Impacts of Nuclear Data on the Core Characteristics of a Compact Breed-and-Burn Fast Reactor (B&BR)

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1. Introduction

A compact sodium-cooled Breed-and-Burn Fast Reactor (B&BR) has been investigated from neutronics perspective in order to improve the core performance and safety [1,2,3,4]. Similar to the CANDLE reactor configuration [5], the blanket fuels are located on the top of the initial core. In our study, the spent nuclear fuels (SNF) from light water reactors after being metallized are used as the blanket fuels. Meanwhile, metallic low-enriched uranium (LEU) is located in the initial core. Besides providing the core first criticality, the initial core will supply neutrons to the blanket fuels so the fertile fuels can be converted to fissile fuels.

In this paper, the impact of using different nuclear data libraries on the core characteristics of the B&BR was addressed. The performance of the B&BR depends on how many fissile fuels can be bred and utilized from the blanket fuels. On the other hand, the core inherent safety characteristic depends strongly on the TRU (transuranic) fuel composition generated in the fuel region. Three present major evaluated nuclear data libraries were considered in this study, i.e., ENDF/B-VII.0 [6], ENDF/B-VII.1 [7] and JENDL-4.0 [8]. The neutronics analyses were all performed by using the continuous-energy Monte Carlo code Serpent [9].

2. Compact B&BR Concepts

The compact sodium-cooled B&BR core has thermal power of 250 MWth. Assuming the balance of plant thermal efficiency is about 40%, then the electric power output is about 100 MWe. The fuel assemblies and the reflector assemblies are arranged in an 8-ring hexagonal core. The core consists of 78 fuel assemblies, 78 reflector assemblies and 7 control rod assemblies.

The initial core is a concave-shaped core as shown in Fig. 1. The inner initial core which is loaded with LEU-10Zr has height of 60 cm. Meanwhile, the outer initial core is loaded with LEU-7Zr and has height of 85 cm. The combination of the concave-shaped core and dividing the initial core into 2 different zirconium contents has the advantage of reducing the core maximum excess reactivity and the radial power peaking [4]. The required U-235 enrichment for the inner and outer initial core is about 12.42% just adequate to make the reactor slightly supercritical. The blanket region is also divided into several regions with different zirconium contents. The purpose of the region division is to reduce the radial power peaking at end of life (EOL). There are 3 zirconium zones in the blanket region considered in this work, i.e. SNF-10Zr, SNF-8Zr, and SNF-6Zr. With these 3 zirconium zones in the blanket region, the radial power peaking at EOL can be reduced from 1.42 to 1.33 [4].

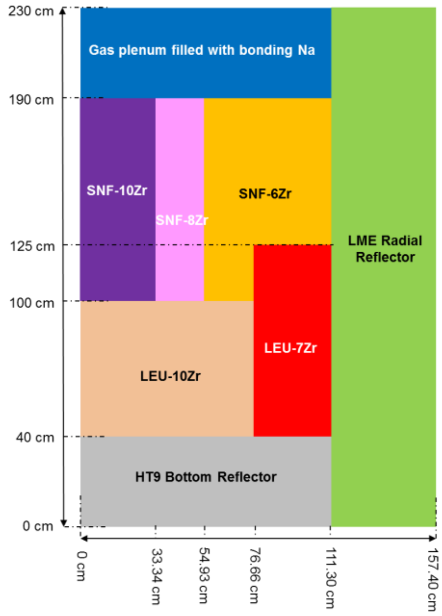

Fig 1. B&BR core arrangement

In the axial direction, a 40 cm axial HT-9 reflector is located at the bottom of the core, while a 40 cm thick gas plenum filled with sodium bonding is placed at the top of the core. The total active core height is 150 cm.

The fuel assembly (FA) consists of 127 fuel pins. The fuel pin diameter and P/D ratio are 1.9 cm and 1.064, respectively. The HT-9 cladding thickness is 0.06 cm. The assembly flat-to-flat distance is 23.75 cm. The assembly pitch is 24 cm. The resulting volume fraction of fuel, coolant and structure are 63.34%, 22.65 and 14.01, respectively.

As for the reflector assembly, a tighter lattice is adopted and 91 reflector pins are placed in each reflector assembly. The reflector pin diameter is 2.32 cm and the thickness of the HT-9 cladding is 0.10 cm. The reflector material is LME (lead magnesium eutectic) [3]. The cladding of reflector pin is thicker than that of the fuel pin in order to accommodate the possibility of corrosion caused by LME. In the reflector assembly, the volume fractions of LME, coolant, and structure are 68.06%, 18.47% and 13.47%, respectively.

3. Analysis Results and Discussion

The first sensitivity considered in this study is the impact of different nuclear data libraries on the core lifetime. Before performing the depletion calculations, the neutron reaction sub-library, decay sub-library, and neutron-induced fission product yields sub-library for each nuclear data library were prepared so that they can be read by the Serpent code. Since the decay reaction sub-library of JENDL-4.0 is not complete, the decay reaction sub-library of ENDF/B-VII.0 was used in the JENDL-4.0 calculations. In the depletion calculations, 100,000 neutron histories/cycle and 200 cycles with 50 skip cycles were adopted. Under this calculation conditions, the standard deviation of the multiplication factor (k-eff) is about 30~35 pcm. In the Serpent continuous-energy depletion calculations, the Chebyshev Rational Approximation Method (CRAM) [9] was chosen to solve the Bateman equations, and the option for predictor-corrector method was turned on. The depletion time step was varied from 5 to 100 days from the beginning of life (BOL) to 500 EFPDs (Effective Full Power Days) and after that, the depletion time step was increased to 500 days until EOL. Meanwhile, there are 2,340 depletion zones in the calculations since each fuel was divided into 30x5 cm axial regions.

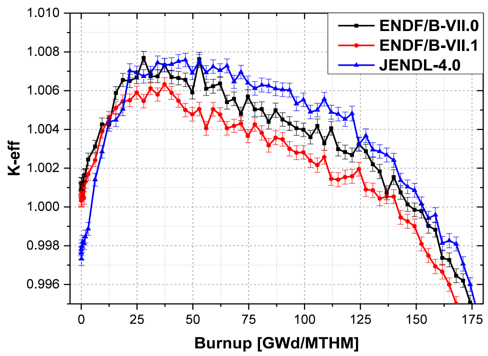

Fig 2. k-eff evolution with different nuclear data libraries

The change of the k-eff with burnup is shown in Fig. 2. At the BOL, the k-eff values by both of the ENDF/B-VII.0 and ENDF/B-VII.1 are slightly above critical, but the prediction by JENDL-4.0 is subcritical by ~200 pcm. As the burnup increases, the JENDL-4.0 prediction overestimates both the ENDF/B-VII.0 and ENDF/B-VII.1. It is noteworthy that JENDL-4.0 is closer to ENDF/B-VII.0. The k-eff evolution shows fluctuations because larger time steps were used in the depletion calculations in order to save computation time.

The second sensitivity considered is to the reactivity coefficients. The important reactivity coefficients evaluated here are Doppler reactivity coefficient α_D, coolant density reactivity coefficient α_{Na}, coolant void worth CVR, axial expansion reactivity coefficient α_H, and radial expansion reactivity coefficient α_R. Each reactivity coefficient was evaluated at 0 GWd/MTHM (BOL) and 150 GWd/MTHM (EOL). The coolant density reactivity coefficient and coolant void worth were evaluated by completely voiding the core. The axial expansion reactivity coefficient were calculated by increasing the height of fuel and cladding by 1% but the mass of the fuel and cladding were kept to be the same as the unexpanded core. On the other hand, the radial expansion coefficient were calculated by increasing the fuel assembly pitch by 1% but keeping the mass of the fuel and structure materials to be the same as the unexpanded core. The results of the reactivity coefficients with different nuclear data libraries are listed in Table I. It can be observed that the coefficients evaluated by each library were very close to each other and within the standard deviations except for the CVR.

Table I: Important reactivity coefficients with different nuclear data libraries.

Coefficients	BOL	EOL
α_D (¢/°C)		
ENDF/B-VII.0	-0.043 ± 0.004	-0.039 ± 0.009
ENDF/B-VII.1	-0.047 ± 0.004	-0.032 ± 0.009
JENDL-4.0	-0.041 ± 0.004	-0.037 ± 0.009
α_{Na} (¢/°C)		
ENDF/B-VII.0	-0.058 ± 0.001	0.185 ± 0.002
ENDF/B-VII.1	-0.060 ± 0.001	0.182 ± 0.002
JENDL-4.0	-0.061 ± 0.001	0.188 ± 0.002
CVR (¢)		
ENDF/B-VII.0	-204.01 ± 2.91	655.59 ± 6.19
ENDF/B-VII.1	-211.85 ± 2.92	644.11 ± 6.38
JENDL-4.0	-217.82 ± 1.89	667.68 ± 6.14
α_H (¢/°C)		
ENDF/B-VII.0	-0.029 ± 0.005	-0.074 ± 0.011
ENDF/B-VII.1	-0.030 ± 0.005	-0.067 ± 0.011
JENDL-4.0	-0.030 ± 0.005	-0.076 ± 0.010
α_R (¢/°C)		
ENDF/B-VII.0	-0.320 ± 0.006	-0.217 ± 0.012
ENDF/B-VII.1	-0.312 ± 0.006	-0.205 ± 0.012
JENDL-4.0	-0.309 ± 0.006	-0.227 ± 0.012

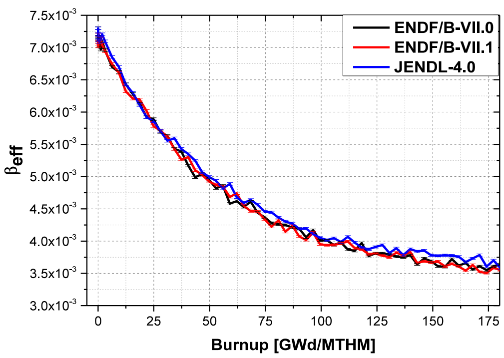

Fig 3. β_{eff} evolution with different nuclear data libraries

We also investigated the nuclear data library impact on the effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (*l*). Both of these integral kinetic parameters are important especially for transient and accident analyses. From the Serpent code calculation, the adjoint-weighted β_{eff} and *l* were provided and are shown in Figs. 3 and 4, respectively. One can observe that the three nuclear data libraries provide very similar β_{eff}, while the prompt neutron lifetime is slightly higher from JENDL-4.0.

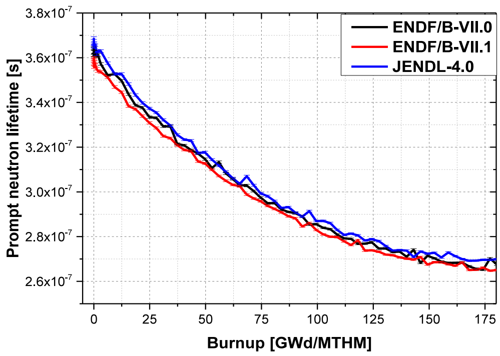

Fig 4. Prompt neutron lifetime evolution with different nuclear data libraries

The last sensitivity considered here is the fuel compositions, i.e. the buildup of the important actinides at EOL such as Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242, Am-241, and Am-242m. The result is given in Table II. The isotope composition at the EOL or at 150 GWd/MTHM is quite similar except for the Am-242m from JENDL-4.0 calculation.

Table II: Uranium and actinides buildup at EOL.

Isotope	At BOL [kg]	At EOL (ENDF/B-VII.0 / ENDF/B-VII.1 / JENDL-4.0) [kg]
U-235	2,627.8	326.01 / 326.77 / 330.01
U-238	37,216.8	30,128.4 / 30,127.64 / 30,162.9
Pu-238	4.17	41.68 / 43.59 / 42.35
Pu-239	112.00	2,513.6 / 2,512.6 / 2,488.2
Pu-240	52.90	414.2 / 414.1 / 413.9
Pu-241	17.98	22.87 / 23.22 / 23.68
Pu-242	13.45	15.84 / 16.03 / 16.10
Am-241	12.48	29.81 / 29.78 / 30.90
Am-242m	0.03	0.69 / 0.70 / 0.0015

4. Conclusions

A B&BR core has been characterized with three different evaluated nuclear data libraries, ENDF/B-VII.0, ENDF/B-VII.1 and JENDL-4.0. The comparison results show that the core excess reactivity is rather sensitive to the type of the library which in turn affects the estimated core life time of the B&BR core. However, a good agreement within the standard deviations on the integral kinetic parameters, Doppler reactivity coefficient, coolant density reactivity coefficient, axial expansion reactivity coefficient and radial expansion reactivity coefficient can be observed among the libraries. The coolant void reactivity (CVR) values of the three libraries slightly deviate beyond the standard deviations. With the different neutron libraries, the buildup of the fission products and minor actinides (especially Am-242m) are slightly different. In the future, depletion calculations with fine time steps should be performed to eliminate the error from large time steps.

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