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A PRELIMINARY DESIGN
FOR A PHOENIX FUELED HFIR REACTOR

U. P. Jenquin

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Reactor Technology

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FOR A PHOENIX FUELED HFIR REACTOR

U. P. Jenquin

Analytical Physics Section
Reactor Physics Department

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ABSTRACT

Calculations were performed to determine potential advantages of utilizing Phoenix fuel for the High Flux Isotope Reactor (HFIR). The Phoenix Fuel Program, under which this study was performed, is a research and development project concerned with the use of plutonium fueled systems for extending core reactivity lifetime. The neutronic characteristics of a plutonium fueled HFIR are compared to a uranium fueled HFIR. Utilizing the results of the neutronic investigations, a preliminary plutonium fueled HFIR design is derived.

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* Ref: ORNL-3572 Rev 2 Vol. 1B

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A PRELIMINARY DESIGN
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I. INTRODUCTION

The Phoenix Fuel Program is a research and development project concerned with the use of plutonium fueled systems for extending core reactivity lifetime. Analytical studies have been made⁽¹⁾ 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 .

The High Flux Isotope Reactor⁽²⁾ (HFIR) seems to be potentially well suited for utilization of Phoenix fuels. Preliminary investigations⁽³⁾ indicate that the thermal and total fluxes in the flux trap would be higher for a Phoenix fueled HFIR than for the current HFIR system. Since the ^{240}Pu is a poison to the system initially, the shim control requirements should be less than currently required. The present HFIR design utilizes a burnable boron poison in addition to control rods. Reduced peak xenon and samarium poisoning in the plutonium core would also reduce shim requirements. Since the present HFIR fuel is complex and very expensive to make, using plutonium rather than uranium should not increase the cost significantly. Thus, if the core reactivity lifetime is increased appreciably, the net fuel cost will be lowered.

One disadvantage of plutonium fuel could be a larger spatial power peaking effect. Also, the delayed neutron fraction is lower for plutonium,

which might introduce kinetics problems. The amount of control material necessary for a given reactivity span would have to be increased because the neutron cross sections for plutonium are larger than the cross sections for uranium.

Preliminary calculations⁽³⁾ for a Phoenix fueled HFIR core verified the neutronic characteristics listed above. More detailed neutronic, thermal hydraulic, and economic calculations have been done and will be discussed in detail in the following sections. Based on the calculations, a preliminary HFIR Phoenix fueled core design has been developed.

The calculations were made for a HFIR core and plant with the same characteristics as currently exist for the ^{235}U fueled core. The general plant and core are described in Sections II and III, respectively. These sections contain the same information as in Reference (2), but is repeated here for completeness. The calculational methods used in the analysis are described in Section IV.

Reactivity and burnup results for various Phoenix fueled HFIR systems are also discussed in Section IV. Results for the standard ^{235}U core are included for comparison. A discussion of the feasibility of manufacturing plutonium fuel for HFIR is given in Section V. A potential Phoenix fueled HFIR design based on the results of the analysis is presented in Section VI. The conclusions drawn from the study are summarized in Section VII.

II. PLANT DESCRIPTION

The HFIR is a beryllium-reflected, light-water-cooled and moderated, aluminum-clad fuel plate, flux-trap type reactor which utilizes highly

enriched ^{235}U fuel. The design power level is 100 Mw. The major areas and important equipment items are shown in a cutaway view of the plant in Figure 1.

The reactor core assembly is contained in an 8-ft-diameter pressure vessel, which is located in an 18-ft-diameter cylindrical pool of water. The top of the pressure vessel is 17-ft below the pool water level, with the reactor center line 27 1/2 ft below the pool level.

The primary coolant enters the pressure vessel through two 16-in. pipes above the core, passes through the core, and exits through an 18-in. pipe beneath the core. The flow rate is about 16,000 gpm, of which approximately 13,000 gpm flows through the fuel region and the remainder through the target, reflector, and control regions. The system is designed to operate at an inlet pressure up to 1000 psi; however, normal operation at 100 Mw requires an inlet pressure of only 600 psi. Under these conditions the inlet temperature is 120° F, the corresponding exit temperature is 164° F and the pressure drop through the core is 110 psi.

From the reactor the coolant flow is distributed to three of four identical heat-exchanger circulation-pump combinations, each located in a separate cell adjacent to the reactor and storage pools. Each cell also contains a let-down valve which controls the primary coolant pressure. A secondary coolant system removes heat from the primary system and transfers it to the atmosphere by passing water over a four-cell, induced-draft cooling tower.

Although the primary purpose of the HFIR is the production of trans-uranium isotopes, several other experimental facilities have been provided. These include (1) four horizontal beam holes which originate in the reflector;

FIGURE 1. Isometric Cutaway of Reactor Building

(2) four slant access facilities, called "engineering facilities," which are located adjacent to the outer reflector at an angle with the vertical; and (3) 38 vertical facilities of various sizes located in the reflector.

III. CORE DESCRIPTION

The reactor core, shown in Figure 2, consists of four concentric regions, each approximately 2 ft. high. The central region, often called the "island," has the highest thermal-neutron flux and constitutes the flux trap. The island is approximately 5 inches in diameter and contains the target, which is in the form of a bundle of rods.

The fuel region, located immediately outside the island, has an inside diameter of approximately 5 inches and an outside diameter of approximately 17-1/8 inches. It consists of two cylindrical fuel elements made up of vertically oriented curved plates. To minimize the radial peak-to-average power density ratio, the fuel concentration is radially graded.

The control region, with control plates in the form of two thin poison-bearing concentric cylinders, is located between the outer fuel element and the beryllium reflector. Each cylinder contains an axially varied poison to minimize the axial value of the peak-to-average power density ratio throughout the core lifetime.

The reflector region is a concentric ring of beryllium reflector approximately 1 ft. thick. This, in turn, is reflected by water of effectively infinite thickness. Water above and below the core serves as an end reflector.

A. Target Assembly

The target assembly is located in the island and contains the ^{242}Pu

ORNL-LR-DWG-65776

FIGURE 2. Illustration of Core

target material. It consists of 30 rods (3/8 in. O.D.) spaced in a triangular array. The rods contain pellets fabricated from a mixture of PuO_2 and aluminum powder. The pellets are then packed in aluminum tubes and capped, and the tubes are collapsed onto the pellets to provide good heat transfer. Appropriate leak tests are applied to each completed tube. The target tubes are centered inside 21/32 inch diameter circular shrouds, thereby providing each rod with its own cooling channel. The shrouds change to a 5/8 inch hexagonal shape near the upper end in order to provide for spacing and clamping.

B. Fuel Elements

The fuel region of the HFIR, shown in Figure 3, is composed of two concentric, cylindrical fuel elements containing vertically oriented curved plates extending in the radial direction. The inner element, which contains 171 plates, is initially loaded with 2.6 kg of ^{235}U and 2.8 g of ^{10}B in the standard core design. The inner diameter is 5.067 inch and the outer diameter is 10.590 inch. The outer element contains 369 plates and initially contains 6.8 kg of ^{235}U but no burnable poison in the standard core. Its inner diameter is 11.250 inch and the outer diameter is 17.134 inch.

The individual plates are of a sandwich-type construction composed of a fuel-bearing cermet hermetically sealed in covers of type 6061 aluminum. In the standard core design the fuel-bearing cermet is a mixture of U_3O_8 and aluminum, approximately 30 wt% U_3O_8 in the case of the inner annulus and 41 wt% U_3O_8 in the case of the outer annulus. Uranium containing at least 93% ^{235}U is used as fuel. Initially, each inner annulus plate contains

FIGURE 3. Fuel Element

15.18 g $\pm$ 1% of ^{235}U ; each outer annulus plate contains 18.44 g $\pm$ 1% of ^{235}U . In addition, each inner annulus plate initially contains 0.0164 g of ^{10}B . The finished plates are 0.050 inch thick and 2 ft. long, with a nominal active length of 20 inches. The thickness of the fuel plate core is 0.030 inch and the cladding thickness is 0.010 inch. The plates are curved in the form of an involute which provides a constant cooling channel width between plates; the width is 0.050 inch. Before bending, the inner annulus plates are approximately 3.6 inch wide, and the outer annulus plates are approximately 3.2 inches wide. The fuel cores are centered to provide adequate edge cladding. Both inner and outer plates are fastened between cylindrical aluminum side plates by welding.

To minimize the radial peak-to-average power density ratio, the fuel surface density in each plate is varied along the arc of the involute curve. The ^{10}B included in the inner plates in the form of B_4C , is added to an aluminum filler piece, which together with the fuel-bearing cermet makes up a composite, rectangular fuel plate core, as shown schematically in Figure 4.

A fuel element has a typical lifetime of about 23 days at 100 Mw. Under normal operating conditions, flow through the fuel is 13,000 gpm, which corresponds to a velocity of 51 fps. The core pressure drop at this velocity is 110 psi. At a total heat power of 100 Mw, the average heat flux is $0.78 \times 10^6 \text{ Btu ft}^{-2} \text{ hr}^{-1}$, and the calculated maximum hot spot heat flux is $1.97 \times 10^6 \text{ Btu ft}^{-2} \text{ hr}^{-1}$. The minimum incipient boiling power level under anticipated conditions (inlet pressure, 600 psi; flow, 51 fps; inlet temperature, 120°F) is calculated to be 142 Mw at the beginning of the operating cycle.

FIGURE 4. Fuel and Burnable-Poison Distribution in the As-Built Element

C. Beryllium Reflector

The fuel region is radially reflected by a 1-ft-thick beryllium reflector. The inner portion of the reflector is readily removable to permit replacement, when necessary due to radiation damage or other causes, and to allow access to the control plate drives. The reflector is water cooled; but in order to avoid excess neutron absorption, the volume of water in the reflector is held to the minimum consistent with the cooling requirements.

The inner portion of the reflector consists of a 3.689 inch thick beryllium section composed of four concentric cylinders with cooling water flowing axially between them. The outer part of the reflector is an 8.375 inch thick beryllium annulus provided with axial circular coolant channels. Three of the four inner cylinders, called the removable reflector, may be removed from the reactor as a unit, thus permitting access to the control plate shock absorbers. The fourth cylinder, called the semipermanent reflector, is removed only when necessary because of radiation damage. Four small pieces of the semipermanent beryllium, called control plate access plugs, are easily removable to facilitate access to the control drives. Although the thick outer annulus, called the permanent reflector, may be removed, it is anticipated that this would be done very infrequently.

To minimize the chance, should the beryllium crack, of interference with the control plates which are located between the removable reflector and the fuel region, a 1/16 inch thick aluminum liner covers the inner surface of the removable beryllium. Coolant for these removable reflector cylinders is provided by water flowing through 1/8 inch axial grooves on the surfaces of

these cylinders. A 1/16 inch thick water gap between the removable and semipermanent reflectors facilitates removal and replacement of the removable reflector, furnishes cooling, and provides for expansion of the removable reflector. Eight 1/2 inch diameter vertical irradiation facilities are provided in the middle removable reflector cylinder, and eight others are located in the removable control plate access plugs. They are provided with beryllium plugs which are in place when the facilities are not in use.

The permanent reflector is cooled by water which flows axially through 1/8 inch diameter holes which are spaced on circles concentric with the reflector. There are five such circles, each containing 80 cooling holes, except where the coolant hole pattern is interrupted by the experimental facilities.

The permanent reflector is penetrated by 16 vertical 1.58 inch diameter and 6 vertical 2.83 inch diameter experimental facilities similar to those in the removable reflector. Two 4 inch beam tubes penetrate the permanent reflector and terminate in the semipermanent beryllium. One 4 inch diameter tangential beam tube penetrates all the way through the reflector assembly. Four slanted grooves in the outer surface of the reflector accommodate the engineering test facilities.

D. Control Plates

The control plates are located in a 0.869 inch thick annular region between the outer fuel element and the removable beryllium. The control plates consist of two 1/4 inch thick concentric cylinders. The coolant

passages, listed radially outward, are 0.104 inch, 0.170 inch, and 0.095 inch. The coolant water velocity in the control region is about 16 fps. The inner cylinder, which is a single piece, is used both as a shim and a regulating rod. The outer cylinder is divided into four quadrants, each used as a shim-safety rod and each having its own drive rod and scram mechanism.

The chain reaction is controlled by altering the efficiency of the beryllium reflector as the control plates are moved vertically between the core and the reflector. The single inner cylinder is driven downward, thus driving the poison out of the reactor to increase reactivity while the outer quadrants are driven upward out of the reactor to increase reactivity.

In order to reduce axial variations in the power distribution and to prolong the life of the control elements the cylinders are divided into three longitudinal sections, each of which has different neutron absorbing characteristics. The lower section of the inner cylinder and the upper section of the outer cylinder contain 33 vol % Eu_2O_3 dispersed in aluminum. These sections are highly neutron absorbing and are called the black regions. The central section of each plate contains 40 vol % tantalum dispersed in aluminum. They are less absorbing than the Eu_2O_3 sections and are called the gray regions. The upper section of the inner cylinder and the lower section of the outer cylinder are made of aluminum and are called the white regions. In normal operation criticality is achieved by driving in unison the inner cylinder down and the outer quadrants up.

The inner control cylinder has an outer radius of 8.921 inch and an inner radius of 8.671 inch. The overall length is 68-3/8 inch. The lower end of the black region is 14-5/16 inch above the lower edge of the cylinder,

and the upper end of the white region is 7-1/16 inch below the upper edge of the cylinder. The outer control plates have an outer radius of 9.300 inch and an inner radius of 9.050 inch. The overall width of each is 13-39/64 inch and the overall length is 66-3/16 inch. The upper end of the black region is 8-5/8 inch below the upper end of the plate, and the lower end of the white region is 10-9/16 inch above the lower end of the plates.

The cylinder and plates are driven from beneath the reactor by drive rods which extend into the subpile room, where the drive mechanisms are located. Guidance for each outer control plate is furnished by six ball bearings. Four bearings are attached to each outer control plate, two at each end, and are run in stationary tracks which extend above and below the core. The other two bearings are attached to the lower track assembly and bear against the control plate. The inner control cylinder is guided by eight bearings, four of which are attached to the track assemblies at each end of the reflector and extend through the slots between the four outer plates.

IV. NEUTRONIC INVESTIGATIONS

The calculations were made for a HFIR core with the same characteristics as currently exist for the ^{235}U fueled core. The calculational model must be quite detailed due to the complicated geometry of the system. Various methods of calculating the value of the average ^{240}Pu absorption cross section at 1 eV were investigated. The sensitivity of the results to the ^{240}Pu cross section was determined. Reactivity and burnup characteristics were determined for a number of Phoenix fueled HFIR systems as well as the uranium fueled HFIR core. From these results the effect of various parameters on the neutron flux in the target and core reactivity lifetime were determined.

A. Calculational Technique

The calculational model used to determine the reactivity and burnup characteristics of the HFIR fuel is a one-dimensional, multi-region description of the reactor core. The composition of the fuel annuli and the target region were obtained from ORNL-3572.⁽²⁾ The compositions and radii used for the other regions are those which ORNL personnel use in their calculations.

The fuel thickness varies along the plate in the radial direction and the flux magnitude varies significantly across each annulus. To account for the effects of these variations, it was necessary to use a relatively complicated calculational model for the system. The inner annulus was subdivided into five regions and the outer annulus was subdivided into four regions. The radii of the regions and the thickness of the fuel within each region are listed in Table I.

A slab THERMOS⁽⁴⁾ calculation of each region containing fuel gives the thermal flux in the coolant, clad, filler, and fuel relative to the cell average flux. To obtain interactions in the radial direction, the cell averaged, flux weighted nuclear densities from the preceeding slab THERMOS calculations were used in radial THERMOS calculations. Each homogenized slab calculation represents one region in the radial calculation. Since there are too many regions in the radial direction for one THERMOS calculation, three calculations were done assuming reflecting boundaries in the first and second annuli. Region average macroscopic cross sections from these radial THERMOS calculations were used for the reactivity calculations with HFN.⁽⁵⁾

For the burnup calculations, the microscopic cross sections from the radial THERMOS calculations were multiplied by the appropriate flux

TABLE IREGION DESCRIPTION FOR HFIR CALCULATIONS

<u>Region</u>	<u>Description</u>	<u>Fuel Thickness (mils)</u>	<u>Inside Radius (cm)</u>	<u>Outside Radius (cm)</u>
1	Target		0.00	4.32
2	Inner H ₂ O		4.32	6.40
3	Inner Clad		6.40	7.14
4	Inner Fuel	11	7.14	7.78
5	Inner Fuel	15.5	7.78	8.50
6	Inner Fuel	19	8.50	9.33
7	Inner Fuel	23	9.33	11.74
8	Inner Fuel	19	11.74	12.60
9	Middle Clad		12.60	15.15
10	Outer Fuel	16	15.15	16.07
11	Outer Fuel	23.5	16.07	18.77
12	Outer Fuel	16	18.77	19.90
13	Outer Fuel	11	19.90	21.00
14	Outer Clad		21.00	24.15
15	Inner Be		24.15	33.00
16	Outer Be		33.00	54.00
17	Outer H ₂ O		54.00	60.00

depression factors from the slab THERMOS calculations in order to preserve the isotopic macroscopic cross sections. Thus the flux depressed microscopic cross sections times the cell averaged densities (not flux depressed) preserve the macroscopic cross sections calculated with the radial THERMOS calculations. These cross sections were then used in the burnup calculation using ZODIAC G.⁽⁶⁾

Epithermal cross sections for each radial region were computed with HRG⁽⁷⁾ (1967 version) assuming no interactions between regions in the radial direction. The Dancoff correction for HRG was obtained using the Bell prescription.⁽⁸⁾ Resonance integral calculations were done for ^{239}Pu and ^{240}Pu . The "wings" ^{240}Pu ⁽⁹⁾ cross section prescription was used for the 1 eV ^{240}Pu resonance. A more detailed discussion of ^{240}Pu cross sections is presented in the next subsection.

The four group cross sections were used in a one-dimensional radial diffusion theory calculation with HFN to determine the spatial distribution of the four-group fluxes and the effective neutron multiplication. The HFN calculation is also part of the ZODIAC G calculation. The transmutation calculation in ZODIAC G is done with the ALCHEMY⁽¹⁰⁾ code. The four group cross sections calculated for the unirradiated core were used throughout burnup. An HFN calculation was done at each burnup step with the updated isotopic densities.

Three isotopes were included in the calculations to account for fission products. Xenon-135 and samarium-149 were included as fission products in equilibrium throughout most of the core life. In addition, a pseudo fission product based on the information published by Nephew⁽¹¹⁾ was used to represent fission products not in equilibrium.

B. Evaluation of ^{240}Pu Cross Sections

The 1967 version of HRG usually used in conjunction with the "standard" ^{240}Pu cross section data allocates the entire 1 eV resonance to one fine group, and hence the wings of the resonance are subject to the same amount of self-shielding as the peak of the resonance. For systems containing a large amount of ^{240}Pu , the net result is a spectrum averaged ^{240}Pu cross section which is too small. The "wings" prescription for ^{240}Pu cross section data is an allocation of the ^{240}Pu cross section which places a portion of the 1 eV resonance in the adjacent energy groups in HRG. The resulting spectrum average cross section is larger because the data in these adjacent groups is not self-shielded using resonance parameters. With this data, reactivity calculations for the MTR-Phoenix critical experiment are in good agreement with experimental results.^(12,13) The two sets of ^{240}Pu cross section data used with the 1967 version of HRG give significantly different results as shown in Table II for region 7 with double Pu (23.3% ^{240}Pu). This region contains more fuel than any other region in the inner annulus.

Since the allocation of the data in the "wings" ^{240}Pu cross section set is a prescription and is system independent, more sophisticated epithermal calculations with THERMOS and EGGNIT,⁽¹⁴⁾ were performed to determine the validity of using the "wings" ^{240}Pu cross section data for Phoenix fueled HFIR systems. The average ^{240}Pu cross section from 0.683 to 2.38 eV and the infinite neutron multiplication, k_{∞} , calculated with the various methods are summarized in Table II for region 7. The THERMOS calculation has 26 energy groups between 0.683 and 2.38 eV and calculates an average ^{240}Pu cross section (σ_a^{240}) using point cross sections and the calculated

Table II

CALCULATED RESULTS FOR REGION 7

FUELED WITH DOUBLE Pu-23.3% ^{240}Pu

<u>Parameter</u>	<u>HRG (1967 Version)</u>		<u>THERMOS</u>	<u>EGGNIT</u>	<u>HRG (1969 Version)</u>	
	<u>"Standard" ^{240}Pu</u>	<u>"Wings" ^{240}Pu</u>			<u>Allocation</u>	<u>All Changes</u>
Microscopic ^{240}Pu cross section from 0.683 to 2.38 eV, σ_a^{240} (barns)	438	927	885	749	697	597
Infinite medium neutron multiplication, k_∞	1.52	1.40		1.45	1.46	1.47

space-energy flux distribution for the 26 groups. The average ^{240}Pu cross section calculated with THERMOS is nearly the same value as the one calculated with HRG using the "wings" ^{240}Pu cross section data, thus implying that the "wings" ^{240}Pu cross section data is sufficiently accurate for these calculations. Since THERMOS does not calculate a scattering cross section, k_{∞} cannot be determined.

A calculation with EGGNIT gives a smaller value for σ_a^{240} and hence a larger k_{∞} than either the "wings" ^{240}Pu HRG calculation or the THERMOS calculation gives. The differences in σ_a^{240} are not large enough to significantly affect the reactivity slope with burnup; however, the initial reactivity of the system may be changed enough to alter the calculated core lifetime appreciably.

A new version of HRG (1969) which allocates the resonance cross section to more than one fine group⁽¹⁵⁾ has become available for routine use. Using the new version of HRG is theoretically more sound than using the "wings" ^{240}Pu cross section data with the 1967 version of HRG; however, this investigation was initiated before the new version was available for routine use. The new version of HRG gives a larger σ_a^{240} and a smaller k_{∞} relative to the 1967 version with "standard" ^{240}Pu cross section data. These results are also listed in Table II.

Other changes have also been made to the HRG calculation. These include: using an updated cross section data tape with the best cross sections available, including the upscattering effects of H_2O ,⁽¹⁶⁾ and using the intermediate resonance integral formulation for the admixed scatterers in the fuel.⁽¹⁷⁾ Results incorporating all of these changes are listed in the last column of Table II. The effects of each change are discussed in

more detail in Reference 9. Thus, the "wings" ^{240}Pu cross section data may give slightly high values for σ_a^{240} for the most heavily fueled systems, but the results are more accurate than using the "standard" ^{240}Pu cross section data in the 1967 version of HRG. The resulting calculated neutron multiplications are expected to be slightly low if σ_a^{240} is high.

C. Reactivity and Burnup Results

Reactivity and burnup results are presented in Tables III-VI and Figures 5-8 for the various systems. All fluxes are axial averages assuming 100 MW reactor power. Target region fluxes are also radial averages from 0.0 to 4.32 cm assuming a target load of 265 grams of ^{242}Pu . Fluxes for Region 7 are radial averages from 9.33 to 11.74 cm. This region contains most of the inner fuel and has the lowest thermal flux of the regions in the inner annulus. The last set of fluxes, at 30.05 cm, occurs at a point in the beryllium reflector approximately where the thermal flux peaks in the reflector. Neutron multiplications and fluxes at time zero are for operating systems but containing no fission products. Isotopic concentrations, flux distributions, and neutron multiplications were calculated after 1, 4, 10, 20 and 30 days of full power operation. As mentioned previously, the same four group microscopic cross sections were used throughout burnup.

The normal HFIR U_3O_8 core lifetime was calculated to be 24 days assuming an average burnup in the axial direction and no reactivity control. This is in good agreement with the actual operating lifetime for a HFIR core of about 23 days.

A second burnup calculation for the uranium system was done with the reactor just critical at each time step. To make the system just critical,

a search was made on the tantalum concentration in the control region, which is just outside of the outer fuel annulus. When the tantalum is included in the system, the power is shifted toward the center of the reactor and the fluxes in the target region are increased by about 20% as shown in Table III. The fluxes in Region 7 are increased about 15% but at 30.05 cm the thermal flux is decreased 15% and the total flux is decreased 10%. The core lifetime is calculated to be 24 days, so the tantalum has a negligible effect on it.

Previous calculations⁽³⁾ with high exposure plutonium substituted for uranium indicated that at 100 MW a Phoenix loading might give higher fluxes in the target and/or longer core lifetime for the High Flux Isotope Reactor. However, the geometrical model was not detailed enough to show the proper trends. Similar calculations have been repeated using the more complex geometrical model. The first burnup calculation was done with an atom for atom substitution of plutonium for uranium (single Pu); the second burnup calculation was done with two atoms of plutonium replacing each atom of uranium (double Pu). Both calculations were done without tantalum control. The plutonium composition used in these calculations is 65.5/23.3/7.7/3.5 at% $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$. Results for these two systems are also shown in Table III.

Substituting the high exposure plutonium for uranium in HFIR increases both the thermal flux and the total flux in the target at time zero for the same total power. The fluxes are increased 25-30% in the single Pu system and 10-20% in the double Pu system. In Region 7, the thermal flux is much lower for the plutonium systems because the fuel is blacker. However, the

TABLE III
REACTIVITY AND BURNUP RESULTS
FOR URANIUM AND PLUTONIUM - 23.3% ²⁴⁰Pu FUELED SYSTEMS

Parameter	Uranium		Single Pu 23.3% ²⁴⁰ Pu	Double Pu - 23.3% ²⁴⁰ Pu	
	Without Control	With Control		"Wings" σ_a^{240}	"Standard" σ_a^{240}
Total flux in target (10 ¹⁵ n/cm ² -sec)	3.475	4.184	4.485	4.196	3.974
Thermal flux in target (10 ¹⁵ n/cm ² -sec)	1.636	1.955	2.002	1.784	1.702
Total flux in Region 7 (10 ¹⁵ n/cm ² -sec)	2.802	3.303	3.323	3.035	2.890
Thermal flux in Region 7 (10 ¹⁵ n/cm ² -sec)	0.359	0.411	0.196	0.081	0.084
Total flux at 30.05 cm (10 ¹⁵ n/cm ² -sec)	2.078	1.872	2.408	2.283	2.160
Thermal flux at 30.05 cm (10 ¹⁵ n/cm ² -sec)	1.352	1.151	1.526	1.419	1.346
Ave. reactivity loss rate, $\Delta mk/k/day$	-2.8	-2.8	-4.6	-2.6	-3.5
Xe-Sm worth, mk	-53		-52	-45	-45
Initial neutron multiplication (no fission products)	1.113	1.000	1.011	1.032	1.093
Days of Full Power Operation	24	24	0	1	13

total flux in Region 7 is higher for the plutonium systems than for the uranium system. Both the thermal and total flux are higher in the reflector for the plutonium systems than for the uranium system.

The reactivity loss rate for the single Pu system is larger than for the uranium system because there are fewer atoms of fissile material in the single Pu system. Analogously, the reactivity loss rate is smaller for the double Pu system than for the uranium and single Pu systems. However, the initial reactivity for both plutonium systems is too low to give any appreciable burnup before reaching end of core life.

The calculations done previously⁽³⁾ utilized the "standard" instead of the "wings" ^{240}Pu cross section data. The two sets of data result in a large difference in the initial neutron multiplication and hence, a large difference in the calculated lifetime. Results of a second burnup calculation for the double Pu system using the "standard" HRG cross section data for ^{240}Pu are also shown in Table III. The total flux in the target, fuel, and reflector is lower than calculated previously⁽³⁾ and lower than calculated using the "wings" ^{240}Pu cross section data. The reactivity loss rate is larger when using the "standard" ^{240}Pu cross sections, but since the excess reactivity available for burnup is also larger, the lifetime is calculated to be 13 days which is still shorter than the lifetime of the uranium fueled core. The differences between the two calculations with the two types of ^{240}Pu cross section data indicate the sensitivity of the results to the ^{240}Pu cross section data. As indicated earlier, the results using the "wings" ^{240}Pu cross section data should be more accurate than the results using the "standard" ^{240}Pu cross section data.

To evaluate the validity of using constant cross sections in the burnup calculation, a reactivity calculation was done for Region 7 using isotopic concentrations at 30 days for the single plutonium system. This fuel region is one of the most important in terms of reactivity, and the effect should be most significant for the single Pu system because the fractional change in plutonium concentration is largest for this system. The infinite medium neutron multiplication for Region 7 is about 1% $\Delta k/k$ higher when the cross sections are recalculated at 30 days. Since the reactivity loss of the system is about 15% $\Delta k/k$ for 30 days of operation, recalculating cross sections at each time step could increase the estimated core lifetime only about 2 days.

The Phoenix fueled HFIR systems containing 23.3% ^{240}Pu give higher fluxes in the target, which enhances the production of transplutonium isotopes. However, if this type of fuel is used in the present HFIR system, the core reactivity lifetime would be shorter than with uranium fuel.

To increase the excess reactivity available for burnup, either plutonium with a lower percentage of ^{240}Pu must be used or the total Pu loading must be increased. Increasing the total Pu loading does not increase reactivity appreciably and is beyond the range of matrix fuel. Cross sections for Region 7 with a double Pu load were calculated assuming an isotopic composition of 80.1/14.4/5.1/0.4 atom% $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$. The infinite medium neutron multiplication for this system is 1.53, or 9.3% higher than with the other type of plutonium. Thus, the neutron multiplication of a Phoenix fueled HFIR with this type of plutonium was expected to be comparable to the uranium system.

Results of burnup calculations with and without tantalum control for this type of loading (double Pu; 14.4% ^{240}Pu) are compared with the results for the uranium system in Table IV. Without the tantalum control, the thermal flux in the target at time zero is the same for the double Pu system as for the uranium fueled system. The total flux in the target is increased 12%, so the production of transplutonium isotopes should be enhanced to some extent. A reactivity calculation for the double Pu system with tantalum control indicates that the thermal flux is increased by 4% and the total flux is increased by 15% in the target region relative to the uranium fueled system with tantalum control. The fluxes in the target for the double Pu system with 14.4% ^{240}Pu are lower than the fluxes for the double Pu system with 23.3% ^{240}Pu because the concentration of fissile atoms is larger in the double Pu system with 14.4% ^{240}Pu .

The change in thermal flux with irradiation for the controlled and uncontrolled uranium and double Pu (14.4% ^{240}Pu) fueled systems is shown in Figure 5. As a system produces power, the concentration of fissionable isotopes decreases and the thermal flux in the fuel must increase to maintain the same power. When the thermal flux in the fuel increases, the thermal flux in the target also increases; thus, the thermal flux in the target increases as a function of irradiation. In the controlled system the change in power distribution with burnup significantly affects the flux in the target region. The thermal flux increases faster in the uranium system than in the double Pu system because there is a larger percentage change in the fissile atom concentration of the uranium system than in the double Pu system as a function of burnup. The average thermal flux throughout lifetime

TABLE IV

REACTIVITY AND BURNUP RESULTS
FOR URANIUM AND DOUBLE Pu -14.4% ²⁴⁰Pu FUELED SYSTEMS

Parameter	Uranium		Double Pu (14.4% ²⁴⁰ Pu)	
	Without Control	With Control	Without Control	With Control
Total flux in target (10 ¹⁵ n/cm ² -sec)	3.475	4.184	3.880	4.819
Thermal flux in target (10 ¹⁵ n/cm ² -sec)	1.636	1.955	1.649	2.033
Total flux in Region 7 (10 ¹⁵ n/cm ² -sec)	2.802	3.303	2.812	3.401
Thermal flux in Region 7 (10 ¹⁵ n/cm ² -sec)	0.359	0.411	0.072	0.085
Total flux at 30.05 cm (10 ¹⁵ n/cm ² -sec)	2.078	1.872	2.112	1.825
Thermal flux at 30.05 cm (10 ¹⁵ n/cm ² -sec)	1.352	1.151	1.312	1.064
Ave. reactivity loss rate, Δmk/k/day	-2.8	-2.8	-3.0	-3.0
Xe-Sm worth, mk	-53		-50	
Initial neutron multiplication (no fission products)	1.113	1.000	1.112	1.000
Days of Full Power Operation	24	24	19	19

FIGURE 5. VARIATION OF THERMAL FLUX IN TARGET WITH IRRADIATION TIME FOR URANIUM AND DOUBLE Pu-14.4% ^{240}Pu HFIR SYSTEMS

is higher in the uranium system than in the double Pu system. Thus, on the average, the double Pu system with 14.4% ^{240}Pu does not produce adequate thermal flux in the target even though the thermal flux level is adequate at the beginning of life.

The calculated lifetime for the double Pu (14.4% ^{240}Pu) system is only 19 days (Table IV), so the system is also inadequate from the standpoint of core reactivity lifetime. In comparison with the double Pu (23.3% ^{240}Pu) system, the average reactivity loss rate is larger; the reason being that the ^{240}Pu concentration increases with burnup. Based on these burnup calculations, it was estimated that a double Pu system with 12% ^{240}Pu would give a lifetime of 25 days.

A burnup calculation for a double Pu system with isotopic compositions of 84.0/12.0/3.7/0.3 atom% $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$, but using the microscopic cross sections for the double Pu system with 14.4% ^{240}Pu , does predict a core lifetime of 25 days. The fluxes are still lower in the target region as indicated in Table V. However, if the power distribution is flattened, the fluxes in the target region would be expected to increase a few percent.⁽³⁾ Another burnup calculation was done for the double Pu (12% ^{240}Pu) system with 10% of the plutonium preferentially removed to flatten the initial power distribution. The thermal flux in the target was increased 7% but the lifetime was reduced to 16 days. Reducing the plutonium content by 10% should increase the reactivity loss rate and decrease the initial neutron multiplication to a very small degree. The calculated reduction of 2% in the initial neutron multiplication occurs because the fuel removed contained the highest value of eta ($\eta = \nu \sigma_f / \sigma_a$). The remaining

TABLE V
REACTIVITY AND BURNUP RESULTS
FOR VARIOUS PHOENIX FUELED SYSTEMS WITHOUT CONTROL

Parameter	Uranium Without Control	Single Pu 23.3% ²⁴⁰ Pu	Double Pu 23.3% ²⁴⁰ Pu	Double Pu 14.4% ²⁴⁰ Pu	Double Pu - 12% Original Fuel Distribution	²⁴⁰ Pu Power Flattened
Total flux in target (10 ¹⁵ n/cm ² -sec)	3.475	4.485	4.196	3.880	3.804	3.962
Thermal flux in target (10 ¹⁵ n/cm ² -sec)	1.636	2.002	1.784	1.649	1.618	1.745
Total flux in Region 7 (10 ¹⁵ n/cm ² -sec)	2.802	3.323	3.035	2.812	2.759	2.878
Thermal flux in Region 7 (10 ¹⁵ n/cm ² -sec)	0.359	0.196	0.081	0.072	0.070	0.084
Total flux at 30.05 cm (10 ¹⁵ n/cm ² -sec)	2.078	2.408	2.283	2.112	2.069	2.136
Thermal flux at 30.05 cm (10 ¹⁵ n/cm ² -sec)	1.352	1.526	1.419	1.312	1.286	1.347
Ave. reactivity loss rate, Δmk/k/day	-2.8	-4.6	-2.6	-3.0	-3.2	-3.9
Xe-Sm worth, mk	-53	-52	-45	-50	-51	-51
Initial neutron multiplication (no fission products)	1.113	1.011	1.032	1.112	1.134	1.115
Days of Full Power Operation	24	0	1	19	25	16

fuel should see a softer spectrum and hence a higher value of η . Since this effect was not accounted for in the calculation, the results are expected to be conservative. This effect indicates another possible error in using constant cross sections throughout burnup. An error of 2% in reactivity is worth 6 days in lifetime for this system.

The calculations done for the various plutonium fueled HFIR systems are summarized in Table V and Figure 6. The results indicate that the initial reactivity is most sensitive to the amount of ^{240}Pu in the plutonium, and the fluxes in the target region are most sensitive to the total amount of plutonium in the system. Thus, to increase the fluxes in the target region, the quantity of plutonium in the system should be reduced. However, one would expect to need more plutonium than in a single Pu system because η and hence k_{eff} in all plutonium mixtures except nearly pure ^{239}Pu is lower than η for the uranium system (93% ^{235}U). Alpha (α) is higher for ^{239}Pu than for ^{235}U thus creating more ^{240}Pu than ^{236}U , which would give a larger reactivity loss rate for plutonium.

Therefore, it was estimated that a 1.5 Pu loading (one and one-half atoms of plutonium replacing each atom of uranium) with 8% ^{240}Pu would give a lifetime of 25 days and fluxes in the target region comparable to the uranium fueled system. The ^{240}Pu concentration is much less for this system than the other systems considered thus far, so it is worth less in terms of reactivity. Because of the lower ^{240}Pu concentration the cross sections calculated with the "wings" ^{240}Pu cross section data should be more valid. Even if the calculated ^{240}Pu cross sections are in error, the neutron multiplication will not be significantly affected because of the lower reactivity worth of ^{240}Pu .

FIGURE 6. VARIATION OF THERMAL FLUX IN TARGET WITH IRRADIATION TIME FOR VARIOUS UNCONTROLLED PHOENIX FUELED HFIR SYSTEMS

A burnup calculation was done with a 1.5 Pu loading containing 93.5/6.0/0.5/0.0 atom% $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$. The results are summarized in Table VI. The thermal flux in the target is the same as for the uncontrolled uranium system while the total flux in the target is 10% higher than for the uncontrolled uranium system. The lifetime is calculated to be 21 days and would probably be larger if the spectrum average cross sections were recalculated at each burnup step.

Since this system seems to be a fairly good candidate as a HFIR core, another burnup calculation was done with the fuel redistributed and the appropriate tantalum concentration in the control region to keep the system just critical. In redistributing the fuel, the total amount of fuel was increased by 15% in order to maintain the calculated core lifetime. An estimate of the proper fuel distribution to give a flat power distribution with control was made from the initial power distribution of the uncontrolled system with the original fuel distribution.

The calculated lifetime of this system is 20 days, just slightly less than for the previous system. The initial power distribution is still not quite flat. It would probably be necessary to include boron in the fuel to flatten the power distribution and to control the excess reactivity at the beginning of burnup. The thermal flux in the target region is 14% larger than in the controlled uranium system and the total flux in the target region is 20% larger than in the controlled uranium system. The averages of the fluxes throughout lifetime are about equivalent to the uranium system. The variation of thermal flux and total flux in the target as a function of irradiation for the uranium and 1.5 Pu (6% ^{240}Pu) systems are shown in Figures 7 and 8, respectively.

TABLE VI

REACTIVITY AND BURNUP RESULTS
FOR URANIUM AND 1.5 Pu-6% ^{240}Pu FUELED SYSTEMS

Parameter	Uranium		1.5 Pu - 6% ^{240}Pu	
	Without Control	With Control	Original Fuel Distribution Without Control	Power Flattened With Control
Total flux in target (10^{15} n/cm ² -sec)	3.475	4.184	3.788	4.974
Thermal flux in target (10^{15} n/cm ² -sec)	1.636	1.955	1.643	2.238
Total flux in Region 7 (10^{15} n/cm ² -sec)	2.802	3.303	2.764	3.546
Thermal flux in Region 7 (10^{15} n/cm ² -sec)	0.359	0.411	0.097	0.135
Total flux at 30.05 cm (10^{15} n/cm ² -sec)	2.078	1.872	2.051	1.693
Thermal flux at 30.05 cm (10^{15} n/cm ² -sec)	1.352	1.151	1.285	0.966
Ave. reactivity loss rate, Δ mk/k/day	-2.8	-2.8	-4.8	-4.8
Xe-Sm worth, mk	-53		-55	
Initial neutron multiplication (no fission products)	1.113	1.000	1.160	1.000
Days of Full Power Operation	24	24	21	20

FIGURE 7. VARIATION OF THERMAL FLUX WITH IRRADIATION TIME FOR URANIUM AND 1.5 Pu-6% ^{240}Pu HFIR SYSTEMS

FIGURE 8. VARIATION OF TOTAL FLUX IN TARGET WITH IRRADIATION TIME FOR URANIUM AND 1.5 Pu-6% ^{240}Pu HFIR SYSTEM

The initial total flux in Region 7 is 6% higher in the plutonium system than in the uranium system. The thermal flux in Region 7 is much lower in the plutonium system because of the fuel blackness. At 30.05 cm the total flux is 10% lower and the thermal flux is 18% lower in the plutonium system relative to the uranium system. This occurs because of the larger amount of tantalum control in the plutonium system at time zero. If boron were included with the fuel in the inner annulus as it is in the uranium system, the amount of tantalum required for reactivity control would be reduced and the fluxes in the reflector would be increased.

As indicated previously, the calculated lifetime would probably be longer if the cross sections were recalculated as a function of burnup. The cross sections for the 1.5 Pu system with 6% ^{240}Pu and power flattened with Ta control were recalculated using isotopic concentrations at 20 days. The neutron multiplication is increased 2.4% $\Delta k/k$ which implies an increase in lifetime from 20 days to 26 days. The thermal flux in the target is decreased 3%.

The results for the 1.5 Pu (6% ^{240}Pu) system show comparable target fluxes and core reactivity lifetime to the uranium system. More detailed calculations are needed to accurately design a Phoenix fueled HFIR system with either or both core lifetime and target flux level advantages.

V. FUEL MANUFACTURING CONSIDERATIONS

The standard HFIR fuel element is a plate fuel consisting of an aluminum-clad dispersion of 30 and 40 wt% 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 wt% U_3O_8 , the cermet consists of 19 vol% U_3O_8 and 81 vol% 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 metallic matrix and requires a minimum matrix content of 50 to 60 volume percent. The weight content for several candidate plutonium fuel materials⁽³⁾ at matrix contents of 50 and 60 volume percent is shown in Table VII. 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.

TABLE VII

FUEL CONTENTS OF PLUTONIUM COMPOUNDS IN AN ALUMINUM MATRIX

Density of Plutonium Compound	Matrix Volume	
	50%	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%

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. PuO₂ 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.

From the irradiation damage standpoint, Idaho Nuclear Corporation has shown⁽³⁾ that the uranium-aluminum intermetallic UAl₃ is superior to U₃O₈, and the dispersion of UAl₃ 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 contains the intermetallic compounds PuAl₂, PuAl₃, and PuAl₄. If the plutonium intermetallics are as resistant to irradiation damage as UAl₃, 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.

Some development work has been done at PNL 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.⁽³⁾

In summary, it looks like 70 wt% 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 would be independent of both the fuel material and fuel content. From a cost standpoint only, a 70 wt% 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.

VI. POTENTIAL DESIGN

A Phoenix fueled HFIR system should utilize the overall plant without major modifications. There should be no problems in using the current plant because quantities and parameters external to the core are relatively independent of the type of fuel used. Radiation damage within the core may be the largest problem, but the current plant should handle any foreseeable changes very adequately.

As much of the current core design as possible should be retained to minimize new design efforts and hold down costs. The current design has been proven and much of the experience can be useful in a Phoenix fueled core. Thus, basic descriptions of the target, fuel elements, control system and reflector are not changed in the potential design. The composition of the fuel

plates can be changed with a minimum effort on overall core design and a negligible effort on plant design.

The primary objective of fueling the HFIR with Phoenix fuel is to increase the core lifetime. The uranium fuel is very expensive and has a rather short operational lifetime. The fuel elements are very complex and costly to manufacture. Using plutonium as the fuel should not increase the manufacturing cost appreciably. Thus, the average cost per operational day should be reduced if the core lifetime is increased.

The primary purpose of HFIR is to produce transplutonium isotopes - namely ^{252}Cf . The production rate of ^{252}Cf is approximately proportional to the third power of the flux in the target. Therefore, the flux in the target must not be lowered in a new core design. Increasing the flux in the target is a strong secondary objective.

A Phoenix fueled HFIR core should also be designed so the control system is not changed greatly. The control rods are not as effective when plutonium is used because plutonium cross sections are larger than uranium cross sections. Also, since plutonium has a smaller delayed neutron fraction than uranium, the control system may have to be modified to make it compatible with the kinetics of a plutonium system. This problem was not evaluated in detail.

Utilizing the results of the neutronic investigations and following the ground rules just presented, a preliminary Phoenix fueled HFIR design was derived. Plutonium with a low concentration of ^{240}Pu must be used in order to have a large amount of excess reactivity available for burnup. The plutonium loading will have to be larger than the uranium loading in order

to get a flatter reactivity loss rate. However, the plutonium loading cannot be too large or the flux in the target will be too low. Boron will have to be included in the fuel plates to help control the excess reactivity at the beginning of life as is done in the uranium core. The distribution of fuel within the plates will have to be altered in order to maintain a relatively flat power distribution. The plutonium loading designated 1.5 Pu (6% ^{240}Pu) satisfies the ground rules. This system should be close to an optimum design. The fluxes in the target are higher at the beginning of life but lower at the end of life so that the average throughout lifetime is about the same.

VII. CONCLUSIONS

The results of the study indicate that the initial reactivity of a Phoenix fueled HFIR core is most sensitive to the amount of ^{240}Pu in the plutonium, and the fluxes in the target region are most sensitive to the total amount of plutonium in the system. Thus, to obtain high fluxes in the target region, the quantity of plutonium in the system must be as small as possible. On the other hand, more plutonium is needed than in a single Pu system because η and hence k_{eff} in all plutonium mixtures except nearly pure ^{239}Pu is lower than η for the uranium system (93% ^{235}U). In addition, α is higher for ^{239}Pu than for ^{235}U thus creating more ^{240}Pu than ^{236}U , so the reactivity loss rate would be larger for plutonium than for uranium.

A HFIR core fueled with plutonium will give a longer core lifetime with comparable flux levels in the target if the plutonium is about 1-1/2 times the uranium loading and contains about 6% ^{240}Pu . Power distributions, kinetic parameters and controllability are larger problems in a plutonium loading, but should not require a great effort for a HFIR core.

Plutonium fueled systems can give higher neutron fluxes than uranium fueled systems because of the greater number of neutrons liberated per fission. The fissile isotopes of plutonium have relatively high capture-to-fission ratios so a large amount of nonfissile isotopes is produced and the reactivity loss rate tends to become larger. The nonfissile isotope ^{240}Pu converts into ^{241}Pu by the capture of a neutron; however, this process requires very economical neutron utilization.

Small systems like HFIR have a large amount of neutron leakage which competes with the sequential capture-fission process for neutrons. Neutron absorption in nonfuel materials also utilizes many of the neutrons in HFIR. Due to the competition for neutrons, not many are left for the process of converting ^{240}Pu to ^{241}Pu and then fissioning the ^{241}Pu . Therefore, the fuel must consist of mostly fissionable isotopes and the ^{240}Pu concentration must be relatively low.

Consideration of design changes was limited to the plates in the fuel elements so the additional costs of any potential design would be minimized. A more optimum plutonium core could probably be designed by changing the other parameters such as core size, fuel plate thickness, and coolant channel thickness.

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ADDENDUM

Since the HFIR study was completed, a burnup experiment utilizing Phoenix fuel has been performed in the MTR. For the same control margin, the core lifetime of the Phoenix fueled MTR is longer than the core lifetime of the uranium fueled MTR by approximately a factor of two. However, the lifetime of the Phoenix core in the MTR was about half of what had been predicted⁽¹⁸⁾ using calculational methods similar to the methods used in the HFIR study. This occurred because the measured reactivity slope was twice as great as the calculated reactivity slope. The fission product model used in the calculations is suspected to be the major contributor to the discrepancy.

Despite the discrepancy between the MTR experiment and calculation, most of the results and conclusions for the HFIR systems are still valid. The neutron flux in the target is not influenced by the fission products. Thus, the sensitivity of neutron flux in the target to the plutonium content is valid even if the fission product model is in error. The sensitivity of core lifetime to the plutonium composition is not affected by the fission product model either; although the reactivity slopes would be steeper if the fission product model were changed as indicated by the MTR results.

The conclusions reached in the study should be valid also because the results for Phoenix fueled HFIR cores were compared with the calculated results of the uranium core. Any errors due to the fission product model should exist in the uranium calculation as well as in the plutonium calculation. However, the fact that the uranium results agree with operational data indicates that there may be compensating errors in the calculations.

The general trends observed for the plutonium systems are valid even if some unknown problem exists with the plutonium calculations but not with the uranium calculations. The neutron flux in the target is very sensitive to the

plutonium content but relatively insensitive to the plutonium composition, whereas the initial reactivity and core lifetime are most sensitive to the plutonium composition. Thus, a plutonium core with a loading 1.5 times the uranium loading and utilizing plutonium containing 6% ^{240}Pu would still be the best Phoenix fueled HFIR system.

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