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Design and Analysis of Nuclear Transportation Cask Using Phase Changing Material

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Abstract: In the nuclear sector, secure transportation of radioactive materials is crucial in the nuclear sector, and cask design and analysis are crucial for maintaining fuel integrity. Peak cladding temperature (PCT) is a significant factor in fuel failure likelihood. This research provides a thorough approach to designing and analysing nuclear transportation casks, focusing on PCT. Advanced computer techniques are used in the study, which includes a number of variables including the thermal properties of the cask and external heat sources. In this work, a composite PCM made of polyethylene glycol (PEG) and octadecanol is employed to provide equal heat transfer. The proposed composite PCM is reinforced by (Carbon Nano Tubes) CNTs to ensure homogeneous heat conduction within the PCM particles. The thermal behaviour of the cask is simulated using ANSYS, which takes into consideration conduction, convection, and radiation heat transfer among other aspects. The PCT for spent fuel assemblies for light water reactors operating in helium settings with varying decay heat loads and temperature boundary circumstances was calculated using these simulations. Finally yet importantly, the outcome has been validated by comparison with related studies.

Keywords: ANSYS; Peak Cladding Temperature (PCT); Phase Changing Material (PCM); transportation cask

1 INTRODUCTION

At the conclusion of the nuclear fuel cycle, a conveyance task will be needed to transport spent nuclear fuel (SNF) from temporary containers in utilizing crops (reactors of nuclear power plants or scientific sites) to reprocess along with permanently stored sites [1]. Radioactive compounds are normally transported in a specially constructed large container known as a transportation cask. The container might transported through a number of modes, namely road, rail, sea, and potentially air [2]. The procedure for shipping must fulfill a number of duties to assure the security of people, citizens, along with the environment during all situations, namely nuclear material confinement, mechanical security, thermal safeguarding, along with radiologic shielding. It additionally has to allow for simple loading, drying, tightening up, & handling as a way to maintain operational time as well as expenses [3].

SNF gatherings for simple water reactors are typically composed of fuel rods that are joined in square arrays by headers, footings, and periodical spacer plates. Every rod is a Zircaloy-coated tube filled with highly radioactive fuel pellets with fission gases [4]. Spent components are carried far from nuclear power plant zones in thick-walled vessels. Before the US Nuclear Regulatory Commission (NRC) issuing a Certificate of Compliance for a shipment, Federal Regulations (10CFR71) need the supplier to produce a safety analysis report (SAR) showing that the package will continue to provide its containment, shielding, along with criticality handling abilities under normal transport conditions (NCT) afterwards following a series of complicated cases [5]. Amongst the situations was a 9-meter drop onto an unyielding work layer, a 1-meter drop onto a steel piercing rod, full engulfment in an 800 °C fire for 30 minutes, and water submersion. The fuel cladding contains the extremely radioactive fuel pellets along with fission gases, but its integrity needs to be preserved with the goal to keep the components in the expected configuration throughout regulatory review. Radial hydrides can form within the fuel cladding under NCT, making it brittle if the temperature rises beyond 400 °C. If the cladding temperature rises over 570 °C or 750 °C after a fire, it may undergo creep deformation or burst rupture [6].

A transportation cask's body consists of a hollow cylinder with a minimum of three concentric levels: an outer steel shell that interacts with the outside atmosphere, a thick shielding material coating providing thermal insulation along with neutron absorption operations, along with an interior steel shell that lines the cavity where hazardous substances are maintained [7]. These may include vitrified trash packaged in steel canisters or used fuel rods wrapped in metal and grid-stacked into a basket housing the fuel assembly. The container is kept secure by a tight-fitting lid. Shock absorbers are mounted to both ends of the cask to maintain mechanical integrity in the event of a significant accident [8].

PCMs have generated curiosity regarding their use in thermal security, storing power, thermal regulation, and conserving electricity because of their high quantities of enthalpy and of their transitions from liquid to solid [9]. Since they collect heating/cooling thermal energy in the thermal circuit in addition to safeguarding the back-end equipment/materials from thermal runaway/overcooling, PCMs are frequently used as heat capacitors in thermal shielding. Organic PCMs are suitable for thermal barrier operations due to their poor thermal conductivity along with elevated latent heat [10]. However, organic PCM leakage through the solid-liquid phase transition procedure shows a significant impact on their application effectiveness. To address this problem, stable composite PCMs could be made by mixing PCMs with a variety of porous supporting substances [11]. Because of their significant heat conductivity, porous based on carbon supports including extended graphite, carbon nanotube sponge, along with graphene aerogels are not conducive to increasing thermal safeguarding efficacy [12].

The design of casks is crucial in the field of nuclear transportation for assuring the secure and safe transfer of radioactive materials. Our addition, which builds on the current cask design, attempts to improve its characteristics for increased security and effectiveness. We may further enhance the current cask to meet changing industry requirements by concentrating on important factors like structural integrity, thermal management, real-time monitoring, modular design, and regulatory compliance. Primarily, it is crucial to reinforce the cask's structural integrity. We can increase its resistance to impact, corrosion, and environmental factors by utilizing cutting-edge materials like high-strength metals or composites. This improvement will reduce the risk of radioactive material escape by strengthening the cask's ability to withstand mishaps or occurrences during transit. Furthermore, the integrity of the radioactive elements depends on efficient thermal control inside the cask. The cask will be able to maintain constant inside temperatures by including better insulation materials and cooling technologies, notably for high-level radioactive waste. We can increase the overall reliability of the cask design by introducing passive cooling techniques that lessen the need for active cooling systems.

Therefore, the present paper does a thermal study of dry storage facilities for spent nuclear fuel at various conditions, relying on designing foundation estimates and inputs to ensure that the peak cladding temperatures do not go over an established limit. Therefore, a distinctive substance for thermal shielding is determined depending on the thermal characteristics of the entire system.

2 LITERATURE SURVEY

Csontos et al. [13] provided the results of a double-blind benchmark created to estimate the accuracy of simulations of dry storage structures and obtain knowledge of the main causes of modelling inaccuracy. The DOE/EPRI High Burn up (HBU) project's experimental results are contrasted with the four entrants from this round robin. The temperature within the fuel is being monitored by thermocouples while it is kept in a bolted cask together with 32 HBU pressurized water reactor components. The results highlighted the need of improving essential uncertainty component assessments in modelling solutions, especially those pertaining to internal gap size changes.

Zinet et al. [14] used finite element modelling to examine the secondary impacts of the vapour transfer mechanisms unique to these endothermic shielding materials. The framework was developed to study heat transport within a standard cask's wall. The framework considers mechanisms including evaporation, diffusion, and re-condensation of the water content that induce a transfer of latent heat within the medium. The materials under consideration include plaster, phenolic foam, and polyester resin compound. This sophisticated model's forecasts are contrasted with the results of a framework that just includes conduction heat transfer. It has been established that heating kinetics are significantly influenced by gas transport phenomena. Following that, the effects of modelling components such as material porosity and condensation coefficient are addressed.

Two thermal simulations of the SNF storing casks CASTOR 440/84 and CASTOR 440/84M were created utilizing the COBRA-SFS code, an internationally recognized and tested code used for dry storage cask licensing and safety evaluation, according to Seveceka et al. [15]. At the Nuclear Power Plant Dukovany, the WWER-440 SNF is being dry stored in the casks. The SUJB examined and utilized both versions for assessing multiple problems in different configurations, primarily for licensing reasons. The Czech Republic's State Office for Nuclear Safety is the regulatory agency in charge of awarding licenses for spent nuclear fuel storage and transportation containers (SCs) (SNF).

Schwartz [16], According to research, the buffer's auxiliary minerals have a big influence on how quickly the copper shield corrodes. According to groundwater chemistry, a buffer containing calcium sulphates (gypsum or anhydrite) produces a Cu° corrosion depth three to eight times bigger than one without calcium sulphates. The existence or absence of iron oxides (including hematite, goethite, or lepidocrocite) has no influence on reactive-transport models. Iron oxides, however, drastically reduce corrosion depth when present, according to mineral equilibrium thermodynamics.

Dong-Won Lim et al. [17] to prevent the intrusion of an inert sodium area into the air-cooling space built a double-cylinder cask. When both the weight and the shielding thickness were considered, the PSO technique produced a shielding thickness of 26 cm. We were able to assess the effectiveness of the shielding by calculating the radiation dosage of wasted fuel retrieved before and after a year of cooling. Two distinct fuel locations discovered throughout transportation are being studied in order to address a functional issue in a cask drive system. As a result of the analysis, it is determined that the current cask design satisfies legal criteria when used ordinarily.

Alsmadi et al. [18] studied the shielding capabilities of the majority of prevalent structurally amorphous metals (SAMs) for potential usage as coating barriers for radiation shielding and corrosion on spent fuel dry cask canister uses. SAM1651 possessed the greatest attenuation coefficients along with smallest rates of exposure at low photon frequencies owing to its substantial density (7.6 g/cm3) along with elemental content in molybdenum (Mo) and yttrium (Y), whereas SAM2X5 possessed the opposite property at elevated photon energies owing to its high density (7.6 g/cm^3) and elemental content in manganese (Mn) and tungsten (Si).

Hyungjin Kim et al. [19] says that depending on the environment they are in, spent fuel assemblies in nuclear power plants may reach different peak cladding temperatures. Manteufel and Todreas calculate peak cladding temperature using three models: the Wooton-Epstein correlation, the current thermal conductivity approach by Bahney Lotz, and the two-region model. Using a two-dimensional CFD simulation, the study measured the peak cladding temperatures of a Babcock and Wilcox (15 × 15 PWR) reactor. The simulation is helpful for determining peak cladding temperature since it overestimates the measured peak cladding temperature in a used fuel dry storage cask. The 17 × 17 array has a more aggressive effective thermal conductivity than the 16 × 16 PWR spent fuel arrangement.

Jie Li et al. [20] says that for spent nuclear fuel assemblies in commercial light water reactors, the peak cladding temperature (PCT) is a significant metric. An effective thermal conductivity model, a computational fluid dynamics model, and a linked active thermal conductivity as well as edge conductance model were utilized to determine the PCT for spent fuel assemblies in storage or transit casks. The outcomes showed that the vacuum environment is more interesting than other gas atmospheres, as the PCT limit is exceeded at lower boundary temperatures for a specified decay heat load. This essay illustrates the PCT computations and contrasts the outcomes produced by various models.

Several issues have been observed regarding the PCT in previous studies:

1) *Inadequate Thermal Insulation:* Higher temperatures within the barrel during shipment may result from inadequate thermal insulation in the cask design. This may lead to higher PCT values that may exceed the safety thresholds established for the fuel rod cladding. Inadequate insulation materials or subpar insulation design may exacerbate this issue.
2) *Insufficient Cooling Mechanisms:* To reduce the heat that the radioactive materials produce while being transported, suitable cooling systems are required. Temperatures may rise and PCT values may increase if the cooling systems inside the cask are improperly constructed or do not operate as intended. Poor design, broken equipment, or shoddy maintenance can all contribute to insufficient cooling.
3) *Inaccurate Temperature Predictions:* For constructing and assessing the performance of nuclear transportation casks, accurate temperature distribution and PCT prediction is essential. However, prior study has demonstrated cases in which the temperature forecasts were wrong, resulting in an underestimating of the PCT. This might jeopardize the fuel rod cladding's transportation safety.
4) *External Environmental Factors:* The external environment may affect the PCT during transit, including the temperature and sunlight exposure. Higher temperatures and higher PCT values can occur because of inadequate consideration of these parameters during barrel design or inadequate preventive measures.

Consequently, it is evident from the study that it is important to construct a nuclear transportation cask with a high temperature resistant capacity.

3 RESEARCH METHODOLOY

This study suggested a unique thermal shielding for the secure transportation of Spent Nuclear Fuel (SNF) and tests its effectiveness at various temperatures using a 3D of a vertical dry cask built for computer simulation utilizing ANSYS/FLUENT. Based on suggested cask, the SNF is initially covered with steel, and next to that is a lead-based radiation shielding. The role of thermal shielding, which is essential for cask safety, is next discussed. The thermal shielding must be strong enough to prevent both the transmission of inside heat and the transmission of outside heat outdoors. Many low thermal conductive materials are in use, but recently PCM materials have seen significant application for greater SNF applicability. Uneven heat transmission caused by large volumetric change is one of the main issues with PCM. In order to provide uniform heat transmission, a composite PCM made of polyethylene glycol (PEG) and octadecanol is utilized. The suggested composite PCM is strengthened with CNTs to provide uniform heat conduction inside the PCM molecules. The CNT aids in the quick and equal transfer of heat energy; as a result, the volumetric change is uniform throughout the body of the cask, extending life.

The next step is to provide a low thermal conductivity encapsulation in order to bind the thermal energy from a free flow convection process. Thus, the suggested system used 60% of Silica aero gel together with 20% fibre glass and 20% TiO_2 to greatly reduce convection. Low thermal conductivity silica aerogels are utilized to encapsulate PCMs in polyethylene glycol (PEG) and octadecanol with reinforced CNT to create composite PCMs. The suggested aerogel may exhibit low thermal conductivity, high latent heat, big compressive strength, good hydrophobicity, and higher thermal cycle stability when compared to traditional aerogel. Due to the synergistic effect between the composite PCMs' high latent heat also the low thermal conductivity, the proposed silica aerogel mix-based CNT reinforced composite PCMs have the potential to directly apply to the thermal insulation and protection device. Monolithic mechanically reinforced glass fibre TiO_2-silica aerogel composites have been successfully made using two-step sol-gel system. The main silica precursor used was methyl-tri-meth-oxy-silane, which was joint with an environmentally friendly ethanol/water mixture as the solvent and dried at ambient pressure. In this method, the two phases are chemically bound together, and the silica particles take on the elongated shape of fibers, improves the mechanical properties and Young's modulus when compared to the traditional thermal shielding, thereby supplying an impact shield.

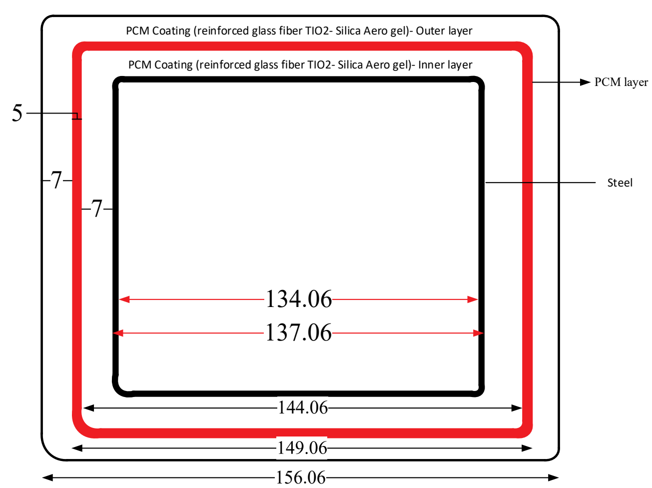

Figure 1 Dimensional parameter of proposed nuclear cask (all the dimension are in mm)

3.1 Modelling of Peak Cladding Temperature

BWR assembly geometries: In Fig. 2, a transverse cross-section of a B&W 9 × 9 fuel assembly, or one-fourth of an assembly, is shown. There are 81 wasted fuel rods in all. Their sheathing has an outside diameter and thickness of 10.77 mm and 0.76 mm, correspondingly. The fuel pellet measures 9.06 mm in diameter. The wasted fuel assembly's active fuel length is 3810 mm. Tab. 1 lists the assembly dimensions that were utilized.

Table 1 Parameter used for modelling spent nuclear cask

Parameter	
Number of rods	79
Rod Pitch	14.53 mm (0.572 in)
UO_2 Pellet Diameter	9.06 mm (0.3565 in)
Cladding Outer Diameter	10.77 mm (0.424 in)
Cladding thickness	0.76 mm (0.030 in)
Number of water Rods	2
Water Rod Outer Diameter	10.82 mm (0.426 in)
Water Rod Thickness	0.81 mm (0.032 in)
Channel Inside Width	134.06 mm (5.278 in)
Channel thickness	2.03, 2.54, or 3.05 mm (0.080, 0.100, or 0.120 in)
Channel Corner Radius	9.65 mm (0.380 in)
Active Fuel Length	3810.0 mm (150 in)

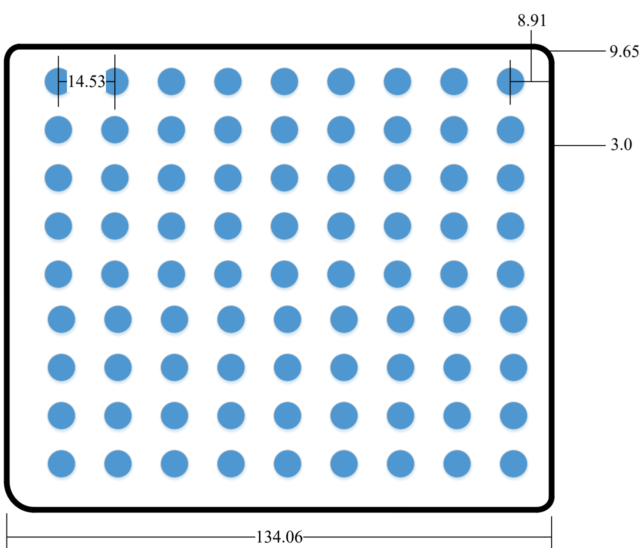

Figure 2 9 × 9 Fuel cell Assemblies (all the dimension are in mm)

3.2 ANSYS Thermal Model

The B&W 9 × 9 BWR assembly's two-dimensional Thermal mesh model is shown in Fig. 4. Using symmetry, this model represents one quarter of an assembly. The method comprises the basket wall with the space between fuel cladding and fuel rods, supposing two planes of symmetry.

A steady-state energy equation was solved consuming a finite volume approach and a second-order upwind discretization methodology, yielding the conduction and radiation temperature findings. The differential equations of the mathematical model are discretized consuming the finite volume technique (FVM), and the resulting algebraic equation system is solved by means of a segregated implicit solver. In order to speed up Gauss-Seidel procedure, calculations are first linearized and then sequentially solved (Hutchinson and Raithby, 1986). Using the Semi-Implicit Method for Pressure-Linked Equations (SIMPLE) technique, the pressure velocity coupling is accomplished (Patankar, 1980).

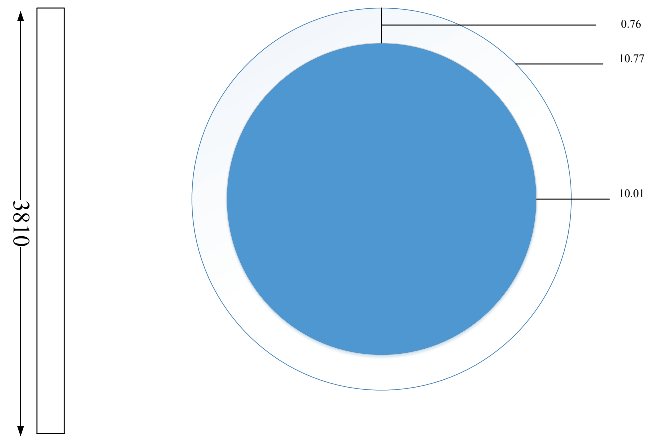

Figure 3 Single fuel rod parameters (all the dimension are in mm)

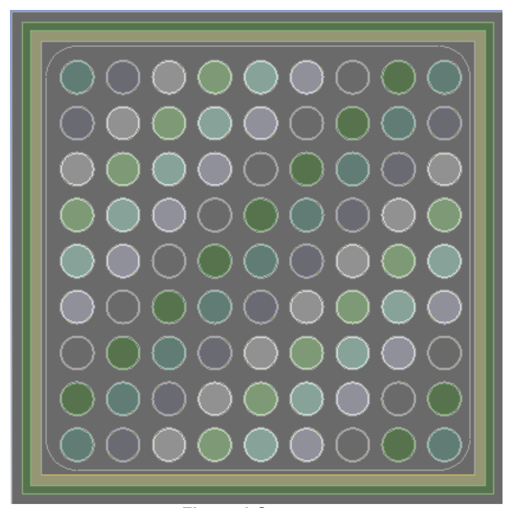

Figure 4 Geometry

The mass and momentum residuals must be 10^{-3} and the energy equation must be 10^{-14} for the convergence criterion to be met. The number 10^{-14} for the energy formula was chosen since the net heat transfer error must be very tiny, signifying that a complete heat balance has been reached. Helium, fuel cladding, and fuel rod's thermal characteristics (specific heat and thermal conductivity) were believed to have a temperature-dependent value (KHNP, 2008).

Radiation model: The radiation of the spent fuel assembly was solved using the discrete ordinates (DO) radiation model. For a finite number of discrete solid angles, each associated with a vector direction fixed in the global Cartesian system (x, y, and z), the discrete ordinates (DO) radiation model solves the radioactive transfer equation (RTE). The user controls the degree of angular discretization that is comparable to selecting the amount of rays in the discrete transfer radiation model (DTRM). Ray tracing is absent from the DO model. The DO algorithm using a transport calculation instead represents the radiation intensity in the spatial coordinates (*x*, *y*, and *z*). Similar to how the energy and fluid flow equations were solved, so was this one.

The DO model has the benefit of being able to be used for modelling non-gray banded radiation, radiation in semi-transparent material, and all types of optical thickness. It remains computationally expensive and has a restricted amount of radiation directions, resulting in numerical smearing. The discrete ordinates radiation model's angular discretization and pixilation values have a big impact on how temperature behaves. Therefore, it is critical to select an appropriate pixilation and discretization value. In ANSYS (2013) and US NRC (2013), the impact of these values in relation to a pixilation or discretization constant is noted.

3.2.1 Model Creation

The ANSYS models are built for the spent fuel shown in Fig. 4. The PCTs were calculated using the fluent code. The steady-state energy formula was determined by employing the finite volume method and a second-order discretization technique, producing values for radiation as well as conduction temperatures. For obtaining the pressure and velocity fields, the steady-state conservation of momentum formulas was addressed utilizing a second-order upwind technique for natural convection/radiation computations. All conduction/radiation and radiation/natural convection scenarios were conducted to determine the impacts of natural convection on the BWR (9 × 9) spent fuel assembly with helium backfill.

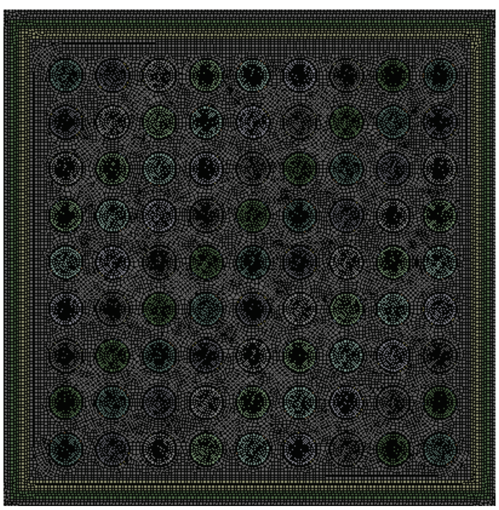

Figure 5 Meshing

3.2.2 Meshing of Geometry

In ANSYS, meshing is the process of turning a geometry into a network of nodes and elements called a finite element mesh. The discretization of the geometry into manageable, small components during the meshing process is a critical stage in the finite element analysis (FEA). A variety of meshing tools and methods are offered by ANSYS to produce high-quality meshes for precise and effective simulations.

Table 2 Meshing size

Meshing size	
Nodes	413971
Elements	76196

3.2.3 Boundary Condition

Boundary conditions in ANSYS are used to specify how a finite element model behaves at its edges. They are necessary for correctly replicating real-world settings and producing insightful findings. You may simulate a variety of physical circumstances by using boundary conditions, which define the limits and loads that are applied to the model.

Table 3 Boundary condition for material

Steel	Density	7850 kg/m^3
	Thermal conductivity	45 W/mK
	Specific heat	0.49 kJ/kgK
PCM with CNT (0.5% of grapheme with erythritol)	Density	1480 kg/m^3
	Thermal conductivity	1.095 W/mK
	Specific heat	1.35 kJ/kgK
Silica Aerogel (Silica + Glassfibre + TiO_2)	Density	0.12 kg/m^3
	Thermal conductivity	0.12 W/mK
	Specific heat	0.023 kJ/kgm^3
Spent Nuclear Fuel (SNF)	Density	2000 kg/m^3
	Thermal conductivity	0.135 W/m^3
	Specific heat	2.640 kJ/kgK
Helium	Density	69.7 kg/m^3
	Thermal conductivity	0.151 W/m^3
	Specific heat	5.1926 kJ/kgK

ANSYS has the tools and capabilities necessary to model time-dependent behaviour and heat generation. Let's examine transient analysis and heat production modelling, two crucial ANSYS concepts relating to time and heat generation.

1) *Transient Analysis:* To examine a system's time-dependent behaviour, transient analysis is performed. It is appropriate for simulations of dynamic events, transient thermal phenomena, or time-varying loads when the model's reaction varies over time. You may carry out transient analysis in ANSYS by taking into account the following:
2) *Time Steps*: The time intervals at which the analysis will be conducted must be specified. You may choose the time step size, overall analysis time, and a number of other time-dependent options using ANSYS.
3) *Loads and Boundary Conditions:* Changing forces or temperature profiles are examples of time-varying loads that may be imposed at various time increments. Similar boundary conditions that rely on time include temperature changes and specified displacements.
4) *Solution Method:* The transient equations are solved numerically by ANSYS. Up until the analysis time is reached, the program use iterative algorithms to calculate the response of the model at each time step.

Modelling of heat generation: Thermal analysis frequently involves heat generation when there are heat sources present, such as electrical components, chemical processes, or frictional heat. ANSYS offers numerous ways to mimic the production of heat:

1) *Heat Sources:* At particular points in the model, the power density or rate of heat generation can be specified. To do this, either provide the heat generation value or take into consideration internal heat generation by employing the right material attributes.
2) *Joule Heating:* The resistance of conductive materials to electrical current causes joule heating in electrical systems. By defining electrical current and resistive qualities, ANSYS enables you to add the impacts of joule heating.
3) *Radiation and Convection:* Heat transmission by radiation and convection may be modelled using ANSYS. Boundary conditions that take radiation or convective heat transfer coefficients into account for heat exchange with the surroundings can be defined.
4) *Thermal Material Properties:* For a realistic representation of material thermal behaviours, ANSYS offers a wide variety of material models and attributes. These characteristics, including thermal conductivity, specific heat, and thermal expansion, are very important in models of heat generation and transport.

ANSYS enables you to simulate and evaluate time-dependent thermal phenomena, including heat sources, temperature changes, and transient behaviours in diverse engineering applications. This is done by integrating transient analysis with suitable heat generating modelling approaches.

3.2.4 Setup

Configure the thermal solver settings, such as selecting the appropriate solver (e.g., steady-state or transient), specifying convergence criteria, and defining solution methods.

Transient Thermal: Transient thermal analysis in ANSYS Thermal is a powerful tool for studying temperature changes over time in a system. It allows engineers and researchers to simulate and analyse transient heat transfer phenomena, such as how a system responds to sudden temperature changes or how long it takes a system to reach thermal equilibrium. The process begins with creating or importing the system's geometry into ANSYS and generating an appropriate mesh. Next, thermal properties like conductivity, specific heat, and density are defined for the materials involved. Boundary conditions are then specified, including initial temperature distribution, prescribed temperature changes, heat fluxes, and radiation effects. Solver settings are configured, such as the time step size and convergence criteria. The transient analysis is then solved, with ANSYS iterating through time steps to solve the heat transfer equations and update the temperature distribution. Finally, post-processing tools in ANSYS allow for in-depth analysis and visualization of the results, including temperature distribution, heat fluxes, and other relevant quantities at different time points. By following these steps, engineers can gain valuable insights into the transient thermal behaviour of their systems and make informed decisions for design improvements or optimization.

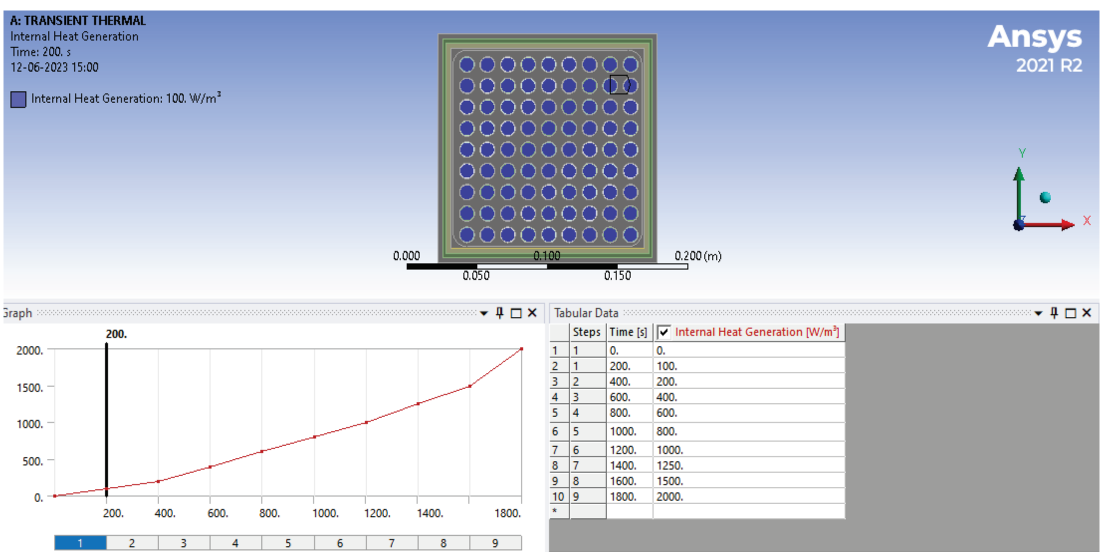

Figure 6 Time step with heat generation

4 RESULT AND DISCUSSIO

Peak cladding temperature (PCT) is the highest temperature that the protective covering of a nuclear fuel rod can reach under regular or unusual operational circumstances. Inside a nuclear reactor, the fuel rods house fuel pellets that produce heat through fission. This protective covering, often crafted from materials such as zirconium alloy, envelops the fuel pellets, acting as a shield to keep them in place and prevent the dispersion of radioactive substances.

4.1 For 25 °C and 100 W Heat Generation

In Fig. 7, the output of the nuclear cask is presented under conditions of a 25 °C ambient temperature and an internal heat production of 100 W. The graph indicates that the highest temperature reached by the cladding is 23 °C.

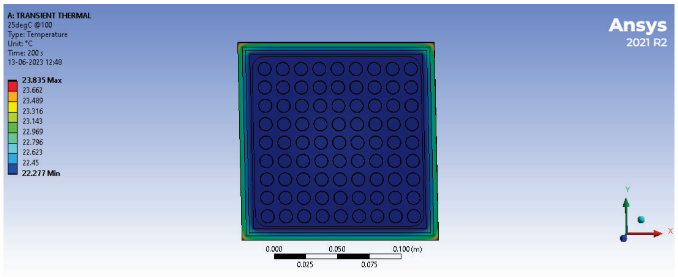

Figure 7 Analysis for 25 °C and 100 W heat generation

4.2 For 25 °C and 800 W Heat Generation

The graph in Fig. 8 displays the performance of the nuclear cask under specific conditions: an ambient temperature of 25°C and an internal heat generation of 800 W. It reveals that the Peak cladding temperature reached 24°C.

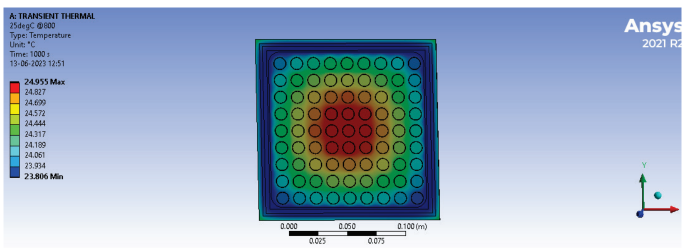

Figure 8 Analysis for 25 °C and 800 W heat generation

4.3 For 25 °C and 2000 W Heat Generation

In Fig. 9, the graph illustrates the output of a nuclear cask when subjected to an ambient temperature of 25 °C and an internal heat production of 2000 W. As indicated by the graph, the highest temperature reached by the cladding is 28 °C.

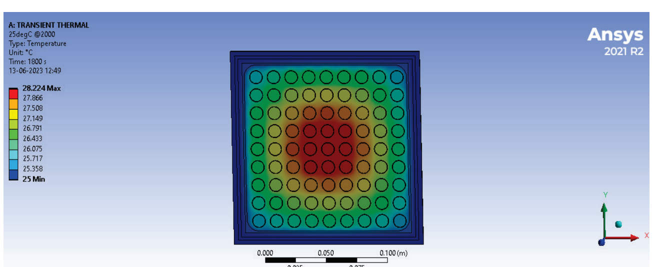

Figure 9 Analysis for 25 °C and 2000 W heat generation

4.4 For 400 °C and 100W Heat Generation

In Fig. 10, the nuclear cask's performance is depicted, indicating an internal heat generation rate of 100 W and an ambient temperature of 400 °C. The graph exhibits a maximum cladding temperature of 253 °C.

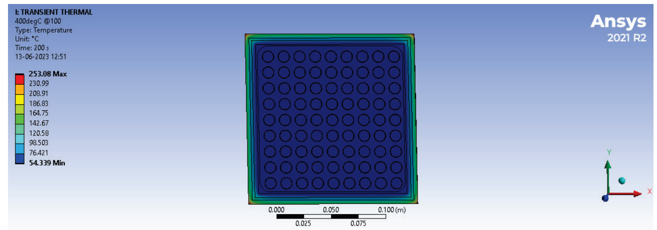

Figure 10 Analysis for 400 °C and 100 W heat generation

4.5 For 400 °C and 800 W Heat Generation

In Fig. 11, the graph illustrates the output of the nuclear cask when subjected to a 400 °C ambient temperature and an internal heat production of 800 watts. Specifically, it shows a high cladding temperature of 320 °C as depicted on the graph.

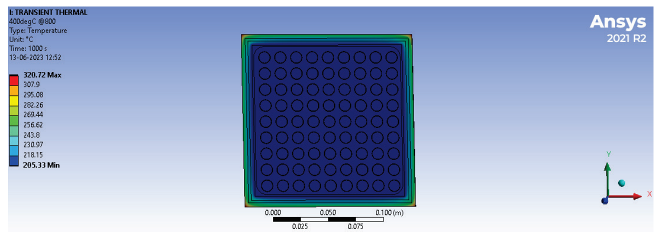

Figure 11 Analysis for 400 °C and 800 W heat generation

4.6 For 400 °C and 2000 W Heat Generation

In Fig. 12, the output of the nuclear cask is depicted under specific conditions: an internal heat generation of 2000 W and an ambient temperature of 400 °C. The graph indicates that the cladding temperature reaches a high of 356 °C.

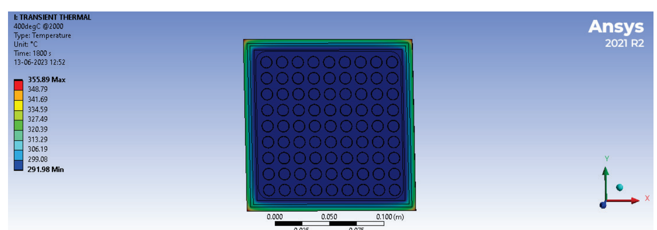

Figure 12 Analysis for 400 °C and 2000 W heat generation

4.7 For 800 °C and 100 W Heat Generation

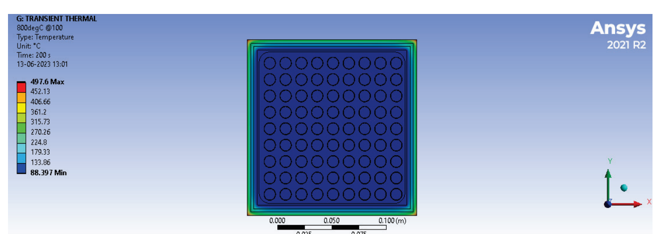

Figure 13 Analysis for 800 °C and 100 W heat generation

In Fig. 13, the graph illustrates the output of the nuclear cask under specific conditions: an ambient temperature of 800 °C and an internal heat production of 100 W. The graph highlights a notable high cladding temperature, reaching 497 °C.

4.8 For 800 °C and 800 W Heat Generation

In Fig. 14, the data illustrates the output of a nuclear cask under specific conditions. With an internal heat generation of 800 W and an ambient temperature of 800 °C, the graph indicates a peak cladding temperature of 636 °C.

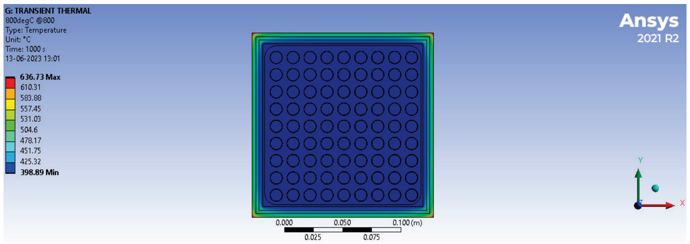

Figure 14 Analysis for 800 °C and 800 W heat generation

4.9 For 800 °C and 2000 W Heat Generation

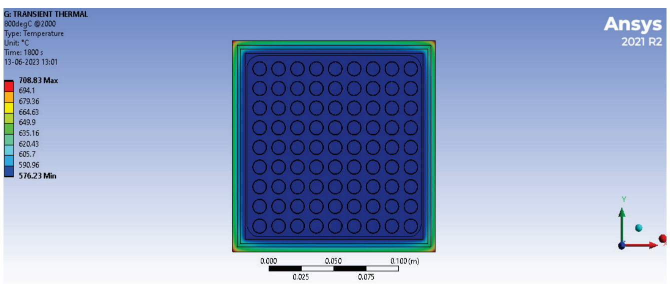

Figure 15 Analysis for 800 °C and 2000 W heat generation

Fig. 15 illustrates the performance of the nuclear barrel, showing an internal heat generation of 2000 W and an ambient temperature of 800 °C. The graph displays a maximum cladding temperature of 708 °C.

The Peak Cladding Temperature (PCT) for a BWR (9 × 9) spent fuel assembly in each environment are shown in Tab. 4 based on the ETC correlations. Again, the vacuum environment produces the greatest PCTs, whereas the helium atmosphere produces the lowest PCTs.

Table 4 Calculated PCTs for a BWR spent fuel assembly

Temperature (°C)/Internal heat (W)	100	200	400	600	800	1000	1250	1500	2000
25	23	24	24	24	24	25	26	27	28
100	69	74	78	81	83	85	87	89	91
200	130	141	150	157	163	168	172	176	179
300	192	210	222	233	242	250	257	263	268
400	253	277	294	308	320	331	340	349	356
500	314	343	365	384	400	413	425	435	444
600	375	411	438	460	478	495	509	522	532
700	436	478	509	535	557	576	593	608	620
800	497	545	581	611	636	658	677	694	708

5 COMPARATIVE ANALYSIS

The suggested study has been evaluated against prior research, as displayed in the Tab. 5. We can conclude from this work that PCM is a suitable material for use in nuclear transportation casks because the table clearly shows that our proposed cask with PCM has higher thermal shielding than other materials like zirconium, which has higher thermal resistance and corrosion resistance but was a bit unstable for higher and longer use.

Table 5 Comparative analysis

Research work	Temperature (ºC)/Internal heat (W)	200	400	800	1000	1250	1500
Hyungjin Kim et al. [19]	300	315.7	330.6	344.8	358.5	371.5	384.1
Jie Li et al. [20]	300	303	305	310	313	319	325
Bahney et al. [21]	300	303	306	309	313	316	319
Proposed work	300	210	222	242	250	257	263

6 CONCLUSION

In a 2D ANSYS simulation based on the FLUENT code, the suggested framework was used to estimate the peak cladding temperature along with the efficient thermal conductivity for up to 9 × 9 BWR spent fuel assemblies under helium backfill gas. They were evaluated using a range of assembly heat loads with basket wall temperatures. The ANSYS simulation's peak cladding temperatures were compared to the traditional technique's cladding temperatures.

The peak cladding temperatures as well as the efficient thermal conductivity of spent fuel assemblies utilized in Koodankulam NPPs are being investigated utilizing ANSY Thermal modeling. They were calculated using the 9 × 9 PWR spent fuel assemblies. For this simulation, the assembly heat load ranged from 100 to 2000 W. The simulation findings for peak cladding temperatures are identical, but the temperature drop variance among the three arrays happens very minimally at all basket wall temperatures. The efficient thermal conductivity computed from the 9 × 9 BWR spent fuel assembly data is more conservative than typical shielding found in the literature at a normal operating temperature for spent fuel transportation/storage casks. In the future, we would like to concentrate on the development of improved thermal management technologies, such as passive or active cooling approaches, to effectively disperse heat created during travel.

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7 REFERENCES

[1] Kurniawan, T. A., Othman, M. H. D., Singh, D., Avtar, R., Hwang, G. H., Setiadi, T. & Lo, W. H. (2022). Technological solutions for long-term storage of partially used nuclear waste: A critical review. *Annals of Nuclear Energy, 166*, 108736.

https://doi.org/10.1016/j.anucene.2021.108736

[2] Alyokhina, S. (2018). Thermal analysis of certain accident conditions of dry spent nuclear fuel storage. *Nuclear Engineering and Technology, 50*(5), 717-723. https://doi.org/10.1016/j.net.2018.03.002

[3] Kim, T., Kim, K., Lee, D., Na, T., Chung, S. & Kim, Y. (2022). Conceptual Design, Development, and Preliminary Safety Evaluation of a PWR Dry Storage Module for Spent Nuclear Fuel. *Applied Sciences, 12*(9), 4587. https://doi.org/10.3390/app12094587

[4] Burns, J. R., Hernandez, R., Terrani, K. A., Nelson, A. T. & Brown, N. R. (2020). Reactor and fuel cycle performance of light water reactor fuel with 235U enrichments above 5%. *Annals of Nuclear Energy, 142*, 107423. https://doi.org/10.1016/j.anucene.2020.107423

[5] Shama, A., Rochman, D., Caruso, S. & Pautz, A. (2022). Validation of spent nuclear fuel decay heat calculations using Polaris, ORIGEN and CASMO5. *Annals of Nuclear Energy, 165*, 108758. https://doi.org/10.1016/j.anucene.2021.108758

[6] Lee, D. & Diaconeasa, M. A. (2022). Preliminary Siting, Operations, and Transportation Considerations for Licensing Fission Batteries in the United States. *Eng., 3*(3), 373-386.

[7] Jeong, G., Kim, S. & Lee, S. (2022). Development of Model to Evaluate Thermal Fluid Flow around a Submerged Transportation Cask of Spent Nuclear Fuel in the Deep Sea. *Journal of Nuclear Fuel Cycle and Waste Technology*, Corpus ID: 256163523. https://doi.org/ https://doi.org/10.7733/jnfcwt.2022.043

[8] Rodríguez-Penalonga, L. & Moratilla Soria, B. Y. (2017). A review of the nuclear fuel cycle strategies and the spent nuclear fuel management technologies. *Energies, 10*(8), 1235. https://doi.org/10.3390/en10081235

[9] Abdelgaied, M., Attia, M. E. H., Kabeel, A. E. & Zayed, M. E. (2022). Improving the thermo-economic performance of hemispherical solar distiller using copper oxide nanofluids and phase change materials: Experimental and theoretical investigation. *Solar Energy Materials and Solar Cells, 238*, 111596. https://doi.org/10.1016/j.solmat.2022.111596

[10] Yousefi, E., Nejad, A. A., & Rezania, A. (2022). Higher power output in thermoelectric generator integrated with phase change material and metal foams under transient boundary condition. *Energy, 256*, 124644. https://doi.org/10.1016/j.energy.2022.124644

[11] Zhang, J., Cao, Z., Huang, S., Huang, X., Liang, K., Yang, Y. & Wen, C. (2022). Improving the melting performance of phase change materials using novel fins and nanoparticles in tubular energy storage systems. *Applied Energy, 322*, 119416. https://doi.org/10.1016/j.apenergy.2022.119416

[12] Li, Y., Huang, X., Wang, J., Lv, F., Jiang, S. & Wang, G. (2022). Enzymolysis-treated wood-derived hierarchical porous carbon for fluorescence-functionalized phase change materials. *Composites Part B: Engineering, 234*, 109735. https://doi.org/10.1016/j.compositesb.2022.109735

[13] Csontos, A. A., Waldrop, K., Durbin, S., Hanson, B. D., Broussard, J. E. & Lenci, G. (2018). Spent Fuel Dry Storage Cask Thermal Modelling Round Robin. https://api.semanticscholar.org/CorpusID:211483369.

[14] Zinet, M., Ghazal, R., Issard, H. & Bardon, O. (2019). Spent fuel transportation cask under accidental fire conditions: Numerical analysis of gas transport phenomena affecting heat transfer in shielding materials. *Progress in Nuclear Energy, 117*, 103045. https://doi.org/10.1016/j.pnucene.2019.103045

[15] Ševečeka, M., Valacha, M. & Leeb, C. H. (2020). Thermal analysis of dry storage and transportation casks castor using COBRA-SFS. *Acta Polytechnica CTU Proceedings, 28*, 32-41. https://doi.org/10.14311/APP.2020.28.0032

[16] Schwartz, M. O. (2021). Corrosion-Enhancing and Corrosion-Reducing Accessories in Bentonite Surrounding Copper-Shielded Containers for Nuclear Waste. *Journal of Hazardous, Toxic, and Radioactive Waste, 25*(4), 04021024. https://doi.org/10.1061/(ASCE)HZ.2153-5515.0000628

[17] Lim, D. W., Lee, C. W., Lim, J. Y. & Hartanto, D. (2019). On the Particle Swarm Optimization of cask shielding design for a prototype Sodium-cooled Fast Reactor. *Nuclear Engineering and Technology, 51*(1), 284-292. https://doi.org/10.1016/j.net.2018.09.007

[18] Alsmadi, Z. Y. & Bourham, M. A. (2022). An assessment of protective coating dry cask canisters with structurally amorphous metals (SAMs) for enhanced radiation shielding. *Nuclear Engineering and Design, 388*, 111647. https://doi.org/10.1016/j.nucengdes.2022.111647

[19] Kim, H., Kwon, O. J., Kang, G. U. & Lee, D. G. (2014). Comparisons of prediction methods for peak cladding temperature and effective thermal conductivity in spent fuel assemblies of transportation/storage casks. *Annals of Nuclear Energy, 71*, 427-435. https://doi.org/10.1016/j.anucene.2014.04.004

[20] Li, J., Murakami, H., Liu, Y., Gomez, P. E. A., Gudipati, M. & Greiner, M. (2007). Peak cladding temperature in a spent fuel storage or transportation cask. *Proceedings of the 15th International Symposium on the Packaging and Transportation of Radioactive Materials*, 21-26.

[21] Bahney, R. H. & Lotz, T. L. (1996). *Spent nuclear fuel effective thermal conductivity report*. Prepared for the US DOE, Yucca Mountain Site Characterization Project Office by TRW Environmental Safety Systems, Inc.

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