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EIGHTEENTH NUCLEAR ACCIDENT DOSIMETRY INTERCOMPARISON STUDY: August 10-14, 1981

R. E. Swaja
C. S. Sims
R. T. Greene

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EIGHTEENTH NUCLEAR ACCIDENT DOSIMETRY INTERCOMPARISON STUDY

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HIGHLIGHTS

The Eighteenth Nuclear Accident Dosimetry Intercomparison Study was conducted August 10-14, 1981, 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 showed that about 56% of the reported neutron doses measured using foil activation, thermoluminescent, or sodium activation methods and about 53% of the gamma doses measured using thermoluminescent methods met nuclear accident dosimetry guidelines which suggest accuracies of ±25% for neutron dose and ±20% for gamma dose. The greatest difficulties in measuring accident doses occurred in radiation fields with large fractions of low energy neutrons and a high gamma component (>40%). Results of this study indicate that continued accident dosimetry intercomparisons are necessary to test dosimetry systems and training programs are needed to improve the technical competence of evaluating personnel.

INTRODUCTION

The eighteenth in a series[1-3] of nuclear accident dosimetry (NAD) intercomparison studies was conducted at the Oak Ridge National Laboratory's (ORNL) Dosimetry Applications Research (DOSAR) Facility during August 10-14, 1981. 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 nuclear accidents produced by operating the Health Physics Reactor (HPRR)[4] in the pulse mode. These results were compared with those of the participants who made similar measurements under identical conditions and with reference doses[5] based on reactor characteristic data. In addition to the experimental intercomparison, the study included lectures and discussions on relevant subjects such as calculation of dose from nuclear accidents, medical aspects of nuclear accidents, radiation dose determination based on chromosome aberrations, biological effects of radiation, and problems associated with accident monitoring at participating facilities. The program for the entire intercomparison is included in Appendix A of this report. This study was approved for 8 units of continuing education credit (No. 81-26) by the American Board of Health Physics.

PARTICIPATION

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

DESCRIPTION OF EXPERIMENTS

Nuclear accidents were simulated by operating the HPRR in the pulse mode. To expose dosimeters to different neutron energy spectra and neutron-to-gamma dose ratios, three separate pulses were performed: an unshielded pulse, a pulse with the dosimeters shielded by 12 cm of Lucite, and one shielded by 20 cm of concrete. The latter configuration was not known to participants prior to preliminary dose estimation but was made known for final dosimeter evaluation. 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 ringstands in air to simulate area monitoring stations and on the fronts (side facing the reactor) of Bomab phantoms for personnel monitoring. Area stations and the centerlines of phantoms were located 3 m from the vertical centerline of the HPRR. Figure 1 shows the arrangement of the reactor, phantoms, and area monitoring stations used for the unshielded pulse. This is also the basic experimental configuration used for all three pulses. Figures 2 and 3 show experimental arrangements for the Lucite and concrete shielded pulses, respectively, including shield identifications and orientations.

All phantoms were arranged with their fronts facing the reactor, and phantom A was filled with a saline solution with a sodium concentration approximately equal to that found in human blood (1.51 mg/ml). Samples of the irradiated saline solution were made available to participants for sodium activation analysis[6-7] after each pulse. Phantoms B and C were filled with tap water.

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 activation-based systems 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 Appendices C through G of this report. Detailed descriptions of nuclear accident dosimetry systems and methods are available in the literature.[8-9]

Gamma Dosimeters

Thermoluminescent Dosimeters (TLD) - All gamma dosimeters used in this study were based on thermoluminescent properties of certain materials (LiF, CaF, $CaSO_4$). Metastable centers are produced when these materials are irradiated and, upon heating, light is emitted in proportion to the absorbed dose.

Neutron Dosimeters

1. Neutron Activation Systems (ACT) - Some materials (e.g., gold, copper, indium, sulfur) become radioactive when exposed to neutrons. By measuring the activity of exposed foils, neutron fluences over differential energy ranges can be estimated for the incident spectrum. Associated neutron doses can be obtained by applying fluence-

to-dose conversion factors to the estimated fluences and summing over the range of energies encompassed by the activation foils. Some activation systems also use foils made of fissionable materials (e.g., plutonium, neptunium, uranium) which have fission cross sections with thresholds at different neutron energies. These systems are called Threshold Detector Units (TDU's)[3] and are generally used for area monitoring.

2. Thermoluminescent Dosimeters (TLD) - two types of thermoluminescent material (chips), one sensitive to gammas (^{7}LiF) and the other sensitive to neutrons and gammas (^{6}LiF), are simutaneously exposed to the nuclear accident radiation fields. The response due to neutrons can be determined after both chips are analyzed. Various shields and absorbers are often near the chips to limit their exposure from a given direction to a selected range of neutron energies.

3. Sodium Activation (NaACT) - Samples from irradiated, saline-filled, 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 activity, per unit volume of solution and orientation of the phantom.

4. Human Hair Activation (HACT) - Samples of human hair are analyzed for ^{32}p activity following irradiation.

REFERENCE DOSIMETRY

Calculated neutron and gamma reference doses in air and on phantoms are given in Tables 2 and 3, respectively. Reference neutron doses in air given in Table 2 were obtained using fission yields measured by sulfur pellet activation analysis and calculated dose-per-fission conversion factors at 3 m from the reactor for the various HPRR energy spectra.[5] Calculated neutron doses in air are given in terms of wet tissue kerma,[10] which was the convention used by most participating agencies, and element 57 absorbed dose with the capture gamma [primarily due to the $^{1}H(n,\gamma)^{2}H$ reaction] component excluded. Element 57 refers to the central volume element of a cylindrical phantom[11] used to calculate the radiation dose distribution in a tissue equivalent volume exposed to an external neutron field. Neutron dose in this volume element is the highest of all volume elements considered in the mathematical model and represents the expected maximum measured value for each exposure conducted in this study. Element 57 reference doses average 16% higher than corresponding kerma values for air station measurements. Reference gamma doses in air were obtained by dividing neutron kerma in air by the neutron-to-gamma dose ratio at 3 m from the reactor.[12]

The reference neutron and gamma doses on phantoms given in Table 3 were calculated by multiplying doses in air by air-to-phantom conversion factors. The indicated factors are the ratio of phantom-to-air dose based on measured results from the first seventeen NAD intercomparisons.[1-3] Neutron phantom-to-air conversions were applied to kerma values only since element 57 data represent absorbed in a particular interior volume element of a tissue equivalent phantom. Element 57 neutron doses average 8% higher than corresponding kerma values for the phantom measurements considered in this study.

MEASUREMENT RESULTS AND ANALYSIS

Measured data and evaluation results are shown in Tables 4 through 20 of this report. Tables 4-6 give results of preliminary dose estimates obtained during the intercomparison study. The remaining tables are based on final results reported by participants after detailed evaluation at their facilities. The following analysis is primarily aimed at comparing results of individual participants, comparing measured doses to calculated reference values, and evaluating the performance of participants relative to regulatory criteria for nuclear accident dosimetry.

Preliminary Results

Preliminary neutron and gamma dose estimates obtained during the intercomparison study are summarized in Tables 4 and 5 for air station and phantom measurements, respectively. The shield configuration for pulse 3 (concrete shield) was unknown to participants prior to presentation of these preliminary data. Reference neutron doses are given in terms of wet kerma which was the convention used by all participants who reported preliminary results. Neutron dose estimates were based on activation dosimetry - foil activation, threshold detector units, and sodium activation (phantom measurements only). Gamma doses were measured using thermoluminescent dosimeters (mostly TLD-700) for all air station and phantom measurements.

Monitoring standards for nuclear accident dosimetry systems suggest that neutron and gamma doses be determined with an accuracy of ±50% within 24 hours after an accident.[13] Table 6 summarizes the performance of participants' preliminary results relative to this standard. Considering all preliminary data for the three pulses, about 84% of the neutron measurements and approximately 68% of the gamma measurements were within ±50% of the reference values. For pulses 1 (unshielded) and 3 (concrete shield), all reported preliminary neutron dose estimates met the subject criteria and almost 90% of the gamma dose measurements met the subject criteria. However, only 54% of the neutron doses and none of the gamma doses were within ±50% of the reference values for pulse 2 (Lucite shield). Mean neutron energy and neutron-to-gamma dose ratio for the radiation field associated with this pulse are significantly lower than corresponding values for the unshielded and concrete shielded pulses. These results indicate that some participants had difficulty measuring neutron and gamma doses in mixed radiation fields which contain large numbers of low energy neutrons and a large gamma component (>40%). Tables 4 and 5 show that almost all reported preliminary results for pulse 2 were higher than reference values which is consistent with results observed in previous NAD intercomparisons.[3] The fact that the shield configuration for pulse 3 was unknown to participants prior to the presentation of preliminary dose estimates apparently had no effect on the accuracy of the reported results since all neutron doses and 86% of the gamma doses satisfied regulatory criteria for this case.

Final Results

Tables 7-12 summarize final reported results of individual measurements made during the Eighteenth NAD Intercomparison Study. Results of measurements made at air stations for each of the three pulses are shown in Tables 7-9 and include neutron and gamma doses, neutron-to-gamma dose ratios (D_n/D_γ), neutron fluences determined by foil activation methods, and types of detection systems used by the reporting agencies. Tables 10-12 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. Since almost all neutron dose results were reported in terms of wet tissue kerma, reference values used in the subsequent analysis are based on this convention.

Average measured neutron doses, experimental standard deviations from the mean, and reference values for measurements made at air stations and on phantoms are summarized in Table 13. 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, or sodium activation methods.

Table 14 shows average measured neutron doses normalized to the kerma reference values and associated percent standard deviations from the mean (in parenthesis) based on data given in Table 13. 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 labeled "All") indicate that doses were more accurately measured for unshielded relative to shielded pulses for air station and phantom locations. Measured neutron doses for unshielded pulses averaged about 0.92 times reference values for air and phantom measurements compared to corresponding averages of 1.18 and 1.74 times reference values for the concrete and Lucite shielded pulses, respectively. These results also indicate that participant neutron dose measurement accuracy decreased with decreasing mean energy of the incident neutron spectrum; i.e., with increasing spectral softness. Standard deviations from the means varied from 10 to 38% (average = 21%) for air station measurements and from 19 to 26% (average = 23%) for phantom measurements. Average standard deviations for unshielded and shielded measurements were 14 and 26% of the means, respectively, which indicates that unshielded neutron 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 which is consistent with performance observed in prior NAD intercomparisons.[14-15]

With regard to various neutron dosimeter types used in the study, Table 14 shows that average foil activation measurements varied from 0.92 to 1.79 times reference values for air station and phantom measurements with average unshielded doses being about 0.93 times the reference values and shielded doses averaging approximately 1.48 times the references. At both locations, measured neutron doses become more discrepant (less accurate) with increasing spectral softness. Percent standard deviations varied from 10 to 38% for air station measurements and from 19 to 42% for phantom doses with air station results being more precise than corresponding phantom measurements for each shield configuration. Average standard deviations from the mean were 19 and 31% for unshielded and shielded foil activation results, respectively. Neutron doses on phantoms measured using TLD systems averaged 1.12 times the reference for the unshielded case and 1.56 times the reference value for the shielded pulses. The TLD-measured doses were higher and less accurate than corresponding activation-measured values for all three pulses. Associated average standard deviations of 16% for unshielded and shielded pulses indicate that TLD-measured phantom doses were more precisely measured than those obtained using activation methods. Neutron doses determined using ^{24}Na activation methods were 0.84 to 1.52 times the reference values for all three pulses. For the two shielded cases, ^{24}Na activation produced the most accurate dose estimates of the three methods shown in Table 14. However, for the unshielded pulse, this technique provided the least accurate average measured dose. Average standard deviations of 16% for unshielded doses and 18% for shielded doses indicate that sodium activation methods were more precise than foil activation techniques and equally as precise as TLD systems. In general, no one type of NAD system used for measuring neutron doses in this study exhibited significantly better performance characteristics than any other type for all pulses.

Average measured gamma doses, experimental standard deviations from the mean, neutron-to-gamma dose ratios, and reference values are summarized in Table 15 for measurements made at air stations and on phantoms. All gamma doses were measured using TLD systems (mostly TLD-700) which included LiF, $CaSO_4$, and CaF material. Measured neutron-to-gamma dose ratios are within two experimental standard deviations of the reference values for air station measurements and within one standard deviation of reference values for phantom measurements.

Table 16 is a summary of average measured gamma dose normalized to the reference values and associated percent standard deviations from the mean for air station and phantom locations. For all three pulses, average measured gamma doses were higher than reference values by factors of 1.49 and 1.39 for air station and phantom measurements, respectively. The most descrepant measurements were obtained for the Lucite shielded cases (average normalized dose = 1.90) which produced the softest neutron energy spectrum and the lowest D_η/D_γ ratio of any configuration used in this study. Standard deviations from the means varied from 23 to 29% (average = 26%) for air station measurements and from 17 to 22% (average 19%) for phantom locations w . phantom doses being more precisely measured than corresponding air doses for each pulse.

Considering the composite of all reported results, Tables 14 and 16 show that neutron dose measurements were more accurate and precise than gamma dose measurements at air stations. Average neutron and gamma doses measured at air stations were 1.32 and 1.49 times the reference values, respectively, with associated standard deviations of 21% (neutron) and 26% (gamma) of the means. With regard to doses measured on phantoms, average neutron results were 1.24 times reference values and were more accurate than measured gamma doses which averaged 1.39 times the references. However, gamma doses on phantoms (average standard deviation = 19%) were more precisely measured than neutron doses (average standard deviation = 23%). Tables 14 and 16 also show that measured neutron doses were more accurate than corresponding gamma doses for each of the three pulses. Also, unshielded neutron dose measurements (average standard deviation = 14%) were more precise than unshielded gamma doses (average standard deviation = 22%). Neutron and gamma doses for shielded pulses were measured with almost the same precision (average standard deviation = 26% and 23%, respectively). The Lucite-shielded pulse yielded the least accurate results for neutron and gamma dose measurements at air stations and on phantoms.

The average of neutron fluences measured at air stations are summarized in Table 17. Fluences associated with activation of the various elements provide definition of the HPRR neutron spectra resulting from the three pulses. This spectral information is valuable in dose determination since the relative contribution of a neutron to the total dose depends on its energy. Since all participants did not report fluence results, no detailed analysis of the data presented in Table 17 is given in this report.

Ratios of doses measured on phantoms to doses measured at air stations for each pulse are given in Table 18. Doses measured on phantoms were larger than those obtained at air stations by an average of about 4% for neutrons and 43% for gammas for all shield configurations. Measured neutron dose on a phantom is increased relative to air by reflected (albedo) neutrons. Gamma dose is enhanced relative to air by the $^{1}H(n,\gamma)^{2}H$ reaction in the water that fills the phantom. Measured phantom-to-air dose ratios given in Table 18 for the Eighteenth NAD Study are within one experimental standard deviation of reference values based on data obtained from the previous seventeen intercomparisons.[1-3]

Table 19 shows neutron and gamma doses for each pulse normalized to the number of fissions per pulse and to the unshielded value (in parenthesis). This information can be related to the radiation attenuation characteristics of the various shield materials used during the intercomparison. For example, the table shows that a 20-cm concrete shield placed 1 m from a bare U^{235} fission source reduces the neutron dose at 3 m to 30% (10.9/36.9) of its unshielded value. Neutron and gamma normalized doses were within one experimental standard deviation of the reference values of the unshielded pulse. Both normalized dose measurements were more than one standard deviation higher than normalized reference values for the Lucite shield which is consistent with results observed during prior intercomparison studies.[12] Although these data are not directly related to NAD intercomparison, they are presented here as information for DOSAR users and staff, and for reactor shield designers.

DOSIMETER PERFORMANCE RELATIVE TO REGULATORY CRITERIA

Nuclear criticality accident dosimetry guidelines[16-17] suggest accuracies of ±25% for neutron dose and ±20% for gamma dose. Table 20 summarizes performance of the study participants relative to these standards. The composite results for all three pulses show that slightly more than half of the individual neutron (56%) and gamma (53%) dose measurements satisfied the subject criteria relative to the reference values. Best results were obtained for the unshielded pulse in which 86% and 73% of the neutron and gamma dose measurements, respectively, met the suggested guidelines. Poorest results were obtained for the Lucite-shielded pulse which provided the softest neutron energy spectrum and the lowest neutron-to-gamma ratio encountered in this study. For this case, only 10% of the reported final neutron doses and none of the gamma doses satisfied the NAD criteria. It is concluded from these results that accident dose measurement techniques must be improved for radiation spectra with a large fraction of low energy neutrons and a high gamma component.

COMPARISON TO PREVIOUS INTERCOMPARISON STUDIES

Results presented in the preceeding text for the Eighteenth NAD Intercomparison Study are consistent with the following statements which are based on an analysis of results from the previous studies.[1-3]

1. The precision of dose measurements based on 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 Eighteenth NAD Intercomparison Study (14 and 22%, respectively) are almost equal to the average of all seventeen previous intercomparisons (15% for unshielded neutron and 21% for unshielded gamma).[1-3] Average percent standard deviations for shielded neutron and gamma dose measurements (∿22%) are consistent for all NAD 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

Results of the Eighteenth NAD Study show that slightly more than half (56%) of the final reported neutron doses measured using foil activation, thermoluminescent, or sodium activation methods satisfied accident dosimetry performance criteria relative to the reference values. Also, more than half (53%) of the final reported gamma doses measured using thermoluminescent methods satisfied dosimetry performance standards. The greatest difficulties in measuring accident doses occurred for mixed radiation fields with large numbers of low energy neutrons and a high gamma component (>40%). Although the gamma dosimetry performance represents an improvement over that observed in the previous study,[1] the fact that approximately 45% of the participating agencies do not meet existing neutron or gamma measurement criteria indicates that continued accident dosimetry development and evaluation are required to upgrade existing NAD programs.

RECOMMENDATIONS

Discussions conducted during the study program indicated a need for continuing accident dosimetry intercomparisons and training programs to test NAD techniques and increase the technical competence of evaluating personnel. In response, the DOSAR staff will develop an accident dosimetry training course to be initally conducted in 1983. This course will include lectures on detailed aspects of accident dosimetry and experimental work which will permit participants to perform dosimetric analyses based on activation (foil, blood sodium, hair, clothing) and thermoluminescent methods. These techniques can then be used by participants in subsequent NAD intercomparisons.

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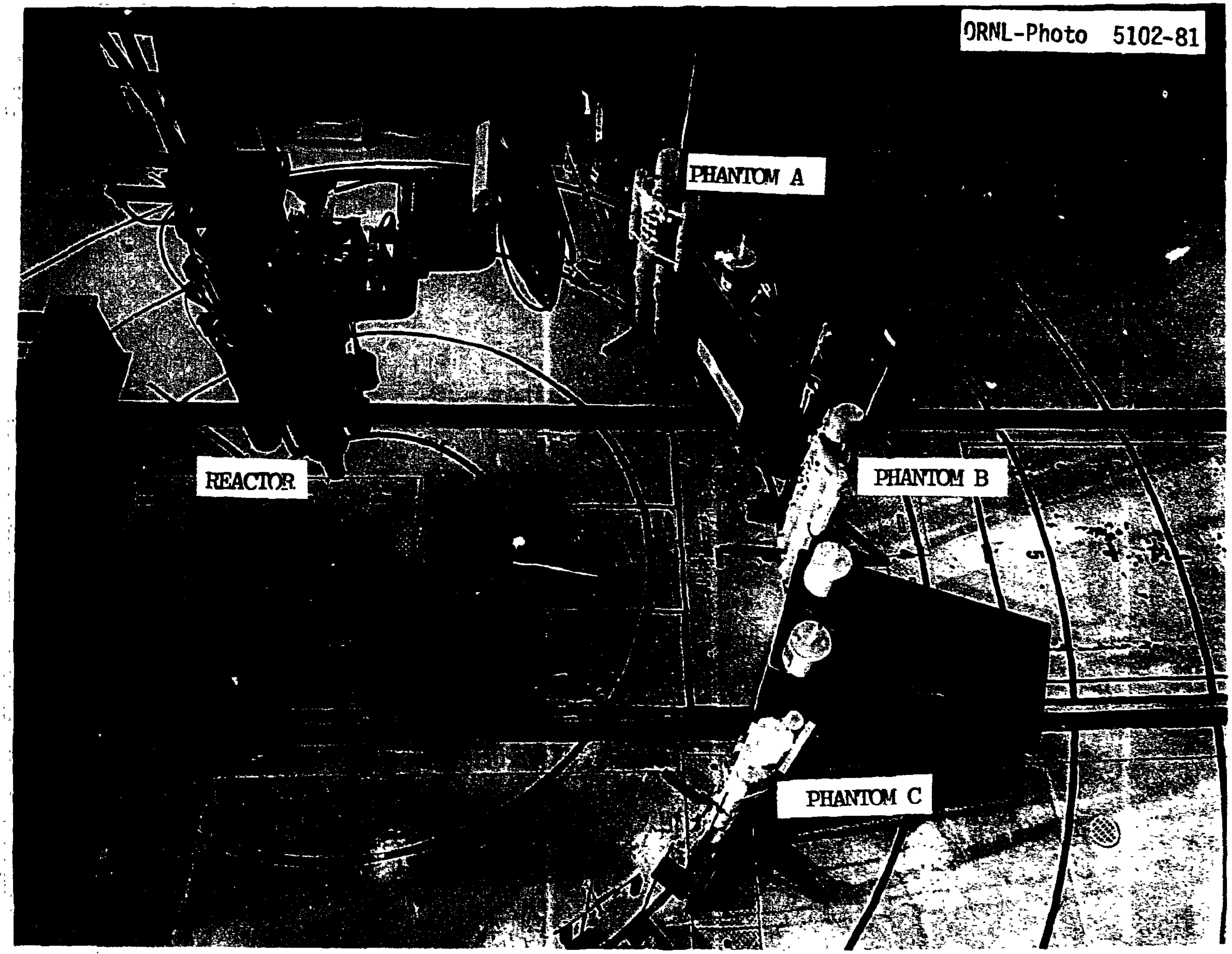

Fig. 1 Experimental arrangement of reactor, phantoms and area monitoring stations for all pulses.

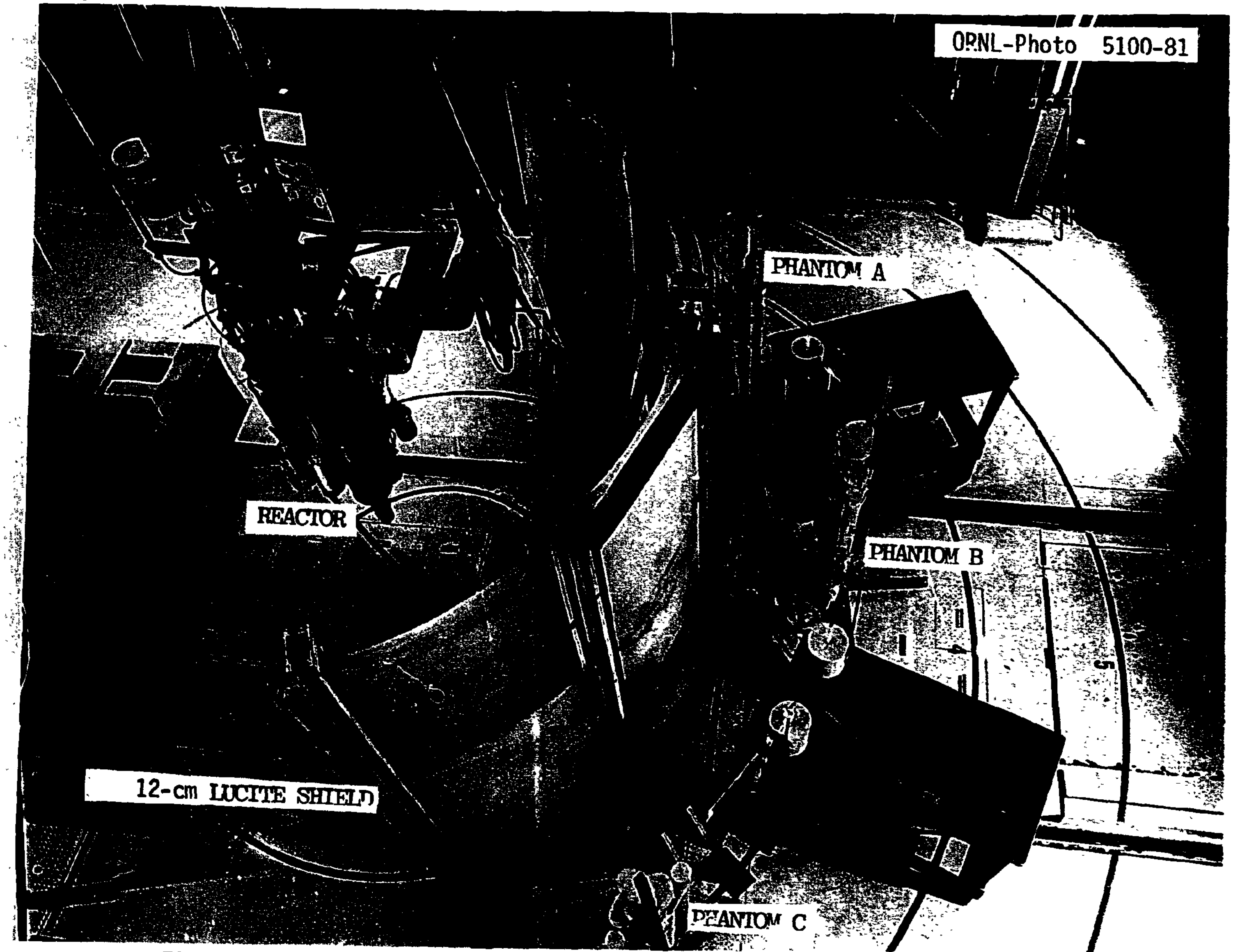

Fig. 2 Experimental arrangement for pulse No. 2 (12-cm Lucite shield).

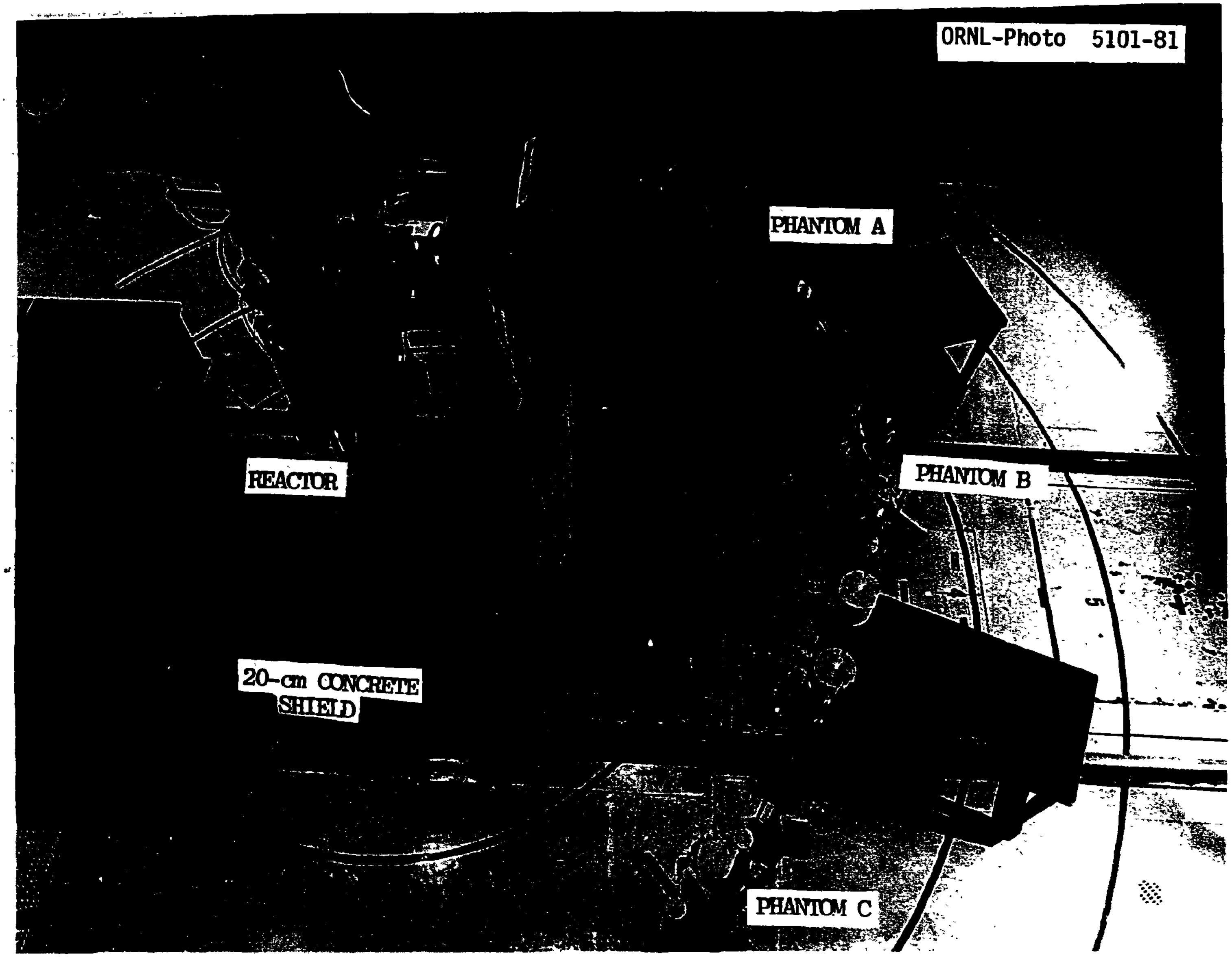

Fig. 3 Experimental arrangement for pulse No. 3 (20-cm concrete shield).

Table 1. Summary of experimental conditions

Pulse No.	Date	Eastern Daylight Time	Pulse yield,[a] 10^{16} fissions	Shield	Reactor to shield distance, m	Reactor to dosimeter distance,[b] m
1	8/11/81	1049	7.07	None		3
2	8/12/81	1155	5.64	12-cm Lucite	2	3
3	8/13/81	1105	5.71	20-cm concrete	1	3

[a]Based on sulfur pellet activation analysis.

[b]Dosimeters at 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. Reference neutron and gamma doses at air stations

Pulse No.	Shield	Pulse yield, 10^{16} fissions	Neutron dose, 10^{-2} Gy[a]		Neutron-to-gamma dose ratio[b]	Gamma dose, 10^{-2} Gy[a]
			Kerma	Element 57		
1	None	7.07	283	325	6.2	46
2	12-cm Lucite	5.64	28	33	1.2	23
3	20-cm concrete	5.71	49	57	2.7	18

[a]Calculated dose at 3 m from the reactor centerline based on HPRR reference dosimetry document ORNL/TM-7748. Units are 10^{-2} Gy (1 rad).

[b]Dose ratio at 3 m from the reactor based on measured results from the first seventeen nuclear accident dosimetry intercomparison studies.

Table 3. Reference neutron and gamma doses on phantoms

Pulse No.	Neutron air-to-phantom conversion[a]	Neutron dose, 10^{-2} Gy		Gamma air-to-phantom conversion[a]	Gamma dose, 10^{-2} Gy[b]
		Kerma[b]	Element 57		
1	1.05	297	325	1.69	78
2	1.03	29	33	1.38	32
3	1.17	57	57	1.60	29

[a]Ratio of phantom-to-air dose based on measured results from the first seventeen nuclear accident dosimetry intercomparison studies.

[b]Product of conversion factor times the dose in air given in Table 2.

Table 4. Preliminary measurements at air stations

Pulse Number	Study group	Radiation dose, 10^{-2} Gy Neutron[a]	Gamma	Basis for estimating Neutron	Gamma
1	REFERENCE	283	46		
	BPNL	220		ACT	
	DOSAR	263	47	TDU	TLD-700
	GAC		54		TLD
	INEL		50		TLD-700
	SRP	246	51	ACT	TLD
	Y12	286		ACT	
2	REFERENCE	28	23		
	BPNL	33		ACT	
	DOSAR	40	34	TDU	TLD-700
	GAC	45		ACT	
	INEL		42		TLD-700
	Y12	54		ACT	
3[b]	REFERENCE	49	18		
	BPNL	54		ACT	
	DOSAR	45	17	TDU	TLD-700
	INEL		20		TLD-700
	SRP	48	29	ACT	TLD

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

[b]Shield configuration unknown to participants for the preliminary evaluation.

Table 5. Preliminary measurements on phantoms

Pulse Number	Study group	Radiation dose, 10^{-2} Gy Neutron[a]	Radiation dose, 10^{-2} Gy Gamma	Basis for estimating Neutron	Basis for estimating Gamma
1	REFERENCE	297	78		
	ANL	282		NaACT	
	DOSAR	264	91	NaACT	TLD-700
	GAC		92		TLD
	INEL		89		TLD-700
	RFP	300		ACT	
	SRP	289	70	ACT	TLD
	SRP	242	78	NaACT	TLD
	USN	295		ACT	
	Y12	295	125	ACT	TLD
2	REFERENCE	29	32		
	ANL	39		NaACT	
	DOSAR	50	59	NaACT	TLD-700
	GAC	28		ACT	
	INEL		56		TLD-700
	RFP	91		ACT	
	SRP	32		NaACT	
	USN	57		NaACT	
	Y12	42	76	ACT	TLD
3[b]	REFERENCE	57	29		
	ANL	73		NaACT	
	DOSAR	69	33	TDU	TLD-700
	INEL		34		TLD-700
	RFP	62		ACT	
	SRP	56	28	ACT	TLD
	SRP	54	31	NaACT	TLD
	USN	51		ACT	

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

[b]Shield configuration unknown to participants for the preliminary evaluation.

Table 6. Summary of preliminary measurement results relative to regulatory criteria[a]

Pulse Number	Neutron measurements[b]		Gamma measurements[b]	
	Number of measurements	Number meeting criteria	Number of measurements	Number meeting criteria
1	11	11	10	9
2	11	6	5	0
3	9	9	7	6
Total	31	26	22	15

[a]Criteria presented in ICRP Report No. 12 which require that neutron and gamma doses be determined with an accuracy of ±50% within 24 hours after an accident.

[b]Includes air station and phantom measurements shown in Tables 4 and 5.

Table 7. Final measurements at air stations for pulse No. 1
Yield: 7.07 (10^{16}) fissions, Shield: None

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	D_n/D_γ	10^{-10} x Neutron fluence, n/cm²								Detector system	
				Au, thermal	Pu, >1 keV	Np, >0.75 MeV	U, >1.5 MeV	S, >2.9 MeV	Cu	In, thermal	In, fast	Neutron	Gamma
REFERENCE	283	46	6.2										
REFERENCE	325[b]												
BPNL	220											ACT	
DOSAR	263	47	5.6	0.7	10.4	7.9	3.9	2.2[c]				TDU	TLD-700
GAC		54											TLD
INEL	285	50	5.7	2.4				1.5	5.5	2.2	7.0	ACT	TLD-700
SRP	246	52	4.7					1.8	6.8	1.3	4.5	ACT	TLD-700
USN	264[b]	80	3.3									ACT	TLD-700
Y12	286			1.0				1.6			4.4	ACT	

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

[b]Neutron dose represents element 57 dose with the $^{1}H(n,\gamma)^{2}H$ component excluded.

[c]Fluence >2.5 MeV.

Table 8. Final measurements at air stations for pulse No. 2
Yield: 5.64 (10^{16}) fissions, Shield: 12-cm Lucite

Study group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	D_n/D_γ	10^{-10} x Neutron fluence, n/cm^2								Detector system	
				Au, thermal	Pu, >1 keV	Np, >0.75 MeV	U, >1.5 MeV	S, >2.9 MeV	Cu	In, thermal	In, fast	Neutron	Gamma
REFERENCE	28	23	1.2										
REFERENCE	33[b]												
BPNL	33											ACT	
DOSAR	40	34	1.2	1.3	1.5	1.2	0.6	0.4[c]				TDU	TLD-700
GAC	45											ACT	
INEL	39	42	0.9	4.1				0.3	2.2	6.4	0.1	ACT	TLD-700
USN	85[b]	57	1.5									ACT	TLD(CaF)
Y12	55			2.0				0.3			0.6	ACT	

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

[b]Shield configuration unknown to participants for the preliminary evaluation.

[c]Fluence >2.5 MeV.

Table 9. Final measurements at air stations for pulse No. 3
Yield: 5.71 (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} x Neutron fluence, n/cm²								Detector system	
				Au, thermal	Pu, >1 keV	Np, >0.75 MeV	U, >1.5 MeV	S, >2.9 MeV	Cu	In, thermal	In, fast	Neutron	Gamma
REFERENCE	49	18	2.7										
REFERENCE	57[b]												
BPNL	54											ACT	
DOSAR	45	17	2.6	0.8	2.0	1.2	0.6	0.3[c]				TDU	TLD-700
INEL	63	20	3.2	3.1				0.3	3.4	4.0	0.6	ACT	TLD-700
SRP	66	24	2.8					0.3	4.4	1.8	0.9	ACT	TLD-700
USN	66[b]	33	2.0									ACT	TLD (CaF)
Y12	73			2.2				0.2			0.5	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]Fluence >2.5 MeV.

Table 10. Final measurements on phantoms for pulse No. 1
Yield: 7.07 (10^{16}) fissions, Shield: None

Study Group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	^{24}Na activity, Bq/ml[b]	Basis for estimating Neutron dose	Basis for estimating gamma dose
REFERENCE	297	78			
REFERENCE	325[c]				
ANL	283	--	62.9	NaACT	
BPNL	265	110		ACT	TLD-700
BPNL	176		39.2	NaACT	
DOSAR	264	91		NaACT	TLD-700
GAC		92			TLD
INEL		89			TLD-700
KK	339	73		TLD[d]	TLD
KK	380	73		TLD[e]	TLD
NTHU		74			TLD ($CaSO_4$)
ORAU	241			HACT	
RFP	254	118		ACT	TLD-700
RFP	277			TLD	
RFP	233		37.0		
SRP	230	74		ACT	TLD-700
SRP	252		35.9	NaACT	
USN	295			ACT	
Y12	358	125		ACT	TLD
Y12	286		48.8	NaACT	

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

[b]3.7×10^{10} Bq = 1 Ci.

[c]Neutron dose represents element 57 dose with the $^1H(n,\gamma)^2H$ component excluded.

[d]Karlsruhe personnel albedo dosimeter.

[e]Single sphere (30-cm diameter) albedo dosimeter system.

Table 11. Final measurements on phantoms for pulse No. 2
Yield: 5.64 (10^{16}) fissions, Shielded: 12-cm Lucite

Study Group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	^{24}Na activity, Bq/ml[b]	Basis for estimating Neutron dose	Basis for estimating gamma dose
REFERENCE	29	32			
REFERENCE	33[c]				
ANL	39		8.9	NaACT	
BPNL	40	51		ACT	TLD-700
BPNL	38		8.5	NaACT	
DOSAR	50	59		NaACT	TLD-700
GAC	28			ACT	
INEL		56			TLD-700
KK	53	59		TLD[d]	TLD
KK	56	51		TLD[e]	TLD
NTHU		52			TLD ($CaSO_4$)
RFP	67	73		ACT	TLD-700
RFP	52			TLD	
RFP	44		13.3	NaACT	
SRP	40		7.1	NaACT	
USN	57			NaACT	
Y12	74	76		ACT	TLD
Y12	42		10.3	NaACT	

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

[b]3.7×10^{10} Bq = 1 Ci.

[c]Neutron dose represents element 57 dose with the $^1H(n,\gamma)^2H$ component excluded.

[d]Karlsruhe personnel albedo dosimeter.

[e]Single sphere (30-cm diameter) albedo dosimeter system.

Table 12. Final measurements on phantoms for pulse No. 3
Yield: 5.71 (10^{16}) fissions, Shield: 20-cm concrete

Study Group	Neutron dose, 10^{-2} Gy[a]	Gamma dose, 10^{-2} Gy	^{24}Na activity, Bq/ml[b]	Basis for estimating Neutron dose	Basis for estimating gamma dose
REFERENCE	57	29			
REFERENCE	57[c]				
ANL	55		12.2	NaACT	
BPNL	49	34		ACT	TLD-700
BPNL	44		9.9	NaACT	
DOSAR	69	33		NaACT	TLD-700
INEL		34			TLD-700
KK	62	28		TLD[d]	TLD
KK	59	28		TLD[e]	TLD
NTHU		26			TLD ($CaSO_4$)
RFP	55	45		ACT	TLD-700
RFP	95			TLD	
RFP	73		22.4	NaACT	
SRP	56	30		ACT	TLD-700
SRP	54		20.0	NaACT	
USN	51			ACT	
Y12	97			ACT	
Y12	46		12.9	NaACT	

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

[b]3.7×10^{10} Bq = 1 Ci.

[c]Neutron dose represents element 57 dose with the $^1H(n,\gamma)^2H$ component excluded.

[d]Karlsruhe personnel albedo dosimeter.

[e]Single sphere (30-cm diameter) albedo dosimeter system.

Table 13. 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	Sodium	All[c]	Reference, kerma/element 57
1	Air	261 ± 25[d]			261 ± 25	283/325
2	Air	50 ± 19			50 ± 19	28/33
3	Air	62 ± 10			62 ± 10	49/57
1	Phantom	280 ± 49	332 ± 52	249 ± 41	276 ± 52[e]	297/325
2	Phantom	52 ± 22	54 ± 2	44 ± 7	49 ± 12	29/33
3	Phantom	62 ± 20	72 ± 20	57 ± 12	62 ± 16	57/57

[a]Values are average doses based on data shown in Tables 7-9 (air) and Tables 10-12 (phantoms) and are given in units of 10^{-2} Gy (1 rad). Doses represent wet tissue kerma unless otherwise indicated.

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

[c]Average of results for all measurement methods.

[d]Mean ± one standard deviation.

[e]Includes one measurement based on hair activation.

Table 14. Normalized average measured neutron doses and associated percent standard deviations[a]

Pulse No.	Shield	Dosimeter location	Normalized dose (percent standard deviation)[b]			
			Activation	TLD	Sodium	All[c]
1	None	air	0.92(10)			1.00(14)
2	12-cm Lucite	air	1.79(38)			1.79(38)
3	20-cm concrete	air	1.26(16)			1.26(16)
1	None	phantom	0.94(18)	1.12(16)	0.84(16)	0.94(19)
2	12-cm Lucite	phantom	1.79(42)	1.86(4)	1.52(16)	1.69(24)
3	20-cm concrete	phantom	1.09(32)	1.26(28)	1.00(21)	1.09(26)

[a]Based on data shown in Table 13.

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

[c]Includes results for all measurement methods.

Table 15. Summary of results of gamma dose measurements at air stations and on phantoms

Pulse No.	Dosimeter location	Gamma dose, 10^{-2} Gy[a] TLD[b]	Gamma dose, 10^{-2} Gy[a] Reference	D_n/D_γ Measured[c]	D_n/D_γ Reference
1	air	57 ± 13[d]	46	4.6 ± 1.1	6.2
2	air	44 ± 12	23	1.1 ± 0.5	1.2
3	air	24 ± 7	18	2.6 ± 0.1	2.7
1	phantom	92 ± 20	78	3.0 ± 0.9	3.8
2	phantom	60 ± 10	32	0.8 ± 0.2	0.9
3	phantom	32 ± 6	29	1.9 ± 0.6	2.0

[a]Values are average doses based on data shown in Tables 7-9 (air) and Tables 10-12 (phantoms) and are given in units of 10^{-2} Gy (1 rad).

[b]Includes results of LiF, $CaSO_4$:Dy, and CaF:Mn dosimeters.

[c]Average of all reported neutron dose measurements from Table 13 divided by the average of all reported gamma dose measurements.

[d]Mean ± one standard deviation.

Table 16. Normalized average measured gamma doses and associated percent standard deviations[a]

Pulse No.	Shield	Dosimeter location	Normalized dose (percent standard deviation)[b]
1	None	air	1.24(23)
2	12-cm Lucite	air	1.91(27)
3	20-cm concrete	air	1.33(29)
1	None	phantom	1.18(22)
2	12-cm Lucite	phantom	1.88(17)
3	20-cm concrete	phantom	1.10(19)

[a]Based on data given in Table 15 which considers only TLD systems.

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

Table 17. Summary of neutron fluence measurements at air stations

Pulse No.	10^{-10} × Average neutron fluence, n/cm^2 [a]								
	Au, thermal	Pu, >1 keV	Np, >0.75 MeV	U, >1.5 MeV	S, >2.5 MeV	S, >2.9 MeV	Cu	In, thermal	In, fast
1	1.4 ± 0.9[b]	10.4	7.9	3.9	2.2	1.6 ± 0.2	6.2 ± 0.9	1.8 ± 0.6	5.3 ± 1.5
2	2.5 ± 1.5	1.5	1.2	0.6	0.4	0.3	2.2	6.4	0.4 ± 0.4
3	2.0 ± 1.2	2.0	1.2	0.6	0.3	0.3 ± 0.1	3.9 ± 0.7	2.9 ± 1.6	0.7 ± 0.2

[a] Average fluences based on data given in Tables 7-9.

[b] One standard deviation from the mean. No standard deviation indicates that results were reported by only one participant.

Table 18. 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[b]	Measured[c]	Reference[b]
1	None	1.07 ± 0.21[d]	1.05	1.61 ± 0.51	1.69
2	12-cm Lucite	1.04 ± 0.33	1.03	1.36 ± 0.44	1.38
3	20-cm concrete	1.00 ± 0.37	1.17	1.33 ± 0.46	1.60

[a]Based on data given in Table 13 for all reported dose measurements.

[b]Based on experimental data obtained during the previous 17 intercomparison studies

[c]Based on data given in Table 15 for all reported dose measurements.

[d]One standard deviation from the mean.

Table 19. 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[e]	Measured	Reference[e]
1	7.07	None	36.9 ± 3.5[f](1.00)[g]	40.0(1.00)	8.1 ± 1.8(1.00)	6.5(1.00)
2	5.64	12-cm Lucite	8.9 ± 3.4(0.24)	5.0(0.12)	7.8 ± 2.1(0.96)	4.1(0.63)
3	5.71	20-cm concrete	10.9 ± 1.8(0.30)	8.6(0.21)	4.2 ± 1.2(0.52)	3.2(0.48)

[a]The Lucite shield was located 2 m from the reactor centerline and the concrete shield was located 1 m from the reactor centerline.

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

[c]Based on data given in Table 13.

[d]Based on data given in Table 15.

[e]Reference dose divided by the fission yield given in Table 1 (reference dose divided by the unshielded reference value).

[f]One standard deviation from the mean.

[g]Normalized dose divided by the unshielded value.

Table 20. Summary of final measured results relative to regulatory criteria[a]

Pulse number	Dosimeter location	Neutron measurements		Gamma Measurements	
		Number of measurements	Number meeting criteria	Number of measurements	Number meeting criteria
1	Air	6	6	5	4
2	Air	6	1	3	0
3	Air	6	2	4	2
1	Phantom	15	13	10	7
2	Phantom	14	1	8	0
3	Phantom	14	11	8	7
Total		61	34	38	20

[a]Criteria presented in ANSI N13.3 which suggest accuracies of ±25% for neutron doses and ±20% for gamma doses.

APPENDIX A

PROGRAM

EIGHTEENTH NUCLEAR ACCIDENT DOSIMETRY INTERCOMPARISON STUDY

August 10-14, 1981

Date	Time	Activity
August 10	9:00 AM	Welcome, P. S. Rohwer (ORNL)
	9:15 AM	Orientation, C. S. Sims (ORNL)
	9:30 AM	Review of Nuclear Accident Dosimetry Intercomparison Program R. E. Swaja (ORNL)
	10:00 AM	Description of DOSAR Facility and assignment of workspace R. T. Greene (ORNL)
	10:30 AM	Tour of Control Room and Reactor Building
		LUNCH
	1:00 PM	Lecture: *Nuclear Accident Dosimetry: Purpose and Performance* R. E. Swaja (ORNL)
	2:00 PM	Lecture: *Radiation Doses Due to Nuclear Accidents* - C. S. Sims (ORNL)
	3:00 PM	Preparation for Pulse No. 1
	7:00 PM	Dinner at the Holiday Inn, Oak Ridge Speaker: H. W. Dickson (ORNL), *Dosimetry-Is it Worth All the Effort?*
August 11	8:00 AM	Final setup of dosimetry for Pulse No. 1
	9:00 AM	Observation of pulse operation of HPRR
	10:00 AM	Pulse No. 1 (unshielded)
	10:30 AM	Review of participant dosimety system J. P. Cusimano (INEL)
	11:00 AM	Collect dosimeters
		LUNCH
	1:00 PM	Lecture: *Medical Aspects of Nuclear Accidents* R. C. Ricks (ORAU)

Date	Time	Activity
August 11	2:00 PM	Analysis of data and preparation for Pulse No. 2
August 12	8:00 AM	Final setup of dosimeters for Pulse No. 2
	9:00 AM	Review of participant dosimetry system C. N. Wright (SRL)
	10:00 AM	Pulse No. 2 (12-cm Lucite shielded)
	10:15 AM	Lecture: *Occupational Exposure at Nuclear Power Plants in Japan* Akira Imahori (ORAU)
	1:00 PM	Lecture: *Determination of Radiation Dose based on Chromosome Aberrations* L. G. Littlefield (ORAU)
	2:00 PM	Analysis of data and preparation for Pulse No. 3
August 13	8:00 AM	Final setup of dosimeters for Pulse No. 3
	9:00 AM	Review of participant dosimetry system E. A. Putzier and J. M. Aldrich (RFP)
	9:30 AM	Lecture: *Nuclear Accident Photon Dosimetry* V. Gupta (INEL)
	10:00 AM	Pulse No. 3 (unknown to participants)
		Discussion: Requirements and problems associated with nuclear accident monitoring at participating facilities.
	11:00 AM	Collect dosimeters
		LUNCH
	1:00 PM	Lecture: *Biological Effects of Radiation* T. D. Jones (ORNL)
	2:00 PM	Analysis of data
August 14	9:00 AM	Discussion: Reporting final doses for analysis of intercomparison study results C. S. Sims and R. E. Swaja (ORNL)
	9:30 AM	Presentation of preliminary dose estimates and discussion of results
	10:30 AM	Final Critique

APPENDIX B

List of Participants and Observers

Name	Affiliation
E. H. Dolecek	Argonne National Laboratory OHS/HP, Bldg. 14 9700 S. Cass Avenue Argonne, Illinois 60439 (ANL)*
G. A. Rieksts R. Robinson R. H. Wilson	Battelle Pacific Northwest Laboratories P.O. Box 999 Richland, Washington 99352 (BPNL)
Jen-Shu Hsieh	Defense Nuclear Agency Radiological Physics Division Bethesda, Maryland 20014 (DNA)
J. R. Martin	Department of Energy Oak Ridge Operations Oak Ridge, Tennessee 37830 (DOE)
L. W. Gilley R. T. Greene G. R. Patterson C. S. Sims R. E. Swaja	Oak Ridge National Laboratory Dosimetry Applications Research P.O. Box X, Building 7710 Oak Ridge, Tennessee 37830 (DOSAR)
L. D. Bloomfield A. Jeffries E. Litteral J. Ortman H. Steinhauer	Goodyear Atomic Corporation P.O. Box 628 Piketon, Ohio 45661 (GAC)
F. N. Flakus	International Atomic Energy Agency Division of Nuclear Safety Wagramerstrasse 5 P.O. Box 100 A-1400 Vienna AUSTRIA (IAEA)

*Abbreviation by which this participant organization is referred to in this report.

Name	Affiliation
J. P. Cusimano V. P. Gupta J. S. Morton	Department of Energy Idaho Operations Office 550 Second Street Idaho Falls, Idaho 83401 (INEL)
B. Burgkhardt E. Piesch	Kernforschungszentrum Karlsruhe GmbH Postfach 3640 D7500 Karlsruhe 1 Federal Republic of Germany (KK)†
P. C. Hsu P. S. Weng	National Tsing Hua University Hsinchu, Taiwan 300 Republic of China (NTHU)†
Akira Imahori	Oak Ridge Associated Universities Medical and Health Sciences Division P.O. Box 117 Oak Ridge, Tennessee 37830 (ORAU)
J. M. Aldrich E. A. Putzier	Rocky Flats Plant Health Sciences and Industrial Safety Division P.O. Box 464 Golden, Colorado 80401 (RFP)
D. Lutkenhoff C. Strain C. N. Wright	E. I. DuPont de Nemours and Company Savannah River Plant Aiken, South Carolina 29801 (SRP)
S. D. Tarnecky	U.S. Navy Charleston Naval Shipyard Building 590 Charleston, South Carolina 29408 (USN)
B. Gose	Union Carbide Corporation Y-12 Plant Building 9704-02 Oak Ridge, Tennessee 37830 (Y12)

†Mail-in participant.

APPENDIX C

PARTICIPANT FURNISHED DOSIMETRY SYSTEM - ANALYSIS TECHNIQUE DESCRIPTION

Argonne National Laboratory (ANL)

The activation of blood sodium produces ^{24}Na, which can readily be counted in a NaI well counter. The dps/ml of ^{24}Na at the time of the neutron pulse can be determined by:

$$k \text{ (dps/ml)} = \frac{\text{cps}}{(E)e^{-\lambda t}},$$

where cps is the counts per second as determined by the ^{24}Na full energy 2.75 MeV photon peak, E is the counting efficiency for the size of sample counted and is empirically determined, t is the time interval between the pulse and the midpoint of the counting interval, and λ is the decay constant for ^{24}Na (0.0462 h^{-1}). The approximate neutron kerma dose is then calculated by

$$D \text{ (rads)} = 4.5\ k$$

This formula is derived from the Y-12 burro experiment which gave 1.65 x 10^5 rad/μCi/ml of serum as the ratio of blood activity to the first collision dose.

APPENDIX D

PARTICIPANT FURNISHED DOSIMETRY SYSTEM ANALYSIS TECHNIQUE DESCRIPTION

Dosimetry Applications Research Facility (DOSAR)
Oak Ridge National Laboratory

Neutron Dose Measurements

1. Threshold Detector Units

The Threshold Detector Unit (TDU) contains fission foils of plutonium, neptunium, and uranium enclosed in a ^{10}B 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}

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) which competes with ^{32}P during the first 10 to 12 hours.

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

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

2. 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:

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

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

$$D\ (\mathrm{Gy}) = 0.1076 \times A\ (\mathrm{Bq/mg})$$

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

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

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

Gamma Dose Measurements

Gamma ray dose was measured using Harshaw LiF thermoluminescent (TLD) dosimeters. These dosimeters (TLD-700) are enriched in ^{7}Li (99.993%) and have a negligible neutron sensitivity.

APPENDIX E

PARTICIPANT FURNISHED DOSIMETRY SYSTEM - ANALYSIS TECHNIQUE DESCRIPTION

Idaho National Engineering Laboratory (INEL)

The Idaho National Engineering Laboratory (INEL) NAD system contains the following: In, Au, and Cu foils covered with 0.5 mm thick cadmium; In and Au bare foils; sulfur pellet; TLD-700 and RPL gamma dosimeters. All the neutron detectors are in the form of circular 1.27 cm diameter discs and their thicknesses are as follows:

In	-	0.13	mm
Au	-	0.025	mm
Cu	-	0.13	mm
S	-	2.00	mm

Two NADs were used for each pulse and were hung in free air at a distance of 3 meters from the reactor.

Sulfur pellets were burned to remove S^{32} and to increase the counting sensitivity. Neutron tissue kerma doses were taken as the average of doses calculated from In and Au data. Gamma dosimeters were readout after 24 hours to allow proper fading.

To estimate neutron doses we define neutron fluences in five different energy ranges as follows:

ϕ_{th} = < 0.4 eV from In^{116m} or Au^{198}

ϕ_{epi} = 0.4 - 2 eV from In^{116m}(Cd) or

ϕ_{epi} = 0.4 - 10 eV from Au^{198} (Cd)

ϕ_{Cu} = 2 eV - 1 MeV from Cu^{64} (Cd)

ϕ_{In} = 1 - 2.9 MeV from In^{115m} (Cd)

ϕ_S = > 2.9 MeV from $S(P^{32})$

These fluences are calculated as,

$$\phi_{th} = \frac{\text{dpm/g In}^{116m} - 1.31 \times \text{dpm/g In}^{116m}\text{ (Cd)}}{1.06 \times 10^{-2}}$$

or $$\phi_{th} = \frac{\text{dpm/g Au} - \text{dpm/g Au (Cd)}}{5.4 \times 10^{-5}}$$

$$\phi_{epi} = \frac{1.31 \times \text{dpm/g In}^{116m}\text{ (Cd)}}{0.14}$$

or $$\phi_{epi} = \frac{\text{dpm/g Au (Cd)}}{1.4 \times 10^{-3}}$$

$$\phi_{Cu} = \frac{\text{dpm/g Cu (Cd)}}{1.6 \times 10^{-6}} - 2.2\phi_{epi}$$

$$\phi_{In} = \frac{\text{dpm/g In}^{115m}\text{ (Cd)}}{2.6 \times 10^{-6}} - 1.7\,\phi_S$$

$$\phi_S = \frac{\text{dpm/g S}(P^{32})}{1.53 \times 10^{-7}}$$

$$\text{dpm/g} = \frac{\text{Net total } \gamma \text{ or } \beta \text{ counts}}{T \cdot \varepsilon \cdot m \cdot \gamma \cdot \alpha \cdot s \cdot N(t)}$$

where: dpm/g = disintegrations per minute per gram of detector material

T = total counting time in minutes

ε = detector counting efficiency

m = mass of activation foil in grams

γ = abundance of measured radiation

α = isotope abundance

s = self-shielding factor (for ϕ_{th} and ϕ_{epi} calculations only)

$N(t)$ = decay correction factor

Finally, tissue kerma doses are calculated using the following empirical formula:

$$\text{Dose} = \left[0.02\,\phi_{th} + 0.003\,\phi_{epi} + 1.1\phi_{Cu} + 2.4\,\phi_{In} + 3.7\,\phi_S\right] \times 10^{-9} \quad \text{(rads)}$$

APPENDIX F

PARTICIPANT FURNISHED DOSIMETRY SYSTEM - ANALYSIS TECHNIQUE DESCRIPTION

Rockwell International - Rocky Flats Plant (RFP)

The personnel badge consists of two separate systems: The TLD elements to determine gamma and neutron dose amounts, and an activation system consisting of a sulfur tablet and three different types of metal foils which are used to evaluate neutron fluence, neutron energy spectrum, and neutron doses received by personnel.

The TLD dosimetry badge is designed to serve as a container for LiF dosimeters which are used to measure external radiation. The TLD system uses Harshaw TLD-700 and TLD-600 dosimeters made from LiF chips with dimensions of 1/8 x 1/8 x 0.035 inch. The 700 series are depleted in ^{6}Li and are used to measure radiation exposure other than neutrons. The 600 series are enriched in ^{6}Li and measure all the radiation which is present.

The dosimeter badge is loaded for use in radiation areas by placing a TLD-700 in three cavities across the bottom of the badge, Cacity numbers two and three measure penetrating gamma while cavity number four measures skin exposures. Another TLD-700 is placed in the first cavity at the bottom of the badge. This crystal is under an open window and measures beta in addition to x-ray and soft gamma.

The TLD neutron system consists of four crystals, two TLD-600's, and two TLD-700's. One pair, a -700 and a -600, are shielded from the front by a cadmium strip; and the other pair, a -700 and a -600 from the back by cadmium. When all the crystals are in place a plastic insert with the brass shield for the two gamma crystals (cavities three and four) and front cadmium for the neutron system is installed. The insert is then secured by a plastic rivet.

The neutron activation system consists of a copper foil, cadmium-shielded indium foil, unshielded indium foil, and a sulfur tablet. The nuclear reactions in use are:

^{115m}Indium, ^{116m}Indium, ^{63}Copper $(n,\gamma)^{64}$Copper, and
^{32}Sulfur $(n,p)^{32}$Phosphorus.

These reactions cover the neutron energy ranges: 0 to 1.4 eV; 2 eV to 1 MeV; 1 MeV to 2.9 MeV; and >2.9 MeV. By using principles of neutron activation analysis, the neutron fluence, a five different energy group spectrum, and the dose in rads can be determined.

By counting the activated foils, taking into consideration decay constants, time since exposure, background, mass of foils, geometry of systems, efficiency of detectors, and appropriate conversion factors, the neutron fluence for each configuration of foil can be determined. The fluence is then either multiplied by a conversion factor to determine rads or subtracted from one of the other foil fluences to obtain the fluence in a specification portion of the energy spectrum. A resultant fluence (found by subtracting two fluences) is multiplied by a constant or conversion factor to determine the rad exposure in that given energy band. The five resulting different energy band rad dose calculations are then summed and represent the overall neutron exposure to the individual.

The fluence calculations for the different activation foils are as follows:

Bare Indium $$\phi t = 1.25 \times \frac{N_r}{MG\,\gamma\,(e^{-\lambda T_1} - e^{-\lambda T_2})}$$

Cd-shielded Indium $$\phi t = .091 \times \frac{N_r}{MG\,\gamma\,(e^{-\lambda T_1} - e^{-\lambda T_2})}$$

Copper $$\phi t = 555 \times \frac{N_r}{MG\,\gamma\,(e^{-\lambda T_1} - e^{-\lambda T_2})}$$

Scin. Indium $$\phi t = 102 \times \frac{N_r}{MG\,\gamma\,(e^{-\lambda T_1} - e^{-\lambda T_2})}$$

Sulfur $$\phi t = 353 \times \frac{N_r}{MG\,\gamma\,(e^{-\lambda T_1} - e^{-\lambda T_2})}$$

where

N_r = Total counts observed

M = Foil weight (grams)

G = Detector geometry

λ = Decay constant of the product isotope

T_1 = Elapsed time from exposure to counting time

T_2 = Elapsed time from exposure to completion of counting

γ = Foil size correction factor

To convert foil and sulfur neutron fluence values (ϕT) to total rad dose, use the following equations:

1. (ϕT Bare indium - ϕT Cd-indium) x (0.32×10^{-9}) = Thermal neutron dose

2. (ϕT Cd-indium) x (0.45×10^{-9}) = Epithermal neutron dose

3. (ϕT Copper foil) x (1.37×10^{-9}) = 2-eV - 1-MeV neutron dose

4. (ϕT Scin. indium - ϕT Sulfur) x (4.2×10^{-9}) = 1-MeV - 2.9-MeV neutron dose

5. (ϕT Sulfur) x (5.8×10^{-9}) > 2.9-MeV neutron dose

Upon completion of the neutron dose calculations, the operator records the results on a Personnel Dosimeter Data Sheet. The five rad doses in each of the energy regions shown above are added to obtain the total neutron dose.

APPENDIX G

PARTICIPANT FURNISHED DOSIMETRY SYSTEM - ANALYSIS TECHNIQUE DESCRIPTION

Union Carbide Corporation-Y-12 Plant (Y12)

Personnel and area dosimeter measurements were made using foil and sodium activation analysis for neutron doses and TLD for gamma doses. Neutron activation analysis methods are described in J. C. Bailey, *New Methods for Interpreting Neutron Data Obtained from Y-12 NAD, ORGDP Film, and Hurst Threshold Detector Units,* Report No. K-1821, October 15, 1971.