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PHOENIX FUEL PROGRAM PROGRESS REPORT

D. D. Lanning
G. J. Busselman

November 1967

AEC RESEARCH & DEVELOPMENT REPORT

BATTELLE NORTHWEST
BATTELLE MEMORIAL INSTITUTE
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PHOENIX FUEL PROGRAM PROGRESS REPORT

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PHOENIX FUEL PROGRAM PROGRESS REPORT

1. INTRODUCTION AND PROGRAM SUMMARY

1.1 - Phoenix Fuel Description

The Phoenix Fuel Program is a research and development project concerned with the use of plutonium fueled systems for extra long reactivity lifetime. Computational studies have been made[19] that show the potential inherent in the utilization of ^{240}Pu as a convertible poison. The conversion of the fertile ^{240}Pu into the fissionable ^{241}Pu helps balance the loss of the initial ^{239}Pu during power operation. Thus, a plutonium fueled system can be developed with ^{240}Pu as a reactivity control much as boron is used as the burnable poison in ^{235}U fueled systems; however, the ^{240}Pu has the added advantage of conversion to a fissionable isotope ^{241}Pu.

A preliminary study of the use of Phoenix fuel, in a pressurized water compact reactor reveals the following general neutronic characteristics:

1. The utilization of Phoenix fuels in water moderated cores permits reduction in shim control requirements. Further reductions come about because of the reduced peak xenon poisoning in plutonium cores. The reduced control requirements should make it possible, in principle, to obtain simple, more compact, core designs at less cost.
2. The use of burnable poisons such as boron with the plutonium fuel can result in further improvements in reactivity-lifetime characteristics. The large cross sections of plutonium isotopes make it possible to avoid the extensive use of self-shielded (lumped) burnable poison elements and thus, again, lead to simpler core designs.

3. The negative ^{240}Pu Doppler coefficient can be utilized to improve the safety characteristics of the core.
4. The neutronic advantages of the plutonium burnup must be balanced with the reduced control rod worths and smaller delayed neutron fraction to offset some of the reduction in control requirements, and the spatial power peaking which might arise in the short diffusion length of plutonium cores.

Because of the favorable nuclear aspects, the Phoenix fuels afford an opportunity to improve the operating characteristics of compact reactor systems, such as marine propulsion special purpose compact power plants and research or test reactors.

The development of these ideas into a technically sound concept involves coordinating several groups: reactor physics, reactor engineering, and fuels engineering, together with reactor operations groups for criticality and burnup study. Reactor Physics provides the criticality experiments, the theory-experiment correlation, and the predictions of core reactivity, together with power distributions as a function of core operating conditions including control systems and fuel burnup. Reactor engineering combines the reactor physics results with thermal hydraulic analyses to develop the region of practical interest for operating cores, including cost evaluations of possible applications. Fuels engineering develops the plutonium fuel fabrication techniques complete with irradiation testing and evaluation of promising fuel designs. In addition to the criticality experiments, it is important to have an easily analyzed burnup experiment with a Phoenix fuel core. And finally, a proof-of-principle testing of the reactor concept is needed at high power levels and at temperatures characteristic of power reactors.

Thus, the Phoenix Fuel Program consists of three general phases designed to develop the required technology and to demonstrate the concepts:

Phase I: is concerned with acquiring the fundamental data and developing the technology basic to the Phoenix Fuel Program. It also includes broad application studies exclusive of fuels development.

Phase II: represents the MTR-Phoenix fuel burnup experiment. This includes the development of a full core loading (19 elements in a 3x9 array with 8 shim follower fuels) of Phoenix fuel for the MTR, to be operated in such a manner that theory-experiment correlations can be made to establish a firm calculational technique for burnup prediction of Phoenix fueled cores.

Phase III: represents design, fabrication, and proof-of-principle testing of the reactor concept at high power levels and temperatures characteristic of power reactors.

1.2 - Program Schedule

Figure 1.2-1 indicates the program schedule in terms of individual projects. The experiments designated as CAF, PRCF, and MTR, involve the use of various critical approach experiments and critical experiments that are described in more detail within this document. The critical approach facility experiment (CAF) and the mockup of the MTR in the Plutonium Recycle Critical Facility (PRCF) have been completed by PNL. The Phoenix fuel burnup experiment in the Materials Testing Reactor (MTR) is in the planning stages and will be operated under the direction of INC.

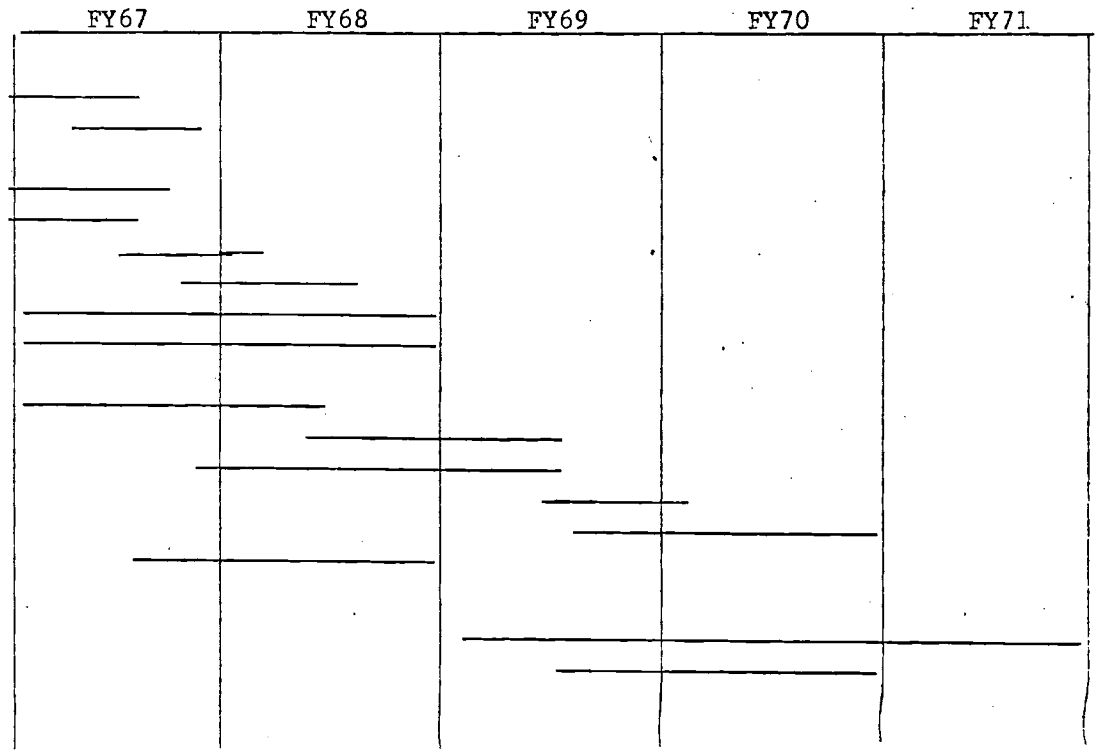
FIGURE 1.2-1

Phoenix Fuel Program Schedule

Phase I and II Tasks

PhF-1. CAF Experiment
PhF-2. CAF Analysis
PhF-3. PRCF Experiment (Planning and Preparation)
PhF-4. PRCF Fuel Fabrication
PhF-5. PRCF Experiment
PhF-6. PRCF Experiment Analysis
PhF-7. MTR Core Nuclear Analysis
PhF-8. MTR Core Thermal Hydraulic Design
PhF-9. MTR Fuel Development and Final Fuel Specifications
PhF-10 MTR Fuel Fabrication
PhF-11 MTR Experiment Preparation
PhF-12 MTR Burnup Experiment
PhF-13 MTR Experiment Analysis
PhF-14 Phoenix Fuel Applications

Phase III (Preliminary Schedule)
Core Analysis
Fuel Development



1.3 - MTR-Phoenix Fuel Specification Schedule

During the past year, major efforts by the Phoenix Fuel Program have been to develop specifications for the fuel elements to be used in the MTR burnup experiment. Figure 1.3-1 shows the interaction of the various groups involved in the development of these specifications, and the flow of information to develop a final design and a completed set of fuel specifications. This work is now finished and fabrication is about to begin.

1.4 - MTR-Phoenix Fuel Burnup Experiment Schedule

As indicated in Section 1.2, it is expected that the Phoenix fuel burnup experiment will start about the beginning of CY-1969. In order to develop the necessary program plans and to provide the safety and operating analysis documents, it is expected that a flow of information will be made as shown in Figure 1.4-1.

The laboratory responsibilities for acceptance of the fuel to be operated in the MTR and for obtaining the proper approvals to operate the Phoenix fuel core in the MTR fall to the MTR operating group of the Idaho Nuclear Corporation. As shown in the Figure, any information that has been developed at PNL will be supplied to INC, and assistance that they request will be given for preparation of appropriate documents. These coordinations have been discussed in a meeting between INC and PNL. The results of the burnup experiment will then be gathered by PNL and published as a program document.

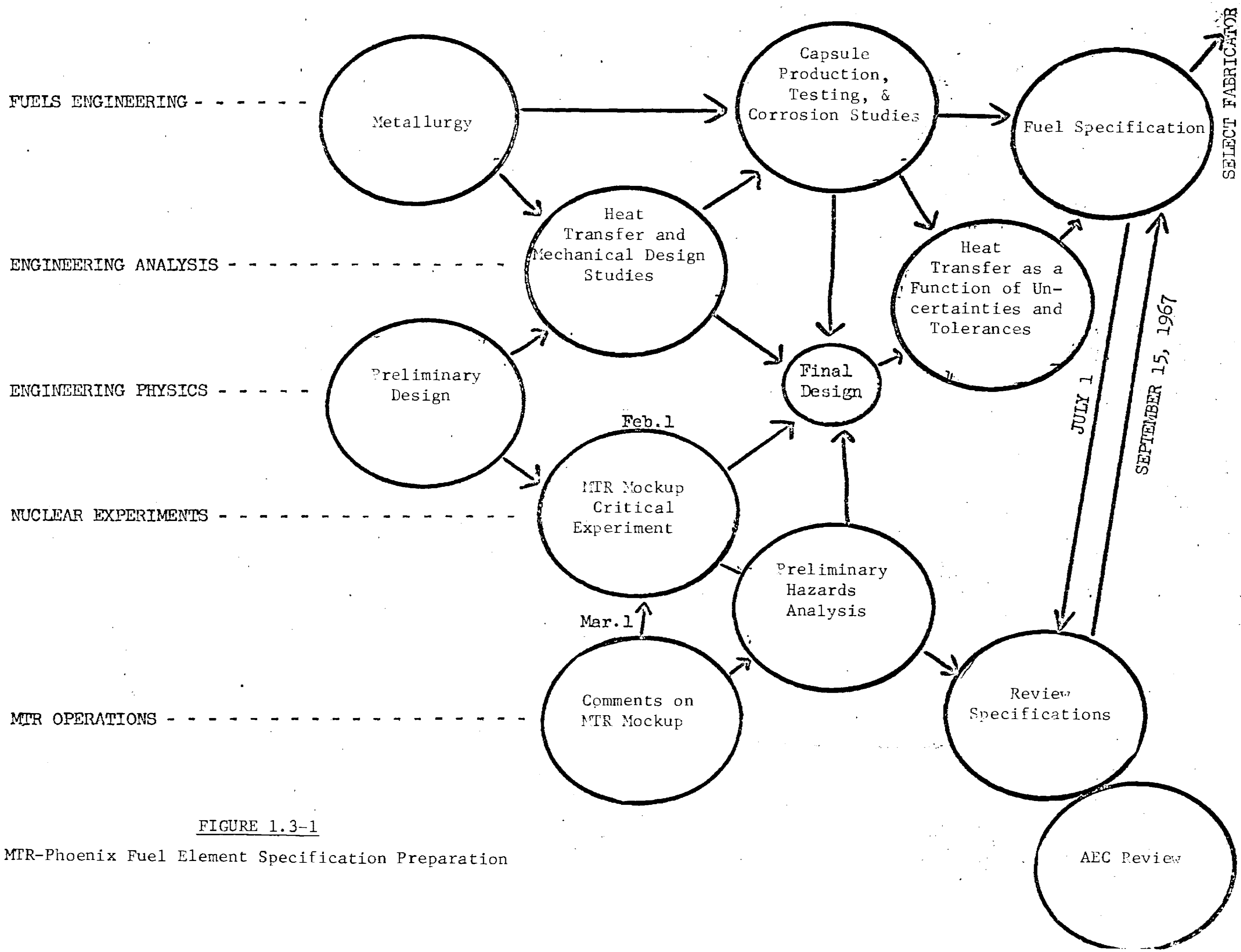
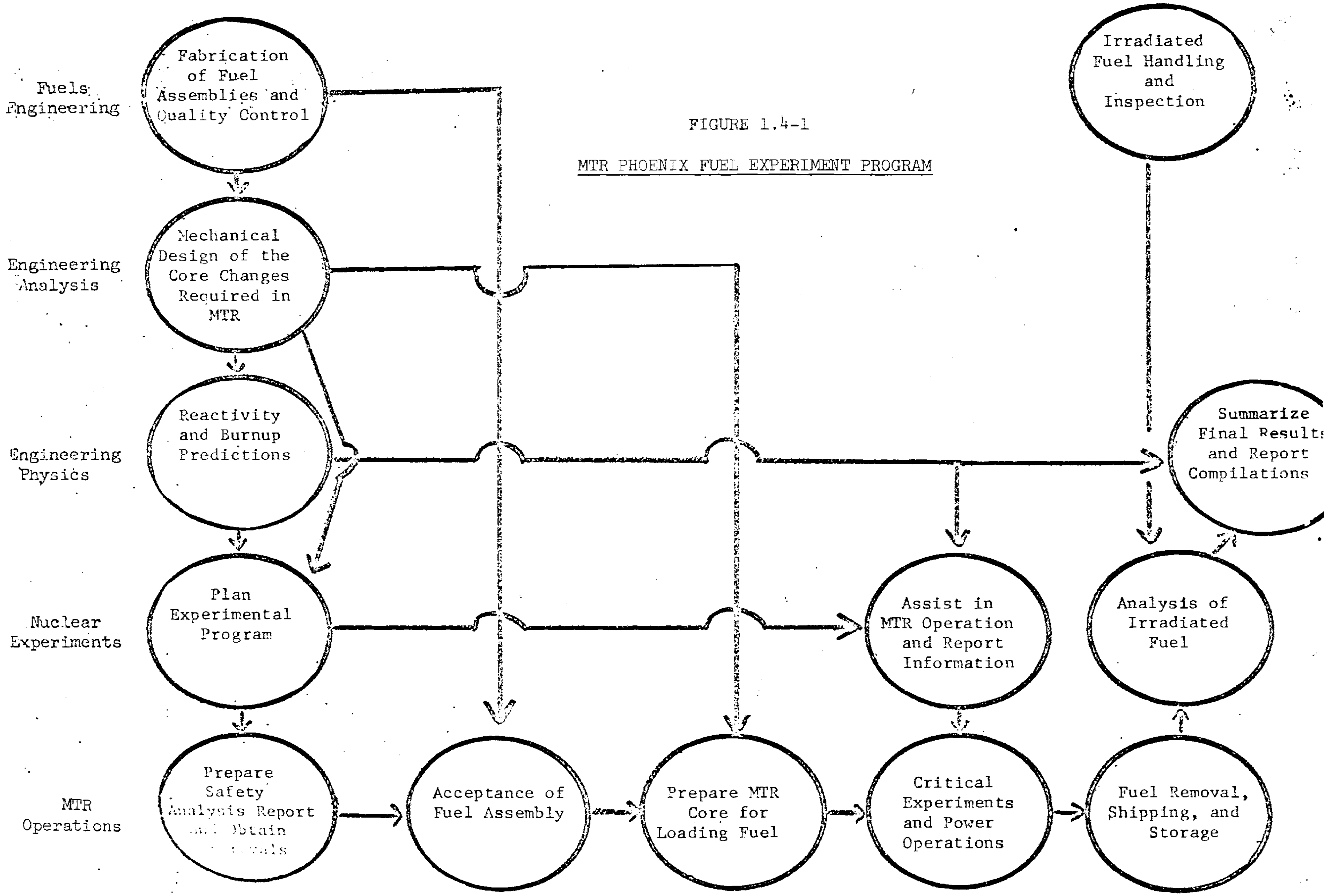


FIGURE 1.3-1

MTR-Phoenix Fuel Element Specification Preparation

FIGURE 1.4-1

MTR PHOENIX FUEL EXPERIMENT PROGRAM



2. HIGHLIGHTS

2.1 - CNSG-Phoenix Conceptual Studies

Preliminary calculations show some economic incentive for substituting plutonium for uranium in a typical reactor core for maritime application (CNSG). More detialed calculations are being made to determine if further optimization is possible and what economic advantages accrue.

2.2 - PRTR-Phoenix Conceptual Studies

A calculational study was performed to determine the physics characteristics of plutonium Phoenix fuel in the Plutonium Recycle Test Reactor (PRTR). The reactivity effects associated with fuel enrichment, nuclear heating, and boron dissolved in the moderator were investigated. Core lifetime calculations were made for a 55-element batch core of Phoenix fuel, and an 85-element zoned core containing irradiated UO_2-PuO_2 fuel and Phoenix fuel.

2.3 - HFIR-Phoenix Conceptual Studies

A calculational study is underway to examine the possible benefits of replacing the ^{235}U fuel in the HFIR with plutonium. Preliminary calculations indicate the possibility of increased core lifetime and total flux, but a lower thermal flux in the center trap.

More detailed calculations now underway show a thermal neutron flux advantage in the center for Pu fuel over U fuel. An increase of 20% in thermal flux and 25% in total flux is calculated. Burnup calculations have not yet been carried out.

2.4 - CAF -Phoenix Fuel Experiment

Approach-to-critical experiments have been conducted in the Critical Approach Facility (CAF) with a hydrogen-moderated, Al-20.3 wt% Pu-fueled system. The plutonium isotopic composition was 90.48% ^{239}Pu, 8.19% ^{240}Pu, 1.25% ^{241}Pu, and

0.8% ^{242}Pu. The fuel was in the form of disks, which were assembled together with additional aluminum and polyethylene disks to form cylindrical fuel columns. The results of the experiments, which were comprised of extrapolated heights for critical of a symmetric 19-column array, either clean or with the center column replaced by a poison column, have been used to test the computational methods currently being applied to Phoenix fuel cores. The methods used are, in general, able to come to within 1% in k_{eff}.

2.5 - PRCF-Phoenix Experiment and Analysis

A critical mockup was constructed in the Plutonium Recycle Critical Facility which closely duplicated the physical configuration planned for the MTR-Phoenix fuel burnup experiment. The core was comprised of a three-by-nine array containing nineteen fuel elements and eight shim rods with fuel followers. The fuel comprised 19.95 wt% Pu with nominal plutonium isotopic composition of 76.92% ^{239}Pu, 19.31% ^{240}Pu, 3.18% ^{241}Pu, and 0.59% ^{242}Pu.

Experiments have included shim-free critical size, banked shim height for full core critical, temperature coefficient of reactivity, and power distributions throughout the core. The results of the experiments are being used to test computational methods currently applied to the MTR-Phoenix fuel experiment.

The rough calculations carried out prior to and during the experiment were about 6% high in k with about half of this due to various approximations. The banked shim position (after normalizing the calculations to the shim-free core) at critical was predicted to within one-half an inch. More detailed calculations are now underway.

2.6 - Fuel Development and Fabrication Summary

The Phoenix fuel element and shim control rod are similar in design to their standard MTR counterparts. The major physical differences are the use of a plutonium-aluminum alloy core, a thicker plate, and fewer plates. The fuels will operate under similar conditions to the existing MTR fuels, and no radiation damage or swelling problems are anticipated. The fuel core is fabricated by casting a billet and extruding it to shape, and the ends of the core are shaped to provide a tapered core in the finished plates. 6061 aluminum alloy has been selected as the cladding material; however, the 6061 will be clad with 1100 aluminum alloy to promote interface bonding. The plates are fabricated by hot rolling, blister testing, cold rolling, and annealing, with a total reduction of 12 to 1. All plates are X-rayed and ultrasonically tested for bonding. The X-rays are read with a densitometer to determine homogeneity of the core and end taper. Destructive samples are checked for end taper shape, clad and core thickness, bond integrity and plutonium content. The fuel section of the shim control rod will be roll swaged and welded rather than brazed, and it is planned to use flux monitors in some of the fuel elements. Both of these features are nonstandard, and dummy elements will be prepared for hydraulic and shock testing prior to production. An irradiation test element utilizing Phoenix type plates has been prepared and is scheduled for charging in October, 1967. Fuel element and shim control rod specifications have been prepared and issued for AEC and INC review.

2.7 - MTR Core-Mechanical Design

The Phoenix Fuel Experiment will require some mechanical modification of the MTR. The shifting of the fuel array to the center of the reactor core will necessitate rearranging the lattice pieces and changing the temperature and pressure monitoring system to accomodate the shifted core. The presently used lattice pieces with internal basket holes will have to be replaced with the existing solid beryllium pieces.

The Phoenix loading calls for the use of eight shim control rods, thus an existing KAPL loop now in the No. 42 position will need to be removed. The shim rod in this position may not be scramable as the lower shock absorber has probably been altered. Efforts to date to determine the condition of this absorber have not been successful. The shim rod motors now in use will need to be changed to slower speed motors which were used in cycle 108 and are still available.

During the course of the experiment, a method of monitoring the neutron flux within the core is desired. This can be accomplished by inserting a small flux wand into the fuel boxes which will be designed to accomodate this wand. These wands will be removed for analysis during the various reactor outages.

2.8 - MTR Phoenix Burnup Calculations

The best estimate of burnup in the MTR is that 50% of the initial ^{239}Pu will be destroyed in the highest flux zone. This is based on the most pessimistic estimate of initial reactivity which would yield a 55-day core life at 40 MW.

2.9 - MTR Core-Thermal and Hydraulic Analysis

A preliminary study of the thermal and hydraulic properties of a Phoenix core in the MTR has been completed. The analysis indicates that a Phoenix core operating at a 29 MW_t reactor power level will approximate the same heat flux, fuel, and clad temperatures, and coolant conditions as a standard MTR core operating at 40 MW_t. A power level of 37 MW_t is within the established operating limits for the MTR when the pressure head and maximum nominal heat flux are increased from 60 psia and $8.2*10^5$ Btu/hr-ft^2 to 73 psia and $1*10^6$ Btu/hr-ft^2, respectively; however, mechanical limitations may prevent such a possibility.

3. PHOENIX FUEL APPLICATIONS

3.1 Marine Propulsion (Consolidated Nuclear Steam Generator - CNSG)

3.1-1 Introduction

The Consolidated Nuclear Steam Generator design which has been considered for a possible Phoenix fuel application is a pressurized water reactor with the steam generator inside the pressure vessel and produces 66,000 shp (176 MW_{th}) for high speed ship propulsion.

In an attempt to find a core geometry more suited to Phoenix fuel, a number of heat transfer limited cores with various pin diameters and fuel volume fractions were simulated. Initial fuel loadings and end of life inventories were obtained from a nuclear analysis survey done by Reactor Physics. The fuel cycle costs for all of the Phoenix cores were then compared to the fuel cycle cost for the enriched uranium core originally designed for the CNSG II.

3.1-2 Conclusion

This survey of core designs for the CNSG application indicates that Phoenix fuel may have an economic advantage when compared with the enriched uranium core of the original CNSG II design.

The geometry of core finally selected would be quite close to the base case uranium because the low pressure drop would be desirable for a low pumping power requirement. Nuclear advantages with reasonable decreases in moderator to fuel ratio were quite small.

3.1-3 Reactor Physics Calculations

The maritime reactor considered was the Babcock and Wilcox (B&W) designed Consolidated Nuclear Steam Generator (CNSG-II).[1] The reactivity, multiplication, and core lifetime characteristics were calculated for various plutonium Phoenix loadings in the CNSG-II as well as for the B&W enriched uranium loading. The results of these calculations were utilized in an economic evaluation of the various plutonium loadings and the enriched uranium loading.

The reactivity and multiplication calculations were performed using the HRG,[3] THERMOS,[4] TEMPEST,[5] and HFN[9] codes while the burnup calculations were done with the ZODIAC(2+2)[19] code. Parameters varied in the study were: enrichment, plutonium composition, moderator-to-fuel volume ratio, and rod size. A brief summary of the results is given here.

The initial reactivity is nearly independent of enrichment, as shown in Figure 3.1-1; however, the core lifetime is essentially proportional to the enrichment as shown in Figure 3.1-2. The effect of increasing the amount of ^{240}Pu in the plutonium is to reduce the initial values of k_∞ and k_{eff}, and to reduce the core lifetime as shown in Figures 3.1-1 and 3.1-2. However, the reduction in core lifetime is caused by the reduction in the fissile atom content (i.e., ^{239}Pu and ^{241}Pu). The system with 10 vol% enriched fuel and a plutonium composition of 43/40/10/7 contains 21% more fissile plutonium atoms than the 6 vol% enriched system with a plutonium composition of 65.5/23.3/7.7/3.5. As seen in Figure 3.1-2, the high ^{240}Pu content fuel has the lowest initial reactivity value and a core lifetime which is comparable to that of the 6 vol% fuel; thus, the plutonium composition is an important consideration in the design of these systems.

The effect of varying the moderator-to-fuel volume ratio and the rod size is relatively small on initial reactivity and core

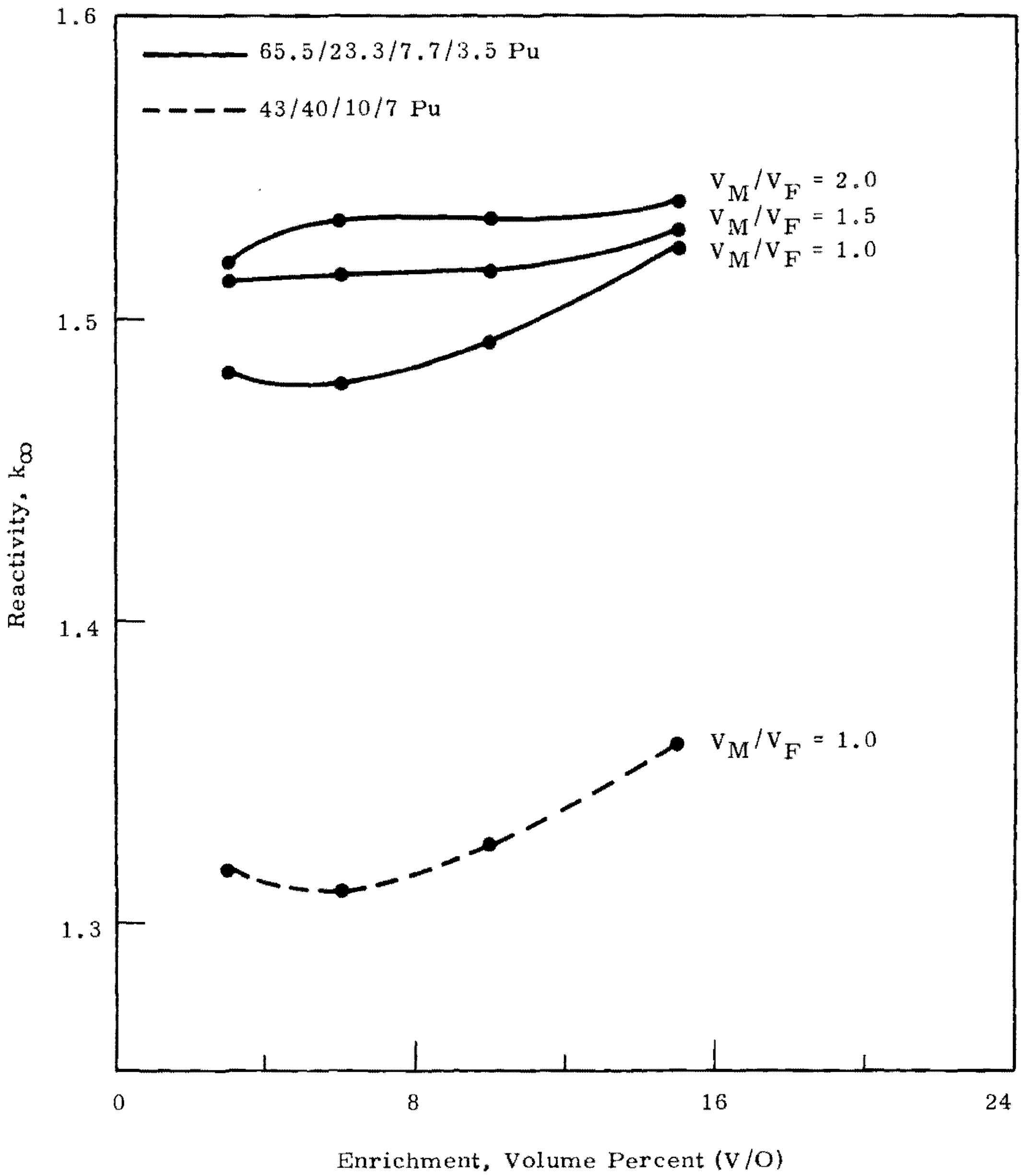


FIGURE 3.1-1

Plutonium Fueled CNSG:Variation of Reactivity with Enrichment for 0.4 in. Diameter Rods

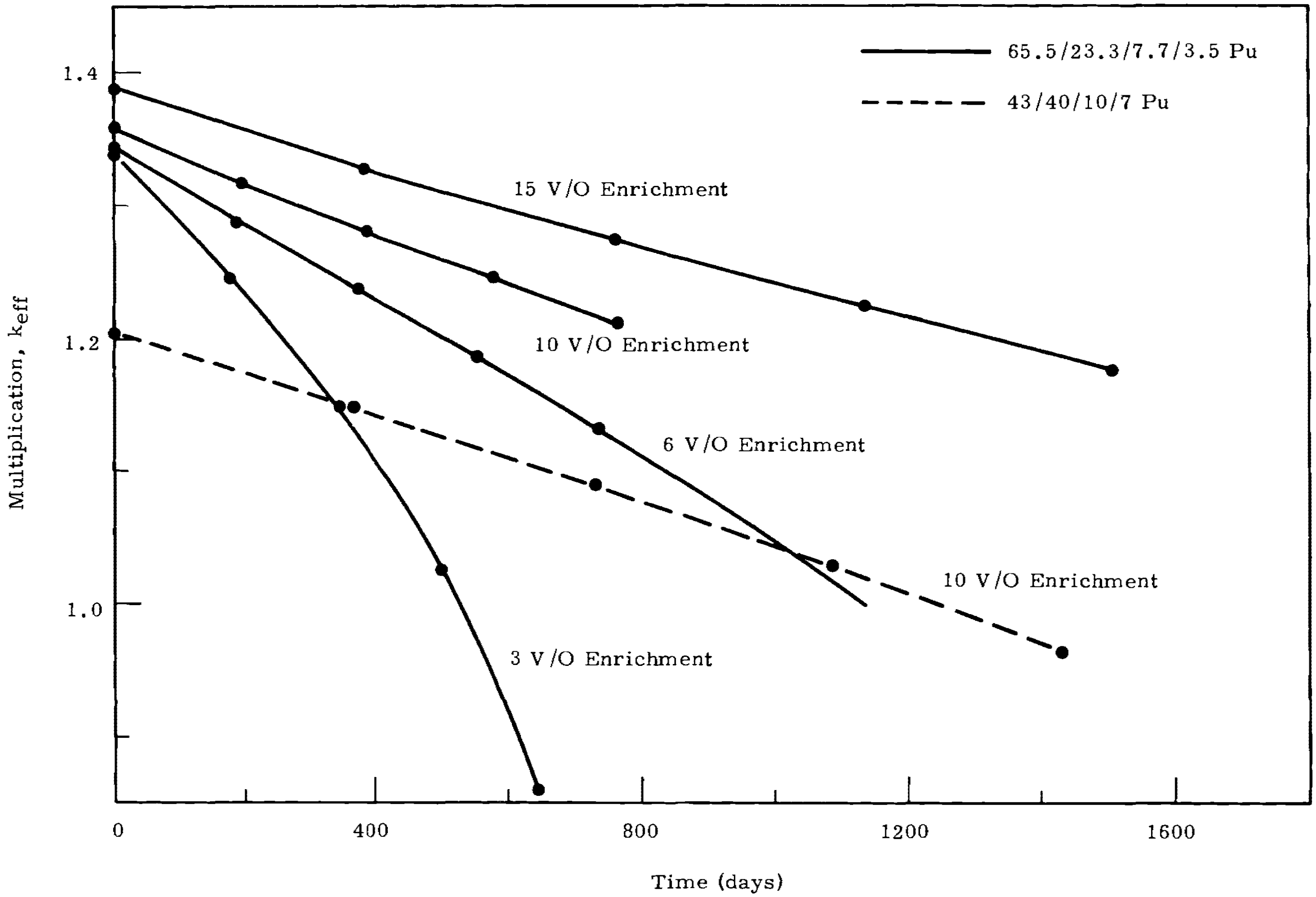


FIGURE 3.1-2

Plutonium Fueled CNSG:Variation of Effective Multiplication with Irradiation for 0.4 in. Diameter Rods at $V_M/V_F = 1.5$

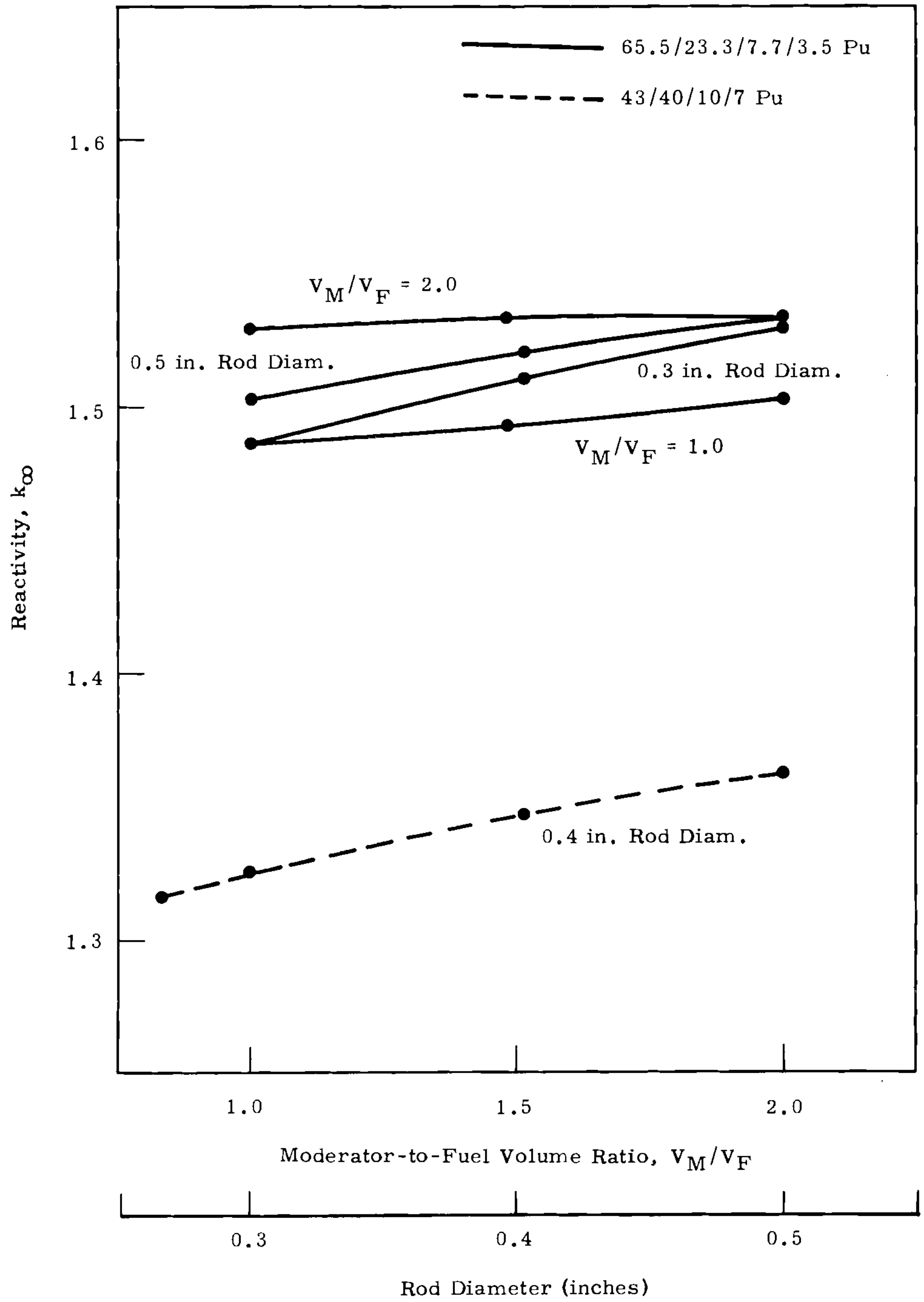


FIGURE 3.1-3

Plutonium Fueled CNSG:Variation of Reactivity with Moderator-to-Fuel Volume Ratio and Rod Diameter for 10 V/0 Enrichment lifetime. The variation in initial reactivity with moderator-to-fuel volume ratio and rod size is shown in Figure 3.1-3. The effect of moderator-to-fuel volume ratio on the core lifetime is shown in Figure 3.1-4 . As the moderator-to-fuel volume ratio is decreased, the amount of plutonium in the system is increased; thus accounting for the increase in core lifetime. The rod size has almost a negligible effect on core lifetime (Figure 3.1-5).

The results of these calculations indicate that the enrichment and the plutonium composition are key parameters to the design of plutonium Phoenix fueled compact H_2O reactors.

3.1-4 Results of Fuel Cycle Cost Analysis

The uranium enriched base case fuel cycle cost was calculated to be 2.9 mils/shp-hr. This compares with 1.84 mils/shp-hr, the cost indicated in the Babcock and Wilcox design summary. The B&W document does not include the assumptions made in computing their fuel cycle cost, our assumptions must be different, causing the discrepancy. The Phoenix cores examined have fuel cycle costs ranging up from 2.6 mils/shp-hr. These Phoenix core costs were calculated using the same assumptions as used in calculating the base case cost. The enriched uranium fuel for the base case was assumed to cost 8 dollars/lb U_3O_8. Plutonium was assumed to be worth 10 dollars/gram fissile.

Two core geometry variables, pin size and fuel volume fraction, were varied to find their effect on the core design for Phoenix fuel.

Increasing the fuel volume fraction decreases the lattice spacing, removing coolant and hardening the spectrum. Increases in the moderator to fuel ratio caused increases in fuel cycle costs.

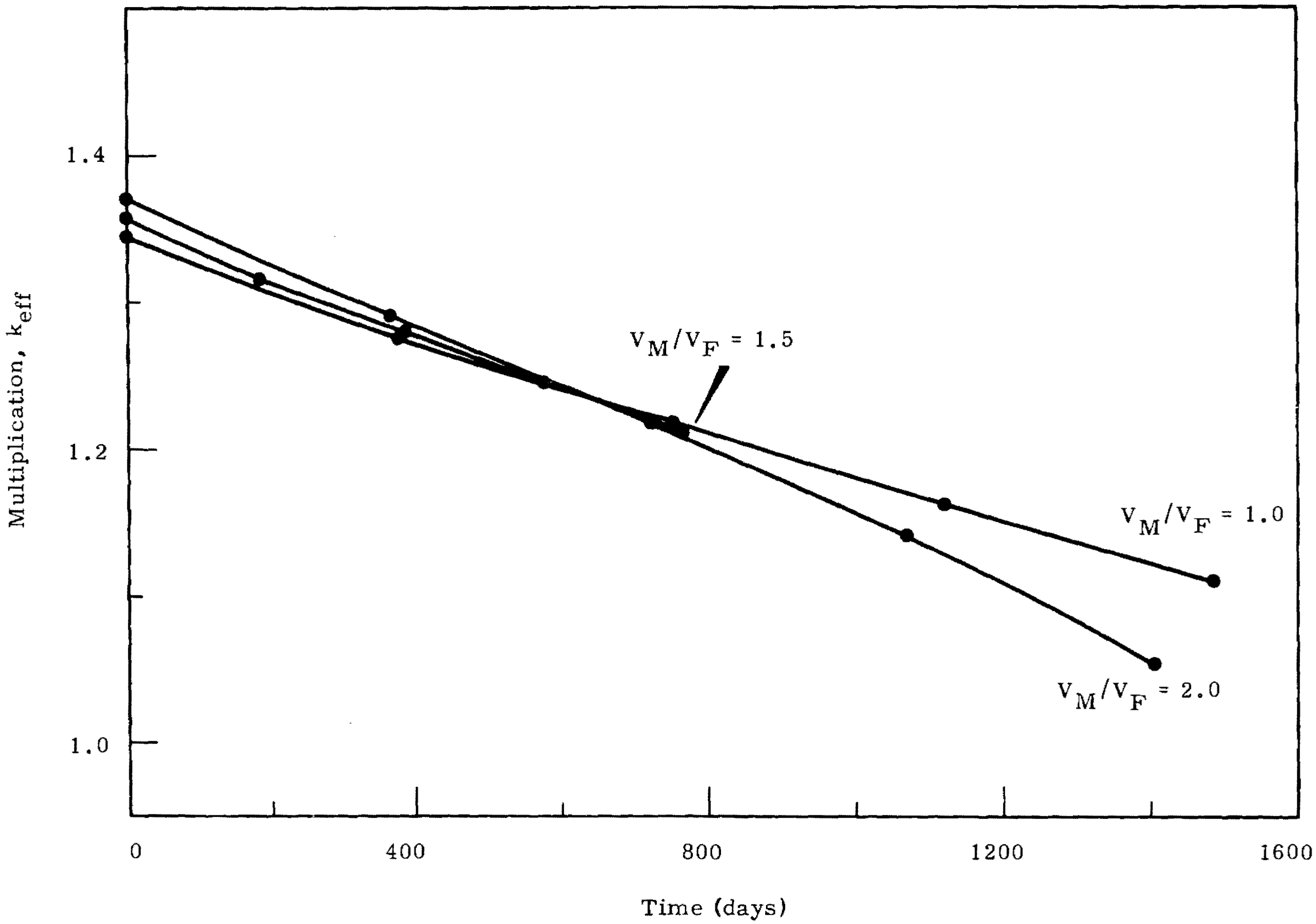


FIGURE 3.1-4

Plutonium Fueled CNSG:Variation of Effective Multiplication with Irradiation for 0.4 in. Diameter Rods with 10 V/O Enrichment

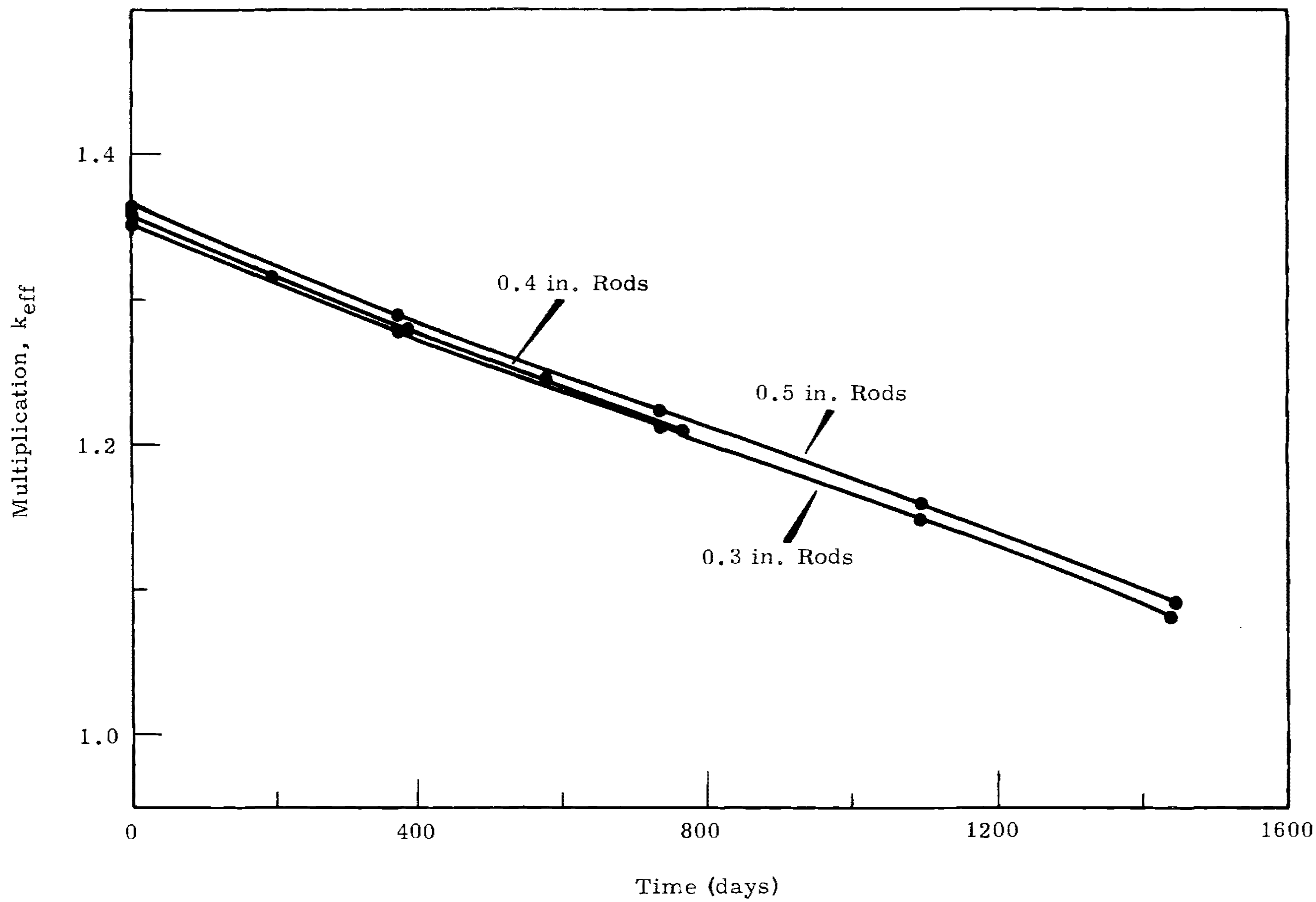


FIGURE 3.1-5

Plutonium Fueled CNSG: Variation of Effective Multiplication with Irradiation for 10 V/O Enrichment at $V_M/V_F = 1.5$

The net mass of fissile atoms used decreases only three percent when the moderator to fuel ratio is decreased from 1.75 to .67. Pressure drop increases considerably with decreasing moderator to fuel ratio and this power requirement, which was not factored into the fuel cycle cost calculations, will tend to keep desirable moderator to fuel ratios high since the effect on nuclear characteristics of going to tighter lattices is minimal.

Smaller pin sizes increased the pressure drop for heat transfer limited cores. Core volume also decreased as pin size decreased. The decrease in core volume due to smaller pins required slightly larger initial loadings of plutonium due to increased leakage. The number of pins which must be fabricated increases with decreasing pin size.

Two plutonium compositions were considered; one having 23.3% ^{240}Pu, the other having 40% ^{240}Pu. Both compositions were assumed to have the same cost, $10 per gram fissile.

3.1-5 Discussion

Plutonium fuel may pay a fabrication cost penalty because it is more difficult to handle. In this study, the penalty was assumed to be 25%. This fabrication cost penalty must be paid for by a decrease in other components of the total power cost in order for the Phoenix fueled core to be economically attractive. In some applications, a decrease in core volume may be worth a considerable amount. For this application, because the steam generator is located inside the pressure vessel, a decrease in core volume is of minor significance. A decrease in core volume also increases pumping requirements and any decrease in core volume would be offset by increased pump size.

The capital cost advantages of smaller core size were assumed to be offset by increasing pump costs. Thus, the assumption made for this study was that the fabrication cost penalty for plutonium fuel must be offset by a decrease in other components of the fuel cycle cost.

The cost of boron control was neglected in the cost calculations. The boron can be mixed with the fuel in the Phoenix cores but must be shielded in separate rods for a uranium core. The plutonium cores using plutonium with 23.3% ^{240}Pu and the clean uranium core had initials k's of ~1.30. The lower initial k for the cores with plutonium with 40% ^{240}Pu will require less control, and the boron can be added directly to the fuel mixture.

The first look at this application for Phoenix fuel shows it to be comparable in cost to the enriched uranium core. The survey portion of the study has shown that changes in moderator to fuel ratio are not warranted in pressurized water reactors. Plutonium composition does appear to be an important variable and some gain in cost may be made by using different plutonium compositions. Since only two compositions were considered an extension of this study might include other fuel compositions.

The code used to calculate the fuel cycle costs needed some changes which made a more complete report impossible at this time. The problems in the code are being corrected and the fuel cycle cost analysis will be completed.

3.2 Physics Characteristics of Plutonium Phoenix Fueled Loadings in PRTR

Previously the systems studies for possible Phoenix fuel application have been light water (H_2O) reactor systems. A recent study[2] on the use of Pu fuel in heavy water (D_2O) reactor systems shows that a "Phoenix effect" is attainable in a pressure tube type D_2O reactor. To aid in scoping applications of Phoenix fuel, a study was made to determine the physics characteristics of this type of fuel in a pressure tube type D_2O reactor, namely the Plutonium Recycle Test Reactor (PRTR). The results of these calculations are given in this report.

3.2-1 Analysis Methods

The fuel was assumed to be nineteen (19) rod clusters of aluminum-plutonium (Al-Pu) alloy, eighty eight (88) inches in length. The isotopic composition of the plutonium was taken as 65.5/23.3/7.7/3.5 percent of $^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu$, respectively. The operating temperatures assumed for the plutonium Phoenix fuel loadings in the PRTR were 58°C for the moderator, 260°C for the coolant, and 377°C for the fuel.

The reaction rates and average cross sections for a representative lattice cell of the PRTR were computed using the HRG[3], THERMOS[4], and TEMPEST[5] codes. The techniques utilized in mocking up the 19-rod cluster are identical to those used in a previous analysis[6]. The spectrum averaged cross sections obtained are used in a calculation of the reactivity k_∞ and k_{eff}. Isotopic transmutations are computed using the code ALCHEMY[8]. The chain of codes used for the burnup calculations, HRG, TEMPEST, HFN, and ALCHEMY, is known as ZODIAC (2+2).[9]

3.2-2 Calculational Results - Lattice Characteristics

The calculated reactivity variation with fuel enrichment is shown in Figure 3.2-1. The reactivity is fairly constant for

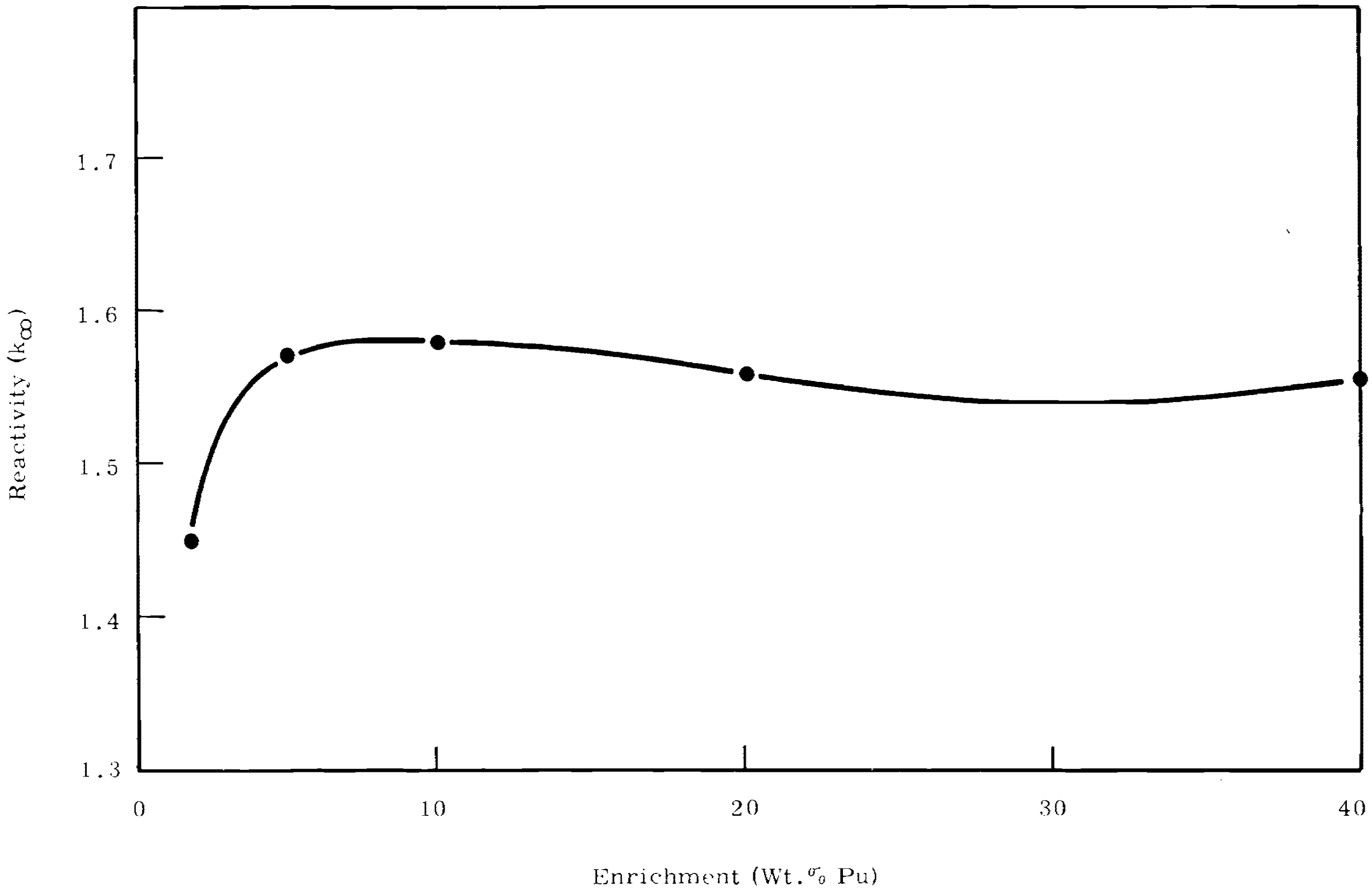


FIGURE 3.2-1

Plutonium Fueled PRTR: Variation of Reactivity with Enrichment at Operating Temperatures enrichments in excess of 4-5 wt% Pu. The largest value of k_∞ is obtained with an enrichment of 10 wt%; thus , in principle, a fuel with an enrichment in excess of 10 wt% Pu will gain reactivity with the depletion of plutonium atoms (i.e., k_∞ increases with burnup) for this system.

The effective multiplication, k_{eff}, of a 55-element loading in the PRTR has been calculated for the same range of fuel enrichment. The values are plotted in Figure 3.2-2, along with values of k_∞. The difference in shape of the curves indicates the variation of neutron leakage with enrichment for these loadings. The leakage is seen to be concentration dependent, the ratio k_∞/k_{eff} varying from 1.10 to 1.20 over the range 40 to 1.8 wt% Pu, respectively.

The curve shows k_{eff} increases with enrichment up to ~15 wt% Pu, decreases between 15 and 30 wt% and then increases again with increasing enrichment. The increase in k_{eff} with decreasing enrichment at around 20 wt% Pu is somewhat smaller than the increase in k_∞. Thus, the merit of the principle stated in the preceeding paragraph is diluted to some extent.

Criticality calculations were performed to determine the critical number of elements for each enrichment considered. Figure 3.2-3 shows the variation of critical radius with enrichment. The area of a unit cell of the PRTR, which is used to convert from critical radius to numbers of elements, is 357.6 cm^2. The critical number of elements given in Figure 3.2-3 are all expected to be underestimates of the actual number required because of assumptions made in the calculations. For example, the effects of various neutron absorbers such as process and shroud tubes in non-fueled channels have been neglected.

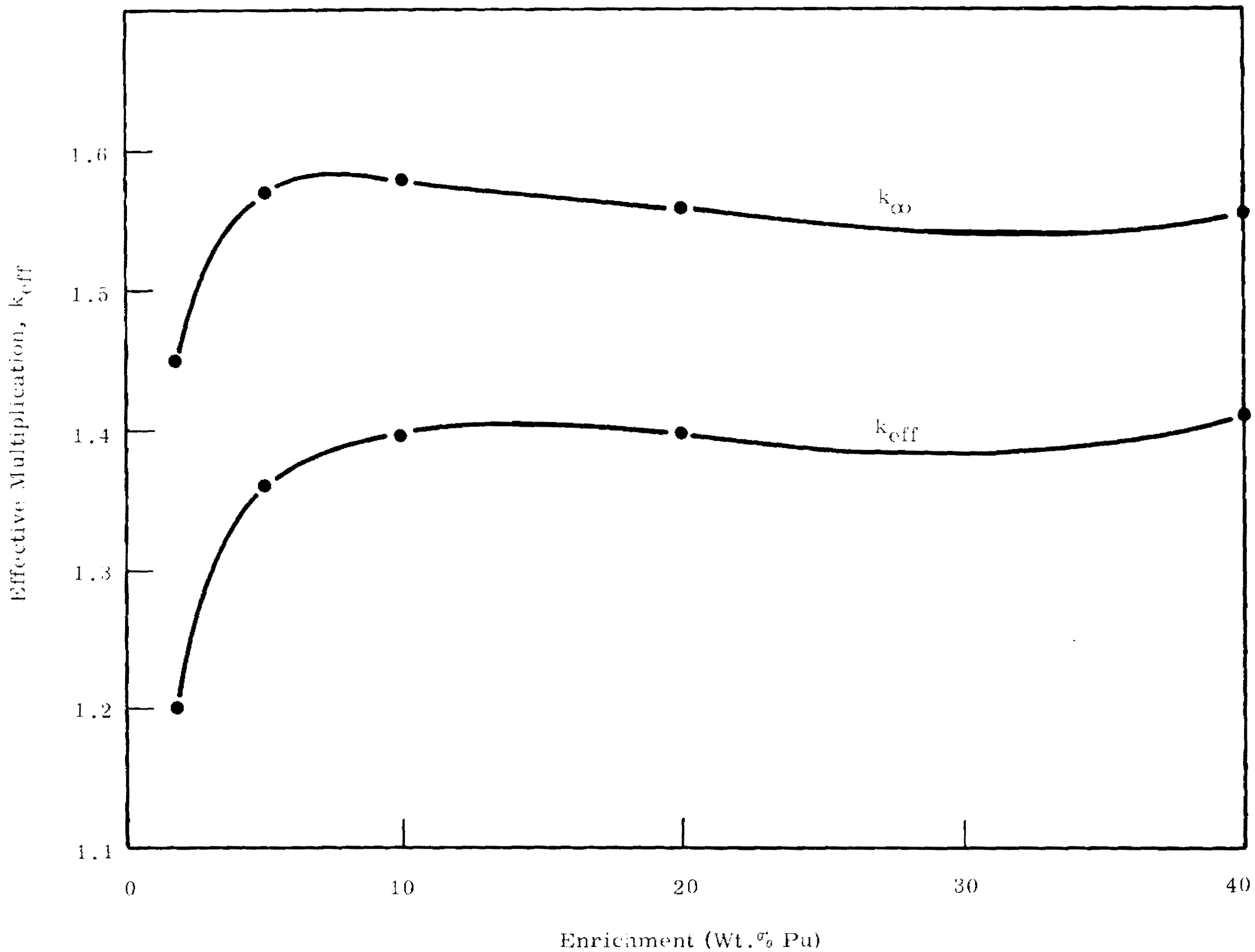


FIGURE 3.2-2
Variation of the Effective Multiplication with Enrichment for a 55 Element Loading in PRTR at Operating Temperatures

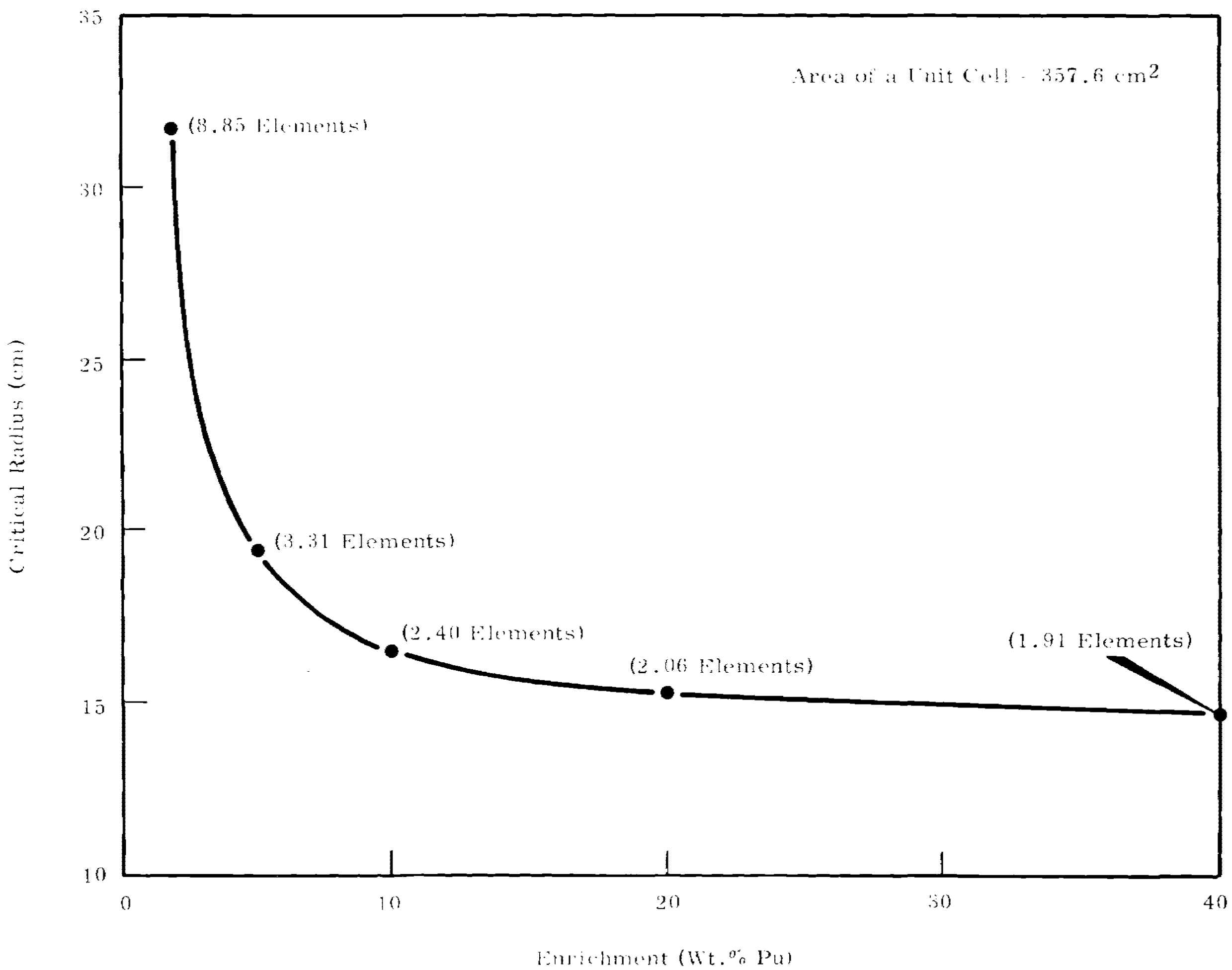


FIGURE 3.2-3

Critical Radius versus Enrichment for PRTR at Operating Temperatures

The moderator, coolant, and fuel temperature coefficients of reactivity were calculated for a Al-20 wt% Pu fueled cell. The values are given in Table 3.2-1. The coefficients are all negative in sign and relatively small. The reactivity invested in the sum of moderator, coolant, and fuel heating is less than 1% $\Delta k/k$.

A sizeable value of excess reactivity, k_{ex}, is shown (Figure 3.2-2, $k_{eff}=1.0$) for 55 element loadings in the PRTR. Some of this excess reactivity would be required to compensate for saturation fission products. The rest could be controlled by use of soluble poison in the moderator. A calculation was performed to determine the amount of boron required to control the excess reactivity of a 55-element loading of Al-20 wt% Pu elements. The result of the calculation shows that about 400 atoms of natural boron per million molecules of D_2O are required to control the 400 mk of excess reactivity. Thus, a worth of the boron is roughly

$$\frac{1\ \text{mk}}{\text{atom of natural boron per million molecules of } D_2O}$$

3.2-3 Calculational Results - Burnup

A core lifetime calculation was performed for the 55 element loading of Al-20 wt% fuel in the PRTR. The reactivity loss rate with irradiation time is shown in Figure 3.2-4. The calculated lifetime of this core is 100,000 megawatt days (MWd). Reducing the length of the elements from 88 inches to 60 inches (the length of Batch Core fuel[10]) results in a 35-40% reduction in core lifetime.

A burnup calculation was performed for a two-region core containing 19 irradiated Batch Core fuel elements (UO_2-PuO_2)[6] surrounded by 66 Al-10 wt% Pu fuel elements. The UO_2-PuO_2 fuel was assumed to have an exposure of 13,000 megawatt days per metric ton of mixed oxide

TABLE 3.2-I

AVERAGE TEMPERATURE COEFFICIENTS OF REACTIVITY
FOR THE Al-20 wt.% Pu FUEL

Type of Coefficient	Temperature Range (°C)	Ave. Temp. Coeff. $\frac{1}{k_\infty}\frac{dk_\infty}{dT}$ $(10^{-3}/°C)$
Moderator	20-58	- 0.123
Coolant	20-260	- 0.003
Fuel	20-377	- 0.007

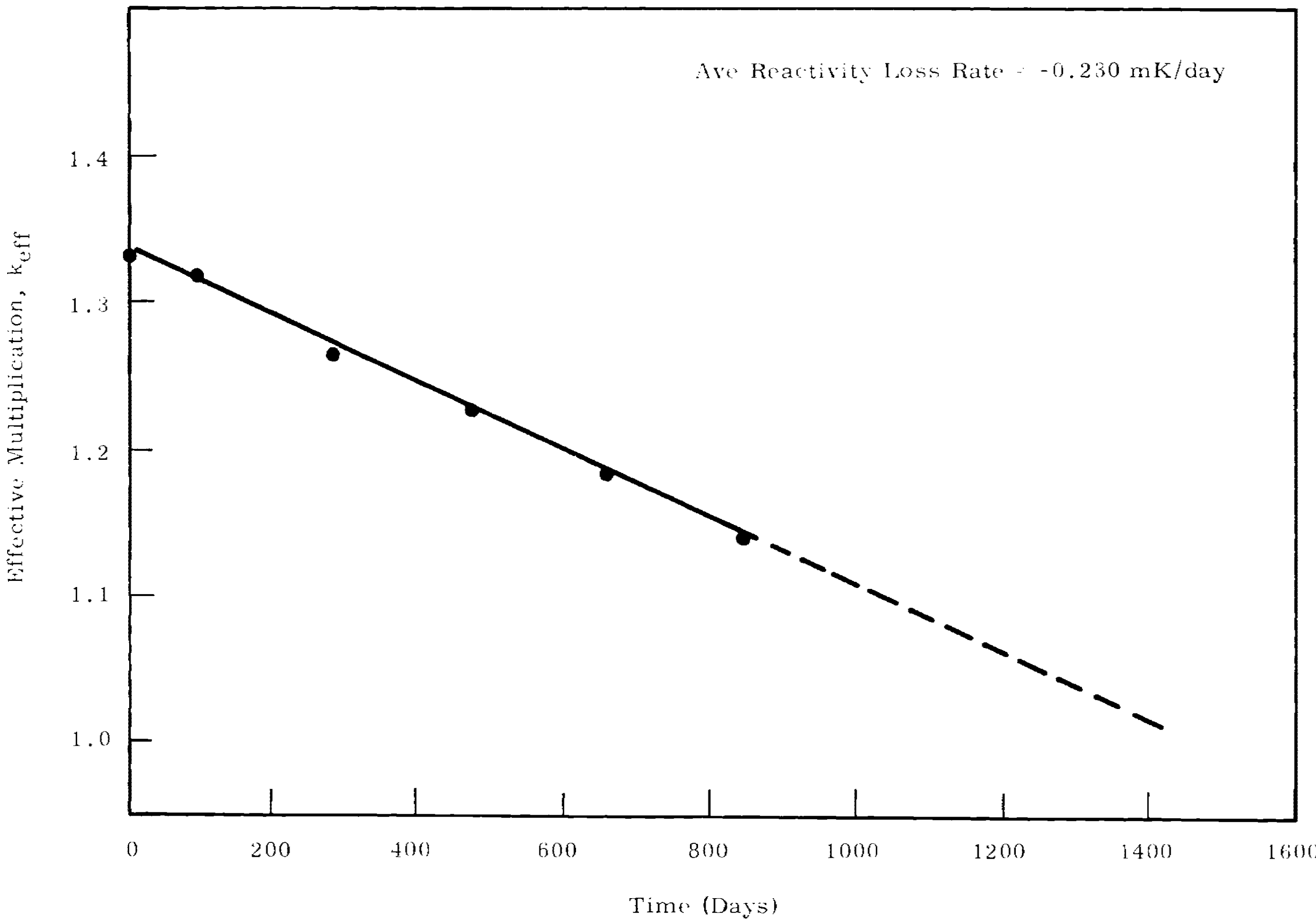


FIGURE 3.2-4

Variation of Reactivity with Exposure for 55 PRTR Elements of Al-20 Wt.% Pu
at a Power Level of 70 MW

(MWd/tonne). The calculated reactivity loss with irradiation for this core is shown in Figure 3.2-5. The core lifetime was calculated to be 28,000 MWd and an additional 7,000 MWd/tonne average expsoure is obtained on the UO_2-PuO_2 fuel. Utilizing an Al-Pu fuel containing more plutonium (i.e., > 10 wt%) for this purpose would result in a slight decrease in the power density of the UO_2-PuO_2 zone, but the core lifetime would be increased.

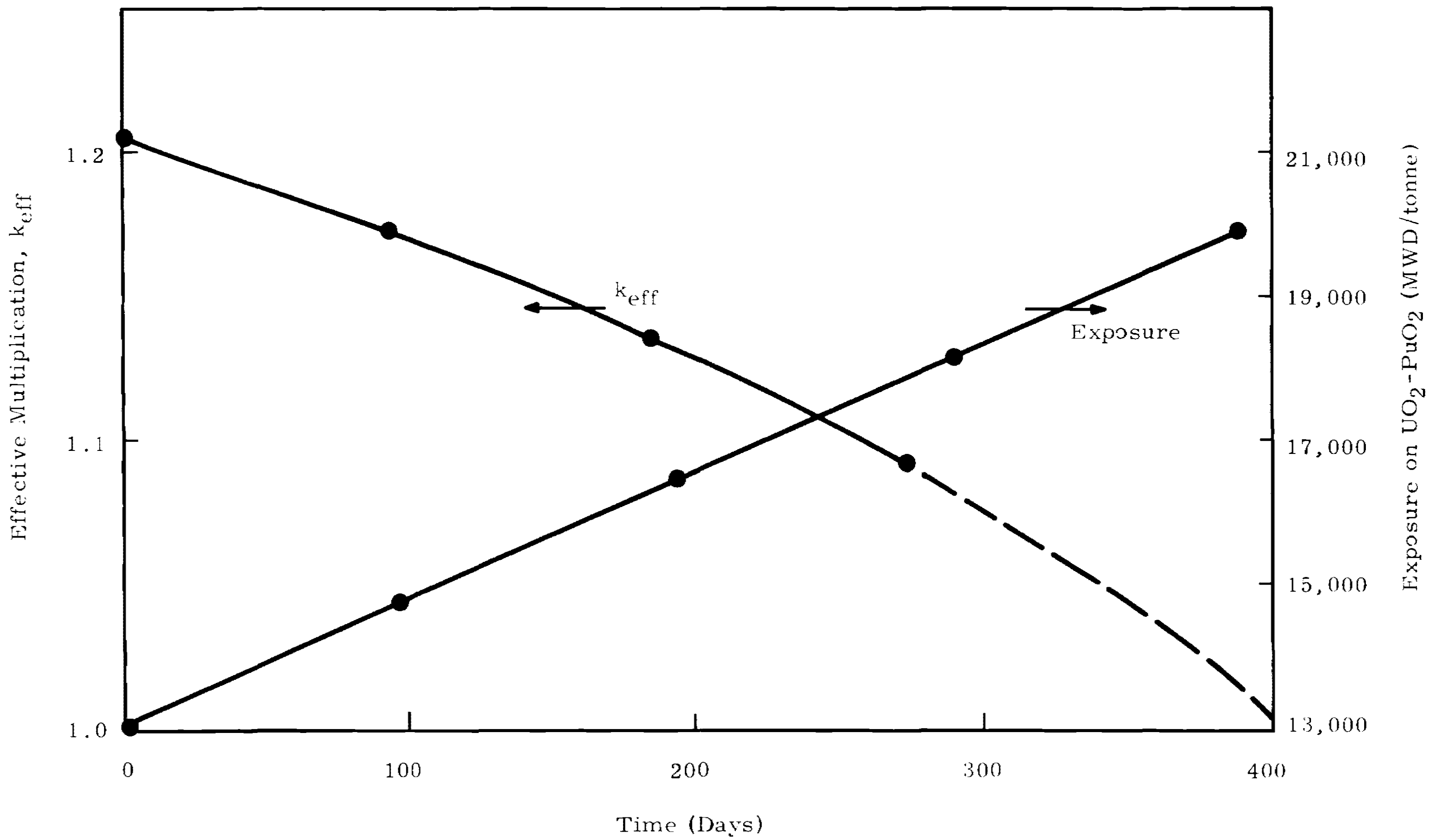


FIGURE 3.2-5

Phoenix Fuel Drivers for the PRTR Batch Core

3.3 Research and Test Reactors

3.3-1 HFIR - First Approximate Calculations

The High Flux Isotope Reactor (HFIR) (Figure 3.3-1) seems to be potentially well suited for utilization of Phoenix fuels. The reasons behind this conclusion are as follows: First, the fuel is very expensive and has a rather short (23 day) operational lifetime; therefore, the Phoenix fuel may give a cost saving by extending the core reactivity lifetime. Second, the fuel is very complex (Figure 3.3-2); fuel shaping and local flux suppressors are already used in the standard core. The additional cost to modify these items to be optimum for Phoenix core should be small.

These initial considerations led to a series of rather crude calculations to test out the use of Phoenix fuel in the HFIR. An approximate geometrical model was selected and a standard HFIR core was calculated using the ZODIAC code augmented by THERMOS for thermal energies. For 23 full power days a $\Delta\rho$ of 0.063 was obtained, see Figure 3.3-3, which is very close to the expected value of 0.059 as given in ORNL-3572. The same methods were used to calculate the required control margins for two Phoenix loadings. The first was an atom for atom replacement of ^{235}U with Pu of an isotopic mixture 60:24:13:3 for ^{239}Pu, ^{240}Pu : ^{241}Pu : ^{242}Pu. The second was a double load of Pu. Burnable poisons were not used in either Pu run. Figure 3.3-4 shows that an atom for atom replacement results in about the same exposure while the double Pu load might give as much as 60 percent more fuel lifetime from a reactivity limitation standpoint.

An additional advantage of Phoenix fuel for the HFIR is the expected increase in total flux level for comparable power densities.

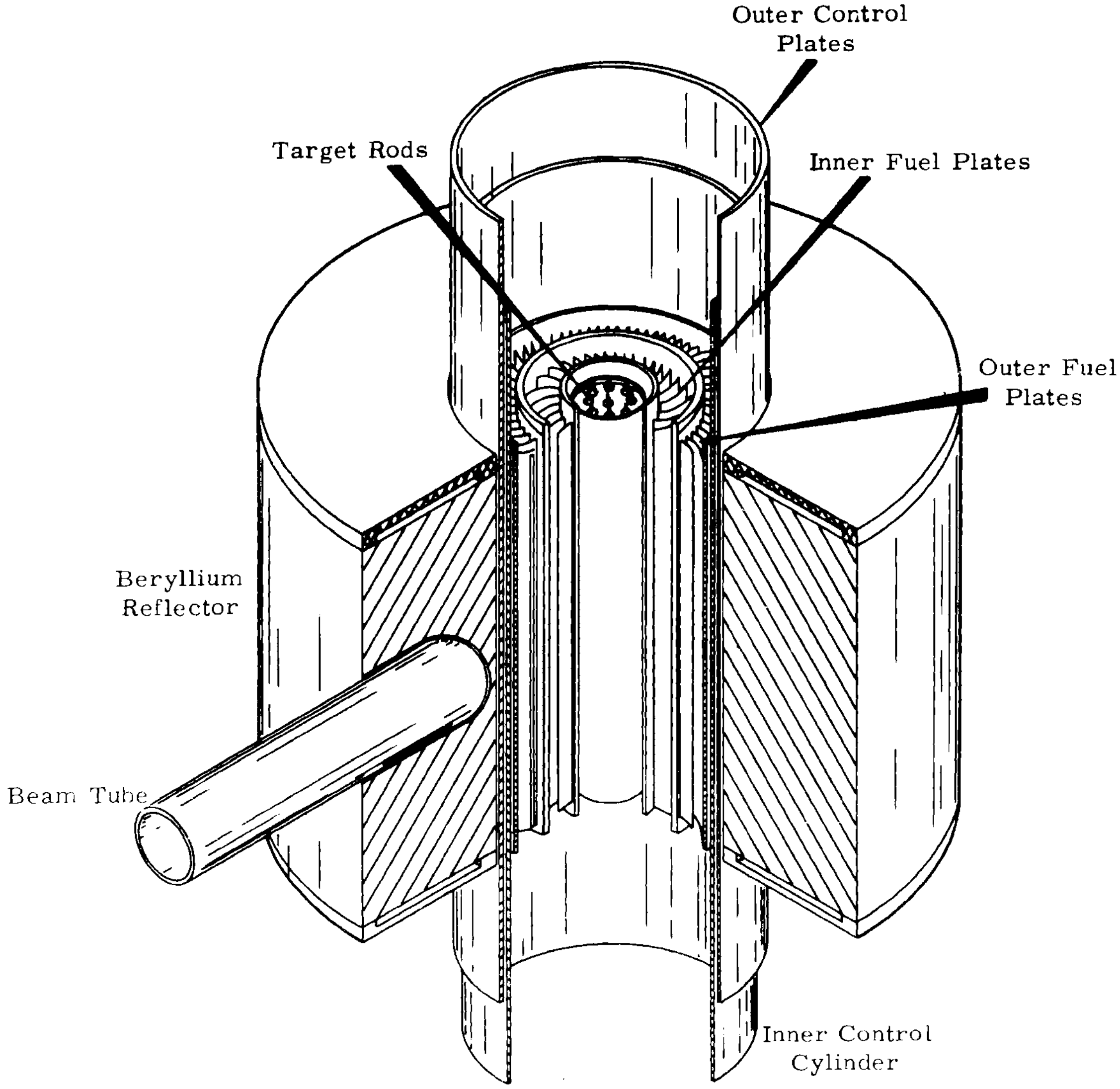


FIGURE 3.3-1
HFIR Core

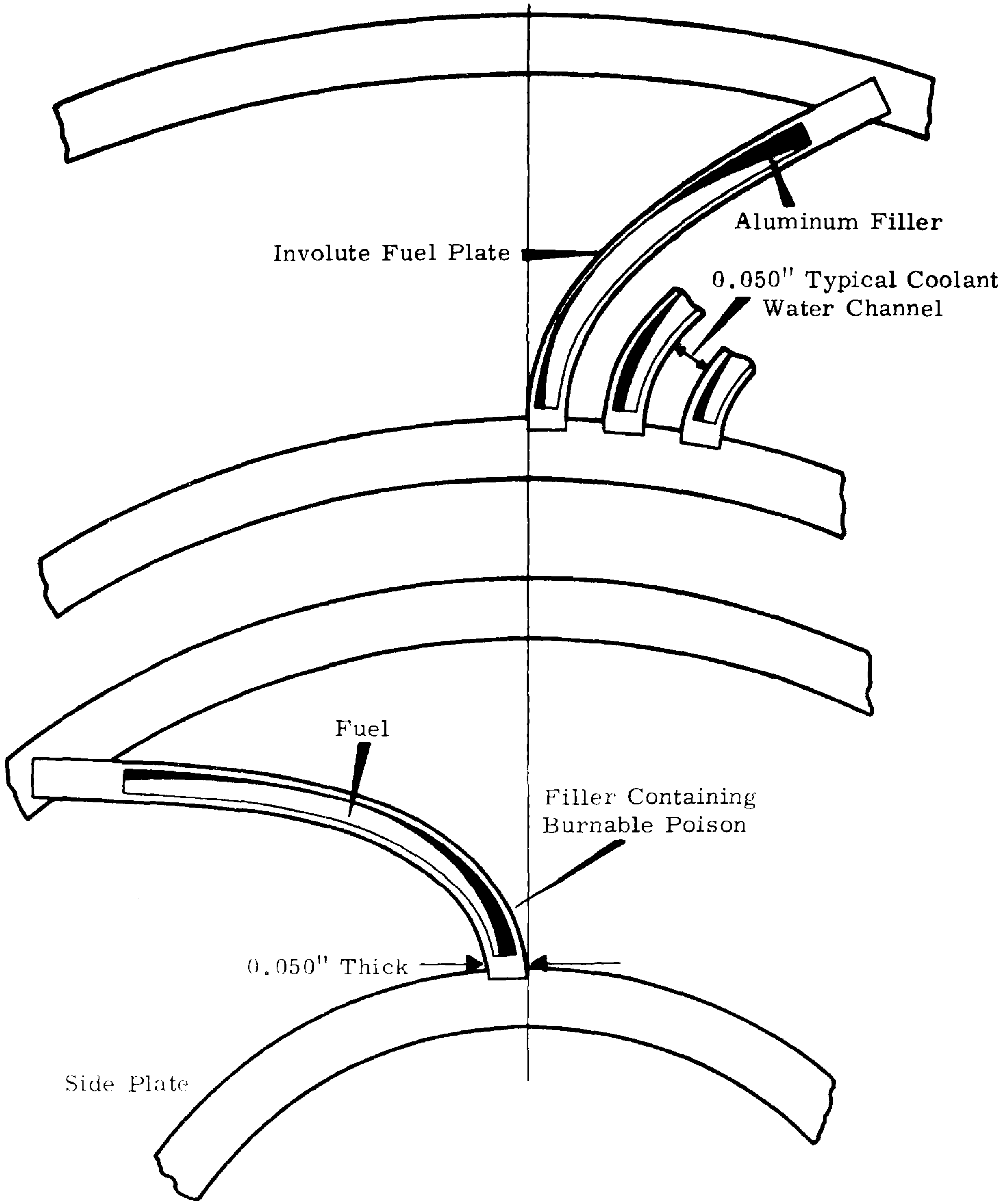


FIGURE 3.3-2
HFIR Fuel Plates

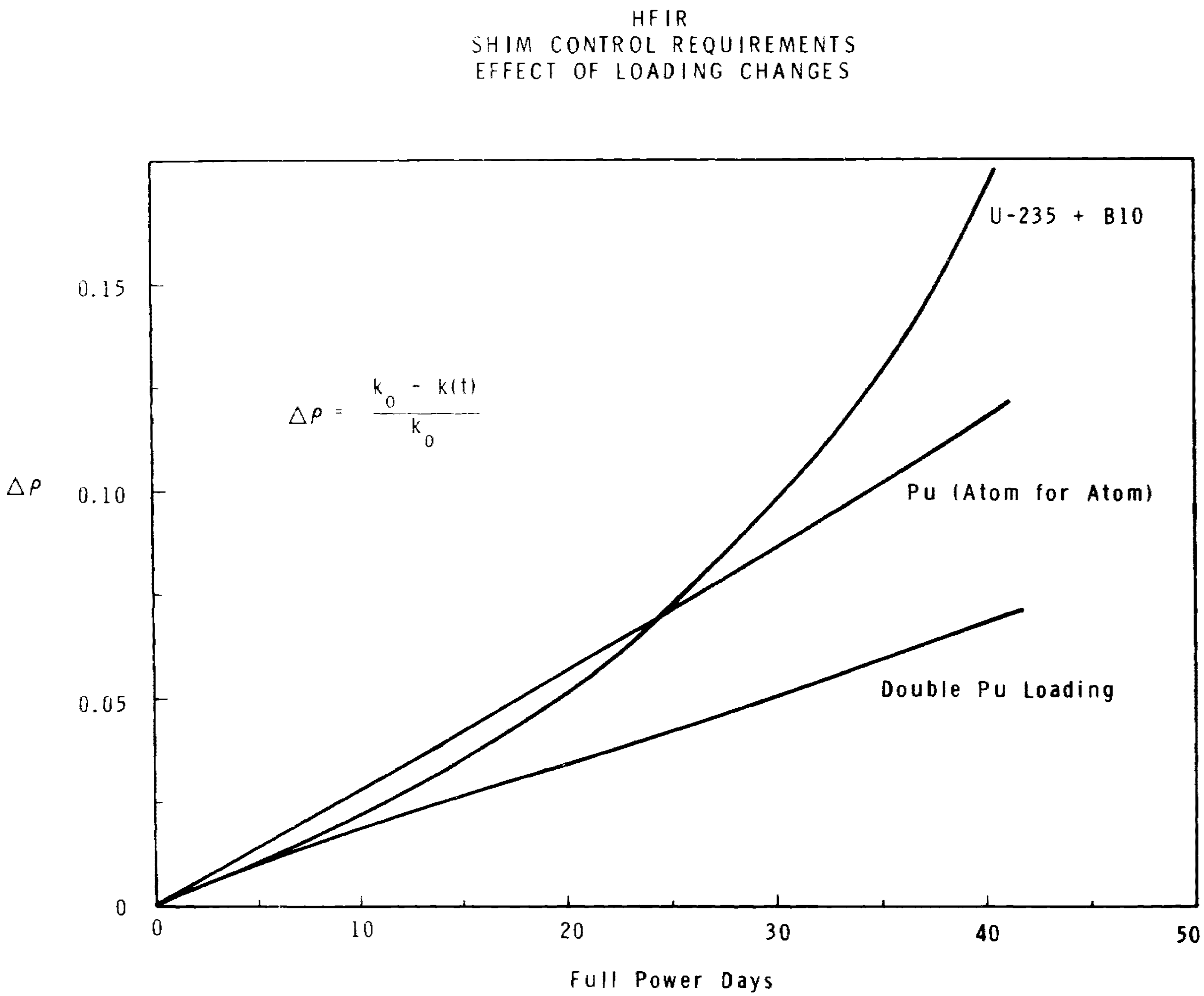


FIGURE 3.3-3
HFIR Shim Control Requirements Effect of Loading Changes

The increase comes about because of the greater yield of fission neutrons in a Pu system. The Pu to U thermal fission yield is 2.89 to 2.43 or 1.19 for constant fission density. This ratio is, effectively reduced by the greater energy liberation of Pu fission, that is 211 Mev/fission for Pu versus 200 Mev/fission for ^{235}U. At a constant power density, the overall flux ratio is 1.13 in favor of the Pu system. It is apparent from these calculations that the best Pu loading will be determined by a compromise between long core life and high neutron density in the flux trap.

3.3-2 HFIR - Improved Calculations

The results of the approximate calculations were discussed with both ORNL and ANL personnel and the flux distributions were questioned. The concensus of opinion was that the geometrical model was not detailed enough to show the proper trends. ORNL subsequently provided the calculational geometry which they have employed.

It should also be noted that the reactivity lifetime is not the only present limiting factor for HFIR cores. If means of improving the reactivity lifetime are to be used it will be necessary to provide a fuel with improved radiation damage and corrosion limits as well.

The work now underway is first to recalculate the ^{235}U B-10 system then substitute Pu and check the flux distribution. At present, the improved calculations include both ^{235}U + ^{10}B cases and the so-called double Pu without ^{10}B. Table 3.3-I shows the comparison of the ^{235}U fuel and the Pu fuel for a consistent set of calculations; comparison of this ^{235}U calculation with experiment has not yet been made. Recent contact with ORNL indicates that the diffusion coefficients (D) used in their calculations yield a good power distribution as compared to experiment; however, their value is neither the D given by the computer code HAMMER nor the special D used in the PNL calculations shown in Table 3.3-I. The special values of D chosen for the PNL calculations are based on a standard averaging system and have been found to give better agreement with experiment in the PRCF-Phoenix experiment than the D evaluated in the HAMMER code. The double Pu case has a maximum peak-to-average power of 2.79 compared to 1.26 for the ^{235}U case. A total of 7.6% of the Pu was removed in reshaping the fuel plate tips with a resulting maximum peak-to-average power of 1.50. This shaping can be further optimized.

The results in Table 3.3-I indicate about a 20% gain in thermal flux and 25% gain in total flux at the flux trap center when Pu fuel is used. Burnup calculations have not yet been carried out.

TABLE 3.3-I

HFIR Improved Calculation Flux Levels
(Flux in Units of 10^{15} Neutrons/cm^2 sec)

	Thermal Flux Fuel Average	Thermal Flux Trap Center	Total Flux Fuel Average	Total Flux Trap Center	Peak-to-Average Power
^{235}U and ^{10}B Special D	0.355	2.38	2.53	3.66	1.26
$^{235}U \rightarrow$ 2Pu No ^{10}B, Special D	0.0818	2.75	2.58	4.48	2.79
2 Pu Power Flattened, No ^{10}B, Special D	0.117	2.88	2.70	4.57	1.50

3.3-3 Consideration of HFIR Phoenix Fuel Fabrication

The HFIR fuel element is a plate fuel consisting of an aluminum-clad dispersion of 30 and 40 w/o enriched U_3O_8 in an aluminum matrix. The plate fabrication process is basically a roll-cladding process similar to the MTR, ETR, and ATR fuel processes; however, the HFIR fuel distribution, plate fabrication, assembly and inspection are considerably more complicated.

In applying the Phoenix concept to the HFIR fuel, three problems are encountered: 1) The substitution of plutonium for enriched uranium, 2) Increasing the fuel content, and 3) Increased radiation damage. Increasing the fuel content should prove the most difficult problem. At 41 w/o U_3O_8, the cermet consists of 19 v/o U_3O_8 and 81 v/o Al matrix. For the longer burnup Phoenix elements, an increase up to double the quantity of fuel is desirable. The increase in the fuel content decreases the matrix content and, because the matrix is the ductile phase, fabrication becomes more difficult by conventional roll-cladding techniques. In general, a cermet fuel implies a continuous metalic matrix and requires a minimum matrix content of 50 to 60 volume percent. In the following table, the weight content for several candidate plutonium fuel materials at matrix contents of 50 and 60 volume percent is shown. The higher density compounds have the advantage of greater fuel contents with less reduction in the matrix volume. Exceeding the fuel contents in the table may require the adoption of ceramic fabrication techniques as opposed to metallic techniques.

FUEL CONTENTS OF PLUTONIUM COMPOUNDS IN AN ALUMINUM MATRIX

Density of Plutonium Compound	Matrix Volume 50%	Matrix Volume 60%
14 (PuO, PuN, PuC, or delta stabilized Pu metal)	84 wt%	77 wt%
11 (PuO_2)	80 wt%	73 wt%
8 ($PuAl_2$)	75 wt%	67 wt%
7 ($PuAl_3$)	72 wt%	64 wt%
6 ($PuAl_4$)	69 wt%	60 wt%

From the irradiation damage standpoint, Idaho Nuclear has shown that the uranium-aluminum intermetallic UAl_3 is superior by a factor of 5 to U_3O_8, and the dispersion of UAl_3 in aluminum is now the reference fuel for the ATR and ETR. The plutonium-aluminum alloy system is nearly identical to the uranium-aluminum system, and the intermetallic compounds $PuAl_2$, $PuAl_3$, and $PuAl_4$ are formed. If the plutonium intermetallics are as resistant to irradiation damage as UAl, the higher burnups required in a Phoenix-HFIR fuel should be achievable with a Pu-Al intermetallic fuel. With the other plutonium compounds irradiation damage may be a problem, and compatibility problems may occur between the fuel and the matrix, the coolant, or both.

The substitution of plutonium, per se, for enriched uranium in the HFIR fuel should present no significant fabrication problem. The major increase in fuel cost will occur in the preparation of the plutonium compounds. PuO_2 would have a definite cost advantage over the other plutonium compounds in this area. The plate fabrication process for the plutonium fuels will be slightly more expensive, but the assembly and inspection costs should be about the same.

The high exposure plutonium required for Phoenix fuels will result in levels of radiation similar to those experienced with the development of the MTR Phoenix fuel (see Section 4.3). However, close control and minor shielding can be used to reduce radiation to a minor problem.

PNL has done some development work with roll bonded zirconium-clad plutonium zirconium alloy plate fuels. From a cost standpoint this fabrication process looked attractive and may be of some interest for HFIR applications.

In summary, it looks like 70 w/o $PuAl_2$ dispersed in aluminum is the best candidate for a Phoenix HFIR fuel fabricated by the present techniques. Irradiation damage may not be a problem if the extrapolation of UAl_3 experience is applicable to 1) $PuAl_2$ and 2) the higher fuel content. Fabrication cost should not be prohibitive, because a large fraction of the HFIR cost is probably incurred in the assembly and inspection steps which should be independent of both the fuel material and fuel content. From a cost standpoint only, a 70 w/o PuO_2-Al dispersion would be the best candidate. A metallic plutonium base alloy clad with zirconium or aluminum is also attractive from the fabrication cost standpoint.

3.4 - General Compact Reactor Systems

3.4-1: Characteristics Available

The major benefits that might be obtainable in plutonium fueled burners, as compared to ^{235}U systems, are:

1. More favorable reactivity-life characteristics
2. Potentially simpler, more compact core designs
3. More favorable temperature coefficient characteristics
4. Potential cost savings

The favorable life characteristics stem from the exploitation of the ^{240}Pu fertility. The burnout of ^{240}Pu not only results in the fissile ^{241}Pu isotope, but in addition, the ^{240}Pu burnout itself is of benefit to the reactivity lifetime characteristics of the reactor. Burnable poisons can also be used in plutonium cores, as they are in long endurance, fully enriched ^{235}U cores. Since plutonium has large cross sections, high cross section burnable poisons can be used in intimate contact with the fuel and without the need of extensive poison "lumping." (In ^{235}U cores, self-shielding of burnable poisons, particularly B-10, is often a necessity.) The resulting design simplification can lead to potentially more reliable, more compact core designs.

Since ^{240}Pu is a resonance absorber, its use leads to a negative Doppler coefficient. For some core designs this feature can be of importance.

The current price of Pu is $10/gm fissile compared to $12/gm fissile for ^{235}U . The price of ^{235}U in the lower or intermediate enrichment range is much lower while the Pu cost is fixed. This tends to make Pu fueling more attractive in the very high enrichment range and less attractive in cases of low enrichment requirements.

Small power sources (i.e., 1 to 30 Mw_e) are a possible application if compactness and long life are important. Such cases might be life support systems for both oceanographic and space work. The initial placement of the power system is expensive in both cases and is size dependent enough to require a compact reactor. Refueling is difficult or impossible which places long life at a premium for continually inhabited situations.

If small size is of real premium, such as in a deep submersable or space craft, it should be possible to design a smaller equivalent lifetime core using Pu rather than ^{235}U.

4. MTR-PHOENIX FUEL BURNUP EXPERIMENT

4.1 CAF-Phoenix Experiments

4.1-1: Measurements

Approach-to-critical experiments with a hydrogen-moderated Al-20 wt% Pu-fueled system have been conducted(11) in the Critical Approach Facility (CAF).(12) The experimental results have been used to check the computational methods currently being applied to Phoenix fuel cores. The results of this work allow recommendations to be made for calculational models to be used for small, strongly-reflected Phoenix fuel cores.

The fuel was in the form of disks of Al-20.3 wt% Pu alloy. The isotopic composition of the plutonium was 90.48% ^{239}Pu, 8.19% ^{240}Pu, 1.25% ^{241}Pu, and 0.08% ^{242}Pu. Two fuel disks, 0.020 inches thick and 1.96 inches in diameter, were placed between two 0.020 inch thick aluminum disks. Polyethylene disks 0.060 inches thick were placed on each side of the aluminum. Thus, a cell was formed which was 0.20 inches thick and contained 1.24 ± 0.01g Pu. The fuel cells were stacked in polyethylene tubes and sealed in aluminum cans 2-1/4" O.D. with 0.125 inch wall thickness to form fuel columns. A maximum of 30 inches of fuel (145 fuel cells) could have been loaded in a tube. For shorter fuel columns, the remainder of the tube was filled with polyethylene disks.

The array consisted of 19 fuel columns in a hexagonal lattice of 2-1/4" pitch. This array was reflected in the radial direction with a 2-inch layer of beryllium and in the axial direction with H_2O. Light water filled the interstices between the cans and surrounded the beryllium reflector.

Experiments were conducted with a clean core and with the contents of the center fuel column replaced by a poison column (2 inches O.D. by 0.040 inches cadmium tube filled with polyethylene). Each experiment was comprised of radial approaches-to-critical to the 19-can array, with successive experiments using greater fuel heights. The end result of each radial approach was a point on an axial approach curve to determine the critical height of the 19-columm array. The results of the measurements are presented in Table 4.1-I. The listed uncertainties are one standard deviation from the extrapolated critical size, based on a linear least-squares fit to the inverse multiplication data. The data, $(\text{count rate})^{-1}$ are multiplied by N, the number of fuel cans, which accounts for the increased neutron source from the fuel, and a cylindrical shell correction, $I_o(A\sqrt{N})$, a modified Bessel function of the first kind, which accounts for the varying thickness of moderator between the core and the detectors. The constant, A, is chosen such that the expression

$$\frac{N\ I_o(A\sqrt{N})}{CR}$$

is a linear function of N. Examples of the adjusted data are shown in Figure 4.1-1.

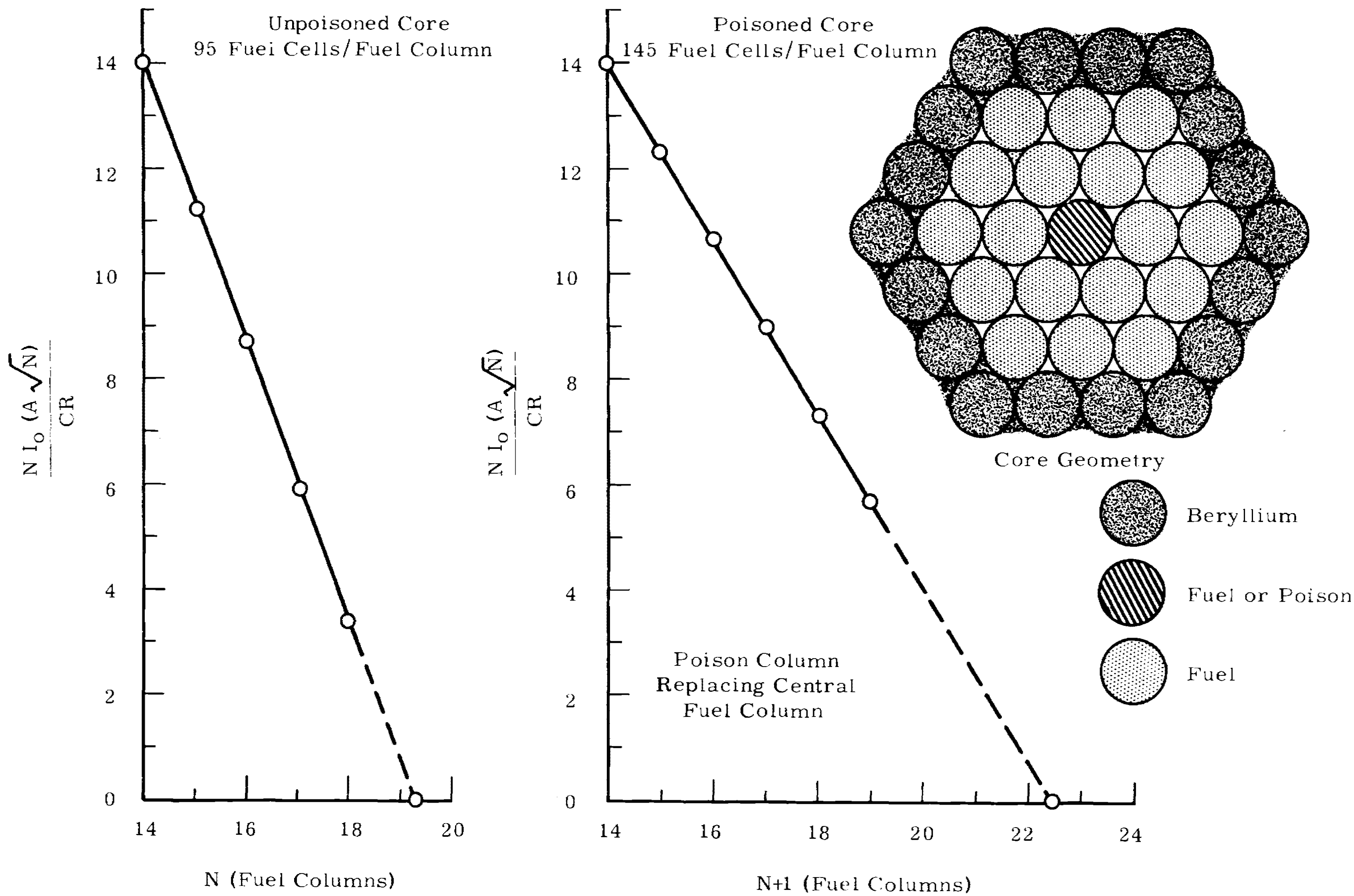


FIGURE 4.1-1

CAF-Phoenix Fuel Approach-to Critical Experiments

Recordings of reactor noise were made with the 19-column array at several fuel heights in the poisoned, beryllium reflected core. The measured values of $\frac{\beta-\rho}{\ell}$ for the subcritical assemblies have been used to extrapolate to the critical condition, (ρ=0), and a value of 623 $\pm$ 4 for β/ℓ.

TABLE 4.1-I

Results of Approach-To-Critical Experiments

Radial Approaches, Fixed Heights, Extrapolated Critical Number of Fuel Columns

Cells/Fuel Column	H_2O Reflected Unpoisoned Core	Beryllium Reflected Unpoisoned Core	Beryllium Reflected Poisoned Core
75		21.81 $\pm$ 0.02	
95		19.27 $\pm$ 0.04	
115			24.58 $\pm$ 0.09
126	23.76 $\pm$ 0.03		
130			23.41 $\pm$ 0.06
145	25.24 $\pm$ 0.03		22.43 $\pm$ 0.06

4.1-2: Theory-Experiment Correlations

For the analysis of the three CAF-Phoenix fuel experiments, it was convenient to interpret the extrapolated critical buckling of the first assembly in terms of an experimental k_{eff}. To do this, an axial buckling corresponding to the 30-inch height of an oversize radius sufficient to make the total buckling equal to the experimental critical buckling was used in a 17-group diffusion theory calculation of k_{eff}. It was found that k_{eff}=0.997, thus, the calculated k_{eff} for this assembly was three milli-k below the experimental k_{eff}=1.0. Keeping the same axial buckling, the radius was reduced to that corresponding to 19 cans. A calculated k_{eff} of 0.945 resulted. Assuming that the bias between diffusion theory and experimental remained constant at 3 milli-k low, an experimental k_{eff} of 0.948 (0.945 + .003) is reasonable. The results of the transport theory application for the simplest experimental assembly have been reported. Diffusion theory was used in the analysis of the three basic experimental geometries because better agreement with experiment was obtained than with transport theory for the nuclear data used.

The nuclear data used will be presented in detail in a formal document. The ^{239}Pu cross sections used were Sher 1965 values from BNL-325, supplement 2.

Spectrum average cross sections were computed using HRG,[(3)] a site revision of GAM-I and THERMOS.[(4)] The upper boundary of the thermal group was 0.683 ev. Thermal spectra for the core were calculated from two successive passes through THERMOS. First, a calculation was done in slab geometry with the internal fuel and moderator comprising the unit cell. These region average fluxes were used to adjust the nuclear densities for the next calculation. This second calculation in cylindrical geometry was a representation of the unit cell for the core in the axial direction. The can, the can wall, and the surrounding water were the regions used. The can region nuclear densities were adjusted according to the region average to cell average flux ratios obtained from the first case.

The sensitivity of k_{eff} to the number of energy groups is shown in Figure 4.1-2. The one dimensional diffusion theory code, HFN, [7] was used in the 4, 7, 10, 13, and 17 group calculations which define Figure 4.1-2. The 17-group structure was selected as standard for the analysis.

The calculated and experimental values of k_{eff} are shown in Table 4.1-II.

TABLE 4.1-II

Seventeen Group Diffusion Theory

Model	Experimental k_{eff}	Calculated k_{eff}
1. Water reflected	0.948	0.945
2. Beryllium reflector, homogeneous reflector	1.0	1.025
3. Beryllium reflected assembly, two-region reflector	1.0	1.012
4. Beryllium reflected assembly, central control rod	1.0	0.991

The poorest agreement was obtained for the partially beryllium reflected assembly with the ring of beryllium metal logs represented as a single homogenized beryllium water region. An improvement was made by using a two-region model for the reflector in which the water and beryllium were separated, half of the interstitial water in the reflector being lumped into a pure water region immediately surrounding the core. The

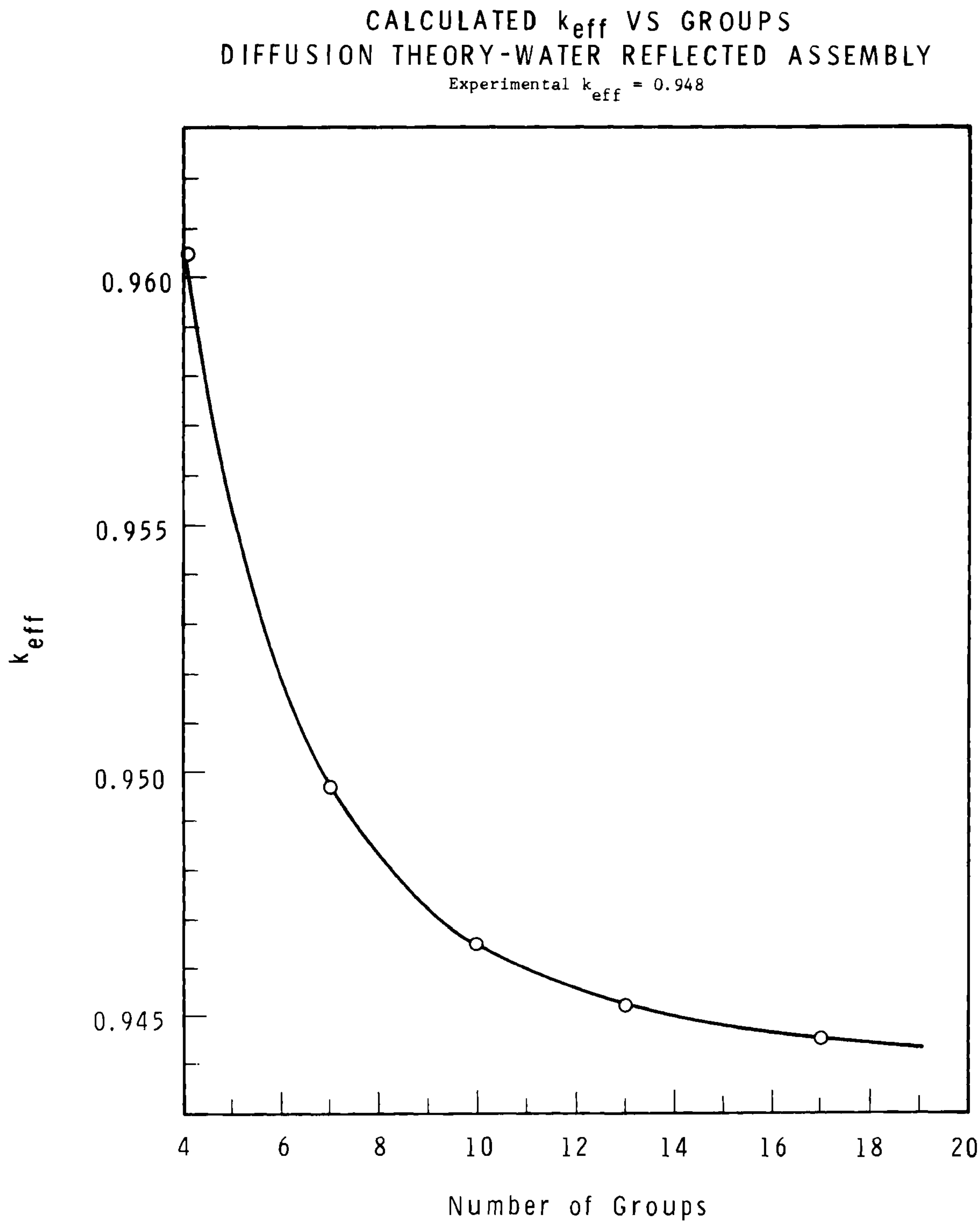


FIGURE 4.1-2
Calculated k_{eff} Versus Groups Diffusion Theory-Water Reflected Assembly Experimental k_{eff} 0.948

improvement obtained by this more accurate reflector model illustrated the great importance of the region adjacent to the core. The computed value of k_{eff} is heavily dependent upon the space and energy detail in this region because of the high fast and epithermal leak rate and because of in-leaking thermal neutrons. Many of these have first leaked out, been thermalized in the reflector, and then returned to the core.

In the poisoned core model, the cadmium region was smeared into the aluminum can wall region with appropriate self-shielding in the thermal region obtained from cell region fluxes from a THERMOS calculation. The THERMOS cell model included a large portion of the surrounding core in order to compute the correct spectrum.

In general, the three experiments and the calculational models developed provide a good test of nuclear data for use in small Phoenix reactor study. The data used in this particular study allows fairly accurate predictions of k_{eff} with a diffusion theory calculation. The diffusion theory models should be reasonably detailed both in energy and in space. This is particularly true of the core-reflector interface regions.

4.2: MTR Mockup Critical Experiment

4.2-1: Measurements

A critical mockup was constructed in the Plutonium Recycle Critical Facility (PRCF) which duplicated as closely as possible the physical configuration planned for the MTR experiment. The core was comprised of a three-by-nine array containing nineteen fuel elements and eight shim rods with fuel followers (see Figure 4.2-1). The core coolant and the top and bottom reflectors were H_2O. The radial reflector was beryllium. Each fuel element was comprised of sixteen plates of Al-20.3 wt% Pu, 0.040 inch thickness x 2.50 inch width x 23.50 inch length, and clad with 0.020 inch aluminum (see Figure 4.2-2.). Each fuel follower was comprised of twelve plates, identical with the fuel element plates except reduced in width to 2.34 inches. The nominal isotopic composition of the plutonium in the fuel was: ^{239}Pu-76.92%; ^{240}Pu-19.31%; ^{241}Pu-3.18%; and ^{242}Pu-0.59%.

Measurements in the mockup have included the shim-free critical size (14-1/4 element in a 3x4-3/4 element array, 5.65 kg plutonium), height of banked shims for critical with the full core (59% withdrawn), and the temperature coefficient of reactivity over the range 30-40°C (-1.97 $\pm$ 0.09¢/$^{\circ}$C). The effect on the power peaking at the bottom edge of the fuel plates of tapering the fuel cores was also investigated (Figure 4.2-3). Tapering the bottom edge of the fuel plate cores essentially eliminated the power peak at that location.

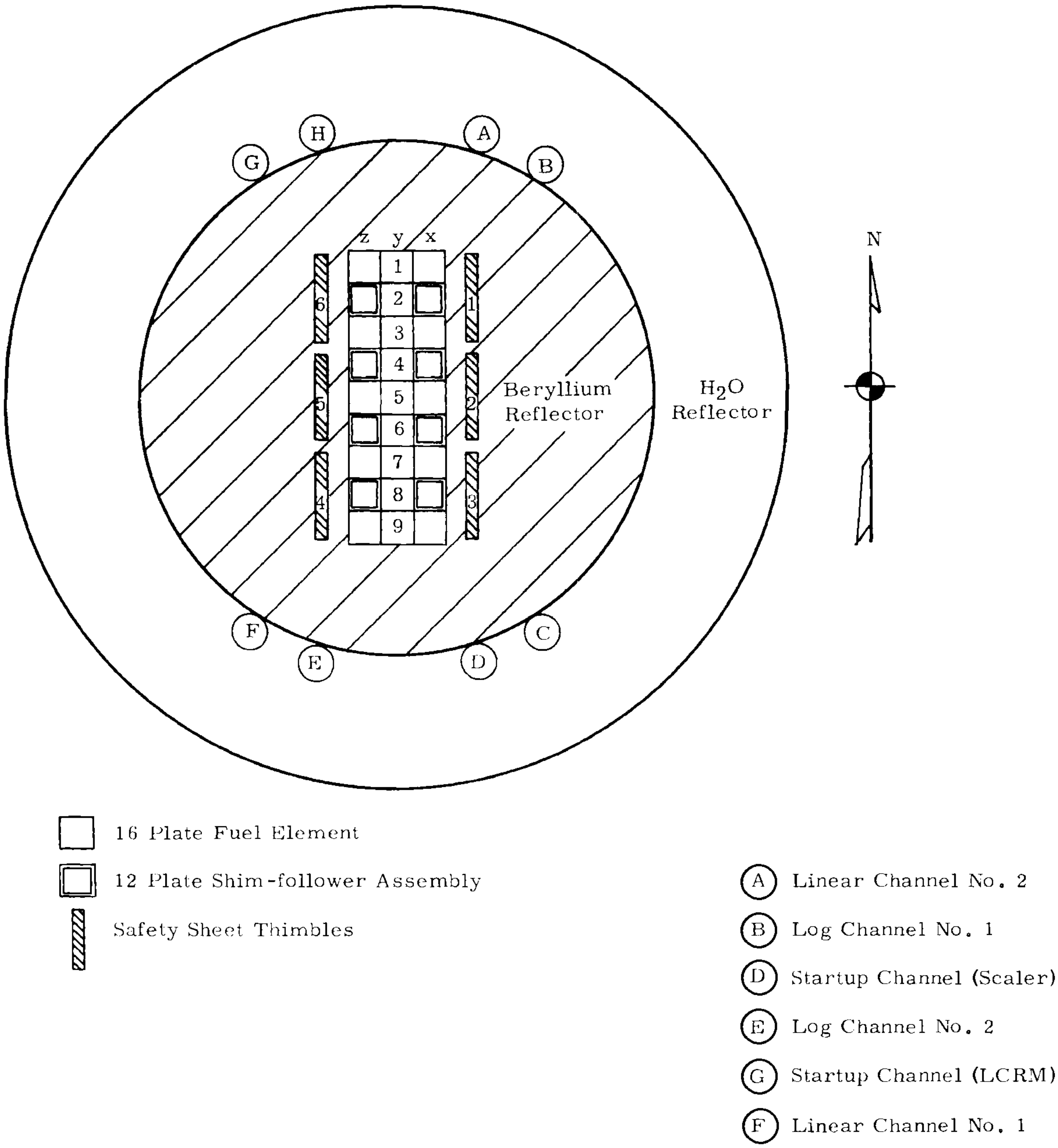


FIGURE 4.2-1

PPCF Mockup of the MTR-Phoenix Fueled Core

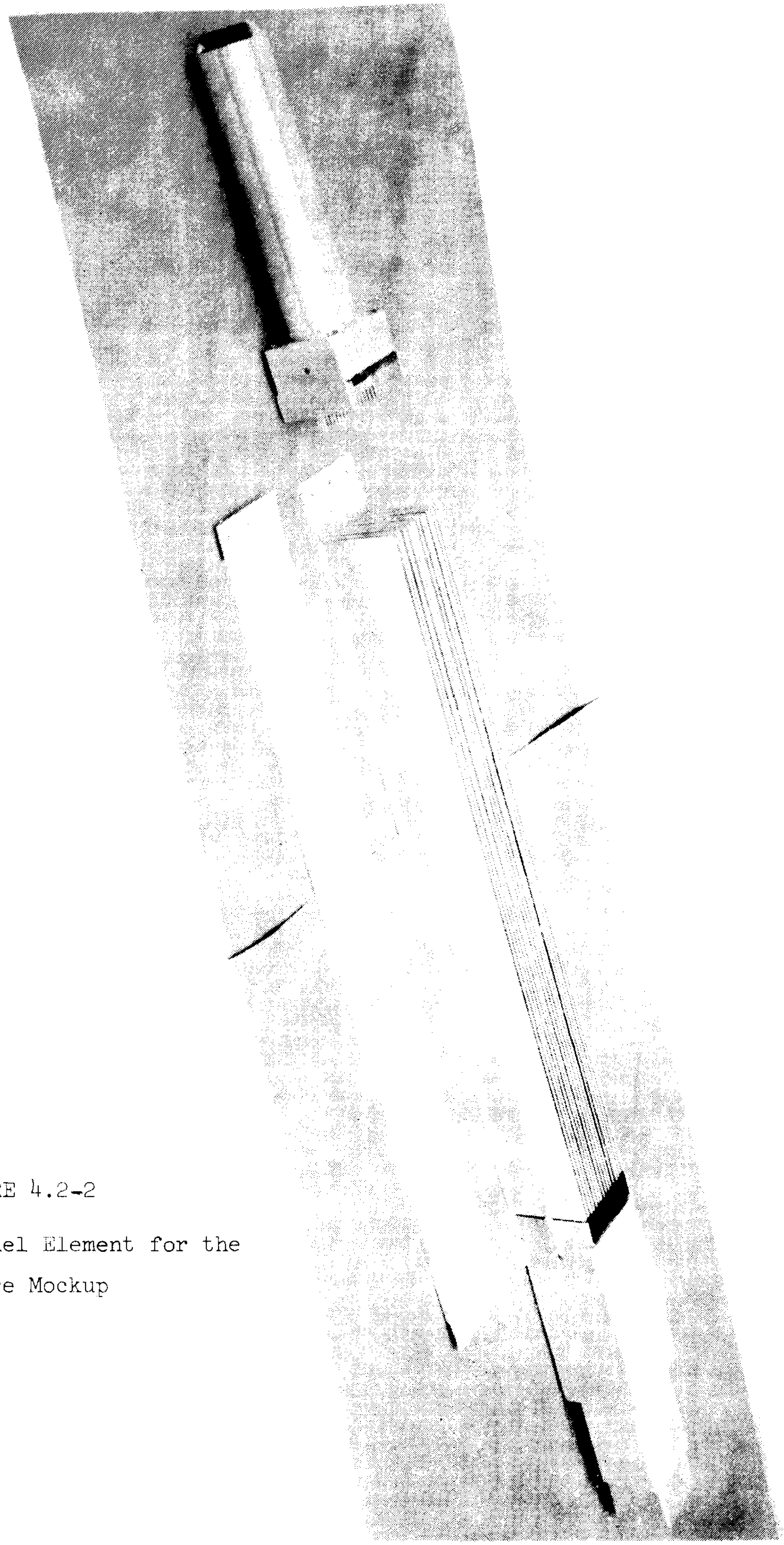


FIGURE 4.2-2

PRCF-Phoenix Fuel Element for the MTR Core Mockup

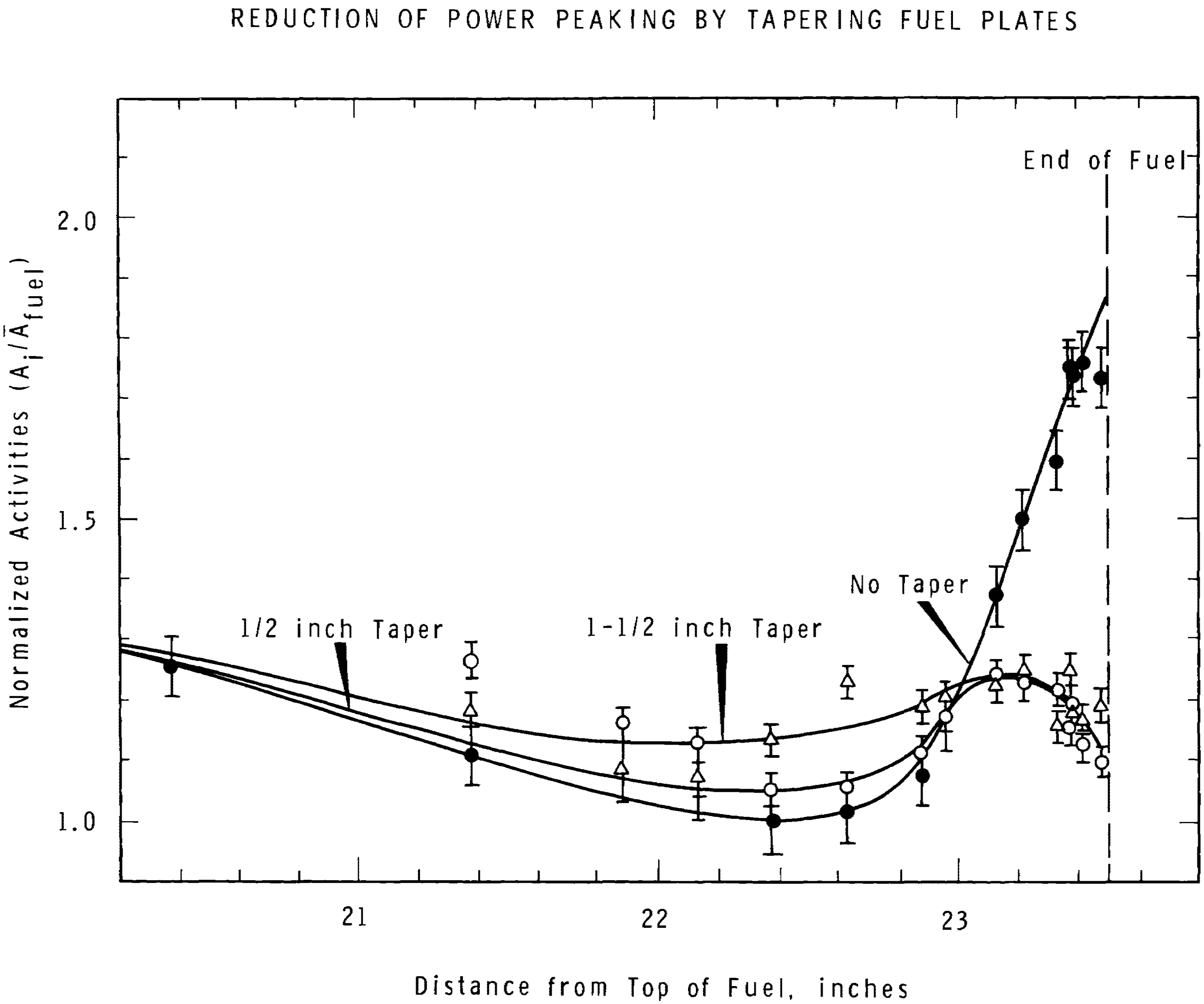


FIGURE 4.2-3

Power Distributions Near the Bottom of a PRCF-Phoenix Fuel Element

4.2-2: Theory-Experiment Correlations

The theory-experiment correlation for the MTR-Phoenix mockup in the PRCF is divided into two distinct portions; first, the pre-experiment planning phase, together with working calculations needed during the experiment; second, the post-experiment work now in progress.

First PRCF-Phoenix Fuel Loading

At the start of the PRCF experiment, the then preliminary analysis of the CAF experiments indicated that k was being over-estimated by about 3.5%. It was also expected that the PRCF Be reflector would contain about 10% H_2O and be relatively uniform throughout the system. Diffusion theory calculations using 4 energy groups and 2 dimensions gave a k of 1.035 for 11.6 fuel boxes. This value was taken from a curve drawn through calculations of the following loading patterns, always centered in the core: 2x3; 3x3; 4x3; and 9x3.

A critical loading of 16.5 boxes was determined. This corresponds to a k of 1.12 from the previous curve and was felt to be far from an acceptable result.

The 2-D calculation was gradually made more geometrically correct. The following table summarizes the results:

TABLE 4.2-I

PRCF-Phoenix Preliminary Loading Estimates

Calculated Critical Number of Boxes	Calculated k at Measured Critical	Calculated k Adjusted by CAF Experience	Condition of Reflector in Calculation
11.0	1.132	1.09	2% H_2O in Be, (i.e., MTR condition)
11.5	1.124	1.03	10% H_2O in Be, approx. PRCF condition
11.75	1.120	1.08	Above plus Be partially removed and Al beam tubes added in a homogenized representation.
12.5	1.102	1.06	Above plus water gap adjacent to core caused by interstices of Be tube reflectors.
14.0	1.075	1.03	Above plus water gap at control sheet position.

Additional 2-D calculations in the vertical plane were carried out to examine the effect of the Al beam tube simulators immediately adjacent to the core. These calculations showed that a drop of up to 3% k might be expected from this heterogeneity. This would lower the final calculated critical to 1.045. In retrospect (based on the final CAF analysis) 17 energy groups and a region mixed energy spectrum at the core reflector boundary might further lower this value to 1.02, which is more or less reasonable for this stage of the calculation.

After normalizing the control rod worth curve using the critical loading of 16.5 boxes, the predicted banked shim position was about 50% withdrawn. The experimental position was 65% withdrawn.

Second PRCF-Phoenix Fuel Loading

At this point in the experiment, it was decided that the reflector near the core should be restacked to more nearly represent the MTR. In particular, the beam tubes should not be immediately adjacent to the core and large water gaps should be avoided, if possible.

The PRCF-Phoenix core with restacked Be reflector was calculated using 4-group diffusion theory in two dimensions. These calculations indicated a clean critical configuration of about 11.2 fuel boxes. If a correction of between 2 and 3.5% in k as determined by the CAF experiment is applied, the clean critical configuration should be from 12.4 to 13.1 fuel boxes.

The experimental value of 14.25 fuel boxes corresponds to an error in k of about 5.5% from the 4-group calculations or between 2 and 3.5% greater difference between the calculated and experimental values than for the CAF-Phoenix experiments.

The fully loaded core with banked control rods was calculated by a combination of 2 and 3-D codes. The following predictions of the critical rod position were made prior to the experiment:

Calculated position:	51% withdrawn
Calculation adjusted by CAF experience:	56% withdrawn
Calculation adjusted by previous PRCF experience:	59.6% withdrawn

The experimental value is 58.5% withdrawn. The total rod travel is 78cm. The difference between the highest and lowest prediction is about 6.7cm, which is about 6% in k.

Another equally important theory experiment correlation is that of power profiles in the Phoenix core. The 3-D diffusion theory code WHIRLAWAY in modified form (see Section 5.2) has been used to calculate the power distribution throughout the core. Figure 4.2-4 shows measured and calculated fission densities for both an internal and external plate in a stationary fuel element. The external plate is adjacent to a control rod whose tip is about 30 cm down from the top of the fuel meat. The diffusion coefficient from the Battelle THERMOS code gives significantly better results than from the HAMMER THERMOS version. The reactivity is essentially unchanged from one thermal spectrum code to another.

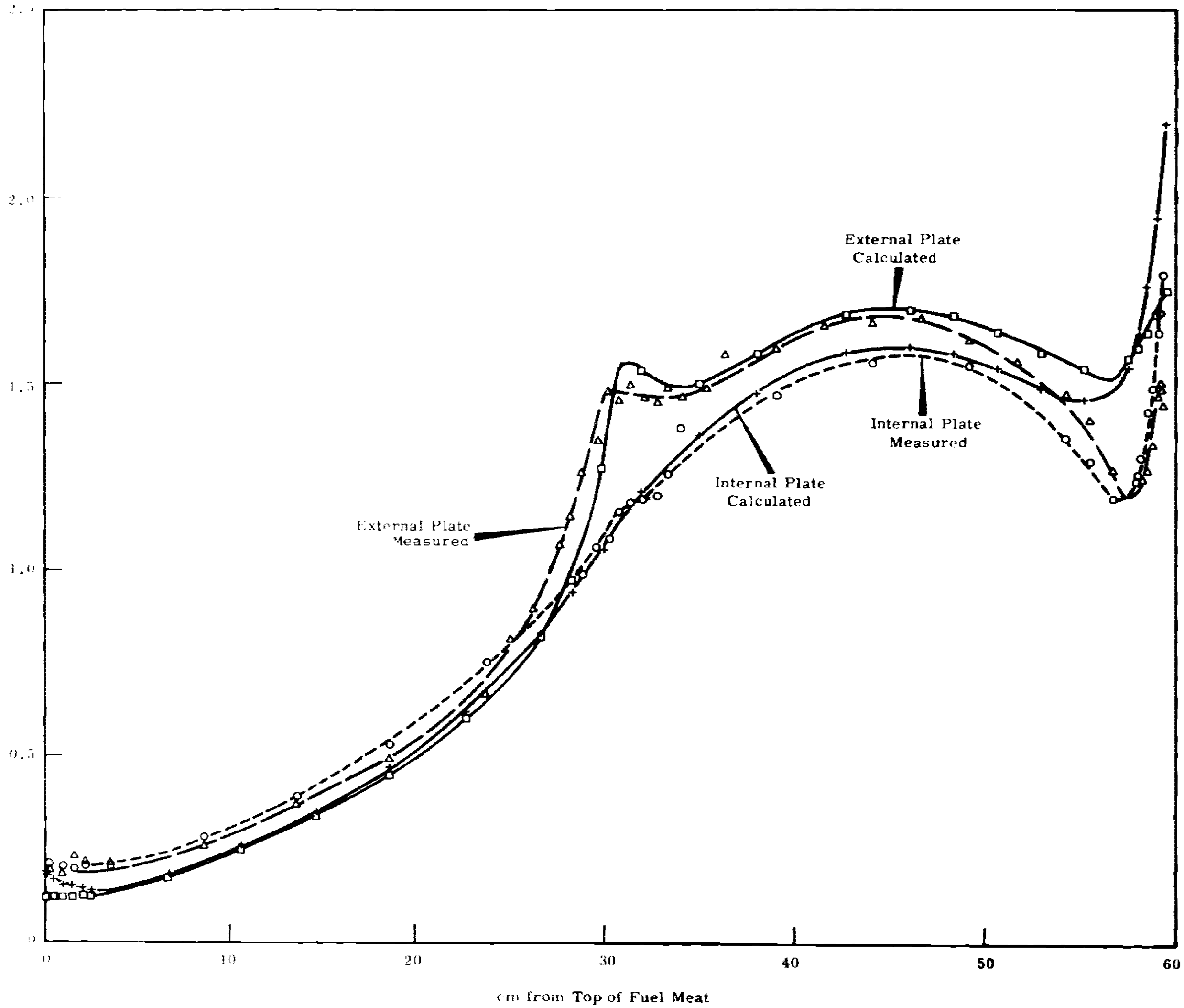


FIGURE 4.2-4
Comparison of Measured and Calculated Power PRCF Distribution for Phoenix Fuel Elements

4.3: Fuel Development and Fabrication

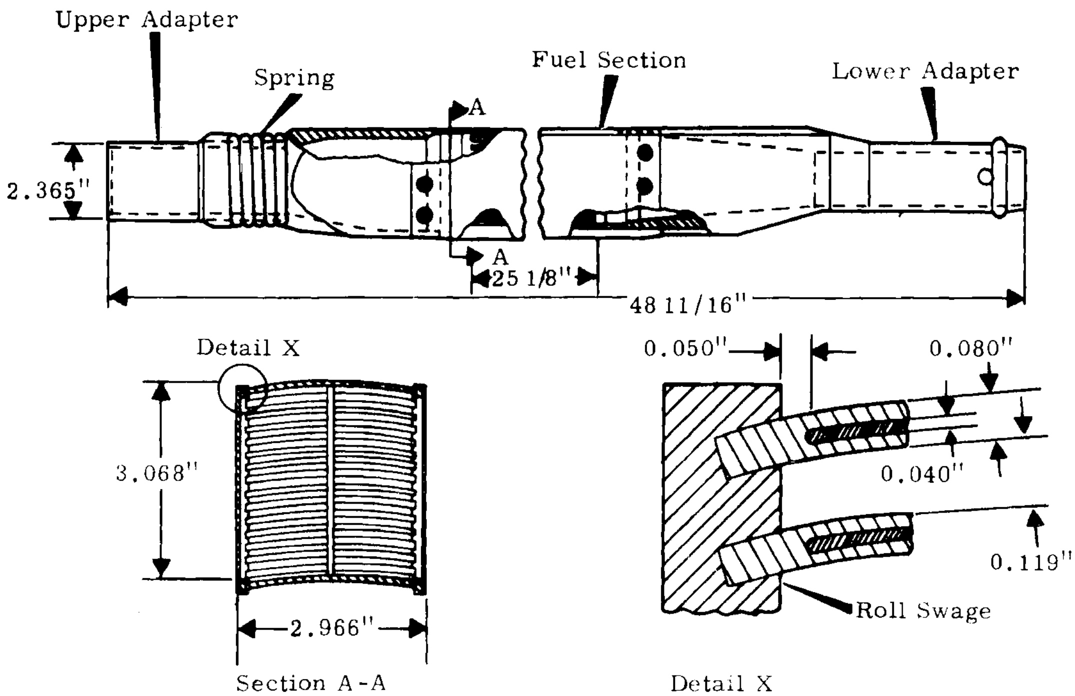
4.3-1: General Description of Fuel Element and Shim Control Rod Design

The Phoenix experiment fuel element and shim control rod are similar in design to their existing MTR counterparts. The fuel design was selected to meet operating requirements of the Phoenix experiment and at the same time remain as close to a standard MTR fuel as possible. The fuel element will consist of sixteen 0.080 inch thick, equally spaced, roll fabricated fuel plates in the standard MTR configuration. Each plate will have a 0.040 inch thick 21 wt% plutonium, 79 wt% aluminum core clad with 0.020 inches of 6061 alloy aluminum. The water channels will be 0.119 inches, providing flows similar to those in standard MTR elements (see Figure 4.3-1). The plutonium utilized will be Shippingport blanket core I plutonium, and will have the following approximate isotopic content:

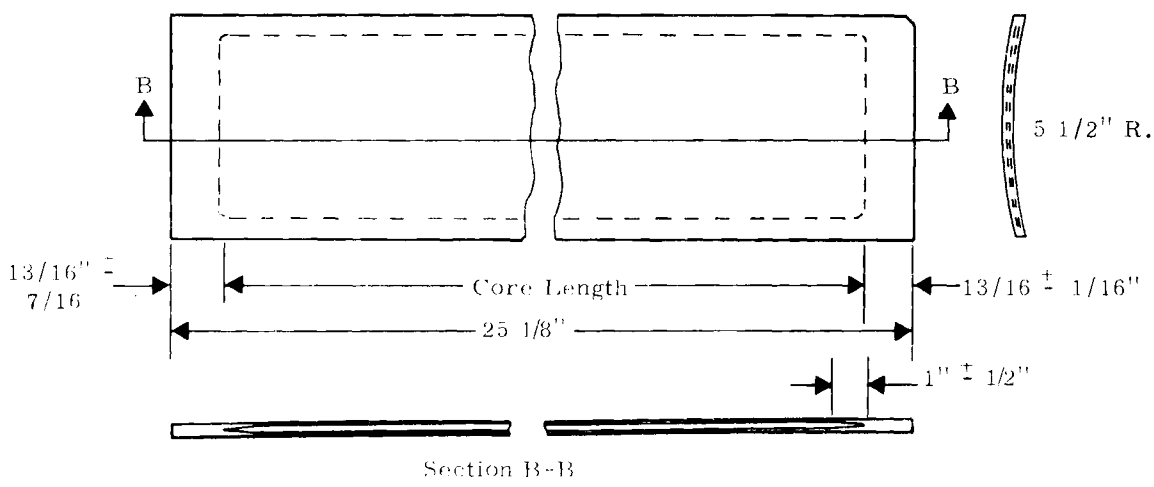
^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
0.50	65.3	23.2	7.6	3.4

To control peaking problems at the bottom of the fuel element, the fuel cores will have a 1/2 inch to 1-1/2 inch end taper and the core bottom located within the element within $\pm$ 1/8 inch. The experiment requires the use of flux monitors in some of the fuel elements. The flux monitor will consist of a fuel bearing wire located between two plates extending the length of the plates. These two plates will have narrow cores to avoid hot spot problems and pins passing through them to hold the flux monitor wire in location.

The shim control rod fuel plates will be similar to the fuel element fuel plates utilizing the same concentration plutonium aluminum alloy, 0.040 inches thick cores in 0.080 inch thick plates. The shim control rod fuel section will consist of thirteen fuel plates of standard MTR configuration, spaced with 0.119 inches of water channels in a roll swaged and welded but otherwise standard MTR assembly.



PHOENIX MTR FUEL ASSEMBLY



PHOENIX FUEL PLATE

FIGURE 4.3-1

Fuel element loadings will be 25.0 grams of plutonium per standard plate and 22.0 grams per plate for narrow core plates. This will result in assembly loadings of 400 grams of plutonium for standard elements and 394 grams for flux monitor elements. Shim rod fuel section loadings will be 23.2 grams of plutonium per plate and 302 grams per shim rod. It is planned to fabricate 23 fuel elements and 10 shim rods with the reactor loading requiring 19 fuel elements and 8 shim rods.

4.3-2: Design Performance Conditions

The fuel will operate under similar conditions to the existing MTR fuel. The average fuel exposure will be $4\text{-}5\times10^{20}$ fission/cc. The maximum fuel temperature will be $\sim 360^{\circ}F$ and the maximum cladding surface temperature will be $\sim 332^{\circ}F$.

Under these irradiation conditions, it is predicted from the computer code[13] that the maximum film oxide formation will be ∿1.6 mils. which corresponds to a metal (clad) penetration of ∿1.1 mils with a maximum localized attach of twice that value, based on the observation of Griess, et.al.[14]

From observations of analogous U-Al alloy fuels and some limited data on plutonium-aluminum fuels, irradiated to these exposures, under similar conditions, we anticipate no radiation damage problems. We expect fuel swelling on the order of one to two percent. This should be accommodated by the matrix without the formation of blisters or nonbonds and without the distortion of the fuel plates.

4.3-3: Fabrication Development

Core Fabrication and Homogeneity

The homogeneity specification established for the Phoenix fuel plate requires the Pu content of the core to vary no more than 10%.

The first cores fabricated for roll plate assemblies were resistance melted in air and cast into graphite molds. The first cores were cast to width and thickness and machined to length. Plates fabricated from these cores met homogeneity limits, but contained micro-discontinuities caused by either porosity or inclusions. All other cores fabricated to date have been induction melted, cast to billets in graphite molds, and the billets extruded to core shape. The extrusions are cut to length to provide a finished core. Plate assemblies rolled with these cores are generally free from voids and discontinuities and well within homogeneity limits. In addition to providing excellent core characteristics, the extrusion process provides high yields with 97% of the original melt cast and 80% of the extrusion usable. Using this fabrication process it is estimated that the total plutonium content of individual fuel plates will vary 4% and fuel elements less than 2%.

Taper Control

The peaking problems associated with the Phoenix fuel require the use of a tapered core on the bottom end of the fuel plate. Four types of end configurations were rolled at 6 to 1 reductions, using uranium aluminum alloy cores as a stand-in for the plutonium aluminum core and 1100 cladding. From the results of this test, the eliptical end shape was selected as best, and four plutonium aluminum plates fabricated. The Pu-Al core plates exhibited marked differences from the U-Al cores, indicating the U-Al is a poor stand-in for Pu-Al. These first Pu-Al fuels exhibited dog-earing at corners, triangular voids at core ends, and dog-boning and clad thinning at core ends. Additional development plates were fabricated using 1100 clad plates with 1100 inserts at the core ends, 1100 clad plates with 6061 inserts at the core ends, and 6061 clad plates. Dog-earing was eliminated on all plates by chamfering the edges of the cores. The 1100 alloy clad plates with 1100 inserts still exhibited extreme dog-boning. The 1100 alloy clad plates with 6061 inserts reduced dog-boning, but did not eliminate it. The 6061 alloy clad plates exhibited no dog-boning and had tapers close to those desired. An additional problem involving inserts was the length of insert required. It was necessary to enclose the entire end of the core to reduce dog-boning, and this resulted in a vertical bond line extending from the core to the fuel plate edge that does not receive full reduction pressure during rolling. This bond interface often does not bond providing an unexceptable water path to the core. It is possible that additional core shaping could make 1100 clad acceptable or that inserts could be redesigned to gain bond line reduction, but at this point 6061 was selected as the preferred clad and no further 1100 clad development undertaken.

Cladding Selection

The choice of cladding depended upon corrosion conditions, strength requirements, and fabricability. Two choices examined in detail were 6061 and 1100 alloys. 1100 alloy is the standard MTR cladding and 6061 the standard HFIR and ATR cladding. 6061 alloy aluminum was selected as the preferred Phoenix fuel cladding for the following reasons:

1. 6061 alloy eliminates dog-boning and allows the fabrication of a fuel plate with a tapered core.
2. If it is necessary to increase heat flux beyond existing MTR standards, the 6061 alloy is preferable based on a greater volume of favorable in-reactor and ex-reactor data.

3. The 6061 alloy is stronger than 1100 providing greater resistance to fuel deformation.
4. The 6061 alloy has more potential for advanced applications.
5. The 6061 clad fuels are less susceptable to scratches and other surface defects.

Bonding and Destructive Testing

The Phoenix fuel plate requires a metallurgical bond on the clad to core and clad to frame interfaces. The quality of the metallurgical bond obtained with rolled plate fuels depends upon the total reduction ratio utilized, reduction per rolling pass, rolling temperature, cleanliness of components, and material at interface. Test coupons were rolled at temperatures of 500, 530, and 560^{0}C and reduction ratios from 6:1 to 12:1 to determine desirable fabrication conditions. The test coupons consisted of three 6061 plates with a layer of 1100 aluminum inserted at one interface. The results of the rolling tests as determined by metallography indicated that reductions of 10 to 1 are required to completely break-up bond line oxide layers. The higher temperatures appear only slightly better bonded, and the use of a 1100 alloy interface greatly enhances bonding. No grain growth across the bond line occurs on the hot rolled plate, but on good bonds the oxide layer is well broken up providing metal to metal contact and there are no voids. Grain growth across the bond line will occur if the plates are cold rolled and annealed as is required on HFIR and ATR fuels. Since all initial Phoenix plates were hot rolled only at reductions of 6:1, grains did not grow across the bond line and the oxide layer was clearly visible. However, void-free bonds meeting standard MTR requirements were obtained at the 6-1 reduction.

A convenient method of utilizing 1100 alloy aluminum at the interface is to use 6061 Al-clad with 1100 aluminum as plate assembly components. This technique is used on HFIR and ATR fuels and permits the fabrication benefits of a 1100 to 1100 bond to be used on 6061 clad fuels. As a result of this test work and HFIR and ATR experiences, the 6061 Al-clad material was selected as the preferred cladding material and 12:1 selected as the reduction ratio for future development plates.

Destructive testing is used to qualify fabrication processes and as an in-process control on representative samples. Phoenix destructive test samples are checked for clad and core thickness, end taper shape, bond integrity, and plutonium content. A 2% sample rate will be utilized during Phoenix plate manufacturing as an in-process control.

Non-Destructive Testing

All Phoenix development plates have been non-destructively tested to evaluate the plates and testing method. The non-destructive tests used were ultrasonic scanning for voids, unbonds, and discontinuities, radiographs for core shape, densitometer scans of radiographs for density homogeneity, gamma counting for homogeneity, dye penetrant for non-bonds and blister tests.

The ultrasonic scanning utilized both through-transmission and pulse echo techniques to find and locate discontinuities. The system proved very sensitive and found stringers in as-cast cores as small as .005" in diameter and voids at core ends as small as .010". The pulse echo technique can be used to determine the location of discontinuities and is more valuable for evaluation of development plates than as a manufacturing test.

X-ray radiographs were used to locate cores and evaluate core shape as well as provide radiographs for densitometer evaluation of plate density. The radiograph method of locating cores is adequate, but not as handy as the flourescope systems normally used for plate manufacture. A densitometer with a 1/6" diameter aperature was used to measure film density. The plate center was utilized as one standard and the frame as another and the variation in density across the plate plotted. The system provided a good measure of density variances and an excellent plot of core end tapers. Gamma counting was also utilized as a homogeneity measurement technique on some development plates. The system worked well, and is considered more accurate than the film density system on a point basis. However, the radiograph film density method meets process requirements and was selected as the primary homogeneity measurement tool due to availability of equipment, the resulting visual record of entire plate, and rapid measurement of core taper.

Zy-glow liquid dye penetrant tests were used to evaluate suspected non-bonds on plate edges. The test is good when used to locate suspect defects, but does not provide a measurement of depth of defects. All actual defects found with this technique were also located with ultrasonic techniques.

All test plates were blister tested by heating to rolling temperature for one hour after the completion of hot rolling. After cooling, the plates were examined for blisters and any plates having blisters in the finished plate area considered rejects.

The standard non-destructive tests selected for use on all future Phoenix plates are the blister test, through transmission ultrasonic tests, and the X-ray radiograph with film densitometer test. Standards will be required for the ultrasonic and radiograph tests.

Fuel and Shim Rod Assemblies

Use of a flux monitor in some of the fuel assemblies has required a slight modification of the standard MTR fuel design to allow the insertion of the monitor wire. Two dummy assemblies using the flux monitor design have been ordered from a commercial fuel fabricator and will be used for hydraulic testing prior to the fabrication of plutonium bearing elements. The use of 6061 as the fuel plate cladding has also required a change in the design of the shim rod fuel section. The standard MTR shim rod has 1100 clad plates and has a brazed fuel section. The Phoenix shim rod fuel section will be roll swaged and welded or pinned and welded since the 6061 melt temperature is too close to brazing temperature. A dummy shim rod fuel section will also be fabricated for hydraulic and shock testing prior to the assembly of plutonium bearing elements.

4.3-4: Radiation Levels and Contamination Control

The high exposure plutonium required for the Phoenix experiment will result in higher radiation levels during fabrication, assembly, and inspection than with normal MTR fuel elements. Contact radiation levels will be on the order of 2000 mrad/hr and the unshielded radiation level at one foot will be approximately 1/10 of the contact levels. Although these radiation levels are quite high, close control and minor shielding can be used to allow fabrication with no exceptional problems or hazards. Fuels with higher levels of radiation have been fabricated at PNL using methods similar to those used in plate fabrication.

Careful controls are required during plate fabrication to prevent alpha contamination. Cores are carefully cleaned and the assembly kept free of smearable contamination. During plate rolling and machining precautions are taken to minimize contamination, should the cladding be ruptured. However, no incidents occurred during development fabrication, and no problems should be encountered during assembly and inspection operations.

4.3-5: Irradiation Testing

An irradiation test of prototype Phoenix fuel plates is scheduled for charging in October of 1967. Six full length test plates will be irradiated in the MTR L-51 position to exposures similar to those planned for the full core experiment. The plates will be .080" thick and have .040" thick cores of 20 wt% plutonium, 80 wt% aluminum alloy. Fabrication techniques will be those planned for the full core loading. Hot spot conditions on the irradiation test plates will be close to those expected during the experiment, and core tapers will be held as close as possible to the minimum taper required for the Phoenix experiment plates. The assembly will contain Co-Al and Ni flux monitor wires and the element inspected between cycles. It is planned to irradiate the assembly from 4 to 6 cycles.

4.3-6: Fuel Element and Shim Rod Specifications

Specifications have been prepared covering both the Phoenix fuel element and shim control rod and have been issued for AEC and INC review. The specifications are based on the results of development work completed to date and the results of MTR, HFIR, and ATR fuel fabrication development work performed by INC and ORNL. While the specifications are in final format, they are subject to modification depending upon the results of Physics tests and fabrication development work now in progress.

4.4 - Mechanical Design of the MTR-Phoenix Core

4.4-1: Introduction

The use of the Materials Test Reactor (MTR) at the Idaho Test Site for the Phoenix Fuel Experiment requires some mechanical modification of the reactor. These modifications, under the direction of INC, are envisioned as follows: (1) shifting the active core; and (2) altering the control system as necessary to control the Phoenix fuel load. Also, the Phoenix loading calls for the use of all the shim control rods requiring the removal of a loop from one shim rod position and preparing the position to accept a shim rod.

During the course of the experiment a method of monitoring the neutron flux activity within the core is desired. This can be accomplished by inserting a flux monitoring device into the active core.

4.4-2: Shifting the Active Core

Lattice Piece Re-arrangement

The shifting of the fuel array to the center of the reactor core will require a rearrangement of the lattice fill pieces. As can be seen in Figure 4.4-1 the lattice fill pieces are now all on the south side and consist mostly of "LB" pieces (1-3/8 diameter basket hole) with some "LB-4X" and "LB-4N" beryllium pieces. It is desired that these be replaced with solid beryllium "L" pieces with a 3/16" diameter coolant hole. The centered core will require nine lattice pieces on each side (rows 1 and 5) or a total of 18 for both sides. The proposed MTR-Phoenix core is shown in Figure 4.4-2. A check of the lattice piece inventory shows that 18 "L" pieces are available. In the event that not all 18 "L" pieces are usable, the "LB" pieces could be used provided the larger hole could be filled with a beryllium insert. Twenty-two "LB" pieces are available.

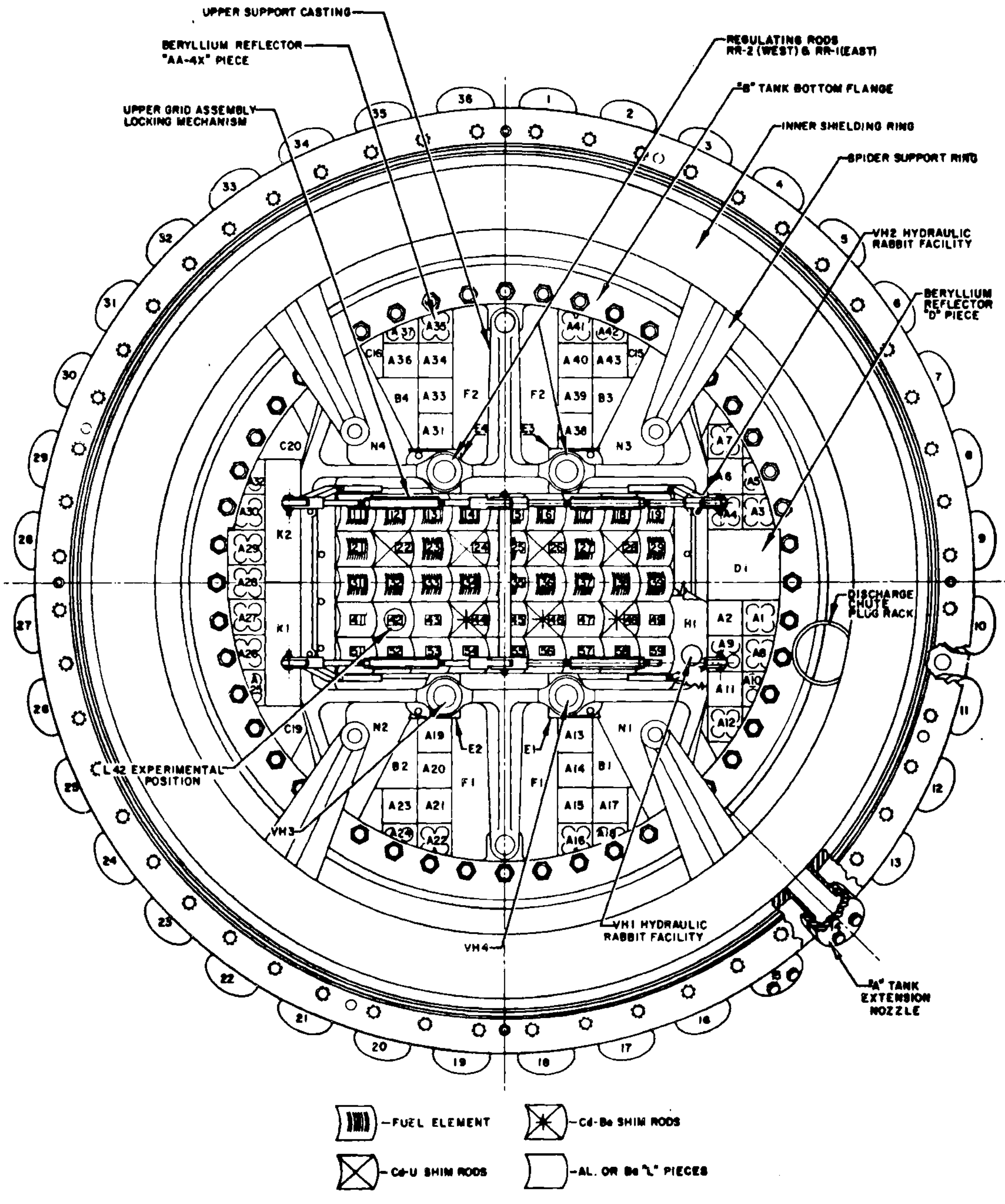


FIGURE 4.4-1

MTR Core

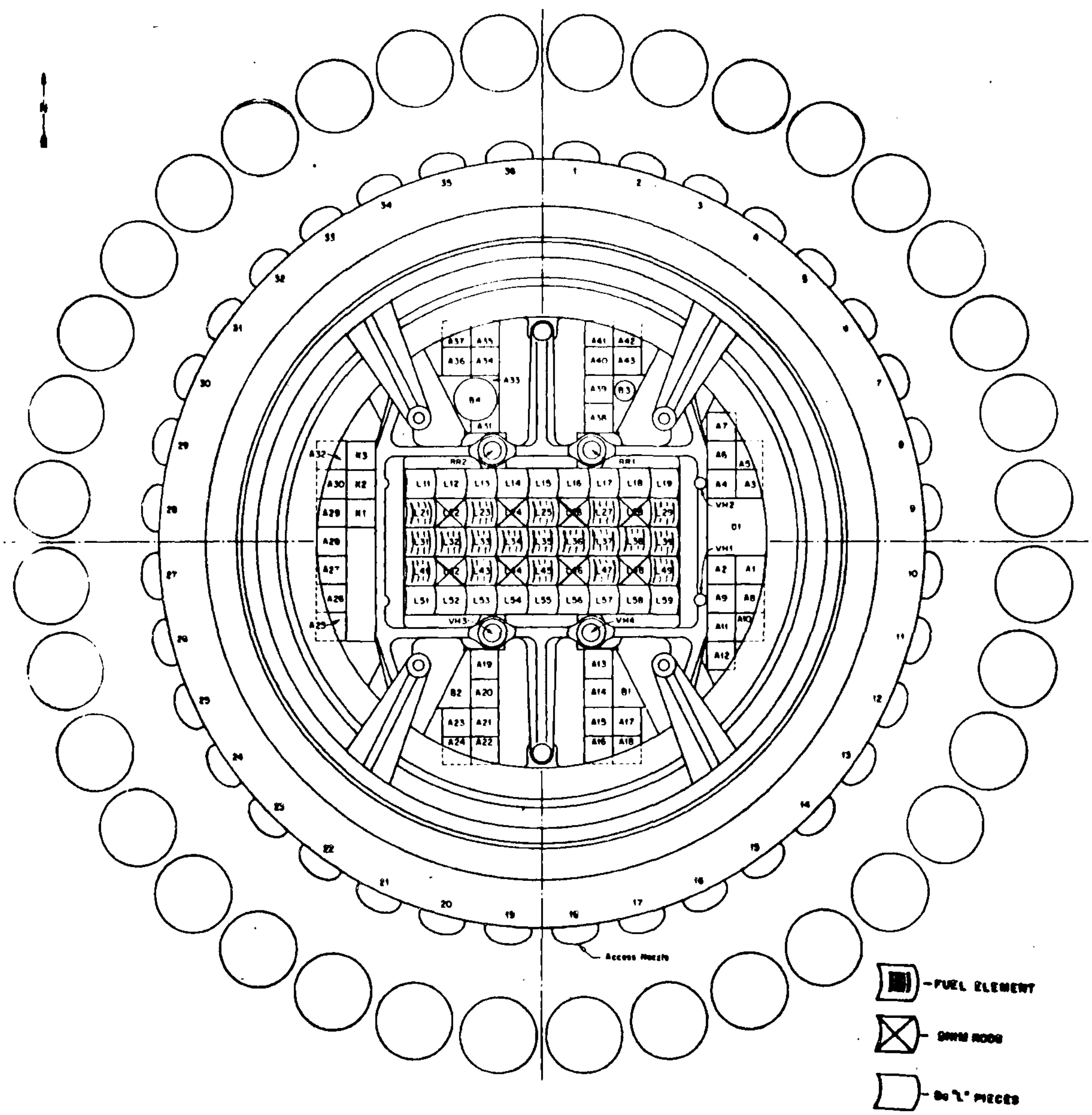


FIGURE 4.4-2

Proposed arrangement of MTR active core with Phoenix Fuel load.

Fuel Element Monitor Probes

The monitor tube system is designed to measure flow, temperature, and activity of the cooling water leaving each fuel assembly. This is accomplished with 37 monitor tubes which pass up through the bottom plug into the end boxes of the fuel elements. The present slab loading utilizes only 23 of the available 37 monitor tubes. The shifting of the core will require only 19; however, some will be different than now used. None of the tubes on the north side (row 1) will be monitored but 5 in row 4 will be connected up and monitored.

The monitor tubes in row 4 are all believed to be in place; however, their condition will need to be determined. The change over to the other monitor probes is not believed to present any difficult problem. However, some of the probes which are not now in use will undoubtedly need changing out.

4.4-3: Control Modification

Lattice Position No. 42

The Phoenix loading will call for the use of all eight shim rods with fuel followers. One of the present positions now has a KAPL loop installed and has had little or no use as a shim rod position. The condition of the lower bearing and shock absorber is uncertian. Recent photos show that the lower bearing is in place but the condition cannot be determined until the loop is removed.

The existence of the shock absorber or its condition, if one is in place, is completely unknown. The location of these shock absorbers make removing and installing a new shock absorber very difficult. If one is in place, it probably is drilled out to allow passage of the existing KAPL loop piping. If the shock is gone or cannot be made usable, this shim rod would have to be removed from the scram circuit as dropping the rod without adequate shock absorbing features would seriously damage the rod. The standard magnetic clutch feature which holds the shim rods could be retained but being removed from the scram circuit will require a separate power source and also an integral shick absorber on the end of the shim rod in the event the clutch should accidentally let go.

An alternate approach would be a disconnect with a quick release ball coupling joint such as used on the regulating rods. The latter would probably not require any shock features and the end of the rod would only need to be altered to hold the rod in the proper position when the drive is disconnected for removing the top reactor plug.

Shim Rod Drive Motors

A slower speed shim rod drive motor will be used for the Phoenix fuel load. The motors which will be used to replace the GE 1/6 hp 1425/1720 RPM motors now used are Delco 1/4 HP 360 RPM motors. These motors are now in storage as they were used for a previous (cycle 108) Pu fuel load.

The slower speed motors having been in storage and not used for several years will need to be inspected and tested to insure that they are in satisfactory operating condition. A description of these motors is given in Table 4.4-1, along with the 1/6 HP standard drive motors now being used.

Possible Regulating Rod Modifications

Tests will be conducted in the PRCF mockup critical experiment. Shifting the core 3 inches away from the existing regulating rod will affect the worth of these rods sufficiently to where some alteration may be necessary. The present regulating system is a dual rod system on the north side of the core with one rod giving the "fine" reactor control and the other rod held on standby.

Should a greater regulating rod worth be needed as a result of shifting the active core, one aternative would be to install regulating rods in the positions provided on the south side of the core. If the original bearings are still in place, the original size (1-1/2 inch O.D.) regulating rod could be used to provide greater Δk control. The regulating rod worth could be increased by utilizing two rods. There is a question as to whether the output of the amplidyne units have sufficient capacity to handle two rods. The two regulating rod drives would have to be linked together to insure simultaneous movement. A similar standby regulating rod system is also desirable. A description of three extra regulating rod drive motors now in storage is given in Table 4.4-I.

TABLE 4.4-I

No. of Units	Description
3 ea.	Motors, SR Drive, GE, 1/6 HP, 3 PH., Cycle-50/60, RPM-1425/1725, V-220-208, AMP-.8/.7, Temp. Rise -55^{0}C, Time Rating.
8 ea.	Motors, SR Drive, (For Pu Run) Delco, A.C., Model-14153, SN-L-55, 40^{0}C Rise Cont., Frame-66, V-220/240, Cycle 60, 3 PH, Amps-1.50/.75, HP-1/4, RPM-860, Code-L, Design-A.
3	Motors, Reg Rod, 2 s/mounting brackets, 1 w/o mounting brackets, 1 HP, 250 volt D/C, 4 Amps, 1200 RPM Max.

4.4-4: Flux Monitoring Device

Purpose

To obtain the maximum information from the Phoenix fuel experiment, a method is needed to determine the flux within the active core initially and as the experiment proceeds. Also, a means is needed to determine fuel burnup after desired exposure intervals. Several positions within the core should be monitored due to the flux variation within the reactor core so that fuel in areas of different neutron flux magnitudes can be evaluated.

Location of Flux Monitoring System

The MTR active core presents very few locations where a flux monitoring system can be placed during full power operation. Our proposed flux monitor concept consists of inserting a fueled flux monitor wire directly into the fuel boxes to be placed at the desired locations. The flux monitor wire made up into a wand will be inserted down the corner of the fuel box as shown in Figure 4.4-3. The wand will go into and down the inside corner of the upper end box and on down beside the side plate between the second and third fuel plates from the concave side of the fuel box as shown in Figure 4.4-4. The wand is held against the side plate by a series of pins spanning the second and third fuel plates shown in the blow-up in Figure 4.4-5. The two fuel plates adjacent to the flux wand will have a non-fueled strip approximately 1/4 inch wide nest to the wand as the coolant flow will be insufficient between these two fuel plates due to the wand to provide adequate cooling.

Description of Flux Wand

The fueled flux monitor wand originally conceived consisted of a fuel core wire within an aluminum tube, with coolant introduced into the head of the wand which flowed between the tube I.D. and wire to cool the fueled wire. However, calculations were made to determine if a solid 0.115 inch O.D. aluminum rod with a 0.020 inch diameter fuel core would operate satisfactorily as this would be a stronger and simpler wand to construct than the wire and tube wand. The solid fueled flux wire is shown in Figure 4.4-6. Results of the calculations showed that the

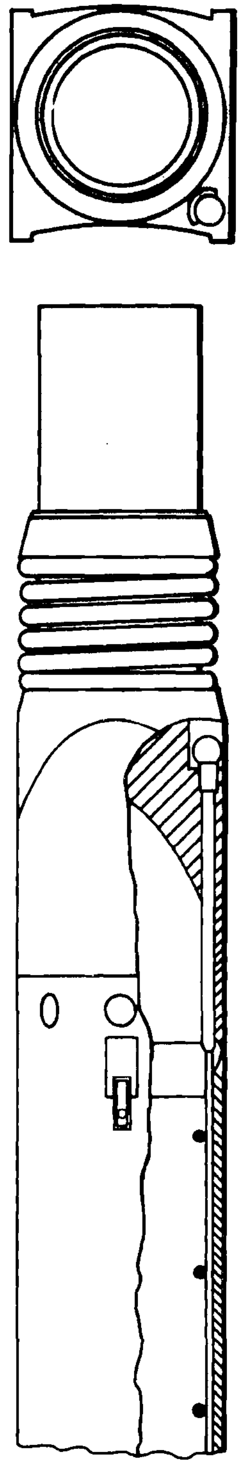


FIGURE 4.4-3
MTR Fuel Box Showing the Location of the Flux Monitor Well with the Flux Wand Inserted

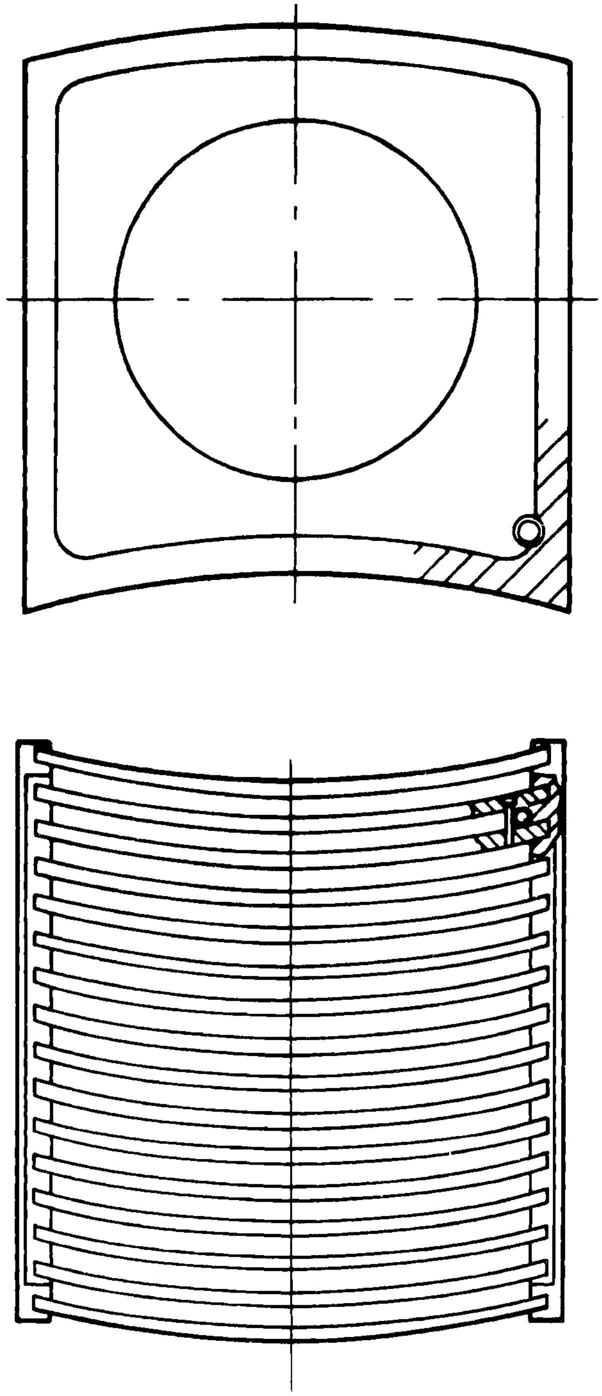


FIGURE 4.4-4
Views Showing the Location of the Flux Monitor Well Down the Corner of the End Box and Down Between the Fuel Plates of an MTR Fuel Box

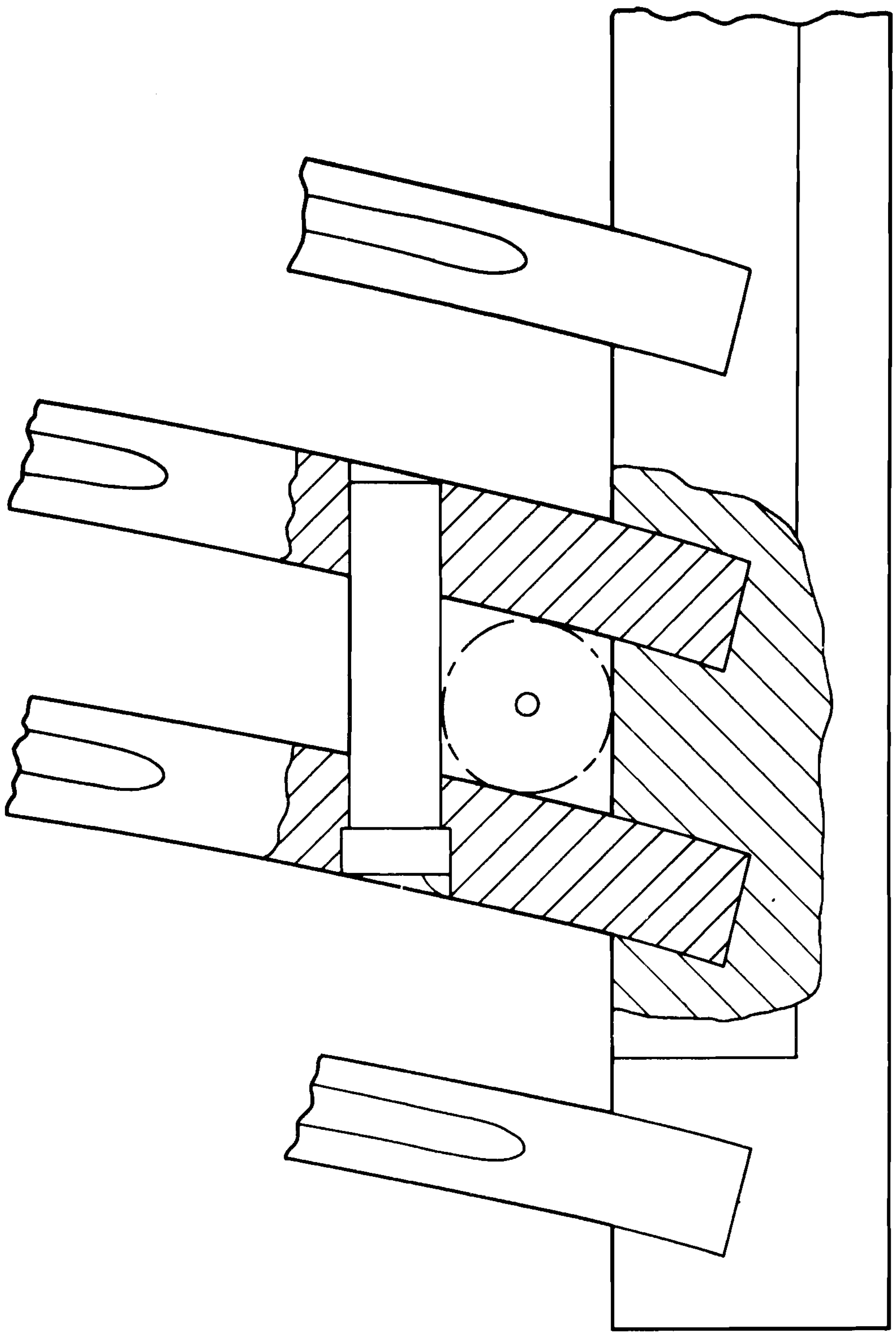


FIGURE 4.4-5
Enlarged View of the Flux Wand Position

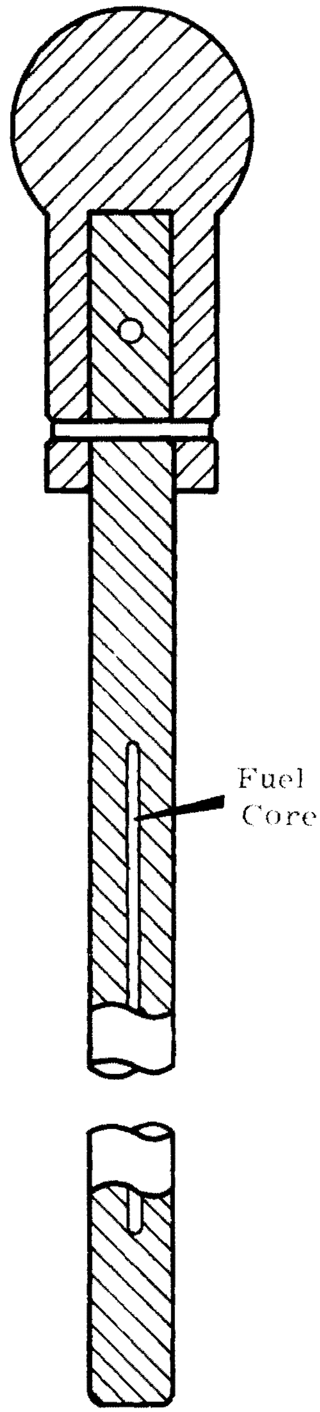


FIGURE 4.4-6
Solid Aluminum Flux Monitor Wand with a 0.020 inch Diameter
Phoenix Fuel Core

maximum heat flux would not exceed 1.87×10^5 BTU/hr-ft^2 if all the heat was transformed out one side of the rod into the coolant canal between the 2nd and 3rd fuel plates (approximately 1/4 of the circumference of the rod was assumed to be transferring which is very conservative). Also, a peaking factor of 3.5 times the average heat generation rate in the reactor (18.45×10^7 BTU/hr-ft^3) was assumed along the full length of the wand. The maximum nominal heat flux which is now used on the reactor is 8.2×10^5 BTU/hr-ft^2.

Handling Procedure

A handling tool is being designed which will allow the wands to be removed from the core after removing the upper grid and hopefully without removing the fuel boxes from their positions. If it is desired to reinsert wands into the fuel element flux wand wells while in the reactor, it will probably require lifting the fuel to facilitate insertion of the flux wand into the small hole.

Wands from selected locations will be pulled at various times during the experiment. Upon removing, the wands will be dropped into a fuel transport canister and transported into the reactor canal. They will then be moved to the underwater scanner at INC or shipped in a cask to Richland for analysis. Facilities are also available for analysis work.

4.4-5: Present Status of Monitoring System

Two mockup fuel elements are being fabricated by Nuclear Metals Division of National Lead which will contain 16 simulated dummy fuel plates. The simulated fuel assemblies will have the provisions to accomodate a flux monitor wand. These simulated fuel boxes will be used to evaluate the proposed flux monitor well and to test a handling device for removing and reinserting the wands. The handling device is designed based upon a flux wand handling device now used at the MTR. The flux wands are being fabricated at BNW; however, they will not have a fueled core for these tests.

With the completion of the above tests, the fuel boxes will be sent to Idaho for hydraulic testing in the fuel element hydraulic testing facility. These tests will be run at 140% of the rated MTR flow and the fuel assemblies will have the flux monitor wands in place.

4.4-6: <u>Flux Depressors</u>

Present calculations indicate that no special flux depressors will be needed to reach a reasonable power level within the reactor if the fuel core is tapered on the lower end of the plates. However, should any additional flux depressor be needed, they would have to be incorporated into the fuel box. A method which is used to suppress the flux at the bottom of the fuel in the ETR could possibly be used. This consists of placing 1/2 inch wide, 0.080 inch thick boronated stainless steel plate in the grooves immediately below the fuel plates. Any side suppression would have to be done by tapering the fuel core at the edges of the fuel plates as incorporating boronated flux suppressors into the fuel side plate would be very difficult.

4.4-7: <u>Handling and Storage of Unirradiated Fuel</u>

The expected 1-r radiation level on the fuel may pose some problems in handling and storage. However, experience at the BNW PRCF indicates that the radiation is quite soft and can easily be shielded. If necessary, the fuel might be stored in the shipping containers, although there is now an internal directive at INC against doing this.

Close inspection at INC will probably require some light shielding due to the close proximity of the inspector. Flow testing and loading of the fuel is not expected to require any special precautions.

4.5 - Core Physics Design and Performance Characteristics

4.5-1: MTR-Phoenix Core Design Evolution from 1964 to 1967

The possibility of using the MTR as a Phoenix fuel test bed was proposed by the AEC. It was suggested that after startup of the ATR (which would take over the work load of the MTR) there would be time for such an experiment.

In December of 1964 a proposal was made by Hanford Laboratories to use either a 4x9 or a 5x9 loading in the MTR. The use of 27% Pu^{240} Plutonium of 20 wt% in aluminum was postulated. Calculations showed that power peaking at the core reflector interface would be a problem and that a 3x9 core at 0.1 kg/ℓ of Pu would have a 3000 full power hour lifetime while a 5x9 core at 0.15 kg/ℓ would have about a 6000 FPH lifetime.

In June of 1965, R. S. Marsden of Phillips Petroleum reviewed the proposal in PTR-757, "Review of Proposal to Operate the MTR on HX-Pu Fuel." HX stands for high expsoure and denotes any plutonium isotopic mixture containing more than about 15% ^{240}Pu. The general tenor of this report is that a Phoenix loading is probably possible but that the proposed time schedule (fuel loading in the MTR during FY-1967) could not be met. A detailed design proposal would be necessary to really evaluate the difficulties and finally a Phoenix test would probably disrupt the existing research reactor work.

In response to this review, BNW prepared a more detailed design. The following few paragraphs are excerpts from that rough draft report:

"These constraints result in the following set of conditions:

1. 115^{o}F average water
2. Al clad and matrix
3. 20 wt% Pu in Al
4. The water channel thickness shall remain standard; i.e., 117 mils.

The experimental core must, of course, fit within the MTR test region (see Figure 4.5-1). The shaded areas represent the shim rods which may have either fuel or Be followers. A small core will have the advantage of less excess reactivity to control as well as a high power density, which leads to a short in-reactor time. On the other hand, the peak-to-average power values could easily be too high to permit full power operation. A large core will have more excess reactivity, but it will also require more rods for control. It will have a much flatter power distribution, but it will take longer to reach a specified percent burnup."

Certain core configurations appear to be more desirable than others. For example, symmetry should be maintained if possible, as this will allow one quarter of the reactor to serve as a unit for calculations. Cylindrical (idealized) geometry is a great advantage, since one dimensional burnup codes can then be used for preliminary calculations, with fewer approximations for reflector savings, etc.

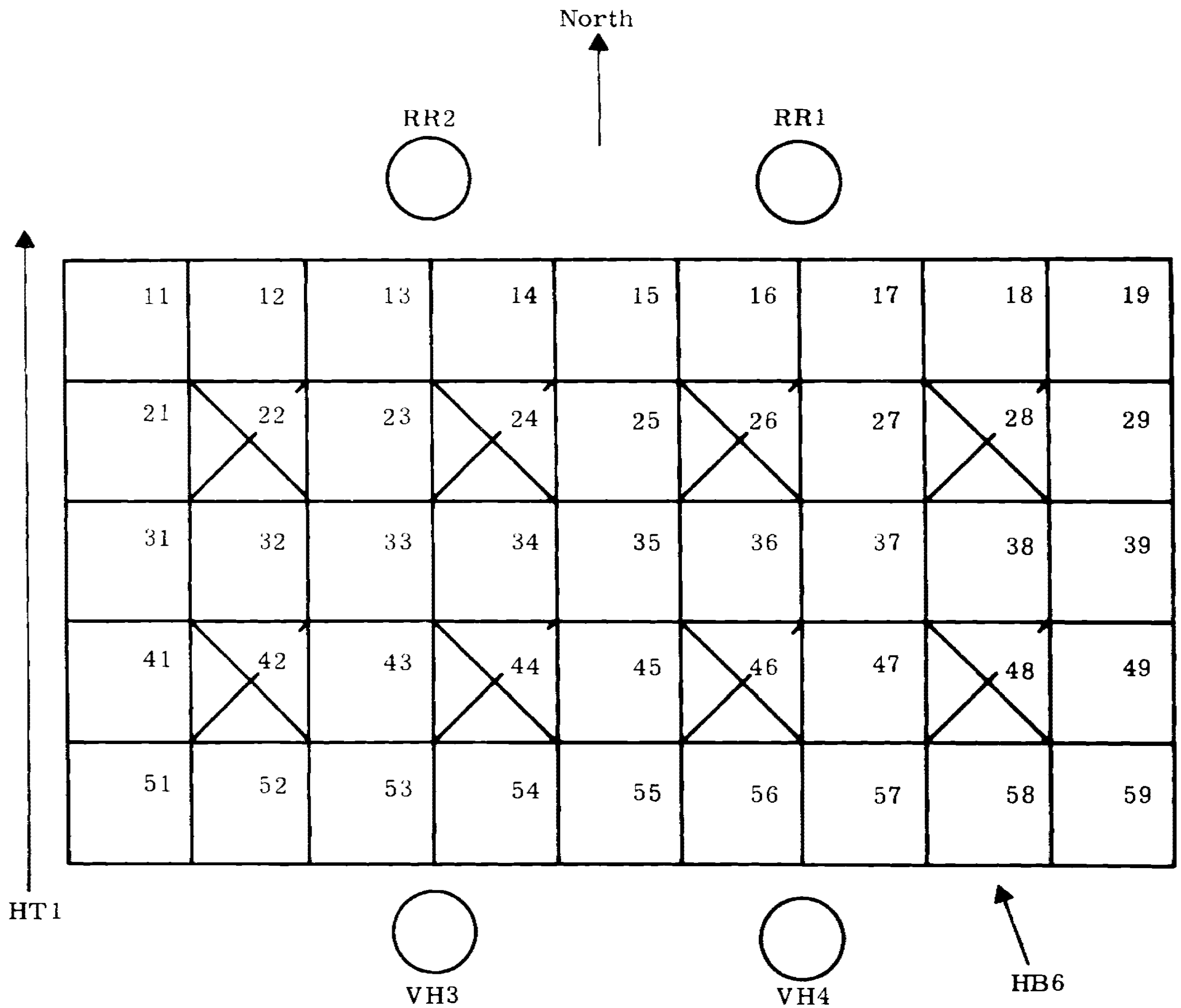


FIGURE 4.5-1
MTR Core Layout

Four possible full length (i.e., 23.5") core configurations have been examined. No detailed calculations were made. Instead, available nuclear data were used for extrapolation purposes.

The MTR control rods have been calibrated for a variety of core loads including the cycle 108 plutonium load. A 4-rod fully inserted worth of 0.26 $\frac{\Delta k}{k}$ is reported in IDO-16712. It will be assumed in this report that each in-core has a worth of 0.06 $\frac{\Delta k}{k}$.

The 3x3 core consists of 5 fuel elements and 4 control rods with fuel followers. There are no adjacent control rods so the 4 core rods must be almost fully inserted to overcome the initial reactivity of 1.226. The core could be run at the following power levels if the corresponding power distributions could be obtained.

MTR Power	$\hat{P}/\overline{P}$	
	9 Elements (Rods Out)	5 Elements (Rods In)
10 MW	1.85	1.10
8	2.35	1.30
5	3.60	2.10

The 3x3 core with all rods in would, without a doubt, have a much worse $\hat{P}/\overline{P}$ than 2 but might have an average value over the core lifetime of about 2. Calculations show that 50% burnup occurs in this core at about 23 FPD, or at 180 days at 5 MW. The advantage of this core is, however, that the 1.5 kg limit of fissioned fuel is not exceeded.

The 5x5 core has 4 adjacent control rods with Be followers and 4 adjacent locations (RR1, RR2, VH3, VH4) to be used for control and to lower the power spikes at the core reflector inner face. It was estimated that these 8 reflector rods can override the initial k excess of 0.264 hot and about 0.275 cold. If this is true, the entire 4x5 array is available as heat transfer area. The following table lists allowable $\hat{P}/\overline{P}$ for both the 25 box core (rods out) and the 21 box core (rods in).

Power	$\hat{P}/\overline{P}$	
	25 Elements (Rods Out)	21 Elements (Rods In)
40 MW	1.30	1.10
30	1.75	1.45
20	2.70	2.20

These values appear to allow a startup power level of at least 20 MW with an increase to 40 MW as soon as the power spikes at the core corners burn off. Assuming a 30 MW average power for the core lifetime, 50% burnup is achieved in less than 90 days. The 3.6 kg fissioned exceeds the current limit by about a factor of two, which is probably a reasonable extension. This core, like the 3x3, is very easy to convert to a cylinder for preliminary calculations.

The control rods could in theory be operated in two banks; central and reflector. At time zero the reflector rods would be fully inserted and the central rods move out until they are fully withdrawn. The reflector rods are now withdrawn perhaps 50% and the centrals reinserted until the core is just critical. Again, as the core burns, the centrals are withdrawn. This mode of operation should yield a reasonably flat power distribution and a control rod configuration which is reasonable to calculate.

The 5x7 core utilized all available control rods although the outer ones must be worth a good deal less than the central rods. The full length (23.5") has the following peaking factors at rods-out and rods-in configurations:

Power	$\hat{P}/\overline{P}$	
	35 Elements (Rods Out)	27 Elements (Rods In)
40 MW	1.80	1.40
30	2.40	1.85
20	3.60	2.80

This core is plagued by the 5 kg of fissioned Pu required for 50% burnup as well as a minimum in-reactor time of 90 days assuming 40 MW at all times.

The 5x9 core is the largest possible and, of course, has the best heat transfer capability; however, the 6.5 of fissioned Pu is about a factor of 4 over the present limit.

The crude reference core that was selected after review of the above calculations was the full length 5x5 core. The possibility of reflector control was the major factor in this decision. Other points in favor of this core over a shortened larger core were:

1. Smaller fabrication cost (fewer units to manufacture).
2. A core whose real size and hydraulic characteristics closely resemble the standard 3x9 ^{235}U core.
3. Almost cylindrical geometry for calculational simplicity.
4. The use of only four (standard number) control rods with fuel followers.

This design also does not require that the 42 position be reconverted to be a movable rod. The loading in the 42 position could be changed in a stepwise fashion as will be done in VH3 and VH4.

The fuel plate for this core would have either 40 or 60 mil meat with 20 mil Al clad on each side. These two thicknesses with a 0.117" water gap give a gross M/W ratio of 0.93 and 1.12. The M/W of 1.0 calculation by Holeman and Hofmann [(15)] assumed a meat thickness of 58 mils. No major change in reactivity is expected in changing to either 40 or 60 mils.

In the fall of 1965, this design was presented to Phillips at an Idaho Falls meeting and was rejected mainly on two counts. First, the "extra" control spots were not large enough to accomodate a significant amount of poison and second, such a configuration would completely eliminate the possibility of using the beam tubes for a continuing physics program. At that meeting, D. R. de Boisblanc suggested a small Pu zone driven by a standard ^{235}U zone with the core in the standard 3x9, 1, 2, and 3 row location. PNL agreed to look into this and determine how large a Pu zone was large enough.

As before, the following paragraphs are clipped from the next design document, BNWL-CC-722, "Design Calculations for a Partial Pu-MTR Phoenix Experiment."

> "The design task then became one of finding the smallest Pu zone within a standard core which would meet the objectives of the test. The main criteria to be followed were:
>
> 1. The typical Phoenix neutron spectrum must cover at least half of a fuel box from which burnup samples could be taken.

2. The core design must be as symmetric as possible to simplify the multidimensional core analyses. Failure to meet this criterion could mean the inability to properly analyze the experiment.
3. An adequate shutdown margin must be achieved with a single rod fully withdrawn.
4. The power capability of the core should be near 40 MW to allow more or less standard operation of the beam hole experiments.
5. There must be enough Pu present to markedly affect the reactivity versus burnup relationship."

A single Pu fuel box in a standard core could be symmetrically located and controllable but it would hardly provide a suitable spectrum since the thermal mean free path in the Pu zone is about 2 inches and the fuel box is about 3 inches square. In addition, the presence of an isolated 400 grams of Pu in a core containing 5.2 kg of ^{235}U would hardly cause a marked change in its neutronic characteristics.

The next largest symmetric system is a 3x3 Pu zone. This configuration would have a 3x3 U zone on each end to form a 3x9 array which could contain either 4 or 8 control rods. The 3x9 array was selected to provide core spectrum neutrons to the HT1 beam hole and still allow moderated neutrons to reach the HB6 tube (see Figure 4.5-1. The characteristics of the fuel in each zone are shown in Table 4.5-I. The core would contain about 3.6 kg of Pu and 3.6 kg of ^{235}U so that the presence of the Pu does affect the reactivity-time behavior (Figure 4.5-2. The approximately 9" Pu zone dimensions would appear to be sufficient to

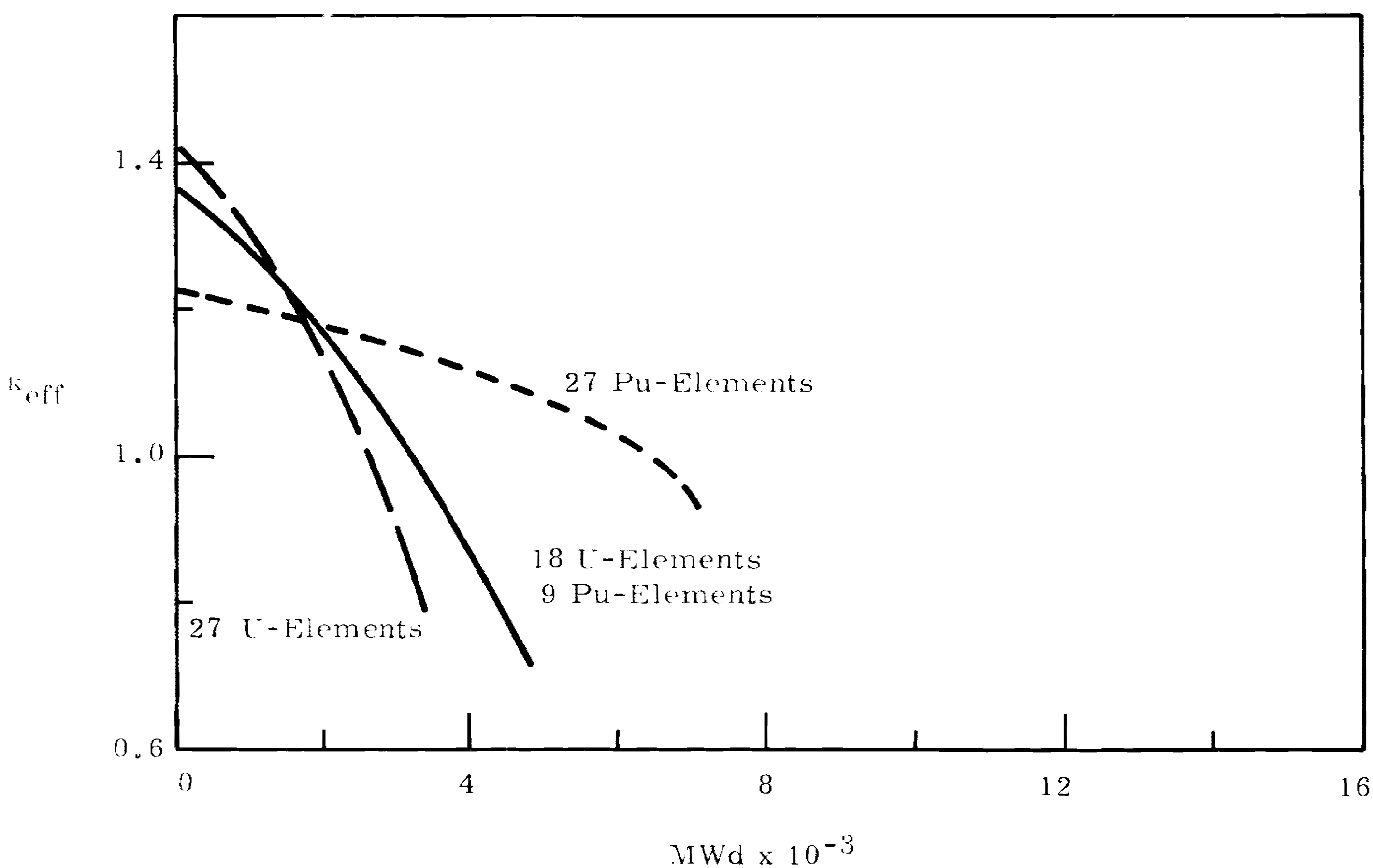


FIGURE 4.5-2
MTR-Phoenix Fuel Possible Loading Studies

produce an equilibriated spectrum over the entire central box. This is demonstrated in Figure 4.5-3, which shows a burnup cross section traverse calculated by means of one and two dimensional diffusion theory.

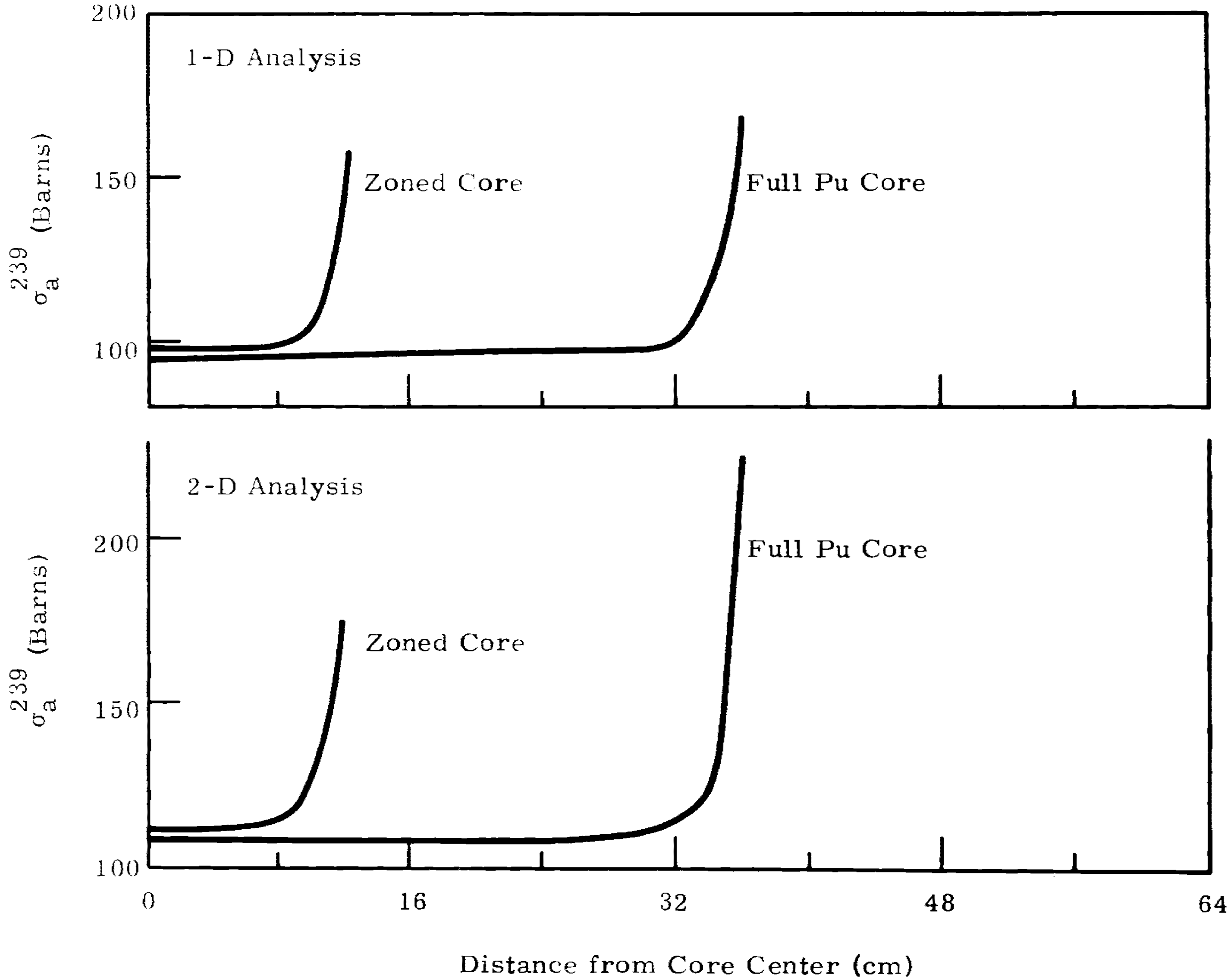


FIGURE 4.5-3
Comparison of Zoned and Fuel Core MTR-Phoenix Fuel Loadings

TABLE 4.5-I

Characteristics of Core Zones

	Pu Zone	U Zone
Plates/box	16	19
Meat thickness	0.040"	0.020"
Clad thickness	0.020"	0.015"
Water channel thickness	0.119"	0.117"
Gross M/W	0.9219	0.6412
Grams fissile mat'l/box	313	200
At% Pu^{240}	20	

TABLE 4.5-II

Results of 2-D Calculations

Fuel	Configuration	Rods	k_{eff}
Pu	Full 3x9 core	No	1.225
Pu	Full 3x9 core	8	0.948
Pu	Full 3x9 core	4 core	1.109
U ("Phillips" core)	Full 3x9 core	No	1.249
U ("Phillips" core)	Full 3x9 core	4 core, 4 ref.	0.891
Pu and U	Zoned 3x3 Pu	No	1.264
Pu and U	Zoned 3x3 Pu	8	0.904
Pu and U	Zoned 3x3 Pu	4	1.097
Pu and U	Zoned 3x3 Pu, 165g U	4	1.068
Pu and U	Zoned 3x3 Pu, 200g U	4 (2xCd)	1.0687

A 3x3 Pu zone contains a large number of control rods, either 2 out of 4 or 4 out of 8. This is fairly typical of most compact cores which generally have rods on a rather tight spacing. Hence, the rod problem is not unique to this design and would also be present in a full Pu core design.

The curve on Figure 4.5-4 shows that more than 50% burnup occurs at 3000 MWD. Figure 4.5-2 shows that this exposure could be reached.

The following tentative conclusions were drawn from these design calculations:

1. A meaningful Phoenix fuel burnup experiment could be carried out in a partially Pu loaded MTR.
2. The power spiking problem in this partial Pu core would present a very difficult design problem.
3. While a 3x3 Pu zone would have an observable effect on the reactivity time behavior of the core, a fully Pu loaded core should yield a better experiment.

In the early summer of 1966 the AEC made the decision to support the 3x9 full Pu loading with 8 control rods in the core. This was based first on the preceeding Figure 4.5-2 , which shows a pronounced Phoenix effect for a full Pu core and only a slight effect for the partial load. Second, Table 4.5-II predicts that a 3x9 full Pu core in row 1, 2, and 3 (i.e., 4 control rods) would not have a sufficient control margin so a shift to the central 3x9 location was made. It should be noted that even the zoned core would have the required all 8 control rods and the central location.

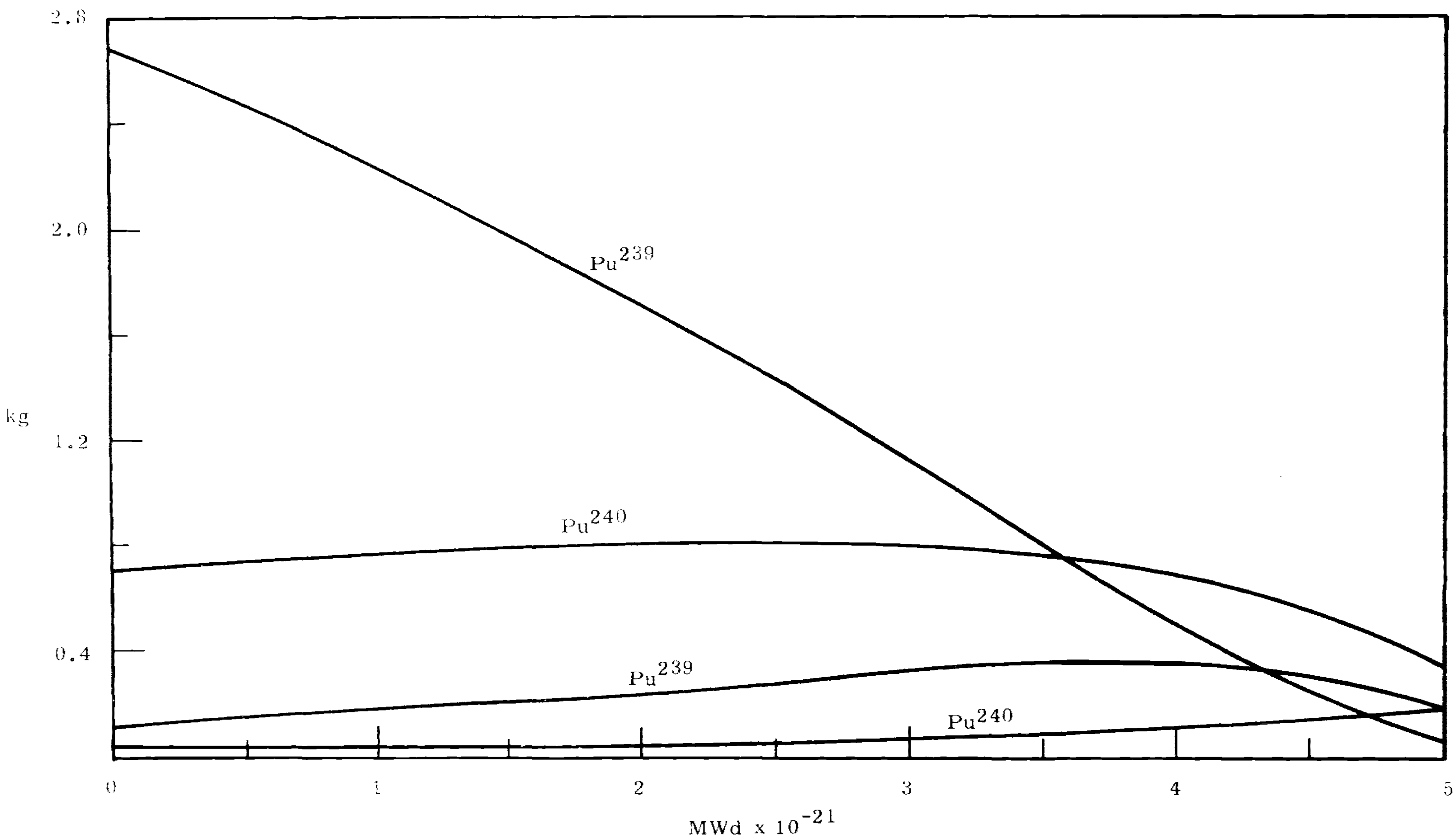


FIGURE 4.5-4
Plutonium Isotope Variations in a 3 x 3 Zoned Core Loading of a 3 x 9 MTR Core

The only change made within the last year is the use of tapered fuel meat to lower the power spikes at the bottom of the core. A detailed discussion of this design feature is contained in a latter section on power profiles.

4.5-2: Burnup Prediction

The best burnup data currently available for the full 3x9 core operation is presented in Figures 4.5-5 through 4.5-9. Figure 4.5-5 shows the reactivity as a function of burnup at 40 MW. This calculation does not include the equilibrium Xenon or Samarium. If the calculation currently incluced every other detail, the core lifetime would be between 80 and 85 days.

Comparison with experiment shows that the early PRCF-Phoenix calculations were high in k by about 5.5%. Since the same methods were used for the MTR fuel burnup predictions, the same sort of error is expected in Figure 4.5-5. The predicted full power capability is thus cut to between 55 and 60 days; however, part of the 5.5% difference between the PRCF calculation and experiment is due to experimental differences in mocking up the MTR reflector and a final burnup prediction is yet to be made after the theory-experiment correlations have been completed.

Figure 4.5-6 shows the burnup of ^{239}Pu for the various calculational zones of the full Pu core. The reactor has quarter core symmetry with calculational zone 1 being in the center. As can be seen, a burnup of 80 days yields between 47% and 63% initial ^{239}Pu burnup for zones 4 and 1. At 60 days the burnups are 36% and 49%, respectively.

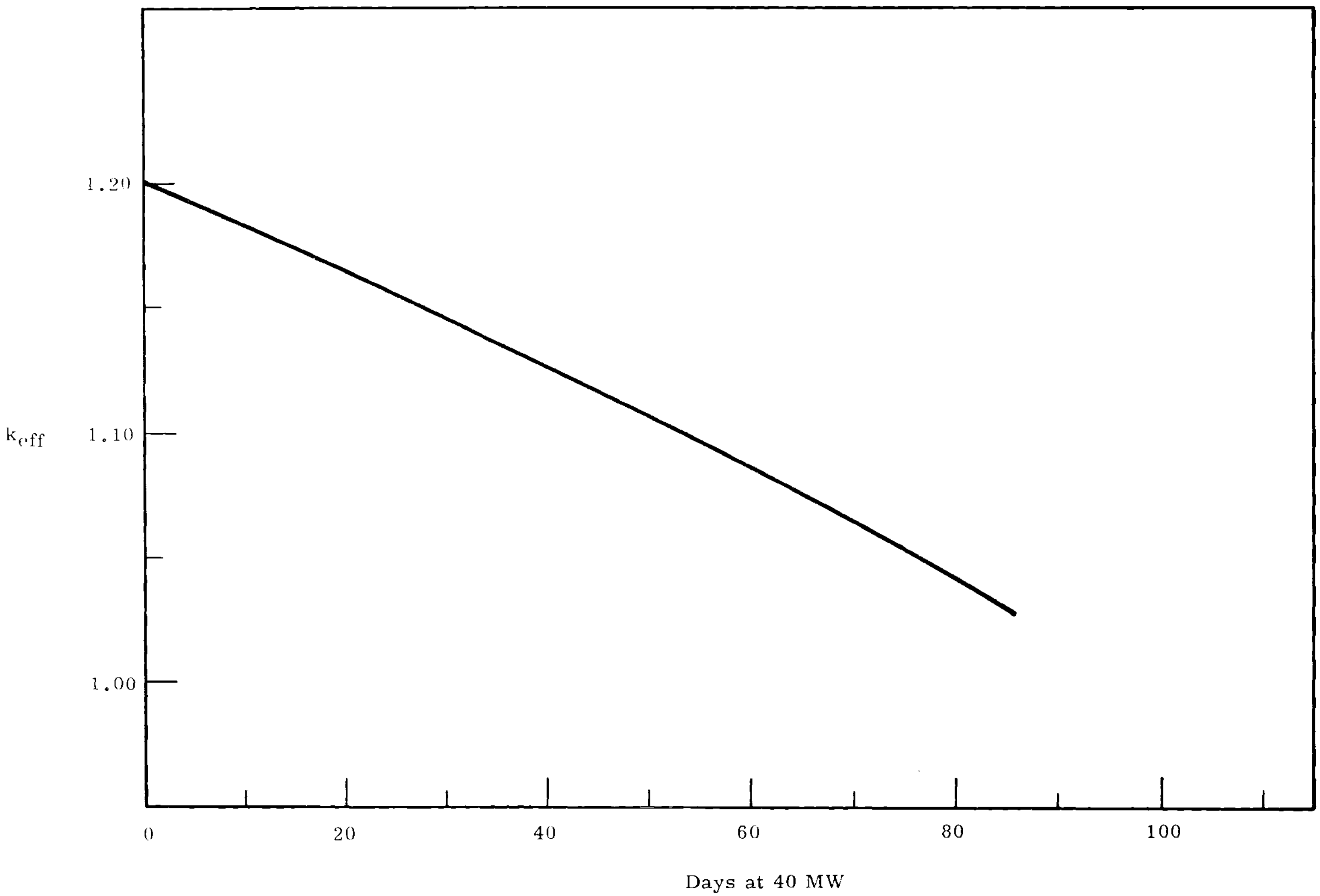


FIGURE 4.5-5
MTR-Phoenix Fuel Burnup Calculation (Full 3 x 9 Core Operation)

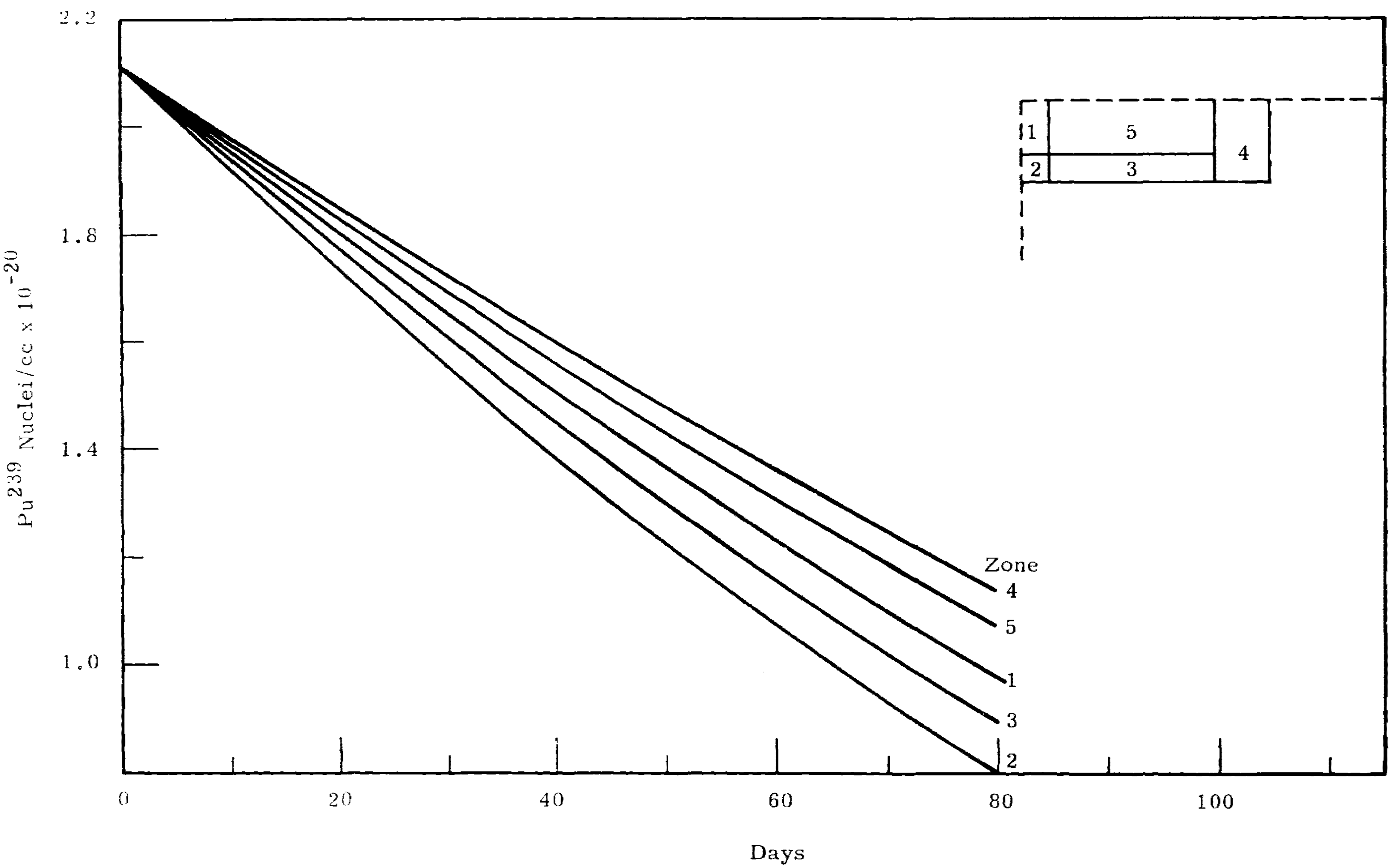


FIGURE 4.5-6
MTR-Phoenix Fuel Burnup Calculations (3 x 9 Full Core Loading Burnup Variations by Region)

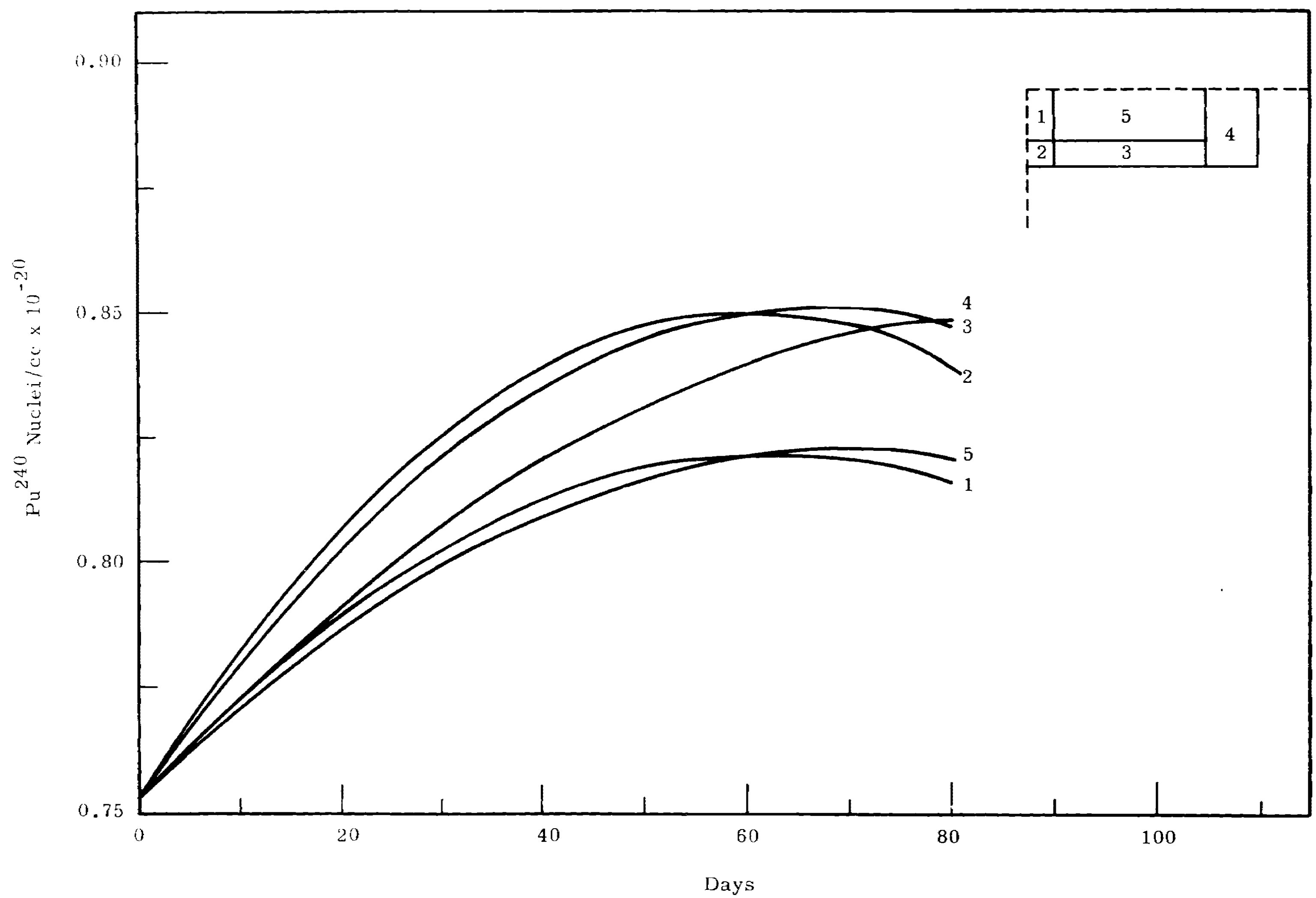


FIGURE 4.5-7
Variation of Pu^{240} by Region in 3 x 9 Loading of Phoenix Fuel

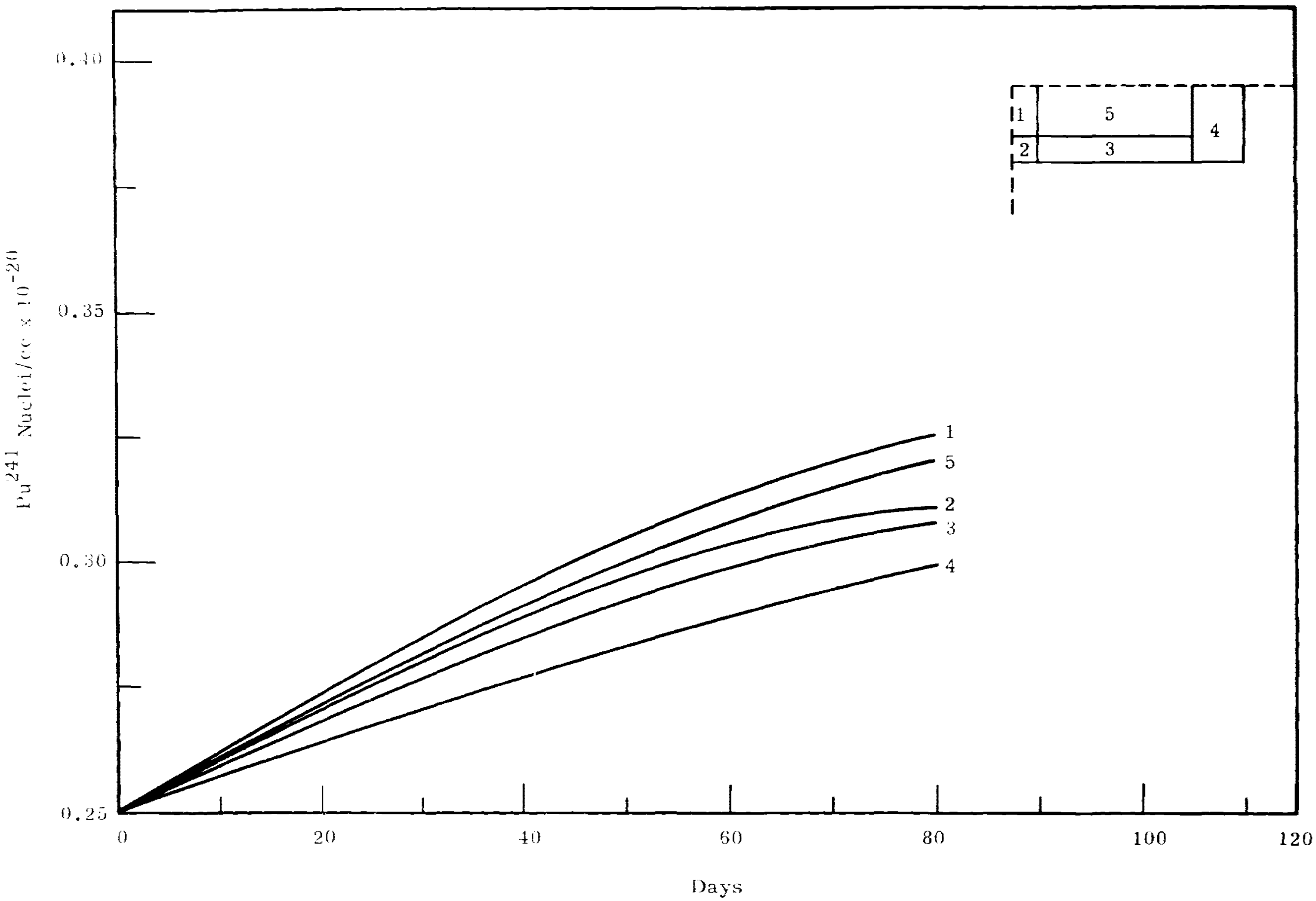


FIGURE 4.5-8
Variation of Pu^{241} by Region in 3 x 9 Loading of Phoenix Fuel

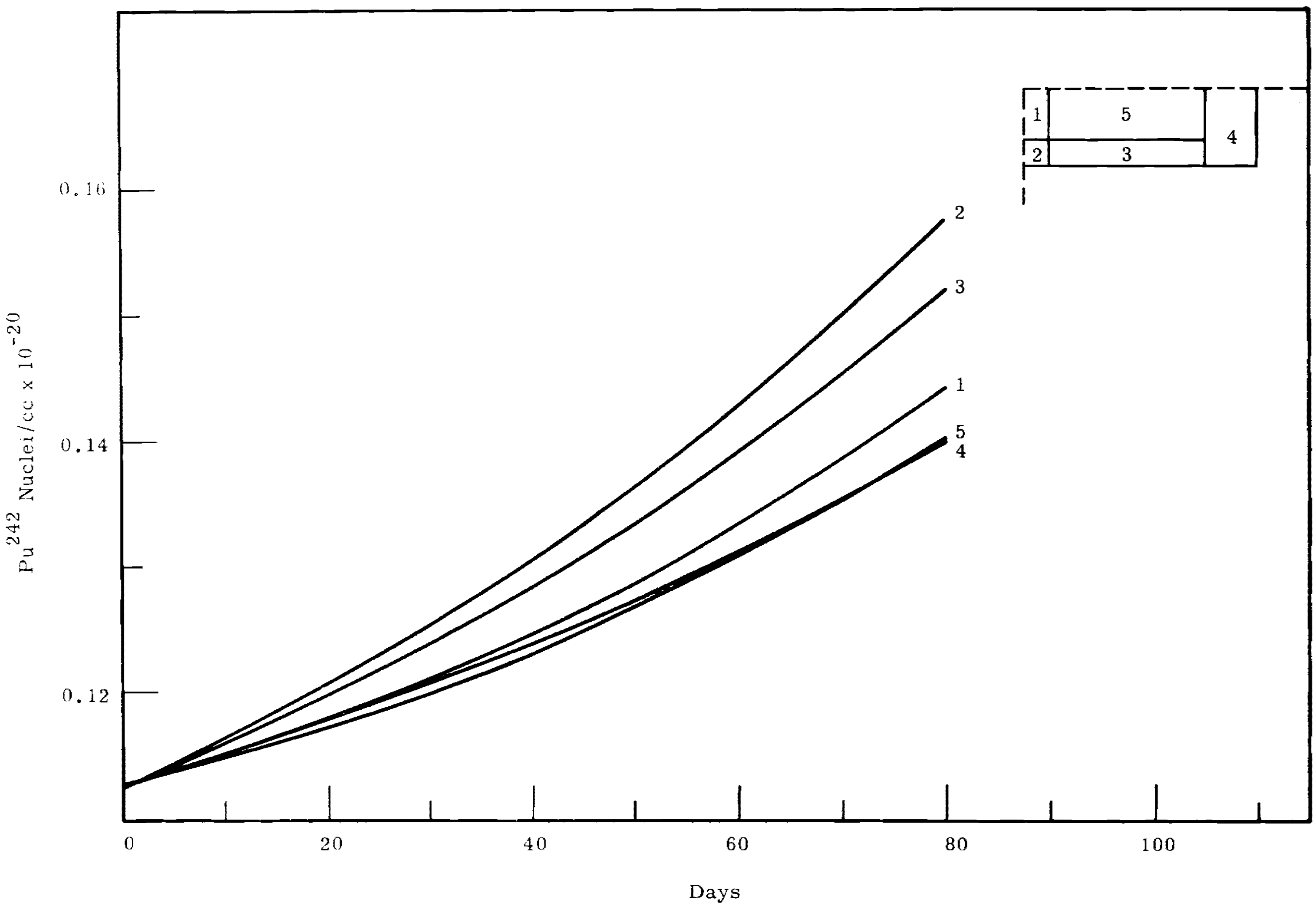


FIGURE 4.5-9

Variation by Region of Pu^{241} in 3 x 9 Loading of Phoenix Fuel

In Figure 4.5-7 the different burnup characteristics of ^{239}Pu and ^{240}Pu are evident. The net ^{240}Pu as shown here is the product of ^{239}Pu burnup producing ^{240}Pu with a spectrum dependent alpha value and ^{240}Pu burnup with a different spectrum dependance. The accurate prediction of effects such as these is an important portion of the Phoenix program.

These burnup calculations utilize the 2-D multigroup code ASSAULT supplied with cross sections from THERMOS and HRG. The multigroup cross sections are recalculated at several steps during the burnup.

Figure 4.5-10 shows the variation in reactivity lifetime for some possible changes to the MTR-Phoenix fuel. The cross section set used here should be better than that used in the previous calculation but this has yet to be verified. The initial k has dropped about 1.5%, the calculational error should therefore decrease from 5.5 to about 4.0. Case 1, the standard, still results in a lifetime of about 60 days. As more fuel is added through thicker fuel meat, the lifetime increases but the percent of initial ^{239}Pu burned remains relatively constant. Case 4 is a softer spectrum, more water, and standard fuel meat. This core would start with a higher excess k which is harder to control and drop faster than the standard. The percent burned is higher here by about 3%. Cases 5 and 6 show the result of lowering the core load with the result that 5% in inventory is worth about 4 days or about 7% in lifetime for this system.

If the MTR fuel loading in the PuAl core was increased from 20 wt% to say 25 wt% Pu in Al (i.e., a 25% increase in loadings) the lifetime should increase about 20 days with a few percent gain in the ^{239}Pu burnup. This possibility is being studied, although at this time it is expected

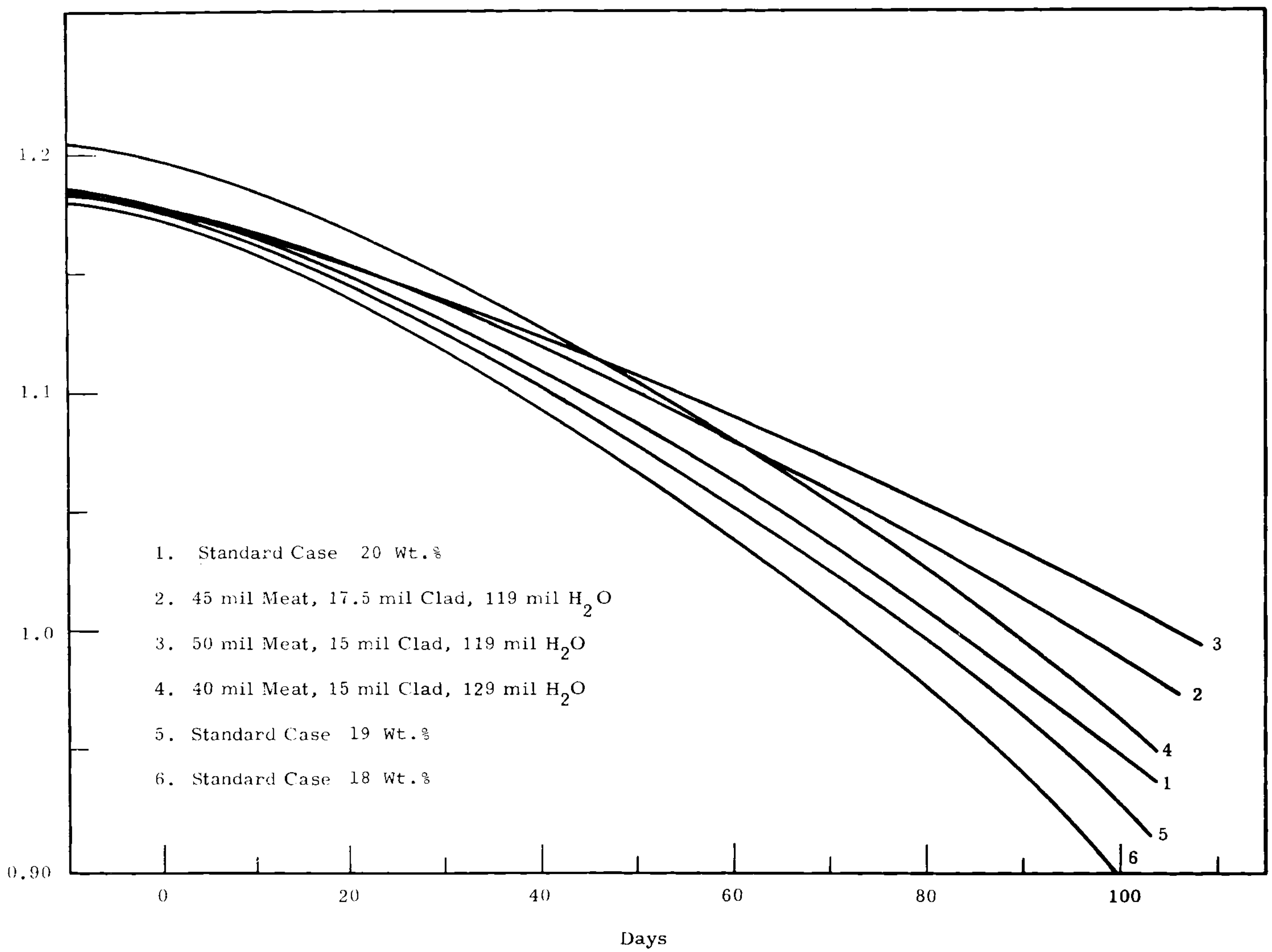


FIGURE 4.5-10
Effect of Fuel Design Variations on Core Burnup Reactivity

for several reasons that the plutonium concentration in the PuAl will be specified at a value of about 21 $\pm$ 1 wt%.

4.5-3: Control Margins

The latest calculations show an initial k of about 1.20; if this is correct, the full shutdown k should be about 0.85. The 1.20 is more likely really about 1.16, which gives an even larger shutdown margin. Expected banked critical position is about 51% withdrawn, or about 15 inches withdrawn. This is near the normal MTR operating range.

If the total rod bank strength of about 0.35 Δk was evenly divided between all 8 rods, the average rod strength would be 0.04. Assume that some rods have twice the average strength or 0.08, it appears that at least three of these strong rods could be stuck out and the reactor could still be scrammed. The stuck rod experiments in the PRCF mockup agree with this result.

Burnup of the Cd poison in the control rods could be a potential problem due to the longer core life of the Phoenix experiment. It is, however, standard practice of the MTR to use the same control rods in from 2 to 3 ^{235}U cores which results in exposure comparable to the Phoenix core. A crude calculation indicates that as much as one quarter of the Cd of the rod tips might be burned in 3000 hours at 40 MW. An effective reduction of from 40 mils to 30 mils in Cd thickness should have no noticeable effect on the control worth of a rod. The document, IDO-16712, "MTR Shim Rod Calibrations," makes use of both burned and fresh control rods and there is no noticeable deviation in the data between cases.

The worth of the regulating rod has not yet been calculated, but the experiments point to a very small value. Procedures for operation of the MTR with either a small worth regulating rod or use of a shim rod for control will be discussed with Idaho Nuclear in the near future.

4.5-4: Power Profiles - Problems and Current Solutions

Power peaking at core boundaries is recognized as a problem in all compact cores. The Phoenix cores with a higher than usual fuel loading are subject to more of a peaking problem than cores with a lower loading. Early in the MTR-Phoenix program, power profiles such as the partial Pu loading shown in Figure 4.5-11 were calculated. This shows power peaking along the long axis of the MTR core. The uranium zone has a more thermal spectrum than the Pu zone. The influx of thermal neutrons from the soft zone to the hard Pu zone caused an effect very reminiscent of that experienced at the U core - Be reflector interface of 36.5 cm on the figure.

The Pu core - Be reflector or the Pu core - top or bottom water reflector interface also produces such an effect. Figure 4.5-11 is a one-dimensional representation of a somewhat idealized situation. As the core design takes on a more definite shape, three dimensional calculations become necessary to account for all the details.

Figures 4.2-4 in Section 4.2 and Figure 4.5-12 are power profile calculations down specific channels in the MTR and experimental data taken from the PRCF-Phoenix core mockup. Figure 4.2-4 shows the variation of power down the fixed fuel bundle loaded on the core edge between two shim rods. The adjacent control rods were inserted to 30cm. The effect can be clearly seen on the external fuel plate. Notice the overcalculation of the lower power peak. The profile shown in Figure 4.5-12 is for the fuel follower on one of the eight control rods. In this case, the top 29cm of the follower fuel is adjacent to the stationary fuel while the remainder hangs into the lower reflector. The power spikes shown at the

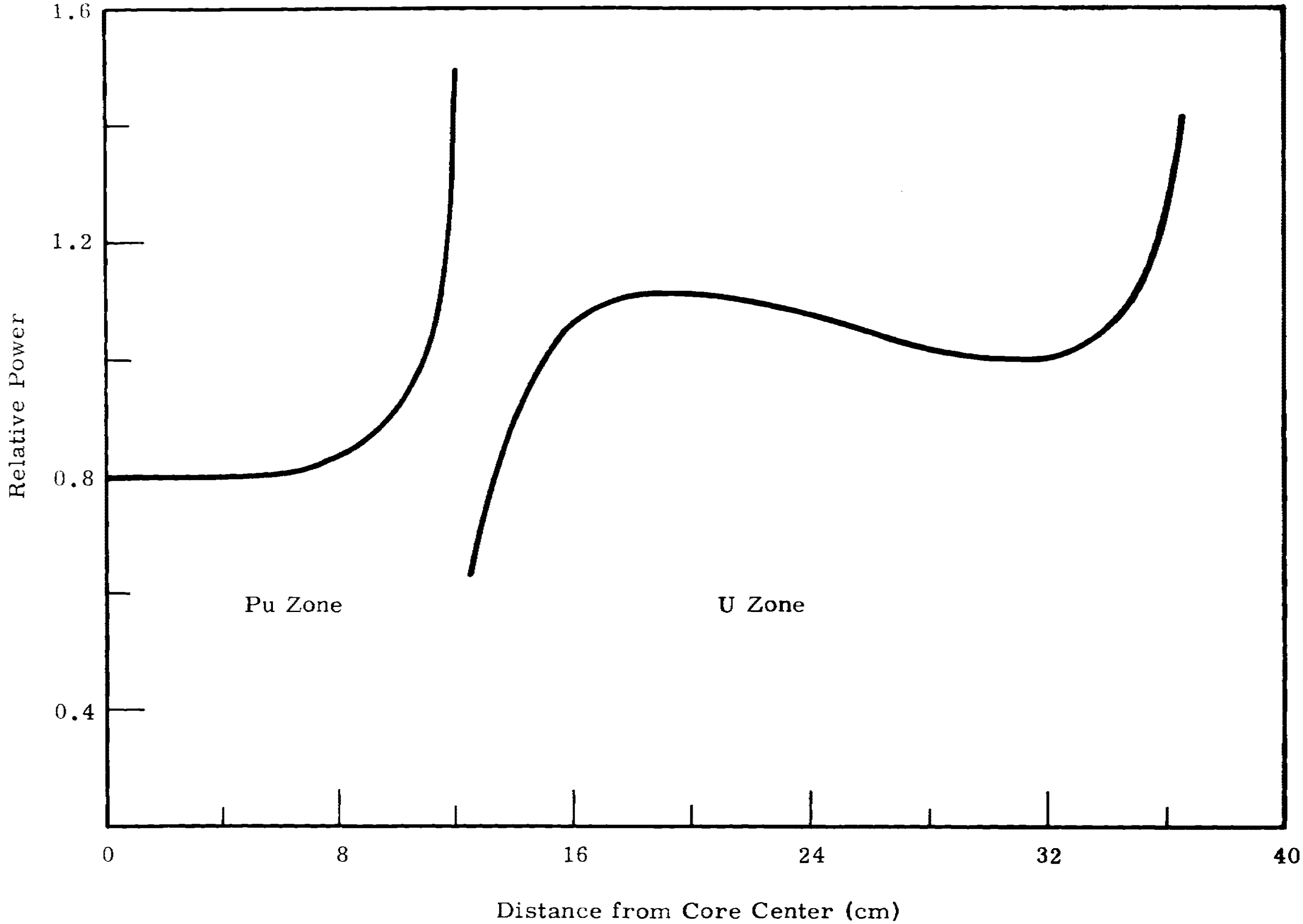


FIGURE 4.5-11
Partial (3 x 3 Zone) Phoenix Fuel Loading Showing the Power Peaking at the U-Pu Boundary

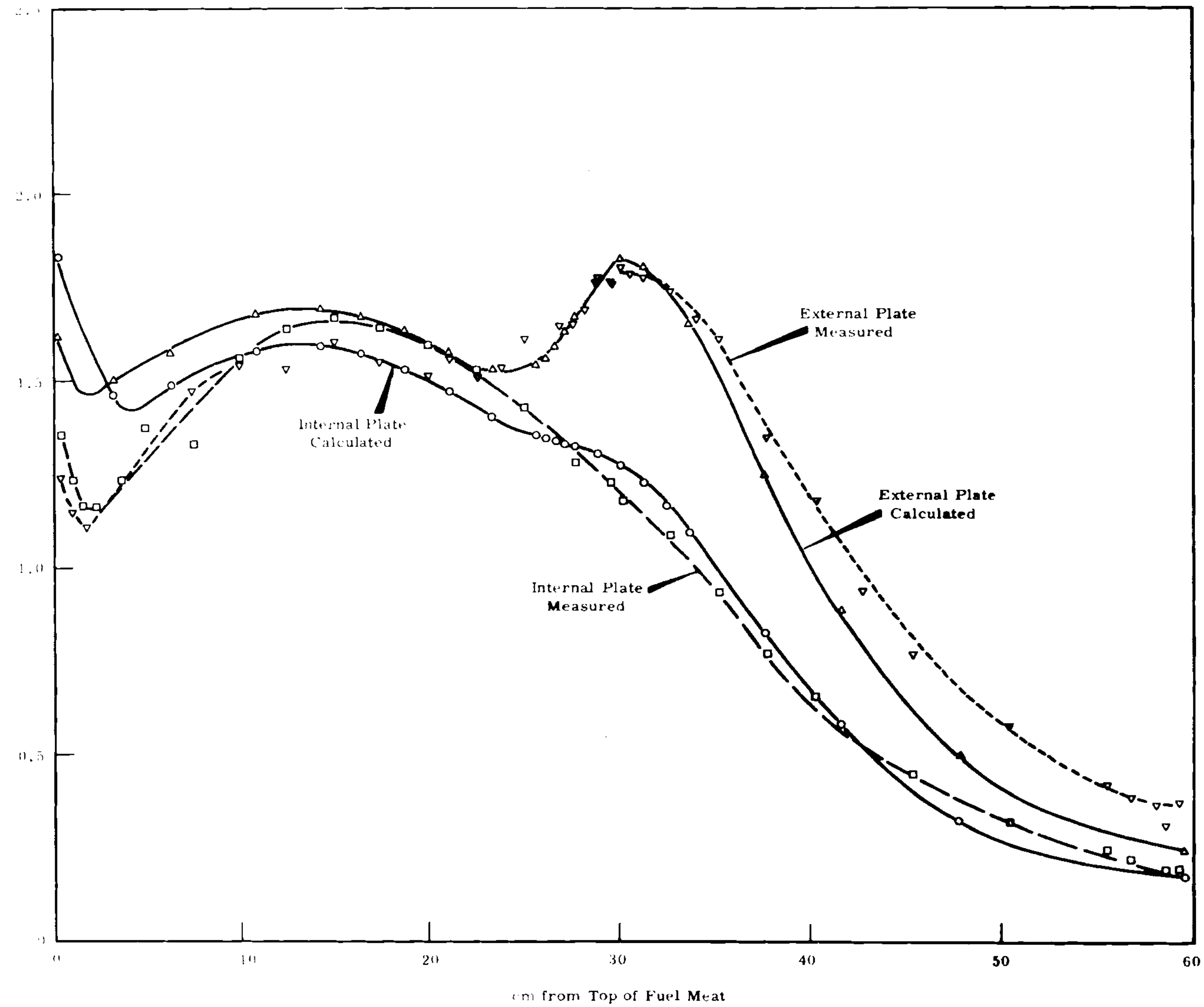


FIGURE 4.5-12
Power Distribution in a Fuel Plate of the Shim Rod Fuel Follower

top of the follower fuel are due to the influence of the large amount of water in the hollow poison section immediately above it. These curves have been normalized such that the area under each pair (calculation and experiment) is equal. The ratio of the inner plate curve to the outer plate curve is based on the calculation.

The curves shown in Figures 4.2-4 and 4.5-12 are from a 2-energy group 3-dimensional calculation of one quarter of the MTR core. This calculation indicates that the peak power density occurs at the bottom of the fuel element, adjacent to the Be reflector, and in the center of the core along the long dimension. The ratio of the local power at that point to the core average power is 3.44. Comparison to the experiment as in Figure 4.2-4 indicates that this peak should be lowered to about 2.6. The power bump in the fueled rod followers is a magnitude of 3.08 but is not limiting because of its more favorable hydraulic position.

The peak at the bottom of the stationary fuel was eliminated by the use of tapered fuel plates (see Figure 4.2-3). The taper is only on the fuel meat, and is not visible externally. The length of the taper to a point may vary between 1/2 inch and 1-1/2 inches with very little experimental difference between them. This scheme places the peak power density at a calculated point to reactor average ratio of 2.44, and occurs about 6cm up from the bottom of the fuel. Figures 4.2-4 and 4.5-12 show that we have good agreement with experiment in this area. The peak in the control rod fuel follower is still not limiting, although the power density is higher.

4.5-5: Uncertainties in the Prediction of the Nuclear Characteristics

The main problem with calculations on the Phoenix core arises from the extreme heterogeneity of the system. The sharp flux spikes and rapidly varying spatial spectrum are a result of this problem. The Pu itself with the very spectrum dependent cross sections is, of course, an added difficulty. As these effects tend to multiply, the k error of 5.5% is not too bad for preliminary calculations.

Experience has shown us that the use of more energy groups will lower k by about 1.5%. We expect an equal amount from a better geometrical model of the boundaries and perhaps another 1% or so from a very detailed calculation of the spectrum at the core reflector interface. This would leave us within the neighborhood of 1% which is nearing the limit of the experiment because of such things as fuel inhomogeneities, bubbles in the water, and uncertain reflector composition.

The use of the PRCF-Phoenix mockup experiment changes the role of the calculations to a tool for the conversion of experimental results. to the predicted MTR conditons. However, a program to improve the methods is continuing to better understand the problem so as to minimize the uncertainties and to develop the Phoenix fuel calculational techniques for the application studies.

4.6 - Thermal Hydraulics of the MTR-Phoenix Core

4.6-1: Introduction

Thermal and hydraulic calculations for the Phoenix core in the MTR have been completed based on power profiles calculated by Engineering Physics personnel using a three-dimensional neutron diffusion code (WHIRLAWAY). In addition, the experimentally evaluated flux shapes, as determined from the MTR-Phoenix fuel simulation in the Plutonium Recycle Critical Facility (PRCF), were used to modify the power at the higher power peaks. Reassessment of these results will be required as more experimental data becomes available and can be incorporated into the study.

The initial thermal and hydraulic conditions for the Phoenix core were developed from the standard MTR core to insure design characteristics within the capability of the MTR facility. Film buildup due to corrosive influence was based on a 3000 MWD operation at full power. Since burnup data were not available, it was assumed that the initial power peaks existed throughout the time increment, which is, of course, a pessimistic assumption. Also, head loss calculations through the end fittings of an MTR fuel assembly involve the use of an approximate method based on data in Reference 16 in order to obtain satisfactory agreement with actual MTR operating data. A mockup of the Phoenix core is being fabricated and will be used in the HTF (Hydraulic Test Facility) at INC to more closely fix these coefficients and secure other pertinent hydraulic data.

Hot channel analysis included incorporation of various uncertainty factors into the study. The uncertainty factors used were those developed at INC for a standard MTR core. This assumes comparable fabrication techniques and design specifications for the Phoenix core. However, development of factors unique to the Phoenix core will be made when design specifications have been firmly established. It is expected that they will have a minor effect on the final results of this effort.

4.6-2: Method of Analysis

"Macabre II," a steady-state thermal and hydraulic digital computer code developed at Idaho Nuclear Corporation, was used in this investigation. The code computes the heat split between the two coolant channels adjacent to a fuel plate allowing variation in thermal resistances for both conduction and convection in the direction normal both to the fuel plate and coolant flow. The axial and azimuthal conduction within the plate is disregarded but was accounted for in the uncertainty factors. The Dittus-Boelter forced convection heat transfer correlation equation and the kinetic energy and pressure energy equations are solved exactly as the demanded convergence on pressure loss and mesh distance goes to zero.[13] The oxide accumulation due to corrosive influence is calculated over a specified length of time with an added feature of analyzing hot spots caused by oxide flaking. The oxidation rate is a function of the aluminum or oxide to coolant interfacial temperature, acidity enrivonment, time, and current oxide thickness. Since the boehmite corrosion product formed and retained on the aluminum heat transfer surface is a significant resistor to heat transfer, the code offers an attractive package for analyzing long reactor operating cycles.

The code was originally written to analyze thermal and hydraulic conditions in the ATR core and was not compatible with the MTR fuel element design. Code modifications were made to make it suitable for this purpose. The MTR fuel element is fitted at each end to box shaped reducers, which, in turn, are attached to an upper and a lower grid plate in the reactor. This fueled assembly has coolant flowing over its outer surfaces as well as between the fuel plates within the assembly. The modified version of "Macabre II" is used to calculate both the flow through the assembly and its outer surfaces and is especially useful for shim rod applications. The addition of the Bernath and Jens' and Lottes' correlations in subroutines to the code make it possible to obtain incipient nucleate boiling data and burnout heat flux comparison at any desired point along the core.

4.6-3: <u>MTR Operating Limits</u>

The major thermal and hydraulic limits placed on MTR operation are summarized as follows:

- The nominal heat flux* at the hot spot in the hot channel should not exceed $1.0*10^6$ BTU/hr-ft^2.
- There should be a 95% confidence level that subcooled incipient nucleate boiling does not occur in the reactor. This is defined to be two standard deviations below the predictions obtained using the Jens' and Lottes' correlation.
- The maximum reactor heat flux should be at least three standard deviations less than the burnout heat flix as predicted from the Bernath correlation and calculated at the reactor trip power.
- Temperatures should be maintained at levels that will safeguard the structural integrity of the core.
- The maximum nominal power level is 40 MW_t when permitted by incipient nucleate boiling limits and heat flux conditions.

---

* Nominal heat flux is defined to be the heat flux calculated by standard procedures without the influence of core uncertainty factors.

- Inlet pressure at the reactor vessel will not exceed 75 psig.
- The differential pressure across the reactor core will not exceed 50 psi nor will it be less than 30 psi.
- The inlet coolant temperature should not exceed 130°F. The minimum temperature may be limited from environmental considerations since the heat sink is the atmosphere. However, the reactor currently operates with an inlet temperature of 105°F[(16)] at a power level of 40 MW_t.

The thermal and hydraulic limits listed above are consistent with operating limits established by INC.[(17)] However, not all are within previous operating experience. For example, the nominal heat flux at the hot spot in the hot channel has been limited to 8.2×10^5 $BTU/hr\text{-}ft^2$ instead of 1×10^6. Where deviations from standard operating experience such as this occur, graphical plots will be presented to show the effect on reactor operation of changing the given limit.

4.6-4: Power Distributions

The power distribution shows maximum power peaks occurring at the bottom of fuel elements number 25 and 45 along the center of the elements. The calculated peaking factor at this location is 3.47 but was modified to 2.44 to agree with results from the experiment in the PRCF (Figure 4.6-1). Shim rod positions 24, 26, 44, and 46 show maximum power ratios of 3.08 which is less limiting than the stationary fuel element. Therefore, fuel element number 45 was selected as the critical element upon which to base this investigation. The experimental investigation in the PRCF confirmed this decision.

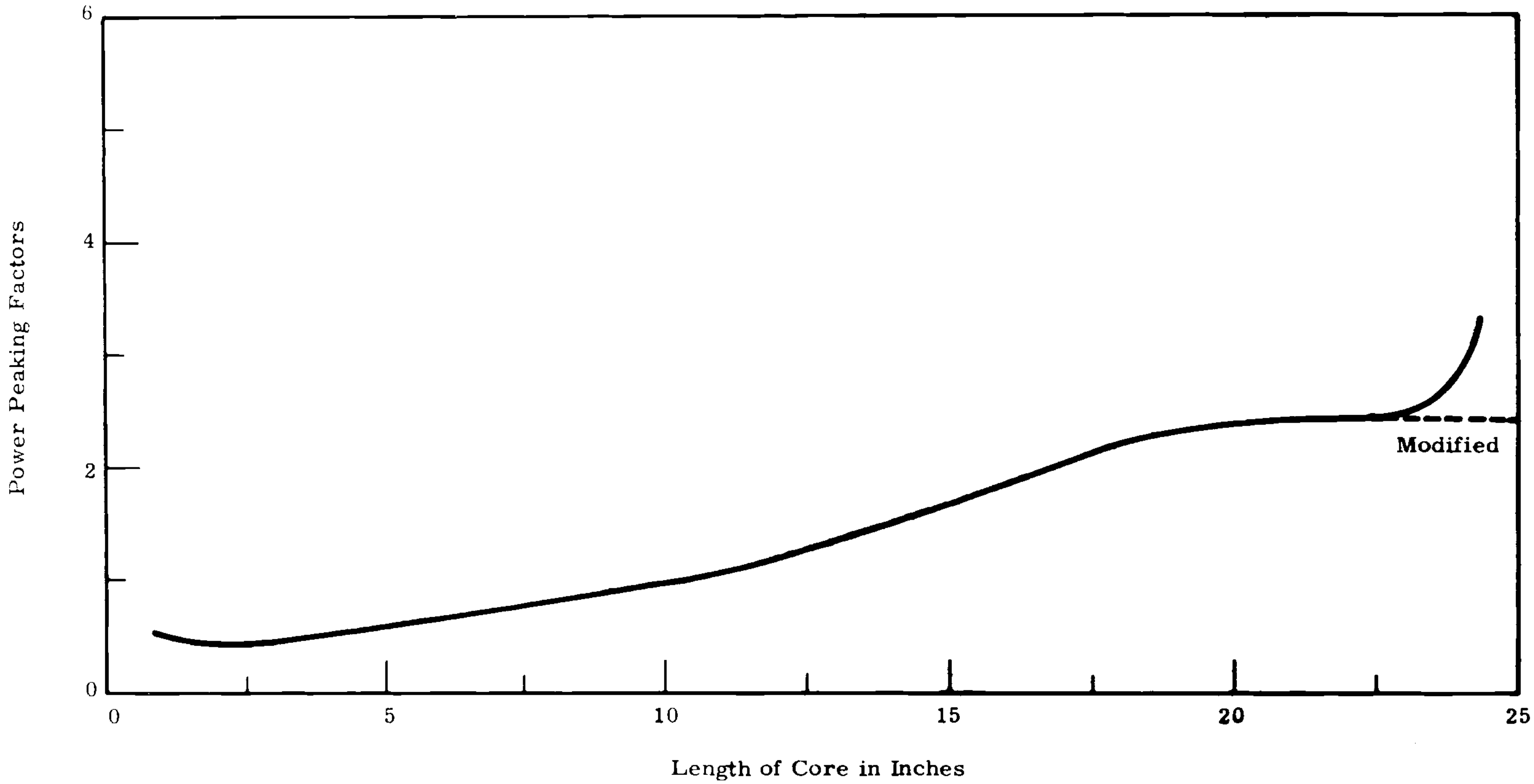


FIGURE 4.6-1
Vertical Power Peaking Factors for MTR Fuel Element Number 45 Loaded with a Phoenix Fuel. (Preliminary, doew not Include Effects of Fuel Element Tape)

4.6-5: Results

The major difference affecting thermal and hydraulic properties between the proposed Phoenix core and the present MTR core is the total heat transfer area (Table 4.6-I). The Phoenix core contains approximately 20% less effective heat transfer surface and 10% less coolant flow through the core. Detailed analysis as described above indicates that if the reactor is operated at 29 MW_t with the Phoenix core, the temperatures and heat fluxes within the core will not differ significantly from corresponding values in the present core (Table 4.6-II).

Thus, it is seen that operation of the MTR at a power level of 29 MW_t with the Phoenix core is well within limits implied by present operation. On the other hand, there are alternatives available for raising the reactor power and at the same time operating within specified limits. For example, increasing the inlet coolant pressure to 73 psia (maintaining a pressure drop of 40 psi across the core) would allow operation at 37 MW_t (Figure 4.6-2). Furthermore, the limiting factor for this latter case is the nominal heat flux limit of $1*10^6$ BTU/hr-ft^2. Incipient nucleate boiling and burnout heat fluxes were no longer the controlling criteria.

Further effects of changing the pressure head are illustrated in Figures 4.6-3 to 4.6-6. Figure 4.6-3 illustrates the maximum heat flux versus reactor power level and included the influence of a 25% velocity disparity across the fuel assembly. The velocity disparity effects a skewed heat split in plate #2 sufficiently to cause the maximum heat flux to occur on its surface and thus away from the reactor power peaks. For this condition the maximum nominal heat flux is $1.065*10^6$ BTU/hr-ft^2 at a power level of 37 MW_t, which slightly exceeds the $1.0*10^6$ BTU/hr-ft^2.

TABLE 4.6-I

COMPARISON OF CORE CHARACTERISTICS FOR A STANDARD MTR FUELED CORE VERSUS AN MTR PHOENIX FUELED CORE

	Standard MTR Core	Phoenix Fueled Core
Core Power (MW_t)	40	29
Inlet Coolant Temperature (°F)	105	105
Inlet Reactor Pressure (psia)	60	60
Core Pressure Drop (psi)	40	40
Average Coolant Flow per Element (gpm)	620	554
Average Coolant Velocity in Fuel Element $\frac{ft}{sec}$	31.4	33.2
Hydraulic Diameter of Fuel Element (in)	0.223	0.236
Fuel Element Flow Area (ft^2)	0.0406	0.035
Heat Transfer Area per Element (ft^2)	15.49	13.33
Total Core Heat Transfer Area (ft^2)	398.87	325.43
Number of Fuel Plates in Fuel Element	19	16
Number of Elements in Core	23	19
Element Fuel Plate Thickness (in) Inside Outside	0.050 0.065	0.080
Coolant Channel Thickness (in)	0.116	0.119
Number of Fuel Plates in Shim Rod	14	12
Number of Shim Rods in Core	4	8
Shim Rod Fuel Plate Thickness	0.060	0.080
Shim Rod Coolant Channel Thickness (in)	0.117	0.126
Position of Core in Reactor	North Side	Center

TABLE 4.6-II

HEAT TRANSFER CONDITIONS FOR A MTR CORE FUELED WITH A PHOENIX FUEL AND A STANDARD FUEL AND OPERATING AT 29 AND 40 MW_t RESPECTIVELY

	Standard MTR Core	Phoenix Fuel Core
Coolant Temperatures		
Entering Reactor	105	105
Average Exit Hottest Element	128	129
Average Exit Hottest Channel	137	132
Maximum Exit Hottest Channel	148	141
Saturation Temperature at Hot Spot	248	242
Heat Fluxes ($Btu/Hr\text{-}Ft^2$)		
Average for all Elements	$3.1*10^5$	$2.97*10^5$
Nominal Hot Spot in Hottest Channel	$8.2*10^5$	$8.1*10^5$
Maximum Hot Spot in Hottest Channel	$1.13*10^6$	$1.12*10^6$
Burnout Heat Flux	$1.56*10^6$ *	$1.41*10^6$
Hot Channel Factors		
Variation in Velocity From Channel to Channel	1.25	1.25
Fraction of Total Reactor Power Generated in Fuel Plus Coolant	0.95	0.95
Hot Channel Uncertainties		
Reactor Power Measurement Error	1.05	1.05
Plate to Plate Fuel Variation	1.02	1.02
Dimensional Effects on Mass Flow	1.08	1.08
Fuel Element Power Calculation Error	1.10	1.10
Heat Flux Uncertainties		
Vertical Maximum-to-Average Flux Ratio Measurement	1.05	1.05
Reactor Power Measurement Error	1.05	1.05
Fuel Element Power Calculation Error	1.10	1.10
Fuel Variation Plate to Plate	1.02	1.02
Fuel Core Alloy Thickness Variation	1.05	1.03
Fuel Core Alloy Area Variation	1.06	1.06
Heat Transfer Coefficient Factors		
Variation in Velocity Within the Channel	0.96	0.96
Heat Transfer Coefficient Uncertainties		
Correlation Equation	0.80	0.80
Local Dimensional Effect	0.96	0.96

*The probability that the hot spot will exceed the burnout heat flux is less than $0.0^{11}1$. The three standard deviations require a burnout heat flux of 1.066×10^6 $Btu/hr\text{-}ft^2$.

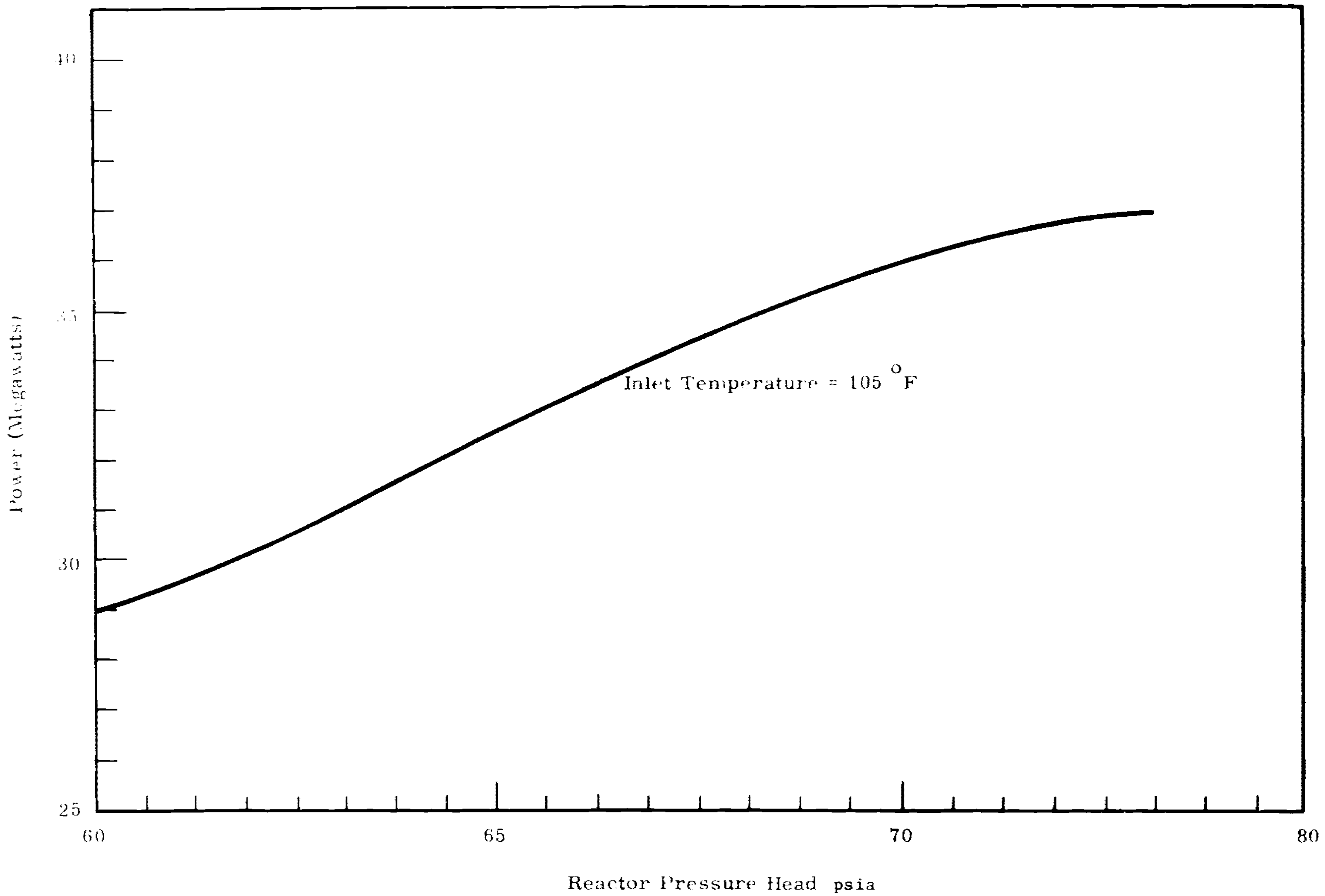


FIGURE 4.6-2
Reactor Power Versus Pressure Head at Maximum Thermal Hydraulic Operating Limits for MTR Application

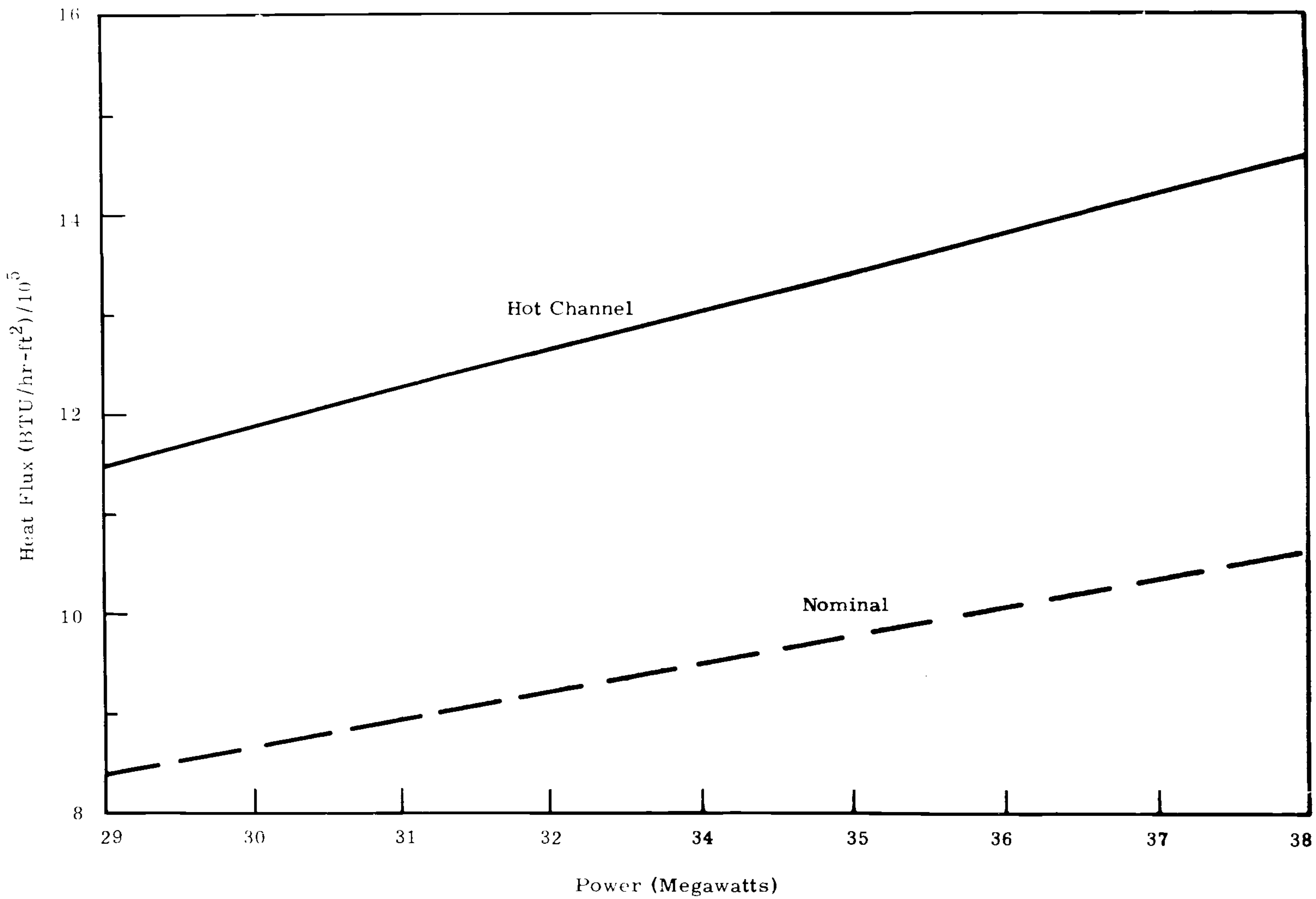


FIGURE 4.6-3
Maximum Heat (Away from Hotspot due to a Velocity Disparity) Versus Reactor Power at Maximum Thermal Hydraulic Operating Limits for MTR Application

However, the maximum nominal heat flux occurring at the hot spot for the same reactor power but with a uniform velocity profile across the assembly is $9.253*10^5$ Btu/hr-ft^2 and is well within the thermal and hydraulic limits established previously. This is illustrated in Figure 4.6-4. Since there exists a degree of uncertainty in the coolant velocity from channel to channel, this phenomena is included in the hot channel uncertainty factors and the 37 MW_t is assumed to be a conservative power level within the limits specified.

Figure 4.6-5 illustrates a comparison between nominal wall temperatures, hot channel wall temperatures, and critical wall temperatures at various reactor power levels. The critical wall temperature is that temperature which causes incipient nucleate boiling to occur and was determined from the Jens' and Lottes' correlation. Approximately 60^{o}F[16] was considered to be equivalent to two standard deviations from incipient nucleate boiling conditions. Figure 4.6-5 shows the nominal wall temperature below the critical wall temperature by two standard deviations, and was the limiting influence at reactor power levels less than 37 MW_t. As before in the case of the maximum nominal heat flux, the influence of the velocity disparity caused critical conditions to occur away from the fuel plate containing the highest power spikes shown in Figure 4.6-1. The hot channel wall temperature illustrated in Figure 4.6-5 has no physical significance since it lies above the critical wall temperature where local boiling begins.

Figure 4.6-6 illustrates the relationship between nominal heat flux, hot channel heat flux, and departure from nucleate boiling heat flux for various reactor power levels. For a 37 MW_t reactor power there exists a $1.5*10^5$ Btu/hr-ft^2 heat flux differential between hot channel

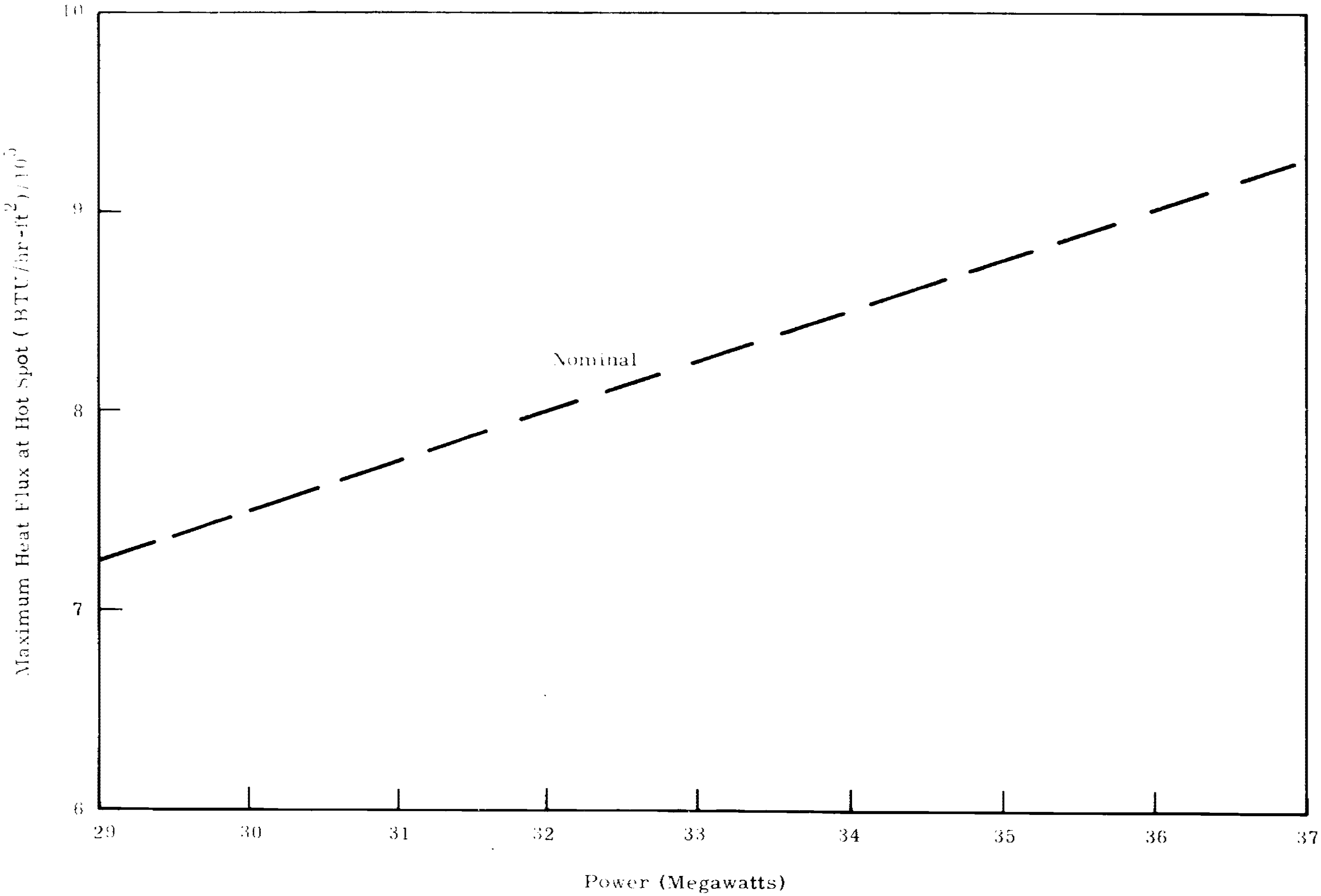


FIGURE 4.6-4
Maximum Heat Flux (at Hotspot for a Uniform Velocity Profile) Versus Reactor Power at Maximum Thermal Hydraulic Operating Limits for MTR Application

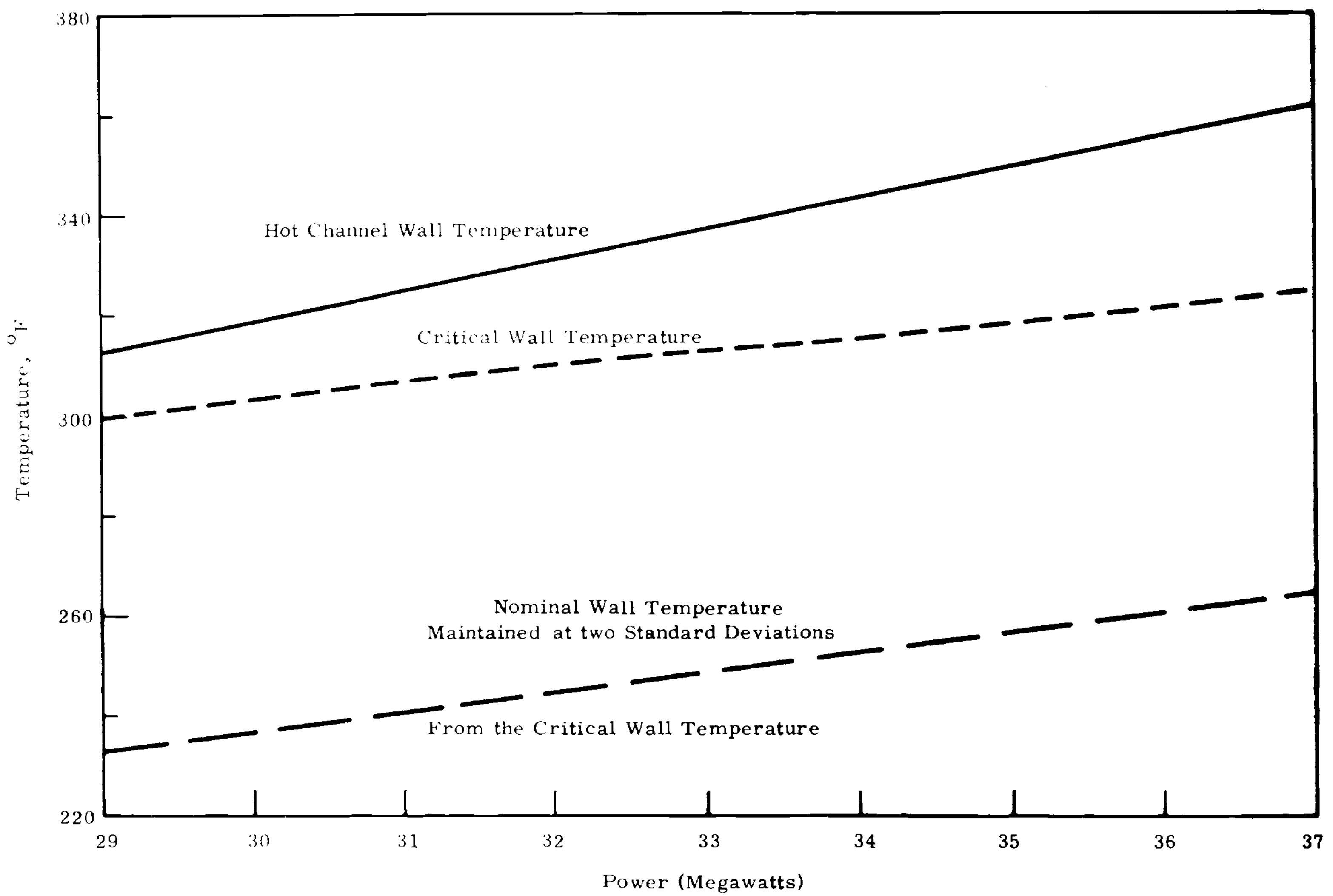


FIGURE 4.6-5
Critical Wall Temperature Versus Reactor Power at Maximum Thermal Hydraulic Operating Limits for MTR Application

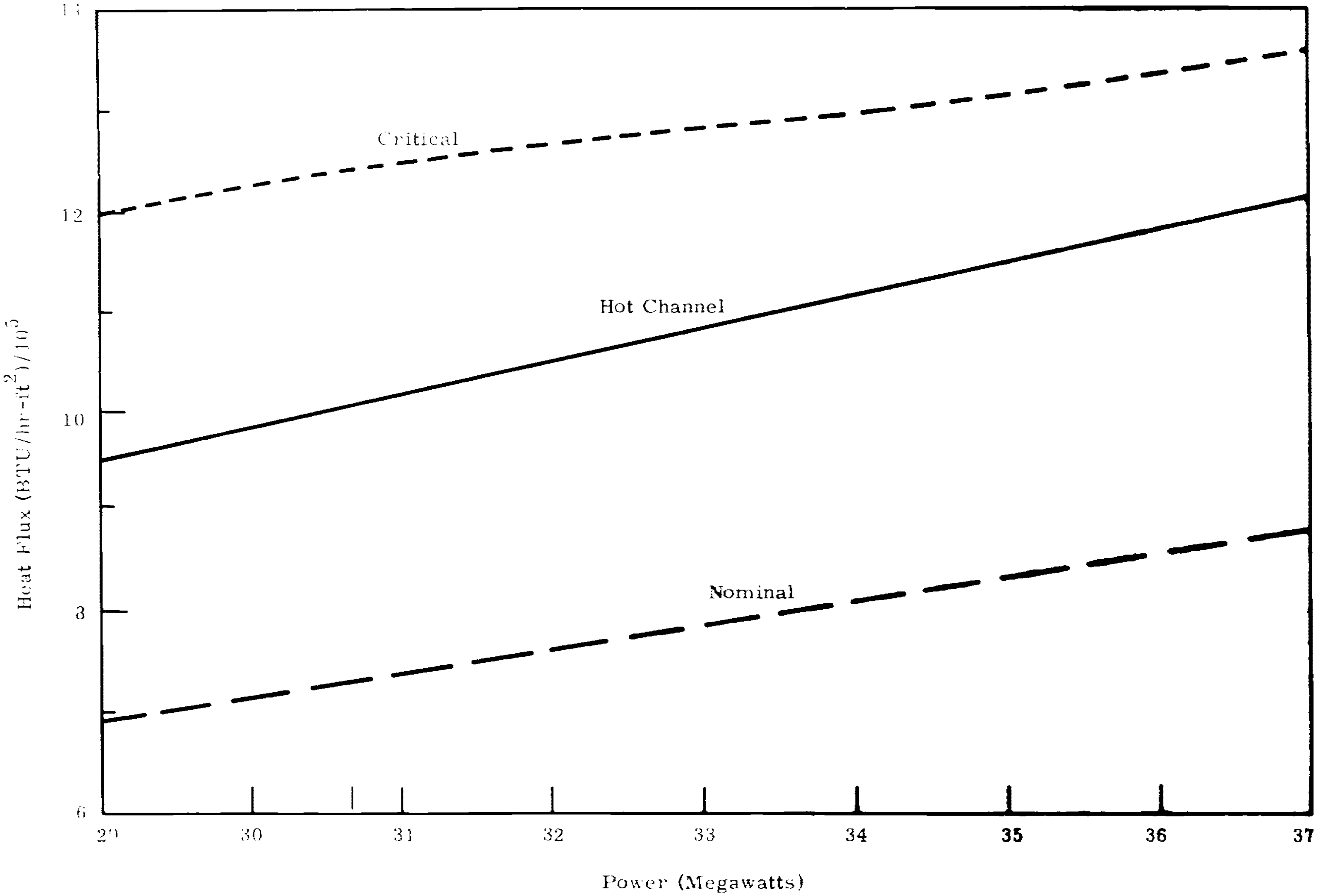


FIGURE 4.6-6
Critical Heat Fluxes Versus Reactor Power at Maximum Thermal Hydraulic Operating Limits for MTR Application

flux and the burnout heat flux as calculated from the Bernath correlation. The exact differential required according to the thermal and hydraulic limits established previously has not firmly been established. However, the above margin is considered conservative and will be re-evaluated when a standard deviation from the reactor trip power has been determined for a Phoenix core.

The corrosive film buildup during reactor operation is illustrated in Figure 4.6-7. The maximum film thickness occurring for a 37 MW_t reactor power was approximately $1.73*10^{-3}$ inches which is thick enough to begin flaking and thus causing an increase in wall temperature potential at the flaking location. No attempt was made in this study to evaluate the flaking influence on temperatures but is left for a later analysis.

The main consideration is that the film did cause increased wall temperatures and affected the heat transport properties but within reasonable limits.

It should be noted that the foregoing analysis does not include the effect of the fuel element taper. Thus, these results should be considered as conservative. The final results are expected to raise the allowable power.

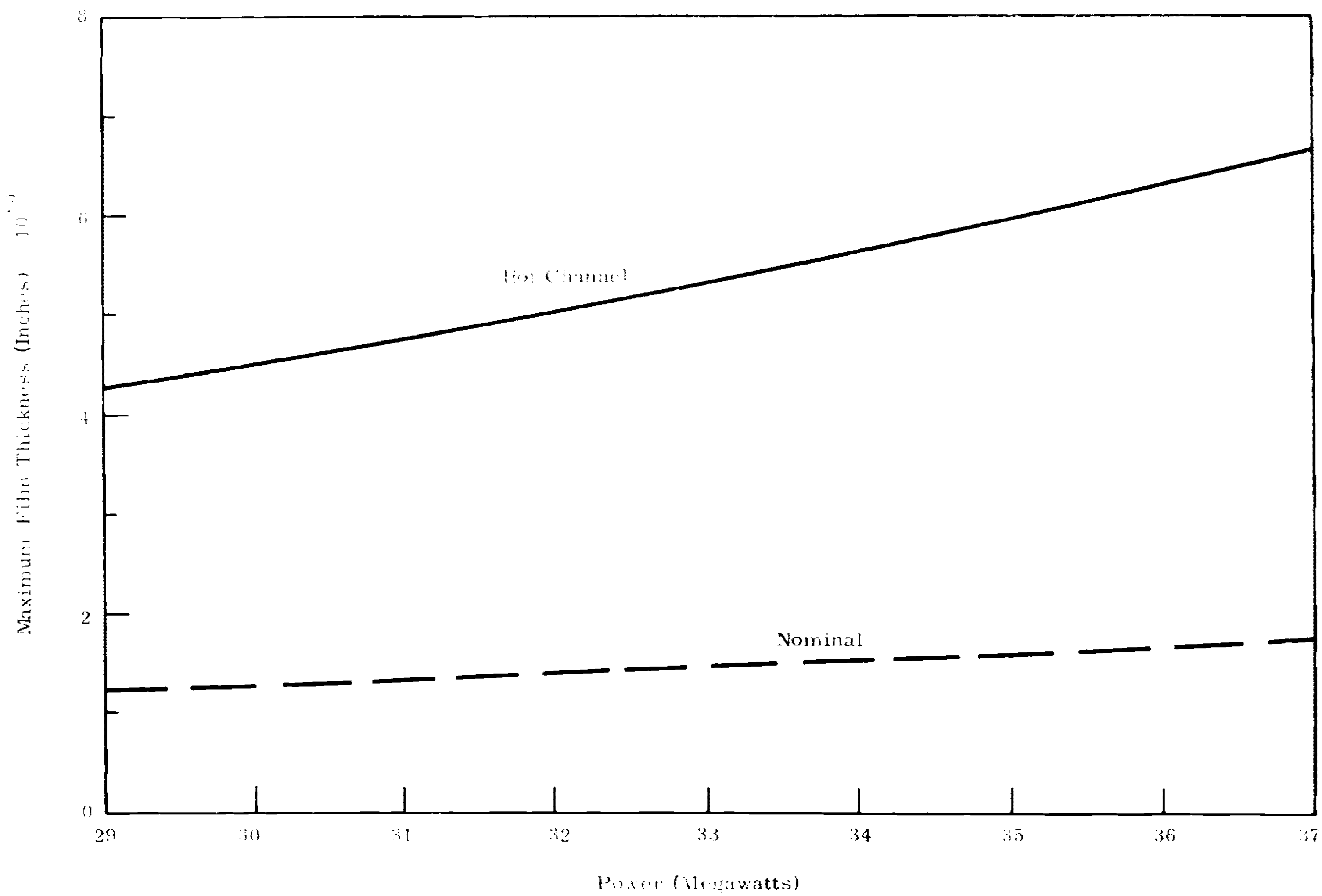


FIGURE 4.6-7
Corrosive Film Buildup Versus Reactor Operating Power for a Phoenix Fuel Operating Cycle in the MTR of 3000 MWd

4.6-6: Conclusions

It should be understood that this effort is only a preliminary study and is based on analytical power profiles which will be re-evaluated as experimental data becomes available. Also, uncertainty factors were used that were developed for a standard MTR fuel loading and will be modified to agree with conditions unique to a Phoenix core as the core specifications are determined. Based on these data and conditions, the following conclusions can be made:

- A maximum reactor power of 37 MW_t is within thermal and hydraulic limitations.
- Corrosion products are not severely limiting over a 3000 MWD time cycle.
- A 25% velocity disparity exerts a substantially large influence on power limits and causes limiting conditions to occur away from the power spikes. This results directly from the skewed power split in the fueled plate caused by the differential velocity in adjacent channels.
- The development of uncertainty factors unique to the Phoenix core will require reassessment of this study. It is expected that this will have only minor effects on the final results.

5. REFERENCES AND APPENDICES

5.1 - Code Development

5.1-1: WHIRLAWAY Modifications

Program WHIRLAWAY is a modification to the three dimensional, two-group neutron diffusion code WHIRLAWAY.[18] The modifications to the WHIRLAWAY code permit (1) the interruption and subsequent restart of a calculation; (2) the calculation of the total fission rate in the core; and (3) an increase of the maximum number of mesh points to 20,000, which necessitated the removal of the adjoint calculation capability.

Program POWERWRITER utilized the intermediate output from the WHIRLPOWER code to calculate and print a normalized power density distribution over a region of interest in three dimensions. It will find the maximum peak to average ratio and print this value along with the coordinates of the mesh point which it occurs.

Both codes are written in UNIVAC 1108 FORTRAN-IV-CSC. Input instructions and sample cases are given in the document, "WHIRLPOWER/ POWERWRITER - Two Groups, Three-Dimensional Codes to Calculate Normalized Flux and Power Distributions," BNWL-CC-1302, by W. W. Porath.

5.1-2: ZODIAC 2+2

The ZODIAC2 Code has been modified and enlarged to increase the capabilities of this code to handle burnup analysis. The primary change was the inclusion of the superior epithermal cross section averaging code HRG as an optional alternative to GAM. Further changes were made to COMBO, TEMPEST, SIGMA-3H, and REFIRE.

Libraries for the codes TEMPEST, HRG (GAM) and SIGMA-3H have undergone reorganization and expansion to the extent that Composite Library Tapes (CLT's) previously used are not compatible with current versions of these codes. At present, both current and obsolescent versions of these three codes are included on the chain tape. Hence, the name ZODIAC(2+2).

A brief statement describing the nature of the changes and their effect on the input and output is given in the document, "ZODIAC(2+2): A Revision to ZODIAC2," BNWL-459, by R. H. Holeman and D. D. Matsumoto.

REFERENCES

1. The CNSG-II - A Conceptual Merchant Ship Nuclear Reactor Design, BAW-1280; The Babcock and Wilcox Company, Lynchburg, Virginia; September 1963.

2. S. Yasukawa and R. Shindo, "Uses of Plutonium Fuel in Pressure Tube-Type, Heavy Water Moderated Thermal Reactors," IAEA Symposium on the Use of Plutonium as a Reactor Fuel; Brussels, Belgium, March 13-17, 1967.

3. J. L. Carter, Jr., "Computer Code Abstracts, Computer Code-HRG," Reactor Physics Department Technical Activities Quarterly Report, July, August, September 1966, USAEC Report BNWL-340, October 15, 1956

4. H. C. Honeck, "THERMOS - A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5826, Brookhaven National Laboratory.

5. R. H. Shudde and J. Dyer, NAA Program Description, "TEMPEST - A Neutron Thermalization Code," North American Aviation Corp. (1960).

6. J. R. Worden, W. L. Purcell, and L. C. Schmid, "Physics Experiment High Power Density Core of the PRTR," BNWL-221, January 1966.

7. J. R. Lilley, "Computer Code HFN-Multigroup, Multiregion Neutron Diffusion Theory in One Space Dimension," USAEC Report HW-71545, November 1961.

8. B. H. Duane, "Time Variant Isotopic Transmutation GE-HL Program ALCHEMY; Physics Research Quarterly Report, October, November, December 1963, HW-80020, January 1964.

9. R. H. Holeman and D. D. Matsumoto, "ZODIAC(2+2): A Revision to ZODIAC2," BNWL-459, May 1967.

10. M. D. Freshley, F. E. Panisko, and R. E. Skavdahl, "Irradiation of UO_2-PuO_2 High Power Density Fuel Elements in PRTR," Trans. Am. Nucl. Soc., 8, (365) 1965.

11. W. P. Stinson, C. M. Heeb, "Approach-To-Critical Experiments with Phoenix Fuel," Trans. Am. Nucl. Soc., 10, 1, 187 (1967)

12. H. L. Henry, F. Swanberg, Jr., "Critical Approach Facility Final Safeguards Analyses," BNWC-40, Rev.2, (in press).

13. M. L. Griebenow and K. D. Richert, MACABRE II User's Manual.

REFERENCES (Continued)

14. J. C. Griess, "Effect of Heat Flux on Corrosion of Aluminum by Water," Part 4, ORNL-3541, February 1964.

15. R. H. Holeman and P. L. Hofmann, "Nuclear Design Calculations for an Hx-Pu Fueled MTR," HW-84575, December 1964.

16. W. S. Little, "Heat Transfer in the MTR Core," PTR-689, April 15, 1964.

17. MTR Operating Limits.

18. T. B. Fowler and M. L. Tobias, "WHIRLAWAY - A Three-Dimensional, Two-Group Neutron Diffusion Code for the IBM 7090 Computer," ORNL-3150, August 15, 1961.

19. P. L. Hofmann, G. J. Busselman, and R. H. Holeman, "Nuclear Characteristics of Some Compact, Water Moderated, Plutonium Burners," HW-79977, April 1964.

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