Dynamic Burnup Studies of Seaborg Compact Molten Salt Reactor by Serpent 2

Vutheam Dos[a], Kyeongwon Kim[a], Eirik Eide Pettersen[b], Jacob Groth Jensen[b], Deokjung Lee[a]*
[a]*Department of Nuclear Engineering, Ulsan National Institute of Science and Technology 50 UNIST-gil, Ulsan 44919, Republic of Korea*
[b]*Seaborg Tech Co., Ltd, Titangade 11 2200 Copenhagen N, Denmark*
* *Corresponding author: deokjung@unist.ac.kr*

1. Introduction

Current interest in advanced nuclear energy and molten salt reactor (MSR) concepts characterized by its use of the liquid-fuel salts serving as a fuel and as a coolant has enhanced interest in developing new core design and building the tools necessary to analyze the reactor core system. Seaborg Technologies is actively developing a design for its compact molten salt reactor (CMSR)[1], a liquid-fuel MSR with a thermal power output of 250 MWth. The liquid-fuel of the reactor provides important benefits over the traditional solid-fuel reactors with passive safety and fuel utilization. However, during the fuel circulation, the position of delayed neutron precursors continuously changes both in the core and in the external loop, and the fission products (FPs) are extracted by an online fuel reprocessing unit. The online fuel reprocessing limits to most of the simulation tools used with the solid-fuel in the conventional reactor. One exception is the Monte Carlo code Serpent 2 [2] which has recently included the capability to do on-the-fly modification of the material definitions model online fuel reprocessing.

In this paper, the burnup characteristics of CMSR is studied by performing different burnup calculations with and without continuously removing FPs using Serpent 2. Three burnup calculations are performed 1) without removing any FPs, 2) six noble gases are continuously removed, 3) six noble gases and other noble metals are continuously removed. Reactivity control by inducing soluble boron into the moderator is also investigated. The results of this study will help to steer the design efforts as well as provide insight into future R&D targets.

2. Seaborg CMSR Core Design

Figure 1 displays a horizontal view (left) and vertical view (right) of the 3D model of CMSR reactor geometry as implemented in Serpent. The 3D geometry incorporates 235 slots of about 10 centimeters in diameter through which the fuel salt (visible in pink) flows in and out the core. The fuel salt is made up of a NaF-KF-UF_4 mixtures in which the uranium enrichment is adjusted in this study such that the reactor able to reach the equilibrium in 12 years of continuous operation. The system shutdown rods, composed of 5 slots of carbon carbide, is not studied in this paper and thus replaced by air (visible in white). The NaOH moderator is shown in blue, and the Hastelloy-N shown in grey. The core was 2.5 meters in diameter and 2 meters in height; and the core size was kept constants in call simulation. The material properties for fuel and moderator are obtained from the Seaborg salt properties database are shown in Table 1.

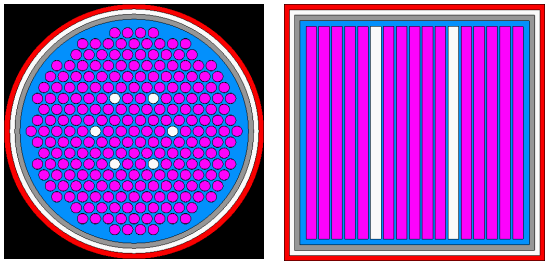


Figure 1. The layout of the CMSR reactor in Serpent 2 code.

Table 1. Detail of material composition in CMSR.

Type	Materials	Density (g/cm^3)
Fuel - pink	NaF-KF-UF_4	4.261
Moderator - blue	NaOH	1.637
Cladding - grey	Hastelloy N	8.86

3. Simulation Details

In this study, we performed the depletion simulation by using the Serpent2 code [3], a 3D continuous energy Monte Carlo particle transport code developed by the VTT Research team of Finland. Serpent 2 is the first commonly used Monte Carlo code capable of performing the continuous material reprocessing by allowing the user to define multiple materials flows into and out of the fuel. The simulations employed the ACE (A Compact ENDF) format cross-section library with continuous energy based on the ENDF/B-VII.1 nuclear data library evaluated at 900K on a Linux Cluster. For fission yield and decay, data are based on the JEFF-3.1.1 library.

For all simulations, 50 inactive cycles are employed to converge the fission source before starting the active cycles. Then, the calculation is performed with 200 active cycles and 50,000 neutrons per cycle to achieve low relative statistical uncertainty on the calculated k-eff of 20 pcm. The total power level is assumed to be constant (250 MWth) in each depletion time interval. The burnup simulation is performed by removing the FPs and actinides every 60 seconds. For the first 10 depletion time steps which highly affect the change of the material compositions are set to be very small as 0.1, 0.3, 0.6, 1, 3, 8, 17, 27, 40, and 51 days.

After that, the burnup time steps are set to have 73 days interval and in total, there are 69 cycles to reach the final specified final period of 12 years. The three burnups simulated in this study are:

1. **No-pro**: No FP is removed
2. **Degas1**: six noble gases (He, Ne, Ar, Kr, Xe, and Rn) are being removed continuously
3. **Degas2**: six noble gases and with more noble metals (Nb, Mo, Tc, Ru, Rh, Pd, Ag, Sb, and Te) are being removed continuously. This scheme is highly unfeasible and only included as sort of a best-case scenario. This reason for this is that there is no indication that its possible to remove noble metals at the same rate as noble gasses.

4. Simulation Results and Discussion

4.1. Effective Multiplication factor

Figure 2 illustrates the improvement in performance for various degassing schemes as explained in the previous section. With the fuel enrichment of 12.73% of uranium-235, the results with no-processing of this CMSR model indicate that the multiplication factor slowly decreases and reaches unity after 12 years of operation. However, one clear difference between the no-processing and with degassing is that the reactivity decreases faster in the no-processing scheme. This is due to the continuous removal of neutron poisons leading to a relative improvement of the k-eff at EOC by 1.5% for Degas1 and 3% for Degas2 compared to the no-processing scheme. It is also can be explained that as the neutron poisons are continuously removed, the thermal fission fraction $\left(f = \frac{\Sigma_a^F}{\Sigma_a} \right)$ is increased as the flux spectra get softer as illustrated in **Figure 3**.

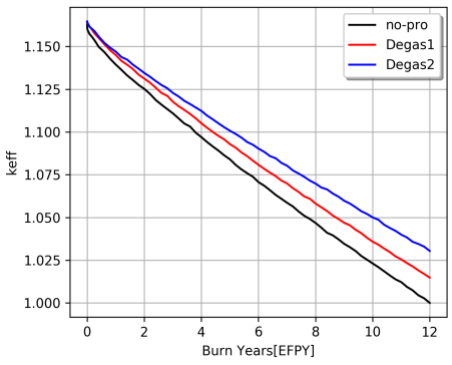


Figure 2. Deterioration of the effective multiplication factor with various depletion schemes

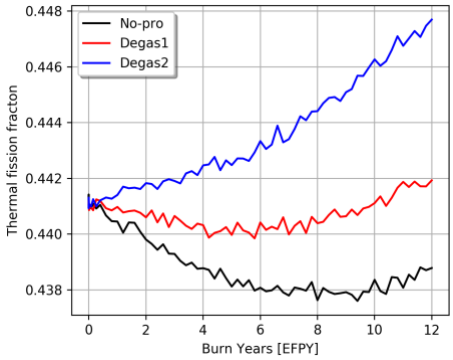


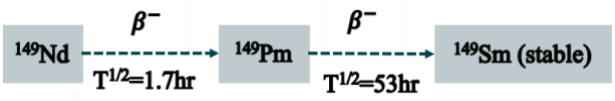
Figure 3. Development of the thermal fission fraction with various depletion schemes.

4.2. Evolution of fission products

One important aspect to investigate the behavior of the FPs inventories in the fuel salt is the reactivity poisoning effects since there are several hundred FPs are produced during operation. Those FP isotopes show different behaviors in the reactor core based on their nuclear properties and the rate of their production/consumption. Many isotopes with small absorption cross-section or minute concentration tend to have a low effect on the reactivity; and some isotopes like xenon-135 or samarium-149 which has large cross-sections and considerable effective fission yield will have a large negative effect in the reactor. Thus, it is important to know the behavior of these poisons. In the present CMSR model Serpent 2 tracks 1,642 different FP isotopes. In the following, the evolution of the most widely produced isotopes is analyzed for three different depletion schemes. The FPs of interest are xenon-135, samarium-149, molybdenum-99, strontium-90 and cesium-137.

Figure 4-Figure 7 show the atomic density and mass of the FPs of interest in the different depletion schemes as a function of time. All figures show that some FPs in the fuel salt buildup to an equilibrium value and then remains constant. In addition, the FP gasses reach the equilibrium value much faster than FP metals. As shown in **Figure 4** and **Figure 5** for the result in the no-processing scheme, the xenon-135 reaches equilibrium around 5 days whereas cesium-137 and strontium-90 do not saturate in the fuel cycle. The xenon-135 quickly reaches its saturation due to its extremely large capture cross-section (2.6 x10$_6$ barns) and short half-life of 9.1 hours, and its formation directly from fission production and the beta decay of other fission products. In contrast, the strontium-90 and cesium-137 are the major radioactive product of nuclear fission, but due to their long half-life of 28 years for strontium-90 and 30.2 year for cesium-137, they are not fully saturated yet even after 12 years and therefore make them less important compared to xenon-135. Another important FP poison encountered in the reactor is samarium-149 which also has large thermal

absorption cross-section of 4x10$_4$ barn. It is produced through beta-decay of its precursor as follow:



The samarium-149 has a smaller cross-section than xenon-135 and the fission yield of its precursors is relatively small. The noble gas like xenon-135 is fairly easy to remove by helium bobbling or some other degassing method. However, that's not the case for samarium-149 because it is not a noble gas. The rate at which noble gasses are removed in reprocessing scheme Degas1 and Degas2 are similar and both significantly smaller that the half-life of xenon-135. This means that only a small difference in atomic density of xenon-135 is expected between the two schemes. Therefore, xenon-135 contributes the most negative impact in CMSR compared to other isotopes in terms of poisoning effect.

Noble-gasses degassing (Degas1): Xenon-135 which has the largest negative impact on the reactor core is continuously removed along with other noble gases. As seen in **Figure 4** and **Figure 5**, the xenon-135 concentration is very small compared to the no-processing scheme. The changed amount of xenon-135 concentration in this noble-gasses degassing schemes also affected the cesium-137 isotope which is produced from the decay of xenon-135 after 9 hours.

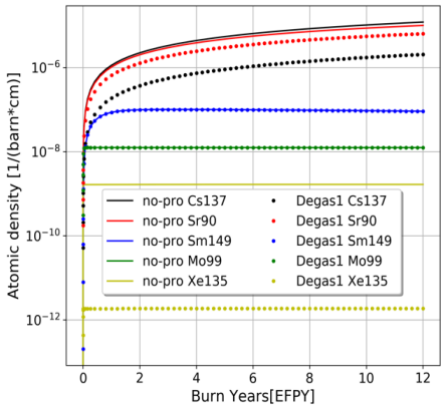


Figure 4. The atomic density of FPs in fuel salt of no-pro and Degas1.

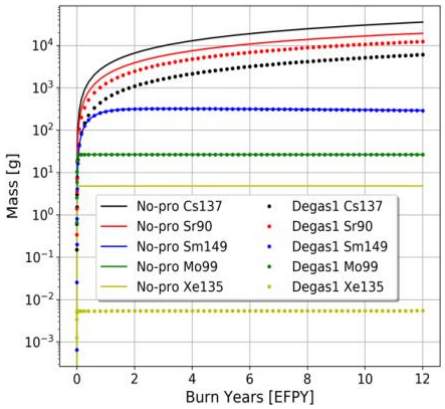


Figure 5. The mass of FPs in fuel salt of no-pro and Degas1.

Noble-gasses and noble-metal degassing (Degas2): the FPs concentrations illustrated in **Figure 6** and **Figure 7** shows that, for the noble-gases only/and noble-metal degassing schemes do not change much because those noble-metals do not have large cross-section compared to xenon-135. Molybdenum-99 and technetium-99 are among the noble-metals to be continuously removed; therefore, the molybdenum-99 concentration has become very small.

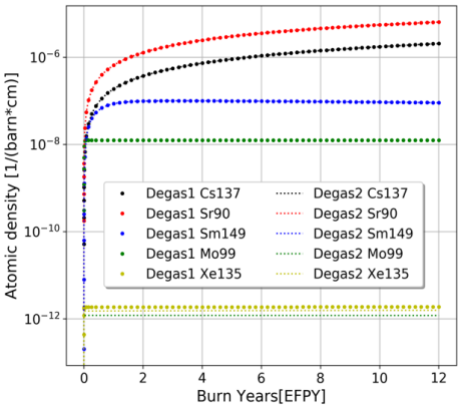


Figure 6. The atomic density of FPs in fuel salt of Degas1 and Degas2.

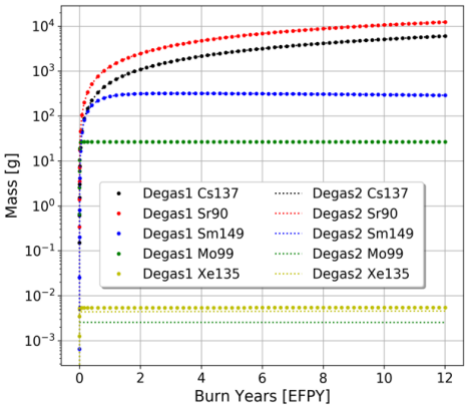


Figure 7. The mass of FPs in fuel salt of Degas1 and Degas2.

4.3. Reactivity Control

In this work, we use Serpent2 code with python-script to estimate an equilibrium boron concentration search during the depletion simulations by inducing the boric acid $B(OH)_3$ to the moderator salt NaOH. In this section, three simulations with soluble boric acid to moderator salt are performed for three depletion schemes such as no-pro, Degas1, and Degas2. At BOC, a high boron concentration is required to compensate for the excess reactivity into the core and there is very little soluble boron in the moderator at the end of the fuel cycle (EOC). **Figure 8** shows the drop in boron concentration during the fuel cycle as a function of burnup time or commonly called "boron letdown curve". It notices that at BOC, the boron concentration in the simulation without degassing (no-pro) is very rapidly dropping at first as xenon and samarium build-up to an equilibrium level. However, for the boron concentration of Degas1 and Degas2 reprocessing schemes the drop is small BOC due to the continuous removal of FPs such as xenon-135.

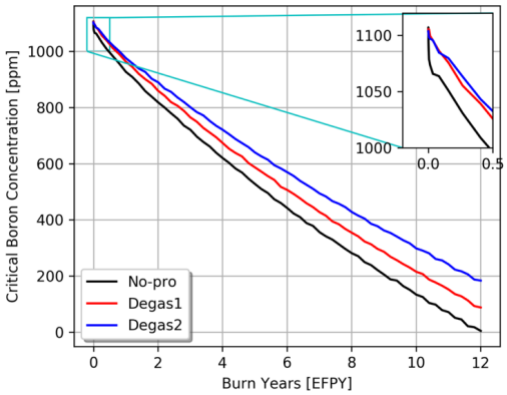


Figure 8. Boron letdown curve in CMSR.

4. Conclusion and Perspective

The capability to continuously extract FPs in the fuel salt during normal operation by Serpent 2 for the Seaborg CMSR model with different degassing schemes is demonstrated. Online-processing schemes, clearly improve the neutron economy performance by reducing the concentration of fission products such as strontium-90, cesium-137, samarium-149, and xenon-135 and thereby minimizing the depletion of uranium. It has been successfully demonstrated using Serpent 2 with python-script a critical boron concentration can be obtained when using boric acid mixed in the moderator. In the future, the temperature coefficients which is important for safety will be studied. In addition, the results of fuel composition at EOC would be used to characterize the CMSR spent fuel and propose an appropriate design for a spent fuel storage.

ACKNOWLEDGEMENTS

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