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Validation of a fuel management code MCODE-FM against fission product poisoning and flux wire measurements of the MIT reactor

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ABSTRACT

The Massachusetts Institute of Technology Research Reactor (MITR), with thermal power of 6 MW, is one of the five high-performance research reactors operated in the U.S. An in-house program MCODE-FM (MCNP-ORIGEN Coupled Depletion Program – Fuel Management) as the reference code supporting routine operation is currently developed by the MITR staff. Adopting Monte Carlo methods enables exact modeling of the MITR core geometry with use of continuous-energy nuclear data. A criticality search algorithm to track control blade movement is implemented in MCODE-FM. The code also features automation of input file generation, data manipulation, and post-processing of output data for fuel cycle analysis. Some verification and validation runs have been carried out by comparisons with the multiplication factor and the reactivity worth induced by control blades. In this study, further validation runs are performed, with two sets of measured data: 1) reactivity effects of fission product poisoning and fuel depletion and 2) reaction rate based thermal and fast neutron flux. Fission product poisoning effects calculated by MCODE-FM are in good agreement with the measured data obtained during the reactor start-up process (within 10%). The trend of peak xenon reactivity after the reactor shutdown is predicted accurately as well. As compared to the neutron activation experiment, MCODE-FM is found to slightly over-estimate the thermal flux with less than 10% discrepancy and the fast neutron flux with less than 20% discrepancy.

1. Introduction

The Massachusetts Institute of Technology Research Reactor (MITR) has been in operation since 1958. The original version, MITR-I, was a heavy-water moderated and cooled nuclear research reactor. After a reevaluation of needs and further core optimization studies, the current reactor design, MITR-II, underwent a major upgrade and began operation in 1976. It is moderated and cooled by light-water and has a heavy-water reflector (Newton, 2006). MITR-II uses rhomboid-shaped fuel elements (see Fig. 1 left). There are 27 in-core positions for fuel elements and/or irradiation experiments (see Fig. 1 right). These positions are divided into three radial rings with 3, 9 and 15 rhomboid-shaped areas in each of them. The edge-to-edge distances of the three concentric hexagons of the fuel region are 12.4 cm, 25.5 cm, and 38.4 cm, respectively. Each fuel element contains 15 fuel plates, which consist of ~93% enriched uranium sandwiched between sides of aluminum cladding (Newton et al., 2007). The fuel meat is 55.9 cm high, 5.3 cm wide, and 0.076 cm thick. The nominal thickness of cladding is 0.038 cm, with longitudinal fins running down the length of each plate to improve heat transfer. There are six control blades and one regulation rod located at the core periphery. They are withdrawn and/or inserted to maintain criticality. All the above-mentioned components are contained in the core tank, which has a cylindrical shape, 52 cm in diameter and 73 cm in height. There is a heavy water reflector surrounding it from the sides and the bottom (see Fig. 2). In addition, a light-water plenum sits above the core. The thermal power of the MITR-II was uprated from 5 MW to 6 MW in 2011 in conjunction with a 20-year license renewal (MITR-Staff, 2011).

There are normally 24 fuel elements in the 27 in-core positions, with the remaining three locations available for experiments. Recent in-core experiments include the Advanced Cladding Irradiation (ACI) experiment, HYdride Fuel Irradiation (HYFI) experiment, and In-Core Sample Assembly (ICSA) (Kim et al., 2013). ACI tests the performance of SiC cladding material under light-water reactor (LWR) temperature, pressure, and chemistry conditions. The HYFI experiment tests the irradiation performance of hydride fuel pellets with a lead–bismuth eutectic (LBE) gap inside zircaloy cladding. The ICSA facility, which is mostly involved in the current study, provides a flexible high temperature irradiation environment for advanced materials and sensor testing. It consists of an S-

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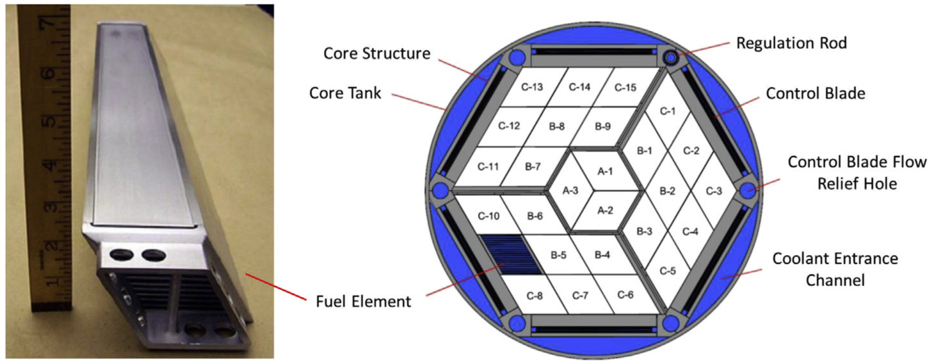

Fig. 1. A photo of the rhomboid-shaped fuel elements (left) and the horizontal cross-section of the MITR-II core with indication of different components (right).

shape guide thimble (in order to prevent direct radiation streaming up through the core tank), irradiation capsules and gamma susceptors. A desired temperature range (typically from 400 to 900 °C) can be achieved by controlling the gas mixture of helium and neon between the capsule outer surface and the thimble inner surface. Titanium was selected as the primary gamma susceptor material due to favorable neutron transparency. It results in limited reactivity penalty and lower activation compared to other candidate metals such as steel. The mechanical design was driven by the requirement that the capsule has sufficient radial clearance to pass through the S-shape guide thimble, but is positioned concentrically in the irradiation position to provide the necessary gas gap. The irradiation capsule normally sits on a dummy spacer in order to fix its axial center to the maximum neutron flux position, i.e. ~5 cm below the core centerline (see Fig. 2).

Due to steady advances of computer power in recent decades, continuous-energy Monte Carlo method-based codes are becoming more widely used in reactor calculations, especially for small cores such as the MITR. Recognizing this, an MCNP model of the MITR-II, which is highly detailed and contains very few approximations, was constructed in the early 1990s (Redmond et al., 1994). In addition, MCNP has been externally coupled with the point-depletion code ORIGEN for burn-up calculations. The link is established by an in-house code package, MCODE (MCNP-ORIGEN Coupled Depletion Program) (Xu, 2003), and more recently, by the extended MCODE-FM (Fuel Management) (Romano, 2007) (Horelik et al., 2009).

Some code-to-code benchmark studies have been performed in the past for both the original MCODE code package and the extended version. MCODE was verified against the state-of-the-art neutronic code CASMO-5, in terms of BWR fuel assembly depletion (Xu et al., 2007); whereas MCODE-FM was benchmarked against REBUS-PC for MITR's power peaking calculations (Romano, 2007). Moreover, the MCODE MITR model has been validated — the reactivity worth of the control blades and of the fuel burn-up were found agree well to experimental data (Wilson et al., 2010) (Newton et al., 2004). In the current paper, additional validation runs are carried out for MCODE-FM, to provide further confidence that MCODE-FM can be adopted as the reference code for the fuel management routine of the MITR. MCODE-FM results are compared against two sets of measured data: 1) reactivity effects of fission product poisoning and fuel depletion and 2) reaction rates based on thermal and fast neutron flux.

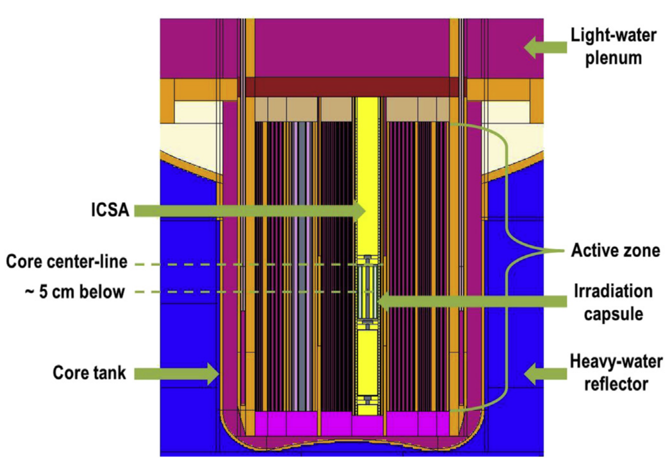

Fig. 2. The vertical cross-section of the MITR-II core and the illustrations of the ICSA and the irradiation capsule (MCNP plot).

2. MCODE-FM model of the MITR

The focus of the present study is using experimental data to validate the fuel management code MCODE-FM. The main workhorse of this code package consists of the Monte Carlo transport code MCNP5-v1.40 and the point-depletion code ORIGEN2.2. The former provides snapshot physics results of the studied system, in particular the continuous-energy neutron flux distribution of the entire MITR core model. The one-group macroscopic cross-section is generated by weighting the point–wise microscopic cross-section (included in nuclear and atomic data libraries) with the energy dependent neutron flux. MCODE is the external link to transfer the macroscopic cross-sections of all the involved isotopes to ORIGEN. The latter adopts the matrix exponential method to solve a large system of coupled linear first-order ordinary differential equations with constant coefficients. The material compositions can be therefore updated by taking into account neutron absorption and radioactive decay within a pre-defined time frame. MCODE then sends back the updated material compositions to MCNP and initiates a new round of neutron transport calculation. It should be noted that MCODE adopts a predictor-corrector method to calculate nuclide concentrations for each time-step, which has been shown to be more accurate than the methods used in other similar codes that adopt only the beginning-of-step or middle-of-step reaction rates for the depletion matrix in the Betaman equations (Xu et al., 2007). The wrapper code MCODE-FM is specifically designed for MITR fuel management as the extension of MCODE. It contains all geometry information of the MITR core and enables the rhomboid-shaped MITR fuel elements to be rotated and flipped. As mentioned, a criticality search algorithm to track critical blade movement has been implemented. In addition, MCODE-FM features automation of input file generation, data manipulation, and post-processing of the output data for fuel cycle analysis.