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Assessment of fission product inventory considering axial burnup of a fuel assembly

Hae Sun Jeong *, Hyo Joon Jeong, Eun Han Kim, Moon Hee Han, Won Tae Hwang

Korea Atomic Energy Research Institute, Daedeok-daero 989beon-gil 111, Yuseong-gu, Daejeon 305-353, Republic of Korea

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ABSTRACT

The fission product inventory for a loss-of-coolant accident (LOCA) analysis is generally evaluated based on the average burnup of the fuel at the end of life regardless of the variation of the actual burnup with time. In this study, we analyze the variation of the fission product inventory based on the accident occurrence cycle and the degree of fuel meltdown, which reflect the axial burnup of the fuel at the beginning, middle, and end of the cycle (BOC, MOC, and EOC). The fuel assembly for the Hanul unit 6, which is an OPR1000-type pressurized water reactor (PWR) in Korea, was chosen as an analysis model for the source-term assessment. The burnup cross-section library was generated using the TRITON module of the SCALE6.1 code system, and the depletion calculations were carried out using ORIGEN-ARP. The first scenario, which is referred to as CASE (I), is applied with an average burnup at the EOC, which is a commonly recommended conservative assumption in accident analyses. The CASE (II) scenario includes various cases that consider the accident occurrence cycle and fuel meltdown from the top to the bottom of the core at the BOC, MOC, and EOC. As a result, the inventory of the main nuclides gradually decreased as the reactor core operation was performed. In addition, the activity intensity for the main fission products of CASE (II) was shown to be higher than that of CASE (I) in the range above a fuel meltdown of approximately 30%. Therefore, this study concludes that not all of the results obtained by the CASE (I) assumption are guaranteed to be more conservative than those obtained by CASE (II).

1. Introduction

The evaluation of fission product releases from a reactor core into containment is necessary to guarantee a site's safety in the event of an accident involving a substantial meltdown of the core (Yang, 2014). Over the past few decades, a lot of information pertaining to the physical and chemical behavior of accident source terms has been developed to enable the radiological assessment of a design-based accident (DBA) through a variety of documents, which are published from the U.S. Nuclear Regulatory Commission (NRC), including TID-14844, Regulatory Guide (RG) 1.4, RG 1.183, and RG 1.195 (DiNunno et al., 1962; U.S. AEC, 1974; U.S. NRC, 2000; U.S. NRC, 2003). These regulatory guides all propose the need to make assumptions that can produce results that are more conservative than those realistically expected, although they each present different guidelines that consider different perspectives of the accident consequence analysis.

Because the starting point of all radiological consequences is defined by the initial core inventory, it is important to apply a suitable conservative assumption to obtain more precise results and assessments that enable us to estimate the activity of fission products. To determine the core inventory, most documents recommend the use of appropriate isotope generation and depletion computer codes such as ORIGEN2 or ORIGEN-ARP. In addition, the use of the maximum inventory at the end of life is also required because for some radionuclides, such as Cs-137, equilibrium is not reached prior to the fuel offload. Therefore, the average burnup at the end of life is applied to calculate the initial core inventory for a loss-of-coolant analysis (LOCA) in a safety analysis reports.

Meanwhile, the actual fuel burnup of the reactor core is nonuniformly distributed and fluctuates along the fuel height with the axial power distributions, which vary with time from the beginning of the cycle (BOC) to the middle and end of the cycle (MOC and EOC) (see Fig. 1). In addition, the LOCA procedure causes the fuel to melt vertically from the top to the bottom of the structure as the coolant level decreases. Realistically, the fission product inventory released during the meltdown of fuel in the LOCA event will therefore depends on the accident occurrence cycle and the degree of the meltdown. Because the axial power related to the fuel burnup often exceeds 100% in the middle area of the fuel for the entire fuel cycle, as shown in Fig. 1 (Kim et al., 2010), the

* Corresponding author. Tel.: +82 2 868 4516; fax: +82 2 868 2370.
E-mail address: haesunin@kaeri.re.kr (H.S. Jeong).

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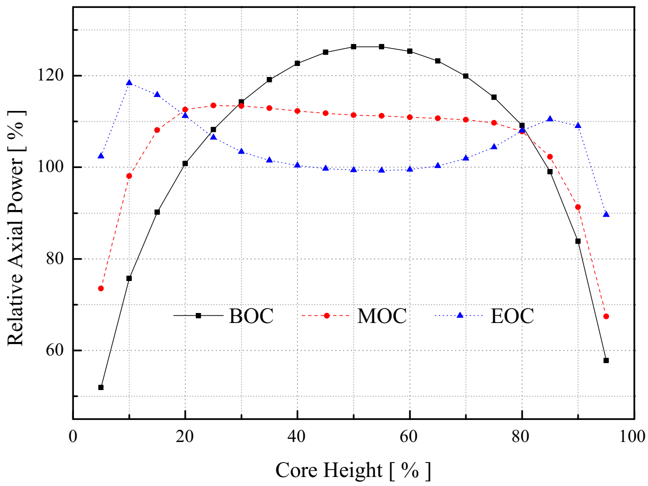

Fig. 1. Relative axial power distributions along the core height of Hanul unit 6 at BOC, MOC, and EOC.

conventional assumption for source-term calculations, which applies the average burnup condition at the EOC, needs to be verified to determine whether it is always sufficiently conservative.

In the present study, we analyzed the variation of the fission product inventory considering the LOCA occurrence cycle (BOC, MOC, or EOC) and the degree of fuel meltdown. The aims are to reflect the axial power distribution along the core height for each cycle, after which we derive a more realistic accident analysis. For this study, the TRITON (DeHart, 2009a,b) and ORIGEN-ARP (Gauld et al., 2009a,b) modules in the SCALE6.1 package code system (ORNL, 2011) are used to perform the depletion calculation based on the axial power distributions of the Hanul unit 6.

2. Methods and materials

2.1. Fuel assembly model

The analysis model used for the source-term assessment is the Optimized Power Reactor 1000 (OPR1000)-type Hanul nuclear power plant (NPP) unit 6. The reactor core is composed of 177 fuel assemblies with a 16 × 16 array, each of which includes 236 fuel rods and 5 guide tubes for the control rods and a tube for the in-core instruments, which are held at the top and bottom by end fittings. The active fuel length is about 3.8 m tall and just over 20 cm wide. The fuel assembly consists of UO_2 with 4.51 wt% U-235 for 184 rods and 4.00 wt% for 52 rods. The detailed design characteristics of the fuel rods, moderator, and guide tubes are summarized in Table 1.

2.2. Generation of burnup cross-section library

Most of the NRC's regulatory guides for evaluating radiological consequences recommend that the core inventory be determined using isotope generation and depletion computer codes such as ORIGEN2 or ORIGEN-ARP (U.S. NRC, 2000; U.S. NRC, 2003). Therefore, in this study, the radiological source terms were calculated using the ORIGEN-ARP module in the SCALE6.1 code system.

ORIGEN-ARP uses the ORIGEN-S isotope point-depletion code (Gauld et al., 2009a,b) to solve the time-dependent fission product concentrations. Because this module reads the prepared libraries incorporating reactor parameters that are specified by modeling the fuel assembly design, after which it solves the problem, a burnup-dependent cross-section library of the model to be analyzed should be produced in advance. There are several kinds of procedures that generate the ARP library with SCALE, and in this study, we used the TRITON module to produce the burnup cross-section library of ARP.

The TRITON module, which was developed by the Oak Ridge National Laboratory (ORNL), is used for various depletion calculations based on the two-dimensional lattice physics code NEWT (DeHart, 2009a,b) using discrete ordinate transport methods, the three-dimensional KENO, and ORIGEN-S (Zwermann et al., 2014). This basically provides five kinds of depletion sequences, namely T-XSEC, T-NEWT, T-DEPL, T5-DEPL, and T6-DEPL; in particular, T-NEWT and T-DEPL are used for the generation of the ARP's burnup cross-section library. These two sequences produce the three-group weighted cross-section library by deriving the volume average flux at each part in the model from NEWT, and the new library is finally imported into ORIGEN-S using the COUPLE module (Gauld and Hermann, 2009). As shown in Table 2, in this study, we applied the T-DEPL depletion sequence, and the ENDF/B-VII 238 cross-section set was used for radiation transport calculations. The T-DEPL sequence iteratively performs the process of transport and depletion calculations using the predictor–corrector approach. During the iterative phase, the cross-section processing and neutron transport solution are performed in the transport calculations, and the COUPLE and ORIGEN-S modules are used in the depletion calculations. At each mid-cycle point, the fluxes and cross sections are predicted and updated, after which the depletion calculations are performed over the full cycle based on the appended data.

The PRISM utility was used to create several input decks of TRITON by automatically converting the parameters defined in the template input file into new integer numbers. Fig. 2 shows the 1/4-sized fuel assembly model of the reactor core developed by TRITON. The maximum burnup was set to 72 GWD/MTU

Table 1
Design data for fuel assembly.

	Parameter	Data
	Fuel assembly	Combustion engineering 16 × 16
	Active fuel length	381 cm
	Fuel pin pitch	1.285 cm
	Number of fuel pins	236
	Number of guide tubes	5
	Assembly pitch	20.574 cm
Fuel rod	Fuel material	UO_2
	UO_2 mass	501.67 kg
	Fuel pellet outer diameter	0.826 cm
	Effective fuel temperature	961 K
	Clad material	Zircaloy-4
	Inner clad diameter	0.843 cm
	Outer clad diameter	0.97 cm
	Average clad temperature	612 K
Moderator	Moderator material	H_2O
	Average moderator temperature	585 K
	Moderator density	0.705 g/cm^3
	Average boron concentration	565.63 ppm
Guide tube	Guide tube material	Zircaloy-4
	Guide tube inner diameter	2.29 cm
	Guide tube outer diameter	2.49 cm

Table 2
SCALE modules with T-DEPL depletion sequence.

	SCALE modules
centrm 2region	CRAWDAD, BONAMI, WORKER, CENTRM, PMC, NEWT, COUPLE, ORIGEN-S, OPUS
nitawl	BONAMI, NITAWL, NEWT, COUPLE, ORIGEN-S, OPUS
nocentrm	BONAMI, WORKER, NEWT, COUPLE, ORIGEN-S, OPUS