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Evaluation of the U.S. Army DT-236 Battlefield Personnel Dosimetry System*

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EVALUATION OF THE U.S. ARMY DT-236 BATTLEFIELD PERSONNEL DOSIMETRY SYSTEM*

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Highlights

Performance characteristics of the U.S. Army DT-236 battlefield personnel dosimetry system were evaluated using the Health Physics Research Reactor at Oak Ridge National Laboratory. The DT-236 dosimeter is designed to measure total (neutron plus gamma) radiation dose using a radiophotoluminescent (RPL) detector for gamma rays and a silicon diode for fast neutrons. Areas considered in this evaluation included pre-irradiation dose indication; accuracy and precision of total, gamma, and neutron dose measurements; fading; angular response; temperature dependence; and relative dosimeter response in air and on various body locations. Experimental results for a variety of radiation fields and dose levels indicate that the existing system overestimates total, neutron, and gamma radiation doses in air by about 20 to 60% relative to reference values. Associated measurement precisions were about ± 5% of the means for doses above approximately 0.5 Gy. Fading characteristics, angular dependence, and temperature dependence of the RPL and diode systems were consistent with results expected based on detector characteristics and previous performance studies. Recommendations to improve existing reader performance and measurement accuracy are also presented.

INTRODUCTION

Performance characteristics of the U.S. Army DT-236 battlefield personnel dosimetry system[1,2] were evaluated at Oak Ridge National Laboratory (ORNL) during September and October of 1985. This system is being considered by the U.S. Army as a means of estimating total neutron and gamma radiation doses to combat forces in locations where tactical weapons could be used. Radiation fields and battlefield exposure conditions for this study were produced by operating the Health Physics

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Research Reactor (HPRR) at ORNL in the pulse mode with several spectrum-modifying shields[3]. Areas considered in this evaluation included pre-irradiation dose indication; reproducibility of results; accuracy and precision of gamma, neutron, and total dose measurements; fading; angular response; temperature effects; and relative dosimeter response in air and on various body locations. The following text provides a summary and evaluation of the results obtained during this study.

DESCRIPTION OF THE DT-236 DOSIMETRY SYSTEM

The U.S. Army DT-236 personnel dosimeter is a wristwatch-style detector which is designed to measure total (gamma plus neutron) radiation doses received by individual soldiers operating on a nuclear battlefield. Although specifically developed for military applications, this system could also be applied for area or personnel monitoring anywhere that high level (greater than about 0.25 Gy) gamma and/or neutron doses are possible; e.g., criticality accidents.

The DT-236 dosimeter badge, which is shown in Figure 1, consists of two independent solid-state detecting elements[4] used to measure gamma dose and fast neutron dose. A 12 x 15 x 3.5 mm rectangular parallelepiped of silver-activated phosphate glass is used as the gamma-sensitive element. Gamma dose estimation is based on radiophotoluminescent (RPL) properties in which the phosphate glass fluoresces with an intensity proportional to the absorbed gamma dose when stimulated with ultraviolet (UV) light. The neutron detector is a wide-based silicon junction diode. When exposed to fast neutrons, the crystal lattice structure of the diode is damaged and the resistivity of the material increases. Neutron dose estimation is based on measuring the increase in voltage drop across the diode at constant current. Gamma and neutron readings using these methods are non-destructible and the dosimeter will maintain the cumulative dose received by the individual. Both elements are packaged in a wristwatch-sized container which can be worn on the wrist or on an identification tag chain. The DT-236 dosimeters used in this study had serial numbers between B005600 and B006199.

Total gamma plus neutron doses are evaluated using a CP-696 dosimeter reader which is shown in Figure 2 and consists of two separate evaluation circuits contained in one instrument. The gamma portion is a UV flashtube source, optical filters, and a photodiode sensitive to the RPL glass fluorescent light. The neutron portion consists of a peak-reading voltmeter and a pulsed constant current generator. The neutron and gamma channels have check standards to indicate proper reader operation. For this study, the CP-696 reader was designated type 3146-1 and had serial number 19-B [HR C21 A5]. Power for the reader was supplied by a 24 volt DC power supply (ORNL Model X-93776) connected to the power input.

Although the reader has analog indication of total radiation dose, measurements in this study were based on digital indication from a voltmeter (ORNL Model I 009772) connected internally in parallel with the analog meter. Figure 3 shows the complete instrument setup used to evaluate the dosimeters including the ORNL digital voltmeter, the CP-696 reader, and the 24 volt DC power supply. The use of the digital voltmeter permitted more accurate readings and allowed estimation of readings which were off-scale on the analog meter. To convert from indicated voltage to total dose, a calibration curve was developed by comparing analog dose and digital readings. This correlation has two distinct linear segments which are described by the following equations:

$$\text{Dose (cGy)} = \begin{cases} 4 + 1478\ V & (V < 0.168) \\ -504 + 4496\ V & (V \geq 0.168) \end{cases}$$

where the dose is the analog indication in rads (i.e., cGy) tissue kerma and V is the voltmeter reading in volts. These equations are very similar to previous analog-digital correlations determined for other CP-696 readers[1]. Calibration and operating procedures for the reader were in accordance with those specified in the technical manual[5].

Since the reader is designed to give a single total dose reading when the operation mode switch is in the "read" position, separate neutron and gamma dose indications required a second evaluation cycle. This second reading was performed with the mode switch set in the "gamma test" position which provided an indication of the gamma-only dose. The neutron dose component was obtained by subtracting the total and gamma-only indications because no direct reading of neutron-only dose is available.

PRE-IRRADIATION CHARACTERISTICS

Prior to the HPRR exposures, 188 of the unirradiated DT-236 dosimeters were evaluated using the CP-696 reader. Since there was some variation between different readings of the same badge, the indicated dose was taken to be the middle value of three successive readings. The variation in successive readings for the same badge was approximately

±3% about the middle value with most dosimeters being better than ± 2% for total, gamma, or neutron doses.

All unirradiated dosimeters gave non-zero dose readings. Table 1 summarizes ranges, means, and standard deviations of the pre-irradiation total, neutron, and gamma dose indications for the 188 badges. Total doses ranged from 0.16 to 0.66 Gy with a mean of 0.40 Gy and one standard deviation of 0.09 Gy (22% of the mean). Neutron doses varied between 0.02 and 0.38 Gy with a mean of 0.17 Gy and one standard deviation of 0.07 Gy (45% of the mean). Pre-irradiation gamma doses varied from 0.08 to 0.38 Gy with a mean of 0.23 Gy and one standard deviation of 0.08 Gy (34% of the mean). Based on the observed standard deviations and suggested calculational conventions[6], the theoretical lower limits of detection for this system are 0.41, 0.34, and 0.36 Gy for total, neutron, and gamma doses, respectively. The ranges, means, standard deviations, and lower limits of detection obtained for the unirradiated badges in this study are within about 10% of corresponding values obtained in previous DT-236 dosimeter evaluations[3]. In the subsequent analyses, unexposed dosimeter responses for each individual badge were recorded and subtracted from the exposed dosimeter readings for the same badge to account for background levels.

During evaluations of the pre-irradiated badges, occasional readings much lower than the mean observed for several successive readouts were obtained for many of the individual dosimeters. These aberrant readings have also been observed in previous performance tests[3]. The cause of these occasional low readings was identified as being the UV flashtube which did not always function when the "read" switch was depressed. Without the flashtube, the gamma dose component which is based on RPL glass detection would not be included in the total dose indication. During battlefield operations, the underestimation of the total dose caused by this potential malfunction could be more than 50% depending on the relative neutron and gamma dose components of the radiation field.

It was observed that when the "read" switch was depressed and the UV tube functioned properly, a clearly audible "click" which originated inside the reader was obtained. However, when the switch was depressed and no "click" was heard, the UV source did not function and the total dose reading was low. During evaluation of the irradiated badges, readings in which the ultraviolet source was not heard to function were neglected and a reproducibility of about ± 3% about the middle of three readings with the flashtube functioning properly was obtained. Since the "click" which characterizes proper UV operation may not be audible under battlefield conditions, a design change such as an indicator light or UV lightmeter may be necessary to indicate flashtube operation. Without such an indication, evaluation personnel will need to consider the maximum three values out of about 10 readings to ensure that the gamma component has been included in the total dose estimate.

IRRADIATION CONDITIONS

The source of radiation for this evaluation was the Health Physics Research Reactor operated in the pulse mode. The HPRR is a fast pulsed reactor which can be used to simulate nuclear battlefield conditions and provide acute, high-level, neutron and gamma doses in times as short as 60 microseconds. A variety of radiation fields with the neutron and gamma characteristics given in Table 2 can be produced by using spectrum-modifying shields to simulate various weapon and material attenuation spectra. The fields range from the unshielded reactor which has a hard (nearly U^{235} fission) neutron energy spectrum with a low gamma component to a Lucite-shielded condition which has a soft (hydrogen-moderated) neutron spectrum with a relatively high gamma component.

A total of seven pulses was conducted for this study between September 13 and 24, 1985. Dates, HPRR pulse numbers, shield conditions, fission yield, and reference neutron, gamma, and total radiation doses (tissue kerma) at 3 m from the reactor are summarized in Table 3 for these operations. Fission yields ranged from 3.91 to 9.28 x 10^{16} fissions with corresponding pulse half-widths between about 120 and 65 microseconds, respectively. Associated neutron doses at 3 meters from the HPRR vertical centerline varied from 0.40 to 3.10 Gy (tissue kerma), gamma doses varied from 0.12 to 0.50 Gy, and total doses varied from 0.77 to 3.60 Gy. Radiation doses given to some dosimeters were more or less than those values since some badges were located closer or farther than 3 m from the HPRR. Reference neutron doses, gamma doses, and fission yields were determined using standard HPRR reference dosimetry techniques[7,8] and neutron differential spectrum measurements. For these irradiations, the reactor was operated over Pit 1 at a height of 1.4 m above the floor.

Dosimeters were exposed in air (attached to ring stands) at a height of 1.4 m above the floor for most tests. When simulation of the human torso was required, 40-cm-high polyethylene BOMAB phantoms with 20-cm by 30-cm elliptical cross sections filled with tap water were used. A 10-cm-diameter, 40-cm-high cylindrical polyethylene BOMAB arm section filled with water was used to simulate the wrist. At least five badges mounted side-by-side were used for the air station and phantom measurements in each irradiation,

ACCURACY AND PRECISION

Accuracy and precision associated with total, neutron, and gamma dose measurements made with the DT-236 system were determined by comparing measured and reference doses for a wide range of dose levels (0.04 to 13.98 Gy tissue kerma) and a variety of HPRR pulsed radiation fields. Accuracy is reflected by the mean of the individual measurements made at a particular location and precision is given by one standard deviation of the individual results about the mean. Dose measurements presented in the following text were made at air stations, and reference and measured results were reported in terms of tissue kerma.

Total Dose Measurements

Table 4 summarizes accuracy and precision results for 21 measurements of total dose which is what would be determined during battlefield application of this system. Data shown in this table include date of pulse, shield condition, dosimeter distance from the reactor, reference total dose, measured total dose in air, measured result divided by the reference, one standard deviation about the mean, and the percent of the mean of one standard deviation. Most indicated results are for the unshielded HPRR with dosimeters placed at various distances from the reactor. Data for the steel-, concrete-, and Lucite-shielded pulses are for the badges located at 3 meters from the HPRR which is the distance at which the shielded reference doses are best known. Reference total doses given in the table are the sums of the reference gamma and neutron doses in air and vary between 0.20 and 13.98 Gy.

Average measured divided by reference total doses as a function of reference dose are shown in Figure 4. Error bars indicate one standard deviation about the mean. These data show that the DT-236 system overestimates reference values by about 20 to 60% for doses between approximately 0.2 to 14.0 Gy for all considered HPRR radiation fields. These results are consistent with DT-236 system accuracy observed during previous dosimeter tests at pulsed reactor facilities[1].

Performance specifications[1,2] for this system require ± 40% accuracy at doses between 0.5 and 10.0 Gy and ± 0.2 Gy accuracy at doses below 0.5 Gy. Table 4 and Figure 4 show that the DT-236 system does not meet these criteria relative to the HPRR reference values. However, by adjusting the reader output to indicate 40% lower total doses (i.e., decrease the calibration curves for digital readout or decrease the meter indication for analog readout), measured results will be within ± 20% of reference results for a wide range of spectra and doses between 0.2 and 14.0 Gy. Figure 4 shows an adjusted reference line at a measured-to-reference ratio of 1.4 and the ± 20% limits about this line. Although the figure indicates that the suggested 40% adjustment will provide ± 20% accuracy at dose levels below 0.5 Gy, the practical system accuracy at low doses will still be limited by the 0.40 Gy theoretical lower limit of detection and the 0.09 Gy standard deviation observed for the pre-irradiated badges. A measurement accuracy of ± 20% would satisfy ± 25% accuracy criteria specified by the American National Standards Institute[9], the U.S. Department of Energy [10], and the International Atomic Energy Agency[11] for criticality accident dosimetry systems.

With regard to measurement precision, Table 4 shows that single standard deviations were within 4% of the mean values for total doses greater than about 1.0 Gy. For doses below this value, standard deviations ranged from 4 to 12% of the means. These results are consistent with data obtained in previous DT-236 performance tests[2] which indicated one standard deviation values of about 5% of the means for doses above 0.5 Gy.

Gamma Dose Measurements

Accuracy and precision results for gamma dose measurements are summarized in Table 5 for the same exposures considered in the preceding analysis of total dose measurements. Indicated measured data are the background-corrected gamma doses in air based on the RPL detection system. Reference gamma doses given in the table are the products of the reference neutron doses in air times the neutron-to-gamma dose ratios at the measurement locations. Reference values vary between 0.04 and 1.70 Gy.

Average measured divided by reference gamma doses as a function of reference dose are shown in Figure 5. Error bars indicate one standard deviation about the mean. The figure shows that, except for one measurement, the RPL system overestimates gamma doses by about 20 to 60% for reference values above 0.35 Gy. This overestimation is expected based on the observed overresponse of the gamma detection system to hard gamma rays and the neutron sensitivity of the RPL glass[1, 11]. Below approximately 0.35 Gy, measured gamma doses show significant variations relative to reference values (between 0.5 to 1.9 times references) with relatively large standard deviations about the measured means. Thus, at gamma doses below about 0.35 Gy, which is very close to the theoretical lower limit of detection determined from unirradiated dosimeter results, the RPL system does not provide accurate gamma dose estimates in the fields considered in this study. Figure 5 shows that by adjusting the reader output to indicate 40% lower, gamma doses between about 0.35 and 1.70 Gy can be measured to within ± 20% of reference values.

Table 5 shows that single standard deviations were within approximately 5% of the means for gamma doses above 0.50 Gy. Below this level, standard deviations ranged from about 5 to 31% of the means with most values being in the 15 to 25% range. These results are slightly more precise than results obtained in previous DT-236 performance tests[3] which indicated one standard deviation values of about 20% of the means for gamma doses above 0.50 Gy.

Neutron Dose Measurements

Table 6 presents accuracy and precision results for neutron dose measurements in air. Measured data are the background-corrected neutron doses which were determined by subtracting the indicated total and gamma doses for each dosimeter. Reference neutron doses were based on sulphur pellet activation analysis and dose-per-fission correlations[7-8]. Reference doses given in the table vary between 0.16 and 12.28 Gy (tissue kerma).

Figure 6 shows measured divided by reference neutron doses as a function of reference dose based on data given in Table 6. Error bars represent one standard deviation about the measured mean. Except for one point, average measured neutron doses overestimate reference values by 20 to 60% over the entire range of reference doses and all HPRR spectra. Overestimation is expected based on the observed overresponse of the silicon diode detection system to fast neutrons in air[1]. Figure 6 also shows that by adjusting the reader output to indicate 40% lower, neutron doses can be estimated to within ± 20% of reference values between about 0.20 and 12.00 Gy. At doses below about 0.20 Gy, neutron measurement accuracy is significantly affected by relatively large uncertainties in corresponding low gamma dose measurements which must be subtracted from total dose readings.

Table 6 shows that single standard deviations for the estimated neutron doses vary from 0.3 to 15.7% of the means over the entire range of reference doses. Most standard deviations are between 1 and 6% of the means which is significantly more precise than the range of values obtained for the gamma measurements. These results are consistent with measurement precisions obtained in previous battlefield dosimeter performance studies[2].

FADING

Figure 7 shows measured total, neutron, and gamma doses at various times after exposure relative to the doses measured at two hours after irradiation for times up to 15 days. Each point represents the average result of five dosimeters irradiated to total, neutron, and gamma doses of 3.30, 1.88, and 0.42 Gy, respectively, in the unshielded HPRR spectrum. Single standard deviations associated with the indicated points are about 4%, 4%, and 10% about the means of the five readings for the total, neutron, and gamma measurements, respectively.

Over the 15 day evaluation period, the average total dose decreased by only about 7% relative to the value obtained two hours after exposure. Most of this fading occurred within the first seven days after irradiation. Very little fading was exhibited for total dose beyond the initial seven day period. Neutron dose results decreased by about 13% over the 15 day evaluation time with most fading (approximately 8%) occurring in the first two days after irradiation. The RPL-measured gamma doses showed an increase of about 11% over initially measured results in 15 days. Most of this increase (approximately 7%) occurred in the first day after irradiation. The increased gamma response after irradiation, which is characteristic of RPL materials[11], partly compensated for the decrease in neutron response due to fading to reduce the decrease in total dose indication that might be expected in a strong neutron field. The qualitative and quantitative performance observed in this study for fading of the total, neutron, and gamma components of the DT-236 system are consistent with results of pervious performance tests[2].

ANGULAR RESPONSE

To determine the effect of angnlar orientation on dosimeter response, groups of five badges were placed on the centers of three BOMAB torso sections and exposed to the unshielded HPRR spectrum with the minor axis of the elliptical phantoms positioned at 0° (front-facing), 45°, and 90° (side-facing) relative to the incident field. In all cases, badge centerlines were located 3 meters from the reactor. The three dosimeter components (total, neutron, and gamma responses) showed similar performance characteristics for the three orientations. Average total dose responses decreased by 4% and 35% at 45° and 90°, respectively, compared to the direct irradiation. Mean neutron doses decreased by 6% and 37% at 45° and 90°, respectively, relative to direct incidence. Gamma responses decreased by 1% at 45° and 33% at 90° compared to the 0° orientation. Uncertainties associated with these results are about ± 4% for the total and neutron measurements and ± 10% for the gamma measurements for one standard deviation about the mean. Thus, at angles of incidence between direct and 45° relative to the incident field, the dosimeter exhibits low sensitivity to angular orientation. At 90° incidence, the dosimeter response decreases by about 35% relative to direct incidence.

TEMPERATURE EFFECTS

Effects of temperature changes on dosimeter response were evaluated by storing separate sets of badges exposed at room temperature (about 20°C) in cold (0°C) and in hot (45°C) environments for 24 hours and then reading the dosimeters while at the reduced or elevated temperatures and after return to room temperature. These cycles were repeated for three days to determine if observed changes were permanent. Temperature limits chosen for this test correspond with those specified in performance standards for routine personnel dosimetry systems[12].

Table 7 summarizes results obtained for the cold and hot tests. Data shown in the table are average indicated total neutron and gamma doses relative to the values measured at room temperature two hours after exposure. Single standard deviations associated with these data are about ± 4% of the means for total and neutron values and about ± 10% of the means for gamma results. For total doses, the hot tests indicate that increasing from room temperature to 45°C for about 24 hours causes a reduction of approximately 23% if the badges are read hot. Allowing the badges to cool to room temperature before reading results in a 14-19% reduction in measured total dose. These data were consistent for all three hot temperature cycles. The cold tests showed only a 3% maximum increase in dosimeter response relative to the response after room temperature exposure if the badges are cooled to 0°C for 24 hours and read cold. Reading the dosimeters after allowing them to return to room temperature resulted in decreases in average response of from 1 to 4% compared to the initial total dose.

Data for the neutron and gamma components of the dosimeter response showed that the RPL gamma system was much more sensitive to temperature changes than the diode neutron detector. Considering the hot tests, neutron and gamma indications decreased by 24% and 16%, respectively, following heating to 45°C and reading at the elevated temperature. However, subsequent heatings and coolings to room temperature produced almost no variation in neutron dose estimation following the initial decrease while the gamma dose estimates increased by about 20 to 40% between the hot and room temperature readings. Cold tests indicated that cooling the irradiated badges to 0°C and reading at cold or room temperature had almost no effect on neutron response. However, the gamma system was much more sensitive to temperature variations in that the gamma dose estimate after the initial cooling and reading at 0°C increased by 22% relative to the original room temperature reading. Subsequent cooling and heating cycles produced approximately 20 to 40% variations in gamma response between 0°C and room temperature with a higher response obtained at the cold temperature.

Those data indicate that for strong neutron fields, heating the badge by 25°C after exposure at room temperature can cause a significant decrease (about 23%) in total dose response if the dosimeter is read hot. Even if the heated badge is allowed to cool to room temperature before reading, a permanent reduction in dosimeter response of about 14-19% can be expected. While some of this reduction can be attributed to fading, much of it can be attributed to temperature sensitivity of the neutron portion of the dosimeter. Temperature variations below room temperature produce much smaller effects on total and neutron response; i.e., changes which are within experimental uncertainties of the dose read at room temperature after exposure. After a response decrease following initial heating, the neutron detector indicates significantly lower sensitivity to temperature than the RPL gamma system.

AIR-TO-PHANTOM RESPONSE

Total doses measured with the badges mounted on the centers of BOMAB torso and arm sections relative to values obtained in-air (on ringstands) are given in Table 8 for four HPRR spectra. For these irradiations, all badges were located with their vertical centerlines at 3 meters from the reactor and all were positioned with their tops perpendicular to the incident field. Dosimeters mounted on the BOMAB torso center (i.e., worn on an identification chain around the neck) indicated total doses which were 7 to 18% higher than those measured in-air for the same exposure conditions. The largest air-to-phantom increases were exhibited for the hardest neutron energy spectra with the lowest gamma components (unshielded and steel-shielded). Badges mounted on the arm section (i.e., worn on the wrist) indicated total doses 7 to 12% higher than those obtained with dosimeters on ringstands. No obvious correlations between radiation field characteristics and observed results are evident for the air-to-arm-phantom results. Uncertainties associated with the ratios given in Table 8 are about ± 4% for one standard deviation.

Based on these data, total doses measured with the badge worn on the chest or the wrist can be at least 7% and as much as 18% higher than values measured in air for HPRR or similar spectra. The increase in total detector response on a polyethylene phantom relative to air is due primarily to contributions of incident neutrons scattered by the phantom and secondary gamma rays produced by neutron captures in the hydrogenous phantom (body)material[8].

SUMMARY

The following summary statements concerning performance of the U.S. Army DT-236 personnel dosimetry system are based on results presented in the preceding text:

1. There was about ± 3% variation in different readings of the same badge. The indicated dose was taken to be the middle value of three successive readings.
2. All unirradiated dosimeters gave non-zero dose readings. Pre-exposure total dose readings ranged from 0.16 to 0.66 Gy with a mean of 0.40 Gy and one standard deviation of 0.09 Gy. Corresponding gamma readings varied between 0.08 and 0.38 Gy with a mean of 0.23 Gy and one standard deviation of 0.08 Gy. Preirradiation neutron doses ranged from 0.02 to 0.38 Gy with a mean of 0.17 Gy and one standard deviation of 0.07 Gy. Based on these results, theoretical lower limits of detection for this system are 0.41, 0.34, and 0.36 Gy for total, neutron, and gamma doses, respectively.
3. Under pulse irradiation conditions using the HPRR, the DT-236 system overestimates total doses in air by between 20 and 60% relative to reference doses between about 0.2 to 14.0 Gy and a wide range of incident radiation fields. Neutron doses are also overestimated by this amount for reference neutron values between 0.20 and 12.00 Gy. For reference gamma doses between approximately 0.35 and 1.70 Gy, the system also overestimates gamma doses by 20 to 60%. Below 0.35 Gy, measured gamma results show significant variations relative to reference data

with large standard deviations. The total dose performance does not satisfy suggested ± 40% accuracy requirements for this system between 0.5 and 10.0 Gy.

4. Measurement precisions for the DT-236 system were about 4%, 4%, and 5% for one standard deviation about the mean for total, neutron, and gamma doses greater than about 0.5 Gy, respectively. Below this value, standard deviations increased significantly for all three dose measurements. Also, precision for neutron measurements was generally better than that obtained for corresponding gamma measurements.

5. Observations of dosimeter response over a 15-day period indicated that the measured total dose decreased by approximately 7% relative to that obtained two hours after exposure. Neutron results decreased by about 13% while gamma measurements increased by 11% over the 15 day evaluation period.

6. Average dosimeter responses for total, gamma, and neutron measurements decreased by about 4% and 35% at exposure angles of 45° and 90°, respectively, compared to the response for a direct incidence irradiation.

7. Heating the dosimeter following exposure can result in a significant reduction in total and neutron response. Temperature variations below room temperature following irradiation produce relatively small effects on measured total and neutron doses. For the radiation fields considered in this study, the RPL gamma system exhibits much greater sensitivity to temperature changes than the neutron system.

8. Based on results obtained with dosimeters mounted on polyethylene phantoms, total doses measured with the badge worn on the

chest or wrist can be at least 7% and as much as 18% higher than values measured in air depending on the incident spectrum.

9. Pre-irradiation, precision, and fading characteristics observed in this study are consistent with results obtained in previous DT-236 system performance evaluations.

RECOMMENDATIONS

Based on results of this study, the following recommendations are submitted:

1. Some method to indicate proper UV flashtube operation during dosimeter readout should be considered to prevent possible significant underestimation of the total dose. Without the flashtube, which does not always operate when the "read" switch is depressed, the gamma dose component based on RPL detection is not included in the indicated total dose. A design change such as an indicator light or UV lightmeter may be necessary to indicate flashtube operation under battlefield conditions.

2. The CP-696 reader output should be adjusted to indicate about 40% lower total doses to ensure compliance with performance standards. This can be accomplished by decreasing the meter indication for analog readout or decreasing the calibration curves or reader output for digital readout. Such a correction is recommended based on observed overresponses of the diode detection system to fast neutrons in air[1] and of the RPL system in mixed-field conditions[11]. This change will provide

measured total doses (tissue kerma) within ± 20% of reference results for a wide range of incident spectra and doses between about 0.2 and 14.0 Gy.

3. The conven'ion associated with the indicated doses should be reviewed and, if necessary, changed to correspond to reporting requirements. The present tissue kerma convention is recommended if doses in air are desired. However, if doses to personnel are required, the convention should be changed so that indicated values represent maximum absorbed dose to the body; e.g., element 57 dose[11]. These conventions are used in accident dosimetry experimental studies[8] and are recommended by international scientific agencies[11].

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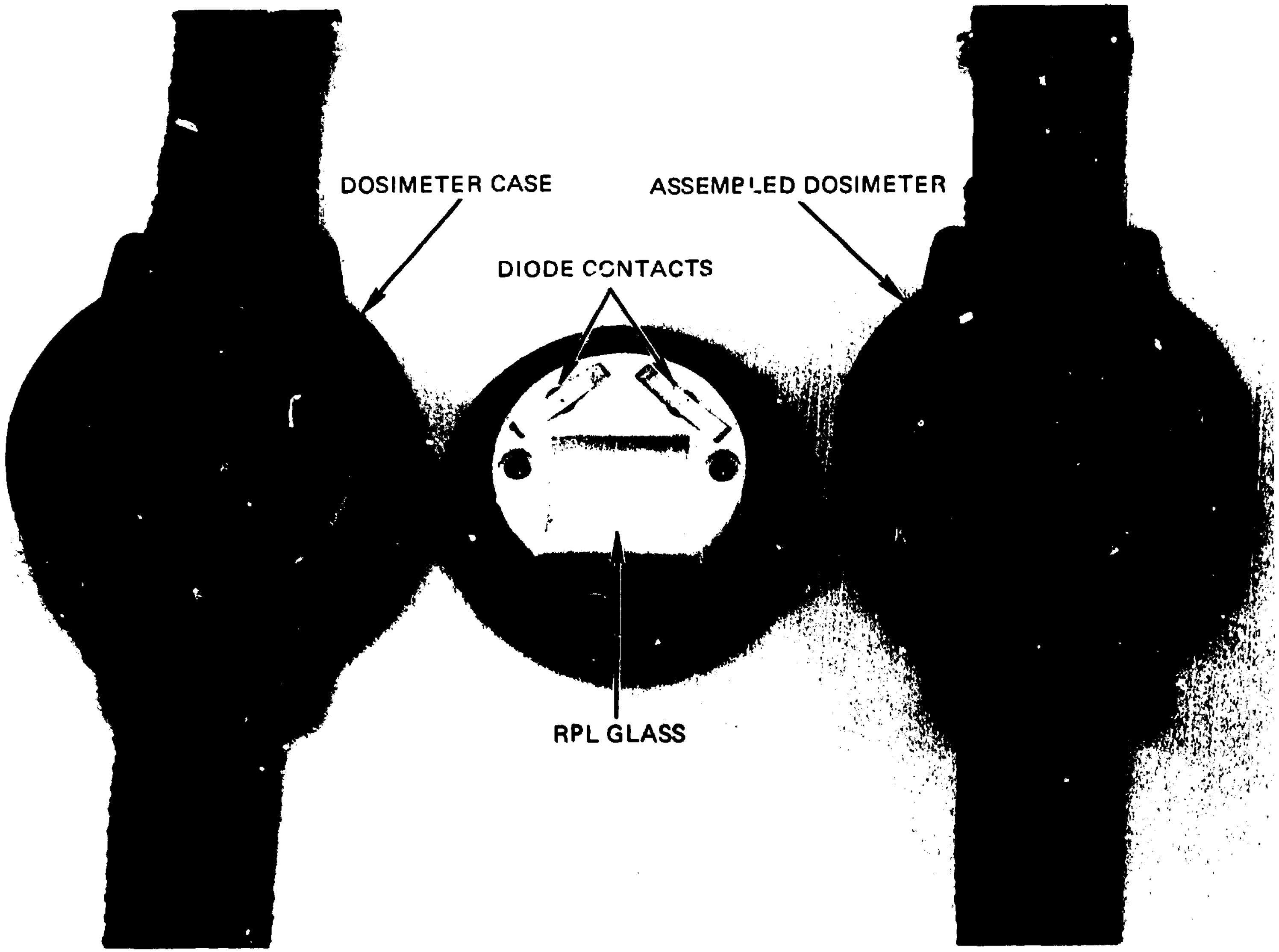

Figure 1. The U. S. Army DT-236 personnel dosimeter

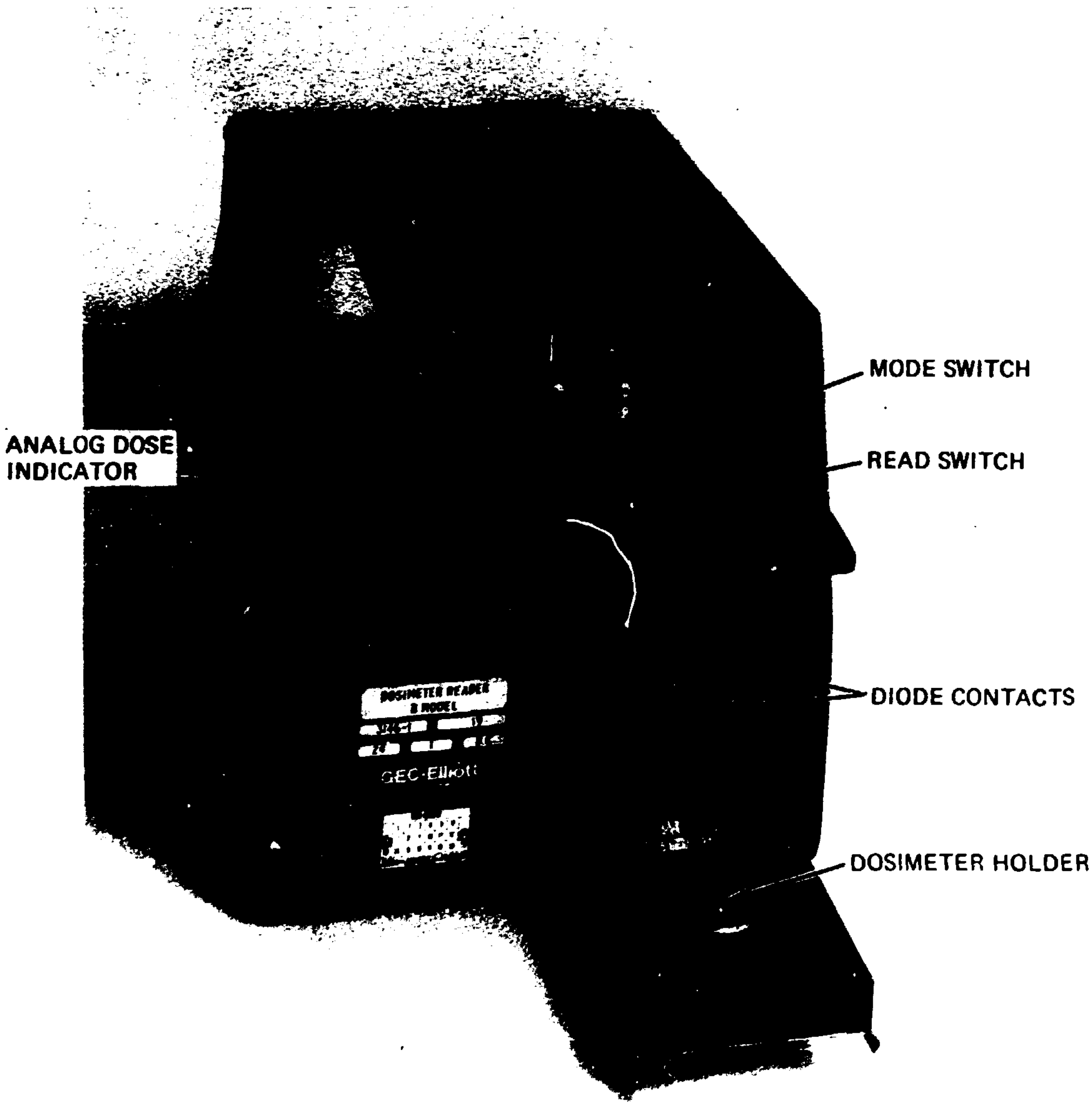

Figure 2. The U. S. Army CP-696 battlefield dosimeter reader.

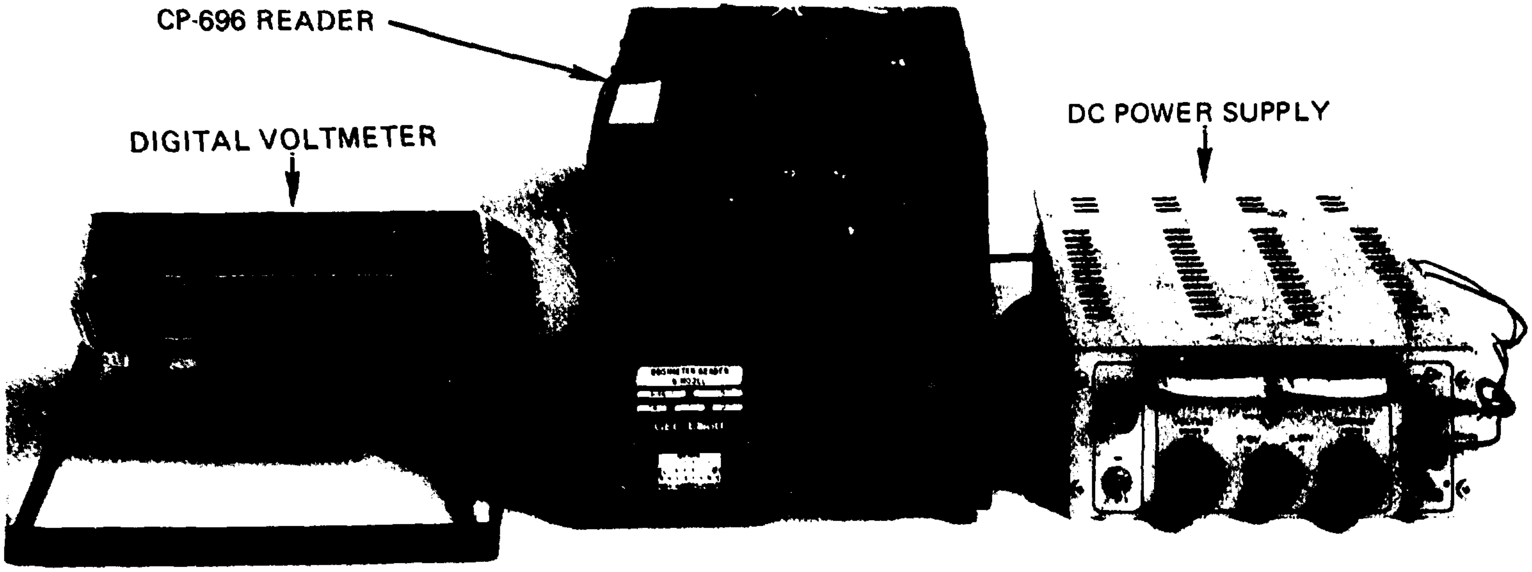

Figure 3. Experimental setup used to evaluate the battlefield dosimeters.

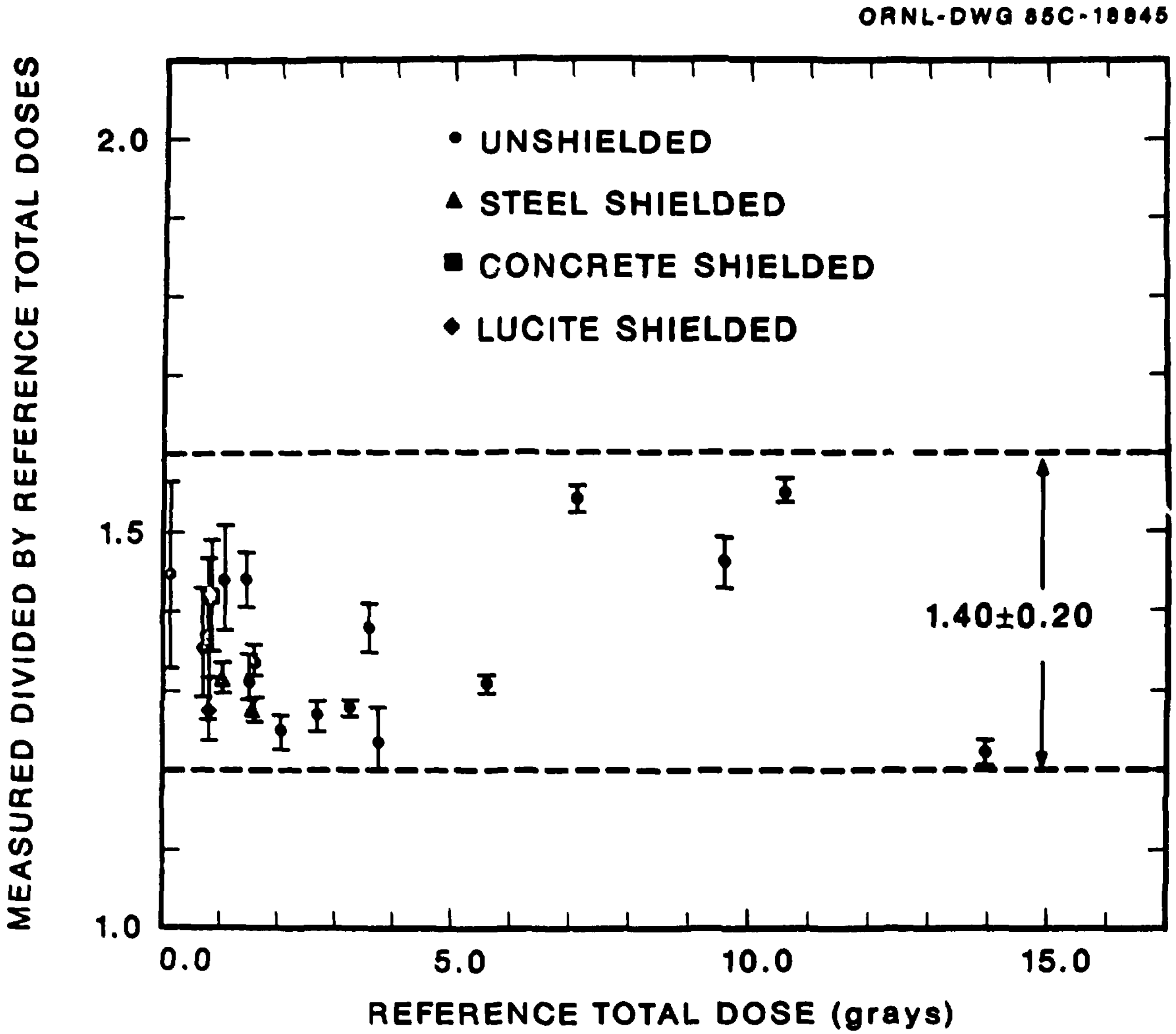

Figure 4. Measured divided by reference total doses as a function of reference dose.

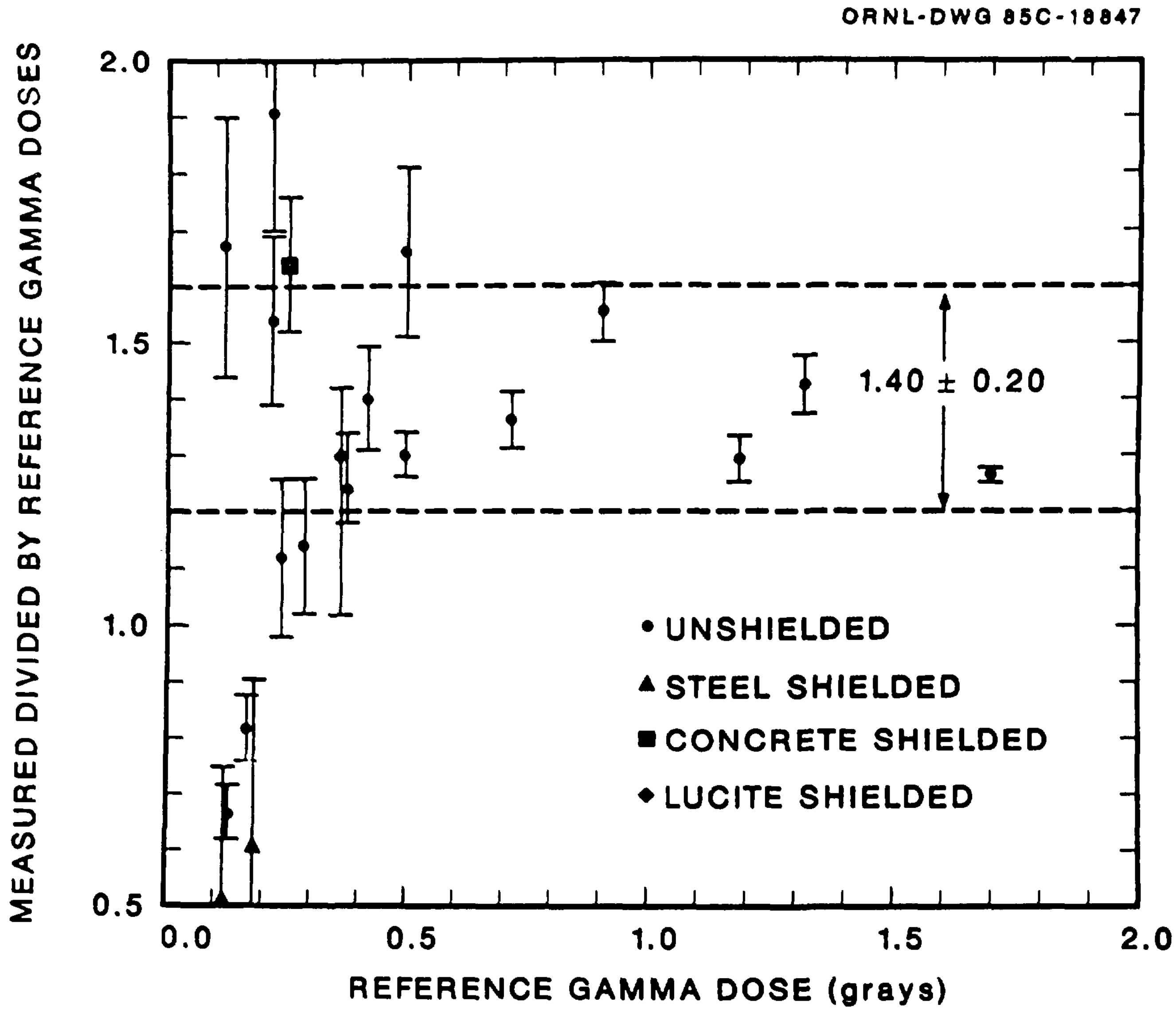

Figure 5. Measured divided by reference gamma dose as a function of reference dcse.

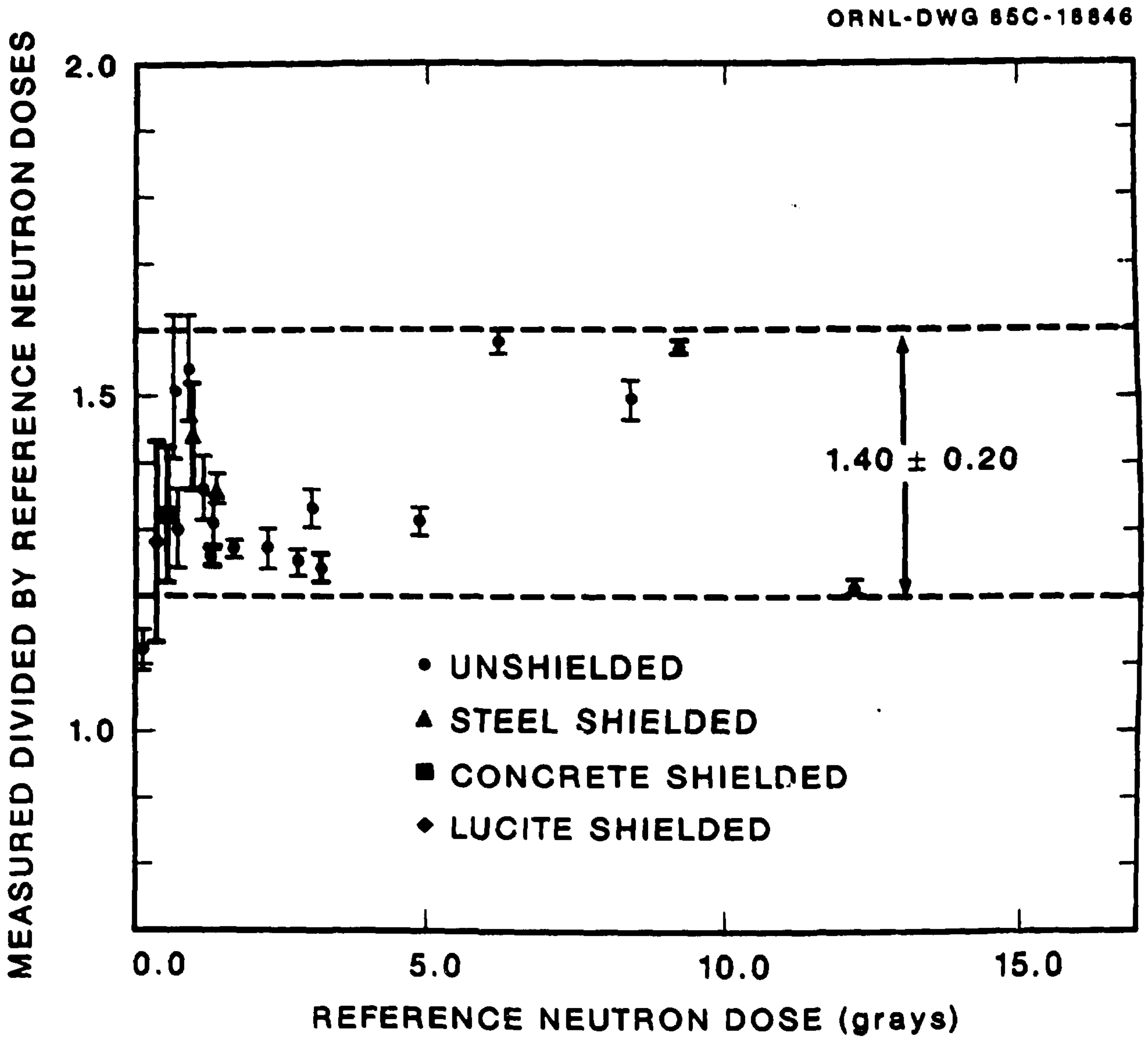

Figure 6. Measured divided by reference neutron dose as a function of reference dose.

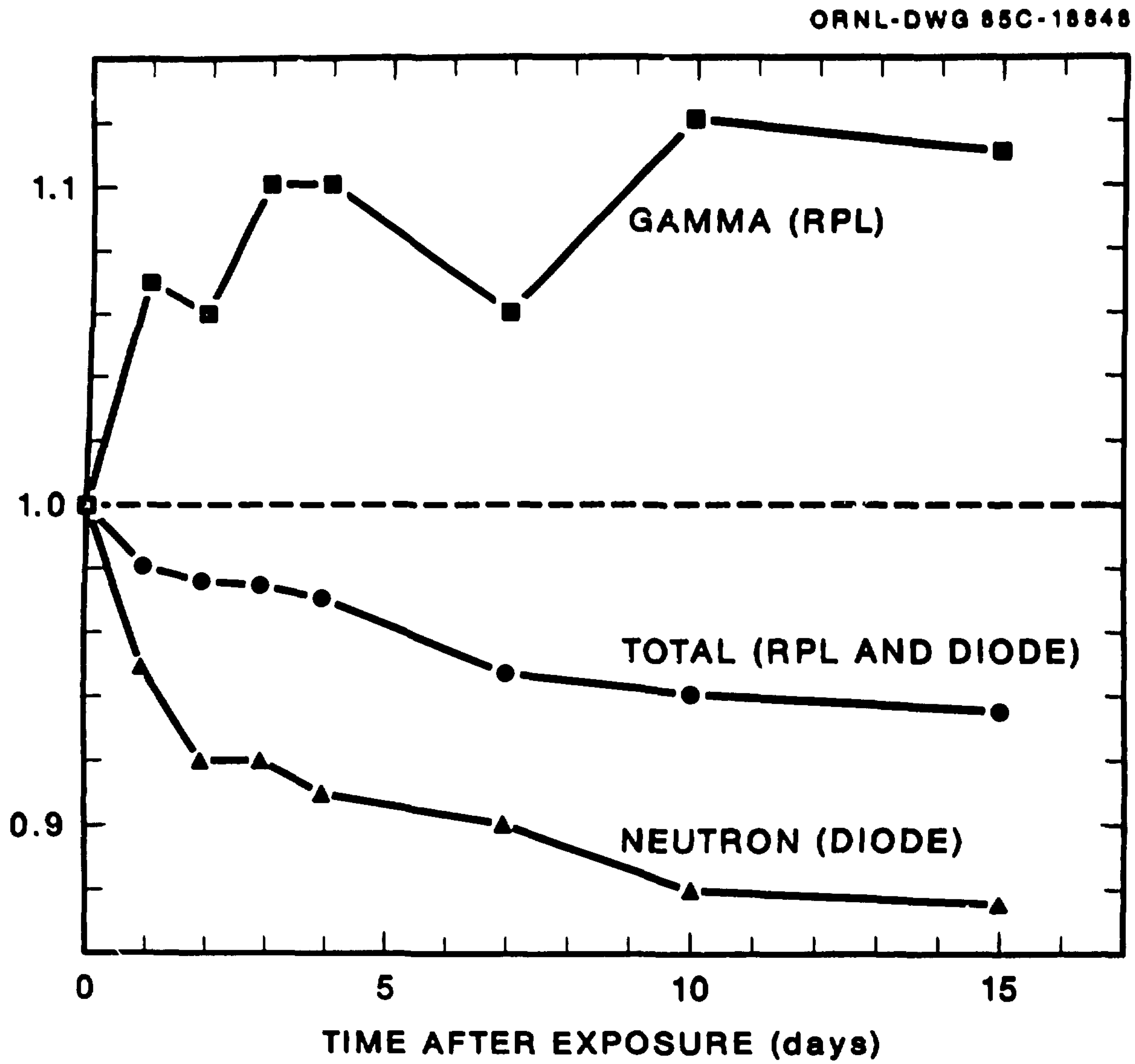

Figure 7. Measured total, gamma, and neutron dose fading characteristics for the DT-236 dosimeter.

Table 1. Indicated doses for 188 DT-236 dosimeters before irradiation

Dose	Digital dose indication, Gy		σ (%σ)[a]
	Range	Mean	
Total	0.16–0.66	0.40	0.09 (22)
Neutron	0.02–0.38	0.17	0.07 (45)
Gamma	0.08–0.38	0.23	0.08 (34)

[a] One standard deviation about the mean in Gy (percent of the mean of one standard deviation).

Table 2. Characteristics of HPRR radiation fields[a]

Shield	Neutron data		Neutron-to-gamma
	Mean energy, MeV	Median energy, MeV	dose ratio[b]
None	1.306	0.790	6.2
13-cm steel	0.780	0.430	7.8
20-cm concrete	0.885	0.167	2.2
12-cm Lucite	0.951	0.183	1.1

[a]Data at 3 meters from the HPRR with the reactor operated over Pit 1 at 1.4 meters above the floor.

[b]Ratio of neutron and gamma doses (tissue kerma) in air at 3 meters from the reactor.

Table 3. Pulse data for the battlefield dosimeter evaluation

Date	Pulse number	Shield	Fission yield, x 10^{16}	Reference doses, Gy[a]		
				Neutron	Gamma	Total
9/13/85	1004	None	3.91	1.37	0.22	1.59
9/16/85	1005	None	8.85	3.10	0.50	3.60
9/18/85	1006	Concrete	8.67	0.56	0.25	0.81
9/19/85	1007	Steel	9.28	1.41	0.18	1.59
9/20/85	1008	Lucite	7.11	0.40	0.37	0.77
9/23/85	1009	None	6.69	2.34	0.38	2.72
9/24/85	1010	Steel	6.10	0.93	0.12	1.05

[a]Reference values (tissue kerma) in air at 3 meters from the reactor.

Table 4. Measured and reference total dose data[a]

Date	Shield	Dosimeter distance, m[b]	Total dose, Gy[c] Reference	Total dose, Gy[c] Measured	Measured/ Reference	σ(%σ)[d]
9/16/85	None	1.43	13.98	17.01	1.22	0.20(1.2)
9/16/85	None	1.66	10.61	16.49	1.55	0.11(0.7)
9/23/85	None	1.50	9.67	14.16	1.46	0.43(3.0)
9/16/85	None	2.06	7.13	11.22	1.54	0.20(1.8)
9/23/85	None	2.00	5.65	7.41	1.31	0.78(1.1)
9/23/85	None	2.50	3.77	4.69	1.24	0.10(3.9)
9/16/85	None	3.00	3.60	4.96	1.38	0.13(2.6)
9/13/85	None	2.00	3.30	4.20	1.28	0.03(0.7)
9/23/85	None	3.00	2.72	3.45	1.27	0.07(2.0)
9/23/85	None	3.50	2.06	2.58	1.25	0.05(2.0)
9/13/85	None	3.00	1.59	2.13	1.34	0.03(1.6)
9/19/85	Steel	3.00	1.59	2.03	1.28	0.01(0.6)
9/23/85	None	4.00	1.52	2.01	1.32	0.05(2.5)
9/16/85	None	4.97	1.45	2.09	1.44	0.05(3.2)
9/23/85	None	5.00	1.09	1.57	1.44	0.10(6.4)
9/24/85	Steel	3.00	1.05	1.39	1.32	0.01(1.0)
9/18/85	Concrete	3.00	0.81	1.15	1.42	0.09(7.1)
9/23/85	None	6.00	0.78	1.07	1.37	0.11(10.0)
9/20/85	Lucite	3.00	0.77	0.99	1.28	0.05(4.0)
9/16/85	None	7.35	0.72	0.98	1.36	0.07(7.4)
9/16/85	None	15.00	0.20	0.29	1.45	0.04(12.0)

[a]Total neutron and gamma doses measured in air (on ringstands).

[b]Distance from reactor vertical centerline to the dosimeter centerline.

[c]Doses given in terms of tissue kerma.

[d]One standard deviation about the mean in Gy (percent of the mean of one standard deviation).

Table 5. Measured and reference gamma dose data[a]

Date	Shield	Dosimeter distance, m[b]	Gamma dose, Gy[c] Reference	Gamma dose, Gy[c] Measured	Measured/ Reference	σ(%σ)[d]
9/16/85	None	1.43	1.70	2.15	1.26	0.02(0.7)
9/16/85	None	1.66	1.32	1.87	1.42	0.09(4.8)
9/23/85	None	1.50	1.19	1.54	1.29	0.06(4.1)
9/16/85	None	2.06	0.91	1.41	1.55	0.07(4.8)
9/23/85	None	2.00	0.72	0.98	1.36	0.05(5.2)
9/23/85	None	2.50	0.50	0.65	1.30	0.03(4.0)
9/16/85	None	3.00	0.50	0.83	1.66	0.12(14.8)
9/13/85	None	2.00	0.42	0.59	1.40	0.06(9.4)
9/23/85	None	3.00	0.38	0.47	1.24	0.04(9.6)
9/23/85	None	3.50	0.29	0.33	1.14	0.04(11.8)
9/13/85	None	3.00	0.22	0.34	1.54	0.05(15.0)
9/19/85	Steel	3.00	0.18	0.11	0.61	0.03(30.9)
9/23/85	None	4.00	0.24	0.27	1.12	0.04(13.7)
9/16/85	None	4.97	0.22	0.42	1.91	0.10(23.1)
9/23/85	None	5.00	0.17	0.14	0.82	0.01(6.4)
9/24/85	Steel	3.00	0.12	0.06	0.50	0.02(25.0)
9/18/85	Concrete	3.00	0.25	0.41	1.64	0.05(12.0)
9/23/85	None	6.00	0.12	0.08	0.67	0.04(4.6)
9/20/85	Lucite	3.00	0.37	0.48	1.30	0.06(12.2)
9/16/85	None	7.35	0.12	0.20	1.67	0.05(23.0)
9/16/85	None	15.00	0.04	0.11	2.75	0.02(20.0)

[a]Gamma doses measured in air (on ringstands).

[b]Distance from reactor vertical centerline to the dosimeter centerline.

[c]Doses given in terms of tissue kerma.

[d]One standard deviation about the mean in Gy (percent of the mean of one standard deviation).

Table 6. Measured and reference neutron dose data[a]

Date	Shield	Dosimeter distance, m[b]	Neutron dose, Gy[c] Reference	Neutron dose, Gy[c] Measured	Measured/ Reference	σ(%σ)[d]
9/16/85	None	1.43	12.28	14.86	1.21	0.18(1.2)
9/16/85	None	1.66	9.29	14.62	1.57	0.05(0.3)
9/23/85	None	1.50	8.48	12.62	1.49	0.42(3.3)
9/16/85	None	2.06	6.22	9.81	1.58	0.23(2.3)
9/23/85	None	2.00	4.93	6.44	1.31	0.13(2.0)
9/23/85	None	2.50	3.27	4.04	1.24	0.08(1.9)
9/16/85	None	3.00	3.10	4.13	1.33	0.12(3.0)
9/13/85	None	2.00	2.88	3.61	1.25	0.07(1.9)
9/23/85	None	3.00	2.34	2.98	1.27	0.10(3.4)
9/23/85	None	3.50	1.77	2.25	1.27	0.02(1.0)
9/13/85	None	3.00	1.37	1.79	1.31	0.07(4.1)
9/19/85	Steel	3.00	1.41	1.92	1.36	0.03(1.8)
9/23/85	None	4.00	1.38	1.74	1.26	0.03(1.8)
9/16/85	None	4.97	1.23	1.67	1.36	0.09(5.5)
9/23/85	None	5.00	0.92	1.42	1.54	0.11(7.9)
9/24/85	Steel	3.00	0.93	1.34	1.44	0.11(8.3)
9/18/85	Concrete	3.00	0.56	0.74	1.32	0.08(10.4)
9/23/85	None	6.00	0.66	1.00	1.51	0.11(11.2)
9/20/85	Lucite	3.00	0.40	0.51	1.28	0.08(15.7)
9/16/85	None	7.35	0.60	0.78	1.30	0.05(6.0)
9/16/85	None	15.00	0.16	0.18	1.12	0.06(3.2)

[a]Neutron doses measured in air (on ringstands).

[b]Distance from reactor vertical centerline to the dosimeter centerline.

[c]Doses given in terms of tissue kerma.

[d]One standard deviation about the mean in Gy (percent of the mean of one standard deviation).

Table 7. Temperature effects on dosimeter response

Condition	Time after exposure, hours	Relative measured dose[a]					
		Total		Neutron		Gamma	
		Hot[b]	Cold[c]	Hot	Cold	Hot	Cold
Read at room temperature[d]	2	1.00	1.00	1.00	1.00	1.00	1.00
Stored and read at new temperature	24	0.77	1.00	0.76	1.00	0.84	1.22
Stored and read at room temperature	28	0.86	0.99	0.79	0.99	1.25	0.93
Stored and read at new temperature	48	0.77	1.01	0.76	0.98	0.81	1.13
Stored and read at room temperature	52	0.81	0.99	0.76	1.00	1.08	0.92
Stored and read at new temperature	72	0.77	1.03	0.75	0.97	0.85	1.27
Stored and read at room temperature	76	0.81	0.96	0.76	0.97	1.12	0.87

[a]Measured dose divided by the value measured at room temperature two hours after exposure.

[b]Hot temperature = 45° C.

[c]Cold temperature = 0° C.

[d]Room temperature = 20° C.

Table 8. Total dose response on torso and arm phantoms relative to the response in air[a]

	Ratio of phantom-to-air measured total doses	
HPRR Spectrum	Torso phantom[b]	Arm phantom[c]
Unshielded	1.15	1.07
Steel	1.18	1.10
Concrete	1.07	1.08
Lucite	1.07	1.12

[a]Measurements made with dosimeter centerlines at 3 meters from the HPRR.

[b]Standard BOMAB torso section - elliptical 20 cm x 30 cm cross section and 40 cm high. Phantom is made of polyethylene and filled with tap water.

[c]BOMAB arm section - right circular cylinder 10 cm in diameter and 40 cm high. Phantom is made of polyethylene and filled with tap water.

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