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ANALYSIS OF ZPPR/FTR
SHIELD EXPERIMENTS
GAMMA DISTRIBUTION

FEBRUARY 1971

HANFORD ENGINEERING DEVELOPMENT LABORATORY
UNITED STATES ATOMIC ENERGY COMMISSION

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**ANALYSIS OF ZPPR/FTR
SHIELD EXPERIMENTS:
GAMMA DISTRIBUTION**

February 1971

E. T. Boulette
W. L. Bunch

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ABSTRACT

The coupling of one-dimensional transport theory with two-dimensional diffusion theory is shown to be a practical method for realistically predicting measured gamma flux distributions throughout the core, reflectors, shield, and stored fuel zone of assembly FTR-2 in the Zero Power Plutonium Reactor (ZPPR). Two-dimensional diffusion theory calculations were used to generate the neutron flux distribution upon which the volumetric gamma source was based. It is also demonstrated that gamma yields for thermal neutron capture will realistically predict the gamma flux distribution in a fast reactor environment, such as the FTR.

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ANALYSIS OF ZPPR/FTR SHIELD EXPERIMENTS:
GAMMA DISTRIBUTION

INTRODUCTION

Shielding experiments were conducted in the Zero Power Plutonium Reactor (ZPPR) by Argonne National Laboratory (ANL) in support of the design of shields for the Fast Test Reactor (FTR). To the extent possible, the ZPPR loading approximated the basic features of the FTR. The loading included a 1000-liter, two-zone core with a nickel reflector and a sodium-stainless steel shield. To approximate the in-vessel fuel storage capability of the FTR, a stored fuel zone was introduced into the experimental shield. Neutron and gamma distribution measurements were made in two dimensions. Analysis of the neutron measurements has been reported in a separate document.⁽¹⁾ This report presents the analysis of the gamma measurements.

In addition to providing an indication of the gamma heat generation rate throughout the FTR core and its environs, the experiments serve as an evaluation of the nuclear data and analytical methods being used to design and evaluate radiation shields in the FTR.

SUMMARY

The results presented in this report indicate that the gamma flux distribution in the FTR core and its surroundings can be calculated realistically with one-dimensional gamma transport theory using the gamma source distribution calculated with two-dimensional diffusion theory. Comparisons with experimental results indicate good agreement in the core and reflectors; however, the calculations underestimate the measured gamma dose in the radial shield sector by up to a factor of two. When the stored fuel zone is present in the shield, the agreement becomes much better. This inconsistency cannot be easily resolved within this analysis since it is not clear that the source of disagreement is calculational rather than experimental. A comparison of the gamma measurements at locations where the gamma dose should not be influenced by the stored fuel zone indicates a scatter in the measurements by as much as 30 to 40 percent. In addition, it is not clear from the limited measurements that the uncertainty in the measured results does not increase as the dose decreases, i.e., in the outer periphery of the radial shield sector where comparison with calculations is in greatest disagreement. More detailed experiments are planned for the FTR Engineering Mockup Critical (EMC) loading in ZPR-9. These later irradiations will provide a broader basis for determining whether the disagreement shown in this report is due to scatter in the measurements or whether it is a numerical problem.

It is also shown in this report that the gamma dose at any point in a fast reactor environment, such as the FTR, is relatively insensitive to the transport of photons, and is determined principally by the magnitude of the local gamma source strength. Therefore, except for scatter in the measured

results, it is believed that any additional discrepancy between the calculations and the measurements is primarily due to uncertainties in the gamma source distribution, or more specifically, the neutron flux distribution upon which the gamma source depends.

Thermal neutron capture gamma yield spectra were used to generate the neutron capture gamma source distribution. The spectra were normalized to conserve the binding energy of the captured neutron. The relatively good agreement demonstrated in this work indicates that the use of thermal neutron capture gamma yield spectra for calculating gamma source distributions in the FTR system is adequate so long as the captured neutron binding energy is conserved.

DISCUSSION

BACKGROUND

Shielding experiments were carried out in the Zero Power Plutonium Reactor (ZPPR) by Argonne National Laboratory (ANL) to provide a partial evaluation of the shield design of the Fast Test Reactor (FTR). The gamma experiments consisted of irradiating thermoluminescent dosimeters (TLD's) throughout the FTR-2 assembly. In this report, these measurements have been used to assess the adequacy of the nuclear data and numerical methods being employed to calculate the gamma flux distributions in the FTR.

The FTR-2 loading was arranged to approximate the basic features of the FTR. It included a 1000-liter, two-zone core surrounded by a 1-foot thick nickel reflector and a 2-foot thick stainless steel-sodium radial shield. The azimuthal span of the radial shield was limited to about 80° to minimize material requirements. Preliminary calculations indicated that azimuthal leakage into or from the shield sector would not contribute to the attenuation of the flux at the shield midplane.

The present design of the FTR includes in-vessel storage locations within the sodium annulus for storage of fuel sub-assemblies and other core components. Since the ZPPR facility is not large enough to include this feature of the FTR, provisions were made to include a stored-fuel zone in the radial shield sector. Measurements were carried out both with and without a fuel zone in the shield.

Measurements of the two-dimensional gamma flux distribution were obtained by irradiating thermoluminescent dosimeters (TLD's) at various elevations and radial locations in the plane axially bisecting the radial shield sector. The TLD's used

were enriched in ^7Li to eliminate the thermal neutron flux response associated with ^6Li . After irradiation, the TLD's were read out by Battelle-Northwest to provide an absolute measurement of the gamma dose distribution in the FTR-2 loading.

EXPERIMENT DESCRIPTION

The FTR-2 loading ⁽²⁻⁴⁾ in ZPPR is depicted in Figure 1. The stationary half and the movable half were identically loaded. The areal profile of the loading is depicted in Figure 2, whereas the cross-sectional profile through the shield midplane is shown in Figure 3. The TLD irradiation locations are indicated in both figures. The stored-fuel zone is shown in the shield sector in both figures; however, measurements were made with and without the stored fuel in position.

The ZPPR facility was loaded by positioning drawers ($\sim 2\text{ in.} \times \sim 2\text{ in.} \times \sim 48\text{ in.}$) into both halves of the stainless steel matrix. Each drawer contained an array of platelets varying in dimensions from about $1/8\text{ in.} \times 2\text{ in.} \times 1\text{ in.}$ to $1/2\text{ in.} \times 2\text{ in.} \times 8\text{ in.}$ ^(2,4,5) Platelets were made of Pu, U_3O_8 , Ni, stainless steel, Fe_2O_3 , and Na and Na_2CO_3 canned in stainless steel. Each drawer was loaded with the appropriate platelets and positioned in the ZPPR matrix to approximate the material composition of the different zones in the FTR. The result is a heterogeneous mixture in each zone; however, the heterogeneity is sufficiently fine to allow a zone-dependent homogeneous treatment in the analysis. ⁽¹⁾

GAMMA MEASUREMENTS

Thermoluminescent dosimeters (TLD's) seemingly are well suited for use in measuring the gamma distribution within a

FIGURE 1. ZPPR Assembly with FTR Shield Loading

FIGURE 2. Area Profile of ZPPR Assembly FTR-2 (Section BB)

FIGURE 3. Axial Cross Section of ZPPR Assembly FTR-2 (Section AA)

zero power reactor system. Ionizing radiations interact with thermoluminescent material to raise electrons from the valence band to the conducting band. Rather than recombining, some of the electrons (or holes) that are produced are able to move until they become trapped. When the material subsequently is heated, the trapped electrons (or holes) are released and able to recombine, giving up energy in the form of light. The light output is directly proportional to the amount of energy stored, that is, to the dose in rads absorbed by the material. In practice, calibration curves are employed to relate the light output from a given material to the absorbed dose, eliminating the need for absolute light emission measurements.

A number of different materials have been employed successfully as TLD's. For the present application, lithium fluoride was used. This material has a useful response range from a few mrad up to about 10^5 rads. There appears to be no measurable dependence of the response on the dose rate nor on the angular distribution of the incident photons. The response of lithium fluoride is essentially independent of energy in the range from about 0.01 MeV to 10 MeV. The effective atomic number of lithium fluoride is 8.2; therefore its response is almost equivalent both to air and to tissue. By employing lithium enriched to 99.993 percent in the isotope ^7Li , the response to neutrons was minimized.

Normal environmental conditions, such as light, temperature, and humidity, have little if any effect on the response of lithium fluoride. Fading of the stored information is reported to be less than 5% per twelve week period. These factors were an important consideration in the use of TLD's, since the ZPPR is located in Idaho and the irradiated TLD's were mailed to Washington for readout.

A calibration curve for the TLD's used in this experiment was obtained using a ^{60}Co gamma source. Three of the TLD ribbons (1/8 in. $\times$ 1/8 in. $\times$ 0.035 in.) were irradiated at each calibration point with the mean light output being employed to develop the dose-response curve shown in Figure 4. The calibration exposures ranged from 0.5 to 5000 roentgens to cover the range of interest. A Conrad TLD reader, modified to read out through a Dymec analog-to-digital converter, was used for measuring and recording the light output from the calibration and experimental ribbons. Glow curves were taken for each TLD to assure that the total light output was measured. A typical glow curve is illustrated in Figure 5. The main light output peak occurs at a temperature of about 200°C whereas a smaller peak is exhibited below about 100°C. Since the gamma field around the cobalt calibration source was established using an ionization chamber, the dose response of the TLD's is given in roentgens. This assumes that the presence of the small TLD's do not affect the gamma field during the calibration irradiations.

Pairs of TLD ribbons were loaded at selected positions throughout the ZPPR/FTR configuration and irradiated to an integrated exposure of about 750 watt-hours. The TLD's were placed in small labelled paper packets, which were taped to a plate at the appropriate position within a drawer. The TLD's were removed from the reactor within a few hours following the irradiation and subsequently mailed to the Pacific Northwest Laboratory for readout. Care was taken throughout to prevent contamination of the ribbon surfaces, even though this was deemed unnecessary.

One of the TLD's in each pair contained lithium enriched in ^7Li , whereas the other contained natural lithium. In the

FIGURE 4. TLD Dose-Response Calibration Curve

FIGURE 5. Typical LiF Glow Curve

case of the TLD's that were irradiated within the reactor, the light output of the natural and the enriched sample in each pair was comparable. In the reflector and shield, a progressively higher response was induced in the natural lithium ribbons as a function of distance from the core fuel. This expected effect is related to the neutron energy spectrum. In the core, the most probable neutron energy is about 0.2 MeV and there are essentially no thermal neutrons. In the reflector and shield, the neutron spectrum softens appreciably and a significant low energy and thermal component exists. Natural lithium contains 7.42 atom-percent ^6Li , which has a cross section of 950 barns for the thermal neutron (n, α) reaction. In the presence of low energy neutrons, a significant amount of energy can be stored in a natural LiF ribbon because of this reaction. It is possible to estimate the thermal neutron flux distribution from the difference in response of the natural and enriched TLD's. However, this was not done in the present experiment because foil activation data were employed to measure the neutron flux distribution.

No attempt was made to account for potential differences in the environment in which the experimental and the calibration ribbons were irradiated. The TLD response is known to be relatively independent of the energy of the incident photons and electrons; therefore, the known spectral differences were assumed to be negligible. In both the experiment and the calibration the ribbons were encased in paper; however, the calibration ribbons were then surrounded by air, whereas the measurement ribbons were embedded in the mockup. Other than the use of the enriched ribbons to eliminate low energy neutron reactions, no correction was attempted for energy deposited by neutrons. The error introduced into the measured results by these effects is believed to be small, and probably of no

greater importance than the omission in the photon transport calculations of fluorescent radiation, annihilation radiation, and secondary bremsstrahlung.

Results of the gamma distribution measurements obtained from the TLD irradiations are contained in Appendix A together with the results of the analyses described in the following sections.

ANALYSIS

Gamma Source Calculations

The analysis of the gamma experiments requires a knowledge of the neutron flux distribution, which in turn establishes the volumetric gamma source. Analysis of the neutron distribution was treated in another report⁽¹⁾ and will not be discussed in great detail here. The neutron distribution was calculated in two-dimensional geometry with the diffusion code 2DBS⁽⁶⁾ using the 31-group transport corrected HOMSET cross-section library.⁽¹⁾ Validity of the calculations was based on agreement with experimental reaction rates for several isotopes. X-Y and R-Z solutions were coupled to account for the transverse leakage of the stored fuel zone. Further details on this coupling approach are reported in Reference 1.

Once the two-dimensional neutron flux distribution was obtained, the multigroup gamma source distribution was calculated by forming the product of the multigroup neutron fluxes and the gamma production cross sections. The gamma production cross sections fall into three categories: 1) fission-product gamma production cross sections, 2) prompt fission gamma production cross sections, and 3) neutron capture gamma production cross sections. Gammas produced from neutron inelastic scattering events were ignored in this analysis for two reasons:

1) the majority of inelastic scattering events take place in the fuel regions where other gamma producing events (fission and capture) predominate, and 2) the gamma energy release associated with inelastic scattering is but a fraction of the release associated with fission and capture. A similar rationale resulted in the elimination from this analysis of neutron capture decay gammas. Estimates of the gamma dose associated with the decay of radioactive stainless steel and sodium (assuming activity at complete saturation) indicate that decay gammas would contribute no more than ~0.1% to the total dose at full power. In a facility such as ZPPR, induced activation is kept at a minimum. The TLD's were loaded just prior to the irradiation and unloaded within a few hours following the irradiation. Thus, decay gamma could not contribute significantly to the TLD dose.

The experimental prompt fission gamma yield spectrum⁽⁷⁾ for ^{235}U was used to approximate the yield spectra for all fissionable materials in the FTR-2 loading. Literature^(8,9) suggests that such an approximation is quite reasonable. Since the gamma transport calculations were to be done in an 18-group approximation, the measured spectrum was energy weighted and partitioned into the appropriate 18-group structure. The resulting spectrum is listed in Table B-1 of Appendix B. The weighting was such as to conserve the total gamma energy released by fission as reported by the experiment.

In addition to the prompt fission gamma yield spectrum, a fission-product gamma yield spectrum was also required. To generate this spectrum, the RIBD⁽¹⁰⁾ code was run to obtain the total fission-product activity associated with plutonium oxide-uranium oxide fuel such as is found in the FTR. The code ISOSHL⁽¹¹⁾ then was used to calculate a 16-group gamma spectrum based on the total activity calculated by RIBD. After

appropriate normalization, energy weighting, and fitting to the 18-group structure, the spectrum listed in Table B-1 of Appendix B was obtained for the fission-product gamma yield spectrum.

The gamma source contribution due to neutron capture events requires knowledge of gamma yield spectra as a function of the kinetic energy of the capture neutron. Since very little data exist for nonthermal neutron capture, it was decided that only the spectral yield for thermal neutron capture would be used in this study. A compilation of the 18-group spectra for materials of interest to the FTR is listed in Appendix B. These were all based on data reported in the literature. (12-15) In all cases, the spectra were uniformly adjusted so as to yield a total gamma energy release equal to the binding energy of the captured neutron.

Once the gamma yield spectra are known for the various nuclear events of interest, the microscopic gamma production cross sections may be calculated by the following relationship:

$$\sigma_{i,j} = \sigma_{f,j} [Y_{pf,i} + Y_{fp,i}] + \sigma_{c,j} Y_{c,i},$$

where the indices i and j refer to gamma group i and neutron group j , respectively; $\sigma_{f,j}$ and $\sigma_{c,j}$ are the group j microscopic fission and (n,γ) capture cross sections, respectively; and $Y_{pf,i}$, $Y_{fp,i}$ and $Y_{c,i}$ are the prompt fission, fission-product and neutron capture gamma yield values in gamma group i . The following equation yields the energy dependent gamma source, S_i , in gamma group i :

$$S_i = \sum_j \sum_k N_k \sigma_{i,j,k} \phi_j,$$

where ϕ_j is the neutron flux in group j and N_k is the atom density of nuclide k . The volumetric gamma source distribution

was calculated in R-Z geometry using the discrete ordinate code DOT.⁽¹⁶⁾ However, the gamma transport solution was not made with DOT because of core storage limitations. The geometric models used for the source calculations are depicted in Figures 6(a) and (b). The material compositions by zone are given in Table 1. The empty matrix and the ZPPR support structure were included in the calculation to approximate the actual boundaries. The material compositions used for these zones are 13 vol% stainless steel for the matrix and 100 vol% stainless steel for the support structure. The bottom and left boundary conditions imposed on the geometric models were reflective; whereas, the top and right boundary conditions were represented by a vacuum. The azimuthal leakage of the stored fuel zone was treated as described in Reference 1. In Figure 7 are plotted the gamma source components (prompt fission, fission-product, and capture gammas) as well as the total gamma source at $Z = 0$. Note the source components have been integrated over energy. In performing the transport calculation, the gamma source distribution was in multi-group form (18 groups) and has units of gammas/cm²-sec. It is interesting to note that the gamma source components are essentially equal in the core. The same situation exists in the stored fuel zone.

Gamma Transport Calculations

All gamma transport calculations for this analysis were done in one-dimensional geometry. The justification for this approach was twofold: 1) if the experiment validated the one-dimensional treatment such an approach could represent a considerable computer time savings for FTR shield analysis; (2) at the time of the analysis, the computer high speed core storage capability required for two-dimensional discrete ordinates calculations of this complexity was not available.

FIGURE 6(a). R-Z Map of Material Zones (With Stored Fuel)

FIGURE 6(b). R-Z Map of Material Zones (Without Stored Fuel)

TABLE 1. Material Compositions by Zone (atoms/barn-cm)

Nuclide	Depleted Zone	Core	Radial Reflector	Axial Reflector	Shield	Stored Fuel
^{235}U	0.0000120	0.0000131	-----	-----	-----	0.0000133
^{238}U	0.0096771	0.0059766	-----	-----	-----	0.0061128
^{239}Pu	-----	0.0012737	-----	-----	-----	0.0014153
^{240}Pu	-----	0.0001688	-----	-----	-----	0.0001876
^{241}Pu	-----	0.0000278	-----	-----	-----	0.0000310
C	0.0008514	0.0010944	0.0001547	0.0000284	0.0000264	0.0007295
O	0.0152671	0.0152206	-----	-----	-----	0.0152920
Cr	0.0020978	0.0025348	0.0019747	0.0022823	0.0102972	0.0025351
Fe	0.0121299	0.0130662	0.0071580	0.0081683	0.0380490	0.0143480
Ni	0.0009453	0.0011631	0.0460037	0.0189744	0.0045539	0.0011633
Na	0.0080440	0.0085302	0.0065078	0.0133588	0.0059617	0.0008003

Empty matrix - 13 vol% stainless steel

Support structure - 100 vol% stainless steel

FIGURE 7. Radial Gamma Source Distributions at the ZPPR/FTR-2 Midplane

The one-dimensional transport theory code used for all calculations was the discrete ordinates S_n code, ANISN.⁽¹⁸⁾ The gamma transport cross sections used were generated by the MUG⁽¹⁹⁾ code with a source-compatible 18-group structure and a P_5 expansion of the scattering cross sections.

From Figures 6(a) and 6(b), it is obvious that two different one-dimensional geometrical models may be used in the analysis: 1) a cylindrical model for radial traverse calculations, and 2) a plane model for axial traverse calculations. Thus, for a radial traverse calculation at $Z = 0$, one extracts the midplane source distribution from the two-dimensional source array, and uses this one-dimensional source distribution as an input volumetric source for ANISN. The geometrical model used in this case is that of a cylinder. After appropriate point-wise convergence on the flux, the flux distribution was converted to a dose distribution by using the conversion curves in Reference 18. The calculations were then normalized to an integrated exposure of 750 watt-hours⁽²⁰⁾ and compared with the experiment. The result of such a comparison is depicted in Figure 8, where the distribution is along the midplane of ZPPR through the core, reflector, shield, and stored fuel. The agreement is very good and leads one to conclude that the gamma dose at any point within the assembly is strongly dependent upon the local source strength and that the gamma transport mechanism contributes insignificantly to the dose. This phenomenon can readily be seen in Figure 9, where the one-dimensional source extracted from the two-dimensional array was zeroed in the radial shield sector and not changed in the other zones. An S_8P_3 transport calculation was then made and the solution was converged to 10^{-5} on the point-wise flux. The flux attenuation in the shield is so severe when compared to the source distribution, that it is obvious that

FIGURE 8. Radial Gamma Dose Distribution at Z = 0 in. (With Stored Fuel)

FIGURE 9. Radial Dose Distribution with Source in Shield Set Equal to Zero

the transport mechanism is unimportant in this analysis. As a final check on this point, the discrete ordinate quadrature and the Legendre polynomial expansion of the scattering cross sections were varied (S_4P_0 , S_8P_3 , and $S_{12}P_5$) and the results were intercompared. Insignificant differences in the dose distributions resulted from this comparison, indicating that the transport phenomenon is unimportant in this type of shield configuration. Since the S_8P_3 calculation was not significantly more costly than the S_4P_0 calculation, all of the analyses were made employing an S_8P_3 approximation. In addition to varying the quadrature order and the scattering cross section expansion representation, the multigroup structure was also varied. Calculations with the 18-group structure were compared with those made with an older 10-group set. Differences between the two calculations were relatively minor.

The source distributions depicted in Figure 7 indicate severe discontinuities in the radial direction. Similar discontinuities exist in the axial direction. One-dimensional cylindrical calculations to obtain radial dose traverses easily accommodate radial discontinuities in the source distributions. To accommodate axial discontinuities with a radial one-dimensional calculation, special treatment of the source distribution must be made. For example, to obtain a calculation of the dose distribution at the 18-inch elevation (core-axial reflector interface), one must decide upon the appropriate source distribution and the material composition through which gamma transport takes place. In Figure 10, two calculations of the radial dose distribution are compared with the experiment. The calculations were done in one-dimensional cylindrical geometry, where the radial variation of the material composition corresponds to the R-Z model at an elevation within the core. No stored fuel was present in the shield for these calculations. The source distribution used for the calculation of the upper curve was taken from the two-dimensional source array at an elevation corresponding to the last axial mesh point within the core. Thus, this source distribution included fission-product and prompt fission gammas within the core radius. The complication arises because one-dimensional cylindrical calculations assume no axial variation in the source. In reality, the source distribution of gammas at the core-axial reflector interface goes through an axial discontinuity. To resolve this, a weighted source distribution was calculated to account for the discontinuity. The source distribution used as input for ANISN was the sum of one-half of each of the radial source distributions immediately contiguous to both sides of the axial interface. The dose distribution obtained using this source is also plotted in Figure 10. The agreement within the core radius is greatly improved; whereas, the results are not perturbed at all in the radial reflector and shield, where axial discontinuities do not exist.

FIGURE 10. Radial Dose Distributions at Core-Axial Reflector Interface

The same procedure and weighting scheme was used to generate the radial source distribution at the 30-inch elevation (axial reflector-empty matrix interface). Results of these calculations are in rather poor agreement with the experimental results--with more than a factor of 2 disagreement throughout. This is probably due to 1) the homogeneous treatment of the empty matrix tubes, 2) inappropriate weights assigned to each source component on either side of the interface.

An alternative approach for accommodating axial discontinuities in the source distributions is to perform one-dimensional calculations in plane geometry to obtain axial dose distributions. These can be made at radii which are distant from radial discontinuities, such as in the center of the enriched zone of the core, the center of the radial reflector, the center of the shield and center of the stored fuel zone. In Figure 11 is plotted the results of an axial traverse calculation at a radius of ~ 13 inches from the core center (radial midpoint of the enriched zone of the core). The agreement between the calculation and the measurements are seen to be within 25% at all points, including locations of axial discontinuity in the source. Similar axial calculations were done at the radial midplane of the reflector and shield, and the good agreement is demonstrated in Figures 12(a) and 12(b).

The results of the experiments are plotted in Appendix A. Also plotted are the corresponding radial traverse calculations.

FIGURE 11. Axial Dose Distribution at $R \sim 13$ in.

FIGURE 12(a). Axial Gamma Dose Distribution at $R \sim 44$ in. FIGURE 12(b) Axial Gamma Dose Distribution at $R \sim 28$ in.

ACKNOWLEDGMENTS

The authors acknowledge the computer programming assistance afforded them by Duane R. Marr. They also acknowledge Leo Kocher (BNW) who processed the TLD's, and Arthur Derosier and Matt Schramer, both AEC Trainees, who assisted in developing the gamma transport and production cross sections.

APPENDIX A

COMPILATION OF MEASUREMENTS

The measured gamma dose distributions are plotted in Figures A-1 through A-6. Radial traverse calculations are also depicted for comparison. All calculations were made in one-dimensional cylindrical geometry using an S_8P_3 approximation with the discrete ordinates code ANISN. The calculations at the 18-inch and 30-inch elevations employed a weighted source distribution as discussed in the report.

Above each figure are indicated the material zones in which the measurements were made. The abbreviations used are as follows:

DEP	Central depleted zone
CORE	^{239}Pu enriched peripheral zone
R.REFL.	Radial reflector
A.REFL.	Axial reflector
SHIELD	Radial shield sector
FUEL	Stored fuel zone
MATRIX	Empty matrix
S	Structual support plate

The lower figures are labeled completely; whereas the upper figures are labeled to show the different zone introduced by the stored fuel.

FIGURE A-1(a). Radial Gamma Dose Distribution at $Z = 0$ in. (With Stored Fuel)

FIGURE A-1(b). Radial Gamma Dose Distribution at $Z = 0$ in. (Without Stored Fuel)

FIGURE A-2(a). Radial Gamma Dose Distribution
at Z = 6 in. (With Stored Fuel)

FIGURE A-2(b). Radial Gamma Dose Distribution at
Z = 6 in. (Without Stored Fuel)

FIGURE A-3(a). Radial Gamma Dose Distribution at Z = 12 in. (With Stored Fuel)

FIGURE A-3(b). Radial Gamma Dose Distribution at Z = 12 in. (Without Stored Fuel)

FIGURE A-4(a). Radial Gamma Dose Distribution at Z = 18 in. (With Stored Fuel)

FIGURE A-4(b). Radial Gamma Dose Distribution at Z = 18 in. (Without Stored Fuel)

FIGURE A-5(a). Radial Gamma Dose Distribution at
Z = 24 in. (With Stored Fuel)

FIGURE A-5(b). Radial Gamma Dose Distribution at
Z = 24 in. (Without Stored Fuel)

FIGURE A-6(a). Radial Gamma Dose Distribution at $Z = 30$ in. (With Stored Fuel)

FIGURE A-6(b). Radial Gamma Dose Distribution at $Z = 30$ in. (Without Stored Fuel)

APPENDIX B

GAMMA YIELD SPECTRA

The gamma yield spectra reported in this appendix are based on data found in the literature. The prompt fission gamma yield spectrum is listed in Table B-1. It was derived from Reference 7 and was used^(8,9) in this study for fission events in uranium as well as plutonium isotopes. In fitting the literature data to the 18-group structure, energy weighting was used to conserve energy. Also listed in Table B-1 is the fission product gamma yield spectrum. Its derivation is discussed in the report. The fission-product gamma yield spectrum was based on the inventory appropriate for all fission events in the mixed plutonium-uranium fuel.

The neutron capture gamma yield spectra were taken from References 12 through 16. All spectra were based on thermal neutron capture, because data for other neutron energies are very scarce. Energy weighting was used to partition the data into the 18-group structure, and the resulting spectra were normalized to conserve the total binding energy of the captured neutron. The resulting 18-group gamma yield spectra are tabulated in Table B-2.

TABLE B-1
PROMPT FISSION AND FISSION-PRODUCT
GAMMA YIELD SPECTRA (γ /fission)

<u>Group No.</u>	<u>Energy Limits (MeV)</u>	<u>Prompt Fission</u>	<u>Fission- Product</u>
1	10.00-8.00	0.0010	0.0
2	8.00-7.00	0.0005	0.0
3	7.00-6.00	0.0093	0.0
4	6.00-5.00	0.0197	0.0
5	5.00-4.00	0.0586	0.0
6	4.00-3.50	0.0805	0.0
7	3.50-3.00	0.0805	0.0
8	3.00-2.50	0.2480	0.2554
9	2.50-2.00	0.2480	0.2554
10	2.00-1.60	0.4997	0.4455
11	1.60-1.20	0.6425	0.5728
12	1.20-0.90	0.6425	0.5728
13	0.90-0.60	1.1120	1.3783
14	0.60-0.40	1.6685	2.0681
15	0.40-0.21	2.7528	3.4120
16	0.21-0.12	0.0	0.0
17	0.12-0.07	0.0	0.0
18	0.07-0.01	0.0	0.0

TABLE B-2
THERMAL NEUTRON CAPTURE GAMMA YIELD SPECTRA (γ /capture)

Group No.	Energy Limits (MeV)	H	Be	B	C	O	Na
1	10.00-8.00	0.0	0.0	0.1100	0.0	0.0	0.0
2	8.00-7.00	0.0	0.0	0.0	0.0	0.0	0.0
3	7.00-6.00	0.0	0.0	0.4505	0.0	0.0	0.2298
4	6.00-5.00	0.0	0.7840	0.0	0.0	0.0	0.0918
5	5.00-4.00	0.0	0.0	1.6840	0.0	0.0	0.0987
6	4.00-3.50	0.0	0.0	0.0	1.1577	0.2959	0.2753
7	3.50-3.00	0.0	0.4870	0.0	0.4238	0.2959	0.2753
8	3.00-2.50	0.4451	0.0	0.0	0.0	0.2762	0.4291
9	2.50-2.00	0.4451	0.0	0.0	0.9889	0.2762	0.4291
10	2.00-1.60	0.0	0.0	0.0	0.0	0.1395	0.0902
11	1.60-1.20	0.0	0.0	0.0	0.0	0.1793	0.1159
12	1.20-0.90	0.0	0.0	0.0	0.0	0.1793	0.1159
13	0.90-0.60	0.0	0.0	0.0	0.0	0.0	0.0
14	0.60-0.40	0.0	0.0	0.0	0.0	0.0	0.0
15	0.40-0.21	0.0	0.0	0.0	0.0	0.0	0.0
16	0.21-0.12	0.0	0.0	0.0	0.0	0.0	0.0
17	0.12-0.07	0.0	0.0	0.0	0.0	0.0	0.0
18	0.07-0.01	0.0	0.0	0.0	0.0	0.0	0.0

		Al	Si	Cr	Fe	Ni	U and Pu
1	10.00-8.00	0.0	0.0258	0.4710	0.0424	0.5442	0.0400
2	8.00-7.00	0.3512	0.0980	0.2207	0.6075	0.1463	0.0
3	7.00-6.00	0.0842	0.1664	0.1119	0.1079	0.1854	0.0
4	6.00-5.00	0.0985	0.0709	0.0492	0.1192	0.0779	0.0
5	5.00-4.00	0.3532	0.7788	0.0111	0.1084	0.0275	0.1511
6	4.00-3.50	0.1573	0.3431	0.0133	0.0648	0.0185	0.9715
7	3.50-3.00	0.1573	0.3431	0.0133	0.0648	0.0185	0.9715
8	3.00-2.50	0.1678	0.1000	0.0135	0.0533	0.0332	0.4550
9	2.50-2.00	0.1678	0.1000	0.0135	0.0533	0.0332	0.4550
10	2.00-1.60	0.0875	0.0616	0.0246	0.0693	0.0250	0.5394
11	1.60-1.20	0.1126	0.0792	0.0316	0.0891	0.0321	0.6935
12	1.20-0.90	0.1126	0.0792	0.0316	0.0891	0.0321	0.6935
13	0.90-0.60	0.0	0.0	0.0	0.0	0.0	0.0
14	0.60-0.40	0.0	0.0	0.0	0.0	0.0	0.0
15	0.40-0.21	0.0	0.0	0.0	0.0	0.0	0.0
16	0.21-0.12	0.0	0.0	0.0	0.0	0.0	0.0
17	0.12-0.07	0.0	0.0	0.0	0.0	0.0	0.0
18	0.07-0.01	0.0	0.0	0.0	0.0	0.0	0.0

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