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Preliminary Shielding Analysis of the Concrete Cask for Spent Nuclear Fuel Under Dry Storage Conditions

건식저장조건의 사용후핵연료 콘크리트 저장용기 예비 방사선 차폐 평가

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The Korea Radioactive Waste Agency (KORAD) has developed a concrete cask for the dry storage of spent nuclear fuel that has been generated by domestic light-water reactors. During long-term storage of spent nuclear fuel in concrete casks kept in dry conditions, the integrity of the concrete cask and spent nuclear fuel must be maintained. In addition, the radiation dose rate must not exceed the storage facility's design standards. A suitable shielding design for radiation protection must be in place for the dry storage facilities of spent nuclear fuel under normal and accident conditions. Evaluation results show that the appropriate distance to the annual dose rate of 0.25 mSv for ordinary citizens is approximately 230 m. For a 2×10 arrangement within storage facilities, rollover accidents are assumed to have occurred while transferring one additional storage cask, with the bottom of the cask facing the controlled area boundary. The dose rates of 12.81 and 1.28 mSv were calculated at 100 m and 230 m from the outermost cask in the 2×10 arrangement. Therefore, a spent nuclear fuel concrete cask and storage facilities maintain radiological safety if the distance to the appropriately assessed controlled area boundary is ensured. In the future, the results of this study will be useful for the design and operation of nuclear power plant on-site storage or intermediate storage facilities based on the spent fuel management strategy.

Keywords: PWR spent nuclear fuel, Concrete cask, Dry storage condition, Radiation shielding analysis, Interim storage facility.

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한국원자력환경공단에서는 국내 경수로 원전에서 발생된 사용후핵연료를 건식으로 저장할 수 있는 콘크리트 용기를 개발하였다. 본 저장용기는 사용후핵연료가 건식환경에서 장기간 저장되는 동안 용기 및 사용후핵연료의 건전성이 유지되며, 방사선량률이 저장시설의 설계기준을 초과하지 않도록 설계되어야 한다. 특히, 저장시설은 정상 및 사고조건에서 적절한 방사선 방호를 위한 차폐설계가 이루어져야 한다. 이를 위해 본 연구에서는 미국 10CFR72 및 10CFR20의 기술기준과 NRC의 표준심사지침 NUREG-1536에서 제시한 평가방법에 따라 건식저장조건하에서 단일 콘크리트용기 및 2×10 용기배열조건의 선량율을 평가하였다. 평가결과, 일반인에 대한 연간선량 한도인 0.25 mSv를 만족하는 통제구역 경계까지의 거리는 약 230 m로 도출되었다. 콘크리트 저장용기의 설계사고는 2×10 배열의 저장시설에서 한 개의 저장용기가 이송 중 전도사고가 발생하여 용기의 바닥면이 통제구역 경계로 향하는 상황으로 가정하였다. 전도된 저장용기의 바닥면으로 부터 100 m 및 230 m 지점에서 각각 12.81 mSv 및 1.28 mSv로 평가되었다. 본 연구를 통해 건식저장조건에서 콘크리트 저장용기 및 저장시설은 적절하게 평가된 통제구역경계까지의 거리가 확보된다면 방사선적 안전성이 유지됨을 확인할 수 있었다. 본 평가결과만으로 건식환경의 저장용기(시설) 설계에 직접 적용하기는 어렵겠으나, 향후 '국가 고준위폐기물 관리 전략' 에 근거한 원전내 저장시설 또는 중간저장 시설의 설계 및 운영에 유용한 자료가 될 것으로 사료된다.

중심단어: 경수로 사용후핵연료, 콘크리트 저장용기, 중간저장시설, 건식저장조건, 방사선 차폐해석

1. Introduction

The Korea Radioactive Waste Agency (KORAD) has developed a concrete storage cask for the dry storage of spent nuclear fuel that has been generated by domestic light-water reactors [1]. During long-term storage of spent nuclear fuel in concrete casks kept in dry conditions, the integrity of the concrete cask and spent nuclear fuel must be maintained. The radiation dose rate must not exceed the storage facility's design standards. A suitable shielding design for radiation protection must be in place for the dry storage facilities of spent nuclear fuel under normal and accident conditions. The controlled area boundary where spent nuclear fuel is handled and stored must be situated at least 100 m away from the storage pad's casks. Furthermore, in terms of the shielding assessment and radiation protection, the As Low As Reasonably Achievable (ALARA) requirement must be considered for protecting radiation workers and the general public against radiation exposure [2].

This paper describes the assessment and results of the dose rate limit requirement set by code and standard 10CFR72 and 10CFR20, as well as the assessment method presented by the United States NRC standard review plan pertaining to the 2×10 arrangement for multiple concrete casks and single casks [3]. The analysis design requirements for safety assessment for radiological protection for dry storage casks and facilities of spent nuclear fuel under the Standard Review Plan are as follows.

a. A 100 m minimum distance must be maintained from the controlled area boundary to the spent fuel storage facility.
b. Never surpass an annual dose rate of 0.25 mSv in the controlled area boundary.
c. In accident conditions, never surpass an annual personal dose rate of 5 rem in the nearby controlled area boundary.
d. In normal and off-normal conditions, the annual dose rate for ordinary citizens near the controlled area boundary must not surpass 0.25 mSv, 0.75 mSv, 0.25 mSv for each whole-body, thyroid gland, body tissue, respectively.

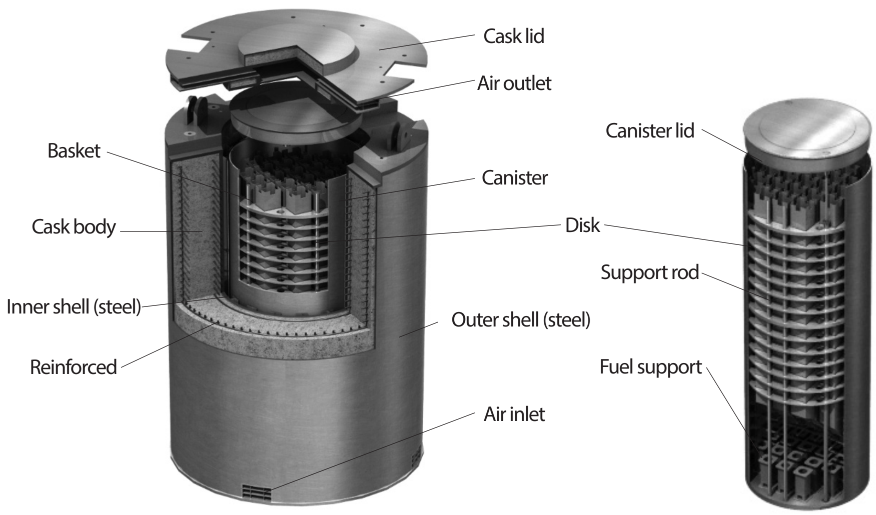


Fig. 1. The conceptual design of the concrete cask and canister.

2. Materials and methods

2.1 Source specification

To interpret the radiation shielding analysis of the concrete cask for spent nuclear fuel, a radiation source specification evaluation of the design basis fuel was executed. For radiation protection, radiation sources of spent nuclear fuel are as follows.

○ Gamma Rays

· Primary gamma rays occur from the radioactive decay of fission products and actinides.

· During nuclear reactor operation, activation of the fuel assembly structure results in ^{60}Co Radioisotope gamma rays.

· Reaction of neutron capture by fissile and non-fissile material results in secondary gamma rays.

○ Neutrons

· Neutrons resulting from spontaneous fission

· Neutrons resulting from (α, n) reactions

· Delayed neutrons

The Westinghouse ('WH') and Combustion Engineering ('CE') type fuel assembly has a cooling period of more than 10 years. Design basis fuels are divided into two categories: gamma-ray and neutron flux emitted from the efficient (fuel) region and radioactivity due to the radiation of the fuel assembly structural material. Therefore, we selected the fuel type source that emits the most radiation for each item.

Gamma ray flux resulting in effective fuel has the largest U-metal mass; source radiation was selected from WH type 17 RFA fuel, which releases the most radiation. The structure area of WH type fuel is relatively larger in mass than that of CE. CE type PLUS7™ standard/guardian fuel has a large quantity of ^{59}Co. Among these, the largest value was applied to each area. Therefore, the design basis fuel and fuel assembly structure that applies to each type of spent nuclear fuel (WH and CE) was selected and a shielding evaluation was applied. Characteristics of the design basis fuel using source specifications and shielding analysis are listed in Table 1.

The following are the assumed values for calculating the source radiation of the design basis fuel.

Table 1. Characteristic of the design basis fuel assembly

Item		Specification
Enrichment of initial ^{235}U [w/o]		4.5
Burnup [MWD/MTU]		45,000
Cooling time [years]		10
U-metal mass [kg]		461.5
UO_2 mass [kg]		523.55
Fuel rod	material of pellet	UO_2
	material of cladding	Zircaloy-4
	density of pellet [g·cm^{-3}]	10.286
Guide / Instrument tube		Zircaloy-4
Mass of the fuel assembly Structural material [kg]	weight of Zircaloy per F/A	139.30
	weight of stainless steel per F/A	20.95
	weight of Inconel per F/A	6.70

○ Initial ^{235}U enrichment: 4.5 w/o
○ Burn up: 45,000 MWD/MTU
○ Cooling time: 10 years
○ Burning period in the reactor: 3cycle burning

Table 2 shows primary gamma rays occurring from the radioactive decay of fission products and actinides by each energy section (18 groups) and the energy section of neutron flux (27 groups).

2.2 Calculation model design

The presumed factors for fuel assembly modeling are as follows:

○ Rectangular homogenization model application of effective fuel area and upper/lower structural components
○ Conservative fuel assembly design applied
· Width of WH 17 RFA nuclear fuel specifications were applied to the effective fuel area and the height in the CE type nuclear fuel specifications was applied.
· Upper/lower structural components were applied to CE type fuel data because there was less ^{59}Co content and longer than with the WH type fuel length.
○ Effective fuel area: Composed of UO_2 fuel rods and cladding

The internal canister of the concrete cask has a capacity of 21 spent nuclear fuel assemblies. A basket assembly that can accommodate dissemblance of the fuel assembly is installed. The thickness of the top cover of the canister is 240 mm. The thickness of the shell, which functions as an extra shield, is 25 mm. The basket cell in the internal canister, basket supporting disk, and tie rod were designed on the basis of nominal dimensions. The neutron absorber length of the external basket shell is 4,390 mm, and it is longer than the effective fuel area of the design basis fuel

Table 2. Gamma flux and neutron flux of the active fuel region

Gamma flux		Neutron flux	
Energy range [MeV]	$Gamma \cdot sec^{-1} \cdot FA^{-1}$	Energy range [MeV]	$Neutrons \cdot sec^{-1} \cdot FA^{-1}$
0.01 ~ 0.05	9.56×10^{14}	1.00×10^{-11}~1.00×10^{-8}	8.40×10^{-5}
0.05 ~ 0.10	2.67×10^{14}	1.00×10^{-8}~3.00×10^{-8}	3.32×10^{-4}
0.10 ~ 0.20	1.94×10^{14}	3.00×10^{-8}~5.00×10^{-8}	4.70×10^{-4}
0.20 ~ 0.30	5.71×10^{13}	5.00×10^{-8}~1.00×10^{-7}	1.60×10^{-3}
0.30 ~ 0.40	3.74×10^{13}	1.00×10^{-7}~2.25×10^{-7}	5.89×10^{-3}
0.40 ~ 0.60	1.53×10^{14}	2.25×10^{-7}~3.25×10^{-7}	6.16×10^{-3}
0.60 ~ 0.80	1.72×10^{15}	3.25×10^{-7}~4.00×10^{-7}	5.31×10^{-3}
0.80 ~ 1.00	7.96×10^{13}	4.00×10^{-7}~8.00×10^{-7}	3.62×10^{-2}
1.00 ~ 1.33	4.60×10^{13}	8.00×10^{-7}~1.00×10^{-6}	2.23×10^{-2}
1.33 ~ 1.66	6.18×10^{12}	1.00×10^{-6}~1.13×10^{-6}	1.57×10^{-2}
1.66 ~ 2.00	1.26×10^{11}	1.13×10^{-6}~1.30×10^{-6}	2.20×10^{-2}
2.00 ~ 2.50	4.65×10^{10}	1.30×10^{-6}~1.77×10^{-6}	6.82×10^{-2}
2.50 ~ 3.00	2.83×10^{9}	1.77×10^{-6}~3.05×10^{-6}	2.32×10^{-1}
3.00 ~ 4.00	2.68×10^{8}	3.05×10^{-6}~1.00×10^{-5}	2.05
4.00 ~ 5.00	7.26×10^{6}	1.00×10^{-5}~3.00×10^{-5}	1.04×10
5.00 ~ 6.50	2.92×10^{6}	3.00×10^{-5}~1.00×10^{-4}	6.57×10
6.50 ~ 8.00	5.72×10^{5}	1.00×10^{-4}~5.50×10^{-4}	9.35×10^{2}
8.00 ~ 10.0	1.21×10^{5}	5.50×10^{-4}~3.00×10^{-3}	1.19×10^{4}
Total	3.52×101^{5}	3.00×10^{-3}~1.70×10^{-2}	1.61×10^{5}
		1.70×10^{-2}~1.00×10^{-1}	2.24×10^{6}
		1.00×10^{-1}~4.00×10^{-1}	1.52×10^{7}
		4.00×10^{-1}~9.00×10^{-1}	3.31×10^{7}
		9.00×10^{-1}~1.40	3.31×10^{7}
		1.40~1.85	2.66×10^{7}
		1.85~3.00	4.99×10^{7}
		3.00~6.43	4.51×10^{7}
		6.43~2.00×10	4.25×10^{6}
		Total	2.10×10^{8}

Table 3. Chemical composition of canister components

Item	Material	Density [g·cm^{-3}]	Nuclide	Weight Fraction
Neutron Absorber	B_4C	2.646	^{10}B	0.04113
			^{11}B	0.18192
			C	0.06195
			Al	0.71500
Basket, Basket Disk, Disk rods	SA-240 Type 304	8.03	C	0.00080
			N	0.00100
			Si	0.00750
			P	0.00045
			S	0.00030
			Cr	0.19000
			Mn	0.02000
			Fe	0.68745
			Ni	0.09250
Canister Shell, Canister lid	SA-240 Type 316 L	8.03	C	0.00030
			N	0.00100
			Si	0.00750
			P	0.00045
			S	0.00030
			Cr	0.17000
			Mn	0.02000
			Fe	0.65545
			Ni	0.12000
			Mo	0.02500

(CE type, 3,810 mm). The thickness of the neutron absorber is 3 mm. Material properties used for the canister shell, top cover, and basket assembly are listed in Table 3.

The structural composition is as follows: the total height of the concrete cask is 6,030 mm, the thickness of the side concrete shielding wall is 625 mm, the thicknesses of the internal and external concrete casing are 50 mm and 25 mm, and the thicknesses of the upper lid and lower base is 200 mm and 480 mm, respectively. Fig. 2 and Fig. 3 represent the vertical/horizontal cross-sectional drawing of normal condition modeling using 2 dimensional plotters of MCNP5; Table 4 represents the material construction of each section of the model used to build the concrete casks. There is an air path inside the cask for heat release and there are four entrances/exits each for the inflow and release of air on the upper and lower sides of the cask (Fig. 4). In particular, there is a shield grid made of 6 mm thick steel to decrease the release of radiation created at the air entrances/exits.

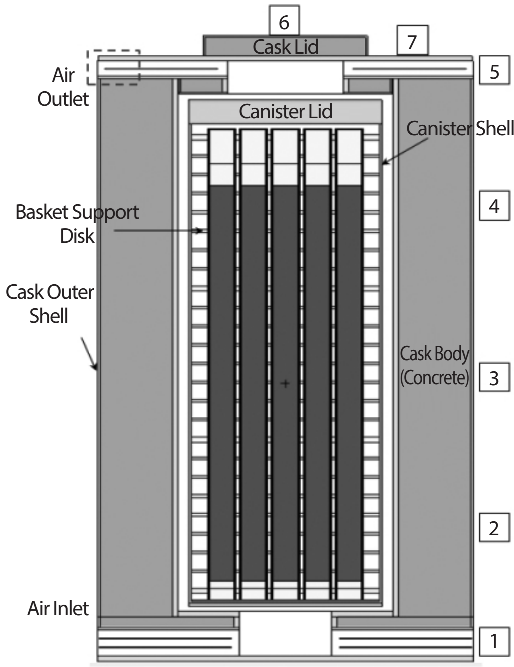


Fig. 2. The vertical cross section of the concrete cask.

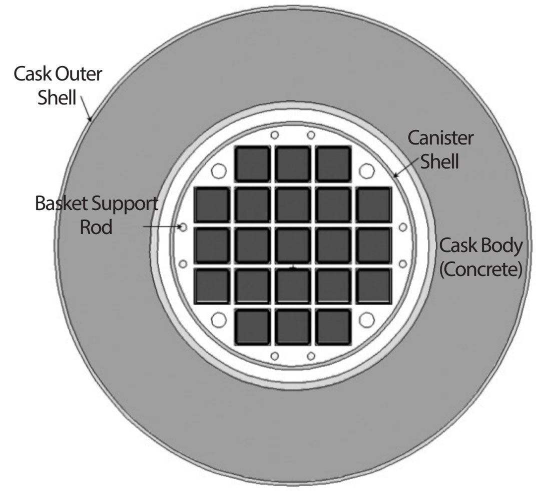
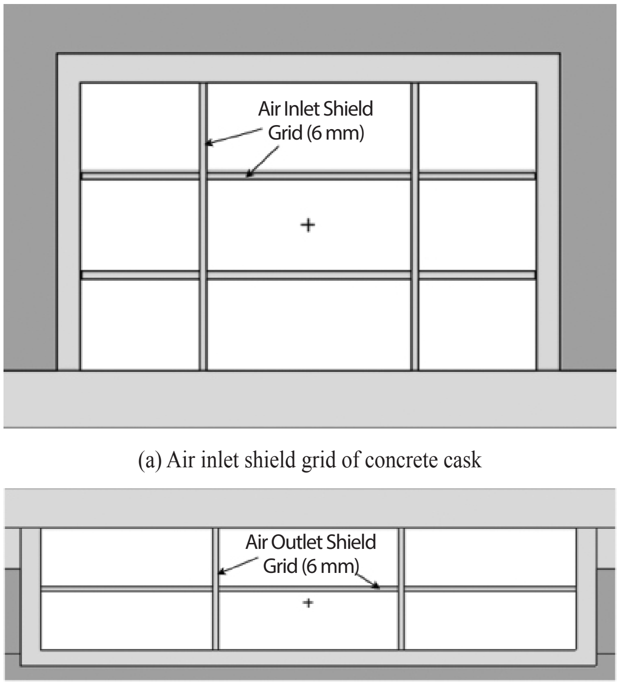


Fig. 3. The horizontal cross section of the concrete cask.

Table 4. Chemical composition of cask components

Item	Material	Density [g·cm^{-3}]	Nuclide	Weight fraction
Cask Body	Concrete	2.30	H	0.010
			O	0.532
			Na	0.029
			Al	0.034
			Si	0.337
			Ca	0.044
			Fe	0.014
Cask Casing, Air Inlet/Outlet Shield Grid, Cask Lid Casing	SA-36	7.75	C	0.0026
			Mn	0.0010
			P	0.0004
			S	0.0005
			Si	0.00275
			Cu	0.0020
			Fe	0.98175



(a) Air inlet shield grid of concrete cask

(b) Air outlet shield grid of concrete cask

Fig. 4. The shield grids of the air inlet and outlet for the concrete cask.

2.3 Analysis method

To minimize radiation injury caused by possible radiation exposure, design standards of the storage facility consider several factors: the storage cask arrangement and form, the daily duties of the worker within the controlled area (transportation and installation of storage casks, visual inspection, monitoring radiation, as well as maintenance and repair, etc.), work procedures, and the number of workers. These types of storage facility design requirements and dose limits can be regulated by the surface dose rate of single storage casks. Therefore an assessment of the surface dose rate of casks should be conducted before the shielding assessment of storage facilities and single storage casks. Following the recommendations of the standard screening guidelines from the U.S. regulatory authority, the US Nuclear Regulatory Commission (NUREG-1536), under the conditions of an arrangement of at least 20 storage casks (normally a 2×10 array), the radiation dose rate should be calculated from the outermost storage cask and correspond to the annual radiation dose rate limit of the controlled area boundary.

2.3.1 Single cask under normal conditions

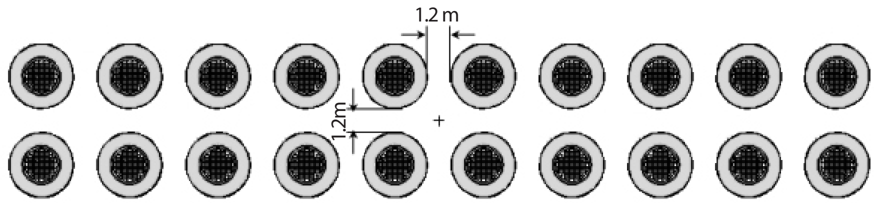
Conditions in which a concrete cask is kept vertically above the storage facilities pad is assumed because concrete casks are designed for the sole purpose of storing spent nuclear fuel. The radiation dose rate of single casks is calculated 1~2 m from the cask's surface according to the daily duties of the radiation workers, such as transportation and installation of storage casks, monitoring radiation, maintenance, other related tasks.

2.3.2 2×10 array casks under normal condition

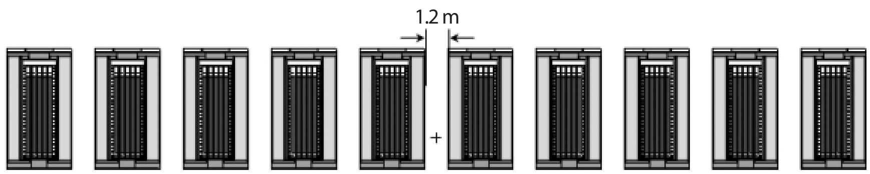
The distance between the spent nuclear fuel storage facility's controlled area boundary must be at least 100 m from the storage facility's outermost storage cask. The shielding assessment of the spent nuclear fuel storage facility is performed with at least 20 casks (2×10 array) (Fig. 5). After evaluating the individual distance to the radiation dose rate, the distance to the controlled area is calculated while the annual dose rate limit (0.25 mSv). The storage facility's radiation dose rate assessment is conducted according to the center point of the 20 casks or the pivot point (2.5 m above the storage pad). From that point, the annual (8,760 hours) dose rate is calculated from the outermost cask to the general public located at the controlled area boundary.

2.3.3 2×10 array casks under a design basis accident

Accidental conditions are assumed to occur to the single storage cask, and when an accident occurs, the damage is assumed to be restricted and localized to the outer surface of the cask. Therefore, the impact of radiation occurring under accidental conditions has an insignificant influence on the design standards because of the considerable distance (at least 100 m) between the storage casks and the controlled area boundary and can therefore be disregarded [4][5].

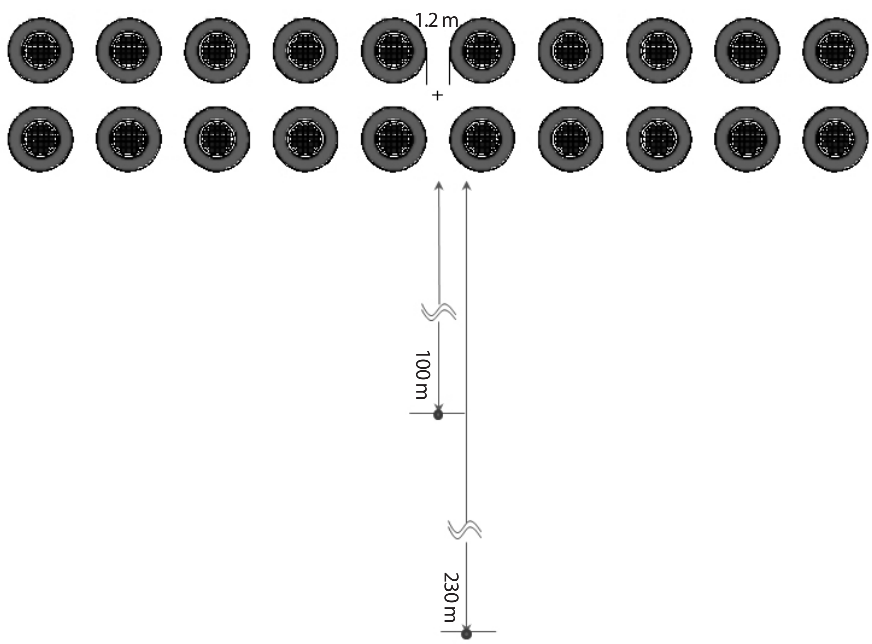


(a) The horizontal cross section of 2×10 array casks

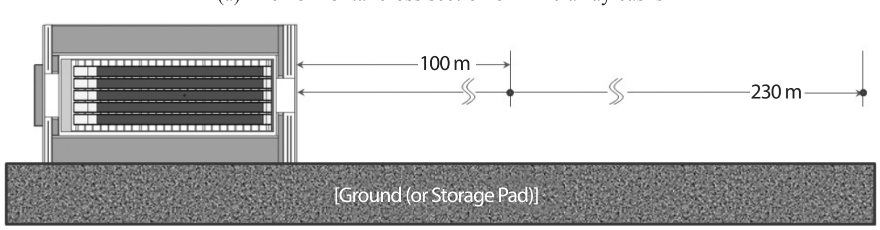


(b) The vertical cross section of 2×10 array casks

Fig. 5. The horizontal and vertical cross section of 2×10 array casks.



(a) The horizontal cross section of 2×10 array casks



(b) Accident by cask rollover

Fig. 6. The shielding analysis model for rollover of a concrete cask.

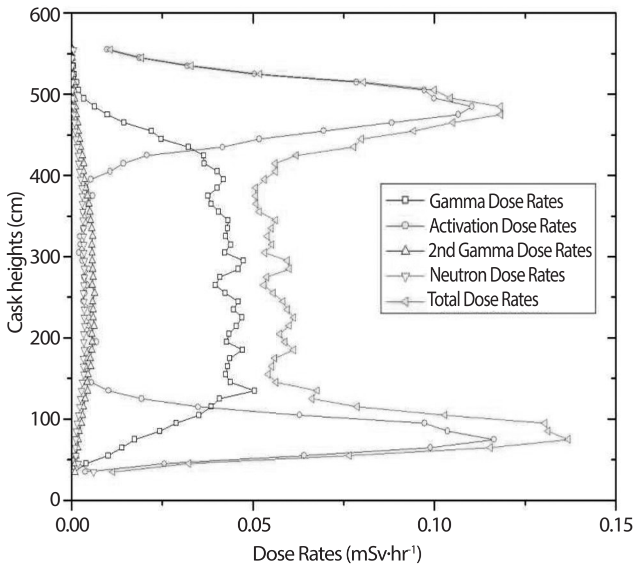


Fig. 7. Dose rates at the external surface of the concrete cask.

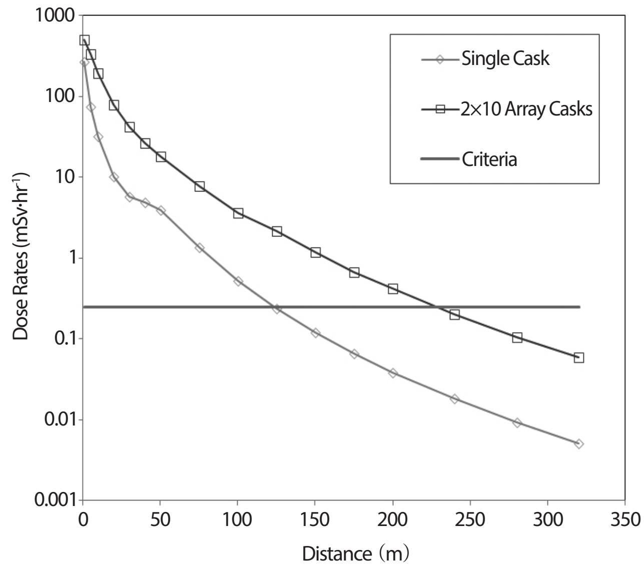


Fig. 8. Annual dose versus distance for single and 2×10 arrays.

Table 5. Result of shielding analysis of a single cask surface

(mSv·h^{-1})

Position		Surface	1 m	2 m
Cask side surface	1	0.4930	0.3400	0.2068
	2	0.1029	0.0476	0.0328
	3	0.0529	0.0343	0.0260
	4	0.1183	0.0405	0.0230
	5	0.0087	0.0179	0.0151
Top of the cask	6	0.0377	0.0136	0.0062
	7	0.0384	0.0101	0.0052

In this paper, rollover accidents, which represent the biggest radiation effects in the handling of concrete casks, were selected as accident conditions. It is assumed that a rollover occurs at the center of the 2×10 array in the process of transferring the cask to the storage pad.

Radioactive safety is assessed by determining whether the dose rate has exceeded 50 mSv for 30 days within the controlled area boundary (Fig. 6).

3. Result and discussion

The calculated cask surface dose rate for a single concrete cask at the air inflow entrance was 0.493 mSv·h^{-1}, the largest value among all the parts. The side of the cask is divided into upper, middle, lower (Fig. 2, Points 4, 3, 2). The calculated dose rate for each point was 0.0777, 0.0529, and 0.0662 mSv·h^{-1}, respectively. However, the dose rate from the top and bottom parts of the storage cask is largely influenced by radiated gamma rays coming from structural materials of the top and bottom of the fuel assembly. Gamma rays occurring from the active fuel region contribute 80% to the dose rate of the middle part of the cask (Fig. 7). The dose rate from the top cover of the concrete cask at the middle and the absence of the concrete part was respectively 0.0377 mSv·h^{-1} and 0.0384 mSv·h^{-1} (Table 5).

When evaluating the concrete cask in a 2×10 arranged condition, the distance between each cask was assumed as 1.2 m. Considering the case analysis of a foreign spent fuel storage facility and the facility operation aspects, a distance between the casks of 1.2 m was applied [6].

To consider the sky-shine effect, the air boundary was

Table 6. Result of shielding analysis of single cask surface

($mSv \cdot 30day^{-1}$)

Item	Dose results (Cask bottom)	
	100 m	230 m
Design basis accident by cask rollover	12.81	1.28
Criteria		50.0

taken into consideration. Twenty casks were arranged and the dose rate was calculated from the middle of the casks to the controlled area boundary. Evaluation results show that the appropriate distance to the annual dose rate of 0.25 mSv for ordinary citizens is approximately 230 m. Fig. 8 shows the dose rate by the controlled area boundary distance under the single or arranged casks conditions.

For a 2×10 arrangement within storage facilities, rollover accidents are assumed to have occurred during the transfer of one additional storage cask, with the bottom of the cask facing the controlled area boundary. Dose rates were calculated for accidents occurring in the same location with the minimum separation distance of 100 m under 10CFR72 and 230 m, which was derived from shielding analysis. Dose rates of 12.81 and 1.28 mSv were calculated at 100 m and 230 m, respectively, from the outermost cask in the 2×10 arrangement, and the results are shown in Table 6.

Source specifications of a concrete cask for spent nuclear fuel must not surpass the limit of the predesigned standard fuel amount. The actual value of the annual radiation dose rate must be smaller than 0.25 mSv within 230 m of the controlled area boundary. The radiation dose result for design basis accidents was determined to be about 25.6% of the individual dose limit of 50 mSv at the control zone boundary.

4. Conclusion

The radiation shielding analysis results of a concrete cask comply with U.S. 10CFR72 and NUREG-1536 revealed that the radiation dose rate satisfies the dose rate limit in an arrangement of 2×10 casks and an accident condition. This study, for the first time in Korea, provides evaluation methodology and results for normal and design basis accidents in dry conditions of the spent fuel storage cask. And, it will be helpful to assist in making a 'Safety Analysis Report' in accordance with the recently announced NSSC Notice No.2015-20. Korea does not yet have clear plans and information on dry storage of PWR nuclear fuel. Therefore, this study performed only a shielding analysis for limited conditions with information on nuclear fuel and casks. So it would be difficult to apply this result directly to the domestic situation (e.g., a dry storage facility construction plan) which has not yet been confirmed. As further shielding analysis of various dry storage conditions progresses, the analysis methodology can be validated. In particular, if more precise environmental conditions are presented in the 'Accident Evaluation' described in the cask's 'Safety Analysis Report', the shielding analysis method and the results of this study will be of value. Thus, in the future, the results of this study will be useful for the design and operation of on-site storage facilities or intermediate storage facilities based on a spent fuel management strategy.

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