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FAST FLUX IRRADIATION FACILITY FOR THE ETR

E. Fast and G. J. Russell

IDAHO NUCLEAR CORPORATION
NATIONAL REACTOR TESTING STATION
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ABSTRACT

The purpose of this report is to document the results of a study initially intended to provide a basis for design and fabrication of an experimental facility. The facility was to be used in the ETR to provide fast flux irradiation of various materials of interest to the Reactor Experiments Section's experimental programs. However, before the study was complete, the direction of the experimental program had changed, and it is now expected that initial fast flux irradiations will be done in EBR-II. This study has shown the feasibility of providing fast flux irradiation for experiments in ETR, and may be used in the future as a basis for further related studies by Idaho Nuclear or other experimenters.

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FAST FLUX IRRADIATION FACILITY FOR THE ETR

I. INTRODUCTION

A study was made to design a relatively inexpensive fast flux irradiation facility for use in the ETR. The facility would filter and tailor the neutron flux spectrum to make it similar to that anticipated in fast power reactors. Previous studies of this type [1,2,3] have focussed on filtering the reactor spectrum through various amounts of cadmium or boron. This leaves a neutron spectrum in the sample region which has the fission energy peak (at about 2 MeV) and distribution, modified by various amounts of lower energy contamination. The peak of a fast power reactor spectrum is, however, typically in the 100 to 500 kV range. Thus the purpose here was to lower the fission peak energy as well as to deplete the spectrum of its lower energy neutrons. The results of the study are given in this report.

The general concept was to combine a heavy element scatterer with a filter of high cross section for low energy neutron absorption. This combination in annular form tailors the neutron spectrum of a thermal reactor in the center of a facility designed for irradiation of samples. Because of convenience initial screening studies assumed an ARMF type core. Subsequent results were obtained using an actual ETR position and an approximation to the ETR core composition surrounding this position. A multigroup diffusion code, MONA^[4], was used with flux averaged cross sections obtained with the PHROG^[5] and INCITE^[6] codes. The facility model was designed to occupy one lattice position of the ETR, a region about 3 inches square. The preliminary work was covered briefly in a recent annual report.^[7]

In outline, the following report will discuss the results of the studies and present the conclusion on the feasibility of such a facility. Emphasis will be placed on the final model arrived at in these studies. Steps leading to the final model will be given as necessary for completeness. This study is calculational only and does not include experimental data.

II. CALCULATIONAL PROCEDURE

The Code MONA: Reactor calculations were made with the multigroup one dimensional diffusion code MONA. Twenty-six neutron energy groups were used including one thermal group (below 0.523 eV). Details of the group structure are given in Table I. The groups were chosen to emphasize spectral detail in the higher energy regions which were of primary interest.

The ETR model was divided into 25 regions of which 12 were within the region of the facility under study. Further subdivision was made to give a total of 182 mesh points of which 73 were in the region of interest. Calculated fluxes were determined at each of these mesh points for each energy group. A detailed neutron balance for each region included the number of absorptions, fissions, average flux and various other data. A search was made of the fuel to poison ratio of the control rod regions to obtain criticality. This approximated the actual situation of shim or control rod adjustment in the ETR for each case. The power was normalized to 1 MW from which a direct scaling to 175 MW full power operation was assumed valid. Flux plots were obtained from the plot program PLTONE^[8] designed to be compatible with the MONA data output.

In the preliminary calculations similar results were obtained with a model of the ARMF core. These models had 24 to 28 regions 12 to 14 of which were in the region of the facility. Criticality was obtained by adjusting the boundary between the fuel and reflector regions. The adjustment simulated the actual adjustment in number of loose leaf fuel plates to make the reactor critical within the limits of the shim and control rod capability. The ETR and ARMF cores differed in fuel element length (36 and 24 inches respectively) and in core composition. The ETR fuel has a greater concentration of ²³⁵U and some boron as a burnable poison.

Model of the Facility: The calculational model assumed cylindrical symmetry for both the facility and the reactor core. The actual irradiation facility will be cylindrical in all regions except the outside which will be the geometry and dimensions of one ETR element. Cylindrical regions were calculated to approximate the actual region area of the horizontal cross section. Figure 1 shows a schematic of the calculational regions of the facility itself as used in the final model with the ETR core. Radial dimensions and annular region widths are tabulated (Table II) for both core and facility.

Temperature: Two computer codes were used in the calculation of temperature distribution in the irradiation facility components. HEAT-1^[9] is a one-dimensional code well adapted to the cylindrical symmetry used. A profile plot is obtained as well as the listing of temperatures at specified mesh points. SIMIR is a two-dimensional code, an adaptation of the PDQ reactor calculation code.^[10] It was used to supplement the HEAT-1 code for the shaper region since the outer geometry of the shaper is actually square rather than cylindrical. Two-dimensional isothermal plots are obtained from this program.

TABLE I

ENERGY AND LETHARGY GROUP STRUCTURE USED IN MONA CALCULATIONS

<u>Group No.</u>	<u>Lower Energy, eV</u>	<u>Upper Lethargy</u>	<u>Lethargy Increment</u>
1	6.07×10^6	0.5	0.5
2	3.68×10^6	1.0	0.5
3	2.23×10^6	1.5	0.5
4	1.35×10^6	2.0	0.5
5	8.21×10^5	2.5	0.5
6	4.98×10^5	3.0	0.5
7	3.02×10^5	3.5	0.5
8	1.83×10^5	4.0	0.5
9	1.11×10^5	4.5	0.5
10	6.74×10^4	5.0	0.5
11	4.09×10^4	5.5	0.5
12	2.48×10^4	6.0	0.5
13	1.50×10^4	6.5	0.5
14	9.12×10^3	7.0	0.5
15	5.53×10^3	7.5	0.5
16	2.61×10^3	8.25	0.75
17	1.23×10^3	9.0	0.75
18	583	9.75	0.75
19	275	10.5	0.75
20	130	11.25	0.75
21	47.9	12.25	1.0
22	17.6	13.25	1.0
23	6.48	14.25	1.0
24	2.38	15.25	1.0
25	0.532	16.75	1.5
26	0.0		

TABLE II

CYLINDRICAL CALCULATIONAL MODEL H FOR THE FAST
FLUX IRRADIATION FACILITY AND ASSOCIATED ETR CORE

Region No.	Material	Outer Radius Cm.	Region Width Cm.	No. of Mesh Spacings
1	Sample	0.15875	0.15875	4
2	Sample	0.31750	0.15875	4
3	SS, Clad	0.3556	0.0381	2
4	SS, B-10 tube	0.43497	0.07937	4
5	Water	0.59372	0.15875	2
6	SS, B 10 filter	0.75247	0.15875	8
7	SS, B-10 filter	0.91122	0.15875	8
8	Water	1.06997	0.15875	2
9	Shaper	2.10967	1.0397	12
10	Shaper	3.14937	1.0397	12
11	Shaper	4.18907	1.0397	12
12	Water	4.3478	0.15873	2
13	ETR fuel	6.1392	1.7914	10
14	ETR fuel	7.9306	1.7914	10
15	ETR fuel	9.7220	1.7914	10
16	Control Rods	11.382	1.66	6
17	Control Rods	13.042	1.66	6
18	ETR fuel	15.744	2.702	10
19	ETR fuel	18.446	2.702	10
20	Experiments	21.739	2.393	4
21	ETR fuel	24.595	2.856	10
22	Control Rods	27.50	2.905	6
23	Experiments	30.435	2.935	4
24	ETR fuel	37.0	1.565	8
25	Reflector (H ₂ O, Al, Be)	50.0	13.0	15

III. PARAMETER STUDIES

Variations in three parameters, ie, water coolant channels, filter materials and thickness, and shaper materials were studied as to the effect on the flux spectrum and intensity. Results may be weighed in the light of the dimensional and thermal restrictions imposed on the facility to arrive at an optimum design. Some of these results were obtained only in the preliminary phase with the ARMF core model. The shaper material calculations were repeated in the ETR core model along with a variation in the coolant channel study.

Water Coolant Channels: Studies were made of the effect of coolant channels because the high heat generation rate in the proposed irradiation position might require additional cooling. The model studied had coolant channels between the filter and shaper, and, of course, outside the shaper (next to the ETR fuel) regions 8 and 12 in Figure 1. The inner water channel (region 5) was not present in this model. The region 8 channel was varied from 0 to 6 times the normal 1/16 inch (0.15875 cm) width. The change in water channel width was made at the expense of the shaper thickness.

A comparison of the calculated flux spectra of three cases is given in Figure 2, namely zero, 1/16 (normal), and 3/8 inch channel width. Qualitatively there is a small buildup of higher and lower energy neutrons with channel width relative to and at the expense of the peak neutrons. The weighted average of the peak neutrons shifts toward higher energies. Table III gives a summary of changes in neutron absorptions in a ^{239}Pu samples in three broad energy groups. The middle group preferably should have as high a proportion of the total absorptions as possible.

TABLE III

EFFECT OF VARIOUS COOLANT CHANNEL THICKNESSES ON ^{239}Pu
ABSORPTIONS IN THREE ENERGY REGIONS, ARMF MODEL

Neutron Energy Range	% Absorptions for Coolant Channel Thickness (1/16 in. units)						
	0	1	2	3	4	5	6
>1.35 MeV	17.3	16.9	16.8	16.9	17.2	17.5	18.7
1.23 keV 1.35 MeV	74.3	72.5	70.4	68.5	66.8	65.3	62.6
<1.23 keV	8.4	10.6	12.7	14.6	16.0	17.2	18.8

Thermal Neutron Filter: The study of thermal neutron filters considered various amounts of cadmium and ^{10}B in regions 4 - 7 of Figure 1, ie, between the shaper and sample. The model used (ARMF model) did not have the coolant channel in region 5 (Figure 1). The normal thicknesses of the filter materials assumed previously and in the study of water channel width was: cadmium, 0.15875 cm and ^{10}B , 0.3175 cm. The latter was assumed 60 percent of theoretical density as a practical compaction of powdered boron. In studying the effect of various filter thicknesses the actual region dimension was maintained and the atom density alone changed. The "filter thickness" is given as the fraction of "normal" thickness. A "normal" thickness of one filter material was maintained while the other material was varied in the study. The shaper material in all cases was tungsten. The effect on the neutron spectrum in the sample region was the basis of judgment.

The cadmium atom density was varied from 0 to 1.5 times normal while maintaining a normal boron density. Very little difference was noted in the spectrum between the two extreme cases.

The reason cadmium was originally introduced into the neutron filter was to lower the rate of the boron burnout. The effectiveness of cadmium in this role is obtained from the MONA tabulation of absorptions in the filter regions. Comparing the "normal boron" case with and without a cadmium shield on the outside the numbers show a difference of less than 15 percent in boron absorptions. Of these, 90 percent are in the lower energy range below 130 eV. The initial burnup rate of boron was calculated to be less than 0.08 percent per day at full power operation of the ETR.

The effect of a change in boron (^{10}B) content of the filter on the sample flux is shown in Figure 3 for 0.0, 0.2, and 1.0 normal. These results were obtained likewise with the initial (ARMF) model and included normal cadmium. The relative change in spectrum only should be compared in the three cases. The case "0.2 normal" has an atom density of ^{10}B very nearly that of the ^{10}B stainless steel filter in the latest version of the ETR model facility (regions 4, 6, and 7 of Figure 1). Figure 4 shows the variation of relative absorptions in a ^{239}Pu sample for several ranges of neutron energies.

Heavy Element Flux Shapers: Preliminary considerations of nuclear properties and practical aspects limited the candidates for heavy element flux shaper material to three: tantalum, tungsten, and depleted uranium. (Lead and bismuth were also subjected to preliminary calculation but eliminated early). These three were originally studied with the initial model, with and without a sample of ^{239}Pu [7]. Final calculations were then made with the ETR model using updated values of cross sections from ENDF-B compilations for tungsten and uranium. Only the final calculations are presented here.

Comparison may be made of the relative effectiveness of these three materials in spectral shaping. In Figure 5 the flux spectra at the sample region are plotted for the three cases. Table IV compares the absorptions in a UO_2 -30 wt% PuO_2 sample for several broad energy groups and for the same cases of tungsten, tantalum, and depleted uranium as

shaper materials. The thermal neutron filter was ^{10}B stainless steel alloy (regions 4, 6, and 7 of Figure 1).

TABLE IV

EFFECT OF VARIOUS SHAPER MATERIALS ON
NEUTRON ABSORPTION IN $\text{UO}_2\text{-30 PuO}_2$

Neutron Energy Range	Percent Absorptions in $\text{UO}_2\text{-30 PuO}_2$ for Shaper Material			
	Tungsten	Tantalum	Depleted Uranium	Tungsten*
1.35 - 10 MeV	10.4	14.4	15.1	13.5
1.23 keV - 1.35 MeV	39.6	48.6	48.8	49.6
130 eV - 1.23 keV	19.1	20.2	17.9	15.2
Below 130 eV	30.9	16.8	18.2	21.7
Relative Total Absorptions in Sample	1.48	1.0	1.35	0.834

*Tungsten, in this case, includes region 8 which contains water in the other cases.

Heat Generation: In a high flux reactor like the ETR heat is generated by nuclear reactions of the material itself and by gamma-ray absorption. Measurements in the ETR core show that gamma heat is about 25 watts per gram for heavy elements and 15 watts per gram for lighter elements. These quantities were used in the following calculations. Attenuation of gamma rays in passing through the heavy element material of the shaper is appreciable and energy dependent, thus must be considered in heat calculations to obtain reasonable results.

The prompt gamma photon spectrum of ^{235}U fission was assumed applicable along with attenuation factors at various energies, given in the same reference.^[11] The low energy gamma photons are more strongly attenuated leaving only a fraction of the higher energy photons, which have relatively low absorption coefficients, at the sample position. The average energy in watts per cm^3 generated in the facility material by gamma rays is summarized in Table V. The gamma photons generated within the facility materials, including the sample, and absorbed within the facility produce heat energy well below the uncertainty of the calculation, and hence are neglected.

TABLE V

SUMMARY OF HEAT GENERATION DATA

Region See (Figure 1)	Material	Heat Generated in watts/cm ³		
		<u>Gamma</u>	<u>Nuclear 1</u>	<u>Total</u>
10	W	328		328
10	W	87.4		87.4
9	W	32.4		32.4
(8)*	(W)	(32.4)		(32.4)
7		0.5	106.8	107
6		0.5	30.5	31
4		0.2	28.0	28
1 & 2	Sample			
	Pu	9.3	2726	2736
	UO ₂ -PuO ₂	4.5	477	482
	Phoenix	1.14	92.8	93.9

*The values in parenthesis are for the case when tungsten replaces water in the coolant channel of Region 8.

**Phoenix is a Al-21 wt% Pu fuel in which the primary isotopes are 65% - ²³⁹Pu and 23% - ²⁴⁰Pu.

Nuclear reactions which contribute significantly to the heat generation are primarily fission reactions in fissionable materials and the (n, α) reaction in boron. The energy contribution in each case is likewise compiled in Table V. The values obtained were based on the total neutron absorptions given for the respective regions by the MONA reactor calculations.

Temperature distributions were calculated for the various regions in the irradiation facility. The pertinent results are summarized in Table VI. A two-dimensional isothermal map of the shaper and filter region is given in Figure 6, where the two are combined, eliminating the water channel of region 8, Figure 1.

TABLE VI

CALCULATED TEMPERATURES IN THE FAST FLUX IRRADIATION FACILITY

<u>Regions</u>	<u>Material</u>	<u>Maximum Temperatures °C</u>	
		<u>Center</u>	<u>At Water Channel Surface</u>
1 & 2	Sample		
	Normal Pu	897	162
	UO ₂ -30 PuO ₂	411	70
	Phoenix Fuel*	66	55
6 & 7	¹⁰ B-SS Filter	60	54
9 - 11	Tungsten Shaper	269	163
6 - 11	¹⁰ B-SS Filter and		
	Tungsten Shaper**	343	185

*Phoenix is a Al-21 wt% Pu fuel in which the primary isotopes are 65% - ²³⁹Pu and 23% - ²⁴⁰Pu.

**Case with no water channel between filter and shaper regions.

IV. DISCUSSION

To show the effectiveness of the heavy element scatterer in shaping the neutron flux spectrum, Figure 7 shows a plot of the facility with and without tungsten present in regions 9, 10 and 11 of Figure 1. The modification obtained is in the desired direction. More such material appears desirable but cannot be used due to the space limitation. The spectrum obtained in the CFRMF [12,13,14], which is based on a similar concept, shows the peak of the flux degraded to about 100 keV and broadened. The shaper material thickness is about twice as great in that case. An identical facility in the high flux of the ETR would encounter considerable heat removal problems.

The optimum choice of shaper material to use is not entirely obvious from the calculated spectra. Uncertainties enter with the cross sections of the keV and MeV regions. For example, a marked difference was observed in the spectra obtained with tungsten before and after the latest ENDF-B cross sections were used. In this report the updated cross sections for tantalum were not yet available in a form in which they could be used.

The physical properties of the material must also be considered. ^{238}U would need to be clad, adding to the difficulty of fabrication, and heat transfer might be less efficient as a result. Tungsten and tantalum could be used without being clad. Tungsten would probably pose a greater machining problem, while the cost of tantalum metal is approximately five times greater than tungsten.

It seems obvious that a minimum of water is desirable. Chronologically in the study, the water channel between the filter and shaper sections (region 8) was included initially. Additional cooling of the sample was found necessary, hence the coolant channel of region 5 was inserted. However, the most recent heat calculations show that the coolant channel of region 8 can then be eliminated, releasing additional space for shaper or filter material. The only case calculated was that with tungsten in region 8. The improvement in spectrum as seen in Table III (compare first and last data columns) is gratifying.

It appears that more efficient removal of lower energy neutrons is desirable. In initial studies an error in shaper material cross sections showed excessive absorption of these low energy neutrons. Thus the filter section was simplified by substituting a high-boron stainless steel for the Cd-SS clad ^{10}B powder, which presumably would not require cladding. This steel contained 3 wt% ^{10}B . The updated calculations given here would indicate that further removal of low energy neutrons is desirable.

In conclusion then, a facility of the type proposed appears entirely feasible. Additional calculations are needed, however, to determine the optimum choice of filter and shaper materials within the available space. The maximum possible amount of heavy element scattering material should be used, together with the minimum water required to cool the system to acceptable levels.

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Region No.	Material
1	} Sample
2	
3	S. S.
4	S.S. and B-10
5	Water
6	} S.S. and B-10 Filter
7	
8	Water (shaper)
9	} Shaper
10	
11	
12	Water

Figure 1. Calculational Model H for Fast Neutron Flux Irradiation Facility

Figure 2. Variation of Neutron Flux Spectrum in the Sample Region With Width of Water Coolant Channel

Figure 3. Sample Region Flux Spectra for Several Concentrations of ^{10}B in the Thermal Neutron Filter, Fast Flux Irradiation Facility

Figure 4. Variation of Relative Neutron Absorptions in ^{239}Pu of Three Energy Ranges as a Function of ^{10}B Filter Thickness

Figure 5. Comparison of Neutron Flux Spectra in the Sample Region for W, Ta, and ^{238}U Shaper Materials

Figure 6. Isothermal Map of the Fast Flux Irradiation Facility (1/4 Typical) for the ^{10}B -SS, W, Filter-Shaper Regions and No Water Channel Between

Figure 7. Flux Spectrum of Sample With and Without Tungsten Shaper and Superpose W from Figure 5