# Radiation characterization summary for the WSMR fast burst reactor environment at the 6-inch location

*Danielle* Redhouse[1,*], *Edward S.* Lum[2], and *Johnathon* Koglin[3]

with *Edward J.* Parma[1], *Curtis D.* Peters[1], *Mikhail* Finko[3], *Jesse M.* Roebuck[1], *David W.* Vehar[1], *Frank* Sage, *Andrew M.* Tonigan[1], *Ryan* Mulcahy[1], *Thomas A.* Ball[1], *Elliott* Pelfrey[1], *Melissa* Moreno[1], *Karissa* Currie[1], and *Patrick J.* Griffin[1]

[1]Sandia National Laboratories, Albuquerque, NM, USA
[2]Los Alamos National Laboratory, Los Alamos, NM, USA
[3]Lawrence Livermore National Laboratory, Livermore, CA, USA

**Abstract.** The characterization of the neutron, prompt gamma-ray, and delayed gamma-ray radiation fields for the White Sands Missile Range (WSMR) Fast Burst Reactor, also known as molybdenum-alloy Godiva (Molly-G) has been assessed at the 6-inch irradiation location. The neutron energy spectra, uncertainties, and common radiation metrics are presented. Code-dependent recommended constants are given to facilitate the conversion of various dosimetry readings into radiation metrics desired by experimenters. The Molly-G core was designed and configured similarly to Godiva II, as an unreflected, unmoderated, cylindrical annulus of uranium-molybdenum-alloy fuel with molybdenum loading of 10%. At the 6-inch position, the axial fluence maximum is about $2.4\times10^{13}$ n/cm$^2$ per MJ of reactor energy; about 0.1% of the neutron fluence is below 1 keV and 96% is above 100 keV. The 1-MeV Damage-Equivalent Silicon (DES) fluence is estimated at $2.2\times10^{13}$ n/cm$^2$ per MJ of reactor energy. The prompt gamma-ray dose is roughly 2.5E+03 rad(Si) per MJ and the delayed gamma-ray dose is about 1.3E+03 rad(Si) per MJ.

[*] Corresponding author: drredho@sandia.gov

# 1 Introduction

In the late 1950's, Los Alamos National Laboratory (LANL) designed and developed the Moly-Godiva reactor during the Rover program period at the Pajarito Site. The core was designed as an unreflected, unmoderated, cylindrical annulus of uranium-molybdenum alloy. The assembly was moved to the U.S. Army's Survivability, Vulnerability, and Testing Directorate (SVAD) at White Sands Missile Range (WSMR) in June of 1964 [7]. The Moly-Godiva reactor was renamed the WSMR Fast Burst Reactor (FBR), nicknamed Molly-G. Molly-G was designed and configured similarly to Godiva-II but had a molybdenum loading of 10%. Molly-G was developed to conduct neutronic studies by providing an intense neutron burst over a narrow pulse width, with high dose capabilities. Initial criticality of the Molly-G at WSMR was achieved on July 7$^{th}$, 1964 [18].

Characterization of the neutron and gamma-ray environments in WSMR FBR include the 6-inch (152.4 mm) and 24-inch (609.6 mm) leakage free-field environment external to the FBR core radially spaced on y-plate at the axial fluence maximum and includes both accurate neutronic modeling and experimental efforts [1]. The neutronics model for the WSMR FBR was redundantly modeled in MCNP by both LANL and Sandia National Laboratories (SNL) and modeled by Lawrence Livermore National Laboratory (LLNL) in their proprietary neutronic code MERCURY. Due to the proprietary nature of MERCURY, these results will not be enclosed in this report. Experimental observations, including passive and active dosimetry, are important to determine the accuracy of the model. Presented are the energy dependent neutron fluence, some of the conversion constants for radiation metrics, the axial and radial neutron fluence profiles, and the time-dependent responses for different pulse sizes.

# 2 Environment Description

WSMR FBR was designed by Dr. Thomas F. Whimett and Dr. Gordon E. Hansen to perform neutron studies in a burst environment [7]. WSMR FBR can operate in a steady-state mode with the operating power level limited to 8kW. In pulse mode, a maximum pulse size of 2.0 MJ ($\Delta T$ of 250°C) with a full-width half-maximum (FWHM) of 50 μs can be attained [16].

The WSMR FBR is fueled by six uranium-molybdenum solid metal ring elements. The fuel uses uranium enriched to 93.0 weight percent U-235 and 10.0 weight percent molybdenum. The fuel elements are clad in 6061-aluminum at 0.5 cm thick. Each fuel rings dimensions are an internal diameter of 2 inches (5.08 cm) and an external diameter 4 inches (10.18 cm), with an individual height of 1.28 inches (3.25 cm). WSMR FBR is controlled by two fuel control rods, one fuel burst rod, and a fueled safety block. The control rods are positioned axially in 5 fuel elements, while the burst rod is positioned axially in 4 fuel elements in the normal core configuration [7][16][17].

Historically, the WSMR FBR has been used for a wide variety of experiment campaigns including those for space applications and radiation effects sciences. The main irradiation location for the WSMR FBR is within the reactor room on an elevated table. The table features a y-plate with standard reproduceable test positions in 1/5-inch increments along three separate 'legs'. Figure 1 shows the WSMR FBR along with its irradiation table.

Figure 2 shows the y-plate with three legs that extend out to the edges of the test table; these include the 'A-Leg', 'B-Leg', and 'C-Leg'. By convention, 'A-Leg' or 'Long Leg' extends from the core west, ‘B-Leg’ extends N-NE (~60° angle), and ‘C-Leg’ extends S-SE (~60° angle). Experiments are classically placed at the axial fluence maximum outside of the core, roughly 3 inches from the bottom of the core. The WSMR FBR is situated in an open-air cell approximately 48 ft feet (30.48 cm) by 48 feet (30.48 cm), with a 23 ft height to the ceiling. WSMR FBR features a new state-of-the-art cooling system allowing continued safe pulsing operations. Cooling is done in both steady-state mode and after pulses using chilled/dried compressed air.

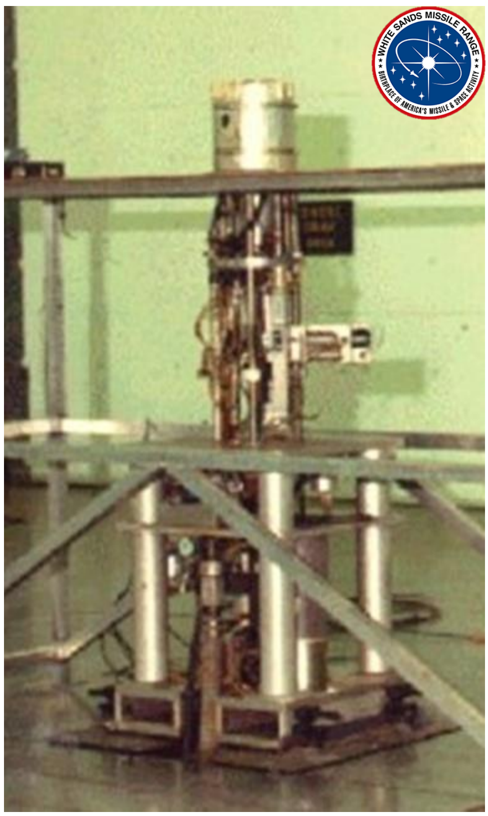


**Figure 1.** WSMR “Molly-G” Fast Burst Reactor and test table.

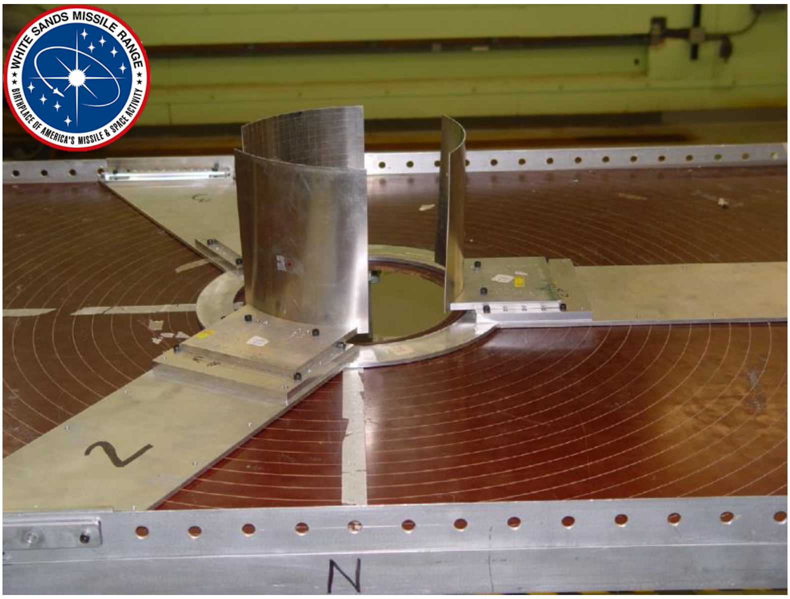


**Figure 2.** WSMR test table with the y-plate installed. Curved dosimetry plates are shown at the 6-inch position on the 'A-Leg', B-Leg, and 'C-Leg' of the y-plate.

## 3 Neutronic Models

The MCNP model of the WSMR FBR is shown in XY and XZ plane views in Figure 3. The MCNP model includes the uranium-molybdenum fuel elements, safety block, control rods, and burst rod. The control rods and burst rods can be vertically adjusted in the model to any position desired. The model is typically run at standard temperature and pressure, using room-temperature cross sections (293.6 K) from ENDF/B-VIII.0. Various cross section temperatures can be modeled if desired [11].

Neutron energy spectra and fluence per fissions were calculated using a 6-cm diameter tally sphere located in-core and ex-core on the long leg (i.e., A Leg) level with the fluence maximum centerline, and in mirror positions on the opposite side of the reactor from the long leg. Calculations were performed using the MCNP k-code mode for the neutron fluence and metrics. For the SNL MCNP neutronic model of the WSMR FBR, the system $k_{eff}$ was noted as 1.00019 ± 0.00001. This value coincides with a system configuration of the burst rod at full insertion, with control rod 1 inserted at 2838 mils and control rod 2 inserted at 2800 mils.

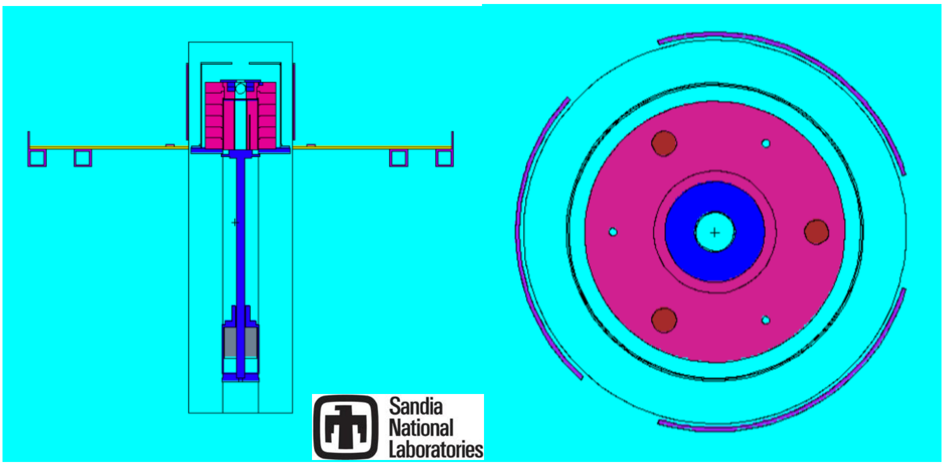


**Figure 3.** SNL MCNP Model views showing the above and side profile of the WSMR Fast Burst Reactor.

The neutron energy spectrum was calculated for 640-group and 89-group energy structures using the MCNP model presented in Section 2 for the 6-cm diameter tally sphere [4]. MCNP6.2 version was used with the ENDF/B-VIII.0 cross sections. Room temperature cross sections were used for the calculations [3][11][15]. The model was run in the k-code mode using both neutrons and photons. This arrangement allowed for both the neutron energy spectra and the prompt gamma-ray energy spectrum to be generated in a single run. The gamma-ray spectrum is not presented here. The results generated in the tally sphere were in units of neutron fluence per source neutron. These results were then converted to fluence per fission. The neutron fluence results are then converted from fluence per fission to fluence per MJ of reactor power using 192.4 MeV per fission. This value represents the fission fragment, neutron, prompt gamma-ray, capture gamma-ray, and delayed gamma-ray energy deposition in the reactor core per fission event. These energy deposition values were calculated using MCNP. The 89-group neutron energy spectrum result was used as the trial function for LSL-M3, a least-squares unfolding code [9][12][14].

Figure 5 and Figure 6 show the MCNP-generated 640-group (black) and 89-group (grey) neutron energy fluence at the 6-inch position on a linear and logarithmic y-axis, respectively. The units on the y-axis are in lethargy fluence or energy fluence, equal to $E d\phi/dE$ (MeV/MeV-cm$^2$-MJ). With the energy fluence represented linearly on the y-axis and the neutron energy on the x-axis represented logarithmically, the area under the curve represents the total neutron fluence. This representation allows for the best visual depiction of the fluence over the complete neutron energy range.

The energy fluence is found to peak near 1 MeV. The thermal peak occurs at ~0.07 eV (7.0E-8 MeV). The results for the 640-group and 89-group neutron energy fluence are in very close agreement, as expected. The 640-group energy fluence calculation shows higher fidelity in the spectrum, which is more notably observed in the energy range of 1E-4 to 1 MeV. This structure is considered to be real and not an artifact of the code or cross section set [10][13]. It is caused by resonances in the cross sections, especially from the oxygen elastic scattering cross section. The calculated standard deviation for each energy bin in the 640-group energy fluence in the energy range from 0.005 eV (5.0E-9 MeV) to 6 MeV is less than 1%. A similar but less resolved structure is also seen in the 89-group energy fluence. The 89-group energy structure is the energy grouping used in the NuGET code [5].

## 4 Experiments

The results of the MCNP and MERCURY calculations of the WSMR environment represent a good initial representation for the neutron spectrum characterization. However, a considerable uncertainty in the results can exist due to model representation uncertainty (geometry, density, and composition) and uncertainty in the transport cross sections. An improved spectrum can be obtained by combining this initial trial spectrum with measured integral values that correspond to the reactions rates from high-fidelity passive dosimetry reactions. For this work, the LSL-M3 code was used to generate an 89-group neutron energy spectrum that is used in the NuGET code [5].

The selection of passive dosimetry foils and activation reactions has been studied and evaluated over many years. A summary of the work can be found in noted references of this manuscript [1][6][10][11][13]. No complete and perfect set of activation reactions exists that allows the neutron fluence energy spectrum to be calculated by dosimetry alone. However, there are enough reactions to cover the relevant energy range with high-fidelity dosimetry cross sections to allow for adjusted neutron fluence results to be generated with a quantified accuracy. The passive dosimetry foils and the associated neutron activation reactions used to perform the neutron fluence characterization for the WSMR 6-inch (152.4 mm) and 24-inch (609.6 mm) environments are shown in Table 4 and Table 5.

The foils and activation reactions chosen for the analysis represent expert judgment and references to previous work [1][10][13]. A complete set of dosimetry foils and reaction data will vary for a given neutron environment being characterized [2]. Typically, neutron activation resulting in the emission of protons (n,p), neutrons (n,2n), (n,n'), or alpha particles (n,$\alpha$) represent high-neutron-energy reactions of 1 MeV or greater. Neutron activation resulting in prompt gamma-ray emission from radiative capture (n,$\gamma$) or fission reactions determine the shape of the thermal and epithermal region of the neutron spectrum. Covering foils with cadmium (Cd) and/or boron (B) can allow for resonances above the associated cutoff energies to become more prominent, allowing for additional integral quantities to be included in the analysis. Typically, only thermal activation or fission foils are covered in Cd or B.

A total of 23 different foil types, resulting in 35 different reactions, were irradiated [2]. Three fission foils (U-235, U-238, and Np-237) were irradiated individually in a bare or cadmium cup configuration in a single 24 MJ steady-state operation. An historic Pu-239 foil activity was used from an older dosimetry set characterized and run with a nickel foil for tracking purposes. This historic activity was nickel corrected to match the 24 MJ steady-state operation conducted in 2022 [2].

The free-field environment was used for all the foil irradiations noted previously. A high variation in the flux is known to exist radially and axially within the reactor cell. However, a test region was identified where the neutron fluence varies ± 5%, this region is the highlighted (yellow) section on the curved plates shown in Figure 4. Additional uncertainty in the analysis was included to account for possible variation due to geometry effects.

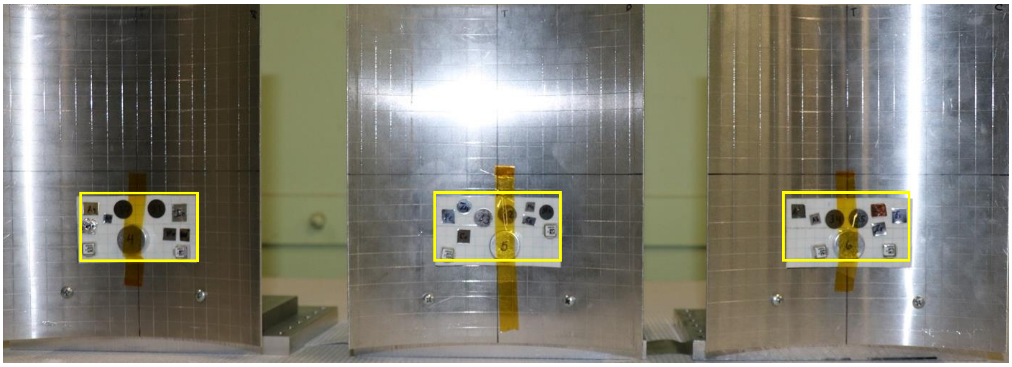

**Figure 4.** Typical Dosimetry Foils – Characterization Test Setup.

## 5. Spectrum Characterization Results

The neutron energy spectrum is first calculated using MCNP and then a least-squares spectrum or genetic algorithm adjustment is performed using passive neutron activation dosimetry measurements to produce a “characterized” neutron spectrum. The resulting spectrum can then be considered “characterized” in that it quantifies the “true” neutron-fluence energy spectrum with a stated accuracy, including energy dependent uncertainties and a covariance matrix. The process for determining a characterized neutron spectrum is to first generate an *a priori* 640-group and 89-group neutron energy trial spectra from the MCNP models presented in Section 2 at the 6-cm diameter tally sphere location. If the least-squares spectrum is found to have inconsistent inputs when comparing the MCNP *a priori* trial spectrum and the measured dosimetry results, as determined by a measure of the chi-squared per degree of freedom ($\chi^2$/dof) in the spectrum adjustment, the dosimetry measurements would be reexamined and/or the MCNP model would be reexamined and modified or modeled with greater fidelity. The results of the MCNP calculations of the WSMR environment represent a good initial representation for the neutron spectrum characterization.

Also required in the analysis was an initial uncertainty estimate in the neutron spectrum as a function of energy, an initial energy-dependent correlation matrix, the energy dependent self-shielding factors, and the dosimetry cross section library that also included uncertainties and covariance matrices [15]. In addition to the counting uncertainty, an additional 2% uncertainty was included for the foils to address uncertainty contributions due to positioning and possible geometrical effects in the central region of the core. The output was in the 89-group NuGET format described earlier.

The resulting values for $\chi^2$/dof are 0.74 for the 6-inch testing position, which represents a highly acceptable value. Figure 5 and Figure 6 for the 6-inch location show the same representation of the neutron energy spectra described in the Experimentation section, with the additional 89-group adjusted spectrum result from the LSL analysis (blue). The adjusted neutron fluence in Figure 6 shows very close agreement to that calculated using the MCNP model.

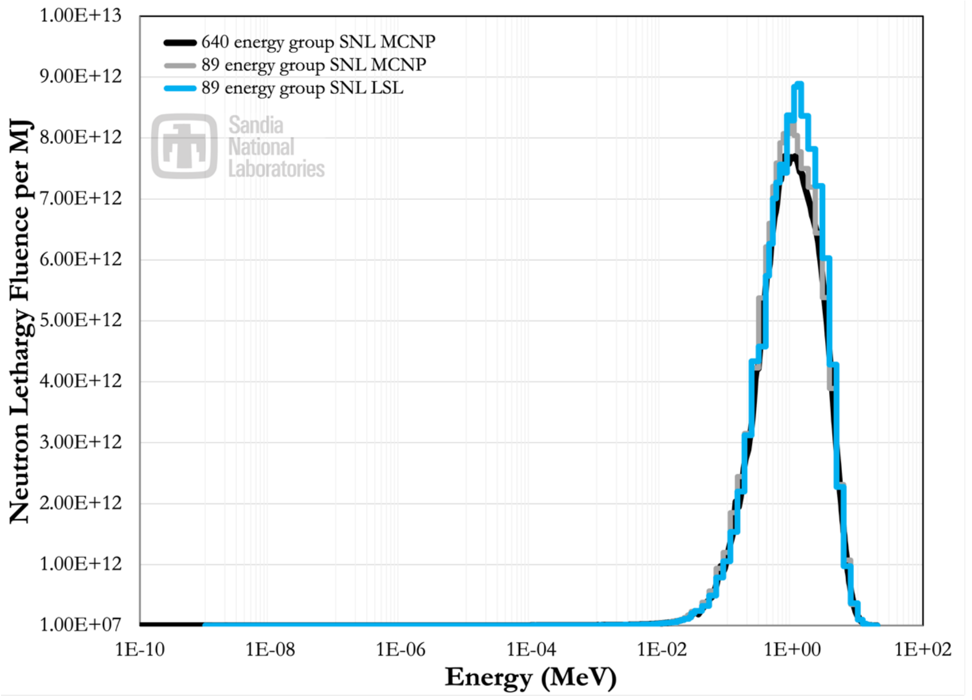


**Figure 5.** WSMR 6-inch – SNL LSL Adjusted 89-Group Neutron Lethargy Fluence Energy Spectrum Compared to the MCNP Calculated Results (linear–log).

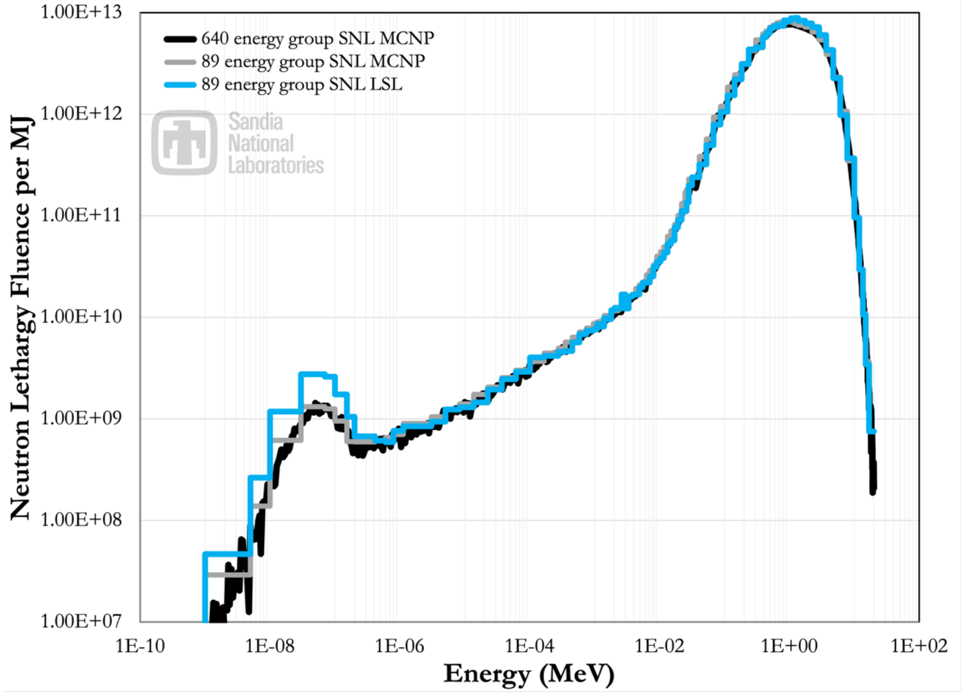


**Figure 6.** WSMR 6-inch – SNL LSL Adjusted 89-Group Neutron Lethargy Fluence Energy Spectrum Compared to the MCNP Calculated Results (log–log).

The results for some integral metrics and conversion factors are shown in Table 1. The total neutron fluence is often normalized to 1.00 and the other values for fluence are in reference to this value. These values were calculated as part of the LSL analysis. Conversion values to translate to n/cm$^2$ are given for fissions in the reactor, MJ of reactor energy, and $^{58}$Ni(n,p)$^{58}$Co activity at the characterized location on the Y-plate. It should be noted that portions of the energy spectrum are highly correlated, so the uncertainty in the integral metric can be much less than the average uncertainty [9][14].

**Table 1.** WSMR 6-inch Location – SNL Integral Neutron Spectrum Metrics and Associated Uncertainties.

| Metric | Integral Response | Standard Deviation (%) |
|---|---|---|
| Total Neutron Fluence<br>Average Neutron Energy = 1.445 MeV | 1.00 | --- |
| Fluence > 3 MeV | 0.137 | 5.2 |
| Fluence > 1 MeV | 0.513 | 3.6 |
| Fluence > 100 keV | 0.968 | 0.58 |
| Fluence > 10 keV | 0.997 | 0.46 |
| Fluence < 1 keV | 0.001 | 10.7 |
| Fluence < 1 eV | 0.0002 | 31.6 |
| Fluence 1-MeV(Si) Eqv. E722-19<br>(Ref. 1-MeV value = 95 MeV-mb) | 83.8 MeV-mb<br>0.897 | 1.1 |
| Total Fluence Conversion ([n/cm$^2$]/fission) | 6.791E-04 | 0.3 |
| Total Fluence Conversion ([n/cm$^2$]/MJ) | 2.403E+13 | 0.5 |
| Total Neutron Silicon Dose<br>(rad[Si]/MJ) | 1.677E+03 | --- |
| Ionizing Si Dose (rad[Si]/[n/cm$^2$]) | 9.951E+02 | --- |
| Percent Neutron Si Dose Ionizing (%) | 59.3 | --- |
| Total Prompt Gamma-Ray Ionizing Silicon Dose<br>(rad[Si]/MJ) | 9.476E+02 | |
| Total Delayed Gamma-Ray Ionizing Silicon Dose<br>(rad[Si]/MJ) | 2.162E+03 | |
| Total Ionizing Dose (rad[Si]/MJ) | 4.105E+03 | |

Figure 7 shows the results for the sulfur radial neutron fluence profile for the free-field environment at an axial position of 3 inches from the bottom of the test table for the sulfur pellets arranged perpendicular from the core. The nickel foils and sulfur tablets were arranged on an extended piece of tape that spanned from the 6-inch position to end of the test table on all three legs of the y-plate. The solid line represents the average value for all of the pellets measured on the 'B-Leg' at standard 1-inch increments. The counting uncertainties are within each data point symbol. The results show that for the nickel and sulfur activation reactions, there is no significant variation in the fast neutron fluence between the legs on the y-plate.

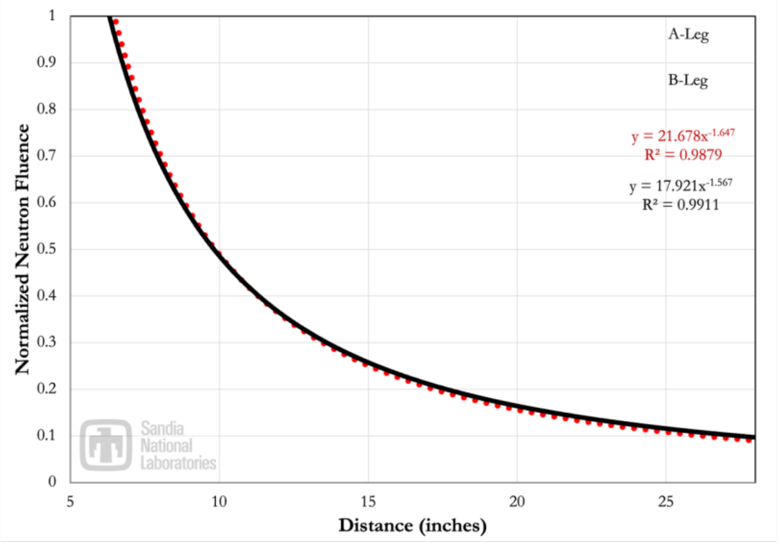


**Figure 7.** A-Leg (Red Dotted) and B-Leg (Black Solid) Nickel and Sulfur Normalized Radial Neutron Fluence Profiles, referenced from the center of the FBR.

## 7 Conclusions

This report presents the characterized neutron radiation environments for the free-field environment for the WSMR FBR, called Molly-G [1]. The characterized 6-inch location of the y-plate on the test table is presented. 640-group and 89-group neutron energy spectra were calculated using MCNP, utilizing a newly created high-fidelity model of WSMR FBR by SNL. Each neutron spectrum was adjusted to align more closely with neutron-activation dosimetry. The adjustment was performed using the least-squares code LSL-M3, a modified version of LSL-M2 [9][14]. Neutron conversion factors are presented for various dosimetry readings into radiation metrics desired by experimenters.

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