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Radiation Shielding Analysis for a Spent Fuel Storage Cask under Normal Storage Conditions

Dheya Shujaa Al-Othmany
Department of Nuclear Engineering, Faculty of Engineering, King Abulaziz University, Jeddah, Saudi Arabia

ABSTRACT
In most cases, gamma radiation is the dominant dose contributor, but in specific configurations, neutron radiation can become significant for the overall dose rate. This occurs for canister storages where the amount of spent fuel is large and thick concrete shields or entry mazes are used for radiation protection. The design of the cask is based on the safety requirements for normal storage conditions under 10 CFR Part 72. A radiation shielding analysis of the spent fuel storage cask optimized for loading design basis fuels was performed for a single cask. For the single cask, dose rates at the external surface of the spent fuel storage cask, some distance away from the cask surface, were evaluated. The results of the shielding analysis for the single cask show that dose rates were considerably higher at the lower side (from the bottom of the cask to the bottom of the neutron shielding) of the cask. However, this is not considered to be a significant issue since additional shielding will be installed at the storage facility. The shielding analysis results showed exponential decrease with distance off the sources. The controlled area boundary was calculated to be approximately 280m from the array, with a dose rate of 20 mrem/yr. Actual dose rates within the controlled area boundary would be lower than 25mrem/yr, due to the decay of radioactivity of spent fuel in storage. Another finding of the study is that the burnup distribution of the spent fuel needs to be taken into account when assessing the yield of the neutron radiation source, because use of the assembly average burnup leads to underestimation of it.
Keywords: disposal of spent nuclear fuel, radiation shielding, storage condition, dual-purpose cask, spent fuel assemblies
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INTRODUCTION

The pools that were designed initially for short-term storage have now become quasi-permanent storages because of the recent prohibition on the reprocessing of spent fuel. Wet storage needs special care to maintain good water chemistry pH-values, chloride and sulphate impurity concentrations and conductivity. Currently, concrete canisters identical to the Point Lepreau's dry storage system are in use for an interim storage of spent fuels at Wolsong nuclear power plant unit (a CANDU-PHWR). A major factor in spent fuel dry storage design and one of the key thermal safety issues for licensing a spent fuel dry storage system is dissipation of the residual heat generated by the spent fuel. That is, the spent fuel temperature in a dry storage canister must be kept below 160 °C to avoid fuel oxidation [1].

In the storage cask method fuel is stored in units, which are self-shielding, cooling and protecting. Each cask is a hollow reinforced concrete cylinder sitting on a concrete base. The spent fuel is contained in a steel can at the center of the cylinder, with possibly a thin lead layer between the fuel can and the concrete walls. In this application, natural draft air circulation through an annulus between the shield and the fuel can was necessary to remove the heat being generated at the design rate.

The geometry and shape of fuel assemblies (cylindrical or spherical) is an important parameter in the design of casks. The arrangement of fuel elements in a fuel assembly is a major concern in the design of cask. The cask physical layout is evaluated in the perspective of fuel elements arrangement. Heat transfer models and shielding design greatly vary with the arrangement of fuel elements. The design of cask should be compatible with the environmental conditions (wind velocity, humidity, temperature etc.). The geological factors are to be considered which include natural disasters like earthquakes, storms etc [2, 3].

The design of casks shall be such that when they are loaded with fuel, the external radiation fields do not exceed the criteria or limits recommended. Loading and unloading of spent fuel into casks, silos or vaults in a storage configuration shall be done using equipment and methods designed to limit skyshine and the reflection of radiation towards uncontrolled areas, in accordance with the ALARA principle.

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The design of fuel baskets or canisters intended for use with casks and the design of the casks within which the fuel baskets or canisters are to be placed, shall ensure that the fuel will remain in a configuration that has been predetermined to be subcritical during loading, storage and retrieval [4].

The design of dry fuel storage facilities should allow for any consequences likely to result from the redistribution or the introduction of a moderator as a consequence of an internal or external event. If sub-criticality under these conditions cannot be assured, then arguments should concentrate on why they are unlikely. This requires substantial consideration of site conditions with supporting analysis and/or demonstration that the stored fuel can remain effectively isolated from the external environment.

The cask shall be adequately constructed of suitable materials, using appropriate design and construction methods, to maintain shielding and containment functions under the environmental and loading conditions expected during its design lifetime unless adequate maintenance and/or replacement during operation can be demonstrated. The design of concrete or metal casks shall incorporate containment barriers acceptable to the Regulatory Body. Containment barriers to prevent the release of radionuclides include liners that might form integral parts of the cask structures [5].

MATERIALS AND METHODS For the shielding analysis, independent analyses for source terms and shielding are required. To perform these analyses, proper tools could be chosen such as ORIGEN2.1, ORIKAN, MCNP, QAD, DORT, SCALE, etc. It is also important to do decay heat estimation for the spent fuel after it has been cooled for a certain period of time in the wet storage pool, is calculated for a fuel bundle. This determined heat load is then used in further calculations. In order to carry out criticality assessment of the facility, the multiplication factor has been calculated for the dry storage and in accident conditions. For this purpose the two computer codes are usually used [6]:

- WIMS-D/4: generates the necessary group constants for the homogenized cell and for the materials.
- CITATION: calculates the infinite multiplication factor for the assembly.
- SCALE 4.4: used on the spent fuel nuclide composition calculation with the core operating parameters, and the keff calculation.

A major factor in spent fuel dry storage design and one of the key thermal safety issues for licensing a spent fuel dry storage system is dissipation of the residual heat generated by the spent fuel. That is, the spent fuel temperature in a dry storage canister must be kept below 160°C to avoid fuel oxidation. In order to evaluate the maximum fuel rod temperature in the canister under design conditions, one may need to solve a multi-dimensional heat transfer problem with an extremely complicated geometry where three modes of heat transfer are superimposed [7].

Two types of cask basket structure are considered for criticality control. The basket using boral is a stainless steel honeycomb structure with rectangular fuel load cells and poison (boral) plates attached to their outside surfaces. Boral as a neutron absorbing material is mechanically attached to the cells such that the fuel assemblies are fully surrounded by the neutron absorbing material. The neutron absorbing material extends the full height of the active fuel. The boral contains the B-10 content. The other type of basket uses borated stainless steel (BSS) as a basket structure material. This basket consists of interlocking flat BSS plates which form square fuel load cells, and a loaded fuel shares the plate walls of basket fuel cell with adjacent fuels [8].

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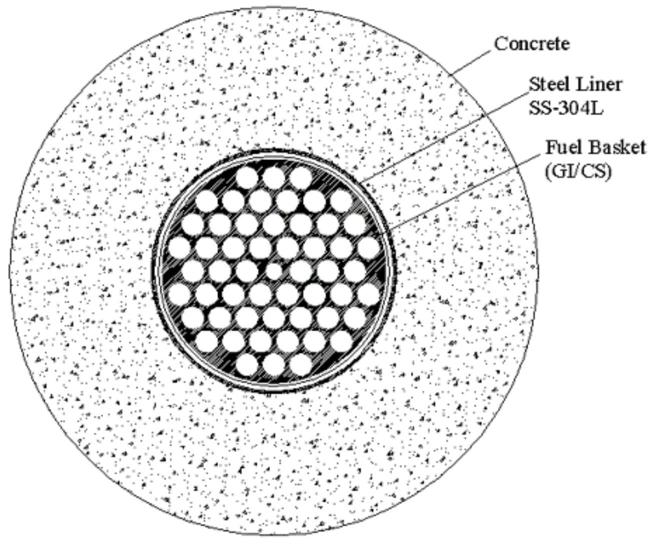

Figure 1: General layout of the spent fuel cask

Most countries use helium as the storage atmosphere, although both air and helium are used in Canada, while South Korea uses an air atmosphere only. The best worldwide summary of light water reactor-related temperature limit determination for inert dry storage of spent fuel assemblies was published by EPRI [9]: cladding creep from internal gas overpressure constitutes the rate-determining degradation and failure mechanism for setting maximum allowable storage temperature limits in dry inert storage. If creep were allowed to proceed to rupture, the fracture mode would most likely be of a pinhole type. Nevertheless, it is desirable to avoid any degradation. This can be accomplished by confining the creep degradation mechanism to its primary and early secondary stages. If it can be shown that creep strain never exceeds that critical strain domain during inert dry storage, tertiary creep and subsequent fuel rod defection can be ruled out. The inner shell is usually of forged carbon steel or stainless steel material. It is provided as a containment barrier for fission products. It also serves as a shielding material against gamma and neutrons as well as it helps in the removal of decay heat from the spent fuel. It also provides the structural support to the spent fuel assembly.

There are two types of shielding materials required: one for the fast neutrons and the other for gamma radiations. Fast neutrons are highly energetic particles and give a large dose in a little exposure. So, their shielding is of major concern. Gamma rays are highly penetrating radiations and can not be stopped completely but can be attenuated to a safe limit. Thus, there are two types of shields used in a cask; neutron shield and gamma shield.
Concrete is the most widely used neutron shield. Usually, reinforced concrete is used to give structural strength to the cask and provide for the time-dependent degradation. Stainless steel is also used as a neutron shield but it faces a problem of "neutron streaming". Hydrogenous materials like water is very efficient in the shielding of fast neutrons. It slows down the neutrons and the boron present in it absorbs these neutrons.
Lead is mostly used for the shielding of highly energetic gamma rays. Concrete is also very much effective in the shielding of gamma rays. Not all the types of cask possess the outer shell. It only serves as a final containment barrier against the release of radioactivity. It is usually made up of stainless steel. The upper lid is a removable head with lifting lugs in order to facilitate the easy retrieve ability of spent fuel assemblies.
The upper lid is also made up of shielding material to prevent the release of radioactivity. A storage cask is usually provided with a basement. This usually consists of concrete [10].

RESULTS AND DISCUSSIONS

Spent fuels have different burnups along the axial height of the assembly, giving off different levels of heat and radiation. In order to analyze the radiation shielding capability of storage casks in practical conditions, the shielding analysis should include varying emission ratios along the emission axis In the shielding analysis for a single cask, dose rates were evaluated at the external surface of the cask; 1m and 2m away from the cask surface. These results can be used as data to control handling time, work procedures, and radiation worker dose rates in accordance with then ALARA principle for such work as the installation, visual examination, radiation monitoring, and maintenance of casks.

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For the Single Layered Shielding Analysis, the source term has been considered as line source and relations for calculations of dose are defined as follows:

For shielding calculations, it is assumed that cladding provides no gamma attenuation and the shielding is exposed equally from all booster elements. The following parameters are used for calculations:

Gamma Energy = E = 0.63 Mev
Source Height = H= 200 cm
Source Strength = S = 1.6 x 10^{10} γ/cm^2 . sec
S=Distance of dose point = R = 12 cm
Lead shield thickness = a = 6 cm
Taylor's coefficient of Lead = A = 2.007
$\alpha_1 =$ -0.04
$\alpha_2 =$ -0.30

Linear attenuation coefficient = μ = 1.3 cm^{-1}

$X_1 = (1+\alpha_1)\mu a = 6.9$
$X_2 = (1+\alpha_2)\mu a = 9.1$

$\theta_1 = \theta_2 = 80°$

Build up flux is found from the following [11]:

$$\varnothing_b = \frac{S}{4\pi R}\sum_{n=1}^{2} A_n \{F\,[\theta_1, \quad (1+\alpha_n)\,\mu R] + [\theta_2, \quad (1+\alpha_n)\,\mu R]\}$$

Using Sievert integral functions, the buildup flux is found to be

$$\varnothing_b = 1.7 \times 10^{10} \ \frac{\gamma}{cm^2\ sec}$$

Similarly the build up flux for other energy groups are as follows:

Energy (Mev)	Source Strength γ/cm. sec	Build up flux γ/cm^2. sec
0.63	1.60E+10	1.66E+05
1.15	4.38E+7	9.03E+03
1.60	4.25E+6	3.54E+03
2.05	1.07E+6	1.17E+02
2.45	9.01E+4	4.65E+01
2.80	6.53E+3	5.58E+00
3.30	2.14E+2	1.35E-01

The analysis of dose rates was performed for members of the public outside of the controlled area boundary. In particular, this analysis was performed by reflecting a sufficient air-layer to consider the effect of scattering radiation caused by the skyshine effect. In this shielding analysis, the self-shielding effect between casks were taken into account, because the amount of radiation from the longer side of the array is greater than that of the shorter side of the array since there are more casks positioned lengthwise. Thus, the directional orientation of the array affects the calculation of the distance to the controlled area boundary.

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CONCLUSIONS

A radiation shielding analysis of a metal storage cask, optimized for loading on design basis fuel assemblies, under normal storage conditions for a single cask. For the single cask, dose rates were evaluated on the external surface of the cask, and at points of one and two meters away from the cask surface. The results of the shielding analysis for the single metal cask showed that dose rates were considerably higher at the lower side (from the bottom of cask to the bottom of the neutron shielding) of the cask, at over 2mSv/hr at the external surface of the cask. Actual dose rates in the controlled area boundary will be lower than 25mrem/yr, due to the decay of radioactivity of spent fuel in storage. In particular, if more precise environmental conditions, the shielding analysis method and the results of this study will be of value. Thus, in the future, the results of this study will be useful for the design and operation of on-site storage facilities or intermediate storage facilities based on a spent fuel management strategy.

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