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ScienceDirect

Energy Procedia 71 (2015) 3 – 13

Energy Procedia

The Fourth International Symposium on Innovative Nuclear Energy Systems INES-4

Criticality uncertainty dependence on nuclear data library in Fast Molten Salt Reactors

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Abstract

To increase the sustainability of the nuclear fuel cycle, and increase security of nuclear energy, we have been investigating Molten Salt Fast Reactors (MSFR) for transmutation of Minor actinoid (MA) isotopes. In the present work we describe the reactor physics analysis of a Th-TRU MSFR using a LiF-ThF_4-$TRUF_3$-fuel salt. We investigated the uncertainty of major reactor physics parameters using 3 sets of evaluated nuclear data: JENDL-4.0, JEFF-3.1.2, and ENDF/B-VII.1. The result of our work is that the spread in the multiplication factor is rather large between the sets of nuclear data, while other parameters are by and large the same. The uncertainties due to cross section covariance are large, with Th-232, U-233, and F-19 giving the most important contributions. The isotopic contributions to the uncertainties are quite different between the sets of nuclear data, giving a suspicion that the covariance data may is very different between the evaluations, and a review of the covariance data may be needed.

Selection and peer-review under responsibility of the Tokyo Institute of Technology

Keywords: Nuclear Transmutation, Molten Salt Reactor, Nuclear Data, Uncertainty Analysis, Cross Section Processing.

1. Introduction

For the long-term sustainability of the nuclear fuel cycle, and to increase security of nuclear power as a source of energy, it is important to address the issue of long-lived nuclear waste (HLW, High Level Waste). In the present paper we investigate a Molten Salt Reactor for MA transmutation: Molten Salt Fast Reactor (MSFR). The reactor uses a thorium-TRU (TRans-Uranium) fuel. The Molten Salt Reactor concept offers several attractive features for transmutation of Minor actinoid (MA) isotopes: online refueling and reprocessing, negative reactivity coefficients for fuel temperature and density, and the potential to discharge the fuel to a safe container in case of a severe accident. In the present work, reactor phsyical parameters (reactivity, Doppler coefficient, fuel density coefficient, delayed neutron fraction) are calculated and their uncertainties, using three modern sets of evaluated nuclear data: JENDL-4.0, JEFF-3.1.2, and ENDF/B-VII.1. The result is that all three of these sets of nuclear data require improvements for application to MSFR.

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Selection and peer-review under responsibility of the Tokyo Institute of Technology
doi:10.1016/j.egypro.2014.11.849

2. Fuel salt composition

For the present work, we focus on a fluoride molten salt. While there is much information about salt compositions for tetravalent fluoride compounds (ThF_4 and UF_4), there is much fewer information about the trivalent fluoride compounds (PuF_3, AmF_3, etc). After literature review [1, 2, 3, 4]), we decided on the following salt compostion: LiF−ThF_4−$TRUF_3$ (78/17/5 mole%, TRU = TRans Uranium isotopes). In Figure 1 is given phase diagram of this fuel salt. From Figure 1 it is seen that the chosen composition is near the eutectic point, and the melting point was estimated as 850 K. The density was estimated as $\rho(T) = 5.54 - 1.25 \times 10^{-3} T$ g cm^{-3}, with T the temperature in Kelvin [5].

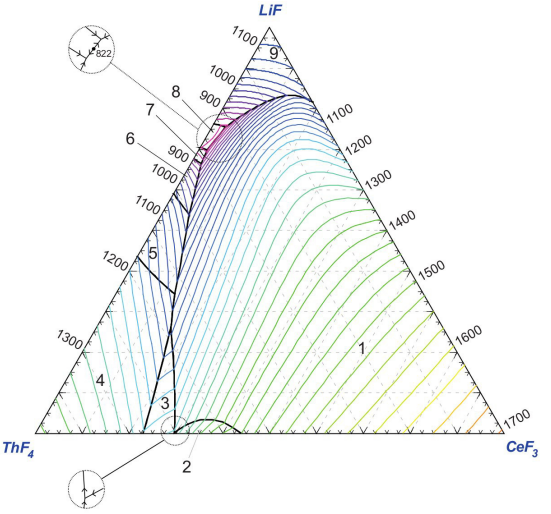

Fig. 1. Phase diagram of the LiF−ThF_4−CeF_3-system, taken from [3]. CeF_3 is an analogon for PuF_3, and in the present work it is assumed that Np, Am, and Cm have similar behavior. From this diagram it is clear that addition of PuF_3 to the fuel salt rapidly increases the melting point. The composition for the present work, LiF−ThF_4−$TRUF_3$ at 78/17/5 mole%, is in the upper right hand corner of the upper circle.

A representative core composition was derived from the following considerations: to maintain the melting point of the fuel salt, the fraction of TRU should be constant, as should the fraction of Th + U (the chemical behavior of ThF_4 is similar to UF_4 [4]). Also, the concentration of fission products in the fuel salt should remain low. During operation, Th and TRU are continuously added to the reactor. The fissile isotopes in the TRU will deplete very quickly. Therefore, sustained operation is possible if the production of U-233 from Th-232 balances the reactivity loss of the TRU material. The equilibrium ratio of U-233 to Th-232 is given by $N_{23}/N_{02} \approx \sigma_{c,02}/\sigma_{c,23} \approx 0.14$, where index "02" indicates Th-232 and "23" indicates U-233. Thus, to maintain the salt composition in an acceptable range, the rate of addition of TRU should be the same as the rate of consumption; to maintain the fraction of Th + U, one adds Th at a sufficient rate, and any surplus U-233 is to be continuously removed (note: as long as the "enrichment" in U-233 is below the equilibrium value, there will be an overproduction of U-233).

3. Scoping calculations for the fuel cycle

In the present work, we have assumed that the TRU originates from recycled BWR-MOX fuel. The TRU composition was calculated assuming an ABWR-type reactor (specifically, the ABWR under construction in Oma, Japan), with a full MOX core. The fuel is irradiated to 55000 GWd/t, and subsequently a cooling period of 15 year is assumed. At present, further details, such as shuffling patterns and/or fuel batching are not taken into account. The core-averaged composition of the TRU is given in Table 1. The basic mode of