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NEUTRON DOSIMETRY FOR THE MOTA-1F EXPERIMENT IN FFTF

L. R. Greenwood
L. S. Kellogg

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Pacific Northwest Laboratory
Richland, Washington 99352

NEUTRON DOSIMETRY FOR THE MOTA-1F EXPERIMENT IN FFTF - L. R. Greenwood and L. S. Kellogg, Pacific Northwest Laboratory[a]

OBJECTIVE

To provide dosimetry and damage analysis for fusion materials irradiation experiments.

SUMMARY

Neutron fluence and spectral measurements are reported for the MOTA-1F experiment in the Fast Flux Test Facility (FFTF). The irradiation was conducted from November 18, 1987, to January 8, 1989, for a total exposure of 335.4 EFPD. The maximum fluence was 12.7 x 10^{22} n/cm^2, 9.56 x 10^{22} above 0.1 MeV producing 39.1 dpa in iron. Neutron energy spectra were adjusted at three positions and gradients were measured at nine other locations.

PROGRESS AND STATUS

Twelve dosimetry capsules were positioned at different axial locations in the MOTA-1F assembly. Each stainless steel capsule measured about 2.1 cm long by 0.48 cm o.d. and contained dosimetry wires for either spectral or gradient measurements. The three spectral capsules contained Fe, Ti, Ni, Cu, 0.1% Co-Al alloy, 0.13% Sc in MgO, 0.825% ^{235}U in V, 0.936% ^{239}Pu in MgO, and ^{237}Np oxide monitors; the gradient capsules contained Fe and 0.1% Co-Al alloy wires. The Co, Sc, U, Pu, and Np materials were separately encapsulated in V.

Following irradiation, each dosimetry capsule was opened in a hot cell and each individual monitor was identified and mounted for gamma analysis. The measured activities were then converted to saturated activities by correcting for the sample weight, atomic weight, isotopic abundance, gamma absorption, reactor power history, and fission yield, as needed. Neutron self-shielding effects were not significant since the Co, Sc, U, and Pu were dilute alloys and the FFTF neutron spectrum has few neutrons at lower neutron energies.

Neutron burnup effects were found to be quite significant for the fission monitors and a small correction was necessary for the ^{59}Co(n,g) reaction. In the case of ^{59}Co, this correction can be applied using an iterative procedure since the reaction itself is the sole source of the burnup. Hence, the burnup must be at least as great as predicted by the measured value, namely exp(-st), where s is the reaction rate and t the total exposure time. Having corrected the reaction rate, the procedure can thus be repeated until it converges. This iterative approach works quite well except for very lengthy exposures where we go beyond the equilibrium activity. The procedure also includes the burnup of the product isotope (e.g., ^{60}Co). If we can determine the approximate ratio of the burnup reaction rate of the product isotope to that of the target isotope, then we can simply maintain this ratio during the iterative procedure. This approach is generally sufficient since the net corrections are not so sensitive to the reaction rates for the product isotope. In the case of the ^{59}Co(n,g) reaction, the maximum burnup effect was 2.8%.

Neutron burnup effects were found to be much larger and more difficult for the fission reactions. In all three cases, the fission reaction is not the sole source of the burnup since the (n,gamma) cross section must also be included. Unfortunately, we have no measure of the (n,gamma) reaction rate. However, we can readily calculate the rates using calculated neutron spectra. At each spectral location, neutron spectra were provided by R. Simons (Westinghouse Hanford) based on the cycle 9A irradiation.[1] These spectra were used with the STAY'SL computer code[2] to determine reaction rates for both the fission and gamma reaction. Since the absolute measured fission rates were not known, we assumed that the ratios of gamma to fission were the same as the calculations and determined the total burnup cross sections from the measured fission rates. Corrections were done separately for each fission product in order to account for the separate decay and burnup rates for each product isotope. The results were then averaged to obtain a first order correction to the burnup effects. An iterative procedure was used to determine a second order correction, which was found to be sufficient. The burnup corrections were thus determined to be between 10 and 25%. It should be noted that although metal fuel was used in the 1F irradiation, the 9A calculations were based on the previous oxide fuel loadings. The effects of these changes are unknown.

In the case of ^{237}Np, there is also concern that we can breed ^{238}Pu, which will then also contribute to the fission yields. Using the calculated reaction rates derived from the calculated neutron spectra, we estimate that this effect may be as large as 4%. However, if the ^{238}Pu fission yields are similar to those from

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^{237}Np, then there may be only a small difference in the true fission rates for ^{237}Np. It is also possible that (gamma,n) effects are present; however, past experiments have shown that this effect is not significant at in-core positions.

The corrected reaction rates are listed in Tables 1 and 2. The values have an estimated absolute uncertainty of about 5%, although there is an additional uncertainty of about 5% for the fission reactions due to problems in determining the burnup rates. These rates appear to be in reasonable agreement with previous measurements[3,4] except for the fission reactions. Neither of the previous experiments made any attempt to determine neutron burnup corrections for the fission reactions. Hence, our values are about 20-30% higher than previous measurements, allowing for differences in reactor power and core changes. As noted previously, the present results are for metal fuel, whereas both previous experiments were for oxide fuels.

The measured activities were used as input to the STAY'SL computer code to adjust the neutron spectra calculated for cycle 9A. STAY'SL performs a generalized least-squares adjustment of all measured and calculated values including the measured activities, calculated spectra, and neutron cross sections. Neutron cross sections and their uncertainties were generally taken from ENDF/B-V, although new data were available from ENDF/B-VI for the $^{46}Ti(n,p)$ and $^{63}Cu(n,a)$ reactions. The adjusted neutron fluence values are listed in Table 3. The neutron spectral adjustments for the three positions are shown in Figures 1-3 where the solid line is the calculated spectrum and the dotted line the adjusted spectrum. In all three cases, there were significant spectral adjustments. In general, the adjusted spectra reduce the flux at lower neutron energies but increase the flux from about 0.1 up to several MeV. The fast neutron flux is thus higher than previously reported, mainly due to the burnup effects for the fission detectors.

Damage calculations were also performed for the three spectral positions using the SPECTER computer code.[5] Dpa and helium rates for Fe are shown in Table 3. Damage parameters for other elements or compounds are also available.

Table 1
Activity Measurements at Spectral Positions (values in atom/atom-s at 291 MW)

Material Reaction	M66 -59.4 cm	M67 -2.6 cm	M68 +66.3 cm
$^{54}Fe(n,p)^{54}Mn$	3.21E-12	2.72E-11	1.06E-12
$^{58}Fe(n,g)^{59}Fe$	3.17E-11	3.50E-11	1.72E-11
$^{46}Ti(n,p)^{46}Sc$	3.55E-13	2.92E-12	1.14E-13
$^{45}Sc(n,g)^{46}Sc$	9.05E-11	1.15E-10	4.61E-11
$^{58}Ni(n,p)^{58}Co$	4.34E-12	4.05E-11	1.48E-12
$^{59}Co(n,g)^{60}Co$	1.41E-09	2.15E-10	8.13E-10
$^{63}Cu(n,a)^{60}Co$	1.76E-14	1.75E-13	5.54E-15
$^{58}Ni(n,x)^{57}Co$	8.89E-15	8.17E-14	3.13E-15
$^{60}Ni(n,p)^{60}Co$	7.81E-14	6.96E-13	2.67E-14
$^{235}U(n,f)$	7.58E-09	8.18E-09	3.19E-09
$^{239}Pu(n,f)$	5.99E-09	8.16E-09	3.32E-09
$^{237}Np(n,f)$	5.65E-10	2.02E-09	1.75E-10

Table 2
Activity Gradients for FFTF MOTA-1F (values in atom/atom-s at 291 MW)

Capsule	Ht., cm	$^{54}Fe(n,p)^{54}Mn$	$^{58}Fe(n,g)^{59}Fe$	$^{59}Co(n,g)^{60}Co$
69	-67.6	1.317E-12	3.264E-11	1.785E-09
66	-59.4	3.213E-12	3.168E-11	1.442E-09
70	-47.1	1.068E-11	3.295E-11	9.688E-10
71	-39.0	1.854E-11	2.758E-11	4.196E-10
72	-23.2	2.783E-11	3.172E-11	2.452E-10
67	-2.7	2.719E-11	3.497E-11	2.153E-10
73	15.7	2.205E-11	3.147E-11	2.201E-10
74	24.1	1.850E-11	2.833E-11	1.877E-10
68	66.3	1.062E-12	1.719E-11	8.237E-10
75	74.8	5.103E-13	1.314E-11	6.624E-10
76	108.0	4.015E-14	5.000E-12	2.751E-10
77	122.3	1.626E-14	3.607E-12	1.927E-10

Table 3
Neutron Fluences and Damage Parameters (MOTA-1F, 335.4 EFPD, 291 MW)
(Uncertainties in % included in parentheses)

Position	Ht., cm	Neutron Fluence $x10^{22}$ n/cm^2			Damage in Fe	
		Total	>0.1 MeV	>1 MeV	dpa	He, appm
66	-59.4	5.62 (29)	3.24 (34)	0.295 (22)	11.3	0.32
67	-2.6	12.7 (13)	9.56 (18)	1.74 (17)	39.1	3.07
68	+66.3	2.16 (28)	1.27 (35)	0.096 (21)	4.22	0.102

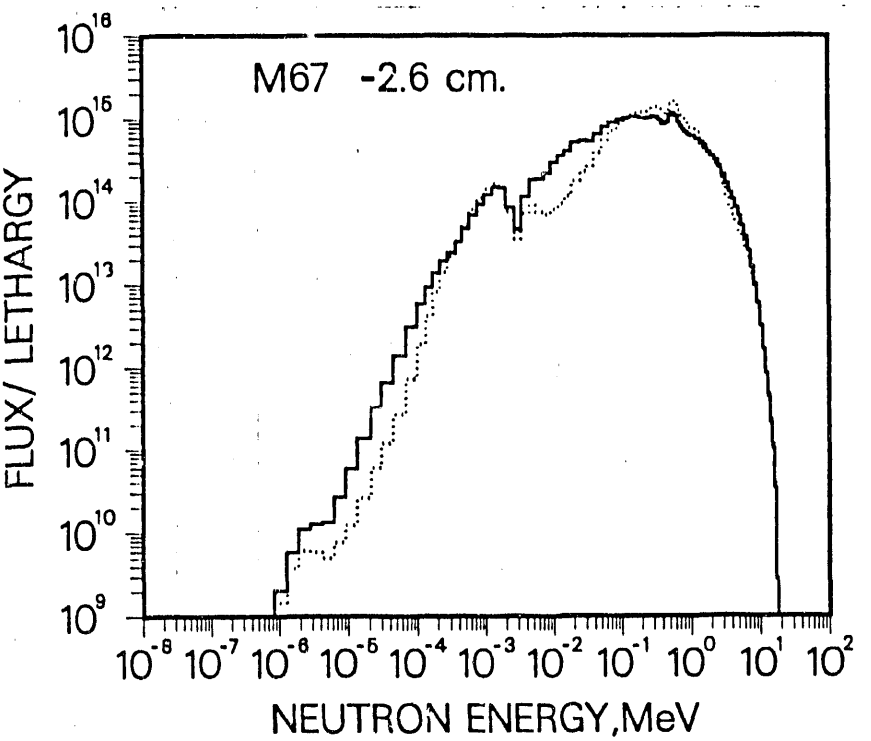

Figure 1. Calculated (solid line) and STAY'SL-adjusted (dotted line) neutron energy spectrum for capsule 67 at -2.6 cm in the FFTF MOTA-1F assembly.

The flux and damage gradients at other positions can be found from the activity gradient data in Table 2, which is shown in Figure 4. Note that the ^{58}Fe and ^{59}Co(n,g) reactions peak outside of the core since they are sensitive to the lower energy neutrons. The unusual behavior of the Co reaction is due to resonance effects.

The fluence above 0.1 MeV was determined from the activity for the ^{54}Fe(n,p) reaction. The spectral-averaged cross section was determined at each of the three spectral positions. Since the spectral changes are quite large, we assumed a linear dependence of the spectral-averaged cross section on the height. The fluence above 0.1 MeV was then determined by dividing the ^{54}Fe(n,p) activity by the calculated spectral-averaged cross sections. The resultant fast fluences are shown in Figure 5. Similarly, the ^{58}Fe(n,g) and ^{59}Co(n,g) reactions were used to determine the fluence below 0.1 MeV and the total fluence gradients are also shown in Figure 5. Dpa in Fe was then determined assuming a fixed ratio of dpa to fluence above 0.1 MeV; the dpa gradients are also shown in Figure 5. These procedures are only approximate since we only had three spectral measurements. Additional spectral monitors have been included with future runs in cycles 11 and 12 so that we can determine the fluence and damage gradients more precisely.

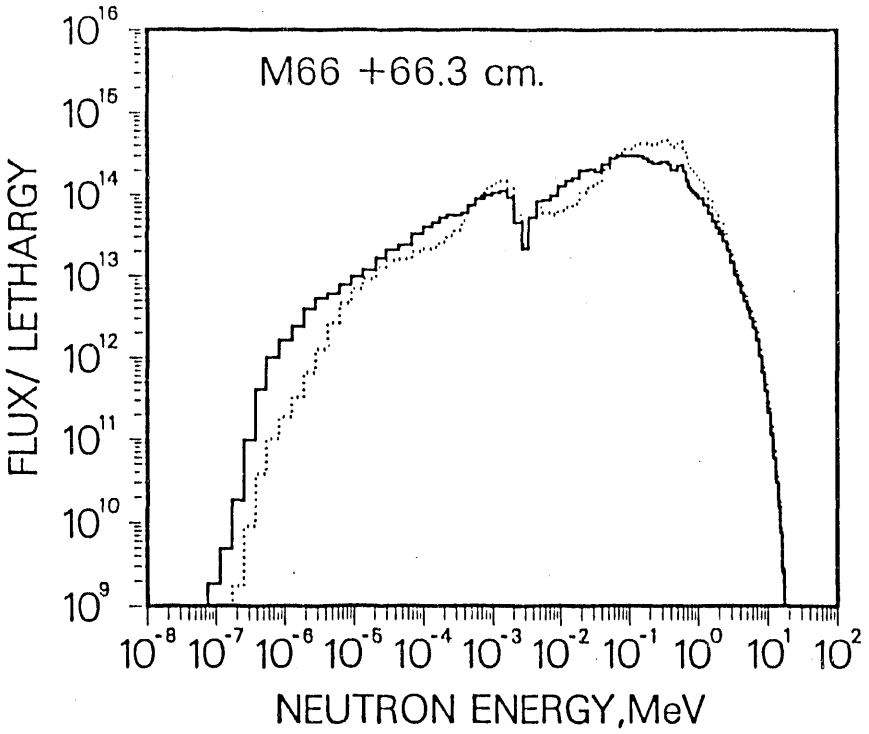

Figure 2. Calculated (solid line) and STAY'SL-adjusted (dotted line) neutron energy spectrum for capsule 66 at -59.4 cm in the FFTF MOTA-1F assembly.

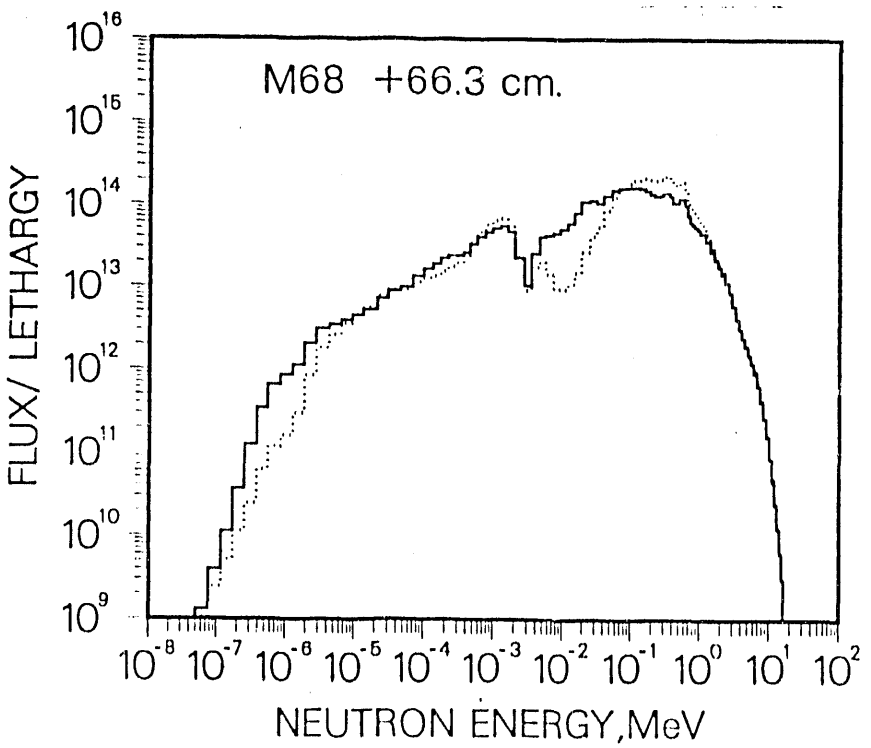

Figure 3. Calculated (solid line) and STAY'SL-adjusted (dotted line) neutron energy spectrum for capsule 68 at +66.3 cm in the FFTF MOTA-1F assembly.

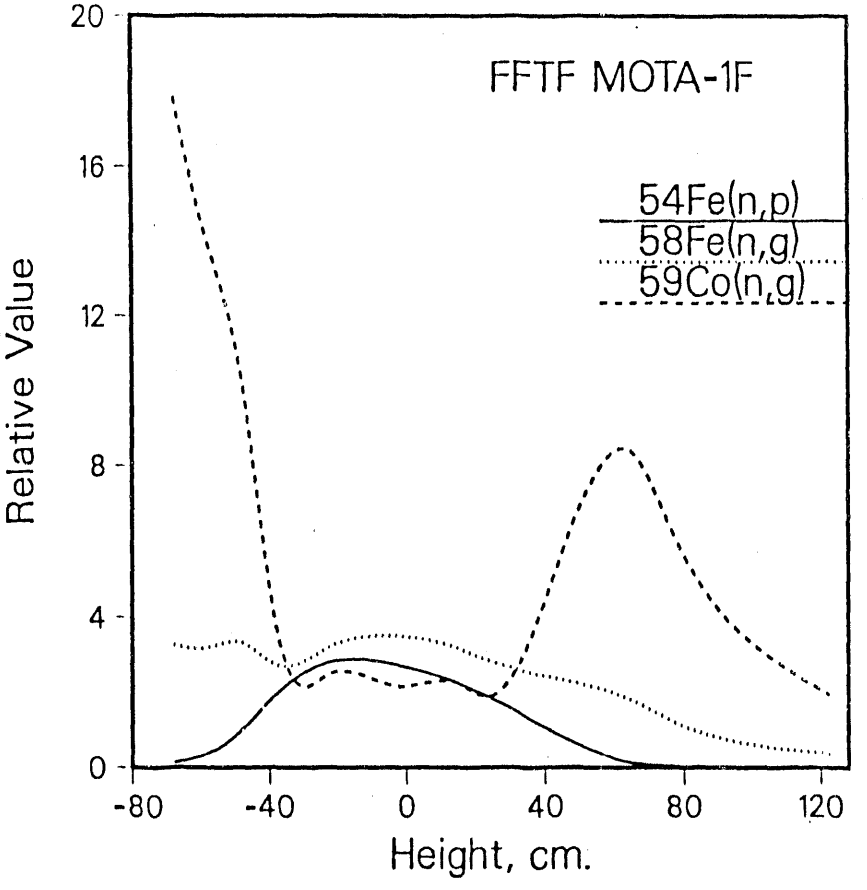

Figure 4. Activity gradients are shown as a function of height for the $^{54}Fe(n,p)$, $^{58}Fe(n,g)$, and $^{59}Co(n,g)$ reactions.

FUTURE WORK

Dosimetry is now being fabricated for the MOTA-2B in FFTF for irradiation with cycle 12 starting in about May 1991. More comprehensive dosimetry will be available from cycle 11, which contains 12 spectral and 21 flux gradient sets. These samples should be received in about June 1991. We are also providing dosimetry for the MFE-RB-200J1, -400J1, and JP-17, 18, and 19 experiments in the High Flux Isotopes Reactor at Oak Ridge National Laboratory.

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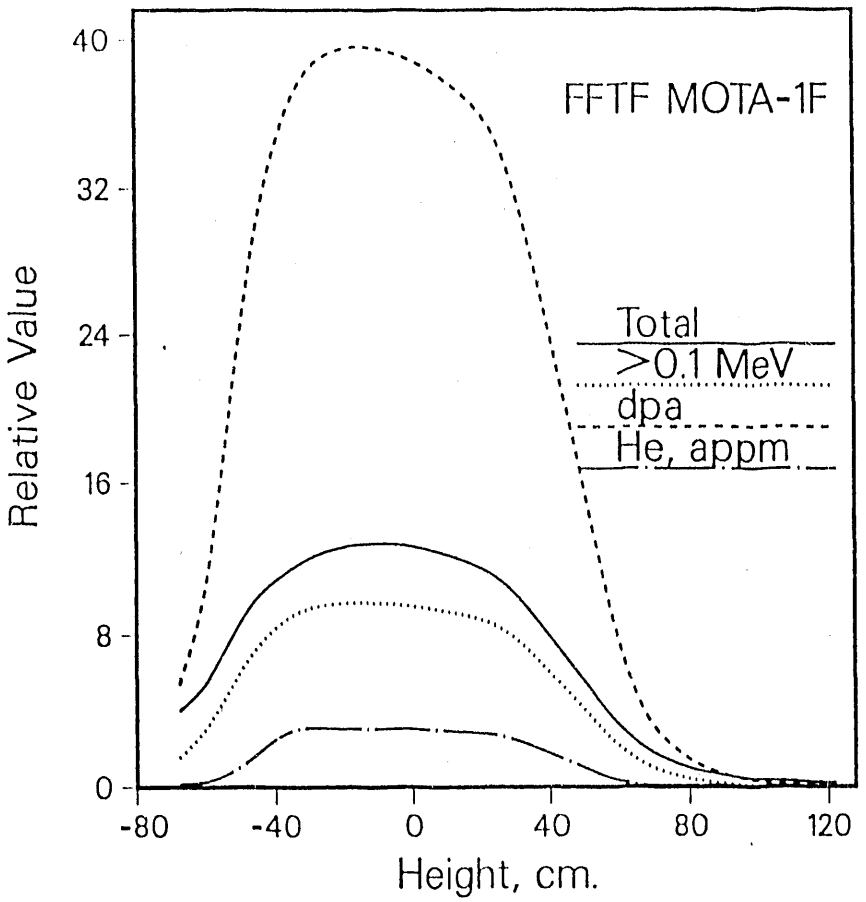

Figure 5. Fast (> 0.1 MeV) and total neutron fluences and Fe (dpa) and He (appm) gradients are shown as a function of height in the MOTA-1F assembly.

PUBLICATIONS

The following papers were presented at the Seventh ASTM-EURATOM Symposium on Reactor Dosimetry in Strasbourg, France, August 27-31, 1990, and will be published in the proceedings.

L. R. Greenwood and D. L. Bowers, Measurements of Activation Cross Sections for Fusion Reactor Applications.

D. L. Smith, J. W. Meadows, and L. R. Greenwood, Microscopic Integral Cross Section Measurements in the Be(d,n) Neutron Spectrum for Applications in Neutron Dosimetry, Radiation Damage, and the Production of Long-Lived Radionuclides.

E. M. Zsolnay, H. J. Nolthenius, L. R. Greenwood, and E. J. Szondi, Reference Data File for Neutron Spectrum Adjustment and Related Radiation Damage Calculations.

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