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Highly Perturbed Operational Test Configurations at the WSMR Fast Burst Reactor

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Abstract. The White Sands Missile Range (WSMR) MoLLY-G reactor has a long history of producing a well characterized environment for testing electronic systems/devices in fission environments. As an unmoderated, unreflected, bare critical assembly, it provides a slightly degraded fission spectrum with a 1/E tail. For radiation hardness testing of electronics, the neutron fluence is usually reported as the 1-MeV Equivalent Neutron Fluence for Silicon. In this paper, we examine additional neutron environments and characterizations ranging from low intensity neutron fields to more extreme modifications of our normal test environment.

1. White Sands Missile Range MoLLY-G Reactor

The WSMR MoLLY-G reactor is an unmoderated, unreflected bare critical assembly. The reactor can be operated in either the steady state mode at the several kilowatt level or made super prompt critical, producing a pulse with a full-width half-maximum time of 45 microseconds or longer. The neutron spectrum leaking from the reactor is a slightly degraded fission spectrum. The reactor is operated in an exposure cell approximately 15 m by 15 m by 6.1 m high. The exposure cell has thick concrete walls lined with gypsum and borated gypsum wall board. The free field neutron environment at any position in the cell is a combination of the slightly degraded fission spectrum leaking from the reactor and a wall-return component which ties to the fission spectrum at a few keV in energy [1–3].

The MoLLY-G core is a cylindrical assembly consisting of six annular fuel rings made of enriched uranium-molybdenum alloy. The safety block is a large fuel element which is inserted into the central cavity of the core. The fuel rings are bolted to a stainless steel support plate by three Inconel metal bolts. There is a stainless steel retaining plate at the top of the core. The power level of the core is adjusted by two control rods of the same fuel alloy which are inserted into voids in the body of the rings. A third control rod can be pneumatically driven into the core for burst operations. The core is covered by a cylindrical decoupling shield containing a ^{10}B loaded silastic. The decoupling shroud minimizes the effect of reactivity changes due to experiments and other environmental considerations external to the core. The core geometry is shown schematically in Fig. 1 with a typical MCNP [4] model of the core shown in Fig. 2. A more comprehensive model used for spectral characterization at experimental

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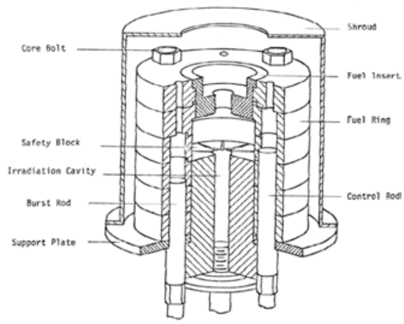

Figure 1. MoLLY-G Core.

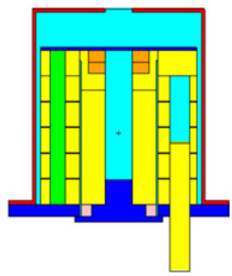

Figure 2. MCNP Model of the Core.

positions includes explicit modelling of the exposure cell walls and experimental table. The complete model was used for all configurations described in this paper.

For routine exposures, the neutron fluence is measured by activating sulfur pellets which are sensitive to neutrons with energies above approximately 3 MeV. The sulfur pellets and gas flow proportional counter are calibrated and traceable to the National Institute of Standards and Technology ^{252}Cf standard [5]. From calculations or activation foil measurements, we can determine the fraction of the spectrum above 3 MeV which then allows us to determine the total neutron fluence. For tests involving the effects of neutron radiation on silicon devices, we calculate the 1-MeV equivalent fluence using the Si displacement damage function [6, 7]. The general shape of the neutron spectrum has changed slightly through the years because of increased accuracy in the activation foil measurements and the continuous updating of the cross section libraries [8].

Figure 3 shows a comparison of calculated and measured spectra at our reference position at 61 cm from the core centerline. The MCNP model includes a complete description of the reactor, the exposure table, and the exposure cell walls, ceiling and floor. MCNP B-V is an MCNP4 calculation using primarily ENDF B-V cross sections. The SAND-II [9] unfolded spectrum is an adjusted spectrum using activation data from 14 activation and 3 fission foils, the SNL-RML cross section library [10], and the MCNP4 46 group spectrum as the trial. The MCNP B-VI calculation is a more recent full spectrum calculation and was used to evaluate the effects of changing from earlier cross section sets to the newer SNL-RML activation data library.

2. Neutron Spectra for Low Intensity Dose Measurements

An increasing number of test programs require pulsed operations delivering relatively small neutron dose. A nominal pulse at our reference position of 61 cm from core centreline produces a total neutron

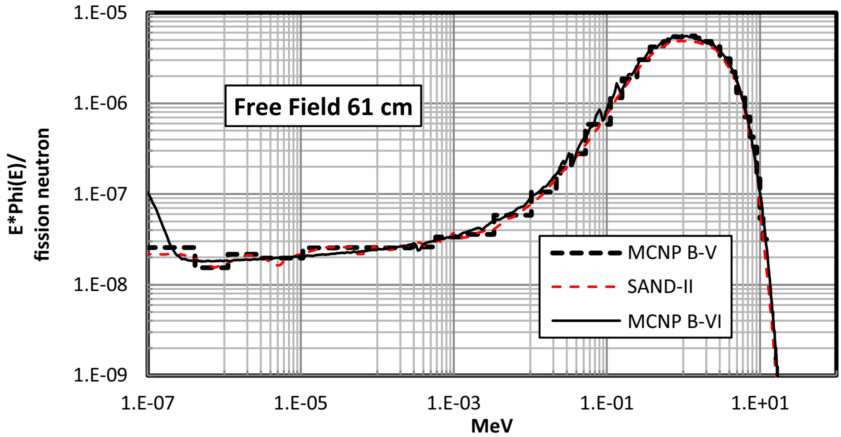

Figure 3. Neutron spectra at the reference position 61 cm from the core centreline. The SAND-II curve is the best experimental measurement of the spectrum using the 46 group MCNP B-V calculation as the trial input for activation foil data. The solid black line is an updated MCNP calculation using newer cross sections.

fluence in the 10^{12} n/cm^2 range. To provide lower test fluence, the pulse yield becomes smaller, increasing pulse variability for a given configuration. Wide variation in pulse size and resulting fluence is often experimentally unacceptable. We re-examined our characterization of the neutron spectrum at 254 cm from the core centerline. This environment is useful for testing equipment designed to monitor radiation dose.

The detailed model of the reactor core and the exposure cell was used to calculate the spectrum at 254 cm from the core centerline. At this position, the spectrum is considerably softer (more low energy neutrons due to reflection from the wall surfaces). Using the detailed neutron spectrum calculated by MCNP, we then folded the spectrum with a variety of tissue response functions to estimate tissue kerma, first collision dose, absorbed dose, and dose equivalent. This provides a "response" per unit fluence that can be applied to experimentally measured neutron fluence for the specific test. A limited set of activation foils were used in the spectral adjustment of the trial data from the MCNP calculation. Table 1 shows the C/E ratio before and after adjustment with the SAND-II code. The cross sections used for the spectrum adjustment were from the SNL-RML library. Figure 4 shows a comparison of the calculated to adjusted spectra at 254 cm.

A sampling of various tissue response functions in their native group structure or format were plotted and fitted with piece-wise polynomial functions. The resulting set of response functions were folded with the MCNP calculated spectral shapes. Table 2 gives the kerma or dose per unit fluence for a variety of response functions. We are anticipating re-measuring and re-calculating the spectrum at 61 and 254 cm using updated cross sections. Additional changes to structures (reactor support stand and an updated experimental table) will cause changes to our calculated spectra. The calculations using the updated cross sections are expected to change in the region of 1 to 3 MeV. We anticipate the newer calculations will result in a decrease of most response function factors of 3 to 5%.

3. Thermalized Test Environments

Recent test programs have required modification of the neutron spectral shape. One example is to produce a higher thermal to fast neutron ratio. The desired thermal neutron environment would be more easily accomplished in a different facility. However, program managers may desire to perform the entire test at a single facility, using consistent dosimetry methods. In order to partly accommodate these tests,

Table 1. Calculated to Measured Ratios for the MoLLY-G spectrum at 254 cm.

Foil Reaction	Cover	Ratio C/E	
		Before Adjustment	After Adjustment
^{24}Mg(n,p)^{24}Na		1.11	1.05
^{27}Al(n,a)^{24}Na		1.08	1.02
^{45}Sc(n,g)^{46}Sc		0.61	0.94
^{45}Sc(n,g)^{46}Sc	Cd	1.15	1.04
^{47}Ti(n,p)^{47}Sc		0.94	0.95
^{55}Mn(n,g)^{56}Mn	Cd	1.15	1.00
^{56}Fe(n,p)^{56}Mn		0.98	0.95
^{58}Ni(n,p)^{58}Co		0.98	0.99
^{98}Mo(n,g)^{99}Mo	Cd	1.02	0.99
^{127}I(n,2n)^{126}I		1.00	0.99
^{197}Au(n,g)^{198}Au		1.07	1.03
^{197}Au(n,g)^{198}Au	Cd	1.12	1.01
^{235}U(n,f)FP		0.95	0.94
^{238}U(n,p)FP	Cd, B	1.03	1.04
^{237}NP(n,f)FP	Cd, B	1.03	1.03
^{237}Np(n,f)FP		0.98	0.98
^{239}Pu(n,f)FP	Cd, B	1.08	1.07

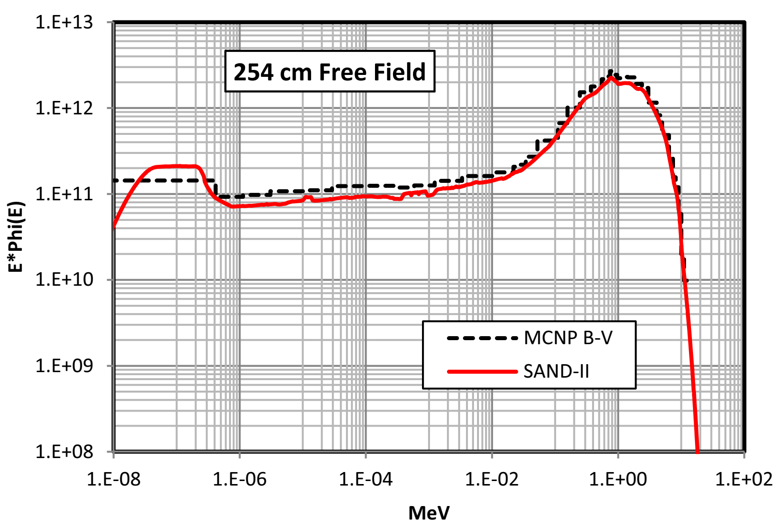

Figure 4. Calculated and adjusted spectral shape at 245 cm from core centerline. The SAND-II adjustment is based on SNL-RML cross sections and 17 reactions.

we characterized an environment using thick layers of graphite, sometimes with additional polyethylene layers, to build a box producing an environment with a significantly enhanced thermal fraction.

In 2004 a test program required a short pulse of thermal neutrons. The test did not require a detailed description of the whole spectrum, only the thermal region of the neutron spectrum. A large box of graphite with outer walls of polyethylene was constructed. Limited activation foil data was obtained in order to evaluate the fraction of the total neutron fluence in the thermal region. In 2013 a much larger test cavity was required and in this case enough spectral information had to be obtained in order to determine the dose delivered to an array of active detector systems inside the box. The original

Table 2. Dose conversion factors for the calculated MoLLY-G spectrum at 61 and 254 cm (Note: We expect new spectra with updated cross sections to be 3–5% lower).

Response Function		61 cm	254 cm
		pGy-cm^2	
E722 1-MeV Equiv. [7]	Si Displacement Damage	0.2848 (82.9 MeV-mbarn)	0.2246 (79.12 MeV-mbarn)
Kerr [11]	Tissue Kerma	23.74	18.6
Henderson [12]	First Collision (> 10 keV)	22.80	21.01
Snyder-Auxier [12]	Absorbed Dose	29.29	24.00
NBS Handbook 75 [13]	First Collision (> 10 keV)	23.15	21.33
NCRP 38 [14]	Absorbed Dose	29.81	24.37
ICRP 21 [15]	Absorbed Dose	29.37	23.89
ICRP 74 [16]	Absorbed Dose (using W_R^*)	18.90	15.21
		pSv-cm^2	
Snyder-Auxier	Dose Equivalent	281.4	224.7
NCRP 38	Dose Equivalent	289.3	230.5
ICRP 21	Dose Equivalent	278.5	220.4
ANS 6.1.1-1977 [17]	Deep Dose Equivalent	286.3	227.8
ICRP 74 H*(10)	Ambient Dose Equivalent	339.7	272.2

$^*W_R = 5+17\exp[-(\ln 2E)^2/6]$.

moderating configuration could not accommodate the full set of detectors simultaneously so the box was quickly resized for a larger cavity. The dimensions of this box were 61 cm wide by 41 cm wide by 61 cm tall with the size of the cavity increased to 10 cm (front to back) in the shield. The front face of the shielding box was 89 cm from the centerline of the reactor.

To estimate the neutron spectrum inside the graphite box, a model of the box was combined with the full model of the reactor in the exposure cell. The calculated spectrum was folded with the desired response functions and estimated the dose fraction due to thermal neutrons. The thermal neutron dose fraction was increased to about 50% of the total dose. Although a higher thermal fraction was desirable, the increase in dose due to thermal neutrons was sufficient to demonstrate the response of the device to thermal neutrons.

Because of the small cavity volume in the graphite shield, a limited activation foil set was used with the MCNP calculation providing the trial spectrum for the SAND-II adjustment. Several foil pairs covered the thermal region. Because of the very low fast fluence only a few high threshold foils received sufficient activation to be statistically significant. As a result, from the epithermal region and up, the spectrum is more heavily dependent on the input trial spectrum. Figure 5 shows the graphite shielded spectrum compared to a normalized free field spectrum at the same position.

4. Severely Moderated Neutron Environment: Iron/Graphite/ Polyethylene Shield Box

An example of a more severely modified environment is one typical of neutrons escaping a containment volume and penetrating an adjacent structure. Desired spectral shapes provided by the experimenter are shown in Fig. 6 (Spectrum 1 and Spectrum 2) with free field spectra at various distances from the core. Multiple shielding calculations were done estimating variations of spectra with different shielding configurations. Previous work to develop the thermal pulse in a graphite shield were useful as starting points, but requirements for a much larger internal cavity and lack of sufficient graphite precluded using a variant of the graphite box configuration. Considering the spectral softening as you move farther away from the reactor core, you would like to build the shield as far away as possible to take maximum advantage of the scattering and softening of the environment. However, since the reactor

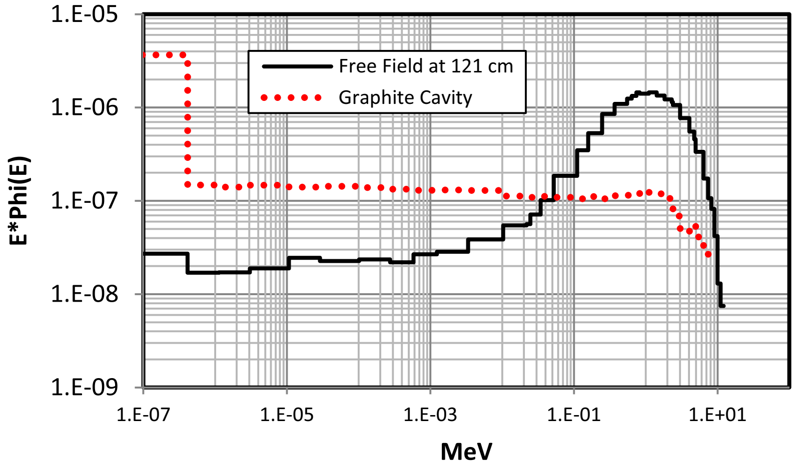

Figure 5. Free field and graphite moderated spectra, normalized to 1 fission neutron.

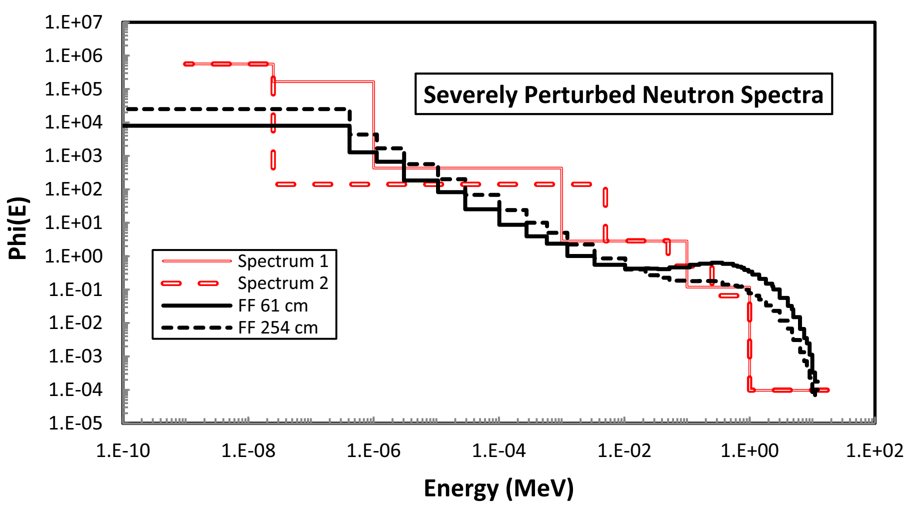

Figure 6. Target spectral shapes for a severely moderated neutron spectrum. Normalized free field spectra at 2 distances are shown for comparison.

still behaves somewhat as a point source at distant ranges, there is a $1/r^2$ penalty. The position at 121 cm was selected as the optimum starting point to obtain the desired total fluence in a reasonable reactor run time, taking into account there would be significant additional losses due to the shield box.

Selected materials for the shield box were constrained by the inventory on hand, but initially focused on graphite, polyethylene, and borated polyethylene. A generic shield box was modelled with MCNP using various thicknesses of the shielding materials. Inspecting the shape of the neutron spectrum showed significant moderation, but the fast neutron portion of the spectrum remained far too high. Eventually, a thick layer of mild steel improved the spectral degradation providing a much improved agreement with the target spectra. The MCNP model provided the guidelines for the experimenter to construct the shield box. Following a rather loose interpretation of specifications, a more accurate model

Table 3. Calculated to Measured Ratios for the iron/graphite/polyethylene shield box.

Foil Reaction	Cover	Ratio C/E	
		Before Adj	After Adj
^{56}Fe(n,p)^{56}Mn		0.735	0.959
^{47}Ti(n,p)^{47}Sc		0.841	0.995
^{58}Ni(n,p)^{58}Co		0.921	1.098
^{238}U(n,p)FP	Cd, B	0.850	1.004
^{237}Np(n,f)FP	Cd, B	0.795	0.953
^{239}Pu(n,f)FP	Cd, B	1.014	1.008
^{63}Cu(n,g)^{64}Cu	Cd	1.320	1.036
^{55}Mn(n,g)^{56}Mn	Cd	1.133	0.978
^{45}Sc(n,g)^{46}Sc	Cd	1.221	0.990
^{197}Au(n,g)^{198}Au	Cd	1.263	0.957
^{197}Au(n,g)^{198}Au		1.438	1.050

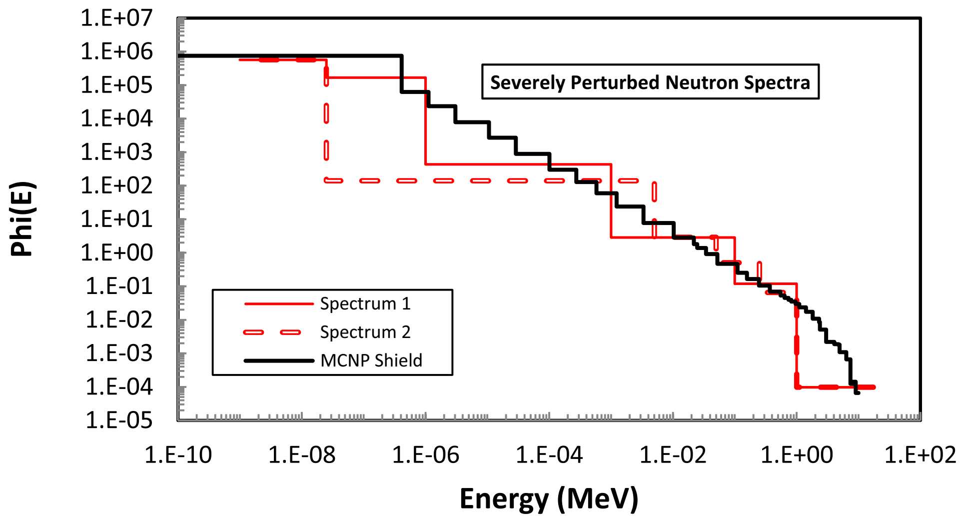

Figure 7. Severely moderated neutron spectrum inside an iron/graphite/borated polyethylene shield box.

with the shield box, reactor and exposure cell was used to calculate the final neutron spectrum inside the shield box. The final specifications for the shield box were:

Outer Front Face: 15.24 cm mild steel and 20.32 cm graphite.
Side Walls: 3.8 cm mild steel and 10.2 cm graphite.
Top/Bottom: 3.8 cm mild steel and 10.2 cm 5% borated polyethylene.
Cavity: Centered at 122 cm, 40.64 cm wide, 30.5 cm deep, 61 cm high.
Supported by wood framework.

The initial trial spectrum for adjustment of activation foil data for the experimental measurement in the shield box used an idealized model of the shielding box developed during an earlier phase of the experiment. Eleven reactions were used to characterize the spectrum inside the box. The C/E ratio before and after adjustment are shown in Table 3. The final calculated spectrum is shown in Fig. 7. The SAND-II experimental spectrum tracks the MCNP calculation closely, but it should be noted only a limited number of foils was used for the adjusted spectrum.

5. Conclusion

After many years of performing calculations and measurements primarily for radiation hardness testing of electronics, we are now reporting environment characterizations for a more diverse range of neutron test programs. Using a combination of calculations and activation foils with spectrum adjustment, we characterize experimental configurations in a very wide range of environments. This provides a wider range of test capabilities to the experimenter, including severely perturbed and moderated spectra. The neutron spectra obtained can be folded with any response function to provide the metric of interest, for example a 1-MeV equivalent fluence or a dose equivalent measurement.

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