**Omaha Public Power District**
1623 Harney Omaha, Nebraska 68102
402/536-4000

May 30, 1984
LIC-84-150

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

References:
1) Docket No. 50-285
2) Letter from James R. Miller to W. C. Jones, dated November 23, 1983
3) Letter from W. C. Jones to J. R. Miller, LIC-84-022 dated January 24, 1984
4) TR-O-MCM-002, Evaluation of Irradiated Capsule W-265, March 1984

Dear Mr. Miller:

Pressurized Thermal Shock (PTS)

In Reference 2 the Commission requested that Omaha Public Power District submit data related to reactor vessel fluence associated with current and future core configurations. The District committed to providing this data in Reference 3. This letter provides the most recent data relative to reactor vessel fluence.

Reference 4 contains the Cycles 1 through 7 reactor vessel flux distribution and the corresponding peripheral assembly power distribution. The District has utilized the DOT-IV model described in Reference 4 to calculate the reactor vessel flux distribution for Cycles 8 and 9. These flux distributions at the vessel/clad interface are shown in Figure 1 where they are plotted relative to the peak (axial and azimuthal) flux for the Cycles 1 through 7 distribution. The peripheral assembly power distributions for Cycles 8 and 9 are shown in Figures 2 and 3. The vessel/clad interface flux distribution for Cycles 1 though 9 is shown in Figure 4. The length of time during which each flux distribution was present at the reactor vessel/clad interface is given in the following table.

| Flux Distribution | Length of Time at 1500 $MW_t$ |
|---|---|
| Cycles 1-7 | 5.92 EFPY (1.868 + 8 sec) |
| Cycle 8 | .88 EFPY (2.775 + 6 sec) |
| Cycle 9 | 1.15 EFPY (3.627 + 7 sec) |
| Cycles 1-9 | 7.95 EFPY (2.507 + 8 sec) |







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Figure 5 shows the flux distribution for one fuel loading scheme being considered for future fuel cycles relative to the Cycles 1-7 flux distribution. The corresponding peripheral assembly power distribution is shown in Figure 6. This fuel loading scheme is one of the several under consideration for Cycle 10. The District will include reactor vessel flux distribution data for the core loading scheme chosen for Cycle 10 in the Cycle 10 reload analysis.

It is the District's goal to reduce the flux at the reactor vessel welds to a minimum level. To accomplish this goal, the District is exploring the utilization of part length CEA fingers in peripheral fuel assemblies and full length dummy assemblies in selected peripheral locations. The District intends to implement a fuel loading scheme beginning with Cycle 10 which maximizes the period of time between now and the time at which the PTS screening criteria is reached. We intend to implement this loading scheme irrespective of the fact that it may be judged not "reasonable and practical" as discussed in the proposed PTS rule. However, if the loading scheme initiated in Cycle 10 is not "reasonable and practical" the District intends to submit an analysis to demonstrate that the Fort Calhoun reactor vessel can be safety operated with an $RT_{NDT}$ greater than the screening criteria as soon as possible. Such an analysis would allow us to operate with a "reasonable and practical" flux reduction program.

The District has recently completed a preliminary probabilistic risk assessment based study of the PTS risk at the Fort Calhoun Station. The study shows that the projected risk due to a PTS event for Fort Calhoun Station is approximately $5 \times 10^{-5}$/RY at the end of design life. The projected end of design life $RT_{NDT}$ used in the study was 310°F for the limiting longitudinal weld. This risk is lower than the risk of $10^{-4}$/RY corresponding to the NRC staff screening criterion for longitudinal welds with an $RT_{NDT}$ of 270°F. SECY 82-465 presents the conclusions: "The risk from PTS events for reactor vessels with $RT_{NDT}$ values less than the proposed screening criterion (270°F for axial welds, and 300°F for circumferential welds) is acceptable."

The results of this study indicate that for all PTS events considered, the risk to Fort Calhoun vessel integrity at the end of design life is less than that which has been stated as being acceptable by the NRC staff. A detailed plant specific analysis of the risk of PTS to Fort Calhoun is expected to show an even lower risk of exceeding vessel integrity limits. A summary of this study is enclosed.

The District has also studied the possibility of shielding the reactor vessel welds. A preliminary design was developed for a shield located between the core shroud and core barrel. However, measurements taken during our 1984 refueling outage indicate that there is insufficient space between the core shroud former plates and the core barrel to allow installation of this shield. The study did identify a material, $TiH_2$, which has considerable shielding potential because two inches of the material reduces the neutron flux by a factor of two. The District is considering the possibility of attaching this material to the thermal shield if the thermal shield is sufficiently stable or to the core barrel if it ever becomes necessary to remove the thermal shield.

The District is continuing to discuss the possibility of sampling the reactor vessel welds with our reactor vessel manufacturer. The manufacturer has a limited effort underway to assess the feasibility of this sampling.

The District is continuing to aggressively pursue resolution of the PTS issue for the Fort Calhoun Station. This is evidenced by the extensive programs discussed in this letter. The District will continue to keep you informed of our plans to resolve this issue.

Sincerely,

WC Jones

W. C. Jones
Division Manager
Production Operations

WCJ/JKG/lp

Enclosures

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Mr. E. G. Tourigny, Project Manager
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Inspector

# SUMMARY OF A PRELIMINARY EVALUATION OF PRESSURIZED THERMAL SHOCK RISKS FOR THE FORT CALHOUN REACTOR VESSEL

A study of the risks of certain pressurized thermal shock (PTS) conditions to the integrity of the Ft. Calhoun reactor vessel at its projected end of life conditions has been performed by Combustion Engineering. This study evaluated the risk to vessel integrity by combining event probabilities from C-E and industry sources with the corresponding conditional probabilities of crack extension without arrest calculated by the CEPFM computer code.

Five transients were evaluated in this study: three steam line breaks, a small break LOCA with loss of feedwater and a steam generator tube rupture. These transients were selected to correspond to the types of transients considered by the NRC staff in establishing the PTS screening limit. The transients present a spectrum of postulated challenges to the Ft. Calhoun reactor vessel from the mild overcooling of a steam generator tube rupture event to the very conservative and severe cooldown and repressurization which is associated with a large (6.3 $ft^2$) steam line break.

Event probabilities for these scenarios were conservatively estimated from C-E and other industry sources. Since the severe steam line break and small break LOCA transients include unrealistic assumptions such as the presence of high head HPSI pumps, it is difficult to determine definitive event probabilities. Detailed plant-specific analyses would reduce the conservatisms of these values.

The CEPFM computer code, which performs probabilistic linear elastic fracture mechanics, was used to evaluate the conditional probabilities of crack extension without arrest for the Ft. Calhoun reactor vessel for each of the five transients. Plant-specific values were used for peak fast neutron fluence, initial $RT_{NDT}$ and weight percentages of copper and nickel in the limiting weld, and pre-existing flaw sizes. These variables were modeled as random variables characterized by probability distributions. Repeated random sampling of these probability distributions is used as input to linear elastic fracture mechanics algorithms. The number of crack extensions without arrest are tallied and weighted by the probabilities of flaw size ranges. Combining these probabilities for all flaw size ranges yields the conditional probability of crack extension without arrest, given that the transient under consideration has occurred.

The flaw size probability distribution used in this analysis was separated into an evaluation of the flaw density in weld and heat affected zone and the distribution of flaw size given that a flaw exists. The distribution of flaw size provided by the Marshall study was used to model the assumed variation of pre-existing flaw sizes given that one or more flaws exist in the beltline region of the vessel. This distribution is the most current and is believed to be conservative for nuclear pressure vessels. The results of the inservice inspection which was performed at Ft. Calhoun in early 1983 were used along with available industry wide data to evaluate the density of flaws in the inspected region of the vessel. The probability of one or more flaws was then determined for the inspected regions of the Ft. Calhoun vessel. In this manner, inservice inspection results were used to provide a flaw distribution for the Ft. Calhoun reactor vessel which accounts for uncertainties in the inspection technique in a conservative fashion.

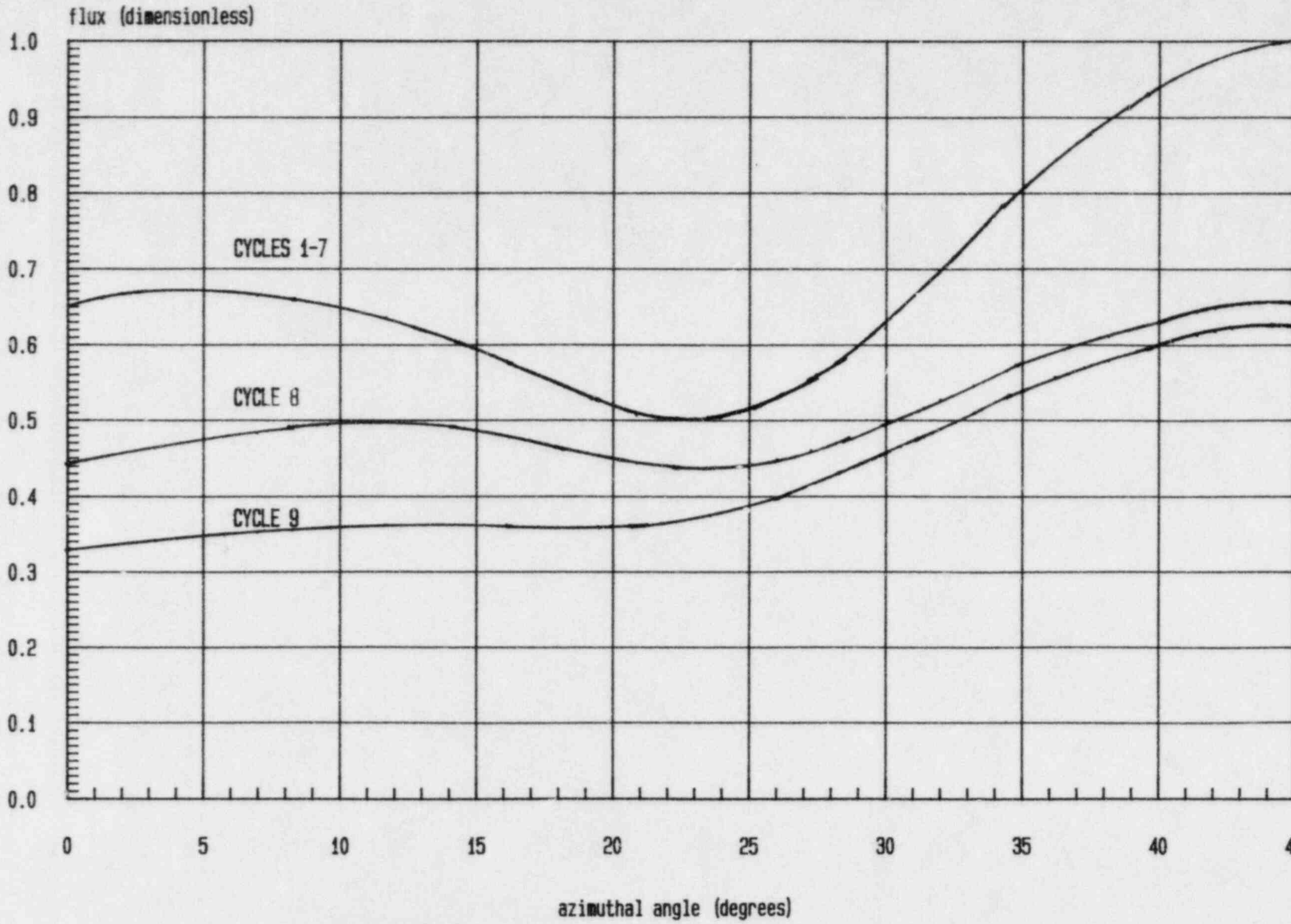
The result of this study consists of an estimate of risk for each of the five transients evaluated. The risk is defined to be the product of the probability of each transient event and the conditional probability of crack extension without arrest given that the event has occurred. The maximum conditional probability was calculated to be $4.8 \times 10^{-3}$ for a 6.3 $ft^2$ steam line break with excess auxiliary feedwater.

Combining this maximum value with the NRC staff estimate of the probability of all PTS events of ($\sim 10^{-2}$/reactor-years) yields a projected risk of $\sim 5 \times 10^{-5}$/RY at end of the Ft. Calhoun vessel design life. This risk is lower than the risk corresponding to the NRC staff screening criterion for longitudinal welds at 270°F of $10^{-4}$/RY.

The transient events analyzed represent very severe challenges to reactor vessel (integrity) at the projected end of life $RT_{NDT}$ of 310°F. Although more severe events might be postulated, it is unlikely that the overall risk to vessel integrity will be found to vary more than a factor of ten from the results of this study. A more detailed PTS risk assessment for Ft. Calhoun can be expected to predict even lower risk of loss of vessel integrity. These results were obtained without assuming further modification in core design or system configuration and no credit was taken for possible warm pre-stress effects.

# FIGURE 1: PAST AND CURRENT CYCLES FLUX DISTRIBUTION

[FLUX RELATIVE TO 4.733E10 n/(cm**2*sec)]



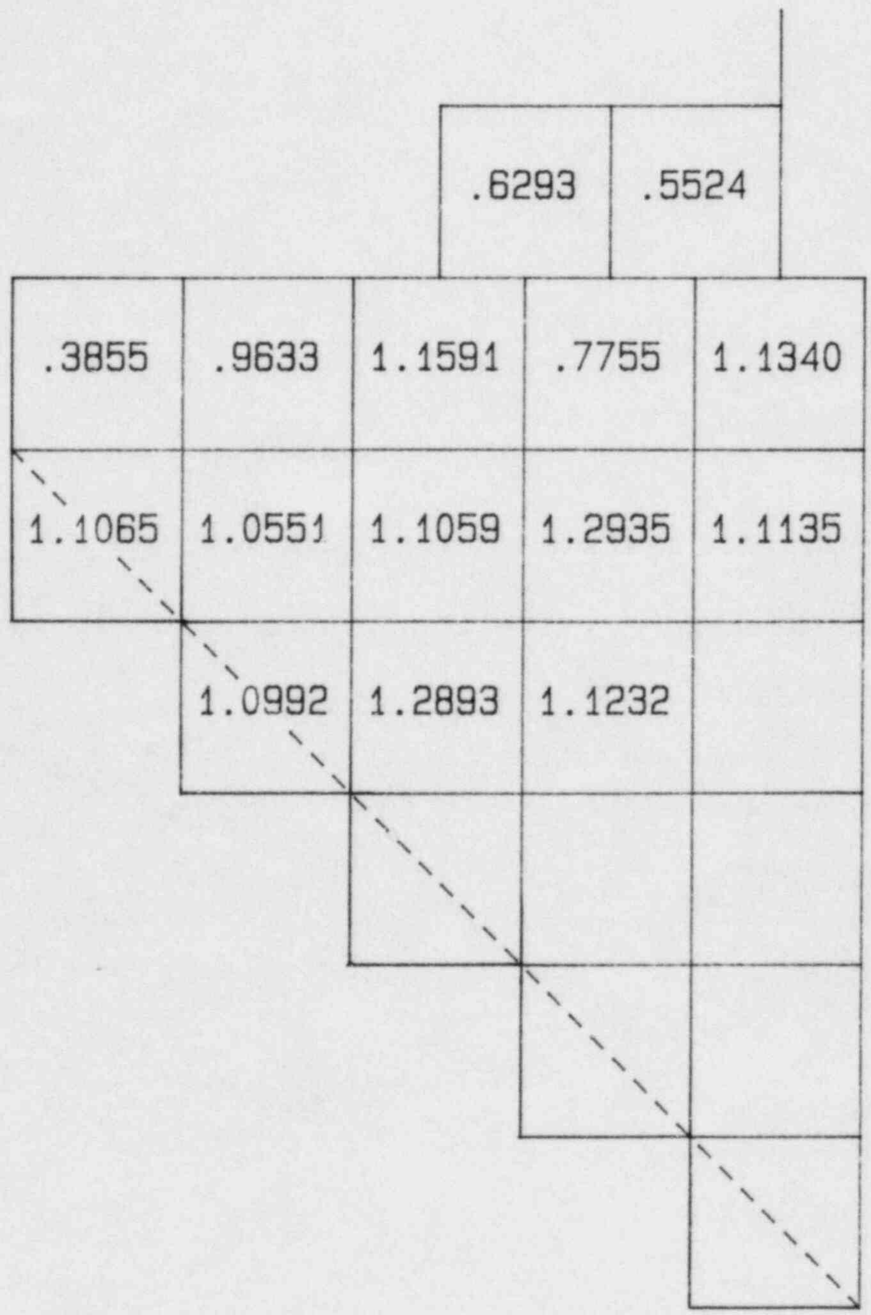



FIGURE 2: CYCLE 8 PERIPHERAL ASSEMBLY POWER DISTRIBUTION

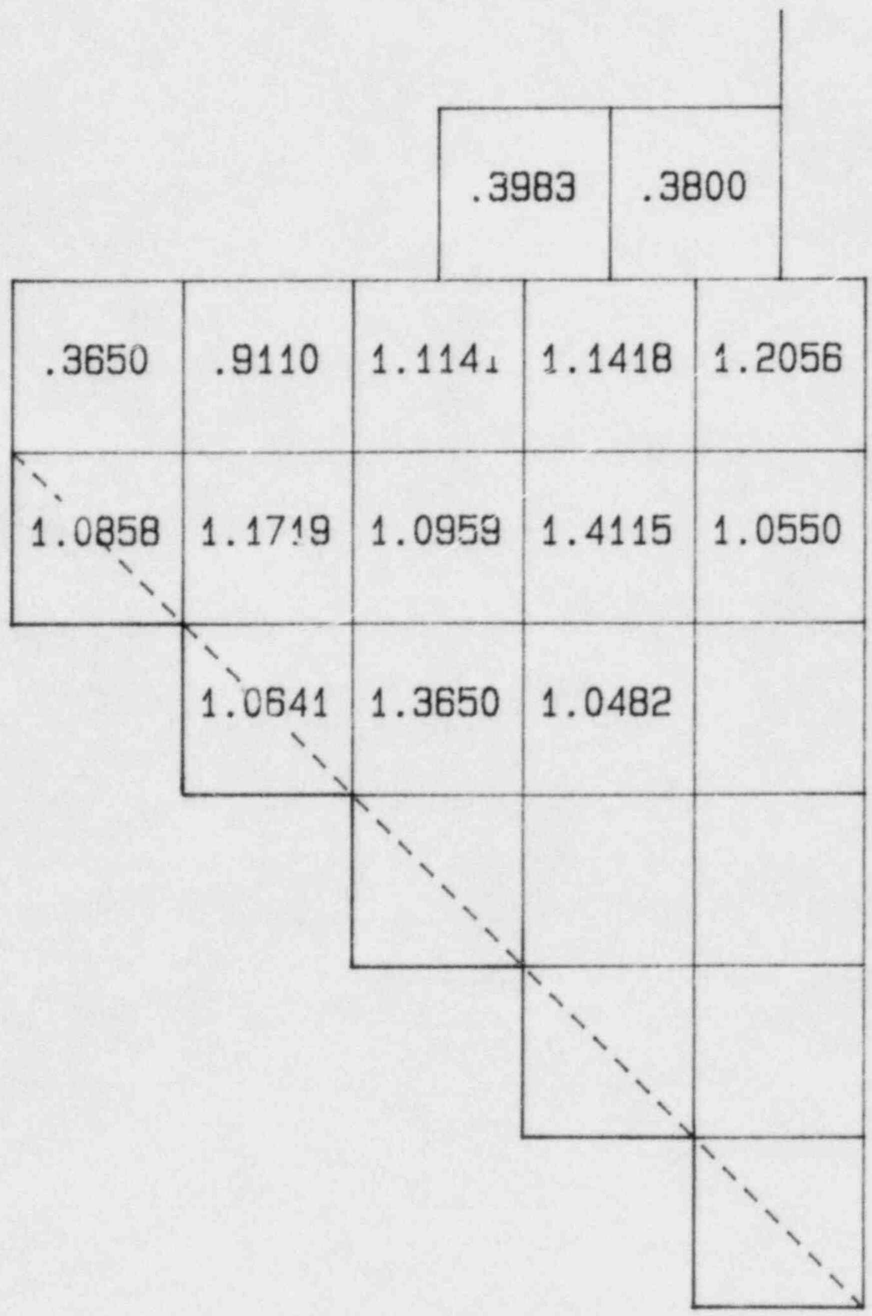



FIGURE 3: CYCLE 9 PERIPHERAL ASSEMBLY POWER DISTRIBUTION

FIGURE 4: FT. CALHOUN CYCLES 1 THROUGH 9 FLUX DISTRIBUTION

[FLUX RELATIVE TO 4.733E10 n/(cm**2*sec)]

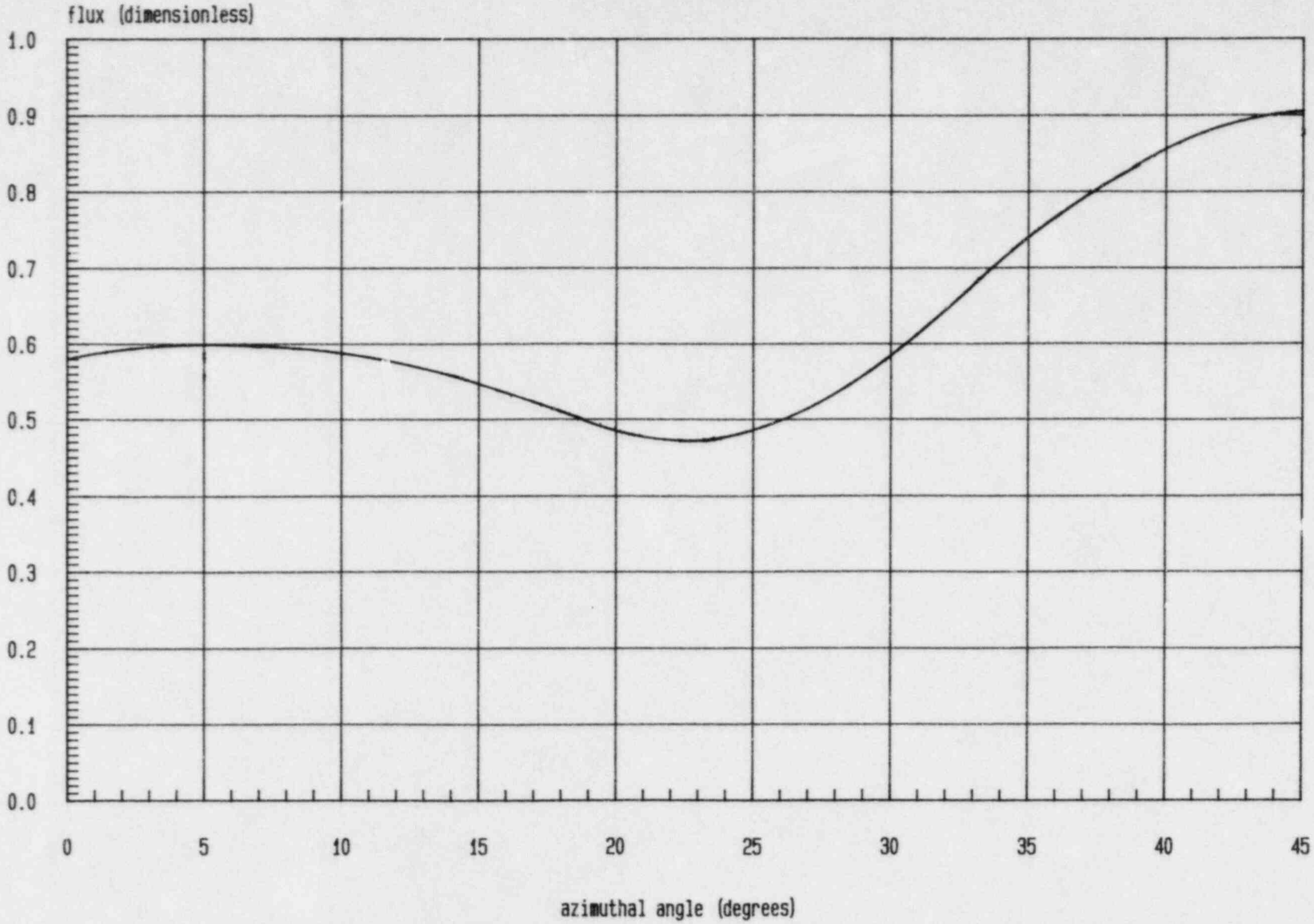
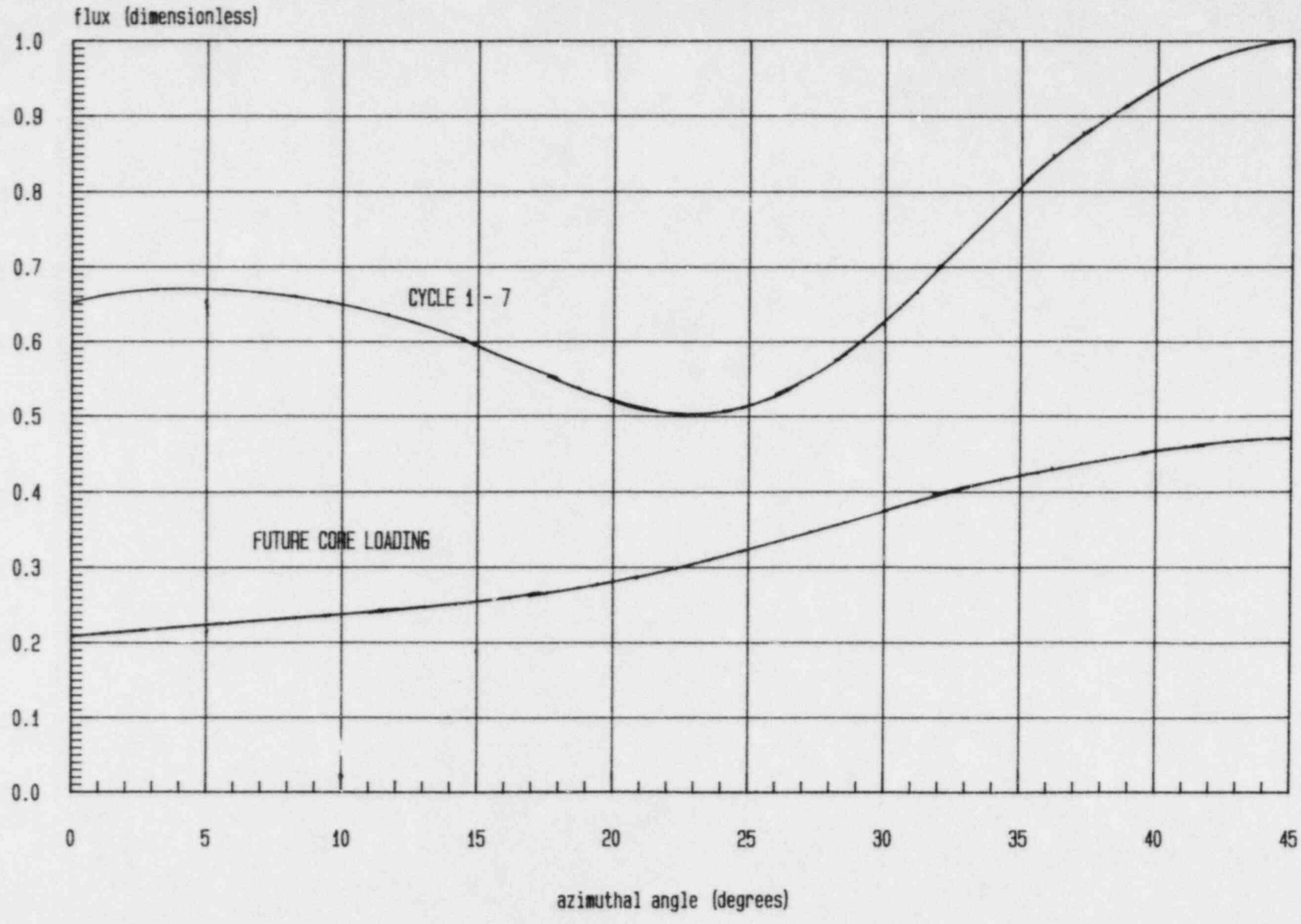


FIGURE 5: POTENTIAL FUTURE FT. CALHOUN CORE LOADING FLUX DISTRIBUTION

[FLUX RELATIVE TO 4.733E10 n/(cm**2*sec)]



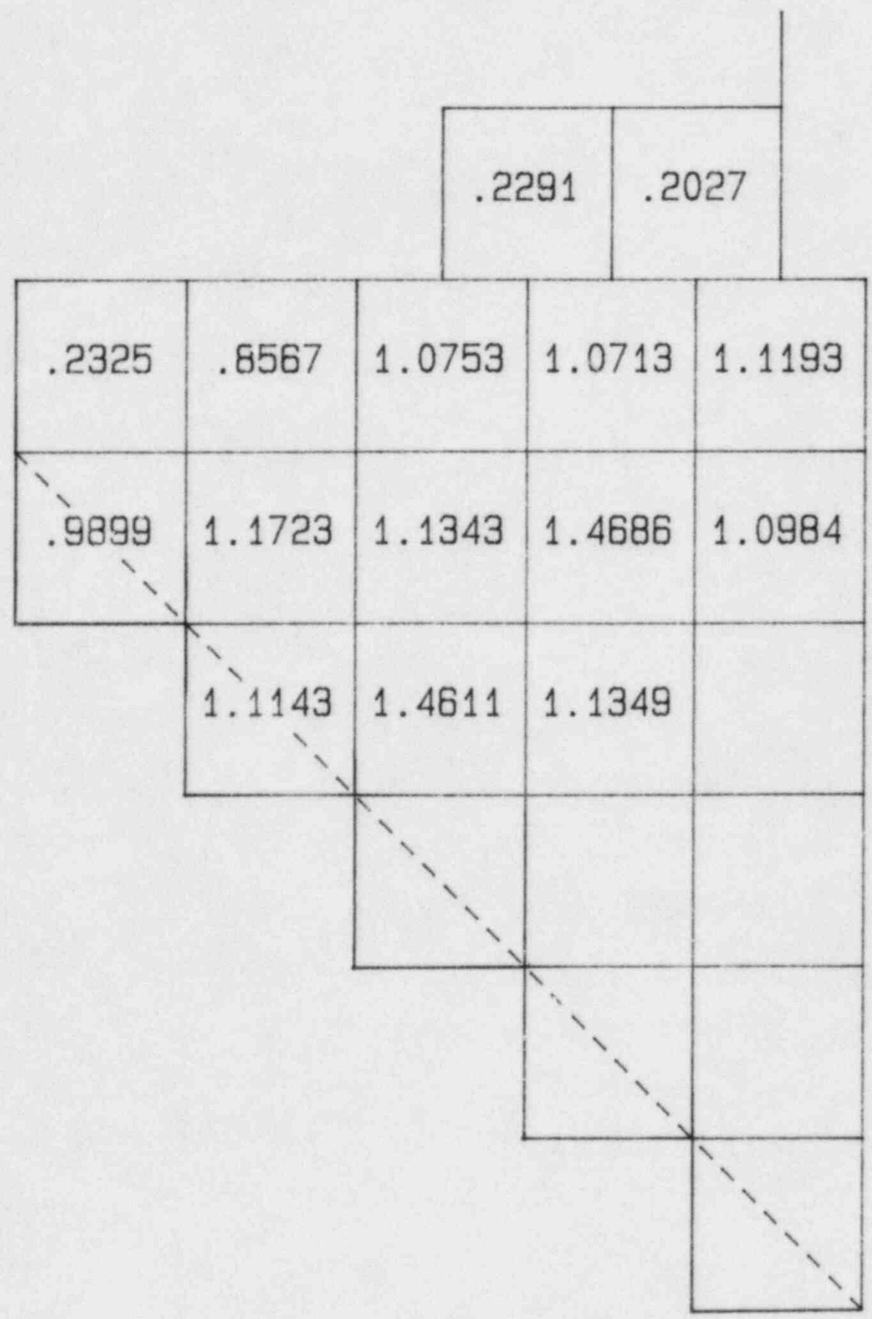



FIGURE 6: PERIPHERAL ASSEMBLY POWER DISTRIBUTION FOR POTENTIAL FUTURE FLUX REDUCTION CORE LOADING