Justification of the Ignalina NPP Model on the Basis of Verification and Validation

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1 INTRODUCTION

The state-of-the-art code RELAP5/MOD3 was originally designed for Pressurised Water Reactors. Because of unique RBMK designs, the application of these codes to RBMK-1500 encountered several problems. A successful best estimate RELAP5 model of the Ignalina NPP has been developed. This model includes the reactor Main Circulation Circuit (MCC) and Reactor Control and Protection System (CPS) required for this kind transient analysis. In process of development of the RELAP5 model for RBMK type reactors comparative analyses of actual operational events are essential because this allows to establish realistic resistances of different MCC components and realistic behaviour of the controllers of the reactor systems.

Benchmark analysis of

- one and all operating MCPs trip events,
- three Main Safety Relief Valves LOCA event,
- inadvertent actuation of Emergency Core Cooling System (ECCS)

has been performed. The calculations performed with RELAP5 models on the Ignalina NPP specific base compare favourably with the plant data.

At the same time validation of RELAP5 code models against test facilities (Elektrogorsk 108 and PNC SEL) experimental data was performed. Specifically, Critical Heat Flux (CHF) and oscillatory behaviour phenomena were analysed.

2 VERIFICATION OF IGNALINA NPP RELAP5 CODE MODEL

The Ignalina Nuclear Power Plant is a twin-unit with two RBMK-1500, graphite moderated, boiling water, multichannel reactors. Several important design features of RBMK-1500 are unique and extremely complex with respect to western reactors [1]. The RBMK-1500 coolant loop, having a very long flow length of more than 200 m, consists of 1661 of parallel pressure tubes and numerous components, such as headers, pumps, valves, etc. This complex MCC is essentially divided into two loops. Schematic representation of one MCC loop is given in Figure 1. Each loop of the MCC is cooled by water provided from three operating Main Circulation Pumps (MCPs). A fourth MCP in each side of the MCC is normally idle and in standby mode. A detailed description of the MCC and plant safety systems of Ignalina NPP is given in [1].

The state-of-the-art code RELAP5/MOD3 was originally designed for Pressurised Water Reactors. Because of the unique RBMK design, the application of this code to RBMK-1500 encountered several problems. A successful best estimate RELAP5 model of the Ignalina NPP has been developed in the Lithuanian Energy Institute. This model includes the reactor MCC and reactor control and protection system required for this kind of transient analysis. In process of development of the RELAP5 model for RBMK type reactors, comparative analyses of actual operational events are essential because this allows to establish realistic resistances of different MCC components and realistic behaviour of the controllers of the reactor systems.

This paper deals with RELAP5 benchmark analysis of

- one and all operating MCPs trip events,
- three Main Safety Relief Valves LOCA event,
- inadvertent actuation of ECCS.

2.1. RELAP5 MODEL OF THE IGNALINA NPP

This section describes the RELAP5/MOD3.2 model of the Ignalina NPP used for analyses of thermal-hydraulic response of plant to various transients. The RELAP5 computer code has been developed by Idaho National Engineering Laboratory [2]. This is one-dimensional non-equilibrium two-phase thermal-hydraulic system code. The RELAP5 code has been successfully applied to PWR and BWR reactors. Because of the uniqueness of the thermal-hydraulic system of RBMK design, the assessment study is required to adapt the RELAP5 code to RBMK reactors.

The RELAP5/MOD3.2 model of the Ignalina plant is schematically illustrated in Figure 2. Key features of this model are as follows:

- Both loops of the MCC are represented. Flow paths within a loop are modelled by one or more passes. In turn, a core pass model uses one or more equivalent fuel channels. The equivalent fuel channel is an abstract that conserves the heat generated in a group of real channels, as well as hydraulic properties of this group. The equivalent fuel channels are modelled by multiple axial and radial control volumes.
- Heat transfer among the equivalent fuel channels is approximated by means of heat exchange through the graphite moderator gaps to the reactor cavity gas circuit.
- Steam paths that serves for vapour removal from steam separators are represented explicitly, including steam lines, steam relief valves, etc.
- Feed water system and ECCS are represented explicitly.

The model of the left loop is shown in Figure 3. Loop model consists of one equivalent core pass. The fuel channels of this core pass are represented by an equivalent channel (21-23) operating at an average power and coolant flow rate. It is connected on one end to the Group Distribution Header (GDH) (19) by the lower water communication line (21) via its flow control valve (20). The other end of the equivalent channel is connected to the steam separator volume (3) by the steam-water communication line (23).

The steam separator is represented by four control volumes (1-4). All downcomers are represented by a single equivalent pipe (6) further subdivided into a number of control volumes. The pump suction header (7) and the pump pressure header (13) are represented by branch objects. Three operating MCPs are represented by one equivalent element (8-10) with check (11) and regulating (12) valves. The stand-by MCP is not modelled. The bypass line between the pump suction header (7) and the pump pressure header (13) is modelled, but manual valves (15) are closed. This is in agreement with a modification done recently at the Ignalina NPP. The loop model includes feed water injection into the steam separator through the mixer (5) and ECCS injection into the GDH (19). Elements 16, 17, and 18 represent the flow-limiting device, GDH inlet pipe and check valve. ECCS bypass line (24) is modelled, too.

The right loop model is shown in Figure 4. Compared to the model for the left loop, this model includes a more detailed representation of the loop section between the pressure header

(13) and the steam separator (1-4). The MCP system is modelled in more detail manner also (it is modelled with three equivalent pumps).

Loop model consists of two equivalent core passes. One core pass represents GDH. Fuel channels from this GDH are represented by four equivalent channels of three power levels (one single channel with maximum power level, one single channel with average power level, one single channel with minimum power level and 40 channels with average power level). The other core pass represents the other 19 GDHs. The channels of this pass are simulated by an equivalent fuel channel of average power.

In the RELAP5 model of Ignalina NPP the core is represented by a number of equivalent channels. Heat structures of the equivalent fuel channel simulate only the active region in the reactor core (top and bottom reflectors are not modelled). The fuel element is modelled with an equivalent four radial node model. One of these radial nodes is for the fuel pellet, one for the gap region and two for the cladding. The fuel channels and graphite stacks are modelled with an equivalent six radial node model. Two of these radial nodes are for the fuel channel wall, one for the gap and graphite rings region and three for the graphite blocks. The fuel element, fuel channel, graphite rings and the graphite blocks are modelled with 14 axial segments, 0.5 m long each. The square graphite stack is represented by an equivalent cylindrical volume. Energy deposition in the graphite is calculated as a fixed fraction (4.15%) of the fission power plus a fixed fraction (11.12%) of the decay power. This largely represents the gamma-ray energy deposition in the graphite. The net result is that at full power approximately 5% of the total energy is deposited in the graphite.

2.2. INTEGRAL PHENOMENA VERIFICATION

2.2.1. Single MCP trip

On May 14, 1996 one MCP at Ignalina Unit 2 was inadvertently tripped. The plant conditions prior to the event were as follow:

- The reactor operated at the 3400 MW_{th};
- Pressure in the steam separators was maintained at 6.86 MPa by the pressure regulator;
- One turbine generator operated in a pressure maintenance mode and the other operated in power control mode;
- Six MCPs were in operation, each provided a coolant flow of 7860 m^3/h.

At 18:10 alarm signal was generated in response to two preferred electrical buses switch off. It leads that one of six MCPs was tripped. AZ-4 signal was generated due to loss of power to the MCP. The CPS rods started to move. The turbine generator (which before the accident operated in the power control mode) switched from power control mode to steam separators pressure maintenance mode. As the flow through the pump dropped to zero (after about 5 seconds from the beginning of the accident) the check valve (downstream of this MCP) closed, preventing a reverse flow through the tripped pump. The setback in an AZ-4 mode was controlled at two percent per second of the maximum design power by the CPS. The reactor power was reduced to 2700 MW_{th} after about 16 seconds from the switching off of the MCP, Figure 5.

Comparison between calculated flow rates obtained by RELAP5 model and real measured data is presented in Figure 6 and Figure 7. The figures show a favourable coincidence of MCP throughputs and coolant flow rate through the reactor. After one MCP trip, the throughput of two running pumps increased by ~3000 m^3/h. However, the total coolant flow through the affected loop decreased from 23500 m^3/h to 19000 m^3/h.

2.2.2. One MCP trip with failure of check valve

The similar event took place on January 23, 1998. In this case, the one MCP trip with failure of check valve occurred. The plant conditions prior to the event were as follow:

- The reactor operated at the 3700 MW_{th},
- Pressure in the steam separators was maintained at 6.86 MPa by the pressure regulator,
- One turbine generator operated in a pressure maintenance mode and the other operated in power control model,
- Six MCPs were in operation, each provided a coolant flow of 7750 m^3/h (23250 m^3/h for each loop of MCC).

At 10:21 one MCP was switched off by mistake. AZ-4 signal was generated due to loss of power to the MCP. CPS rods started to move. The turbine generator (which before the accident operated in the power control mode) switched from power control mode to steam separators pressure maintenance mode. As the flow through the tripped pump dropped to zero (after about 2 seconds from the beginning of the accident), the check valve (downstream of this MCP) had start to close, however, it remained to be open. The setback in an AZ-4 mode was controlled at two percent per second of the maximum design power by the CPS. The reactor power was reduced to 2860 MW_{th} after about 14 seconds from the beginning of the accident. The operator noticed that coolant flow through the left loop of the reactor calculated as a sum of fuel channel flow rates was 13400 m^3/h. There were flow rates less then 10 m^3/h in some peripheral channels. At 10:23 operator closed the throttling regulating valve of the tripped MCP. At 10:24 reactor power was reduced manually down to 2100 MW_{th}. At 10:25 Operator decreased reactor power down to 1900 MW_{th}. At 10:25 operator increased the throughput of two running pumps by opening their throttling-regulating valves.

Thus, it was noticed that after the MCP trip the two other running pumps provided a coolant flow of 20000 m^3/h while the coolant flow through the channels of the right side of the reactor decreased to 13400 m^3/h. On this base the conclusion has been made that the check valve downstream of the tripped pump failed to close.

The analysis of this transient employing RELAP5 mode was performed. The obtained results are shown in Figure 8 - Figure 11. The reactor power was reduced to 2860 MW_{th} after about 14 seconds from the beginning of the accident, Figure 8. The comparison of calculated and measured pressure behaviour in MCC is presented at Figure 9. Because of the check valve failure, the flow through the tripped pump reversed. At the same time, each of two running pumps increased its throughput because the pressure gradient across the pumps was reduced due to the flow bypass. The increased pump throughput effectively compensated for the reversed flow through the failed valve. The net flow supplied to the affected core side decreased from 2300 m^3/h to 15500 m^3/h (see Figure 10). The throttling-regulating valve of the tripped pump closed after 90 seconds from the beginning of the transient and decreased the reversed flow by 2000 m^3/h. At this moment the real coolant flow through the reactor side reached about 18000 m^3/h. The opening of throttling-regulating valves of running pumps after 220 second increased the coolant flow to 20000 m^3/h. Comparison between measured and calculated MCPs throughputs is presented in Figure 10. A comparison of measured data and calculated flows trough fuel channels is shown in Figure 11.

2.2.3. Loss-of-all-MCPs transient

In order to benchmark the RELAP5 model transient analysis calculations were performed for the loss-of-all-MCPs event. This is an actual transient that occurred at Ignalina NPP. On March 26, 1986 all six operating MCPs at Ignalina Unit 1 were tripped simultaneously. Before this event reactor operate at thermal power level of 4650 MW. In response to multiple pump trip, an emergency protection signal AZ-1 was generated and reactor was shutdown. The MCC flow decreased in response to the MCPs cost-down. Long-term flow was due to natural circulation in the MCC. Analysis results are compared against the plant data. It should be noted that the plant data included strip charts and digital data recorded by plant data collection system. There were some inconsistencies between the strip charts and digital data. In most cases, the digital data were used for comparison.

Figure 12 shows the normalised reactor power. The reactor trips almost immediately, with rapid power decrease to decay heat level. The analysis results agree well with observed reduction in power from the plant data. Figure 13 shows the MCC flow rates. Digital data for the flow rate in twelve individual channels was available. Flow rates through individual channels are presented in Figure 14. Flow rate through the fuel channel decreases due to loss of forced circulation by the MCPs. The coast-down occurred within approximately 40 seconds from the beginning of transient. The flow coast-down to a long-term natural circulation at flow rate equal approximately 15 percent of the initial flow. The total steam flow rate is presented in Figure 15. The analysis results agree well with the measured flow rates.

Figure 16 shows pressure in the MCC. The pressure initially decreases, but later increases when the turbine control valves closes. Based on the benchmarking results, it is concluded that RELAP5 model agrees well with the plant data.

2.2.4. Three MSRVs LOCA event

Three Main Safety Relief Valves (MSRVs) of third group were spuriously opened at the Ignalina NPP Unit 1 at November 27, 1986. Before the event, reactor was operated at thermal power level of 4350 MW. One turbine was operating in the “load following” logic, second - in ”pressure following” logic. At time 5:32 p.m. operator noticed that pressure in the steam separator decreased and as well decreased energy generation for one turbine. At the same time pressure increase in the upper steam reception chamber of right ACS tower. Operator decided that steam discharge valve to ACS pool of the right side has opened and attempted to close this valve. At time 5:34 p.m. the pressure stopped to decrease. At time 5:35 p.m. emergency protection AZ-1 was activated and reactor has been shutdown.

Three MSRVs spuriously opened at 59 s and closed at 261.5 s from the beginning of the accident. Turbine Control Valve of the turbine, which operates in the 'pressure following' logic, immediately after three MSRVs opening, decreases steam flow rate until the value, which is equal to amount of steam through three MSRVs. The comparisons of ATHLET and RELAP5 calculation results to actual plant data from the event are shown in Figure 17 trough Figure 20.

Steam through three MSRVs gets into fifth pool of ACS. Water in the fifth pool evaporates in about three minutes. Steam pushes water from 1 - 4 pools and gets to the reinforced compartments of ACS. It is assumed in calculations, that at the time 260 s due to pressure increase, AZ-1 is activated. After AZ-1 activation both turbines starts to decrease their throughput down to 150 kg/s. Figure 17 shows the steam separators pressure response. After spurious opening of MSRVs, pressure starts to decrease. The pressure decrease causes the closure of both Turbines Control Valves after reaching the pressure of 5.98 MPa and 5.49 MPa, correspondingly. Then pressure in steam separators starts to increase. After AZ-1 activation due to MCP throttling-regulating valves operation, coolant flow rate through each MCP decreases down to 6500 m^3/h, Figure 18. The total steam flow rate to turbines and for local consumers is shown in Figure 19. Reactor scram leads to decrease of the water volume in the steam separator at the initial time of event, but later it is restored. However, the water level is maintained by feed water supply system, Figure 20.

Results of 1986 event, caused by three MSRVs spurious opening, were reviewed. Computed results were compared to actual plant data from the event (Figures 17-20). Both RELAP5 and ATHLET models agree well with the plant data when similar boundary conditions are imposed. The comparison of results generated by the RELAP5 and ATHLET codes shows a very good agreement.

Calculations showed, that three MSRVs were opened for about 200 seconds and about 40 tons of steam were discharged to ACS. The enthalpy of discharged steam was about $2.788 \cdot 10^3$ kJ/kg. Estimated amount of energy, which was discharged to ACS, can be used for further analysis of ACS response by using other codes, such as CONTAIN or DRASYS.

2.2.5. Inadvertent actuation of ECCS

On the November 22, 1995 fast acting valves, which are used for isolation of ECCS accumulators from MCC, at Unit 1 of the Ignalina NPP were spuriously opened. Before this event reactor operated at thermal power level of 3525 MW. As a result of inadvertent actuation of ECCS cold water flowed to the GDHs of one (right) loop of MCC.

Reactor local automatic control system was activated. At about 220 second from the beginning of the event operator started to decrease reactor power with aim to diminish power irregularity between loops. During the next 650 seconds the reactor power was decreased manually by operator to thermal power of 3350 MW.

ECCS water injection of the water takes about 30 seconds and, in accordance with event description, it was terminated by operator. During this period the water level in the ECCS accumulators decreased from 5.60 m to 5.20 m and the pressure decreased from 8.93 MPa to 7.92 MPa. However, in such short term spurious delivery of ECCS water to the primary circuit had a small influence on the main process parameters in the primary circuit, but noticeable changes of the water level in steam separators, steam and feed water flow rates has been observed.

For better presentation of results, it was assumed in calculations that ECCS fast acting valves have spuriously opened at 30th second. After opening of fast acting valves, water from eight ECCS accumulators is supplied to GDH of right loop. The water flow rate from accumulators is shown in Figure 21. Injection of the ECCS water to the right MCC loop takes about 29 seconds in accordance to RELAP5 calculations and only about 18 seconds in accordance to ATHLET analysis. Maximum flow rate is about 600 kg/s. This leads to water level decrease in accumulators in about 0.4 m (see Figure 22). Results of RELAP5 and ATHLET analysis are in good agreement with actual data. Water supply continues until pressure in accumulators and pressure in the GDH become equal, as it is shown in Figure 23. However, this is the main reason why supply of the ECCS water to primary circuit has been terminated. The pressure in accumulators in RELAP5 and ATHLET calculations is higher than in experimental data of Ignalina NPP, although characteristics of diagrams are the same. The comparison of results generated by the ATHLET and RELAP5 codes shows a very good agreement. Very likely, that measured at Ignalina NPP pressure is with calibration error.

Due to activation of ECCS the water level in the steam separators of the right loop increases in about 90 mm (see Figure 24). The steam flow rate to turbine, which operates in the ‘pressure following’ logic, also decreases at the same time (see Figure 25). Later the water level in the steam separators and steam flow rate changes due to change of the reactor power by operator. The flow rate of feed water is presented in Figure 26. Calculations performed both with RELAP5 and ATHLET compare favourably with the plant data.

2.3. SINGLE PHENOMENA VERIFICATION

In addition to the validation of the integral experiments (comparisons of RELAP5 model against plant data - described above), the validation of model against the separate experiments were performed. From the separate effects, the critical discharge coolant flow rate and critical heat transfer from the RBMK-1500 fuel bundles to the coolant were selected.

The Figure 27 represents the comparison of calculated and experimental data of critical discharge flow rate through GDH flow limiter. Calculations were performed for test case with initial pressures from 0.5 to 8.6 MPa and water subcooled by 30 degrees. As it is shown in Figure 27, the values calculated with RELAP5 code are in a good agreement with experimental data.

RELAP5 uses the Groeneveld tables with correction coefficients for CHF calculations. The distribution of the predicted CHF of RELAP5 and experimental one is presented in the Figure 28. In Figure 28 it is shown that the CHF values calculated with Ignalina NPP RELAP5 model are slightly lower than experimental ones, described in [13], [6]. It means, that RELAP5 calculations give a more conservative view.

When coolant flow rate through the channels of the different power is decreasing, CHF is reached at different time moment in these channels. The function of minimum power level, at which the CHF is reached from coolant flow rate through the fuel channel, is presented in Figure 29. First of all CHF appears only in the top part of the fuel channel, which leads to the increase of fuel cladding temperature by approximately 100 - 150 ºC. When coolant flow rate through the channels decreases even more, CHF appears in the entire length of the fuel channel and fuel cladding temperature starts rapidly increase. Dependence between power of fuel channel and coolant flow rate, when CHF is reached, is obtained by employing RELAP5 code in order to create Ignalina NPP model. Figure 29 shows the comparison of calculated dependence (points) and experimental CHF boundary [26]. As it seen from the Figure 29, the points obtained with RELAP5 code are located on the both sides of the experimental curve. Points, which represent coolant flow rate, when CHF just starts to appear, are above the experimental curve. Points, which represents coolant flow rate, when CHF appears in the entire length of the fuel channel, are below the experimental curve. That shows a good agreement between experimental data [26] and calculational results by employing RELAP5 code.

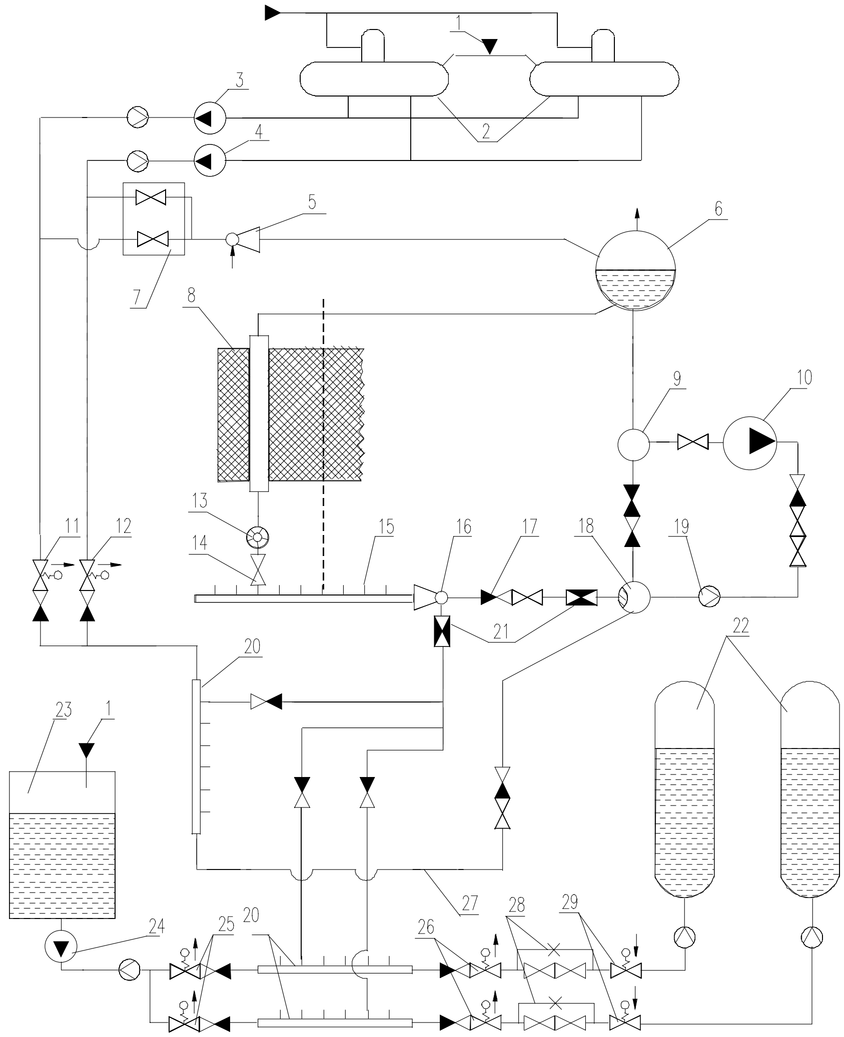


Figure 1 Schematic of the Main Circulation Circuit and Emergency Core Cooling System of Ignalina NPP

1 - make-up from demineralised water reservoir, 2 - deaerators, 3 - auxiliary feed water pump, 4 - main feed water pump, 5 - mixer for the feed water and water from the PCS, 6 - separator drum, 7 - main feeder and auxiliary feeder, 8 - reactor, 9 - suction header, 10 - MCP, 11 - fast acting opening valve on pipelines from auxiliary feed water pumps to ECCS header, 12 - fast acting opening valve on pipelines from main feed water pumps to ECCS header, 13 - ball type flow-rate meter, 14 - isolation and control valve, 15 -GDH, 16 - mixer for the main coolant and the ECCS water, 17 - check valve, 18 - pressure header, 19 - throttling type flow-rate meter, 20 - ECCS header, 21 - flow limiters, 22 - ECCS accumulators, 23 - hot condensate chamber of ACS towers, 24 - ECCS pump, 25 - fast acting opening valve on pipelines from ECCS pumps to ECCS header, 26 - fast acting opening valve in pipelines from ECCS accumulators to ECCS header, 27 - ECCS bypass line, 28 - throttle bypass, 29 - fast acting closing valve in pipelines from ECCS accumulators.

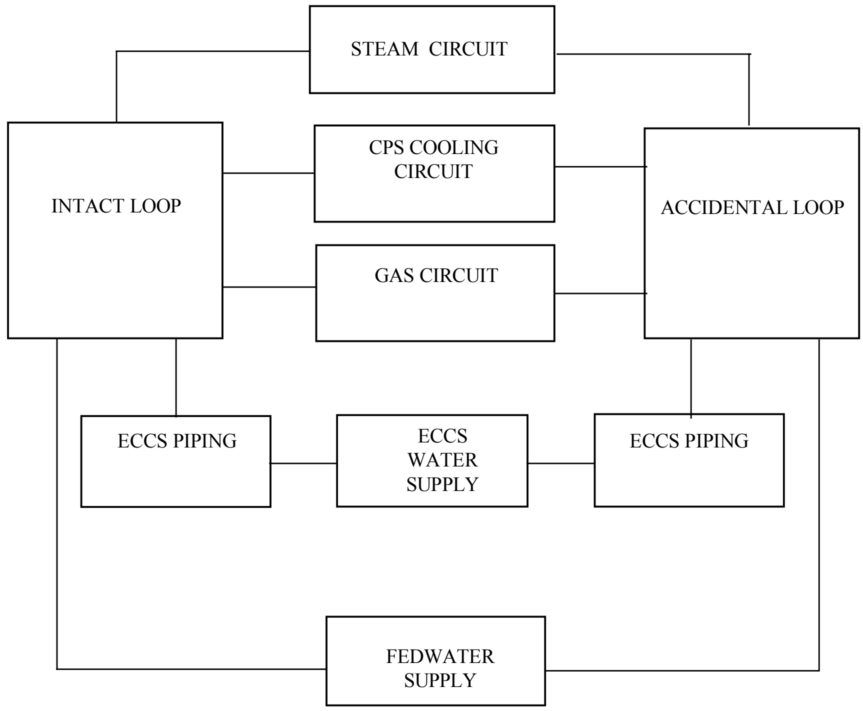


Figure 2. General thermal-hydraulic Ignalina NPP model

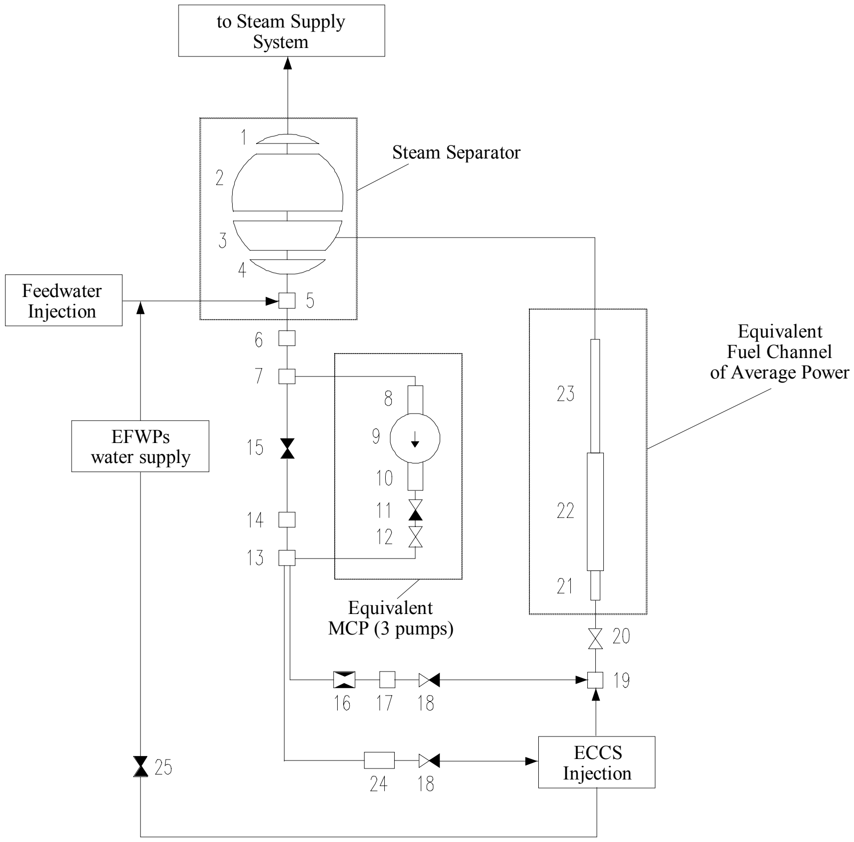


Figure 3. Schematic representation of left loop of the MCC

1 - top part of steam separator, 2 - steam separator part with submerged perforated sheet, 3 - steam separator part with steam-water pipes connection, 4 - bottom part of steam separator, 5 - mixer, 6 - downcomers, 7 - suction header, 8 - suction pipes of MCP, 9 - MCP, 10 - pressure pipes of MCP, 11 - check valve, 12 - throttling regulating valve, 13 - pressure header, 14 - bypass line, 15 - manual valve, 16 - GDH flow limiter, 17 - GDH inlet pipe, 18 - check valve, 19 - GDH, 20 - isolation control valve, 21 - lower water communication line, 22 - equivalent fuel channel (average power), 23 - steam-water communication line, 24 - ECCS bypass line, 25 - valve in the pipeline for ECCS injection into steam separator

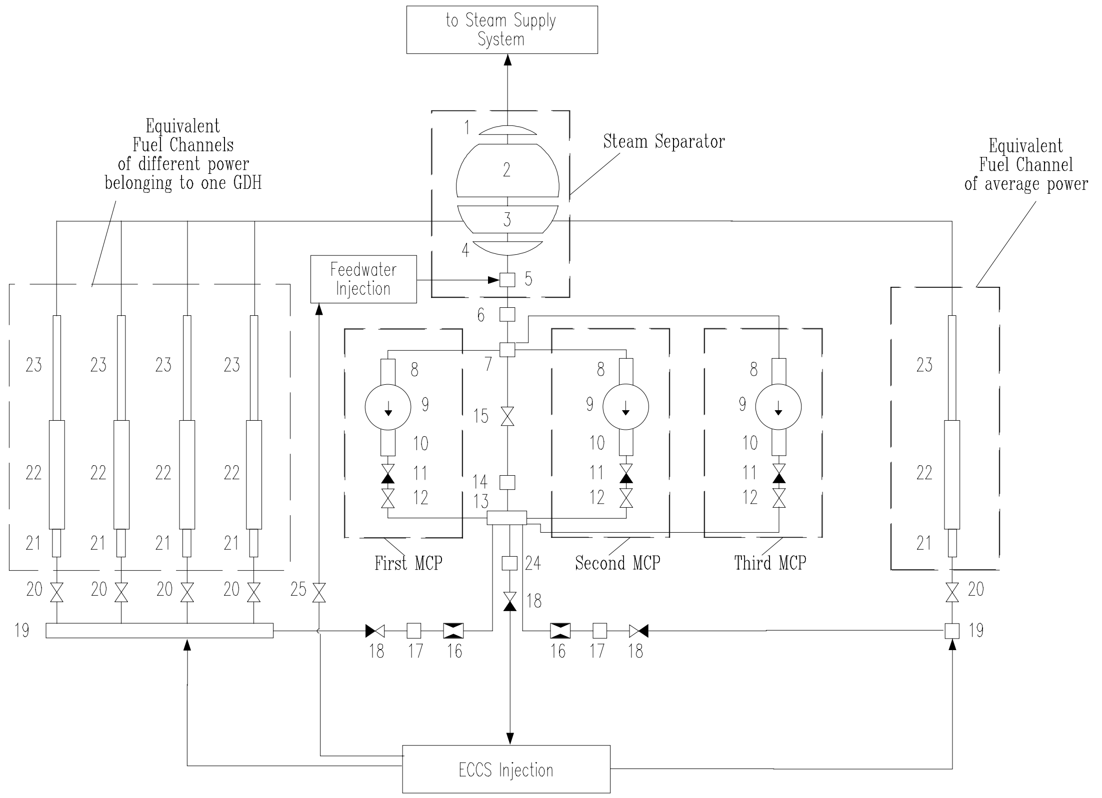


Figure 4. Schematic representation of right loop of the MCC

1 - top part of steam separator, 2 - steam separator part with submerged perforated sheet, 3 - steam separator part with steam-water pipes connection, 4 - bottom part of steam separator, 5 - mixer, 6 - downcomers, 7 - suction header, 8 - suction pipes of MCP, 9 - MCP, 10 - pressure pipes of MCP, 11 - check valve, 12 - throttling regulating valve, 13 - pressure header, 14 - bypass line, 15 - manual valve, 16 - GDH flow limiter, 17 - GDH inlet pipe, 18 - check valve, 19 - GDH, 20 - isolation control valve, 21 - lower water communication line, 22 - equivalent fuel channel (average power), 23 - steam-water communication line, 24 - ECCS bypass line, 25 - valve in the pipeline for ECCS injection into steam separator

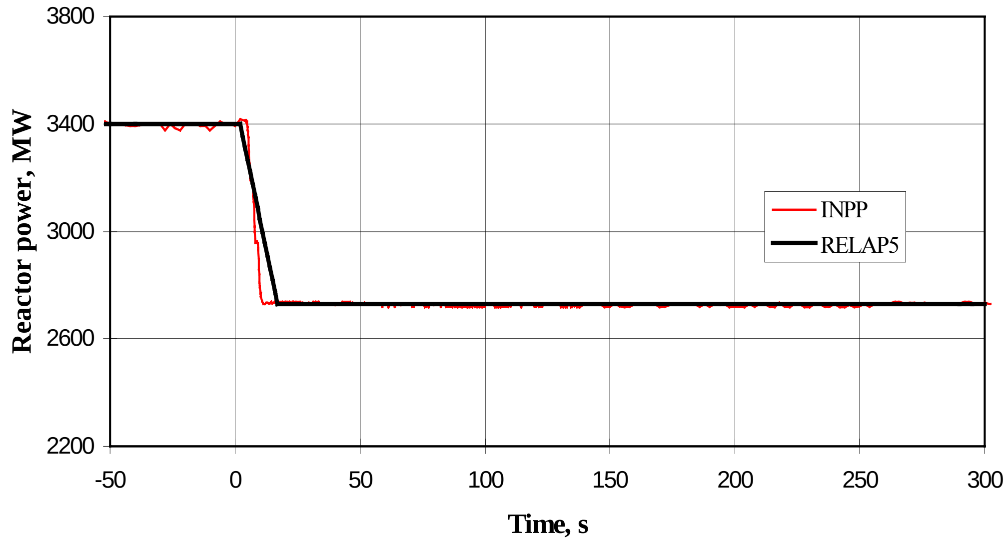


Figure 5. Single MCP trip. Reactor power

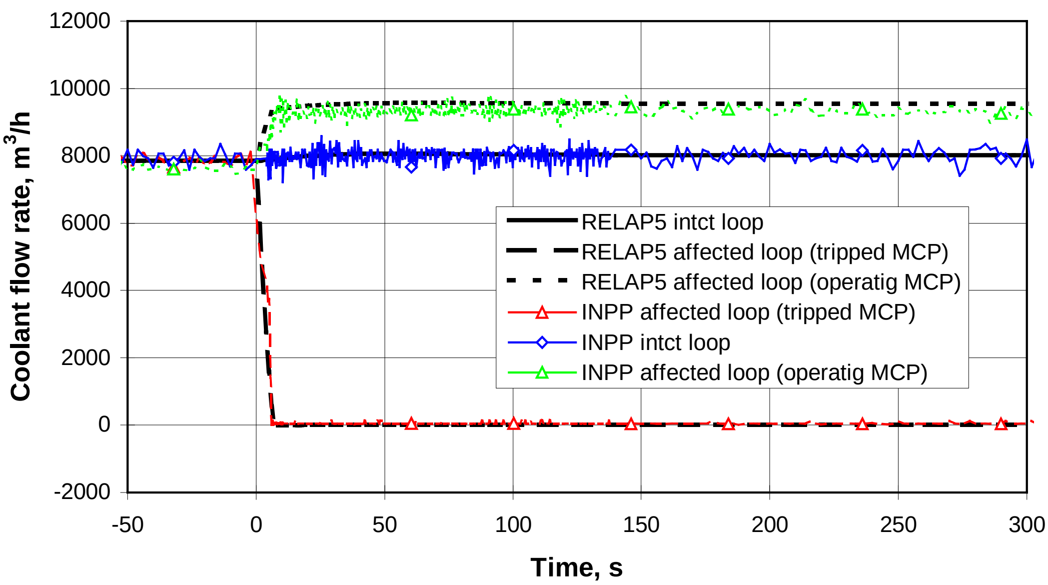


Figure 6. Single MCP trip. MCP throughput

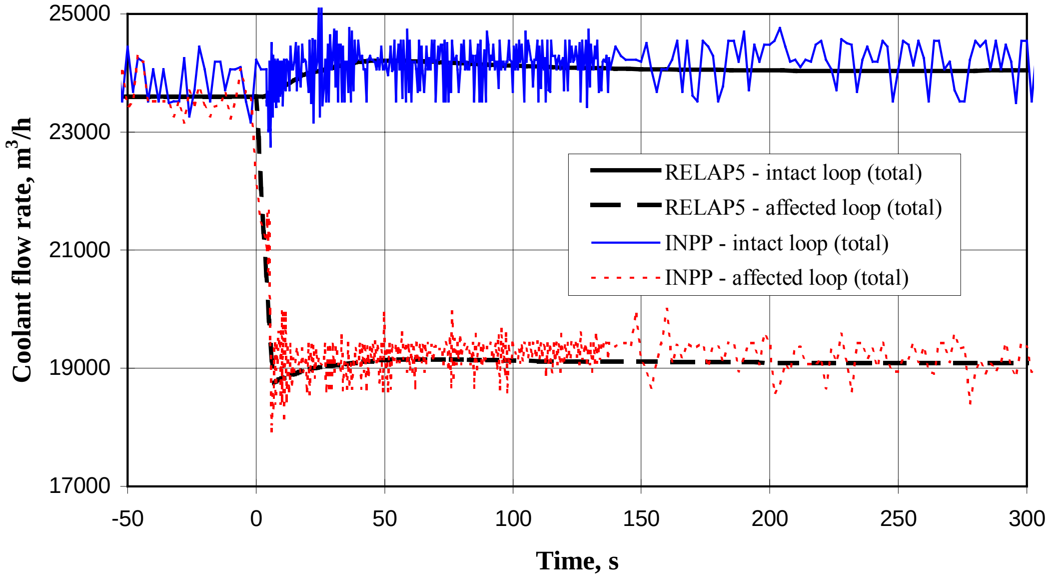


Figure 7. Single MCP trip. Coolant flow rate

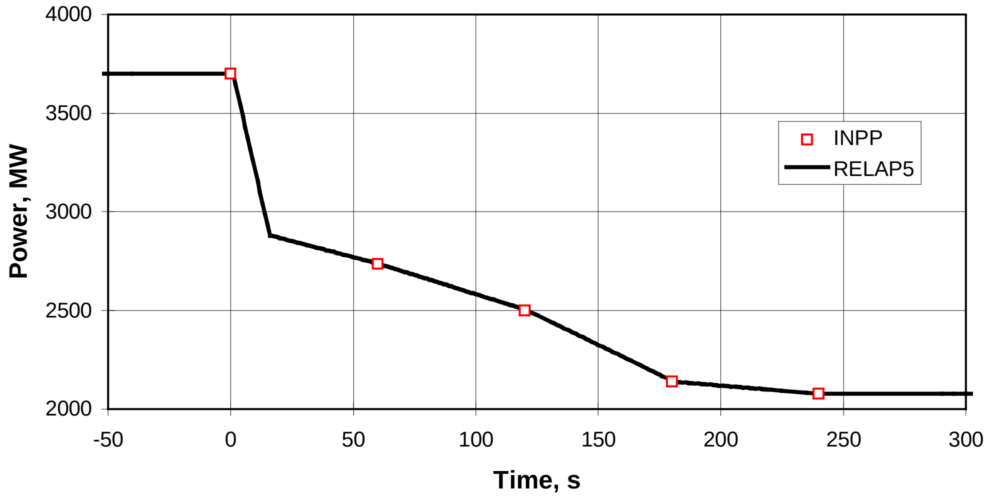


Figure 8. One MCP trip with failure of check valve. Power after MCP trip

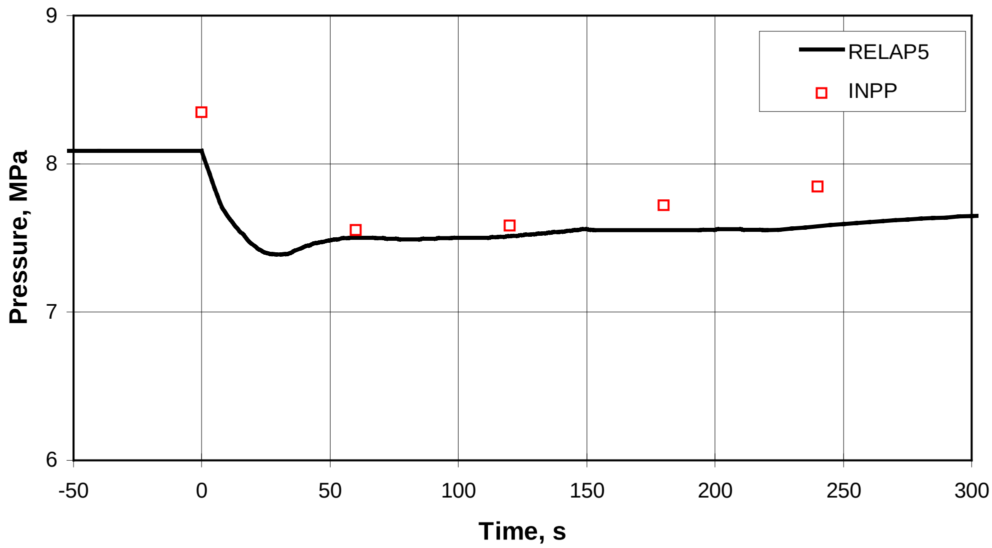


Figure 9. One MCP trip with failure of check valve. Pressure in pressure header of affected loop

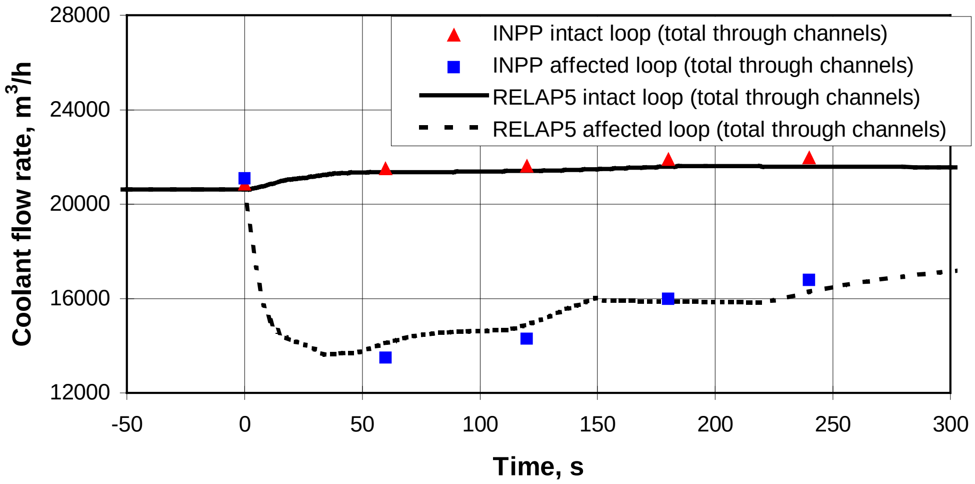


Figure 10. One MCP trip with failure of check valve. Coolant flow rate (as a sum of fuel channel flow rates)

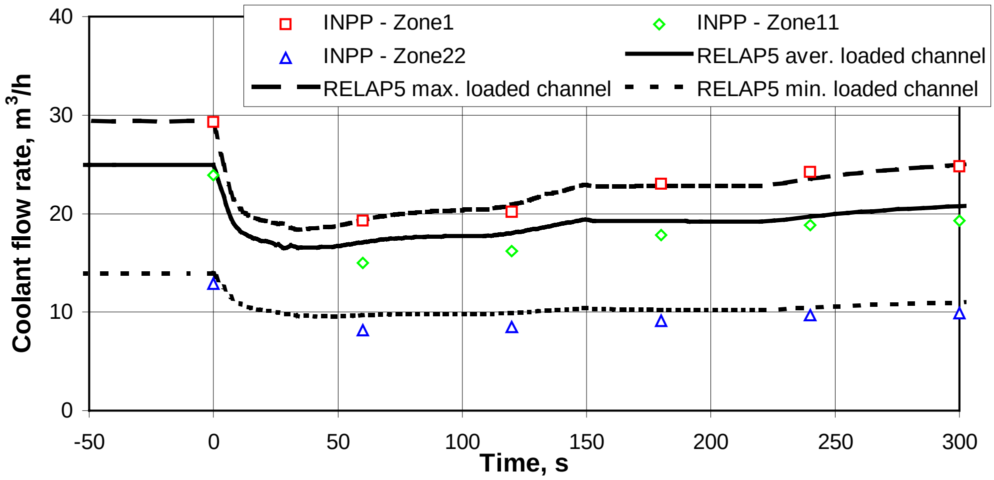


Figure 11. One MCP trip with failure of check valve. Flow rates in separate fuel channels

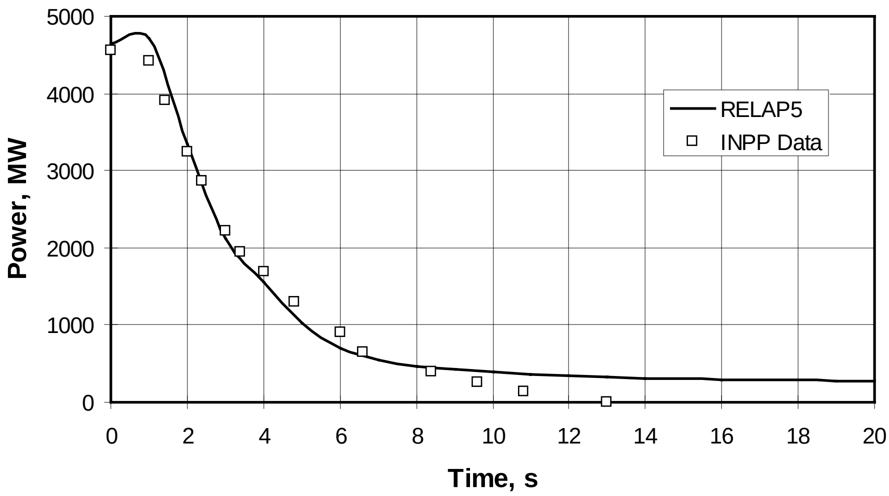


Figure 12. Loss-of-all-MCPs transient. Reactor power

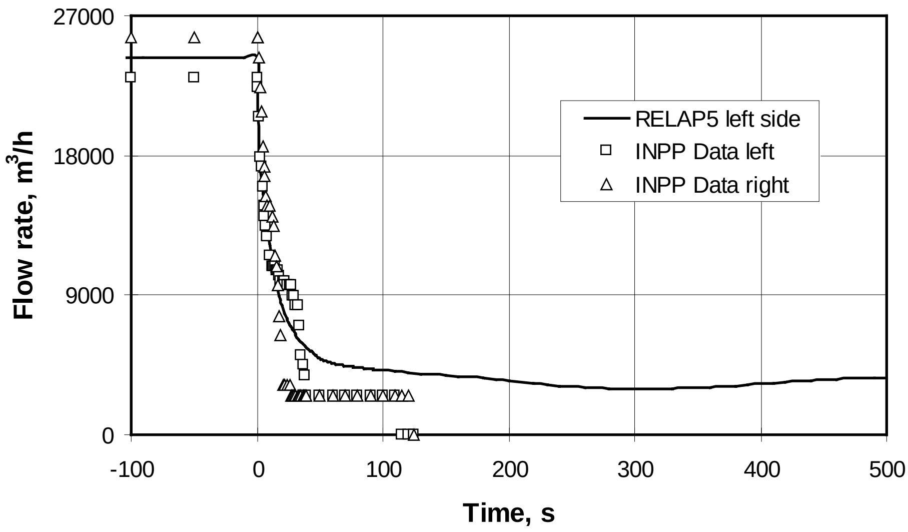


Figure 13. Loss-of-all-MCPs transient. Flow rate through MCP

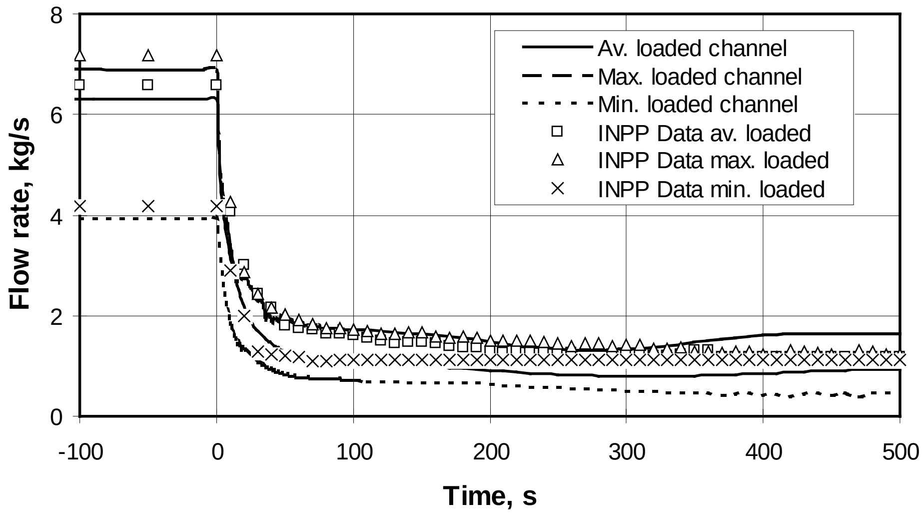


Figure 14. Loss-of-all-MCPs transient. Flow rate through individual channels

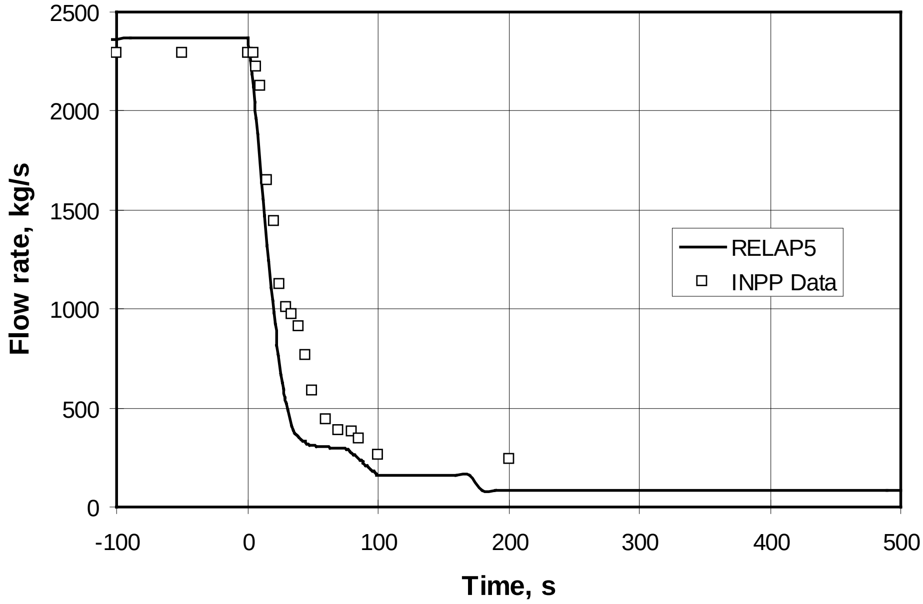


Figure 15. Loss-of-all-MCPs transient. Total steam flow rate to turbines

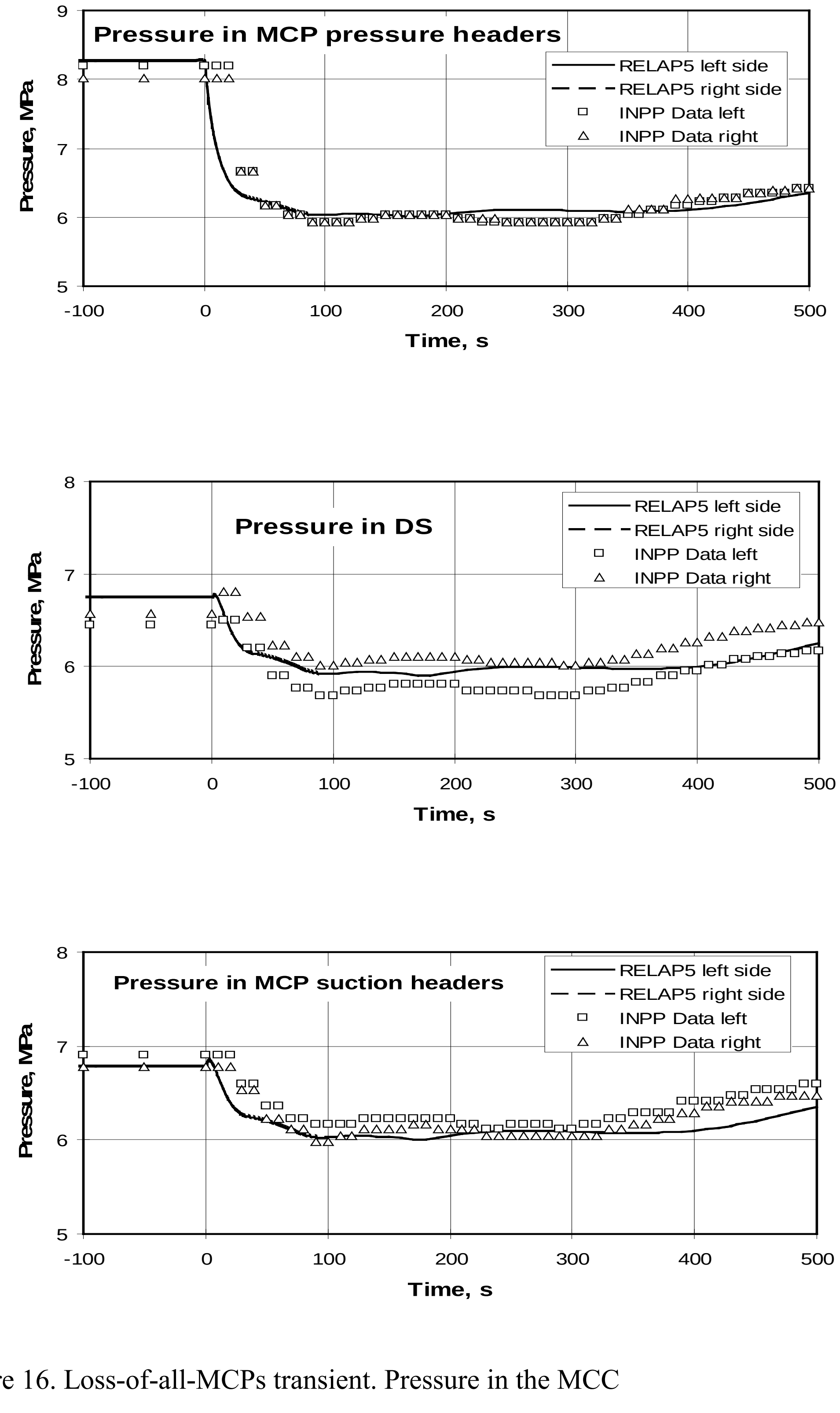


Figure 16. Loss-of-all-MCPs transient. Pressure in the MCC

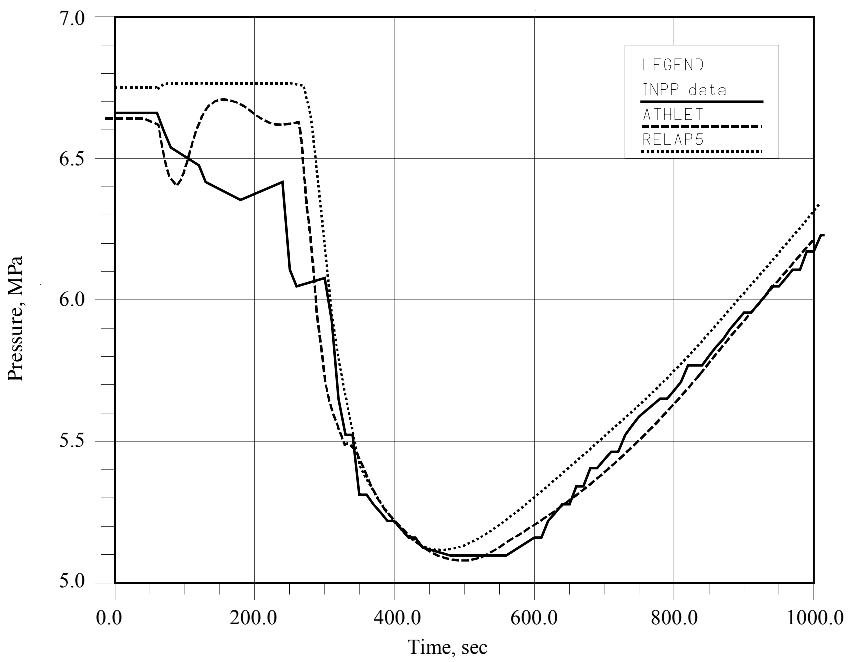


Figure 17. Three MSRVs LOCA event. Pressure in steam separators

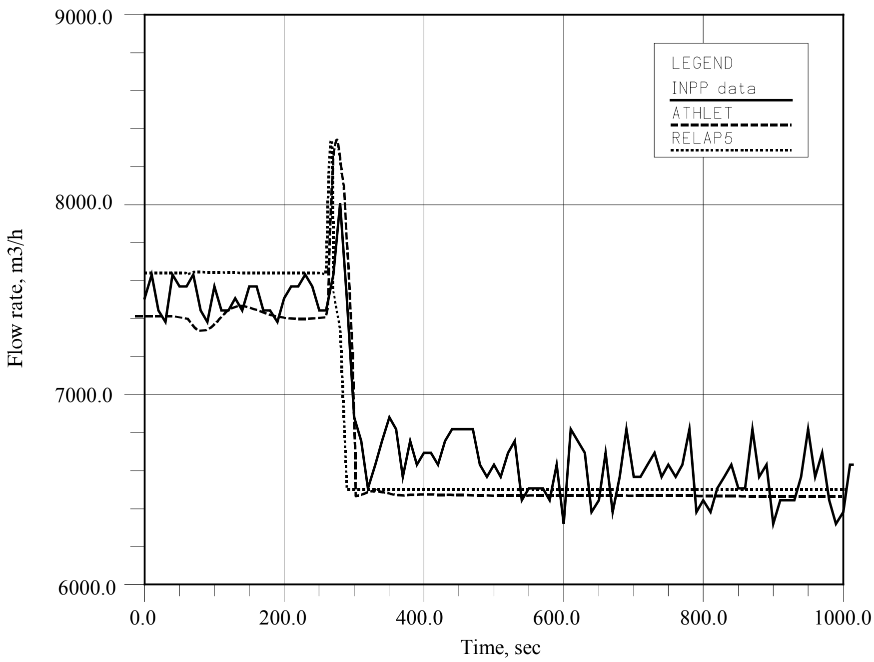


Figure 18. Three MSRVs LOCA event. Water flow through MCP

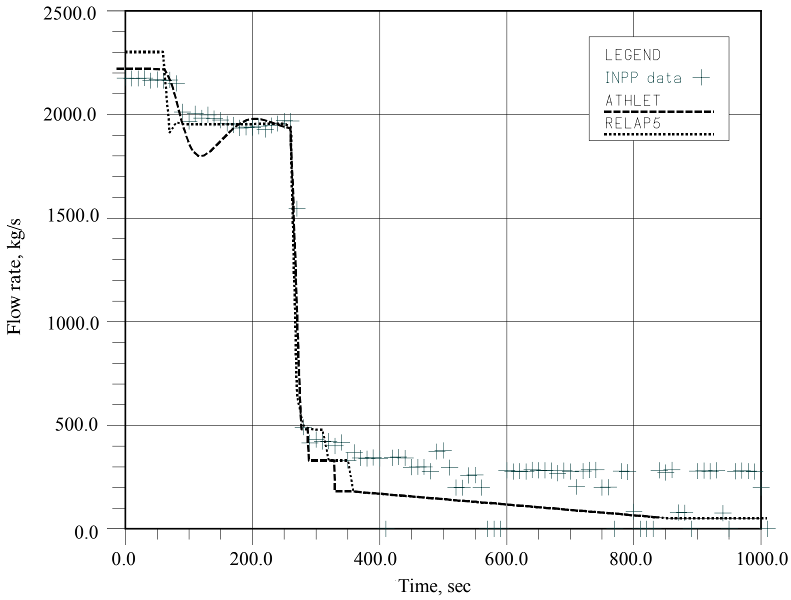


Figure 19. Three MSRVs LOCA event. Total steam flow rate

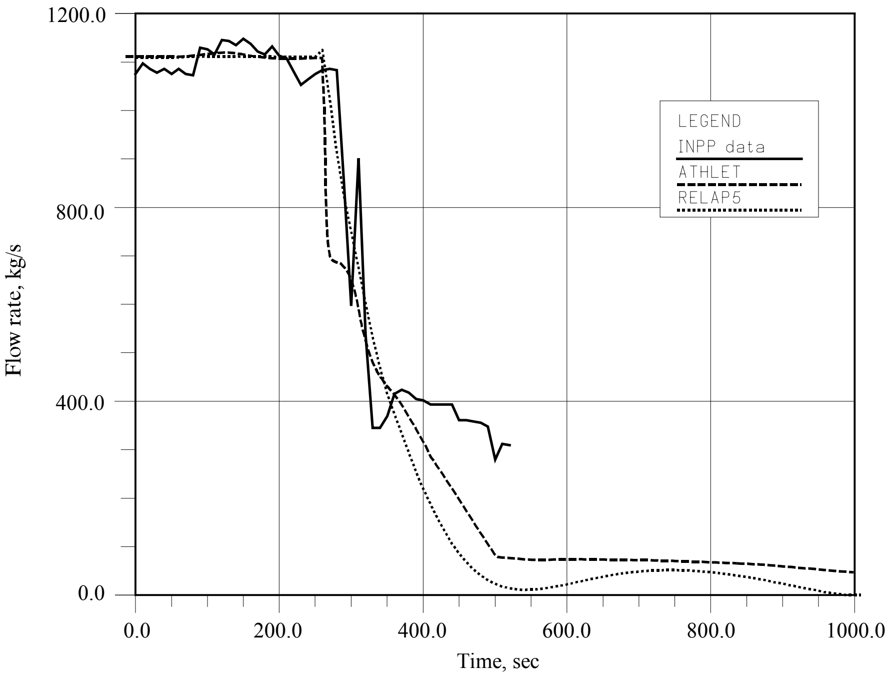


Figure 20. Three MSRVs LOCA event. Feed water flow rate

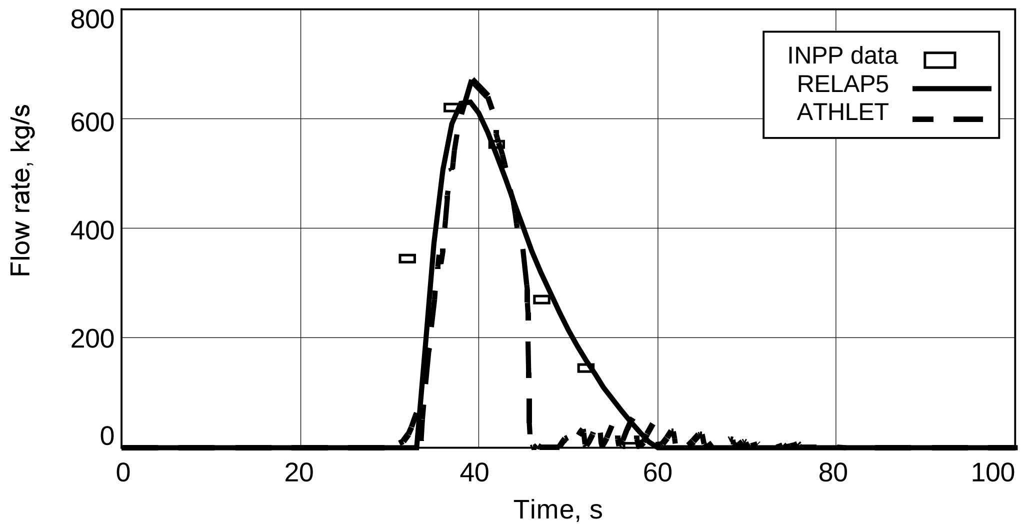


Figure 21. Inadvertent actuation of ECCS. Mass flow from the ECCS accumulators

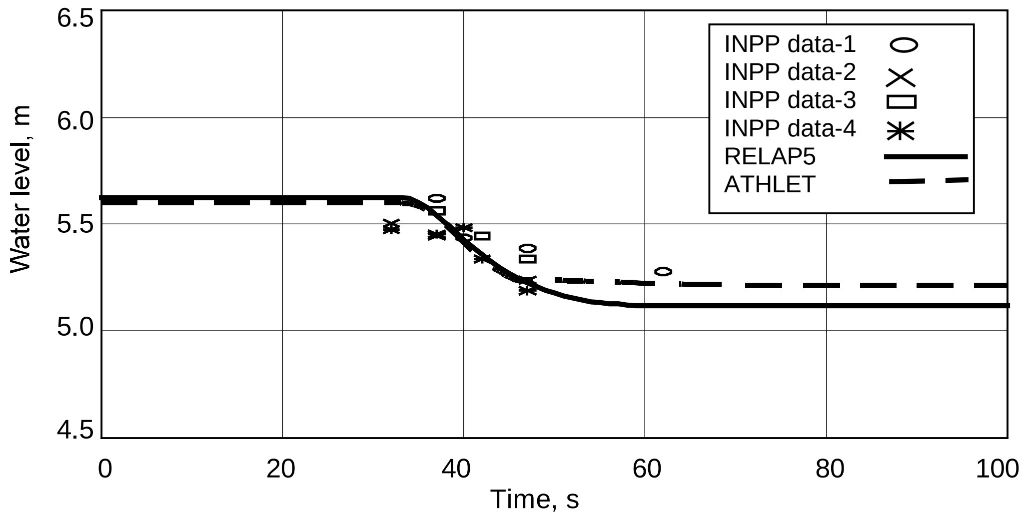


Figure 22. Inadvertent actuation of ECCS. Water level in the ECCS accumulators

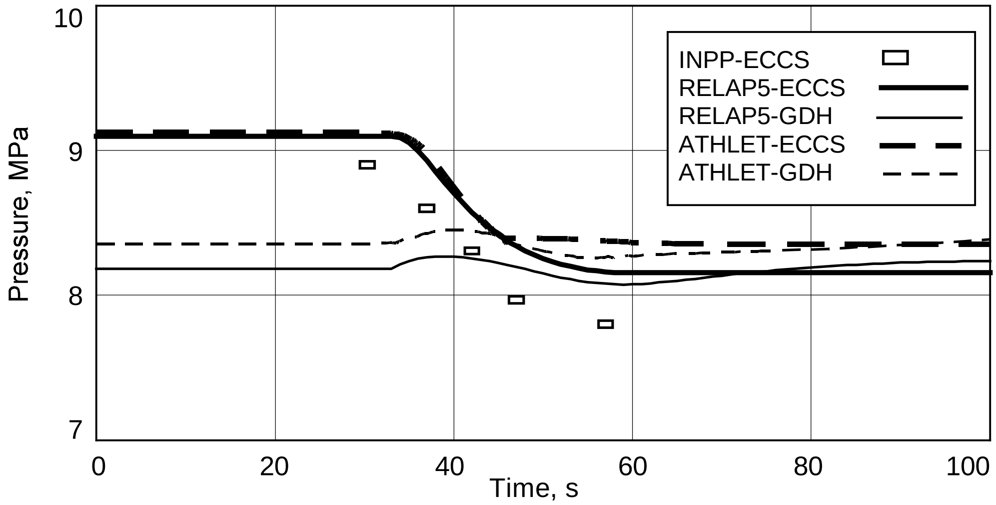


Figure 23. Inadvertent actuation of ECCS. Pressure in the ECCS accumulators

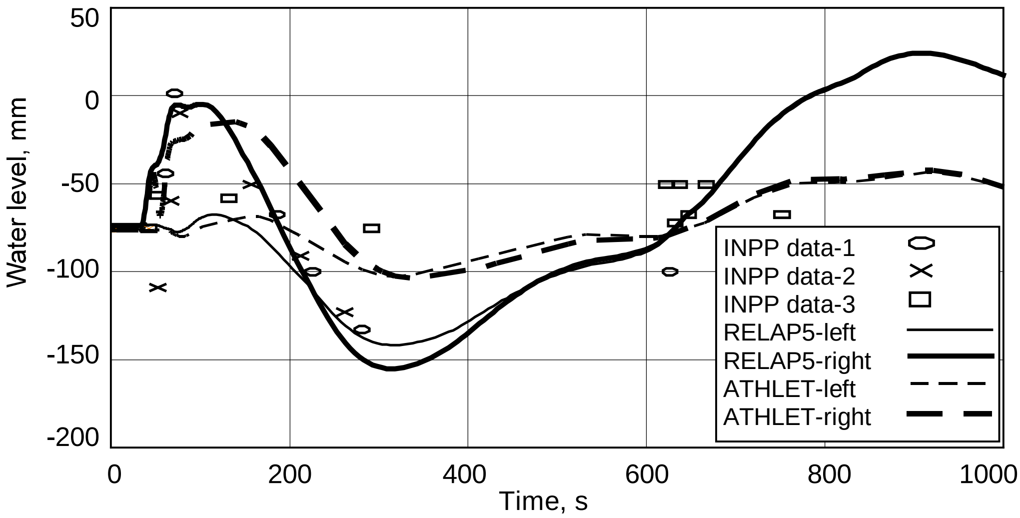


Figure 24. Inadvertent actuation of ECCS. Water level in the steam separators

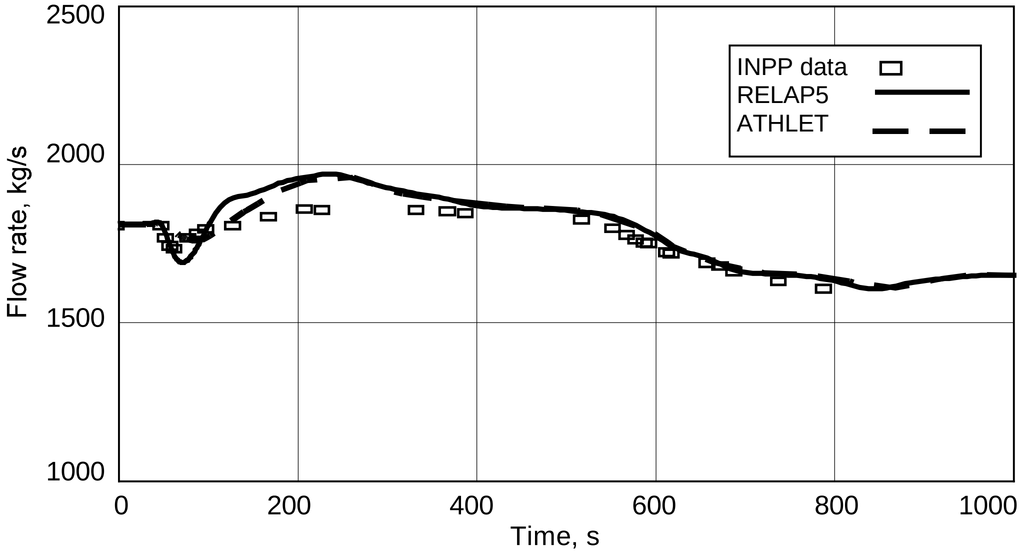


Figure 25. Inadvertent actuation of ECCS. Steam flow rate

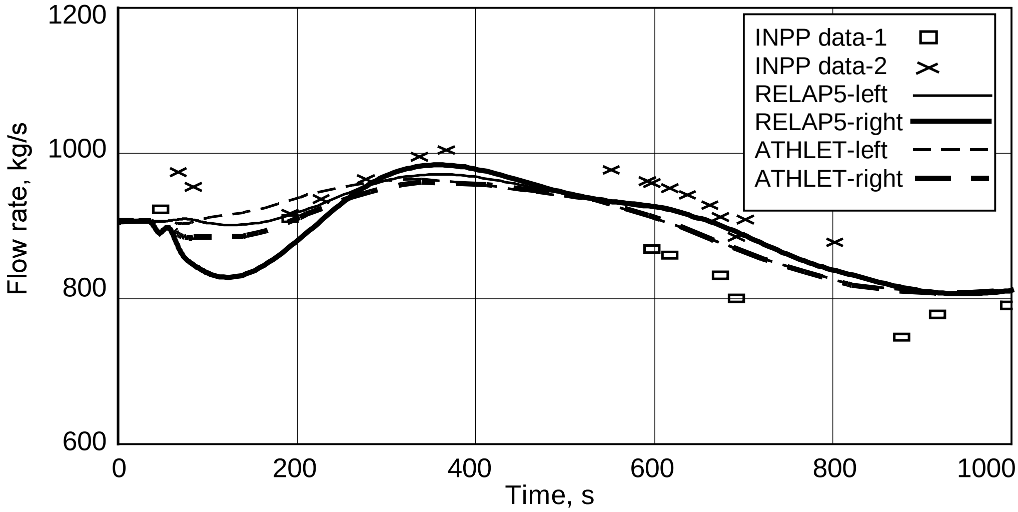


Figure 26. Inadvertent actuation of ECCS. Feed water flow rate

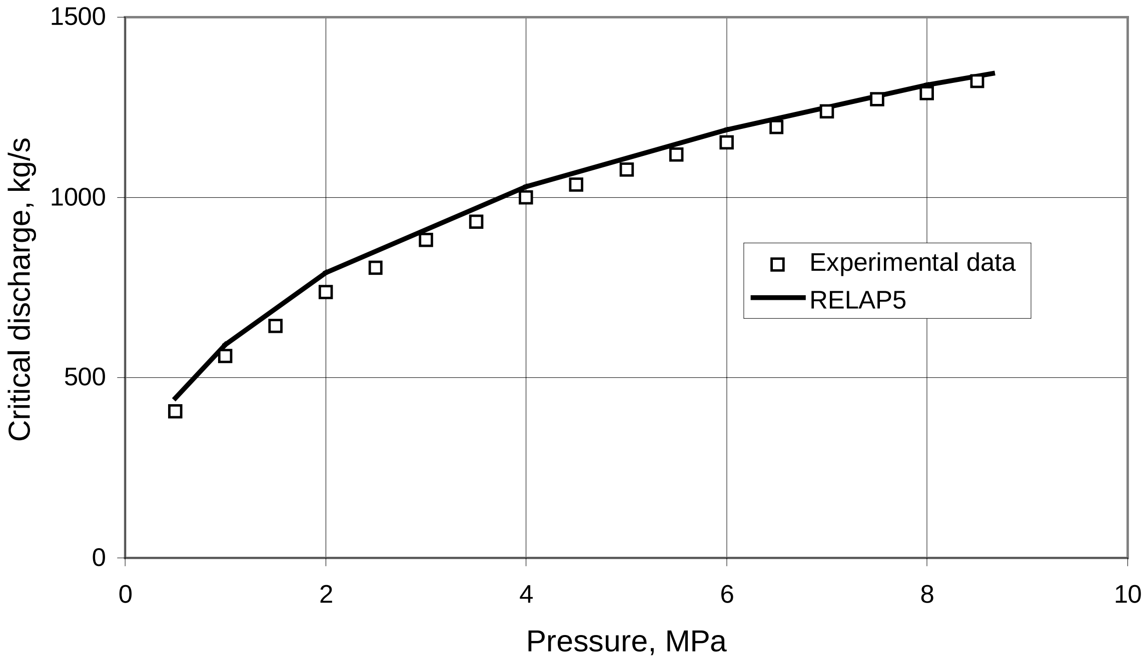


Figure 27. Single phenomena verification by employing Ignalina NPP model. RELAP5 prediction of GDH header flow limiter

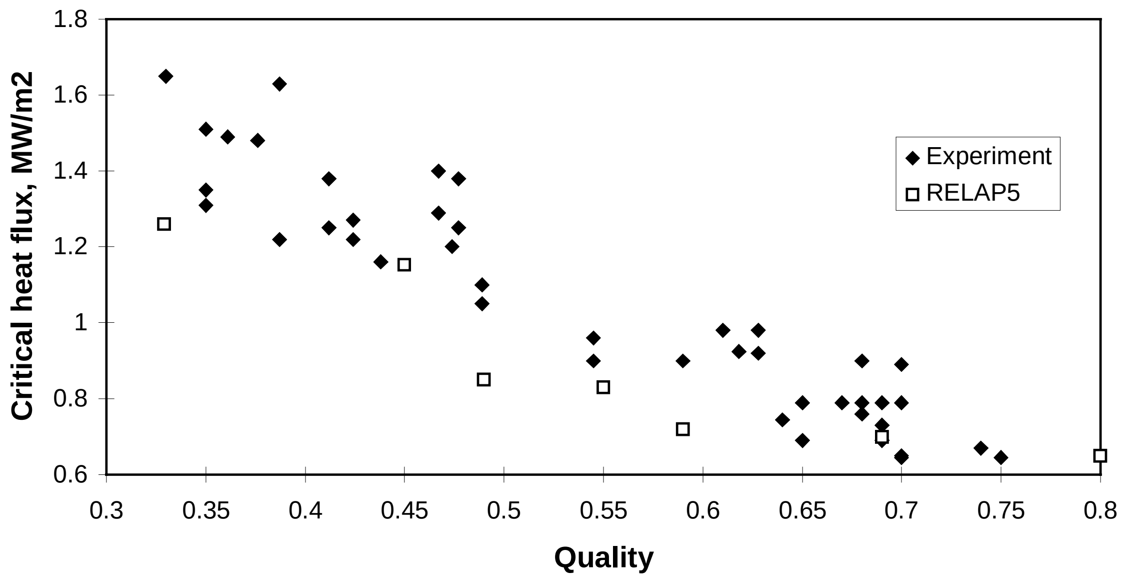


Figure 28. Single phenomena verification by employing Ignalina NPP model. The distribution of the predicted CHF of RELAP5 and experimental one

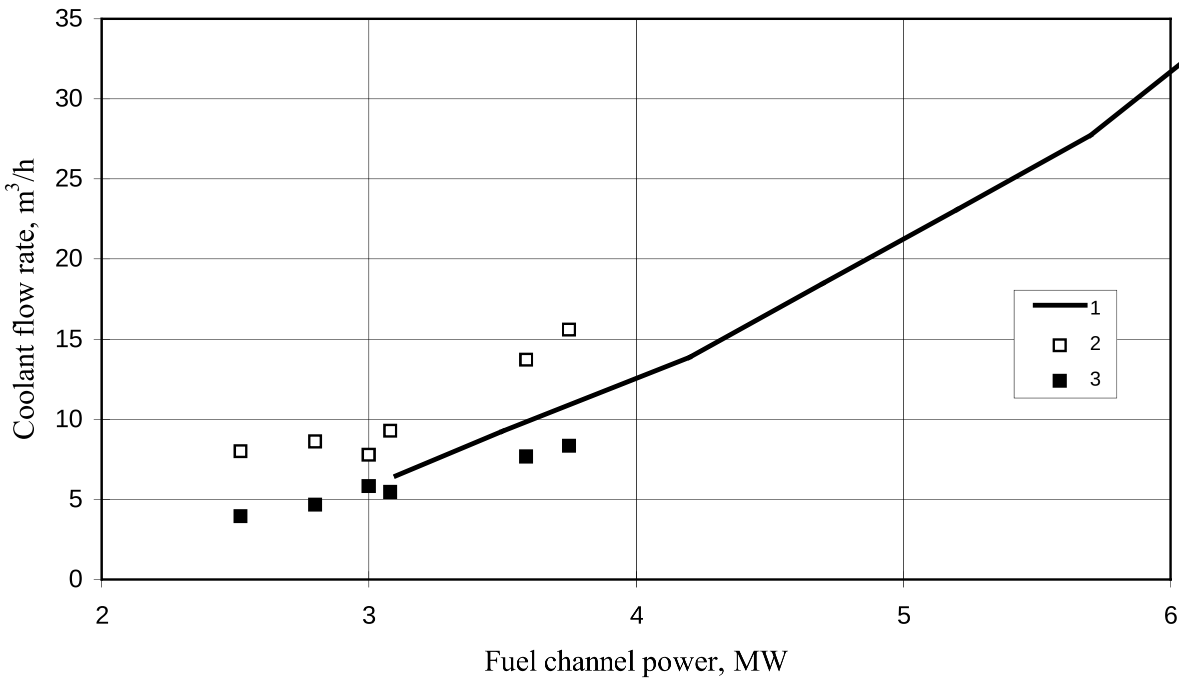


Figure 29. Single phenomena verification by employing Ignalina NPP model. The boundary of CHF in the fuel channels of RBMK-1500

1 - CHC Boundary according experimental data [26], 2 - CHF just starts to appear (calculation data using RELAP5 model), 3 - CHF appears in the entire length of the fuel channel (calculation data using RELAP5 model)

3 SINGLE PHENOMENA VERIFICATION BY EMPLOYING ELEKTROGORSK E-108 TEST FACILITY RELAP5 MODEL

The thermal hydraulic behaviour in long parallel heated channels is if great interest to RBMK design nuclear power reactors. In particular, instabilities caused by high power-to-flow ratio are of particular concern. Oscillatory flow patterns and CHF may result under such conditions.

In this section an evaluation of the experimental data obtained at the Elektrogorsk 108 (E-108) and PNC SEL test facilities are presented. The E-108 test facility consists of six parallel heated channels connected by water and steam drums. These channels are scaled RBMK pressure tubes. The experiments at PNC SEL test facility were performed with two different capacity heated channels.

The Elektrogorsk 108 facility is a scaled model of Russian RBMK design reactor. It was constructed in the town of Elektrogorsk and operated by the Elektrogorsk Research and Engineering Centre of NPP Safety (EREC) to gain understanding of the thermal hydraulic behaviour of the full-scale RBMK reactor. The prime aspects of the facility are the six full-height parallel-heated channels, each of which simulate a single RBMK fuel channel. Test programs were conducted at the E-108 facility in 1982 [14], 1984 [15] and 1985 [16]. One of the features studied with the E-108 model was that of identifying the system conditions at which flow instabilities between the parallel channels developed. This was accomplished by sequentially reducing the forced flow rate of the coolant to the heated channels for a given inlet subcooling level. Tests were conducted for a range of system pressures (1 to 7 MPa) and heated channel power input (0.06 to 0.77 MW/m^2) and inlet subcooling. In addition, and of particular importance with respect to the RBMK pressure channels, the relative flow resistance of the piping sections was varied through the use of flow orifices.

In modelling the Elektrogorsk 108 facility, all known geometrical data was taken into account. The geometrical data was collected from [14], [16], [17] and [18] to match the experimental facility as closely as possible. Additionally, the E-108 test facility geometrical description can be found in [19]. The amount of the heat input to the system, and distribution along the channels, was taken from [14].

Of the tests conducted in the three years (1982, 1984 and 1985), the most comprehensive data have been found for the tests conducted in 1984 [14]. The tests, which have been chosen, for the simulation using RELAP5 code are specified in Table 1.

No. of test	ξ_{LWC}/ξ_{SWC}	Pressure, MPa	q'', MW/m^2	T_{header}, °C;
1	75/14	7	0.6	102.0
2	25/14	7	0.6	100.6
3	3.5/14	7	0.6	102.0

Table 1. E-108 tests specifications

Tests No. 1 and No. 2 correspond closely to the RBMK-1500 conditions with closed check-valve and to the RBMK-1500 conditions under nominal conditions, respectively. Test No. 3 corresponds closely to the RBMK-1000 conditions.

The above listed tests were chosen for simulations to represent all three combinations of the hydraulic resistance coefficient ξ_{LWC} and ξ_{SWC}. The pressure P = 7 MPa was chosen, since this is the nominal pressure in operation of the RBMK reactors operation. The remaining data - heat flux q" and coolant inlet temperature T_{header} - were chosen from the limited data basis.

The calculations were performed with the RELAP5 MOD3.1 code with a representation of the facility configuration shown in Figure 30. The heated channels were represented by either 8, 15 or 29 control volumes. Convergence of the calculated results was obtained, since there was hardly any difference in the results obtained with 29 control volumes, from those obtained with 15 control volumes. Numerical oscillations were observed in the calculated results when the heated channels were divided into 8 control volumes. These oscillations disappeared when the heated channels were divided into 15 and 29 control volumes. Flow oscillations were calculated when the heat transfer conditions approached those of critical heat flux on the tubes. The timing of those calculated oscillations was close to those observed in the tests.

Figure 31 shows the flow rate in the channels for one of the tests. The variation in the flow rate with time resulted from the programmed reductions of the flow during the test. It was found that the coolant flow rate in all channels varied similarly in this test. The calculated void fractions in a tube were also found to be similar to those in the other tubes.

Figure 32 shows the time variation of the calculated temperature at a location in the assembly of six tubes. The Figure shows the calculated results for the choice of the 8, 15 and 29 control volumes in a heated channel. The calculated results are compared to the measured data. The results obtained with the 15 and 29 control volumes appear to slightly lag behind the measured temperatures, i.e. the calculated temperature escalation occurs slightly later than the measured one. It appears, however, that the magnitude of the calculated temperature rise is similar to the measured. The flow oscillations were calculated when the temperature escalations occurred as shown in Figure 32. Thus, in this case the calculated flow oscillations also lag those observed in the tests.

The RELAP5 calculations are able to predict the flow instabilities associated with the CHF conditions in the tube. The code predictions do not include the two-phase density wave instabilities, which may have been observed in the tests.

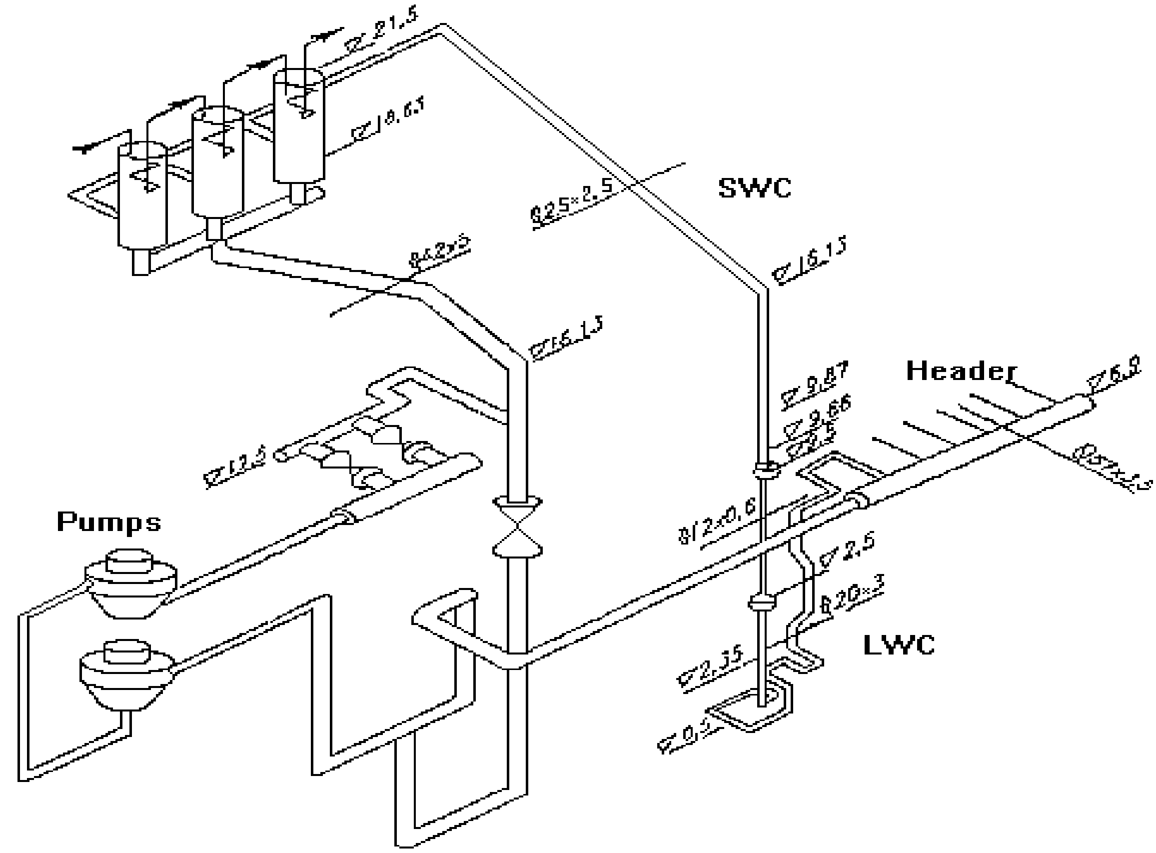


Figure 30 Single phenomena verification by employing Elektrogorsk E-108 test facility RELAP5 model. Elektrogorsk 108 facility configuration

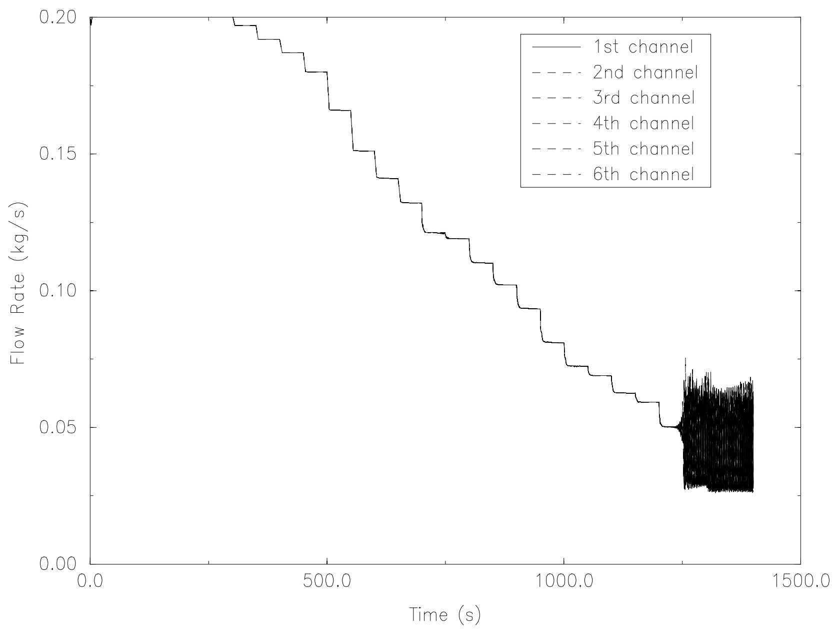


Figure 31. Single phenomena verification by employing Elektrogorsk E-108 test facility RELAP5 model. Flow rate in the channels

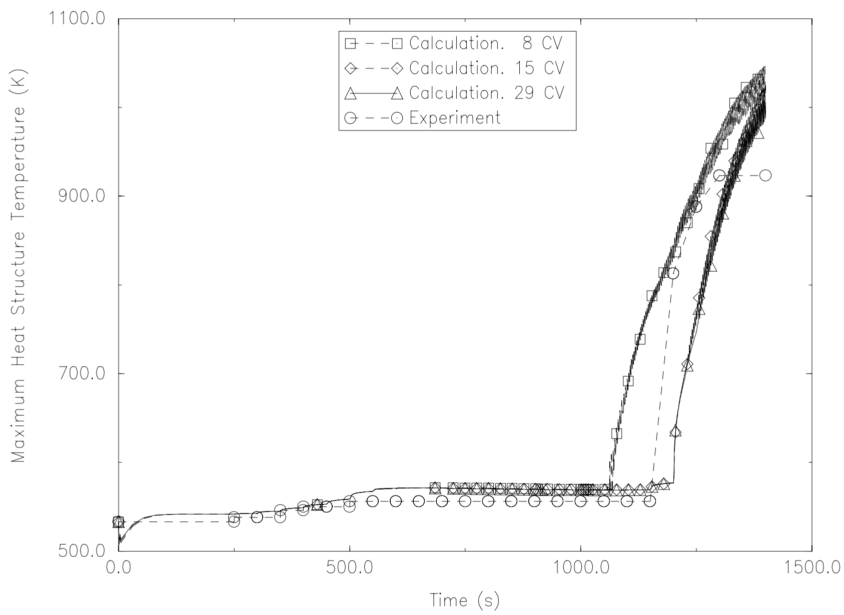


Figure 32. Single phenomena verification by employing Elektrogorsk E-108 test facility RELAP5 model. Comparison of calculated and the measured maximum heated wall temperature

4 SINGLE PHENOMENA VERIFICATION BY EMPLOYING PNC SEL TEST FACILITY RELAP5 MODEL

Several experiments involving flow instabilities, flow fluctuations, channel dryout and channel degradation have been conducted at the Japanese PNC facility in O-arai Engineering Center. The purpose of these experiments was to provide fundamental data with which to validate thermal-hydraulic codes such as RELAP5, CATHARE, ATHLET and RETRAN.

The experiments were conducted in the Advanced Thermal Reactor (ATR) Safety Experimental Loop (SEL) and Heat Transfer Loop (HTL) which are mock-ups of ATR. The ATR boiling heavy water pressure tube reactor design contains some of the features and components, which are similar to RBMK design. Various types of oscillations and thermal hydraulic behaviour under low flow measured at SEL and HTL were presented as a database for thermal hydraulic codes' validation of the RBMK hydraulic system.

The SEL consists of six parallel 3.7 m high, electrically heated channels, outlet pipes, steam and water drums, pump, valves and other instrumentation. The SEL facility configuration is shown in Figure 33. The steam-water mixture generated in the heated channels flows through the outlet pipes and passes into the steam drum where it is separated into steam and water by separators. Each channel contains a full-scale rod cluster. Each cluster contains 36 heater pins enclosed by a pressure tube. Five of the six heated channels employ low power heaters capable of generating 200 kW, while one of the heated channels is capable of producing 6 MW of power. The outlet pipes, which connect the pressure tube outlets to the steam separators (drums), are slanted at an angle of 2º.

The total heat loss at the temperature of 280 ºC for the SEL with two heated channels was measured to be approximately 170 kW.

More information regarding the geometry of the SEL facility can be found in references [19], [20], [21], [22], [23] and [24]. The RELAP5 model of the facility is shown in Figure 34.

In Figure 35 - Figure 37, calculated results are shown.

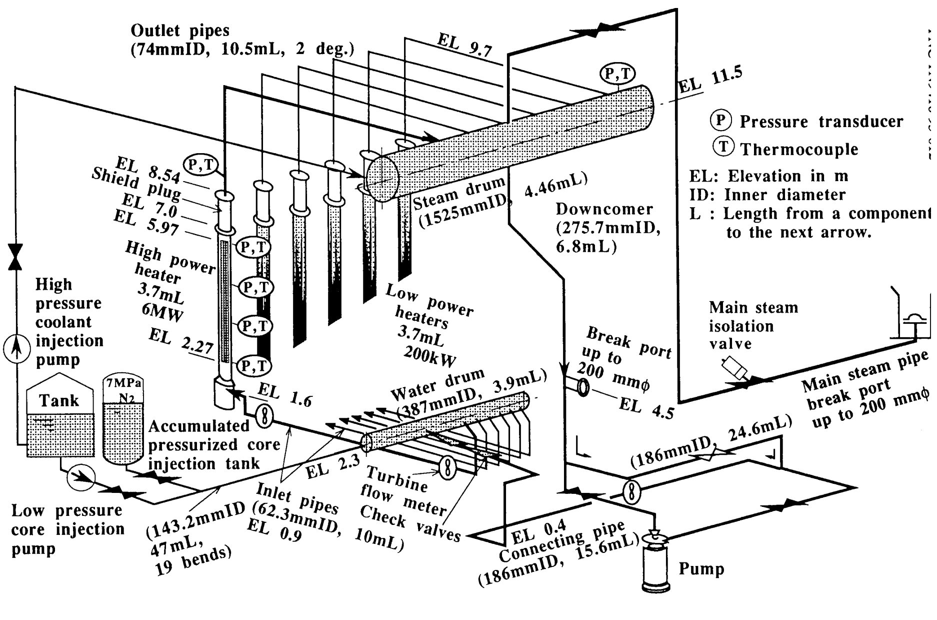
In Figure 35, measured flow rate is compared vs. calculated. The actual coolant flow rate through the high power channel is 3.74 kg/s at the initial stage. The middle ring heated pin "sees" a smaller flow area than average for the high power channel. To account for that in the calculation, it is necessary to normalise the coolant flow rate by the area "seen" by a middle ring pin. In Figure 35, normalised measured coolant flow rate is shown.

In simulation, calculational time was prolonged by 100 seconds, up to 500 seconds. As shown in Figure 35, the oscillations in simulation occur with somewhat of a lag. Two oscillation peaks are within the measured data oscillations. Thus the main oscillation behaviour is observed later. The amplitude of the calculated oscillations is considerably smaller, and constant, compared to that for measured oscillations, which also increases with time.

It is necessary to point out that the RELAP5 code is a 1-D thermal hydraulic code. The code does not represent cross flows between the rods in the 36 rod bundle within the tube. The local pressure drop fluctuations, as well as coolant cross flow, may contribute up to 20% of the difference from the calculated value. In order to obtain the correct coolant flow rate value for the rods in the middle ring, simulated flow rate was varied.

In Figure 36, the total, heated channel upper and lower halves pressure drops are shown. Measured and simulated pressure drops are represented. It is seen that the simulated total pressure is lower than the measured total pressure. However, the simulated heated channel lower and upper halves pressure drops are close to the actual measurements.

Figure 37 represents the heated rod surface temperature. Curves with peaks represent the actual measured surface temperatures. The three other curves represent the calculated heat structure surface temperatures.



Schematic of Safety Experiment Loop (SEL)

Figure 33. Single phenomena verification by employing PNC SEL test facility RELAP5 model. Safety Experimental Loop configuration

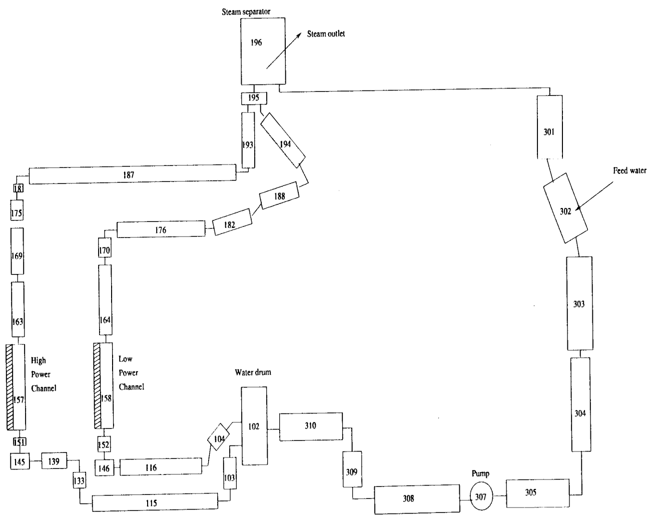


Figure 34. Single phenomena verification by employing PNC SEL test facility RELAP5 model. RELAP5 model of the PNC SEL facility

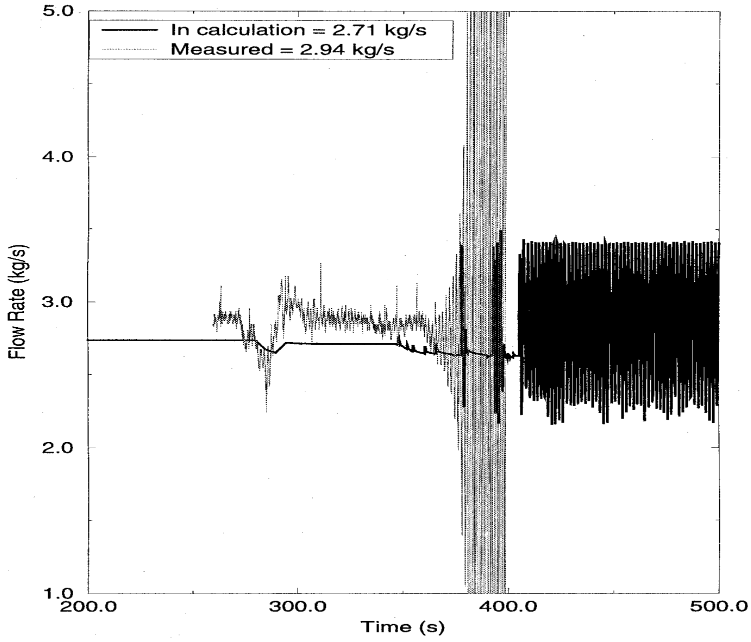


Figure 35. Single phenomena verification by employing PNC SEL test facility RELAP5 model. Flow rate in high power channel

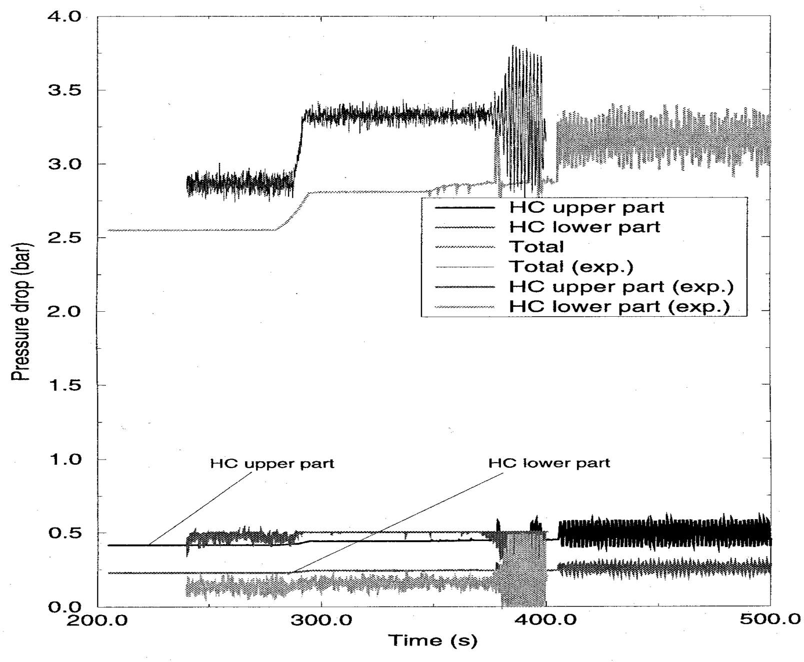


Figure 36. Single phenomena verification by employing PNC SEL test facility RELAP5 model. Comparison of the calculated and the measured pressure drops at flow rate 2.71 kg/s in high power channel

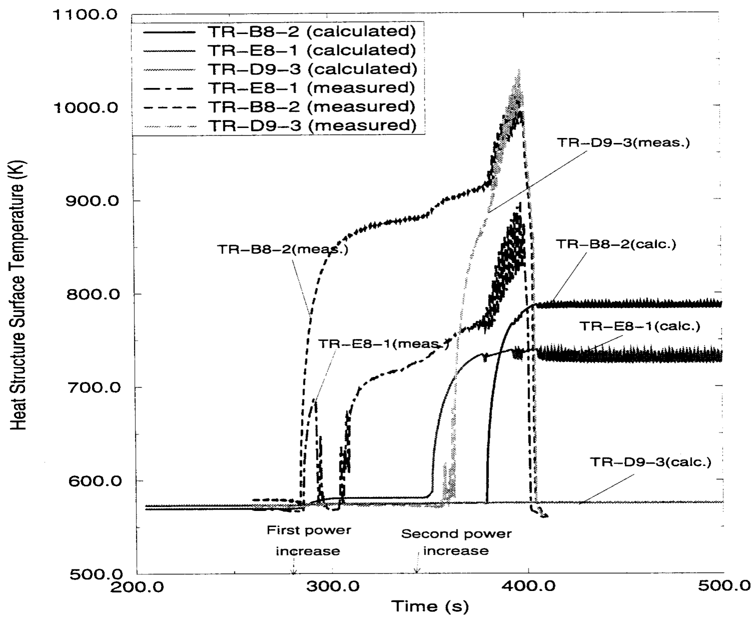


Figure 37. Single phenomena verification by employing PNC SEL test facility RELAP5 model. Comparison of the calculated and the measured maximum heated rod surface temperature at flow rate 2.71 kg/s in high power channel

5 CONCLUSIONS

A successful best estimate RELAP5 model of the Ignalina NPP has been developed. This model includes the reactor MCC and CPS required for this kind transient analysis.

Benchmark analysis of

- one and all operating MCPs trip events,
- three Main Safety Valves LOCA event,
- inadvertent actuation of ECCS

has been performed. The calculations performed with RELAP5 models on the Ignalina NPP specific base compare favourably with the plant data.

Single phenomena analysis of

- critical discharge flow rate
- CHF and CHF boundary
- flow instability

has been performed. Calculations performed by employing RELAP5 code quite well agree with Elektrogorsk E-108 and PNC SEL test facilities experimental data.

Comparative analysis of actual operational events and single phenomena enables RELAP5 code capability evaluation and establishing of actual model, which is used for Ignalina NPP safety assessment.

6 ABBREVIATIONS

ACS	Accident Confinement System
AZ-1	Emergency Protection
CHF	Critical Heat Flux
ECCS	Emergency Core Cooling System
EREC	Elektrogorsk Research and Engineering Centre of NPP Safety
GDH	Group Distribution Header
LOCA	Loss of Coolant Accidents
MCC	Main Circulation Circuit
MCP	Main Circulation Pump
MSRV	Main Safety Relief Valves
NPP	Nuclear Power Plant
RBMK	Russian Acronym for “Channelled Large Power Reactor”
RDIPE	Research and Development Institute for Power Engineering
SEL	Safety Experimental Loop

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