RPE Radiation Physics & Engineering
A Peer-Reviewed Journal
Published by K.N. Toosi University of Technology
http://rpe.kntu.ac.ir
Research Article

Radiation Physics and Engineering 2026; ?(?):?–?

Investigation of required modifications of the manufactured TRR DPC for transport and storage of two-year cooled spent fuel assemblies

Behrooz Rokrok, Zohreh Gholamzadeh*, Atiyeh Jozvaziri, Ebrahim Abedi
Reactor and Nuclear Safety School, Nuclear Science and Technology Research Institute, Tehran, Iran

HIGHLIGHTS

- Dual-purpose cask (DPC) is used for transporting and interim storage of spent fuel assemblies.
- Computational methods help to estimate gamma dose rates before high cost design and constructions.
- Gamma dose rates should be calculated to optimum the cask shield material thickness.
- Measurements help to benchmark the carried out simulations.

ABSTRACT

Dual-purpose cask (DPC) is used for transporting and interim storage of spent fuel assemblies, since such casks are an attractive option due to their flexibility and economy efficiency. In point of economical view, construction of high-capacity DPC is more suitable. The cask could be used to transport the spent fuel assemblies of long cooling time with its full capacity. At emergency situations the same cask potentially could be used to transport the spent fuels with short cooling time using some modifications inside the canister. The present study would investigate the highest capacity of Tehran Research Reactor (TRR) DPC for transport and storage of 2-year cooled spent fuel assemblies. MCNPX2.7.0 computational code was used to calculate the DPC body maximum gamma and neutron dose rates. The obtained results showed that maximum six 2-year cooled SFAs of TRR could be transported by the present DPC. Moreover, filling the empty places of canister with carbon-steel shield blocks, some modifications on the cask door and floor is needed to pass the determined gamma dose rate limit (2 mSv.h^{-1}). By adding 6 cm thick carbon-steel on the DPC door and its bottom the goal is obtained while the modifications increase the cask weight about 1 ton.

KEYWORDS
Dry storage
Nuclear spent fuel
Short cooling time
DPC cask
MCNP simulation

HISTORY
Received:
Revised:
Accepted:
Published:

1 Introduction

Spent fuel storage is an important issue for all the nuclear reactor sites. A range of terms are used to describe the options available for locating spent fuel storage facilities as the following:

- AR (At-Reactor) storage: a pool co-located with the reactor, inside the containment building.
- AFR (Away-from-reactor) storage: a wet or dry storage facility that is not co-located with the reactor. The fuel has to be transferred or transported to the storage facility. There are two classifications of AFR storage facilities: reactor site (RS) and off site (OS):
  - AFR-RS storage occurs in a facility located within the reactor site boundary. Spent fuel is transferred from one facility to the other. A further distinction can be made for an AFR-RS in terms of those that stand alone and can still support operations if the reactor is decommissioned, and those which are reliant on reactor services.
  - AFR-OS storage involves a facility located outside the reactor site boundary. In this case, spent fuel is transported on public roads (IAEA, 2024).

Spent fuels are stored as wet or dry in the above-mentioned storage sites. Newly dry storage of spent fuel assemblies has been taken in attention because of some attractions. Among various existing dry storage concepts, several Member States are utilizing a concept of dual-

---

*Corresponding author: zgholamzadeh@aeoi.org.ir

https://doi.org/

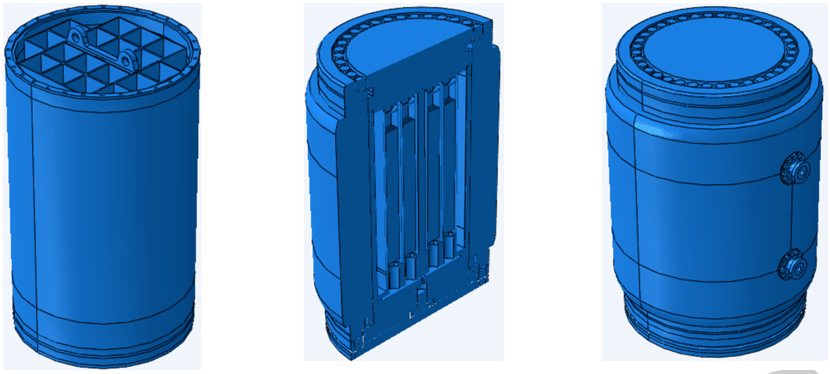

Figure 1: Graphical schematic of the designed TRR DPC.

purpose casks (DPCs). The primary safety objectives of a DPC design relate to national storage regulations and compliance with the transport regulations extant at the time of transport. DPCs are generally designed with a dual containment boundary and are designed and maintained so the primary lid need not be opened for inspection or maintenance during storage or before transport after storage to avoid unnecessary degradation, incidental risks, and radiological exposures (Droste et al., 2020). Also, DPCs has been designed and built by Tehran Research Reactor (TRR) as is shown in Figs. 1 and 2 to store its long-time cooled spent fuels (more than 10 years). The present study aims to investigate such casks potential for transport or storage of short-time cooled TRR spent fuel assemblies (SFAs) (Abedi et al., 2021).

TRR canister material is Stainless Steel ASTM A182 F304, the body material is Low Alloy Steel 1.7225 (ASTM A29-AISI 4140), the TRR DPC weight is 6.25 ton, its height is 1.96 m and its outer diameter is 0.972 m as is shown in Fig. 2 (Abedi et al., 2021).

Figure 2: The constructed TRR DPC photo.

It should be mentioned that dry storage casks, designed to store spent fuel assemblies with a cooling period of only 2 years, pose specific engineering challenges because the decay heat and radioactivity of these fuels are significantly higher than those typically stored in fuel pools for 5 to 10 years. The most important factor in the design of these packages is the ability to dissipate residual heat. Fuels that have only been cooled for two years will produce more heat. Packages must have highly effective passive heat transfer systems. This usually involves the use of large external fins, high thermal conductivity materials, or inert gases (such as helium) inside the container to optimize heat transfer to the outer shell and the surrounding environment. The design must ensure that the fuel cladding temperature does not exceed a certain limit to prevent damage and the release of radioactive material.

A cladding temperature limit of 200 °C ensures that creep or that diffusion, the next most limiting of the degradation mechanisms, do not lead to unacceptable degradation in 50 years of storage (Sindelar et al., 1994).

Due to the higher radioactivity of the newer fuel, these packages require thicker and stronger shielding, especially to absorb gamma rays. DPCs usually made of thick steel, or lead. The design must ensure that the radiation dose at the outer surface of the package complies with international and national nuclear safety regulations.

The dry storage casks designed for the 2-year cooled SFA are structurally very strong, have advanced thermal management, and are much thicker in shielding. These packages represent a more advanced generation of dry storage technology that allows for faster removal of fuel from water pools to preserve operating space at nuclear facilities. These designs are more expensive and more engineering complex than traditional casks, but they are essential to maintaining flexibility in the nuclear fuel cycle (SRIS, 2025; Guide, 2018; Vienna, 2024).

As before mentioned the most important reason for rapid transfer of fuel to dry storage is to maintain sufficient operational space in the plants spent fuel pool. Nuclear safety regulations require that operators be able to quickly evacuate the entire reactor contents (the entire active fuel assembly) into the pool in the event of an emergency or for periodic inspections. If the pool is full, this is not possible. In many older nuclear power plants, existing pools are filling up. By transferring 2-year-old fuel to dry storage casks, the necessary space for future refueling cycles is ensured and the continuous operation of the plant is maintained without interruption. While fuel pools are the safest place for initial cooling, dry storage offers unique passive safety benefits, especially for long-term storage. Pools require active systems (pumps, filters, backup power) to maintain water levels and cooling. Failure of these systems can lead to water boiling, lowering the

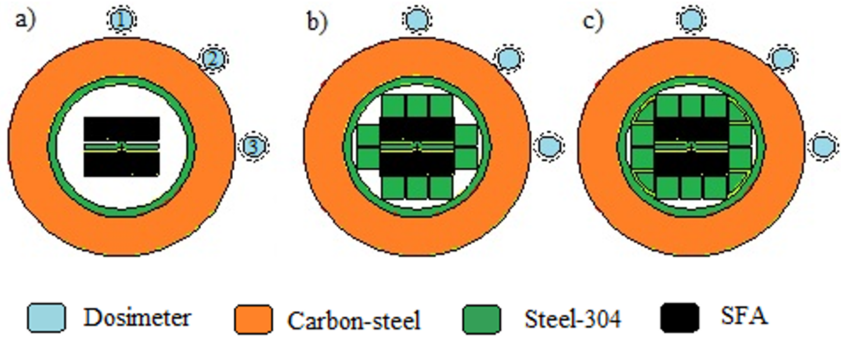

Figure 3: Simulation of TRR DPC filled with 50%-burnup 2-year cooled TRR SPAs a) 6 SFAs loading b) 10 additional carbon-steel shield blocks c) 4 additional wedge-like shield blocks.

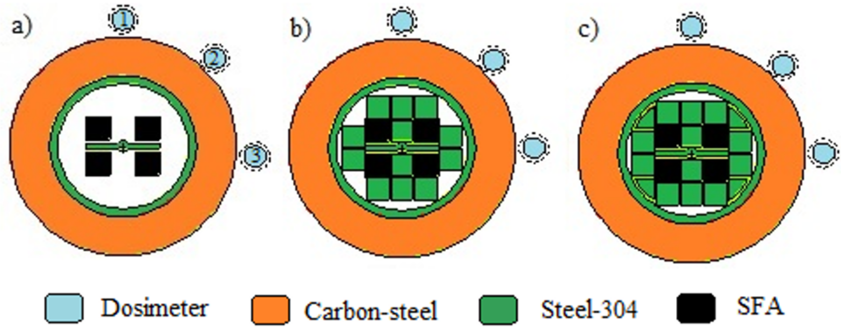

Figure 4: Simulation of TRR DPC filled with 50%-burnup 2-year cooled TRR SFAs a) 4 SFAs loading b) 10 additional carbon-steel shield blocks c) 4 additional wedge-like shield blocks.

water level, and ultimately damaging the fuel. While dry packages are passive and their cooling process is based on natural heat transfer (conduction and convection) in an inert gas chamber (usually helium) and does not require any external systems. Newly dry storage casks are constructed of thick steel and/or concrete walls that provide high protection against severe external events (such as earthquakes, floods, or deliberate attacks). They act as a very strong layer of defense, protecting the fuel from the environment and external factors (IAEA, 2020; NRC, 2025).

In general, to prevent fuel failure, the fuel cladding temperature inside the dry storage cask should be less than 200 °C and the pressure inside the cask should be less than 90 MPa. The maximum gamma dose rate on the transport cask surface should be less than 2 $mSv.h^{-1}$ (USNRC, 2003; Kook et al., 2013).

This study investigates the feasibility of utilizing a cask designed for the dry storage of spent fuel with a cooling time exceeding 10 years for the transport and storage of fuel cooled for only 2 years, while adhering to existing transport cask regulations as the above-mentioned.

2 Materials and Methods

The DPC of TRR has been simulated using MCNPX2.7.0 computational code. MCNPX2.7.0 code is powerfully used to model different nuclear facilities such as nuclear sources, nuclear reactors, nuclear accelerators in details while could transport more than 34 particles including neutron, photon, electron, deuteron etc. (Pelowitz et al., 2008).

Initially, the loading of six 2yearcooled spent fuel assemblies (SFAs) from the TRR into the cask was analyzed, and the gamma dose rate on the external surface of the cask was determined using the DE/DF cards in the computational code. The fluxtodose conversion at various energy groups was performed using ICRP21 coefficients. The gamma spectrum for a 50% burnup, 2yearcooled TRR SFA was generated using the ORIGEN2.1 code (Croff et al., 1980). The burnup value means 50% of initial 235U content of the fresh fuel has been consumed at the end of the fuel cycle in TRR. The TRR fuel assembly isotopic format for ORIGEN2.1 code input is according to Table 1.

0.28 MW value was used as the third term of IRP card of ORIGEN2.1 code implying the highest fuel assembly receiving power. The continuous history was used to obtain a 50% burnup which is equivalent with 450 days TRR operation at this power. 204, 205 and 206 neutron libraries were used which are related with PWR type reactors such as TRR, VVER, etc. Decay term consideration was used as the following after the BUP term of the computational code:

DEC 2 1 2 5 2 so that the first term shows the irradiation time and the second and third terms show the ORIGEN2.1 vector numbers and the fourth term shows the irradiation time unit. The final term shows the starting calculation time. The user could be used the gamma intensity of fission products (reported in ORIGEN2.1 code output) for the gamma dose rate calculations using MCNPX2.7.0 code whereas their values are noticeably overes-

Table 1: TRR fuel content format for ORIGEN code input.

Code	Isotope	Mass g	Isotope	Mass g	Format	Format
2	922350	290.07	922380	11622.8	0	0
4	80160	261.1	130270	621.04	0	0
4	130000	0.727	200000	0.363	0	0
4	290000	0.029	150000	0.145	0	0
4	140000	0.363	110000	0.363	0	0
4	560000	0.015	500000	0.003	0	0
4	480000	0.0007	270000	0.004	0	0
4	300000	0.007	120000	0.145	0	0
4	250000	0.007	190000	0.029	0	0
4	230000	0.007	0	0		

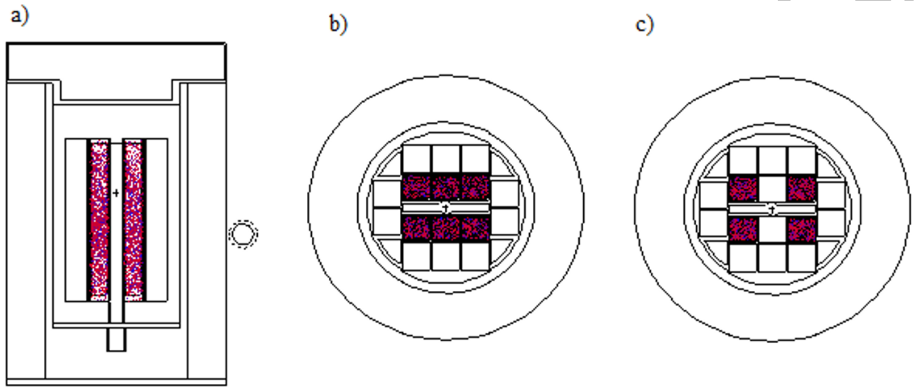

Figure 5: Gamma source distribution on SFAs loaded inside the modeled DPC a) Axial view b) 6 SPAs loading cross sectional view c) 6 SFAs loading cross sectional view.

timated than the emitted gammas from activation products as well as the actinides.

The gamma dose rate was calculated with detectors numbered as 1, 2 and 3, which are equivalent to the exact sensitive volume of the LB1234 dosimeter. Subsequently, the effect of additional shielding on reducing the dose rate at the hottest point of the cask was examined. This point is located on the lateral surface of the cask, directly opposite the midpoint of the SFAs' axial length (Fig. 3). Furthermore, the gamma dose rate was analyzed for the configuration shown in Fig. 3-a, under the assumption that the interior of the cask was filled with water.

Some positions of the above modeled cask could be filled using carbon-steel blocks to provide more shield around the 6 loaded SFAs as the following:

It is estimated that the empty corners also would need to some wedge-like shields. So, the calculations were repeated when four wedge-like blocks were used at the corners according to Fig. 3-c.

As illustrated in Fig. 4, detectors No. 1, 2, and 3 were positioned at the hottest point on the cask body, corresponding to the location opposite the axial midpoint of the spent fuel. The calculations were carried out for 4 SFAs loading scenario as well. The simulations were conducted in three conditions as the above mentioned for 6 SFAs loading scenario (Fig. 4).

In this configuration, the twelve vacant positions were filled with carbonsteel blocks to enhance the shielding around the four loaded SFAs, as illustrated in Fig. 4.

Again, the canister corner filling with wedge-like blocks was investigated as Fig. 4-c shows. Figure 5 shows the gamma source distribution on the SFAs loaded inside the DPC.

To calculate the neutron dose rate on the TRR DPC cask containing six 2-years-cooled SFAs, the LB6411 neutron dosimeter was simulated according to Fig. 6. The neutron intensity from both (α, n) and spontaneous fission was calculated using ORIGEN2.1 code for the 2-years cooled SFA. 244Cm spectrum was used as the most prominent spontaneous fission emitter content of the SFA. ICRP-21 neutron and gamma flux-to-dose rate conversion factors were used to calculate the dose rates using F4 tally and DE/DF cards of MCNPX2.7.0 code.

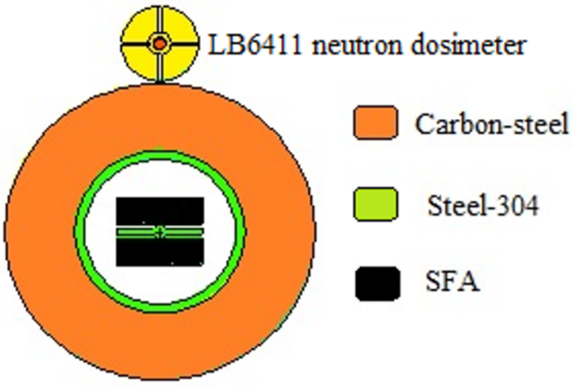

Figure 6: Cross sectional view of MCNPX2.7.0 simulation for the neutron dose rate calculation on the modeled DPC surface containing 6 SFAs.

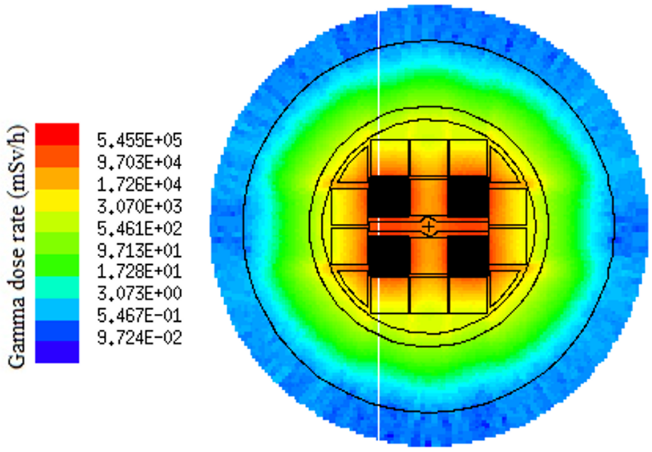

Figure 7: Cross sectional view of gamma dose rate distribution around the DPC cask filled with 4 TRR SPAs and shield blocks.

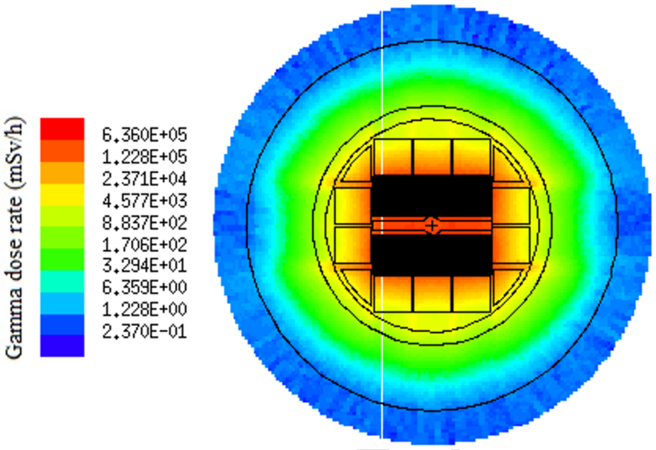

Figure 8: Cross sectional view of gamma dose rate distribution around the DPC cask filled with 6 TRR SPAs and shield blocks.

Table 2: Intensity of gamma rays emitted from TRR SFA with 50% burnup and 2 years cooling.

Energy (MeV)	Intensity (#/s)
1.000E-02	5.38E+12
2.500E-02	1.48E+12
3.750E-02	6.99E+12
5.750E-02	1.16E+12
8.500E-02	1.84E+12
1.250E-01	6.15E+12
2.250E-01	7.51E+11
3.750E-01	6.02E+11
5.750E-01	2.59E+13
8.500E-01	7.13E+12
1.250E+00	1.08E+12
1.750E+00	4.14E+10
2.250E+00	3.63E+11
2.750E+00	1.35E+09
3.500E+00	1.74E+08
5.000E+00	2.31E-07
7.000E+00	1.50E-08
9.500E+00	9.47E-10
Total	5.89E+13

The worst-case condition, in which no carbonsteel block shielding was applied, was considered for calculating the neutron dose rate. This scenario represents the maximum possible neutron dose rate on the lateral surface of the DPC cask.

3 Results and Discussion

The gamma spectrum of the 50%burnup, 2yearcooled TRR spent fuel assemblies (SFAs), calculated using the ORIGEN2.1 code, is presented in Table 2. This spectrum was subsequently used to determine the maximum gamma dose rates on the external surface of the cask.

The calculations performed indicate that if the empty positions within the canister are not filled with carbonsteel blocks, the maximum gamma dose rates on the cask body significantly exceed the established limits for the transport and storage of such casks. The filling water inside the cask also would not decrease the values to less than the determined limit gamma dose rate while its presence inside the cask decreased the gamma dose rate values about 50%. It should be noted the water HLV for 1 MeV and 100 keV gamma rays are about 4 cm and 10 cm. Average gamma intensity of the 2-year cooled SFAs is 35 keV having a HVL about 2 cm. The obtained results showed however water filling inside the cask reduced the gamma dose rate noticeably but it was not concluded in the limit value which is mandatory according to the regulatory rules for transportation of such cask. The utilization of both shield blocks and wedge-like blocks contributes to a noticeable reduction in gamma dose rates (Table 3).

If the cask is to be loaded with six fuel assemblies, it is imperative to reinforce both the cask door and floor with a minimum of 6 cm thick carbon-steel. This configuration would result in a gamma dose rate of 1.27 mSv.h^{-1} at the cask door. Furthermore, the implementation of the third item as detailed in Table 2 is required. This scenario would necessitate approximately 400 kg of internal steel blocks, and the additional door and floor would weigh approximately 600 kg. Consequently, the total weight of the cask would increase by approximately one ton.

The carried-out calculation for the four SFAs loading scenario shows if the empty positions of the canister are not filled with carbon-steel blocks, maximum gamma dose rates of the cask body are extremely higher than the determined limit value again. The filling water inside the cask also would not decrease the values to less than 2 mSv.h^{-1} but its presence inside the cask resulted in 50% gamma dose rate reduction than dry cask involved the SAFs. The utilization of both shield blocks and wedge-like blocks contributes to a noticeable reduction in gamma dose rates (Table 4).

For the cask loaded with four fuel assemblies, it is necessary to reinforce both the cask door and floor with a minimum of 4 cm thick carbon-steel. This configuration yields a gamma dose rate of 0.89 mSv.h^{-1} at the cask door. Furthermore, implementation of the third shielding configuration detailed in Table 3 is required. In this scenario, the internal steel blocks contribute approximately 450 kg to the overall weight, while the additional door and

Table 3: Gamma dose rate of the cask body in the different modeled situations on mSv.h^{-1}.

Detector No.	1	2	3
6 SPAs without carbon-steel blocks	19.87	20.93	20.55
6 SPAs with carbon-steel blocks	1.380	7.380	2.033
6 SPAs with 10 carbon-steel and 4 wedge-like blocks	1.470	1.904	1.406
6 SPAs without carbon-steel blocks water presence inside the cask	9.880	9.290	9.060

Table 4: Gamma dose rate of the cask body in the different modeled situations on mSv.h^{-1}.

Detector No.	1	2	3
4 SPAs without carbon-steel blocks	17.31	17.25	12.72
4 SPAs with carbon-steel blocks	1.170	5.580	1.040
4 SPAs with 10 carbon-steel and 4 wedge-like blocks	1.065	1.720	1.019
4 SPAs without carbon-steel blocks water presence inside the cask	8.640	8.020	6.290

floor add approximately 400 kg. Consequently, the total weight of the cask increases by approximately 0.85 tons. The maximum cross sectional gamma dose rate distribution around the cask, resulting from the third shielding configuration for 2-year-cooled SFAs, is illustrated in Figs. 7 and 8.

The gamma dose rate distributions show using the shield blocks caused the gamma dose rate is kept under the determined level at the DPC external surface. To validate the simulation procedure employed in this study, a benchmark study was conducted. This involved practical dosimetry measurements of a single TRR SFA, and a comparison of this experimental result with simulation data. The test conditions were modeled according to the configuration depicted in Fig. 9. Fuel dose rate calculations are performed by MCNPX2.7.0 and ORIGEN2.1 codes. In order to verify the calculations by these codes, the NRF027 (Logbook code) fuel dose rate was measured so that the SFA top edge was placed at a distance of 90 cm underwater. UMO Lb1234 dosimeter (proportional counter with measuring unit of H*(10) and measuring range of 50 nSv.h^{-1} to 10 mSv.h^{-1}, energy range 30 keV to 1.3 MeV has a 2.25 cm radius and 18.5 cm active length) was used to measure the gamma dose rate. The specifications of the NRF027 fuel assembly are presented in Table 5. Since, the number of assemblies in the core was 28 and the average power of each assembly was about 0.28 MW; the value was used for IPR card of ORIGEN2.1 code to calculate the gamma source rate of the SFA. As can be seen, the relative difference between the calculated dose rate and the measured dose rate is about 15%. The number of NPS used in the MCNPX2.7.0 code was 2×10^9.

Table 5: Benchmark study of the measured and simulated gamma dose rate of a 45% burnup TRR SFA.

Parameter	Values
MWh	2479
Burnup	45
Exit date from core	2024/03/06
Dosimetry date	2025/07/20
Cooling time	501
Measured dose rate (mSv/h)	1.58
Calculated dose rate (mSv/h)	1.83

For cask design or for evaluating the loading of spent fuel assemblies (SFAs) within an existing cask, the worstcase condition should be modeled - that is, a scenario involving the loading of SFAs with the highest burnup. In the case of the TRR, the highest SFA burnup is 50%. No fuel assemblies with identical burnup and cooling time were available in the second pool of the TRR reactor building, since the spent fuel assemblies are typically transported out of the building after several months of decay. Hence, the benchmark study was done using the available SFA according to Table 5.

It should be mentioned the MCNPX2.7.0 computational error was less than 9% using 2×10^9 particle transport and DXT variance reduction for the above modeled SFA. Precise history modeling is effective on reduction of the uncertainties while it is not easy practically whereas a fuel assembly is placed at different positions of the TRR core during its depletion. There is not any geometry tolerance because the SFA has been placed on a fixed stand so the precise intervals and geometry has been modeled using MCNPX2.7.0 code. ORIGEN2.1 code uncertainties is overshadowed by its cross section libraries which are not far from the real values in the case of uranium and other fissionable elements as they were measured several times by different researchers (see Exfor site). Hence, the 15% relative discrepancy reported in this benchmark study mainly arises from different factors such as the MCNPX2.7.0 code computational statistical errors, failure to precise history modeling of the SFA.

Neutron dose rate of the DPC cask containing six 2-year-cooled TRR SFAs was calculated using MCNPX2.7.0 code at the hottest point of the cask surface. The value was 0.21 μSv.h^{-1} with 5% of the statistical errors. All the computation errors were less than 13% for the above carried out simulations. DXT variance reduction method was used to decrease the computation errors.

To investigate the SFAs clad temperature, COMSOL Multi-physics 6.2 is used for conducting the finite element analysis. COMSOL Multi-physics version 6.2 introduces game-changing functionality for simulation apps and digital twins as well as faster solver technology (COMSOL, 1998). Heat Transfer in Solids and Fluids (ht) and Surface-to-Surface Radiation (rad) are the applied physics with a single Stationary study step. An AMG solver using

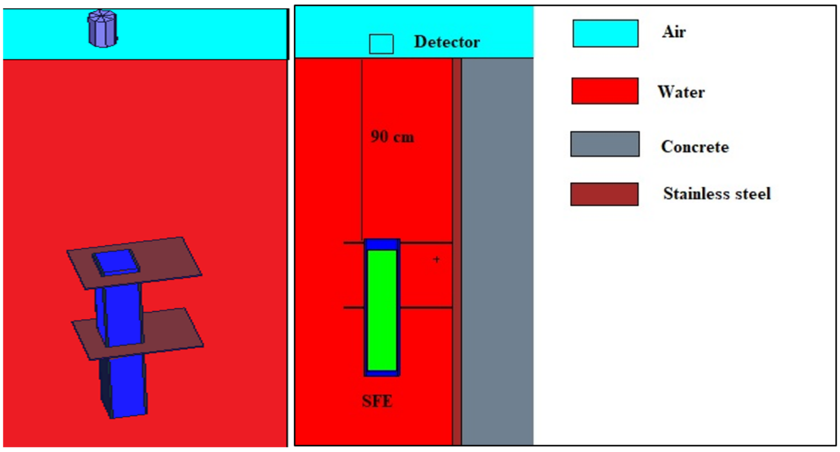

Figure 9: Axial view of the carried out simulation of a TRR SFA dosimetry condition using MCNPX2.7.0 code.

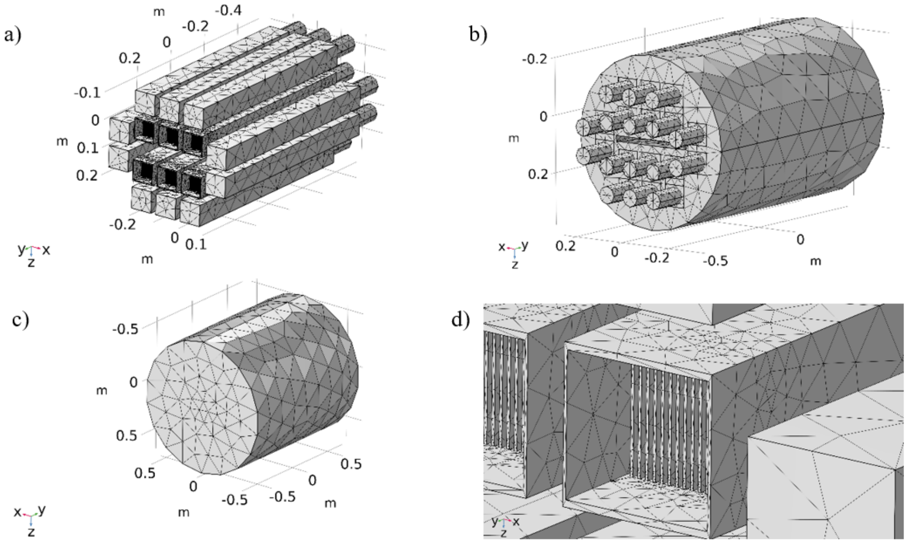

Figure 10: Mesh convergence metric, a) 6 SFAs loading, 10 carbon-steel blocks, b) cask body view showing SFAs end fittings, c) cask body view, d) SFA view containing 19 fuel plates.

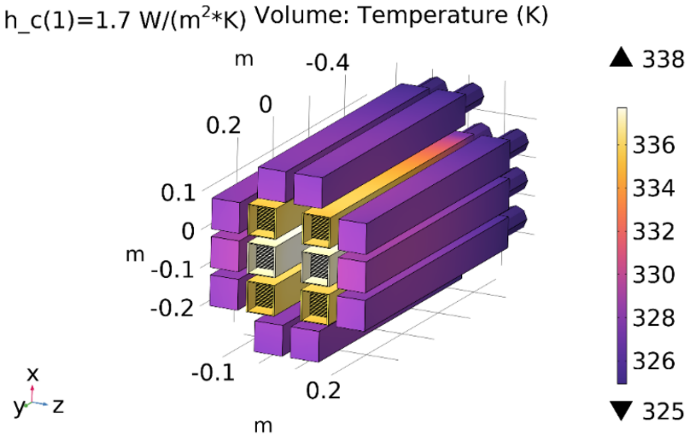

Figure 11: Temperature field visualization for the fuel assemblies in the center of the configuration, which have the maximum temperature of 338 K.

the GMRES algorithm is adopted to solve the finite element problem. A relative tolerance of 0.001 is the default convergence criterion for the solver. Linear tetrahedral elements are used to discretize the geometry. An initial mesh with 439912 elements is used to discretize the geometry. A schematic of the meshed region is shown in

Fig. 10. A reference temperature of 273.15 + 45 (K) is considered for all simulations. 45 K has been assumed for summer temperatures as the conservative value. An absolute pressure of 1 (atm) is further imposed in the Fluid node. The calculations were done for the DPC cask containing six 2-year-cooled SFAs. In the same node, zero velocity field is assumed for the helium gas, suggesting the total discarding of convection in heat transfer modes between the fuel assemblies and their immediate environment. A volumetric Heat Source node is also defined in COMSOL to account for the nuclear-based heat generation inside each fuel plate. All such fuel plates and their corresponding boxes are assigned to be part of the heat generation domain. In total, a heat rate of 121.2 (W) is considered.

The carried out calculations showed that the maximum SFA surface temperature is around 338 K, taking place at the two fuel assemblies at the center of the cask. A detailed plot of the fuel assemblies where the maximum temperature takes place is illustrated in Fig. 11.

As observed in the aforementioned figure, the fuel assemblies in the center have the maximum temperature because they have the least access to cask's surface to transfer heat out of the assembly. The assemblies in the corner have two lateral surfaces of their boxes in proximity of cask's exterior surface, resulting in better heat removal and lower temperature throughout the fuel plates and boxes.

The effect of convective heat transfer coefficient was the only parameter that was studied in this work too. Better insight would be gained by plotting the value of maximum temperature in the model for various values of the convective heat transfer coefficient, hc. This is shown in Fig. 12.

Eventually, the cask's surface temperature is also depicted in Fig. 13, as a measure of effective heat removal from the fuel assemblies into the environment.

As observed in Fig. 13, the bottom part of the cask has the peak temperature throughout the cask's surface, as a result of its closer distance to the fuel assemblies. The maximum temperature appearing on the cask's surface for $h_c = 1.7$ (W.m^{-2}.K^{-1}) equals 325 K.

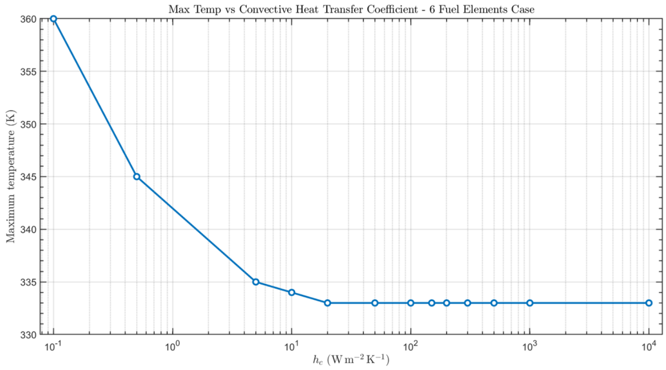

Figure 12: Maximum temperature in the model assembly for various values of h_c.

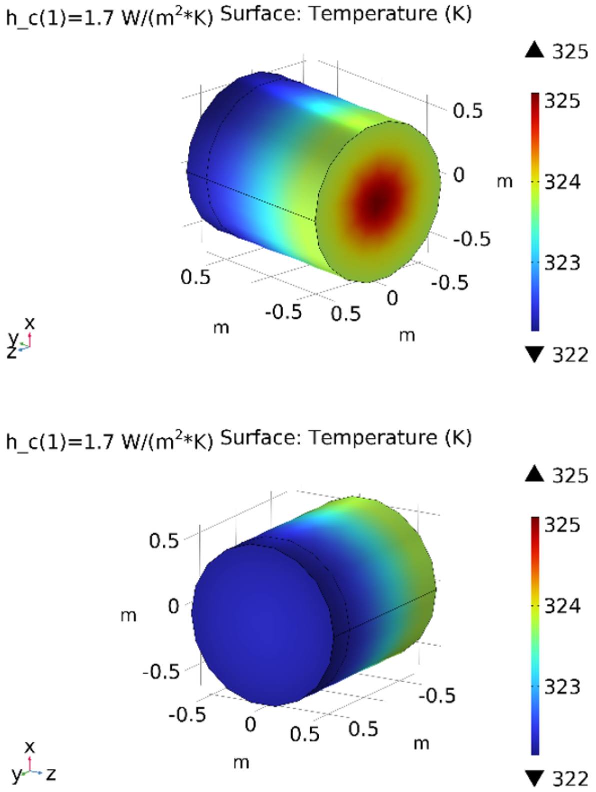

Figure 13: Temperature field for $h_c = 1.7$ (W.m^{-2}.K^{-1}) depicted for the cask and its lid; the maximum temperature.

4 Conclusions

The Dual-Purpose Cask (DPC) is an advanced storage and transportation system for spent nuclear fuel. The basic design philosophy of the DPC is that a single container can perform both critical functions: long-term dry storage and safe transportation. SFAs are initially stored for a period of time (several years) in cooling pools (wet storage) at the power plant site. After they have cooled down, they must be managed in one of two ways: long-term storage at the power plant site or at a central storage site or final disposal. The DPC simplifies this process. Instead of using one container for storage and then transferring the fuel to another container for transport, nuclear fuel is loaded and sealed into the DPC once, where it can remain for decades, and then used directly for transport. The present study applied simulation method to estimate the highest potential of the TRR DPC for storage and transport of 2-year cooled SFAs. The results indicate the feasibility of loading six spent fuel assemblies (SFAs) provided that the cask door and floor are reinforced with 6 cm of carbon-steel. Furthermore, carbon-steel blocks are required for shielding around short-cooled SFAs. These modifications result in an approximate weight increase of 1 ton for the cask. The thermal conductivity calculations showed the 2-year cooled TRR SFAs loaded inside the DPC cask experience the highest temperature of 338 K or 65 °C which is noticeably far from softening and deformation degree for aluminum material (400 °C). The cask body highest temperature for this situation would be 325 K or 52 °C. The carried-out benchmark study confirms the simulation result accuracy in this work. The innovation of this work lies in demonstrating that casks, which are typically designed and manufactured at high cost, can also be effectively utilized for transporting lowcooling fuels during emergency situations. In addition, the benchmark study conducted in this research is entirely new and has not been reported in any of the previously mentioned works or other available online references. Therefore, the benchmark study for SFA modeling represents another key innovation of this work, as such an activity is essential to validate the correctness of the modeled gamma source distribution within the SFAs.

Conflict of Interest

The authors declare no potential conflict of interest regarding the publication of this work.

Funding

The authors declare that no funds, grants, or other financial support were received during the preparation of this manuscript.

References

Abedi, E., Gholamzadeh, Z., and Vaziri, A. (2021). Development of mid and Long Term Strategies for TRR Spent Fuels Dry Storage. In *International Conference on Radioactive Waste Management: Solutions for a Sustainable Future. Book of Abstracts*, number IAEA-CN–294, pages 105–105.

COMSOL (1998). Introduction to comsol multiphysics®. *COMSOL Multiphysics, Burlington, MA, accessed Feb*, 9(2018):32.

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

Droste, B., Wolff, D., Bevilacqua, A., et al. (2020). Methodology for a Safety Case of a Dual Purpose Cask for Storage and Transport of Spent Fuel. In *IAEA TECDOC SERIES*, pages 1–109. IAEA Publishing Section.

Guide (2018). Regulatory Guide 3.71 (Dry Storage of Spent Nuclear Fuel), Issue Date: October 2018. Technical report, Regulatory Guide.

IAEA (2020). IAEA Safety Standards: Storage of Spent Nuclear Fuel, No. SSG-15 (Rev. 1). Technical report, SSG-15.

IAEA (2024). IAEA Guidebook on Spent Fuel Storage Options and Systems, 3rd Edition, Technical report No. 240. Technical report, IAEA.

Kook, D., Choi, J., Kim, J., et al. (2013). Review of spent fuel integrity evaluation for dry storage. *Nuclear Engineering and Technology*, 45(1):115–124.

NRC (2025). Spent Fuel Storage in Pools and Dry Casks - Key Points and Questions, `https://www.nrc.gov/docs/ml1209/ML120970230.pdf`. Technical report, Spent Fuel Storage in Pools and Dry Casks.

Pelowitz, D. et al. (2008). MCNPX user's manual version 2.6. *LANL, USA*.

Sindelar, R., Peacock Jr, H., Iyer, N., et al. (1994). Acceptance criteria for interim dry storage of aluminum-clad fuels. Technical report, Westinghouse Savannah River Co., Aiken, SC (United States).

SRIS (2025). IAEA Spent Fuel and Radioactive Waste Information System (SRIS). Technical report, IAEA.

USNRC (2003). USNRC, Cladding considerations for the transportation and storage of spent fuel. Technical report, ISG-11, Rev. 3.

Vienna (2024). International Conference on Spent Fuel Management, 10-14 June 2024, Vienna, Austria. Technical report, Vienna.

To cite this article:

B. Rokrok, Z. Gholamzadeh, A. Jozvaziri, E. Abedi. Investigation of required modifications of the manufactured TRR DPC for transport and storage of two-year cooled spent fuel assemblies. *Radiation Physics and Engineering*, In Press.

DOI:

To link to this article: