SAND2020-4068C

Method for calculating delayed gamma-ray response in the ACRR Central Cavity and FREC-II Cavity using MCNP

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Abstract. This document presents the process for a new method developed for the characterization of the delayed gamma-ray radiation fields in pulse reactors like the Annular Core Research Reactor (ACRR) and the Fueled Ring External Cavity (FREC-II). The environments used to test this method in the ACRR Central Cavity and in the FREC-II Cavity were FF, LB44, PLG, CdPoly and FF with rods-down, FF with rods-up, CdPoly with rods-down and CdPoly with rods-up. All environment configurations used the same fission product gamma-ray source energy spectrum. This method required the use of fission sites recorded in the MCNP kode source tapes. A FORTRAN script was written to translate and extract the coordinates for the fission sites. The fission sites were then input it into an MCNP source mode script. Using a MATLAB script, a parametric analysis was performed and determined that 10K fission sites are an appropriate number of coordinates to converge to the correct answer. The method gave excellent results as compared to less sophisticated methods. This method can be applied to other pulse research reactors as well.

1 Introduction

To maintain a high degree of fidelity when performing qualification tests at the Annular Core Research Reactor (ACRR) and the Fueled Ring External Cavity (FREC-II), the irradiation environments in the cavity experiment region must be characterized. Environment characterization includes determining the neutron, prompt gamma-ray, and delayed gamma-ray energy spectra and intensity. To characterize the neutron and gamma-ray environments, both experimental and computational modeling work must be performed. To determine the neutron energy spectrum, activation foils are irradiated at the experiment location. The foil activity results are compared to the results attained from the computed neutron energy

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spectrum. Prompt and delayed gamma-ray energy spectra cannot be measured experimentally, but the intensities can be determined using passive and active dosimetry measurement techniques. For photon energies above the K-edges for photon cross sections (~80 keV), most material response functions fail to show any strong energy-dependent structure that can be used to resolve the spectrum. There is simply no effective way to discriminate energy groups in a gamma-ray fluence within a small volume in a reactor core. Integral dose measurements can be performed and are used as benchmarks for calculated integral responses. These integral measurements have value in validation but are not useful for spectrum adjustment. However, both passive dosimeters (e.g., TLDs) and active dosimeters (e.g., calorimeters) have some neutron and delayed gamma-ray contribution effects that make interpreting the results much more complex [1].

Prompt gamma-rays are identified as those produced, immediately from fission and radiative capture. These gamma-rays are considered to be instantaneously generated along with the neutrons. Using the neutronics Monte Carlo code, MCNP [2], the prompt gamma-ray transport can be calculated in conjunction with the neutrons using kcode mode. This allows for a high-fidelity calculation to be performed and the prompt gamma-ray energy spectrum and intensity to be calculated at the irradiation location of interest simultaneously with the neutrons.

Delayed gamma-rays are identified as those produced from the decay of the fission products and by the activation of materials in the core. For this report, and the previous characterization reports, the activation delayed gamma-rays are not considered. Only the fission product delayed gamma-rays are considered dominant for the analysis. Future work may look at the activation gamma-rays. For delayed gamma-rays, there is not a simple method to calculate the transport using MCNP. In theory, the fission locations that are generated in a kcode problem could be saved and used in a source problem using a delayed gamma energy spectrum. However, MCNP does not allow for a methodology to do this. Previous methods, used in the characterization reports, used a simplistic row-by-row fission tally in a kcode mode as a source distribution. The row-by-row fission tally results were used in a source problem with a defined gamma-ray energy spectrum. This methodology was known to be deficient because it did not account for any axial or internal fuel geometric dependency.

The method proposed in this work was to determine the actual fission sites from the kcode problem, and then use those fission locations in a source problem with a defined gamma emission energy spectrum.

2 Methodology

The neutronics ACRR model used to test this method was developed for the Monte Carlo N-Particle transport code [3]. There were eight environments also referred as “buckets” used for the testing of this method. For the ACRR, the environments used were Free Field (FF) [1], 44-Inch Lead-Boron (LB44) [4], Polyethylene-Lead-Graphite (PLG) [5] and Cadmium-Polyethylene (CdPoly) [6], and for the FREC-II, the environments used were FF with rods-up and rods-down [7] and CdPoly with rods-up and rods-down. The unique MCNP models for the analyses are found in the appendices of the characterization reports [1,3-7]. This paper will expand only on the FF environments for both the ACRR and the FREC-II.

The basic concept for calculating the delayed gamma-ray response for this work was to determine the actual fission sites from an MCNP kcode calculation, and then use those locations to run a separate source problem with a defined gamma emission energy spectrum.

First, an MCNP kcode mode calculation for each of the eight environment configurations was done. This mode creates a binary source tape or SRCTP file which at the end of the MCNP run contains the most updated fission sites. The SRCTP files were translated to readable data and all the x, y, z coordinates were stored in a separate file. From these files, 10,000 fission sites were selected and inserted into another MCNP file but this time using the sdef mode. This mode is the general source definition in MCNP. For this experiment the source was defined by a probability distribution requiring a SI (Source Information) and a SP (Source Probability) cards. The selected 10,000 fission sites all had equal probability of being selected to be the starting point for the delayed gamma calculations.

The fission-product delayed gamma-ray energy spectrum used was for fast U-235 fission in the time interval from 0.2 to 0.5 seconds as given in Engle and Fisher [8] [9]. The total delayed gamma-ray energy released from the fission products used was 6.33 MeV per fission for U-235 fission from ENDF/B-VII.1. This value is about half of the value emitted as prompt gamma-rays, and therefore cannot be ignored as a trivial quantity. The number of delayed gamma-rays emitted per fission was then calculated to be 6.57 delayed photons/fission, using the source delayed gamma-ray energy spectrum for fast U-235 fission.

Figure 1 shows the time-integrated energy fraction from the emission of the delayed gamma-rays on a per fission basis. Although fission products continue to decay over many years, most of the energy (~85%) is released from the shorter-lived products over 1E4 seconds (2.8 hours). The fractional energy released up to 0.1 s is ~1%, ~6.5% up to 1 s, ~23.2% up to 10 s, and ~55% up to 300 s (5 minutes) [10].

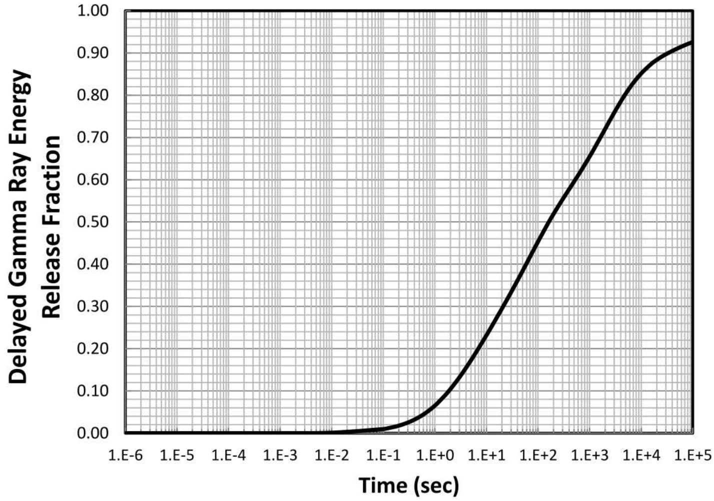

Figure 1. Time Dependent Delayed Gamma-Ray Energy Release Fraction from Fission.

3 Point Determination

When the source tape is generated in the kcode MCNP calculation, it contains the coordinates for the fission sites up to the number of particles per cycle. The determination of how many fission sites to include in the sdef code was an important question that required some parametric analysis. This analysis was approached by using the FF configurations for the ACRR and the FREC-II rods-up and rods-down. Random coordinates were selected in groups of 10, 100, 1K, 10K and 50K and these were plotted in MATLAB. These plots were done in xy, xz and yz coordinates.

Figure 2 shows the XY point results for the coordinates in the ACRR. The groups of 10 and 100 points were too sparse, making it nearly impossible to determine the shape. The group of 1K points had a clearer shape but not quite as defined. Lastly, the 10K and 50K point plots almost looked like they were solid drawn shapes. A similar distribution was found for FREC-II rods-up and rods-down [11].

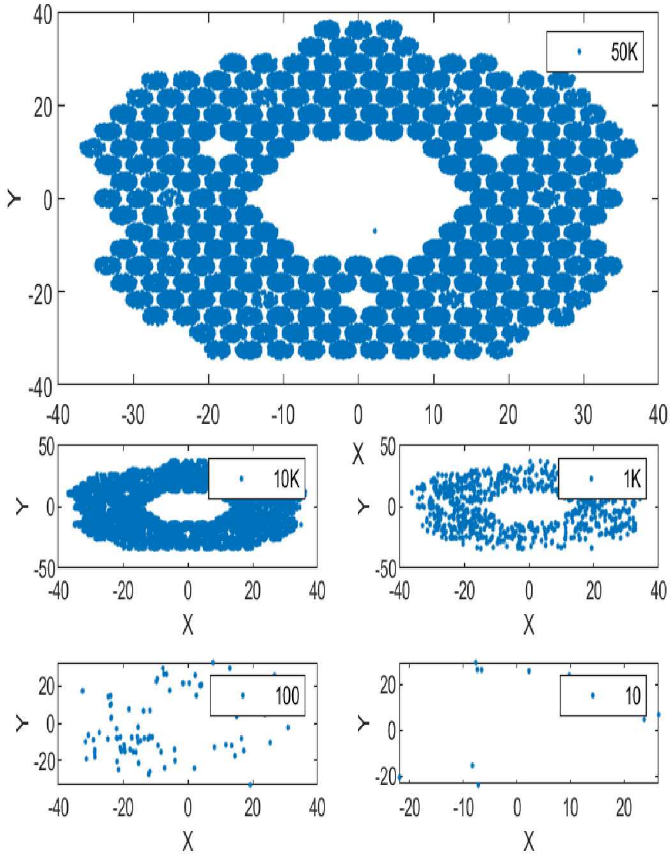

Figure 2. ACRR XY view of the coordinate distribution.

After running the MCNP sdef for all the sets of points, the delayed gamma-ray energy fluence per MJ was plotted and is shown in Figure 3. In this figure it is seen that the 1K, 10K and the 50K results are all converging to the same curve and are somewhat higher in magnitude than the plot generated for the ACRR FF Characterization Report [1]. Since either 10K or 50K seemed acceptable for this method, 10K points was selected for the ease of managing data. After reviewing these plots, it was concluded that the same number of points would work for both the ACRR and the FREC-II since they had similar results for all the groups. Plots generated for the FREC-II can be seen in the full report [11].

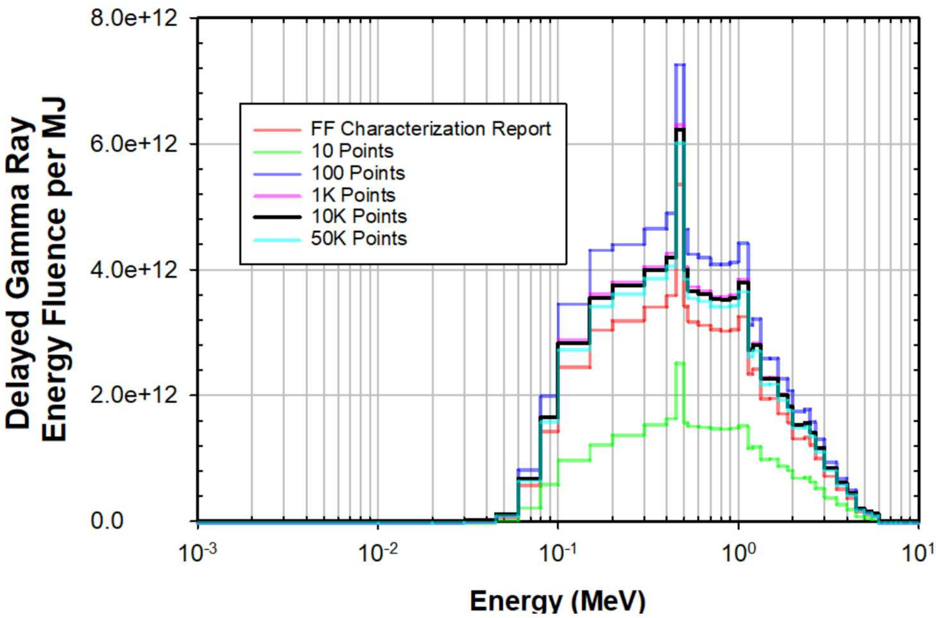

Figure 3. ACRR Free Field plots versus previous report plot.

4 Delayed Gamma-Ray Spectrum Characterization

This report only expands on the ACRR and FREC-II FF but during research the FF, LB44, PLG and the CdPoly environments were investigated for the ACRR and the FF rods-down, FF rods-up, the CdPoly rods-down and CdPoly rods-up environments were investigated for the FREC-II; the results for all environments can be seen in the full report [11].

The MCNP 48-energy group delayed gamma-ray fluence was plotted on a linear y-axis and the energy was plotted on a logarithmic x-axis shown in Figure 4 for the ACRR and Figure 5 FREC-II. The units on the y-axis are in energy fluence equal to $\frac{Ed\Phi}{dE}\left(\frac{MeV}{MeV\ cm^2 MJ}\right)$ and the units in the x-axis are in MeV. The delayed gamma-ray energy fluence has a prominent peak at ~0.5 MeV. This peak represents the electron-positron annihilation photon energy (0.511 MeV). The ACRR plot shows a higher energy and the FREC-II plot shows a slightly lower energy compared to the previous characterization reports.

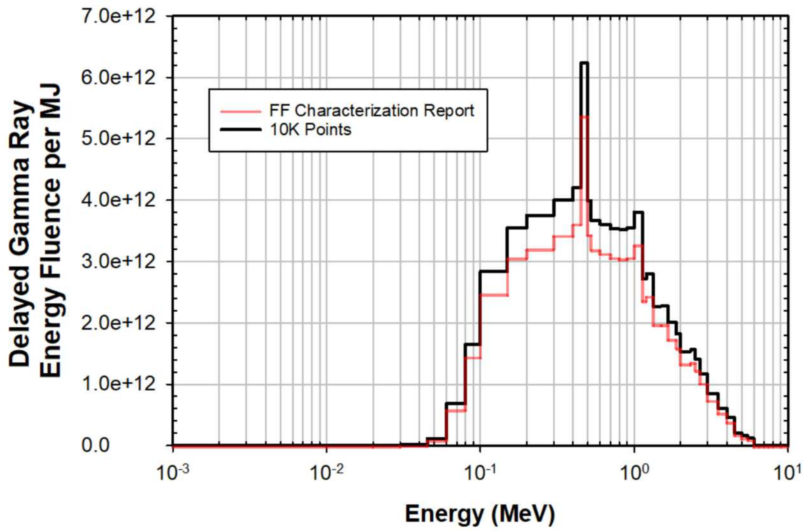

Figure 4. ACRR Free Field 10K plot versus previous report plot.

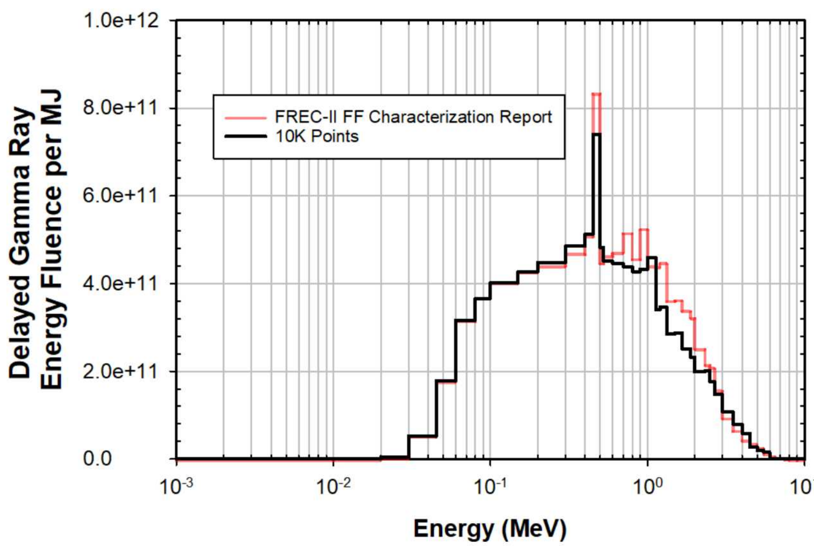

Figure 5. FREC-II rods-up Free Field 10K plot versus previous report plot.

Different tallies were also run to calculate the Si dose, the carbon dose and the TLD Dose for each individual environment, these results and several integral metrics and conversion factors for the ACRR are shown in Table 1 and for the FREC-II in Table 2. These integral metrics may be useful to the experimenter. These values were calculated using the 48-energy group gamma-ray spectrum. For a complete set of tables and conversion methods, refer to the full Sandia report [11].

Table 1. ACRR Metrics.

ACRR / Metric	FF	LB44	PLG	CdPoly
Average delayed gamma-ray (dγ) energy (MeV)	0.863	1.528	1.120	0.876
Fluence Conversion ([dγ/cm^2]/MJ)	1.073E+13	7.114E+11	4.583E+12	9.530E+12
Total (Ionizing) Si Dose (rad[Si]/[dγ/cm^2])	3.538E-10	5.911E-10	4.499E-10	3.599E-10
Total (Ionizing) Carbon Dose (rad[C]/[dγ/cm^2])	3.434E-10	5.772E-10	4.376E-10	3.482E-10
Total (Ionizing) CaF_2:Mn (TLD) Dose (rad[CaF_2:Mn]/[dγ/cm^2])	3.517E-10	5.809E-10	4.452E-10	3.589E-10

Table 2. FREC-II Metrics.

FREC-II / Metric	FF DN	FF UP	CdPoly DN	CdPoly UP
Average delayed gamma-ray (dγ) energy (MeV)	0.770	0.776	0.846	0.840
Fluence Conversion ([dγ/cm^2]/MJ)	1.425E+12	1.510E+12	9.204E+11	9.113E+11
Total (Ionizing) Si Dose (rad[Si]/[dγ/cm^2])	3.213E-10	3.236E-10	3.466E-10	3.444E-10
Total (Ionizing) Carbon Dose (rad[C]/[dγ/cm^2])	3.050E-10	3.074E-10	3.342E-10	3.323E-10
Total (Ionizing) CaF_2:Mn (TLD) Dose (rad[CaF_2:Mn]/[dγ/cm^2])	3.272E-10	3.292E-10	3.469E-10	3.447E-10

5 Conclusion

The method for calculating delayed gamma responses in characterized environments using this technique was successful. After calculating the decay gamma-ray fluence using MCNP for eight different environment configurations, the results showed that this procedure minimizes the assumptions made for the distribution function that was previously done. This method accounts for axial and internal fuel geometric dependencies by improving the sources being used, resulting in more accurate spectral information. This new method could become the formal approach to calculating delayed gamma-ray responses more accurately for any pulse reactor.

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