Burnup and Source Term Analyses for a CANDU Spent Fuel

Seung-Woo, Lee, Dong-Keun Cho, Jong-Won Choi, Heui-Joo Choi
Korea Atomic Energy Research Institute

1. Introduction

Recycling a PWR spent fuel is assumed to be a premise in current R&D for the development of advanced nuclear fuel cycle. However, a direct disposal of CANDU spent fuel is currently being considered because of its low economical potential of fissile materials.

The reference burnup of 7,500MWD/MTU to obtain source terms was simply considered for the CANDU spent fuel to develop Korean Reference Disposal System during the 2nd Long-term R&D Study. However, it was chosen without the statistical analysis for the burnup of the entire spent fuels. In the design step of repository accommodating the high-level waste to be produced from advanced nuclear fuel cycle, the reference burnup representing whole CANDU spent fuels are required to be evaluated based on the actual discharge burnup.

In this paper, the comprehensive burnup analysis to determine the reference burnup for CANDU fuel was performed in statistical method. And then, the source terms needed to design the CANDU panel in the repository were evaluated.

2. Burnup Analysis

The presumption of actual burnup of spent nuclear fuels is crucial in aspect of shielding and safety analyses. For this reason, the statistical burnup analysis for the entire spent fuels produced from four CANDU reactors in Wolsong site from 1985 to 2007 was performed to determine the reference burnup.

An average burnup and the standard deviation of a spent nuclear fuel are listed for each unit in Table 1. An average burnup and the standard deviation were revealed to be 6,937 and 1,167, respectively. Based on these results, the burnup of 8,100 MWD/MTU was chosen as a reference burnup by adding the one standard deviation level. Because this burnup covers about 85% of whole spent fuels, the spent fuels exceeding this burnup can be accommodated by combining the spent fuel with low burnup fuel.

Table 1. Average burnup of CANDU Spent Fuel

	Average (MWD/MTU)	*Standard deviation*
Wolsong1	*6,923*	*941*
Wolsong2	*7,005*	*1,123*
Wolsong3	*6,858*	*1,249*
Wolsong4	*7,010*	*1,188*
Total	*6,937*	*1,167*

3. Source Term analysis Using ORIGEN-ARP

3.1 Features of ORIGEN ARP

ORIGEN-ARP was used to analyze source terms. ORIGEN-ARP is a sequence of SCALE system to perform point-depletion calculations with the ORIGEN-S using problem–dependent cross sections. This sequence allows the ORIGEN-S multi-burnup library for different assembly designs by an interpolation over pre-generated SAS2 cross-section libraries. The code can also provide user-specified energy groups for neutron and gamma spectra. Explicit ENDF/B-VI fission product yields were implemented for 30 actinides. Master photon library was completely updated based on ENDF/B-VI for 2,100 nuclides. The capabilities of a cross-section generation and comprehensive neutron sources are superior features of the ORIGEN-ARP. Code development and V&V activities have been performed extensively by Oak Ridge National Laboratory under the support of DOE and NRC.

3.2 Calculation results

3.2.1 Source Term Intensity and Spectra

Radiation from spent nuclear fuel causes radiolysis to the materials around the canister. Radiolysis produces an oxide and causes corrosion to the canister which degrades the integrity of the canister. In aspect of shielding analysis, radiological dose rate should under the criteria where the radiolysis does not occur. For this reason, photon and neutron intensities are used as input data in shielding analysis. Photon and neutron intensities for 1.0 MTU needed for the shielding analysis are listed in Table 2.

CANDU built-in library was used in analyzing a CANDU fuel. Source intensities and spectra were calculated as a function of the time for the representative CANDU fuel.

The results from this research will be used for future shielding analysis of the canister.

Table 2. Photon and Neutron Intensities

Time(yr)	1	30	40	50
Photon	5.1E+18	4.0E+14	3.2E+14	2.5E+14
Neutron	7.6E+06	3.1E+06	2.9E+06	2.7E+06

3.2.2 Decay Heat

Decay heat emitted from the spent fuel is treated as an important input parameter in the thermo-mechanical analysis, because it can decrease the safety performance of an engineering and natural barrier. Fig. 1 shows the trend of the decay heat as the nuclides decay out. This behaviour can be expressed by Eq. (1). The coefficients which best represent the calculated values were obtained by a statistical analysis, as shown in Table 3. This correlation formula will be used for future thermo-mechanical analysis.

$$P(t) = C1\ EXP(-A1t) + C2\ EXP(-A2t) + C3\ EXP(-A3t) + y_0 \quad (1)$$

Table 3. Coefficients for Correlation Formula

Time (year)	C1	C2	C3	R^2
1 ~ 100	251.12	5431.36	0	0.99
100~ 10^6	11.50	33.26	218.60	0.99

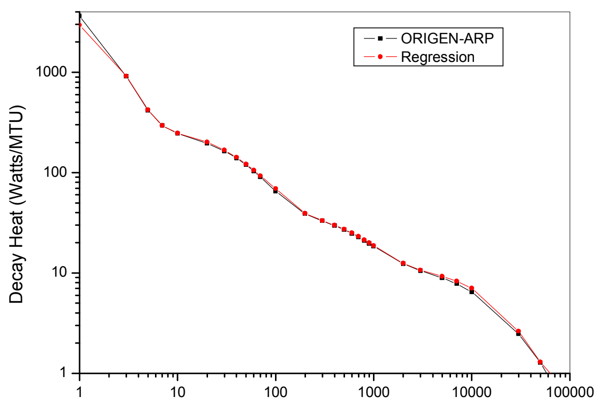

Fig. 1. Decay Heat for Representative CANDU Spent Fuel as a Function of Cooling Time

3.2.3 Nuclide concentration and Irradiation Hazard

All nuclide concentrations including actinides and fission products were estimated to be used as a reference in a safety performance of the repository. Ingestion hazard was also evaluated for long-term safety assessments.

4. Conclusions

In this study, current status of domestic CANDU spent fuels was analyzed and an actual burnup of a CANDU spent fuel was deduced by a statistical method. Source terms of the decay heat, radioactivity and nuclides concentrations for a high burnup CANDU spent fuel were estimated using ORIGEN-ARP.

Source terms evaluated for a high burnup CANDU fuel are expected to be used as input parameters in calculations for designing a conceptual disposal system and also it is expected to be used in optimizing the strategy for the conceptual design.

Moreover, it is expected that the results from this study will be applied to the many fields to evaluate spent fuel performance.

.

REFERENCES

[1] A. G. Croff, “A User’s Manual for the ORIGEN2 Computer Code,” ORNL/TM/7175, Oak Ridge National Laboratory, 1980.

[2] I.C. Gauld, et. al, “ORIGEN-ARP: Automatic Rapid Processing for Spent Fuel Depletion, Decay, and Source Term Analysis,” NUREG/CR-0200, Oak Ridge National Laboratory, 2004.