THERMO-MECHANICAL BEHAVIOUR OF PRESSURE TUBE OF A PHWR DURING POSTULATED SEVERE PLANT CONDITION

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ABSTRACT

The objective of nuclear safety research programme is to develop and verify computer models to accurately predict the behavior of reactor structural components under operating and off normal conditions. One of the postulated accident scenarios for heavy water moderated pressure tube type of reactors (PHWRs) is station blackout with un-availability of heat sinks. In that case, even though the reactor is tripped, the decay heat may not be removed adequately due to steam generator inventory boil-off and later, inventory depletion of primary side. Since the emergency core cooling system as well as moderator cooling system is unavailable, this will result in high temperature of the fuel pins. In PHWRs, the fuel assembly is surrounded by pressure tube, an annulus insulating environment and a concentric calandria tube. During this scenario, the pressure tube is expected to sag as well as balloon out and come in contact with outer cooler calandria tube to dissipate away the heat generated. The amount of heat thus expelled significantly depends on the thermal contact conductance and the nature of contact between the two tubes. This in turn, controls the deformation of the pressure tube thus introducing inter-dependency between the thermal and mechanical contact behavior.

The objective of this paper is to provide insights into this thermo-mechanical behavior by computational studies and to understand the role of underlying parameters (such as material constants, contact heat transfer models and coefficients, boundary conditions, etc.) that control the tube deformation and further damage progression. The thermal hydraulic boundary conditions have been generated for the above scenario using the code RELAP5/MOD3.2. The deformation characteristics of the pressure tube was modeled using finite element based program. Experimental data of pressure tube material, generated earlier, were fit into a multi-linear curve to model and examine the role of nonlinear stress-strain laws in the finite element analyses.

INTRODUCTION

The Pressurized Heavy Water Reactors have engineered safety systems to restore the reactor to safe states in the event of accidents. Most of the safety systems have diverse and redundant features, for example the SDS1 and SDS2, ECCS and double containments. The purpose is to enhance the availability of these safety systems so that failure of one of them does not lead to progression of accidents. Besides, the reactors have several multiple barriers to arrest the release of activity, if a hypothetical accident progresses with the failure of the safety systems.

In spite of all these, an accident of further low probability can be cast beyond the acceptable design basis envelope, which can progress to severe accident. Severe accidents are beyond design basis accidents which arise due to multiple failures of safety systems and lead to significant core degradation. Irrespective of the initiating events, a severe accident occurs only when the ratio of heat generation rate to heat removal rate exceeds a certain threshold for a prolonged duration. The fuel is unable to get cooled and gets damaged leading to its melt down. The severity of such an accident depends on the nature and extent of core damage during the accident progression. Of course, these accidents are very low probable ones, but certainly they have high risk in terms of consequences to the plant personnel and public. In view of this, most of the new generation reactors are designed with built-in severe accident management strategies. The nuclear utilities also plan to extend or build additional safety features (SAMG) in existing plants in order to enhance the defense-in-depth of these plants so as to cope with such severe accidents.

Several beyond design basis scenario can be postulated, which can lead to such melt-down accidents. It can be initiated with a Limited Core Damage Accident (LCDA) which can involve a single or few channels. For example, a feeder break can result in overheating of the fuel in the affected channel, or a LOCA with loss of ECC could lead to widespread fuel damage. In both cases, if moderator is available as a secondary heat sink, it can prevent failure of fuel channels and core degradation. For an accident to proceed to significant core damage, there must be failure of multiple fuel channels. This is possible with the loss of moderator as heat sink. The Severe Core Damage Accident (SCDA) may be initiated from an LCDA and progresses to significant core damage with the non-availability of moderator as heat sink. Hence, the early stage of SCDA in HWRs is different from that of a LWR in that the presence of heavy water moderator slows down progression of core disassembly.

However, the course and time of progression of accident depends to a great extent on the thermo-mechanical behaviour of pressure tube following fuel heatup. The pressure tube may balloon or sag or undergo rupture depending on the coincident pressure and temperature conditions and can lead to different scenarios and alter the progression time.

In the present study, a numerical analysis is performed to evaluate the thermal loads and the stresses due to that in the pressure tube following the above accident scenario in case of a HWR. The deformation characteristics of the pressure tube has been modeled using finite element based program. Experimental data of pressure tube material, generated earlier, were fit into a multi-linear curve to model and to examine the role of nonlinear stress-strain laws in the finite element analyses.

PHENOMENOLOGY OF ACCIDENT PROGRESSION WITH STATION BLACK OUT AS AN INITIATING EVENT

The Station Black Out (SBO) has been considered as the postulating initiating event with unavailability of various safety systems for severe accident progression. The scenarios considered in the present analysis are as follows:

- Class III and IV power unavailable for sufficiently long time
- Moderator cooling and shield cooling are unavailable
- Shut down cooling not available
- Main and auxiliary feed water are unavailable
- Steam generator main steam safety valve (MSSV) are available, they open and close at set point to relieve pressure
- Turbine main stop valves are closed after accident initiation. The valve closure time is assumed to be 20 s.
- ECCS (high, medium and low pressure) unavailable
- SG Crash cool-down not credited
- Local Air Coolers (LACs) not available
- Air-operated Atmosphere Steam Discharge Valves have no back-up air and springs.

With Unavailability of Class IV and Class III power, the Primary Circulating Pumps (PCP) are tripped. Immediately the reactor gets tripped. Though the pumps are not working, because of the flywheel inertia and availability of heat sink in secondary side, decay heat is removed from the core by thermosyphoning. Since the main steam isolation valve gets closed, the secondary side is boxed up and steam generator (SG) pressure starts rising. When it crosses the relief valve set point, the relief valve opens and relieves the steam into environment. As a result of which, the inventory in the SG continuously depletes. When the secondary side inventory is boiled off, the heat sink is lost and primary side pressure starts rising. When primary pressure exceeds the liquid relief valve set point, it opens and the primary side inventory is released to the containment. As the inventory from primary side is lost, the fuel, clad and pressure tube (PT) temperature start rising. At a particular temperature, the PT balloons and touches calandria tube (CT). This causes heat removal through PT and CT to the moderator. The moderator temperature starts rising and ultimately it starts boiling. Increase in calandria pressure causes the rupture disc to blow off releasing the moderator to the containment depleting moderator level in calandria. This causes uncovery of the channels. Due to unavailability of moderator as a heat sink, the fuel, clad, PT and CT temperature start rising very fast which leads to channel disassembly.

THERMAL HYDRAULIC ANALYSIS OF PHWR DURING SBO

The Primary and Secondary Heat Transport System including inlet headers, feeders, outlet headers, feeders, steam generators primary side tubing and reactor coolant pumps of the PHWR has been modeled in RELAP5/Mod3.2 as shown in Fig 1. Moderator system has also been modeled consisting of Calandria as a pipe component and considering heat loss through PT and CT to the moderator. The secondary system consists of stem generator (SG) downcomer, riser, steam separator and steam dome. The turbine has been simulated with a Time Dependent Volume (TDV) so as feed water pumps.

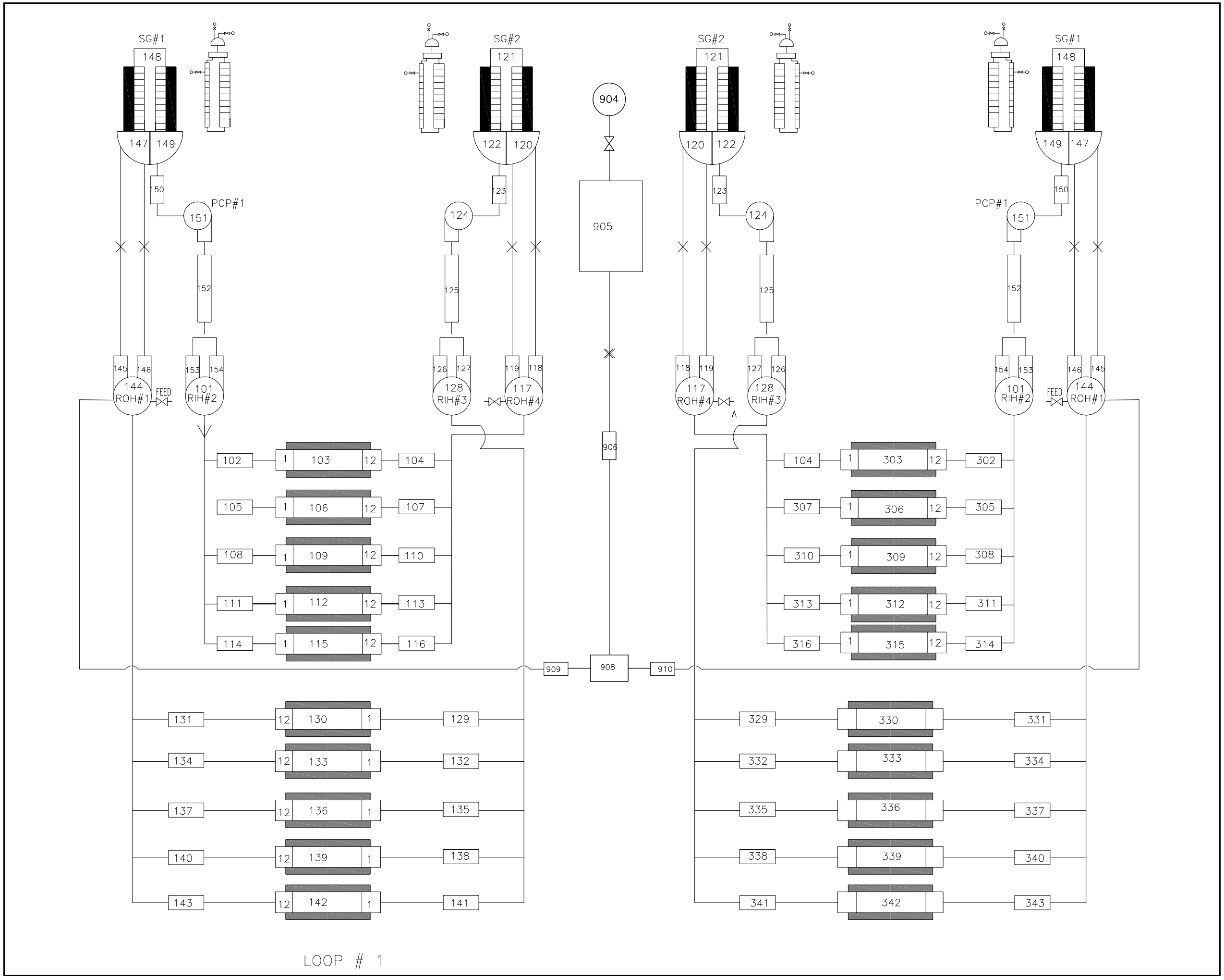

Fig. 1: PHWR Nodalization in RELAP5/MOD3.2

At time t = 0, the SBO is initiated. The boundary conditions are imposed as follows

- All the PCPs tripped
- Reactor is tripped and it is generating decay power
- Feedwater flow rate reduced to zero
- Moderator circulation pump tripped
- Main steam isolation valve is closed

With these boundary conditions, the thermal hydraulic behaviour of the plant was studied. As a result of tripping of pump and reactor shutdown, slight depressurization of primary system is observed. The pressure in the ROH and RIH are shown in Fig 2a. Corresponding secondary side pressures are shown in Fig 2b. Due to box-up of the secondary side, the secondary side pressure rises and is relieved by opening of MSSVs. As a result of which, the inventory in the SG continuously depletes as shown in Fig. 3.

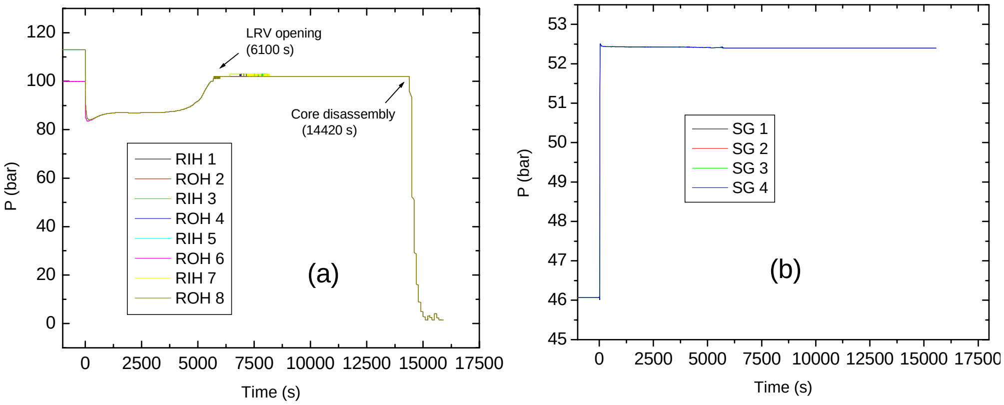

Fig. 1: System pressures a) Primary side b) Secondary side

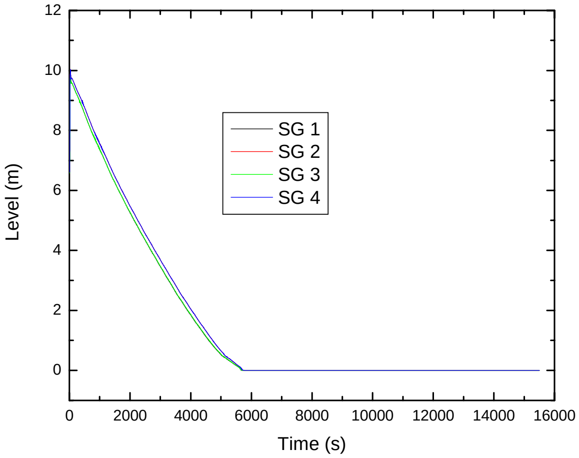

Fig. 3: SG water level

At around 6000 s, the inventory in all the four SG's is completely boiled off. As a result, the primary side pressure starts rising to the set point of the LRVs at around 6100 s as shown in Fig 2. When the pressure increases above the LRV setpoint, it opens and discharges the inventory into the containment. During this time, the inventory in primary side depletes. The mass flow rate in the primary side is shown in Fig 4. After tripping of the PCPs, the mass flow rate reduces due to combined action of the pump coastdown and built up of thermosyphon. However, after nearly 6500 s, the thermosyphon flow rate is very small because of no heat removal from secondary side. Due to this, channels get completely voided.

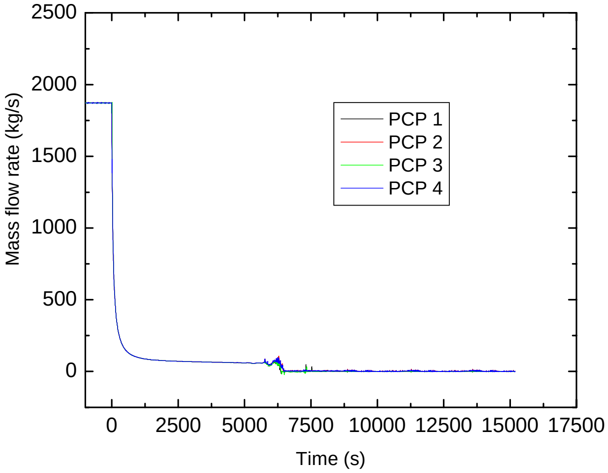

Fig. 4: Primary side mass flow rate

As a result, the fuel and clad temperature start rising. The temperatures of Fuel, clad, PT and CT for low power (Channel 1) and hottest (Channel 3) are shown in Fig 5a and 5b

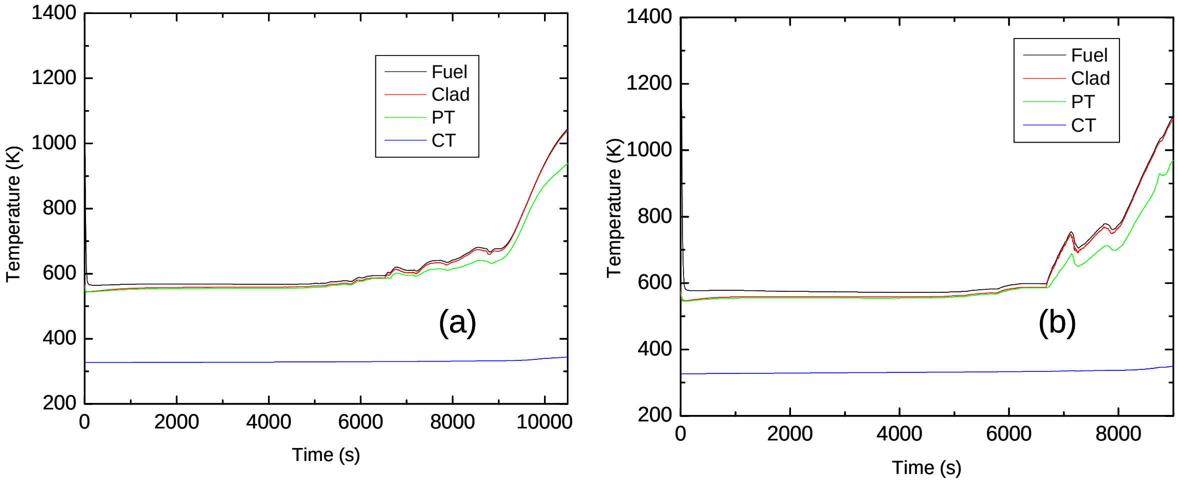

Fig. 5: Temperatures a) Channel 1 b) Channel 3

Since the channels are completely voided and there is no flow, the clad surface temperature is found to start rising after 6000 s. After around 8000 s, the clad temperature starts rising significantly in the hottest channel (Channel 3, Fig 5b) whereas, for low power channels, it starts rising after 9000 s. At around 9200 s, the temperature of PT rises to about 900 K for channel 3. Using these temperature and pressure boundary conditions, the deformation of pressure tube has been studied.

DEFORMATION OF COOLANT CHANNEL ASSEMBLY AT HIGH TEMPERATURE

The coolant channel assembly of PHWR consists of a zirconium alloy pressure tube (PT) which houses the natural uranium fuel bundles. Fig. 6 shows the schematic of a coolant channel assembly of PHWRs.

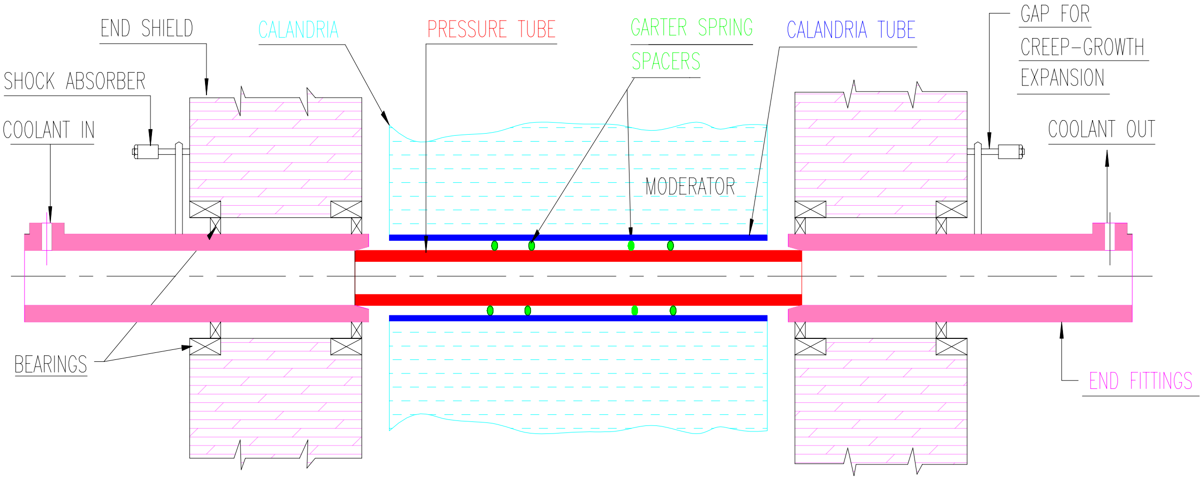

Fig. 6: Schematic of Coolant Channel assembly

This section covers the work undertaken to model the non-linear sag behaviour of the Zr-2.5 Nb alloy pressure tube and Zr-2 calandria tube (CT) coolant channel considering it as a 3D-solid structure using the finite element based computer code. Fig. 7 illustrates the PT-CT assembly at the outer end. As the geometry, loading, material and boundary conditions are symmetric about the central cross section as well as the longitudinal section passing through the central axis of the channel. Only one quarter of the full PT-CT assembly has been modeled. Symmetric boundary conditions have been applied at the respective sections by restraining the appropriate degrees of freedom. Internal pressure and nodal temperatures were applied to the pressure tube material. Fig. 8 illustrates the finite element model of the coolant channel assembly showing symmetric boundary condition.

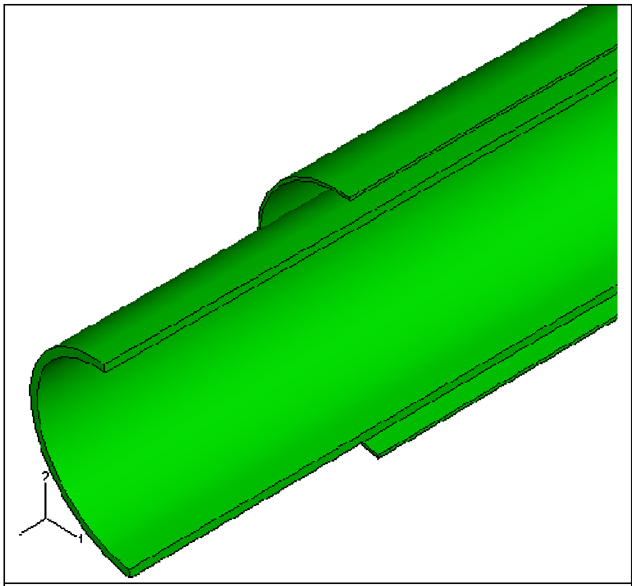

Fig.7 Solid Model of PT-CT assembly

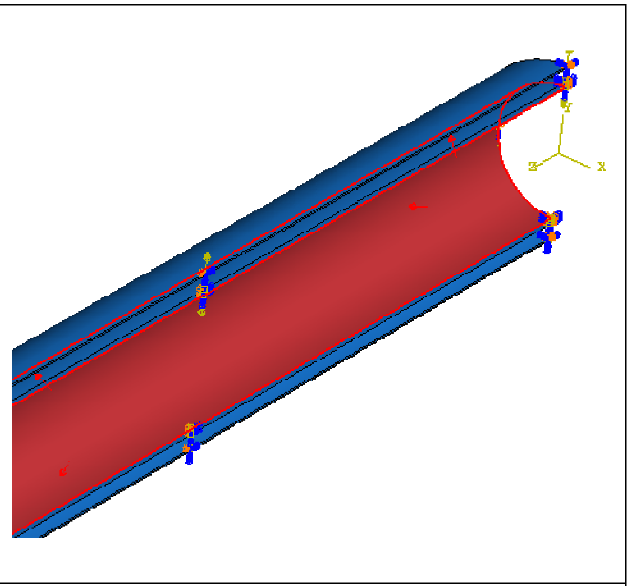

Fig. 8 Application of Symmetric Boundary Condition

In this model, the mechanical properties of the zirconium components have been considered as isotropic. The longitudinal properties have been considered as isotropic. Since the sag behaviour primarily involves the longitudinal properties, it is the nearest approximation. The high temperature tensile properties have been taken from a recent study of Zr-2.5 Nb alloy PT material for Indian reactors [2]. It was shown that the material properties at high temperature become location and direction independent [2] and [3], hence the assumption of using isotropic properties is valid in this case. Figs. 9 and 10 illustrate the flow behaviour of PT material up to operating temperature regime as well as at high temperature.

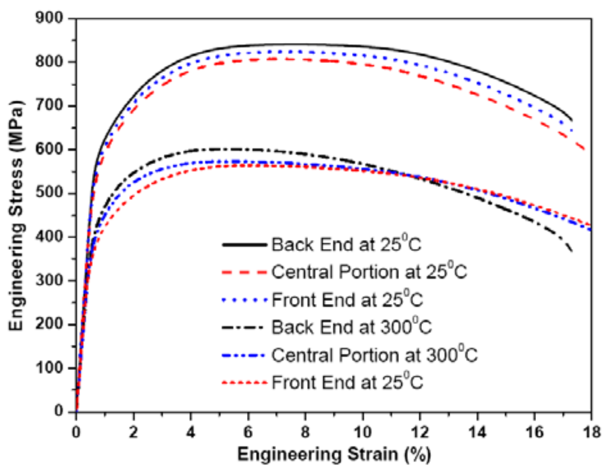

Fig.9: Flow curves for Zr-2.5Nb alloy PT material up to operating temperature regime

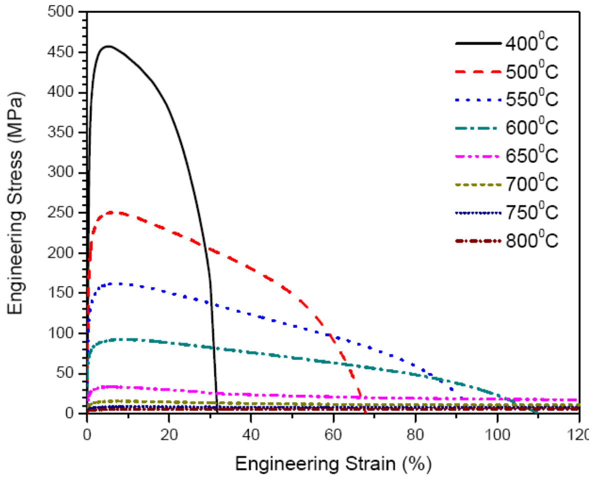

Fig.10: Flow curves for Zr-2.5Nb alloy PT material at higher temperatures

The pressure tube and calandria tube were meshed using 20 noded quadratic hexahedron elements. Four garter spring spacers have been modeled at the design locations. At each such location, a non-linear gap element has been used to simulate the garter spring coil along with the assembly clearance. The finite element mesh is illustrated in Fig. 11. A non-linear large deformation elasto-plastic contact analysis was carried out to find out the time of first contact between PT and CT. It was found that at around 580 ºC, the pressure tube makes its first contact with the calandria tube. The contacting channels are shown in Fig. 12. As the channels are loaded with fuel bundles, sagging becomes the predominant mode of deformation.

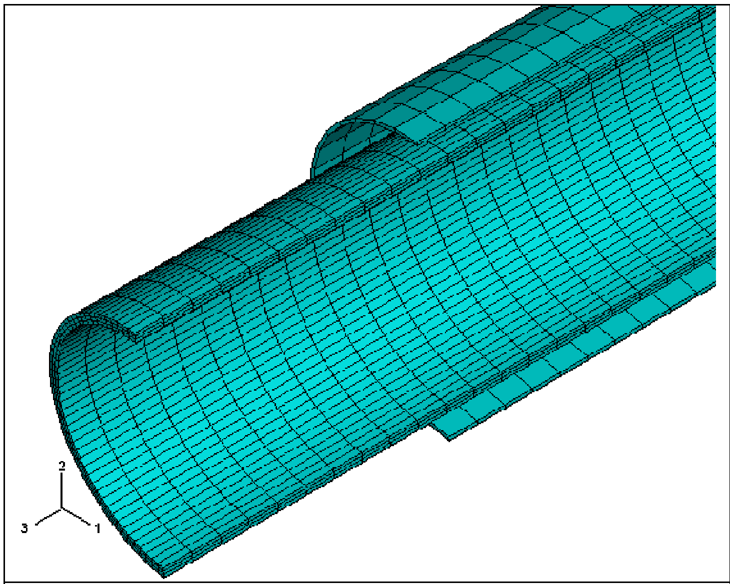

Fig.11 Finite Element Mesh of PT-CT assembly

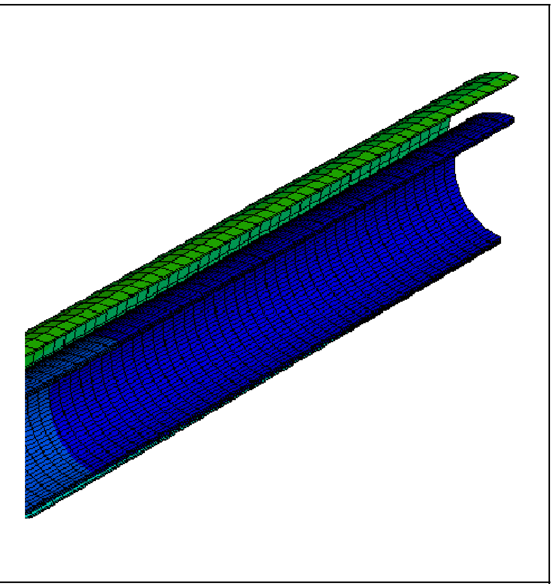

Fig. 12 High Temperature Contact Between PT and CT

DETERMINATION OF THERMAL CONTACT CONDUCTANCE (TCC):

After the first contact, the spread of contact and amount of heat dissipation will be a strong function of thermal contact conductance between PT and CT. A test facility has been designed and experiments have been conducted to determine TCC between PT and CT as a function of contact pressure and interface temperature. Fig. 13 and 14 illustrates the variation of TCC with contact pressure at typical interface temperature.

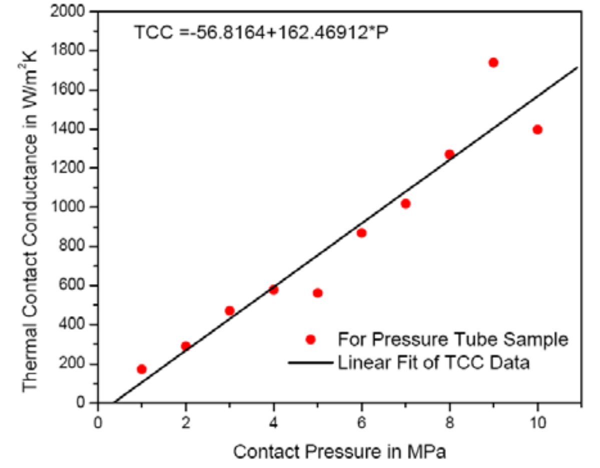

Fig. 13: Variation of TCC with contact pressure for PT

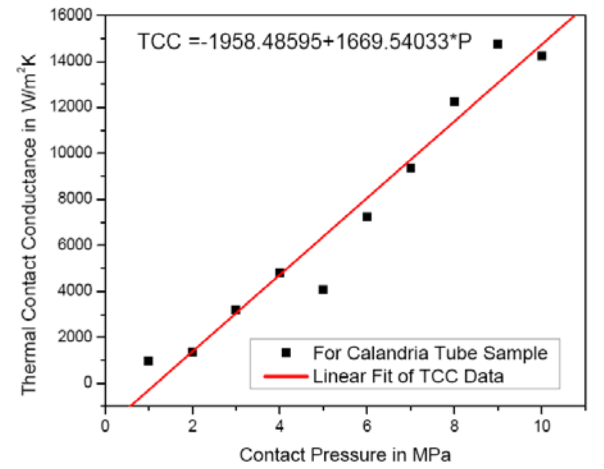

Fig. 14: Variation of TCC with contact pressure for CT

CONCLUSION

In a PHWR, prolonged station blackout leads to substantial damage to the fuel channel and leads to its failure. PT-CT contact as a result of ballooning plays an important role in deciding the course of severe accident progression as the contact can increase the heat removal through PT-CT to the moderator. To study the thermo-mechanical behaviour of PT, non-linear large deformation elasto-plastic contact analysis was carried out. The analysis shows PT-CT first contact occurs at 580^0 C. To simulate the resulting heat transfer after the first contact and ballooning behaviour of PT, detail knowledge of thermal contact conductance and thermo-physical properties of the Zr-Nb alloy is required.

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