THE DEVELOPMENT OF A SPENT NUCLEAR FUEL MANAGEMENT SYSTEM IN KOREA

D. K. Cho, D. H. Kook, J. Y. Lee, H. J. Choi, J. W. Choi
Korea Atomic Energy Research Institute
150, Deokjin-dong, Yuseong-gu, Daejeon-city, Republic of Korea

ABSTRACT

As temporary storage pool at each reactor site is expected to be saturated by 2016, the South Korean government and nuclear industry are trying to find a long-term management strategy for spent fuels in Korea. To establish this strategy, it is necessary to project the amount of spent fuel, and to analyze the historical data for spent fuel characteristics such as inventory, design specifications, irradiation history, and storage history. Therefore, a spent fuel management system was developed in this study to make this cumbersome work simple. The spent fuel management system is comprised of three functional modules: a spent fuel historical data analysis module, a spent fuel arising projection module, and a radwaste characterization module from an advanced fuel cycle. This paper explains the major functions and features of theses modules.

INTRODUCTION

There are now twenty commercial nuclear power reactors operating in the Republic of Korea: four CANDU reactors at the Wolsong site and sixteen PWR reactors at the Kori, Younggwang, and Ulchin sites. The total amount of spent fuel(SF) generated by the end of 2008 was found to be about 10,083 metric tons of uranium (MTU). The annual production rate is about 700 MTU[1]. The SF from PWRs is stored at a reactor wet storage pool. The SF from CANDU reactors is stored at a reactor wet storage pool and dry storage silos. According to the Republic of Korea's national plan for the supply of electricity, 'The 4th Basic Plan for Electric Supply,' 12 additional PWRs will be built by 2022. By achieving this plan, the installation capacity and electricity share of nuclear power plants should reach to 32.6% and 47.9%, respectively.

As storage facilities for PWR and CANDU fuel are expected to be saturated by 2016, the South Korean government and nuclear industry are trying to find a long-term management strategy of SF in the Republic of Korea. To establish this strategy, it is necessary to project the amount of SF, and to analyze the historical data for SF characteristics such as inventory, design specifications, irradiation history, and storage history. Therefore, an SF management system was developed in this study to make this cumbersome work simple.

OVERALL STRUCTURE OF THE MANAGEMENT SYSTEM

Currently, the SF management system is comprised of three functional modules: a SF historical data analysis module(SF-HDAM), an SF arising projection module(SF-APM), and an advanced fuel cycle(AFC) high-level waste(HLW) characterization module(AFC-HLW-CM).

A major function of the SF-HDAM is to give user information on characteristics of SF currently stored at wet and dry storage facilities. It gives the distribution of initial ^{235}U enrichment, discharge burnup, and aging for SF. It also analyzes the correlation of ^{235}U enrichment vs. discharge burnup. An evaluation of the inventory of SF corresponding to the user-defined initial ^{235}U enrichment, discharge burnup, and cooling time is one of the features of this module. The SF-APM can project the SF inventory as a function of upcoming time with three models which will be discussed later. The estimated SF inventory can be used as background data to establish a national strategy for safe management of SF. The AFC-HLW-CM automatically characterizes source terms of radwastes from the AFC with pyroprocessing which generates metal fuel for the sodium fast reactor(SFR), off-gas wastes, and molten salt wastes.

A variety of data regarding SF characteristics are used to analyze the status of SF and to characterize radwastes from AFC with pyroprocessing, as shown in Figure 1.

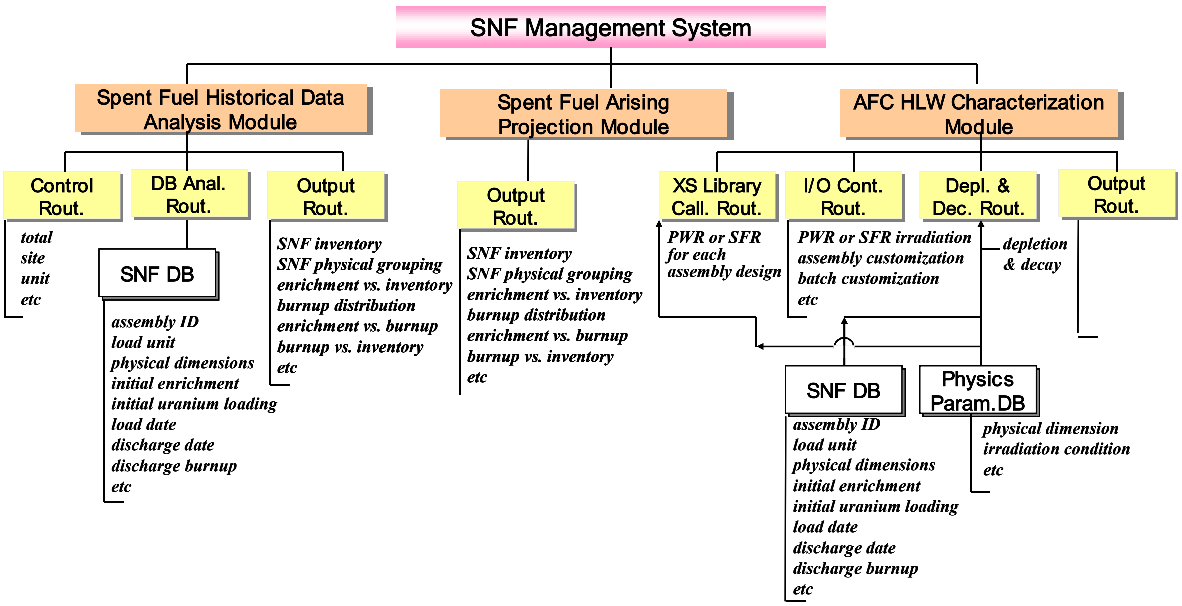

Fig. 1 Overall structure of the SF management system

FUNCTIONS OF EACH MODULE

Historical Data Analysis Module

Not only SF characteristics should be investigated and analyzed for the establishment of a national strategy, but also it should be investigated and analyzed for the design of an interim storage facility and transport cask. Usually, the investigation and analysis on physical parameters, irradiation history, and storage history for SF are carried out prior to the actual design process. To facilitate this work, the SF-HDAM is being developed.

The database for the SF-HDAM established up to now is equipped with physical characteristics such as an assembly design, an array, initial ^{235}U enrichment, initial uranium weight, and structural materials. It also has irradiation characteristics such as a loading date, discharge burnup, a discharge date, and a residence period in a core, and storage characteristics such as the cooling time at a storage facility. This data is recorded for each fuel assembly identification(ID). Each item listed previously can be analyzed for a reactor unit-base or site-base. As was mentioned previously, its major functions are to provide the distribution of initial ^{235}U enrichment, discharge burnup, and aging for SF; to analyze the correlation of ^{235}U enrichment vs. discharge burnup; and to evaluate inventory of SF corresponding to the user-defined initial ^{235}U enrichment, discharge burnup, and cooling time.

To obtain the desired data by using this module, the user defines the upper and lower boundary of initial ^{235}U enrichment, discharge burnup, and cooling time in search options. The user can also define reactor units from which SF is discharged or use search options to determine which sites SF is stored in. All results are displayed in the unit of mass and number of assemblies. An annually counted value, an accumulated value, and an occupying ratio relative to the user-defined reference value are calculated for a desired value as a function of the calendar year. This database management software was developed using Visual Basic in a Windows XP environment.

Figure 2 shows an example of the analyzed result for the number of fuel assemblies and occupying percentile ratio generated from all Kori units. Figure 3 shows the historical trend of average and maximum discharge burnup. The standard deviation for the average burnup is also calculated when a requested job accompanies statistical analysis. Figure 4 represents SF inventory corresponding to each discharge burnup. This value can also be obtained for each reactor unit and site. Figure 5 shows the correlation between the inventory, discharge burnup, and initial ^{235}U enrichment. All data is displayed in a graphical interface and written in a text file.

The quantity of spent fuel with a specific design is also easily obtained through this module. For example, it was found that the number of SF with discharge burnup exceeding 45 GWD/MTU was 3,522, 2,210, and 2,816 PWR assemblies for the Kori, Ulchin, and Younggwang sites, respectively.

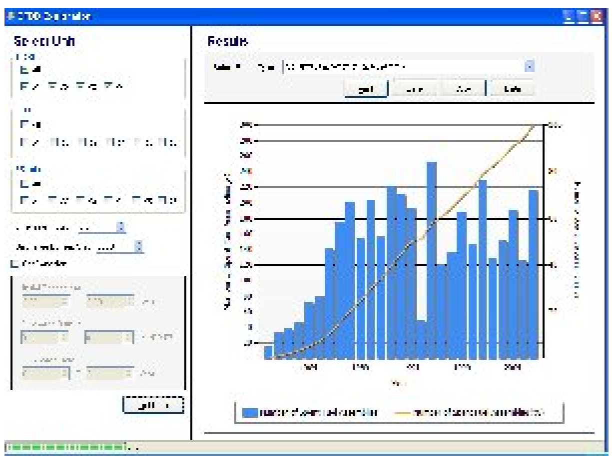

Fig. 2 Total number of fuel assemblies

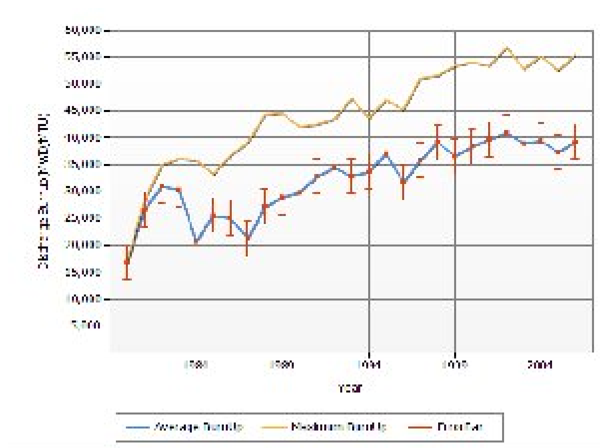

Fig. 3 Discharge burnup distribution

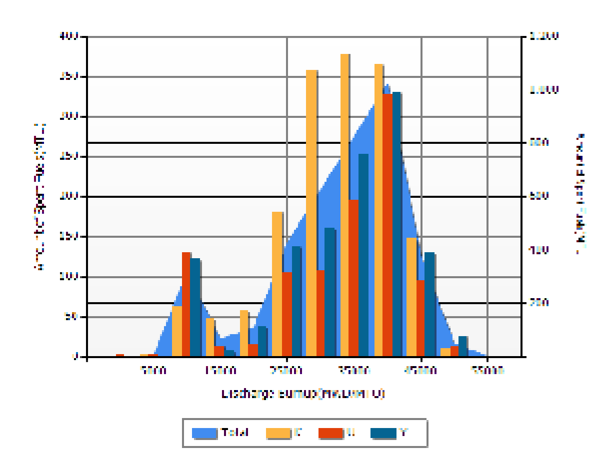

Fig. 4 SF inventory as a function of burnup.

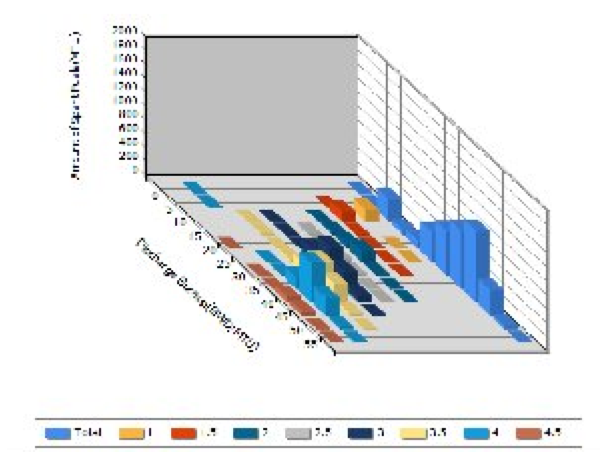

Fig 5. Inventory vs. burnup vs. ^{235}U enrichment

Spent Fuel Arising Projection Module

Currently, the SF-APM can estimate the SF inventory for upcoming time reflecting national nuclear scenario. The projection for occupying ratio for a specific assembly design, ^{235}U enrichment, discharge burnup, cooling time, etc, will be added later.

Three models were introduced to project an SF arising. Following Eq. (1), (2), and (3) represent, respectively, the reactor cycle model, the discharge burnup model, and the annual average discharge model. In all equations, C_f^k, C_c^k, and L_c^k represent the amount of SF to be accumulated, the amount of SF accumulated up to previous time, and the total amount of SF in a core for the reactor unit *k,* respectively. The remaining reactor operation period is declared by *n* on the base year.

Model 1: Cycle length model

$$C_f^k = C_c^k + \sum_{i=1}^{n} \frac{L_c^k}{N_b} \int_t^{t+0.99} \delta \left[\sin\left(\frac{\pi}{M_c^k / 12}(t - t_b) \right) \right] dt + L_c^k \quad \text{--- (1)}$$

where N_b = the number of batchs for unit *k*, t = the calendar year for remaining operation period, M_c^k = the cycle length [month], and t_b = the base year.

Model 2: Discharge burnup model

$$C_f^k = C_c^k + 365 \sum_{i=1}^{n} P^k \frac{L^k}{\varepsilon^k} \frac{1}{B^k} + L_c^k \quad \text{-- (2)}$$

where P^k = the electric power output[MWe] for unit *k*, L^k = the capacity factor for unit *k*, ε^k = thermal efficiency for unit *k*, B^k = discharged burnup[MWD/MTU].

Model 3: Annual average discharge Model

$$C_f^k = C_c^k + \sum_{i=1}^{n} D_{avg}^k + L_c^k \quad \text{-- (3)}$$

where, D_{avg}^k = the annual average discharge rate[MTU/yr] for unit *k*.

Model 1 and 2 can analyze the decreasing amount of SF resulting from a longer cycle operation strategy. Model 1 can assess the amount of SF caused by the cycle length extension with the increasing initial ^{235}U enrichment. Model 2 can simulate a variation from the enhanced discharge burnup.

The SP-APM was developed in Visual Basic in a Windows XP environment. This includes three models explained previously and can deal with PWRs, CANDUs, and the HANARO research reactor. It is also equipped with a graphical user interface.

At first, to evaluate the validity of this module, the reference data was calculated to compare this value with other values that might be available as a result of other reactor operation strategies. The estimation of the spent fuel arising was then done by applying different parameters only for PWR reactors. In the reference case which assumes that the current commercial reactors would be operated by the current licensing period without renewal, it revealed that the amount of spent fuel would be approximately 37,500 tons; ~22,500 tons from PWR reactors and ~15,000 tons from CANDU reactors. For the case where the cycle length for the Westinghouse-type nuclear power plants and Korea Standard nuclear power plants is extended to 18 months, a reduction of 400 tons of spent fuels was expected. The reference case was simulated by applying a 17-month-long cycle for these reactors. For the case where the discharge burnup of those nuclear power plants is extended to 55GWD/MTU, a reduction of 700 tons of spent fuel was expected. The reference case was simulated by applying 50 GWD/MTU for those reactors. If the life time of those reactors is extended to 50 and 60 years, a reduction of 3,400 and 6,700 tons of spent fuels is expected, respectively.

Figure 6 shows an example of the input for the cycle length model, and Figure 7 represents and example of the projected inventory of SF from the cycle length model for each reactor unit.

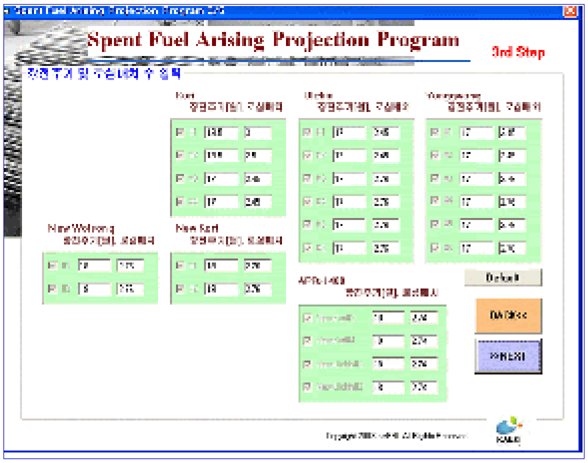

Fig. 6 Example of the input window

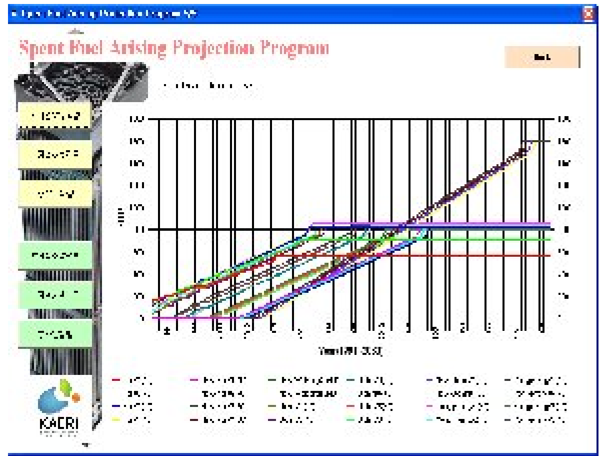

Fig. 7 Example of the estimated inventory for each unit

AFC HLW Characterization Module

A national policy for the long-term management of SF has not been decided yet in Korea. Only in the R&D area, the disposal of an intact SF assembly was considered, and a corresponding disposal system was developed in 2006. The R&D study to develop an AFC with pyroporcessing, however, was launched in 2007. A disposal system to accommodate a variety of radwastes from an AFC has been developed since 2007, correspondingly.

As shown in Figure 8, PWR fuel is recycled as SFR fuel after a pyroprocess. In this process, a variety of radwastes are generated[2]. Because the material balance of a pyroprocess changes as relevant technology develops and the fuel assemblies are supplied into the process as a batch, it was hard and cumbersome to calculate the source terms of many kinds of radwastes from the pyroprocess by using a conventional calculation method. Therefore, the AFC-HLW CM has been developed for automatic source term characterization.

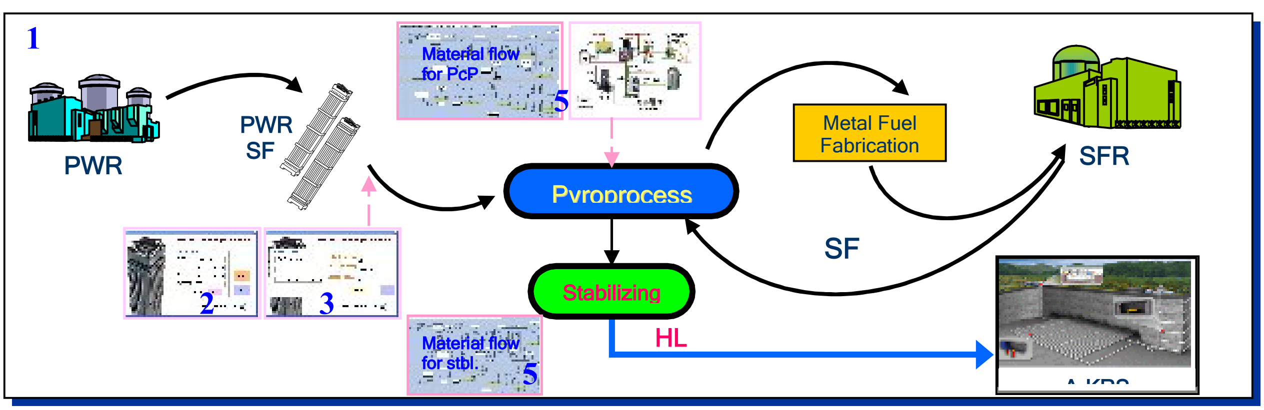

Fig. 8 Schematic diagram for an AFC including pyroprocessing and and SFR

This module can automatically characterize the source term of radwastes from a pyroprocess. It can calculate the source term for radwastes from single-assembly-processing and batch-processing. It uses a built-in SF database to make a batch and to reflect historical irradiation and the decay history of SF generated for commercial reactors. The user can screen and combine the SF to make batch-processing by setting the search options. After combining the batch for the pyroprocess, it automatically characterizes the source terms of radwastes from the AFC by applying the material flow of the pyroprocess. A user-friendly graphic interface is one of the remarkable features of this program. It has several functional modules for the screening and combination of SF, depletion calculation, decay calculation, mixture composition calculation at the beginning of the pyroprocess, etc. It uses ORIGEN-S[3] in a SCALE system[4] for the depletion and decay calculation which solves Eq. (4).

$$\frac{dN_i}{dt} = \sum_j \delta_{ij}\lambda_j N_j + \sum_k f_{ik}\sigma_k N_k - (\lambda_i + \sigma_i \Phi)N_i \text{ -- (4)}$$

where, σ_i = the absorption cross section of nuclide *i*,

δ_{ij} = the fraction of radioactive decay from nuclide *j* to *i*,

f_{ik} = the fraction of neutron absorption by nuclide *k* and transmuted to isotope *i*, and other terms have conventional meanings.

For a depletion calculation of fuel, the cross-sections needed to solve Eq. (4) are called form a built-in cross-section library which was pre-generated for a specific assembly design in Korea. The neutron flux value is the calculated value in ORIGEN-S on the basis of specific power. For an activation calculation of structural components, the cross-sections needed to solve Eq. (4) are called from a built-in activation cross-section library which was pre-generated for each structural component of the specific assembly design. These activation libraries were prepared by using TRITON/KENO-VI module[5], developed in the Oak Ridge National Laboratory. The neutron flux for the activation of structural components is the value calculated by Eq. (5).

$$\overline{\Phi}_{struc.} = \omega \overline{\Phi}_{fuel} \text{ -- (5)}$$

where, ω = the ratio of average neutron flux of the structural component to that of fuel.

Source term characterization for the radwastes from an AFC, although many kinds of SF are pyrpprocessed at the same time, is a remarkable feature of the AFC-HLW CM. The source term characterization of radwastes, which is mixture of radwastes generated at *t=t1*, *t=t2*, and *t=t3*, is also a remarkable feature of the AFC-HLW CM. Figure 9 shows the main window of the AFC-HLW CM and an example of calculated results.

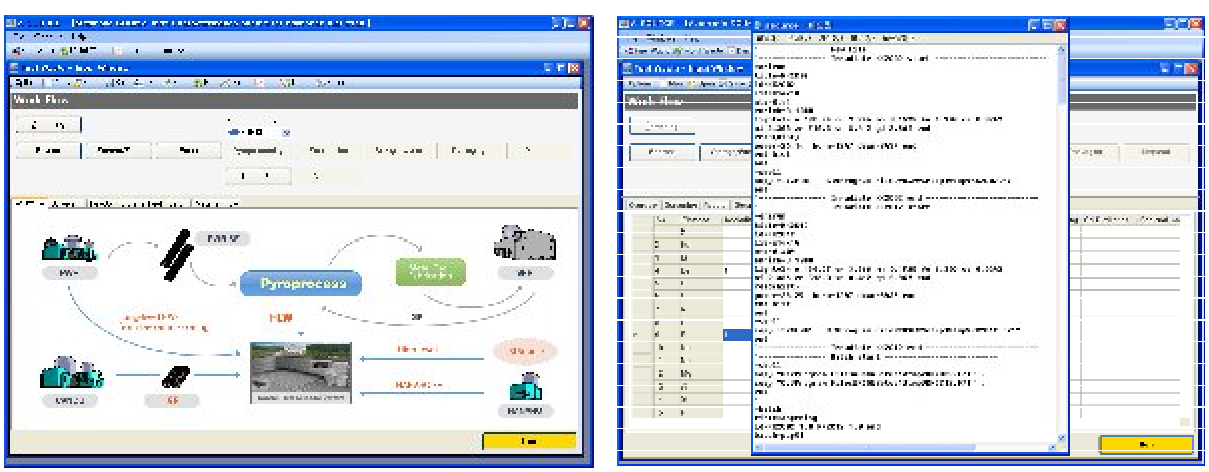

Fig. 9 Main window of the AFC-HLW CM and an example of calculated results

CONCLUSIONS

Since storage facilities for PWR and CANDU fuels are expected to be saturated by 2016, the South Korean government and nuclear industry are trying to find a long-term management strategy of spent fuel in Korea. To establish this strategy, it is necessary to project the amount of spent fuel, and to analyze the historical data for SF characteristics such as inventory, design specifications, irradiation history, and storage history. Therefore, an SF management system was developed in this study. The spent fuel management system is comprised of three functional modules, an SF-HDAM, SF-APM, and AFC-HLW-CM. A major function of the SF-HDAM is to give user information on the characteristics of SF currently stored at wet and dry storage facilities. The SF-APM can project the SF inventory as a function of upcoming time with three models. The AFC-HLW-CM can automatically characterize the source terms of radwastes from the AFC with pyroprocessing which generates metal fuel for the sodium fast reactor(SFR), off-gas wastes, and molten salt wastes.

The SF management system developed in this study will be a useful tool to support the establishment of a national policy for the safe, long-term management of spent fuel.

ACKNOWLOGEMENTS

The authors would like to acknowledge that this work was funded by the Ministry of Science and Technology.

REFERENCES

1. D. K. Cho, et al., Reference Spent Nuclear Fuel for Pyroprocessing Facility Design, J. of the Korean Radioactive Waste Society, **6(3),** (2008).
2. M. S. Lee, et al., Estimation of the Waste Mass from a Pyro-Process of Spent Nuclear Fuel, KAERI/TR-3577/2008, Korea Atomic Energy Research Institute, (2008).
3. I. C. Gauld, et al., ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, And Associated Radiation Source Terms, ORNL/TM-2005/39, Version 5.1, Vol. II, Book 1, Sect. F7, Oak Ridge National Laboratory, (2006).
4. SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, ORNL/TM-2005/39, Version 5.1, Vols. I, II, and III, Oak Ridge National Laboratory, (2006).
5. M. D. DeHart, TRITON: A Tow-Dimensional Transport And Depletion Module for Characterization of Spent Nuclear Fuel, ORNL/TM-2005/39, Version 6, Vol. I, Sect. T1, Oak Ridge National Laboratory, (2009)