Advances in External Dosimetry at the Savannah River Site

Dante' W. Wells

Abstract

External dosimetry is the science of determining the external radiation dose to personnel that work in radiation fields. The determination of dose to the worker is required for beta, gamma, and neutron radiation that personnel are exposed to in the workplace. The Savannah River Site (SRS) has measured the external exposure to radiation workers since November 1951. During this nearly 50-year period, external radiation dosimetry methods and technological advancements have led to many improvements in techniques that are now used to quantify the measured occupational radiation dose. This paper presents the evolution of beta, gamma, and neutron external radiation dosimetry at SRS. This paper will also highlight the accomplishments of the talented personnel responsible for the innovative techniques created over the years to ensure the quality and validity of the external dosimetry results.[1]

Introduction

The science of external dosimetry was developed to determine radiation dose imparted to living tissue by three types of ionizing particles—beta radiation, gamma radiation, and neutron radiation. The region of the body affected by the action of a single ionizing particle is small and the damage caused is insignificant. However, the damage produced by successive particles accumulates, and, if enough energy is imparted, the consequences can become detrimental. External dosimetry programs determine the amount of dose accumulated over an extended period of time (i.e., monthly or quarterly). The Savannah River Site (SRS) has used several types of nuclear emulsion dosimeters and thermoluminescent dosimeters (TLDs) to maintain an increasingly cost-effective, timely, and accurate external dosimetry program.

Beta-Gamma Dosimetry, Film Badge Dosimeter (1951 to 1970)

Beta and gamma radiation monitoring at SRS began on November 12, 1951, with the distribution of 50 Oak Ridge National Lab (ORNL)-designed film badge dosimeters attached to the Site security badge. This original dosimeter used the Du Pont manufactured Type 552 x-ray film; however, a new film packet, Du Pont manufactured Type 558 x-ray film, which was more sensitive than the ORNL type, replaced the Type 552 packet on March 3, 1952. In the mid 1950s, Du Pont created the Type 555 film packet, which replaced the Type 558, and became the dosimeter predominately used at SRS. Beginning in March 1952, the Personnel Monitoring Section began to routinely calibrate beta-gamma personnel monitoring film. In January 1953, the Site could not yet process film badges. The dosimeters were worn for one week, collected, and transported to ORNL for processing. ORNL would then interpret the dose and forward the results to the Health Physics Group. The Site began to process film badges in March 1953 using the ORNL film badge dosimeter. The film had a minimum recordable dose of 30 mrem. The film was changed each week in the operating area and developed. The results were reported in mrem per week, manually recorded on individual exposure record cards, and kept in individual file folders. By early 1954, this operation required 45 people.

A new film badge dosimeter, designed by the Engineering Assistance Section, was introduced into service on November 9, 1959. This dosimeter contained a single packet of standard dosimeter film that could be inserted either manually or automatically using a specially made positioning jig. Manually processing film and data was a time-consuming practice, and human error was a concern because of the number of steps involved. To improve the efficiency of badge processing, an automatic dose computer was developed in 1959 by the Works Technical Department. This device was a combination densitometer and analog computer that determined film exposure. This computer performed satisfactorily in routine service for more than a year before it was replaced with a similar computer that used magnetic amplifiers. The new computer and reader accurately measured personnel exposures up to 1 R and 1 rad. Exposure data were recorded on IBM cards using a digital voltmeter and an IBM card punch. The reader was the key unit in the highly mechanized film badge handling/reading/recording system, which processed 3750 badges a week in the early 1960s.

In 1965, approximately 150 film badge dosimeters developed at SRS were modified to add LiF thermoluminescent dosimeters enriched to 99.91% Li-7, which was relatively insensitive to neutrons. This modification was made to evaluate the accuracy and reliability of TLDs if used for personnel monitoring. Based on previous studies, it was known that unshielded film overresponded twentyfold to 17 keV x-ray or gamma radiation compared to an over-response of 30% for the TLD. When placed in the film badge, the 30% over-response of the LiF was offset by a 0.15-inch plastic shield that effectively reduced the 17 keV radiation to 65% of its initial value. Film filtered with 1 g/cm^2 silver was known to overrespond twofold when exposed to 100 keV gamma radiation. Alternately, LiF dosimeters did not indicate any significant over-response for this energy.

On April 1, 1970, TLDs replaced film as the principal means of measuring external beta-gamma exposure. This change was based on the cost savings from extending low-exposure personnel to a quarterly badge cycle and the improvements in dosimetry accuracy from the near tissue equivalent response of the TLD regardless of photon energy. During this time, approximately 4600 employees were monitored by this system with 1600 participating in the quarterly cycle. This system was manually operated and used electronics and TLDs that were available commercially. Two ^{7}LiF chips were incorporated into the existing film badge. One TLD chip was positioned in the center of the open window, while the other was placed behind the aluminum shield (460 mg/cm^2) on the badge cover. This arrangement distinguished skin from penetrating exposures. These lithium chips were manufactured by Harshaw Chemical Company, and dosimeters were ordered in large uniform batches.

The TLD response to beta radiation is a function of energy, absorber thickness, and TLD self-absorption. The resulting skin dose depends on comparable factors where skin and outer clothing thickness absorb radiation incident on the body. An automatic, computer-controlled TLD badge processing system began operation in January 1973. Badge racks (25-badge capacity) that were compatible with the automated TLD system were used as pickup and deposit points for badges. They were located at convenient locations in areas where employees were assigned.

The TLDs were relatively insensitive to many of the environmental factors and chemicals to which film responded. The change from film to TLDs resulted in a significant reduction in exposure investigations. No significant differences in accumulated exposures were observed in any of the four general work areas during changeover from film to TLD dosimetry.

On July 1, 1982, the TLD badge was replaced by a commercial dosimeter manufactured by Panasonic. Frequent breakdowns of the one-of-a-kind badge reader used prior to this time made it difficult to process the number of TLDs that were in service (~4000 monthly and ~2500 quarterly). At times, personnel exposure data was not available for up to three weeks. The ability to meet impending DOE standards for dosimetry performance also was considered before adopting the Panasonic system. The new badge was one-third the size of the previous one.

Neutron Dosimetry (1951 to Present)

Monitoring personnel for exposure to neutrons began in 1951 using neutron track emulsion (NTA) film and the ORNL film badge. Initially, this film was replaced once per week and shipped to Oak Ridge for processing. On August 3, 1953, the Site assumed the responsibility of processing NTA film. The first badge used at the Site for personnel neutron monitoring was obtained from ORNL. It consisted of a film packet holder with an embossing plate for film identification, a 1-mm sterling silver filter, and a 1/2-inch diameter open window. The film packet holder was connected to a plastic badge equal in size to the SRS security pass. The dosimeter was exactly the same as the one used for beta-gamma personnel monitoring with the exception of the film. An Eastman Kodak NTA dental-size film packet was used in the neutron badge. The packet was inserted in the badge with the emulsion facing the back because it was thought that proton recoil yield from hydrogenous material would increase if the plastic faced the neutron source. A Site-designed film badge dosimeter replaced the ORNL dosimeter for personnel neutron monitoring in November 1959. Effective with the first exchange of NTA film badges after July 14, 1960, the badge cycle was extended to two weeks. The film was wrapped in aluminized mylar packets to prevent track fading from atmospheric conditions.

The procedure used for neutron dosimetry involved counting recoil proton tracks in NTA film. This method was accurate for neutron energies above 500 keV. The neutron film was managed separately from the beta-gamma film. Personnel responsible for reading the film (microscope observers) made all film changes and assisted in the darkroom. Upon return the film was then taken to the darkroom for developing. After developing, the film was read using a microscope to count proton recoil tracks produced in the NTA film emulsion. Considering only the shielded portion of the film, three sets of 40 fields were counted and the average number of tracks determined. By 1959 the NTA counting procedure had been modified. The tracks were counted on the calibration film; tracks produced in a blank film, an "error" figure (the statistical variation at 90% confidence level for the number of tracks observed) was subtracted for each location; and the number of tracks which corresponded to a 300 mrem exposure ("C" number) was determined. The dose recorded on personnel films was then determined by comparing the "C" number to the number of tracks on the film. Neutron film interpretation was never automated. During this time, badges were routinely interpreted by one technician. Approximately 400 NTA dosimeters were used per two-week badge cycle, and only 40-45 badges could be processed per day. Due to NTA film limitation in measuring low energy neutrons and the difficulties in interpreting this film when exposed to gamma fields greater than 0.5 R, an alternative neutron dosimeter had to be developed to detect lower energy neutrons produced by shielding effects in production facilities.

A prototype thermoluminescent neutron dosimeter (TLND), designed to detect albedo neutrons (mainly of thermal energy), was developed in 1968. Evaluation of the prototype for use in personnel monitoring was completed in October 1969.

The prototype TLND did not perform satisfactorily because of its large size and because the mounting clip did not maintain a consistent distance between the body and the dosimeter. This badge responded accurately (±20%) under laboratory conditions; however, it was found that a gap of 1/4-inch between the badge and the body could result in a difference as great as 48% in response. Thus, a new design was developed and completed in the spring of 1970.

A belt was incorporated into the design of the TLND to ensure a reproducible, consistent geometry during exposure in the field. This resulted in a snug fit when mounted against the wearer's body. The badge was modified further by reducing the badge size and weight by incorporating both pairs of TLD chips into to a single 2-inch diameter hemisphere. This arrangement covered the outer surface of the cadmium with stainless steel to increase the durability of the badge, providing a quick disconnect back plate to facilitate access to TLDs, and by modifying the thickness of the cadmium in the dome section to adjust the energy response of the dosimeter.

The redesigned TLND included a 2-inch polyethylene hemisphere that was sliced 1/4-inch from the rounded end (Hoy 1972). These two sections were separated by a cadmium shield plate. A pair of LiF chips, TLD-600 and TLD-700, was placed in the dome section of the hemisphere. Another pair of chips was placed in the hemisphere's base section. The curved surface of the polyethylene hemisphere was covered with a 1/32-inch-thick layer of cadmium, except for a 1/2-inch-radius area in the center of the dome, which was only 0.003 inch thick. This thin area allowed a small fraction of the incident thermal neutrons to reach the dosimeter chips in the smaller section of polyethylene. The cadmium was topped with a shell of 20-gauge stainless steel for protection. The unit was held together by a stainless steel back plate that has slots for attaching a belt to the TLND. When assembled, one pair of LiF chips was held in the dome section of the badge and was entirely surrounded by cadmium. The other pair was shielded by cadmium from the front only. The unit was designed with a small hole in the edge of the assembled badge to lock the unit together. A minimum recordable dose of 10 mrem was used for the TLND throughout its entire use period. Beginning January 1, 1995, the Hoy TLND was replaced with the Panasonic UD-809 albedo neutron dosimeter. The primary reason for the change was the difficulty involved in maintaining quality control for large groups of batch TLD chips, resulting from increased personnel neutron monitoring. This change permitted automated processing of personnel neutron dosimeters.

The UD-809 contains four elements shielded with cadmium and tin, three of ^{6}Li and one of ^{7}Li. The dosimeter is housed in a plastic case and covered with a plastic badge cover. It is worn in conjunction (i.e., issued in the same holder) with the Panasonic UD-812 beta-gamma dosimeter. Technical Specifications of this dosimeter are given in site developed technical manuals. While the Panasonic UD-809 is not as sensitive as the TLND and requires facility correction factors, it has the following advantages:

- Use of individual dosimeter calibration factors.
- Can be read in the same reader as beta-gamma dosimeters.
- Dosimeter results and their algorithms are processed via the same computer as the beta-gamma dosimeters.
- The dosimeters are not as bulky as the TLND.

As a result, the quality control and cost-effectiveness of the neutron dosimetry program has been improved, and many of the issue/control problems have been eliminated. The minimum reportable dose for the Panasonic UD-809 is 15 mrem.

Criticality Dosimetry

The Site has never experienced a criticality accident. However, because operations involve processing fissionable materials, the potential for a criticality accident to occur does exist. Although the chances of an accident of this type are very low because of various engineering and administrative controls, the radiation dose received by personnel must be quickly determined if an accident were to occur. A criticality dosimetry program has been in place since 1960 to assess an accident dose. Before 1960, special combinations of gamma- and neutron-sensitive film along with neutron-sensitive ionization chambers could have been used to measure gamma exposures from 20 mR to 1000 R and neutron exposures up to 1 rem. However, a criticality accident could have possibly resulted in radiation exposures greater than the upper sensitivity limits of these instruments. The Site developed the criticality neutron dosimeter (CND) in the late 1950s. A three-phase dosimetry system was established over the years to respond to a criticality accident. In the first phase, all potentially exposed personnel are screened. Indium foils in the personnel dosimeter and security badge would be activated by neutron exposure. The second phase involves approximating the neutron dose by analyzing ^{24}Na in blood. The third and final phase consists of a more accurate dose determination using a dosimeter capable of measuring dose over a wide range of energies. The CND's design has been modified over the years to improve its functionality.

The CND's components were assembled in a 3.5-4 inch long by 0.5-inch-diameter plastic tube. A clip is attached so it can be placed on the wearer's pocket. Indium, copper, and cadmium foils were shaped into hollow cylinders to lessen directional effects. These foils, along with specific amounts of sodium fluoride and sulfur, are contained in three small polystyrene vials. These materials are pre-weighed to expedite processing after an accident. The neutron fluence would be determined for five neutron energy levels based on the activation of indium, copper, and sulfur. Fluence values are then corrected for the direction of exposure because the CND is worn on the front side of the body, and activation is affected by body shielding and moderation. If left uncorrected, the dose estimate may be low by a factor of 2 for instances where the exposure is received from the backside of the body. The neutron fluence is corrected by comparing the amount of ^{24}Na in the blood of the exposed individual to the amount in the sodium fluoride in the dosimeter. The resulting value is compared to an experimentally derived ratio that provides the relationship between sodium activation in the blood and CND, neutron energy, and orientation of the dosimeter. The total neutron dose in rads is determined by multiplying the fluences in the five energy ranges, as determined from counting CND materials, by the respective dose conversion factors for that energy interval, then summing the five doses. The neutron dose is reported in rad since the quality factor for neutrons in a criticality accident has not been established. A lithium fluoride TLD, contained in a polyethylene vial, is used to measure gamma dose ranging from 25 mR to 10^6 R. Gamma dose measurements are corrected for direction of exposure.

Measurement Quality

Over the years, the Site has participated in many formal and informal intercomparison programs to assess the performance of its dosimetry program against other facilities. Site results usually were formalized in reports to management. In addition, an internal quality assurance and control program has been developed to ensure the accuracy of reported measurements.

A routine audit program was started in 1955 to verify the accuracy of the beta-gamma and neutron film badge monitoring programs. The audit program consisted of introducing a group of previously exposed badges (blind spikes) into the routine monitoring program. Originally, film exposed to known radiation levels was placed in visitor badges, which were then processed normally. Fictitious names were placed on badges to avoid entry of false records into the system.

As the dosimetry program evolved, the blind spike audit remained as an integral part and is still used today.

Calibration dosimeters have always been processed with film and TLD badges. These badges were interspersed with regular badges during processing to verify that the film or TLD reader was operating properly. This on-line quality control check is still used today.

Department of Energy Laboratory Accreditation Program (DOELAP)

In the early to mid 1980s, the ANSI Review Committee was charged with establishing performance standards for external dosimetry. Out of this effort evolved the National Voluntary Laboratory Accreditation Program (NVLAP). NVLAP participants were tested against ANSI standards and accredited in dosimetry services. Because of technical differences of opinion, DOE pursued developing its own performance standard for external dosimetry and accreditation program called DOELAP. Preliminary testing for DOELAP performance was conducted in the mid 1980s. The Site participated in these tests when DOELAP was finalized in the late 1980s. The Site was accredited and has maintained that accreditation, having received its most recent biennial accreditation in August 1999.

Presently, the Site participates in a Quarterly DOELAP/NVLAP External Dosimetry Processors TLD Badge Intercomparison Program. Two DOE facilities presently participate in the program, which serves as a site-specific blind spike check on TLD badge processing and irradiation protocols.

Endnotes

1. The historical information contained in this proceedings paper was derived from WSRC-RP-95-0234, *A History of Personnel Radiation Dosimetry at the Savannah River Site*.

Biography

Dante' W. Wells has been the Group Technical Lead for the Savannah River Site external dosimetry program since 1992. He was the Manager of the Internal and External dosimetry programs at Fernald in Ohio from 1988 to 1992. Prior to that he was the corporate radiation specialist for Entergy in Mississippi (1985-1988) and the REMS site coordinator at the Carolina Power & Light's HB Robinson Nuclear Station (1982-1985). Mr. Wells received his BS in biology/health physics degree from Virginia Polytechnic Institute and State University in 1982. He is a Lead Department of Energy Laboratory Accreditation Program (DOELAP) assessor for the DOE Headquarters EH-51 section through the Radiological and Environmental Sciences Laboratory DOELAP Performance Evaluation Program Administrator.