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Computational model for thermohydraulic analysis of an integral pressurized water reactor with mixed oxide fuel (Th, Pu)O_2

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1. Introduction

The use of nuclear energy has been one of the best alternatives to supply electricity on a large-scale and reduce carbon dioxide emissions into the atmosphere. On the other hand, some challenges emerged, such as the storage or disposal of long-lived radioactive waste and the proliferation of nuclear weapons. The use of advanced generation III+ and IV nuclear reactors, and their applications, has become important, seen as a means capable of contributing to the global transition to more sustainable, accessible and reliable energy systems. This technology, which could be integrated into future carbon-free electricity generation systems and with high proportions of different renewable energy sources, includes small modular reactors (SMR). According to the IAEA, Small Nuclear Reactors are those nuclear reactors with electrical power less than 300 MW(e) [1]. In recent years, the need for proliferation resistance, longer fuel cycles, increased burning, reduced plutonium stores and in situ use of replicated fissile material has led to renewed interest in thorium-based fuel cycles. The thorium fuel cycle is an attractive way to produce long-term nuclear energy with low radioactive waste. ThO_2 has favorable thermophysical properties due to its higher thermal conductivity and lower coefficient of thermal expansion compared to UO_2. Therefore, fuels based on ThO_2 are expected to perform better than mixed oxides based on UO_2. Th-based fuel cycles have intrinsic resistance to proliferation due to the formation of 232U through reactions (n,2n) with 232Th, 233Pa and 233U [2]–[4]. This work focuses on the analysis of a small modular reactor (SMR) of the integral pressurized water (iPWR) type with using thorium-plutonium oxide (Th-Pu MOX) fuel mixtures. The main objective is to develop a computational model of the critical section of the core, where the greatest power is produced, that allows the calculation of the most important thermohydraulic parameters. Using the developed model, the temperature and power profiles released in the hottest channel, the temperature profiles in the envelope and the temperature and density profiles of the water in the cooling zone are calculated.

2. Methodology

The conceptual design of reactor core analyzed was based on the characteristics of the iPWR-type SMR, which has a nominal power of 180 MW (e) per module [5]. The reactor core consists of 69 fuel assemblies standard Westinghouse Company, loaded into a 21.5 cm square lattice pitch, which has a mirror symmetrical configuration octant. Each fuel assemblies contain 264 fuel rods, 24 guide tubes for control rods and an instrumentation tube in the center [1], [6]. In [5], a feasibility assessment of the use of MOX (Th,Pu)O_2 mixtures in the core of an iPWR was studied, considering the core partially loaded with MOX fuel, to ensure an extended cycle of 48 months. One third of the core (24 fuel assemblies) has been loaded with MOX fuel, while the rest has UO_2 fuel (*Figure 1*). Plutonium, obtained from the first recycling of an irradiated fuel originally composed of slightly enriched UO_2, is recycled once into a mixture with Thorium, 13.14% PuO_2 and 86.86% ThO_2. For the neutronic calculations in [5], the probabilistic methods implemented in the SERPENT code (version 2) were used. Serpent code allows you to calculate the axial power distributions in both the fuel assemblies and the subchannels, which will be used in the thermohydraulic calculation to obtain the temperature profiles [7]. For the subchannel axial power calculations, one million stories and 1500 cycles was used; and each fuel element was divided into 100 axial regions of 0.024 m.

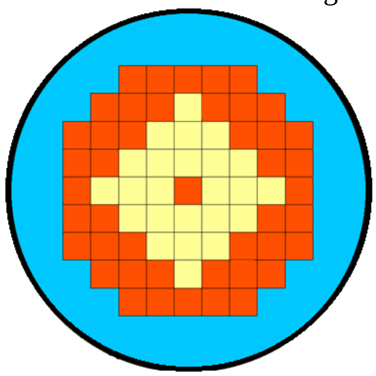

Figure 1. MOX fuel distribution in the core.

Figure 2 shows the radial distribution of the core peak power factors at quarter symmetry at the beginning of the cycle (BOC). The highest power fuel assembly in the core, with a peak factor of 1.39, corresponds to the position of an UO_2 fuel assembly. Among the assemblies loaded with MOX (Th, Pu)O_2 fuel, the one with the highest power has a peak factor of 0.915 [5]. Since both fuels have different behavior for their thermophysical properties, a thermohydraulic study of the subchannel that generates more power in the core was carried out (corresponds to fuel UO2), and a study of the subchannel that generates more power for assemblies with fuel MOX (Th, Pu)O_2. *Figure 2* shows power density distribution calculated by Serpent code for the subchannels to be analyzed.

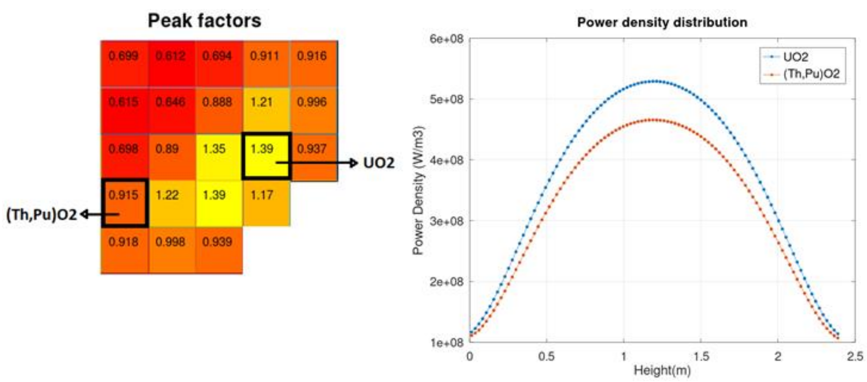

Figure 2. The radial distribution of the core peak power factors for quarter symmetry and power density distribution at the beginning of the cycle (BOC)

Table I shows the main thermohydraulic parameters used in the model to carry out the thermohydraulic study of the hottest section of the core. The thermohydraulic calculations were performed with Ansys CFX. Correlations for thermophysical properties of materials were taken from [8]

Table I. Thermohydraulic parameters.

Parameter	**Value**
Thermal Power	530 MW
Pressure	14.8 MPa
Inlet temperature	290 ºC
Outlet temperature	318.8 ºC
Core mass flow	3345 kg/s

Taking advantage of symmetry conditions to simplify calculations and optimize available computational resources, the hottest subchannel of the highest power fuel assemblies was simulated. The dimensions of the subchannel designed by Westinghouse were taken from [9]. The geometry was decomposed into several parts to obtain a more structured mesh and optimize the computational resources used. With a mesh of 3 876 000 elements and 4 202 801 nodes, a mesh-independent solution is reached. In *Figure 3*, the cross-section of the subchannel is shown. The mesh is generated by the multizone method. To check the quality of the mesh, an evaluation was performed using the quality indicators Skewness, Aspect Ratio and Orthogonal Quality. For the Skewness parameter, a maximum value of 0.5 was obtained. For the Aspect Ratio parameter, 28.66 was obtained for the maximum. The last and main parameter evaluated was Orthogonal Quality, resulting in a value of 0.632 for the minimum. The mesh obtained for the model meets the requirements of the quality indicators, ensuring the reliability of the results.

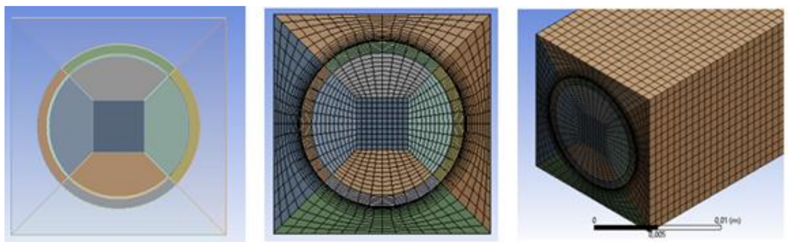

Figure 3. Isometric and superior view of the fuel subchannel geometry and mesh used in the domain discretization.

3. Results and Discussion

With the axial distributions of the power density calculated by the SERPENT code, the thermohydraulic study of the subchannels was carried out. The present study was made at the beginning of the cycle (BOC), that is, with fresh fuel. As a first result, it was possible to obtain the distribution of the outlet coolant temperature. It is observed that the highest temperature is obtained in the area close to the cladding, where heat transfer from the cladding to the water occurs. The water temperature has a variation in this section of 587.07-594.84 K for the MOX fuel subchannel, and a variation in this section of 590.978-598.56 K for the UO2 subchannel. In *Figure 4* the axial water temperature in the casing wall and the axial temperature at the center of the fuel for both subchannels are shown. The maximum water temperature reached is 610.326 K for MOX (Th, Pu)O_2 and 613.3 K for UO_2, lower than the saturation temperature (614.23 K) for the refrigerant pressure. The maximum fuel temperature reached is 1622.24 K for the MOX (Th, Pu)O_2 fuel, well below the melting point (3651 K), and 1544.54 K for the UO_2 fuel, below the melting point (3120 K). *Figure 5* shows the temperature radial variation. There is a sharp drop in temperature as the radius of the fuel element increases, especially in the gap area (helium). The maximum temperature found in the clad was 655.83 K for MOX (Th, Pu)O_2 and 667,024 K for UO_2, well below 1477.59 K, reported as a limit value for accident cases.

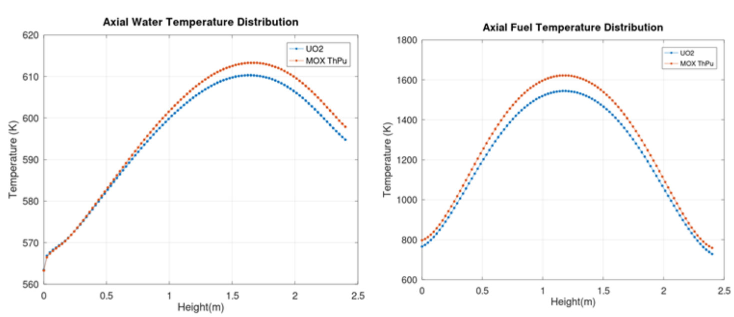

Figure 4. Axial water temperature in the casing wall and the axial temperature at the center of the fuel for both subchannels.

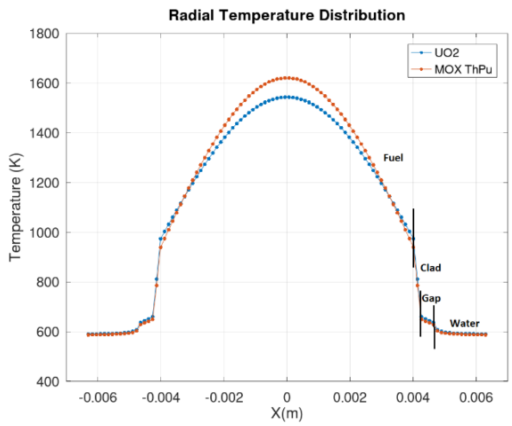

Figure 5. Radial temperature distribution.

4. Conclusions

A three-dimensional model was developed and implemented using the Computational Fluid Dynamics code to evaluate the thermohydraulic behavior of a critical SMR reactor fuel assembly using a mixture of uranium and thorium oxides (Th, Pu)O_2. The axial power distribution was obtained in the critical subchannel of the reactor core, and in the highest power subchannel using MOX fuel, for neutronic calculation was used the Serpent code. The temperature profiles in the subchannels were calculated. It was observed that the MOX fuel reaches a maximum temperature of 1622.24 K being below the melting point (3651 K), while the UO_2 fuel has a maximum temperature of 1544.54 K for the UO_2 fuel, below the melting point (3120 K). The maximum temperature reached by the cladding was also evaluated and a value of approximately 655.83 K and 667,024 K was obtained for MOX (Th, Pu)O_2 and UO_2, respectively, showing a good result against the value of 1477.59 K, reported as a threshold value for accident cases.

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