ANALYTICAL AND EXPERIMENTAL STUDY IN NUCLEAR REACTOR SAFETY

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Abstract

The paper describes some of the key analytical study which are carried out to evaluate safety of Advanced Heavy Water Reactor (AHWR), source term evaluation for postulated accidents in Pressurised Heavy Water Reactor (PHWR), understanding the complex phenomena of severe accident and it’s application for the verification of Severe Accident Management Guidelines (SAMG) for VVER-1000 reactor. The paper also describes the thermal-hydraulics experimental study related to Leak Before Break (LBB) and Molten Material Coolant Interaction and channel heatup.

Key words: nuclear safety, source term, severe accident, critical flow, steam explosion

Analytical Study

Three major studies are briefed in the section which are carried to evaluate of safety of AHWR, PHWR and VVER-1000.

AHWR Safety Analysis

Safety margins like Peak Clad Temperatures, Oxidation thickness etc. are evaluated for different Postulated Initiating Events for AHWR as requirement of to assess the ECCS acceptance criteria. Fig. 1 shows analysis result for evaluating fuel temperature of maximum rated channel to assess the emergency core cooling and fule failure criteria. The analysis work also helped to assess the adequacy of ECSS inventory and mode of injection. The detail work is described by Mukhopadhyay et al. Analysis are also carried out for designing the pressure relieving capacity for calandria vessel, steam pipe line sizing, accumulator sizing etc. Several severe accident scenarios like stagnation channel break, Spectrum of breaks at inlet header without wired shutdown system, steam line break outside containment are analysed. This exercise has been carried out for PSA Level-2 study of AHWR PSA. The analysis has brought out extent of core damage, fission product release in the core and subsequent release to the containment. This study demonstrates that the radiological risk due to postulated severe accidents in AHWR is very low.

PHWR Safety Analysis

Both Design Basis and severe accident scenarios are analysed to support the regulatory board and development...
of a diagnostic system of accident evaluation for PHWR. Analysis like LOCA from feeder break, stagnation channel break, and spectrum of break sizes at inlet header and outlet header are some work carried out. The work is immensely useful in making of regulatory decisions. For severe accident a philosophy of multi physics approach has been laid down to carryout reactor thermal hydraulics, fission product release and transport in the primary heat transport system and containment. Fig. 2 shows such activity for Cs release for different break sizes at Inlet Header without the availibility of Emergency Core Cooling System (ECCS) injection. The analyses are carried out with computer codes like RELAP5, PHTACT and ASTEC to adress the multi physics aspects of the phenomena like blowdown, fuel heat up by different modes of heat transfer, fission product release characteristics and their retention by mechanisms like diffusophoresis, thermophoresis etc.

**Severe Accident Analysis for Light Water Reactors**

Different severe accident scenario for Light Water Reactors like VVER 1000 and TMI-2 are analysed using international severe accident codes ASTEC, ICARE/CATHARE and RELAP/SCDAP. The spectrum of analyses includes high pressure, low pressure and containment bypass scenario for verification of SAMG. Fig. 3 shows such activity for a low pressure event like LOCA with Station Black out (SBO) for VVER-1000 reactor. Analysis for Out-of-Pile severe fuel damage experiments for enhancing the understanding of different phenomena involved in severe accident is also carried out along with reactor calculation. Fig. 4 shows the predicted linear corium profile comparison with measured for PHEBUS-SFD experiment.

**Experimental Study**

Experiments are carried to understand phenomena related to for Design Basis Accidents (DBAs) and Beyond DBAs for PHWR. It comprises of inception and design of facility, conducting experiments and code validation. It involved Pipe Blowdown, Leak Before Break, Thermo-mechanical deformation of reactor channels, Contact Conductance, Fuel Coolant Interactions (FCI) and vapour pull through.
**Critical Flow through Slits and Cracks**

Flow through slits and cracks from high internal pressure of 100-150 bar will be critical as the sound velocity of the medium governs the fluid velocity. Experiments to study the critical flow is carried out under LBB study program. Slits of different dimensions generated on 4” pipes (sch. 80) with wire EDM technique are used to simulate the cracks. Total 27 nos. of experiments were carried out with different thermal hydraulic conditions (70 -90 bar and 220°C - 250°C). Fig. 5 shows slit flow behaviour under different subcooling. The generated data are used to validate in-house code C_SFA and international code RELAP5.

**Molten Fuel Coolant Interaction (MFCI) Study**

Under MFCI study, in the first phase stability of the measured vapor film generated around an internally heated metal ball (50 mm) under different subcooling and pressure pulse was studied along with generation of pool boiling curve to evaluate film heat transfer coefficient. In the second phase dispersion of cold balls (5 mm dia.) was studied to derive the drag forces from studying its dispersion pattern and settling velocity. In the third phase dispersion of hot balls (5 mm dia and at 500°C) was studied to derive the again drag forces, energetics and settling velocity. In the fourth phase experimentation were carried out by discharging molten lead (1kg) and Sn (600 gm) with different jet velocity to study the dispersion pattern and energetics. Following figures show some of the results of different phases,

**Channel Heatup Study for PHWR**

Under Channel Heatup Study, experimentation are carried out for Pressure Tube sagging and balloning under heatup condition. This is to investigate the channel integrity under accident condition. In this study phenomena like thermo-mechanical deformation of PT, PT-CT contact conductance and boiling heat transfer on CT surface are studied. Fig. 7 and 8 show temperature and deflection transients respectively.

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**Fig. 5: Slit Flow Rate at different Subcooling**

**Fig. 6: MFCI Experiments at Different Phases**

**PHASE-1 (film on Sphere) PHASE-2 (cold ball) PHASE-3 (hot ball) PHASE-4 (Pb discharge)**
Analytical model developed, code “PTCREEP” for reactor channel deformation is validated with the generated data.

Conclusion

The safety study carried out analytically and through experiments has helped to understand different aspects of nuclear reactor safety, physics of the associated phenomena, assess the design and safety criteria and verification of proposed operator action in case of accidents.