Arkansas Nuclear One Unit 2

Cycle 2 Reload Analysis Report

8102250223

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* To be supplied at a later date

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1.0 Introduction and Summary

This report provides an evaluation of the design and performance of Arkansas Nuclear One Unit 2 during its second cycle of operation at 100% rated core power of 2815 MWT and NSSS power of 2825 MWT. Operating conditions for Cycle 2 have been assumed to be consistent with those of Cycle 1 and are summarized as full power operation under base load conditions. The core will consist of presently operating Batch A, B and C assemblies, along with fresh Batch D assemblies. The Cycle 1 termination burnup has been assumed to be approximately 12,500 MWD/T. Because of the similarity in basic system characteristics and operation between the initial cycle of ANO-2 and Cycle 2, the Cycle 1 core will hereafter be referred to in this report as the "reference cycle" unless otherwise noted.

A review of all postulated accidents and anticipated operational occurrences has shown that the Cycle 2 core design meets the applicable safety criteria. The evaluations of the reload core characteristics have been examined with respect to the ANO-2 safety analysis described in Chapter 15 of the FSAR (1-1). Specific differences in core fuel loadings have been accounted for in the present analysis. It has been concluded in all cases that either the reference cycle analyses envelope the new conditions or reanalysis has been performed as described in this report. Where dictated by variations from the reference cycle, proposed modifications to the plant Technical Specifications are provided and are justified by the analyses reported herein. Additionally, modifications will be made as required to the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculator (CPC) System.

2.0 Operating History of the Reference Cycle

Arkansas Nuclear One Unit 2 is currently in its first fuel cycle which began with initial criticality on 12/5/78. Low power physics test and power ascension test program results for Cycle 1 have been documented and reported to the NRC (2-1). It is currently estimated that Cycle 1 will terminate during March 1981.

The unit has experienced 4 major outages since initial criticality . The first, for unscheduled maintenance which included Diesel generator and main steam safety valve replacement, extended from 1/31/79 to 6/6/79. The second outage extended from 10/4/79 to 12/1/79 due to an extensive evaluation of an anomalous variation in hot leg temperature. It was determined after a thorough investigation that the anomaly did not present a safety concern (2-1). The adverse impact of the anomaly on operations was alleviated by installing additional temperature sensing Resistance Temperature Detectors (RTD) on each hot leg. The third outage was scheduled for TMI lessons learned modifications and extended from 1/29/80 to 3/19/80. The last major outage extended from 9/4/80 to 10/1/80 due to degraded service water flow to the containment cooling units caused by an accumulation of Asian clams.

3.0 General Description

The Cycle 2 core will consist of those assembly types and numbers listed in Table 3-1. Sixty Batch A assemblies will be removed from the Cycle 1 core to make way for an equal number of fresh, unshimmed Batch D assemblies. All Batch B and C assemblies, along with one A assembly, will be retained for the second cycle.

The reload batch will consist of 40 type D assemblies and 20 type D* assemblies. Both sub-batch types are zone-enriched according to the pattern shown in Figure 3-1. As a result of a Department of Energy (DOE)contract for high burnup test irradiation, two D assemblies will contain fuel rods of various advanced designs. Section 10 of this submittal discusses those assemblies and test rods.

The loading pattern for Cycle 2, showing fuel type and location, is displayed in Figure 3-2. The pattern, given in quarter core, is appropriately expanded to full core through 90° clockwise rotation. Also identified in Figure 3-2 are the planned locations for the Batch D assemblies containing the DOE high burnup test rods, the Batch C assembly containing test rods discussed in Reference 3-1, and the Batch B and Batch C assemblies containing burnable poison test rods discussed in Reference 3-2. The Batch B and C test assemblies have been placed in locations consistent with the criteria of References 3-1 and 3-2, thereby assuring that the test rods will not be in limiting locations nor cause limiting conditions in adjacent rods during Cycle 2.

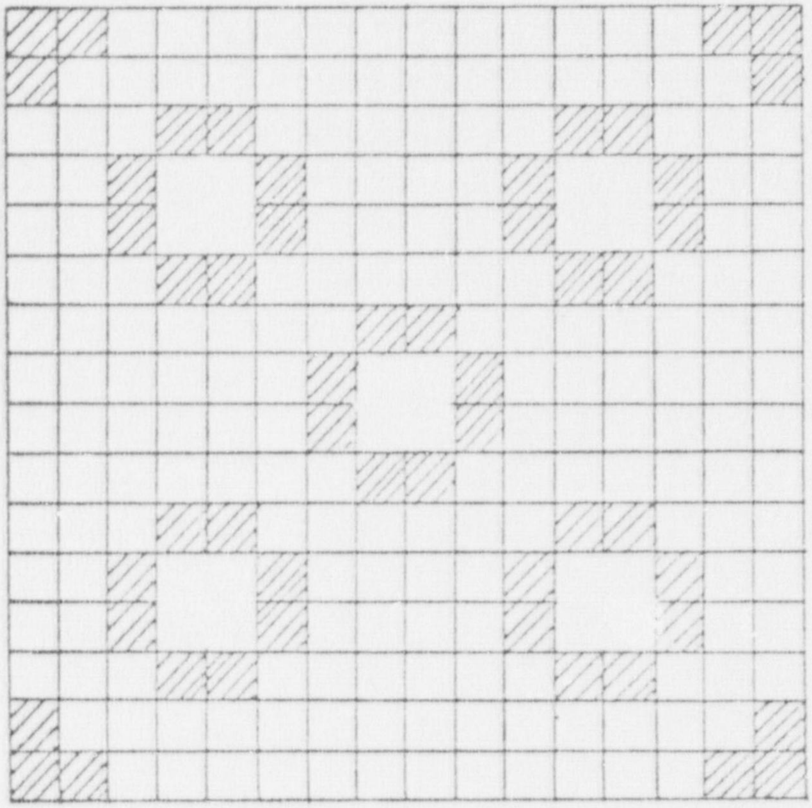
Figure 3-3 displays the beginning of Cycle 2 assembly average burnup distribution along with the initial assembly fuel enrichment. This distribution is based on a core average burnup for Cycle 1 of 12,500 MWD/T.

Control rod patterns and in-core instrument locations will remain unchanged from Cycle 1.

TABLE 3-1

Arkansas Nuclear One Unit 2
Cycle 2 Core Loading

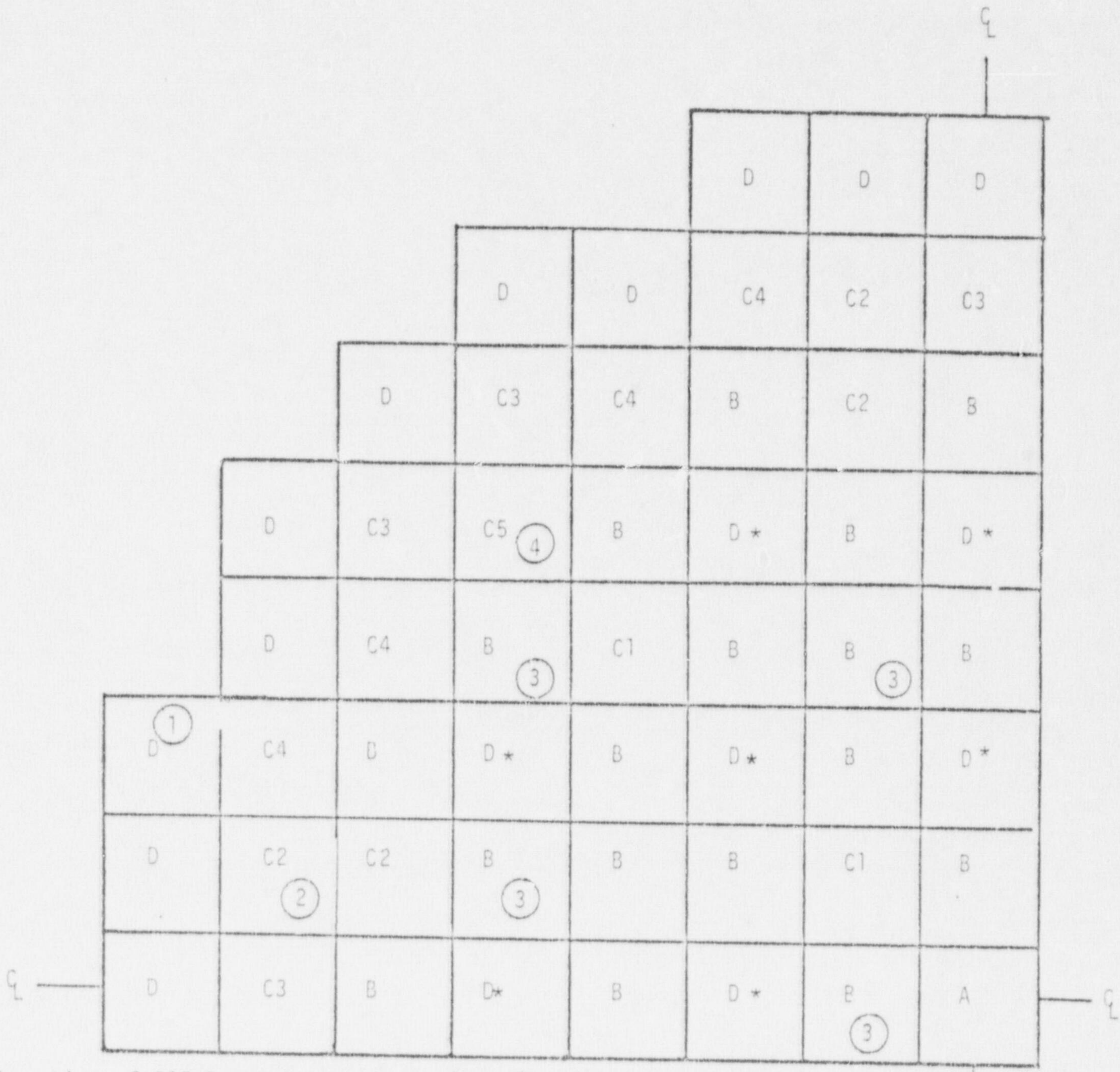
Assembly Desig-nation	Number of Assemblies	Fuel Rods per Assembly	Initial Enrichment (w/o U-235)	Number Shims/ Assembly	Initial Shim Loading (gm B_{10}/cm)	Fuel Rods	Shim Rods
A	1	236	1.93	0		236	0
B	60	224	2.27	12	0.0087	13440	720
C1	8	224	2.94	12	0.0028	1792	96
C2	16	224	2.94	12	0.0047	3584	192
C3	12	234	2.94	2	0.0087	2808	24
C4	16	233	2.94	3	0.0087	3728	48
C5	4	224	2.94	9	0.0047	896	36
				3	0.0087		12
D	40	184	3.48	0		7360	0
		52	3.03			2080	
D*	20	184	3.03	0		3680	0
		52	2.73			1040	
Total	177					40644	1128



FUEL ROD TYPE	FUEL ROD ENRICHMENT (W/O U235) D ASSEMBLY	D*ASSEMBLY
	3.48	3.03
	3.03	2.73

WATER HOLE

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE - UNIT 2	ENRICHMENT ZONING PATTERN FOR D AND D* FUEL ASSEMBLIES	FIGURE 3-1



① Location of DOE Batch D test assemblies (2 of 4 symmetric locations)

② Location of Batch C test assembly (1 of 4 symmetric locations)

③ Location of Batch B burnable poison test rods assemblies (4 of 16 symmetric locations)

④ Location of Batch C burnable poison test rods assemblies (4 of 4 symmetric locations)

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE -UNIT 2	ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2 CORE MAP	FIGURE 3 - 2

X.XX	INITIAL ENRICHMENT W/O U235
YYYY	BOC2 ASSEMBLY AVERAGE BURNUP (MWD/T) EOC 1 = 12500 MWD/T

					3.38 0	3.38 0	3.38 0
			3.38 0	3.38 0	2.94 7,200	2.94 12,500	2.94 9,800
		3.38 0	2.94 9,300	2.94 7,100	2.27 13,500	2.94 10,300	2.27 13,800
	3.38 0	2.94 9,300	2.94 7,700	2.27 12,400	2.96 0	2.27 14,500	2.96 0
	3.38 0	2.94 7,100	2.27 12,400	2.94 13,000	2.27 14,700	2.27 13,900	2.27 15,100
3.38 0	2.94 7,200	2.27 13,500	2.96 0	2.27 14,700	2.96 0	2.27 15,000	2.96 0
3.38 0	2.94 12,500	2.94 10,300	2.27 14,500	2.27 13,900	2.27 15,000	2.94 13,000	2.27 14,800
3.38 0	2.94 9,800	2.27 13,800	2.96 0	2.27 15,100	2.96 0	2.27 14,800	1.93 13,200

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ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE -UNIT 2	ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2 ASSEMBLY AVERAGE BURNUP AND INITIAL ENRICHMENT DISTRIBUTION	FIGURE 3-3

4.0 Fuel System Design

4.1 Mechanical Design

The mechanical design for the standard Batch D reload fuel is essentially identical to that of the standard Batches A, B and C fuel used in ANO-2 and described in the reference cycle analysis (4-1).

C-E has performed analytical predictions of cladding creep-collapse time for all ANO-2 fuel batches that will be irradiated in Cycle 2 and has concluded that the collapse resistance of all standard and test fuel rods is sufficient to preclude collapse during Cycle 2. These analyses utilized the CEPAN computer code (4-2) and included as input conservative values of internal pressure, cladding dimensions, cladding temperatures and neutron flux.

4.2 Hardware Modifications to Mitigate Guide Tube Wear

All fuel assemblies which will be placed in CEA locations in Cycle 2 will have stainless steel sleeves installed in the guide tubes to prevent guide tube wear. A detailed discussion of the design of the sleeves and their effect on reactor operation is contained in Reference 4-3.

4.3 Thermal Design

Using the FATES fuel evaluation model (Reference 4-4), the thermal performance of the various types of fuel assemblies including test assemblies (fuel Batches A, B, C and D) has been evaluated with respect to prior burnup and the proposed burnup during Cycle 2. Burnup dependent fuel performance calculations were used in the ECCS fuel performance calculations presented in Section 8. The NRC burnup enhanced fission gas release factor as given in NUREG-0418(4-5) was used in the FATES analysis.

4.4 Chemical Design

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch D fuel are identical to those of the Batch A, B, and C fuel from Cycle 1. Thus, the chemical or metallurgical performance of the Batch D fuel will be unchanged from the performance of the Cycle 1 fuel.

4.5 Operating Experience

The ANO-2 original core (Batches A, B, and C) represents the first application of the Combustion Engineering 16X16 array fuel design. As such, there will have been no detailed examinations of irradiated fuel performed on this design prior to the EOC1 shutdown.

However, a brief visual inspection was conducted on six fuel assemblies during an interim Cycle 1 shutdown in October, 1979. The mechanical appearance of the fuel bundles was satisfactory.

The fuel assembly design utilized in ANO-2 is basically a scaled version of the C-E 14X14 design in Maine Yankee, Ft. Calhoun, Calvert Cliffs, Millstone Point 2, and St. Lucie 1. Detailed inspections at these plants have shown acceptable mechanical performance out to burnups in excess of 40,000 MWD/MTU. The interim Cycle 1 examination in ANO-2 has indicated that there are no unforseen mechanical problems associated with the conversion from the 14X14 to the 16X16 array.

5.0 Nuclear Design

5.1 Physics Characteristics

5.1.1 Fuel Management

The Cycle 2 loading pattern is characterized by fresh fuel loading on the core periphery and the shuffling to the interior of the fuel assemblies previously on the periphery in Cycle 1. The 20 interior fresh fuel assemblies have a lower assembly average enrichment than those on the periphery and are mixed with once-burned fuel in the central region of the core in a pattern which minimizes power peaking. With this loading, the Cycle 2 reactivity lifetime for full power operation is expected to be 10,500 MWD/T. The Cycle 2 length, along with all physics parameters input to the safety analysis, are based on a Cycle 1 termination burnup of approximately 12,500 MWD/T.

Characteristic physics parameters for Cycle 2 are compared to those of the reference cycle in Table 5-1. The values in this table for both Cycle 2 and the reference cycle are intended to represent nominal core parameters. Those values used in the safety analysis (see Section 7) contain appropriate uncertainties, and in all cases are conservative with respect to the values reported in Table 5-1.

Table 5-2 presents a summary of CEA shutdown worths and reactivity allowances for Cycle 2 with a comparison to reference cycle data. Table 5-2 generally characterizes the changes in reactivity that occur during a trip from full power with a corresponding change in core parameters to the zero power state. It is not intended to represent any particular limiting anticipated operational occurrence (AOO) or postulated accident, although the quantity shown as "Required Shutdown Margin" represents the numerical value of the worth which is required for the hot zero power steam line break accident. The analysis of any specific transient is performed at the limiting conditions which are conservative for the particular transient.

The CEA group identifications remain the same as in the reference cycle. The power dependent insertion limit (PDIL) curve has been changed and is shown in Figure 5-1. Table 5-3 shows the reactivity worths of various CEA groups calculated at full power conditions for Cycle 2 and the reference cycle.

5.1.2 Power Distribution

Figures 5-2 through 5-4 illustrate the calculated all rods out (ARO) planar radial power distributions during Cycle 2. The one-pin power peaks presented in these figures represent the maximum that could be expected between about 20 and 80 percent of core height. Radial peaks outside this axial region were examined and found not to be limiting at any time during the cycle. Time points at the beginning, middle, and end of cycle were chosen to display the decrease in maximum planar radial peak as a function of burnup.

Radial power distributions for rodded configurations are given for BOC and EOC in Figures 5-5 through 5-10. The rodded configurations shown are those allowed by the PDIL at full power: part length CEA's (PLCEA's), Bank 6, and Bank 6 plus the PLCEA's. As is the case for unrodded configurations, the largest radial peak for each of these rodded configuration occurs at the beginning of Cycle 2.

The radial power distributions described in this section are calculated data which do not include any uncertainties or allowances. The calculations performed to determine these radials account for augmented power peaking which is characteristic of fuel rods adjacent to the water holes.

Nominal axial peaking factors are expected to range from 1.20 at BOC2 to 1.13 at EOC2.

5.2 Safety Related Data

5.2.1 Ejected CEA

The maximum reactivity worths and planar radial power peaks associated with an ejected CEA event are shown in Table 5-4 for Cycle 2 and the reference cycle. The Cycle 2 values encompass the worst conditions anticipated during Cycle 2 and are conservative with respect to the actual calculated values.

5.2.2 Dropped CEA

The limiting dropped CEA reactivity worth and maximum increase in radial peaking factor are shown in Table 5-5 for Cycle 2 and the reference cycle. The values shown for Cycle 2 are conservative at any time during the cycle.

5.2.3 Augmentation Factors

Augmentation factors have been calculated for the Cycle 2 core using the calculational model described in Reference 5-1. The input information required for the calculation of augmentation factors that is specific to the core under consideration includes the fuel densification characteristics, the radial pin power distribution and the single gap peaking factors. Augmentation factors for the Cycle 2 core have been conservatively calculated by combining the largest single gap peaking factors with the most conservative (flattest) radial pin power distributon. The calculations yield noncollapsed clad augmentation factors showing a maximum value of 1.049 at the top of the core. The augmentation factors used for Cycle 2 are compared to those of the reference cycle in Table 5-6. These augmentation factors will be incorporated directly into the COLSS and CPC linear heat rate and DNBR calculations.

5.3 Physics Analysis Methods

5.3.1 Analytical Input to In-Core Measurements

In-core detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the manner described in Reference 5-2.

5.3.2 Uncertainties in Measured Power Distributions

The planar radial power distribution measurement uncertainty of 5.3%, based on Reference 5-2, will be applied to the Cycle 2 COLSS and CPC on-line calculations which use planar radial power peaks. The axial and three dimensional power distribution measurement uncertainties are determined in conjunction with other monitoring and protection system measurement uncertainties (See Section 9).

5.3.3 Nuclear Design Methodology

The following changes in nuclear design methodology have been implemented for Cycle 2 relative to the reference cycle:

1. Use of Coarse Mesh Neutronics Calculations (ROCS)
2. Use of Transport Theory (DIT) For Few-group Cross Section Generation
3. Extended Pointwise Doppler Feedback

5.3.3.1 Use of Coarse Mesh Neutronics Calculations (ROCS)

Cycle 1 physics calculations were based on the fine mesh design program PDQ (Reference 5-3). In the Cycle 2 analysis, the coarse mesh computer program ROCS (Reference 5-4) has been used along with PDQ for the calculation of certain parameters. ROCS has been used in a manner consistent with current C-E reload methods as employed in analyses for Calvert Cliffs Units I and II and St. Lucie Unit I reloads (References 5-5, 5-6, 5-7, and 5-8). In particular, ROCS has been used where three-dimensional effects are important and where detailed pin peaking information is not required. As requested in Reference 5-9 this is not an increase in the scope of ROCS usage over that which has appeared in recent C-E reload applications.

5.3.3.2 Use of Transport Theory (DIT) For Few-group Cross Section Generation

Cross sections for both PDQ and ROCS calculations are taken from the DIT assembly Spectrum code (5-10, 5-11) based on integral transport theory.

Local power peaking is calculated with PDQ using few-group fine mesh cross sections from DIT multigroup transport theory calculations. This provides for the increased local power peaking in fuel rods adjacent to the water holes and therefore the factor applied in earlier reload analyses, to account for this phenomenon is no longer required; in earlier reload analyses, this factor had been calculated by DIT but applied to the PDQ results separately.

Coarse mesh cross sections for the ROCS code are also taken from DIT. This has led to agreement with measurements on reactivity, power distributions, rod worths and reactivity coefficients that is substantially improved from previously used cross-section generation methods.

5.3.3.3 Extended Pointwise Doppler Feedback

The extended pointwise Doppler feedback technique, including the improved fuel temperature versus kw/ft correlation, employed for recent C-E reload applications, has been implemented for ANO-2 Cycle 2.

The extended Doppler feedback technique, described in Reference 5-12, involves utilizing iterations between pointwise power distribution and pointwise fuel temperature instead of using precalculated batchwise fuel temperatures.

The improved fuel temperature vs. kw/ft correlation, described in Reference 5-13, is based on the methodology of Reference 5-1 and more accurately accounts for changes in fuel temperature for a given local power density with increased burnup.

TABLE 5-1

ANO Unit 2 Cycle 2

Nominal Physics Characteristics

	Units	Reference Cycle	Cycle 2
Dissolved Boron			
Dissolved Boron Content for Criticality, CEAs Withdrawn, Hot Full Power, Equilibrium Xenon, BOC	PPM	611	865
Boron Worth			
Hot Full Power, BOC	PPM/% $\Delta\rho$	81	100
Hot Full Power, EOC	PPM/% $\Delta\rho$	79	83
Moderator Temperature Coefficients			
Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/°F$	-.5	-.5
End of Cycle	$10^{-4}\Delta\rho/°F$	-2.5	-2.3
Doppler Coefficient			
Hot Zero Power, BOC	$10^{-5}\Delta\rho/°F$	-1.59	-1.59
Hot Full Power, BOC	$10^{-5}\Delta\rho/°F$	-1.23	-1.27
Hot Full Power, EOC	$10^{-5}\Delta\rho/°F$	-1.42	-1.51
Total Delayed Neutron Fraction, β_{eff}			
BOC	-----	.0071	.0063
EOC	-----	.0049	.0054
Neutron Generation Time, ℓ^*			
BOC	10^{-6}sec	30.5	25.7
EOC	10^{-6}sec	31.1	30.6

Table 5-2
Arkansas Nuclear One Unit 2 Cycle 2
Limiting Values of Reactivity Worths and Allowances, % Δρ

	Reference Cycle	Cycle 2 BOC	Cycle 2 EOC
Worth Available			
Worth of all CEAs inserted	11.7	11.4	12.3
Stuck CEA allowance	1.6	2.1	2.4
Worth of all CEAs less highest worth CEA stuck out	10.1	9.3	9.9
Worth Required (Allowances)			
Power defect, HFP to HZP (Doppler and Moderator defects, worth loss, temperature redistribution)	3.4	2.0	2.6
Moderator Voids	0.1	0.0	0.1
CEA bite (Full power PDIL)	0.4	0.5	0.6
Total reactivity allowance	3.9	2.5	3.3
Available Worth Less Allowance	6.2	6.8	6.6
Required Shutdown Margin	5.0	5.0	5.0

TABLE 5-3

Arkansas Nuclear One Unit 2 Cycle 2
Reactivity Worth of CEA Regulating Groups
at Hot Full Power, %Δρ

	Beginning of Cycle		End of Cycle	
Regulating CEAs	Reference Cycle	Cycle 2	Reference Cycle	Cycle 2
Group 6	0.6	0.5	0.6	0.6
Group 5	0.5	0.6	0.5	0.6
Group 4	0.8	0.6	0.7	0.7

Note:

Values shown assume sequential group insertion

TABLE 5-4

ANO-2 Cycle 2
CEA Ejection Data

	Limiting Safety Analysis Values	
	Reference Cycle	Cycle 2
Maximum Radial Power Peak		
Full power with Bank 6 inserted; worst CEA ejected	2.19	2.94
Zero power at ZPDIL; worst CEA ejected	9.2	8.23
Maximum Ejected CEA Worth (% $\Delta\rho$)		
Full power with Bank 6 inserted; worst CEA ejected	0.24	0.30
Zero power at ZPDIL; worst CEA ejected	1.1	0.82

Notes

(1) Uncertainties and allowances are included in the above data.
(2) These Cycle 2 safety analysis values are conservative with respect to the actual Cycle 2 calculated values.
(3) ZPDIL for Cycle 1 is Banks 6+5+4+3+2+PLCEA's
ZPDIL for Cycle 2 is assumed to be Banks 6+5+4+3+PLCEA's

TABLE 5-5

ANO-2 Cycle 2
Full Length CEA Drop Data
(HFP, Equilibrium Xenon)

	Limiting Safety Analysis Values	
	Reference Cycle	Cycle 2
Minimum Worth (% $\Delta\rho$)	0.10%	0.10%
Maximum Percent Increase in Radial Peaking Factor	27%	17%

Note:

1) Uncertainties and allowances are included in the above data.

2) These Cycle 2 safety analysis values are conservative with respect to the actual Cycle 2 calculated values.

TABLE 5-6

ANO-2 Cycle 2
Augmentation Factors and Gap Sizes

		Reference Cycle		Cycle 2	
Core Height (Percent)	Core Height (Inches)	Noncollapsed Clad Augmentation Factor	Gap Size (Inches)	Noncollapsed Clad Augmentation Factor	Gap Size (Inches)
98.7	148.0	1.033	1.913	1.049	1.913
87.9	131.8	1.030	1.706	1.045	1.706
77.0	115.5	1.027	1.499	1.041	1.499
66.2	99.3	1.025	1.292	1.037	1.292
55.4	83.1	1.022	1.085	1.031	1.085
44.6	66.9	1.019	.879	1.027	.879
33.8	50.7	1.016	.672	1.022	.672
23.0	34.5	1.012	.465	1.016	.465
12.1	18.2	1.008	.259	1.011	.259
1.3	2.0	1.002	.052	1.002	.052

Notes:

(1) Values are based on approved model described in Reference 5-1.

(2) The values in this table for Cycle 2 were used in the safety analysis and are incorporated in COLSS and CPC.

Figure 5-1

ANO-2 Cycle 2

Power Dependent Insertion Limit
(PDIL)

TO BE SUPPLIED LATER WITH SECTION 7.0

Assembly Relative Power Density

					0.79	1.07	1.13
			0.78	1.13	1.13	1.14	1.18
		0.80	1.03	1.22	1.00	1.14	0.97
	0.78	1.03	1.14	0.98	1.31	0.92	1.24
	1.14	1.22	0.99	1.03	0.87	0.79	0.80
0.79	1.14	1.01	1.32	0.87	1.12	0.76	1.03
1.07	1.14	1.14	0.92	0.79	0.76	0.79	0.65
1.13 X	1.19	0.97	1.25	0.80	1.03	0.65	0.54

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NOTE: X = Location of Maximum 1-Pin Peak = 1.50

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE - UNIT 2	ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY, HFP AT BOC, EQUILIBRIUM XENON	FIGURE 5-2

Assembly Relative Power Density

					0.75	0.96	1.01
			0.75	1.06	1.06	1.07	1.10
		0.80	1.01	1.17	0.99	1.14	0.98
	0.76	1.01	1.16	1.00	1.30	0.95	1.26
	1.06	1.17	1.01	1.06	0.92	0.87	0.87
0.74	1.06	0.99	X 1.30	0.92	1.20	0.86	1.16
0.96	1.07	1.14	0.95	0.87	0.86	0.93	0.79
1.01	1.10	0.97	1.26	0.87	1.16	0.79	0.68

NOTE: X = Location of Maximum 1-Pin Peak = 1.42

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE - UNIT 2	ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY, HFP AT 5 GWD/T, EQUILIBRIUM XENON	FIGURE 5- 3

Assembly Relative Power Density

					0.75	0.93	0.96
			0.77	1.04	1.04	1.03	1.06
		0.82	1.02	1.15	0.98	1.11	0.97
	0.77	1.01	1.16	1.00	1.26	0.95	1.23
	1.05	1.15	1.01	1.06	0.93	0.91	0.91
0.74	1.04	0.98	X 1.26	0.93	1.22	0.91	1.20
0.93	1.03	1.10	0.95	0.91	0.91	1.01	0.87
0.96	1.06	0.96	1.23	0.91	1.21	0.87	0.78

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NOTE: X = Location of Maximum 1-Pin Peak = 1.34

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT2	ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY HFP AT EOC, EQUILIBRIUM XENON	FIGURE 5- 4

NOTE: X = Location of Maximum 1-Pin Peak - 1.57

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 2	ARKANSAS NUCLEAR ONE UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY WITH PLCEAS INSERTED, HFP, BOC	FIGURE 5-5

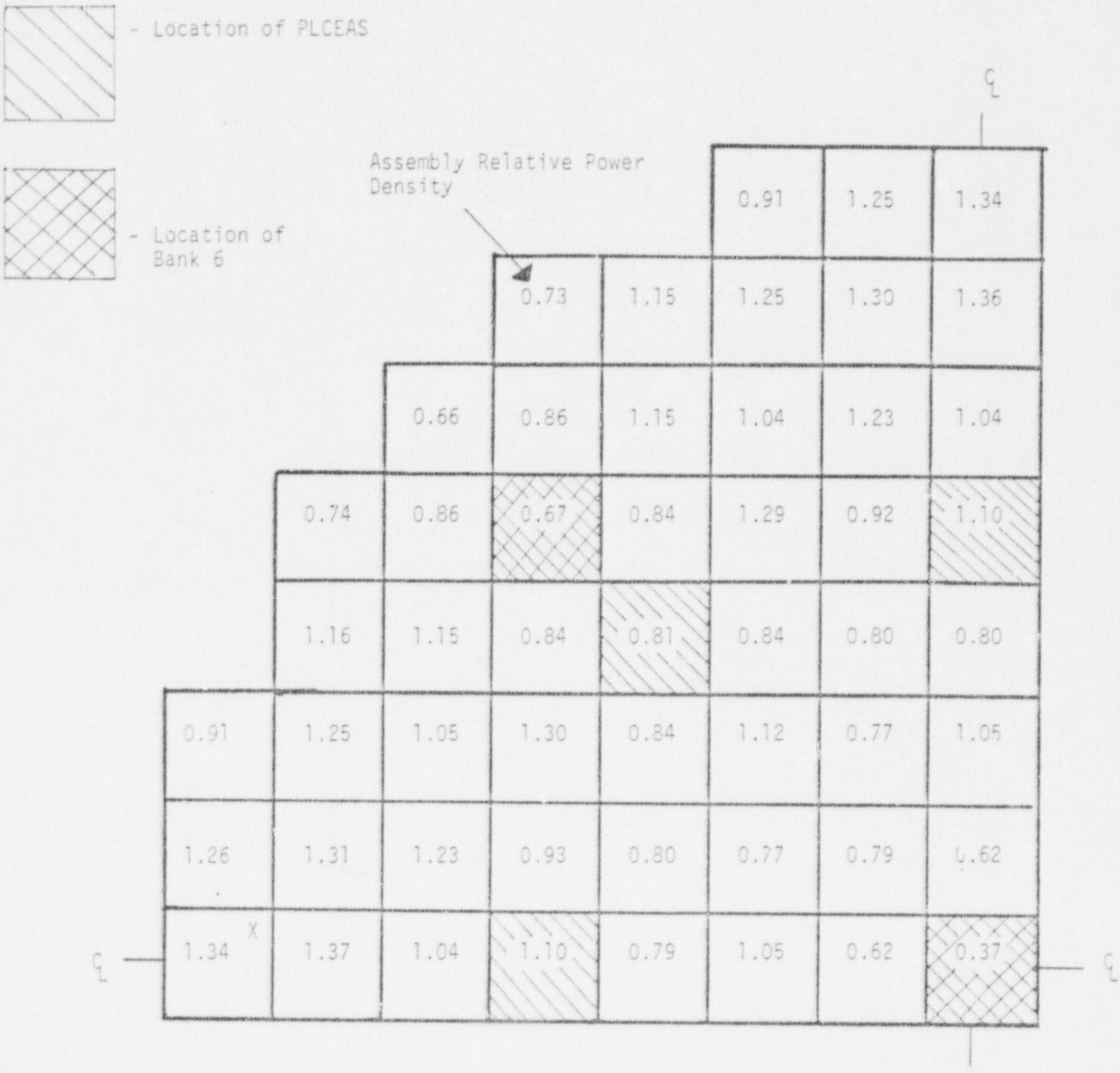
- Location of Bank 6

Assembly Relative Density

					0.86	1.19	1.27
			0.70	1.10	1.19	1.25	1.32
		0.63	0.83	1.11	1.02	1.23	1.07
	0.70	0.83	0.66	0.85	1.31	0.98	1.35
	1.11	1.12	0.86	0.96	0.88	0.84	0.85
0.86	1.20	1.03	1.32	0.88	1.16	0.79	1.08
1.19	1.26	1.23	0.98	0.84	0.79	0.80	0.63
1.27	X 1.32	1.07	1.35	0.85	1.08	0.63	0.37

NOTE: X = Location of Maximum 1-Pin Peak = 1.68

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 2	ARKANSAS NUCLEAR ONE UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 6, HFP, BOC	FIGURE 5-6



NOTE: X = Location of Maximum 1-Pin Peak = 1.77

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 2	ARKANSAS NUCLEAR ONE UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 6 AND PLCEAS, HFP, BOC	FIGURE 5-7

- Location of PLCEAS

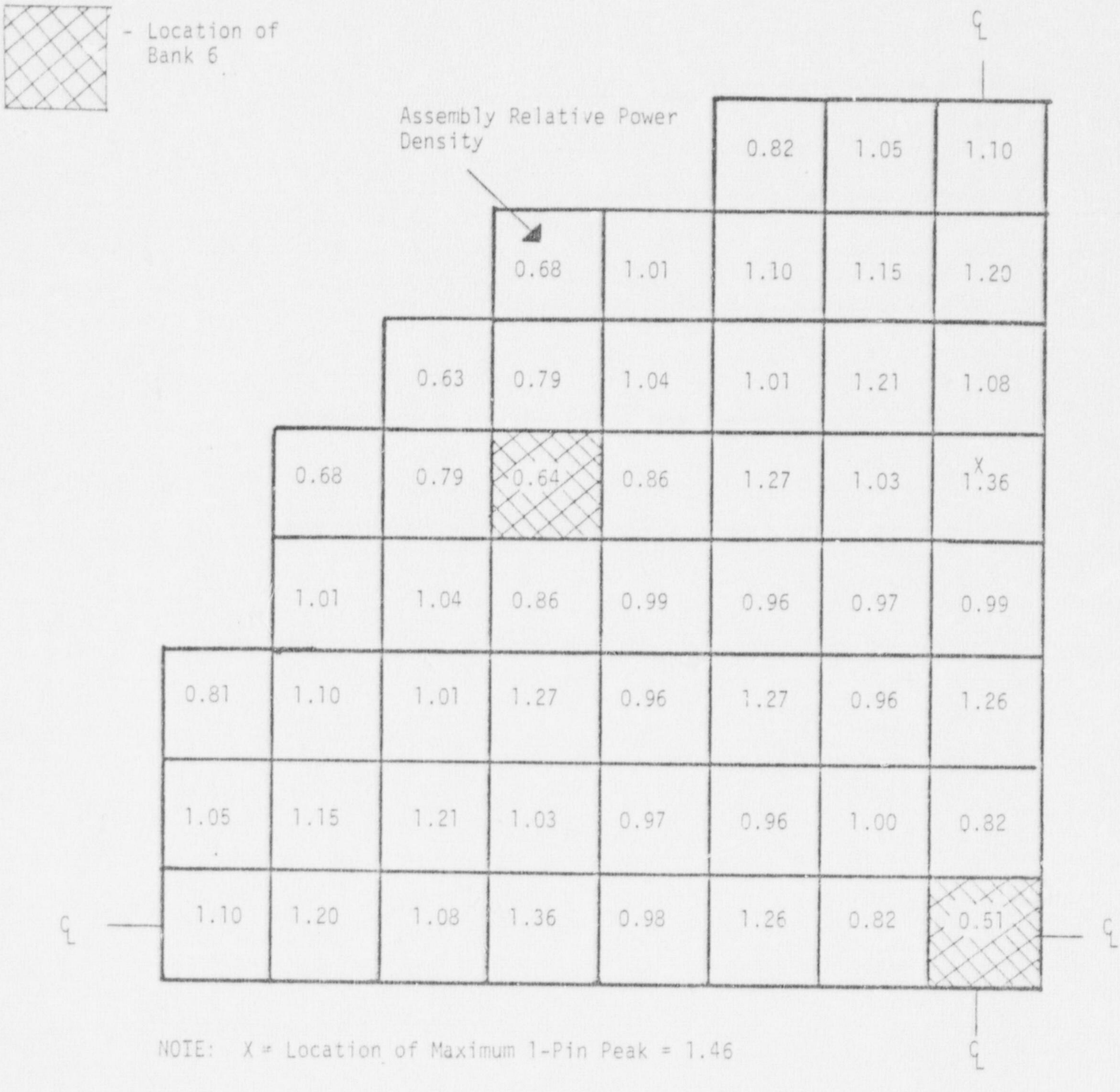
Assembly Relative Power Density

					0.79	0.99	1.02
			0.82	1.10	1.09	1.08	1.09
		0.87	1.06	1.19	1.00	1.10	0.93
	0.82	1.06	1.18	0.98	1.23	0.90	0.98
	1.10 X	1.19	0.99	0.87	0.89	0.87	0.85
0.79	1.09	1.00	1.23	0.89	1.18	0.90	1.20
0.99	1.08	1.10	0.90	0.87	0.91	1.01	0.89
1.02	1.09	0.93	0.98	0.85	1.20	0.89	0.80

℄ ℄ ℄ ℄

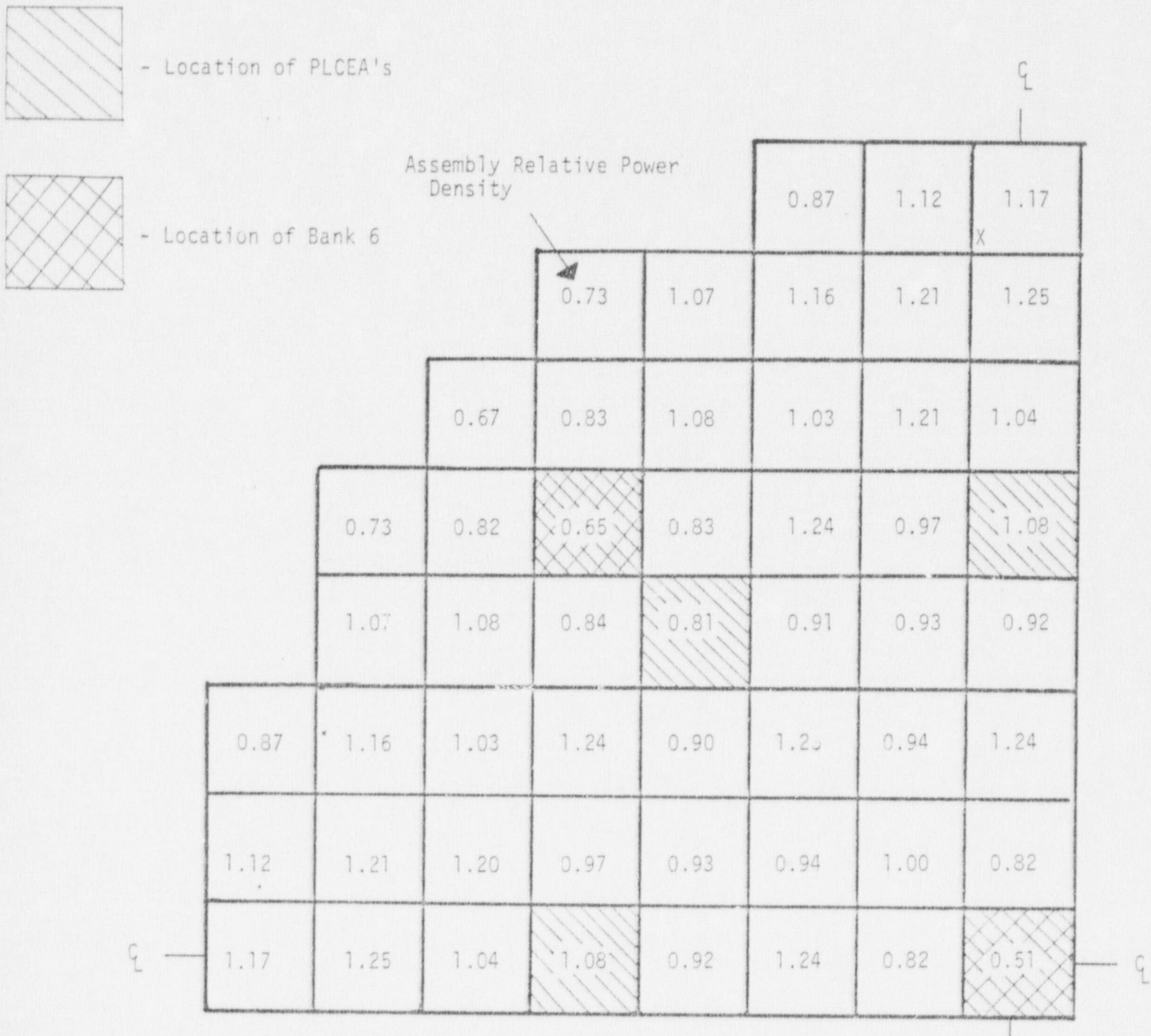
NOTE: X = Location of Maximum 1-Pin Peak = 1.39

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE -UNIT 2	ARKANSAS NUCLEAR ONE UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY WITH PLCEAS, HFP, EOC	FIGURE 5-8



NOTE: X = Location of Maximum 1-Pin Peak = 1.46

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE - UNIT 2	ARKANSAS NUCLEAR ONE UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 6, HFP, EOC	FIGURE 5.9



NOTE: X= Location of Maximum 1-Pin Peak = 1.49

ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE -UNIT 2	ARKANSAS NUCLEAR ONE UNIT 2 CYCLE 2 ASSEMBLY RELATIVE POWER DENSITY WITH BANK 6 AND PLCEAS, HFP, EOC	FIGURE 5-10

6.0 Thermal Hydraulic Design

6.1 DNBR Analysis

Steady state DNBR analyses of Cycle 2 at the rated power level of 2815 MWT have been performed using the TORC computer code described in Reference 6-1, the CE-1 critical heat flux correlation described in Reference 6-2, the simplified modeling methods described in Reference 6-3, and the CETOP code described in Reference 6-4.

CETOP will be employed in Cycle 2 for on-line DNBR calculations (see Section 9). Use of CETOP as the thermal hydraulic code in safety analysis assures consistency between the safety analysis and the monitoring and protection systems.

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters used for safety analyses and for generating reactor protection system setpoint information. Using the methodology presented in Reference 6-5 and 6-6, the calculational factors (engineering heat flux factor, engineering factor on hot channel heat input and the rod pitch, bowing and clad diameter factor) listed in Table 6-1 have been combined statistically with other uncertainty factors at the 95/95 confidence/probability level to define a new design limit on CE-1 minimum DNBR when iterating on power, as discussed in Reference 6-7.

6.2 Effects of Fuel Rod Bowing on DNBR Margin

Effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the same manner as discussed in Reference 6-8. This reference contains penalties on minimum DNBR due to fuel rod bowing as a functi [illegible] p using NRC guidelines contained in Reference 6-9.

TABLE 6-1

Arkansas Nuclear One Unit 2
Thermal Hydraulic Parameters at Full Power

General Characteristics	Unit	Reference Cycle 1	Cycle 2
Total Heat Output (Core only)	MWt 10^6 Btu/hr	2815 9608	2815 9608
Fraction of Heat Generated in Fuel Rod	-	0.975	0.975
Primary System Pressure			
Nominal	psia	2250	2250
Minimum in steady state	psia	2203	2203
Maximum in steady state	psia	2297	2297
Inlet Temperature (Maximum Indicated)	°F	554.7	554.7
Total Reactor Coolant Flow (Minimum Steady State)	gpm 10^6 lb/hr	322,000 120.4	322,000 120.4
Coolant Flow Through Core (Minimum)	10^6 lb/hr	116.2	116.2
Hydraulic Diameter (Nominal channel)	ft	0.039	0.039
Average Mass Velocity	10^6 lb/hr-ft^2	2.60	2.60
Pressure Drop Across Core (Minimum steady state flow irreversible Δp over entire fuel assembly)	psi	15.6	15.5
Total Pressure Drop Across Vessel (Based on nominal dimensions and minimum steady state flow)	psi	35.9	35.9
Core Average Heat Flux (Accounts for fraction of heat generated in fuel rod and axial densification factor)	BTU/hr-ft^2	184,720	184,720*
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	50,707	50,707*
Film Coefficient at Average Conditions	BTU/hr-ft^2°F	6200	6200
Average Film Temperature Difference	°F	30.6	30.6
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for fraction of heat generated in fuel rod)	kw/ft	5.40	5.40*
Average Core Enthalpy Rise	BTU/lb	82.7	82.7
Maximum Clad Surface Temperature	°F	656.5	656.5

TABLE 6-1 (continued)

Calculational Factors	Reference Cycle 1	Cycle 2
Engineering Heat Flux Factor	1.03	1.025++
Engineering Factor on Hot Channel Heat Input	1.03	1.008+, ++
Rod Pitch, Bowing and Clad Diameter Factor	1.05	1.05 ++
Fuel Densification Factor (Axial)	1.002	1.002

NOTES:

* Based on 1128 shims.

\+ Based on "as-built" information.

++ These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6-7.

8.0 ECCS Analysis

An ECCS performance analysis was performed for Arkansas Nuclear One - Unit 2 Cycle 2 to demonstrate compliance with 10CFR 50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled reactors[8-1]. The analysis justifies an allowable peak linear heat generation rate (PLHGR) of 14.5 kw/ft. This PLHGR is equal to the existing limit for ANO-2. The method of analysis and detailed results which support this value are presented herein.

8.1 METHOD OF ANALYSIS

The analysis for ANO-2 Cycle 1 operation[8-5], approved by the NRC, was used as the reference cycle analysis for this evaluation. The method of analysis used the NRC approved C-E evaluation model[8-2]. The model was used to re-evaluate the limiting large break LOCA performance. The blowdown and refill-reflood hydraulic calculations employed in Cycle 1 apply to Cycle 2 since only fuel pin conditions have changed between the two cycles. Therefore, only the STRIKIN-II[8-3] and PARCH[8-6] calculations were necessary to account for the effect of the different fuel pin conditions.

Burnup dependent calculations were performed using the FATES[8-4] and STRIKIN-II[8-3] codes to determine the limiting condition for the ECCS performance analysis.

8.2 RESULTS

Table 8-1 presents the analysis results reported for the 1.0 DEG/PD* break. The results of the evaluation confirm that 14.5 kw/ft is an acceptable value for the PLHGR in Cycle 2. The peak clad temperature and maximum local and core wide clad oxidation values as shown in Table 8-1 are below 10CFR50.46 acceptance limits.

Table 8-2 presents a list of the significant parameters displayed graphically for the limiting 1.0 DEG/PD break.

Table 8-3 presents a comparison of significant parameters for Cycles 1 and 2.

---

*DEG/PD = Double-Ended Guillotine at Pump Discharge

8.3 EVALUATION OF RESULTS

For the Cycle 2 analysis, the peak clad temperature (PCT) was calculated to occur at the rupture location. For Cycle 1 it occurred above the rupture location. The reason for the change in location of the PCT is as follows. For the Cycle 1 analysis, steam cooling heat transfer coefficients (HTC) calculated by the PARCH code were used only at the rupture location. Above rupture, conservatively low steam cooling HTC were used to simplify the calculations in the presence of the large margin in the ECCS performance results. For the Cycle 2 analysis, to counteract the effect of higher stored energy and a more limiting radiation enclosure, PARCH-calculated steam cooling HTC were used at and above the rupture location. This lowered the calculated clad temperatures above the rupture location in the Cycle 2 analysis relative to the Cycle 1 analysis. It also lowered them sufficiently to change the location of PCT to the rupture location for Cycle 2.

The net result was a PCT of 2041°F which is 37°F lower than in Cycle 1 and a lower peak clad oxidation by 0.02% as shown in Table 8-1.

8.4 CONCLUSION

As discussed above, conformance to the ECCS criteria is summarized by the analysis results presented in Table 8-1. The results of the analysis identified the peak clad temperature as 2041°F as opposed to the acceptance limit of 2200°F. The peak local clad oxidation was 11.8% versus the acceptance limit of 17% and the peak core wide clad oxidation was less than .621% versus the acceptance limit of 1.0%. Hence, Cycle 2 operation at a peak linear heat generation rate of 14.5 kw/ft and at a power level of 2871 Mw_t (102% of 2815 Mw_t) will result in acceptable ECCS performance.

8.5 COMPUTER CODE VERSION IDENTIFICATION

The following versions of the NRC approved Combustion Engineering ECCS Evaluation Model computer codes were used in this analysis:

STRIKIN-II: Version No. 77036
PARCH: Version No. 77004

Table 8-1

Arkansas Nuclear One-Unit 2 Cycle 2
ECCS Analysis Results
for Limiting Break Size 1.0 DEG/PD)

Analysis	Maximum Blowdown Clad Temperature	Peak Clad Temperature	Time of Peak Clad Temperature	Time of Clad Rupture	Peak Local Clad Oxidation	Total Core-Wide Clad Oxidation
Cycle 2	1696°F	2041°F	221. sec	45.8 sec	11.8%	<.621%
Reference Cycle (Cycle 1)	1660°F	2078°F	230. sec	42.3 sec	11.82%	<.617%

Table 8-2

Arkansas Nuclear One - Unit 2 Cycle 2

Analysis Plots

Variables	Figure Designation
Peak Clad Temperature	8-1
Hot Spot Gap Conductance	8-2
Peak Local Clad Oxidation	8-3
Clad Temperature, Centerline Fuel Temperature, Average Fuel Temperature and Coolant Temperature for Hottest Node	8-4
Hot Spot Heat Transfer Coefficient	8-5
Hot Rod Internal Gas Pressure	8-6

Table 8-3

Arkansas Nuclear One - Unit 2 Cycle 2

Significant Parameters

Quantity	Cycle 1	Cycle 2
Reactor Power Level (102% of Nominal)	2882**	2882 MWt**
Average Linear Heat Rate (102% of Nominal)	5.66	5.65 kw/ft
Peak Linear Heat Generation Rate (PLHGR) Hot Assembly, Hot Channel	14.5	14.5 kw/ft
Peak Linear Heat Generation Rate (PLHGR) Hot Assembly, Average Channel	13.03	12.77 kw/ft
Gap Conductance at PLHGR*	1633	1411 BTU/hr-ft-°F
Fuel Centerline Temperature at PLHGR*	3478	3580 °F
Fuel Average Temperature at PLHGR*	2192	2278 °F
Hot Rod Gas Pressure*	1206	1007 psia
Hot Rod Burnup*	726	500 MWD/MTU

* Values at limiting hot rod burnup

** This value represents 102% of the core power plus reactor coolant pump power

FIGURE 8-1

ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2

1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG

PEAK CLAD TEMPERATURE

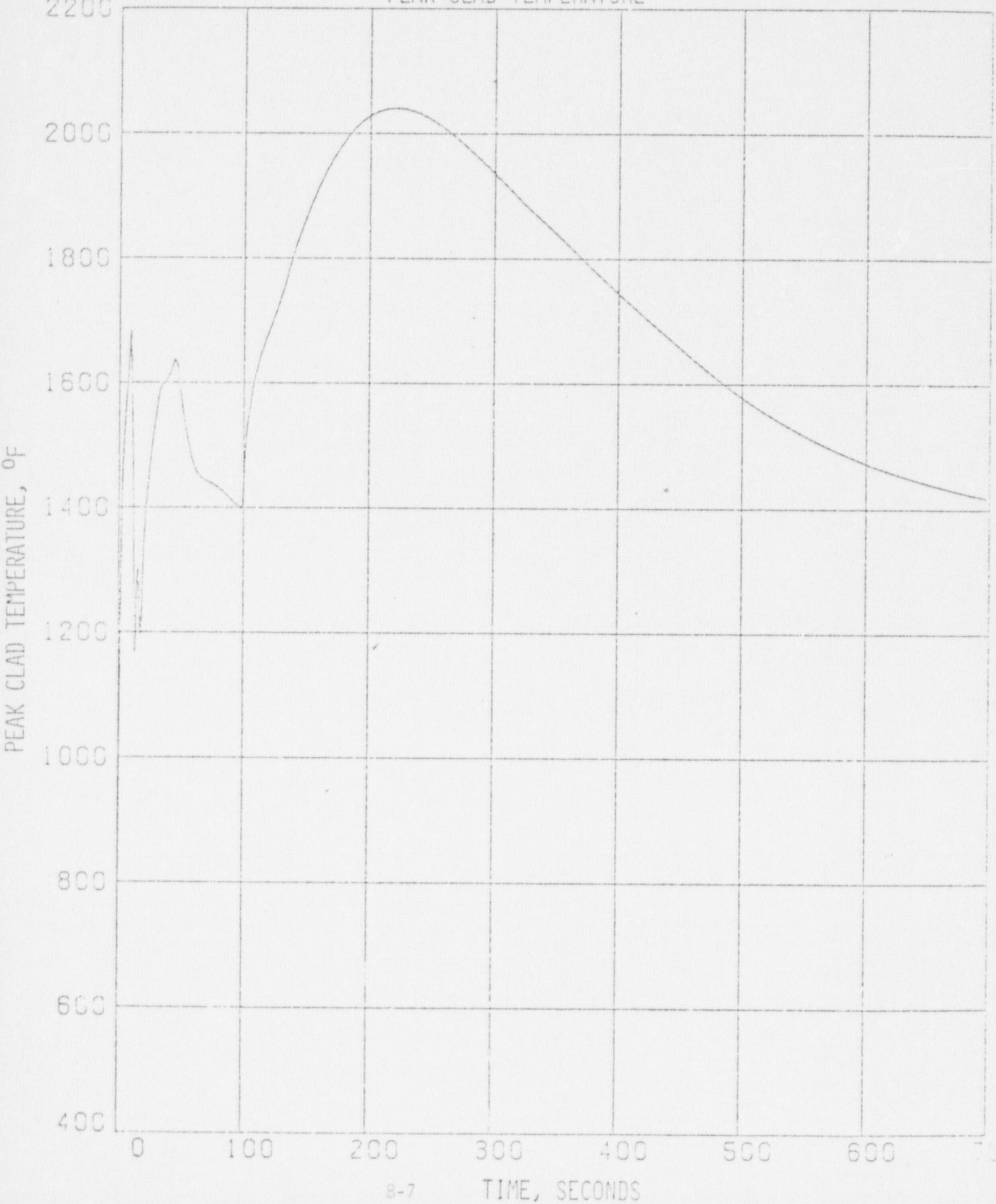


FIGURE 8-2
ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
HOT SPOT GAP CONDUCTANCE

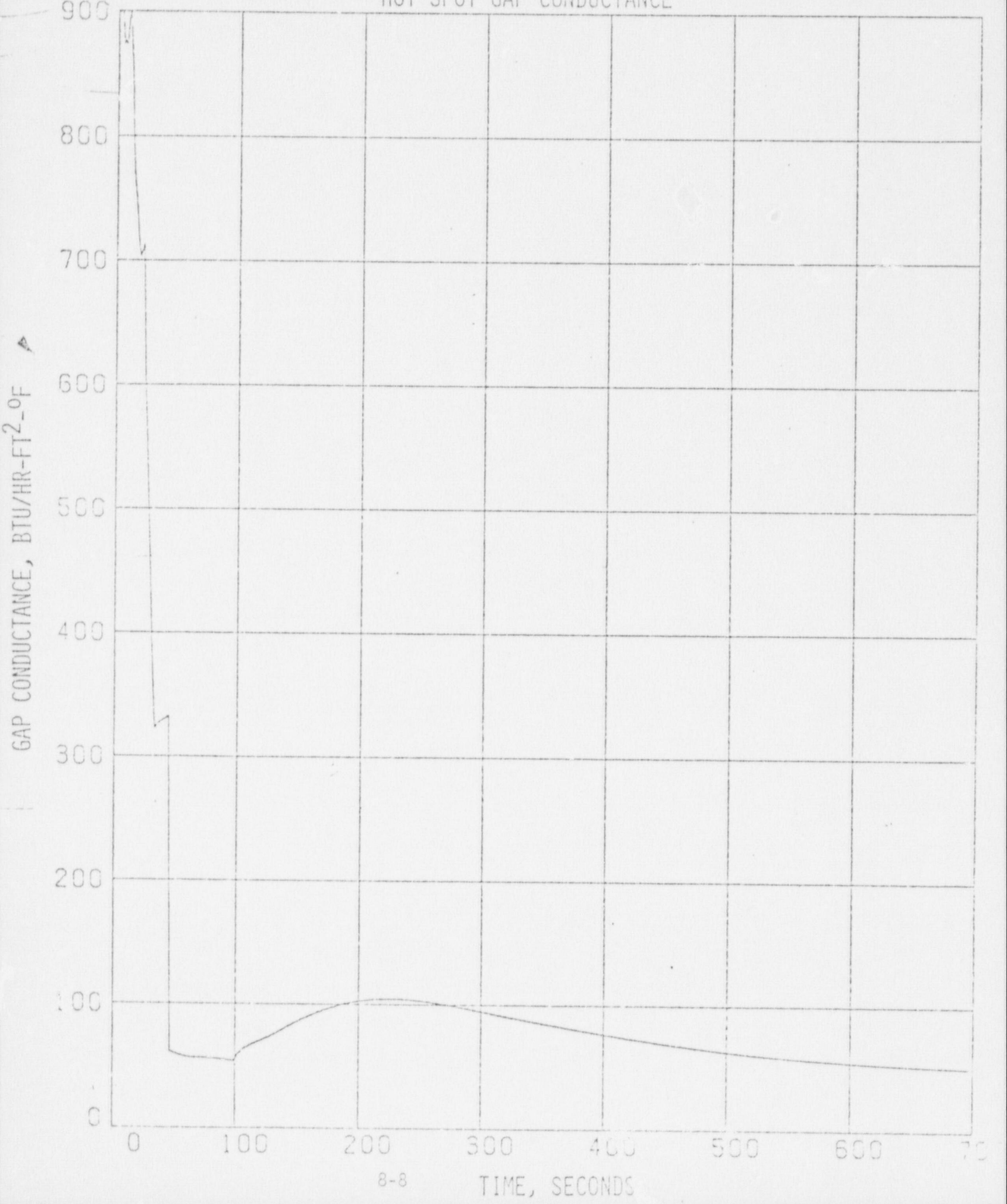


FIGURE 8-5
ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PEAK LOCAL CLAD OXIDATION

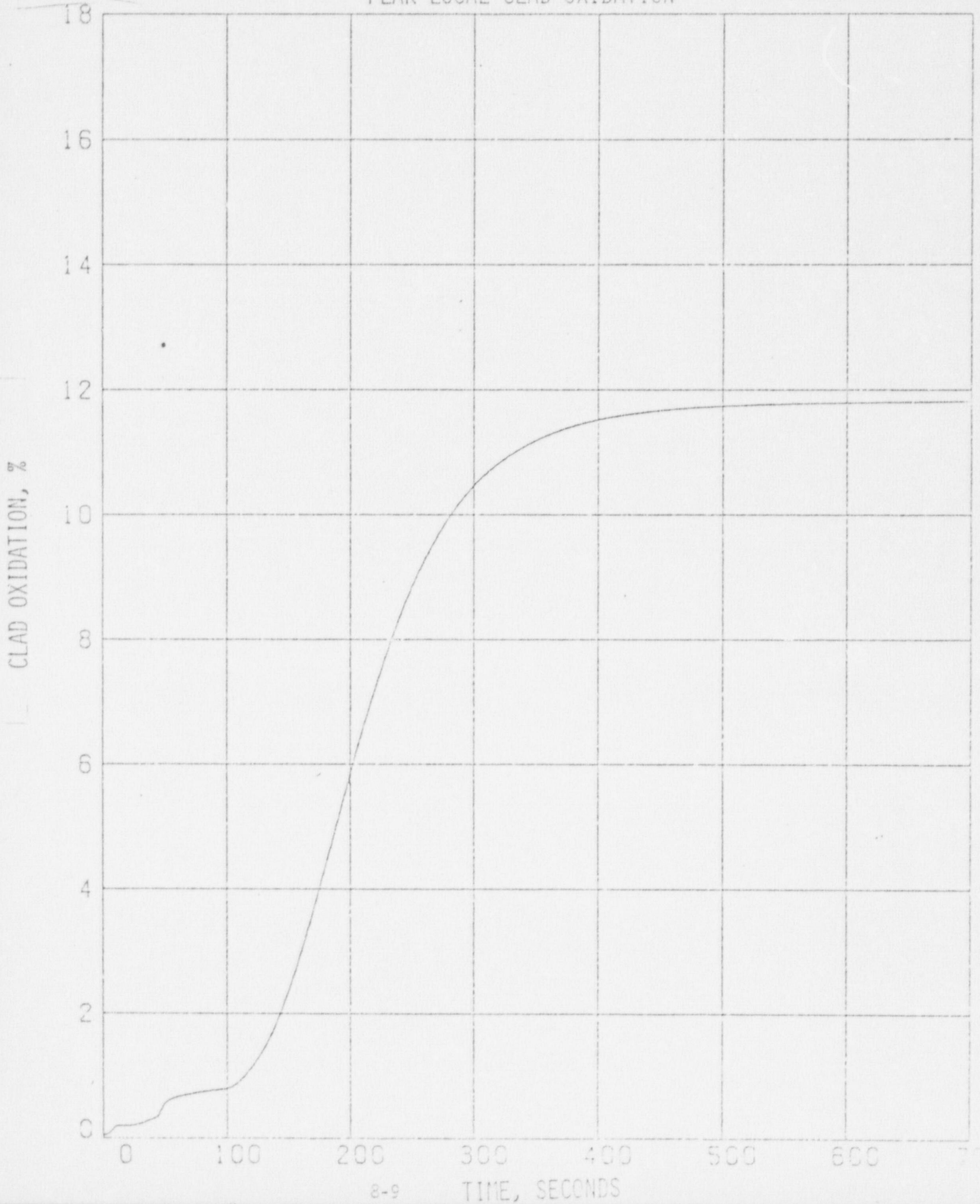


FIGURE 8-4

ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2

1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG

CLAD TEMPERATURE, CENTERLINE FUEL TEMPERATURE, AVERAGE FUEL TEMPERATURE AND COOLANT TEMPERATURE FOR HOTTEST NODE

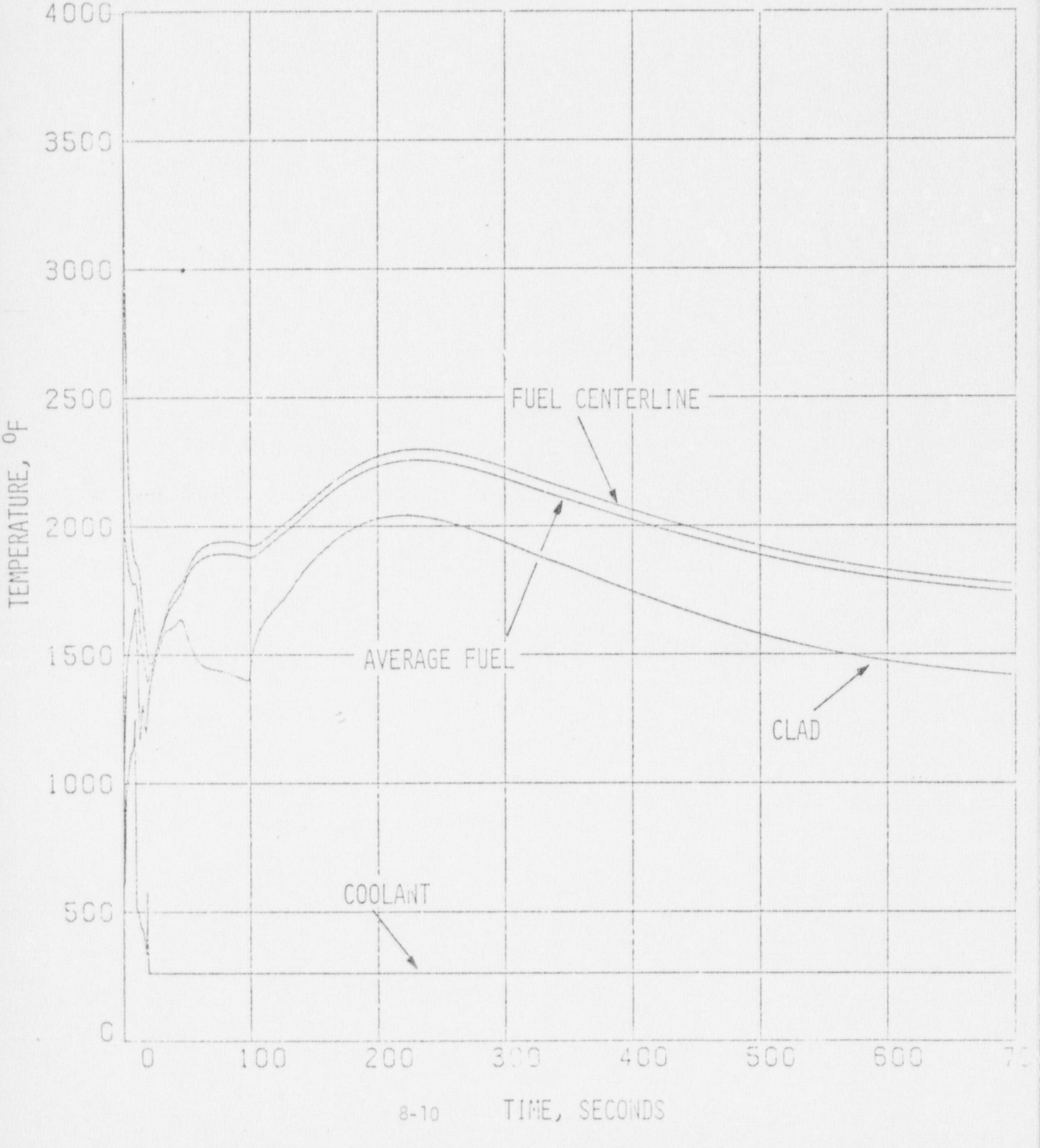


FIGURE 8-5

ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2

1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG

HOT SPOT HEAT TRANSFER COEFFICIENT

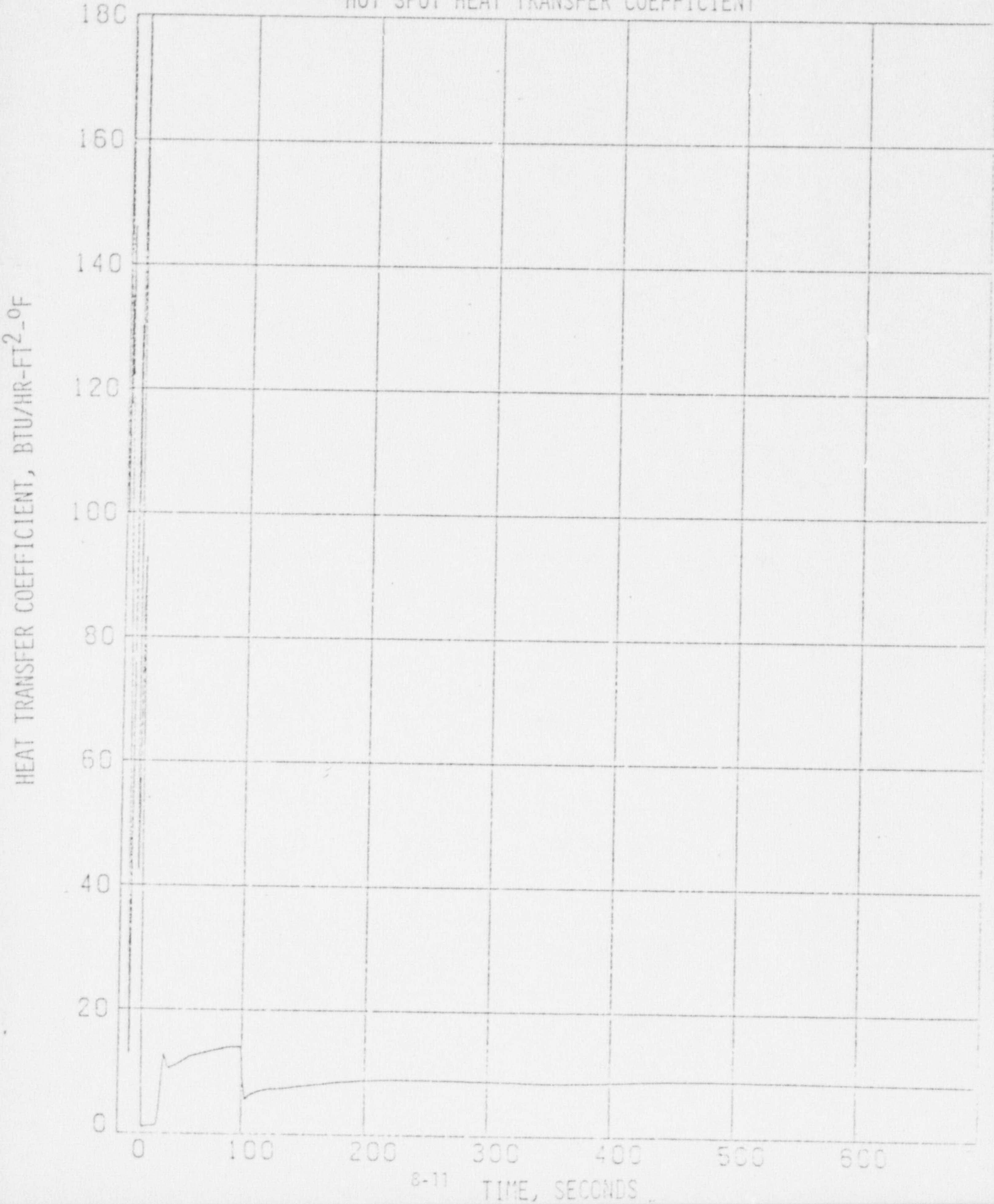
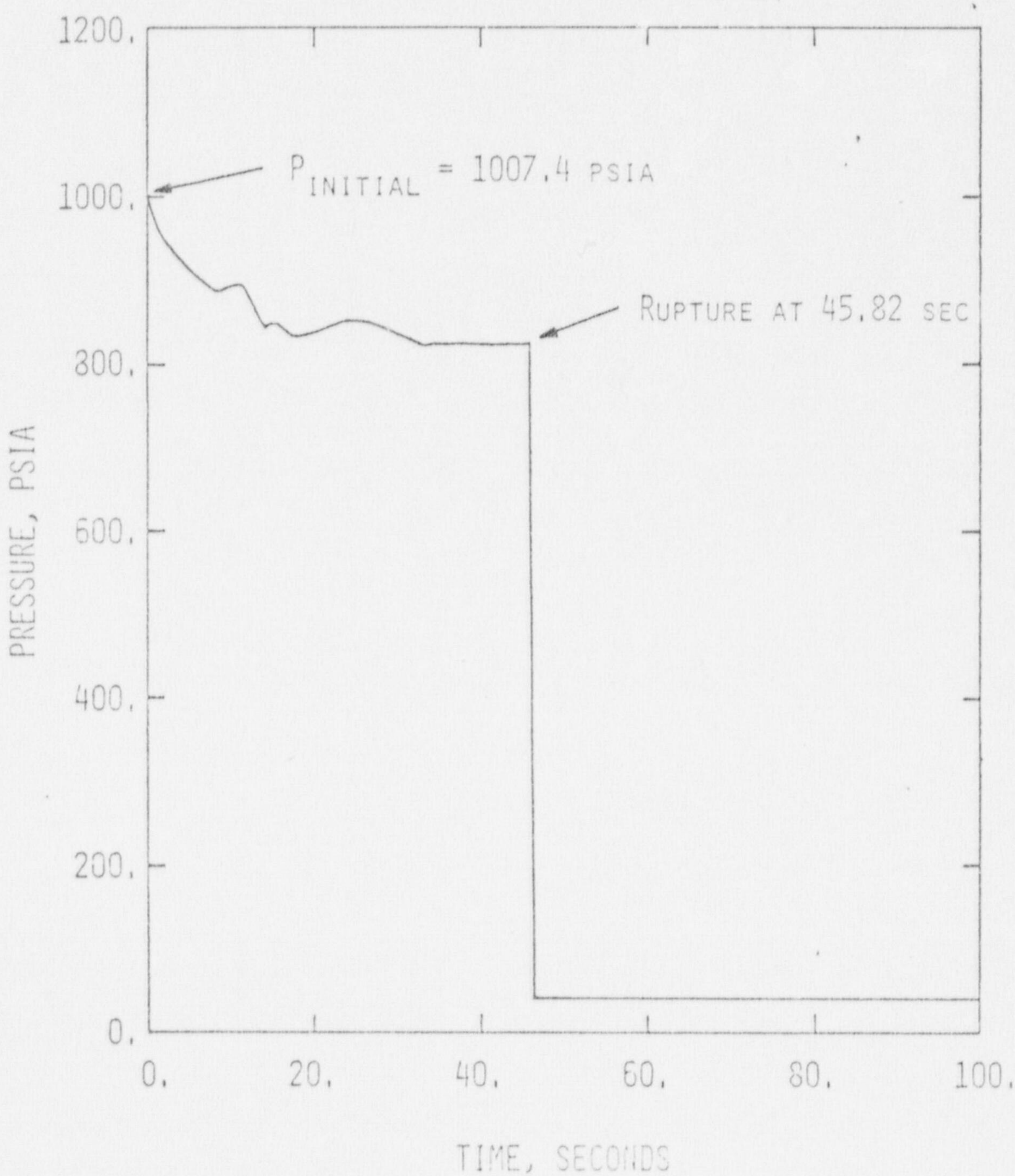


FIGURE 8-6
ARKANSAS NUCLEAR ONE - UNIT 2 CYCLE 2
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
HOT ROD INTERNAL GAS PRESSURE



9.0 Reactor Protection and Monitoring System

9.1 Introduction

The Core Protection Calculator System (CPCS) is designed to provide the low DNBR and high local power density (LPD) trips (1) to ensure that the specified acceptable fuel design limits on departure from nucleate boiling and centerline fuel melting are not exceeded during Anticipated Operational Occurrences (AOO) and (2) to assist the Engineered Safety Features System in limiting the consequences of certain postulated accidents.

The CPCS in conjunction with the remaining Reactor Protection Systems (RPS) must be capable of providing protection for certain specified design basis events, provided that at the initiation of these occurrences the Nuclear Steam Supply System, its systems, components and parameters are maintained within operating limits and limiting conditions for operation (LCO's).

Since the original CPC System licensing and implementation effort for ANO-2, two software modifications have been added to reduce the probability of unnecessary plant trips and to provide the operator with additional diagnostic information. For Cycle 2 additional CPC and CEA Calculator (CEAC) software modifications have been developed and will be verified in accordance with the software change procedures contained in References 9-1 and 9-2. These modifications are due to implementation of the TORC/CE-1 DNBR calculation, experience gained during first cycle operation, implementation of features to make the ANO-2 CPC/CEAC algorithms generic with CPC/CEAC algorithms for future plants, and implementation of improved diagnostic capabilities for the operator.

9.2 CPC Software Modifications

CPC software modifications for Cycle 2 are summarized in this section. A detailed description of the modifications is presented in Reference 9-3.

9.2.1 DNBR Calculation

a. The COSMO/W-3 based DNBR calculation in the STATIC program is replaced with one based on the TORC/CE-1 DNBR calculation (CETOP2).

b. The DNBR update calculation in the UPDATE program has been replaced with a DNBR update calculation which is based on the TORC/CE-1 DNBR calculation being implemented in the STATIC program.

c. New curve fits have been made for determining the core coolant enthalpy/temperature ratio.

9.2.2 Generic Software Changes

a. The pump-dependent uncertainty on the local power density (LPD) is applied in the DNBR and LPD Update (UPDATE) program instead of the Trip Sequence program.

b. Some fixed numbers in the POWER calculation have been changed to Data Base constants.

c. The Control Element Assembly Calculator (CEAC) logic has been modified to allow for a two (2) CEA subgroup.

d. The update period for the CEAC CRT display has been changed to 3.0 seconds.

e. The CPC Point ID table has been revised.

9.2.3 Addressable Constants

a. Addressable constants have been added for:
   1. CEA shadowing factor adjustments,
   2. planar radial peaking factor adjustments, and
   3. boundary point power correlation coefficients; the application of these coefficients has also been simplified.

b. The slope of the coolant temperature shadowing factor has been made an addressable constant.

c. The DNBR and LPD pre-trip setpoints have been made addressable constants.

d. A method for automated reentry of certain CPC addressable constants from a disk has been implemented.

9.2.4 Other Algorithm Changes

a. Planar radial peaking factors are now adjusted by a correction factor based on the value of the normalized core inlet moderator density.

b. New curve fits have been made for core coolant specific volume determination.

c. A pre-selected 20-node axial power distribution is used during low power operation.

9.2.5 Diagnostic Changes

a. The CEAC snapshot buffer is changed to be similar to that for the CPC's.

b. The failed sensor ID numbers used to indicate large differences in the CPC penalty factors have been changed.

c. Logic and an addressable constant have been added to the CEAC to rewrite the entire CEAC display.

9.2.6 Data Constant Changes

In addition to the data constants associated with the above-listed modifications, data constants which could be potentially cycle-dependent are being reviewed for ANO-2 Cycle 2.

9.3 Addressable Constants

Certain CPC constants are addressable so that they can be changed as required during operation. Addressable constants include data that are measured during startup (e.g., shape annealing matrix, temperature shadowing, boundary point power correlation coefficients, and adjustments for CEA shadowing and planar radial peaking factors, see Section 12), uncertainty factors to account for processing and measurement uncertainties in DNBR and LPD calculations (BERRO through BERR4), and miscellaneous items (e.g., pre-trip setpoints, EOL flag, CEAC inoperable flag, etc).

The uncertainty factors to account for processing and measurement uncertainties in DNBR and LPD calculations will be determined for Cycle 2 in a manner similar to that for the reference cycle (Reference 9-4). Measurement uncertainties will be modelled stochastically, including pressure, inlet temperature, flow rate, CEA position, radial peaking factor, processing errors, and noise, to ensure to a 95/95 probability/confidence level that CPC provides a conservative DNBR and LPD calculation.

9.4 Digital Monitoring System (COLSS)

The Core Operating Limit Supervisory System (COLSS) is a monitoring system that initiates alarms if LCO's on DNBR, peak linear heat rate, core power, or core azimuthal tilt are exceeded. To be consistent with the safety analysis and CPC modifications for Cycle 2, COLSS will be modified in several areas. The DNB algorithm will be based on the TORC/CE-1 DNBR calculation (CETOP1). Power distribution and thermal hydraulic data and uncertainties will be updated for Cycle 2 in a manner similar to that used for CPC as discussed above.

10.0 Fuel Utilization Program

10.1 Program Description

AP&L has initiated a Department of Energy (DOE) funded program to improve fuel utilization through more efficient fuel management and an increase in fuel burnup. A complete description of the DOE program is provided in Reference 10-1. Demonstrations and analyses of the improvement in uranium utilization are planned for the ANO-2 reactor including testing of advanced fuel designs which eventually may allow the batch-average exposure to be increased to 53 GWD/T.

The proposed program is planned for implementation in two phases. Phase I involves the development of selected concepts for improved uranium utilization and culminates in demonstration of those concepts in the ANO-2 reactor. If a commitment is made to perform Phase II at ANO-2, it will involve the large scale implementation of the demonstrated fuel utilization improvements. This section addresses the Phase I program as it impacts ANO-2 Cycle 2.

Annular fuel pellets, large grain size pellets and graphite ID coated clad have been selected for inclusion in the advance design demonstration program. These variables are used singly and in combination in segmented and full length rods. The segmented rod design consists of short and long segments to allow for a variety of safety and performance tests which could be performed in future programs. Forty-two (42) test rods will be inserted in two Batch D assemblies to be loaded in the reactor at the start of Cycle 2.

10.2 High Burnup Demonstration Assemblies: Description And Fuel Design

10.2.1 Assembly Description And Placement

The two high burnup demonstration assemblies are Batch D assemblies consisting of 3.48 w/o U-235 fuel zoned with 3.03 w/o U-235 fuel. Twenty one (21) advanced design test fuel rods, consisting of both

full length and segmented rods, will replace standard 3.48 w/o U-235 Batch D fuel rods in each of the two fuel bundles. The matrix of test fuel rods to be placed in each of the fuel bundles is shown in Table 10-1. The positioning of the two demonstration Batch D assemblies is shown on Figure 3-2.

10.2.2 Advanced Fuel Design

Each of the DOE Batch D high burnup demonstration assemblies will contain 21 test fuel rods of various advanced designs which are judged to allow increased fuel burnup. These advanced designs include annular fuel pellets, large grain size pellets and graphite ID coated clad all used either alone or in combinations in either full length or segmented fuel rods. The annular fuel pellets consist of a standard fuel pellet with an 8 volume percent hole through the radial center of the pellet. The large grain size fuel pellets were fabricated by the addition of Nb_2O_5 as a dopant and a controlled sintering process which tailored the grain size to the desired size. (≈40 μ vs. 7 to 8 μ for standard pellets).

Finally, some of the test fuel rods were internally coated with graphite to provide a vehicle for improving dimensional stability and fuel performance.

10.3 Performance Analysis

An analysis was performed to determine the impact of the segmented fuel rods and the advanced design fuel rods to be inserted into the two Batch D demonstration assemblies.

Those advanced design full length rods containing either large grain size pellets or graphite ID coated cladding separately or in combination but without annular pellets do not impact the neutronics and power distribution of the assembly. Mechanical performance characteristics are expected to be better than the standard design Batch D fuel rods.

Fuel rod types that effect power distribution are segmented fuel rods and fuel rods containing annular fuel pellets.

Annular fuel pellets contain less fissile uranium than standard pellets and therefore rods which contain annular pellets are expected to have lower power peaking than standard 3.48 w/o Batch D fuel rods. These fuel rods will be explicitly included in the physics calculations to be used for power distribution measurements and core follow analysis.

The segmented fuel rods (shown in Figure 10-1) contain one large and several smaller nonfuel regions (segment plenums plus connectors) inside the active region of the core. Nonfuel regions such as these tend to cause increased power peaking in adjacent fuel rods. It was determined that the increase in peaking caused by the nonfuel region was not large enough to cause any fuel rod in either demonstration assembly to be limiting.

A helium fill pressure of 475 psig was used in the demonstration segmented fuel rods as compared to 380 psig in the standard Batch D and the demonstration full length rods. This higher fill pressure was used because of the greater void-to-fuel volume ratio for the segmented rods.

C-E has performed analytical predictions of the cladding creep-collapse time for all the demonstration fuel rods that will be irradiated in Cycle 2 and has concluded that the collapse resistance of these demonstration fuel rods is sufficient to preclude collapse during Cycle 2.

As a result of the analyses described in this section, we have concluded that the advanced design fuel rods in the Batch D high burnup demonstration assemblies will not impact the corewide performance in Cycle 2.

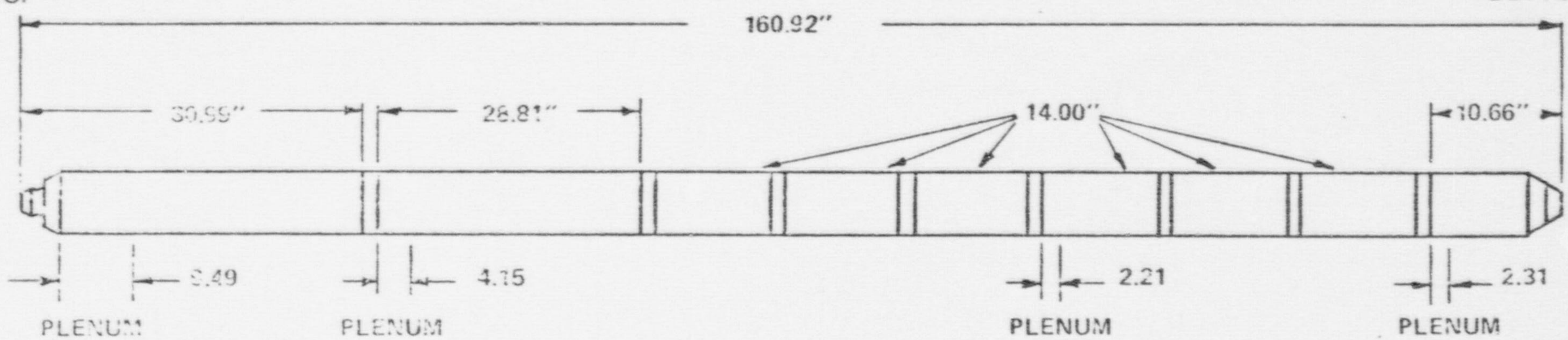
Table 10-1

High Burnup Fuel Designs-Test Variables Matrix

Design Variables	Number of Rods: Segmented Strings	Full Length	Total
Standard Fuel in Standard Cladding (Experimental Control)	1	2	3
Standard Fuel in Graphite ID Coated Cladding	2	4	6
Standard Geometry, Large Grain Size Fuel in Standard Cladding	1	2	3
Annular Geometry Fuel in Standard Cladding	1	2	3
Annular Geometry, Large Grain Size Fuel in Standard Cladding	1	2	3
Annular Geometry Fuel in Graphite ID Coated Cladding	1	2	3
	7	14	21

FIGURE 10-1

SEGMENTED ROD DESIGN



CONNECTOR DESIGN

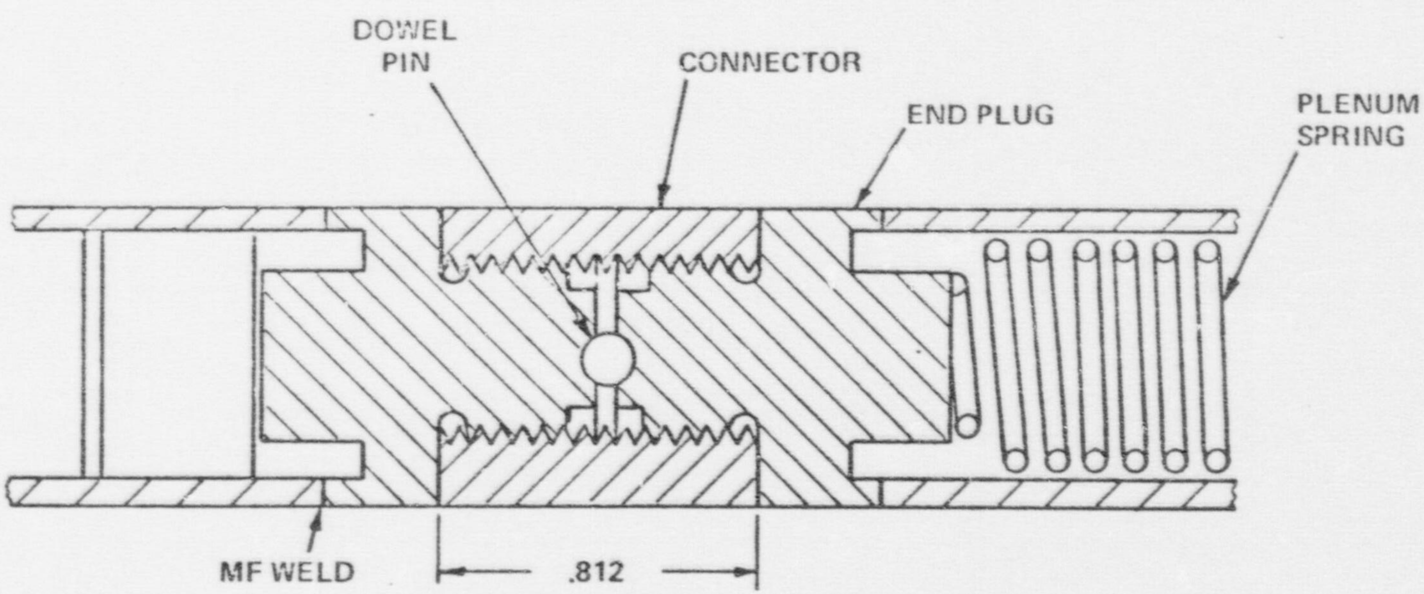


FIGURE 10-1

12.0 Startup Program

The planned startup test program associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide confirmation for continued safe operation of the unit.

12.1 Precritical Tests

12.1.1 Control Element Assembly (CEA) Trip Test

Precritical Control Element Assembly drop times are recorded for all CEA's (81) at hot full flow conditions before low power physics testing begins. Acceptance criteria state that the CEA drop time from fully withdrawn to 90% inserted shall be less than 3.0 seconds at the stated conditions.

12.1.2 Reactor Coolant Flow Coastdown

The coastdown of reactor coolant flow from the onset of a 2 out of 4 reactor coolant pump trip will be measured at hot standby conditions. The response time of the Core Protection Calculators will then be compared to the required response time from Technical Specification 3.3.1.1 to determine if acceptance is satisfied.

12.2 Low Power Physics Tests

12.2.1 Critical Boron Concentration

Criticality is obtained by deboration at a constant dilution rate with all control rods except the lead control group fully withdrawn. Once criticality is achieved, equilibrium boron concentration is obtained. The critical boron concentration is calculated by correcting the actual boron concentration for any deviation of CEA position from the reference CEA position for the predicted critical boron concentration. Acceptance criteria states that the critical boron concentration shall be within 100 ppm of the predicted value.

12.2.2 CEA Symmetry Test

CEA symmetry is verified by inserting a reference CEA of a symmetric group to it's Lower Electrical Limit (LEL) and compensating for the reactivity change by CEA Group 6 withdrawal. CEA's in the symmetric group are then traded with each other and the reactivity change measured to determine the deviation from the reference CEA. The reference CEA is traded for the last CEA in the symmetric group to measure total reactivity drift. For each CEA of the symmetric group, with the exception of the reference CEA, an average drift adjusted deviation from the reference CEA is calculated. Acceptance criteria states that the adjusted deviation for each symmetric CEA shall be within ±1.5¢ of the group average adjusted deviation.

12.2.3 Temperature Reactivity Coefficient

The isothermal temperature coefficient is measured at approximately the all CEA withdrawn configuration and at the Zero Power Dependent Insertion Limit. The average coolant temperature is varied by first decreasing then increasing temperature by 10 degrees F. During the change in temperature, reactivity feedback is compensated by discrete CEA motion; change in reactivity is then calculated by the summation of reactivity (obtained from reactivity calculation on strip chart recorder) associated with the temperature change. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.2 \times 10^{-4}$ $\Delta k/k/$ °F.

The moderator temperature coefficient of reactivity is calculated in conjunction with the isothermal temperature coefficient measurement. After the isothermal temperature coefficient has been measured, a predicted value of fuel temperature coefficient of reactivity is subtracted to obtain the moderator temperature coefficient. The moderator temperature coefficient value must not be more positive than $+0.5 \times 10^{-4}$ $\Delta k/k/$°F.

12.2.4 Part Length Control Element Assembly (PLCEA) Reactivity Worth

PLCEA group reactivity worth is measured at hot zero power conditions using the Boron/PLCEA swap method. The Boron/PLCEA swap method consists of establishing a deboration rate in the RCS and compensating for the reactivity changes of this deboration by inserting PLCEA's in incremental steps. When the PLCEA's have been inserted to the point of maximum worth, the deboration will be stopped. A boration rate will then be established and compensation for the reactivity changes of this boration will be made by PLCEA insertion in incremental steps to their group stop. The incremental reactivity changes that occur during these measurements are calculated based on reactivity calculator data, and differential reactivity worth for each incremental step are obtained from the measured reactivity worth versus the change in PLCEA group position. The incremental reactivity values are then summed to obtain an integral PLCEA group worth. This measurement is made for information only. Therefore, no acceptance criteria are applied.

12.2.5 Regulating CEA Group Reactivity Worth

Regulating CEA group reactivity worths (Groups 2, 3, 4, 5, and 6) are measured at hot zero power conditions using the Boron/CEA swap method. The Boron/CEA swap method consists of establishing a deboration rate in the RCS and compensating for the reactivity changes of this deboration by inserting regulating groups 6, 5, 4, 3, and 2 in incremental steps. The incremental reactivity changes that occur during these measurements are calculated based on reactivity calculator data, and differential reactivity worths for each incremental step are obtained from the measured reactivity worth versus the change in CEA group position. The incremental reactivity values are then summed to obtain integral CEA group worths for each of the regulating groups. Acceptance criteria state that the measured individual group worths shall be within ±15% or ±0.1% $\Delta k/k$ (whichever is larger) of predicted values, and the total worth of all the groups shall be within ±10% of the predicted values.

12.3 Power Escalation Tests

12.3.1 Reactor Coolant Flow at 50% and 100% Full Power

Reactor coolant flow will be measured by calorimetric methods at steady state conditions. Acceptance criteria will require that the measured flow be within allowable limits and that both the Core Operating Limit Supervisory System (COLSS) and Core Protection Calculator (CPC) reactor coolant flowrates will be conservative with respect to the measured calorimetric flowrate.

12.3.2 Core Power Distribution at 50% and 100% Full Power

Steady state reactor power is established with 3-D Equilibrium Xenon. Incore detector data is collected for analysis. The measured results are compared to predicted values in the following manner:

A. The measured radial power distribution is compared to the predicted power distribution utilizing a root mean squared statistical comparison of the relative radial power density distribution for each of the 177 fuel assemblies. The acceptance criteria states that the comparison of the measured radial power distribution with the predicted radial power distribution shall satisfy the following expression:

$$\text{RMS} = \sqrt{\frac{\sum_{1}^{177} (100Z_i)^2}{177}} \leq 5$$

where Z_i is the difference between the predicted and measured relative power density distribution for the i^{th} fuel assembly.

B. The measured radial power distribution is additionally compared to the predicted power distribution utilizing a box-by-box comparison of the relative radial power density distribution for each of the 177 fuel assemblies. The acceptance criteria states that for each assembly with a predicted relative power density ≥ 0.9 the measured and predicted relative power density values must agree within $\pm 10\%$,

and for each assembly with a predicted relative power density < 0.9 the measured and predicted relative power density values must agree within ±15%.

C. The measured axial power distribution is compared to the predicted power distribution utilizing a root mean squared statistical comparison of the relative axial power distribution for each of the 101 axial nodes. The acceptance criteria states that the comparison of the measured axial power distribution with the predicted axial power distribution shall satisfy the following expression:

$$RMS = \sqrt{\frac{\sum_{0}^{100} (100h_i)^2}{101}} \leq .5$$

where h_i is the difference between the predicted and measured relative power density distribution for the i^{th} axial % of height.

D. The measured values of total planar radial peaking factor (F_{xy}), total integrated radial peaking factor (F_R), core average axial peak (F_Z), and 3-D power peak (F_Q) are compared to predicted values. The acceptance criteria states that the measured values of F_{xy}, F_R, F_Z and F_Q shall be within ±10% of the predicted values.

12.3.3 Shape Annealing Matrix (SAM) and Boundary Point Power Correlation (BPPC) Verification at 50% Full Power

The SAM coefficients and BPPC constants are determined from a least squares analysis of the measured excore detector readings and corresponding core power distribution determined from the incore detector signals. Since these values must be representative for a

rodded and unrodded core throughout life it is desirable to use as wide a range of core axial shapes as are available to establish their values. This is done by initiating an axial Xenon oscillation. Data is periodically gathered during the oscillations so that is will be representative of as wide a range of axial shapes as possible. Incore, excore and related data are recorded, and incore analysis is performed which relates the incore detector signals to power distribution and summarizes the necessary power distribution and excore detector data in a form and format which can be easily input to programs used to perform the least squares fitting. The incore analysis results include:

A. Excore detector fractional responses for each CPC;

B. Core peripheral power fractions for the upper, middle and lower third of the core for each quadrant;

C. Core average power fractions for the upper, middle and lower third of the core; and

D. Upper and lower core boundary average power.

The above output is used to determine a "best set" of SAM coefficients and BPPC constants by using least squares analysis. The results of these calculations are then compared to the precalculated values placed in the CPC's and, if necessary, the precalculated values are adjusted based on the "best set" measured results.

12.3.4 Radial Peaking Factor and CEA Shadowing Factor Verification at 50% Power

The performance of the test involves establishing the following CEA configurations:

All CEA's Out

Group 6 at LEL (Lower Electrical Limit)

Group 6 and 5 at LEL

Group 6 and 5 at LEL, Group P at 37.5" WD

Group 6 at LEL, Group P at 37.5" WD

Group P at 37.5"WD

As the various CEA configurations are established, incore detector data and excore detector data are taken. These data are analyzed and planar radial peaking factors (Fxy) and CEA shadowing factors are determined for each CEA configuration. If any of the values calculated are found unacceptable when compared to the predicted values, appropriate CPC and/or COLSS values are modified.

12.3.5 Reactivity Coefficients at 50% and 100% Full Power

Two methods are used to determine the Isothermal Temperature Coefficient and Power Coefficient; one method is performed with no CEA movement and the other method is performed with center CEA movement.

A. No CEA Movement at 50% and 100% Full Power

With the reactor at steady state and equilibrium or near equilibrium Xenon and CEA Group 6 at 120 inches withdrawn, a small step change in the turbine control valve position is made and then adjusted to establish a new coolant inlet temperature. This change produces a small turbine load-reactor power mismatch. The temperature change results in a reactivity feedback and a resultant power change. The power change produces an opposite reactivity feedback and the reactor settles out at a new power and temperature condition. The cycle is then reversed by making a small step change in the turbine control valve position in the opposite direction. The ITC is calculated iteratively using the resultant power and temperature changes along with an assumed power coefficient. The Moderator Temperature Coefficient (MTC) is then calculated by subtracting the predicted Fuel Temperature Coefficient (FTC) from the measured Isothermal Temperature Coefficient (ITC).

6. With Center CEA Movement at 50% Full Power

(1) Isothermal Temperature Coefficient - With the reactor at steady state and equilibrium Xenon and CEA Group 6 at 120 inches withdrawn, CEA 6-1 is withdrawn a specified amount. This reactivity change produces a change in reactor power which in turn causes a change in coolant temperature. The change in coolant temperature results in a reactivity feedback to offset the rod movement. Eventually the system stabilizes at a new power and coolant temperature. The ITC is calculated iteratively knowing the power and temperature changes along with the center CEA integral worth and by using the test predictions as initial estimates for the Isothermal Temperature and Power Coefficients. The MTC is calculated as described previously.

(2) Power Coefficient - A reactivity insertion is made using the center CEA, resulting in a change in reactor power. Average coolant temperature is held constant by changing turbine load to match reactor power. The reactor settles out at a new power when the reactivity feedback due to change in power is equal and opposite to the CEA reactivity insertion. The Power Coefficient is calculated iteratively in a manner similar to the ITC calculation.

Acceptance criteria state the following:

a. The measured ITC shall agree with the predicted values within $\pm 0.3 \times 10^{-4}$ $\Delta k/k/°F$;

b. The measured power coefficient should agree with the predicted values within $\pm 0.3 \times 10^{-4}$ $\Delta k/k/\%$ power; and

c. The MTC shall be less positive than $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ when reactor power is $\leq$ 70% of rated thermal power and less positive than 0.0 when reactor power is > 70% of rated thermal power.

12.4 Procedure If Acceptance Criteria Are Not Met

If the acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. The results of all tests will be reviewed by the plant's nuclear engineering group. If the acceptance criteria of the startup physics tests are not met, an evaluation will be performed by the plant's nuclear engineering group with assistance from general office personnel, Middle South Services and the fuel vendor, as needed. The results of this evaluation will be presented to the onsite Plant Safety Committee. Resolution will be required prior to power escalation. If a safety question is involved, the offsite Safety Review Committee would review the situation and NRC would be notified if an unreviewed safety question exists.

13.0 References

13.1 Section 1.0 References

(1-1) "Final Safety Analysis Report", Arkansas Nuclear One Unit 2, Arkansas Power & Light Company, Docket No. 50-368.

13.2 Section 2.0 References

(2-1) Arkansas Power & Light Company, Arkansas Nuclear One - Unit 2 Startup Test Report to NRC, NFP-6, Docket No. 50-368, July 31, 1979.

13.3 Section 3.0 References

(3-1) CENPD-256-P-A, "Test Fuel Rod Irradiation, 16 X 16 Nuclear Reactor", August 1977.

(3-2) CEN-50-P, Rev. 1-A, "Burnable Poison Irradiation Test", August 1977.

13.4 Section 4.0 References

(4-1) "Final Safety Analysis Report", Arkansas Nuclear One Unit 2, Arkansas Power & Light Company, Docket No. 50-368.

(4-2) CENPD-187, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding", June 1975.

(4-3) CEN-96-(A)-P, Rev. 1, "ANO-2 Reactor Operation with Modified CEA Guide Tubes and Lengthened Upper Guide Structure Flow Channels", July 12, 1978.

(4-4) CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report", July 1, 1974.

(4-5) NUREG-0418, "Fission Gas Release from Fuel at High Burnup", March 1978.

13.5 Secton 5.0 References

(5-1) CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report", July 1, 1974.

(5-2) CENPD-153-P, Rev. 1-P-A, "INCA/CECOR Power Peaking Uncertainty", May 1980.

(5-3) W. R. Caldwell, "PDQ-7 Reference Manual", WAPD-TM,-678, January 1967.

(5-4) T. G. Ober, et al., "Theory, Capabilities and Use of the Three-Dimensional Reactor Operation and Control Simulator (ROCS)", Nucl. Sci. and Eng., 64 (605), 1977.

(5-5) Letter, A. E. Lundvall, Jr. (BG&E) to R. W. Reid (NRC), February 23, 1979.

(5-6) Safety Evaluation Report, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Docket No. 50-317, June 14, 1979.

(5-7) Letter, Robert E. Uhrig, (FP&L) to Victor Stello (NRC), February 22, 1979, "St. Lucie Unit 1, Docket No. 50-335, Proposed Amendment to Facility Operating License DPR-67.

(5-8) Safety Evaluation Report, Florida Power & Light, St. Lucie Unit No. 1, Docket No. 50-335, May 27, 1979.

(5-9) Letter, Robert W. Reid, (NRC) to A. E. Scherer (C-E), December 17, 1979.

(5-10) A. Jonsson, et. al., "Discrete Integral Transport Theory Extended to the Case with Surface Sources", Atomkermenergie, Bd. 24, 1974.

(5-11) A. Jonsson, et al., "Verification of a Fuel Assembly Spectrum Code Based on Integral Transport Theory", Trans. Am. Nucl. Soc., 28 (778), 1978.

13.5 Section 5 References (Continued)

(5-12) Baltimore Gas & Electric Company, Docket No. 50-317 "Calvert Cliffs Unit No. 1, Cycle 2 Amendment to Facility Operating License", March 14, 1977.

(5-13) Omaha Public Power District, Docket No. 50-285 "Fort Calhoun Station Unit No. 1, Cycle 5 Amendment to Facility Operating License", August 3, 1978.

13.6 Section 6.0 References

(6-1) CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core", July 1975.

(6-2) CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution", April 1975.

(6-3) CENPD-206-P, "TORC Code, Verification and Simplified Modeling Methods," January 1977.

(6-4) CEN-143(A)-P "CPC/CEAC Software Modifications for Arkansas Nuclear One - Unit 2", December 1980, Appendix A.

(6-5) CEN-123(F)-P, "Statistical Combination of Uncertainties Part 2", February 1980.

(6-6) CEN-124(B)-P, "Statistical Combination of Uncertainties Part 2" March 1980.

(6-7) CEN-139(A)-P, "Statistical Combination of Uncertainties; Combination of System Parameter Uncertainties in Thermal Margin Analyses for Arkansas Nuclear One Unit 2", November 1980.

(6-8) Supplement 3-P (Proprietary) to CENPD 225P, "Fuel and Poison Rod Bowing", June 1979.

(6-9) Letter, D. B. Vassalo (NRC) to A. E. Scherer (C-E), dated June 12, 1978.

13.7 Section 7.0 References To be Supplied later with Section 7.0

13.8 Section 8.0 References

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(8-2) CENPD-132, "Calculative Methods for the C-E Large Break LOCA Evaluation Model," August 1974 (Proprietary).
CENPD-132, Supplement 1, "Updated Calculative Methods for the C-E Large Break LOCA Evaluation Model,"December 1974 (Proprietary).
CENPD-132, Supplement 2, "Calculational Methods for the C-E Large Break LOCA Evaluation Model", July 1975 (Proprietary).

(8-3) CENPD-135, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1974 (Proprietary)
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CENPD-135, Supplement 4, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976 (Proprietary)

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(8-4) CENPD-139, "C-E Fuel Evaluation Model," July 1974 (Proprietary)

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(8-6) CENPD-138, "PARCH - A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974 (Proprietary).
CENPD-138, Supplement 1, "PARCH- A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup" (Modifications), February 1975 (Proprietary)
CENPD-138, Supplement 2(P), "PARCH - A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup" January 1977 (Proprietary).

13.9 Section 9.0 References

(9-1) CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure," Revision 2, December 21, 1978.

(9-2) CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure Supplement," Supplement 1-P, Revision 1, January 5, 1979.

(9-3) CEN-143(A)-P, "CPC/CEAC Software Modifications for Arkansas Nuclear One - Unit 2," December 1980.

(9-4) CENPD-170-P, "Assessment of the Accuracy of PWR Safety System Actuation as Performed by the Core Protection Calculators (CPC)," July 1975, and Supplement 1-P, November 1975.

13.10 Section 10.0 References

(10-1) CENPD-384, "The Evaluation and Demonstration of Methods for Improved Nuclear Fuel Utilization," First Semi-Annual Progress Report: January-June 1980, for the Department of Energy, October 1980, to be published.