ARTICLE IN PRESS

Nuclear Engineering and Design xxx (xxxx) xxx–xxx

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Validation of computer code 'ATMIKA' against RD-14M Small Break LOCA experiments

Ashutosh Dixit*, S.K. Yadav, N. Kumar, T.A. Khan, S. Hajela, M. Singhal

Nuclear Power Corporation of India Limited, Nabhikiya Urja Bhavan, Directorate of Reactor Safety & Analysis, Anushakti Nagar, Mumbai 400094, India

ARTICLE INFO

Keywords:
SBLOCA
RD-14M
CANDU
IAEA
ICSP
ATMIKA
PHWR

ABSTRACT

The simulation of Small Break Loss-of-Coolant Accident (SBLOCA) experiments in the RD-14M integral test facility is performed under the auspices of International Atomic Energy Agency (IAEA) as an International Collaborative Standard Problem (ICSP) with the objective to benchmark and validate the in-house developed system thermal–hydraulic neutronic computer code 'ATMIKA', extensively used to analyze postulated events in Indian PHWRs. RD-14M is an 11 MW, full-elevation-scaled extensively instrumented thermal hydraulic Canadian test facility, possessing most of the key components of a CANDU (CANada Deuterium Uranium) primary heat transport system (PHTS). The loop configuration is similar to figure-of-eight geometry of a typical CANDU circuit and it is intended to reproduce the important geometric features of a reactor PHTS and the appropriate operating conditions. 'ATMIKA' prediction and its comparison against SBLOCA experimental results are compared in this paper. A specific SBLOCA experiment 'B9006' is selected for the Computer code 'ATMIKA' predictions. Test B9006 is a 7-mm inlet header break experiment with pressurized accumulator emergency coolant injection (ECI) and represents most complete SBLOCA test conducted in RD-14M that includes all the phases of the transient (blow-down, high-pressure ECI, secondary pressure ramp (crash cool), refill, low pressure ECI, exponential pump ramp, and natural circulation). This simulation demonstrates that 'ATMIKA' is adequately capable of predicting the break discharge, PHTS depressurization, channel flow rate, channel voiding, fuel sheath temperatures and high pressure core injection flow through ECCS accumulator and initiation of low pressure ECI for test B9006. 'ATMIKA' predicted results are compared with experimental results and it is seen that predicted results for all phases of transient are in good agreement with experimental results.

1. Introduction

NPCIL participated in an IAEA International Collaborative Standard problem (ICSP) with the objective to benchmark and validate in-house developed system thermal-hydraulic neutronic computer code 'ATMIKA' against qualified data for Small Break Loss of Coolant Accident (SBLOCA) scenarios generated on Canadian RD-14M Test Facility (IAEA-TECDOC-1688, 2012). Computer code 'ATMIKA', is extensively used for safety analysis and licensing of Indian Pressurized Heavy Water Reactors (IPHWRs).

In this IAEA-ICSP activity, two specific Small Break LOCA (SBLOCA) tests (B9006 and B9802) conducted in Canadian test facility RD-14M were selected and case B9006 has been presented in this paper as it represents the most complete SBLOCA test with a 7-mm inlet header break experiment which includes all the phases of the transient (blow-down, high-pressure ECI, secondary pressure ramp (crash cool), refill, low pressure emergency coolant injection (ECI), exponential pump ramp, natural circulation and bidirectional flow in the horizontal parallel channel geometry).

Test facility RD-14M is an 11 MW, full-elevation-scaled thermal–-hydraulic test facility possessing most of the key components of a CANDU primary heat transport system. RD-14M is extensively instrumented facility with over 600 measurements such as temperatures, pressures, flows, levels, and voids at various locations.

This paper presents the modeling of Canadian test facility RD-14M with in-house developed computer code 'ATMIKA' for simulation of one of the selected test i.e. B9006. Analytically predicted transient by computer code 'ATMIKA' has been compared with experimental results for the test B9006.

As seen through the results of PSA studies, performed for Indian PHWR which concludes that SBLOCA is the dominating event influencing the core damage frequency. In view of that, benchmarking for most complete SBLOCA in PHWR becomes more relevant.

* Corresponding author.
E-mail address: adixit@npcil.co.in (A. Dixit).

http://dx.doi.org/10.1016/j.nucengdes.2017.07.042
Received 23 September 2016; Received in revised form 21 July 2017; Accepted 31 July 2017

Please cite this article as: Dixit, A., Nuclear Engineering and Design (2017), http://dx.doi.org/10.1016/j.nucengdes.2017.07.042

2. Brief description and assessment of the code

Computer code 'ATMIKA' (IAEA-TECDOC-984, 1997; IAEA-TECDOC-1149, 2000; IAEA-TECDOC-1688, 2012, and IAEA-TECDOC-1709, 2013) is a system thermal hydraulic neutronic computer code developed in Nuclear Power Corporation of India Limited (NPCIL) for the analysis of Loss-of-Coolant Accident (LOCA), main steam line break and plant transient scenarios. The computer code 'ATMIKA' is based on Unequal Velocity Equal Temperature (UVET) model using three conservation equations with drift flux model. ATMIKA uses four equation model where three conservation equations for homogeneous two phase and drift flux for phasic velocity difference. Phase separation and non-thermal equilibrium in fuel channel is taken care by stratification model. It is seen that with the modeling technique used in ATMIKA, it is equally capable of predicting all associated phenomena of SBLOCA and can be compared with other six equation codes including RELAP5 with local minor deviation in the reactor system involving large source and sinks. A staggered mesh arrangement is adopted where pressure, density and enthalpy are defined at the node; flow is defined along the flow path at the junction of two control volumes. Mass and energy conservation equations are applied on lumped control volume and momentum equation is applied on flow paths. A semi-implicit scheme has been adopted for solving the set of differential equations.

Some of the important models that ATMIKA utilizes are wall heat transfer model for estimation of heat transfer from fuel to coolant and coolant to steam generator/heat exchanger tube including various transition boundaries. Model uncertainties are suitably taken care by selecting reasonably conservative models based on engineering judgment. Critical Heat Flux (CHF) is estimated based on Groeneveld's Look-up table. AECL Look-up table is used for post dry out heat transfer coefficients (IAEA-TECDOC-1203, 2001). Critical discharge through the break is estimated using options from several available correlations e.g. Burnell's (for single-phase liquid), Moody, Homogeneous Equilibrium Model, Frozen slip model. Friction Model is used to calculate pressure drop across pipes. Among various options, model suggested by Chisholm based on Lockhart-Martinelli and Friedel correlation is used to evaluate two-phase frictional pressure drop. There are options for various thermal hydraulic models related to two phase modeling in the code but during detailed validation exercise for special effects, appropriate models are selected and kept same for remaining analysis. Based on sensitivity studies, Lockhart-Martinelli model is dropped against Friedel/Chisholm. Pump Model is used for evaluating pump behavior under single phase and two-phase conditions. Heat Conduction model is used for estimation of radial temperature profile in fuel rod and pipes. Computer code identifies the stratified flow regime in horizontal channel based on Taitel and Duckler model. The ability to calculate the heat transfer from individual groups of pins in a fuel bundle subjected to stratified flow is incorporated into the code. Fuel temperature in case of stratified condition is estimated based on actual non-thermal equilibrium vapor enthalpy conservatively. Thermodynamic properties of the coolant have been simulated in all range of sub-cooled, two phase and superheated conditions. A model has been incorporated for simulation of indirectly heated Fuel Element Simulator (FES). In the experiment, fuel bundles were electrically heated and (reactor) shutdown was simulated by tripping these electric heaters (which were tripped on process signal of primary system pressure low). Therefore, no neutronics were involved in the experiment. However, computer code ATMIKA is coupled with point/3D neutron kinetics code to predict neutronic and thermal hydraulic behavior during postulated accident scenario.

In the numeric, computer code ATMIKA, first solve the momentum equation assuming previous value of pressure which gives the flows from one node to another. The updated flows are used by the mass and energy equations to update the mass and energy contents at each location. The new mass and energy are given to the equation of state to update the pressure distribution at each node. The new pressure, along with the new densities and energies are used by the momentum equation, and so on. In this manner, a time history of the fluid evolution is obtained. This method is evolved to get maximum numerical stability with higher time step size.

As a part of code development above models have been checked, verified and compared for their correctness with logical trends. The computer code 'ATMIKA' as an integration of all the above individual models was assessed with respect to published results in open literature on Canadian experimental facilities. It is observed that overall predictions of ATMIKA match reasonably well with the experimental results. Simulation of wide range of break sizes and locations on these facilities demonstrate the predicting capability of computer code ATMIKA. Computer code ATMIKA is being extensively used in NPCIL to analyze loss of coolant accidents in Indian PHWRs.

Computer code ATMIKA is in-house developed thermal hydraulic code which is well validated with experimental results as well as inter-code comparisons. Verification of ATMIKA code is done intensively in-house within NPCIL. This code is used for licensing analysis of Indian PHWRs and has been peer reviewed by the regulatory body, Atomic Energy Regulatory Board (AERB) in India through various Committees/ Task Groups as a part of regulatory submission involving different scientific organizations and academic institutions.

3. RD-14M facility description

The RD-14M facility, shown schematically in Fig. 1, is a pressurized-water loop with essential features similar to the primary heat transport loop of a typical CANDU reactor (RD-14M, 2010). The facility is designed so that reactor typical conditions, such as fluid mass flux, transit time, pressure and enthalpy can be achieved in the primary-side for both forced and natural circulation. The design incorporates the basic "figure-of-eight" geometry of a CANDU reactor, with five horizontal channels per pass and a 1:1 scaling of the vertical elevations throughout the loop. Each six-meter-long channel contains 7 electrically heated Fuel Element Simulators (FES), connected to end-fitting simulators. The thermal characteristics of the FES are similar to CANDU fuel in terms of power density, heat flux and heat capacity. Five reactor channel/feeder geometries were selected, representing three middle channels, one top channel, and one bottom channel. Similar hydrostatic pressures maintained between RD-14M and a typical reactor by preserving the 1:1 vertical scaling.

The steam generators (SG) are scaled approximately 1:1 with typical CANDU steam generators, in terms of tube diameter, mass flux, and heat flux. Spiral-arm steam separators in the steam dome and flow restricting orifices in the external downcomer of the SGs are used to produce reactor-typical recirculation each of the steam generators.

Primary fluid circulation is provided by two centrifugal pumps. These deliver full reactor typical head (about 225 m) at flow rates similar to a single reactor channel (about 24 kg/s). Primary circuit pressure is maintained by a loop pressurizer that contains an electrical heater.

The RD-14M facility is equipped with an Emergency Core Coolant (ECC) system that provides cooling to the FES under postulated LOCA conditions. The ECC system injects emergency coolant into the primary heat transport system through any combination of the four headers. The ECC system is controlled by the primary loop pressure with the isolation valves automatically opening when header pressure drops below a predetermined pressure.

RD-14M is extensively instrumented facility to measure parameters such as temperatures, pressures, flows, levels, and voids at various locations.

4. Nodalization scheme

The ATMIKA nodalization of the RD-14M test facility for tests B9006 consisting of primary, secondary and ECI system is shown in