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Uncertainty and sensitivity analyses for fuel temperature evaluations of U-Mo/Al plate-type dispersion fuel

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ABSTRACT

U-Mo/Al plate-type dispersion fuel is a promising candidate for the conversion of research reactor fuels from highly enriched to low-enriched uranium due to its high uranium density. The fuel temperature is a very important parameter, as it affects the performance of the fuel through various aspects, such as the formation of an interaction layer (IL) between the fuel particles and the matrix, swelling, and the release of fission gas. For these reasons, the fuel temperature as a function of the fission density was calculated for two representative heat flux profiles using best-estimate values and Monte Carlo simulations. Uncertainty and sensitivity analyses which utilized the uncertainties of the critical parameters were then conducted to determine the upper (maximum) and lower (minimum) bounds of the fuel temperature for the selected heat flux profiles. The uncertainty analysis used common uncertainty propagation approaches and a probabilistic sensitivity analysis (Monte Carlo simulation), randomly sampling numbers following a Gaussian distribution. Lastly, the Pearson correlation coefficient was used to identify the input uncertainties which influence the fuel temperature most in the sensitivity analysis. These analyses contribute to safety analyses and to the licensing process, as they are used in best-estimate approaches that apply realistic assumptions complemented with uncertainty analyses, such as the Best Estimate Plus Uncertainty (BEPU) approach.

1. Introduction

A large number of high-performance research reactors around the world still use highly-enriched uranium (HEU) to provide the high neutron fluxes that are required for their application. However, both apprehensions over the use of HEU and efforts to enhance the proliferation resistance of fuel cycles have simultaneously increased. Accordingly, several programs have been initiated in order to reduce the proliferation risks and threats due to the use of HEU, one example being Reduced Enrichment for Research and Test Reactors (RERTR). These programs aim to replace HEU fuel in research reactors with low-enriched uranium (LEU) fuel without sacrificing the performance of the reactors (Van den Berghe and Lemoine, 2014; Kim, 2012). To achieve this goal, the fuel loading must be increased or a fuel with a higher uranium density must be utilized. However, given that the maximum achievable fuel loading is limited, there is a need for a new uranium alloy fuel with a higher uranium density to compensate for the decreased enrichment (Cho et al., 2017). U-Mo/Al plate-type dispersion fuel is a promising candidate for fuel conversion from the HEU to the LEU type. It provides acceptable performance, a high uranium density level, and good irradiation stability under low and intermediate heat flux conditions (Kim, 2012). U-Mo/Al dispersion fuel has an advantage in that the fuel thermal conductivity is proportional to the amount of aluminum in its matrix (Kim et al., 2015; Burkes et al., 2015), which has significantly high thermal conductivity.

The most common geometry of U-Mo/Al dispersion fuel is the plate type. The plate geometry is weaker than that of the rod in terms of the structure, but it performs better in terms of heat transfer. Fig. 1 shows a schematic cross-section of plate-type U-Mo/Al dispersion fuel.

Generally, the temperature of the nuclear fuel is a significant factor, as it mainly influences the overall fuel performance. Particularly for U-Mo/Al dispersion research reactor fuel, the temperature profile controls the fuel integrity and degradation, as it accelerates the formation of an interaction layer (IL) between the fuel particles and the matrix and controls the kinetics of the IL growth process (Kim et al., 2013). The interaction layer, in turn, degrades the thermal conductivity of the fuel. In addition to the IL, other changes during operation, such as swelling, is affected by the fuel temperature, and vice versa (Kim et al., 2015; Wachs,

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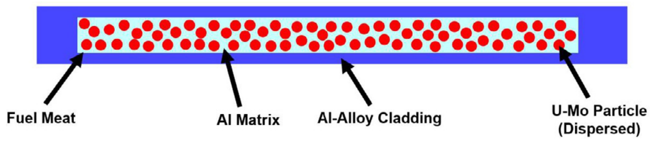

Fig. 1. U-Mo/Al dispersion fuel.

2016). Therefore, precisely evaluating the probable temperature ranges where the fuel operates is a critical aspect during a fuel performance evaluation.

Recently, many studies, including papers by Burkes et al. (2016) and Tahk et al. (2013), have evaluated the thermal behavior and the temperature profile of U-Mo/Al plate-type dispersion fuel, taking into account all of the parameters needed and phenomena which may arise with regard to the fuel. These studies utilized the nominal values of parameters and applied them in the governing heat transfer equations. However, the effects of uncertainty in these parameters were not taken into account in these studies, and no detailed analyses of the uncertainties and reactor-related parameters are provided for this type of fuel. As the uncertainties of these parameters change the fuel temperature profile significantly, uncertainty and sensitivity studies are required to evaluate the effect of uncertainties on the fuel temperature and to obtain the maximum fuel temperature.

Such uncertainty and sensitivity studies are a major concern with regard to safety analyses, specifically the computational safety analyses that are currently used in the licensing process. Simulation of operational scenarios are among the key aspects of safety analyses of nuclear reactors. These types of simulations are typically done using best-estimate (BE) approaches. Recently, the global trend with regard to nuclear reactor safety analysis and licensing has shifted from the use of conservative approaches, which are dependent on conservative assumptions of boundary and initial conditions, to the use of BE approaches with realistic assumptions alongside an uncertainty analysis. This is commonly referred to as the Best Estimate Plus Uncertainty (BEPU) approach. The BEPU approach provides reliable safety margin calculations, secures greater effectiveness of the reactor, and eliminates any unnecessary conservatism, as it utilizes the relevant uncertainties of the key parameters, 95% confidence intervals, and the corresponding sensitivity analyses coefficients (Yum and Park, 2017). In this regard, the U.S. Nuclear Regulatory Commission (U.S. NRC) regulations have been revised, and the IAEA has published supporting documents pertaining to the procuring of operating licenses using final safety analysis reports based upon BEPU for nuclear power plants as well as research reactors (Yum and Park, 2017; International Atomic Energy Agency, 2008).

The aim of this study is to evaluate the fuel temperature profile as a function of the fission density using the best-estimate values followed by the application of the BEPU methodology, through uncertainty and sensitivity analyses, to determine the upper and lower bounds of the fuel temperature and to identify the parameters with the greatest effects on the fuel temperature. Therefore, in this paper, several parameters associated with neutronics, thermal hydraulics, materials, and fabrication processes were used to calculate the fuel temperature profile. The uncertainties of these parameters, on the other hand, were utilized to evaluate the effects of uncertainties on the fuel temperature ranges by calculating the probable maximum and minimum temperature bounds. Finally, the relative impact of each parameter on the fuel temperature was estimated by a sensitivity analysis.

Although some of the required values, profiles, and uncertainties of the parameters are scarce, the analyses in this study are performed using all available reliable data pertaining to U-Mo/Al plate-type dispersion fuel parameters. IAEA documents alongside the ATR Full-size plate In-center flux trap Position (AFIP-1) experiment of the RERTR program – through its irradiation summary report (INL/EXT-11-22045 et al., 2011) – are the main sources of the fuel data and its operational conditions. The AFIP experiments were designed to demonstrate the performance of second-generation U-Mo/Al dispersion fuels (INL/EXT-11-22045 et al., 2011). The report from the AFIP-1 experiment summarizes the experiment in terms of irradiation, safety, neutronics and hydraulics analyses, and thermal analyses results. Thus far, these experiments are most representative of the operation conditions of U-Mo/Al dispersion fuel and are most reliable in terms of data and results.

It is worth mentioning that the AFIP-1 report is used to provide the fission density and heat flux conditions for the fuel plate, which were also determined using MCNP. However, the AFIP-1 report does not discuss the uncertainty associated with the calculations, nor the factors that fed the MCNP as-run calculations. However, as the calculation of the heat flux relies heavily on the fission rate density estimation by MCNP, the possible MCNP calculations uncertainties are taken into account in the surface heat flux uncertainty calculations (Section 3.3).

The structure of the paper is as follows. Section 2 discusses the general methodology of the analyses, the parameter selection process, and the required equations. Data mining of the values, profiles, functions, and uncertainties of the selected parameters is described in Section 3. Sections 4 and 5 correspondingly present results and a discussion of the fuel temperature calculations, uncertainty propagation, and the sensitivity study. These sections are followed by the concluding remarks in Section 6.

2. Methodology, parameter selection, and equations

2.1. General methodological scheme

To evaluate the fuel temperature, a typical systematic procedure was followed. First, the governing heat transfer equations for the fuel temperature calculation were reviewed and determined, after which the required values and profiles of the parameters as functions of fission density (burnup) were collected and used in the equations to calculate the fuel temperature profile.

Realistic heat flux profiles were used in the analyses. Two representative heat flux positions from the AFIP-1 experiment data, referred to as the high and low heat flux profiles, were selected for this purpose. The high heat flux case is represented through the flux of a sample located 4.5 cm from the bottom of the fuel plate (1T2/A-6) in Advanced Test Reactor (ATR). On the other hand, the low heat flux profile is presented through the flux of a sample located 23 cm from the top of the fuel plate (1T2/A-3) (INL/EXT-11-22045 et al., 2011). The methodology used to obtain these two heat flux profiles is discussed thoroughly in Section 2.3. Table 1 shows the fuel plate specifications of the AFIP-1 experiment.

Before conducting the uncertainty and sensitivity analyses, data mining of the parameters' uncertainties was done, as shown in Section 3. Thereafter, the propagation of the uncertainty and sensitivity analyses was carried out to determine the upper and lower bounds of the fuel temperature and to ascertain the relative impact of the potential parameters. These analyses utilize a common methodology, which includes the application of uncertainty propagation, Monte Carlo simulation, 95% confidence intervals, and the use of a sensitivity analysis coefficient. This methodology has widely been used for nuclear power plants and research reactors for safety analyses and licensing, including the Best Estimate Plus Uncertainty (BEPU) approach, as was previously discussed. For instance, Cabellos et al. (2014) and Helgesson et al. (2017) utilized