Annals of Nuclear Energy 110 (2017) 265–281

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

A method for predicting fuel maintenance in once-through MSRs

Gavin Ridley, Ondrej Chvala *

Department of Nuclear Engineering, University of Tennessee at Knoxville, TN, USA

ARTICLE INFO

Article history:
Received 9 January 2017
Received in revised form 15 June 2017
Accepted 20 June 2017

Keywords:
DMSR
Molten salt reactor
Serpent
On-line refueling
Chemistry control
Depletion

ABSTRACT

Liquid fuel molten salt reactors allow reactivity control by material addition. This paper presents a method to adjust material flows in a molten salt reactor to keep the core critical, and to maintain desired reduction-oxidation potential in the core salt melt. The method is aimed at low-enriched uranium fueled thermal systems. It is developed as a Python library and uses Serpent2 Monte-Carlo transport and depletion code. A toy 300 MW(th) reactor with a FLiBe carrier salt is employed to demonstrate the performance of the method over 10 full power years. Results of the calculation are presented, including material flows, conversion ratio, effective delayed neutron fraction, and expected limits on trifluoride concentrations and graphite lifetime are investigated. This method lays a foundation for future studies including fuel cycle performance of molten salt reactors and dynamic behavior of the core during depletion.

1. Introduction

There are currently six reactor concepts that have been chosen by the Generation IV International Forum (GIF). One of these six modern designs is the molten salt reactor (MSR). The MSR class encompasses a huge variety of designs. One of particular interest is the denatured molten salt reactor (DMSR), originally proposed by Engel et al. (1980) at Oak Ridge National Laboratory (ORNL) in 1980.

DMSRs offer several features that many proposed MSRs cannot provide. Proliferation resistance is attained by keeping any fissile material diluted by fertile material. For the purpose of this study, DMSR means that the uranium is kept at low enrichment, such that the fissile material is less attractive for misuse. Designing the core to have relatively low power density and possibly flattened flux allows the moderator graphite to remain in-core for the lifetime of the reactor core. Planning on minimal reprocessing of the salt simplifies the power plant design. These features indicate that the DMSR may be the most cost competitive MSR power plant.

One of the MSR advantages recognized in the 1950s during the Aircraft Reactor Experiment was the ability of the reactor to easily follow the load, without control rod action. Cottrell et al. (1955) Can such reactor be controlled entirely without movable absorbers, using material flows alone?

This paper presents a code that facilitates efficiently computing continuous low-enriched uranium (LEU) refuel rates for DMSR while simultaneously managing fuel chemistry to maintain desirable oxidation potential of the fuel salt. The details of how chemistry can be expected to change in time are given in Section 2.3. The core reactivity is maintained via material additions to the fuel. Reactivity decrease due to depletion is compensated by adding fuel salt with 20%LEU. This takes advantage of MSRs' ability to continuously refuel, thus mitigating the need for neutron-wasting absorbers. As a result of these simultaneous processes, the refueling and chemistry control problems become inherently coupled since both must be addressed through changing the fuel composition. Chemistry calculations can be enormously complicated. For this reason a simple method is used, relying upon presumptive oxidation states of various elements in the fuel.

The thermal fission of a $^{235}U^{4+}$ in fluoride salts can be shown to be an overall oxidizing process to the salt. This is because the uranium nucleus balances the charge of four fluorine ions in the salt, but fission products tend to not bind to all the four fluorines released after the uranium fissions. Fig. 1 sketches an example of an oxidative fission reaction. This excess of fluorine must be mitigated, otherwise chemical reactions deleterious to reactor components would occur. In this study, natural uranium metal was chosen to chemically reduce the salt to adequately overcome fission's oxidizing effects.

Lithium-bearing salts always contain some fraction of ^{6}Li. For this study, a fuel using lithium depleted to 99.995% ^{7} Li was used. As it turns out, even small concentrations of ^{6}Li being depleted to a lower, steady-state concentration can cause significant reactivity

* Corresponding author.
E-mail addresses: gridley@vols.utk.edu (G. Ridley), ochvala@utk.edu (O. Chvala).
URL: http://www.engr.utk.edu/nuclear (O. Chvala).

http://dx.doi.org/10.1016/j.anucene.2017.06.043
0306-4549/

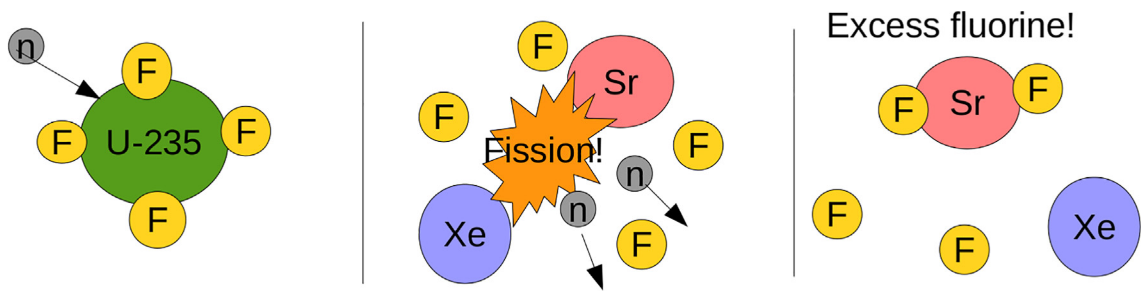

Fig. 1. Fission process creates an excess of fluorine.

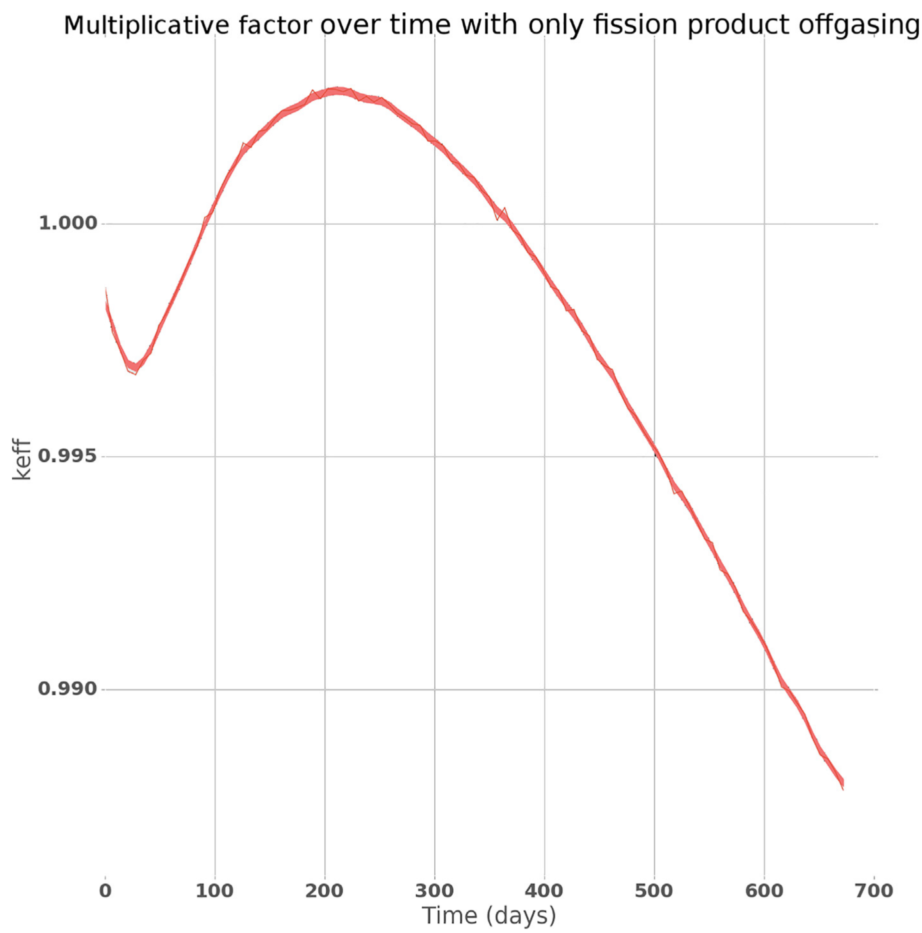

Fig. 2. Evolution of the multiplication factor k_{eff} over the first two full power years with no reprocessing other than gaseous fission product removal. The confidence interval $\pm\sigma$ is shaded.

rises. This effect can be seen in Fig. 2, where a Serpent depletion simulation was done on a FLiBe-fueled MSR with off-gas processing only. It is proposed herein that additions of gadolinium fluoride mitigate this reactivity rise when necessary. The authors acknowledge that using movable absorbers such as control rods would likely be preferable to burnable absorber dissolution in a real system. The aim of this paper is to develop methodology for reactivity and chemistry control using flow adjustments alone.

2. Methodology

The rate of addition of fresh fuel, burnable absorber, and reducing agent to the core is varies over depletion time. These rates vary in a way such that reactivity remains bounded within user-defined margin, and the excess fluorine gets mitigated by a reducing agent to maintain global charge balance. Material inflows increase density of the respective material in Serpent, so the material volume is correspondingly increased at the end of each depletion step in an external script. The resulting effect is akin to filling a bucket with water. Another option is to move the excess fuel salt into a bleed-off tank. Both options are covered by the developed method, and the former is used for the method demonstration in this paper.

The most appropriate tool to simulate this coupled depletion and neutronics at present is the Serpent2 code (Leppänen et al., 2015) from VTT in Finland. Version 2.1.28 was used, but certain bug fixes implemented following this research will be available in version 2.1.29. At the moment, this code does not allow the variation of reprocessing flows over depletion, so many input files are constructed and run sequentially to imitate variable refuel rates over depletion steps. It was determined through trial that 7 days is a good depletion step size over the course of several years of overall depletion.