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SCALE and SERPENT solutions of the OECD VVER-1000 LEU and MOX burnup computational benchmark

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ABSTRACT

The loading of hybrid cores with Mixed Uranium Plutonium Oxide (MOX) and Low Enriched Uranium (LEU) fuels in commercial nuclear reactors requires well validated computational methods and codes capable of providing reliable predictions of the neutronics characteristics of such fuels in terms of reactivity conditions (k_{inf}), nuclide inventory and pin power generation over the entire fuel cycle length. Within the framework of Joint United States/Russian Fissile Materials Disposition Program an important task is to verify and validate neutronics codes for the use of MOX fuel in VVER-1000 reactors. Benchmark analyses are being performed for both computational benchmarks and experimental benchmarks. In this paper new solutions for the (UO_2 + Gd) and (UO_2 + PuO_2 + Gd) fuel assemblies proposed within the "OECD VVER-1000 Burnup Computational Benchmark" are presented, these being representative of the designs which are expected to be used in the plutonium disposition mission. The objective is to test the SERPENT and SCALE codes against previously obtained solutions and to provide new reference solutions to the benchmark with modern nuclear data libraries.

1. Introduction

Mixed oxide (MOX) fuel has only been used on a large scale in the nuclear industry since the 1980's and to a much smaller extent with respect to the more conventional UO_2 fuel. In particular, the irradiation of MOX fuel in existing commercial nuclear reactors (essentially LWRs) as a disposition method for surplus plutonium from the weapons programs is being pursued by the United States and the Russian Federation within the framework of a mutual agreement (Gehin et al., 2004). Due to the relatively scarce experience with MOX fuel in these two countries, particularly if compared with the one accumulated in Europe and Japan, an international Expert Group has been established at the OECD/NEA to facilitate the sharing of existing information and experience in the physics and fuel behavior of MOX fuel as it relates to the disposition of weapons-grade (WG) plutonium. The Russian federation is pursuing the deployment of WG MOX fuel in the VVER-1000 reactors and R&D activities are needed for the certification of the calculation codes which are envisioned to be used. As part of the efforts performed by the OECD/NEA Expert Group to support this certification process, a burnup computational benchmark exercise has been launched in 2000 (Kalugin et al., 2002) based on the prevailing concept which utilizes U–Gd fuel pins to provide an effective means of introducing burnable absorbers into the MOX assemblies. This is a standard problem for VVER-1000 core physics in which two assemblies are considered, namely a Low Enriched Uranium (LEU) fuel assembly and a MOX fuel assembly, and which will provide a good indication of the current computational methods.

In the present work the VVER-1000 benchmark test cases have been solved by means of the SCALE (ORNL, 2011) and SERPENT (Leppänen, 2013) codes. The SCALE (Standardized Computer Analyses for Licensing Evaluation) code is a software package developed at Oak Ridge National Laboratory (ORNL) which provides a comprehensive, verified and validated tool set for criticality safety, reactor physics, radiation shielding and sensitivity and uncertainty analysis. The SERPENT code is a three-dimensional continuous-energy Monte Carlo reactor physics burnup calculation code, developed at VTT Technical Research Centre of Finland since 2004. The objective of our study is to provide new solutions with modern nuclear data libraries (NDLs) for the VVER-1000 MOX and LEU computational benchmark. This study allows us to validate the SCALE calculation schemes for VVER-type reactors and to compare deterministic solutions with Monte Carlo ones at steady state.

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2. Benchmark description

2.1. Benchmark models

The benchmark exercise consists of two different fuel assemblies: a uniform LEU fuel assembly with 12 U–Gd rods (LEU variant) and a profiled MOX fuel assembly with 12 U–Gd rods (MOX variant). These assemblies are typical of the advanced designs under active development in Russia for the VVER-1000 reactors and similar to the designs which are expected to be used in the plutonium disposition mission. The assemblies under investigations are hexagonal in design and consist of one central tube, 312 fuel pin locations (12 of which are U/Gd rods), and 18 guide tubes. The clad and structural materials are composed of a Zr–Nb alloy. The LEU assembly (Fig. 1) consists of fuel rods with 3.7 wt.% enrichment. The 12 U–Gd pins have an enrichment of 3.6 wt.% and a Gd_2O_3 content of 4.0 wt.%. The MOX assembly (Fig. 2) contains fuel rods with three different plutonium loadings. The central region contains MOX pins with 4.2 wt.% fissile plutonium (consisting of 93 wt.% ^{239}Pu), two rings of fuel rods with 3.0 wt.% fissile plutonium, and an outer ring of fuel rods with 2.0 wt.% fissile plutonium. The 12 U–Gd rods are in the same locations as in the LEU assembly configuration and have the same design. The geometrical characteristics of the fuel cells for the LEU and MOX cases are identical and they are provided in Fig. 3. Three different moderator materials are used in the benchmark, namely MOD1, MOD2 and MOD3, these representing light water with different densities, temperatures and boron concentrations.

The benchmark requires solutions for a variety of reactor states, these covering both cold and operational conditions as indicated in Table 1. According to the benchmark specifications, depletion calculations are requested for State 1 up to 40 MWd/kgHM along with several branch calculations at specific burnup points for state S2–S5 using the isotopic compositions from the burnup calculations of state S1. The requested results include k_{inf} values, pin-by-pin fission rate distributions according to the cell numeration depicted in Fig. 4 and nuclide concentrations for the ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{155}Gd, ^{157}Gd, ^{149}Sm, ^{135}Xe isotopes in cells 1 and 24 as well as averaged over the whole assemblies.

Several solutions of the benchmark are available in literature, each of which using different methods and combinations of nuclear data libraries (Kalugin et al., 2002; Petrov et al., 2013; Thilagam et al., 2009). Three of the solutions have been obtained by means of Monte-Carlo methods, while the remaining ones are based on collision probability (or similar) methodologies.

3. Codes, data and methods

The deterministic calculations presented in this paper have been performed with the lattice physics capabilities of the SCALE 6.1 code system. In particular the TRITON sequence has been used, which couples the discrete-ordinates code NEWT (New ESC-based Weighting Transport code) to the depletion code ORIGEN. Cross-section self-shielding is carried out by the BONAMI and CENTRM solvers in the unresolved and resolved resonance regions respectively. The CENTRM module performs transport calculation using data on an ultrafine energy grid to generate effectively continuous energy neutron flux solutions in the resonance and thermal ranges. This is used to weight the multi-group cross-sections to be utilized in the subsequent transport calculations which are performed with the NEWT code. In our analysis the 238 energy groups structure implemented in the SCALE system has been used and a TRITON model of the hexagonal fuel assemblies described by the benchmark specifications have been built. As far as the spatial discretization, in the NEWT model the cylinder surfaces are approximated with a 12-sided regular polygon. Furthermore, the adopted level of symmetric quadrature is set to 6 (SN6).

All the Monte Carlo calculations have been performed using the beta version 2.1.22 of the SERPENT code and results have been obtained by simulating 4.0E+7 neutrons distributed over 400 cycles. The Shannon entropy criterion has also been applied and for the correct convergence of the fission source the first 100 criticality cycles have been skipped. The correspondent statistical errors are in the order of 8E−5 and 2E−3 for k_{inf} and pin power respectively. The SERPENT code uses built-in calculation routines for the burn-up calculation which is usually divided into two steps (Pusa and Leppänen, 2010). The first step is the transport cycle in which the rates of all neutron-induced transmutation reactions are calculated. These data are then combined with radioactive decay constants and fission product yields read from nuclear data libraries. In the second stage the Bateman equations describing the isotopic changes in the irradiated materials are solved by means of

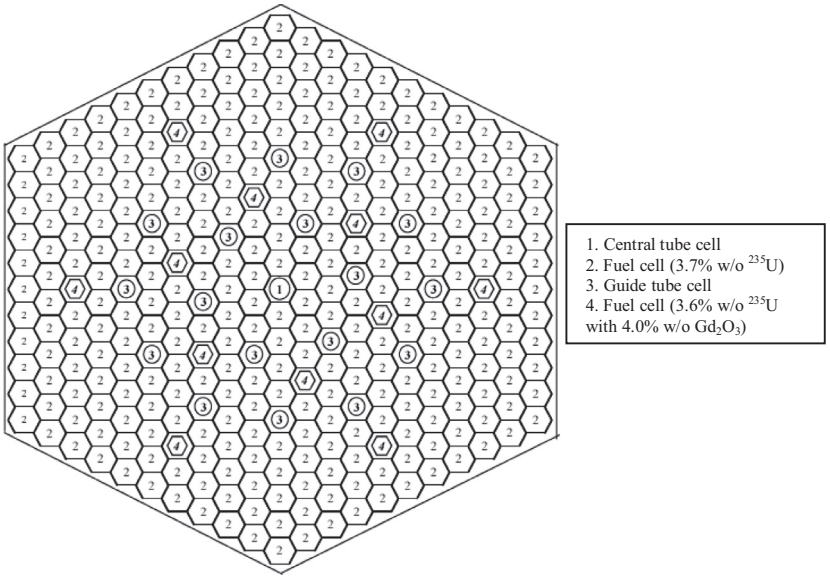

Fig. 1. LEU assembly configuration (Kalugin et al., 2002).