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A Feasibility Study on Improvement of Storage Density Through a Modification of Shielding Material in the CANDU Concrete Canister

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Abstract

The depleted uranium(DU) was used for increasing storage capacity in spent fuel dry storage. It was investigated that the storage capacity of canister would be improved by almost two times compared with the previous one through increasing the shielding capability against the dose rate. In this study, the material such as the DU was only treated to increase the storage density of canister. It would be better to improve the efficiency of dry storage facilities that the shape of canister is considered together with the kind of material. The shielding analyses were performed with MCNP-4/B Monte-Carlo code. There are several reasons available and useful to use the DU as the shielding material. The density of DU is very higher than any material. In addition, the use of the DU is a side of recycling waste.

Ⅰ. Introduction

The discharged spent fuel bundles from CANDU nuclear plant are stored on-site for several years in water-filled pools. After spent fuel has been in a pool of water for about seven years, the radioactivity and heat have decreased enough to allow the fuel to be transferred to dry storage canister shielded by concrete. In Korea, about 20,000 used-fuel bundles will be discharged from 4 Wolsong Power Reactor each year. There has been wet storage system to store the used-fuel bundles removed from reactor for about 10 years each site. The current storage system will be full of spent fuel bundle by 2006. To accommodate the increased spent fuel storage requirements, our country should have additional CANDU storage facility by 2006. However, it has been very difficult to obtain the additional storage site of spent fuel because of several domestic problems. These days, many studies have been carried out to develop the high-density and high-capability dry storage system.

The interim storage canister in Wolsong unit 1 is using a reinforced concrete of density 2.231g/cm^3 as the outer shielding material. The depleted uranium(DU) was considered to enhance the shielding effect of storage canister. In this study, it was assumed that depleted uranium is mixed with the reinforced concrete. The total density of outer shielding part of canister will be increased as the amount of DU increase because DU's density is very high. It was expected to reduce the thickness and to enhance storage efficiency of fuel bundles. The estimation were performed for dose rate on the surface of canister using a Monte Carlo code MCNP-4/B with ENDF/B-VI cross-section library which has merit of explicit three-dimensional geometric representation.

Ⅱ. Analysis of Radiological Hazards of DU

The radiological hazards of DU are a consequence of the properties of three isotopes of U-234, U-235 and U-238. The relative abundance of these three isotopes in naturally occurring uranium are 99.27%(U-238), 0.72%(U-235), and 0.0057%(U-234). Their abundances in DU vary somewhat but are typically 99.80%(U-238), 0.2%(U-235) and 0.0005%(U-234).

The daughter products of these isotopes of uranium are also radioactive and form "decay chain" that contain many possible radionuclides. Uranium ore and its concentrates can contain a large number of these radionuclides including some, such as Ra-226, that present significant radiological hazards. However, the production of DU by gaseous diffusion results in essentially pure uranium without any decay products.

The only radionuclides that occur in sufficient abundance to have an impact on radiological hazards are Th-234 and Pa-234 from U-238 and Th-231 from U-235. In a few months following production of DU, these isotopes will have built up to their maximum concentration. The radiological properties of these uranium isotopes and decay products are presented in Table 1.

Table 1. Radiological Properties of Uranium Isotopes and Decay Products

Radionuclides	Half-life	Principal Radiation Types
U-234	4.5×10^9 years	α
U-235	7.1×10^8 years	α,γ
U-238	2.5×10^5 years	α
Th-230 (from U-234)	8.0×10^4 years	α,γ
Th-231 (from U-235)	1.17 minutes	β,γ
Th-234 (from U-238)	24.1 days	β,γ
Pa-234 (from U-238)	25.5 hours	β,γ

The radiological hazards of any radioactive material are proportional to the amount of radioactivity present. The various uranium isotopes, and mixtures of those isotopes, can be characterized by their "specific activity", defined as the amount of radioactivity (in Curies) per grams. The specific activities of various mixtures of uranium isotopes are presented in Table 2.

Table 2. Uranium Specific Activities

Mixture	U-235(%)	Specific Activity
U-238(pure)	0	3.33×10^{-7}
DU	0.2	4.0×10^{-7}
NU	0.72	7.0×10^{-7}
Enriched Uranium	2.0	1.0×10^{-6}
Enriched Uranium	20.0	9.0×10^{-6}

There are some beta and gamma emissions from the isotopes of uranium and their decay products that require control in the workplace. However, the external radiation hazards associated with uranium handling and storage are generally not a major concern whether in the workplace or in the environment, the radiological hazards from DU are primarily due to alpha particle emission. This means that the internal radiation dose from ingestion or inhalation of uranium compounds is the limiting hazards under almost all circumstances.

Ⅲ. Estimation of the Canister Dose Rate

In 1991, AECL submitted the technical document related to the spent fuel dry storage system installation and operation. Figure 1 and 2 shows cross-sectional view and axial view of canister in Wolsong unit.

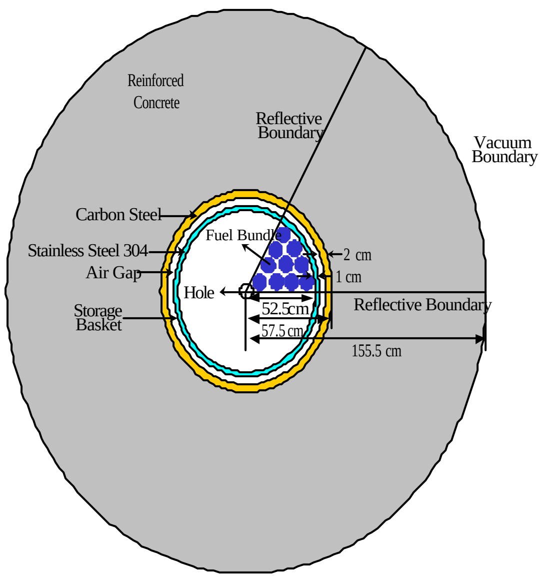

Figure 1. Cross-Sectional View of the Concrete Canister of Wolsong Unit

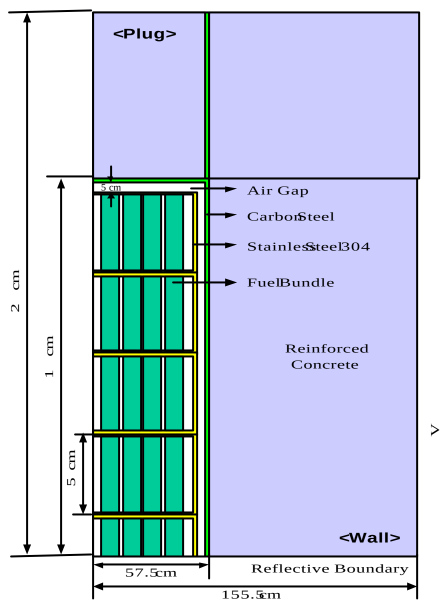

Figure 2. Axial View of the Concrete Canister in Wolsong Unit

The AECL estimated the dose rate of spent fuel dry storage system in Wolsong unit as submitted the technical report. The ONEDANT and TWODANT known as the discrete ordinates transport codes was used for radial and axial radiation calculations. It used an S16 angular quadrature with a 45-neutron group and 16-photon group library. The fuel designated as average has a burnup of 7800 MWd/tU and a 8-year cooling time. The isotope depletion code ORIGEN-S was used together with the burn-up-dependent library for CANDU fuel. Fuel with a cooling time of 8 years was used in the shielding calculations.

The surface dose rate of concrete canister built in Wolsong unit is required less than 25μSv/h. To estimate the dose rate of canister surface, the MCNP-4/B code using Monte Carlo method was employed in this study. Figure 1 shows cross-sectional view of one-sixth of the concrete canister in Wolsong unit with reflective angular boundaries at 0 and 60 degrees. Figure 2 shows axial view of one-second of the canister with reflective boundary at the midst of canister. The wall and plug thickness of the concrete canister is 94 cm and 104 cm respectively. The modeling made in MCNP took advantage of the AECL's source term generated by ORIGEN-S to calculate the dose-rate. The dose rate by the thickness at the wall and plug.

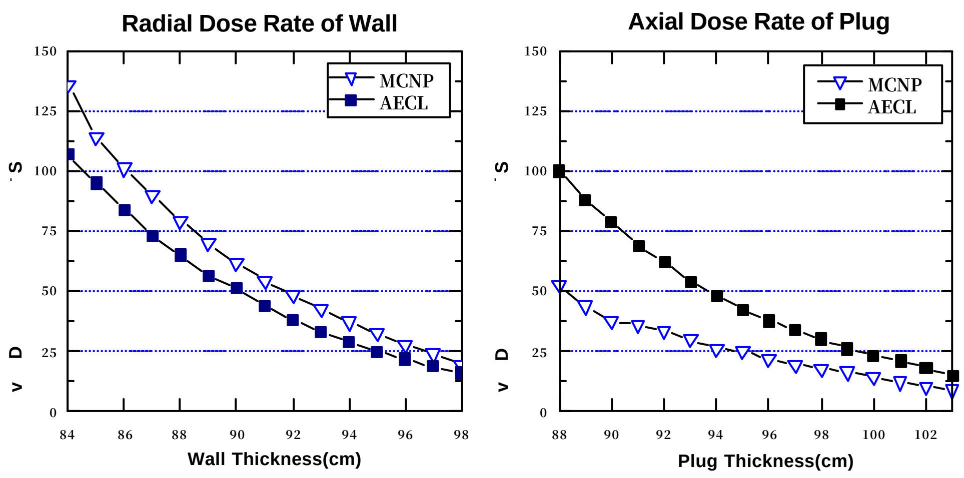

Figure 3. The Comparison of the Dose Rate between AECL and MCNP at the Wall and Plug

Figure 3 shows the radial and axial dose attenuation through concrete. The MCNP as well as the ONEDANT results illustrate that a wall thickness of 94 cm are required to reduce the radial dose rates, to below the 25μSv/h target dose rate. The dose rate of MCNP was 25% on the average over it of AECL at the wall. But on the contrary, the AECL underestimated the dose rate on the average by about 44% compared with the MCNP result at the plug. It was thought that the difference between the ONEDANT and the MCNP was caused by the old cross-section data. After sufficient neutron and photon transport simulations were done, the average relative errors of MCNP tallies was achieved within ±5%.

Ⅳ. A Modification of Shielding Material Using the DU

Currently, the shielding material of canister was made of reinforced concrete having about density of 2.231 g/cm^3. This study assumes the concrete shield was mixed with the depleted uranium(DU) for increase of density. The use of the DU have several merits. Especially, the use of the DU is a side of recycling the waste product. In addition, the density of DU is so higher than the reinforced concrete. The decrease of mean free path of the neutron and photon is given by the increase of the density of shielding material. It was found that the mixture with the DU had the higher shielding effect than the previous one due to increased density.

To consider simplicity of calculation, the homogeneous mixture was assumed. It was considered that the maximum mixture density was like two times of the original concrete density. The DU as the shield was gradually added to the concrete by 10% to 50% of total density. The shielding effect will increase as function of density of mixture as shown in Figure 4, the mean free path for its next collision is greatly shortened. It was investigated that the shielding effect would be enhanced to the maximum two times in the case of DU 50%.

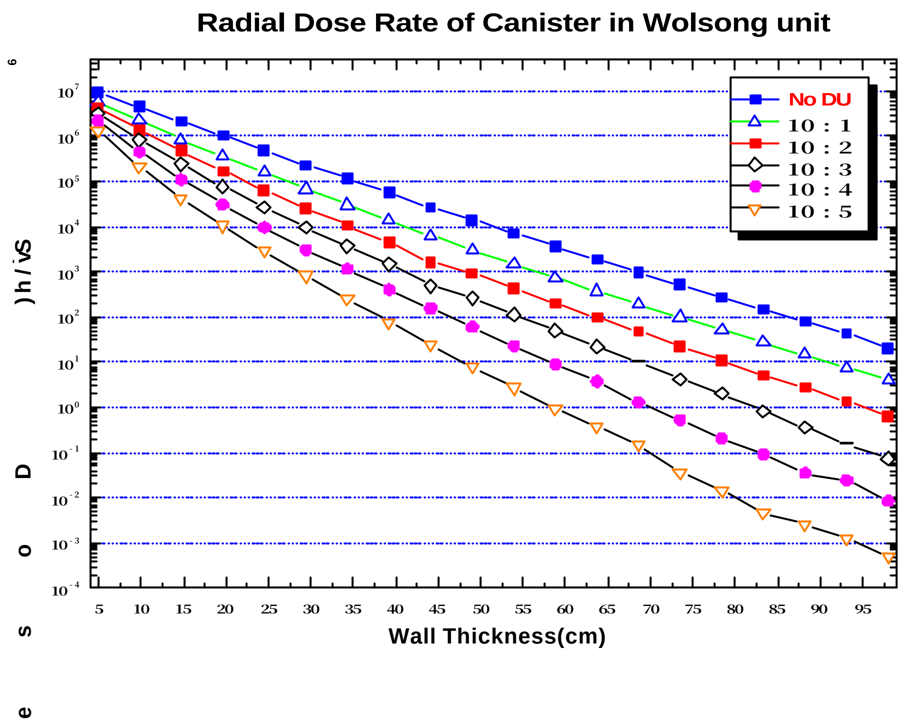

Figure 4. Variation of the Dose Rate on the Concrete Mixture at the Wall

Table 3. The Radial Dose Rate of Canister at the Wall using the DU

Thickness of Wall (cm)	No DU (μSv/h)	DU 10%	DU 20%	DU 30%	DU 40%	DU 50%
4.90	8.99E+06	5.59E+06	4.21E+06	3.08E+06	2.14E+06	1.36E+06
9.80	4.37E+06	2.21E+06	1.35E+06	8.10E+05	4.45E+05	2.13E+05
14.70	2.11E+06	8.32E+05	4.65E+05	2.37E+05	1.09E+05	4.24E+04
19.60	1.02E+06	3.64E+05	1.67E+05	7.53E+04	3.04E+04	1.04E+04
24.50	4.70E+05	1.54E+05	6.38E+04	2.60E+04	9.49E+03	2.81E+03
29.40	2.29E+05	6.66E+04	2.54E+04	9.56E+03	3.06E+03	8.17E+02
34.30	1.16E+05	2.96E+04	1.05E+04	3.71E+03	1.13E+03	2.49E+02
39.20	5.68E+04	1.35E+04	4.54E+03	1.48E+03	4.05E+02	7.69E+01
44.10	2.63E+04	6.29E+03	1.62E+03	4.83E+02	1.53E+02	2.36E+01
49.00	1.40E+04	3.01E+03	9.15E+02	2.60E+02	5.90E+01	7.76E+00
53.90	7.06E+03	1.47E+03	4.27E+02	1.13E+02	2.28E+01	2.72E+00
58.80	3.60E+03	7.31E+02	2.01E+02	4.90E+01	8.90E+00	9.13E-01
63.70	1.85E+03	3.70E+02	9.63E+01	2.16E+01	3.71E+00	3.68E-01
68.60	9.66E+02	1.89E+02	4.66E+01	1.02E+01	1.30E+00	1.48E-01
73.50	5.10E+02	9.74E+01	2.27E+01	4.22E+00	5.30E-01	3.59E-02
78.40	2.72E+02	5.19E+01	1.08E+01	2.01E+00	2.03E-01	1.47E-02
83.30	1.48E+02	2.79E+01	5.18E+00	8.18E-01	9.11E-02	4.54E-03
88.20	8.04E+01	1.44E+01	2.71E+00	3.62E-01	3.48E-02	2.58E-03
93.10	4.30E+01	7.55E+00	1.36E+00	1.57E-01	2.41E-02	1.28E-03
98.00	2.01E+01	3.87E+00	6.38E-01	7.29E-02	8.41E-03	4.96E-04

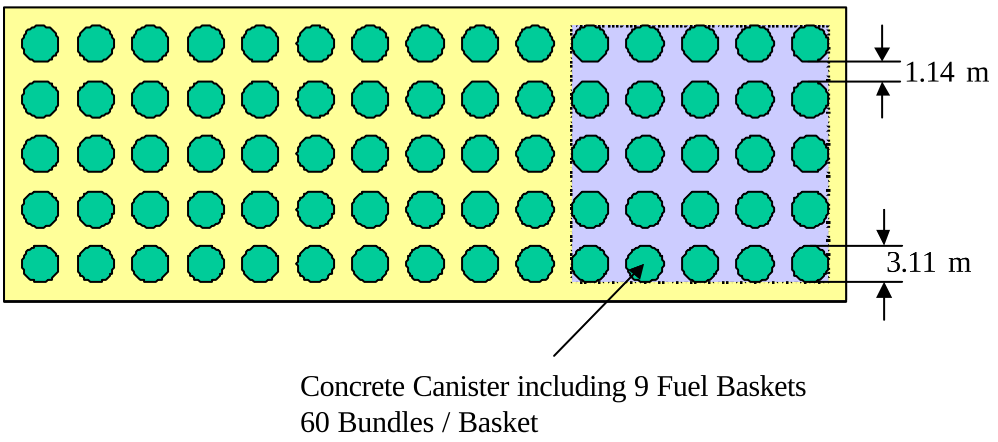

Figure 5. The Cross-Sectional View of the Dry Storage Facilities in Wolsong Unit

Table 3 shows the values of wall dose rate of mixture concrete canister. As shown in Table 3 and Figure 4, it was found that the thickness of wall satisfying the requirement 25μSv/h would be decreased due to increasing the density of shield. The storage efficiency was calculated based on the estimation of dose rate by the MCNP tallies. The cross-sectional view of dry storage facilities in Wolsong unit is shown in Figure 5. There are nine baskets a canister. Each basket has 60 spent fuel bundles. The bundles in one canister amount to 540.

Table 4. Storage Density as Function of Thickness satisfying the requirement

Thickness of the Wall (cm)	98.00	84.36	73.03	63.09	53.60	43.97
Bundles/m^2	33.38	38.42	43.52	48.88	54.98	62.41

The increase of the storage density is connected with the reduction of volumes of concrete canister. Table 4 shows how much area per bundle was occupied as satisfying the requirement 25μSv/h. The storage density increases as the thickness of wall satisfying the requirement decreases. It is expected that the use of the DU shorten the wall thickness satisfying the requirement about 44cm as density of the shield is like two times (4.462g/cm^3) of original reinforced concrete. At that time, the storage density is almost doubled, 62.41 bundles/cm^2. The estimation of effective area of fuel bundle can be carried out in many ways. The storage density of the site was calculated based on the canister own space and management space(1.14m) using the standard simple calculation method.

Ⅴ. Results and Discussions

Using the DU as the shielding material, it was investigated that the storage capacity of canister would be enhanced by almost two times compared with the previous one. In this study, the material such as the DU was only treated to increase the storage density of canister. It would be better to improve the efficiency of storage facilities that the shape of canister is considered together with the kind of material. There are several things available and useful to use the DU as the shielding material. The density of the DU is very high. And use of the DU is a side of recycling waste as well. The industry continue to yield the DU in the future.

The economic analysis related to using the DU as the shielding material would be needed to be carried out. Meanwhile, although the secondary radiation induced by the DU is occurred in the shield, it would be small. To confirm the safety of the shield including the DU, the detailed study would be needed.

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