Radiological Assessment of the Dose Distribution around the Steam Generator

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1. Introduction

A Steam Generator (SG) is replaced because the design life is expired or the cost of maintenance is too high. In Korea, the SGs of KORI-1 and ULJIN-1, 2 were already replaced and other SGs will continually be replaced according to each design life [1]. Furthermore, SG was considered as a high activated component in the decommissioning work. [2]. In particular, some workers had gone directly into the decommissioning workplace when the SG was replaced or decommissioned. Therefore, the assessment of the space dose rate in the facility is important to design a reasonable decommissioning scenario and secure the radiological safety of workers. In this study, the space dose distribution around the SG in KORI-1, which is the main components in a nuclear power plant, was assessed. Furthermore, the method to calculate the space dose rate for a multi-source was assessed for the two Steam Generators in KORI-1.

2. Methods and Results

2.1 Materials

The SG (Red) is an overall cylindrical-type structure and separated by two different diameters, where the upper part is larger than the lower part. Its head and bottom are a hemispherical-type. The dimensions of the SG are as follows: The height is 2,005 cm and the thickness is 9 cm. The upper diameter is 223 cm and the lower diameter is 171 cm. A concrete shield (Green) combines two square-type structures of different size (Fig. 1). The material and density of each of the structures (SG and Concrete shield) are carbon steel (7.86 g/cm^3) and concrete (2.242 g/cm^3). The data of the geometrical structure are based on the blueprint of the nuclear power plant.

The main effective radionuclide in this study is ^{60}Co, which is greatly contributed to the exposure dose of workers.

The radioactivity inventory of ^{60}Co at the SG was obtained from the existing research data [1, 3]. It was assumed that the nuclear power plant operated for 40 years and no cooling time occurred after shutdown.

The source term of SG is the lower part of the cylinder-type because the tube bundle is located at this part and 95% of the total SG radioactivity exists [3].

An MCNP5 code (build 1.40) was used to calculate the space dose distribution around the SG and concrete shield.

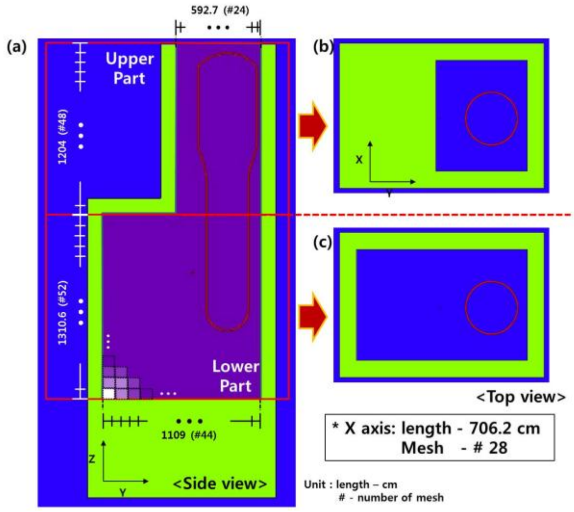

Fig. 1. Geometric structure of KORI-1 SG using MCNP((a) Side view (b) Top view(upper part) and (c) Top view(lower part))

2.2 Assessment of the dose distribution around SG

To measure the space dose rate at the inside of the concrete shield, where the SG exists, meshes were set. The size of each mesh was approximately 25 cm × 25 cm × 25 cm. The output of the fmesh-tally was converted into the dose rate by a flux-to-dose conversion factor from ICRP 60.

The range of the space dose rate was 0.02 - 32.5 mSv/hr and the height of the maximum space dose rate is 24.65 m from the bottom of the containment (Fig. 2). The range of the space dose rate near the SG from 19.86 m to 29.45m in height is 16.9 – 32.5 mSv/hr. This height was set as the source term, and the tube bundle which is the most activated part in SG existed at this height. Heights under 1356.36 m and over the 3820.8 m showed a relatively low space dose rate in Fig. 2. The reason is that the top and bottom of the SG functioned as a shielding material and thus relatively low values were shown at this height.

Some meshes showed a relatively high space dose rate. The reason is that the SG body, which has high radioactivity, was involved in these meshes. Therefore, these meshes were excluded in this assessment.

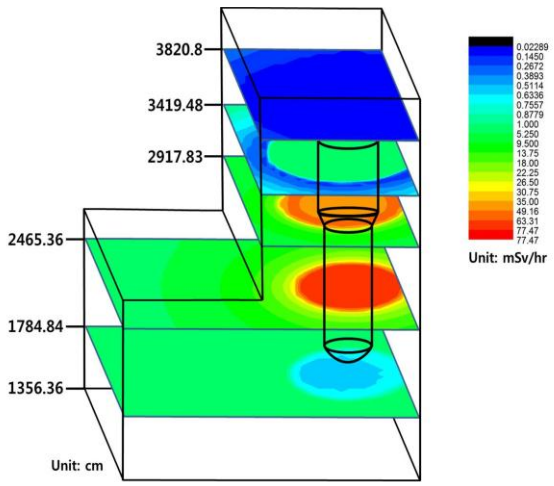

Fig. 2. Space dose rate of each height

2.3 Assessment of the space dose rate for the multi-source

Two methods are used to calculate the space dose rate for a multi-source. One is a Summing method; the space dose rate for each of the main components such as RPV, one of the SG, and one of the RCP, was calculated at the inside of the full containment and the results added to each space dose rate. The other is the Total calculation method; all considered components were involved when the MCNP input file was made. The target area of the space dose rate was the whole of the inside at the containment, and the size of each mesh was approximately 50 cm × 50 cm × 50 cm. The representative height is 24.65 m, which is the maximum space dose rate point in the preceded investigation.

The two methods (Summing and Total calculation methods) showed similar space dose rates. To compare these results numerically, the ratio of the result of the Total calculating method comparison to the result of the summing method was found (Fig. 3). The average value of the ratio of space dose rate between the two results was 0.99. Therefore, the space dose rates for the two methods similarly match.

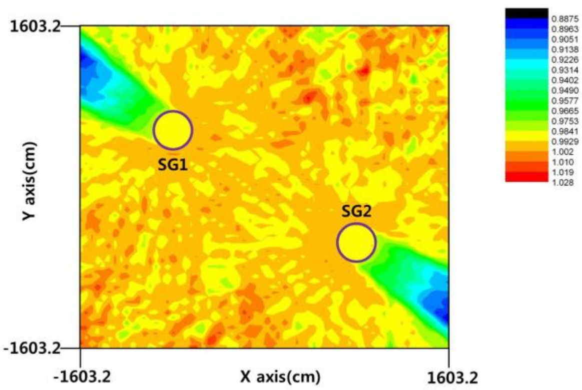

Fig. 3. The ratio of the Total calculating method with the Summing method

3. Conclusions

In this study, the distribution of dose rates around the SG at KORI-1 was calculated using an MCNP code, and the method to calculate the space dose rate for the multi-source was assessed about the two Steam Generators in KORI-1.

Assuming that the nuclear power plant operated for 40 years, the range of the space dose rate near the SG from 19.86 m to 29.45m height is 16.9 – 32.5 mSv/hr. However, these values can be decreased through several decontamination steps to reduce the radioactivity.

The results of the two methods to calculate the space dose rates for the multi-source were shown to be similar. In the case of the Total calculation method, a new MCNP input was made and the space dose rate was recalculated whenever some components were removed. However, in the case of the Summing method, each space dose rate of the existing components were only added, and not recalculated, after some components were removed. Therefore, the Summing method is more useful for evaluating the space dose rate in the decommissioning work.

4. Acknowledgement

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