1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.1 Safety Limits - Reactor Core (Continued)

would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.18. A DNBR of 1.18 corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.(1)

The curves of Figure 1-1 represent the loci of points of reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure, and cold leg temperature for which the DNBR is 1.18. The area of safe operation is below these lines.

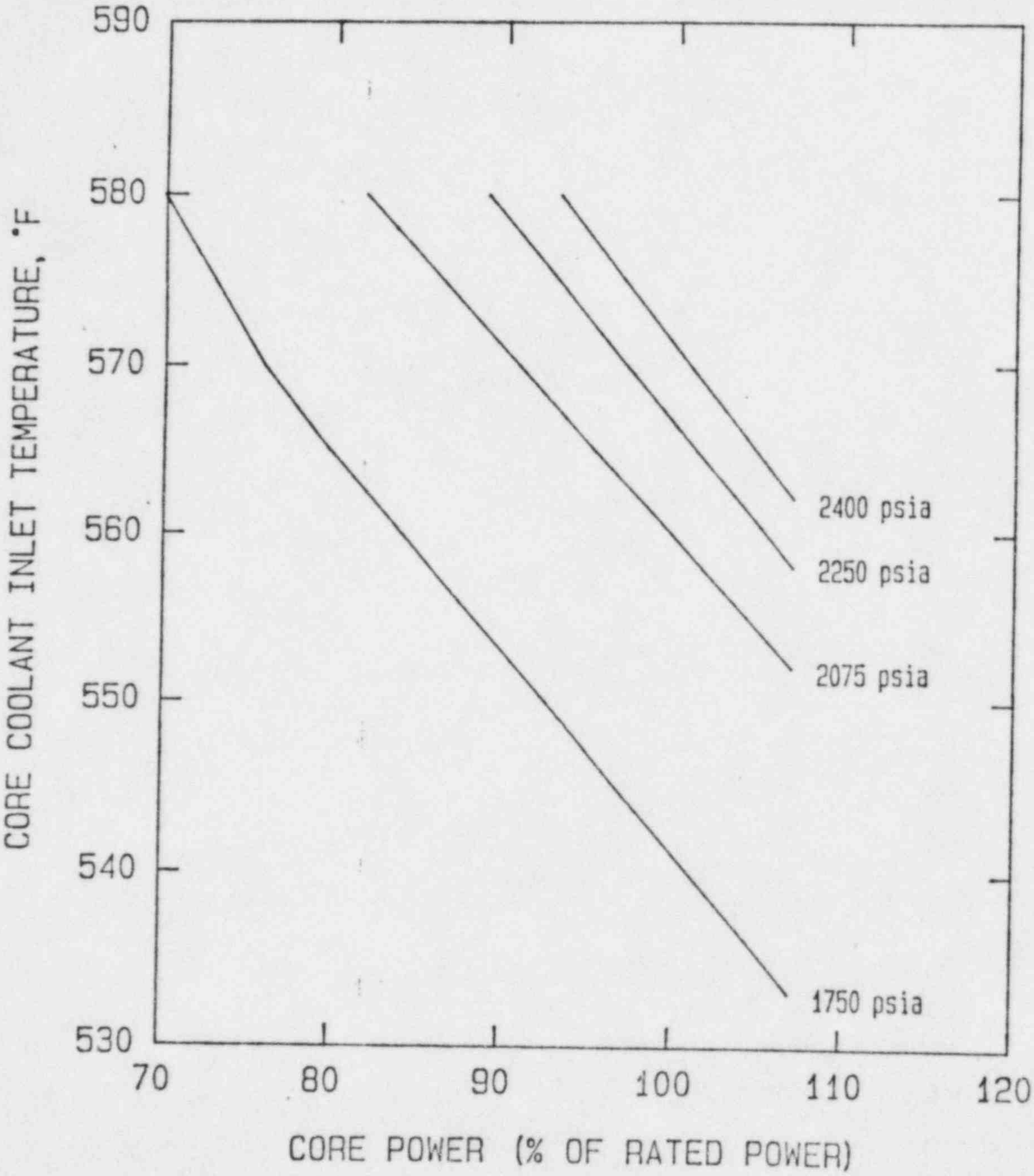
The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2-10 and axial shapes within the axial power distribution trip limits in Figure 1-2 and a total unrodded planar radial peak of 1.85. The LSSS in Figure 1-3 is based on the assumption that the unrodded integrated total radial peak (F_R^T) is 1.80. This peaking factor is slightly higher (more conservative) than the maximum predicted unrodded total radial peak during core life, excluding measurement uncertainty.

Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model tests.(2) The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figure 1-1. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than 1.18.(3)

References

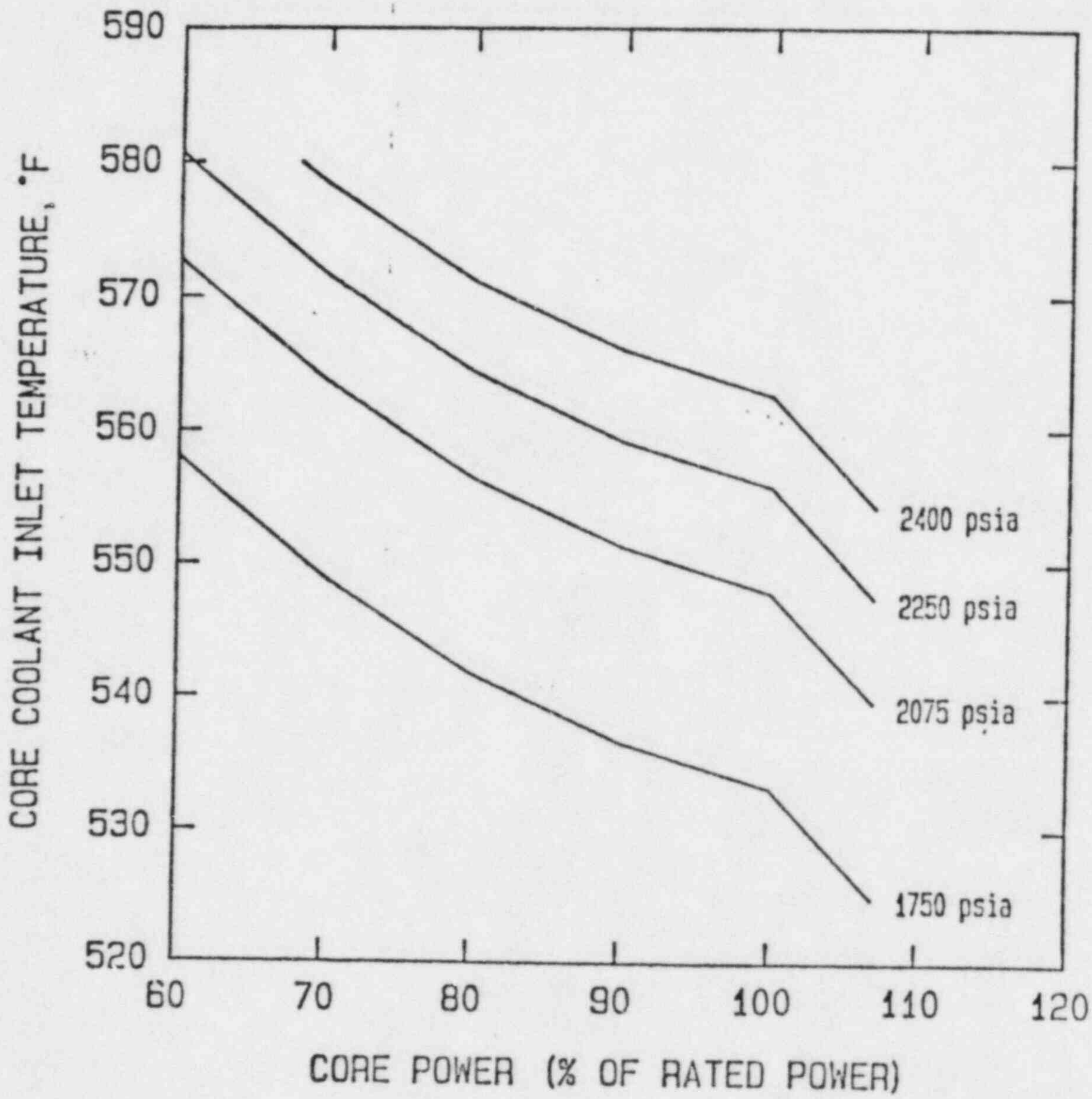
(1) USAR, Section 3.6.7
(2) USAR, Section 1.4.6
(3) USAR, Section 3.6.2

ATTACHMENT A



Thermal Margin/Low Pressure Safety Limits 4 Pump Operation	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 1-1

Amendment No. ~~47~~, ~~70~~, 77



$P_{VAR} = 22\ PF(B)\ A1(Y)\ B + 22.1 T_{IN} - 12674$

$PF(B) = 1.0 \quad B \geq 100\%$

$= -.008\ B + 1.8 \quad 50\% < B < 100\%$

$= 1.4 \quad B \leq 50\%$

$A1(Y) = -.5Y_I + 1.125 \quad Y_I \leq .25$

$= .5Y_I + .875 \quad Y_I > .25$

Thermal Margin/ Low Pressure LSSS 4 Pump Operation	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 1-3

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS
1.3 Limiting Safety System Settings, Reactor Protective System (Continued)

During reactor operation at power levels below 19.1% rated power, a reactor trip will occur in the event of a reactivity excursion that results in a power increase up to the lower fixed set point of the VHPT circuit of 19.1% of rated power.[3] During normal power increases below 19.1% reactor trip would be initiated at 19.1% of rated power unless the set point is manually adjusted.

(2) Low Reactor Coolant Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.

Flow in each of the four coolant loops is determined from a measurement of pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is measured by summing the loop pressure drops across the steam generators and correlating this pressure sum with the pump calibration flow curves.

The percent of normal core flow as follows:[6]

4 Pumps	100%

During four-pump operation, the low flow trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors.

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.3 Limiting Safety System Settings, Reactor Protective System (Continued)

(3) High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the reactor and steam system safety valves to prevent reactor coolant system overpressure (Specification 2.1.6). In the event of loss of load without reactor trip, the temperature and pressure of the reactor coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is 100 psi below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.[1]

(4) Thermal Margin/Low Pressure Trip - The thermal margin/low pressure trip is provided to prevent operation when the DNBR is less than 1.18, including allowance for measurement error. The thermal and hydraulic limits shown on Figure 1-3 define the limiting values of reactor coolant pressure, reactor inlet temperature, axial shape index, and reactor power level which ensure that the thermal criteria[8] are not exceeded. The low set point of a 1750 psia trips the reactor in the unlikely event of a loss-of-coolant accident. The thermal margin/low pressure trip set points shall be set according to the formula given on Figure 1-3. The variables in the formula are defined as:

B = High auctioneered thermal (ΔT) or nuclear power in % of rated power.
T_{IN} = Core inlet temperature, °F.
P_{VAR} = Reactor pressure, psia.
Y_I = Axial Shape Index, asiu

(7) Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down simultaneously with the initiation of the safety injection system. The setting of this trip is identical to that of the containment high pressure signal which indicates safety injection system operation.

(8) Axial Power Distribution - The axial power trip is provided to ensure that excessive axial peaking will not cause fuel damage. The Axial Shape Index is determined from the axially split excore detectors. The set point functions, shown in Figure 1-2 ensure that neither a DNBR of less than 1.18 nor a maximum linear heat rate of more than 21 kW/ft (deposited in the fuel) will exist as a consequence of axial power maldistributions. Allowances have been made for instrumentation inaccuracies and uncertainties associated with the excore symmetric offset - incore axial peaking relationship.

(9) Steam Generator Differential Pressure - The Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF) utilizes a trip on steam generator differential pressure to ensure that neither a DNBR of less than 1.18 nor a peak linear heat rate of more than 21 kW/ft occurs as a result of the loss of load to one steam generator.

(10) Physics Testing at Low Power - During physics testing at power levels less than 10^{-1}% of rated power, the tests may require that the reactor be critical. For these tests only the low reactor coolant flow and thermal margin/low pressure trips may be bypassed below 10^{-1}% of rated power. Written test procedures which are approved by the Plant Review Committee will be in effect during these tests. At reactor power levels less than 10^{-1}% of rated power the low reactor coolant flow and the thermal margin/low pressure trips are not required to prevent fuel element thermal limits being exceeded. Both of these trips are bypassed using the same bypass switch. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown if a steam line break were to occur during the tests.

References

(1) USAR, Section 14.1
(2) USAR, Section 7.2.3.3
(3) USAR, Section 7.2.3.2
(4) USAR, Section 3.6.6
(5) USAR, Section 14.6.2.2, 14.6.4
(6) USAR, Section 14.7
(7) USAR, Section 7.2.3.1
(8) USAR, Section 3.6
(9) USAR, Section 14.10

TABLE 1-1

RPS LIMITING SAFETY SYSTEM SETTINGS

No.	Reactor Trip	Trip Setpoints
1	High Power Level (A) 4-Pump Operation	≤107.0% of Rated Power
2	Low Reactor Coolant Flow (B)(F) 4-Pump Operation	≥95% of 4 Pump Flow
3	Low Steam Generator Water Level	31.2% of Scale (Top of feedwater ring; 4'10" below normal water level)
4	Low Steam Generator Pressure (C)	≥500 psia
5	High Pressurizer Pressure	≤2400 psia
6	Thermal Margin/Low Pressure (B)(F)	1750 psia to 2400 psia (depending on the reactor coolant temperature as shown in Figure 1-3)
7	High Containment Pressure (D)	≤5 psig
8	Axial Power Distribution (E)	(Figure 1-2)
9	Steam Generator Differential Pressure	≤135 psid

(a) A pressurizer steam space of 60% by volume or greater exists, or

(b) The steam generator secondary side temperature is less than 50°F above that of the reactor coolant system cold leg.

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table 2-9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition.*

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

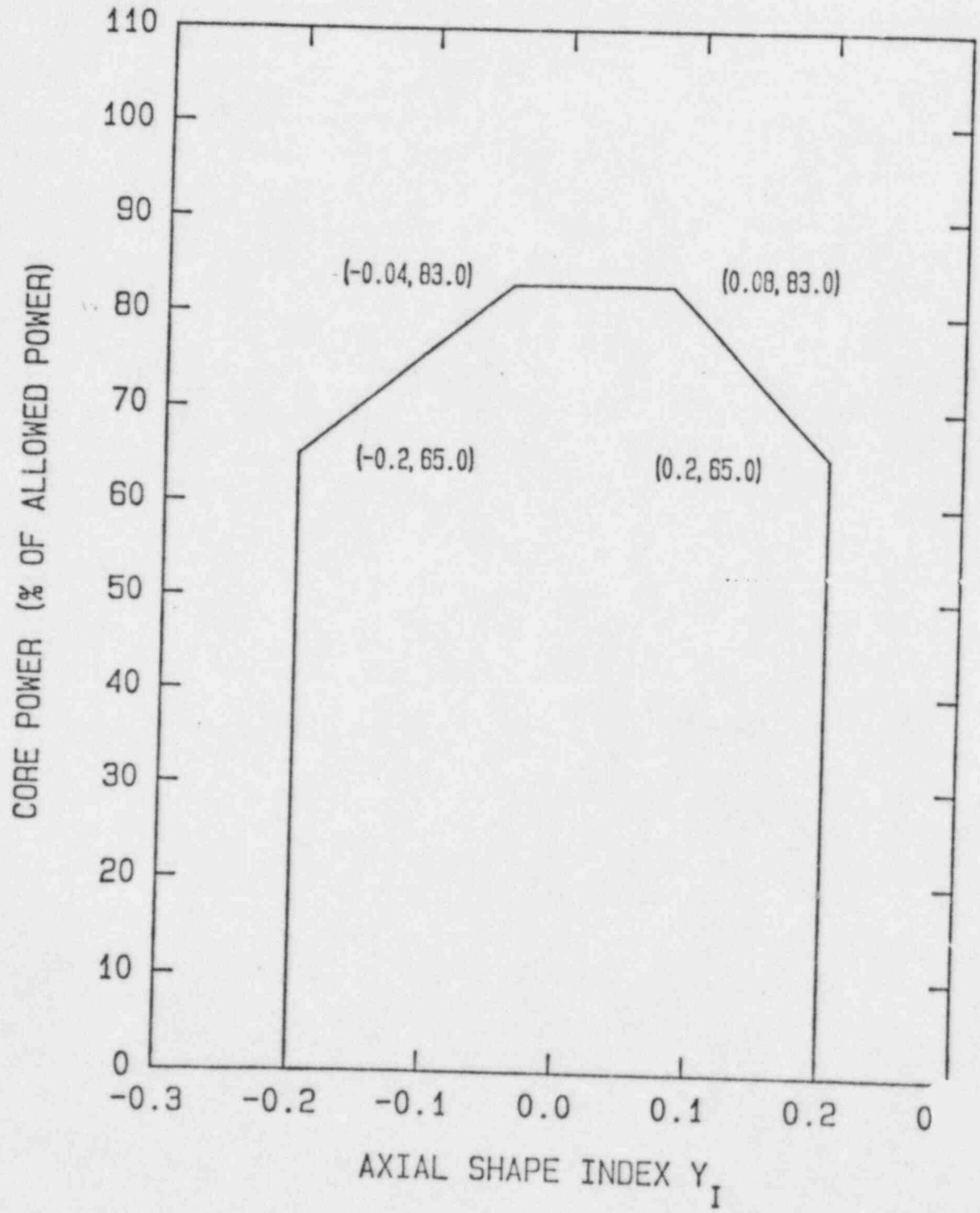
The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above 1.18 during all normal operations and anticipated transients. |

In the hot shutdown mode, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be operable.

In the cold shutdown mode, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two shutdown cooling pumps to be operable.

The requirement that at least one shutdown cooling loop be in operation during refueling ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 210°F as required during the refueling mode, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

*Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplied deenergized.



Limiting Condition for Operation for Excore Monitoring of LHR	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 2-6

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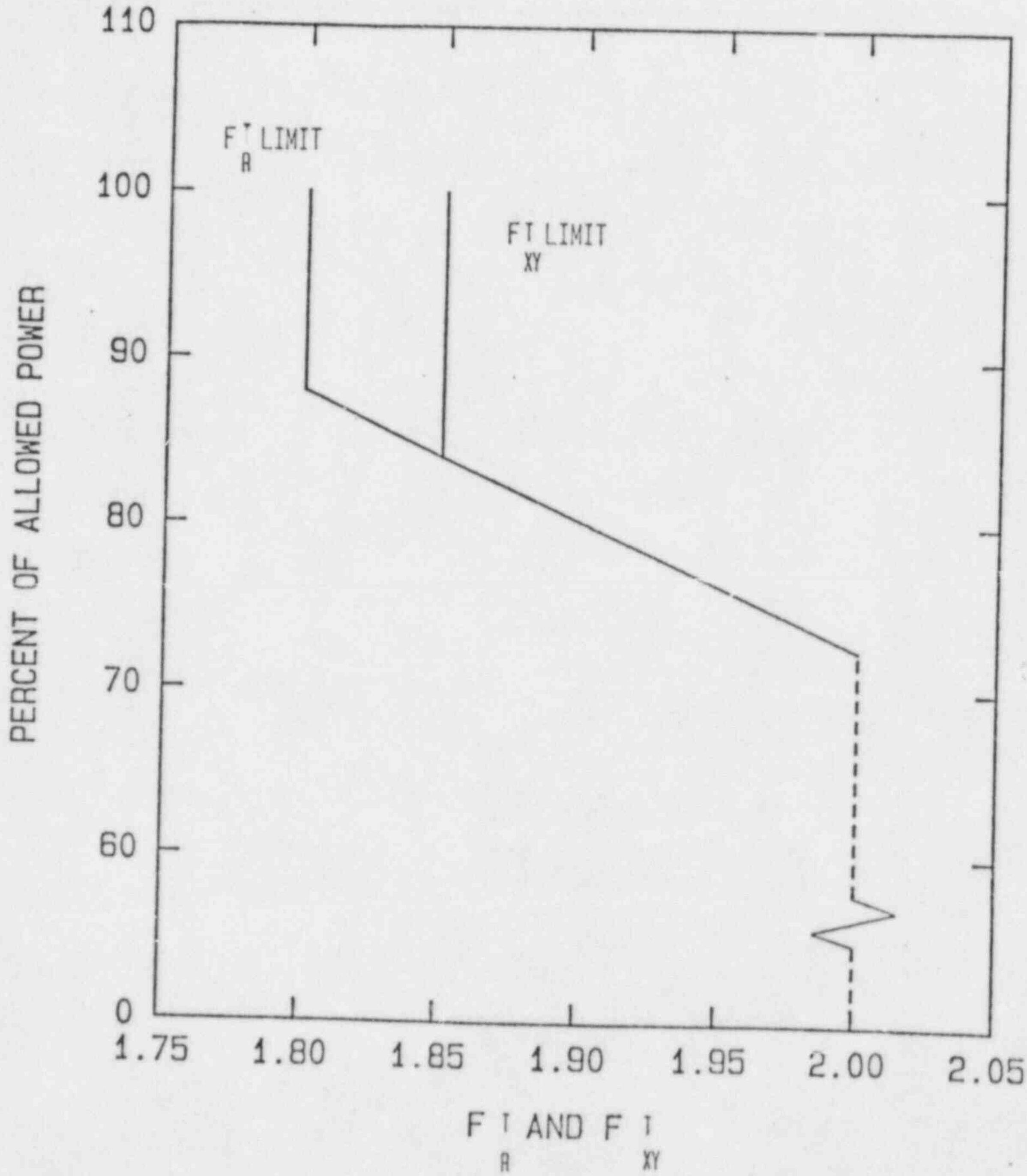


$F_R^T F_{XY}^T$ and Core Power Limitations	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 2-9

Amendment No. 8, 20, 32, 43, 47, 70, 77

(ii) Be in at least hot standby within the next 6 hours.

(2) Total Integrated Radial Peaking Factor

The calculated value of F_R^T defined by $F_R^T = F_R (1+T_q)$ shall be limited to ≤ 1.80. F_R is determined from a power distribution map with no part length CEAs inserted and with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. The azimuthal tilt, T_q, is the measured value of T_q at the time F_R is determined.

With $F_R^T > 1.80$ within 6 hours:

(a) Reduce power to bring power and F_R^T within the limits of Figure 2-9, withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the PLCEAs, or

(b) Be in at least hot standby.

(3) Total Planar Radial Peaking Factor

The calculated value of $F_{xy}T$ defined as $F_{xy}T = F_{xy} (1+T_q)$ shall be limited to ≤ 1.85. F_{xy} shall be determined from a power distribution map with no part length CEAs inserted and with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height in-inclusive and shall exclude regions influenced by grid effects. The azimuthal tilt, T_q, is the measured value of T_q at the time F_{xy} is determined.

With $F_{xy}T > 1.85$ within 6 hours:

(a) Reduce power to bring power and $F_{xy}T$ to within the limits of Figure 2-9, withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the PLCEAs, or

(b) Be in at least hot standby.

(4) Azimuthal Power Tilt (Tq)

When operating above 70% of rated power,

(a) The azimuthal power tilt (Tq) shall not exceed 0.10 whenever Mini CECOR/BASSS is operable, the CEA's are at or above the Long Term Insertion Limit and Mini CECOR/BASSS is being utilized to monitor F_{xy}^T and F_R^T.

(b) The azimuthal power tilt (Tq) shall not exceed 0.03 whenever the provisions of 2.10.4(4)(a) do NOT allow Mini CECOR/BASSS to be utilized to monitor F_{xy}^T and F_R^T. With the indicated azimuthal power tilt determined to be >0.03 but <0.10, correct the power tilt within two hours or determine within the next 6 hours and at least once per subsequent 8 hours, that the total integrated radial peaking factor, F_R^T, is within the limit of Specification 2.10.4(2) and that the total planar radial peaking factor, F_{xy}^T, is within the limit of 2.10.4(3), or reduce power to less than 70% of rated power within 8 hours of confirming Tq >0.03.

(c) With the indicated power tilt determined to be >.10, power operation may proceed up to 2 hours provided F_R^T and F_{xy}^T do not exceed the power limits of Figure 2-9, or be in at least hot standby within 6 hours. Subsequent operation for the purpose of measurement to identify the cause of the tilt is allowable provided the power level is restricted to 20% of the maximum allowable thermal power level for the existing reactor coolant pump combination.

2.0 LIMITING CONDITIONS FOR OPERATION
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(5) DNBR Margin During Power Operation Above 15% of Rated Power

(a) The following DNB related parameters shall be maintained within the limits shown:

(i)	Cold Leg Temperature	≤ 540°F*
(ii)	pressurizer Temperature	≥ 2075 psia*
(iii)	Reactor Coolant Flow	≥ 197,000 gpm**
(iv)	Axial Shape Index Y_I	≤ Figure 2-7***

(b) With any of the above parameters exceeding the limit, restore the parameter to within its limit within 2 hours or reduce power to less than 15% of rated power within the next 8 hours.

Basis

Linear Heat Rate

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System, or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the axial shape index with the operable quadrant symmetric excore neutron flux detectors and verifying that the axial shape index is maintained within the allowable limits of Figure 2-6 as adjusted by Specification 2.10.4(1).(c) for the allowed linear heat rate of Figure 2-5, RC Pump configuration, and F_{xy}^T of Figure 2-9. In conjunction with the use of the excore monitoring system and in establishing the axial shape index limits, the following assumptions are made: (1) the CEA insertion limits of Specification 2.10.2(6) and long term insertion limits of Specification 2.10.2(7) are satisfied, (2) the flux peaking augmentation factors are as shown in Figure 2-8, (3) the total planar radial peaking factor does not exceed the limits of Specification 2.10.4(3).

*Limit not applicable during either a thermal power ramp in excess of 5% of rated thermal power per minute or a thermal power step of greater than 10% of rated thermal power.

**This number is an actual limit and corresponds to an indicated flow rate of 202,500 gpm. All other values in this listing are indicated values and include an allowance for measurement uncertainty (e.g., 540°F, indicated, allows for an actual T_c of 542°F).

***The AXIAL SHAPE INDEX. Core power shall be maintained within the limits established by the Better Axial Shape Selection System (BASSS) for CEA insertions of the lead bank of <65% when BASSS is operable, or within the limits of Figure 2-7.

3.10 Reactor Core Parameters (Continued)

(6) Azimuthal Power Tilt (Tq)

Whenever the core power is above 70% of rated power, the azimuthal power tilt shall be determined to be within its limits by calculating the tilt at least once every day using either:

a. The excore detectors with at least four safety channels operable, or

b. The incore detectors with at least two strings of three rhodium detectors per full core height quadrant operable.

(7) DNB Parameters

a. The cold leg temperature, pressurizer pressure, and axial shape index shall be verified to be within the limits of Section 2.10.4(5) at least once per shift.

b. The reactor vessel coolant total flow rate shall be determined to be within its limit by measurement at least once per month.

JUSTIFICATION, DISCUSSION, AND SIGNIFICANT HAZARDS CONSIDERATIONS FOR CYCLE 10 RELOAD

The Fort Calhoun Technical Specifications are being amended to reflect changes which are a result of the Cycle 10 core reload. Table B-1 presents a summary of the Technical Specification changes and the explanation for the changes. Justification for the changes is contained in the attached Fort Calhoun Cycle 10 Core Reload Evaluation.

Significant Hazards Considerations:

It has been determined, based on the analytical information supplied in the Cycle 10 Core Reload Evaluation, that this amendment request does not involve a significant hazards consideration. This conclusion was derived by applying the Commission's guidance for implementation of 10 CFR 50.92. The Commission provided this guidance concerning the application of these standards through certain examples in the Federal Register, Volume 48, Number 67, Wednesday, April 6, 1983, Rules and Regulations. Example iii of actions involving no significant hazards considerations, on page 14870 of the Federal Register, is quoted below:

> "For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable."

As described in the Cycle 10 Core Reload Evaluation, no fuel assemblies to be loaded into the Cycle 10 core will be of new or different design than those used previously and found to be acceptable to the NRC. No proposed change to the Technical Specifications for Cycle 10 involve acceptance criteria which are significantly different from those previously found acceptable to the NRC. The minimum acceptable DNBR limit has been decreased to 1.18 from 1.22 (using the CE-1 correlation) to be consistent with both the NRC Staff's Safety Evaluation Report approving a 1.15 CE-1 correlation limit (rather than the 1.19 interim value) and also in accordance with the employment of the Statistical Combination of Uncertainties methodology used by the District since Cycle 9. The analytical methods used to demonstrate conformance with the Technical Specifications and regulations are consistent with previous NRC approvals (or documented in report OPPD-NA-8301-P, Rev. 01, submitted to the Commission in June, 1985) and involve no significant changes.

It is concluded that the proposed license amendment does not include significant hazards considerations in that:

1. The probability or consequences of accidents previously evaluated are not increased. All events/accidents not enveloped by Cycle 9 parameters were evaluated and shown to have acceptable consequences, with violation of no safety limits.

2. The Cycle 10 core reload does not create the possibility of a new or different kind of accident from any previously evaluated. The core loading utilizies fuel management techniques which have previously been proven acceptable.

3. The Cycle 10 core reload does not result in a significant reduction in the margin of safety because the Cycle 10 reload evaluation, which uses NRC approved methodologies, demonstrates that the margin of safety is maintained in the revised Technical Specifications limits.

TABLE B-1

Explanation for Cycle 10 Technical Specification Changes

Tech. Spec. Number	Changes	Reasons
1. 1.1, 1.3(2), 1.3(4), 1.3(8), 1.3(9), 2.1.1 Pg. 1-2, 1-7, 1-8, 1-9, 2-2b	Change the minimum DNBR value from 1.22 to 1.18.	The CE-1 limit for 14 x 14 fuel was approved by the NRC at 1.15. The SCU analysis was revised to reflect the approved limit.
2. 1.1 Pg. 1-2	Change the total unrodded planar radial peak from 1.78 to 1.85.	The unrodded planar radial peak is being raised for Cycle 10.
3. 1.1 Pg. 1-2	Change the unrodded integrated radial peaking factor from 1.73 to 1.80.	The unrodded integrated radial peaking factor is being raised for Cycle 10.
4. Figure 1-1	Replace Figure 1-1 with enclosed Figure 1-1.	The TM/LP safety limits have been changed to reflect changes in peaking factors and inclusion of ASI input into the TM/LP calculators.
5. Figure 1-3	Replace Figure 1-3 with enclosed Figure 1-3.	The TM/LP trip LSSS equation has been adjusted to reflect ASI input into the TM/LP calculators. (Item 11, Reload Evaluation).
6. 1.3.(2) - Pg. 1-7	Delete references to less than 4-Pump operation.	The Fort Calhoun License is limited to 4-Pump operation.
7. 1.3(4) Pg. 1-8	Include axial shape index as a thermal-hydraulic parameter.	The modified TM/LP calculators monitor ASI.
8. Table 1-1 No. 1 and 2 Pg. 1-10	Delete references to 3- and 2- Pump Operation.	The Fort Calhoun License is limited to 4-Pump operation.
9. Figure 2-6	Replace Figure 2-6 with Enclosed Figure 2-6.	The LHR excore LCO has been changed to reflect higher radial peaking factors.
10. Figure 2-9	Replace Figure 2-9 with enclosed Figure 2-9.	The $F_{xy}T$ and $F_R T$ limits have been changed to reflect higher peaking factors. The asymetric loading pattern impacted the shape of the $F_{xy}T$ limit line.

11.	2.10.4(2) Pg. 2-57a	Change limited to ≤ 1.73 to limited to ≤ 1.80 and with $F_R T \geq 1.73$ to with $F_R T \geq 1.80$.	The $F_R T$ changes have been made to reflect proposed changes in Tech. Spec. 1.1.
12.	2.10.4(3) Pg. 2-57a	Change limited to ≤ 1.78 to limited to ≤ 1.85 and with $F_R T \geq 1.78$ to with $F_R T \geq 1.85$.	The $F_{xy} T$ changes have been made to reflect proposed changes in Tech. Spec. 1.1.
13.	2.10.4(4) Pg. 2-57b	Mini-CECOR/BASSS Tilt Limits	Adds a change in tilt when Mini-CECOR/BASSS is being used to monitor Technical Specifications.
14.	2.10.4(5)(a)(i) Pg. 2-57c	Change $\leq 545°F$ to $\leq 540°F$.	The Cold Leg Temperature limits have been changed from 545°F to 540°F.
15.	2.10.4(5)(a)(iv) Pg. 2-57c	Add *** footnote to identify incore monitoring with Better Shape Selection System.	The Fort Calhoun Station has added incore DNB LCO monitoring system in addition to the excore LCO.
16.	3.10(6)a Pg. 3-63b	DELETE two symmetric safety channels and two symmetric control channels.	Asymmetric fuel loading pattern for Cycle 10 prevents this combination from properly detecting azimuthal tilts.

FORT CALHOUN

UNIT 1

CYCLE 10

RELOAD EVALUATION

Fort Calhoun
Cycle 10
License Application

CONTENTS

1. INTRODUCTION AND SUMMARY

2. OPERATING HISTORY OF THE REFERENCE CYCLE

3. GENERAL DESCRIPTION

4. FUEL SYSTEM DESIGN

5. NUCLEAR DESIGN

6. THERMAL-HYDRAULIC DESIGN

7. TRANSIENT ANALYSIS

8. ECCS PERFORMANCE ANALYSIS

9. STARTUP TESTING

10. Mini-CECOR/BASSS LCO Monitoring System

11. TM/LP MODIFICATION

12. REFERENCES

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of the design and performance for the operation of Fort Calhoun Station Unit 1 during its tenth fuel cycle at full rated power of 1500 MWt. All planned operating conditions remain the same as those for Cycle 9 with the exception of a reduction in core inlet temperature from 545°F to 540°F. This reduction is being made to minimize the susceptability to steam generator "U" tube stress corrosion cracking. The reduction in inlet temperature was implemented by Operations at the startup of Cycle 9, so this Technical Specification change for Cycle 10 reflects controls already in effect.

The core will consist of 68 presently operating J and K assemblies, 44 fresh Batch L assemblies and 12 G and 9 H assemblies discharged from previous cycles.

The Cycle 10 analysis is based on a Cycle 9 termination point between 12,500 MWD/T to 13,500 MWD/T. In performing analyses of design basis events, determining limiting safety settings and establishing limiting conditions for operation, limiting values of key parameters were chosen to assure that expected Cycle 10 conditions would be enveloped, provided the Cycle 9 termination point falls within the above burnup range. The analysis presented herein will accommodate a Cycle 10 length of up to 13,000 MWD/T.

The evaluation of the reload core characteristics have been conducted with respect to the Fort Calhoun Unit No. 1 Cycle 9 safety analysis described in the 1985 update of the USAR, hereafter referred to as the "reference cycle" in this report unless otherwise noted.

Specific core differences have been accounted for in the present analysis. In all cases, it has been concluded that either the reference cycle analyses envelope the new conditions or the revised analyses presented herein continue to show acceptable results. Where dictated by variations from the previous cycle, proposed modifications to the plant Technical Specifications have been provided.

The Cycle 10 core has been designed to further reduce fluence to critical reactor pressure vessel welds through the use of part-length poison rods inserted in the CEA guide tubes of selected peripheral assemblies and thus minimize the RT_{PTS} shift of these welds. This will delay the time when the reactor vessel welds reach the Pressurized Thermal Shock RT_{PTS} screening criteria contained in 10 CFR 50.61. The Cycle 10 use of a low radial leakage core design has resulted in increased radial peaking factors. The increased peaking factors have been accommodated in the safety analysis through the continued use of the Statistical Combination of Uncertainties methodology (Reference 1), incorporation of axial shape index as an input to the Thermal Margin/Low Pressure Trip Function, and the use of a Mini-CECOR/Better Axial Shape Selection System (BASSS) for incore monitoring of the linear heat rate and DNB LCO's. The Mini-CECOR/BASSS methodology change is an improvement over the previous excore monitoring system and will be installed during the upcoming outage. A full description is supplied in Section 10.

The analysis presented in this report was performed utilizing the methodology documented in the District's reload analysis methodology reports (References 2, 3 and 4). These methodologies were previously transmitted in References 5 and 6.

2.0 OPERATING HISTORY OF THE PREVIOUS CYCLE

Fort Calhoun Station is presently operating in its ninth fuel cycle utilizing Batch G, H, I, J and K fuel assemblies. Fort Calhoun Cycle 9 operation began on July 8, 1984, and reached full power on August 3, 1984. The reactor has operated up to the present time with the core reactivity, power distributions and peaking factors having closely followed the calculated predictions.

It is estimated that Cycle 9 will be terminated on or about September 27, 1985. The Cycle 9 termination point can vary between 12,500 MWD/T and 13,500 MWD/T and still be within the assumptions of the Cycle 10 analyses. As of August 1, 1985, the Cycle 9 burnup had reached 11,311 MWD/T.

3.0 GENERAL DESCRIPTION

The Cycle 10 core will consist of the number and type of assemblies and fuel batches shown in Table 3-1. The primary change to the core in Cycle 10 is the addition of part-length poison rods in selected peripheral assemblies for power reduction to minimize the reactor vessel weld RT_{NDT} shifts. The part-length poison rods are essentially full strength CEA fingers active only in the middle 50% of the core height. Four G assemblies, 21 H assemblies, and 40 I assemblies will be discharged this outage. They will be replaced by 24 fresh unshimmed Batch L assemblies (3.80 w/o enrichment), 12 fresh shimmed Batch L assemblies (3.80 w/o enrichment, 0.01904 gm B_{10}/inch), eight fresh shimmed Batch L assemblies (3.80 w/o enrichment, 0.01190 gm B_{10}/inch), twelve Batch G assemblies (3.03 w/o initial enrichment) discharged from Cycle 8, and nine Batch H assemblies (3.50 w/o initial enrichment) discharged from Cycle 8.

Figure 3-1 shows the fuel management pattern to be employed in Cycle 10 including the location of the part-length poison rods. The location within an assembly of the part-length poison rods is shown in Figure 3-2. Figure 3-3 shows the locations of the poison pins within the lattice of shimmed assemblies and the fuel rod locations in unshimmed assemblies.

Figure 3-4 shows the beginning of Cycle 10 assembly burnup distribution for a Cycle 9 termination burnup of 13,000 MWD/T. The average discharge exposure at the End of Cycle 9 fuel is projected to be 35,239 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-4. Figure 3-5 shows the end of Cycle 10 assembly burnup distribution. The end of Cycle 10 core average exposure is approximately 27,500 MWD/T.

Table 3-1
Fort Calhoun Cycle 10
Core Loading

Assembly Designation	Number of Assemblies	BOC Batch Average Burnup (MWD/T) [EOC 9 = 13,000 MWD/T]	EOC Batch Average Burnup (MWD/T) [EOC 10 = 12,500 MWD/T]	Poison Rods per Assembly	Initial Poison Loading gm B_{10}/inch
G(1)	12	35,038	38,952	0	0
H(1)	9	29,421	33,838	0	0
J	28	24,491	37,242	0	0
J(2)	8	14,407	29,093	0	0
K	12	12,429	27,468	0	0
K/	20	17,643	30,796	8	.0238
L	24	0	13,247	0	0
L/	12	0	14,675	8	.01904
L*	8	0	17,010	8	.01190
TOTAL	133				

(1) Assemblies Discharged From Cycle 8

(2) Assemblies Delivered for Cycle 8, But First Loaded Into Cycle 9

Figure 3-1

FORT CALHOUN CYCLE 10
CORE LOADING PATTERN

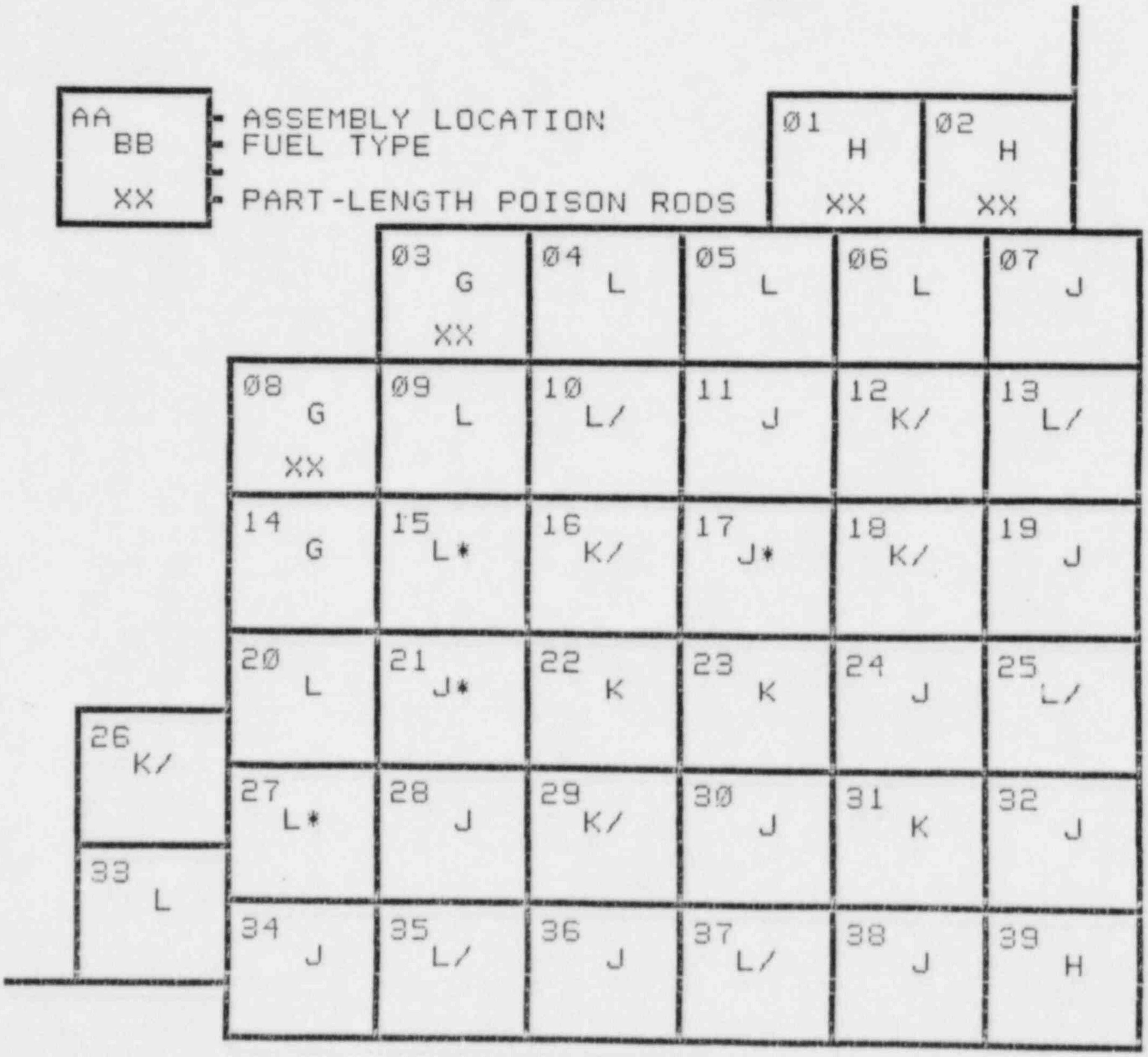
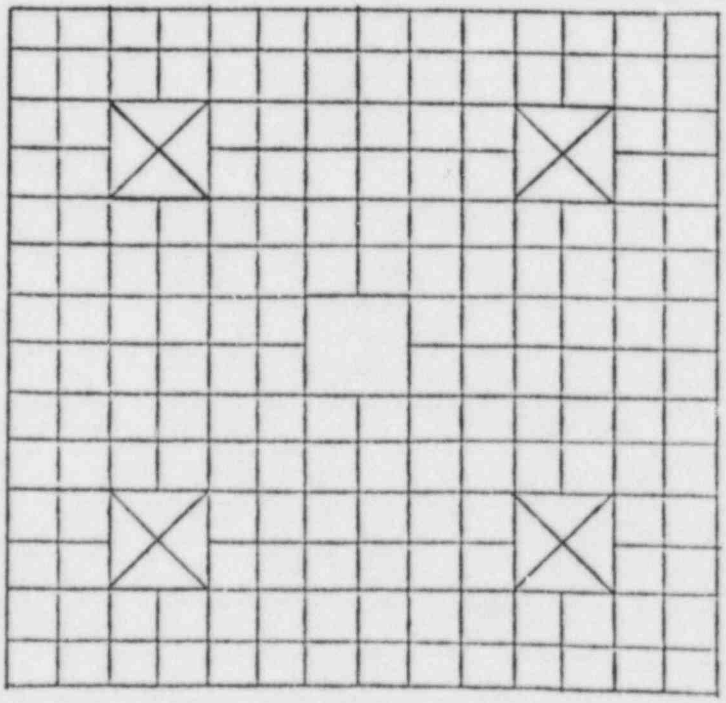


Figure 3-2
FORT CALHOUN CYCLE 10
PART LENGTH POISON ROD
LOCATIONS

4 PART LENGTH POISON ROD ASSEMBLY
ASSEMBLY LOCATIONS 1, 2, 3, & 8 IN QUARTER CORE




FUEL ROD LOCATION

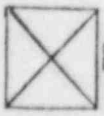

POISON ROD LOCATION

Figure 3-3
FORT CALHOUN CYCLE 10
ASSEMBLY FUEL AND POISON
ROD LOCATIONS

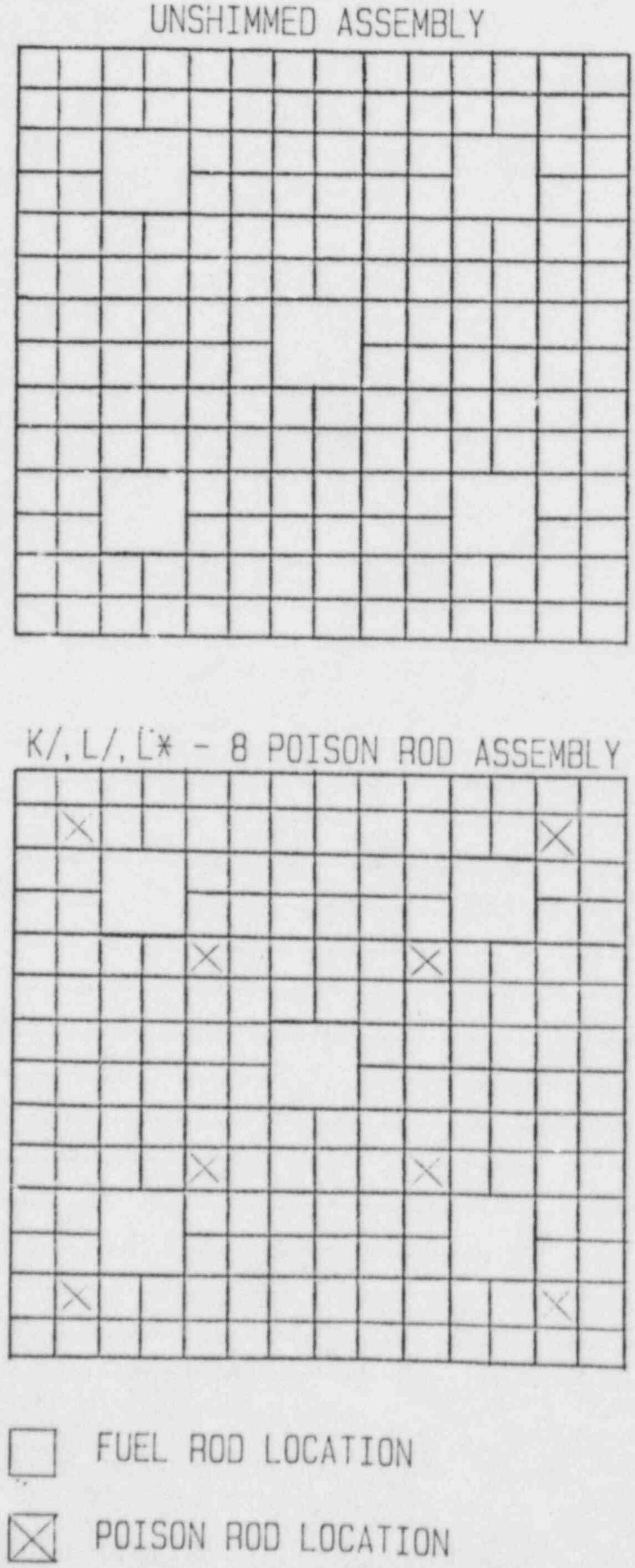


Figure 3-4

FORT CALHOUN CYCLE 10
BOC10 ASSEMBLY AVERAGE EXPOSURE
AND INITIAL ENRICHMENT

- AA - ASSEMBLY LOCATION
- BB - FUEL TYPE
- C.CC - ENRICHMENT, W/O U-235
- DDDDD - ASSY AVG EXPOSURE, MWD/T

					01 H 3.50 29984	02 H 3.50 29929
		03 G 3.03 33082	04 L 3.80 0	05 L 3.80 0	06 L 3.80 0	07 J 3.50 24833
	08 G 3.03 38953	09 L 3.80 0	10 L/ 3.80 0	11 J 3.50 26785	12 K/ 3.50 15430	13 L/ 3.80 0
	14 G 3.03 33077	15 L* 3.80 0	16 K/ 3.50 18412	17 J* 3.50 14385	18 K/ 3.50 17962	19 J 3.50 26253
26 K/ 3.50 18498	20 L 3.80 0	21 J* 3.50 14429	22 K 3.50 11767	23 K 3.50 13779	24 J 3.50 20953	25 L/ 3.80 0
	27 L* 3.80 0	28 J 3.50 26833	29 K/ 3.50 17913	30 J 3.50 20927	31 K 3.50 11742	32 J 3.50 24854
33 L 3.80 0	34 J 3.50 24833	35 L/ 3.80 0	36 J 3.50 26253	37 L/ 3.80 0	38 J 3.50 24854	39 H 3.50 25140

NOTE: EOC 9 CORE AVERAGE
BURNUP = 13000 MWD/T

Figure 3-5

FORT CALHOUN CYCLE 10
ASSEMBLY AVERAGE BURNUP
AT EOC10 (MWD/T)

AA BBBBB	ASSEMBLY LOCATION ASSY AVG EXPOSURE, MWD/T

				Ø1 33386	Ø2 33666	
		Ø3 36643	Ø4 11713	Ø5 13957	Ø6 14432	Ø7 35Ø93
	Ø8 41593	Ø9 13242	1Ø 17427	11 393Ø4	12 3Ø192	13 17684
	14 3862Ø	15 16832	16 33626	17 2911Ø	18 3227Ø	19 39Ø7Ø
	2Ø 13998	21 29Ø75	22 27367	23 28827	24 34613	25 17712
26 25662	27 17187	28 39658	29 32231	3Ø 34572	31 262Ø9	32 37155
33 12141	34 37793	35 17797	36 3893Ø	37 17583	38 37Ø54	39 36336

4.0 FUEL SYSTEMS DESIGN

The mechanical design for the Batch L reload fuel is identical to that of the Batch K fuel described in the 1985 update of the USAR and the Cycle 9 reload submittal. The fuel system design and analysis for ENC fuel in the Fort Calhoun reactor is described in Reference 7.

5.0 NUCLEAR DESIGN

5.1 PHYSICAL CHARACTERISTICS

5.1.1 Fuel Management

The Cycle 10 fuel management uses a low-radial leakage design, with twice, thrice and fourth burned assemblies predominately loaded on the periphery of the core. This low-radial leakage fuel pattern is utilized to minimize the flux to the pressure vessel welds. In addition, selected assemblies located adjacent to critical welds will contain part-length poison rods to further shield the welds. While this type of fuel management results in reduced pressure vessel flux over a standard out-in-in pattern, the radial peaking factors are increased.

As described in Section 3.0, the Cycle 10 loading pattern incorporates 44 fresh Batch L assemblies (12 shimmed L/, 8 shimmed L*, and 24 unshimmed, L) with an enrichment of 3.80 w/o. Twelve 4-cycle burned Batch G assemblies and 9 thrice burned Batch H assemblies, all of which were removed at EOC8, are combined with 32 once burned Batch K assemblies, 8 once burned Batch J assemblies, and 28 twice burned Batch J assemblies to produce a Cycle 10 pattern with a cycle energy of 12,500 ± 500 MWD/T. The Cycle 10 core characteristics have been examined for a Cycle 9 termination between 12,500 MWD/T and 13,500 MWD/T and limiting values established for the safety analysis. The loading pattern is valid for any Cycle 9 endpoint between these values.

Physics characteristics including reactivity coefficients for Cycle 10 are listed in Table 5-1 along with the corresponding values from the reference cycle (Cycle 9). It should be noted that the values of parameters actually employed in safety analyses are different from those displayed in Table 5-1 and are typically chosen to conservatively bound predicted values with accommodation for appropriate uncertainties and allowances.

Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for the end of Cycle 10 Hot Zero Power Steam Line Break accident. The EOC HZP SLB is the most limiting accident of those used in the determining of the required shutdown margin. The Cycle 10 values calculated for minimum scram worth exceed the required Technical Specification limit and thus provide an adequate shutdown margin.

5.1.2 Power Distribution

Figures 5-1 through 5-3 illustrate the all rods out (ARO) planar radial power distributions at BOC10, MOC10

5.0 NUCLEAR DESIGN (Continued)

5.1 PHYSICAL CHARACTERISTICS (Continued)

5.1.2 Power Distribution (Continued)

and EOC10, respectively, that are characteristic of the high burnup end of the Cycle 9 shutdown window. These planar radial power peaks are characteristic of the major portion of the active core length between about 25 and 75 percent of the fuel height. The high burnup end of the Cycle 9 shutdown window tends to increase the power peaking in this axial central region of the core for Cycle 10. The planar radial power distributions for the above region with Bank 4 fully inserted at beginning and end of Cycle 10 are shown in Figures 5-4 and 5-5, respectively, for the high burnup end of the Cycle 9 shutdown window.

The radial power distributions described in this section are calculated data without uncertainties or other allowances. However, the single rod power peaking values do include the increased peaking that is characteristic of fuel rods adjoining the water holes in the fuel assembly lattice. For both DNB and kw/ft safety and setpoint analyses in either rodded or unrodded configurations, the power peaking values actually used are higher than those expected to occur at any time during Cycle 10. These conservative values, which are used in Section 7 of this document, establish the allowable limits for power peaking to be observed during operation.

Figures 3-3 and 3-4 show the integrated assembly burnup values at 0 and 12,500 MWD/T, respectively, based on an EOC9 burnup of 13,000 MWD/T.

The range of allowable axial peaking is defined by the limiting conditions for operation covering the axial shape index (ASI). Within these ASI limits, the necessary DNBR and kw/ft margins are maintained for a wide range of possible axial shapes. The maximum three-dimensional or total peaking factor anticipated in Cycle 10 during normal base load, all rods out operation at full power is 1.99, not including uncertainty allowances.

5.1.3 Safety Related Data

5.1.3.1 Ejected CEA Data

The maximum reactivity worth and planar power peaking factors associated with an ejected CEA event are shown in Table 5-3 for both beginning and end of Cycle 10. These values encompass the worst conditions anticipated during Cycle 10 for any expected

5.0 NUCLEAR DESIGN (Continued)

5.1 PHYSICAL CHARACTERISTICS (Continued)

5.1.3 Safety Related Data (Continued)

5.1.3.1 Ejected CEA Data (Continued)

Cycle 9 termination point. The values shown for Cycle 10 are calculated in accordance with Reference 4. In addition, Table 5-4 lists those values used in the Reference Analysis (Cycle 6) for comparison.

5.1.3.2 Dropped CEA Data

The Cycle 10 safety related data for the dropped CEA analysis were calculated identically to that used in Cycle 9. The data is reported in the dropped CEA analysis.

5.2 ANALYTICAL INPUT TO IN-CORE MEASUREMENTS

In-core detector measurement constants to be used in evaluting the reload cycle power distributions will be calculated in the same manner as for Cycle 9.

5.3 NUCLEAR DESIGN METHODOLOGY

Analyses have been performed in the manner and with the methodologies documented in References 2 and 3.

5.4 UNCERTAINTIES IN MEASURED POWER DISTRIBUTIONS

The power distribution measurement uncertainties which are applied to Cycle 10 are the same as those presented in Reference 2.

TABLE 5-1

FORT CALHOUN CYCLE 10
NOMINAL PHYSICS CHARACTERISTICS

	Units	Reference Cycle*	Cycle 10
Critical Boron Concentration			
Hot Full Power, ARO, Equilibrium Xenon, BOC	PPM	1108	1066
Inverse Boron Worth			
Hot Full Power, BOC	PPM/%$\Delta\rho$	110	110
Hot Full Power, EOC	PPM/%$\Delta\rho$	88	88
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients			
Beginning of Cycle, HZP	$10^{-4}\Delta\rho/°F$	+0.36	+0.25
End of Cycle, HFP	$10^{-4}\Delta\rho/°F$	-2.5	-2.41
Doppler Coefficient			
Hot Zero Power, BOC	$10^{-5}\Delta\rho/°F$	-1.78	-1.79
Hot Full Power, BOC	$10^{-5}\Delta\rho/°F$	-1.40	-1.42
Hot Full Power, EOC	$10^{-5}\Delta\rho/°F$	-1.58	-1.59
Total Delayed Neutron Fraction, β_{eff}			
BOC		0.00640	0.00631
EOC		0.00545	0.00541
Neutron Generation Time, ℓ^*			
BOC	10^{-6} sec	23.1	23.8
EOC	10^{-6} sec	30.7	30.7

* Cycle 9

TABLE 5-2

FORT CALHOUN UNIT 1 CYCLE 10 LIMITING VALUES OF PEACTIVITY WORTHS AND ALLOWANCES FOR HOT ZERO POWER STEAM LINE BREAK, %$\Delta\rho$ END-OF-CYCLE

		Reference Cycle (Cycle 9)	Cycle 10
1.	Worth of all CEA's Inserted	9.95	9.08
2.	Stuck CEA Allowance	1.88	3.06
3.	Worth of all CEA's Less Worth of Most Reactive CEA Stuck Out	8.07	6.02
4.	Power Dependent Insertion Limit CEA Worth	1.11	1.25
5.	Calculated Scram Worth	6.96	4.77
6.	Physics Uncertainty plus Bias	0.69	0.48
7.	Net Available Scram Worth	6.27	4.29
8.	Technical Specification Shutdown Margin	4.00	4.00
9.	Margin in Excess of Technical Specification Shutdown Margin	2.27	0.29

TABLE 5-3

FORT CALHOUN UNIT 1 CYCLE 10
CEA EJECTION DATA

	Cycle 6 Value	BOC10 Value	EOC10 Value
Maximum Radial Power Peaking Factor			
Full Power with Bank 4 inserted; worst CEA ejected	6.00	3.30	3.96
Zero power with Banks 4+3 inserted; worst CEA ejected	13.00*	4.03	4.89
Maximum Ejected CEA Worth (%$\Delta\rho$)			
Full power with Bank 4 inserted; worst CEA ejected	0.30	0.21	0.29
Zero Power with Banks 4+3 inserted; worst CEA ejected	0.90*	0.32	0.44

*Banks 4+3+2 inserted

Figure 5-1

FORT CALHOUN CYCLE 10
ASSEMBLY RELATIVE POWER DENSITY
0 MWD/T, HFP, EQ. XENON

AA	ASSEMBLY LOCATION
B.BBBB	RELATIVE POWER DENSITY

				01 0.2496	02 0.2683	
		03 0.2606	04 0.9304	05 1.1514	06 1.1837	07 0.8052
	08 0.1861	09 1.0420	10 1.3822	11 1.0170	12 1.2031	13 1.4010
	14 0.4084	15 1.3173 X	16 1.2539	17 1.2307	18 1.1749	19 1.0298
	20 1.0952	21 1.1857	22 1.3065	23 1.2730	24 1.1276	25 1.4166
26 0.5113	27 1.3020	28 1.0004	29 1.1608	30 1.1203	31 1.1887	32 0.9917
33 0.9171	34 0.9827	35 1.3534	36 0.9965	37 1.3935	38 0.9791	39 0.8877

X=MAXIMUM 1-PIN PEAK=1.7380

Figure 5-2

FORT CALHOUN CYCLE 10
ASSEMBLY RELATIVE POWER DENSITY
6000 MWD/T, HFP, EQ. XENON

AA — ASSEMBLY LOCATION
B.BBBB — RELATIVE POWER DENSITY

					01 0.2714	02 0.2969
		03 0.2922	04 0.9369	05 1.1086	06 1.1418	07 0.8155
	08 0.2171	09 1.0601	10 1.3935	11 0.9967	12 1.1738	13 1.4050
	14 0.4571	15 1.3555 X	16 1.2146	17 1.1673	18 1.1338	19 1.0167
26 0.5846	20 1.1293	21 1.1761	22 1.2394	23 1.1881	24 1.0792	25 1.4005
33 0.9863	27 1.3981	28 1.0334	29 1.1416	30 1.0806	31 1.1411	32 0.9714
	34 1.0538	35 1.4352	36 1.0134	37 1.3944	38 0.9644	39 0.8821

X=MAXIMUM 1-PIN PEAK=1.7569

Figure 5-3

FORT CALHOUN CYCLE 10
ASSEMBLY RELATIVE POWER DENSITY
13000 MWD/T, HFP, EQ. XENON

AA B.BBBB	
ASSEMBLY LOCATION RELATIVE POWER DENSITY	

					Ø1 Ø.3081	Ø2 Ø.345Ø
		Ø3 Ø.33Ø3	Ø4 Ø.961Ø	Ø5 1.1Ø55	Ø6 1.1485	Ø7 Ø.8562
	Ø8 Ø.25Ø8	Ø9 1.Ø755	1Ø 1.4Ø1Ø	11 Ø.9968	12 1.1598	13 1.4251
	14 Ø.4965	15 1.3383	16 1.1694	17 1.1279	18 1.1139	19 1.Ø261
	2Ø 1.118Ø	21 1.1365	22 1.1779	23 1.1375	24 1.Ø666	25 1.4144
26 Ø.6236	27 1.3825	28 1.Ø274	29 1.1135	3Ø 1.Ø656	31 1.1329	32 Ø.9894
33 Ø.9981	34 1.Ø565	35 1.4422 X	36 1.Ø198	37 1.4Ø78	38 Ø.9837	39 Ø.9142

X=MAXIMUM 1-PIN PEAK=1.6851

Figure 5-4

FORT CALHOUN CYCLE 10
RPD WITH BANK 4 INSERTED
0 MWD/T, HFP, EQ. XENON

AA B.BBBB	
AA	ASSEMBLY LOCATION
B.BBBB	RELATIVE POWER DENSITY

					01 0.2617	02 0.2895
		03 0.1674	04 0.7928	05 1.1378	06 1.2566	07 0.8727
	08 0.1117	09 0.5224	10 1.1345	11 1.0016	12 1.2774	13 1.5204
	14 0.3640	15 1.1109	16 1.1589	17 1.2484	18 1.2460	19 1.1061
26 0.5864	20 1.1684	21 1.2180	22 1.3454	23 1.3309	24 1.1793	25 1.4769
33 1.0757	27 1.4811	28 1.1081	29 1.2569	30 1.1811	31 1.1787	32 0.9176
	34 1.1381	35 1.5360 X	36 1.0967	37 1.4684	38 0.9078	39 0.5625

X=MAXIMUM 1-PIN PEAK=1.8050

CEA BANK 4 LOCATION

Figure 5-5

FORT CALHOUN CYCLE 10
RPD WITH BANK 4 INSERTED
13000 MWD/T, HFP, EQ. XENON

AA	- ASSEMBLY LOCATION
B.BBBB	- RELATIVE POWER DENSITY

					01 0.3244	02 0.3753
		03 0.2035	04 0.7928	05 1.0800	06 1.2222	07 0.9335
	08 0.1482	09 0.5105	10 1.1145	11 0.9694	12 1.2329	13 1.5518
	14 0.4382	15 1.1062	16 1.0603	17 1.1359	18 1.1829	19 1.1055
	20 1.2112	21 1.1754	22 1.2153	23 1.1906	24 1.1155	25 1.4720
26 0.7368	27 1.6150	28 1.1642	29 1.2244	30 1.1301	31 1.1170	32 0.9028
33 1.2072	34 1.2589	35 1.6772 X	36 1.1430	37 1.4932	38 0.9024	39 0.5503

X=MAXIMUM 1-PIN PEAK=1.9530

CEA BANK 4 LOCATION

6.0 THERMAL-HYDRAULIC DESIGN

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 10 at the rated power of 1500 MWt have been performed using the TORC computer code described in Reference 1, the CE-1 critical heat flux correlation described in Reference 2, and the CETOP-D computer code described in Reference 3. This combination was used in the Cycle 8 and 9 Fort Calhoun reload analyses (References 4 and 5) and the reload methodology can be found in Reference 6.

Table 6-1 contains a list of pertinent thermal-hydraulic parameters used in both safety analyses and for generating reactor protective system setpoint information. The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) listed in Table 6-1 have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level (Reference 7) to define the revised design limit on CE-1 minimum DNBR. The MDNBR limit was revised from 1.22 to 1.18 to reflect the NRC approval of a 1.15 limit for 14 x 14 CE type fuel (Reference 8). The interim limit for the CE-1 correlation had been 1.19.

6.2 FUEL ROD BOWING

The fuel rod bow penalty accounts for the adverse impact on MDNBR of random variations in spacing between fuel rods. The penalty at 40,000 MWD/MTU burnup is 0.5% in MDNBR. This penalty was applied to the new design limit in the statistical combination of uncertainties (Reference 7).

TABLE 6-1

Fort Calhoun Unit 1

Thermal-Hydraulic Parameters at Full Power

	Unit	Cycle 10*
Total Heat Output (Core Only)	MWt 10^6 BTU/hr	1500 5119
Fraction of Heat Generated in Fuel Rod		.975
Primary System Pressure		
Nominal	psia	2100
Minimum In Steady State	psia	2075
Maximum In Steady State	psia	2150
Inlet Temperature	°F	540
Total Reactor Coolant Flow	gpm	202,500
(Steady State)	10^6 lbm/hr	76.98
(Through the Core)	10^6 lbm/hr	73.55
Hydraulic Diameter (Nominal Channel)	ft	.044
Average Mass Velocity	10^6 lbm/hr-ft^2	2.26
Core Average Heat Flux (Accounts for Heat Generated in Fuel Rod)	BTU/hr-ft^2	179,722
Total Heat Transfer Surface Area	ft^2	28,485**
Average Core Enthalpy Rise	BTU/lbm	70.1
Average Linear Heat Rate	kw/ft	6.1**
Engineering Heat Flux Factor		1.03***
Engineering Factor on Hot Channel Heat Input		1.03***
Rod Pitch and Bow		1.065***
Fuel Densification Factor (Axial)		1.01***

*Design inlet temperature and nominal primary system pressure were used to calculate these parameters.

**Based on Cycle 10 specific value of 320 shims.

***These factors were combined statistically (Reference 7) with other uncertainty factors at 95/95 confidence/probability level to define a design limit on CE-1 minimum DNBR.

7.0 TRANSIENT ANALYSIS

This section presents the results of the Omaha Public Power District Fort Calhoun Station Unit 1, Cycle 10 Non-LOCA safety analysis at 1500 MWt.

The Design Bases Events (DBEs) considered in the safety analysis are listed in Table 7-1. These events were categorized in the following groups:

1. Anticipated Operational Occurrences (AOOs) for which the intervention of the Reactor Protection System (RPS) is necessary to prevent exceeding acceptable limits.

2. AOOs for which the intervention of the RPS trips and/or initial steady state thermal margin, maintained by Limiting Conditions for Operation (LCO), are necessary to prevent exceeding acceptable limits.

3. Postulated Accidents

The Design Basis Events (DBEs) considered in the Cycle 10 safety analyses are listed in Table 7-1. Core parameters input to the safety analyses for evaluating approaches to DNB and centerline temperature to melt fuel design limits are presented in Table 7-2.

As indicated in Table 7-1, no reanalysis was performed for the DBEs for which key transient input parameters are within the bounds (conservative with respect to) of the reference cycle values (Fort Calhoun Updated Safety Analysis Report including Cycle 9, Reference 1). For these DBEs the results and conclusions quoted in the reference cycle analysis are valid for Cycle 10.

For the events reanalyzed, Table 7-3 shows the reason for the reanalysis, the acceptance criterion to be used in judging the results and a summary of the results obtained. Detailed presentations of the results of the reanalyses are provided in Sections 7.1 through 7.3.

TABLE 7-1

FORT CALHOUN UNIT 1, CYCLE 10
DESIGN BASIS EVENTS CONSIDERED IN THE NON-LOCA SAFETY ANALYSIS

		Analysis Status
7.1	Anticipated Operational Occurrences for which intervention of the RPS is necessary to prevent exceeding acceptable limits:	
7.1.1	Boron Dilution	Reanalyzed
7.1.2	Startup of an Inactive Reactor Coolant Pump[1]	Not Reanalyzed
7.1.3	Loss of Load	Not Reanalyzed
7.1.4	Excess Load	Reanalyzed
7.1.5	Loss of Feedwater Flow	Not Reanalyzed
7.1.6	Excess Heat Removal due to Feedwater Malfunction	Not Reanalyzed
7.1.7	Reactor Coolant System Depressurization	Reanalyzed
7.2	Anticipated Operational Occurrences for which RPS trips and/or sufficient initial steady state thermal margin, maintained by the LCOs, are necessary to prevent exceeding the acceptable limits:	
7.2.1	Sequential CEA Group Withdrawal[2]	Reanalyzed
7.2.2	Loss of Coolant Flow[3]	Reanalyzed
7.2.3	Full Length CEA Drop	Reanalyzed
7.2.4	Part Length CEA Drop[5]	Not Reanalyzed
7.2.5	Transients Resulting from the Malfunction of One Steam Generator[4]	Not Reanalyzed
7.3	Postulated Accidents	
7.3.1	CEA Ejection	Not Reanalyzed
7.3.2	Steam Line Break	Reviewed[6]
7.3.3	Steam Generator Tube Rupture	Not Reanalyzed
7.3.4	Seized Rotor[3]	Reviewed[6]

[1]Technical Specifications preclude this event during operation.

[2]Requires High Power and Variable High Power Trip

[3]Requires Low Flow Trip

[4]Requires trip on high differential steam generator pressure

[5]Bounded by Full Length CEA Drop

[6]Event bounded by reference cycle analysis. A negative 10 CFR 50.59 determination was made for this event.

TABLE 7-2

FORT CALHOUN UNIT 1, CYCLE 10
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Reference Cycle (Cycle 9) Values	Cycle 10 Values
Radial Peaking Factors			
For DNB Margin Analyses ($F_R T$)			
Unrodded Region		1.75*	1.80*
Bank 4 Inserted		1.79*	1.83*
For Planar Radial Component ($F_{xy} T$) of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.78*	1.85*
Bank 4 Inserted		1.93*	2.15*
Maximum Augmentation Factor		1.057	1.057
Moderator Temperature Coefficient	$10^{-4} \Delta\rho/°F$	-2.7 to +0.5	-2.7 to +0.5
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)	$\%\Delta\rho$	-4.0	-4.0

*For the Loss of Coolant Flow and CEA Drop Events, the effects of uncertainties on these parameters were accounted for statistically in the DNBR and CTM calculations. The DNBR analysis utilized the methods discussed in Section 6.1 of this report. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2a, 2b, 2c, 2d.

TABLE 7-2
(Continued)

Safety Parameters	Units	Cycle 9 Values	Cycle 10 Values
Power Level	MWt	1530*	1530*
Maximum Steady State Temperature	°F	547*	542*
Minimum Steady State Pressurizer Pressure	psia	2053*	2053*
Reactor Coolant Flow	gpm	202,500*	202,500*
Negative Axial Shape LCO Extreme Assumed at Full Power (Ex-Cores)	I_p	-0.18	-0.18
Maximum CEA Insertion at Full Power	% Insertion of Bank 4	25	25
Maximum Initial Linear Heat Rate for Transient Other than LOCA	KW/ft	15.22	15.22
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	KW/ft	21.0	21.0
CEA Drop Time to 100% Including Holding Coil Delay	sec	3.1	3.1
Minimum DNBR (CE-1)		1.22*	1.18*

*For the Loss of Coolant Flow and CEA Drop Events, the effects of uncertainties on these parameters were accounted for statistically in the DNBR and CTM calculations. The DNBR analysis utilized the methods discussed in Section 6.1 of this report. The procedures used in the Statistical Combination of Uncertainties (SCU) as they pertain to DNB and CTM limits are detailed in References 2a, 2b, 2c, 2d.

TABLE 7-3

DESIGN BASIS EVENT REANALYZED FOR FORT CALHOUN CYCLE 10

Event	Reason for Reanalysis (changes relative to reference cycle)	Acceptance Criterion	Summary of Results
Boron Dilution	Increased critical boron concentrations from Cycle 9.	Dilution to critical time limits of 30 minutes for refueling and 15 minutes for all other subcritical modes must be met.	Acceptance criteria met. See Section 7.1.1 for details.
Excess Load	Change in TM/LP trip function (P_{var}) trip equation. Reevaluate P_{bias} term.	-	P_{bias} = 35.5 psia which is more limiting (as in Cycle 9) than the RCS Depressurization.
RCS Depressurization	Reevaluate P_{bias} term.	-	P_{bias} = 20.7 psia which is less limiting than that of Excess Load event.
Sequential CEA Group Withdrawal	Increased Tech. Spec. limits on radial peaking factors.	Minimum DNBR greater than 1.18 using CE-1 correlation. Transient PLHGR < 21 kw/ft.	MDNBR = 1.39 PLHGR < 21 kw/ft.

TABLE 7-3
(Continued)

DESIGN BASIS EVENT REANALYZED FOR FORT CALHOUN CYCLE 10

Event	Reason for Reanalysis (changes relative to reference cycle)	Acceptance Criterion	Summary of Results
Loss of Coolant Flow	Increased Tech. Spec. limits on radial peaking factors.	Minimum DNBR greater than 1.18 using CE-1 correlation.	Minimum DNBR = 1.50
Full Length CEA Drop	Increased Tech. Spec. limits on radial peaking factors.	Minimum DNBR greater than 1.18 using CE-1 correlation.	Minimum DNBR = 1.47
Seized Rotor	Increased Tech. Spec. limits on radial peaking factors.	Site boundary dose within 10CFR100 limits, specifically less than 1% failed fuel.	Site boundary dose acceptable. Less than 1% failed fuel.

7.0 TRANSIENT ANALYSIS

7.1 (Continued)

7.1.1 Boron Dilution Event

The Boron Dilution event was reanalyzed for Cycle 10 to determine if sufficient time is available for an operator to identify the cause and to terminate an approach to criticality for all subcritical modes of operation. It was also analyzed to verify corresponding shutdown margin requirements for modes 2 through 5 as they are defined by the Technical Specifications. The event was analyzed using the methods of Reference 3.

Table 7.1.1-1 compares the values of the key transient parameters assumed in each mode of operation for Cycle 10 and the reference cycle (Cycle 9).

Table 7.1.1-2 compares the results of the analysis for Cycle 10 with those for Cycle 9. The key results are the minimum times required to lose prescribed negative reactivity in each operational mode. As seen from Table 7.1.1-2, sufficient time exists for the operator to initiate appropriate action to mitigate the consequences of this event.

TABLE 7.1.1-1

FORT CALHOUN CYCLE 10
KEY PARAMETERS ASSUMED IN THE BORON DILUTION ANALYSIS

Parameter	Cycle 9	Cycle 10
Critical Boron Concentration, PPM (All Rods Out, Zero Xenon)		
Mode		
Hot Standby	1560	1540
Hot Shutdown	1560	1540
Cold Shutdown - Normal RCS Volume	1360	1400
Cold Shutdown - Minimum RCS Volume*	1190	1070
Refueling	1290	1330
Inverse Boron Worth, PPM/%$\Delta\rho$		
Mode		
Hot Standby	-90	-90
Hot Shutdown	-55	-55
Cold Shutdown - Normal RCS Volume	-55	-55
Cold Shutdown - Minimum RCS Volume	-55	-55
Refueling	-55	-55
Minimum Shutdown Margin Assumed, %$\Delta\rho$		
Mode		
Hot Standby	-4.0	-4.0
Hot Shutdown	-4.0	-4.0
Cold Shutdown - Normal RCS Volume	-3.0	-3.0
Cold Shutdown - Minimum RCS Volume	-3.0	-3.0
Refueling	**	**

* Shutdown Groups A and B out, all Regulating Groups inserted except most reactive rod stuck out.

** 1700 ppm initially

TABLE 7.1.1-2

FORT CALHOUN CYCLE 10
RESULTS OF THE BORON DILUTION EVENT

Mode	Time to Lose Prescribed Shutdown Margin (Min)		Criterion For Minimum Time to Lose Prescribed Shutdown Margin (Min)
	Cycle 9	Cycle 10	
Hot Standby	92.7	93.8	15
Hot Shutdown	45.2	45.8	15
Cold Shutdown - Normal RCS Volume	39.3	38.2	15
Cold Shutdown - Minimum RCS Volume	16.4	18.2	15
Refueling	35.0	31.2	30

7.0 TRANSIENT ANALYSIS (Continued)

7.1 (Continued)

7.1.4 Excess Load Event

The Excess Load event was reanalyzed for Cycle 10 to determine the pressure bias term for the TM/LP trip setpoint.

The Excess Load event is one of the DBEs analyzed to determine the maximum pressure bias term input to the TM/LP trip. The methodology used for Cycle 10 is described in References 3 and 4. The pressure bias term accounts for margin degradation attributable to measurement and trip system processing delay times. Changes in core power, inlet temperature and RCS pressure during the transient are monitored by the TM/LP trip directly. Consequently, with TM/LP trip setpoints and the bias term determined in this analysis, adequate protection will be provided for the Excess Load event to prevent the acceptable DNBR design limit from being exceeded.

The assumptions used in the analysis to maximize the pressure bias term are consistent with those described in Reference 3 and include:

(1) The event is assumed to occur due to the inadvertent opening of the steam dump and bypass valves due to a failure of the steam dump control interlock. This results in a decreasing core inlet temperature which produces an increase in core power due to the assumption of the most negative moderator and fuel temperature coefficients during the cycle.

(2) The pressurizer control systems are assumed to be inoperative thus maximizing the rate of pressure decrease and the rate of approach to the DNBR limit.

(3) The initial axial power shape and the corresponding scram worth versus insertion used in the analysis is a bottom peaked shape. This power distribution maximizes the time required to terminate the decrease in DNBR following a trip.

The analysis of this event shows that a pressure bias term of 35.5 psia is required. This is greater than that input from the RCS Depressurization event, the other event for which a pressure bias term is calculated. Hence, the use of the pressure bias factor determined by this event in conjunction with the TM/LP trip, will prevent exceeding the DNBR design limit for AOO's which require TM/LP trip protection.

7.0 TRANSIENT ANALYSIS (Continued)

7.1 (Continued)

7.1.7 RCS Depressurization Event

The RCS Depressurization event was reanalyzed for Cycle 10 to determine the pressure bias term for the TM/LP setpoint.

The RCS Depressurization event is one of the DBEs analyzed to determine the maximum pressure bias term input to the TM/LP trip. The methodology used for Cycle 10 is the same as that used for Cycle 9 and is described in References 3 and 4. The pressure bias term accounts for margin degradation attributable to measurement and trip system processing delay times. Changes in core power, inlet temperature, and RCS pressure during the transient are monitored by the TM/LP trip directly. Consequently, with TM/LP trip setpoints and the bias term determined in this analysis, adequate protection will be provided for the RCS Depressurization event to prevent the acceptable DNBR design limit from being exceeded.

The assumptions used to maximize the rate of pressure decrease and, consequently, the fastest approach to DNBR limits are consistent with those described in Reference 3 and include:

(1) The event is assumed to occur due to an inadvertent opening of both pressurizer relief valves while operating at rated thermal power. This results in a rapid drop in the RCS pressure and, consequently, a rapid decrease in DNBR.

(2) The charging pumps, the pressurizer heaters, and the pressurizer backup heaters are assumed to be inoperable. This maximizes the rate of pressure decrease and, consequently, maximizes the rate of approach to the DNBR limit.

(3) The initial axial power shape and the corresponding scram worth versus insertion used in the analysis is a bottom peaked shape. This power distribution maximizes the time required to terminate the decrease in DNBR following a trip.

The analysis of this event shows that a pressure bias term of 20.7 psia is required. This is less than that input from the Excess Load event, the other event for which a pressure bias term is calculated. Hence, the use of the Excess Load pressure bias term in conjunction with the TM/LP trip, will provide adequate DNBR margin for this and other AOO's which require TM/LP trip protection.

7.0 TRANSIENT ANALYSIS (Continued)

7.2 (Continued)

7.2.1 CEA Withdrawal Event

The CEA Withdrawal event was reanalyzed for Cycle 10 to determine the initial margins that must be maintained by the LCOs such that the DNBR and fuel centerline to melt (CTM) design limits will not be exceeded in conjunction with the RPS (Variable High Power, High Pressurizer Pressure, or Axial Power Distribution Trips).

The methodology contained in Reference 3 was employed in analyzing the CEA Withdrawal event. This event is classified as one for which the acceptable DNBR and centerline to melt limits are not violated by virtue of maintenance of sufficient initial steady state thermal margin provided by the DNBR and Linear Heat Rate (LHR) related Limiting Conditions for Operations (LCOs). Depending on the initial conditions and the reactivity insertion rate associated with the CEA Withdrawal, the Variable High Power Trip and High Pressurizer Pressure Trip in conjunction with the initial steady state LCOs, prevents DNBR limits from being exceeded. An approach to the CTM limit is terminated by either the Variable High Power Trip or the Axial Power Distribution Trip. The analysis took credit for only the Variable High Power Trip (utilizing input from the excore detectors) and High Pressurizer Pressure Trip in both the determination of the required initial overpower margin for DNBR using CETOP/CE-1 and the peak linear heat generation rate for the CTM SAFDL.

For the HFP CEAW DNBR analysis, an MTC identical to that utilized in Reference 5 and the gap thermal conductivity consistent with the assumption of Reference 3 were used in conjunction with a variable reactivity insertion rate. The range of reactivity insertion rates examined is given in Table 7.2.1-1.

For the HFP CEAW CTM analysis, the maximum reactivity insertion rate and the most positive MTC were assumed.

The zero power case was analyzed to demonstrate that acceptable DNBR and centerline melt limits are not exceeded. For the zero power case, a reactor trip, initiated by the Variable High Power Trip at 29.1% (19.1% plus 10% uncertainty) of rated thermal power, was assumed in the analysis.

The zero power case initiated at the limiting conditions of operation results in a minimum CE-1 DNBR of 7.48. Also, the analysis shows that the fuel-centerline temperatures are well below those corresponding to the acceptable fuel centerline melt limit. The sequence of events for the zero power case is presented

7.0 TRANSIENT ANALYSIS (Continued)

7.2 (Continued)

7.2.1 CEA Withdrawal Event (Continued)

in Table 7.2.1-2. Figures 7.2.1-1 to 7.2.1-4 present the transient behavior of core power, core average heat flux, RCS coolant temperatures, and the RCS pressure for the zero power case.

Protection against exceeding the DNBR limit for a CEA Withdrawal at full power is provided by the initial steady state thermal margin which is maintained by adhering to the Technical Specification LCOs on DNBR margin and by the response of the RPS which provides an automatic reactor trip on high power level. The minimum DNBR for this event, when initiated from the extremes of the LCOs, is 1.39.

The HFP maximum reactivity insertion rate analysis shows that the fuel centerline temperatures are well below those corresponding to the acceptable CTM limit. The sequence of events for the full power case with the maximum reactivity insertion rate is presented in Table 7.2.1-3. Figures 7.2.1-5 to 7.2.1-8 present the transient behavior of core power, core average heat flux, RCS coolant temperatures, and the RCS pressure for this full power case.

It may be concluded that the CEA withdrawal event when initiated from the Tech. Spec. LCOs (in conjunction with the Variable High Power Trip if required) will not lead to a DNBR or fuel temperature which exceed the DNBR and centerline to melt design limits.

TABLE 7.2.1-1

FORT CALHOUN CYCLE 10
KEY PARAMETERS ASSUMED IN THE CEA WITHDRAWAL ANALYSIS

Parameter	Units	HZP	HFP
Initial Core Power Level	MWt	1	102% of 1500*
Core Inlet Coolant Temperature	°F	532*	542*
Pressurizer Pressure	psia	2053*	2053*
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	+0.5	+0.5**
Doppler Coefficient Multiplier		0.85	0.85
CEA Worth at Trip	$10^{-2}\Delta\rho$	-4.65	-5.70
Reactivity Insertion Rate Range	$x10^{-4}\Delta\rho/sec$	0 to 1.0	0 to 1.0
CEA Group Withdrawal Rate	in/min	46	46
Holding Coil Delay Time	sec	0.5	0.5

*The effects of uncertainties on these parameters were accounted for deterministically and the DNBR calculations used the methods discussed in Section 6.1 of this document and detailed in References 2a, 2b, 2c and 2d.

**DNBR analysis assumes MTC consistent with Reference 5.

TABLE 7.2.1-2

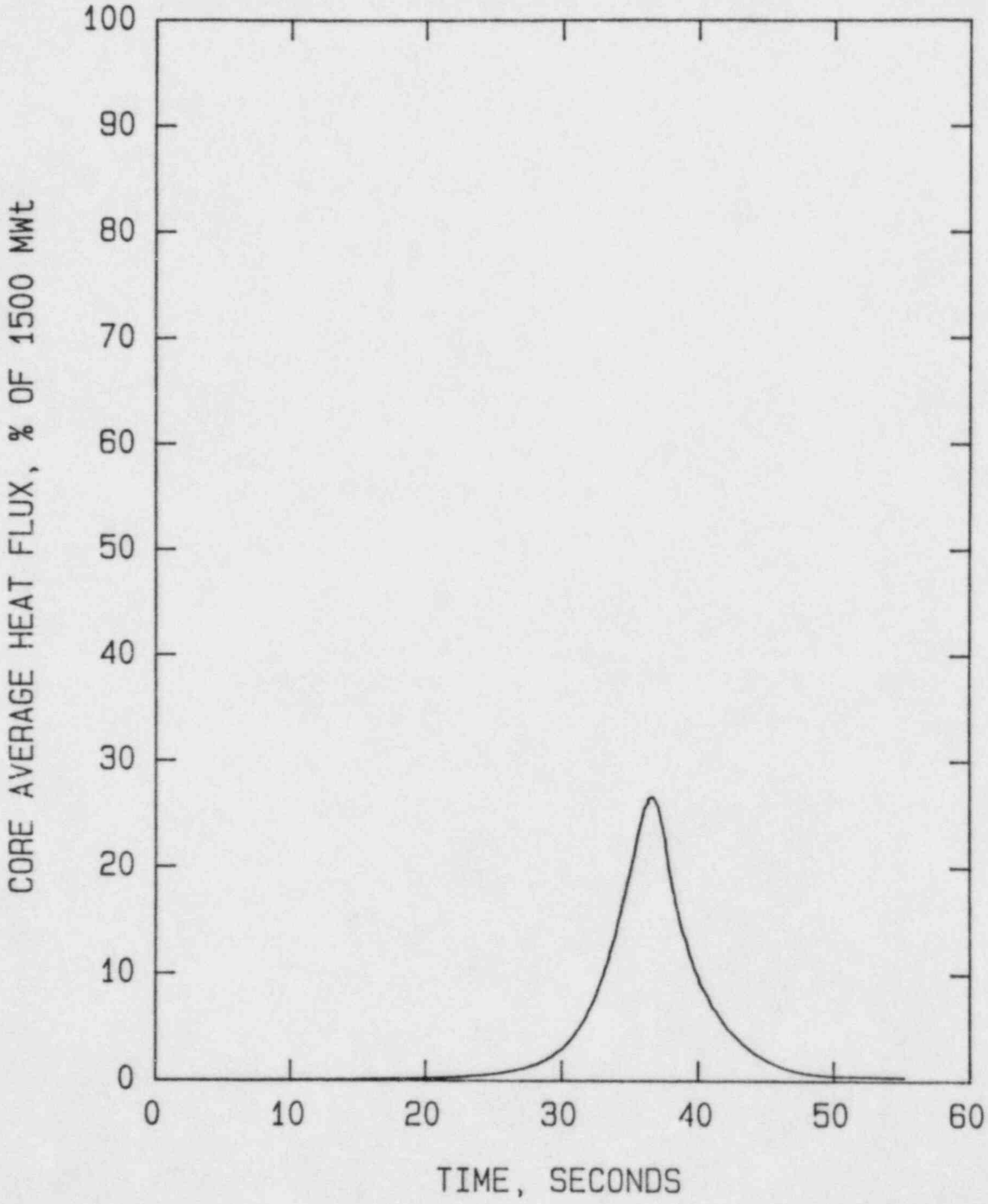
FORT CALHOUN CYCLE 10
SEQUENCE OF EVENTS FOR
CEA WITHDRAWAL FROM ZERO POWER

Time (sec)	Event	Setpoint or Value
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	---
34.4	Variable High Power Trip Signal Generated	29.1% of 1500 MWt
34.8	Reactor Trip Breakers Open	---
35.3	CEAs Begin to Drop Into Core	---
35.8	Maximum Core Power	39.3% of 1500 MWt
36.7	Maximum Heat Flux	26.7% of 1500 MWt
36.7	Minimum CE-1 DNBR	7.48
39.9	Maximum RCS Pressure, psia	2224

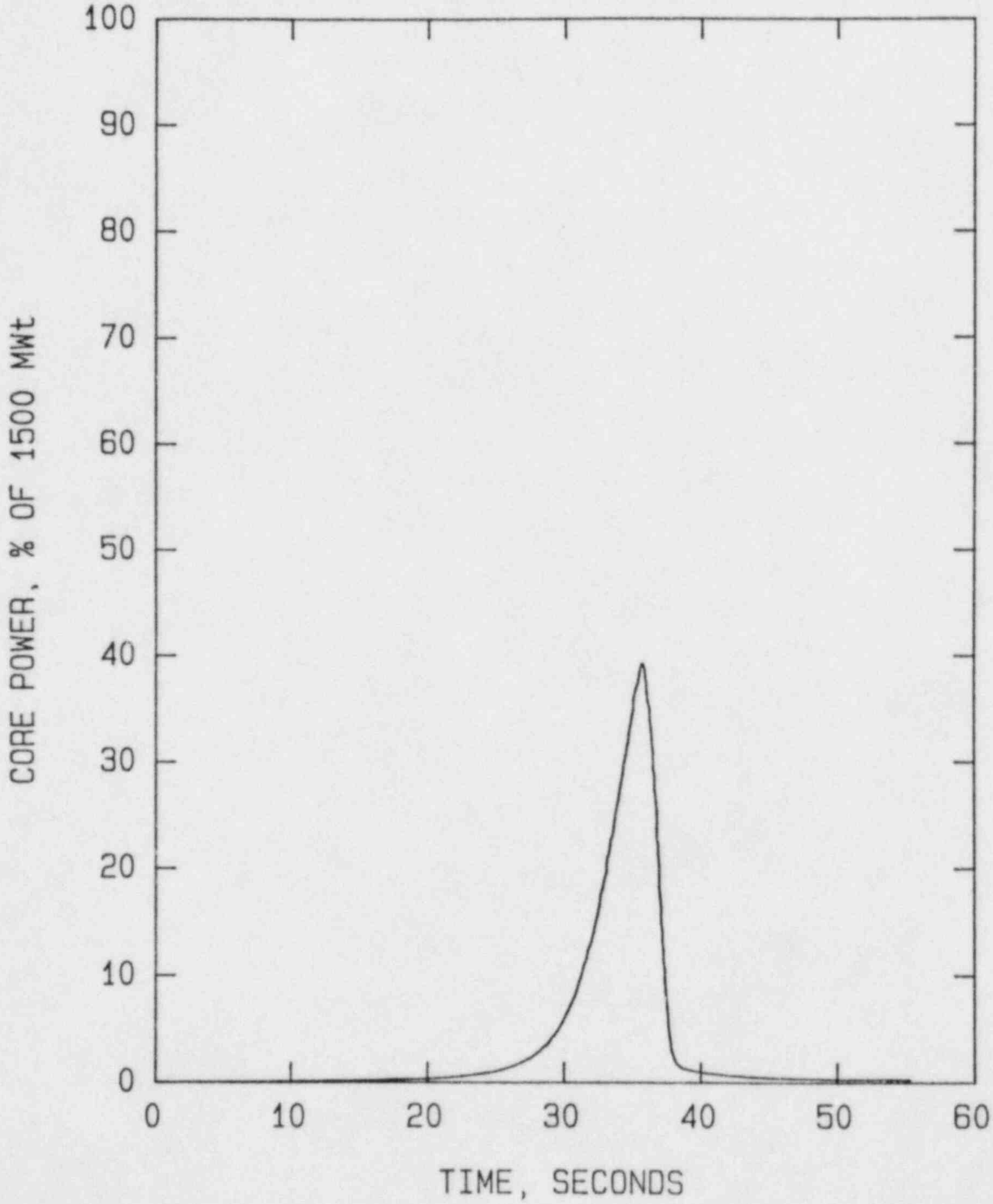
TABLE 7.2.1-3

FORT CALHOUN CYCLE 10
SEQUENCE OF EVENTS FOR
CEA WITHDRAWAL FROM FULL POWER
(MAXIMUM REACTIVITY INSERTION RATE)

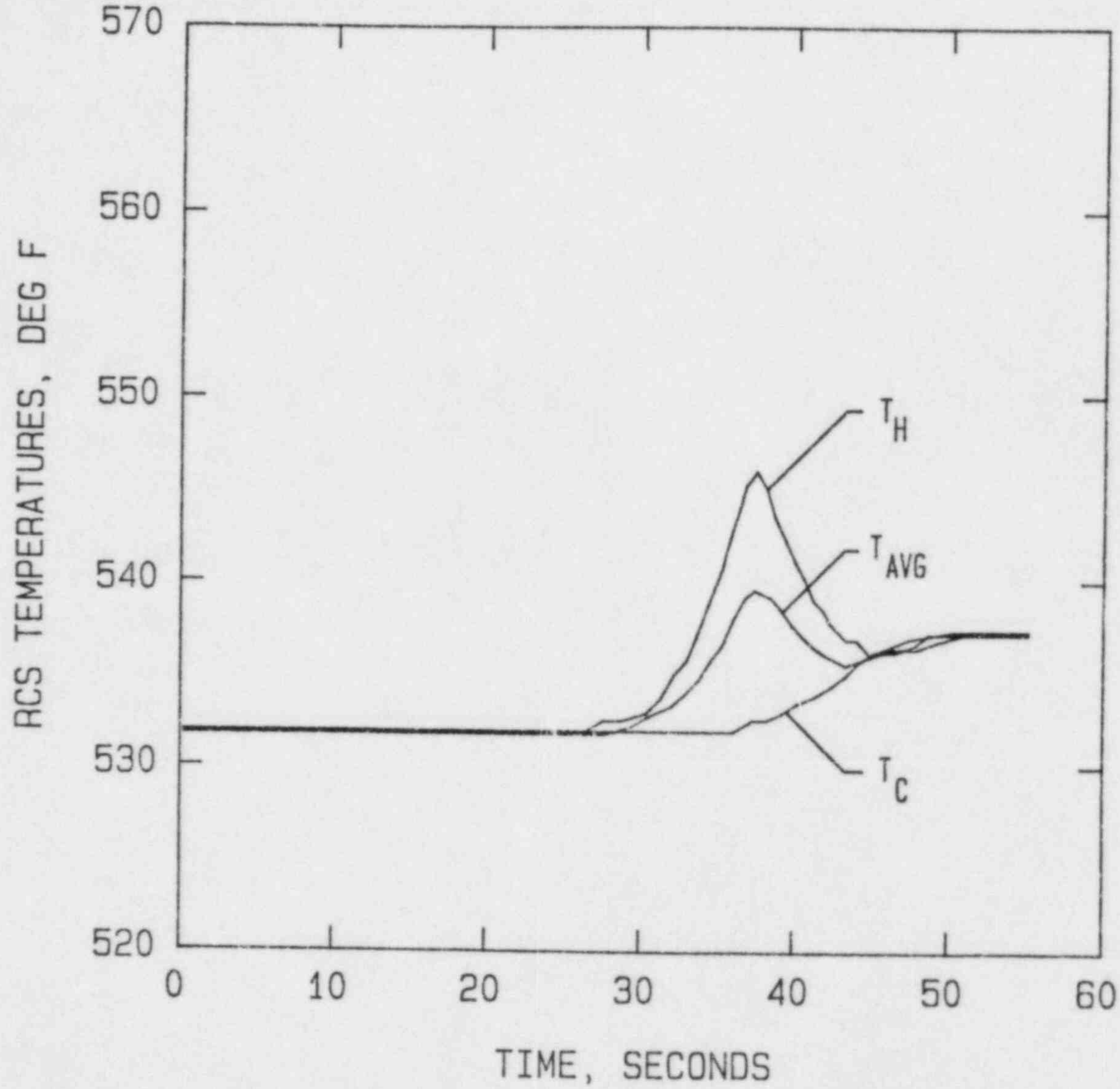
Time (sec)	Event	Setpoint or Value
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	---
4.5	High Power Trip Signal Generated	112% of 1500 MWt
4.9	Reactor Trip Breakers Open	---
5.4	CEAs Begin to Drop Into Core	---
5.5	Maximum Core Power	113.92% of 1500 MWt
5.8	Maximum Heat Flux	108.51% of 1500 MWt
6.7	Maximum RCS Pressure, psia	2098



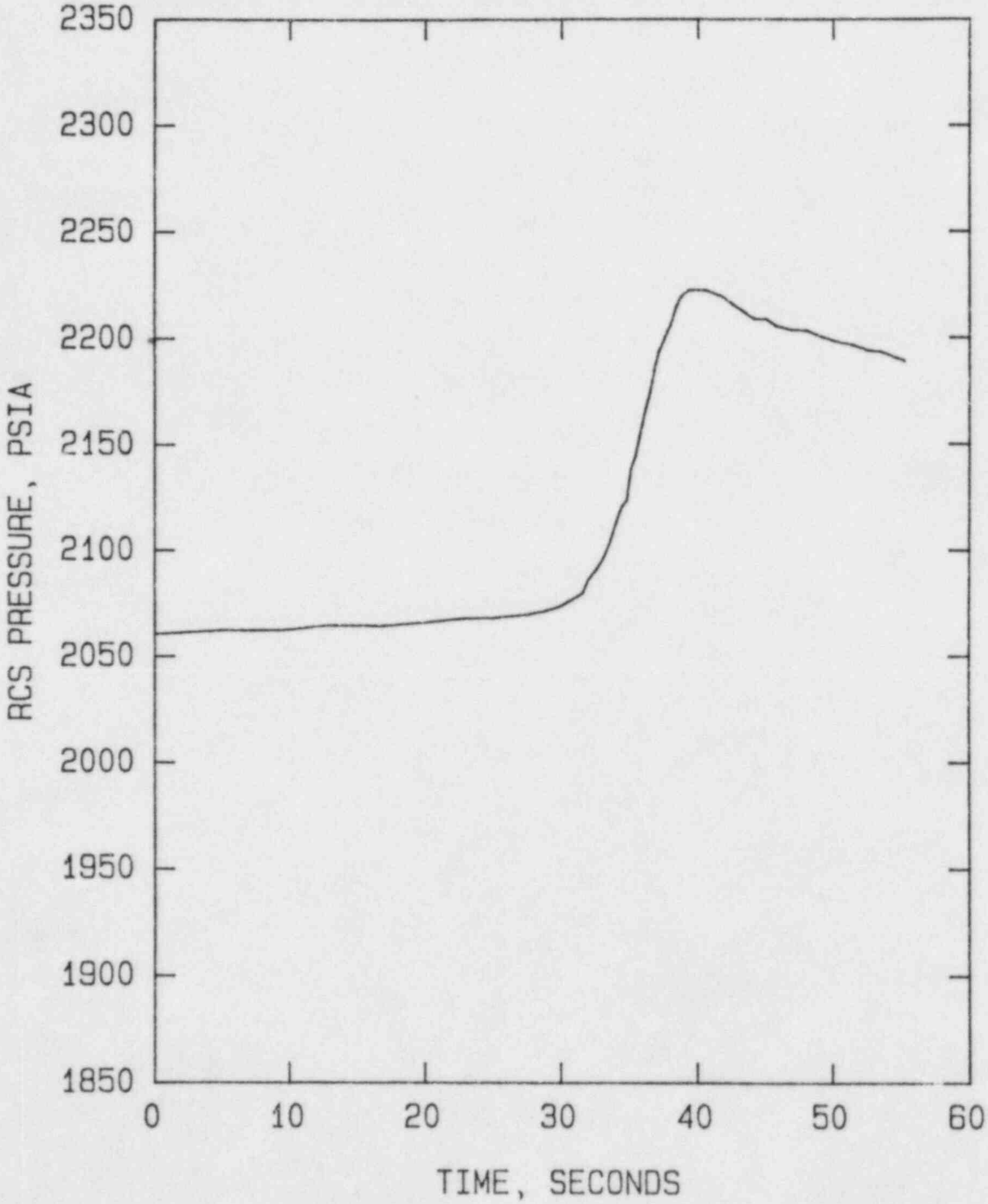
CEA Withdrawal (Zero Power) Core Average Heat Flux vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-2



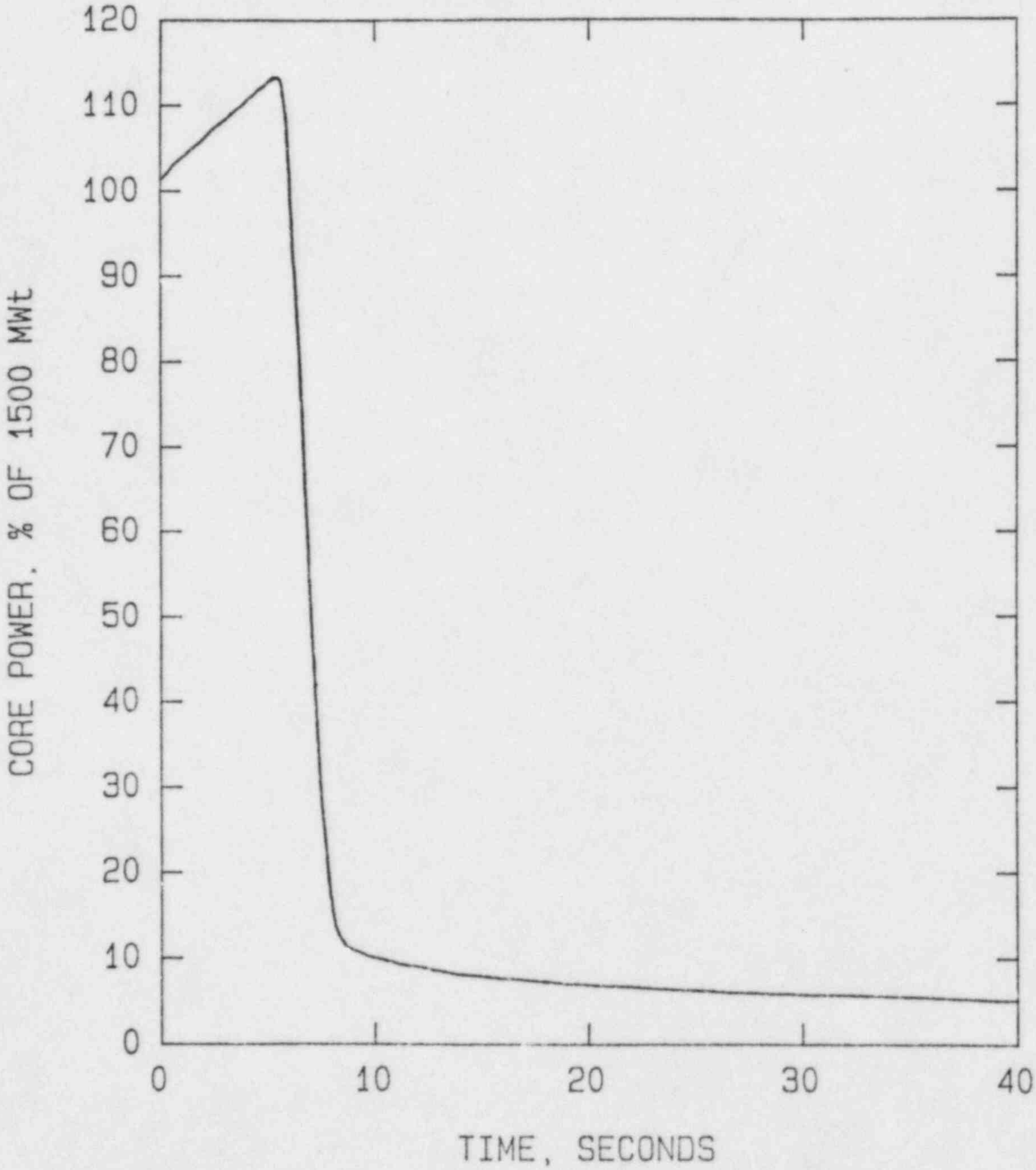
CEA Withdrawal (Zero Power) Core Power vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-1



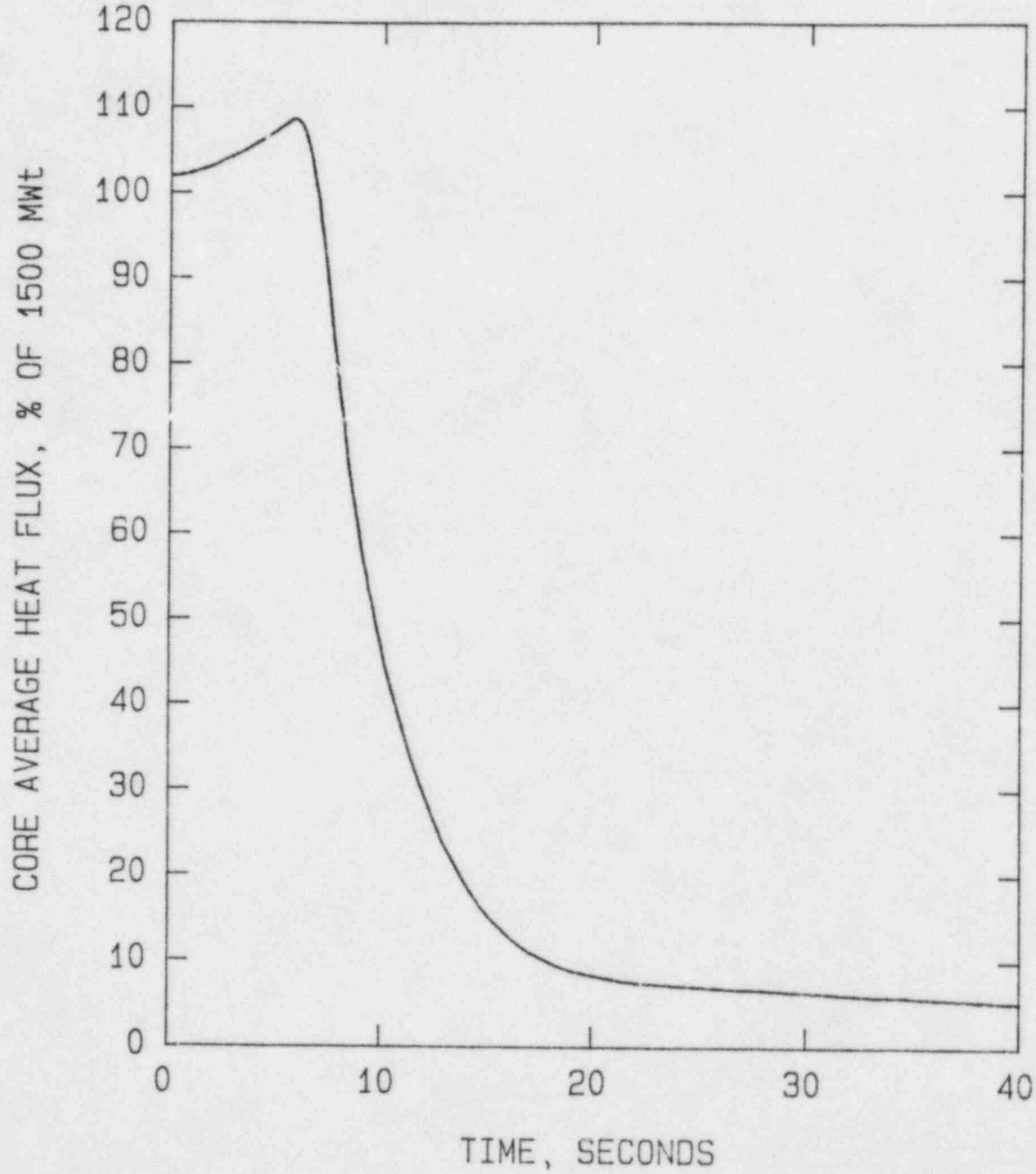
CEA Withdrawal (Zero Power) RCS Temperatures vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-3



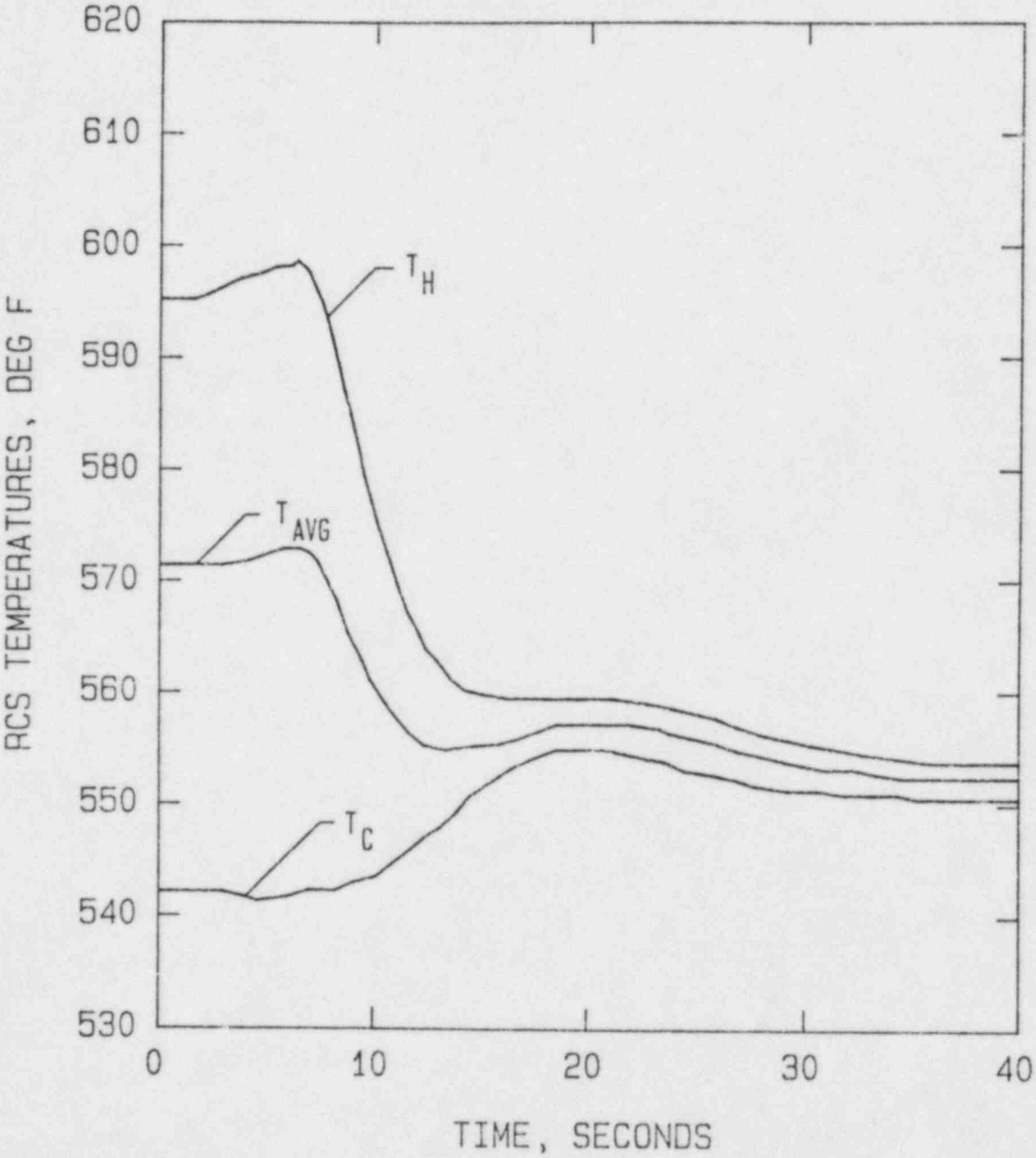
CEA Withdrawal (Zero Power) RCS Pressure vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-4



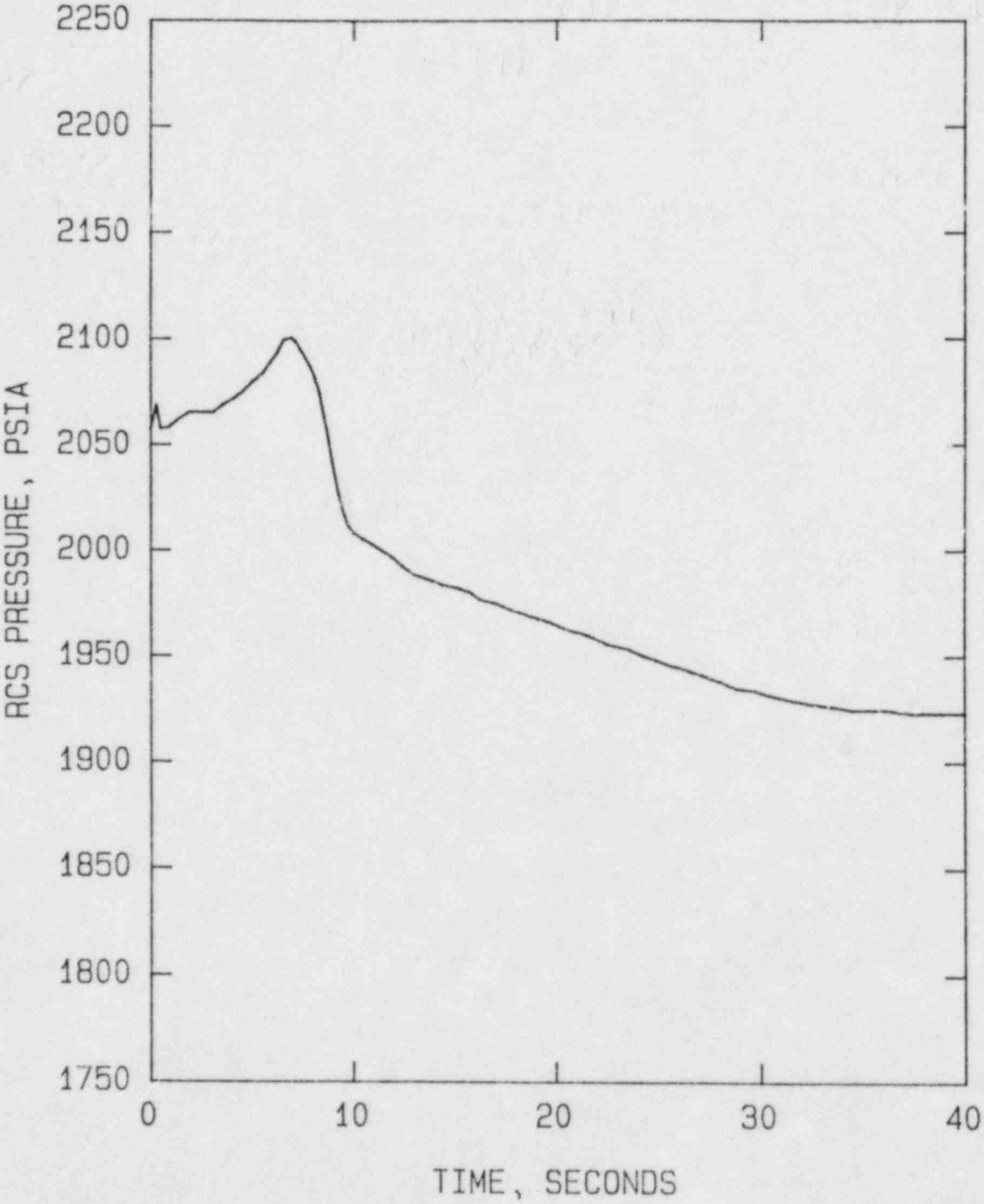
CEA Withdrawal (Full Power) Core Power vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-5



CEA Withdrawal (Full Power) Core Average Heat Flux vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-6



CEA Withdrawal (Full Power) RCS Temperatures vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-7



CEA Withdrawal (Full Power) RCS Pressure vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-8

7.2.2 Loss of Coolant Flow Event

The Loss of Coolant flow event was reanalyzed for Cycle 10 to determine the minimum initial margin that must be maintained by the Limiting Conditions for Operations (LCOs) such that in conjunction with the RPS (low flow trip), the DNBR limit will not be exceeded.

The event was analyzed parametrically in initial axial shape and rod configuration using the methods described in Reference 3 (which utilizes the statistical combination of uncertainties in the DNBR analysis as described in Appendix C of Reference 2c and 2d).

The 4-Pump Loss of Coolant Flow produces a rapid approach to the DNBR limit due to the rapid decrease in the core coolant flow. Protection against exceeding the DNBR limit for this transient is provided by the initial steady state thermal margin which is maintained by adhering to the Technical Specifications' LCOs on DNBR margin and by the response of the RPS which provides an automatic reactor trip on low reactor coolant flow as measured by the steam generator differential pressure transmitters.

The flow coastdown is generated by CESEC-III (References 6 and 7) which utilizes implicit modeling of the reactor coolant pumps. This coastdown is shown in Figure 7.2.2-1. Table 7.2.2-1 lists the key transient parameters used in the Cycle 10 analysis and compares them to the reference cycle (Cycle 9) values.

Table 7.2.2-2 presents the NSSS and RPS responses during a four pump loss of flow initiated at an axial shape index of -0.182 which bounds the DNBR related axial shape index LCO. The low flow trip setpoint is reached at 2.00 seconds and the scram rods start dropping into the core 1.15 seconds later. A minimum CE-1 DNBR of 1.50 is reached at 3.88 seconds. Figures 7.2.2-2 to 7.2.2-5 present the core power, heat flux, core coolant temperatures , and RCS pressure as a function of time.

It may be concluded that the Loss of Flow event when initiated from the Tech. Spec. LCOs in conjunction with the Low Flow Trip will not exceed the design DNBR limit.

TABLE 7.2.2-1

FORT CALHOUN CYCLE 10
KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

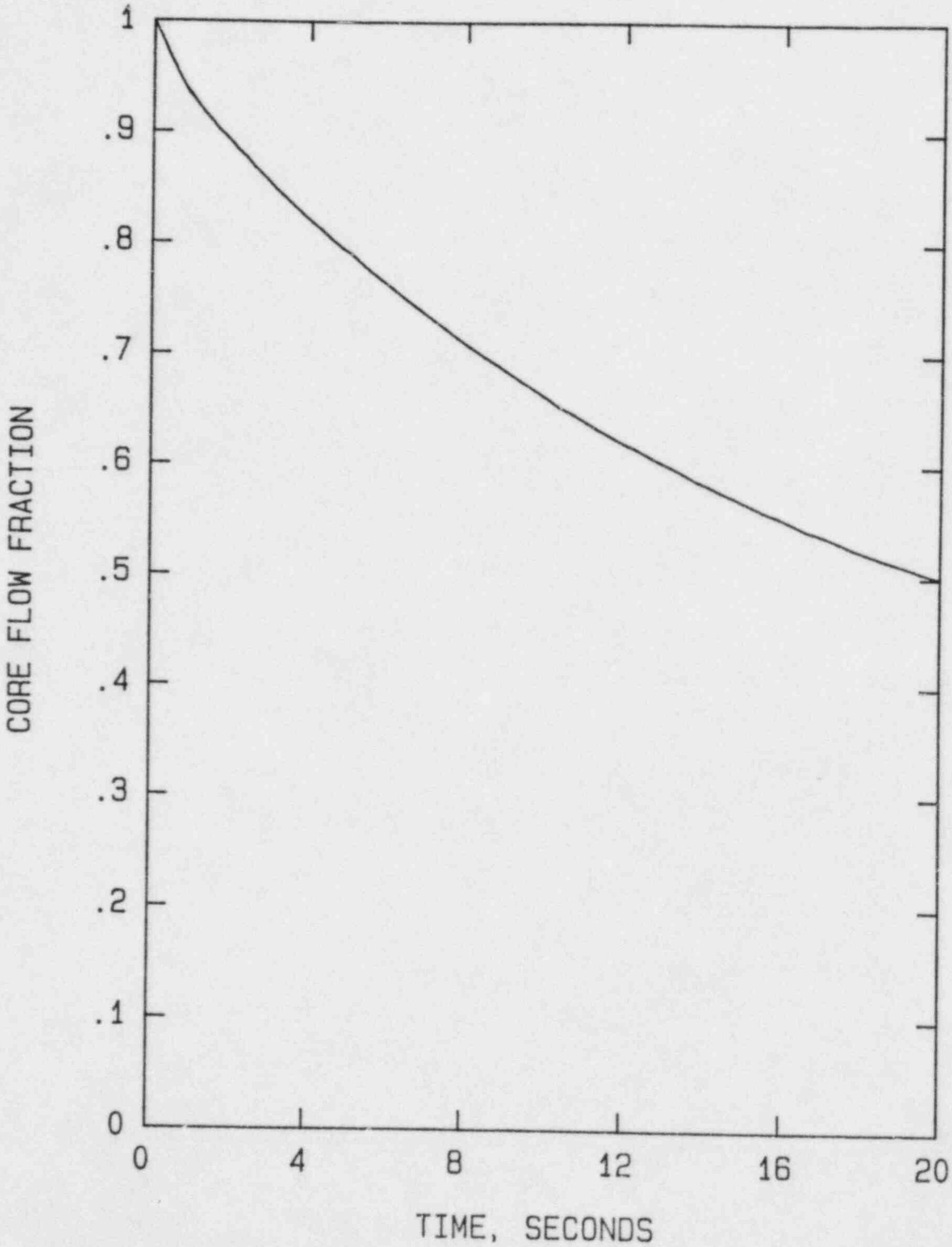
Parameter	Units	Cycle 9	Cycle 10
Initial Core Power Level	MWt	1530*	1530*
Initial Core Inlet Coolant Temperature	°F	547*	542*
Initial RCS Flow Rate	gpm	202,500*	202,500*
Pressurizer Pressure	psia	2053*	2053*
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5	+0.5
Doppler Coefficient Multiplier	- -	0.85	0.85
LFT Analysis Setpoint	% of initial flow	93	93
LFT Response Time	sec	0.65	0.65
4-Pump RCS Flow Coastdown		Fig. 7.2.2-1	Fig. 7.2.2-1
CEA Holding Coil Delay	sec	0.5	0.5
CEA Time to 100% Insertion (Including Holding Coil Delay)	sec	3.1	3.1
CEA Worth at Trip (all rods out)	$\%\Delta\rho$	-6.87	-5.58
Total Unrodded Radial Peaking Factor (F_R^T)		1.75	1.80

*The uncertainties on these parameters were combined statistically rather than deterministically. The values listed represent the bounds included in the statistical combination.

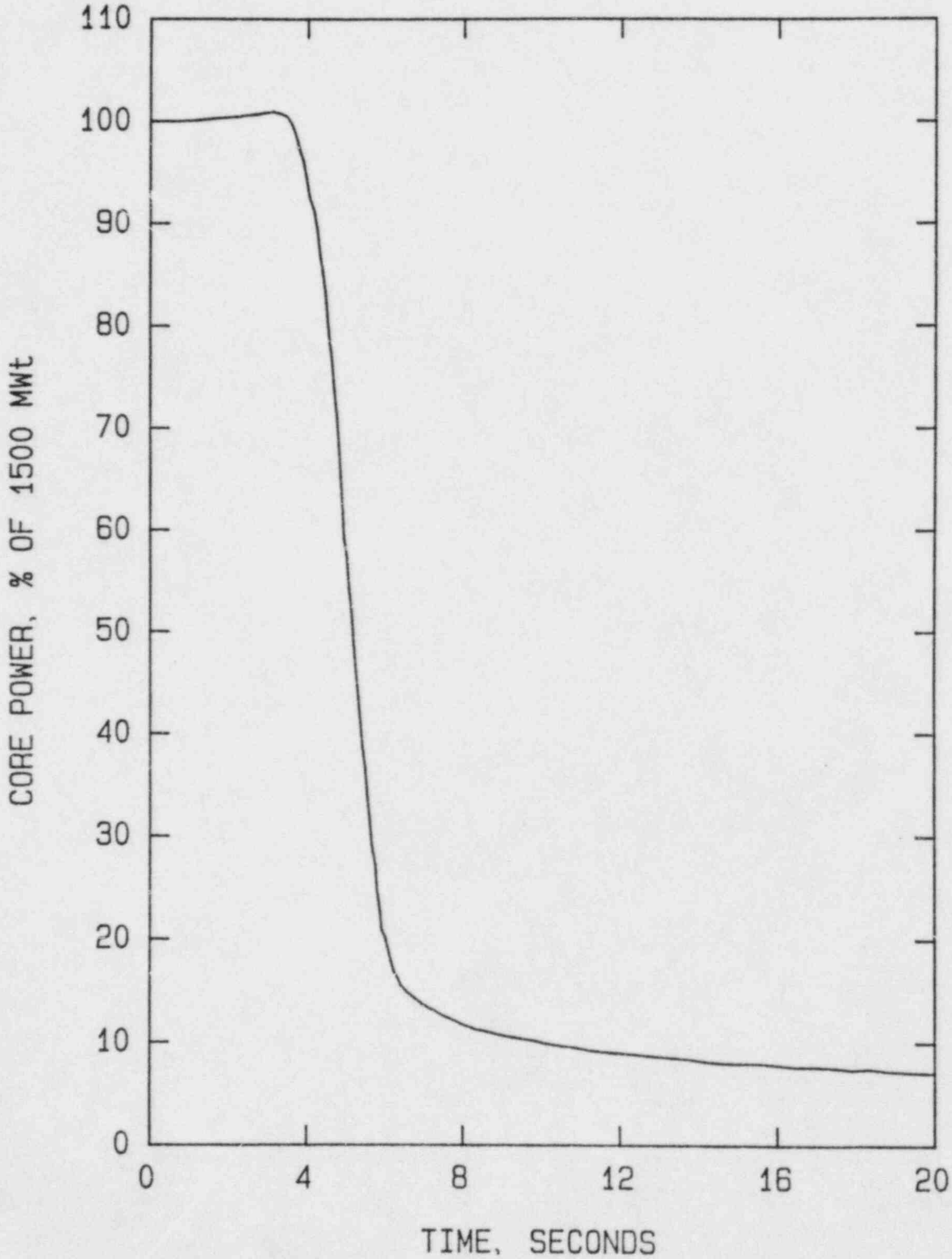
TABLE 7.2.2-2

FORT CALHOUN CYCLE 10
SEQUENCE OF EVENTS FOR LOSS OF FLOW

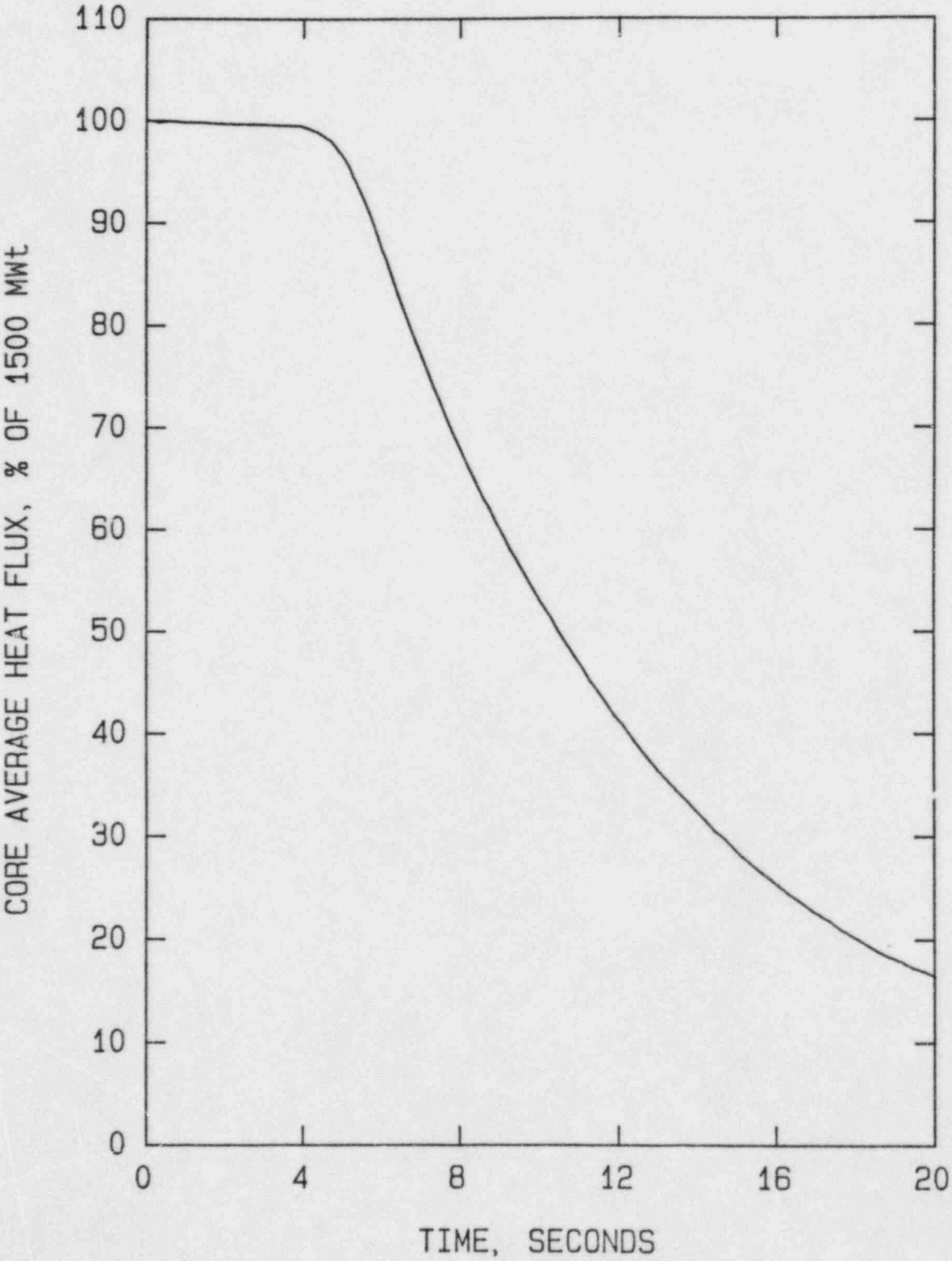
Time (Sec)	Event	Setpoint or Value
0.0	Loss of Power to all Four Reactor Coolant Pumps	- - - -
2.0	Low Flow Trip Signal Generated	93% of 4-Pump Flow
2.7	Trip Breakers Open	- - - -
3.2	Shutdown, CEAs Being to Drop into Core	- - - -
3.9	Minimum CE-1 DNBR	1.50
5.6	Maximum RCS Pressure, psia	2116



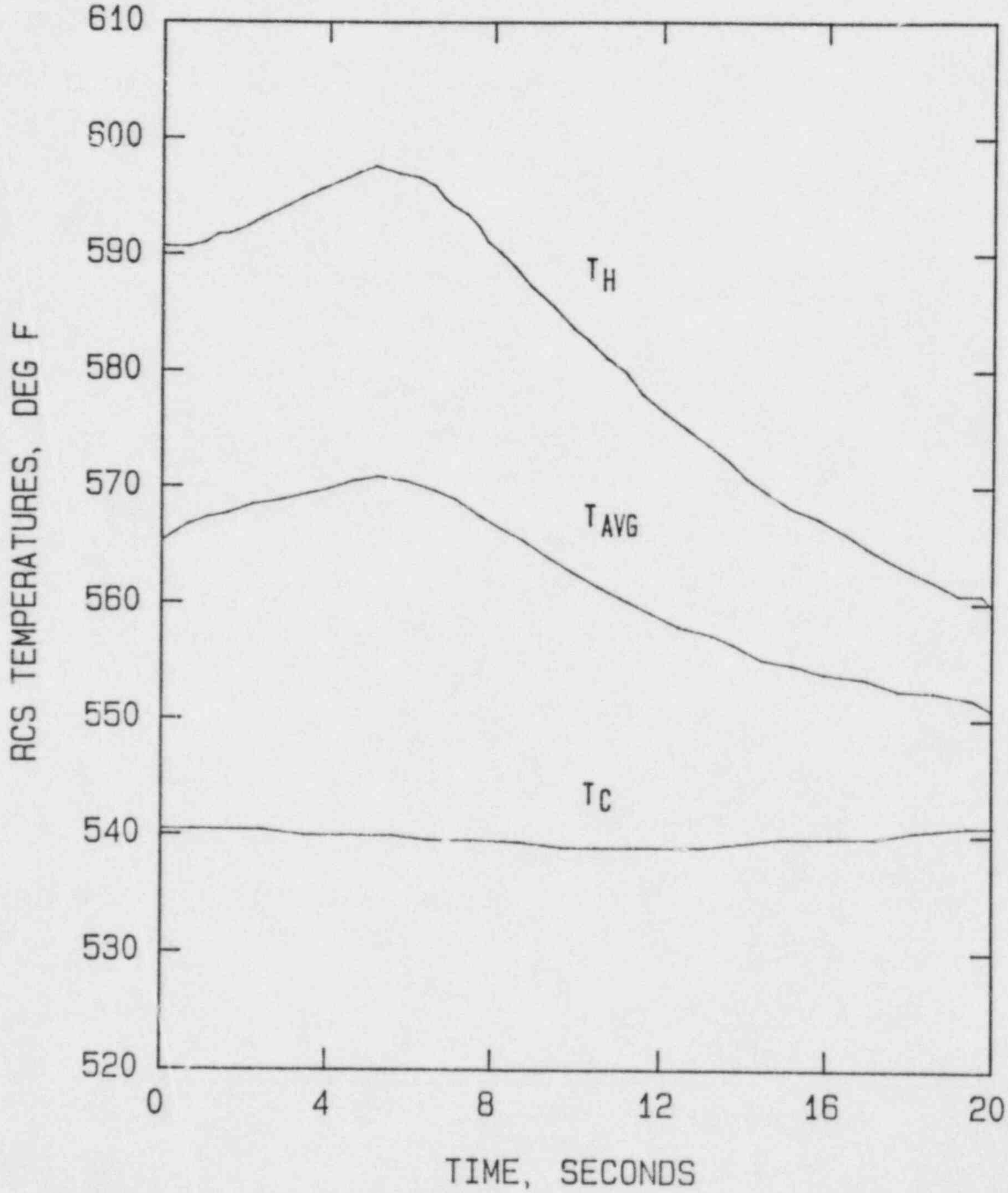
Loss of Coolant Flow Core Flow Fraction vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.2-1



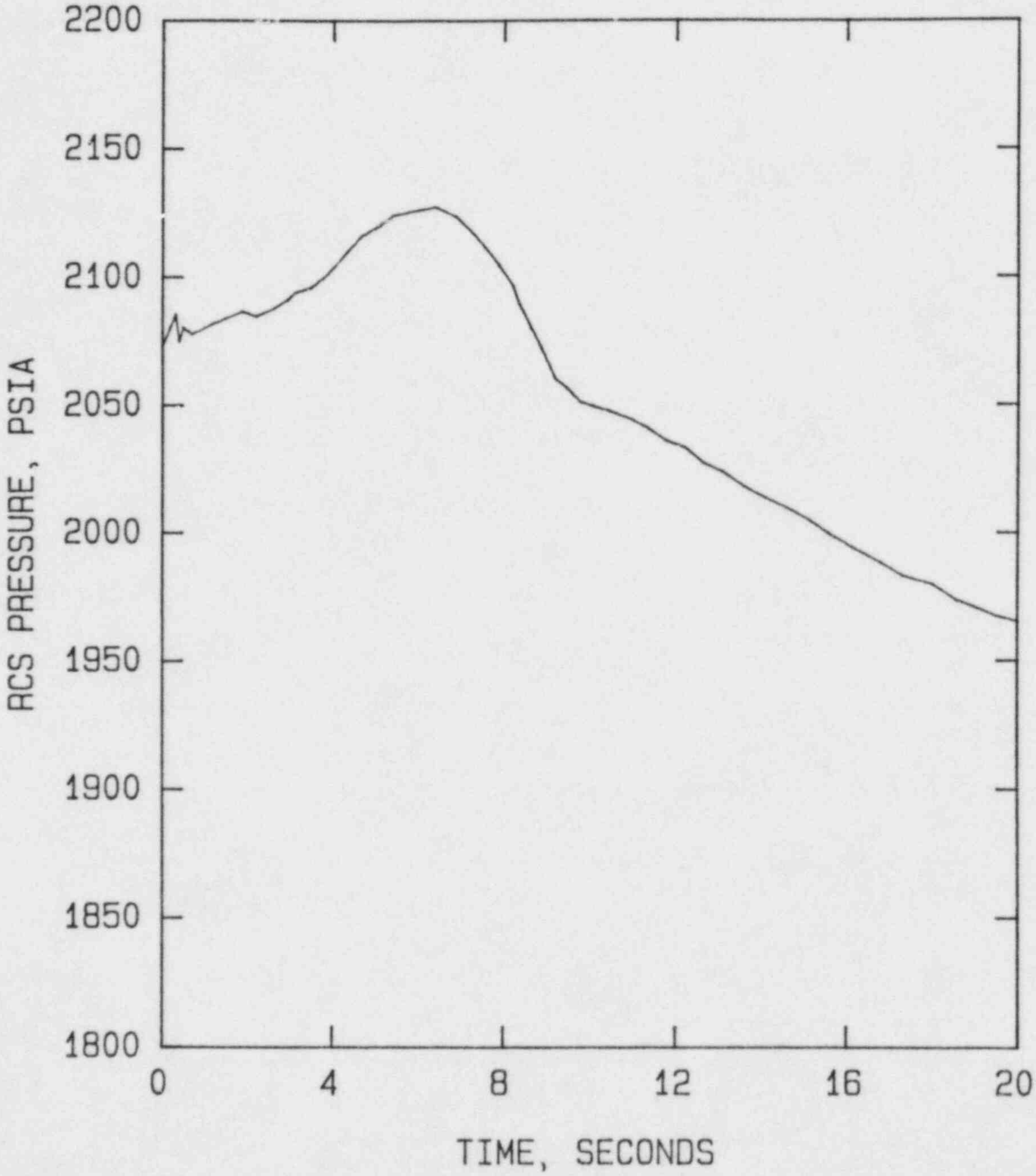
Loss of Coolant Flow Core Power vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.2-2



Loss of Coolant Flow Core Average Heat Flux vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.2-3



Loss of Coolant Flow RCS Temperature vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.2-4



Loss of Coolant Flow RCS Pressure vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.2-5

7.0 TRANSIENT ANALYSIS (Continued)

7.2 (Continued)

7.2.3 Full Length CEA Drop Event

The Full Length CEA Drop event was reanalyzed for Cycle 10 to determine the initial thermal margins that must be maintained by the Limiting Conditions for Operation (LCOs) such that the DNBR and fuel centerline to melt design limits will not be exceeded.

This event was analyzed parametrically in initial axial shape and rod configuration using methods described in Reference 3.

Table 7.2.3-1 lists the key input parameters used for Cycle 10 and compares them to the reference cycle (Cycle 9) values. Conservative assumptions used in the analysis are consistent with those discussed in Reference 3 and include:

1) The most negative moderator and fuel temperature coefficients of reactivity (including uncertainties), because these coefficients produce the minimum RCS coolant temperature decrease upon return to power and lead to the minimum DNBR.

2) Charging pumps and proportional heater systems are assumed to be inoperable during the transient. This maximizes the pressure drop during the event.

3) All other systems are assumed to be in the manual mode of operation and have no impact on this event.

Table 7.2.3-2 presents the sequence of events for the Full Length CEA Drop event producing the minimum DNBR. This event was initiated at an ASI of -0.182 with Group 4 at the PDIL and with the other conditions described in Table 7.2.3-1. The transient behavior of key NSSS parameters are presented in Figures 7.2.3-1 to 7.2.3-4.

The transient was conservatively analyzed at full power with an ASI of -0.182, which is outside of the LCO limit of -0.06. This results in a minimum CE-1 DNBR of 1.47. A maximum allowable initial linear heat generation rate of 16.8 KW/ft could exist as an initial condition without exceeding the acceptable fuel centerline to melt limit of 21 KW/ft during this transient. This amount of margin is assured by setting the Linear Heat Rate related LCOs based on the more limiting allowable linear heat rate for LOCA.

It can be concluded that the CEA Drop event when initiated from the Tech. Spec. LCOs will not exceed the DNBR and centerline to melt design limits.

TABLE 7.2.3-1

FORT CALHOUN CYCLE 10
KEY PARAMETERS ASSUMED IN THE FULL LENGTH CEA DROP ANALYSIS

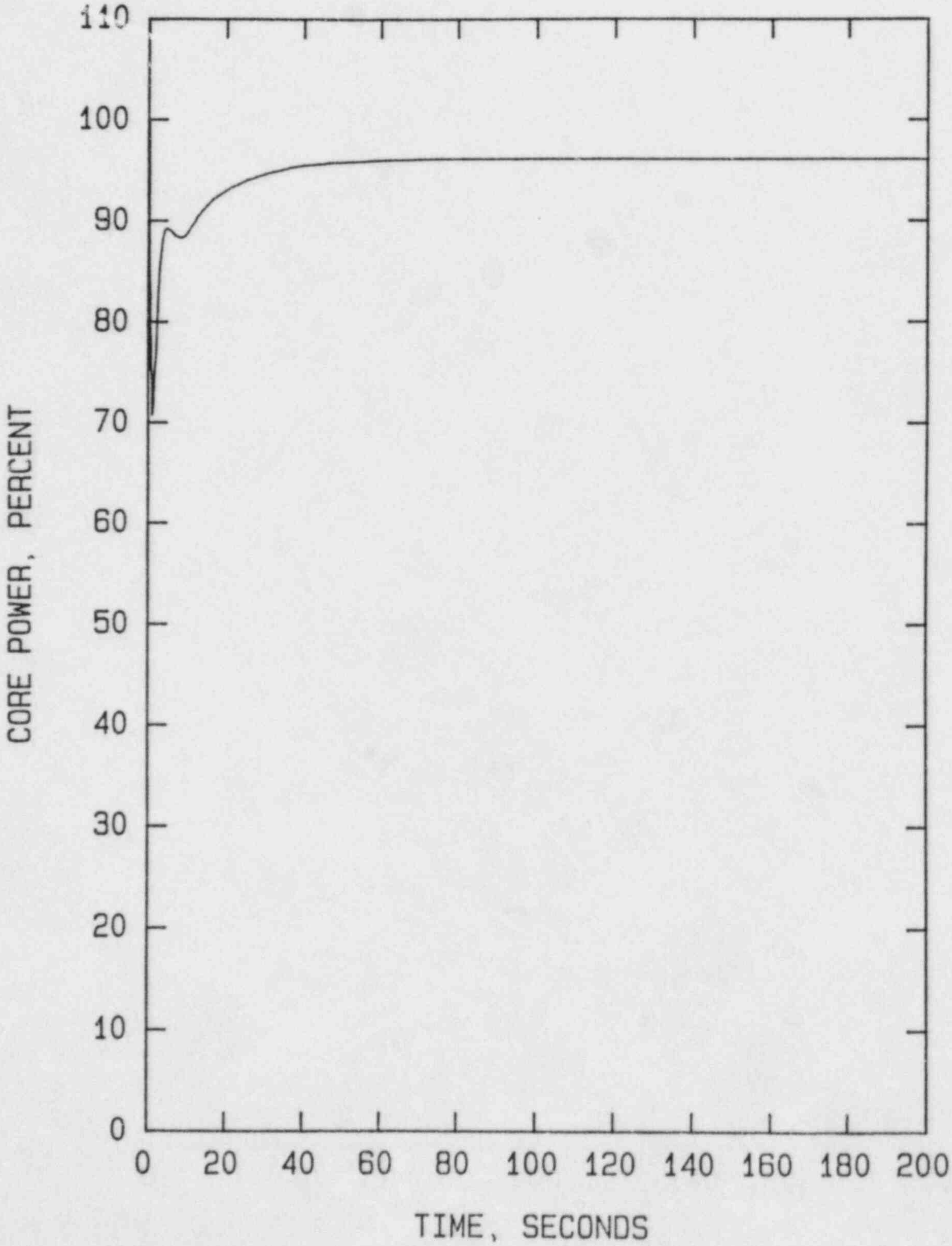
Parameter	Units		Cycle 9	Cycle 10
Initial Core Power Level	MWt		102% of 1500*	102% of 1500*
Core Inlet Temperature	°F		547*	542*
Pressurizer Pressure	psia		2053*	2053*
Core Mass Flow Rate	gpm		202,500*	202,500*
Moderator Temperature Coefficient	$x10^{-4} \Delta\rho/°F$		-2.7	-2.7
Doppler Coefficient Multiplier	--		1.15	1.15
CEA Insertion at Maximum Allowed Power	% Insertion of Bank 4		25	25
Dropped CEA Worth	$\%\Delta\rho$	unrodded	-0.2261	-0.2308
		PDIL	-0.2238	-0.2238
Maximum Allowed Power Axial Shape Index at Negative Extreme of LCO Band			-0.18	-0.18
Radial Peaking Distortion Factor				
Integrated Radial Peaking	Unrodded Region		1.1585	1.1741
	Bank 4 Inserted Region		1.1557	1.1727
Planar Radial Peaking	Unrodded Region		1.213	1.251
	Bank 4 Inserted Region		1.205	1.221

*The uncertainties on these parameters were combined statistically rather than deterministically. The values listed represent the bounds included in the statistical combination.

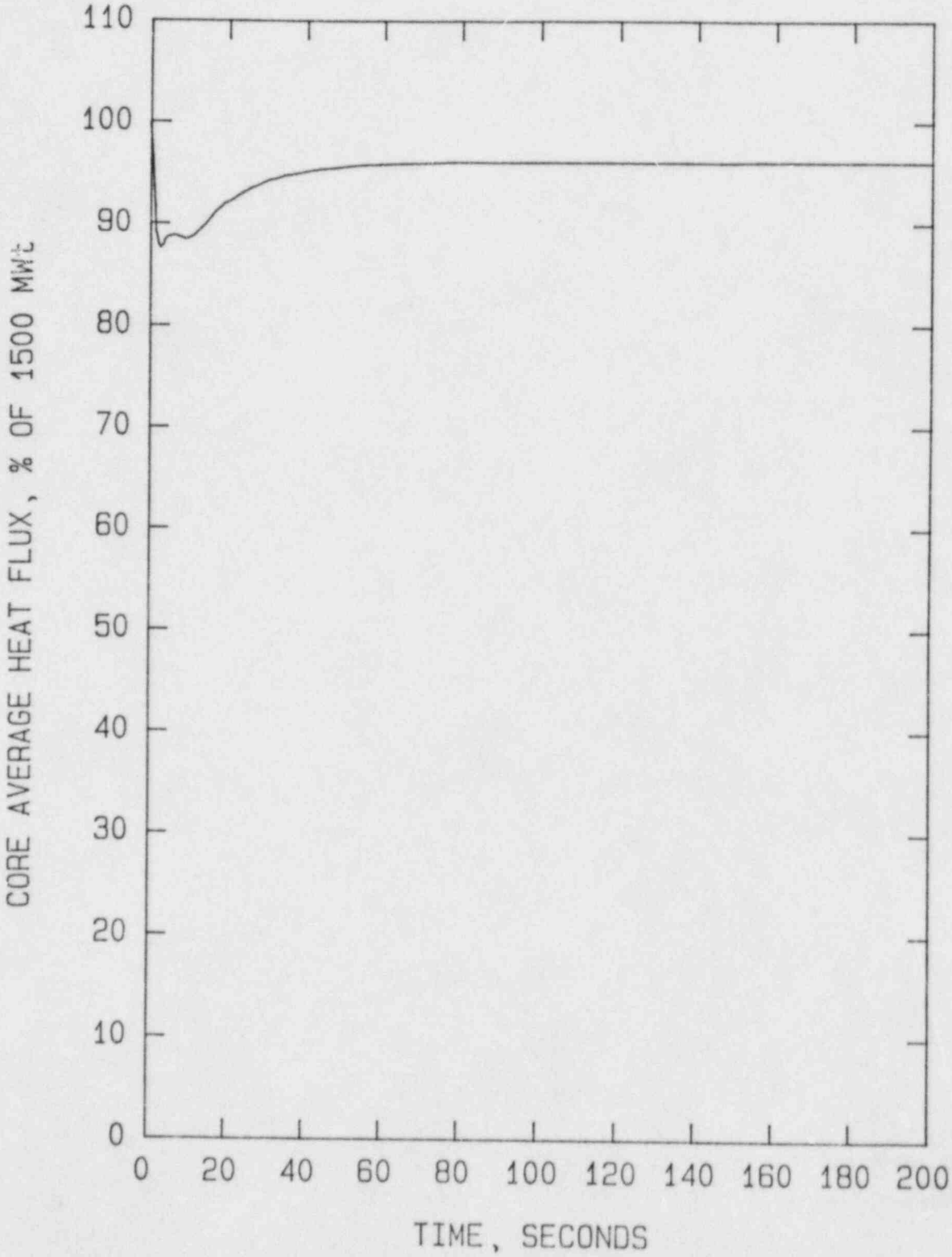
TABLE 7.2.3-2

FORT CALHOUN CYCLE 10
SEQUENCE OF EVENTS FOR FULL LENGTH CEA DROP

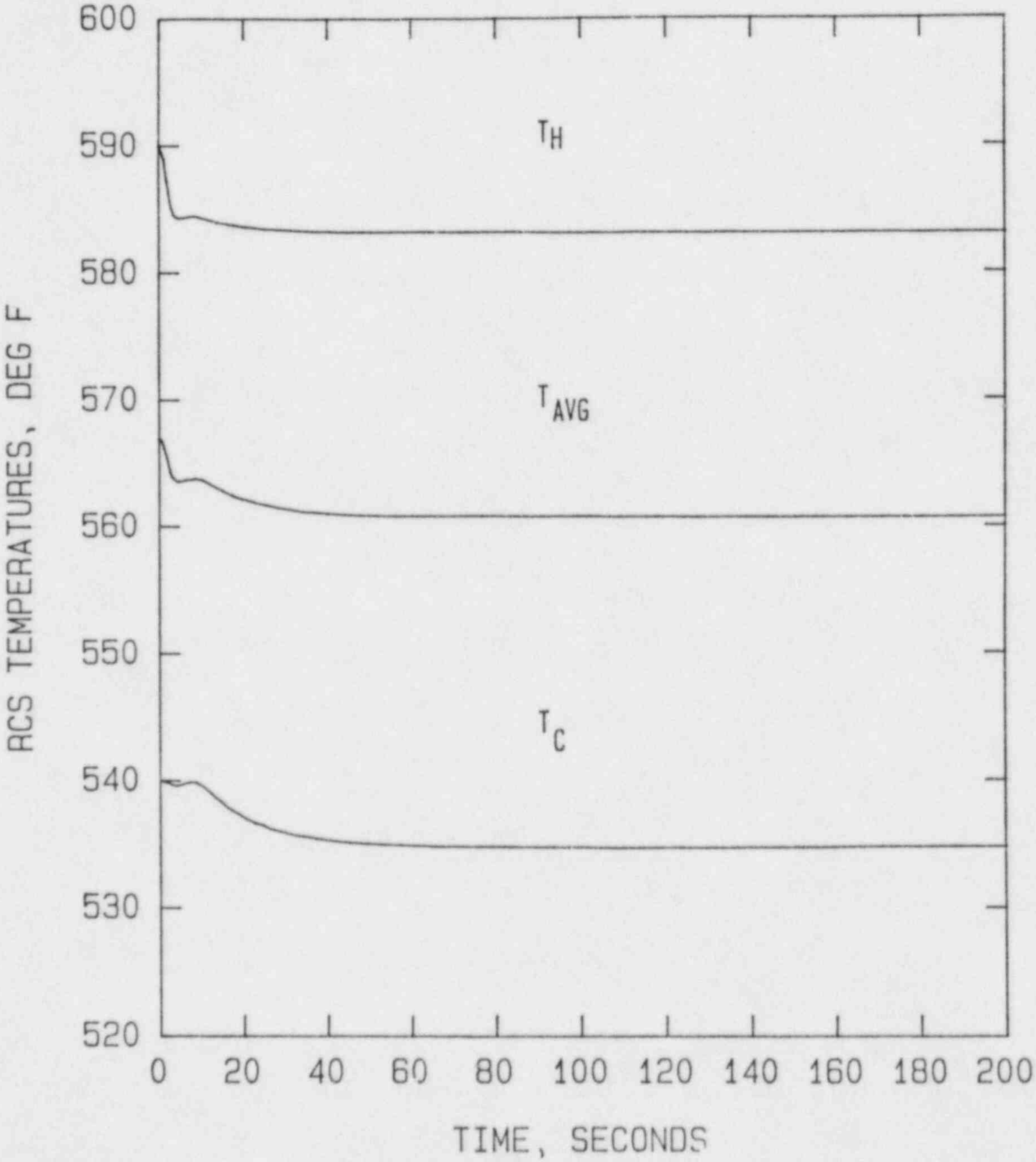
Time (Sec)	Event	Setpoint or Value
0.0	CEA Begins to Drop into Core	--
1.0	CEA Reaches Full Inserted Position	100% Inserted
1.1	Core Power Level Reaches Minimum and Begins to Return to Power due to Reactivity Feedbacks	70.7% of 1500
96.5	Core Inlet Temperature Reaches a Minimum Value	534.9°F
198.9	Reactor Coolant System Pressure Reaches a Minimum Value	2015
200.0	Core Power Returns to its Maximum Value	96.2% of 1500 MWt
200.0	Minimum DNBR is Reached	1.47



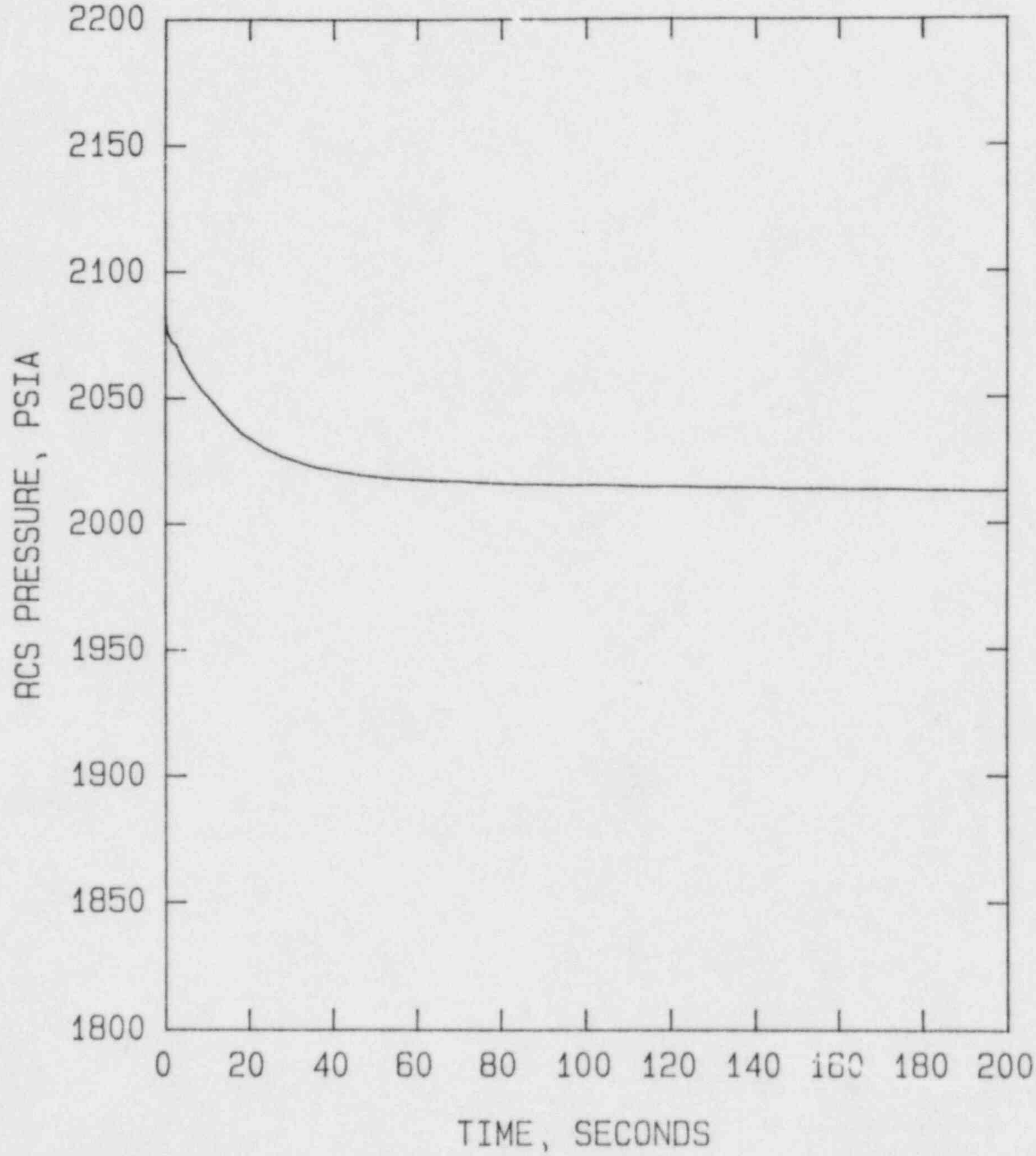
Full Length CEA Drop Core Power vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.3-1



Full Length CEA Drop Core Average Heat Flux vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.3-2



Full Length CEA Drop RCS Temperatures vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.3-3

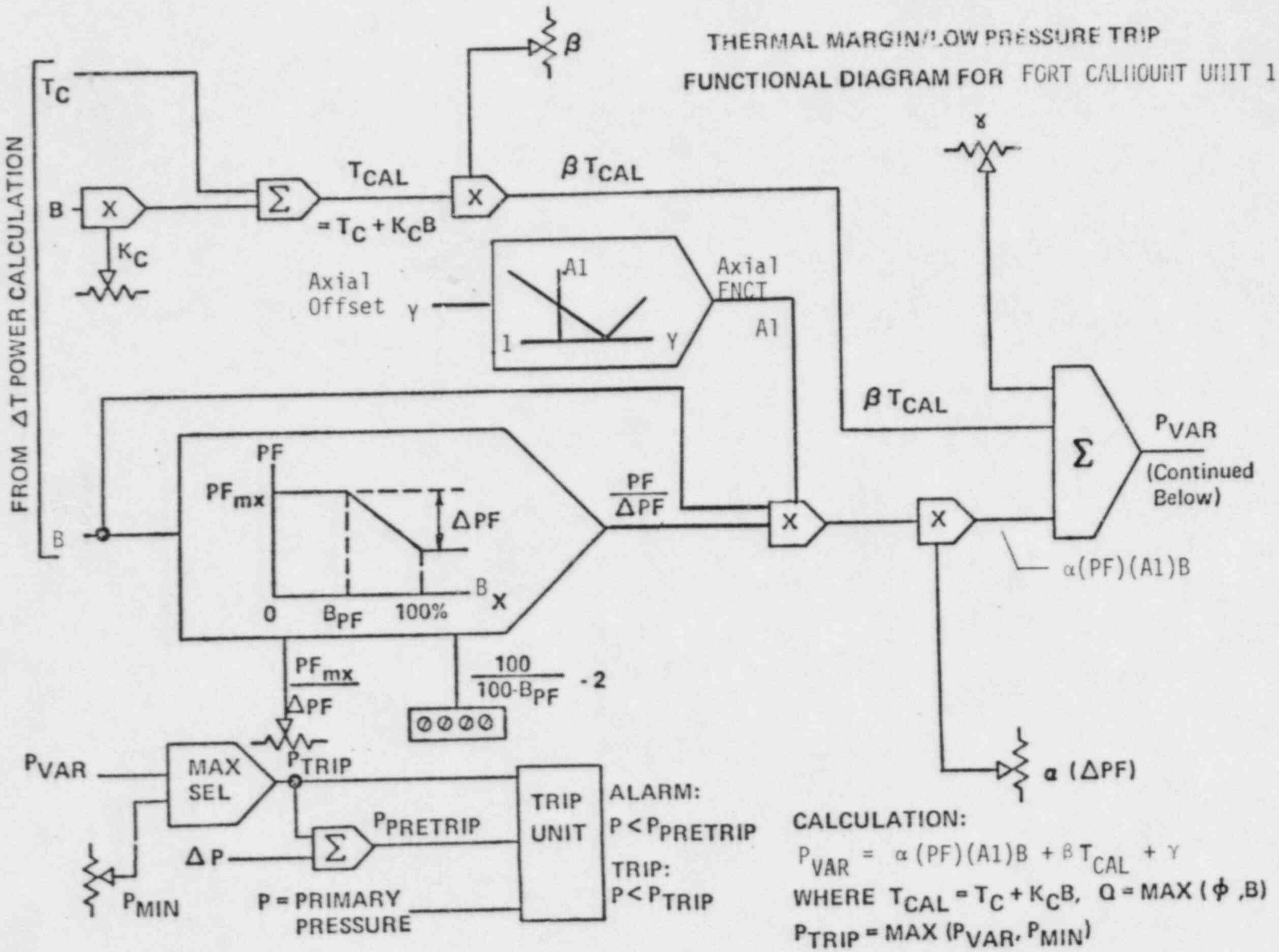


Full Length CEA Drop RCS Pressure vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.3-4

FIGURE 11-1

THERMAL MARGIN/LOW PRESSURE TRIP

FUNCTIONAL DIAGRAM FOR FORT CALHOUN UNIT 1



7.0 TRANSIENT ANALYSIS (Continued)

7.3 (Continued)

7.3.1 CEA Ejection

Input parameters to the CEA Ejection accident were examined and found to be bounded by the previous analysis of Cycle 6. Therefore, under the guidelines of 10CFR 50.59 no reanalysis for Cycle 10 was performed.

7.3.2 Steam Line Break Accident

This accident was evaluated for Cycle 10 using the methodology discussed in References 8 and 3. The Steam Line Break accident was previously analyzed in the Fort Calhoun FSAR and satisfactory results were reported therein. The Steam Line Break accidents at both HZP and HFP were examined in the reference cycle (Cycle 8) safety evaluation with acceptable results obtained. Both the FSAR and reference cycle evaluations are reported in the 1985 update of the Fort Calhoun USAR.

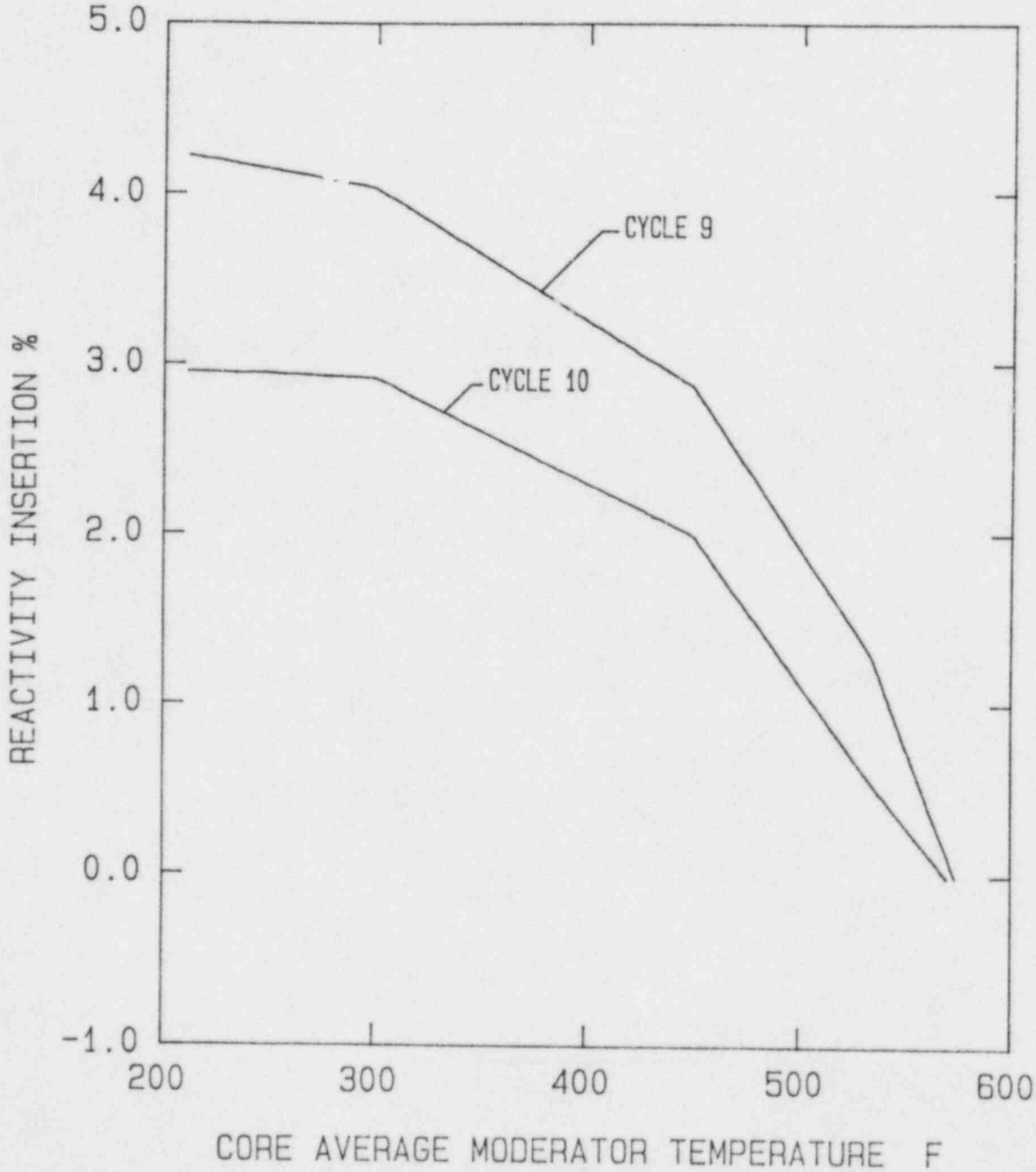
The Cycle 10 Full Power Steam Line Break accident was evaluated for a more negative effective MTC of $-2.7 \times 10^{-4} \Delta\rho/°F$ than the $-2.5 \times 10^{-4} \Delta\rho/°F$ value that was used in the Cycle 8 analysis. The Cycle 10 negative MTC limit of $-2.7 \times 10^{-4} \Delta\rho/°F$, however, remains unchanged from Cycle 9, and the cooldown curve for Cycle 10 is bounded by that of Cycle 9 (as shown in Figure 7.2.3-1). The cooldown curves for Cycles 1, 8 and 10 are shown in Figure 7.2.3-2. This figure shows that the reactivity insertion for the Cycle 10 core with an MTC of $-2.7 \times 10^{-4} \Delta\rho/°F$ due to a Steam Line Break accident at full power is substantially less than the value used in the Cycle 8 analysis. (This smaller reactivity insertion is due to the use of the DIT cross-sections which are valid for a range of moderator temperatures from room temperature to 800°K while the analyses prior to Cycle 9 were performed with cooldown curves derived by conservatively extrapolating CEPAK cross-section values to low temperatures.) The fuel temperature coefficient used in the Cycle 8 analysis is conservative with respect to the fuel temperature coefficient calculated for the Cycle 10 core including uncertainties. The Cycle 10 minimum available shutdown worth is $6.45\%\Delta\rho$ compared to a Cycle 8 value of $6.68\%\Delta\rho$. The reduction of $0.23\%\Delta\rho$ in scram worth from Cycle 8 to Cycle 10 is offset by the $3.83\%\Delta\rho$ gain in moderator cooldown reactivity. The net gain assures that the overall reactivity insertion for a Cycle 10 Steam Line Break is less than that of the reference cycle analysis. Therefore, the return to power is less than that of the reference cycle and Cycle 1 FSAR analyses.

A similar evaluation was performed for the Zero Power Steam Line Break accident. Again the Cycle 10 cooldown for an MTC of $-2.7 \times 10^{-4} \Delta\rho/°F$ shows a substantially smaller reactivity insertion than was used in the Cycle 8 analysis (as seen in

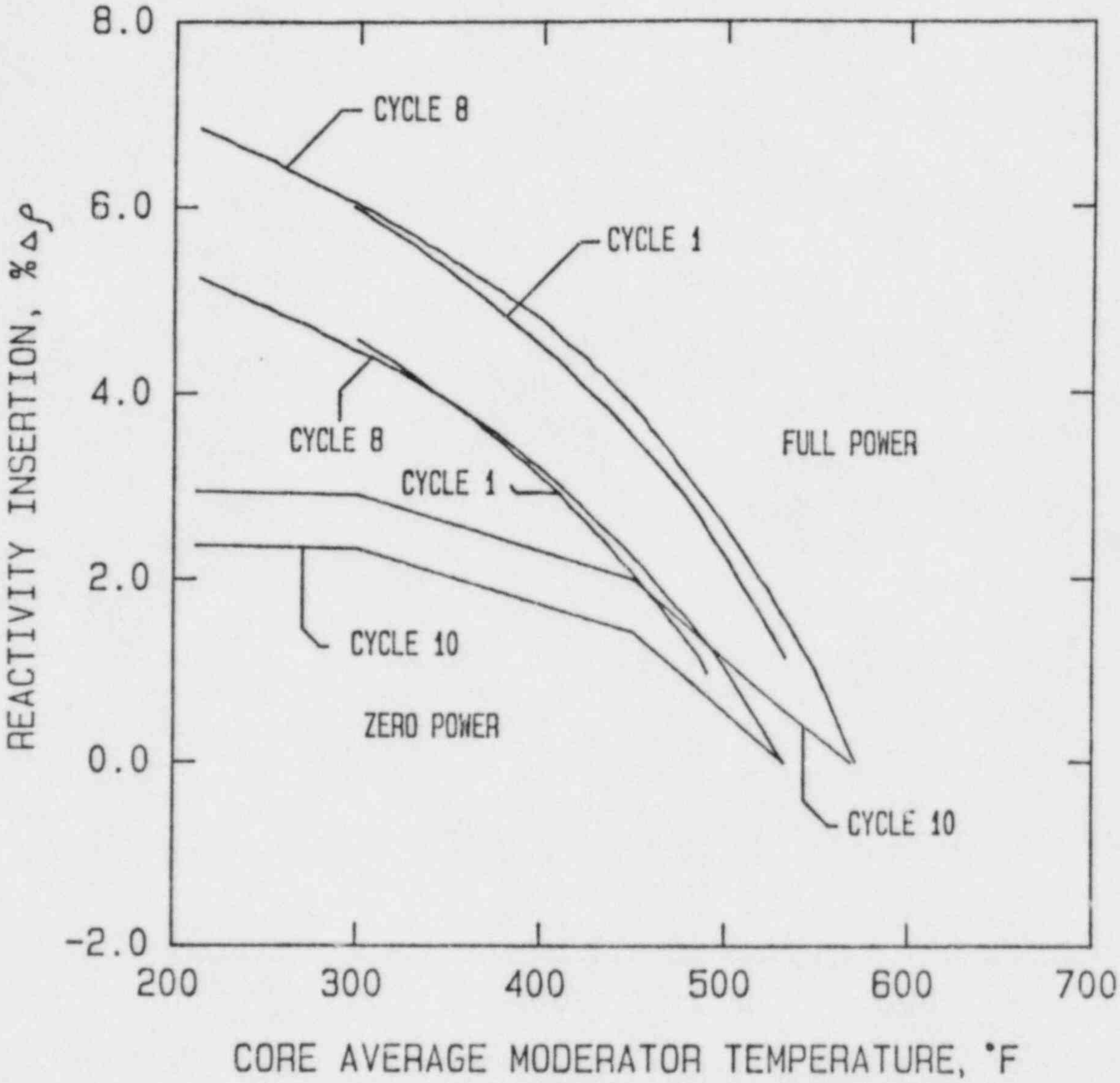
Figure 3.2.2-2. Since the minimum available shutdown margin for Cycle 10 remains unchanged from the reference cycle value (4%$\Delta\rho$), the overall reactivity insertion for the Cycle 10 Steam Line Break accident will be substantially less than that of the reference cycle. Therefore, the consequences of a zero power Steam Line Break accident for Cycle 10 will be less severe than that reported for the reference cycle and the FSAR (Cycle 1) cases.

Based on the evaluation presented above, it is concluded that the consequences of a Steam Line Break accident initiated at either zero or full power are less severe than the reference cycle and FSAR (Cycle 1) cases.

Since a negative 10 CFR 50.59 determination was made for the Cycle 10 Steam Line Break Accident, no reanalysis was performed.



Steam Line Break Incident Reactivity vs Moderator Temperature	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.3.2-1



Steam Line Break Incident Reactivity vs Moderator Temperature	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.3.2-2

7.0 TRANSIENT ANALYSIS (Continued)

7.3 (Continued)

7.3.4 Seized Rotor Event

The Seized Rotor event was evaluated for Cycle 10 to demonstrate that only a small fraction of fuel pins are predicted to fail during this event. Cycle 10 is bounded by the Cycle 9 analysis because an F_R^T of 1.85 was assumed in the Cycle 9 analysis and the Cycle 10 Technical Specification of 1.80 remains conservative with respect to the F_R^T value used in the Cycle 9 analysis.

Therefore, the total number of pins predicted to fail will continue to be less than 1% of all of the fuel pins in the core. Based on this result, the resultant site boundary dose would be well within the limits of 10CFR100.

Since a negative 10 CFR 50.59 determination was made for the Cycle 10 Seized Rotor Event, no reanalysis was performed.

8.0 ECCS PERFORMANCE ANALYSIS

The Loss of Coolant accident evaluation was performed using the methodology discussed in Reference 1. The District has verified that the physics input assumptions and the maximum rod burnup are within the bounds assumed in the large break analysis for the reference cycle (Cycle 8) as reported in the 1985 update of the USAR. Therefore, under the guidelines of 10CFR50.59 no reanalysis for Cycle 10 was performed.

9.0 STARTUP TESTING

The startup testing program proposed for Cycle 10 is identical to the program outlined in the Cycle 6 Reload Application, with two exceptions. First, a CEA exchange technique for zero power rod worth measurements will be performed, in addition to the normal boration/dilution technique. Also, low power CECOR flux maps and psuedo-ejection rod measurements will be substituted for the full core symmetry checks.

The CEA exchange technique is a method for measuring rod worths which is both faster and produces less waste than the typical boration-dilution method. The substitution of a zero power pseudo-ejection rod worth measurement and low power CECOR maps for full core symmetry checks will provide a better assessment of any azimuthal power tilts because of the asymmetric design of the Cycle 10 core. The combination of the pseudo-ejection technique at zero power and low power CECOR maps provides a less time consuming but equally valid technique for detecting azimuthal power tilts during reload core physics testing. The psuedo-ejection rod measurement involves the dilution of a bank into the core, borating a CEA out, and then exchanging (rod swap) the CEA against other symmetric CEA's within the bank to measure rod worths. The acceptance and review criteria for these tests are:

Test	Acceptance Criteria	Review Criteria
Pseudo-ejection rod worth measurement	None	The greater of: 2.5¢ deviation from group average or 15% deviation from group average.
Low Power CECOR maps	Technical Specification limits on F_R^T, F_{xy}^T, and T_q	Azimuthal tilt less than 20%.

OPPD has reviewed these tests and has concluded that no unreviewed safety question exists for implementation of these procedures.

10.0 MINI-CECOR/BASSS LCO MONITORING SYSTEM

The Better Axial Shape Selection System (BASSS) monitors the Limiting Conditions for Operation on peak linear heat rate and departure from nucleate boiling using as input the data available from the MINI-CECOR code and the ERF plant computer. This arrangement is similar to the one used by Baltimore Gas and Electric at their Calvert Cliffs Units, and described in the Combustion Engineering Setpoint Methodology Topical, CENPD-199-P, Revision 1-P, dated April 1982.

MINI-CECOR is a mini-computer version of Combustion Engineering's CECOR code. It uses the same algorithms as the mainframe version but has a reduced level of editing options to enable it to fit on a mini-computer. Combustion Engineering developed MINI-CECOR and will install and benchmark the code on the District's ERF computer system prior to the beginning of Cycle 10 operation. It is used in this application to synthesize the following parameters from readings of the fixed incore detectors:

1. The three-dimensional power peaking factor (F_q)
2. The core average axial shape index ($\bar{I}$)
3. The total planar radial peaking factor ($F_{xy}T$)
4. The total integrated radial peaking factor (F_RT)

These inputs to BASSS are descriptive of the existing core power distribution.

The inputs to BASSS obtained from the plant computer are the following:

1. Measured core power level
2. Percent insertion of the lead CEA regulating group.

BASSS consists of two algorithms: one for peak linear heat rate monitoring and another for DNB monitoring. The peak linear heat rate algorithm uses the 3-D power peaking factor and the measured core power level to calculate the core peak linear heat rate. The algorithm applies appropriate uncertainties and allowances (per the Technical Specifications) to the 3-D peaking factor. The measured peak linear heat rate is compared to the monitoring limit, which is based on both LOCA and AOO transient analysis considerations, and an alarm is activated when the monitoring limit is exceeded. The power operating limit on linear heat rate is also calculated and displayed as an indication of the available operating margin. The DNB algorithm is an improvement over the excore ASI monitoring system in that it uses the incore axial shape index, CEA group position and the radial peaking factors to establish the plant's power operating limit. An alarm is activated when the power operating limit is exceeded. A gain in operating margin results from the following:

1. A reduction in ASI uncertainty due to the use of incore ASI versus excore ASI.
2. Knowledge of the actual CEA group position versus the excore system's assumption that the CEAs are inserted to the PDIL's transient insertion limit.

3. Knowledge of the actual radial peaking factors versus the excore system's assumptions that radial peaks are at the Technical Specification limits.

BASSS is also provided with the capability to monitor the Limiting Conditions for Operation on $F_{xy}T$ and F_RT. If the Technical Specification limits of $F_{xy}T$ or F_RT are exceeded during normal plant operation, BASSS will activate an alarm and calculate the proper tradeoff with maximum allowed power that ensures that the Axial Power Distribution and Thermal Margin/Low Pressure Trips remain conservative. An alarm is activated if the measured power is higher than the allowed power level.

The uncertainties for the use of Mini-CECOR/BASSS to monitor DNB & LHR has been developed by Combustion Engineering and documented in Reference 1.

11.0 TM/LP MODIFICATION

The TM/LP calculators at the Fort Calhoun Station originally monitored core power, reactor coolant inlet temperature and core coolant pressure. The axial shape index was not a monitored parameter, but was assumed (in the setpoint analysis) to always be at the Axial Power Distribtuion LSSS value. For Cycle 10, the TM/LP calculators were modified to include modules for the ASI monitoring. This change makes the TM/LP calculators functionally like the CE "standard system" (Ref. 1).

Figure 11-1 is a simplified functional diagram of this system. The signal representing the core power (B) is the auctioneered highest of the neutron heat flux and the delta-T power. This signal is used to calculate the PF function. The PF term multiplied by the core power is the equivalent to the QR1 term in the "standard" TM/LP equation. The measured axial shape index signal (Y), which includes the adjustment for shape annealing and represents the peripheral axial shape index, is used to calculate A1. The A1, PF, B, and the constant α are multiplied together to generate a signal representing the first term in the PVAR equation.

The second and third terms of the P_{var} equation remain the same. The second term is the product of the constant β and T_{cal}. The third term is the constant γ . These terms are added to the first term to calculate the variable low pressure trip limit. The calculated limit is then compared to a fixed low pressure trip limit (P_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured reactor coolant pressure (P) and a trip signal is generated when P is less than or equal to P_{trip}.

The change to the TM/LP calculators was the addition of the A1 function. The balance of the system is unchanged.

The uncertainty associated with ASI monitoring has been combined statistically into the overall uncertainty of the TM/LP trip system. This work was performed by Combustion Engineering and documented in Reference 2.

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2b. "Statistical Combination of Uncertainties Methodology, Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analysis for Fort Calhoun Unit 1", CEN-257(O)-P, November 1983.

2c. "Statistical Combination of Uncertainties Methodology, Part 3: Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Fort Calhoun", CEN-257(O)-P, November, 1983.

2d. "Statistical Combination of Uncertainties Methodology for Fort Calhoun, Supplement 1-P," CEN-257(O)-P, August 1985.

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6. "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System", Enclosure 1-P to LD-82-001, January 6, 1982.

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