Annals of Nuclear Energy 112 (2018) 322–327

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

Experimental study on heat-removal performance in accordance with mesh size of screen installed at opening of inlets and outlets of concrete storage cask

Ki-Seog Seo, Kyoung-Sik Bang *, Seung-Hwan Yu, Ju-Chan Lee, Woo-Seok Choi

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989 beon-gil, Yuseung-gu, Daejeon 305-353, South Korea

ARTICLE INFO

Article history:
Received 29 December 2016
Received in revised form 29 September 2017
Accepted 1 October 2017
Available online 17 October 2017

ABSTRACT

Spent nuclear fuel generated at nuclear power plants must be safely stored during interim storage periods. A concrete storage cask developed for this purpose should be able to adequately emit the decay heat from the spent nuclear fuel; it should also maintain the temperatures of the spent nuclear fuel assemblies within allowable values under normal and off-normal conditions and during an accident. However, since the thermal conductivity of concrete is low and the allowable temperature of concrete is lower than that of steel, the concrete storage cask must be designed to have heat-removal capability with appropriate reliability. A passive heat-removal system was designed to maintain the temperatures of the fuel assembly cladding material and concrete storage cask components within the allowable limits; it consists of four air inlets and four air outlets with openings that are covered by mesh screens to prevent debris or wildlife from entering the ventilation ducts. Thermal tests were performed to evaluate the effect of the mesh size of each screen on heat-removal performance of the concrete storage cask. The screen mesh size was estimated to have an insignificant effect on the temperature rise of the canister surface and the over-pack surface, but it had a considerable effect on the temperature rise of the components of the over-pack body. As the screen mesh size decreased, the heat-removal by natural convection cooling through the passive heat-removal system was reduced, and the temperature of the concrete storage cask rose.

1. Introduction

The management of spent nuclear fuel generated in nuclear power plants has become a major policy issue in Korea owing to continued delays in the approval and construction of a safe and permanent disposal facility. Most nuclear power plants store their spent nuclear fuel in wet storage pools. After decades of use, however, most of the storage pools have reached their maximum capacities. It is therefore essential to find suitable alternatives for storing spent nuclear fuel if these nuclear power plants are to continue their operation.

Concrete storage casks are one possible solution for the interim storage problem. Therefore, a concrete storage cask containing 21 spent nuclear fuel assemblies is currently being developed by the Korea Radioactive Waste Agency (KORAD).

A typical concrete storage cask consists of three separate components: an over-pack, a canister, and a transfer cask. The spent nuclear fuel assemblies are loaded and sealed inside the canister. Using the transfer cask, the canister is then transferred into a cylindrical over-pack for on-site dry storage. The canister may be removed from the over-pack at any time and placed in the transport cask for relocation to a permanent disposal facility.

Fig. 1 shows a schematic diagram of a concrete storage cask. The structural casing of the over-pack is made of carbon steel, and the inner cavity of the casing is filled with concrete, which acts as a radiation shield. The temperatures of the spent nuclear fuel assemblies must be maintained within the allowable values under normal and off-normal conditions, and during an accident.

According to a report from Pacific Northwest Lab., USA, (PNL-4835, 1983), spent fuel claddings have a typical temperature of 380 °C or lower at the beginning of dry storage for a 5-year cooled fuel assembly under normal conditions and a minimum of 20 years of cask storage. According to the United States Nuclear Regulatory Commission's standard review plan for spent fuel dry storage systems at a general license facility (NUREG-1536, 2010), the maximum cladding temperature of low burn-up spent nuclear fuel (less than 45,000 MWd/MtU) should not exceed 400 °C under normal storage conditions and for short-term loading operations, including cask drying and backfilling; for off-normal and accident conditions, the maximum cladding temperature should not exceed 570 °C.

* Corresponding author.
E-mail address: nksbang@kaeri.re.kr (K.-S. Bang).

https://doi.org/10.1016/j.anucene.2017.10.005

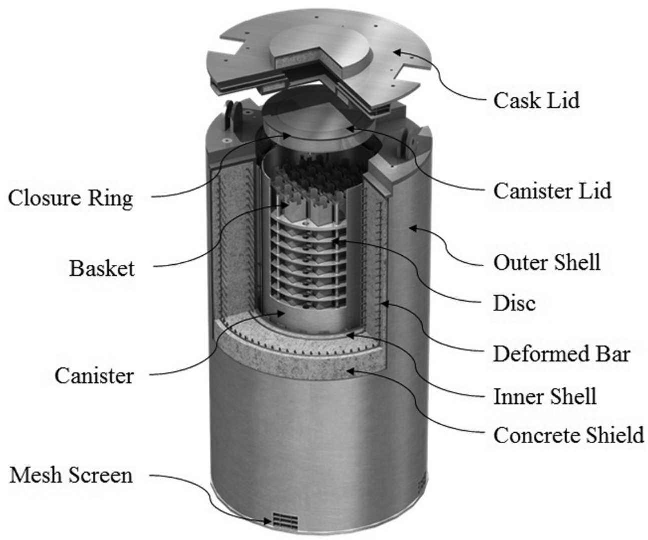

Fig. 1. Configuration of the concrete storage cask.

Therefore, the concrete storage cask must be designed to have heat-removal capability with appropriate reliability (10CFR72, 2005). However, the thermal conductivity of concrete is not adequate for this purpose, and the allowable temperature of concrete is lower than that of steel. The American Concrete Institute (ACI 349-13, 2013) specifies a limit of 66 °C for the normal operating temperature of concrete, except for local areas, which may not exceed 93 °C, and a short-term or accident temperature limit of no more than 177 °C. Therefore, a passive heat-removal system was designed to maintain the temperatures of the fuel-assembly cladding material and concrete storage cask components within these allowable limits. The passive heat-removal system consists of four inlets and four outlets, and their openings are covered by screens of mesh structure to prevent debris or wildlife from entering the ventilation ducts. Depending on its mesh size, each screen will have a different effect on the heat removal of the concrete storage cask.

The features of the concrete storage cask examined in this study are provided in Table 1. Its outer diameter and overall height are 3306 and 6180 mm, respectively, and it weighs approximately 148 t. The concrete storage cask accommodates 21 pressurized water reactor (PWR) spent fuel assemblies with a burn-up of 45,000 MWd/MtU and a cooling time of 10 years. The decay heat from the 21 PWR spent fuel assemblies is 16.8 kW.

Table 1
Description of the concrete storage cask.

Item	Description
Storage capacity	21 PWR assemblies
Components	Over-pack Canister
Dimension	Over-pack: 3306 mm(Ø) × 6180 mm(l) × 700 mm(t) Canister: 1686 mm(Ø) × 4880 mm(l) × 25 mm(t)
Weight	Over-pack: 148 t (loaded canister) Canister: 24.1 t (loaded fuel)
Material	Over-pack: Stainless steel housing & concrete Canister: Stainless steel & boral (B_4C + Al)
Design basis fuel	Burn-up: 45,000 MWd/MtU Cooling time: 10 years Enrichment: 4.5 wt% ^{235}U Decay heat: 16.8 kW

This paper discusses the experimental approach used in the present study to evaluate the heat-removal performance under normal conditions in accordance with the mesh size of the screen installed at the openings of the inlets and outlets.

2. Thermal test

2.1. Description of the test model

The test model is a full-scale model of the concrete storage cask. During actual storage, the lid of the canister is welded to the body of the canister to secure the contents within the confinement. During the thermal tests, however, the lid was only bolted to the body of the canister to allow it to be opened after the test. The lid had one hole for an electric heater, which was used to simulate the heat dissipated by the 21 PWR spent fuel assemblies. The electric heater was accommodated within the canister and fixed onto the top of the lid with a Swage lock connector.

2.2. Heat transfer mode and measurement system

During storage, heat is generated by the spent nuclear fuel assemblies within the canister and transferred to the surface of the canister by conduction, convection, and radiation. This heat is then transferred from the surface of the canister to the inner surface of the over-pack by convection and radiation. The over-pack is designed to dissipate the heat from the canister through a passive heat-removal system, which involves a natural convective air flow through the annular area between the canister and the inner surface of the over-pack. Therefore, heat transfer from the over-pack to the ambient atmosphere is accomplished through two mechanisms: (1) the heat, which is conducted through the over-pack's body, is dissipated from the exterior surface of the over-pack to the ambient atmosphere by convection and radiation; (2) the air, which is heated in the annular area, is vented to the ambient atmosphere through the outlets of the passive heat-removal system.

The overall mass flow rate of air, $\dot{m}$, through the passive heat-removal system is calculated as follows (Street et al., 1996):

$$\dot{m} = \rho A V \tag{1}$$

where ρ and V are the density and velocity of the air, respectively, and A is the cross-sectional area of the outlet.

The heat transferred to the ambient atmosphere through the passive heat-removal system can be obtained from the inlet and outlet fluid temperatures through an energy balance equation (Incropera and DeWitt, 2002),

$$q_A = \dot{m} C_P \Delta T \tag{2}$$

where q_A is the heat transferred to the air, $\dot{m}$ is the mass flow rate, C_p is the specific heat of the air at constant pressure, and $\triangle T$ is the differential air temperature from the inlet to the outlet.

The heat transfer from the exterior surface of the over-pack to the ambient atmosphere is (Holman, 1985):

$$q_S = hA(T_S - T_a) + \sigma \varepsilon A\left(T_S^4 - T_a^4\right) \tag{3}$$

where q_s is the heat flow rate from the exterior surface of the over-pack to the ambient atmosphere (W), h is the natural convective heat transfer coefficient (W/m^2 K), A is the surface area of the over-pack (m^2), T_s is the temperature at the surface (K), T_a is the ambient temperature (K), σ is the Stefan-Boltzmann constant (=5.669 × 10^{-8} W/m^2 K^4), and ε is the emissivity of the exterior surface of the over-pack.