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Neutron Dosimeter Performance and Associated Calibrations at Nuclear Power Plants

Prepared by R. B. Schwartz/NBS
G. W. R. Endres, F. M. Cummings/PNL

National Bureau of Standards

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Prepared by
R. B. Schwartz*
G. W. R. Endres**, F. M. Cummings**

*National Bureau of Standards
Gaithersburg, MD 20760

**Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Facility Operations
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ABSTRACT

This report addresses problems associated with the calibration and use of personnel neutron dosimeters and monitoring instruments. Four particular items are addressed:

- The threshold response of NTA film. NTA film is not recommended for use at reactors.
- A discussion of dosimeter and remmeter calibrations performed using a D_2O-moderated ^{252}Cf source. Use of the moderated Cf source is recommended for calibrating dosimeters and instruments used at reactors.
- The edge effect created by placing the neutron-sensitive elements of albedo dosimeters close to the phantom edge. It is recommended that dosimeters be no closer than 7 cm effective distance from the edge of the phantom on which they are irradiated.
- The response of various personnel neutron dosimeters inside containment at nuclear power plants. It is recommended that dosimeters which demonstrate adequate sensitivity be used and be corrected for variations due to neutron energy spectral differences. Dosimeters that were found to be adequate were TLD-albedo dosimeters and polycarbonate track-etch dosimeters which utilized (n,α) radiators. NTA films, CR-39 films and polycarbonate films which did not use radiators are inadequate for personnel neutron dosimetry at nuclear power plants.

SUMMARY

We have investigated the threshold response of NTA film and the responses of other types of neutron dosimeters irradiated using bare and moderated ^{252}Cf sources. The effect of position of albedo dosimeters on the phantom was also investigated. Additionally, dosimeters were irradiated inside containment of nuclear power plants and the results of those irradiations are interpreted.

Several types of NTA film badges were irradiated at the National Bureau of Standards (NBS) Van de Graaff facility to monoenergetic neutrons with energies between 0.5 and 1.2 MeV. The response of the dosimeters at 0.5 MeV was negligible. Under laboratory conditions, the response threshold occurred at 0.6 MeV (the response threshold is defined as that point at which the response is 1/e below the response to a bare ^{252}Cf source). Inside reactor containment, however, the conditions are very nearly "worst case." The threshold was determined to be 1.2 MeV for those conditions. Since the neutron energy spectra inside containment are predominately below 1 MeV, it is recommended that NTA film not be used as personnel neutron dosimeters in containment.

Several types of dosimeters were irradiated using bare and moderated ^{252}Cf. The results of those irradiations show that the D_2O-moderated ^{252}Cf source more closely approximates the neutron energy spectra measured inside nuclear power plants than does the bare ^{252}Cf source. The albedo dosimeters and moderated instruments had observed responses close to calculated responses. It is therefore recommended that the moderated californium source be used to test and calibrate dosimeters and remmeters which are to be used at nuclear power reactors, or in any other environment where much of the exposure is to low and intermediate energy neutrons.

The response of albedo dosimeters depends on the homogeneity of the reflected neutron spectrum. An array of albedo dosimeters was placed on a phantom and irradiated using the bare and moderated ^{252}Cf sources to investigate the effects of inhomogeneities. The responses were compared as a function of position. The results show that the "edge effect" is minimized when the neutron-sensitive TLD chip in the dosimeter is an effective distance greater than 7 cm from the edge of the phantom. In cases where the location of the

TLD chip inside the dosimeter is not known, no part of the dosimeter should be less than the 7 cm effective distance from the edge of the phantom.

The last part of the report compares the responses of dosimeters used in nuclear power plants to the responses of a rem meter. Dosimeters were irradiated on water phantoms inside one boiling water reactor (BWR) and inside containment of two pressurized water reactors (PWR) while the reactors were at 100% power. When compared to the rem meter measurements: 1) TLD albedo dosimeters calibrated using moderated ^{252}Cf exhibited good accuracy and precision; 2) TLD-albedo dosimeters whose responses were corrected for spectral variations exhibited good accuracy and precision; 3) TLD-albedo dosimeters which utilized other calibration sources and were not corrected for spectral variations responded high; 4) polycarbonate track etch film used by itself with no radiators failed to respond; 5) polycarbonate track etch film used in conjunction with multiple boron-loaded radiators responded high, and 6) CR-39 responded very low inside reactor containment when it responded at all. It is therefore recommended that TLD-albedo dosimeters and polycarbonate dosimeters which employ (n,α) radiators be the dosimeters of choice for nuclear power plants and further that the dosimeter response be corrected by proper calibration techniques or by field measurements which yield corrections for spectral variations.

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1.0 THE RESPONSE OF KODAK NTA FILM TO NEUTRONS NEAR THRESHOLD

1.1 INTRODUCTION

The disadvantages of NTA film as a neutron dosimeter are by now quite well known. As pointed out in a recent study (Brackenbush et al. 1980) these disadvantages include poor sensitivity at low energies, latent image fading from high humidity (e.g., 50% latent image loss after two weeks at 60% relative humidity, and 22°C temperature) (Bartlett 1978), film fogging from high temperatures, and film fogging from exposure to gamma rays. In addition, the readout can be quite tedious. Despite these difficulties, NTA is still being used for neutron dosimetry at several facilities; in particular, at the time this work was started, film was being used at several nuclear power plants. Since it is now known that the neutron spectra encountered by personnel at nuclear plants consist primarily of intermediate and low energy neutrons (Endres et al. 1981), it was felt to be important to make careful measurements of the response of NTA film as a function of neutron energy near its threshold, to determine whether film would be at all useable in these low energy environments.

1.2 EXPERIMENTAL METHOD

The film badges were a conventional type supplied by the U.S. Army Signal Corps. The film itself was Kodak Personal Neutron Monitoring Film Type A, generally referred to as "NTA Film." This is by far the most commonly used emulsion for neutron personnel dosimetry. The film badges were irradiated at several different neutron energies between 0.5 and 1.2 MeV, using monoenergetic neutrons from the NBS Van de Graaff (Wasson 1977). The irradiation details are given in Table 1.1. We had originally intended to do a measurement at 14 MeV, but the NBS 14 MeV facility was not yet operational at the time this work was done. As explained later in this report, however, we were able to normalize our data to higher energy data appearing in the literature and thus tie in a 14 MeV datum point to our results (Eisen et al. 1980).

TABLE 1.1. Film Badge Irradiations on the NBS Van de Graaff

Neutron Energy MeV	Neutron Fluence Neutrons/cm^2	Conversion Factor rem/(n/cm^2)	Dose Equivalent rem	Number of Observed Tracks Tracks/(mm^2-rem) ≥4 grains	≥7 grains
0.49 to 0.53	4.89×10^7	2.57×10^{-8}	1.26	23	2.7
0.59 to 0.64	3.75×10^7	2.72×10^{-8}	1.02	48	7.6
0.68 to 0.75	3.80×10^7	2.92×10^{-8}	1.11	50	13.5
0.81 to 0.85	3.24×10^7	3.20×10^{-8}	1.03	54	18.5
0.90 to 0.97	2.02×10^7	3.48×10^{-8}	0.70	49	23.5
0.91 to 0.97	7.60×10^7	3.49×10^{-8}	2.65	56	32
1.01 to 1.07	3.30×10^7	3.65×10^{-8}	1.21	63	35
1.11 to 1.18	2.20×10^7	3.64×10^{-8}	0.80	66	47
^{252}Cf (bare)	2.95×10^7	3.39×10^{-8}	1.00	116	110

The ^7Li (p,n) reaction was used as the neutron source, with the neutron beam at 0° to the incident proton beam. The neutron fluence was determined with a calibrated plastic scintillator, viewed by a photomultiplier tube (so-called "Black Detector").

The experimental arrangement is shown in Figure 1.1.

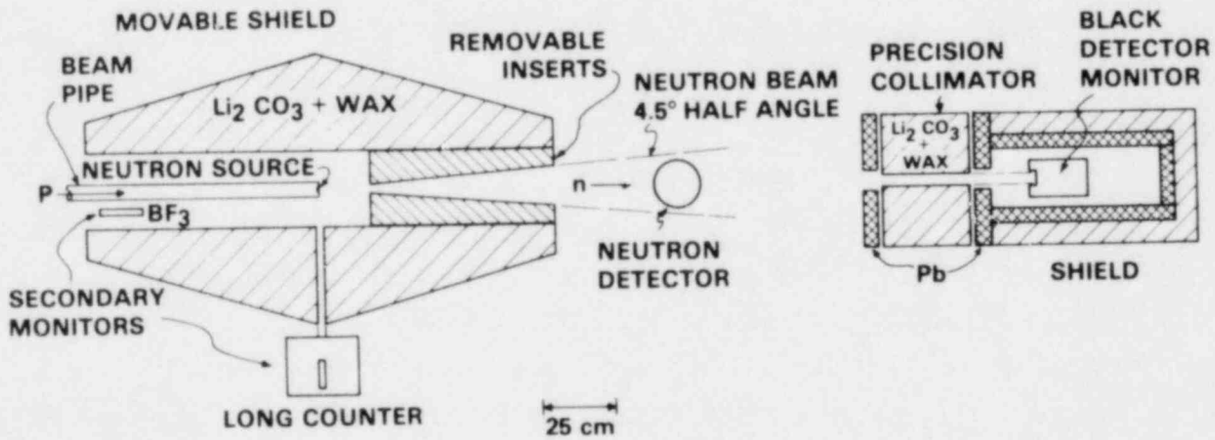

FIGURE 1.1. Irradiation Configuration for Irradiations Using the NBS Van de Graaff (Wasson 1977)

Four badges were irradiated simultaneously at each of the monoenergetic neutron energies and, in addition, a group of five badges was irradiated at the NBS ^{252}Cf facility. This facility is described in a recent publication (Grundl et al. 1977). Badges were also irradiated with the D_2O-moderated ^{252}Cf source, and the results of these irradiations are presented and discussed in the section of this report on the response of neutron personnel dosimeters and neutron measuring instruments to the D_2O-moderated source. All measurements reported here refer only to those done with bare ^{252}Cf.

The film badges were all mounted in air, perpendicular to the neutron beam. They were irradiated to dose equivalents between 0.7 and 2.6 rem, with most of the irradiations being approximately one rem.

The films were processed by the Lexington-Blue Grass Army Depot using normal film processing procedures. While at NBS, the films were kept in an air-conditioned humidity-controlled laboratory, but no other special precautions were taken to safeguard the test films from the effects of heat and humidity. It was reported, however, that there was no evidence of track fading from high humidity, nor film fogging from high temperature. Since the gamma doses varied between about 25 mrem and 70 mrem, there was no serious fogging from gammas. About one rem of photons can darken the film enough to prevent tracks from being counted (Brackenbush et al. 1980).

The first column in Table 1.1 lists the neutron energies; the spread in energy is that due to the thickness of the lithium target. The second column is the measured neutron fluence; the estimated uncertainty is $\pm$ 4%. The third column is the fluence to dose equivalent conversion factor (NCRP 1971 and Hankins 1977a). The fourth column, the dose equivalent, is just the product of the second and third columns.

For each irradiation described in Table 1.1, results were obtained for the number of tracks as a function of the number of grains making up the track. Now, the question of how many grains it takes to make an identifiable track does not have a clear, objective, answer. Under laboratory conditions, the lower detectable limit of track length is often defined as 3 or 4 grains in a row (Dudley 1966). (For the results reported here, 4 grains did suffice to

identify a track.) Under less ideal conditions, however, this definition may not be justified. For example, fading due to humidity could cause a 3-grain track to be missed, while a longer track would still be seen. On the other hand, if 3 grains is defined as a neutron track, then a high gamma dose could be interpreted as being partly due to neutrons. Finally, one always depends on the judgement of the technician reading the film and the time he is willing to spend following a track along its length. (This is particularly important for tracks which are nearly perpendicular to the surface of the emulsion and thus require great care in refocusing the microscope to follow the track.) The role of the technician is particularly important in a test like this. The technicians were aware that it was a test, due to the high level of exposure and the particular track length distribution. Thus, despite the subjective desire to treat these films the same as the other (routine) films, it is very likely that these films were read with more care than usual. In short, track counting is still more of an art than a science, and the word "ambiguous" best describes the criteria for identifying tracks. On the basis of discussions with people in track counting laboratories[(a)] it would seem that 3 or 4 grains suffice to identify a track under ideal conditions, but that it may take 5 to 7 grains to identify a neutron track under more usual conditions where there may be fading, confusion with gammas, and, possibly, less painstaking study.

We thus consider two groups of track lengths in column 5 of Table 1.1: those consisting of 4 or more grains, and those consisting of 7 or more grains. The former group represents a slightly conservative "best case" and the latter, perhaps, a "worst case."

1.3 RESULTS AND DISCUSSION OF RESULTS

The results are shown in Figure 1.2. There is a clear separation between the two groups of tracks ($\geq$4 grains and $\geq$7 grains) for the monoenergetic irradiations. With increasing energy the tracks get longer and thus the difference between the two groups decreases with increasing energy. At ^{252}Cf the difference in the number of tracks for the two groups is only $\sim$5%, and hence

(a) AE Abney and RV Wheeler.

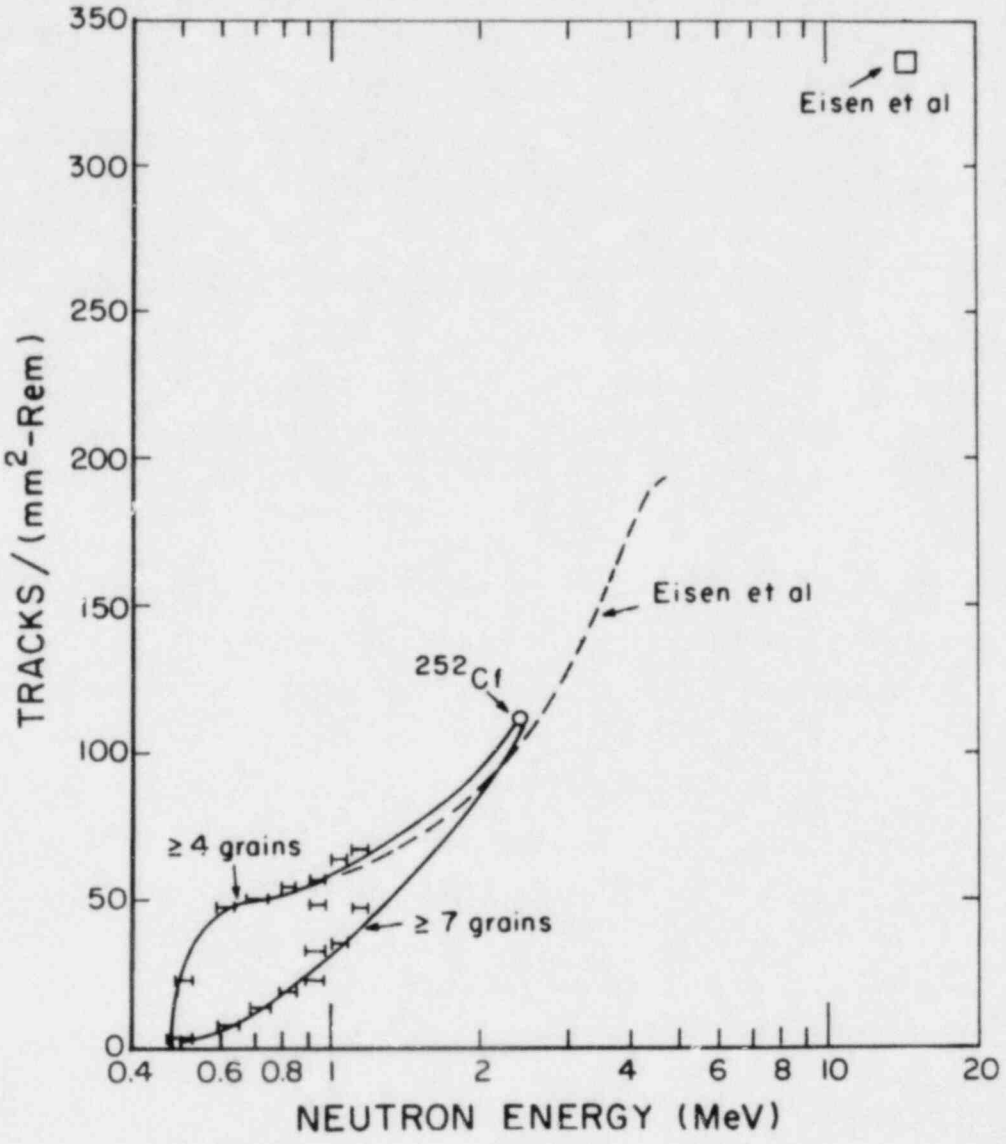

FIGURE 1.2. The Energy Response of Film Badges

only the average is plotted, at the effective energy of 2.4 MeV. The lines are eyeguides. Also shown are the results of Eisen et al. normalized to the present results at ^{252}Cf (Eisen et al. 1980). The dotted line is a smooth curve through their data between ~1 and 4 MeV, and their point at 14 MeV is also shown. Several interesting facts emerge from Figure 1.2. First, it is clear that there is essentially no response at all for neutrons with energies below ~500 keV, as has been observed in the past. The neutron energy

threshold, however, depends upon the definition of "threshold," and on the conditions surrounding the irradiation and film processing. For example, a threshold criterion of 1/e may be used, i.e., the energy at which the response of the film to monoenergetic neutrons is a factor of 1/e below the response to a bare ^{252}Cf source. Thus, under ideal conditions (represented by track lengths $\geq$4 grains) the response is a factor of 1/e below the response to ^{252}Cf at $\sim$600 keV; for conditions which are probably more nearly representative of normal personnel monitoring (track lengths $\geq$7 grains), the 1/e point occurs at a higher value of $\sim$1.2 MeV. Moreover, the response of film to ^{252}Cf neutrons is itself only $\sim$1/e of its response to 14 MeV neutrons (Eisen et al. 1980).

1.4 CONCLUSIONS

Conditions inside reactor containment often represent a "worst case" for NTA film: temperatures >90°F,[a] and relative humidity >90%[a] (Endres et al. 1981). It would appear, then, that such films would be closer to the 7 grains than the 4 grains category, and that $\sim$1.2 MeV probably represents a realistic threshold for these films. Typically, approximately 1% of the neutron fluence inside reactor containment consists of neutrons with energies greater than $\sim$1.2 MeV[b] (Endres et al. 1981). In a few locations the fraction of neutrons greater than 1.2 MeV was as much as 3%, but in several locations it was less than 0.1% (Endres et al. 1981). If we accept 1.2 MeV as representing a "realistic" threshold for NTA film, then it is clear that it is totally unsuited for neutron dosimetry at a power reactor, and in fact, Endres states that NTA film did not record neutron doses even when the dose equivalent was greater than 1 rem. NTA film may be appropriate for environments where there are more high energy neutrons, but its use should be discouraged at power reactors.

1.5 ACKNOWLEDGEMENTS

We would like to thank Edward Abney of the Lexington-Blue Grass Army Depot for his help and cooperation in this project. In particular, we are grateful

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(a) GWR Endres, personal communication.
(b) Since the quality factor is higher for the high energy neutrons, this 1% of the fluence typically represents approximately 10% of the dose equivalent.

for his careful analysis of the film, for pointing out the significance of the number of grains and the difficulty in defining the number of grains required to consistently identify a track.

We should also like to thank Dr. O. A. Wasson and Mr. K. C. Duvall for running the Van de Graaff accelerator and skillfully measuring the neutron fluence for us.

2.0 THE RESPONSE OF NEUTRON PERSONNEL DOSIMETERS AND NEUTRON MEASURING INSTRUMENTS TO THE D_2O-MODERATED ^{252}Cf SOURCE

2.1 INTRODUCTION

In April 1980, we completed the construction of a 15-cm radius D_2O-moderated ^{252}Cf source, to be used for calibrating and testing neutron personnel dosimeters used, for example, at nuclear power reactors (Schwartz and Eisenhauer 1980). The reasoning behind that particular choice of moderator was discussed in detail in that reference, and will not be repeated here. Suffice it to say that it was felt that the spectrum so produced would be much closer to that found inside reactor containment than the ^{252}Cf fission neutron spectrum which had been used for dosimeter testing (Plato and Hudson 1978). (By "closer" is meant that integral responses such as average dose equivalent per unit fluence, albedo dosimeter response, etc., for the moderated source are significantly closer to those for the reactor containment spectrum).

Since completing the moderating assembly, we have tested several neutron dosimeters and neutron measuring instruments. It is the purpose of this report to present these measurements, compare them with our calculations, and to use these data as background to discuss the use of the moderated source for testing and calibration.

2.2 EXPERIMENTAL METHOD

All irradiations were made in pairs: first to a "bare" californium source, and then to the moderated source. The dosimeters used for this test consisted of three different types of albedo dosimeters, two different types of NTA film badges, one polycarbonate dosimeter (used without radiators) and one CR-39 dosimeter. In this report, the various suppliers of these dosimeters are referred to as "processors," i.e., Processor A, Processor B, etc. In each case, the processor furnished us with samples of his own dosimeters, read them after we irradiated them, and reported the results to us. The dosimeters were mounted on a 30 cm x 30 cm x 15 cm thick water phantom for the irradiations. Five or six dosimeters from each processor were irradiated to dose equivalents of, typically, 750 mrem, and the results averaged separately for each of the

two irradiations. (No obvious outliers were seen in any of the irradiations). Corrections were made for air scattering (Eisenhauer 1967) and room return.[a] In the case of the bare ^{252}Cf, corrections were also made for the scattering from the source support and capsule. All of these corrections were less than 1-1/2% except for room return. In the worst case, (bare californium irradiation of albedo dosimeters), the room return correction amounted to ∿10%.

In addition to the dosimeters, measurements were made of the responses of a 9-inch sphere remmeter, an Andersson-Braun remmeter, and a set of Bonner spheres. The instruments were mounted on low mass stands for the irradiations, and the same corrections were made as for the dosimeters. The californium source strengths were measured by Dr. V. Spiegel, NBS, to an accuracy of ±1-1/4%.

It is estimated that the overall uncertainty in the dose equivalent rate from the bare californium is ±3%, exclusive of any uncertainties in the fluence to dose equivalent conversion factors (Eisenhauer and Schwartz 1981). The uncertainty in the dose equivalent rate from the moderated source is, however, estimated to be ∿±15%. This is almost entirely due to uncertainties in the details of the spectrum shape.

Details of the irradiations are given in Tables 2.1, 2.2 and 2.3. Table 2.1 lists the source strengths for the two sources, typical distances used, and typical flux densities and dose equivalent rates. The source strengths are listed as of a particular date. The irradiations were actually made over a period of several months, and the source strengths were corrected for decay (0.07%/day) in all of the results reported.

In Table 2.2, the first three columns simply list the processor, the dosimeter type, and whether the source was bare or moderated. The fourth column, the "Free Field Dose Equivalent," gives the dose equivalent delivered to the dosimeter in the absence of background. That is, it is the dose equivalent delivered by the source itself, without taking account of the neutrons scattered by the walls of the room, the air, and so on. Column 5 are the observed results

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(a) Eisenhauer et al. to be published.

TABLE 2.1. Irradiation Set-Up

Source	Source Strength on 8/27/80 n/sec	Dosimeter Irradiation		Bonner Sphere Irradiation	
		Distance[(a)] cm	Free Field Dose Equivalent Rate, mrem/hr	Distance[(b)] cm	Free Field Flux Density $n/(cm^2\text{-sec})$
Bare ^{252}Cf	2.17×10^8	50	840	75	3.01×10^3
Moderated ^{252}Cf (15 cm D_2O)	1.74×10^9	65	880	75	2.42×10^4

(a) The distance is from the ^{252}Cf source (or center of the moderating sphere) to the center of the front face of the phantom.
(b) The distance is from the ^{252}Cf source (or center of the moderating sphere) to the center of the Bonner sphere.

TABLE 2.2. Dosimeter Irradiation Data

Processor	Dosimeter Type	^{252}Cf Source Configuration	Free Field Dose Equivalent rem	Observed Dose Equivalent rem	Corrected Response rem	Calibration Factor	Response Ratio
A	Albedo	Bare	1.00	1.07	0.95	0.95	12.4
		Moderated	1.01	12.72	11.92	11.8	
B	Albedo	Bare	0.981	0.038	0.034	0.035	12.9
		Moderated	0.841	0.401	0.377	0.448	
C	Albedo	Bare	0.922	1.017	0.924	1.00	22.4
		Moderated	0.870	20.7	19.5	22.4	
D	NTA Film	Bare	0.970	0.550	0.530	0.55	0.87
		Moderated	0.840	0.410	0.400	0.48	
E	NTA Film	Bare	1.00	110[(a)]	106[(a)]	106	1.10
		Moderated	1.00	120[(a)]	117[(a)]	117	
F	CR-39	Bare	0.775	0.546	0.531	0.68	1.66
		Moderated	0.484	0.570	0.553	1.14	
G	Poly-Carbonate	Bare	0.950	0.360	0.350	0.37	1.21
		Moderated	0.830	0.380	0.370	0.45	

(a) Tracks (mm^2-rem)

TABLE 2.3. Remmeter Irradiation Data

Type	Source	Free Field Dose Equivalent Rate mrem/hr	Observed Dose Equivalent Rate mrem/hr	Corrected Response mrem/hr	Calibration Factor	Response Ratio
9" sphere	Bare	956	752	728	0.762	1.83
	Moderated	1120	1609	1560	1.39	
Andersson-Braun	Bare	758	510	494	0.65	1.66
	Moderated	788	880	853	1.08	

reported to us by the processor, and Column 6 are these observed results corrected for the room scattered background mentioned above. The reponses are all in units of rem, except for NTA film processor E, who reported the results in tracks/(mm^2-rem). Column 7, "Calibration Factor" is the corrected response divided by the Free Field Dose Equivalent, and the final column, "Response Ratio," is calibration factor for the moderated source divided by the calibration factor for the bare source.

Table 2.3 gives the data for the remmeters; it is organized in the same way as Table 2.2.

2.3 RESULTS

For each type of dosimeter, the results are given in the tables below as the ratio of the response per unit dose equivalent to the moderated ^{252}Cf neutrons divided by the response per unit dose equivalent to bare californium neutrons. Where the neutron energy response function was known, we also calculated this ratio by folding the response function with the calculated neutron spectrum, and that value is simply listed as "calculated" (Ing and Cross 1977; Grundl and Eisenhauer 1975).

1. Albedo Dosimeter

Calculated	11.4
Processor A	12.4
Processor B	12.9
Processor C	22.4

2. NTA Film

Calculated	1.0
Processor D	0.87
Processor E	1.10

3. Polycarbonate

Processor G	1.21

4. CR-39

   Processor F 1.66

5. 9" Sphere Remmeter

   Calculated 1.54

   Measured 1.83

6. Andersson-Braun Remmeter

   Measured 1.66

7. Bonner Spheres

Since Bonner sphere sets are generally used as spectrometers rather than as remmeters, it was felt to be more appropriate to give the response ratios per unit fluence, rather than per unit dose equivalent. Accordingly, the table below gives the response ratios per unit fluence. The calculated values are based on Sanna's response functions (Sanna 1976).

Sphere Size	Calculated Response Ratio	Measured Response Ratio
12"	0.28	0.30
10"	0.34	0.33
8"	0.44	0.49
5"	1.00	1.07
3"	2.6	2.8
2"	6.7	6.3

2.4 DISCUSSION OF RESULTS

2.4.1 Albedo Dosimeter

The much higher readings of the albedo dosimeters for the moderated source can be understood from Figures 2.1 and 2.2 (Schwartz and Eisenhauer 1980). Figure 2.1 shows the over-lap between the bare californium neutron spectrum and the Hankins albedo dosimeter response function (Hankins 1977b). It is clear that the over-lap is not good, whereas the moderated spectrum (Figure 2.2) has a large fraction of its flux in the energy range where the albedo dosimeter

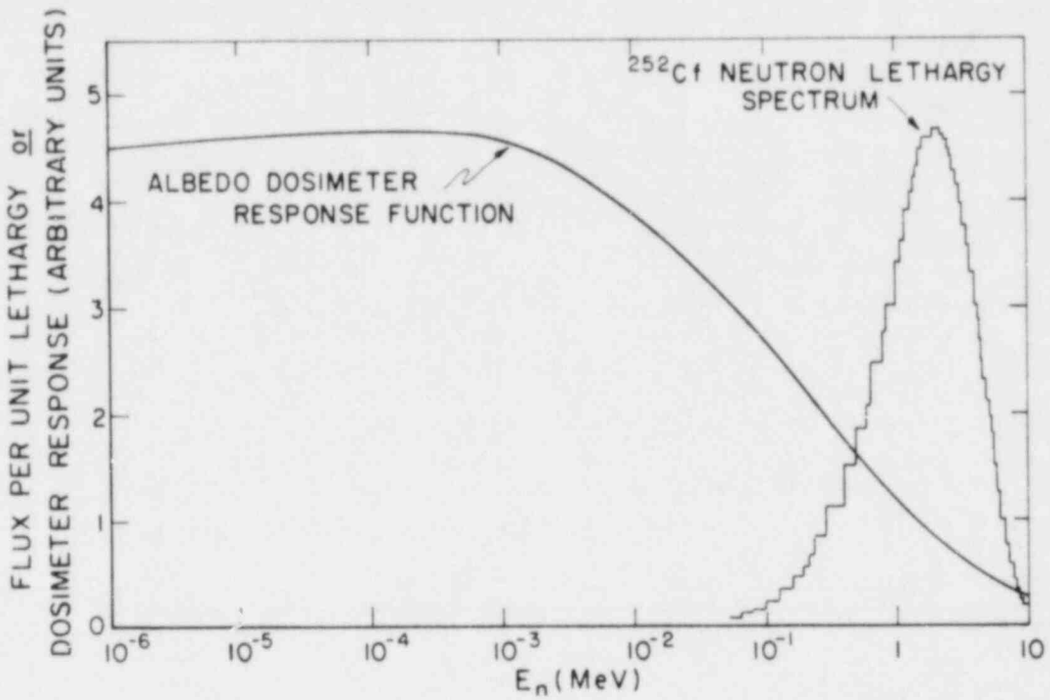

FIGURE 2.1. Flux Density per Unit Lethargy for Bare ^{252}Cf Source, and Albedo Dosimeter Response Function

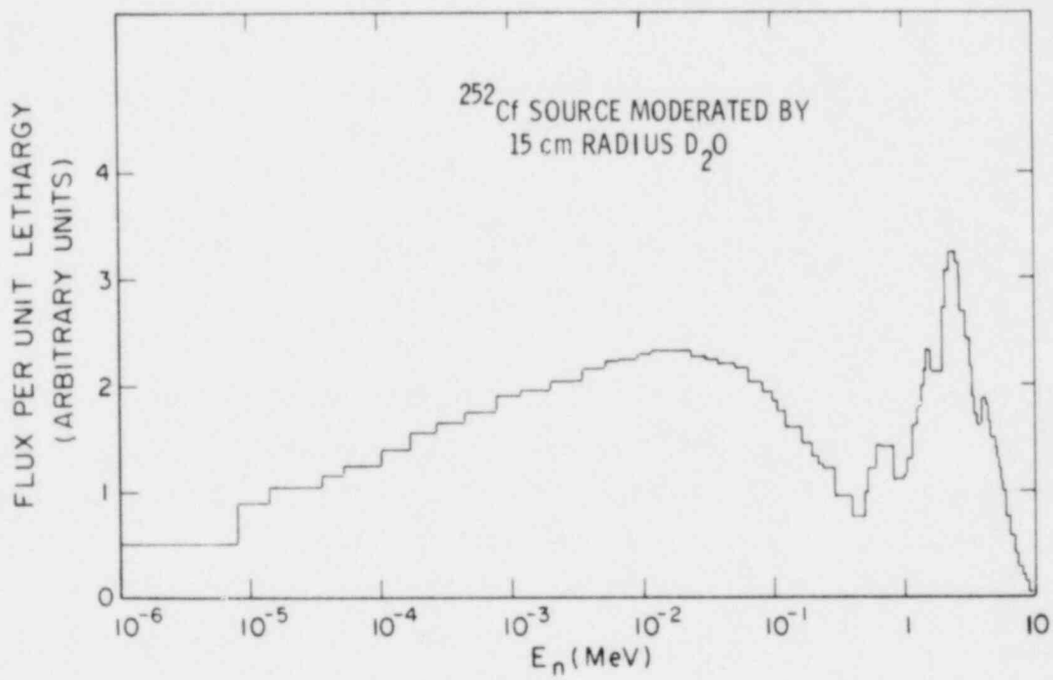

FIGURE 2.2. Flux Density per Unit Lethargy for ^{252}Cf Source Moderated by 15-cm Radius D_2O

has its highest response. In fact, one of the justifications for using this source is that its neutron spectrum "probes" the entire albedo dosimeter response function, rather than just its high energy tail.

Albedo dosimeters A and B give results in good agreement with the calculation. It is important to note that these two dosimeters use different methods for separating the gamma from the neutron dose equivalents. One uses a ^{6}LiF TLD chip for the neutron dose, with a ^{7}LiF chip to subtract the gammas; the other uses a natural LiF chip and reads the 250°C and 325°C glow curve peaks to distinguish neutrons from gammas. Clearly, then, the relative response does not depend on the type of TLD chip.

Dosimeter C also uses ^{6}LiF and ^{7}LiF chips, but the physical arrangement is different from the other two dosimeters. The fact that dosimeter C has a ratio ∿1.8 times as high as A or B suggests that the ratio of the low-and-intermediate energy response relative to high energy response is greater for this dosimeter than for the others. Although measurements of the energy dependence of a few different types of albedo dosimeters had shown them to be generally very similar, (Piesch and Burgkhardt 1978), it would not be too surprising to find differences in detail which might account for this increased ratio. It would clearly be worth-while to measure the neturon energy response function of dosimeter C.

2.4.2 NTA Film

Calculation and measurement both show that NTA film responds, on a rem basis, essentially identically to bare californium and moderated californium. In fact, even the distribution of track lengths on the film is approximately the same for the two sources.(a) While this result seems to be contrary to our intuitive expectations, it may be understood by remembering that for the moderated source ∿75% of the dose equivalent comes from neutrons above 1 MeV, even though 85% of the fluence is in the intermediate and low energy region. Figure 2.3 shows the dose equivalent spectrum for bare californium; Figure 2.4 shows the dose equivalent spectrum for the moderated source. Comparison of Figures 2.1 and 2.3 shows that transforming the plot of flux density per unit

(a) A. E. Abney, personal communication.

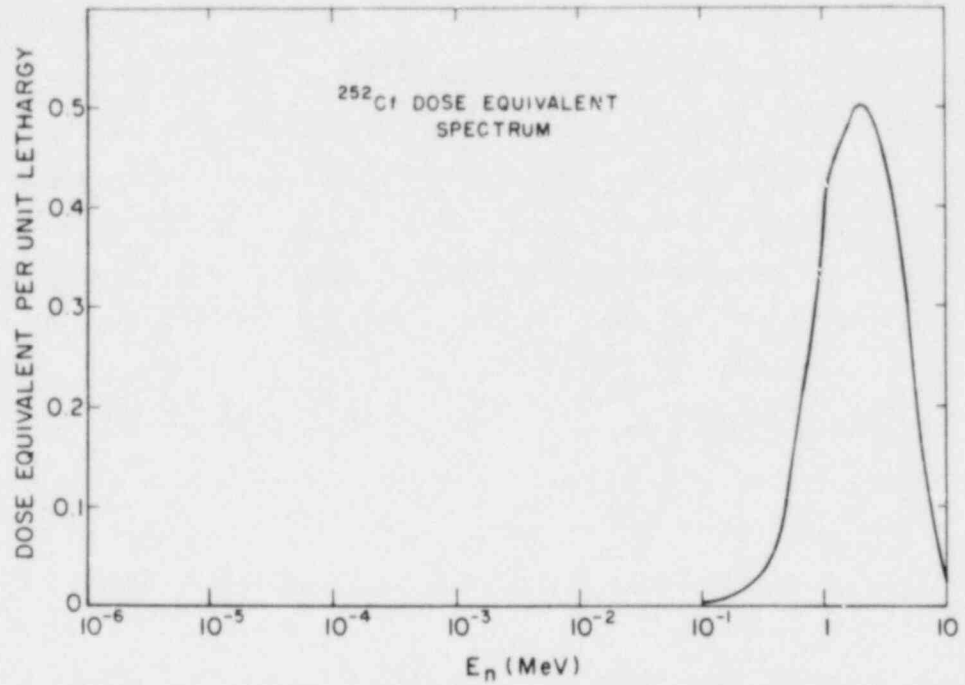

FIGURE 2.3. Dose Equivalent per Unit Lethargy for Bare ^{252}Cf Source

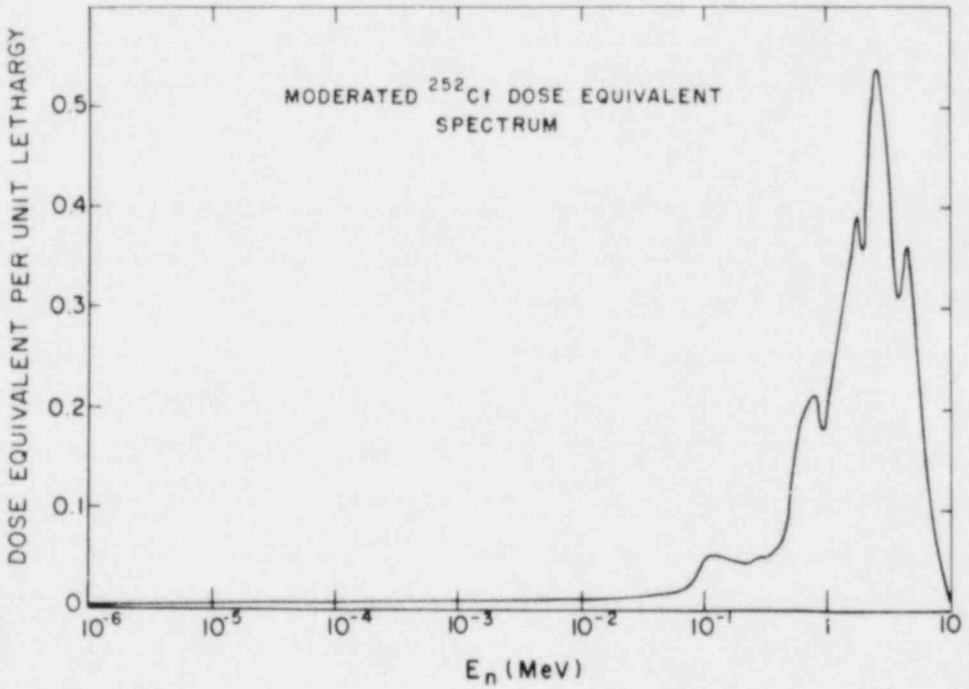

FIGURE 2.4. Dose Equivalent per Unit Lethargy for ^{252}Cf Source Moderated by 15-cm Radius D_2O

lethargy to dose equivalent per unit lethargy does not change the shape of the bare californium spectrum very much. Comparing Figures 2.2 and 2.4 shows, however, that the intermediate energy neutrons contribute only an insignificant amount to the dose equivalent. In fact, for both of these sources, approximately 75% of the dose equivalent comes from neutrons with energies above one MeV. Now, the energy threshold for NTA film may be considered to be somewhere between 0.6 and 1.2 MeV (see Section 1.3). The results should not be very sensitive to the details of the threshold: Figures 2.1 and 2.2 show that there are not many neutrons between 0.6 and 1.2 MeV, and that the two spectra are not greatly different above these energies. Figures 2.3 and 2.4 show that it is just these higher energies that produce most of the dose equivalent.

In short, NTA film responds only to the higher energy neutrons and it is just these higher energy neutrons that give most of the dose equivalent. There is not enough difference between these two spectra at neutron energies >600 keV for film to tell them apart, and thus, when normalized to the same dose equivalent, the two spectra will give the same result on NTA film.

2.4.3 Polycarbonate Track Etch

The neutron response of polycarbonate is roughly similar to that of NTA film (Eisen et al. 1980). Hence, approximately the same arguments can be made about polycarbonate as were made for NTA film, and we expect (and get) approximately the same results.

2.4.4 CR-39 Track Etch

The neutron response for CR-39 extends from high energies down to approximately 100 keV (Brackenbush et al. 1980). Hence, we expect CR-39 to have a higher response to the moderated spectrum, but not nearly as high as an albedo dosimeter.

2.4.5 Nine-Inch Sphere Remmeter

The agreement between the measured and calculated ratios is satisfactory in view of the uncertainties in the details of the moderated spectrum. The fact that the ratio is significantly greater than unity simply reflects the

well-known fact that the 9-inch sphere remmeter over-responds to intermediate energy neutrons (Hankins 1977b). We note that Hankins and Griffith observed an over-response of a factor of 1.82 for their 9-inch remmeter when used at the Alabama Power and Light Reactor (Hankins and Griffith 1978). Since their remmeter was calibrated to read correctly for bare californium, their over-response of 1.82 is directly comparable to our measured response ratio of 1.83.

2.4.6 Andersson-Braun Remmeter

The measured response ratio of 1.66 for the Andersson-Braun remmeter is not very different from the ratio of 1.82 for the 9-inch remmeter, and also reflects the known over-response of this instrument to intermediate energy neutrons (Schwartz 1978).

2.4.7 Bonner Spheres

The Bonner Sphere ratios show the expected behaviour: the ratios increase with decreasing sphere size, since the smaller spheres have a higher response to the softer spectrum. The agreement between calculated and measured ratios is quite good, and, in fact, may be somewhat fortuitous for the small spheres since there is some uncertainty in the room return correction for the 2-inch and 3-inch sizes.

2.5 ROLE OF THE MODERATED SOURCE IN TESTING AND CALIBRATING

One of our goals was to come as close as was practical to duplicating the measured spectrum inside the Alabama Power and Light Reactor (Hankins and Griffith 1978). This spectrum, shown in Figure 2.5, was believed to be typical of spectra in reactor containment. Recently several more reactor spectra measurements have become available (Endres et al. 1981). Some of these spectra are considerably softer than Alabama Power and Light, and some are harder. It is therefore questionable whether it is very useful to mock-up one particular spectrum, even if it were possible to do so. As shown in Figures 2.2, the 15 cm D_2O-moderated spectrum does contain many neutrons in the low and intermediate energies, as do most reactor spectra (Endres et al. 1981). However, unlike the reactor spectra, it has a significant component of neutrons above one MeV. One consequence of the higher energy component is that NTA film and

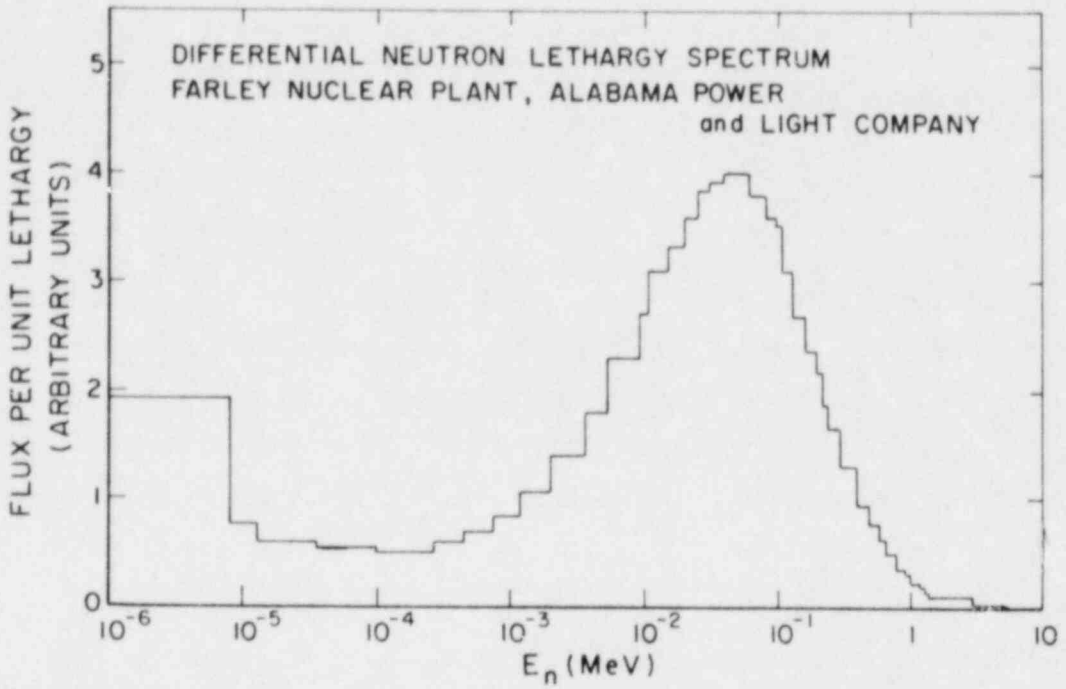

FIGURE 2.5. Flux Density per Unit Lethargy at the Farley Nuclear Power Plant, Alabama Power and Light Company

polycarbonate track etch dosimeters respond quite well to the moderated spectrum, but show little or no response to neutron spectra found inside reactor containment (Endres et al. 1981).

The high energy neutrons (the fission neutrons which have undergone little or no energy loss in traversing the moderator) will be present in any system relying on a small moderator to produce low energy neutrons from a fission source. The high energy peak is a prominent feature of D_2O-moderated spectra out to 50 cm (radius) of moderator, and the average energy of the high energy component ($E > 0.1$ MeV) actually increases with increasing moderator thickness (Ing and Cross 1977). This same general behavior occurs with the H_2O-moderated spectra (Ing and Cross 1975). If the radius of the moderating sphere is increased to 25 cm, the fraction of the dose equivalent due to the high energy component changes from 82% to 60% (Griffith et al. 1977). This would be more desirable, but the cost of this rather modest improvement would

be to increase the weight of the moderating sphere from about 40 pounds to almost 200 pounds, and the cost of the heavy water from $4,500 to over $20,000. This does not seem a desirable trade-off. Thus, the presence of the high energy component prevents the moderated source spectrum from exactly matching either the spectrum shown in Figure 2.5 or the other measured reactor spectra. Choosing a different, or a larger, moderator would not substantially improve the situation.

The moderated source does provide a neutron fluence spectrum which varies only by a factor of plus-or-minus two over the energy range 10 eV to 5 MeV, and thus should be well suited for testing any kind of neutron instrument. In particular, it is an excellent test source for albedo dosimeters since it tests the entire response function of this dosimeter rather than just the high energy tail.

It is clear that any dosimeter in use today should be calibrated for the particular working environment in which it is used. Too often, they are not so calibrated. To illustrate the disasterous results which can follow when a calibration source spectrum is very mismatched to the spectrum in the working environment and to the dosimeter response function, we repeat an earlier example (Schwartz and Eisenhauer 1980). An albedo dosimeter calibrated with the D_2O-moderated source and used at the Alabama Power and Light reactor will over-estimate the neutron dose equivalent by a factor of ~3; the same dosimeter used at the same reactor, but calibrated with a bare ^{252}Cf source would over-estimate the dose equivalent by a factor of 85.

2.6 CONCLUSION

We have constructed a 15-cm radius D_2O-moderated ^{252}Cf source for calibrating and testing neutron instruments and dosimeters used at nuclear power reactors. The measured calibration factors obtained for this source agree quite well with the predicted values.

The presence of a high energy neutron component (E_n >1 MeV) prevents the moderated spectrum from exactly matching the measured reactor spectra. The moderated spectrum is, however, very rich in low and intermediate energy neutrons,

as are the reactor spectra. The moderated spectrum is thus much closer to reactor spectra than is the bare californium source, and should therefore be a more valid source for testing and calibrating instruments to be used in reactor environments.

We therefore recommend that the moderated californium source be used to test and calibrate dosimeters and remmeters which are to be used at nuclear power reactors, or in any other environment where much of the exposure is to low and intermediate energy neutrons.

3.0 INVESTIGATION OF THE "EDGE EFFECT" IN ALBEDO NEUTRON DOSIMETER TESTING

3.1 INTRODUCTION

The neutrons moderated and back-scattered by the body form an important part of the response of an albedo dosimeter. For testing, albedo dosimeters must therefore be mounted on a "phantom" which may approximate the scattering in the human body. This immediately raises several questions. First, there is no universal agreement on the preferred phantom to be used for testing (Hankins 1980). Second, albedo dosimeters placed too close to the edge of the phantom will not "see" as many back-scattered neutrons as those placed near the center. It is the second of these problems, the "edge-effect," which we investigated. This problem was also the subject of a recent publication (Nash and Johnson 1980) and we shall compare our results and conclusions to those of Nash and Johnson. We had originally planned to investigate the edge effect for NTA film dosimeters, and for polycarbonate track etch dosimeters. These devices, however, respond only to high-energy neutrons and respond very slightly, if at all to the back-scattered neutrons. Hence, the edge effect is not a problem for these dosimeters and we therefore did not make any measurements with them.

3.2 EXPERIMENTAL METHOD

The phantom used was identical to that originally used for the NRC pilot study conducted by the University of Michigan: a 30 cm x 30 cm x 15 cm thick box made of 0.64 cm thick plexiglass and filled with water. Two irradiations were made: first to a "bare" ^{252}Cf source, and then to our D_2O-moderated californium source (see Section 2.2). The distance from the bare source to the front face of the phantom was 50 cm; the distance from the center of the moderated source to the front of the phantom was 65 cm. The irradiations in both cases were approximately one rem. Twelve dosimeters were mounted at a time, in the positions shown in Figure 3.1.

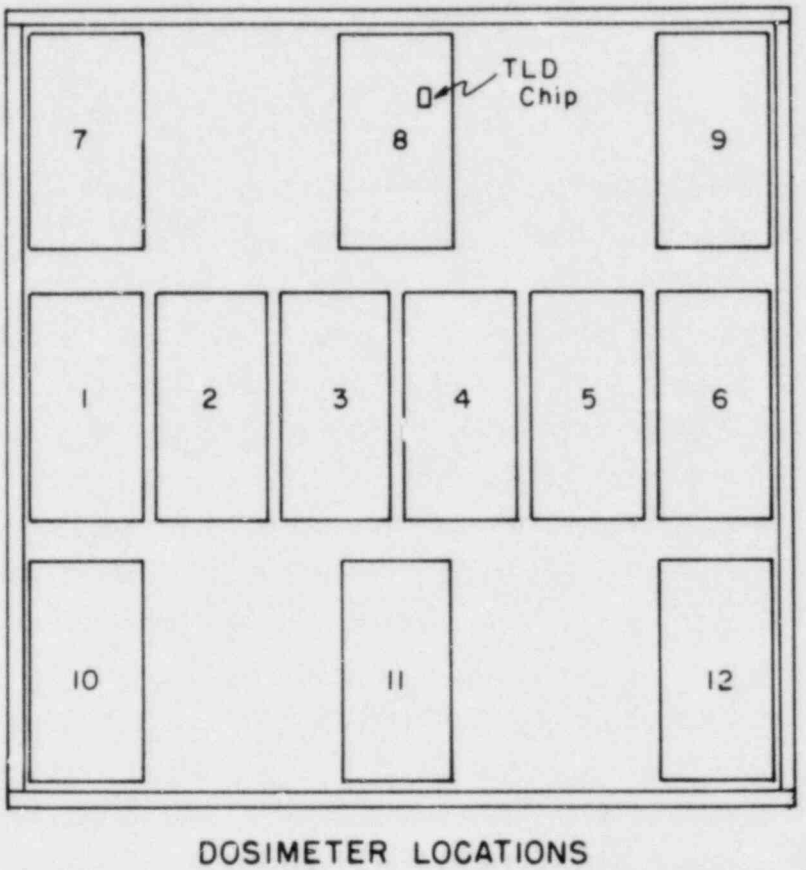

FIGURE 3.1. Dosimeter Locations on the 30 cm x 30 cm x 15 cm Thick Water Phantom. The location of the neutron responsive TLD chip is shown for position 8; the location is the same for the other positions.

3.3 RESULTS

The irradiation data are given in Table 3.1 and 3.2. For each of the two tables, the first column gives the badge number, which is associated with a particular dosimeter location, (see Figure 3.1). The second column gives the free field dose equivalent delivered at each of the dosimeter positions. "Free field" refers to the dose equivalent from source neutrons alone, not including background caused by room and air scattering. Although, for each source, all the dosimeters were irradiated simultaneously for the same length of time, the dose equivalents varied slightly due to the differing distances of the dosimeters from the source. This difference in $1/r^2$ (as much as 10%

TABLE 3.1. Bare ^{252}Cf Irradiation Data

Badge Number	Free Field Dose Equivalent mrem	Observed Dose Equivalent mrem	Corrected Response mrem	Calibration Factor[a]	Relative Dosimeter Response[b]
1	945	743	661	0.70	0.74
2	980	1144	1017	1.04	1.10
3	998	1046	930	0.93	0.98
4	998	1084	964	0.97	1.02
5	980	942	838	0.86	0.91
6	945	606	539	0.57	0.60
7	907	326	290	0.32	0.34
8	962	408	363	0.38	0.40
9	907	178	158	0.17	0.18
10	907	660	587	0.65	0.69
11	962	1085	965	1.00	1.05
12	907	462	411	0.45	0.47

(a) Calibration factor = corrected response/free field dose equivalent
(b) Relative dosimeter response = calibration factor/(average calibration factor from dosimeters 3 and 4).

for some positions) accounts for the variations in free field dose equivalents. The next column is the observed response of the dosimeter, and the fourth column is the corrected response. The "corrected response" corrects for the room and air scatter background mentioned above. It amounts to ∿6-1/2% for the moderated source at the NBS facility. The fifth column is the ratio of the corrected response to the free field dose equivalent. For the centrally located dosimeters (positions 3 and 4) this ratio is the calibration factor for this dosimeter and the source in question. The fact that this ratio is close to unity for the bare source (i.e., 0.93 and 0.97 for Positions 3 and 4, respectively) means that these dosimeters are correctly calibrated for ^{252}Cf. The corresponding ratio for moderated californium is ∿12 (12.03 and 11.55), which is close to the expected value, as explained elsewhere (Schwartz and Eisenhauer 1980). However, if this factor is ignored, the response measured for the moderated source

TABLE 3.2. Moderated ^{252}Cf Irradiation Data

Badge Number	Free Field Dose Equivalent mrem	Observed Dose Equivalent mrem	Corrected Response mrem	Calibration Factor[a]	Relative Dosimeter Response[b]
1	0.979	11.54	10.82	11.05	0.94
2	1.000	10.93	10.24	10.24	0.87
3	1.011	12.97	12.16	12.03	1.02
4	1.011	12.46	11.68	11.55	0.98
5	1.000	12.04	11.28	11.28	0.96
6	0.979	6.68	6.26	6.39	0.54
7	0.955	5.45	5.11	5.35	0.45
8	0.988	7.17	6.72	6.80	0.58
9	0.955	5.20	4.87	5.10	0.43
10	0.955	8.92	8.36	8.75	0.74
11	0.988	12.29	11.52	11.66	0.99
12	0.955	9.29	8.71	9.12	0.77

(a) Calibration factor = corrected reponse/free field dose equivalent
(b) Relative dosimeter response = calibration factor/(average calibration factor of dosimeters 3 and 4)

would overestimate the dose equivalent by a factor of 12. Finally, for each source, we average the calibration factor thus determined for the two central dosimeters (3 and 4) and take ratios for the calibration factors listed in column 5 to the average calibration factor to get the relative dosimeter response listed in the last column, and plotted in Figure 3.2.

For each irradiation, the relative dosimeter responses were analyzed as a function of the "effective distance" from the edge of the phantom to the TLD chip in the dosimeter. The parameter "effective distance" is defined as

$$\text{effective distance} \equiv \frac{1}{\sqrt{\frac{1}{\chi^2} + \frac{1}{\mu^2}}},$$

where χ and μ are the distances from the TLD chip to the nearer vertical and horizontal phantom edges, respectively.

Within the spread of the data, there was no significant difference between the results for the bare and for the moderated source irradiations. The two sets of data were therefore averaged, and the results plotted in Figure 3.2. The numbers on each point refer to the badge positions in Figure 3.1.

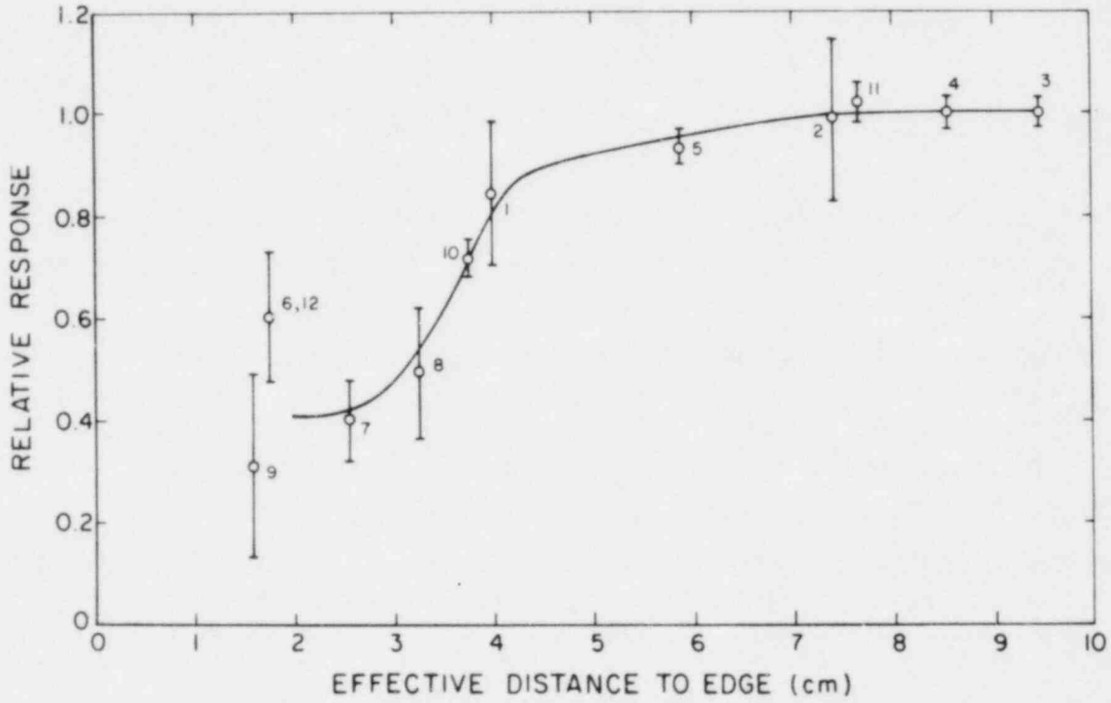

FIGURE 3.2. Relative Dosimeter Response for Different Positions on the Phantom

3.4 DISCUSSION OF RESULTS

Figure 3.2 shows the expected qualitative behaviour: the dosimeter response falls off as it gets too close to the edge of the phantom. Although there is scatter in the data, it is clear that positions 2, 3, 4 and 11 show little or no fall-off. (Note that since the TLD chip is in the upper portion of the dosimeter (see Figure 3.1), position 11 is satisfactory whereas position 8 is not).

An earlier experiment indicated a positional dependence for albedo response by mounting pairs of LiF TLD chips (one ^{6}LiF paired with one ^{7}LiF to subtract the signal due to photons) at one centimeter intervals along the horizontal and vertical axes of the phantom (Nash and Johnson 1980). In most other respects, the two experiments were very similar: the actual TLD material was the same, the phantom had the same dimensions, and both sets of TLD chips were covered with 0.51 mm thick cadmium on the side facing the source. (In our measurement, cadmium was part of the dosimeter; in the earlier measurement a 5 cm wide cadmium strip covered the TLD chips.)

The arrangement of LiF TLD chips allowed the experimenters to take data on a fine grid (i.e., 1 cm spacing) whereas in our measurement the TLD chips could be no closer than the 4.4 cm width of the dosimeter body. On the other hand, the earlier measurement would necessarily miss any effects caused by the plastic and other materials in the dosimeter body (although one expects any such effects to be fairly small). Hence, the two measurements should compliment each other.

The two sets of results are generally in satisfactory agreement. (Compare Figure 2 of Nash and Johnson 1980 with Figure 3.2 of the present report). The agreement indicates that the presence or absence of the dosimeter body makes little difference to the edge effect, and thus implies that results similar to the present results would be obtained for any other size or shape of TLD albedo dosimeter that responds primarily to thermal neutrons reflected from the phantom. Specifically, both sets of data show that the relative response is greater than 0.95 for effective distances from the edge greater than 7 cm, and that the relative response is down to 0.8 for an effective distance from the edge of 4 cm.

3.5 CONCLUSIONS

Since our results, and the earlier results, both indicate that for effective distances from the edge greater than 7 cm, the relative response is greater than 0.95, we would recommend that in future testing the dosimeter be positioned on the phantom so that the TLD chip is at an effective distance

from the edge greater than 7 cm. We note that the latest (as yet unpublished) Health Physics Society Standard for Testing Personnel Dosimetry Performance specifies that the TLD chip be placed no closer than 10 cm from the phantom edge. In the case of a dosimeter mounted near a corner of the phantom, 10 cm from both a horizontal and a vertical edge, the effective distance from the edge is 7 cm. Hence, our recommendation is consistent with that of the Health Physics Society Standard. In cases where the location of the TLD chip inside the dosimeter is not known, no part of the dosimeter should be less than the 7 cm effective distance from the edge of the phantom.

Finally, we note that the phantom size in the new Health Physics Society Standard is specified as 40 cm x 40 cm x 15 cm thick, rather than 30 cm x 30 cm. Nash and Johnson observed discrepant results when the TLD chip in their dosimeter was 10.25 cm from the center of the 30 cm x 30 cm phantom; i.e., 4.75 cm from the edge. With the larger phantom size, 10.25 cm from the center would still be 9.75 cm from the edge, and thus there should be no edge effect problem if the same dosimeter placement were to be used on the larger phantom (Nash and Johnson 1980).

4.0 DOSIMETER IRRADIATIONS INSIDE CONTAINMENT OF COMMERCIAL NUCLEAR POWER PLANTS

4.1 INTRODUCTION

This study of personnel neutron dosimeter response is part of a larger program being conducted at PNL to characterize neutron radiation fields and neutron dosimeter response to neutrons encountered at commercial nuclear power plants. The study is designed to provide a data base for the decision as to which types of neutron dosimeters may be used to monitor personnel neutron dose equivalent at nuclear power plants.

This report was prepared after a series of dosimeter irradiations were conducted to determine the response and precision of several types of neutron dosimeters. These measurements are a continuation of studies conducted in FY-79 and FY-80 to characterize neutron radiation fields at both PWR and BWR power plants.

Several types of thermoluminescence (TLD)-albedo and track etch neutron dosimeters (including polycarbonate track etch films used in conjunction with (n, α) radiators) show promise of accurate neutron dosimetry at commercial nuclear power plants. These dosimeters may be used individually or in combination to cover the neutron energy spectra found at the plants. There are also types of dosimeters which do not respond to the neutron spectra found at nuclear power plants, namely, nuclear track emulsion (NTA) and polycarbonate track etch films which do not employ (n, α) radiators.

4.2 EXPERIMENTAL METHOD

4.2.1 Irradiation Conditions

The nuclear power plants at which irradiations were performed are designated as Site E (BWR), Site G (PWR) and Site I (PWR). All irradiations were performed in locations where routine entry is made and while the reactors were at 100% power. The dosimeters were irradiated at two locations inside the BWR plant both of which were near sample-line pipe penetrations through the

biological shield. Since dose equivalent rates were on the order of a millirem per hour (mrem/hr) at both locations, long irradiation times were required. Dosimeters were irradiated at four locations inside containment of each of the PWR plants. The neutron fields at these locations in each of the three reactors were characterized during an earlier part of this study (Endres et al. 1981).

All the dosimeters were placed centrally on a 37 x 37 x 18 cm thick water-filled phantom. Since five dosimeters of each type were irradiated together to improve the precision of measurement, there was only enough space on the phantom for four types of dosimeters at a time. Hence, two sets of irradiations had to be performed at each location in order to include all the dosimeters.

After the irradiations were made, the dosimeters were mailed back to the processors for analyses. The dosimeter results shown in Table 4.1 are the average and one standard deviation of the processor-supplied results for each group of five dosimeters. The results of individual dosimeters for each particular irradiation are listed in the Appendix.

4.2.2 Personnel Neutron Dosimeter Descriptions

The irradiations of NTA film inside reactor containment performed over the last four years have failed to produce a positive response of personnel neutron dose equivalent. A general statement is made to this effect rather than to provide a table showing a lot of M's representing "minimal" detection. The origin of the use of "minimal" is unknown, but it means that the dose equivalent was below the limit of detection for the dosimeter.

Each type of dosimeter used in this study is described briefly below.

Vendor A

Vendor A dosimeters are a special type of TLD-albedo system which makes use of a deep trap in natural LiF to determine fast neutron dose equivalent. The dosimeters always respond to neutrons in reactor spectra, but the response needs to be corrected by factors between 1 and 3.5 in order to convert to dose equivalent. The results in Table 4.1 for Vendor A are uncorrected reader units as a calibration factor was not supplied to the vendor for these irradiations.

TABLE 4.1. Neutron Dosimeter Response for Irradiations Inside Reactor Containment

Site	Location	Integrated Dose Equivalent SNOOPY mrem	Integrated Dose Equivalent TEPC mrem	Vendor A[a] Counts	Vendor B TLD D_2O:^{252}Cf mrem	Vendor B Bare-^{252}Cf mrem	Vendor B CR-39 mrem	Vendor B Poly-carbonate mrem	Vendor C mrem	Vendor D[b] Counts	Vendor E mrem	HMPD mrem	LLL[c] mrem
E	First Irradiation												
	1x-29	26	28	48(5)	15(1)	316(28)	<10	<10	55(26)	65(7)	13(6)		
	3x-29	15	19	28(4)	9(1)	189(22)	<10	<10	8(10)	40(0)	13(6)		
	Second Irradiation												
	1x-29	32	30							96(15)		102(11)	35(15)
	3x-29	15	20							68(4)		39(?)	16(1)
G	First Irradiation												
	2	40	24		27(1)	575(22)	<10	<10	10(8)	140(7)		164(11)	
	3	47	58		46(2)	979(31)	<10	<10	1(3)	182(8)		290(16)	
	9	300	350		297(31)	6352(668)	<10	<10	275(28)	1102(13)		1662(302)	
	15	350	580		521(21)	11167(444)	4(9)	<10	510(76)	1212(18)		2702(172)	
	Second Irradiation												
	2	220	130	330(42)							157(32)		135(6)
	3	260	320	546(91)							193(32)		225(6)
	9	1700	2000	3688(707)							1100(416)		1433(138)
	15	2000	3300	8050(1509)							3280(382)		2390(50)
I	First Irradiation												
	4	100	22	260(37)	97(3)	2079(55)	4(9)	<10			103(15)		110(0)
	8	160	33	298(44)	148(6)	3170(134)	4(9)	<10			115(21)		160(10)
	10	1200	170	1224(169)	725(29)	15528(625)	13(18)	<10			700(406)		713(31)
	12A	310	67	528(47)	271(29)	5813(559)	<10	<10			405(50)		270(10)
	Second Irradiation												
	4	110	22						105(3)	562(31)		538(107)	
	8	150	32						160(12)	752(31)		870(106)	
	10	1300	200						610(56)	1260(20)		3968(744)	
	12A	370	80						355(11)	1148(13)		1712(432)	

(a) Values are given as net neutron response in units of counts or reader units. Fast neutron dose equivalent cannot be evaluated from these results.
(b) Values are given as mR equivalent. Fast neutron dose equivalent cannot be evaluated from these results.
(c) Corrected for spectral response using the 3-inch to 9-inch sphere ratio technique.
NOTE: Numbers in parentheses indicate one standard deviation.

Vendor B

Vendor B dosimeters were the most sophisticated used in this study. These dosimeters contain a lithium fluoride (LiF) TLD-albedo system and two types of track etch film (polycarbonate film which does not make use of an (n, α) radiator and CR-39 film). The TLD-albedo system had good precision, although only when the TLD responses were calibrated using a D_2O-moderated ^{252}Cf source did did Vendor B achieve accurate results. The polycarbonate track etch film in this dosimeter did not detect neutrons for any of the irradiations. The CR-39 film responded low and erratically.

Parenthetically, CR-39 film is made of a monomer plastic which must be electrochemically etched in order to observe the tracks produced by neutron interactions. While the particular results for CR-39 in this study are discouraging, it must be noted that the neutron response of CR-39 may be improved. Presently one commercial dosimeter supplier is striving to improve both the manufacture and etch procedure of CR-39 film. Additionally, CR-39 may be used in conjunction with TLD-albedo systems to provide a dosimeter sensitive to neutrons with energies between 0.02 eV and tens of MeV's.

Vendor C

Vendor C dosimeters consist of a TLD-albedo system which responded with good precision to the neutrons found inside reactor containment. Vendor C uses its own technique for energy response correction, which seems to work in most cases within a factor of two. Some problems arose when the neutron dose equivalent was less than 50 mrem.

Vendor D

Vendor D uses a TLD-albedo dosimeter which is not designed to directly determine fast neutron dose equivalent. Results shown in the table are for thermal and fast neutrons in terms of "mR equivalent." These dosimeters do detect the neutrons and have good precision in most cases. Vendor D is in the process of developing a dosimeter which will evaluate fast neutron dose equivalent.

Vendor E

Vendor E supplied a dosimeter which employs a multielement polycarbonate track etch system with natural boron-loaded radiators. While this system is still in the development stage, it was adequately sensitive to neutrons found in containment; in fact, a version of the dosimeter which used boron radiators which had been enriched with ^{10}B saturated at all of the exposure locations. The results are rather random in some cases, but that is attributed to the fact that the system is still under development and there are still some problems with the boron radiator uniformity. In cases where the neutron dose equivalent was <50 mrem (Site E only) the dosimeter's accuracy was poor. More work needs to be done to better quantify the results at very low dose equivalent levels.

The Livermore (LLL) and Hanford (HMPD) dosimeters are both TLD-albedo systems and both have adequate sensitivity to neutrons found in reactor containment. The results in Table 4.1 for the LLL dosimeters have been corrected using the 9" to 3" sphere ratios, and these results show good agreement with the SNOOPY measurements. The HMPD results are uncorrected for spectral variations and are high for all irradiations compared to the SNOOPY measurements.

4.2.3 Instruments

The instruments used in this study include the tissue equivalent porportional counter (TEPC), SNOOPY, PNR-4®, Rascal® and the cutie pie (CP).

The TEPC is described in detail in an earlier report (Endres et al. 1981). Briefly it is a proportional counter made of Shonka (tissue equivalent) plastic filled with a methane-based tissue equivalent gas. The dose deposition distritution in the counter is analyzed by computer as a function of lineal energy transfer to determine an average quality factor and dose equivalent.

The SNOOPY, PNR-4 and Rascal are moderated boron-trifluoride (BF_3) counters whose responses (counts/sec) are proportional within a factor of five to the

®Both instruments made by Eberline Instruments, Santa Fe, New Mexico.

neutron dose equivalent rate (mrem/hr) irrespective of neutron energy. The SNOOPY consists of cylindrical moderator around the BF_3 tube while the PNR-4 and Rascal are spherical in design. The difference between the PNR-4 and Rascal remmeter is in the metering of the instrument; the PNR-4 has a dual rate meter readout and the Rascal has a digital display.

The CP is a hand-held ionization chamber used to measure the exposure rate in air from x and gamma radiations.

Individual instrument measurements made at each particular location are listed in the Appendix.

4.3 DISCUSSION OF DOSIMETER RESPONSES

Table 4.1 summarizes the average responses of the dosimeters for each irradiation. These values are the average and one standard deviation of five dosimeters in each case. The integrated dose equivalent was determined by multiplying the dose equivalent rate measurements by the irradiation time. The best estimates of dose equivalent at this time are the SNOOPY measurement although the TEPC measurements are included for comparison.

4.3.1 Dosimeter Precision

The precision of a dosimeter is a measure of how often a dosimeter will give the same result, or be in a given range of results, for a given irradiation. It is defined as the ratio of the standard deviation to the average result. Using one standard deviation, the precision would estimate the spread of 67% of the dosimeter results around an average for a given irradiation. A precision of 10% for field irradiation is considered to be very good for most dosimeter systems. Based on that, all the dosimeter systems used in this study, except the CR-39 track etch film and polycarbonate track etch, generally exhibited excellent precision.

4.3.2 Dosimeter Accuracy

The accuracy of a dosimeter indicates how closely it measures a value relative to a standard (i.e. the SNOOPY in this case). For neutron dosimeters it is desirable to arrive at the identical dose equivalent as the instrument

measures (assuming the instrument approximates the true dose equivalent). Two methods may be used to achieve that end: 1) calibrating the dosimeter using a source which elicits a dosimeter response similar to the response elicited by the field irradiation, and 2) correcting the dosimeter response to account for differences between different radiation fields.

Only two dosimeters consistently yielded accurate responses: 1) Vendor B's TLD-albedo dosimeter which was calibrated using a D_2O-moderated ^{252}Cf source, and 2) LLL's TLD-albedo dosimeter which was corrected using the 9" to 3" sphere ratio technique. These two sytems exactly illustrate the two methods of achieving accuracy described above.

Generally, the other TLD-albedo dosimeters all responded high compared to the instrument measurements while the CR-39 and polycarbonate films (which did not use (n, α) radiators) failed to respond adequately. Vendor E's dosimeter which utilizes the boron-loaded raditors to produce (n, α) tracks in polycarbonate film responded similarly to the TLD-albedo systems.

4.4 CONCLUSIONS

The conclusions and recommendations from this portion of the study are listed by dosimeter type below.

4.4.1 NTA Film

The response of NTA film is discussed at length in Section 1 of this report. Additionally, we have not seen a positive response from NTA film irradiated inside reactor containment during the last four years. Hence, we recommend that NTA film not be used for personnel neutron dosimetry at commercial nuclear sites.

4.4.2 TLD Albedo

TLD-albedo dosimeters showed good precision and accuracy under two conditions: 1) calibration using a D_2O-moderated ^{252}Cf source, and 2) 9"/3" sphere response ratio corrections. We recommend the use of TLD-albedo systems inside containment at commercial nuclear sites using either of the two methods mentioned above to determine dose equivalent.

4.4.3 CR-39 Track Etch Film

The responses of CR-39 film used in this study were disappointingly inaccurate when the film responded at all. At this time we recommend that it not be used for personnel neutron dosimetry at commercial nuclear sites with the caveat that further developments in manufacturing and analysis (i.e. etching procedures and pit identification) may improve its response. CR-39 may be used presently in conjunction with a TLD-albedo system to provide a dosimeter which responds to neutrons with a wide range of energies.

4.4.4 Polycarbonate Track Etch Film

Two types of polycarbonate track etch dosimeters were evaluated in this study: 1) the Vendor B dosimeter which used polycarbonate by itself without an (n, α) radiator, and 2) the Vendor E dosimeter which uses multiple boron loaded radiators to produce (n, α) tracks in the film. Vendor B's polycarbonate film did not respond to neutrons inside containment, so this use of polycarbonate track etch film is not recommended for personnel neutron dosimetry inside containment. Vendor E's dosimeter had adequate sensitivity although the precision was erratic. While the dosimeter had some problems accurately assessing the small neutron dose equivalents (< 50 mrem) and the precision of measurements was as high as 58%, it performed well enough that it may be recommended for use inside reactor containment as a personnel neutron dosimeter over the other types of dosimeters which had poorer responses.

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APPENDIX

DOSIMETER DATA FROM IRRADIATION INSIDE CONTAINMENT OF NUCLEAR POWER PLANTS

TABLE A.1. Neutron Dosimeter Data for Site E, Location IX-29

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Counts	Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	0	26	7046	0	52					
			7047	0	42					
			7048	0	45					
			7049	0	43					
			7050							
B	0	26	46				14	295	<10	<10
			47				17	354	<10	<10
			48				15	319	<10	<10
			49				15	330	<10	<10
			50				13	283	<10	<10
C	0	26	E6							
			E7							
			E8	25		55				
			E9							
			E10							
D	0	28	1546	12		76				
			1547	13		102				
			1548	11		111				
			1549	14		111				
			1550	11		83				
E	0	26	E-3							Saturated
			E-4							Saturated
			N-4							15
			N-5							11
			N-6							10
HMPD	0	28	E6	39		96				
			E7	41		102				
			E8	41		106				
			E9	40		119				
			E10	38		106				
LLNL	0	28	235	47		36				
			236	45		32				
			237	38						

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.2. Neutron Dosimeter Data for Site E, Location 3X-29

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Counts	Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	0	15	7041	0	34					
			7042	0	29					
			7043	0	30					
			7044	0	30					
			7045	0	22					
B	0	15	41				7	153	<10	<10
			42				9	189	<10	<10
			43				9	201	<10	<10
			44				9	189	<10	<10
			45				10	212	<10	<10
C	0	15	E-1							
			E-2							
			E-3	18		8				
			E-4							
			E-5							
D	0	15	1541	11		69				
			1542	13		65				
			1543	12		58				
			1544	9		67				
			1545	12		69				
E	0	15	E-1							Saturated
			E-2							Saturated
			N-1							8
			N-2							12
			N-3							16
HMPD	0	15	E-1	32		35				
			E-2	34		43				
			E-3	38		28				
			E-4	38		50				
			E-5	38		50				
LLNL	0	15	232	47		16				
			233	52		17				
			234	39		16				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.3. Neutron Dosimeter Data for Site G, Location 2

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Counts	Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	170	220	7021	103	405					
			7022	146	299					
			7023	130	305					
			7024	137	312					
			7025	112	340					
B	30	40	21				28	590	<10	<10
			22				25	543	<10	<10
			23				26	566	<10	<10
			24				28	600	<10	<10
			25				27	578	<10	<10
C	30	40	G-1							
			G-2							
			G-3	29		10				
			G-4							
			G-5							
D	30	40	1521	32		133				
			1522	28		144				
			1523	35		140				
			1524	30		136				
			1525	34		147				
E	170	220	E1							Saturated
			E2							Saturated
			N1							119
			N2							165
			N3							176
HMPD	30	40	G1	28		162				
			G2	28		154				
			G3	29		162				
			G4	30		179				
			G5	32		166				
LLNL	170	220	216	162		137				
			217	171		131				
			218	165		132				
			219	166		137				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.4. Neutron Dosimeter Data for Site G, Location 3

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Counts	Neutron Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	140	260	7026	25	434					
			7027	21	555					
			7028	23	506					
			7029	15	548					
			7030	0	675					
B	25	47	26				44	944	<10	<10
			27				45	968	<10	<10
			28				45	968	<10	<10
			29				46	990	<10	<10
			30				48	1027	<10	<10
C	25	47	G6							
			G7							
			G8	22		1				
			G9							
			G10							
D	25	47	1526	15		185				
			1527	15		182				
			1528	16		187				
			1529	14		169				
			1530	14		182				
E	140	260	E3							Saturated
			E4							Saturated
			N4							225
			N5							167
			N6							180
HMPD	25	47	G6	16		274				
			G7	16		282				
			G8	15		282				
			G9	16		310				
			G10	17		301				
LLNL	140	260	220	93		228				
			221	96		220				
			222	95		229				
			223	95		224				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.5. Neutron Dosimeter Data for Site G, Location 9

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Counts	Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	430	1700	7031	0	2877					
			7032	0	4741					
			7033	381	3911					
			7034	212	3250					
			7035	0	3655					
B	75	300	31				337	7220	<10	<10
			32				318	6810	<10	<10
			33				293	6280	<10	<10
			34				273	5850	<10	<10
			35				262	5600	<10	<10
C	75	300	G11							
			G12							
			G13	64		275				
			G14							
			G15							
D	75	300	1531	62		1115				
			1532	66		1099				
			1533	59		1105				
			1534	60		1094				
			1535	56		1088				
E	430	1700	E5							Saturated
			E6							Saturated
			N7							986
			N8							1562
			N9							747
HMPD	75	300	G11	67		1720				
			G12	61		1888				
			G13	60		1333				
			G14	56		1995				
			G15	52		1368				
LLNL	430	1700	224	416		1600				
			225	399		1490				
			226	387		1340				
			227	373		1300				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.6. Neutron Dosimeter Data for Site G, Location 15

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Response, Counts	Neutron Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	600	2000	7036	0	7155					
			7037	0	7901					
			7038	0	8285					
			7039	0	10439					
			7040	445	6456					
B	110	350	36				496	10630	<10	<10
			37				518	11140	20	<10
			38				512	10970	<10	<10
			39				528	11300	<10	<10
			40				553	11830	<10	<10
C	110	350	G16							
			G17							
			G18	84		510				
			G19							
			G20							
D	110	350	1536	81		1203				
			1537	87		1201				
			1538	91		1199				
			1539	94		1219				
			1540	91		1243				
E	600	2000	E7							Saturated
			E8							Saturated
			N10							Saturated
			N11							3012
			N12							3552
HMPD	110	350	G16	93		2667				
			G17	88		2437				
			G18	100		2724				
			G19	96		2907				
			G20	98		2267				
LLNL	600	2000	28	553		2340				
			29	577		2380				
			30	580		2380				
			31	580		2460				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.7. Neutron Dosimeter Data for Site I, Location 4

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Counts	Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	65	100	7001	65	232					
			7002	15	311					
			7003	54	239					
			7004	100	294					
			7005	25	231					
B	65	100	1				97	2077	20	<10
			2				98	2089	<10	<10
			3				100	2148	<10	<10
			4				98	2089	<10	<10
			5				93	1994	<10	<10
C	66	110	IT1							
			IT2							
			IT3	45		105				
			IT4							
			IT5							
D	66	110	1501	57		555				
			1502	57		593				
			1503	59						
			1504	56						
			1505	57						
E	65	100	E1							Saturated
			E2							Saturated
			N1							115
			N2							93
			N3							95
HMPD	66	110	1	46		582				
			2	45		640				
			3	48		506				
			4	50		601				
			5	53		369				
LLNL	65	100	204	56		111				
			205	66		108				
			206	67		107				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.8. Neutron Dosimeter Data for Site I, Location 8

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Response, Counts	Neutron Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	50	160	7006	112	217					
			7007	27	328					
			7008	13	318					
			7009	35	306					
			7010	23	311					
B	50	160	6				127	2926	<10	<10
			7				150	3210	<10	<10
			8				150	3210	<10	<10
			9				151	3233	20	<10
			10				152	3245	10	<10
C	50	150	IT6							
			IT7							
			IT8	45		160				
			IT9							
			IT10							
D	50	150	1506	45		731				
			1507	43		737				
			1508	42		715				
			1509	44		781				
			1510	44		790				
E	50	160	E3							Saturated
			E4							Saturated
			N4							134
			N5							95
			N6							Saturated
HMPD	50	150	6	39		975				
			7	42		905				
			8	44		909				
			9	44		881				
			10	41		687				
LLNL	50	160	207	66		161				
			208	46		173				
			209	59		149				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.9. Neutron Dosimeter Data for Site I, Location 10

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Response, Counts	Neutron Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	240	1200	7011	340	1362					
			7012	603	1042					
			7013	262	1415					
			7014	338	1232					
			7015	505	1071					
B	240	1200	11				722	15470	<10	<10
			12				732	15680	35	<10
			13				755	16180	<10	<10
			14				738	15800	30	<10
			15				678	14510	<10	<10
C	270	1300	IT11							
			IT12							
			IT13	124		610				
			IT14							
			IT15							
D	270	1300	1511	103		1228				
			1512	109		1280				
			1513	105		1267				
			1514	108		1266				
			1515	108		1245				
E	240	1200	E5							Saturated
			E6							Saturated
			N7							386
			N8							1158
			N9							553
HMPD	270	1300	11	105		4132				
			12	104		4614				
			13	113		3885				
			14	105		4487				
			15	107		2746				
LLNL	240	1200	210	178		682				
			211	180		739				
			212	182		724				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.10. Neutron Dosimeter Data for Site I, Location 12A

Vendor	CP mR, γ	SNOOPY mrem,n	Dosimeter ID Number	Gamma Response, mR	Neutron Counts	Neutron Response, mrem	D_2O-Cf Response, mrem	Bare-Cf Response, mrem	CR-39 Response, mrem	Polycarbonate mrem
A	64	310	7016	208	508					
			7017	320	496					
			7018	177	517					
			7019	275	499					
			7020	154	609					
B	64	310	16				246	5275	<10	<10
			17				253	5428	<10	<10
			18				258	5534	<10	<10
			19				296	6340	<10	<10
			20				303	6490	<10	<10
C	76	370	IT16, IT17, IT18, IT19, IT20	94		355				
D	76	370	1516	85		1140				
			1517	86		1154				
			1518	84		1144				
			1519	90		1138				
			1520	84		1169				
E	64	310	E7							Saturated
			E8							Saturated
			N10							
			N11							370
			N12							442
HMPD	76	370	16	94		1274				
			17	83		1738				
			18	80		1287				
			19	81		2025				
			20	79		2237				
LLNL	64	310	213	83		259				
			214	73		279				
			215	75		271				

Note: <10 means less than the detection limit of the dosimeter.

TABLE A.11. Instrument Measurements of Dose Rate, Dose Equivalent Rate and 9"/3" Sphere Ratios

Location	Gamma Meas. CP mR/hr	Neutron Measurements TEPC mrem/hr	SNOOPY mrem/hr	PNR-4 mrem/hr	RASCAL mrem/hr	AVERAGE 9"/3" Ratio
Site E						
IX-29	0	1.3	1.2		1.6	0.22
3X-29	0	0.9	0.7		0.6	0.22
Site G						
2	10	7.8	13	8		0.10
3	8	19	15	13		0.10
9	25	120	100	75		0.12
15	35	190	120	100		0.12
Site I						
4	50	17	80	100	62	0.11
8	40	26	130	150	97	0.11
10	250	180	1300	1000	930	0.09
12-A	75	78	360	300	240	

Note: Site G 9:3 sphere ratios were used from an earlier trip as the Rascal malfunctioned during this time.

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R.B. Schwartz/NBS
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This report addresses problems associated with the calibration and use of personnel neutron dosimeters and monitoring instruments. Four particular items are addressed: 1) The threshold response of NTA film. NTA film is not recommended for use at reactors. 2) A discussion of dosimeter and remmeter calibrations performed using a D_2O-moderated ^{252}Cf source. Use of the moderated Cf source is recommended for calibrating dosimeters and instruments used at reactors. 3) The edge effect created by placing the neutron-sensitive elements of albedo dosimeters close to the phantom edge. It is recommended that dosimeters be no closer than 7 cm effective distance from the edge of the phantom on which they are irradiated. 4) The response of various personnel neutron dosimeters inside containment at nuclear power plants. It is recommended that dosimeters which demonstrate adequate sensitivity be used and be corrected for variations due to neutron energy spectral differences. Dosimeters that were found to be adequate were TLD-albedo dosimeters and polycarbonate track-etch dosimeters which utilized (n,α) radiators. NTA films, CR-39 films and polycarbonate films which did not use radiators are inadequate for personnel neutron dosimetry at nuclear power plants.

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