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Structural and Thermal Analysis of Fuel Rod of VVER-1200 Nuclear Reactor Using ANSYS Software

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ABSTRACT

This study focuses on analyzing the thermo-mechanical behavior of fuel rods in VVER-1200 nuclear reactors to ensure their safe and efficient operation. Bangladesh plans to operate two VVER-1200 reactors at Rooppur by 2025/2026, aligning with global efforts to shift toward low-carbon energy sources. The analysis, performed using ANSYS software, evaluates key parameters such as deformation, stress, strain, temperature distribution, and heat flux. A detailed geometric model of the UO_2 fuel rod with zirconium alloy cladding and helium gas in the gap has been developed. Thermal distribution simulations under steady-state conditions reveal critical temperature gradients, while structural analysis identifies high-stress regions due to thermal expansion and operational loads.

1. INTRODUCTION

Nuclear fuels operate under extremely demanding conditions within the reactor core, where they are subjected to a combination of corrosive environments, mechanical stresses, high temperatures, and intense radiation effects on the fuel elements. The VVER-1200 is a pressurized Water Reactor (PWR) design. As primary coolant, water is used in this system and by circulation remove generated heat. This heated water then flows to a secondary circuit, where it transfers its heat to generate steam for electricity production. The coolant flow within the core is designed to efficiently pass through the fuel assemblies, ensuring effective heat removal from the fuel rods. Research aimed at understanding the irradiation behavior of nuclear reactor fuels has revealed significant changes in the geometry, dimensions, composition, and micro-structure of the fuels both during and after exposure to ionizing radiation (IAEA, 2015).

There are several critical issues that need to be investigated, with pellet-Cladding Interaction (PCI) being a prominent one. Additionally, obtaining detailed dimensions of core elements can be challenging. To ensure fuel safety, it is crucial to evaluate the stress, strain, and displacement distribution within the pellet and cladding for individual fuel pellets. However, for the sake of simplicity in core design and analysis, the complexity must be minimized. Due to the limited availability of research and papers on this topic, a structural analysis is necessary to understand the mechanical behavior of the fuel under normal operating conditions. This analysis is essential for addressing potential problems in the future. One effective approach to designing and evaluating the behavior of such fuel elements is through simulation using specialized software, and in this case, utilizing ANSYS for the analysis.

The ANSYS model necessitates specifying the material properties for each component, along with defining the geometry, mesh configuration, and the applied thermal and static structural loads. The structural analysis evaluates mechanical parameters, including elastic and plastic strains or stresses, under static loading conditions (Gojan, A. 2016).

A steady-state thermal analysis is first conducted, and the thermal results are then utilized as input for the structural model. To maintain brevity, this study focuses solely on the structural analysis results, although ANSYS predictions for temperature distribution are available where ANSYS predictions for temperature distribution can be found (Romanet et al., 2019). The main research gap is the lack of information about the structural dimensions of the entire reactor and the research field is not very enriching enough. The objectives of the study are to analyze the sustainability of fuel rods in the high pressure and temperature ambient and enrich information about the VVER reactors can help solve problems that will be faced in the future.

The objective of this work is to analyze structural and thermal analysis of a fuel rod of VVER-1200 reactor under normal condition and conclude the safety and integrity and plot temperature distribution among fuel, gap and cladding.

2. GEOMETRY MODELING

The VVER reactor, a Russian design, primarily uses uranium dioxide (UO_2) as fuel, enriched with uranium-235 to around 3-5%. This fuel is shaped into small pellets and loaded into long cylindrical fuel rods made of zirconium-4 alloy, chosen for its low neutron absorption and high resistance to corrosion and radiation. In VVER-1200 reactors, these fuel rods are arranged into hexagonal fuel assemblies, allowing efficient coolant flow and optimal core performance. Each rod is sealed with end caps to prevent radioactive fission products from escaping into the coolant, with a small gap inside to accommodate thermal expansion and fission gas release. The cladding serves as a barrier to protect the reactor's coolant from contamination. Thermal and structural analyses of these fuel rods are essential to ensure that they can withstand the extreme conditions inside the reactor, including high temperatures, pressures, and neutron bombardment. These analyses are critical for the reactor's operational efficiency and safety (Status report 108 - VVER-1200). General model of the VVER-1200 fuel rod is shown in Figure 1. and the geometry is created according to Table 1.

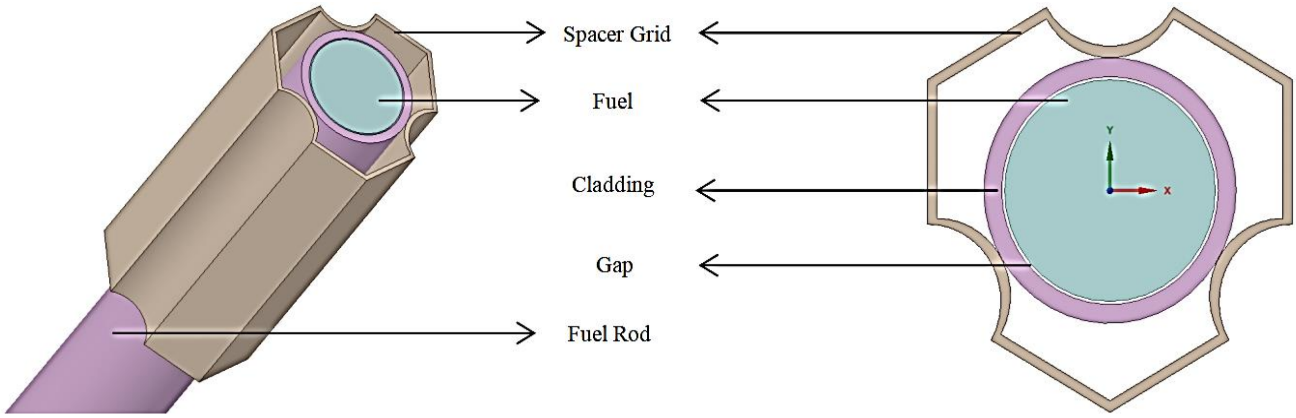


Figure 1: Model of a fuel rod with spacer grid created in ANSYS Design Modular

Table 1: Technical Data Considered For Analysis (Status report 108 - VVER-1200), (Yassin, K. M., Hassan, M. H., Ghoneim, M. M., Elkolil, M. S., Alyan, A., & Agamy, S. A. (2023))

Pellet material: uranium oxide (UO2).	
Clad and Spacer Grid material: zircaloy-4.	
Gap =Consist of Helium	
Pellet diameter	7.6 mm
Gap thickness	0.1 mm
Clad inner diameter	7.8 mm
Clad outer diameter	9.1 mm
Length of fuel rod	3750 mm
Thickness of spacer Grid and Plate	30 mm
No of spacer grid with 2 plate	15

Table 1 shows some structural dimension, which is used to design a fuel rod and table 2 consists of some value which is used to define the structural material (UO2 and Zirconium-4) an ANSYS workbench.

3. MATERIAL PROPERTIES

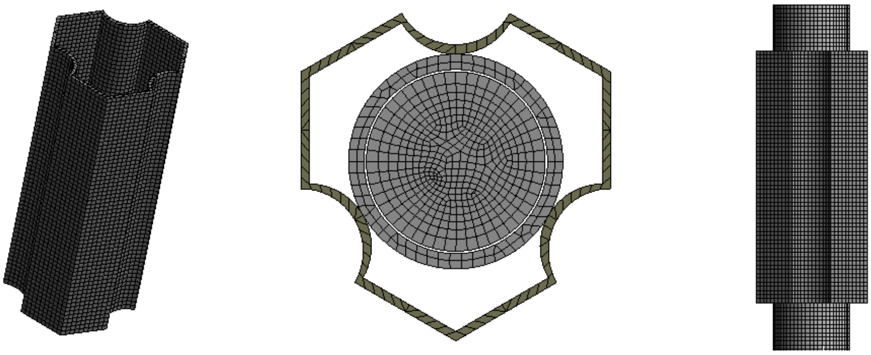
To specify used material structural and thermal properties of Zr-4 and UO2 have to be inserted in the engineering data library as user-defined material. This is already defined in the ANSYS data library (Unrivaled Materials). Table 2 material properties of Zircaloy-4 and UO2 are used in the work.

Table 2: Technical Data Considered For Analysis (Mihaela Roxana Roman; K.M.Pandey&Amrit Sarkar,2011)

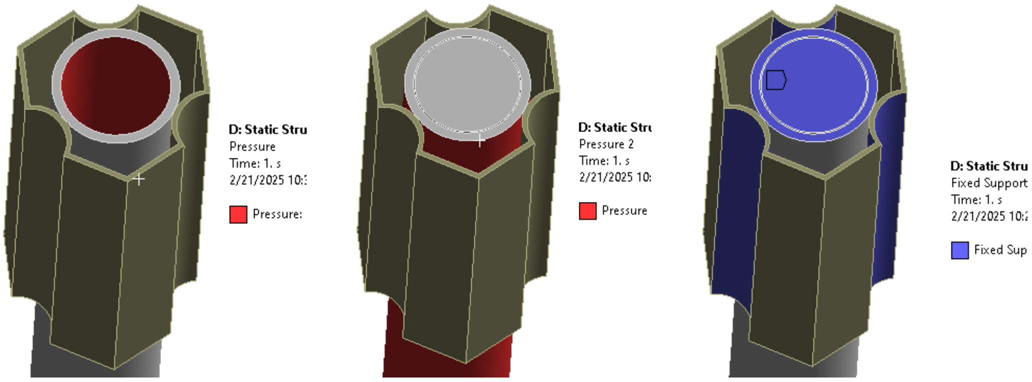
Material Property	Zircaloy-4	Uranium Dioxide (UO_2)
Density (kg/m^3)	6.505	10.97
Young's Modulus (GPa)	88	182
Thermal Conductivity (W/m·K)	22.6	29
Poisson's Ratio	0.34	0.295
Bulk Modulus (GPa)	91.1	265
Shear Modulus (GPa)	33	0.95
Heat Generation(w/m^3)	N/A*	$8x10^8$
Thermal co-efficient of Expansion	$20x10^{-6}$	$10.8x10^{-6}$

4. FINITE ELEMENT MODELLING AND BOUNDARY CONDITIONS

Proper meshing helps to get accurate results. The skewness of the generated mesh is 0.1524 and according to reference (Nor Mariah Adam, 2020) skewness<0.25 is excellent. In this work an adaptive meshing technique is used. Figure 2 represents the finite element modeling of fuel cladding and spacer grid.

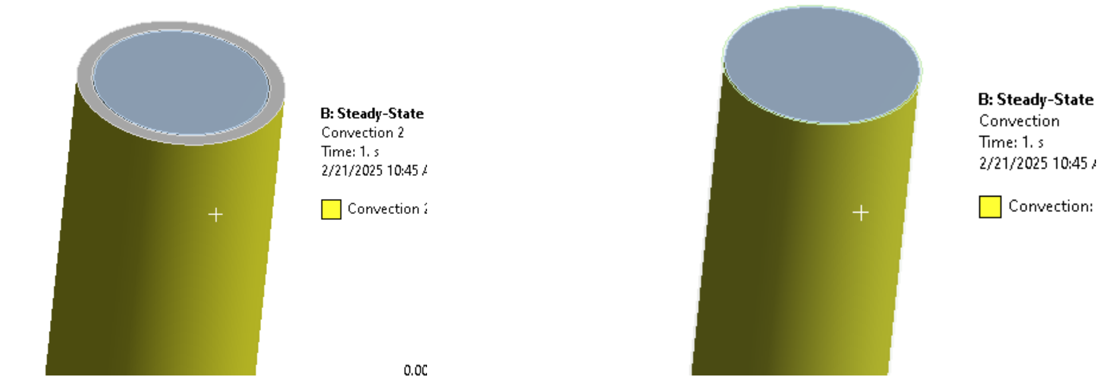


(a) Finite Element Modelling

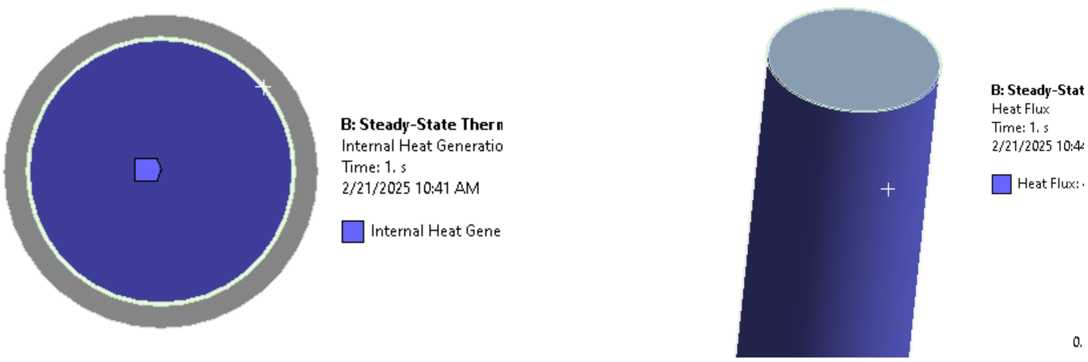


i) Fixed Support ii) Internal Pressure iii) Outer Surface Pressure

(b) Boundary condition for structural analysis (i, ii & iii)



iv) Convection between fuel and cladding v) Convection between cladding and coolant



vi) Heat Flux vii) Internal Heat Generation (Fuel Pellet)

(c) Boundary condition for Thermal analysis (iv, v, vi & vii)

Figure 2: Finite Element Modelling (a), Boundary condition for structural analysis (b), and Boundary condition for Thermal analysis (c)

After generating a mesh, boundary conditions have to apply. For the structural analysis fixed support on the spacer grid and pressure is applied. And in the figure 2, Pictorially represents the location of all applied boundary conditions and meshing. The internal pressure of the fuel element is exerted on the outer surface of the pellets and the inner surface of the cladding, with values obtained from TRANSURANUS (Mihaela Roxana Roman). For the thermo-mechanical study of nuclear fuel rods, TRANSURANUS is a well-known and extremely specialized software program. It is frequently used to forecast fuel behavior under varied operating settings in both research and the nuclear industry. TRANSURANUS is especially well-suited for in-depth simulation of processes including fuel reformation, fission gas release, and pellet-cladding interaction. A more specialized treatment of fuel-specific phenomena is provided by TRANSURANUS (licensed), but ANSYS (free student edition), the program utilized in this study has strong general-purpose finite element analysis capabilities. And clad external pressure is obtained by,

$$\text{Pressure, } p = (p_i - p_o)\,\frac{\delta^2}{\delta^2-1} - p_i \qquad (1)$$

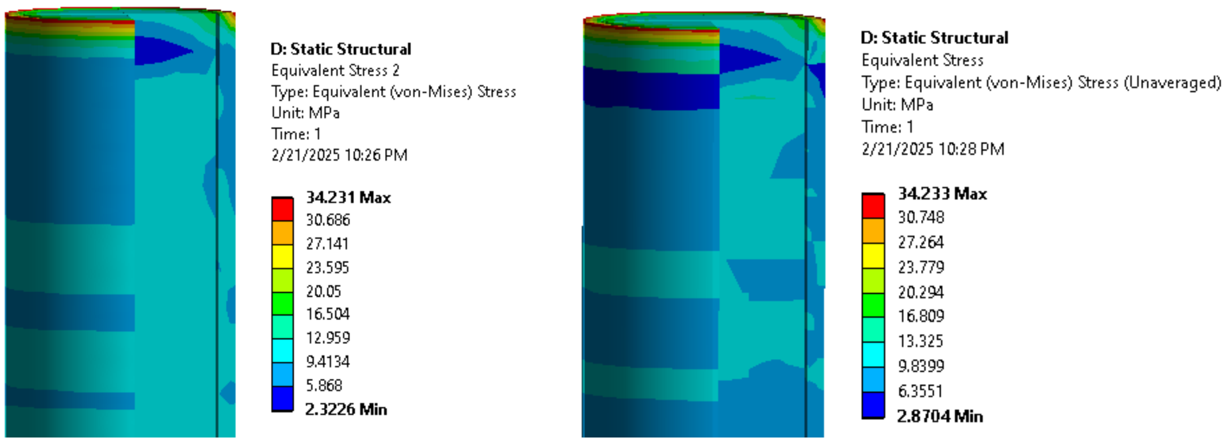
Where, p_i = internal pressure = 5.5 MPa (Hagrman, D., & Reymann), p_o = Coolant pressure, 16.2 MPa (Status report 108 - VVER-1200), δ = ratio between inner and outer diameter.

For thermal analysis, Fuel is used to generate heat. Convection and heat flux is used for heat transfer. These loads are imposed at every time step. In the thermal analysis, heat generation within the fuel pellet is modeled as a uniform volumetric heat source with a rate of 8x108 W/m3. Heat transfer from the cladding to the coolant is simulated using a convection boundary condition applied to the outer surface of the cladding. A constant convection coefficient of 2800W/m2·K is used, assuming a bulk coolant temperature of 320 0C. This steady-state thermal analysis calculates the temperature distribution within the fuel rod under normal operating conditions.

5. RESULTS AND DISCUSSION

At first, to validate the result, analysis result of a single fuel rod without spacer-grid is compared with another research result previously which is shown in Figure 8. And Figure 9 repents the structural analysis result of a single fuel rod without adding additional spacer grid.

And results of averaged and unaveraged value (Figure 3 (i & ii)) repents the accuracy, mesh independence and the reliability of the result. Here, deviation between these two values $\frac{(34.233-34.211)}{34.233} \times 100\% = 0.067\%$, which states that this result is reliable with respect to mesh selection and method of meshing.



(a) Averaged (b) Unaveraged

Figure 3: Von-Mises Stress without considering the spacer grid

Table 3 represents the numerical comparison of Structural and thermal analysis results with previous research work This validates all of the applied boundary condition for structural and thermal analysis. In the further analysis spacer grid is applied to experience more practical flavour in the real scenario.

The structural analysis of the VVER-1200 fuel rod reveals that in figure 5(a) the equivalent stress (von Mises stress) is 43.227 MPa, which is well below the yield strength of Zircaloy-4 is 381 MPa (Zircaloy-4(Alloy Zr4)). This indicates that the material is operating within its elastic range.

Table 3: Validation of Structural anthermal analysisis results with previous research work

Parameters	Experimental Result	Previous research work	Error
Von-Mises Stress (MPa) (without spacer grid)	34.231	32	5%
Maximum Temperature (K)	2093.1	2087.4	0.27%

The calculated safety factor for the equivalent stress is approximately 8.81, (safety Factor $= \frac{\text{Yield Strength}}{\text{Equivalent Stress}} = \frac{381}{43.227}$), highlighting a substantial margin against failure. The cumulative number of strain fatigue cycles on each fuel assembly structural member is assumed to be significantly less than the design fatigue lifetime, which in turn is based on appropriate data, and usually includes a safety factor (2) on stress amplitude and a safety factor (20) on the number of cycles (Review of the Margins for ASME Code). So, the safety factor is in the acceptance limit.

According to Figure 5(b), the maximum principal stress is 31.712 MPa, also significantly lower than the yield strength, confirming the absence of localized tensile failure risks. Additionally, the total deformation of 1.6705 µm is shown in Figure 3 and according to reference (Vladimir et al., 2024), 4-5 µm can withstand a fuel rod in high temperature (Higher than 612°C), verifying that the fuel assembly maintains its geometric stability and operational integrity cause the total deformation is in safety limit.

Figure 6 represents the radial temperature distribution into the fuel rod and Figure 6 is the graphical representation of the variation of temperature with radius of the fuel rod. Also from the figure, the maximum temperature is in the center of the fuel which is 2093.1 K called hot spot of the rod. And the minimum temperature is in the cladding outer surface which is 612.46 K. Also, the temperature cure is more step in the gap region due to low heat transfer capability of helium gas with respect to fuel and cladding material.

Figure 8 also shows the radial temperature distribution of fuel rod for general PWR, where maximum temperature is 2087.4 K. Deviation of result $= \frac{2093.1-2087.4}{2093.1} \times 1\,00\% = 0.27\%$.

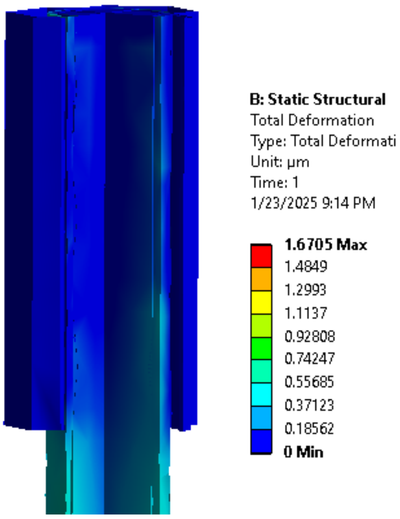
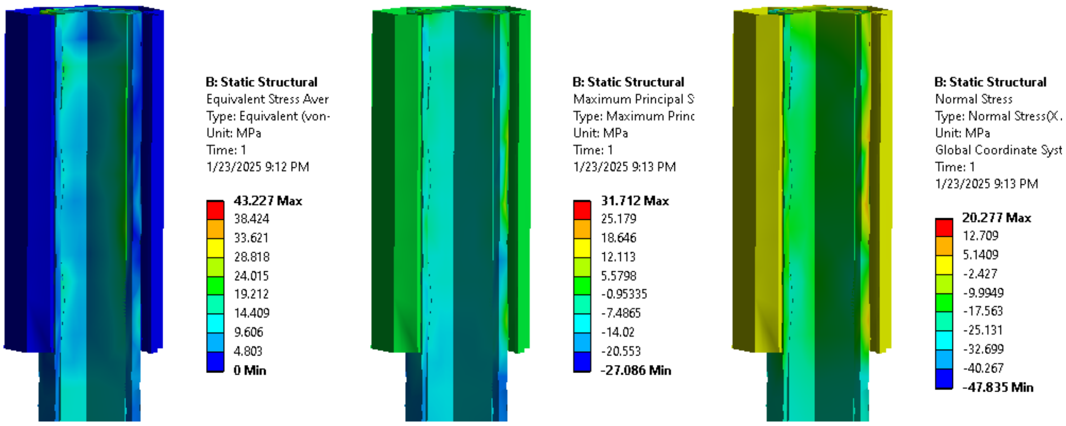


Figure 4: Obtained Total Deformation (With spacer grid)



(a) (b) (c)

Figure 5: Obtained stress a. Equivalent von Mises, b. Maximum Principle and c. Normal Stress (With sacer grid)

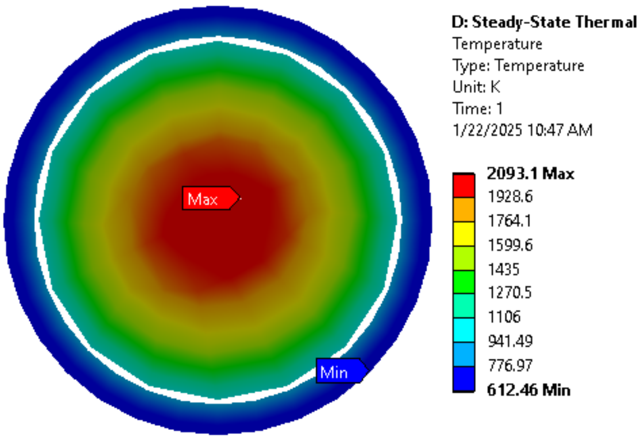


Figure 6: Radial temperature distribution in the fuel rod

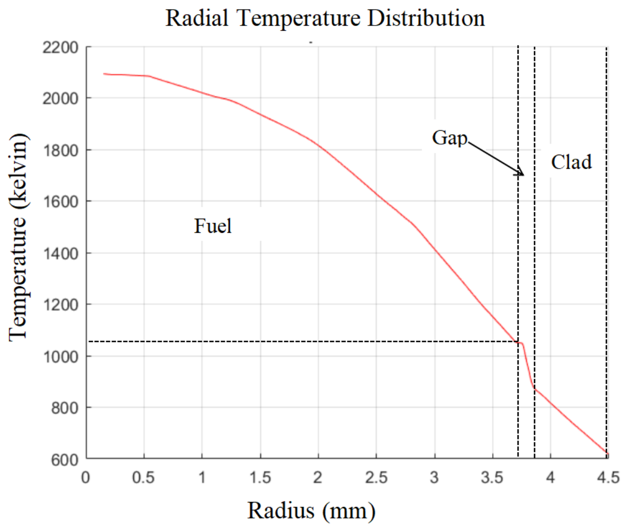


Figure 7: Radial temperature distribution graph using ANSYS

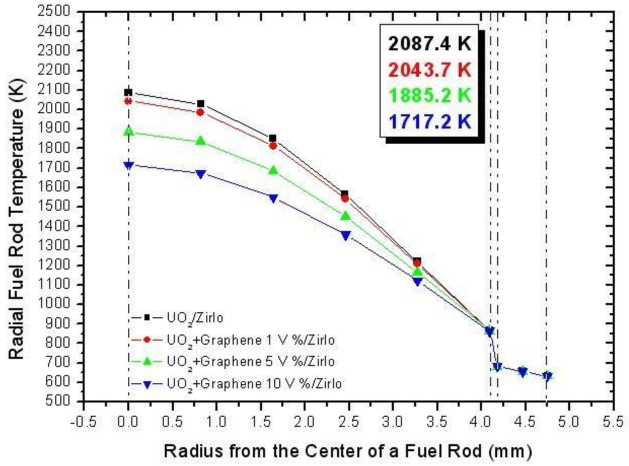


Figure 8: Radial temperature distribution graph using COMSOL [Thermal Validity] (Bang, I. C., Lee, S., & Kim, H. (2011))

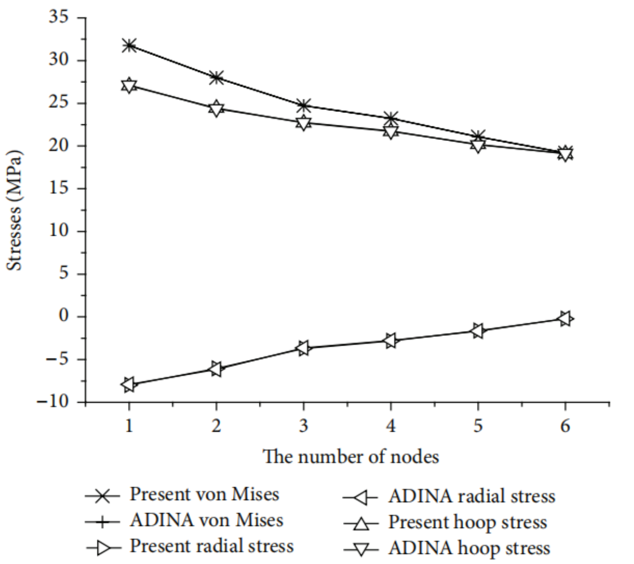


Figure 9: Comparison of stress [Structural Validity] (Kwon, Y., Lee, D., & Yun, T. (2016))

6. CONCLUSION

In this study, ANSYS 19.2 is used for analysis of the structural and thermal properties by using the finite element method (FEM) to solve the governing equation and estimate results. In the structural and thermal analysis, the results are extracted after applying all the boundary conditions properly. All solved values, such as total temperature, deformation, structural error, stress, and strain are at the expected level. ANSYS workbench is used for the solution, which is a reliable and advanced simulation software for analyzing 3D mechanical structures.

Also, this study can enrich the research field of the VVER-1200 reactor and can help to find more details about the structural analysis of the core of the VVER-1200 NPP, which is new and important from our country's perspective. Also, the structural and thermal analysis of the fuel rod in the VVER-1200 nuclear reactor, conducted using ANSYS software, provides valuable insights into the performance and safety of the reactor's core. Since in the result mention that for nuclear components safety factor above 2 is excepted and we find the safety factor of this study is around 8.81.

In the conclusion, the structural and thermal analysis of the fuel rod is reliable because the error is in accepted level. The analysis confirms that the fuel assembly is structurally sound and has a significant safety margin to withstand operational loads without failure. And the safety margin results which is the safety factor and the value is 8.81 also total deformation is in limit which is discussed in the results. Overall, this study contributes to a deeper understanding of the fuel rod's behavior, supporting the reactor's long-term reliability and safe operation.

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AUTHOR DECLARATION

The authors declare that there is no conflict of interest

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