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Neutron Dose Rate Analysis of PWR Spent Fuel Transport Cask Using Monte Carlo Method

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Abstract

A shielding analysis for KSC-7, the shipping cask for transporting the 7 PWR spent fuel assemblies, has been carried out. Radiation source term has been calculated on spent fuel with burnup of 50,000 MWD/MTU and 1.5 years cooling time by ORIGEN2 code. The shielding calculation for the cask has been made by using MCNP4A code with continuous cross section data library from ENDF/B-V. As a result of neutron dose rate analysis, another shielding calculational model on spent fuel shipping cask was provided which is using the Monte Carlo method.

1. Introduction

Nuclear power plants in Korea lacks spent fuel storage facilities. Thus transportation of nuclear spent fuel is inevitable over the coming years. The technologies related with spent fuel transport are currently underway in Korea. Shielding of radiation from spent fuel is one of the main factors to be considered in cask design technology. KSC-7, the shipping cask for transporting the 7 PWR spent fuel assemblies is designed to meet these demand. And the design of larger cask, which has an ability to transport more than ten PWR spent fuel assemblies at once, is being drawn up plans for the future. However, to ensure the safety of such transport, computational models must be established that are capable of evaluating the radiation exposure outside the containers in which spent fuel is shipped. Thus design benchmarking study on KSC-7 is of great importance. In the calculation of neutron dose rate at the surfaces of cask, Monte Carlo method was employed since the method generally offers more accurate results than the computer code using traditional discrete ordinate method.

2. Evaluation of Neutron Source

The source strengths of neutron and photon emitted from spent fuel must be evaluated first to observe radiation effect outside the cask. The neutron production rate from PWR spent fuel assemblies in the case of 50,000 MWD/MTU burnup and 1.5 years cooling time is calculated with ORIGEN2 code[1], which uses a matrix exponential method, using updated PWR model library.[2] Table 1 shows neutron production rates from (α,n) reaction and spontaneous fission for each isotope.

ORIGEN2 code produces neutron emission rate, but cannot generate neutron source spectrum. The isotopes ^{242}Cm and ^{244}Cm characteristically produce all except a few percent of the spontaneous fission and (α,n) neutron source in PWR spent fuel over a 10-year decay time. The measured spontaneous fission neutron spectrum of ^{244}Cm was found to be quite similar to that from ^{235}U and ^{252}Cf.[3] Thus the spectrum of ^{252}Cf is usually used to describe neutron spectrum from PWR spent fuel and is also used in this calculation.

3. Calculational Model and Method

Spent fuels generate enormous decay heats as well as radiations. Thus, shipping cask should be cooled with cooling system. KSC-7 cask has two cooling modes, wet and dry. The neutron shielding efficiency would be increased by water in the cavity with wet cooling mode. However, to consider the case of dry cooling system, which would result some higher dose rate, dry cooling condition was taken in this study.

The atom densities of each element in the structural materials of KSC-7 and the homogenized fuel zone are shown in Table 2. The 7 spent fuel assemblies in the cask are homogenized for each, and it is expected that such homogenization method will give more conservative results than in the exact source medium description. In fuel homogenization, various fission products such as ^{136}Xe are excluded because they have low fraction in percentage and are known as strong neutron absorber. This condition makes cask shielding calculations more conservative also. In shielding analysis calculations, the neutron shielding material, silicon mixture(SM-1) is assumed to be got lost under the hypothetical conditions such as fire accident or other abnormal accident which would occur during transportation of cask. The MCNP calculation model is described in Figure 1 and 2.

The Monte Carlo N-Particle transport code (MCNP4A) developed at Los Alamos National Laboratory (LANL)[4] was employed in this work. The calculation was performed with the continuous energy neutron cross section and gamma ray production cross section library, RMCCS2.

Specific safety and transport regulations on transportation of radioactive materials recommend that radioactive quantity such as dose rate at the surface of cask should not exceed limited value. In MCNP4A code, the calculated fluences are converted to dose rate using user specified conversion factor. Fluence-to-dose rate conversion factors from ANSI/ANS-6.1.1[5] was used to obtain dose rate at various detection positions in this calculation.

4. Conclusion and Further Studies

The dose rates calculated under normal transport and hypothetical accident condition are shown in Table 3 and Figure 3~8. This shielding calculation is conservative in selection of cask cooling method, description of source region, and consideration of detailed structure. However, it was shown that calculated neutron dose rate is allowable. As a result of neutron dose rate analysis, another shielding calculational model on spent fuel shipping cask was provided which is using the Monte Carlo method and continuous cross section data, and it is expected that more reliability in radiation shielding analysis would be given by comparing results from Monte Carlo method to those from discrete ordinate method. Further studies which will evaluate both neutron and photon dose rate and perform the overall shielding analysis should be made on KSC-7. And these tasks would verify further cask design and have a great importance in developing radioactive material transportation technologies.

References

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2. C. V. Parks *et al.*, "Assessment of Shielding Analysis Methods, Code, and Data for Spent Fuel Transport/Storage Applications," ORNL/CSD/TM-246, 1988.
3. S. B. Ludwig and J. P. Renier, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018, 1989.
4. J. F. Briesmeister, "MCNP-A General Monte Carlo N-Particle Transport Code," Version 4A, LA-12625-M, Los Alamos National Laboratory, 1993.

5. "Neutron and Gamma-Ray Flux-to-Dose-Rates Factors," ANSI/ANS-6.1.1-1977 (N666), American National Standard Institute, 1977.

Table 1. Neutron Production Rate from 7 PWR Spent Fuel Assemblies (50,000 MWD/MTU burnup, 1.5y cooling time)

Isotope	Assembly Discharge	1.5y	3.0y	5.0y	10.0y
(α,n)					
Pu238	2.090×10^{7}	2.184×10^{7}	2.168×10^{7}	2.136×10^{7}	2.054×10^{7}
Pu239	9.951×10^{5}	1.010×10^{6}	1.010×10^{6}	1.010×10^{6}	1.010×10^{6}
Pu240	1.804×10^{6}	1.806×10^{6}	1.809×10^{6}	1.813×10^{6}	1.820×10^{6}
Am241	6.266×10^{5}	1.803×10^{6}	2.894×10^{6}	4.228×10^{6}	7.028×10^{6}
Am243	1.148×10^{5}	1.149×10^{5}	1.149×10^{5}	1.149×10^{5}	1.148×10^{5}
Cm242	3.146×10^{8}	3.095×10^{7}	3.106×10^{6}	2.293×10^{5}	9.189×10^{4}
Cm243	2.161×10^{5}	2.083×10^{5}	2.009×10^{5}	1.913×10^{5}	1.694×10^{5}
Cm244	3.082×10^{7}	2.910×10^{7}	2.748×10^{7}	2.545×10^{7}	2.102×10^{7}
Totals	3.701×10^{8}	8.685×10^{7}	5.833×10^{7}	5.442×10^{7}	5.181×10^{7}
Spontaneous Fission					
Isotope	Assembly Discharge	1.5y	3.0y	5.0y	10.0y
Pu238	3.407×10^{6}	3.562×10^{6}	3.537×10^{6}	3.482×10^{6}	3.349×10^{6}
Pu240	9.509×10^{6}	9.521×10^{6}	9.538×10^{6}	9.554×10^{6}	9.593×10^{6}
Pu242	3.889×10^{6}	3.889×10^{6}	3.889×10^{6}	3.889×10^{6}	3.889×10^{6}
Cm242	1.527×10^{9}	1.502×10^{8}	1.507×10^{7}	1.113×10^{6}	4.457×10^{5}
Cm244	3.711×10^{9}	3.504×10^{9}	3.307×10^{9}	3.065×10^{9}	2.531×10^{9}
Cm246	2.434×10^{7}	2.434×10^{7}	2.433×10^{7}	2.432×10^{7}	2.431×10^{7}
Cf252	7.735×10^{6}	5.216×10^{6}	3.517×10^{6}	2.079×10^{6}	5.588×10^{5}
Totals	5.287×10^{9}	3.701×10^{9}	3.369×10^{9}	3.109×10^{9}	2.574×10^{9}
Totals	5.655×10^{9}	3.789×10^{9}	3.427×10^{9}	3.164×10^{9}	2.625×10^{9}

Table 2. Atom Densities of Homogenized Fuel Region and Shielding Materials (atoms/barn-cm)

Material Element	Homogenized Fuel	S.S-304	Lead	Borated S.S	Silicon Mixture
H					4.371×10^{-2}
B					1.277×10^{-3}
C				2.362×10^{-4}	1.604×10^{-2}
O	1.264×10^{-2}				2.489×10^{-2}
Al					1.985×10^{-3}
Si				1.683×10^{-3}	5.751×10^{-3}
Cr	5.100×10^{-6}	2.672×10^{-2}		1.591×10^{-2}	
Mn				8.860×10^{-3}	
Fe	1.000×10^{-5}	6.060×10^{-2}		5.489×10^{-2}	
Ni	4.050×10^{-3}	9.880×10^{-3}		8.860×10^{-3}	
Zr	1.566×10^{-4}				
Pb			3.300×10^{-2}		
U-235	1.896×10^{-4}				
U-238	6.131×10^{-3}				

Table 3. Summary of Maximum Neutron Dose Rates under Normal Transport and Hypothetical Accident Conditions

Condition	Neutron Dose Rate (mrem/hr)								
	Cask Surface			1m from Surface			2m from Surface		
	Side	Top	Bottom	Side	Top	Bottom	Side	Top	Bottom
Normal Transport Condition	15.91	32.82	38.09	5.52	6.53	5.92	3.11	2.17	1.94
Hypothetical Accident Condition	671.35	263.31	345.78	210.76	36.92	51.79	106.40	11.95	17.09

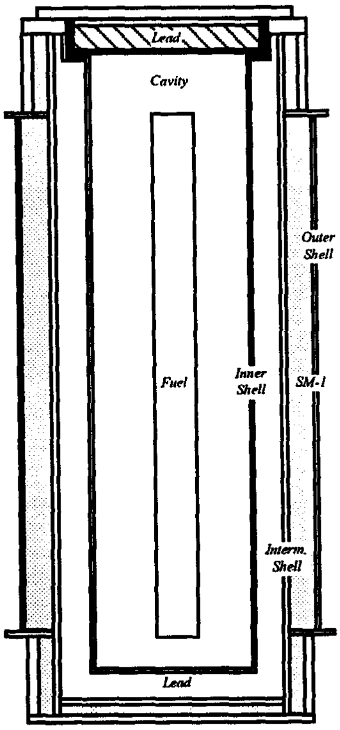

Fig 1. Calculational Model of KSC-7

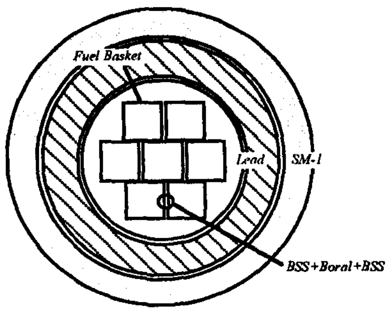

Fig 2. Cross Section View of KSC-7

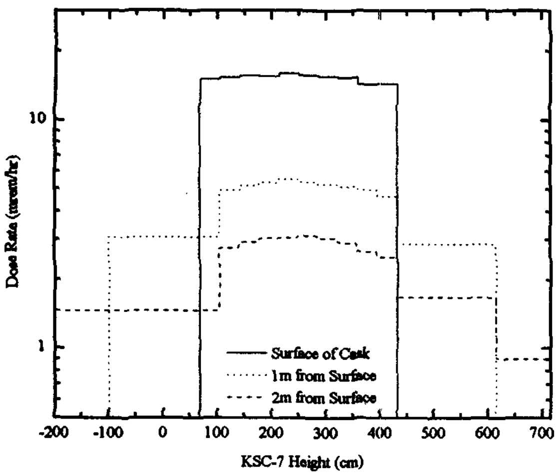

Fig 3. Dose Rate at the Side of KSC-7 (Normal Condition)

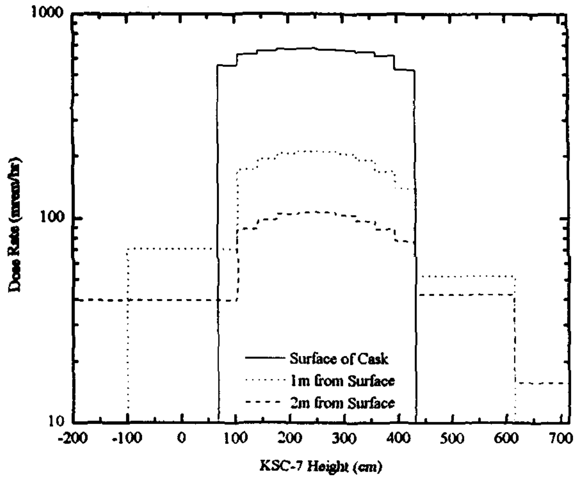

Fig 6. Dose Rate at the Side of KSC-7 (Hypothetical Accident)

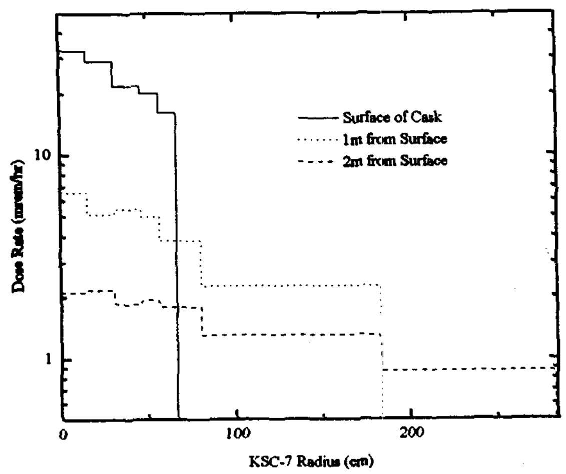

Fig 4. Dose Rate at the Top of KSC-7 (Normal Condition)

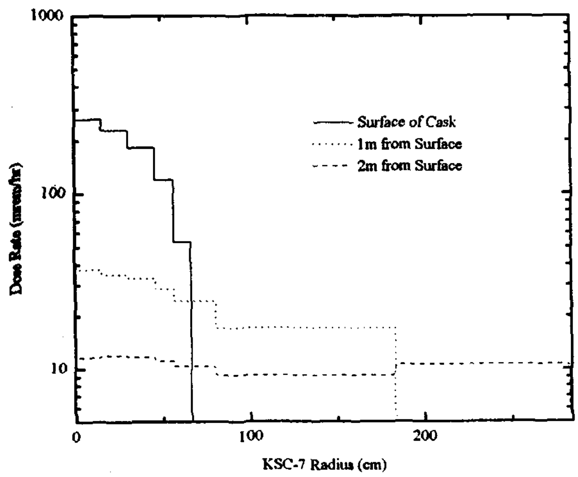

Fig 7. Dose Rate at the Top of KSC-7 (Hypothetic Accident)

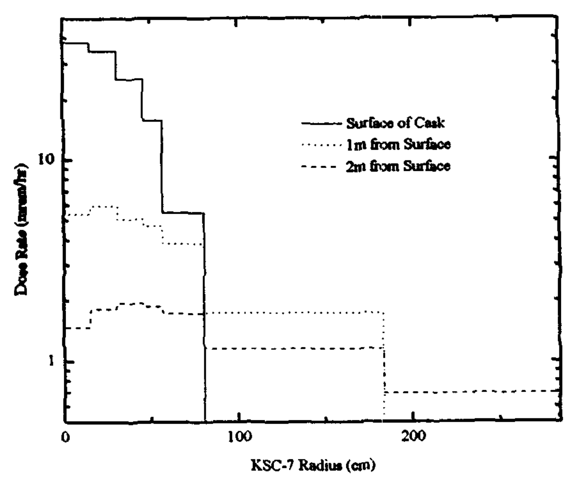

Fig 5. Dose Rate at the Bottom of KSC-7 (Normal Condition)

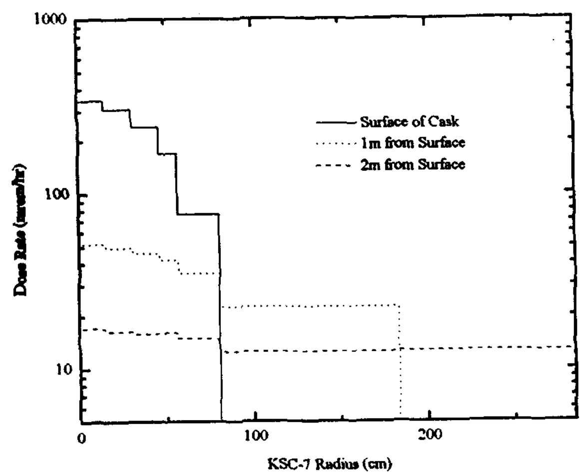

Fig 8. Dose Rate at the Bottom of KSC-7 (Hypothetic Accident)