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Verification of source term estimation method against measured data for spent fuel hardware characterization

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ABSTRACT

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. ASOURCE has the following functions: (a) generation of inflow and outflow source terms of mixed spent fuel (SF) in each process for the design of the pyroprocess facility; (b) overall inventory estimation for Transuranics (TRUs) 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. ASOURCE is comprised of three analysis sequences, a Fuel Waste Characterization Sequence, Metal Waste Characterization Sequence, and Grand Source Term Characterization Sequence. In this study, the Metal Waste Characterization Sequence was verified by comparing the nuclide inventory estimated by ASOURCE with the measured nuclide inventory in an irradiated grid plate. It was found that the values calculated by ASOURCE agreed with the measured data within 35%, indicating that the developed program supplies viable source term data.

1. Introduction

There are now 23 commercial nuclear reactors operating in the Republic of Korea (International Atomic Energy Agency, 2012). The total amount of spent fuel (SF) generated by the end of 2010 was revealed to be about 11,371 metric tons of uranium (MTU). The annual production rate is about 700 MTU (Cho et al., 2008). According to the *Basic Plan of Electricity Supply and Demand* (Ministry of Knowledge Economy, 2010) announced by the South Korean government, eleven PWRs will be built by the end of 2024. By achieving this plan, the installation capacity and electricity share of nuclear power plants should reach 31.9% and 48.5%, respectively.

The Korean government and nuclear industry have sought to propose a national policy for the safe management of SF (Lee, 2005; Hwang et al., 2007). In 2007, the *3rd Comprehensive Nuclear Energy Promotion Plan* (Atomic Energy Commission, 2007), 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 SF. By adopting this fuel cycle, the SF inventory can be greatly reduced through a recycling process in which Transuranics (TRUs) are burned in the SFR and cesium and strontium are discarded after sufficient interim storage to lower the eventual decay heat of radwaste in the final repository. Additionally, the period needed for the radiological toxicity of SF to be reduced to that of natural uranium can be shortened to hundreds of years through burning of the recovered TRU in the SFR (Ko et al., 2007).

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 (Gauld and Murphy, 2010). 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 (Choi et al., 2011; Kook et al., 2012). It was found that about 90% of the PWR assembly hardware should be deposed of at a deep geological repository (Cho et al., 2011a).

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 the Republic of Korea

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(Cho et al., 2012a,b). In the previous study (Cho et al., 2012b) verification of the developed program was carried out, however, it was not performed for hardware analysis module, called Metal Waste Characterization Sequence. Therefore, this sequence 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

2.1. Features of advanced 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 the Republic of 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.

To carry out these functions, ORIGEN-S (Gauld et al., 2009a,b) in the SCALE code package (Oak Ridge National Laboratory, 2009) is used as a depletion and decay chain solver to avoid both unnecessary engineering time for preparation of a variety of peripheral parameters such as cross-section values and nuclear data, and inevitable verification of the newly developed solver. Currently, several additional functional modules, called *Screening*, *DeplDec*, *DecRes*, *ReproRun*, *MetalRun*, and *Batch* are also being prepared for completeness of the program.

The *Screening* module extracts SFs, to which calculations are to be performed, from a SF database. Provision of the SF database includes physical characteristics, irradiation characteristics, and storage characteristics for each fuel assembly identification (ID). The role of the *DeplDec* module is to irradiate and decay SF for specified irradiation and cooling time considering appropriate physics parameters by calling ORIGEN-S. The role of the *DecRes* module is to decay SF or radwaste through a restart calculation with pre-calculated information. The role of the *ReproRun* module is to separate radwastes generated from the pyroprocess considering the removal ratio of each nuclide for each unit process specified by the user. The role of *MetalRun* is to irradiate and decay structural material comprising assembly hardware. The role of the *Batch* module is to calculate the mixture composition when many kinds of SFs or structural components are processed or combined at the same time *t*.

ASOURCE currently has three analysis sequences, a Fuel Waste Characterization Sequence, Metal Waste Characterization Sequence, and Grand Source Term Characterization Sequence. By calling the appropriate functional modules outlined above, it can carry out user-defined tasks. Refer to the reference (Cho et al., 2012b) for more detailed explanation.

2.2. 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.

2.2.1. Analysis method

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

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

where N_i is the atom density of nuclide *i*, δ_{ij} the fraction of radioactive decay from nuclide *j* to *i*, λ_i the radioactive decay constant of nuclide *i*, f_{ik} the fraction of neutron absorption by nuclide *k* to *i*, σ_k the spectrum-average neutron absorption cross-section of nuclide *k*, and ϕ is the space and energy-averaged neutron flux.

Therefore, when *N* nuclides are considered, *N* equations of the above general form, that is, one for each nuclide, should be solved. The solution of this set of simultaneous differential equations yields the inventory of each nuclide present at the end of the time step. The terms multiplied by the neutron flux are important in the irradiation calculation when the production of nuclide *i* by the neutron absorption of nuclide *k* occurs, whereas they are meaningless in the decay calculation after discharge from a reactor.

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) (Cho et al., 2011b,c). 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, as shown in

$$\bar{\sigma} = \frac{\iint \sigma(E)\phi(E)^i \, dEdV}{\iint \phi(E)^i \, dEdV} \tag{2}$$

The neutron flux to solve Eq. (1) for the depletion calculation of the fuel itself is always retrieved by Eq. (3). 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. (4).

$$\phi_{fuel} = \frac{6.242 \times 10^{18}(\overline{P})}{\sum_i N_i^f \sigma_i^f R_i} \tag{3}$$

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

$$\phi_{hardware} = \omega\phi_{fuel} \tag{4}$$

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

2.2.2. Production of cross-section library

The cross-section library generated using the neutron spectrum of the appropriate assembly design is utilized in ASOURCE for the activation analysis of the cladding, grid plate, plenum spring, and guide tube. For the top-end and bottom-end pieces, the cross-section library was prepared by weighting the neutron spectrum of each component with the *t-depl* sequence, specifically the KENO-VI (Hollenbach et al., 2009)/ORIGEN-S module in the SCALE code package. The *t-depl* sequence calculates the neutron spectrum for each region of the fuel assembly by KENO-VI, and generates and updates the cross-section library needed for ORIGEN-S calculation by weighting the previously calculated neutron spectrum to the AMPX master library. The neutron flux needed to depletion is also calculated using KENO-VI, which uses a Monte Carlo method to transport neutrons for a criticality analysis. A three-dimensional model including the fuel rods and top-end and bottom-end pieces was established when the calculation for cross-section generation was performed.

Fig. 1a and b shows example of the geometry model when the cross-section library for top-end and bottom-end pieces of the