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The effect of the number of spent fuel casks on the dose of the outer part of the hall concrete wall

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HIGHLIGHTS

- Storage casks were simulated using the computational code.
- Gamma and neutron dose rates were calculated for a 25-capacity storage hall.
- Shielding effect of the cask material has important role on external wall dose rates of the hall.

ABSTRACT

The management of high radioactive spent nuclear fuel (SNF) from research and power reactors has become a key topic of discussion in the nuclear communities. Metal casks are used for the management and disposal of spent fuel and all types of radioactive waste worldwide. The spent fuel assemblies-contained casks are stored in interim storage facilities. The present study aims to show the neutronic behavior and neutron/gamma dose rates of a designed hall for storage of the casks as a current technical, economic, safe and flexible solution, adaptable to any long and short-term SNF storage strategy. The hall structure was considered as ordinary concrete with an internal dimension of $5\times6\times5$ m^3. The concrete wall thickness was discussed to keep the dose rate limit of 10 μSv.h^{-1} (neutron and gamma) at its external side when 25 casks are available inside the hall. ORIGEN and MCNPX computational codes were used to model the storage hall contained 25 Tehran Research Reactor spent fuel casks. The carried out calculations showed 30 cm thickness would fulfil total gamma and neutron dose rate limitation after the external surface of the concrete wall. When the hall contains 25 casks (any contains 16.55%-burnup 10-years cooled spent fuel assembly), maximum gamma and neutron dose rates at the external surface of the hall are 2.6 nSv.h^{-1} and 1.16 μSv.h^{-1} respectively. In addition, the carried out calculations showed natural circulation of air could powerfully remove the deposited heat of neutron and gamma rays.

KEYWORDS

Spent fuel cask
Gamma and neutron dose rate
Interim storage hall
MCNPX simulation

HISTORY

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1 Introduction

Spent fuel generated by nuclear reactors is directly sent to spent fuel pool at first after its final use. The spent fuel stays at the spent fuel pool for a certain cooling time. After the proper cooling time in wet storage sites, additional storage options as dry storage capacities in the designed casks may be selected. Usually decision on the interim or long storage or even direct disposal depends on any country strategy for spent nuclear fuel (SNF) management.

Diaz-Maurin et. al. investigated a socio-technical multi-criteria evaluation (STMCE) framework and method that supports the search for compromise solutions for commercial spent fuel management (Diaz-Maurin et al., 2021). Many countries select interim storage of SNFs using dry casks because of some reasons such as reuse of the spent fuel in their further nuclear plans or unavailability a standard and suitable direct disposal area.

Dual-purpose metal casks have proven a current technical, economic, safe and flexible solution, adaptable to any long and short-term SNF strategy (Palacio et al., 2014). Spent nuclear fuel must be transported and stored in normal transport conditions to avoid its degradation during long-term storage,. In order to transport the spent fuel safely, a tie-down structure for supporting and transporting a cask containing the spent fuel is essential (Jeong et al., 2021).

In the following, some of the design characteristics of the storage casks such as their capacity, shielding material as well as the used computational codes for calculating of their gamma and neutron dose rate would be reviewed.

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In Korea, a dual-purpose cask (DPC) that can be used for both transportation and storage has been developed with domestic technology (Dho et al., 2016; Baeg and Cho, 2018). Korea designed a 21-design basis fuel assembly's cask, which composes of a main body made of carbon steel and a stainless steel dry-shielded canister. In the carried out work for calculation of gamma dose rates, primary gamma sources generated from the active fuel region were calculated using the SAS2H/ORIGENS module of SCALE 5.1 (Ko et al., 2014). Many metal casks have been built for the spent fuel storage. For example, the TN-24 cask is a dual-purpose transport and storage package that its designed vessel in forged steel provides the main gamma shielding while an external layer of neutron shielding resin covered by an outer steel shell is used for neutron shielding (Garcia, 2014). In addition, a dual-purpose metallic cask, able to accommodate 21 fuel assemblies has been designed and licensed at Spain. The cask body is stainless steel with inner and outer shell, which is separated by a lead gamma shield (Narkūnas et al., 2021). Narkunas et al. presented the neutron dose rate analysis of the new CONSTOR®RBMK-1500/M2 storage cask intended for the spent nuclear fuel. Neutron dose rate modeling was performed using the MCNP 5 computer code with very detailed geometrical representation of the cask and the fuel. They explained that after no less than 5 years of wet storage in the pools, SNF assemblies could be loaded into the storage casks for the dry interim storage. The cask has a capacity of 182 SNF. The obtained modeling results were compared with the measurement results and it was revealed, that modeling results are generally in good agreement with the measurements. The measured neutron dose rates were in the range from 14 to 132 μSv.h^{-1} (Gago et al., 1998).

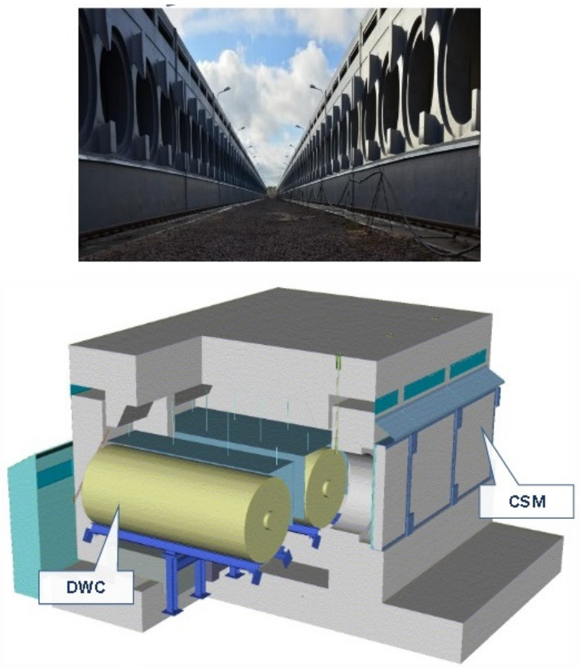

Figure 1: Concrete Storage Modules (CSM) (https://www.ornl.gov/division/rnsd/projects/spent-nuclear-fuel-storage-interim-storage-facilities).

Ebiwonjumi et al. examined the individual relationships between a fuel assembly decay heat and its burnup, enrichment, cooling time and uranium mass (Ebiwonjumi et al., 2021).

The effective multiplication factor or keff for a set of the spent fuels in the cask in particular circumstances, and probable situations should be investigated while ensuring the contents of the fuel remain intact. The circumstances include the followings:

- Leakage or penetration of water into or out of the cask,
- Decrease or damage of the neutron absorber material used in the cask structure (if any),
- Changes of the fuel assemblies gap because of an accident,
- Temperature changes, and so on.

In overall, the effective multiplication factor of cask for all types of fuels that contain a percentage of the enrichment with different burnups, as well as various materials available in the cask should not be greater than 0.95 regarding the worst incident conditions that may be occurring (Mennerdahl, 2012). The maximum dose value on the outer surface of the fuel cask, except for casks designed to carry in special arrangements by rail, sea transport or road transport, at no point in the outer surface of the cask shall not exceed 2 mSv.$^{-1}$ (IAEA., 2013). Pervious factors, i.e. gamma dose rate, neutron dose rate, deposited heat and effective multiplication should be calculated not only for the storage cask but for the storage areas too.

Different storage areas are designed for the dry spent fuel casks. Figure 1 shows the concrete storage area and its schematic view, which contains Double-Walled Canister (DWC) (https://www.nwmo.ca/en/Canadas-Plan/Canadas-Used-Nuclear-Fuel/How-Is-It-Stored-Today).

In addition, some interim storage facilities build big halls to storage the casks for 50 to 60 years (Fig. 2). Some of the underground cavities are used as an interim site by some countries to storage their casks (Fig. 3).

Figure 2: The cask storage hall at Canada (https://www.powermag.com).

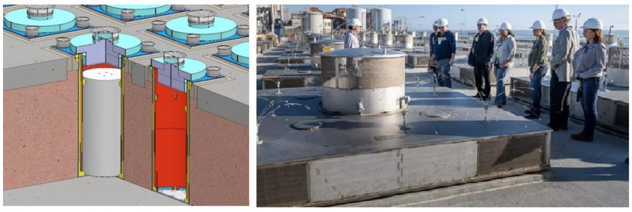

Figure 3: The cask storage hall at Mexico (https://www.powermag.com).

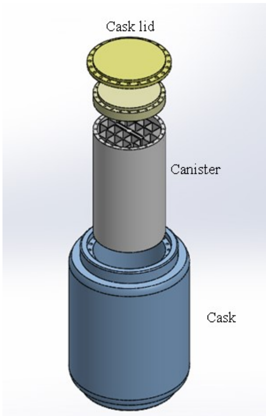

Figure 4: Schematic view of the canister and cask.

The hall dimension is different depends on the number of the casks are to be loaded inside the cask storage hall as well as the cask dimension. For example, a cask storage hall in Switzerland has 68 m long, 41 m wide and almost 20 m high. This hall has room for around 200 standing casks. The heat can always dissipate by the natural circulation of air through openings in the sidewalls of the hall and in the roof. Initially, the heat output of a steel casks filled with high-level radioactive waste can amount to between 40 and 50 kW. The heat output drops continuously and equals only around 25 to 30 kW after ten years in interim storage. The hot air released through the openings in the roof is not radioactive and does not have any negative impact on the environment (https://www.zwilag.ch/en/cask-storage-hall-_content---1--1054.html). Therefore, the present work aims to design a cask storage hall for the Tehran Research Reactor (TRR) spent fuel casks.

2 Materials and Methods

The plan of the supposed dual-purpose cask was given in Fig. 4. The cask height is 130 cm and its diameter is 97 cm.

The canister is used for the spent fuel loading and the spaced-grids inside it prevents any contact between the spent fuel assemblies.

In the present work, MCNPX2.6.0 code is used to calculate the gamma dose rates (Pelowitz et al., 2005; Fensin, 2008). The carbon-steel cask with a capacity of 16 spent fuels was modeled using the MCNPX code (Fig. 2). A hall with 10×12 m^2 length and widths dimensions and 5 m height was modeled to accommodate 25 casks. A Geiger Mller dosimeter was placed at the hottest point of the storage cask hall in front of its external surface. The detector level is in front of the spent fuel middle level. ORIGEN code was used to calculate the spent fuel assembly gamma emission rates for 55%- burnup and 10-years cooling time (Croff, 1980). The gamma source extracted from the ORIGEN code for Tehran Research Reactor (TRR) spent fuel assemblies. The gamma emission rates are to be used in MCNPX code input as the gamma source for the dose rate calculations. In these calculations, the thickness of the storage hall wall was optimized so that the outer surface dose rate (includes gamma and neutron) of the wall is less than 10 μSv.h^{-1} (Hamzah et al., 2019). DE/DF card and ANSI/ANS-6.1.1-1977 flux to dose conversion factors were used to calculate the gamma dose rates. The U-235 enriched UO_2 library on 204 205 206 for neutrons was used in this work, which is modified for a PWR with 33 GWd/MtU. The library is suitable to calculate gamma dose rate of the TRR spent fuel.

The neutron dose rate of the hottest point of the wall surface was calculated using LB6411 sphere. Mass fractions of the cask body and canister materials are presented in Table 1. The cask simulation view by MCNPX code is presented in Fig. 5. Delayed neutrons were extracted from ORIGEN code and were used as a neutron source in MCNPX code for delay neutron dose rate calculations. Flux to dose conversion factor of NCRP-38, ANSI/ANS-6.1.1-1977 was used to calculate the neutron dose rates.

Figure 6 shows the modeled hall, which the casks near to the dosimeter detector have about 10 cm interval of the internal sections of the concrete wall. The detector was placed in the center of the right wall of the hall and could be seen as a small point on the right wall of Fig. 6-b. Concrete wall was selected as ordinary concrete with a density of 2.2 g.cm^{-3}.

The gamma source distribution on the cask positions was modeled in MCNPX. The red color shows the gamma source distribution along the fuel assemblies (Fig. 7).

Table 1: Mass fraction and density of the cask body and canister components.

Material	Density (g.cm^{-3})	Weight fraction (%)	
Cask: Low Alloy Carbon Steel	7.7	Fe	98.31
		Mn	0.6
		C	0.4
		S	0.05
		P	0.04
		Si	0.72
Canister: Stainless Steel	8	Fe	70.17
		Cr	18.17
		Ni	9.25
		Mn	1.0
		Si	0.5
		C	0.04
		S	0.212

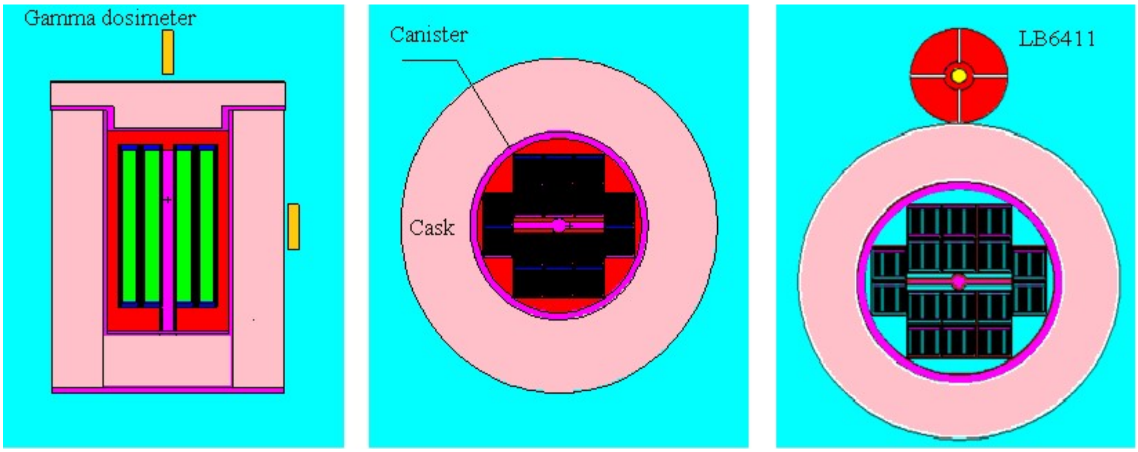

Figure 5: View of the simulated cask using MCNPX code.

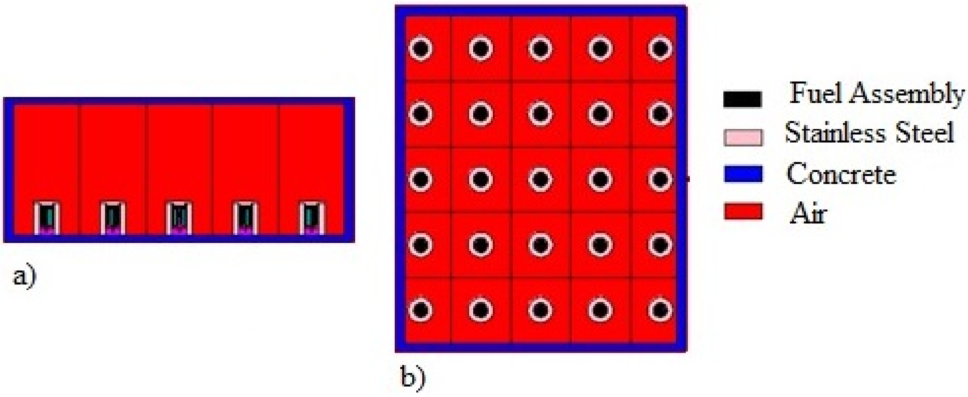

Figure 6: View of the simulated casks inside the storage hall: a) vertical view, b) cross-sectional view.

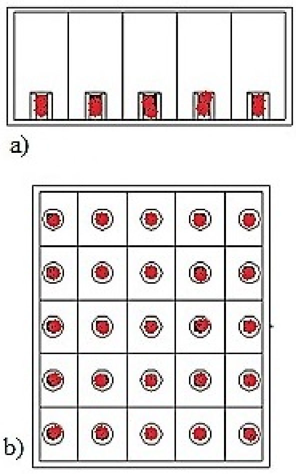

Figure 7: Distribution of gamma source in the a) vertical, and b) radial direction of the fuel assemblies inside the cask.

The effect of the number of the spent fuel casks on of the concrete wall external part gamma dose rate was investigated by changing the number of casks available inside the hall.

A Geiger Muller gamma dosimeter was placed near to the external part of the concrete wall and dxt card (the dashed lines around the detector) of MCNPX code was used as a variance reduction method for the calculations (Fig. 8).

The gamma and neutron deposited heats inside the hall was calculated using F6 card of MCNPX code. The gamma heat distribution inside the hall was calculated using the MESH tally card of MCNPX code. By consideration of water inside the hall volume, the effective multiplication of the hall was calculated using KCODE card of MCNPX. The transport history of particles inside the modeled geometry was chosen so that the calculation errors reach less than 5%.

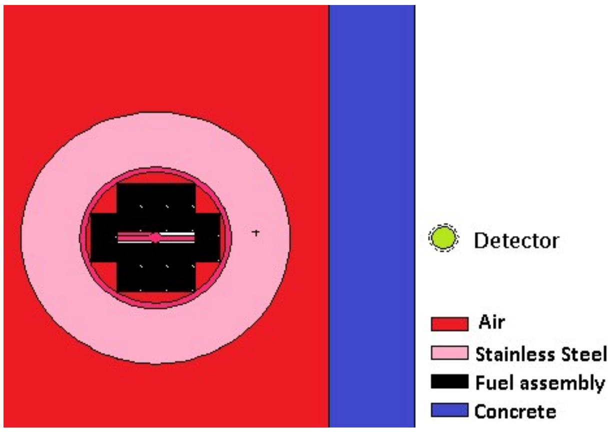

Figure 8: Detector location behind the concrete wall.

3 Results and Discussion

To calculate the gamma and neutron dose rates, the gamma spectra exited from the described fuel assembly was calculated using ORIGEN code. The spectra according to Fig. 9 was used in MCNPX code input as gamma source.

In general, only gammas with energies from approximately 0.8 to 2.5 MeV will contribute significantly to the dose rate through typical types of shielding for the spent fuel storage areas. The neutron source term for the spent nuclear fuel (SNF) may be specified in terms of the constituent radionuclides with their respective neutron yields and spectra. The SNF neutron source will generally result from both spontaneous fission and alpha-n reactions in the fuel. Depending on the method used to calculate these source terms, the applicant may need to define the energy group structure separately. This is often accomplished by selecting the nuclide with the largest contribution to spontaneous fission (e.g., Cm-244) and using that spectrum for all neutrons, since the contribution from alpha-neutron reactions is generally small. Especially for the SNFs of less than 5-years cooling, this is a satisfying method to calculate the neutron dose rate with this neutron source (Bozic et al., 1996).

Delayed neutron spectra of the 55%-burnup fuels was obtained from the MCNPX code via burnup card of the computational code (Fig. 11). The obtained spectra was used for the modeled cask to calculate the delayed neutron dose rates on the concrete hall external surface.

First, the concrete wall thickness was discussed by considering 25 casks inside the hall. The gamma dose rate of the external part of the concrete wall was calculated for different thicknesses of the ordinary concrete wall. The calculations showed that the maximum gamma dose rate changes from 82 nSv.h^{-1} to 2.6 nSv.h^{-1} for 10 to 30 cm thicknesses, respectively (Fig. 12).

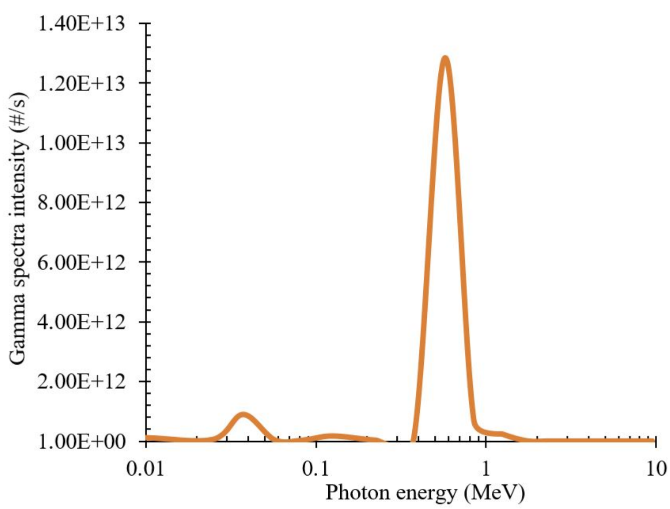

Figure 9: Gamma spectra of the 55%-burnup spent fuel with 10-years cooling.

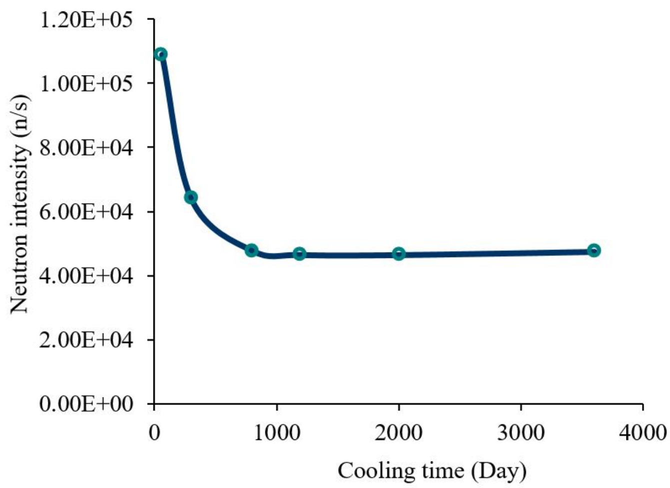

Figure 10: Change in the number of delayed neutrons emitted from the spent fuel by 55% burnup in terms of cooling time.

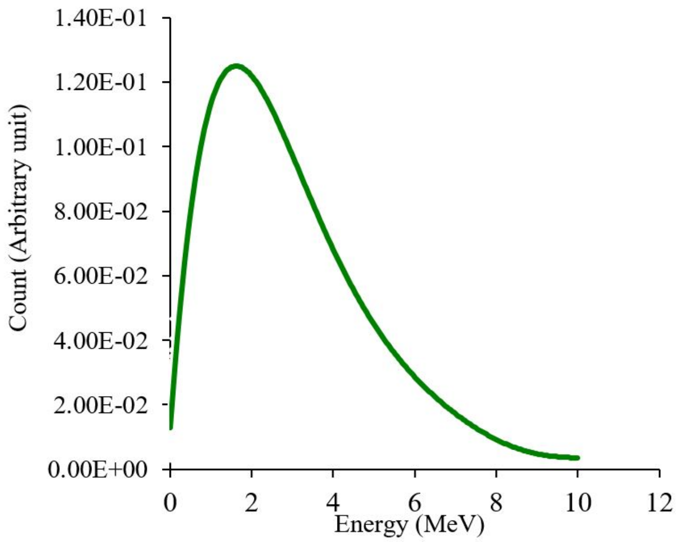

Figure 11: Delayed neutron spectra obtaied for the 55%-burnup spent fuels.

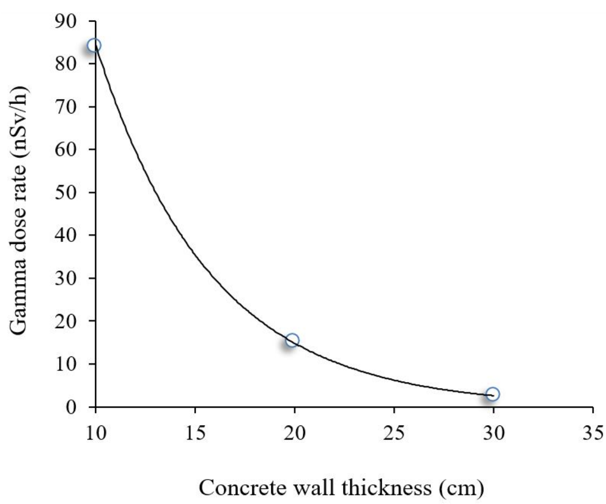

Figure 12: Gamma dose rate on the concrete wall thickness enhancement.

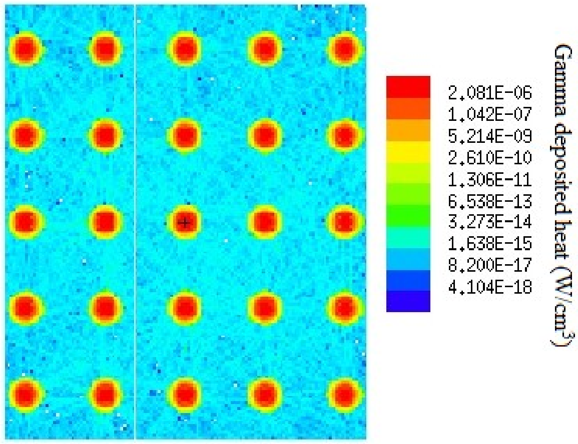

Figure 13: Gamma dose rate density distribution inside the hall.

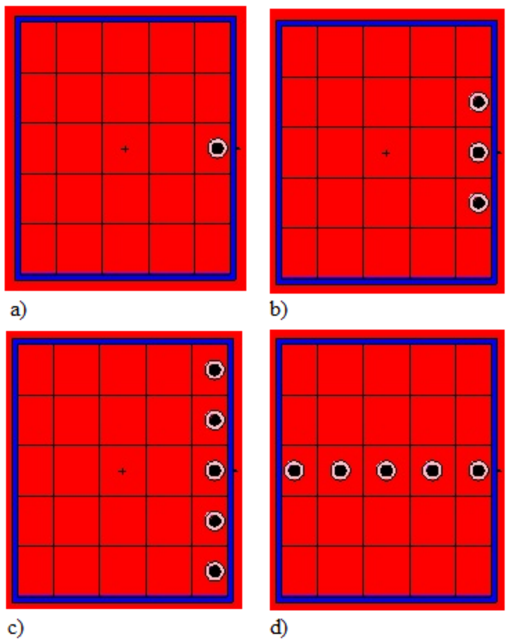

Figure 14: Gamma dose rate on the external surface of the hall, the number of casks inside the hall: a) one, b) three, c) five,and d) five casks aligned with the detector center.

The storage hall concrete wall thickness was considered as 30 cm and the following calculations were carried out by this situation.

The total deposited heat of the gamma rays calculated by F6 tally of MCNPX code inside the hall is about 38 W for loading 25 casks inside the hall (every cask contains 16 55%-burnup 10-years cooled fuel assemblies of TRR); neutron, gamma and secondary gamma were accounted in the deposited heat calculations. The gamma deposited heat distribution inside the hall is shown in Fig. 13. The highest deposited heat density is about 2.08E-06 W.cm^{-3} that is not significant which confirms no air-forced conditioner system installation is not mandatory for the hall. That means the natural circulation of air is enough to cope heat removal from the hall, as it was possible for the storage hall of Switzerland with total 50 kW deposited heat.

The total gamma, secondary gamma, and neutron deposited heats inside the hall are 25.64 W, 12.7 W, and 8.5 μW, respectively.

The effect of the number of casks loaded inside the hall on the maximum external wall gamma dose rate was investigated. The calculations were carried out for 1, 3, 5 casks beside the wall and (Fig. 14-a to Fig. 14-c) also 5 casks aligned in front of the detector (Fig. 14-d).

The calculations showed that by loading only 1 cask the maximum gamma dose rate of the external wall surface is 9.2 nSv.h^{-1} according to Fig. 14-a. By loading 3 casks according to Fig. 14-b the value is 5 nSv.h^{-1}. By the loading of 5 casks according to the Fig. 14-c the value is 3.49 nSv.h.h^{-1} and the configuration of Fig. 14-d would result in a gamma dose rate 3.45 nSv.h^{-1}.

The neutron dose rate was calculated using LB6411 sphere and the obtained result showed the maximum neutron dose rate is 1.16 μSv.h^{-1} on the external wall of the hall contained 25 casks. When only one cask is available inside the hall, the neutron dose rate after the 30 cm concrete wall is 1.87 μSv.h^{-1}. A benchmark study was done in the case of our available 8 cm-thick lead cask with a capacity of nine spent fuel loading. The neutron dose rate measured on the cask body, which involves 10-years cooled spent fuels, was 1.2 μSv.h^{-1} while the simulation data was 2 μSv.h^{-1}. Therefore, there is very fine conformity between the simulation and experimental data. So the procedures were used for calculation of the neutron dose rate inside the hall is reliable. The calculations showed a full hall would result less noticeably gamma dose rates after the concrete wall because the cask body materials could finely shield gamma rays from any source (cask that was filled with fuel assemblies). In addition, the same behavior is observed in the case of neutron dose rate but its reduction by shielding effect of the cask material is less than gamma. Clearly, the cask material is a powerful shield for gamma ray than neutron.

The modeled hall multiplication factor was calculated regarding any flooding is happening, which causes water fills completely the hall. The calculations showed the effective multiplication factor is 0.62229$\pm$0.00067 with this situation that is much farther from the limit value (0.95) (NRC, 2009). Hence, no critically accident would not happen with this situation.

4 Conclusion

Independent storage of spent fuel out of nuclear power plants is required for all the nuclear reactor sites. In the present work, we proposed to use a dual-purpose dry metal cask for transport and storage for this interim storage. Clearly spent fuel management requires technical solutions that allow secure, temporary dry storage of SNF and versatility to transport and retrieve it, until governments decide its long-term strategies and permanent installations start operating. Hence, the designed and modeled hall is a place for the safe disposal of casks contained the spent fuel elements from the Tehran research reactor. The carried out calculations showed that 30 cm-thick ordinary concrete wall would ensure total gamma and neutron dose rates of less than 10 μSv.h^{-1} on the external surface of the designed hall when it filled with the casks contained the 55%-burnup 10-years cooled SNFs. The neutron dose rates are less than 2 μSv.h^{-1} on the external wall surface; clearly it decreases with the storage time enhancement. The carried out calculations showed the deposited heat of the emitted gamma rays is low and the deposited heat of neutrons can be ignored, so that the hall does not need to any cooling system. In addition, if the hall is filled with water any criticality accident will not happen because the effective multiplication factor of the hall would be much less than the limit value of 0.95. Such calculations are mandatory for designing any short interim storage halls to ensure enough safety of a site personnel as well as public.

References

Baeg, C.-Y. and Cho, C.-H. (2018). Development status for commercialization of spent nuclear fuel transportation and dry storage system technology. *Journal of Nuclear Fuel Cycle and Waste Technology (JNFCWT)*, 16(2):271–279.

Bozic, H., Bace, M., and Grgic, D. (1996). Concrete spent fuel cask criticality calculation.

Croff, A. G. (1980). User's manual for the ORIGEN2 computer code. Technical report, Oak Ridge National Lab.

Dho, H.-S., Kim, T.-M., and Cho, C.-H. (2016). The evaluation of minimum cooling period for loading of PWR spent nuclear fuel of a dual purpose metal cask. *Journal of Nuclear Fuel Cycle and Waste Technology (JNFCWT)*, 14(4):411–422.

Diaz-Maurin, F., Yu, J., and Ewing, R. C. (2021). Sociotechnical multi-criteria evaluation of long-term spent nuclear fuel management strategies: A framework and method. *Science of The Total Environment*, 777:146086.

Ebiwonjumi, B., Cherezov, A., Dzianisau, S., et al. (2021). Machine learning of LWR spent nuclear fuel assembly decay heat measurements. *Nuclear Engineering and Technology.*

Fensin, M. L. (2008). *Development of the MCNPX depletion capability: A Monte Carlo linked depletion method that automates the coupling between MCNPX and CINDER90 for high fidelity burnup calculations.* University of Florida.

Gago, J., Grandal, R., Lapucci, A., et al. (1998). Design and licensing of a dual purpose cask in Spain. *International Journal of Radioactive Materials Transport*, 9(1):25–34.

Garcia, J. (2014). Dual purpose casks in operation. *Workshop on the Development and Application of a Safety Case for Dual Purpose Casks for Spent Nuclear Fuel IAEA headquarter.*

Hamzah, A., Adrial, H., et al. (2019). Preliminary analysis of dose rates distribution of experimental power reactor 10 MW using MCNP. In *Journal of Physics: Conference Series*, volume 1198, page 022038. IOP Publishing.

IAEA. (2013). *REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL: 2005 Edition.* International ATOMIC ENERGY AGENCY.

Jeong, G.-E., Choi, W.-S., and Cho, S. S. (2021). Topology optimization of tie-down structure for transportation of metal cask containing spent nuclear fuel. *Nuclear Engineering and Technology.*

Ko, J.-H., Park, J.-H., Jung, I.-S., et al. (2014). Shielding analysis of dual purpose casks for spent nuclear fuel under normal storage conditions. *Nuclear engineering and Technology*, 46(4):547–556.

Mennerdahl, D. (2012). Review of the nuclear criticality safety of SKBs licensing application for a spent nuclear fuel repository in Sweden. *Technical Note-2012*, 65.

Narkūnas, E., Šmaižys, A., Poškas, P., et al. (2021). Neutron dose rate analysis of the new CONSTOR® storage cask for the RBMK-1500 spent nuclear fuel. *Nuclear Engineering and Technology*, 53(6):1869–1877.

NRC, U. (2009). Standard review plan for dry cask storage systems, NUREG-1536.

Palacio, A., Garrido, D., Castrillón, D., et al. (2014). High radioactive spent fuel management. Dual purpose metal casks as a current optimal solution. *Proceedings for the 8th International Youth Nuclear Congress.*

Pelowitz, D. B. et al. (2005). MCNPXTM users manual. *Los Alamos National Laboratory, Los Alamos.*