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Determination of the optimal few-energy group structure for the Canadian Super Critical Water-cooled Reactor

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ABSTRACT

Most deterministic neutron transport codes implement the multi-group approximation theory to solve the linear Boltzmann equation by which the neutron flux and the interaction cross-sections are averaged over discretized energy groups. In order to further reduce the computational cost for large scale and/or time dependent problems, this multi-group information undergoes additional energy group condensation into the so-called few energy group structure which may be used in a broad range of diffusion based full-core analyses. The accuracy and efficiency of few-group based computations is dependent on the number and the boundaries of the discrete energy group structure. Since the flux spectrum used in the homogenization process may not be known a priori and indeed may evolve in space, with burnup, and during transients, the optimal energy group structure depends on reactor type, design, operating conditions, fuel type, and composition. The Canadian Pressure Tube Super-Critical Water-cooled Reactor (PT-SCWR) is a Generation IV advanced reactor system that uses light water above its thermodynamic critical point as coolant and a plutonium-driven thorium fuel mixture. Considering that the anticipated flux spectrum for such a design deviates significantly from the thermal-neutron dominated CANDU designs, there may be a need for improvements in the number and boundaries of the few-group nuclear data. This paper presents a systematic methodology for the delineation of few-group energy structure for the Canadian PT-SCWR. The methodology used the SCALE (Standardized Computer Analysis for Licensing Evaluation) code package to examine the effect of energy group homogenization and determine the structure which minimizes the sensitivity of the results to energy group partitioning. As such, it utilizes normal and off-normal operating conditions to determine the effect of energy homogenization and determines the minimal group of energy boundaries which can accurately capture the lattice physics phenomena within the lattice cell over a wide range of operating and accident conditions.

1. Introduction

The Generation IV International Forum (GIF) has identified six advanced reactor systems to be developed through an international collaboration of research and development (R&D) (U.S. DOE Nuclear Energy Research Advisory Committee, 2002). GIF has proposed four main goals for Generation IV advanced energy systems including enhancements in safety, economics, proliferation resistance, and sustainability (U.S. DOE Nuclear Energy Research Advisory Committee, 2002). The Canadian Pressure Tube Super-Critical Water-cooled Reactor (PT-SCWR) is a pressure tube design that uses supercritical light water as coolant and a separate low-pressure and temperature heavy water moderator (Leung et al., 2011).

Some of the key distinctions of the Canadian PT-SCWR from the fleet of the well-established CANDU reactors are the high-temperature supercritical light water coolant, the vertically orientated re-entrant channel, and the enriched plutonium driven thorium fuel mixture. The Canadian PT-SCWR is also characterized by a tight lattice pitch (25 cm lattice pitch) which results in an under-moderated lattice cell with a direct impact on the neutron energy spectrum (Moghrabi and Novog, 2016). Moreover, the fuel channel design features a High-Efficiency Re-entrant Channel (HERC) where the coolant flows downward through the central flow tube and then reverses direction to flow upward around the fuel elements to the channel outlet (Yetisir et al., 2013; Pencer et al., 2013; Pencer and Colton, 2013). Such design is characterized by strong variations in coolant density and temperature along the vertical fuel assembly. With such substantial changes the

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neutronic behavior is expected to be different from the conventional CANDU reactors. Consequently, many studies have been performed to investigate the important lattice physics phenomena (Moghrabi and Novog, 2016, 2017b,c) and for benchmarking lattice level neutronic behavior (Sharpe et al., 2015).

Deterministic physics codes solve for neutron transport within a reactor core using a number of approximations (Lewis and Miller, 1993). In general, the reactor physics calculations are carried out in three stages. First nuclear data from an established source (e.g., ENDF VII.1) are obtained and processed using a code such as NJOY in order to generate a large number of discrete energy-dependent cross sections (e.g., the 238-group library included in the SCALE code package). Such generic and large energy-group structure are often referred to as multi-group libraries. Secondly, transport simulations are performed utilizing this fine-group nuclear data for a specific lattice geometry. From these results, a further energy condensation is performed wherein the transport solution (flux and reaction rates) are used to derive equivalent macroscopic cross sections for a much smaller number of energy groups and for a homogenized region of space (i.e., the lattice cell). Third, these few-group homogenized cross sections are then passed to a core-level diffusion code such as PARCS or DONJON for full-core steady state and/or transient calculations. Such a procedure is applied in deterministic-full-core analysis packages such as SCALE (Laboratories, 2011) followed by PARCS (Downar et al., 2012) or DRAGON (Marleau et al., 1994) followed by DONJON (Varin et al., 2005). Although many deterministic lattice level codes are capable of generating the few-group homogenized cross-sections automatically, users are required to provide the energy boundaries for the few-group structure. Such few-group structure depends on the problem (reactor design, fuel type, coolant and moderator properties) and is often determined based on experience. In reactor physics analysis of thermal reactors, a general approach is to split the neutron energy spectrum into thermal and fast ranges with a cut-off energy of 0.625 eV. However, the energy barriers are not universally fixed and vary depending on the problem/model specifications. The proposed changes in Generation IV designs have driven renewed interest in the optimization of the few-energy group domain (Yi and Sjoden, 2013; Aristodemou et al., 2006; Akbari et al., 2012; Druska et al., 2009). This paper investigates the optimum few group-energy structure for the Canadian PT-SCWR 64-element design with fresh and depleted fuel.

Previous studies have suggested that the traditional two-energy group approach may not be sufficient to capture the steady and transient phenomena in the Canadian PT-SCWR given the fuel enrichment, flux spectrum, complex geometry and feedback behavior (Moghrabi and Novog, 2016, 2017b,c; Shen, 2012). It is also desirable that such a structure show minimal numeric sensitivity to boundary selection and/or group subdivision, a facet often referred to as discretization convergence in numerical methods literature (Fleming et al., 2005). For example, the shift in a boundary or the subdivision of energy groups should not cause an unreasonable change in the accuracy of a solution relative to the multi-group case, i.e., it is desirable to minimize the sensitivity of the solution to the movement of, or the addition/subtraction of a boundary. A proposed structure should also examine the effects of fuel depletion, delayed neutron effects, fuel temperature and coolant/moderator perturbations such that it captures such phenomena with acceptable accuracy as compared to the multi-group solution. In addition, the accuracy of the few-group boundaries needs to be evaluated against a consistent basis such as to avoid cancelation of errors. In particular, boundaries can be numerically selected as optimal in order to cancel a bias resulting from another boundary placed at a different energy. Such selection would then erroneously allow for boundaries with high bias, but opposite sign, as compared to the multi-group solution. Finally, selection of boundaries solely to eliminate a code-to-code bias between the diffusion code and a transport code should be avoided since such biases are code dependent, a function of fuel conditions, and may evolve in the course of a transient. As such it is desirable to assess potential boundaries within a single computational tool such as SCALE, rather than assessing behavior using full-core diffusion codes since the diffusion calculations are subjected to many more approximations (leakage, assembly discontinuities, etc...).

2. Materials and modelling methods

2.1. Description of the fuel channel and core geometry

The Canadian PT-SCWR core contains 336 pressurised fuel channels with 5 m long vertical fuel assemblies and zirconium modified stainless steel cladding (Pencer and Colton, 2013). The most recent design of the Canadian PT-SCWR 64-element fuel assembly geometry (Pencer et al., 2013; Pencer and Colton, 2013) with material atomic densities and geometry specifications were used in this study (Hummel, 2015). The coolant flow is characterized by a bi-directional flow path within the fuel channel (Pencer et al., 2013; Pencer and Colton, 2013). The coolant enters the inlet plenum at 350 °C and 25.8 MPa and flows downward through the central flow tube within each fuel channel, then reverses direction at the bottom of the fuel channel and flows upward through fuel containing outer assembly. The coolant exits the channel at 625 °C and 25.0 MPa (Fig. 1). The fuel is arranged in two concentric fuel rings of 32 elements each that are 12% and 15% by weight PuO_2 in ThO_2 in the outer and inner rings, respectively, creating a balanced radial power distribution (Pencer and Colton, 2013).

The PT-SCWR shares characteristics with the traditional Pressurized Heavy Water Reactor (PHWR) and Boiling light Water Reactors (BWR) and operates above the thermodynamic critical pressure. Consistent with PHWR designs the SCWR has a separate high pressure coolant and low pressure moderator. The low-temperature and low-pressure moderator is thermally isolated from the hot coolant through a ceramic insulator within the pressure tube. Similar to BWRs, the turbine is directly coupled to the reactor coolant outlet. The PT-SCWR utilizes a batch fuelled vertical core with enriched fuel similar to Light Water Reactor (LWR) technologies.

2.2. Description of computational codes

One of the primary goals in nuclear reactor analyses is to provide an accurate estimate of the neutron density in the nuclear reactor which itself is a function of time, space, energy, and direction. For practical reasons, traditional reactor analyses have used multi-scale methods wherein simulations at the lattice-level scale with high spatial and energy fidelity are used to derive inputs for full-core space-time kinetics calculations at lower spatial resolution. In this study, the lattice transport calculations were performed using the NEWT (Dehart and Jessee, 2011) code which is part of the SCALE (Standardized Computer Analysis for Licensing Evaluation) 6.1.3 package (Oak Ridge National Laboratories, 2013). Lattice cell simulations were performed using the 238 energy group ENDF/VII.0 library distributed in SCALE version 6.1.3. Fuel burnup simulations were performed with the TRITON (Jessee and Dehart, 2011) module that employs NEWT as a multi-group solver. NEWT is a deterministic transport solver that uses the discrete ordinate approach to solve the neutron transport equation. However, the discrete ordinate approach is difficult to apply directly to complicated non-orthogonal geometries due to the nature of the finite difference approximations. Consequently,