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Nuclear data sensitivity for reactor physics parameters in a lead-cooled reactor

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ABSTRACT

Sensitivity of fast reactor physics coefficients to nuclear data differs from thermal reactors. A conceptual lead-cooled reactor has been modelled and assessed for nuclear data sensitivity. A TSUNAMI sensitivity analysis was performed to estimate uncertainty and sensitivity of the neutron multiplication factor to the nuclear data of all nuclides and reactions relevant to the reactor concept. The results were compared against MCNP6, which also employs an adjoint-based method for sensitivity calculations. Sensitivity of the neutron multiplication factor was found to agree closely between MCNP6 and TSUNAMI, for lead cross sections and other important nuclear data. The TSUNAMI assessment of nuclear data impact was also examined through a brief survey of nuclear data evaluations. The evaluations were found to have differences of interpretation that impact the TSUNAMI calculations. The consequent uncertainties in fuel-temperature and coolant-voiding coefficients were also examined in TSUNAMI. Based on the TSUNAMI ranking of important data, a subset of nuclides and reactions were selected for stochastic sampling of cross-section data using NJOY. The sampled cross-section data were then used in SERPENT simulations to assess uncertainties in calculated parameters, which were consistent with the TSUNAMI results.

1. Introduction

Reactors using lead–bismuth coolant were first designed in the Soviet Union in the 1950s (Tarantino et al., 2012) to power nuclear submarines. Further development of the technology was hindered by technical difficulties such as corrosion and the buildup of highly radioactive polonium from activation of bismuth in the coolant. However, lead-cooled reactor concepts have received closer attention from reactor vendors and research institutions in the last two decades. Proposals for experimental reactors (MYRRHA (Engelen et al., 2015) and ELECTRA (Wallenius et al., 2012)), large power plants (BREST-300 (Dragunov et al., 2015), BREST-1200 (Orlov et al., 2005), and Dual Fluid Reactor (IFK Berlin, 2017)), and small modular reactors or SMRs (Gen4 Module (Gen4 Energy, 2017), SSTAR (Smith et al., 2008), and SEALER (Wallenius et al., 2014)) exist at the stages of white paper, conceptual evaluation, material testing, regulator evaluation, or further advancement.

Each new lead-cooled reactor conceptual design is based on advanced technologies, which lack the nuclear power plant operating experience that is ubiquitous for conventional designs such as LWRs and PHWRs. Their limited history of operation presents a challenge to evaluation and approval by regulators and social acceptance, relative to LWRs and PHWRs. As such, reactor design evaluation and acceptance may be premised on the quality, completeness and correctness of the reactor safety analysis. Reactor physics modelling of normal operation and accident conditions is an essential constituent of that analysis, and neutron transport codes are the tools by which the physics is typically modelled. Specific requirements may be imposed for the validation of transport code accuracy against experiment or other sources of data that have a suitable level of accuracy (Shin, 1999).

Safety parameters calculated in neutron transport codes are impacted by the uncertainties in the nuclear data that are used with the codes, in addition to any approximations or other limitations in the methods of simulation and solution on which the codes are based. Quantification of the uncertainty impact on reactor safety parameters is therefore an essential element of rigour in the safety analysis of reactor designs. It is especially important to the acceptability of safety analyses for novel design concepts that lack an extensive operating history, such as reactors that are cooled by molten lead. An appropriate level of effort must therefore be demonstrated in the evaluation of the nuclear data and its impact. By extension, appropriate care must also be shown in the selection of the methodology that frames and focuses the evaluation of the nuclear data.

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Nomenclature			
β_{eff}	Effective delayed neutron fraction	*LWRs*	Light Water Reactors
Λ	Neutron generation lifetime	*PHWRs*	Pressurized Heavy Water Reactors
k_{eff}	Effective neutron multiplication factor	*SMRs*	Small Modular Reactors
LHS	Latin Hypercube Sampling		

The objective of this study was to quantify the impact of nuclear data uncertainties on predictions of reactor physics safety parameters for a conceptual design of a small, modular lead-cooled reactor. It does not constitute the full validation of the calculation codes for those purposes, but provides insight into issues that may be encountered. In the numerical simulation of a reactor design, uncertainties in nuclear data may be propagated to key reactor safety parameters. The present study investigates the propagation of the nuclear data uncertainties to the effective neutron multiplication factor, the fuel temperature reactivity coefficient, and the coolant void reactivity coefficient for the conceptual reactor design.

Two fundamental methods are used to propagate uncertainties from nuclear data to quantities characterizing nuclear systems (Rochman et al., 2011): the perturbation method and the Total Monte Carlo method. The first applies first-order perturbation theory to forward/adjoint flux weighted reaction rates over a multigroup grid. This method is implemented in both the TSUNAMI (Rearden, 2004) and MCNP6 code (Goorley et al., 2016) packages. In the second method, nuclear data uncertainties in the parameters describing basic nuclear models are sampled and propagated, and the overall variance of the reactor physics safety parameters is determined. In the present work, a stochastic Monte Carlo method is used. This method is based on the SERPENT code (Leppänen et al., 2015), with auxiliary software presented in this paper. It propagates nuclear data uncertainties encoded at the level of the nuclear data evaluation file, as opposed to the more fundamental level of the nuclear model parameters.

In the perturbation method, nuclear data sensitivity profiles are calculated for response functions using methods based on first-order perturbation theory. It is used in the TSUNAMI and TSAR modules of the SCALE suite of codes (Foad and Takeda, 2016) to calculate sensitivity matrices as a function of isotope, reaction and energy bin. The sensitivity profiles are then combined with covariance data for the nuclear data evaluation to obtain uncertainties in the response functions, which may include nuclear system safety parameters. The perturbation method has been implemented with the perturbation card of the MCNP code, which can be used to calculate sensitivity coefficients for reactivity responses (Rochman et al., 2011). ERANOS, the European Reactor Analysis Optimized Calculation System for neutronic calculations of thermal and fast reactors, also has tools for uncertainty analysis (Tamagno and van Rooijen, 2013). Use of these codes in nuclear data uncertainty propagation can be found in Tamagno and van Rooijen (2013) and Engelen et al. (2015) for fast reactors and in Diez et al. (2015) and Fiorito et al. (2015) for PWRs.

The Stochastic method presented in this work is based on performing a large number of Monte Carlo simulations using the same nuclear system description but different random nuclear data files for the selected isotopes. The result is a probability distribution from which different moments can be extracted (Rochman et al., 2011). Several software packages and techniques have been developed for uncertainty and sensitivity analyses based on this statistical sampling method (Fiorito et al., 2017) (Wan et al., 2017) (Koning and Rochman, 2012) (Koning and Rochman, 2017) (Zwermann et al., 2014) (Tamagno and van, 2015). These codes and methods have been used to investigate the propagation of nuclear data uncertainties to the reactor safety parameters in PWRs ((Cabellos et al., 2014; Wan et al., 2017)), high-temperature reactors (Tamagno and van, 2015), and lead-cooled reactors (Alhassan et al., 2015). The statistical sampling methodology requires a large number of Monte Carlo simulations, with extensive usage of computational resources.

Based on characteristics of the methodology, TSUNAMI was selected for sensitivity analysis, and to estimate the uncertainty of the neutron multiplication factor due to the nuclear data for nuclides in the conceptual reactor design. MCNP6 was chosen for cross-check the TSUNAMI uncertainty results. The collective results were used to perform a brief survey of the current state of the nuclear data for a lead-cooled reactor concept in the context of its impact on computational uncertainty.

Recent work (Ayres and Eaton, 2015) has showed that polynomial chaos expansions could be used to significantly reduce the computing requirements of Monte Carlo-based nuclear data propagation studies. That motivated the Authors to also perform a limited comparative exercise in which selected data were investigated using the Stochastic Monte Carlo technique.

2. Reactor conceptual design

SMRs are considered an attractive source of energy due to their claims of inherent safety systems and lower total capital cost as compared to larger conventional reactor designs, especially in smaller markets or off-grid installations.

The SEALER (Swedish Advanced Lead Reactor) design is a recent concept for a small modular reactor developed by Dr. J. Wallenius, founder of LeadCold Reactors (Wallenius et al., 2014). It is a fast neutron spectrum reactor, in which the fuel is composed of UO_2 enriched to 19.9%, cooled by molten lead under pressure-driven flow. Long term reactivity control relies on the slow extraction of twelve control-rod assemblies during core operation. The reactor concept targets 3 to 10 MW electrical output, with a core-life span of between 10 and 30 full power years without refueling (at 90% availability) (LeadCold, 2017). The SEALER design effort is focused on off-grid installations, which aligns it with possible applications in the northern regions of Canada. It is physically unlike other reactors operated or designed in Canada, and might present new challenges to existing analysis toolsets or their underlying nuclear data.

The present work is based on open data related to earlier iterations of the SEALER concept (Zhuchkova et al., 2016). Results of this study should be considered applicable to lead-cooled reactor systems in general. They should not be considered representative of real systems designed by LeadCold Reactors.

The conceptual reactor design is structured as a vertical, prismatic core with 85 hexagonal sites. The sites are defined by hexagonal stainless steel wrapper cans, which are mounted onto the bottom grid plate. The 85 component assemblies of the reactor core are installed into the wrapper cans. The core is surrounded by a cylindrical stainless steel core wall that fits closely around the hexagonal lattice of wrapper cans. Dimensions are shown in Table 1.