January 16, 2002
Contributed ANS Summary
To be published in ***TRANSACTIONS***
Track 3, 3i or 3l

Nuclear Science and Technology Division (94)

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Submitted to the
American Nuclear Society 2002 Annual Meeting
"The Revival of the Nuclear Energy Option",
Embedded Topical Meeting:
International Congress on Advanced Nuclear Power Plants (ICAPP)
June 9–13, 2002,
The Diplomat Hotel,
Hollywood, Florida

*Managed by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the U.S. Department of Energy.

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In the United States (U.S.) and abroad, the diversity of fissile-material combinations and processes under consideration is increasing in the absence of applicable and comparable benchmark experimental data. Consequently, the implementation of safe, efficient and optimized fissile-material operations is increasingly dependent upon the use of radiation transport methods with accurate cross-section data. In the absence of accurate nuclear data, radiation transport methods cannot provide accurate results and will significantly hinder the analysis of fissile-material operations. In the U.S., the Evaluated Nuclear Data File (ENDF) system[1] is the repository for evaluated cross-section data. Because of the complexity and volume of data in a cross-section evaluation, these files cannot be accessed directly by radiation transport codes. Therefore, a cross-section processing code system must be used to process the ENDF evaluation for each material and generate cross-section libraries for modern radiation transport codes. Prior to the current work, the NJOY code system[2] was the only complete software package that could be used to process data for all versions of ENDF/B data through Version VI. Because of the paucity of critical experimental data, nuclear analysts need independent radiation transport capabilities to evaluate new and emerging fissile material operations. With equal importance, there is a need for an independent nuclear data processing capability. At the Oak Ridge National Laboratory (ORNL), a new version of the AMPX cross-section processing system (AMPX-2000) has been developed[*] to process the full range of ENDF formats used to describe the physics associated with neutron, gamma and neutron-gamma interactions up to 20 MeV. The objective of this paper is to describe the processing capabilities of AMPX-2000 for criticality safety applications.

AMPX-2000 can be used to generate a variety of multigroup and/or continuous-energy (point) cross-section libraries that can be used to perform nuclear analyses. For example, the SCALE[3] system, which is used to perform licensing evaluations, is designed to use AMPX-formatted libraries. The AMPX code system is extremely modular in construction and is comprised of more than 80 distinct processing modules that perform a variety of data operations. A detailed description of each module is beyond the scope of the paper; however, some of the prominent functional capabilities include:

- Generate temperature-dependent pointwise cross-sections,
- Provide resonance self-shielding for the resolved- and unresolved-resonance range (i.e., RRR and URR, respectively),
- Generate probability tables for the URR,
- Generate energy and angle distributions for secondary particles,
- Process $S(\alpha,\beta)$ data for thermal moderator evaluations (e.g., H in H_2O, H in ZrH, etc.),
- Generate free-gas $S(\alpha,\beta)$ data for other nuclides,
- Process particle-yield data,

[*]Development funding provided by the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC).

- Generate continuous-energy weighting spectra,
- Perform multigroup averaging operations,
- Process cross-section uncertainty data for use in sensitivity and uncertainty (S/U) analyses.

Although the AMPX system has existed since the early 1970s with the most recent release being AMPX-77,[5] AMPX-2000 represents a complete revolution in construction and functional capability of the code system. The following discussion highlights some of the new features of the system that are pertinent for criticality safety applications.

Continuous-Energy Cross Section Data

AMPX-2000 has the capability to process ENDF/B resonance parameters and generate temperature-dependent cross sections as a function of energy in the RRR and the URR.[6] Work is also in progress to develop pointwise cross sections for a continuous-energy version of KENO that will be released in a future version of SCALE. A distinct advantage of AMPX-2000 is the capability to read and export cross-section libraries in a variety of formats. For example, AMPX can be used to produce MCNP[7] cross sections. Currently, the MCNP-library generation must be completed by coupling AMPX with NJOY. Temperature-dependent pointwise data can be generated in a PENDF format with AMPX-2000, and the NJOY module ACER is used to generate the ACE-formatted cross-section library. Note that the "tools" are essentially in place to generate an MCNP library completely independent of NJOY; however, additional development work is needed to finalize the independent MCNP-library production capability.

Particle Interactions

One of the novel and robust features of AMPX-2000 is the treatment of the collision kinematics that provide exiting energy and angle distributions for particles emerging from a reaction of interest. In AMPX, a uniform forward kinematics structure has been developed to describe all possible secondary energy-angle distributions including thermal scattering collisions. As a result, multigroup and pointwise libraries can be produced that are independent of ENDF laws and procedures. Numerous radiation transport codes have the ENDF laws and procedures programmed in the code. Unfortunately, as changes are made to ENDF, the radiation transport code and cross-section processing code system must be updated to process the new ENDF laws and procedures. The recent developments in AMPX have removed the burden of processing ENDF formats from the transport code and transferred the "ENDF processing" to the cross-section generation code. As new updates are implemented in ENDF, AMPX must change, but the radiation transport code does not need modification thereby reducing code maintenance costs.

Resonance Self-Shielding

Historically, problem-dependent resonance self-shielding has been accomplished in AMPX and SCALE using either the Nordheim integral treatment method (**NITAWL-III** module) or the Bondarenko factor method (**BONAMI** module) for the RRR or URR, respectively.[3] One of the newest additions to AMPX and SCALE is a continuous-energy discrete ordinates capability (**CENTRM**) that can be used to calculate pointwise neutron spectra for a one-dimensional system.[8] **CENTRM** uses pointwise data to solve the Boltzmann transport equation to obtain problem-dependent neutron fluxes. AMPX-2000 is used to generate pointwise cross sections that are needed by **CENTRM**. In SCALE, Criticality Safety Sequences (CSAS) have been developed to use **CENTRM** to generate problem-dependent neutron spectra that are subsequently used to calculate problem-dependent self-shielded multigroup cross-sections.

Probability Tables

Because of the statistical nature of the unresolved-resonance parameters, probability tables can be used to provide cross-section probability distribution functions for energy ranges at specific temperatures within the URR. In AMPX-2000, a Monte Carlo approach is used to calculate probability tables for the URR. As a distinction, AMPX does not use the historical or conventional "ladder" approach to generate a series of resonances that are subsequently used to calculate a probability table.[9] In the AMPX approach, pairs of resonances are sampled around the reference energy or energies for a table. The resonance spacings are sampled from a Wigner spacing distribution, and the parameters for each resonance are sampled from a Chi-square distribution with a designated number of degrees of freedom. The probability tables that are produced by AMPX can subsequently be used in Monte Carlo transport calculations.

Cross-Section Uncertainty

Since the release of ENDF/B-IV, standards and formats have been in place to permit the communication of estimated uncertainties in the evaluated cross-section data. By including the uncertainty or covariance information, the analyst can propagate cross-section data uncertainties through sensitivity studies to a final calculated quantity of interest. AMPX-2000 can be used to process ENDF covariance data for a specific material, and generate covariance information associated with $\bar{\nu}$, resonance parameters and neutron cross section data.[10] AMPX-generated covariance data have been used to provide uncertainty information for use in S/U analyses for criticality safety.[11]

In short, the AMPX-2000 cross-section processing system can be used to process ENDF/B cross-section formats through Version VI and generate pointwise or multigroup nuclear data libraries for modern radiation transport codes. The AMPX code system provides an independent data-generation capability that can be used to provide cross-section data for criticality safety analyses.

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