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International Agreement Report

Assessment Study on the PMK-2 Total Loss of Feedwater Experiment Using RELAP5 Code

Prepared by
J. Bánáti
Lappeenranta University of Technology
Department of Energy Technology
P.O. Box 20
FIN-53851 Lappeenranta, Finland

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ABSTRACT

The main objective of the present report is to evaluate the predictability of RELAP5/MOD3.2.2 Beta computer code for the thermal-hydraulic system behavior during a total loss of feed water (LOFW) transient. The relevant experiment was conducted in the PMK-2 facility at the KFKI Atomic Energy Research Institute in Budapest, Hungary. The test simulated a beyond design-basis accident scenario with unavailability of the hydro-accumulators. For prevention of core damage, accident management strategies were applied, including a primary side bleed-and-feed procedure with intentional depressurization of the secondary side. After a brief description of the facility, the profile of the experiment is presented. Modeling aspects are discussed in the RELAP analysis of the test. Emphasis is placed on the ability of the code to reproduce the system response to opening of the pressurizer safety valve and feeding of coolant by the high pressure injection system. The calculations revealed that the code was capable of predicting the parameters with a sufficiently good overall agreement.

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ACKNOWLEDGMENTS

The author is grateful to the personnel of the PMK-2 team at the KFKI Atomic Energy Research Institute for conducting the tests and providing the experimental database to us. Special thanks are due to Dr. György Ézsöl and Dr. László Perneczky for their valuable suggestions and instructions.

NOMENCLATURE

BRU-A	Steam generator atmospheric relief valve
CL	Cold Leg
DC	Downcomer
ECC	Emergency Core Cooling
HA	Hydro Accumulator
HL	Hot Leg
HF	Henry-Fauske model
HPIS	High Pressure Injection System
IAEA	International Atomic Energy Agency
LOCA	Loss Of Coolant Accident
LP	Lower Plenum
LPIS	Low Pressure Injection System
MV	Motor Valve
NPP	Nuclear Power Plant
PMK	Paks Model Circuit (in Hungarian)
PRZ	Pressurizer
PV	Pressure Valve
PWR	Pressurized Water Reactor
SG	Steam Generator
SIT	Safety Injection Tank (synonym for hydro-accumulator)
SPE:	Standard Problem Exercise
TLFW:	Total Loss of Feed Water
UP:	Upper Plenum
VVER:	Water cooled Water moderated Energetic Reactor (in Russian)
dt:	Time step

1. INTRODUCTION

There are a few geometrical and structural differences between the Russian-design VVER-440 reactors and standard Western types pressurized water reactors widely used in the world. Due to the specific features of these nuclear power plants, the computer codes originally developed for typical PWRs have to be validated for VVER transients. In order to set up experimental databases, the PMK-2 facility was constructed in the early 1980s and upgraded during the past few years.

Performance of the Hungarian Paks NPP has been excellent since the beginning of their commercial operation and this PWRs are among the highest in the ranking of load factors. Looking at the status of system code verification process of VVER-440/213 in the thermal hydraulic area, there is a great need for enlargement of existing data base with new experiments. Without the proper benchmarking against experimental data, a computer code cannot be used with any degree of confidence. With this purpose in mind, the International Atomic Energy Agency organized test series, such as the Standard Problem Exercises (SPE-1 to 4) in the PMK-2 facility. Results of these large projects demonstrated the applicability of various computer codes for small break LOCA and primary-to-secondary leakage transients (IAEA 1987, 1988, 1991 and 1996).

The PMK-2 total loss of feed water test (TLFW) simulated a beyond design basis accident. This initiating event was selected, because loss of feed water accidents give a relatively high contribution to core melt frequency. Furthermore, this test served as an experimental basis to promote the implementation of accident management measures, e.g. primary and secondary side bleed and primary side feed, at the Paks NPP. More specifically, the principal aim of the present test was to reveal wether the reactor level can decrease to the elevation of the of the hot-leg and consequently, interrupting the natural circulation. On the other hand, the database is used in the evaluation of the effects of the HPIS injection on the pressure decrease in the primary circuit. The project was supported by the National Committee of Technological Development in 1995.

The goal of the current study is to contribute to validation process with an analysis using RELAP5/Mod 3.2.2 Beta version of the code against a specific type of accidental situation, when the feed water is lost and the operator intervenes. In particular, the code has been challenged by a few characteristic phenomena in the test, such as the simulation of the blowdown of the pressurizer and the heat transfer in the horizontal tube bundle in the steam generator during the evaporation of the secondary side inventory.

A thorough description on the findings of the TLFW experiment was worked out by Ézsöl et al. (1996). The main parameters of the PMK-2 facility and the reference reactor are compared in Table 1.

Parameter	**PMK-2**	**VVER-440**
Volumetric scaling ratio	1:2070	-
Vertical scaling factor	1:1	-
Number of primary loops	1	6
Max. heating power	0.7 MW	1375 MW
Number of fuel rods	19	44000
Diameter of heater rods	9.1 mm	9.1 mm
Heated length of the core	2.50 m	2.42 m
Axial power distribution	uniform	cosine
Axial peaking factor	1	-
Number of SG	1	6
Number of SG tubes	82	5500
Average SG tube length	3.715 m	9.26 m
Number of SG tube rows	82	76
Pitch of the SG tubes	18-32 mm	24 mm
Max. operating pressure	12.3 MPa	12.3 MPa
Nominal prim. temperature	573 K	573 K
Nominal. sec. pressure	4.65 MPa	4.65 MPa
Nominal sec. temperature	533 K	533 K
Accumulator pressure	5.5 MPa	5.5 MPa
LPIS pressure	0.7 MPa	0.7 MPa
HPIS pressure	12.0 MPa	12.0 MPa

Table 1: Main parameters of the PMK-2 facility and the VVER-440 reactor

2. MODELING OF THE VVER-440 REACTOR

2.1 The Reference Reactor

The VVER-440/213 PWRs of Russian design are located in mainly Eastern and Central Europe, among others, at Loviisa, in Finland and at Paks, in Hungary. This type of reactor has a number of unique features, such as horizontal steam generators, combined safety injection, accumulator injection to upper plenum and to the downcomer with higher set-point pressure than the secondary pressure, loop-seal in both hot-leg and cold-leg, and fuel bundles in hexagonal arrangement. Consequently, compared with Western type PWRs, the system's response to the disturbances can be different. The following phenomena have also been reproduced in most of the PMK-2 experiments:

- Heat transfer in the horizontal tube bundle of the steam generator: because of the relatively large secondary side inventory, the temperature and density distribution is different and internal circulation may develop.
- Loop-seal effect: phase separation in the horizontal components is different, with the possible formation of loop-seals in the hot-legs and cold-legs. Loop-seals in the hot-leg may reduce or completely stop primary coolant flow to the SG during natural circulation or in boiler-condenser mode.
- Different mixing and condensation effects, especially during emergency core cooling (ECC) injection to the upper plenum.

2.2 Characteristics of the PMK-2 Facility

The PMK-2 is a full-pressure, integral-type model of the four VVER-440 type pressurized water reactors used in the Paks NPP in Hungary. The facility (Figure 1) is located in the KFKI Atomic Energy Research Institute in Budapest in Hungary. The aspect ratio for power and volumes is 1:2070. Relative elevations are preserved the same as in the reference reactor, (except for the pressurizer and lower plenum), in order to maintain the driving head for natural circulation. The primary and secondary sides of the PMK-2 correspond to six loops in the NPP. The facility is equipped with similar safety systems, e.g. two hydro-accumulators (safety injection tanks,

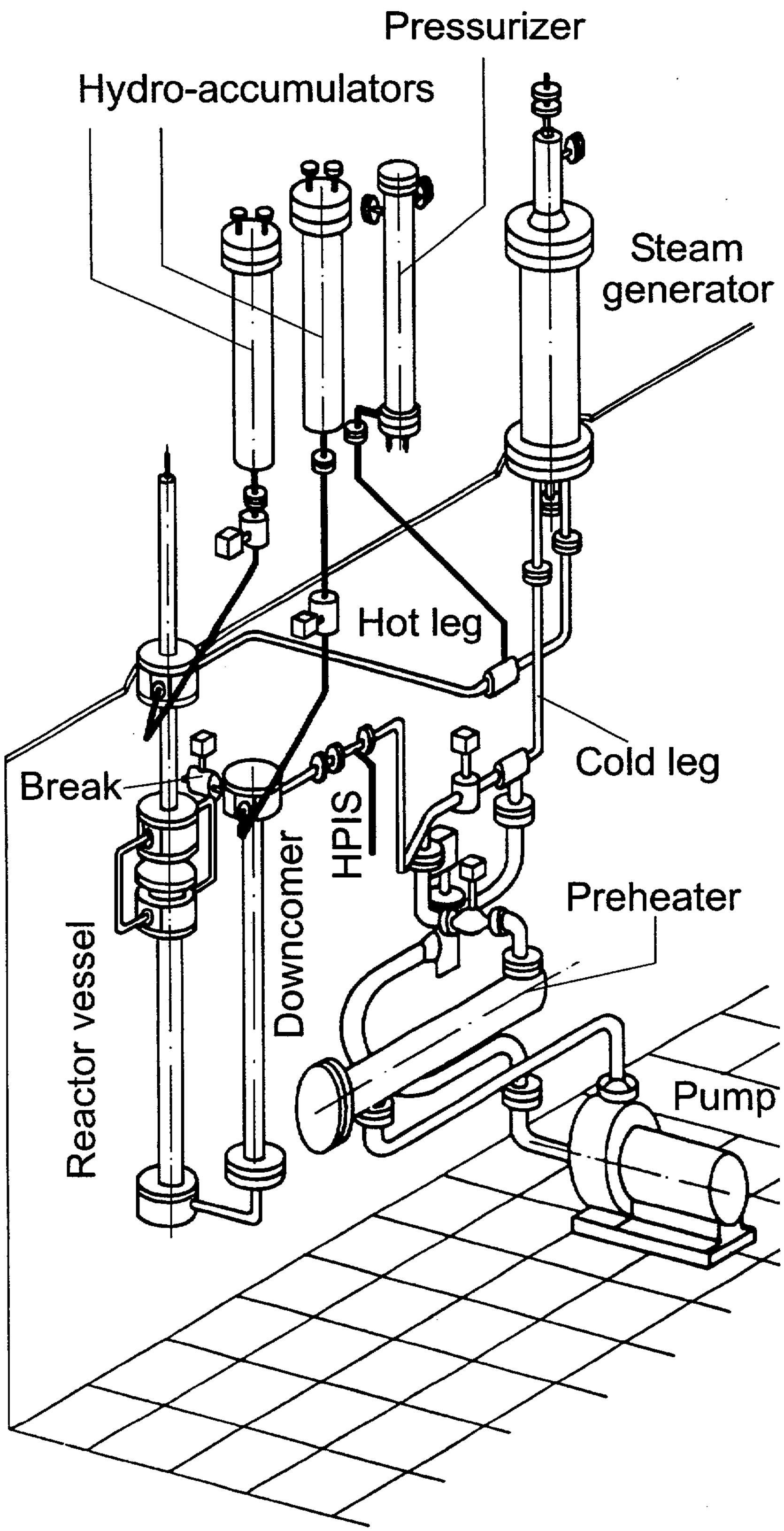

Figure 1: Axonometric view of the PMK-2 facility

SITs), high and low pressure injection systems (HPIS and LPIS). The SIT-1 injects water to the top of the downcomer, while the SIT-2 is connected to the upper plenum. The LPIS is attached to the downcomer head. The setpoint values of these components are identical to those of the reference plant. The reactor vessel is simulated with a U-tube construction, consisting of the core model and an external downcomer. The core itself comprises 19 directly heated fuel rod simulators embedded in a hexagonal ceramic shroud (Figure 2). The diameter and pitch of the heater rods are 9.1 mm and 12.2 mm, respectively and the active length is 2.5 m. The nominal power of 664 kW is distributed uniformly along the height of the rods.

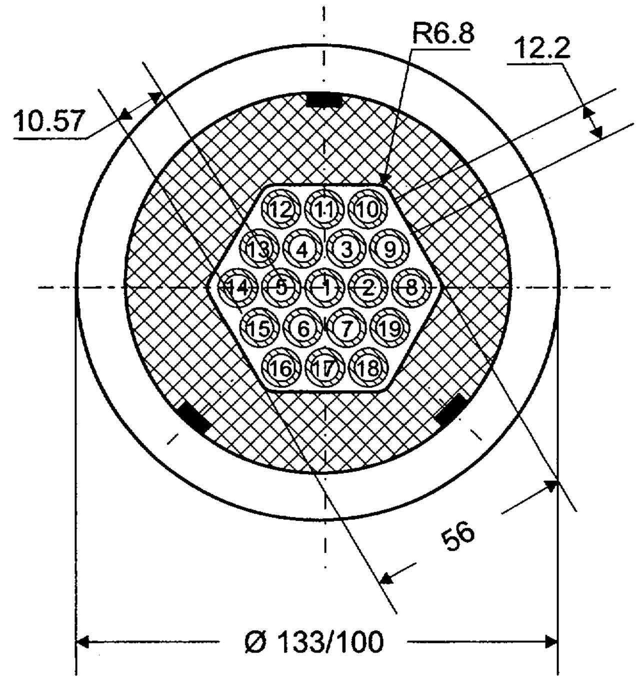

Figure 2: Cross section of the core

The steam generator model (Figure 3) is a full height, vertically divided section of the horizontal VVER-440 steam generator, with 82 tubes bent in serpentine shape. The ratio between the volumes of steam and water is kept in the secondary side. The PV23 valve acts as a steam relief valve, which opens and closes at the setpoint secondary pressures or by operator intervention. Steam output from the SG can be controlled by the PV22 valve, which is used before the transient initiation.

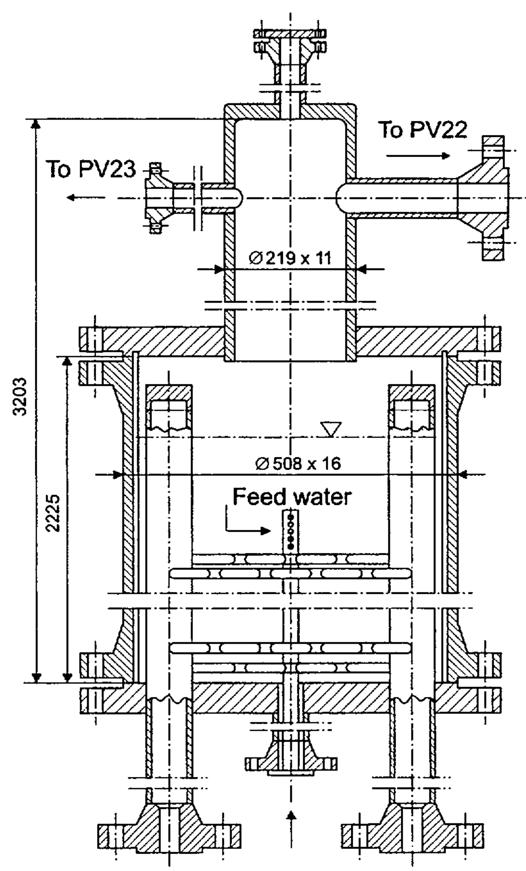

Figure 3: The SG model of the PMK-2 facility

3. PROFILE OF THE PMK-2 TLFW EXPERIMENT

The total loss of feed water transient started from the nominal operating parameters of the reference plant at time=0, when the feed water is lost. The initial conditions and the main occurrences of the PMK-2 TLFW test are listed in Tables 2 and 3, respectively. As a conservative assumption, the hydro-accumulators were considered to be inoperable. The core input power used in this test followed the decay heat curve after the scram. The test included the modeling of the pump trip and coast-down. In order to promote the implementation of the accident management measures in the Paks NPP, primary bleed-and-feed (using the pressurizer safety valve and the HPIS, respectively) and bleed of the sec. side by opening the SG relief valve (BRU-A) were modeled. The primary feed was assured by only 1 operable HPIS and it was triggered from the signal of "PRZ level 1 m below nominal". The primary and secondary bleed valves were modeled with orifices of 1 mm and 4 mm of diameter, respectively. These valves remained open after their actuation. Location of the transducers and the measured initial values are shown on the simplified flow diagrams in Figures 4, 5 and 6.

Parameter	Code	Value
Primary pressure	PR21	12.23 ± 0.05 MPa
Loop flow	FL53	5.44 ± 0.06 kg/s
Core inlet temp.	TE63	530.3 ± 1 K
Core power	PW01	649.6 ± 3 kW
Level in PRZ	LE71	9.308 ± 0.02 m
SG sec. pressure	PR81	4.22 ± 0.02 MPa
Level in SG secondary	LE81	9.03 ± 0.05 m
Feed water flow	FL21	0.37 ± 0.02 kg/s
Feed water temp.	TE21	500.6 ± 1 K

Table 2: Initial conditions

Occurrence	Time	Actuated by
SG isolation	0 s	
Loss of feed water	0 s	
Scram	6 s	PR21=12.0 MPa
Sec. bleed	127 s	PR81=4.96 MPa
Primary feed	513 s	LE71=8.33 m
Pump trip	1523 s	PR21=7.6 MPa
Primary bleed	1523 s	PR21=7.6 MPa
Pump stopped	1673 s	Pump trip + 150 s
Test terminated	7886 s	

Table 3: Occurrences in the test

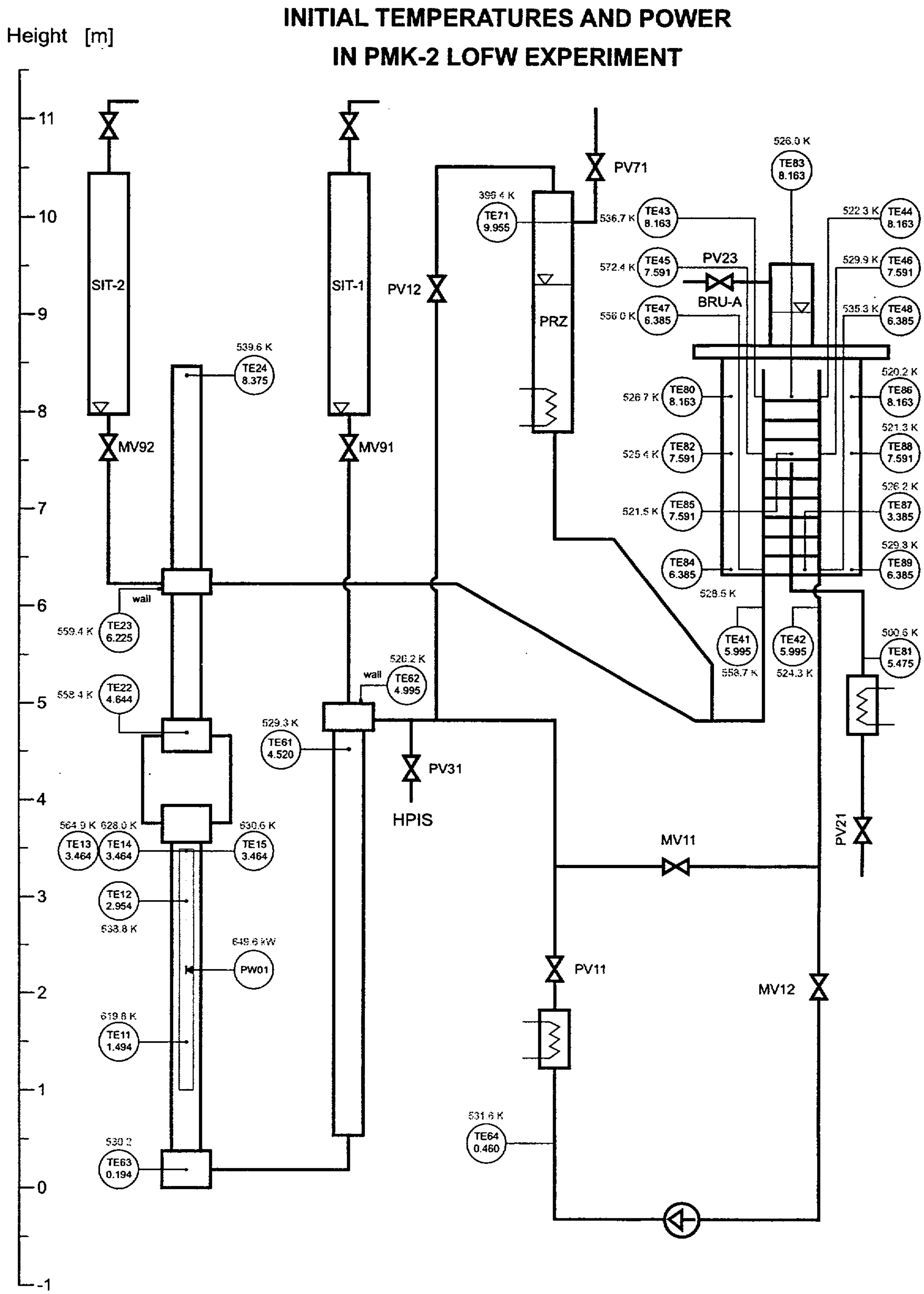

Figure 4: Initial temperatures and the heating power

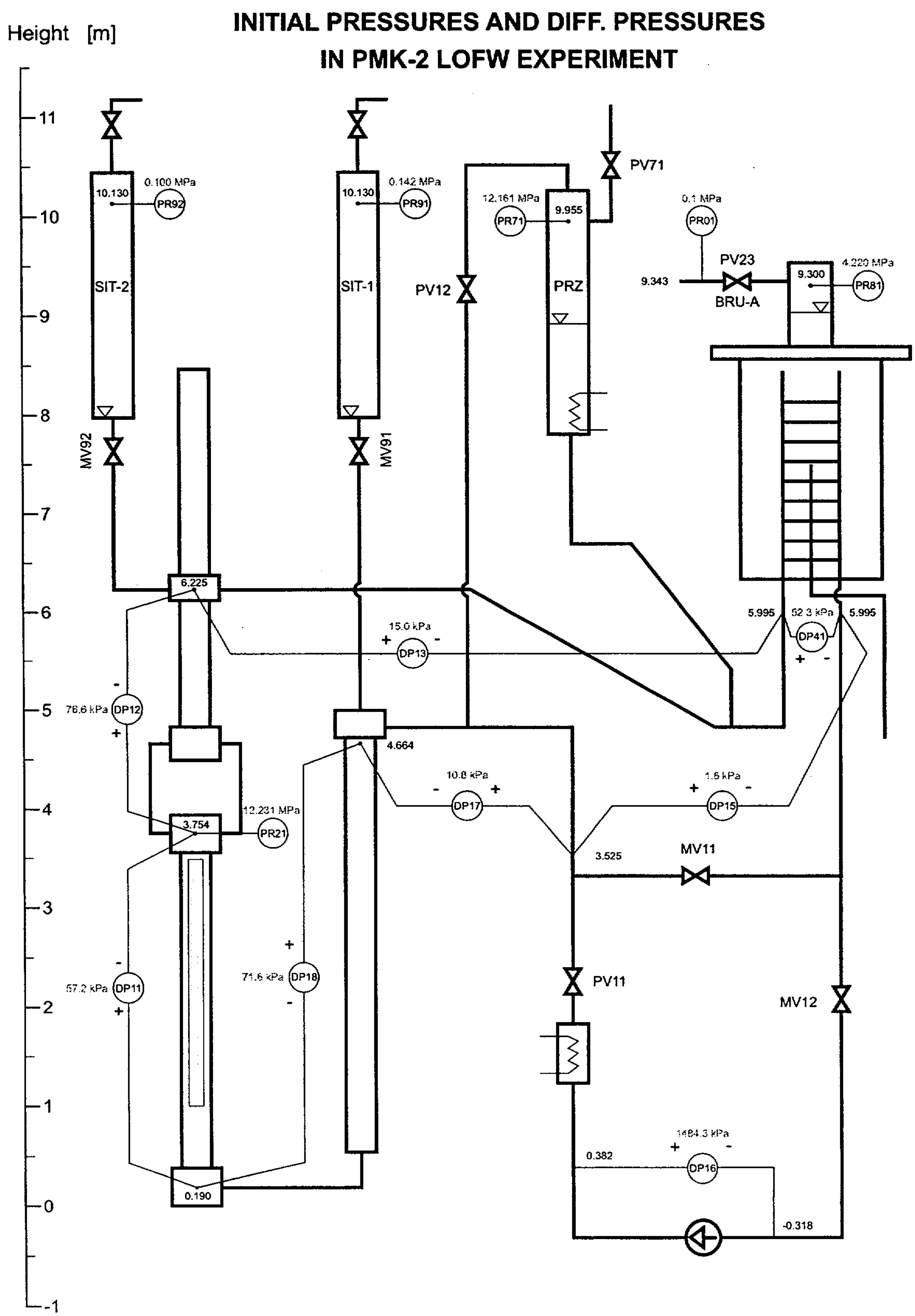

Figure 5: Initial pressures and differential pressures

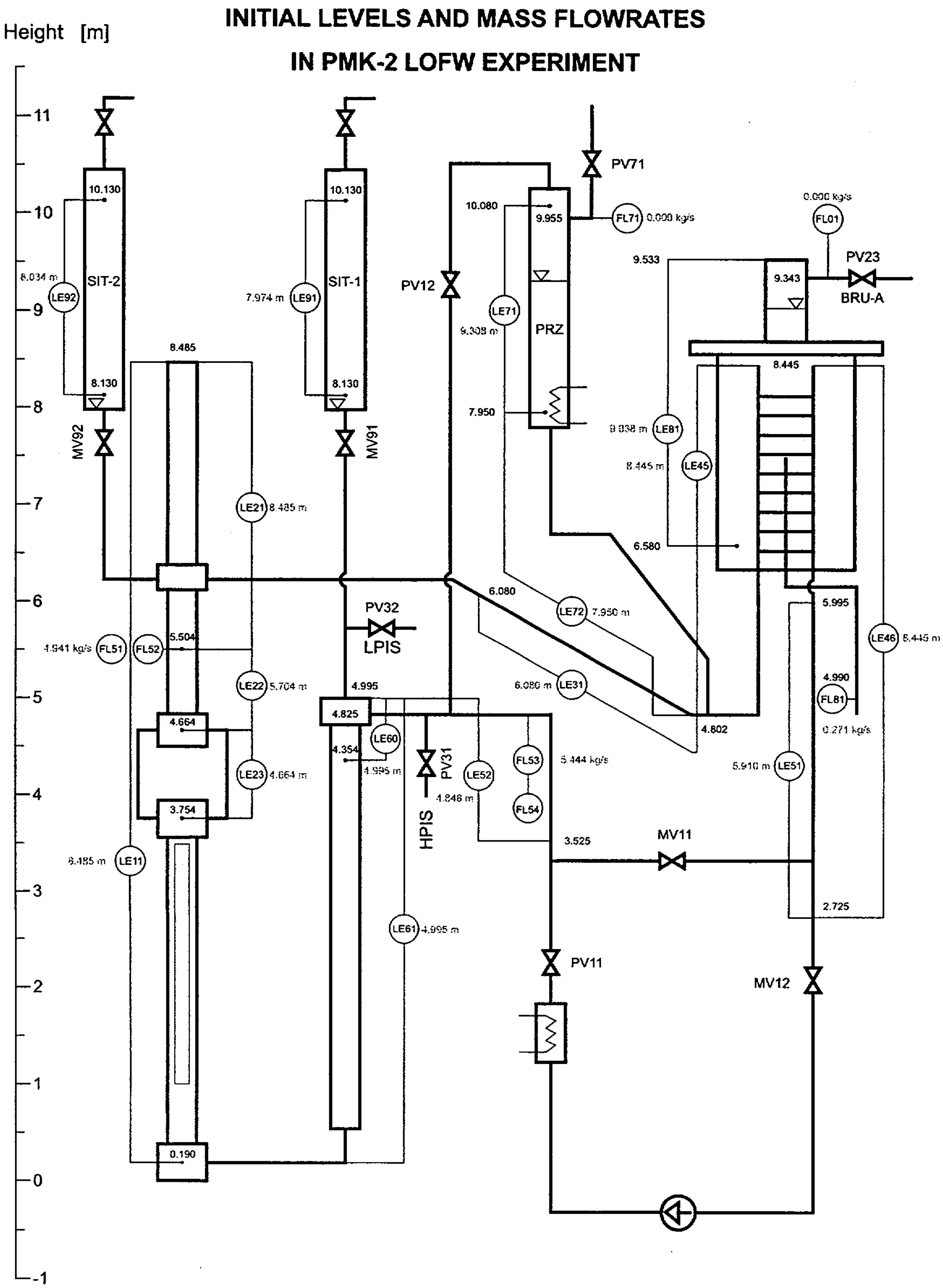

Figure 6: Initial collapsed levels

The transient can be characterized by the variation of pressure in the upper plenum. As it is shown in Figure 7, PR21 decreases rapidly after the scram and this trend slows down until the secondary bleed (opening of PV23, t=127 s). Due to the heat sink and the lower secondary pressure, further decrease

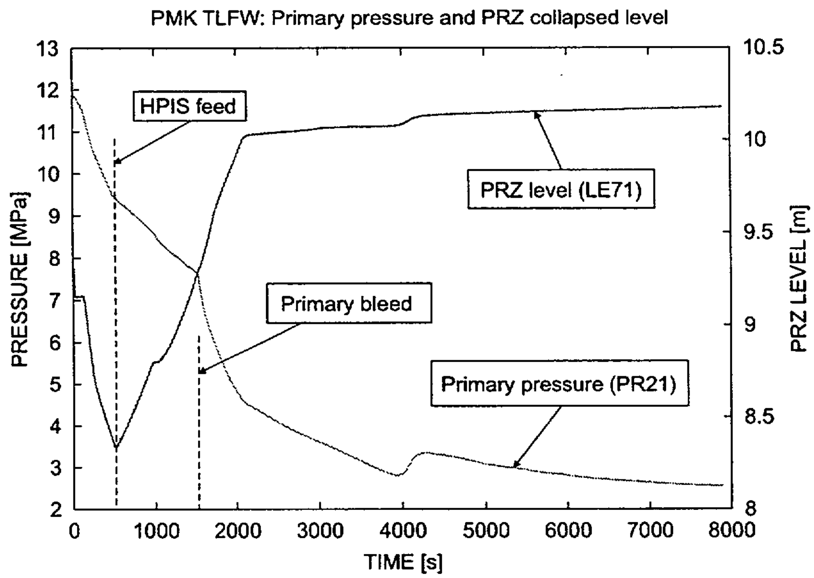

Figure 7: Primary pressure and the collapsed level in the PRZ

was observed in PR21. The first characteristic point can be seen at the actuation of high pressure injection system (primary feed, t=513 s). Initially, the PRZ level decreases due to the lower coolant temperature. After the start-up of the HPIS, the PRZ level starts to rise again. Reaching the nominal level in the PRZ, the safety valve PV71 opens and the primary bleed begins at 1523 s. The fast depressurization continues until approximately 2100 s, when two-phase flow starts to be discharged. The maximal level in the PRZ is approached at 4000 s, followed by a nearly stabilized state. In this phase, the inventory loss through the safety valve is roughly compensated by mass of the primary feed.

The pressure in the steam generator secondary side is depicted in Figure 8. The test results show that the SG is effective for the whole duration of the transient. The core remains covered, since changing of the level in the primary side takes place only in the PRZ.

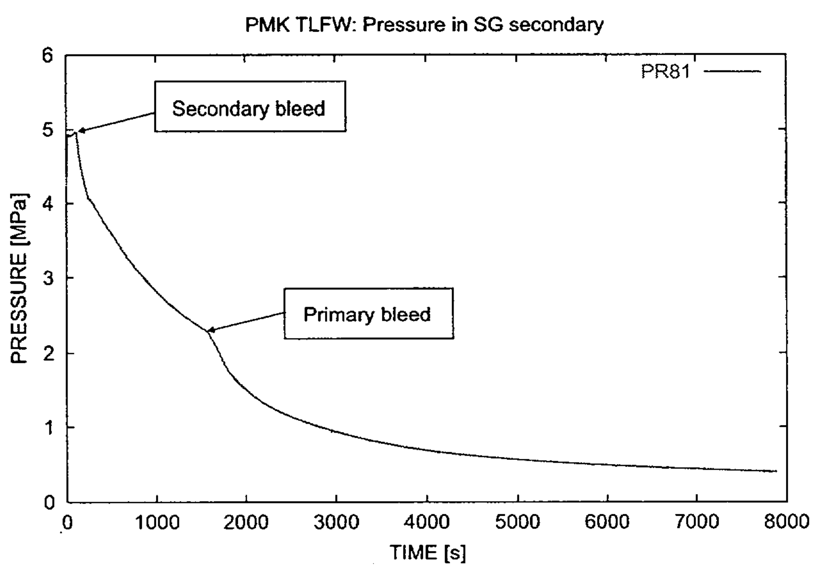

Figure 8: Effects of the bleed on the secondary side pressure

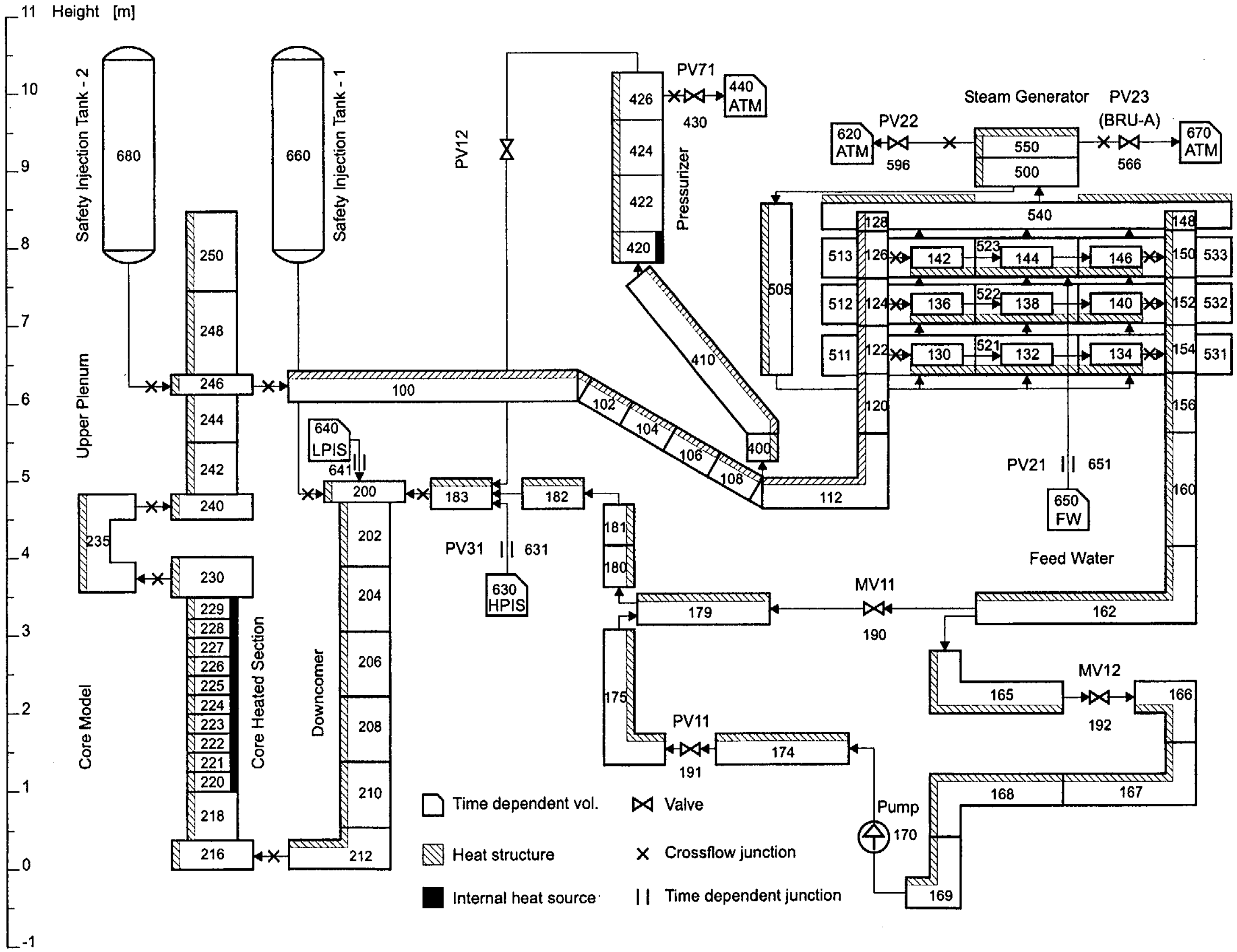

Figure 9: Nodalization scheme of the PMK-2

4. ANALYSIS OF THE RELAP CALCULATIONS

4.1 Features of the RELAP5 input

The basic structure of the nodalization scheme (Figure 9) was adopted from the former IAEA Standard Problem Exercise calculations, with a few modifications according to the needs of the TLFW test. (E.g. the steam line was removed and the pressurizer surge line was connected to the bottom of the hot leg). The model includes 96 volumes and 105 junctions.

The main features of the input can be summarized as follows: most of the hydrodynamic volumes were modeled with branches. Crossflow junctions are applied at connecting the following parts: downcomer to the lower plenum, upper plenum U-tubes, upper plenum to the hot leg, SIT connections, cold leg connection to the downcomer top, valve connections to the steam dome, the pressurizer safety valve and finally the steam generator hot and cold collector connections to the heat exchange tubes.

The core is axially subdivided into 10 parts. Rod bundle interphase friction model is used in the volumes of the heated section. Counter-current flow limitation (CCFL) model is turned on only in the core and the downcomer. Choking option is applied in the vicinity of the pressurizer safety valve (PV71) and the steam generator relief valve (PV23), in order to simulate the primary and secondary bleed processes.

In fact, the pressurizer safety valve behaves like a break in the primary side during the bleed phase. Consequently, it is necessary to assure the adequate value for the valve discharge coefficient to match the correct amount of released primary coolant. The same requirement is valid for the SG depressurization.

The steam generator model includes basically three primary side horizontal layers for the heat exchanger tubes and three vertical volumes in the secondary side. The steam dome consists of a separator component near the initial secondary water level. The original PMK-2 steam generator does not contain any steam separator. However, this arrangement helped to intensify the internal circulation and resulted in a better heat transfer. Based on the experiences of earlier calculations, void fraction limitation quantities of 0.35 and 0.55 yielded to the best results for the vapor exit junction and the liquid fall-back junction, respectively.

The current RELAP5 model allows the code user to have full control over appropriate simulation of the pump. The steady-state loop flow rate is achieved by regulation of the pump speed. In the transient, the by-pass line is closed by the MV11 valve and the pump is running at constant speed until the pump trip. When the pump is tripped, the PV11 valve starts to close while the MV11 valve is gradually opening in a prescribed manner. Simulation of the pump coast-down is terminated by closing of the MV12 valve. The nominal length of this process is 150 s.

Heat structures are used in the heater rod simulators, in the PRZ heaters, in the steam generator tubes and in all the pipe walls along the facility, describing the heat exchange between the environment. For heat losses, convective boundary conditions were considered, with a constant heat transfer coefficient of 5 W/m2K.

The present model of the PMK-2 includes the safety injection tanks and the pressurizer spray line, however, these components had no specific role during this particular transient.

The initial data set and set-point values were derived from the nominal operating parameters of the VVER-440 plant. The strategies used to achieve steady-state with the code were similar to those of applied in earlier analyses. Auxiliary components were connected to primary and secondary sides to stabilize the initial parameters nearly at the same values as it was observed in the test.

4.2 The Base Case Calculation

The base case calculation focused on the system behavior during the steam dump at the pressurizer and SG relief valve. For this purpose, homogeneous (h=2) choking flow (c=0) option was selected at the bleed valve junctions, with default discharge coefficients (1.0, 1.0, 1.0).

The search run took only a few seconds with using the "steady-state" option of the code. The requested time step was 0.1 s for the whole duration of the transient calculation. The transient started by the isolation of the SG secondary side. Following the scram at 6 s of transient time, the depressurization of the primary system started rapidly. Figure 10 illustrates the time variation of measured and calculated values of the pressure in the SG secondary side. The set-point of the SG relief valve was reached at 127 s. At this time the valve PV23 opened and remained in open position for the rest of the transient. Bleeding of the secondary side resulted in a fast decreasing of pressure.

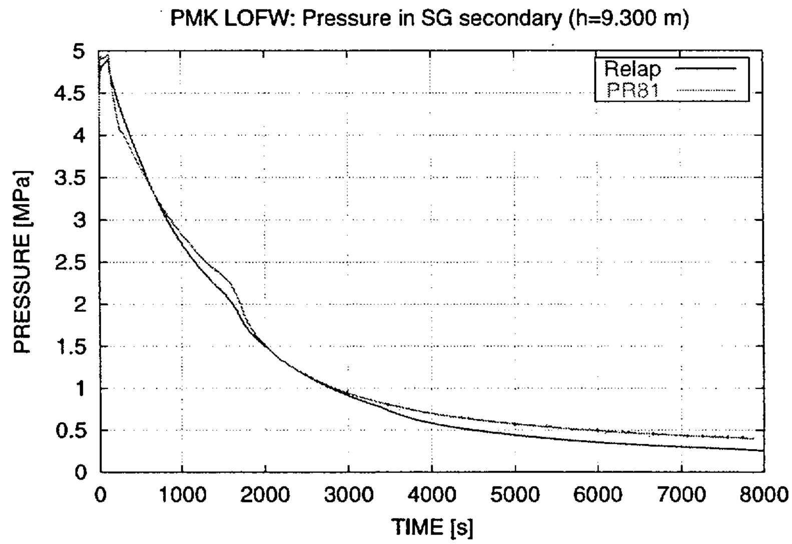

Figure 10: Pressures in the SG secondary side

While RELAP5 predicted the SG pressure at a good accuracy, the pressure in the upper plenum was well simulated only in the first part, until about 4000 s (Figure 11). The first characteristic point can be observed at 513 s, when the HPIS (feed) was initiated. Beginning of the primary side bleed is well matched at 1523 s. However, the slight increase of the pressure after 4000 s can be explained by the water level reaching the elevation of PRZ safety valve.

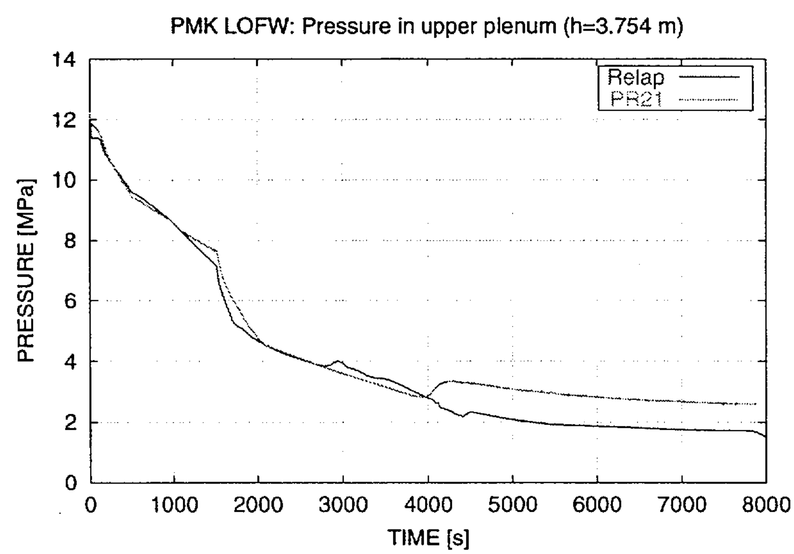

Figure 11: Pressures in the upper plenum

By examination of Figure 12, the following events have particular interest. There was a continuous dropping of the collapsed level in the pressurizer until 513 s. At this time the primary side feed was actuated. Injection of cold water by HPIS started to fill up the pressurizer. The collapsed level reached the nominal value at 1523 s. At this moment the primary side bleed process was initiated by opening the pressurizer safety valve PV71. The fill-up process was terminated at approximately 2100 s, when the collapsed level reached the elevation of valve PV71. It can be concluded that RELAP5 predicted the variation of the level in the PRZ fairly well. In the period, while there is one phase flow through the safety valve, is well simulated. The nearly-stabilized level is only about 0.1 m higher in the calculation than in the experiment. (Rather complex thermal-hydraulic process took place in the vicinity of the PRZ safety valve during the primary bleed. Arrangement of the valve and the ducts of the DP and level measurements should be considered in this respect).

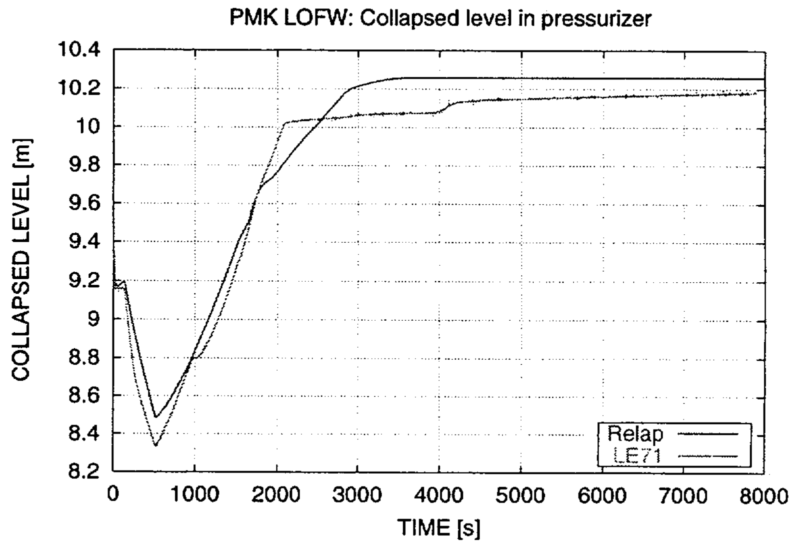

Figure 12: Collapsed level in the PRZ

The mass flow rate fell back to zero twice soon after triggering the bleed. As Figure 13 shows, RELAP could not reproduce the short stagnations of the steam flow between 1500 and 2000 s. However, the general trend is sufficiently well calculated with using default values for the valve.

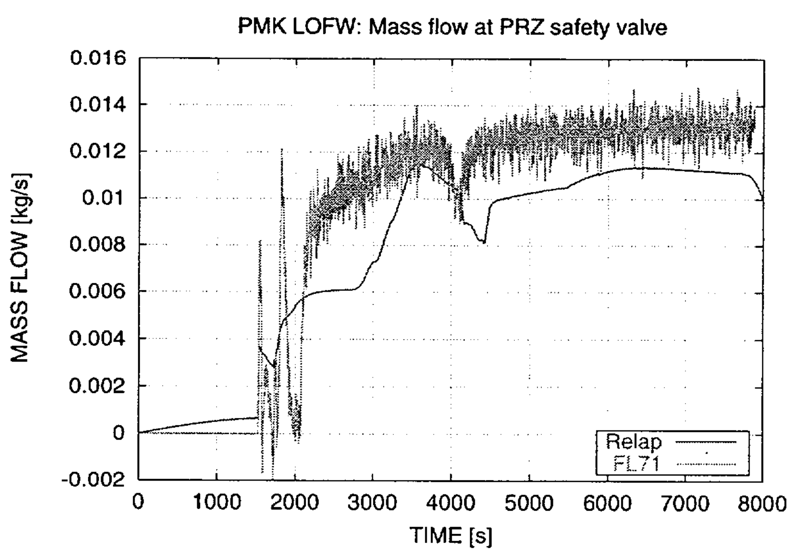

Figure 13: Mass flow rate at PRZ safety valve

The total inventory loss through the PRZ safety valve is depicted in Figure 14. It is visible that this parameter is underestimated. The final calculated value is approximately 60 kg, instead of the measured amount of 70 kg. (A minor initial discharge of the safety valve was assumed before the valve opening in the calculation).

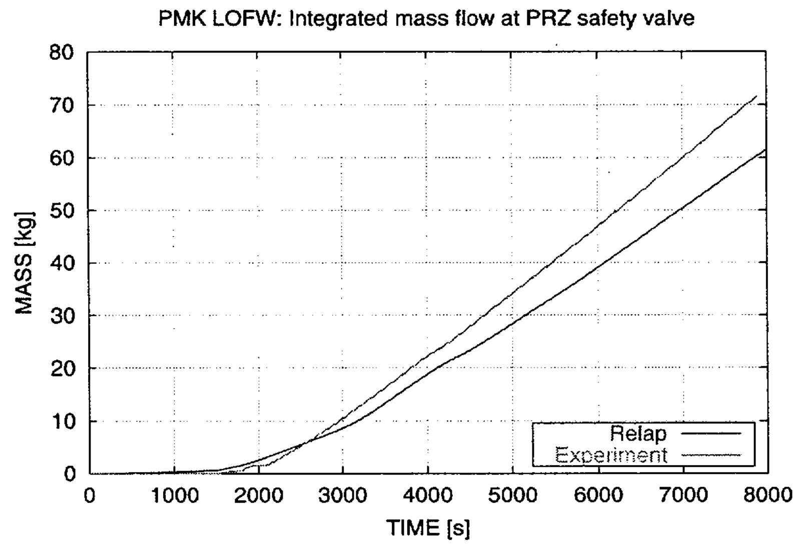

Figure 14: Integrated mass flow at the PRZ safety valve

Bleeding of the secondary side can be seen in Figure 15. Opening of the SG safety valve occurred at 127 s of simulation time. The BRU-A valve was intentionally left open for the rest of the transient. This resulted in a continuous dumping of the steam through an orifice with a diameter of 4 mm. This value was selected considering the scaling factor of the PMK-2 and the fact that one steam generator simulates 6 ones in the reference reactor.

The bleeding process is very well matched by the code calculation. Using the default values for the orifice assured an excellent reproduction of the total mass in the first half of the transient. The discrepancy increased just a few percent near the end of the simulation.

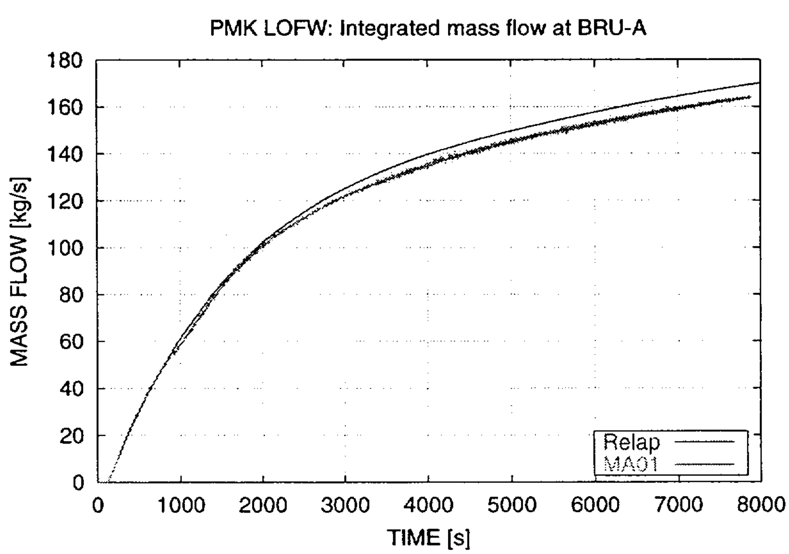

Figure 15: The total steam mass dumped from the SG

The history of the temperature of the heater rods at the elevation of 3.464 m, near the exit of the core, is shown in Figure 16. Due to the scram, the temperature started to drop very rapidly. A stagnation can be observed at nearly 540 K for a short duration, followed by a decreasing. This parameter was also simulated with a good degree of accuracy until approx. 500 s. However, it started to be slightly underestimated before a minor heat-excursion at 1800 s. The timing and the magnitude of the minor were well predicted by RELAP.

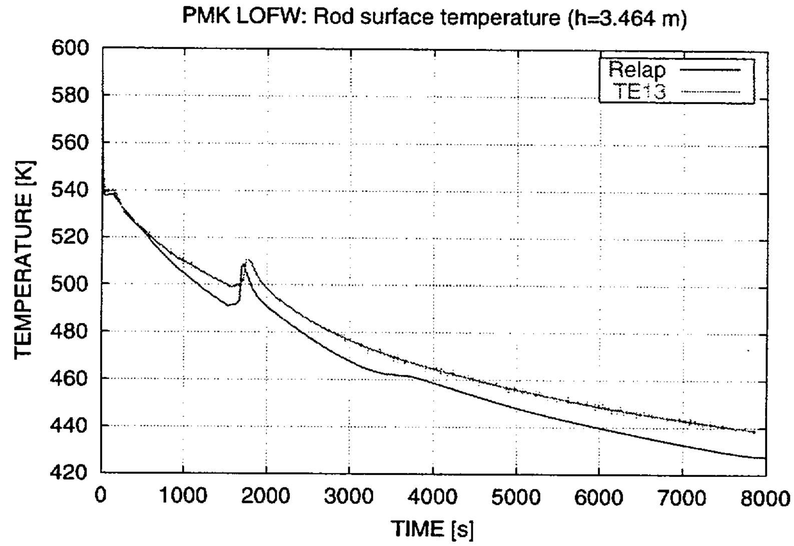

Figure 16: Rod surface temperatures

The primary fluid temperature has similar characteristic points as the secondary pressure curve. Figure 17 shows the coolant temperature at the inlet of the core. Reaching the maximum at the initiation of the SG bleed, the process is driven by the energy release in the secondary. Decrease is accelerated after opening of the PRZ valve. This is the break point of the curve at ~1800 s.

The calculation has confirmed that the reactor level remained stable. Voiding did not take place in the upper plenum and consequently, the core was constantly covered.

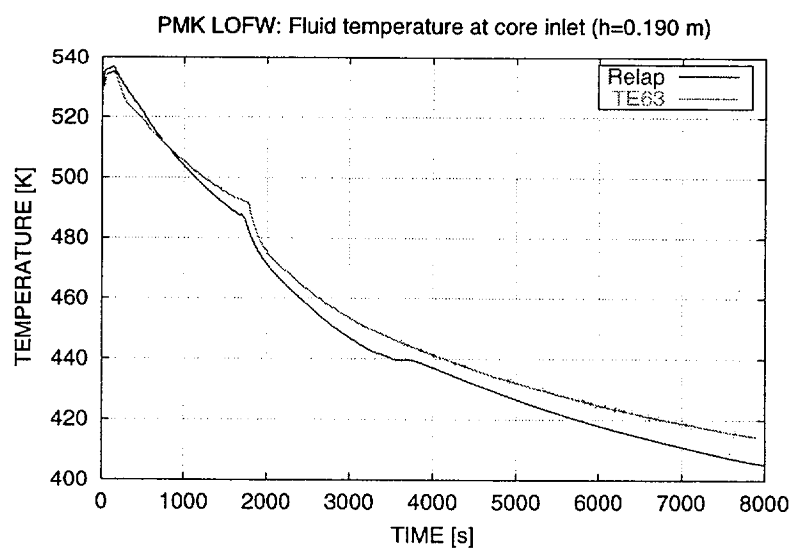

Figure 17: Fluid temperature at the core inlet

4.3 Sensitivity Study

It was found in the base-case calculations that the process was sensitive for both the primary and secondary bleed. Using default discharge coefficients underestimated the PRZ bleed flow. Increasing the value resulted in wrongly simulated primary pressure. There is an option in the currently applied version of the code to use the Henry-Fauske critical flow model. In order to evaluate the effects of the new model, sensitivity calculations were performed with varied parameters. Values of 0.85 and 1.2 were used for the discharge coefficient, while the thermal non-equilibrium parameter was left constant (0.14, as recommended in the code manual). Figures 18, 19 and 20 show the results of comparison to the previous calculation, where the H-F model was not applied. It can be seen in Figure 18 that the smallest discrepancy is achieved without the HF model, until approx. 4000 s, but c=0.85 shows better result after that. A similar conclusion can be drawn for the PRZ level but all the 3 parameters resulted in identical calculated levels after 4600 s.

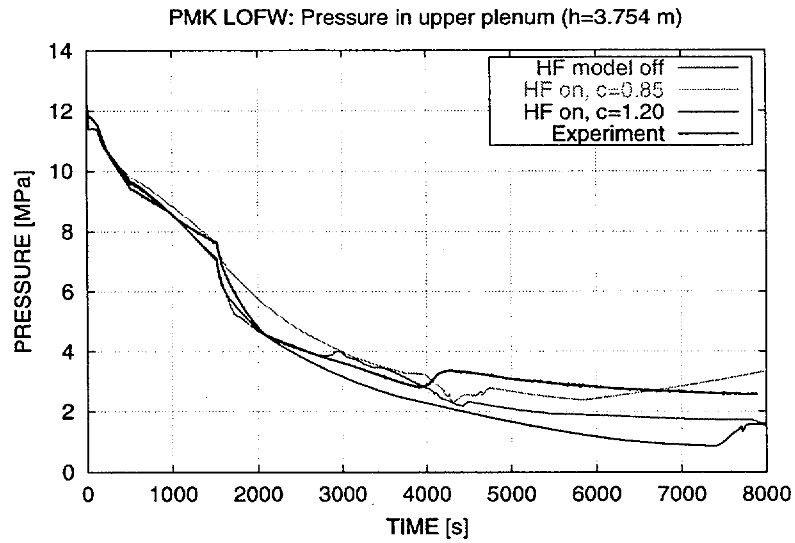

Figure 18: Primary pressures

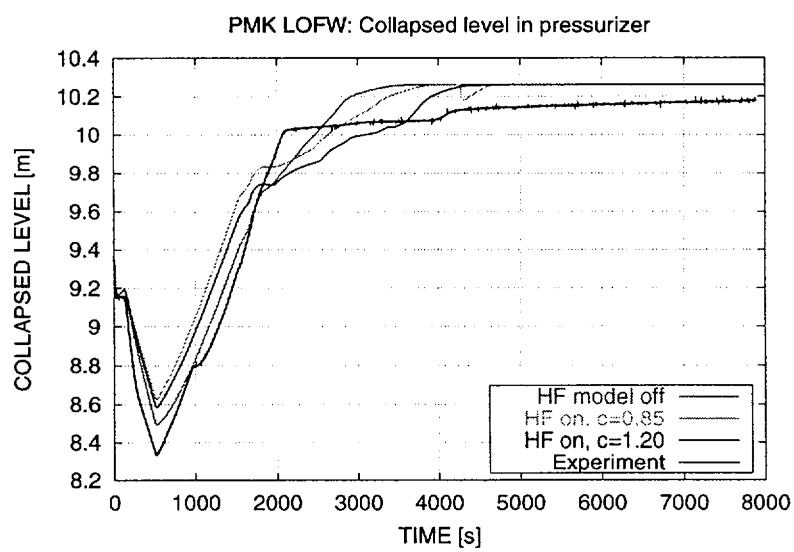

Figure 19: Collapsed levels in the PRZ

It is interesting to note that the highest discharge coefficient gives the best correlation in the first half of the transient but the flow becomes more underestimated until the end of the calculation (Figure 20). Looking at the plots of the sensitivity study, it cannot be stated that the results of Henry-Fauske model would be in clearly better agreement with the experimental data. The ordinary critical flow model shows closer matching for some of the parameters.

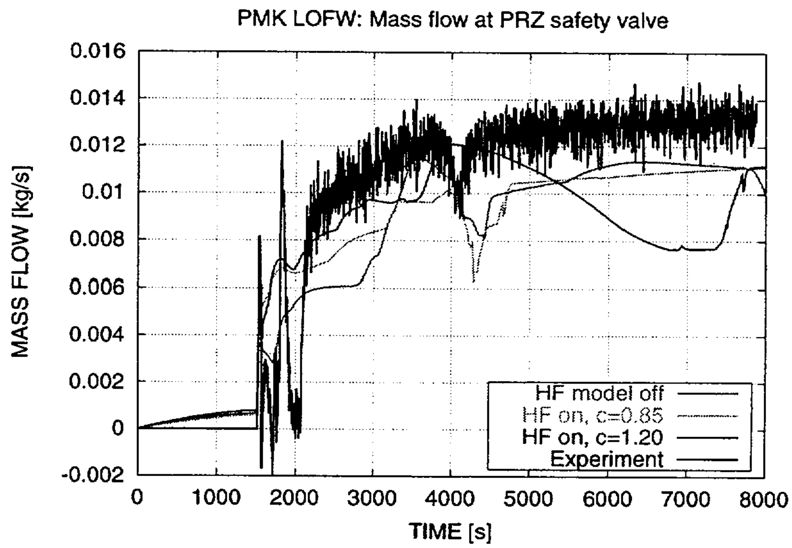

Figure 20: Mass flow rate in PRZ safety valve

4.4 Run Statistics

The calculations were performed on a Silicon Graphics Octane workstation with 175 MHz R10000 processor and 256 MB central memory.

With requested time step of 0.1 s, the 8000 s long transient consumed 1371.07 s CPU time. It means the CPU time / transient time ratio was 1371.07 s / 8000 s = 0.17138375.

The mass error ratio (emass / tmass) was reasonably low: -0.354311 kg / 361.207 kg = -9.809086E-04

5. CONCLUSIONS

The analyses with RELAP5/MOD3.2 computer code showed that code is capable of predicting the key events in the experiment. It can be stated generally, that main trends have been repeated in the calculation and the qualitative agreement is good.

Matching of the calculated and measured conditions was excellent in the pressurizer. From thermal hydraulic aspects, very complex processes took place in this relatively small volume: feeding with cold ECC water and bleeding by the safety valve, furthermore, phase separation near the elevation of the valve. The inventory loss through the bleed valve challenged the code because it behaved as a small break in the top of PRZ. RELAP5 calculated this process with a good accuracy.

In spite of the fact that the code was originally developed for simulation of reactors with vertical steam generators, the secondary side model has been proven to be adequate. The horizontal heat exchanger tubes were lumped into 3 layers. Even with this simplified arrangement, the results produced by the code were in very good agreement with the experimental data. The bleed phase was better simulated in the secondary side.

The sensitivity study revealed that the application of the Henry-Fauske critical flow model did not necessarily improved the prediction of the two-phase flow at the orifices.
Dependence of the results on the requested time step value was not experienced. The results obtained by this simulation has confirmed that the built-in automatic time step selection method of RELAP5/MOD3.2.2beta could be used with confidence.

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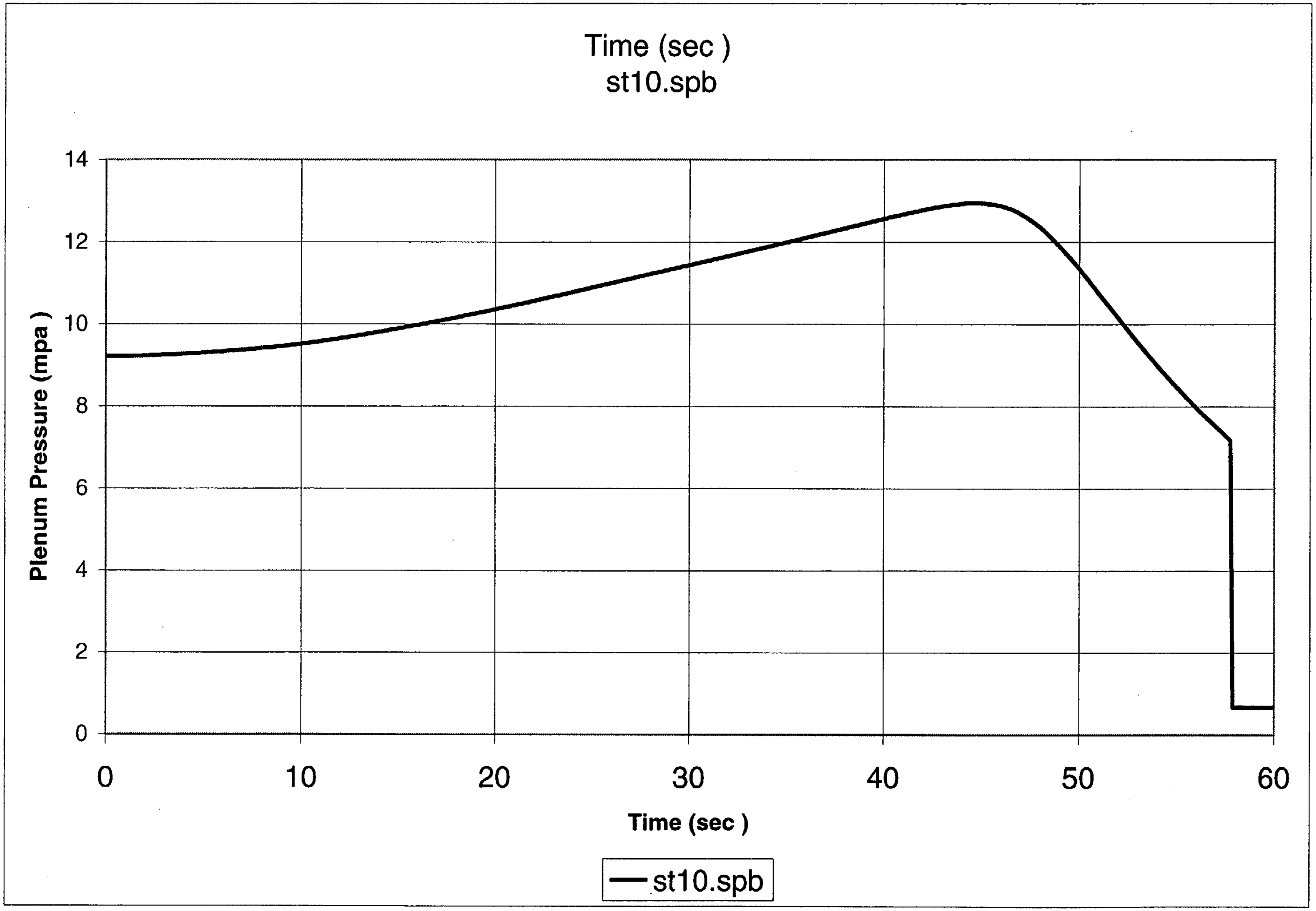
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6
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0
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10
20
30
40
50
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Time (sec)
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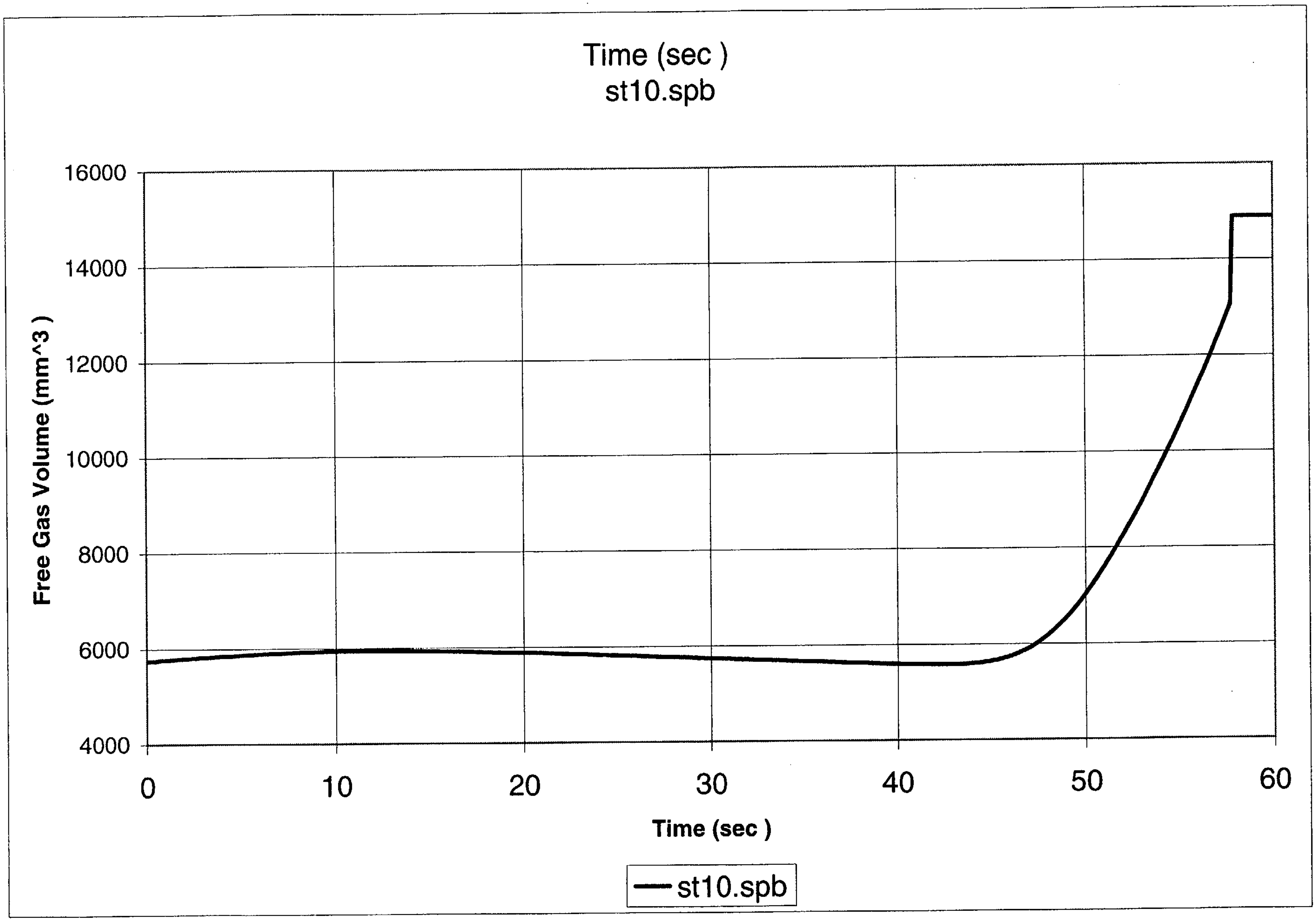
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12000
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4000
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30
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50
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Time (sec)
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FIN-53851 Lappeenranta, Finland

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11. ABSTRACT *(200 words or less)*

The main objective of the present report is to evaluate the predictability of RELAP5/MOD3.2.2 Beta computer code for the thermal-hydraulic system behavior during a total loss of feed water (LOFW) Transient. The relevant experiment was conducted in the PMK-2 facility at the KFKI Atomic Energy Research Institute in Budapest, Hungary. The test simulated a beyond design-basis accident scenario with unavailability of the hydro-accumulators. For prevention of core damage, accident management strategies were applied, including a primary side bleed-and-feed procedure with intentional depressurization of the secondary side. After a brief description of the facility, the profile of the experiment is presented. Modeling aspects are discussed in the RELAP analysis of the test. Emphasis is placed on the ability of the code to reproduce the system response to opening of the pressurizer safety valve and feeding of coolant by the high pressure injection system. The calculations revealed that the code was capable of predicting the parameters with a sufficiently good overall agreement.

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