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EFFECTIVE PU-238 AND NP-237 NEUTRON ABSORPTION
CROSS SECTIONS IN N AND K REACTORS

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EFFECTIVE PU-238 AND NP-237 NEUTRON ABSORPTION
CROSS SECTIONS IN N AND K REACTORS

INTRODUCTION

The efficiency of producing Pu-238 from Np-237 depends on the relative absorption cross sections of the target, Np-237, and the product, Pu-238.¹ A lowering of the Pu-238/Np-237 cross section ratio reduces the destruction of Pu-238 formed during reactor operation.

The absorption cross sections of both these isotopes are strongly neutron energy dependent; the absorption cross section of Pu-238 decreases more rapidly with thermal neutron temperature increases than the conventional v^{-1} dependency and exhibits negligible resonance absorption. On the other hand, Np-237 is a very nearly v^{-1} thermal neutron absorber and has dominant low level resonance cross sections. Thus, for higher thermal neutron temperatures and a more epithermal neutron spectrum, depletion of the Pu-238 product is reduced relative to Np-237 burnout.

The relative absorption cross sections of these two isotopes were calculated for the K and N Reactors as a function of graphite temperature for two linear densities of neptunium in the target elements for each reactor type.

The method of calculation was as follows: The HAMMER integral transport code was used to obtain region and cell averaged four-group cross sections for both the fuel and target cells. These cross sections were then used in the HFN diffusion theory code to obtain reactor-effective cross sections.

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RESULTS

For representative operating graphite temperatures of the K and N Reactors, the following cross sections for Pu-238 and Np-237 were obtained:

Pu-238 and Np-237 Cross Sections

	Graphite T (°C)	Linear Density Np (g/ft unexposed)	Microscopic Absorption Cross Sections			
			Calculated		Pu-238/Np-237	
			Np-237	Pu-238	Calculated	Experimental*
N Reactor	700	45	67.7	124.5	1.84	1.8
	700	70	63.4	116.7	1.84	1.8
K Reactor	700	50	73.5	140.3	1.91	1.8
	700	120	60.6	109.0	1.80	

*These values of the cross section ratios most closely match the reported Np-237 burnout and Pu-238 formation data from the experimental program (See Reference 3 for the N irradiation and Reference 4 for the K irradiation results).

Complete results of parametric analysis are given in Figures 1 and 2. The curves for a K Reactor show greater dependence on graphite temperature than for N Reactor due to a higher graphite-to-fuel ratio in a K Reactor, and because the water moderation is more dominant in N Reactor. (The water temperatures were held at constant operating levels in the calculations.)

Although comparative calculations were not performed for the Savannah River D₂O lattice, a cross section ratio of about 2.5 has been reported² for the Savannah River lattice which is consistent with the values calculated for lower temperature moderators.

DISCUSSION

The problem was formulated to obtain effective flux-weighted absorption cross sections generated by the HAMMER code for the neptunium target cells and enriched driver cells over a range of graphite temperatures in N and K Reactors.

Linear densities of neptunium in the elements were: 50 g/ft. and 120 g/ft for the K Reactors, and 45 g/ft and 70 g/ft for N Reactor.

Region cross sections were found for the neptunium elements at various graphite temperatures. The calculations were performed for exposed elements having 83 percent Np-237 left with 13 percent forming Pu-238, and one percent each forming Pu-239, Pu-240, Pu-241, and Pu-242. Homogenized lattice cross sections were found for the drivers at graphite temperatures near their operating range. The temperatures of the driver cells were not changed because variation in the moderator temperature in these cells has a small effect on the flux spectrum in the target compared to temperature variations in the graphite immediately adjacent to the targets.

In-reactor flux spectrums were calculated with the HFN code by surrounding regions of the neptunium elements by a cylindrical driver region. The outer boundary of the driver was made reflective and its radius was adjusted to give the system a xenon-free k_{∞} of 1.015 and 1.030 for the K and N Reactors, respectively.

Microscopic cross sections from HAMMER were flux weighted by HFN fluxes giving effective in-reactor absorption cross sections for Pu-238 and Np-237. These cross sections, and their ratios, are plotted vs. graphite temperature in Figures 1 and 2. Numeric description of the lattices is given in Tables A and B.

CONCLUSION

Figures 1 and 2 clearly show dependency of the Np-237 and Pu-238 neutron absorption cross sections on graphite temperature. They also indicate the effect of a harder spectrum; due to a greater thermal flux depression, higher density targets have a more epithermal spectrum which favors the Np-237 resonance absorptions.

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These results indicate the desirability of high moderator temperatures and a high epithermal to thermal flux ratio in optimizing Pu-238 production. However, other technical and economic factors outside the scope of this study also have a strong bearing on optimum irradiation conditions and target design.

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TABLE A
DRIVER LATTICES

N REACTOR

<u>Region</u>	<u>Material</u>	<u>OD (inches)</u>	<u>Density (gm/cm³)</u>	<u>Temp (°C)</u>
1	Water	0.438	0.7985	247
2	Zircaloy-2	0.493	7.718	269
3	Uranium*	1.164	18.067	406
4	Zircaloy-2	1.25	6.725	267
5	Water	1.763	0.7985	247
6	Zircaloy-2	1.815	7.128	277
7	Uranium**	2.351	17.752	408
8	Zircaloy-2	2.406	6.918	270
9	Water	2.708	0.7985	247
10	Zircaloy-2	3.258	6.40	277
11	Graphite	9.552	1.300	430
12	Water	9.575	0.98	65

K REACTOR

1	Water	0.432	0.9857	55
2	Aluminum	0.545	2.699	105
3	Uranium*	1.419	17.339	252
4	Aluminum	1.509	2.699	88
5	Water	1.724	0.9857	55
6	Zircaloy-2	1.807	6.400	62
7	Graphite	8.4628	1.5955	550

* 0.947 wt% U-235

** 1.25 wt% U-235

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TABLE B

NEPTUNIUM TARGET LATTICES

N REACTOR (45 and 70 gm/ft)

<u>Region</u>	<u>Material</u>	<u>OD (inches)</u>	<u>Density (gm/cm³)</u>	<u>Temp (°C)</u>
1	Graphite	1.685	1.6285	200
2	Aluminum	1.755	3.2383	210
3	Np-Pu-Al*	2.192	2.95	210
4	Aluminum	2.327	2.699	210
5	Zircaloy-2	2.410	8.4986	200
6	Water	2.708	0.8714	200
7	Zircaloy-2	3.258	6.400	277
8	Graphite	9.552	1.300	(20,430,700)
9	Water	9.575	0.8714	200

K REACTOR (50 and 120 gm/ft)

1	Aluminum	1.020	2.163	50
2	Np-Pu-Al**	1.503	2.322	50
3	Aluminum	1.593	2.699	50
4	Water	1.724	0.9857	56
5	Zircaloy-2	1.807	6.400	62
6	Graphite	8.4628	1.5955	(20,560,700)

*	<u>Target (gm/ft)</u>	<u>wt% Np-Pu</u>	<u>wt% Al</u>
	45	5.520	94.480

	70	8.586	91.414
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**	<u>Target (gm/ft)</u>	<u>wt% Np-Pu</u>	<u>wt% Al</u>
	50	16.584	83.416
	120	39.802	60.198

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1. RL-3-4 "Richland Five Year 02 R&D Program, Pu-238," Richland Operations Office, et al, June 1, 1967 (Secret)
2. AECOP-00060, "Report of the Neptunium Irradiation Meeting, Germantown, Maryland, November 14-15, 1967," F. E. Driggers, December 27, 1967 (Secret)
3. "Final Report PT-68, Results of N Reactor Neptunium Irradiation," D. W. Constable, July 31, 1967
4. DUN-4354, "K Reactor Pu-238 Demonstration Results," J. C. Nosler, June 13, 1968 (Secret)

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Microscopic Absorption Cross Sections of Np-237 and Pu-238 vs. Neutron Temperature

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Figure 1

Absorption Ratio of Pu-238 and Np-237 vs. Graphite Temperature

Figure 2