NEUTRON DOSIMETRY AND DAMAGE CALCULATIONS FOR THE HFIR-RB-11J,12J IRRADIATIONS - L. R. Greenwood (Pacific Northwest National Laboratory)* and C. A. Baldwin (Oak Ridge National Laboratory)

OBJECTIVE

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

SUMMARY

Neutron fluence measurements and radiation damage calculations are reported for the joint U.S. Japanese experiments RB-11J and RB-12J which were conducted in the removable beryllium (RB) position of the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). These experiments employed a Eu_2O_3 thermal shield for the first time to reduce the thermal neutron fluence. The maximum total neutron fluence at midplane was 1.85×10^{22} n/cm^2 (9.5×10^{21} n/cm^2 above 0.1 MeV), resulting in about 5.5 dpa and 3.6 appm helium in type 316 stainless steel.

PROGRESS AND STATUS

The RB-11J and –12J experiments were irradiated in the RB* position of HFIR during cycles 352 through 361 starting February 7, 1997, and ending July 17, 1998, for a net exposure of 223.92 effective full-power days at 85 MW. The experiment was a collaborative effort of the U.S. Fusion Materials Program at ORNL and Monbusho in Japan. Complete descriptions of the specimen matrices and irradiation assemblies as well as the reactor operating history have been published previously. [1,2]

Neutron dosimetry capsules were inserted at 6 different elevations located at various radial positions in each assembly. The assemblies were rotated after every cycle to minimize any radial flux gradients. The dosimetry capsules consisted of small aluminum tubes measuring about 1.3 mm in diameter and 6.4 mm in length. Each tube contained small monitor wires of Fe, Cu, Ni, Ti, Nb, 0.1% Co-Al alloy, and 80.2% Mn-Cu alloy. Following irradiation, the monitors were removed from the assemblies and analyzed for gamma activities at ORNL.

Upon examination at ORNL, it was found that four of the dosimetry capsules from the 12J experiment were ruptured, as shown in Figure 1. For these capsules, it appears that the copper wires melted through the aluminum capsules. A likely explanation for this behavior is that the copper was not in good thermal contact with the surrounding aluminum capsule, and, therefore, heat was transferred from the copper inefficiently. The resulting elevated temperature of the copper could have been sufficient to melt the aluminum at the point of contact. The copper was then lost in disassembly of the experiment capsule. The formation of a eutectic of aluminum and copper is also likely once melting began. This could reduce the melting point by about 110 ^{o}C. However, since the capsule temperature was at 500 ^{o}C, the copper only had to be about 50 ^{o}C higher than its surroundings to melt through the aluminum capsule, which has a melting point of 660 ^{o}C. Fortunately, only the copper monitors were lost in two of the ruptured capsules, and all of the other monitors were recovered. The monitor weights were checked against the fabrication records to ensure no loss of material on the other monitors. In the case of capsule 12J-126, the Co-Al alloy monitor could not be located after disassembly.

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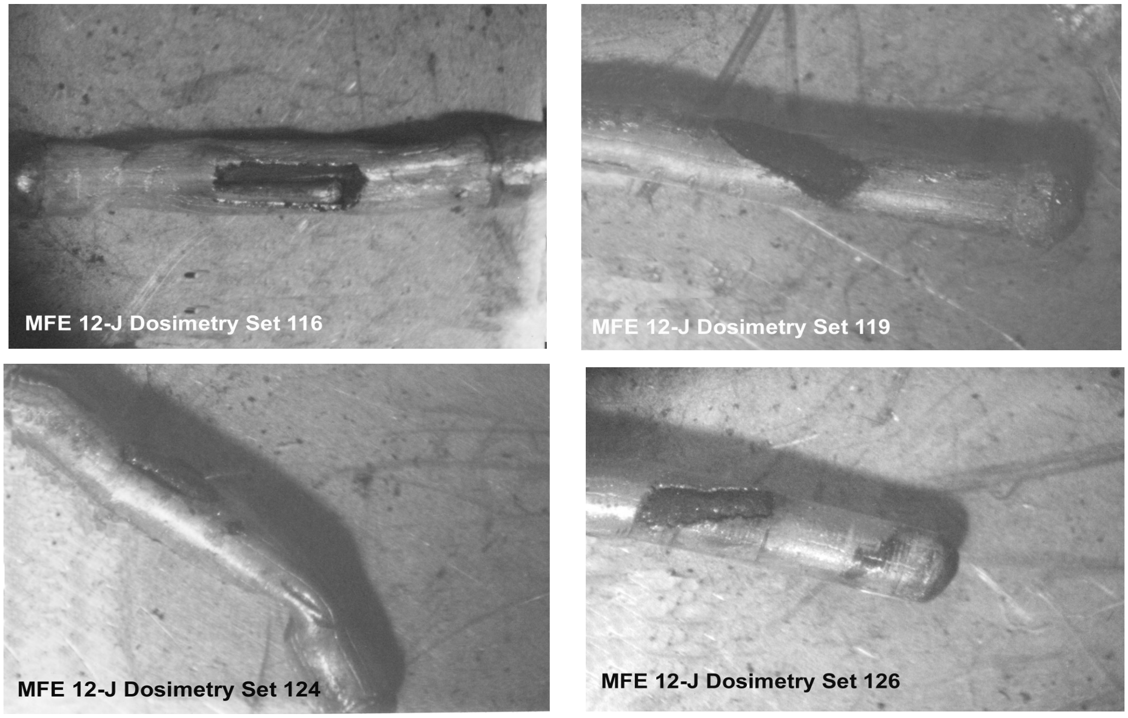

Figure 1. Pictures of ruptured dosimetry monitors from the RB-12J experiment.

The measured gamma activities were subsequently analyzed at Pacific Northwest National Laboratory. The activities were converted to activation rates, as listed in Table 1, by correcting for nuclear burnup, gamma self-absorption, decay during and after irradiation, isotopic abundance, and atomic weight. Burnup corrections were based on an iterative procedure for the thermal/epithermal monitor reactions. The resulting estimates of the thermal/epithermal neutron fluences were then used to calculate burnup corrections for the threshold fast neutron monitor reactions. Burnup corrections were quite small, averaging 1-5% for the thermal/epithermal reactions and < 1.5% for the threshold reaction rates. The activation rates listed in Table 1 are normalized to full reactor power of 85 MW and have a net absolute uncertainty of about 3%.

It should be noted that there was a discrepancy in the records concerning the placement of some of the dosimetry capsules in the 11J subassembly. The positions assigned in Table 1 lead to a consistent dependence of the activation rates with elevation about midplane for both experiments. Consequently, this placement was adopted and the data for both experiments were analyzed together. The activation rates were fit to a polynomial function of form $f(x) = f(0) [1 + ax + bx^2]$, where x is the vertical height from reactor centerline in cm. All of the data are reasonably well fit by the average polynomial with coefficients a = -2.744E-3 and b= -1.674E-3. The best fit to the data, as given by these coefficients, predicts that the maximum flux position is located about –0.8 cm below midplane. The flux gradient is so shallow that this only produces a flux difference of 0.1% from the midplane position. However, the asymmetry increases to a flux difference of about 12% between the top and bottom of each assembly.

Midplane activation rates were used in the STAY'SL [3] computer code to adjust the neutron flux spectrum. STAY'SL performs a generalized least-squares adjustment of all measured and

Table 1. Activation Rates (atom/atom-s) - HFIR-RB-11J,12J

Position /Monitor	Ht,cm	$^{54}Fe(n,p)^{54}Mn$ (E-11)	$^{46}Ti(n,p)^{46}Sc$ (E-12)	$^{55}Mn(n,2n)^{54}Mn$ (E-14)	$^{63}Cu(n,\alpha)^{60}Co$ (E-14)	$^{59}Co(n,\gamma)^{60}Co$ (E-9)	$^{93}Nb(n,\gamma)^{94}Nb$ (E-10)
11J-128	-13.4	0.998	1.36	3.07	6.42	1.45	2.44
11J-76	-6.6	1.27	1.72	3.82	8.15	1.82	3.11
11J-92	0	1.30	1.66	3.90	8.51	1.95	3.33
11J-94	0	1.33	1.81	4.02	8.57	1.95	3.31
11J-115	6.6	1.23	1.58	3.68	7.89	1.94	3.10
11J-96	13.4	0.844	1.07	2.59	5.59	1.30	2.11
12J-29	-13.4	0.953	1.36	2.87	6.40	1.45	2.44
12J-126	-6.6	1.27	1.71	3.82	7.76*	**	3.24
12J-119	0	1.34	1.73	3.88	*	1.92	3.44
12J-124	0	1.36	1.83	4.01	*	1.92	3.44
12J-116	6.6	1.26	1.62	3.66	8.00*	1.91	3.23
12J-28	13.4	0.878	1.20	2.69	5.76	1.40	2.20

*Cu samples could not be recovered for capsules 12J-119 and –124. The Cu monitors were recovered for capsules 12J-116 and –126; however, some sample loss may have occurred.
**Co-Al sample could not be relocated.

calculated values, including the measured activities, calculated spectra, and neutron cross sections. Neutron cross-sections and their uncertainties were generally taken from the ENDF/B-V evaluation [4]. The resulting neutron fluence values are listed in Table 2.

The calculated neutron flux spectrum was obtained from R. A. Lillie (ORNL, private communication). The Eu_2O_3 thermal neutron shield initially provides high suppression of the thermal neutron flux and extends into the epithermal energy range. However, due to the high thermal neutron flux in HFIR, the Eu suffers significant burnup as the irradiation progresses, thereby making the thermal shield less effective as the irradiation progresses. The thermal neutron flux, as calculated by R. A. Lillie, is shown as a function of irradiation time in Figure 2. Due to this unusual increase in the thermal neutron flux as well as the shielding effects on the epithermal neutrons, it is difficult to determine a unique solution to the thermal and epithermal fluxes given the dosimetry measurements. Consequently, the thermal neutron fluence in Table 2 has been assigned a high uncertainty. However, the net contribution of the thermal neutrons represents a fairly small perturbation on the mainly epithermal reaction rates for activation and helium production from Ni, as discussed below. The neutron spectral adjustment performed with STAY'SL, as shown in Figure 3, represents an average neutron spectrum over these irradiations.

Neutron damage calculations were performed using the SPECTER computer code [5] at the midplane position of HFIR. Midplane dpa and helium (appm) values are also listed in Table 2.

Table 2. Midplane Fluence and Damage Values for HFIR-RB-11J,12J

Neutron Fluence, $\times 10^{21}$ n/cm^2	
Total	18.5 $\pm$ 7%
Thermal (<.5 eV)	0.26 $\pm$ 50%
0.5 eV - 0.1 MeV	8.76 $\pm$ 13%
> 0.1 MeV	9.48 $\pm$ 11%
> 1 MeV	3.31 $\pm$ 13%

Element	dpa	He,appm
C	6.9	7.0
Al	10.9	2.4
V	7.0	0.10
Cr	5.9	0.63
Fe	5.3	1.0
Ni Fast	5.90	15.7
^{59}Ni	0.01	5.8
Total	5.91	21.5
Cu	7.1	0.86

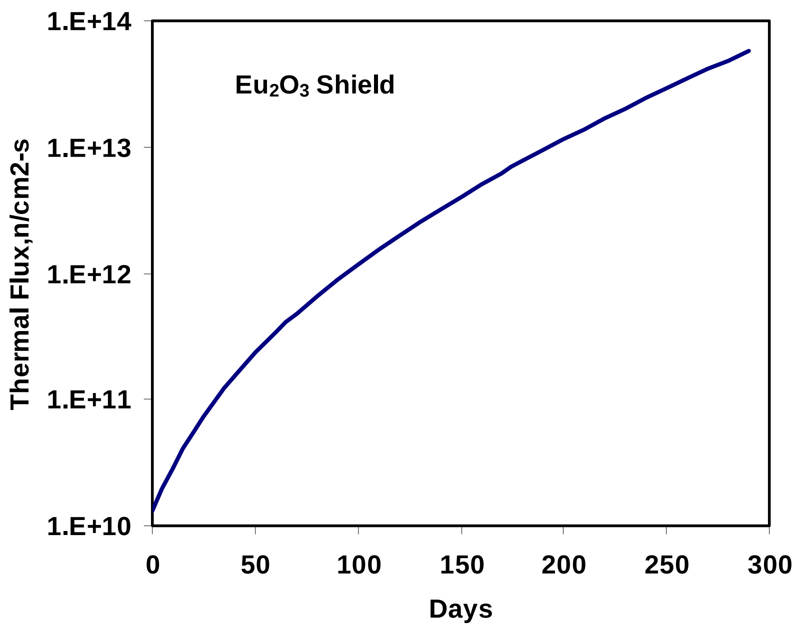

Figure 2. Time-dependent thermal neutron flux in the HFIR-RB showing the burnout of Eu_2O_3 shielding, as calculated by R. A. Lillie.

The fluence and damage values at other experimental positions can be calculated by the gradient equation given above. Damage parameters for other elements or compounds have been calculated and are readily available on request.

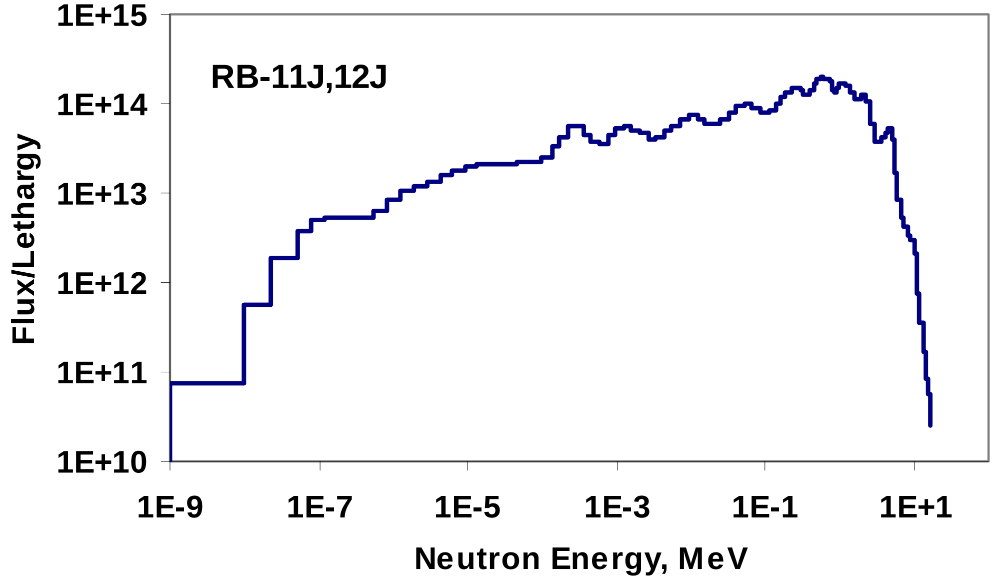

Figure 3. Neutron energy spectrum adjustment for the HFIR-RB-11J/12J experiments using the STAY'SL computer code.

Table 3. DPA and Helium Values for 316 SS in HFIR-RB-11J,12J (includes ^{59}Ni effect)

Height, cm	He,appm	dpa
-16	2.0	3.4
-14	2.4	3.9
-12	2.7	4.3
-10	3.0	4.7
-8	3.2	5.0
-6	3.4	5.2
-4	3.5	5.4
-2	3.6	5.5
0	3.6	5.5
2	3.5	5.4
4	3.4	5.3
6	3.3	5.1
8	3.0	4.8
10	2.8	4.4
12	2.5	4.0
14	2.1	3.5
16	1.7	2.9

316SS = Fe(0.645), Ni(0.13), Cr(0.18), Mn(0.019), Mo(0.026) wt%

Helium production in nickel and nickel alloys requires a more complicated non-linear calculation [6]. Helium production in stainless steel is thus detailed separately in Table 3.

FUTURE WORK

Additional experiments that are still in progress or not yet analyzed are detailed in the Fusion Reactor Materials Semiannual Progress Reports.

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