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Verification of a Subgroup Generation Method for PWR Fuel Pin with Am-241

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1. Introduction

A new procedure of generating multi-group cross section data from the evaluated nuclear data file (ENDF) and/or other nuclear data files such as JEF, JENDL, BROND and CENDL without allowing any forced adjustment of the resonance integral, has been established in order to apply it in building the direct whole core transport code, nTRACER's own cross section library. In this consistent method, only shield subgroup cross section level dependent conservation condition is imposed instead of use of the resonance integral conservation principle.[1] As a Separate Effects Test (SET) for code validation, and improvement, and for verification of proper processing of its own cross section library starting from any basic nuclear data files, adding one specific isotope in conventional UO_2 fuel pin cell will be the easiest way for the sake of nTRACER library files generations for primary nuclides included in fuel pin cell before investigation into further complex geometry configurations or the innovative fuels in current PWRs.

Some nuclear waste actinides, such as Pu239, Pu241, Am242m and Cm245 have high thermal fission cross sections, and all of them become fissile fuel under fast neutron spectrum. Although Am242m has the highest thermal fission cross section values among known isotopes, some other MAs such as Pu238, Am241 are moderately fissile even under thermal neutron spectrum. It is worth to investigate SET of normal PWR UO_2 fuel pin mixed with Am241 for subgroup parameter generation and new cross section library data verification because there is also a potential of MA for future use as nuclear fuel leading to their incineration in conventional commercial reactors.

Thus, the extensive efforts for actual generation and verification of individual nuclides data for Am241 in addition to five primary nuclides U235, U238, H1, O16, natural Zr, are performed by solving simple pin cell problem. The calculation results such as multigroup cross sections and the multiplication factors are then compared with the continuous energy Monte Carlo calculation ones achieved by using same ENDF-B/VII and JENDL-4 cross section data. It is found that the nTRACER calculation results of reactivity differences from Monte Carlo solution for same $(U+Am241)O_2$ fuel pin by using same ENDF and JENDL, are about 11 and 26 pcm respectively at 764 K and also meet in good agreement for reactivity swing with temperature variations. In addition, the accuracy of RMET21 calculation of shielded cross sections of each resonant isotope is analyzed from combined resonant isotopes effect under the assumption of resonant isotope as resonant or non-resonant ones in a mixture.

2. Pin Cell Model under Investigation

After we decided to select Am241 isotope to mix in typical fresh UO_2 pin cell with enrichment of 4.9 w/o for benchmark calculation, as a next step, how much volume percentage of Am241 will be added is one of concern though we do not emphasize on optimization of fuel pin since we targeted on validation and verification of newly generated nuclear library data. From basic understanding of Am241 behavior as regards excess reactivity of fuel pin, only slightly mixture of Am241 (0.2 volume percentage in 4.95 w/o UO_2 fuel pin), is taken into account to generate new cross section library firstly for Monte Carlo calculation from ENDF/B-VII and JENDL-4.0. The cross section libraries at various temperatures for the Monte Carlo codes are processed by mainly NJOY and additional ANJOYMC program developed by KAERI [2] which facilitates generation of a batch file and the NJOY input files for further continuous cross section library for MCNP processing. The multiplication factors by nTRACER and MCNP calculation for pin cell model starting from two libraries; ENDF/B-VII and JENDL-4.0 are compared in Table I. The reactivity difference with JENDL is slightly larger than the one with ENDF. Later on, same libraries at more temperatures are used to validate library data generation for nTRACER.

Table I. K-effective values between two libraries at 764 K

Libraries	JENDL-4.0	ENDF/B-VII
nTRACER	1.2638	1.2785
MCNP	1.2634	1.2783
$\Delta\rho$ (pcm)	26	11

3. Verification of Library Generation

2.1 Self-shielded Multigroup Cross Section Generation

Despite that the actual configuration to be used in the core calculation is a square cell, RMET21 [3] requires only

circular cell geometry by the Winger–Seitz approximation for its calculation and thus, it was converted into required geometry. The resonant isotope effects are investigated for all resonant isotopes by assuming all isotopes contained in fuel pin cell as resonant ones (Case-C) and by assuming only resonant isotope of interest in mixture as resonant (Case-D) shown in Figure 1. In doing so, the rigorously shielded cross sections for resonant isotopes (U238, U235, Am241) which are included in this pin cell model are generated through RMET21 runs solving neutron slowing down equation in heterogeneous 1D cylindrical geometry on an ultra-fine energy grid provided by execution of GEXSCO. All are compared with MCNP results for each resonant isotope. The maximum relative errors of U238 resonant in both Case-C and -D are 0.11% at 22.6 eV. While the rest two fissile isotopes; U235 and Am241 manifest very good agreement in Case-C, each one has maximum relative errors of 0.77% at 6.87 eV and 0.39% at 37.27 eV correspondingly. Nevertheless, it was found that all resonant isotopes in Case-C had no distinctive errors, and they were negligible.

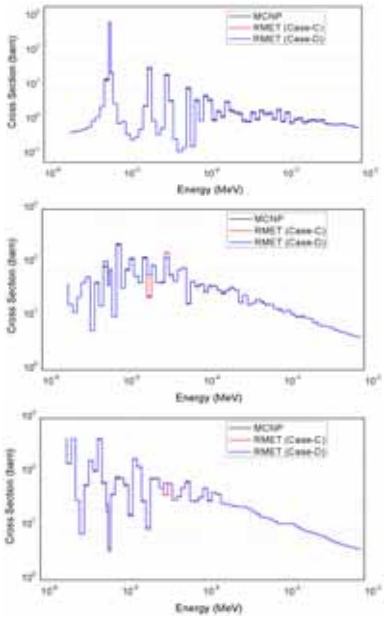

Figure 1. Multigroup absorption cross sections of U-238 (top), U-235(middle) and Am241(bottom) obtained by MCNP and RMET21

Since the self-shielding that determines the groupwise effective cross section is strongly dependent on the material composition and also to the geometrical configuration, 10 pin cells with different dilution (geometry and density variation) for 69 resonance groups at 5 temperature are considered. Again, shielded subgroup level dependent background cross sections are generated through subroutine nTRACER-FSP transport solver which solves a fixed source problem given the effective cross section. The subgroup parameters can be determined by the utility program called GENOME which implements the method of Lagrange multiplier by solving a constrained minimization problem. As prerequisites, it requires NJOY for pointwise and groupwise cross section generation as PENDF and GENDF, and LIBDEC aimed to organize and integrate well-defined, complete nuclide-wise library file for use in nTRACER simulation.

2.2 Benchmark Results of Multiplication Factor

The multiplication factors are calculated by using newly established library data of fuel pin model $(U+Am241)O_2$ for nTRACER for two cases and compared with MCNP results as described in Figure 2. At 764 K, the reactivity difference of case-C with ENDF is about 11 pcm and that with JENDL, 26 pcm whereas one with ENDF in case-D shows 91 pcm. It is interestingly found that the case-D at higher temperature after 959 K is closer to MCNP results. In short, both cases still meet in good agreement with Monte Carlo simulation.

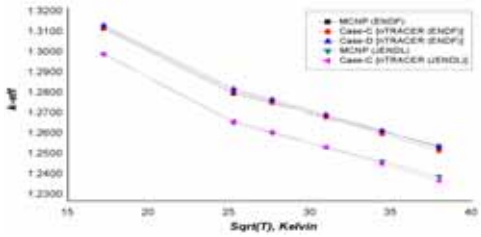

Figure 2. Multiplication Factor vs. Sqrt(T) evaluated by nTRACER and MCNP

3. Conclusions

The accuracy of library data is determined through comparison of the results obtained by MCNP4c calculation with same ENDF/B-VII library data. The reactivity differences of reference model pin cell between them are small. As future work, it still requires investigation into the innovative fuels in current PWRs or further complex geometry configurations (Fuel Assembly or Whole Core). The ‘‘super-cell’’ calculation modeling the fuel pellet which may contain burnable absorbers, and all other materials found in fuel assembly, will be proposed to get involved in more materials and complex geometry arrangement.

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