Structural Analysis of Nuclear Fuel Element with Ansys Software

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***Abstract*— Prediction of thermo-mechanical behavior of fuel element in a nuclear power plant is very much necessary to design and prevent the failure of element in operating conditions. In this work an attempt is made to find out the structural analysis of a fuel element in operating condition of a nuclear power plant. For structural analysis, first thermal analysis is carried out to find out the non-uniform temperature in fuel element, which is then used as a thermal load in structural analysis. The stress, strain and displacement of the fuel element are found out for different thickness of gap in between pellet and claddings. Then analysis is also carried out for different values of heat transfer co-efficient. From structural analysis it is found that though it is expanding due to thermal effects, it is within limit. A nuclear fuel rod of boiling water reactor is considered. Standard thermo-physical data are considered for study purposes. Separate governing equations are considered for pellet and cladding. Heat generation is considered only in the pellet. Since the height of the pellet is much larger than the diameter of rod axial distribution of temperature is neglected. That means heat transfer equations comes to one-dimensional form. The Finite Element Method (FEM) is used in ANSYS to descretize the computational domain. A coupled analysis is carried out to find out the temperature, heat flux, stress strain and thermal expansion.**

***Index Terms*— clad, fuel element, gap, PCI, pellet, thermal expansion.**

NOMENCLATURE

$\rho_F =$ Density of fuel (kg/m^3)

$C_{PF} =$ Specific heat of fuel (J/k$_g$. K)

$T_F =$ Fuel pellet temperature (K)

$t =$ Time (sec)

$r =$ Radius (m)

$k_F =$ Thermal conductivity of fuel (W/m. K)

$\dot{g} =$ Heat generation (W/m^3)

$\rho_C =$ Density of clad (kg/m^3)

$C_{PC} =$ Specific heat of clad (J/k$_g$. K)

$T_C =$ Clad temperature (K)

$k_C =$ Thermal conductivity of clad (W/m. K)

$R_F =$ Radius of fuel (m)

$R_C =$ Clad inner radius (m)

$R_W =$ Clad outer radius (m)

$h_{gap} =$ Heat transfer co-efficient for gap (W//m^2. K)

$h_{wall} =$ Heat transfer co-efficient for coolant (W//m^2. K)

I. INTRODUCTION

Nuclear power plants today supply up to 7% of the world's primary electricity. They can be two or three times more expensive to run than coal-fired power plants, and have themselves become a kind of radioactive waste by the time they are closed down. In the mean time nuclear rod which release the energy for generation of power using heat exchanger and turbine has tendency to failure due to many more reasons, and one of the most important phenomena is due to thermal characteristics or behavior of the nuclear fuel rod, when the time passes pellet swells and sometimes come in contact with cladding hence need is arises to predict the temperature variation in the nuclear fuel rod so that study and analysis can be done on the predicted result and to design the rod so that heat transfer is optimized to reduce the failure of rod. That’s why fuel rod is considered here for prediction of temperature distribution and analysis purpose. Nuclear fuel rod of a BWR is considered which is made of UO2 (pellet) and Zirconium-2 (cladding). Initially helium gas is consists in between pellet and cladding. Main objective of this work is prediction of temperature distribution from a nuclear fuel rod using ANSYS. Then analysis of the computational data under varying boundary conditions and thickness of gap. For analysis purpose a fuel element of BWR is considered here. A brief description of nuclear power plant is written in the following pages. It is summarized here how this fuel element is made and what are the main problems which insists me to work on this. After many studies around the world in various nuclear power plants it was found that cladding and outer surface of the pellet are swelling and in few cases cracking is there. There are many reasons for which fuel element may fails, amongst them pellet-cladding interaction is prevailing one which causes mainly due to the thermal expansions of the pellet and cladding. So the need arises to predict the temperature distribution and to find out the stress, strain and displacement of the pellet and cladding.

II. REVIEW OF LITERATURE

Numerous scientist and researchers have done many more works on nuclear fuel element around the world. Maximum work is on prediction of temperature distribution from nuclear fuel element. In few cases mechanical analysis were carried out. Some works were done on prediction of temperature distribution using different numerical method. Few works were done on stability analysis. Brief histories of few works are gathered here. The accurate prediction of the fuel rods’ temperature in light water reactor cores is required during both the uncovery and refloods phases of severe accidents. An accurate prediction of the rod's temperature is needed and a solution of the heat conduction in the fuel rod is required. The numerical solution of the transient heat conduction in the fuel rods of light water reactors is studied for the purpose of selecting an efficient, economic and accurate solution procedure to be used in fast running codes in the work of S.M. Ghiaasiaan, A.T. Wassel and J.L. Farr. In their work radial heat conduction equations are presented in terms of ordinary time dependent differential equations that are subsequently integrated. Comparisons were carried out using the lumped capacitance method, the method of lines, and the integral method with parabolic temperature profiles. The temperature-time history of the fuel pellet and cladding, predicted by the various methods, is summarized for slow and fast transients representing the uncovery and reflood phases of a loss of coolant accident. They found that the integral method, with parabolic temperature profiles in both the fuel pellet and cladding, was the most economic and gave sufficiently accurate predictions [1]. A finite difference technique is used here for obtaining the solution of the in-depth heat conduction. The accuracy and computational cost of the procedure are directly proportional to the number of nodes selected. Under slow transient conditions, the problem has been simplified by assuming flat temperature profiles in the fuel pellet and cladding. The radial heat conduction in fuel rods was investigated for the purpose of selecting a fast and accurate solution procedure to be used in fast running system codes. The formulation was carried out for the lumped capacitance method (2 nodes), the integral method with parabolic temperature profiles, and the method of lines (arbitrary number of nodes). The numerical tests have led to the following conclusions:

1) For slow transients the difference between the predictions of all the methods was relatively small. For swollen fuel pellets, however, the predictions by the integral method are more accurate than the lumped capacitance method.
2) For fast transients typical of core reflooding and for an unswollen fuel pellet, the predictions of the lumped capacitance technique and the integral method are relatively close, with the integral method predicting the clad surface temperature more accurately. The method of lines requires many radial nodes in order to give predictions more accurate than those of the parabolic integral method. For a swollen fuel pellet, the predictions of the lumped capacitance method are grossly inaccurate. The integral method predicts the clad surface temperature with the same accuracy as the method of lines with twenty radial nodes. In conclusion, the integral method with parabolic temperature distributions in the clad and fuel pellet is relatively simple, and computationally economic. It is as fast as the lumped capacitance method and as accurate as the method of lines with many radial nodes[1].

Clarisa, Renota Jian su (2000) has analyzed an improved lumped parameter approach to find out transient heat conduction in a nuclear fuel rod. The lumped parameter approach has been widely used in the thermo hydraulic analysis of nuclear reactors. As in the analysis of other complex thermal systems, this classical approach is extremely useful and sometimes even mandatory when a simplified formulation of the transient heat conduction is sleeked. Together with the neutron point kinetics model, the lumped parameter approach for fuel rod heat conduction is essential in the simplified models of pressurized water reactors (PWRs) and in real-time simulators of nuclear power plants [2].The circumferential symmetry is assumed with the heat transfer through the gap modeled by a heat transfer coefficient. Here Hermite approximation for integration is used to obtain the average temperature and heat flux in the radial direction. Significant improvement over the classical lumped parameter formulation has been achieved. Recently, the dynamics of chaotic instabilities in boiling water reactors has aroused increased interests. In such studies, the lumped parameter approach has been the unique option in the fuel dynamics models. For example, Rao et al.[3] performed a linear stability analysis in the frequency domain to study the basic mechanism of coupled nuclear-thermal instabilities in a boiling channel, using a one-node lumped parameter model for the fuel dynamics. Even with the simplest fuel dynamics model, they found that the fuel-time constant was one of the parameters determining the density-wave instability. Chang and Lahey [4] used one-dimensional homogeneous equilibrium assumptions for diabatic two-phase flow, a one-node lumped parameter approach for heated wall dynamics, and neutron point kinetics for the consideration of nuclear feedback in a boiling water reactor (BWR) loop. They found that a boiling channel coupled with a riser could experience chaotic oscillations. Linet al.[5] found a strip of limit cycle oscillation of a nuclear-coupled boiling channel with a two-node lumped parameter model for the fuel dynamics, where one node was for the fuel and the other for the cladding. The proposed fuel rod heat conduction model can be used in stability analysis of BWR, simplified model of PWR or real-time simulator of nuclear power plants. [10]. As an inherent limitation of the lumped parameter approach, moderate to low temperatures gradients within the region are assumed, which through the associated problem parameters, governs the accuracy of such approximate formulations. As a rule of thumb, the classical lumped parameter approach, where uniform temperature is assumed within the region, is in general restricted to problems with Blot number less than 0.1. In most nuclear reactor engineering problems, the Biot number is much higher. In other words, the moderate to low temperature gradient assumption is not reasonable in such applications, thus more accurate approach should be adopted. Cotta and Mikhailov [6] proposed a systematic formalism to provide improved lumped parameter formulation for steady and transient heat conduction problems based on Hermite approximation for integrals that define averaged temperatures and heat fluxes. This approach has been shown to be efficient in a great variety of practical applications [7-9].

K.M.Pandey and Amit Kumar [11] worked on Studies on Base Pressure in Suddenly Expanded Circular Ducts: a Fuzzy Logic Approach and their findings are given below. An optimum L/D ratio is evaluated in the present study using fuzzy-set theory. The fuzzy set based methodology could easily consider many attributes concurrently, while deciding the specifications of the suddenly expanded supersonic fluid flow through a straight circular duct. The methodology can be easily extended to a situation involving diverse conflicting objectives. This study can be extended to different nozzles having different geometries with variations in Mach numbers, primary pressure ratio and area ratio. It is observed that L/D ratio is 6 for base pressure for Mach numbers of 1.58, 1.74, 2.06 and 2.23 , which is in very close agreement with the experimental results cited in the literature. This has been discussed with fuzzy logic as a tool for three area ratios 2.89, 6.00 and 10.00. The primary pressure ratio has been varied from 2.10 to 3.48 and L/D ratio has been varied from 1 to 6. From this analysis it is observed that L/D ratio 6 is the optimum needed keeping in view all the parameters like wall static pressure and pressure loss including base pressure.

III. Mathematical Modelling of Equations and Solutions

Power reactor cores are composed of cylindrical fuel elements that contain fuel pellets, gap and cladding. Our goal will be to calculate the temperature drop from the centerline of the fuel where the maximum temperature occurs to the surface of the clad in terms of the various physical properties of the fuel elements, while the fuel element geometry, thermal properties, and physical characteristics have been known. Further we consider the thermal analysis of a BWR fuel element. The general procedure is to solve the classical heat conduction equation.

A. *Definition of Problem*

A fuel rod consists of UO2 pellets in a Zircaloy-2 cladding tube and a small gap between the surface of the fuel pellets and the inside surface of the cladding. The heat generated by nuclear fission is conducted through the fuel rod and convected by the surrounding coolant in the flow channel. A radial heat conduction model is used to calculate the fuel heat flux and temperature distribution. The thermal energy storage and transport is modeled with the following

B. *Basic Assumptions*

1) Any axial heat transfer is neglected; this is justified because the length of the rod is much greater than its diameter and also because the pellets' interfaces offer a high thermal resistance.
2) The only active heat transfer process is conduction: convection due to a gas flow through the pellet cracks is neglected. This is a good assertion because neither the gas amount, nor its flow speed reaches the level required modifying appreciably the temperature field.
3) The strain effects on the temperature field in the fuel is not taken into account.
4) The gap heat transfer coefficient, which generally depends on the gap width, the temperature at the fuel outer surface and at the cladding inner surface, the inner gas pressure, and the mean temperature, is modeled by a given function of time. The latter can also include effects arising from radiation.
5) In the same way, the heat transfer coefficient in the film between cladding and coolant is also approximated by a given function of time.

The fuel pellets and the cladding receive separate descriptions so that their temperature distributions are obtained separately and independently. The reasons for adopting an uncoupled description are the following. First, it is believed that the computing time will be appreciably reduced. Moreover, this uncoupling allows the use of the same subprograms in both the fuel and the cladding. Finally, the thermal properties of the fuel rod are given within a certain error percentage; therefore, it would be illusory to search for a highly accurate description. The temperature distribution inside the fuel and the cladding is found by solving the set of transient heat equations.

C. *Properties of UO2 and cladding material*

TABLE 1. Technical Data Considered for Analysis

Pellet material: uranium oxide (UO2).	
Clad material: zircaloy-2.	
Gap =consist of Helium	
Pellet radius	0.004782m
Gap thickness	0.000193m
Clad outer radius	0.005582m
Clad inner radius	0.004975m.

TABLE 2. Constant Property for Pellet

Thermal conductivity	29 W/M.K
Specific heat	268J/Kg.k.
Density	11000kg/m3
Heat generation	8x108 w/m3
Modulus of elasticity	1.82x1011Pa
Poisson’s ratio	0.295
Thermal co-efficient of expansion	10.8x10-6

TABLE 3. Constant Property for Cladding

Thermal conductivity	13 W/M-K
Specific heat	330J/Kg.k.
Density	6500kg/m3
Heat generation	None
Modulus of elasticity	0.99283x1011Pa
Poisson’s ratio	0.33
Thermal co-efficient of expansion	20x10-6

TABLE 4. Gap

Thermal conductivity for gap material.	50w/m-k
Pressure	10 atm of He

TABLE 5. Coolant

Heat transfer coefficient	=40000w/m^2-k.
Temperature	=400^{0}K.
Pressure	=7171087 Pa.

D. *Governing Equations*

The transient heat conduction equation for pellet neglecting axial conduction is given by:

$$\rho_F C_{PF} \frac{\partial T_F}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_F \frac{\partial T_F}{\partial r}\right) + \dot{g}$$

and the governing heat equation for cladding is given by:

$$\rho_C C_{PC} \frac{\partial T_C}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_C \frac{\partial T_C}{\partial r}\right)$$

The boundary condition and initial condition for the nuclear fuel rod problem is given by:

$$\text{at} \quad r = R_F : -k_F \frac{\partial T_F}{\partial r} = h_{gap}\left[T_F(r = R_F) - T_C(r = R_C)\right]$$

at $\quad r = R_C : -k_C \dfrac{\partial T_C}{\partial r} = h_{gap}\left[T_F(r = R_F) - T_C(r = R_C)\right]$

at $\quad r = R_W : -k_C \dfrac{\partial T_C}{\partial r} = h_W\left[T_F(r = R_W) - T_f\right]$

E. Element types and meshing

In Finite Element Analysis (FEA) different types of element are generally used for different purposes, in this work also separate element type is used for particular application. For 2-D thermal analysis PLANE 77 is used and for 2-D structural analysis PLANE 82 is used.

1) PLANE 77- 2-D 8-Node Thermal Solid

PLANE77 is a higher order version of the 2-D, 4-node thermal element (PLANE55). The element has one degree of freedom, temperature, at each node. The 8-node elements have compatible temperature shapes and are well suited to model curved boundaries. The 8-node thermal element is applicable to a 2-D, steady state or transient thermal analysis which is shown in fig.1.

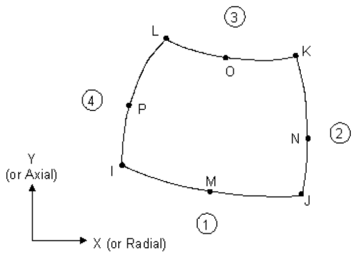

Fig.1 PLANE77 Geometry

2) PLANE82- 2-D 8-Node Structural Solid

Fig.2 shows PLANE82 a higher order version of the 2-D, four-node element (PLANE42). It provides more accurate results for mixed (quadrilateral-triangular) automatic meshes and can tolerate irregular shapes without as much loss of accuracy. The 8-node elements have compatible displacement shapes and are well suited to model curved boundaries. Eight nodes having two degrees of freedom at each node define the 8-node element: translations in the nodal x and y directions. The element may be used as a plane element or as an axisymmetric element.

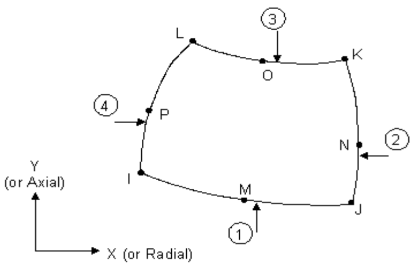

Fig.2 PLANE82 Geometry

In this work PLANE82 is considered because it is necessary to take similar element for coupled analysis.

F. Meshing the geometry

After choosing the element type meshing is done for 3-D and 2-D geometry an elemental size length of 0.0004m and 0.0001m is considered for meshing purposes. Later on 0.000050m and 0.000025m is considered for grid independence test.

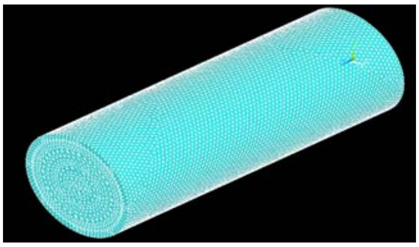

Fig.3 3-D meshing geometry of fuel rod

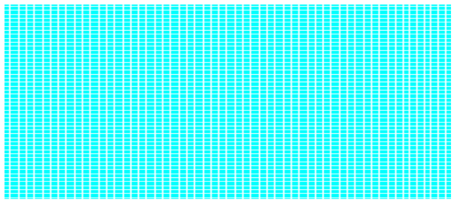

Fig.4 2-D meshing geometry of fuel rod

The meshing geometry of 3-D and 2-D are shown in fig. 3 and 4.

G. Apply boundary conditions and loads

For thermal analysis to find out the temperature distribution and heat flux from fuel element a heat transfer co-efficient of 40000 W/m^2- ^0K. And a temp of 400K is applied at the right boundary. A heat generation rate of $4x10^8$ W/m^3 is applied only in the pellet area. Heat generation is neglected in cladding. First steady state heat conduction solution is carried out to find out the initial temperature at the pellet surface, cladding inner and outer surface, which was later, used in transient heat conduction analysis. For other boundary heat transfer co-efficient are taken 0 that means no heat conduction from at those boundary. Fig. 5 shows the computational domain with boundary conditions.

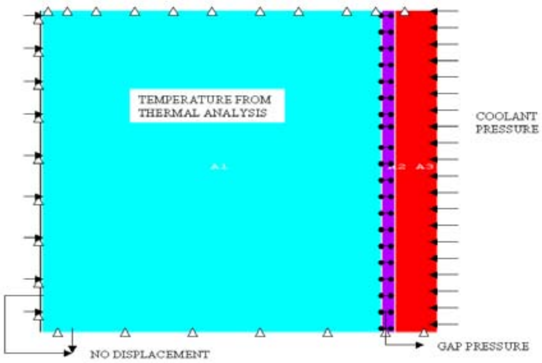

Fig.5 boundary conditions and loads for structural analysis

For structural analysis to find the thermal stress, strain and expansion of the pellet and cladding, temperature distribution that was found from thermal analysis is considered as thermal load throughout the geometry. Coolant pressure is applied at the right boundary. And within the gap an initial pressure is applied.

IV. Results and Discussion

The objective of this work is to carry out structural analysis of the nuclear fuel element in operating condition using thermal analysis. From the history of nuclear reactor, in few cases it was found that after certain period fuel elements are swelling. That means structural displacements are taking place. This is a serious condition to be taken into consideration because if the pellet section or cladding section expands due to the temperature difference and pressure difference it was found that in few cases there is a chance to failure of the fuel element. And as we all know what will be happen after that. There is chance of different types of failure of fuel element, amongst them PCI (pellet-cladding Interactions) is important one. So here in this work an attempt is taken to analyze the thermal and mechanical behaviors of the fuel elements.

A. Early prediction of temperature for structural analysis

A heat conduction analysis is carried out to know the non-uniform temperature in the fuel element. From the fig.6 it is found that solution is converges after some period.

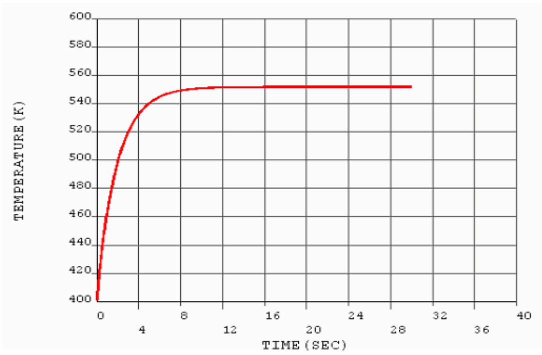

Fig. 6 temp with time for node 14452

It is found that the maximum temperature in the pellet center is 703K. It is found that initially temperature rises gradually then steady state condition attains. The resulted non-uniform temperatures are then used in structural analysis as a thermal load.

B. The results of coupled analysis

In coupled analysis the thermal and structural analysis are coupled to find out the displacement, stress, and strain at different sections of the fuel element. Where the temperature distribution, which found from thermal analysis, is used in structural analysis to find out various parameters which is due effect of temperature variations.

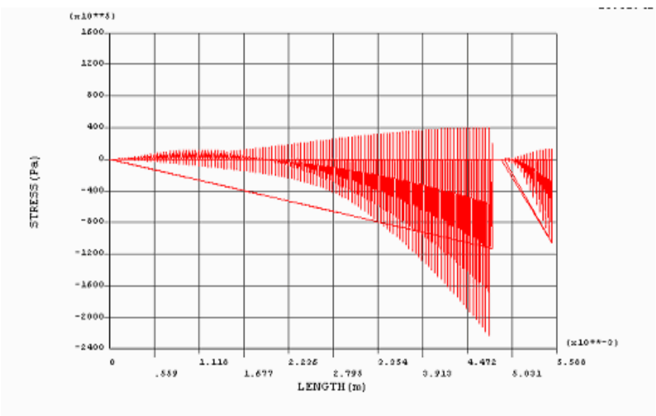

Fig. 7 stress distribution in fuel element

Coolant pressure and gap pressure are considered as structural loads and temperature as thermal loads. Fig.9 shows the thermal expansion of the pellets and cladding. From fig. it shows that the maximum displacement occurs at the outer surface of pellets and in the claddings.

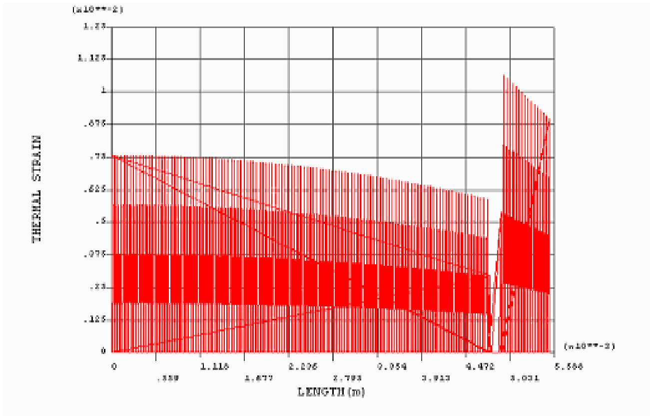

Fig.8 Strain distribution in fuel element

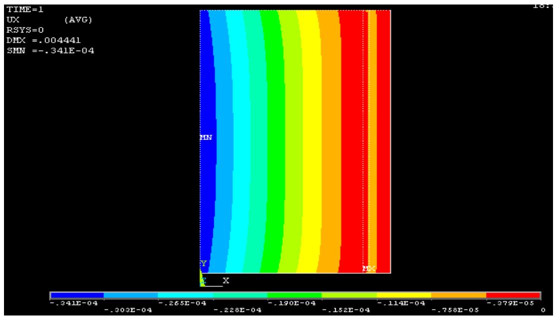
Fig.9 displacements of different sections of nuclear fuel rod

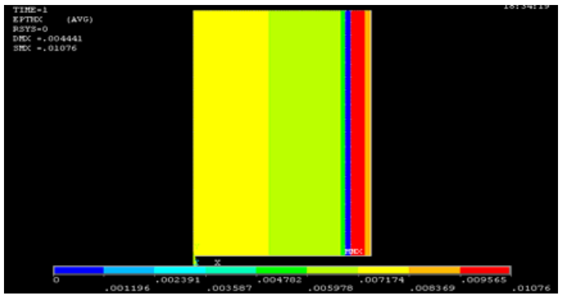
Fig.10 Thermal strain at different sections of nuclear fuel rod

C. Stress, strain and thermal expansion due to thermal effects

Fig. 7, 8 and 10 shows the thermal stress and strain in fuel element. From fig.7 it is found that the maximum stress is developed near the pellet surface. For different thickness of gap it is seen that stress distribution is not changing so much. For strain from fig.8 it shows that the maximum strain occurs at clad sections. But in this work it is found that pellet –cladding interactions is not happening for considered design.

V. CONCLUSION

Early prediction of thermo-mechanical behavior is necessary to prevent the failure of a nuclear fuel rod. In this work a coupled analysis is carried out to find the thermal stress, strain and thermal expansion of a nuclear fuel element. In operating condition of a nuclear power plant there is chance to fail fuel elements. A fuel elements may fails due to various reasons, out of them Pellet-Cladding Interactions (PCI) is prevailing one. In this work a BWR fuel element is considered, which is made of uranium oxide (UO_2) and surrounded by a zircaloy cladding. The gap between pellet and cladding initially filled with helium gas. An attempt is made to see what is happening to the fuel element due to thermal differentiation and pressure difference between gap and coolant. From this analysis it is found that though the pellet and cladding expanding due to thermal effect but it is within safe zone. Stress developed also in permissible limit.

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