Technical Paper https://doi.org/10.7733/jnfcwt.2022.032

A Study on Thermal Load Management in a Deep Geological Repository for Efficient Disposal of High Level Radioactive Waste

Jongyoul Lee*, Heuijoo Choi, and Dongkeun Cho
Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon 34057, Republic of Korea

(Received July 7, 2022 / Revised August 26, 2022 / Approved September 21, 2022)

Technology for high-level-waste disposal employing a multibarrier concept using engineered and natural barrier in stable bedrock at 300–1,000 m depth is being commercialized as a safe, long-term isolation method for high-level waste, including spent nuclear fuel. Managing heat generated from waste is important for improving disposal efficiency; thus, research on efficient heat management is required. In this study, thermal management methods to maximize disposal efficiency in terms of the disposal area required were developed. They efficiently use the land in an environment, such as Korea, where the land area is small and the amount of waste is large. The thermal effects of engineered barriers and natural barriers in a high-level waste disposal repository were analyzed. The research status of thermal management for the main bedrocks of the repository, such as crystalline, clay, salt, and other rocks, were reviewed. Based on a characteristics analysis of various heat management approaches, the spent nuclear fuel cooling time, buffer bentonite thermal conductivity, and disposal container size were chosen as efficient heat management methods applicable in Korea. For each method, thermal analyses of the disposal repository were performed. Based on the results, the disposal efficiency was evaluated preliminarily. Necessary future research is suggested.

Keywords: Spent nuclear fuels, High-level waste, Geological disposal, Decay heat, Thermal management, Bentonite block, Disposal container, Engineered barrier, Natural barrier

*Corresponding Author.
Jongyoul Lee, Korea Atomic Energy Research Institute, E-mail: njylee@kaeri.re.kr, Tel: +82-42-868-2071

ORCID

Jongyoul Lee	http://orcid.org/0000-0001-8482-9008	Heuijoo Choi	http://orcid.org/0000-0001-9253-7697
Dongkeun Cho	http://orcid.org/0000-0003-4152-8605		

1. Introduction

High-level radioactive waste including spent nuclear fuel generated by utilizing nuclear energy should be managed in such a way that it is safely isolated from the general public and the natural environment for a long period of time to prevent harm by high heat and radiation in the present or future. Currently, the most widely accepted disposal method for long-term isolation is to disposal in a deep geological repository designed and constructed with multiple barriers composed of engineered and natural barriers so that the waste can be completely isolated in a stable deep geological environment. An important consideration, both in the near field of the engineered barrier and in the far field of the natural barrier, is the heat generated from the waste due to the large amount of fission products present in the high-level waste loaded in the disposal container. Many technical uncertainties in the design of high-level waste disposal systems are related to the temperature regime due to the release of this thermal energy and the effect of this temperature region on the disposal system. The heat generated and released from the radioactive waste will eventually be transferred to the surrounding rock through all containers and buffers around the waste. Additionally, the rate and duration of heat release can affect the physical and chemical properties of barriers in a repository system [1].

Thermal changes in a disposal container or buffer material may affect its integrity or effectiveness. For example, as the disposal container surface temperature increases, the susceptibility to corrosion increases, and the container may come into contact with groundwater earlier than if the waste was not generating heat [1]. The stability of underground openings in thermally heated strata is also an important consideration, and when the bedrock is thermally affected, groundwater movement patterns and the environment of transport routes for radionuclides may be altered [1].

As for the thermal effect in the near field of the repository, if heat emitted from radioactive waste disposed of in a deep geological formation diffuses through the near field to the host rock, which is the far field, cracks may occur because of the large-scale thermal expansion and this will create thermal stress because of the displacement. In addition, new groundwater flow paths may be created according to the deformation of the bedrock in the far field, and the borehole and the excavated opening may deteriorate. On the contrary, if this effect manifests in a rock salt or clay rock repository, it may also have a desirable effect as it can close the opening and seal the flow of fluid in the repository [2].

As a thermal effect in the near field of the repository, the disposal container for containment of disposed radioactive waste, which is an important component of the engineered barrier, could be corroded by evaporation of the surface water on the disposal container due to the heat emitted from the waste in the early stage after the repository is closed. This is because the evaporation could result in the deposition of potentially corrosion-accelerating salts on the disposal container surface [3]. It is known that the corrosion rate of copper, a disposal container material, approximately doubles for each 10°C rise in temperature [4].

In addition, in the case of bentonite buffer, which is an engineered barrier component, the temperature increases due to heat diffused from the waste, and the transformation process in which smectite in the buffer material becomes illite, a non-swellable clay, commonly takes place in petrification and hydrothermal chemical reactions. Because the illitization of bentonite affects the swelling properties, hydraulic conductivity, and plasticity, it also affects the safety and performance requirements of the buffer material.

In developing the concept of a repository for high-level radioactive waste that generates heat, thermal management is considered as one of the important factors because it limits key factors such as repository layout, waste disposal container specifications, and design and operation of other engineered barrier components. Thermal limitations for thermal management due to thermal effects in a repository may limit the temperature or limit the thermal behavior of components of the disposal system, and such limitations may be imposed on the host rock (or other natural features) or engineered barrier systems. Therefore, in order for the decay heat generated by the disposed high-level waste to be controllable, the heat load of the repository must be systematically analyzed and managed, and continuous research is being conducted for this purpose.

In this paper, a conceptual analysis was performed on the effect of heat in a geological repository system for high-level radioactive waste and on the thermal management method. This conceptual analysis analyzes the thermal stability of the repository at the initial stage of concept development, followed by thermal-hydraulic-mechanical complex behavior analysis in the next stage [5, 6]. For this purpose, the analysis was performed according to the thermal effect on the behavior of engineered barriers and natural barriers, and various thermal management methods were reviewed to analyze different thermal management concepts for efficient geological disposal. In addition, the current status of thermal load management research in overseas countries conducting disposal research and thermal limitations in various repositories were reviewed. Based on the results, the characteristics of various thermal management methods were analyzed, and efficient thermal management methods applicable in Korea and their efficiency in terms of the disposal area where they would be applied were conceptually analyzed.

2. Review of Current Status of Thermal Management

2.1 Crystalline Rocks

2.1.1 Sweden

SKB, a Swedish radioactive waste management organization, plans to use bentonite as a buffer material for the KBS-3V [7] concept, a concept for a spent nuclear fuel disposal system developed in Sweden (Fig. 1). The reference bentonite is MX-80, installed in pre-compacted blocks. The main goal of the bentonite block and ring manufacturing process and the subsequent buffer installation process of the KBS-3V concept is, once equilibrated after deployment, to achieve a predefined final bulk density in a water-saturated buffer.

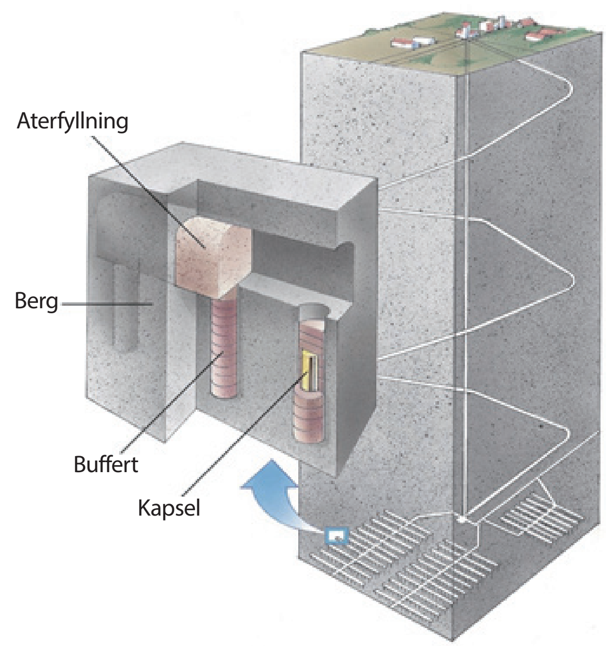


Fig. 1. The Swedish disposal concept [8].

The limit of the surface temperature of the waste disposal container of the Swedish KBS-3V concept is 100°C [8]. This limit was specified for the following reasons.

- To avoid having to consider the boiling effect of water on emplaced waste disposal containers.
- To prevent excessive drying of the bentonite buffer material.
- To limit the copper canister corrosion rate (copper corrosion rate approximately doubles for every 10°C increase in temperature).
- To prevent the build-up of potentially corrosion-accelerating salts on the surface of the canister.
- To prevent illitization of the bentonite buffer material.

2.1.2 Finland

In the case of Finland, Posiva, an implementation

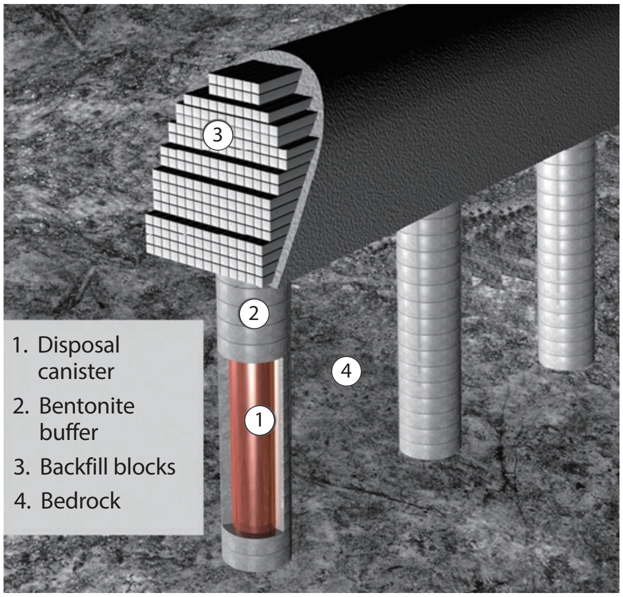


Fig. 2. Finnish disposal concept [12].

agency for the disposal of spent nuclear fuel, considers KBS-3V [9] as a reference concept (Fig. 2) and KBS-3H [10] as an alternative concept for the disposal of spent fuel in the crystalline rock of the Olkiluoto site. The thermal constraint in this concept is keeping the temperature of the bentonite buffer below 100°C. The temperature limit has existed since the early 1980s after Pusch [11] reviewed the smectite stability of bentonite buffers. Posiva describes the reasons for the thermal management temperature limit at the repository as follows.

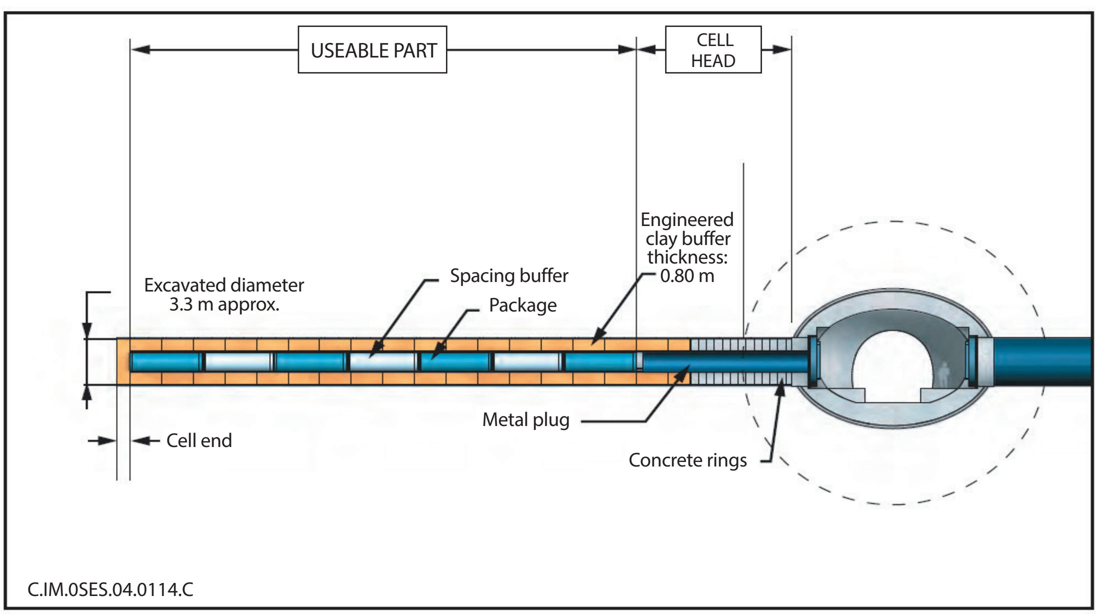
- To avoid the boiling of water and enrichment of solids near the surface of the canister during saturation, which might cause cementation of the bentonite and enhance corrosion of the copper canister.
- To avoid evaporation of water in bentonite.
- To avoid chemical changes in the buffer material.

Posiva therefore currently includes the 100°C temperature limit in the development of the disposal system through the requirements management system.

2.2 Sedimentary Rocks

2.2.1 France

The French concept for the disposal of spent fuel in a clay host rock (Callovo-Oxfordian Clay) considered the



Longitudinal cross section of a CU cell

Fig. 3. French concept for spent fuel disposal [14].

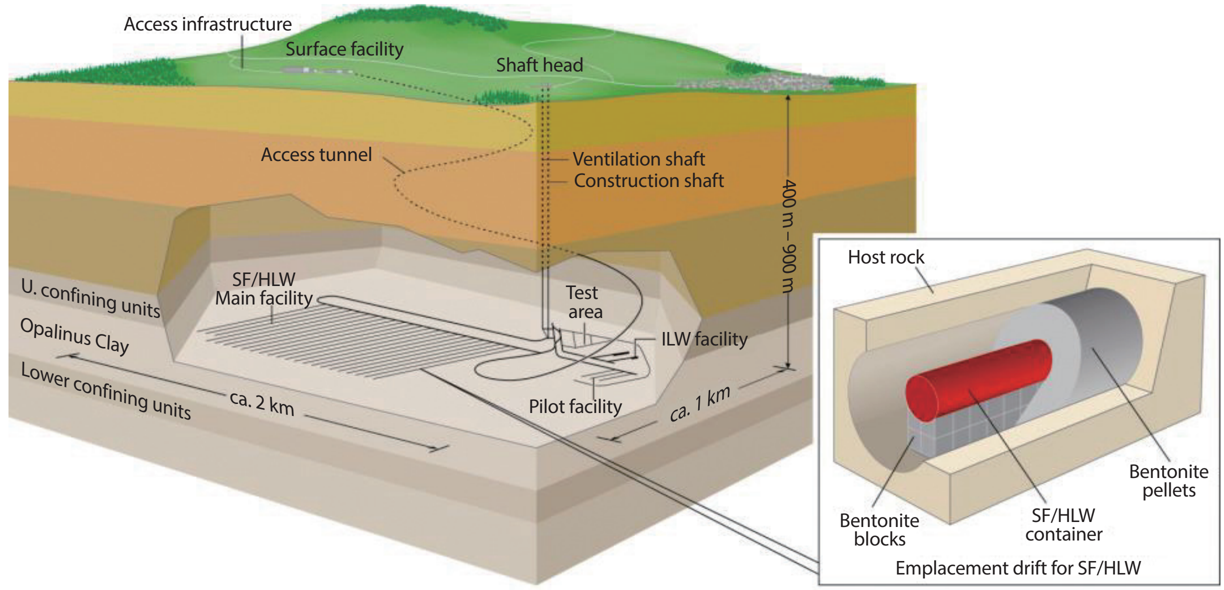


Fig. 4. Swiss concept of repository for SNF/HLW [17].

concept of placing a steel container loaded with spent fuel waste in a horizontal tunnel of the clay rock and enclosing it with bentonite between steel liners in consideration of the heat emitted by the spent fuel (Fig. 3). Bentonite buffers are not installed around the disposal containers for the high-level radioactive solidified waste from the spent nuclear fuel reprocessing process [13].

The thermal criterion for the disposal concept was initially defined with a maximum temperature of 100℃ for the clay components of the disposal system (clay host rock and swollen clay). Taking into account other uncertainties (e.g. thermal conductivity of the clay host rock), the maximum temperature of the steel liner was set at 90℃ [14].

2.2.2 Swiss

The Swiss reference concept developed in the Opalinus Clay project envisions placing a buffer material around the waste containers in a combination of compacted bentonite blocks and pellets with powder. The reference bentonite is MX-80, and the Swiss concept is illustrated in Fig. 4 [15].

Safety-related properties of buffer materials are their swelling pressure and hydraulic conductivity for a period of time after resaturation is completed. In addition to the maximum repository temperature, other factors can influence these characteristics. In the Swiss disposal concept, the main role of the buffer material in terms of long-term safety is to prevent horizontal flow of groundwater along the disposal tunnel and with bentonite backfilled tunnels in tight clay stone (Opalinus Clay), temperatures higher than 100℃ is tolerated for the bentonite, because the host rock is a primary barrier [16]. Therefore, NAGRA of Switzerland set the design basis temperature at the center of the bentonite blocks as 125℃ based on the research result that the decrease in the swelling capability of bentonite at 125℃ is negligible [15].

2.3 Other Host Rocks

2.3.1 Germany

The reference concept of a geological repository for the disposal of high-level waste and spent nuclear fuel in Germany (Fig. 5) considers salt rock as the host rock [18]. In this concept, if the disposal canister is degraded or damaged, isolation, containment, and delay functions are provided by

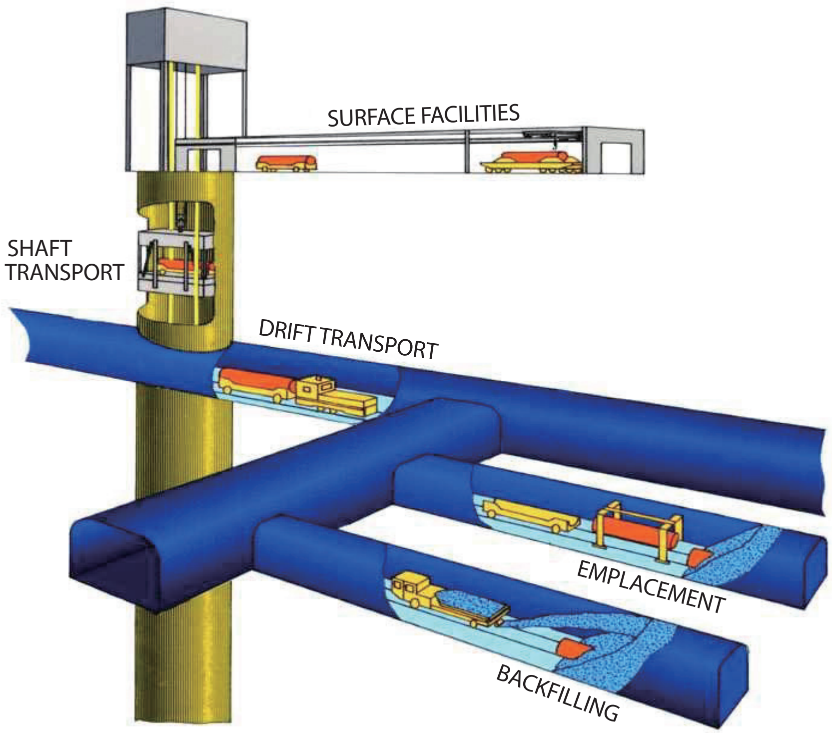


Fig. 5. German disposal concept [19].

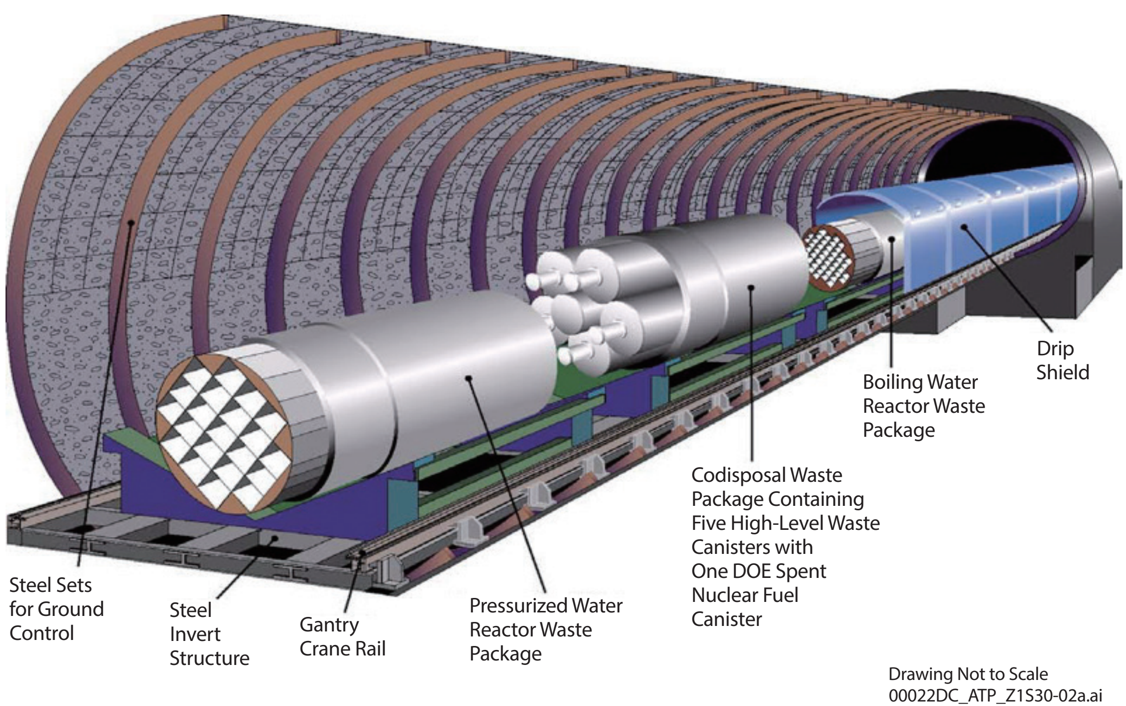


Fig. 6. US YMP disposal concept [21].

the salt rock, which is a geological barrier. A thermal limitation of the German salt rock repository is that the temperature of the salt rock is maintained below 200°C [19]. This temperature limit is based on the conclusion that decomposition crystals were found only where the rock was heated to 230°C, and the decomposition was insignificant at 200°C, based on the results of a temperature analysis according to the results of heater tests performed at the Asse salt mine in the early 1980s.

2.3.2 The United States

The United States submitted an application for a construction permit to dispose of high-level waste and spent nuclear fuel in Mt. Yucca, Nevada, in 2008, but the Obama administration suspended the project. The concept of disposal was that waste disposal containers would be placed in a disposal tunnel in an unsaturated area above the groundwater level and would not be backfilled (Fig. 6). The temperature constraint on the disposal facility was defined as follows [20]:

The waste package surface temperature shall be kept below 572°F (300°C) for the first 500 years and below 392°F (200°C) for the next 9,500 years to eliminate post-closure issues (i.e. phase stability).

High temperatures are regarded as a beneficial feature of the disposal concept as they contribute to sustaining dry conditions in the near-field. In addition, the temperature of the disposal tunnel wall and the intermediate position between the disposal tunnels were limited to 200°C and 96°C, respectively, to keep temperatures below the boiling point of water in the unsaturated area [21].

2.4 Summary of Repository Thermal Management Concepts

The current status of thermal management by host rock in each country described above is summarized in Table 1 below.

Table 1. Thermal management concepts associated with nations and host rocks

Host Rock	Nation	Thermal Conductivity ($W \cdot mK^{-1}$)	Disposal Tunnel Spacing	Deposition Hole Spacing	Thermal Limits
Granite	Finland [22]	2.3–3.2	25 m	11 m	Bentonite buffer < 100°C
	Sweden [23]	3.4–4 2.45–2.9	40 m	6 m 7.2 m	Bentonite buffer < 100°C
Clay	France [24]	1.9–2.7 parallel, 1.3–1.9 perpendicular	8.5–13.5 m	2.5–4 m	Argillaceous host rock < 100°C
	Swiss [25]	1.8	40 m	3 m	Clay buffer < 125°C
Salt	Germany [26]				Salt rock < 200°C
	USA	3.09 @ 100°C, 3.37 @ 29°C [27]	HLW: 120 ft. (36.6 m) SNF: 170 ft. (51.8 m)	SNF: 28−85 ft (8.5−25.9 m) Defense HLW glass: 10 ft. (3.05 m)	Fuel cladding < 375°C Salt rock < 250°C HLW glass < 500°C
Tuff		0.99–2.07 [28]			Between tunnel temperature < 96°C Tunnel Wall temperature < 200°C

3. Thermal Management Methods for Deep Geological Repository

The standards related to decay heat generated from high-level waste encapsulated in a disposal container of a deep geological disposal facility according to the Public Notice on General Standards for Deep Geological Disposal Facilities for High-Level Radioactive Waste [29] are described below.

Article 13 Paragraph 2 related to engineered barriers stipulates that, in connection with other design features and characteristics of natural barriers, they shall be able to withstand decay heat and ambient pressure caused by radioactive waste during operation and after closure of the disposal facility. In addition, article 18 relating to the radioactive waste form characteristics defines that in the disposal environment in conjunction with engineered barriers against decay heat and pressure, they shall maintain a physically and chemically stable solid form and function for a long time.

In this study, a method was described to satisfy the standards related to decay heat stipulated in this public notice on deep geological disposal facilities.

3.1 Extension of the Spent Nuclear Fuel Cooling Time

One of the ways to manage spent nuclear fuel, which is a high-level waste with high temperature characteristics, is to store it for a longer period before disposal so that it can decay and that the heat output will be lowered to an acceptable level, and thereby as much spent nuclear fuel as possible can be loaded in the disposal container. Cooling by storage of spent nuclear fuel or high-level waste as part of the preparation and operation of the repository can significantly reduce the decay heat released from the waste during the operation of the repository and after permanent closure.

As a thermal management strategy, the effectiveness of cooling, also called decay storage, depends on the type of waste and is usually limited to short-lived radionuclides (e.g., a half-life of less than 100 years).

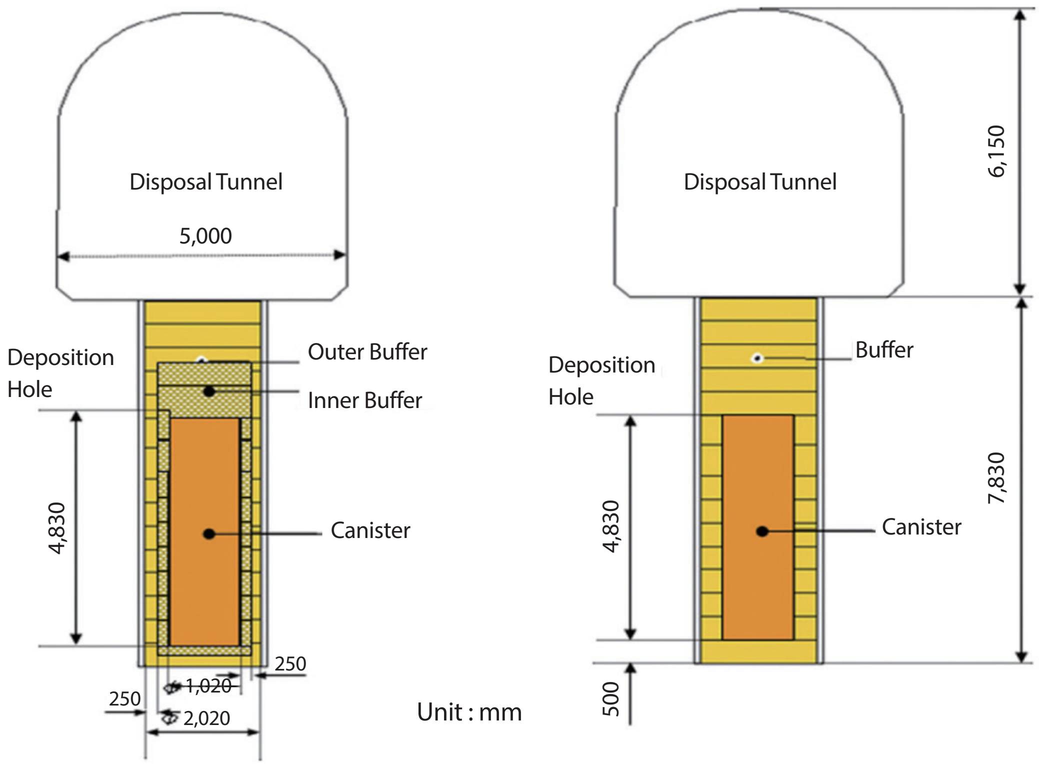
In the case of such an extension of the cooling time, related cost factors such as storage facilities and licenses increase according to the extension of the storage period of spent nuclear fuel, and an integrity evaluation of the spent nuclear fuel is required. Therefore it is necessary to establish the optimal storage period. In addition, as the size of the disposal container increases due to the increase of spent nuclear fuel loaded in the disposal container, it is necessary to analyze the related handling and economic characteristics.

3.2 Changes in the Characteristics of Disposal Containers

In order to improve the disposal efficiency in terms of disposal area, the amount of waste per unit element must be increased thermally and structurally compared to the reference disposal system. The most obvious solution to reduce the excessive thermal load in a disposal container is to limit the amount of spent nuclear fuel that can be accommodated in a disposal container.

When a nuclear fuel cycle produces wastes of different characteristics with respect to heat output due to inventory, cooling time, or other characteristics, they can be disposed of together within individual packages to serve as a thermal management tool. Thus, high burn-up spent nuclear fuel assemblies can be combined with low burn-up assemblies or other cooler waste types. Therefore, mixing according to the thermal properties of the waste in the disposal container can have an important effect on thermal management.

In addition, as a way to change the design of the disposal container, using a low thermal conductivity material for the disposal container in order to lower the temperature of the surface of the disposal container and a bentonite buffer material may be considered. In this case, the corrosion rate of the disposal container may increase at high temperatures, and the potential effect in this case can be offset simply by increasing the thickness of the container, but other factors such as weight should also be evaluated.



(a) a double-layered buffer material

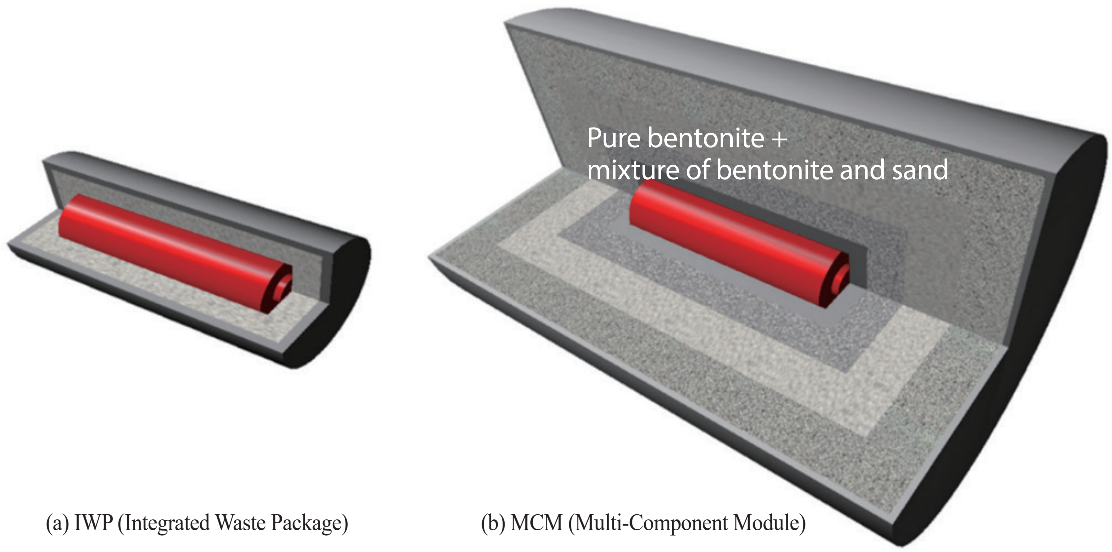
(b) a single-layered buffer material

Fig. 7. Concept of a double-layered buffer [31].

3.3 Changes in the Structure or Material Properties of Engineered Barriers

In order to meet the design requirements for the maximum temperature of the buffer bentonite, which is a component of the engineered barrier in the disposal system, the structure of the buffer material surrounding the disposal container could be modified. Regarding a method to increase heat transfer from high-level waste in the disposal container to bentonite by installing a material with high thermal conductivity around the disposal container, sand and graphite are being considered as materials that can be used. Studies have been conducted in Japan to use a mixture of bentonite and sand as a buffer material instead of installing pure bentonite for the entire buffer material. Japan employed compressed pure Kunigel bentonite as a buffer material in the initial deep disposal concept, but in the H-12 deep disposal concept [30], the Kunigel bentonite (70%)-sand (30%) mixture was compressed and used as a buffer material to improve the efficiency of heat transfer and the mechanical performance. In this case, since the hydraulic conductivity of the mixed buffer material may be increased and the characteristics of the buffer material may be deformed, a detailed evaluation of the related characteristics is required in terms of long-term safety.

In addition, for efficient thermal management of the repository, the initial design of a single-layered buffer material could be changed to a double-layered buffer material to accelerate heat diffusion from the waste, as shown in Fig. 7. This concept improves the thermal conductivity of the buffer material by mixing bentonite, sand, and graphite in the buffer layer close to the disposal container. It was analyzed that the temperature of the surface of the disposal container could be lowered by about 7°C relative to the initial



(a) IWP (Integrated Waste Package) (b) MCM (Multi-Component Module)

Fig. 8. PEM Concept : no scale [35].

concept (98.8 → 91.6°C [31]).

The use of graphite was also considered in the Belgian repository concept to improve the thermal conductivity of the buffer material. When using a mixture of bentonite and sand, adding 5wt% of graphite increased the thermal conductivity of the buffer material from 1.5 to 4 $W \cdot mK^{-1}$ [32]. However, the use of additives can increase the hydraulic conductivity of the buffer material, and the high heat causes the buffer material near the disposal container to dry, which can decrease the thermal conductivity. It is therefore necessary to carefully consider whether the use of additives meets other performance requirements of the buffer material from the viewpoint of disposal safety. Additionally, factors affecting the long-term stability of the material (such as colloid formation) must first be identified before graphite can be used as part of a barrier material.

The use of dried bentonite packed in super containers can manage the potential impact of hot waste on bentonite properties. Because the kinetics of smectite-illite conversion depend on the availability of potassium ions in the groundwater [33], keeping bentonite dry can reduce the possibility of illitization. As a similar disposal concept, PEMs (Prefabricated EBS Modules) involve prefabricating a steel container with waste, buffer materials, and engineered barrier materials such as sand (Fig. 8) [34]. However, although the temperature at the surface of the PEM will be lowered, heat is not well discharged to the surroundings, and the temperature of the high-level waste itself increases. This may deteriorate the integrity of the PEM, and thus a thorough safety evaluation should be preceded.

3.4 Change in Design Limit of Temperature

Since the maximum temperature at the interface between the disposal container and the buffer material limits the amount of waste that can be disposed of in the unit area of the disposal site to determine the disposition efficiency, allowing a higher temperature limit than 100°C directly leads to a reduction in the disposal area. Characterization of buffer materials under high temperature conditions over 100°C is a relatively new and challenging area of study, and many countries are actively conducting research to verify the performance of their engineered barrier, especially bentonite, at high temperature. Interest in this topic has increased in some countries and institutions, and a project has recently been carried out in GTS, Switzerland, after an international joint research planning meeting called HotBENT (High Temperature Bentonite Project) [36]. The main goal of HotBENT is to improve the understanding of the behavior of buffers and near field bedrock under high-temperature environmental conditions up to 200℃ (surface temperature of disposal containers) and secure databases through related experiments. In Korea, the Korea Atomic Energy Research Institute is conducting laboratory experiments and running chemical models under high temperature conditions (up to 150℃) as part of its nuclear technology development project to identify the high temperature properties of buffers. This will serve as an opportunity to secure safety cases for high-temperature buffers and can contribute to improving the thermal design criteria applied to deep geological repositories.

3.5 Changes in the Disposal Methods of Spent Nuclear Fuel

One option for managing high-temperature waste is to store high-level waste for a longer period of time to lower the heat dissipation value of the container for packaging the high-level waste below the thermal limit. Hicks and Crawford [37] reported that the amount of heat emitted from the surface of the disposal container could be reduced to an acceptable level through about 40–50 years of storage. Thus, in order to meet the thermal limit, securing an additional intermediate storage period for spent nuclear fuel may be an alternative. In addition to the aforementioned measures, Japan’s Nuclear Waste Management Organization (NUMO) is developing a variety of new disposal concepts to meet thermal limitations in a limited disposal area [38]. Among them, the concept of CARE (CAvern REtrievable) puts high-level waste in a multipurpose container, store it in a cave at least 300 m underground rock to reduce heat, and then close the disposal cave (Fig. 9) [39, 40].

The cave remains open for about 300 years, and active management is conducted under the institutional system during this period. The CARE concept was developed as part of the Repository Design Option Study, and it has the advantage of enabling a more precise repository layout than the Japanese reference disposal concept [30, 38]. In addition to the cooling period for the first 50 years, through the introduction of the CARE concept for 300 years, the thermal load would be reduced by more than 20-fold [30].

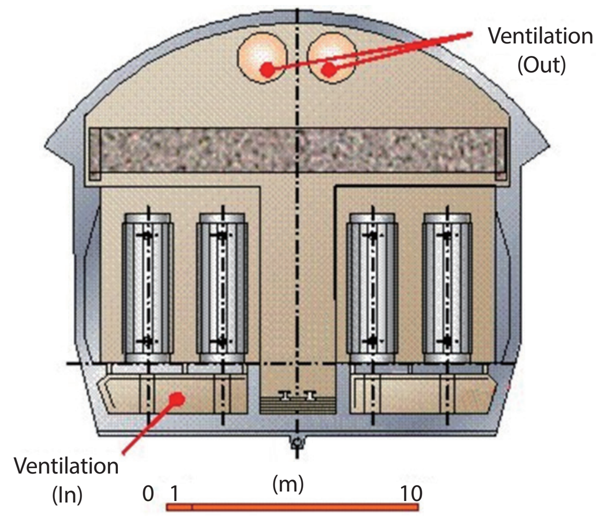


Fig. 9. CARE concept [40].

4. Analyses of the Efficiency of the Thermal Management Methods

Among various measures for efficient thermal management related to the disposal site, engineered barrier, and waste characteristics, the efficiency of thermal management methods according to the cooling time of spent nuclear fuel, thermal conductivity of a bentonite buffer material, and dimensions of the disposal container were analyzed in this study.

4.1 Cooling Time of Spent Nuclear Fuel

Various studies are being conducted in relation to the cooling time of spent nuclear fuel or high-level waste and the arrangement of underground disposal areas for the

Table 2. Characteristics of high burn-up spent nuclear fuel [42]

Type	Reference high burn-up spent nuclear fuel		
	Enrichment	Burn-up	Cooling time
PWR spent nuclear fuel	4.5wt%	55 GWd·tU^{-1}	40 years
CANDU spent nuclear fuel	0.711wt%	8.5 GWd·tU^{-1}	30 years

design of a deep geological repository. If the cooling period is extended before the spent nuclear fuel is disposed of, radioactive decay can be further increased, and thus the radiotoxicity and decay heat value are reduced, thereby reducing the area required for the repository.

4.1.1 Thermal Characteristics of Reference Spent Nuclear Fuels

The reference spent nuclear fuel applied to the concept of the spent nuclear fuel disposal system developed in Korea reflects the characteristics (Table 2) of high burn-up spent nuclear fuel generated by nuclear power plants in Korea [41]. The reference spent nuclear fuel for the conceptualization of the deep geological disposal system emits high heat due to the continuous decay reaction of the radioactive nuclide remaining inside. Fig. 10 below shows the decay heat history over time of the reference spent nuclear fuel (see Table 2) based on high burn-up, and the efficiency of these high burn-up nuclear fuels was analyzed by calculating the temperature in the disposal site related to the cooling time of spent nuclear fuels.

4.1.2 Preliminary Efficiency Analysis According to the Cooling Time

The spent nuclear fuel released after generating electricity from a nuclear power plant emits a high level of radioactivity and high heat and, as shown in Fig. 10, the decay heat will be attenuated over time. Table 3 shows the time required to reduce 100 W of decay heat based on the 40-year cooling time of four assemblies of PWR spent nuclear fuel per disposal container set in the reference disposal system and the amount of heat reduction in each 10 years' cooling

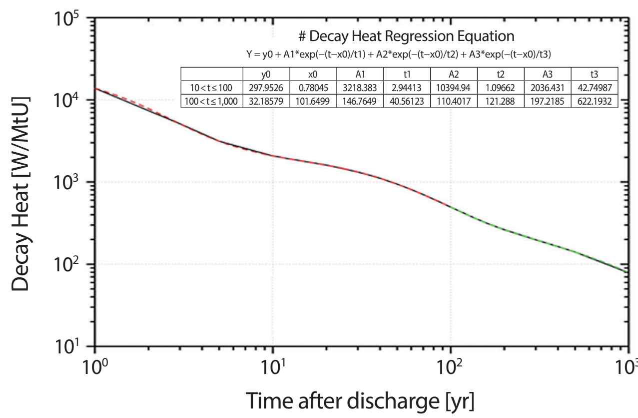


Fig. 10. The history of decay heat over time of PWR spent nuclear fuel [42].

of four assemblies of spent nuclear fuel in the disposal container. As shown in the table, the time required to reduce the amount of heat shows that the lower the amount of decay heat is, the longer the period of time required for the reduction of a certain amount (100 W). In addition, in the case of cooling time, the amount of heat reduced per disposal container decreases as the cooling period becomes longer. Therefore, it is necessary to derive an appropriate cooling time for disposal in consideration of changes in cost and waste characteristics according to the storage period.

Fig. 11 shows the temperature history and maximum temperature in the repository according to the cooling period. As shown in the figure, in the case of a 40-year cooling period, which is the reference concept, the maximum temperature is lowered from 97.5°C to 87°C, 78.7°C, 72.3°C, and 67.4°C, as the cooling time is extended to 50, 60, 70, and 80 years, respectively. Therefore, it is necessary to increase the amount of thermal load by adding spent nuclear fuel per disposal container to meet the temperature limitation requirement of 100°C according to the cooling period,

Table 3. Decrease or change of heat according to cooling time from spent fuel per disposal container

Time required to reduce 100 W per disposal container			Amount of thermal load reduced during 10 years' cooling per disposal container		
Decay heat (W)	Cooling time (yr.)	Required year for 100 W reduction (yr.)	Cooling time (yr.)	Decay heat (W)	Reduced decay heat (10 years, W)
1,915	40	-	40	1,915	-
1,900	40.5	-	50	1,623	292
1,800	43.7	3.2	60	1,391	232
1,700	47.15	3.5	70	1,208	183
1,600	50.9	3.8	80	1,063	145
1,500	55	4.1	90	948	115
1,400	59.6	4.6	100	857	91
1,300	64.7	5.1	110	774	83
1,200	70.5	5.8	120	709	65
1,100	77.2	6.7	130	656	53
1,000	85.2	8.0	140	611	45
900	95	9.8	150	574	37
800	106.5	11.5	160	542	32
700	121.7	15.2	170	515	27
600	142.9	21.2	180	491	24
500	176	33.1	190	470	21
400	237	61.0	200	452	18
300	365	128.0	-	-	-
200	640	275.0	-	-	-
100	1,360	720.0	-	-	-

or to maximize the disposal efficiency in terms of the disposal area by reducing the disposition tunnel/disposal space.

4.2 Buffer Thermal Conductivity

4.2.1 Concept of Buffer for the Reference Disposal System

In a deep geological repository located in the deep crystalline bedrock, it is essential to install a buffer material to prevent the inflow of groundwater through cracks in the rock and corrosion of the disposal container and the outflow of radionuclides [43, 44]. In addition, the buffer material serves to protect the disposal container from external stress and to dissipate the decay heat generated from the waste to the host rock. In order to meet the design requirements of the disposal site according to the decay heat, it is necessary to increase the thermal conductivity of the buffer surrounding the disposal container to efficiently dissipate the diffused decay heat. So, the thermal conductivity of the buffer is one of the important factors.

In order to establish and design the concept of a buffer in the reference disposal concept, not only qualitative

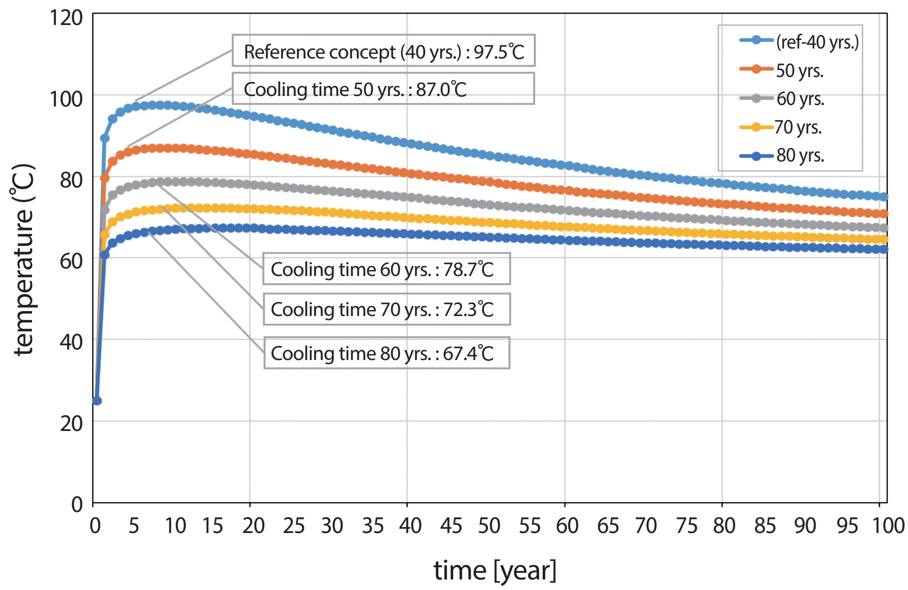


Fig. 11. The temperature history of the disposal site according to the cooling time (40 years−80 years) of spent nuclear fuel in the disposal container [44].

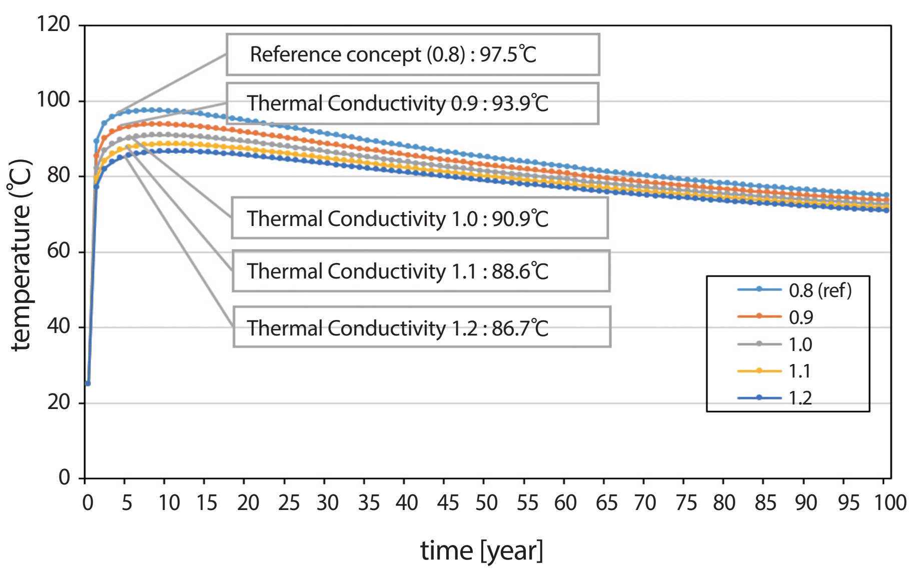


Fig. 12. The temperature history of the repository according to the thermal conductivity of the buffer material (the thermal conductivity of the buffer material is 0.8−1.2).

requirements but also quantitative performance standards are required. In general, the quantitative performance standards of buffer materials are determined by analyzing technical review items for buffer requirements based on the results of related studies. The main technical items are hydraulic conductivity, nuclide adsorption, swelling ability and swelling pressure, thermal conductivity, long-term integrity, organic matter content, mechanical properties, etc. Among these items, the criteria related to thermal stability are as described in Table 4 below.

Table 4. Criteria for thermal stability of buffer materials

Characteristics	Ca-Bentonite (KJ bentonite)
Dry density ($Mg \cdot m^{-3}$)	1.6
Initial water content (%)*	13
Thermal conductivity ($W \cdot mK^{-1}$)	0.8
Hydraulic conductivity ($m \cdot s^{-1}$)	1×10^{-12}
Diffusion coefficient ($m^2 \cdot s^{-1}$)	1×10^{-11} (for cation)
	1×10^{-9} (for anion)
Swelling pressure (MPa)	5−7

* The ratio of the weight of water to the weight of the bentonite with water

4.2.2 Analysis of Disposal Efficiency According to Thermal Conductivity of Buffer

In the deep geological disposal concept of crystalline bedrock, the bentonite block, which prevents the flow of leaked radionuclides from the breakage of the disposal container and groundwater flowing through cracks in the bedrock, and serves as a buffer from external stress, has a significant impact on the repository temperature depending on the thermal conductivity.

Fig. 12 shows the temperature history and maximum temperature of the repository according to the thermal conductivity of the bentonite buffer material. In the case of a buffer material with thermal conductivity of 0.8 $W \cdot mK^{-1}$, the reference concept, the maximum temperature is 97.5℃, and as the thermal conductivity increases to 0.9, 1.0, 1.1, and 1.2 $W \cdot mK^{-1}$, the maximum temperature of the repository is lowered to 93.9, 90.9, 88.6, and 86.7℃, respectively, as shown in Fig. 12. Therefore, improving the thermal conductivity of the buffer material in a direction that does not degrade the isolation and delay performance of the buffer material, e.g., the disposal safety in the disposal system, is considered a good way to improve disposal efficiency. In other words, according to the thermal conductivity of the buffer material, it is necessary to maximize the disposal efficiency in terms of the required area by adding spent nuclear fuel per disposal container to meet the 100℃ temperature limit requirement of the repository or by reducing the disposition tunnel/disposal space.

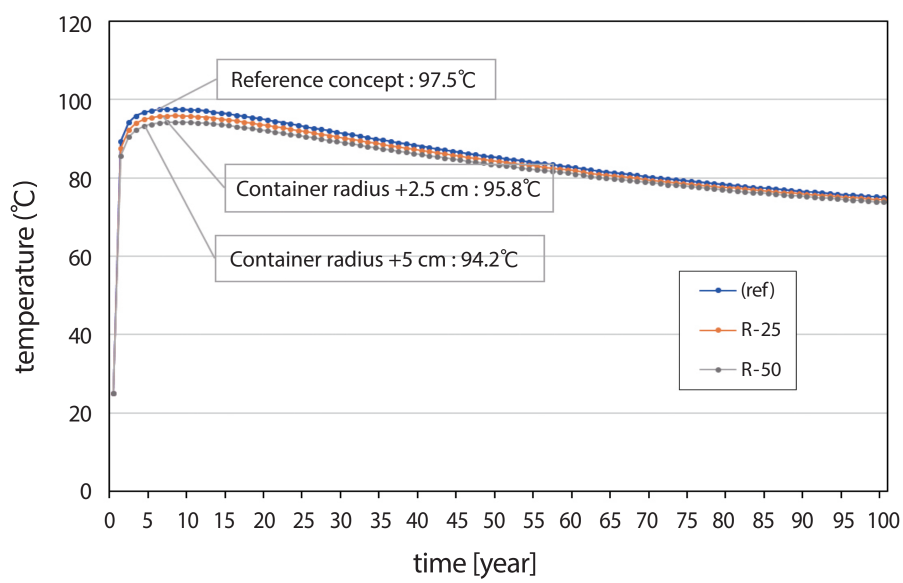


Fig. 13. Repository temperature history according to the size of the container (increased in container radius by 2.5 cm and by 5 cm).

4.3 Disposal Container Size

In a deep geological repository, the decay heat of the spent nuclear fuel loaded in the disposal container diffuses to the disposal container, the buffer material, and the host bedrock. This is mainly accomplished by conduction rather than convection or radiation when the repository is closed [1]. Therefore, the heat transfer energy according to conduction can be expressed by Fourier's law of heat conduction, as shown in the following equation.

$$Q_n = -\mathrm{kA}\frac{\partial \mathrm{T}}{\partial \mathrm{n}} \tag{1}$$

where Q is the heat flux, k is the thermal conductivity, A is the heat transfer area, and $\frac{\partial \mathrm{T}}{\partial \mathrm{n}}$ is the temperature gradient in each direction [45].

As shown in the equation, the heat transfer rate is proportional to the thermal conductivity and the area through which heat is transmitted, and inversely proportional to the thickness of the area through which the heat is transmitted. Therefore, if the heat transfer surface area of the disposal container is increased, it is judged that the heat transfer rate will increase and efficient thermal management of the repository will be possible.

The heat management efficiency in the case of expanding the heat transfer area of the disposal container was thus analyzed. Fig. 13 shows the temperature history at the repository when the radius of the disposal container is increased by 2.5 cm and 5 cm.

As shown in Fig. 13, in the case of the reference concept, the maximum temperature of the repository is 97.5°C. If the radius of the disposal container is increased by 2.5 and 5 cm, the maximum temperature of the repository is lowered to 95.8°C and 94.2°C, respectively. It is therefore judged that the disposal efficiency in terms of disposal area can be improved to some extent in terms of the required area.

Table 5. Results of the disposal efficiency analyses

Thermal management method	Value of factors	Maximum temperature (℃)	Remarks (℃)
Cooling time of spent nuclear fuel (years)	40 (Reference concept)	97.5	
	50	87.0	↓ 10.5
	60	78.7	↓ 8.3
	70	72.3	↓ 6.4
	80	67.4	↓ 4.9
Buffer thermal conductivity ($\mathrm{W{\cdot}mK^{-1}}$)	0.8 (Reference concept)	97.5	
	0.9	93.9	↓ 3.6
	1.0	90.9	↓ 3.0
	1.1	88.6	↓ 2.3
	1.2	86.7	↓ 1.9
Disposal container size (radius, cm)	radius (Reference concept)	97.5	
	radus+2.5	95.8	↓ 1.7
	radus+5	94.2	↓ 1.6

Therefore, it is possible maximize the disposal efficiency in terms of the disposal area by adding spent nuclear fuel per disposal container or by reducing the disposition tunnel/disposal space to meet the temperature limitation requirement of 100℃.

However, the improvement in efficiency compared to the disposal efficiency according to the cooling time of spent nuclear fuel or the thermal conductivity of buffer materials is relatively small, and when the size of the disposal container is increased, structural supplementation and integrity of the container are also required.

4.4 Results of Disposal Efficiency Analyses With Thermal Management Methods

As described above section, among various measures for efficient thermal management, in this study the efficiency of thermal management methods according to the cooling time of spent nuclear fuel, thermal conductivity of a bentonite buffer material, and dimensions of the disposal container were analyzed. Table 5 showed the results of the disposal efficiency analyses in terms of disposal area.

5. Conclusions and Future Plans

Currently, the most widely accepted disposal method for safe isolation of radioactive waste is to dispose of it in a deep geological repository designed and constructed with multiple barriers composed of engineered and natural barriers so that it can be completely isolated for very long time in a stable and deep geological environment. The safe disposal of high-level radioactive waste depends on the performance of engineered barriers and natural barriers, which are components of the entire disposal system. A key consideration for both the near field around the engineered barriers and natural barrier are the heat generated from the waste due to the large amount of fission products in the disposal container.

In this paper, considering the situation in Korea in which a relatively large amount of spent nuclear fuel has to be disposed of in a small area of land due to the use of nuclear energy that is an essential component of energy production because of the lack of resource, thermal management technology in the repository to improve disposal efficiency in terms of disposal area is presented and analyzed. The findings of this study are summarized below.

- A conceptual analysis was performed on the effect of heat on the geological repository system for high-level radioactive waste and thermal management methods. The analysis was performed mainly according to the thermal effect on the behavior of engineered barriers and natural barriers.
- By reviewing various thermal management methods, thermal management concepts for efficient geological disposal were analyzed. In particular, we reviewed the current status of research on thermal load management in some countries conducting disposal research on thermal limitations in the repository.
- From the results of the analyses on various thermal management methods, several thermal analyses of the repository for factors such as spent nuclear fuel cooling time, buffer bentonite thermal conductivity, and disposal container size was carried out to develop an efficient thermal management method applicable in Korea. Based on the results, efficiency in terms of disposal area was conceptually analyzed and additional research that should be performed for each factor was described.

The results of this study can be used as data to establish an efficient concept of a deep geological repository for spent nuclear fuel, and they are expected to contribute to securing flexibility because of a long-term disposal project in the development of disposal technology. In the future, in order to secure applicability and validity of the results of this study in Korea, a detailed analysis using data obtained from actual disposal sites is required.

Acknowledgements

This work was supported by the Ministry of Science and ICT within the framework of the national long-term nuclear R&D program (NRF-2021M23A2041312).

REFERENCES

[1] International Atomic Energy Agency, Effects of Heat From High-level Waste on Performance of Deep Geological Repository Components, 14-26, IAEA-TEC-DOC-319 (1984).

[2] T.W. Hicks, M.J. White, and P.J. Hooker. Role of Bentonite in Determination of Thermal Limits on Geological Disposal Facility Design, Galson Sciences Ltd. Report, 12-26, Report 0883-1, version 2 (2009).

[3] J.S. Kim, W.J. Cho, S. Park, G.Y. Kim, and M.H. Baik, "A Review on the Design Requirement of Temperature in High-level Nuclear Waste Disposal System: Based on Bentonite Buffer", J. Korean Tunn. Undergr. Sp. Assoc., 21(5), 587-609 (2019).

[4] S. Vomvoris, J. Birkholzer, L. Zheng, I. Gaus, and I. Blechschmidt, "THMC Behavior of Clay-based Barriers Under High Temperature - From Laboratory to URL Scale", Proc. of 15th International High-Level Radioactive Waste Management Conference, 678-687, Charleston, NC, USA (2015).

[5] K. Ikonen. Thermal Analyses of Spent Nuclear Fuel Repository, Posiva Oy Report, Posiva 2003-04 (2003).

[6] A. Lempinen. THM Model Parameters for Compacted Bentonite, Posiva Oy Working Report, 2006-79 (2006).

[7] H. Harald and B. Faelth. Thermal Dimensioning of the Deep Repository. Influence of Canister Spacing, Canister Power, Rock Thermal Properties and Nearfield Design on the Maximum Canister Surface Temperature, Svensk Kärnbränslehantering AB Technical Report, SKB-TR-03-09 (2003).

[8] R. Christiansson, Design of the Radioactive Waste Repository in Sweden Swedish Nuclear and Fuel Waste Management Co. (2011).

[9] K. Rasilainen. Localisation of the SR 97 Process Report for Posiva's Spent Fuel Repository at Olkiluoto, Posiva Technical Report, Posiva 04-05 (2004).

[10] P. Smith, L. Johnson, M. Snellman, B. Pastina, and P. Gribi. Safety Assessment for a KBS-3H Spent Nuclear Fuel Repository at Olkiluoto Evolution Report, Posiva Report, POSIVA 2007-8 (2007).

[11] L. Ahonen, P. Korkeakoski, M. Tiljander, H. Kivikoski, and R. Laaksonen. Quality Assurance of the Bentonite Material, Posiva Oy Working Report, 2008-33 (2008).

[12] J. Palomäki and L. Ristimäki. Facility Description 2012 Summary Report of the Encapsulation Plant and Disposal Facility Designs, Posiva Oy Working Report, 2012-66 (2013).

[13] Agence nationale pour la gestation des déchets radioactifs. Dossier 2005 Argile SYNTHESIS Evaluation of the Feasibility of a Geological Repository in an Argillaceous Formation, Andra Report Series (2005).

[14] Agence nationale pour la gestation des déchets radioactifs. Dossier 2005 Andra Research on the Geological Disposal of High-level Long-lived Radioactive Waste. Results and Perspective, Andra Report (2005).

[15] National Cooperative for the Disposal of Radioactive Waste. Demonstration of Disposal Feasibility for Spent Fuel, Vitrified High-level Waste and Long-lived Intermediate-level Waste (Entsorgungsnachweis), Nagra Technical Report, 02-05 (2002).

[16] O.X. Leupin, P. Smith, P. Marschall, L. Johnson, D. Savage, V. Cloet, J. Schneider, and R. Senger. High level Waste Repository induced Effects, Nagra Technical Report, 14-13 (2016).

[17] I. Blechschmidt, Radioactive Waste Management in Switzerland Short Update, KAERI-NAGRA Technical Meeting, 9-16, Daejeon (2017).

[18] W. Bollingerfehr, W. Filbert, and J. Wehrmann, "Emplacement Technology for the Direct Disposal of Spent Fuel Into Deep Vertical Boreholes", International Conference Underground Disposal Unit Design & Emplacement Processes for a Deep Geological Repository, June 16-18, 2008, Prague.

[19] R. Graf and W. Filbert, "Disposal of Spent Fuel From German Nuclear Power Plants – Paper Work or Technology", Presentation at the Topseal Conference, September 17-20, 2006, Finland.

[20] Sandia National Laboratories, Total System Performance Assessment Data Input Package for Requirements Analysis for DOE SNF/HLW and Navy SNF Waste Package Overpack Physical Attributes Basis for Performance Assessment, TDR-TDIF_ES-000009 REC.00, Las Vagas, Nevada (2007).

[21] U.S. Department of Energy Office of Civilian Radioactive Waste Management. Yucca Mountain Science and Engineering Report, U.S. Department of Energy Report, DOE/RW-0539-1 (2002).

[22] Posiva Oy. Interim Summary Report of the Safety Case 2009, Posiva Oy Report, 10-02 (2010).

[23] Svensk Kärnbränslehantering AB. Long-term Safety for KBS-3 Repositories at Forsmark and Laxemar – a First Evaluation, SKB Technical Report, TR-06-09 (2006).

[24] Agence nationale pour la gestation des déchets radioactifs. Dossier 2005 Argile – Architecture and Management of a Geological Disposal System, Andra Report (2005).

[25] L.H. Johnson and F. King. Canister Options for the Disposal of Spent Fuel, Nagra Technical Report , NTB 02-11 (2003).

[26] H.N. Kalia, “Simulated Waste Package Test in Salt”, International High Level Radioactive Waste Management Conference, May 22-26, 1994, Las Vegas.

[27] U.S. Department of Energy. Site Characterization Plan Conceptual Design Report for a High-Level Nuclear Waste Repository in Salt, Horizontal Emplacement Mode: Volume 1, U.S. Department of Energy Report, DOE/CH-46656-14(1) (1987).

[28] U.S. Department of Energy. Yucca Mountain Repository License Application for Construction Authorization, U.S. Department of Energy Report, DOE/RW-0573, Rev.0, Section 1.3.1.2.5 (2008).

[29] Nuclear Safety & Security Commission, General Standards for Deep Geological Disposal Facility of High-level Radioactive Wastes, Public Notice of NSSC 2017-74 (2017).

[30] Japan Nuclear Cycle Development Institute. H12: Project to Establish the Scientific and Technical Basis for HLW Disposal in Japan. Second Progress Report on Research and Development for the Geological Disposal of HLW in Japan, Supporting Report 2: Repository Design and Engineering Technology, JNC Report, JNC TN1410 2000-003 (2000).

[31] H.J. Choi and J. Choi, “Double-layered Buffer to Enhance the Thermal Performance in a High-level Radioactive Waste Disposal System”, Nucl. Eng. Des., 238(10), 2815-2820 (2008).

[32] ONDRAF/NIRAS. Technical Overview of the SAFIR 2 Report, Safety Assessment and Feasibility Interim Report 2, ONDRAF/NIRAS Report, NIROND 2001-05 E (2001).

[33] P. Wersin, L.H. Johnson, and I.G. McKinley, “Performance of the Bentonite Barrier at Temperatures Beyond 100°C: A Critical Review”, Phys. Chem. Earth, 32(8-14), 780-788 (2007).

[34] H. Kawamura, I.G. McKinley, and F.B. Neall, “Practical and Safe Implementation of Disposal With Prefabricated EBS Modules”, Proc. of International Technical Conference on Practical Aspects of Deep Radioactive Waste Disposal Session 2, No. 8, June 16-18, 2008, Prague.

[35] S. Masuda, H. Kawamura, I. McKinley, F. Neall, and H. Umeki, “Optimising Repository Design for the CARE Concept”, 11th International High-Level Radioactive Waste Management Conference, 507-514, Las Vegas, USA (2006).

[36] L. Zheng, J. Rutqvist, J.T. Birkholzer, and H.H. Liu, “On the Impact of Temperatures up to 200°C in Clay Repositories With Bentonite Engineer Barrier Systems: A Study With Coupled Thermal, Hydrological, Chemical, and Mechanical Modeling”, Eng. Geol., 197, 278-295 (2015).

[37] T.W. Hicks and M.B. Crawford. Co-disposal of HLW and Spent Fuel With ILW and LLW Viability Study: Hydro-Thermo-Mechanical Effects, Galson Sciences

Ltd. Report, 9902-1 (2002).

[38] Nuclear Waste Management Organization of Japan. Development of Repository Concepts for Volunteer Siting Environments, NUMO Technical Report, NUMO-TR-04-03 (2004).

[39] I.G. McKinley, F.B. Neall, P.A. Smith, J.M. West, and H. Kawamura, “Evolution of the Cavern-Extended Storage (CES) Concept for Flexible Management of HLW”, MRS Online Proceedings Library, 807, 695-700 (2003).

[40] S. Masuda, H. Umeki, I. McKinley, and H. Kawamura, “Management With CARE”, Nucl. Eng. Int., 49(604), 26-29 (2004).

[41] D.K. Cho, J.W. Kim, I.Y. Kim, and J.Y. Lee, “Investigation of PWR Spent Fuels for the Design of a Deep Geological Repository”, J. Nucl. Fuel Cycle Waste Technol., 17(3), 339-346 (2019).

[42] I.Y. Kim, H.A. Kim, and H.J. Choi. Evaluation on Thermal Performance and Thermal Dimensioning of Direct Deep Geological Disposal System for High Burn-up Spent Nuclear Fuel, Korea Atomic Energy Research Institute Technical Report, KAERI/TR-5230/2013 (2013).

[43] Svensk Kärnbränslehantering AB, Final Storage of Spent Fuel – KBS-3, Stockholm (1983).

[44] H.J. Choi, J.Y. Lee, and D.K. Cho. Korean Reference HLW Disposal System, Korea Atomic Energy Research Institute Technical Report, KAERI/TR-3563 (2008).

[45] Y.A. Cengel, A.J. Ghajar, and M. Kanoglu, Heat and Mass Transfer: Fundamentals and Applications, 4th ed., McGraw-Hill, New York (2011).