Method for calculating delayed gamma-ray response in the ACRR Central 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). The environments used to test this method in the ACRR Central Cavity were various standard irradiation configurations that used various amounts of moderators and shielding. All environment configurations used the same fission product gamma-ray source energy spectrum. This method required the use of fission sites recorded in the Monte Carlo N-Particle (MCNP) KCODE source tape. A FORTRAN script was written to translate and extract the coordinates for the fission sites. The fission sites were then input into an MCNP source mode input file. 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 previous 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), 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 spectrum. Prompt and delayed gamma-ray energy spectra

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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, radiative capture, or inelastic scattering. These gamma-rays are considered to be instantaneously generated along with the neutrons. Using the particle transport Monte Carlo N-Particle (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 fission produced beta particles and their associated bremsstrahlung were not analyzed. The activation delayed gamma-rays were also not considered in this analysis; only fission delayed gamma-rays were considered in this study. Future work may look at the fission product beta particles and 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 directly. 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 MCNP [3]. This paper expands only on the Free Field (FF) environment for the ACRR, but there were eight total environments used for the testing of this method [4]. For the ACRR, the environments tested were FF [1], 44-Inch Lead-Boron (LB44) [5], Polyethylene-Lead-Graphite (PLG) [6] and Cadmium-Polyethylene (CdPoly) [7], and for the FREC-II, the environments tested were FF with rods-up and rods-down [8] 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].

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) card. 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 [9] [10]. This spectrum was used since the ACRR is an epithermal system and approximately 87% of the neutron population is above 1 eV and 65% above 1 keV [1]. 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 [11]. 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) [12].

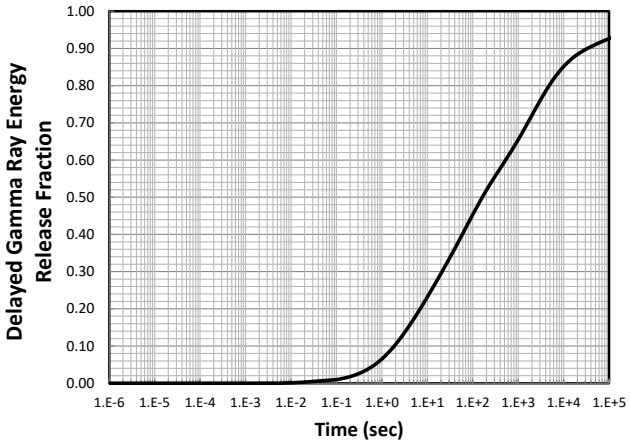

Figure 1. Time Dependent Delayed Gamma-Ray Energy Release Fraction from Fission.

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. Random coordinates were selected in groups of 10, 100, 1K, 10K, 50K and 100K, 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 FF. The groups of 10 and 100 points were too sparse, making it nearly impossible to determine the shape of the reactor. The group of 1K points had a clearer shape but not quite as defined. Lastly, the 10K, 50K, and 100K point plots almost looked like they were solid drawn shapes. A similar distribution was found for FREC-II rods-up and rods-down [4].

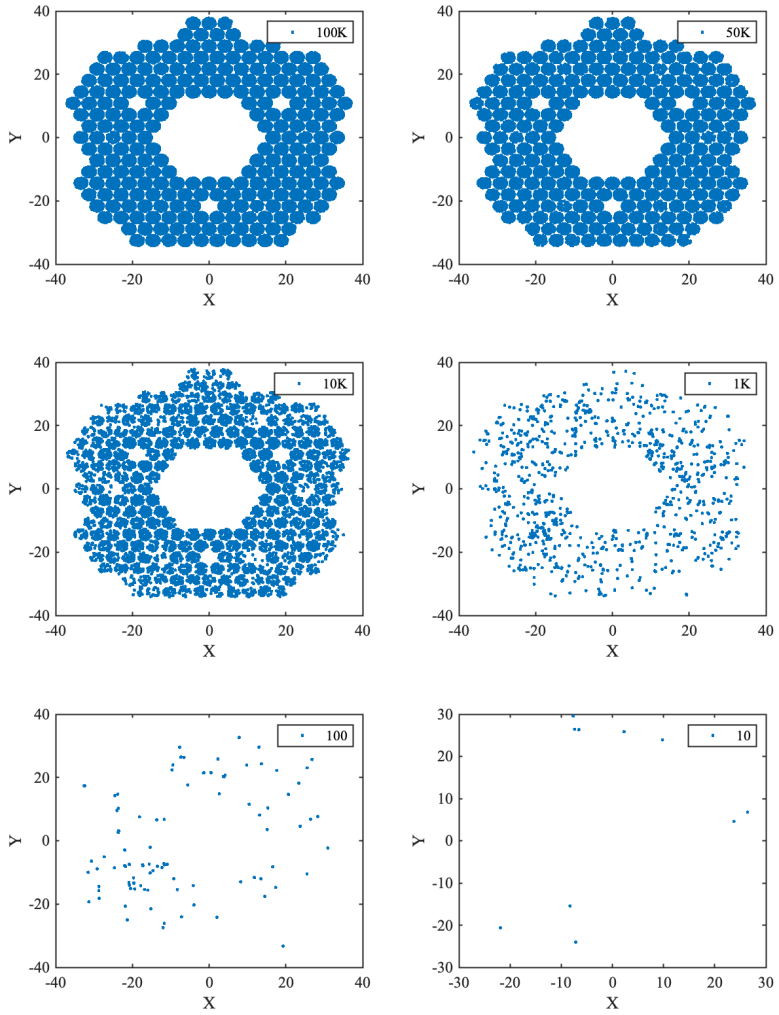

Figure 2. ACRR XY view of the coordinate distribution.

After running the MCNP SDEF for all the sets of points, the 48-energy group delayed gamma-ray fluence per MJ was plotted and is shown in Figure 3. These spectra were calculated using 5 billion particles resulting in an average total uncertainty of 0.05%. In this figure it is seen that the 1K, 10K, 50K, and the 100K results are all converging to the same curve and are higher in magnitude than the plot generated for the ACRR FF Characterization Report [1]. At the 0.5 MeV peak, the 10 Point was 112% below the characterized value, the 100 Points was 26% above the characterized value, 1K was 15% above the characterized value, 10K points was 14% above the characterized value, 50K points was 11% above the characterized value, and 100K points was 13% above the characterized value.

Since either 10K, 50K, or 100K seemed acceptable for this method, 10K points was selected for the ease of managing data. 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 [4].

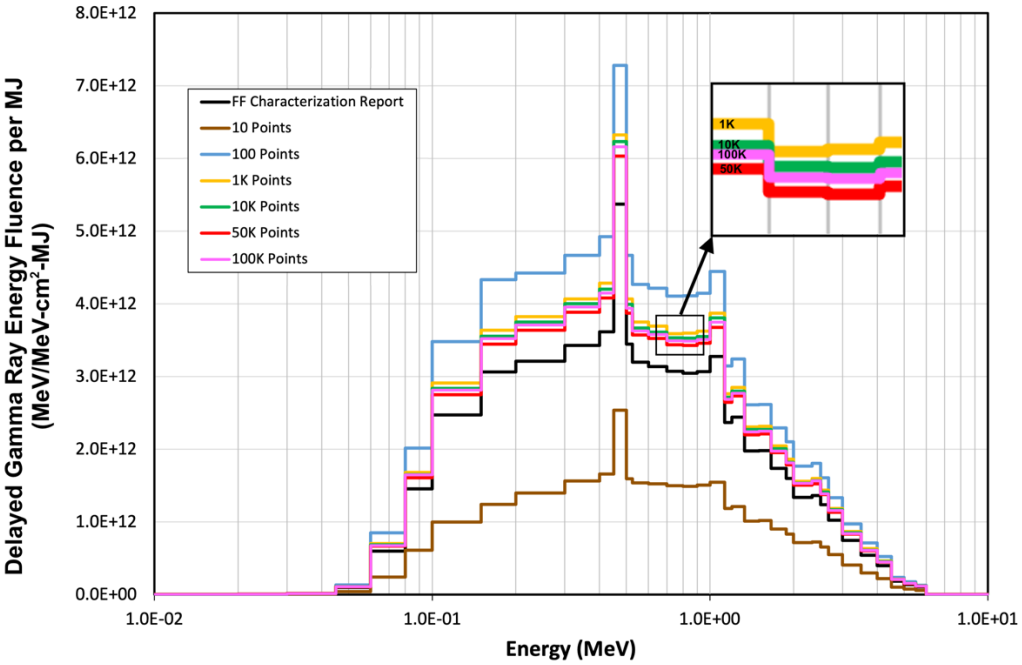

Figure 3. ACRR FF MCNP plots versus ACRR FF Characterization Report plot.

4 Delayed Gamma-Ray Spectrum Characterization

The MCNP 48-energy group delayed gamma-ray fluence is shown in Figure 4 for the FF ACRR using the 10K points. 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 FF 10K plot shows a 14% higher intensity compared to the previous characterization report. This means the previous method was underestimating the delayed gammas generated due to fission in the ACRR, thus misrepresenting the delayed gamma spectra.

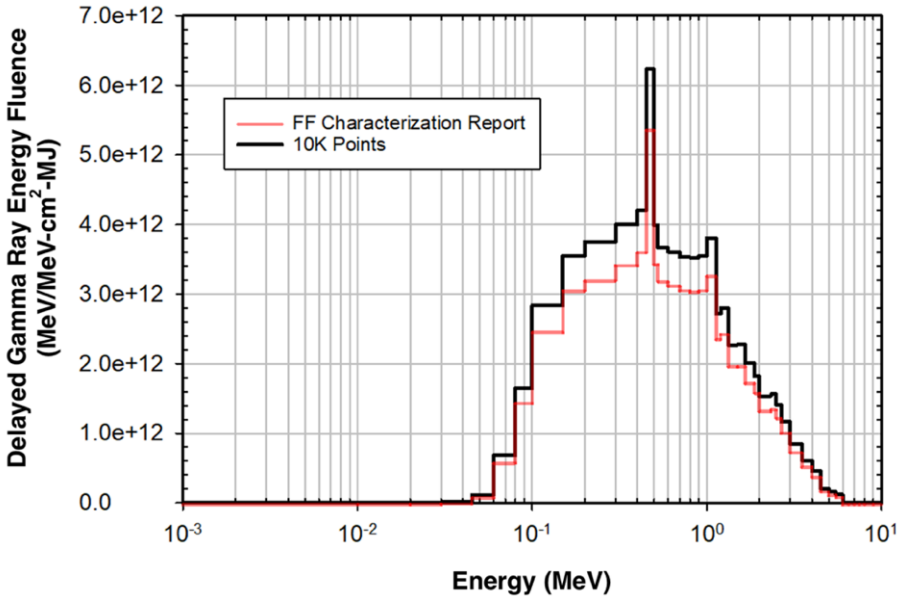

Figure 4. ACRR Free Field 10K plot versus previous report plot.

The MCNP ACRR models are run to calculate different tallies for each individual environment using the 48-energy group gamma-ray spectrum. These results and several integral metrics and conversion factors may be useful to the experimenter. Among the metrics calculated are Average delayed gamma-ray (dγ) energy (MeV), Fluence Conversion ([dγ/cm^2]/MJ), Total (Ionizing) Si Dose (rad[Si]/[dγ/cm^2]), Total (Ionizing) Carbon Dose (rad[C]/[dγ/cm^2]), and Total (Ionizing) CaF_2:Mn (TLD) Dose (rad[CaF_2:Mn]/[dγ/cm^2]). For a complete set of tables for all environments and conversion methods, refer to the full Sandia report [4].

Table 1. ACRR Metrics.

ACRR FF / Metric	10	100	1K	10K	50K	100K
Average delayed gamma-ray (dγ) energy (MeV)	0.937	0.849	0.862	0.863	0.865	0.863
Fluence Conversion ([dγ/cm^2]/MJ)	4.30E+12	1.27E+13	1.09E+13	1.07E+13	1.04E+13	1.06E+13
Total (Ionizing) Si Dose (rad[Si]/[dγ/cm^2])	3.80E-10	3.49E-10	3.54E-10	3.54E-10	3.54E-10	3.57E-10
Total (Ionizing) Carbon Dose (rad[C]/[dγ/cm^2])	3.69E-10	3.38E-10	3.43E-10	3.43E-10	3.44E-10	3.47E-10
Total (Ionizing) CaF_2:Mn (TLD) Dose (rad[CaF_2:Mn]/[dγ/cm^2])	3.77E-10	3.47E-10	3.51E-10	3.52E-10	3.52E-10	3.55E-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. Future work may look at the fission product beta particles and activation gamma-rays, and MCNP card investigations to assess if a more established method can be utilized to accomplish the same results without assumptions being made on the fission sources.

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