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TRACE VALIDATION AGAINST FIX-II TEST NO. 3025

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Abstract

The paper describes the results of the validation of the TRACE code against FIX-II LOCA Blowdown and Pump Trip Heat Experiment No. 3025. The FIX-II facility was a scaled down model of the Swedish type Boiling Water Reactor (BWR) with external recirculation pumps. The experiment simulated the 31% break in one of the recirculation lines. The experimental facility consisted of test section with model of one fuel assembly, spray condenser, bypass and downcomer, and two recirculation lines in which one simulated the broken loop. The results obtained with TRACE v5.0 Patch 2 are in general in a good agreement with the experimental measurements.

Introduction

The aspect of safety in the nuclear industry is of paramount importance. To assess whether the nuclear facility fulfills the safety criteria appropriate safety analysis need to be performed. Part of the safety analysis consists of calculations done by a systems code, thermal-hydraulics engineering tool. In order to assess the thermal-hydraulics code applicability and accuracy, a validation procedure needs to be performed. The validation purpose is to show the codeĝs ability to predict relevant parameter of physical experiment or full-scale facility.

In this paper the main results from the TRACE code validation against a FIX-II experiment are presented. The calculations were performed with TRACE 5.0 patch 2. FIX-II LOCA (Loss of Coolant Accident) Blowdown and Pump Trip Heat Experiment was a series of experiments that were intended to test the behavior of the Swedish type Boling Water Reactor (BWR) with external recirculation pumps. The FIX-II facility was a volumetrically scaled down model of the existing reactor. The scaling refers to Oskarshamn-2 reactor and gives volume ratio of 1:777. The total number of 14 LOCA tests were performed in the FIX-II facility. Basic properties of these tests are shown in Table 1.

Table 1. Experimental matrix of FIX-II experiment.

Break type	Split break							Guillotine break		
Break area (%)	10	31	31	48	100	150	200	155	200	200
Break diameter (mm)	6.8	12.0	12.0	15.0	21.6	26.4	30.5	16.0+21.6	21.6+21.6	21.6+21.6
Test no.	3051	3013	3024 3025 (ISP 15) 3026 3027	3031	3061	3071	3041	4011	5061	5051 5052

The main goals of the FIX-II experiment were:

- to determine the time to dryout in a simulated electrically heated fuel rod bundle under conditions which resemble those in an external pump reactor during a LOCA,
- to investigate the effect during simulated LOCA conditions of different areas and geometries for bottom breaks, and of different process and on the cladding temperature during the blowdown period,
- to measure fuel rod temperatures and use these measurements to calculate heat transfer coefficients between cladding and coolant during the blowdown period,
- to determine the time to dryout and measure post-dryout temperatures in fuel rod bundle during simulated pump trip transients for internal pump reactors.

Test no. 3025 was chosen for a calculation exercise by the NEA/CSNI as an international standard problem (ISP-15) for assessment of computer codes used for thermal-hydraulics analysis of transient events in reactors. In this test a split break LOCA located in the main recirculation line was simulated. The break size was equal 31% of the pipe inner diameter.

1. Description of the experimental facility

FIX-II facility diagram is shown in Fig. 1. The main components of the facility are:

- pressure vessel with a 36-rod test section and a spray condenser,
- piping system with recirculation pumps and arrangements for simulating breaks,
- auxiliary systems, i.e. cooling loops for steady-state operation of the loop,
- power supply and power control systems,
- process control system,

- measured systems.

Total volume of fluid in primary loop was 1.205 m^3 at operating temperature, 1.190 m^3 at 20 ºC.

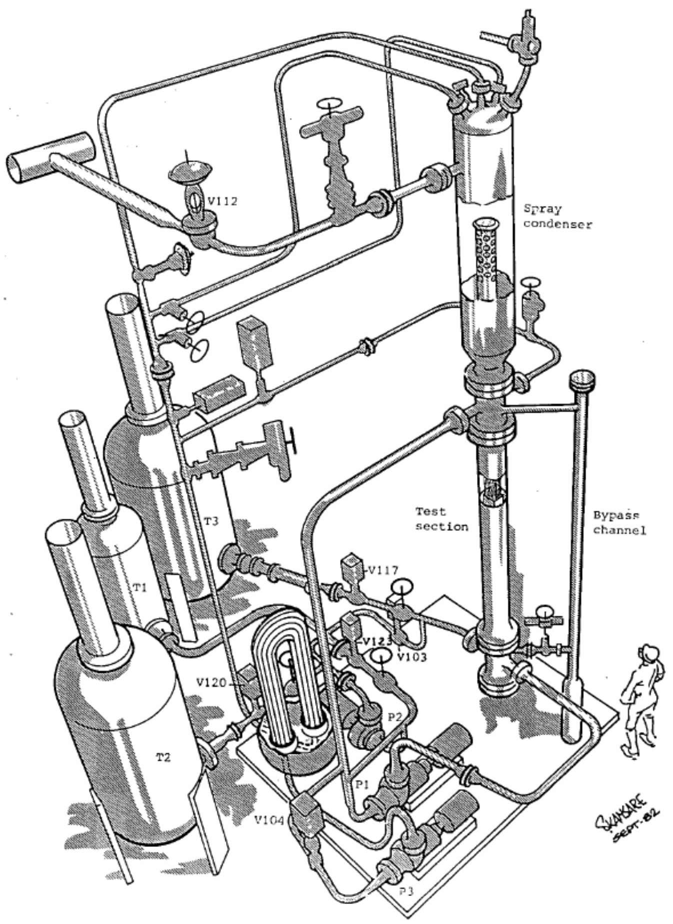

Figure 1. FIX-II facility diagram.

The four main recirculation pumps in the Oskarshamn-2 reactor were represented in the FIX-II loop by one large pump, P1, and one small pump P2. The larger pump delivered 3/4 of the total flow, and when simulating a pipe break in the main recirculation line this pump corresponded to the three pumps in the intact lines. The smaller pump corresponded to one reactor pump, which was in the line where pipe breaks can be simulated for different LOCA cases.

Opening valve V120 simulated split break LOCA in the system.

Sequence of main events

The whole test lasted 75.2 sec. The beginning of the test started from the opening of V120 valve (simulating the brake, LOCA), initiating decay power in the test section (36 heaters made of Inconel 600 and stainless steel), and the coast down of the main recirculation pump P1. The SRV (Steam Relief Valve) open-close sequences took place in time intervals of: 0.4-1.0 sec, 1.6-2.3 sec, and from

11.8 until the test was terminated. The decay power of bypass heaters started at 1.1 sec. The spray water flow and feedwater flow were closed at 1.8 sec and 1.9 sec, respectively. Termination of data acquisition was terminated after the experiment was finished, at 85.2 sec.

2. TRACE nodalization

The nodalization of FIX-II facility is presented in Fig. 2.

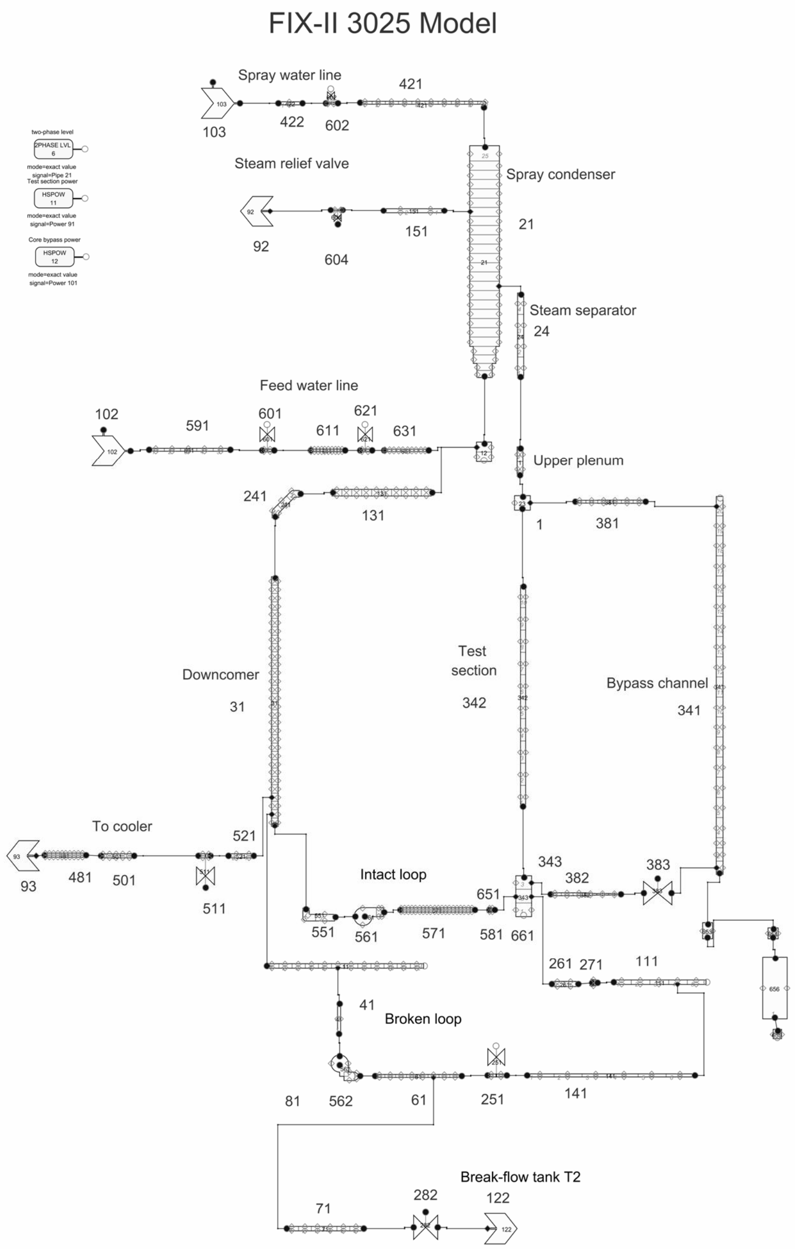

Figure 2. Nodalization of FIX-II facility.

The spray water line, feedwater line and water to the cooler are represented by two fill components and a break component, respectively. In the fill components the mass flow rate, pressure and temperature of water are set as boundary conditions. The test section is modelled by the pipe component. The fuel rods are modelled by the heat structure components and generate heat that corresponds to the hot channel, which is 3.385 MW. Channel wall heat transfer between the test section and bypass is simulated by the heat structure component with the power of 43.7 kW. The axial shape of the core power and bypass were simulated according to the experiment description. The initial conditions are presented in the Table 2.

Table 2. Initial conditions for experiment no. 3025.

Pressure in the steam dome	7.00	MPa
Power to the 36-rod bundle (incl. connections)	3.385	MW
Power to the bypass heaters	57.4	kW
Cooling power in the filler body space	196	kW
Mass flow rate through pump P1	4.56	kg/s
Mass flow rate through pump P2	1.55	kg/s
Mass flow rate in the bypass	0.60	kg/s
Mass flow rate in the 36-rod bundle	5.51	kg/s
Mass flow rate in the spray line	5.36	kg/s
Mass flow rate in the feed water line	2.49	kg/s
Temperature of water at the bundle inlet (TE2)	269.0	°C
Temperature of feed and spray water	181.0	°C
Water level in the spray condenser	0.815 (z = 6.329)	m
Rotational speed of pump P1	160	rad/s
Rotational speed of pump P2	210	rad/s

The heat from the heaters in the test section and in the bypass was modeled in TRACE taking into account the axial power shape provided in reference [1].

Default critical flow code option was used to calculate a critical flow passing through the valve V120 (the brake location). This model uses discharge coefficient (multiplier) of 1.0 for both subcooled and two-phase flow. No CCFL option was used in the calculations.

Concerning the experiment uncertainties, the evaluated measurement errors (ñprobable errorò and error corresponding to 95% confidence level) of the measurement quantities are [1], respectively:

- pressures, 0.014 MPa and 0.04 MPa,
- fluid temperatures, 1 °C and 2 °C,
- cladding temperature, 1.6 °C and 3.2 °C,
- small range differential pressures (5 to 7.5 kPa), 0.13 kPa and 0.3 kPa,
- medium range differential pressures (25 to 50 kPa), 0.22 kPa to 0.5 kPa,
- high range differential pressures (100 to 700 kPa), 0.26 kPa to 0.65 kPa.

The mass flow rates in the recirculation lines and in the bypass channel were determined (in the experiment) based upon differential pressures. These data are valid as long as the fluid conditions are subcooled.

3. Results

After steady-state initialization the transient calculation was performed. The results are shown in the following figures.

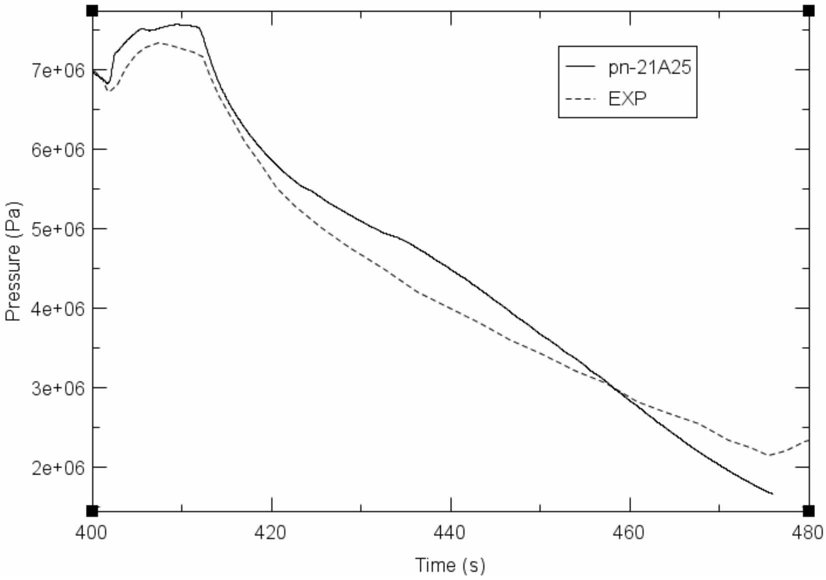

Figure 3. Pressure in the steam dome.

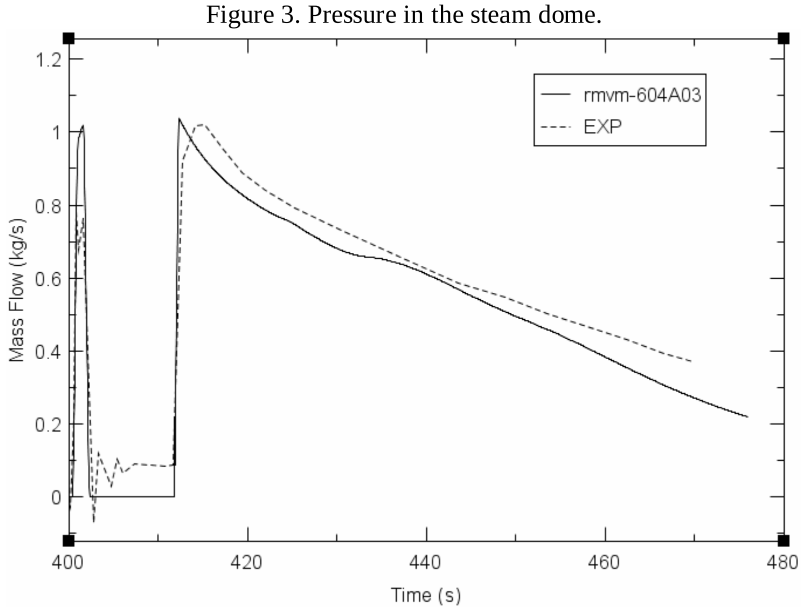

Figure 4. Steam relief valve mass flow rate.

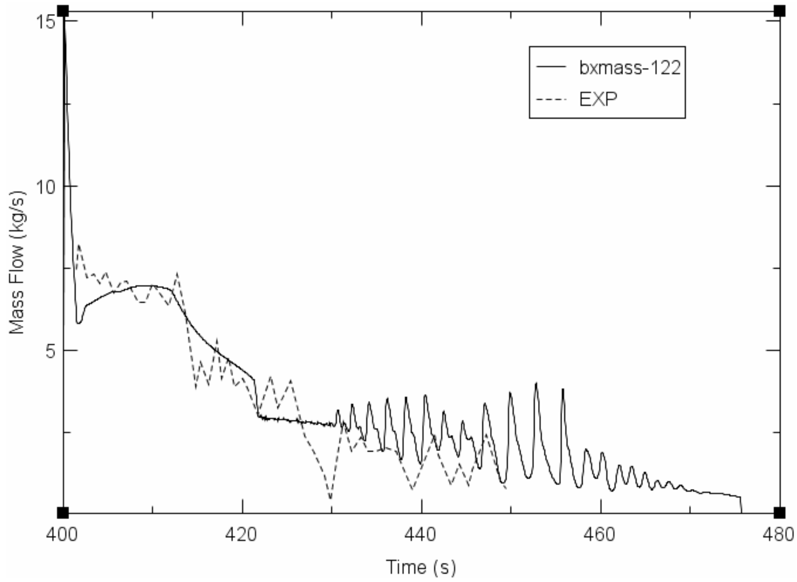

Figure 5. Break mass flow rate.

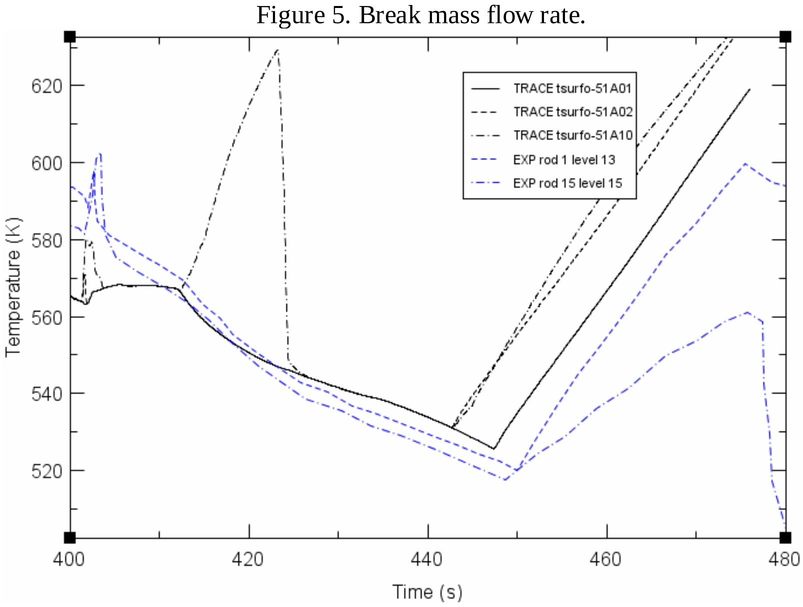

Figure 6. Cladding temperatures.

4. Discussion and Conclusions

The results are in a good agreement with the experimental results. It can be observed that pressure in the steam dome is slightly over-predicted in the first seconds of the transient. In the later stadium of transient it converges to the experimental value. Steam relief flow tends to be slightly under-predicted in the final stage. The break flow to the tank T-2 shows large oscillations, but the general trend is similar to that of experimental data. TRACE predicted cladding temperatures deviate compared to experimental data, but in a conservative way. The steady-state temperature deviation is due to heat transfer coefficient, probably because of the uncertainties in the initial mass flow rate. The transient fuel temperature deviates due to lower mass flow rate, which causes initial dryout at 405 sec and 415 sec (not observed in the experiment) and an earlier final dryout, at around 445 sec (observed in the experiment at 450 sec).

Based on the above results, the TRACE code appears capable of simulating BWR LOCA transient.

Acknowledgment

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5. References

[1] Lars Nilsson et all, ñFIX-II LOCA Blowdown and Pump Trip Heat Transfer Experimentsò, Experimental results from LOCA test No. 3025, Swedish Nuclear Power Inspectorate, Project B28/82, Reference G.2.1/2461/77, Studsvik/NR-83/318.