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Assessment of the Dry Process Oxide Fuel in Sodium-Cooled Fast Reactors

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ABSTRACT

In order to assess the feasibility of the dry reprocessed oxide fuel in a sodium-cooled fast reactor, the reactor core analysis for the equilibrium fuel cycle was performed for two selected reference cores: the Hybrid BN-600 benchmark core with a enlarged lattice pitch (Case 1) and the modified BN-600 core (Case 2). The dry reprocess technology is based on the molten-salt process, which was developed by Russian scientists for recycling oxide fuels. The core calculation was performed by the REBUS-3 code and the reactor characteristics such as the transuranic enrichment, breeding ratio, peak linear power, burnup reactivity swing, etc. were calculated for the equilibrium core under a fixed fuel management scheme. The results showed that a self-sustainable breakeven core was achieved without blanket fuels when the fuel volume fraction was ～50% and almost all the fission products were removed.

I. INTRODUCTION

The development of a Generation-IV (Gen-VI) reactor system was initiated in order to extensively increase the safety and economical efficiency and to drastically minimize the radioactive waste of the conventional reactors by considering different fuel and coolant materials.[1] Considering the electricity demand after 20～30 years, six reactor concepts were selected by the Gen-IV International Forum (GIF), which were the gas-cooled fast reactor (GFR), lead-cooled fast reactor (LFR), sodium-cooled fast reactor (SFR), supercritical-water-cooled reactor (SCWR), very high temperature reactor (VHTR), and the molten salt reactor (MSR).

The dry reprocess technology can be utilized as a significant mean to control the pressurized water reactor (PWR) spent fuel disposition if a proper fuel cycle strategy and a safeguards system are adopted. In this study, the feasibility of the dry reprocessed fuel (DRF) cycle is assessed for the SFR which is recommended by the GIF. The feasibility calculation was performed for the equilibrium fuel cycle to assess the nuclear fuel cycle characteristics and reactor physics parameters of the SFR loaded with the DRF.

II. PHYSICS ANALYSIS MODEL

II.1. Reference Core Selection

In order to assess the applicability of the DRF to the SFR, it is required to determine the reference core model. The requirements of the reference core performance were set as the no-blanket and breakeven (breeding ratio $\geq$ 1.05) core, which avoided the separation process of transuranic (TRU) elements from the spent fuel and the provision of additional fissile material, respectively. The reference core for the DRF was determined to be BN-600. The BN-600 is a pool-type sodium-cooled prototype reactor, which produces an electric power of 600 MWe (1470 MWth) and uses UO_2-PuO_2 fuel. The dry reprocess technology used in this study is based on the molten-salt process, which has been developed by the Russian scientists for recycling the mixed oxide fuel.[2]

As shown in Figs. 1 and 2, two core models are considered as candidates of the reference core for the DRF. The first one (Case 1) is based on the BN-600 Hybrid core which is used as the benchmark core of the Co-ordinated Research Project (CRP) sponsored by the International Atomic Energy Agency (IAEA) started in 1999. [Ref. 3] The second one (Case 2) is based on the original BN-600 core in order to consider the real operating reactor conditions. For both cases, the reactor core is composed of two regions which are called "inner-core" and "outer-core" without the blanket region. The Gas Expansion Module (GEM) is deployed in the out-core region to reduce the coolant void reactivity. The axial configuration of the reference core is simplified from the BN-600 core such that the lower shield in the bottom of the core is 100 cm thick, the sodium bond area is 20 cm thick and the fission gas plenum in the top of the core is 120 cm thick. The axial active heights are 100 cm for Case 1 and 110 cm for Case 2, respectively. The selected reference core specifications are given in Table 1.

II.2 Codes and Libraries

The core calculation is performed by the REBUS-3 code[4] (including DIF3D) using the KAFAX-F22 library.[5] The KAFAX-2.2 is a neutron 80-group and gamma 24-group cross-section library based on JEF-2.2 for the fast reactor applications, which is a MATXS formatted library containing 89 isotopes. The TRANSX and TWODANT codes[6,7] are used to generate the 9-group region-wise effective macroscopic cross-sections in an ISOTXS form. The fission products not included in the burnup chain are represented by lumped fission products (LFP). The cross-sections for the LFP are generated from the KAFAX-E6FP cross section library[8] by collapsing 80-group cross sections into 9-group ones with the typical fast reactor neutron spectrum. The core analysis is then performed by the REBUS-3 using the ISOTXS cross section data. In this study, the coarse-mesh nodal diffusion method is used for the depletion calculation in the hexagonal-z reactor configuration and the finite difference method is used for the reactivity coefficient calculation in the triangular-z configuration.

III. EQUILIBRIUM CORE ANALYSIS

The breeding ratio of the SFR changes depending on the fuel pin diameter (or fuel volume fraction). At first, in order to find the appropriate fuel volume fraction to maintain the core performance requirement on the breeding ratio, the reactor core analysis was performed for the reference core which has the same TRU enrichment in the inner-core and outer-core at the beginning of the equilibrium cycle (BOEC). The reactor characteristics such as the TRU enrichment, breeding ratio, peak linear power, burnup reactivity swing, etc. were calculated for the equilibrium core under a fixed fuel management scheme. Secondly, for the selected volume fractions of the fuel, cladding and coolant, the equilibrium core calculation was performed to reduce the maximum power density or peak linear power in the core, depending on the TRU enrichment of the inner-core and outer-core at the BOEC. Based on the reactor characteristics obtained from the equilibrium core calculation, an appropriate TRU enrichment was determined. Then the fuel temperature coefficients and coolant void reactivity were calculated for the reference core.

In this study, only the neutronics calculation was performed for the reference core. In other words, the detailed design calculations of the fuel assembly and fuel channel are beyond the scope of this study. The external fuel cycle strategies for the equilibrium core are as follows:

- 95% of the rare-earth and all other fission products are removed and sent to the waste stream.
- All uranium isotopes and 99.9% of the TRU are recovered; and 0.1% of the TRU is lost and sent to the waste stream.
- All surplus fuel materials after the reprocessing process are sold.
- Assumed periods simulating the external fuel cycle are as follows:
  - Cooling time after discharge: 2 months
  - Reprocessing time: 8 months
  - Refabrication time: 8 months
  - Pre-loading storage time: 2 months.

The isotopic compositions of the external feed to achieve the equilibrium cycle are composed of the TRU recovered from the typical light water reactor (LWR) spent fuel after cooling for 3.17 years and the depleted uranium. The recycling equilibrium mode calculation was performed to establish a self-sustaining breakeven core without an external source of fissile material, aiming at the breeding ratio of 1.05 under the condition that 99.9% of the TRU were recovered during the reprocessing process. For the equilibrium core, the core criticality was maintained with all the control rods withdrawn.

III.A. Sensitivity Calculation on the Physics Design Parameters

III.A.1 Reactor characteristics depending on the fuel volume fraction

The breeding ratio is a ratio of the fissile material produced to the fissile material destroyed during a fuel cycle, which is a measure of self-sustaining of the reactor with no additional supply of the fissile material. Because the breeding ratio strongly depends on the neutron spectrum of the core, appropriate volume fractions of the fuel, cladding and coolant of the fuel assembly were searched for the selected cores under the condition that the breeding ratio was maintained at 1.05. As shown in Fig. 3, the breeding ratio is maintained over 1.05 when the fuel volume fractions are over 47% and 48% for Case 1 and Case 2, respectively.

Burnup reactivity swing is the magnitude of the reactivity variation during a fuel cycle, which has a direct impact on the available shutdown margin of the control system and the manipulation in case of several reactivity-induced transients. So, the burnup reactivity swing should be kept at the minimum value during the cyclic operation. The burnup reactivity swing is also affeacted by the volume fraction of the fuel. The burnup reactivity swings for Case 1 and Case 2 are shown in Fig. 4, in which the magnitude is greater if the volume fraction is higher.

III.A.2 Reactor characteristics depending on the TRU content

Linear power density is closely related to the center temperature of the fuel rod and is directly proportional to the power density (power generation per unit volume of the fuel). To reduce the peak linear power density and flatten the power distribution of the core, the equilibrium core calculations were performed for various TRU enrichment distributions in the inner-core and outer-core at the BOEC. As described in Sec. II.4.1, the breeding ratio is over 1.05 when the fuel volume fraction is greater than 47% and 48% for Case 1 and Case 2, respectively. However, because the breeding ratio has a decreasing tendency as the difference of the TRU enrichment of the inner-core and outer-core increases, the reactor characteristics calculations for the TRU enrichment were performed for 49% and 51% of the fuel volume fraction for Case 1 and for 51% and 53% of the fuel volume fraction for Case 2, respectively.

For Case 1, the reactor core characteristics are given in Table 2 for 49% and 51% of the fuel volume fraction. When the TRU enrichments of the inner-core and outer-core are the same, the power peaking occurs at the core center and moves to the outer core region as the out-core enrichment increases. In order to achieve the target breeding ratio of 1.05, the TRU enrichments of the inner-core and outer-core should be 12.63% and 15.84%, respectively, for the fuel volume fraction of 49% and 10.30% and 16.48%, respectively, for the fuel volume fraction of 51%. For Case 2, the breeding ratio is near 1.05 if the TRU enrichments of the inner-/outer-core are 11.99%/16.43% and 10.67%/17.63% for the cases of the fuel volume fractions of 51% and 53%, respectively.

For Case 1, the average linear power density is 114.44 W/cm; and the peak linear power densities for the selected fuel volume fractions are 170.89 W/cm and 205.38 W/cm when the TRU enrichments of the inner-/outer-core are 12.63%/15.84% and 10.30%/16.48%, respectively. For Case 2, the average linear power density is 142.58 W/cm; and the peak linear power densities for the selected fuel volume fractions are 245.41 W/cm and 221.73 W/cm when the TRU enrichments of the inner-/outer-core are 11.99%/16.43% and 10.67%/17.63%, respectively. On the other hand, the average and peak linear power densities of the BN-600 are 360 W/cm and 530 W/cm, respectively. Because the reactor cores considered in this study have no blankets, the power load for a fuel assembly is considerably reduced and the average and peak power densities are appreciably decreased when compared to those of the BN-600 reactor.[9]

III.B Reactivity coefficient

The DIF3D code is used for the core reactivity calculations for various TRU enrichments of the inner-core and outer-core. The isotopic compositions of the BOEC and the end of equilibrium cycle (EOEC) states are obtained from the equilibrium cycle calculation of the REBUS-3 code for each TRU enrichment. Then the fuel temperature coefficients and the coolant void reactivity at the BOEC and EOEC are calculated by the DIF3D using the finite difference method in the triangular-z node. The reactivity calculations are performed directly for the perturbed and unperturbed systems.

III.B.1 Coolant void reactivity

The coolant void reactivity is an important parameter for the reactor design. The loss of sodium from the Liquid Metal Reactor (LMR) results in a decreased moderation of the neutrons such that the average neutron energy increases, i.e., the neutron spectrum is hardened, which causes a positive reactivity effect due to the increase of neutron importance with increasing energy. Since the void reactivity is positive for the SFR, it is important to accurately estimate the coolant void reactivity. The coolant void reactivity is calculated at the operating temperature by reducing the coolant density from the nominal value to a perturbed value such as

$$\alpha_V(\mathrm{pcm}) = 10^5 \times \left(\frac{1}{k_{nominal}} - \frac{1}{k_{void}} \right).$$

It is assumed that the voiding of sodium coolant occurs in the active core consisting of all fuel and GEM assemblies.

As shown in Table 3, the coolant void reactivity generally shows a decreasing tendency as the difference of the TRU enrichment between the inner-core and outer-core increases. Sodium loss in the core center region results in a highly positive reactivity effect, while the sodium loss from the core periphery causes a negative effect due to the increased neutron leakage. As the difference of the TRU enrichment between the inner-core and outer-core increases, the reactivity from the core center decreases due to the relatively low TRU enrichment of the inner core. At the same time, the coolant void reactivity decreases as the fuel volume fraction increases because the fraction of the sodium coolant relatively decreases in the core. In addition, the void reactivity of the EOEC core is larger than that of the BOEC core, as the difference of the TRU enrichment between the inner-core and outer-core increases. The plutonium buildup through the fuel irradiation also contributes to the void reactivity increase.

For Case 1, the TRU enrichments of the inner-/outer-core to achieve the breeding ratio of 1.05 is 12.63%/15.84% and 10.30%/16.48% for the fuel volume fractions of 49% and 51%, respectively. The void reactivity of 49% and 51% fuel volume fractions are 2636 pcm and 2087 pcm for the BOEC and 2696 pcm and 2267 pcm for the EOEC, respectively. For Case 2, the void reactivity of 51% and 53% fuel volume fractions are 2636 pcm and 2087 pcm for the BOEC and 2696 pcm and 2267 pcm for the EOEC, respectively. These results are comparable with the full core void reactivity (~3000 pcm) of the KALIMER-600 reactor.[10]

III.B.2 Fuel Temperature Coefficient

Like the coolant void reactivity, the fuel temperature coefficient is one of the most important parameters for the reactivity-induced transient analysis of the LWRs and LMRs. The fuel temperature coefficient is generally negative owing to the Doppler effect of the capture resonances in the fertile material of the fuel. However, the fuel temperature coefficient tends to be positive in a core with a harder neutron spectrum, because the neutron population is reduced in the resonance region, which contributes to the resonance capture to U-238. The fuel temperature coefficient is defined as the change of the effective multiplication factor per fuel temperature change such as

$$\alpha_T(\text{pcm/K}) = \frac{k_1 - k_2}{T_1 - T_2} \times 10^5,$$

where k_1 and k_2 are the effective multiplication factors corresponding to fuel temperatures T_1 (operating temperature: 1273K) and T_2 (room temperature: 293K), respectively.

In contrast with the void reactivity, the fuel temperature coefficient in general increases as the difference of the TRU enrichment between the inner-core and outer-core and the fuel volume fraction decrease. At the same time, the fuel temperature coefficient of the EOEC core is lower than that of the BOEC core, because the fissile plutonium isotopes are significantly built up and U-238 transmutes as the fuel is irradiated. For the TRU enrichment of the inner-/outer-core achieving the breeding ratio of 1.05, the fuel temperature coefficient ranges from -0.93 pcm/K to -1.05 pcm/K for both the Case 1 and Case 2.

III.C Fuel Mass Flow and Inventory

The breeding ratio is closely related to the fuel mass flow of the equilibrium cycle. In this study, the fuel mass flow for each step of the external fuel cycle is calculated to estimate the amount of fuels required and/or removed in each external reprocessing step when achieving the equilibrium core.

For Case 1, the fissile plutonium gains during one cycle of the equilibrium core are 32.84 kg and 41.40 kg for the fuel volume fractions of 49% and 51%, respectively, which satisfy the self-sustaining breakeven core without an excess fissile material. The amount of minor actinides is slightly reduced during one cycle in the core, and built up a little during the external reprocessing process by the decay. The total amount of minor actinides to be sold is 1.43 kg and 1.55 kg per cycle for the fuel volume fraction of 49% and 51%, respectively. The increase in the amount of fission products is 710 kg per cycle; 5% of the rare-earth fission products are recovered and all other fission products are removed during the reprocess. The amount of depleted uranium to be supplied through the external feed is about 740 kg per cycle.

For Case 2, the fissile plutonium gains during one cycle of the equilibrium core are 32.24 kg and 32.16 kg for the fuel volume fraction of 51% and 53%, respectively. Like Case 1, the minor actinides are slightly reduced during the cycle in the core and built up a little during the external fuel cycle by the decay. The amount of surplus minor actinides is 1.42 kg and 1.39 kg per cycle for the fuel volume fractions of 51% and 53%, respectively. The fission products are produced by 710 kg per cycle. The required depleted uranium through the external feed is about 745 kg per cycle.

III.D Sensitivity of the Breeding Ratio to the Fission Products Recovery Factor

So far, the equilibrium fuel cycle calculations were performed by assuming that 95% of the rare-earth and all other fission products were removed, and all uranium isotopes and 99.9% of the TRU isotopes were recovered. In addition, a target breeding ratio of 1.05 was set to maintain a self-sustaining breakeven core, considering 99.9% recovery of TRU isotopes during the reprocessing process. It is however difficult to completely extract rare-earths during the reprocessing process, which consist of lanthanide series isotopes and Y (Yttrium). The amount of non-rare-earth fission products is ~80% of the total fission products when the fuel is discharged from the core. In order to assess the feasibility of applying a more simple dry process which may not remove all the fission products from the spent fuel, sensitivity calculations were performed on the fission products removal rate of the dry process.

In order to estimate the effect of the fission products removal rate, the equilibrium cycle calculations were performed for various removal rates of the rare-earth and non-rare-earth fission products. In the previous calculation, the breeding ratio was used as a measure of achieving the breakeven core. In this section, however, the amount of surplus TRU materials to be sold during the reprocessing is searched instead of the breeding ratio. The surplus TRU material is a more accurate factor for estimating the self-sustaining breakeven core and the amount of surplus TRU material is calculated for one cycle. In the calculation, the removal rate of the rare-earth was changed from 0% to 90%, and the removal rate of the non-rare-earth was 100%, 80%, and 60%. The criticality was maintained through out the fuel cycle.

When the fuel volume fraction is 49% for Case 1, 30%, 35% and 55% of the rare-earth fission products have to be removed for 100%, 80% and 60% removal of the non-rare-earth components, respectively, in order to achieve an equilibrium cycle core. For the 51% fuel volume fraction, 35%, 45% and 85% of the rare-earth fission products have to be removed for 100%, 80% and 60% removal of the non-rare-earth components, respectively. When the fuel volume fraction is 51% for Case 2, 35%, 45% and 90% of the rare-earth fission products have to be removed to achieve an equilibrium cycle core for 100%, 80% and 60% removal of the non-rare-earth components, respectively. For the 51% fuel volume fraction, 35%, 40% and 85% of the rare-earth fission products have to be removed for 100%, 80% and 60% removal of the non-rare-earth components, respectively.

VI. SUMMARY AND CONCLUSION

In order to verify the applicability of the DRF to the SFR, the equilibrium core calculation was performed for the selected reference cores. The dry fuel reprocessing technology used in this study was based on the molten-salt process, which was developed for recycling oxide fuels. At first, two kinds of fuel volume fractions for achieving 1.05 of the breeding ratio were searched for the two reference cores with the same TRU enrichments in the inner-core and outer-core at the BOEC. Then, appropriate TRU enrichments of the inner-core and outer-core were searched for the reference cores with the selected fuel volume fractions. The fuel temperature coefficient, void reactivity, fuel mass flow per cycle and the sensitivity of the fission products removal rate were calculated for the finally selected reference cores.

In this study, four kinds of reference cores were analyzed for the fuel volume fraction and TRU enrichment, without considering the detailed design of the fuel assembly and fuel channel. However, if the design criteria used in this study are proved to be acceptable through the detailed physics design and thermal hydraulic analysis in the future, it is practically possible to construct an equilibrium fuel cycle of the SFR based on the oxide fuel utilizing the dry reprocessing technology. Especially, the oxide fuel was used for the PHENIX reactor of France and the JOYO and MONJU reactors of Japan, etc.; and the performance of the oxide fuel was established through a lot of irradiation experiments. However, because the oxide fuel has a lower thermal conductivity, it is required to consider the center line temperature of the fuel rod. In addition, because the oxide fuel core has a softer neutron spectrum when compared to the metallic fuel and nitride fuel cores, it may be necessary to study the applicability of the Pb-Bi coolant to have a harder neutron spectrum core in the future.

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REFERENCE

1. "Generation 4 Roadmap Fuel Cycle Cross Cut Croup Executive Summary," U.S. DOE Nuclear Energy Research Advisory Committee and the Generation VI International Forum, 2001.

2. A. V. Bychkov et al., "Fuel Cycle of Actinide Burner-Reactor. Review of Investigation by <<DOVITA>> Program," Proc. of Int. Conf. on Future Nuclear Systems, Challenge Towards Second Nuclear Era with Advanced Fuel Cycles: Global'97, Yokohama, Oct.5-10, 1997.

3. IAEA, "Working Material: Updated Codes and Methods to Reduce the Calculational Uncertainties of Liquid Metal Fast Reactors Reactivity Effects, The fourth Research Co-ordination Meeting," Obninsk, 19-23 May 2003, IAEA-RC-803.4, TWG-FR/113, 2003.

4. B. J. Toppel, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, Argonne National Laboratory, 1983.

5. J. D. Kim and C. S. Gil, "KAFAX-F2.2: Development and Benchmark of Multi-group Library for Fast Reactor Using JEF-2.2," KAERI/TR-842/97, Korea Atomic Energy Research Institute, 1997.

6. R. E. MacFarlane, "TRANSX 2: A Code for Interfacing MATXS Cross-Section Libraries to Nuclear Transport Codes," LA-12312-MS, Los Alamos National Laboratory, 1993.

7. R. E. Alcouffe et al., "User's Guide for TWODANT: A Code Package for Two-Dimensional, Diffusion-Accelerated, Neutron Transport," LA-10049-M, Los Alamos National Laboratory, 1990.

8. J. D. Kim, "Generation of Lumped Fission Product Cross Sections for Fast Reactors," NDL-25/99, Korea Atomic Energy Research Institute, 1999.

9. A. E. Walter and A. B. Reynolds, *Fast Breeder Reactors*, Pergamon Press, New York, 1981.

10. H. Song et al., "Nuclear Design of a NA Cooled KALIMER-600 Core with No Blanket," Advances in Nuclear Fuel Management III (ANFM 2003), Hilton Head Island, Oct. 5-8, 2003.

Table 1. Specification of the Reference Core

	Modified BN-600 Hybrid Core (Case 1)	Modified BN-600 Core (Case 2)
• Power	600 MWe / 1470 MWth	600 MWe / 1470 MWth
• Fuel	$U_xPu_{(1-x)}O_2$	$U_xPu_{(1-x)}O_2$
- Density	10.5 g/cm^3	10.5 g/cm^3
- Fuel Smeared Density	85.0%	85.0%
• Clad Material	HT9	HT9
• Coolant Material	Na	Na
• Refueling Interval	18 Months	18 Months
• Effective Full Power Day	465 Days	465 Days
• Plant Capacity Factor	85.0%	85.0%
• Number of Batches	3	3
• Active Core Height	100 cm	110 cm
• Duct Pitch	16.1 cm	9.9 cm
• Number of Fuel Assemblies		
- Inner-Core	180	372
- Outer-Core	294	366
• Pins per Fuel Assembly	271	127
• Fertile Feed	Depleted Uranium	Depleted Uranium
• Fissile Feed	Reprocessed LWR Spent Fuel	Reprocessed LWR Spent Fuel

Table 2. Sensitivity of Reactor Characteristics to the TRU Content (Case 1)

Fuel volume faction (%)	TRU enrichments (%) (inner-/outer-core)	Breeding ratio	Burnup reactivity swing (pcm)	Peak linear power density (W/cm)	Max. discharge burnup (MWd/kg)
49.0	8.54/18.24	1.01239	1076.24	215.94	90.53
	9.80/17.80	1.01940	710.85	206.00	86.18
	11.02/17.14	1.02871	247.00	192.97	80.53
	12.15/16.27	1.04095	247.39	175.73	73.94
	12.63/15.84	**1.04711**	**414.66**	**170.89**	**72.48**
	13.06/15.10	1.05563	529.61	196.03	80.72
	12.14/12.14	1.07150	428.92	254.72	108.56
51.0	7.68/17.28	1.02863	971.17	229.47	88.80
	9.00/17.00	1.03575	624.18	219.08	84.68
	10.30/16.48	**1.04531**	**165.04**	**205.38**	**79.25**
	11.44/15.60	1.05827	372.90	186.54	72.11
	12.36/14.42	1.07440	738.06	202.15	76.32
	13.00/13.00	1.09164	678.78	269.37	106.12

Table 3. Sensitivity of Reactivity to the TRU Enrichment (Case 1)

Fuel volume fraction (%)	TRU enrichment (%) (inner-/outer-core)	Coolant void reactivity (pcm)		Fuel temperature coefficient (pcm/K)	
		BOEC	EOEC	BOEC	EOEC
49.0	8.54/18.24	1802.74	2033.06	-0.91127	-0.90651
	9.80/17.80	2006.89	2233.26	-0.94643	-0.94283
	11.02/17.14	2265.99	2457.89	-0.99235	-0.98565
	12.15/16.27	2540.93	2645.70	-1.04359	-1.02324
	12.63/15.84	**2636.83**	**2696.13**	**-1.06214**	**-1.03290**
	13.06/15.10	2708.39	2725.91	-1.07635	-1.03705
	12.14/12.14	2688.91	2704.22	-1.07286	-1.02542
51.0	7.68/17.28	1661.44	1862.25	-0.89122	-0.88469
	9.00/17.00	1844.50	2045.99	-0.92406	-0.91929
	10.30/16.48	**2087.72**	**2267.40**	**-0.96845**	**-0.96258**
	11.44/15.60	2368.25	2472.43	-1.02157	-1.00442
	12.36/14.42	2558.78	2568.70	-1.05844	-1.02139
	13.00/13.00	2546.91	2547.68	-1.05419	-1.00782

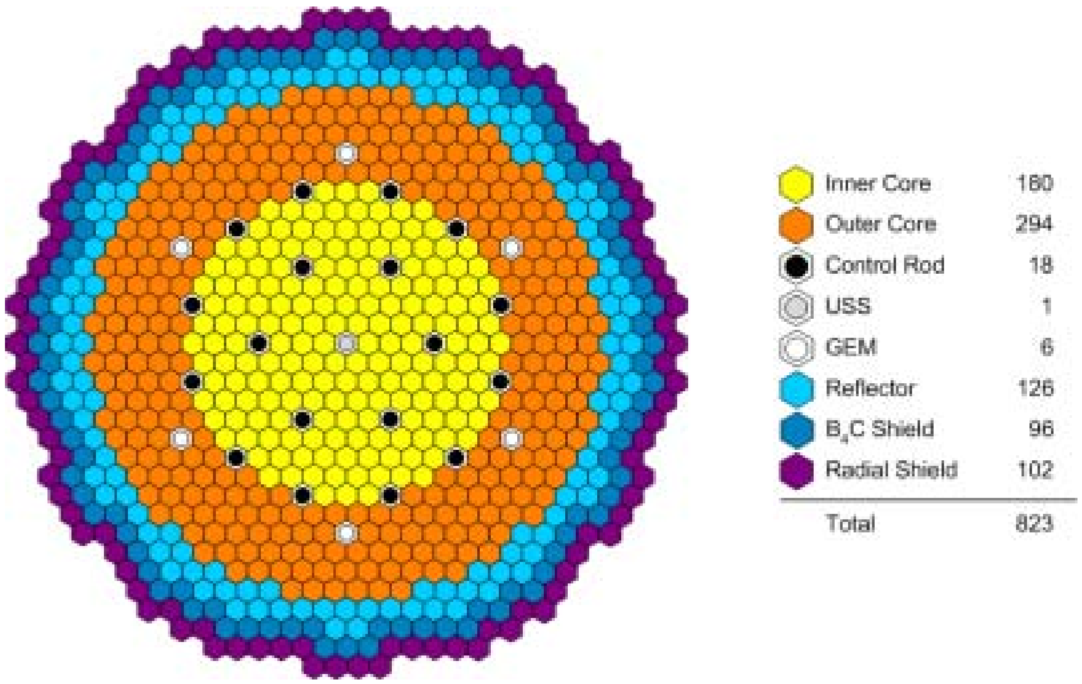

Fig. 1. Radial Configuration of the Modified BN-600 Hybrid Core (Case 1)

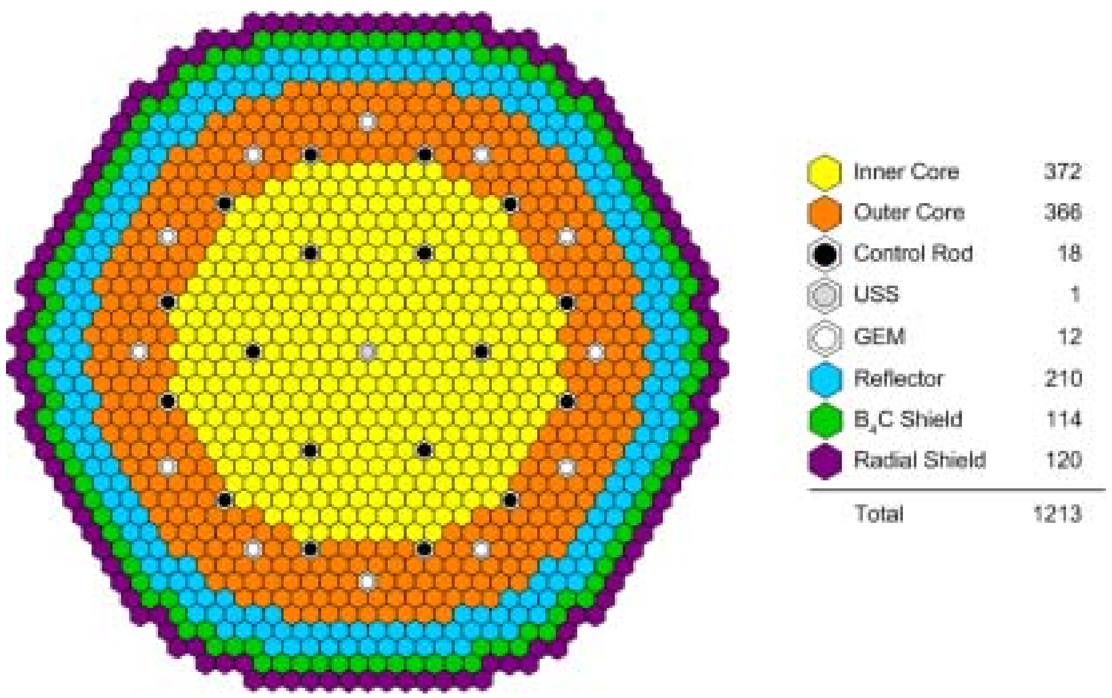

Fig. 2. Radial Configuration of the Modified BN-600 Core (Case 2)

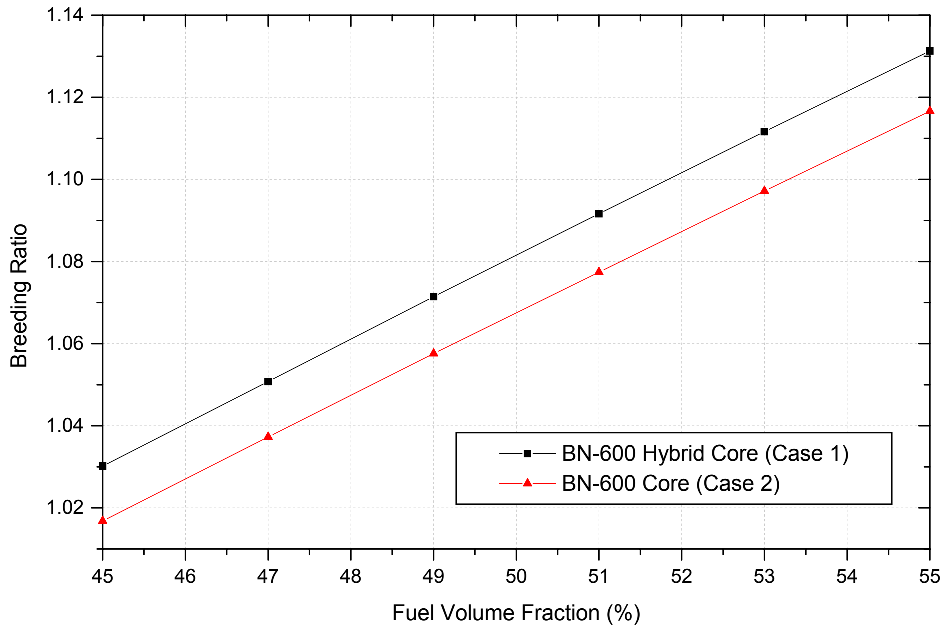

Fig. 3. Breeding Ratio vs. Fuel Volume Fraction

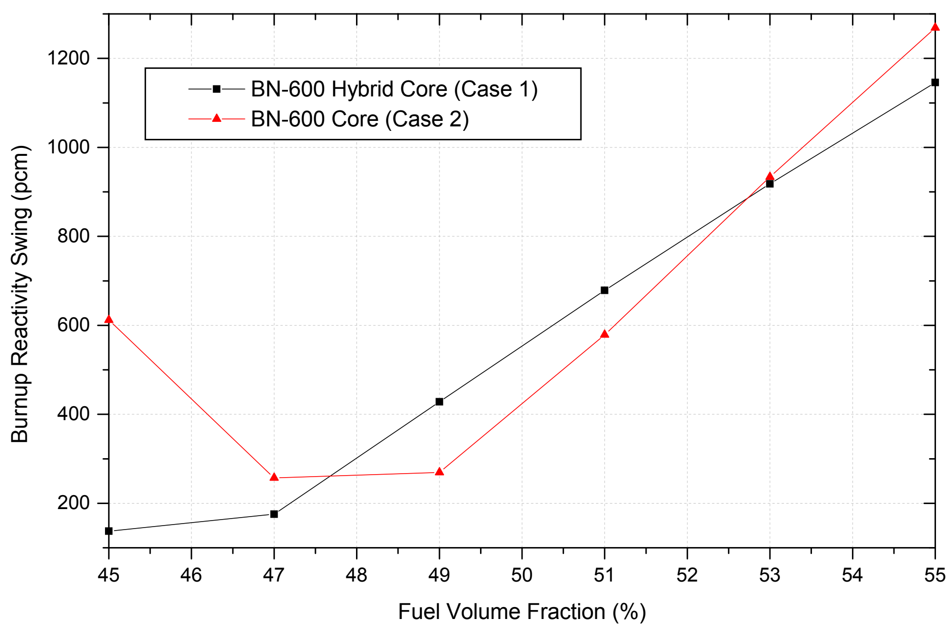

Fig. 4. Burnup Reactivity Swing vs. Fuel Volume Fraction