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Nuclear data sensitivity and uncertainty for the Canadian supercritical water-cooled reactor

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ABSTRACT

Accurate and complete nuclear data are a fundamental requirement for any nuclear reactor model. Present nuclear data evaluations have been tested and validated against experiments that cover operating conditions and materials that are typical of conventional reactors (e.g. thermal light and heavy water moderated reactors, and some fast reactors). The Canadian supercritical water-cooled reactor (SCWR) is an advanced reactor concept for which, like all advanced GEN-IV reactor concepts, operating conditions and materials differ significantly from conventional reactors. Consequently, current nuclear data evaluations may not be suitable for modeling the Canadian SCWR, either due to assumptions or gaps in the evaluations or gaps in the experimental data on which the evaluations are based. In this paper, a simplified model of an SCWR fuel channel with fresh fuel is analyzed in order to determine the sensitivities and uncertainties of the model to nuclear data. The results are used to determine which isotopes and which reactions make the largest contributions to the sensitivity and uncertainties in key reactor physics parameters such as the neutron multiplication factor, k, and various lattice reactivity coefficients.

1. Introduction

Advanced Generation-IV (GEN-IV) reactor concepts are being developed through an international collaboration, the GEN-IV International Forum (GIF), in order to provide future nuclear energy systems with enhanced safety, improved resource sustainability, improved economic benefit and enhanced proliferation resistance (U.S. DOE Nuclear Energy Research Advisory Committee, 2002). Canada's primary contribution to the GIF is the Canadian SCWR concept (Leung et al., 2011), which is a heavy water moderated, pressure tube reactor that uses supercritical light water (SCW) as a coolant. The use of SCW coolant significantly increases the thermodynamic efficiency of the SCWR over present heavy water reactors (HWR), from approximately 33% to as high as 48% efficiency. However, the harsh environment of the SCW coolant necessitates the use of modified stainless steels or other advanced materials rather than more typical zirconium-based alloys as in-core materials (Ru and Staehle, 2013). SCWR operating conditions and materials, therefore, present significant differences from conventional HWR and light water reactors (LWR). Since current nuclear data evaluations and reactor modeling methods are optimized for conventional HWR and LWR, further development of both the nuclear data and modeling methods is likely needed for accurate SCWR modeling.

Regardless of the modeling approach, the accuracy of a nuclear reactor physics model is ultimately limited by the accuracy of the nuclear data used in it. The development of accurate models of any advanced or novel reactor systems therefore requires an assessment of the sensitivity of calculations to the nuclear data as well as the quality of the data itself (Chang, 2003; Driscoll and Hejzlar, 2005; Forrest, 2010; Pelloni and Mikityuk, 2012). A number of such studies for advanced nuclear systems have been performed. Uncertainties in actinide cross sections were studied in advanced systems for transmutation of nuclear waste in Artisyuk et al. (2008), and the use of the ERANOS code to examine uncertainties in modeling of advanced fast reactors was presented in (Aliberti et al., 2006; Pelloni and Mikityuk, 2012). A comparative study of nuclear data libraries for the Canadian SCWR was given in (Kozier and Dyck, 2005), and a similarity study against previous and proposed experiments in the ZED-2 critical facility (located at AECL's Chalk River Laboratories) was presented in (Langton et al., 2012).

This paper combines and expands upon work presented at two recent conferences, the 3rd China-Canada Joint Workshop on Supercritical-Water-Cooled Reactors (Blomeley and Pencer, 2012a) and the 33rd Canadian Nuclear Society Annual Meeting (Blomeley and Pencer, 2012b). The TSUNAMI (Tools for Sensitivity and Uncertainty Analysis Methodology Implementation) codes in SCALE (Standardized Computer Analyses for Licensing Evaluation) 6.0 (Oak Ridge National Laboratories, 2009; Reardon et al., 2011) are used to study the effects of nuclear data on calculations for SCWR fresh fuel. Nuclear data related sensitivities of and uncertainties in the neutron multiplication factor of the system (k), as

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well as coolant voiding, fuel temperature, coolant temperature and moderator temperature reactivity coefficients are examined. These results are used to determine the isotopes and reactions that make the largest contributions to the nuclear data sensitivities and uncertainties.

2. Calculation methods

The sensitivity and uncertainty analyses presented in this work were performed using the codes TSUNAMI-3D for sensitivity calculations of k and TSAR (Tools for Sensitivity Analysis of Reactivity) for sensitivity calculations of reactivity parameters. The TSUNAMI code set, nuclear data library and nuclear data covariance data library used for this study were the versions distributed with SCALE 6.0 (Oak Ridge National Laboratories, 2009). The libraries and codes used are described in more detail below.

TSUNAMI-3D is a code sequence in SCALE that executes KENO V.a to generate forward and adjoint neutron transport solutions followed by SAMS (Sensitivity Analysis Module for SCALE) to produce sensitivity coefficients (Oak Ridge National Laboratories, 2009; Reardon et al., 2011).

Calculations in TSUNAMI-3D were performed using the 238-group ENDF/B-VII Rel.0 library and the single, comprehensive cross-section-covariance data library distributed with SCALE 6.0. The covariance library combines evaluated covariance data from a number of sources (Reardon et al., 2011). The application of this set of covariance data is based on a number of assumptions (Williams et al., 2009), which will not be discussed here. Nevertheless, the uncertainties determined below depend directly on the covariance data and consequently, specific conclusions made based on those uncertainties should take into account the relevant assumptions made in the evaluations of the covariance data used.

SAMS determines a sensitivity profile consisting of a dimensionless quantity at each energy group, which is calculated as the fractional change in the neutron multiplication factor, k, with respect to the fractional change in the macroscopic cross section, Ω, i.e.,

$$Sensitivity = \frac{\partial k}{k}\left(\frac{\partial \Omega^i_{x,g}}{\Omega^i_{x,g}}\right)^{-1}, \tag{1}$$

where the nuclear data component for process x of nuclide j in energy group g is denoted $\Omega^i_{x,g}$. The contributions to sensitivity from various isotopes discussed later are also dimensionless and expressed as fractions of k. The complete sensitivity including the implicit component from the resonance self shielding calculations and the explicit component of k due to perturbations of the nuclear data $\Omega^i_{x,g}$, (representing both the cross section data Σ and other nuclear data terms) can be defined as,

$$\left(S_{k,\Omega^i_{x,g}}\right)_{complete} = \frac{\Omega^i_{x,g}}{k}\frac{\partial k}{\partial \Omega^i_{x,g}} + \sum_j \sum_h \frac{\Omega^j_{y,h}}{k}\frac{\partial k}{\partial \Omega^j_{y,h}} \times \frac{\Omega^i_{x,g}}{\Omega^j_{y,h}}\frac{\partial \Omega^j_{y,h}}{\partial \Omega^i_{x,g}}, \tag{2}$$

where the nuclear data component for process y of nuclide j in energy group h is denoted $\Omega^j_{y,h}$. This formulation includes the explicit sensitivity in the first term and the implicit sensitivity due to the effect of changes in $\Omega^j_{x,h}$ on the crosss section $\Omega^j_{y,h}$ based on resonance shielding. The complete effect then incorporates all terms for which this effect is non-zero.

The nuclear data covariance matrix between all cross sections (and all other nuclear data terms) is defined as,

$$C_{\alpha\alpha} = \left[\frac{COV(\alpha_m, \alpha_p)}{\alpha_m \alpha_p}\right], \quad \alpha = 1,2\ldots M; \quad p = 1,2\ldots M, \tag{3}$$

where M is the number of nuclide-reaction pairs multiplied by the number of energy groups. This matrix can then be used in conjunction with the sensitivity data to find the uncertainties in k due to the nuclear data covariances,

$$\sigma_k^2 = S_k C_{\alpha\alpha} S_k^T. \tag{4}$$

In this formulation, the diagonal terms are the nominal cross section variances for each nuclide, reaction and energy, whereas the off diagonal terms relate terms at different energies and in a few cases different reactions and/or nuclides. SAMS can provide the total or partial uncertainties in k due to the uncertainty in individual nuclides.

The TSAR code computes reactivity sensitivities based on the k sensitivities of two states. The reactivity sensitivity profile is calculated from the k sensitivity profiles S_k using

$$S_{\rho,\alpha} = (\lambda_2 S_{k_2,\alpha} - \lambda_1 S_{k_1,\alpha}), \tag{5}$$

for a particular nuclide-reaction cross section α where the reactivity change between the two states is $(\lambda_1 - \lambda_2)$ and $\lambda = 1/k$.

When these sensitivities are combined with the covariance data, TSAR can also be used to compute uncertainties in calculated reactivity coefficients due to uncertainties in the nuclear data. TSAR can provide the total uncertainty, or partial uncertainties in reactivity coefficients due to individual nuclide-reaction uncertainty.

3. Modeling methods

For this study, a single fuel channel loaded with fresh fuel is modeled. Examination of a fresh fuel assembly facilitates the isolation of major contributions to uncertainties in neutron multiplication factor k and reactivity coefficients without the complication of fission and activation products. The fuel assembly and fuel channel specifications used in this study are based on those used in (Boczar et al., 2010; Magill et al., 2011) in order to facilitate comparisons with other related studies (e.g., Harrison and Marleau, 2012; Shen, 2012). More recent SCWR fuel assembly designs (e.g., McDonald et al., 2011; Pencer et al., 2013) are not considered here. A cross-sectional view of the fuel channel is shown schematically in Fig. 1, with geometric specifications listed in Table 1. The fuel assembly contains 54 fuel elements with a centre pin of zirconia. The fuel channel consists of a perforated steel liner, a porous zirconia insulator and a Zircaloy pressure tube.

The component material compositions used in the models are listed in Table 2. The fuel is 14% by weight PuO_2 in ThO_2. The fuel cladding and perforated liner are made of modified 310 stainless steel (310 SS). The liner and insulator are modeled as homogeneous volume-weighted mixtures of coolant with liner material and insulator material, respectively.

Estimated values of temperatures for the components at each nominal axial location are also taken from (Magill et al., 2011) and are listed in Table 3. While these temperatures are expected

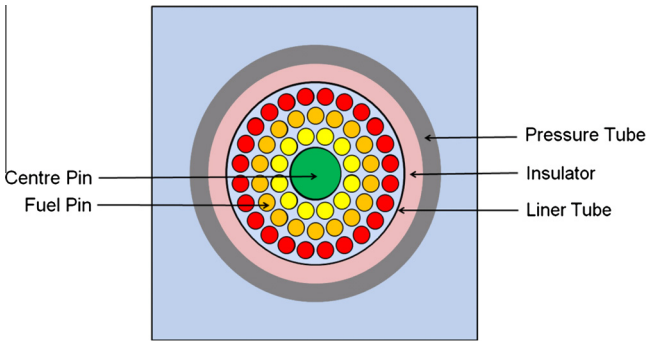

Fig. 1. Cross-sectional view of the SCWR fuel assembly and fuel channel.