THERMAL HYDRAULIC & NEUTRONIC ANALYSIS OF DRY CASK STORAGE SYSTEMS FOR SPENT NUCLEAR FUELS

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ABSTRACT

Interim spent fuel storage systems must provide for the safe receipt, handling, retrieval and storage of spent nuclear fuel before reprocessing or disposal. In the context of achieving these objectives, the following features of the design were taken into consideration for metal shielded type storage systems; to maintain fuel subcritical, to remove spent fuel residual heat, to provide for radiation protection.

These features in the design of a dry cask storage system were analyzed by employing COBRA-SFS and SCALE4.4 (ORIGEN, XSDOSE, CSAS6) codes for normal operation of the system under study.

In accordance with safety assurance limits of International Atomic Energy Authority (IAEA), appropriate designs for Dry Cask Storage Systems (DCSS) were reached for 33000, 45000, and 55000 MWd/t bumup values and 5 and 10 years of cooling periods for spent fuel to be stored (Table 1).

INTRODUCTION

The design of a metal-shielded DCSS is under study. Three main requirements for a DCSS are (1) subcriticality must be maintained all the time, (2) Spent Fuel residual heat must be removed, and (3) protection against radiation must be ensured [2],

Thermal hydraulic and neutronic analysis of the DCSS under study are performed for normal operation conditions and for several selected bumup values (33000, 45000 and 55000 MWd/t) and cooling times of spent fuel (5 and 10 years) in order to determine if these requirements are met.

MODEL

Radial configuration of DCSS is presented in Fig 1, which consists of an inner steel shell, a Pb gamma shield, an outer steel shell, a neutron shield, and an outside barrel. The three steel components are assumed to be SS-304 (69.5% Fe, 19% Cr, 9.5 % Ni, and 2% Mn by weight). The neutron shield consists of 28.5% water (1.0 g/cc), 66.0% ethylene glycol ($HOCH_2CH_2OH$ at 1.11 g/cc), and 5.5% potassium tetraborate ($K_2B_40_7$*$8H_20$ at 1.74 g/cc) by volume. By weight, this common mixture of water and antifreeze contains ~1% natural boron. The interior of the cask contains a large monolithic aluminium cylinder having 24 square holes, each measuring 22.1 cm x 22.1 cm, and each containing a dry PWR spent fuel assembly. Each 17 x 17 assembly has 25 pins removed, leaving a total of 264 Zircaloy-clad fuel pins per assembly. The spent fuel consists of 10.33 g/cc UO_2 (0.84 wt % ^{235}U and 99.16 wt % ^{238}U) and 0.077 g/cc PuO_2 (70.77 wt % ^{239}Pu and 29.23 wt % ^{240}Pu). The fuel pins are spaced on a 1.260-cm pitch and are 0.950 cm in diameter. The fuel pellets are 0.820 cm in diameter, and the Zircaloy clad is 0.057 cm thick. The neutron/gamma source strength in the homogenized fuel/basket region depends on the number of spent fuel assemblies (24) and the age of the spent fuel.

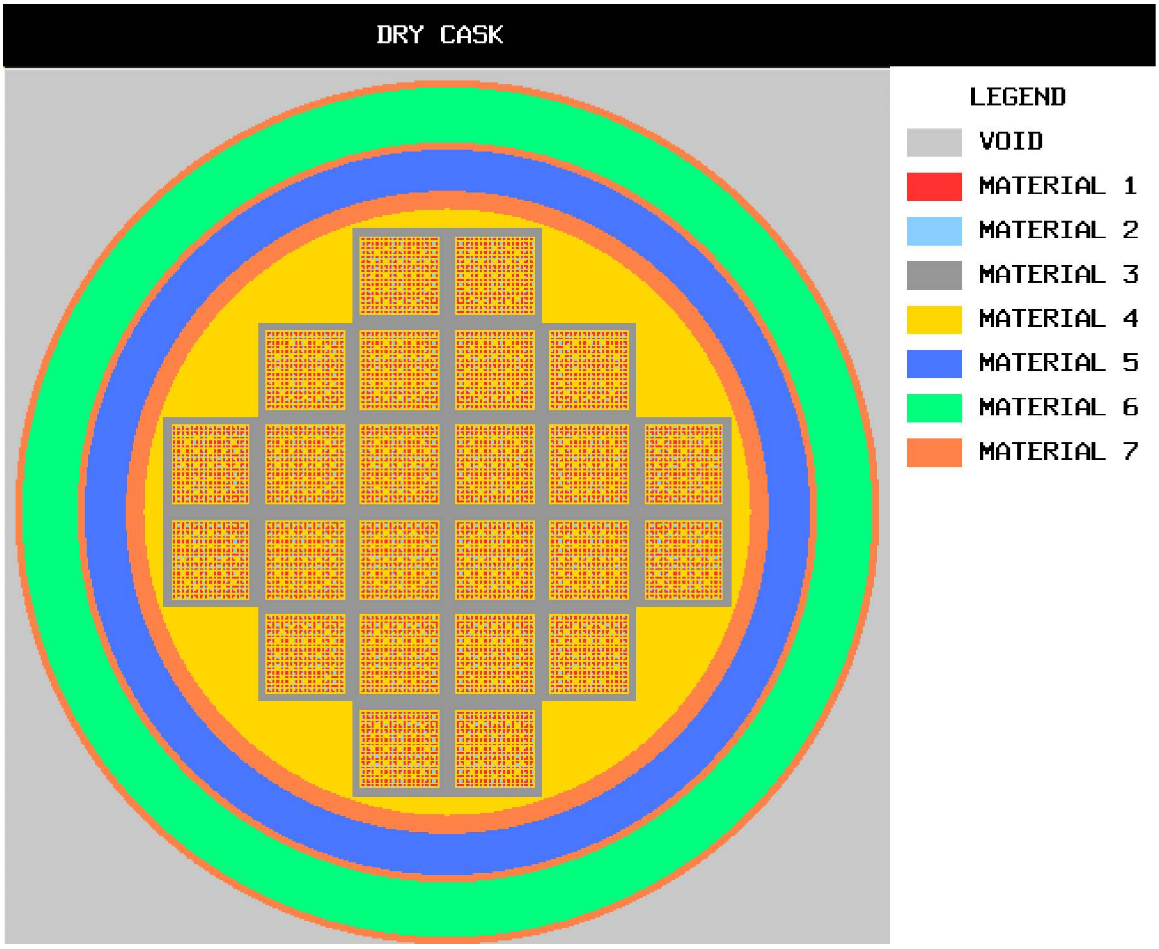

FIGURE 1: Radial Configuration of Metal Shielded Type Storage System

Void = Air

Material 1 = UO_2 (% 3.3 enriched U)

Material 2 = Zircalloy

Material 3 = Aluminum (Al)

Material 4 = Helium (He)

Material 5 = Gamma Shield, Lead (Pb)

Material 6 = Neutron Shield

Material 7 =Stainless Steel (SS304)

COMPUTATIONS AND CODES USED

Six cases for DCSS design (for different bumup values and cooling times of spent fuel which is as shown in Table 1), which are subject to the safety assurance limits of IAEA [5] given in Table 2, are to be investigated.

Table 1 Cases for different burnup values and cooling times of spent fuel

Case	Fuel Burnup (MWd/tU)	Spent Fuel Cooling Time (Years)
1	33000	5
2	33000	10
3	45000	5
4	45000	10
5	55000	5
6	55000	10

Table 2 Design Limitations

	DCSS that maintain 5 years cooled Spent Fuel.	DCSS that maintain 10 years cooled Spent Fuel
Total Dose Rate at the Boundary of Controlled Area (100 m away from cask)= Dose	Dose < 25 mrem/yr	Dose < 25 mrem/yr
Criticality = k_{eff}	keff < 0.95	keff < 0.95
Maximum Clad Temperature = T_c	$T_c < 380^oC$	$T_c < 340\ ^o$ C

For each case, burnup, shielding, criticality and thermal hydraulic calculations are performed.

Burnup calculations yield neutron and gamma spectra (22 n and 18 y group by ORIGEN), which are used in shielding calculations (by XSDOSE).

Criticality calculations are performed using CSAS6 with fresh nuclear fuel assumption and no boron concentration in the neutron shield.

Thermal-hydraulic calculations are performed using COBRA-SFS. Decay heat of spent fuel for six cases given in Table 1 are provided from burnup calculations, (as output of ORIGEN). Decay heat is transferred to the coolant (He) by natural circulation in the fuel basket region, and also to the ambient by natural circulation of air at the outside surface of cask [3],[1].

Radiation protection calculations are performed using XSDOSE. Neutron and gamma spectra are provided from burnup calculations (as output of ORIGEN). Dose rates are calculated as functions of shield thicknesses at the edge of controlled boundary. SSTL thickness was not varied and was taken from REA 2023 [4], since it is determined by structural analysis. Shielding material weight and cost are considered for selecting the final design among those which satisfy radiation protection criterion.

RESULTS

Geometric description of the metal-shielded DCSS designs for six different cases are presented in Table3 and normal operating conditions are given in Table4. For each case of design assuring radiation protection according to the IAEA limit, keff and maximum clad temperature are sufficient far below the limit of IAEA. Accident analysis should be performed far to determining the behavior under transient conditions and then thus for reaching final design satisfying criteria.

Table 3 : Geometric Description of DCSS

Case	1	2	3	4	5	6
Number of Assembly DCSS Contains	24	24	24	24	24	24
Height of DCSS (cm)	462.0	462.0	462.0	462.0	462.0	462.0
Outer Radius of DCSS (cm)	123.63	118.43	127.73	124.33	132.93	130.83
Neutron Shield Thickness (cm)	18.0	15.3	21.2	20.6	26.2	27.1
Gamma Shield Thickness, Pb (cm)	12.5	10.0	13.4	10.6	13.6	10.6
Weight of Neutron Shield (kg)	5892	4782	7156	6717	9152	9220
Weight of Gamma Shield (kg)	55651	46082	59848	49686	62116	51644
Weight of SSTL SS304 (kg)	21814	21197	22251	21775	22765	22395
Weight of Material Used (kg)	83357	72061	89255	78178	94033	83259
Cost of Material Used ($)	54003	48477	56626	50973	58376	52696

Table 4 : Operating Conditions of DCSS

Case	1	2	3	4	5	6
Burnup of S.F (MWd/t)	33000	33000	45000	45000	55000	55000
Cooling Time (Years)	5	10	5	10	5	10
Neutronic Parameters						
Criticality k_{eff}	0.5800	0.5770	0.5812	0.5794 0.5767	0.5823 0.5791	0.5812
Gamma Dose Rate At Controlled Boundary (mrem/yr)	22.708	21.242	21.406	20.589	22.070	22.000
Neutron Dose Rate At Controlled Boundary (mrem/yr)	2.045	3.722	3.526	4.351	2.914	2.887
Total Dose Rate At Controlled Boundary (mrem/yr)	24.753	24.964	24.932	24.940	24.984	24.887
Thermal Hydraulic Parameters						
Coolant(He) Pressure (atm)	1	1	1	1	1	1
Ambient Temperature (°C)	43	43	43	43	43	43
Maximum Clad Temperature (°C)	96	75	118	86	136	95
Heat Transfer Method in the Baskets	Natural Circulation	Natural Circulation	Natural Circulation	Natural Circulation	Natural Circulation	Natural Circulation

REFERENCES

1) U.S. Nuclear Regulatory Commission, NUREG-1536 : Standart Review Plan for Dry Cask Storage Systems.
2) Safety Series “Design of Spent Fuel Storage Facilities” International Atomic Energy Agency, Vienna, 1994.
3) I.S. Levy, *et al.,* Pacific Northwest Laboratory, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircalloy Clad Fuel Rods in Inert Gas," PNL-6189, May 1987.
4) A. Alan Moghissi, Herschel W. Godbee, Sue A. Hobart “Radioactive Waste Technology”, 1986.
5) U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste," Part 72, Title 10, "Energy."