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3D-thermo-structural simulation of pressure tube–calandria tube behaviour under accident conditions in PHWR using ABAQUS

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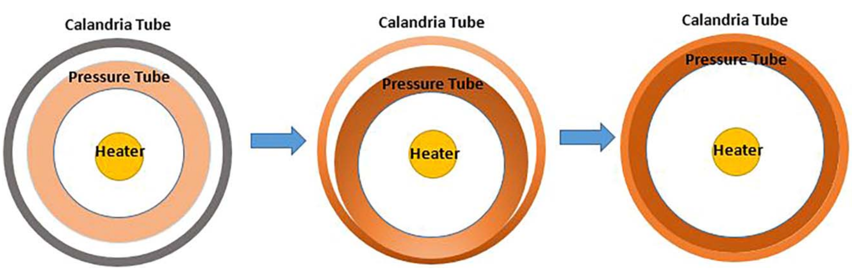
GRAPHICAL ABSTRACT



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ABSTRACT

In-depth understanding of the thermo-structural behaviour of structure system and components (SSCs) of a nuclear power plant is necessary for both safety margin assessment, and for devising appropriate prevention and accident mitigation strategies. In a pressurised heavy water reactor (PHWR), the fuel in the form of fuel bundles resides inside pressure tube (PT). The pressure tube is surrounded by a co-axial calandria tube (CT). The heat generated by nuclear fission in fuel bundle is carried away by heavy water coolant, which flows inside pressure tube. The coolant is at high temperature and high pressure inside the pressure tube. The surrounding calandria tubes are at low temperature and low pressure under normal operating conditions. Under certain accident scenario, the fuel bundles cooling may be lost. This results in heating up of the fuel bundles which leads to increase in temperature of the pressure tube. The pressure tube starts deforming as its temperature increases. The deformed pressure tube may contact the calandria tube either by sagging or ballooning. This leads to increase in the calandria tube temperature. If the temperature on the outer surface of the calandria tube exceeds the critical heat flux, film boiling may occur on the surface of the calandria tube. If the area in dry-out is sufficiently large and the dry-out is prolonged, the pressure-tube/calandria-tube combination can continue to strain radially and may challenge fuel-channel integrity. The International Collaborative Standard Problem (ICSP) on Heavy Water Reactor (HWR) Moderator Sub-cooling Requirements is organized by the IAEA to facilitate the development and validation of computer codes for the analysis of fuel channel integrity. The objective was to assess the capability of safety analysis computer codes in predicting the associated phenomena namely; radiation heat transfer from fuel to the pressure tube (PT), from PT to calandria tube (CT) heat transfer, PT deformation or failure, CT to moderator heat transfer and CT deformation or failure. The current analysis is carried out as a part of international collaborative standard problem exercise. This work simulates the heat transfer phenomena along with resulting deformation of the pressure tube and calandria tubes in a finite element framework. The insights obtained from the detailed modelling of the structural aspects of the phenomena can be used to update the

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system safety analysis codes. In this paper, a coupled heat transfer and structural analysis of pressure tube -calandria tube assembly is carried out using the finite element code, ABAQUS. The heat transfer analysis implements radiation heat transfer from heater to PT and from PT to CT. Contact conductance between PT-CT is modeled based on contact pressure. Convection from outer surface of CT to water is also considered. Structural analysis included three cases elastic–creep, elasto-plastic and elasto-plastic with creep of the individual PT-CT assembly under thermal and mechanical loading. It is observed that the results matched well for the case where only elastic creep constitutive model was considered. The temperature variation of pressure tube and calandria tube along with resulting deformation is obtained. The simulation is able to capture and predict the progressive contact of PT with CT and the deformation of CT along with the PT using FEM model. It was observed during the experiment and captured in the simulation that the PT-CT assembly deformation lead to removal of heat to the moderator and channel integrity was maintained for the given moderator sub cooling margin.

1. Introduction

In nuclear reactors, heat produced by nuclear fission reaction is used to generate steam which in turn is utilized to generate electricity using turbines and generator. The 220 MWe PHWRs consists of a horizontal reactor core of 306 parallel reactor channels. Each reactor channel consists of a pressure tube (PT) which is concentrically placed in a calandria tube (CT). PT and CT are made of Zr 2.5 wt% Nb and Zircaloy-4 material respectively. Fuel pins in the form of fuel bundles are housed inside PT. The gap between PT and CT is filled with CO_2 for thermal insulation. The PT is supported along its length through tight garter springs as shown in Fig. 1. All the reactor channels are submerged in a pool of heavy water called moderator maintained at around 65 °C.

Under steady state conditions the fuel heat is transferred mainly by conduction and convection to the coolant. During the loss of coolant accident scenarios along with loss of emergency core cooling system, the reactor shutdowns and the reactor power reduces to decay power (2–1% of nominal power). Under such conditions the coolant is unavailable therefore the convective mode of heat transfer from the fuel to coolant comes down and is limited to convection by steam. However the heat transfer by radiation mode increases as the fuel temperature increases. Therefore the fuel heat is removed mainly by radiation heat transfer to the pressure tube. As the pressure tube temperature rises, it starts to deform under the internal pressure and fuel bundle weight till it contacts the outer calandria tube. As the pressure tube contacts, the heat transfer increases from pressure tube to calandria tube to outside low temperature moderator (International Atomic Energy Agency, 2008). To achieve a final safe configuration/state of the reactor, it is important to simulate the phenomena involved and understand the behaviour of each component and its final state. Experimental investigations on PT-CT behaviour has been done in the past for pressurized PT or completely voided conditions (Nandan et al., 2010; Nandan et al., 2010; Yadav et al., 2012). Combined sagging and ballooning of PT under LOCA at different heat-up rates has been carried out by Nandan et al. (2010) and Nandan et al. (2010). In order to obtain temperature profile over pressure tube of Indian PHWR under full voided and partially voided conditions, an experiment has been carried out using a 19 pin fuel element simulator by Yadav et al. (2012). However, simulation of the experiments on the PT-CT assembly to predict the various phenomena involved is limited to simplistic methodology (Majumdar et al., 2004) or 2D plain strain simulation to obtain the contact time (Dureja et al., 2013). Attempts have been made to simulate the phenomena using simplistic user defined codes and integrate with system codes like ASTEC (Majumdar et al., 2014). These codes are validated for specific experimental conditions. It is important to have detailed simulation of PT-CT deformation in the postulated scenarios so that the insights developed can be used in the system thermal hydraulic codes. Experiments can be conducted to understand a given scenario but it is limited to specific range of parameters considered in the experiment. It is prudent to have a benchmarked simulation code specifically for the structural deformation that result from the thermal loadings encountered in the accident scenarios. This work demonstrates the analytical capability of 3D models to capture an extremely complex CT-PT phenomenon which are absolutely beyond the reach of commercial Lumped Parameter (LP) system codes.

Pressure tube and calandria tube contact experiment for pressurized heavy water reactor (CANDU 6) geometry was conducted as an International Collaborative Standard Problem (ICSP) (Niteanandan, 2012) exercise for moderator sub-cooling requirements. The objective was to assess the capability of safety analysis computer codes in predicting the associated phenomena namely; radiation heat transfer from fuel to the PT, from PT to CT heat transfer, PT deformation or failure, CT to moderator heat transfer and CT deformation or failure. The current analysis is carried out as a part of ICSP exercise. In the current work, coupled thermal and structural analysis is performed in a finite element framework. The heat transfer analysis implements radiation heat transfer from heater to PT and from PT to CT. Contact conductance between PT-CT is modeled based on contact pressure. Convection from outer surface of CT to water is also considered. Structural analysis included three cases elastic –creep, elasto-plastic and elasto-plastic with creep of the individual PT-CT assembly under thermal and mechanical loading.

2. The objective

To perform coupled thermo structural analysis in a finite element framework using ABAQUS to predict the associated phenomena namely; radiation heat transfer from fuel to the pressure tube (PT), from PT to calandria tube (CT) heat transfer, PT deformation or failure, CT to moderator heat transfer and CT deformation or failure during the experiment conducted as a part of ICSP (Niteanandan, 2012) exercise and assess the PT-CT behaviour and its integrity for the given postulated accident scenario.

3. The ICSP benchmark

The schematic of the experimental setup (Niteanandan, 2012) is shown in Fig. 2. The test section consists of Zr 2.5Nb pressure tube (PT) of length 1749 mm and concentrically mounted inside Zircaloy 2 calandria tube (CT) of 1700 mm long. A 38 mm diameter graphite rod

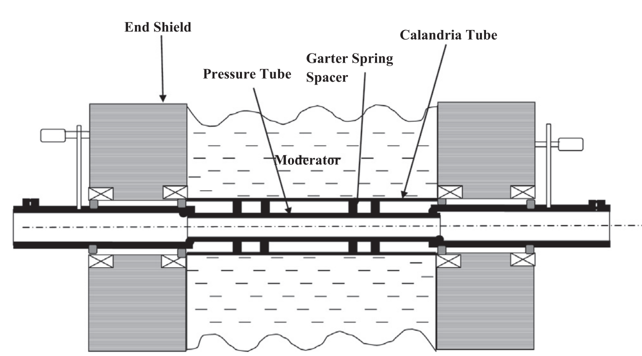

Fig. 1. Schematic of PHWR Reactor Channel.

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