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Personal Nuclear Accident Dosimetry at Sandia National Laboratories

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PERSONAL NUCLEAR ACCIDENT DOSIMETRY

AT

SANDIA NATIONAL LABORATORIES

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ABSTRACT

DOE installations possessing sufficient quantities of fissile material to potentially constitute a critical mass, such that the excessive exposure of personnel to radiation from a nuclear accident is possible, are required to provide nuclear accident dosimetry services. This document describes the personal nuclear accident dosimeter (PNAD) used by SNL and prescribes methodologies to initially screen, and to process PNAD results. In addition, this report describes PNAD dosimetry results obtained during the Nuclear Accident Dosimeter Intercomparison Study (NAD23), held during 12-16 June 1995, at Los Alamos National Laboratories. Biases for reported neutron doses ranged from -6% to +36% with an average bias of +12%.

† currently with

ACKNOWLEDGMENT

The authors would like to thank Patrick J. Griffin for his invaluable consultation and advice in regards to the development of the spectrum weighted cross sections used in this report. We also thank Daniel J. Thompson for providing guidance and encouragement for the development of this report. In addition, we want to express our special appreciation to William Casson who organized and supervised the 23rd Nuclear Accident Dosimetry Intercomparison Study at Los Alamos National Laboratories, 12-16 June, 1995.

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SUMMARY

This report was prepared to describe the materials used in the SNL Personal Nuclear Accident Dosimeter (PNAD) and to document the procedures needed to process data obtained from the dosimeter. PNADs are used by the Criticality Dosimetry Program at Sandia National Laboratories to help assess neutron doses received by SNL employees or contractors who might be involved in a criticality accident. Results obtained during the Nuclear Accident Dosimeter Intercomparison Study (NAD23), held at Los Alamos National Laboratories during 12-15 June 1995 are also presented to give an indication of the accuracy and precision that can be obtained with this system, and as a verification of performance.

NOMENCLATURE

FNAD	Fixed Nuclear Accident Dosimeter
LANL	Los Alamos National Laboratory
NAD23	Nuclear Accident Intercomparison Study (23rd), held at Los Alamos National Laboratories, 12-15 June 1995
PNAD	Personal Nuclear Accident Dosimeter
rad	Radiation Absorbed Dose. This work uses the term "rad" to mean the dose absorbed by tissue.
dps	Disintegrations per second
g	gram
SHEBA-II	Solution High Energy Burst Assembly
SNL	Sandia National Laboratories
TLD	Thermoluminescent Dosimeter
μCi	microCurie
HPRR	Health Physics Research Reactor

PERSONAL NUCLEAR ACCIDENT DOSIMETRY AT SANDIA NATIONAL LABORATORIES

1. INTRODUCTION

A system of permanently mounted, or fixed nuclear accident dosimeters (FNADs) (Coats et al. 1990) and individually worn personal nuclear accident dosimeters (PNADs) are used at Sandia National Laboratories (SNL). The purpose of this dosimetry system is to allow the estimation of absorbed neutron dose received by individuals who may have been involved in a criticality accident at SNL controlled facilities. Both the FNAD and PNAD employ a variety of activation reactions which allow the total neutron fluence to be determined. By combining the total neutron fluence with information about the neutron energy spectrum, an estimate of an individual's absorbed neutron dose can be made. Individuals are also required to wear a Thermoluminescent Dosimeter (TLD) which is used to estimate absorbed dose due to photon radiation. An estimate of the total absorbed dose (in rads) can be obtained by summing the PNAD and TLD results.

Data obtained from the PNAD worn by each individual is considered to be the most reliable indicator of absorbed dose due to neutron radiation. Figure 1 shows an opened PNAD and the materials it contains. If PNAD data is not available, e.g., PNAD lost/damaged during the accident, estimates of neutron absorbed dose can be made using data obtained from the FNADs.

Documentation of the number of FNAD units, their locations, the effect of intervening shielding, and an analysis demonstrating performance limits is contained in a previous report (Coats, et al. 1990). Figures 2 and 3 contain pictures of an FNAD and the foils/TLD chips it contains.

In an effort to obtain dose assessments as efficiently as possible, dosimetry tables have been prepared (Appendix A) which list many representative types of neutron spectra that could be created during a criticality accident (Ing and Makra, 1978). From this table, the dosimetrist can obtain spectrum weighted cross sections and fluence-to-dose conversion factors. The dosimetrist must however, be supplied with sufficient information about the accident in order to select the most appropriate neutron spectra from the table. In addition to the data supplied by Ing and Makra, measured spectra for the Godiva and SHEBA reactors, which are operated at Los Alamos National Laboratories (LANL) have been included (Casson, 1995). Spectrum-weighted cross-section values and dose conversion factors have been calculated for the LANL spectra using the SAND II code (McElroy et al., 1967). The LANL spectra were included because future dosimetry Intercomparisons will most likely use the LANL Godiva and SHEBA neutron sources.

Both the FNAD and PNAD units have indefinite shelf lives and do not require any periodic calibrations. They are designed to remain "inert" until their foils are activated. If an accident occurs, the dosimeters must be retrieved within a few hours (2-8) and sent to a suitable counting facility. The Radiation Protection Sample Diagnostics Lab is the designated facility within SNL.

It is assumed that the reader is familiar with the basis physics of neutron activation and radiation dosimetry (Attix 1986, Knoll 1979 and IAEA Technical Report 211, 1982). This is done to eliminate the need for detailed derivations of equations and to focus the main effort of this work toward the description and performance of the criticality dosimeters used at Sandia National laboratories.

2. DESCRIPTION OF DOSIMETERS

2.1 Personal Nuclear Accident Dosimeter (PNAD).

PNADs are designed to be worn by individuals who enter locations in which installed criticality alarm systems are required. Figure 1 shows a PNAD that has been opened. Each PNAD contains the following:

Table 1. Data Regarding PNAD Elements[a]

PNAD Element	Diameter (inches)	Thickness (inches)	Approximate Wt. (g) $\pm 1\sigma$	Purity (%)
NaF pellet[b]	0.500	0.100	0.406 ± 0.007	50% NaF by weight[e]
Cadmium[c] covered Copper foil	0.406	0.032	$0.610^{d} \pm 0.003$	99.9
Titanium foil	0.500	0.036	0.525 ± 0.004	99.9
Nickel foil	0.500	0.032	0.835 ± 0.006	99.9
Aluminum foil	0.500	0.032	0.271 ± 0.001	99.9
Indium foil	0.500	0.010	0.228 ± 0.003	99.9

[a] Monte Carlo calculations with MCNP were performed to determine the error in the fluence estimate introduced by stacking the Cu, Ti, Ni, and In activation foils. The error was less than 3% (Griffin, 1989)

[b] NaF pellet used to determine an individual's orientation when compared with blood Na activity.

[c] Thickness of cadmium cover is 30 mils.

[d] Weight of copper foil without the cadmium cover.

[e] See Appendix E for a detailed description of the manufacturing process and composition of the NaF pellet. The ^{23}Na mass is 27.4% of the pellet mass.

Each PNAD is a sealed unit. Dosimetry materials are removed by breaking the front cover away from the plastic body/holder. A pair of pliers and a strong screwdriver are

recommended for use during opening. The dosimetry materials should be removed and counted per instructions contained in Appendix C.

Table 2 shows the reactions which are used for dosimetry purposes (Griffin et al., 1993, Knoll, 1979). The two indium reactions shown in Table 2 have different uses. The $^{115}In(n,\gamma)^{116m}In$ reaction has a very large cross section for low energy neutrons but a short (54 m) half-life. This reaction produces a relatively large foil activity that dies away quite rapidly. This activity is most useful for determining if the wearer was involved in a criticality accident. Portable health physics survey equipment is used to "Quick Scan" the PNAD. The measured exposure rate (mR h^{-1}) and time since the criticality event are used with the nomogram in Appendix B to determine if the PNAD should be collected for further analysis. The second indium reaction, $^{115}In(n,n')^{115m}In$, has a longer half-life (4.36 h) and is used for dosimetry purposes.

Table 2. Dosimetry Reactions

Material	Reaction	Half-Life	E_γ (keV)	Gamma Yield (%)	Threshold (MeV)
Al	$^{27}Al(n,\alpha)^{24}Na$	15 h	1368.633	100.0	8
Ti	$^{47}Ti(n,p)^{47}Sc$	3.4 d	159.381	67.9	2
Ni	$^{58}Ni(n,p)^{58}Co$	70.9 d	810.775	99.4	3
Cu	$^{63}Cu(n,\gamma)^{64}Cu$	12.9 h	1345.77	0.47	Epithermal
In	$^{115}In(n,n')^{115m}In$	4.36 h	336.241	45.9	1
	$^{115}In(n,\gamma)^{116m}In$	54.4 m	1293.54	84.4	Thermal[b]
Au[a]	$^{197}Au(n,\gamma)^{198}Au$	64.8 h	411.80	95.5	Thermal
Na	$^{23}Na(n,\gamma)^{24}Na$	15 h	1368.633	100.0	Epithermal

[a] Gold foils used only in FNADs

[b] Not used for dosimetry purposes, but used for Quick Scan purposes, see Appendix B.

The indium foil in a PNAD cannot be immediately counted on laboratory equipment, even for a relatively small neutron dose of 10 rads (tissue). The short-lived ^{116m}In has so much initial activity that counting system dead time becomes a major problem. It was empirically determined that at least 4 to 8 hours of decay time was needed to allow accurate determination of the ^{115m}In gamma activity. The half-lives of all the other activation products are long enough that a 4 to 8 hour wait prior to counting will not significantly affect dosimetry results. Half-lives for all isotopes of interest are shown in Table 2.

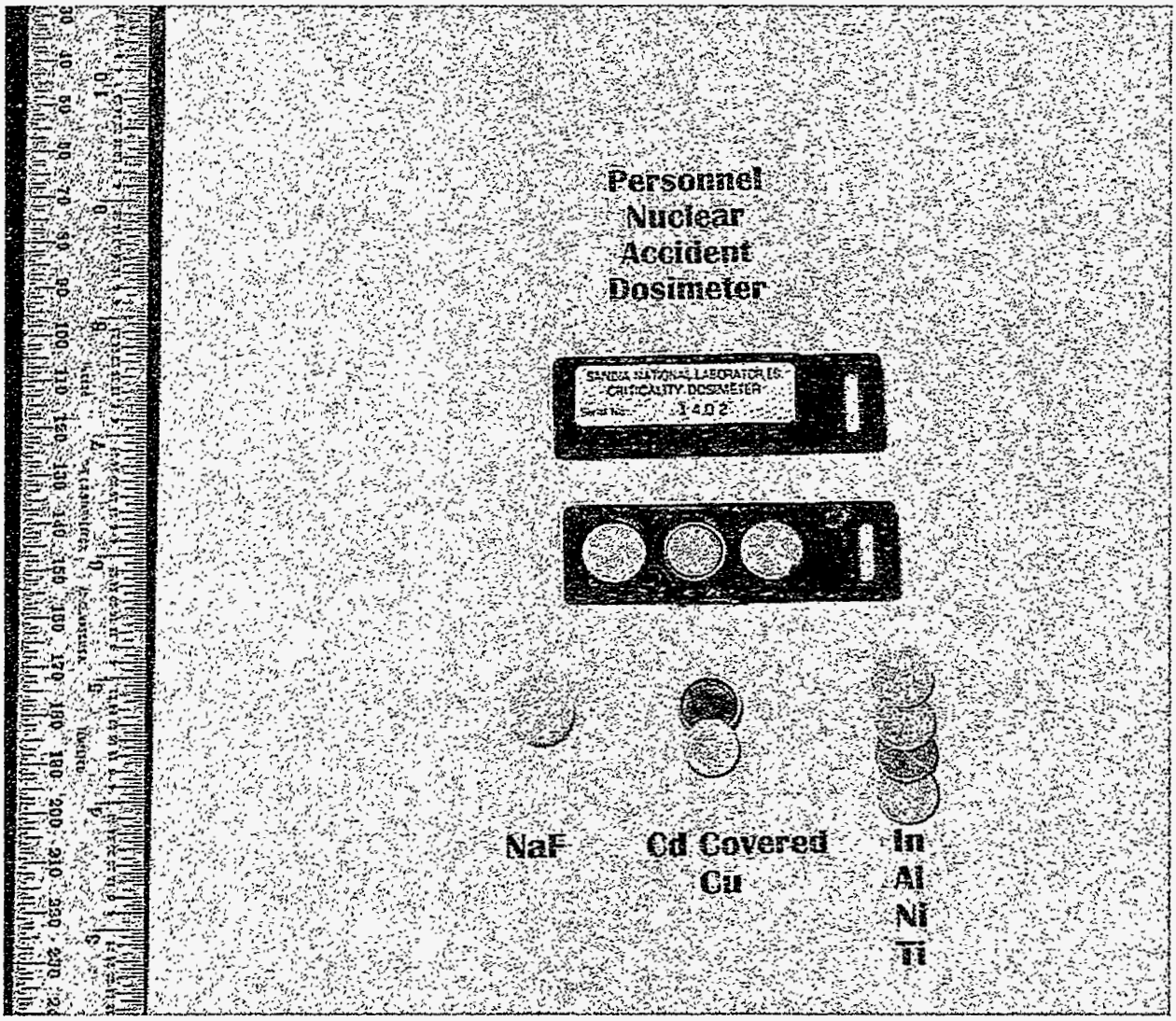

Figure 1. PNAD Components. The indium (In), aluminum (Al), nickel (Ni), titanium (Ti) and copper (Cu) materials are in the form of thin foils. The sodium (Na) material is in the form of a pellet consisting of 50% NaF and 50% microthene. The copper foil is covered by a 30 mil thick cadmium cup.

2.2 Fixed Nuclear Accident Dosimeter (FNAD).

The FNAD consists of two main components: (1) A removable portion containing the activation materials and three each TLD600/TLD700 ribbons[1] and (2) an eight inch polyethylene sphere. The removable portion consists of an activation foil packet mounted on one end of a polyethylene rod and the TLD600/TLD700 ribbons enclosed within the other end of the rod. The end containing the TLD material is inserted into the eight inch sphere. The activation foil packet remains outside of the sphere. The activation foils are: gold, aluminum, indium, titanium, nickel, and cadmium-covered copper. A single TLD700 chip is also included in the foil packet. The entire assembly is suspended from the ceiling or a wall. The FNAD is shown in Figures 2 and 3. Following a nuclear accident, the removable portion is retrieved and should be sent to the Radiation Protection Measurements Sample Diagnostics Laboratory for analysis. FNADs are positioned in accordance with written instructions (Coats et al. 1990; Petraglia 1991, Petraglia 1992). FNADs are not to be removed or re-positioned without permission of the SNL Criticality Safety Committee.

3. ANALYSIS OF DOSIMETER MATERIALS

3.1 General

The analysis of dosimeter materials requires two items of information.

- The first item is **induced activity per gram of material**. The PNAD must be opened and each foil weighed individually. The activity in each foil is then determined by using gamma spectral analysis. A high purity germanium detector and a multichannel analyzer are required.
- The second item is **knowledge of the neutron spectrum**. In theory, the combination of activities induced in each of the different foils will allow the incident neutron spectra to be deduced. In practice, this is a time consuming task that requires extensive analytical support. A more efficient method is to use a reasonable approximation of the incident neutron spectrum, for which spectrum-weighted cross-section values (for each foil) and fluence-to-dose conversion factors have already been calculated. Appendix A contains a table of representative spectra for which

[1] TLD/600 material is LiF enriched in ^{6}Li (approximately 95.6% ^{6}Li). The TLD/600 ribbons measure 0.125" by 0.125" and are 0.035" thick (square shape). TLD/700 material is LiF enriched in ^{7}Li (approximately 99.9% ^{7}Li). The TLD/700 ribbons measure 0.125" by 0.094" and are 0.035" thick (rectangular shape). Because the TL signal resulting from a neutron exposure fades at a rate of approximately 10% per month, a fade correction will have to be applied to TLDs read out more than two weeks after exposure (Coates et al. 1990)

spectrum-weighted cross-section values and fluence-to-dose conversion values have been calculated.

3.2Estimation of an Appropriate Neutron Criticality Spectra.

The determination of an appropriate neutron spectra is important for dosimetry purposes (Ing & Makra, 1978). The amount of activation per unit mass of foil material depends upon both the total fluence and energy distribution of the neutron spectrum. Because the precise determination of any particular neutron spectrum is quite difficult, a series of representative spectra have been pre-selected and evaluated. The dosimetrist should select the most appropriate spectra in Table A-1 and then use the resulting spectrum-weighted cross-section values and fluence-to-dose conversion factor as shown in Section 3.4 below. The selection of an appropriate spectrum is based upon the following general guidance:

- If the criticality accident involves a bare metal assembly and there was no shielding between the source and the individual concerned, select spectra No. 2 (Godiva).
- If the criticality accident involves a bare metal assembly and there was shielding between the source and the individual concerned, use one of the spectra numbered 22 through 30.
- If the accident was a solution criticality, select one of the spectra numbered 16 through 19. The selection should be based upon the diameter of the solution container at the time of the accident. Different diameters result in varying amounts of moderation.
- If detailed information regarding the physical situation immediately prior to the criticality event is available, and time permits, additional reference spectra can be consulted (Ing and Makra 1978, Griffin et al. 1990).

3.3Foil Activity.

Appendix C contains instructions for determining specific foil activity (μCi/g).

3.4Neutron Fluence Conversion Equations.

The equations listed in Table 3 yield estimates of the total neutron fluence (Φ, in units of n/cm^2) based upon measured specific foil activities (A, in units of μCi/g) and use of the appropriate spectrum-weighted cross-section value (σ, in units of barns) from Table A-1. It is recommended that the foil with the highest specific activity (i.e. the foil with the lowest counting error) be used for fluence determination. See Appendix D for an

example of a fluence calculation. The equation given for ^{197}Au yields an estimate of the thermal neutron fluence Φ_{Th}, and was derived using a value of 98.69 barns[2] for σ.

3.5Determination of Absorbed Dose.

The absorbed dose due to neutron radiation is calculated by multiplying the estimated neutron fluence Φ times the appropriate fluence-to-dose conversion factor ($\bar{D}$) from Table A-1. See Appendix D for an example of a dose calculation.

Table 3. Neutron Fluence Conversion Equations For PNAD

Reaction	E_γ used for Analysis	Fluence Conversion Equations*
$^{115}In(n,n')^{115m}In$	336 keV	$\Phi = \frac{A}{5.985x10^{-12}\sigma}$
$^{47}Ti(n,p)^{47}Sc$	159 keV	$\Phi = \frac{A}{6.076x10^{-14}\sigma}$
$^{58}Ni(n,p)^{58}Co$	810 keV)	$\Phi = \frac{A}{2.167x10^{-14}\sigma}$
$^{27}Al(n,\alpha)^{24}Na$	1369 keV	$\Phi = \frac{A}{7.707x10^{-12}\sigma}$
$^{63}Cu(n,\gamma)^{64}Cu$	1346 keV	$\Phi = \frac{A}{2.661x10^{-12}\sigma}$
$^{197}Au(n,\gamma)^{198}Au$	411 keV)	$\Phi_T = \frac{A}{2.454x10^{-11}}$
$^{23}Na(n,\gamma)^{24}Na$	1369 keV	$\Phi = \frac{A}{4.986x10^{-12}\sigma}$

* The derivation of these equations is presented in a previous report (Griffin et al. 1990)

[2] This value refers to the 2200 m/sec, E = 0.025 eV cross section $\Phi_{Th}(^{198}Au) = 98.69 \pm 0.14\%$.

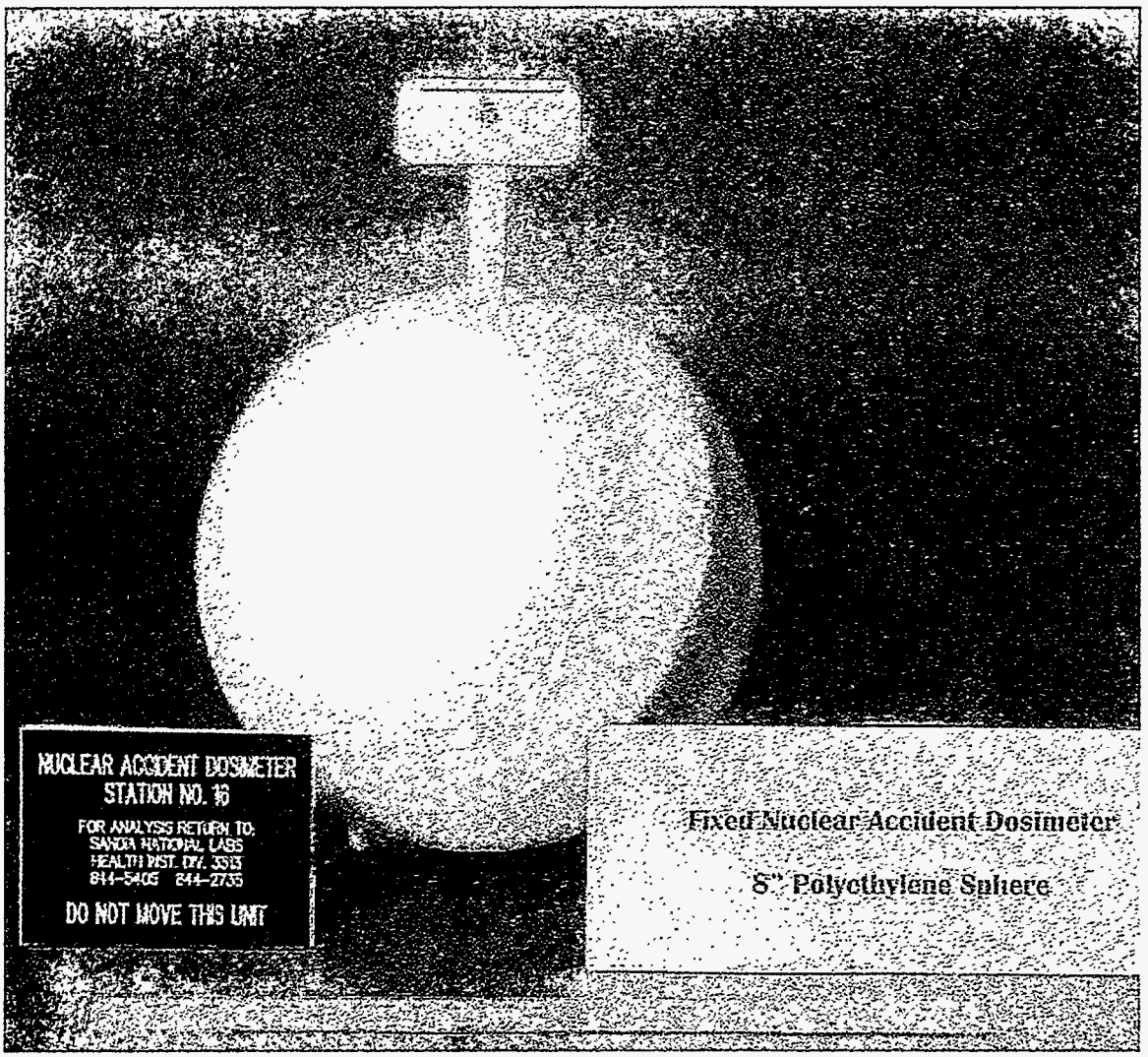

Figure 2. Fixed Nuclear Accident dosimeter, Assembled Configuration

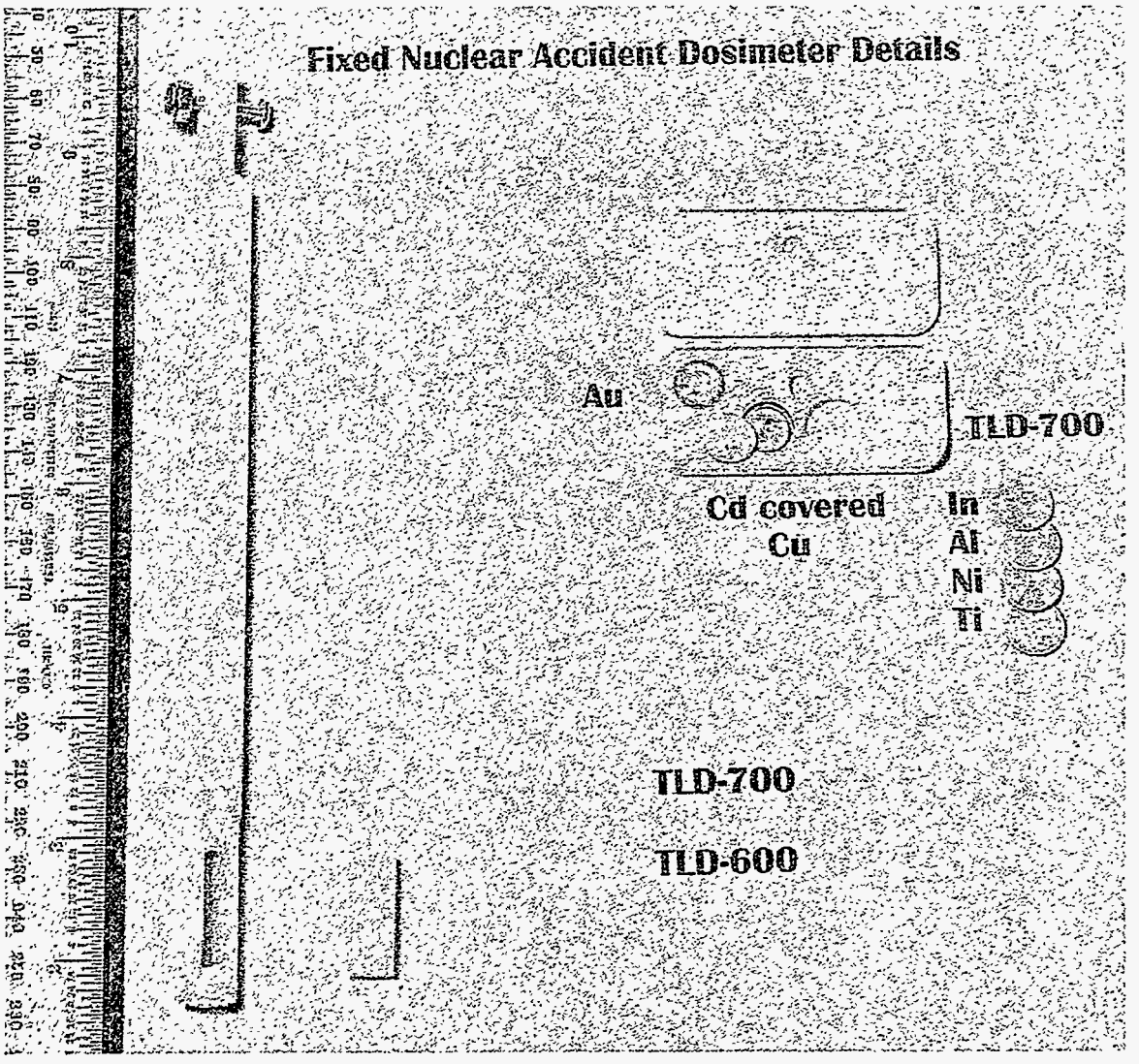

Figure 3. Components in the Removable Portion of a FNAD.

3.6Blood Sodium Analysis

When an unknown spectrum of neutrons enters the body, it is moderated and looses energy. As the neutrons loose energy, they pass through an energy region (approximately 3 keV) where ^{23}Na exhibits a high cross section for the following reaction: $^{23}Na(n,\gamma)^{24}Na$. The fraction of neutrons captured by ^{23}Na is relatively independent of spectrum shape. Although the amount of ^{24}Na created may be unequally distributed at the time of exposure, circulation of the blood evenly distributes this isotope in a short period of time. The ^{24}Na activity detected in a blood sample will be substantially independent of body orientation and neutron spectrum. The total number of ^{23}Na atoms contained in whole blood may differ considerably according to body weight, but the concentration is in most cases relatively constant at about 1.5 mg/ml of whole blood, or 3.2 mg/ml of blood plasma, for humans.

Blood sodium analysis can be performed by placing a 10 ml sample of whole-blood into a disposable covered Petri Dish with a diameter of 50 mm. The ^{24}Na activity of the whole-blood sample (μCi/ml) can be determined using a high purity germanium detector or equivalent. The volume of the whole-blood sample should be carefully measured and as close to 10 ml as possible. A 50 mm diameter petri dish is used to produce a sample geometry very similar to the geometry of standard NIST traceable calibration sources used in counting facilities. Blood samples should be obtained before any additional sodium is administered, either through food or medical treatment.

If time permits, the sodium concentration of the whole-blood sample may be determined because the dose conversion factors listed in Appendix I are keyed to a 1.5 mg/ml sodium concentration in whole blood.

Information regarding the orientation of the individual during the criticality accident can also be obtained by comparing the ratio of activities between blood sodium and the NaF tablet contained within the PNAD. The orientation of the individual can affect the final dose determination. See Appendix I for a more detailed discussion of blood sodium analysis and orientation of the PNAD during exposure

4.PERFORMANCE VERIFICATION

An assessment of PNAD performance was obtained through participation in the Nuclear Accident Dosimeter Intercomparison Study (NAD23), held at Los Alamos National Laboratory 12-16 June, 1995. Neutron sources consisted of the highly moderated Solution High Energy Burst Assembly (SHEBA-II) and the Godiva bare-metal assembly. The neutron spectra from the Godiva bare-metal assembly was also modified by a variety of separate shields (12 cm Lucite, 20 cm concrete and 13 cm steel). PNADs were exposed in a direct-line fashion with each source.

Appendix E contains a summary of the raw PNAD data obtained during NAD23.

Table 4 below summarizes the NAD23 performance results

Table 4. Comparison of Measured Dose to Reference

Neutron Source	Reference Neutron Dose (rad)	Reported Neutron Dose (rad)	Bias[a] (%)	Spectrum No. Used (from Table A)
SHEBA high power	108	114	6	16
SHEBA low power	11	12	9	16
SHEBA free-run	94	99	5	16
Bare Godiva	200	272	36	2
Godiva through 12 cm Lucite	26	32	23	27, 28[b]
Godiva through 20 cm concrete	34	32	-6	24
Godiva through 13 cm steel	N/A[c]	48	N/A	

[a] Bias = 100*[1-Reported/Reference)]

[b] Neutron absorbed dose calculated using interpolated values from spectrums #27 and #28.

[c] Reference dose not made available to NAD23 participants.

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APPENDIX A: DOSIMETRY DATA

Table A- 1. Dosimetry Data for PNAD Materials and Dose Conversion Factors for Selected Neutron Spectra

Spectrum-Averaged Cross Section (barns)	Spectrum Type								
	#1 WATT	#2 GODIVA (Taken from Ing and Makra, 1990)	#3 1/E + WATT	#4 ZPR-6 4Z+ Thermal	#5 FISSION through 10 cm Carbon	#6 FISSION through 90 cm Water	#7 20 cm Li-H Slab	#8 Graphite Test Lattice	#9 FERMI Reactor Test
$^{63}Cu(n,\gamma)^{64}Cu$	0.0107	0.0130	0.194	0.112	0.734	0.127	0.0955	0.184	0.231
$^{115}In(n,n')^{115m}In$	0.183	0.129	0.0412	0.0420	0.0903	0.138	0.185	0.00571	0.000436
$^{58}Ni\ (n,p)^{58}Co$	0.104	0.0649	0.0230	0.0233	0.0417	0.160	0.155	0.00231	0.000141
$^{47}Ti(n,p)^{47}Sc$	0.0176	0.0113	0.00391	0.00400	0.00739	0.0257	0.0253	0.000419	0.0000262
$^{27}Al(n,\alpha)^{24}Na$	0.000686	0.000468	0.000142	0.000168	0.000209	0.00509	0.00302	0.000010	0.00000140
$^{23}Na(n,\gamma)^{24}Na$	0.000278	0.000338	0.617	0.0239	0.0138	0.0324	0.0267	0.233	0.00780
$^{197}Au(n,\gamma)^{198}Au$	0.0774	0.103	173	29.3	19.6	46.6	1.44	82.7	11.8
$\overline{K}$ $\frac{nrad}{n/cm^2}$	2.81	2.42	0.652	0.985	1.67	2.03	2.22	0.141	0.255
$\overline{D}$ $\frac{nrad}{n/cm^2}$	3.30	2.77	0.762	1.05	1.90	2.43	2.66	0.160	0.229
$\overline{DE}$ $\frac{nrem}{n/cm^2}$	31.7	28.7	7.68	11.6	19.8	20.1	22.9	1.91	3.17
$\overline{D_\gamma}$ $\frac{nrad}{n/cm^2}$	0.215	0.237	0.222	0.294	0.263	0.247	0.231	0.118	0.352

Table A-1 (continued): **Dosimetry Data for PNAD Materials and Dose Conversion Factors for Selected Neutron Spectra**

Spectrum-Averaged Cross Section (barns)	Spectrum Type								
	#10 SPR-III 17" Leakage	#11 ACRR Central Cavity	#12 GODIVA Bare, Measured at LANL	#13 GODIVA with 12 cm Lucite, Measured at LANL	#14 GODIVA with 20 cm Concrete Measured at LANL	#15 SHEBA Bare, Measured at LANL	#16 H_2O Solution, 50 cm Diameter	#17 H_2O Solution, 30 cm Diameter	#18 H_2O Solution, 10 cm Diameter
$^{63}Cu(n,\gamma)^{64}Cu$	0.0206	0.148	0.101	0.143	0.172	0.0662	0.0994	0.100	0.0900
$^{115}In(n,n')^{115m}In$	0.110	0.0579	0.101	0.0728	0.0437	0.0933	0.120	0.122	0.125
$^{58}Ni\ (n,p)^{58}Co$	0.0595	0.0256	0.0607	0.0364	0.0208	0.0681	0.0718	0.0729	0.0720
$^{47}Ti(n,p)^{47}Sc$	0.0101	0.00463	0.0103	0.00640	0.00374	0.0116	0.0121	0.0123	0.0122
$^{27}Al(n,\alpha)^{24}Na$	0.000386	0.000151	0.000693	0.000257	0.000284	0.00173	0.000561	0.000577	0.000504
$^{23}Na(n,\gamma)^{24}Na$	0.00203	0.0414	0.0191	0.0696	0.0549	0.0117	0.00441	0.00423	0.00384
$^{197}Au(n,\gamma)^{198}Au$	1.76	52.4	24.9	66.2	58.2	15.7	25.9	23.9	20.8
$\overline{K}$ $\frac{nrad}{n/cm^2}$	2.15	1.15	1.79	1.20	0.962	1.79	1.97	1.98	2.03
$\overline{D}$ $\frac{nrad}{n/cm^2}$	2.42	1.30	2.07	1.40	1.09	2.03	2.29	2.31	2.36
$\overline{DE}$ $\frac{nrem}{n/cm^2}$	25.2	13.8	20.7	14.0	11.8	20.2	22.4	22.5	23.1
$\overline{D_\gamma}$ $\frac{nrad}{n/cm^2}$	0.251	0.273	0.255	0.228	0.253	0.260	0.265	0.265	0.262

Table A-1(continued): Dosimetry Data for PNAD Materials and Dose Conversion Factors for Selected Neutron Spectra

Spectrum-Averaged Cross Section (barns)	Spectrum Type						
	#19 H_2O Solution, 5 cm Diameter	#20 SPRIII Central Cavity	#21 Cf-252 from NBS Compendium	#22 Fission Neutrons through Concrete (10 cm)[a]	#23 Fission Neutrons through Concrete (20 cm)[a]	#24 Fission Neutrons through Concrete (30 cm)[a]	#25 Fission Neutrons through Concrete (40 cm)[a]
$^{63}Cu(n,\gamma)^{64}Cu$	0.0742	0.0162	0.0104	0.0731	0.165	0.202	0.208
$^{115}In(n,n')^{115m}In$	0.131	0.108	0.188	0.119	0.0861	0.0737	0.0743
$^{58}Ni\ (n,p)^{58}Co$	0.0756	0.0589	0.166	0.0628	0.0443	0.0371	0.0367
$^{47}Ti(n,p)^{47}Sc$	0.0128	0.0101	0.0194	0.0108	0.00776	0.00656	0.00654
$^{27}Al(n,\alpha)^{24}Na$	0.000569	0.000365	0.00106	0.000310	0.000246	0.000202	0.000176
$^{23}Na(n,\gamma)^{24}Na$	0.00295	0.000457	0.000271	0.00275	0.00716	0.00927	0.00974
$^{197}Au(n,\gamma)^{198}Au$	12.5	0.170	0.0742	6.61	42.0	73.9	88.6
$\overline{K}$ $\frac{nrad}{n/cm^2}$	2.16	1.15	2.87	2.02	1.45	1.20	1.18
$\overline{D}$ $\frac{nrad}{n/cm^2}$	2.51	1.30	3.38	2.33	1.67	1.39	1.37
$\overline{DE}$ $\frac{nrem}{n/cm^2}$	24.5	13.8	31.9	23.3	16.8	14.0	13.7
$\overline{D_\gamma}$ $\frac{nrad}{n/cm^2}$	0.253	0.273	0.214	0.258	0.296	0.314	0.317

[a] Taken from page 88, Ing & Makra, 1978.

Table A-1(continued): **Dosimetry Data for PNAD Materials and Dose Conversion Factors for Selected Neutron Spectra**

Spectrum-Averaged Cross Section (barns)	Spectrum Type				
	#26 Fission Neutrons through Polyethylene (5 cm)[b]	#27 Fission Neutrons through Polyethylene (10 cm)[b]	#28 Fission Neutrons through Polyethylene (20 cm)[b]	#29 Fission Neutrons through Polyethylene (40 cm)[b]	#30 Fission Neutrons through Polyethylene (60 cm)[b]
$^{63}Cu(n,\gamma)^{64}Cu$	0.284	0.643	0.914	0.853	0.749
$^{115}In(n,n')^{115m}In$	0.110	0.0901	0.0872	0.105	0.116
$^{58}Ni\ (n,p)^{58}Co$	0.0720	0.0631	0.0687	0.0971	0.113
$^{47}Ti(n,p)^{47}Sc$	0.0119	0.0103	0.0110	0.0155	0.0184
$^{27}Al(n,\alpha)^{24}Na$	0.000639	0.000606	0.000873	0.00194	0.00312
$^{23}Na(n,\gamma)^{24}Na$	0.0267	0.0696	0.103	0.0961	0.0839
$^{197}Au(n,\gamma)^{198}Au$	30.2	44.3	45.4	40.7	37.7
$\overline{K}$ $\frac{nrad}{n/cm^2}$	1.80	1.46	1.38	1.62	1.79
$\overline{D}$ $\frac{nrad}{n/cm^2}$	2.09	1.70	1.62	1.92	2.13
$\overline{DE}$ $\frac{nrem}{n/cm^2}$	20.0	20.0	15.0	20.0	20.0
$\overline{D_\gamma}$ $\frac{nrad}{n/cm^2}$	0.247	0.218	0.187	0.184	0.193

[b] Taken from page 80, Ing & Makra, 1978

APPENDIX A

Table A-1(continued): Dosimetry Data for PNAD Materials and Dose Conversion Factors for Selected Neutron Spectra

Spectrum-Averaged Cross Section (barns)	Spectrum Type #31 Fission Neutrons through Iron (5 cm)[c]	#32 Fission Neutrons through Iron (10 cm)[c]	#33 Fission Neutrons through Iron (20 cm)[c]	#34 Fission Neutrons through Iron (30 cm)[c]	#35 Fission Neutrons through Iron (50 cm)[c]
$^{63}Cu(n,\gamma)^{64}Cu$	0.0131	0.0156	0.0221	0.0283	0.0453
$^{115}In(n,n')^{115m}In$	0.133	0.0947	0.0421	0.0178	0.00373
$^{58}Ni\ (n,p)^{58}Co$	0.0603	0.0357	0.0117	0.00368	0.000334
$^{47}Ti(n,p)^{47}Sc$	0.0107	0.00662	0.00233	0.000784	0.0000905
$^{27}Al(n,\alpha)^{24}Na$	0.000324	0.000150	0.0000466	0.0000117	5.92 E -8
$^{23}Na(n,\gamma)^{24}Na$	0.000320	0.000418	0.000650	0.000797	0.00136
$^{197}Au(n,\gamma)^{198}Au$	0.101	0.133	0.224	0.275	1.38
$\overline{K}$ $\frac{nrad}{n/cm^2}$	2.44	2.14	1.68	1.40	1.08
$\overline{D}$ $\frac{nrad}{n/cm^2}$	2.81	2.43	1.80	1.42	1.02
$\overline{DE}$ $\frac{nrem}{n/cm^2}$	29.2	26.8	22.0	18.4	13.7
$\overline{D_\gamma}$ $\frac{nrad}{n/cm^2}$	0.234	0.252	0.280	0.298	0.317

[c] Taken from page 100, Ing & Makra, 1978

APPENDIX B: QUICK-SCAN PROCEDURES FOR PNADS

Initial screening of personnel involved in a criticality accident, to determine whether significant exposures to neutron radiation occurred, can be performed using the following procedure. The screening (or Quick-Scan Procedure) is accomplished by checking the PNAD for induced activity using portable health physics survey equipment and the nomogram contained in Figure B-1.

When the indium foil in a PNAD is exposed to neutron radiation, two different reactions are of importance. The $^{115}In(n,n')^{115m}In$ reaction has an effective 1 MeV threshold and does not produce a level of activity that is easily measurable with portable survey equipment. The product of this reaction has a 4.5 h half-life and is, however, the most important isotope for laboratory analysis purposes. The second reaction, $^{115}In(n,\gamma)^{116m}In$, has a very large capture cross section for low energy neutrons. This reaction produces a relatively large foil activity that decays with a 54 min. half-life. It is the activity produced by this second reaction that is detectable with portable survey equipment. The significant gamma radiation associated with the decay of ^{116m}In is as follows:

Table B- 1. Significant Gamma Energies Associated With ^{116m}In

Reaction	Half-Life	Eγ (keV)	Yield (%)
$^{115}In(n,\gamma)^{116m}In$	54.4 m	1293.54	84.4
		1097.3	56.2
		818.7	11.5
		2112.1	15.5

The SAND II computer code and the Bare Godiva spectrum, supplied by LANL, were used to calculate an initial indium-foil activation. Both ^{115m}In and ^{116m}In reaction products were considered. Total foil activity (μCi) versus time was then calculated. A Specific Gamma Ray Constant of 0.006295 mrem/h/μCi @ 1 m was used for ^{115m}In, and a value of 12.4577 mrem/h/μCi @ 1 m was used for ^{116m}In (Shleien, Table 6.1.2, 1992). Exposure rate values were then calculated assuming a point-source configuration with a source-to-detector distance of 0.01 m. This source-to-detector distance corresponds to a configuration in which the long axis of the PNAD is aligned with the long axis of an HP270 probe and the two items placed in physical contact. The movable shield on the HP270 probe should be in the "closed" position. The results of the SAND II calculations were used to develop the nomogram shown in figure B-1.

The nomogram shown in Figure B-1 should be used to determine if the individual wearing the PNAD was exposed to any significant amount (> 10 rad) of neutron radiation. The measured PNAD activity (mR h^{-1}) is located on the "Y" axis, and the time since the criticality event (min.) is located on the "X" axis. These two data items determine a reference point on the nomogram. The "isodose line" which passes closest to this reference point is an estimate of the neutron dose received by the wearer and is used for screening purposes. When the dose estimate is equal to or greater than 10 rad, the PNAD should be forwarded to the Sample Analysis Laboratory to allow for a more accurate dose determination.

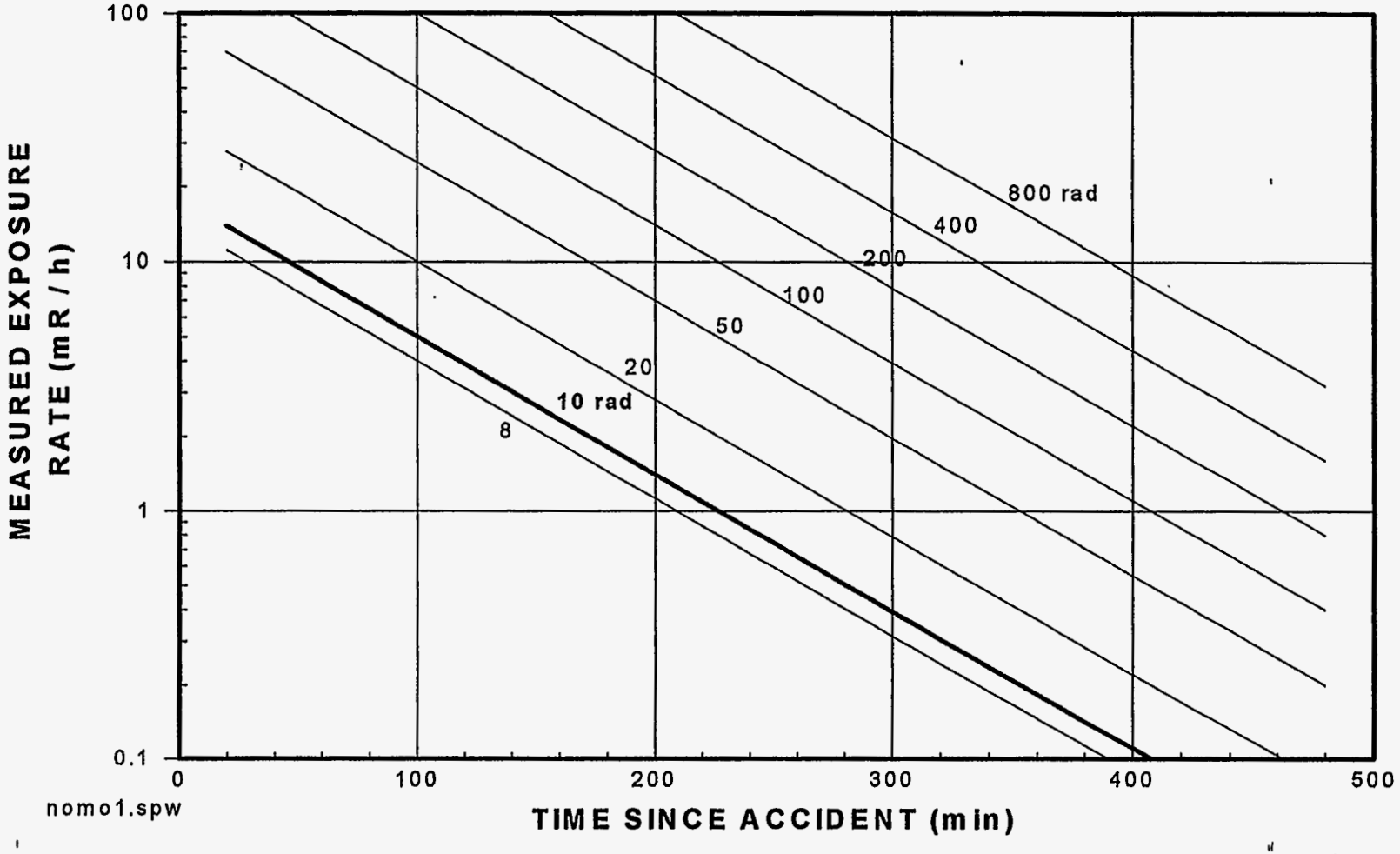

Figure B-1. Nomogram for "PNAD Quick-Scan" measurements. The X axis represents the amount of time that has elapsed since the criticality event. The Y axis represents the measured exposure rate from the PNAD. Exposure rate measurements should be made using a HP270 probe (calibrated to ^{137}Cs), with the long axis of the PNAD aligned with the long axis of the HP270 probe., the probe shield in the closed position and the PNAD in contact with the probe shield. To obtain an estimate of the neutron dose, use the values of time and exposure rate to determine a reference point on the nomogram, and then use the isodose lines to estimate absorbed neutron dose (rad).

The Quick-Scan procedure mentioned above was performed during the 23rd Criticality Intercomparison Study at Los Alamos National Laboratories. PNAD number 1412 was attached to a 20 cm X 20 cm X 10 cm Lucite phantom, and was irradiated during shot number IV (Godiva reactor, using the 20 cm concrete shield). The reference neutron dose was 34 rad. The following data were obtained using an Eberline model HP270 probe attached to a Bicron model Surveyor 2000 (exposure rate calibrated to ^{137}Cs):

Table B- 2. Quick-Scan Data For PNAD No. 1412[a].

Irradiation Time (AM)	Measurement Time (AM)	Elapsed Time (min.)	Measured Exposure Rate (mR/h)
06:15	07:52	97	25
	08:38	143	15
	08:59	165	11
	09:22	187	8.0
	09:42	207	6.5
	10:09	234	4.0
	10:30	255	3.5
	10:56	281	2.5
	11:35	320	1.2

[a] Data obtained on 6/14/95 during the 23rd Intercomparison at Los Alamos National Laboratories.

The measured exposure rate data from Table B-2 is displayed in Figure B-2. Figure B-2 also includes 10 rad, 34 rad, and 100 rad isodose lines for data comparison. Figure B-2 demonstrates that the Quick-Scan technique is reliable and useful for sorting purposes.

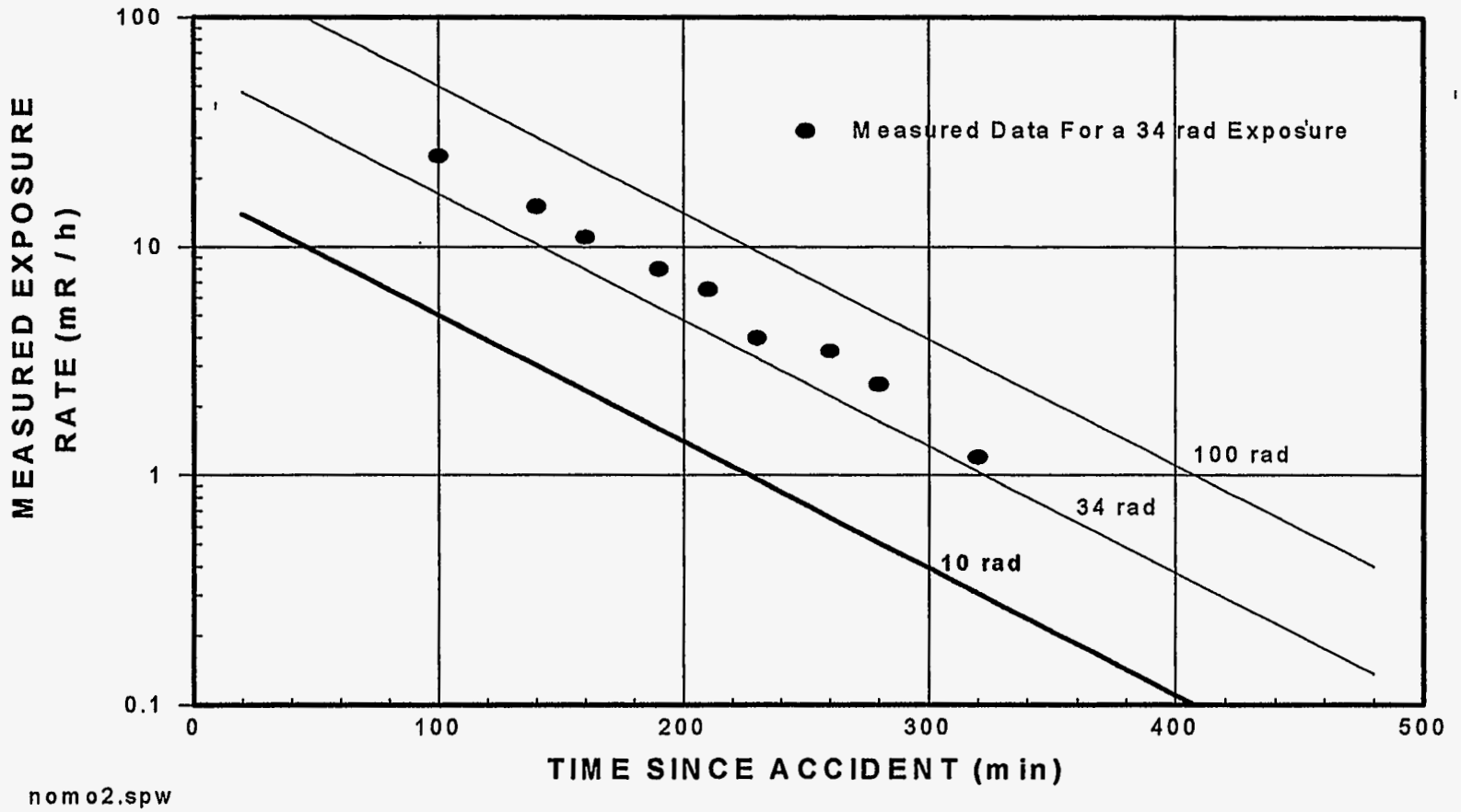

Figure B-2. Results of PNAD Quick-Scan procedure. The data points shown as ovals represent exposure rate measurements performed on an intact PNAD that was exposed to 34 rad of neutron radiation. The PNAD was located on a 20 cm X 20 cm X 10 cm Lucite phantom and was exposed to a Godiva spectrum which passed through 20 cm of concrete. In this situation, the Quick-Scan technique slightly overestimates the actual neutron dose.

APPENDIX C: PNAD FOIL ANALYSIS

A High Purity Germanium (HPGE) detector or equivalent should be used to count the PNAD foils.[C1] A mixed-nuclide radioactive source in a similar foil geometry should be used to calibrate the detector.[C2] The first calibration step is to determine the constants for converting the MCA channel number to energy in keV. This is done by placing the known source with known gamma energies on the detector and finding the channels where the corresponding photopeaks appear. The data is used to find the slope and offset for the linear equation: Energy = m*channel + b. The second step is to document how the FWHM of the photopeaks depends upon energy. The FWHM of each peak is determined and fit to the theoretical form: $FWHM = m^*\sqrt{E} + b$. This equation plays an important role in assisting peak search routines to perform accurately. The third and last step in the calibration process is to relate the observed net counts of a peak to its activity. This is done by placing the multi-nuclide source on the detector, computing the observed efficiency, and fitting the data to an empirical equation. The efficiency equation is a function of gamma energy and is used to determine the detector efficiency for each observed photopeak.

The counting sequence consists of placing a foil on the detector (in the same position as the calibration source), taking a spectrum, saving the spectrum, and then analyzing the saved file. The analysis consists of the following steps:

- Perform a peak search on the spectrum.
- Subtract background peaks.
- Use a gamma isotopic library to identify and quantify the observed peaks.

Each PNAD contains a sodium pellet and five metallic foils. All items are removed from the PNAD and are counted in the following order:

- The sodium pellet is counted first for 10 minutes.
- The copper foil is removed from the cadmium cup and is counted for 20 minutes.
- The titanium, nickel, and aluminum foils are counted together[C3] for 60 minutes.
- The indium foil is counted last for 20 minutes.

[C1] The counting system components used during NAD23 consisted of a Canberra portable n-type HPGE with a relative efficiency of 50%, a Canberra Inspector module, and IBM Notebook running the Canberra's Genie-PC Gamma Spectroscopy analysis and control software. The Inspector Module houses the high voltage supply, a linear amplifier, an ADC with a maximum gain of 8192 channels, and an MCA. A serial (RS-232) connection is used to connect the Inspector module to the computer.

[C2] The mixed radionuclide source used during NAD23 was a NIST traceable source which contained a number of radionuclides providing gamma energies from 60 to 1836 keV.

[C3] The uncertainty introduced by stacking the foils was evaluated and determined to be less than 1%.

It is important to wait at least four hours from the time of activation before counting the In foil. This allows most of the ^{116m}In activity to decay away and makes possible a more accurate determination of ^{115m}In activity.

APPENDIX D: SAMPLE CALCULATIONS

1. General

The absorbed dose received as a result of a criticality accident has two major components. The first component is due to neutrons, and the second component comes from X and gamma ray photons. Data from the SNL PNAD is used to estimate the neutron dose component, and data from the SNL TLD can be used to most accurately estimate the photon component.

A series of sample calculations is presented below to document the process of dose determination for criticality accidents. The indium foil always has a relatively large specific activity and has given the best overall dosimetry results of any material used in the SNL PNAD. For these reasons, neutron fluence and reported dose should be obtained from the indium counting data. Data from other foils should be used for confirmation purposes.

The cadmium-covered copper foil has a unique use. Neutron dose and kerma (in tissue) result primarily from neutrons in the energy region above 15 keV, whereas ^{64}Cu activity is mainly determined by the neutron fluence in the region between 1 eV and 15 keV. Uncertainties in the fluence spectrum in the region between 1 eV and 15 keV may lead to large deviations in the measured ^{64}Cu activity and, therefore in the derived dose (Eisen et al, 1985). The sensitivity of the copper foil to neutrons of thermal and epithermal energies can, however, be used. In a highly thermalized field, the threshold type reactions (Table 2, main text) will not occur. The SNL TLD, however, has an element which is also very sensitive to thermal and epithermal neutrons. In such a situation, the SNL TLD should be used to obtain both photon and neutron dose information.

2. Absorbed Dose Due to Neutrons (Tissue)

2.1 Fluence Determination: The following table contains SNL PNAD data obtained during the Nuclear Accident Dosimeter Intercomparison Study (NAD23), held at Los Alamos National Laboratories 15-18 July 1995. The incident neutron spectrum was best approximated by spectrum #16, from Table A-1. The PNAD was located on a 40 cm X 40 cm X 15 cm Lucite phantom during the exposure.

Table D- 1. Measured PNAD Data[a,b]

Reaction	Measured Specific Activity (Bq/g)	Measured Specific Activity (μCi/g)	Uncertainty in Measured Specific Activity (%)[c]
$^{115}In(n,n')^{115m}In$	1270	0.0343	21.4
$^{47}Ti(n,p)^{47}Sc$	1.19	0.0000322	29.3
$^{58}Ni(n,p)^{58}Co$	1.81	0.0000489	18.0
$^{27}Al(n,\alpha)^{24}Na$	10.74	0.00290	24.0
$^{63}Cu(n,\gamma)^{64}Cu$	3340	0.0903	21.6

[a] Data obtained from run #2, SHEBA High Power, NAD23
[b] Reference dose was 108 rads (to tissue, from neutrons), uncertainty not reported
[c] Uncertainty is 2σ

From Table A-1, spectrum #16 was selected as the best estimate of the incident neutron spectrum. The following values were obtained:

- Spectrum-weighted cross-section (σ) for $^{115}In(n,n')^{115m}In$ 0.0120 barns
- Fluence-to-dose conversion factor $\overline{D}$ for Spectrum # 1:2.29 $\frac{nrad}{n/cm^2}$

The neutron fluence Φ (n/cm^2) is calculated by selecting the indium foil equation shown in Table 2. This equation is as follows:

$$\Phi = \frac{A}{5.985\,x10^{-12}\,\sigma} \qquad \text{Eqn. D-1}$$

where A is the measured, specific foil activity (μCi/g). Substituting the above mentioned values yields:

$$\Phi = \frac{0.0343}{5.985x10^{-12}\,x(0.120)} = 4.78x10^{10}\,n/cm^2 \qquad \text{Eqn. D-2}$$

2.2 Absorbed Dose (tissue): The absorbed dose due to neutron radiation is given by the following equation:

$$Dose(rad) = \Phi(\frac{n}{cm^2})\bar{D}(\frac{nrad}{n/cm^2})x10^{-9}\frac{rad}{nrad} \qquad \text{Eqn. D-3}$$

Substituting values from section 2.1 above yields:

$$Dose = (4.78x10^{10} n/cm^2)(2.29\frac{nrad}{n/cm^2})x10^{-9}\frac{rad}{nrad} \qquad \text{Eqn. D-4}$$

$$= 109 \text{ rad}$$

3. Absorbed Dose Due to Photons (Tissue)

Photon deep dose (rad) is obtained by processing the SNL TLD using standard procedures. The photon deep dose obtained for this example was 100 rad.

4. Combining Neutron and Photon Dose Values

The neutron and photon dose values should be summed and reported as absorbed dose (rad).

For this example, the total reportable dose was (109 +100) = 209 (rad).

NOTE:

The concept of dose equivalent is of limited use in accidents when doses in excess of 25 rads have been received. The quality factor in such situations is not uniquely defined, although it is generally assumed (RBE Committee, 1963) to be between one and two. The dose equivalent ($\overline{DE}$) values presented in Table A-1 have been calculated using the quality factors normally adopted for radiation protection. These values are provided for the convenience of determining the dose equivalent for those individuals exposed to doses less than 25 rem and for whom the dose equivalent received in a criticality incident must be added to the already accumulated record (Ing & Makra, 1978).

APPENDIX E: PRODUCTION SPECIFICATIONS FOR NAF TABLETS

The NaF tablets (or disks) used in the SNL PNAD have been fabricated at Sandia National Laboratories. Since these items are not commercially available, the specifications and processing requirements needed to fabricate the compressed NaF disks have been included in this work for completeness of documentation.

Disks manufactured using the following specifications will contain 50% by weight of NaF.

12/11/91

CAGE CODE 14213

SS396585
-000

THOMPSON 7710
APPEL 2473
MORGAN 2413
MANLEY 2833
CARTRIDGE 1

PROCESSING OF SODIUM FLUORIDE DISKS FOR CRITICALITY DOSIMETER MONITORING DEVICES (U)

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Issue	A	A	A	A

1. GENERAL

1.1 Scope. This specification describes the processing requirements needed to fabricate compressed NaF disks for criticality dosimeter monitoring devices. These devices provide directional location of personnel and indicate the percent irradiation by absorption.

2. EQUIPMENT

2.1 Ball Mill. US Stoneware Y5023 Ball Mill.

2.2 MTP-8 Tetrahedran Press. Materials Test Press/Hydraulic Lamination Press.

2.3 Strippit Punching Equipment. Manually Operated.

3. MATERIALS

3.1 Sodium Fluoride (NaF). One pound reagent grade powder.

3.2 Microthene. One pound of binder.

3.3 Nalgene Mill Jar. One gallon size.

3.4 Mill Balls (5 each). 0.833" x 0.815" cylindrical Alumina.

3.5 Weighing Dish. 40 cc disposable Al dish.

3.6 Caul Plates. 8.0" x 9.0" x 0.0625" 304 stainless steel plates.

3.7 Metal Shims (4 each). 1.5" x 1.5" x 0.113".

3.8 Spatula. Lab Spoon.

3.9 Mold Release. Ram 225 Liquid Mold Release.

3.10 Cotton/Paper Cloths.

3.11 1/2" Punch and Die Set. Close tolerance/0.002" clearance.

3.12 Alcohol.

3.13 Catch Bag. Olefin Plastic Bag.

3.14 Gloves.

3.15 Micrometer Measuring Instrument.

4. SAFETY REQUIREMENTS

4.1 Use proper ventilation/exhausting when applying mold release.

4.2 Use gloves when applying the mold release.

4.3 Use gloves when handling the NaF powder.

4.4 Follow safety procedures in using equipment.

5. PROCESS REQUIREMENTS

5.1 Powder Preparation.

5.1.1 Weigh one pound of sodium fluoride and one pound of microthene.

5.1.2 Place equal parts (by weight) of sodium fluoride powder and microthene in mill jar.

5.1.3 Put five each mill balls in the mill jar.

5.1.4 Place mill jar on ball mill and mill for 24 hours.

NOTE: Intermittently shake the jar to prevent settling.

5.2 Caul Plates Preparation.

5.2.1 Apply mold release on both sides of two cleaned caul plates using a soft cloth (apply just enough to wet surfaces).

NOTE: Use gloves and work in a properly ventilated area.

5.2.2 Allow the plates to air dry for five minutes.

5.2.3 Place the plates in a conventional convection oven set at 230°F for two hours.

5.2.4 Remove the plates from the oven and allow to cool.

5.3 Powder Slug Preparation.

5.3.1 Spoon the NaF powder into the weighing dish using a spatula. Compact the powder intermittently with the flat portion of the spatula. Approximately 42 g of powder will fill the dish.

5.4 Press Preparation.

5.4.1 Open the platens of the press.

5.4.2 Place the mold-released caul plate on the bottom platen (center).

5.4.3 Place the metal shims on the corners of the platens (thickness control for pressed slugs).

5.4.4 Invert the weighing dish of prepared powder on the center of the bottom caul plate.

5.4.5 Remove the weighing dish.

5.4.6 Place the mold-released caul plate on top of the powder slug (center).

5.5 Pressing.

5.5.1 Use the following parameters for the press run:

a. Set the heat for 325°F.

b. Set the heat rate for 50°F per minute.

c. Set the pressure to 1000 pounds of force.

d. Set the pressure force to increase at 500 pounds per minute.

e. Set the dwell time to 10 minutes.

f. Set the cooling cycle to cool rapidly to 160°F under pressure.

5.5.2 Close the platens.

5.5.3 Run the press/lamination cycle to press the powder slug.

5.5.4 Remove the top caul plate after the press cycle is complete and the platens have opened.

5.5.5 Remove the bottom caul plate with the powder slug which has been pressed flat and let it cool for a few minutes.

5.5.6 Remove the flat powder slug and clean both caul plates with a soft cloth to wipe residual powder off the caul plate.

5.6 Interim Inspection of Pressed Powder Slug.

5.6.1 Measure the thickness of the pressed powder disk. It should measure 0.10" +0.0/-0.01".

5.7 Punch Disks.

5.7.1 Degrease punch and die set with alcohol.

5.7.2 Clean working surface(s) of the punching equipment by wiping with alcohol and cloth.

5.7.3 Insert catch bag (disk collection).

5.7.4 Insert punch and die into punching equipment.

5.7.5 Turn power on and set the equipment to Single Stroke (Manual Stroke Mode).

5.7.6 Place flattened powder slugs on working surface of punching equipment.

5.7.7 Punch out 0.50" sized disks.

a. Manually align the flattened powder slug under the punch head.

b. Push the foot pedal (or actuating device) to punch out a disk.

c. Rotate the flattened powder slug.

d. Continue the punching process (steps 5.7.7.a through 5.7.7.c) until all the available surface area of the slug is exhausted.

5.7.8 Remove the punch and die set.

5.7.9 Remove the catch bag.

5.7.10 Clean and wipe the working area.

5.8 Final Inspection and Measurement of Powder Disks.

NOTE: Wear gloves when handling the disks.

5.8.1 Visual Inspection of Disks. Visually inspect the individual disks (look for contamination, voids, edge and shape quality, uniform disk compaction, etc.).

5.8.2 Dimensional Inspection of Disks. Measure the thickness of the disks with a calibrated micrometer. Assure that the thickness is 0.10" +0.0/-0.01".

APPENDIX F: PNAD INTERCOMPARISON RESULTS FROM NAD23

This appendix contains the detailed calculations and graphical summaries of PNAD performance during NAD23. Data tables and figures are linked as shown below: All calculated doses due to neutron exposure are dose to tissue.

Some exposures, such as the SHEBA "High Power" incorporated two different PNADs. One PNAD was mounted on a 40 cm X 40 cm X 15 cm Lucite phantom while the second PNAD was located nearby but without any backscatter material. This was done to obtain data for "On phantom" and "Free-in-air" geometries.

Reference dose values are also shown in Figures 1 through 7. These are the values reported by Los Alamos for absorbed neutron dose to tissue. These values were given to the participants in NAD23 in order to assess dosimeter performance. Uncertainties associated with the reference values were not made available.

RUN #	NEUTRON SOURCE	DATA PRESENTED IN TABLE #	GRAPHICAL SUMMARY SHOWN IN FIGURE #
1	SHEBA-"Low Power"	F-1	F-1
2	SHEBA-"High Power"	F-2	F-2
3	GODIVA-"Bare"	F-3	F-3
4	GODIVA-"20 cm Concrete Shielding"	F-4	F-4
5	SHEBA-"Free Run"	F-5	F-5
6	GODIVA-"12 cm Lucite Shielding"	F-6	F-6
7	GODIVA-"13 cm Steel Shielding	F-7	F-7

Values of uncertainty listed in Tables F-1 through F-7 are $\pm 2\sigma$.

Table F- 1. Data From NAD23 Run # 1: "SHEBA-Low Power[a]"

PNAD #	Set Up		Material				
			In	Cu	Ti	Ni	Al
1405	On Phantom	Foil Wt. (g)	0.226	0.612	0.521	0.830	0.271
		MDA[b] (μCi g^{-1})	0.00624[c]	0.0424	0.000042	0.00011	0.00039
		Foil Activity (μCi g^{-1})	<MDA	<MDA	<MDA	<MDA	<MDA
		Uncertainty					
		Fluence (n cm^{-2})					
		Absorbed Dose (rad)					
		Uncertainty					
004	Free In Air	Foil Wt. (g)	0.229	0.609	0.531	0.836	0.272
		MDA[b] (μCi g^{-1})	0.000784	0.0424	0.000042	0.0000776	0.000714
		Foil Activity (μCi g^{-1})	0.00349	<MDA	<MDA	<MDA	<MDA
		Uncertainty	0.000795				
		Fluence (n cm^{-2})	4.86 x 10^{9}				
		Absorbed Dose[d] (rad)	11.1				
		Uncertainty	2.53				

[a] Exposure date 06/12/95
[b] Minimum Detectable Activity for the counting system
[c] Indium foil counted without allowing several hours for decay of ^{116m}In
[d] Absorbed dose calculated using spectrum #16 data, from Table A-1.

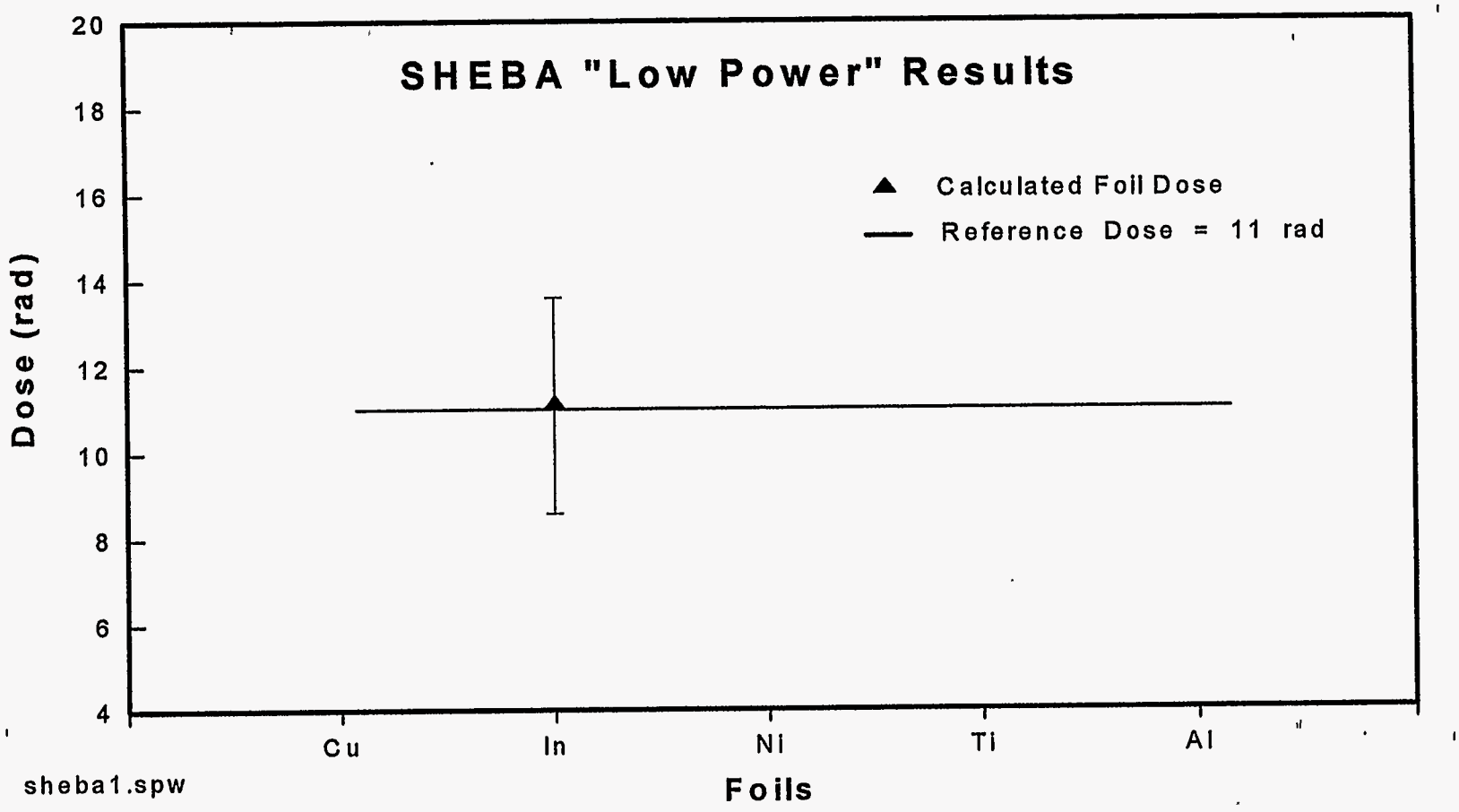

Figure F- 1. Results of SHEBA "Low Power" exposure. The Cu, Ni, Ti and Al foils did not have activities above the MDA of the counting system

Table F- 2. Data From NAD23 Run # 2: "SHEBA-High Power[a]"

PNAD #	Set Up		Materials				
			In	Cu	Ti	Ni	Al
1410	On Phantom	Foil Wt. (g)	0.229	0.608	0.523	0.831	0.270
		MDA[b] (μCi g^{-1})	0.00576	0.0213	0.0000127	0.0000081	0.0000773
		Foil Activity (μCi g^{-1})	0.0343	0.124	0.0000319	0.0000501	0.000290
		Uncertainty	0.00735	0.0268	0.00000935	0.0000090	0.0000697
		Fluence (n cm^{-2})	4.78×10^{10}	4.68×10^{11}	4.34×10^{10}	3.22×10^{10}	6.71×10^{10}
		Absorbed Dose[c] (rad)	109	1070	99	74	154
		Uncertainty	23.4	231	29	13	37
1409	Free In Air	Foil Wt. (g)	0.230	0.611	.526	0.833	0.271
		MDA[b] (μCi g^{-1})	0.0107	0.125	0.000136	0.000117	0.000889
		Foil Activity (μCi g^{-1})	0.0346	<MDA[b]	<MDA[b]	<MDA[b]	<MDA[b]
		Uncertainty	0.00978				
		Fluence (n cm^{-2})	4.82×10^{10}				
		Absorbed Dose[c] (rad)	110				
		Uncertainty	31.2				

[a] Exposure date 06/12/95: end exposure @ 23:37.
[b] Minimum Detectable Activity for the counting system
[c] Absorbed dose calculated using spectrum #16 data, from Table A-1.

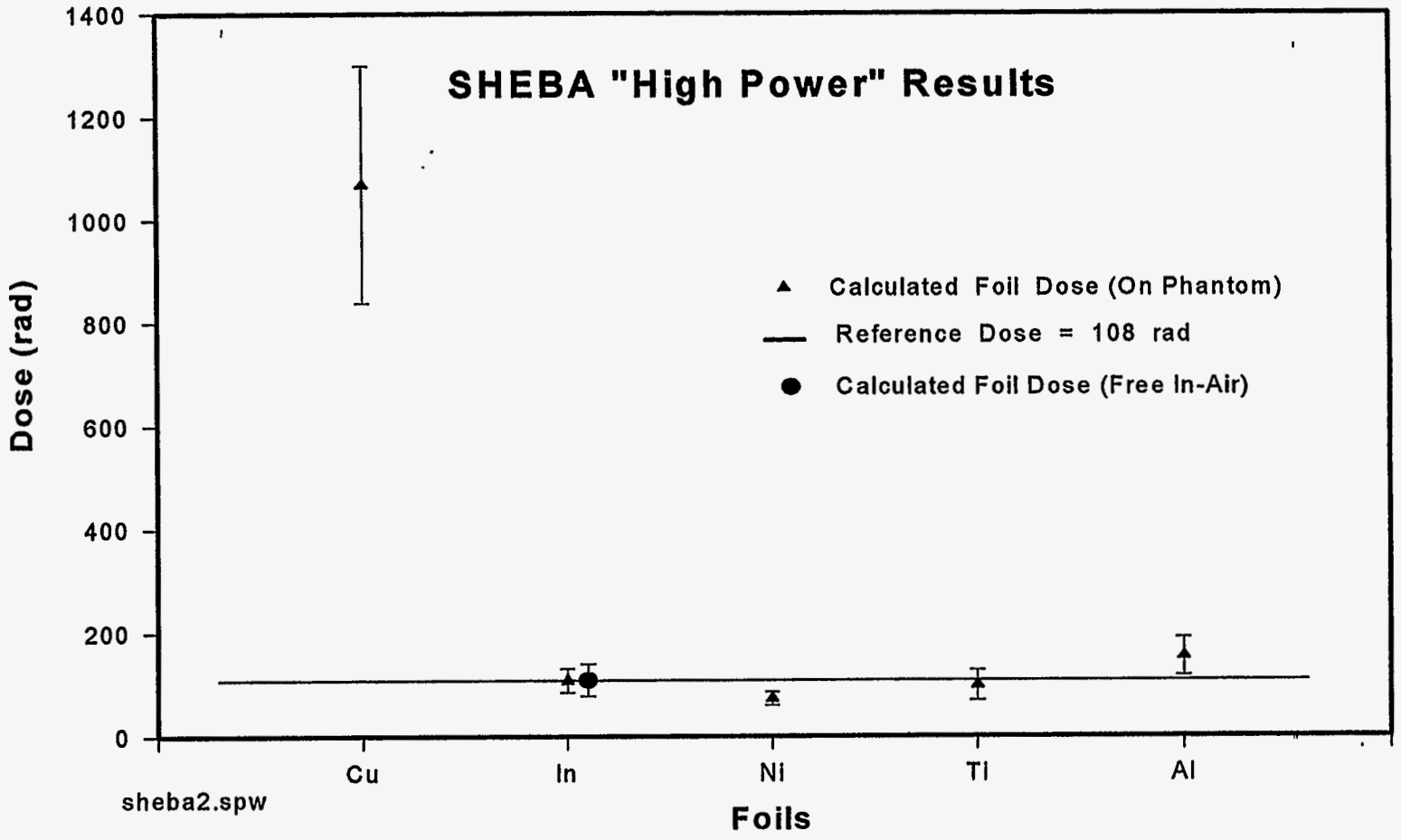

Figure F- 2. Results of the SHEBA "High Power" exposure.

Table F- 3. Data From NAD23 Run # 3: Godiva-Bare[a]

PNAD #	Set Up		Materials				
			In	Cu	Ti	Ni	Al
1344	On Phantom	Foil Wt. (g)	0.232	0.609	0.523	0.843	0.270
		MDA[b] (μCi g^{-1})	0.0145	0.0224	0.0000284	0.0000527	0.0000780
		Foil Activity (μCi g^{-1})	0.0735	0.127	0.0000730	0.000108	0.000244
		Uncertainty	0.0159	0.0403	0.0000094	0.0000090	0.0000697
		Fluence (n cm^{-2})	9.52×10^{10}	3.67×10^{12}	1.06×10^{11}	7.68×10^{10}	6.76×10^{10}
		Absorbed Dose[c] (rad)	264	10200	295	213	187
		Uncertainty	57	3230	38	18	53
None	Free In Air	Foil Wt. (g)					
		MDA[b] (μCi g^{-1})					
		Foil Activity (μCi g^{-1})					
		Uncertainty					
		Fluence (n cm^{-2})					
		Absorbed Dose[c] (rad)					
		Uncertainty					

[a] Exposure date 06/13/95:
[b] Minimum Detectable Activity for the counting system
[c] Absorbed dose calculated using spectrum #2 data, from Table A-1.

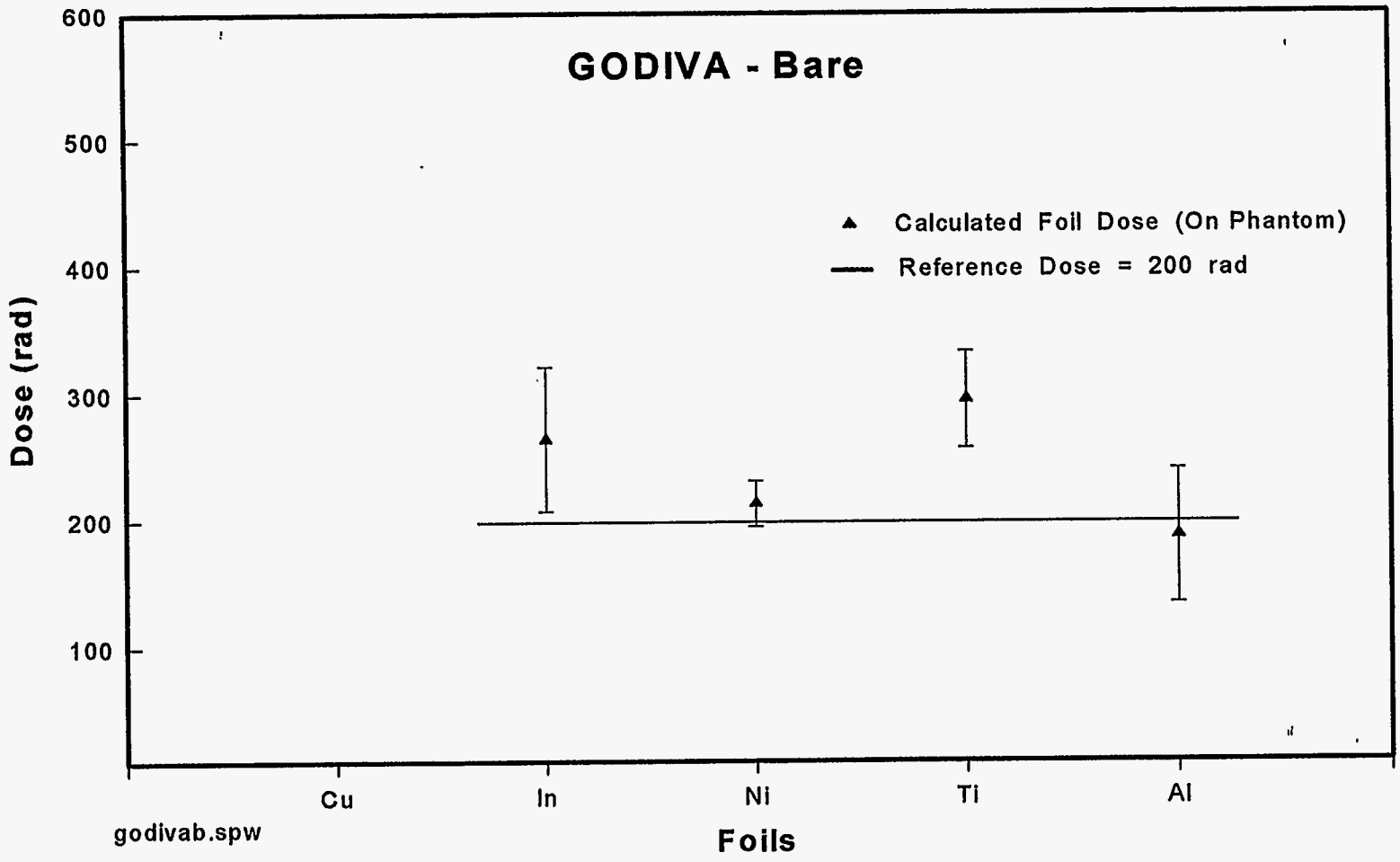

Figure F- 3. Results of the GODIVA Bare exposure.

Table F- 4. Data From NAD23 Run # 4: Godiva-20 cm Concrete[a]

PNAD #	Set Up		Materials				
			In	Cu	Ti	Ni	Al
1347	On Phantom	Foil Wt. (g)	0.226	0.606	0.526	0.833	0.271
		MDA[b] (μCi g^{-1})	0.00278	0.0759	0.0000284	0.0000527	0.0000780
		Foil Activity (μCi g^{-1})	0.0118	<MDA[b]	<MDA[b]	<MDA[b]	<MDA[b]
		Uncertainty	0.00273				
		Fluence (n cm^{-2})	2.05 x 10^{10}				
		Absorbed Dose[c] (rad)	37.7				
		Uncertainty	8.7				
1419	Free In Air	Foil Wt. (g)	0.217	0.606	0.526	0.834	0.272
		MDA[b] (μCi g^{-1})	0.00264	0.0295	0.0000267	0.0000403	0.0000622
		Foil Activity (μCi g^{-1})	0.00970	0.0859	<MDA[b]	<MDA[b]	<MDA[b]
		Uncertainty	0.00250	0.0338			
		Fluence (n cm^{-2})	1.69 x 10^{10}	TBD			
		Absorbed Dose[c] (rad)	31.0				
		Uncertainty	8				

[a] Exposure date 06/13/95:
[b] Minimum Detectable Activity for the counting system
[c] Absorbed dose calculated using spectrum #24 data, from Table A-1.

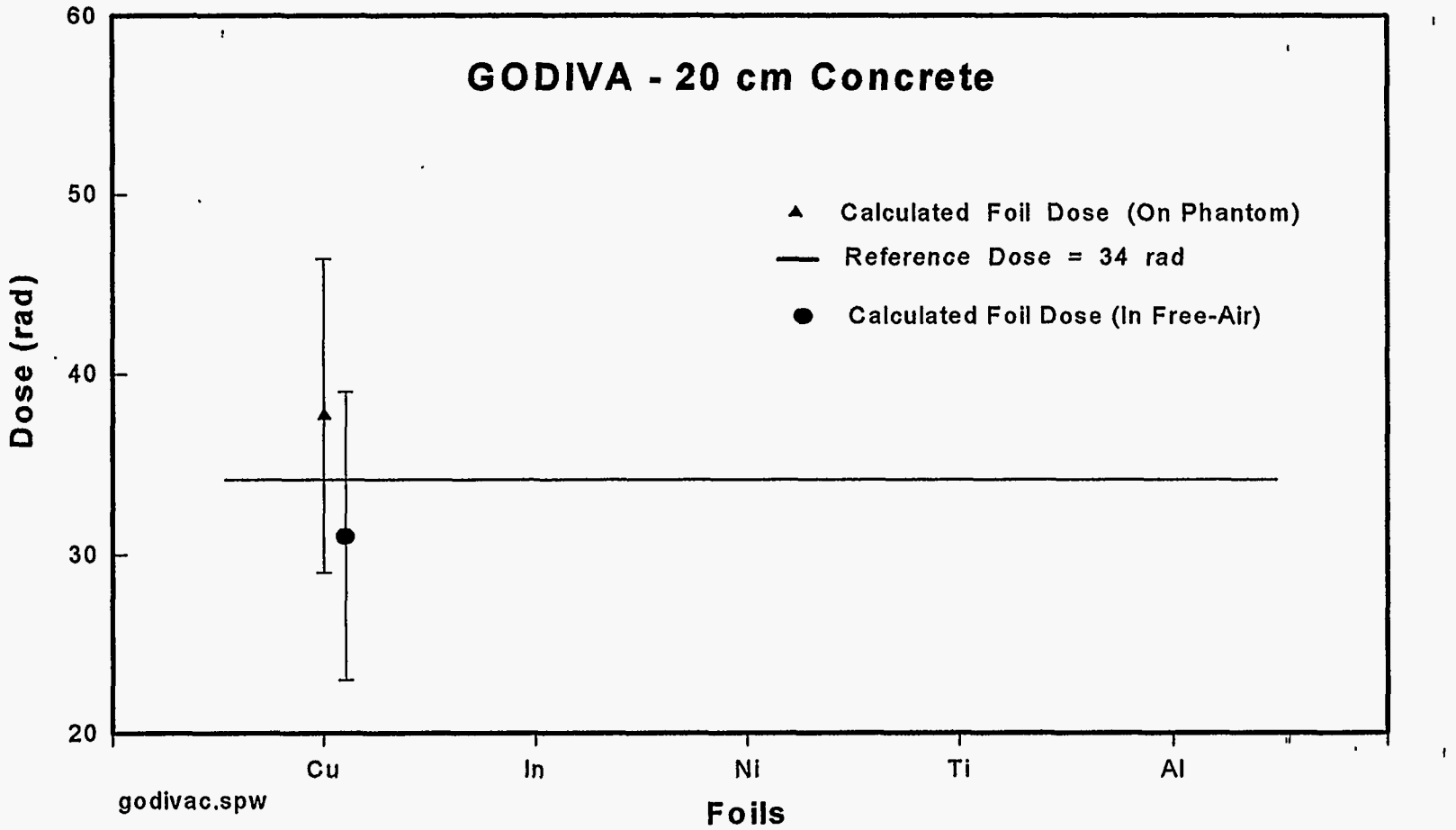

Figure F- 4. Results of the GODIVA 20 cm Concrete exposure.

Table F- 5. Data From NAD23 Run # 5: SHEBA-"Free Run[a]"

PNAD #	Set Up		Materials				
			In	Cu	Ti	Ni	Al
1418	On Phantom	Foil Wt. (g)	0.217	0.606	0.521	0.830	0.271
		MDA[b] (μCi g^{-1})	0.00995	0.078	0.0000560	0.0000449	0.000325
		Foil Activity (μCi g^{-1})	0.0314	0.519	<MDA[b]	0.0000547	<MDA[b]
		Uncertainty	0.00857	0.117		0.0000303	
		Fluence (n cm^{-2})	4.37 x 10^{10}	1.96 x 10^{12}		3.52 x 10^{10}	
		Absorbed Dose[c] (rad)	100	4490		81	
		Uncertainty	27	1010		45	
1342	Free In Air	Foil Wt. (g)	0.231	0.604	0.521	0.830	0.271
		MDA[b] (μCi g^{-1})	0.00700	0.0886	0.0000489	0.0000242	0.00298
		Foil Activity (μCi g^{-1})	0.0273	<MDA[b]	<MDA[b]	0.0000297	<MDA[b]
		Uncertainty	0.00714			0.0000169	
		Fluence (n cm^{-2})	3.8 x 10^{10}			1.91 x 10^{10}	
		Absorbed Dose[c] (rad)	87			44	
		Uncertainty	23				

[a] Exposure date 06/14/95:
[b] Minimum Detectable Activity for the counting system
[c] Absorbed dose calculated using spectrum #16 data, from Table A-1.

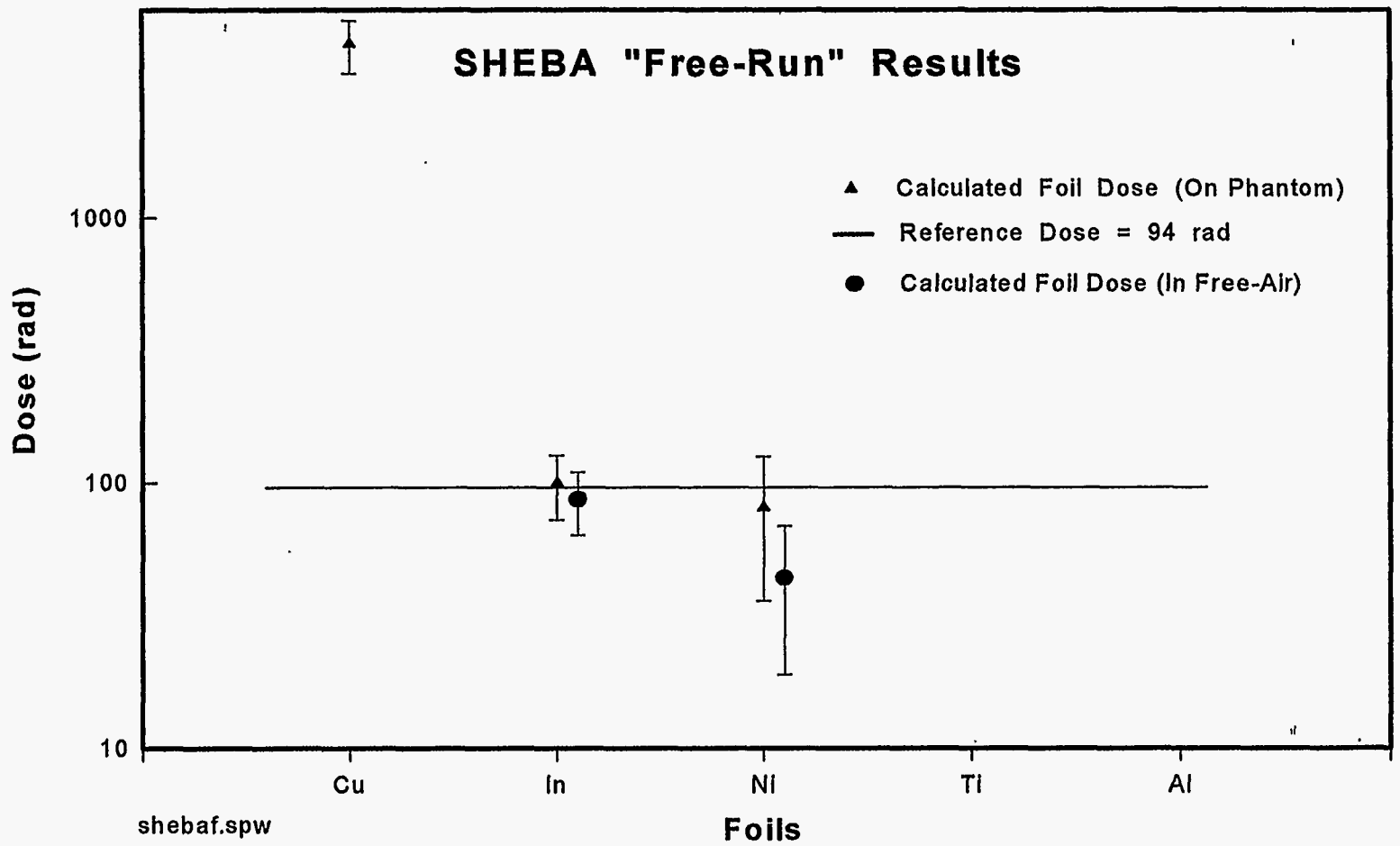

Figure F- 5. Results of the SHEBA "Free Run" exposure.

Table F- 6. Data From NAD23 Run # 6: Godiva- "12 cm Lucite[a]"

PNAD #	Set Up		Materials				
			In	Cu	Ti	Ni	Al
1350	On Phantom	Foil Wt. (g)	0.230	0.613	0.521	0.830	0.271
		MDA[b] (μCi g^{-1})	0.00368	0.0689	0.0000503	0.0000521	0.000153
		Foil Activity (μCi g^{-1})	0.00962	<MDA[b]	<MDA[b]	<MDA[b]	<MDA[b]
		Uncertainty	0.00292				
		Fluence (n cm^{-2})	1.47×10^{10}				
		Absorbed Dose[c] (rad)	32				
		Uncertainty	10				
1348	Free In Air	Foil Wt. (g)	0.230	0.607	0.521	0.830	0.271
		MDA[b] (μCi g^{-1})	0.00516	0.0824	0.0000444	0.0000345	0.000269
		Foil Activity (μCi g^{-1})	0.00608	<MDA[b]	<MDA[b]	<MDA[b]	<MDA[b]
		Uncertainty	0.00349				
		Fluence (n cm^{-2})	9.32×10^{9}				
		Absorbed Dose[c] (rad)	20				
		Uncertainty	11				

[a] Exposure date 06/14/95:
[b] Minimum Detectable Activity for the counting system
[c] Absorbed dose calculated using interpolated values from spectrums #27 and #28, from Table A-1.

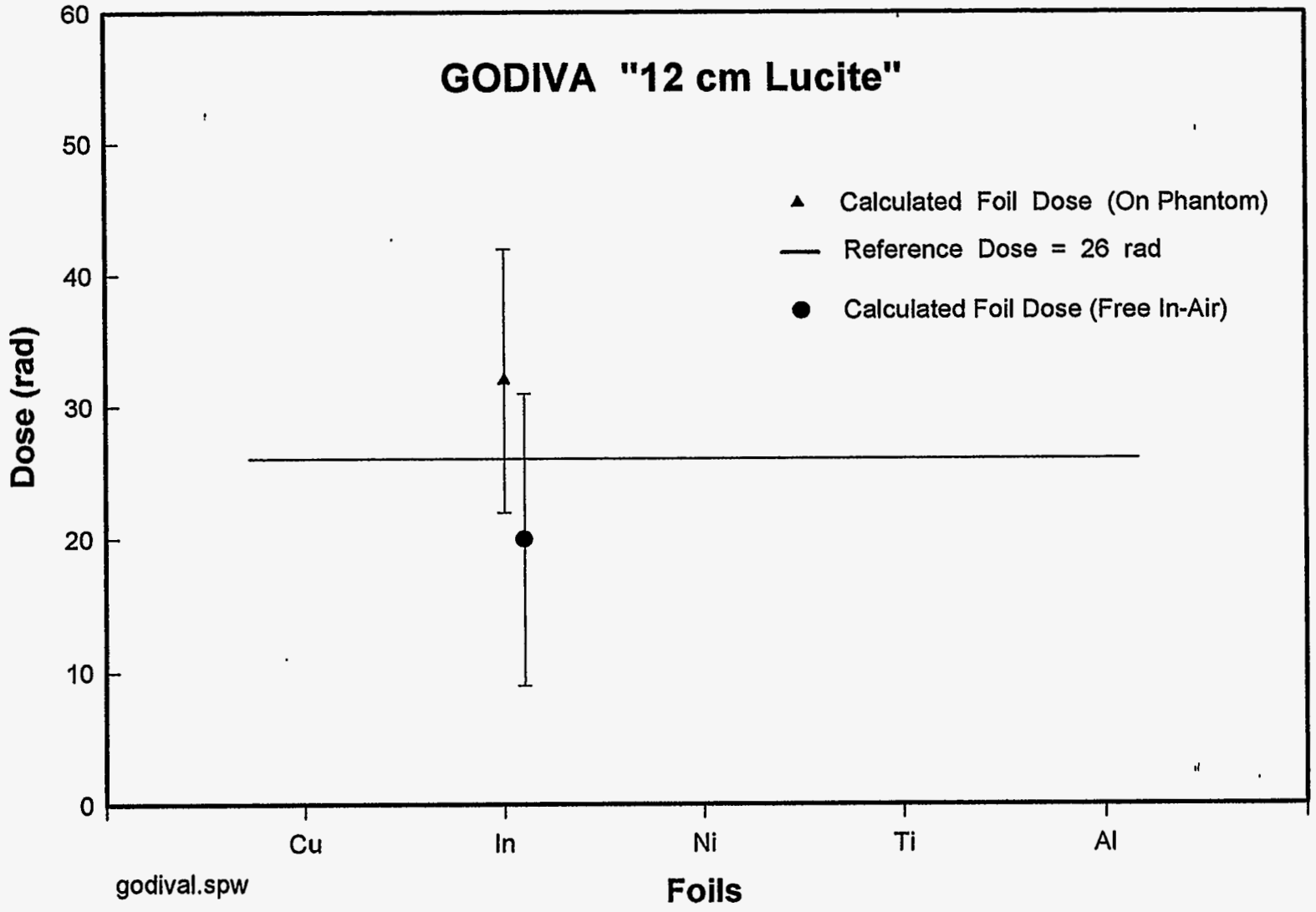

Figure F- 6. Results of the GODIVA "12 cm Lucite" exposure.

Table F- 7. Data From NAD23 Run # 7: "Godiva- 13 cm Steel"

PNAD #	Set Up		Foils				
			In	Cu	Ti	Ni	Al
1350	On Phantom	Foil Wt. (g)	0.233	0.608	0.526	0.833	0.271
		MDA[a] ($\mu Ci\ g^{-1}$)	0.000778	0.00400	0.00000213	0.00000208	0.0000117
		Foil Activity ($\mu Ci\ g^{-1}$)	0.0116	0.0221	0.00000673	0.0000116	0.0000303
		Uncertainty	0.00200	0.00686	0.00000174	0.00000214	0.00000954
		Fluence ($n\ cm^{-2}$)	2.46×10^{10}	4.71×10^{11}	2.08×10^{10}	1.89×10^{10}	3.30×10^{10}
		Absorbed Dose[b] (rad)	55.1	1060	46.6	42.3	73.9
		Uncertainty	9.5	330	12	7.8	23
1348	Free In Air	Foil Wt. (g)	0.227	0.601	0.526	0.833	0.271
		MDA[a] ($\mu Ci\ g^{-1}$)	0.000565	0.00773	0.00000280	0.00000157	0.0000141
		Foil Activity ($\mu Ci\ g^{-1}$)	0.0106	0.0150	0.00000791	0.0000124	0.0000311
		Uncertainty	0.00201	0.00684	0.00000218	0.00000247	0.0000108
		Fluence ($n\ cm^{-2}$)	2.25×10^{10}	3.20×10^{11}	2.44×10^{10}	2.02×10^{10}	3.39×10^{10}
		Absorbed Dose[b] (rad)	50.4	717	54.7	45.2	75.9
		Uncertainty	9.6	327	15.1	9.0	26.4

[a] Minimum Detectable Activity for the counting system

[b] Absorbed dose calculated using interpolated values from spectrums #32 and #33, from Table A-1.

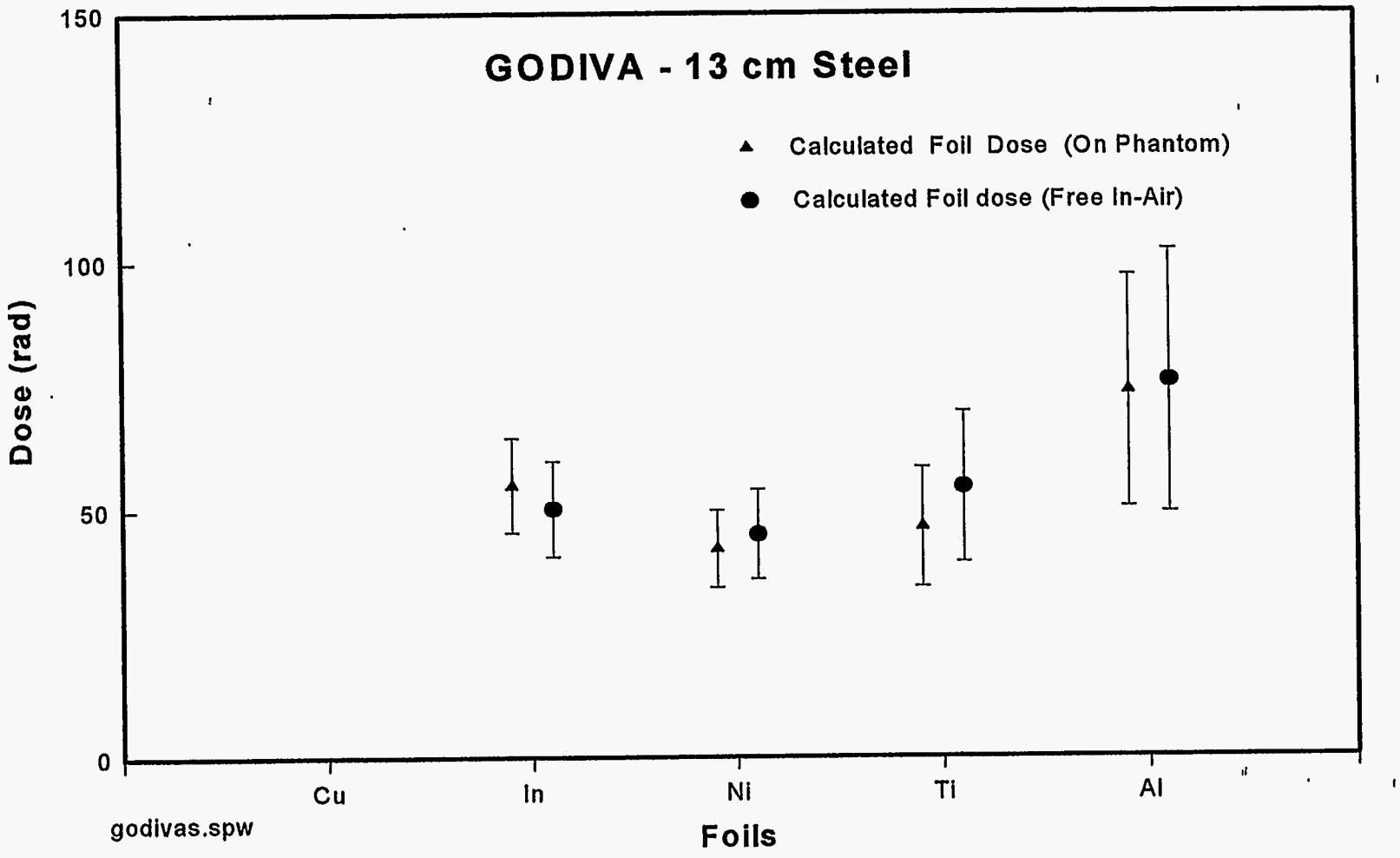

Figure F- 7. Results of GODIVA "13 cm Steel" exposure. The results for the copper foils were 1060 rads (on-phantom) and 717 rads (free in-air). The reference dose for this exposure was not given to the NAD23 participants.

APPENDIX G: NAD REFERENCE SPECTRA

The dosimetry table shown in Appendix A contains several neutron spectra not previously considered by Coates et al (1990). The additional spectra are described below. Spectra number is the same as given in Appendix A.

1 through 11:Listed in Coates et al. (1990).

12. GODIVA Bare, measured at LANL: This spectrum was supplied to all NAD23 participants by LANL. The uncertainty associated with this spectrum was not given. Spectrum is displayed in Figure G-1.

13. GODIVA with 12 cm of lucite, measured at LANL. This spectrum was supplied to all NAD23 participants by LANL. The uncertainty associated with this spectrum was not given. Spectrum is displayed in Figure G-1.

14. GODIVA with 20 cm concrete, measured at LANL. This spectrum was supplied to all NAD23 participants by LANL. The uncertainty associated with this spectrum was not given. Spectrum is displayed in Figure G-1.

15. SHEBA, Bare, measured at LANL. This spectrum was supplied to all NAD23 participants by LANL. The uncertainty associated with this spectrum was not given. Spectrum is displayed in Figure G-2.

16. Spectrum from critical solution, Fissile H_2O, 50 cm radius. Spectrum obtained from Ing & Makra (1978). Spectrum is displayed in Figure G-2 and G-3.

17. Spectrum from critical solution, Fissile H_2O, 30 cm radius. Spectrum obtained from Ing & Makra (1978). Spectrum is displayed in Figure G-3.

18. Spectrum from critical solution, Fissile H_2O, 10 cm radius. Spectrum obtained from Ing & Makra (1978). Spectrum is displayed in Figure G-3.

19. Spectrum from critical solution, Fissile H_2O, 5 cm radius. Spectrum obtained from Ing & Makra (1978). Spectrum is displayed in Figure G-3.

20. Listed in Coates et al. (1990).

21. Cf-252. Taken from NBS Compendium.

22. Fission neutrons through 10 cm concrete. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-4.

23. Fission neutrons through 20 cm concrete. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-4.

24. Fission neutrons through 30 cm concrete. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-4.

25. Fission neutrons through 40 cm concrete. Taken from Ing & Makra(1978). Spectrum is displayed in Figure G-4.

26. Fission neutrons through 5 cm polyethylene. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-5.

27. Fission neutrons through 10 cm polyethylene. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-5.

28. Fission neutrons through 20 cm polyethylene. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-5.

29. Fission neutrons through 40 cm polyethylene. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-5.

30. Fission neutrons through 5 cm iron. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-6.

31. Fission neutrons through 10 cm iron. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-6.

32. Fission neutrons through 20 cm iron. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-6.

33. Fission neutrons through 30 cm iron. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-6.

34. Fission neutrons through 50 cm iron. Taken from Ing & Makra (1978). Spectrum is displayed in Figure G-6.

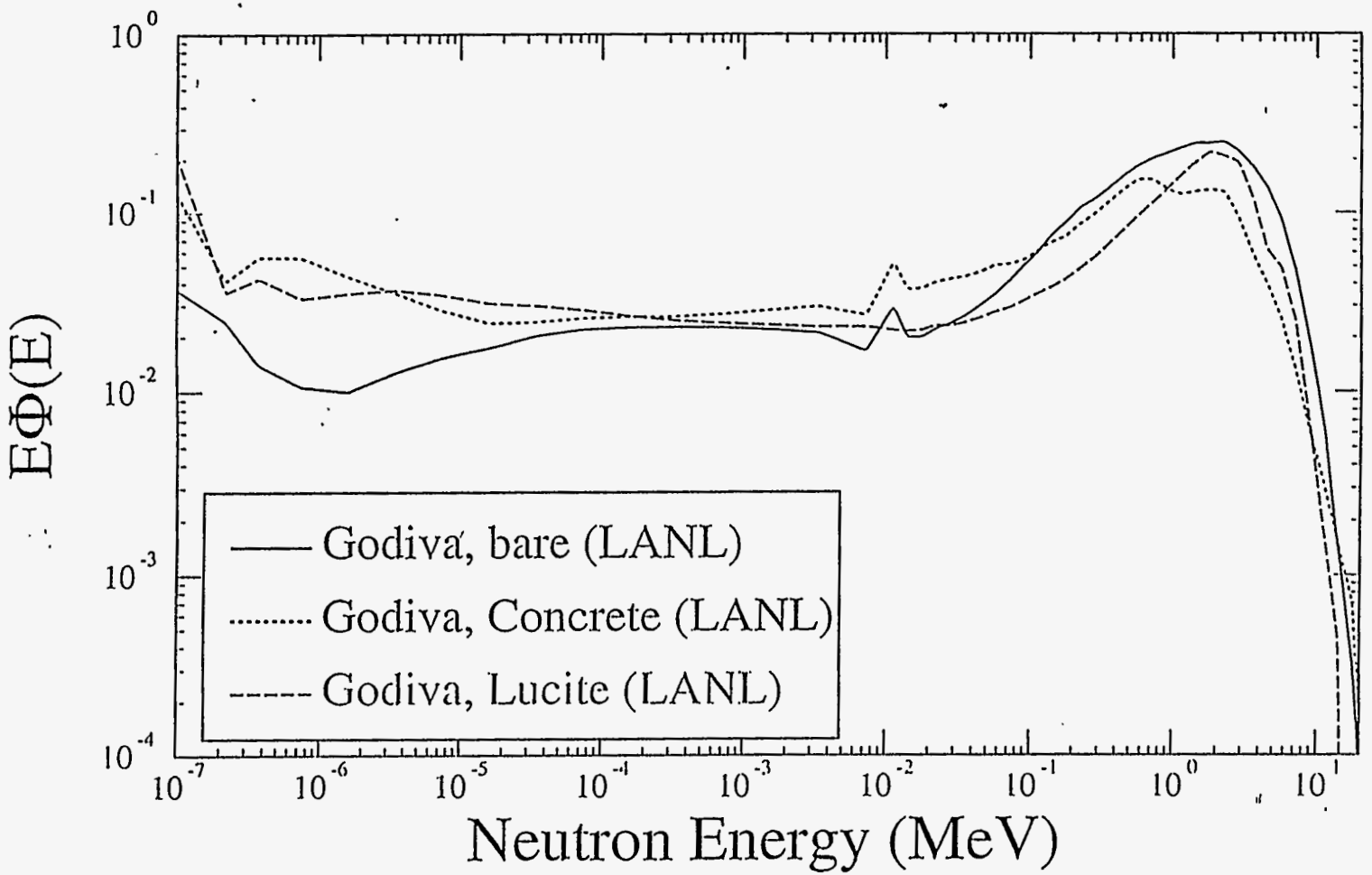

Figure G- 1. Neutron spectra supplied to NAD23 participants.

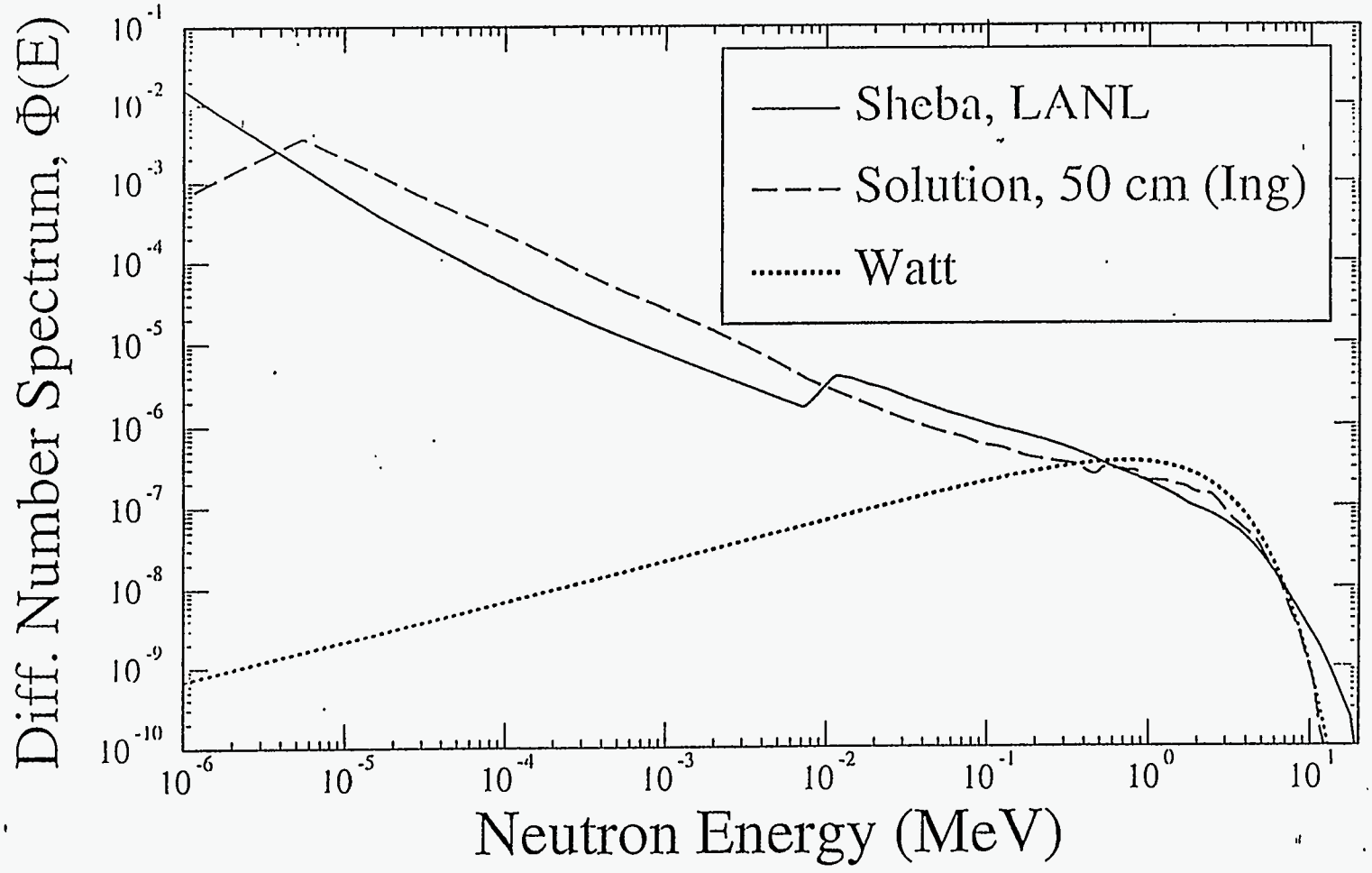

Figure G- 2. Solution criticality spectra. The SHEBA spectrum was measured at LANL. A calculated spectrum (Ing & Makra, 1978) is shown for comparison. A Watt fission spectrum has been included for comparison.

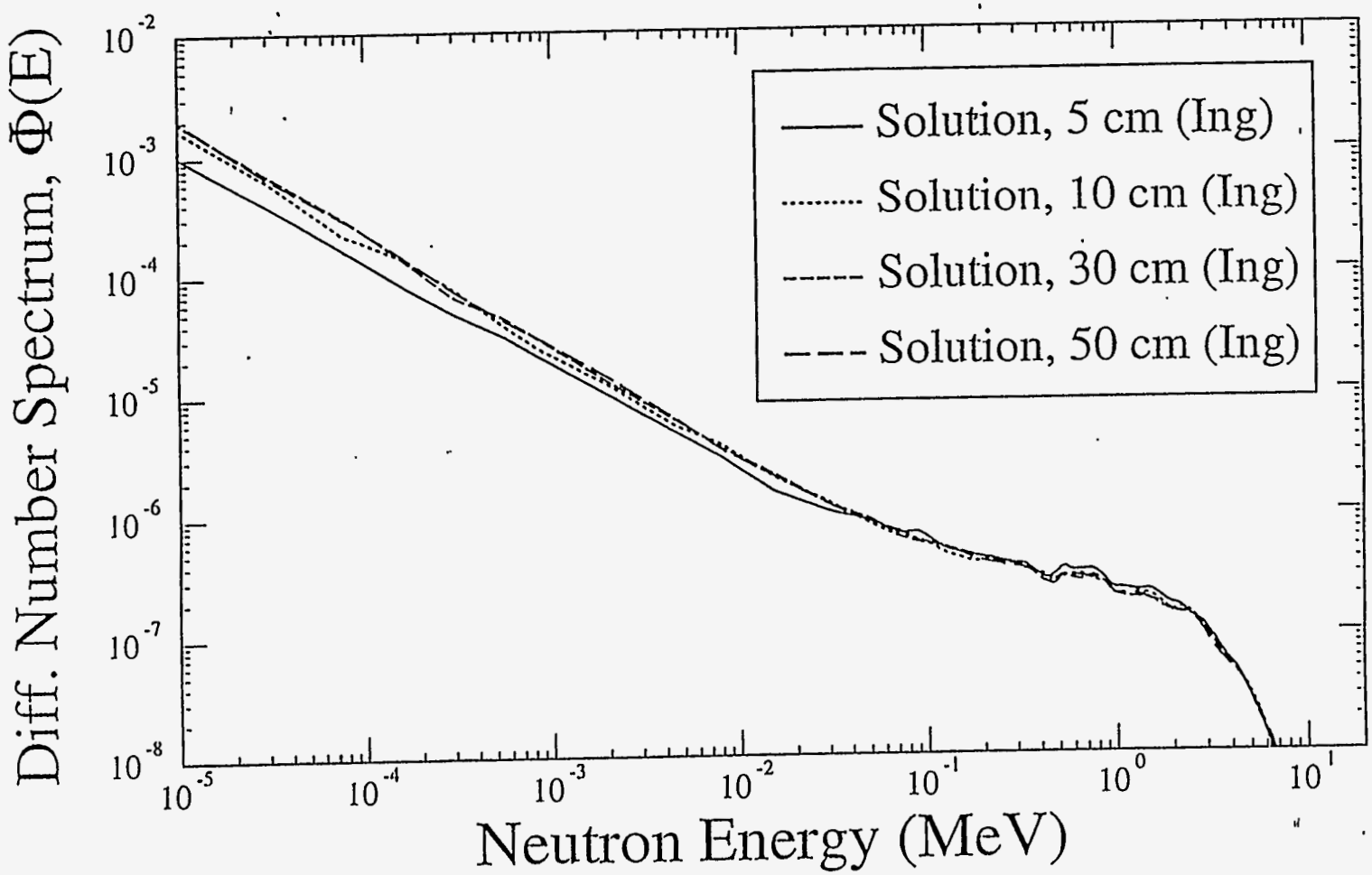

Figure G- 3. Solution criticality spectra for a spherical geometry. Numbers listed correspond to different radii (Ing & Makra 1978).

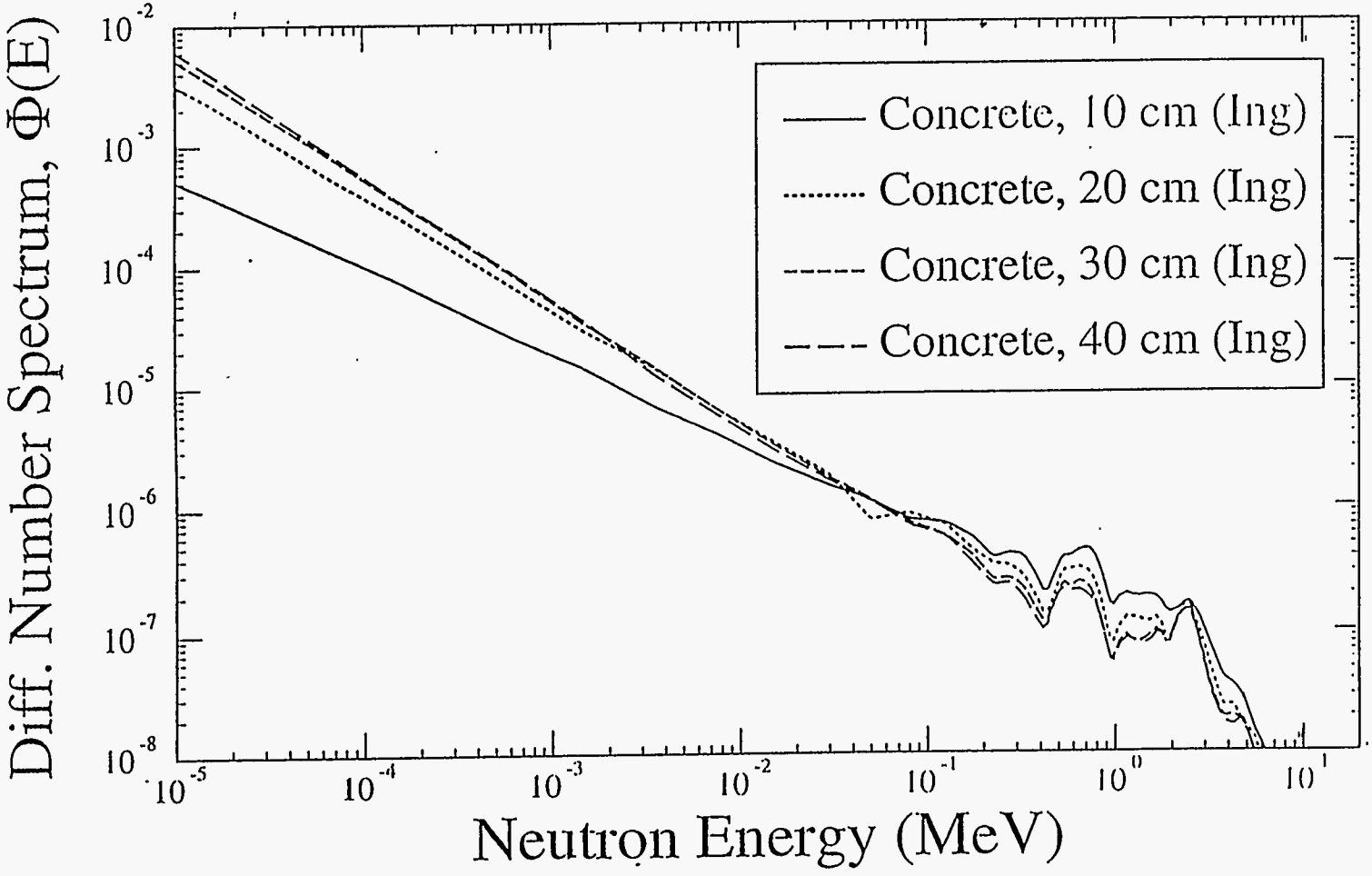

Figure G- 4. Spectra for fission neutrons through different thicknesses of concrete (Ing & Makra, 1987).

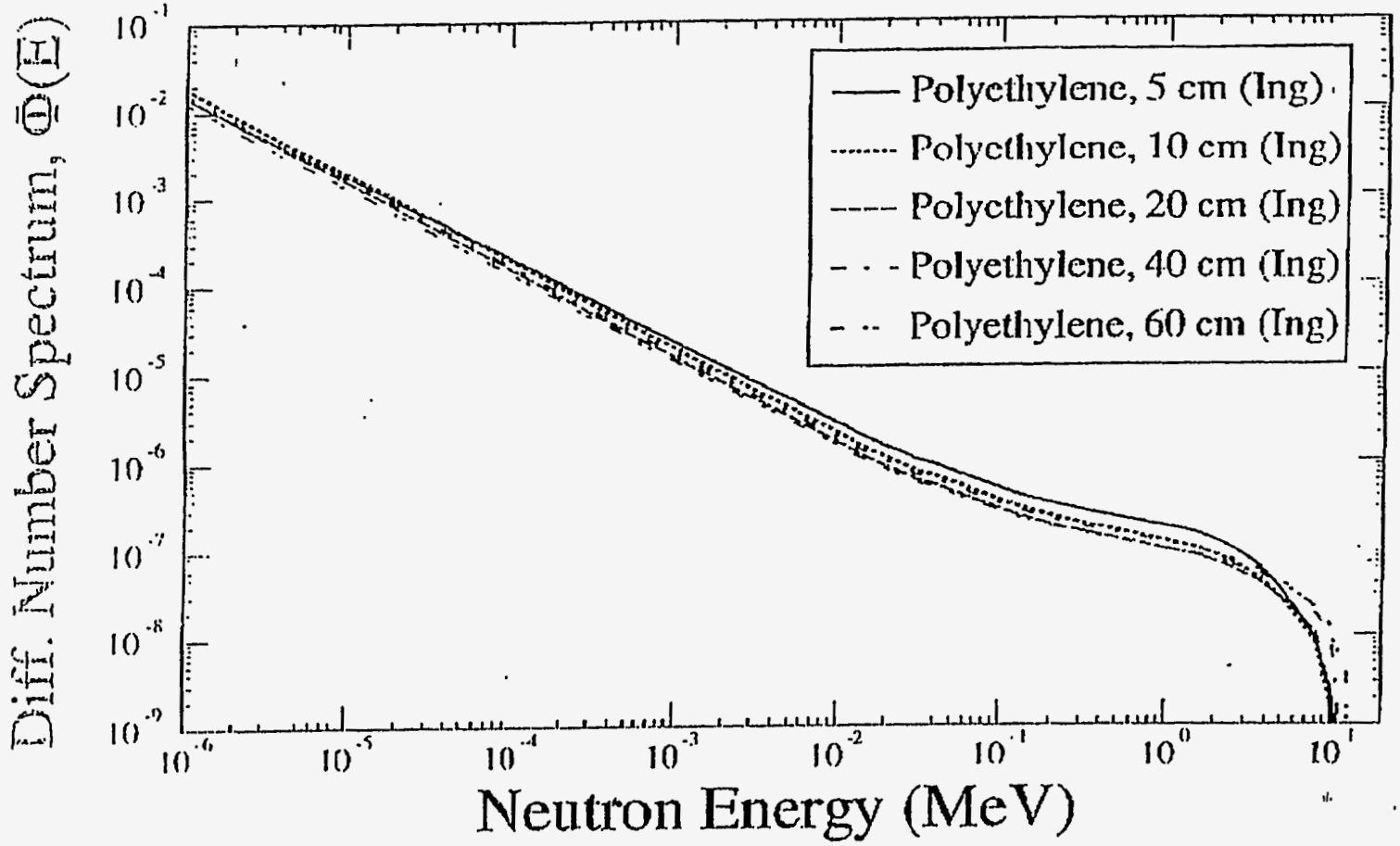

Figure G- 5. Fission neutrons through different thicknesses of polyethylene (Ing & Makra, 1978).

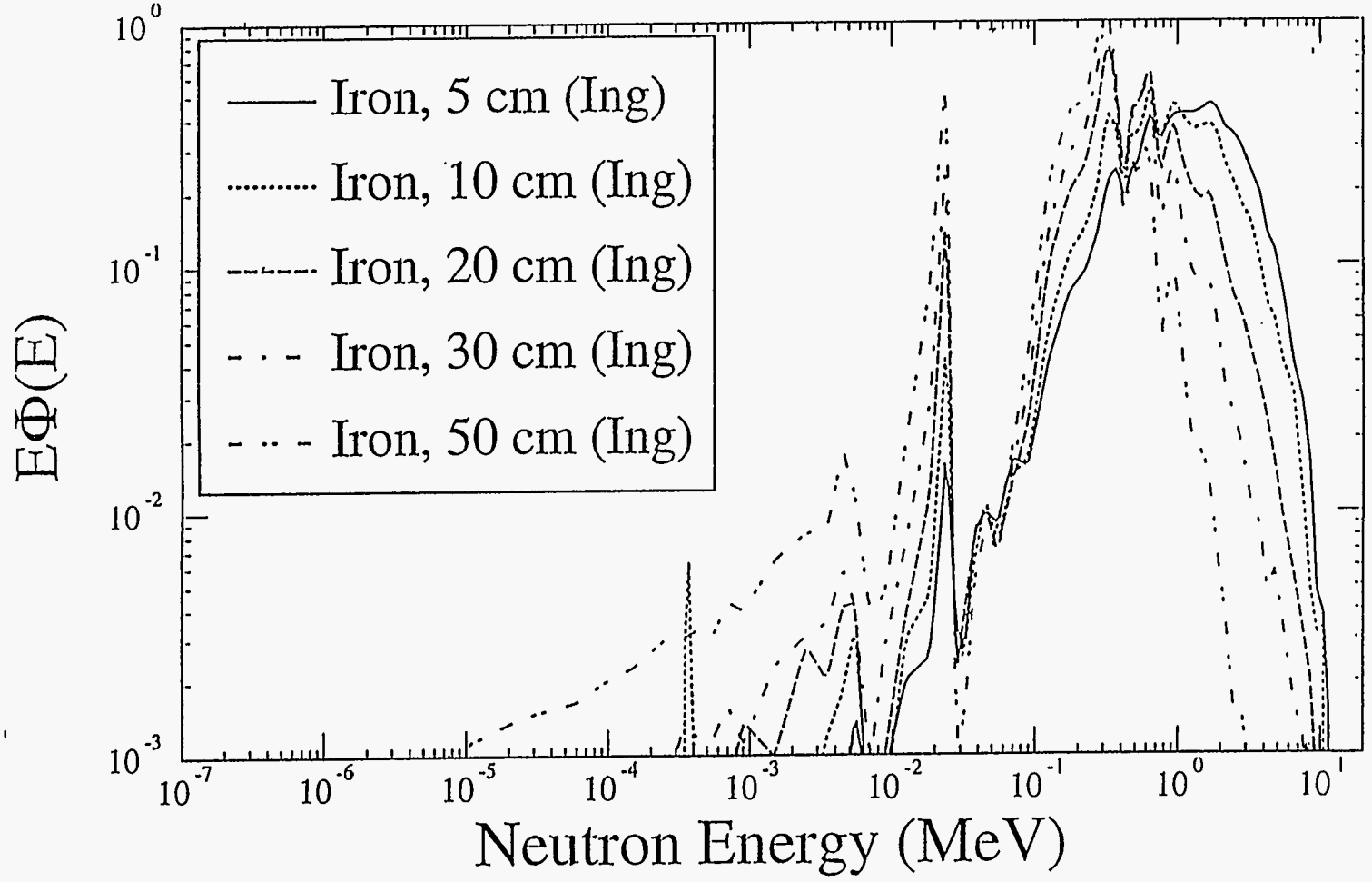

Figure G- 6. Spectra for fission neutrons through different thicknesses of iron (Ing & Makra, 1978).

APPENDIX H: NEUTRON/GAMMA DOSE RATIOS

Table H-1 summarizes the reported neutron and gamma doses. Reported neutron doses were obtained using indium foil results. Reported gamma doses were obtained using Sandia National Laboratorys' TLD badges.

Neutron-to-gamma ratios can be used when only one dosimeter (PNAD or TLD) is retrieved from the accident victim.

Table H- 1. Summary of Reported Neutron and Gamma Doses.

Source	Reference Neutron Dose (rad)	Reported Neutron Dose (rad)	Reported Gamma Dose (rad)	Neutron-to-Gamma Ratio
SHEBA high power	108	114	108	1.06
SHEBA low power	11	12	11	1.09
SHEBA free run	94	104	92	1.13
Bare Godiva	200	273	73	3.74
Godiva through 12 cm lucite	26	32	27	1.19
Godiva through 20 cm concrete	34	32	25	1.28
Godiva through 13 cm steel	N/A[a]	48	19	2.53

[a] Reference dose not made available to NAD23 participants

APPENDIX I: CRITICALITY DOSIMETRY USING BLOOD SODIUM ACTIVITY

1. General

Blood ^{24}Na activity can supply information regarding the total incident neutron flux. However, the dose to the body is strongly dependent upon neutron energy. There is therefore, no simple relation between blood ^{24}Na activity and average whole body neutron dose. If, however, various neutron spectra can be categorized into Fast, Semi-moderated and Moderated spectra, the calculation of average whole body neutron dose becomes a simple matter. In this situation, an uncertainty of ± 30% may apply (Hankins, 1968). Blood-sodium activation has been used to assist in the evaluation of the dose from most criticality accidents.

If an individual might have been involved in a criticality accident and his/her PNAD and TLD were not retrieved, whole-body counting for ^{24}Na can be used as a screening method.

2. Estimation of Whole Body Neutron Dose From Blood Sodium Activation

Test exposures to neutron spectra, ranging from an unmoderated fission spectrum at the ORNL Health Physics Research Reactor (HPRR) (Auxier, 1965) to a thermal spectrum at the Standard Pile (SP) reactor at the Savannah River Plant (Axtmann et al., 1953) have provided factors for converting ^{24}Na activity in blood plasma to a rad dose. For estimates of dose, an assumption of the neutron spectrum must be made from knowledge of the type of critical assembly involved. The broad characterizations used are "Fast", "Semi-moderated" and "Moderated". Conversion factors for these cases are given in Table I-1.

Table I- 1. Blood Sodium Activity-to-Dose Conversion Factors[a]

Spectrum Type	Conversion Factor for Blood Plasma	Conversion Factor for Whole Blood[b]
Fast	1.65×10^5 rad/μCi/ml	3.52×10^5 rad/μCi/ml
Semi-moderated	1.00×10^5 rad/μCi/ml	2.13×10^5 rad/μCi/ml
Moderated	0.75×10^5 rad/μCi/ml	1.60×10^5 rad/μCi/ml

[a] Units are rad absorbed dose due to neutrons per μCi of ^{24}Na per ml of sample.

[b] The ^{23}Na content of blood plasma is approximately 3.2 mg/ml, and the ^{23}Na content of whole blood is approximately 1.5 mg/ml.

The category of "Fast" corresponds to bare metal assemblies regardless of size or shape. Principal criterion is the absence of moderating materials such as water and organic solvents or thick concrete shields. "Semi-moderated" is applicable to process tanks or small vessels filled with either light water or organic solvents, or bare metal assemblies with limited shielding e.g., < 30 cm concrete. "Moderated" refers to reactors, basins, or large tanks containing considerable volumes of moderator.

The ability of this technique to determine personnel dose has been tested in a series of exposures to the Health Physics Research Reactor (HPRR) at Oak Ridge National Laboratory (Auxier, 1965).

The gamma dose can be assessed from the individual's TLD or an estimate can made by using the neutron-to-gamma dose ratio values contained in Appendix H.

3. Orientation of The PNAD and TLD During Exposure

The PNAD and TLD are normally worn on the front of the person. The response of these items is affected by the body's shielding and moderation. It is important to know the direction from which the person was exposed. If a correction for direction of exposure is not made, the dose estimate may be low by a factor of as much as six in the case where the exposure was received from the back of the wearer. The PNAD incorporates a NaF pellet for determining the direction from which the majority of the exposure was received. By comparing the activity of the ^{24}Na in the individual's blood with the activity of the ^{24}Na produced in the NaF pellet, the direction of exposure can be determined.

Table I-2 contains sodium pellet test data obtained during The Fourteenth Nuclear Accident Dosimetry Intercomparison Study (Sims, 1979). The Health Physics Research Reactor was operated in the pulsed mode to simulate nuclear accidents. Different shielding configurations were used to create a variety of neutron spectra. Two of the three phantoms used in the experiments were filled with a saline solution (the third was water filled) such that the ^{23}Na concentrations approximated that in human blood. Pellets containing sodium were placed on the front, side, and back surfaces of the phantoms in order to assess activation as a function of position.

Table I-3 contains the measured-activity ratios between the sodium pellets and the saline solution. Ratio values were calculated using data shown in Table I-2.

The orientation of the PNAD (and TLD) is evaluated as follows:

- Determine the ^{24}Na activity in the NaF pellet contained within the PNAD. This activity should be in units of μCi of ^{24}Na activity per mg of ^{23}Na contained within the pellet. Decay correct the activity to the time of exposure.
- Determine the ^{24}Na activity per ml of whole blood. Decay correct the activity to the time of exposure.

- Calculate the activity ratio ($\frac{pellet[\mu Ci/mg]}{solution[\mu Ci/ml]}$).
- Estimate the type of neutron spectrum involved and determine the orientation of the wearer.
  - ◊ Fast Spectrum:
    - ⇒ If the activity ratio is≥ 0.35, the majority of the neutron fluence came from the front of the wearer.
    - ⇒ If the activity ratio is ≥ 0.20 but <0.35 assume a side exposure.
    - ⇒ If the activity ratio is < 0.20 assume the exposure was from behind.
  - ◊ Semi-moderated Spectrum
    - ⇒ If the activity ratio is≥ 0.80, the majority of the neutron fluence came from the front of the wearer.
    - ⇒ If the activity ratio is ≥ 0.30 but <0.80 assume a side exposure.
    - ⇒ If the activity ratio is < 0.30 assume the exposure was from behind.
  - ◊ Moderated Spectrum
    - ⇒ If the activity ratio is≥ 0.90, the majority of the neutron fluence came from the front of the wearer.
    - ⇒ If the activity ratio is ≥ 0.40 but <0.90 assume a side exposure.
    - ⇒ If the activity ratio is < 0.40 assume the exposure was from behind.

In all cases where the exposure was from behind the wearer, use the blood sodium estimate for neutron exposure and the neutron-to-gamma ratios from Appendix H to estimate a gamma dose.

In all cases where the exposure was from the front, use the neutron dose derived from the indium foil in the PNAD and the gamma dose derived from the TLD. These values will not require any correction for orientation of the wearer.

In all cases where the exposure was from the side, correction of the neutron dose derived from the indium foil is not required. Apply a correction factor (multiplicative) of 1.25 to the gamma dose obtained from the TLD (Friedmann 1991).

Table I- 2. Sodium Pellet Test Data, DOSAR, July 1977.

Pulse #	Shield Type	Spectrum Classification[b]	Sodium Solution, Reference Dosimetry (μCi/ml)[a]	Sodium Pellet Front (μCi/mg)[b]	Sodium Pellet Side (μCi/mg)[b]	Sodium Pellet Back (μCi/mg)[b]
1	15 cm concrete + 5 cm steel	moderated	5.1×10^{-4}	5.5×10^{-4}	3.2×10^{-4}	0.46×10^{-4}
2	none	fast	20.6×10^{-4}	7.8×10^{-4}	6.8×10^{-4}	1.40×10^{-4}
3	20 cm concrete	moderated	4.7×10^{-4}	4.9×10^{-4}	3.4×10^{-4}	0.44×10^{-4}

[a] ^{24}Na activity per ml of solution.
[b] ^{24}Na activity per mg of ^{23}Na contained within the pellet.

Table I- 3. Sodium Activity Ratios As A Function of Position.

Pulse #	Shield Type	Spectrum Type	ACTIVITY RATIO ($\frac{pellet[\mu Ci/mg]}{solution[\mu Ci/ml]}$) Front	Side	Back
1	15 cm concrete + 5 cm steel	Semi-moderated	1.08	0.63	0.09
2	none	fast	0.38	0.33	0.07
3	20 cm concrete	Semi-moderated	1.04	0.73	0.09

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