EXPERIMENTAL AND ANALYTICAL SIMULATION OF THERMAL HYDRAULIC SAFETY SCENARIOS FOR INDIAN NUCLEAR REACTORS

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Abstract
This paper briefly describes the simulations and mathematical model developments carried out for simulating thermal-hydraulic safety scenarios for Indian nuclear reactors. Analyses were also carried out pertaining to design of experiments on reactor safety. These experiments are being conducted to generate data for validating in-house built computer models, understand the phenomenon and generate correlations. Two-phase thermal-hydraulic safety analysis code such as RELAP5/MOD3.2 [1] was used in modeling the reactor system and simulating the accidents. Beyond design basis accidents and its progression were simulated through ASTEC code [2]. Some computer models, pertaining to severe accidents, were also developed for PHWRs and PWRs which could not be analyzed otherwise with existing commercial codes.

Keywords: Safety analysis, Creep, Fragmentation, Corium stratification, Vapour pull-through

Introduction
Indian nuclear reactors namely – Pressurized Heavy Water Reactor, Advanced Heavy Water Reactor and VVER, widely various in the system thermal-hydraulics and accident progression under both design basis and severe accident scenarios. The objective of the thermo-hydraulics analysis of these reactors are – 1) to ascertain that some viable reactor anomalies or accidents do not breach the various acceptance criteria of the reactor; 2) to understand the reactor response under certain accident for which no engineered safety features have been devised and 3) to formulate emergency preparedness based on the analysis. Some of the phenomenon which occur during reactor accidents are not well understood and/or mathematically complex to model and simulate. In such cases part or integrated experiments are undertaken to generate simplified correlations or data which can be used to broaden the scope of existing thermal-hydraulic system codes. Some of the analyses were carried out in collaboration with different members of Core Safety Studies Section of BARC, Institute of Radio-protection in Nuclear Safety, France and the different universities involved in the experiments.

PHWR : Analysis and Modelling
A simulation model of a typical 220MWe PHWR system in generated in RELAP5 code for level-2 calculations and for conventional safety analysis. The model includes primary heat transport system with two channels (average and hot) in each pass, secondary system, auxiliary systems like bleed and feed, boiler pressure controller, reactor kinetics and emergency core cooling system. The kinetics of iodine and xenon concentrations in the reactor and its feedback on the power is programmed explicitly through user defined programmable feature of RELAP5 code. This
model was used in analyzing the over power incident which occurred in KAPS-1 reactor on March 10, 2004. It was found that an external reactivity addition of 0.03 to 0.04 mK (because of rod withdrawal) within a period of 30 sec might had initiated the event. Review of the incident indicated that the removal of xenon due to sudden rise in the neutron density induced additional positive reactivity into the system. The power obtained through the simulation closely resembles the plant-recorded data (Fig.1).

**AHWR : Analysis and Modelling**

Advanced Heavy Water Reactor is a vertical type of reactor wherein the heat transport from the core depends on the natural circulation of two-phase flow. A system model of the plant is developed in RELAP5 code to – a) generate and provide the process parameters during normal and transient conditions to the designers and b) carry out safety analysis with the best estimate models. Simulated results of loss of regulation accident (LORA) with the xenon feedback is shown in Fig.2. Separate RELAP5 models were also developed for design analysis of purification system and AHWR containment system. For AHWR critical facility, a point kinetics model was developed for simulating the ATWS transient.

**RD14M : Modelling, Analysis and Validation**

An exercise was undertaken to validate the thermal-hydraulic computer code (RELAP5/MOD3.2) against the benchmark experiments carried out with RD-14M Test facility[3]. RD-14M is a full-elevation-scaled thermal hydraulic test facility simulating most of the key components of a CANDU including the ECI system. RELAP5 modelling involves the discretization of the total system into number of volumes and inter-connecting junctions, valves, heat slabs and component specific models such as pump, separator etc. Heat losses from the PHT pipings were also simulated. Fig.3 shows the comparison of RELAP5 calculation against the experimental data for the same LOCA transient.

**Code Development for Severe Accident Analysis**

Following section discusses some important code developments carried out for simulating certain thermo-mechanical and thermo-chemical phenomenon which occur during severe accident scenario.
**Pressure Tube Creep and Heat Transfer**

In PHWR, core channels, which include pressure tube (PT) and calandria tube (CT), are horizontal and hence amenable to sagging creep deformation during high temperature transient owing to the fuel bundle weight. Also, pressure tube is expected to deform radially and balloon during high pressure and high temperature conditions. These deformations are time dependent and plastic in nature. Large deformation of the pressure tube will lead to PT – CT contact and rejection of excess heat to the moderator. A thermo-mechanical computer code (PTCREEP) was developed which predicts the deformation based on Shewfelt’s strain rate correlation [4,5]. This code is coupled with a heat transfer module which predicts the temperature distribution in PT and CT, and also the heat dissipation core to the moderator. Fig. 4 depicts the predicted results for an experiment carried out on CANDU channel [6].

**Jet Fragmentation and Debris Heat Transfer**

During the late phase of severe accident in PWRs, molten corium would flow through openings in the lower core plate and fall into the coolant contained in the lower plenum. The interactions of hot jet with relatively cold liquid yields melt fragments which causes rapid thermal and hydrodynamic exchange. These affect the accident progression and determines the thermal and pressure load on the reactor vessel. A simplified model has been developed for jet fragmentation[7]. The jet steady state length and the fragment diameter is obtained from a statistical correlation developed based on instability analysis of jet surface. The fragments thus produced exchange heat, under film boiling condition, with the surrounding water and settles on a bed. In the bed, heat transfer is based on 1D Lipinski model developed for deep bed configuration. The model has been well validated against FARO-L14 and FARO-28 Tests[8] and has been implemented in severe accident code ASTEC.

**Corium Stratification**

Large molten pool may form in the lower plenum of PWRs and in the Calandria vessel of PHWRs due to slumping of molten core material during late phase of severe accidents. The liquid corium, depending on elements content, dissociates into two distinct immiscible fluids (metallic and oxidic) having different material properties and composition. Such stratified layer has important bearing on the thermal load of vessel due to focusing effect of high conducting metallic layer (if it is thin enough) and causing a limited zone of high heat flux on the vessel surface. The situation may aggravate due to re-distribution of heat generating Fission Products within the layers. A computer code has been developed which calculates the immiscible liquid phase fractions having different compositions, movement of these phases due to density difference and distribution of fission products among the phases. The model has been implemented in ASTEC code. Fig. 5 shows the validation of the code prediction against MASCA experiments [9].
Experimental Investigations

Vapour Pull-Through Experiment

Vapour pull-through occurs in horizontal components where liquid and gas phases stratify with distinct interface. Such situation is expected in Header-Feeder configuration of PHWRs during abnormal conditions. Feeders may ingress vapour along with the water even though its location is below the interface. Vapour pull through experiments were carried out in the facility developed at M.S University, Baroda. The data generated was used to develop a generalised correlation for predicting the level at which ingestion begins. Fig. 6 shows the comparison between well known Smoglie’s model and the developed correlation for a typical feeder position.

PT Ballooning Creep Experiment

This facility has been developed at IIT, Roorkee. The setup primarily simulates the PT ballooning under set internal pressure and heat up condition. In the experiment, a single channel, consisting of PT and CT, is submerged in a large pool of water. Three ballooning experiments were carried out with internal pressure of 2, 4 and 6 MPa at heater power of 17, 21 and 14 kW respectively. The data generated is used in validating the in-house built computer code ‘PTCREEP’. Fig. 7 compares the PT-CT gap predicted by the code with experimental data [10].

Concluding Remarks

Synergy is achieved between experimental and analytical simulation of number of safety scenarios for Indian nuclear reactors. This work and its further extension are helpful in enhancing defence in depth of Indian Nuclear Power Plants.

References


