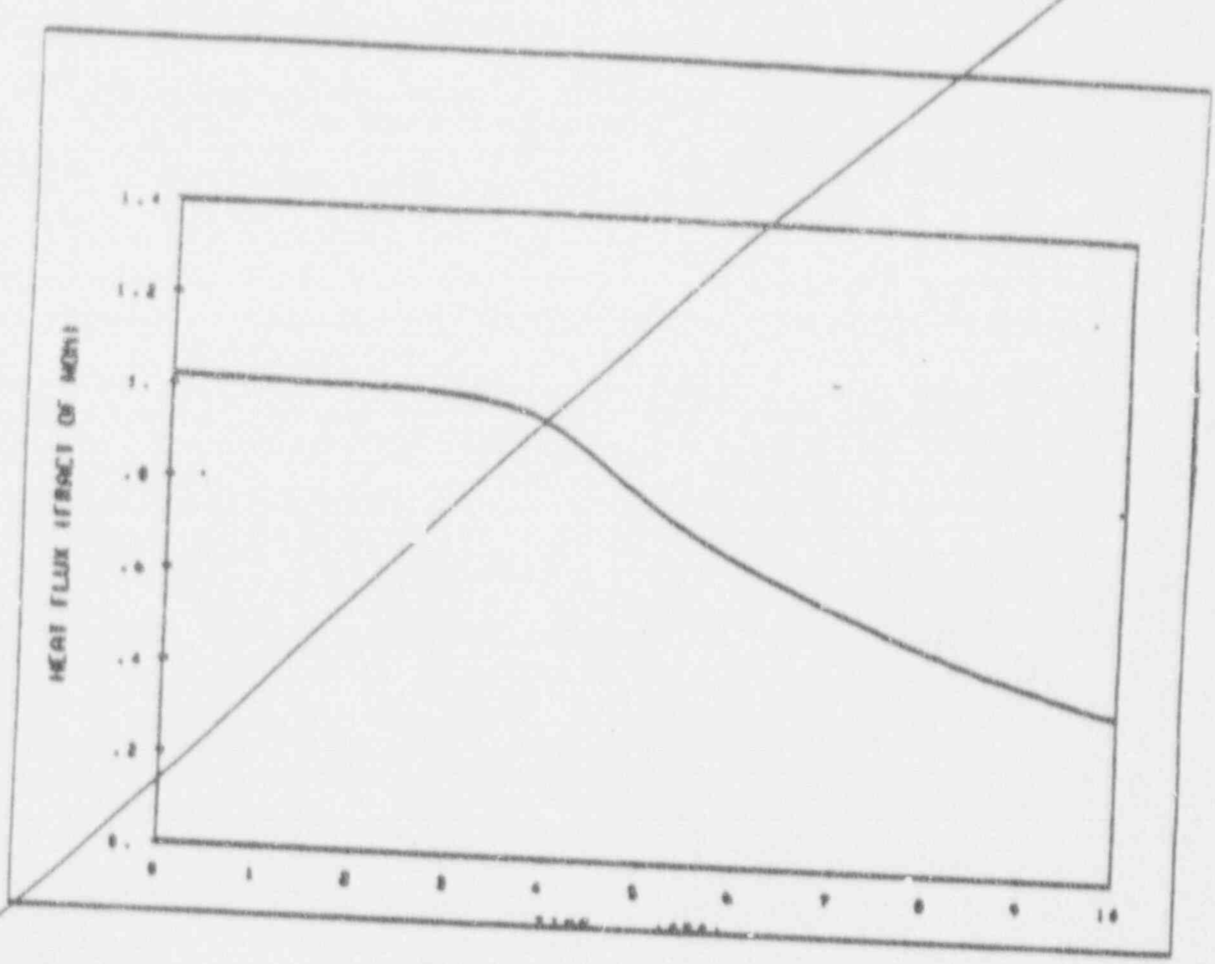
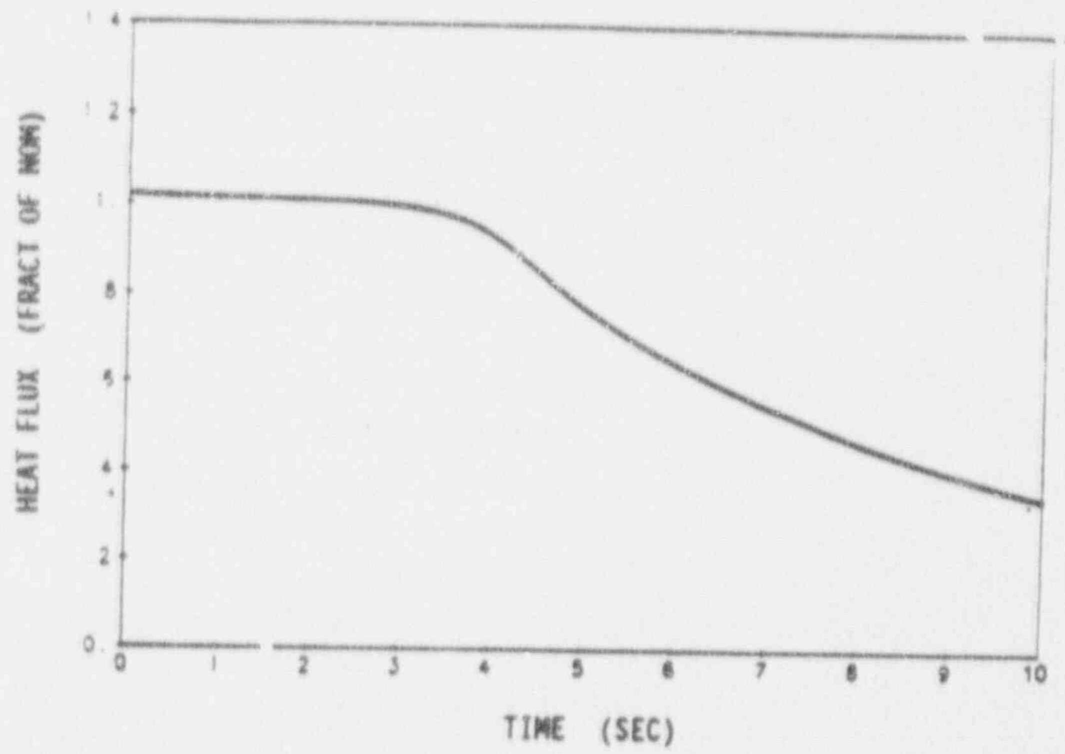


FIGURE 15.3.4-2c



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UNITS 1 and 2

Complete Loss of Forced Reactor Coolant Flow
Heat Flux vs. Time (Hot Channel)

Figure 15.3.4-2c

REPLACE WITH THE FOLLOWING

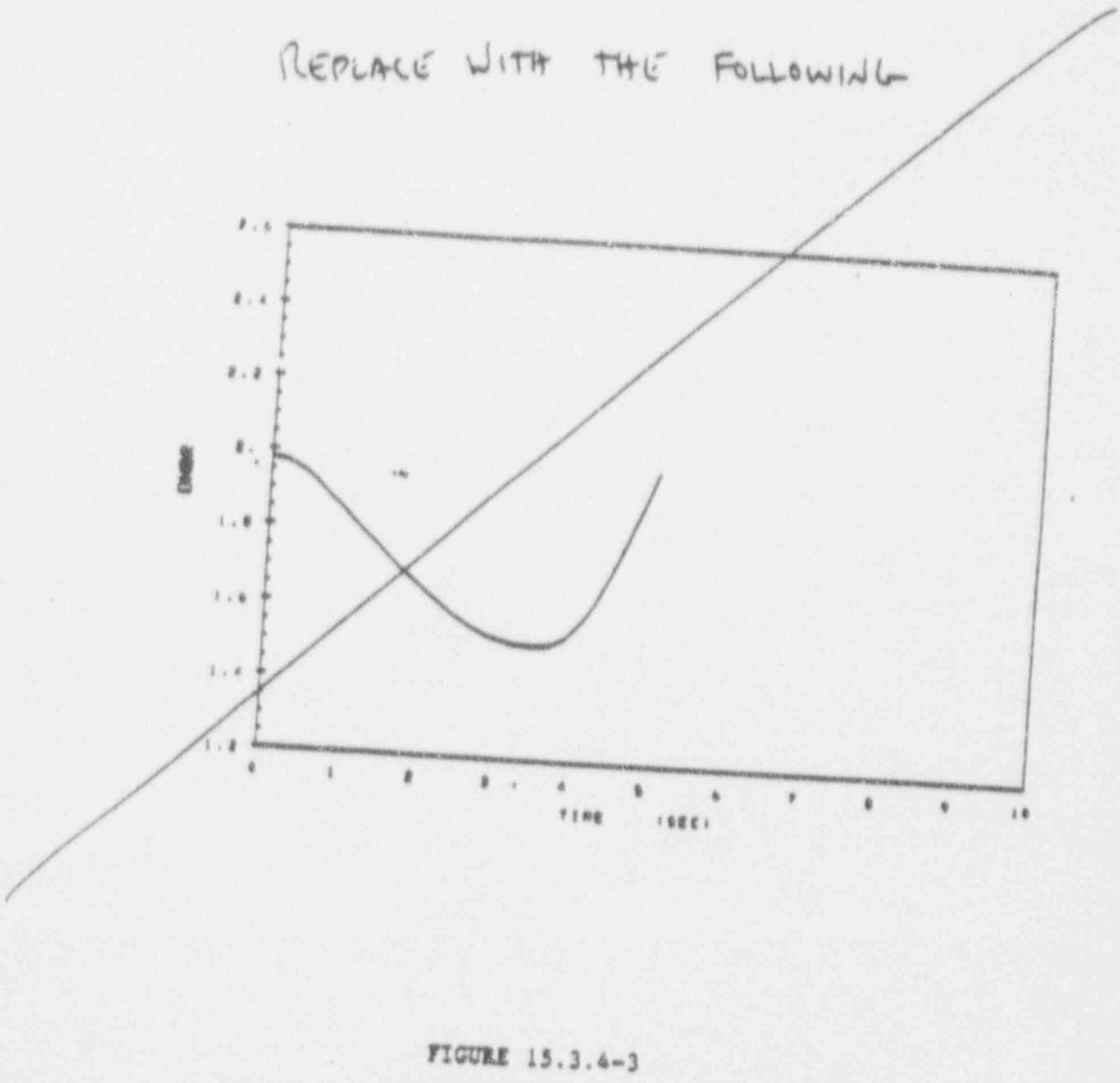
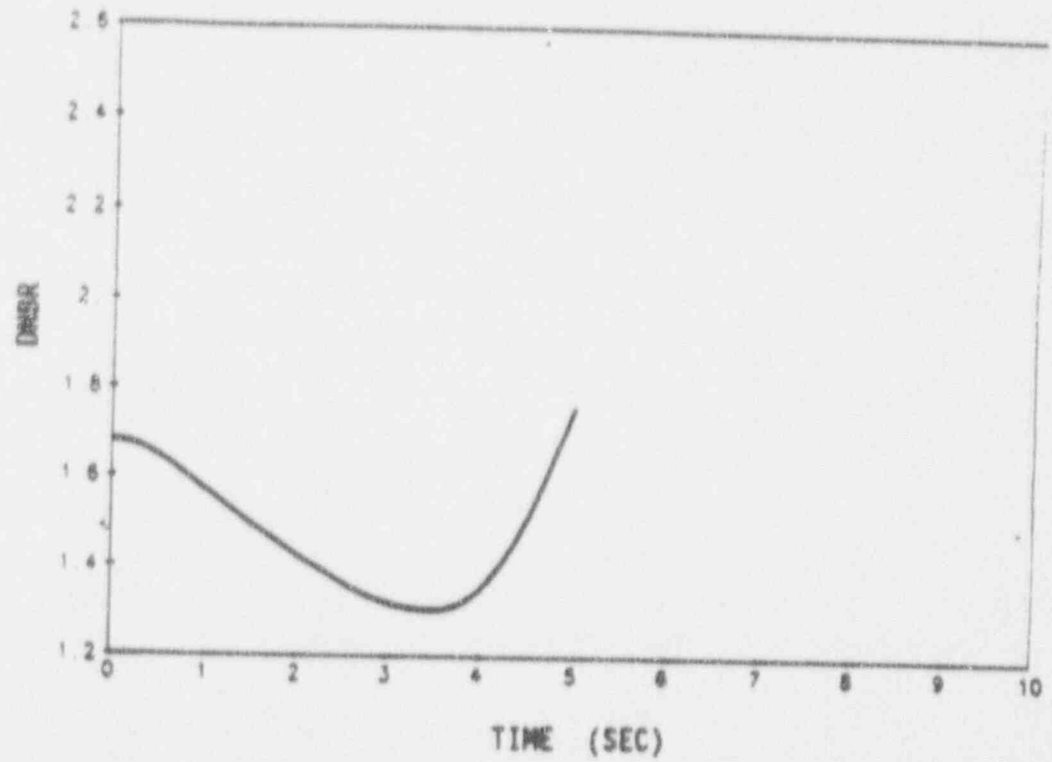


FIGURE 15.3.4-3

Complete Loss of Forced Reactor Coolant Flow
DNBR versus Time



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UNITS 1 and 2

Complete Loss of Forced Reactor Coolant Flow
DNBR vs. Time

Figure 15.3.4-3

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Figure 15.2.4-3 shows the information available to the operator on the core relative power based on the Nuclear Instrumentation System for the Eq Xe case. As shown there is essentially an instantaneous decrease in nuclear power from 100% to 7.5% (< 5 seconds). From 7.5% the standard 80 second period is used until the precursor isotopes have been depleted. From the point shown, an 18 day half life is assumed. For the case without Eq Xe, the NIS stable reading on source range is achieved very rapidly, < 5 minutes as opposed to 21 minutes for the Eq Xe case.

Sequence of events Tables are attached for both cases (see Tables 15.2.4-1, 15.2.4-2). These show that for both cases > 15 minutes of operator action time is available. Therefore the acceptance criteria for this event is met. In addition to the High Flux at Shutdown Alarm, there is also the High Pressurizer Level Trip and alarm available. In order to return critical a very large total dilution volume is required. The only means of accommodating this large volume is to allow the pressurizer to start filling. As shown, however, this results in a High Pressurizer Level Alarm very early in the transient.

These two alarms would provide the operator an adequate set of indications that a boron dilution event was in progress and also allow adequate time for operator corrective action.

Retained

15.2.5 Partial Loss of Forced Reactor Coolant Flow

15.2.5.1 Identification of Causes and Accident Description

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

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip which is actuated by two out of three low flow signals in any reactor coolant loop. Above approximately 35% power (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. A reactor trip signal from the pump undervoltage relay is provided as an anticipatory signal which serves as a backup to the low flow signal. It functions essentially identically to the low flow trip so that above Permissive 7 an undervoltage relay trip signal from any two pumps will actuate a reactor trip.

Normal power for the pumps is supplied through buses connected to the generator and each pump is supplied from a different bus. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

15.2.5.2 Analysis of Effects and Consequences

Method of Analysis

Partial loss of flow involving loss of two pumps with four loops in operation has been analyzed.

This transient is analyzed by three digital computer codes. First, the LOFTRAN Code (Reference 4) 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 Code (Reference 3) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC Code is used to calculate the departure from nucleate boiling ratio (DNBR) during the transient, based on the heat flux from FACTRAN and the flow from LOFTRAN. The WRB-1 correlation is used for DNBR calculation. The DNBR transients presented represent the minimum of the typical or thimble cell.

Typical Initial Conditions

Initial operating conditions assumed are the most adverse with respect to the margin to DNB, i.e., maximum steady state power level, minimum steady state pressure, and maximum steady state coolant average temperature. See Subsection 15.1.2 for explanation of initial conditions. In addition to the initial average temperature condition in Subsection 15.1.2, 1.5°F was added to the initial average temperature for conservatism. ~~With all but one loop operating, the maximum power level (including errors) allowed for that mode of operation is assumed.~~

(Delete, N-1 NOT CONSIDERED)

Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (See Table 15.1.2-2). The total integrated Doppler reactivity from 0 to 100% power is assumed to be 0.016 Δk. The lowest absolute magnitude of the moderator temperature coefficient (0.0 ~~Δk/°F~~ ΔK/°F) is assumed since this results in the maximum hot-spot heat flux during the initial part of the transient when the minimum DNBR is reached.

Flow Coastdown

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

Results

Insert B

The calculated sequence of events is shown on Table 15.2-1 for the case~~s~~ analyzed. ~~Figures 15.2.5-1 through 15.2.5-3 show the loop coastdowns, the core flow coastdowns, the nuclear power coastdowns and the average and hot channel heat flux coastdowns for each of the cases.~~ The minimum DNBR ~~for each of the cases~~ is not less than the safety analysis limit. 8 8

15.2.5.3 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit at any time during the transient. Thus there will be no cladding damage and no release of fission products to the Reactor Coolant System. 8 4

Not Reviewed

~~15.2.6 Startup Of An Inactive Reactor Coolant Loop~~

~~15.2.6.1 Identification of Causes and Accident Description~~

~~If the plant is operating with one pump out of service, there is reverse flow through the inactive loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.~~

~~Administrative procedures require that the unit be brought to a load of less than 25% of full power prior to starting a pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without~~

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6. Turbine Load - Turbine load was assumed constant until the electro- hydraulic governor drives the throttle valve wide open. Then turbine load drops as steam pressure drops.

7. Reactor Trip - Reactor Trip was initiated by low pressurizer pressure assumed at a conservatively low value of 1775 psia.

Results

The transient response is shown in Figures 15.2.14-1 and 15.2.14- 2. Nuclear power starts decreasing immediately due to boron injection but steam flow does not decrease until 15 seconds into the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The low pressure trip set point is reached at 64 seconds and rods start moving into the core at 66 seconds.

After trip, pressures and temperatures slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat.

15.2.14.3 Conclusions

Not Reviewed

Results of the analysis show that spurious safety injection with or without immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System.

DNB ratio is never less than the initial value. Thus there will be no cladding damage and no release of fission products to the reactor coolant system.

If the reactor does not trip immediately, the low pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident.

15.2.15 References

1. W. C. Gangloff, "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486, May 1971.

2. D. H. Risher, Jr., R. F. Barry, "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A (Proprietary), WCAP-8028-A (Non-Proprietary), January 1975. B

Replace with Insert A

~~3. C. Hunin, "FACTRAN, A Fortran Code for Thermal Transients in UO_2 Fuel Rod," WCAP-7908, June 1972.~~

4. Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non_Proprietary), April 1984. B

FSAR 15.2.5 - Partial Loss of Forced Reactor Coolant Flow

Inserts for reanalysis due to increased $F_{\Delta H}$

Insert A : Change Reference 3 to the following:

3. Hargrove, H. G., "FACTRAN - A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

Insert B:

Figures 15.2.5-1 through 15.2.5-3 show the resulting transient conditions for the 2/4 Partial Loss of Flow analysis. Included in these figures are total RCS flow, faulted loop flow, average and hot channel heat flux, nuclear power, and DNBR, each as a function of time.

TABLE 15.2-1 (Sheet 2)
(Continued)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

NOT Reviewed

Accident	Event	Time (Sec.)	
Uncontrolled Boron Dilution			3
1. Dilution during refueling and startup	Dilution begins	0	
	Operator isolates source of dilution; minimum margin to criticality occurs	refueling - precluded (by administrative controls) startup - >1140	8
2. Dilution During Full Power Operation			3
a. Automatic Reactor Control	Shutdown margin lost	2520	8
b. Manual Reactor Control	Dilution begins	0	3
	Reactor trip setpoint reached for over temperature ΔT	<120	8
	Shutdown margin is lost (if dilution continued after trip)	>2400	
Partial Loss of Forced Reactor Coolant Flow			3
All loops operating, two pumps coasting down			8
	Coastdown begins	0	3
	Low flow reactor trip	1.42	
	Rods begin to drop	2.42	8
	Minimum DNBR occurs	3.[illegible]	

REPLACE WITH THE FOLLOWING

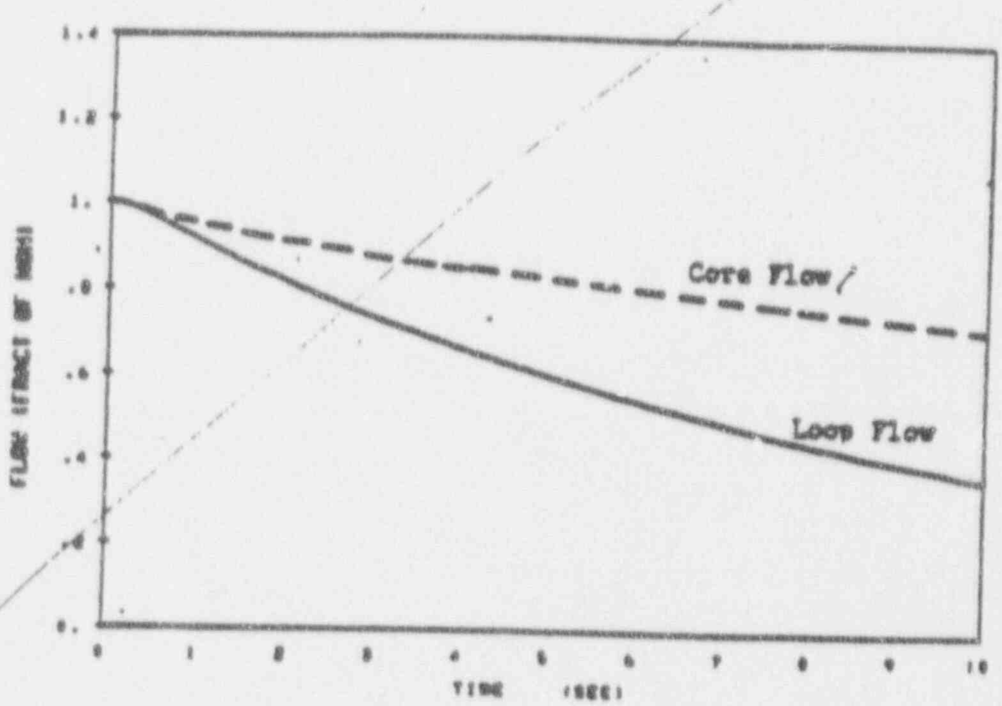
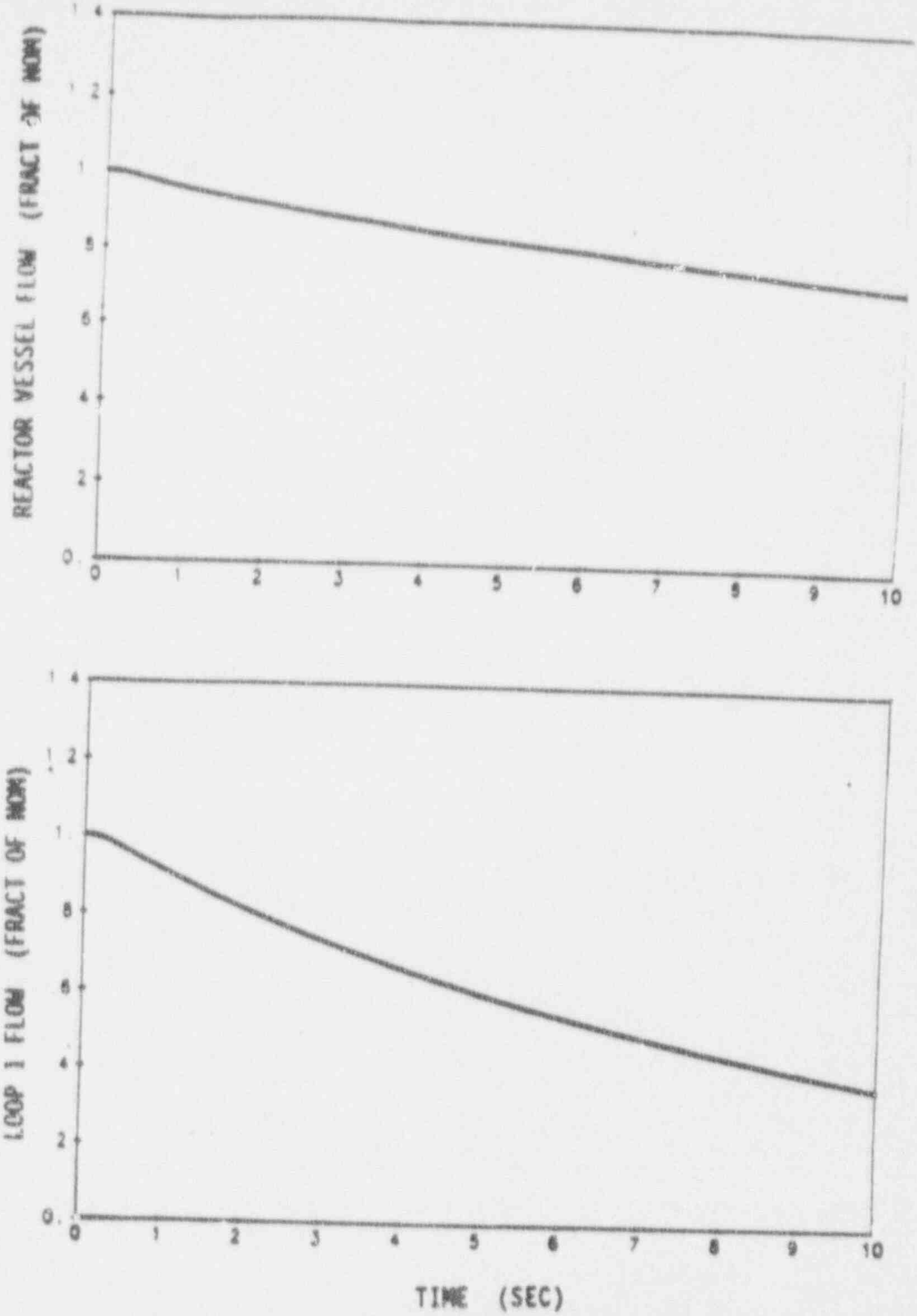


FIGURE 15.2.5-1

Partial Loss of Forced Reactor Coolant Flow
~~Core~~ and Loop Flow versus Time

Reactor Vessel



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Partial Loss of Forced Reactor Coolant Flow
Reactor Vessel and Loop Flow vs. Time

Figure 15.2.5-1

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REPLACE WITH THE FOLLOWING

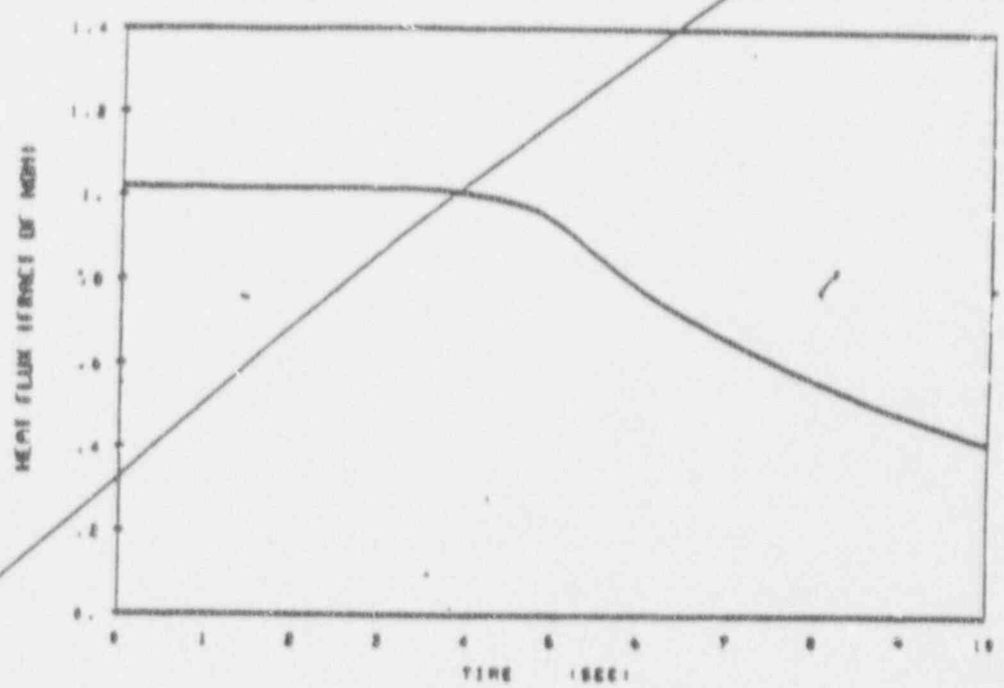
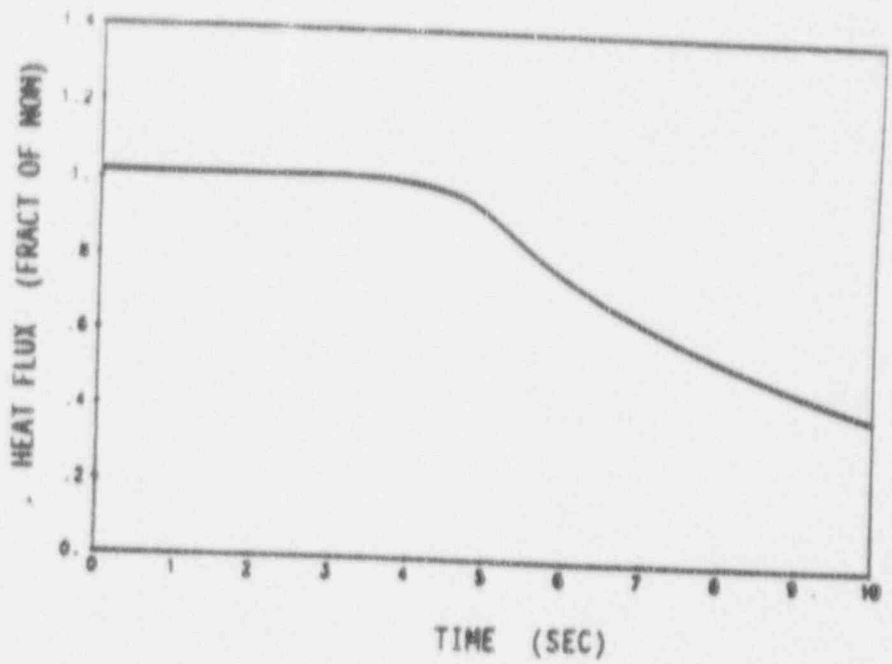


FIGURE 15.2.5-2a

Partial Loss of Forced Reactor Coolant Flow
Heat Flux versus Time



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Partial Loss of Forced Reactor Coolant Flow
Heat Flux vs. Time (Hot Channel)

Figure 15.2.5-2a

REPLACE WITH THE FOLLOWING

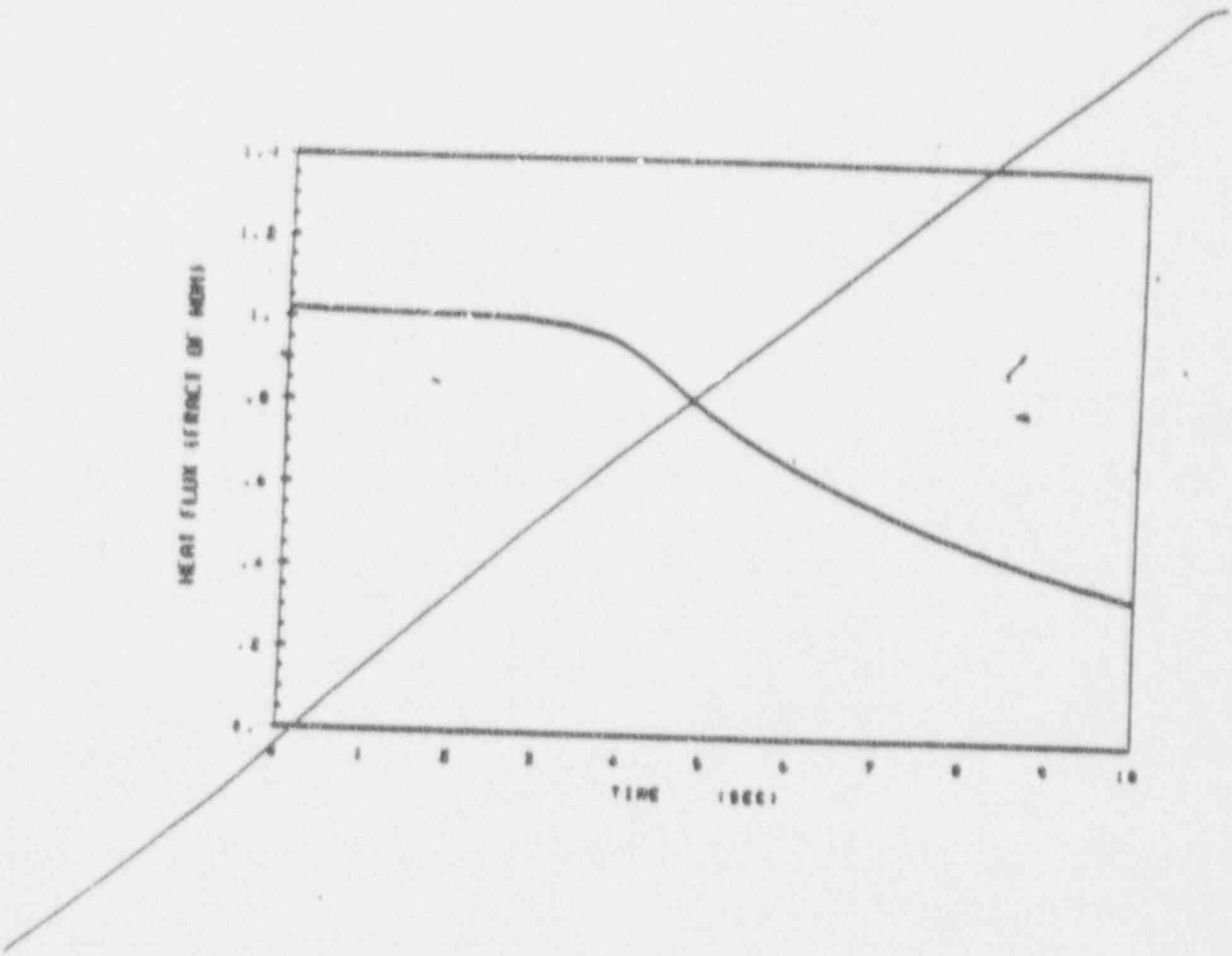
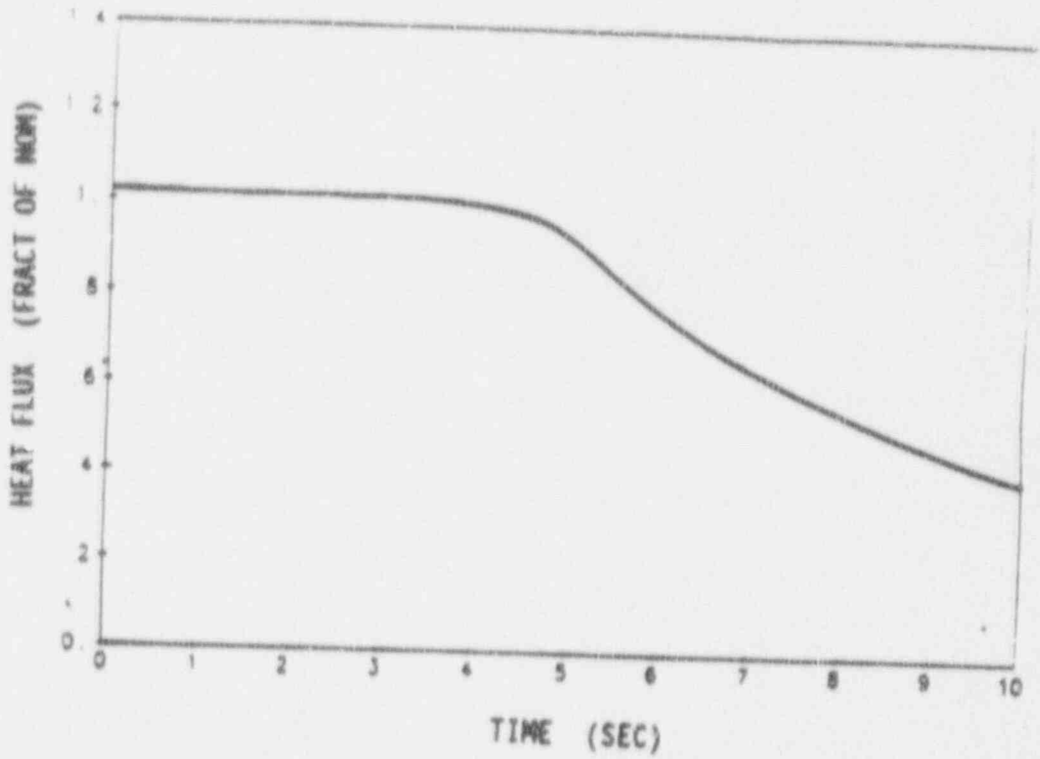


FIGURE 15.2.5-2b

Complete Loss of Forced Reactor Coolant Flow
Heat Flux versus Time



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Partial Loss of Forced Reactor Coolant Flow
Heat Flux vs. Time (Average Channel)

Figure 15.2.5-2b

REPLACE WITH THE FOLLOWING

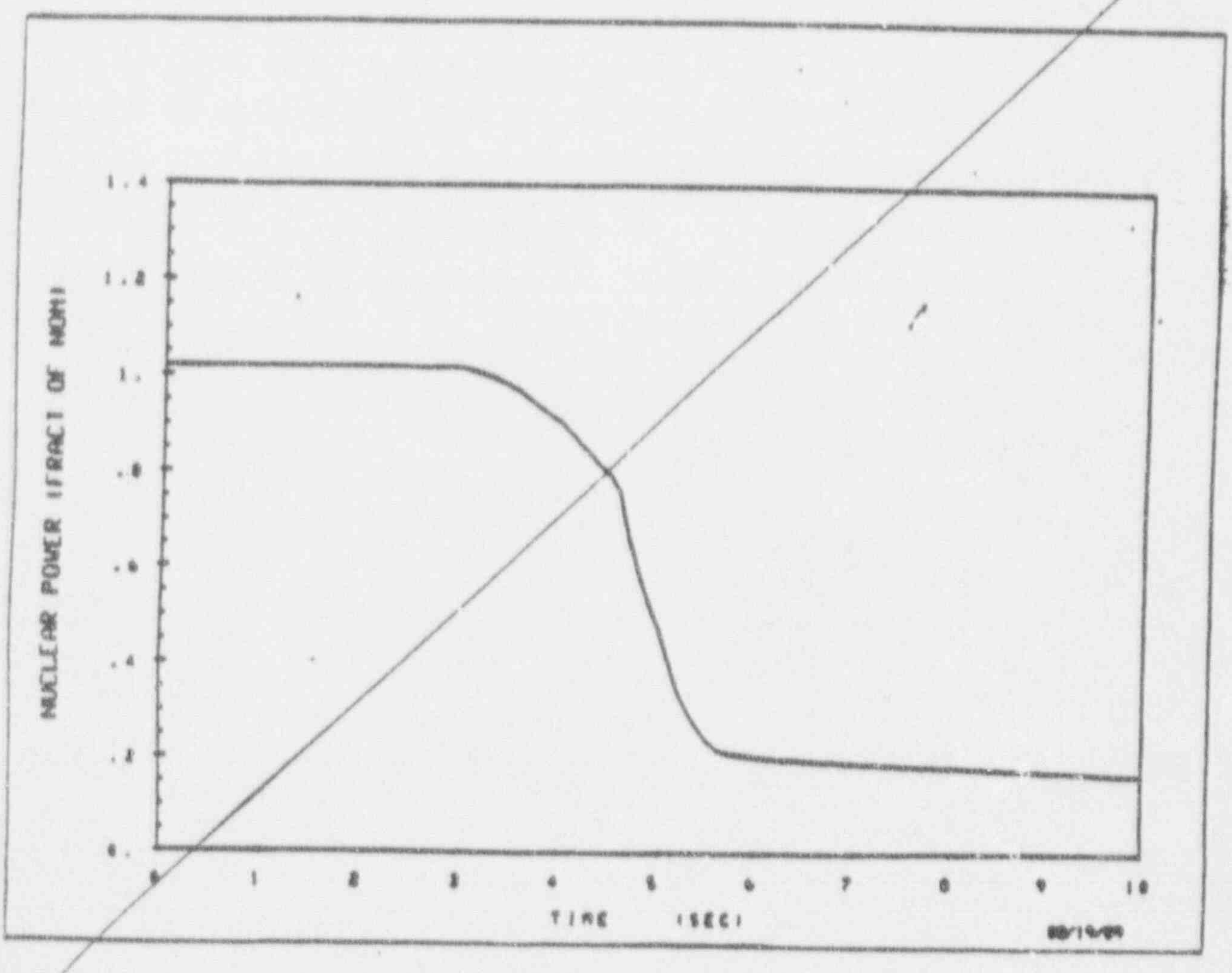
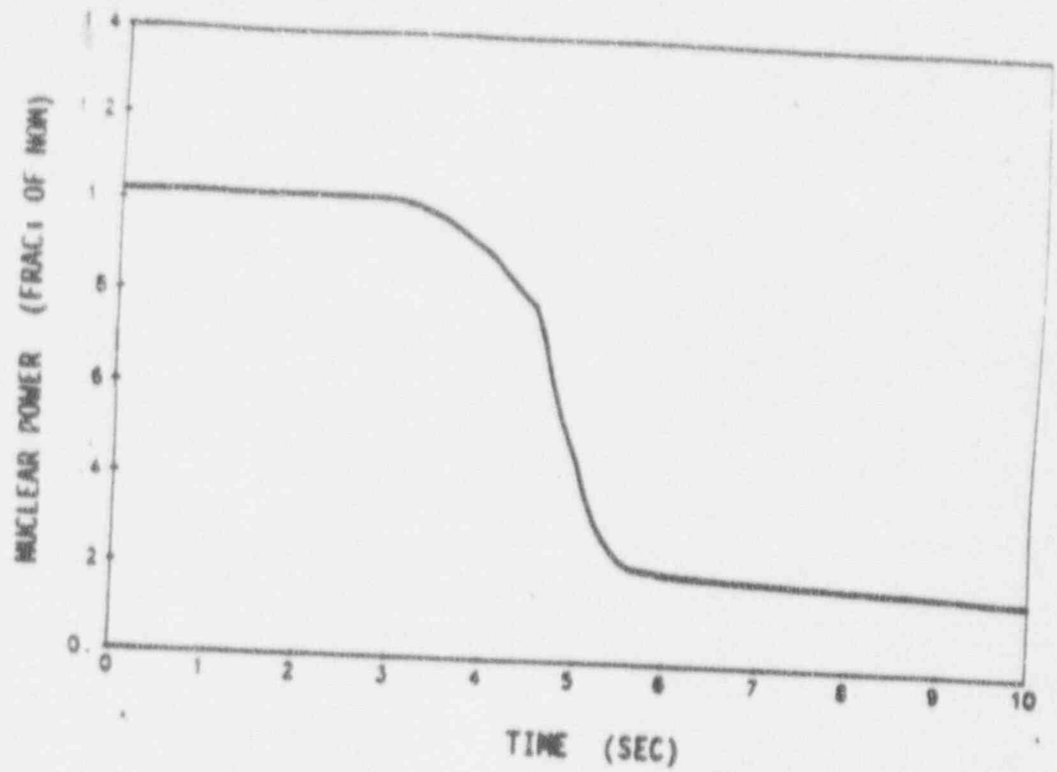


FIGURE 15.2.5-2c

Partial Loss of Forced
Reactor Coolant Flow
Nuclear Power versus Time



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Partial Loss of Forced Reactor Coolant Flow
Nuclear Power vs. Time

Figure 15.2.5-2c

Replace with the following

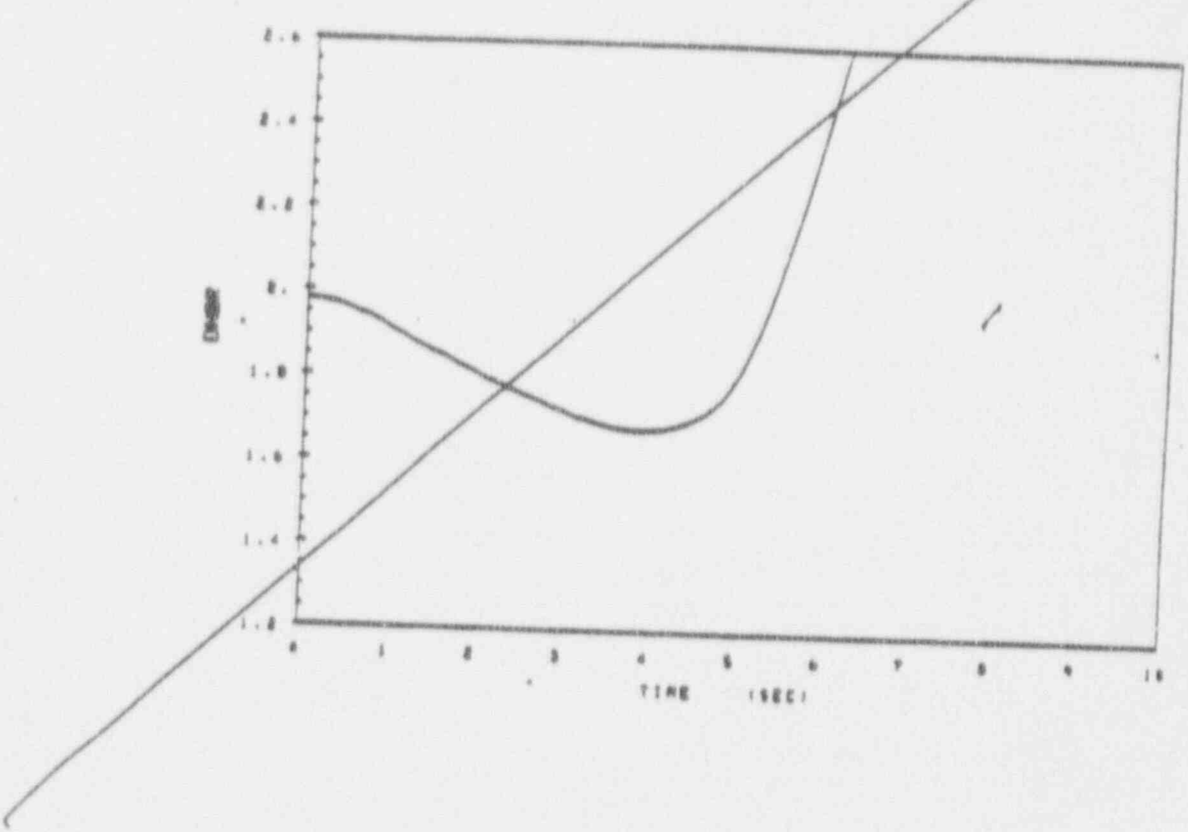
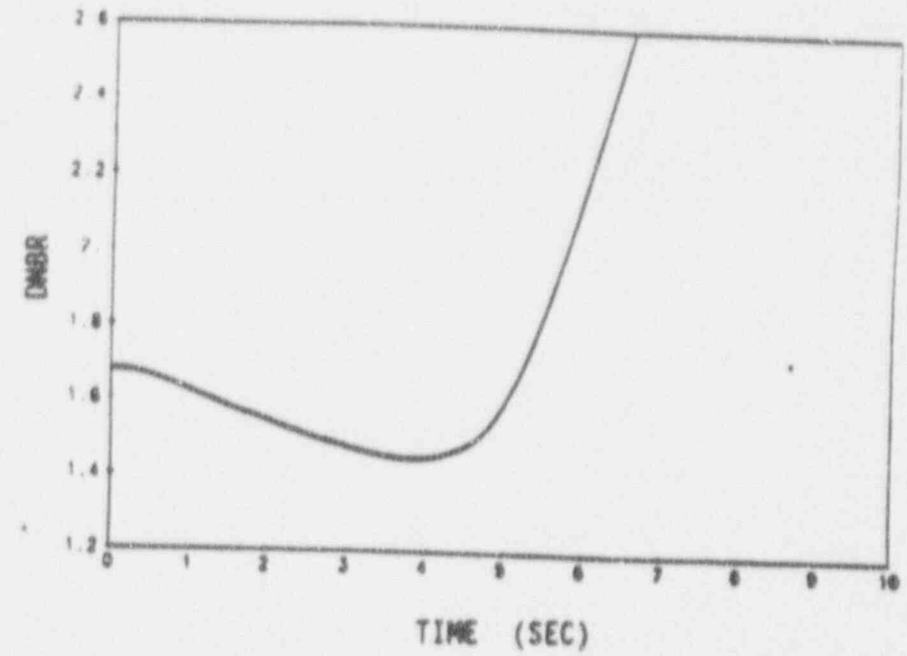


FIGURE 15.2.5-3

Partial Loss of Forced Reactor Coolant Flow
DNBR versus Time



SEQUOYAH FINAL SAFETY ANALYSIS REPORT UNITS 1 and 2
Partial Loss of Forced Reactor Coolant Flow DNBR vs. Time
Figure 15.2.5-3

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Locked Rotor Results

Transient values of pressurizer pressure, reactor vessel flow coastdown, nuclear power, and hot channel heat flux are shown in Figures 15.4.4-1 through 15.4.4-3.

Maximum Reactor Coolant System pressure, maximum clad temperature and amount of zirconium-water reaction are contained in Table 15.4.4-1. Figure 15.4.4-4 shows the clad temperature transient for the worst case.

15.4.4.3 Conclusions

1. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

2. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than the regulatory limit and the amount of Zirconium-water reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability. 6

NOT Reviewed

15.4.5 Fuel Handling Accident

15.4.5.1 Identification of Causes and Accident Description

The accident is defined as dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor.

15.4.5.2 Analysis of Effects and Consequences

For the analysis and consequences of the postulated fuel handling accident, refer to Subsection 15.5.6.

15.4.6 Rupture Of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

15.4.6.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster

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control assembly and drive shaft. 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.

Design Precautions and Protection

Certain features in the Sequoyah Nuclear Plant pressurized water reactor are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design which lessens the potential ejection worth of rod cluster control assemblies and minimizes the number of assemblies inserted at power.

Mechanical Design

The mechanical design is discussed in Section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of rod cluster control assembly drive mechanism housing failure sufficient to allow a rod cluster control assembly to be rapidly ejected from core are listed below:

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

2. The mechanism housings are individually hydrotested as they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed RCS.

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

4. The latch mechanism housing and rod travel housing are each a single length of forged Type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures which will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy type rod welds. Administrative regulations require periodic inspections of these (and other) welds.

Nuclear Design

Even if a rupture of a rod cluster control assembly drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected rod cluster control assembly is inherently limited. In general, the reactor is operated with the rod cluster control changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control rod banks are selected during the nuclear design to lessen the severity of a rod cluster control assembly ejection accident. Therefore, should a rod cluster control assembly be eject from its normal position during high power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the rod cluster control assemblies above this limit guarantees shutdown capability and acceptable power distribution. The position of all rod cluster control assemblies is continuously indicated in the control room. An alarm will occur if a bank of rod cluster control assemblies approaches its insertion limit or if one assembly deviates from its bank. There are low and low-low level insertion monitors with visual and audio signals. Operating instructions require boration at low level alarm and emergency boration at the low-low alarm.

Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Reference 26. The protection for this accident is provided by the power range high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in Section 7.2.

Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of a rod cluster control assembly mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking is not expected to cause damage to adjacent housings leading to increased severity of the initial accident.

Limiting Criteria

Due to the extremely low probability of a rod cluster control assembly ejection accident, limited fuel damage is considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy, have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Reference 27). Extensive tests of UO_2 zirconium clad fuel rods representative of those in Pressurized Water Reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT (Reference 28) results, which indicted that this threshold decreases by about 10% with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ration of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm. | 6

In view of the above experimental results, conservative criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

2. Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits. | 8

3. Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion (1) above.

15.4.8.2 Analysis of Effects and Consequences

Method of Analysis

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

A detailed discussion of the method of analysis can be found in Reference 29.

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Average Core Analysis

The spatial kinetics computer code, TWINKLE (Reference 30), is used for the average core transient analysis. This code uses cross sections generated by LEOPARD (Reference 31) to solve the two group neutron diffusion theory kinetic equations in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation pointwise Doppler and moderator feedback effects. 4

In this analysis, the code is used as a one dimensional axial kinetics code since it allows a more realistic representation of the special effects of axial moderator feedback and rod cluster control assembly movement and the elimination of axial feedback weighting factors. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Subsection 15.1.9.

Hot Spot Analysis

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN (Reference 25). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power generation is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong correlation (Reference 32) to determine the film boiling coefficient after DNB. The DNB heat flux is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state temperature distribution to agree with that predicted by design fuel heat transfer codes presently used by Westinghouse.

For full power cases, the design initial hot channel factor (F_qT) is input to the code. The hot channel factor during the transient is assumed to increase from the steady state design value to the maximum transient value in 0.1 seconds, and remain at the maximum for the duration of the transient. This is conservative, since detailed spatial kinetics models show that the hot channel factor decreases shortly after the nuclear power peak due to power flattening caused by preferential feedback in the hot channel (Reference 29). Further description of FACTRAN appears in Subsection 15.1.9.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer spray and pressure relief valves. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of calculated values for this type of core. The more important parameters are discussed below. Table 15.4.6-1 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

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

The total transient hot channel factors F_QT, is then obtained by combining the axial and radial factors.

Appropriate margins are added to the results to allow for calculational uncertainties including an allowance for nuclear power peaking due to fuel densification.

Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single channel analysis. Physics calculations were carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which when applied to single channel feedbacks correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not used. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors were shown to be conservative compared to three dimensional analysis (Reference 29).

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life were adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using the one dimensional steady state computer code with a Doppler weighting factor of 1.0. The resulting curve is conservative compared to design predictions for this plant. The Doppler weighting factor should be larger than 1.0 (approximately 1.3), just to make the present calculation agree with design predictions before ejection. This weighting factor will increase under accident conditions, as discussed above. The transient weighting factor used in the analysis is presented in Table 15.4.6-1.

Delayed Neutron Fraction, β

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values of 0.70% at beginning of life and 0.50% at end of life for the first cycle. The accident is sensitive to the ejected rod when its worth is nearly equal to or greater than β_{eff} as in zero power transients. In order to allow for future fuel cycles, pessimistic estimates were used in the analysis (0.55% at beginning of cycle and 0.45% β_{eff} at end of cycle). 6 8

Trip Reactivity Insertion

The trip reactivity insertion is assumed to be 4% from hot full power and 2% from hot zero power including the effect of one stuck rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point is reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip 8 breaker to open and 0.15 seconds for the coil to release the rods. The curve of rod insertion versus time which was used is shown in Figure 15.1.5-1. The time to full insertion assumed together with the 0.5 second delay overestimates the time for significant insertion of shutdown reactivity into the core. This is particularly important conservatism for hot full power accidents.

Results

The values of the parameters used in the analysis, as well as thu results of the analysis, are presented in Table 15.4.6-1 and discussed below.

Beginning of Cycle, Full Power

Control bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were 0.20% ΔK/K and 7.11 respectively. The peak hot spot fuel center temperature reached the beginning of life melt temperature of 4900°F. However, melting was restricted to less than 10% of the pellet. 8

Beginning of Cycle, Zero Power

For this condition, control bank D was assumed to be fully inserted and C was at its insertion limit. The worst ejected rod is located in control bank D and has a worth of 0.75% Δk/k and a hot channel factor of 14.05. 8

End of Cycle, Full-Power

Control bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were 0.21% Δk/k and 7.88 respectively. The peak hot spot fuel temperature exceeded the end of life melt temperature of 4800°F. However, melting was restricted to less than 10% of the pellet. The variation in melt temperature with burnup is discussed in Paragraph 4.4.1.2. 6 8 4

End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control bank D to be fully inserted and bank C at its insertion limit. The results were 0.97% Δk and 26.0 respectively. The peak fuel center temperature was 4391°F. These EOL zero power results are from an analysis for Mode 2 operation, which is more limiting than Mode 3 operation. 8

EOL

A summary of the cases presented above is given in Table 15.4.6-1. The nuclear power ~~and hot spot~~ fuel and clad temperature transients for the worst case in terms of fuel melt (BOC full power) are presented in figures 15.4.6-1 and 15.4.6-2. The same transients for the worst case in terms of clad temperature (EOL zero power) are presented in figures 15.4.6-3 and 15.4.6-4. 8

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Fission Product Release

It is assumed that fission products are released from the gaps of all rods having a DNBR of less than the safety analysis limit. In all cases considered, less than 10% of the rods entered DNB based on a detailed 3 dimensional THINC analysis. Although limited fuel melting at the hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident. | 3

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth 1 dollar at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits (Reference 29). Since the severity of the present analysis does not exceed this "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce disturtion. However, the temperature gradients across individual rods may produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect conservatively ignored in the analyses. | 4

15.4.8.3 Conclusions

Even on a pessimistic basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it isconcluded that there is no danger of further consequential damage to the primary system. The analyses have demonstrated that upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10%.

15.4.7 References

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Replace with Insert A

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Replace with Insert B

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FSAR 15.4.6 - Rupture of a Control Rod Drive Mechanism Houseing (Rod Cluster Control Assembly Ejection)

Inserts for reanalysis due to increased F_Q

Insert A : Change Reference 25 to the following:

25. Hargrove, H. G., "FACTRAN - A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

Insert B: Change Reference 29 to the following:

29. D. H. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1-A, January, 1975.

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TABLE 15.4.1-12 (Sheet 3)
(Continued)

TIME SEQUENCE OF EVENTS FOR
CONDITION IV EVENTS

Accident	Event	Time (Sec)
a. End of Cycle, Zero Power	RCCA ejected	0
	Reactor trip setpoint reached (High Neutron Flux, high setting)	0.16
	Rods begin to drop	0.66
	Peak clad average temperature reached	0.73
	Peak fuel center temperature reached	2.49

8

[ADD following INSERT C]

FSAR 15.4.6 - Rupture of a Control Rod Drive Mechanism Houseing (Rod Cluster Control Assembly Ejection)

Insert for reanalysis due to increased F_Q

Insert C : Add the following into Table 15.4.1-12 (Sheet 3)

b. End of Cycle, Full Power	RCCA ejected	0
	Reactor trip setpoint reached (High Neutron Flux, high setting)	0.05
	Rods begin to drop	0.55
	Peak clad average temperature reached	2.36
	Peak fuel center temperature reached	3.99
c. Beginning of Cycle, Full Power	RCCA ejected	0
	Reactor trip setpoint reached (High Neutron Flux, high setting)	0.05
	Roas begin to drop	0.55
	Peak clad average temperature reached	2.29
	Peak fuel center temperature reached	4.36

TABLE 15.4.6-1

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

Time in Life	Beginning	Beginning	End	End
Power Level	102 pct	0 pct	102 pct	0 pct
Ejected rod worth, %Δk/k	.20	.75	.21	0.97
Delayed neutron fraction, %	.55	.55	.44	0.45
Feedback reactivity weighting	1.3	2.4	1.6	3.63
Trip Reactivity, %Δ/k	4.0	2.0	4.0	2.0
F_q before rod ejection	~~2.50~~ 2.62	--	~~2.52~~ 2.62	--
F_q after rod ejection	7.11	14.05	7.00	26.0
Number of operational pumps	4	2	4	4
Max. fuel pellet average temperature, °F	~~4089~~ 4121	3156	~~4004~~ 4056	3744
Max. fuel center temperature, °F	4971	3610	~~4894~~ 4879	4391
Max. fuel stored energy, cal/gm	~~180~~ 181	132	~~174~~ 177	161

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Replace with the following

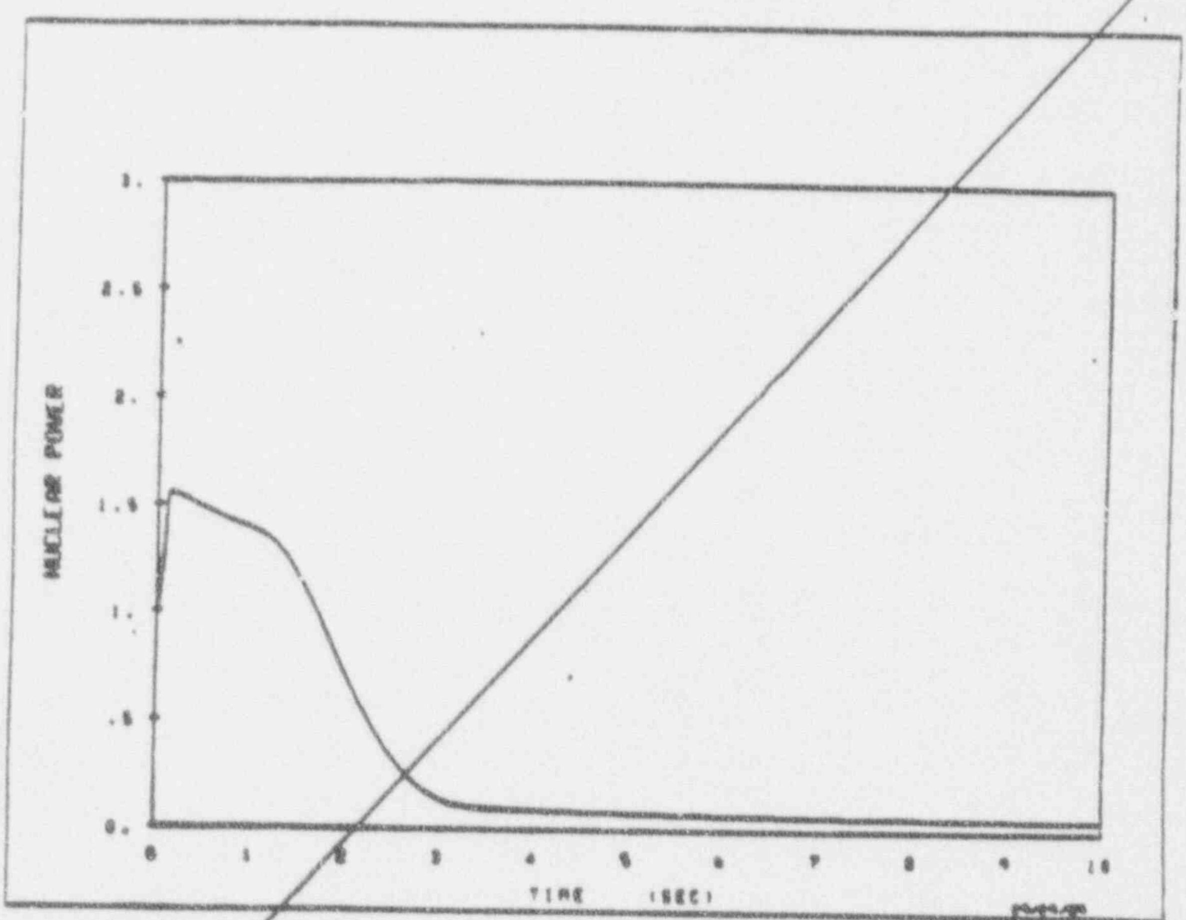
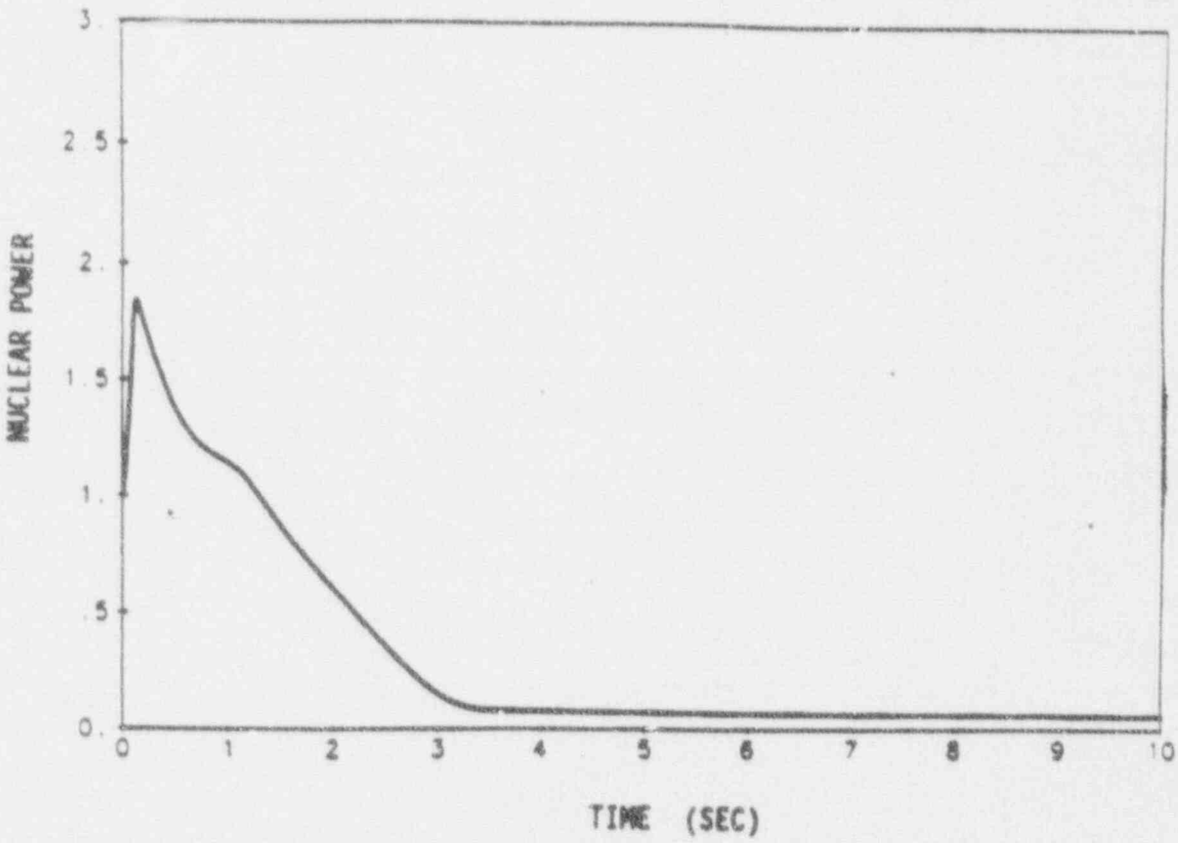


FIGURE 15.4.6-1
Nuclear Power versus Time for VSH, BOL, HFP



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Rod Cluster Control Assembly Ejection
Nuclear Power vs. Time (EOL, HFP)

Figure 15.4.6-1

Replace with the following

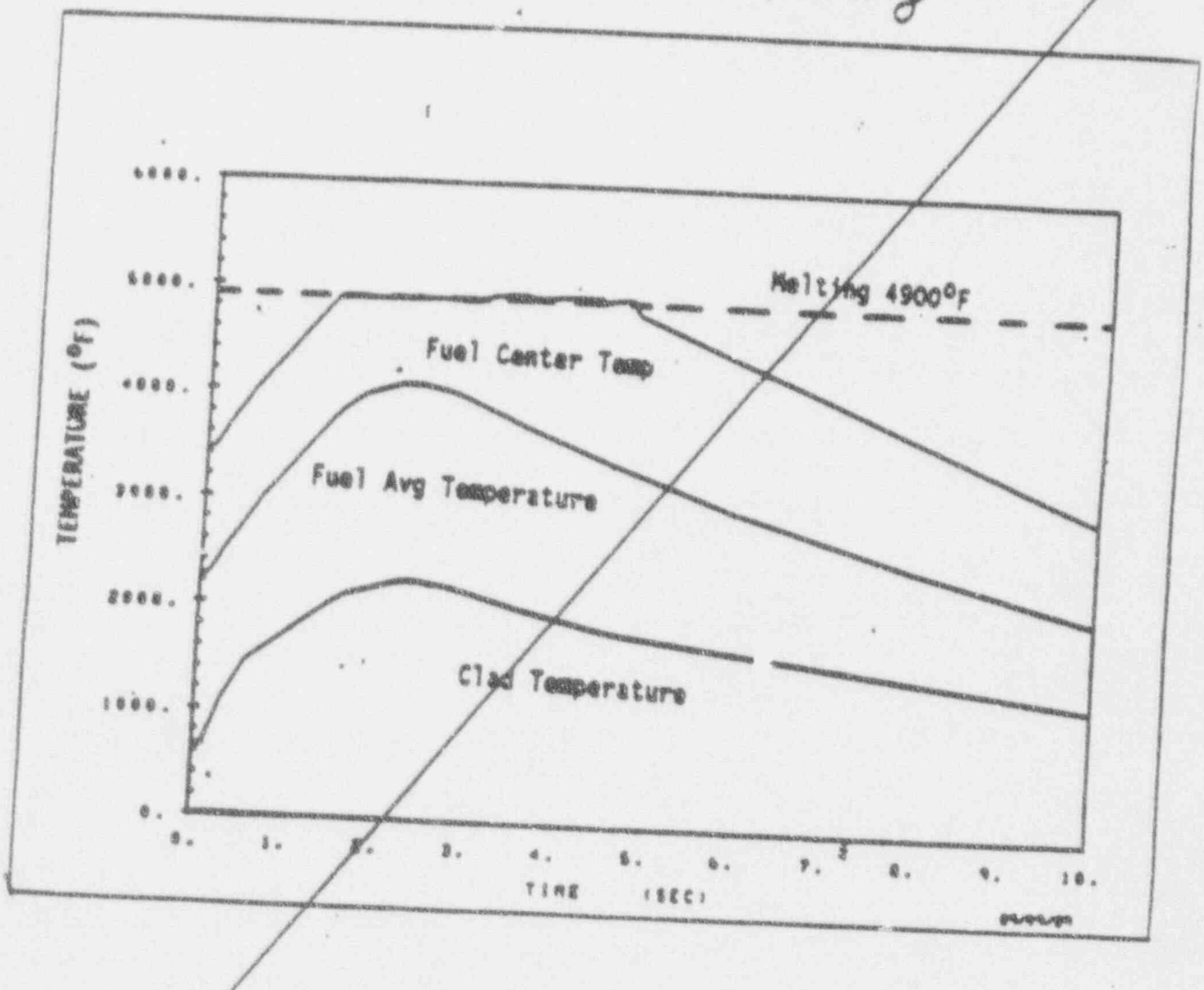
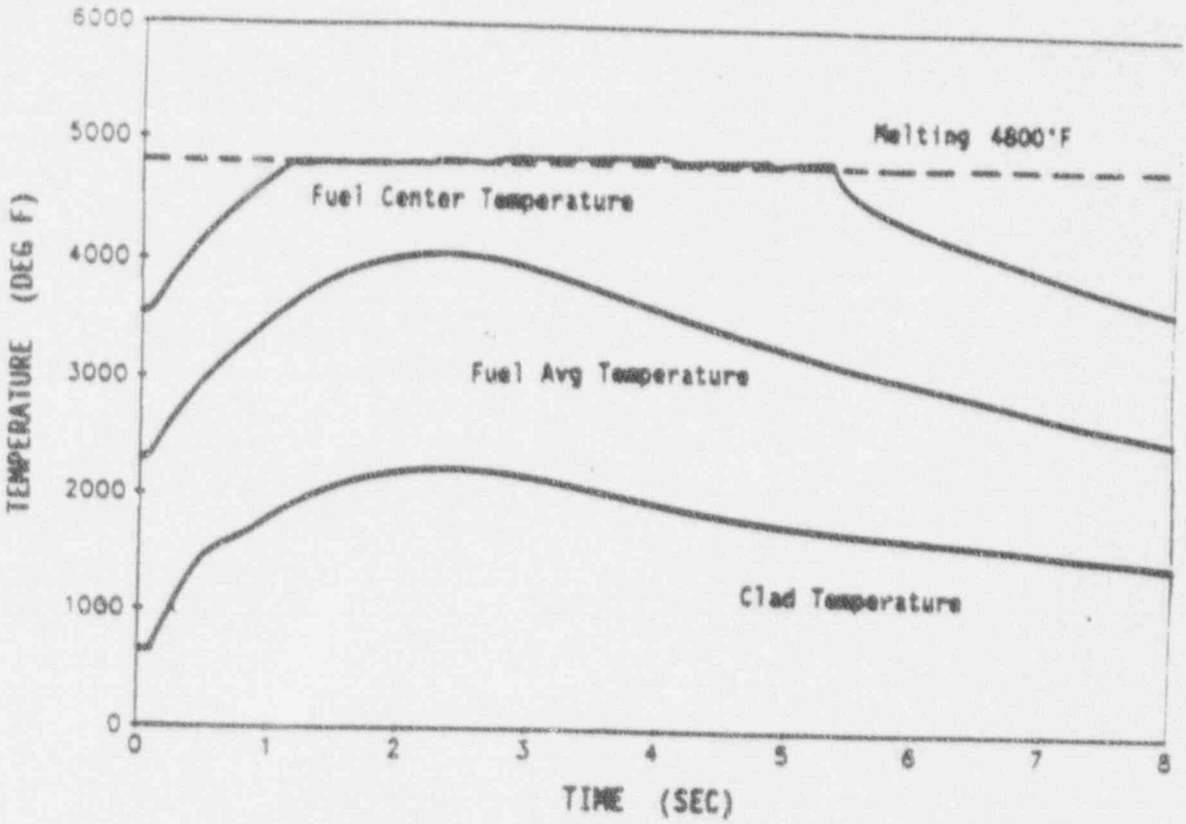


FIGURE 15.4.6-2
Fuel and Clad Temperature versus
Time for V5H, BOL, HZP



SEQUOYAH
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Rod Cluster Control Assembly Ejection
Fuel and Clad Temperature vs. Time (EOL, HFP)

Figure 15.4.6-2

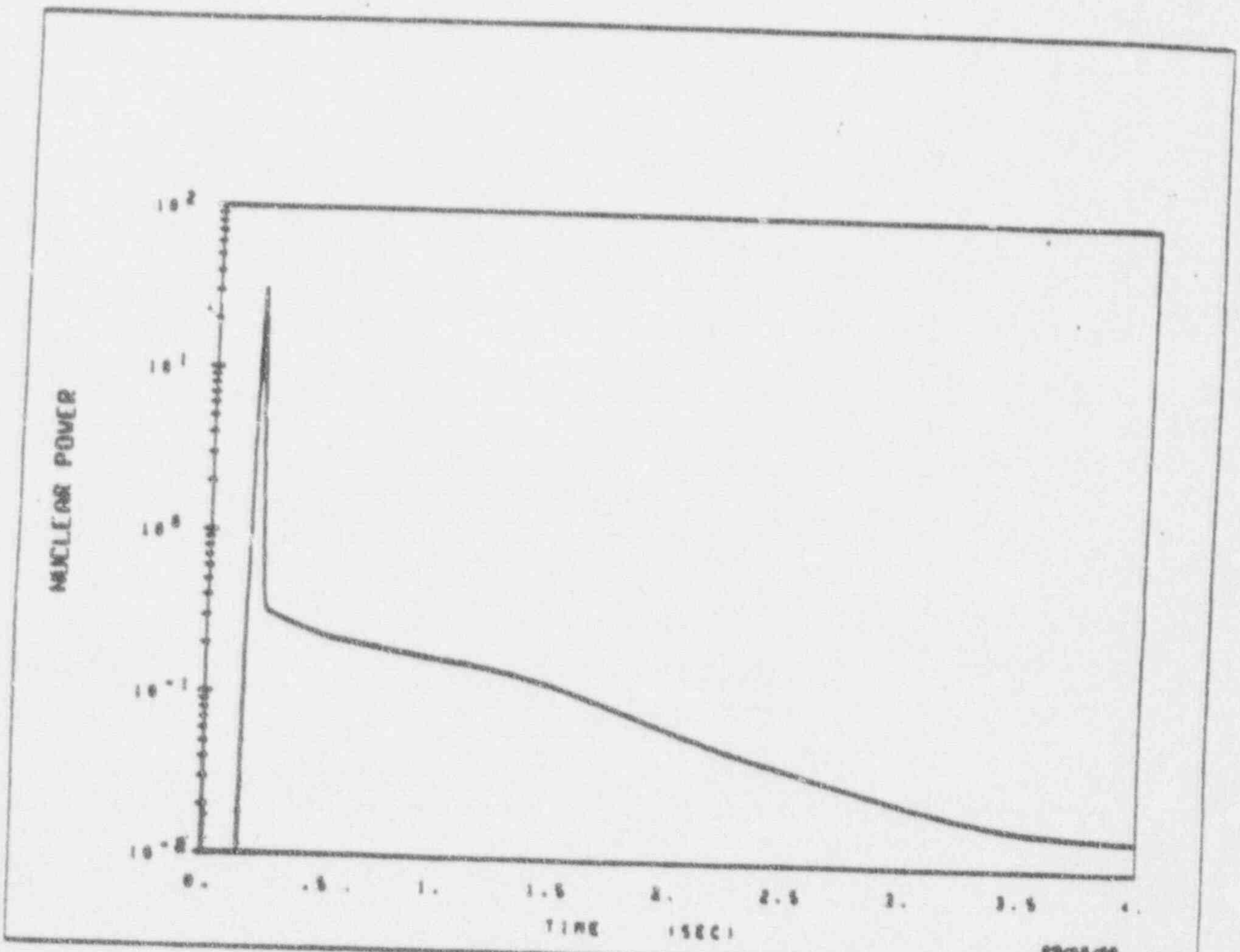


FIGURE 15.4.6-3
Nuclear Power versus Time
for V5H, EOL, HZP

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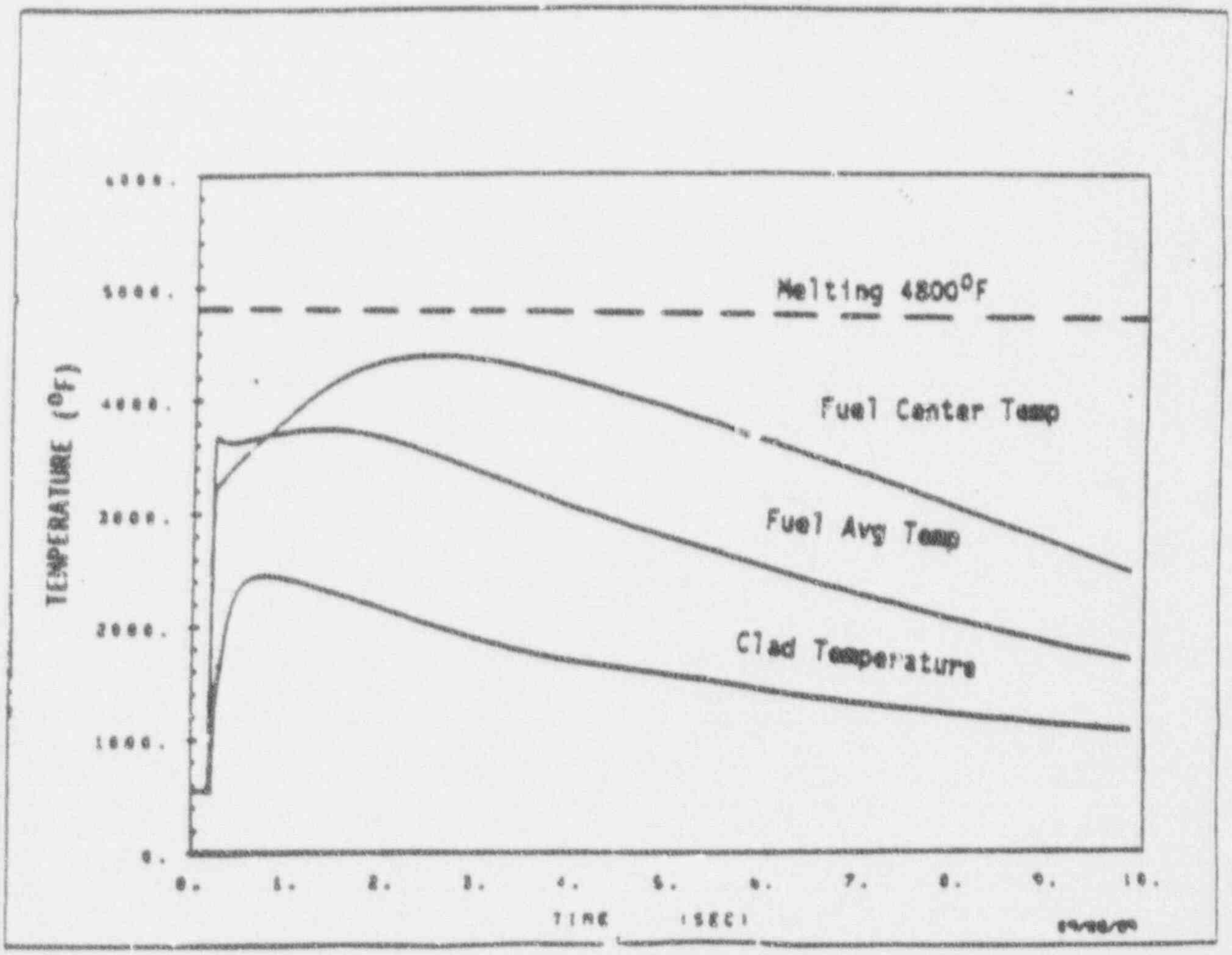

FIGURE 15.4.6-4
Fuel and Clad Temperature
versus Time for V5H, EC.., HZP

10.1.2 LOCA

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic which must be designed against and thus represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

1. Major rupture of pipes containing reactor coolant up to and including double ended rupture of the largest pipe in the Reactor Coolant System (loss of coolant accident).
2. Major secondary system pipe ruptures.
3. Steam generator tube rupture.
4. Single reactor coolant pump locked rotor.
5. Fuel handling accident.
6. Rupture of a control rod mechanism housing (rod cluster control assembly ejection).

The analysis of thyroid and whole body doses, resulting from events leading to fission product release, appears later in the Safety Analysis Report. The fission product inventories which form a basis for these calculations are presented in Chapter 11 and Section 15.1. The Safety Analysis Report also includes the discussion of systems interdependency contributing to limiting fission product leakages from the containment following a Condition IV occurrence.

15.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

The analysis specified by 10 CFR 50.46 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors", is presented in this section. The results of the loss of coolant accident analysis ~~is~~ are shown in Table 15.4.1-1 and ~~shows~~ compliance with the Acceptance Criteria. The description of the various aspects of the LOCA analysis is given in References 33, 48, and 49. 9

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The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the Acceptance Criteria (1).

15.4.1.1.2 Method of Thermal Analysis

Descriptions of the various aspects of the LOCA analysis are provided in References 2 and 49. These documents describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which serve to maintain compliance with the acceptance criteria of 10 CRF 50.46.

The analysis of a large break LOCA transient is divided into three phases: Blowdown, Refill, and Reflood. A series of computer codes has been developed to analyze the transient based on the specific phenomena which govern each phase. During the blowdown portion, the SATAN-VI code (Reference 35) is used to calculate the RCS pressure, enthalpy, density, and mass and energy flows in the primary system, as well as the heat transfer between the primary and secondary system. At the end of the blowdown, information on the state of the system is transferred to the WREFLOOD code (Reference 36) which performs the calculation of the refill period to bottom of core (BOC) recovery time. Once the vessel has refilled to the bottom of the core, the reflood portion of the transient begins. The BASH code (Reference 2) is used to calculate the thermal-hydraulic simulation of the RCS for the reflood phase.

Information concerning the core boundary conditions is taken from all of the above codes and input to the LOCBART code (Reference 2) for the purpose of calculating the core fuel rod thermal response for the entire transient. From the boundary conditions, LOCBART computes the fluid conditions and heat transfer coefficient for the full length of the fuel rod by employing mechanistic models appropriate to the actual flow and heat transfer regimes. Conservative assumptions ensure that the fuel rods modeled in the calculation represent the hottest rods in the entire core.

15.4.1.1.3 Containment Analysis

The containment pressure analysis is performed with the LOTIC-2 (37) code. The transient pressure computed by the LOTIC code can be entered in the BASH code for the purpose of computing the reflood transient. The containment pressure transient input to BASH from LOTIC is presented in Figure 15.4.1-4. The containment data used in the containment pressure analysis to determine the ECCS backpressure are presented in Tables 15.4.1-4 and 15.41-5. 9

The mass and energy release rates used for the containment backpressure calculation as a function of time during blowdown, are given in Table 15.4.1-6. 9

15.4.1.1.4 Results of Large Break Spectrum

Calculations of cold leg double-ended guillotine pipe breaks were performed over a range of Moody discharge coefficients (C_D), for a plant similar in design to the Sequoyah Units, to identify the case which produces the highest peak clad temperature. For that analysis, calculations were performed for discharge coefficients of 0.4, 0.6, and 0.8. This spectrum of breaks was performed assuming the availability of only minimum safety injection flow capacity, in accordance with the single failure criteria of 10 CFR 50, Appendix K. A break discharge coefficient of 0.6 was found to result in the highest peak clad temperature. Based on these results, this discharge coefficient was chosen to be analyzed as the limiting break size for Sequoyah Units 1 and 2. This case was found to result in the limiting peak clad temperature of ~~2084.2~~ 2069°F, which is below the 2200°F limit of 10 CFR 50.46 9

15.4.1.1.5 Effect of Containment Purging

To ~~assess~~ assess the impact of purging on the calculated post-LOCA Sequoyah containment pressure, a
calculation was first performed to obtain the amount of mass which exits through three available 7
sets of purge lines during the initial portion of a postulated LOCA transient. Purge-line isolation
closure time is assumed at 4.0 seconds after receipt of signal; during this interval, the full flow area
is presumed to be available. In addition, the time to reach the S.I. signal setpoint and the delay
necessary to generate the S.I. signal are conservatively assessed as 1.5 seconds total. Thus, flow
through three pairs of fully open available purge lines was evaluated from 0.0 to 5.5 seconds for
the postulated double-ended cold leg break.

The calculation employed the 50-node TMD computer code model which is described in Section 6.2.1.3.4. The 24-inch purge supply lines are connected to Volumes 34, 37, and 25; purge exhaust lines are connected to 36 and 25. Possible combinations of supply lines and exhaust lines open to the atmosphere were considered. Each of these purge lines is represented by a flow path of cross section area equal to 2.948 ft^2 and a total flow resistance factor equal to 3.98 (entrance and exit loss, three fully open butterfly valves and a debris screen). The most conservative two pairs of 24-inch purge and supply lines were assumed to be open in this calculation. In addition, two 12-inch lines connected to TMD node 29 were modeled as open.

In a computation for ECCS performance, the greatest impact on containment pressure occurs for the purge case of maximum air mass loss, which is based upon the two 12-inch lines being open and involves three open purge lines in the lower compartment (TMD elements 34, 36, and 37) and one purge line open in the upper compartment together with a cold leg break in TMD Volume 1. A total of 2820 pounds of air are calculated to be lost in this case. The maximum air loss case is the limiting case because any steam lost through purging in an ECCS backpressure elevation would otherwise be calculated to condense in the ice bed. Therefore, any steam lost through purging is ultimately of no consequence in the containment pressure determination, while any air loss directly reduces calculated pressure. To incorporate the TMD-calculated results, the initial compression peak of the LOTIC code was adjusted to consider the mass lost through purging. The corrected LOTIC containment pressure thus reflects the loss of mass through purging during the first few seconds of the LOCA transient.

The impact of the reduced containment pressure on ECCS performance is included in the calculated peak cladding temperature of ~~2084~~ 2067 °F. Basing the plant Technical Specification peaking factor on this result permits purging of the Sequoyah containment during normal operation to be conducted through three sets of purge lines. 9

TABLE 15.4.1-1

LARGE BREAK

Results	CD = 0.6 DECLG
Peak cladding temp (°F)	2169 *
Peak cladding location (ft)	7.0
Local Zr/H2O reaction (max)	6.79
Local Zr/H2O location (ft)	7.0
Hot rod burst time (seconds)	57.13
Hot rod burst location (ft)	5.75
Core-wide Zr/H_2O reaction (%)	<1.0

Calculation Assumptions	
Core power (MWt), 102 percent of	3411
Peak linear power (kw/ft), 102 percent of	13.067
Total Peaking factor (at license rating)	2.40
Hot channel enthalpy rise peaking factor	1.62
Accumulator water volume (cold leg delivered)(ft^3)	3 @ 1050 per accumulator
Steam generator tube plugging level	10 percent, uniform

* This value is applicable until such time as the standard fuel with inconel grids is removed from the core. At that time, this value will be 2069°F.

TABLE 15.4.1-3

BACKPRESSURE TRANSIENT USED IN ANALYSIS

Time (sec)	Pressure (psia)
36.3	17.25
38.7	16.56
42.0	16.1
50.0	15.56
55.0	15.41
57.1	15.36
64.0	15.29
85.0	14.97
100.0	14.66
10[illegible].5	14.54
150.1	15.74
200.0	15.91
250.0	15.96

TABLE 15.4.1-6 (Sheet 1)

MASS AND ENERGY RELEASE RATES
$C_D = 0.6$

TIME (sec)	$\dot{M}$ (lb/sec)	$\dot{e}$ (BTU/sec)
2.00E+00	5.5819E+04	2.9663E+07
4.00E+00	3.7638E+04	2.0528E+07
6.00E+00	2.9473E+04	1.6673E+07
8.00E+00	2.4009E+04	1.4579E+07
1.00E+01	2.0598E+04	1.2424E+07
1.20E+01	1.8380E+04	1.0753E+07
1.24E+01	1.7432E+04	1.0253E+07
1.40E+01	1.5121E+04	9.1177E+06
1.50E+01	1.3548E+04	8.3830E+06
1.60E+01	1.2148E+04	7.7059E+06
1.70E+01	1.0779E+04	7.0083E+06
1.80E+01	8.9009E+03	6.0757E+06
1.90E+01	7.6423E+03	5.3858E+06
2.00E+01	6.5315E+03	4.7371E+06
2.10E+01	5.6643E+03	4.2114E+06
2.20E+01	4.7930E+03	3.5873E+06
2.30E+01	3.6129E+03	2.5890E+06
2.40E+01	3.4061E+03	2.2302E+06
2.50E+01	4.7350E+03	2.3539E+06
2.60E+01	5.1390E+03	2.2818E+06
2.70E+01	5.0637E+03	2.0867E+06

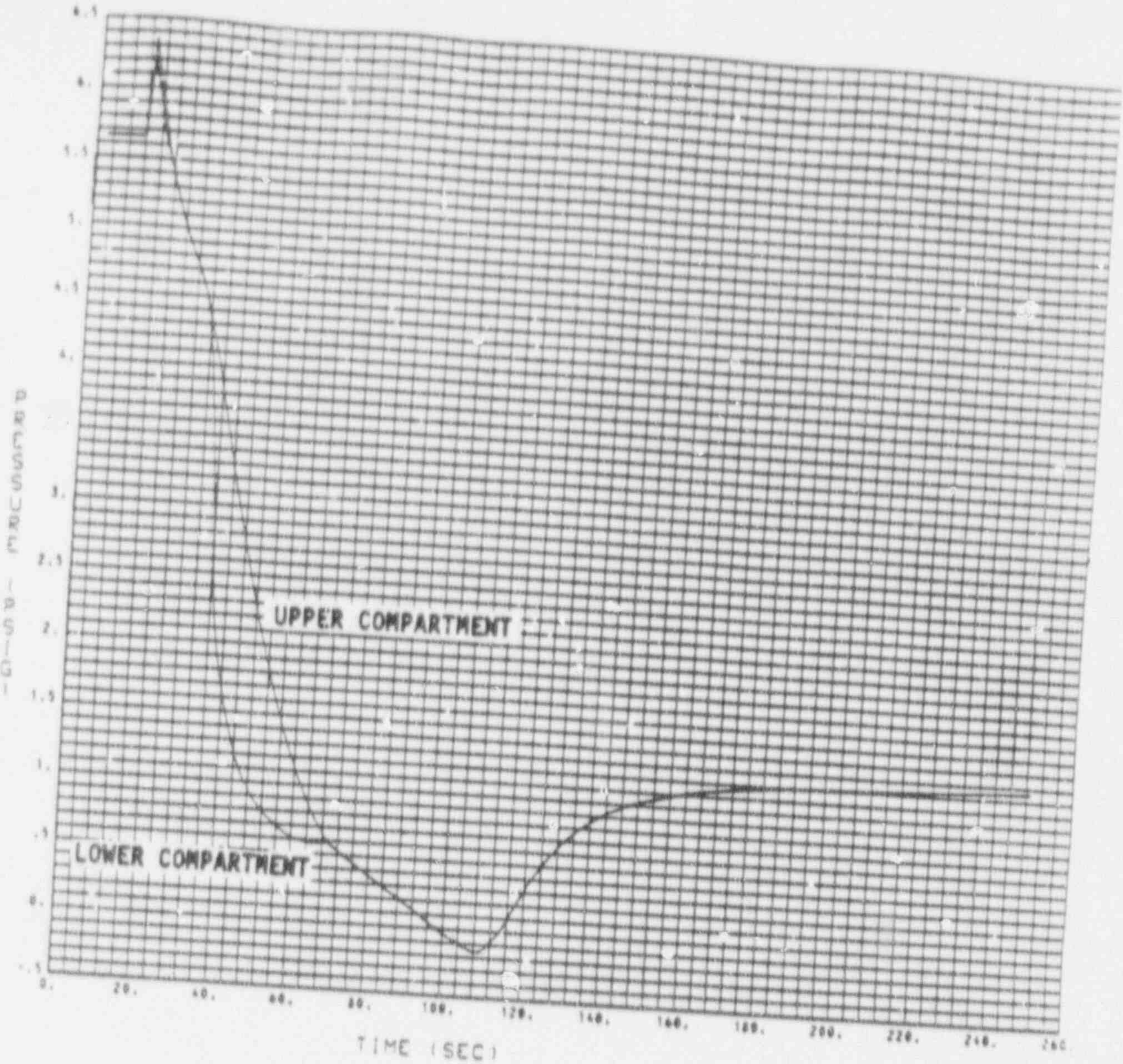
TABLE 15.4.1-6 (Sheet 2)
(Continued)

TIME (sec)	$\dot{M}$ (lb/sec)	$\dot{e}$ (BTU/sec)
2.80E+01	5.0177E+03	1.9106E+06
2.90E+01	4.8275E+03	1.6756E+06
3.00E+01	4.6839E+03	1.4576E+06
3.10E+01	4.0353E+03	1.1095E+06
3.20E+01	6.0525E+03	1.5903E+06
3.40E+01	2.9313E+03	6.8024E+05
3.60E+01	6.4982E+03	1.1822E+06
3.80E+01	1.0229E+03	5.9349E+04
4.00E+01	1.0070E+03	5.8401E+04
4.20E+01	9.9170E+02	5.7488E+04
4.40E+01	9.7660E+02	5.6587E+04
4.60E+01	9.6240E+02	5.5740E+04
5.00E+01	9.3620E+02	5.4177E+04
5.40E+01	9.1190E+02	5.2727E+04
5.79E+01	8.8975E+02	5.1463E+04
5.86E+01	8.9061E+02	5.6369E+04
6.37E+01	8.8159E+02	7.6607E+04
7.28E+01	2.1933E+02	5.4335E+04
9.63E+01	2.2707E+03	2.9896E+05
1.08E+02	2.1768E+03	2.8944E+05
1.55E+02	5.6515E+02	2.4903E+05
1.96E+02	5.7374E+02	2.3885E+05
2.38E+02	6.1940E+02	2.5198E+05
3.16E+02	6.3191E+02	2.4965E+05

TABLE 15.4.1-7

LARGE BREAK
TIME SEQUENCE OF EVENTS

	DECL C_D = 0.6 (seconds)
START	0.0
Rx trip signal	0.451
S.I. signal	2.7
Acc injection (CL)	19.9
End of blowdown	36.27
Bottom of core recovery	57.03
Acc empty (CL)	108.47
Pumped Injection	34.7
End of Bypass	34.5



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NUCLEAR PLANT UNITS 1 & 2
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Double Ended Cold Leg Guillotine Break, C_D=0.6
Compartment Pressure

Figure 15.4.1-1

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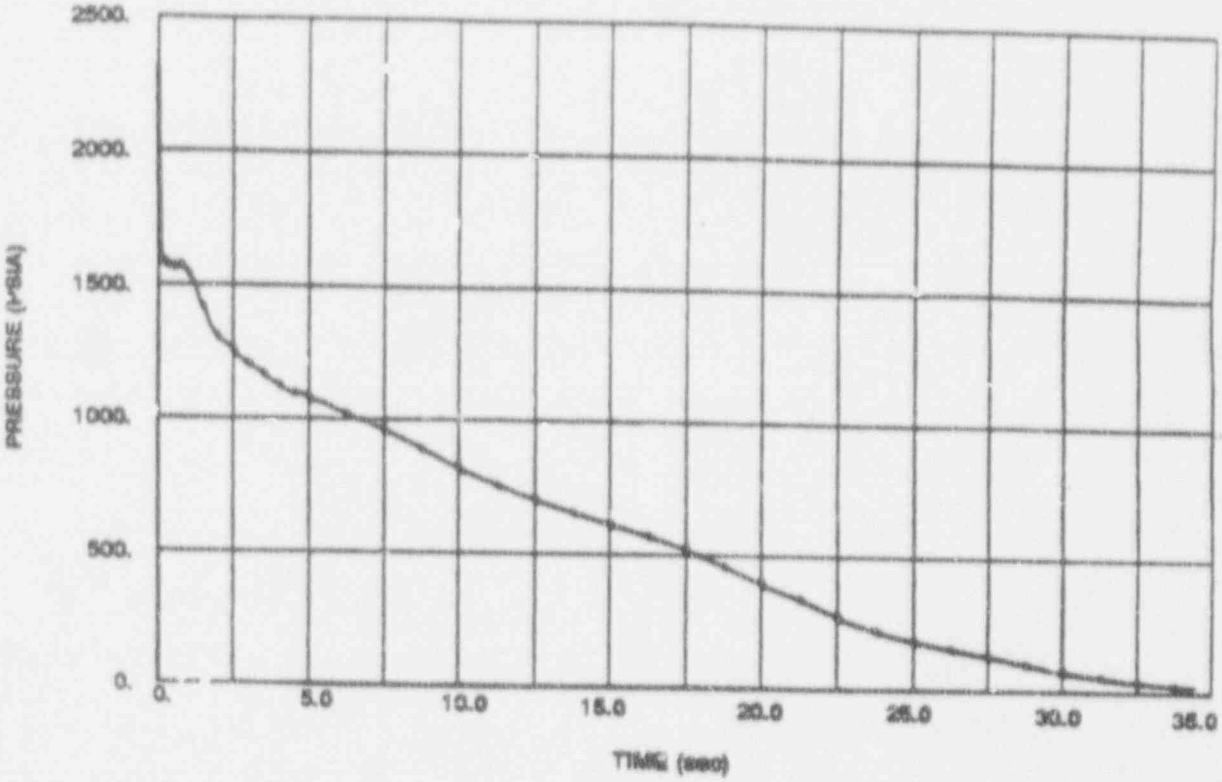

SEQUOYAH NUCLEAR PLANT UNITS 1 & 2 *Final Safety Analysis Report*
Double Ended Cold Leg Guillotine Break, C_D=0.6 RCS Pressure
Figure 15.4.1-2

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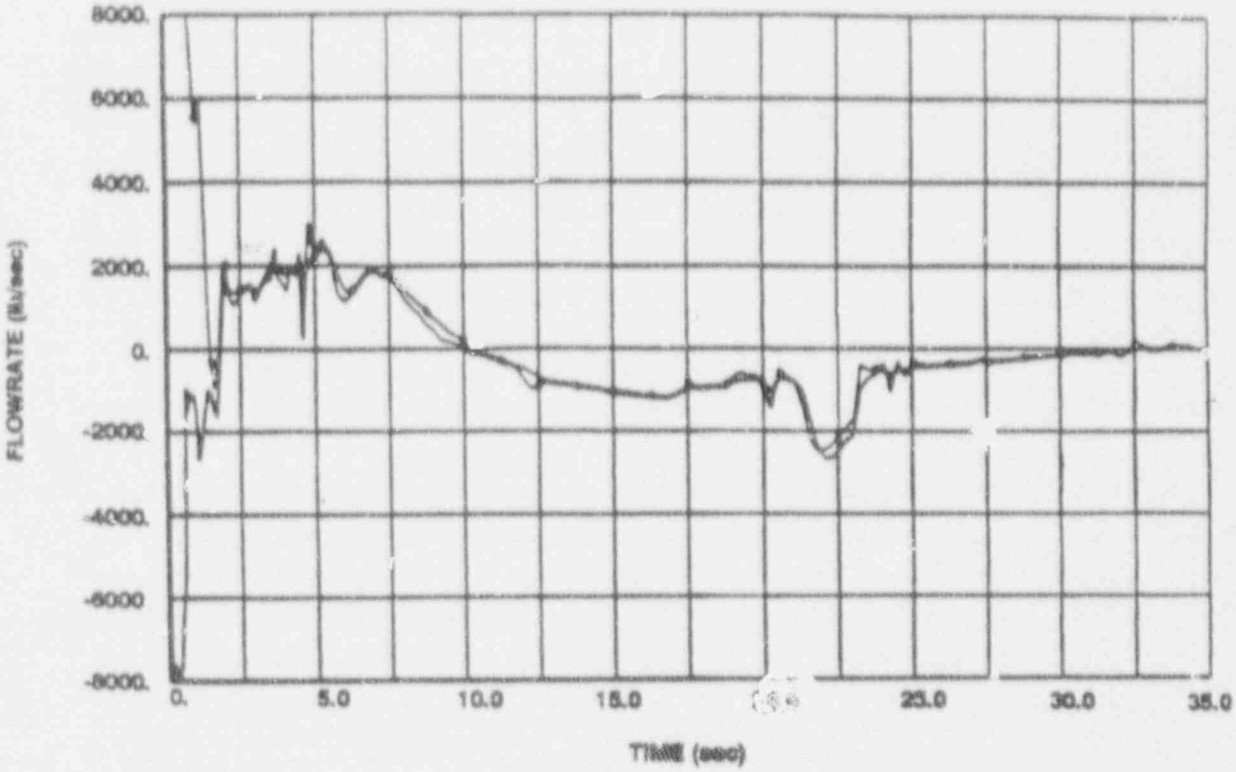

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Double Ended Cold Leg Guillotine Break, C_D=0.6
Core Flowrate

Figure 15.4.1-3

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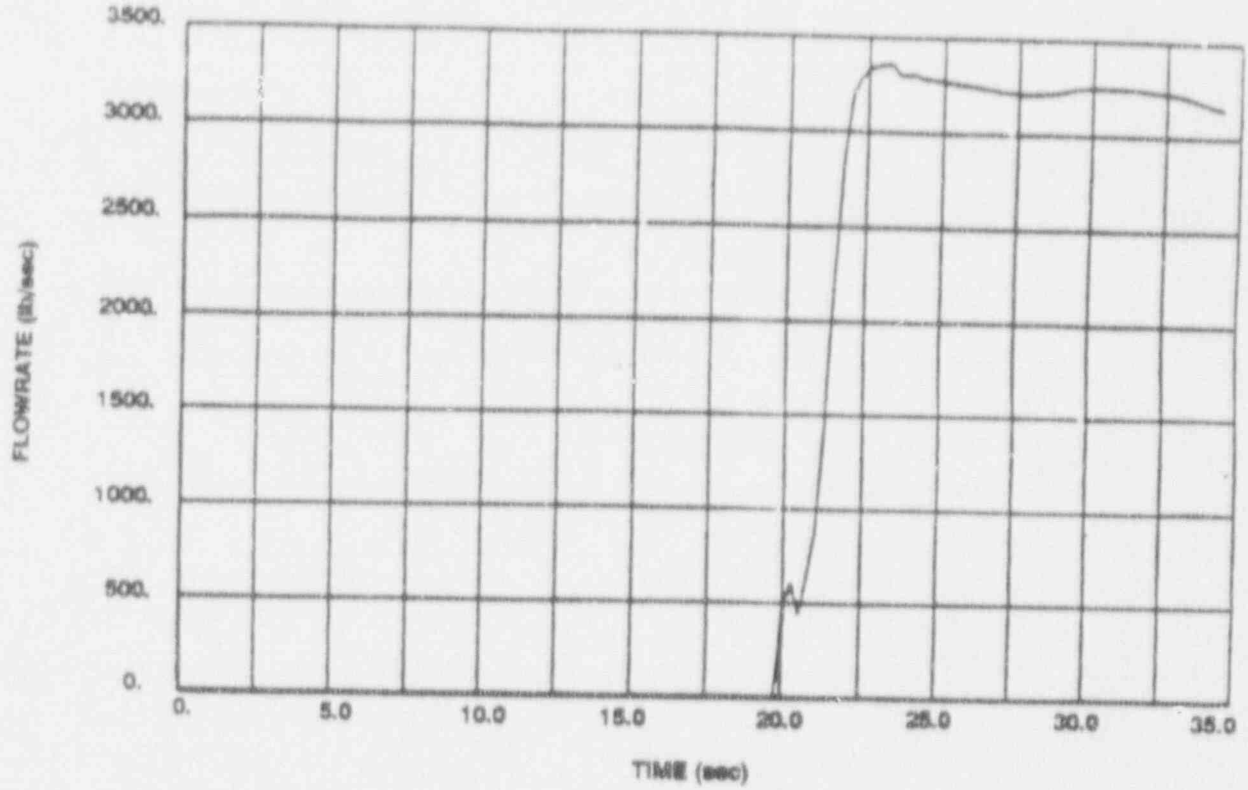

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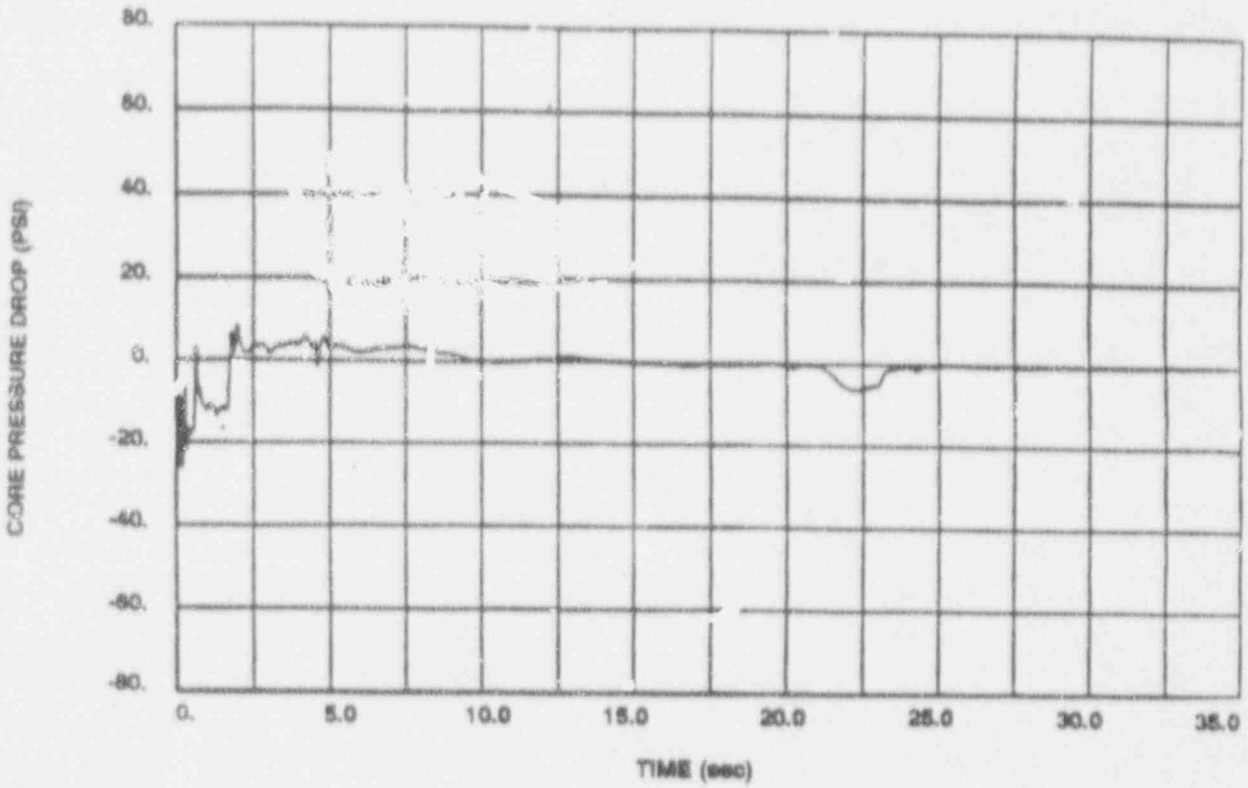
Double Ended Cold Leg Guillotine Break, C_D=0.6
Cold Leg Accumulator Flowrate

Figure 15.4.1-4



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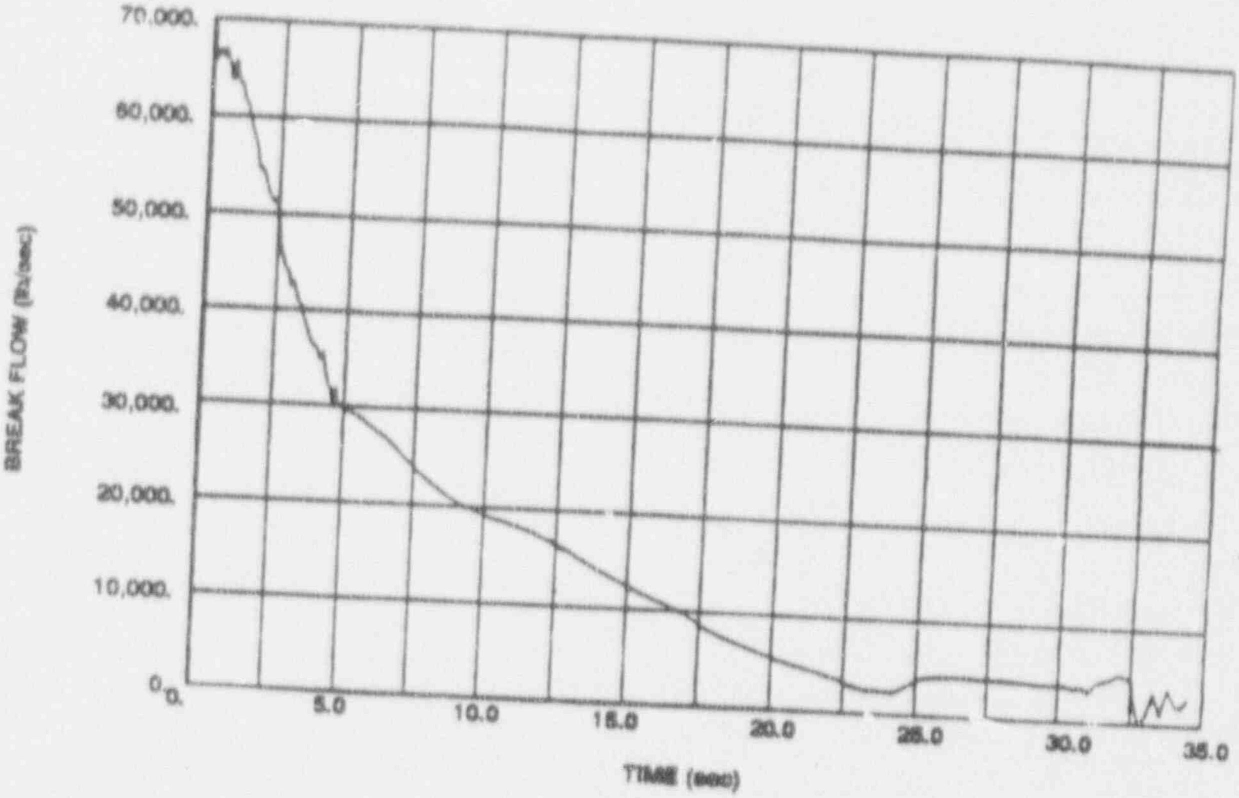
Double Ended Cold Leg Guillotine Break, C_D=0.6
Core Pressure Drop

Figure 15.4.1-5



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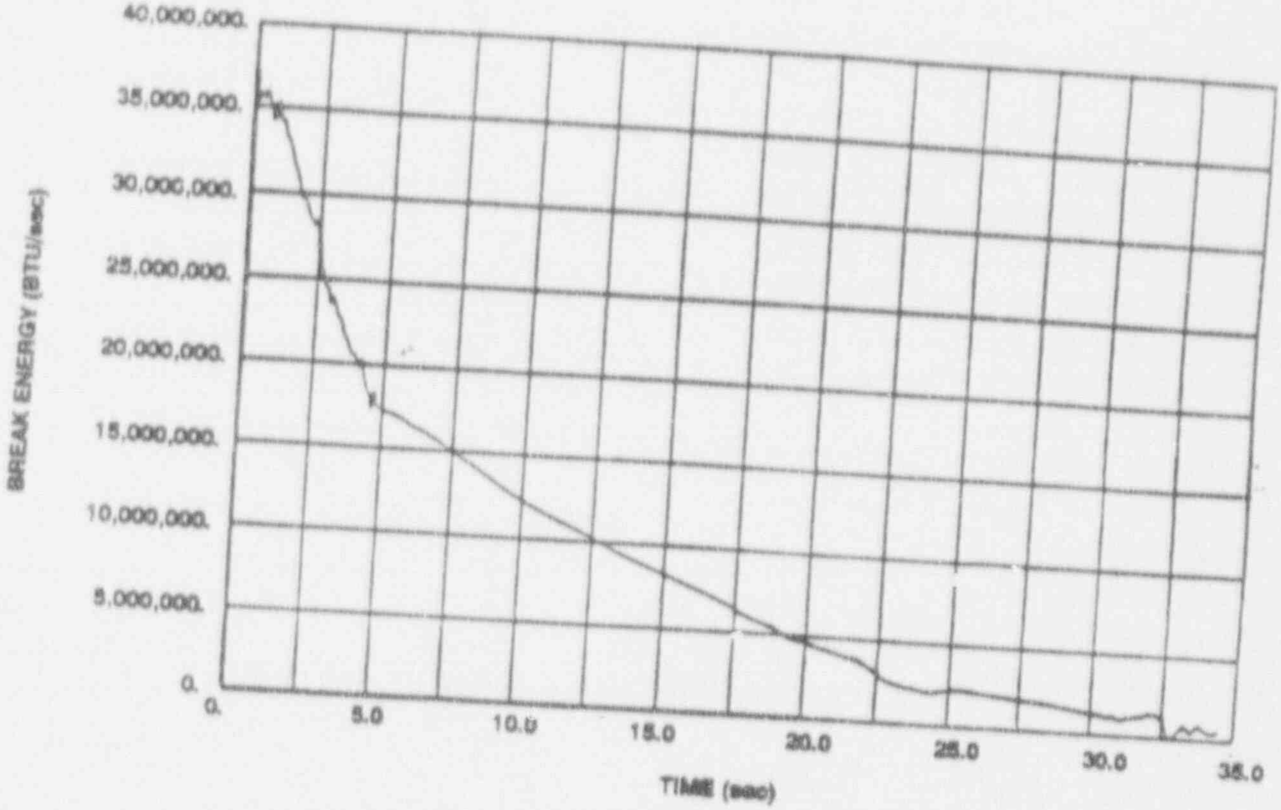
Double Ended Cold Leg Guillotine Break, C_D=0.6
Break Mass Flowrate

Figure 15.4.1-6



SEQUOYAH NUCLEAR PLANT UNITS 1 & 2 *Final Safety Analysis Report*
Double Ended Cold Leg Guillotine Break, C_D=0.6 Break Energy Flowrate
Figure 15.4.1-7

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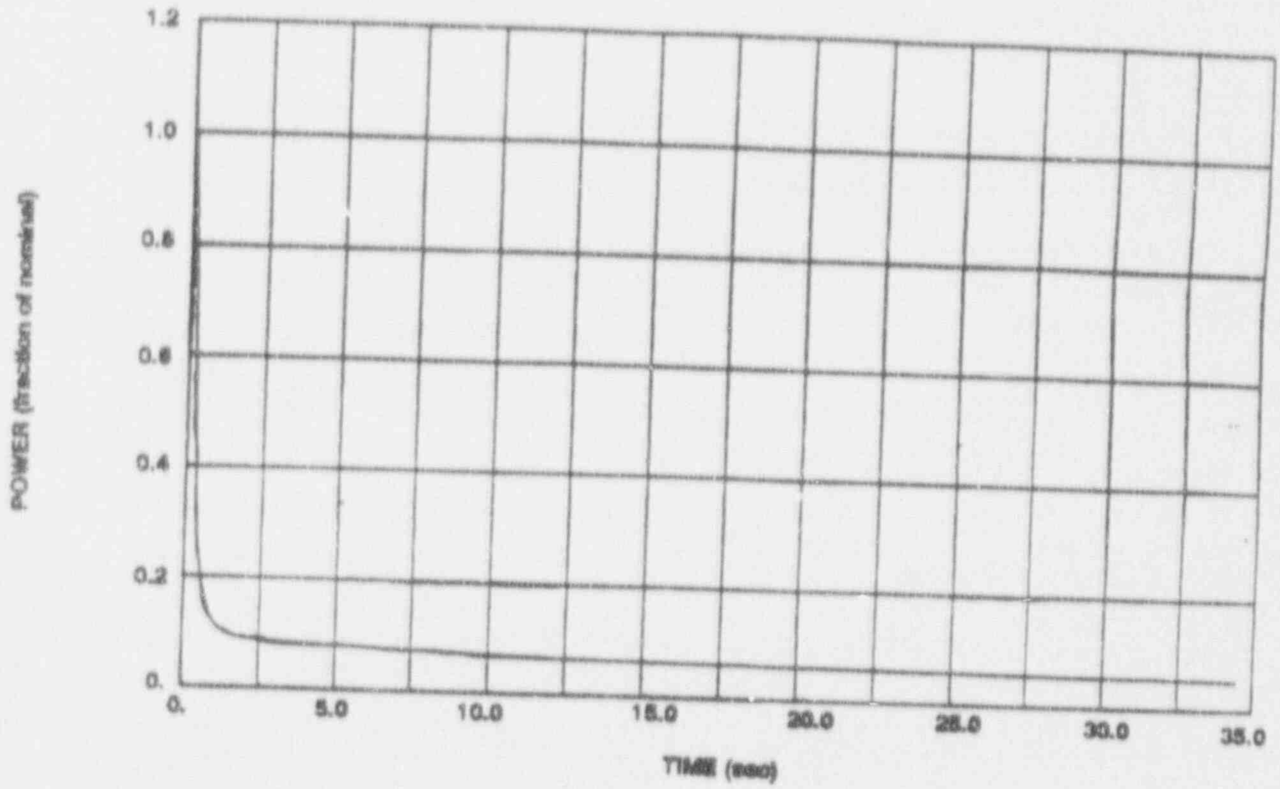

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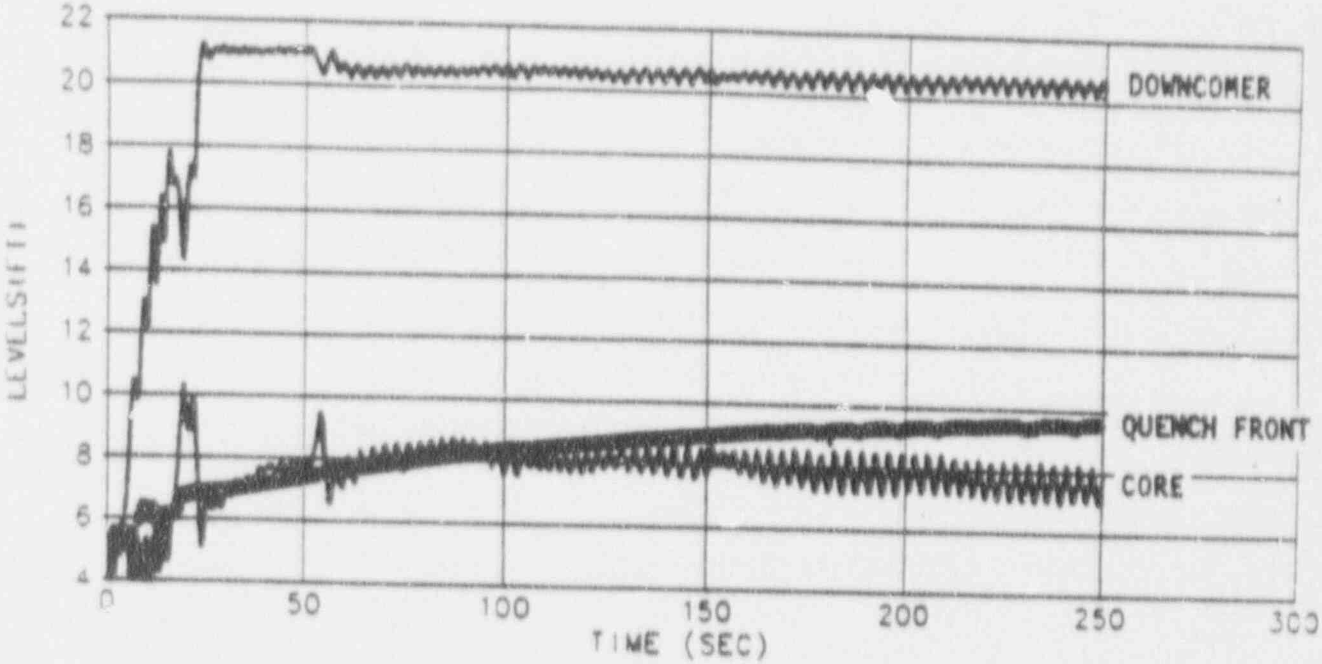
Double Ended Cold Leg Guillotine Break, C_D=0.6
Normalized Core Power

Figure 15.4.1-8



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Double Ended Cold Leg Guillotine Break, C_D=0.6
Core and Downcomer Liquid Levels

Figure 15.4.1-9

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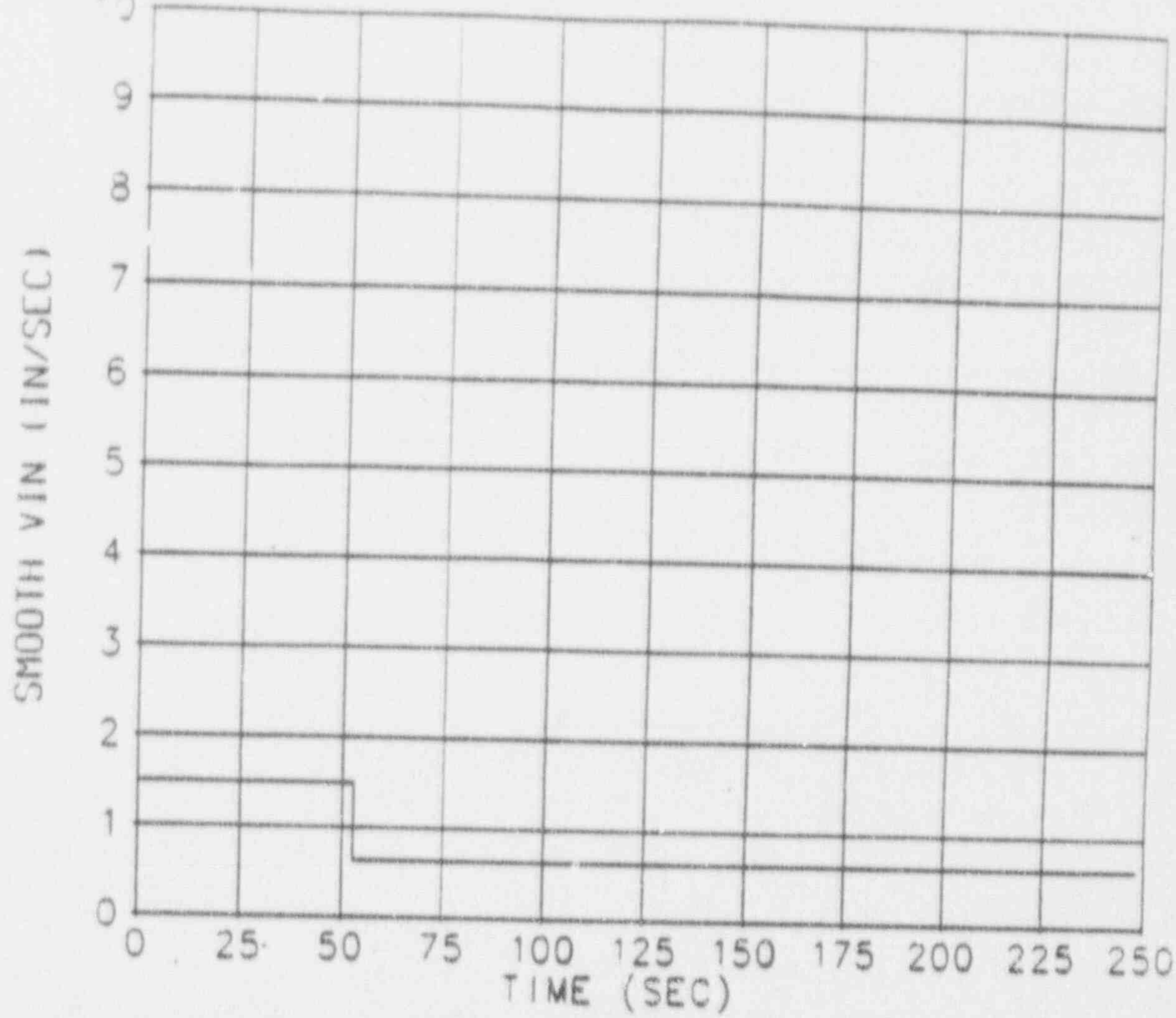

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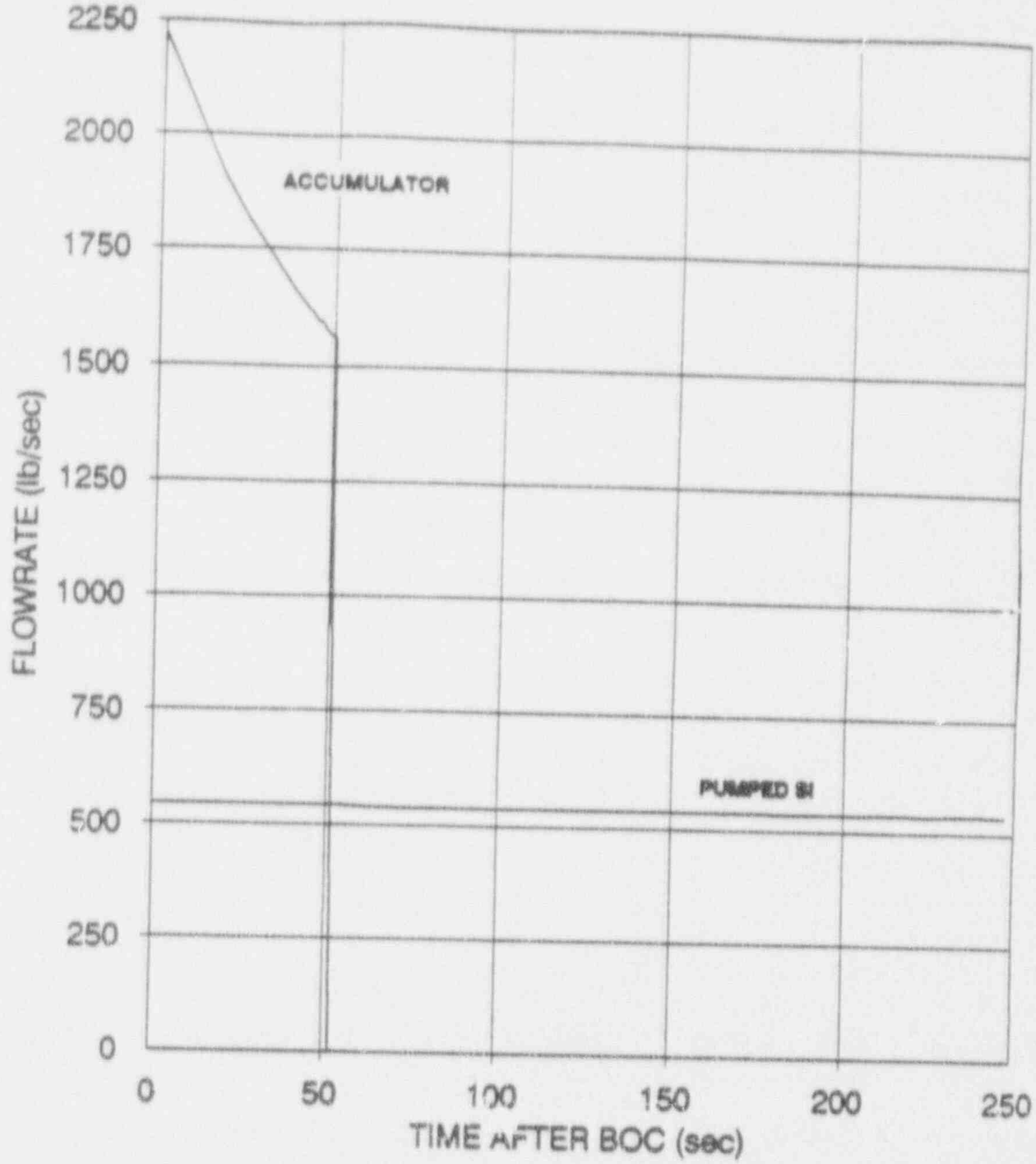
Double Ended Cold Leg Guillotine Break, C_D=0.6
Core Inlet Fluid Velocity

Figure 15.4.1-10



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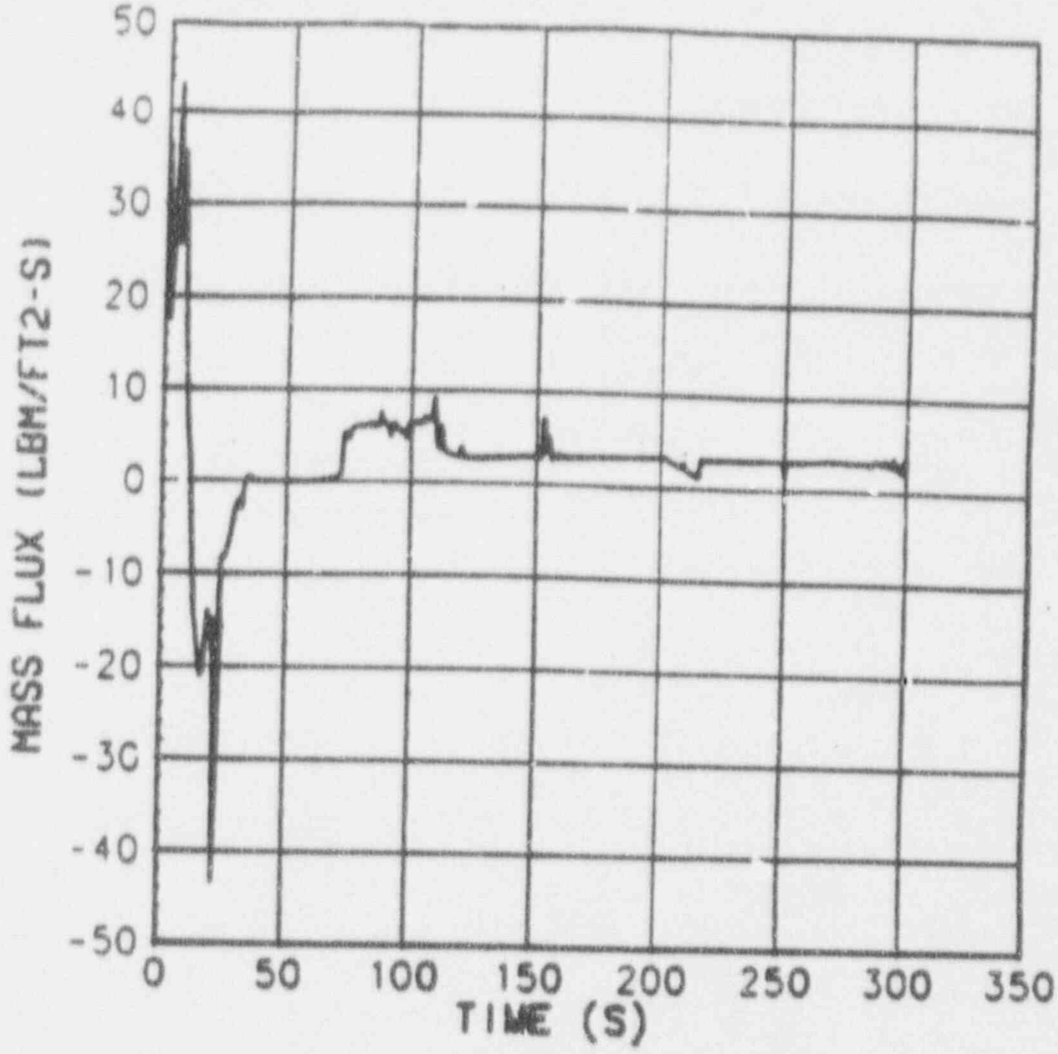
Double Ended Cold Leg Guillotine Break, C_D=0.6
Accumulator and Pumped SI Flowrate

Figure 15.4.1-11



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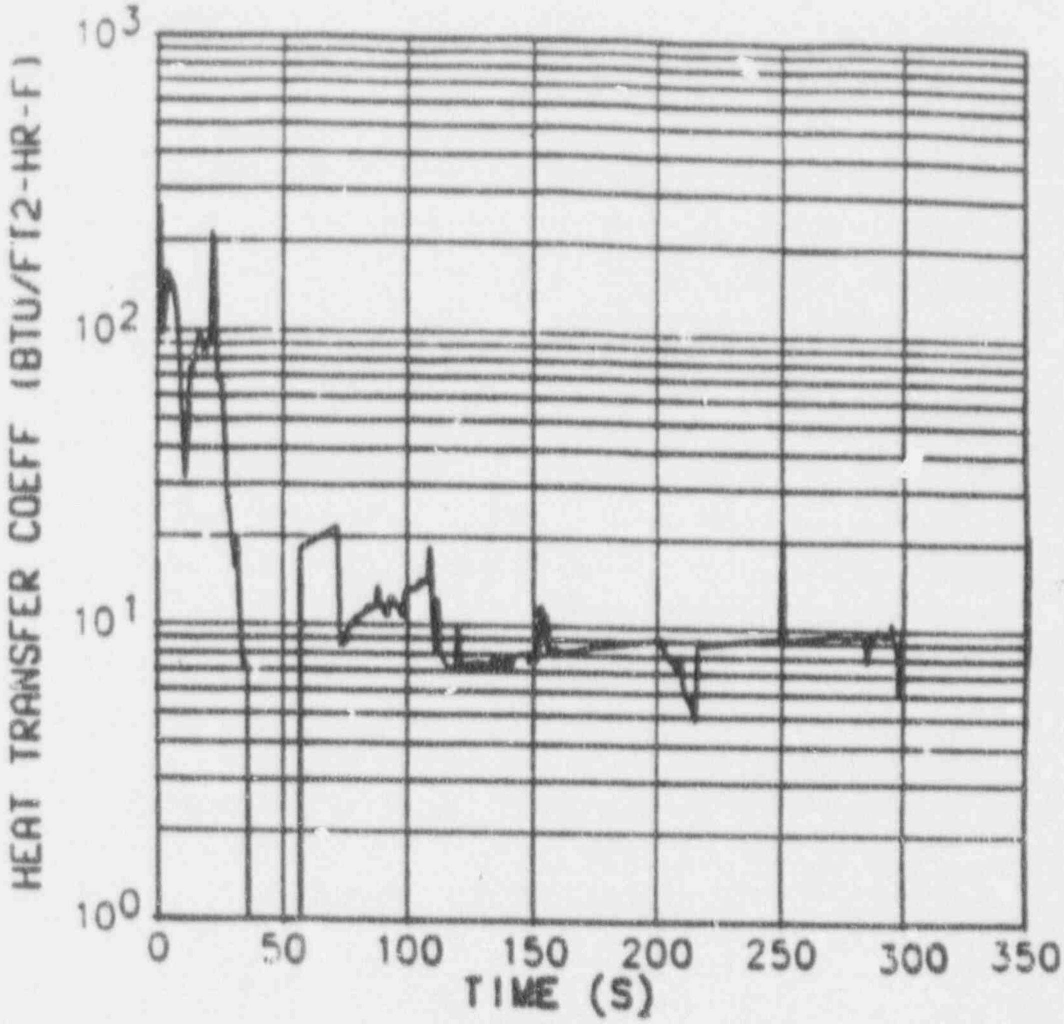
Double Ended Cold Leg Guillotine Break, C_D=0.6
Fluid Mass Flux

Figure 15.4.1-12



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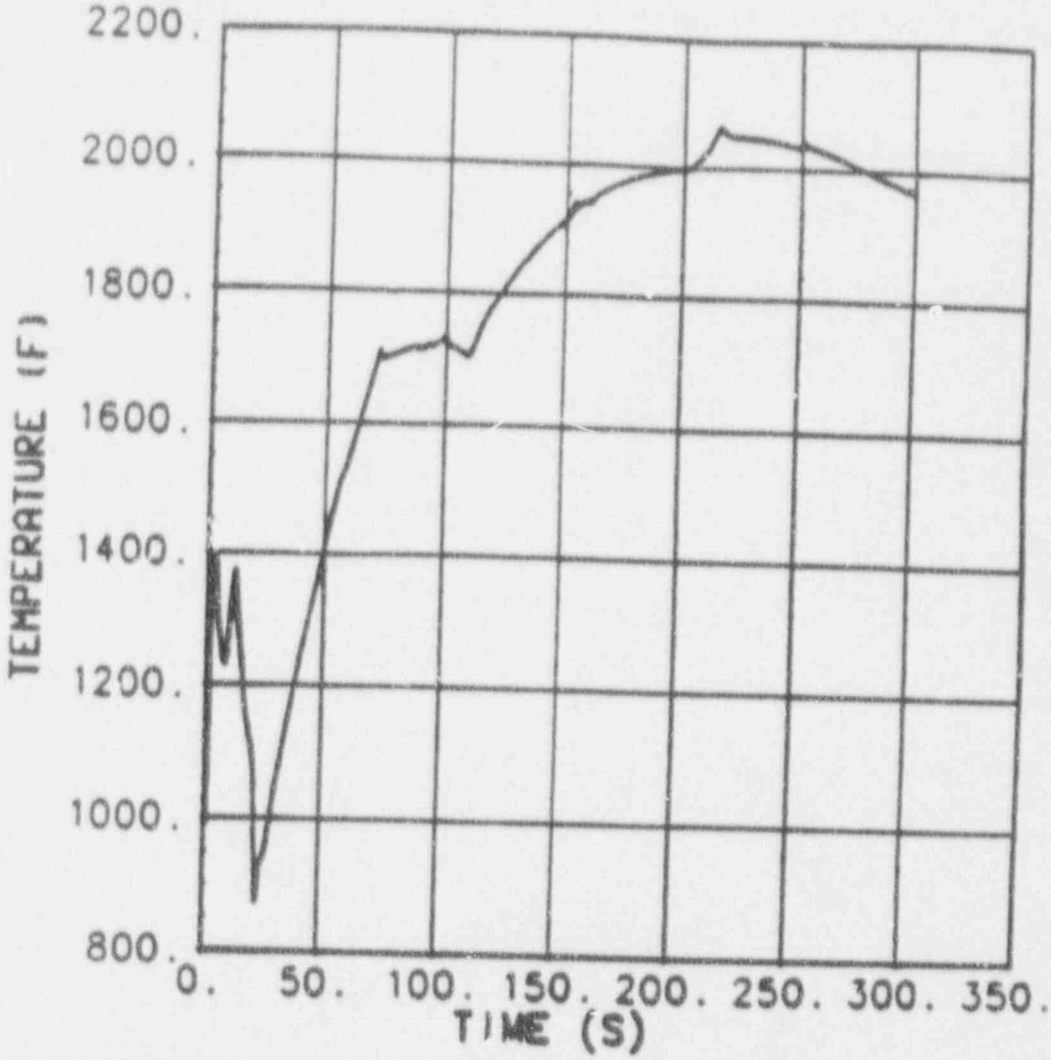
Double Ended Cold Leg Guillotine Break, C_D=0.6
Rod Heat Transfer Coefficient

Figure 15.4.1-13



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Double Ended Cold Leg Guillotine Break, C_D=0.6
CTad Peak Temperature

Figure 15.4.1-14

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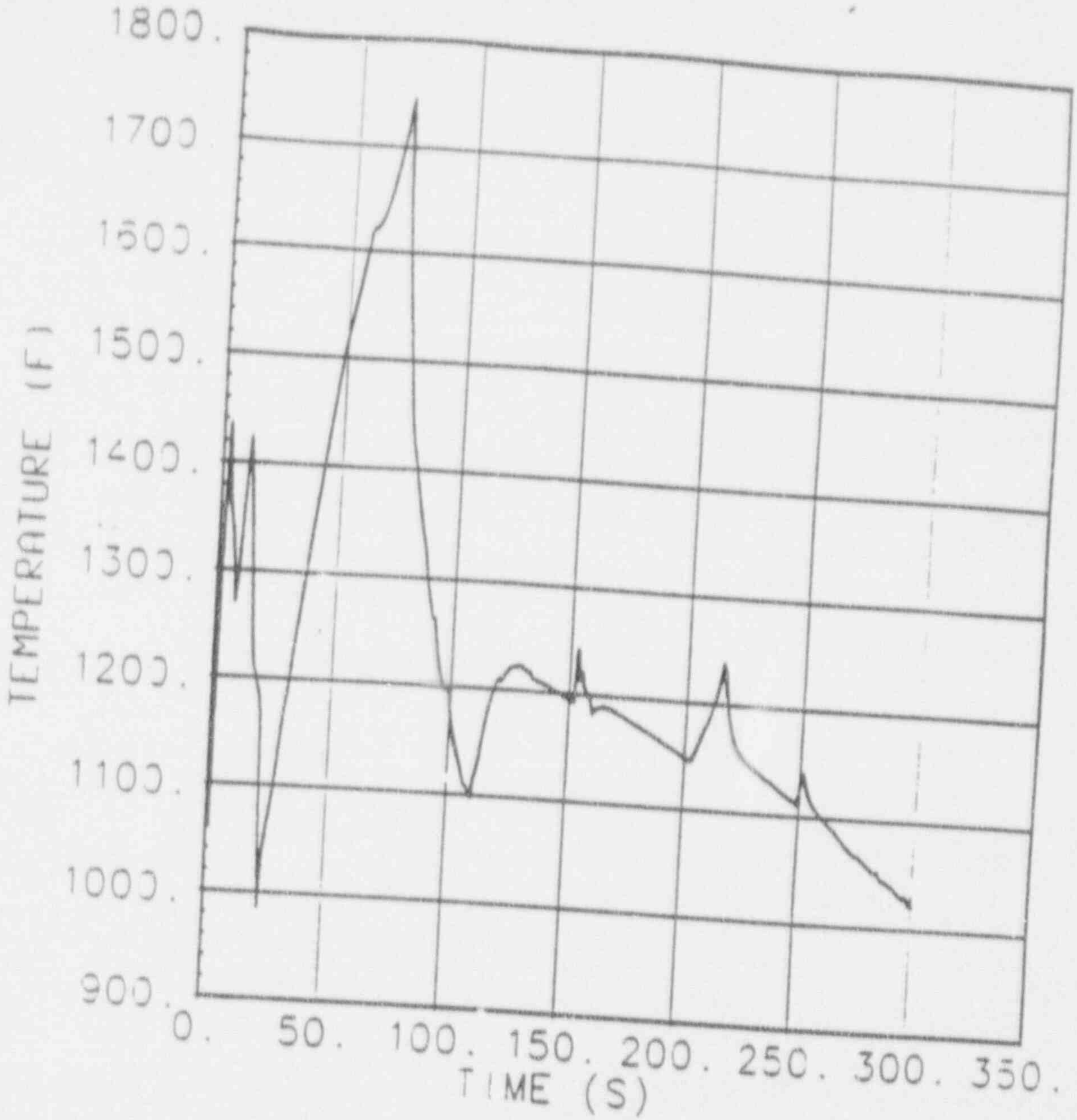

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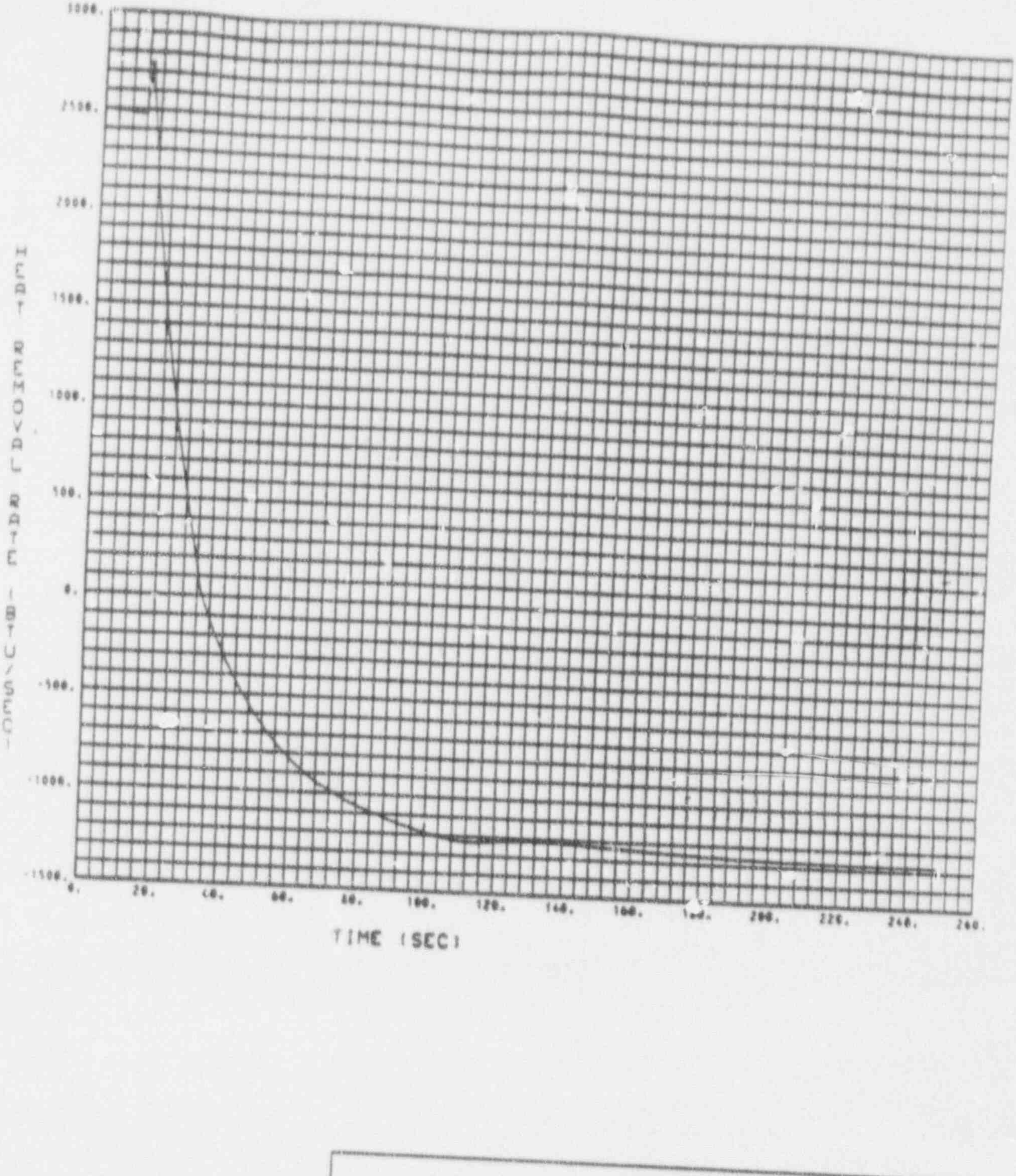
Double Ended Cold Leg Guillotine Break, C_D=0.6
Clad Temperature at the Burst Elevation

Figure 15.4.1-15



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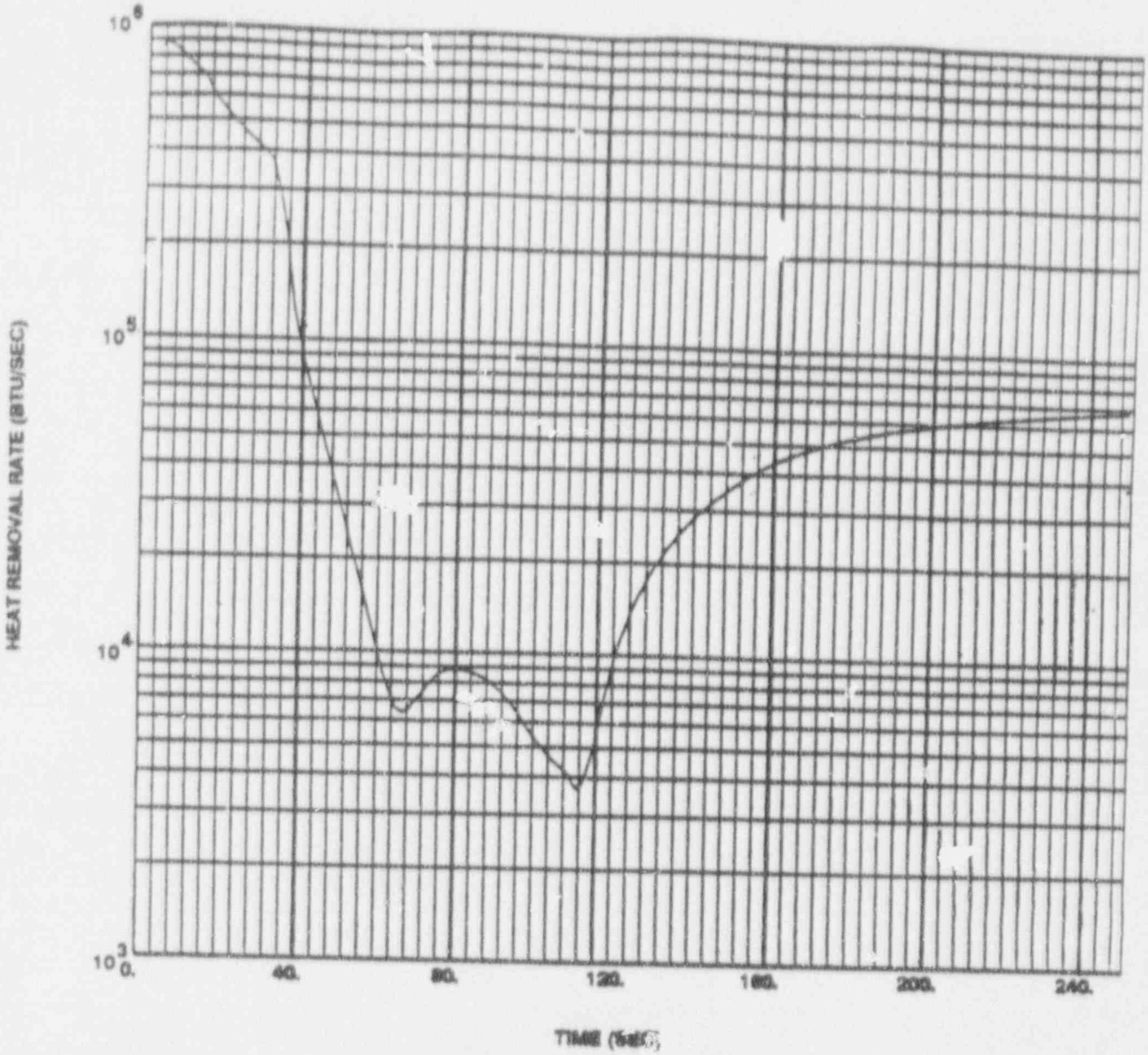
Double Ended Cold Leg Guillotine Break, C_D=0.6
Upper Compartment Structural Heat Removal Rate

Figure 15.4.1-16



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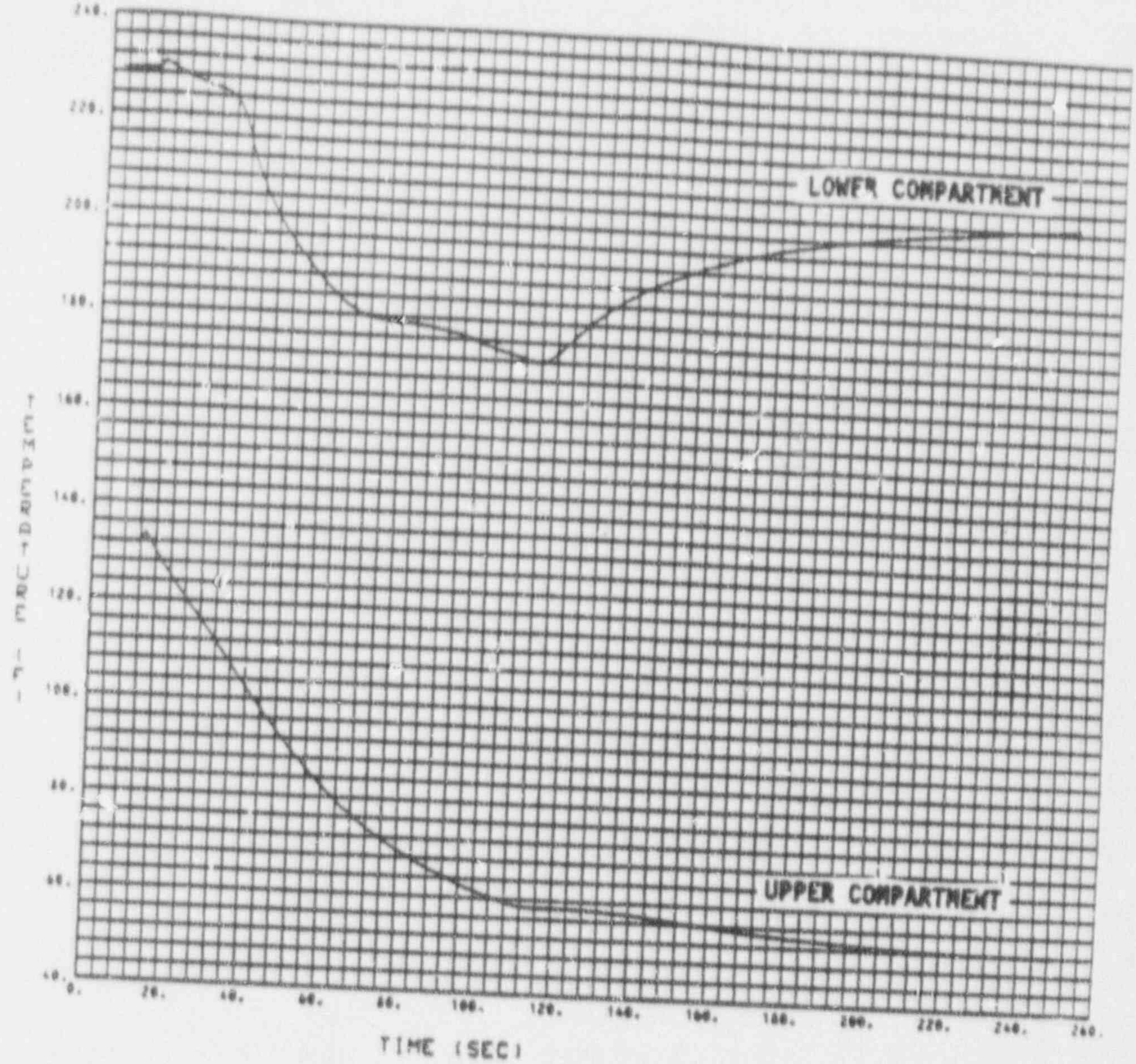
Double Ended Cold Leg Guillotine Break, C_D=0.6
Lower Compartment Structrual Heat Removal Rate

Figure 15.4.1-17



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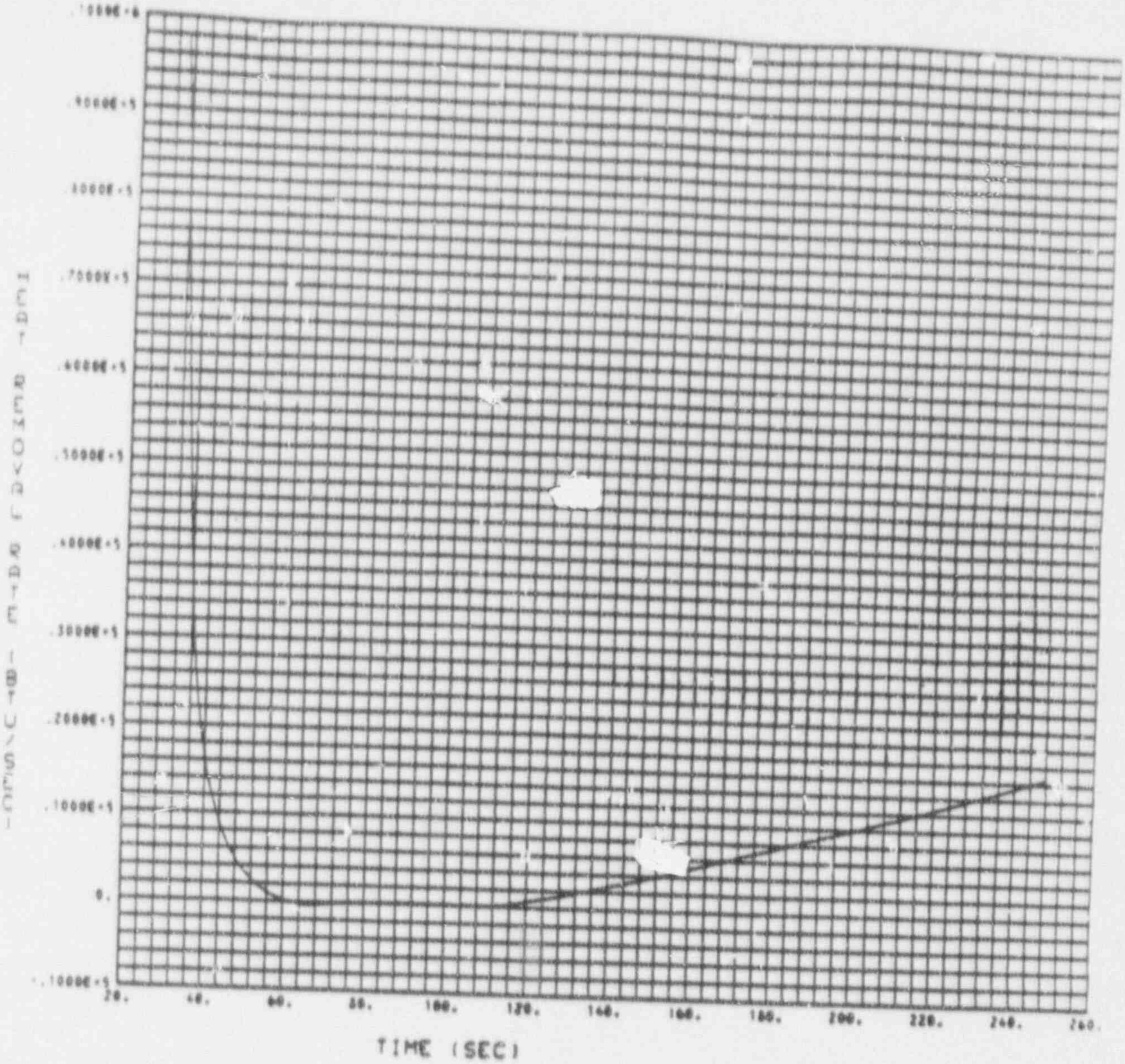
Double Ended Cold Leg Guillotine Break, C_D=0.6
Compartment Temperature

Figure 15.4.1-18



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NUCLEAR PLANT UNITS 1 & 2
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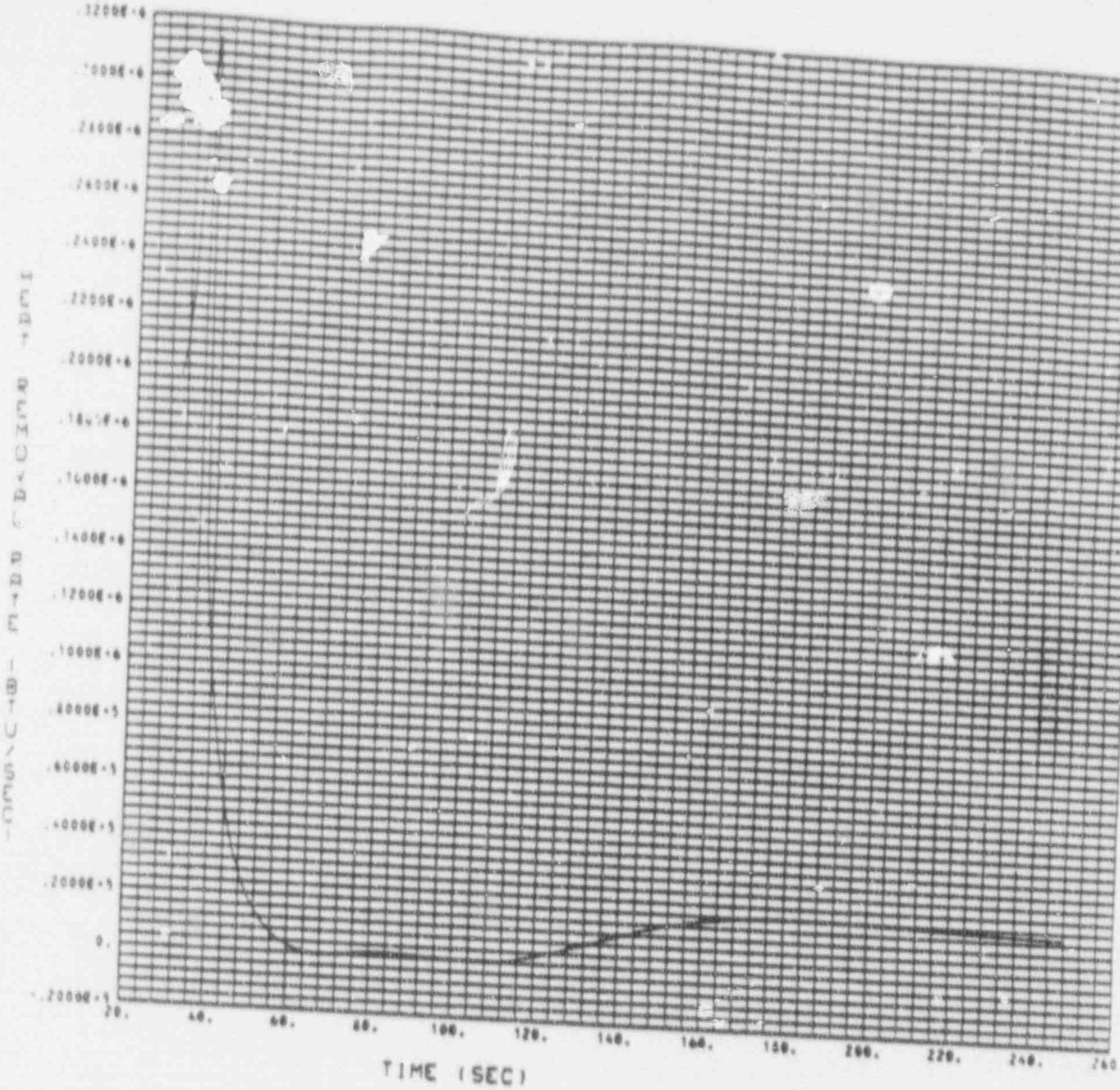
Double Ended Cold Leg Guillotine Break, C_D=0.6
Heat Removal by Sump

Figure 15.4.1-19



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NUCLEAR PLANT UNITS 1 & 2
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Double Ended Cold Leg Guillotine Break, C_D=0.6
Heat Removal by IC Drain

Figure 15.4.1-20

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10.1.3 Chapter 4

TABLE 4.1-1 (Sheet 1)

REACTOR DESIGN COMPARISON TABLE

THERMAL AND HYDRAULIC DESIGN PARAMETERS	SEQUOYAH UNITS 1 & 2 17x17 FUEL ASSEMBLY WITH DENSIFICATION EFFECTS ~~(d)~~	REFERENCE PLANT 17x17 FUEL ASSEMBLY WITH DENSIFICATION EFFECTS
1. Reactor Core Heat Ouput, MWt	3411	3411
2. Reactor Core Heat Output, Btu/hr	11,641.7 x 10^6	11,641.7 x 10^6
3. Heat Generated in Fuel, %	97.4	97.4
4. System Pressure, Nominal, psia	2250	2250
5. System Pressure, Min. Steady State, psia	2200	2220
6. Minimum DNBR for Design Transients Coolant Flow	>1.30	>1.30
DNB Correlation	WRB-1 ~~"R" (W-3~~ with modified spacer factor	"L" (W-3 with modified spacer factor
7. Total Thermal Flow Rate, lb/hr	138.0 x 10^6	132.7 x 10^6
8. Effective Flow Rate for Heat Transfer, lb/hr	127.7 x 10^6	126.7 x 10^6
9. Effective Flow Area for Heat Transfer, ft^2	51.1	51.1
10. Average Velocity Along Fuel Rods, ft/sec	15.6	15.7
11. Average Mass Velocity, lb/hr-ft^2	2.50 x 10^6	2.48 x 10^6
Coolant Temperature, °F		
12. Nominal Inlet	546.7	552.5
13. Average Rise in Vessel	63.1	64.2
14. Average Rise in Core	67.6	66.9
15. Average in Core	582.2	585.9
16. Average in Vessel	578.2	584.7
Heat Transfer		
17. Active Heat Transfer, Surface Area, ft^2	59,700	59,700
18. Average Heat Flux, Btu/hr-ft^2	189,800	189,800
19. Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	~~440,300(a)~~ 455,560	474,500(b)
20. Average Thermal Output, kw/ft	5.44	5.44
21. Maximum Thermal Output for Normal Operation, kw/ft	~~12.2(a)~~ 13.0	13.6(b)
22. Peak Linear Power for Determination of Protection Setpoints, kw/ft	18.0(c)	18.0(c)
23. Heat Flux Hot Channel Factor, F_Q	~~2.32~~ 2.40	2.50

~~(a) This limit is associated with the value of F_Q = 2.32~~
(b) This limit is associated with the value of F_Q = 2.50
(c) See Subparagraph 4.3.2.2.6
~~(d) Initial core design~~

0264F/COC4

including the above factors, provided the assumed error in operation does not continue for a period which is long compared to the xenon time constant. The calculation results shown on Figure 4.3.2-23 which are greater than 18 kW/ft result from transients which would proceed without operator intervention for greater than 0.25 hour and would result in violation of the control rod insertion limits in the Technical Specifications. Since the peak kW/ft is below the above limit, no flux imbalance penalties are required for overpower protection. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error. Additional detailed discussion of these analyses is presented in Reference 7 and 30. 4

Analyses of possible operating power shapes show that the appropriate hot channel factors F_Q and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are the values given in Table 4.3.2-2 and addressed in the SQN Technical Specifications.

1.62

The maximum allowable F_Q can be increased with decreasing power, as shown in the SQN Technical Specifications. Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits, as described in Section 4.4.3. The allowance for increased $F^N_{\Delta H}$ permitted is $F^N_{\Delta H}$ = 1[redacted] (1 + 0.3 (1-P)). This becomes a design basis criterion which is used for establishing acceptable control rod patterns and control bank sequencing. Likewise, fuel loading patterns for each cycle are selected with consideration of this design criterion. The worst values of $F^N_{\Delta H}$ for possible rod configurations occurring in normal operation are used in verifying that this criterion is met. Typical radical factors and radial power distributions are shown in Figure 4.3.2-6 through 4.3.2-11. The worst values generally occur when the rods are assumed to be at their insertion limits. Maintenance of axial offset control establishes rod positions which are above the allowed rod insertion limits, thus providing increasing margin to the $F_{\Delta H}$ criterion. Section 3.2 of Reference 8 discusses the determination of $F_{\Delta H}$. These limits are taken as input to the thermal hydraulic design basis, as described in Section 4.4.3.2.1. 6

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition. These alarms are described in detail in Chapter 7.0.

4.3.2.2.7 <u>Experimental Verification of Power Distribution Analysis</u> 5

This subject is discussed in depth in Reference 2. A summary of this report is given here.

In a measurement of peak local power density, F_Q with the moveable detector system described in Subsection 7.7.1 and 4.4.5, the following uncertainties have to be considered.

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4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 Design Bases

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that heat removal by the Reactor Coolant System or the Emergency Core Cooling System (when applicable) assures that the following performance and safety criteria requirements are met:

1. Fuel damage' is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the plant cleanup system and are consistent with the plant design bases.

2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged' although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

In order to satisfy the above criteria the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure from Nucleate Boiling Design Basis

Basis

There will be at least a 95% probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events) at 95% confidence level. Historically, this criterion has been conservatively met by adhering to the following thermal design basis: there must be at least a 95% probability that the minimum departure from nucleate boiling ratio (DNBR) of the limiting power rod during Condition I and II events is greater than or equal to the Safety analysis DNBR limit ~~of the DNB correlation being used. The DNBR limit for the correlation is established based on the variance of the correlation such that there is a 95% probability with 95% confidence that DNB will not occur when the calculated DNBR is at the DNBR limit.~~

insert A

Discussion

Historically, the DNBR limit has been 1.30 for Westinghouse applications. In this application, the WRB-1 correlation (Reference 86) is employed. With the significant improvement in the accuracy of the critical heat flux prediction by using the WRB-1 correlation instead of previous DNB correlations, a DNBR limit of 1.17 is applicable for the 17x17 Standard fuel assembly (Reference 86) and for the VANTAGE 5H fuel assembly (Reference 87).

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Insert A

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The design limit DNBR is set at 1.22 for the typical cell and 1.21 for the thimble cell. Plant specific margin to accomodate rod bow and other DNB penalties and allowance for flexib lity in the design, operation and analysis of the plant is provided by performing the safety analyses to a safety analyses DNBR limit value of 1.38.

Insert B

DNBR margin is maintained for the Standard and VANTAGE 5H fuel by performing the DNB safety analysis to a DNBR limit of 1.38. Comparing this limit of 1.38 to the WRB-1 correlation limit of ~~1.17 results in 15.2% DNBR margin.~~

By preventing departure from nucleate boiling, adequate heat transfer is assured between the fuel clad and the reactor coolant, thereby preventing clad damage as a result of inadequate cooling. Maximum fuel rod surface

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[1]Fuel damage as used here is defined as penetration of the fission product barrier (i.e., the fuel rod clad).

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Insert B

THINC-IV and Transient Code uncert.

The design method used to meet the DNB design basis is the MINI-Revised Thermal Design Procedure (Reference *101*) which is a conservative application of the Revised Thermal Design Procedure (Reference *102*). In the MINI-RTDP method, uncertainties in the nuclear peaking factors and fuel fabrication parameters are combined statistically with the ~~x~~DNB correlation uncertainties to define the DNBR design limit such that there is at least a 95 percent probability (with 95 percent confidence) that DNB will not occur when the calculated minimum DNBR is equal to or greater than the design limit. The uncertainties included in the MINI-RTDP method are for the nuclear enthalpy hot-channel factor, F(N, H); the enthalpy rise engineering hot-channel factor, F(E, H); and the THINC-IV and transient codes. Since the uncertainties in these parameters are considered in determining the design DNBR value, the plant safety analyses are performed using input values without uncertainties for these parameters. For this application, the DNBR design limit value is *1.22 for the typical cell and 1.21 for the thimble cell*

In addition to the considerations above, a specific plant allowance has been considered in the present analysis. In particular, a DNBR limit value of 1.18 has been used in the safety analyses for the plant. The difference between the DNBR value used in the safety analyses and the design DNBR value (1.18 vs. *1.22*/1.21) provides plant specific DNB margin to offset the rod bow penalty and other DNB penalties that may occur. This DNB margin may also be used for flexibility in the design, operation or analysis of the plant.

For conditions outside the range of parameters for the WRB-1 correlation (refer to Section 4.4.2.3.1), the W-3 correlation is used with a DNBR correlation limit of 1.30 for pressure equal to or greater than 1000 psia. For low pressure applications (500-1000 psia), the W-3 DNBR correlation limit is 1.45 (Reference *103*).

4.4.2.2.4 <u>Surface Heat Transfer Coefficients</u>

The fuel rod surface heat transfer coefficients during subcooled forced convection and nucleate boiling are presented in Subparagraph 4.4.2.8.1.

4.4.2.2.5 <u>Fuel Clad Temperatures</u>

The outer surface of the fuel rod at the hot spot operates at a temperature of approximately 660°F for steady state operation at rated power throughout core life due to the onset of nucleate boiling. Initially (beginning-of-life), this temperature is that of the clad metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod surface causes the clad surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. Since the thermal-hydraulic design basis limits DNB, adequate heat transfer is provided between the fuel clad and the reactor coolant so that the core thermal output is not limited by considerations of the clad temperature.

4.4.2.2.6 <u>Treatment of Peaking Factors</u>

The total heat flux hot channel factor, F_Q, is defined by the ratio of the maximum to core average heat flux as discussed in Subparagraph 4.3.2.2.1, the design value F_Q for normal operation is ~~2.32~~ 2.40, including fuel densification effects. Subparagraph 15.4.1.1.7 discusses the F_Q value used in LOCA analyses.

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13.0

This results in a peak local power of 12.6 kW/ft at full power conditions. The peak linear power for determination of protection setpoints is 21.1 kW/ft. The centerline temperature at this kW/ft must be below the UO_2 melt temperature over the lifetime [illegible], including allowances for uncertainties. The fuel temperature design basis is discussed in Subsection 4.4.1.2 and results in a maximum allowable calculated centerline temperature of 47[illegible]. The peak linear power for prevention of centerline melt is > 21.1 kW/ft. The centerline temperature at the peak linear power resulting from overpower transients/overpower errors (assuming a maximum overpower of 118%) is below that required to produce melting.

4.4.2.3 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology

The minimum DNBRs for the rated power, and anticipated transient conditions are given in Table 4.4.2-1. The minimum DNBR in the limiting flow channel will be downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

DNBR's are calculated by using the correlation and definitions described in the following Subparagraphs 4.4.2.3.1 and 4.4.2.3.2. The THINC-IV computer code (discussed in Subparagraph 4.4.3.4.1) is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in Subparagraph 4.4.3.2.1 (nuclear hot channel factors) and in Subparagraph 4.4.2.3.4 (engineering hot channel factors).

4.4.2.3.1 Departure from Nucleate Boiling Technology

The WRB-1 DNB correlation is applicable to VANTAGE 5H fuel since, from a DNB perspective, the VANTAGE 5H assembly is virtually identical to the 17x17 Inconel R-Grid design. As documented in Reference 87, the use of the WRB-1 DNB correlation with a 95/95 limit DNBR of 1.17 is applicable to the VANTAGE 5H fuel assembly.

For conditions outside the range of applicability of the WRB-1, the W-3 correlation is used.

The W-3 correlation, and several modifications of it, have been used in Westinghouse CHF calculations. The W-3 was originally developed from single tube data, (Reference 39) but was subsequently modified to apply to the 0.422 inch O.D. rod "R" grid, (Reference 42) and "L" grid, (Reference 38) as well as the 0.374 inch O.D., (Reference 84, 40) rod bundle data. These modifications to the W-3 correlation have been demonstrated to be adequate for reactor rod bundle design.

For the W-3 correlation, the 95/95 limit DNBR is 1.30 at system pressures greater than or equal to 1000 psi. For low pressure application (500-1000 psi), the 95/95 limit DNBR is 1.45 (Reference 92).

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4.4.3.2.1 Nuclear Enthalpy Rise Hot-Channel Factor, $F^N_{\Delta H}$

Given the local power density q' (kW/ft) at a point x, y, z in a core with N fuel rods and height H,

$$F^N_{\Delta H} = \frac{\text{hot rod power}}{\text{average rod power}} = \frac{\text{Max} \int_0^H q'(x_0, y_0, z)\,dz}{\frac{1}{N} \sum_{\text{all rods}} \int_0^H q'(x, y, z)\,dz} \quad (4.4\text{-}19)$$ 5

The way in which $F^N_{\Delta H}$ is used in the DNB calculation is important. The location of minimum DNBR depends on the axial profile and the value of DNBR depends on the enthalpy rise to that point. Basically, the maximum value of the rod integral is used to identify the most likely rod for minimum DNBR. An axial power profile is obtained which when normalized to the design value of $F^N_{\Delta H}$, recreates the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with rod average powers which are typical of distributions found in hot assemblies. In this manner worst case axial profiles can be combined with worst case radial distributions for reference DNB calculations. It should be noted again that $F^N_{\Delta H}$ is an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal power shapes throughout the core. The sensitivity of the THINC-IV analysis to radial power shapes is discussed in Reference 52.

For operation at a fraction P of full power, the design $F^N_{\Delta H}$ used is given by:

$$F^N_{\Delta H} = \cancel{1.55}\ 1.62\ (1 + 0.3\,(1-P)) \quad (4.4\text{-}20)$$

The permitted relaxation of $F^N_{\Delta H}$ is included in the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits, thus allowing greater flexibility in the nuclear design. 6

4.4.3.2.2 Axial Heat Flux Distributions

As discussed in Paragraph 4.3.2.2, the axial heat flux distribution can vary as a result of rod motion, power change, or due to spatial xenon transients which may occur in the axial direction. Consequently, it is necessary to measure the axial power imbalance by means of the ex-core nuclear detectors (as discussed in Subparagraph 4.3.2.2.7) and protect the core from excessive axial power imbalance. The Reactor Trip System provides automatic reduction of the trip setpoint in the Overtemperature ΔT channels on excessive axial power imbalance; that is, when an extremely large axial offset corresponds to an axial shape which could lead to a DNBR which is less than that calculated for the reference DNB design axial shape.

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DNB With Return to Nucleate Boiling - Additional DNB tests have been conducted by Westinghouse (Reference 80) in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once in single rod in the bundles for short periods of time. Each time, a reduction in power of approximately 10% was sufficient to reestablish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

4.4.3.9 Energy Release or Rupture of Waterlogged Fuel Elements

A full discussion of waterlogging including energy release is contained in Paragraph 4.4.3.6. It is noted that the resulting energy release is not expected to affect neighboring fuel rods.

4.4.3.10 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockages can occur within the coolant channels of a fuel assembly or external to the reactor core. The effects of fuel assembly blockage within the assembly on fuel rod behavior is more pronounced than external blockages of the same magnitude. In both cases the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, where it occurs in the reactor, and how far along the flow stream the reduction persists are considerations which will influence the fuel rod behavior. The effects of coolant flow blockages in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as computer programs similar to the THINC-IV program. Inspection of the DNB correlation (Paragraph 4.4.2.3 and Reference 44) shows that the predicted DNBR is dependent upon the local values of quality and mass velocity.

The THINC-IV code is capable of predicting the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. In Reference 63, it is shown that for a fuel assembly similar to the Westinghouse design, THINC-IV accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked. Full recovery of the flow was found to occur about 30 inches downstream of the blockage. With the reactor operating at the nominal full power conditions specified in Table 4.4.2-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor violating the [redacted] safety analysis DNBR limit.

From a review of the open literature it is concluded that flow blockage in "open lattice cores" similar to the Westinghouse cores cause flow perturbations which are local to the blockage. For instance, A. Ohtsubo, et al., (Reference 81), show that the mean bundle velocity is approached asymptotically about 4 inches downstream from a flow blockage in a single flow cell. Similar results were also found for 2 and 3 cells completely blocked.

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TABLE 4.4.2-1 (Sheet 1)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	Sequoyah Units 1 & 2 17 x 17 With Densification	Reference Plant 17 x 17 With Densification
Reactor Core Heat Output, MWt11	3411	3411
Reactor Core Heat Output, BTU/hr	$11{,}641.7 \times 10^6$	$11{,}641.7 \times 10^6$
Heat Generated in Fuel, %	97.4	97.4
System Pressure, Nominal, psia	2250	2250
System Pressure, Minimum Steady State, psia	2220	2220
Minimum DNBR at Nominal Conditions		
Typical Flow Channel	~~2.51~~ 2.43	2.04
Thimble (Cold Wall) Flow Channel	~~2.27~~ 2.29	1.71
Minimum DNBR for Design Transients	>1.38	>1.30
DNB Correlation	WRB-1	"L" (W-3 with modified spacer (factor)
Coolant Flow		
Total Thermal Flow Rate, lb/hr	138.0×10^6	132.7×10^6
Effective Flow Rate for Heat Transfer, lb/hr	127.7×10^6	126.7×10^6
Effective Flow Area for Heat Transfer, ft^2	51.1(STD), 51.3 (V-5H)	51.1
Average Velocity Along Fuel Rods, ft/sec	15.6(STD), 15.5 (V5H)	15.7
Average Mass Velocity, lb/hr-ft^2	2.50×10^6 (STD) 2.49×10^6(V5H)	2.48×10^6
Coolant Temperature		
Nominal Inlet, °F	546.7	552.5
Average Rise in Vessel, °F	63.1	64.2
Average Rise in Core, °F	67.6	66.9
Average in Core, °F	582.2	585.9
Average in Vessel, °F	578.2	584.7

TABLE 4.4.2-1 (Sheet 2)
(Continued)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	Sequoyah Units 1 & 2 17 x 17 With Densification	Reference Plant 17 x 17 With Densification
Active Heat Transfer, Surface Area, Ft^2	59,700	59,700
Average Heat Flux, $BTU/hr\text{-}ft^2$	189,800	189,800
Maximum Heat Flux, for normal operation $BTU/hr\text{-}ft^2$	440,300[a]	440,300[a]
Average Thermal Output, kW/ft	5.44	5.44
Maximum Thermal Output, for normal operation kW/ft	~~12.8[a]~~ 13.0	12.8[a] (d)
Peak Linear Power for Determination of protection setpoints, kW/ft	21.1[c]	18.0[c]
Pressure Drop[b]		
Across Core, psi	23.4 ± 2.3	25.7 ± 2.6
Across Vessel, including nozzle psi	46.86 ± 4.6	45.1 ± 4.5

(a) This limit is associated with the value of F_Q = ~~2.32~~ 2.40
(b) Based on best estimate reactor flow rate as discussed in Section 5.1.
(c) See Subparagraph 4.3.2.2.6.
(d) This limit is associated with the value of F_Q = 2.32

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10.2 Tech Spec Mark-Ups

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 WRB-1 correlation and the W-3 correlation for conditions outside the range of WRB-1 correlation. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. R142

Design DNBR Limit.

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 correlation in this application). 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 limit. R142

The Design

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, specified in the Core Operating Limit Report (COLR) 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: R159

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}[1+ PF_{\Delta H} (1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, R159

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

Design DNBR

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis ~~(1.38)~~ and the ~~WRB-1 correlation~~ limit ~~(1.17)~~ to completely offset the rod bow penalty. R142

The applicable value of rod bow penalty is referenced in the FSAR. R15

Margin in excess of the rod bow penalty is available for plant design flexibility. R142

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor, W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function is specified in the COLR. R15

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 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.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R142

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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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 WRB-1 correlation and the W-3 correlation for conditions outside the range of WRB-1 correlation. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. R130

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 Design DNBR Limit. ~~DNBR limit of the DNB correlation being used (the WRB-1 or W-3 correlation in this application).~~ ~~The correlation~~ The Design 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 limit. R130

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. R104 R130

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, specified in the Core Operating Limit Report (COLR) 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: R146 R21

$$F^N_{\Delta H} = F^{RTP}_{\Delta H}[1+ PF_{\Delta H}\ (1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

$F^{RTP}_{\Delta H}$ = the $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and R146

$PF_{\Delta H}$ = the power factor multiplier for $F^N_{\Delta H}$ specified in the COLR.

Design DNBR

Fuel rod bowing reduces the value of DNB ratio. Margin has been retained between the DNBR value used in the safety analysis ~~(1.38)~~ and the ~~WRB-1 correlation~~ limit ~~(1.17)~~ to completely offset the rod bow penalty. R130

The applicable value of rod bow penalty is referenced in the FSAR. R146

Margin in excess of the rod bow penalty is available for plant design flexibility. R130

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor W(z), to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function is specified in the COLR. R21 R146

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. R2

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.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the safety analysis DNBR limit throughout each analyzed transient. R130

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. R21

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10.3 COLR Markups

2.2 Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)
[3/4.1.3.5]

2.2.1 The shutdown rods shall be withdrawn to a position as defined below:

Cycle Burnup (MWD/MTU)	Steps Withdrawn
≤ 2,000	≥ 226 to ≤ 231
> 2,000 to < 14,000	≥ 222 to ≤ 231
≥ 14,000	≥ 226 to ≤ 231

2.3 Control Rod Insertion Limit (Specification 3/4.1.3.6)
[3/4.1.3.6]

2.3.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.4 Axial Flux Difference (Specification 3/4.2.1)
[3/4.2.1]

2.4.1 The axial flux difference (AFD) limits are provided in Figure 2.

2.5 Heat Flux Hot Channel Factor - $F_Q(Z)$ (Specification 3/4.2.2)
[3/4.2.2]

$$F_Q(Z) \le \frac{F_Q^{RTP}}{P} * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{F_Q^{RTP}}{0.5} * K(Z) \qquad \text{for } P \le 0.5$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

2.5.1 $F_Q^{RTP} = \boxed{2.32}$ 2.40

2.5.2 K(Z) is provided in Figure 3.

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2.5.3 Note that the W(Z) values required by TS SR 4.2.2.2 are provided in Figures 4 through 7. This information is sufficient to determine W(Z) versus core height for all cycle burnups through the use of three point interpolation.

2.6 <u>Nuclear Enthalpy Rise Hot Channel Factor</u> - $F^N_{\Delta H}$ (Specification 3/4.2.3) [3/4.2.3]

$$F^N_{\Delta H} \leq F^{RTP}_{\Delta H} * (1 + PF_{\Delta H} * [1-P])$$

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

2.6.1 $F^{RTP}_{\Delta H} = \boxed{\cancel{1.55}}$ 1.62

2.6.2 $PF_{\Delta H} = 0.3$

2.2 <u>Shutdown Rod Insertion Limit</u> (Specification 3/4.1.3.5)
[3/4.1.3.5]

2.2.1 The shutdown rods shall be withdrawn to a position greater than or equal to 225 steps withdrawn.

2.3 <u>Control Rod Insertion Limit</u> (Specification 3/4.1.3.6)
[3/4.1.3.6]

2.3.1 The control rod banks shall be limited in physical insertion as shown in Figure 1.

2.4 <u>Axial Flux Difference</u> (Specification 3/4.2.1)
[3/4.2.1]

2.4.1 The axial flux difference (AFD) limits are provided in Figure 2.

2.5 <u>Heat Flux Hot Channel Factor</u> - $F_Q(Z)$ (Specification 3/4.2.2)
[3/4.2.2]

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} * K(Z) \qquad \text{for } P > 0.5$$

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

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

2.5.1 $F_Q^{RTP} = \boxed{\cancel{2.33}} \leftarrow 2.40$

2.5.2 K(Z) is provided in Figure 3.

2.5.3 Note that the W(Z) values required by TS SR 4.2.2.2 are provided in Figures 4 through 6. This information is sufficient to determine W(Z) versus core height for all cycle burnups through the use of three point interpolation.

2.6 Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$ (Specification 3/4.2.3) [3/4.2.3]

$$F^N_{\Delta H} \le F^{RTP}_{\Delta H} * (1 + PF_{\Delta H} * [1-P])$$

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

2.6.1 $F^{RTP}_{\Delta H} = \boxed{\cancel{1.55}}$ 1.62

2.6.2 $PF_{\Delta H} = 0.3$

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