ATTACHMENT A

Technical Specifications

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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. 1.80 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

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.2 Safety Limit, Reactor Coolant System Pressure (Continued)

References

(1) USAR
~~FSAR~~, Section 4

(2) USAR
~~FSAR~~, Section 4.3.3

(3) USAR
~~FSAR~~, Section 4.3.4

(4) USAR
~~FSAR~~, Section 4.3.9.5

(5) USAR
~~FSAR~~, Section 7.4.5.1

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.3 Limiting Safety System Settings, Reactor Protective System

Applicability

This specification applies to RPS Limiting Safety System settings and bypasses for instrument channels.

Objective

To provide for automatic protection action in the event that the principal process variables approach a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be within the Limiting Safety System Setting as stated in Table 1-1.

Basis

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

(1) High Power Level - A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding resulting from some reactivity excursions too rapid to be detected by pressure and temperature measurements (in addition, thermal signals are provided to the high power level trip unit as a backup to the neutron flux signal).

During normal plant operation, with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 107.0% of indicated full power. Adding to this the possible variation in trip point due to calibration and measurement errors, the maximum actual ~~stready~~ steady-state power at which a trip would be actuated is 112%, which was used for the purpose of safety analysis.[1] Provisions have been made to select different high-power level trip points for various combinations of reactor coolant pump operation as described below under "Low Reactor Coolant Flow".[2]

During reactor operation at power levels between 19.1% and 100% of rated power, the Variable High Power Trip (VHPT) will initiate a reactor trip in the event of a reactivity excursion that increases reactor power by 10% or less of rated power. The high power trip set point can be set no more than 10% of rated power above the indicated plant power. Operator action is required to increase the set point as plant power is increased. The set point is automatically decreased as power decreases.

2.10.4 Power Distribution Limits

Applicability

Applies to power operation conditions.

Objective

To ensure that peak linear heat rates, DNB margins, and radial peaking factors are maintained within acceptable limits during power operation.

Specification

(1) Linear Heat Rate

The linear heat rate shall not exceed the limits shown on Figure 2-5 when the following factors are appropriately included:

1. Flux peaking augmentation factors are shown in Figure 2-8,

2. A measurement-calculational uncertainty factor of 1.062,

3. An engineering uncertainty factor of 1.03,

4. A linear heat rate uncertainty factor of 1.002 due to axial fuel densification and thermal expansion, and

5. A power measurement uncertainty factor of 1.02.

The linear heat rate shall be monitored by the incore detector system in accordance with specifications 2.10.4(1)(a) or 2.10.4(1)(b), or maintain the Axial Shape Index, Y_I, within the limits of Figure 2-6 in accordance with specification 2.10.4(1)(c)

(a) When the linear heat rate is continuously monitored by the incore detectors, and the linear heat rate is exceeding its limits as indicated by four or more valid coincident incore detector alarms, either:

(i) Restore the linear heat rate to within its limits within one hour, or

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

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

(b) If while operating under the provisions of part (a), the plant computer incore detector alarms become inoperable, operation may be continued without ~~reducing power provided each of the following~~ conditions is satisfied:

for seven days from the date of the last valid core power distribution

(i) A core power distribution was obtained utilizing incore detectors within 7 days prior to the incore detector alarm outage and the measured peak linear heat rate was no greater than 90% of the value allowed by (1) above.

(ii) The Axial Shape Index as measured by excore detectors remains within $\pm .05$ of the value obtained at the time of the last measured incore power distribution.

(iii) Power is not increased nor has it been increased since the time of the last incore power distribution.

(c) When the linear heat rate is continuously monitored by the excore detectors, withdraw the full length CEA's beyond the long term insertion limits of Specification 2.10.2.7, ~~If the linear heat rate is exceeding its limits~~ as determined by the Axial Shape Index, Y_I, being outside the limits of Figure 2-6, where 100 percent of the allowable power represents the maximum power allowed by the following expression:

and maintain the Axial Shape Index, Y_I, within the limits of Figure 2-6

$$\frac{L}{15.22} \times M$$

where

1. L is the maximum allowable linear heat rate as determined from Figure 2-5 and is based on the core average burnup at the time of the latest incore power map.

2. M is the maximum allowable fraction of rated thermal power as determined by the F_{xy}^{T} limit curve of Figure 2-9 when monitoring by excore detectors. M = 1 when monitoring kw/ft using incore detectors.

(i) Restore the reactor power and Axial Shape Index, Y_I, to within the limits of Figure 2-6 within 2 hours, or

(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 non-trippable CEA's inserted and with all full length CEA's 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 CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the NTCEA's, 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 non-trippable CEA's inserted and with all full length CEA's 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 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~~ 1.80 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 CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the NTCEA's, or

(b) Be in at least hot standby.

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	$\leq$ 545°F*
(ii)	Pressurizer Pressure	$\geq$ 2075 psia*
(iii)	Reactor Coolant Flow	$\geq$ 197,000 gpm**
(iv)	Axial Shape Index, Y_I	$\leq$ 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.

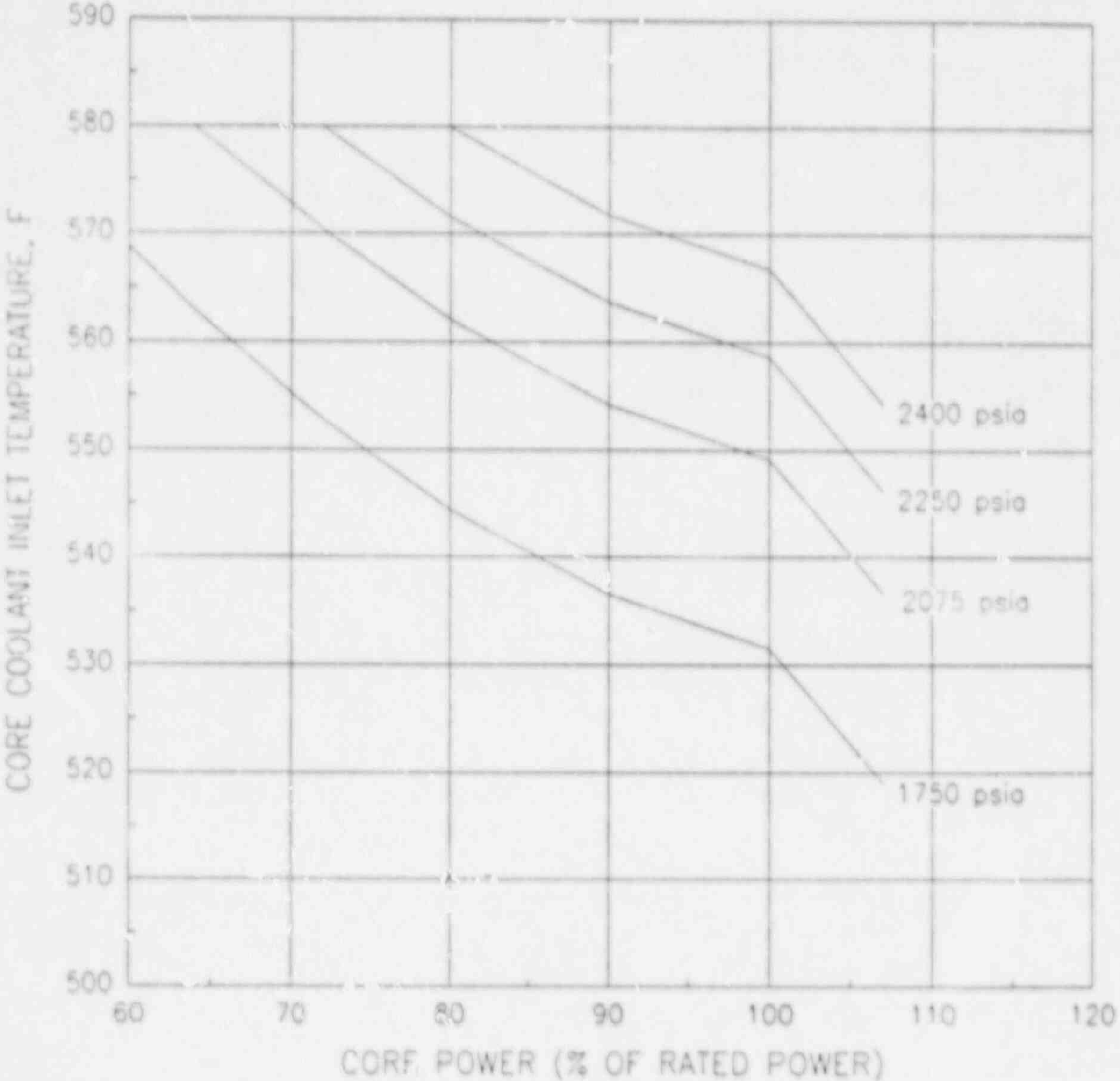
Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System, or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector 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, and (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., 545°F, indicated, allows for an actual T_c of 547°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.

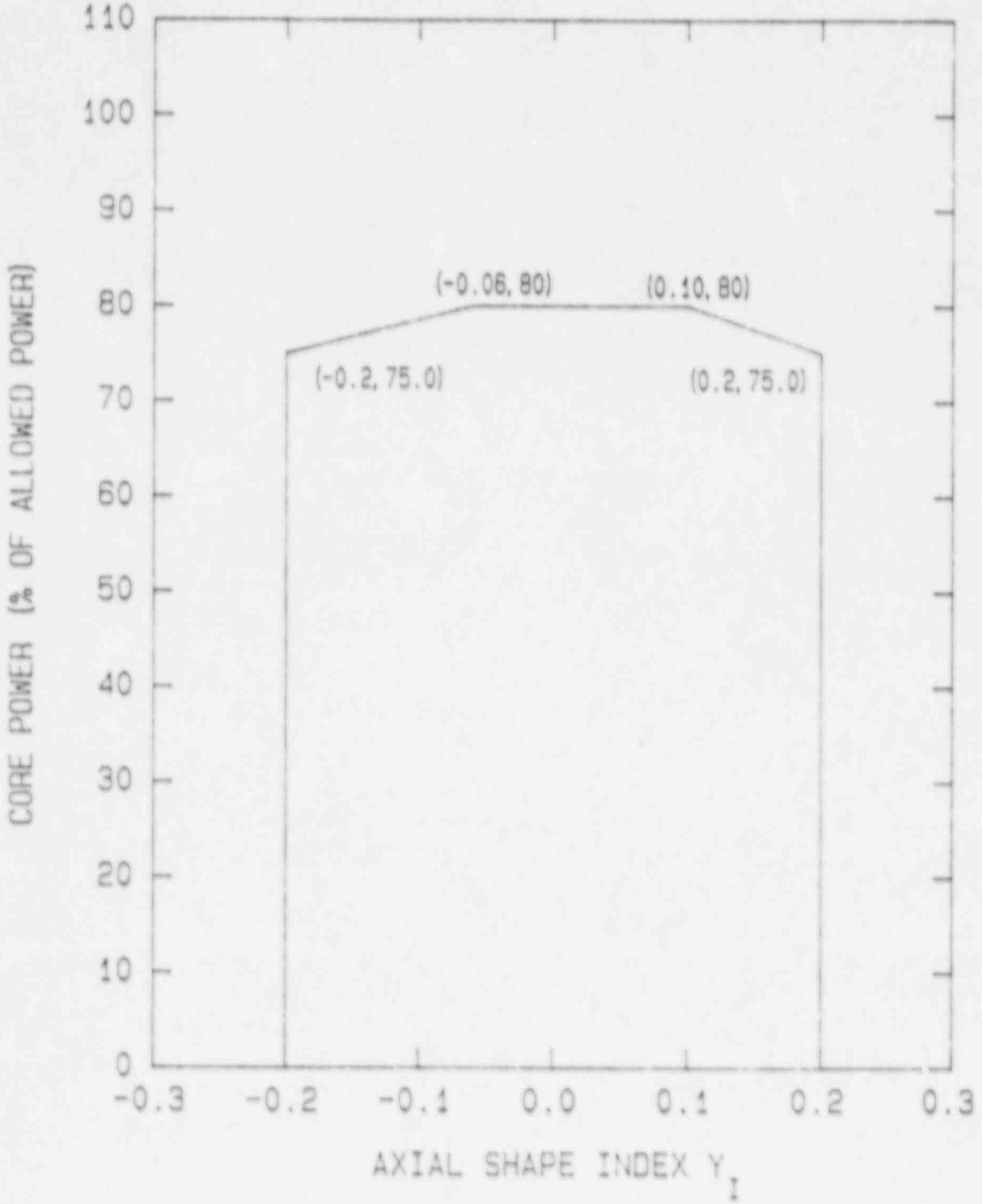


$$P_{VAR} = 29.73PF(B)A1(Y)B + 18.44T_{IN} - 11350$$

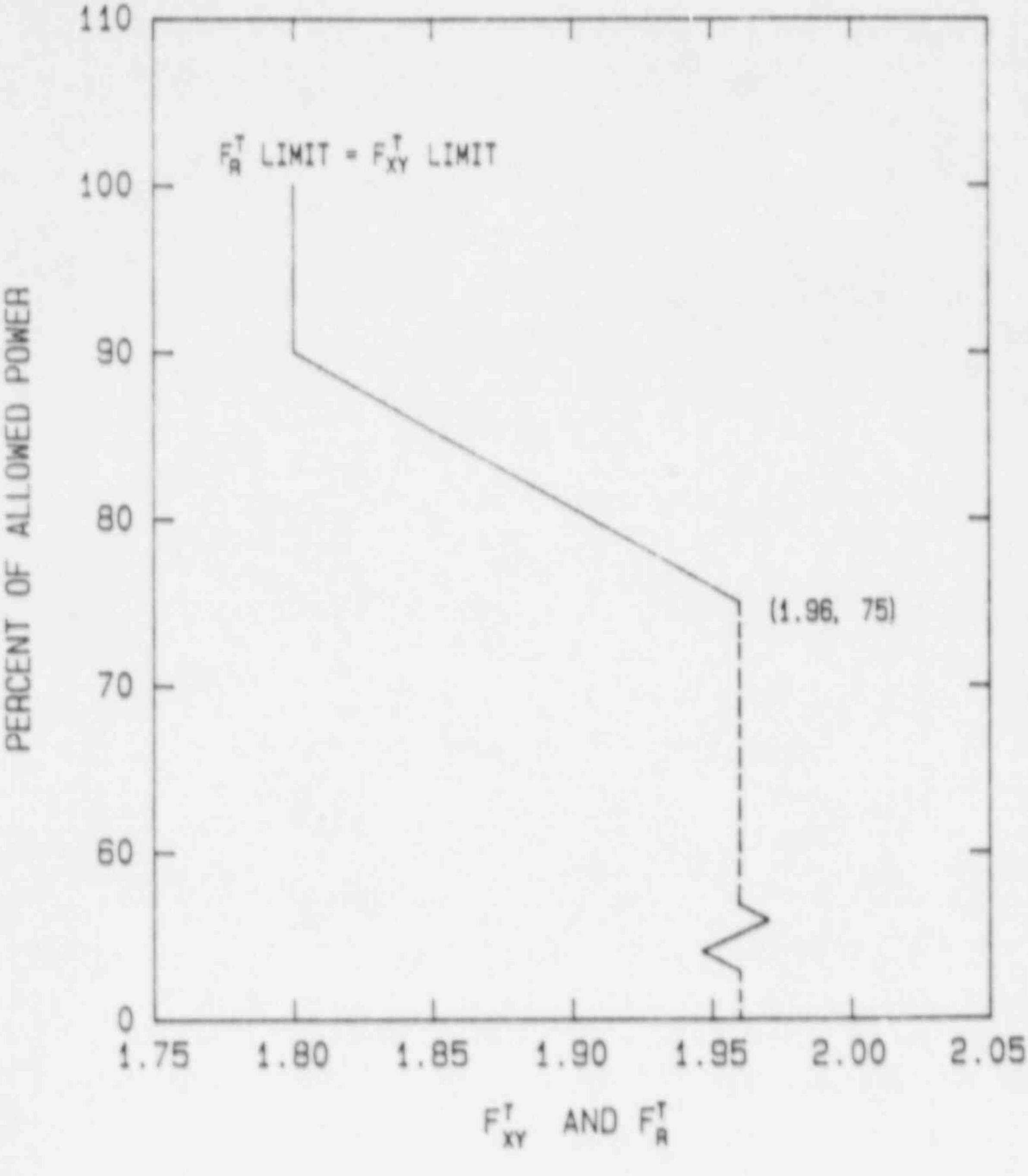
$$\begin{aligned} PF(B) &= 1.0 \quad B \geq 100\% \\ &= -.008B + 1.8 \quad 50\% < B < 100\% \\ &= 1.4 \quad B \leq 50\% \end{aligned}$$

$$\begin{aligned} A1(Y) &= -0.35294Y_I + 1.08824 \quad Y_I \leq .25 \\ &= 0.57143Y_I + 0.875 \quad Y_I > .25 \end{aligned}$$

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



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



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

FORT CALHOUN STATION

UNIT NO. 1

CYCLE 12

RELOAD EVALUATION

Fort Calhoun
Cycle 12
License Application

CONTENTS

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of the design and performance for the operation of Fort Calhoun Station Unit No. 1 during its twelfth fuel cycle at full rated power of 1500 MWt. All planned operating conditions remain the same as those for Cycle 11.

The core will consist of 89 presently operating J, K, L and M assemblies and 44 fresh Batch N assemblies.

The Cycle 12 analysis is based on a Cycle 11 termination point between 13,100 MWD/T and 14,100 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 12 conditions would be enveloped, provided the Cycle 11 termination point falls within the above burnup range. In accordance with Reference 1, the fuel burnup limitations on Batch K fuel further restrict the Cycle 11 upper bound to 13,860 MWD/MTU. The analysis presented herein will accommodate a Cycle 12 length of up to 13,450 MWD/T.

The evaluation of the reload core characteristics have been conducted with respect to the Fort Calhoun Unit No. 1 Cycle 11 safety analysis described in the 1987 update of the USAR, hereafter referred to as the "reference cycle" in this report unless noted otherwise.

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 12 core has been designed to reduce fluence to critical reactor pressure vessel welds to minimize the RT_{PTS} shift of these welds. This will preclude the reactor vessel welds reaching the Pressurized Thermal Shock RT_{PTS} screening criteria of the current 10 CFR 50.61 regulations and maximize the time to reaching the screening criteria if the Reg. Guida 1.99, Rev. 02, methods are used to revise 10 CFR 50.61.

The analysis presented in this report was performed utilizing the methodology documented in the District's reload analysis methodology reports (References 1, 2, and 3). These methodologies were previously transmitted in References 4, 5 and 6.

2.0 OPERATING HISTORY OF THE PREVIOUS CYCLE

Fort Calhoun Station is presently operating in its eleventh fuel cycle utilizing Batch H, I, J, K, L and M fuel assemblies. Fort Calhoun Cycle 11 operation began on June 8, 1987, and reached full power on June 30, 1987. 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 11 will be terminated on or about September 23, 1988. The Cycle 11 termination point can vary between 13,100 MWD/T and 14,100 MWD/T and still be within the assumptions [illegible] the Cycle 12 analyses. In accordance with Reference 1, the fuel bu[illegible]up limitations on Batch K fuel further restrict the Cycle 11 upper bou[illegible] to 13,860 MWD/MTU. As of July 24, 1988, the Cycle 11 burnup had reached 12,334 MWD/T.

3.0 GENERAL DESCRIPTION

The Cycle 12 core will consist of the number and type of assemblies and fuel batches shown in Table 3-1. One H assembly, one I assembly, 6 J assemblies, 21 K assemblies and 15 L assemblies will be discharged this outage. They will be replaced by 20 fresh unshimmed Batch N assemblies (3.70 w/o enrichment) and 24 fresh shimmed Batch N assemblies (3.70 w/o, 0.020 gm B_{10}/inch).

Figure 3-1 shows the fuel management pattern to be employed in Cycle 12. The primary change to the core in Cycle 12 is the reduction of the initial enrichment by 0.1 w/o of U-235. The locations of the poison pins within the lattice of shimmed assemblies and the fuel rod locations in unshimmed assemblies are shown in Figure 3-2.

Figure 3-3 shows the beginning of Cycle 12 assembly burnup distribution for a Cycle 11 termination burnup of 13,600 MWD/T. The fuel average discharge exposure at the end of Cycle 11 is projected to be 38,211 MWD/T. The initial enrichment of the fuel assemblies is also shown in Figure 3-3. Figure 3-4 shows the end of Cycle 12 assembly burnup distribution. The end of Cycle 12 core average exposure is approximately 29,224 MWD/T.

Table 3-1
Fort Calhoun Cycle 12
Core Loading

Assembly Designation	Number of Assemblies	BOC Average Burnup (MWD/T) [EOC 11 = 13,600 MWD/T]	EOC Average Burnup (MWD/T) [EOC 12 = 13,450 MWD/T]	Poison Rods per Assembly	Initial Poison Loading gm B_{10}/inch
J*(1)	8	34,858	39,819	0	0
K	8	39,188	43,823	0	0
L	21	25,196	39,129	0	0
L/	8	32,300	45,812	8	.01904
M	20	14,817	26,801	0	0
M/	24	17,792	33,361	8	.024
N	20	0	14,087	0	0
N/	24	0	17,502	8	.020
TOTAL	133				

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

FIGURE 3-1
FORT CALHOUN STATION CYCLE 12
CORE LOADING PATTERN

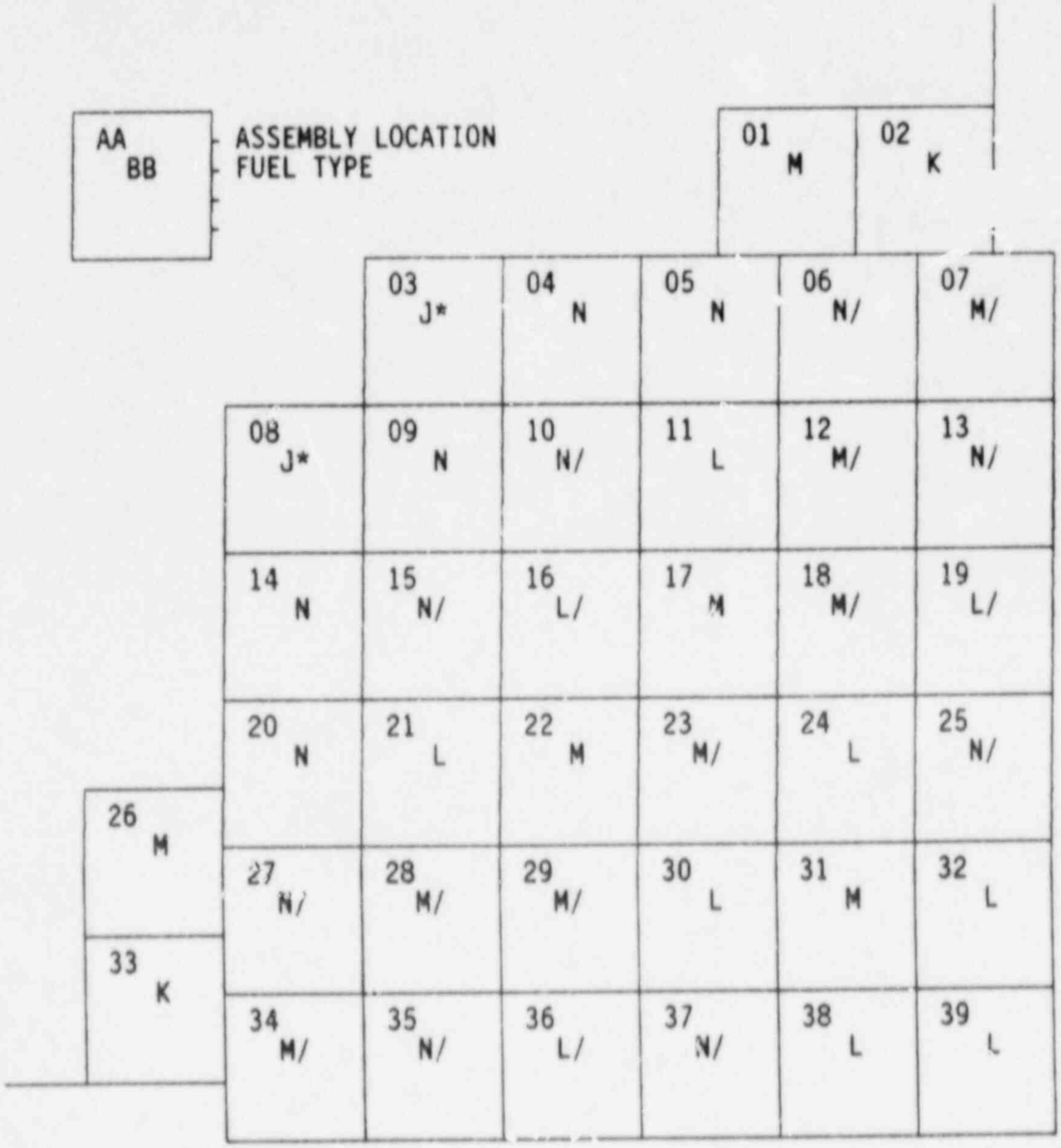
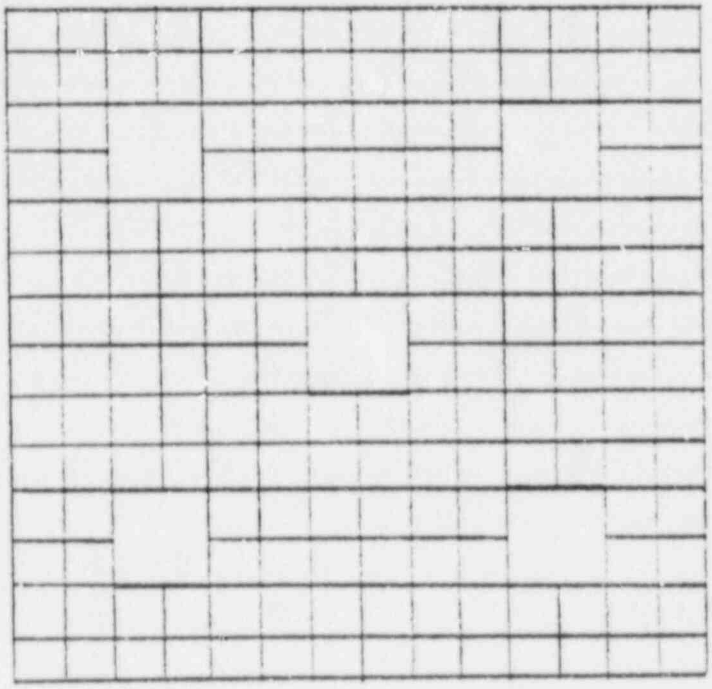
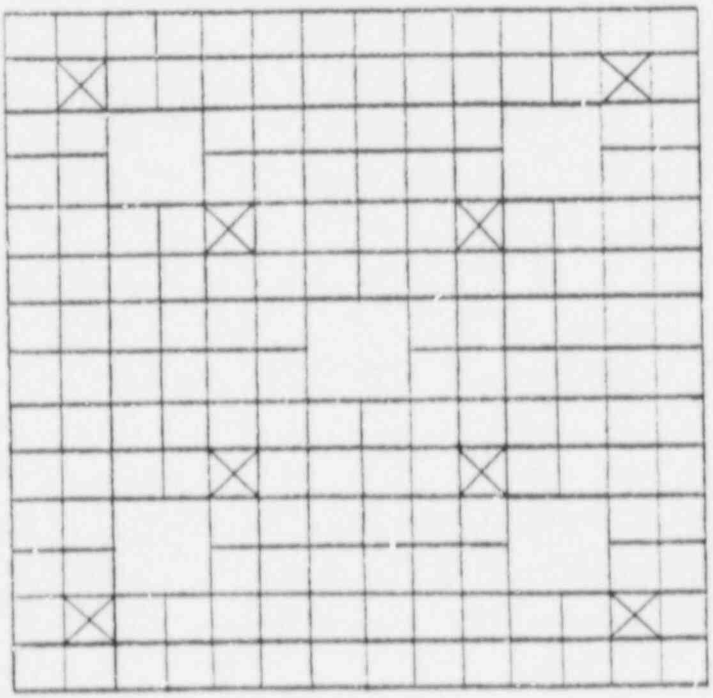


FIGURE 3-2
FORT CALHOUN STATION CYCLE 12
ASSEMBLY FUEL AND POISON
ROD LOCATIONS

UNSHIMMED ASSEMBLY



L/, M/, N/ - 8 POISON RODS PER ASSEMBLY



FUEL ROD LOCATION

POISON ROD LOCATION

FIGURE 3-3, FORT CALHOUN STATION
CYCLE 12 BOC ASSEMBLY AVERAGE EXPOSURE
AND INITIAL ENRICHMENT

AA	ASSEMBLY LOCATION
BB	FUEL TYPE
C.CC	ENRICHMENT (W/O U-235)
DD,DDD	ASSY AVG EXP (MWD/T)

					01 M 3.80 15,523	02 K 3.50 39,644
		03 J* 3.50 34,843	04 N 3.70 0	05 N 3.70 0	06 N/ 3.70 0	07 M/ 3.80 19,293
	08 J* 3.50 34,874	09 N 3.70 0	10 N/ 3.70 0	11 L 3.80 28,353	12 M/ 3.80 18,505	13 N/ 3.70 0
	14 N 3.70 0	15 N/ 3.70 0	16 L/ 3.80 31,476	17 M 3.80 13,403	18 M/ 3.80 16,553	19 L/ 3.80 33,028
	20 N 3.70 0	21 L 3.80 27,926	22 M 3.80 13,419	23 M/ 3.80 17,257	24 L 3.80 20,392	25 N/ 3.70 0
26 M 3.80 15,549	27 N/ 3.70 0	28 M/ 3.80 18,532	29 M/ 3.80 16,569	30 L 3.80 20,741	31 M 3.80 16,189	32 L 3.80 26,067
33 K 3.50 38,732	34 M/ 3.80 19,383	35 N/ 3.70 0	36 L/ 3.80 33,218	37 N/ 3.70 0	38 L 3.80 28,876	39 L 3.80 29,587

NOTE: EOC 11 CORE AVERAGE
BURNUP = 13,600 MWD/T

FIGURE 3-4
FORT CALHOUN STATION CYCLE 12
EOC ASSEMBLY AVERAGE EXPOSURE

AA	ASSEMBLY LOCATION
BB	FUEL TYPE
CC,CCC	ASSY AVG EXP (MWD/T)

				01 M 21,710	02 K 44,234	
		03 J* 39,797	04 N 12,569	05 N 15,024	06 N/ 15,825	07 M/ [illegible]2,412
	08 J* 39,841	09 N 15,213	10 N/ 18,019	11 L 41,879	12 M/ 34,344	13 N/ 18,792
	14 N 12,581	15 N/ 18,015	16 L/ 45,119	17 M 29,645	18 M/ 32,739	19 L/ 46,429
	20 N 15,047	21 L 41,533	22 M 29,641	23 M/ 33,527	24 L 35,561	25 N/ 18,592
26 M 21,763	27 N/ 15,863	28 M/ 34,376	29 M/ 32,710	30 L 35,774	31 M 31,245	32 L 39,232
33 K 43,411	34 M/ 32,526	35 N/ 18,790	36 L/ 46,580	37 N/ 18,401	38 L 41,517	39 L 41,215

4.0 FUEL SYSTEMS DESIGN

The mechanical design for the Batch N fuel is essentially the same as the Batch M fuel supplied by Combustion Engineering, Inc. in Cycle 11.

The Batch N fuel is similar in design to the Batch G fuel supplied by Combustion Engineering in Cycle 5 and is mechanically, thermally, and hydraulically compatible with the Advanced Nuclear Fuel (ANF) supplied fuel remaining in the core. Reference 2 describes the Batch M fuel characteristics and design. This report was previously transmitted in Reference 3. References 4 and 5 remain valid for describing the design of the ANF-supplied fuel. Thirty-six (36) total fuel pins from Batch K (20) and Batch L (16) are projected to exceed the burnup limits established in Cycle 11 for extended burnup of ANF fuel.

ANF has reanalyzed the subject fuel pins in accordance with References 5 and 6 and established that operation of the fuel pins in Cycle 12 will not violate any of the extended burnup criteria. The maximum burnup for these pins was increased from 49,000 MWD/MTU to 50,000 MWD/MTU for Batch K and from 51,600 MWD/MTU to 52,000 MWD/MTU for Batch L fuel.

5.0 NUCLEAR DESIGN

5.1 PHYSICAL CHARACTERISTICS

5.1.1 Fuel Management

The Cycle 12 fuel management uses a low-radial leakage design, with once, twice, and thrice 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 and achieve the maximum in neutron economy. Use of this type of fuel management to achieve reduced pressure vessel flux over a standard out-in-in pattern results in higher radial peaking factors. The peaking factors for Cycle 12 are consistent with previous cycles in which low radial leakage patterns have been utilized.

As described in Section 3.0, the Cycle 12 loading pattern incorporated 44 fresh Batch N assemblies (24 shimmed N/ and 20 unshimmed N) with an enrichment of 3.70 w/o. Eight thrice burned Batch J* assemblies, which were delivered for Cycle 8, but initially loaded into the core for Cycle 9, are being returned to the core to be combined with 8 thrice burned K assemblies, 29 twice burned L assemblies, and 44 once burned M assemblies to produce a Cycle 12 pattern with a cycle energy of 13,450 ± 500 MWD/T. The Cycle 12 core characteristics have been examined for a Cycle 11 termination between 13,100 MWD/T and 14,100 MWD/T and limiting values established for the safety analysis. The Cycle 12 loading pattern is valid for any Cycle 11 endpoint between these values.

Physics characteristics including reactivity coefficients for Cycle 12 are listed in Table 5-1 along with the corresponding values from Cycle 11. It should be noted that the values of paramaters 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 beginning of Cycle 12 Hot Zero Power Steam Line Break accident. The BOC HZP SLB is the most limiting accident of those used in the determination of the required shutdown margin. The Cycle 12 values, calculated for minimum scram worth, exceed the minimum value required Technical Specifications 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 BOC12, MOC12 and ECC12, respectively, and are characteristic of the high burnup end of the Cycle 11 shutdown window. These planar radial power peaks are representative of the major portion of the active core length between about 20 and 80 percent of the fuel height. The high burnup end of the Cycle 11 shutdown window tends to increase the power peaking in this axial central region of the core for Cycle 12. The planar radial power distributions for the above region, with Bank 4 fully inserted at beginning and end of Cycle 12, are shown in Figures 5-4 and 5-5, respectively.

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 12. 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 13,450 MWD/T, based on an EOC11 burnup of 13,600 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 12 during normal base load, all rods out operation at full power is 1.97, 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 12. These values encompass the worst conditions anticipated during Cycle 12 for any expected Cycle 11 termination point. The values shown for Cycle 12 are calculated in accordance with Reference 7. In addition, Table 5-3 lists those values used in the Reference Analysis (Cycle 11) for comparison.

5.0 NUCLEAR DESIGN (Continued)

5.1 PHYSICAL CHARACTERISTICS (Continued)

5.1.3 Safety Related Data (Continued)

5.1.3.2 Dropped CEA Data

The Cycle 12 safety related data for the dropped CEA analysis were calculated identically with the methods used in Cycle 11.

5.2 ANALYTICAL INPUT TO INCORE MEASUREMENTS

Incore detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the same manner as for Cycle 11.

5.3 NUCLEAR DESIGN METHODOLOGY

Analyses have been performed in the manner and with the methodologies documented in References 8 and 9.

5.4 UNCERTAINTIES IN MEASURED POWER DISTRIBUTIONS

The power distribution measurement uncertainties which are applied to Cycle 12 are the same as those presented in Reference 9.

TABLE 5-1

FORT CALHOUN CYCLE 12
NOMINAL PHYSICS CHARACTERISTICS

	Units	Cycle 11	Cycle 12
Critical Boron Concentration			
Hot Full Power, ARO, Equilibrium Xenon, BOC	PPM	1081	1081
Inverse Boron Worth			
Hot Full Power, BOC	PPM/%$\Delta\rho$	113	113
Hot Full Power, EOC	PPM/%$\Delta\rho$	90	90
Reactivity Coefficients (CEAs Withdrawn)			
Moderator Temperature Coefficients			
Beginning of Cycle, HZP	$10^{-4}\Delta\rho/^\circ F$	+0.23	+0.25
End of Cycle, HFP	$10^{-4}\Delta\rho/^\circ F$	-2.47	-2.49
Doppler Coefficient			
Hot Zero Power, BOC	$10^{-5}\Delta\rho/^\circ F$	-1.96	-1.97
Hot Full Power, BOC	$10^{-5}\Delta\rho/^\circ F$	-1.42	-1.47
Hot Full Power, EOC	$10^{-5}\Delta\rho/^\circ F$	-1.54	-1.57
Total Delayed Neutron Fraction, βeff			
BOC		0.00609	0.00607
EOC		0.00522	0.00521
Neutron Generation Time, ℓ^*			
BOC	10^{-6} sec	22.3	22.2
EOC	10^{-6} sec	28.0	28.0

TABLE 5-2

FORT CALHOUN UNIT 1 CYCLE 12 LIMITING VALUES OF REACTIVITY WORTHS AND ALLOWANCES FOR HOT ZERO POWER STEAM LINE BREAK, %$\Delta\rho$

		Cycle 11 (EOC)	Cycle 12 (EOC)
1.	Worth of all CEA's Inserted	10.07	8.70
2.	Stuck CEA Allowance	2.80	1.42
3.	Worth of all CEA's Less Worth of Most Reactive CEA Stuck Out	7.27	7.28
4.	Power Dependent Insertion Limit CEA Worth	1.35	1.41
5.	Calculated Scram Worth	5.92	5.87
6.	Physics Uncertainty plus Bias	0.59	0.59
7.	Net Available Scram Worth	5.33	5.28
8.	Technical Specification Shutdown Margin	4.00	4.00
9.	Margin in Excess of Technical Specification Shutdown Margin	1.33	1.28

TABLE 5-3

FORT CALHOUN UNIT 1 CYCLE 12
CEA EJECTION DATA

	BOC11 Value	EOC11 Value	BOC12 Value	EOC12 Value
Maximum Radial Power Peaking Factor				
Full Power with Bank 4 inserted; worst CEA ejected	3.74	3.21	2.38	2.15
Zero power with Banks 4+3 inserted; worst CEA ejected	5.74	5.27	4.85	4.82
Maximum Ejected CEA Worth (%$\Delta\rho$)				
Full power with Bank 4 inserted; worst CEA ejected	0.39	0.38	0.39	0.29
Zero Power with Banks 4+3 inserted; worst CEA ejected	0.65	0.66	0.56	0.62

FIGURE 5-1, FORT CALHOUN STATION
CYCLE 12 ASSEMBLY RELATIVE POWER DENSITY
0 MWD/T, HOT FULL POWER, EQ. XENON

AA	ASSEMBLY LOCATION
B.BBBB	ASSEMBLY RPD'S
C.CCC	MAXIMUM 1-PIN PEAK ASSY

					01 .4057	02 .2778
		03 .3389	04 .9425	05 1.1267	06 1.0903	07 .9019
	08 .3402	09 1.1497	10 1.3332 1.670	11 1.0026	12 1.1812	13 1.3379
	14 .9447	15 1.3338	16 1.0213	17 1.2845	18 1.2570	19 .9847
	20 1.1305	21 1.0111	22 1.2831	23 1.3089	24 1.1807	25 1.4149
26 .4083	27 1.0949	28 1.1831	29 1.2532	30 1.1682	31 1.1973	32 1.0094
33 .2839	34 .9052	35 1.3392	36 .9830	37 1.3966	38 .9638	39 .8824

MAXIMUM 1-PIN PEAK AT 30% CORE HEIGHT

FIGURE 5-2, FORT CALHOUN STATION
CYCLE 12 ASSEMBLY RELATIVE POWER DENSITY
7000 MWD/T, HOT FULL POWER, EQ. XENON

AA	ASSEMBLY LOCATION
B.BBBB	ASSEMBLY RPD'S
C.CCC	MAXIMUM 1-PIN PEAK ASSY

				01 .4691	02 .3423	
		03 .3655	04 .9401	05 1.1230	06 1.2012	07 .9999
	08 .3664	09 1.1383	10 1.3587 1.658	11 .9932	12 1.1933	13 1.4252
	14 .9407	15 1.3582	16 .9990	17 1.2033	18 1.2079	19 .9826
	20 1.1242	21 .9987	22 1.2017	23 1.2034	24 1.1000	25 1.3875
26 .4710	27 1.2037	28 1.1932	29 1.2043	30 1.0903	31 1.1098	32 .9543
33 .3488	34 1.0014	35 1.4245	36 .9794	37 1.3733	38 .9169	39 .8422

MAXIMUM 1-PIN PEAK AT 30% CORE HEIGHT

FIGURE 5-3, FORT CALHOUN STATION
CYCLE 12 ASSEMBLY RELATIVE POWER DENSITY
13,450 MWD/T, HOT FULL POWER, EQ. XENON

AA – ASSEMBLY LOCATION B.BBBB – ASSEMBLY RPD'S C.CCC – MAXIMUM 1-PIN PEAK ASSY				01 .5289	02 .4021	
		03 .3975	04 .9507	05 1.1275	06 1.2681	07 1.0504
	08 .3982	09 1.1394	10 1.3610	11 .9867	12 1.1765	13 1.4293 1.691
	14 .9510	15 1.3605	16 .9868	17 1.1487	18 1.1567	19 .9694
	20 1.1283	21 .9913	22 1.1478	23 1.1340	24 1.0521	25 1.3520
26 .5307	27 1.2702	28 1.1768	29 1.1550	30 1.0452	31 1.0699	32 .9369
33 .4091	34 1.0520	35 1.4294	36 .9676	37 1.3439	38 .9054	39 .8414

MAXIMUM 1-PIN PEAK AT 22% CORE HEIGHT

FIGURE 5-4, FORT CALHOUN STATION
CYCLE 12 ASSY RPD'S WITH BANK 4 INSERTED
0 MWD/T, HOT FULL POWER, EQ. XENON

AA	ASSEMBLY LOCATION
B.BBBB	ASSEMBLY RPD'S
C.CCC	MAXIMUM 1-PIN PEAK ASSY
XXXXXX	CEA BANK 4 LOCATION

					01 .4391	02 .3104
		03 .2086	04 .7909	05 1.1327	06 1.1949	07 1.0154
	08 .2096	09 .4724 XXXXXX	10 1.0708	11 1.0101	12 1.3006	13 1.5092
	14 .7934	15 1.0718	16 .9434	17 1.3474	18 1.3917	19 1.1074
	20 1.1373	21 1.0191	22 1.3462	23 1.4225	24 1.2998	25 1.5573 1.797
26 .4422	27 1.2005	28 1.3030	29 1.3877	30 1.2863	31 1.2637	32 1.0030
33 .3174	34 1.0196	35 1.5112	36 1.1059	37 1.5378	38 .9575	39 .5292 XXXXXX

MAXIMUM 1-PIN PEAK AT 50% CORE HEIGHT

FIGURE 5-5, FORT CALHOUN STATION
CYCLE 12 ASSY RPD'S WITH BANK 4 INSERTED
13,600 MWD/T, HOT FULL POWER, EQ. XENON

AA	ASSEMBLY LOCATION
B.BBBB	ASSEMBLY RPD'S
C.CCC	MAXIMUM 1-PIN PEAK ASSY
XXXXXX	CEA BANK 4 LOCATION

				01 .5873	02 .4595	
		03 .2401	04 .7905	05 1.1493	06 1.4172	07 1.2036
	08 .2406	09 .4500 XXXXXX	10 1.0763	11 .9925	12 1.3088	13 1.6326 1.937
	14 .7911	15 1.0762	16 .8924	17 1.1940	18 1.2828	19 1.0948
	20 1.1510	21 .9975	22 1.1935	23 1.2241	24 1.1525	25 1.4820
26 .5898	27 1.4206	28 1.3098	29 1.2814	30 1.1452	31 1.1134	32 .9099
33 .4679	34 1.2062	35 1.6336	36 1.0933	37 1.4741	38 .8792	39 .4721 XXXXXX

MAXIMUM 1-PIN PEAK AT 22% CORE HEIGHT

6.0 THERMAL-HYDRAULIC DESIGN

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 12 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 through 11 Fort Calhoun reload analyses (References 4 through 7) and the reload methodology can be found in Reference 8.

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 9) to define the design limit on CE-1 minimum DNBR.

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 45,000 MWD/MTU burnup is 0.5% in MDNBR. This penalty was applied to the MDNBR design limit of 1.18 (References 6 and 10) in the statistical combination of uncertainties (Reference 9).

TABLE 6-1

Fort Calhoun Unit 1

Thermal-Hydraulic Parameters at Full Power

	Unit	Cycle 12*
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	545
Total Reactor Coolant Flow	gpm	202,500
(Steady State)	10^6 lbm/hr	76.49
(Through the Core)	10^6 lbm/hr	73.08
Hydraulic Diameter (Nominal Channel)	ft	.044
Average Mass Velocity	10^6 lbm/hr-ft^2	2.24
Core Average Heat Flux (Accounts for Heat Generated in Fuel Rod)	BTU/hr-ft^2	181,189
Total Heat Transfer Surface Area	ft^2	28,255**
Average Core Enthalpy Rise	BTU/lbm	70.5
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 12 specific value of 448 shims.

***These factors were combined statistically (Reference 8) 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 12 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 12 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 (i.e., conservative with respect to) of the reference cycle values (Fort Calhoun Updated Safety Analysis Report including Cycle 11 analyses, Reference 1). For these DBEs the results and conclusions quoted in the reference cycle analysis remain valid for Cycle 12.

For those analyses indicated as reviewed, calculations were performed in accordance with Reference 6 until a 10 CFR 50.59 determination could be made that Cycle 12 results would be bounded by Cycle 11.

All events were evaluated for up to a total of 6% steam generator tube plugging in Cycle 11. Fort Calhoun Station currently has 1.08% steam generator tubes plugged, thus; no additional analysis is required.

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 12
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	Reviewed[5]
	7.1.2	Excess Load	Reviewed
	7.1.3	Reactor Coolant System Depressurization	Reviewed
	7.1.4	Loss of Load	Not 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	Startup of an Inactive Reactor Coolant Pump	Not Reanalyzed[1]
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	Reanalyzed[2]
	7.2.2	Loss of Coolant Flow	Reanalyzed[3,5]
	7.2.3	CEA Drop	Reviewed
	7.2.4	Transients Resulting from the Malfunction of One Steam Generator	Not Reanalyzed[4]
7.3	Postulated Accidents		
	7.3.1	CEA Ejection	Reviewed[5]
	7.3.2	Steam Line Break	Reviewed[5]
	7.3.3	Steam Generator Tube Rupture	Not Reanalyzed
	7.3.4	Seized Rotor	Reviewed[3,5]

NOTE: All events evaluated or reanalyzed for the effect of increased steam generator tube plugging to 6%/SG.

[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]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 12
CORE PARAMETERS INPUT TO SAFETY ANALYSES
FOR DNB AND CTM (CENTERLINE TO MELT) DESIGN LIMITS

Physics Parameters	Units	Cycle 11 Values	Cycle 12 Values
Radial Peaking Factors			
For DNB Margin Analyses (F_R^T)			
Unrodded Region		1.80*	1.80*
Bank 4 Inserted		1.98*	1.90*
For Planar Radial Component (F_{xy}^T) of 3-D Peak (CTM Limit Analyses)			
Unrodded Region		1.85*	1.85*
Bank 4 Inserted		2.04*	1.94*
Maximum Augmentation Factor		1.000	1.000
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{\circ}F$	-2.7 to +0.5	-2.7 to +0.5
Shutdown Margin (Value Assumed in Limiting EOC Zero Power SLB)		-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 2-5.

TABLE 7-2
(Continued)

Safety Parameters	Units	Cycle 11 Values	Cycle 12 Values
Power Level	MWt	1530*	1530*
Maximum Steady State Temperature	°F	547*	547*
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)	Ip	-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	22.0	22.0
CEA Drop Time to 100% Including Holding Coil Delay	sec	3.1	3.1
Minimum DNBR (CE-1)		1.18*	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 2-5.

TABLE 7-3

DESIGN BASIS EVENT REANALYZED FOR FORT CALHOUN CYCLE 12

Event	Reason for Reanalysis	Acceptance Criterion	Summary of Results
Sequential CEA Group Withdrawal	Change in rod worth nonconservative with lower reactivity insertion rate.	Minimum [illegible] greater than [illegible] using CE-1 correlation. Transient [illegible] w/ft.	MDNBR = 6.99 (HZP) MDNBR = 1.28 (HFP) PLHGR < 22 kw/ft.
Loss of Coolant Flow	Change in rod worth nonconservative with lower reactivity insertion rate.	[illegible]	Minimum DNBR = 1.43

7.0 TRANSIENT ANALYSIS

7.1 ANTICIPATED OPERATIONAL OCCURRENCES

7.1.1 Boron Dilution Event

The Boron Dilution event was reviewed for Cycle 12 to verify that sufficient time is available for an operator to identify the cause and to terminate an approach to criticality for all subcritical modes of operation.

Table 7.1.1-1 compares the values of the key transient parameters assumed in each mode of operation for Cycle 12 and the reference cycle (Cycle 11).

As noted in this table, the critical boron concentration for Cycle 12 is less than the corresponding Cycle 11 values for all operating modes. Therefore, the time to lose critical shutdown margin will increase from Cycle 11 results due to the inverse relationship between response time and critical boron concentration. Since all criteria were met in the Cycle 11 analysis, it is concluded that the criterion for minimum time to lose prescribed shutdown margin will be met for Cycle 12.

TABLE 7.1.1-1

FORT CALHOUN CYCLE 12
KEY PARAMETERS ASSUMED IN THE BORON DILUTION ANALYSIS

Parameter	Cycle 11	Cycle 12
Critical Boron Concentration, PPM (All Rods Out, Zero Xenon)		
Mode		
Hot Standby	1580	1560
Hot Shutdown	1580	1560
Cold Shutdown - Normal RCS Volume	1480	1430
Cold Shutdown - Minimum RCS Volume*	1290	1250
Refueling	1400	1350
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	1800	1800

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

7.0 TRANSIENT ANALYSIS (Continued)

7.1 ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

7.1.2 Excess Load Event

The Excess Load event was reviewed for Cycle 12 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 12 is described in References 6 and 7. 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 analysis of this event shows that a pressure bias term of 58.4 psia is required compared to the 61.0 psia value in Cycle 11. This is greater than that input from the RCS Depressurization event, the other event for which a pressure bias term is calculated. However, the current pressure bias term from the TM/LP P_{var} equation is 65 psia which bounds the 58.4 psia calculated for Cycle 12. This yields a negative 10 CFR 50.59 result for this event.

7.0 TRANSIENT ANALYSIS (Continued)

7.1 ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

7.1.3 RCS Depressurization Event

The RCS Depressurization event was reviewed for Cycle 12 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 12 is the same as that used for Cycle 11 and is described in References 6 and 7.

The evaluation of this event shows that a pressure bias term of 25.8 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.

7.2 ANTICIPATED OPERATIONAL OCCURRENCES

7.2.1 CEA Withdrawal Event

The CEA Withdrawal event was analyzed for Cycle 12 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 6 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).

For the HFP CEAW DNBR analysis, an MTC identical to that utilized in Reference 8 and the gap thermal conductivity consistent with the assumption of Reference 6 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.

The HFP case for Cycle 12 is considered to meet the 10 CFR 50.59 criteria since the results show that the required overpower margin is less than the available overpower margin required by the Technical Specifications for DNB and PLHGR LCO's.

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 10 CFR 50.59 criteria is satisfied for the HZP event if the minimum DNBR is greater than that reported in the reference cycle.

The zero power case initiated at the limiting conditions of operation results in a minimum CE-1 DNBR of 6.99 which is less than the Cycle 11 value of 7.35. 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 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.

7.0 TRANSIENT ANALYSIS (Continued)

7.2 ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

7.2.1 CEA Withdrawal Event (Continued)

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 12
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*	547*
Pressurizer Pressure	psia	2053*	2053*
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho$/sec	+0.5	+0.5**
Doppler Coefficient Multiplier		0.85	0.85
CEA Worth at Trip	$10^{-2}\Delta\rho$	5.28	6.33
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

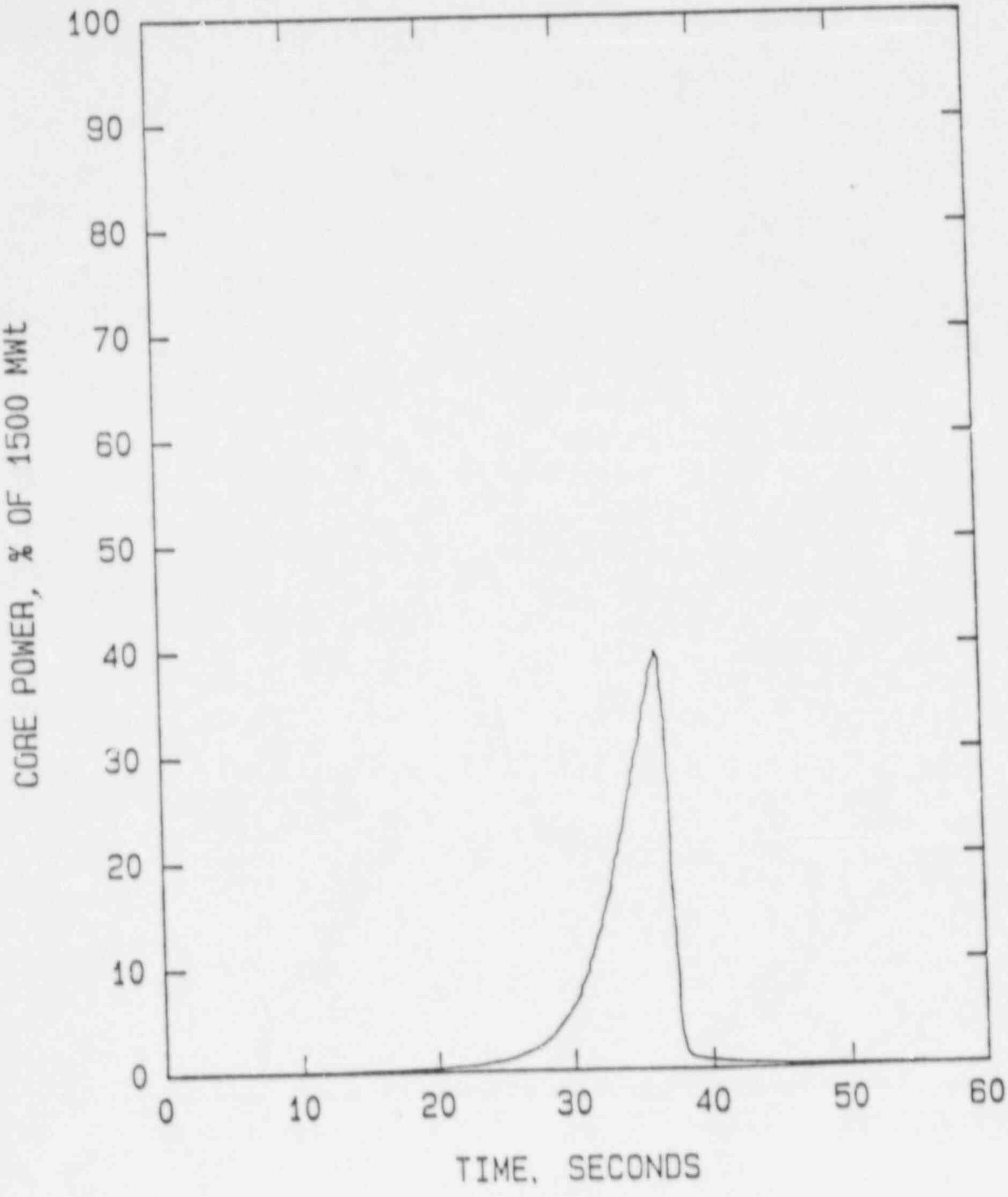
*The effects of uncertainties on these parameters were accounted for deterministically and the DNBR calculations used the methods discussed in Section 6.1 of this document and detailed in References 2-5.

**DNBR analysis assumes MTC consistent with Reference 8.

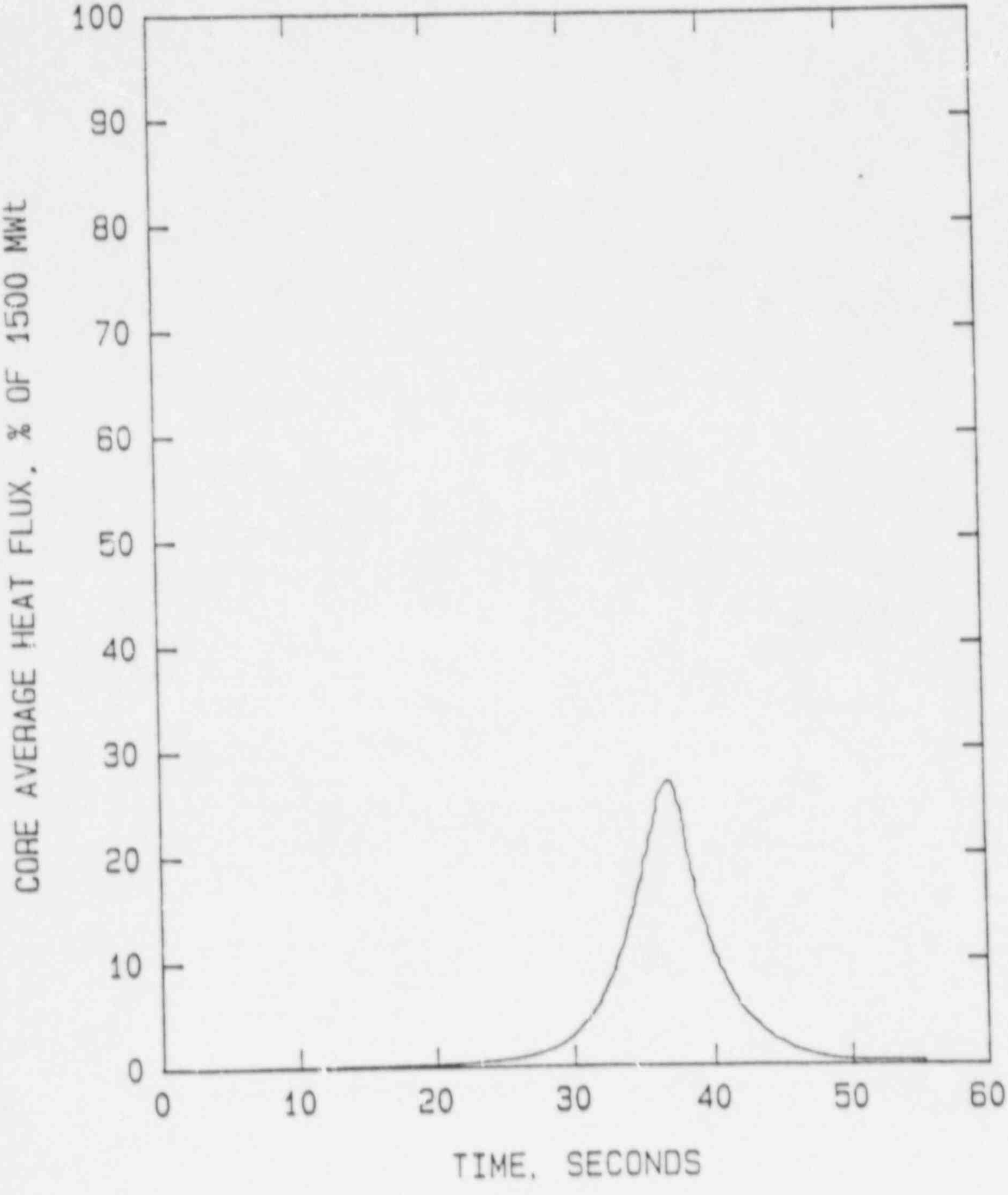
TABLE 7.2.1-2

FORT CALHOUN CYCLE 12
SEQUENCE OF EVENTS FOR
CEA WITHDRAWAL FROM ZERO POWER

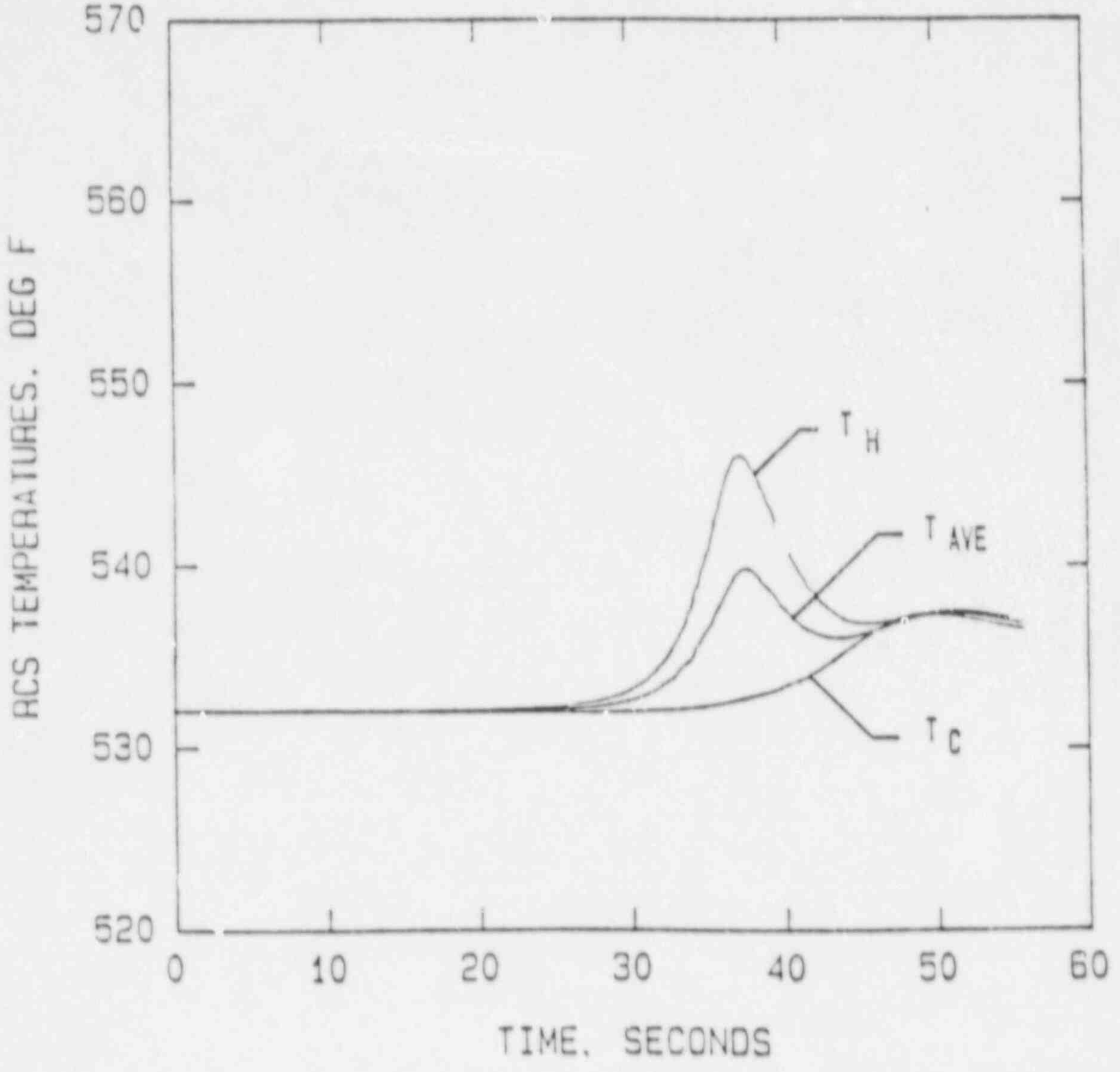
Time (sec)	Event	Setpoint or Value
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	---
33.7	Variable High Power Trip Signal Generated	29.1% of 1500 MWt
34.1	Reactor Trip Breakers Open	---
34.6	CEAs Begin to Drop Into Core	---
35.05	Maximum Core Power	41.6% of 1500 MWt
35.92	Maximum Heat Flux	28.1% of 1500 MWt
38.98	Minimum CE-1 DNBR	6.99
40.2	Maximum RCS Pressure, psia	2230



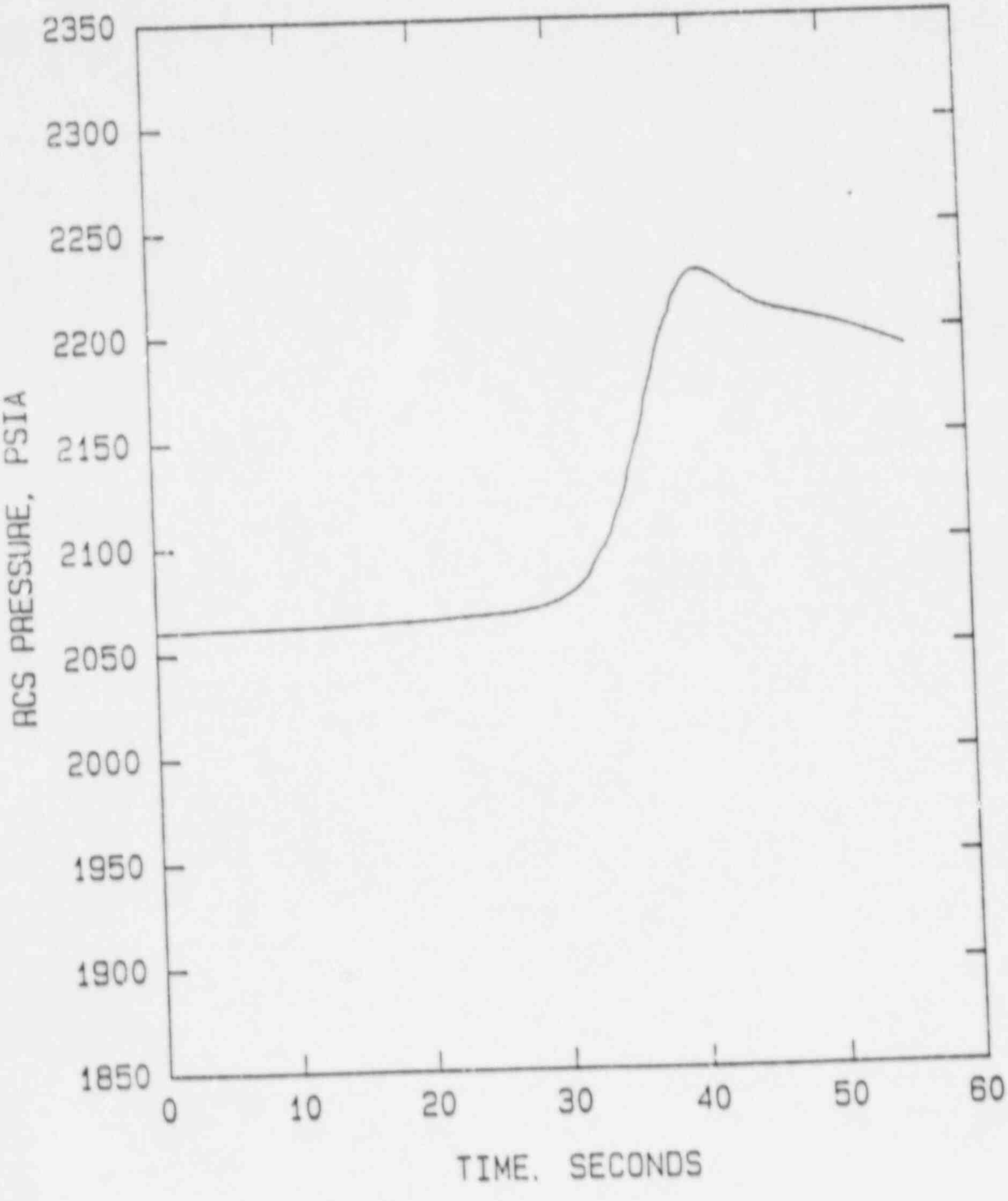
CEA Withdrawal (Zero Power) Core Power vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-1



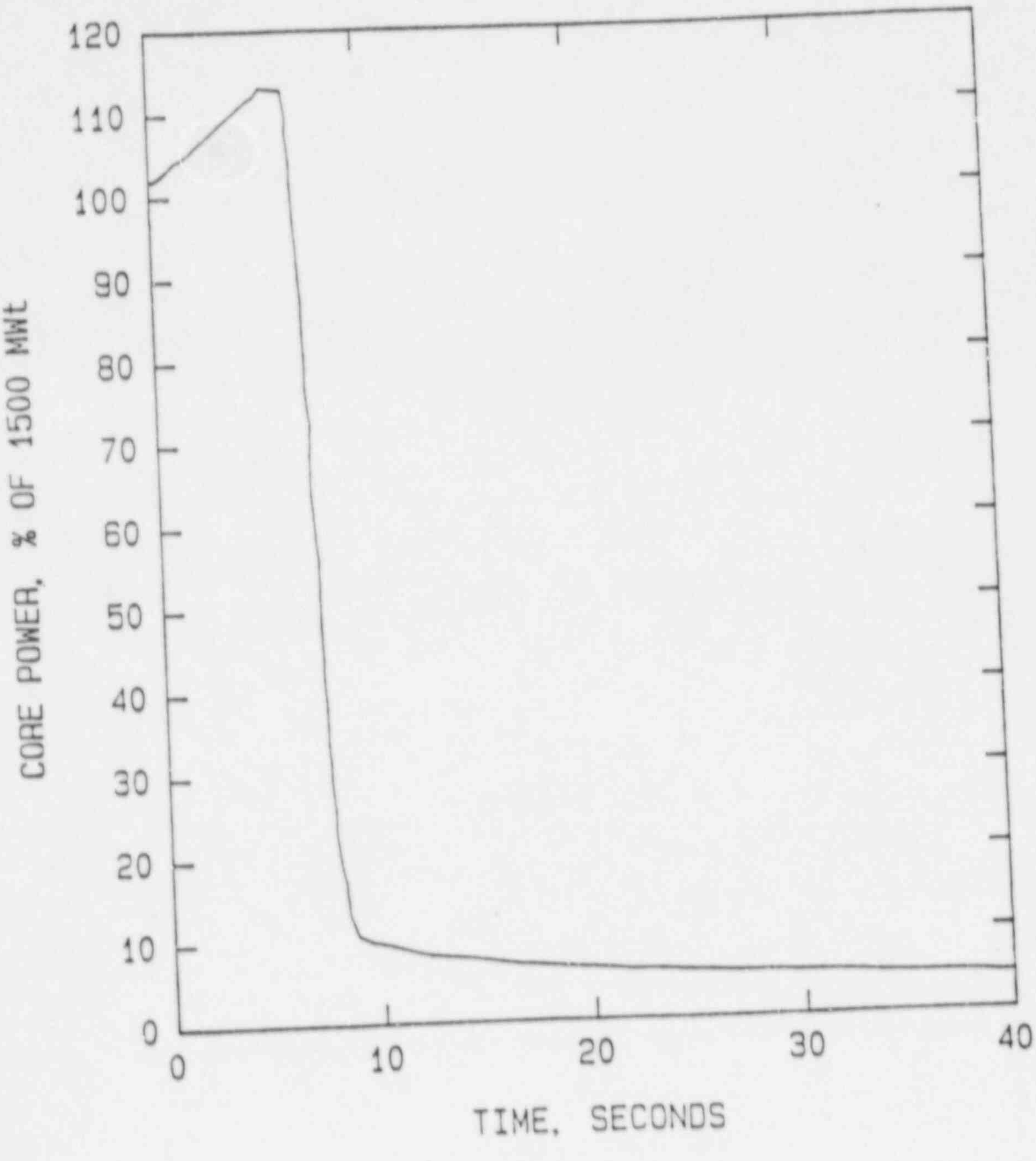
CEA Withdrawal (Zero Power) Core Average Heat Flux vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-2



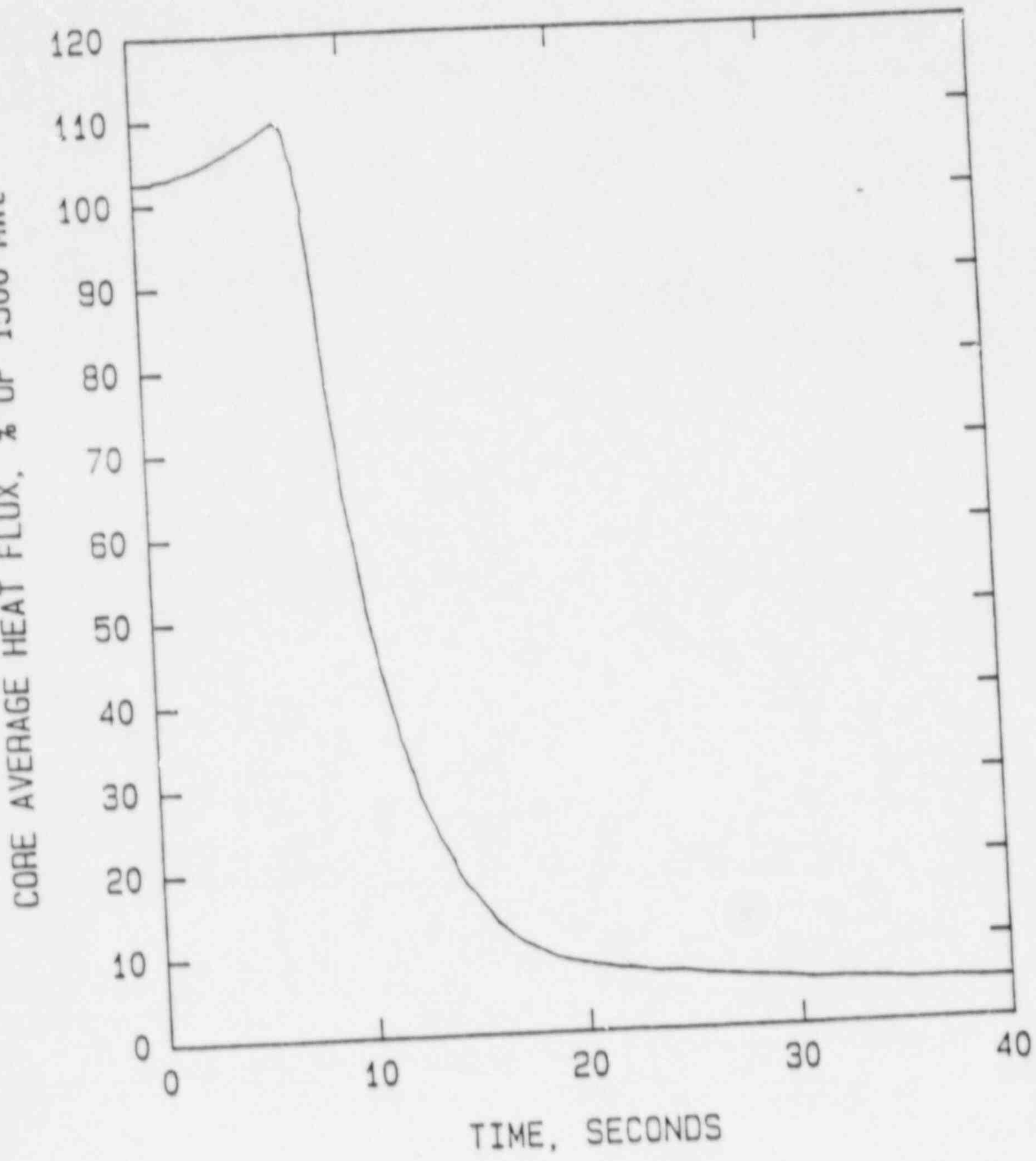
CEA Withdrawal (Zero Power) RCS Temperatures vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-3



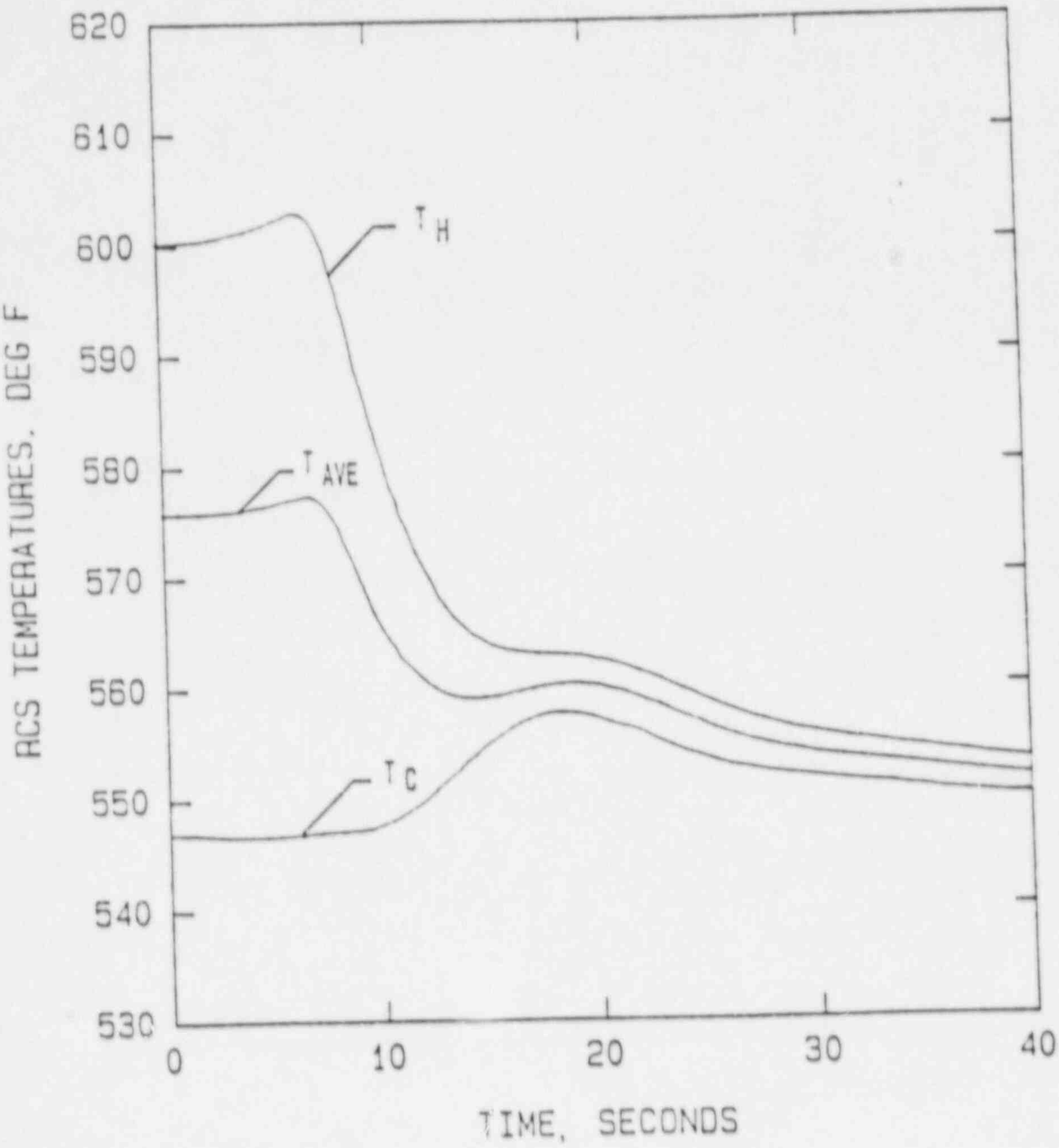
CEA Withdrawal (Zero Power) RCS Pressure vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-4



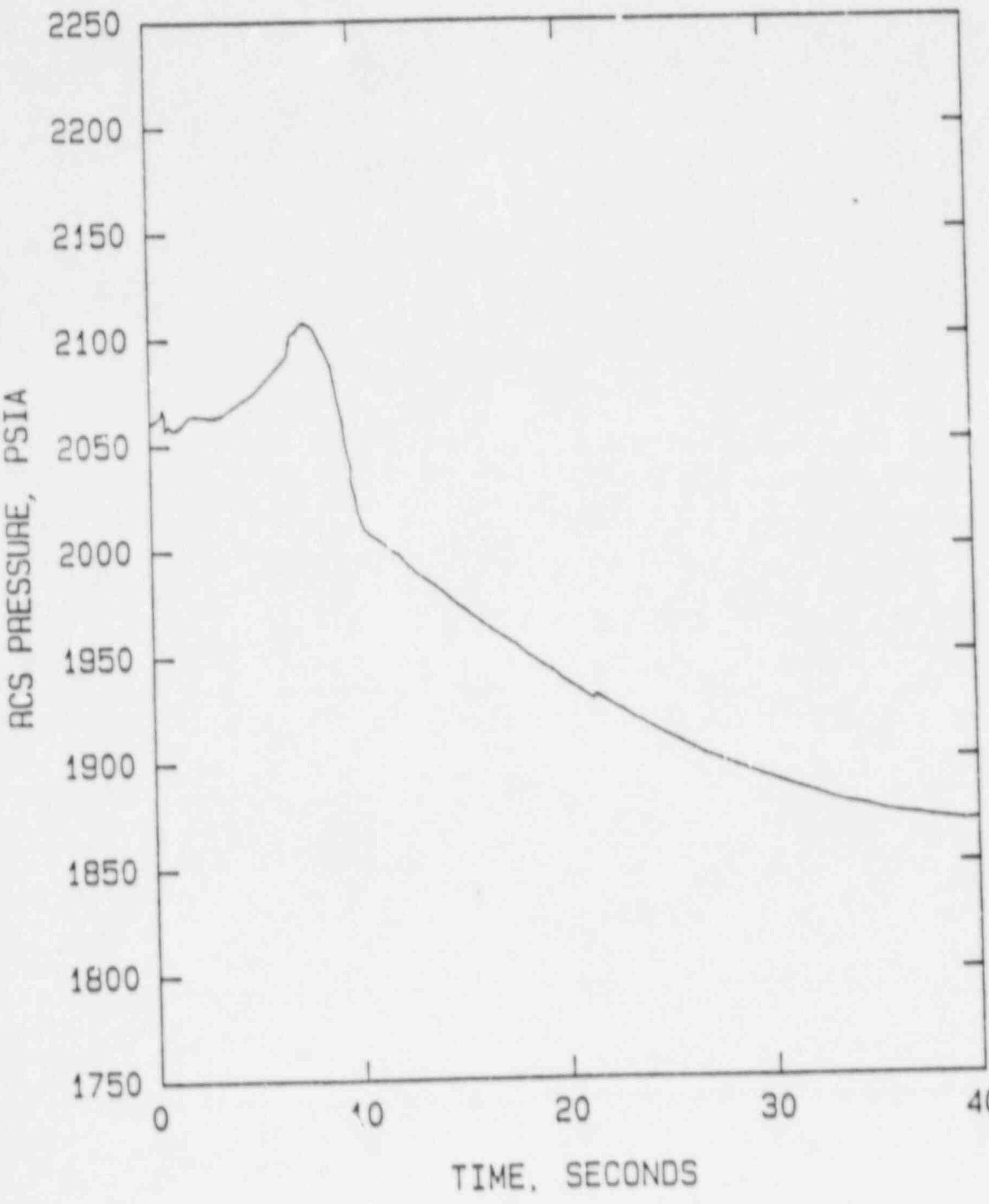
CEA Withdrawal (Full Power) Core Power vs. Time	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 7.2.1-5



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 12 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 6 (which utilizes the statistical combination of uncertainties in the DNBR analysis as described in Appendix C of References 4 and 5).

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 9 and 10) 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 12 analysis and compares them to the reference cycle (Cycle 11) values.

The low flow trip setpoint is reached at 2.80 seconds and the scram rods start dropping into the core 1.15 seconds later. A minimum CE-1 DNBR of 1.43 is reached at 4.56 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 for Cycle 12 the Loss of Flow event when initiated from the Tech. Spec. LCOs in conjunction with the Low Flow Trip, will not exceed the minimum DNBR design limit.

TABLE 7.2.2-1

FORT CALHOUN CYCLE 12
KEY PARAMETERS ASSUMED IN THE LOSS OF COOLANT FLOW ANALYSIS

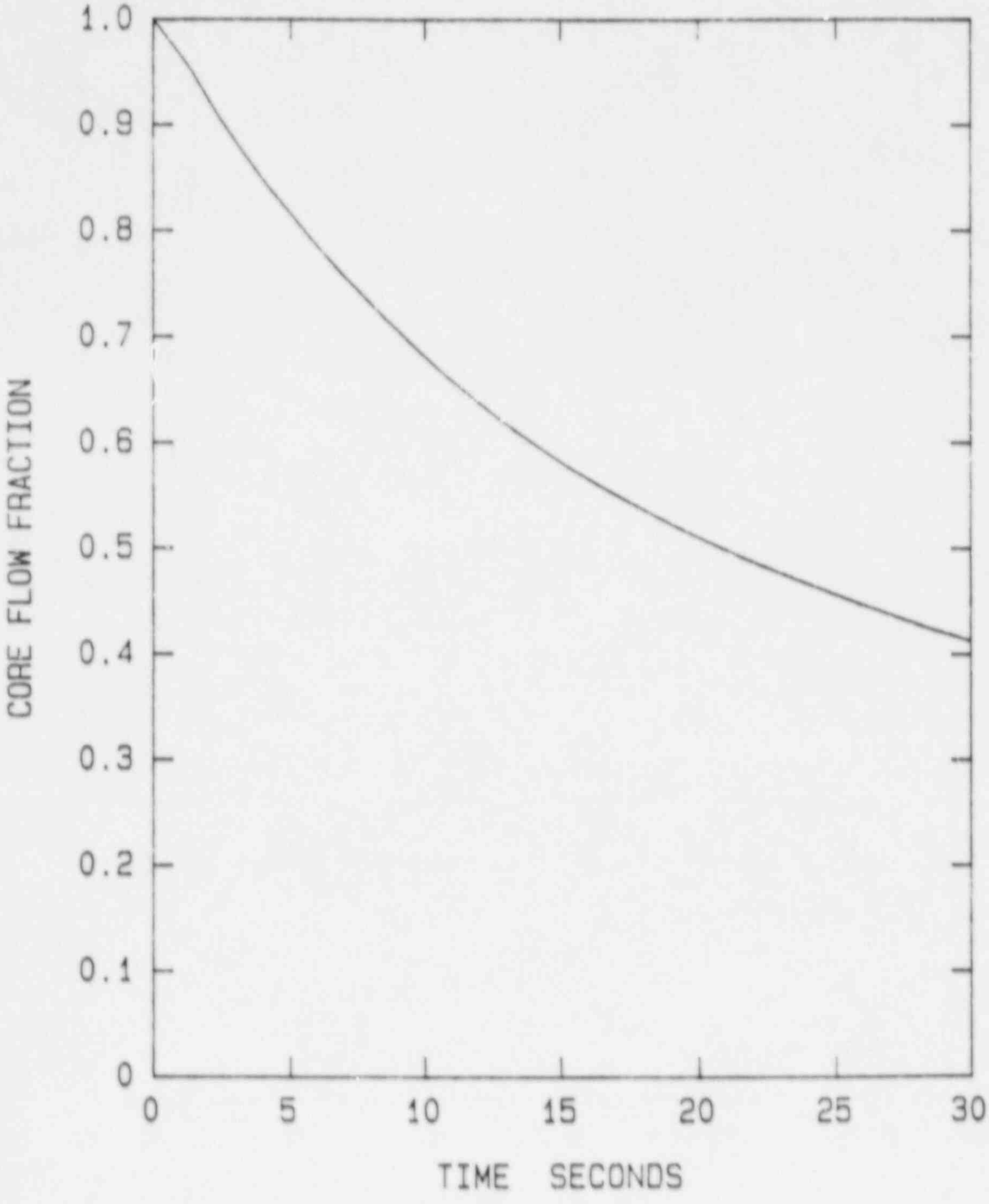
Parameter	Units	Cycle 11	Cycle 12
Initial Core Power Level	MWt	1500*	1500*
Initial Core Inlet Coolant Temperature	°F	545*	545*
Initial RCS Flow Rate	gpm	208,280*	208,280*
Pressurizer Pressure	psia	2075*	2075*
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	+0.5	+0.5
Doppler Coefficient Multiplier	...	0.85	0.85
LFT Analysis Setpoint	% of initial flow	93	93
LFT Response Time	sec	0.65	0.65
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)	$\%\Delta\rho$	-6.85	-6.50
Total Unrodded Radial Peaking Factor ($F_R T$)		1.80	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.[illegible] 2-2

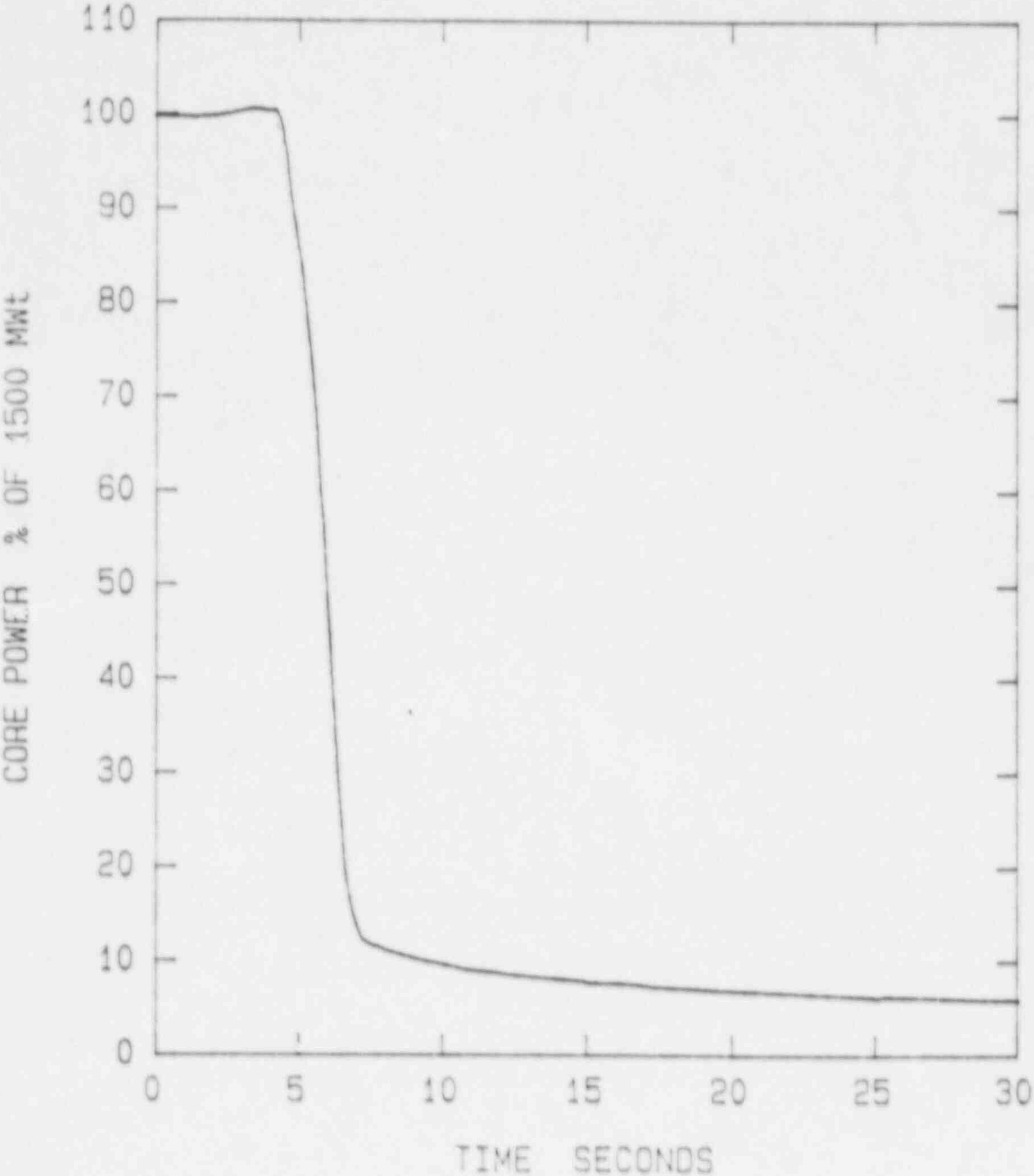
FORT CALHOUN CYCLE 12
SEQUENCE OF EVENTS FOR LOSS OF FLOW

Time (Sec)	Event	Setpoint or Value
1.0	Loss of Power to all Four Reactor Coolant Pumps	- - - -
2.80	Low Flow Trip Signal Generated	93% of 4-Pump Flow
3.35	Trip Breakers Open	- - - -
4.0	Shutdown, CEAs Begin to Drop into Core	- - - -
4.56	Minimum CE-1 DNBR	1.43
6.4	Maximum RCS Pressure, psia	2113



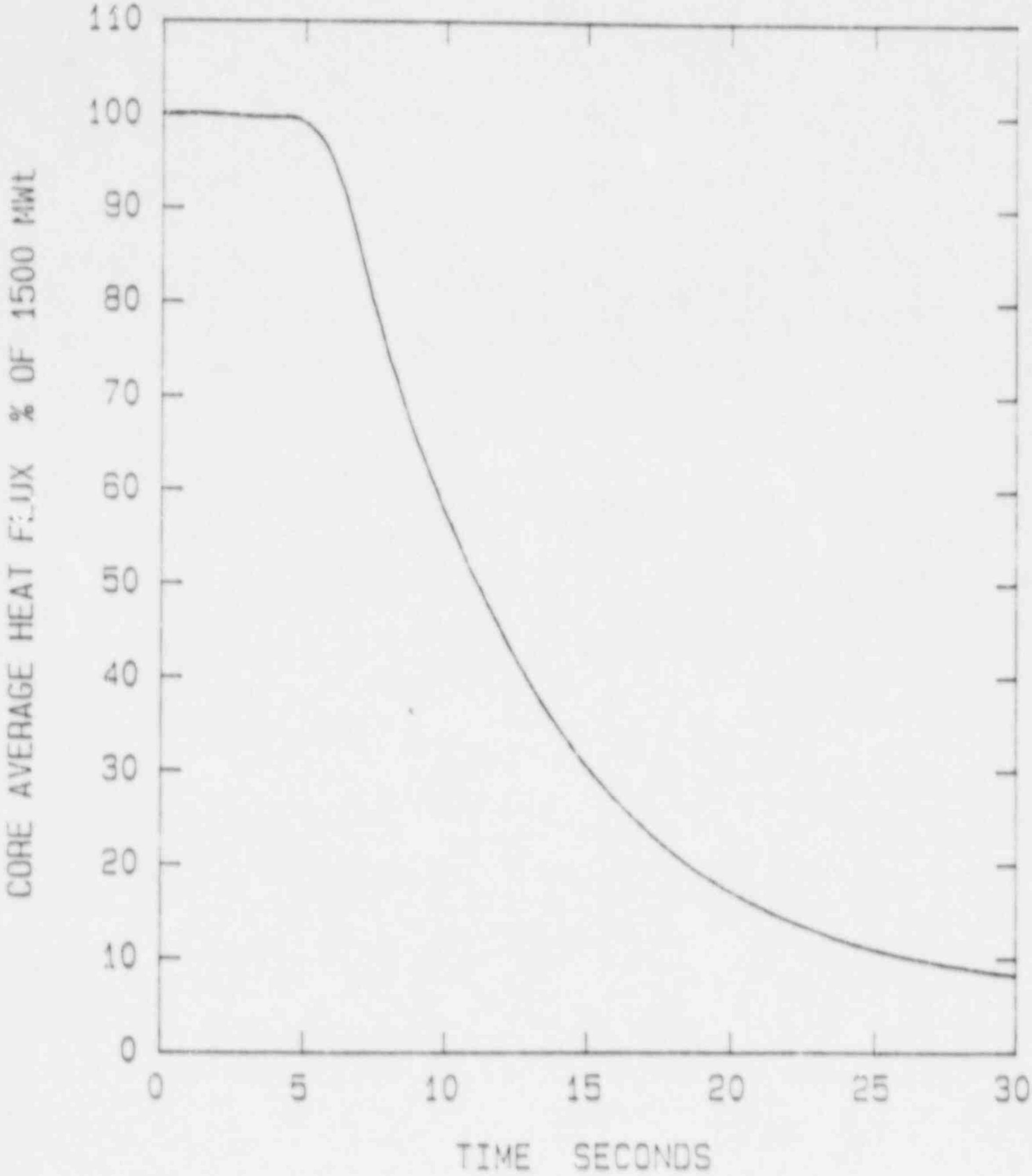
NOTE: CYCLE 12

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



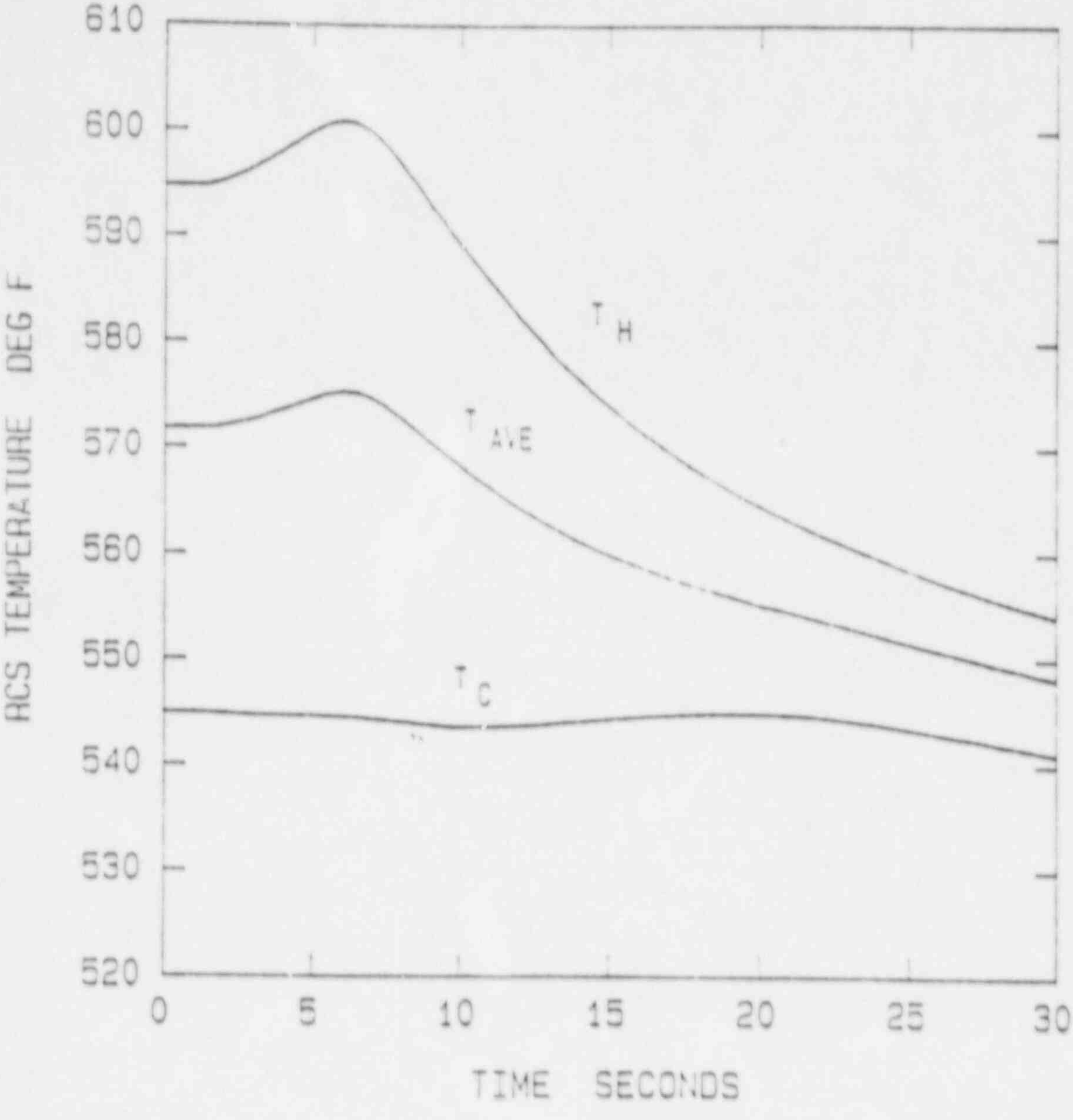
NOTE: C'CLE 12

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



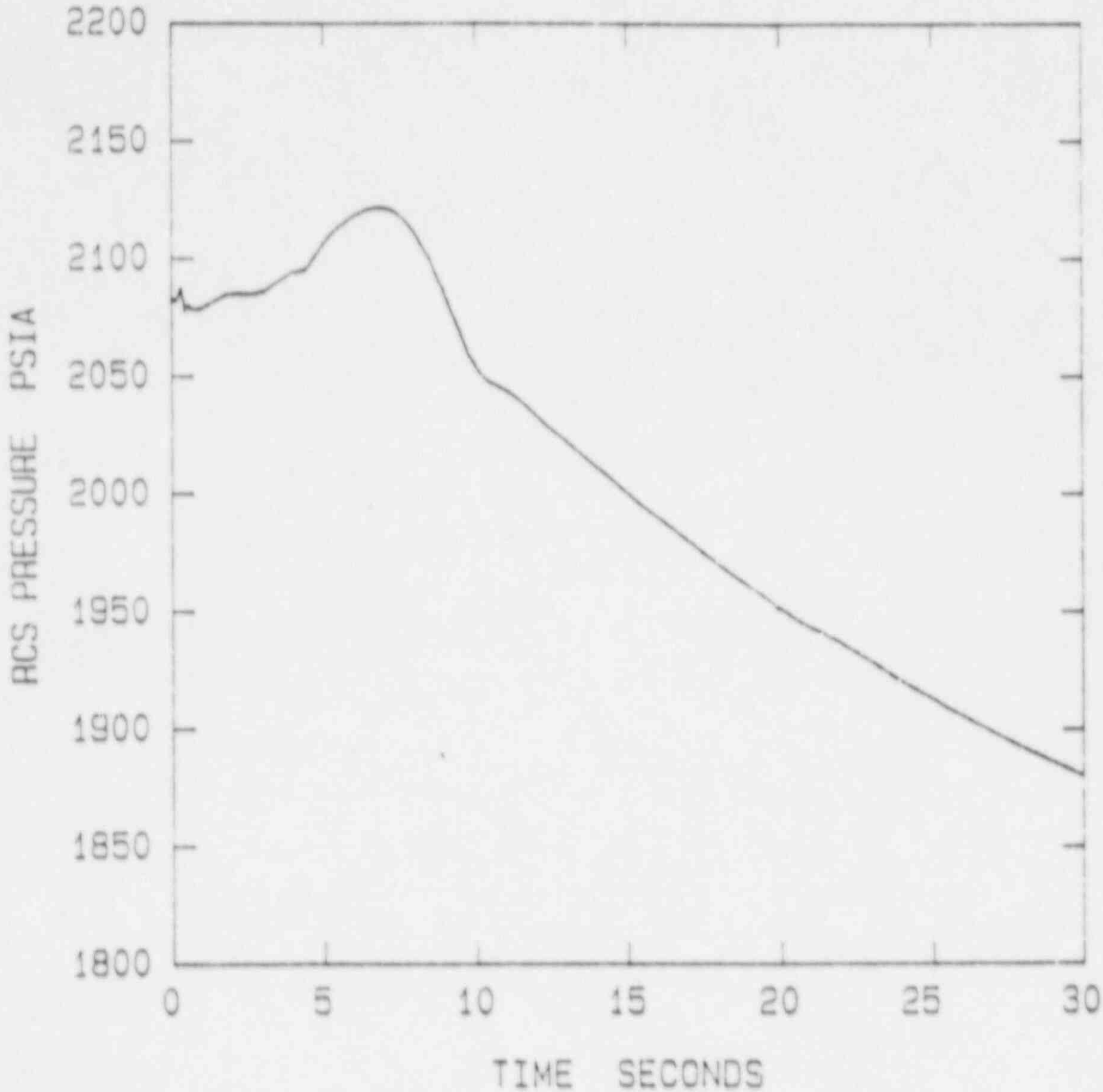
NOTE: CYCLE 12

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



NOTE: CYCLE 12

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



NOTE: CYCLE 12

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 ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

7.2.3 Full Length CEA Drop Event

The Full Length CEA Drop event was reviewed for Cycle 12 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 6.

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 1.45. A maximum allowable initial linear heat generation rate of 18.5 KW/ft could exist as an initial condition without exceeding the acceptable fuel centerline to melt limit of 22 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.

The CEA drop incident was reviewed for Cycle 12 and found to be bounded by Cycle 11. Since a negative 10 CFR 50.59 determination was made for Cycle 12, the conclusions from Cycle 11 remain valid and applicable to Cycle 12.

7.3 POSTULATED ACCIDENTS

7.3.1 CEA Ejection

The CEA Ejection event was reviewed for Cycle 12. A summary containing the results of the analysis was submitted in Reference 11 for Cycle 11 and has been validated for use in Cycle 12.

Since a negative 10 CFR 50.59 determination was made for the Cycle 12 CEA Ejection event, no reanalysis was performed.

7.3.2 Steam Line Break Accident

This accident was evaluated for Cycle 12 using the methodology discussed in References 6 and 12. 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 1986 update of the Fort Calhoun Station Unit No. 1 USAR.

The Cycle 12 Full Power Steam Line Break accident was evaluated for a more negative effective MTC of $-2.7 \times 10^{-4} \Delta\rho/^{\circ}F$ than the $-2.5 \times 10^{-4} \Delta\rho/^{\circ}F$ value that was used in the Cycle 8 analysis. However, the cooldown curve for Cycle 12 is

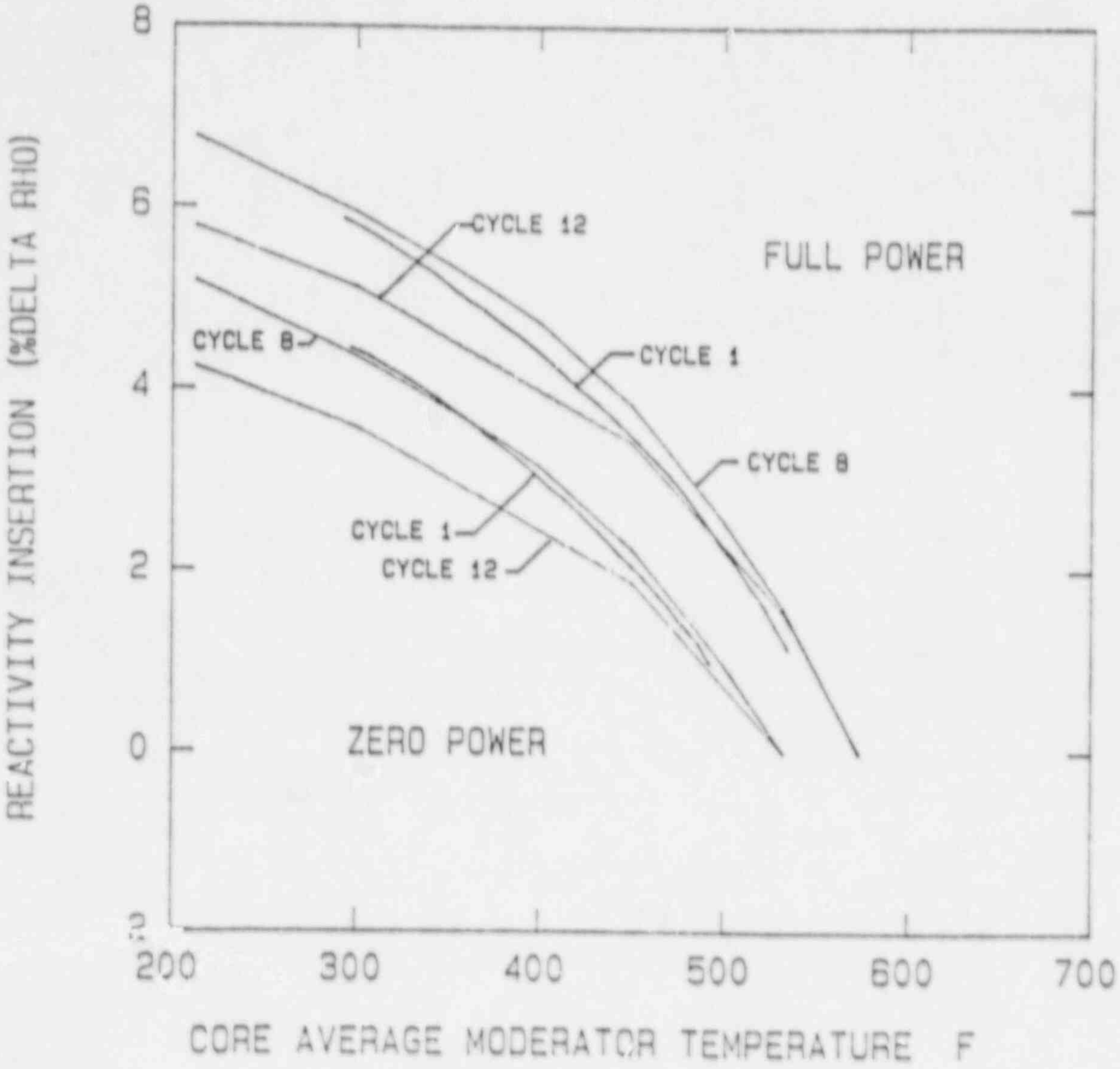
7.3.2 Steam Line Break Accident

bounded by that of Cycle 8 (as shown in Figure 7.3.2-1). This figure shows that the reactivity insertion for the Cycle 12 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 600°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 12 core including uncertainties. The Cycle 12 minimum available shutdown worth is 6.53%$\Delta\rho$ compared to a Cycle 8 value of 6.68%$\Delta\rho$. The reduction of 0.15%$\Delta\rho$ in scram worth from Cycle 8 to Cycle 12 is offset by the 0.98% $\Delta\rho$ reduction in moderator cooldown reactivity. The net gain assures that the overall reactivity insertion for a Cycle 12 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 12 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 7.3.2-1). Since the minimum available shutdown margin for Cycle 12 remains unchanged from the reference cycle value (4%$\Delta\rho$), the overall reactivity insertion for the Cycle 12 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 12 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 12 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

7.0 TRANSIENT ANALYSIS (Continued)

7.3 POSTULATED ACCIDENTS (Continued)

7.3.4 Seized Rotor Event

The Seized Rotor event was evaluated for Cycle 12 to demonstrate that only a small fraction of fuel pins are predicted to fail during this event. Cycle 12 is bounded by the reference cycle (Cycle 9) analysis because an $F_R T$ of 1.85 was assumed in the Cycle 9 analysis and the Cycle 12 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 10 CFR 100.

Since a negative 10 CFR 50.59 determination was made for the Cycle 12 Seized Rotor Event, no reanalysis was performed.

8.0 ECCS PERFORMANCE ANALYSIS

Both Cycle 11 Large and Small Break Loss of Coolant accident analyses were performed using the methodology discussed in Reference 1. A summary containing the results of the analyses was submitted in Reference 2. The Cycle 11 revised ECCS analysis was verified to be valid for use in Cycle 12 given the bounding input assumptions.

Since a negative 10 CFR 50.59 determination was made for the Cycle 12 ECCS analysis, no reanalysis was performed.

9.0 STARTUP TESTING

The startup testing program proposed for Cycle 12 is identical to that used in Cycle 11. It is also the same as the program outlined in the Cycle 6 Reload Application, with two exceptions. First, a CEA exchange technique (Reference 1) for zero power rod worth measurements will be performed in accordance with Reference 2, replacing the boration/dilution method. 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. Cycle 11 startup testing exclusively used the CEA exchange technique. Results from the CEA exchange technique were within the acceptance and review criteria for low power physics parameters. The combination of the pseudo-ejection technique at zero power and low power CECOR maps provides for 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
CEA Group Worths	± 15% or predicted	± 15% of predicted
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 REFERENCES

References (Chapters 1-5)

1. Letter A. Bournia (NRC) to R. L. Andrews (OPPD), dated March 11, 1988.

2. "Omaha Public Power District Batch M Reload Fuel Design Report," CEN-347(O)-P, November 1986.

3. Letter LIC-86-677, R. L. Andrews (OPPD) to A. C. Thadani (NRC), dated December 15, 1986.

4. "Generic Mechanical Design Report for Exxon Nuclear Fort Calhoun 14 x 14 Reload Fuel Assembly," XN-NF-79-70-P, September 1979.

5. "Extended Burnup Report for Fort Calhoun Reloads XN-4 and XN-5 (Batches K and L)," ANF-87-139(P), October, 1987.

6. "Qualification of Exxon Nuclear Fuel for Extended Burnup," XN-NF-82-06(P)(A) & Supplements 2, 4 & 5, Revision 1, October 1986.

7. "Omaha Public Power District Reload Core Analysis Methodology - Transient and Accident Methods and Verification," OPPD-NA-8303-P, Revision 02, April 1988.

8. "Omaha Public Power District Reload Core Analysis Methodology Overview," OPPD-NA-8301-P, Revision 03, April 1988.

9. "Omaha Public Power District Reload Core Analysis Methodology - Neutronics Design Methods and Verification," OPPD-NA-8302-P, Revision 02, April 1988.

10.0 REFERENCES (Continued)

References (Chapter 6)

1. "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-[illegible], July 1975.

2. "[illegible]al Heat Flux Correlation For CE Fuel Assemblies with [illegible] Spacer Grids, Part 1, Uniform Axial Power Distribution," [illegible]2-PA April 1975.

3. [illegible]-D Code Structure and Modeling Methods for Calvert Cliffs [illegible]s 1 and 2," CEN-191(B)-P, December 1981.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 70 to Facility Operating License No. DPR-40 for the Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, March 15, 1983.

5. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 77 to Facility Operating License No. DPR-40 for the Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, April 26, 1984.

6. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 92 to Facility Operating License No. DPR-40 for the Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, November 29, 1985.

7. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 109 to Facility Operating License No. DPR-40 for Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, May 4, 1987.

8. "Omaha Public Power District Reload Core Analysis Methodology Overview," OPPD-NA-8301-P, Revision 02, November 1986.

9. "Statistical Combination of Uncertainties, Part 2," Supplement 1-P, CEN-257(O)-P, August 1985.

10. Safety Evaluation Report on CENPD-207-P-A, "CE Critical Heat Flux: Part 2 Non-Uniform Axial Power Distribution," letter, Cecil Thomas (NRC) to A. E. Scherer (Combust[illegible] Engineering), November 2, 1984.

10.0 REFERENCES (Continued)

References (Chapter 7)

1. "Amendment to Operating License DPR-40, Cycle 11 License Application", Docket No. 50-285, May 4, 1987.

2. "Statistical Combination of Uncertainties Methodology, Part 1: Axial Power Distribution and Thermal Margin/Low Pressure LSSS for Fort Calhoun", CEN-257(0)-P, November 1983.

3. "Statistical Combination of Uncertainties Methodology, Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analysis for Fort Calhoun Unit 1", CEN-257(0)-P, November 1983.

4. "Statistical Combination of Uncertainties Methodology, Part 3: Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Fort Calhoun", CEN-257(0)-P, November, 1983.

5. "Statistical Combination of Uncertainties Methodology for Fort Calhoun," CEN-257(0)-P, Supplement 1-P, August 1985.

6. "Omaha Public Power District Reload Core Analysis Methodology - Transient and Accident Methods and Verification", OPPD-NA-8303-P, Revision 02, April 1988.

7. "CE Setpoint Methodology", CENPD-199-P, Rev. 1-P, March 1982.

8. "CEA Withdrawal Methodology", CEN-121(B)-P, November [illegible]9.

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

10. "Response to Questions on CESEC", Louisiana Power and Light Company, Waterford Unit 3, Docket 50-382, CEN-234(C)-P, December 1982.

11. Letter LIC-86-675, R. L. Andrews (OPPD) to A. C. Thadani (NRC), dated January 16, 1987.

12. "Omaha Public Power District Reload Core Analysis Methodology - Neutronics Design Methods and Verification", OPPD-NA-8302-P, Revision 02, April 1988.

10.0 REFERENCES (Continued)

References (Chapter 8)

1. "Omaha Public Power District Reload Core Analysis Methodology - Transient and Accident Methods and Verification", OPPD-NA-8303-P, Rev. 02, April 1988.

2. Letter LIC-86-675, R. L. Andrews (OPPD) to A. C. Thadani (NRC), dated January 16, 1987.

References (Chapter 9)

1. "Control Rod Group Exchange Technique," CEN-319, November 1985.

2. "Acceptance for Referencing of Licensing Topical Report CEN-319 - Control Rod Group Exchange Technique," letter, Dennis M. Crutchfield (NRC) to Rik W. Wells (Chairman - CE Owners Group), April 16, 1986.

ATTACHMENT B

Justification, Discussion, and Significant Hazards Considerations for Cycle 12 Reload

The Fort Calhoun Technical Specifications are being amended to reflect changes which are a result of the Cycle 12 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 12 Core Reload Evaluation.

TABLE B-1

Explanation for Cycle 12 Technical Specification Changes

Tech. Spec. No.	Change	Reasons
1) Page 1-2	Change total unrodded planar radial peak from 1.85 to 1.80.	Revised value is conservative with respect to previous value of 1.85. The reduced value will provide additional operating margin.
2) Figure 1-3	Replace Figure 1-3 with enclosed Figure 1-3.	The Cold Leg Temperature limit has been changed from 545°F to 543°F. This has resulted in a change to the Alpha, Beta and Gamma term of the TM/LP equation.
3) Page 1-5	Change the reference from FSAR to USAR.	Reflect updated reference.
4) Page 1-6	"strady" to "steady"	Corrected typo.
5) Figure 2-6	Replace Figure 2-6 with enclosed Figure 2-6.	The LHR-LCO has been revised to reflect the use of the more limiting LOCA ROPM limit in the analysis vs. the transient analysis ROPM. (see Letter LIC-88-620, K. J. Morris (OPPD) to NRC Document Control Desk, dated July 25, 1988).
6) Figure 2-9	Replace Figure 2-9 with enclosed Figure 2-9.	The $F_{xy}{}^T$ and $F_R{}^T$ limits as a function of power have been revised to maintain consistency with changes to Items 1 and 5, above.
7) Page 2-56	Add... "The linear heat rate shall be monitored by the incore detector system in accordance with Specifications 2.10.4(1)(a) or 2.10.4(1)(b), or maintain the Axial Shape Index, Y_I, with the limits of Figure 2-6 in accordance with Specification 2.10.4(1)(c)." after 2.10.4(1)5.	This change clarifies how the linear heat rate should be monitored and what parameters apply to bound the limits.

TABLE B-1
(Continued)

Tech. Spec. No.	Change	Reasons
8) Page 2-57	Add "... for seven days from the date of the last valid power distribution ..." to Specification 2.10.4(1)(b).	Clarify the point at which the ex-core LHR-LCO is entered during operation.
9) Page 2-57	Add "... and maintain the Axial Shape index, Y_I, within the limits of Figure 2-6..." to the first sentence of Specification 2.10.4(1)(c).	This is to clarify the requirements of maintaining the Axial Shape Index within the requirements of Figure 2-6.
10) Page 2-57a	Change $F_{xy}^T \leq 1.80$.	Revised value is conservative with respect to previous value of 1.85. The reduced value will provide additional operating margin.
11) Page 2-57c	Change "≤545°F" to ≤ "543°F"	The Cold Leg Temperature is being changed to more accurately reflect actual operating conditions and to gain additional margin.
12) Page 2-57c Footnote **	Change "545°F" to "543°F" and "547°F" to "545°F"	The Cold Leg Temperature is being changed to more accurately reflect actual operating conditions and to gain additional margin.

Description of Amendment Requests to Reduce the Planar Radial Peaking Factor F_{xy}^T to 1.80:

The proposed Technical Specification changes in Table B-1 corresponding to Items 1, 5, 6 and 10 for Technical Specifications Section 1.1, Figure 2-6, Figure 2-9 and Section 2.10.4(3) on Page 1-2 and 2-57 concern reducing the F_{xy}^T value from 1.85 to 1.80.

An error in the Cycle 11 setpoint evaluation has reduced the excore LHR-LCO tent from 90% power to 80% power during operation with the excore detectors. By reducing F_{xy}^T, additional operating margin is gained in the LHR-LCO operating tent. The F_R^T and F_{xy}^T limits as a function of power in Technical Specifications Figure 2-9 have been revised to maintain consistency with the change to Figure 2-6.

Basis for No Significant Hazards Determination:

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change merely allows utilization of the additional margin available with the reduction of maximum F_{xy}^T value with no changes in administrative specifications. On the basis of technical safety evaluation, operating with gain in margin for Cycle 12 LHR-LCO would be no more limiting than operating with the Cycle 11 LHR-LCO. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different type of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the same Technical Specification administration controls prevents the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Administrative specifications involving the LHR-LCO ensure that operating with the extra margin gained from the reduction of F_{xy}^T conforms to current plant conditions and, therefore, preserves the margin of safety. Reducing the LHR-LCO tent does not affect the available margin and, therefore, will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

Description of Amendment Requests Reducing Cold Leg Temperatures to 543°F:

The proposed Technical Specification changes in Table B-1 corresponding to Items 2, 11 and 12 for Technical Specifications Figure 1-3, 2.10.4(5) on page 2-57c and Footnote ** on Page 2-57c concern lowering the current cold leg temperature (T_c) from 545°F to 543°F.

The operation of the unit with a reduced cold leg T_c will provide additional margin for the TM/LP P_{var} equation. The Alpha, Beta and Gamma terms of the TM/LP P_{var} trip equation were optimized given the reduced allowable T_c and the unchanged F_R^T operating parameters.

All of the safety analyses and Cycle 12 design analyses were calculated at 545°F for conservative reasons; this bounds the use of a 543°F inlet temperature during Cycle 12 operations.

<u>Basis for No Significant Hazards Determination</u>

This proposed amendment does not involve a significant hazards consideration because the operation of the Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change allows the reduction of T_c to 543°F. The temperature change is bounded by the previous technical safety analysis which addressed the 545°F inlet temperature. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the same Technical Specification administrative controls prevents the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Administrative specifications involving T_c ensure that operating at a T_c of 543°F conforms to current plant conditions and, therefore, preserves the margin of safety. The temperature change is bounded by previous technical safety analysis which addressed the 545°F inlet temperature and, therefore, will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

Description of Amendment Requests for Changing References from FSAR to USAR:

The proposed Technical Specification changes in Table B-1 corresponding to Item 3 for Technical Specification 1.2 on page 1-5 concern changing all references mentioning "FSAR" to the correct reference "USAR."

One of the numerous post-TMI related changes was to require that all licensed commercial nuclear power plants perform an annual revision to the FSAR. This updated FSAR became officially recognized as the USAR (Updated Safety Analysis Report) to avoid any confusion with the FSAR. Needed reference changes in the Technical Specifications are generally made at the time when the related change is made.

Basis for No Significant Hazards Determination:

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change merely allows the Technical Specifications to reference the proper updated document with no changes in administrative specifications. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the same Technical Specification administrative controls prevents the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Administrative specifications involving the referencing of the USAR will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

Description of Amendment Request for Correcting a Typographical Error:

The proposed Technical Specification changes in Table B-1 corresponding to Item 4 for Technical Specifications 1.31(1) on Page 1-6 concerns correcting a typographical error by changing the word "strady" to "steady."

During the evaluation of Technical Specification changes for Cycle 12, a misspelled word was discovered in Technical Specification 1.3(1). The word in question is spelled "strady," however, the correct spelling of the word is "steady." This error is obviously typographic in nature and, therefore, poses no significant hazards consideration.

Basis for No Significant Hazards Determination:

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change merely allows for correct spelling of a word. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different kind of accident is not created because no new or different modes of operation are proposed for the plant. The continued use of the same Technical Specification administrative controls prevents the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Neither this typographical error nor its correction will reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.

Description of Amendment for Revising Section 2.10.4:

The proposed Technical Specification changes in Table B-1 corresponding to Items 7, 8, and 9 for Technical Specification Section 2.10.4 concern changes to instructions for entering into the excore LHR-LCO.

A review of Technical Specification 2.10.4 with the NRC Senior Resident Inspector indicated that the requirements for entering into the excore LHR-LCO (Figure 2-6) were unclear. The changes made herein more accurately define when the LHR-LCO should be entered, to allow sufficient time for a power reduction to the maximum power allowed by Technical Specification Figure 2-6, should the reactor be in excess of that power level at the time the LHR-LCO was entered.

Basis for No Significant Hazards Determination:

This proposed amendment does not involve a significant hazards consideration because the operation of Fort Calhoun Station in accordance with this amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated. This change clarifies the point at which the LHR-LCO (Figure 2-6) must be entered and provides better guidance for plant operation. The basis for the technical safety evaluation would be no more limiting than operating with the Cycle 11 basis. Therefore, this change does not increase the probability or consequences of a previously evaluated accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated. It has been determined that a new or different type of accident is not created because no new or different modes of operation are proposed for the plant. The continued used of the Technical Specification administrative controls prevents the possibility of a new or different kind of accident.

3) Involve a significant reduction in a margin of safety. Administrative specifications involving the LHR-LCO ensure that the operators enter the LCO with sufficient time to reduce power, if necessary, prior to utilizing the excore instruments to monitor core power. The changes have been implemented through strict administrative procedures and, therefore, will not reduce the margin of safety.

Based on the above considerations, OPPD does not believe that this amendment involves a significant hazards consideration.