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Seventeenth Nuclear Accident Dosimetry Intercomparison Study: August 11-15, 1980

R. E. Swaja
R. T. Greene

HEALTH AND SAFETY RESEARCH DIVISION
OAK RIDGE NATIONAL LABORATORY

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Health and Safety Research Division

SEVENTEENTH NUCLEAR ACCIDENT DOSIMETRY INTERCOMPARISON STUDY: AUGUST 11-15, 1980

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Date Published: April 1981

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SEVENTEENTH NUCLEAR ACCIDENT DOSIMETRY INTERCOMPARISON STUDY

AUGUST 11-15, 1980

R. E. Swaja
R. T. Greene

HIGHLIGHTS

The Seventeenth Nuclear Accident Dosimetry Intercomparison Study was conducted August 11-15, 1980, at the Oak Ridge National Laboratory. Nuclear criticality accidents with three different neutron and gamma ray energy spectra were simulated by operating the Health Physics Research Reactor in the pulse mode. Participants from 13 organizations exposed dosimeters set up as area monitors and mounted on phantoms for personnel monitoring. Analysis of experimental results reported by participants showed that less than 60% of the neutron dose measurements using foil activation, thermoluminescent, or sodium activation methods and less than 20% of the gamma dose measurements using thermoluminescent dosimeters met nuclear criticality accident dosimetry guidelines which suggest accuracies of ±25% for neutron dose and ±20% for gamma dose. This indicates that continued development and evaluation of criticality accident dosimetry systems for area and personnel monitoring are required to improve measurement accuracy so that existing standards can be met.

INTRODUCTION

The seventeenth in a series[1-6] of nuclear accident dosimetry (NAD) intercomparison studies was conducted at the Oak Ridge National Laboratory (ORNL) Dosimetry Applications Research (DOSAR) facility during August 11-15, 1980. Participants measured unshielded and shielded neutron and gamma radiation doses greater than 0.1 Gy (10 rads) at air stations and on anthropomorphic phantoms following simulated criticality accidents produced by operating the Health Physics Research Reactor (HPRR)[7] in the pulse mode. These results were compared with those of other participants who made similar measurements under identical conditions and with expected doses based on reactor characteristic

data. In addition to the experimental work, the study included lectures and discussions on relevant subjects such as medical aspects of nuclear accidents, nuclear criticality safety, dosimetry requirements in the nuclear power industry, and calculation of dose from criticality accidents. The program for the entire study is included in Appendix A of this report.

PARTICIPATION

Individual participants in the Seventeenth NAD Intercomparison Study and their affiliations are listed in Appendix B. A total of eighteen different organizations, fourteen domestic and four foreign, were represented by active participants, speakers, or observers. Thirteen agencies actually made measurements during this study with twelve reporting final results. Abbreviations used in this report to identify participant organizations are also included in Appendix B.

DESCRIPTION OF EXPERIMENTS

Nuclear criticality accidents were simulated by operating the HPRR in the pulse mode. To expose the various types of dosimeters to different neutron energy spectra[8,9] and neutron-to-gamma ratios,[2] three separate pulses were performed: an unshielded pulse, a pulse with the dosimeters shielded by 5 cm of steel, and one shielded by 20 cm of concrete. Table 1 is a summary of experimental conditions for the three pulses. In each case, the fission yields were sufficient to provide neutron and gamma doses greater than 0.1 Gy.

Dosimeters were mounted on Bomab[10-11] phantoms and at area monitoring stations (also called air stations) which were located 3 m from the reactor vertical centerline. Figure 1 shows the experimental arrangement of the phantoms and area monitoring stations used for all three pulses. Two phantoms (A and C) were front-facing and one (B) was side-facing relative to the HPRR. Figures 2 and 3 show experimental arrangements for the steel and concrete shielded pulses, respectively, including shield and phantom identifications and orientations.

Sodium concentration data for each of the three phantoms are presented in Table 2. Two of the phantoms (A and B) were filled with a saline

solution such that the sodium concentrations approximated that in human blood. The third phantom (C) was filled with tap water. Samples of the irradiated saline solutions were made available to participants shortly after each pulse for sodium activation analysis.[10-11]

DOSIMETERS USED IN THE INTERCOMPARISON

The general types of radiation dosimeters used by the participants in this intercomparison study are briefly described below. Abbreviations used to identify these dosimeter types in the remainder of this report are also included. The majority of participants used foil activation dosimeters to measure neutron dose and thermoluminescent dosimeters (mostly TLD-700) to measure gamma dose. Participant furnished information regarding analysis techniques and NAD systems used in this study is included in Appendices C through G of this report. Detailed descriptions of nuclear accident dosimeters are available in the literature.[1-3, 5-6, 10-15]

Gamma Dosimeters

1. Thermoluminescent Dosimeter (TLD) — Irradiation of TLD material produces metastable centers which, upon heating, emit light in proportion to the dose received by the material.

2. Radiophotoluminescent Glass Dosimeter (RPL) — When specially prepared glass is exposed to radiation, luminescent centers are produced. Upon subsequent exposure to ultraviolet light, the luminescent centers emit visible light which is proportional to the dose.

Neutron Dosimeters

1. Foil Activation (ACT) — Foils (e.g., gold, copper, indium) exposed to neutrons become radioactive. The activity is measured and the dose obtained from knowledge of the incident spectrum and weighted dose per unit fluence.

2. Threshold Detector Unit (TDU) — Gram quantities of fissionable material (e.g., plutonium, neptunium, uranium) having fission cross sections with thresholds at different neutron energies are enclosed in a boron shield. Threshold Detector Units also generally contain activation foils such as gold and sulfur for additional spectral definition. Neutron fluences within certain energy intervals are determined by measuring foil activities subsequent to exposure. The associated dose is obtained using dose per unit fluence conversion factors for each energy interval.

3. Sodium Activation (Na ACT) — Saline solution samples from irradiated phantoms are analyzed for ^{24}Na activity by any of a variety of counting techniques. The dose received by a phantom is proportional to the specific activity of the solution and orientation of the phantom.

4. Thermoluminescent Dosimeter (TLD) — Two types of TLD material (chips), one sensitive to gammas and the other sensitive to neutrons and gammas, are simultaneously exposed to reactor radiation. The response due to neutrons can be determined after both chips are analyzed. Various shields and absorbers are often placed near the chips to limit their exposure from a given direction to a selected range of neutron energies.

MEASUREMENT RESULTS AND ANALYSIS

Measured data and evaluation results are shown in Tables 3 through 14 of this report. Data analyses in prior intercomparison studies[1-6] have been primarily aimed at determining measurement precision of NAD systems based on experimental results reported by individual participants. Associated measurement accuracy has not been determined because of large uncertainties in data used to calculate doses from pulsed reactor operation. As a result of extensive experimental data analysis,[16] the predictability of doses from pulses has improved sufficiently to permit evaluation of overall NAD performance based on measurement precision and accuracy relative to a reference dose.[17] Reference values given in the following text were calculated as described in Appendix H using the number of fissions per pulse, HPRR characteristic data,[1-2, 16] and known shield attenuation factors.[1-3]

Tables 3-8 summarize final reported results of individual measurements made during the Seventeenth NAD Intercomparison Study. Results of measurements made at air stations for each of the three pulses are shown in Tables 3-5 and include neutron and gamma doses, neutron-to-gamma dose ratios (D_n/D_γ), neutron fluences, and types of detection systems used by the reporting agencies. Tables 6-8 summarize results of individual measurements made on phantoms for each of the three pulses. Data contained in these tables include neutron and gamma doses, sodium activities, and associated detection systems. Although most neutron dose results were reported in terms of kerma (kinetic energy released to

the medium), some participants determined dose due to protons and recoils using conversion factors for central volume element 57 of the cylindrical Auxier phantom.[18] Since these conversion factors include a component due to the $H^1(n,\gamma)H^2$ reaction, this capture gamma contribution was subtracted from reported values to obtain the neutron doses due only to protons and recoils which are shown in the tables.

Table 9 is a summary of average measured neutron doses, experimental standard deviations from the mean, and reference values for measurements made at air stations and on phantoms. Data shown in the table do not include results reported by NTHU who determined the dose due only to thermal neutrons and TPC who reported results in terms of neutron dose equivalent. Measurements at air stations were made using activation methods (foil activation or TDU) for each of the three pulses. Phantom doses were measured using foil activation, TLD, and sodium activation methods. Doses obtained based on kerma and on protons plus recoils [without the $H^1(n,\gamma)H^2$ capture gamma contribution] are within 10% for all air station measurements and within 5% on phantoms for pulse Nos. 1 and 2 with neither method being consistently higher or lower than the other. The low average dose due to protons plus recoils for pulse No. 3 includes results from only two agencies which were significantly lower than results reported by all other participants using various measurement techniques. Neglecting the phantom results for pulse No. 3, the reported data indicate that neutron doses based on kerma and on protons plus recoils are consistent for the measurements made during this NAD study.

Table 10 shows average measured neutron doses normalized to the reference values and associated percent standard deviations from the mean (in parentheses) for several NAD systems. Normalized dose indicates the accuracy of the mean of a particular set of reported results relative to the reference value. Percent standard deviation from the mean is a measure of precision and reflects agreement among individual measurements of the same dose. The averages of reported neutron doses for all dosimeter types (column labelled "All") were within 14 and 27% of the reference values for air station and phantom measurements, respectively, with shielded doses being lower than the reference values by an average

of 14% and unshielded doses being almost equal to the reference values. Standard deviations from the means varied from 12 to 17% (average = 14%) for air station measurements and 17 to 38% (average = 24%) for phantom measurements with air station doses being more precisely measured than corresponding phantom doses for each shield configuration. Average standard deviations for unshielded and shielded measurements were 16 and 21% of the means, respectively, which indicates that unshielded doses were more precisely measured than shielded doses. In general, the composite data show that unshielded neutron doses were measured with more accuracy and precision than shielded doses and that measurements at air stations were more accurate and precise than those on phantoms.

With regard to various neutron dosimeter types used in the study, Table 10 shows that foil activation measurements were within 14 and 42% of the reference values for air station and phantom measurements, respectively, with average unshielded doses being almost equal to the reference values and shielded doses being lower than the references by an average of 18%. Percent standard deviations varied from 12 to 17% for air station measurements and 19 to 67% for phantom doses with air station measurements being more precise than corresponding phantom results for each shield configuration. Average standard deviations from the mean were 17 and 29% for unshielded and shielded foil activation results, respectively. Average neutron doses on phantoms measured using TLD's (mostly TLD-albedo) were within 11% of the reference value for the unshielded case and within 28% of the reference for the shielded cases. Considering only phantom measurements, TLD-measured doses were more accurate than activation-measured values for shielded cases and less accurate for the unshielded case. Associated average standard deviations of 6% for unshielded and 7% for shielded doses indicate that TLD-measured doses were more precisely measured than corresponding values using activation methods. Neutron doses determined using ^{24}Na activation methods were within 12% of the reference values for all three pulses, which is the best overall accuracy exhibited by the dosimetry methods considered in this study. Standard deviations of 8% for unshielded doses and 16% for shielded doses indicate that sodium activation methods were slightly less precise than thermoluminescent dosimeters and more

precise than activation methods. In general, unshielded doses were more accurately and precisely measured than shielded doses for all dosimeter types with sodium activation and thermoluminescent dosimeters exhibiting better overall performance than foil activation methods on phantoms.

Average measured gamma doses, experimental standard deviations from the mean, neutron-to-gamma dose ratios, and reference values are shown in Table 11 for measurements made at air stations and on phantoms. Gamma doses were measured using thermoluminescent (TLD) and radiophotoluminescent (RPL) dosimeters. No standard deviations are included with the RPL-measured data because only one participant reported results using this method. Measured neutron-to-gamma dose ratios are within one standard deviation of the reference values for air station measurements and within two standard deviations at the reference values for phantom measurements.

Average measured gamma doses normalized to the reference values and associated percent standard deviations from the mean for air station and phantom measurements are summarized in Table 12. The composite of all reported gamma dose results (column labelled "All") were within 16 and 6% of the reference values for air station and phantom measurements, respectively, with unshielded doses being within an average of 4% of the reference values and shielded doses being within an average of 10% of the reference values. Standard deviations from the means varied from 10 to 20% for air station measurements and from 26 to 34% for phantom measurements with air station doses being more precisely measured than corresponding phantom doses for each shield configuration. Average standard deviations from the mean were 22% for unshielded and shielded gamma dose measurements which indicates that, on the average, unshielded and shielded gamma doses were measured with equal precision. Thermoluminescent dosimeter results indicate the same trends as the composite data since only one other result was reported for another measurement technique. Gamma doses measured using RPL dosimeters were within 44% of the reference values for all pulses. The RPL results at air stations (average accuracy = 9%) were more accurate than corresponding data measured on phantoms (average accuracy = 32%) for all shield configurations.

Considering the composite of all reported results, Tables 10 and 12 show that the accuracy and precision of neutron and gamma dose measurements at air stations were nearly equal. All average neutron and gamma doses measured at air stations were within 14 and 16% of the reference values, respectively, with associated average standard deviations of 14% (neutron) and 15% (gamma) of the means. With regard to doses measured on phantoms, gamma results were within 6% of the reference values and were more accurate than measured neutron doses which were within 27% of the references. However, neutron doses (average standard deviation = 24%) were more precisely measured than gamma doses on phantoms (average standard deviation = 32%). Measured gamma doses were slightly less accurate than corresponding neutron doses for unshielded cases and somewhat more accurate for shielded cases. Unshielded gamma doses were less precise than corresponding unshielded neutron dose measurements (average standard deviation = 16%) while shielded neutron and gamma dose measurements were equally precise (average standard deviation = 22%). Although average neutron doses were lower than the reference values for all shielded pulses, no similar behavior is exhibited for gamma results.

The averages of neutron fluences measured at air stations are summarized in Table 13. Fluences associated with activation of the various isotopes provide definition of the HPRR radiation spectrum resulting from the three different pulses. This spectral information is valuable in dose determination since the relative contribution of a neutron to the total dose depends upon its energy. Some of the percent standard deviations shown in the table are high because of the small number and large variation of individual measurements available for comparison on several foils. Thermal neutron fluences measured by gold activation exhibited the smallest percent standard deviations ($\leq$26% of the mean) for the three pulses which indicates that participants measured thermal fluence more precisely than any of the other energy ranges. In general, percent standard deviations associated with the fluence measurements are consistent with values obtained during prior intercomparison studies.[1-3]

Ratios of doses measured on phantoms to doses measured at air stations for each pulse are given in Table 14. Doses measured on phantoms were larger than those obtained at air stations by an average of about 10% for neutrons and 80% for gammas for all shield configurations. Gamma dose on a phantom is enhanced relative to air by the $H^1(n,\gamma)H^2$ reaction in the water that fills the phantom. Measured neutron dose is increased relative to air by reflected (albedo) neutrons from the phantom. The large phantom-to-air gamma dose ratio for the steel shield compared to the other cases is a result of steel being the least effective neutron shield used in this NAD study and the large D_n/D_γ ratio for this shield as shown in Table 11. Measured phantom-to-air dose ratios given in Table 14 are within one standard deviation of values calculated using reference doses and are consistent with data obtained during previous intercomparison studies.[1-2]

Table 15 contains information which can be related to the effectiveness of various shield configurations used during the intercomparison. For example, the table shows that a 5-cm steel shield placed 1 m from the HPRR reduced the neutron dose at 3 m to 62% (23.8/38.2) of its unshielded value. Also, the steel shield more effectively attenuated gamma rays than neutrons relative to the unshielded case while the concrete shield exhibited the opposite effect. Measured normalized doses are within one standard deviation of the reference values in every case and are consistent with results obtained during previous intercomparison studies.[1,3] Although these data are not directly related to dosimeter intercomparisons, they are presented here for information to the DOSAR staff and reactor shield designers.

COMPARISON TO PREVIOUS INTERCOMPARISON STUDIES

Results presented in the preceding text for the Seventeenth NAD Intercomparison Study are consistent with the following statements which are based on analyses of results from the previous sixteen studies:[1-2, 17]

1. The precision of dose measurements based on the composite data has not improved as a function of time. The average

percent standard deviations for unshielded neutron and gamma dose measurements made during the Seventeenth NAD Intercomparison Study (16 and 22%, respectively) are almost equal to the average of all sixteen previous intercomparisons (15% for unshielded neutron and 20% for unshielded gamma).[1-2] Average percent standard deviations for shielded neutron and gamma dose measurements (∿22%) are consistent for all intercomparison studies to date.

2. Neutron doses from unshielded pulses have been measured more precisely than those from shielded pulses.
3. Unshielded neutron dose measurements have been more precise than unshielded gamma dose measurements.
4. Shielded neutron and gamma dose measurements have been equally precise.
5. Considering precision and accuracy, overall performances of neutron and gamma dosimeters are better for unshielded pulses than for shielded pulses.
6. Neutron and gamma doses measured at air stations are more accurate and precise than corresponding measurements made on phantoms.

CONCLUSIONS

Nuclear criticality accident dosimetry guidelines[19-20] suggest accuracies of ±25% for neutron dose and ±20% for gamma dose. The NAD study results show that slightly more than half (58%) of the neutron doses measured using foil activation, thermoluminescent, or sodium activation methods and less than 20% of the gamma doses measured using TLD's satisfy these criteria relative to the reference values. Considering precision and accuracy, none of the three neutron dosimetry methods satisfied the NAD guidelines for air station or phantom measurements during all of the pulses. Gamma dose precision alone, as reflected by percent standard deviation from the mean for TLD-measured results, exceeded the guideline limits in approximately 70% of the measurements. Although

some individual participants reported results within one standard deviation from the mean for most of their measurements, no organization reported doses within one standard from the mean for all of their measurements. It is therefore concluded that continued development and evaluation of nuclear accident dosimetry systems for area and personnel monitoring are required to improve measurement techniques so that existing standards can be met.

RECOMMENDATIONS

Some problems were encountered in the analysis and interpretation of measured results because all participants did not include units or evaluation methods when reporting results on forms provided for this study. These difficulties may have contributed to the relatively large standard deviations from the mean for some of the average doses and fluences presented in this report. To ensure clarity and uniformity, subsequent NAD intercomparison studies should include sessions or documentation to define explicitly the convention to be used to report measured results.

Although agencies who have participated in annual NAD intercomparison studies have benefited from this experience, they generally are not able to meet present accuracy criteria. Since intercomparison study participants are among the best at determining doses from nuclear accidents, it is presumed that many less experienced organizations who have never participated in such a study and have responsibilities in the area of criticality dosimetry also cannot satisfy existing NAD guidelines. The DOSAR staff should consider designing, developing, and conducting a periodic NAD training course in addition to the annual intercomparison study. The ultimate objective of this education and training would be to standardize NAD techniques and increase the technical competence of persons responsible for nuclear accident dosimetry throughout the nuclear industry. The HPRR, the DOSAR facility, and the dosimetry expertise at ORNL provide excellent facilities and personnel for conducting such

a course. Government sponsorship (e.g., Department of Energy or Nuclear Regulatory Commission) should be sought on the basis of upgrading and standardizing nuclear accident dosimetry methods in the nuclear industry.

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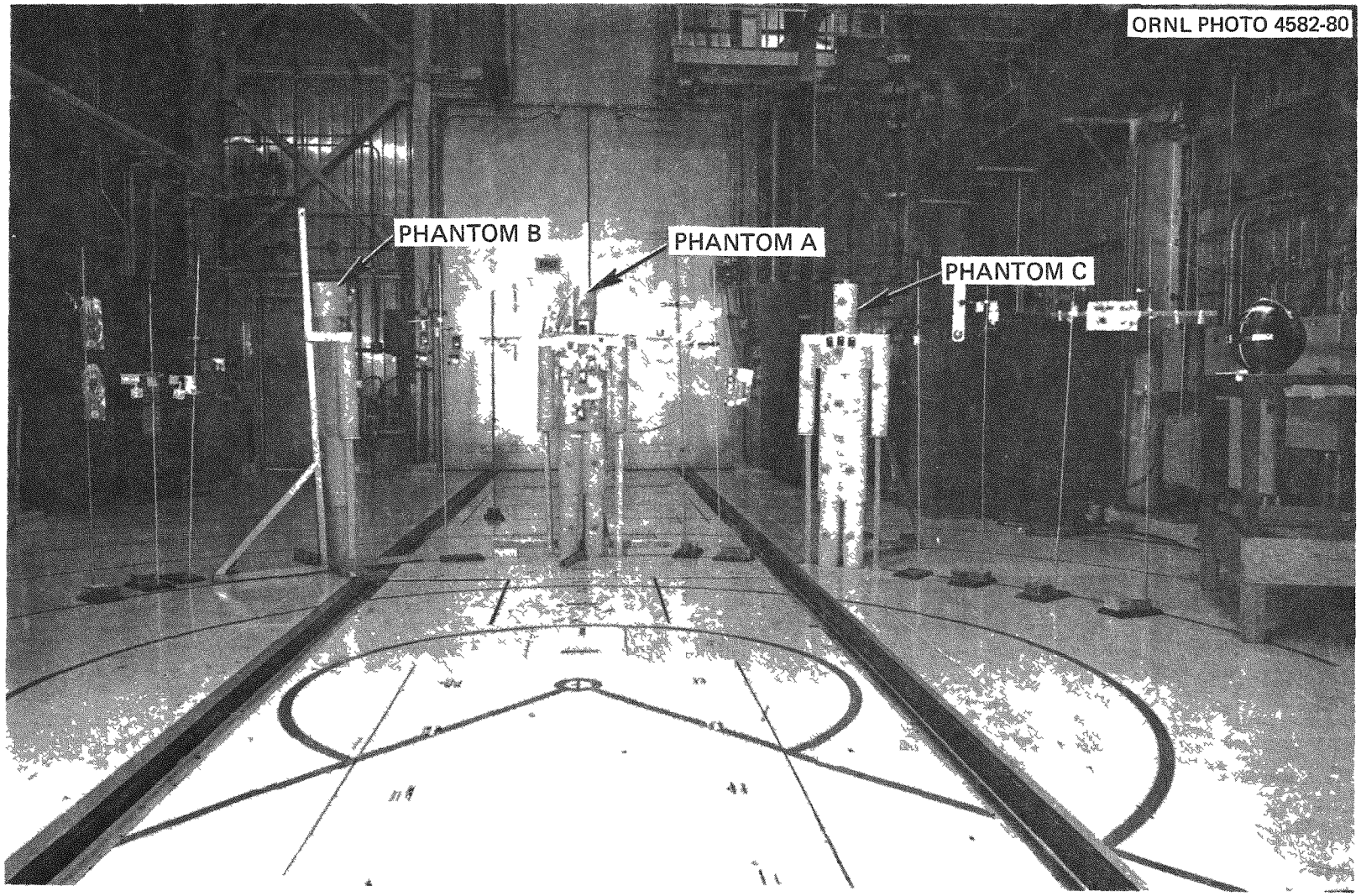

Fig. 1. Experimental arrangement of phantoms and area monitoring stations for all pulses.

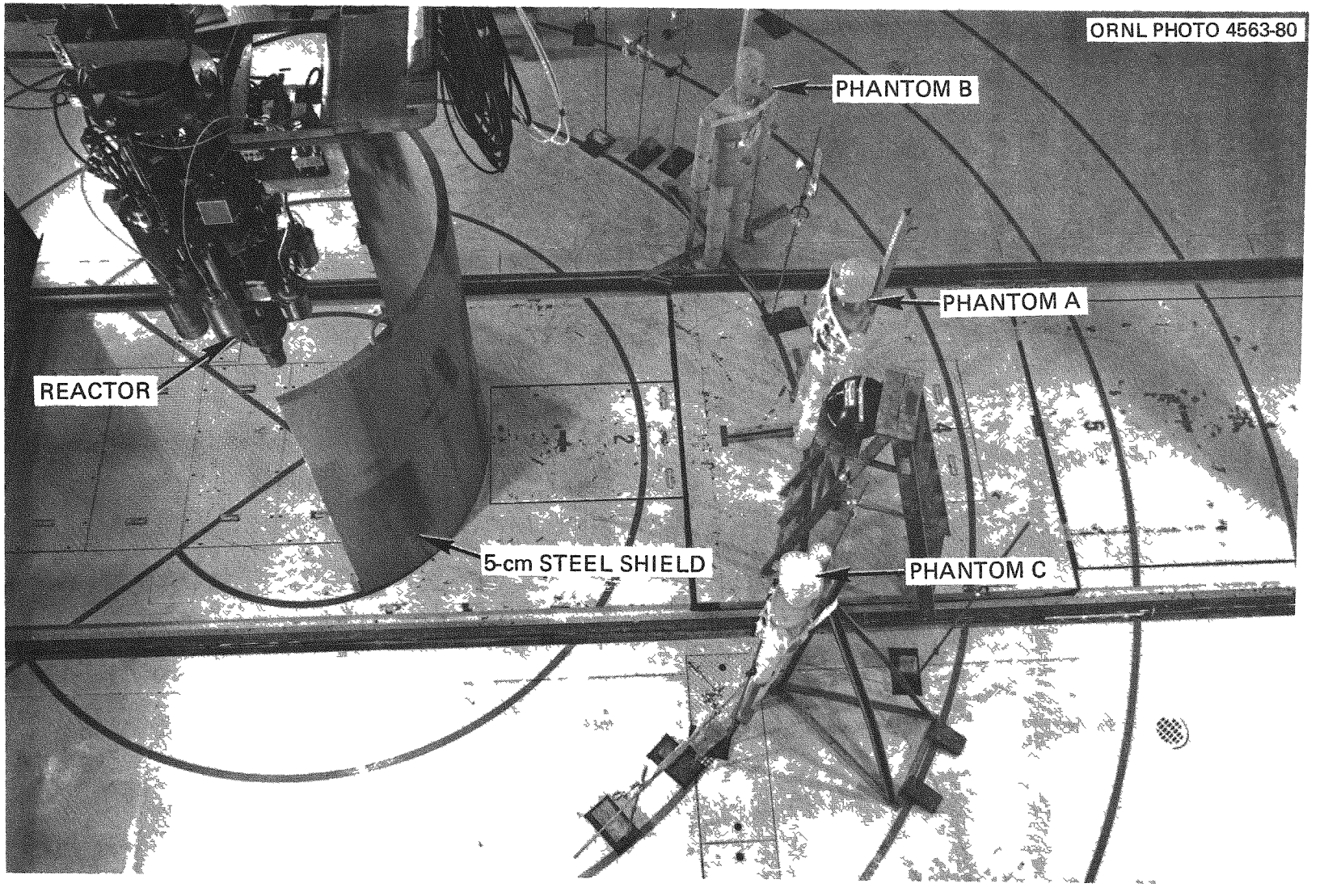

Fig. 2. Experimental arrangement for pulse No. 2 (5-cm steel shield).

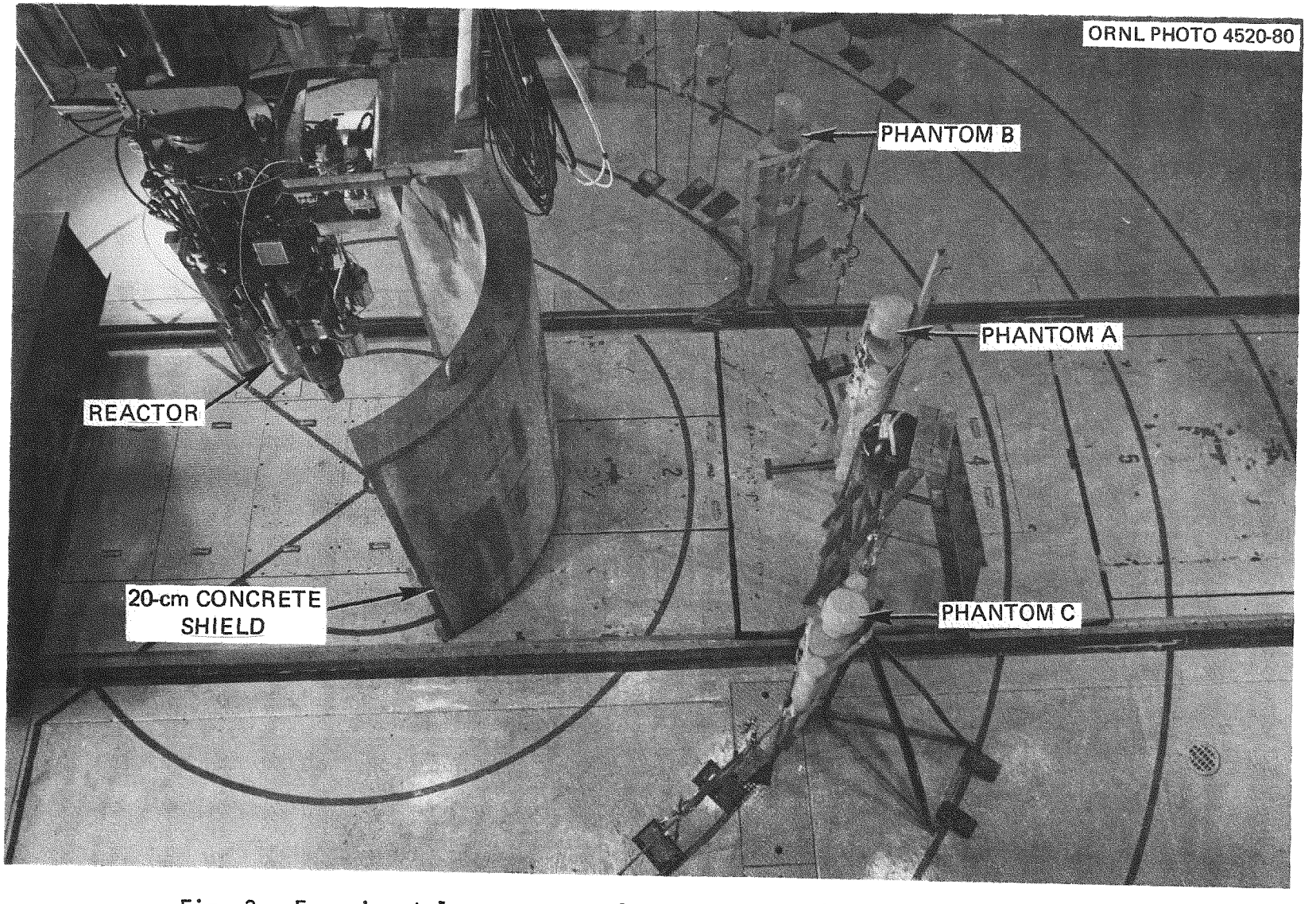

Fig. 3. Experimental arrangement for pulse No. 3 (20-cm concrete shield).

Table 1. Summary of experimental conditions

Pulse No.	Date	Eastern Daylight Time	Pulse yield, 10^{16} fissions	Shield	Reactor to shield distance, m	Reactor to dosimeter distance,[a] m
1	8/12/80	1036	7.61	None		3
2	8/13/80	1120	4.54	5-cm steel	1	3
3	8/14/80	1112	5.18	20-cm concrete	1	3

[a]Dosimeters associated with area monitoring stations were located 3 m from the centerline of the HPRR. The centerlines of phantoms on which dosimeters were exposed were 3 m from the centerline of the HPRR.

Table 2. Sodium concentrations in Bomab phantoms[a] irradiated during the Seventeenth Nuclear Accident Dosimetry Intercomparison Study

Pulse No.	Sodium concentration, mg/mℓ		
	Phantom A[b]	Phantom B[c]	Phantom C[b]
1	1.53	1.56	Tap water
2	1.55	1.60	Tap water
3	1.55	1.60	Tap water

[a]See Fig. 1 for locations and orientations of phantoms A, B, and C.

[b]Front-facing phantom.

[c]Side-facing phantom.

Table 3. Measurements at air stations for pulse No. 1
Yield: 7.61 (10^{16}) fissions, Shield: None

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	D_n/D_γ	10^{-10} × Neutron fluence, n/cm^2								Detector system	
				Au, thermal	Pu, >1 keV	Np, >0.75 MeV	U, >1.5 MeV	S, >2.5 MeV	Cu	In, thermal	In, fast	Neutron	Gamma
BAPL	260	53	4.9					1.40	2.90	0.16	6.90	ACT	TLD
BAPL	273[b]		5.2									ACT	
BW	270							1.50[c]	3.50	1.40	8.00	ACT	
DOSAR	284	54	5.3	0.81	11.00	8.10	3.80	2.60				TDU	RPL
DOSAR		45	6.3										TLD
GAC	290[b]			0.76				1.70		3.95		ACT	
LASL	291	45	6.5	0.83				2.35[c]	8.70	1.47	4.60	ACT	TLD
LASL	240[b]		5.3									ACT	
ORGDP	294			1.08				1.79[c]			4.70	ACT	
ORGDP	261[b]											ACT	
REECO	342	54	6.3					2.02	1.70	1.34	8.10	ACT	TLD
REECO	392[b]		7.3									ACT	
USN	274	45	6.1									ACT	TLD
Y-12	306			1.20				1.60				ACT	

[a]Neutron doses represent tissue kerma based on dosimeter data alone unless otherwise indicated and are given in units of 10^{-2} Gy (1 rad).

[b]Protons plus recoils with the $H^1(n,\gamma)H^2$ component subtracted for volume element 57 of the cylindrical Auxier phantom.

[c]>2.9 MeV.

Table 4. Measurements at air stations for pulse No. 2
Yield: 4.54 (10^{16}) fissions, Shield: 5-cm steel

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	D_n/D_γ	10^{-10} × Neutron fluence, n/cm^2								Detector system	
				Au, thermal	Pu, >1 keV	Np, >0.75 MeV	U, >1.5 MeV	S, >2.5 MeV	Cu	In, thermal	In, fast	Neutron	Gamma
BAPL	92	14	6.6					0.39	1.50	0.53	2.50	ACT	TLD
BAPL	95[b]		6.8									ACT	
DOSAR	111	12	9.2	0.40	4.80	3.20	1.30	0.77				TDU	RPL
DOSAR		13	8.5										TLD
LASL	125	18	6.9	0.40				0.28[c]	6.00	0.63	1.70	ACT	TLD
LASL	94[b]		5.2									ACT	
REECO	107	14	7.6					0.19	0.50	1.61	1.10	ACT	TLD
REECO	116[b]		8.3									ACT	
USN	124	16	7.8									ACT	TLD

[a]Neutron doses represent tissue kerma based on dosimeter data alone unless otherwise indicated and are given in units of 10^{-2} Gy (1 rad).

[b]Protons plus recoils with the $H^1(n,\gamma)H^2$ component subtracted for volume element 57 of the cylindrical Auxier phantom.

[c]>2.9 MeV.

Table 5. Measurements at air stations for pulse No. 3
Yield: 5.18 (10^{16}) fissions, Shield: 20-cm concrete

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	D_n/D_γ	10^{-10} × Neutron fluence, n/cm^2								Detector system	
				Au, thermal	Pu, >1 keV	Np, >0.75 MeV	U, >1.5 MeV	S, >2.5 MeV	Cu	In, thermal	In, fast	Neutron	Gamma
BAPL	48	13	3.7					0.13	1.40	0.92	1.10	ACT	TLD
BAPL	50[b]		3.8									ACT	
BW	47							0.20[c]	1.90	1.80	0.90	ACT	
DOSAR	38	22	1.7	1.10	1.50	0.95	0.49	0.33				TDU	RPL
DOSAR		16	2.4										TLD
GAC	45[b]			0.90				0.21		0.45		ACT	
LASL	38	22	1.7	1.50				0.31[c]	0.60	0.98	0.70	ACT	TLD
LASL	31[b]		1.4									ACT	
REECO	46	16	2.9					0.53	0.60	0.70	2.60	ACT	TLD
REECO	50[b]		3.1									ACT	
USN	31	17	1.8										

[a]Neutron doses represent tissue kerma based on dosimeter data alone unless otherwise indicated and are given in units of 10^{-2} Gy (1 rad).
[b]Protons plus recoils with the $H^1(n,\gamma)H^2$ component subtracted for volume element 57 of the cylindrical Auxier phantom.
[c]>2.9 MeV.

Table 6. Phantom studies for pulse No. 1
Yield: 7.61 (10^{16}) fissions, Shield: None

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	$10^8 \times {}^{24}$Na activity[b] Bq/mℓ	Bq/mg	Basis for estimating Neutron dose	Gamma dose
DOSAR	238[c]	63[c]	33.9	22.2	Na ACT	RPL
DOSAR		88[d]				TLD
GAC	271				ACT	
KK	314	82			TLD[e]	TLD
KK	325[f]				TLD[e]	
KK	358[f]	66			TLD[g]	TLD
LASL	317	45			ACT	TLD
LASL	249[f]				ACT	
LASL	285[c]		38.7	25.3	Na ACT	
LASL	288[d]		26.2	16.8	Na ACT	
NTHU	6.2[h]				TLD[i]	TLD[i]
REECO	350	145			TLD[e]	TLD
REECO	367[c]	61			ACT	TLD
REECO	427[c,f]				ACT	
REECO	240[d]	66			ACT	TLD
REECO	279[d,f]				ACT	
TPC	318[j]	100			TLD[e]	TLD
USN	323	91			ACT	TLD
Y-12	266[c]		51.8	33.8	Na ACT	

[a]Neutron doses represent tissue kerma unless otherwise indicated and are given in units of 10^{-2} Gy (1 rad).

[b]Sodium concentrations in phantoms (mg/mℓ) are given in Table 2. Activity units are Bq (3.7×10^{10} Bq = 1 Ci).

[c]Front-facing phantom.

[d]Side-facing phantom.

[e]TLD-albedo dosimeter.

[f]Protons plus recoils with the $H^1(n,\gamma)H^2$ component subtracted for volume element 57 of the cylindrical Auxier phantom.

[g]Karlsruhe 30-cm sphere with TLD-albedo dosimeters. A quality factor of 9.4 was used for this spectrum.

[h]Thermal neutron dose equivalent in rem reported by the participating agency. A quality factor of 9.4 was used to convert to dose.

[i]Powder $CaSO_4$:Dy.

[j]Reported in rem by the participating agency. A quality factor of 9.4 was used to convert to dose.

Table 7. Phantom studies for pulse No. 2
Yield: 4.54 (10^{16}) fissions, Shield: 5-cm steel

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	$10^8 \times {}^{24}Na$ activity[b] Bq/mℓ	Bq/mg	Basis for estimating Neutron dose	Gamma dose
DOSAR	160[c]	29	23.0	14.8	Na ACT	RPL
DOSAR	124[d]	36	13.6	8.5	Na ACT	TLD
KK	129	29			TLD[e]	TLD
KK	126[f]				TLD[e]	
KK	124[f]	23			TLD[g]	TLD
LASL	135	18			ACT	TLD
LASL	111[f]				ACT	
LASL	103[c]		20.8	13.4	Na ACT	
LASL	96[d]		12.6	7.9	Na ACT	
NTHU	2.6[h]	24			TLD[i]	TLD[i]
ORGDP	129				ACT	
ORGDP	155[f]				ACT	
ORGDP	173[c]		24.9	16.1	Na ACT	
REECO	125	45			TLD[e]	TLD
REECO	123[c]	40			ACT	TLD
REECO	136[c,f]				ACT	
REECO	91[d]	22			ACT	TLD
REECO	103[d,f]				ACT	
TPC	148[j]	32			TLD[e]	TLD
USN	166				ACT	

[a]Neutron doses represent tissue kerma unless otherwise indicated and are given in units of 10^{-2} Gy (1 rad).

[b]Sodium concentrations in phantoms (mg/mℓ) are given in Table 2. Activity units are Bq (3.7×10^{10} Bq = 1 Ci).

[c]Front-facing phantom.

[d]Side-facing phantom.

[e]TLD-albedo dosimeter.

[f]Protons plus recoils with the $H^1(n,\gamma)H^2$ component subtracted for volume element 57 of the cylindrical Auxier phantom.

[g]Karlsruhe 30-cm sphere with TLD-albedo dosimeters. A quality factor of 9.0 was used for this spectrum.

[h]Thermal neutron dose equivalent in rem reported by the participating agency. A quality factor of 9.0 was used to convert to dose.

[i]Powder $CaSO_4$:Dy.

[j]Reported in rem by the participating agency. A quality factor of 9.0 was used to convert to dose.

Table 8. Phantom studies for pulse No. 3
Yield: 5.18 (10^{16}) fissions, Shield: 20-cm concrete

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	$10^8 \times {}^{24}Na$ activity[b] Bq/mℓ	Bq/mg	Basis for estimating Neutron dose	Gamma dose
DOSAR	59[c]	39	8.5	5.5	Na ACT	TLD
DOSAR	55[d]	33	6.0	3.8	Na ACT	TLD
GAC	60				ACT	
KK	44	24			TLD[e]	TLD
KK	40[f]				TLD[e]	
KK	39[f]	15			TLD[g]	TLD
LASL	32	22			ACT	TLD
LASL	23[f]				ACT	
LASL	49[c]		9.4	6.0	Na ACT	
LASL	54[d]		5.9	3.7	Na ACT	
NTHU	2.5[h]	25			TLD[i]	TLD[i]
REECO	50	35			TLD[e]	TLD
REECO	13	34			ACT	TLD
REECO	14[f]				ACT	
TPC	120[j]	31			TLD[e]	TLD
USN	67				ACT	
Y-12	56[c]		8.1	5.2	Na ACT	

[a]Neutron doses represent tissue kerma unless otherwise indicated and are given in units of 10^{-2} Gy (1 rad).

[b]Sodium concentrations in phantoms (mg/mℓ) are given in Table 2. Activity units are Bq (3.7×10^{10} Bq = 1 Ci).

[c]Front-facing phantom.

[d]Side-facing phantom.

[e]TLD-albedo dosimeter.

[f]Protons plus recoils with the $H^1(n,\gamma)H^2$ component subtracted for volume element 57 of the cylindrical Auxier phantom.

[g]Karlsruhe 30-cm sphere with TLD-albedo dosimeters. A quality factor of 9.0 was used for this spectrum.

[h]Thermal neutron dose equivalent in rem reported by the participating agency. A quality factor of 9.0 was used to convert to dose.

[i]Powder $CaSO_4$:Dy.

[j]Reported in rem by the participating agency. A quality factor of 9.0 was used to convert to dose.

Table 9. Summary of results of neutron dose measurements at air stations and on phantoms

Pulse No.	Dosimeter location	Neutron dose, 10^{-2} Gy[a]						
		Activation[b]			TLD[c]	Sodium	All[d]	Reference
		Kerma	Protons plus recoils[e]	All ACT[f]				
1	Air	290.1 ± 25.5[g]	291.2 ± 59.2	290.5 ± 39.4			290.5 ± 39.4	291
2	Air	111.8 ± 13.6	101.7 ± 12.4	108.1 ± 13.3			108.1 ± 13.3	111
3	Air	41.3 ± 6.7	44.0 ± 9.0	42.4 ± 7.4			42.4 ± 7.4	50
1	Phantom	303.6 ± 49.2	318.3 ± 95.3	309.1 ± 63.5	336.8 ± 20.7	268.0 ± 21.7	305.8 ± 52.0	303
2	Phantom	128.8 ± 26.8	126.2 ± 23.8	127.7 ± 24.0	126.0 ± 2.2	131.2 ± 34.1	128.3 ± 23.4	144
3	Phantom	43.0 ± 25.1	18.5 ± 6.4	34.8 ± 23.3	43.2 ± 5.0	54.6 ± 3.65	43.7 ± 16.7	60

[a]Values are average doses based on data shown in Tables 3-5 (air) and Tables 6-8 (phantom) and are given in units of 10^{-2} Gy (1 rad).

[b]Includes foil activation and threshold detector unit data.

[c]NTHU and TPC results not included.

[d]Average of results for all measurement methods.

[e]Dose due to protons and recoils with the $H^1(n,\gamma)H^2$ component subtracted for volume element 57 of the cylindrical Auxier phantom.

[f]Includes data based on kerma and on protons plus recoils.

[g]One standard deviation from the mean.

Table 10. Normalized average measured neutron doses and associated percent standard deviations[a]

Pulse No.	Shield	Dosimeter location	Normalized dose (percent standard deviation)[b]			
			Activation[c]	TLD[d]	Sodium	All[e]
1	None	air	1.00(14)			1.00(14)
2	5-cm steel	air	0.97(12)			0.97(12)
3	20-cm concrete	air	0.86(17)			0.86(17)
1	None	phantom	1.02(20)	1.11(6)	0.88(8)	1.01(17)
2	5-cm steel	phantom	0.89(19)	0.88(2)	0.91(26)	0.89(18)
3	20-cm concrete	phantom	0.58(67)	0.72(12)	0.91(7)	0.73(38)

[a]Based on data shown in Table 9.

[b]Average reported measured dose divided by the reference dose (percent of standard deviation from the mean).

[c]Includes results based on kerma and on protons plus recoils.

[d]NTHU and TPC results not included.

[e]Includes results for all measurement methods.

Table 11. Summary of results of gamma dose measurements at air stations and on phantoms

Pulse No.	Dosimeter location	Gamma dose, 10^{-2} Gy[a]				D_n/D_γ	
		TLD	RPL[b]	All[c]	Reference	Measured[d]	Reference
1	Air	48.4 ± 4.7[e]	54	49.3 ± 4.8	47	5.9 ± 1.0	6.2
2	Air	15.0 ± 2.0	12	14.5 ± 2.2	13	7.4 ± 1.4	8.5
3	Air	16.8 ± 3.3	22	17.7 ± 3.6	21	2.4 ± 0.6	2.4
1	Phantom	81.2 ± 27.7	63	79.5 ± 26.9	78	3.8 ± 0.4	3.9
2	Phantom	29.9 ± 9.1	39	30.8 ± 9.0	29	4.2 ± 0.4	5.0
3	Phantom	27.4 ± 7.0	39	28.7 ± 7.6	27	1.5 ± 0.5	2.2

[a]Values are average doses based on data shown in Tables 3-5 (air) and Tables 6-8 (phantoms) and are given in units of 10^{-2} Gy (1 rad).

[b]Used by only one participant.

[c]Average of results for all measurement methods.

[d]Average of all reported neutron dose measurements from Table 9 divided by the average of all reported gamma dose measurements.

[e]One standard deviation from the mean.

Table 12. Normalized average measured gamma doses and associated percent standard deviations[a]

Pulse No.	Shield	Dosimeter location	Normalized dose (percent standard deviation)[b]		
			TLD	RPL[c]	All[d]
1	None	air	1.03(10)	1.15	1.05(10)
2	5-cm steel	air	1.16(13)	0.92	1.12(15)
3	20-cm concrete	air	0.80(20)	1.05	0.84(20)
1	None	phantom	1.04(34)	0.81	1.02(34)
2	5-cm steel	phantom	1.03(30)	1.34	1.06(29)
3	20-cm concrete	phantom	1.01(26)	1.44	1.06(26)

[a]Based on data shown in Table 11.

[b]Average reported measured dose divided by the reference value (percent of standard deviation from the mean).

[c]Used by only one participant.

[d]Includes results for all measurement methods.

Table 13. Summary of neutron fluence measurements at air stations

Pulse No.	10^{-10} × Average neutron fluence, n/cm^2 [a]								
	Au, thermal	Pu,[b] >1 keV	Np,[b] >0.75 MeV	U,[b] >1.5 MeV	S, >2.5 MeV	S, >2.9 MeV	Cu	In, thermal	In, fast
1	0.94 ± 0.19[c]	11.00	8.10	3.80	1.86 ± 0.47	1.88 ± 0.43	4.20 ± 3.09	1.66 ± 1.39	6.46 ± 1.72
2	0.40[d]	4.80	3.20	1.30	0.45 ± 0.29	0.28[b]	2.67 ± 2.93	0.92 ± 0.60	1.77 ± 0.70
3	1.17 ± 0.30	1.50	0.95	0.49	0.30 ± 0.17	0.26 ± 0.08	1.12 ± 0.64	0.97 ± 0.51	1.32 ± 0.86

[a]Average fluences are based on data given in Tables 3-5.

[b]Results reported by only one participant.

[c]One standard deviation from the mean.

[d]Based on two fluence measurements which produced the same result.

Table 14. Comparison of doses measured on phantoms with those measured at air stations

Pulse No.	Shield	Ratio of phantom dose to air station dose			
		Neutron		Gamma	
		Measured[a]	Reference	Measured[b]	Reference
1	None	1.1 ± 0.2[c]	1.0	1.6 ± 0.6	1.7
2	5-cm steel	1.2 ± 0.3	1.3	2.1 ± 0.7	2.2
3	20-cm concrete	1.0 ± 0.6	1.2	1.6 ± 0.5	1.3

[a]Based on data given in Table 9 for all reported dose measurements.

[b]Based on data given in Table 11 for all reported dose measurements.

[c]One standard deviation.

Table 15. Normalized dose at air stations

Pulse No.	Pulse yield, 10^{16} fissions	Shield[a]	Normalized dose, 10^{-2} Gy/10^{16} fissions[b]			
			Neutron[c]		Gamma[d]	
			Measured	Reference	Measured	Reference
1	7.61	None	38.2 ± 5.2[e](1.00)[f]	38.2(1.00)	6.5 ± 0.6(1.00)	6.2(1.00)
2	4.54	5-cm steel	23.8 ± 2.9(0.62)	24.4(0.64)	3.2 ± 0.5(0.49)	2.9(0.47)
3	5.18	20-cm concrete	8.2 ± 1.4(0.22)	9.6(0.25)	3.4 ± 0.7(0.52)	4.0(0.64)

[a]Shields were located 1 m from the reactor vertical centerline.

[b]Calculated using the average of all reported doses which were measured at 3 m from the reactor vertical centerline.

[c]Based on data given in Table 9.

[d]Based on data given in Table 11.

[e]One standard deviation.

[f]Normalized dose divided by the unshielded value.

APPENDIX A

PROGRAM

Seventeenth Nuclear Accident Dosimetry Intercomparison Study

August 11-15, 1980

Date	Time	Activity
August 11	9:00 AM	Welcome
	9:15 AM	Orientation, H. W. Dickson (ORNL)
	9:30 AM	Review of Nuclear Accident Dosimetry Intercomparison Program, H. W. Dickson (ORNL)
	10:00 AM	Assignment of equipment and workspace
	10:15 AM	Tour of Control Room and HPRR Building
		LUNCH
	1:00 PM	Lecture: Principles of Nuclear Criticality Safety G. R. Handley (Y-12)
	2:00 PM	Seminar: D. E. Hankins (LLNL)
	3:00 PM	Preparation for Pulse No. 1
	7:00 PM	Evening social — Alexander Motor Inn
August 12	8:00 AM	Final setup of dosimetry for Pulse No. 1
	9:00 AM	Group photo
	9:10 AM	Observation of pulse operation of HPRR
	10:00 AM	Pulse No. 1 (unshielded)
	10:30 AM	Collect dosimeters
		LUNCH
	1:30 PM	Lecture: Medical Aspects of Nuclear Accidents R. C. Ricks (ORAU)
	2:00 PM	Analysis of data and preparation for Pulse No. 2
August 13	8:00 AM	Final setup of dosimeters for Pulse No. 2
	9:00 AM	Review of participant dosimetry system (speaker to be selected from among participants)
	10:00 AM	Pulse No. 2 (5-cm steel shield)
	11:00 AM	Collect dosimeters
		LUNCH
	1:00 PM	Discussion: Dosimetry Requirements in the Nuclear Power Industry
	2:00 PM	Analysis of data and preparation for Pulse No. 3

Date	Time	Activity
August 14	8:00 AM	Final setup of dosimeters for Pulse No. 3
	10:00 AM	Pulse No. 3 (20-cm concrete shield)
	10:30 AM	Collect dosimeters
		LUNCH
	1:00 PM	Lecture: Calculation of Dose from Criticality Accidents G. E. Whitesides (ORNL)
	2:00 PM	Analysis of data
August 15	9:00 AM	Discussion: Methods of Reporting Dose Due to Nuclear Accidents
	10:00 AM	Presentation of dose estimates
	11:00 AM	Final critique

APPENDIX B

List of Participants, Speakers, and Observers

Name	Affiliation
M. W. Weisfield	Bettis Atomic Power Laboratory P. O. Box 79 West Mifflin, Pennsylvania 15122 (BAPL)*
Raymond L. Vinton	Babcock and Wilcox Commercial Nuclear Fuel Plant P. O. Box 800 Lynchburg, Virginia 24505 (BW)
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*Abbreviation by which this participant organization is referred to in this report.

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Name	Affiliation
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Boyd Gose G. R. Handley David Jones Alan King	Union Carbide Corporation Y-12 Plant P. O. Box Y Oak Ridge, Tennessee 37830 (Y-12)

APPENDIX C

PARTICIPANT FURNISHED DOSIMETRY SYSTEM — ANALYSIS TECHNIQUE DESCRIPTION

Bettis Atomic Power Laboratory (BAPL)

All neutron dose rates based on dosimeter results are calculated using the conversion factors in Savannah River Report DP-1006 (Revision 1). One dosimeter was exposed for each of the three test pulses. Neutron spectra corrections to the doses were made for the unshielded pulse and the pulse with 20-cm of concrete for shielding using the HPRR spectra supplied by Oak Ridge. All results required are supplied on the data sheets provided by Oak Ridge. These results include KERMA (kinetic energy released in the medium) and "element 57" dose (dose at the central volume element of a tissue equivalent phantom). The "KERMA" and "ELEMENT 57" doses were calculated from expressions provided by your facility.

Two gamma doses are reported. Those not in parentheses are the doses obtained directly from the Li^7F TLD chips in the dosimeters. These doses have not been corrected for the thermoluminescence caused by neutron interaction in the Li^7F TLD chips. This factor has the effect of causing the gamma dose to be overestimated unless the neutron contribution is accounted for. The gamma doses in parentheses have been corrected for the neutron response of the Li^7F TLD chips.

In addition to the requested doses, the reaction rates as requested on the Oak Ridge data sheet are provided. These reaction rates were calculated as indicated on the data sheet.

APPENDIX D

PARTICIPANT FURNISHED DOSIMETRY SYSTEM — ANALYSIS TECHNIQUE DESCRIPTION

Dosimetry Applications Research Facility (DOSAR)
Oak Ridge National Laboratory

Threshold Detector Unit

The Threshold Detector Unit (TDU) contains fission foils of plutonium, neptunium, and uranium enclosed in a B_4C sphere whose thickness is 1 cm. In addition, bare gold and cadmium-covered gold foils are used to determine thermal neutron fluence. A sulfur pellet is used to determine neutron fluence above 2.5 MeV.

Foil	Energy threshold	Cross section
Au	At thermal energy	98 b
Pu	1 keV	1.8 b
Np	750 keV	1.6 b
U	1.5 MeV	0.55 b
S	2.5 MeV	0.23 b

Energy interval	Dose conversion factor $Gy\text{-}n^{-1}\text{-}cm^2$
At thermal energy	2.4×10^{-13}
0.001-0.75 MeV	1.4×10^{-11}
0.75-1.5 MeV	2.4×10^{-11}
1.5-2.5 MeV	3.0×10^{-11}
2.5 MeV and above	3.7×10^{-11}

Gamma-Rays

Gamma ray dose was measured using radiophotoluminescent (RPL) and thermoluminescent (TLD) dosimeters. Toshiba FD-P8-1 glass blocks containing 50% Al $(PO_3)_3$, and 50% $LiPO_3$ with $AgPo_3$ added were used. In the mixed neutron and gamma ray field of the HPRR, these blocks were exposed inside a ^6Li container lined with 0.040 in. tin for a more independent energy response at low-photon energies. Harshaw TLD-700

LiF chips were also used to obtain the gamma ray dose. The TLD-700 dosimeters contain 0.007% ^{6}Li and 99.993% ^{7}Li and have negligible neutron sensitivity.

Analysis of TDU's

The fluence, ϕ, in each energy interval is determined from the activation or fission product activity produced in each of the foils. The following equations are used to solve for the fluence:

In general,

$$\phi = \frac{C\ (cpm) \times P \times 10^{10}}{g \times N(t)} ,$$

where

C = count rate measured from the foil,

P = perturbation factor to correct for attenuation in the boron shield,

g = the weight of the foil in grams,

N(t) = factor including decay correction.

$$R = \phi_{Pu} = \frac{C \times 10^{10}}{3 \times N(t)}$$

$$S = \phi_{Np} = \frac{C \times 1.15 \times 10^{10}}{0.4 \times N(t)}$$

$$T = \phi_{u} = \frac{C \times 1.1 \times 10^{10}}{5 \times N(t)}$$

$$U = \sigma_{s} = \frac{C \times 1.3 \times 10^{7}}{e^{-\lambda t}}$$

One needs to correct the sulfur count for ^{31}Si activity ($T_{1/2}$ = 2.62 h) that competes with ^{32}P during the first 10 to 12 h.

The fast neutron dose determination (tissue kerma in free air) is made by multiplying the fluence in each energy interval by the appropriate dose conversion factor and summing the individual doses.

$$D\ (Gy) = [1.4\ (R-S) + 2.4\ (S-T) + 3.0\ (T-U) + 3.7\ (U)] \times 10^{-11}$$

The thermal neutron dose determination is found in a similar manner by

$$D\ (Gy) = 2.4 \times 10^{-13}\ \phi_{n_{th}},$$

where

$$\phi n_{th} = 10.24 \times 10^{5} \left[\frac{C_{bare}}{e^{-\lambda t}} - \frac{C_{Cd\ covered}}{e^{-\lambda t}} \right].$$

Analysis of Na Activation

The activation of blood sodium produces ^{24}Na, which can readily be counted in a NaI well counter. This activity may be determined using the equation:

$$A\ (Bq/mg) = \frac{C\ (cpm) \times 0.017\ (min/s)}{Eff \times \rho(mg/ml) \times vol\ (ml) \times e^{-\lambda t} \times (s^{-1}/Bq)}$$

For a phantom facing the neutron source, the dose can be determined using the equation:

$$D\ (Gy) = 0.1076 \times 10^{8} \times A\ (Bq/mg)$$

or for a phantom with its side to the source, the dose is given by

$$D\ (Gy) = 0.1454 \times 10^{8} \times A\ (Bq/mg).$$

These are empirically determined factors found from dosimetry studies at the HPRR and may not apply to other sources.

APPENDIX E

PARTICIPANT FURNISHED DOSIMETRY SYSTEM — ANALYSIS TECHNIQUE DESCRIPTION

Kernforschungszentrum Karlsruhe (KK)

Neutron doses on phantoms were measured using TLD-albedo dosimeters and the Karlsruhe single sphere albedo system described in *Nuclear Instruments and Methods*, 175 (1980), pp. 180-2. Quality factors of 9.4 and 9.0 were assumed for the unshielded and shielded pulses, respectively, for the HPRR spectra.

APPENDIX F

PARTICIPANT FURNISHED DOSIMETRY SYSTEM — ANALYSIS TECHNIQUE DESCRIPTION

Reynolds Electrical and Engineering Company (REECO)

Nevada Test Site (NTS) Albedo Neutron Dosimeter

The personnel neutron dosimeter used at the (NTS) is of the design described in Dale Hankins report, LA-5261, "A Small, Inexpensive Albedo Neutron Dosimeter." It consists of pairs of ^{6}LiF and ^{7}LiF encased in cadmium and polyethylene. It is worn close to the body so that neutrons reflected from the body (albedo neutrons) significantly increase the response of the dosimeter per unit of direct neutron fluence.

The response of this dosimeter can be calibrated to personnel neutron dose by several methods. Hankins described one method in several of his papers — the use of the response of a BF_3 tube in different moderators in relation to predetermined calibration factors to graph a calibration curve. Another method mentioned was to determine the calibration factors for each neutron spectrum by measuring the dosimeter response vs a known neutron spectrum dose. This was accomplished for the NTS dosimeter at DOSAR in the Fifth and Sixth Personnel Neutron Dosimetry Intercomparisons. These low dose tests provided the calibration factors for two of the neutron exposures of this NAD intercomparison — unmoderated and concrete-moderated reactor spectra. Other spectral data indicated that the calibration factor of the dosimeter when exposed to the steel-moderated spectrum was on the order of 30% higher than the unmoderated reactor calibration factor.

NTS Personnel Dosimeter - NAD Description and Use

Nuclear accident dosimetry devices provided by REECO at the NTS consist of Hurst Threshold Detector Units (TDU) and NTS personnel dosimeters in fixed positions at possible nuclear criticality accident sites. In addition, of course, the latter are also worn by NTS personnel.

The NTS personnel dosimeter is a combination security credential, film dosimeter, and nuclear accident dosimeter. The NAD portion contains one teflon TLD-700 microdosimeter. The neutron dosimetry components are as follows: one 10.5 gm sulfur pellet; one Cd-clad and one unclad indium

foil 0.0254 cm thick; and one 0.0381 cm thick, Cd clad, Cu foil. All Cd covers are 0.0381 cm thick.

With the TDU it is possible to determine the neutron fluences having energies within the following energy intervals: <0.5 eV (considered thermal), 1.0 keV to 0.75 MeV, 0.75 MeV to 1.5 MeV, 1.5 MeV to 3.0 MeV, and >3.0 MeV. The neutron fluences (n/cm^2) and the kerma in equivalent tissue rad are calculated from induced activities of the NTS NAD components as follows:

<0.5 eV

$$F_1 \text{ (n/cm}^2\text{)} = 144\ (0.98\ S_1 - S_2) \tag{1}$$

$$D_1 \text{ (rad)} = 2.5 \times 10^{-11}\ F\ . \tag{2}$$

0.5 eV to 1.0 keV (epithermal)

$$F_2 \text{ (n/cm}^2\text{)} = 280\ S_2 \tag{3}$$

$$D_2 \text{ (rad)} = 2.71 \times 10^{-12}\ F \tag{4}$$

1.0 keV to 1.0 MeV

Case I:

$$F_3 \text{ (n/cm}^2\text{)} = 74.23\ (210.88\ S_3 + S_2) \tag{5a}$$

Case II:

$$F_3 \text{ (n/cm}^2\text{)} = 17.79\ (7259.5\ S_3 - S_2) \tag{5b}$$

Case III:

$$F_3 \text{ (n/cm}^2\text{)} = 704.6\ (1075.4\ S_3 - S_2) \tag{5c}$$

$$D_3 \text{ (rad)} = 1.62 \times 10^{-9}\ F_3 \tag{6}$$

>1.0 MeV

$$F_4 \text{ (n/cm}^2\text{)} = 3.27 \times 10^{5}\ S_4 \tag{7}$$

1.0 MeV to 3.0 MeV

$$D_4 \text{ (rad)} = 2.87 \times 10^{-9} (F_4 - F_5) \tag{8}$$

>3.0 MeV

$$F_5 \text{ (n/cm}^2\text{)} = 6.37 \times 10^6 S_5 \tag{9}$$

$$D_5 \text{ (rad)} = 4.00 \times 10^{-9} F_5 \tag{10}$$

Each subscripted S in the above equations denotes a specific activity in disintegrations per gram-minute. In order of increasing value, the subscripts refer to ^{116m}In, ^{116m}In (Cd), ^{64}Cu (Cd), ^{115m}In (Cd), and ^{32}P from sulfur.

Selection of the appropriate equation from 5a, 5b, and 5c is made on the basis of observing the configuration and composition of the nuclear assembly and the scattering and/or shielding elements affecting the spectrum at the dosimeter. All of these elements are viewed in the light of previous experience with the dosimeter in neutron fields from several critical assemblies.

APPENDIX G

PARTICIPANT FURNISHED DOSIMETRY SYSTEM — ANALYSIS TECHNIQUE DESCRIPTION

Los Alamos Scientific Laboratory (LASL)

PND and LASL CD Dosimeters

The Personnel Dosimetry Dosimetry (PND) packet is used to estimate the neutron dose to the person carrying the device. The PND packets are issued to persons working in areas where a nuclear criticality accident is possible. The LASL Criticality Dosimetry (LASL CD) packet is an area monitor which is provided at various LASL locations where a potential exists for a nuclear accident. For this intercomparison study, the LASL CD packet was used for in-air measurements of neutron and gamma doses. The PND packet was used for all phantom studies.

Neutron Dosimetry

In Table G-I are shown the threshold detectors that are used in the PND and LASL CD dosimeters. The particular reactions of interest and the applicable energy ranges for fluence determinations are shown. The threshold detectors are treated as if they were ideal detectors with exact thresholds, T_{th}, and with only one cross-section, $\bar{\sigma}$. This is possible only if one knows the shape of the neutron energy spectrum. Then the effective cross-section for the detector is defined by:

$$\bar{\sigma} = \frac{\int_{E_o}^{\infty} \frac{dn}{dE}(E)\sigma(E)dE}{\int_{E_{th}}^{\infty} \frac{dN}{dE}(E)dE} , \qquad (1)$$

where

$\bar{\sigma}$ = the effective cross-section for the detector,

$\frac{dN}{dE}(E)$ = the irradiating neutron energy function,

$\sigma(E)$ = the detector cross-section function,

E_o = the absolute threshold energy for the detector, and

E_{th} = the effective threshold energy that one chooses for the detector.

For this intercomparison study, appropriate neutron spectra for HPRR were available to determine the effective cross-sections for each burst. The neutron fluence, ϕ, in each energy interval is determined from the activation produced in each foil along with the appropriate cross-sections.

The fast neutron dose determination (first collision dose or kerma in tissue) is made by multiplying the fluence in each energy region by the appropriate fluence-to-dose conversion factor and summing the individual doses. Fluence-to-dose conversion factors were determined by a similar weighting technique described by Eq. 1. The thermal and epithermal neutron doses are determined in a like manner.

Photon Dosimetry

Photon dose was measured using radiophotoluminescent (RPL) and TLD dosimeters.

Blood Sodium Activation

Neutron dose estimates were made from the sulfur fluence/blood sodium activation ratios for each burst. Dose estimates were made for the phantom facing the neutron source and with its side to the source.

Table G-1. PND and LASLCD dosimeter threshold foils

No.	Foil type	Packet location(s)	Energy range	Nuclear reaction of interest
1	Cadmium covered indium	PND and LASLCD	1-9 MeV	$^{111}_{49}In + ^{1}_{0}n \rightarrow ^{115m}_{49}In + ^{1}_{0}n'$
2	Sulfur tablet	PND and LASLCD	2.9-9 MeV	$^{32}_{16}S + ^{1}_{0}n \rightarrow ^{32}_{15}P + ^{1}_{1}p$
				$^{32}_{15}P \xrightarrow{\beta-} ^{32}_{16}S$
3	Cadmium covered copper	PND and LASLCD	0.5 eV-1 MeV	$^{63}_{29}Cd + ^{1}_{0}n \rightarrow \left(^{64}_{29}Cu\right)^{*} \xrightarrow[\beta+]{} ^{64}_{28}Ni + \gamma's$
4	Bare indium cadmium covered indium	PND and LASLCD	Thermal and epithermal	$^{115}_{49}In + ^{1}_{0}n \rightarrow \left(^{116m}_{49}In\right)^{*} \xrightarrow[\beta-]{} ^{116}_{50}Sn + \gamma's$
5	Bare gold cadmium covered gold	LASLCD	Thermal and epithermal	$^{197}_{79}Au + ^{1}_{0}N \rightarrow \left(^{198}_{79}Au\right)^{*} \xrightarrow[\beta-]{} ^{198}_{80}Hg + \gamma's$

APPENDIX H

REFERENCE NEUTRON AND GAMMA DOSE CALCULATIONS

Reference doses at area monitoring stations were calculated based on the number of fissions per pulse and HPRR characteristic data as shown in Table H-1. The fission yield for each pulse was determined by measuring the ^{32}P beta activity in a 22 gram sulfur pellet located at a fixed position near the reactor core.[H1] Reference neutron doses at 3 m from the reactor were calculated for the unshielded and concrete shielded pulses by relating the number of fissions to the unshielded tissue kerma at an area monitoring station using dose vs distance curves which have been developed at the DOSAR facility over 16 years of operational experience.[H2] Neutron dose for the 5-cm steel shield was calculated by applying a shield attenuation factor of 0.64 for pulse operation to the unshielded neutron dose[H3] and adjusting the result to account for the fission yield of pulse 2. Gamma doses were obtained by dividing calculated neutron doses by neutron-to-gamma dose ratios at 3 m from the reactor for the various shields.[H2,H3]

Reference doses on phantoms were calculated by multiplying the neutron and gamma doses at air stations (Table H-1) by air-to-phantom dose conversion factors for the various shield configurations as shown in Table H-2. Air-to-phantom neutron and gamma dose conversions were based on previous NAD intercomparison study experience.[H4,H5]

Table H-1. Reference neutron and gamma doses at air stations

Pulse No.	Shield[a]	Fission yield,[b] 10^{16} fissions	Neutron dose,[c] 10^{-2} Gy	D_n/D_γ [d]	Gamma dose, 10^{-2} Gy
1	None	7.61	291	6.2	47
2	5-cm steel	4.54	111[e]	8.5	13
3	20-cm concrete	5.18	50	2.4	21

[a]Steel and concrete shields were located 1 m from the reactor centerline.

[b]Based on sulfur pellet activation analyses.

[c]Dose at 3 m calculated using dose vs distance curves and the number of fissions per pulse.

[d]Neutron-to-gamma dose ratio at 3 m from the reactor.

[e]Based on a measured steel shield attenuation of 0.64 relative to the unshielded dose.

Table H-2. Reference neutron and gamma doses on phantoms[a]

Pulse No.	Shield	Air-to-phantom conversion[b]		Dose on phantom, 10^{-2} Gy[a]	
		Neutron	Gamma	Neutron	Gamma
1	None	1.04	1.66	303	78
2	5-cm steel	1.30	2.20	144	29
3	20-cm concrete	1.20	1.28	60	27

[a]Phantoms were located 3 m from the reactor.

[b]Based on previous NAD intercomparison study results.

[c]Phantom dose obtained by multiplying the dose at an air station given in Table H-1 by the appropriate air-to-phantom conversion.

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