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Self-sustaining Core Design for 200 MWe Molten-Salt Reactor with Thorium-Uranium Fuel :FUJI-U3-(0)

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Abstract: *In this paper, an improved design for a small Molten-Salt Reactor (MSR) by using neutron flux flattening is proposed, which is referred as FUJI-U3-(0). This reactor is a 200 MWe power reactor, and its core contains graphite as the moderator and fuel salt. The fuel salt is composed of ThF_4 as the fertile material, $^{233}UF_4$ as the fissile material, and LiF-BeF_2, working as both fuel and heat transfer medium.*
A basic improvement in FUJI-U3-(0) is the optimization of the three-region core design in order to avoid the replacement of graphite, which is achieved by reducing the maximum neutron flux and minimizing the radiation damage of the graphite.
Based on the above concept, 2 million cases were investigated using the nuclear analysis code SRAC95, and the final case was proposed as the optimized three-region core. For the burnup calculation, the burn-up analysis code ORIGEN2 was also used,
It is concluded that there is no need to replace the graphite for 30 years with a load factor of 75%. The conversion ratio (CR) varies with burn-up of fuel, but the lifetime average CR is 1.01. The initial fissile inventory is 1.1 t and the feed for 30 years is 0.4 t, and the residual is 1.5 t. This means that this design is essentially a self-sustaining core. It is also shown that FUJI-U3-(0) produces very small amount of plutonium and Minor Actinides owing to the thorium-uranium fuel cycle application.

Keywords: Molten-Salt Reactor (MSR), Thorium cycle, Core design.

1. INTRODUCTION

The Molten-Salt Reactor (MSR) [1,2] uses a molten-salt fluid fuel, which contains thorium (Th) and uranium-233 (^{233}U) as the fertile and fissile materials, respectively. Further, the MSR can use plutonium (Pu) as the fissile material, as shown in our previous paper [3]. Besides the benefit with regard to the nuclear fuel cycle, the MSR has many other advantages such as flexibility in the power capacity, which is shown in our earlier papers for a capacity range of 150 MWe [4,5] to 1,000 MWe [6]. In addition, other advantages are the effective incineration of minor actinides (MA) [7], excellent safety [8,9], and cost-effectiveness [9], which are also shown in the authors' publications.

With regard to the nuclear fuel cycle, the Th cycle produces very small amounts of Pu and MA when compared with the conventional U fuel cycle. Some of the results are shown in this paper. Furthermore, ^{233}U accompanied by high gamma-ray emissions contributes to the nuclear proliferation resistance [9].

The MSR was studied at the Oak Ridge National Laboratory (ORNL), and the Molten-Salt Reactor Experiment (MSRE) showed excellent operation for four years during 1965–69 [10]. Subsequently, the conceptual design work was initiated for Molten-Salt Breeder Reactor (MSBR) with a capacity as large as a 1,000 MWe plant [1,2].

In order to deploy the MSR widely in the world, Dr. Furukawa and the authors have proposed a small MSR with a power output of 150 MWe (FUJI-II [4] and FUJI-12 [5]). FUJI-12 had a simple single-region core configuration, and it attained a high conversion ratio of 0.92 by employing batch chemical processing every 7.5 years. However, FUJI-12 required the replacement of the graphite moderator every 15 years due to limited irradiation growth in the graphite, as explained in the next section, and this may cause additional maintenance work and cost.

In this paper, an improved design for a small MSR (referred to as FUJI-U3-(0), or just as FUJI-U3) is proposed by using neutron flux flattening. The basic improvement is the introduction of the design concept of a three-region core in order to avoid the replacement of graphite; which is achieved by reducing the maximum neutron fluence. Therefore, even if FUJI-U3 is operated for 30 years, there will be no need to replace the graphite moderator.

2. DESIGN CONDITIONS

The electric power of FUJI-U3-(0) is 200 MWe, and the thermal power is 450 MW(th), which implies a thermal efficiency of 44.4%. The fuel salt is initially composed of mixed fluoride of ThF_4 (12 mol%), BeF_2 (16 mol%), and $LiF + {}^{233}UF_4$ (72 mol%). Here, Li is enriched lithium corresponding to ^7Li (99.995 mol%), and the relative proportion between LiF and $^{233}UF_4$ is a variable in this design.

The graphite moderator is used without being replaced for 30 years of the reactor lifetime. Since FUJI-U3 is a small reactor, load-following operation is assumed according to the electric power demand. Assuming 100% power during 14 hours of operation in the daytime, 50% power during 8 hours of operation in the nighttime, and a linear change in between, the average load factor is 81%. Further, assuming 12 months of operation and 1 month of downtime, the average load factor amounts to 75%. Therefore, the average load factor in this study is assumed to be 75%. The abovementioned 30 years operation period is based on the assumption that the core internals are enclosed in the reactor vessel that is not opened during the period. If a longer operation period is required, this vessel may be replaced along with the core internals such as the graphite moderator.

Assuming 30 years of operation with a 75% load factor, the irradiation limit of the graphite moderator equals 4.2×10^{13} 1/(cm^2·s) for the fast neutron flux of energy higher than 52 keV. Furthermore, the irradiation limit of the vessel, which is made of Hastelloy-N (Ni-based alloy with Mo/Cr/Nb/Si), is 1.4×10^{11} 1/(cm^2·s) for the fast neutron flux of energy higher than 0.8 MeV, and 7.1×10^{12} 1/(cm^2·s) for the thermal neutron flux of energy lower than 1.0 eV. These limits are based on the MSBR design [1]. These limits are slightly old, and may be revised for recent improvements in the manufacturing technology.

When ^{232}Th captures a neutron, ^{232}Th transmutes to ^{233}Pa (Protoactinium-233), and it decays to ^{233}U with 27 days half-lifetime. In the actual burnup calculation shown in Section-5, full power operation is assumed, but, in the actual FUJI operation, there is an abovementioned 1-month shutdown time, and almost half of ^{233}Pa in the core decays to ^{233}U in this down time. Since the reactivity-loss due to Pa is about 0.5% delta-K, it means that about 0.25% delta-K higher reactivity is added due to this effect, and this will improve neutron economy a little bit. But, after the reactor starts again, the core becomes equilibrium condition soon, and Pa content reaches the previous equilibrium value. Therefore, this Pa effect is neglected in the calculation.

3. CALCULATION PROCEDURE FOR CRITICALITY

The criticality of FUJI-U3 is calculated using the nuclear analysis code SRAC95 [11]. At first, a collision probability routine PIJ [12] is applied with 107 energy groups for the unit fuel cell model, which is shown in Fig. 1. Nuclear cross sections of the 107 groups are compressed into 30 groups, which are composed of 24 fast neutron groups and 6 thermal neutron groups. Finally, a diffusion calculation by the CITATION [13] routine is performed using the cross sections of the abovementioned 30 groups. In the two-dimensional RZ diffusion calculation, the core is divided into 65 radial regions and 32 axial zones. In these calculations, JENDL3.2 [14] is used as a nuclear library.

In the fuel cell calculation, the temperature of the fuel cell (comprising the fuel salt and graphite moderator) is assumed to be 900 K, which is the mean value between the core inlet and outlet temperatures. With regard to the RZ calculation for the core, the temperature of the entire core is also assumed to be 900 K. The effect of the axial temperature distribution has been evaluated in a separate report [15]. According to this report, the actual temperature is higher in the upper part of the core than in the lower part, and this causes a slightly lower neutron flux at the upper part of the core due to a decrease in temperature reactivity. However, the influence of the assumption of a constant temperature on the neutron flux is small, i.e., approximately 2% to 3% difference. Therefore, in this paper, a constant temperature is assumed for the entire core.

As shown in Fig. 1, the FUJI core is composed of graphite moderator and fuel salt. In this study, hexagonal graphite (p=0.19m) is modeled as a cylindrical element (D=0.20m), and the flow hole in the graphite is designed to be cylindrical (d=variable); this flow hole acts as a passage for the flow of the fuel salt. The lower part of Fig. 1 shows the fuel cell model considered in this calculation.

For the calculation of criticality, the size of the reactor core and the graphite reflector (see Fig. 3) are varied. The volume fraction of the fuel salt in each region (Fr) is varied by changing the flow-hole diameter (=d) of graphite. With regard to the fuel salt, the total percentage concentration of (LiF + $^{233}UF_4$) is fixed as 72 mol%, while the $^{233}UF_4$ concentration is varied in order to achieve criticality.

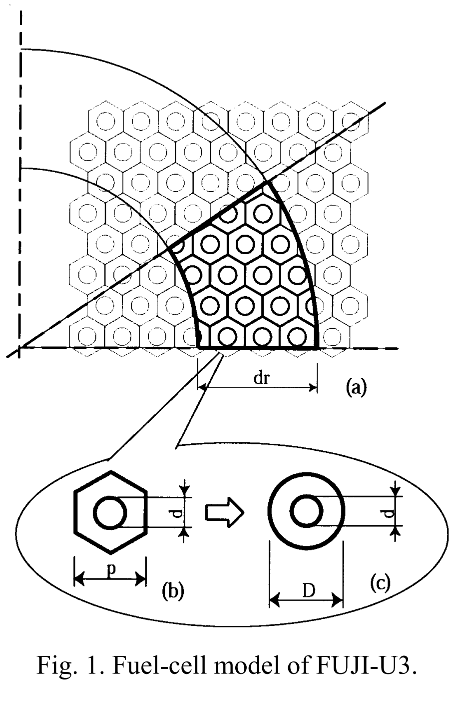


Fig. 1. Fuel-cell model of FUJI-U3.

The eigenvalue calculation is performed using SRAC95, and in this calculation, the neutron effective multiplication factor (Keff), fuel conversion ratio (CR), temperature coefficient of the reactivity, and neutron flux distribution (ϕ) are calculated. The obtained value of the neutron flux distribution is compared with the irradiation limit, and if it does not meet the design criteria, the above core parameters are varied until all design conditions are met.

The calculation procedure discussed above is shown in Fig. 2, where both the criticality calculation (discussed in this section) and the burnup calculation (discussed in the next section) are described.

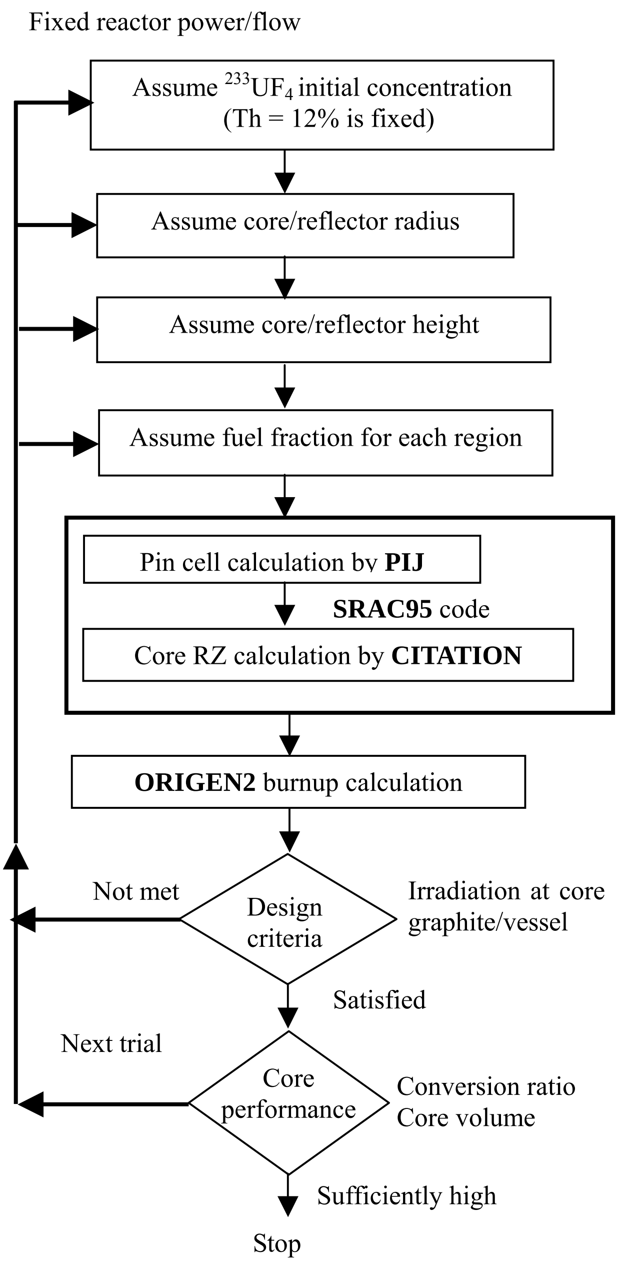
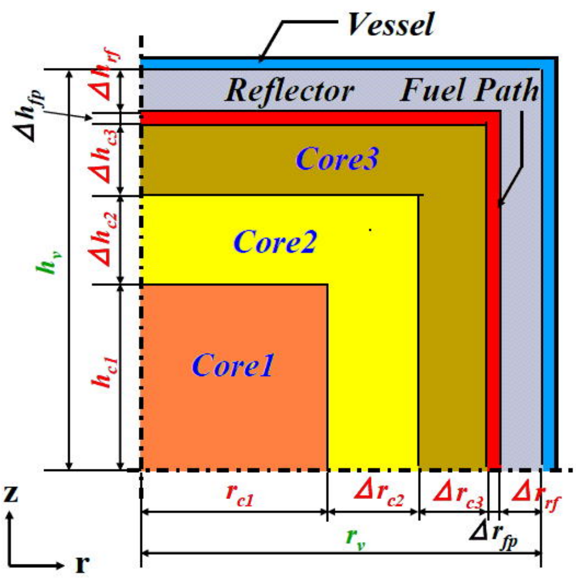


Fig. 2. Calculation procedure for FUJI-U3.

4. CRITICAL PROPERTY AND MAIN RESULTS

Fig. 3 shows the core configuration with dimensions and the volume fractions (Fr) of the fuel salt in each region.



	Core 1	Core 2	Core 3
r or Δr (m)	1.16	0.80	0.40
h or Δh (m)	1.23	0.70	0.40
Fuel volume fraction	0.39	0.27	0.45

Fig. 3. Core configuration of FUJI-U3 (vertical section)

As described in the discussion on design conditions, the fast neutron fluence is limited in the graphite of the core; hence, it is important to reduce the fast neutron flux at the center of the core. In this study, the core is designed to have 3 regions. The following is the explanation of this 3-region concept.

First of all, let us consider the 2-region core. In order to reduce the peak neutron flux, it is necessary to reduce K-infinity (Kinf) in the center region of the core, and to increase the Kinf at the outer region. But, increasing Kinf at the outer region causes more neutron leakage, and then, the conversion ratio (CR) becomes worse. Since our targets are both to flatten the neutron flux and to achieve high CR, the above 2-region core cannot satisfy these conflicting targets.

Therefore, the third region is introduced at the peripheral of the core, where Kinf is low as the center of the core, in order to reduce the neutron leakage from the peripheral of the core.

In the above concept, changing Kinf is achieved by changing the fuel volume fraction in the graphite. Fig. 4 shows the relation between Kinf and the Graphite/^{233}U (G/U) atom density ratio. In this fuel cell calculation, which is explained in section 3 and Fig. 1, the concentration of $^{233}UF_4$ is fixed at 0.24 mol%, which corresponds to that in the beginning of the cycle condition. The hatched region corresponds to the range for the core design of FUJI-U3, which is designed at under-moderated region (left side of the curve). Since the center region of the core in FUJI-U3 has a higher fuel fraction and a lower graphite fraction, which means it has a lower G/U ratio, Kinf in this region becomes lower than that in the outer region. This effect causes the flattening of the neutron flux.

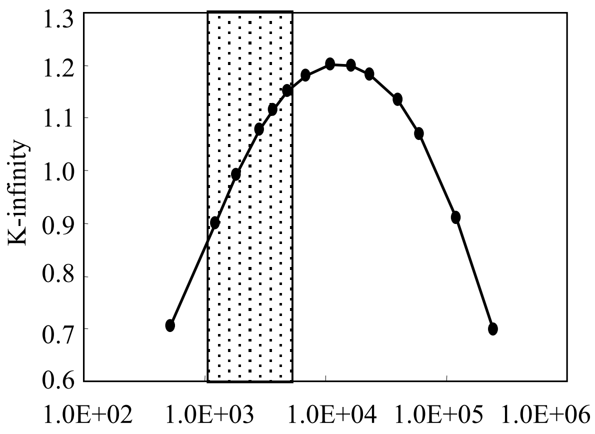


Fig. 4. K-infinity vs. Graphite/^{233}U atom density ratio ($^{233}UF_4$ = 0.24 mol%).

Based on the above concept, 2 million cases were investigated using the nuclear analysis code SRAC95, and the final case was proposed as the optimized three-region core, as is shown in Fig. 3 and Table 1.

The region located closest to the center is Core 1, and it has a fuel fraction of 39 vol%. Core 2 and Core 3 have fuel fractions of 27 vol% and 45 vol%, respectively. The core diameter is 4.72 m, and its height is 4.66 m, then the power density in the core region is 5.5 MW/m^3; on the other hand, the MSBR was designed with a power density of 22 MW/m^3 [1]. There is a fuel path between Core 3 and the graphite reflector; further, there are fuel ducts at the bottom and top of the core. The width of fuel path or duct is 0.04 m. The thickness of radial and axial graphite reflector is both 0.30 m. The entire core is enclosed in a reactor vessel having a thickness of 0.05m with an inner diameter of 5.40 m and a height of 5.34 m.

Table 1 shows the main characteristics of the core of FUJI-U3 for a capacity of 200 MWe. In the criticality calculation, the effective multiplication factor (Keff) is maintained close to unity, actually about 1.01.

Since SRAC95 is a static nuclear code, delayed neutron is not considered. Delayed neutron fraction of FUJI-U3 is small as 0.21%, because of a smaller value of ^{233}U than ^{235}U. Since 20% of fuel inventory exists out of core, 20% of delayed neutron is lost out of core. Therefore, calculated Keff by SRAC95 should subtract 0.04% delta-K. But, this value is very small, and it does not affect design results.

The mean conversion ratio (CR) is 1.01, and the temperature coefficient of reactivity is -2.7×10^{-5} 1/K in average.

Table 1 Design parameters of FUJI-U3.

Electric output Thermal output Thermal efficiency	200 Mwe 450 MW(th) 44.4%
Reactor vessel Diameter /Height (inner) Thickness	 5.40 m/5.34 m 0.05 m
Core Diameter /Height Fuel volume fraction (av.)	 4.72 m/4.66 m 36 vol.% (See details in Fig.3)
Fuel path/Duct Width Fuel volume fraction	 0.04 m 90 vol%
Reflector Thickness Fuel volume fraction	 0.30 m 0.5 vol%
Power density within core	5.5 MW/m^3
Multiplication factor	ca. 1.01
Conversion ratio (av.)	1.01
Temperature coefficient (av)	-2.7×10^{-5} 1/K
Maximum neutron flux Graphite (>52 keV) Vessel (>0.8 MeV) (<1.0 eV)	 4.1×10^{13} 1/(cm^2·s) 1.4×10^{11} 1/(cm^2·s) 2.5×10^{12} 1/(cm^2·s)
Fuel salt Composition mol% Volume in reactor Volume in primary loop Flow rate Temperature; in/out	 $LiF-BeF_2-ThF_4-UF_4$ 71.76-16.0-12.0-0.24* 33.6 m^3 38.8 m^3 0.711 m^3/s 833 K/973 K
Inventory in primary loop ^{233}U Th Graphite	 1.133 t* 56.4 t* 163.1 t

(*: Initial condition)

Fig. 5 and Fig. 6 show the FUJI-U3 radial and axial distribution of the fast neutron flux with energy higher than 52 keV at the center of the core, compared with the previous FUJI-II and FUJI-12 design. These are the flux distributions at the beginning of operation.

In Fig.5 and Fig.7, the X-axis is normalized to 1.0 for each vessel outer radius (R_V) in the different designs. Also, in Fig. 6 and Fig. 8, the X-axis is normalized to 1.0 for each vessel half-height (H_V).

Since the three-region core design is applied to FUJI-U3, the maximum value of the fast neutron flux is fairly reduced compared with FUJI-II or FUJI-12. Since FUJI-U3 distribution is less than the irradiation limit, which is specified to be 4.2×10^{13} $1/(cm^2 \cdot s)$ in the design condition section, FUJI-U3 can be operated for 30 years without replacing the graphite moderator.

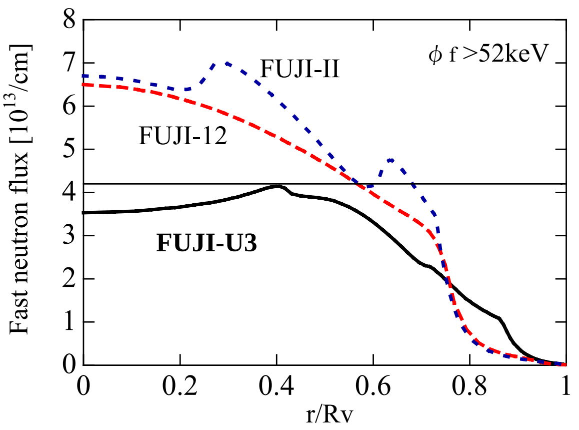


Fig. 5. Radial distribution of fast neutron flux in each FUJI designs at center of the core.

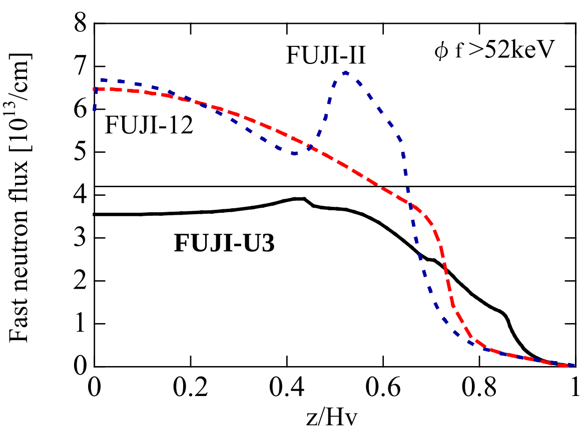


Fig. 6. Axial distribution of fast neutron flux in each FUJI designs at center of the core.

Fig. 7 and Fig. 8 show the radial and axial distribution of the thermal neutron flux with energy lower than 1.0 eV. It is observed that the thermal neutron flux is also flattened due to the three-region core design of FUJI-U3. Compared with FUJI-II, FUJI-U3 has a lower graphite fraction at the center of the core, and then, thermal neutron flux becomes lower there and flattened. This brings flatter heat generation throughout the core.

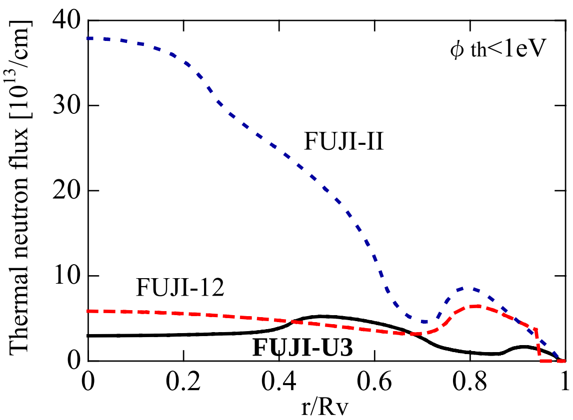


Fig. 7. Radial distribution of thermal neutron flux in each FUJI designs at center of the core.

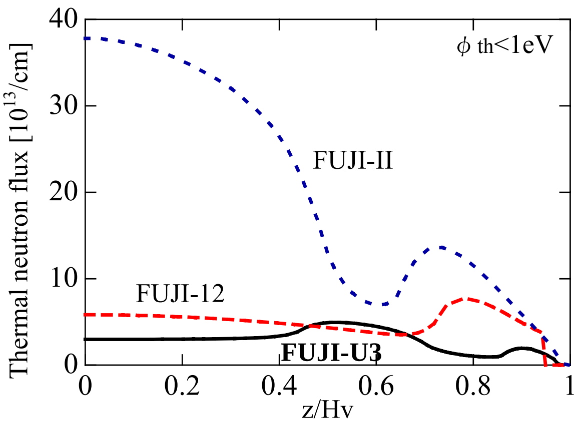


Fig. 8. Axial distribution of thermal neutron flux in each FUJI designs at center of the core.

5. COMPUTATIONAL PROCEDURE FOR BURNUP CHARACTERISTICS

The burnup characteristics are calculated using the nuclear analysis code SRAC95 [11] and the burnup/decay analysis code ORIGEN2 [16]. The cross-section library for SRAC95 is JENDL3.2 [14]. Since ORIGEN2 requires one-group cross sections, it is necessary to reduce the number of energy groups for the cross sections of SRAC95. Using 107-group cross sections and neutron flux of SRAC95 for each region of FUJI-U3, a one-group cross section is obtained for the 211 nuclides (70 actinides and 141 fission products); and these 211 nuclides are selected because they have larger total cross sections and more importance among the 1297 nuclides in the core. For the other 1086 nuclides having smaller total cross sections, the ORIGEN2 cross-section library for PWR is used. A total of 1297 nuclides are considered in this burnup calculation.

In the burnup calculation, the continuous removal of FP gas from the fuel salt is assumed to occur through the gas removal system, as proposed for the MSBR [1]. These gaseous FPs are Kr, Xe, H, T, Ne, Ar, and Rn. The removal speed in the system is estimated to be 1.4% per minute of gaseous FPs in the fuel salt. But, in the actual calculation, it is assumed that 99.9% of gaseous FP is removed and the residual 0.1% remains in the core at the Keff calculation point of every 40 days interval. Since Xe reactivity loss is about 1.6% delta-K which is dominant among gaseous FPs, the above 0.1% residual effect is negligibly small.

Keff is calculated every 40 days, and it is maintained at approximately 1.01 by feeding the reactor with new fuel salt. This makeup fuel for replenishment is composed of the mixture of LiF (73 mol%) and $^{233}UF_4$ (27 mol%), according to a proposal of ORNL [1]. LiF is added to the mixture to achieve a low melting point. In this makeup fuel, the recovered U is used at first, and this U is obtained from the chemical processing of the fuel salt, which is described later. If the amount of this recovered U is not enough, then new ^{233}U is used as an additional makeup fuel.

The fuel salt composition in the core changes with the burnup. The composition is examined every 40–200 days, and Li, Be, Th, and F are then added, if necessary. As a result, the chemical composition of the fuel salt is maintained close to that of the initial core. At this makeup timing when Li/Be/Th/F are added, the one-group cross sections are also updated to those of the chemical composition of the new fuel salt. In the makeup fuel, Li is supplied in the form of LiF; Be in the form of BeF_2 or metallic Be; and F as F_2 in the gaseous form. Furthermore, Th is supplied in a mixture of LiF (71 mol%) and ThF_4 (29 mol%), according to a proposal of ORNL [1].

6. CHEMICAL PROCESSING OF FUEL SALT

The fission products are accumulated in the fuel salt during the operation of the FUJI, and their precipitation or extraction occurs when the concentration of the FPs is too high after a long-term operation. Therefore, it is necessary to process the fuel salt chemically before such a situation occurs. In this study, it is assumed that the chemical processing of the fuel salt is performed every 2,000 EFPD (Effective Full Power Days), which corresponds to a 7.5 years interval with a 75% load factor. The details of the chemical processing are described in the MSBR report [1].

In our study, it is assumed that 100% of the actinides and FPs are removed from the discharged fuel salt. The reactivity loss due to all FPs is about 5% delta-K, and if a 95% removal is assumed, it will result in a small penalty of 0.25% delta-K in the above calculation.

In the chemical processing, LiF, BeF_2, ThF_4, and UF_4 are recovered and purified from the discharged salt and are reused as fresh fuel salt. Moreover, the remainder is used as a makeup fuel. Since an element such as Th decreases with burnup, it is added. On the other hand, a small amount of U is removed in order to decrease Keff, because the removal of FPs increases Keff just after the chemical processing. Since ^{233}Pa decays to ^{233}U with a half-life of 27 days, all the PaF_4 is recovered after the chemical processing and reused as a potential fissile material of fuel salts.

7. BURNUP BEHAVIOR OF REACTOR CHARACTERISTICS

Fig. 9 shows the time behavior of Keff and CR. The X-axis shows EFPD, and 8213 EFPD corresponds to 30 years of operation with a load factor of 75%.

FUJI-U3 is designed to maintain criticality mainly by the makeup of $^{233}UF_4$ fuel salt at every 40 days (30 EFPD) for the 30 years operation.

Since the decrease in reactivity within 40 days is about 0.2% delta-K, this small change can be regulated either by the movement of the graphite rod or by an adjustment of the core flow. The above graphite rod was proposed in the MSBR design, and when the graphite rod is inserted into the flow hole (Fig.1), it causes a small positive reactivity. Also, when the core flow is increased, it causes a small positive reactivity, and this method is applicable to FUJI as shown by the authors [17]. This technology is the same as the power control system in current BWRs. Therefore, a small reactivity decrease due to burnup can be compensated either by the insertion of a graphite rod or by the increase of the core flow.

Recently, the authors proposed another control method, which is to change the temperature of fuel salt by changing the turbine/generator output. Its principle is the same as the power control system in current PWRs [18].

The mean value of the CR throughout the lifetime for the abovementioned case is 1.01. Meanwhile the CR of old FUJI-12 value was 0.92, showing the effect of FUJI-U3's 3-region concept.

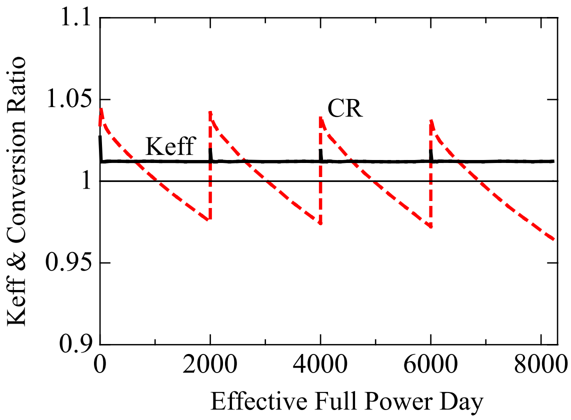


Fig. 9. Time behavior of Keff and Conversion Ratio.

Table 2 shows the change in the reactor characteristic parameters when FUJI-U3 is operated for 30 years. In this table, it is assumed that chemical processing is performed between the date of 2000 and 2001, and so on. Keff is kept about 1.01, except at the starting point. The maximum neutron flux (ϕ_G) of energy higher than 52 keV in the graphite moderator is 4.1 × 10^{13} 1/(cm^2·s), which is restricted by neutron radiation damage. The maximum neutron flux ϕ_V on the inner wall of the vessel is 1.4 × 10^{11} 1/(cm^2·s) for the fast neutron flux with energy higher than 0.8 MeV, and 2.5 × 10^{12} 1/(cm^2·s) for the thermal neutron flux with energy lower than 1.0 eV. Both values are below the irradiation limit of this study, and the FUJI-U3 design satisfies the design condition for the neutron irradiation of 30 years operation. As a future study, borated graphite can be used to reduce the neutron flux at the vessel.

The temperature coefficient of reactivity is –2.7 × 10^{-5} 1/K in average, and this temperature coefficient is sufficient for stable control. The reactor dynamics for FUJI with one-region core design is described in a separate report [19].

Table 2. Time behavior of FUJI-U3 characteristics.

Operation period (EFPD)	Keff	CR	α_T [1/K] ($\times10^{-5}$)	ϕ_G [1/cm^{2}s] >52keV ($\times10^{13}$)	ϕ_V [1/cm^{2}s]	
					>0.8MeV ($\times10^{11}$)	<1.0eV ($\times10^{12}$)
0	1.027	1.034	-3.10	4.10	1.34	2.46
2000	1.012	0.975	-2.39	4.12	1.40	2.33
2001	1.019	1.042	-2.97	4.10	1.36	2.42
4000	1.012	0.974	-2.43	4.10	1.41	2.30
4001	1.019	1.040	-3.00	4.09	1.36	2.38
6000	1.012	0.972	-2.47	4.09	1.41	2.27
6001	1.018	1.038	-3.03	4.08	1.37	2.36
8213	1.012	0.965	-2.50	4.07	1.42	2.24

8. MATERIAL BALANCE OF ACTINIDES

Table 3 shows the material balance of actinides such as U/Th/Pu and the MA in FUJI-U3. This table shows the calculated results for the initial inventory, reload fuel, and total feed for 30 years of operation. The final remaining amount of actinides and their net production are the values after 30 years of operation. In Table 3, “Initial inventory” is the weight of Th and U-fissile at the beginning of the FUJI-U3 operation, and “Net feed” is the weight of charged Th and U-fissile after startup and up to the closing of the reactor, which is 30 years later. “Total demand” is the sum of “Initial inventory” and “Net feed”. “Final remaining amount” is the weight of the actinides at the closing of the reactor. Finally, “Net production” is the difference between “Final remaining amount” and “Total demand.” The negative value for the net production implies consumption. In the column of “Final remaining amount” of this table, ^{233}Pa is added to U-fissile, because ^{233}Pa decays to ^{233}U with a half-life of 27 days.

As shown in Table 3, the net production of Pu is 0.70 kg and that of MA (mostly Np) is 4.55 kg. Therefore, it is confirmed that there is very little production of Pu and MA, and it shows the benefits of the Th cycle.

Table 3 Material balance of FUJI-U3 for a 30 years operation.

	Th (t)	U-fissile (t)	Pu (kg)	MA (kg)
Initial inventory	56.4	1.133	--	--
Net feed	5.2	0.426	--	--
Total demand	61.6	1.559	--	--
Final remaining amount	57.5	1.584	0.70	4.55(*)
Net production	-4.1	0.025	0.70	4.55(*)

(*) Np=3.16kg, Pa(except ^{233}Pa) =1.39kg, Am=0.2g, Cm=0.02g

9. FISSION PRODUCT

Table 4 shows the amount of FP, which is accumulated in FUJI-U3 at the point of 30 years. These FPs are classified into gaseous FP, liquid FP, and solid FP.

Since the gas solubility of the fuel salt is sufficiently low, the FP gas is easily and continuously removed from the fuel salt by the He gas bubbling system. As is shown in Table 4, the residual gaseous FPs is only 178g.

As for the concentration of the 3rd valence FP, it must be less than 1 mol%, considering the solubility of the FP in the fuel salt, as is evaluated in the MSBR report [1].

As shown in Table 4, the maximum concentration of the 3rd valence FP is 0.11 mol%. Therefore, precipitation does not occur for the 3rd valence liquid FP.

The concentration of other metals, except 3rd valence FP, is 0.16 mol%. Since this concentration is less than 1 mol%, it is considered that there is no chemical influence on the graphite and the vessel, as is evaluated in the MSBR report [1].

As for the solid FP, the total amount is 158 kg, and they are Mo/Ru/Tc and so on. Most of them are removed by the filter system, which is equipped in the primary loop; subsequently, there will be little precipitation of the solid FP on the wall of the primary loop. However, this should be confirmed in the future.

Based on the above consideration, the operation of FUJI-U3 will not face serious problems due to the FP.

Table 4. Fission products in the fuel salt of the core at the point of 30 years.

Gaseous fission products	g	178
Liquid fission products 3rd valence metal Other metal, except 3rd valence	 mol% kg mol% kg	 0.11 323 0.16 364
Solid (precipitated) fission products	kg	158

10. FUEL REQUIREMENT AND ACTINIDES FOR 1-GWe PLANT

In order to compare the MSR performance with a 1-GWe BWR [20], the abovementioned FUJI-U3 results are multiplied by 5 (1000 MWe/200 MWe = 5). The fuel requirement and the amount of Pu/MA are shown in Table 5. In this comparison, it is assumed that the BWR uses uranium oxide as the fuel and has an operation period of 25.9 years with an 87% load factor. This means that the operation period is 22.5 EFPY (Effective Full Power Years) for both reactors.

The initial fissile (^{233}U) amount for FUJI-U3 is extrapolated to be 5.7 t for a capacity of 1-Gwe, and that (^{235}U) for the BWR is 3.9 t.

The amount of fissile material fed as a reload fuel is 2.1 t of ^{233}U. This value is 10% of the BWR amount of 20.7 t of ^{235}U.

The total requirement of fissile material for the entire operation period is 7.8 t of ^{233}U. This value is 32% of the BWR amount of 24.6 t of ^{235}U. One of the big benefit of FUJI-U3 is that since CR is almost 1.0, the same amount of fissile is discharged at the end of FUJI operation, and this fissile (7.9 t) can be used to the next FUJI.

FUJI-U3 produces 3.5 kg of Pu for a 30 years operation period, whereas BWR produces 5080 kg for the same period. This implies that Pu production in FUJI-U3 is only 0.1% of that in BWR.

The production of minor actinides (MA), which are long-lived nuclear wastes, is 23 kg, whereas BWR produces 543 kg for the same period. This implies that MA production in FUJI-U3 is 4% of that in BWR.

Table 5. Production of U/Pu/MA in a 1-GWe reactor.

	FUJI-U3	BWR
Output power (Gwe)	1.0	1.0
Reactor operation time (year)	30.0	25.9
Load factor	0.75	0.87
Initial inventory of U-fissile (t)	5.7	3.9
Net feed of U-fissile (t)	2.1	20.7
U-fissile total demand (t)	7.8	24.6
Final remaining U-fissile (t)	7.9	--
Net production in reactor life U-fissile (t) Pu-total (kg) Minor actinides (kg)	 0.1 3.5 23	 -17.7 5080 543

11. CONCLUSION

In this paper, we propose a small Molten-Salt Reactor FUJI-U3 with a three-region core design. By this design, the flattening of the fast neutron flux is achieved. The core performance of the reactor is obtained by calculations performed using the reactor analysis code SRAC95 and the burnup analysis code ORIGEN2. The main results obtained are as follows.

(1) It is possible to operate the reactor with an electric power of 200 MWe (thermal power output of 450 MW(th), 44.4% thermal efficiency) together with a load factor of 75% for 30 years without replacing the graphite moderator.
(2) With regard to the average conversion ratio, a high value of 1.01 is achieved, and it is almost self-sustaining. Also, the residual fissile at the end of the reactor life can be used to the next FUJI.

For comparison with a 1-GWe BWR, the results of FUJI-U3 for a 30 years operation period are scaled to those of a 1-GWe capacity.

(3) The ^{233}U requirement for FUJI-U3 is 7.8 t, and this value is 32% of the BWR. At the end of FUJI-U3 operation, 7.9 t of fissile is discharged, and this fissile can be used to the next FUJI.
(4) FUJI-U3 produces 3.5 kg of Pu for a 30 years operation period, whereas BWR produces 5080 kg for the same period. This implies that Pu production in FUJI-U3 is only 0.1% of that in BWR.
(5) The production of minor actinides (MA), which are long-lived nuclear wastes, is 23 kg, whereas BWR produces 543 kg for the same period. This implies that MA production in FUJI-U3 is 4% of that in BWR.

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