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■ 論 文 ■

Analysis of Key Parameters for Designing the Spent Nuclear Fuel Disposal Container in Korea

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사용후핵연료 처분용기 설계를 위한 주요인자 분석

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Abstract - For the first step to develop a reference disposal container of spent fuel to be used in a deep geological repository, this paper examined safe dimensions of the disposal container on the points of nuclear criticality and radiation safety and mechanical structural safety and provided basic information for dimensioning the container and configuration of the container components, and establishing the favorable and safe disposal conditions.

When the safety factor for stress due to the external loads (hydrostatic and swelling pressure) is taken as 2.0, the safe diameter of the filler material to provide enough container strength under the assumed external loads is found to be 112cm with 13cm spacing between inner baskets in PWR container. Also the thickness of the thinner section between the fuel basket and the surface of the cast insert is determined to be 150 mm. Regarding these dimensions of the container, the PWR fuel container is sketched to accommodate 4 square assemblies or 297 CANDU fuel 297 bundles (33 circle tubes x 9 stacks). However the top and bottom parts need to be checked again through the detail radiation shielding analysis with respects to the emplacement position and handling processes of the disposal container.
Key words : Geological Repository, Disposal Container, Criticality, Radiation shielding, Spent fuel

요약 - 본 연구에서는 심지층처분장에서 사용될 사용후핵연료 처분용기 개발을 위한 첫 시도로서 핵임계 및 방사선 안전성과 열역학적 구조안정성 관점에서 만족하는 처분용기 크기를 도출하였으며, 처분용기 구성요소의 적절한 배열과 안전한 처분조건 등을 설정하기 위한 기본정보도 수록하였다. 처분용기에 주어지는 외압에 대한 응력해석을 위한 안전계수를 2.0으로 하였을 때, 13cm의 사잇거리를 갖는 사용후핵연료 저장통을 둘러싸고 있는 내부충전물의 직경은 112cm로 평가되었으며, 저장통과 용기외부의 가장 얇은 부분의 최소두께는 15cm로 결정되었다. 이러한 크기를 갖는 처분용기는 가압경수로 사용후핵연료 집합체 4개 또는 중수로형 사용후핵연료는 297다발을 수용할 수 있는 것으로 평가되었다. 그러나 향후 처분작업의 방사선적 안전성 확보를 위하여 용기의 상하단 부위에 대한 상세 방사선차폐해석이 필요하다
중심어 : 심지층처분장, 처분용기, 핵임계, 방사선차폐, 구조안정성, 사용후핵연료

INTRODUCTION

Since 1978 there are twenty Nuclear Power Plant in operation, four Canadian deuterium-uranium (CANDU) and sixteen pressurized-water reactor (PWR) facilities. At the end of 2005, the total amount of spent fuels accumulated was about 8,062 tHM (3,400 tHM for PWR and 3,900 tHM for CANDU). According to the "2nd Basic Plan for Electric Power Demand and Supply", which was finalized by MOCIE (Ministry of Commerce, Industry and Energy), eight units will be commissioned by the year 2017. Thus growth in nuclear energy production in Korea, coupled with the aging reactors, planned decommissioning of existing reactors and the long lead time required to bring a geological repository on line, highlights the need to develop the high-level waste (HLW) disposal system. Since 1997 Korea has performed a national R&D program for HLW disposal technology development in order to establish a Korea Standard Reference Disposal System by the year of 2006. The preliminary disposal concept being conceived at present is to encapsulate the spent nuclear fuel into corrosion resistant containers. The packaged disposal containers are then to be emplaced into a mined underground facility located about 500m below the surface in a crystalline rock mass. No site for the deep geological repository has been specified yet, but a generic site with granitic rock is considered for this study. The container has being designed for spent nuclear fuel disposal in a deep repository in the crystalline bedrock, which entails an evenly distributed hydrostatic pressure from underground water and high swelling pressure from saturated bentonite buffer. And also it should protect spent fuels during the desired lifetime from any anticipated external impact under the disposal conditions.

This paper presents key parameters should be considered in designing the disposal container and their effects on the aspects of safety and technical availability under the geologic disposal conditions. The main purpose of this study is to provide basic information for design and optimization of the disposal container. The discussing points are the nuclear criticality and radiation shielding analysis and thermo mechanical structural analysis of container composition and configuration.

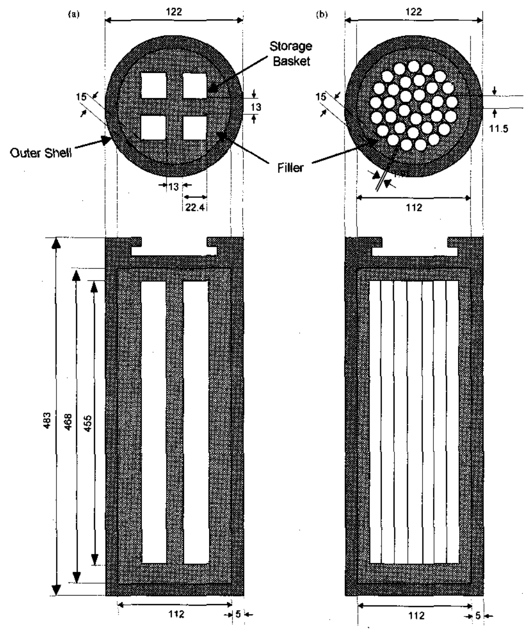
DESIGN BASES

The design bases and controlling parameters for developing the disposal container are:

- Discharge burnup of the reference spent fuels [1] : 45 GWd/tHM for PWR and 7.5 GWd/tHM for CANDU
  Decay heat of spent fuel(40yr cooling): 385 watt/assembly for PWR, 2.28 watt/bundle for CANDU
- The container filling material for void space within the container: carbon steel
- Spent fuel container criteria :
  - Temperature of spent fuel cladding in the container < 200 ℃ (in air)
  - Surface temperature of the disposal container < 100 ℃
  - The maximum dose rate on the surface of the container < 500 mGy/hr
  - The container should be subcritical ($K_{eff} \leq 0.95$)
  - The container has to withstand the mechanical loads consisting of 5 MPa of hydrostatic pressure from groundwater and 10 MPa of swelling pressure of bentonite buffer.

As noted above, the reference PWR spent fuel has the average burnup of 45,000 MWd/tHM (initial enrichment of 4.0 wt%) and is cooled for 40 years after irradiation before the encapsulation and disposal. PWR assembly weight and dimensions are 665 kg and 21.4 cm^2(cross-section) x 453 cm (length), respectively. The reference CANDU fuels have the average burnup of 7,500 MWd/tHM and the fuel dimensions are 10cm (diameter) x 49.5 cm (length).

CONTAINER DESCRIPTION



Canister Outer shell	Ni-Alloy (or Copper or SUS)	Ni-Alloy (or Copper or SUS)
Filler (void space in the container)	Carbon steel	Carbon steel
Capacity	4 PWR Spent Fuel Assemblies	297 CANDU Spent Fuel Bundles
Total No. of the Required Containers	11,375	2,926
Residual Heat in Canister	1,540 Watt	760 Watt
Total Volume of the container	5.64 m^3	5.64 m^3
Total Surface Area	19.67 m^2	19.67 m^2
Total Weight	39,112 kg	39,099 kg
☞Fuel wt.	2,660 kg	7,425 kg
☞Cast Insert	27,345 kg	22,567 kg
☞Outer shell (Case for Ni-alloy)	9,107 kg	9,107 kg
(Case for Copper)	9,211 kg	9,211 kg

Fig. 1. Schematic diagram of the reference disposal container for spent PWR(a) and CANDU(b) fuels.

As illustrated in Figure 1, PWR and CANDU spent fuels are separately encapsulated into the reference container due to the significantly different properties of both fuels and the retrieval potential of PWR fuel for reuse or a certain reason for safety. The overall sizes and component materials of the containers for both spent fuels are designed to be exactly identical to make the encapsulation and handling processes in the repository simplify. The container consists of two major components: a massive cast insert and a corrosion resistant outer shell. The insert provides mechanical strength and radiation shielding, and it keeps the fuel assembles in a fixed configuration. For the insert, carbon steel is considered as the design basis material. For the complete isolation of waste for a long time, high nickel alloy (Alloy22), stainless steel or copper are considered as the candidate corrosion resistant materials for the outer shell. As shown in the figure, the outer shell contains fuel storage baskets (4 square tubes for spent PWR fuel and 33 circular tubes for spent CANDU fuel) and the void space between the fuel storage basket and the outer shell is filled with carbon steel called as cast insert. The loading capacity of the container was determined from the thermal analysis to confirm that the maximum thermal load on the container satisfies the thermal constraint of the bentonite buffer surrounding container. The temperature at the buffer should be lower than 100 ℃ to keep the physical and chemical properties of bentonite. Four spent PWR fuel assemblies and 297 CANDU fuels are loaded in each container. The heat load from the PWR and CANDU containers are about 1.54 kW and 0.68 kW, respectively. The dimensions of the container will be determined from the mechanical structure analysis under the expected mechanical loads in underground repository conditions. The spent fuel received at the encapsulation facility, co-located with the underground repository, is transferred by a remote-handling system to the hot cell for the packaging process where the spent fuels are inserted into the disposal containers. The packaged disposal containers are then to be emplaced into a mined underground facility located about 500m below the surface in a crystalline rock mass.

CRITICALITY AND RADIATION SHIELDING ANALYSES

For the radiation safety in the packaging and emplacing processes of the disposal containers containing spent fuels, a set of radiation shielding and criticality analysis were carried out, of which results are also useful for conceptual design of the disposal system.

Calculation methodology and assumptions

Two calculation steps were applied in the criticality analysis. In the first step, burnup calculations up to each average discharged burnup were undertaken using HELIOS code[2]. Neutron multiplication factor and sets of nuclide number densities corresponding to each average burnup and cooling period were determined under typical reactor conditions. Cross-section library with 45 neutron energy group was used in this calculation. Assumptions to perform the burnup calculations are:

- Specific power : 37.5 W/gU for the PWR fuel assemblies without showdown periods
- Fuel and coolant temperature: 600 ℃, 310 ℃
- Boron concentration : 600 ppm

White boundary condition was applied in the lattice calculation. These burnup calculations provided initial conditions for the second step of the analyses, in which MCNP [3] calculations with continuous nuclear data library were done to determine the reactivity of spent fuel assemblies placed in containers in an underground repository.

For the shielding analysis, ORIGEN2[4] calculation was carried out as a preliminary step to determine source intensity and spectrum of photon and neutron emitted from spent fuel in disposal container. And then, MCNP calculation was performed to evaluate gamma and neutron dose rate from the surface of the disposal container.

Results of Criticality Analysis

Criticality analysis was performed under the assumption that the disposal container containing 4 PWR fuel assemblies was placed in boreholes in crystalline rock and surrounded by saturated bentonite buffer. Spent fuel assemblies were assumed to be centered axially and radially within a storage basket of the container. Each container was assumed to be surrounded by 50cm thick bentonite buffer, with 150 cm of bentonite above the container and 50 cm of bentonite below the container. The containers were assumed to form an infinite two-dimensional planner lattice on a pitch of 6m in the host rock. The model included a rock thickness of 5 m above and below the boreholes. For cases in which water was assumed to have entered the container, the bentonite was considered to be water saturated. Only actinide nuclides were considered for burnup credit.

First, isolation of each borehole with disposal container was checked from the preliminary study. It showed that interaction between the holes in the criticality aspect was negligible when borehole spacing is longer than 4m. Secondly, the optimum moderating condition was searched. It is the most highly reactive when the container is filled with water with the density of 1.0 g/cc. For the repository with PWR fuel assemblies, infinitive multiplication factor appeared to be 0.25 far below 0.95 at normal condition. In case of the container void space was filled with water with the density of 1.0g/cc, infinitive multiplication factor was maintained below ~0.78 after operating of repository, as shown in fig. 2. The reason for the increase of the infinitive multiplication factor after 80 years is due to increase of ^{235}U caused by decay of ^{239}Pu with a 24,000-year half-life and decrease of neutron absorber of ^{240}Pu and ^{241}Am. For the CANDU fuel assemblies, nuclear criticality is no problem under the water-flooded condition, although fresh fuel is placed in a disposal container. Therefore, if the fuel assemblies are intact and fissile nuclide is confined in a fuel rod, the nuclear criticality in a repository is not a big controlling parameter in current design concept.

Results of radiation shielding analysis

To prevent formation of oxidizing chemical

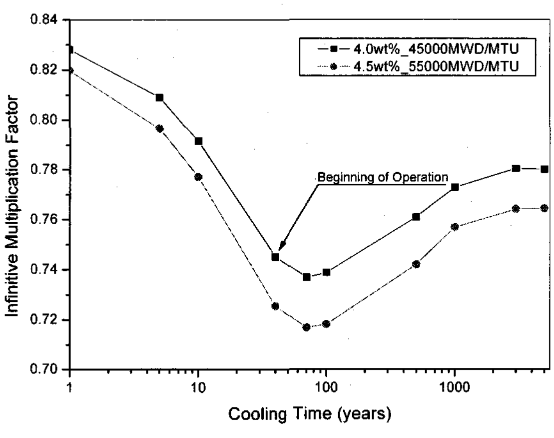


Fig. 2. Infinitive multiplication factor as a function of cooling period.

Table 1. Photon and neutron dose Rates at surface of container.

Fuel Type, Burnup(MWD/MTU)	Absorbed Dose Rate (Gy/hr)			
	Photon		Neutron	
	Side	Bottom(Top)	Side	Bottom(Top)
PWR, 45,000	6.28E-04(6%)[a]	4.48E-01(1%)	3.94E-05(1%)	1.03E-04(2%)
PWR, 55,000	7.97E-04(7%)	5.50E-01(1%)	7.07E-05(1%)	1.81E-04(2%)
CANDU, 7,500	2.63E-04(4%)	1.14E-01(1%)	9.26E-07(1%)	2.19E-06(4%)

[a] Percentage standard deviation

species, neutron and gamma ray shielding is inevitable. Absorbed dose rates on the surface of disposal container estimated based on repository model with canister are discussed. Photon and neutron absorbed dose rates calculated by MCNP with MCPLIB2 library are given in Table 1. It is clear that photons dominate the absorbed dose rate at the surface of the canister. For photon results, absorbed dose rate at bottom of canister is about 500 times higher than that at side. Absorbed dose rate, however, was assured below limit of 500 mGy/hr. Although the proposed dimensions of the container is satisfied with the limit value to prevent radiolysis caused by neutron and photon, the design on the top and bottom part of the container will be examined again with respect to the handling processes of the container.

THERMAL AND STRUCTURAL ANALYSIS

Model formulation

The heat transfer analysis was conducted to provide basic information for conceptual design of disposal container. Heat sources were calculated using a computer program, ORIGEN2, for PWR spent fuels. NISA (Numerically Integrated elements for System Analysis) computer program based upon FEM was used for the numerical solution. The temperature distribution in the composite system of 「canister + buffer + tunnel + rock」 due to heat generation from the spent fuel was obtained. Input parameters for calculations are listed in Table 2. As shown in Fig. 3, top and bottom boundary to calculate temperate distribution were set to the ground surface and 500m below the underground repository. It was assumed that the surface temperature is constantly 20 ℃ and the temperature increases by 3 ℃ every 100 m below. Also the initial temperature of the rock body surrounding the disposal container was assumed to be 50.4 ℃.

The structural strength of the disposal container depends on the structural shape and dimensions of the container and the material[5]. The design variables in this study to perform the

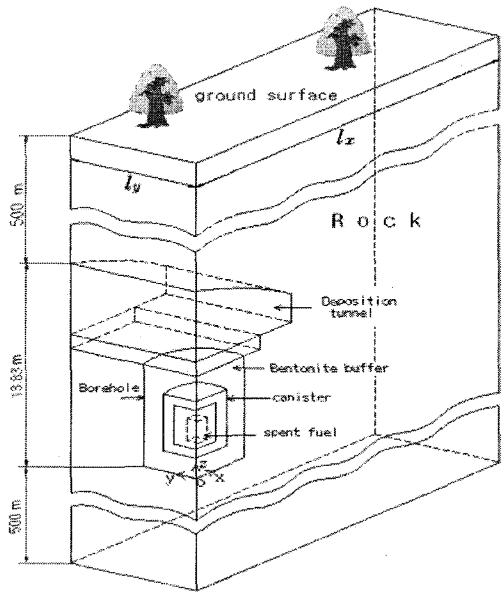
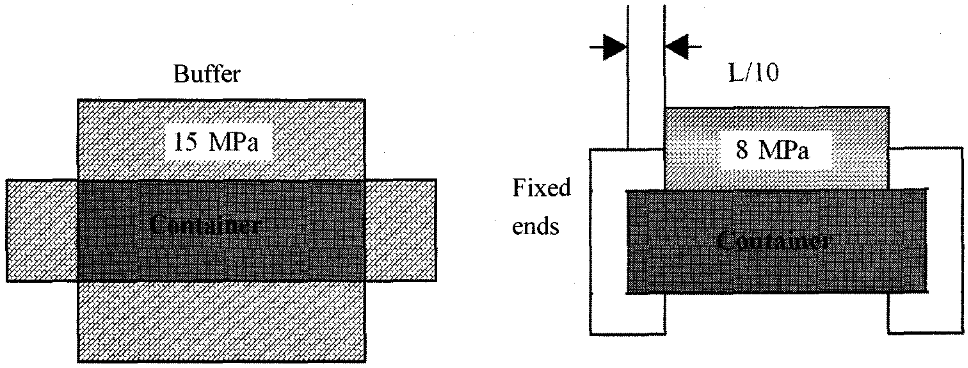


Fig. 3. Model for heat transfer analysis of the underground disposal system.

Table 2. Material properties for the structural stress analysis.

Properties \ Materials	Cast iron	High Ni alloy	Stainless steel	Copper (Outer shell)	Buffer	Back-fill	Granitic rock	Spent fuel
Young's modulus, E (GPa)	83-170	210	190-200					
Poisson's ratio, ν	0.2-0.3	0.31	0.3					
Thermal exp. coeff., α (10^{-6}/C)	9.9-12	13	17					
Mass density, ρ (kg/m^3)	7,400	8,800	7,857	8,900	1,800	2,100	2,660	2,000
Yield stress, σ_y (MPa)	200-290	114-624	128-700					
Ultimate stress, σ_u (MPa)	340-1,400	310-760	400-					
Thermal conductivity, k (W/m · K)	52	26	1,000 31	386	1.3	2.6	3.5	0.135
Specific heat, C (kcal/kg · ℃)	420	460	460	383	1,000	870	1.2	2.64



Case 1 Case 2

Fig. 4. External load conditions due to the hydrostatic pressure and the swelling pressure of bentonite buffer : (a) load case 1 (normal case, (b) load case 2 (abnormal case).

mechanical structural analysis of the proposed container and the dimensioning work are the thickness of the outer shell and the lid/bottom, the minimum thickness of the thinner section between the storage basket and the surface of the cast insert. Other issues studied here include structural integrity due to the temperature build-up within the container and the unexpected rock movement. Values of material properties for the structural stress analysis are also listed in Table 2. Mean values are adopted for some indefinite property values in the calculation.

As mentioned previously, the maximum design load for the container is assumed to be 15 MPa of external pressure. In the case of a normal load, a total of 15 MPa of external load is assumed to be evenly distributed and acting on the whole surfaces of the container as shown in Fig.4 (Load Case1). In the abnormal cases during the groundwater saturation of the bentonite buffer, the swelling pressure could unevenly develop due to the different direction of ground water intrusion as illustrated as Load case 2 of Fig.4. The abnormal case may caused a malpositioned container in the hole, heterogeneous rock properties, or a banana-like curved disposal hole. In this case, uniform pressure of 8MPa is applied perpendicularly on the upper half outer surface of the container with fixed ends.

For the structural integrity of the proposed container, the equivalent von Mises defined as the yield stress of carbon steel divided by the maximum von Mises stress occurred inside the filler is calculated. The safety factor (S) allowable for the safe design of the container is considered as 2.0 in this study. The structural strength of the disposal container is analyzed by the finite element analysis method, using the commercial finite element code (NISA), with respect to the normal and abnormal load conditions as specified in Fig.4. In the numerical analysis for load case 1, some symmetric boundary conditions are used for displacements due to the symmetric loading condition and to prevent a rigid body motion.

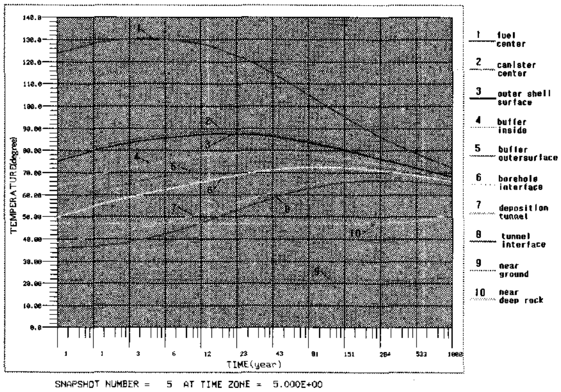
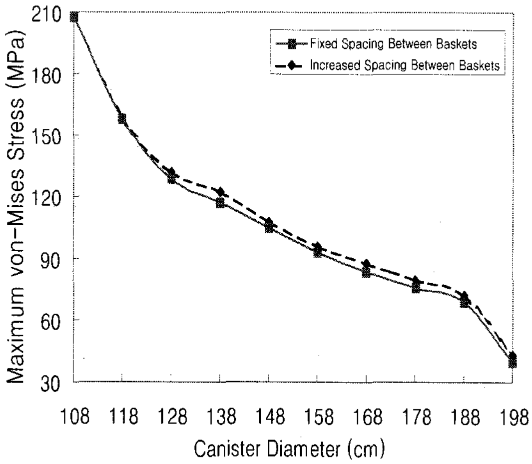


Fig. 5. Temperature distribution of the HLW disposal system as a function of time.

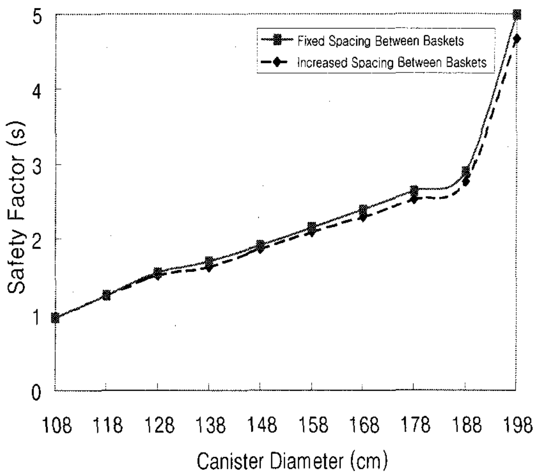
Results and discussion

As shown in Fig.5, in the case of 40 m tunnel spacing and 6 m borehole spacing the peak temperature (87.5℃) showed around 15~16 years after disposal and gradually decreased. The peak temperature on the surface of the container was absolutely lower than 100℃ required for bentonite buffer. Also, the temperature near the ground surface does not change very much.

Figure 6 shows the maximum von Mises stress occurred in the filler as the diameter of the filler diameter increases. When the safety factor for the stress due to the external loads (hydrostatic and swelling pressure) is taken as 2.0, the safe diameter of the container is found to be more than 112 cm to secure the structural



(a) **Maximum von Mises stress occurred in the cast insert of the canister for the PWR fuel versus the canister diameter**



(b) Safety factor versus the canister diameter

Fig. 6. Maximum von Mises stress occurred in the cast insert of the container for PWR fuel versus the container diameter (Load case 1&2).

Table 3. Structural stress analysis results of CANDU fuel container (Load case 2).

Number of baskets inside cast insert of container	17	19	25	33	37
Maximum von Mises stress occurred inside cast insert (MPa)	93.06	93.11	99.75	108.2	112.7
Maximum deflection occurred inside container (cm)	0.097	0.099	0.106	0.11	0.112

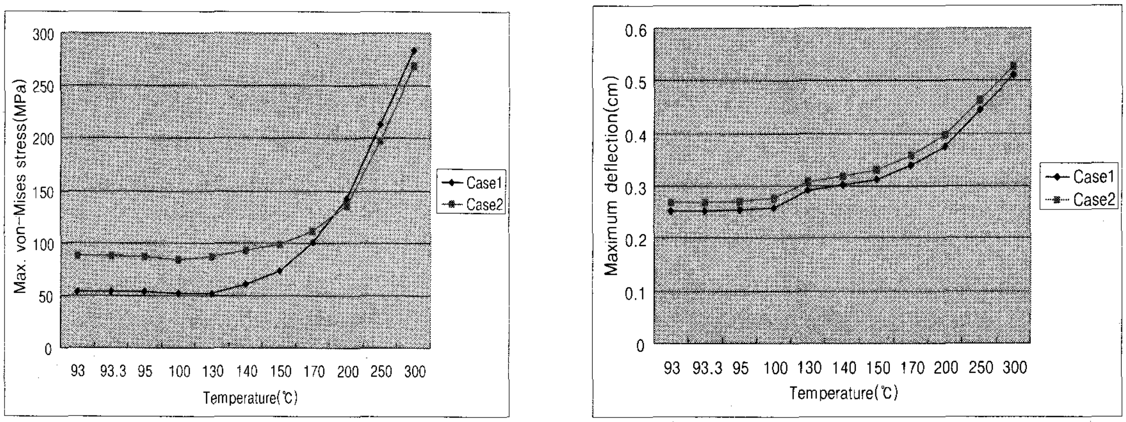


Fig. 7. The maximum von Mises stress (a) and the maximum deflection in the filler (cast iron) as a function of temperature (PWR container, D=122 cm).

safety. This result was applied for the structural analysis of the CANDU container to find the possible numbers of inner baskets to accommodate CANDU fuels. The results for the abnormal load case (Load case 2) are summarized in Table 3. The table shows that the CANDU container with 3-circular baskets may give the safety factor of 2.0 when the diameter of the container is 122 cm. From this result, the thickness of the thinner section of the filler between the fuel basket and the surface of the filler (or inner surface of the outer shell) is determined to be 15 cm, which is evaluated to provide enough container strength under the assumed external loads, for both PWR and CANDU fuel containers. Regarding the dimensions of the filler and the fuel, the PWR fuel container could accommodate 4 fuel assemblies and the CANDU fuel container could accommodate 297 bundles (33 circle tubes x 9 stacks). Of course the configurations of the containers satisfy the thermal constraints mentioned in the previous section.

According to the stress distributions inside container, the stress is concentrated at the outer corner of the square basket in PWR fuel container with Load case 1, but not in other cases. But this stress concentration may not be serious in the actual case. Thermal stresses of the container due to thermal loads of the heat generation of spent nuclear fuels inside baskets may be another reason affecting the structural safety of the container emplaced in the underground repository. Fig.7 shows the thermal stress analysis result of the container. Even though some high thermal stresses occur in the container, the container appears to be still structurally safe less than about 170℃. Because the maximum stress occurred in the container is smaller than the yield strength of the cast iron.

Fig.8 shows the nonlinear structural analysis of the container with 50cm-bentonite buffer that is to predict the structural safety of the container while the host rock is unexpectedly moved up by 10cm. This case may be caused by the earthquake etc. at a deep underground. For the

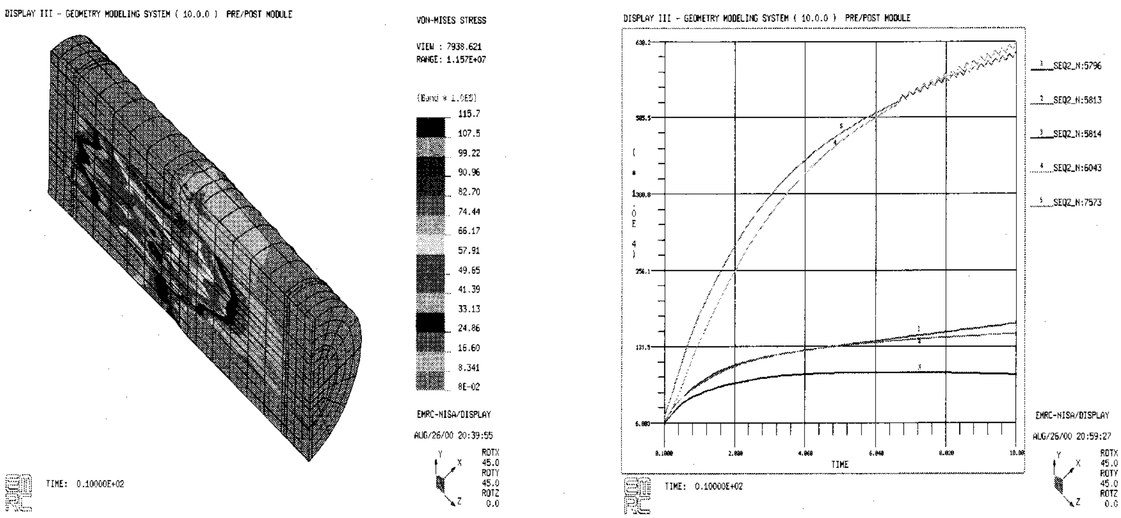


Fig. 8. The maximum von Mises stress of the container (PWR container, D=122 cm).

analysis, horizontal symmetric rock movement is assumed to simulate this case and the elastoplastic material model is adopted. Drucker-Prager yield criterion is used for the material yield prediction of the bentonite buffer and von-Mises yield criterion is used for the material yield prediction of the container(cast iron insert, copper outer shell and lid and bottom). The analysis result shows that even though very large deformations occur beyond the yield point in the bentonite buffer, the container structure still endures elastic small strains and stresses below the yield strength. Hence, the 50cm thick bentonite buffer can protect the container safely against the 10cm sudden rock movement by earthquake etc.. Analysis results also show that bending deformations occur in the container structure due to the shear deformation of the bentonite buffer.

CONCLUSIONS

After emplacing the disposal container in the underground repository, it appeared that the containers may get structural deformations by the external load due to the hydrostatic and bentonite swelling pressure and the high thermal load built-up in the container, the rock movement, etc.. For the first step to develop a reference disposal container under such an expected underground disposal conditions, this paper examined safe dimensions of the disposal container on the points of nuclear criticality and radiation safety and mechanical structural safety and provided basic information for dimensioning the container and configuration of the container components, and establishing the favorable and safe disposal conditions.

When the safety factor for stress due to the external loads (hydrostatic and swelling pressure) is taken as 2.0, the safe diameter of the filler material to provide enough container strength under the assumed external loads is found to be 112cm with 13cm spacing between inner baskets in PWR container. Also the thickness of the thinner section between the fuel basket and the surface of the cast insert is determined to be 150 mm. Regarding these dimensions of the container, the PWR fuel container is sketched to accommodate 4 fuel assemblies and the CANDU fuel container 297 bundles (33 circle tubes x 9 stacks). However the top and bottom parts may be checked again through the detail radiation shielding analysis with respects to the emplacement position and handling processes of the disposal container.

Acknowledgement

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