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Dosimetry in Mixed Neutron-Gamma Fields

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Abstract

The gamma field accompanying neutrons may, in certain circumstances, play an important role in the analysis of neutron dosimetry and even in the interpretation of radiation induced steel embrittlement. At the High Flux Isotope Reactor pressure vessel the gamma-induced reactions dominate the responses of ^{237}Np and ^{238}U dosimeters, and ^9Be helium accumulation fluence monitors. The gamma induced atom displacement rate in steel is higher than corresponding neutron rate, and is the cause of "accelerated embrittlement" of HFIR materials. In a large body of water, adjacent to a fission plate, photofissions contribute significantly to the responses of fission monitors and need to be taken into account if the measurements are used for the qualification of the transport codes and cross-section libraries.

1. INTRODUCTION

The primary function of the dosimetry used in nuclear reactor pressure vessel surveillance programs, studies of irradiation-induced steel embrittlement, and related experiments is to provide information on neutron flux, fluence and spectrum. The gamma field accompanying neutrons is usually not analyzed, because steel embrittlement is generally accepted as being caused by neutron-induced reactions, and because neutron-induced reaction rates (RRs) are usually much higher than gamma-induced reaction rates, which can, therefore, be neglected. However, there are situations in which gamma rays contribute significantly and need to be accounted for. Two such examples will be presented: an analysis of the dosimetry experiments at the pressure vessel (PV) of the Oak Ridge National Laboratory High Flux Isotope Reactor (HFIR), and a calculation of the radiation field in a large light water cube adjacent to a fission plate.

2. DOSIMETRY EXPERIMENTS AT THE HFIR PRESSURE VESSEL

2.1 HFIR, ACCELERATED EMBRITTLEMENT, AND THE NEED FOR IMPROVED RADIATION FIELD CHARACTERIZATION

The High Flux Isotope Reactor is a 100 MW flux-trap type reactor which began operation in 1966. A horizontal cross section of the HFIR is shown in Fig. 1. The reactor core, contained in a 2.44-m-diameter pressure vessel, consists of a series of ~0.61-m-high concentric annular regions. A target region (12.7-cm-diameter) at the core center is surrounded by two concentric fuel elements, each consisting of numerous curved fuel plates. The fuel is highly enriched ^{235}U in the form of U_3O_8 cermet. A typical core loading of 9.4 kg of ^{235}U provides a core cycle of ~22 days at 85 MW. The fuel is surrounded by a ring of beryllium reflector, approximately 0.30 m thick, and a ~0.6-m thick water annulus. The PV embrittlement surveillance is based on testing of Charpy specimens irradiated in locations referred to as keys, which are shown in Fig. 1. Each key provides locations for several surveillance capsules.

In 1986 tests of surveillance specimens revealed that the HFIR PV surveillance materials experienced a significantly larger increase in Charpy transition temperature than the increase observed in other test reactors, for a given fast neutron fluence ($E > 1\text{MeV}$). The HFIR was temporarily shut down; however, it resumed operation in 1989, after additional studies¹ proved that it could be operated safely for at least additional 10 years, at reduced power of 85 MW. The studies into the causes of "accelerated embrittlement" were initiated, and received considerable attention, because the neutron flux levels and the irradiation temperature in the HFIR are similar to the conditions at the PV supports of the commercial power reactors.² Different effects were considered, but the accelerated embrittlement remained unexplained.²⁻⁵ The original HFIR surveillance capsules included only one neutron dosimeter—a stainless steel wire, and the original transport calculations with the ELXSIR⁶ cross-section library were designed to predict fast fluxes only.¹ Therefore, to improve characterization of the neutron field, a dosimetry experiment and new transport calculations were initiated.

2.2 RESULTS FROM DOSIMETRY EXPERIMENTS AND ANALYSIS

The transport calculations were repeated using updated cross sections^{7,8} and a dosimetry experiment, referred to as

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Fig. 1 Horizontal cross section of the HFIR reactor.

DOS-1, was performed at key 7 (see Fig. 1). The primary goal of the DOS-1 was to provide reliable thermal fluxes.⁹ The thermal fluxes derived from cobalt and silver dosimeters, and from the lithium and boron helium accumulation fluence monitors (HAFMs) were consistent and gave a "measured" thermal flux of $\sim 2.3 \times 10^8 \text{ cm}^{-2} \cdot \text{s}^{-1}$ which agreed within $\sim 45\%$ with the thermal flux of $4.0 \times 10^8 \text{ cm}^{-2} \cdot \text{s}^{-1}$ from the new transport calculations. Surprisingly, the threshold dosimeters showed conflicting results. The fast neutron flux determined from nickel dosimeters was $1.5 \times 10^8 \text{ cm}^{-2} \cdot \text{s}^{-1}$, and was in good agreement with $1.8 \times 10^8 \text{ cm}^{-2} \cdot \text{s}^{-1}$ fast flux obtained earlier from stainless steel monitors, and with a calculated value of $1.2 \times 10^8 \text{ cm}^{-2} \cdot \text{s}^{-1}$. The fast flux derived from the ^{237}Np dosimeter and beryllium HAFM were, however, 17 and 15 times higher, respectively. This disparity triggered additional dosimetry experiments, named DOS-2 and DOS-3, in which comprehensive sets of threshold, thermal, and fission radiometric monitors, solid state track recorders (SSTRs), and beryllium HAFMs were irradiated.¹⁰ Besides at key 7, the irradiations were done at keys 2 and 4 also. The experiments at keys 2 and 4—which are rings around the neutron beam tubes—did not show any new effects not observed at key 7; however, their analysis was more complicated due to severe flux gradients. For these reasons, and because most of the measurements were done at key 7, only the key 7 results will be used in the discussion in this paper. The dosimetry capsule in key 7 was located in the core midplane and away from the beam tubes; therefore the neutron flux was relatively uniform across the capsule. The measured reaction rates (RR) are given in Table 1. The RR from different locations within the capsule—referred to in Table 1 as slots D, J, B, and A—show only small differences. Good reproducibility of the measurements was obtained, therefore eliminating the possibility of experimental errors. However, the disparity between ^{237}Np , beryllium HAFMs and other dosimeters was confirmed.

The procedure used to calculate the neutron multigroup fluxes at the locations of measurements was quite complex, and was described earlier.⁷ Here only the essential features will be repeated. A one-dimensional (1-D) HFIR model¹¹ and the AMPX¹² code system were used to self-shield and collapse the cross sections from the 99 neutron group ANSL-V⁸ library into 64-neutron energy groups; several energy groups with upscattering were retained in the thermal energy range. The DORT code,¹³ the 64-group cross sections, and a two-dimensional HFIR model¹⁴ were used to calculate directional fluxes throughout the HFIR. The directional fluxes were then reformatted and transformed into boundary sources for the three-dimensional models,¹⁵ which covered the vicinity of the surveillance locations. Three-dimensional calculations were performed with the TORT computer code.¹⁶ The 64-neutron-group library, a P_3 approximation to the angular dependence of the anisotropic scattering cross sections, and an S_{10} directional quadrature set were used.

With the neutron fluxes from transport calculations the calculated RRs were determined. The calculated-to-measured RR ratios (C/M) are also given in the Table 1. The arithmetic averages of the C/M ratios are 0.67 ± 0.07 (0.07 is the standard deviation of the average) for the threshold dosimeters, 1.50 ± 0.31 for the gadolinium-covered thermal monitors, and 1.38 ± 0.14 for the bare thermal monitors. This indicates that the calculation underpredicted the fast neutron fluxes by approximately 30% and overpredicted the thermal fluxes by $\sim 40\%$. The ^{237}Np , ^{238}U and ^9Be monitors (shown in bold print in Table 1) were excluded from the averages given above; for these monitors the average C/M

Table 1. Measured reaction rates and calculated-neutron-induced-to-measured (C/M) reaction rate ratios.

REACTION	Measured Reaction Rates (RRs)				Average Measured	SS*	C/M
	DOS 1	DOS 2	DOS 3	DOS 4			
	$s^{-1} \cdot atom^{-1}$				$s^{-1} \cdot atom^{-1}$		
Slot D							
Al27(n,p)Na24		1.30E-18	1.31E-18		1.30E-18		0.63
Ti48(n,p)Sc48		5.85E-19	4.95E-19		5.40E-19		0.56
Cu63(n, α)Co60		6.55E-19	6.26E-19		6.40E-19		0.67
Ti46(n,p)Sc46		9.76E-18	8.82E-18		9.29E-18		0.62
Fe54(n,p)Mn54		3.29E-17	3.41E-17		3.35E-17		0.75
Ti47(n,p)Sc47		7.64E-18	6.89E-18		7.26E-18		0.71
Ni58(n,p)Co58, in Gd	4.13E-17			4.41E-17	4.27E-17		0.74
Np237(n,f), in Gd	4.98E-15				4.98E-15		0.05
Sc45(n, γ)Sc46, bare		6.33E-15			6.33E-15		1.38
Co59(n, γ)Co60, bare	7.16E-15				7.16E-15	1.13	1.55
Fe58(n, γ)Fe59, bare		2.90E-16			2.90E-16		1.45
Sc45(n, γ)Sc46, in Gd			2.37E-16		2.37E-16		1.69
Co59(n, γ)Co60, in Gd	3.35E-16				3.35E-16	1.96	1.63
Co59(n, γ)Co60, in Gd, diluted				6.61E-16	6.61E-16		1.62
Fe58(n, γ)Fe59, in Gd			1.73E-17		1.73E-17		1.61
Slot J							
U238(n,f), SSTR, in Gd			1.36E-15		1.36E-15		0.05
Np237(n,f), SSTR, in Gd			4.75E-15		4.75E-15		0.05
Be9(n,x)He		1.43E-15	1.46E-15		1.45E-15		0.05
Co59(n, γ)Co60, bare	7.47E-15				7.47E-15	1.13	1.48
Slot B							
U235(n,f), in Gd			7.75E-15		7.75E-15		1.05
U238(n,f), in Gd			1.60E-15		1.60E-15		0.04
Np237(n,f), in Gd			4.59E-15		4.59E-15		0.05
Be9(n,x)He	1.46E-15				1.46E-15		0.05
Li6(n, α)H3, bare	2.04E-13				2.04E-13		1.48
B10(n, α)Li7, bare	8.54E-13				8.54E-13		1.44
Slot A							
Ni58(n,p)Co58	4.29E-17	4.30E-17	4.63E-17	4.48E-17	4.42E-17		0.72
Co59(n, γ)Co60, bare	7.91E-15	7.92E-15			7.91E-15	1.13	1.40
Co59(n, γ)Co60, in Gd			5.33E-16		5.33E-16	1.96	1.02
Co59(n, γ)Co60, bare, diluted				8.88E-15	8.88E-15		1.41
Au197(n, γ)Au198, bare		2.38E-14			2.38E-14	1.68	1.14
Au197(n, γ)Au198, bare, diluted				3.89E-14	3.89E-14		1.17

* All reaction rates are given as measured. The measured RRs, for which the self-shielding correction factors (SS) are listed, were multiplied by SS factors before the C/M ratios were determined.

was around 0.05. Clearly, the neutron field consistent with the responses of other monitors created only small fractions of the responses of these three monitors; the majority of their responses must come from sources other than neutron reactions.

A feature common to the ^{237}Np , ^{238}U and ^9Be monitors is that the same reaction products can be generated not only by neutron reactions but also by gamma-ray reactions, that is photofissions in ^{237}Np and ^{238}U , and $^9\text{Be}(\gamma,n)^8\text{Be}(2\alpha)$ reaction in ^9Be . The contributions from gamma-ray reactions were estimated from the coupled neutron-gamma calculations with 39-neutron groups and 44-gamma groups,⁸ and with a 1-D model of HFIR (which was used in the cross-section preparation). These calculations were originally performed to investigate the effects of neutrons generated by gamma rays in the beryllium reflector.⁷ The neutron- and gamma-induced RR versus distance from the core axis, as obtained from the 1-D calculation, are plotted in Fig. 2. The calculations showed that the gamma flux ($E > 1 \text{ MeV}$) in the core is

only about 2.5 times higher than the neutron flux ($E > 1\text{ MeV}$), and gamma reaction rates are negligible compared to neutron reaction rates. However, going outward from the core towards the pressure vessel, the neutrons are effectively slowed down by beryllium and water, and the fast flux decreases rapidly with distance from the core. On the other hand, since both beryllium and water contain only light elements (low Z), they have low electron density and consequently low gamma absorption. Therefore, outside the core the gamma flux decreases much slower than the neutron flux. At $\sim 40\text{ cm}$ and $\sim 50\text{ cm}$ from the core the photofission and photoneutron reaction rates, respectively, become equal to corresponding neutron reaction rates and then grow increasingly dominant towards the pressure vessel. Inside the pressure vessel wall, however, the gamma flux decreases rapidly. The calculations indicated that practically all high energy gamma rays that were important for photofission, photoneutron and atom displacements at the pressure vessel originated from the core and the beryllium reflector. The contributions from gamma rays from neutron capture in water were mainly below $\sim 2.4\text{ MeV}$ and were mostly below 10%, and the contributions from captures in iron (in the pressure vessel) were negligible. The 1-D calculation showed the trends, but could not predict the absolute fluxes, probably because the neutron beam tubes and other geometrical features could not be included. For this reason the gamma-to-neutron reaction rates were determined at the location closer to the core, where the fast neutron flux from the 1-D calculation was equal to the fast flux from three-dimensional calculations, and not at the actual radial location of the dosimeters. These gamma RRs are compared with the calculated neutron-induced reaction rates and measured RRs in Table 2. The gamma induced RRs were calculated to be $\sim 15\text{--}21$ times higher than neutron RRs in ^{237}Np , ^{238}U , and ^9Be monitors. Photofission contributes only $\sim 20\%$ to the Uranium-235 response, much less than in the other two fission dosimeters, because the ^{235}U neutron-induced fission rate is much higher than those for other dosimeters. After the contribution of gamma-induced reactions are taken into account, the C/M ratios of ^{237}Np and ^{238}U monitors are close to the values for other dosimeters, while the beryllium response is slightly overestimated.

Table 2. Comparison of measured (R_m) and calculated neutron (R_n) and gamma (R_γ) reaction rates.

	R_m	R_n	R_γ	R_γ/R_n	$(R_n + R_\gamma)/R_m$
	$\text{s}^{-1} \cdot \text{atom}^{-1}$				
^{237}Np , fission	4.77 E-15	2.25 E-16	3.44 E-15	15.3	0.77
^{238}U , fission	1.48 E-15	6.24 E-17	1.07 E-15	17.1	0.77
^9Be to ^4He	1.45 E-15	7.60 E-17	1.60 E-15	21.0	1.16
^{235}U , fission	7.75 E-15	8.13 E-15	1.76 E-15	0.22	1.28

To further verify the calculated gamma field, an additional experiment, named DOS 4, was performed at key 7. Several sets of polychlorostyrene films were irradiated together with neutron dosimeters (see Table 1). Polychlorostyrene films, analyzed by the National Institute of Standards and Technology,¹⁷ provided a measured absorbed gamma-dose rate of 36.4 Gy·s⁻¹, in excellent agreement with the calculated value of 36.6 Gy·s⁻¹. However, most of the dose in silicon is due to the gamma rays below the threshold for the photon-induced reactions in ²³⁷Np, ²³⁸U, and ⁹Be.

Finally, to further test the consistency of the calculations and measurements, the spectrum adjustment technique was used. The adjustment technique, which was traditionally applied to the neutron spectrum only, was extended to allow simultaneous adjustment of the neutron and gamma spectra.¹⁸ The calculations were performed with the LSL-M2 computer code.¹⁹ A first adjustment run, with all the measured responses and neutron spectrum only, clearly indicated the inconsistencies between the ²³⁷Np, ²³⁸U, and ⁹Be monitors and the other dosimeters and the calculated neutron field. Hence the ²³⁷Np, ²³⁸U, and ⁹Be monitors were omitted and adjustment was performed with the remaining dosimeters. No further inconsistencies in the input data were found; the adjusted irradiation parameters are given in Table 3, in the column labelled "Adjusted *n-only*". For the simultaneous neutron and gamma (*n* + γ) adjustment the spectra and covariances were arranged as shown schematically in Fig. 3. The same RRs as for the *n-only* adjustment were used and the ²³⁷Np, ²³⁸U, and ⁹Be responses were input into the adjustment procedure without any prior corrections (i. e., as measured). The measured dose rate in silicon was also added. Detailed comparison of the *n-only* and *n* + γ runs showed similar adjustments of the measured reaction rates. In the *n* + γ run the average of the absolute values of the adjustments of dosimeter responses was ~2.8% for the neutron threshold dosimeters and ~2.8% for the bare and gadolinium-covered thermal dosimeters, while in the *n-only* run the adjustments for these two groups of monitors were 2.8% and 2.6%. For the dosimeters with contributions from gamma reactions the average RR adjustment was ~5.5%. The last group of responses, used only in the *n* + γ run, experienced the largest adjustment; however, it included mostly fission dosimeters for which the scatter of different fission products activities is usually larger than the scatter of the other dosimeter responses.

Fig. 3. Structure of the input data for the simultaneous *n* + γ adjustment.

An example of the results from the simultaneous *n* + γ and *n-only* adjustments is given in Table 3. The neutron irradiation parameters from the two runs are practically identical; differences are less than 1%. Therefore, the ²³⁷Np,

Table 3. Neutron and gamma irradiation parameters and their standard deviations at key 7. Adjusted values are for neutron only (*n-only*) and for simultaneous neutron and gamma (*n* + γ) spectrum adjustment.

Irradiation parameter	Before adjustment*		Adjusted*				Ratio**
			<i>n-only</i>		<i>n + γ</i>		
	Value	± σ	Value	± σ	Value	± σ	
Φ _{>1MeV}	1.17 E+08	± 38	1.58 E+08	± 8	1.57 E+08	± 8	1.34
Φ _{>0.1MeV}	1.75 E+08	± 38	2.34 E+08	± 11	2.32 E+08	± 11	1.33
Φ _{<0.4 eV}	3.97 E+08	± 137	2.70 E+08	± 6	2.70 E+08	± 6	0.68
<i>n-dpa/s</i>	1.82 E -13	± 37	2.47 E -13	± 7	2.46 E -13	± 7	1.35
<i>γ-dpa/s</i>	1.24 E -12	± 45			1.21 E -12	± 7	0.98
Dose rate in Silicon	3.66 E+01	± 46			3.28 E+01	± 13	0.90

* Units are cm²·s⁻¹ for fluxes, s⁻¹ for dpa, Gy·s⁻¹ for dose rate, and percentages for standard deviation (σ).

** Ratio of the adjusted value and the value before the adjustment.

^{238}U , and ^9Be responses used in the $n + \gamma$ adjustment did not distort the adjustment of the neutron spectrum, because it was determined by the large number on monitors sensitive to neutrons only. The ^{237}Np , ^{238}U , and ^9Be monitors, with about 95% of their responses coming from gamma-induced reaction, acted essentially as gamma-ray monitors. This allowed adjustment of the gamma field. The measured gamma dose rate in silicon, and to a smaller degree the ^{235}U monitor, also contributed to the adjustment of the gamma field.

The simultaneous $n + \gamma$ adjustment contributed to the verification of the consistency of the measurements and the calculated neutron and gamma fields and provided best estimate of the neutron and gamma irradiation parameters. The parameter of particular interest is the gamma-induced displacement per atom (γ -dpa) rate which was calculated with the cross sections from Baumann.²⁰ At key 7, the γ -dpa rate was found to be approximately 5 times higher than the neutron dpa (n -dpa) rate. In other locations analyzed (i. e., key 2 and 4) the γ -dpa was less dominant, probably because of the higher neutron fluxes due to leakage from the beam tubes, but was still about 1.1 to 1.5 times higher than n -dpa. Taking into account the contributions from the gamma field was essential for the explanation of accelerated embrittlement of the HFIR materials: when the sum of neutron and gamma dpa is used for the interpretation of the HFIR surveillance results, the HFIR data are in agreement with data from other test reactors.²¹ Accelerated embrittlement in the HFIR was caused therefore by a strong gamma field, which was not accounted for in previous analyses and which contributes significantly to the neutron-induced radiation damage in steel.

The ratio of γ -dpa rate to n -dpa rate, obtained from one-dimensional calculations, changes more than four orders of magnitude, from $\sim 6 \times 10^{-4}$ at the core center to ~ 25 just in front of the pressure vessel. The ratios of $\text{Np}(\gamma, f)/\text{Np}(n, f)$, $^{238}\text{U}(\gamma, f)/^{238}\text{U}(n, f)$, and $\text{Be}(\gamma, n)\text{He}/\text{Be}(n, x)\text{He}$ experience similar variations, as shown in Fig. 4. The γ -dpa/ n -dpa ratio was, however, within 0.20 to 2.1 of the $\text{Np}(\gamma, f)/\text{Np}(n, f)$ ratio, within 0.10 to 1.16 of the $^{238}\text{U}(\gamma, f)/^{238}\text{U}(n, f)$ ratio and below ~ 0.4 of the $\text{Be}(\gamma, n)\text{He}/\text{Be}(n, x)\text{He}$ ratio over the whole geometry range, as is also shown in Fig. 4. The ^{237}Np , ^{238}U and Be HAFM can therefore be used, in combination with other threshold dosimeters, to check for the presence of gamma fields of such intensity and high energy that they may contribute considerably to radiation damage in ferritic metals. The beryllium HAFM appears to be the most suitable, since it is relatively more sensitive to gamma rays—the $\text{Be}(\gamma, n)\text{He}/\text{Be}(n, x)\text{He}$ ratio was always higher than the γ -dpa/ n -dpa ratio. The threshold for the gamma-induced reaction on Be is ~ 1.6 MeV—similar to the γ -dpa threshold at 1 MeV—and much lower than photofission thresholds of ^{237}Np and ^{238}U , which are at ~ 5.5 MeV. In the gamma field at the HFIR PV $\sim 90\%$ of the γ -dpa is caused by gamma rays with energies between ~ 1.6 and 7.5 MeV, and those gamma rays also contribute about 90% of the γ -induced response in Be HAFM. Gamma rays with energies above ~ 5.5 MeV, which cause photofissions in ^{237}Np and ^{238}U , contributed less than 50% to the γ -dpa. Finally, more than 50% of the gamma dose rate in silicon was due to < 1 MeV gamma rays, which do not contribute to γ -dpa at all. This is illustrated in Fig. 5, which shows the cumulative relative responses of γ -dpa, Be HAFM, and photofission reactions in ^{237}Np , and ^{238}U , versus gamma-ray energy.

Fig. 4 On the left the ratios of gamma-to-neutron-induced RR are plotted versus distance from the core axis. The γ -dpa to n -dpa ratio is also given. On the right the "doble ratios" are shown.

Fig. 5. Cumulative relative responses of γ -dpa, Be HAFM, gamma dose rate in silicon, and photofission reactions in ^{237}Np , and ^{238}U , versus gamma-ray energy in the gamma-ray field at the HFIR pressure vessel.

3. FISSION-PLATE DRIVEN RADIATION FIELD IN LIGHT WATER CUBE

An experiment, named WINES, was recently performed at Mol, Belgium, to provide data for testing the capabilities of transport codes and cross sections to predict neutron transport through large thicknesses of water.²² Neutrons from a thermal-neutron-beam driven fission plate were allowed to propagate into a large (1 m³) cube of water. Fission rates in ^{232}Th , ^{235}U (bare and cadmium covered), ^{237}Np , and ^{238}U were measured with fission chambers at several locations in the water, on the axis perpendicular to the fission plate. Preliminary calculations, performed with deterministic (DORT) and Monte Carlo (TRIPOLI²³) codes, and ENDF/B-IV- and ENDF/B-VI-based cross sections underpredicted the measurements by ~15–45% at 25 cm in the water and by factors ~5–20 at 50 cm. The only exception was ^{235}U , which was overpredicted.

The experiment investigated radiation transport through water (low Z material). Consequently, photofissions could contribute to the measured RRs. Therefore, it is of interest to calculate the gamma field and the corresponding RRs. Since details about the WINES experiment were not available, the test calculations used an approximate model. A fission plate (~20-cm diameter) made of ^{235}U , clad with aluminum, was placed in a cylindrical (~27-cm diameter) beam tube, at ~36 cm from the face of the water cube. The beam tube was surrounded by concrete and iron shielding, and a 1.4-cm thick boral thermal filter was placed between the fission plate and the water cube. The DORT computer code was used with the r -z geometry model. Calculations were done for a fixed source, defined with a uniform fission density in the fission plate. The ENDF/B-VI-based BUGLE-96 library, with 47 neutron groups and 20 gamma groups²⁴ was used. The library included upscattering below ~5eV (i.e., for groups 43–47), and numerous outer as well as inner iterations were necessary to converge the fluxes. Besides the coupled n - γ calculation, the prompt and delayed fission gamma fields were calculated separately; however, their contribution to photofissions were found to be small compared to secondary ("non-fission") gamma rays. The ratios of gamma- and neutron-induced fission rates are given in Table 4.

The relative importance of photofissions of course increases with the thickness of water penetrated. For example, at 50 cm in the water, the photofission rate is about 50% of the neutron-induced fission rate in ^{232}Th and ~20% in ^{238}U and in ^{237}Np . At 95 cm, photofission rates are about 40 times higher than neutron-induced fission rates in ^{232}Th , about 20 times higher in ^{238}U and ^{237}Np , and about 2 times higher in cadmium-covered ^{235}U . Since the intensity and other specifics of the thermal-neutron beam were not available, the beam was not included in the calculations, and the gamma rays

Table 4 Ratios of gamma-to-neutron induced fission rates versus distance in the water.

Fission monitor	20 cm	30 cm	40 cm	50 cm	60 cm	75 cm	85 cm	95 cm
^{237}Np	0.007	0.02	0.06	0.18	0.53	2.52	6.86	20.4
^{238}U	0.011	0.03	0.07	0.22	0.64	2.91	7.74	21.6
^{235}U in Cd	0.000	0.001	0.003	0.01	0.03	0.16	0.47	1.98
^{232}Th	0.027	0.06	0.15	0.47	1.34	5.99	15.6	42.5

arising from the interactions of the beam with the fission plate, thermal neutron filters and other structures are not accounted for in Table 4. The gamma-to-neutron fission-rate ratios given above should therefore be regarded as lower bounds for the actual experiment. They indicate that the gamma contributions are substantial, and a complete analysis, including the thermal-neutron beam, is necessary. Given the magnitude by which the preliminary calculations underestimated the measurements,²² it does not appear probable that photofission contributions will be sufficient to explain the discrepancies; however, the photofissions need to be taken into account if measurements are to be used for the qualification of the transport codes and cross-section libraries.

4. CONCLUSIONS AND RECOMMENDATIONS

In the radiation field near the HFIR PV with the gamma-to-neutron fast flux ($E > 1\text{ MeV}$) ratio of $\sim 1 \times 10^4$, the gamma-ray induced reactions dominate the responses of ^{237}Np and ^{238}U dosimeters, and ^9Be HAFMs. These monitors performed essentially as gamma-ray dosimeters. It is therefore suggested that they may be used as such in high-intensity, high-energy gamma fields, especially for long-term, high-dose irradiations, which may be difficult to monitor with other gamma-ray dosimeters. Due to its lower gamma-reaction threshold, beryllium HAFM appears to be the most suitable. Improved photofission and photoneutron cross sections would be needed for such applications.

The gamma-induced *dpa* in the HFIR surveillance specimens was found to be higher than neutron-induced *dpa* and it was essential for understanding the higher than expected embrittlement of HFIR materials. Evidence was found that if the gamma field affects significantly the (above mentioned) dosimeter responses, it should be taken into account also in the assessment of radiation embrittlement in metals.

Radiation fields with strong gamma-ray components are likely to be found whenever the neutron source and the target region are separated by thick layers of light materials, which attenuate neutron fluxes much faster than gamma-ray fluxes.

REFERENCES

1. R. D. Cheverton et al., Oak Ridge National Laboratory Report, ORNL/TM-10444 (April 1988).
2. R. D. Cheverton, F. B. K. Kam, R. K. Nanstad and G. C. Robinson, Nuc. Eng. Des. 117 (1989) 349.
3. R. K. Nanstad, K. Farrell, D. N. Braski and W. R. Corwin, J. Nucl. Mater. 158 (1988) 1.
4. L. K. Mansur and K. Farrell, J. Nucl. Mater. 170 (1990) 236.
5. K. Farrell, S. T. Mahmood, R. E. Stoller and L. K. Mansur, J. Nucl. Mater. 210 (1994) 268.
6. R. E. Maerker et al., Oak Ridge National Laboratory, EPRI NP-3654 (1984).
7. J. V. Pace III, C. O. Slater and M. S. Smith, Reactor Dosimetry, ASTM STP 1228, pp. 140–146 (1994).
8. W. E. Ford et al., Oak Ridge National Laboratory Report, ORNL-6618 (1990).
9. K. Farrell et al., Oak Ridge National Laboratory Report, ORNL/TM-12511 (1994).
10. I. Remec and F. B. K. Kam, NUREG/CR-6117, ORNL/TM-12484 (Nov. 1993).
11. R. T. Primm III, Oak Ridge National Laboratory Report, ORNL/TM-11956 (Oct. 1991).
12. N. M. Greene, et al., Oak Ridge National Laboratory Report, ORNL/CSD/TM-283 (Oct. 1992).
13. W. A. Rhoades and R. L. Childs, Nucl. Sci. Engr. 99, 1, pp. 88–89, (May 1988).
14. W. W. Engle, RZ Base case, Shielding Input to the HFIR SAR, EPM.HFIR.91.001, (Jun. 1991).
15. R. L. Childs et al., "Three-Dimensional Calculations of Neutron Streaming in the Beam Tubes of the ORNL HFIR Reactor", Proc., 7th Internat. Conf. on Radiation Shielding, VOL II, (Sept. 1988).
16. W. A. Rhoades and R. L. Childs, Oak Ridge National Laboratory Report, ORNL-6268 (Nov. 1987).
17. J. Grundl, Report of Special Measurement of HFIR Pressure Vessel by use of NIST FWT-67-20 Chlorostyrene Dosimeters, personal comm. to F. Kam, ORNL; in NUREG/CR-6117.
18. I. Remec, "Simultaneous Neutron and Gamma Spectrum Adjustment", Proc. 9th ASTM-Euratom Symposium on Reactor Dosimetry, to be published.
19. F. W. Stallmann, NUREG/CR-4349, ORNL/TM-9933 (1986).
20. N. P. Baumann, Proc. 7th ASTM-Euratom Symp. on Reactor Dosimetry, Strasbourg, p. 689, (1990).
21. I. Remec et al., J. Nucl. Mater. 217, pp. 258–268, (1994).
22. H. A. Abderrahim et al., "WINES: Water Inelastic Neutron Scattering Experimental Study," Proc. 9th ASTM-Euratom Symposium on Reactor Dosimetry, to be published.
23. J. C. Nimal et al., CEA report, DMT/93/022 SERMA/LEPP/93/1468.
24. J. E. White et al., BUGLE-96, Oak Ridge National Laboratory, RSIC DLC-185 (1996).

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