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Workshop 5 – Cross Section, Nuclear Data, and Uncertainties

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The Cross Section, Nuclear Data, and Uncertainties Workshop met with 30 participants. There was a lively discussion that covered several different topics. Some of the topics had resonance with topics brought up in other workshop at this symposium.

One major issue was how to assign an *a priori* uncertainty to the trial spectrum used in spectrum adjustment techniques, or, more specifically, the need for a more rigorous and automated coupling of uncertainties into the radiation transport codes that are used to generate the *a priori* (trial) spectrum that is used in neutron spectrum adjustment approaches. Approaches based on discrete ordinate transport provide a means of generating sensitivity coefficients for specific variations, but fall short of the required task. While some Monte Carlo-based radiation transport codes permit the user to vary the cross sections for specific reactions or to vary material densities, a more universal and automated variation of all relevant inputs to the radiation transport code is required. One approach mentioned was a Total Monte Carlo (TMC) approach provided by a coupling into radiation transport codes of the recently generated statistical sample of random ENDF nuclear data evaluations produced by the TALYS code and incorporated into the TENDL library (for some isotopes). This would provide a set of self-consistent nuclear data files that can be sampled in a transport calculation. This variation of nuclear data would need to be coupled with a variation in material composition, density, and spatial dimensions.

Another discussion topic was the need for the community to revisit the consistency of current approaches to spectrum unfolds/adjustment, i.e. a new round of the older REAL-84 and REAL-88 exercises to assess our status.

While the dosimetry community has made great strides with the IRDFF library in establishing an international community consensus for dosimetry cross sections and associated nuclear data, the differences within the community in the treatment of the uncertainty in the trial spectrum and the availability of new spectrum adjustment techniques, such as Maximum Entropy codes like MAXED and the use of genetic algorithms, warrant a revisit of these international exercises to assess the current consistency (within stated uncertainty bonds) of approaches within the community to the characterization of neutron spectra and towards expressing the uncertainty in integral metrics based on these spectrum uncertainties. Since the IAEA/NDS has done such a good job in coordinating an international consensus in dosimetry cross sections and nuclear data, and since they coordinated the older REAL-8X exercises, we look to them to coordinate a new international assessment of our consistency in treating spectrum adjustments and characterizing the resulting uncertainties.

In the discussion on the IRDFF library, some users noted seeing issues with some of the current covariance files and expressed a need for the community to validate the covariance data along with the cross section data. The IAEA/NDS will be requested to review the status of the covariance files for the released IRDFF files – verifying that the matrices are positive semi-definite and can be processed with current codes, such as NJOY-2012. The need for compatibility between the recommended decay data and the cross section evaluation was also addressed in the discussion. Since there is an ongoing IAEA Cooperative Research Project (CRP), #F41031, which has the mission to "test, validate, and improve the IRDFF library", the IAEA will be requested to ensure that the scope of this "library validation" effort includes validation of both the cross sections, covariance files, and the associated nuclear decay data.

A important issue for the reactor dosimetry community is the lack of gamma dosimeters that facilitate the determination of the gamma spectrum in reactors. Many metrics for radiation effects are sensitive to photon as well as neutron damage mechanisms.

Current approaches use a calculated gamma spectrum, but the quality of the photon spectral characterization has not been validated and it does not typically come with any uncertainty quantification. The gamma environment is often dominated by secondary gammas generated by neutron interactions with nearby materials, and there is a large variation in the secondary gamma spectrum as modeled by different evaluations of nuclear data, e.g. between ENDF/B, JEFF, and JENDL libraries. As was demonstrated in work at the High Flux Isotope Reactor (HIFR) reactor, the effect of gammas through (γ,γ') reactions and (γ,f) photofission reactions can sometimes have a significant impact on the interpretation of reactor dosimetry.

There was discussion on the need for new multi-group neutron/gamma cross section libraries. While processing codes such as NJOY have been used to generate multi-group libraries based on most recent nuclear data evaluations, the energy group structure for the current collapsed-group cross section libraries has been tailored for use in light water reactors. It is not clear that this energy structure is appropriate for all applications supporting advanced reactor concepts such as thorium-fueled sodium-cooled fast reactors, lead-bismuth eutectic fast reactors, and very high temperature reactor (VHTR) designs. The adequacy of the collapsed energy grid is addressed in the ANS 6.1.2 standard and can only be determined through the comparison of results produced by fine-group with that from collapsed-group cross sections – and through a range of comparison for relevant calculated benchmark problems. This consideration generated a need for a wider range of calculated benchmarks to support multi-group calculations for advanced reactors. The community is looking for an agency to lead the way in the development/maintenance of a SINBAD-type compilation of benchmarks that are applicable for advanced reactor concepts. The community suggested that Radiation Safety Information Computational Center (RSICC) is a candidate for this role since they are currently responsible for the Shielding Integral Benchmark Archive and Database (SINBAD).

There was discussion about support for continuous energy or point cross section libraries. While the generation of ACE-type point cross section libraries as are used in the MCNP code is supported by the NJOY processing code, it was not clear that there were openly available codes that supported the generation of point cross section library formats used in other codes from ENDF-6 format nuclear data evaluations. Discussion revealed the following status for some other codes: the AMPX-2000 code supported cross section generation for the point-wise version of KENO V.a; PREPRO-2012 can process cross sections for the TART code (which really uses a very finely gridded multi-group/multi-band set of cross sections rather than truly continuous energy cross section representation); the MCBEND code uses an "ultrafine" 13,193 group cross section representation and has data libraries based on most nuclear data files, but does not appear to have an openly available code that permits users to generate their own cross sections; COG can directly use ENDF-format cross section representations; TRIPOLI can be used with any ENDF-6 format representation. The conclusion was that point cross section formats are generally code-specific, in which case the developer generally supplies codes that can be used to generate the cross sections from available ENDF-6 format files, or that the codes can directly utilize the standard ENDF-6 cross section format.

At its conclusion, there was a consensus that this was a very productive workshop and the group looked forward to a future workshop in this area at the ISRD16 where participants may have updates regarding some of the issues raised here.