Evaluation of Decay Heat Distribution in the OASIS-32D Cask

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1. Introduction

The decay heat distribution from spent nuclear fuels in the OASIS-32D cask has been evaluated. The OASIS-32D is a dual purpose cask for spent fuel transport and storage under developing by KEPCO E&C. It is designed to store 32 PWR spent fuel assemblies (FA) that have been cooled for 10 years. The decay energy from spent fuels is used as a heat source for the thermal analysis of the cask design. In thermal analysis, it is generally assumed that all the decay heat is released from the spent fuel. However, in reality, gamma rays emitted from the spent fuel react with adjacent substances, and some of the decay heat is generated in the surrounding structure. For the evaluation of the spatial distribution of decay heat from spent fuel in the cask, the gamma energy depositions in each structure of the cask were obtained by carrying out gamma ray transport calculation.

2. Methods and Results

2.1 Source term calculation

The spent fuels for the decay heat distribution calculation in the OASIS-32D are PLUS7 with an initial enrichment of 3.0 wt%. The PLUS7 fuels are assumed to be burned for three cycles at a power level of 40 MWt/MTU in the reactor core and have an average discharge burnup of 45 GWD/MTU. The amounts of decay energy from each particle (α, β, and γ) and decay gamma spectrum from spent nuclear fuels are calculated by ORIGEN-ARP module in the SCALE 6.1 code [1].

2.2 Gamma transport calculation

The alpha and beta particles have very short ranges so it can be considered to deposit all of their energies into the fuel pellet. The neutron and gamma rays emitted from spent fuel pellets react with surrounding materials and distribute their energies in the fuel regions and in the structure materials of the cask. The deposition energy of neutrons is negligible compared to that of other particles (α, β, or γ), so gamma transport calculation has been performed by MCNP5 code [5] to obtain the decay heat distribution in the components of OASIS-32D cask. The ratio of gamma decay energy to total decay energy (30.9%), derived from ORIGEN-ARP depletion calculation results, is used in conjunction with the total gamma deposition energy to evaluate the decay heat confined in the fuel region.

In the MCNP modeling, the PLUS7 fuel assemblies are homogenized as the materials of UO_2 fuels and Zircaloy clads. And all the cask components such as basket, neutron absorber, reinforcement structure, canister, cask shell, and external neutron shields are modeled as their actual shapes as shown in Fig. 1.

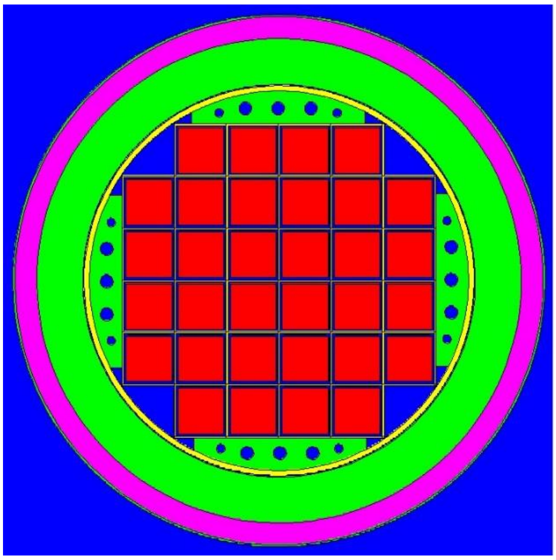

Fig. 1. MCNP model for transport calculation.

The cross-section data used in the MCNP5 gamma transport calculation is continuous cross-section data based on the ENDF/B-VI nuclear data library. The source energy distribution is based on the gamma spectra from spent fuel with a cooling time of 10 years and the axial burnup profile is reflected in the spatial distribution of the source.

Table 1 shows the deposition fractions of gamma energy to each material region and the deposition fractions of total decay energy generated from spent fuel assemblies as a result of gamma-ray transport calculation.

Table 1. Energy deposition ratio at each component

Component	Gamma Energy	Total Decay Energy
FA	93.22%	97.88%
Neutron Absorber	0.24%	0.07%
Basket	4.90%	1.51%
Reinforcement Structure	0.49%	0.15%
Canister	0.84%	0.26%
Cask Shell	0.31%	0.10%
Neutron Shield	0.00%	0.00%

3. Conclusions

The decay heat deposited in the spent fuel assemblies was evaluated to be 98% of the total decay heat. When realistic heat source distribution is required for the OASIS-32D thermal analysis, it is possible to apply the decay heat distribution results in this paper. However, the analysis result that the 98% heat source is distributed within the fuel assemblies suggests that the current thermal analysis assumption that all the heat sources are present in spent fuel is appropriate and guarantees some safety margin of the cask thermal analysis.

REFERENCES

[1] "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, June 2011.

[2] "MCNP-A General Monte Carlo N-Particle Transport Code, Version 5-1.51," Los Alamos National Laboratory, LA-UR-09-00384, January 2009.