Omaha Public Power District

1623 HARNEY ▪ OMAHA, NEBRASKA 68102 ▪ TELEPHONE 536-4000 AREA CODE 402

January 26, 1983
LIC-83-018

Mr. Robert A. Clark, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Clark:

Resolution of the Pressurized
Thermal Shock Issue

Attached please find Omaha Public Power District's plan for resolving the pressurized thermal shock (PTS) issue for the Fort Calhoun Station's reactor vessel.

As we have discussed with your staff, the District is fully aware of both the safety and commercial features of the PTS issue. The District has implemented actions which encompass a broad program to resolve this issue. We believe this program contains all actions which are reasonable and prudent in order to resolve PTS on a schedule consistent with the safety concerns for the Fort Calhoun Station's pressure vessel.

For the past two years, the District has actively participated with the NRC staff, industry groups, and specifically Combustion Engineering (our nuclear steam system supplier for the Fort Calhoun Station) on PTS activities. During the District's participation in these activities, specific corrective measures were identified which would assist in resolving the PTS issue. As these measures were identified and evaluated, the District implemented actions in several important areas. One of these included the design and planning for the installation of a "low-leakage" fuel pattern for the next operating cycle beginning in late March or early April, 1983. Details of this action are provided in the attachment. Arrangements have also been completed to perform an enhanced inservice inspection of all welds in the belt line region of the reactor vessel. This inspection will utilize state-of-the-art procedures and equipment and will be performed during the present refueling outage. This inspection will be performed in accordance with Regulatory Guide 1.150.

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The analysis of a reactor vessel surveillance capsule scheduled for removal during the present refueling outage will provide valuable information regarding the condition of the Fort Calhoun Station's pressure vessel. This will be the second vessel surveillance capsule removed since plant startup in 1973. Two new capsules will be installed during the present refueling outage to assist in evaluating the effectiveness of the fluence redu ion program being implemented for future operating cycles. These surveillance capsule activities are in addition to technical specification requirements.

Another significant improvement was the completion of additional training and revisions to station procedures relating to PTS. The District's technical support staff and personnel from Combustion Engineering conducted training for the operating staff to provide additional assurance that operating personnel were trained to prevent and/or mitigate potential PTS events. This training and associated improvements in the operating procedures have been evaluated by the NRC. The recommendations resulting from this evaluation were promptly implemented by the District.

The District has also completed or implemented actions on other tasks which have made or will make substantial improvements in the operating staff's ability to prevent and/or mitigate potential PTS events. These include improvements to the auxiliary feedwater system controls and installation of a subcooling margin monitor. The planned installation of the Safety Parameter Display System (SPDS) will provide the operating staff with additional information to evaluate and mitigate potential PTS events. A portion of this system will be installed during the present refueling outage and the remainder is scheduled for installation in 1984.

Other measures which have been implemented include additions to the operating crews on each shift. These additions provide an increased level of technical expertise and thereby facilitate the diagnostic capabilities of the operating crews.

In conclusion, the District has already implemented many measures which have made substantial contributions toward resolving safety concerns for potential PTS events at the Fort Calhoun Station. The planned fluence reduction measure assures that the present RT_{NDT} screening criteria for plates and axial welds (270°F) will not be exceeded for approximately 14 years. The attachment describes the District's planned efforts to resolve the PTS issue and thereby assure plant operation to the full design lifetime of the vessel.

Sincerely,

W C Jones

W. C. Jones
Division Manager
Production Operations

WCJ/KJM:jmm

Attachment

cc: LeBoeuf, Lamb, Leiby & MacRae

PLANNED EFFORTS TO RESOLVE PTS CONCERNS
FOR
FORT CALHOUN STATION

The District plans to implement measures during the next year which should provide the basis for the resolution of PTS concerns at the Fort Calhoun Station. These measures include a reduction in the neutron flux to reactor vessel welds, further determination of reactor vessel material properties, determination of a flaw distribution in the belt line welds, further improvements in emergency operating procedures, and a probabilistic analysis of PTS. A brief discussion of each of these items is addressed below.

FLUX REDUCTION

Highest priority is being placed on reducing the neutron flux to reactor vessel welds which currently have a significant RT_{NDT} shift. Figure 1 identifies the current (EOC-7) RT_{NDT} values using the Combustion Engineering (CE) azimuthal flux distribution. The figure shows that the zero degree middle course axial weld has a current RT_{NDT} value which is nearest to the Commission's screening criteria. The RT_{NDT} rate of increase for this weld is approximately 7.5°F per Effective Full Power Year (EFPY). Based on the Fort Calhoun Station's historical annual capacity factor of 70%, the Commission's screening criteria would be exceeded in December, 1987 using the CE azimuthal flux distribution or in December, 1989 using the Brookhaven distribution.

Figure 2 identifies the location of the axial welds relative to the core. The marked core locations are those locations for which the assembly power must be reduced in order to reduce the neutron flux at the reactor vessel welds. Figure 3 shows the Cycle 8 IN-IN-OUT (i.e., assemblies are loaded in the interior of the core for two cycles and then are peripherally loaded for the third cycle) core loading scheme pattern which provides for inserting irradiated fuel assemblies in core locations which previously were the largest source of neutrons to the reactor vessel welds and, thus, have the largest RT_{NDT} values. This loading pattern reduces the power in these locations compared to the power in previous cycles. Figure 4 shows the Cycle 7 core loading for these core locations and is typical of the OUT-IN-IN loading scheme used in previous cores. Figures 5 and 6 demonstrate the reduction in the peripheral assembly powers for BOC-8 and EOC-8. It should be noted that the power in the peripheral pins (the dominant source of neutrons to the reactor vessel wall) is decreased more than the average assembly power.

The OUT-IN-IN core loading scheme used in Cycles 1 through 7 was the standard PWR refueling scheme utilized in the 1970's. The IN-IN-OUT core loading scheme has been implemented at several PWR's to reduce fuel cycle costs. The Cycle 8 core, currently scheduled for a late March, 1983 startup, utilizes an IN-IN-OUT loading which has been optimized for flux reduction rather than fuel cycle economics. To accommodate the increased one pin peaks expected in a core optimized for flux reduction, the District performed the Cycle 8 reload safety analysis using the CE reload methodology. This methodology was previously utilized and approved for the Cycle 5 core with the exception of the use of the CE-1 correlation.

The District has several ongoing studies aimed at achieving the maximum flux reduction while maintaining full power capability for future cycles. The goal of these studies is to determine if an economically feasible core loading scheme, which will allow the reactor vessel to reach the end of its design life, can be derived without ever exceeding the screening criteria.

The first of these studies is the derivation of a Cycle 9 core loading pattern and the associated safety analysis. The District hopes to achieve at least a factor of three flux reduction while maintaining the core parameters within the proposed Cycle 8 Technical Specification limits. Figure 7 shows a preliminary core loading for the peripheral core locations, and Figure 8 shows the corresponding assembly power reductions. Preliminary analysis results have lead the District to believe that a factor of three flux reduction can be achieved. A study will also be undertaken to attempt to further reduce the peripheral assembly powers by utilizing more advanced CE safety analysis methodologies such as the Statistical Combination of Uncertainties. If further flux reduction is determined to be feasible, the results of the study will be discussed with the staff. At that time the District will also discuss the implementation of this improved methodology to determine the feasibility of including it in the Cycle 9 reload.

The next step will be to evaluate future cycles beyond Cycle 9 to determine the maximum flux reduction achievable in an "equilibrium cycle". Parameters such as batch size, cycle length, burnable shim configurations, and further analytical improvements in the reload safety analysis will be considered. This study will also include an economic assessment of the flux reduction scheme(s).

AZIMUTHAL FLUX DISTRIBUTION PREDICTION BENCHMARKING

Concurrent with the flux reduction studies identified above, the District will commence a program to improve the ability to predict the flux distribution at the reactor vessel wall. This involves the removal of a surveillance capsule during the current refueling outage, measurement of the capsule fluence, and completing a benchmark calculation using the DOT code. Figure 9 shows the location of the surveillance capsules. The 225^{o} capsule was removed at 2.59 EFPY, and the 265^{o} capsule will be removed during the current outage after receiving a fluence corresponding to 5.92 EFPY. Comparison of these measured fluences with the fluences calculated by the DOT code should improve the accuracy of the calculated reactor vessel flux distribution.

REACTOR VESSEL INSPECTIONS

A third program with the same priority as the flux reduction program is to more accurately determine the reactor vessel material properties. The District is currently performing a comprehensive ultrasonic inspection of the reactor vessel. This inspection includes both an examination in accordance with ASME code criteria and a near-surface examination of all reactor vessel welds in the belt line region. The near-surface examinations will utilize the 70^{o} compression wave technique. These examinations will detect both near-surface and deep flaws. The weld inspections will also include all material 1/2 T

either side of the centerline of the belt line welds. Figure 10 shows the welds which will be inspected. The District is confident that all flaws down to one-half inch in depth can be detected utilizing these techniques.

SURVEILLANCE CAPSULE MATERIAL

The District will also perform metallurgical tests on the surveillance capsule specimens removed during the present outage. The District will insert two new surveillance capsules in the reactor vessel during the outage. These capsules contain weld material specimens which will allow improved measurement of material properties and contain improved fluence monitors.

EMERGENCY PROCEDURES

The District will implement the symptom-oriented Emergency Operating Procedures in accordance with our schedule to be provided in response to Generic Letter 82-33. The District will utilize the CE Emergency Procedure Guidelines, which are based on work performed by CE and the CE Owners Group member utilities and include PTS guidance. These guidelines have undergone several reviews by the NRC and are currently under final review. The District will use the CE Emergency Procedure Guidelines to prepare Fort Calhoun Emergency Procedure Guidelines and, during this process, PTS concerns will be given a high priority.

PROBABILISTIC RISK ASSESSMENT (PRA)

Finally, the District expects to perform preliminary PRA studies for PTS scenarios at Fort Calhoun. The aim of these studies is to determine if the PTS risk at Fort Calhoun is significantly less than that identified in the generic studies. If the risk is found to be low, the District may undertake a comprehensive PRA study for PTS at Fort Calhoun to determine an operating limit on RT_{NDT} for Fort Calhoun. In order to perform such a study, the District will require the use of the risk acceptance criteria for PTS which we understand will be developed during the PTS rulemaking process. The comprehensive PRA will evaluate the plant as it presently exists and the impact of proposed modifications, such as SIRWT heating, to determine the current risk of PTS and the cost/benefit ratio for proposed modifications.

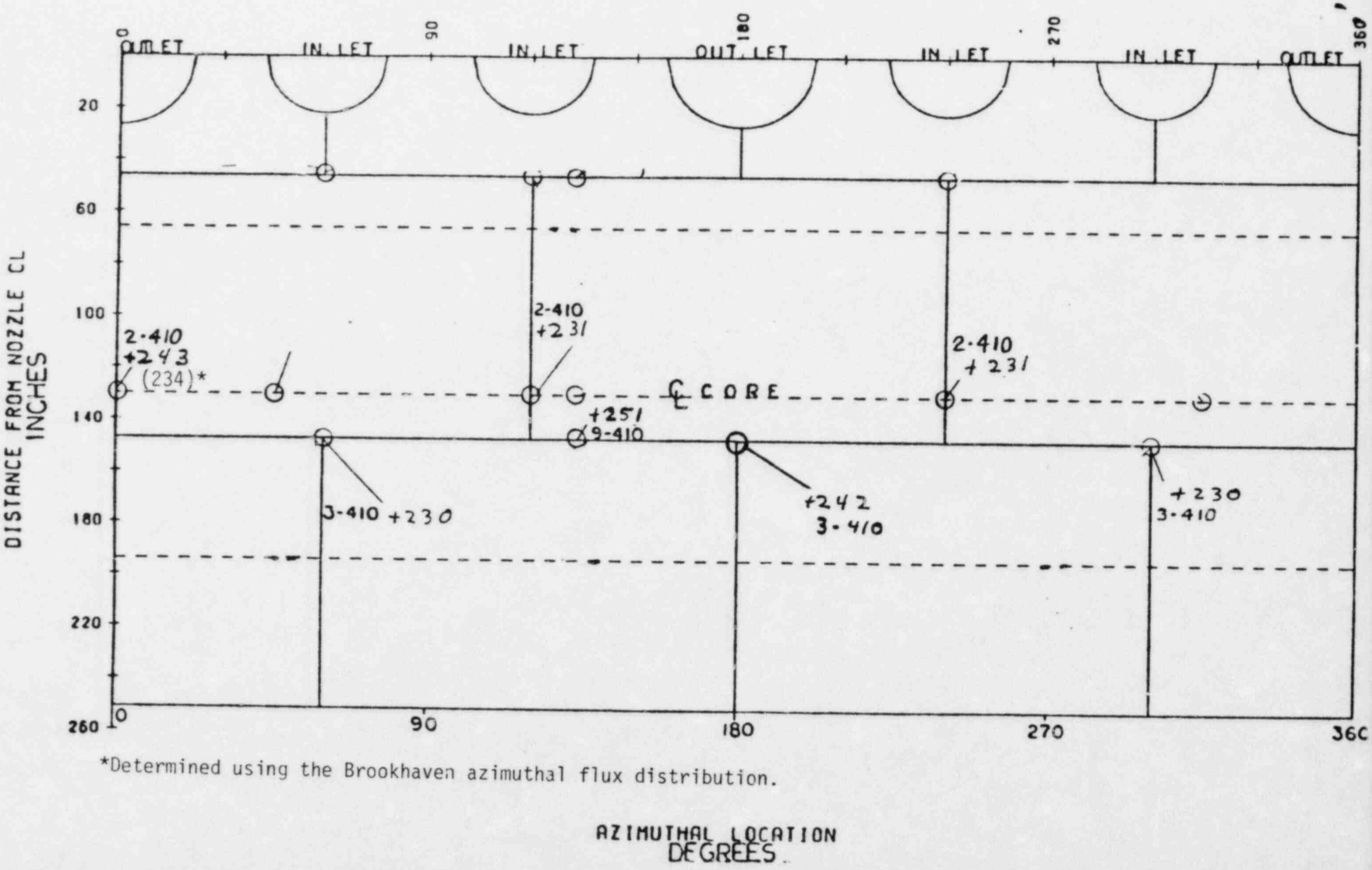
SUMMARY

The District has or will undertake a number of programs to resolve PTS concerns at Fort Calhoun. We have identified those areas which show the most promise and are actively pursuing programs in these areas. We have also prioritized those programs to assure that resolutions which require near-term action can be implemented promptly.

FIGURE 1

FORT CALHOUN REACTOR PRESSURE VESSEL MAP
ADJUSTED RT_{NDT} IN °F (12/31/82)

5.92 Effective Full Power Years
$0.823 \times 10^{19} n/cm^2$ Peak Surface Fluence



*Determined using the Brookhaven azimuthal flux distribution.

FIGURE 2 - VESSEL WELD LOCATION

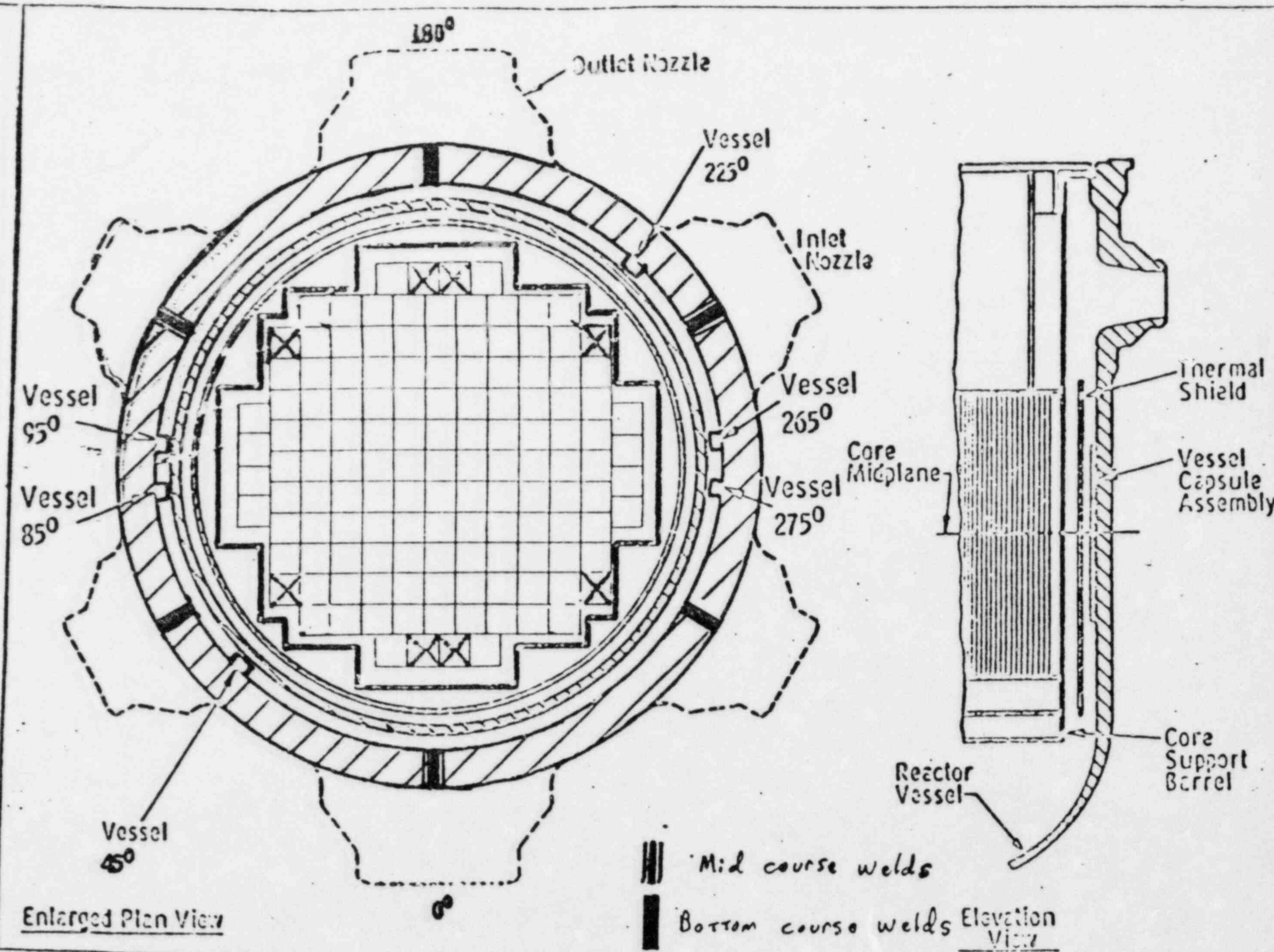
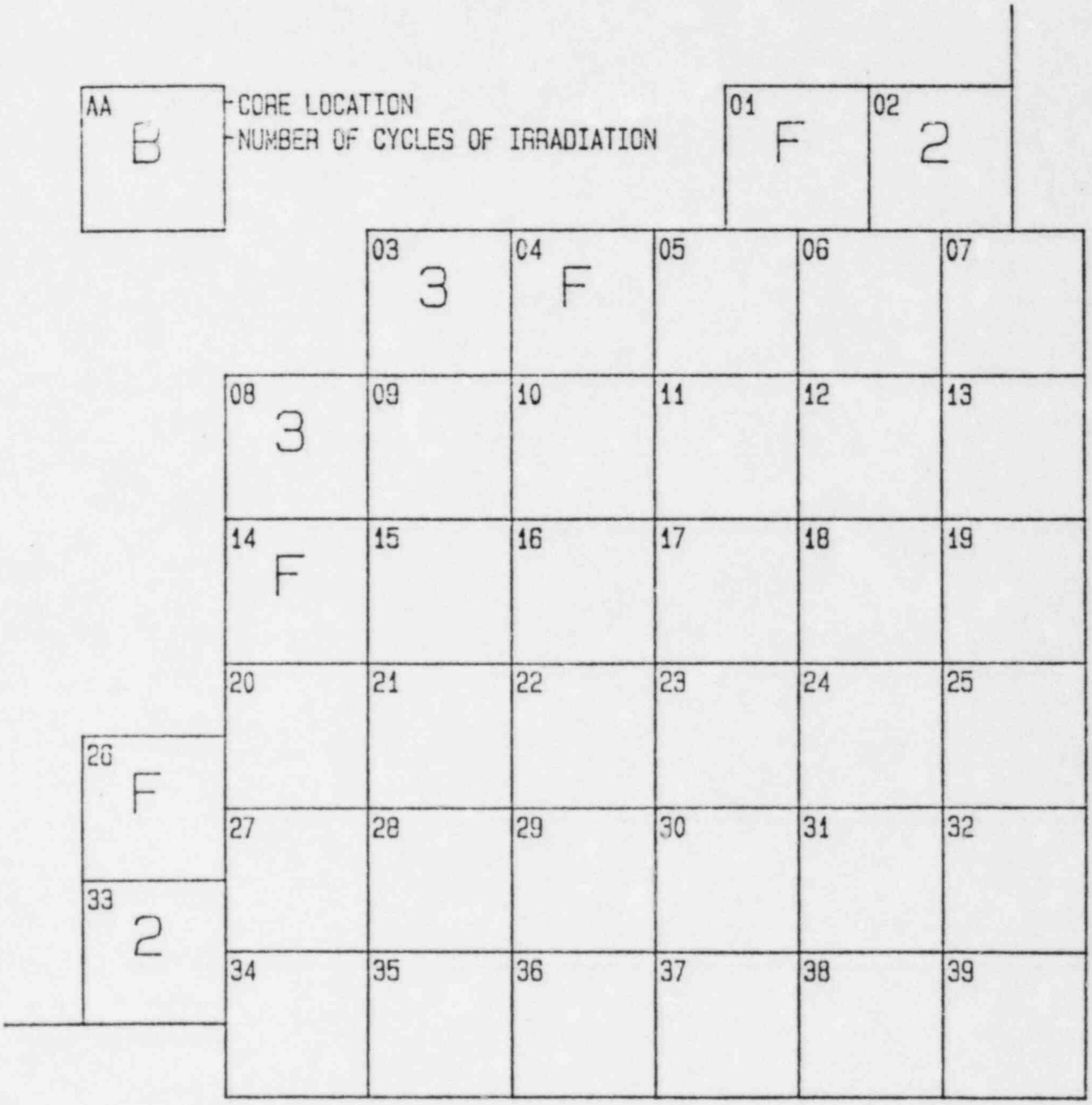


FIGURE 3

CYCLE 8

PATTERN 830TA



F is unirradiated fuel at BOC

FIGURE 4

CYCLE 7

PATTERN 715E

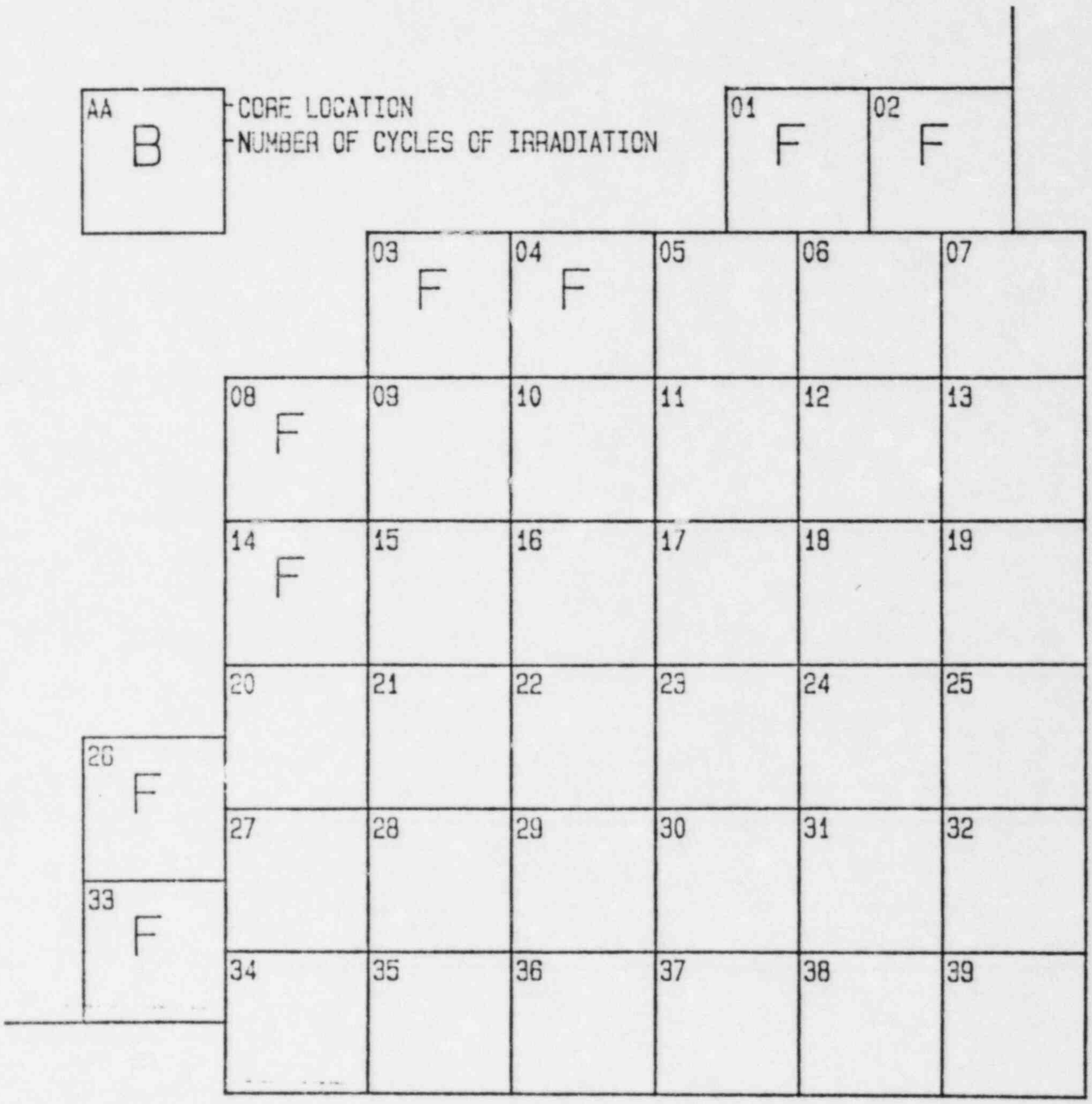


FIGURE 5

BOC 8 TO BOC 7

POWER RATIOS

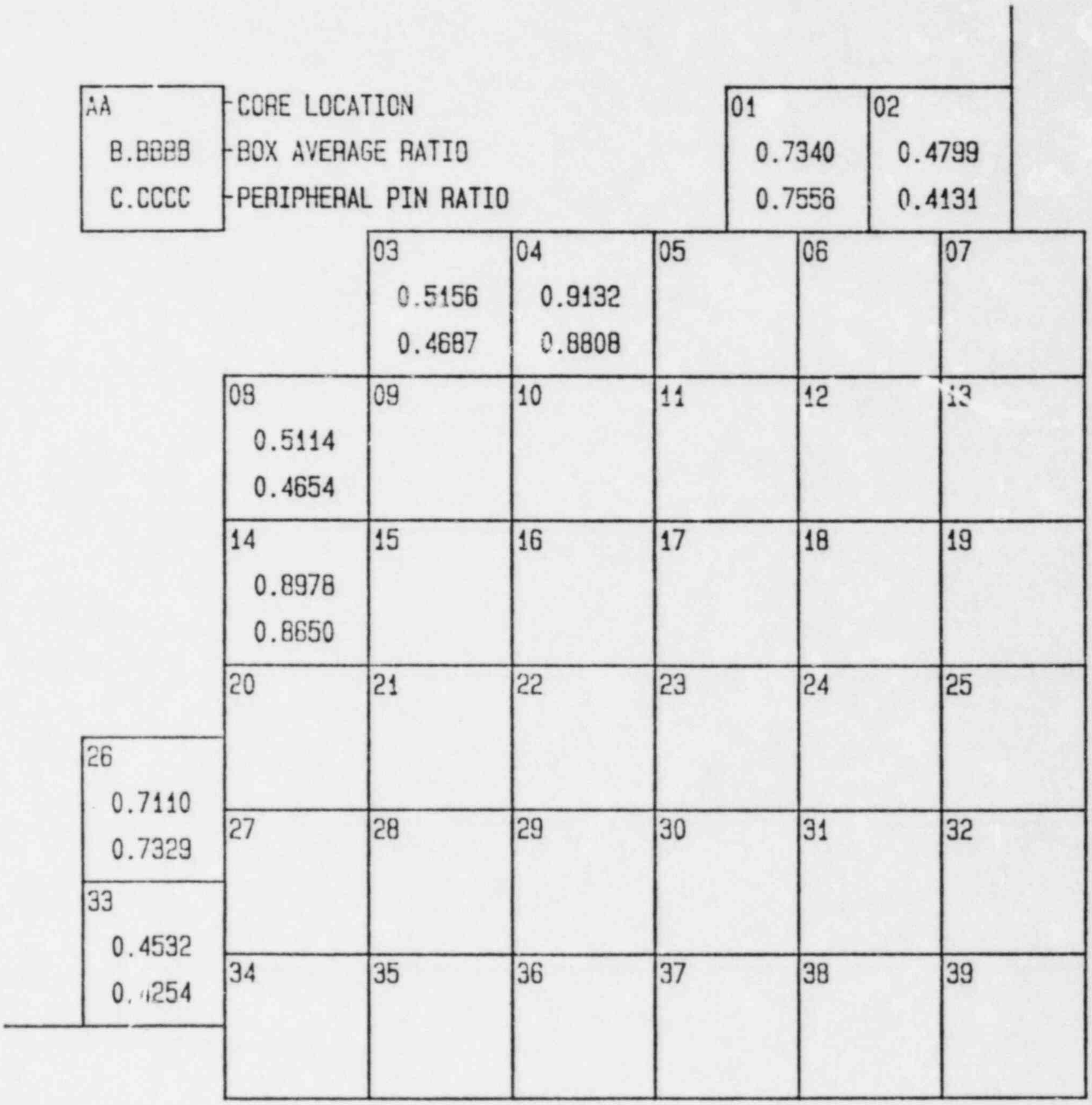


FIGURE 6

EOC 8 TO EOC 7

POWER RATIOS

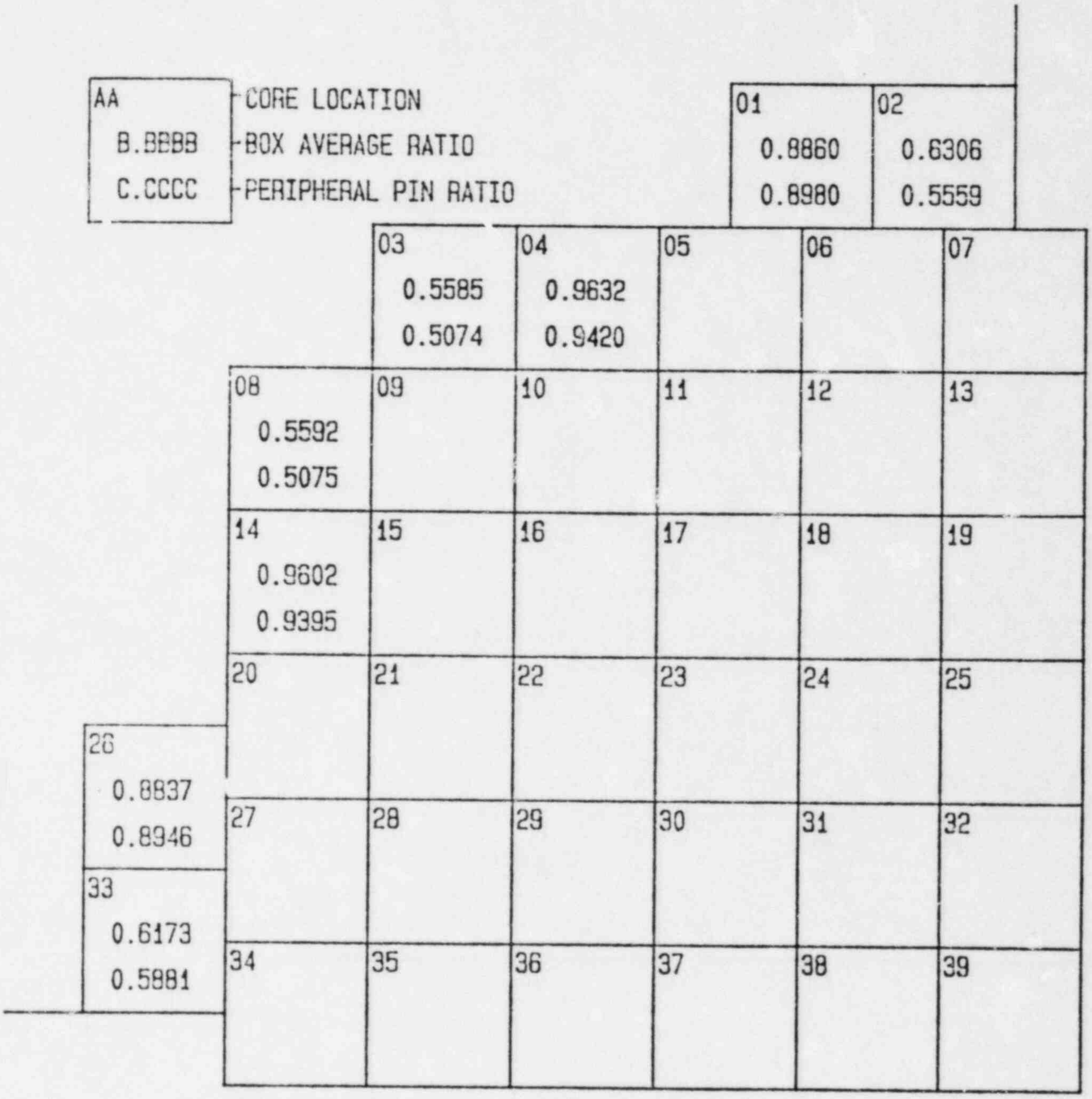


FIGURE 7

CYCLE 9

PATTERN 9 PRELIMINARY

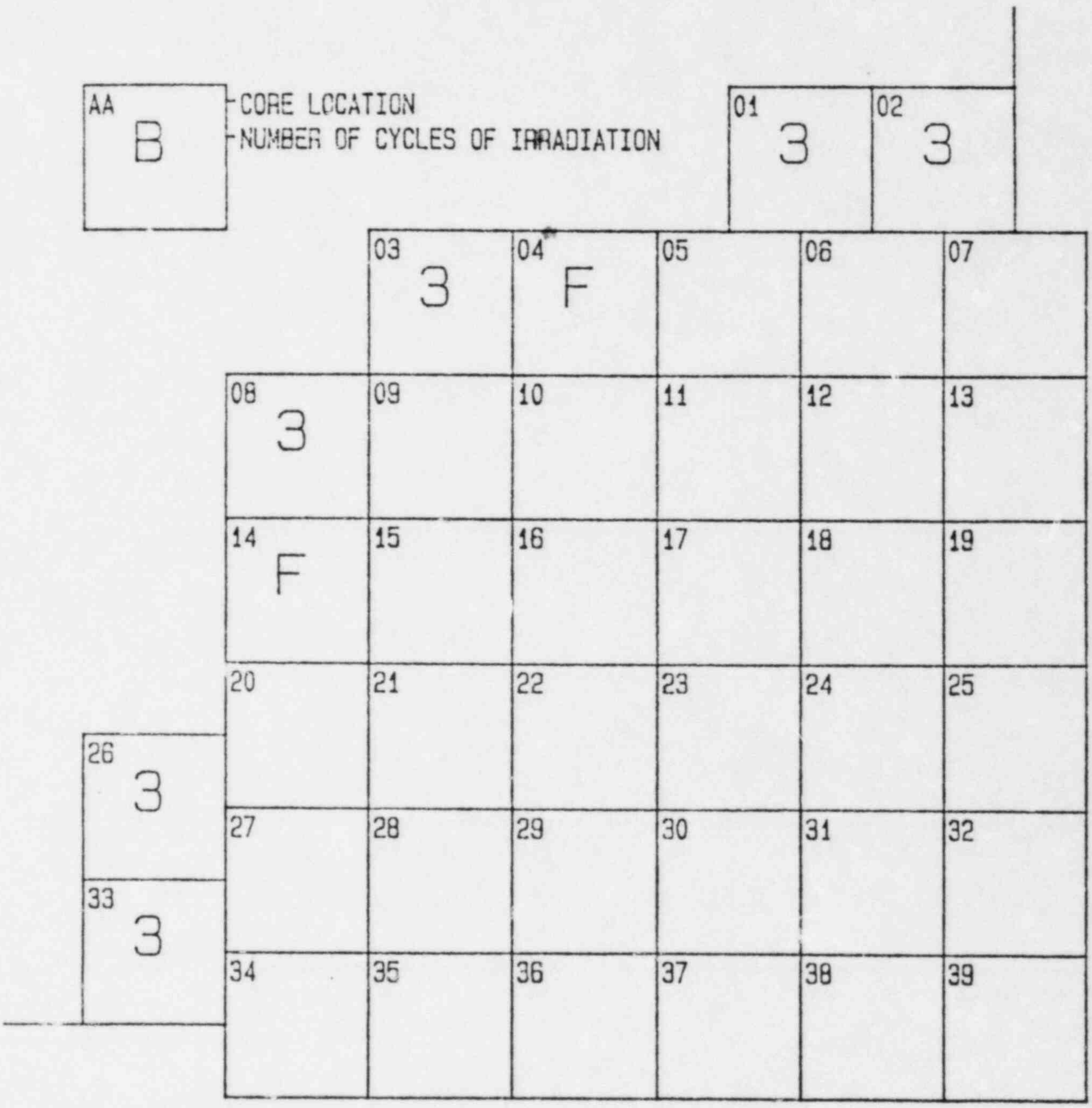


FIGURE 8

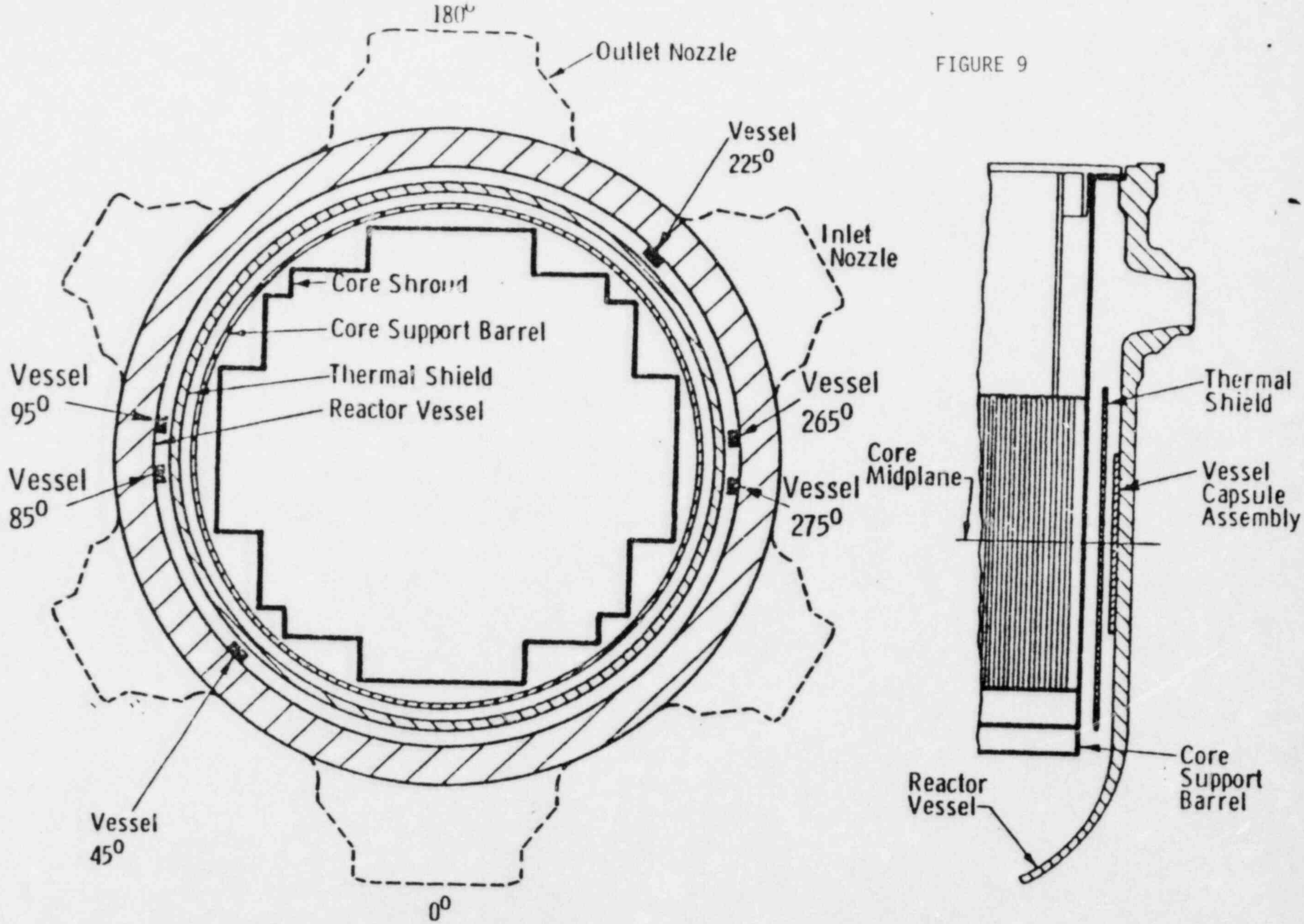
CYCLE 9 AT BOC

POWER REDUCTION FROM CYCLE 7

AA	CORE LOCATION
B.BBBB	RPD FOR PATTERN 715E
C.CCCC	RPD FOR PATTERN 9 PRELIMINARY
DD.DDD	PERCENTAGE POWER REDUCTION

				01 0.7515 0.2846 62.129	02 0.9881 0.3233 67.281	
		03 0.6654 0.3413 48.708	04 1.0083 0.8750 13.220	05	06	07
	08 0.6692 0.3395 49.268	09	10	11	12	13
	14 1.0125 0.8647 14.598	15	16	17	18	19
26 0.7517 0.2779 63.030	20	21	22	23	24	25
33 0.9880 0.3104 68.583	27	28	29	30	31	32
	34	35	36	37	38	39

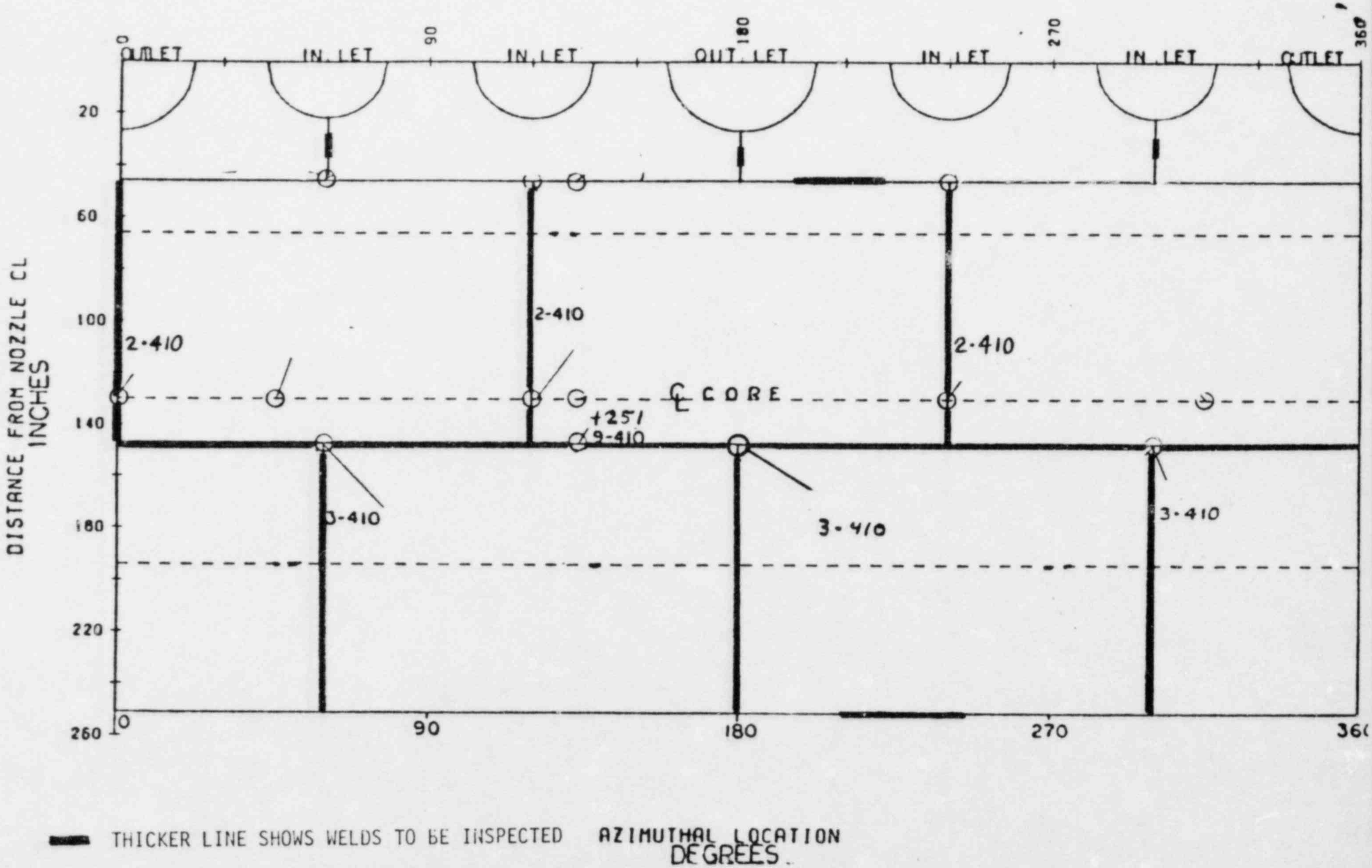
FIGURE 9



Location of Surveillance Capsules

FIGURE 10

FORT CALHOUN REACTOR PRESSURE VESSEL MAP



THICKER LINE SHOWS WELDS TO BE INSPECTED