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Original Article

Evaluation of source terms using the BESNA code for reference spent fuels for future underground disposal system of South Korea

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ABSTRACT

In this work, the source-terms such as nuclide inventories, radioactivities, neutron and gamma spectra, and decay heats were evaluated using new code BESNA in which the burnup dependent one-group cross sections were evaluated based on MCNP6. This work started with the selection of the reference spent fuels through a statistical analysis of all the accumulated PWR spent fuels in South Korea for safety analysis of a geological repository, which led to the two reference spent fuels for each of 16x16 and 17x17 fuel assembly types considering representative low and high burnups. Finally, the source terms were evaluated using our inhouse code BESNA for the reference spent fuels, and they were compared with those from ORIGEN in SCALE6.2 for validation purposes. The results of the comparison showed that BESNA gives good agreements with ORIGEN for all the considered source terms.

1. Introduction

In South Korea, the electricity production by PWRs and CANDUs has significantly contributed to the economic growth while a huge amount of nuclear spent fuels have been generated and accumulated. Currently, all the spent fuels generated from PWRs are stored in the spent fuel storage pools in the nuclear power plant sites but the ones from CANDUs are in dry storage facilities such as MACSTOR and SILO. However, it is expected that the storage capacities of the PWR spent fuel storage pools will exceed in the near future. For example, the capacities of the spent fuel storage pools for HANBIT, HANWOL, and KORI units are expected to exceed their limits from 2030, 2031, and 2032, respectively [1]. So, this spent fuel storage issue is more severe for PWRs than for CANDUs, and there is a strong need to consider the dry storage for PWR spent fuels.

Under these situations, the government of South Korea has actively driven a long-term program to develop the technologies needed for the deep disposal of spent nuclear fuels in a geological repository. As a part of the national program, KORAD (Korea Radioactive Waste Agency) has initiated a long-term research and development project to enhance technical capabilities for a deep underground high-level radioactive waste disposal system [2]. One of the goals of the project is to evaluate the long-term safety and performance of a deep underground disposal system, which requires an understanding of the complicated coupled processes of thermal (T), hydraulic (H), mechanical (M), and chemical (C) mechanisms. This kind of evaluation starts with the source term evaluation, which reflects the characteristics of the spent nuclear fuels.

Recently, we have developed an in-house source-term code called BESNA (Bateman Equation Solver for Nuclear Applications) [3], which can solve the Bateman equation with the advanced numerical scheme called CRAM (Chebyshev Rational Approximation method) [4] for matrix exponentials and a new predictor-corrector scheme in which the simple Talyor expansion method is used in the corrector step but CRAM in the predictor step. Actually, this code was developed for various purposes such as point depletion analysis, activation analysis, and source terms' evaluation. The source terms include radionuclides' inventories, radioactivities, decay heats, and neutron and gamma emission spectra, including emission rates. In particular, BESNA was developed to be used as a main module in the several depletion and activation analysis-related codes by our group.

The objective of this work is to evaluate the basic source terms with our in-house code BESNA, which are the basic quantities required for the analysis of the transport of the radionuclides, and to verify the basic source terms by comparing the well-known ORIGEN code [5] of the newest version. Additionally, this work describes the selection procedure of the reference spent nuclear fuels through a statistical analysis based on the accumulated spent nuclear fuels in South Korea for the long-term safety and performance analysis of the underground disposal

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system. The source term evaluation is done for the selected nuclides in the reference spent nuclear fuels.

In Section 2, the computational models and libraries related to BESNA are simply reviewed. Section 3 describes the statistical analysis of the accumulated PWR spent nuclear fuels in South Korea to establish the reference spent nuclear fuels. Section 4 describes the results of the source term evaluations including verification. Finally, the summary and conclusion are given in Section 5.

2. Review of computational models in BESNA

In this section, the computational models of BESNA for the source term evaluation are reviewed briefly, and details of them are given in Ref. [3]. Actually, the basic equation BESNA solves is the following Bateman equation:

$$\frac{dN_i}{dt} = -(\lambda_i + \sigma_i\varphi)N_i + \sum_{j\neq i}^{M}(\lambda_j b_{j,i} + \sigma_{j,i}\varphi)N_j + \sum_{j=1}^{L}\gamma_{j,i}\sigma_{f,j}\varphi N_j, \quad (1)$$

where, N_i, σ_i and λ_i are the atomic number density, microscopic effective one-group removal cross section, and decay constants of nuclide i, respectively. In Eq. (1), φ is the total integrated flux over neutron energy and, $\sigma_{j,i}$ and $b_{j,i}$ represent the microscopic effective one-group cross section of the reactions of nuclide j which, produces nuclide i and the decay branching of nuclide j which, produces nuclide i, respectively. $\sigma_{f,j}$ is the effective one-group fission cross section of nuclide j. $\gamma_{j,i}$ is the yield of the fission product nuclide i produced by the fission of actinide nuclide j. The BESNA code solves the vector form of Eq. (1), which is given by

$$\frac{d\vec{N}}{dt} = A\vec{N}. \quad (2)$$

If the coefficient matrix A is specified as the independent quantities on time over a time interval Δt, the formal solution is given by

$$\vec{N}(t+\Delta t) = e^{\Delta tA}\vec{N}(t). \quad (3)$$

In Eq. (3), it should be noted that the coefficient matrix A is not constant because it depends on the effective one-group cross sections and flux, but they are changing over the time interval. For solving Eq. (2) over a time interval, BESNA separates the reaction part from the whole matrix after applying the predictor-corrector method as follows:

$$\vec{N}(t+\Delta t) = e^{\Delta t\left(A^R_{MOS}\varphi_{MOS} - A^R_{BOS}\varphi_{BOS}\right)} e^{\Delta t\left(A^D + A^R_{BOS}\varphi_{BOS}\right)}\vec{N}(t). \quad (4)$$

where A^R_{MOS} and A^R_{BOS} represent the reaction components of the coefficient matrix at the middle and beginning points of the time interval, respectively, while A^D represents the decay component of the coefficient matrix. φ_{BOS} and φ_{MOS} are the neutron fluxes at the beginning and middle points of the time interval, respectively. This separation significantly reduces the matrix norm of the first term of Eq. (4), and TEM (Talyor Expansion Method) can be effectively used for this term while the second term is be solved by CRAM, which improves the computational efficiency. BESNA also adopts the sub-step calculations as in ORIGEN to improve the estimation of neutron flux changes.

BESNA has functionalities for calculating the source terms such as the radioactivity, decay heat, and neutron and gamma emission spectra, including emission rates over a user defined energy group structure. The photon data in BESNA includes the line-energy and intensity data for the X-rays, gamma from decay, spontaneous fission gamma, (α, n) reaction gamma, and bremsstrahlung from negatron and positrons slowing down in the UO_2 mixture. The neutron emission source in BESNA considers the neutrons generated from (α, n) reactions and spontaneous fissions. The methodologies used in the neutron and gamma emission spectra, including emission rate, are similar to those in ORIGEN.

The decay library of BESNA considers 2237 nuclides and eleven decay types, including isometric transitions, neutron decay, and delayed neutron decay. Actually, BESNA uses the decay data given in the ENDF/D data. In this work, the burnup-dependent one-group cross sections for BESNA are evaluated using MCNP6 [6] depletion calculation for fuel assembly types where the total reaction rates and total fluxes are tallied at several burnup points. Currently, the MCNP6 depletion calculations are performed using the ENDF/B-VII.1 and ten reactions (i.e., (n,γ) to ground or isomeric states, $(n,2n)$ to ground or isometric states, (n,α), (n,p), (n,d), (n,t), $(n,3n)$, and $(n, fission)$). The cross sections for the reactions producing the products at the isomeric state are calculated by combining the 63-group neutron fluxes from MCNP6 at various burnups with the branching ratios calculated from the CINDER [7] library. The fission product yield library for 30 fissionable actinides in BESNA was prepared from the one given in the ENDF/B-VII.1 data.

Table 1
Status of the accumulated amounts of spent nuclear fuels in South Korea (as of March 2023).

Types	Site	Storage Types	Number of spent fuel assemblies or bundles
PWR	Kori	Wet	5,216
	Shin Kori	Wet	1,823
	Sewol	Wet	496
	Hanbit	Wet	7,093
	Hanwol	Wet	6,455
	Sin Hanwol	Wet	156
	Sin Wolsung	Wet	794
	Sub-total		22,033
CANDU	Wolsung	Wet	152,008
	Dry Storage (MACSTOR & Silo)	Dry	347,760
	Sub-total		499,768

3. Reference spent fuel assemblies

3.1. Statistical analysis of PWR spent nuclear fuels

As of March of 2023, 22 PWRs and 3 CANDUs, of which total designed capacity is 24.7 GWe, are operating in South Korea. Furthermore, based on the 10^{th} Basic Supply and Demand Plan of South Korea [8], the designed electricity capacity will increase by 31.7 GWe through the construction of new PWRs such as Shin Hanwol Units 2, 3, 4, and Shin Kori Units 5 and 6. Such operations of nuclear power plants have generated huge amounts of spent nuclear fuels. Table 1 summarizes the accumulated amounts of spent nuclear fuels in South Korea as of March 2023. As shown in Table 1, a total of 22,033 spent fuel assemblies from PWRs are stored in the storage pools as of March 2023, while 152,008 spent fuel bundles from CANDUs are stored in the storage pools, and 347,760 bundles are in the dry storage facilities such as MACSTOR and Silo.

In South Korea, many different types of spent fuel assemblies with different design specifications have been generated. Also, the spent fuels have different irradiation histories and different loading of initial uranium. So, it is not desirable (impractical) to generate the source terms for all the individual spent fuels or for all types of spent fuels, and so it is a usual way to select or set up the reference spent fuels and to use the source terms from the reference spent fuels in the safety or performance analysis of the underground disposal system [9]. In this work, the selection of the spent fuels will be confined to the spent fuels from PWRs because the burnups of spent fuels from CANDUs are much lower than those of PWR spent fuels. The statistics of the PWR spent fuels generated from 1979 to 2012 were analyzed based on the database from KHNP (Korea Hydro & Nuclear Power). The data from the database include initial uranium enrichment, initial uranium loading, average discharge burnup, specific power, lattice type, the final discharge date, and effective full power days.

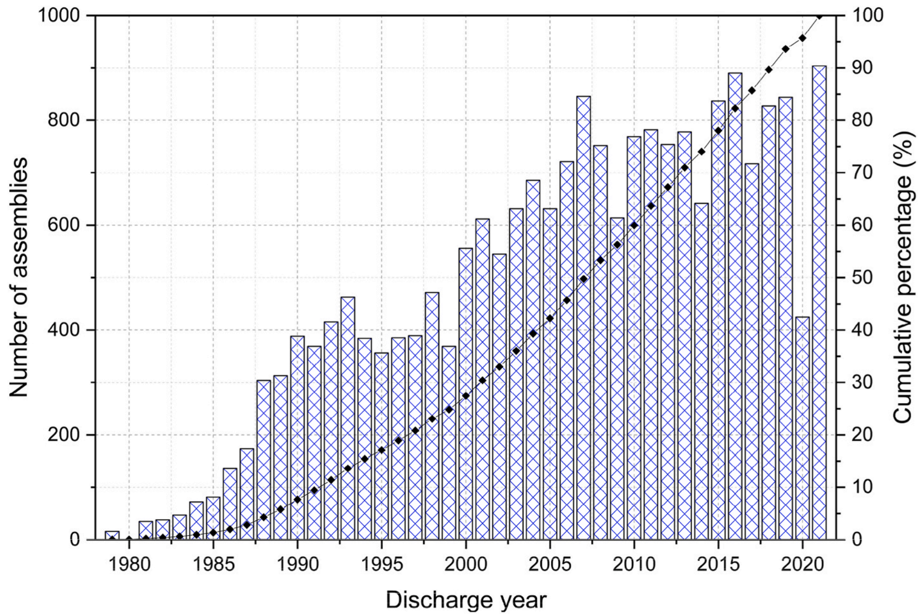


Fig. 1. Change of the number of annually discharged spent fuel assemblies from PWRs in South Korea.

The PWR spent fuels from South Korea can be classified into four different types depending on their lattice structures as follows: 1) WH (Westinghouse) 14×14, 2) WH 16×16, 3) WH 17×17, and 4) CE (Combustion Engineering) 16×16. The WH 14×14 and WH 16×16 type assemblies were discharged from Kori Units 1 and 2, respectively, while the WH 17×17 ones were from the Kori Units 3, 4, Hanbit Units 1, 2, and Hanwol Units 1, 2. On the other hand, the CE 16×16 type assemblies were discharged from Hanbit Units 3–6, Hanwol Units 3–6, Shin Kori Units 1, 2, Sewol Units 1, 2, Shin Wolsung Units 1, 2, and Shin Hanwol Unit 1.

The WH 14×14 type fuel assembly is further divided into four different types (WH standard, JDFA, OFA (WH), and OFA (KNFC)). The lattice pitches, number of rods, and active fuel lengths of these types are the same, but the uranium loadings are different depending on the outer radii of the uranium pellets and cladding dimensions. Of them, the WH standard one has the largest uranium loading of 405.8 kg. The WH16×16 type fuel assembly is divided into three different types (WH standard, JDFA, and KNFC standard). They also have the same values of lattice pitch, number of rods, and active fuel length but their uranium loadings are slightly different due to their different cladding dimensions and uranium pellet radii. The WH standard one has the largest uranium loading of 410.7 kg. In South Korea, there are five different types of WH 17×17 type fuel assembly (WH standard, OFA, KOFA, V5H, and RFA). They have different uranium loadings, which are much higher than the previously addressed assemblies due to their larger number of fuel rods in spites of the same active fuel lengths. Of them, RFA has the largest uranium loading of 461.5 kg. There are three different spent fuel assemblies (CE standard, Guardian, and PLUS7) of CE 16×16 lattice structures in South Korea. The CE standard and Guardian types have all the same dimensions, so they have the same value of uranium loading.

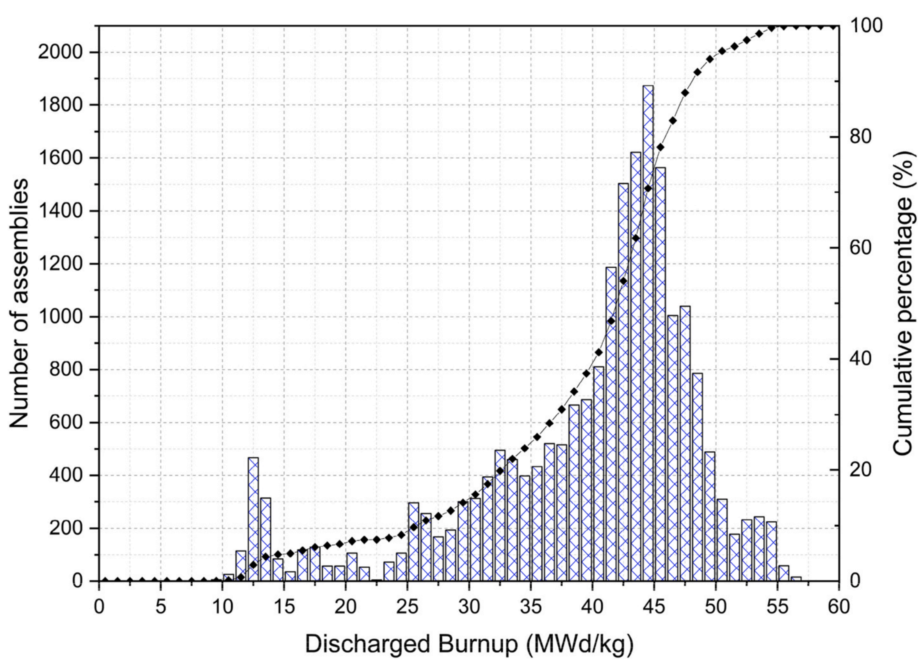
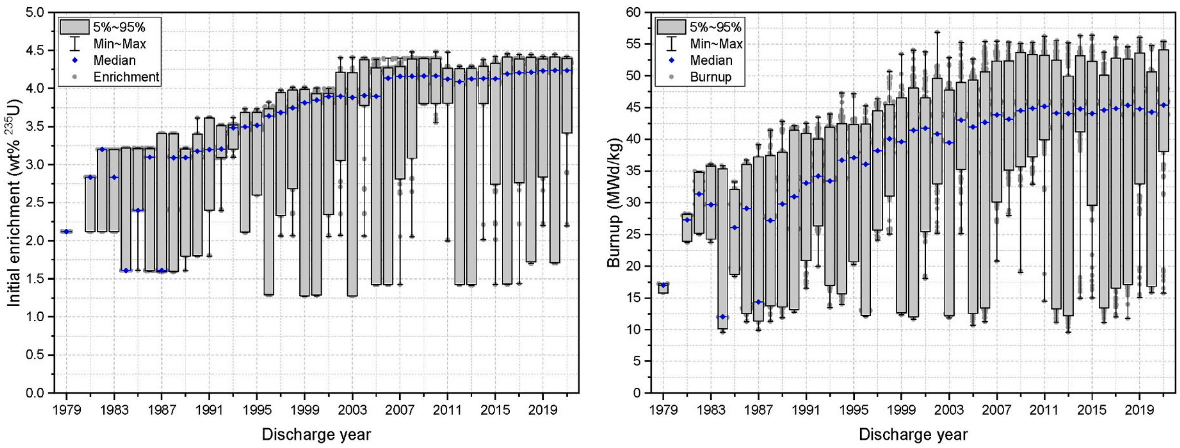


Fig. 2. The number of the PWR spent fuel assemblies for given burnups in South Korea.



(a) Uranium enrichment (b) discharge burnup

Fig. 3. Changes of the statistical values of uranium enrichment and discharge burnup in South Korea.

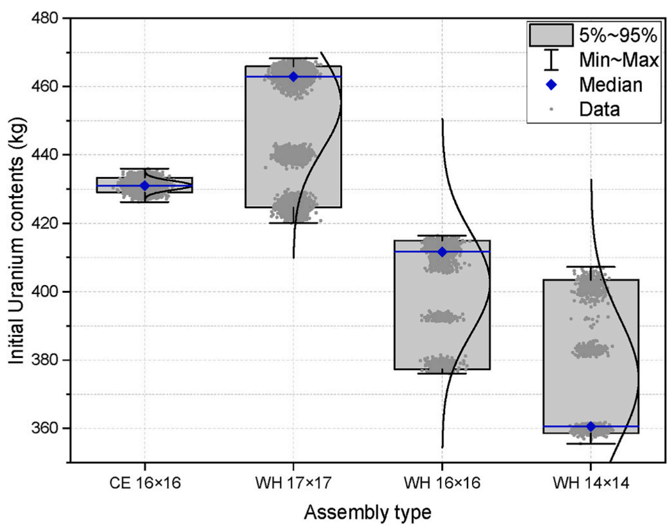


Fig. 4. Statistical values of uranium loadings of PWR spent fuel assemblies in South Korea.

Still, the PLUS7 one has larger uranium loading due to higher pellet density than the CE standard and Guardian ones. The uranium loading of the PLUS7 assembly is about 430.8 kg, which is smaller than that of RFA. Of the PWR spent fuel assemblies, WH 17×17 and CE 16x16 type ones occupy 42.6 % and 43.5 %, respectively, while the WH 14×14 and WH 16×16 ones occupy 6.6 % and 7.3 %, respectively. Considering the status of the operating PWRs and nuclear power plant construction plan, the majority of the PWR spent fuel assemblies will still be the WH17×17 and CE 16×16 type ones.

After the commercial operation of Kori Unit 1, the amount of the discharged spent fuel assemblies has been rapidly increased up to the beginning of 2006 except for the period of 1990 to 1999. Fig. 1 shows the change in the annual number of PWR spent fuel assemblies in South Korea. Fig. 2 shows the number of spent fuel assemblies for given discharge burnup values. As shown in Fig. 2, most of the spent fuel assemblies have discharge burnups of 25~55 MWd/kg, and statistical analysis shows that the average and median values of the discharge burnup are 40 and 42 MWd/kg. In particular, the spent fuel assemblies having lower burnups than 45 MWd/kg occupy ~70 % of all the spent fuel assemblies, while the ones having higher burnup than 55 MWd/kg occupy just ~0.35 %.

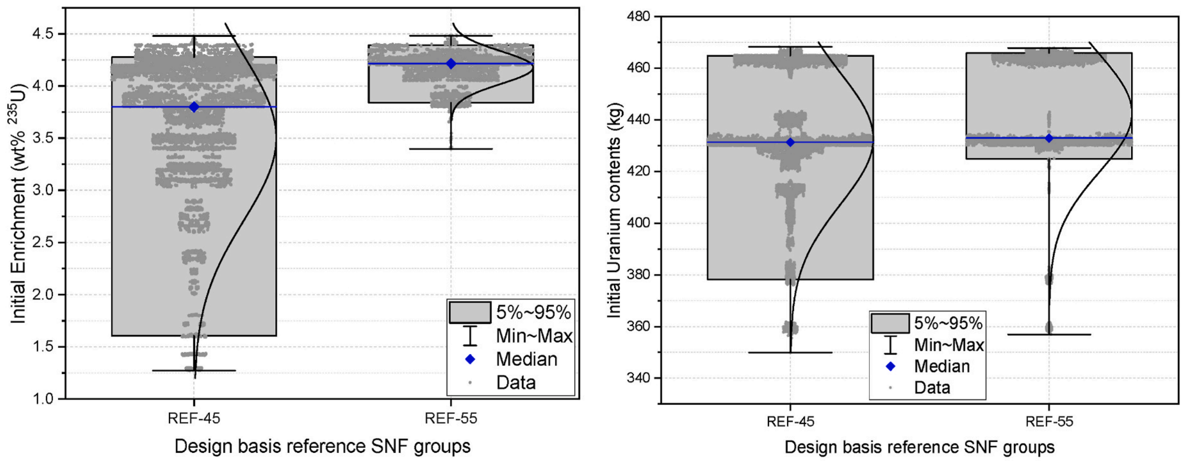
Fig. 3(a) and (b) compare the changes of the initial uranium enrichments and discharge burnups with respect to the discharged year. As expected, the median values of discharge burnup and initial uranium enrichment overall increase over time because South Korea has pursued long-cycle operations to improve the economy.

Next, the statistics of the uranium loadings are analyzed. Fig. 4 shows the minimum, maximum, median, upper 5 %, and lower 5 % values of the uranium loadings for the CE 16×16, WH 17×17, WH 16×16, and WH 14×14 type fuel assemblies. The median values of uranium loadings are 431, 463, 412, and 360 kg for the CE 16×16, WH 17×17, WH 16×16, and WH 14×14 type fuel assemblies, respectively.

3.2. Determination of reference spent fuel assemblies

In this section, the reference spent fuel assemblies are determined considering the statistical analysis of the spent fuel assemblies given in Sec. 3.1. Additionally, a disposal scenario in which 70 % of the underground disposal system is occupied with low burnup spent fuel assemblies and the remaining 30 % with high burnup ones. As shown in Fig. 2, 70 % of the discharge burnups correspond to 44.9 MWd/kg, so the spent fuel assemblies are classified into two groups based on 45 MWd/kg. The low burnup group includes the spent fuel assemblies having lower burnup than 45 MWd/kg, and the higher burnup group includes the ones having higher burnup than 45 MWd/kg. The representative burnup of the low burnup group assemblies is set to 45 MWd/kg, but the one of the high burnup group is to 55 MWd/kg. The statistical analysis given in Sec. 3.1 showed that only 73 assemblies of 20,970 spent fuel assemblies have burnups higher than 55 MWd/kg, and the maximum burnup is 56.9 MWd/kg. Based on this classification, 14,826 and 6144 spent fuel assemblies are included in the low and high burnup groups, respectively. The initial uranium enrichment and loading are also important parameters. Fig. 5(a) and (b) show the statistics of the initial uranium enrichments and loadings, respectively, for low and high burnup groups. For the low burnup group, the median and maximum values of initial uranium enrichment are 3.8 % and 4.48 %, respectively, while they are 4.2 % and 4.48 % for the high burnup group.

However, the initial uranium enrichment has been increased recently for long-cycle operations, and as of December 2021, the number of spent fuel assemblies having higher uranium enrichments than 4.6 % is about 2347. So, considering conservatism, the initial uranium enrichments of the reference fuel assemblies are set to be 4.5 and 4.65 % for low and high burnup groups, respectively. As shown in Sec. 3.2, the initial uranium loading depends on the type of fuel assemblies so it is desirable to set up the reference fuel assemblies separately for the WH



(a) Uranium enrichment (b) Uranium loading

Fig. 5. Statistical values of uranium enrichments and loadings for low and high burnup groups.

Table 2
Specification of the reference spent fuel assemblies.

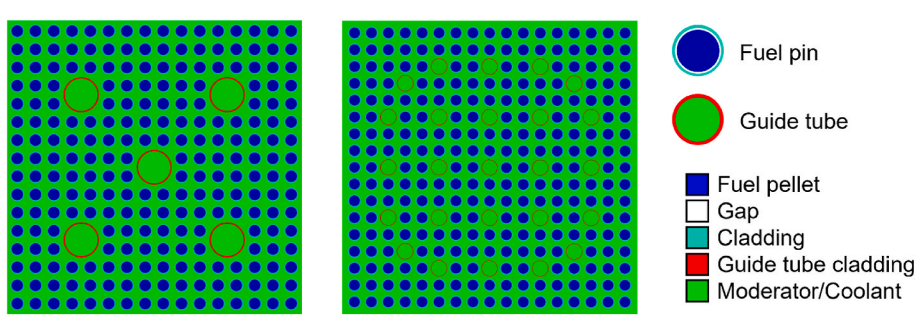
Name	Low burnup group		High burnup group	
	REF-PLUS7-LB	REF-ACE7-LB	REF-PLUS7-HB	REF-ACE7-HB
Lattice type	16×16	17×17	16×16	17×17
Burnup (MWd/kg)	45		55	
Initial uranium enrichment (%)	4.5		4.65	
Initial uranium loading (kg)	436	468	436	468
Specific power (MW/MTU)	40			
Depletion interval (EFPDs)	1,125		1,375	

17×17 and CE 16×16 type assemblies. Based on the statistical analysis, the low and high burnup groups for CE 16×16 type assemblies have maximum uranium loadings of 435,984 g and 435,581 g, respectively, while they have 468,259 g and 467,770 g for WH 17×17 type assemblies. As a result, the initial uranium loadings for CE 16×16 and WH 17×17 type assemblies are set to be 436 kg and 468 kg, respectively, independently of the burnup groups. Finally, the evaluation of the source term requires irradiation history. Usually, the fuel assemblies are discharged after a repeated cycle of depletion and cooling. However, it is not practical to consider all the detailed irradiation histories in the source term evaluation. In this work, a single depletion followed by a single cooling time period is considered for simplification. The specific power for the depletion is set to be 40 MW/MTU, which is slightly higher than the typical value of the PWR fuel assemblies, which leads to 1,125 EFPDs and 1,375 EFPDs depletion times, for the low and high burnup groups, respectively. Table 2 summarizes the specification of the reference spent fuel assemblies.

4. Results of the source term evaluations

4.1. Modeling of reference spent fuel assemblies

In this section, the burnup-dependent one-group cross sections of four reference spent nuclear fuels selected in Sec. 3.2 for BESNA were produced. In addition, ORIGEN, where the burnup-dependent one-group cross sections were evaluated with TRITON [10], was used to compare the source terms results from BESNA. Nuclear data used in TRITON calculations are neutron cross sections library 252 groups and gamma cross section library 47 groups of the ENDF/B-VII.r1. In particular, we divided the fuel pellet region into three sub-regions to consider the self-shielding effect in the TRITON depletion calculations for the generation of the ORIGEN cross section libraries. The burnup-dependent effective one-group cross sections for BESNA were generated through the MCNP6 depletion calculations for the fuel assembly models, where the fuel pellet regions were divided into three sub-regions. We used 100 inactive cycles followed by 5,100 active cycles with 10,000 particle histories per cycle. The depletion calculations of MCNP6 were performed using the following burnup steps: 0.5, 1.0, 1.5, 2.0, 2.5, 3.0, 5.0,



(a) PLUS7 (b) ACE7

Fig. 6. Configuration of the reference spent nuclear fuel assemblies.

Table 3
Specification of the reference spent fuel assemblies for effective one-group cross sections for depletion calculations.

Parameter	PLUS7	ACE7
Lattice type	16 × 16	17 × 17
Number of fuel pins	236	264
Active fuel height (cm)	381	365.76
Number of guide tubes	5	25
Fuel assembly height (cm)	452.8	406.3
Fuel assembly pitch (cm)	20.7772	21.5040
Fuel pin height (cm)	409.4	388.1
Fuel pin pitch (cm]	1.285	1.260
Fuel pellet diameter (cm]	0.8192	0.8192
Cladding material	ZIRLO	ZIRLO
Cladding inner diameter (cm]	0.8357	0.8357
Cladding outer diameter (cm)	0.95	0.95
Guide tube material	ZIRLO	ZIRLO
Guide tube inner diameter (cm)	2.286	1.008
Guide tube inner diameter (cm)	2.4892	1.224

7.5, 10.0, 15.0, 20.0, 25.0, 30.0, 35.0, 40.0, 45.0, 50.0, 55.0, and 60.0 MWd/kg. The MCNP6 calculations were performed with ENDF/B-VII.r1 point-wise cross section library.

Fig. 6 (a) and (b) show the configuration of reference spent nuclear fuel assemblies. Table 3 shows the design specification of the reference spent nuclear fuel assemblies.

4.2. Results of source term evaluation

In this subsection, the source terms evaluated using BESNA for the reference spent fuel assemblies were presented and compared with those obtained using ORIGEN. The source terms considered in this work include decay heat, radioactivities, neutron and photon emission intensities, and neutron and photon emission spectra. First, the decay heats from the reference spent fuel assemblies over 1,000,000 years after discharge from the reactor are presented in Fig. 7.

Fig. 7 also shows the discrepancies (%) of the decay heat evaluated using BESNA compared with the one evaluated using ORIGEN. The discrepancies are defined as the absolute relative percent discrepancies between the BESNA and ORIGEN calculation results. The ORIGEN results were used as the reference values. From Fig. 7, it was shown that the decay heat evaluated using BESNA give very good agreements with those using ORIGEN over 1,000,000 years of cooling time. The discrepancies in decay heats are higher for the high burnup reference spent fuel assemblies than for the low burnup ones, but the discrepancies are less than 7 %. In particular, BESNA underestimates the decay heats relative to ORIGEN and the discrepancies are relatively large over 200~100,000 years of cooling time, while the discrepancies are quite small over the other cooling time periods.

Next, the radioactivities evaluated using BESNA for the reference spent fuel assemblies are presented in Fig. 8 and their discrepancies to the ORIGEN results are also compared in Fig. 8. Fig. 8 shows that the BESNA results of radioactivity also give good agreements with the ORIGEN ones and the discrepancies are similar levels to the ones for the

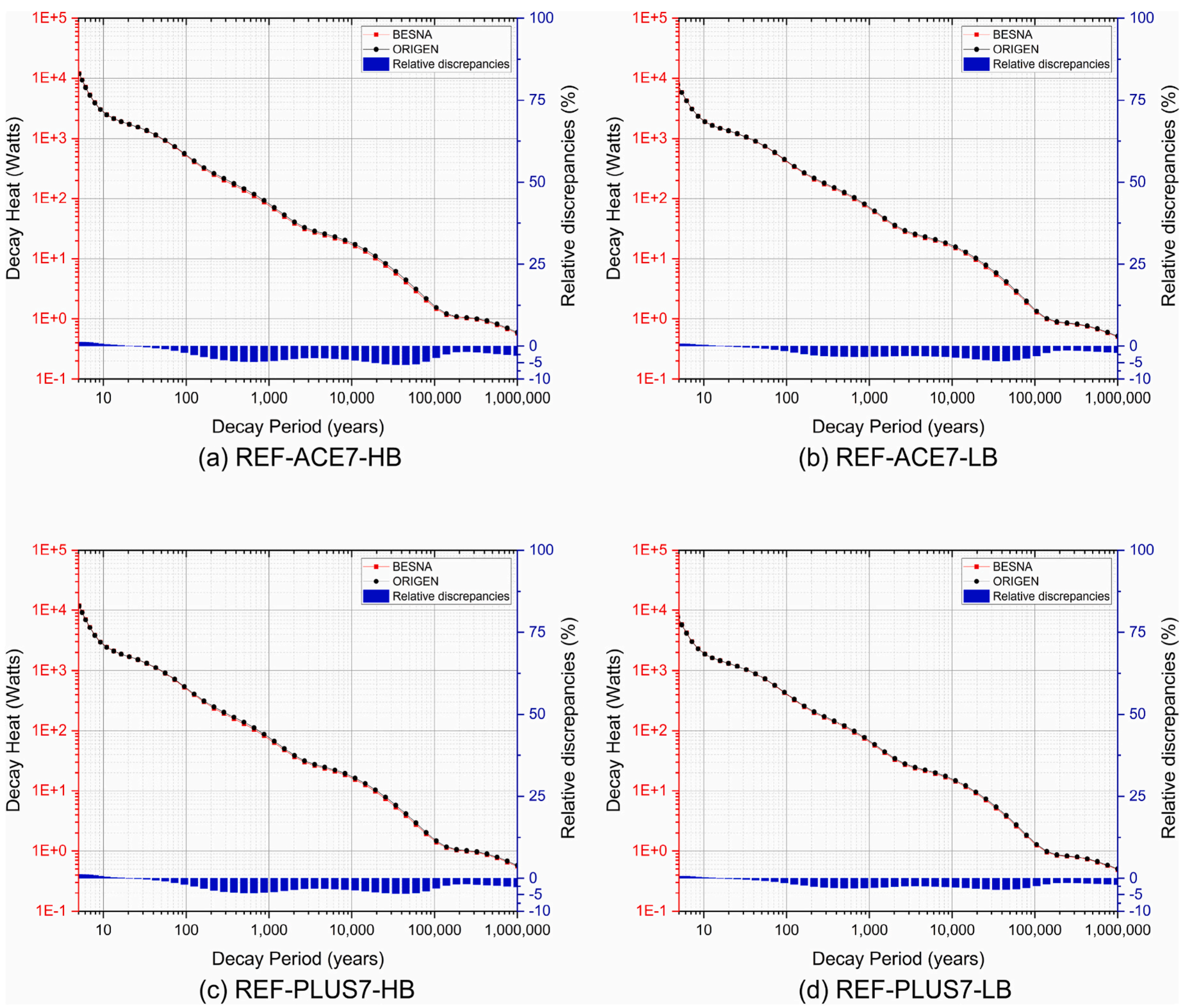


Fig. 7. Decay heat of the reference spent nuclear fuels and relative discrepancies (%) between BESNA and ORIGEN.

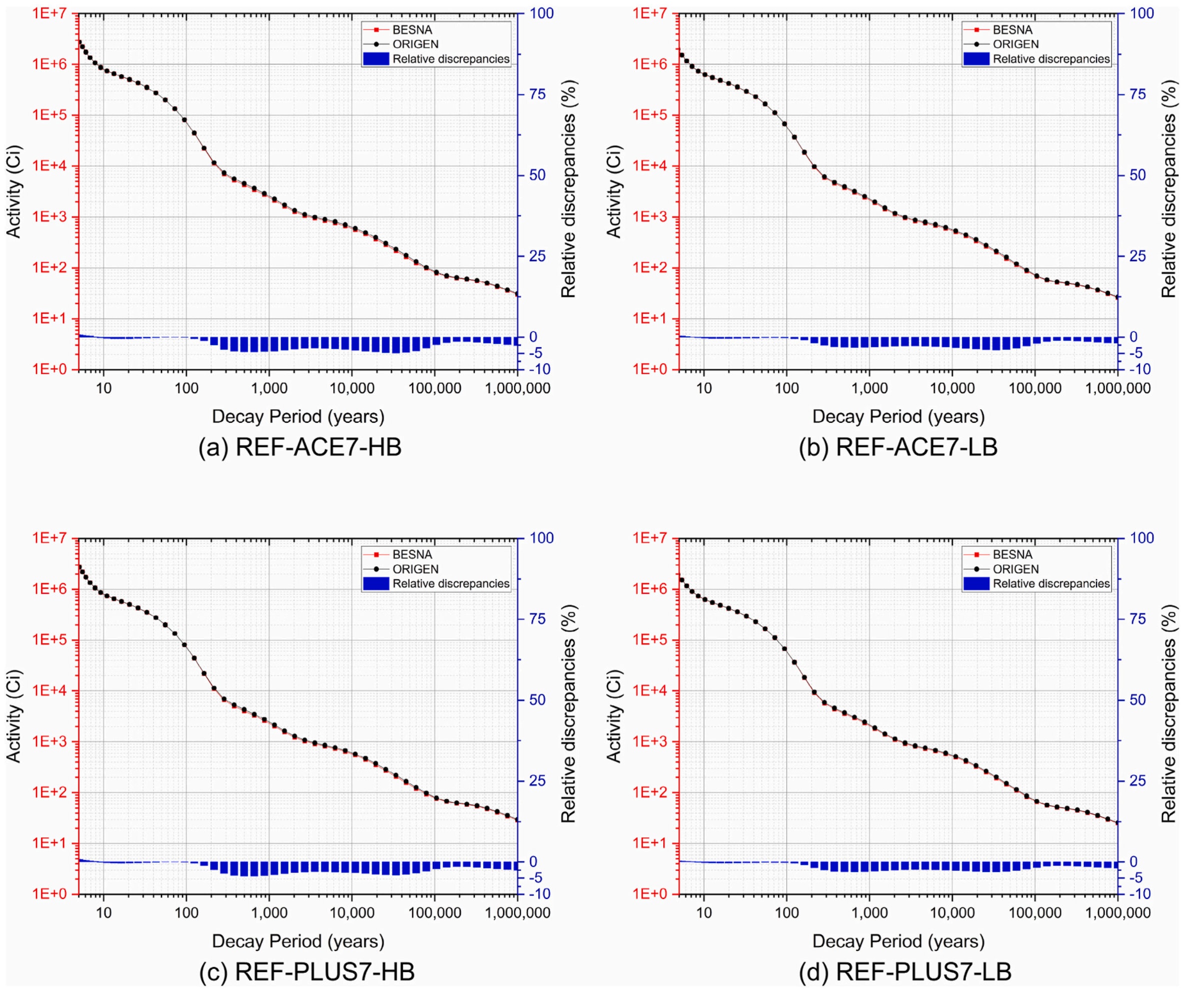


Fig. 8. Radioactivity of reference spent nuclear fuels and relative discrepancies (%) between BESNA and ORIGEN.

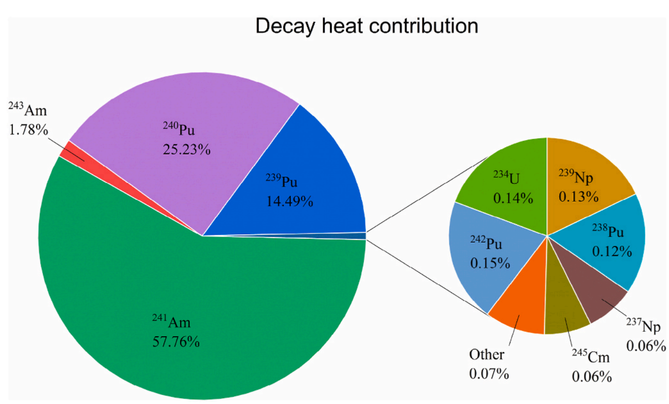


Fig. 9. Contributions of ten large contributor nuclides to the total decay heat at 1000 years cooling time.

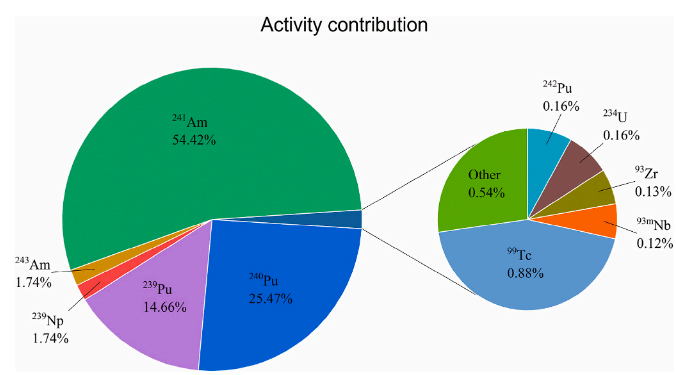
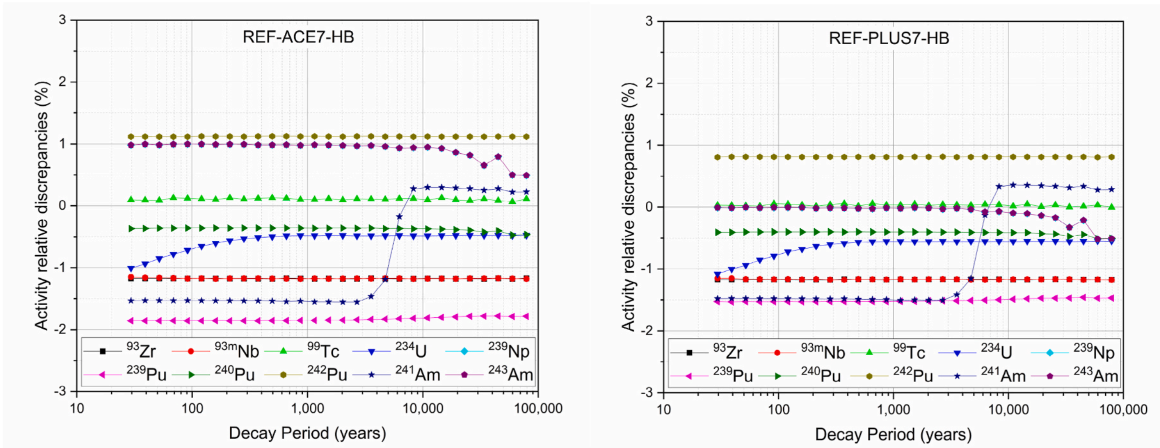


Fig. 10. Contributions of ten large contributor nuclides to the total radioactivity at 1000 years cooling time.

decay heats.

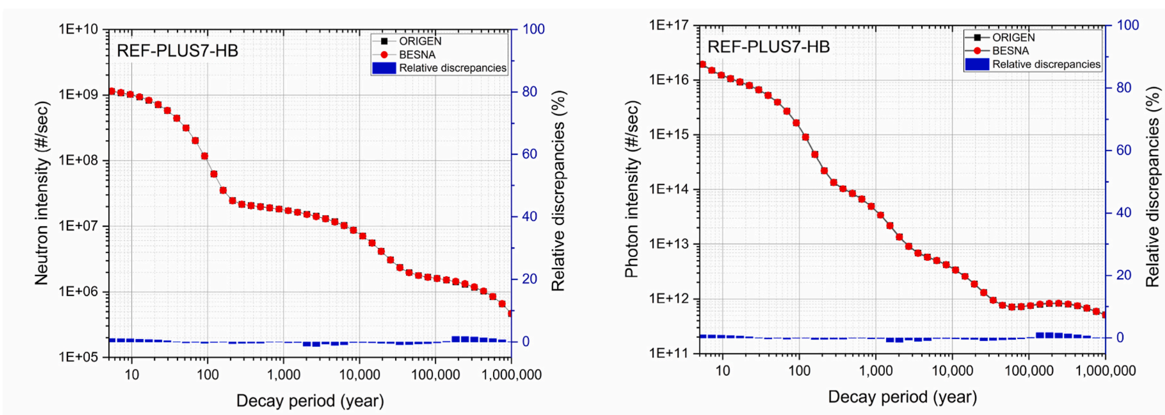
Next, the contributions from the major contributor nuclides to total decay heat and radioactivity are analyzed using BESNA at 1000 years of cooling time, which is considered as the duration time of corrosion resistance for the disposal canister developed by KAERI (Korea Atomic Energy Research Institute) [11]. These analyses are done for the PLUS7 high burnup reference fuel assembly. Fig. 9 shows the contributions (%) from the top ten contributor nuclides to the total decay heat at 1,000 years after discharge. As shown in this figure, the largest contribution of 57.8 % comes from ^{241}Am, and the following largest contributions of 25.2 % and 14.5 % from ^{240}Pu and ^{239}Pu, respectively. The other large contributor nuclides are ^{243}Am, ^{242}Pu, ^{234}U, ^{239}Np, ^{238}Pu, ^{237}Np, and ^{245}Cm.

Fig. 10 shows the contributions (%) from the top ten contributor nuclides to the total radioactivity at 1,000 years after discharge. As shown in this figure, the largest contribution of 54.4 % comes from ^{241}Am, and the following largest contributions of 25.5 % and 14.7 % from ^{240}Pu and ^{239}Pu, respectively. The other large contributor nuclides are ^{243}Am, ^{239}Np, ^{99}Tc, ^{242}Pu, ^{234}U, ^{93}Zr, and ^{93m}Nb.



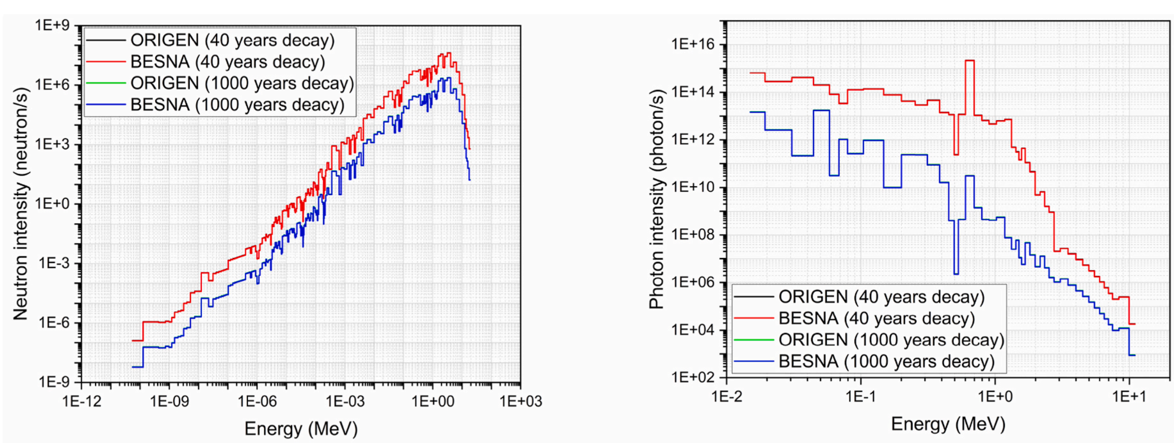
(a) REF-ACE7-HB (b) REF-PLUS7-HB

Fig. 11. Relative discrepancies (%) in radioactivities between BESNA and ORIGEN for top ten contributor nuclides for high burnup reference spent fuels.



(a) Neutron (b) Photon

Fig. 12. Total neutron and photon intensities and discrepancies (%) between BESNA and ORIGEN for the reference high burnup spent fuel assembly of PLUS7.



(a) Neutron (b) Photon

Fig. 13. Comparison of the neutron and photon emission spectra for the PLUS7 high burnup reference spent fuel assembly.

Fig. 11(a) and (b) compare the discrepancies in radioactivities of ten nuclides giving large contributions between BESNA and ORIGEN for the high burnup reference spent fuel assemblies (i.e., REF-ACE7-HB and REF-PLUS7-HB, respectively) over 100,000 years after discharge. From these figures, it is shown that BESNA accurately estimates the radioactivities in comparison with ORIGEN and the maximum discrepancy is less than 2 %. Actually, the small discrepancies in nuclide-wise radioactivities mean the small discrepancies in the inventories of these nuclides.

Next, the neutron and photon emission intensities from the high burnup reference spent fuel assemblies evaluated using BESNA of PLUS7 type lattice (i.e., REF-PLUS7-HB) are presented in Fig. 12(a) and (b), respectively. These results show that the discrepancies in neutron and photon emission intensities between BESNA and ORIGEN are quite small, and the maximum discrepancy is less than 1.74 % both for neutron and photon.

Finally, the spectra of the neutron and photon emission from BESNA are compared with those from ORIGEN in Fig. 13(a) and (b), respectively, for the PLUS7 high burnup spent fuel assembly. These neutron and photon emission spectra are evaluated at 40 and 1,000 years after discharge in 252 neutron energy group and 47 photon group structures. From these figures, it is shown that the emission spectra from BESNA, both for neutron and photon, give quite good agreements with the ones from ORIGEN both for 40 and 1,000 years of cooling time. Also, these figures show that there are no considerable changes in neutron emission spectra at 40 and 1,000 years of cooling time, but there are considerable differences in low energy ranges in photon emission spectra at between 40 and 1,000 years of cooling time.

5. Summary and conclusion

This work presented the source term evaluation results using the inhouse code BESNA developed by the authors for reference PWR spent fuels in South Korea. Also, the source terms evaluated using BESNA were compared with those obtained using ORIGEN for validation purposes. The burnup-dependent effective one-group cross sections were generated tallying the reaction rates and neutron fluxes during the MCNP6 depletion calculations considering the detailed fuel assembly dimensions of the reference spent fuels, while the effective one-group cross sections for ORIGEN were produced using TRITON in SCALE6.2.

The reference spent fuels that will be used in the safety analysis of the future deep geological disposal system of South Korea were determined through a statistical analysis of all the accumulated spent fuels discharged from all the PWRs in South Korea. As the results of the statistical analysis, two reference fuels for each type of 16×16 and 17×17 fuel assembly lattice structures were determined considering low and high burnups (i.e., 45 MWd/kg and 55 MWd/kg for low and high burnups, respectively). The high and low burnup reference spent fuels have initial uranium enrichments of 4.5 and 4.65 %, respectively. The 17×17 ACE7 reference spent fuel has slightly higher initial uranium loading by 32 kg than the 16×16 PLUS7 one.

The comparison of the source terms from BESNA and ORIGEN showed that the total decay heat and radioactivity from BESNA for all the reference spent fuels give agreements with ORIGEN within a 7 % discrepancy over 1,000,000 years after discharge. For detailed validation, the top ten contributor nuclides to the total radioactivity were analyzed, and their nuclide-wise radioactivities from BESNA over 100,000 years after discharge were compared with those from ORIGEN. From the comparison of these nuclide-wise radioactivities, it was shown that BESNA gives very good agreements with ORIGEN within a 2 % discrepancy. There were also very good agreements within a 1.74 % discrepancy in the total neutron and photon emission rates from the high burnup reference spent fuel and the spectra of neutron and photon emission rates also showed good agreements between BESNA and ORIGEN.

CRediT authorship contribution statement

Kyu Jung Choi: Formal analysis, Investigation, Methodology, Writing – original draft. **Duy Long Ta:** Methodology. **Shin Sung Oh:** Data curation. **Ser Gi Hong:** Conceptualization, Funding acquisition, Investigation, Methodology, Supervision, Writing – original draft, Writing – review & editing.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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