J.Korean Asso. Radiat. Prot.
Vol.26 No.3 : 207-214(2001)

RADIATION SAFETY ASSESSMENT FOR KN-12 SPENT NUCLEAR FUEL TRANSPORT CASK USING MONTE CARLO SIMULATION

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Abstract - The KN-12 spent nuclear fuel (SNF) transport cask is designed for transportation of up to 12 assemblies and is in standby status for being licensed in accordance with Korea Atomic Energy Act. To evaluate radiation shielding and criticality safety of the KN-12 cask, each case of study was carried out using MCNP4B Code. MCNP code is verified by performing benchmark calculation for the KSC-4 SNF cask designed in 1989. As a result of radiation safety evaluation for the KN-12 cask, calculated dose rates always satisfied the standards at the cask surface, at 2 m from the surface in normal transport condition, and at 1 m from the surface in hypothetical accident condition. Maximum dose rate was always arisen on the side of the cask. For normal transport condition, photons primarily contribute to dose rate between two kinds of released sources, neutrons and photons, from spent nuclear fuel but for hypothetical accident condition, contrary case was resulted. The level of calculated dose rate was 27.8% of the limit at the cask surface, 89.3% at 2 m from the cask surface, and 25.1% at 1 m from the cask surface. For criticality analysis, keff resulting from the criticality analysis considering the condition of optimum partial flooding with fresh water is 0.89708(0.00065. The results confirm the standards recommended by all regulations on radiation safety.

INTRODUCTION

The KN-12 cask is a new facility designed to allow transportation up to 12 assemblies from PWRs in dry and wet conditions. The cask design should satisfy the regulation standards of IAEA Safety Standard Series No. ST-1 [1], US 10 CFR part 71 [2] and Korea Atomic Energy Act [3]. The containment system of the KN-12 cask consists of a forged thick-walled carbon steel cylindrical body with an integrally-welded carbon steel bottom and is closed by a lid made of stainless steel, which is fastened to the cask body by lid bolts. The steel thickness of the cask body wall and of the lid should meet the dose rate limits of the related regulations with neutron shielding material. General standards in radiation shielding analysis for SNF transport cask of IAEA, 10 CFR, and Korean Act are given as follows; the radiation level should not exceed 2 mSv/hr at any point on, 0.1 mSv/hr at 2 m from the surface of the transport cask in normal transport conditions, and 10 mSv/hr at 1 m from the surface of the transport cask in hypothetical accident conditions. For criticality safety analysis, it is recommended for keff not to exceed 0.95 at a 95% confidence level in the conservative condition given from NUREG-1617 [4].

In this context, radiation shielding and criticality analysis for the KN-12 cask were performed, using MCNP code in this study. Simultaneously, in order to benchmark MCNP run, radiation safety analysis was also performed for the KSC-4 cask and compared with ANISN and DOT4.2 run for shielding calculation and KENO-IV for criticality analysis.

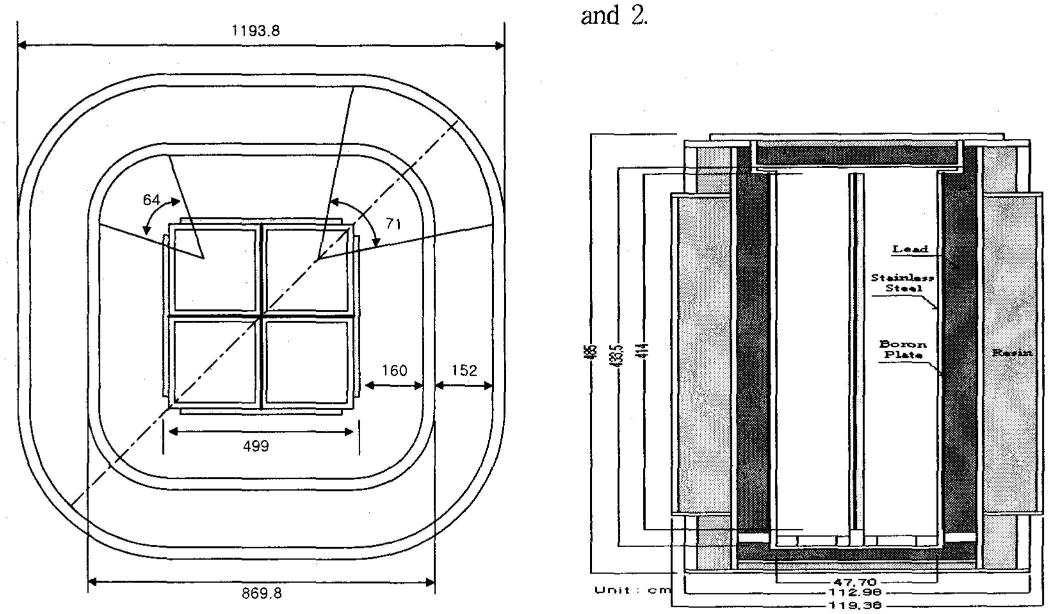
MCNP CODE VERIFICATION

To evaluate the cask, 1-D or 2-D SN codes, ANISN and DOT/DORT have been used for radiation shielding analysis. Since these codes have limitation to describe its geometry there are attemption to use 3-D SN code, TORT, and Monte Carlo code, MCNP, to describe the geometry more precisely [4]. The results from these codes are evaluated to be more reliable compared with the existing tool for 1-D or 2-D. Since MCNP uses continuous cross section library, it may give more accurate results. So, criticality analysis is also performed. MCNP code verification was carried out using the KSC-4 cask model that was designed by the Korea Atomic Energy Research Institute (KAERI) in 1989 for transportation up to 4 assemblies from PWRs. The KSC-4 cask was evaluated using ANISN for dose rate at the side surface and DOT4.2 for dose rate at the top and the bottom surfaces for radiation shielding analysis and KENO-IV for criticality analysis. KSC-4 cask has its shape of a square type but four edges are rounded. It is very difficult to describe it exactly by using 1-D or 20D.

In MCNP calculation for radiation shielding and criticality analyses, the same spent fuel assembly model of design report was used except the follows; for radiation shielding analysis, Westinghouse 17x17 PWR assembly of 3.2 w/o U-235 after 3 years on site cooling and for criticality analysis, Westinghouse 17x17 fresh PWR fuel of 3.3 w/o U-235.

1. Radiation Shielding Analysis

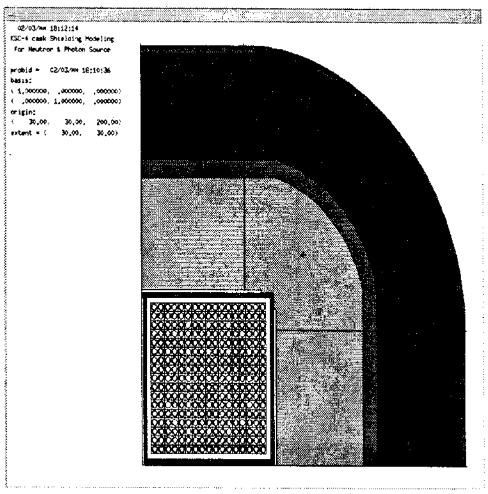
MCNP modeling for radiation shielding analysis is shown in Fig. 2. The calculation results are presented in Tables 1 and 2. In the cask side, the discrepancies between MCNP and ANISN results are mainly come from geometrical modeling. As shown in Fig. 2.3, the modeling for ANISN run was assumed as a circular type and it is expected that dose rate of the side surface has average value. Actually, it is expected that the plane surfaces of the cask have maximum dose rate and the edge surfaces have minimum dose rate as shown in Figs. 1 and 2.



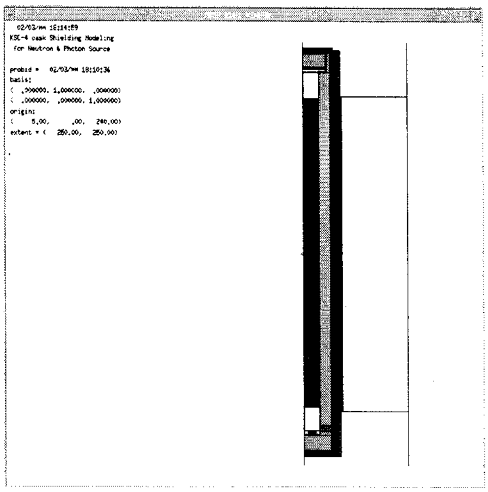
a) Radial Cross Sectional View　　b) Axial Cross Sectional View

Fig. 1. General Arrangement of KSC-4

Dose rate at the bottom and the top surfaces are dominated by neutrons due to thinner neutron shielding wall than in the side, and biased active fuel to the bottom gives much higher dose rate at the bottom. On the other hand, dose rate at the side surface is dominated by photons. At 2 m from the top and the bottom surfaces, dose rates by MCNP tend to be reduced by half compared with DOT4.2.



a) Radial Cross Sectional View of KSC-4



b) Axial Cross Sectional View of KSC-4

Fig. 2. KSC-4 Modeling for MCNP Run

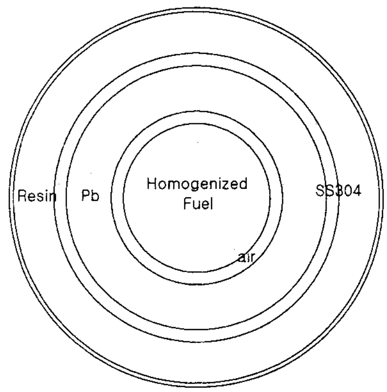


Fig. 3. KAERI Model of KSC-4 for ANISN and DOT4.2 Runs

Table 1. Maximum Dose Rate of KSC-4 at the Surface

[mSv/hr]

	Side		Top		Bottom	
	*MCNP	**ANISN	MCNP	**DOT4.2	MCNP	DOT4.2
Neutron	0.052	0.034	0.265	0.206	0.580	0.577
Gamma	0.221	0.176	0.047	0.023	0.047	0.044
Total	0.283	0.21	0.312	0.229	0.627	0.621

*Hanyang Univ. Model [8]
**KAERI Model [6]

Table 2. Maximum Dose Rate of KSC-4 at 2 m from the Surface

	Side		Top		Bottom	
	*MCNP	**ANISN	MCNP	**DOT4.2	MCNP	DOT4.2
Neutron	0.007	0.006	0.010	0.021	0.0164	0.036
	0.0401	0.037	0.0036	0.005	0.0032	0.006
Total	0.0471	0.043	0.0136	0.026	0.0196	0.042

*Hanyang Univ. Model [8]
**KAERI Model [6]

2. Criticality Analysis

Criticality analysis model is similar with radiation shielding model except wet cavity type. The calculation results are presented in Table 2.3 and show large difference in between.

The discrepancy is due to geometrical modeling as shown in Fig. 2.4. For criticality analysis the cask is assumed to be wet type such that the cask cavity is filled with water. The fuel assembly model using KENO-IV at KAERI was assumed as a circular type enclosing fuel assemblies of square type having actual size. The volume of the cask cavity was thus over-estimated and much more water inside compared to actual transport condition was assumed. It expects that neutrons were over-moderated and keff was under-estimated. But in criticality analysis using MCNP, geometry was modeled realistically and water volume was also assumed as actual quantity.

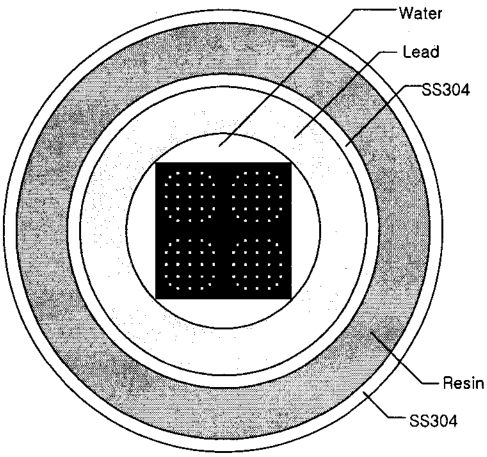


Fig. 4. Radial Cross Sectional View of KSC-4 for KENO-IV Run Modeled in 1989

Table 3. Results for Criticality Analysis for KSC-4

	k_{eff}
*MCNP	0.93139±0.00074
**KENO-IV	0.84345±0.0041

*Hanyang Univ. Model [8]
**KAERI Model [6]

RADIATION SHIELDING

The thick-walled cask body and the lid provide shielding for the KN-12 cask. For neutron shielding, polyethylene rods are arranged in the longitude boreholes in the vessel wall and polyethylene plates are inserted around the cask cavity and the bottom plate. Additional shielding is provided by basket structures. In this study, impact limiters on the top and the bottom are modeled. Other structures around the fuel basket slightly decreasing dose rate are not considered for conservative calculation. Hypothetical accident conditions, moreover, assume the absence of the neutron moderator and impact limiters on the top and the bottom.

ORIGEN-S code, the fuel depletion code that is the part of SCALE4.4a code system, was used for the source term calculation for radiation shielding analysis [7]. The Westinghouse 17x17 fuel type is used. For the source term calculations the spent fuel is characterized by the following parameters; specific burnup of 50 GWD/MTU, initial enrichment of 4.5 w/o, and minimum cooling time of 7 years that is satisfied with the thermal limit - 12.6 kW per an assembly. Detailed assumptions for the source term calculation using ORIGEN-S code are given as follows; (a) fuel assembly with 464 kg of U-metal, (b) the fuel is burned during 3 cycles with 396.825 days each at an average specific power of 42 MW/MTU (weighted by 1.2 during first cycle and 0.8 during the third cycle), and (c) a 60 day shut down period is assumed. The results from ORIGEN-S are presented in Table 4. The total neutron release from the 12 spent PWR fuel is calculated as 4.62E+9 neutrons/sec. Especially, 98.5% of the total neutron release is charged with spontaneous fission events. The total photon release is 7.89E+19 photons/sec. Most part of the photon release, 96.7%, comes from decay of fission products.

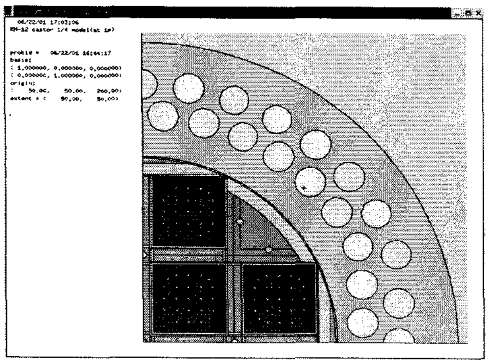
The radial and axial views of the analysis model for MCNP run are shown in Fig. 5.
The cask was modeled with a quadrant type using reflective boundary supplied by MCNP. For photon shielding calculation the cask body and the lid are segmented by the thick of about 2 cm for geometry splitting. Surfaces are also divided into 14 segments in axial direction and 23 segments in azimuth. The cask outside was assumed to be composed of air. F2 Tally is used for the surface flux and that is converted by the flux to dose conversion factor from ICRP 74 [9].

The calculation results are presented in Tables 5, 6 and 7. In normal transport conditions, maximum dose rate was obtained from the side, 0.557 mSv/hr at the surface, and 0.0893 mSv/hr at 2 m from the surface. This is due to the thicker shielding on the top and the bottom than the side. Photons are more dominant to all direction due to polyethylene rod and plates as neutron moderator. Biased active fuel to the bottom results higher dose rate at the bottom. In hypothetical accident conditions, dose rates by neutrons are dominant at every surface due to the absence of moderator and impact limiters. Maximum dose rate, 2.5144 mSv/hr, was also obtained from the side. Biased active fuel also gives higher dose rate to the bottom in hypothetical accident conditions.

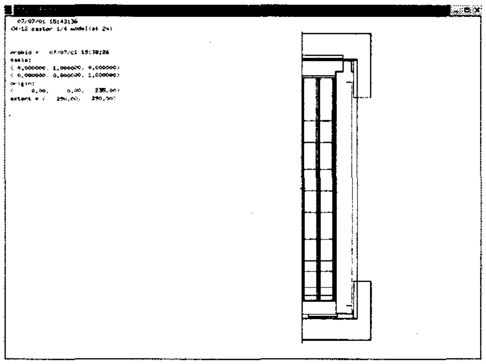
Table 4. Neutron and Photon Release from 12 Assemblies Based on Enrichment of 4.5 w/o, Burnup of 50 GWD/MTU, and 7 Years Cooling Time

Neutron Source	Neutrons/sec
(α, n) Reaction	7.54E±07
Spontaneous Fission	4.55E±09
Total	4.62E±09

Photon Source	Photons/sec
Activation Products	1.88E+15
Actinides	7.01E+14
Fission Products	7.63E+16
Total	7.89E+16



(a) Radial Cross Sectional View



(b) Axial Cross Sectional View

Fig. 5. MCNP Model for Shielding Analysis

Table 5. Maximum Dose Rates at the Cask Surface in Normal Transport Conditions

[mSv/hr]

Tally Location	Gamma Dose Rate	Neutron Dose Rate	Total Dose Rate
Top	0.0002	0.001	0.0012
Side	0.366	0.191	0.557
Bottom	0.0095	0.0014	0.0109
*Limit	-	-	2

* IAEA Safety Standard Series No. ST-1, US 10 CFR part 71, and Korea Atomic Energy Act

Table 6. Maximum Dose Rates at 2 m from the Cask Surface in Normal Transport Conditions

[mSv/hr]

Tally Location	Gamma Dose Rate	Neutron Dose Rate	Total Dose Rate
Top	0.00005	0.00006	0.00011
Side	0.0651	0.0242	0.0893
Bottom	0.00113	0.00014	
*Limit	-	-	0.00127

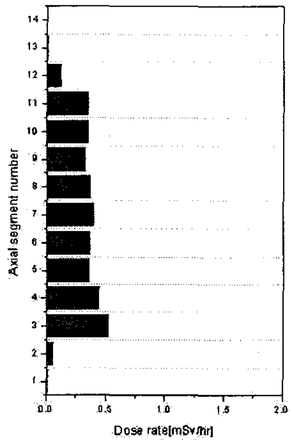
* IAEA Safety Standard Series No. ST-1, US 10 CFR part 71, and Korea Atomic Energy Act

Table 7. Maximum Dose Rates at 1m from the Cask Surface in Hypothetical Accident Conditions

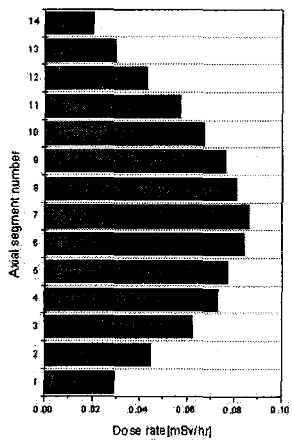
[mSv/hr]

Tally Location	Gamma Dose Rate	Neutron Dose Rate	Total Dose Rate
Top	0.0218	0.1903	0.2120
Side	0.2721	2.2423	2.5144
Bottom	0.2079	0.8927	1.1005
*Limit	-	-	10

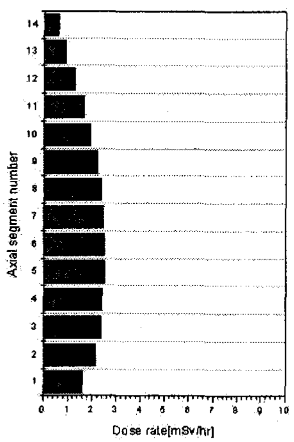
* IAEA Safety Standard Series No. ST-1, US 10 CFR part 71, and Korea Atomic Energy Act



a) At the Surface



b) At 2 m



c) At 1 m(in Accident Condition)

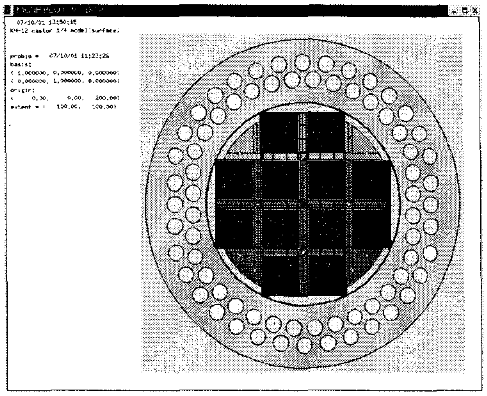
Fig. 6. Dose Rate Distributions of the Cask Side

CRITICALITY

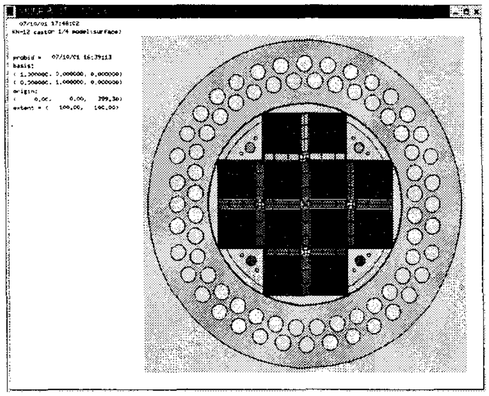
The criticality analysis was performed with the Westinghouse 17x17 type PWR assemblies. The cask is designed for both wet and dry transport. For wet transport, 80% of the volume of the cavity is filled with water. But the cask for loading and unloading operations is totally flooded. The flooded state is more limiting in terms of reactivity than the dry. The condition that results in the highest reactivity is not the fully flooded condition but the condition in which the cask is laying on its side and the water level is on the top of second row of assemblies (counting from the top)[10]. The fuel pellet-to-clad air gaps are also assumed to be flooded with the fresh water. The fuel has the initial enrichment of 5.0 w/o and the stack density is assumed to be 95%. Hypothetical accident conditions are not considered because they have no effect on design parameters important to criticality safety. Therefore, these conditions are identical to those for the normal conditions. The cask modeling assumptions are given as follows; (a) Partial flooded condition is used for the criticality analysis [Fig. 7] and (b) The minimum B-10 content of the borate aluminum will be manufactured to be 0.11 g/cc. This amount of B-10 is then conservatively reduced in the model by 20%. This 20% reduction is slightly less than 25% recommended by NUREG-1617 [4]. NUREG/CR-5661 suggests that this reduction is to account for self-shielding, grain size, and as-built boron content [11]. However, criticality experiment with borated aluminum shows that this 25% penalty is unreasonable [12]. Therefore, a 20% penalty is sufficiently conservative.

The tool for performing the criticality analysis is MCNP likewise shielding analysis but full scope model is used. In the modeling for criticality analysis, some factors like steel sheet in the flux trap, material compositions, partially flooded system, and so on, which can increase the reactivity, are added.

keff resulting from the criticality analysis considering the condition of optimum partial flooding with fresh water is 0.89708(0.00065. The result confirms the standard recommended by NUREG-1617; do not exceed 0.95 at a 95% confidence level.



a) Overall Radial Cross Sectional View



b) Radial Cross Sectional View at Reinforcing Plate

Fig. 7. Criticality Analysis Model of the Cask - Partial Flooded Condition

CONCLUSIONS

Radiation shielding and criticality safety for the KN-12 cask is evaluated using MCNP. For radiation shielding analysis, the level of maximum dose rates is between 25.1% and 89.3% of the standard limit. For criticality analysis, the level of calculated keff is 94.4% of the standard limit. Synthetically, the KN-12 cask provides radiation shielding and criticality control in normal transport conditions and hypothetical accident conditions.

ACKNOWLEDGEMENT

The authors wish to acknowledge the financial support both from the 2001 Research Program of the Innovative Technology Center for Radiation Safety (iTRS) at the Hanyang University and from the Korea Institute of Nuclear Safety (KINS).

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