Journal of NUCLEAR SCIENCE and TECHNOLOGY, Supplement 1, p. 334–336 (March 2000)

Model and Method for Reasonable Shielding Calculation of Spent Fuel Casks

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For the purpose of performing the reasonable shielding calculation of transport/storage spent fuel cask, the discussion of the cause of the discrepancy between the measured and calculated dose rate of a spent fuel cask is important. This paper shows the several items that may have large discrepancy between the measured and calculated dose rate of a spent fuel cask with respect to the calculation models and methods. The items include 1) the expression of source terms such as burnup profile, distribution of activated gamma-ray source, and gamma-ray source spectra, 2) the usage of a detailed geometrical model that is able to calculate the distribution of fission reaction precisely, and 3) the difference between detector response and dose conversion factor. The consideration of these effects may improve the agreement between the measurement and the calculation of a spent fuel cask.

KEYWORDS: Spent fuel cask, burnup profile, Source term, Effective multiplication factor, Source distribution and intensity of ^{60}Co, Gamma-ray source spectra, detector response

I. Introduction

Many dose rate measurement of the transport and storage spent fuel casks and comparison of these results with calculations have been made. But, there are large variations of the agreement between the measurement and calculation of these spent fuel casks. Some results show good agreement but the others show large difference. For example, the Nuclear Energy Agency Committee for Reactor Physics of the Organization for Economic Cooperation and Development obtained a factor of two variation in the calculated results by several delegates from the member nations for the actual cask measurement[1)]. One of the cause of the discrepancy between the measured and calculated dose rates of a spent fuel cask may come from the inappropriate assumption in the calculation models and methods including source terms.

This paper discusses the calculation models and methods which may be the cause of the discrepancy between the measured and calculated dose rates of a spent fuel cask. The effect of each item is also estimated. This kind of discussion is very important to obtain a reasonable calculated results of a spent fuel cask. Selection of calculation codes and cross-section libraries is out of scope of this paper.

II. Source Term

The precise definition of source terms is the most important, but the most difficult matter for the calculation of a spent fuel cask. A spent fuel assembly has a burnup profile, especially for axial direction. The importance of this profile is well known for the calculation of the neutron dose rate at the radial direction of a cask, but it is also important for fission product(FP) gamma dose rates at both ends of a cask. So far, the evaluation of an intensity of the activated ^{60}Co gamma-ray source is overestimated because of using the conservative impurity level of ^{59}Co in endfittings of fuel assembly.

Furthermore, appropriate expression of FP gamma-ray source spectra should be considered. But, the selection of neutron fission source spectra from actinide such as ^{235}U, ^{239}Pu or ^{244}Cm, affects not so much to the results. The neutron source spectrum by the (α ,n) reaction is not necessary to consider in the typical case, because the source intensity by the (α , n) reaction is very small for the typical spent fuel assembly, even though the (α , n) reaction spectrum is much harder than the fission spectrum. When the detailed neutron spectrum is necessary for the calculation, it is easy to obtain by using the source intensity calculation codes such as ORIGEN-S[2)].

1. Effect of Burnup Profile

A typical axial burnup profile derived from Ref.3 and the corresponding neutron and FP gamma-ray source intensity distributions are shown in **Fig.1**. A calculation with a cask geometry using a flat burnup profile shows about 0.7 times the neutron dose rate using the typical burnup profile for the radial direction and about 1.7 times that for the bottom region[4)]. Therefore, the consideration of burnup profile is very important to the neutron calculation. Also the difference of the peaking factor (a ratio of maximum burnup and average burnup) affects the dose rare of the radial direction very much but not so much to the both end.

The FP gamma-ray dose rate is not so sensitive with this profile for the radial direction, but for the bottom region, the flat burnup profile gives a dose rate a factor of 4 higher. The axial neutron source intensity varies greatly depending on the

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Table 1 Contribution of each source to calculated gamma-ray dose rates around TN-12/2 package.

Measured Point	Gamma-ray dose rate (mR/h)							
	Surface				1 m from surface			
	FP	^{60}Co	(n, γ)	Total	FP	^{60}Co	(n, γ)	Total
Side								
1(top)	1.64	0.56	0.40	2.61 6.31	1.53	0.30	0.22	2.06 2.43
2	5.69	0.11	0.51	7.87 7.08	2.05	0.13	0.25	2.85 3.53
3	7.28	0.00	0.59	8.07 9.85	2.51	0.05	0.29	3.36 3.04
4	6.41	0.00	0.67	8.19 5.34	3.22	0.03	0.29	2.92 2.41
5(center)	7.35	0.00	0.72	2.41	3.04	0.02	0.30	2.11
6	9.16	0.00	0.69		2.73	0.03	0.28	
7	7.53	0.00	0.66		2.59	0.08	0.25	
8	4.89	0.02	0.43		1.88	0.32	0.20	
9(bottom)	0.76	1.32	0.33		1.05	0.90	0.16	
Top								
16(center)	0.25	1.43	0.16	1.84	0.17	0.50	0.14	0.81
Bottom								
18	2.18	5.30	0.83	8.31	0.78	1.81	0.48	3.07
19(center)	2.26	5.96	1.01	9.23	0.83	2.16	0.59	3.58
20	2.16	5.30	0.83	8.29	0.78	1.81	0.47	3.07

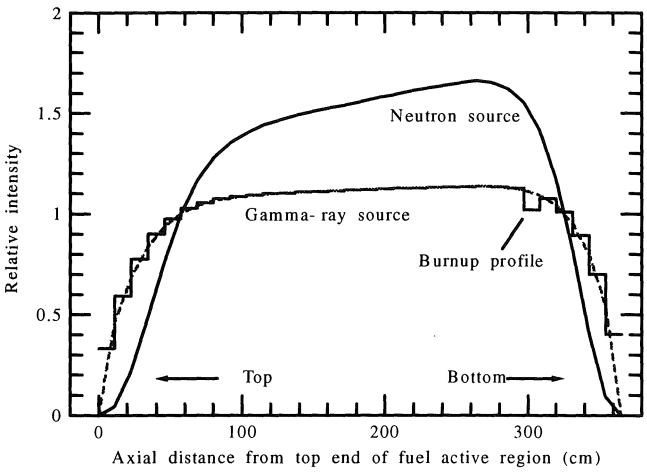

Fig.1 Axias Burn-up Profile and Evaluated Source Distribution

burnup profile, and the top and bottom gamma-ray dose rates are closely correlated with the burnup profile .Therefore, it is very important to know precise burnup profile in order to get a reasonable dose rate by calculation.

2. Source Distribution and Intensity of ^{60}Co

The activated gamma-ray source intensities at the endfittings of a spent fuel assembly are calculated on the basis of an assumed maximum impurity level of ^{59}Co in each material, but the actual average impurity level is believed to be less. Furthermore, the actual ^{60}Co source may be biased toward the fuel region corresponding to the thermal neutron flux distribution during operation in reactor core.

The calculated dose rate contribution of each gamma-ray source for the TN-12A cask is shown in Ref 4. In these results, the ratio of the gamma-ray dose rate from the activated endfitting and that from FP for the bottom and the top region is more than two as shown in **Table 1**. (In this case, the typical burnup profile mentioned above is considered for the FP gamma-ray and secondary gamma-ray calculations) Therefore, the evaluation of ^{60}Co source is very important for the end regions. The more detailed information to obtain the ^{60}Co source, such as the impurity level of ^{59}Co in each materials and the detailed axial thermal neutron flux profile of fuel assemblies during the burnup in a reactor, is required for getting more precise results.

3. Gamma-Ray Source Spectra

As there are huge number of FP gamma-ray source energy lines in a spent fuel, usually these source energy lines are grouped into broad energy group spectra. This means many different gamma-ray energy lines are replaced by the representative energy line. This method has a possibility of making a large error in some cases. For example, gamma energy replacement from 1.0MeV to 0.9MeV or 1.1MeV for 30cm thick iron sphere, the dose rate is underestimated 40% when using 0.9MeV, and overestimated 60% when using 1.1MeV even though the energy conservation is considered as mentioned in Ref.5.

Therefore, it is preferable to use a combination of some important discrete gamma-ray energy lines and the representative broad energy grouped spectra for the other gamma-ray energy lines to avoid of making a large error by the expression of source term. Of course, this method is only applicable to the calculation using Monte Carlo codes or point kernel codes. The gamma-ray energy lines such as 2.186MeV of ^{144}Pr, 1.562MeV of ^{106}Rh, 1.365MeV of ^{134}Cs and 1.17 and 1.33MeV of ^{60}Co are very important for the shielding calculation

III. Calculation Model

For a wet type spent fuel cask, the neutron dose rate may be evaluated conservatively, because the distribution of the fission inside of the cask cavity is usually not considered at all. The higher fission rate, which corresponds to higher neutron source intensity, is usually at the center of the cavity. The distribution of fission inside of the cask cavity is important for the wet type casks.

1. Effective Multiplication Factor

For the neutron dose rate calculation, increasing of the neutron source intensity by the fission with the neutrons from the spontaneous fission of the actinides such as ^{242}Cm and ^{244}Cm, or from the (α , n) reaction with light elements in the spent fuel assemblies must be considered. The distribution of the fission is directly corresponding to the increased source intensity by the fission. For considering this increased neutron source intensity for the shielding calculation, an averaged effective multiplication factor(Keff) of the spent fuel cask is used to increase total neutron source intensity. For the wet type casks, the fission occurs much at the central region than at the outer region, and the neutrons emitted at central region can be easily shielded by the cavity water. For the dry type casks, the Keff is small and this shielding effect is also small because there is no cavity water. The effect of this fact is more than 30% for the wet type cask, but less than 5% for the dry type cask according to Ref.5. Usage of the detailed geometrical model for the region including a fuel assembly, which can be applied to a criticality analysis is recommended for the wet type spent fuel cask.

2. Modeling of Geometry

For the usage of two dimensional calculation codes to a spent fuel cask, it is necessary to make some simplified model, because the cask has very complicated geometry, especially in the basket with spent fuels, the neutron shield region with fins and the cooling fin region at the outer surface of the cask. Some discussion for the model of calculation geometry is shown in Ref.5.

IV. Detector

Usually the response of detectors used in the measurement and the conversion factors used for a calculation are different because it is difficult to reproduce the conversion factor by the arrangement of the neutron moderator, and/or absorber of the detector. The dose rate measurement of a cask surface does not measured the exact surface dose rate.

1. Difference in Detector Response

The response of the neutron rem-counter does not have the same energy dependence specified in the standard such as the ANSI/ANS 6.1.1-1977 flux-to-dose conversion factor used in the calculation. The comparison of the calculated neutron dose rates using the neutron spectra obtained by measurements with measured ones with several types of neutron rem-counters in shown in Ref.6. The effect is up to 40% according to the results.

An angular dependence of the response for the neutron measurements shows an effect on the surface dose rate. According to Ref.7, the surface neutron dose rates turn out to be about 0.9 times the calculated ones when the angular dependence of the response is considered.

2. Surface Dose Rate

In the measurements, the rem-counter can not actually measure the exact surface dose rate because the size of the rem-counter is usually large and the dose rate decreases rapidly near the surface compared to that at the point 1m from the surface. As the effective center of the rem-counter used in these measurements is about 6cm from the surface of the rem-counter based on the rem-counter manual, the dose rates at 6cm from the surface of the cask showed a dose rate 10% lower than those at the surface for neutron according to the calculations by a Monte Carlo calculation.

This means that the calculated surface dose rate is always overestimated around 10% from the measured values obtained by the rem-counter. The modification of the distance between the actual surface to the effective center of a detector is necessary

V. Summary

The cause of the discrepancy with respect to the calculation models and methods are discussed. The following effects are shown in this paper.

1) Expression of source terms such as shape of burnup profile, evaluation of the distribution of activated gamma-ray source, and expression of gamma-ray spectra is the most important,
2) calculation model which can be consider distribution of fission reaction precisely is also important for the wet type casks, and
3) difference between detector response and dose conversion factor must be in mind. The consideration of these effects improve the agreement between a measurement and calculation of spent fuel casks.

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