Preliminary Study of Long-life GFR 100 and 150 MWth

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Abstract. The breeding capability of fast reactor allow the reactor to operate in a long life time, but with the risk of a large reactivity swing and high power peaking in the center of the core. Both of these factors can be reduced by adjusting the fuel with a right configuration. This is down in this study, where optimization is performed on 100 and 150 MWth reactors, with the aim to getting the reactor to operate for more than 20 years with Power Peaking factor (PPF) is less than 1.3 and reactivity swing less than 1%. Based on the optimization result, a maximum configuration of fuel in inner, mid, and outer core of reactor I and II is given by 6.6%, 11.6%, 12.4% and 6.6%, 9.8%, 14.0%. Neutronic analysis results of this design indicate that the reactor II can only operate for 29 years, while reactor I can operate for more than 30 years because it has a larger conversion ratio (CR). The PPF value of reactor II is greater than reactor I caused by fission rate in the center of the core of this reactor is larger as a result of the short buildup process. Reactivity swing of reactor I and II are 0.25% and 0.16%, where the maximum reactivity of reactor I is 1.02% and occur in BOL, and for reactor II, the maximum reactivity is 1.03% and occurred in the 18th year.

1. Introduction

Seventy percent of commercial nuclear reactors are used until this time is generation two reactor like Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR). This reactor had several drawbacks, such as low uranium utilization and low efficiency of reactors. Efficiency of reactors is less than 35%, caused by the maximum temperature output is less than 330 °C [1]. Ninety percent of fuel waste of reactors still have the same form with original fuel, composed of 35% fission products, 1% waste of 235U and 1% plutonium as transmutation yield [2].

This weakness becomes a problem in the nuclear industry, thus needed new nuclear technology is more promising. One of the more modern nuclear technology were able to overcome this problem is Gas Cooled Fast Breeder Reactor (GFR) is a fourth generation reactor. This reactor takes the advantage of breeding process to produce plutonium from 238U that the reactor can operate within a
longer time. This means uranium utilization and radioactive isotopes destroyed of this reactor is more than efficient [3]. This reactor also uses helium gas as a coolant, so possible to produce high-temperature gas output until 850°C and net thermal efficiency is 42% [4]. As the pay, neutronic analysis of this reactor become more complicated due to high conversion ratio (high breeding) of the core.

Breeding process in the cores depends on the amount of fertile material and hard neutron spectrum. To keep the fast neutrons and the fissile materials in sufficient quantity, fuel in the active core must be arranged in a particular configuration where fuel with the lowest percentage of plutonium is placed at the center of core whereas a material with the highest percentage of plutonium is located at outer of the core. Fuel with a low percentage of plutonium serves as a provider of the fertile material needed for breeding in a long time. Fuel in outer core is used as driver fuel.

When the reactor operates, most fission reactions occur in the outer region that has the higher amount of fissile material. Neutrons produced by this fission reaction are then used to subsequent fission reaction and also for breeding. The highest probability of breeding occurs in the inner core due to the greater number of neutron and fissile material. High breeding in the inner core has an impact on increasing of power distribution in this region as a result of an increase of the amount of fissile material. The increase of power distribution in a long time at this region can result in the increasing value of Power Peaking Factor (PPF) of the core until 1.5, this is dangerous for the reactor. In addition to PPF, the other neutronic problems were appeared as a consequence of the high breeding velocity of GFR, especially for small power GFR is the values of maximum reactivity and reactivity swing more than 1%.

To keep the reactor operable for a long time with neutronic parameters such as PPF, maximum reactivity, and reactivity swing remains within safe limits, the fuel in the core must be arranged in the optimum configuration.

In this paper, optimization is done to get small power GFR which capable of operating more than 20 years with a low PPF and small values of reactivity swing and maximum reactivity. As a comparison, do the neutronic analysis of the reactor with a power 100 and 150 MWth. These reactors are suitable to use in the remote area like Indonesia islands which has a relatively low electricity requirements. In addition, due to the high outlet temperature, these reactors can be used to produce hydrogen gas and useful in chemical industry.

2. Design

2.1 Assembly Design.

The reactor core is composed of some several types of assemblies such as fuel, control, shutdown, reflector, and shielding assembly.

Fuel assemblies were designed base on allegro 75 MWth assembly design [5]. This assembly is filled by 91 cylinder fuel pin with arranged in six line hexagonal concentric and surrounded by hexagonal duct assembly with thickness 3.0 mm and distance of duct flat-to-flat are 10.4 cm. Every fuel pin is filled by fuel pellet with diameter 8.56 mm, and wrapped cladding with thickness 0.91 mm. 50% of this assembly is fill up by fuel, 23.6% is cladding and the remaining part is coolant.

The material of fuel pellet is (U,Pu)C, where natural uranium mixed with plutonium waste from PWR with a certain percentage plutonium. Plutonium used as driver fuel and 239Pu and 241Pu which result of breeding from 238U and 240Pu would be used as next fissile fuel. The cladding and assembly
duct are formed of the same material is SiC. The design of fuel pin and fuel assembly is given by figure 1.

![Fuel assembly design](image1)

**Figure 1.** a) Fuel assembly design. b) Fuel pins design

The reflector and shielding assembly have the same form but different compound of material. Material of reflector is Zirconium Carbide (ZrC) while shielding is Boron Carbide. Any assemblies are filled by 19 pin rods with radius size is 2.0 cm. The design of shielding/reflector assembly and shielding/reflector pins is given by figure 2.

![Reflector/Shielding assembly design](image2)

**Figure 2.** a) Reflector/Shielding assembly design. b) Reflector/Shielding pins design
2.2 Core Design

The reactor core is designed in three-dimensional hexagonal, where the core consists of three main parts: active core, reflector, and shielding. Active core is located in the center of the core and covered by reflector and shielding in the outer of the core.

Active core is filled by 127 assemblies where it is arranged in 7 lines hexagonal concentric. Active core is divided into III regions based on the percentage of plutonium of fuel assembly. Line 1-3 of active core is region I (inner core) consisting of one inner shut down assembly (S1) placed in the center of region, six inner control assemblies (C1) placed in vertex in line 3, and twelve inner fuel assemblies (F1) filled in another place in line 2 and 3. Region II (mid core) is filled by six outer shut down assemblies (S2), and thirty-six mid fuel assembly (F2). Outer shut down assemblies were placed in the middle of every side of line 5, and another area of line 4 and line 5 is filled by mid fuel assembly. In region III (outer core) outer fuel (F3) was placed in line six and seven except in every vertex in line seven was filled by outer control assemblies (C2). Reflector (R) of core in the radial direction is filled by one hundred and forty-four assemblies which arranged in three line hexagonal concentric. In the axial direction, there is a reflector at the top and bottom of active core with the same thickness of 30 cm. The shielding (SH) was placed as wrappers of the reflector. In axial direction it has a thickness 50 cm, and in radial direction is arranged in four line assemblies. Number of shielding assembly is two hundred and seventy-six assemblies. Core views in radial and axial directions are given by figure 3.

![Core Design](image)

Figure 3. Core design. a) Radial view. b) Axial view

This reactor was designed to operate more than 20 years with reactivity and swing reactivity is less than 1.0%, and power peaking factor is less than 1.3 and applies to two different power: 100 MWth and 150 MWth. Some design limitation of the reactor must be operated with average
power density 50 - 100 MW/m$^3$ [4], with inlet coolant temperature is less than 500$^\circ$C and outlet 850$^\circ$C or more in pressure 7 MPa. General specification of reactor is described in Table.1.

### Table 1. Some result of neutronic analysis

| Parameter                        | Specification     |
|----------------------------------|-------------------|
| Thermal power                    | 100 – 150 MWth    |
| Refueling period                 | 20 – 30 years     |
| Core geometry                    | Hexagonal 3D      |
| Coolant                          | Helium gas        |
| Fuel type                         | (U,Pu)C          |
| Number of fuel assembly           | 96                |
| Duct flat-to-flat of fuel assembly| 104 mm           |
| Duct thickness of fuel assembly   | 3 mm              |
| Fuel pin pitch                   | 10.8 mm           |
| Fuel pin OD                      | 8.56 mm           |
| Cladding thickness               | 1.84 mm           |
| Number of control assembly       | 24                |
| Number of shut down assembly     | 7                 |

3. Analysis Method

In this paper, neutronic analysis of reactor design calculation was analyzed by used SRAC (Standard Reactor Analysis Code). JENDL-4.0 from JAEA (Japan Atomic Energy Agency) is used as a library of this code. The primal data of this library is diffusion coefficient, and the cross section data of 406 nuclides with incident neutron energy range is from $10^{-5}$ eV ~ 20 MeV in the discretization to 107 groups energy (fine groups)[6].

In SRAC, the calculation is divided into two steps those are the lattice calculation and core calculation [7]. The purpose of lattice calculation is to prepare a few-group homogenized macroscopic cross section of two-dimensional (2D) infinite arrangement of assemblies. This cross section would be used to solve diffusion equation in core calculation. The dependence of energy has been calculated in here so that calculation in the core will be easier because only focus on the region such as power distribution in the core.

In the lattice calculation, the calculation started from the preparation of effective microscopic cross section of any group energy in every segment area of the lattice from the self-shielding factor and infinite dilution cross section where the self-shielding factor are calculated from background cross section. All of the parameters are calculated based on reactor design like temperature or other characteristic of the material and calculated by used Narrow Resonance (NR) approximation. Base on the value of atomic density number and effective microscopic cross section can be calculated the value of the effective macroscopic cross section. This value will be used to solve neutron transport equation to find neutron flux in any group energy and any segment of the lattice. This equation solved iteratively use Collision Probability Method (CPM) with convergence criterion of neutron flux is 0.0001. The iteration known as inner iteration.

To reduce the calculation cost in times and memory due to a huge amount of neutron flux and microscopic cross section, do homogenization to minimize the space-dependent information and group-collapsing to reduce the energy-dependent information. Homogenization of neutron flux is performed at the first step, and then this result can be used in homogenization calculation of
microscopic cross section and macroscopic cross section. In the same way, group-collapsing calculation of result of neutron flux, microscopic cross section, and macroscopic cross section homogenization was started from neutron flux, where 107 groups energy would grouping to 10 group energy. This result will be used to calculate the few-group homogenization microscopic cross section and macroscopic cross section.

The next step in lattice calculation is burnup calculation. In this step, the cycle calculation of few-group homogenized cross section will be repeated for the different atomic number density of fuel material in every time burnup. The atomic number density in every time burnup was calculated used burnup equation.

To solve the multi-group diffusion equation in the core calculation is required macroscopic cross section that has been obtained in lattice calculation and also needed the diffusion coefficient was contained in the JENDL-40 and transmutation few-group macroscopic cross section. This diffusion equation solved iteratively to get the value of neutron flux and effective multiplication factor that convergence. This iteration process is known as outer iteration. In this paper, the convergence criterion of neutron flux and effective multiplication factor are 0.00001 and 0.001. After the neutron flux of all segment in reactor core obtained and also the value of multiplication factor, the calculation continued to get both value in another burnup steps along 30 years. Furthermore, based on the result of multiplication factor effective at BOL and EOL can be calculated the reactivity swing of the reactor and then power in every segment in the reactor core can be calculated based on neutron flux in each segment as a function of fission reaction rate. Diagram block of neutronic calculation with SRAC is given by figure 4.

The power density in every mesh of core can be averaged in axial direction so that the radial power density distribution in the core is easier to be analyzed. Based on this value, the power peaking factor can be calculated as a ratio of peak power to average power. The CR value of reactor is also can calculated from CR value and power in regions.
4. Result
The optimization result of reactor 100 MWth and 150 MWth provide, the maximum fuel configuration of reactor 100 MWth is 6.6%, 11.6% and 12.4% percentage of plutonium in the inner core, mid core, and outer core. Maximum fuel configuration of reactor 150 MWth in the inner, mid, and outer region is 6.6%, 9.8% and 14.0%.

Some result of neutronic calculation of both reactors is given by table 2.

| Reactor parameter                     | Reactor I | Reactor II |
|---------------------------------------|-----------|------------|
| Power (MWt)                           | 100       | 150        |
| Refueling time (years)                | 30        | 28         |
| Average average power density (MW/m³) | 62.61     | 93.92      |
| Reactivity swing (%)                  | 0.25      | 0.16       |
| Power peaking factor at BOL, MOL, EOL | 1.09, 1.19, 1.26 | 1.16, 1.18, 1.30 |

Besides this result are also given details of the result such as atomic density change every burnup, neutron flux, distribution of average power density, conversion ratio and a multiplication factor ($k_{eff}$) of any reactor.

The atomic density change and neutron flux in BOL, MOL, and EOL, and CR of reactor I is given by figure 5, 6 and 7.
Figure 5. Atomic density change of reactor I. a) Inner core. b) Mid core. c) Outer core

Figure 6. Neutron flux of reactor I. a) Inner core. b) Mid core. c) Outer core
As shown in figure 5, the atomic density number in BOL is proportional to the percentage of plutonium who used in each region. Based on this figure, it is known that the atomic density number of uranium is highest in the inner core, and the atomic density number of plutonium is highest in the outer core where the atomic density change of every nuclide is dependent on breeding, and fission reaction rate in the core, and vice versa, the core fission and breeding reaction is dependent on atomic density number of every nuclide and neutron flux each time burnup.

Base on figure 5.a and 6.a, due to the high number of atomic density of $^{238}$U and the high value of neutron flux especially from epithermal to the fast region, allow to happen high breeding at the start of the operation. The high breeding resulted in decreased atomic density number of $^{238}$U at BOL to MOL is also impact on the increase of $^{239}$Pu. From BOL to MOL the number density of $^{239}$Pu not only increase due breeding of $^{238}$U but also decrease due to fission reaction and neutron capture to become $^{240}$Pu, but because the breeding rate is more dominant so that the amount of $^{239}$Pu is significant to grow. The significant grow of $^{239}$Pu is a direct impact on increasing the number density of $^{240}$Pu due to capture and it is more dominant than the breeding of $^{240}$Pu to $^{241}$Pu. Different with $^{239}$Pu and $^{241}$Pu, atomic density number of $^{241}$Pu is decrease due to fission rate is higher than breeding rate. The same thing also happens on $^{235}$U who not have breeding.

From MOL to EOL, neutron flux and atomic number density do not change drastically, the significant changes occur only in a view years after MOL and after that, it became saturation condition. In this state, the atomic density number of $^{239}$Pu has slightly increased due to an increase of fission reaction of $^{239}$Pu as a consequence of the declining amount of fission reaction of $^{235}$U and $^{241}$Pu.

The change of atomic density of this region also seen in CR (figure 7), where have high value in the BOL but drastically change towards MOL and in EOL the value to 1.116. The change of CR is accordance with the rate of reduction atomic density of $^{238}$U or the rate of increase atomic density $^{239}$Pu which is the dominant nuclides of inner core.

Different with the inner core, in the BOL the atomic density of uranium in the mid-core is slightly little but the atomic density of plutonium is slightly larger (figure 5.b). The neutron flux of this region (figure 6.b) was much smaller than the previous region resulted in CR of this region is becomes smaller as shown in figure 7. This was resulted in the rate of atomic density change to be slower but still with the same principle as in the inner core, where the atomic density has significant change in BOL to MOL and change be slower from MOL to EOL. It is also seen in neutron flux,
where the neutron flux from BOL to MOL have the little decrease and from MOL to EOL the reduction to be very slightly. The significant change of CR only happen in BOL and view years after MOL and after that CR be more stable until in EOL value to 1.049. Another thing that can be seen from this region is the nuclides number of $^{235}$U is so small that the main source of fission is become from $^{239}$Pu and $^{241}$Pu, where the burner of $^{241}$Pu in the operation started is dominant but drastically decrease after 11 years operating.

Different from the previous two regions, atomic density of uranium is far fewer lead to the breeding capability of the outer core to be much lower. On the other contrary, the fission reaction rate of this region in BOL is higher. The lower breeding capability of this region seen in slowly rate change of atomic density and neutron flux value which almost unchanged from BOL to EOL. Overall it appears that each region requires a long time to buildup and only in five years before EOL all regions in equilibrium states. In addition, it is also seen that CR value of every region is more than one until the end of the operation so that allows the reactor to operate more than 30 years. It also looks at the result of the CR calculation of reactor I, where at EOL reactivity is still positive and CR value still greater than 1.05 as shown in figure 8 and 7.

From figure 8, the buildup process takes a long time resulting value of the reactivity swing reaches 0.25%, where maximum reactivity is 1.02% in the beginning of the operation and minimum reactivity occurs in year 7. This reactor needs 25 years to buildup cause high breeding process and low power of reactor.

![Figure 8. Multiplication factor of reactor I.](image)

Another impact of the high breeding of reactor is a high transition of average power density distribution of reactor as shown in figure 9. In figure 9, average power density distribution is given in radial direction as shown at red line in figure 3. The minimum of average power density in the radial direction is at the center of the core and at a distance of 22 and 66 cm from the center which is the shutdown and control assemblies. The power density of this reactor is 62.61 MW/m$^3$. 
Figure 9. Average power density distribution of reactor I.

In this figure, in BOL the greatest power contributed by mid core. This is slightly different when compared with the analysis result of atomic density change in every core regions in the figure 5, but this is acceptable if the effect of neutron leakage at outer core is calculated. From BOL to MOL the transition of average power density are significant in the inner core and outer core. In inner core, the number of fission reactions increased due to the growing of number fissile nuclides as a consequence of the high breeding of this region thus resulting in accumulation of power in the center of the core. In contrast to this region, the power in the outer core is decreasing because the number of fission decreased due to the reduce amount of fissile material as the slow process of breeding. In mid of core, there was a difference between the line 4 and line 5 assembly arrangement, where has a decrease of power at the outermost regions with the highest possibility of neutron leakage and otherwise has additional power on the inner side of mid-core were has a number of more neutrons. The transition of power distribution from MOL to EOL has the same principle like from BOL to MOL but with no significant changed. It can be explained by figures 5, 6 and 7, where the breeding rate in this time range is not as high as in BOL to MOL. Furthermore, can be calculated PPF of this reactor in BOL, MOL, and EOL are 1.09, 1.19, and 1.26. This result shows that even though the reactor can operate more than 30 years, but the PPF in the inner core will continue to increase do to the accumulation of excess power in this region so it is not allowed in the thermal hydraulic analysis.

Atomic density change, neutron flux and the conversion ratio of reactor II have the same principle like in reactor I. These parameters are given by figure 10, 11 and 12.
Figure 10. Atomic density change of reactor II. a) Inner core. b) Mid core. c) Outer core.
The inner core of reactor II has the same fuel percentage with the same region of reactor I so that the atomic density, neutron flux, and CR of both reactors are same, as a shown in figures 10.a, 11.a, and 12.a.
12. When the reactor operates, the reduction rate of atomic density $^{238}$U and the increase rate of atomic density $^{239}$Pu of reactor II is higher than reactor I, also seen the neutron flux and CR value of this reactor are drastically decrease in the period from BOL to MOL. This is due to the fission reaction needs of reactor II is higher than reactor I, so that the number of neutrons generated per reactor period is also higher, but most of this neutrons are used for capture $^{238}$U and for fission reactions $^{235}$U, $^{239}$Pu, and $^{241}$Pu consequently the neutron flux of this region is decreased sharply in each period. The rapidly decreasing of the neutron flux and the atomic density $^{238}$U result in a rapidly decreasing of CR value of the region.

From MOL to EOL, the breeding capability of this region is still decrease like shows at atomic density change of $^{239}$Pu which significant only along 5 years after MOL, and thereafter the addition rate of this nuclide being very slow and finally reaches saturation state. The same thing happened to the neutron flux which not changes from MOL to EOL when comparable to the first core.

The situation of mid core is as same as like in inner core of this reactor, where atomic density and neutron flux is higher than the same region in core I, but due to greater power, the rate of change of neutron flux, atomic density, and also CR value of this region is higher than reactor I. The rate of change of three parameters is highest from BOL to MOL. From BOL to MOL, the shifting value of neutron flux in mid core of reactor II is greater than reactor I. From MOL to EOL, the neutron flux values do not to seem to shift. However, overall neutron flux values of this region from BOL to EOL is higher than in the same region of reactor I.

Atomic density change of $^{238}$U from BOL to MOL is significant, in MOL an amount of Atomic density $^{238}$U is less than in mid core of reactor I. In EOL this number is much less than in reactor I. Instead, the atomic density growth of $^{239}$Pu of this region is very vast, even the saturated of this region is 5 years faster than saturated of the inner core of the same reactor.

The impact of the change of neutron flux and atomic density $^{238}$U to CR value of this region is shown in figure 12, where the most drastic change of CR occurs in interval BOL to MOL. In MOL the value of CR has been lower than the CR of the same region in reactor I in the same time, and in EOL this value becomes much lower.

Much different from two pervious regions, the outer region of reactor II has higher percentage of plutonium than another region so that the breeding rate of this region is very small and only occurred in some years in the beginning of operation. The low breeding rate plus the higher fission rate of this region resulting $^{239}$Pu growth is slow. Only in some years, this region has reached saturation, and even the amount of atomic density number of $^{239}$Pu has decreased at the end of the operation. This corresponds to a neutron flux which is almost no change over time burnup, and the value of CR is less than 1 after MOL.

The combination of these three regions in reactor II resulted in the buildup process is shorter compared to the first reactor where the buildup process in this reactor occurred for 17 years as shown in figure 13. From this figure also seen that the reactivity in BOL is lower than reactor I, where minimum reactivity occurs in the third year and maximum reactivity occurs in the 18th year at 1.03%. This reactor only able to operate for twenty-nine years with the reactivity swing of this reactor is 0.16%. The negative reactivity value at year 30 in figure 13 is proportional to the CR value at the same time as shown in figure 12.
The high of the breeding rate in the beginning of the operation and high fission reaction along the process direct impact on the power distribution of reactor. The power density of this reactor is 93.92 MW/m³, it is bigger than the previous reactor. The power distribution in BOL, MOL, and EOL is given by figure 14.

As shown in figure 14, the peak power of this reactor at BOL occurred in the outer core. This is caused by the percentage of plutonium in inner and mid-core is very low if compared with at outer core, so that although the neutron leakage is maximum in the outer core region, this region still become the main fission reaction source when the reactor in operating. Furthermore, the higher breeding rate in the beginning of the operation, especially in the inner and mid-core resulted in an accumulation of fissile material like $^{239}$Pu and $^{241}$Pu in these regions. Otherwise, in the outer core, the amount of fissile material continues to decrease due high fission process and low breeding rate in this region. Consequently, when MOL power in the inner and mid-core rose sharply, while the outer core is significantly reduced power. In this time, the peak power occurred in the second line of the inner region.
From MOL to EOL, the impact of low breeding rate in each region of reactor II looks at the power distribution in the EOL which had a slight increase of distribution power in MOL. The PPF of the reactor in BOL, MOL, and EOL is given by 1.16, 1.18, and 1.30.

5. Summary and Conclusions
The high breeding velocity of fast reactor impacted on the high value of reactivity swing and significant change of power density distribution especially for small power reactor who can operate more than 20 years. However, by adjusting fuel configuration in a good composition this effect can be reduced as was done in this study. As a comparison, optimization is performed on the reactor power 100 MWth (reactor I) and 150 MWth (reactor II), where maximum fuel configuration of any reactor in inner, mid and outer core are given by 6.6%, 11.6%, and 12.4% plutonium for reactor I, and 6.6%, 9.8%, 14.0% for reactor II. Based on the optimization result, it is known that the both of reactor have high breeding rates in the BOL, but due to the greater power, the buildup process in the second reactor takes place shorter. A shorter of buildup process on reactor II resulted in the decrease rate of CR value of this reactor being sharper than reactor I, where in the thirty years the CR value of this reactor had been less than 1, while the CR value of reactor I remained higher than 1. A CR value of more than 1 allows reactor I to operate more than thirty years, while the second reactor can only operate for twenty-nine years.

The shorter buildup process in reactor II also impacts the power distribution shift of reactor, where the shift of power distribution of reactor II from BOL to MOL is greater than reactor I, whereas the high rate of breeding in the MOL to EOL period in reactor I results in a power distribution shift over large compared to reactor II. Over all the PPF value of reactor II is higher than reactor I, but due to the reactor I can continue to operate, its PPF value can increase even bigger than reactor II so it is dangerous for the reactor.

Maximum reactivity of both reactors is still more than 1%. Maximum reactivity of reactor I occur on BOL of 1.02% so that reactivity swing of this reactor is greater than reactor II. The reactivity value of reactor I is 0.25%. The maximum reactivity of reactor II occurred in the year 18 is equal to 1.03% with reactivity swing of 0.16%.

References
[1] Rajan KS. Comparison of characteristics of fast and thermal reactors, Role of fast reactors in Indian Nuclear [Internet]. [cited 2017 Feb 28]. Available from:
http://www.nptel.in/course/103106101/Module - 12/Lecture - 2.pdf

[2] Processing of Used Nuclear Fuel – World Nuclear Association [Internet]. [2016 Nov; 2017 Feb 2]. Available from:
http://www.world-nuclear.org/information-library/nuclear-fuel-cycle/fuel-recycling/processing-of-used-nuclear-fuel.aspx

[3] Won JH, Cho NZ, Park HM, Jeong YH. Sodium-cooled fast reactor (SFR) fuel assembly design with graphite-moderating rods to reduce the sodium void reactivity coefficient. Nuclear Engineering and Design. 2014; 280:223-232.
[4] Rouault J, Wei TYC. The GEN IV Gas Cooled Fast Reactor: Status of Studies [Internet]. 2016 Feb 16 [cited 2016 Aug 8]. Available from:
http://www.oecd-nea.org/science/meetings/ARWIF2004/2.01.pdf

[5] Zajac R. Serpent Code Using in ALLEGRO Project [Internet]. 2014 Sep [cited 2016 Mar 23]. Available from:
http://www.montecarlo.vtt.fi/mtg/2014_Cambridge/Radoslav_Zajac.pdf

[6] Japan Atomic Energy Agency. JENDL [Internet]. [2017 Aug 1]. Available from:
http://wwwndc.jaea.go.jp/jendl/j40/j40..html

[7] Oka Y, Madarame H, Uesaka M, editors. An Advanced Course in Nuclear Engineering. Tokyo: Springer; 2014.