A Feasibility Study of the Burnup Determination Based on the Neutron Interrogation of a Spent Fuel Assembly

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Abstract

A neutron interrogation has been performed to verify the burnup of the C15 Assembly which has an initial enrichment of 3.2 wt% and a NPP-declared burnup of 31.9 GWd/tU and has cooled for 21 years. The nuclide compositions were calculated with the ORIGEN2 code on the basis of the estimated burnup at the axial position of the assembly, which were then used in the MCNP calculation to simulate the spent fuel assembly and the interrogation process in order to obtain the background and induced fission neutrons, respectively. The 16 control rod guide tubes, a instrument thimble and 22 rod-empty positions as well as 157 spent fuel rods were described in the MCNP model. The ratios of the induced fission neutron to the background neutron were obtained from the measurement and the ORIGEN2-MCNP calculation, respectively, which were then used for the burnup determination of the C15 assembly. The results show that the measured burnups are in the range from 31.6 to 32.2 GWd/tU which are quite close to the declared burnup, 31.9 GWd/tU.

KEYWORDS: neutron interrogation, spent fuel assembly, burnup measurement, ORIGEN2 code, MCNP4C code

1. Introduction

The radionuclide compositions in PWR spent fuel are mainly estimated on the basis of the spent fuel burnup which is usually determined through a gamma spectrometry in which Cs-137 and Cs-134 are important isotopes. That is, the ratio of two isotopic gamma counts is used to determine the burnup with the curve of the burnup-to-count ratio obtained from a fuel irradiation simulation[1] by the ORIGEN2 code[2]. But it is not easy to apply this method to a spent fuel assembly. In this paper, the applicability of a neutron interrogation to a burnup measurement[3-5] for a PWR spent fuel assembly has been studied.

2. Experimental system

The experimental system consists of a neutron source (Cf-252), a neutron detector of a fission chamber type, electronic equipments and a PC including a spectrum analysis program. The neutron detector is extremely small, ϕ6.3 x 25.4 mm and it is a fission chamber type in which 93

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wt% U-235 is coated inside the chamber. The detector is inserted into a ϕ9.5 stainless steel tube which can be loaded into the control rod guide tube of a PWR spent fuel assembly[6]. The Cf-252 neutron source is encapsulated in a stainless steel cylinder size of ϕ6.8 x 10 mm. The intensity is about 1.32 x 10^7 n/s. The neutron source is inserted into a ϕ9.2 x 50 mm capsule and welded. The capsule is connected to a ϕ6 x 10 mm stainless steel bar which can be inserted into the control rod guide tube of a spent fuel assembly loaded at the PIEF storage pool which is ten meters deep.

The neutron-detecting signal passes a pre-amplifier and a main-amplifier and it is transmitted to the MCA, and finally the neutron counts can be read on the PC with a counting analysis software. The experimental system is automatically controlled with the integrated control system which consists of a Programmable Logical Controller(PLC) and a stepping motor set, which has been developed on the basis of the GENIE 2000 Library [7,8].

3. Burnup Measurement Procedure

The burnup determination procedure is divided into three parts. The first is that the background and the Cf-252-induced fission neutron counts for a PWR spent fuel assembly are measured at a given point, and the measured ratio is determined as the ratio of the fission neutron count to the background. The measurement is carried out by using the neutron source and detector inserted in one of the control rod guide tubes of the assembly, respectively. The second is that the measurement is simulated through the ORIGEN2-MCNP[9] calculations. The isotopic compositions at the given burnup are calculated with the ORIGEN2 code. Then by using the compositions, with the MCNP code, the background and neutron interrogation are simulated to calculate the counts and finally the calculated ratio is determined in the same manner as the measurement. The third is that the burnup is determined by a comparison of the measured and calculated ratios.

4. Neutron Measurements

A measurement of the neutron counts has been performed for the C15 Assembly which is stored at the KAERI Post Irradiation Examination Facility(PIEF) after being discharged from Kori Unit 1 in Korea, which has an initial enrichment of 3.2 wt% and a NPP-declared burnup of 31.9 GWd/tU and has cooled for 21 years. Presently 157 rods remain in the assembly because 22 fuel rods have been removed for a post irradiation examination as shown in Figure 1. The positions of the neutron source and detector are also shown in Figure 1, respectively.

The background neutron which is emitted from the spent assembly is counted at a given point with moving the fission chamber detector from the center of the C15 assembly to the upper end at an interval of 2 cm. The counting time is an hour at each position. In the case of the neutron interrogation with the Cf-252 source, the detector and source are simultaneously moved in the axial in the same interval as the background measurement and the counting time is thirty minutes per a measurement position.

The net induced fission neutron count is obtained by a subtraction of the background from the total measured neutron count obtained by the interrogation, which is presented with the background measurements in Figure 2. The background neutron distribution shape is similar to an axial gamma-ray distribution except that the end part is decreasing rapidly. In the case of the net induced fission neutron, the shape is very different from the background and especially at the end part, the fission neutron count is increased in contrast to the rapid decrease of the background.

Figure 1: Cross-sectional view of the C15 assembly drawn by the MCNP code

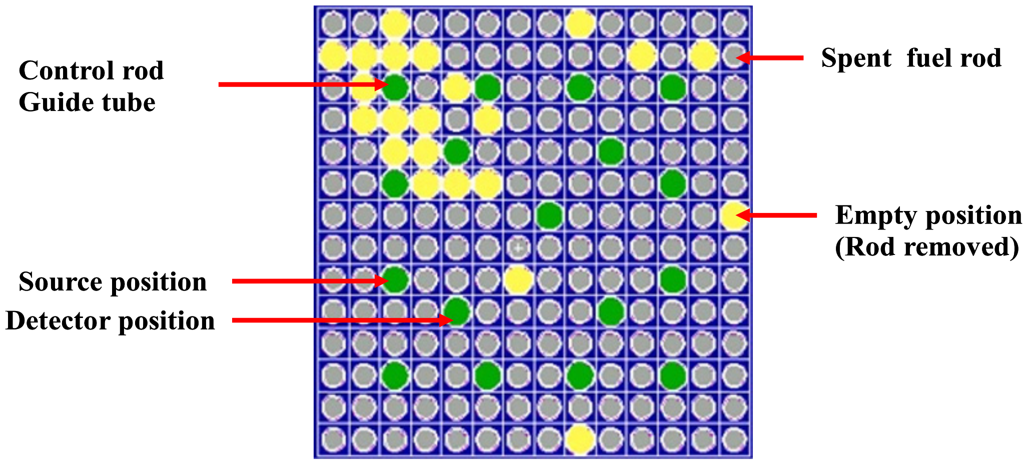
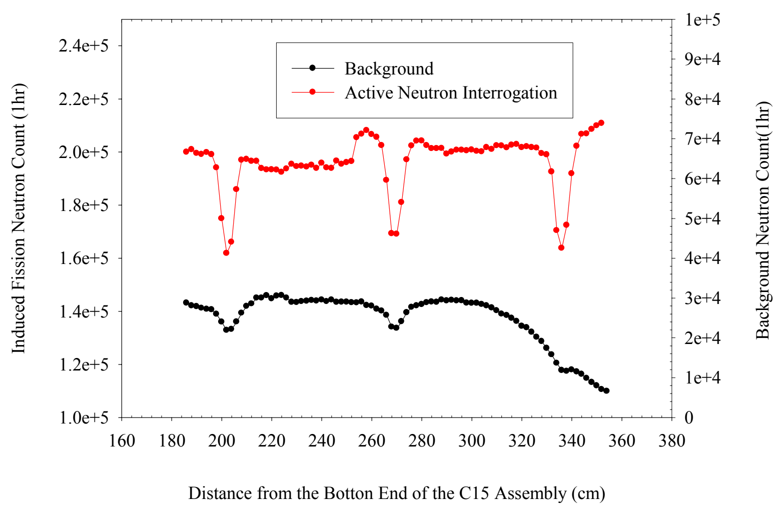


Figure 2: Measured background and neutron interrogation counts for the C15 assembly.



5. Calculation and Burnup Determination

The assembly-average burnup of the C15 assembly, which is declared by NPP, is 31.9 GWd/tU. As presented in Figure 3, the burnup at each axial position of the assembly was obtained from an estimation based on an axial gamma-ray distribution in a spent fuel rod of the C15 assembly with the same burnup. The nuclide compositions were calculated at each position by using the ORIGEN2 code on the basis of the estimated burnup at each position. Twenty nuclides with a large contribution to the neutron fission or absorption were selected at each burnups and one of them is presented in Table 1. The material card for the MCNP input was created on the basis of the nuclides amounts at each burnup. The upper half of the assembly was modeled with a reflected boundary condition which was adapted on the bottom end. The 16 control rod guide tubes, an instrument thimble and 22 rod-empty positions as well as 157 spent fuel rods were described well in the MCNP model shown in Fig. 1. The background and interrogation neutron counts were calculated at four positions near the upper end part of the assembly where the burnup was decreasing rapidly. The calculation results and the measurements are presented with the ratios of the interrogation counts to the background in Table 2.

Figure 3: An estimated axial burnup distribution for the C15 assembly

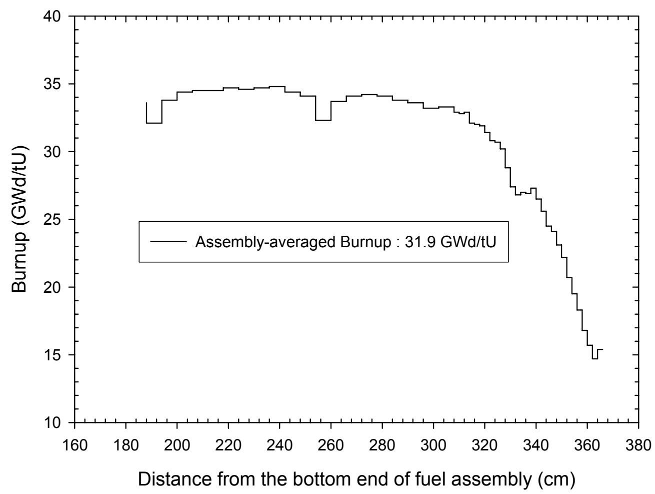


Table 1: Twenty selected nuclides and their compositions at the burnup of 31.9 GWd/tU

Nuclide	Amount(g/1MTHM)	Nuclide	Amount(g/1MTHM)
U-235	8030	Tc-99	756.5
U-236	3930	Rh-103	442.3
U-238	945200	Xe-131	417.3
Pu-238	124.8	Cs-133	1098
Pu-239	5156	Nd-143	775.4
Pu-240	2016	Sm-149	3.3
Pu-241	390.	Sm-151	11.5
Pu-242	438.1	Sm-152	126
Am-241	874.6	Eu-153	114.4
O-16	134100	Gd-155	11.4

The M/C values, which are presented in Table 3, were obtained from the measurement ratios divided by the calculation ratios in Table 2. What the M/C value is larger than 1.0 indicates that the measured burnup is lower than the reference burnup, the C15 assembly-averaged burnup of 31.9 GWd/tU, on which the ORIGEN2 irradiation calculation has been based. In the reverse case, the estimated burnup is higher than the reference burnup. The burnup of the C15 assembly is estimated from the ratios of the induced fission neutron to the background neutron obtained from the measurement and the ORIGEN2-MCNP calculation, respectively. The burnup determination result shows that the measured burnups are in the range from 31.6 to 32.2 GWd/tU, which are quite close to the declared burnup, 31.9 GWd/tU. From the results, the feasibility of a burnup measurement by using an interrogation seems to be confirmed. The burnups at the four positions are determined in the same manner and the measured burnups are very similar to the estimated values based on the declared burnup of the C15 assembly-averaged as shown in Table 3.

Table 2: Background and induced fission counts at the four positions of the C15 assembly

Measured Position (cm)	Measurement(n/hr)			ORIGEN2-MCNP Calculation(n/s)		
	Background	Interrogation	Ratio	Background	Interrogation	Ratio
316	24895	202599	8.14	3230	26656	8.25
324	22537	201973	8.96	2981	26331	8.83
346	9797	206819	21.11	1362	28261	20.7
352	7001	210807	30.11	903	28528	31.6

Table 3: Burnup determination for the upper part of the C15 assembly

Measured Position (cm)	Ratio(M/C)	Burnup(GWd/tU)	
		Based on NPP-Declared Burnup	Measured(This Study)
316	0.99	32.5	32.6
324	1.02	31.1	31.0
346	1.02	25.1	25.0
352	0.95	22.6	22.9

6. Conclusion

The applicability of a neutron interrogation to a burnup measurement of a PWR spent fuel assembly has been carried through a verification of the declared burnup of the C15 spent fuel assembly discharged from Kori unit 1 in Korea. As the measured burnups appear in the range from 31.6 to 32.2 GWd/tU which seem to be consistent with the declared burnup, 31.9 GWd/tU, it is concluded that the neutron interrogation method is applicable to a burnup measurement of a PWR spent fuel assembly. But, for more concrete conclusion, it is necessary to prove that the results are reproducible for various PWR fuel assemblies.

Acknowledgements

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