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DESIGN GUIDES FOR IRRADIATION EXPERIMENTS WITH FAST SPECTRUM REACTOR FUEL ELEMENTS IN THERMAL TESTS REACTORS

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16. Abstract

Multigroup calculations provide a comparison of various techniques for simulating the effects of a fast reactor spectrum during the irradiation of fueled experiments in thermal test reactors. Spectral requirements are met by thermal flux spectra that are modified by removing thermal neutrons with cadmium and/or boron-10 absorbers. Power profile requirements in fueled samples are met by reducing the uranium-235 enrichment and/or reducing the sample size.

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SUMMARY

Fast spectrum reactor fuel elements must presently be irradiated in thermal spectrum test reactors. Therefore, modifications to the thermal spectrum test environment are necessary to simulate the fast spectrum or its effects.

The simulation of a fast reactor spectrum, the absorption activity induced in fuel cladding, and the power level and profile generated in a fuel element while in a thermal spectrum were studied by using multigroup neutron transport calculations. A conceptual design of a fast reactor provided a representative fast reactor spectrum and a model fuel element, and the NASA Plum Brook Reactor provided a typical thermal spectrum test environment.

In a core test hole of the thermal reactor, the flux is high enough to allow the use of thermal neutron filters to simulate a fast spectrum or its induced power distributions and still produce accelerated burnup. A 0.25-centimeter filter of $B_4^{10}C$ simulates the fast spectrum above 5 keV, but below that energy (and above 0.2 keV) the simulated spectrum has a much larger fraction of neutrons than a typical fast reactor spectrum. Generally then, for irradiation mechanisms that are not important for neutron energies between 5.5 and 0.2 keV or for irradiation effects that are relatively independent of neutron energy, such as power, filtering is a good method of simulation.

In a reflector test hole of the thermal reactor, the flux is too low to allow both spectral simulation and accelerated burnup. The required flat power profiles of fast reactor fuel elements can be simulated in this location, however, by reducing both the sample size and the uranium-235 enrichment.

INTRODUCTION

At the present time, the development of fuel elements for fast spectrum reactors requires that irradiation tests be performed in thermal test reactors. Thus, for the irradiation of fueled experiments, the spectral characteristics and effects provided by a thermal test reactor must be modified to simulate the spectral characteristics and effects of a fast reactor. Reference 1 describes neutron filters which are used to simulate fast spectra by absorbing thermal neutrons and transmitting epithermal neutrons. However, under some circumstances it may be practical to simulate only an effect, such as power profile, of the fast spectrum rather than the spectrum itself.

In this work, various techniques for simulating fast reactor irradiation characteristics in thermal test reactors have been studied by using multigroup calculations. The irradiation characteristics and effects considered are the flux spectrum "seen" by the fuel, the power level and profile in the fuel, and the absorption rate in the fuel cladding. A conceptual design of a fast reactor provides the reference fast reactor spectrum and a model fuel element. The NASA thermal test reactor at Plum Brook provides the reference thermal reactor spectrum. The generalized results provide design guides for fueled experiments.

DESCRIPTION OF REACTORS

Reference Fast Reactor

A cylindrical array of cylindrical fuel elements reflected by an annulus of molybdenum was chosen as the reference fast spectrum reactor (ref. 2). Figure 1 shows a plan

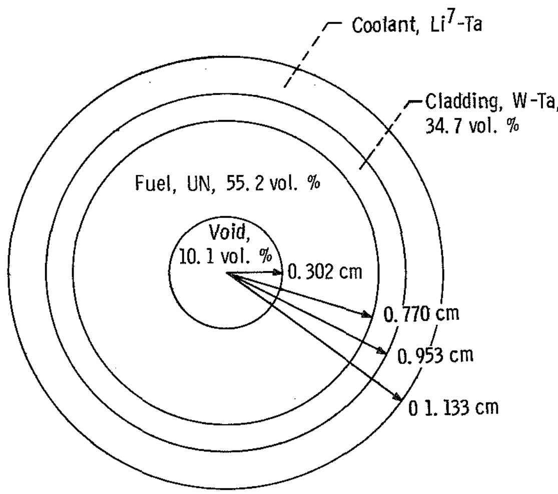


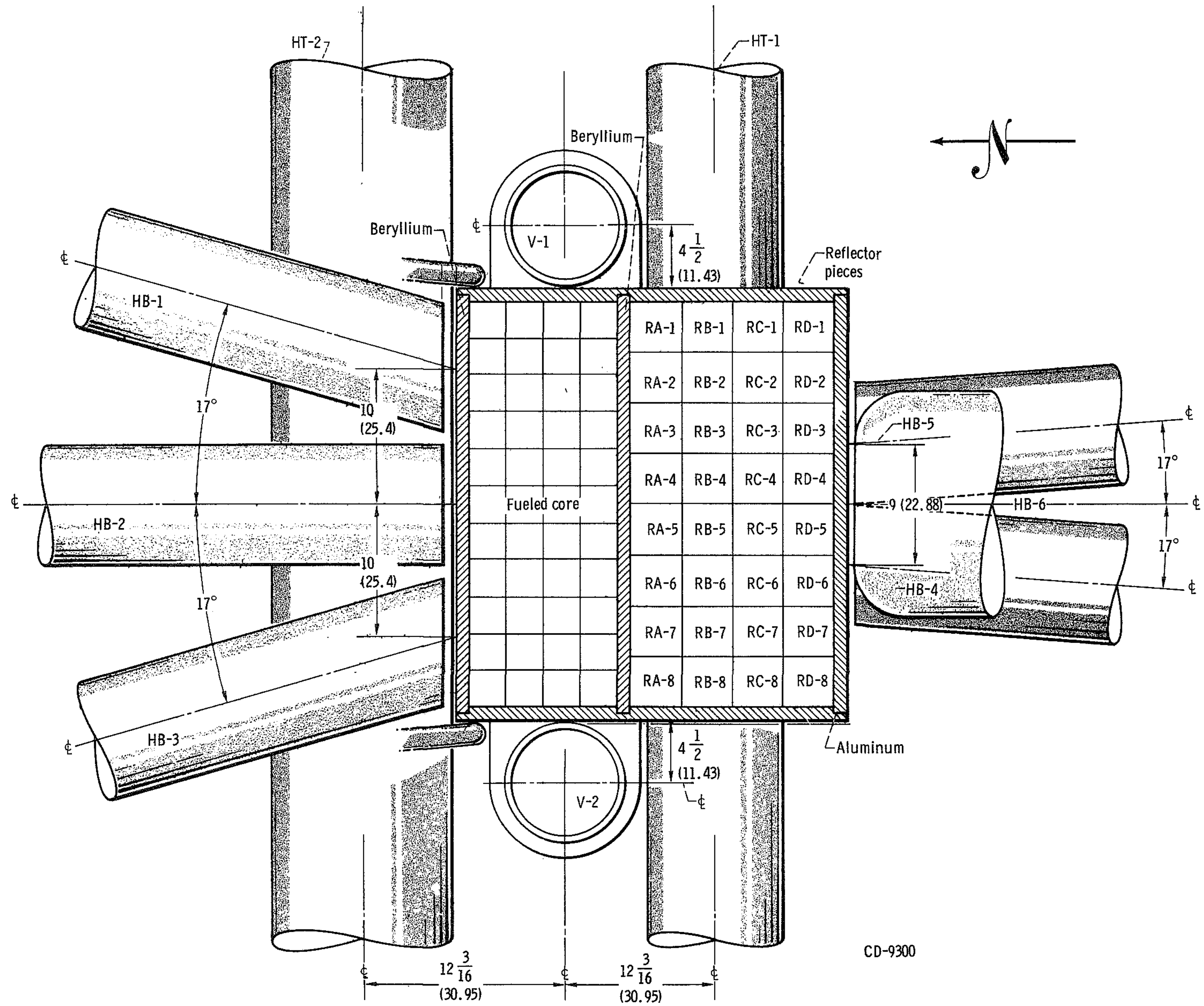
Figure 1. - Plan view of fuel element (see table I for composition of regions).

TABLE I. - COMPOSITION OF MATERIALS

Component	Material	Composition, vol. %	Nuclear density, atoms/(b)(cm)
Fuel element	Uranium nitride fuel	93.2 U^{235} 6.8 U^{238} N	0.03253 .002373 .0349
	Tungsten liner Tantalum cladding	8.79 W 91.21 Ta	0.00555 .05044
	Lithium-7 coolant Tantalum channels	85.65 Li^7 14.35 Ta	0.03753 .007936
Driver	Homogenized fast spectrum reactor (fully enriched UN)	10 Void 25 Li^7 15 Mo 15 Ta 35 $\left\{\begin{array}{l} U^{235} \\ U^{238} \\ N \end{array}\right.$	-------- 0.01096 .0096 .008295 .01138 .0008306 .01222
	Homogenized PBR core	58.3 Water $\left\{\begin{array}{l} H \\ O \end{array}\right.$ 41.7 Al Bal. $\left\{\begin{array}{l} U^{235} \\ U^{238} \end{array}\right.$	0.03906 .01953 .0251 .0001454 .00001095
	Homogenized PBR reflector	11.1 Water $\left\{\begin{array}{l} H \\ O \end{array}\right.$ 88.9 Be	0.007437 .003718 .1099
Filter	Natural cadmium	-------------	0.0464
	Boron-10 carbide $\left(B_4^{10}C\right)$	B^{10} C	0.1164 .0291

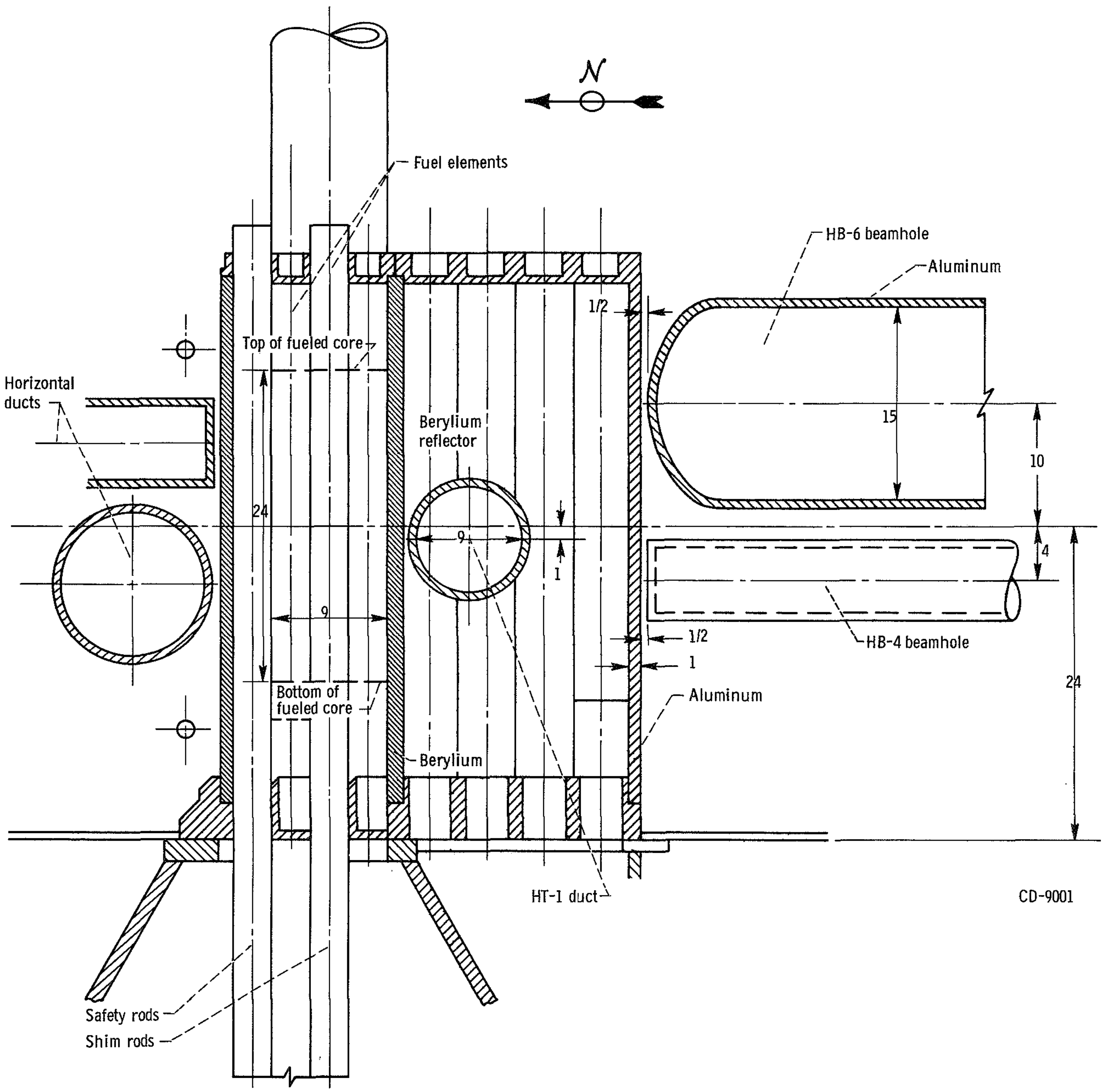
view of a homogenized calculation model of the cylindrical fuel-element pin; table I lists the composition of the regions. The central void region provides for expansion of the fuel with burnup. The fuel itself is an annulus of uranium nitride (UN) enriched to 93.2 percent uranium-235 (U^{235}) and encircled by a cladding of tantalum with a tungsten liner. The coolant passages are tantalum and are considered to be homogenized with the lithium-7 (Li^7) coolant. The actual fuel element will use a tantalum alloy (T-111) instead of pure tantalum and the coolant would contain some lithium-6.

The fast spectrum reactor has an average total unperturbed flux (above 5.5 keV) of about 10^{14} neutrons per square centimeter per second (neutrons/(cm^2)(sec)). The maximum flux at the center of the fast reactor is 1.6×10^{14} neutrons/(cm^2)(sec).



(a) Horizontal plan view.

Figure 2. - Plum Brook Reactor (PBR) core, reflectors, and experiment test holes. (Dimensions are in inches (cm).)



(b) Vertical section (cutaway at north-south core vertical midplane).

Figure 2. - Concluded.

Thermal Test Reactor

The Plum Brook Reactor (PBR) of NASA was chosen as a representative thermal spectrum reactor. The PBR, which can generate 60 megawatts-thermal power, has uranium-aluminum fuel elements with the uranium fully enriched to about 93-percent U^{235}. The PBR is cooled and moderated by light water (H_2O), and it has a primary beryllium (Be) reflector and a secondary water reflector.

Plan and side views of the PBR core are shown in figures 2(a) and (b). The figures indicate the variety of test locations and, hence, the variety of thermal spectra that is

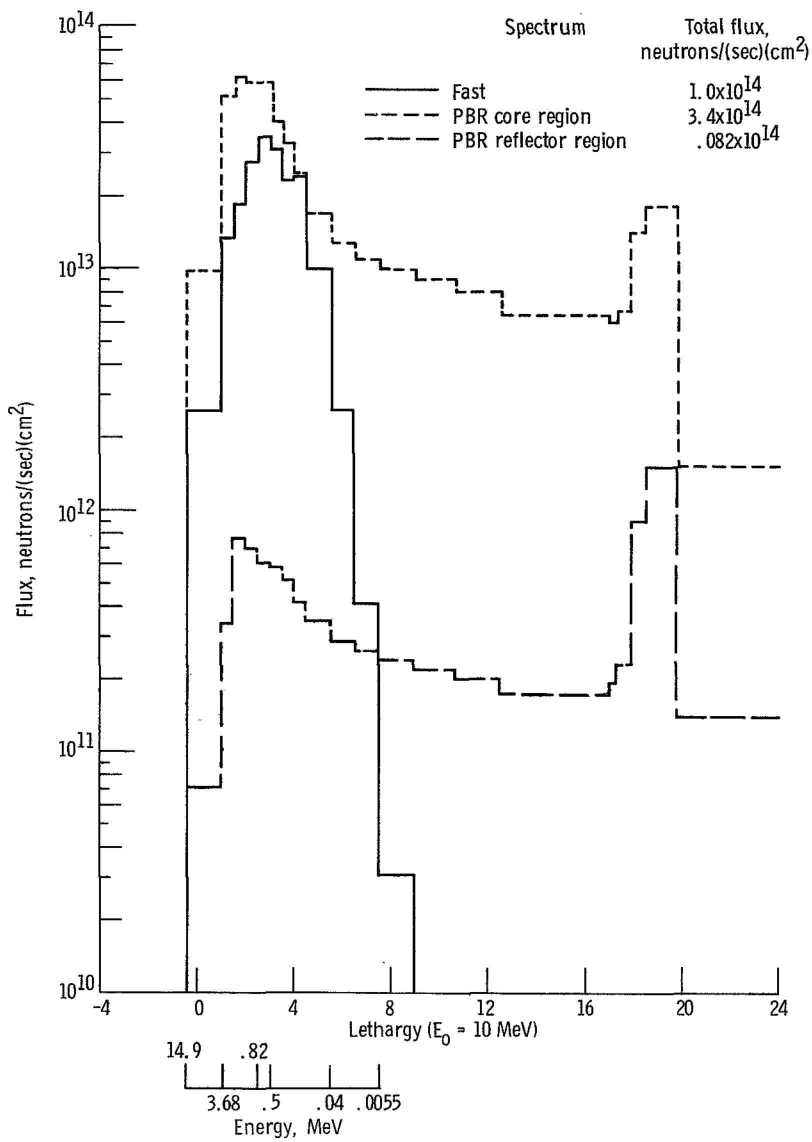


Figure 3. - Comparison of representative fast and PBR spectra. (Numbers in parentheses denote total flux.)

available. For a description of the PBR facility and its neutronic characteristics refer to "Information for Experiment Sponsors for the NASA Plum Brook Reactor Facility," parts I to III.

A PBR spectrum with total unperturbed flux of 3.4×10^{14} neutrons/(cm^2)(sec) can be expected in a typical core-region test location. Similarly, the total flux in a typical reflector location in PBR is 0.082×10^{14} neutrons/(cm^2)(sec) (see fig. 3).

METHODS OF SPECTRAL SIMULATION

The flux spectrum of a fast reactor may be simulated by removing thermal and epithermal neutrons from a thermal reactor spectrum by means of absorbing materials called filters. Reference 1 compares the effectiveness of several filter materials in simulating a fast spectrum. Filtering as it applies to this study is discussed in a later section, entitled FUEL ELEMENT IRRADIATION CHARACTERISTICS.

If the irradiation effect to be examined is highly dependent on the energy of the neutrons, such as resonance absorption, close spectral simulation through filtering will be necessary. However, in some cases, it may be practical to simulate only an integrated effect of a fast spectrum rather than the spectrum itself. For example, the main criterion for simulation of the power profile of the fast spectrum is simply that the profile be flat.

In a fueled experiment, the power profile resulting from the thermal spectrum can be partially flattened by reducing the fuel enrichment, but the consequent drop in fission rate must be offset or accepted. One way to offset a reduced fission rate is to use physically smaller samples in the thermal spectrum than would be used in the fast spectrum. The smaller-scaled samples depress the flux less than the full-scale samples, and thus provide a relatively higher fueled experiment fission rate in the thermal spectrum.

Because of the variation of thermal spectrum with test location within PBR, the conditions for an acceptable simulation of a fast spectrum or its effect will vary throughout the thermal spectrum reactor. Representative flux spectra of the PBR core and reflector regions and the fast spectrum reactor to be simulated are shown in figure 3 (with absolute flux levels indicated).

Figure 3 shows that it is not practical to use a filter in the reflector region because the fast flux level is too low there. If the thermal neutrons were filtered out, there would be no chance of even approaching the irradiation rate attainable in the fast spectrum.

METHODS OF CALCULATION

Because of the radial symmetry of the fast reactor fuel element (fig. 1), its neutronic characteristics are readily calculated in a one-dimensional cylindrical geometry (fig. 4). In figure 4, the thermal neutron filter is represented by an annulus immediately surrounding the fuel element. The reactor, either fast reference or thermal test, is treated as a homogenized annular region surrounding the fast fuel element and, when appropriate, the thermal neutron filter as well. This annular reactor serves as a driver, providing the particular flux spectrum for the test element.

The fast reference driver is a homogenized array of fuel elements. However, because of the irregular geometry (fig. 2) of the PBR it is difficult to determine an equivalent single homogeneous region that allows a reasonably close calculation of the various neutron distributions within PBR. Thus, a desire for a simple geometric model of PBR subject to the calculational limitations led to the choice of two independent homogenized regions which are used in the calculations.

One region, formed from the fueled core (shown in figs. 2(a) and (b)), contains homogenized fuel, water, and aluminum. The other region, which contains homogenized beryl-

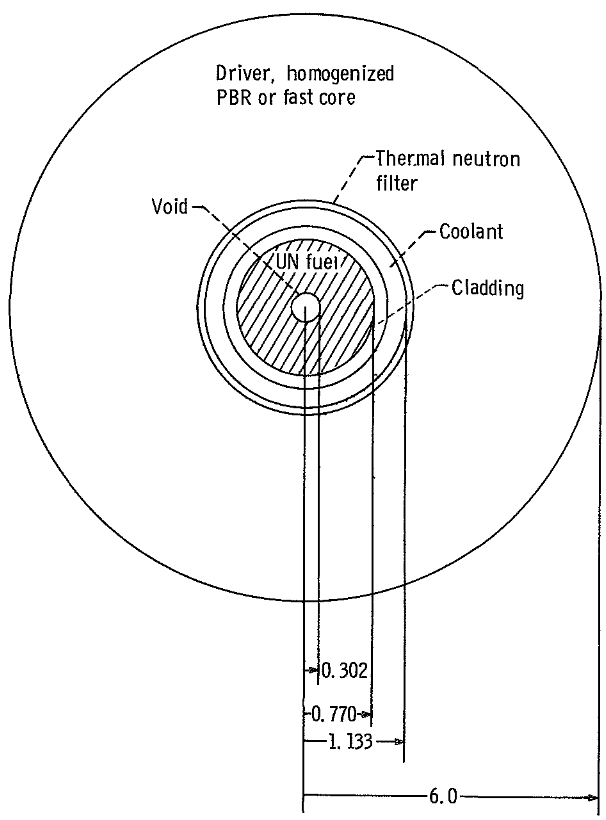


Figure 4. - Geometry used for calculations - one-dimensional cylindrical. (Dimensions are in cm.)

lium and water, is formed from the reflector (fig. 2(b) and the R-region of fig. 2(a)). Either of the two regions can be used as the driver (reactor) region of figure 4, depending on the location of the test in PBR.

Table I lists the nuclear densities of the constituent nuclides of the various driver regions and filters. The GAM-II (ref. 3) and GATHER-II (ref. 4) programs provided multigroup cross sections; the structure of the 20 energy groups used is given in table II. The TDSN program (ref. 5) performed the bulk of the calculations, those to obtain the spatial distribution of neutron flux and fission power.

From figure 2 it is evident that the PBR is a complex three-dimensional array of fuel, reflector, and miscellaneous materials. The approximations for one-dimensional calculations cannot then be expected to provide a good neutronic description of a cylindrical fuel pin in such an environment. A two-dimensional calculation would be more accurate but would require a disproportionate increase in computational time. Furthermore, a two-dimensional model of PBR would still require considerable approximation (ref. 6).

TABLE II. - ENERGY GROUP STRUCTURE

Computer program	Group number	Reduced group number	Low energy boundary	Corresponding lethargy
GAM-II (fast)	1	1	[a]3.68 MeV	1.0
	2		2.23 MeV	1.5
	3		1.35 MeV	2.0
	4		821 keV	2.5
	5	2	498 keV	3.0
	6		302 keV	3.5
	7		183 keV	4.0
	8		111 keV	4.5
	9	3	40.8 keV	5.5
	10		15.0 keV	6.5
	11		5.53 keV	7.5
	12	4	1.23 keV	9.0
	13		214 eV	10.75
	14		37.2 eV	12.50
	15		0.414 eV	17.0
GATHER (thermal)	16	5	0.300 eV	17.322
	17		0.160 eV	17.951
	18	6	0.080 eV	18.644
	19		0.025 eV	19.795
	20		0.0001 eV	25.328

[a]Upper energy boundary is 14.9 MeV (lethargy = -0.4).

However, a comparison of one- and two-dimensional calculations is useful to indicate what bias the simpler treatment has. This comparison is provided in figure 5, which shows the one- and two-dimensional perturbed flux spectra that occur halfway through the fuel annulus and that result from a representative thermal core spectrum. The profile in the fuel, rather than in the driver, was chosen because in the fuel the flux results from the effects of both the axial and radial driver regions. The two-dimensional calculations were performed using six energy groups (table II) in r-z geometry.

The fuel element has a length-diameter ratio of 2.25. The large difference between end and center thermal flux levels indicates that this representative fuel element does not have negligible end effects. It is likely, especially if a radial filter were present, that an axial filter would be used in an experiment to provide more uniform thermal flux over

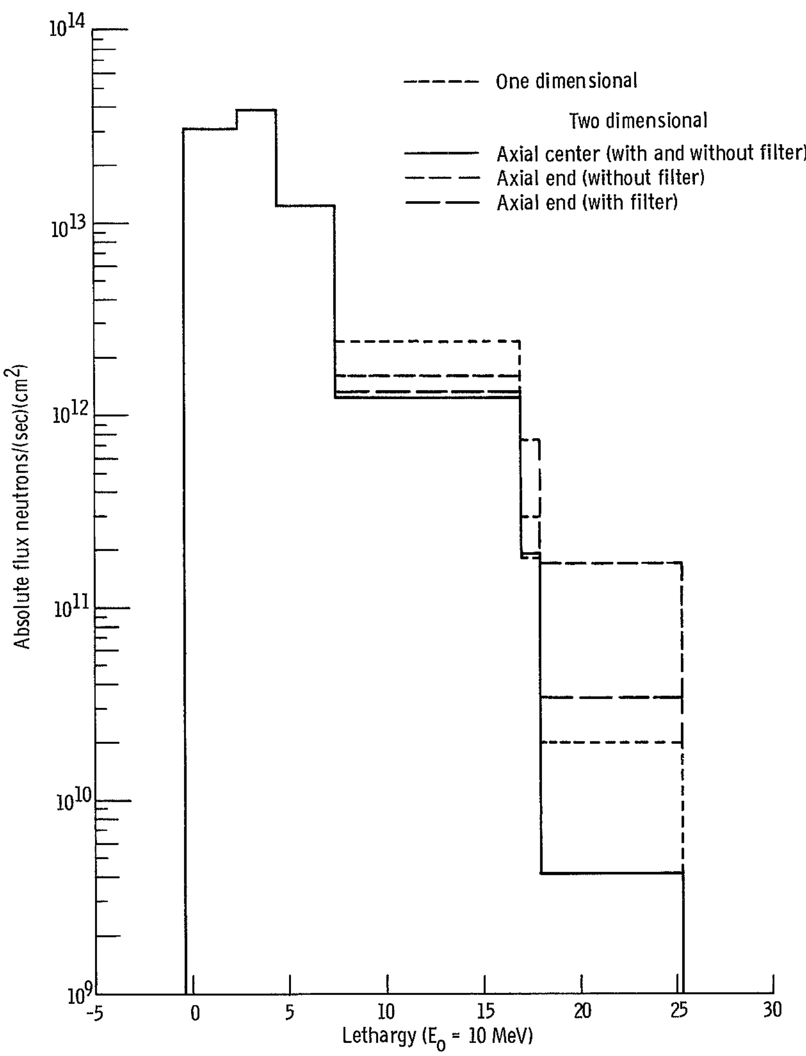


Figure 5. - Six-energy-group neutron flux halfway through fuel annulus. PBR core spectrum; axial filter, 0.05-centimeter boron-10 carbide; no radial filter.

the length of the fuel element. Therefore, the effect of an axial filter (0.05-cm $B_4^{10}C$) is also shown in figure 5.

However, whether or not an axial filter is used, the thermal flux level from the one-dimensional calculation lies between the thermal flux levels at the center and end of the cylinder from the two-dimensional calculation. (The one- and two-dimensional fluxes in epithermal groups 4 and 5 bear no such relation.) Thus, the one-dimensional calculations roughly indicate the average thermal flux over the length of the cylindrical element. The one-dimensional model and calculations which were used for this study offer simply a first-order approximation to the irradiation characteristics.

FUEL-ELEMENT IRRADIATION CHARACTERISTICS

The life of a nuclear reactor might be limited by fission product buildup in a fuel element or by the structural properties of the fuel cladding. Accordingly, some of the irradiation effects in two components, the fuel and the cladding, of a fuel element designed for a fast spectrum reactor were examined. For one component, the fuel cladding, the concern was embrittlement because of element transmutation; and, hence, it was necessary to simulate the flux spectrum. For the other component, the fuel, a spatially uniform fission rate or power profile was sought.

Simulation of Flux Spectrum

Figure 6 shows the neutron flux spectrum at the inside of the driver and/or filter. Included are several simulated fast spectra, the reference fast spectrum, and a thermal reactor core spectrum. Table III lists flux ratios for most of these configurations. These ratios represent the fraction of neutrons above 1.35 MeV, above 0.3 MeV, and below 5.5 keV at the inside edge of the driver or filter. These particular ratios provide a means of measuring the degree of spectral simulation, and some can also be compared to experimental indicators of the degree of simulation (ref. 1). For example, the fission rate of neptunium-237 is a measure of the neutron flux above about 0.3 MeV, and the fission rate of U^{238} is a measure of flux above about 1.3 MeV.

Boron-10 carbide ($B_4^{10}C$) provided the best simulation of the fast spectrum. Even though simulation of the fast neutrons is good, a thermal spectrum filtered by 0.25 centimeter of $B_4^{10}C$ permits a much larger fraction (about 5 percent) of neutrons below 5.5 keV, but above 0.2 keV, than exists in the actual fast spectrum (about 0.1 percent). The simulation could be improved by a greater thickness of $B_4^{10}C$, but the limit of 0.25 centimeter was set in anticipation of physical limitations in the actual experiments.

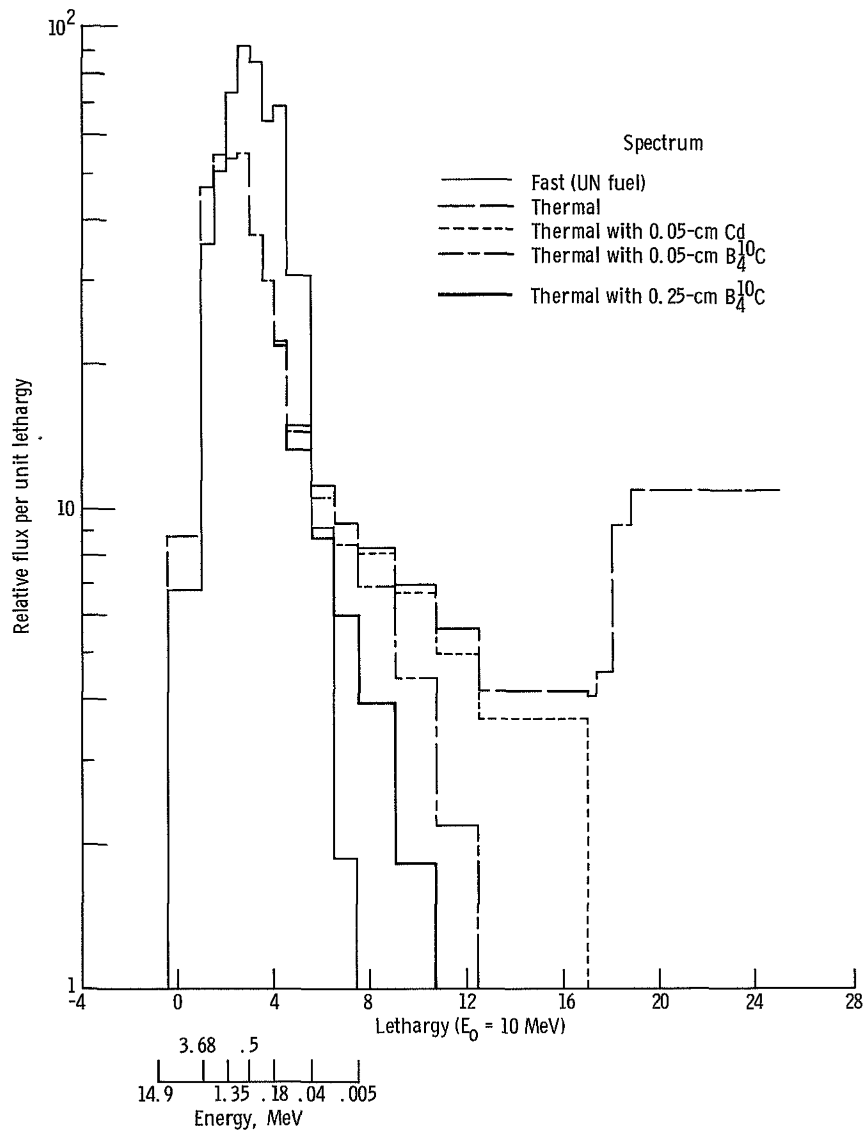


Figure 6. - Relative flux at inside of driver or filter normalized to some total unperturbed flux.

TABLE III. - FLUX RATIOS OF VARIOUS SPECTRA AT

OUTER EDGE OF FUEL REGION

Configuration	Fraction of flux above 1.35 MeV	Fraction of flux above 0.302 MeV	Fraction of flux below 5.5 keV
Uranium nitride fast reactor	0.18	0.62	0.0012
PBR reflector	.08	.19	.64
PBR core:			
No filter	.23	.49	.29
0.05-cm Cd filter	.25	.55	.20
0.05-cm $B_4^{10}C$ filter	.28	.61	.11
0.25-cm $B_4^{10}C$ filter	.31	.68	.049

The high absorption cross section for thermal neutrons which makes B^{10} a good filter also results in a high B^{10} burnup rate and accentuates the problem of helium generation and consequent swelling. Cadmium offers an interesting option because its absorption cross section is virtually all thermal; cadmium could be used in conjunction with B^{10} to retard the B^{10} burnup from thermal neutrons and, once the bulk of thermal neutrons was absorbed, the thickness of the cadmium could be increased to offset its own burnup without significantly changing the simulated spectrum.

Cladding Irradiation

One prospective nuclear fuel cladding is an alloy containing tungsten, rhenium, and molybdenum. But the element rhenium (Re) causes concern because it transmutes into osmium (Os) by the Re(n, β)Os reaction and because transmutation beyond a certain percent of Re atoms (about 0.3 percent) causes the particular alloy to become brittle. Therefore, we would like to provide the same amount of transmutation of Re into Os in the simulated fast spectrum as would be expected in the fast spectrum. Furthermore, damage from fast neutrons (energies greater than 0.82 MeV) will also contribute to the degradation of cladding material properties, and thus close simulation is necessary to get transmutation in proper proportion to fast damage.

In figure 6, the presence of a "low energy tail" (below 5.5 keV and above 0.2 keV) for the simulated spectrum is significant because of the Re absorption cross section in this region. Figure 7 shows the absorption rate or activity in Re for the fast spectrum and two simulated fast spectra. The total unperturbed flux above 5.5 keV is the same

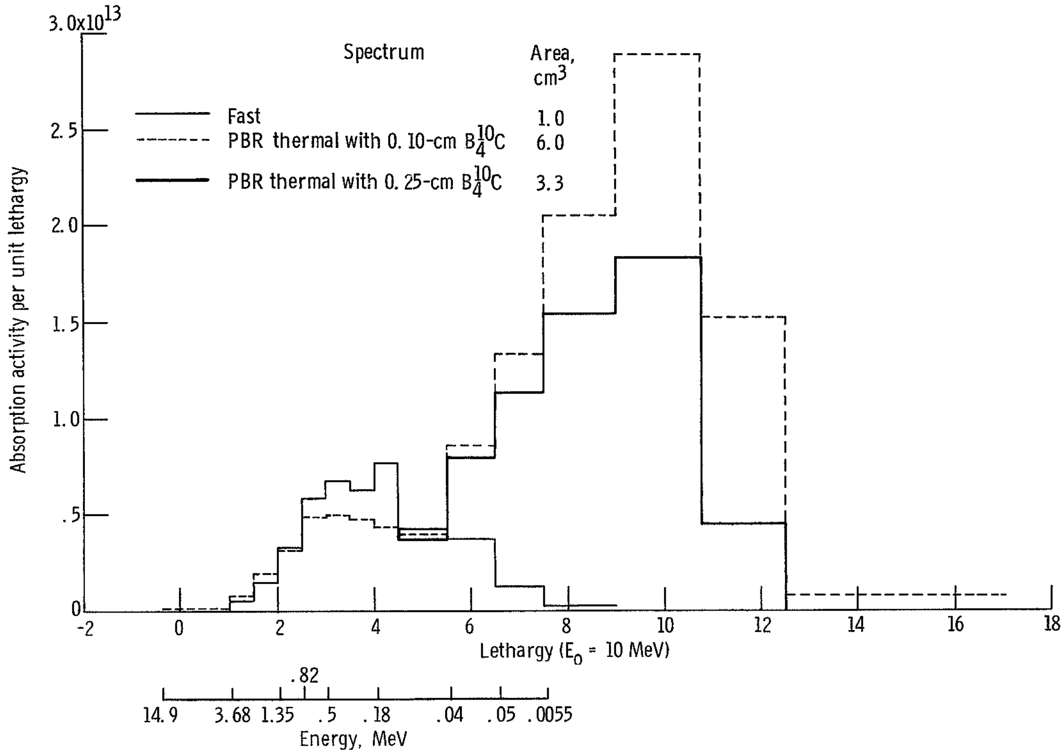


Figure 7. - Rhenium absorption activity in cladding. Unperturbed flux above 5.5 keV is the same for each spectrum.

for each spectrum. The figure shows graphically the added transmutation in the simulated spectrum compared to the fast spectrum.

Assume that the absorption activity in the fast reactor (essentially all above 5.5 keV) is normalized to 1. The 0.10-centimeter $B_4^{10}C$ filter limits the PBR-induced activity below 5.5 keV to about five to six times that above 5.5 keV and to about six times that in the fast reactor. With a 0.25-centimeter $B_4^{10}C$ filter, the PBR-induced activity below 5.5 keV is only about three times that above 5.5 keV and about three times the activity of the fast reactor. In either case, in obtaining the same fast fluence (time-integrated flux above 0.82 MeV) in PBR as the fast spectrum provides, the transmutation in the simulated spectrum will be substantially greater than in the fast spectrum. Practically then, two cladding specimens must be tested, one irradiated to the same fast fluence and another irradiated to the same total number of transmutations (a much shorter time).

Because of the "low energy tail" in this simulated spectrum, the Re cross section can introduce considerable uncertainty in determining the absolute amount of transmutation. Figure 8 shows (as a solid-line histogram) the Re absorption activity for the 0.25-centimeter $B_4^{10}C$ filter using cross sections from the GAM-II compilation. The shaded regions indicate the variation in activity if other cross sections are used. The upper extreme results from the cross sections of reference 7; the lower extreme comes from a recent unofficial Evaluated Nuclear Data File (ENDF/B) evaluation.

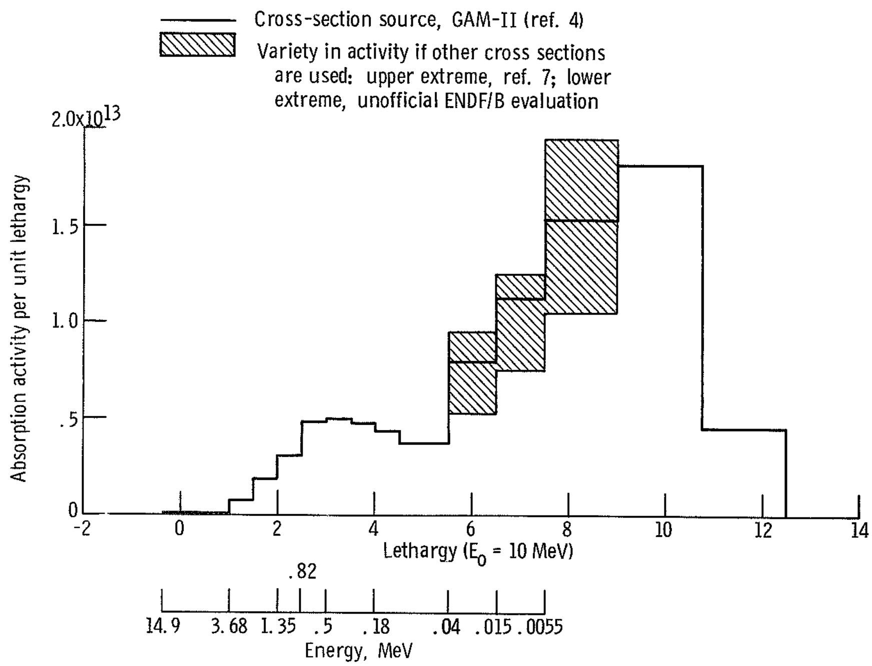


Figure 8. - Rhenium absorption uncertainty due to cross section uncertainty. PBR thermal spectrum with 0.25-centimeter boron-10 carbide filter.

Simulation of Power Profile

This section emphasizes the spatial dependence of the energy-integrated fission rate (power profile) rather than the energy dependence. Because the geometry is one-dimensional, the radial power profile is of interest.

Reflector test location. - Considerations other than the absolute flux level (fig. 3) may determine the location of the test in a thermal reactor. This section discusses the problem of simulation if the reflector region, where thermal neutron filters are generally not acceptable, is used.

Figure 9 shows the absolute fission power profile across the fully enriched fuel pin in the fast spectrum and across somewhat lower-enrichment pins in the reflector region. Note that different fuel-pin enrichments are used and that the fast spectrum provides a greater and somewhat flatter fission rate than can be obtained in the reflector region. (As on most figures with power profiles, the outside-inside power ratio is given.)

However, this rough simulation of fission power profile without a neutron filter has two principal disadvantages: (1) there is still a substantial outside-inside power ratio because of the thermal absorption, and (2) the overall power level gets smaller if we try to improve this power ratio by decreasing fuel enrichment. Furthermore, the fact that U^{238} transmutes to plutonium-239 (Pu^{239}), which is fissionable, may place a lower bound on the enrichment.

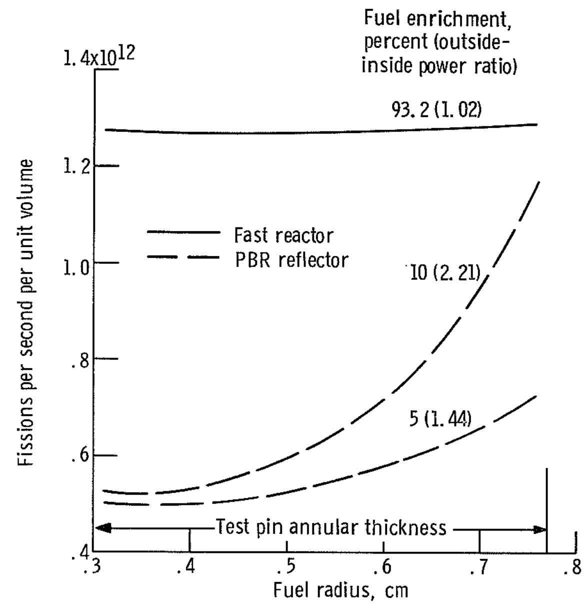


Figure 9. - Absolute fission rates for fast core and PBR reflector region.

Thus, at this typical reflector location, there is no chance of obtaining a relatively flat power profile at a power level as high as that in the fast spectrum. On the other hand, the size of the low-enrichment test element could be reduced in order to increase power density in the element. Figure 10 shows the power distributions for a full-size fuel element and a smaller fuel element with approximately half the diameter (see table IV for comparative dimensions). The power density in the half-diameter pin is a little over three times as great as in the full-diameter element. In addition, the outside-inside power ratios become smaller as the pin size is reduced. Table V compares the power characteristics of these fuel pins. From this table can be determined the relative advantages of reducing fuel-element size or changing the enrichment or cladding thickness in order to obtain a certain power density.

Thus, in a reflector region, a combination of these two techniques (reducing the fuel enrichment and reducing the fuel-element diameter) can provide a sufficiently flat power profile without reducing the power density. In fact, the power density may be greater than that in the fast spectrum.

Core test location. - From figure 3, it is evident that, if the core region were used for testing, the fast flux level would be high enough that the thermal neutrons could be filtered out and there would still be acceptable irradiation rates. Thus, in a core test location, filtering as well as changing the fuel-element size or enrichment can be used to simulate the fast spectrum power profile.

Because using neutron filters may allow more highly enriched fuel to be used in the core than in the reflector, the effect of enrichment on power profile in the core region

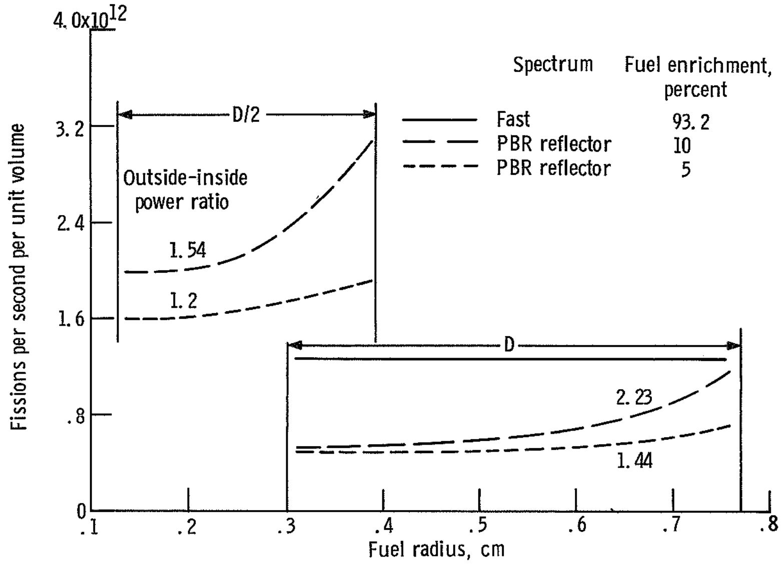


Figure 10. - Fission rates in reflector region for full- and half-size pins.

TABLE IV. - COMPARISON OF FUEL-ELEMENT SIZES

Region	Outer radius, cm	
	Full size	Half size
Void	0.302	0.127
Fuel	.770	.394
Clad	.953	.475
Coolant	1.133	.678
Driver	6.0	6.0

TABLE V. - RELATIVE POWER CHARACTERISTICS OF FUEL ELEMENTS IN PBR REFLECTOR SPECTRUM - NO FILTER

Characteristic	Full size		Half size		Full size with half clad	Absolute value for relative value of 1.0
	Uranium-235 enrichment, percent					
	5	10	5	10	5	
Relative quantity:						
Fuel radius, cm	1.0	1.0	0.5	0.5	1.0	0.302
Fuel volume, cm^3	3.6	3.6	1.0	1.0	3.6	0.437
Cladding thickness, cm	1.0	1.0	0.5	0.5	0.67	0.183
Cladding volume, cm^3	4.48	4.48	1.0	1.0	2.88	0.221
Average power density[a]	1.0	1.28	2.91	4.04	1.18	-----
Integrated power	3.59	4.61	2.91	4.04	4.26	-----

[a]Average power density of the fully enriched full-sized element in the fast spectrum is 2.12.

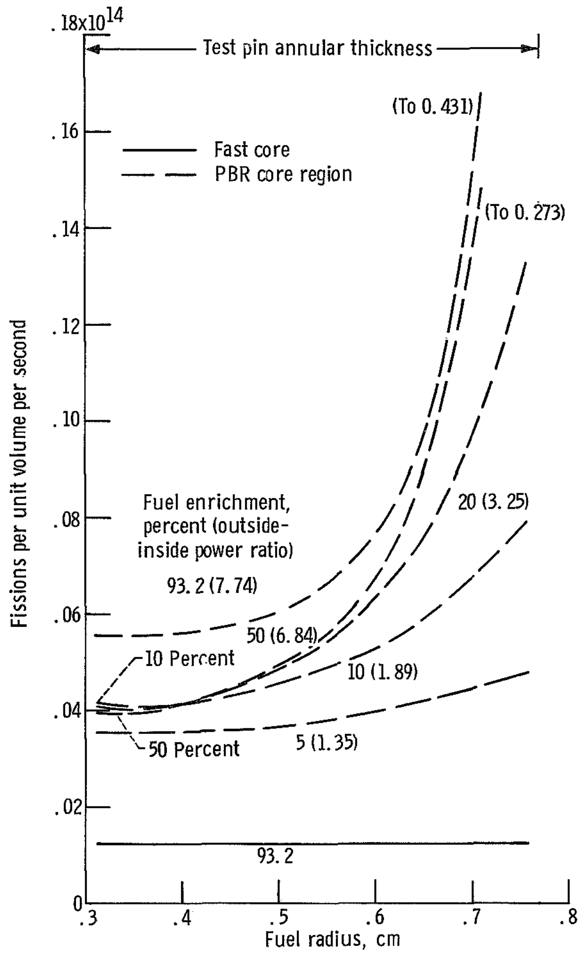


Figure 11. - Absolute fission rates for fast core and PBR core region.

was examined. Figure 11 shows the radial power profile (absolute fission rate per unit volume) in the PBR core region through the full-size test pin for several U^{235} enrichments varying from 5 to 93.2 percent (fully enriched). For the lowest enrichment (5 percent), the power level is still about three times that of the fully enriched pin in the fast spectrum.

The crossover of the 10-, 20-, and 50-percent enrichment curves at the pin interior is caused by different rates of change for two separate effects. As the enrichment is increased, the thermal neutrons are increasingly absorbed in the outer region of the fuel pin and the number of fast fissions throughout the entire fuel pin increases. But because the fission cross section of U^{235} at fast energies (~2 to 3 b) is so much lower than at thermal energy (~577 b), the decrease in thermal fissions overshadows the increase in fast fissions at the interior radius, at least up to 50-percent enrichment.

The power-flattening effect of $B_4^{10}C$ and Cd filters is shown in figure 12. The fuel

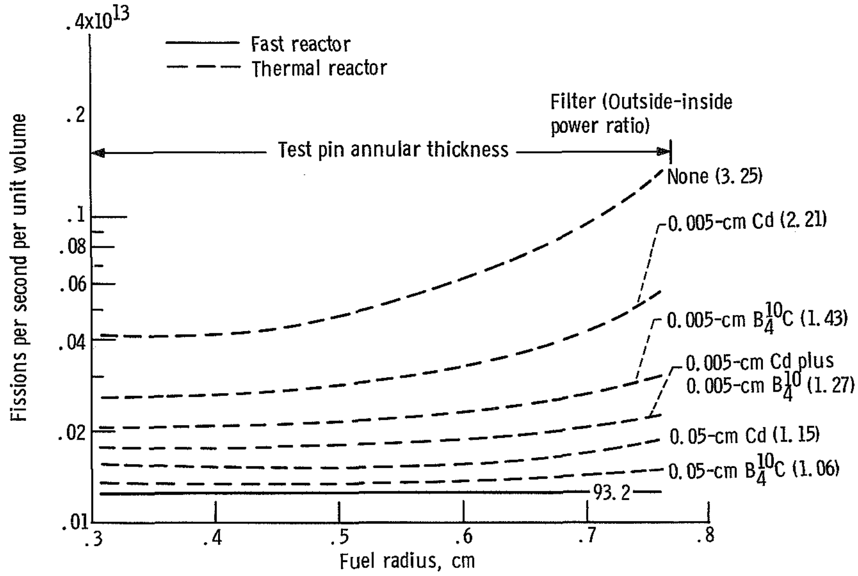


Figure 12. - Radial power profile for 20-percent-enriched uranium-235 nitride with various filters. Fully enriched element profile in fast spectrum shown for comparison.

enrichment in this case is 20 percent, and the maximum filter thickness is 0.05 centimeter. With 0.05-centimeter $B_4^{10}C$, the profile is quite flat (an outside-inside power ratio of 1.06). In fact, it is very nearly the same as the power profile of the fully enriched element in the fast spectrum, which is shown.

Figure 13 contains a summary of power profiles using the various options for simulation. From this figure, the relative results of the simulation techniques of reducing fuel enrichment, reducing element size, filtering, and changing the test location can be compared.

Filtering may also be required to simulate the axial power profile. In the two-dimensional calculation, an axial filter of 0.05-centimeter $B_4^{10}C$ reduced the thermal fission power produced at the end of the fuel element (fig. 5). In an experiment then, it may be desirable to use an axial filter to provide more uniform thermal fission power over the length of the element.

The volume-integrated fission activity resulting from this representative PBR spectrum in the one-dimensional case is 0.374×10^{14} fissions/(sec)(cm^3); in the two-dimensional unfiltered case it is 0.491×10^{14} fissions/(sec)(cm^3); and in the two-dimensional axial filtered case it is 0.427×10^{14} fissions/(sec)(cm^3). Hence, the one-dimensional calculations tend to underestimate the fission rate in this example.

Furthermore, partial axial and radial filters may also be needed to minimize the gradient of the unperturbed thermal flux across the test region. This would make the calculational assumption of no circumferential flux gradient more nearly true.

Experimental comparison. - Direct normalization of the analytical thermal flux to an experimentally measured thermal flux would provide the most useful comparison be-

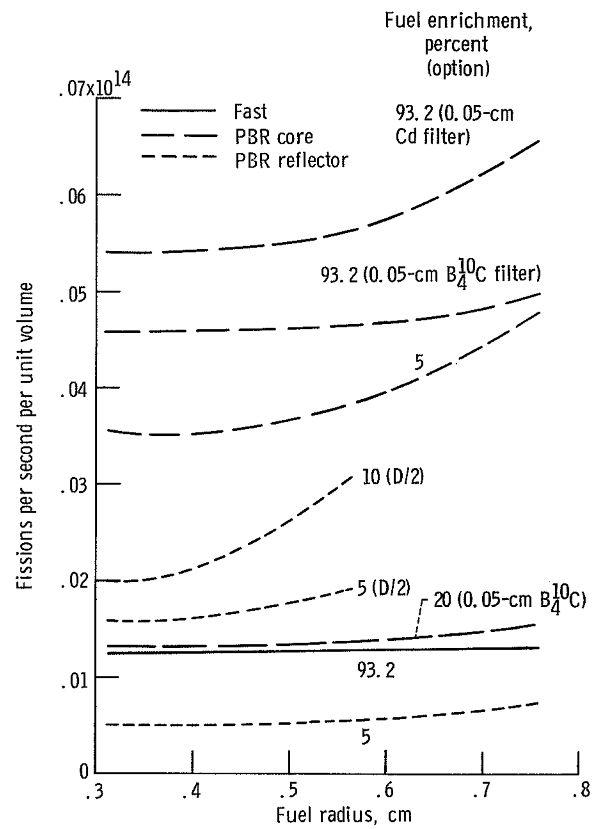


Figure 13. - Fission rate (power distribution for various simulation options).

cause the absolute as well as the relative levels of performance could be derived from the basic data. But this procedure is not always reliable (1) because the energy group structure of the calculations may not coincide with the thresholds and cutoffs of the actual flux detector, and (2) because the calculations usually must assume a simplistic one-dimensional geometry in order to obtain an averaged spectrum, whereas the flux detector is in a particular spectrum influenced by local three-dimensional inhomogeneities.

However, if for analytical comparison generalized flux information not characteristic of a particular position is needed, the calculational techniques and models should be satisfactory. For a quantity that is energy-integrated, such as power, the derived analytical values may be normalized to the corresponding experimental values. This makes the precise location and the calculation of detailed energy dependence less important.

In figure 14 this procedure is demonstrated using power density as the energy-integrated quantity. Experimental points were obtained from the mockup reactor of the PBR; the test pin was placed in the RC-3 position (reflector region) 13.9 inches above the core horizontal midplane (figs. 2(a) and (b)). The calculated smooth curve (for the reflector driver) is normalized to a probable experimental value at 7.5-percent enrich-

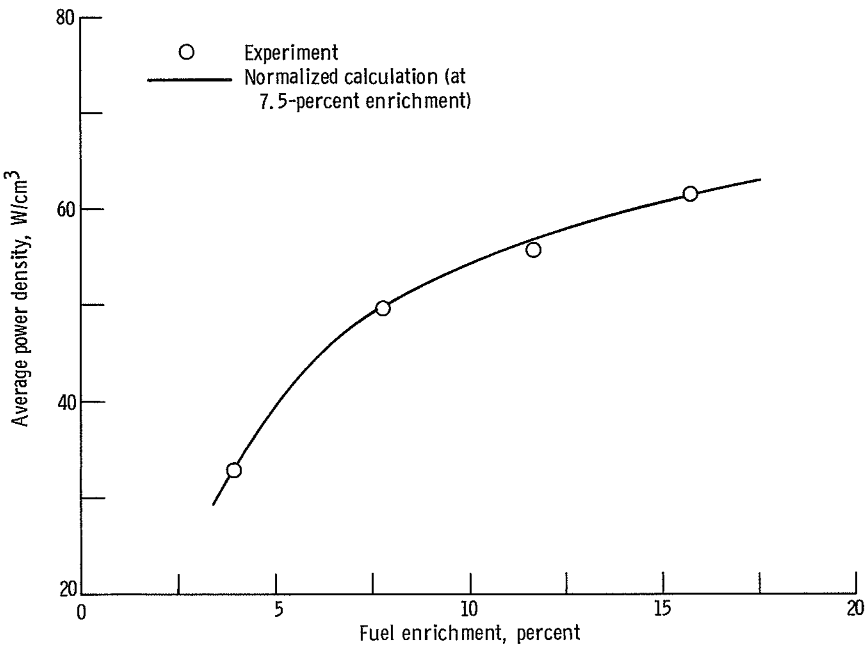


Figure 14. - Average power density in test pin (reflector region).

ment.

The agreement of the calculated profile in the reflector region with the experimental points indicates that the one-dimensional technique described in this report may be used to determine relative flux levels. But complete analytical determination of absolute flux levels will require a more elaborate analysis.

SUMMARY OF RESULTS

The simulation of a fast reactor spectrum, the absorption activity induced in fuel cladding, and the power profile generated in a fuel element while in a thermal spectrum were studied by using multigroup neutron transport calculations. A conceptual design of a fast reactor provided a typical fast reactor spectrum and a model fuel element, and the NASA Plum Brook Reactor provided a typical thermal spectrum test environment. The results of this study are as follows:

1. The only method of simulating the fast reactor flux spectrum and the most effective method of simulating the power distribution is to filter out the thermal neutrons with strong absorbers. Boron-10 is the best filter, but cadmium may be useful because it absorbs only thermal neutrons. Hence, cadmium could be used as a thermal neutron shield for a boron-10 filter in order to retard the boron-10 burnup from thermal neutrons.

2. In a typical core test region, the flux is high enough that filters can generally be used to simulate the fast spectrum. The fast spectrum has about 0.1 percent of its flux below 5.5 keV, but the best filter configuration considered (0.25-cm $B_4^{10}C$) still allowed the simulated spectrum to have about 5 percent of its flux below this energy. Generally then, for irradiation mechanisms that are important between 5.5 and 0.2 keV (e.g., resonance absorption), the simulated fast spectrum will exaggerate the effect of the irradiation relative to what would occur in the fast spectrum. There are several ways of simulating the required power profiles and levels in a core test region, namely, use of filters, reduced enrichment, and reduced element size.

3. In a typical reflector test region, the fast flux levels are so low that neutron filters generally cannot be used, and hence, the fast spectrum itself cannot be simulated. However, the power profiles can be simulated by reducing the fuel enrichment and sample size. Although reducing the fuel enrichment flattens the power profile, it also lowers the power density. Therefore, in conjunction with reduced enrichment, the diameter of the sample fuel element may have to be reduced in order to provide an offsetting increase in the power density.

4. To summarize, the options available (and the conditions affected) for simulating the irradiation characteristics of a fast spectrum reactor while in a thermal spectrum reactor are (1) reduced enrichment (flattens power profile but reduces burnup rate), (2) neutron filters (flattens power and tailors flux profiles but reduces burnup rate), (3) test location (provides various flux profiles and burnup rates), and (4) reduced pin size (increases power density and burnup rate).

5. The type or degree of simulation to be used will depend on the thermal spectrum that is available, the irradiation effect to be simulated, and the burnup rate that is desired. Optimum spectral simulation will require knowledge of the important thresholds, cutoffs, and resonances in the cross sections of both the irradiated materials and the thermal neutron filters.

6. The approximations made in order to treat simply a complex geometry make it difficult to determine the precise shape and absolute level of the flux spectrum at a particular location in the reactor. However, comparative information about flux profiles and energy-averaged quantities, such as power, may be obtained.

Lewis Research Center,
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Cleveland, Ohio, March 4, 1970,
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