OMAHA PUBLIC POWER DISTRICT

NUCLEAR ANALYSIS

RELOAD CORE ANALYSIS METHODOLOGY OVERVIEW

OPPD-NA-8301-NP

September 1983

ABSTRACT

This document is a Topical Report describing Omaha Public Power District's reload core analysis methodology for application to the Fort Calhoun Station Unit No. 1.

The report provides an overview of the District's reload core methodology. Analyses performed by the District and its contractors are described. Details of the thermal hydraulic methodology which were previously submitted to the NRC are provided.

PROPRIETARY DATA CLAUSE

This document is the property of Omaha Public Power District (OPPD) and contains proprietary information, indicated by brackets, developed by Combustion Engineering (CE) anbd Exxon Nuclear Company, Inc. (ENC). The CE and ENC information was purchased by OPPD under proprietary information agreements.

TABLE OF CONTENTS

	SECTION	PAGE
1.0	INTRODUCTION	1
2.0	FUEL SYSTEM DESIGN	1
3.0	NUCLEAR DESIGN	1
4.0	THERMAL HYDRAULIC DESIGN	3
5.0	POSTULATED ACCIDENTS AND TRANSIENTS	11
6.0	SETPOINT GENERATION	11
7.0	REFERENCES	12

OMAHA PUBLIC POWER DISTRICT
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1.0 INTRODUCTION

Analyses done to license reload cores for Fort Calhoun Station consists of the analysis performed by the Omaha Public Power District and the analysis performed by the nuclear fuel vendor. The current nuclear fuel vendor is Exxon Nuclear Company, Inc. (ENC); however, future reload fuel may be supplied by any of the four PWR nuclear fuel vendors: Exxon Nuclear Company, Inc. (ENC), Combustion Engineering (CE), Westinghouse, or Babcock and Wilcox. The following sections discuss the reload analyses and consolidate information about the District's methodology previously submitted.

2.0 FUEL SYSTEM DESIGN

The fuel assembly mechanical design and analysis are performed by the nuclear fuel vendor. The fuel mechanical design and design methods currently utilized for Fort Calhoun Station by Exxon Nuclear Company, Inc. are described in Reference 1.

3.0 NUCLEAR DESIGN

The District's nuclear design methodology is discussed in Reference 2.

3.1 Fuel Management

The reload core fuel management is performed by the District. Current fuel management schemes are selected to reduce flux to the reactor pressure vessel welds.

3.0 NUCLEAR DESIGN (Continued)

3.2 Power Distribution Measurement

The District utilizes the CE methodology (Reference 3) to measure the power distributions. This methodology is discussed in the Cycles 5 and 6 reload submittals and approved in the SER's for these fuel cycles (References 4 and 5).

3.3 Uncertainties and Allowances

The power distribution uncertainties which are included in the overall analysis of reload cores are:

Parameter	Uncertainty
3D Peak, F_q3-D	6.2%
Integrated Radial Peak, F_R	6.0%
Planar Radial Peak, F_{xy}	5.3%

These values are approved for use in CENPD-153-P (Reference 6). A more detailed discussion of the treatment of uncertainties and allowances can be found in Reference 7.

3.4 Physics Safety Related Data

The physics safety related data are produced using the methodology discussed in Reference 2.

4.0 THERMAL HYDRAULIC DESIGN

4.1 Steady State DNBR Analysis

The steady state DNBR analysis is performed by the District using the TORC/CETOP/CE-1 methodology (References 8, 9, 10, and 11). This methodology was approved for use by the District in Reference 12.

4.1.1 Grid Spacer Loss Coefficients

The analysis utilizes a D-TORC model with explicit representation of the loss coefficients associated with ENC and CE fuel assemblies. The nominal grid loss coefficients used in thermal hydraulic analysis are:

	ENC Spacer	CE Spacer
Loss Coefficient (K)	[	]

RE = Reynolds Number

These values were obtained by ENC using single phase pressure drop testing of an ENC test assembly and a typical CE 14 x 14 assembly. Single phase hydraulic loss coefficients, previously transmitted in Reference 13, contained a [ ] while the values given above are best estimate values. Because of the sensitivity of DNBR calculations to the difference in spacer grid loss coefficients, the District utilizes the Reynold's number expression for loss coefficients. This provides the most

4.1.1 Grid Spacer Loss Coefficients (Continued)

accurate representation of the pressure drop across each spacer grid in the assembly. Thus, the cross flows between adjacent assemblies in the region of the spacer grid are accurately modeled.

The spacer grid geometries for the CE and ENC spacer grids are shown in the attached Figures 1 and 2. The spacer grid envelope for both the CE and ENC grids is 8.115 inches by 8.115 inches. The axial location of the CE and ENC spacer grids is shown in Figure 3.

In D-TORC calculations, the spacer grid loss coefficient for a channel corresponds to the assembly type whenever a channel represents a single assembly or a portion of an assembly. The choice of loss coefficient for lumped channels in D-TORC is made such that the minimum flow is provided to the limiting fuel assembly. The CETOP model employs the spacer grid loss coefficient for limiting the assembly calculated in D-TORC. The inlet flow fraction of the CETOP model is tuned such that the CETOP model produces conservative results with respect to the D-TORC model, which models all fuel assemblies.

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.2 CE-1 Correlation

The District utilizes the CE-1 correlation for DNBR calculations. The range of data in the data base for the CE-1 correlation is contained in References 8 and 9. The range of parameters for the CE-1 correlation and corresponding ranges for the CE and ENC assemblies are shown in Table 1. Because the data for the ENC fuel assembly is within those specified in CE-1 data base, the use of CE-1 correlation is appropriate for the ENC fuel.

4.1.3 D-TORC and CETOP Models

The District utilizes the D-TORC code (Reference 10) and the CETOP code (Reference 11) to perform thermal hydraulic analysis for the Fort Calhoun reload core. The fraction of inlet flow to the hot assembly in the CETOP model is adjusted such that the model yields appropriate MDNBR results when compared with results of D-TORC analysis for a given range of operating conditions. The fraction of inlet flow is determined for each reload core. The use of this methodology was approved for use by the District in Reference 12.

TABLE 1

PARAMETER RANGES OF THE SOURCE DATA FOR THE CE-1 CHF CORRELATION AND THE RANGE OF EXXON AND CE 14 x 14 FOR CALHOUN VALVES

PARAMETER	CORRELATION RANGE	CE RANGE	EXXON RANGE
Pressure (psia)	1785 to 2415	N/A	N/A
Local Coolant Quality	-.16 to .20	N/A	N/A
Local Mass Velocity ($lbm/hr\text{-}ft^2$)	0.87×10^6 to 3.21×10^6	N/A	N/A
Subchannel Wetted Equiv. Diameter (in)	.3588 to .5447	.4043 to .5449	.4010 to .5402
Subchannel Heated Equiv. Diameter (in)	.4713 to .7837	.5334 to .7040	.5270 to .7760
Heated Length (in)	84,150	128	128
Grid Spacing	14.2" to 18.25"	16.8	16.8

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.3 D-TORC and CETOP Models (Continued)

The following paragraphs discuss the application of the CETOP code to the Fort Calhoun reactor. Examples are for the Cycle 8 core.

Thermal margin analysis utilizing the CETOP model is supported by comparing its predictions for Fort Calhoun Station with those obtained from a detailed TORC analysis. Several operating conditions were arbitrarily selected for this demonstration; they are representative, but not the complete set, of conditions which would be considered for a normal DNB analysis.

A thermal margin model for 1500 MWT for Fort Calhoun Unit No. 1 was developed for the following operating ranges:

- Inlet Temperature	450 to 600°F
- System Pressure	1750 to 2400 psia
- Primary System 4-Pump Flow Rate, (LCO = 197,000 gpm)	80% to 120%
- Axial Power Distribution	-0.517 to +0.526 ASI

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.3 D-TORC and CETOP Models (Continued)

The detailed thermal margin analyses were performed for the sample core using the radial power distribution and detailed TORC model shown in Figures 4, 5, and 6. The appropriate spacer grid loss coefficient was applied to each "assembly" channel or partial assembly channel in each stage. In stage 1, lumped channel 28 utilized the CE spacer grid loss coefficient because the channel was predominantly composed of CE fuel. Lumped channels 26 and 27 utilized the ENC spacer grid loss coefficient because either the channel was composed of entirely ENC fuel or contained a single CE assembly not on a boundary between channels. The axial power distributions are given in Figure 7. These distributions were the most limiting ones generated for the length of the cycle and for the various power dependent insertion limits examined. The core inlet flow and exit pressure distributions used in the analyses were based on flow model test results given in Figures 8 and 9. The results of the detailed TORC analyses are given in Table 2.

The CETOP design model has a total of four thermal hydraulic channels to model the open-core fluid phenomena. Figure 10 shows the layout of these

TABLE 2

COMPARISONS BETWEEN TORC AND CETOP-D

Operating Parameters					MDNBR		Quality at MDNBR		Axial Elev. of MDNBR(in)	
Pressure (psia)	Inlet Temperature (°F)	Avg Mass Velocity (10^6 lbm/hr-ft)	Core Avg. Heat Flux (BTU/hr-ft^2)	Shape Index (ASI)	Detailed TORC Relative Flow in Location 5	CETOP-D Inlet Flow Factor [.84 .76]	Detailed TORC Relative Flow in Location 5	CETOP-D Inlet Flow Factor [.76]	Detailed TORC	CETOP-
1750	450	1.7432	242409	-.517						
2100	450	1.7432	257008	-.517						
2250	450	1.7432	261195	-.517						
2400	450	1.7432	264118	-.517						
2100	545	2.1790	216494	-.517						
1750	600	1.7432	149283	-.206						
2100	600	1.7432	168727	-.206						
2250	600	1.7432	176398	-.206						
2400	600	1.7432	184260	-.206						
2100	545	2.1790	257118	-.206						
2100	545	2.1790	282778	.004						
2100	545	2.1790	298644	.203						
1750	450	1.7432	295262	.527						
1750	545	1.7432	227063	.527						
1750	600	1.7432	157319	.527						
2100	545	2.1790	255014	.527						

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.3 D-TORC and CETOP-D Models (Continued)

channels. Channel 2 is a quadrant of the hottest assembly which represents the average coolant conditions for the remaining portion of the core. The boundary between channels 1 and 2 is open for crossflow; the remaining outer boundaries of channel 2 are assumed to be impermeable and adiabatic. Channel 2 includes channels 3 and 4. Channel 3 lumps the subchannels adjacent to the MDNBR hot channel 4. The "hot" assembly determined from D-TORC analysis was an ENC assembly. Since CETOP models a quandrant of the "hot" assembly, the ENC spacer grid loss coefficient was used in the analysis.

The CETOP model described above was applied to the same cases as the detailed TORC analyses. The results from the CETOP model analyses are compared with those from the detailed analyses in Table 2. It was found that a constant inlet flow split providing hot assembly inlet mass velocity of [ ] of the core average value is appropriate for 4-pump operation so that MDNBR results predicted by the CETOP model are either conservative or accurate for the Cycle 8 core.

4.0 THERMAL HYDRAULIC DESIGN (Continued)

4.1 Steady State DNBR Analysis (Continued)

4.1.3 D-TORC and CETOP-D Models (Continued)

The uncertainties associated with the thermal hydraulic analysis are combined statistically (Reference 14). In this method, the impact of component uncertainties on DNBR is assessed and the SAFDL is increased to include the effects of the uncertainties.

5.0 POSTULATED ACCIDENTS AND TRANSIENTS

The postulated accidents and transients are analyzed using the methodology discussed in Reference 15.

6.0 SETPOINT GENERATION

The District utilizes the methodology discussed in CENPD-199-P (Reference 16) to generate setpoints for Fort Calhoun Station. The District's reactor physics methodology is discussed in Reference 2.

The scram reactivity curves are produced using the QUIX code. The power-to-fuel design limit on centerline melt is derived using the QUIX code with the appropriate combinations of planar radial peaking factor, F_{xy}^T, and axial power distribution.

The thermal margin analysis is done using the CETOP code with the appropriate combinations of the integrated radial peaking factor, F_R^T, axial power distribution, RCS inlet temperature, and RCS pressure.

6.0 SETPOINT GENERATION (Continued)

The Fort Calhoun RPS utilizes the "early system" local power density trip and TM/LP trip. The TM/LP trip does not monitor the axial shape index. The sequential CEA withdrawal is analyzed using the methods described in Reference 16 and not included in the TM/LP trip considerations. The RCS depressurization and excess load events provide the transient analysis input into the TM/LP trip.

The uncertainties are treated statistically in the District's setpoint analysis.

7.0 REFERENCES

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

2. "Reload Core Analysis Methodology, Neutronics Design Methods and Verification," OPPD-NA-8302, September 1983.

3. "INCA, Method of Analyzing Incore Detector Data in Pressurized Water Reactors," CENPD-145-P, April 1, 1975.

4. Letter, R. W. Reid (NRC) to T. E. Short (OPPD), December 5, 1978.

5. Letter, R. W. Reid (NRC) to W. C. Jones (OPPD), April 1, 1980.

6. "INCA/CECOR Power Peak Uncertainty," CENPD-153-P, Revision 1-P-A, May 1980.

7.0 REFERENCES (Continued)

7. "Statistical Combination of Uncertainties," CEN-124(0)-P, Parts 1 and 3, 1983.

8. "CE Critical Heat Flux," CENPD-162-P-A, Part 1, Combustion Engineering, September 1976.

9. "CE Critical Heat Flux," Part 2, CENPD-207-P-A, Combustion Engineering, June 1976.

10. "A Computer Code for Determining the Thermal Margin of a Reactor Core," CENPD-161-P, TORC Code, Combustion Engineering, July 1975.

11. "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 & 2," CEN-191(B)-P, Combustion Engineering, December 1981.

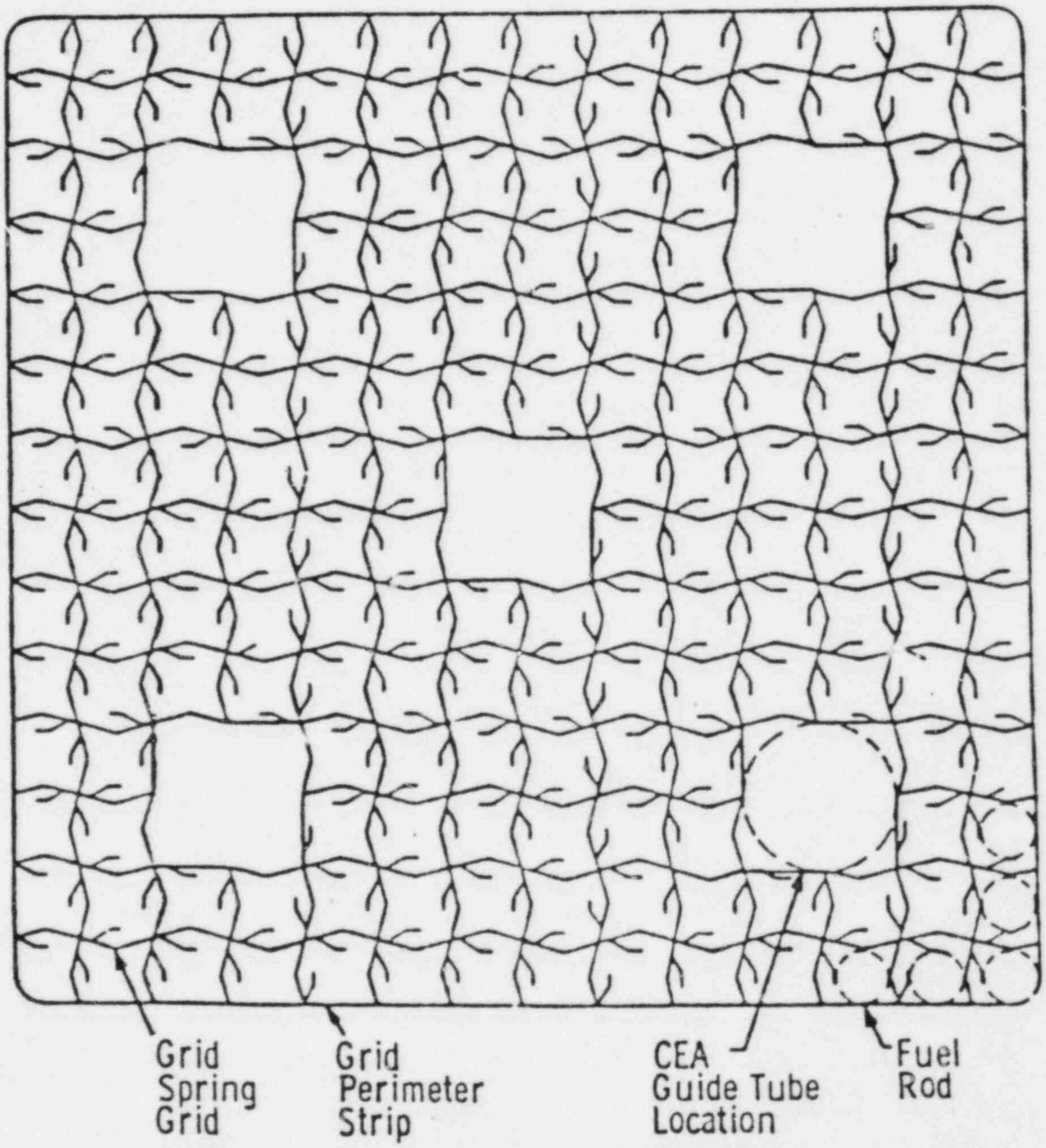
12. Letter from R. A. Clark (NRC) to W. C. Jones (OPPD), March 15, 1983.

13. Letter from W. C. Jones (OPPD) to R. W. Reid (NRC), December 4, 1979.

14. "Statistical Combination of Uncertainties," CEN-124(0)-P, Part 2, 1983.

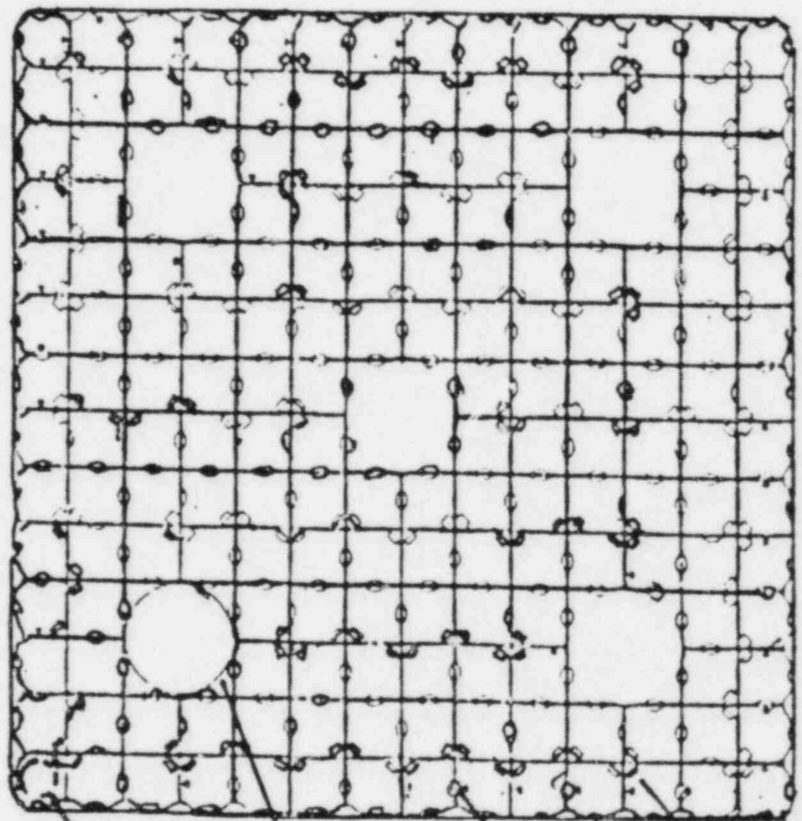
15. "Reload Core Analysis Methodology, Transient and Accident Analysis Methods and Verification," OPPD-NA-8303, September 1983.

16. "CE Setpoint Methodology," CENPD-199-P, Revision 1-P, April 1982.



CE SPACER GRID GEOMETRY

FUEL SPACER GRID	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 1



ENC SPACER GRID GEOMETRY

FUEL SPACER GRID	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 2

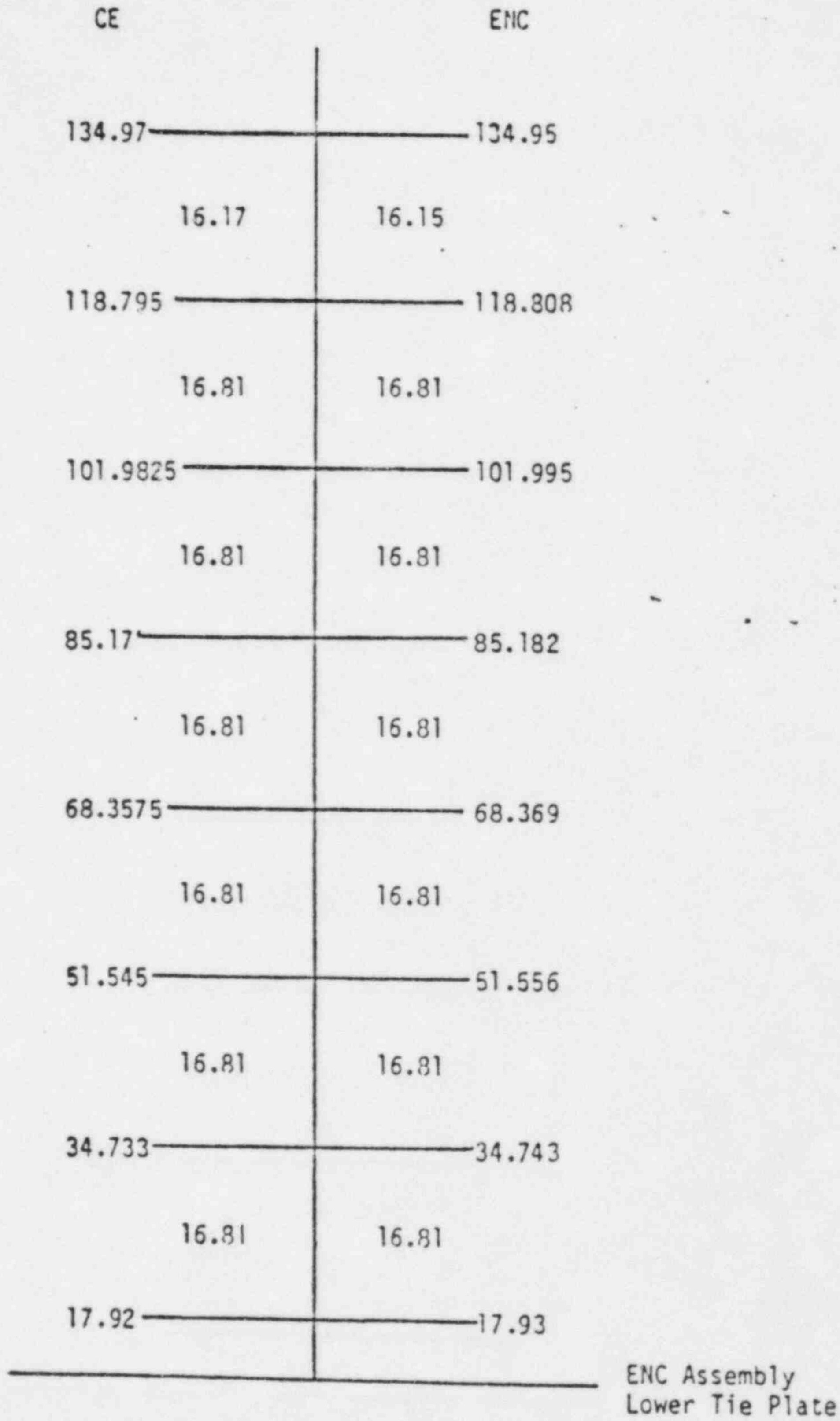


FIGURE 3
AXIAL LOCATION OF FUEL ASSEMBLY SPACER GRIDS

FIGURE 4
STAGE 1 TORC CHANNEL GEOMETRY FOR FORT CALHOUN UNIT NO. 1

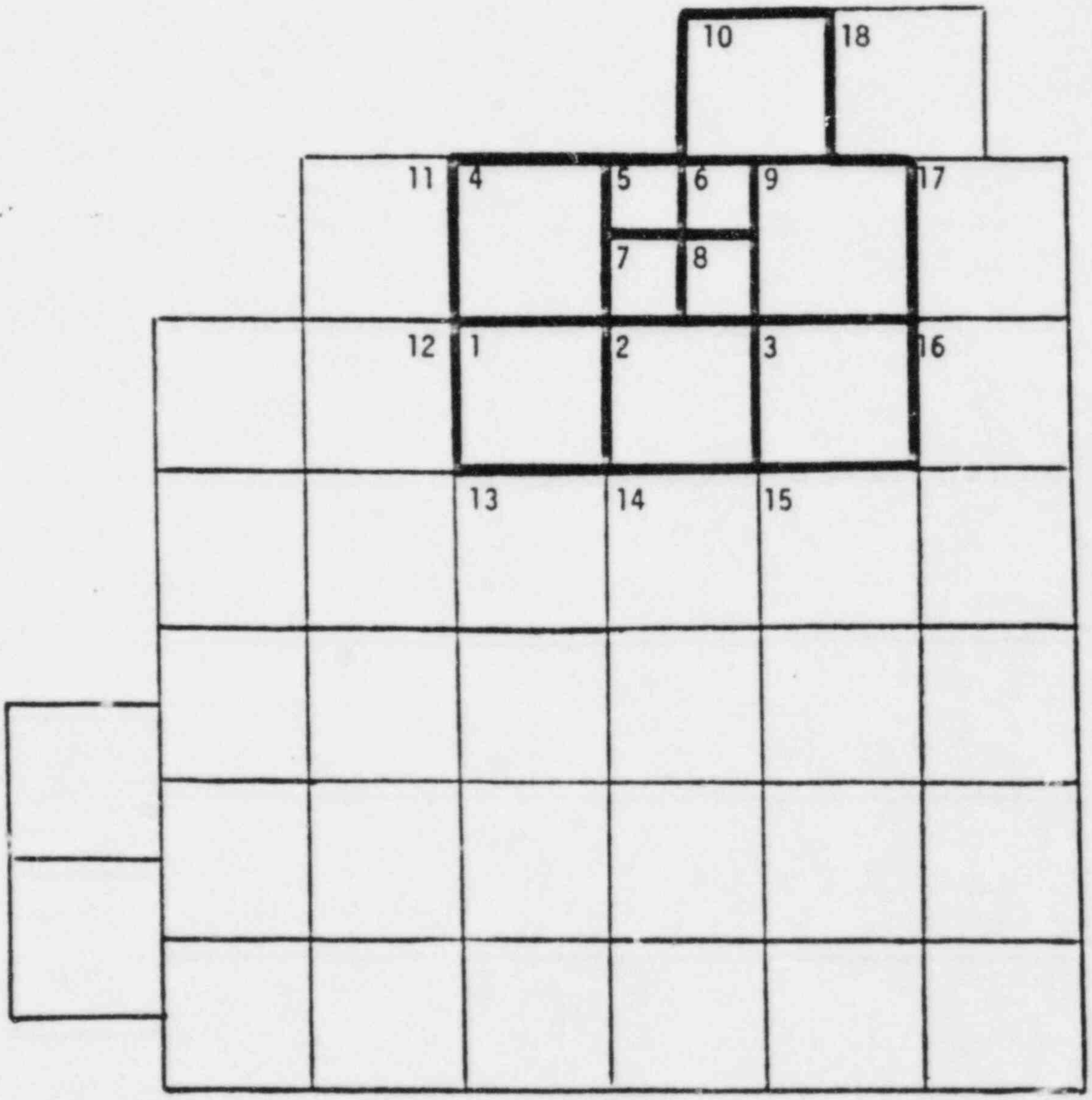


Figure 5

STAGE 2 TORC CHANNEL GEOMETRY FOR FORT CALHOUN UNIT NO. 1

FIGURE 6
STAGE 3 TORC CHANNEL GEOMETRY FOR FORT CALHOUN UNIT NO. 1

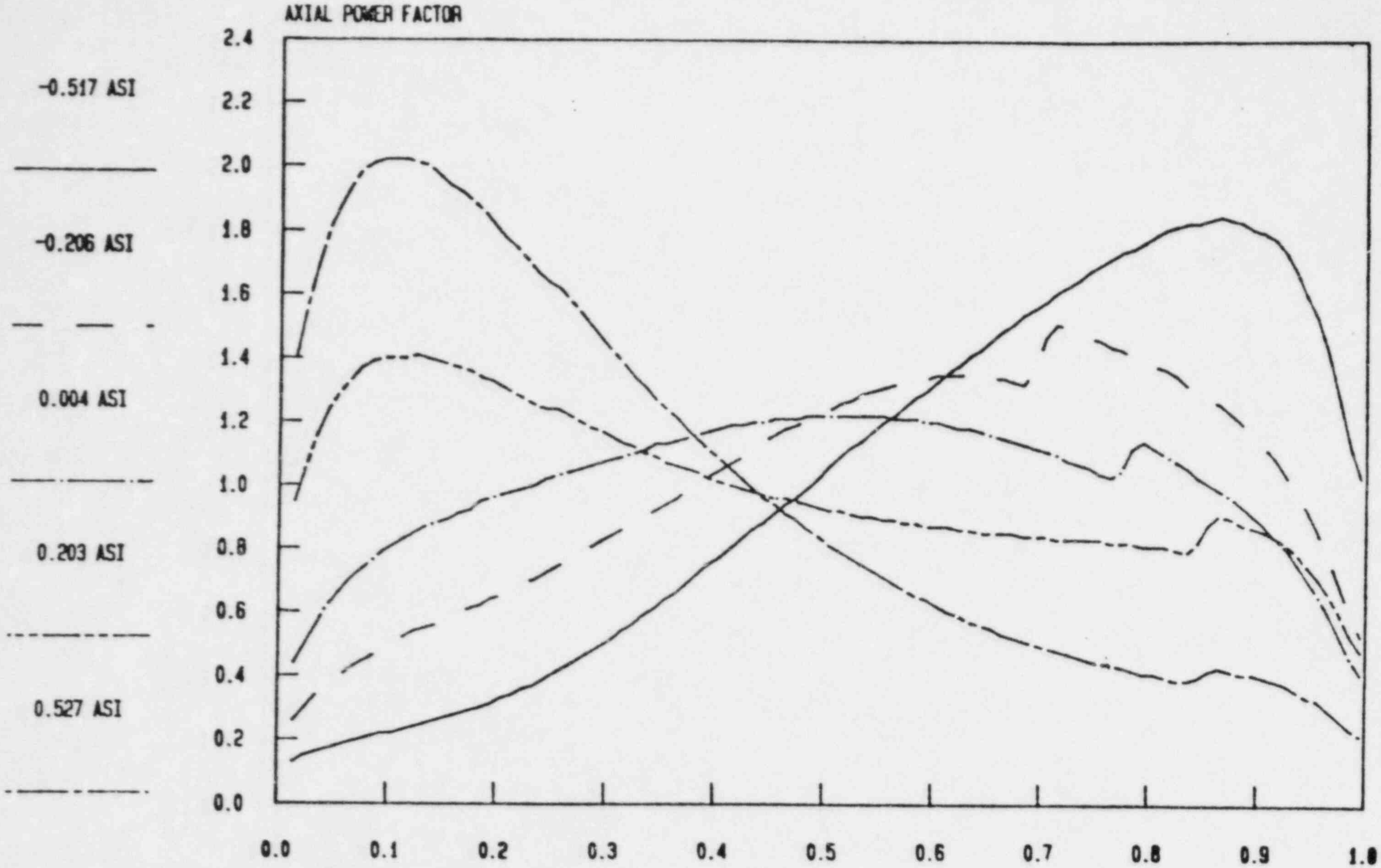


FIGURE 7
AXIAL POWER DISTRIBUTIONS
FORT CALHOUN CYCLE 8

FIGURE 8
INLET FLOW DISTRIBUTION FOR OMAHA 4-PUMP OPERATION

FIGURE 9
EXIT PRESSURE DISTRIBUTION FOR OMAHA, 4-PUMP OPERATION

FIGURE 10
CETOP-D CHANNEL GEOMETRY FOR FORT CALHOUN UNIT NO. 1
(CHANNEL 1 NOT SHOWN)