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NUCLEAR DATA FOR CRITICALITY SAFETY - CURRENT ISSUES

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INTRODUCTION

Traditionally, nuclear data evaluations have been performed in support of the analysis and design of thermal and fast reactors. In general, the neutron spectra characteristic of the thermal and fast systems used for data testing are predominantly in the low- and high-energy range with a relatively small influence from the intermediate-energy range. In the area of nuclear criticality safety, nuclear systems arising from applications involving fissionable materials outside reactors can lead to situations very different from those most commonly found in reactor analysis and design. These systems are not limited to thermal or fast and may have significant influence from the intermediate energy range. The extension of the range of applicability of the nuclear data evaluation beyond thermal and fast systems is therefore needed to cover problems found in nuclear criticality safety.

Before criticality safety calculations are performed, the bias and uncertainties of the codes and cross sections that are used must be determined. The most common sources of uncertainties, in general, are the calculational methodologies and the uncertainties related to the nuclear data, such as the microscopic cross sections, entering into the calculational procedure. The aim here is to focus on the evaluated nuclear data pertaining to applications in nuclear criticality safety.

CURRENT ISSUES

In the winter of 1993, an article¹ in the *Criticality Safety Quarterly* pointed out differences in the calculated infinite multiplication factor (k_{∞}) of calculations performed with the SCALE code system² and the MCNP code.³ The systems considered in the calculations were fictitious mixtures made of metal mixed with ^{235}U , namely, Al/ ^{235}U , Fe/ ^{235}U , and Zr/ ^{235}U . Since the MCNP calculational methodology is based on the continuum energy approach, an assumption was made that the MCNP results were correct. In the absence of experimental data for these metal/ ^{235}U systems, to understand the cause of the discrepancies between the results, it was decided to calculate the k_{∞} of the metal/ ^{235}U systems using a variety of computer codes and cross-section libraries. It turned out that the k_{∞} results of these calculations were not in agreement with MCNP. Since several sources of uncertainties contribute to the overall uncertainty in the integral results, the lack of agreement among the different calculational methodologies led us to investigate the methods and data used in the calculations. A detailed examination of the MCNP results suggested that the pointwise cross-section data of its library used in the calculations were not adequate for the systems under consideration. In particular, it was discovered that the aluminum cross sections in the ENDF library were given as averaged smooth cross sections which cannot be energy self-shielded even though aluminum has resonance structure. It is well known that resonance parameter representations are needed to properly account for resonance self-shielding effects. The neutron spectrum characterizing the Al/ ^{235}U system studied was in the intermediate energy range where the resonance self-shielding effects on the aluminum cross sections play an important role, and, therefore, strongly indicate the pointwise cross-section representation of ENDF/B is inadequate. In a separate study, a criticality safety assessment of the fuel cycle facility⁴ at Argonne National Laboratory, Idaho Falls, also indicates a similar problem related

to the pointwise cross-section representation. In this study, it was verified that the cross-section representation of chlorine (Cl), a material with resonance structure, is not represented with resonance parameters, but rather with averaged smooth cross sections. These issues highlight areas where there appears to be a deficiency in the ENDF evaluations, specifically data for materials used in criticality safety which have no resonance representation.

NUCLEAR DATA NEEDS FOR CRITICALITY SAFETY ANALYSES

Nuclear systems commonly found in criticality safety applications encompass a wide range of fissionable, moderator, and absorber materials. The characteristic neutron spectrum of these systems spans a wide range from thermal to fast. Of particular interest for many current applications are systems with a neutron spectrum which peaks in the intermediate energy range. In general, the intermediate energy range is between 3 eV and 3 keV. A system is considered an intermediate system when the energy corresponding to the average lethargy of the neutron-causing fission (AEF) falls in this range. This energy range, from the viewpoint of differential data, is extremely important since resonance effects in the cross sections are dominant. The status of nuclear data for the majority of isotopes of interest to criticality safety applications in the basic nuclear data files has to be examined. As an example, Table 1 lists eight materials of interest in criticality safety identified as having questionable ENDF/B-VI evaluations.

There are a variety of isotopes that can be added to the list in Table 1. One specific need is to improve the ^{235}U capture-to-fission ratio (alpha). Existing ^{235}U evaluations do not reproduce the measured alpha value and consequently impact criticality safety analyses of highly enriched intermediate spectrum systems.

From the point of view of integral data, there is a need for high-quality critical experiments with intermediate energy spectra (3 eV to 3 keV) to validate methods and data. An assessment of existing experimental data relevant to criticality safety applications has to be performed. Table 2 shows four experiments of interest for data assessment for criticality safety applications.

CONCLUDING REMARKS

The issues mentioned above indicate a need for new data and evaluations. To satisfy specific needs for criticality safety applications, closer collaboration between the nuclear criticality safety community and the Cross Section Evaluation Working Group (CSEWG) must be established. In this framework, a broader set of CSEWG benchmark testing should be developed to help extend the energy range of cross-section validation. Detailed evaluations of many isotopes with resonances in the intermediate energy range are clearly needed.

Table 1. Isotopes for which measurements and evaluations are needed

Isotope	Cross-section representation	
	Last evaluation	Resonance data
²⁰ Ca	Aug71	No ^a
²⁷ Al	Dec73	No
¹⁷ Cl	Feb67	No ^a
¹⁹ K	Feb67	No ^a
³¹ P	Oct77	No
¹⁴ Si	Feb74	No ^a
¹² Mg	Feb78	No ^a
²³³ U	Dec78	Yes

^aIsotopic evaluations needed.

Table 2. Critical experiments with intermediate energy spectra

Critical experiments	Description
HISS(HUG)	Homogeneous uranium-graphite
HISS(HPG)	Homogeneous plutonium-graphite
UH ₃ -UR	High-enriched uranium with low-enriched uranium reflector
PCTR	Low-enriched UO ₃ /H ₂ O with a range of H/U

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