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Shielding Design of Shipping Cask for 4 PWR Spent Fuel Assemblies

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PWR집합체 4개 장전용 수송용기의 차폐설계

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Abstract

A Shielding analysis of the shipping cask designed conceptually, of which shielding material are lead and resin, for containing 4 PWR spent fuel assemblies, has been made with the help of a computer code, ANISN. The shielding materials being used in the cask have been selected and arranged to minimize cask weight while maintaining an overall shielding effectiveness. Radiation source terms have been calculated by means of ORIGEN-2 code under the assumptions of 38,000 MWD/MTU burnup and 3-year cooling time. A calculation of gamma-ray and neutron dose rates on the cask surface and 1 m from the surface has been done.

It is revealed that the total dose rates under the normal transport and hypothetical accident conditions meet the standards specified.

요　　약

PWR사용 후 핵연료 집합체 4개를 장전할 수 있는 납/Resin 차폐체형 수송용기에 대한 방사선 차폐해석을 수행하였다. 이때 차폐효과를 유지하면서도 전체중량이 최소화되도록 차폐재를 선택하였다. 방사선원은 ORIGEN 전산코드로 계산하여 얻었으며, 사용후 핵연료의 연소도를 38,000 MWD/MTU 그리고 냉각기간을 3년으로 가정하였다. 수송용기의 외부 표면에서 1m거리에서 나타나는 감마선 그리고 중성자의 선량율은 ANISN전산코드로 계산하여 얻었다. 계산된 총방사선 선량율은 정상 및 가상 사고조건하에서도 국내 법규에 규정된 기준치이내에서 만족하는 것으로 나타났다.

1. Introduction

The need for transportation of radioactive spent fuels is increasing with the capacity of nuclear power reactors being installed. After removal from the core the irradiated fuels are stored at the on-site storage pool for a period of time in which short-lived fission products decay to the level at which the fuel can be shipped. The on-site storage capcity at some of nuclear power plants is expected to be full sooner or later in Korea. With growing inventory of spent fuels stored at power plants throughout the country it is necessary to transport those spent fuels to other storage facility. Only one cask which had been designed and fabricated for 1 PWR spent fuel asembly shipment,[1] which is not suitable in the viewpoint of the transport capacity and economy. Therefore, a large capacity cask is needed with consideration of transport route conditions. The optimized large spent fuel casks have been studied in other countries since early 1980's.[2]

Spent fuels to be shipped in the cask have in usual high radioactivity. Therefore, all shipping casks should be designed to protect from radiation. The external radiation level of the cask from the spent fuel must be limited to meet standards specified by the regulation. Primary reliance for safety in transport of spent fuels is placed on the cask.

A number of spent fuel casks in use had to be designed to handle a fairly large amount of decay heat and to shield very intensive neutron and gamma-ray based on less than 1 year cooling time. Frequently it required the use of very thick neutron and gamma shields whose size and weight further limited to the transport and quantity of fuel per shipment.

The purpose of this study is to provide an information for the optimum shielding design by calculating the total gamma-ray and neutron dose rates outsite the cask under normal transport and hypothetical accident conditions. The calculation was based on 4-PWR 17×17 spent fuel assemblies with 3-year cooling time.

2. Source Term Calcuations

The radiation source term of the spent fuel is necessary for the multigroup shielding analysis. For the typical PWR in equilibrium, fresh fuel at 3.2 wt.% U-235 is loaded. It is assumed that the PWR fuel was irradiated at a specific power of 37.5 KW/KgU for 1,013 full power days during 3-year period, resulting in an average discharge burnup of 38,000 MWD/MTU.

The ORIGEN-2 code[3] was used to calculate the buildup of the various fission products, activation products, and higher-order actinides during the irradiation. The fuels under consideration in this study are the assemblies of Uljin unit 1 & 2 with the average burnup of 37,000 MWD/MTU.[4] Each PWR fuel assembly was assumed to be contained initially 0.46 metric tonnes of heavy metal and to have a maximum burnup 38,000 MWD/MTU.

ORIGIN-2 generates an 18-group gamma-ray source spectrum for fission products, activation products of light structure materials and actinides of heavy metals. Each of these was weighted by the corresponding yield to obtain the net gamma-ray spectrum. The gamma-ray source spectrum from 4-PWR assemblies with 3-year cooling time is shown in Table 1.

Unfortunately, the ORIGEN-2 group structure does not correspond exactly to the 18-group gamma-ray structure used in DLC-23/CASK cross section library.[5] To make the necesary transformation, the gamma-ray sources were considered to be constant within each ORIGEN energy group It is assumed that lower energy 1-4 groups of ORIGEN structure are transformed to 18 group of DLC-23 structure because the lower groups less than 0 1 MeV is not important for the shielding analysis. Figure 1 shows the normalized gamma-ray spectra of DLC-23 and ORIGEN-2 energy group structure. The gamma-ray energy spectrum of DLC-23 normalized by 18 group is tabulated in Table 2.

The neutron sources are from either (α , n) reaction or spontaneous fisson of heavy metals. The (α , n) neutron source is almost entirely due to the Pu-238, Cm-242 and Cm-244. Likewise,

Table 1. Gamma-ray Energy Spectrum per 4-PWR Assemblies with 3-Year Cooling Time Calculated by ORIGEN-2.

Group	Mean Energy (MeV)	Photon Spectrum (Photons/sec) Total	F.P	L.S	H.M
1	0.015	1.49 +16	1.48 +16	4.98 +11	1.06 +14
2	0.025	3.47 +15	3.47 +15	1.35 +11	1.38 +12
3	0.0375	3.52 +15	3.52 +15	5.46 +10	4.58 +11
4	0.0575	3.05 +15	3.03 +15	3.90 +10	1.98 +13
5	0.085	2.09 +15	2.09 +15	2.01 +10	1.91 +12
6	0.125	2.27 +15	2.27 +15	1.29 +10	1.63 +12
7	0.225	1.82 +15	1.82 +15	7.14 +09	1.30 +12
8	0.375	1.06 +15	1.06 +15	2.65 +10	8.00 +11
9	0.575	1.65 +16	1.65 +16	5.50 +10	1.46 +09
10	0.85	5.05 +15	5.05 +15	2.90 +10	2.84 +09
11	1.25	9.06 +14	8.94 +14	1.22 +13	1.34 +09
12	1.75	4.91 +13	4.91 +13	9.96 +08	3.70 +08
13	2.25	5.44 +13	5.44 +13	6.37 +07	1.86 +08
14	2.75	1.33 +12	1.33 +12	1.96 +05	3.91 +08
15	3.5	1.68 +11	1.68 +11	—	9.70 +07
16	5.0	4.15 +07	—	—	4.15 +07
17	7.0	4.79 +06	—	—	4.79 +06
18	11.0	5.50 +05	—	—	5.50 +05
Total (P/s)		5.47 +16	5.46 +16	1.31 +13	1.33 +14

Table 2. Gamma-ray Energy Spectrum per 4-PWR Assemblies according to DLC-23 Energy Structure (38,000 MWD/MTU Burnup, 3-Year Cooling Time)

Group	Upper Energy (MeV)	Normalized Spectrum	Spectrum (Photons/s)
1	10.0	1.004 −11	5.495 +05
2	8.0	6.556 −11	3.587 +06
3	6.5	4.008 −10	2.193 +07
4	5.0	3.798 −10	2.074 +07
5	4.0	3.074 −06	1.682 +11
6	3.0	2.423 −06	1.327 +11
7	2.5	9.398 −04	5.438 +13
8	2.0	6.098 −04	3.337 +13
9	1.66	5.918 −03	3.238 +14
10	1.33	0.0109	5.981 +14
11	1.0	0.0616	3.367 +15
12	0.8	0.1515	8.289 +15
13	0.6	0.1875	1.026 +16
14	0.4	0.0129	7.060 +14
15	0.3	0.0222	1.215 +15
16	0.2	0.0527	2.882 +15
17	0.1	0.0832	2.091 +15
18	0.05	0.455	2.490 +16
Total		1.000	5.427 +16

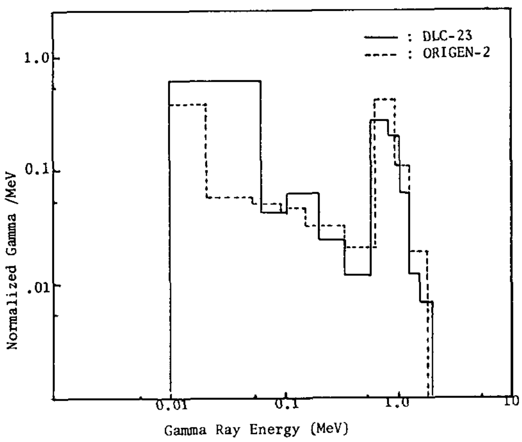

Fig. 1. Normalized Gamma Spectrum of DLC-23 and ORIGEN-2 Energy Structure.

the spontaneous fission neutron source is almost due to the Cm-242, Cm-244 and Cm-246. The energy spectrum of both neutron sources is similar to that of Cf-252 fission in the PWR spent fuel.[6] The resulting 22-group neutron spectrum to be used in the multigroup shileding calculations is

Table 3. Neutron Source Spectrum per 4-PWR Assemblies according to LDC-23 Energy Structure (38,000 MWD/MTU Burnup, 3-Year Cooling Time)

Group	Upper Energy (MeV)	Normalized Spectrum	Spectrum (Neutrons/s)
1	14.92	0.4653 −3	4.6004 +5
2	12.2	0.1883 −2	1.8617 +6
3	10.0	0.5756 −2	5.6910 +6
4	8.18	0.01924	1.9023 +7
5	6.36	0.04000	3.9548 +7
6	4.96	0.05174	5.1155 +7
7	4.06	0.1094	1.0816 +8
8	3.10	0.08804	8.7045 +7
9	2.46	0.02088	2.0644 +7
10	2.35	0.1156	1.1429 +8
11	1.83	0.2089	2.0653 +8
12	1.11	0.1920	1.8983 +8
13	0.55	0.1327	1.312 +8
14	0.11	0.0135	1.3298 +7
15	3.35 −3	0.0	0.0
16	5.83 −4	0.0	0.0
17	1.01 −4	0.0	0.0
18	2.90 −5	0.0	0.0
19	1.01 −5	0.0	0.0
20	3.06 −6	0.0	0.0
21	1.12 −6	0.0	0.0
22	4.14 −7	0.0	0.0
Total		1.00	9.887 +8

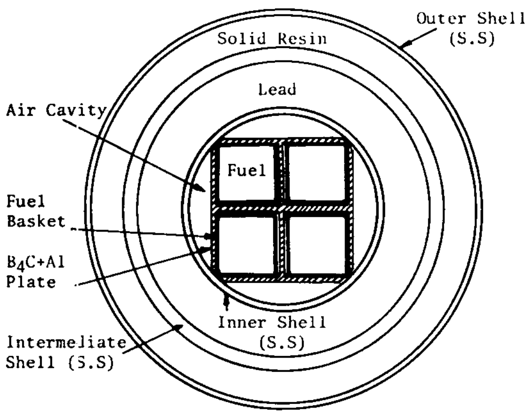

Fig. 2. Geometrical Model of Actual Cask for 4-PWR Spent Fuel Assemblies.

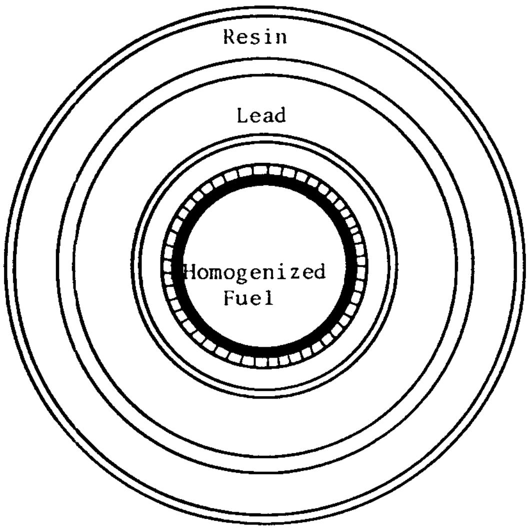

Fig. 3. Homogenized Model of Cask for Shielding Calculation.

shown in Table 3. The neutron source strength per 4-PWR assemblies at 3-year cooling time is 9.89×10^8 neutrons/sec.

3. Shielding Analysis

3.1. Calculation Model

The shipping cask for 4-PWR spent fuels has been designed for two kinds of shilelding materials: one is the high density material for gamma-ray source; the other is the light materials for neutron source. The shielding materials used in cask design have to be selected and arranged to minimize the cask weight while maintaining overall shield effectiveness.

Figure 2 described the conceptually designed model of the spent fuel shipping cask. Inner baskets which house the 4-PWR assemblies are placed in cask cavity. The cask body consists of inner, intermediate and outer stainless shells. The annulus between inner and intermdiate cylinders contains a lead for gamma-ray shielding. Neutron shielding is provided by a solid resin which surrounds the outer cylinder and axially blankets the active fuel region of fuel assemblies. The solid resin as the light materials to make efficiently fast neutrons slowdown. The 4-fuel baskets which measure 230 mm square with stainless steel of 6 mm

Table 4. Atom Number Densities of Homogenized Fuel Region and Shielding Materials (atoms/barn-cm)

Element \ Material		Homogen. Fuel	S.S.-304	Lead	B_4C+Al Plates	Solid Resin
H	1					5.950−2
B	5				1.471−2	
C	6				3.674−3	2.305−2
N	7					1.427−3
O	8	1.297 −2				2.674−2
Al	13				3.269−2	7.966−3
Cr	24	8.50 −6	1.674 −2			
Fe	26	3.07 −5	6.06 −2			
Ni	28	4.05 −3	9.88 −3			
Zr	40	1.566 −4				
Pb	82			0.033		
U-235	92	3.95 −5				
U-238	92	6.05 −2				
Pu	94	3.86 −5				

Table 5. Summary of Dose Rates under the Normal and Accident Conditions (mrem/hr)

Dose rate	Normal condition		Accident
Contents	Surface	1 m	1 m
Primary Gamma	17.6	6.5	28.8
Secondary Gamma	3.7	1.2	0.1
Neutron	4.8	1.7	151.9
Total	26.1	9.4	180.8
(Limit value)	(200.0)	(10.0)	(1,000.0)

thick are enough to load the 17×17 assembly. The baskets are also lined with plates made of sintered B_4C+Al for reducing the criticality. Internal fluid in the cask cavity is filled with air for dry shipment. The internal cavity is at least 706 mm in diameter.

Figure 3 is a calculational model of 4-PWR assemblies cask, which was homogenized for the simplified shielding calculations. The fuel region based on the equivalent circular cross section was assumed to be homogenized together with 4-PWR fuel assemblies. Table 4 gives the atom number densities in the fuel and shield material regions.

3.2. Shielding Calculations

To determine the optimal shielding layers which minimize the total weight of the shipping cask, a number of shielding calculations were performed. The ANISN[7] code was used to perform the one-dimensional multigroup neutron and gamma-ray transport calculations for the flux distribution throughout the cask materials. In this study, the P_3-S_8 approximation to the transport equation was employed with a 40-group(22-neutron, 18-gamma) of coupled neutron and gamma-ray energy spectrum. The multigroup neutron cross sections, the primary gamma-ray transport and the secondary gamma-ray production cross sections were coupled to form a 40 group set. A use of the P_3 cross section data found in DLC-23/CASK cross section library allowed us to account for the forward-direct anisotropic scattering terms which are characteristic of high-energy neutrons and gamma-rays, and which are quite important in deep-penetration shielding problems.

The flux-to-dose conversion factors[8] is used for obtaining the dose rates. The shielding calculations were carried out for a given type of lead and resin shield cask. In addition, the shielding requirement was imposed to be 40 tonnes or so in cask weight for road transport. Therefore, WEIGHT program was used to calculate the total weight of loaded cask using a rather simplified model.[2)]

4. Results and Discussion

In compliance with the regulation specified in the packaging and transport standards of radioactive materials, the pertinent radiation levels under the normal transport and hypothetical accident conditions are as follows:[9)]

200 millirem per hour at any point on the external surface of cask and 10 millirem per hour at 1 m from the external surface of cask under the normal conditions; 1,000 millirem per hour at 1 m from the external surface of cask under the accident conditions.

The dose rates calculated by the ANISN under the normal transport and accident conditions are given in Table 5. Here, the accident conditions were assumed to be loss of neutron shield caused by drop, impact and fire conditions. The dose rates were obtaind for the homogenized 4-PWR assemblies in the cask. The calculated results for the actual geometry model are slightly lower than those of the homogenized model, because the inner plates of fuel basket and B_4C plates divided between the baskets were not considered in the homogenized model.

The total dose rates shown in Table 5 under the normal condition are 26 mrem/hr at the external surface and 9.4 mrem/hr at 1 m from the surface of cask. Also the dose rate under the hypothetical accident conditions is 180.8 mrem/hr at 1 m from the surface of cask. The inner, intermediate and outer shells determined 10, 25.4 and 10 mm thick. The lead and resin thickness obtained are 155 and 136.6 mm. The resultant total weight of loaded cask seems to be approximately 38 tonnes.

5. Conclusion

All the calculated results for the normal transport and hypothetical accident conditions do not exceed the limites specified. It is believed that the conceptually designed 4-PWR spent fuel assemblies shipping cask which contains the lead for gamma-ray shielding and resin for neutron shileding is suitable in a viewpoint of shielding effectiveness. The information of shielding analysis make it possible to go through to a basic and detail design step of the PWR spent fuel shipping cask. This study provides also a basis for further development of the transport shipping cask.

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