Comparison of Source Term Analysis Result with Measured Data for Spent Fuel Hardware Characterization

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1. Introduction

In 2007, the 3rd Comprehensive Nuclear Energy Promotion Plan, passed at the 254th meeting of the Atomic Energy Commission, was announced as an R&D action plan for the development of a sodium-cooled fast reactor (SFR) in connection with a pyroprocess for a sustainable stable energy supply and a reduction of the amount of spent fuel(SF).

The source terms of assembly hardware for an intact SF are not an important concern in relation to deep geological disposal, because major activities are contributed by the decay of nuclides in the irradiated fuel itself. However, the source terms of the assembly hardware in the aforementioned advanced fuel cycle become relatively important, because major nuclides contributing to radioactivity and decay heat are removed by the pyroprocess for recycling and interim storage. It was found that about 90% of the PWR assembly hardware should be deposed of at a deep geological repository[1].

A source term evaluation for assembly hardware with a single irradiation profile can be easily accomplished with the conventional computation tool. However, source term assessment for a batch of assembly hardware or a mixture of metal wastes generated from SFs with different irradiation profiles-a task that is essential to support the source term generation for the design of a disposal system-is impossible with the conventional tool. This shortcoming was overcome through the development of an advanced source term estimation program in our country[2]. However, the developed program has not been sufficiently verified yet. Therefore, the program was verified by comparing the estimated nuclide inventory with the measured nuclide inventory in irradiated assembly hardware in the present study.

2. Explanation of Source Term Characterization Method for Assembly Hardware

II.1. Features of Source Term Estimation Program

An advanced source term evaluation program called ASOURCE has been developed by the Korea Atomic Energy Research Institute to support source term analysis to achieve the advanced fuel cycle being considered in Korea. ASOURCE has the following functions: (a) generation of inflow and outflow source terms of mixed SF in each process for the design of the pyroprocess facility; (b) overall inventory estimation for TRU and long-lived nuclides in SFs stored at each or all reactor sites for the design of the SFR; and (c) grand source terms of a batch of SFs with different irradiation and cooling profiles for the practical design of a temporary or interim storage facility of SFs. Please refer to reference 2 for more information.

II.2. Source Term Estimation Module for Assembly Hardware

Automatic source term characterization for assembly hardware is accomplished by using the *Screening*, *DeplDec*, *ReproRun*, *DecRes*, *MetalRun*, and *Batch* modules.

The rate at which the amount of nuclide i changes as a function of time is expressed as follows

$$\frac{dN_i}{dt} = \sum_j \delta_{ij}\lambda_j N_j + \sum_k f_{ik}\sigma_k \Phi N_k - (\lambda_i + \sigma_i \Phi) N_i \quad \cdots\cdots (1)$$

where, N_i= atom density of nuclide i,
σ_k= spectrum-average neutron absorption cross-section of nuclide k,
δ_{ij}= fraction of decay from nuclide j to i,
f_{ik}= fraction of nuclide absorption by nuclide k to i.

Because the neutron spectrum and flux level in the structural components vary within the assembly hardware, the radioactive nuclide inventory produced by (n, γ) reaction should be estimated by adopting an appropriate cross-section and neutron flux in Eq. (1). The cross-section generated by weighting the neutron spectrum of the core is utilized to solve Eq. (1) for structural components in the active core region such as the cladding, fuel rod end cap, and grid plate. The cross-section generated by weighting the neutron spectrum of region i is used for structural components in the outer core region such as the top-end piece and bottom end piece.

The neutron flux to solve Eq. (1) for the depletion calculation of the fuel itself is always retrieved by Eq. (2). However, the neutron flux for the activation calculation of the assembly hardware is obtained by multiplying the flux scaling factor by the average neutron flux of the fuel, as delineated in Eq. (3).

$$\Phi_{fuel} = \frac{6.242 \times 10^{18} (\overline{P})}{\sum_{i} N_i^f \sigma_i^f R_i} \quad \cdots\cdots\cdots\cdots\cdots\cdots (2)$$

where $\overline{P}$ and R represent the average specific power and recoverable energy of fission of nuclide i, respectively.

$$\Phi_{hardware} = \omega \, \Phi_{fuel} \quad \cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots (3)$$

where ω is a pre-generated or user-supplied flux scaling factor to represent the neutron flux of the structural component.

3. Radionuclide Quantification by Measurement

U3HA03 spent fuel with a 16 x 16 PLUS7 fuel design that was initially loaded in January 2003 and discharged in February 2007 was considered to produce a reference nuclide inventory for verification of ASOURCE. This spent fuel has initial enrichment of 4.5 wt.% and discharge burnup of 53,200 MWd/tU. Sampling was taken at four locations of Zirlo grid plates of aforementioned spent fuel. The sampled specimens were chemically pretreated with 3 mL of a mixed solution of diluted hydrochloric acid and nitric acid at the boiling point for two hours. The gamma rays from the solutions were measured using an HPGe gamma ray spectrometer, which is heavily shielded by lead.

The information on irradiation, overhaul, and decay time was applied to the depletion and decay calculation in ASOURCE. The specific power averaged over each cycle based on historical data was also applied in the depletion calculation of ASOURCE.

4. Discussions on Verification

The calculated nuclide inventory by ASOURCE versus the measured data agreed within 20% located in the upper region, 35% located in the upper middle region, and 20% located in the middle region of the core. Considering the activation calculation is mainly biased by the neutron capture cross-section, indicating that the bias is proportional to deviation of the estimated capture cross-section from the true value, the difference could be remarkably large. When viewed from this perspective, it is judged that the agreement as a whole is acceptable.

5. Conclusion

The Republic of Korea has developed an advanced source term analysis tool, called ASOURCE, to support R&D action plans for the achievement of an advanced fuel cycle employing a pyroprocess in connection with a sodium-cooled fast reactor.

It was found that the nuclide inventory calculated by ASOURCE for irradiated fuel cladding agreed with the measured data within 35%, indicating that the developed program supplies viable source term data.

6. References

[1] Cho, D.K., et al., "Waste Classification of 17×17 KOFA Spent Fuel Assembly Hardware," Nuclear Engineering and Technology 43 (2), 2011.

[2] Cho, D.K., et al., "Advanced Hybrid Analysis System for Nuclear Facility Design with Best Estimate Source Terms," Nuclear Engineering and Design, 2012(in press).