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Monte Carlo based study of radiation field in a deep geological repository for high-level nuclear waste with different host rock types

Bo Pang[a,*], Héctor Saurí Suárez[b], Frank Becker[b]

[a] College of Physics and Energy, Shenzhen University (SZU), Nanhai Avenue 3688, 518060 Shenzhen, PR China
[b] Institute for Nuclear Waste Disposal (INE), Karlsruhe Institute of Technology (KIT), Hermann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany

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ABSTRACT

Disposal in deep geological formations with arrangements for monitoring and retrieval has been considered as one of the most promising management concepts of heat-generating, high-level nuclear waste. Typical design of a deep geological repository requires a shielding cask containing the waste, which is in turn located underground in an emplacement drift of a host rock formation. Various rock types, such as rock salt, clay stone and granite, are possible candidates as host rock for a deep geological repository. The massive host rock formation around the emplacement drift is the most important barrier against radionuclide release into the biosphere. However, the host rock layers have also influence on the radiation field in the emplacement drift, which should be taken into account when assessing occupational exposure in the repository. In the current study, impact of two different materials, i.e. rock salt and concrete (building material of the supporting liner for a drift in clay stone or granite), on radiation field around a nuclear waste package disposed in an emplacement drift was investigated with Monte Carlo method. The high-level nuclear waste was simulated with a ^{252}Cf neutron source. Both neutron and gamma dose rates and spectra around the waste package were calculated. It was found out that neutrons dominate the radiation field in the drift and the dose rates in the drift are enhanced due to backscattered radiation by the surrounding material layers. Furthermore, due to different material compositions of rock salt and concrete, the resulted neutron spectra have also different characteristics. In general, concrete moderates neutrons better than rock salt, which leads to a lower dose rates in the emplacement drift.

1. Introduction

Development of an appropriate concept for a long-term and safe disposal of heat-generating, high-level nuclear waste (HLW) from the use of commercial power plants is nowadays still a complex challenge. In the recent years, disposal in deep geological repositories with arrangements for monitoring and retrieval has been more and more considered as one of the most promising concepts to manage HLW (Kommission, 2016). Typical design of a deep geological repository requires a shielding cask containing the waste, i.e. the nuclear waste package, which is in turn located in an underground emplacement drift of a host rock formation. Various rock types, such as rock salt, clay stone and granite, have been proposed as possible candidates as host rock for deep geological repository (Stahlmann et al., 2015). The massive host rock layers around the emplacement drift provide the most important barrier against possible release of radionuclides into the biosphere. However, the host rock layers have also influence on radiation field in the emplacement drift, which must be taken into account when assessing occupational exposure in the facility.

In the current study, the geological repository was represented with an underground, horizontal emplacement drift. An emplacement drift in rock salt is due to the inherent stability of rock salt without further consolidation measures stable enough for emplacement operations. However, a drift of the same geometry in clay stone or granite must be supported by a concrete liner, which provides the engineered structure mechanical stability (Berner et al., 2013). Therefore, two different materials interacting with radiation in the emplacement drift, i.e., rock salt and concrete, were investigated. For simulations the general-purpose Monte Carlo N-Particle code MCNP6 (Goorley et al., 2012) was applied. The MCNP6 code was first validated for its application to calculate the radiation field in a geological repository, in which an iron-based shielding cask loaded with a ^{252}Cf neutron source simulating a HLW package was placed inside a rock salt drift. Subsequently, radiation fields in the emplacement drift with concrete liner of two different concrete types, i.e. the ordinary Portland concrete (OPC) and the low-pH concrete, were calculated with MCNP6. Finally, results in the emplacement drift with different materials were compared.

* Corresponding author.
E-mail address: bo.pang@szu.edu.cn (B. Pang).

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2. Methods and methodology

2.1. Recalculation of experiments with a ^{252}Cf neutron source in a simulated deep geological repository with MCNP6

In order to validate the MCNP6 code for its application to simulate the radiation field in a geological repository, an active handling experiment (AHE) with a ^{252}Cf neutron source simulating HLW (Engelmann et al., 1995; Knauf et al., 1997) was recalculated. The ^{252}Cf source was chosen due the fact that its spontaneous fission neutron spectrum is comparable to the neutron spectrum of spent nuclear fuel (Engelmann et al., 1995). The ^{252}Cf source consists of 60 small pellets (outer diameter 4.6 mm and length of 12 mm). Detailed information of encapsulation and configuration of the source can be found in Schlömer et al. (2013). The ^{252}Cf source has a total length of 60 cm and a neutron source strength of $1.7 \times 10^8\ s^{-1}$ (mainly stemming from spontaneous fission of ^{252}Cf) at the time when the experiments were performed. A cylindrical, iron-based shielding cask (outer diameter 92 cm and length 135 cm), henceforth referred to as the "AHE cask", containing the ^{252}Cf source was used to simulate a high-level nuclear waste package. The lid side of the shielding cask has a thickness of 40 cm, while that of the bottom side and the mantle side is up to 30 cm. The cask was located directly on the ground of an emplacement drift in a rock salt mine with its bottom surface is at 2 m distance to the end side of the drift. In order to study the impact of the surrounding rock salt layers on the radiation field, measurement were performed both underground in the emplacement drift and also above ground in an open area. Neutron and gamma dose rates and spectra were measured at various positions with different distances to the cask surfaces. The neutron dose rates and spectra were obtained with a Bonner sphere spectrometer (BSS) consisted of spherical proportional counters filled with ^{3}He (Knauf et al., 1997). For the measurement of gamma spectra the NE213 detector was applied, while gamma dose rates were measured with a ionization chamber, a Geiger-Müller (GM) counter and a Tissue Equivalent Proportional Counter (TEPC) (Engelmann et al., 1995).

Fig. 1 shows the MCNP6 model of the AHE experimental setups inside the rock salt drift. The detector positions for measurement of dose rates and spectra are designated with Arabic numbers 1–9 in the figure. In the reality, rock salt layers around the drift extend hundreds of meters. As a simplification, in the MCNP6 model, thickness of the rock salt layers around the drift was reduced to be at least 1 m, which is sufficient to account for possible interaction of the radiation with rock salt. The rock salt investigated was a typical rock salt type available in Germany. Its material composition was reported in Bernnat et al. (1995) with an average density of 2.2 g cm^{-3}. The water content in the rock salt is negligibly small. The air inside the rock salt drift was assumed as dry air at near see level. The ^{252}Cf source neutron spectrum was approximated with the Watt fission spectrum given in the MCNP6 manual (Pelowitz, 2013):

$$f(E_n) \propto exp\left(-\frac{E_n}{1.18000}\right)\cdot sinh\left(\sqrt{1.03419\cdot E_n}\right) \tag{1}$$

where E_n stands for the neutron energy in MeV. Also shown in the figure is the MCNP6 model of the iron-based AHE cask, which consists of a stainless cask (the internal cask) that is in turn covered by a polyethylene cylinder as neutron moderator. Finally, this body was contained in a thick cast iron cask (the external cask). The variance reduction technique "geometrical splitting" in MCNP6 was applied to obtain reasonable statistical errors within an acceptable computational time. The external cask was split into several sub-layers with individually assigned particle importance. As recommended in Shultis and Faw (2011), a layer farther away from the ^{252}Cf source was assigned to have a greater importance than a layer closer to the neutron source. The evaluated Nuclear Data File (ENDF) B-VI-1 continuous energy cross section was used in the simulations. The $S(\alpha,\beta)$ treatment for hydrogen in polyethylene was used to account for the up-scatter in materials with high hydrogen contents (i.e. the polyethylene cylinder as neutron moderator). In order to calculate the dose rate in the rock salt drift, the track-length tally (determination of neutron and photon fluence) was applied together with the fluence-to-ambient dose equivalent conversion coefficient of the ICRP publication 21 (ICRP, 1971), the same as that used in the experiment (Knauf et al., 1997). The same energy bins used in the measurement to unfold the measurement results of BSS (Engelmann et al., 1995) were also applied in the MCNP6 simulations to calculate the neutron spectra. In order to calculate the above ground dose rates and spectra, the same MCNP6 model as shown in Fig. 1 was used with the rock salt layers were replaced by air.

2.2. Calculations of radiation field in deep geological repository of different host rock types with MCNP6

For the purpose of a fair comparison, the same geometrical

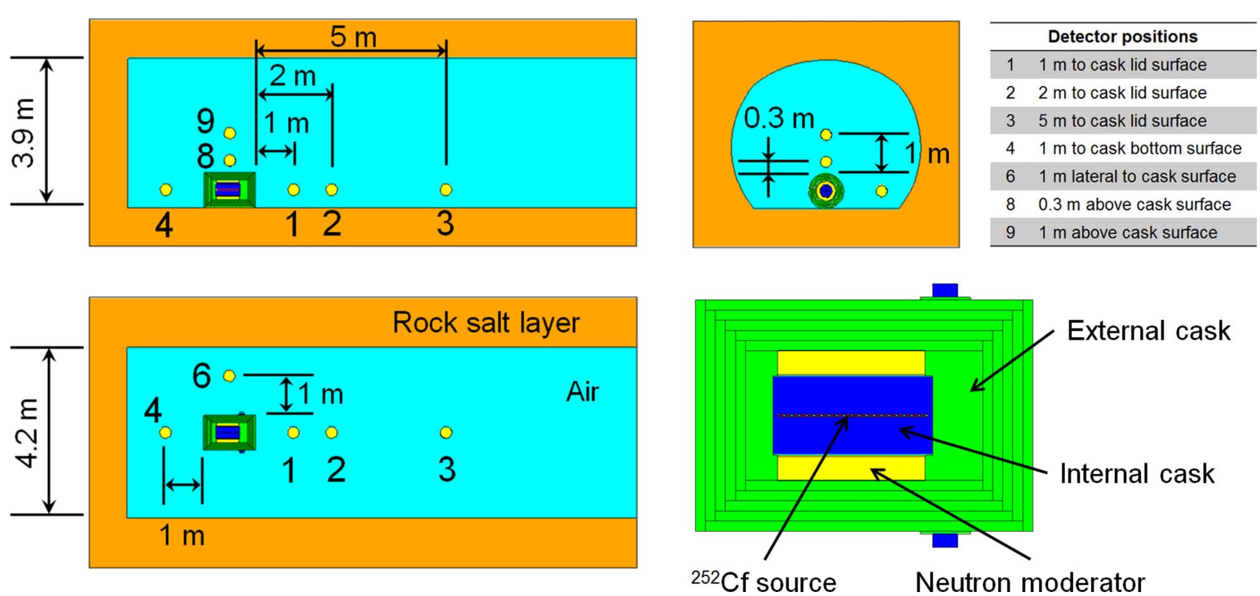

Fig. 1. MCNP6 model of the AHE experimental setups inside a rock salt drift, in which a shielding cask loaded with ^{252}Cf source simulating a high-level nuclear waste package was placed on the ground. Measurement of both neutron and gamma dose rates and spectra were performed at various distances to the shielding cask.