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Neutron Personnel Dosimetry Considerations for Fusion Reactors

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T. P. Barton and C. E. Easterly

ABSTRACT

The increasing development of fusion reactor technology warrants an evaluation of personnel neutron dosimetry systems to aid in the concurrent development of a radiation protection program. For this reason, current "state of knowledge" neutron dosimeters have been reviewed with emphasis placed on practical utilization and the problems inherent in each type of dosimetry system. Evaluations of salient parameters such as energy response, latent image instability, and minimum detectable dose equivalent are presented for nuclear emulsion films, track etch techniques, albedo and other thermoluminescent dosimetry techniques, electrical conductivity damage effects, lyoluminescence, thermocurrent, and thermally stimulated exoelectron emission. Brief summaries of dosimetry regulatory requirements and intercomparison study results help to establish compliance and recent trends, respectively.

Spectrum modeling data generated by the Neutron Physics Division of Oak Ridge National Laboratory for the Princeton Tokamak Fusion Test Reactor (TFTR) Facility have been analyzed by both International Commission on Radiological Protection fluence to dose conversion factors and an adjoint technique of radiation dosimetry, in an attempt to determine the applicability of current neutron dosimetry systems to deuterium and tritium fusion reactor leakage spectra. Based on the modeling data, a wide range of neutron energies will probably be present in the leakage spectra of the TFTR facility, and no appreciable risk of somatic injury to occupationally exposed workers is expected. The relative dose contributions due to high energy and thermal neutrons indicate that neutron dosimetry will probably not be a serious limitation in the development of fusion power.

INTRODUCTION

Development of fusion reactor technology has reached the stage where preliminary evaluations of personnel safety are necessary to establish the groundwork for a comprehensive radiation protection program. While an operating fusion power reactor is still not expected before the early part of the twenty first century, major advances toward attaining the break-even conditions necessary for practical magnetic fusion energy have been sparked by recent progress in plasma containment and heating. This report is part of an ongoing effort at the Oak Ridge National Laboratory (ORNL) to study the health physics aspects of fusion power.

There is still a degree of uncertainty as to which reactor concept will be the first to become feasible for fusion power production. Among the designs under close consideration are the magnetic mirror systems (e.g., tandem mirror), the toroidal system (e.g., tokamak and theta pinch), and the inertial confinement systems (e.g., laser-heated pellet fusion). It is generally accepted, however, that the first generation of fusion reactors will use deuterium and tritium (D-T) as fuel, due to the high reaction cross section and low threshold temperature required (relative to other reactions under consideration). This reaction

$$ {}_{1}^{2}D + {}_{1}^{3}T \rightarrow {}_{2}^{4}He\ (3.5\ MeV) + n\ (14.1\ MeV) \qquad (1) $$

is highly exothermic, giving up a total of about 17.6 MeV (~94,000 kW-hr/g fuel). Although deuterium occurs naturally (about 1 part in 6000 in

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water), and can be separated easily and cheaply, tritium must be obtained by other means, since it occurs naturally in much smaller concentrations. Tritium production can be accomplished by surrounding the plasma of a fusion reactor with a lithium blanket and breeding tritium in the reactions:

$$^{6}_{3}Li + n_{slow} \rightarrow {}^{3}_{1}T + {}^{4}_{2}He + 4.8\ MeV \qquad (2)$$

$$^{7}_{3}Li + n_{fast} \rightarrow {}^{3}_{1}T + {}^{4}_{2}He + n_{slow}' - 2.5\ MeV. \qquad (3)$$

It is certain that a large number of neutrons will be produced in a self-sustaining fusion reaction, and personnel working around a reactor may possibly be exposed to neutron spectra and intensities which to date have not been routinely encountered by large numbers of workers. It will be necessary to quantitatively evaluate these exposures so as to meet radiation protection guidelines and regulations as set forth by national and international agencies. The dosimetry system used for routine personnel neutron monitoring around fusion reactors will have to be portable, inexpensive, easily evaluated, and above all else, accurate. This accuracy will depend on a variety of factors, some of the more important of which are energy dependence, effect of spatial orientation, latent image (information) fading, minimum detectable dose equivalent, and the ability to distinguish between high- and low-linear energy transfer (LET) radiations (i.e., neutrons and gamma rays).

It is the objective of this report to discuss the "state of knowledge" in neutron dosimetry, including currently used dosimetry

systems as well as those being investigated as potential replacements or supplements. Current trends and intercomparisons will be summarized, with a certain degree of emphasis placed on energy dependence, and its effect on fluence to dose equivalent conversion. Since the neutron spectra which will be encountered by workers around a fusion reactor is practically unknown, such spectra will be calculationally modeled, with the associated dose equivalent rates to personnel in the modeled areas. The spectra will also be analyzed by a determination of risk associated with exposure per unit time, keeping in mind the recent data which suggest that occupational exposure to neutrons is significantly more hazardous than is reflected in established quality factors, or the current maximum permissible annual dose equivalent. In summary, this report provides an indication of the applicability of present day neutron dosimetry techniques to the proposed fusion reactor systems.

NEUTRON DOSIMETRY REVIEW

Neutron Dosimeter Requirements

Requirements and recommendations for personnel neutron dosimeters have been established by the United States Nuclear Regulatory Commission (NRC) and the American National Standards Institute (ANSI), respectively. Section 20.202, "Personnel Monitoring," of Title 10, Chapter 1, Code of Federal Regulations (10 CRF Part 20)[1] states the legal requirements of licensees and Regulatory Guide 8.14 of the NRC[2] and ANSI N319-1976[3] provide a discussion of performance and calibration criteria, use factors, and accuracy requirements. Adherence to the standards set forth in these latter two documents will insure compliance of the licensees to all requirements. Some of the important recommendations in these documents include:

1. the dosimetry system shall be capable of measuring dose equivalents between 300 millirem and 10 rem (per quarter).
2. the dosimetry system shall be capable of detecting 1 rem of neutrons in the presence of 3 rem of gamma rays (E_γ > 500 keV).
3. the standard deviation of measured neutron responses from 10 dosimeters exposed to approximately 1 rem under identical conditions shall be $\leq$30%.
4. the average accuracy of 10 dosimeters exposed to an unmoderated ^{252}Cf source (100 millirem to 3 rem) should be ±50%.
5. the dosimeter will meet all above requirements when subjected to the following environmental factors after exposure:

a. temperature extremes of 0°C and 45°C for 1 week.

b. relative humidity of 90% for 1 week.

c. normal artificial room light or sunlight for the extent of the dosimetry period.

d. a drop to a hard surface from a height of 1.5 m.

With these regulations in mind, the desirable features of personnel neutron dosimeters can be identified. These include:

1. relatively low minimum detectable dose equivalent,
2. energy response which approximates that of soft tissue (according to International Commission on Radiological Protection (ICRP) Publication 21),[4]
3. sufficient accuracy and precision,
4. relatively insensitive to other types of radiation,
5. negligible latent image instability,
6. relatively low cost, and
7. low toxicity.

These features, combined with the fact that an acceptable neutron dosimetry system should ideally function over about nine decades of energy (10^{-8}-10^{+1} MeV), place rather stringent requirements on potential neutron dosimeters. Consideration of these parameters is necessary in the evaluation of potential neutron dosimetry systems.

Nuclear Emulsion Film

Nuclear track emulsion film was used for neutron dosimetry as early as 1947[5] and remains the most widely used and commercially available

system for routine personnel neutron monitoring. A measure of neutron dose can be obtained by counting the number of recoil proton tracks produced in nuclear emulsion film as a result of elastic scattering reactions between fast neutrons and hydrogen atoms in the emulsion and film wrapper. A variety of emulsions have been used for fast neutron monitoring (e.g., Ilford C-2, Kodak NTA and NTB), with the Kodak NTA being the most widely used in recent years. The associated problems of neutron films were recognized early, with significant advances made until about 1954, but these problems were never entirely resolved.[6]

Major disadvantages of nuclear emulsions include latent image instability and an extreme energy dependence. The latent image fading phenomenon has been shown to be due to the oxidation of the silver grains by atmospheric oxygen in the presence of water vapor.[7] This process is a function of the temperature and humidity to which the film is subjected. Sealing the film in a variety of packages with and without dessicants has been attempted by many investigators with varying degrees of success.[8-12] Another somewhat successful approach has been to subject calibration films to the same environmental conditions as the monitoring films, thereby building a fading correction into the calibration procedure.[11]

The neutron response of nuclear emulsions (tracks-rem^{-1}) can vary by as much as a factor of 3 or more over the energy range from about 0.5-14 MeV. A complex packaging technique which partially alleviated this problem was designed and marketed;[12] however, this system is no longer commercially available. Nuclear emulsions are generally considered

to have a low-energy cutoff of about 0.5 MeV, but the effective energy cutoff is more on the order of 0.7-1.0 MeV in practical field exposures, depending on the individual technician's ability to discern short tracks.[6,13] In moderated low-energy neutron spectra, such as the leakage spectra from large nuclear reactors, or shielded glove boxes, neutron film will fail to accurately record the neutron dose.

Other problems associated with nuclear-emulsions include the following:

1. considerable angular dependence leading to approximations in calibration procedures,
2. decreasing neutron response with increasing gamma background,
3. relative neutron response which decreases with decreasing neutron energy at a constant gamma background,
4. thermal neutron contamination of the measured field producing proton tracks (as a result of the $^{14}N(n, p)$ reaction in the emulsion) which are indistinguishable from other recoil proton tracks,
5. low statistical accuracy (relatively few tracks are produced for dose equivalents in the millirem range),
6. tracks counted by individuals, leading to an increased probability of variation and error, and
7. the time consuming (15-30 minutes/film) and costly necessity of visual scanning for evaluation.

The minimum detectable dose equivalent (MDDE) of nuclear emulsions depends primarily on the type of spectrum measured and the statistical

error defined as acceptable. These factors lead to a wide range of estimated detection limits for various systems, extending from a MDDE of 2 millirem for fast neutrons from a PuBe source,[11] up to a MDDE of 450 millirem for fission neutrons from ^{252}Cf.[14]

For high energy neutron spectra from accelerators and certain radioactive neutron sources (e.g., PuBe and AmBe), neutron films have proven relatively satisfactory due to the low MDDE and a nearly flat energy response in the high energy range. The problems of latent image instability and energy dependence (in conjunction with other problems) have, however, been almost insurmountable thus far, with little significant progress made over the last twenty years. Nuclear emulsions must, therefore, be judged unsatisfactory for generalized personnel neutron monitoring, especially in the environments of highly moderated neutron spectra such as may be present in large fusion power reactors.

Track Etch Techniques

The application of electrochemically etched solid state nuclear track detectors to the field of neutron dosimetry has mushroomed over the last decade. These devices (usually polycarbonate films) are used to record the products of neutron-induced nuclear reactions [e.g., fission fragments, recoil nuclei, alpha particles from (n, α) reactions]. The damage tracks produced in the detector film can be processed and counted to yield a quantitative evaluation of neutron dose.

Fission fragment dosimeters, generally popular as stationary criticality monitors, have recently been used for personnel neutron

dosimetry.[15-20] When a fissile radiator is placed in contact with a polycarbonate film and exposed to a field of neutrons, heavy charged particles (fission fragments) produced via neutron and fission reactions enter the detector film and cause the formation of damage tracks. Electrochemical etching of the detector films increases track widths to the point of visibility with an ordinary optical microscope.[21] An automated spark counting technique[22] has also been developed which removes the tedium of manual counting, and partially eliminates the statistical error associated with low doses (i.e., low track densities).

Natural and enriched uranium, ^{232}Th, and ^{237}Np have all been incorporated into fission fragment radiator foils, with ^{237}Np exhibiting several distinct advantages. These include a low spontaneous fission rate, relatively low effective energy threshold ($\simeq$0.6 MeV), a response above this threshold that is nearly proportional to dose equivalent, and a very low sensitivity to high-energy gamma rays relative to neutrons.[20]

Disadvantages of fission foil dosimeters include spontaneous fission induced background tracks, high cost, energy threshold and radiotoxicity.[23] The problem of radiotoxicity seems to be a major factor limiting more widespread use of fission foil dosimeters. Calculations by Cross and Ing indicate that the dose equivalent received by external exposure to 0.6 milligrams of ^{237}Np, 5 cm from a critical organ, is 34 millirem over a 2000-hour working year, with the calculated maximum skin dose 5 mm from the radiator about 40 times higher.[20] In many facilities using neutron radiation, either regulation or policy prohibits

the use of radioactive materials for personnel dosimetry. The inherent radioactivity of fission foils becomes a self-induced stumbling block inhibiting their further use.

An interesting development in the field of fission fragment dosimetry is the introduction of the "thin film breakdown counter" by Tommasino, et al.[24] This device utilizes the detection of a fission fragment induced, limited-damage, breakdown phenomenon which occurs during a constant voltage application across a thin film metal-silicon dioxide-silicon structure (MOS capacitor). Although research is preliminary, MOS technology and fabrication skills suggest the practicality of developing a compact, fast time response, automatic counting personnel neutron dosimeter, conceptually capable of alarm neutron detection and quantitative personnel dosimetry.[24,25]

Some of the problems associated with fission fragment dosimeters can be avoided by a dosimetry system designed to detect neutron-induced recoil particle tracks. Production of recoil nuclei and (n, α) reactions in an insulating film eliminates the need for an additional radiator, fissile or otherwise. Electrochemical etching of neutron irradiated films produces tracks which are easily visible with an optical microscope or microfiche reader.[21,26]

A major portion of the research on track detectors has been devoted to development and optimization of etching apparatus and parameters.[21,27-38] A variety of etching chambers have been designed and investigated, ranging from experimental chambers built to evaluate

single films and etching conditions,[39] to practical systems capable of processing 960 dosimeters every four hours.[26] Intensive study of the voltage and frequency of etching current, reagent type, temperature, and concentration, sensitivity enhancement by uv-light exposure, and other parameters has led to differing conclusions by different investigators.

One system, offered as a commercial service by R. S. Landauer, Jr., and Co., is fairly typical of the current track etch techniques.[26,34] Polycarbonate foils (without any external radiator) are used as the detecting material. After exposure, the films are processed by electrochemical etching at 1000 volts rms, 2000 Hertz, 22°C in 28% KOH. This system yields a foil background of 25 millirems (4-5 tracks/cm^2) with no measurable gamma response when evaluated on a microfiche reader. The sensitivity is reported to vary from 40 millirem ±30% to 1600 millirem ±7% for AmBe neutrons, with a minimum detectable dose equivalent claimed of 30 millirem (also AmBe neutrons). Fading is less than 5% in 4 months (at 22°C) and the directional response is stated as 40% with phantom rotated 90° from perpendicular incidence.[26,34]

Unfortunately, the energy response of electrochemically etched dosimeters is not very good. Although the polycarbonate film responds fairly well above about 3 MeV, it has an effective energy threshold of about 1-1.5 MeV; a region of extreme importance in personnel neutron dosimetry.[34,39] Investigation of the energy behavior of other types of plastics is underway, attempting to isolate those with a more desirable energy response.[35,40]

Albedo Dosimeters

When an absorbing and scattering medium such as the human body is placed in a field of neutron radiation, some of the incident neutrons are backscattered. These are called albedo neutrons. Positioning a dosimeter on the body to measure this backscattered flux of particles is a technique called albedo-neutron dosimetry, which has been comprehensively investigated for personnel dosimetry. A variety of albedo dosimeters are currently utilized for personnel neutron dosimetry in a number of organizations.[41-43]

"State of the art" albedo neutron dosimeters are designed to measure the thermal neutron flux leaving the body of a person exposed to fast and intermediate energy neutrons. Although any type of thermal neutron detecting material could be used, LiF thermoluminescent dosimeters (TLD's) are most frequently used in practical applications. These detectors use the $^{6}Li(n, \alpha)^{3}H$ reaction as the mechanism for deposition of energy, with about 4.78 MeV of excess energy shared between the alpha and triton particles. Most of this energy is deposited in the TLD crystal since the dimensions of the TLD's are large when compared to the range of these particles in LiF. This reaction effectively differentiates between fast and thermal neutrons, as the thermal neutron cross section is very high ($\sigma \simeq 950$ barns) compared to the fast neutron cross section ($\sigma \simeq 0.3$ barn at 1.0 MeV).

Although natural LiF TLD's are sensitive to thermal neutrons, this sensitivity can be increased by making the TLD out of ^{6}Li enriched

lithium. Conversely, ^{7}Li enriched LiF is produced which has essentially no response to thermal neutrons. Since both of these types of TLD's have approximately the same response to gamma rays, the ^{7}Li TLD reading can be subtracted from the ^{6}Li TLD reading (after appropriate correction for the small difference in gamma response) to obtain the ^{6}Li TLD thermal neutron response. An attempt can then be made to correlate the thermal neutron response to dose equivalent.

Thermal neutron contamination of the incident radiation field will be primarily responsible for the ^{6}Li TLD neutron response (as opposed to albedo neutrons). To utilize effectively the albedo principle, the incident thermal neutron flux must either be removed or greatly reduced. A shield of either cadmium or borated plastic is generally positioned over the TLD's to attenuate incident thermal neutrons, and under ideal conditions the dosimeter's response is largely from the albedo neutrons. Due, however, to practical considerations of dosimeter size and weight, thermal neutron leakage cannot be eliminated. Wearing the dosimeter backwards or away from the body can cause large errors in response, even for incident thermal neutron doses (rem) as small as 1-2% of the fast neutron dose.[44]

A variety of designs of albedo neutron dosimeters have been evaluated with respect to sensitivity, energy dependence, effect of dosimeter orientation, and other applicable parameters.[41,44-53] It has been found that although sensitivity can be altered dramatically by the amounts of thermal neutron absorber and polyethylene used

in dosimeter construction, the energy dependence is about the same for all albedo designs (except for small variations at intermediate energies).[50]

Albedo neutron dosimeters to date have ranged in size and complexity of design from the simple, two-element Hankins-type[44] to the large, belt worn, four-element dosimeter designed by Hoy.[51] Some use cadmium as the thermal neutron absorber, others use boron-loaded plastics, and still others use a combination of materials. Some of these dosimeters were designed for specific purposes, such as a response independent of orientation,[44] automatic read-out,[42] or a response desirable for a certain energy spectrum (e.g., reactor leakage).[53] The minimum detectable dose equivalent of albedo dosimeters is relatively low; ranging from about 5 to 100 millirems,[42,51] depending primarily on the amount and configuration of construction materials, the magnitude of the associated gamma exposure, and the statistical error defined as acceptable.

The energy dependence of albedo neutron dosimeters has been the major problem inhibiting their widespread use. The response of albedo dosimeters to neutrons with energies <400 MeV was calculated using adjoint neutron transport codes by Alsmiller and Barish,[54] and has been experimentally determined by a number of investigators.[41,42,44-51] Consistent results have shown a dosimeter response which decreases rapidly with increasing energy, falling by a factor of about 1000 from 10 keV to 14 MeV.[54] Although many investigators have attempted to modify albedo dosimeters to correct for this energy dependence, studies

have shown that any LiF albedo neutron dosimeter will have basically the same energy dependence.[50] This energy dependence restricts their use to known energy spectra, and then only after proper calibration.

Calibration of albedo neutron dosimeters is generally performed in one of two ways. The dosimeter may be placed on an appropriate phantom, exposed to a known radiation field for a given time, and calibrated by comparing its reading with the total calculated dose equivalent. A much faster method involves using the ratio of count rates from 3-inch and 9-inch spherical neutron monitors in a technique described by Hankins. Both methods are currently used and have been shown to agree within ±18%.[44]

There are several other problems associated with albedo-neutron dosimetry techniques which warrant consideration. If the gamma to neutron dose ratio is high, large statistical errors can be introduced in the dose evaluation. This can be dealt with to a certain extent by using a dosimeter design with a high neutron sensitivity. Albedo dosimeters display a large degree of directional dependence (e.g., response from side and rear exposures of 60% and 20% of the response to front exposure, respectively).[51] Also, as mentioned earlier, orientation with respect to the body can introduce relatively large errors.

After careful consideration of all the necessary parameters, an albedo-neutron dosimeter can be designed to fit many practical situations. It cannot at this time, however, be designed to cope with anything but

a very well defined, unchanging, neutron energy spectrum. Energy distribution fluctuations produced by streaming effects from neutron sources must be effectively eliminated in order to utilize albedo dosimeters effectively. Albedo dosimeters can provide useful dosimetry information only under known spectral conditions, and their practicality for routine personnel must therefore be questioned.

Other TLD Techniques

Thermoluminescent compounds have received the attention of investigators recently in techniques other than albedo dosimetry, and have shown several interesting possibilities for personnel neutron monitoring. These techniques include the measurement of neutron sensitive, high-temperature glow peaks in TLD materials, and the development of hydrogenous radiators to be used in conjunction with TLDs.

Although rare earth doped TLD materials have been utilized for a number of years in routine personnel dosimetry, investigation of their high-temperature glow peak characteristics relative to neutron radiation is fairly recent; the lower temperature glow peaks of TLDs have been used historically for dosimetry as a matter of convenience. Recent work has shown that some TLD materials exhibit two or more salient glow peaks, one of which may be more sensitive to neutrons than the other.[55] A prime example of this phenomenon and the most promising material investigated to date is calcium fluoride doped with thulium (CaF_2:Tm).[55-57] One mechanism suggested for this high temperature glow peak neutron response is that fast neutron induced scattering reactions transfer energy to the higher temperature traps.[57]

The major advantage of the CaF_2:Tm TLD's is the ability to perform gamma and neutron dose measurements at the same time and with the same materials, leading to a high degree of internal precision.[56,57] Due to the mechanism of energy transfer, this technique does not require the dosimeter to be backed by a phantom.[57] Other reported advantages include a relatively low minimum detectable dose (a few millirads),[55] technical ease of evaluation, and low cost. Kapsar and Lucas also suggest that the technique of high temperature glow peak reading in CaF_2:Tm may be particularly useful when used in conjunction with an albedo system.[55]

Incorporation of hydrogenous proton radiators into TLD materials may be accomplished in any of several ways. One of these methods is mixing powdered TLD material with a hydrogenous powder which can be washed away prior to evaluation.[58,59] Rzyski et al. have mixed pure CaF_2 with sugar, and measurements of the fast neutron response yields encouraging results.[59] This mixture produces a dosimetry system with a neutron to gamma relative response ratio considerably larger (~33 times) than CaF_2 without a hydrogenous radiator. Other advantages claimed include a dosimeter which is relatively free of directional dependence, and a linear dose response from about 1 millirad to 13 rads (AmBe neutrons). The major disadvantage of such a system is probably the amount of time and effort necessary for separation of the radiator from the phosphor before evaluation.

Another technique using a hydrogenous radiator is accomplished by permanently mixing a high melting point radiator (capable of withstandin

the high temperatures of TLD read-out) with a TLD phosphor. Some promising results have been obtained with para-sexiphenyl (melting point, 465°C)[60] hot-pressed into pellets with $CaSO_4$:Dy.[61] These pellets have displayed a recoil proton detection efficiency (relative to gamma rays) of about 40% for both 14 MeV and fission spectrum neutrons.[61] Disadvantages of this mixture include the necessity of further purification of the commercially available para-sexiphenyl before use, high cost, and high sensitivity to uv-light.

Lastly, rare earth activated TLD materials have been doped with hydrogen by heating under a hydrogen atmosphere in the presence of aluminum.[62] A variety of TLD-rare earth combinations have been doped with hydrogen and evaluated with respect to neutron sensitivity. Results of this study indicate that the doping concentration reached thus far (10^{17} atoms/cm^3) must be increased by at least two orders of magnitude before producing a TLD material sensitive enough for routine personnel monitoring.[62]

Although neither high temperature glow peak evaluation nor use of hydrogenous radiators in thermoluminescent materials is at the stage of development necessary for use as personnel dosimeters, these techniques certainly seem promising enough to warrant further investigation. Consideration should also be given to the possibility of combining the techniques, to provide a TLD material in which the effect of a high temperature, neutron sensitive glow peak is utilized, possibly augmented by a hydrogenated TLD.

Electrical Conductivity Damage Effects

Neutron-induced nuclear reactions and scattering of heavy charged particles can cause measurable changes in the electrical conductivity of some materials. Two examples of this phenomenon are the damage effects produced by neutrons in silicon diodes and cellulose acetate. Small size, ease of evaluation, and almost total insensitivity to other types of radiation are attractive features of both of these systems.

Silicon diodes exhibit a response equivalent to the neutron dose in rads (±20%) between ∿0.3 and 15 MeV.[63] Intra- and interbatch fluctuations, read-out errors, temperature dependence, and significant fading have all been cited as contributing factors to a less than desirable overall accuracy.[63] A diode has been developed, however, which seems potentially suitable for annually assessed neutron exposures (minimum detectable dose equivalent of AmBe neutrons ≃ 400 millirems).[64,65] Current research of silicon diode techniques is aimed at development of instrumentation capable of lowering the minimum detectable dose equivalent to a level acceptable for routine personnel monitoring.[65]

Measurements of resistivity changes in cellulose acetate after irradiation with AmBe neutrons have been reported by Fadel.[66] Statistically reproducible data with sufficient accuracy have been claimed for neutron fluences or doses over a relatively wide range (10^{-3} - 7 rads). A pronounced temperature dependence indicates the necessity of precise temperature control during the read-out procedure. Further investigation of both the cellulose acetate and silicone diode techniques of neutron

monitoring is necessary before either can be properly assessed for potential utility in personnel monitoring.

Lyoluminescence

Lyoluminescence is the phenomenon resulting in light emssion upon dissolution of previously irradiated substances. First applied to radiation dosimetry in 1973 by Atari et al.,[67] this property has been shown to be characteristic of a variety of materials including saccharides, amino acids, proteins, and alkali halides.[67-69] The major advantage of such a system for personnel dosimetry is the possibility of a nearly "tissue equivalent" dosimeter resulting from the elemental composition of some of the investigated compounds. This is an attractive feature since the neutron response of these compounds might be expected to approximate the response of soft tissues.[70]

Although a fairly large number and variety of compounds have been investigated for use as lyoluminescent dosimeters,[71] the sensitivity still seems to be inadequate for personnel dosimetry. The addition of sensitizing agents such as luminol (3-aminophthalhydrazide) or lucigenin (N, N-dimethyl-9,9-biacridinum dinitrate) to the solvent system has greatly lowered the minimum detectable dose. The use of these sensitizers, however, is accompanied by a chemiluminescent self-glow, caused by the presence of trace level impurities or oxygen in the lyoluminescent sample or the solvent system. Doses down to about 1 rad can be accurately measured before the sensitizing agent's self-glow

becomes a limiting factor. Promising solutions to this problem include further purification of samples and solvent systems,[68,72] reduction in the amount of solvent necessary for read-out,[71] and preparation of lyoluminescent materials doped with minute amounts of sensitizing agents.

Previous investigation into the response of lyoluminescent dosimeters to neutrons has been superficial. Preliminary experimental data are available, establishing the response of the saccharides mannose and trehalose dihydrate to several types of mixed neutron and gamma fields.[70] In addition, the ratio of the lyoluminescence of mannose per rad in International Committee on Radiation Units (ICRU) muscle tissue relative to ^{60}Co gamma rays has been calculated.[73] It should also be mentioned that an early investigation by Ahnstrom and Ehrenstein[74] indicated that irradiation of crystalline glucose with fast neutrons in the megarad dose range produced five to seven times more luminescent response upon dissolution in alkaline solutions than irradiation with the same dose of ^{60}Co gamma rays. Although these data are cursory, they indicate a certain degree of potential.

It is evident that lyoluminescence is incapable at this time of providing an adequate means of personnel neutron dosimetry. Extensive research and development may overcome the sensitivity problems, at which time more elaborate studies of neutron response would be warranted. The possibility of developing a nearly tissue equivalent dosimeter has always been an inviting one. An interesting sidelight is that LiF has been observed to exhibit lyoluminescence.[75] Therefore, the possibility of evaluating both thermoluminescent and lyoluminescent outputs of irradiated LiF samples has been suggested for mixed field dosimetry.[71]

Thermocurrent

The study of radiation induced thermally activated currents (RITAC) or depolarization (RITAD) for neutron dosimeters has been almost exclusively performed by workers at the University of Wisconsin. Advantages of such dosimeters include possible tissue-equivalent plastic dosimeter materials,[76] simple electronic read-out systems which are very stable (compared to photomultiplier tube systems used in thermoluminescent dosimetry),[77] and a relatively high neutron to gamma relative response ratio.[78]

Investigated in a variety of materials, the RITAC phenomenon has shown the most promise in uv-grade sapphire (Al_2O_3),[76] and a methylpentene polymer plastic. Although sapphire has shown advantages such as a linear dose response from a fraction of a millirad to hundreds of rads (gamma)[79] and a 15% response to 14-MeV neutrons relative to cesium gamma rays on an equivalent tissue-rad basis,[80] it is not sufficiently tissue equivalent for use as a personnel dosimeter. On a tissue-rad basis relative to ^{60}Co gamma rays, the nearly tissue-equivalent methylpentene polymer has demonstrated a 49% and 40% response to 14 MeV and fission neutrons, respectively.[78] Use of the material is limited because of 16-24% fading over 60 hours,[76] and spurious background currents to neutron doses greater than 10 rads.[78]

Local RITAD phenomenon in high purity CaF_2 crystals has been suggested as a new technique of selective radiation dosimetry, and although investigation has been extremely preliminary, these dosimeters have been shown to fade too rapidly for routine personnel dosimetry.[81] In the

light of present data, both the RITAC and RITAD effects must be judged unsuitable for personnel neutron monitoring at this time. Basically, economic considerations (projected cost of development) have forced research efforts in these areas to cease.[84]

Thermally Stimulated Exoelectron Emission

The measurement of thermally stimulated exoelectron emission (TSEE) from the surface of irradiated ionic crystals, first proposed for use as an integrating radiation dosimeter by Kramer,[83,84] has been investigated in a number of materials such as LiF, $Li_2B_4O_7$, BeO, $CaSO_4$, and MgF_2. Some of the early advantages claimed for this type of dosimetry system include relatively low cost, ease of preparation, extremely high sensitivity, complete annealing during read-out process, and an energy response which can be easily adjusted by low-Z additives to any desired dependence.[85,86]

Achieving the most promising results with ceramic beryllium oxide discs (Brush Beryllium Co., Thermolox 995), various researchers ardently pursued this technique until recently as a possible solution to the problem of fast neutron dosimetry. Early work by Becker and Crase[87] using a polyethylene covered BeO detector system indicated a fast neutron to gamma relative response ratio of $\sim$0.18 to $\sim$0.28, from 0.1 to 16 MeV, respectively (expressed as R_γ equivalent/tissue rad n_f). This stimulated additional investigations which lead to a lower ratio of 0.11 by Gammage et al.[88] for Oak Ridge National Laboratory Health Physics Research Reactor fission neutrons.

Since, in most neutron environments, the total absorbed dose is due to a combination of neutron and gamma radiation, resolution of the relative contribution of each type of radiation is necessary for accurate determination of dose equivalent. Mixed fields in which the gamma to neutron ratio is approximately equal to or greater than one, tax the statistical ability of a TSEE dosimeter to resolve the neutron component.

Other problems associated with TSEE dosimetry include an extreme directional dependence[89] and a relatively large error in the read-out procedure.[90] These factors, combined with the observation of a lower than originally stated neutron-to-gamma response ratio, have lead to both the cessation of research in this area and the conclusion that TSEE is unsuitable for generalized fast neutron personnel monitoring.[90]

Energy Dependence of Currently Available Dosimeters

The single most important consideration in the evaluation of a potential personnel neutron dosimeter is the comparison of the dosimeter's energy response with the established dose equivalent conversion factors as set forth in ICRP Publication 21.[4] Neutron spectra commonly encountered by workers can encompass approximately nine decades of energy (10^{-8}-10 MeV), with the recommended conversion factors varying from about 7 to 260. Any potential personnel neutron dosimeter must be evaluated with respect to these conversion factors if it is to be used for routine personnel monitoring. The ICRP dose equivalent conversion factors are plotted as a function of incident neutron energy in Fig. 1. Although given in

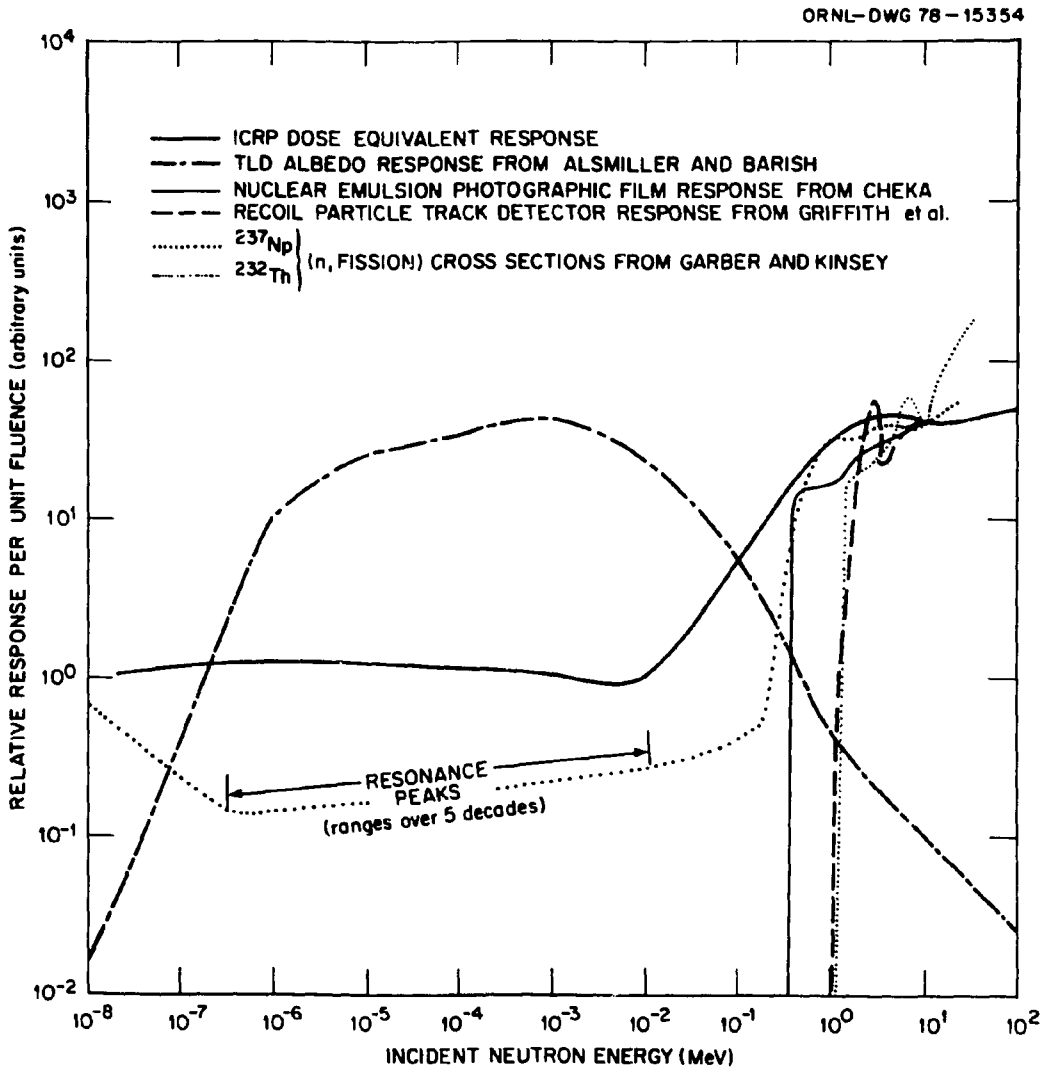

Fig. 1. Relative Energy Response of Neutron Dosimeters.

arbitrary units (for the sake of comparison), this curve becomes the "true" ICRP dose equivalent conversion curve (rem per neutron/cm^2) if the ordinate is assigned values of 10^{-11} to 10^{-6} from the origin to the fifth decade.

The relative response per unit fluence of the five currently used types of personnel neutron dosimeters (nuclear-emulsion film, TLD-albedo, ^{237}NP and ^{232}Th fission fragment detectors, and light-particle recoil track detectors) is plotted in Figure 1 for comparison with the standard ICRP response curve. All dosimeter responses are normalized to the ICRP response at 10 MeV, with the exception of the TLD-albedo detectors, which are normalized at 0.1 MeV. These curves illustrate the less than desirable energy response characteristics demonstrated by each of the currently used dosimeters. In every case, the energy response has been shown to be an inherent characteristic of the type of dosimeter considered (due to the physical mechanisms of the reactions responsible for neutron detection), and attempts to alter this response have met with little success.

Inspection of these response curves indicates the possibility of combining two or more detectors to provide a more usable energy response. Investigation of combined dosimetry systems has proven that these systems can not only reduce spectral dependence to a certain extent, but can also reduce the effects of directional dependence.[3,91] Although a substantial amount of effort has been expended on the evaluation of each type of potential dosimeter, surprisingly little work has been performed on the study of combination systems.

Intercomparison Studies

The current trend in personnel neutron dosimetry is illustrated by the results of the four personnel dosimetry intercomparison studies held at the Oak Ridge National Laboratory. These studies are conducted to allow independent experimenters the opportunity for objectively testing their dosimeters against several "standardized" radiation fields at the low radiation levels typically encountered in personnel monitoring. Personnel Dosimetry Intercomparison Studies (PDIS) were held at the ORNL Dosimetry Applications Research Facility (DOSAR) in 1974, 1976, 1977, and 1978.[92-95] Participants used a variety of dosimetry systems for evaluating mixed field (neutron and gamma) doses, including nuclear-emulsion films, fission foil and recoil particle track detectors, and TLD albedo systems.

The Health Physics Research Reactor (HPRR) was operated in a steady-state mode at a constant power level for time periods necessary to produce dose equivalents of several hundred millirems. Dosimeters were positioned on water-filled phantoms at a distance of 3 meters from the reactor core. When used, spectrum-modifying shields were placed between the detectors and the HPRR core, at a distance of 2 meters from the core. In the first, second, and third intercomparisons (1974, 1976, and 1977), the reactor was operated unshielded, shielded with 12 cm of Lucite, or shielded with 13 cm of steel.[96] During the fourth study (1978) the investigators discontinued use of the steel shield and added a 20-cm concrete shield and a combination 5-cm steel and 15-cm concrete shield.[95] These new shields provide more realistic test spectra as concrete and steel are commonly used for neutron shielding.

Discrete ordinates transport code calculations of the HPRR spectra have been performed for all exposure configurations used in the intercomparison studies. These calculations, combined with measurements of neutron and gamma doses by a variety of dosimetric devices, were used to obtain reference values of the dose and dose equivalent for each exposure. Some of these calculations and comparisons with experimental results are summarized in Table 1.

Results of the first through fourth PDIS indicate that dose equivalent estimates vary over a relatively large range.[95,96] Although the past several years have seen a general trend toward improved neutron dose assessment, this does not seem to be the case with evaluation of gamma dose equivalent. Another interesting tendency seen is that the mean values of the participants' measurements are consistently greater than the corresponding calculated values. Dickson and Gilley have suggested that this trend might be attributed to the conservative philosophy generally applied to personnel dosimetry.[96]

Since there is now and probably will always be some lack of standardization in dosimeter testing criteria and techniques, intercomparison studies such as these are extremely valuable; not only to the individual investigators and commercial services for evaluating their own dosimetry systems, but also to the policy and standards-setting groups which have the task of determining minimum performance criteria. Until the problems of mixed field personnel dosimetry are solved, it would seem advantageous for this type of intercomparison study to be conducted on a regular basis.

Table 1. Summary of ORNL HPRR Dosimeter Intercomparison Studies[a]

Shield	PDIS No. and Year	Reference dosimetry		Participants' results		% Standard deviation of measurements	
		Neutron dose equivalent (millirem)	Gamma dose equivalent (millirem)	Neutron dose equivalent (millirem)	Gamma dose equivalent (millirem)	Neutron	Gamma
None	1 (1974)	436	—	453 ± 213	25 ± 6	47	24
	2 (1976)	545	16 ± 2	550 ± 217	35 ± 29	39	83
	3 (1977)	545		675 ± 168	25 ± 14	25	56
	4 (1978)[b]	496	32 ± 3	484 ± 121	31 ± 13	25	43
Lucite	1 (1974)	338		675 ± 687[d]	75 ± 14	102	19
	2 (1976)	427	41 ± 4	532 ± 154	86 ± 46	29	53
	3 (1977)	427		558 ± 307	83 ± 34	55	41
	4 (1978)[b]	411	41 ± 4	388 ± 152	61 ± 12	39	20
Steel	1 (1974)	529		554 ± 346	18 ± 4	62	24
	2 (1976)	665	8 ± 1	753 ± 226	31 ± 30	30	97
	3 (1977)	665		721 ± 186	25 ± 14	26	56
Concrete	4 (1978)[b]	429	27 ± 3	465 ± 194	46 ± 14	42	30
Concrete/Steel	4 (1978)[b]	444	24 ± 3	564 ± 305	47 ± 17	54	37

[a] See references.

[b] Preliminary results of fourth PDIS (1978). Includes all participants (15) who reported their results as of June 1, 1978 (personal communication with H. W. Dickson, Oak Ridge National Laboratory).

[c] Gamma dose equivalent due to activation product gamma irradiation of dosimeters could only be estimated due to handling time, therefore some of this data is not included.

[d] Some means and standard deviations are unduly influenced by one or more extreme measurements. Although elimination of the outliers would produce much more consistent results, they have not been eliminated in this figure.

FUSION REACTOR SPECTRUM MODELING

Tokamak Fusion Test Reactor: General Information

The Princeton Tokamak Fusion Test Reactor (TFTR) is the first magnetic confinement device expected to achieve a "break-even" deuterium tritium (D-T) fusion reaction and is considered to be the link between large experimental tokamak devices and the first experimental fusion power reactor.[97] Based on the concept of build-up and reaction of an energetic deuteron ion population in a moderately hot tritium plasma (by neutral beam injection), the TFTR will permit energy break-even conditions (Q = 1)[a] to be reached.[98]

Currently under construction at the Forrestal Research Campus of Princeton University, the TFTR is a tokamak-type toroidal device with major and minor radii of 280 and 70 cm, respectively. When completed, it will be limited to 1000 equivalent full tritium pulses per year, each pulse using about 100 curies of tritium. In a typical D-T experiment, the magnetic fields will be on for about 15 seconds, with the important part of the pulse occurring in less than 500 milliseconds. If break-even conditions (Q = 1) are achieved, 7×10^{18} neutrons will be produced, corresponding to 10 MW of D-T power for 0.5 seconds, demonstrating a fusion power density of $\sim$1 W/cm^3.[97,99] The most common mode of D-T

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[a] Q, the "plasma" figure of merit, is defined as the ratio of fusion power produced in plasma and blanket to power which must be added to plasma externally to maintain temperature.[98]

operation is expected to be a sequence of 40 pulses spaced 5 minutes apart per day, one day every 1-2 weeks. This operation will require 35 MW average power from the utility system and will generate approximately 120×10^6 Btu/hour of excess heat, which will be dissipated into the atmosphere by induced-draft evaporative cooling towers.[97]

Modeling, Geometry and Spectrum Calculations

Calculations of dose equivalent rates and neutron spectra in areas accessible by personnel are being performed for the Princeton Plasma Physics Laboratory by the Engineering Physics Division at the Oak Ridge National Laboratory. The purposes of these calculations are to assess the efficacy of shield design and to demonstrate areas in which additional shielding is necessary to provide a safe working environment. Although these calculations are not complete, an example of them will be used here to illustrate an approximation of the spectrum generated by the TFTR during D-T operation to which personnel may be exposed.

Neutronic and photonic calculations have been carried out using the two-dimensional discrete-ordinates code DOT, a P_3 scattering expansion, and an S_8 angular quadrature.[100] The radiation transport was accomplished using cross-section data obtained by collapsing the DLC-37 cross-section library to a 35-neutron, 21-gamma-ray energy group subset.[101,102]

Figure 2 illustrates the geometry used to calculate the neutron leakage spectra of the TFTR to the roof (A) and outside the wall (B) of the facility, areas in which personnel may be present during operation.

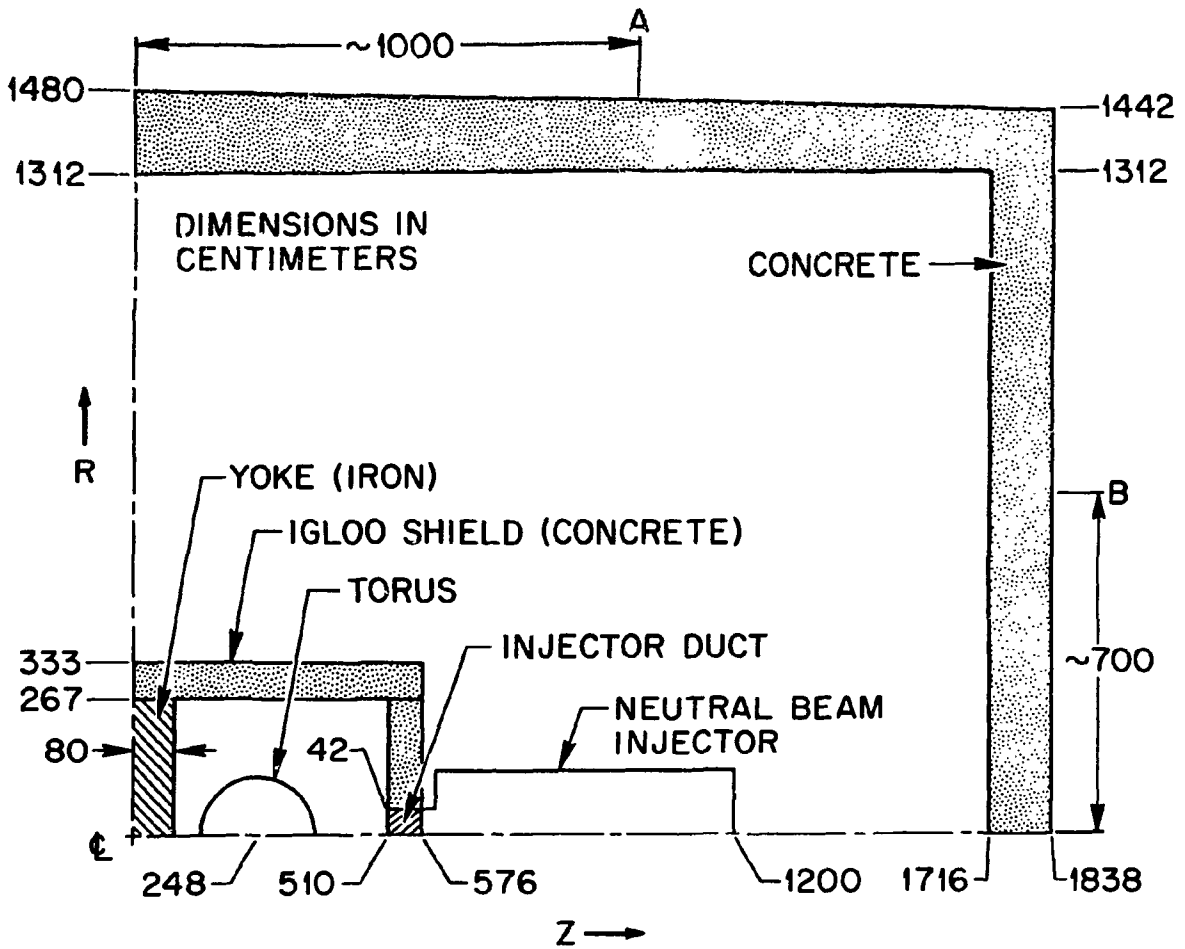


Fig. 2. Schematic Diagram of TFTR Test Cell Calculations Geometry.

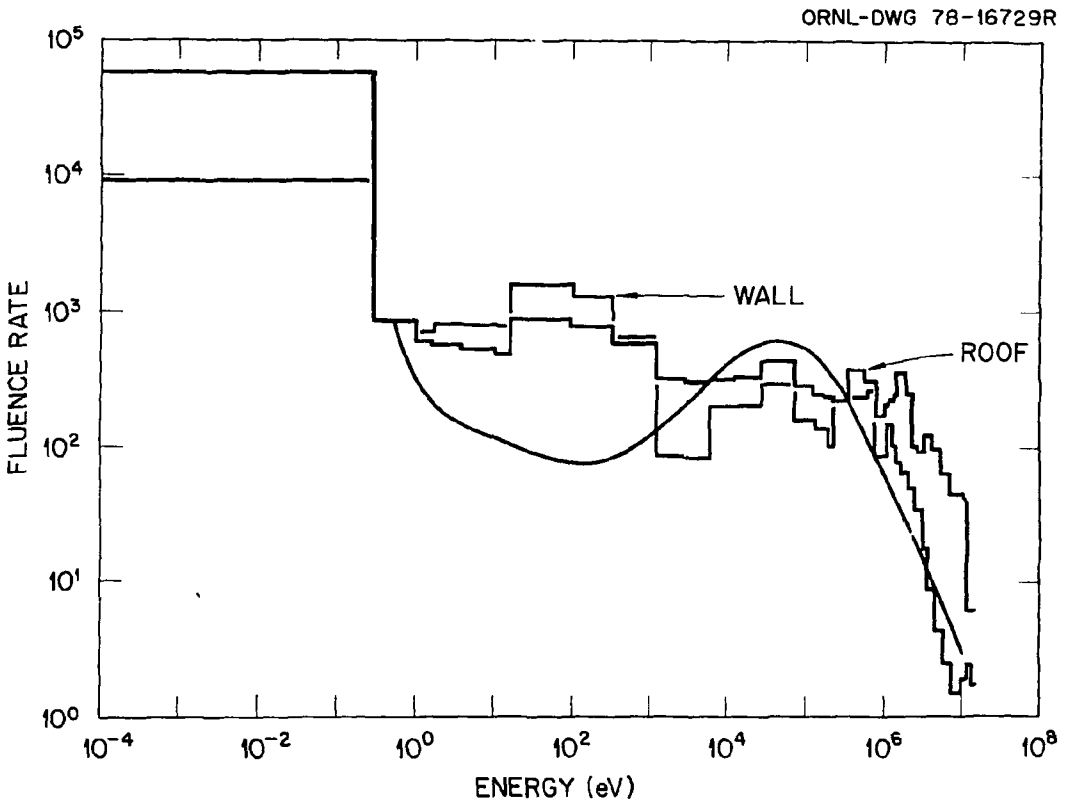
Although other areas of the facility will probably require additional shielding to that outlined in the diagram, it is anticipated that the roof will be shielded in the manner shown, and only minor changes in wall shielding will probably be made.[103] It is also anticipated that the leakage spectra at these points will not vary appreciably from that of other occupied areas (although the dose rates may differ), since the neutron spectrum generated in working areas during D-T operation of the facility will most probably be an equilibrium spectra, and additional shielding should cause attenuation without appreciable modification.[103] It should be recognized at this point that these calculations have been performed with only one neutral beam injector in the geometry. The addition of three more injectors (the facility is expected to operate with four) will cause considerably more neutron leakage (via streaming through injector ducts), the extent of which cannot be evaluated until further calculations are undertaken. This fact, combined with frequent design changes and reevaluation, suggest that these calculated leakage spectra will certainly not be equivalent to the true leakage. It is hoped, however, that these spectra will yield useful information about personnel neutron exposures in fusion reactors, as knowledge of the quality of such a spectrum has to this date been relatively limited.

The calculated neutron spectrum on the roof and wall of the TFTR facility about 1000 cm and 700 cm, respectively, from the center line (C_L) is given numerically and plotted graphically in Table 2 and Fig. 3, respectively.[104] The most prominent feature of these

Table 2. Results of two-dimensional neutron transport analysis to accessible areas of the TFTR facility

Group number	Lower boundary of energy group (eV)	ΔE per group (eV)	Fluence rates of TFTR leakage neutrons $\frac{\text{neutron}}{\text{cm}^2 - \text{pulse}}$	
			Roof	Wall
Upper boundary	1.45E7[a]			
1	1.3499E7	1.0010E6	5.9903E0	1.8350E0
2	1.2214E7	1.2850E6	4.1089E1	2.6464E0
3	1.0000E7	2.2140E6	4.1941E1	1.9806E0
4	8.1873E6	1.8127E6	4.4756E1	1.5654E0
5	6.7032E6	1.4841E6	6.1546E1	2.6291E0
6	5.4881E6	1.2151E6	9.4871E1	4.6066E0
7	4.4933E6	9.9480E5	1.2153E2	9.4023E0
8	3.6788E6	8.1450E5	9.0891E1	1.7983E1
9	3.0119E6	6.6690E5	1.0176E2	3.5785E1
10	2.4660E6	5.4590E5	2.4162E2	5.0837E1
11	2.0190E6	4.4700E5	3.5241E2	6.7525E1
12	1.6530E6	3.6600E5	2.3907E2	8.1847E1
13	1.3534E6	2.9960E5	2.2122E2	1.0675E2
14	1.1080E6	2.4540E5	2.0205E2	1.5168E2
15	9.0718E5	2.0082E5	1.0720E2	8.8112E1
16	7.4274E5	1.6444E5	3.0174E2	2.7159E2
17	4.9787E5	2.4487E5	3.7153E2	2.4400E2
18	3.3373E5	1.6414E5	2.1937E2	2.3359E2
19	2.2371E5	1.1002E5	2.3896E2	1.0687E2
20	1.4996E5	7.3750E4	2.4148E2	1.4433E2
21	8.6517E4	6.3443E4	2.8186E2	1.5742E2
22	3.1828E4	5.4689E4	4.3367E2	3.0541E2
23	1.5034E4	1.6794E4	3.1304E2	2.1179E2
24	7.1018E3	7.9322E3	3.0333E2	2.1084E2
25	3.3546E3	3.7472E3	2.9022E2	8.2583E1
26	1.5846E3	1.7700E3	3.0731E2	8.6979E1
27	4.5400E2	1.1306E3	5.5910E2	6.5777E2
28	1.0130E2	3.5270E2	7.5707E2	1.3711E3
29	2.2603E1	7.8697E1	8.6861E2	1.5786E3
30	1.0677E1	1.1926E1	4.7255E2	7.8580E2
31	5.0435E0	5.6335E0	4.0757E2	7.9882E2
32	2.3824E0	2.6611E0	5.4730E2	8.1252E2
33	1.1254E0	1.2570E0	5.7924E2	7.1046E2
34	4.1400E-1	7.1140E-1	8.4384E2	8.9400E2
35	1.0000E-4	4.1390E-1	5.7262E4	9.2754E3
			Σ = 6.79E4	Σ = 1.9565E4

[a]Read as 1.45×10^7.



—CALCULATED D-T FUSION SPECTRA
(FLUENCE RATE IN UNITS OF neutrons-cm^{-2}- pulse^{-1})
INCLUDING ONE NEUTRAL BEAM INJECTOR

— MEASURED FISSION SPECTRA IN MAIN CONTAINMENT BUILDING OF A PWR (FLUENCE RATE IN ARBITRARY UNITS: neutrons-cm^{-2}- time^{-1})

Fig. 3. TFTR D-T Fusion Spectra in Accessible Areas.

spectra is the tendency toward lower energies, with a large thermal component. This is somewhat surprising as one may have thought a D-T fusion spectrum would have contained a much larger component of high energy neutrons (i.e., up to and including 14.1 MeV). Also shown in Figure 3, for the sake of comparison, is a spectrum generated in the main containment building of a pressurized water reactor (PWR). This area of the PWR contains a significant neutron environment.[105,106]

Neutron Dosimetry Around a Fusion Reactor

Careful analysis of the dose equivalent arising from exposure to the calculated TFTR leakage spectra reveals several interesting details. At first glance, it seems as though the dosimetrist would require a personnel neutron dosimetry system capable of measuring dose equivalent over about nine decades of energy (10^{-2} to 10^{7} eV) since the fluences are relatively large throughout this entire region. However, using ICRP fluence-to-dose equivalent conversion factors to analyze the roof leakage spectrum, it is discovered that 54% of the total dose equivalent arises from neutrons with energies greater than about 0.5 MeV, and 35% of the total dose equivalent arises from neutrons having thermal energies. Therefore, a dosimetry system capable of accurately measuring neutrons of energies >0.5 MeV and thermal, would be satisfactory for use in this type of neutron environment, as these neutron energies account for $\sim$89% of the total dose equivalent. Since techniques like TLD (e.g., using

^{6}LiF) and others are adequate for evaluation of thermal neutron exposures, all that is necessary is development of a dosimeter capable of accurately accessing exposures to the higher energy neutrons. Nuclear emulsion films and electrochemically etched recoil particle track detectors may be able to measure the higher energy component, after some improvements in problem areas such as latent image fading, minimum detectable dose equivalent, and energy response characteristics. Thermoluminescent dosimeter-albedo techniques are inadequate for either the high energy or thermal component, unless the spectrum is very constant (with differences only in dose rates) throughout all accessible areas; in which case the appropriate calibration factor could be applied. The TLD-albedo system could then be used (as could any other sensitive dosimeter with an appropriate calibration factor) with the added advantage of a relatively low minimum detectable dose equivalent. There is probably no future in this or any other type of neutron dosimetry for fission fragment track detectors (e.g., ^{232}Th and ^{237}Np) due to their inherent radiotoxicity.

Fusion Spectra Risk Analysis

The standard approach to expressing the effect of neutron irradiation is conversion of neutron spectra to dose equivalent by established and accepted conversion factors, such as these stated in ICRP Publication 21,

Appendix 6. This is generally considered satisfactory for most instances of low-level exposure to neutrons. It is more meaningful, however, to consider the mean insult to the active bone marrow (the critical organ for external, whole-body neutron irradiation) when assessing the hazards of neutrons. For this reason, analysis of the moderated fusion spectrum by Critical Human Organ Radiation Dosimetry (CHORD) operators has been undertaken to establish the bone marrow insult and relative risk to a worker occupying accessible areas of the TFTR facility during operation.

The CHORD system, developed by Jones,[107] utilizes probability density functions as insult-attenuation operators in a variety of exposure geometries. Its application to active (red) bone marrow dosimetry can be summarized by the simple equation:

$$\overline{D} = \frac{\Sigma\ \phi\ (E)\Delta E \times D_n(E)}{\Sigma\ \phi(E)\Delta E} ,$$

where

$\phi(E)\Delta E$ = neutron fluence in an energy interval,

$D_n(E)$ = active marrow dose per fluence neutron in the same energy interval.

For each energy group, the fluence (e.g., neutrons/cm^2) may be multiplied by the appropriate CHORD operator (rads/fluence neutron) to express the dose to the active bone marrow from incident neutrons in that energy group. Summation of these fluence weighted doses and subsequent division

by the total neutron fluence over the entire energy spectrum yields a fluence weighted mean marrow dose ($\overline{D}$). This mean dose may then be used to obtain an associated mean particle energy by finding the energy corresponding to the CHORD operator which has the same value as the mean dose. At this point, the incident neutron spectrum may be treated as monoenergetic (with respect to active marrow insult), and the risk of long term somatic effects may thereby be calculated.

Evaluation of Katz's cell-survival model for mixed radiation fields may be performed using the time-integrated recoil-ion dose, the time-integrated gamma-ray dose produced by *in vivo* neutron interactions, and their corresponding calculated mean neutron energies.[107,108] This yields an estimate of the degree of cell-killing if the dose was delivered in an acute manner. The literature suggests that the risk from high linear energy transfer (LET) radiations is not significantly decreased for extended exposure times (i.e., lower dose rates), whereas carcinogenic risk from low LET radiations has been inferred to be reduced by about a factor of 3 or more for chronic versus acute exposures.[109] Work submitted for publication indicates a direct relationship between fatal malignancies in man and the degree of cytotoxicity.[110] This has lead to a proposed model which describes the promotion stage of cancer by the degree of cell-killing and the resultant cell proliferation, since it appears that the degree of cytotoxicity can serve as an index for increased cell proliferation, and increased proliferation can be directly associated with increased cancer risk.

The leakage spectrum at accessible areas of the TFTR facility (points on the roof and outside the wall) were analyzed by the CHORD system of dosimetry and by ICRP fluence to dose conversion factors. Table 3 illustrates the CHORD operators and results of multiplication by fluence rates in their respective energy groups. Summation of these fluence weighted doses yields a total recoil ion and autogamma dose (rads/pulse) to the active bone marrow for both the roof and wall spectra. Results of these manipulations are presented in Table 4, along with the calculated ICRP dose equivalent at the two points of interest.

If it is assumed that a worker may occupy one of the described accessible areas for the entire expected operating life of the TFTR facility (1000 pulses per year for 4 years), and that the insult due to accumulated exposure to calculated neutron spectra approximates that of an acute exposure, then the risk of malignancy induction (leukemia or other) may be quantified by the cell survival methodology. Over the four year period, the upper limit of dose accumulation estimated by the CHORD method would be approximately 0.02 and 0.07 rads to the bone marrow for the wall and roof locations respectively. These values are 14 and 50 times greater than the neutron component of the cosmic-ray background radiation. By comparing these occupational doses to cell survival data,[110] it is clear that more than 99% of the red marrow stem cells exposed will survive. (Probably much more than 99% cell survival would be experienced, although presently available data does not allow for accurate extrapolation.) Thus, the four year during operational period of the TFTR, much less than 1% of the steady state inventory

Table 3. Dose to active marrow from recoil ions and autogammas produced by leakage neutrons from TFTR facility as predicted by CHORD distributions

Group	Recoil ion dose to marrow (-10^{-11} rads/fluence neutron)	$\phi(E)\Delta E \times D_n(E)$ (rads/pulse)		Autogamma dose to marrow (-10^{-11} rads/fluence neutron)	$\phi(E)\Delta E \times D_\gamma(E)$ (rads/pulse)	
		Roof	Wall		Roof	Wall
1[a]	540	3.2E-8[b]	9.9E-9	48.2	2.89E-9	8.84E-10
2	480	1.9E-7	1.3E-8	46.1	1.89E-8	1.22E-9
3	420	1.8E-7	8.3E-9	44.0	1.84E-8	8.71E-10
4	390	1.8E-7	6.0E-9	41.9	1.88E-8	6.56E-10
5	360	2.2E-7	9.5E-9	39.8	2.45E-8	1.05E-9
6	330	3.1E-7	1.5E-8	37.8	3.59E-8	1.74E-9
7	300	3.7E-7	2.8E-8	36.0	4.37E-8	3.38E-9
8	250	2.3E-7	4.5E-8	34.0	3.09E-8	6.11E-9
9	210	2.1E-7	7.5E-8	31.8	3.24E-8	1.14E-8
10[a]	190	4.6E-7	9.7E-8	29.8	7.20E-8	1.51E-8
11	165	5.8E-7	1.1E-7	28.6	1.01E-7	1.93E-8
12	140	3.4E-7	1.1E-7	28.4	6.79E-8	2.32E-8
13	115	2.5E-7	1.2E-7	28.2	6.24E-8	3.01E-8
14	95	1.9E-7	1.4E-7	28.0	5.66E-8	4.25E-8
15[a]	75	8.0E-8	6.6E-8	27.8	2.98E-8	2.45E-8
16	55	1.7E-7	1.5E-7	27.8	8.39E-8	7.55E-8
17	35	1.3E-7	8.5E-8	27.8	1.03E-7	6.78E-8
18	28	6.1E-8	6.5E-8	27.8	6.10E-8	6.49E-8
19	21	5.0E-8	2.2E-8	27.7	6.62E-8	2.96E-8
20	15	3.6E-8	2.2E-8	27.7	6.69E-8	4.00E-8
21[a]	9.2	2.6E-8	2.8E-8	27.7	7.81E-8	4.36E-8
22	7.0	3.0E-8	2.1E-8	27.0	1.17E-7	8.25E-8
23	5.0	1.6E-8	1.1E-8	26.2	8.20E-8	5.55E-8
24[a]	3.0	9.1E-9	6.3E-9	26.4	8.01E-8	5.57E-8
25	2.7	7.8E-9	2.2E-9	26.6	7.72E-8	2.20E-8
26	2.5	7.7E-9	2.2E-9	26.8	8.24E-8	2.33E-8
27[a]	2.3	1.3E-8	1.5E-8	26.9	1.50E-7	1.77E-7
28	2.2	1.7E-8	3.0E-8	25.9	1.88E-7	3.55E-7
29	2.1	1.8E-8	3.3E-8	24.8	2.15E-7	3.91E-7
30	2.0	9.5E-9	1.6E-8	23.7	1.12E-7	1.86E-7
31	1.9	9.6E-9	1.5E-8	22.6	1.15E-7	1.81E-7
32	1.8	9.9E-9	1.5E-8	21.5	1.18E-7	1.75E-7
33	1.7	9.9E-9	1.2E-8	20.5	1.19E-7	1.46E-7
34	1.6	1.4E-8	1.4E-8	19.4	1.64E-7	1.73E-7
35	1.6	9.2E-7	1.5E-7	18.3	1.05E-5	1.70E-6
		Σ_n = 5.4E-6	Σ_n = 1.6E-6		Σ_γ = 1.31E-5	Σ_σ = 4.23E-6

[a]Footnoted values given by CHORD distributions, all other interpolated per T. D. Jones.

[b]Read as 3.2×10^{-8}.

Table 4. Summary of TFTR Facility neutron dose calculations

	Chord analysis			
	Recoil ion dose to marrow (rad/pulse)	Autogamma dose to marrow (rad/pulse)	Bone marrow dose — Weighted mean neutron energy (keV)	Calculated ICRP dose equivalent (rem/pulse)
Roof	5.4×10^{-6}	1.31×10^{-5}	Recoil ion: ≈60 Autogamma: ≈0.001	1.75×10^{-4}
Wall	1.6×10^{-6}	4.23×10^{-6}	Recoil ion: ≈90 Autogamma: ≈0.004	6.50×10^{-5}

of stem cells would require replacement due to the potential occupational exposure. This is equivalent to an upper limit demand for roughly one extra cell per 100 over a 4 year time period — hardly enough to be called a proliferation stimulus when compared to the normal cell inventory replacement rate of the order of several per year. Therefore, it can be concluded using the tentative model,[110] that no measurable risk of cancer induction will result from an individual receiving the maximum possible exposure to TFTR neutrons. Other risk estimate procedures are available for application[111] but would not likely be any more appropriate in expressing actual risk, even though they might appear to be more quantitative. It should be emphasized that these calculations are measures of dose and risk from neutrons only (i.e., radiation doses from other sources such as gamma radiation) tritium leakage, and structural activation products are not considered.

SUMMARY AND CONCLUSIONS

Personnel neutron dosimetry continues to be one of the more elusive facets of radiation protection for a variety of reasons. Fluctuating biological response over an extremely large energy range combined with difficulty in measuring the low doses associated with occupational exposures seem to be major factors limiting the applicability of most neutron dosimetry systems. If suggestions to increase the quality factors for neutrons are followed,[112,113] virtually all neutron dosimeters currently available will become incapable of meeting the legal requirements as set forth in 10 CFR 20, due to their inability to detect extremely low doses.

The current trend in personnel neutron dosimetry seems to include the increasing utilization of "mixed" dosimeters (i.e., two or more types of dosimeters used to evaluate spectral quality). Evaluation of these systems can be quite a difficult task, with the type of dosimetry system and method of evaluation tailored many times to the individual's specific exposure conditions. This heterogeneity of neutron dosimetry points to a need for ongoing standardization and intercomparison studies as discussed earlier.

It is evident that there is no "perfect" personnel neutron dosimeter available at present. Barring the advent of any darkhorse solutions, it is also clear that there is little chance in the near future of any one neutron dosimeter becoming capable of accurately monitoring all occupational neutron exposures. Only continuing research and development

can advance the capabilities of neutron dosimetry, and this may be constrained due to economic considerations (i.e., the potential profit to be derived from such a limited market may restrict major companies from large investments in research and development).

The reactor system modeled in this paper illustrates the wide variety of neutron energies to which personnel may be exposed around an operating fusion reactor. It is necessary, of course, to remember that the type spectra leaking from a large fusion power reactor will depend extensively on such factors as the types and amounts of shielding materials used, blanket characteristics, type and size of neutral beam injectors and their respective ducts. Although the spectrum calculated for the TFTR facility certainly cannot be extrapolated to a power reactor, it is fairly evident that personnel working around such a reactor will be exposed to a wide range of neutron energies, with large spectral contributions (relative to dose equivalent) from thermal and very high energies. Even though doses and associated risks due to neutron irradiation at the modeled points of the TFTR are essentially negligible, these calculations should yield useful information about the spectral quality of a highly moderated D-T fusion spectrum. It is thought that although continuing investigation of personnel neutron dosimetry is warranted for both the projected fusion power reactors and other sources of neutron radiation, personnel neutron dosimetry will probably not prove to be an insurmountable obstacle in the development of fusion power.

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