⟨Technical Note⟩

http://dx.doi.org/10.5516/NET.08.2013.039

SHIELDING ANALYSIS OF DUAL PURPOSE CASKS FOR SPENT NUCLEAR FUEL UNDER NORMAL STORAGE CONDITIONS

JAE-HUN KO[1*], JEA-HO PARK[1], IN-SOO JUNG[1], GANG-UK LEE[1], CHANG-YEAL BAEG[2], and TAE-MAN KIM[2]
[1]Korea Nuclear Engineering & Service Co.
#2, Hyundai Plaza., 341-4 Jangdae-dong, Yuseong-gu, Daejeon, Korea
[2]Korea Radioactive Waste Management Corporation
1045 Deadeokdae-ro, Yuseong-gu Daejeon, Korea
[*]Corresponding author. E-mail : kojh4@kones21.com

Received May 16, 2013
Accepted for Publication February 19, 2014

Korea expects a shortage in storage capacity for spent fuels at reactor sites. Therefore, a need for more metal and/or concrete casks for storage systems is anticipated for either the reactor site or away from the reactor for interim storage. For the purpose of interim storage and transportation, a dual purpose metal cask that can load 21 spent fuel assemblies is being developed by Korea Radioactive Waste Management Corporation (KRMC) in Korea.

At first the gamma and neutron flux for the design basis fuel were determined assuming in-core environment (the temperature, pressure, etc. of the moderator, boron, cladding, UO_2 pellets) in which the design basis fuel is loaded, as input data. The evaluation simulated burnup up to 45,000 MWD/MTU and decay during ten years of cooling using the SAS2H/OGIGEN-S module of the SCALE5.1 system. The results from the source term evaluation were used as input data for the final shielding evaluation utilizing the MCNP Code, which yielded the effective dose rate.

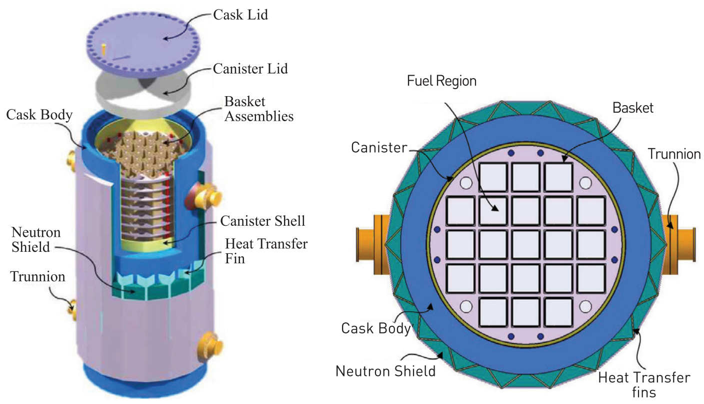
The design of the cask is based on the safety requirements for normal storage conditions under 10 CFR Part 72. A radiation shielding analysis of the metal storage cask optimized for loading 21 design basis fuels was performed for two cases; one for a single cask and the other for a 2x10 cask array. For the single cask, dose rates at the external surface of the metal cask, 1m and 2m away from the cask surface, were evaluated. For the 2x10 cask array, dose rates at the center point of the array and at the center of the casks' height were evaluated. The results of the shielding analysis for the single cask show that dose rates were considerably higher at the lower side (from the bottom of the cask to the bottom of the neutron shielding) of the cask, at over 2mSv/hr at the external surface of the cask. However, this is not considered to be a significant issue since additional shielding will be installed at the storage facility. The shielding analysis results for the 2x10 cask array showed exponential decrease with distance off the sources. The controlled area boundary was calculated to be approximately 280m from the array, with a dose rate of 25mrem/yr. Actual dose rates within the controlled area boundary will be lower than 25mrem/yr, due to the decay of radioactivity of spent fuel in storage.

KEYWORDS : Radiation Shielding, 2x10 Array, Storage Condition, Dual Purpose Cask, Spent Fuel Assemblies

1. INTRODUCTION

Korea expects a shortage in storage capacity for spent fuels at reactor sites. Therefore, a need for more metal and/or concrete casks for storage systems is anticipated for either the reactor site or away from the reactor for interim storage. For the purpose of interim storage and transportation, a dual purpose metal cask that can hold 21 spent fuel assemblies is being developed by Korea Radioactive Waste Management Corporation (KRMC) in Korea. This cask is composed of a main body made of carbon steel and a stainless steel dry shielded canister (DSC) with stainless steel baskets inside to contain spent fuel assemblies as shown in Fig. 1.

The design of the cask is based on the safety requirements for normal storage conditions of 10 CFR Part 72. 10 CFR Part 72 requires that spent fuel storage and handling systems be designed with adequate shielding to provide sufficient radiation protection under normal, off normal, and accident-level conditions. As prescribed in 10 CFR Part 72, the regulatory requirements for dose rates at and beyond the controlled area boundary include radiation from direct radiation and radiation from radionuclides in effluents. To meet the regulatory requirements of 10 CFR Part 72, dry storage facilities for spent fuel should



(a) Outline of the metal cask (b) Cross-sectional view of the metal cask

Fig. 1. Dual Purpose Metal Cask Arrangement

be designed to protect the public and radiation workers from direct radiation from casks. While 10 CFR Part 72 does not impose specific dose rate limits on individual casks, dose rates from 20 to 400mrem/hour have been accepted for storage casks in previous 10 CFR Part 72 evaluations[1]. The shielding analysis for a single storage cask should identify the minimum distance that is required to meet the dose criteria stipulated in 10 CFR 72.104. Dose rates at the controlled area boundary are generally calculated to meet the requirement of less than 25mrem/year for the public under normal storage conditions. In addition, the radiation shielding analysis should include a graph on dose rate versus distance for a single cask to facilitate a site-specific evaluation for general licensees[1].

The design of the spent fuel storage facilities should consider many factors, such as the casks' arrangement, radiation workers' working condition (ex. installation, visual examination, radiation monitoring, and maintenance, etc.), and operating procedures. To take these design factors into consideration, dose rates at the cask's external surface should be calculated. After that, a calculation of dose rates for the cask surface, performed in accordance with the NUREG-1536 dose rates evaluation, should also include a dose rate versus distance curve for a theoretical cask array. The theoretical cask array should consist of at least 20 storage casks (typically in a 2x10 array), and may include the shadowing effect among the casks.

Design basis accidents could result in limited and localized damage to the outer shell and radial external surface of the cask. However, as the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose rate at the site boundary would be negligible[2][3]. This is because damage to a single cask has negligible effects on radiation safety of the whole storage facility.

This paper presents the shielding analysis method and results of dose rate calculations for a single cask and the 20 storage cask arrays under normal storage conditions.

2. RADIATION SOURCE

The storage cask was designed to load either 21 WH or 21 CE type fuel discharged from Korean NPPs prior to 2008. To simulate loading different types of spent fuel in the cask, a hypothetical design basis fuel was assumed in order to be conservative. For the evaluations, the design basis fuel assembly was divided into two parts: one containing fissile material and the other including all other structural components

The active fuel region was assumed to load WH type 17RFA fuel assemblies, with the greatest U-metal mass emitting the most gamma rays and neutrons. Also, structural component of the design basis fuel assembly were assumed to be that of CE type PLUS7^{TM} fuel, which has the largest dimensions.

To calculate dose rates from the spent fuel storage cask, a source term evaluation for design basis fuel was carried out. The following are factors of the source term that should be considered in terms of radiation protection.

- Gamma sources
  - Primary gamma rays emitted from disintegration of fission products and actinides
  - Gamma rays emitted from the decay of Co-60 nuclides generated from the activation of fuel assembly structures
  - Secondary gamma rays generated from (n, γ) reaction of fissile and non-fissile materials
- Neutron sources
  - Neutrons generated from spontaneous fission
  - Neutrons generated from (α, n)reaction of fissile materials

Primary gamma rays and neutrons generated from spontaneous fission and (α, n)reaction of fissile materials were calculated by creating cross section libraries, and by reflecting the burnup and cooling time of the design basis fuel using the SAS2H module of SCALE 5.1[4]. Gamma rays emitted from the decay of Co-60 nuclides, which were generated from the activation of structural components, were calculated using the flux scaling factors and the results generated from the SAS2H module. In addition, secondary gamma rays generated from (n, γ) reactions of fissile and non-fissile materials were calculated using appropriate options during the preparation of shielding analysis input.

The source term evaluation for the design basis fuel is determined the gamma and neutron flux of the fuel assembly using the data of the in-core environment (the temperature, pressure, etc. of the moderator, boron, cladding, UO2 pellet), in which the design basis fuel is loaded at the reactor. The evaluation simulated burnup up to 45,000 MWD/MTU and decay during ten years of cooling using a burnup code (SAS2H/OGIGEN-S module of SCALE5.1 system). The results from the source term evaluation are shown as the flux per energy group of gamma rays and neutrons(18 group gamma flux and 27 group neutron flux).

SAS2H is a control module used for source term evaluation and produces temporary cross section libraries in connection with the "BONAMI" and "NITAWL-II" modules in the SCLAE5.1 code, which can process resonance self-shielding. The temporary cross section libraries which are created are used as base input data for the ORIGEN-S module, which, by adopting the matrix exponential method, can calculate the creation and decay of fission products and actinides after irradiated fuel assemblies go through neutron transformation, fission, and radioactive decay. ORIGEN-S reflects, through the COUPLE module of the SCALE5.2 code, the weight value of the neutron spectrum for the burnup history of each ENDF/B fission reaction cross section library's energy group. Then, it calculates the burnup characteristic of the spent fuel based on the revised multi-group temporary cross section libraries.

The source term of the design basis fuel is time dependent. In determining the time dependence of nuclide concentrations, ORIGEN-S is primarily concerned with developing solutions for the following equation:

$$\frac{dN_i}{dt} = \text{formation rate} - \text{destruction rate} - \text{decay rate} \qquad (1)$$

ORIGEN considers radioactive disintegration and neutron absorption (capture and fission) as the processes appearing on the right-hand side of Eq. 1. The time rate of change of the concentration for a particular nuclide, Ni, in terms of these phenomena can be written as

$$\frac{dN_i}{dt} = \sum_j \gamma_j \sigma_{f,j} N_j \Phi + \sigma_{c,i-1} N_{i-1} \Phi + \lambda_i' N_i' - \sigma_{f,i} N_i \Phi - \sigma_{c,i} N_i \Phi - \lambda_i N_i \qquad (2)$$

Where (i = 1, … I), and

$\sum_j \gamma_{ji} \sigma_{f,j} N_j \Phi$ is the yield rate of N_i due to fission of all nuclides N_j

$\sigma_{c,i-1} N_{i-1} \Phi$ is the rate of transmutation into N_i due to radiative neutron capture by nuclide N_{i-1}

$\lambda_i' N_i'$ is the rate of formation of N_i due to the radioactive decay of nuclides N_i'

$\sigma_{f,i} N_i \Phi$ is the destruction rate of N_i due to fission

$\sigma_{c,i} N_i \Phi$ is the destruction rate of N_i due to all forms of neutron absorption other than fission (n,γ, n,α, n,p, n,2n, n3n)

$\lambda_i N_i$ is the radioactive decay rate of N_i

Eq. 2 is written for a homogeneous medium containing a space-energy-averaged neutron flux, φ, with flux-weighted average cross sections, σ_f and σ_c, representing the reaction probabilities. The flux is a function of space, energy, and time is dependent upon the nuclide concentrations. The mathematical treatment in ORIGEN-S assumes that the space-energy-averaged flux can be considered constant over a sufficiently small time interval, Δt. Similarly, it is assumed that a single set of flux weighted neutron cross sections can be used over the same time step. For a given time step, these assumptions are necessary if Eq. 2 is to be treated as a first-order, linear differential equation.

2.1 Gamma Source

As previously mentioned, the major gamma sources generated from spent fuel consists of gamma rays generated from disintegration of fission products and actinides, and gamma rays emitted from the decay of Co-60 nuclides generated from activation of fuel assembly structures.

Active fuel region

Primary gamma sources generated from the active fuel region were calculated using the SAS2H/ORIGEN-S module of SCALE 5.1. Table 1 and Fig. 2 show the results.

Activation of fuel assembly structures

Co-59 contained in the structural components, such as the steel and Inconel components of the fuel assembly, becomes Co-60. Co-60 emits strong gamma rays of 1.173MeV and 1.332MeV and should be considered when calculating spent fuel dose rates. Activities of radioactive structures are calculated by multiplying the contents of Co-59 in the fuel assembly structures with flux scaling factors of each region and with activities generated from 1g of Co-59.

Co-59 contents in fuel assembly structures were taken from the ORNL-6051 report[5]. In addition, flux scaling factors were taken from Table 1 of NUREG/CR-6802[6]. The activities for 1g of Co-59 were calculated using the SAS2H/ORIGEN-S module of SCALE 5.1. Table 2 shows the gamma flux from the decay of Co-60 generated from the activation of fuel assembly structures.

Table 1. Primary Gamma Ray Flux of the Design Basis Fuel

Energy [MeV]	Gamma Flux [photons/sec·FA]	Energy [MeV]	Gamma Flux [photons/sec·FA]
0.01~0.05	9.556E+14	1.33~1.66	6.175E+12
0.05~0.10	2.672E+14	1.66~2.00	1.262E+11
0.10~0.20	1.936E+14	2.00~2.50	4.647E+10
0.20~0.30	5.712E+13	2.50~3.00	2.831E+09
0.30~0.40	3.744E+13	3.00~4.00	2.683E+08
0.40~0.60	1.526E+14	4.00~5.00	7.264E+06
0.60~0.80	1.724E+15	5.00~6.50	2.915E+06
0.80~1.00	7.964E+13	6.50~8.00	5.718E+05
1.00~1.33	4.600E+13	8.00~10.0	1.214E+05
Total gamma flux			3.520E+15

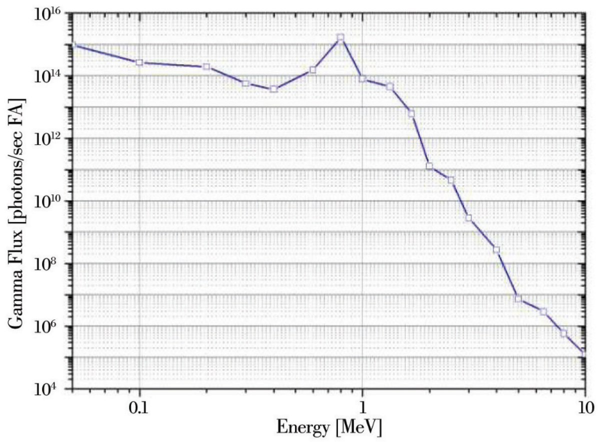


Fig. 2. Gamma Ray Spectrums of the Design Basis Fuel

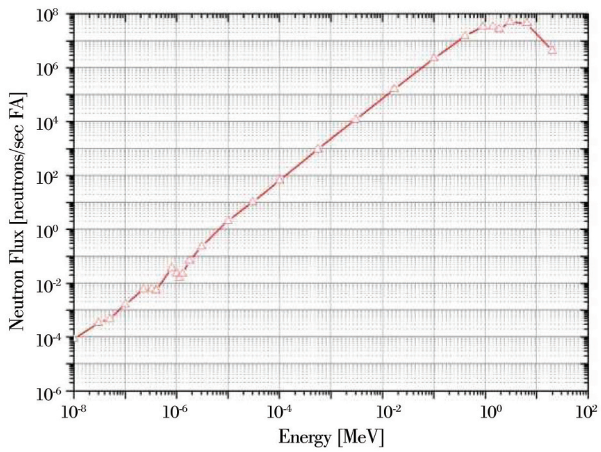


Fig. 3. Neutron Spectrums of the Design Basis Fuel

Table 2. Activation Source (Gamma Source from Co-60)

Structures	Activation Source [photons/sec · FA]
Inconel spring	2.3310E+12
Top end Plug	9.0132E+12
Plenum springs	1.0976E+12
Top grid	1.7013E+12
Cladding	3.4854E+12
Bottom grid	8.1104E+12
Bottom end Plug	3.2431E+12

Table 3. Neutron Flux of the Design Basis Fuel

Emission Form	Neutron Source [neutrons/sec · FA]
Spontaneous fission	2.054E+08
(α, n) reaction	4.299E+06
Total neutron flux	2.097E+08

2.2 Neutron Source

Neutrons generated from spontaneous fission and (α, n) reaction of fissile materials were calculated using the SAS2H module of SCALE 5.1, as shown in Table 3 and Fig. 3.

3. RADIATION SHIELDING EVALUATION

There is a recent trend toward the wider use of the Monte Carlo method in reactor modeling, simulation of radiation measuring instrument, as well as radiation shielding evaluation. In particular, the MCNP code is the most widely used code as a general-purpose program to analyze transportation of electrons, neutrons, and photons. Also, it can be used for three dimensional modeling with complicated geometric structure.

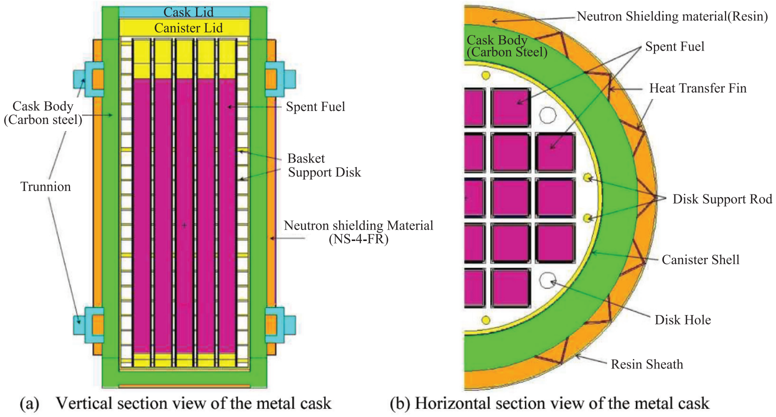


Fig. 4. Shielding Analysis Model for Single Storage Cask

In this study, MCNP5 (ver.1.40) was used to evaluate the cask's radiation safety. For accurate evaluation, MCNP input requires cross section libraries for nuclides of each material. Therefore, the ENDF/B-VI library for continuous energy was used in this analysis[7].

The following assumptions were made for the shielding analysis:

- Conservative specification for the design basis fuel
  - Fuel assembly was assumed to be a homogenized rectangular form
  - Active fuel region consists of UO_2 pellets and cladding
  - Conservative specification for structural components by adopting that of CE type fuel
- Application of a sufficient air layer to calculate scattering radiation effects caused by the collision between direct radiation from casks and air (Skyshine)
- Distance between casks is 1.2m

3.1 Analysis of Cask Model

The canister in the metal cask can accommodate 21 spent fuel assemblies. The canister lid is 240mm thick, considering radiation exposure for workers and crane capacity. The canister shell is 25mm thick, which provides additional radiation shielding. Especially, the basket model is comprised of basket support disks, support rods, and air path holes to simulate the actual model.

The main body of the cask is made of carbon steel with a thickness of 215mm, and of neutron shielding material, which is a 110mm thick layer of resin enclosed in a stainless steel sheath surrounding cask's exterior. There is 4mm thick resin layer in the bottom base plate, but not in the top plate. The resin sheath is made of stainless steel of 10mm thickness. Also, the shielding analysis model considered wedge shaped (">") heat transfer fins, and trunnions to lift and rotate the cask. Fig. 4 shows the shielding analysis model of a single storage cask under normal storage conditions.

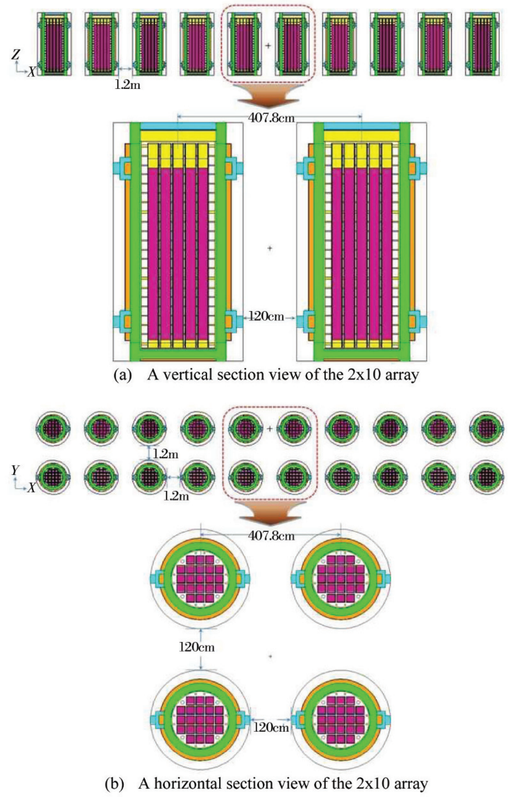


Fig. 5. Shielding Analysis Model for the 2x10 Array

As previously mentioned in the introduction, the shielding analysis was performed using a 2x10 array, as per recommendations of the Standard Review Plan of NUREG-1536. Therefore, the analysis model for storage casks in a 2x10 array adopted assumptions and features applied to the shielding analysis for a single cask. In this analysis model, the distance between casks was 1.2m. Fig. 5 shows the shielding analysis model for storage casks in a 2x10 array under normal storage conditions.

3.2 Input Data for Calculations

Axial burnup profiles

Spent fuels have different burnups along the axial height of the assembly, giving off different levels of heat and radiation. In order to analyze the radiation shielding capability of storage casks in practical conditions, the shielding analysis should include varying emission ratios along the emission axis.

The axial burnup profiles of the burnup range of the spent fuel that was chosen as the design basis was based on foreign data since there is no actual axial data for spent fuel from Korean NPPs. Available foreign data on axial burnup profiles include axial burnup profile data published by the Department of Energy (DOE), data from Table 5 of NUREG/CR-6801, and data from YEAC-1937. Among these, data from the DOE and those from NUREG/CR-6801 are based on the data from YEAC-1937, which used the CASMO-3 code to analyze axial burnup profiles of 3,169 fuel assemblies from 20 NPPs in the US. Therefore the data on emission ratio of gamma rays and neutrons of the design basis fuel along the axis were taken from YEAC-1937 published by the Yankee Atomic Electric Company[8]. The data on the emission ratio of gamma and neutron sources in this report are similar to axial burnup profiles of the design basis fuel. In particular, this study collected burnup data from 1,062 PWR fuel assemblies of WH and CE type and used data from fuel assemblies, that have a burnup range between 42,000 and 46,000MWD/MTU.

The maximum burnup for each node was used as the burnup profile for the design basis fuel, as shown in Table 4.

The emission ratios of gamma ray sources along the axis of the active fuel region change linearly depending on burnup. Also, more than 80% of gamma sources were emitted from fission products. Therefore, gamma sources are directly proportional to the burnup values, because the amount of the fission products is proportional to burnup. The strength of the neutron source is proportional to the burnup raised to the fourth power[6].

The relationship between source terms and the magnitude of some of these burnups is well known[6]. Table 5 shows the modification factors for the neutron source and gamma ray source values along the axis of active fuel region.

Material property

Table 6 shows the material properties and weight fractions of each material used in the cask.

Table 4. Axial Burnup Profile Depending on Fuel Height

Height of Fuel [%]	Axial burnup profile
2.78	0.71815
8.33	0.96558
13.89	1.06851
19.44	1.08916
25.00	1.10562
30.56	1.08885
36.11	1.10110
41.69	1.09912
47.22	1.07577
57.80	1.09239
58.33	1.07436
63.89	1.07799
69.44	1.07985
75.00	1.06999
80.56	1.05631
86.11	1.01978
91.67	0.91500
97.22	0.67478

Table 5. Modification Factors of the Axial Burnup Profiles

Height of Fuel [%]	Neutron Release Ratio	Gamma Ray Release Ratio
2.78	0.265987	0.71815
8.33	0.869267	0.96558
13.89	1.303510	1.06851
19.44	1.407235	1.08916
25.00	1.494251	1.10562
30.56	1.405634	1.08885
36.11	1.469965	1.10110
41.69	1.459420	1.09912
47.22	1.339300	1.07577
57.80	1.424003	1.09239
58.33	1.332292	1.07436
63.89	1.350389	1.07799
69.44	1.359733	1.07985
75.00	1.310747	1.06999
80.56	1.244989	1.05631
86.11	1.081499	1.01978
91.67	0.700946	0.91500
97.22	0.207324	0.67478

Table 6. Material Properties of Cask

Item	Material	Density [g/cm^3]	Nu-clide	Weight fraction[w%]
· Cask shell	SA-350 LF.3	7.80	C	0.001
			Si	0.0028
			Cr	0.0015
			Mn	0.0045
			Fe	0.9542
			Ni	0.035
			Mo	0.0006
			P	0.0002
			S	0.0002
· Cask lid · Trunnion	SA-182 Gr.F6NM	7.80	Si	0.0030
			Cr	0.1275
			Mn	0.0075
			Fe	0.8095
			Ni	0.045
			Mo	0.0075
· Basket · Neutron absorber sheat · Basket support disk · Canister shell and lid	SA-240 Type 304	7.94	C	0.0008
			Si	0.0075
			P	0.00045
			Cr	0.19000
			Mn	0.02000
			S	0.00045
			Ni	0.09250
			Fe	0.6883
· Heattransfer fin	SA-516 Gr.70	7.80	Si	0.0030
			Cr	0.1275
			Mn	0.0075
			Fe	0.8095
			Ni	0.045
			Mo	0.0075
· Environment	Air	0.001225	N	0.760
			O	0.240
· Neutron absorber	Boral	2.6456	B	0.277
			C	0.076
			Al	0.647
· Neutron shield	NS-4-FR (Resin)	1.682	Al	0.215
			H	0.060
			C	0.277
			O	0.428
			N	0.020

Flux to dose conversion factors

In this study, a FMESH tally was used for MCNP, and the results of the MCNP tally is shown as the flux(#/sec · cm^2) to be converted into dose rate. This study used the flux to dose conversion factor in ICRP-74[9].

5. RESULTS OF SHIELDING ANALYSIS

The dose rate results from MCNP calculations contain uncertainty as a relative error. In general the uncertainty is below 10%. In this study, the uncertainty for the single cask was less than 5% and that of the 2x10 array was less than 15%.

5.1 Single Cask

In the shielding analysis for a single cask, dose rates were evaluated at the external surface of the cask; 1m and 2m away from the cask surface. These results can be used as data to control handling time, work procedures, and radiation worker dose rates in accordance with the ALARA principle for such work as the installation, visual examination, radiation monitoring, and maintenance of casks. Table 7~9 and Fig. 6~8 show dose rate results measured at the exterior circumference of the single cask.

In the shielding analysis for the single cask, dose rates were considerably higher at the lower area (from the bot-

Table 7. Dose Rate Results at Cask Surface

[unit: mSv/hr]

Location		Neutron	Gamma	Total
Side of Cask	Upper part	0.2016	0.3297	0.5313
	Middle part	0.3480	0.2196	0.5676
	Lower part	0.9358	1.3216	2.2484

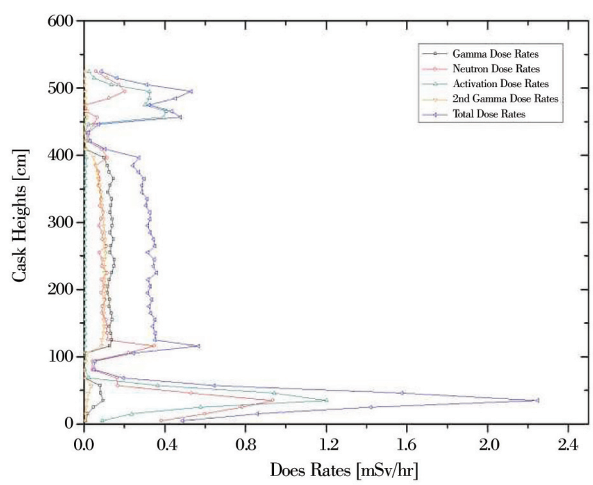


Fig. 6. Dose Rate Results at Cask Surface

Table 8. Dose Rate Results at 1m from Cask

[unit: mSv/hr]

Location		Neutron	Gamma	Total
Side of Cask	Upper part	0.0231	0.0942	0.1173
	Middle part	0.0394	0.1042	0.1436
	Lower part	0.0954	0.2253	0.3207

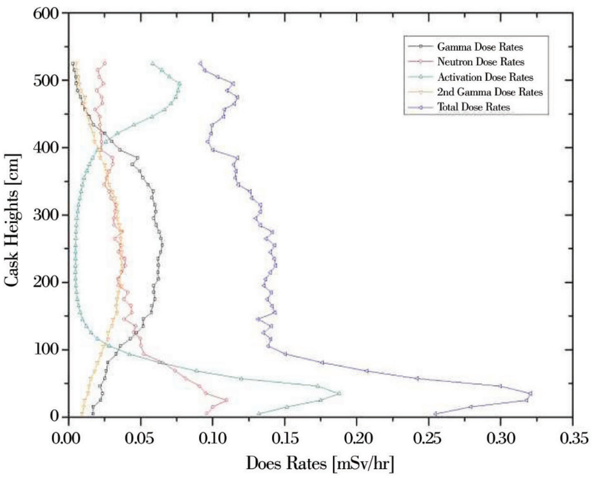


Fig. 7. Dose Rate Results at 1m from Cask

Table 9. Dose Rate Results at 2m from Cask

[unit: mSv/hr]

Location		Neutron	Gamma	Total
Side of Cask	Upper part	0.0166	0.0533	0.0699
	Middle part	0.0288	0.0634	0.0922
	Lower part	0.0414	0.1033	0.1447

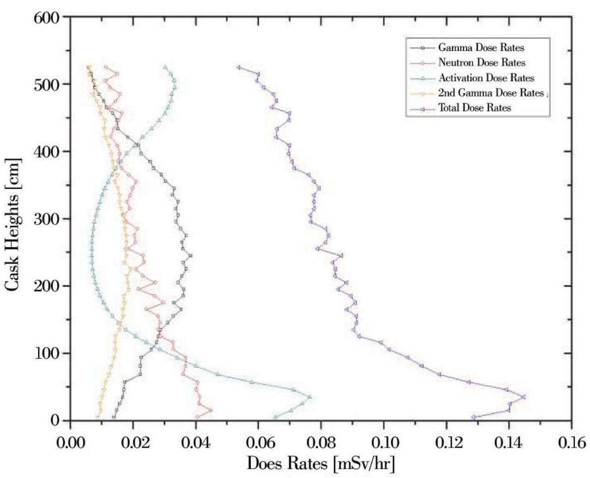


Fig. 8. Dose Rate Results at 2m from Cask

tom of the cask to the bottom of the neutron shielding material) of the cask. These results are due to gamma rays caused by the decay of Co-60 nuclides generated by activation of fuel assembly components and neutrons that cast off in the absence of neutron shielding material.

An evaluation of the dose rates showed that activation sources are responsible for only a negligible portion of the total dose rate. The total dose rate was mainly composed of other sources, which were all similar in strength. Also, the upper and lower parts of the cask showed higher dose rates than that of the middle part. The dose rates measured at the surface of the trunnions were significantly lower than that of other locations. This is because the trunnions are made of steel and contain neutron shielding material.

5.2 2x10 Array

The shielding analysis for multiple casks was performed using a 2x10 array, as per recommendations of the Standard Review Plan of NUREG-1536. It was based on the same assumptions and features applied to the shielding analysis for a single cask. The analysis of dose rates was performed for members of the public outside of the controlled area boundary. In particular, this analysis was performed by reflecting a sufficient air-layer to consider the effect of scattering radiation caused by the skyshine effect.

In this shielding analysis, the self-shielding effect between casks were taken into account, because the amount of radiation from the longer side of the array is greater than that of the shorter side of the array since there are more casks positioned lengthwise. Thus, the directional orientation of the array affects the calculation of the distance to the controlled area boundary. Fig. 9 shows dose rate results of the 2x10 array under normal storage conditions, in two directional orientations; lengthwise and crosswise. Dose rates from a single cask and those of the

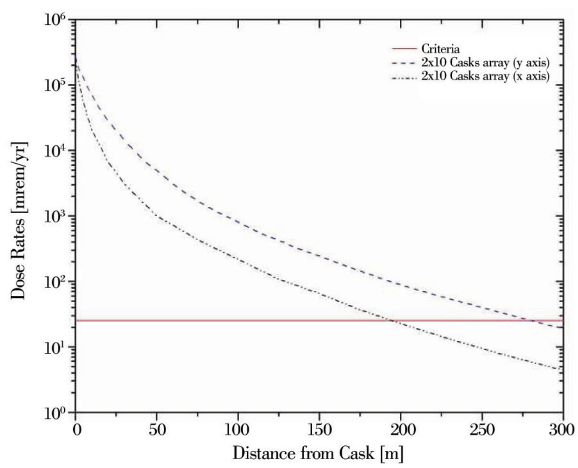


Fig. 9. Dose Rate Results of 2x10 Array in Lengthwise and Crosswise Orientations

2x10 array (lengthwise) were compared as shown in Fig. 10. The distances in Fig. 9 and Fig. 10 were measured from the outer surface of the cask array in each orientation.

The shielding analysis results of the 2x10 array showed that dose rates on surrounding areas (within the controlled area boundary) were higher in the lengthwise direction as shown in the Fig 9. As shown in Fig. 10, the distances from the cask(s) to the controlled area boundary were approximately 155m for a single cask and 280m and for the 2x10 array. The controlled area boundary means the border of a restricted area not accessible to the public, in which the dose rate is limited to 25mrem/yr. This means a minimum separation distance of 280m is required at future interim storage facilities.

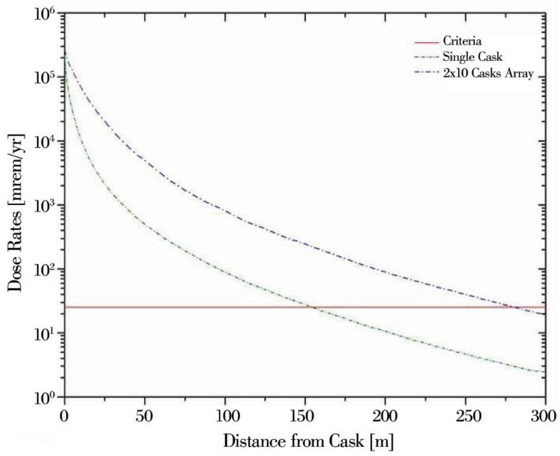


Fig. 10. Dose Rate Results of the Single Cask Versus the 2x10 Array (Lengthwise)

6. CONCLUSION

A radiation shielding analysis of a metal storage cask, optimized for loading 21 design basis fuel assemblies, under normal storage conditions was performed for two cases; one for a single cask and another for a 2x10 cask array. The procedure for the design basis fuel's source term evaluation and the cask's shielding analysis is shown in Fig. 11.

Fig. 11 Procedure for design basis fuel's source term evaluation and cask's shielding analysis

For the single cask, dose rates were evaluated on the external surface of the cask, and at points of one and two meters away from the cask surface. For the 2x10 cask array, dose rates were evaluated at different distances from the outer surface of the cask array in lengthwise and crosswise orientations.

The results of the shielding analysis for the single metal cask showed that dose rates were considerably

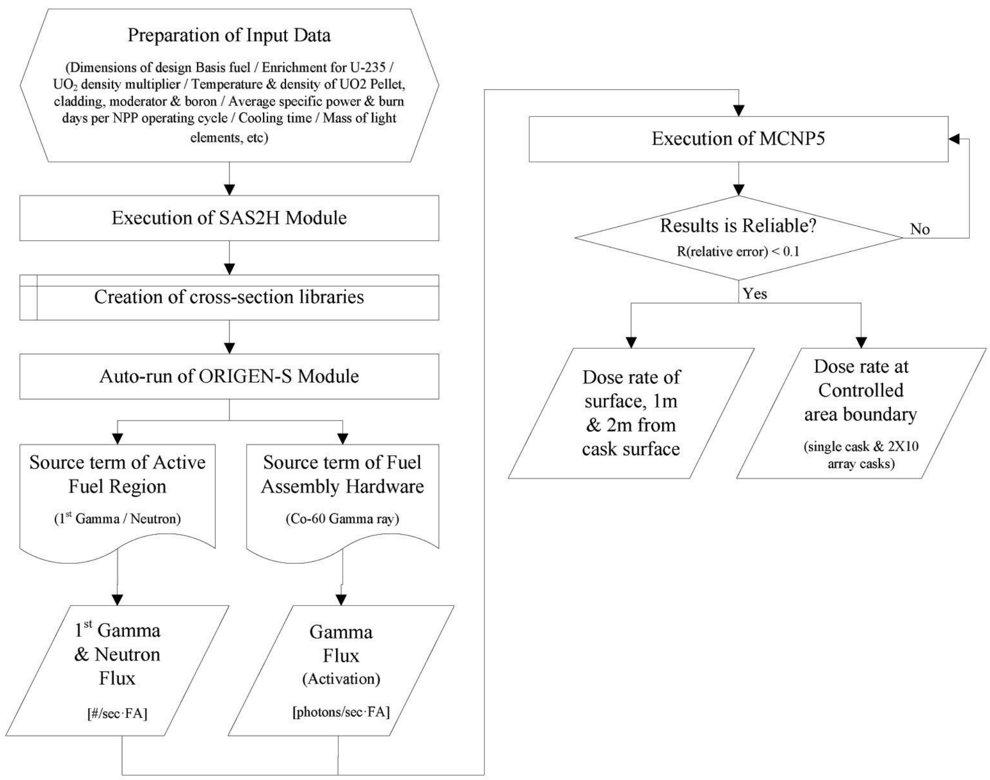


Fig. 11. Procedure for Design Basis Fuel's Source Term Evaluation and Cask's Shielding Analysis

higher at the lower side (from the bottom of cask to the bottom of the neutron shielding) of the cask, at over 2mSv/hr at the external surface of the cask.

The shielding analysis results of the 2x10 cask array showed that at approximately 280m from the array, the dose rate was 25mrem/yr, which is the limit for the public beyond the controlled area boundary. Actual dose rates in the controlled area boundary will be lower than 25mrem/yr, due to the decay of radioactivity of spent fuel in storage.

ACKNOWLEDGEMENTS

This study was supported by The Ministry of Commerce, Industry and Energy (MCIE), Korea, under the Radioactive Waste Management Technology Development Project.

REFERENCES

[1] NUREG-1536, Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility, US NRC, July 2010.
[2] NRC Docket No. 72.1008, “Final Safety Analysis Report for the HI-STAR 100 Cask System. Rev.0”
[3] NRC Docket No. 72.1014, “Final Safety Analysis Report for the HI-STORM 100 Cask System. Rev.4”, June, 2006.
[4] ORNL/TM-2005-39, SCALE A Modular Code for performing Standardized computer Analyses for Licensing Evaluation, Sect. 32 S2 “SAS2:A Coupled 1-D Depletion and Shielding Analysis Module.” Ver. 5.1
[5] A. G. Croff, M. A. Bjerke, G. W. Morrison, and L. M. Petrie, “Revised U-Pu Cycle PWR and BWR Models for the ORIGEN Computer Code”, Oak Ridge National Laboratory Report ORNL-6051, 1978.
[6] NUREG/CR-6802, Recommendations for Shielding Evaluations for Transport and Storage Packages, Oak Ridge National Laboratory, US NRC, May 2010.
[7] LA-UR-03-1987, MCNP – A General Monte Carlo N Particle Transport Code, Version 5, Release 1.40, 2005
[8] YAEC-1937, Axial Burnup Profile Database Pressurized Water Reactors, Yankee Atomic Electric Company, Bolton, Massachusetts, May 1997.
[9] Conversion Coefficients for Use in Radiological Protection against External Radiation, Annals of the ICRP Publication 74, 1996.
[10] 10 CFR Part 72, Licensing requirements for the independent storage of spent nuclear fuel and high-level radioactive waste
[11] NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facility, US NRC, 2000.