⟨Technical Note⟩
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SHIELDING ANALYSIS OF DUAL PURPOSE CASKS FOR SPENT NUCLEAR FUEL UNDER NORMAL STORAGE CONDITIONS

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Korea expects a shortage in storage capacity for spent fuels at reactor sites. Therefore, a need for more metal and/or concrete casks for storage systems is anticipated for either the reactor site or away from the reactor for interim storage. For the purpose of interim storage and transportation, a dual purpose metal cask that can load 21 spent fuel assemblies is being developed by Korea Radioactive Waste Management Corporation (KRMC) in Korea.

At first the gamma and neutron flux for the design basis fuel were determined assuming in-core environment (the temperature, pressure, etc. of the moderator, boron, cladding, UO_2 pellets) in which the design basis fuel is loaded, as input data. The evaluation simulated burnup up to 45,000 MWD/MTU and decay during ten years of cooling using the SAS2H/OGIGEN-S module of the SCALE5.1 system. The results from the source term evaluation were used as input data for the final shielding evaluation utilizing the MCNP Code, which yielded the effective dose rate.

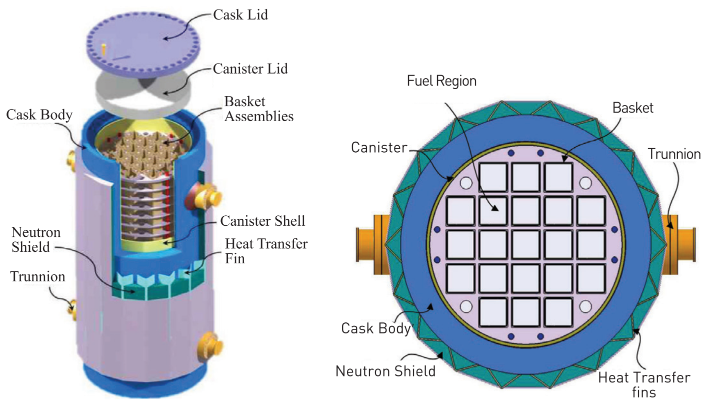
The design of the cask is based on the safety requirements for normal storage conditions under 10 CFR Part 72. A radiation shielding analysis of the metal storage cask optimized for loading 21 design basis fuels was performed for two cases; one for a single cask and the other for a 2x10 cask array. For the single cask, dose rates at the external surface of the metal cask, 1m and 2m away from the cask surface, were evaluated. For the 2x10 cask array, dose rates at the center point of the array and at the center of the casks' height were evaluated. The results of the shielding analysis for the single cask show that dose rates were considerably higher at the lower side (from the bottom of the cask to the bottom of the neutron shielding) of the cask, at over 2mSv/hr at the external surface of the cask. However, this is not considered to be a significant issue since additional shielding will be installed at the storage facility. The shielding analysis results for the 2x10 cask array showed exponential decrease with distance off the sources. The controlled area boundary was calculated to be approximately 280m from the array, with a dose rate of 25mrem/yr. Actual dose rates within the controlled area boundary will be lower than 25mrem/yr, due to the decay of radioactivity of spent fuel in storage.

KEYWORDS : Radiation Shielding, 2x10 Array, Storage Condition, Dual Purpose Cask, Spent Fuel Assemblies

1. INTRODUCTION

Korea expects a shortage in storage capacity for spent fuels at reactor sites. Therefore, a need for more metal and/or concrete casks for storage systems is anticipated for either the reactor site or away from the reactor for interim storage. For the purpose of interim storage and transportation, a dual purpose metal cask that can hold 21 spent fuel assemblies is being developed by Korea Radioactive Waste Management Corporation (KRMC) in Korea. This cask is composed of a main body made of carbon steel and a stainless steel dry shielded canister (DSC) with stainless steel baskets inside to contain spent fuel assemblies as shown in Fig. 1.

The design of the cask is based on the safety requirements for normal storage conditions of 10 CFR Part 72. 10 CFR Part 72 requires that spent fuel storage and handling systems be designed with adequate shielding to provide sufficient radiation protection under normal, off normal, and accident-level conditions. As prescribed in 10 CFR Part 72, the regulatory requirements for dose rates at and beyond the controlled area boundary include radiation from direct radiation and radiation from radionuclides in effluents. To meet the regulatory requirements of 10 CFR Part 72, dry storage facilities for spent fuel should



(a) Outline of the metal cask (b) Cross-sectional view of the metal cask

Fig. 1. Dual Purpose Metal Cask Arrangement

be designed to protect the public and radiation workers from direct radiation from casks. While 10 CFR Part 72 does not impose specific dose rate limits on individual casks, dose rates from 20 to 400mrem/hour have been accepted for storage casks in previous 10 CFR Part 72 evaluations[1]. The shielding analysis for a single storage cask should identify the minimum distance that is required to meet the dose criteria stipulated in 10 CFR 72.104. Dose rates at the controlled area boundary are generally calculated to meet the requirement of less than 25mrem/year for the public under normal storage conditions. In addition, the radiation shielding analysis should include a graph on dose rate versus distance for a single cask to facilitate a site-specific evaluation for general licensees[1].

The design of the spent fuel storage facilities should consider many factors, such as the casks' arrangement, radiation workers' working condition (ex. installation, visual examination, radiation monitoring, and maintenance, etc.), and operating procedures. To take these design factors into consideration, dose rates at the cask's external surface should be calculated. After that, a calculation of dose rates for the cask surface, performed in accordance with the NUREG-1536 dose rates evaluation, should also include a dose rate versus distance curve for a theoretical cask array. The theoretical cask array should consist of at least 20 storage casks (typically in a 2x10 array), and may include the shadowing effect among the casks.

Design basis accidents could result in limited and localized damage to the outer shell and radial external surface of the cask. However, as the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose rate at the site boundary would be negligible[2][3]. This is because damage to a single cask has negligible effects on radiation safety of the whole storage facility.

This paper presents the shielding analysis method and results of dose rate calculations for a single cask and the 20 storage cask arrays under normal storage conditions.

2. RADIATION SOURCE

The storage cask was designed to load either 21 WH or 21 CE type fuel discharged from Korean NPPs prior to 2008. To simulate loading different types of spent fuel in the cask, a hypothetical design basis fuel was assumed in order to be conservative. For the evaluations, the design basis fuel assembly was divided into two parts: one containing fissile material and the other including all other structural components

The active fuel region was assumed to load WH type 17RFA fuel assemblies, with the greatest U-metal mass emitting the most gamma rays and neutrons. Also, structural component of the design basis fuel assembly were assumed to be that of CE type PLUS7^{TM} fuel, which has the largest dimensions.

To calculate dose rates from the spent fuel storage cask, a source term evaluation for design basis fuel was carried out. The following are factors of the source term that should be considered in terms of radiation protection.

- Gamma sources
  - Primary gamma rays emitted from disintegration of fission products and actinides
  - Gamma rays emitted from the decay of Co-60 nuclides generated from the activation of fuel assembly structures
  - Secondary gamma rays generated from (n, γ) reaction of fissile and non-fissile materials
- Neutron sources
  - Neutrons generated from spontaneous fission
  - Neutrons generated from (α, n)reaction of fissile materials

Primary gamma rays and neutrons generated from spontaneous fission and (α, n)reaction of fissile materials were calculated by creating cross section libraries, and by reflecting the burnup and cooling time of the design basis fuel using the SAS2H module of SCALE 5.1[4]. Gamma rays emitted from the decay of Co-60 nuclides,