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Neutronic analysis of MSRE and its study for validation of ARCH code

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HIGHLIGHTS

- Steady-state neutronic evaluation of Molten Salt Reactor Experiment (MSRE) using system of codes DRAGON and ARCH is given.
- Effect of Beryllium fast neutron multiplication in the fluid fuel is highlighted.
- Power and flux map obtained from the evaluation present a more informative radial and axial distribution through core.
- Validation of ARCH core code for molten salt systems

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ABSTRACT

In the present paper, preliminary steady-state neutronic evaluations are carried out to understand the dynamics of a molten salt reactor. Oak Ridge National Laboratory's (ORNL) Molten Salt Reactor Experiment(MSRE) is selected for the case study and the system of nuclear codes *DRAGON* and *ARCH* are used for lattice and core calculations respectively. *DRAGLIB*, an open-source group condensed cross-section library compatible with *DRAGON* is used for transport calculations. Differences in the reactivity values are observed using the two recent libraries namely, JEFF 3.1.2 and ENDF/BVII.1. A reactivity difference of about 4.5 mk, close to a typical beta value for the reactor, is observed for MSRE core using ENDF/BVII.1 and JEFF3.1.2 libraries. This is attributed to the absence of (n,2n) neutron multiplication reaction cross section for Beryllium in the processed JEFF 3.1.2 draglib library. The effect of missing (n,2n) cross-section leads to an underestimated net reactivity. Core calculations for the MSRE are found satisfactory and highlight observations regarding flux and power distribution which were not seen previously in ORNL's calculations. A study on the reactivity coefficients reconfirm that the graphite temperature coefficient as calculated previously was overestimated due to lack of proper data library. The present paper aims to understand the MSR neutronics and validate numerical simulation methodology as well as benchmark the indigenously developed core code *ARCH* on molten salt systems. It also emphasizes the effect of beryllium as a neutron multiplier in the fluid fuel for typical MSRs and highlights its importance in the library evaluations and on the computational correctness of the reactivity of molten salt systems.

1. Introduction

Molten Salt Reactors (MSR) have been considered as one of the Gen-IV reactor concepts and are currently being pursued with renewed interest at major nuclear centers. They are seen worldwide as a promising technology, owing to their dynamics, particularly for thorium utilization. A number of improved MSR designs are being studied with challenges in safety, fuel handling, reprocessing, coupled dynamics etc.

In particular, the neutronic behaviour of moving fluid fuel systems and the respective thermal hydraulic feedbacks are challenges to the design of Molten Salt Breeder Reactors (MSBR Kasten, 1966; Weinberg, 1970). MSBRs are one of the reactor types considered in the Indian nuclear power program. Liquid fuelled reactors have significant advantage like, inherent safety, high neutron economy and possibility of continuous on-line reprocessing (Rosenthal, 1970; Leblanc, 2010; Krepel et al., 2013; Moir (2008)). The MSR core can be maintained practically at very low excess reactivity. The neutronics of such a concept is unique and

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requires accurate modelling. The fuel and coolant admixed in the reactor core provide stronger coupling (Zhangpeng, 2013), so the salt moving out of the core will have to be modelled more precisely with respect to safety, criticality etc. In particular, the flowing fluid fuel renders a complicated reactor kinetics. The hydraulic term alters the distribution of precursors within the core which in turn effects the neutronic behaviour. This leads to a strong coupling and provides a unique neutronic characteristic than the usual solid fuelled cores (Zhang et al., 2009, 2015).

In an attempt to understand and simulate the neutronics of a molten salt reactor system, ORNL's test reactor Molten Salt Reactor Experiment (MSRE) (Leblanc, 2010; Robertson, 1965; Haunbenreich et al., 1964) was chosen for the study. Lattice and core calculations for MSRE core were performed using the system of lattice analysis code DRAGON (Marleau and Hébert, 1989; Marleau and Hébert, 1933; Marleau et al., 1992) and 3D space-time diffusion core code ARCH. While DRAGON is maintained by Ecole Polytechnique de Montreal, ARCH (Anurag, 2012) is an indigenous effort at BARC, India. ARCH has been benchmarked against a wide range of problems and reactor cores for both Hexagonal and Cartesian geometries. The present neutronic analysis serves as an important benchmarking validation of ARCH for future molten salt systems. Calculations have been performed using different set of processed transport libraries, namely ENDF/B and JEFF versions available in DRAGLIB format (www.polymtl.ca/merlin/libraries.htm).

2. Molten Salt Reactor Experiment (MSRE)

The research on molten salt reactors initially began at ORNL in the late 1940s. After years of dedicated research and development and continuous design evolutions, a concept known as the Single Fluid, graphite moderated MSBR (Kasten, 1966; Weinberg, 1970) was adopted. During this phase, a highly successful experimental test reactor was constructed and operated by ORNL under the name MSRE.

MSRE (Robertson, 1965; Haunbenreich et al., 1964) was born as a result of series of experiments done at ORNL in 1960s to develop a nuclear powered Aircraft Engine. During those days, the main emphasis was to develop a cost-effective economical breeder reactor. A typical two-fluid reactor concept had this advantage, however the single fluid reactor concept appeared much simpler at the cost of low breeding ratio. While a lot of importance was placed on two-fluid concept, single fluid reactors were also studied and equally debated.

The key features of MSRE are mentioned below (Robertson, 1965):

- MSRE was a single-fluid reactor concept where fluid fuel was made to flow through graphite channels. Since the fuel salt did not contain thorium, its dynamics could also be tested as a two-fluid breeder. It went critical in 1965.
- It was a 10MWth cylindrical reactor design with a can containing the core placed within the reactor vessel.
- The fuel salt was a mixture of U, Li, Be and Zr fluorides and unclad graphite served as moderator.
- All other parts that contact salt were made of nickel-based Hastelloy (INOR-8).
- At the center of the core were three control rod thimbles and a graphite sample assembly.
- Second phase began in 1968 after a reprocessing facility was attached to the reactor. It was designed to reprocess fuel salt and extract original U by fluorination technique.
- A small fraction of U233 was added to the fuel salt and the reactor again went critical on 2nd October.

2.1. Core specifications

Some of the main core specifications concerning size and geometry are mentioned below in bullets and also shown in Figs. 1 and 2. The remaining data are available from relevant ORNL reference documents (Robertson, 1965; Haunbenreich et al., 1964).

- The height and diameter of the core were fixed at 5.5 ft (1.6 m) and 4.5 ft (1.4 m) respectively.
- Average fuel fraction was set at 0.225.
- The fuel salt inlet and outlet temperature was 908 K (1175 °F) and 936 K (1225 °F) respectively. This gives an average core temperature at 922 K (1200 °F).
- Fuel salt circulated at a rate of 1200 gallons/min.

The fluid fuel salt, enters the reactor vessel through a flow distributor and flows down into the annular region, termed

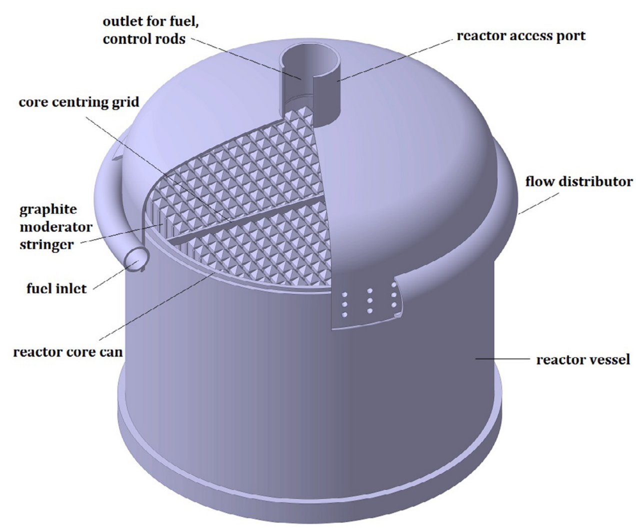

Fig. 1. A 3D representation of MSRE core vessel (Kedl, 1970).

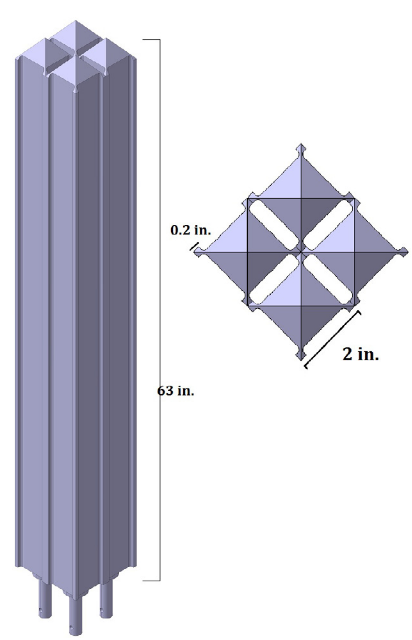

Fig. 2. Typical graphite stringer arrangement in the core (Kedl, 1970).