Pre-conceptual Design of a Spent PWR Fuel Disposal Container

가압경수로형 사용후핵연료 처분용기의 예비 개념설계 평가

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Abstract

In this paper, sets of engineering analyses were conducted to renew the overall dimensions and configurations of a disposal container proposed as a prototype in the previous study. Such efforts and calculation results can provide new design variables such as the inner basket array type and thickness of the outer shell and the lid & bottom of a spent nuclear fuel disposal container. These efforts include radiation shielding and nuclear criticality analyses to check to see whether the dimensions of the container proposed from the mechanical structural analyses can provide a nuclear safety or not. According to the results of the structural analysis of a PWR disposal container by varying the diameter of the container insert, the Maximum Von Mises stress from the 102 cm container meets the safety factor of 2.0 for both extreme and normal load conditions. This container also satisfies the nuclear criticality and radiation safety limits. This decrease in the diameter results in a weight loss of a container by ~20 tons.

Key Words: disposal container, Von Mises stress, spent PWR fuel, safety factor, deep geological repository

요 약

본 연구에서는 사전연구로부터 사용후핵연료의 처분용기 원형모델로 제안된 처분용기의 전체 크기와 배열을 평가하기 위하여 일련의 공학적 분석을 수행하였다. 그러한 노력의 결과 용기 내부 저장통의 배열형태와 외곽쉘과 상하부뚜껑의 두께와 같은 새로운 설계변수를 도출하였다. 공학적 분석 작업에는 처분용기의 기계구조 해석 결과를 근거로 도출된 용기의 규격자료에 대한 방사선 안전성 측면에서의 타당성을 검토하기 위하여 방사선차폐 해석과 핵임계 해석 등이 수행되었다. 처분용기 내부 삽입체의 직경 변화에 따른 구조안정성 해석 결과에 따르면, 직경 102cm일 때 극한 외압조건은 물론 정상적인 외압조건 하에서도 최대 Von Mises 응력이 안전계수 2.0을 만족하는 것으로 나타났다. 이 경우에서도 핵임계 및 방사선차폐 해석 결과 안전기준치를 만족시키며, 무게는 20톤 가량 줄어드는 효과가 있는 것으로 나타났다.

중심단어 : 처분용기, Von Mises stress, 가압경수로형 사용후핵연료, 심지층처분장

I. Introduction

Korea started a national long-term R&D program for high-level waste (HLW) disposal technology development in 1997. The spent PWR and CANDU fuels are regarded as HLW for the time being because there is no further plan except for a spent fuel interim storage. The main purpose of this program was to establish a reference HLW repository system by the end of 2006.

The disposal concept being studied at present is to encapsulate spent nuclear fuels into corrosion-resistant containers. These spent fuel containers are then to be disposed in a mined underground facility located about 500 m below the surface in a crystalline rock mass. The container is designed for a safe disposal of spent fuel in a deep geological repository, which entails an evenly or unevenly distributed hydrostatic pressure from the underground water and a high swelling pressure from the saturated bentonite buffer. Hence, an engineered barrier system (EBS), mainly a disposal container, needs to be designed to withstand these high-pressure loads with the provision of a radiological safety during the proposed lifetime.

There are two kinds of disposal containers, corrosion resistance- or corrosion allowance-type, proposed by several advanced countries advanced with the HLW disposal technology. The corrosion resistance-type container consists of two parts. One is an outer shell for the corrosion resistance and the other is an insert for the mechanical strength. Double-layered composite container was introduced in Sweden [1,2] and Finland [3]. The lifetime of this container consisting of copper and steel was designed to be more than 100,000 years. This long lifetime of the container may guarantee that the radioactivity in the waste decreases to the level of a natural background within the container. The main function of the container is to confine the radionuclides in the waste package during the specified period.

Since 1997, the Korea Atomic Energy Research Institute (KAERI) has been developing a Korea standard Reference disposal Container (KDC) to accommodate all the types of spent PWR and CANDU fuels generated from the domestic Nuclear Power Plants (PWR 16 units and CANDU 4 units), which satisfies the established functional/technical criteria under the normal and abnormal cases which may be found in a deep underground disposal. A prototype of the reference disposal container for the respective PWR and CANDU fuel has been designed through a set of mechanical structure analysis, radiation shielding and nuclear criticality analysis during the past years [4,5,6]. The basic concept in the designing process was to use the same overall sizes and component materials for both the PWR and CANDU fuels containers for a simple encapsulation and handling processes for an above and the underground facility. The previous study [7] showed that the diameter and height of the pre-conceptually designed KDC were 122 cm and 483 cm, respectively. The container size was determined by the diameter of the insert provided through the structural safety analysis rather than the radiation shielding analysis. The insert has to be designed to withstand the hydrostatic pressure and the swelling pressure from the swelling of clay-based buffer material. Consequently the weight of the pre-conceptually designed disposal container was about 40 tons because of too conventional design criteria, which may cause some unexpected difficulties in the handling process and from an economical point of view.

In this paper, some engineering efforts to lessen the weight of the disposal container were examined. Sets of engineering analyses were conducted to renew the dimensions of the container and to remodel all the configurations of the container components. Additionally a new heat source term from the PWR spent fuels and boundary conditions was applied in the assessment of the pre-conceptual design of the insert of the disposal container for the PWR spent fuels. The results of the calculation were compared with the safety factor of 2.0. This study aims to provide the optimum size of the container from the aspects of the structural safety, nuclear criticality and radiation shielding safety.

II. Reference Spent Fuel and Container Description

1. Reference Spent Fuel

In a previous study [8], a reference spent fuel, encompassing a variability of the characteristics of all the existing and future spent fuels of interest was defined. Key parameters in the reference fuel screening processes were the nuclear and mechanical design parameters and the burnup histories for the existing spent fuels as of 2003 and for the future spent fuels with an extended burnup for the initial enrichments and their expected burnups. The selected reference fuel which is characterized in terms of the initial enrichment, burnup, dimension, gross weight and age is summarized as :

- types of the reference spent PWR fuel
  - ·fuel rod array : 17 x 17
  - ·total weight and dimensions : 665 kg, 21.4^2 cm^2 (cross-section) x 453 cm (length)
  - ·decay heat per assembly : 385 W/assembly
- initial enrichment and discharge burnup of the reference PWR fuel
  - ·nominal burnup case : 4.0 wt.% for 45,000MWd/tHM
  - ·cooling time before disposal : 40 years

Figure 1 shows the spent fuel inventories which are annually generated from the Korean Nuclear Power Plants and the trend of the average discharge burnup of the spent PWR fuel. At the end of 2004, the total amount of spent fuels accumulated was about 7,300 tHM (3,400 tHM for PWR and 3,900 tHM for CANDU). According to the National 2nd Basic Plan for Electric Power Demand and Supply, a total of twenty eight units will be in operation in 2017 and the generation rate of the spent fuel is expected to increase even more. Average burnup of the spent PWR fuel was revealed as ~36 GWD/MTU and ~40 GWD/MTU for the period of 1994-1999 and 2000-2003, respectively. From this trend, the average burnup of the spent PWR fuel is expected to exceed 45 GWD/MTU at the end of the 2000's.

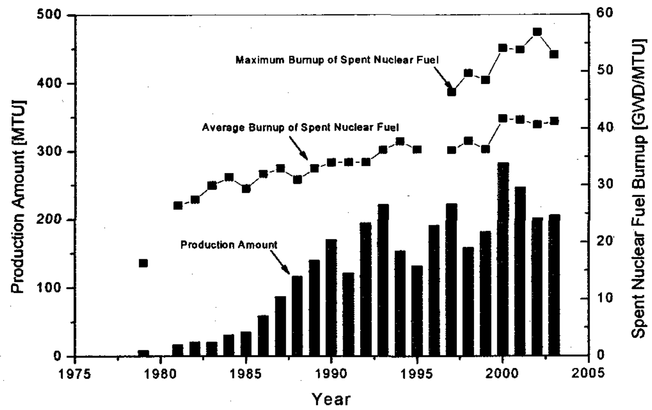
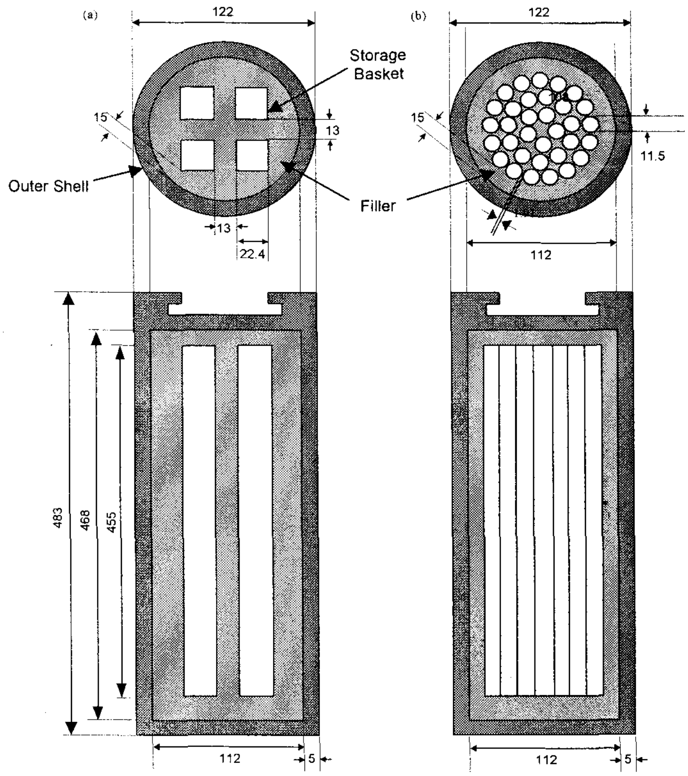


Fig. 1. Spent fuel inventories annually generated from the Korean Nuclear Power Plants and the trend of the average discharge burnup

2. Disposal Container

Figure 2 shows a schematic diagram of the Korea reference spent fuel Disposal Container (KDC) which has been developed to establish a Korea Reference disposal System (KRS), over the past years. As mentioned in the previous section, the overall sizes and component materials of the containers for both spent fuels are designed to be exactly identical to make the encapsulation and handling processes in the repository simple. Table 1 shows the technical criteria for developing the disposal container. In this table, values considered in Finland are also included for the reference.

The container being developed consists of two major components: a massive cast insert and a corrosion-resistant outer shell. The insert provides a mechanical strength and a radiation shielding, and it holds the fuel assembles in a fixed configuration. For the insert, carbon steel is considered as the design basis material. For a complete isolation of the waste for a long time, high nickel alloy, stainless steel or copper are considered as a candidate corrosion-resistant material for the outer shell. As shown in the figure, the outer shell contains the fuel storage baskets (4 square tubes for spent PWR fuel and 33 circular tubes for spent CANDU fuel) and the void space between the fuel storage basket and the outer shell is filled with carbon steel, called a cast insert. The loading capacity of the container was determined from a thermal analysis to confirm that the maximum thermal load on the container satisfies the thermal constraint of the clay-based buffer surrounding the container. The temperature at the buffer should be lower than 100 ℃ to maintain the physical and chemical properties of the buffer as the physical and chemical barriers in the disposal system. Consequently, thermo-mechanical analysis of the container under the expected and unexpected deep geological conditions showed that the proposed container could accommodate four spent PWR fuel assemblies or 297 CANDU fuels. The heat load from the PWR and CANDU containers are about 1.54 kW and 0.68 kW, respectively. The overall dimensions and design characteristics of the container are shown in Fig.2.



Canister Outer shell	Ni-Alloy (or Copper or SUS)	Ni-Alloy (or Copper or SUS)
Filler (void space in the container)	Carbon steel	Carbon steel
Capacity	4 PWR Spent Fuel Assemblies	297 CANDU Spent Fuel Bundles
Total No. of the Required Containers	11,375	2,926
Residual Heat in Canister	1,540 Watt	760 Watt
Total Volume of the container	5.64 m^3	5.64 m^3
Total Surface Area	19.67 m^2	19.67 m^2
Total Weight ☞Fuel wt. ☞Cast Insert ☞Outer shell (Case for Ni-alloy) (Case for Copper)	39,112 kg 2,660 kg 27,345 kg 9,107 kg 9,211 kg	39,099 kg 7,425 kg 22,567 kg 9,107 kg 9,211 kg

Fig. 2. Schematic diagram of the reference disposal container for spent PWR (a) and CANDU (b) fuels

Table 1. Technical Criteria for designing the Disposal Container

	KAERI	Finland
Duration of corrosion resistant	1,000 or 100,000 years	100,000 years
Maximum exposure rates	0.5 Gy/h	1 Gy/h
Criticality	K_{eff} < 0.95	subcritical
Temp. at the container surface	< 100°C	< 100°C
Load (Pressure) condition	Qualitatively stated	7 MPa + 7 MPa
Drop test condition	2.0 m	Not mentioned
Handling	Qualitatively mentioned	Strength of copper
Remaining	Initial defects less than 0.1 %	Gap between outer shell and insert: <5%

III. Calculation and Analysis

1. Structural Safety Analysis

The heat source from the PWR spent fuels was recalculated with the computer program, ORIGEN-ARP [9], whose validation and verification (V&V) have been performed extensively by Oak Ridge National Laboratory under the support of DOE and NRC since 1982. Figure 3 shows the decay heat generated from one ton of the spent PWR fuel material as a function of the cooling time. This decay heat projection was based on the results of the ORIGEN-ARP calculation, from which the heat generation rate could be expressed as a simple formula as follows:

$$q(50) = 1502.1\, W/m^3 \qquad (1)$$

This value was used to calculate the thermal stress from the PWR spent fuels.

Table 2 shows that Von Mises Stress of the normal and abnormal cases with respect to a variation of the diameter of the container. Two cases of a hydrostatic pressure and a swelling pressure were considered in this paper. One is for an extreme case and another for anormal case. The safety factor of 2.0 was used for the extreme case and the normal case.

The diameter of the container was 122 cm in the previous study [7]. Results of the structural analyses for five cases were compared to evaluate the suitability of the container insert as shown in Table 2. In this table, the smallest container with the diameter of 102 cm meets the safety factor of 2.0 for both the extreme load condition and the normal load condition. The distribution of the von-Mises stress for the extreme load condition is shown in Fig. 4.

Table 2. Results of the structural analysis

		102cm	107cm	112cm	117cm	122cm
Load Case 1	Max. Von Mises Stress (MPa)	80.5	77.5	74.7	72.3	70.0
	Safety Factor	2.5	2.6	2.7	2.8	2.8
	Max. Deformation (mm)	2.7	2.7	2.7	2.7	2.7
Load Case 2	Max. Von Mises Stress (MPa)	65.8	62.2	61.7	61.4	57.4
	Safety Factor	3.0	3.2	3.2	3.3	3.5
	Max. Deformation (mm)	2.6	2.6	2.6	2.6	2.6

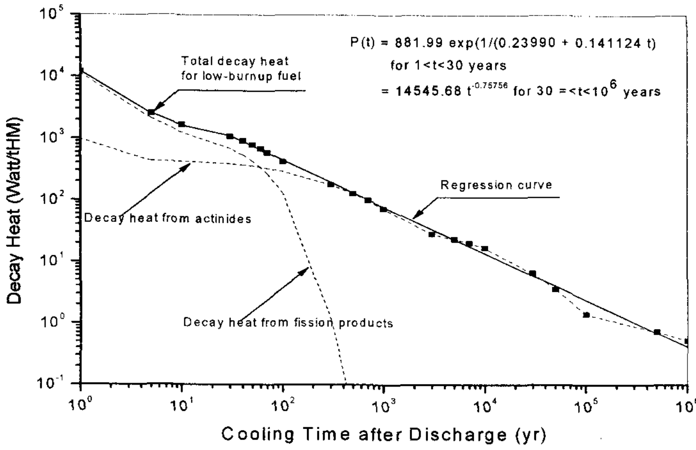


Fig. 3. Estimation of the Decay heat from the Reference spent PWR fuel.

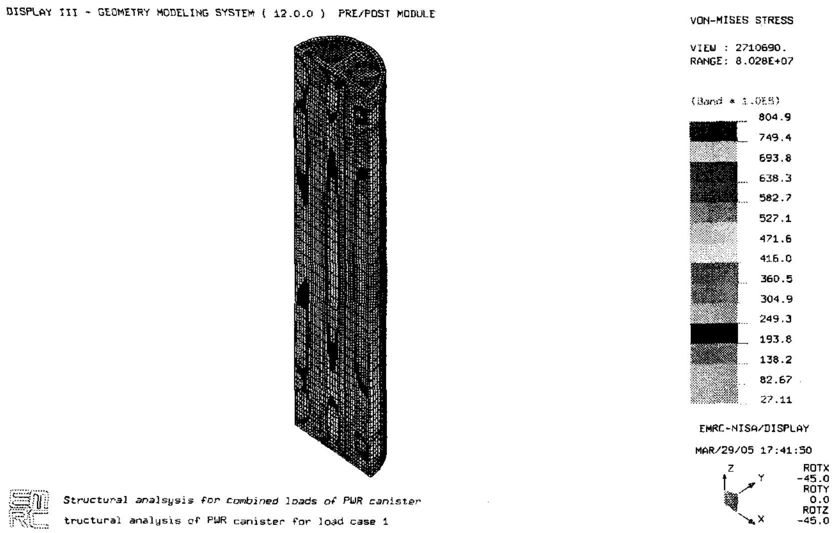


Fig.4. Results of the structural analysis of the container insert for load case 1

2. Nuclear Criticality and Radiation Shielding Analyses

Spent Fuel, Disposal Container, and Computational Tool

The characteristics of the spent PWR fuel and the basic structure of the disposal container for the calculation of a nuclear criticality and a radiation shielding are the same as those previously described. In this study, two steps were applied in the criticality analysis. In the first step, burnup calculations up to each discharge burnup were undertaken with the SCALE [10] control module ORIGEN-ARP. The built-in PWR 17×17 cross section library was applied. Sets of nuclide number densities for each burnup and cooling period were estimated based on the assembly design characteristics, fuel type, and the reactor operating conditions. The assumptions applied to perform the burnup calculations for each fuel assembly are as follows: The reactor was operated with a specific power of 37.5 W/gU for the PWR fuel assemblies with appropriate showdown periods. These burnup calculations provided initial conditions for the second step of the analyses, in which MCNP [12] calculations with a continuous nuclear data library were implemented to evaluate the reactivity of the spent fuel assemblies placed in the containers in an underground repository.

For the shielding analysis, ORIGEN-ARP was also used to obtain the source intensities and spectrum of the photon and neutron emitted from the spent fuel in the disposal container. MCNP calculation was performed to evaluate the gamma and neutron dose rate from the container. A flat source distribution was assumed along the axial and radial direction.

Criticality Analysis

For the nuclear criticality analysis, the disposal container was assumed to be surrounded by 60 cm thick bentonite buffer, with 150cm of bentonite above the container and 50 cm of bentonite below the container. The disposal hole was assumed to form an infinite two-dimensional planner lattice pitch of 6 m in the host rock. The model included a rock thickness of 5 m above and below the boreholes. It was considered that the bentonite was saturated with water and the storage basket was filled with water of 1.0 g/cc, which is the highest reactive condition [5]. Only actinide nuclides were considered for a burnup credit.

The minimum burnups to achieve a criticality safety were evaluated. As a result, the minimum burnups were shown to be 18 GWD/MTU for reference fuel. The fissile and actinide contents were shown to be 1.58 and 0.04 wt%, when it was discharged from the reactor.

The multiplication factor with the cooling time was evaluated to check the criticality trends after operating the repository. Neutron multiplication factor was maintained below ~0.74 at the beginning of emplacement. The multiplication factor increased after emplacement from 100 to 10,000 years after emplacement. However, the peak value was revealed as about 0.78. The reason that the infinitive multiplication factor is increasing after 100 years is due to an increase of ^{235}U caused by a decay of ^{239}Pu with a 2 ,000-year half-life and a decrease of the neutron absorbers of ^{240}Pu and ^{241}Am.

Criticality safety was evaluated with the severe assumption that the container and bentonite disappeared, which allows the spent fuel to be surrounded by water. The distance between the assemblies and moderator density were varied from 0 to 13 cm and from 0.6 to 1.0 g/cc, respectively. When the moderator density is 1.0 g/cc and the assemblies are in contact with each other, the multiplication factor showed the highest value of 0.86717±0.00092.

Therefore, if the fuel assemblies are intact and the fissile nuclide is confined in a fuel rod, a criticality safety in a repository is not possible under the proposed design.

Radiation Shielding Analysis

According to the design requirement of the disposal container, absorbed dose from the surface of a container should be maintained below the limit of 0.5 Gy/hr to avoid a radiolysis and a subsequent corrosion. Photon and neutron absorbed dose rates in the water in the bentonite layer close to the container surface, calculated by the MCNP with MCPLIB2 library are given in Table 3. Absorbed dose rate at the bottom of the container is about 1.5 ~ 3 times higher than that at the side. It cant be concluded that current design satisfies the design limit, but the radiolysis caused by photon needs to be examined in detail in a further analysis.

As a lower dose rate is achieved, the more the benefits regarding a radiation protection during the emplacement operation of a waste package are possible. As the shield thickness increases to reduce the dose rate, however, the cost and weight increases. Therefore, from the aspects of the safety and cost points of view, the dose rate at the surface of the container should be weighed. It is clear that an additional shielding mechanism for a radiation protection should be developed in further work based on the results also listed in Table 3.

Table 3. Dose rate at the surface of the container

	Side		Bottom(Top)	
	Photon	Neutron	Photon	Neutron
Absorbed dose (Gy/hr)	9.86×10^{-2}(3%)[a]	4.75 $\times 10^{-5}$(1%)	5.77 $\times 10^{-1}$(4%)	7.10 $\times 10^{-5}$(1%)
Exposure dose (Rem/hr)	8.11(3%)[a]	1.09×10^{-1}(1%)[a]	5.27×10^{1}(4%)	1.64×10^{-1}(2%)

[a] Percentage standard deviation

IV. Conclusions

The diameter of the container insert for the PWR spent fuels was determined based upon the structural analysis. The calculation shows that a 102 cm container meets the safety factor of 2.0 for both the extreme and normal load conditions. The remodeled container proposed in this study satisfies the nuclear criticality and radiation safety limits. Such a decrease in the diameter when compared to the previous container design results in a weight loss of the container by ~20 tons. And the volume of the buffer material necessary for the PWR spent fuel deposition holes could be reduced by about 3.94×10^{4} m^3 when compared with the previous study.

Acknowledgement

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References

[1] C.G. Andersson, P. Eriksson, and M. Westman, Status report, container fabrication, SKB TR-04-23 (2004).

[2] L.Werme, Design premises for container for spent nuclear fuel, SKB Technical Report TR-98-08, SKB (1998).

[3] H. Raiko, J.P.Salo, Design report of the container for nuclear fuel disposal, POSIVA-96-13 (1996).

[4] J.W. Choi, S.K. Kwon, C.H. Kang, and Y.J. Kwon, "A Reference Container Concept for Spent Fuel Disposal : Structural Safety for Dimensioning of the Reference Container,"Journal of the Korean Association for Radiation Protection, vol. 29, No. 1, pp. 49-55 (2004)

[5] D.K. Cho, J.W. Choi, J.Y. Lee, S.K. Kim, and P.S. Hahn, "A Preliminary Evaluation on Criticality Safety for Spent Fuel Disposal System,"Proc. of Korean Nuclear Society Spring Meeting, May 25-29, Gyeongju, Korea (2004).

[6] D.K. Cho, J.W. Choi, J.Y. Lee, H.J. Choi and S.K. Kim, "Evaluation of Radiation Shielding Performance of Disposal Canister Storing PWR Spent Fuels," Proc. Of Korean Nuclear Society Atomn Meeting, pp. 1188-1189, October 28-29, Yongpyong, Korea (2004).

[7] C.H. Kang, J.W. Kim, K.S. Chun, J.H. Park, W.J. Cho, J.W. Choi, J.O. Lee, Y.M. Lee, S.S. Kim, Y.S. Hwang, S.K. Kim, and S.K. Kwon, High-Level Radwaste Disposal Technology Development, Korea Atomic Energy Research Institute Report,KAERI/RR-2236/2002 (2002).

[8] C.H. Kang, J.W. Choi, W.I. Ko, S.K. Kim, Y.M. Lee, J.H. Park, and Y.S. Hwang, Reference Spent Fuel and Its Characteristics for the concept Development of a Deep Geological Disposal System, Korea Atomic Energy Research Institute Report, KAERI/TR-914/97 (1997).

[9] I.C. Gauld, S.M. Bowman, J.E. Horwedel, and L.C. Leal,ORIGEN-ARP: AUTOMATIC RAPID PROCESSING FOR SPENT FUEL DEPLETION, DECAY, AND SOURCE TERM ANALYSIS, Oak Ridge National Laboratory (2004).

[10] SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, NUREG/CR-0200, ReV. 7(ORNL/NUREG/CSD-2/V1/R7), Vols. I, II, III, May 2004, Oak Ridge National Laboratory (2004).

[11] I. I. C. Gauld, and O.W. Hermann, SAS2: A COUPLED ONE-DIMENSIONAL DEPLETION AND SHIELDING ANALYS MODULE, Oak Ridge National Laboratory (2004).

[12] J. F. Briesmeister, MCNP-A General Monte Carlo N-Particle Transport Code, Version 4C, LA-13709-M, Los Alamos National Laboratory (2000).