An Estimation of Fissile Material Production from the 5 MWe Yong-Byon Graphite-moderated Reactor by MCNP6 and SCALE6 Calculation

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ABSTRACT

The estimation of the fissile material production from the 5 MWe Yong-Byon graphite-moderated reactor is important to evaluate North Korea's capability to produce nuclear weapons. In this work, the amount of fissile material according to each operating period of the Yong-Byon graphite-moderated reactor were estimated using MCNP6 and SCALE6 core modeling, based on the operation histories and the core design of the reactor given in the open literatures. Furthermore, inventories of the Weapon Grade plutonium (WG Pu) were estimated and the two results of MCNP6 and SCALE6 calculation were compared. In addition, variables not provided in the open literature: fuel rod rearrangement, boron contamination change, power change, and graphite moderator density change, etc., were used for estimating production of fissile materials. Also, the estimated plutonium production for each variable is compared with the expected mass of plutonium according to the operation records of the Yong-Byon graphite-moderated reactor published by Heckers et al. The results from SCALE6 and MCNP6 show that the difference in WG Pu production for each variable is within 1% at the longest operating time before removing spent fuel. In addition, it was confirmed that the values calculated by SCALE6 and MCNP6 were in the range of the total amount of plutonium produced from 1989 to 1994 estimated by Albright and Hecker.

INTRODUCTION

North Korea has conducted a total of six nuclear tests until 2017. The six nuclear tests are expected to increase the sophistication and power of the nuclear arsenal. Since highly enriched uranium (HEU) and plutonium are needed to fabricate nuclear weapons, nuclear power plants for plutonium production and uranium enrichment facilities have increased (Hecker et al., 2016). The 5 MWe Yong-Byon graphite moderated reactor is a representative reactor that produces plutonium necessary for the production of plutonium bombs. The 5 MWe Yong-Byon reactor is similar to the model of the Magnox type reactor in the UK, and it can quickly extract plutonium by locating a reprocessing facility near the reactor. The reactor burns natural uranium enriched with 0.72% U^{235} as a source for making a plutonium bomb. Estimating the amount of Pu^{239} and U^{235}, which are fissile materials, is very important to know the extent of North Korea's nuclear strategy and risk. Hecker visited the 5 MWe Yong-Byon reactor in 2010 and published many reports. However, the amount of fissile material produced at the 5 MWe Yong-Byon reactor must be estimated through an indirect method such as satellite imaging, and the amount of fissile material must be estimated through many assumptions. Several experts estimated the fissile material production by estimating the burnup range, reprocess duration, and reprocessing method through the estimation of the operation history of the Yong-Byon reactor (Albright, 1994; Braun et al., 2016), but there are still many uncertainties. Accordingly, in order to reduce the uncertainty of the fissile material production quantity, it is necessary to estimate it considering various variables and scenarios.

Estimating the inventory of WG Pu and HEU (highly enriched uranium) in North Korea is critical to understanding the status of North Korea's nuclear weapons program and its potential risks. U235 and Pu239, which are important fissile materials for WG Pu and HEU, are estimated to have been produced in nuclear reactors, but there are many uncertainties. In order to reduce uncertainty, the fissile material production is estimated for the variables that have not been taken into consideration uncertainty: graphite moderator density, boron contamination, natural uranium enrichment, fuel rods arrange, power, and reloading scenarios. The amounts of U235 and Pu239 were estimated through MCNP6 and SCALE6 calculations. In particular, the estimated WG Pu amount was compared with the WG Pu production estimates for a specific period reported by Hecker.

Core modeling and variable settings

MCNP6 and SCALE6 core modeling

The thermal power of the 5 MWe Yong-Byon graphite moderated reactor is known as 25MWt and its initial uranium loading is about 50 tons of natural uranium (Kang, 2011). The reactor was designed based on the Calder Hall plant model in England and is classified as a Magnox type using Magnox cladding (1% Al and 99% Mg). Also, the reactor uses CO_2 gas as a coolant and graphite as a moderator. The main design parameters of 5 MWe Yong-Byon graphite moderated reactor are given in Table 1(Kang, 2011).

Table 1. Design specification of 5 MWe Yong-Byon graphite moderated reactor.

	Parameter	**Value**		**Parameter**	**Value**
Core Power	Thermal power	25 MWt	**Fuel Properties**	Uranium loaded	50 t
	Electric power	5 MWe		Uranium enrichment	0.71 wt%
	Specific power	0.50 MWt/tHM		Fuel composition	Nat U (0.5% Al)
Core Dimension	Effective core radius	643 cm		Diameter of fuel meat	2.9 cm
	Effective core height	592 cm		Length of fuel meat	52 cm
	Number of channels	812-877		Length of fuel rod	60 cm
	Number of fuel channels	801		Uranium per fuel rod	6.24 kg
	Number of control rod channels	44		Density of fuel meat	18.17 g/cm^3
	Number of fuel rods per channel	10	**Cladding Properties**	Clad composition	Mg (1% Al)
	Distance between channels	20 cm		Clad thickness	0.05 cm
	Diameter of fuel channel	6.5 cm		Cladding density	1.65 g/cm^3
	Diameter of Control rod channel	13 cm	**Coolant Properties**	CO_2 density	0.0068 g/cm^3
Reflector Properties	Graphite-reflector	300 t	**Moderator Properties**	Graphite-moderator	300 t
	Upper reflector	77.5 cm		Graphite density	1.62 g/cm^3
	Bottom reflector	66.5 cm			

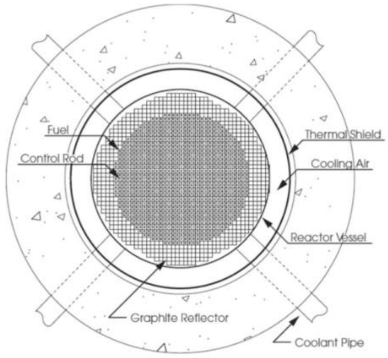


Figure 1. Radial layout of 5 MWe Yong-Byon graphite moderated reactor (Kang, 2011).

Figure 1 shows the radial layout of the reactor (Kang, 2011). The effective core is surrounded by graphite reflectors. The outside pressure vessel area is cooled by air and the vessel is surrounded by a thermal shield region. The SCALE6 and MCNP6 modeling used in this study considered only the graphite reflector for the core modeling as shown in Figure 2. This is because the region outside of the graphite reflector has a small influence on the neutron properties. In addition, since it is assumed that the control rods were all drawn out, the fuels are burned without being inserted. The outside of the cooling channels is filled with graphite, and the center of the control channels are filled with CO_2 gas. The configuration and dimensions of a fuel channel are shown in Figure 3.

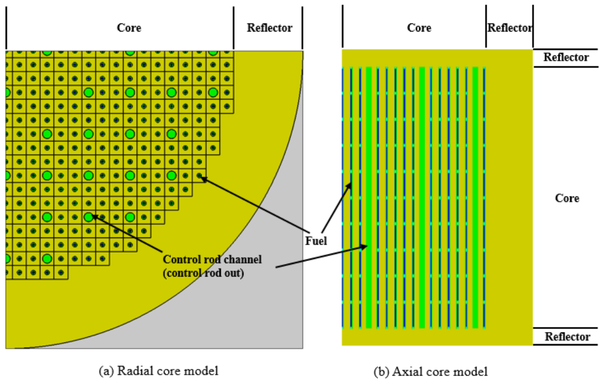


Figure 2. SCALE6 and MCNP6 model of 5 MWe Yong-Byon graphite moderated reactor

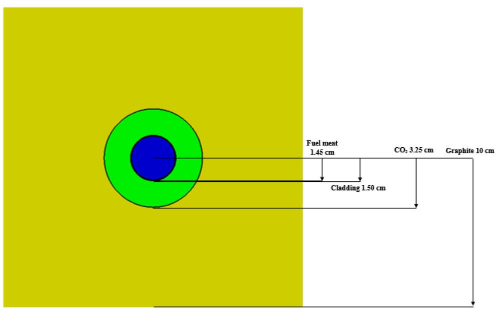


Figure 3. Fuel channel radial layout of the 5 MWe Yong-Byon graphite moderated reactor

Variables and range assumptions

Table 2. Range according to each variable

Variable	Range
Thermal power	20-30 MWt
Graphite density	1.62 – 1.72 g/cm3
Graphite boron contamination	0 – 3.2 ppm
Natural uranium enrichment	0.65 – 0.74% (U235)
Fuel rod arrangement	5 patterns

Among the parameters assumed in the study of the 5 MWe Yong-Byon graphite moderated reactor, graphite density, graphite boron contamination, thermal power, uranium enrichment, and fuel rod arrangement have been considered and studied as generally estimated values. In this work, five variables are considered and shown in Table 2. First, the 5 MWe Yong-Byon graphite moderated reactor is normally operated with a thermal power of 25 MWt, but there may be variations in power during operation. Therefore, thermal power is considered in the range of 20-30 MWt. Second, in the case of graphite used as a moderator, with the development of graphite manufacturing technology, the density of graphite used in graphite moderated reactors increased from 1.62 g/cm3 to 1.82 g/cm3(Marsden, 2001). Since the higher the density of graphite is, the more advantageous it is to produce fissile materials, so it is assumed that an effort was made to use a higher density and the above three density values were considered. Third, in the case of the graphite moderator contamination, graphite purchased from the commercial market outside the regulated nuclear supply chain generally contains 1-3 ppm of boron contamination, so up to 3.2 ppm of boron contamination is considered. Fourth, in the case of natural uranium enrichment, since there is uncertainty in the enrichment of U^{235} used in fuel, the fissile material production is calculated by changing the U^{235} enrichment from 0.66% to 0.76%. Lastly, in the case of the fuel rod arrangement, in the previous modeling, the arrangement of the control channels is arbitrarily arranged, but in reality, it may be a different arrangement, so different five fuel arrangement models are assumed and can be seen in Figure 4.

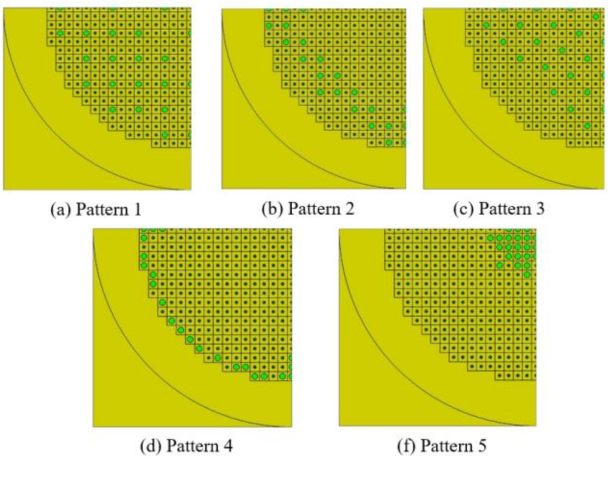
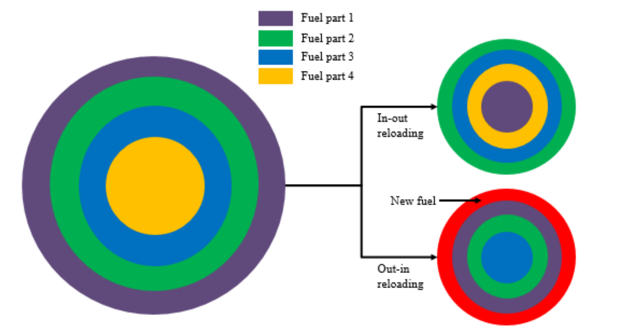


Figure 4. five different fuel arrangement models

Fuel reloading scenario assumptions

In the estimation of the operating history of the Yong-Byon reactor, the burnup and total operating time are estimated, but the record of reloading and the duration of one cycle are unknown. In this study, it is assumed that a total of one cycle is one year based on four-year operation, and fuel reloading takes place three times. Two patterns were assumed for reloading, in-out pattern and out-in pattern. The method of the 2 patterns is shown in Figure 5. Before reloading, the fuel channel is divided into 4 fuel groups and the fissile material production of each group is calculated after fuel depletion for one year corresponding to one cycle. After that, it is divided into two situations. In the case of the in-out reloading pattern, the burned fuel rods have a new arrangement. The new arrangement is after 1 cycle, loading the lowest fuels burnup (1/4 fuels of the core) at the center position and shifting the other fuels to outward. In the case of the out-in reloading pattern, the most burned fuel group is withdrawn and a new fuel group is loaded. The new arrangement is after 1 cycle, discharging the highest fuel group burnup (1/4 fuels of the core), shifting the other fuel groups to inward, and loading a fresh fuel group at the periphery.



(a) Divide fuel chanel in to 4 fuel groups (b) In-out/ Out-in reloading situation

Figure 5. Method of reloading fuels after 1cycle

Results of MCNP6 and SCALE6 core calculations

In this subsection, the difference in fissile material (U^{235}, U^{234}, Pu^{239}) production calculations according to depletion in the core modeled with the parameters in Table 1 of MCNP6 and SCALE6 is presented. Then, the calculation results of fissile material production according to each variable value range shown in Table 2 and reloading scenario are shown. Finally, the estimation of WG Pu according to the known operation history is compared with the estimation of MCNP6 and SCALE6 calculation.

Comparison of MCNP6 and SCALE6 results

In order to compare the results under the same standard, SCALE6 and MCNP6 were modeled identically with the parameters shown in Table1. In the same model, the results of the two codes for the mass of Pu^{239} and U^{235} from depletion fuels, and the Pu quality value, which is a value confirming WG Pu, are compared and shown in Figure 6. The fuel has an irradiation time of 2000 days, which corresponds to 1000 MWd/MTU when the heat power is 25 MWt. To compare the results of each time step, operation days of MCNP6 and SCALE6 are divided into the same step, and each time step is less than 100 days for more accurate comparison. In the case of Pu^{239} and U^{235}, which are fissile materials, the difference in the estimated production was within 1% during the irradiation time of 2000 days. In addition, it was confirmed whether the WG Pu, which plutonium having higher Pu^{239} content than 93 wt%, even when the fuel is burnup to 1000 MWd/MTU.

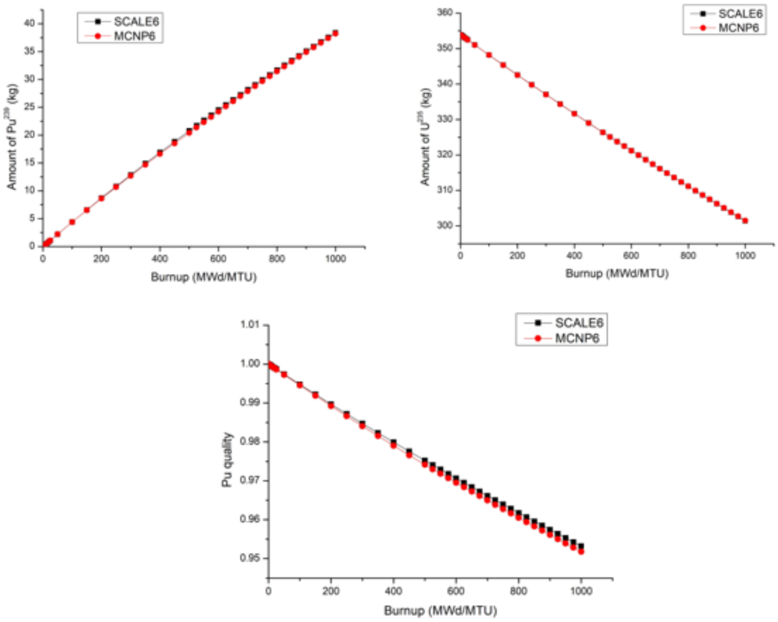


Figure 6. Comparison of the Fissile material production and Pu quality calculated by MCNP6 and SCALE6

<u>Fissile material production results according to each variable</u>

Figure 7 shows the comparison of the amount of fissile material (U^{235}, Pu^{239}) according to the thermal power change. Looking at the result value according to the irradiation time, it seems that the amount of Pu^{239} produced varies according to the thermal power, but there is no difference in the amount of Pu^{239} produced according to the thermal power when considering the burnup. In this study, the amount of fissile material is estimated according to the burnup because the burnup, which is an indicator of how much fuel was actually burned, is more important than the irradiation time because the shutdown period of the reactor was not known exactly.

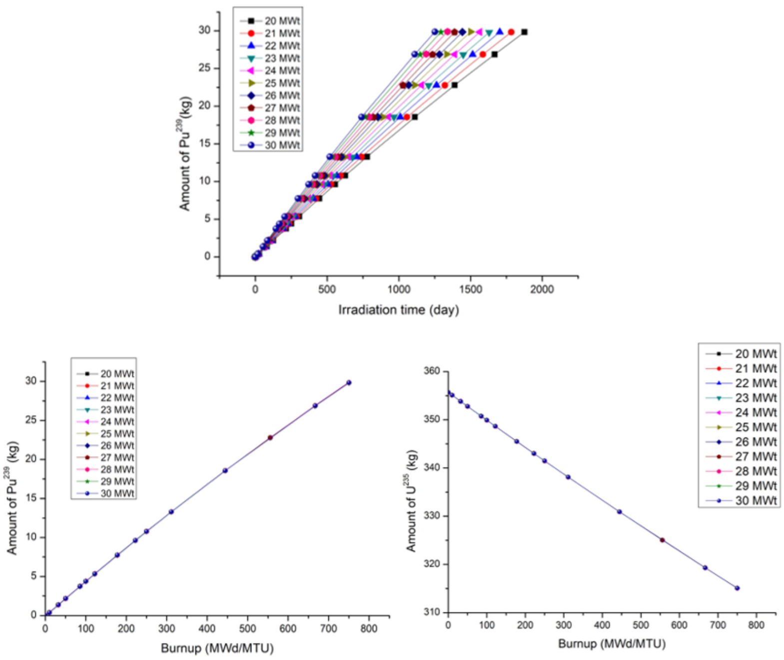
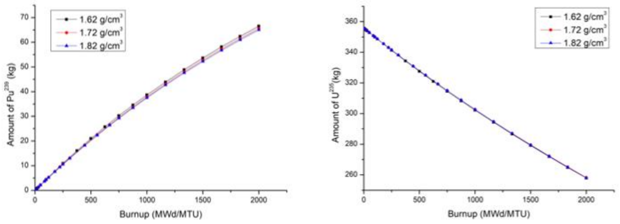
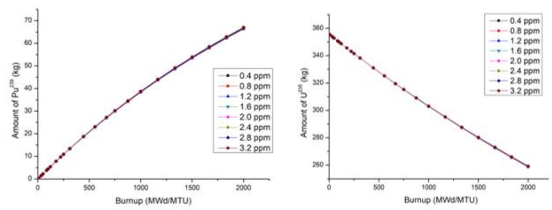


Figure 7. Amount of Fissile material (U^{235}, Pu^{239}) according to thermal power change

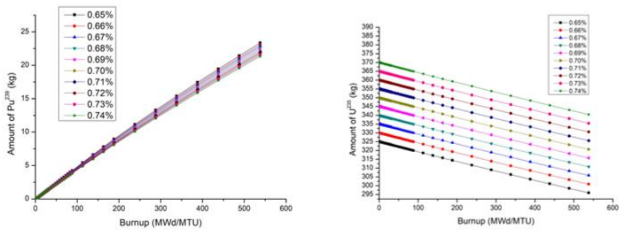
Figure 8 shows the amount of Pu^{239} and U^{235} according to fuel burnup for values in the range of values of the four variables assumed besides thermal power change. Figure 9 shows the range of plutonium production when fuel is irradiated for 1000 days for each variable. Since the Pu^{239} production range in normal operation is the value shown when the fixed parameters shown in Table 1 are considered, only one result is shown. As shown in Figure 8 and Figure 9, it can be seen that the uranium enrichment, graphite density, and thermal power change have the greatest effect on Pu^{239} production. However, it can be seen that there is little difference in Pu^{239} production in other variables.



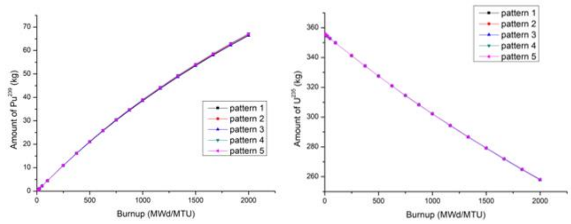
(a) Graphite density change



(b) Boron contamination change



(c) U^{235} enrichment change



(d) Fuel rod arrangement change

Figure 8. Amount of Fissile material (U^{235}, Pu^{239}) according to change of graphite density, boron contamination, U235 enrichment, and fuel rod arrangement

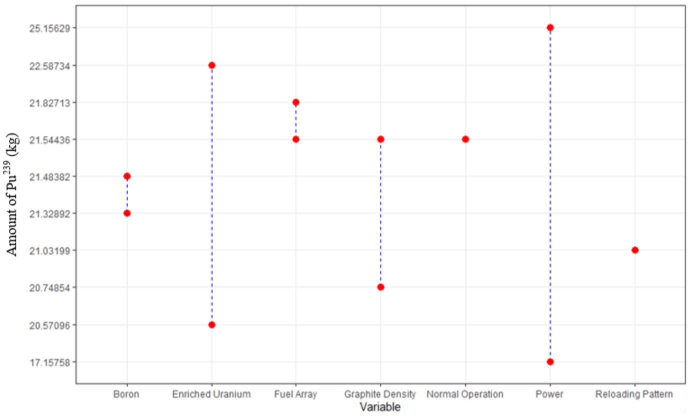
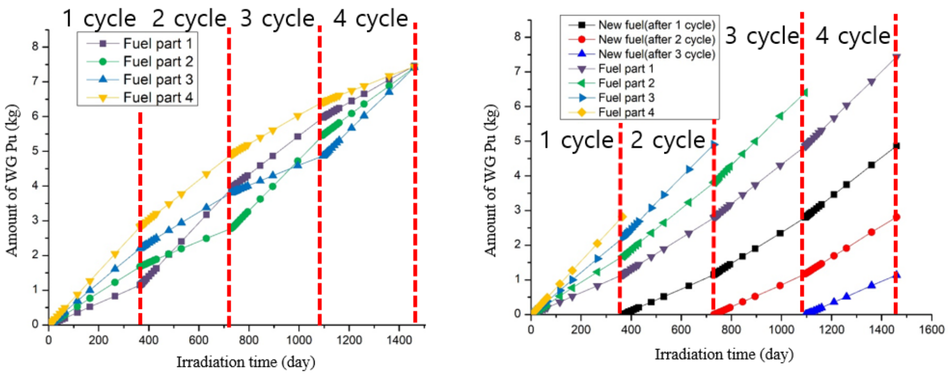


Figure 9. Pu^{239} production range for each variable at 1000 days

Results of fuel reloading scenario assumptions

Figure 10 shows the Pu production of each fuel group according to the two reloading scenarios during a total of four cycles. In the case of Figure 10. (a), since the fuel groups are rearranged after 1 cycle, the WG Pu producing rate of each group is different for each cycle, and it can be confirmed that the WG Pu of all fuel groups becomes the same after the fourth cycle. In Figure 10. (b), new fuel is loaded and the most burned fuel group is withdrawn. Since there are 3 fresh fuel group reloads during 4 cycles, there are 7 burned fuel groups. Figure 11 shows the total amount of WG Pu and quality of Pu in each fuel group in each reloading scenario. There is no significant difference in the amount of WG Pu production in the two scenarios, but in terms of quality, the in-out reloading pattern has the advantage to maintain better quality because new fuel groups are loaded.



(a) In-out reloading scenario **(b) Out-in reloading scenario**

Figure 10. Amount of WG Pu production for each fuel parts

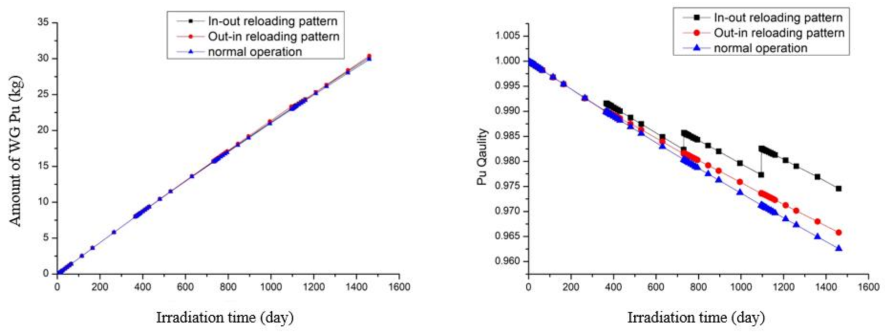


Figure 11. Comparison of WG Pu production and Pu quality of reloading scenarios and normal operation

<u>Comparison with known WG Pu estimates</u>

Among the estimated operating histories published by Heckers, the longest operating time and the highest burnup was from 1989 to 1994 (Heckers et al., 2016). During this period, the WG Pu production estimated by Hecker and Albright are compared with the WG Pu production range calculated by MCNP6 and SCALE6 considering the variables and scenario assumptions.

Table 3. Estimated range of WG Pu production

Year	2003 – 2005	
Burnup Range (MWd/MTU)	600 – 700 MWd/MTU	
Separated WG Pu (kg)	27 -29	David Albright(ISIS)
	25	Siegfried S.Hecker(CISAC)
	23.7 – 29.9	Estimated WG Pu production for MCNP6
	23.9 – 31.2	Estimated WG Pu production for SCALE6

Table 3 shows an estimate of a wider range than the range estimated by Albright and Hecker. In Table 3, the results of MCNP6 and SCALE6 show a difference of 1%.

Variables proportional to WG Pu production are graphite density, and variables inversely proportional to natural uranium enrichment and boron contamination. Also, when the control channels correspond to pattern 3 which are evenly distributed and when not reloading fuels, the WG Pu production is the highest. When the control channels correspond to pattern 4 which locate periphery of the core and when the out-in reloading scenario is used, the WG Pu production is the lowest. As a result of considering changes in all variables simultaneously, the range of SCALE6 is 23.9 kg to 31.2 kg, and the range of MCNP6 is 23.7 kg to 29.9 kg.

CONCLUSIONS

In this work, many experts in the past have estimated the amount of fissile material produced at the 5 MWe Yong-Byon graphite moderated reactor, but there were many parts that were not taken into account, so the uncertainty of the estimate was high. In order to reduce uncertainty, we aim to estimate fissile material production through the assumption that it is a reloading scenario and changes in the values of various variables that have not been considered. The results are cross-verified through two codes, MCNP6 and SCALE6, and the difference in fissile material production between the two codes is within 1%, and it is confirmed that plutonium was WG Pu within the burnup range of the operation history of the Yong-Byon graphite moderated reactor. In addition, as a result of comparing the WG Pu production range of Hecker and Albright with the range calculated using MCNP6 and SCALE6 for the longest operating period among the existing operating history, it seems that the range increased and the uncertainty was lowered. In a later study, fissile material production will be estimated additionally for other variables and scenarios not considered in this study. In addition, other than the 5 MWe Yong-Byon graphite moderated reactor, fissile material production is estimated for other nuclear reactors in North Korea to increase utility of the two codes.

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