MCNP5 Benchmarking Calculations of HI-STAR100 Cask Radiation Shielding

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1. Introduction

We performed radiation shielding calculations with an MCNP5 input file from the Hi-star100 safety analysis report [1] (SAR). Using the source term data given in the SAR, we modified the input file for the fuel gamma source. By running MCNP5 [3] with the modified input file, we obtained gamma dose rates that show reasonable agreement with the results given in the SAR.

2. Description on Calculation Steps

A cask radiation shielding calculation with the MCNP5 code was performed with the sample input file from the Hi-star100 SAR as the first step in the validation study for the used nuclear fuel (UNF) cask shielding calculations. This paper discusses a part of entire steps for the UNF cask shielding calculations. Entire steps for the UNF cask shielding calculations will be firstly presented and after that I'll describe the particular step I performed.

2.1 Shielding Calculation Steps

A typical shielding calculation consists of two individual steps. The first step generates source terms with the SCALE code [4] and the second performs radiation dose rate calculations with the MCNP5 code. At the source term generation step, fuel depletion calculations are performed and the isotopic composition of UNF assemblies is obtained. At the same time neutron and gamma source terms are obtained. Source terms obtained from this step are used in the next step.

For the radiation dose calculation, an input file is written accounting for the geometry, material composition and source distributions. At this step, six different gamma and neutron sources are dealt with in three individual simulations. These three different simulations are named as fuel gamma source simulation, ^{60}Co gamma source simulation and neutron source simulation. Each simulation deals with (1) gamma rays from the decay of radioactive nuclides in the active fuel region, (2) gamma rays from activated impurities in the nonfuel structure in fuel assemblies and (3) neutrons from spontaneous fissions in various Pu and Cm radionuclides, (α,n) reactions and subcritical fissions and gamma rays from (n,γ) reactions. The final dose rate for a shielding simulation is obtained by summing up three results.

2.2 Design Basis Fuel

Fuel assemblies that are stored and transported in the cask can have several different characteristics (enrichment, burnup and cooling time). To deal with these different fuel types and burnup histories, one fuel type and one burnup history are selected that represent the most conservative result (called design basis fuel). For the shielding calculation of MPC-68, GE7x7 fuel assembly, ^{60}Co gamma source, a fuel burnup of 34500 MWD/MTU, cooling time of 11-year, ^{235}U enrichment of 2.6 wt.% is selected.

2.3 Benchmarking Calculation with theMCNC5 code

The MCNP5 sample input file for the MPC-68 shielding calculation is presented in appendix 5.A-3 in the Hi-star100 SAR. However, comparable results for this sample input file is not presented in the Hi-star100 SAR. For that reason we changed some part of this input file. The geometry specification was retained but we put a different source distribution representing the fuel gamma given in the SAR replacing the ^{60}Co gamma distribution in the original input file. Furthermore I modified tally surfaces and the tallies to match the description on the SAR. Through these modifications, I was able to obtain results that show reasonable agreement with the result shown in the SAR.

3. Methods and Results

3.1 Input Specifications

We performed a shielding benchmarking simulation for the fuel gamma source of the MPC-68 fuel cask. The design basis fuel type is GE7x7 model. SAR provides the fuel gamma source term for this assembly type. We normalized the source distribution and put it in the sample input file because MCNP5 input requires probability distributions.

SAR also provides the axial burnup distribution in the fuel region along the vertical axis. We normalized the burnup distribution and added to the sample input file.

Figure 1 shows the geometry of the MPC-68 cask is given in the SAR. This geometry represents the normal condition for the cask. There exist impact limiters on the bottom and the top, while the sample input file (Figure 1.) does not represent them.

Figure 1 also points out surface dose tally positions. Surface A, B and C is divided into 26 segments along vertical axis. Points 1, 3, 4 and 3a correspond to segments that have length of 1 foot or less on the Point 2a and 2 are selected referring the results that represent the highest dose rate among 16 segments on the region shown in Figure 1. Point 3a corresponds to 2 segments on the position shown in the figure.

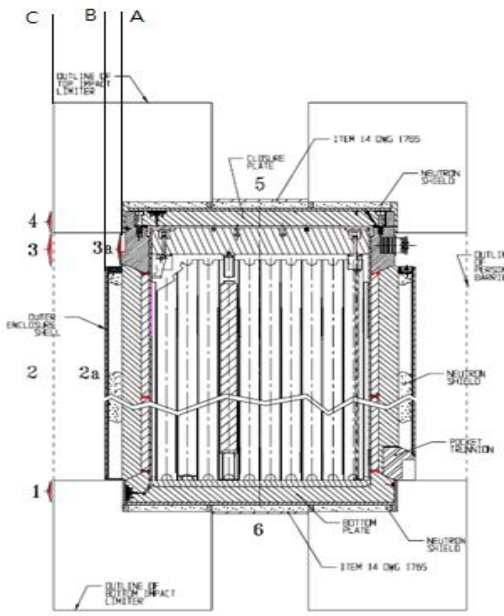

Figure 1. Cross section view of MPC-68 [Ref. 1]

Along with x and y axes the presented probability distribution is uniform.

3.2 Results

By running 1,000,000,000 particles running MCNP5 with the modified input file, we obtained surface dose rates per one source particle for six locations. For a regular F2 tally, this tally gives output data in unit of $(1/cm^2)$. After putting conversion factors $(\frac{rem/hr}{\#/cm^2 s})$ in F2 tally, the unit of output becomes $(\mathrm{rem/hr})$(#/s). Dose rates are calculated by multiplying by $(68 \times 7.25 \times 10^{14} \times 10^{3} \#/s)$ of total particles released from 68 assemblies in a cask. Output data and dose rates are presented in Table 1.

Table 1. Dose rate comparison

Location	Dose rate (mrem/hr)		Relative difference
	my result	SAR	
2a	27.57 ($\pm$1.0%)	27.04	2.0 %
3a	0.34 ($\pm$9.0%)	0.43	-20.9 %
1	3.43 ($\pm$1.4%)	2.15	59.5 %
2	14.37 ($\pm$0.7%)	17.55	-18.1 %
3	0.81 ($\pm$1.8%)	0.67	20.9 %
4	0.44 ($\pm$3.9%)	0.36	22.2 %

MCNP5 dose rates are compared with those from the SAR in Table 1. Although the comparison is generally good, we note significant differences especially at low-dose sites.

The first reason is the difference in geometry. The results in the SAR are calculated with the model which has impact limiters on the top and the bottom, while they are not represented in our model. The difference in the MCNP versions is the second reason. Results in the SAR are obtained with MCNP-4A while we used MCNP5. During the revision of the SAR over several years, geometric parameters, materials and many other parameters were amended but the sample input file remains the same. [2] Some portion of the differences noted in Table 1 is likely due to parametric modifications of the cask model.

4. Conclusions and Discussion

By comparing dose rate results from the SAR and our MCNP5 simulation, we could check the reasonable similarity but we failed getting sufficiently accurate matching results even we ran very large number of particles. Variances of the result is still large, we can conclude that we need further research with proper variance reduction techniques especially for the deep penetrations simulation like this case. However as we checked reasonable accuracy, within a range of $\pm 2\sigma$ only for point 2a we can conclude that the differences in results are not only from the systematic variances.

The difference in geometry, code version, and parametric changes of model are considered to mainly affect the differences. To reduce the relative differences and get more similar results we need to modify the geometry as the description given in the SAR.

In order to calculate final dose rates for every presented source, we are preparing the neutron and gamma coupled simulation and the simulation for the nonfuel gamma source.

ACKNOWLEDGEMENTS

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