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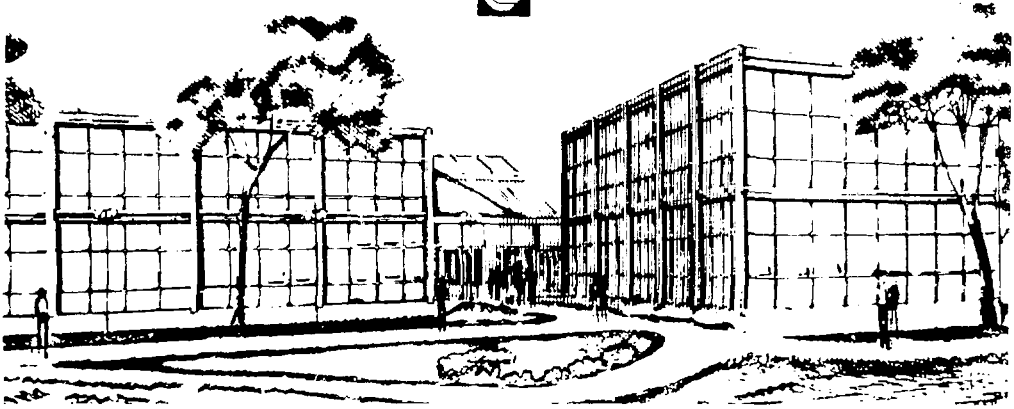

# A SURVEY OF NEUTRONS INSIDE THE CONTAINMENT OF A PRESSURIZED WATER REACTOR*

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## ABSTRACT

A neutron survey was made inside the containment of the Farley Nuclear Plant, Alabama Power and Light Company, Dothan, Alabama, in November 1977. The survey was made to determine the spectra of leakage neutrons and to evaluate the accuracy of albedo neutron dosimeters and a 9-in.-diameter sphere rem meter. The survey also covered variations in the neutron spectra, the ratio of gamma-to-neutron dose rates, and the thermal neutron component of the neutron dose.

## INTRODUCTION

The Farley Nuclear Plant at Dothan, Alabama, reached full operating power in November 1977. The plant management desired an accurate determination of the neutron dose rates in the containment of the reactor. Because moderator rem meters overrespond to reactor leakage neutrons, the management requested that the neutron spectra at various locations in the containment be determined. From these spectra, they could determine how much the rem meters overrespond.

Of equal importance was to determine appropriate personnel neutron dosimetry. Albedo neutron dosimeters were being considered, and the management desired information on their sensitivity and accuracy at this reactor. Because albedo dosimeters are sensitive to variations in neutron spectra, an extensive survey was required to assure that the variations in the neutron spectrum would not seriously affect the accuracy of the dosimeters.

The survey was made by personnel from the Lawrence Livermore Laboratory (LLL) and the Oak Ridge National Laboratory (ORNL). The LLL personnel made measurements with thermoluminescent dosimeters (TLD's), albedo neutron dosimeters, and neutron and gamma instruments. One of the neutron instruments used was the multisphere moderator system,[1]

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and these measurements were used to unfold the spectra. The ORNL personnel used activation foils and detectors utilizing fission-fragment damage to polycarbonates. This paper describes the LLL study. The ORNL results are given in another paper presented at this session.

## PROCEDURE

Because we were not familiar with the reactor and the potential for variations in the neutron-leakage spectrum, our approach was to first make a survey with the PNR-4 neutron survey instrument.* Counting rates were obtained with both the 9- and 3-in.-diameter spheres. Variations in the leakage spectra can be detected by observing differences in the ratio of the count rates from these two spheres; the 9-in. sphere is primarily sensitive to fast neutrons while the 3-in. sphere is sensitive to low-energy neutrons. The survey included points outside the containment, inside the air lock, and at numerous locations inside the containment (Fig. 1). These included shielded locations and locations where the pressure vessel could be seen. While this survey was being made, we used the multisphere system to measure the neutron spectrum at location 1, outside the containment.

The initial survey indicated that the neutron spectrum did not vary significantly at the 27 survey points. Three points inside the containment (6, 9, and 16 in Fig. 1) were selected for subsequent multisphere measurements of the spectrum. The points selected were considered to have a typical exposure condition, where the dose rates were not so high as to cause count-rate losses. Also, one point was behind the shield of steam generator 1-B, where a variation in the spectrum might be expected.

The count-rate ratio of the 9- to 3-in.-diameter spheres was also used to determine the calibration factor for albedo neutron dosimeters. The 9- and 3-in. spheres are not sensitive to the direction of the neutrons; whereas dosimeters worn on the front of the body normally respond primarily to neutrons impinging on the front, with less response to neutrons impinging on the back of the body and elsewhere. Because the neutrons appeared to be coming from many directions at this reactor, we also determined calibration factors at eight locations using Hankins-type albedo neutron dosimeters[2] planted on the LLL chest phantom.[3] Each dosimeter contained four $^{6}Li$ and four $^{7}Li$ TLD's. The locations were selected to provide typical exposure

*Reference to a company or product name does not imply approval or recommendation of the product by the University of California or the U.S. Department of Energy to the exclusion of others that may be suitable.

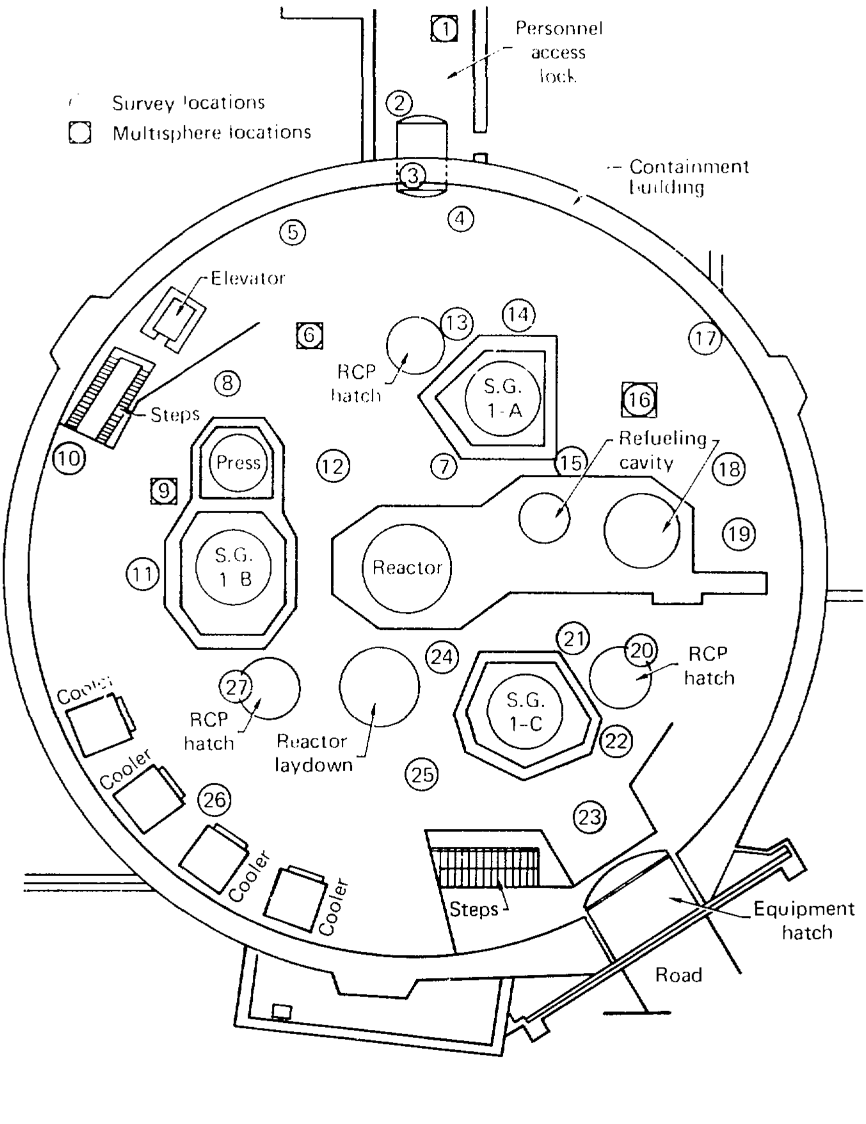


Fig. 1. Floor Plan of the Containment at the 155-ft Level

conditions scattered throughout the enclosure, to use the same three locations studied by the multisphere technique, and to use three of the locations being studied by the ORNL personnel.

We used the bare $BF_3$ tube from the PNR-4 to determine the thermal-neutron contribution to the total neutron dose rate. We obtained gamma dose rates with instruments available at the reactor and by using TLD's.

The multisphere neutron-spectrum measurements were made using a $^6LiI$ scintillation crystal, 0.5 in. diameter by 0.5 in. long, connected to the LLL 1024-channel portable pulse-height analyzer. The detection mechanism in this crystal is the $^6Li(n, \alpha)$ reaction, which causes a distinct peak in the pulse-height spectrum. For the detector response, we used the full-width peak integral with an exponential background continuum subtracted.

We obtained the neutron-energy spectrum by taking counts with the bare scintillation crystal, with the crystal in a 0.020-in. cadmium shell, and with the crystal used sequentially in 3-, 5,- 8-, 10-, and 12-in.-diameter spheres of polyethylene. The fast-neutron response of this system increases with increasing sphere size because the polyethylene removes low-energy neutrons and moderates fast neutrons to make them more detectable. Cadmium shells were placed around the 3- and 5-in. spheres to suppress the thermal-neutron response.

Using the responses from the seven detector configurations (bare, cadmium covered, and 3-, 5-, 8-, 10-, and 12-in. moderated), we unfolded the spectrum with the *LOUHI* computer code.[4] We used response functions calculated by Robert Sanna[5] at the U.S. Department of Energy Environmental Measurements Laboratory as input for the unfolding process. Essentially, we are solving for $\phi_j$ in the equation

$$A_i = \sum_{j=1}^{26} R_{ij}\phi_j \ , \qquad (1)$$

where $A_i$ is the count rate with the ith detector configuration,

$R_{ij}$ is one of the responses calculated by Sanna, and

$\phi_j$ is the neutron flux in the jth energy band.

## RESULTS AND DISCUSSION

The ratio of the 9- and 3-in. sphere responses, shown in Table 1, varies only from 0.12 to 0.15, indicating no significant variation in the neutron spectrum. The neutron dose rates as determined by the

Table 1. Survey Results Obtained with PNR-4 Neutron Instrument and the Plant Gamma Instrument

| Location | Other dosimetry | Ratio 9/3-in. spheres | Neutron dose rate, 9-in. sphere (mrem/hr) | % thermal neutrons in dose | Gamma dose rate (mR/hr) | Ratio n/γ |
|---|---|---|---|---|---|---|
| 1 | Multisphere | 0.13 | 0.34 | 5.2 | | |
| 2 | | 0.12 | 0.92 | 4.0 | | |
| 3 | | 0.14 | 21 | 2.4 | | |
| 4 | | 0.14 | 110 | 2.5 | 15 | 7.3 |
| 5 | | 0.13 | 310 | 2.9 | 17 | 18.2 |
| 6 | Multisphere | 0.12 | 170 | 3.6 | 25 | 6.8 |
| 7 | ORNL[a] | 0.14 | 1160 | 3.3 | 180 | 6.5 |
| 8 | | 0.12 | 190 | 3.7 | 20 | 9.5 |
| 9 | Multisphere | 0.13 | 37 | 4.6 | 10 | 3.7 |
| 10 | | 0.13 | 48 | 4.2 | 9 | 5.3 |
| 11 | | 0.13 | 40 | 4.1 | 40 | 1.0 |
| 12 | ORNL[a] | 0.13 | 580 | 3.4 | 80 | 7.3 |
| 13 | | 0.13 | 140 | 4.4 | 23 | 6.1 |
| 14 | | 0.13 | 87 | 3.6 | 15 | 5.8 |
| 15 | ORNL[a] | 0.14 | 960 | 3.2 | 140 | 6.9 |
| 16 | Multisphere | 0.14 | 420 | 3.2 | 60 | 7.0 |
| 17 | | 0.14 | 350 | 3.3 | 50 | 7.0 |
| 18 | | 0.15 | 520 | 2.8 | 70 | 7.4 |
| 19 | | 0.17 | 910 | 2.5 | 100 | 9.1 |
| 20 | | 0.14 | 620 | 3.4 | 90 | 6.9 |
| 21 | ORNL[a] | 0.15 | 630 | 3.7 | 80 | 7.9 |
| 22 | | 0.14 | 250 | 3.8 | 46 | 5.4 |
| 23 | | 0.15 | 170 | 3.1 | 28 | 6.1 |
| 24 | | 0.14 | 1020 | 3.3 | 150 | 6.8 |
| 25 | | | 260 | 3.4 | 32 | 8.1 |
| 26 | | 0.15 | 180 | 3.1 | 26 | 6.9 |
| 27 | | 0.14 | 190 | 3.2 | 29 | 6.6 |

[a]ORNL made measurements at these locations using fission foils and activations of gold and sulfur.

9-in.-sphere rem meter are also given in Table 1. The neutron dose rates were lowest behind the shielding of steam generator 1-B and highest at locations near the reactor. The neutron dose rates outside the containment were less than 1 mrem/hr.

The thermal-neutron contribution to the total neutron dose is given as percent thermal in Table 1. This value averaged 3.4% over all points inside the containment, indicating that the thermal neutrons contribute only a small part of the neutron dose in this area. Outside the containment, the percent thermal increases to 4.0 and 5.2%.

This increase is the effect expected outside the containment. A small increase is also observed at positions 9, 10, and 11, which are shielded from the reactor.

The gamma dose rates in Table 1 were obtained using the plant instruments and in all but one location (11) were about 1/7 the neutron dose rate. We recognized the abnormally high gamma dose rate at location 11, but were not able to pinpoint the source. The neutron-to-gamma exposure ratio (without location 11) averaged 7.3. The constant ratio made us suspicious that the gamma instrument was responding to neutrons, but the TLD data, discussed later, indicates that the instrument responds primarily to gamma rays.

The ratio of the 9- to 3-in.-sphere count rates can be used with the curve in Fig. 2 to determine the calibration factor for albedo neutron dosimeters.[6,7] (Note: The albedo TLD readings are divided by this calibration factor to determine the dose.) The calibration

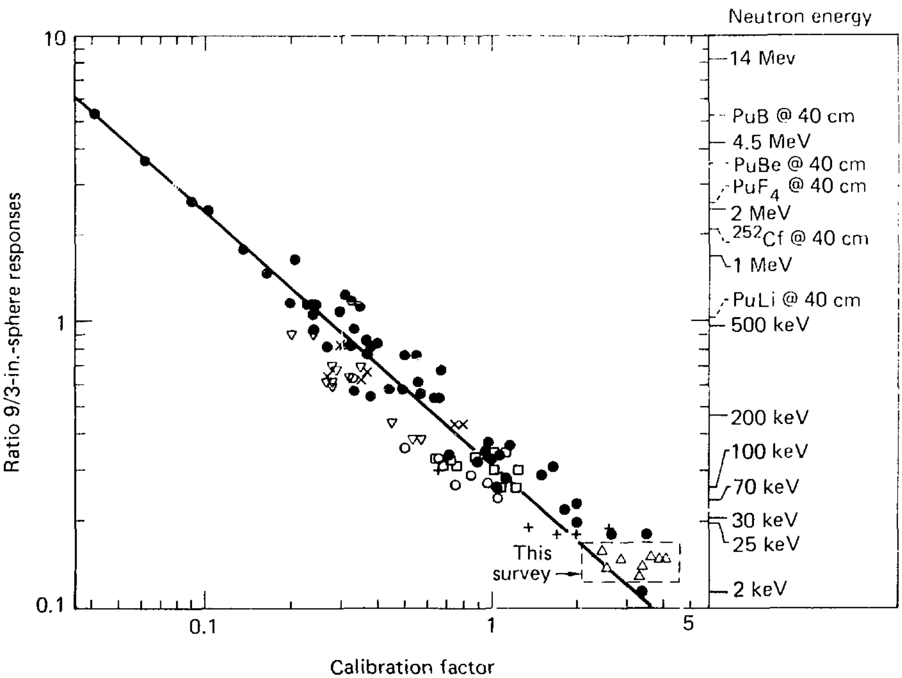


Fig. 2. Curve Used to Obtain the Calibration Factor for Albedo Neutron Dosimeters

factors determined using Fig. 2 have been plotted as a function of the neutron dose rate in Fig. 3. Frequently, these surveys show that the calibration factors change as the neutron dose rate varies, making a plot similar to Fig. 3 useful in determining the appropriate calibration factor. For this reactor, we found no significant difference in the calibration factor at the various dose rates. The average calibration factor of 2.65 is indicated, with the observed ±13% spread in calibration factors from the 9- to 3-in.-sphere ratio.

The results obtained using albedo neutron dosimeters are shown in Table 2. These calibration factors were obtained by taping the dosimeter on phantoms and placing the phantoms at the locations indicated for fixed periods of time. The dose delivered to the dosimeters was calculated using readings from the 9-in.-sphere. The neutron response from the TLD's was divided by the dose to obtain the calibration factors.

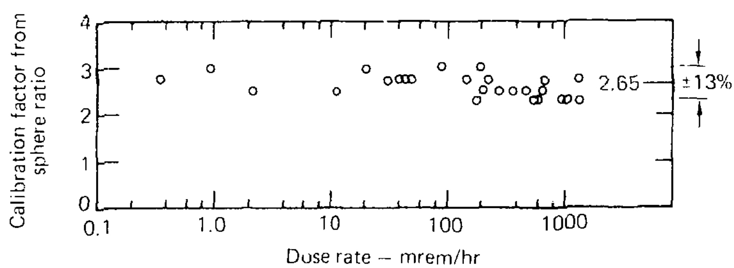


Fig. 3. Lack of Dose-Rate Dependance for Calibration Factors from the 9/3-in.-Sphere Response Ratio

Table 2. Summary of Albedo Neutron Dosimeter Readings

| Location | Calibration factor[a] | | Back:front[b] calib. factor ratio | |
|---|---|---|---|---|
| | Dosimeter on phantom front | Dosimeter on phantom back | Neutron | Gamma |
| 7 | 3.92 | | | |
| 12 | 3.67 | | | |
| 9 | 2.71 | 1.71 | 0.63 | 0.96 |
| 6 | 3.54 | 1.73 | 0.49 | 0.49 |
| 19 | 2.57 | 1.08 | 0.42 | 0.55 |
| 15 | 4.33 | 0.94 | 0.22 | 0.37 |
| 16 | 3.03 | 1.61 | 0.53 | 0.61 |
| 24 | 4.26 | 1.10 | 0.26 | 0.40 |
| Average | 3.50 | 1.36 | 0.43 | 0.56 |
| Deviation | +22%, −27% | +27%, −31% | | |

[a]TLD reading divided by neutron dose from 9-in. sphere rem meter.
[b]Of phantom.

The albedo calibration factors are plotted in Fig. 4, which can be compared to Fig. 3 (results for 9- to 3-in.-sphere responses). The average calibration factor is 3.5, and the spread in points is larger ($\pm$25%). This is more representative of the spread that would exist on a person. We also plotted these results in the dashed box in Fig. 2. With one exception, the points fall to the right of the curve, indicating the TLD dosimeters have a higher sensitivity than predicted by the curve for these spectra. This is also shown by the difference in the average from Figs. 3 and 4 (2.65 compared to 3.50). This difference could be from some increased sensitivity of our TLD's or from changes in our reading procedures. Because TLD's can have various sensitivities, a calibration similar to the above would have to be made with the TLD's to be used at the reactor in order to establish the appropriate calibration factor.

Also shown in Table 2 are the calibration factors from albedo dosimeters placed on the back of the phantom and the ratio of the factors for dosimeters on the back and front. The differences in the ratio indicate that the neutron fluxes on the back of the phantom were always less than and not a constant percentage of the front factors for the various locations. (Note: The front was the side of the phantom facing the center of the containment except for location 9, where this was reversed because the location was behind a shield.) The neutron readings on the back of the phantom averaged 39% of the front readings. Table 2 also shows the ratio of back and front TLD gamma readings. These show the same trend as the neutron response, but the gamma back readings are slightly higher, averaging 56% of the front readings.

The capture of neutrons in cadmium results in gamma rays, which will expose a TLD located next to the cadmium. At locations 7 and 12 we placed a packet of four $^{7}$Li TLD's on the phantom to measure the gamma dose. We then compared the readings of these TLD's to the readings of $^{7}$Li TLD's inside the albedo neutron dosimeter. We found that the gamma readings from TLD's in the dosimeter were a factor of

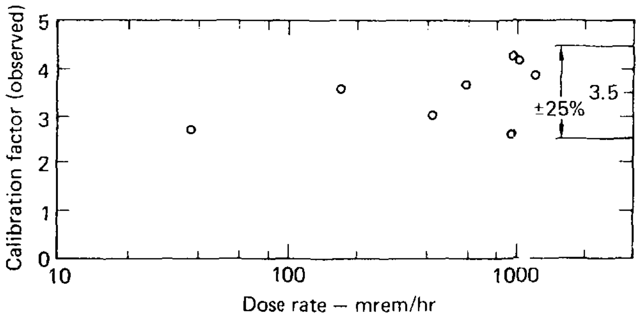


Fig. 4. Lack of Dose-Rate Dependence for Calibration Factors from Albedo Neutron Dosimeters

two greater. We also compared the instrument gamma reading with that from the $^{7}Li$ TLD's and found the TLD readings to be about 7% lower than the instrument reading, which indicates the calibrations of the instruments and TLD's are very similar.

We compared the neutron dose rates determined by the 9-in. sphere to those obtained by the multisphere technique. Because all moderator rem meters overrespond to intermediate-energy neutrons, we expect a 10-in.-diameter sphere rem meter to overrespond by about 60% at a reactor and the 9-in. sphere, which has a higher sensitivity to intermediate-energy neutrons, to overrespond even more. The results of our comparison (see Table 3) indicated a 9-in. sphere overresponse of about 82% for spectra at this reactor.

The ratio of the 9- and 3-in. sphere responses can also be used for a rough estimate of neutron energy (see Fig. 2).[7,8] Neutron energies at this reactor corresponded to a monoenergetic neutron source with an energy of between 2 and 10 keV. Care should be taken in making this type of analysis, however, because a combination of fast and intermediate-energy neutrons could give any average neutron energy between the two extremes.

Because there was no concrete between the reactor vessel and our measurement points, we could expect an abundance of neutrons at the iron reasonances. These neutrons would have energies primarily around 25 keV, with small contributions at 82, 137, and 270 keV. There would also be a contribution of the typical reactor leakage spectrum. The ratios obtained with the 9- and 3-in. spheres and the spectra obtained by the multisphere technique agree with this combined spectrum, as do the ORNL results.

The neutron spectrum at location 9 was unfolded using the multi-sphere technique and is shown in Fig. 5, and the spectrum in flux per unit lethargy is shown in Fig. 6. The tabulated spectrum appears in Table 4. We have also included integral dose equivalent, kerma, and adsorbed-dose information. Although the thermal contribution from the multisphere measurements (8 to 15%) exceeds that determined with the 9-in. sphere instrument, the general character of the spectra show the same kind of information: a large portion of the flux and the dose

Table 3. Comparison of Neutron Dose Rates from the Multisphere Technique and the 9-in. Sphere Remmeter

| Location | Dose rate (mrem/hr) 9-in. sphere | Dose rate (mrem/hr) Multisphere | Ratio of 9-in. to multisphere |
|---|---|---|---|
| 1 | 0.34 | 0.185 | 1.8 |
| 6 | 170 | 107 | 1.6 |
| 9 | 37 | 18.5 | 2.0 |
| 16 | 420 | 229 | 1.8 |
| | | Average | 1.82 |

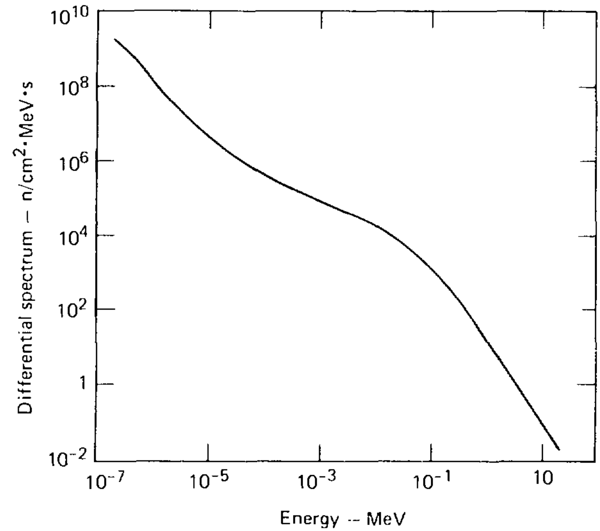


Fig. 5. Differential Neutron Spectrum Obtained at Location 9

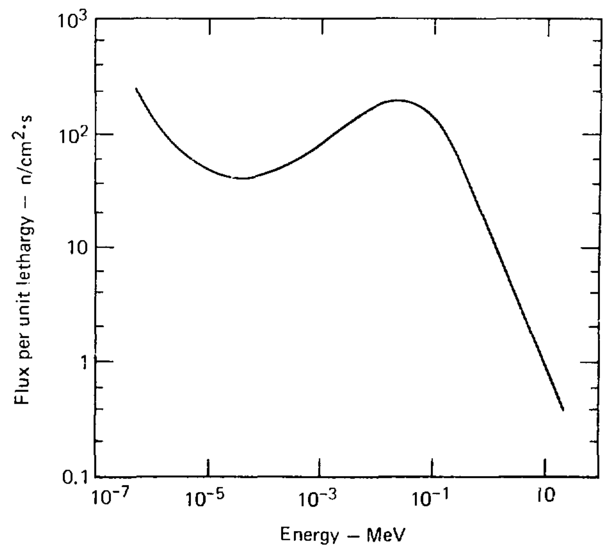


Fig. 6. Differential Lethargy Neutron Spectrum Obtained at Location 9

Table 4. Neutron Spectrum and Dosimetric Data for Location 9[a]

| Energy bin | Av neutron energy (MeV) | Energy band (MeV) | Differential flux (n/cm²·MeV·s) | Integral[b] flux | Integral[b] dose equivalent | Integral[b] kerma | Integral[b,c] element 57 dose |
|---|---|---|---|---|---|---|---|
| 1 | 2.07E-07 | 3.09E-07 | 1.68E+09 | 1.00E+00 | 1.00E+00 | 1.00E+00 | 1.00E+00 |
| 2 | 5.32E-07 | 2.69E-07 | 4.70E+08 | 6.96E-01 | 0.50E-01 | 9.87E-01 | 7.43E-01 |
| 3 | 9.93E-07 | 7.63E-07 | 1.36E+08 | 6.38E-01 | 0.20E-01 | 9.85E-01 | 6.89E-01 |
| 4 | 2.10E-06 | 1.61E-06 | 4.19E+07 | 5.90E-01 | 7.94E-01 | 9.84E-01 | 6.40E-01 |
| 5 | 4.45E-06 | 3.42E-06 | 1.41E+07 | 5.58E-01 | 7.78E-01 | 9.84E-01 | 6.10E-01 |
| 6 | 9.42E-06 | 7.22E-06 | 5.21E+06 | 5.36E-01 | 7.67E-01 | 9.83E-01 | 5.89E-01 |
| 7 | 2.00E-05 | 1.53E-05 | 2.13E+06 | 5.18E-01 | 7.58E-01 | 9.83E-01 | 5.73E-01 |
| 8 | 4.[illegible]2E-05 | 3.23E-05 | 9.68E+05 | 5.03E-01 | 7.51E-01 | 9.83E-01 | 5.60E-01 |
| 9 | 8.94E-05 | 6.89E-05 | 4.82E+05 | 4.89E-01 | 7.44E-01 | 9.83E-01 | 5.49E-01 |
| 10 | 1.89E-04 | 1.45E-04 | 2.62E+05 | 4.73E-01 | 7.36E-01 | 9.83E-01 | 5.37E-01 |
| 11 | 4.04E-04 | 3.18E-04 | 1.52E+05 | 4.56E-01 | 7.28E-01 | 9.83E-01 | 5.23E-01 |
| 12 | 8.55E-04 | 6.40E-04 | 9.26E+04 | 4.33E-01 | 7.18E-01 | 9.82E-01 | 5.07E-01 |
| 13 | 1.80E-03 | 1.38E-03 | 5.77E+04 | 4.06E-01 | 7.06E-01 | 9.81E-01 | 4.87E-01 |
| 14 | 3.80E-03 | 2.91E-03 | 3.56E+04 | 3.69E-01 | 6.90E-01 | 9.77E-01 | 4.60E-01 |
| 15 | 8.05E-03 | 6.20E-03 | 2.10E+04 | 3.20E-01 | 6.69E-01 | 9.65E-01 | 4.27E-01 |
| 16 | 1.70E-02 | 1.30E-02 | 1.12E+04 | 2.60E-01 | 6.44E-01 | 9.34E-01 | 3.83E-01 |
| 17 | 3.61E-02 | 2.77E-02 | 5.26E+03 | 1.92E-01 | 6.00E-01 | 8.68E-01 | 3.27E-01 |
| 18 | 7.64E-02 | 5.86E-02 | 2.06E+03 | 1.24E-01 | 5.22E-01 | 7.48E-01 | 2.60E-01 |
| 19 | 1.58E-01 | 1.13E-01 | 6.63E+02 | 6.82E-02 | 4.10E-01 | 5.64E-01 | 1.88E-01 |
| 20 | 3.18E-01 | 2.27E-01 | 1.76E+02 | 3.35E-02 | 2.91E-01 | 3.78E-01 | 1.27E-01 |
| 21 | 6.40E-01 | 4.56E-01 | 4.05E+01 | 1.49E-02 | 1.83E-01 | 2.30E-01 | 7.87E-02 |
| 22 | 1.29E+00 | 9.20E-01 | 8.64E+00 | 6.35E-03 | 9.81E-02 | 1.29E-01 | 4.51E-02 |
| 23 | 2.59E+00 | 1.85E+00 | 1.82E+00 | 2.66E-03 | 4.32E-02 | 6.77E-02 | 2.40E-02 |
| 24 | 5.22E+00 | 3.73E+00 | 3.95E-01 | 1.09E-03 | 1.86E-02 | 3.40E-02 | 1.25E-02 |
| 25 | 1.05E+01 | 7.50E+00 | 8.84E-02 | 4.10E-04 | 7.27E-03 | 1.51E-02 | 5.89E-03 |
| 26 | 1.96E+01 | 1.09E+01 | 2.02E-02 | 1.02E-04 | 1.88E-03 | 4.06E-03 | 1.85E-03 |

[a]Read E-07, for example, as $\times 10^{-7}$.

[b]Integrals are the fractions of that quantity with energy at or above the bin lower energy limit. Total flux = 215.28 n/cm²·s; dose equivalent rate = 18.452 mrem/hr; kerma rate = 0.12297 ergs/g·hr; element 57 dose rate = $4.4927 \times 10^{-3}$ rads/hr; average energy = $4.3154 \times 10^{-2}$ MeV.

[c]Element 57 of the human body is explained in: F. H. Attix, W. C. Roesch, and E. Tochilin, Radiation Dosimetry, vol. 1 (Academic Press, New York, 1968), p. 295.

equivalent (35 to 45%) is from neutrons between 1 and 200 keV. Because the spectra at the other three measurement locations differed only slightly from this, they are not given in this report.

The total dose-equivalent rates obtained with the multisphere technique are given in Table 3. Based on recent comparisons with moderated $^{252}Cf$ spontaneous-fission neutrons, these values are estimated to be accurate within ±20%.

## CONCLUSIONS

This study indicates that at this reactor the 9-in. sphere rem meter overresponds by about 82%. The calibration of the instruments could be adjusted to correct for this, or they could remain as presently calibrated and have a safety factor of two.

The neutron spectrum is very constant throughout the reactor and probably consists of a 25-keV component superimposed on a 1/E spectrum.

The gamma dose at most locations on the 155-ft level of the reactor is 1/7 the neutron dose (the latter determined by a 9-in. sphere rem meter).

Albedo neutron dosimeters could be used very effectively at this reactor. They would have a high efficiency and the constant neutron spectrum in the reactor would make their interpretation very accurate. The Hankins type albedo dosimeter is ideally suited to this use.

Thermal neutrons contribute about 3 to 4% of the total neutron dose (as determined by a 9-in. sphere rem meter), an insignificant amount.

We stress that the dose obtained by the albedo neutron dosimeters is based on the reading of the 9-in. sphere. Consequently, if the 9-in. sphere reading is 82% high, the dose determined by the albedo neutron dosimeter is also 82% high.

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