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**SHIELDING CALCULATIONS
FOR IRRADIATED FTR FUEL**

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SHIELDING CALCULATIONS
FOR IRRADIATED FTR FUEL

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DECEMBER, 1966

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SHIELDING CALCULATIONS FOR IRRADIATED FTR FUEL

ABSTRACT

Shielding calculations have been made for irradiated FTR fuel to provide systems engineers with a basis for conceptual design of those facilities that are to contain this material. Gamma intensity as a function of shield thickness was obtained for several materials of interest, including lead, tungsten, depleted uranium, concretes of various density, water and air. Heat generation as a function of decay time was also calculated for evaluation of cooling requirements. A preliminary estimate of neutron generation rate in irradiated fuel indicated that special neutron shielding will only be required if beryllium is contained within the fuel.

INTRODUCTION

Calculations in this report serve as a basis for the conceptual design of shields for those Fast Flux Test Facility (FFTF) systems in which irradiated fuel will be handled, transported, stored, cleaned, or examined. It was necessary to make a number of assumptions to define the fuel and the Fast Test Reactor (FTR) environment to carry out this study. Generally the assumptions do not limit application of the results to specific reactor or facility concepts. In those areas where specific concepts are assumed, the results will still provide a basis for preliminary evaluation of alternate concepts. This study also serves to indicate the problem areas requiring investigation in greater detail as the design of the facility progresses. Similar studies are in progress to establish the reactor shield requirements and to determine shield requirements for sodium, which probably will serve as the primary reactor coolant.

SUMMARY

A number of calculations have been made to provide system engineers with basic shield requirements for those FFTF systems that are to contain irradiated fuel. The computer program RIBD was used to determine the fission product inventory and heat generation in prototypic plutonium fuel as a function of exposure history and decay time. The computer program ISOSHLD was used to determine the shielding effectiveness of several materials for various geometric arrangements. Lead, tungsten, and depleted uranium were considered as possible shield materials for a fuel handling machine in which space and weight would be important parameters. Concretes of various composition and density were considered for cells in which the fuel would be cleaned, disassembled, and examined. Effectiveness of water as a shielding material was also calculated. Finally, an estimate of the neutron production rate within the irradiated

fuel was made to determine the extent of the neutron shielding problem that might be encountered.

As the design of the facility evolves and requirements become more clearly defined, it will be necessary to continue these analyses to assure that optimum use is made of shielding materials throughout the FFTF.

BASES FOR STUDY

Although the shielding calculations that have been made are not limited in application to a specific facility concept, the interim reference concept has been used to define the problem areas of interest. In this concept, the fuel would be removed from the reactor into a shielded fuel handling machine, which would be used to transport it to the storage and/or examination facilities. The fuel would remain in sodium while being transported and stored. When placed in the examination facility, the fuel would be cleaned of sodium and, subsequently, would be gas cooled while being examined and/or disassembled. Following disassembly, the individual components could continue to be gas cooled, or could be submerged in water for further examination and processing. The shielding problems associated with these processes have been treated in the present study.

Before calculating the shielding effectiveness of various geometrical arrangements of materials, it is necessary to define the intensity and spectral distribution of the source and to establish the dose rate criteria that are to be met. Since many of the factors that influence these parameters have not been

firmly established, a number of assumptions were made to carry out this study. The major assumptions are as follows:

FUEL ASSEMBLY

One of the basic parameters in any shielding calculation is that of defining the source. Since the fuel and test specimens for irradiation in the FTR will not be specified in detail for some time, it has been necessary to assume the characteristics of a typical assembly.

The first assumption made is that most of the isotopes to be shielded will result from the fission of Pu^{239} . Fast fission events occurring in other plutonium isotopes or in U^{238} would not be expected to affect the results. Although some difference would be expected if the fission events were due to U^{235} , the difference would not be expected to alter the shielding requirements significantly. Likewise, the form of the fuel (cermet, oxide, carbide, or other) would not be expected to affect the gamma shielding requirements. However, as will be shown, the presence of beryllium in the fuel would strongly influence the neutron shielding requirements.

The accepted active length of the fuel material was 90 cm, based on the neutronic calculations for a right circular cylinder core. Variations of a few centimeters in the length of the fuel would not materially influence the shielding requirements. Generally it was believed that the fuel assembly could be approximated by a line source for shielding calculations. To estimate the effect of self absorption, it was assumed the effective diameter of an assembly would be 10 cm. Although the exact diameter, arrangement and composition of the

assembly will influence the amount of self-absorption and decay heat generation within the fuel, self-absorption is expected to be a relatively unimportant factor in establishing basic shielding requirements. This is true because shielding requirements are strongly influenced by the penetrating high energy photons while self-absorption is most pronounced for the low energy photons.

Table I presents the assumed composition of a fuel assembly used for estimating the effect of self-absorption on the shielding requirements.

TABLE I. Assumed Composition of FTR Fuel Assembly

<u>Element</u>	<u>g/cm³</u>
Sodium	0.43*
Nickel	0.16
Plutonium-Uranium	2.44
Oxygen	0.33
Iron	1.82

*not present if the assembly is gas-cooled

The design heat generation rate of the FTR is currently set at a nominal 400 MW. The interim reference concept provides for 61 positions within the core for driver fuel, test loops, and control elements. Present plans are that from three to six of the positions would be utilized for control while six to twelve positions would be reserved for test purposes. That would leave 43 to 52 positions for driver fuel; therefore, the assumption was that a typical driver fuel assembly will generate 10 MW at full power operation. Where necessary, it was further assumed that a fuel assembly would consist of 200 separate fuel pins, with

each pin generating 0.05 MW (50 kW) during operation.

The irradiation lifetime of the driver fuel will depend, in part, on the amount of fertile material that is included, while the irradiation history of a test assembly will depend on test requirements. For the shielding calculations it was assumed that an element would be operated for 100 days continuously at full power. It is anticipated in actual practice that the fuel would be operated for two or more operating cycles with one or more intervening outages of some duration. The assumption of 100 days of continuous operation should realistically account for the buildup of the longer-lived fission product isotopes. The shorter-lived isotopes, which dominate the shielding requirements, would also be approximated realistically by this assumption.

SHIELD MATERIALS

Gamma attenuation efficiency is usually directly related to the specific gravity of the shield. However, because of cost or structural considerations, dense materials are not used in all gamma shields. A number of different materials were used in the study so that thickness considerations could be included in various design concepts. Table II presents the assumed specific gravities of the materials used in the calculations. Although these values are realistic, it should be recognized that allowance will need to be made if the in-place specific gravity of the material is appreciably different than the value tabulated in Table II. Also, no allowance has been made in the calculations for the presence of lower density penetrations through the shield or for occlusions that might be required or that might occur in the actual construction of a shield wall.

TABLE II. Assumed Specific Gravity of Shield Materials

Material	g/cm ³
Air	0.0013
Water	1.0
Concrete	
Ordinary	2.5
Magnetite	3.5
Ferrophosphorus	4.5
Iron (steel)	7.5
Lead	11.3
Tungsten	18.5
Uranium	18.5

were used in this study. Exposure rates were calculated as a function of shield thickness in order that this parameter might be considered in evaluating the design concept. If shield requirements appear to be quite restrictive in a particular region, then alternate concepts might need to be considered to circumvent the problem area. As the design of a particular component becomes more firm, specific shielding criteria can be established.

CALCULATION METHODS

DOSE EXPOSURE CRITERIA

National and international committees concerned with radiation protection have recommended that industrial dose exposure to individuals not exceed 5 rem/yr. It is thought that this exposure would limit biological damage to acceptable levels. In addition to the annual limit, exposure rates have also been recommended for shorter time periods. In designing the FFTF, it is necessary to provide sufficient shielding to assure that the operators exceed neither the annual exposure limits nor the recommended exposure rates.

In normally occupied regions of the facility, an exposure rate of 0.2 mrem/hr has been used as the basis for establishing shielding requirements. At this exposure rate an operator would receive less than one-tenth of the annual recommended limit. Thus, this rate is sufficiently low to leave the operator with additional exposure for the intermittent and nonroutine exposures that will be encountered in the operation of the facility.

In regions where continuous occupancy is not required (either by the operator or by the radioactive source), no firm criteria

FISSION PRODUCT INVENTORY

The fission product inventory in a fuel sample was calculated using the computer code Radio-Isotope Buildup and Decay (RIBD), which is a Hanford developed general purpose, multiple path, multiple chain, grid processor.⁽²⁾ To make the fission product calculations, it is necessary to include a library containing the pertinent nuclear data for the isotopes of interest. In addition to those isotopes formed directly from the fission of U²³⁵ or Pu²³⁹, this also includes those isotopes that can be formed subsequently as a result of neutron capture and/or chain decay.

The various isotopes described by the library are organized into a grid with the linkages established from the chain branching data, yield data, and neutron capture cross sections with their branching ratios. A sample of a portion of the grid is shown in Figure 1. Upper boxes of a pair, where they exist, refer to isomeric states, while the lower boxes refer to ground states. In the figure, Y235 and Y239 refer to the variable direct yield rates from U²³⁵ and Pu²³⁹, respectively. Horizontal transitions between boxes are by neutron capture; vertical transitions are by beta decay and/or isomeric

FIGURE 1. Sample Reaction Grid Used in RIBD

transition. As shown, ground states are allowed to have up to seven source paths and isomeric states up to five. Irradiation and decay time and neutron flux levels are also factors that influence the isotope inventory. RIBD will follow the operating history of the fuel in detail through power level changes and shutdowns. Calculations can be made for fuels containing essentially any combination of U^{235} and Pu^{239} .

It was necessary to generate a fast reactor library for these FTR studies, ⁽³⁾ since RIBD was originally written for thermal reactor analyses. The fast reactor library includes

fast fission yields ⁽⁴⁾ and fast neutron burn-up cross sections. Because of the limited amount of nuclear data of this type that has been obtained, a great deal of extrapolation of data was required. Although significant upgrading of the library is anticipated, it is unlikely that the revisions will invalidate the present preliminary results.

GAMMA ATTENUATION CALCULATIONS

The gamma attenuation calculations were made using the computer program ISOSHLD, ⁽⁵⁾ which includes a number of source-shield geometrical options. The source strength (curies

of each fission product isotope) is calculated within the program using a modified version of RIBD that permits only a single operating period and up to five decay times. The fast reactor library⁽³⁾ was also incorporated into ISOSHLTD for these calculations. A group photon source structure is internally derived from the curie amounts using a library based on current decay schemes.⁽⁶⁾ Buildup factors are calculated by ISOSHLTD based on the number of mean free paths in the shield material located between the source and the detector, where an effective atomic number is used to establish the Taylor buildup coefficients for this material.

In this study, most of the calculations were made using the line source option. This option permits an analytical solution that requires very little computer time. In a few instances the cylindrical source option was employed to more realistically approximate the effect of self-absorption in a fuel assembly. In this option a large number of calculations are required to consider independently many incremental volumes comprising the whole cylinder. The fuel is considered to be homogeneous in the cylindrical calculations, neglecting effects that might be introduced by a pin-like structure. Since self-absorption of the high energy photons that govern the shielding requirements is found to be relatively insignificant, extensive use of the cylindrical option was not felt to be justified for the bulk of this study.

RESULTS OF CALCULATIONS

DECAY HEAT

A knowledge of heat generation within the fuel following a reactor shutdown is necessary to determine cooling

requirements. Immediately following a shutdown, heat generation in the fuel will be dominated by fission events induced by delayed neutrons. After a few minutes, the delayed neutrons will have decayed sufficiently and heat generation will be dominated by the beta-gamma decay of the fission products contained within the fuel. Calculations were made for the reference fuel using the computer code RIBD; the results are presented in Figures 2 and 3. A step function, negative reactivity change was assumed to occur at zero time for these calculations.

INTENSITY IN AIR

Gamma intensity in the vicinity of an unshielded fuel element is of interest in the design of several facilities to assure that equipment will be protected adequately from the radiation. In these calculations, it was assumed the fuel could be approximated by a line source. The grid network of points adjacent to the line source, shown in Figure 4, was used for the calculations. No calculations were made along the imaginary line formed by the extension of the fuel element, since these would require detailed knowledge of the structure of the fuel and fuel end cap. The results of the calculations are presented in Table III. Values at other locations could be estimated by interpolation of these results on the basis of radial distance, as shown in Figure 5.

FUEL HANDLING MACHINE

A shielded fuel handling machine (FHM) for the skewed-tube type reactor being

FIGURE 2. Heat Generation Following Shutdown of Plutonium Fueled Fast Reactor

FIGURE 3. Heat Generation Following Shutdown of Plutonium Fueled Fast Reactor (Percent of Equilibrium Value)

FIGURE 4. Locations for Gamma Intensity Calculations in Air from FTR Fuel Assembly

FIGURE 5. Gamma Intensity as a Function of Radial Distance from FTR Fuel

TABLE III. Calculated Gamma Intensity Near Irradiated FTR Element

Position	Distance cm	Radial Distance from Midpoint (R), cm	Decay Time, days				
			0	1	10	30	60
			Gamma Intensity, R/hr				
A. Midpoint	30	30	8.08×10^7	1.64×10^7	6.95×10^6	3.54×10^6	2.05×10^6
	100	100	1.05×10^6	2.14×10^5	9.06×10^5	4.61×10^4	2.67×10^4
	300	300	1.25×10^6	2.55×10^5	1.08×10^5	5.52×10^4	3.19×10^4
B. End	30	54	5.14×10^7	1.04×10^7	7.86×10^5	4.00×10^5	2.31×10^5
	100	110	9.11×10^6	1.85×10^6	7.86×10^5	4.00×10^5	2.31×10^5
	300	303	1.22×10^6	2.50×10^5	1.06×10^5	5.40×10^4	3.13×10^4
C. 90 cm beyond end	30	138	6.51×10^6	1.32×10^6	5.62×10^5	2.86×10^5	1.66×10^5
	100	168	4.13×10^6	8.41×10^5	3.57×10^5	1.82×10^5	1.05×10^5
	300	329	1.05×10^6	2.14×10^5	9.08×10^4	4.63×10^4	2.68×10^4
D. 180 cm beyond end	30	227	2.27×10^6	4.70×10^5	1.97×10^5	1.00×10^5	5.80×10^4
	100	246	1.92×10^5	3.91×10^5	1.65×10^5	8.43×10^4	4.88×10^4
	300	388	8.11×10^5	1.66×10^5	7.06×10^4	3.59×10^4	2.08×10^4

considered for the FFTF would need to be about 60 ft tall. The spacing between the nozzles limits the diameter of the FHM shielding at the lower end. For these reasons there is an incentive to employ highly efficient gamma shielding material in the FHM cask. Lead is one relatively dense material that is frequently used in gamma shielding applications where space is at a premium. For the FHM, it might be practical or necessary to utilize even more dense material; therefore, calculations have also been made considering both tungsten and depleted uranium as the gamma shielding material.

To compare the different shield materials, a standard cask design was employed. It was assumed that the inside diameter of the cask would be 40 cm. It was further assumed that both the inner and outer walls of the cask would consist of 1 in. thick steel walls. No additional allowance was made for cask structure. It was also assumed that the cylindrical region surrounding the fuel within the cask would be filled with sodium. Calculations were made at a fixed point 100 cm radially

from the midpoint of the line source representing the fuel element. Results of these calculations are presented in Figures 6, 7, and 8.

Calculations were also made using the cylindrical option in which the fuel assembly was considered to have a radius of 5 cm. These calculations indicated that self-absorption of the gamma radiation within the fuel would be expected to reduce the intensity outside a thick cask by less than a factor of two. Actual effect of self-absorption will need to be considered in more detail as the design of the fuel assembly becomes more firm.

EXAMINATION FACILITIES

The examination facilities for irradiated fuel and test specimens form an important part of the FFTF. Because of the radioactivity levels, it will be necessary for the samples to be shielded and cooled at all times. The examination facilities that have been considered would either employ water or an inert gas coolant. Shielding calculations have been made for the following types of examination cells:

FIGURE 6. Gamma Intensity 100 cm from FTR Fuel Assembly (Line Source) as a Function of Depleted Uranium Shield Thickness

FIGURE 7. Gamma Intensity 100 cm from FTR Fuel Assembly (Line Source) as a Function of Lead Shield Thickness

FIGURE 8. Gamma Intensity 100 cm from FTR Fuel Assembly (Line Source) as a Function of Tungsten Shield Thickness

Gas-Cooled Interim Examination Cell

It was assumed that the interim examination cell would provide adequate cooling for decay heat generation rates up to 20 kW, that the fuel assembly would be handled in the vertical position, that it would be positioned at least 120 cm from the cell wall and 300 cm from the ceiling, and that a second element would be permitted 300 cm away from the first on the far side of the cell. The decay heat rate limitation requires that the reference

assembly be permitted to decay for six days prior to entry into the interim cell. Results of the calculations for the fuel assembly near the cell wall are presented in Figure 9. As shown, a slightly greater mass of the more dense concrete would be required to compensate for the reduced wall thickness. Table IV summarizes calculated shield requirements for three different concretes in both the wall and ceiling considering two elements to be in the cell.

FIGURE 9. Gamma Intensity Outside Interim Examination Cell as a Function of Shield Material and Thickness

TABLE IV. Calculated Shield Thickness to Obtain Dose Rates ~ 1.0 mR/hr For Assumed Fuel Test Assemblies and Geometrical Arrangements

Cell	Type of Concrete	Wall		Roof	
		cm	g/cm ²	cm	g/cm ²
Interim Exam Cell	Ordinary	206	515	190	475
	Magnetite	150	525	139	485
	Ferrophosphorus	119	535	110	495
Disassembly Cell	Ordinary	182	455	160	400
	Magnetite	133	465	117	410
	Ferrophosphorus	106	475	93	420
Pin Exam Cell	Ordinary	154	385	134	335
	Magnetite	113	395	99	345
	Ferrophosphorus	90	405	79	355

Gas-Cooled Disassembly Cell

It was assumed that decay heat generation rate in the disassembly cell would be limited to 10 kW in one fuel assembly. Other assumptions were as in the interim examination cell except the assembly would be placed 90 cm from the cell wall. The heat generation rate requires that the reference element be permitted to decay for about 21 days before entry into the cell. Results of the calculations bracketing the desired decay time for an assembly located 90 cm from the cell wall are presented in Figure 10. A summary of wall and ceiling shield requirements are presented in Table IV, for two elements located on opposite sides of the cell.

"Pin" Examination Cell

The assumption was made that the test assembly would be disassembled into components containing no more than 5% of the fission products in a complete reference fuel assembly. To have been disassembled in the cell described above, the fuel would have decayed for at least 21 days. It was further assumed that it would be desirable to move the "pin" or partial assembly to a position 60 cm from the cell wall. As before, it was assumed two samples could be handled at the same time on opposite sides of the cell, vertically, and no closer than 300 cm to the ceiling. Results of the calculations for a single specimen adjacent to a cell wall are shown in Figure 11, while Table IV summarizes the wall and ceiling shielding requirements for two samples in the cell.

FIGURE 10. Gamma Intensity Outside Disassembly Cell as a Function of Shield Material and Thickness

Underwater Examination Cell

An underwater examination cell would permit visual observation of the irradiated specimen with the water acting as both the shield and the coolant. Criticality considerations would limit the amount of material that could be handled underwater, requiring the fuel element to be disassembled before entry into the facility. Thus, the heat generation rate or age of the fuel would be

governed by the capacity of the disassembly cell that is employed. A number of decay times were used in the calculations, where it was assumed the line source would contain 0.5% of the fission product in the reference element. Results are presented in Figure 12 for a position 60 cm above the water surface. It was assumed in the calculations that the line source is parallel to the water surface.

FIGURE 11. Gamma Intensity Outside Pin Examination Cell as a Function of Shield Thickness

FIGURE 12. Calculated Gamma Intensity from Single Fuel Pin as a Function of Water Shielding for Various Decay Times

Calculations were also made to determine the amount of shielding material that would be required in the cell walls. It was assumed the line source was in water parallel to the shielding wall and 60 cm from it. The point of interest was assumed to be 200 cm from the line source. Results of the calculations for three different concretes are given in Figure 13.

NEUTRON SHIELDING CONSIDERATIONS

A brief survey has been made of the neutron sources that might be associated with irradiated FTR fuel elements to determine the magnitude of the neutron shielding problem that might be encountered. The four sources of neutrons that have been considered are: delayed neutrons, spontaneous fission neutrons, neutrons resulting from

FIGURE 13. Calculated Gamma Intensity from Single Fuel Pin as a Function of Concrete Shielding for Various Decay Times

the (α, n) reaction, and neutrons resulting from the (γ, n) reaction. The relative importance of each is discussed briefly.

Delayed Neutrons

Delayed neutrons emitted by fission product isotopes can be neglected insofar as shielding is concerned because of half life considerations. This would not be true if charge-discharge operations were to be performed with the reactor operating. Provision, then would need to be made for shielding the delayed neutrons, or a few minute holdup of the fuel element within a neutron-shielded enclosure would be necessary to permit the neutron activity to decay.

Spontaneous Fission Neutrons

An estimate of the neutron emission rate resulting from spontaneous fission in irradiated

fuel is presented in Table V. Under the assumptions that have been made, there will be about 40,000 g of uranium and plutonium in a fuel assembly. Thus, spontaneous fission would be expected to generate about 10^6 n/sec in a complete fuel assembly. Even with no allowance for the effect of gamma shielding material surrounding the fuel, the dose rate at 100 cm would only be about 1 mrem/hr because of the neutrons from spontaneous fission. It is concluded that no special shielding will be required for this source of neutrons.

Neutrons from (α, n) Reactions

An estimate of the neutron emission rate induced by the (α, n) reaction in the oxygen isotopes contained in a ceramic fuel has been made previously.⁽⁹⁾ This information has been used in Table VI to estimate the neutron emission

TABLE V. Neutrons from Spontaneous Fission in FTR Fuel

Isotope	Spontaneous Fission Rate ⁽⁷⁾ n/g _i -sec ^(a)	Neutrons per Fission ⁽⁸⁾	Assumed Composition ^(b) g _i /g _m	n/g _m -sec
U ²³⁸	7×10^{-3}	2.0	0.7630	0.011
Pu ²³⁸	1.1×10^3	2.2	0.0002	0.48
Pu ²³⁹	1.0×10^{-2}	2.2 ^(c)	0.2100	0.0046
Pu ²⁴⁰	4.6×10^2	2.2	0.0230	23.0
Pu ²⁴²	8.2×10^2	2.2	0.0002	0.36

(a) g_i = grams of isotope

(b) g_m = grams of uranium and plutonium

(c) Estimated from material presented in Reference 8.

TABLE VI. Neutrons from (α, n) Reaction in FTR Fuel

Isotope	Reaction Rate ^(a) n/g _i -sec	Assumed Composition g _i /g _m ^(b)	n/g _m -sec
Pu ²³⁸	1.4 x 10 ⁴	2 x 10 ⁻⁴	2.8
Pu ²³⁹	4.5 x 10 ¹	2.1 x 10 ⁻¹	9.5
Pu ²⁴⁰	1.7 x 10 ²	2.3 x 10 ⁻²	3.9

(a) g_i = grams of isotope
(b) g_m = grams of uranium and plutonium

rate that might be expected from these reactions in irradiated FTR fuel. For a complete fuel assembly, the (α, n) reaction would be expected to generate fewer neutrons than would be created by spontaneous fission. Therefore, it is concluded that no special neutron shielding would be required for this source based on the reference fuel.

Neutrons from (γ, n) Reactions

Photoneutron reactions are possible when the energy of the incident photon exceeds the neutron binding energy within the nucleus. Only four isotopes appear to have photoneutron thresholds sufficiently low to be of interest in reactor applications. These are Be⁹ (1.666 MeV), H² (2.226 MeV), C¹³ (4.9 MeV) and Li⁶ (5.35 MeV). Of these, only beryllium might be present in the FTR fuel in sufficient quantity to permit significant generation of photoneutrons. A brief analysis indicates that special neutron shielding will be required for handling irradiated fuel elements if sufficient beryllium is incorporated in the fuel.

After an 8 hr decay period, the ISOSHLCD calculations indicate that approximately 5×10^{12} photons/cm³ sec

would be released in the fuel at energies in excess of the 1.666 MeV required to induce photoneutron reactions with beryllium. A very preliminary calculation using published geometric relationships⁽¹⁰⁾ assuming a homogeneous cylinder indicates that this source strength would result in a maximum photon flux of about 2×10^{13} photons/cm² sec at the center of a fuel assembly. The beryllium photoneutron reaction cross section has been measured up to 5 MeV;⁽¹¹⁾ although resonances exist, an average value of 5×10^{-28} cm² appears to be a reasonable approximation for photons with energies in excess of 1.666 MeV. A beryllium content of 1 vol% would correspond to approximately 10^{21} atoms Be/cm³, giving a macroscopic photoneutron cross section of 5×10^{-7} cm⁻¹. The maximum neutron generation rate within the fuel would then be approximately 10^7 n/cm³ sec per vol% beryllium. The total neutron generation rate in an assumed assembly would be less than 10^{11} n/sec, so the neutron flux 100 cm from the fuel would be less than 10^6 n/cm² sec, which corresponds to a neutron dose rate less than 10^5 mrem/hr per vol% of contained beryllium. The gamma shielding in a typical FHM cask might be expected to reduce this dose rate to approximately 1000 mrem/hr per vol% of beryllium. Thus, it is concluded that special neutron shielding will be required for FTR elements containing beryllium, and that actual requirements will be quite dependent on the beryllium content. Additional analyses need to be made to determine neutron shield requirements for specific fuel concepts.

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