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Optimization of disposal method and scenario to Reduce High Level Waste Volume and Repository Footprint for HTGR

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ABSTRACT

To reduce volume of High Level Waste (HLW) and the footprint in the geological repository of a High Temperature Gas-cooled Reactor (HTGR), this study optimizes the disposal method and scenario of the HLW.

By virtue of high burn-up, high thermal efficiency and pin-in-block type fuel, the HTGR more effectively reduces the HLW volume and its footprint than those of Light Water Reactor (LWR) in our previous study. In this study, the disposal method and scenario are optimized. To optimize the disposal method, the geological repository layout is the horizontal emplacement based on the KBS-3H concept, rather than the vertical emplacement based on the KBS-3V concept adopted in our previous study.

In comparison with the earlier study, the horizontal emplacement reduced the repository footprint in direct disposal by 20% in the same scenario. By extending the cooling time by 40 years before disposal, the footprint was reduced by 50%. In disposal with reprocessing, extending cooling time by 1.5 years between discharge and reprocessing reduced the number of canister generated by 20%. Extending the cooling time by 40 years pre-disposal reduced the footprint per unit of electricity generation by 80%.

Moreover, by employing four-group partitioning technology without transmutation, the footprint can be reduced by 90% with a cooling time of 150 years.

1. Introduction

High Temperature Gas-cooled Reactors (HTGRs) have attracted a huge attention from a safety point of view (Ohashi et al., 2011), especially from the Fukushima Daiichi nuclear power plant disaster in Japan in 2011. The environmental burden of radioactive waste is the most important consideration in Nuclear Power Generation (NPG). After the disaster, the significance of nuclear technology was frequently questioned and discussed. In this context, the number of High Level radioactive Waste (HLW) packages and the footprint in a geological repository for HTGR were evaluated in the previous study (Fukaya and Nishihara, 2016).

The Gas Turbine High Temperature Reactor (GTHTR300) (Yan et al., 2003) is an annular-core type HTGR that generates 600 MW thermal power from pin-in-block type fuel. The major specifications of the GTHTR300 (Nakata et al., 2003) are listed in Table 1. The burn-up is approximately 120 GWd/t, approximately triple that of Light Water Reactor (LWR) with a burn-up of 45 GWd/t. Moreover the thermal efficiency is 30% higher in an HTGR than that in a LWR (45.6% versus 34.5%). Consequently, HTGR generate less HLW LWRs. The previous study proposed an effective waste-loading method that exploits pin-in-block type fuel. In direct disposal and disposal with reprocessing, this method reduces the number of canisters and the footprint per electricity generation of the HTGR by 60% and 30% respectively, compared with those of PWR case.

Previous study has adopted vertical emplacement based on the KBS-3 V concept (SKB, 2010) named after its proposer, Svensk Kärnbränslehantering AB (SKB), which is the most achievable one. As HTGR waste generates less heat than LWR waste, the geological repository footprint of vertical emplacement is determined by structural limitations, which ensure the structural integrity of the repository. Meanwhile, in horizontal emplacement based on the KBS-3H concept (SKB, 2010), only the drift intervals must be structurally limited. The small drift diameter reduces the structural limitations, thus reducing the repository footprint. The waste package pitches are unrelated to the repository integrity, and are determined only by the dimensions of the engineering barrier. Owing

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Table 1
Main specifications of GTHTR300.

Item	Value
Thermal power (MWt)	600
Thermal efficiency (%)	45.6
Uranium inventory (t)	7.09
^{235}U enrichment (wt%)	14
Fuel particle	SiC coated particle
Kernel diameter (μm)	550
Particle diameter (μm)	1010
Particle packing fraction (%)	28.5
Block across flat (mm)	410
Fuel rod numbers	57
Fuel rod diameter (mm)	26
Coolant hole diameter (mm)	39
Burnable poison	B4C-C composite
Block height (mm)	1050
Cycle length (days)	706.0
Number of batch	2
Discharge burn-up (GWd/t)	119.5
Initial heavy metal inventory per electricity generation (tIHM/TWeh)	0.765

to its lower footprint, horizontal displacement is dominated by thermal limitation, which confers a buffer functionality. Therefore, the disposal scenario should be reconsidered to reduce the footprint with the decaying heat generation. Moreover, the number of waste packages generated in disposal with reprocessing can be reduced by extending the cooling time between discharge and reprocessing.

Partitioning & Transmutation (P&T) is another popular method that reduces the volume and footprint of HLW. Therefore, this study also considers partitioning technology, which has been already demonstrated.

The present study seeks the optimal disposal method and scenario that could reduce the volume and footprint of HTGR-generated HLW as an introductory option, without requiring innovative technology. The disposal method is optimized by changing the disposal layout of horizontal emplacement and the waste pitches. The disposal scenario was optimized by changing the duration between the spent fuel discharge, reprocessing and disposal. To optimize the process, the cooling time before disposal is limited to within 100 years (to within 150 years in the partitioning case). Section 2 summarizes the major results and calculation conditions of the previous study, and Section 3 describes the optimal disposal scenario and calculation method of the present study. The repository layout of each disposal scenario is designed in Section 4. The waste reduction effect of the partitioning technology is investigated in Section 5. Finally, the acceptance of the proposed disposal scenario and the HLW volume and footprint reduction of the scenario are described in Section 6.

2. Evaluation conditions, methods and reference case in the previous study

2.1. Scenario, geological repository design, and safety requirement

In the previous study, the reduction effect on HLW volume and its footprint were evaluated and compared with those of LWR. The scenario, repository design, and specifications for the disposal of HLW generated from the LWR fuel cycle are given in the Japan Atomic Energy Commission (JAEC) report (JAEC, 2004). According to this plan, the Spent Fuels (SFs) are reprocessed 4 years after discharge, and the vitrified wastes are disposed of 50 years after reprocessing (54 years after discharge). Directly disposed SFs are disposed of 54 years after discharge to match the disposal-with-reprocessing plan.

The most achievable configuration, namely, vertical emplacement based on the KBS-3V concept (SKB, 2010) described in Section 1, was selected as the reference case. The repository design depends on two parameters: the tunnel interval and the waste package pitch. These parameters are limited by the safety requirement of structural integrity and maintenance of the buffer function. The limitations imposed by structural integrity were evaluated by structural analysis (JAEC, 2004), and they were taken into the previous study. In maintaining the buffer function, the main problem is the maximum temperature in the bentonite buffer. When the temperature exceeds 100 °C, the high temperature changes its property and loses its ability to delay nuclide migration. To allow for uncertainties, the target upper temperature is set to 90 °C (JAEC, 2004). The maximum temperature of the bentonite for the HTGR case was evaluated by time-dependent thermal conductivity calculations performed in ANSYS code (ANSYS, Inc., 2013), which solves the thermal equation by the finite element method with an implicit time integral technique.

In addition, the waste must never reach criticality in the repository forever. In direct disposal, the waste package includes residual ^{235}U, and generated ^{239}Pu and ^{241}Pu. Criticality safety is also confirmed in MVP calculations (Nagaya et al. 2006). MVP is a neutron transport calculation code based on the Monte Carlo method, and the calculations use evaluated nuclear data of JENDL-4.0 (Shibata et al. 2011). MVP code is suitable for HTGR calculation because it applies a statistical geometry model that handles the double heterogeneity effect, the self-shielding effect caused by the complicated geometry of Coated Particle Fuel (CPF) (Murata et al. 1997).

2.2. Burn-up calculation and characteristics of heat generation

The fuel burn-up composition and decay heat were evaluated in ORIGEN (Croff, 1983) code. However, ORIGEN code cannot evaluate the neutron spectrum in a core and uses a single energy group cross section libraries. Libraries for the major reactors have been already developed. The Japan Nuclear Data Committee (JNDC) has developed ORIGEN library of ORLIBJ40 (Okumura et al. 2012) based on evaluated nuclear data of JENDL-4.0. ORLIBJ40 includes libraries for LWRs and Fast Breeder Reactors (FBRs). In the previous study, the PWR47J40 library in ORLIBJ40 was used for Pressurized Water Reactor (PWR) calculations, and libraries for HTGR (which did not previously exist) with the nuclear characteristics of GTHTR300 were evaluated by MVP code using the evaluated nuclear data of JENDL-4.0, JEFF-3.1.2 (Koning et al., 2011), JENDL/A-96 (Nakajima, 1991), JEFF-3.1/A (Koning et al. 2006) and TENDL-2011 (Koning and Rochman, 2011b).

The burn-up compositions and decay heats of the PWR and HTGR were calculated under the conditions listed in Table 2. The decay heat per burn-up curves are shown in Fig. 1. The decay heats of Fission Products (FPs) from HTGR and LWR coincide. The actinoid decay heats are approximately 20% smaller in the HTGR than in the PWR because HTGR generates fewer TRans Uranium (TRU) nuclides. Apart from neptunium, the TRU nuclides are converted from ^{238}U. The generated weight per burn-up of HTGR is approximately half

Table 2
Conditions of the burn-up calculations.

	PWR	HTGR
Enrichment (wt%)	4.5	14.0
Specific power (MW/t)	38.0	84.6
Burn-up days (day)	1184	1412
Burn-up (GWd/t)	45	119.5