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International Agreement Report

RELAP5/MOD3.3 Assessment by Comparison with PKL III G3.1 Experiment (small break in the main steam line)

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ABSTRACT

The behavior of the PKL facility during a small main-steam-line break was investigated by using the RELAP model of the main facility components, including the reactor pressure vessel, steam generators, pressurizer, primary piping, and main steam lines. The calculations were performed using the computer code RELAP5/MOD3.3.

The main objectives of the study are to compare the simulation results with experimental data and to assess the accuracy of the calculated data. The major parameters and phenomena for simulating a main steam line break, e.g. primary and secondary pressures and temperatures, break flow rate and water level in SG and PZR, were analyzed. The results of the calculations were found to be in good agreement with experimental results.

All the analytical activities were performed as part of the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) projects of Test G3.1.

CONTENTS

Page

FIGURES

Page

TABLES

Page

EXECUTIVE SUMMARY

The main purpose of the report is to assess the capability of RELAP5/MOD3.3 computer code to predict the response of PKL III facility under simulated accidental conditions (PKL III Test G3.1: small break in the main steam line).For this purpose, a RELAP5/MOD3.3 analysis of the experiment is performed and the results of the calculation are compared with the measured data.

The PKL III test facility (AREVA GmbH, Erlangen, Germany) was built to investigate the behavior of German pressurized-water reactors under accident and transient conditions.

A RELAP5 input deck that had been used for earlier PKL analyses was used as the basis for the RELAP5/MOD3.3 model of the PKL III facility.

All calculations were performed in two steps: a steady-state calculation (in order to stabilize the computational process), and a transient calculation.

The calculated values of the most important parameters and phenomena for simulating a main steam line break (e.g., primary and secondary pressure, fluid temperature, break flow rate and the mass flow rate in the primary circuit) were found to be in good agreement with the experimental results.

A deviation between the calculation and the experimental results was found for the water level in the affected steam generator and for the collapsed level in the pressurizer.

ABBREVIATIONS

ECCS	Emergency Core Cooling System
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
MSRV	Main Steam Relief Valve
NPP	Nuclear Power Plant
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
PWR	Pressurized-Water Reactor
SG	Steam Generator

1. INTRODUCTION

The behavior of the PKL facility during a small main-steam-line break was investigated using the RELAP model of the main facility components, including the reactor pressure vessel (RPV), steam generators (SG), pressurizer (PRZ), primary piping, and main steam lines. The calculations were performed using the computer code RELAP5/MOD3.3.

The main objectives of the calculation are to compare the results with experimental data obtained during the PKL G3.1 test and to assess of the accuracy of the calculated data. The most important parameters and phenomena for simulating a main steam line break, e.g., primary and secondary pressures and temperatures, break flow rate and water level in SG and PZR, were analyzed.

2. FACILITY AND TEST DESCRIPTION

2.1. PKL III test facility

The PKL III test facility (AREVA NP GmbH, Erlangen) was built to investigate the behavior of German pressurized-water reactors (PWRs) under accident and transient conditions [2]. The layout of the PKL III facility (Figure 1) is based on the “Vorkonvoi“ type (4-Loop, 1300 MWe) of a KWU pressurized-water reactor, with the Philippsburg 2 nuclear power plant (NPP) serving as the reference plant. The entire reactor coolant system (RCS) primary side and the most significant components of the secondary side (excluding turbines and condenser), including the appropriate control system, are represented.

Following the scaling concept, all geodetic heights are represented in a 1:1 ratio. The entire volume of the primary circuit and the proportions of the individual volumes are scaled by a factor of 1:145. The maximum primary pressure is limited to 4.5 MPa; the maximum core power of 2.5 MW is equivalent to 10 % of the nominal rate.

The core is simulated with a bundle of 314 electrically heated rods. The core geometry is, like the SG geometry, constructed as an “actual section;” that is, the individual heated rods and U-tubes have the actual geometry, but the number of heated rods in the core and the number of U-tubes in the SG are reduced by the scaling factor 1:145 (volume and power scaling), as compared to the reference plant.

The representation of the primary side by four identical and symmetric loops arranged around the reactor pressure vessel allows the realistic investigation of accidents, even with asymmetric boundary conditions in the individual loops. Through the representation of all significant interfaces and auxiliary system functions on the primary and secondary side, the overall system behavior as well as the interaction between the individual systems can be investigated under a wide range of accident conditions and the effectiveness of the automatic or manually-performed counter actions can be ascertained [1].

With approximately 1500 measurement points, the PKL facility is comprehensively instrumented. This allows detailed analysis and interpretation of the phenomena that develop in the course of the tests.

2.2. Boundary conditions: status of the facility before test start

The status of the facility before the accident is summarized in Table 1 and graphically presented in Figure 2. The primary circuit is filled with water and the pressurizer pressure is maintained at 42 bar. The loop flow rate in the primary circuit is set to the 33.7 kg/s and the core power is 260 kW. The fluid temperature in the core inlet / exit is 243°C / 245°C respectively. All steam generators are isolated from the feedwater lines during the last stage of the preparation phase.

The core power of 260 kW includes compensation for appropriate 150kW heat losses of the primary and secondary side components. The remaining power describes the decay heat power and leads to a slow and continuous temperature increase in the primary circuit (~ 12°C/h) before starting the test.

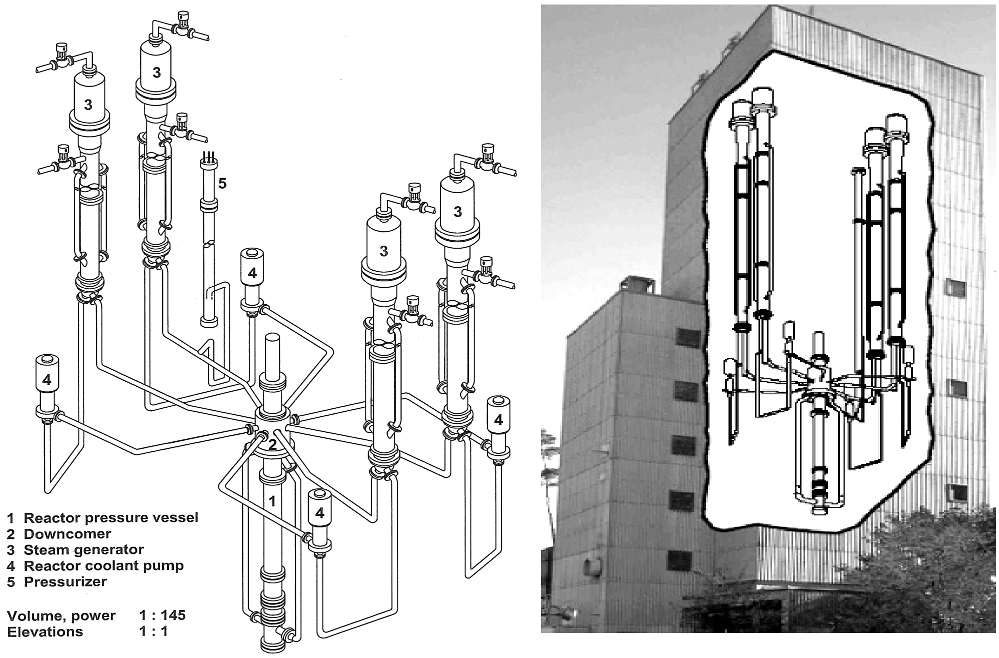

Figure 1: PKL facility (AREVA NP, Erlangen)

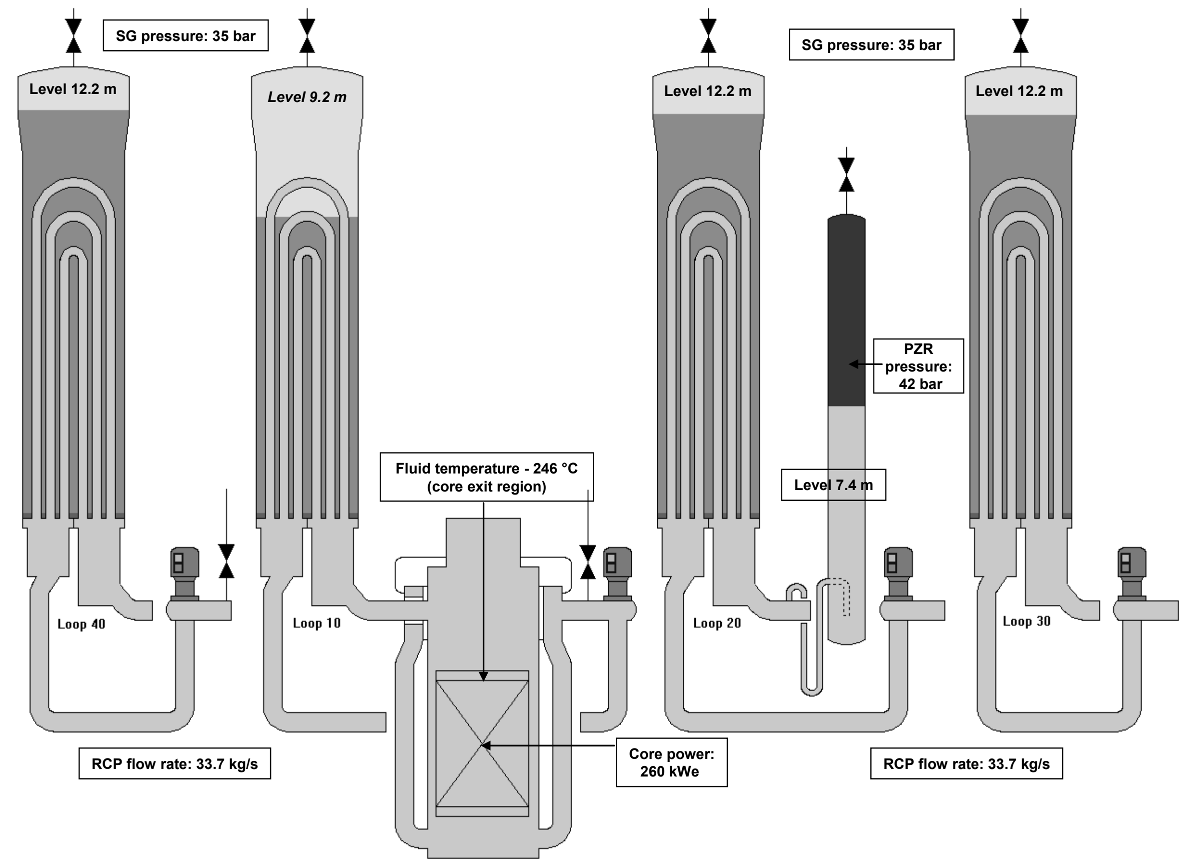

Figure 2: Status of PKL test facility before test start

Table 1: Initial conditions at test start

Primary circuit	**Condition / Value**
Heater rod bundle power (decay heat)	260 kW
RCPs in operation	Mass flow rate: 33.7 kg/s
PZR pressure	42 bar
Collapsed level in PZR	7.4 m
Fluid temperature (core exit)	245°C
Flow pattern	Symmetrical heat removal via 4 SGs with subcooled forced circulation in all 4 loops
Coolant inventory	Primary circuit completely filled with water
Secondary side	
Main steam pressure	SG 1-4: 35 bar
Water temperature (Riser)	SG 1-4: 240°C
Collapsed level (Riser)	SG1: 9.2 m*), SG 2 - 4: 12.2 m

*) Lower collapsed level in affected SG1 to avoid of entrained droplets

2.3. G3.1 Test: Accident description and sequence of events

The G3.1 test can be divided into 2 major phases:

Phase1: starts with opening of the break and ends when the secondary side of the affected SG 10 becomes empty

Phase 2: starts with the initiation of the emergency core cooling system (ECCS) injection into 2 cold legs and includes the pressure limitation (control) by the PZR safety valves.

The following discussion applies only to the first phase of the experiment.

The investigated case is a small break in the main steam line connected to SG 10. The break (break size is 0.1A) is initiated by opening the break valve. The feedwater lines in all SG have been isolated before starting the test. Immediately after the break opens, the coast down of all four reactor coolant pumps is initiated and the non-affected steam generators are automatically isolated by closing the isolation valves.

The break leads to the fast depressurization of the affected steam generator. The temperature difference between primary and secondary side increases and considerably improves the heat transfer in the SG. As a result, the pressure and the temperature in the primary circuit start to decrease. The water level in the affected SG decreases due to the intensive evaporation. Concurrently, the water level in the pressurizer also decreases due to the volume contraction.

After the coast down of the main coolant pumps, the heat removal from the core is provided using natural circulation. The natural circulation flow in loop 10 is greater due to the increased heat transfer from the primary to the secondary side in the affected SG.

The secondary water level in the affected SG further decreases and eventually the affected SG becomes empty. The fluid temperature at the RPV inlet and the PZR water level reach their minimum values and start increasing again.

The information about the major events, measurements and essential system response during the test is given in Table 2.

Table 2: G3.1 test information (sequence of major events)

Time (s)	Event	Remarks
0	Isolation of SG 20, 30, and 40 Start of the test: Opening of the break-valve in the main steam line of SG10 Coast down signal for all RCPs PZR heaters are switched off	Evaporation in SG 10 → decrease in secondary side pressure and water level Decrease of the RCP speed → decreased mass flow rate
210	Blockage of RCPs	Natural circulation in each loop
600		Minimum temperature (153 °C) at RPV inlet of loop 10
1000	End of phase 1	Complete evaporation of secondary inventory in the affected SG Further decrease of the natural circulation Increase of the primary and secondary side temperature and PZR water level

3. RELAP5/MOD3.3 CALCULATIONS

3.1. Numerical code and RELAP5/MOD3.3 input deck preparation

The RELAP5/MOD3.3 computer code is used for the calculations. This advanced computer code is designed for use in realistic studies of accident thermal-hydraulics in pressurized-water reactors. The basis for the RELAP5/MOD3.3 model of PKL III facility was a RELAP5 input deck which had already been used for earlier PKL analyses [3], [4] and was adapted for the main steam line break case [1].

All calculations were performed in two steps: a steady-state calculation in order to stabilize the computational process and the transient calculation after the opening of the break-valve. The details are presented in the following chapters.

3.2. RELAP5/MOD3.3 model of PKL III facility

The analysis uses a full four-loop model of the facility developed by AREVA NP GmbH [4].

The reactor pressure vessel model contains the vessel with its major components: the reactor core, the downcomer, the lower and the upper plenum and the RPV head (Figure 3).

The heat production in the core is realized by a heat structure connected to six pipe volumes. Additional heat structure models are used to simulate the behavior of the vessel wall and the main internals (see Figure 3 for details).

The following components, numbered from 20 to 256, represent the reactor pressure vessel.

B20 and B22 **	Lower plenum
P42	Reactor core (core simulator)
A40	Core bypass
B102 and B112	Core exit region
P130 and P220	Upper plenum and upper head
B242, B244, P246, and P256	Downcomer
B232 and B234	Inlet nozzles
B122 and B124	Outlet nozzles
B230 and B222	Upper head bypass

** — P - PIPE component, B - BRANCH component, A - ANNULUS component

The steam generator and primary piping model is given in Figure 4. The components which represent the steam generator and the primary piping are presented in Table 3.

Table 3: RELAP5/MOD3.3 input deck: primary piping and SG components

	Hot leg	Crossover leg	RCP	Cold leg
Loop 10	P300 and P310	P324, B326, and P328	PP332 and SV330	P338
Loop 20	P400 and P410	P424, B426, and P428	PP432 and SV430	P438
Loop 30	P500 and P510	P524, B526, and P528	PP532 and SV530	P638
Loop 40	P600 and P610	P624, B626, and P628	PP632 and SV630	P638
	SG Inlet	**U-tubes**		**SG Outlet**
SG10 (primary)	B314	P316, P318, and P320		B322
SG20 (primary)	B414	P416, P418, and P420		B422
SG30 (primary)	B514	P516, P518, and P520		B522
SG40 (primary)	B614	P616, P618, and P620		B622
	SG riser	**SG downcomer**		**Steam dome**
SG10 (secondary)	P350 and B372	B362, B364, B366, P368, and B370		P352 and P354
SG20 (secondary)	P450 and B472	B462, B464, B466, P468, and B470		P452 and P454
SG30 (secondary)	P550 and B572	B562, B564, B566, P568, and B570		P552 and P554
SG40 (secondary)	P650 and B672	B662, B664, B666, P668, and B670		P652 and P654

The main reactor coolant pumps (RCPs) are modeled by a special RELAP5/MOD3.3 PUMP component. The primary loops are symmetrical except for loop 20, where the pressurizer surge line is connected. The pressurizer and PZR surge line are represented by P800, B805, and P810 components. The SG main steam lines are modeled by B710, B720, B730, and B740.

Figure 5 presents the nodalization scheme for the whole facility. The PKL III model consists of 146 hydrodynamic components connected by 55 junctions. The facility structures are represented by 145 heat structures with 655 mesh points. The detailed information about the adopted code resources and the nodalization features are given in Table 4.

Table 4: Input deck resources and nodalization features

ADOPTED CODE RESOURCES	
Total number of hydraulic components (whole facility) PIPE / BRANCH / ANNULUS / SNGLVOL / TMDPVOL	146 58 / 65 / 1 / 4 / 18
Total number of hydraulic components in RPV	18
Total number of hydraulic components in SG (per SG; primary and secondary side)	56
Total number of heat structures	145
Total number of mesh points in the heat structures	655
Total number of SNGLJUN / TMDPJUN / VALVE-connections	18 / 21 / 16
NODALIZATION FEATURES	
Number of modeled loops	4
Number of components modeling the DC in RPV	2 Pipes, 6 Branches
Number of U-tubes (per SG)	3
Number of axial meshes of SG U-tubes (per SG)	60
Core model (1-D or 3-D component)	1-D
Number of hydraulic channels in the core region	1
Cross-flow junctions between parallel channels in the core (YES / NO)	NO
CODE OPTIONS	
Break flow model	Henry-Fauske critical flow model **
SEPARATOR or DRYER models in SG dome (YES / NO)	NO
Specific models activated in PZR (YES / NO)	NO

** - for more details see § 3.3

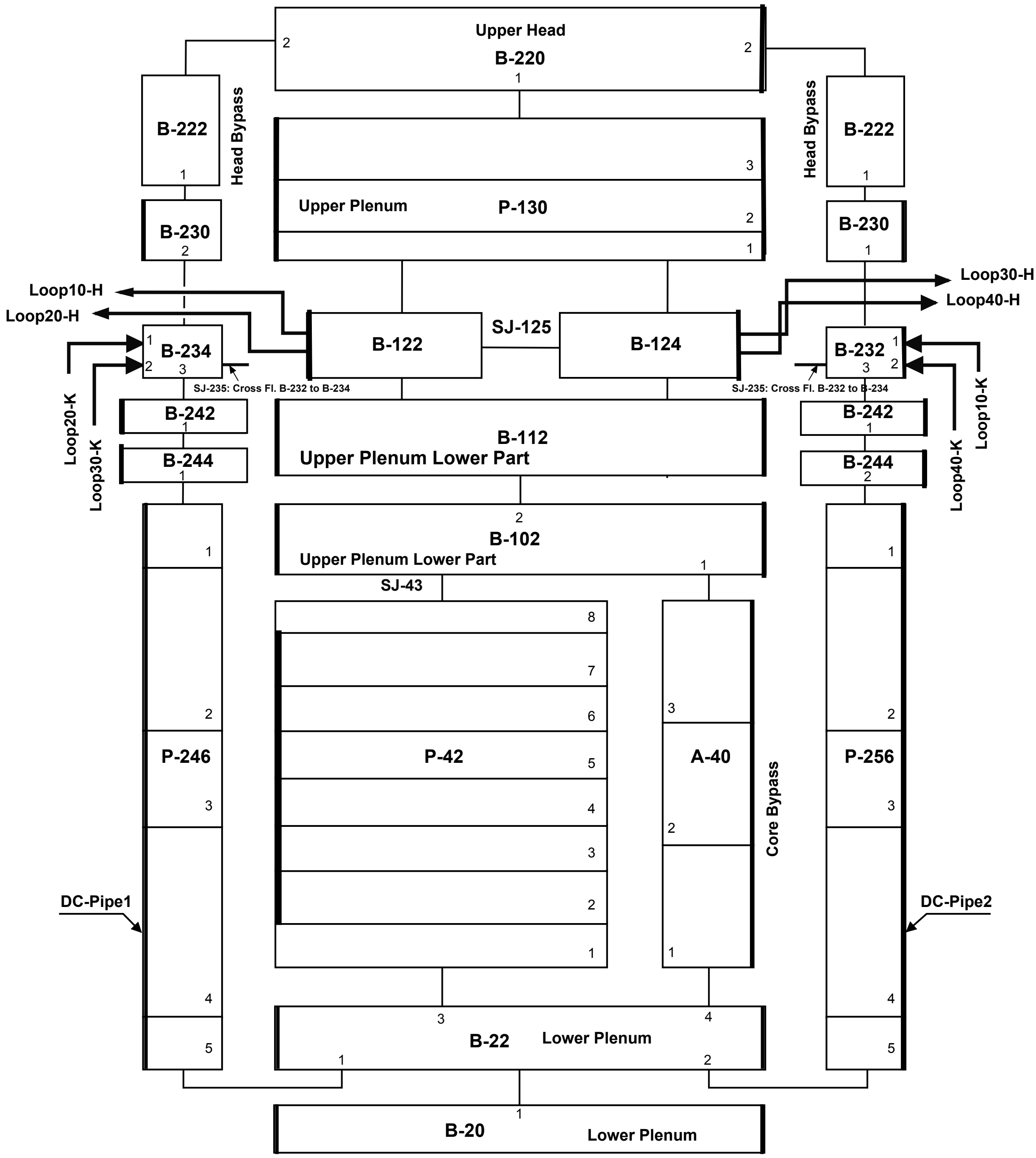

Figure 3: Reactor pressure vessel model

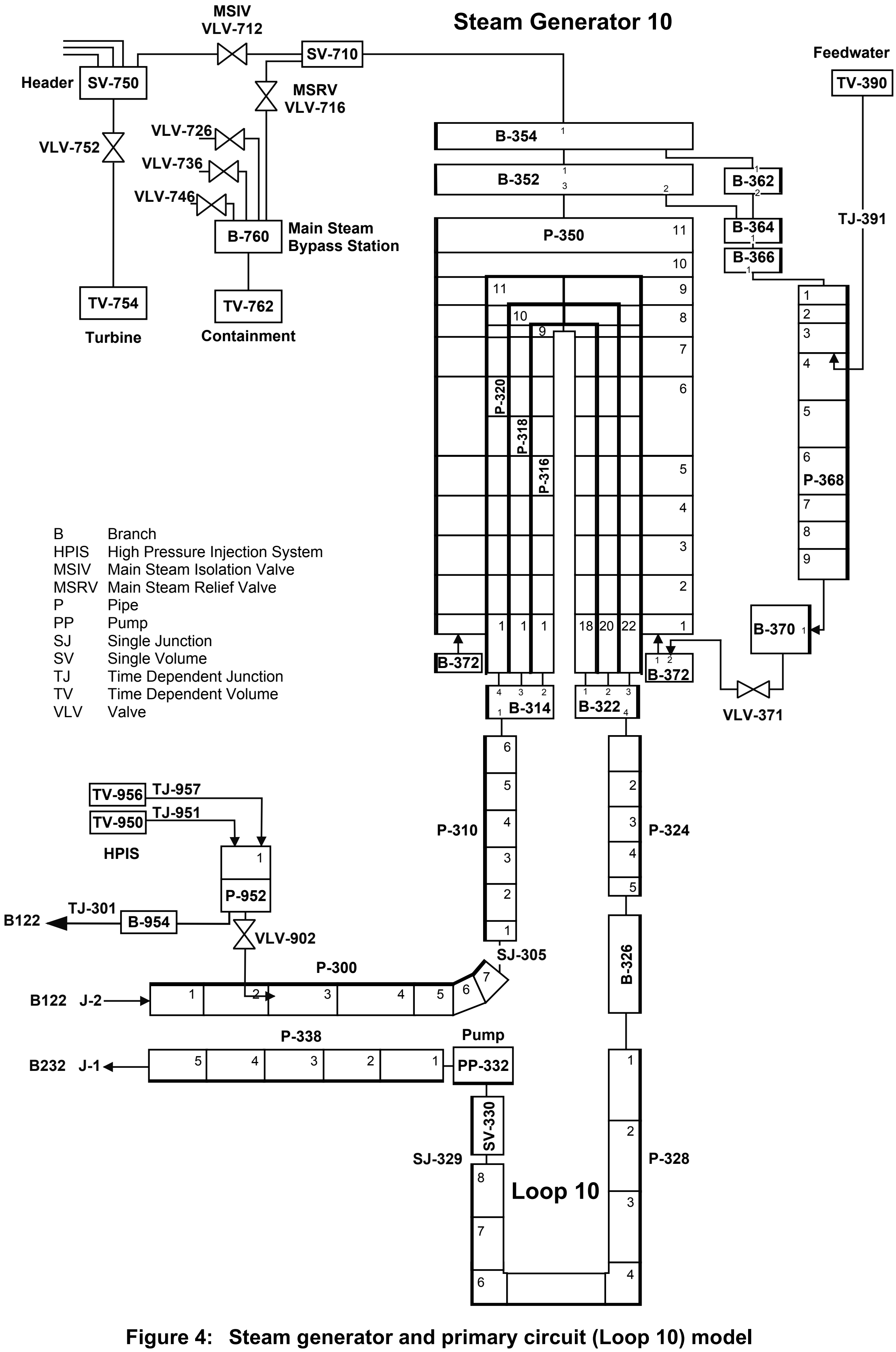

Figure 4: Steam generator and primary circuit (Loop 10) model

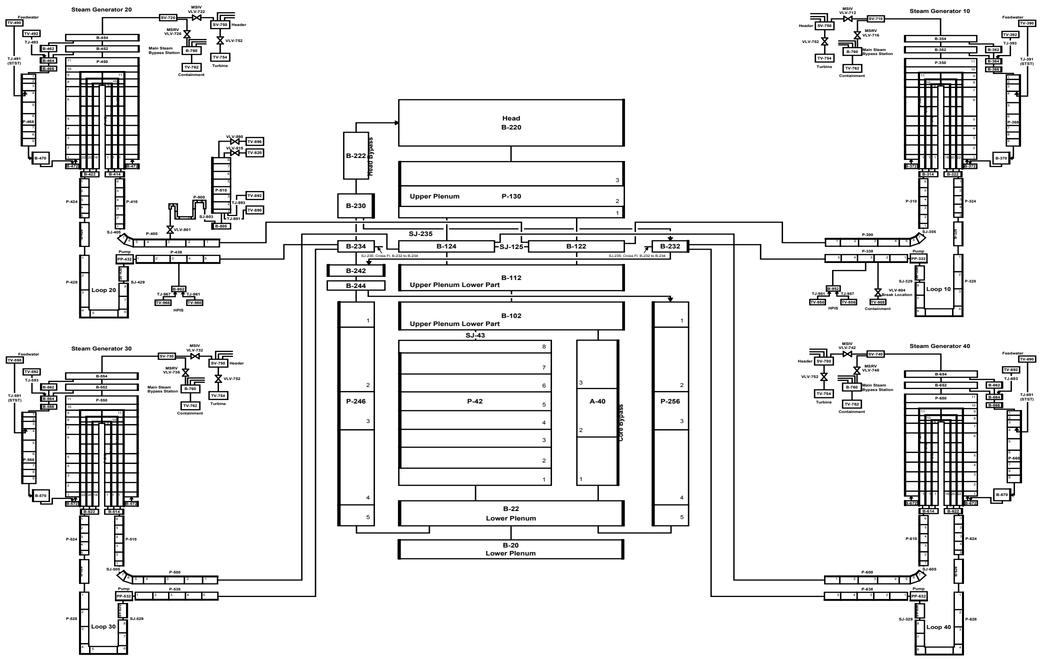

Figure 5: Nodalization of PKL III facility

3.3. Pre-calculation of the break flow rate and the break model

An additional study was performed to validate the break flow rate and the behavior of the steam generator during the depressurization phase after the opening of the break valve without an influence of the heat transfer from the primary to the secondary side. The calculation results were compared with the experimental data obtained during the depressurization test performed at the PKL III facility. The main purpose of this analysis is an assessment of the break model, which can then be used in the calculation of the small MSLB accident.

The results of the RELAP5/MOD 3.3 calculation and the experimental data obtained during the PKL depressurization test are presented in Figure 6. By varying the discharge coefficient of the break valve and the valve-opening time, the computationally determined pressure in the SG and mass flow at the leak were adjusted to meet the experimental results as closely as possible.

The default Henry-Fauske model for two-phase critical flow is used in the calculation. The adjusted discharged coefficient is set to 0.8.

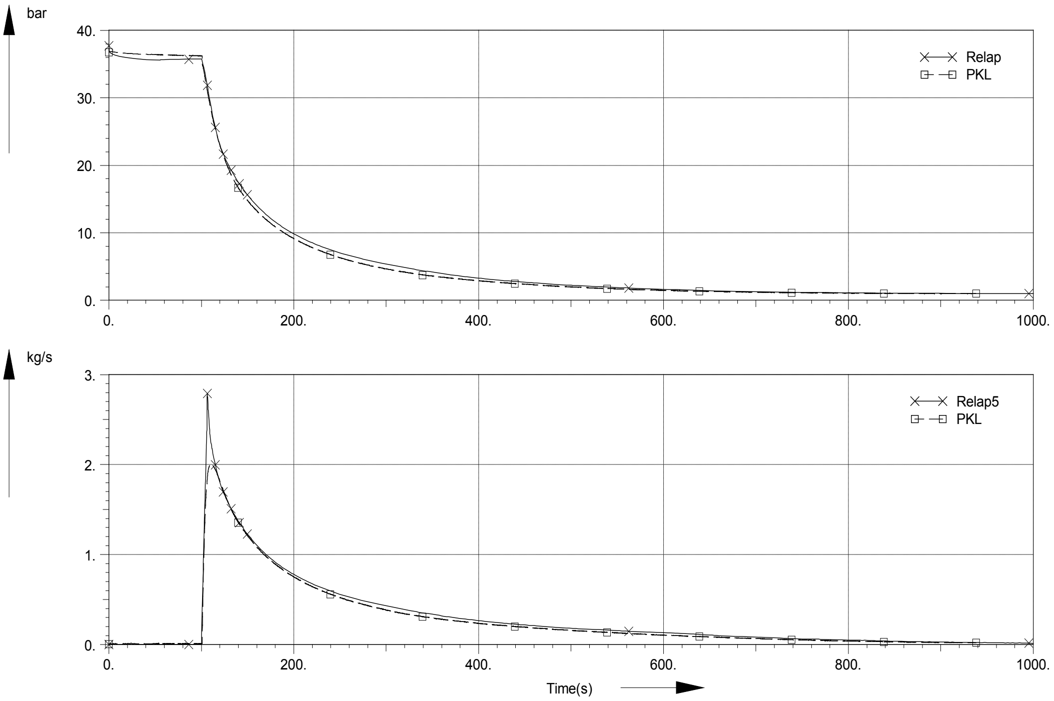

Figure 6: Secondary side pressure and mass flow rate at the leak compared with experimental data

3.4. Boundary conditions and steady-state initialization

The initial conditions of the facility does not represent a “pure” steady-state situation since the heat losses in PKL III facility are smaller than the power supplied to the core simulator (§ 2.2). As a result, the temperature in the primary circuit constantly increases. For this reason, the initial conditions for the calculation were also adjusted to produce a “quasi-steady-state”, which would approximate the behavior of the PKL III facility. The computation of the main steam line break was started when primary-side temperatures of 243.6 °C and pressure of 42.3 bar were reached.

The major parameters representing the initial conditions used in RELAP5 calculations are given in Table 5.

Table 5: Agreement of Boundary conditions (PKL / RELAP5/mod3.3)

	PKL	RELAP5/MOD3.3 Value
Primary circuit		
Heater rod bundle power (decay heat)	260 kW	260 kW
RCPs in operation	Mass flow rate: 33.7 kg/s	Mass flow rate: 34.3 kg/s
PZR pressure	42 bar	42.3 bar
Collapsed level in PZR	7.4 m	7.4 m
Fluid temperature (core exit)	245°C	243.6°C
Secondary side		
Main steam pressure	SG 1-4: 35 bar	SG 1-4: 35 bar
Temperature (Riser)	SG 1-4: 240°C	SG 1-4: 242.5°C
Collapsed level (Riser)	SG1: 9.2 m**, SG 2 - 4: 12.2 m	SG 1: 9.2 m, SG 2 - 4: 12.2 m

** - lower collapsed level in affected SG1 to avoid of entrained droplets

3.5. RELAP5/MOD3.3 results: diagrams of the most representative parameters

The main steam line break computation began with opening of the break valve in the main steam line of the SG 10, simultaneously isolating the non-affected SGs and switching off the main coolant pumps. On the secondary side, the feedwater pumps were shut off. Core heaters continuously supplied a power of 260 kW inclusive heat losses. The pressurizer remained connected during the test.

The diagrams of the most representative parameters of the RELAP5/MOD3.3 calculation together with experimental data are given in Figures 7-11.

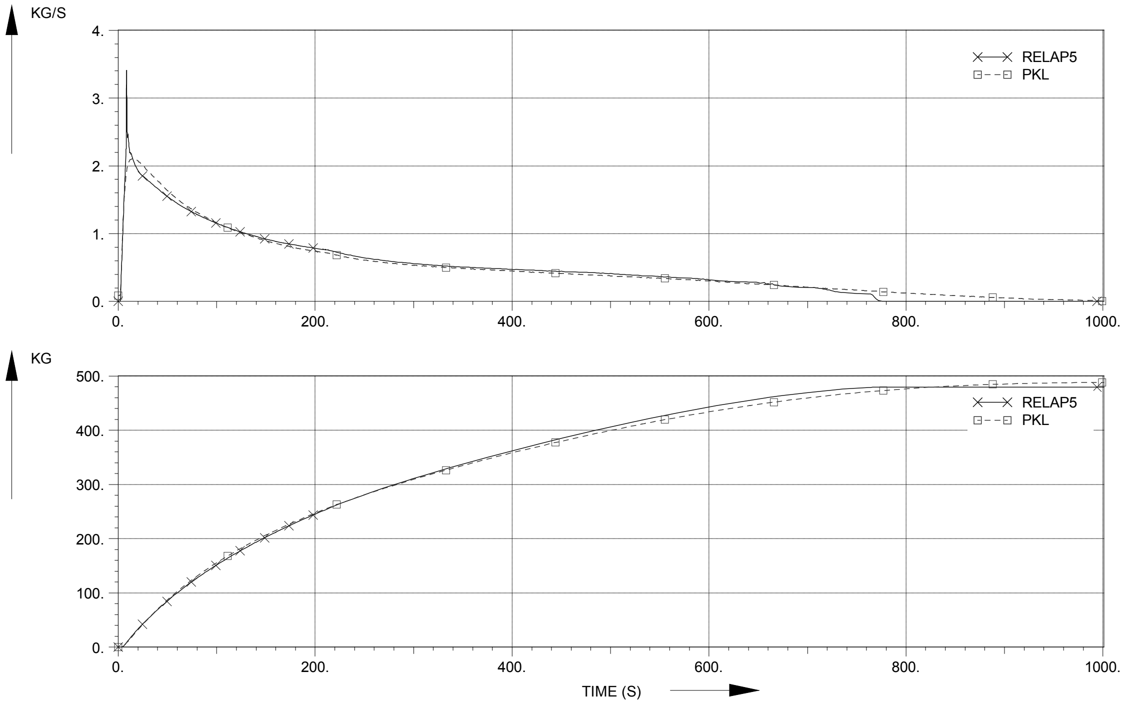

Figure 7: Break mass FLOW RATE/ Integral mass released via break

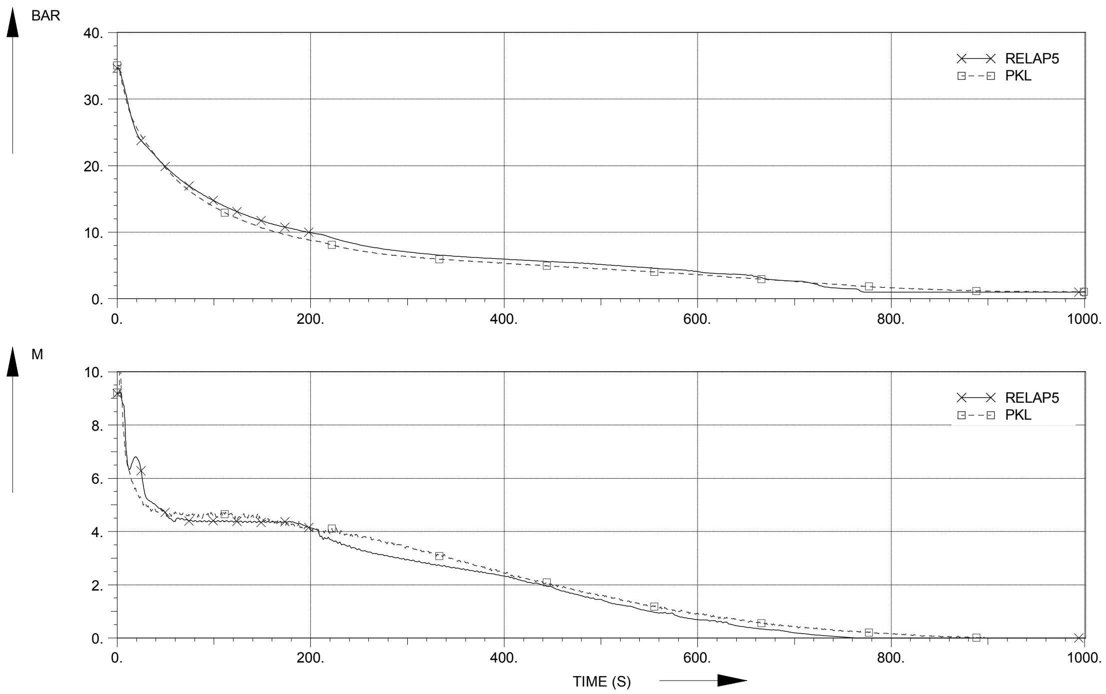

Figure 8: Pressure in SG 10 / SG 10 water level

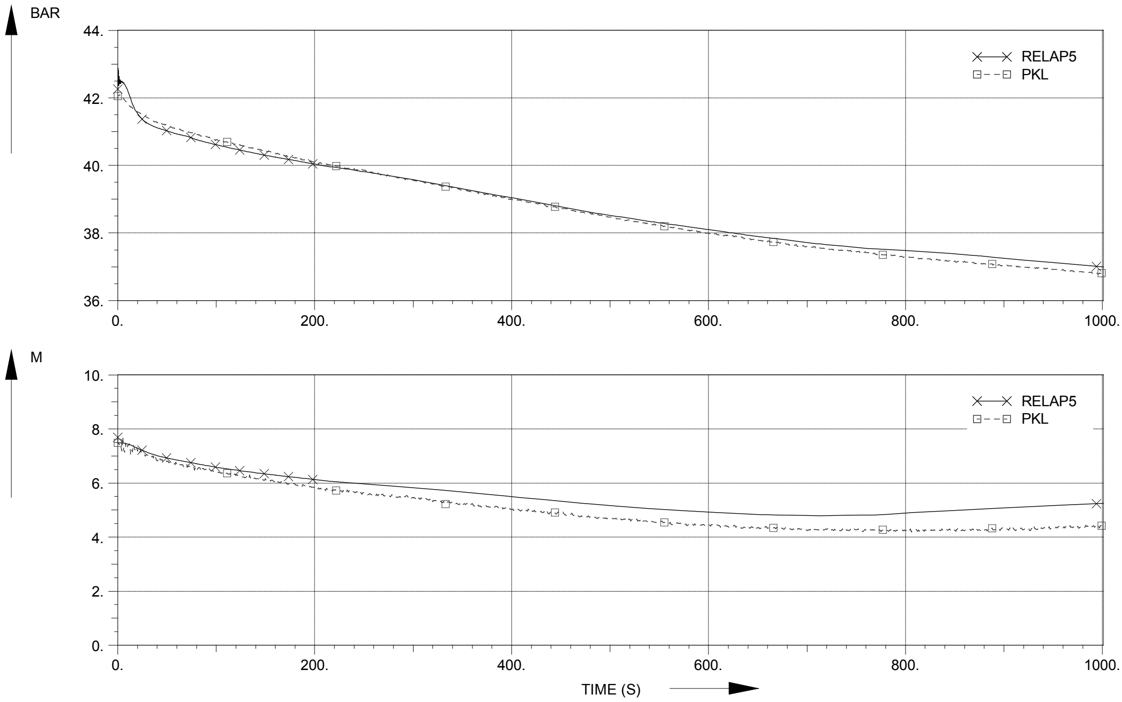

Figure 9: PZR pressure (inlet region) / PZR water level

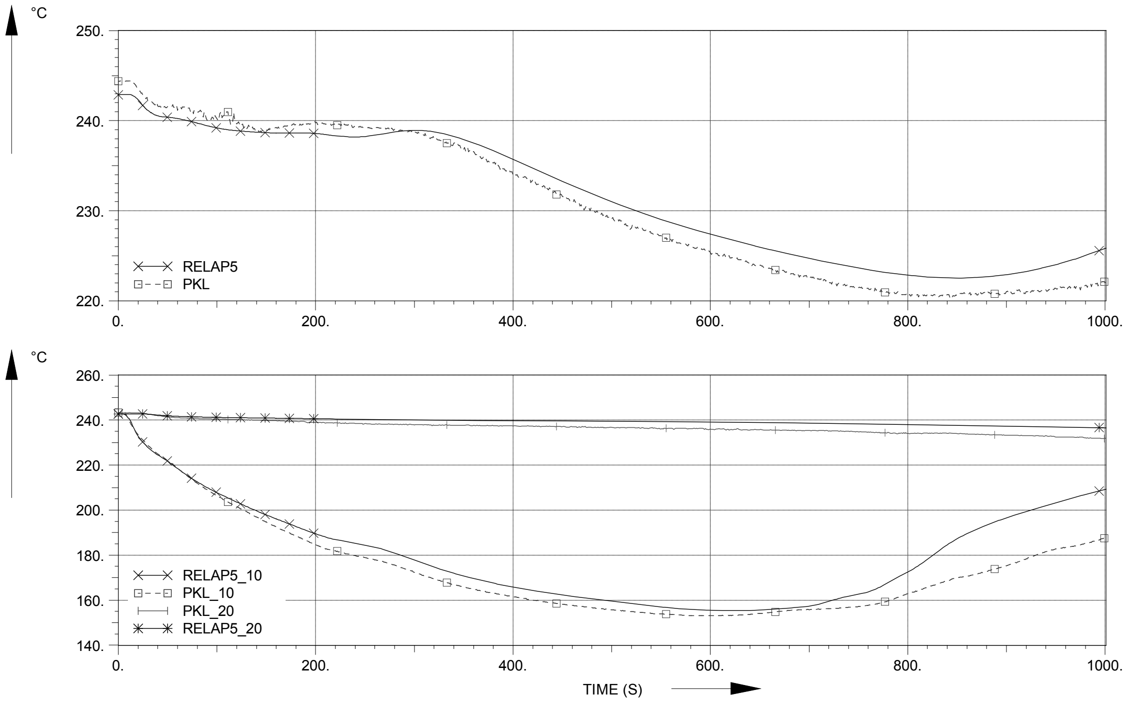

Figure 10: RPV outlet temperature (Loop10) / RPV inlet temperature (Loop 10 and Loop 20)

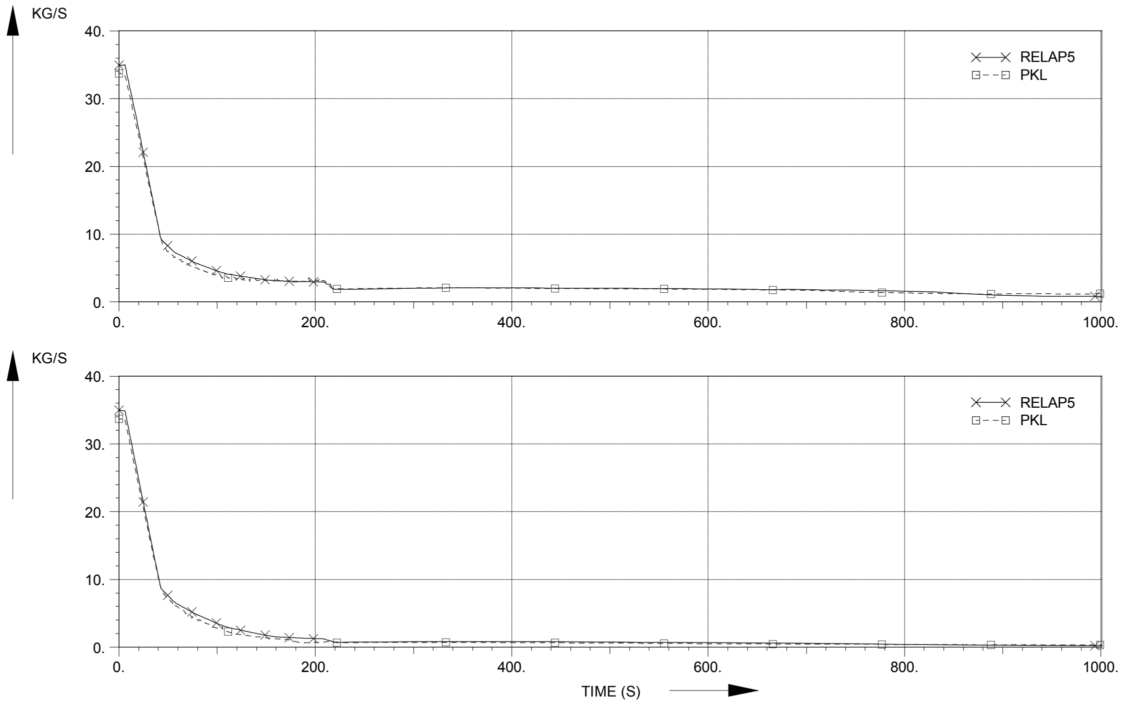

Figure 11: Primary circuit flow rate (Loop 10) / Primary circuit flow rate (Loop 20)

4. DISCUSSION OF RESULTS

4.1. Break flow rate

The predicted break flow rate is found to be in good agreement with the experimental results during the depressurization of the SG10 only and with all facility components, as presented in Figure 6 and Figure 7. A small discrepancy between the experiment and the calculation is observed between 180 and 230 seconds, where the predicted break flow rate is rather higher than that from the experiment (see Figure 7). The difference between the curves becomes significant when the SG 10 in the RELAP5 calculation empties (after approx. 760 seconds) and the calculated break flow rate reduces to zero.

4.2. Secondary-side pressure and SG water level

The secondary side pressure and the water level in the affected SG are presented in Figure 8. The opening of the break valve causes a fast pressure drop and decrease of the collapsed level in the affected SG 10.

The calculated secondary side pressure is close to that measured experimentally during first 80 seconds of the transient. RELAP5 slightly over-predicts the secondary side pressure in the time between 80 and 650 seconds. After approximately 720 seconds, the secondary side pressure from the RELAP5 calculation drops below the experimental curve after the SG 10 becomes empty.

The water level in the affected SG decreases rapidly after break opening due to the intensive evaporation. The results are found to be in a good agreement with the measured data during 200 seconds after the beginning of the transient. After that, the water level in the RELAP5 calculation becomes lower than that in the experiment. This can be explained by the fact that the RELAP5 slightly over-predicts the evaporation in the affected SG. Correspondingly, the break flow rate in the calculation slightly exceeds the measured data after approx. 200 seconds, as presented in Figure 7.

4.3. RPV outlet fluid temperature

The RPV inlet / outlet fluid temperatures and their comparison with experimental data are presented in Figure 10.

The calculated RPV outlet temperature is in good agreement with the experimental results during approx. 250 seconds after the beginning of the transient. During this time, the initial difference between the measured and the predicted curves remains virtually the same. After this, the cooldown on the primary side in the RELAP5 calculation occurs slowly. This can be explained by the fact that the water level in the SG in the RELAP5 calculation is smaller, and therefore, the decay heat removal from the primary circuit is less effective.

After approximately 850 seconds, the RPV outlet temperature starts increasing again as the heat removal by the affected SG terminates. This is consistent with the experimental observations.

4.4. <u>Pressurizer pressure and PZR water level</u>

The fast depressurization of the affected SG causes evaporation and associated turbulent mixing on the secondary side. The latter improves the heat transfer from the primary to the secondary side. This leads to cooling of the water on the primary circuit (cooldown transient). The water density on the primary side increases with decreasing temperature, and the primary pressure decreases (Figure 9).

The RELAP5 PZR pressure is in a good agreement with the experimental results. A small difference between two curves is observed after approx. 760 seconds.

The predicted water level in the PZR matches the experimental data most closely in the initial phase of the transient (<100 sec). Later on, the RELAP5 over-predicts the measured water level in the pressurizer. The discrepancy between the results becomes larger after approx. 760 seconds as SG 10 empties. This is associated with discrepancy of calculated primary temperatures.

5. CONCLUSION

The main purpose of the report is to assess the capability of RELAP5/MOD3.3 computer code to predict the response of PKL III facility under simulated accidental conditions (PKL III test G3.1: small break in the main steam line). For this purpose, a RELAP5/MOD3.3 analysis of the experiment is performed and the results of the calculation are compared with the measured data.

Comparative analysis between the RELAP5 calculations and the measured results shows that the RELAP5/MOD3.3 code is able to predict the thermal-hydraulic parameters of the PKL facility with a sufficient level of precision. The calculated values of the most important parameters and phenomena for simulating a main steam line break, primary and secondary pressure, fluid temperature, break flow rate and the mass flow rate in the primary circuit, are found to be in good agreement with the experimental results.

Deviation between the calculation and the experimental results was noted for the water level in the affected SG10 and for the collapsed level in the pressurizer. RELAP5 slightly under-predicts the water level in the SG and over-predicts the collapsed level in PZR during the transient. This was associated with discrepancy of calculated primary temperatures.

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11. ABSTRACT (200 words or less)

The behavior of the PKL facility during a small main-steam-line break was investigated by using the RELAP model of the main facility components, including the reactor pressure vessel, steam generators, pressurizer, primary piping, and main steam lines. The calculations were performed using the computer code RELAP5/MOD3.3.
The main objectives of the study are to compare the simulation results with experimental data and to assess the accuracy of the calculated data. The major parameters and phenomena for simulating a main steam line break, e.g. primary and secondary pressures and temperatures, break flow rate and water level in SG and PZR, were analyzed. The results of the calculations were found to be in good agreement with experimental results.
All the analytical activities were performed as part of the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) projects of Test G3.1.

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