Applied Radiation and Isotopes 139 (2018) 201–208

Contents lists available at ScienceDirect

Applied Radiation and Isotopes

journal homepage: www.elsevier.com/locate/apradiso

Effects of materials and design on the criticality and shielding assessment of canister concepts for the disposal of spent nuclear fuel

Miguel Morales Gutiérrez, Stefano Caruso*, Nikitas Diomidis

National Cooperative for the Disposal of Radioactive Waste (Nagra), Hardstrasse 73, 5430 Wettingen, Switzerland

HIGHLIGHTS

- Evaluation of a multi-barrier system for spent nuclear fuel geological disposal.
- Development and analysis of advanced disposal canisters concepts.
- Performed criticality safety assessment for proposed canister configurations.
- Shielding assessment to prevent radiolysis processes.
- Results very promising for long term safety assessment.

ABSTRACT

According to the Swiss disposal concept, the safety of a deep geological repository for spent nuclear fuel (SNF) is based on a multi-barrier system. The disposal canister is an important component of the engineered barrier system, aiming to provide containment of the SNF for thousands of years. This study evaluates the criticality safety and shielding of candidate disposal canister concepts, focusing on the fulfilment of the sub-criticality criterion and on limiting radiolysis processes at the outer surface of the canister which can enhance corrosion mechanisms.

The effective neutron multiplication factor (k-eff) and the surface dose rates are calculated for three different canister designs and material combinations for boiling water reactor (BWR) canisters, containing 12 spent fuel assemblies (SFA), and pressurized water reactor (PWR) canisters, with 4 SFAs. For each configuration, individual criticality and shielding calculations were carried out. The results show that k-eff falls below the defined upper safety limit (USL) of 0.95 for all BWR configurations, while staying above USL for the PWR ones. Therefore, the application of a burnup credit methodology for the PWR case is required, being currently under development. Relevant is also the influence of canister material and internal geometry on criticality, enabling the identification of safer fuel arrangements. For a final burnup of 55MWd/kgHM and 30y cooling time, the combined photon-neutron surface dose rate is well below the threshold of 1 Gy/h defined to limit radiation-induced corrosion of the canister in all cases.

1. Introduction and scope

Nagra is the Swiss national cooperative responsible for the disposal of all types of radioactive waste produced in Switzerland. One of its main objectives is the planning and construction of a deep geological repository for the final disposal of the spent nuclear fuel and high level radioactive waste. By about 2024, Nagra intends to apply for a site licence for the repository. This will require documentary evidence that all factors relating the construction, operation and long-term safety of the repository have been considered.

The disposal concept envisaged by Nagra is to emplace canisters containing the SF and HLW in underground drifts which are subsequently completely backfilled with bentonite clay. The radial distance between the canister outer surface and the tunnel wall will be approximately 75 cm, while there will be an axial gap between canisters of 3 m. The regulatory requirement for the lifetime without breach of containment for disposal canisters for SF and HLW is 1000 years. However, Nagra has set increased design target lifetimes to ensure a significant safety margin. To achieve this, the canister development program proceeds by considering factors such as long-term mechanical integrity, requirements related to the sub-criticality safety and the avoidance of radiation-induced corrosion. Various options are being

* Corresponding author.
E-mail address: stefano.caruso@nagra.ch (S. Caruso).

https://doi.org/10.1016/j.apradiso.2018.05.016
Received 13 December 2017; Received in revised form 5 April 2018; Accepted 17 May 2018
Available online 19 May 2018
0969-8043/

explored with respect to selection of materials and design concepts for SF and HLW disposal canisters (Johnson and King, 2003; Landolt et al., 2009; Holdsworth et al., 2014).

It is currently envisaged that HLW repository implementation will occur at about 2060. Given the long time until canister emplacement in the repository, there is the potential to take advantage of developments in materials technology. Nonetheless, carbon steel stands out as a strong candidate due to its mechanical integrity, the expected good corrosion performance under disposal conditions and the advanced state of overall technical maturity related to fabrication and sealing. Prior work in Switzerland on carbon steel canister for the disposal of nuclear waste goes back to 1985, when a first design concept was developed for a HLW canister (Steag and Motor-Columbus, 1985). The design of the carbon steel disposal canister was further refined recently (Patel et al., 2012). Corrosion aspects related to the long-term performance for a carbon steel SNF canister were discussed in work by (Johnson and King, 2003) and in more detail in (Landolt et al., 2009). Current carbon steel canister designs have a target lifetime of 10,000 years.

The application of copper as a coating material could provide significantly longer canister lifetimes (100,000 years) due to the very low corrosion rate under repository conditions (Scully and Edwards, 2013; King et al., 2011). As a result, copper coating technology is being actively developed (Keech et al., 2014; Jakupi et al., 2015). The copper coatings could be applied on a carbon steel internal structure (Bastid et al., 2015; Allen et al., 2016). Alternatively, cast iron, due to its strength, good casting properties, and low cost is also under consideration as a candidate material for the structural element of a coated SNF canister. The use of cast iron as a canister material has been considered by Nagra in the past (Johnson and King, 2003) but there is also significant international experience (SKB, 2010; Nolvi, 2009).

The current approach assumes PWR and BWR SFAs are assorted separately into 4-SFA and 9-SFA carbon steel canisters, respectively (Johnson et al., 2012). Another design is currently considered for the disposal of 12 BWR SFAs instead of the 9 SFAs, aiming to reduce the total number of canisters. Previous studies were carried out to prove criticality and shielding safety for the 9-SFA BWR canister model (Johnson et al., 2012). The inclusion of a larger number of SFAs enforces a re-evaluation of criticality and shielding safety for the new canister concepts.

The 12-SFA BWR canister concept is currently considered by both Swedish and Finnish waste management organisations, respectively SKB (SKB, 2011) and Posiva (Anttila, 2005). Their criticality safety studies, which are the base for their construction licence applications, provide fuel loading curves which indicate the minimal acceptable burnup of the fuel to be loaded in the canister for the 12-SFA BWR as well as the 4-SFA PWR configurations. Long term material degradation effects are also considered for the long-term safety analysis (Agrenius and Spahiu, 2016).

This paper gives an overview of the main results obtained from shielding and criticality calculations for SF disposal canisters, in particular on the investigations of the 12-SFA BWR as well as the 4-SFA PWR cases. These criticality safety analyses assume the fuel contained in the canisters to be unirradiated and intact. The application of a BU-credit methodology, which takes credit for the reduction of the reactivity associated with fuel depletion, is not part of these scoping investigations. It is currently under development and it will be documented elsewhere (Herrero et al., 2016).

2. Methodology

With respect to criticality safety assessment, analyses for all canister designs and FA types are carried out using the state-of-the-art KENO-VI Monte Carlo (MC) code, available as part of the SCALE 6.2 package (ORNL, 2016), in combination with a continuous-energy implementation of the ENDF/B-VII.1 database.

As a MC code, KENO-VI performs eigenvalue calculations for neutron transport in fissile systems, primarily to calculate k-eff and flux distributions, by means of a stochastic iterative process. Neutrons are randomly started in the fissile material regions and tracked throughout the geometry. During the transport, neutrons can undergo different kinds of interactions, which are randomly determined from appropriate probability distributions.

From the comparison between the numbers of neutrons in subsequent generations, the value of k-eff can be extracted. A system in which the neutron population decreases with time (k-eff < 1) and eventually will die off is called subcritical. To achieve a safe subcritical condition, according to international safety standards and international norms (IAEA, 2014), the k-eff of the system, including all uncertainties, must not exceed a defined USL, which for the aim of this scoping study can be preliminary set to 0.95, by assuming a conservative administrative margin of 5% without considering further uncertainty terms (ANS, 1998). Bias and uncertainty terms are not calculated in this preliminary study but intended for the final assessment.

Optimal moderation conditions must be assumed in criticality safety assessment; therefore, the presence of water is considered inside the canister cavity.

Shielding calculations are performed by means of MAVRIC (Peplow, 2011), a well-established sequence based on the MC code Monaco and implementing automatic variance reduction techniques. Included as part of the SCALE 6.2 package, MAVRIC is applied for this study in combination with 27 and 19 energy-dependent cross section (XS) libraries for neutrons and photons, respectively, based on the ENDF/B-VII.1 database.

The MAVRIC sequence allows simulating radiation transport on problems that, due to their complexity, are too challenging to be solved using analog MC methods. It aims to evaluate fluxes and dose rates with low uncertainties in reasonable computational times even for deep penetration problems. Denovo, a three-dimensional, discrete ordinates code using the CADIS (Consistent Adjoint Driven Importance Sampling) or the FW-CADIS (Forward-Weighted CADIS) methodologies, calculates the space and energy-dependent adjoint solution optimized for the considered problem on a user-defined Cartesian grid superimposed onto the geometry. The adjoint information is then used to generate a space and energy-dependent importance map (i.e., weight windows) and a mesh-based biased source distribution. MAVRIC then passes the importance map and biased source distribution to Monaco, a fixed-source 3D Monte Carlo shielding code that evaluates neutron and photon fluxes and response functions for specific point, region and mesh tally detectors.

The fact that both KENO-VI and MAVRIC share the SCALE Generalized Geometry Package (SGGP) for the definition of the geometry simplifies significantly the task of generating different models for criticality and shielding calculations, since the geometries and material definitions are interchangeable.

Fuel compositions and source spectra and strengths needed for shielding calculations are computed by means of ORIGAMI, the new source term generator implemented as part of SCALE 6.2. ORIGAMI relies upon one-group reactor data libraries produced previously with lattice physics methods such as TRITON and it is suitable for performing not only 0D, but also 1D and 3D depletion and decay analysis of SNF. Using a nominal irradiation history for the whole assembly and axial and radial relative power distribution, ORIGAMI evaluates the burnup and composition at each nodal and pin location. For the aim of this study, however, ORIGAMI is used with a 0-D approach, wherein the reactor data library is automatically interpolated for the problem-specific initial fuel enrichment, moderator void fraction (only for BWR fuel), and mid-cycle burnups. The source term for the shielding calculations is obtained for a final burnup of 55MWd/kgHM and 30 years cooling time after the final unloading of the fuel from the reactor (minimum time expected between the last core unload and the time of emplacement in repository).