NUREG/IA-0246

International Agreement Report

RELAP5/MOD3.3 Assessment against PMK Test T3.1 – LBLOCA with Nitrogen in PRZ

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Washington, DC 20555-0001

November 2010

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the International Code Assessment and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

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NUREG/IA-0246

RELAP5/MOD3.3 Assessment against PMK Test T3.1 - LBLOCA with Nitrogen in PRZ

Manuscript Completed: November 2008
Date Published: October 2010

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Abstract

The results of RELAP5 post-test analyses of test T3.1 performed on the PMK experimental facility are presented. The Hungarian facility PMK is a scaled-down model of NPP with VVER-440/213 reactor. The code versions RELAP5/MOD3.3hg (post Patch03) and RELAP5/MOD3.3ef (Patch02) have been assessed against the experimental data from the test T3.1. The test T3.1 was a large-break LOCA with 30% break starting from shutdown conditions with nitrogen in PRZ. Generally, both prediction of system behavior and prediction of nitrogen transport are in very good agreement with measured data.

FOREWORD

The RELAP5 is a very important computational tool for increasing nuclear safety also of the VVER reactors, especially in the Czech Republic. The Nuclear Research Institute (NRI) Rez has assessed the code against numerous experiments and consequently applied it to safety analyses of Czech NPP. The presented report documents one of the assessment works.

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EXECUTIVE SUMMARY

The PMK-2 facility [3] is a scaled down model of the VVER-440/213 and it had been primarily designed for investigation of small-break loss of coolant accidents (SBLOCA) and transient processes of this type of NPP. Nowadays the facility is also widely used for assessment of advanced computer code, that are used for safety analysis in VVER-operating countries.

One of the most important and world-widespread computer codes is the RELAP5 code. In the Czech Republic, the RELAP5 is installed under agreement between US NRC and Czech regulatory body (SONS). The main user of the code is the Nuclear Research Institute (NRI, UJV) Rez, where the code is widely assessed and applied to NPP safety analyses.

The test T3.1 used in this report for assessment of RELAP5/MOD3.3 computer code is large-break LOCA with 30% break in cold leg starting from shutdown conditions with nitrogen in PRZ.

Comparison of the measured test data and the RELAP5/MOD3.3 results showed very good overall agreement of all major system parameters as primary pressure, reactor level, reactor coolant and clad temperature etc. Also the prediction of nitrogen transport in primary system was in very good agreement with the measured data.

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ACKNOWLEDGMENTS

The authors acknowledge the support of the Czech regulatory body - the State Office of Nuclear Safety (SONS) - in acquiring the advanced thermal hydraulic codes. We also acknowledge the support of the Ministry of Industry and Trade of the Czech Republic within the national programs and grants focused on increase of the nuclear safety and the level of knowledge in branch of thermal hydraulics.

ABBREVIATIONS, GREEK LETTERS

BE	best-estimate
CL	cold leg
D	diameter
DC	downcomer
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedures
HA	hydroaccumulator
HL	hot leg
HPIS	High Pressure Injection System
HPSI	high pressure safety injection
ID	inner diameter
LOCA	loss-of-coolant accident
LOOP	loss-of-offsite power
LPIS	Low Pressure Injection System
LPSI	low pressure safety injection
MBLOCA	medium-break LOCA
N/A	not applicable
EOP	emergency operating procedures
PCT	peak clad temperature
PRZ	pressurizer
RCP	reactor coolant pump
SBLOCA	small-break LOCA
SCRAM	reactor trip (“safety control rod ax man”)
SG	steam generator
SIT	safety injection tank
UP	upper plenum
VVER	Russian type of PWR (with horizontal SGs)

1. INTRODUCTION

The test used in this report for assessment of RELAP5/MOD3.3 computer code was carried out in frame of the IMPAM-VVER project. The project was focused on different problems encountered during the development of EOPs for VVER reactors. The participants of the project performed both pre- and post-test analyses of the test with computer codes CATHARE, ATHLET and RELAP5.

Objective the work presented in this report is assessment of RELAP5/MOD3.3 computer code against the PMK test T3.1 performed in frame of the IMPAM project. The test is a large-break LOCA with 30% break starting from shutdown conditions with nitrogen in PRZ.

The objective of our assessment work was at one side verify RELAP5 capability to predict overall system behavior in LOCA conditions, which is a usual objective. And at the other side to test the RELAP5 capability to simulate system behaviour starting from shutdown conditions and to predict nitrogen transport in primary system, which are less usual tasks for a system TH computer codes.

2. DESCRIPTION OF THE PMK FACILITY

The PMK-2 facility [3] is a scaled down model of the VVER-440/213 and it was primarily designed for investigating small-break loss of coolant accidents (SBLOCA) and transient processes of this type of NPP. The specific features of VVER-440/213 are as follows: 6-loop primary circuit, horizontal steam generators, loop seal in hot and cold legs, safety injection tank (SIT) set-point pressure higher than secondary pressure (nowadays modified at majority of VVER-440/213), the coolant from SITs directly injected to the upper plenum and downcomer. As a consequence of the differences the transient behavior of such a reactor system should be different from the usual PWR system behavior.

The volume and power scaling of PMK facility are 1:2070. Transients can be started from nominal operating conditions. The ratio of elevations is 1:1 except for the lower plenum and pressurizer. The six loops of the plant are modeled by a single active loop. In the secondary side of the steam generator the steam/water volume ratio is maintained. The coolant is water under the same operating conditions as in the nuclear power plant.

The core model consists of 19 electrically heated rods, with uniform power distribution. Core length, elevation and flow area are the same as in the Paks NPP.

In the modeling of the steam generator primary side, the tube diameter, length and number were determined by the requirement of keeping the 1:2070 ratio of the product of the overall heat transfer coefficient and the equivalent heat transfer area. The elevations of tube rows and the axial surface distribution of tubes are the same as in the reference system. On the secondary side the water level and the steam to water volume ratios are kept. The temperature and pressure are the same as in the NPP. The horizontal design of the VVER steam generator affects the primary circuit behavior during a small break LOCA in quite a different way to the usual vertical steam generators.

Cold and hot legs are volume scaled and care was taken to reproduce the correct elevations of the loop seals in both the cold and the hot legs. Cold and hot leg cross section areas if modeled according to volume scaling principles would have produced much too high pressure drops. Since, for practical reasons, length could not be maintained 1:1, relatively large cross sections were chosen for the PMK loop. On the one hand this results in smaller cold and hot leg frictional pressure drops than in the NPP, on the other hand, however, it improves the relatively high surface to volume ratio of the PMK pipework. As to the former effect, the small frictional pressure drop of the PMK cold and hot legs will have a negligible effect on small-break processes. However, the pressure drop is increased using orifices around the loop.

For the pressurizer the volume scaling, the water to steam volume ratio and the elevation of the water level is kept. For practical reasons the diameter and length ratios cannot be realized. The pressurizer is connected to the same point of the hot leg as in the reference system. Electrical heaters are installed in the model and the provision of the spray cooling is similar to that of Paks NPP.

For the hydroaccumulators, the volume scaling and elevation is kept. They are connected to the downcomer and upper plenum similar to those of the reference system. The four hydroaccumulators of the VVER-440/213 are modeled by 2 SIT vessels.

The HPIS and LPIS systems are modeled by controlling the coolant flow rate in the lines by control valves. The flow rates measured during the start-up period of the Paks NPP are used to control the valves.

The main circulating pump of the PMK serves to produce the nominal operating conditions corresponding to that of the NPP prior to break initiation as well as to simulate the flow coast-down following pump trip early in the transient. For this reason the pump is accommodated in a by-pass line. Flow coast-down is modeled by closing a control valve in an appropriate manner and if flow rate is reduced to that of natural circulation, the valve in the by-passed cold leg part is opened while the pump line is simultaneously closed.

PMK Test Facility Characteristics:

Reference NPP:
Paks Nuclear Power Plant with VVER-440/213 (6 loops)
1375 MWt - hexagonal fuel arrangement

General Scaling factor:
Power, volumes: 1/2070, loops 1/345
Elevations: 1/1

Primary coolant system (1 loop representation):
- Pressure: 12.3 MPa (nominal), 16 MPa (max.)
- Nominal core inlet temperature: 540K
- Nominal core power: 664 kW
- Nominal flow rate: 4.5 kg/s

Special features:
- 19 heater rods, uniform axial and radial power distribution
- 2.5 m heated length
- External downcomer
- Pump is accommodated in by-pass line
    - -- flow rate 0 to nominal value
    - -- NPP pump coast down simulation
- Loop piping: 46 mm ID

Secondary system:
- Pressure: 4.6 MPa, feed water temperature: 496 K
- Nominal steam and feed water mass flow: 0.36 kg/s

Special features:
- Horizontal steam generator
- Controlled heat removal system

Safety injection systems:
- High Pressure Injection System (HPIS) and Low Pressure Injection System (LPIS)
- Safety Injection Tanks (SITs)
- Emergency feed water

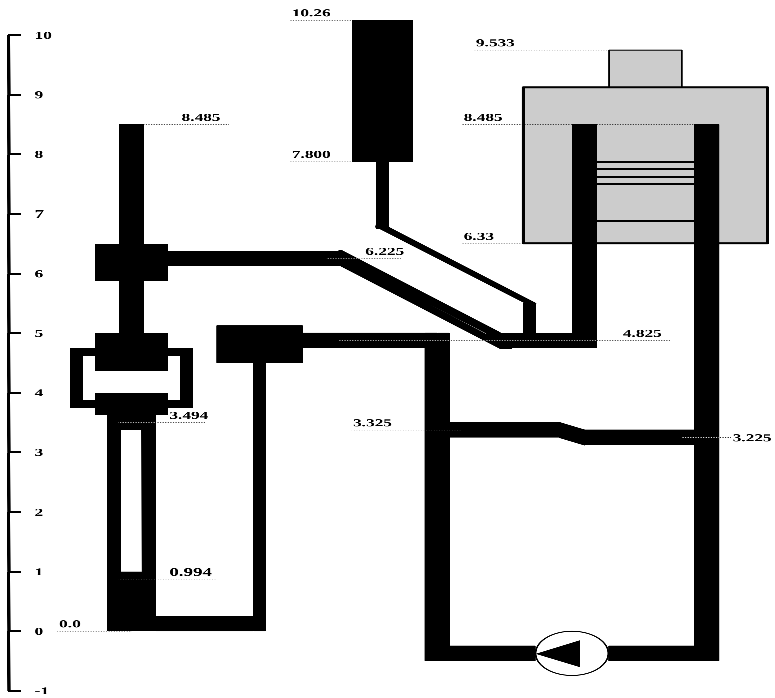

Figure 1 Elevation diagram of the PMK facility

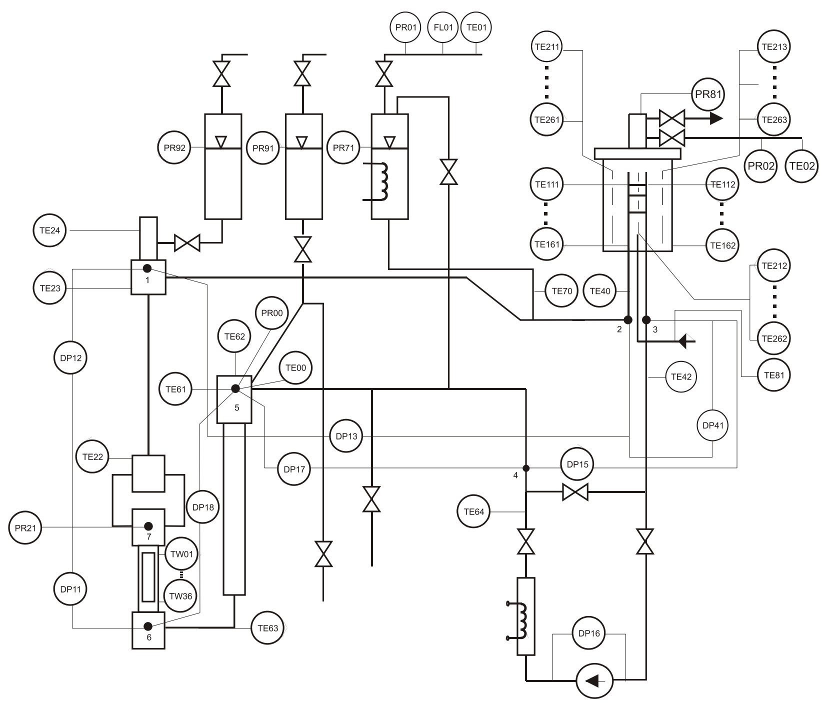

Figure 2   PMK measurement locations #1 – pressure and temperature

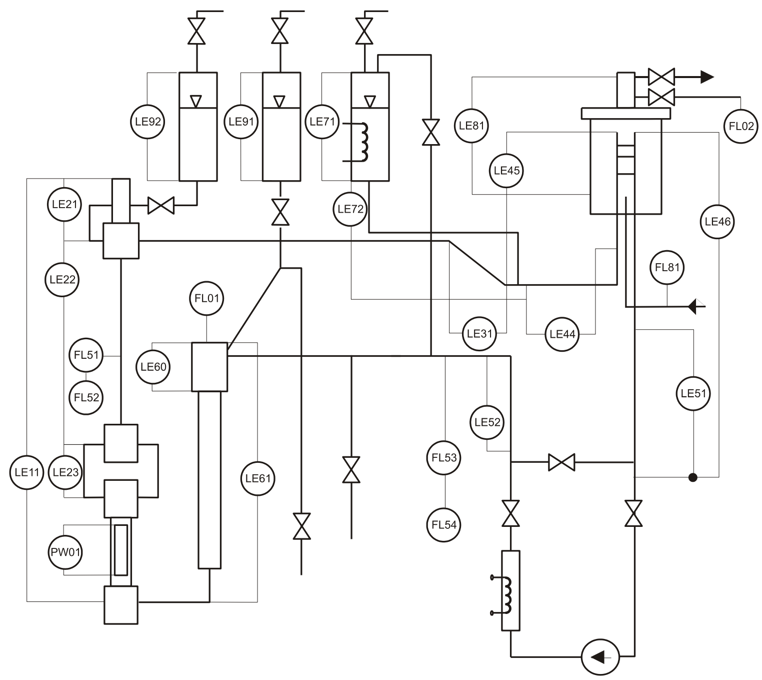

Figure 3 PMK measurement locations #2 – levels and flow

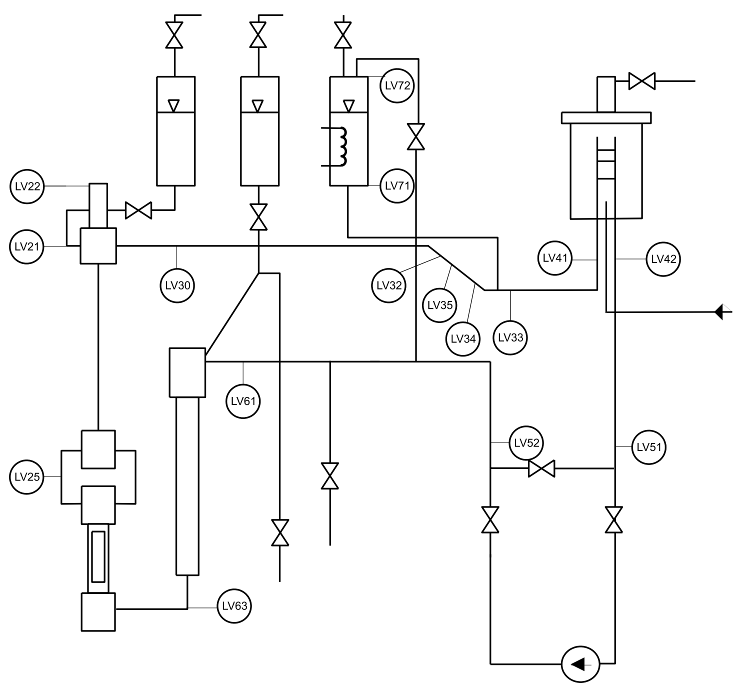

Figure 4 PMK measurement locations #3 - void probes

3. UJV INPUT MODEL OF PMK FACILITY

The RELAP5 input deck of PMK used for the post-test analyses is a modified version of our older deck [1, 2] used for modeling of PMK-NVH in early 90-ties, when we analyzed the IAEA organized SPE tests.

The modeling approach used in development of PMK model is similar to the approach applied in development of input models of Czech NPPs with VVER reactors. Generally, geometry and nodalization of primary circuit except of SG is very similar to those of standard PWR. There are only 3 major specific features of VVER-440/213, that should be reflected in nodalization – horizontal SG (reflected in multi-layer nodalization of SG tubing), loop seal in hot leg (reflected in detailed nodalization of HL), and direct HA/LPIS injection to reactor (we don't expect any multi-dimensional effects in small-scale facility like PMK, so simple 1-D modeling of reactor vessel was used).

Our RELAP5 input model of PMK experimental facility consists of:

- 134 volumes
- 144 junctions
- 126 heat structures (with 553 mesh points)
- 62 control variables
- 68 trips

Nodalization scheme can be seen in **Figure 5**. Comparing to our „old“ model of PMK-NVH [1, 2], the major modifications of PMK nodalization implemented during work on this report, are as follows: more exact modeling of lower plenum, remodeled core outlet and upper plenum, and modified nodalization of PRZ and PRZ surge line (incl. location of PRZ surge line connection to the hot leg).

Listing of the PMK input deck for RELAP5 developed in the NRI Rez can be found in the NUREG/IA-0229 report [12].

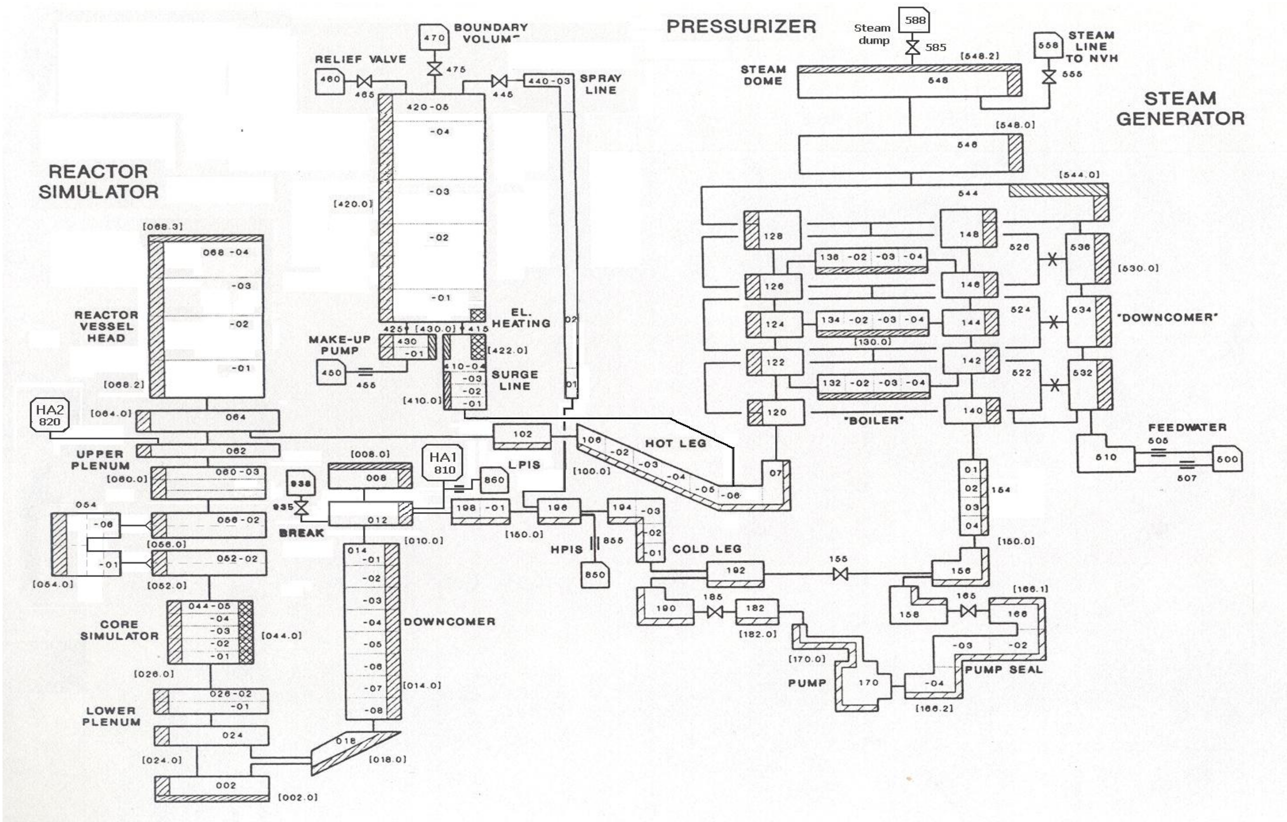

Figure 5 Nodalization scheme of PMK for RELAP5

4. POST-TEST ANALYSIS OF T3.1 EXPERIMENT

4.1 Experiment description

The Test 3.1 (T3.1-CD) [5, 6] experiment simulates a large break LOCA during the cool-down state of the plant. According to the original VVER-440 cool-down procedures neither passive, nor active ECCS could be automatically activated below 2.5 MPa. This may lead to core heat-up in case of a larger LOCA. The test should help to answer the question, whether a single LPSI train started by the operator – as now foreseen in the Paks NPP – can effectively prevent core heat-up. The break size is about 30%.

The test is defined by the following steps:

- Initial conditions correspond to the plant state during cool-down with nitrogen in PRZ,
- The PMK core power relevant for the shutdown state is 6 kW, however the core power in steady state was increased to 21 kW in order to compensate for heat losses of the facility - power “correction” 21⇒6 kW was performed at the beginning of modelled accident,
- The experiment is started by opening the 30% break in the cold leg with simultaneous initiation of
- secondary side isolation,
- switch off of pressurizer heaters,
- pump coast down,
- LPIS starts at $p < 0,7$ MPa and time > 1800 s, or Twall > 450 °C,
- Test is terminated at Twall > 500 °C.

The main objective of the test is to get experimental evidence about the effectiveness of the plant procedure to prevent core heat-up. As a consequence of the large break size the pressurizer is emptied in a few seconds and N2 gas enters the primary circuit. The special void probes installed in PMK make it possible to track the N2 propagation along the circuit.

The initial conditions of the test are nearly the same as the nominal operating parameters of the plant considering the scaling ratio. In **Table 1** below these conditions are given. Specified data are compared with measured data and the steady-state calculation results.

Table 1   Initial conditions of test T3.1

	Unit	Specified	Measured	Calculation NRI
Primary system pressure (PR21)	MPa	2.6	2.82	2.87
Primary loop flow (FL53)	kg/s	4.5	4.54	4.54
Core inlet temperature (TE63)	K	150.0	152.5	152.8
Core power (PW01)	kW	21.0	21.0	21.00
Coolant level in PRZ (LE71)	m	9.27	8.98	8.98
Pressure (PR81)	MPa	1.0	1.32	1.319
Feedwater flow (FL81)	kg/s	0.31	1.0	1.1
Feedwater temperature (TE81)	°C	146.8	148.0	146.8
Coolant level in SG (LE81)	m	completely filled		

The boundary conditions of the test and at the calculations are nearly the same as specified, except for the pump coast-down time, which had to be shortened in order to save the pump from running too long in two-phase conditions. The LPIS flow rate was specified for 0.7 MPa. The increased flow rate is the consequence of the fact that the injection took place at nearly atmospheric conditions. The boundary conditions are listed in **Table 2** below.

Table 2   Boundary conditions of test T3.1

	Unit	Specified	Measured	Calculation NRI
Break orifice diameter	mm	6.0	6.0	6.0
Break opens at	s	0.0	1.0	0.0
Core power linearly reduced to 6 kW	s	0.0	1.0 *[1]	0.0
Isolation of feedwater and steam lines *[2]	s	3.0	0.0	0.0
Pump coast-down initiated at	s	0.0	2.0 *[1]	2.0
Pump coast-down end	s	150	86	150
LPIS flow rate (1 system assumed)	kg/s	0.042	0.070	0.070
LPIS injection starts if clad temperature or time	°C s	450 1800	N/A 1777	N/A 1777

Notes: *[1] There is a slight inconsistency in the test results – start of core power reduction is reported either at 1.0 s or at 2.0 s, start of pump coastdown is reported either at 2.0 or at 4.0 s.

*[2] To get acceptable prediction of secondary pressure (in condition of SG full of water), we modelled for T3.1 not simple steam line isolation, but pressure boundary condition at steam line end.

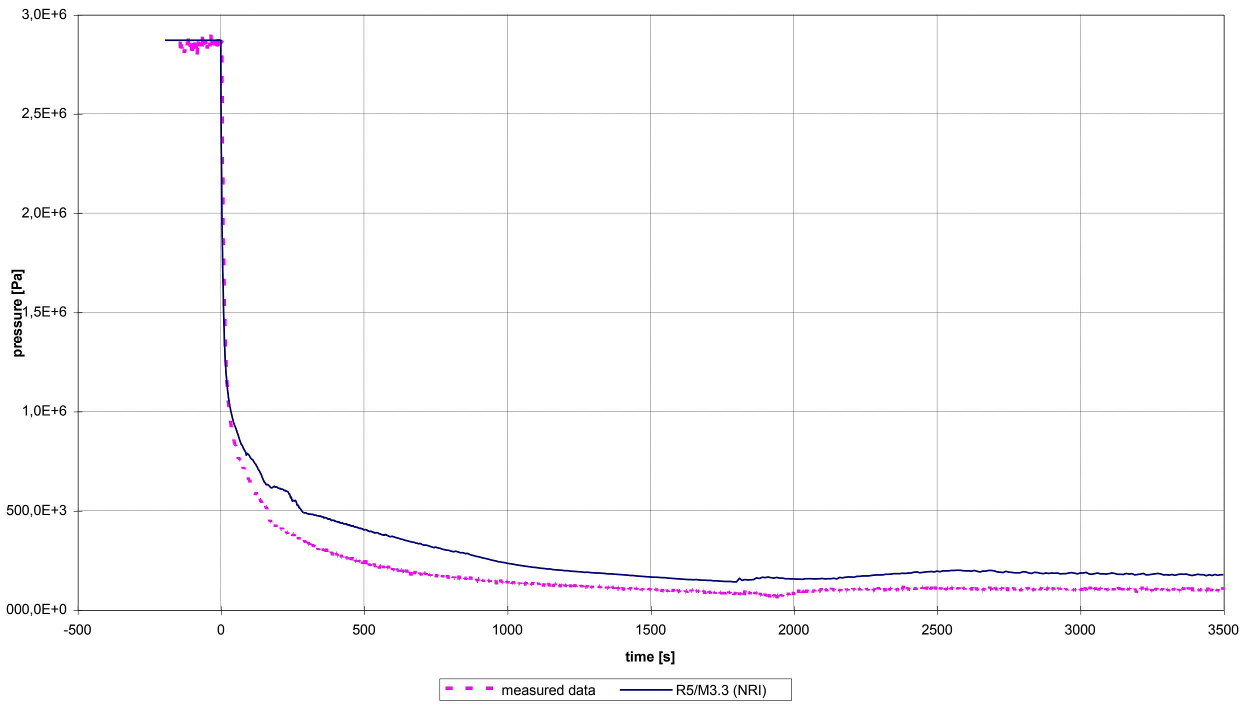

Figure 6 Primary pressure (T3.1)

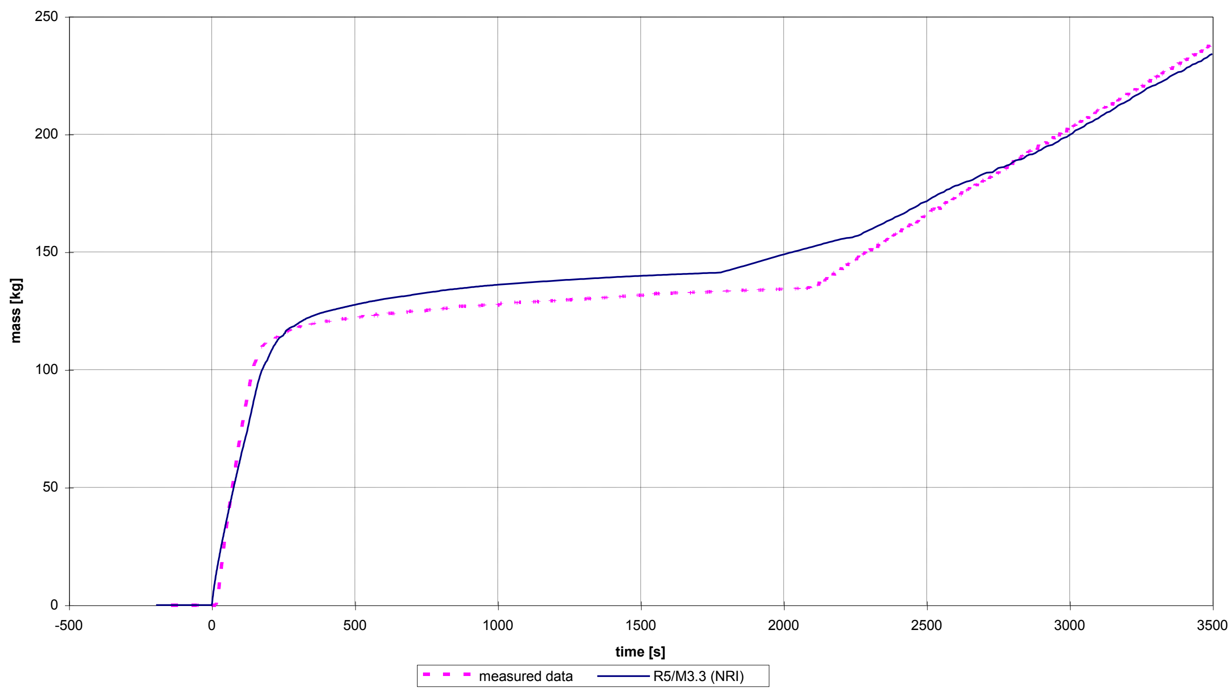

Figure 7 Integrated break mass flow rate (T3.1)

4.2 Results of calculation

The main events of the Test 3.1 and the RELAP5 calculations are listed in **Table 3** below:

Table 3 Timing of main events of test T3.1

Event	Timing [s] Measured	Timing [s] Calculation NRI	Comment
Break opens	0.0	0.0	Break D6 mm (30%)
Core power linearly reduced	(0.0)	0.0	
Isolation of feedwater and steam lines	0.0	0.0	
Pump coast-down initiated	2.0	2.0	
Pressurizer empty	8	8	
Pump coast-down ended	86	150	
Hot collector empty	88	90	
Hot leg loop seal cleared	150	192	
Cold leg loop seal cleared (reactor side)	172	180	
1st core overheat	-	220-250	Maximal PCT in calc. was 186 °C
2nd core overheat starts	1281	1360	
LPIS starts	1777	1777	Flow rate 0.07 kg/s
Vessel level minimum during major core overheating	1763 (1.40 m)	1780 (1.08 m)	
Fuel rod temperature maximum	1806 (405 °C)	1800 (417 °C)	
2nd core overheat end = end of reflood	1950	1920	
Transient end	3543	3500	

The defined LOCA scenario starts with opening of the break valve D6 mm (30% of cold leg flow area) at reactor downcomer top. At the same time the reactor power reduction, trip of RCP, and isolation of SG secondary side occurs.

The pump coast-down time had to be in the actual test reduced to 86 s in contrast to the specified 150 s, in order to save the pump from consequences of cavitations. Due to the large break size and consequently strongly two-phase character of the process, this has limitted effect on the overall system behavior.

Outflow of primary coolant through large break leads to fast decrease of primary pressure. Because of shutdown initial conditions, there is neither hydroaccumulators injection start after pressure drop under 6.0 MPa nor automatic actuation of active safety injection systems.

As there is no compensation of coolant leak through the break, the primary inventory is depleted pretty fast – the reactor collapsed level drops under 3.5 m (approximate elevation of core top) both in experiment and calculation before 200 s. In the calculation, there is even an early temporary core overheat in this phase – in time 220-250 s with clad temperature maximum 186 °C.

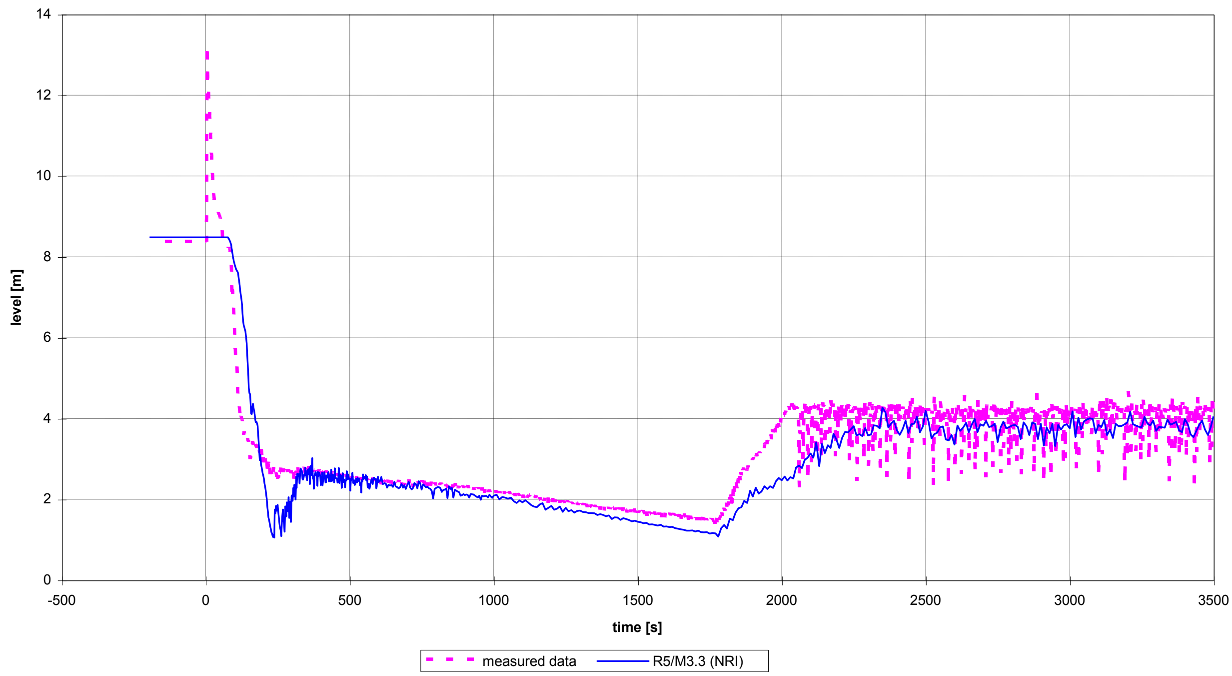

Figure 8 Collapsed level in reactor (T3.1)

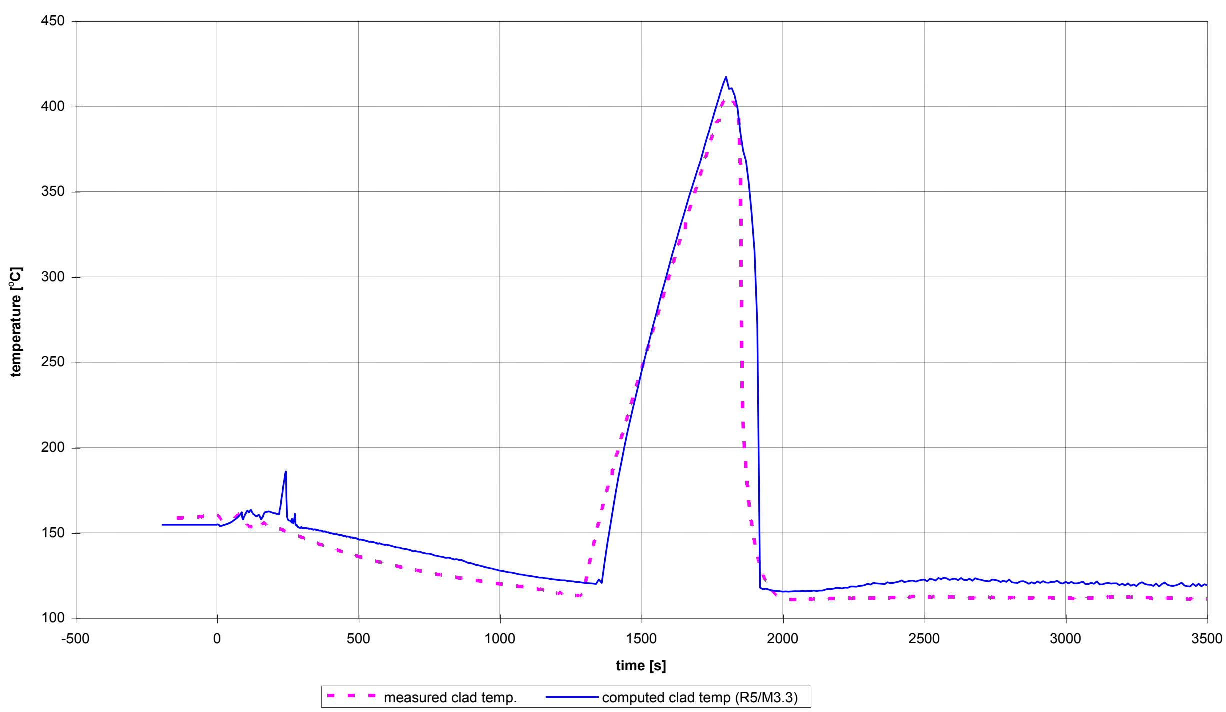

Figure 9 Cladding temperature (T3.1)

The ultimate core uncovery and fuel heat-up starts around 1300 s. Increase of clad temperature is fast and so the operator starts at 1777 s the LPIS injection, which limits the maximal clad temperature in calculation to 417 °C at 1800 s (measured values were similar).

The core reflood was finished at about 1920 s and later on a stable core cooling was ensured up to the end of test at 3500 s.

4.3 Comparison of results

The most important comparison plots of the measured data and the post-test UJV calculations are shown in **Figure 6** - **Figure 9**. Complete set of T3.1 comparison plots can be found in Appendix III.

Most calculated parameters are in very good agreement with measured data, especially the most important system parameters like primary and sec. pressure, coolant and clad temperature etc.

The initial primary pressure drop is well predicted. In later phases of the accident the calculated course is slightly overpredicted against the measured pressure course.

The integrated break flow is slightly overpredicted in the first 200 s of the transient and on the contrary, partially underpredicted in interval 200-2100 s. After start of LPIS and refilling of the system, predicted mass outflow is again higher than the measured one. One can conclude, that calculated break flow is overpredicted in single phase liquid and two phase outflow phase, while underpredicted in single phase steam outflow phase of the accident.

As for the cladding temperature, the major heat-up period was very well predicted, both in timing and in maximal PCT value (417 °C compared to measured 405 °C). In the calculation, there was even an early small core heat-up period in time 220-250 s with clad temperature maximum 186 °C, which was not measured in the test.

Both the experiment and calculations show that in this LBLOCA scenario the Accident Management represented by operators start of LPIS can effectively stop the core heat-up and cool down the system.

Further comments to results:

Correct prediction of core overheat was sensitive on the used break model and discharge coefficients – for the final calculation we used coefs 1.1 and H-Fauske choked flow model.

Results were also very sensitive on initial coolant temperature (connected here strongly with FW temperature), initial PRZ temperature (not properly specified in test data) etc.

A very surprising and positive finding of the analysis was a minimal mass error, although the calculated process (LBLOCA) was very dynamic and further complicated by presence of non-condensable gas in primary system.

5. ADDITIONAL CALCULATIONS AND ANALYSES

5.1 Analysis of NC Gas Behavior during the Test T3.1

The LOCA test T3.1 gives us a chance to assess capability of current version of RELAP5 computer code to compute presence of noncondensable gas(es) in the primary system. The follosing chaptes focus on this topic.

5.1.1 Analysis of NC Gas Behavior during the Test T3.1

Tracking of non-condensable gases in PMK test T3.1 was done with help of special void probes containing micro-thermocouple. There were installed 8 probes of traditional type (measuring void only) and 8 advanced probes with integrated thermocouple.

Advanced void probes with thermocouple enables to distinguish portion of subcooled gas from sub-cooled liquid, which can be quantified as non-condensable gas.

For faster orientation we place also here the figure with PMK void probes – see the **Figure 10** below with location of the 16 void probes. In the following figures, one can see comparison of measured and calculated voids in selected positions of primary circuit.

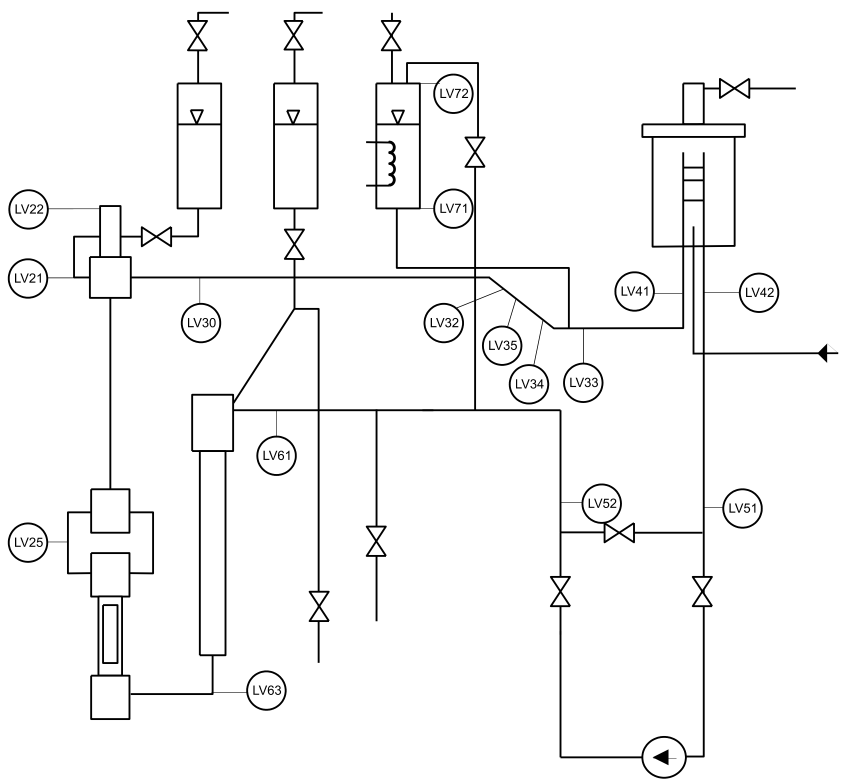

Figure 10 PMK measurement locations #3 - void probes

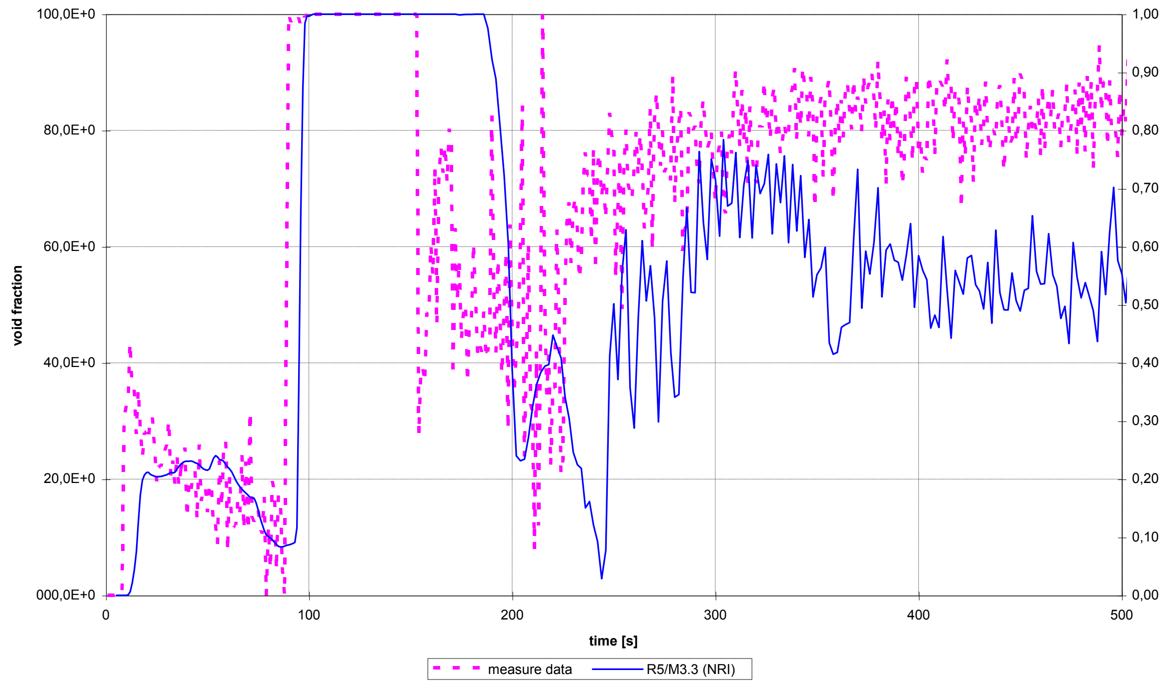

Figure 11 Void fraction in hot leg between PRZ connection and SG inlet (LV41)

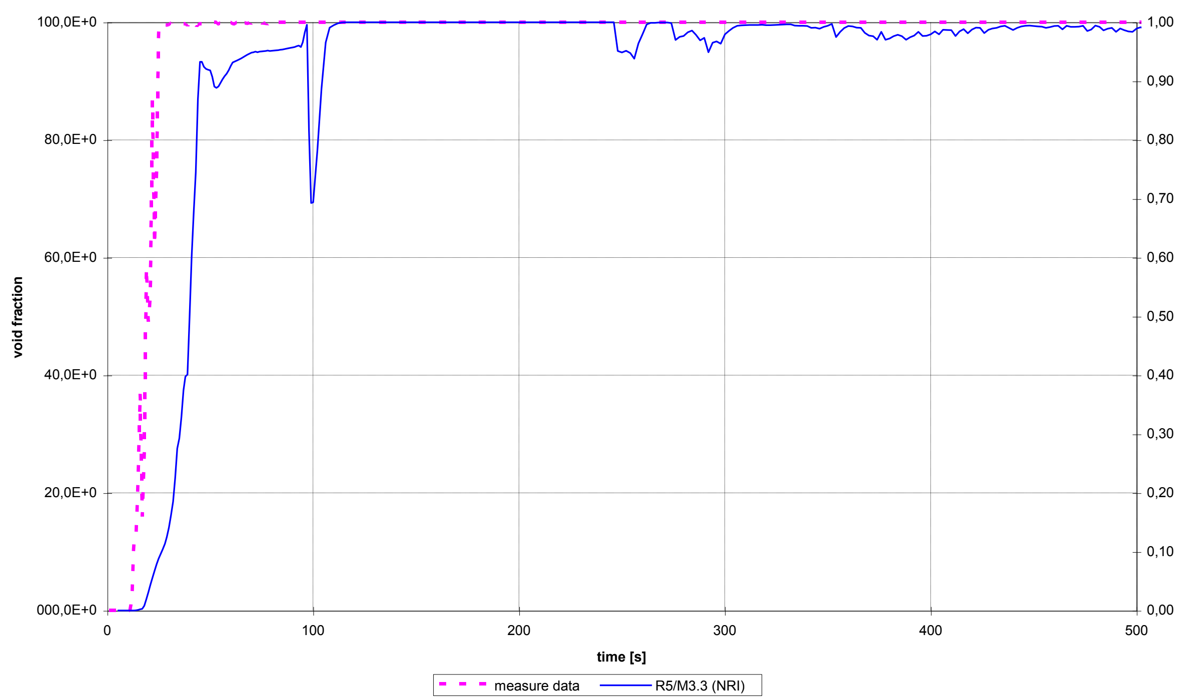

Figure 12 Void fraction in SG outlet to cold leg (LV42)

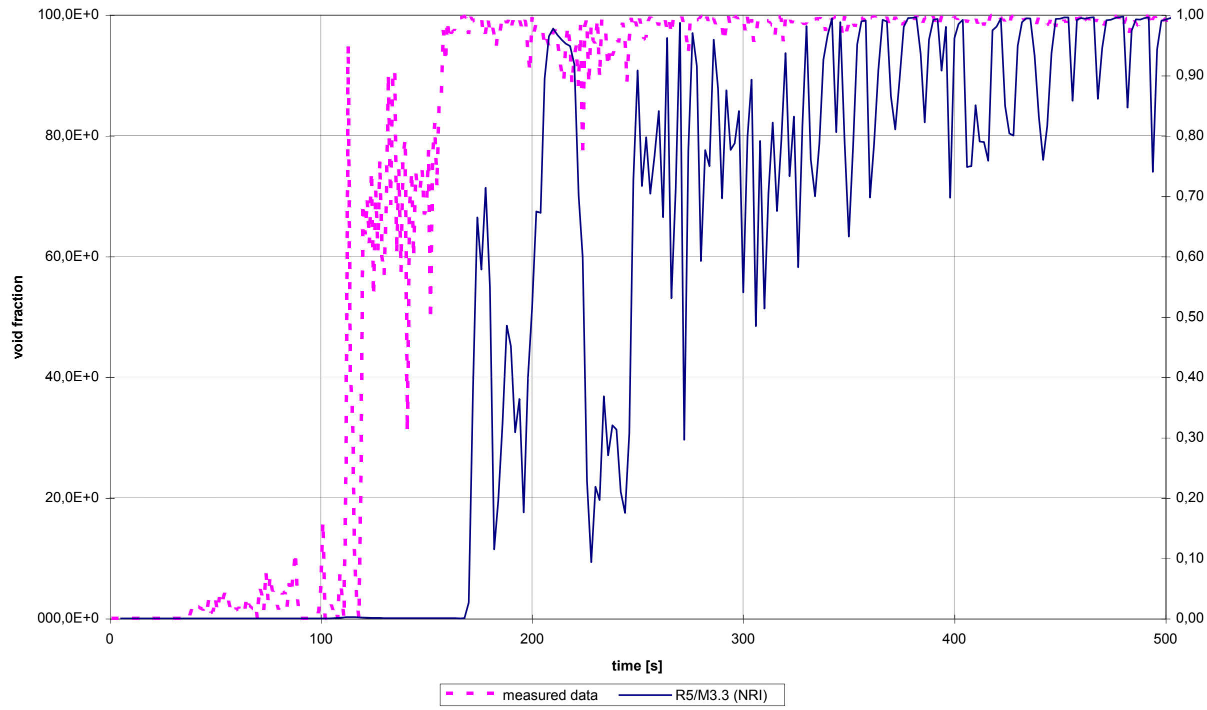

Figure 13 Void fraction in cold leg loop seal upward part (LV52)

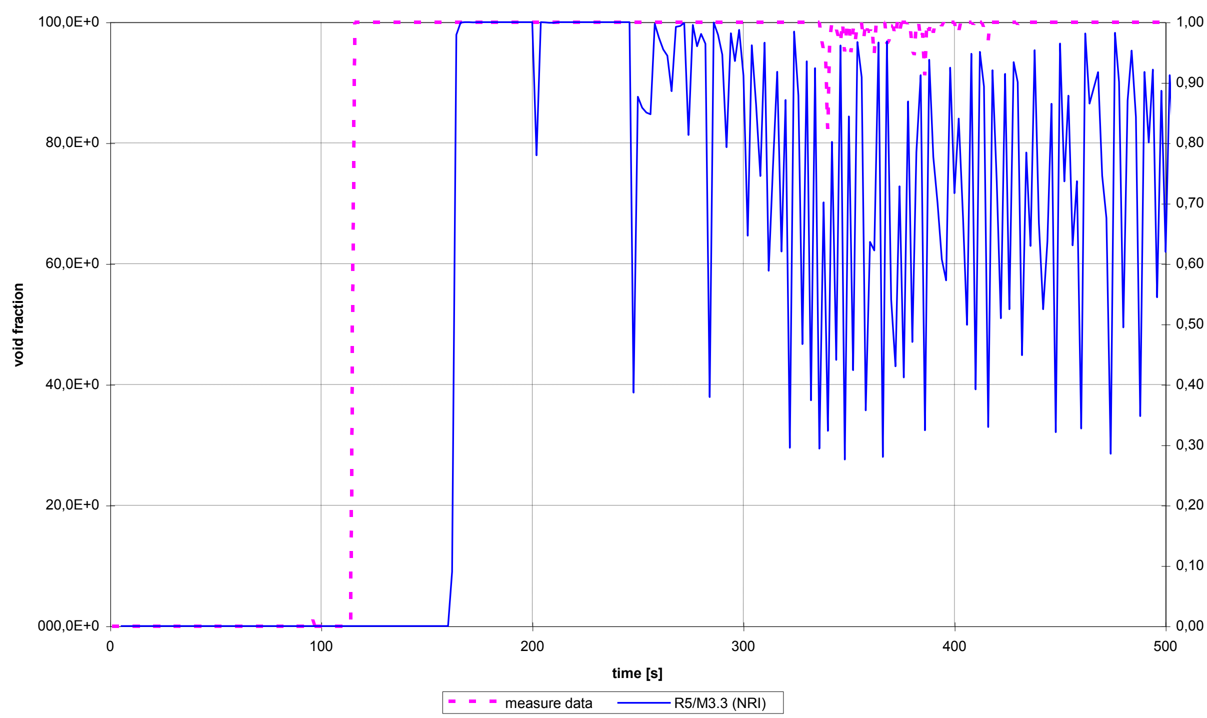

Figure 14 Void fraction at core outlet (LV25)

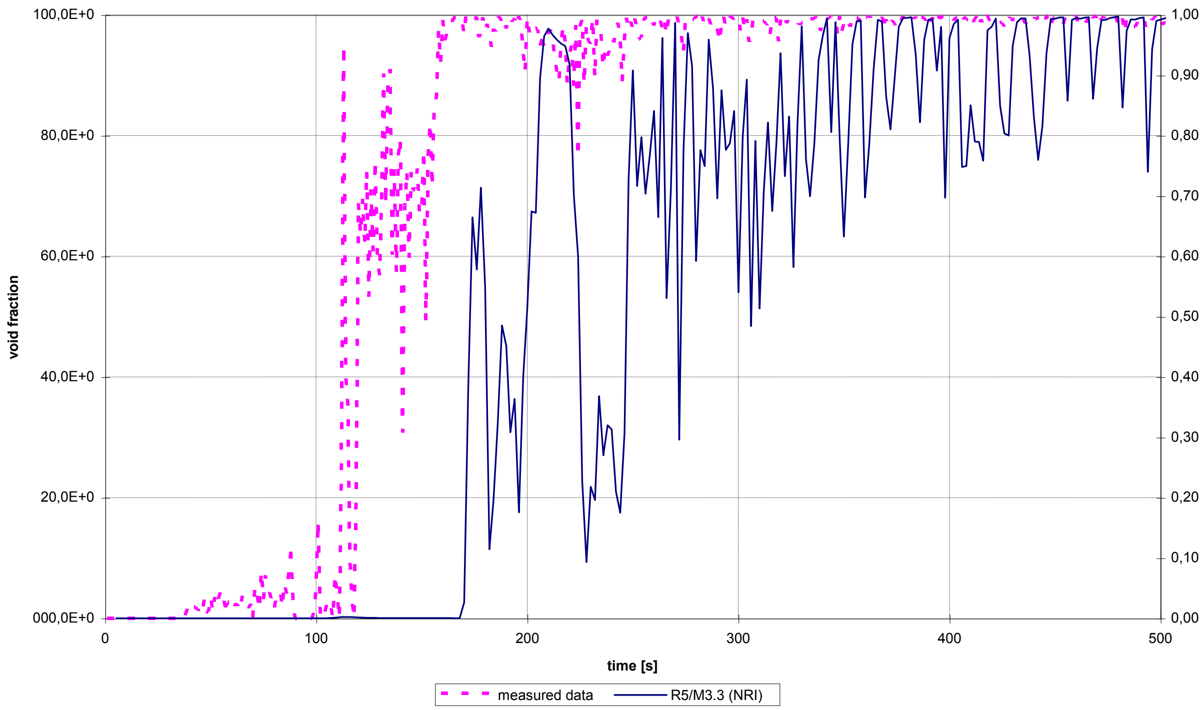

Figure 15 Void fraction in upper plenum by outlet nozzle (LV21)

Tracking of non-condensable gases in PMK by help of detecting of location of void and subcooling is well proved by calculation results, where we can work not only with both VOIDG, TEMPG and SATTEMP variables, but also directly quantify mass fraction of noncondensable gas in vapor phase by help of QUALA variable. See below some examples with NC gas tracking in hot leg.

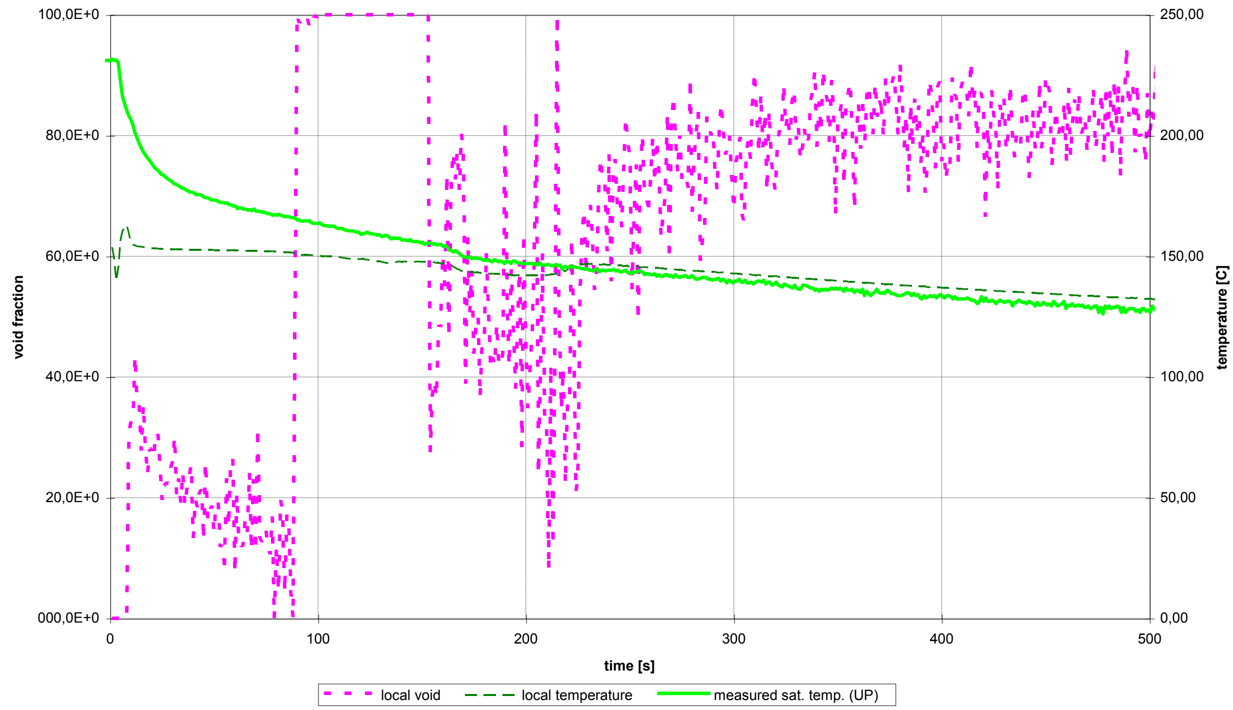

Figure 16 NC-gas tracking in hot leg in EXPERIMENT (LV41)

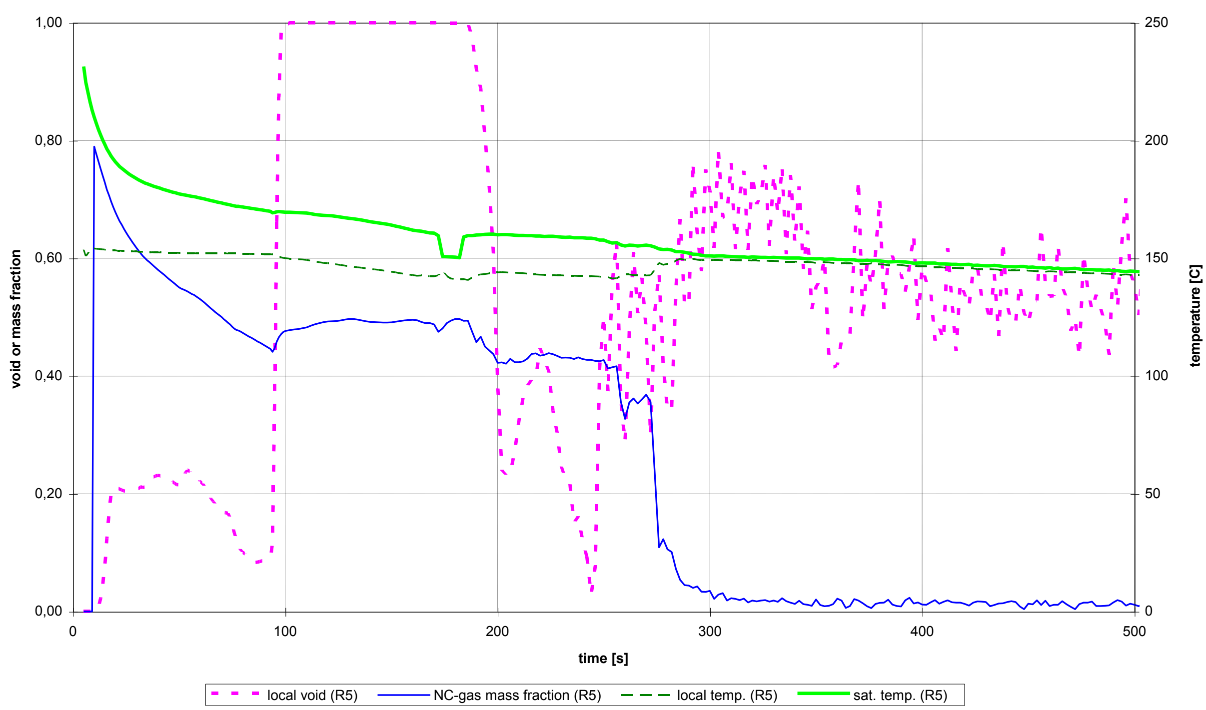

Figure 17 NC-gas tracking in hot leg in CALCULATION (LV41)

5.1.2 Checking NCG Balance along the Primary Circuit

As a next step we added into the PMK input deck a number of control variables for checking of NC gas balance along primary circuit, like the following:

$$MASS_{NCG,VOL} = VVOL \cdot VOIDG \cdot RHOG \cdot QUALA$$

$$MASS_{NCG,JUN} = \int_t \frac{VOIDGJ \cdot VELFGJ \cdot RHOGJ \cdot MFLOWJ \cdot QUALAJ}{VOIDFJ \cdot VELFJ \cdot RHOFJ + VOIDGJ \cdot VELFGJ \cdot RHOGJ} dt$$

(Note: In the latest version of RELAP5/MOD3.3/Patch4, the checking of NC balance and transport would be easier as there are new Minor Edit variables available.)

These variables enable us to watch the nitrogen transport from pressurizer to various parts of primary system and/or to the break:

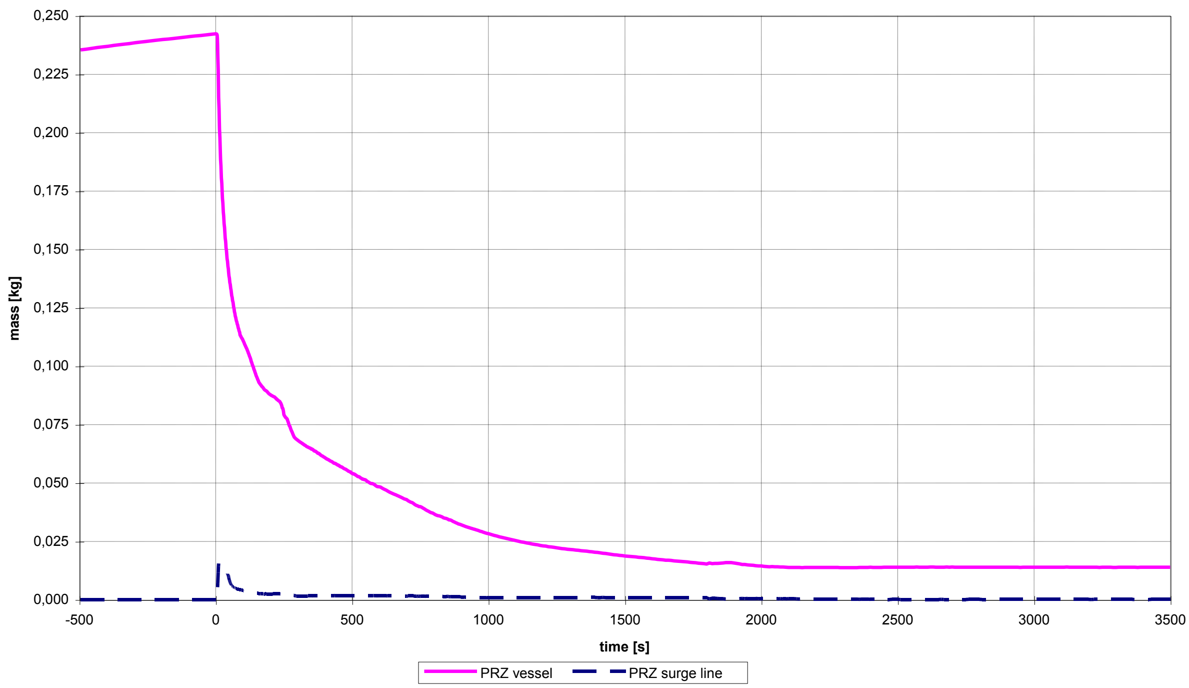

Figure 18 Mass of NC-gas in pressurizer

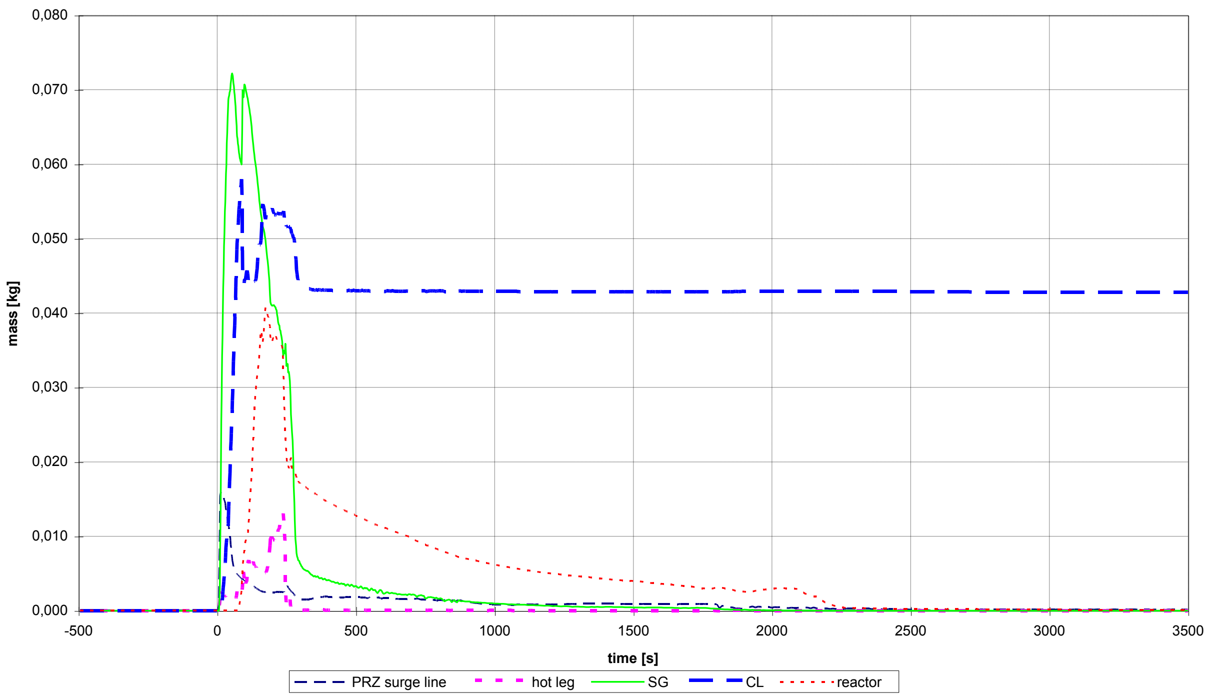

Figure 19 Mass of NC-gas in main parts of primary circuit

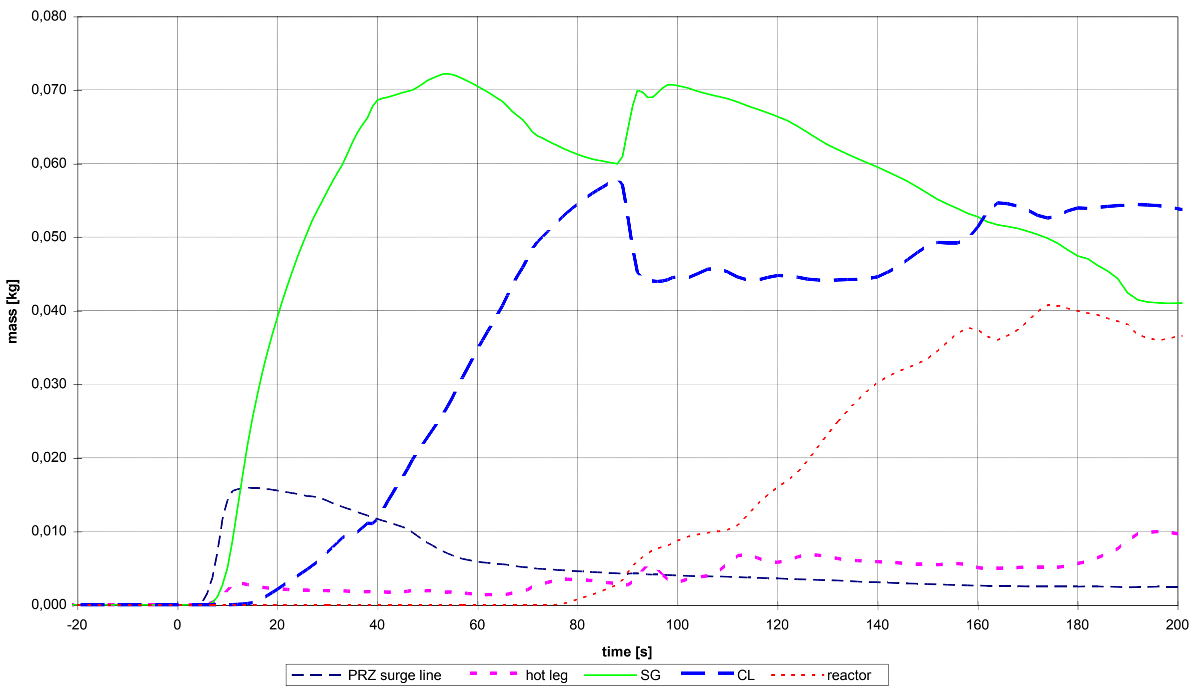

Figure 20 Mass of NC-gas in main parts of primary circuit – DETAIL

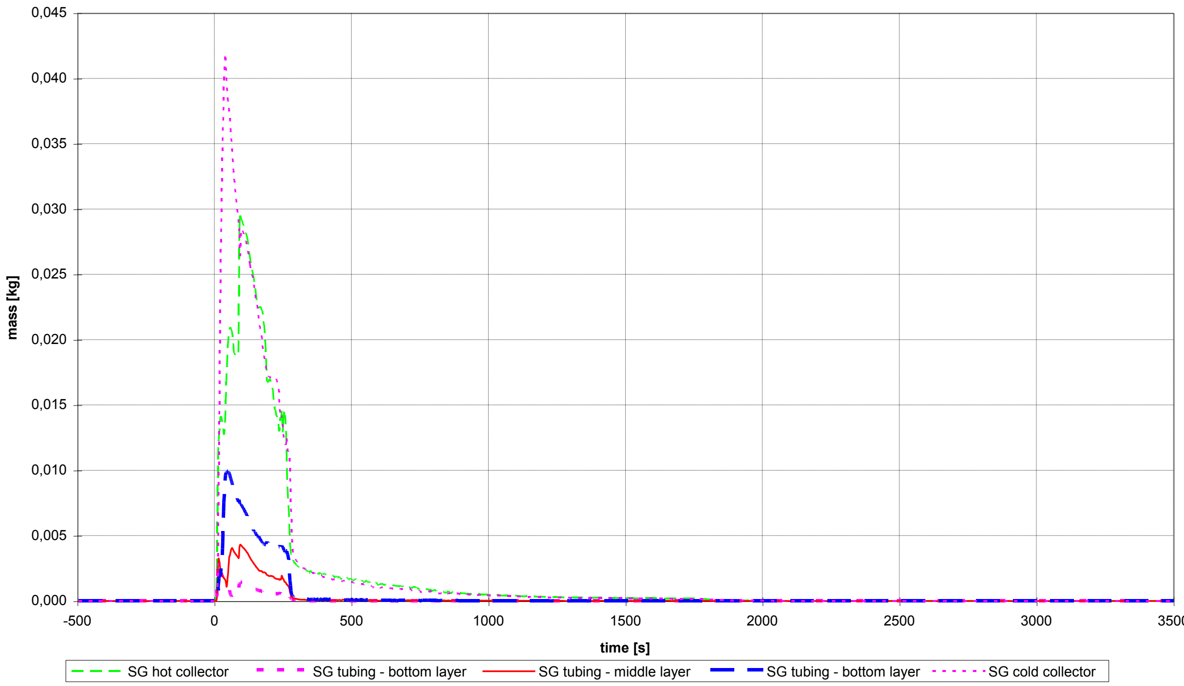

Figure 21 Mass of NC-gas in SG primary

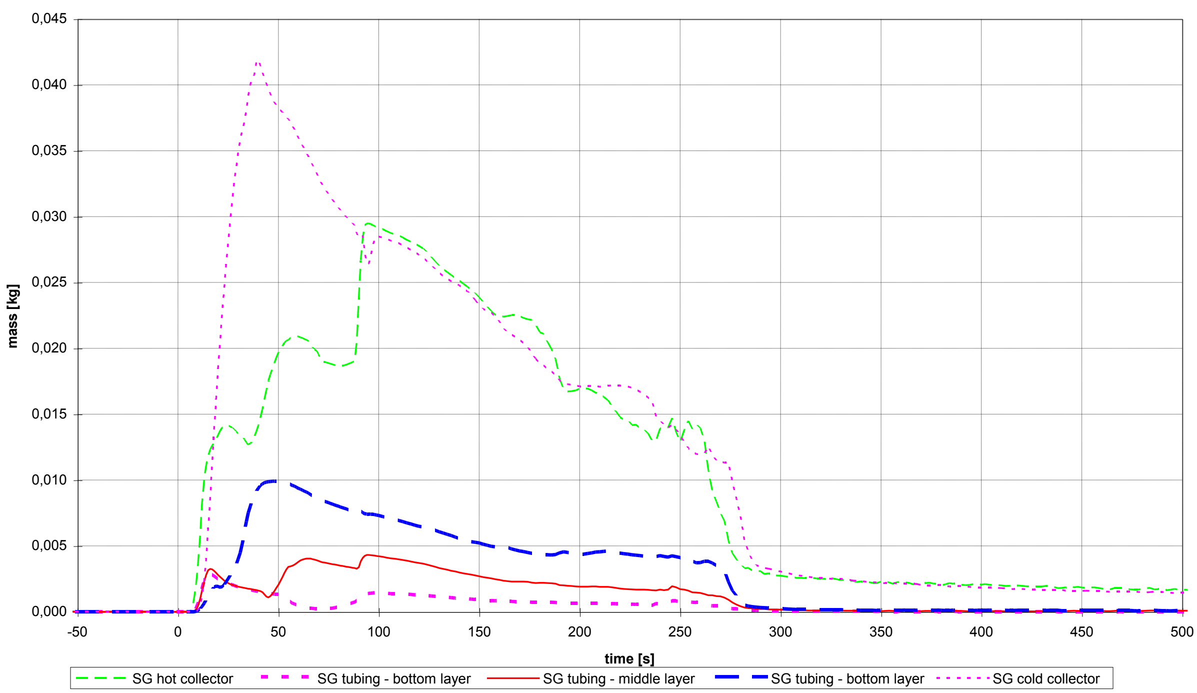

Figure 22 Mass of NC-gas in SG primary - DETAIL

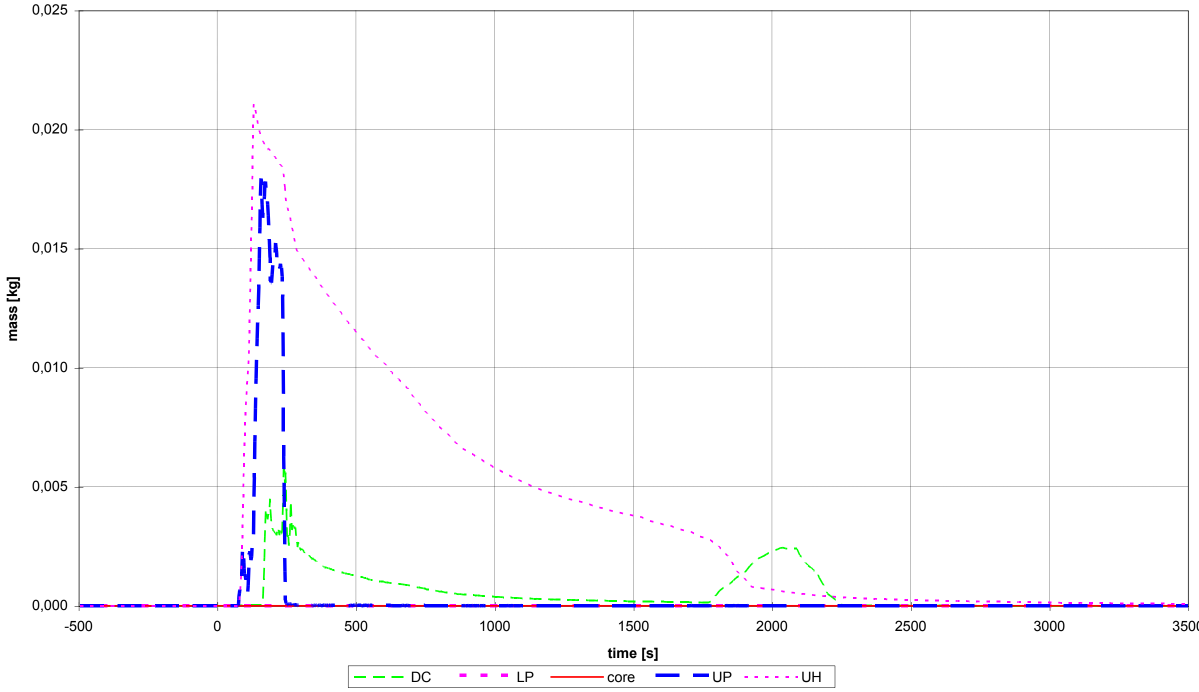

Figure 23 Mass of NC-gas in reactor

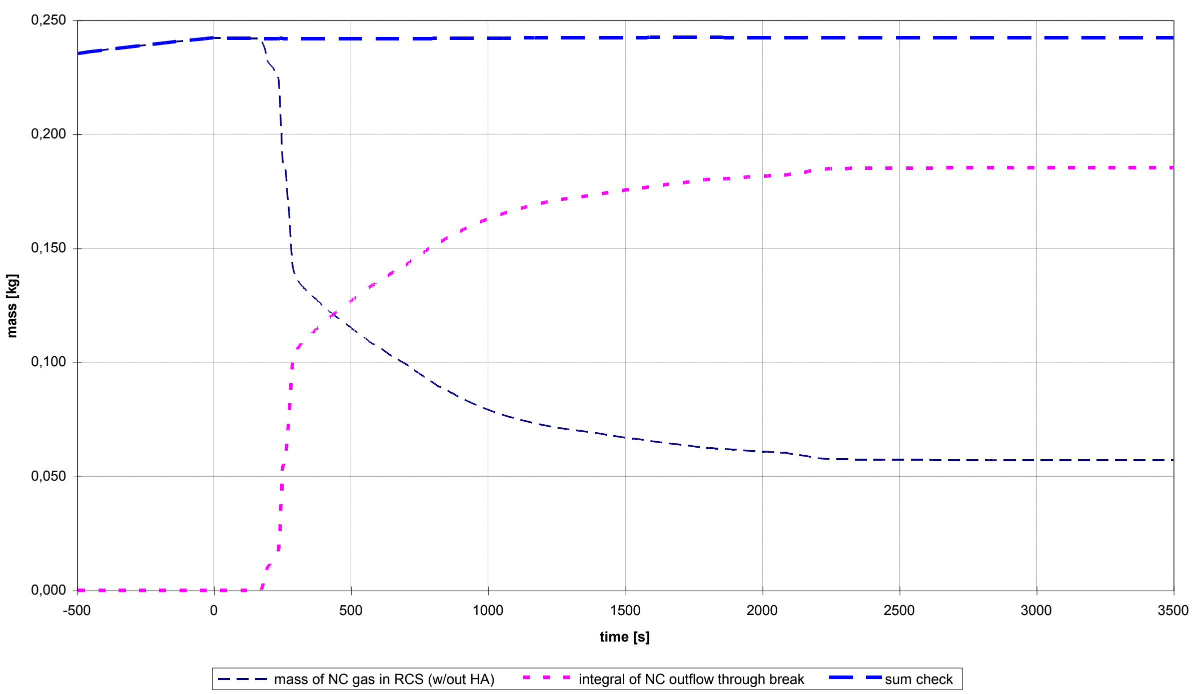

Figure 24 Balance of NC-gas mass in RCS

<u>Conclusions from additional analysis of NC-gas transport in Test T3.1:</u>

- ⇒ Very good qualitative and good quantitative (depending on location) prediction of void and nitrogen mass transport from PRZ to primary circuit and partially out through break.
- ⇒ Verification of non-condensable gases tracking method based on void probes with integrated micro- thermocouples.
- ⇒ No mass error in noncondensable balance in RELAP5/MOD3.3 calculation.

5.1.3 Influence of NCG on SG heat transfer

As for the influence of NC gas on heat transfer in SG (it was naturally not measured), we can compare only the SG temperatures (see the figures below and also the **Figure 21** and **Figure 22** above):

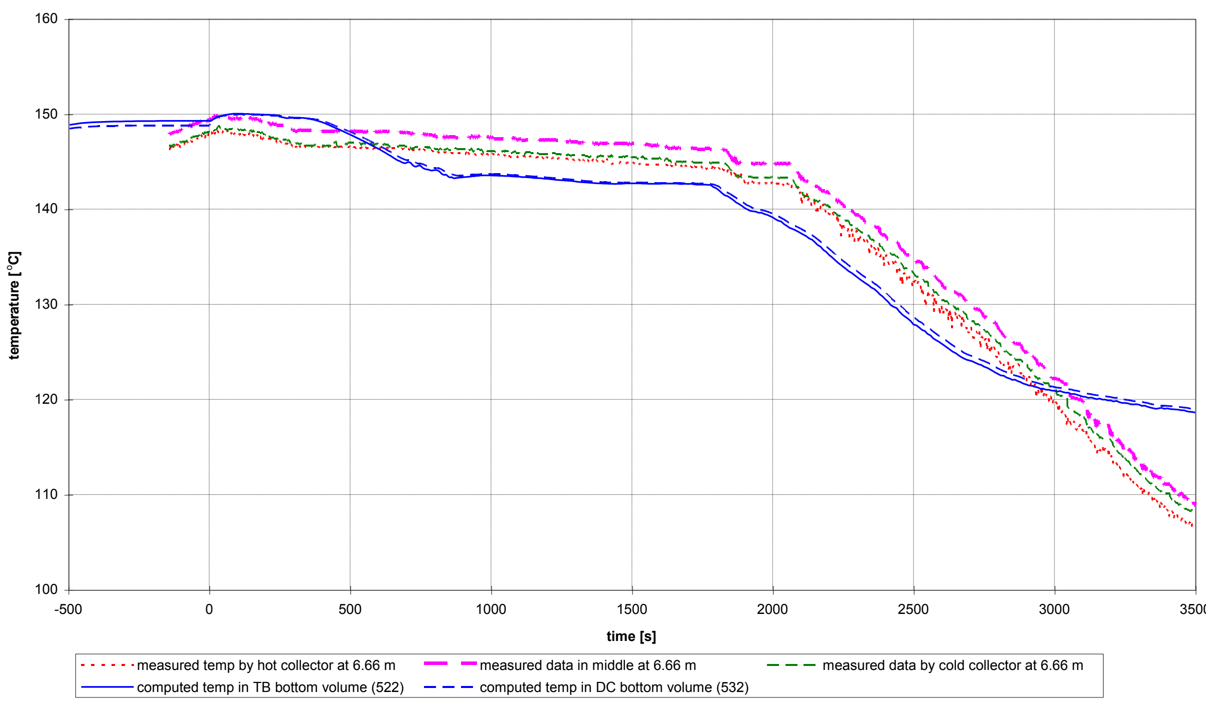

Figure 25 Temperature on SG secondary – bottom layer of TB

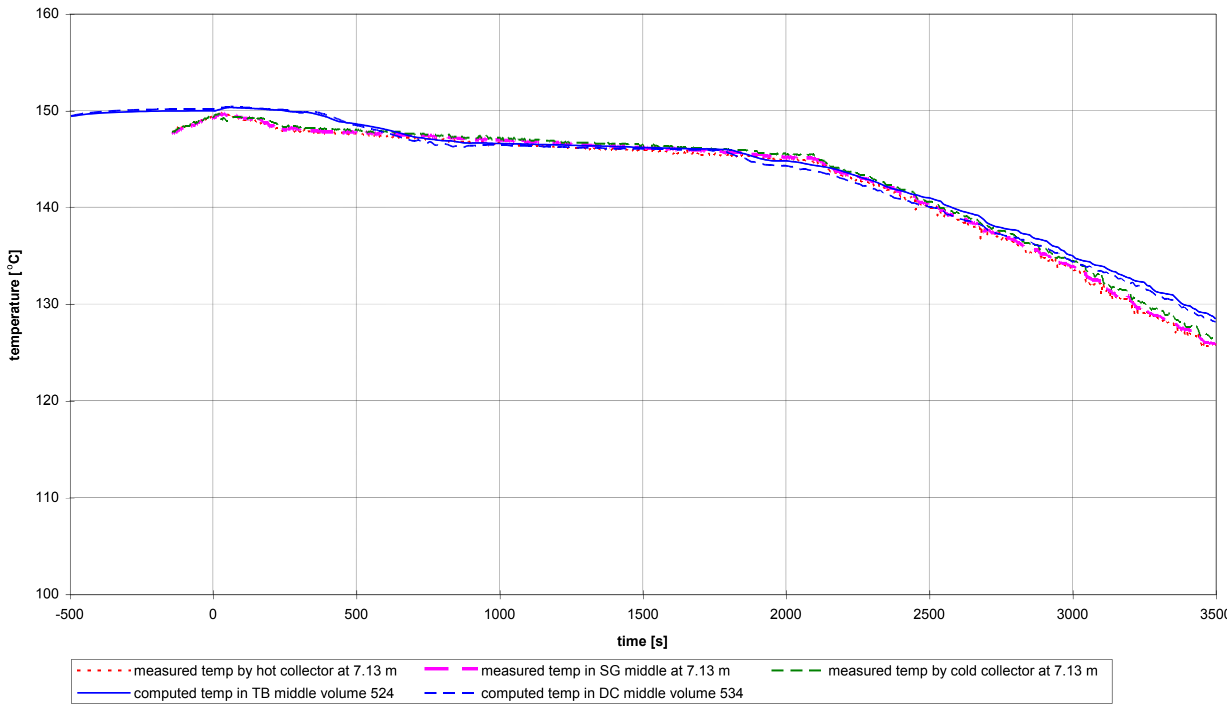

Figure 26 Temperature on SG secondary – middle layer of TB

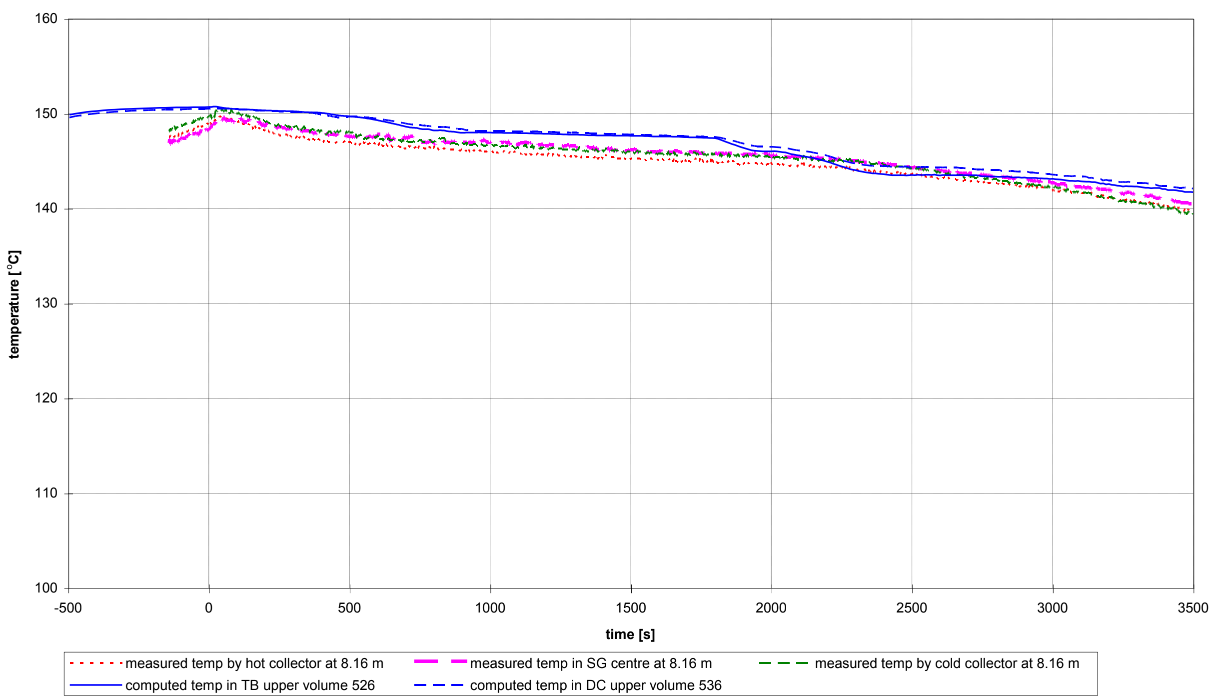

Figure 27 Temperature on SG secondary – upper layer of TB

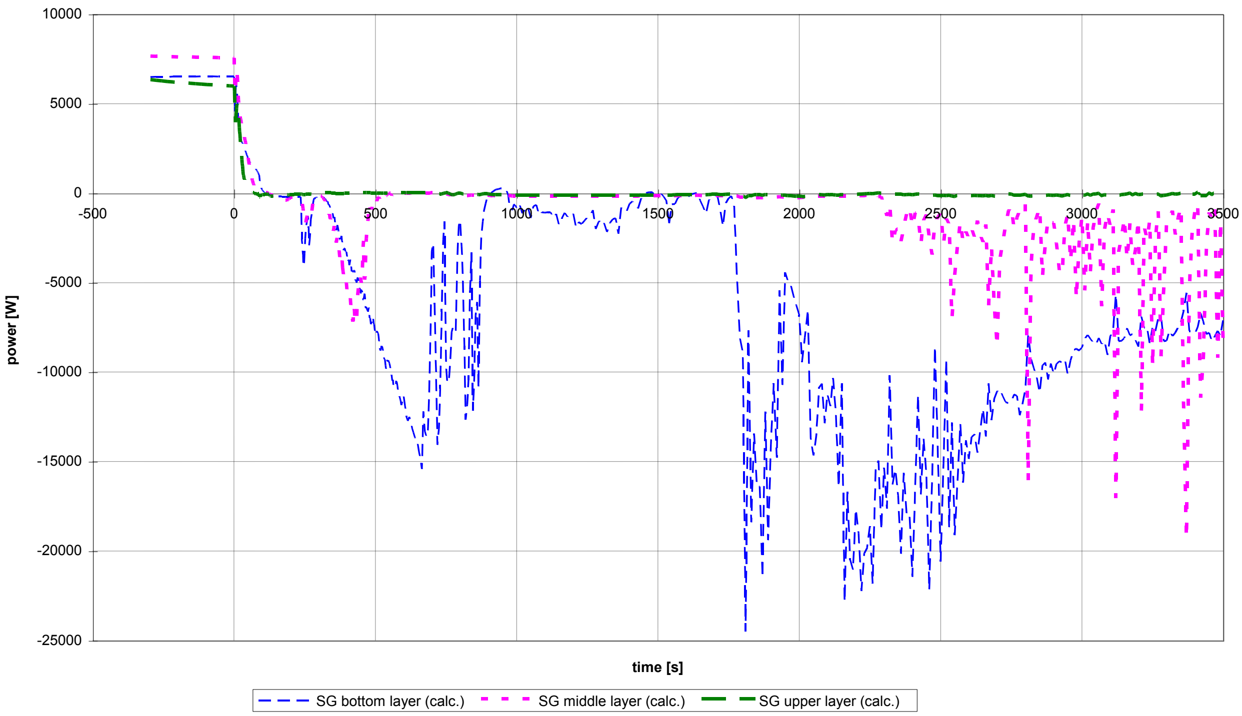

Figure 28 Calculation heat transfer at SG layers

5.2 Auxiliary calculation with older RELAP5 version MOD3.3ef

The base analysis presented in this report (above) was done with help of the RELAP5 version MOD3.3hg (post Patch03 – our latest version at time of T3.1 analysis).

Running the identical input model with older version RELAP5/MOD3.3ef (Patch02) led to substantial differences in initial phase of the process - sudden drop of primary pressure caused probably by NC-gas mass error in PRZ – see comparison figures below.

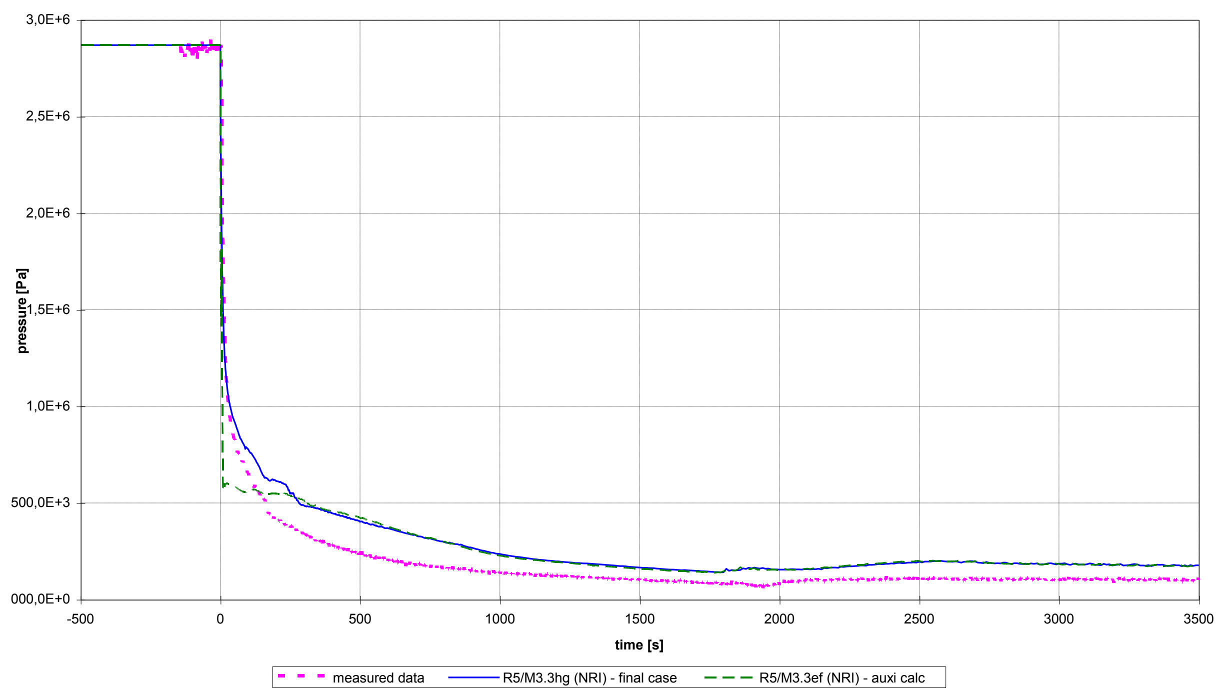

Figure 29 Primary pressure in final (Mod3.3hg) and in auxiliary calc. (Mod3.3ef)

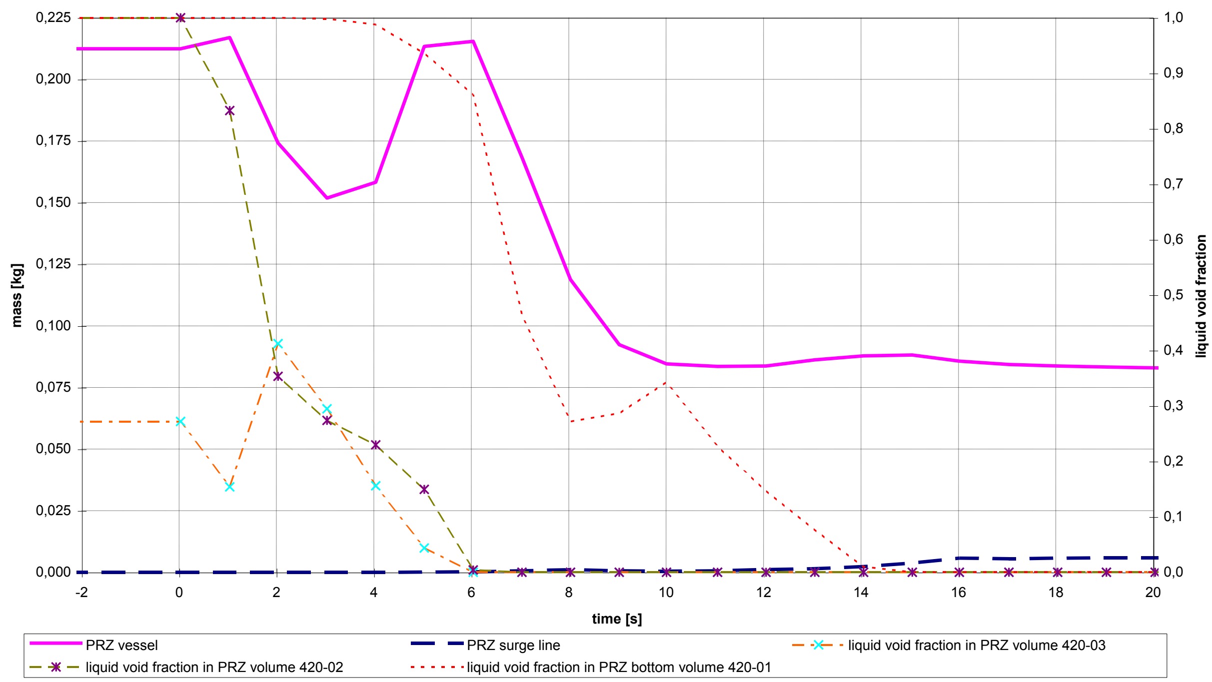

Figure 30 Mass of NC-gas and voids in pressurizer in auxiliary calc. with MOD3.3ef

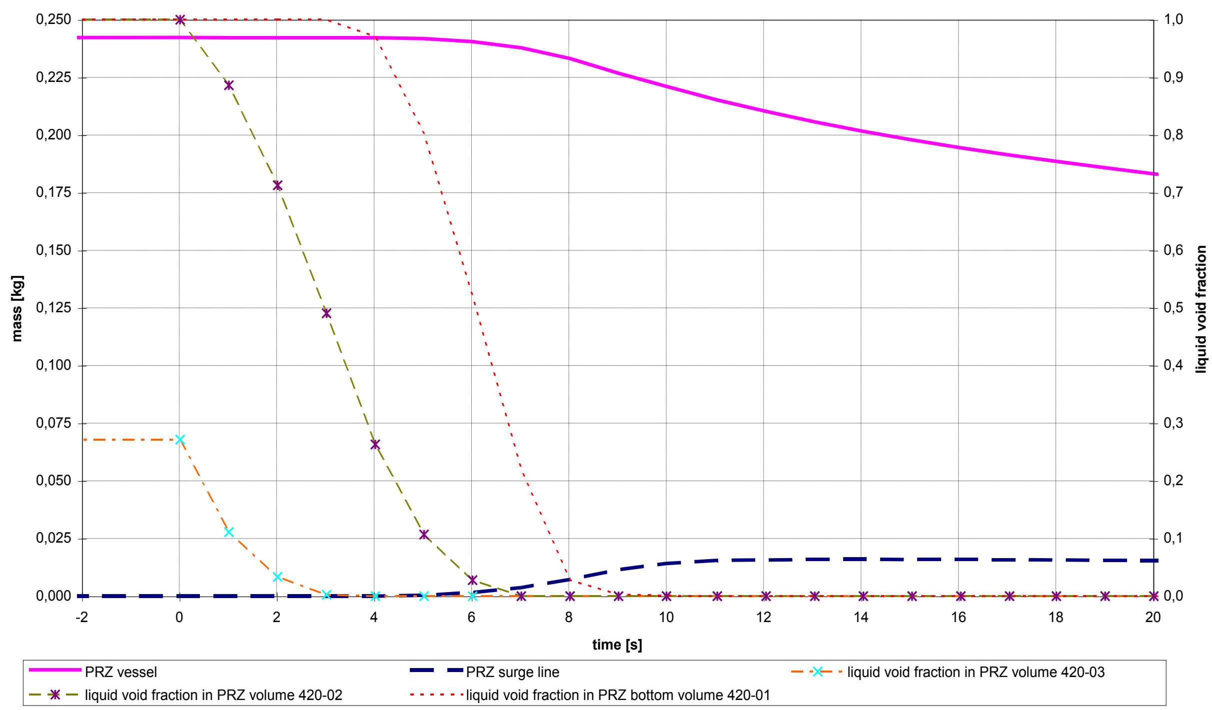

Figure 31 Mass of NC-gas and voids in pressurizer in final calc. with MOD3.3hg

6. CONCLUSIONS

As a part of the assessment of new version of RELAP5 (the MOD3.3) in UJV Rez, we have performed a set of post-test analyses of new PMK experiments. The tests T2.1, T2.2, T2.3, and T3.1 were performed in 2003-2004 in frame of the IMPAM-VVER project and presented in NUREG/IA-0229 report. The presented report is focused on the test T3.1 - a large-break LOCA with 30% break starting from shutdown conditions with nitrogen in PRZ.

The PMK facility is a scaled down model of the VVER-440/213 and it was primarily designed for investigating small-break loss of coolant accidents (SBLOCA) and transient processes of this type of NPP. The volume and power scaling of PMK facility are 1:2070. Transients can be started from nominal operating conditions. The ratio of elevations is 1:1 except for the lower plenum and pressurizer. The six loops of the plant are modeled by a single active loop. In the secondary side of the steam generator the steam/water volume ratio is maintained. The coolant is water under the same operating conditions as in the nuclear power plant.

The RELAP5 input deck used for the post-test analyses is a modified version of the older UJV input deck used for modeling of PMK-NVH in early 90-ties, when we analyzed the IAEA organized SPE tests. Listing of the current version of the deck used for the presented analyses is in the Appendix I.

Comparison of the measured test data and the calculation results showed very good overall agreement of all major system parameters as primary pressure, reactor level, reactor coolant and clad temperature etc. Also, prediction of nitrogen mass balance and transport was in very good agreement with measured data.

7. REFERENCES

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8. Král, P.: Results of RELAP5 Calculations of LOCA D136 and D60 mm for VVER-440/213, IMPAM-VVER, UJV Rez, 2005.
9. Král P.: REL RELAP5/MOD3.3 Assessment Against New PMK Experiments, prezentation at Fall 2006 CAMP Meeting.
10. Král P.: RELAP5/MOD3.3 Assessment Against PMK Test T3.1 - LOCA with Nitrogen in PRZ, prezentation at Fall 2008 CAMP Meeting.
11. Král P.: RELAP5/MOD3.3 Assessment Against PMK Test T3.1 - LOCA with Nitrogen in PRZ, UJV Z 2545 T, November 2008.
12. Král P.: RELAP5/MOD3.3 Assessment Against New PMK Experiments, NUREG/IA-0229, June 2010.

APPENDIX A COMPLETE SET OF COMPARISON PLOTS FOR CASE T3.1

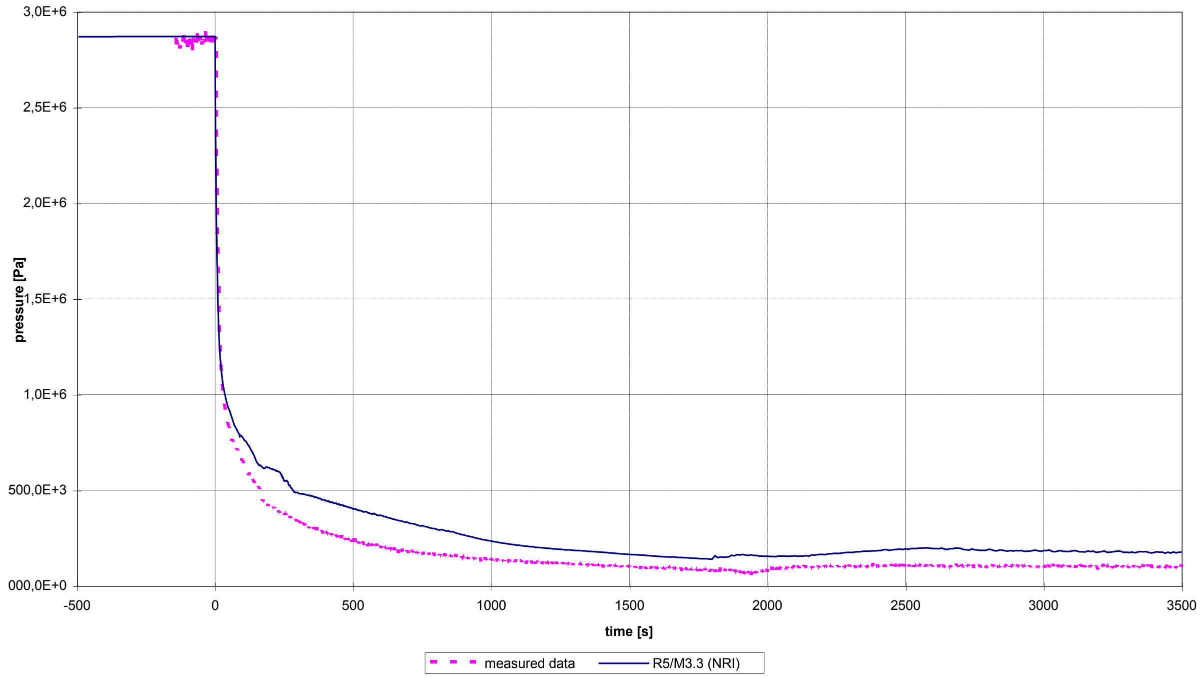

Fig.A-1 Primary pressure (T3.1)

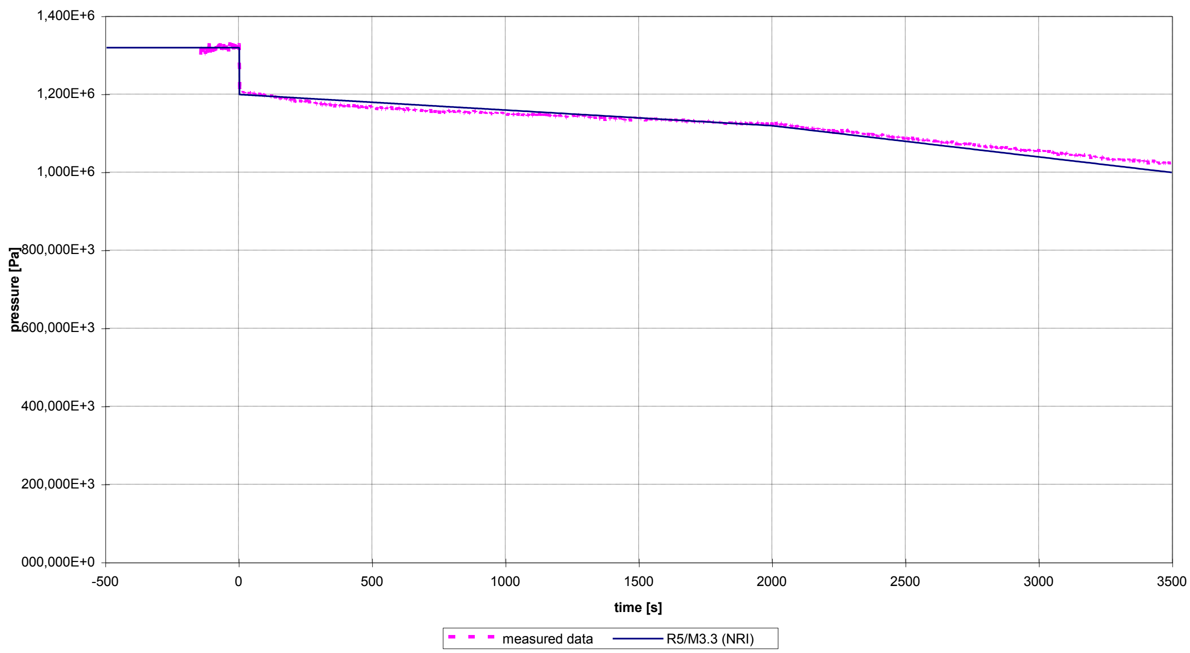

Fig.A-2 Secondary pressure (T3.1)

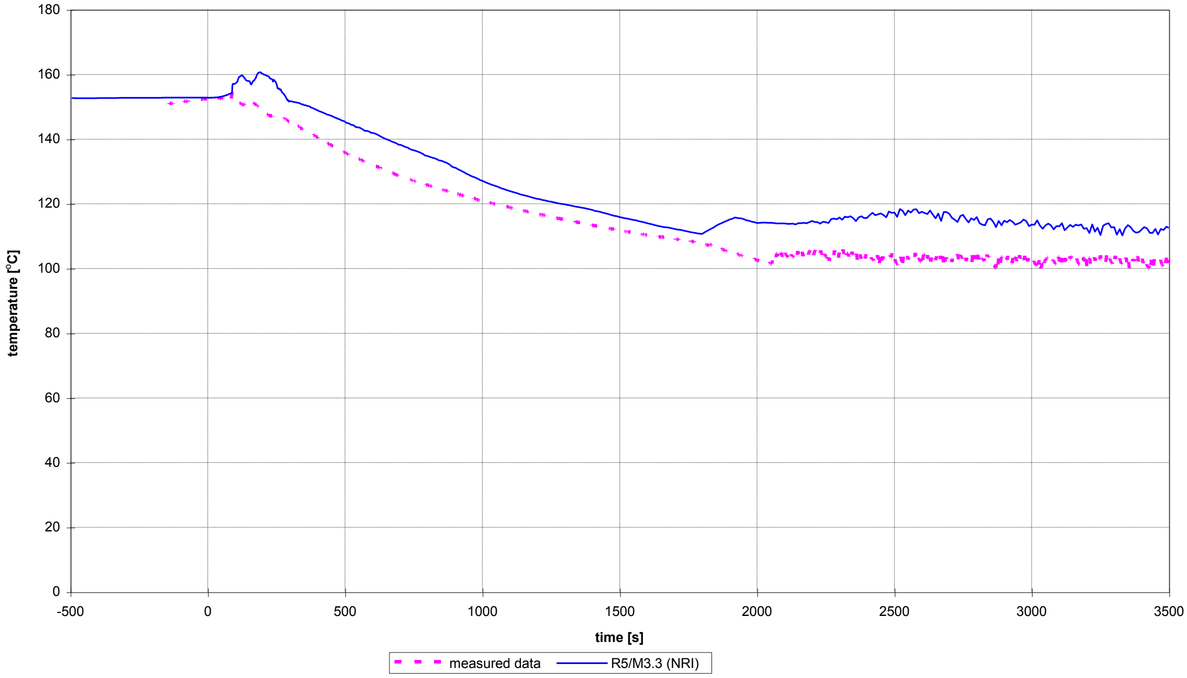

Fig.A-3 Core inlet temperature (T3.1)

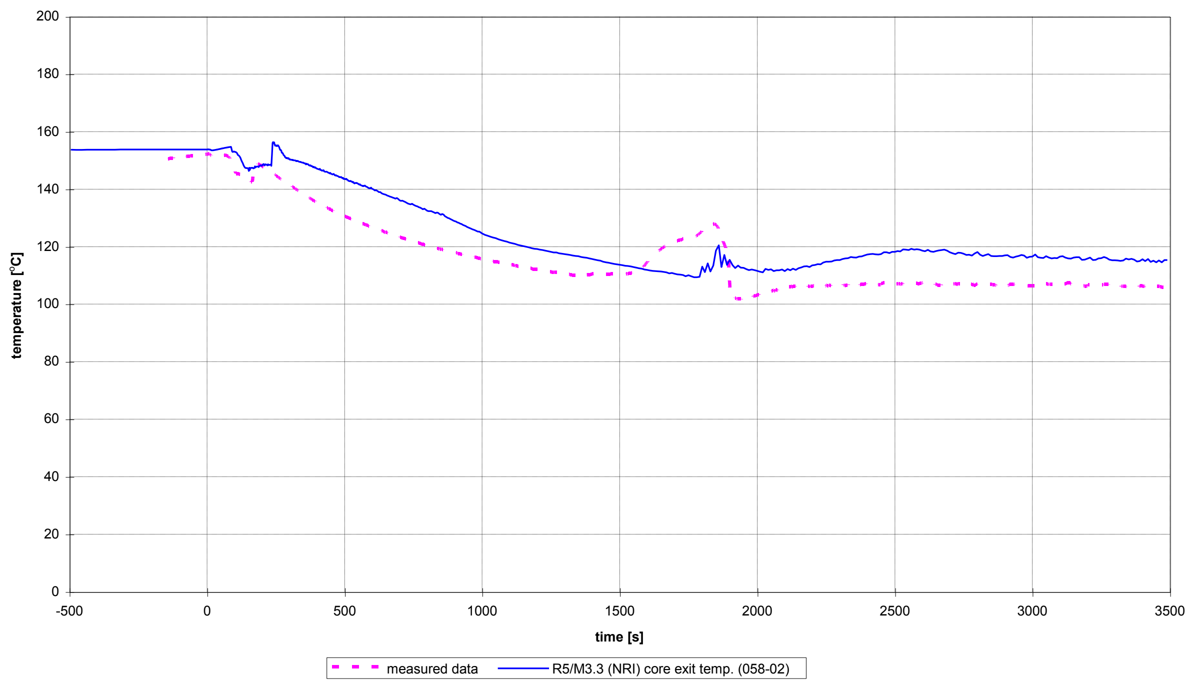

Fig.A-4 Core outlet temperature (T3.1)

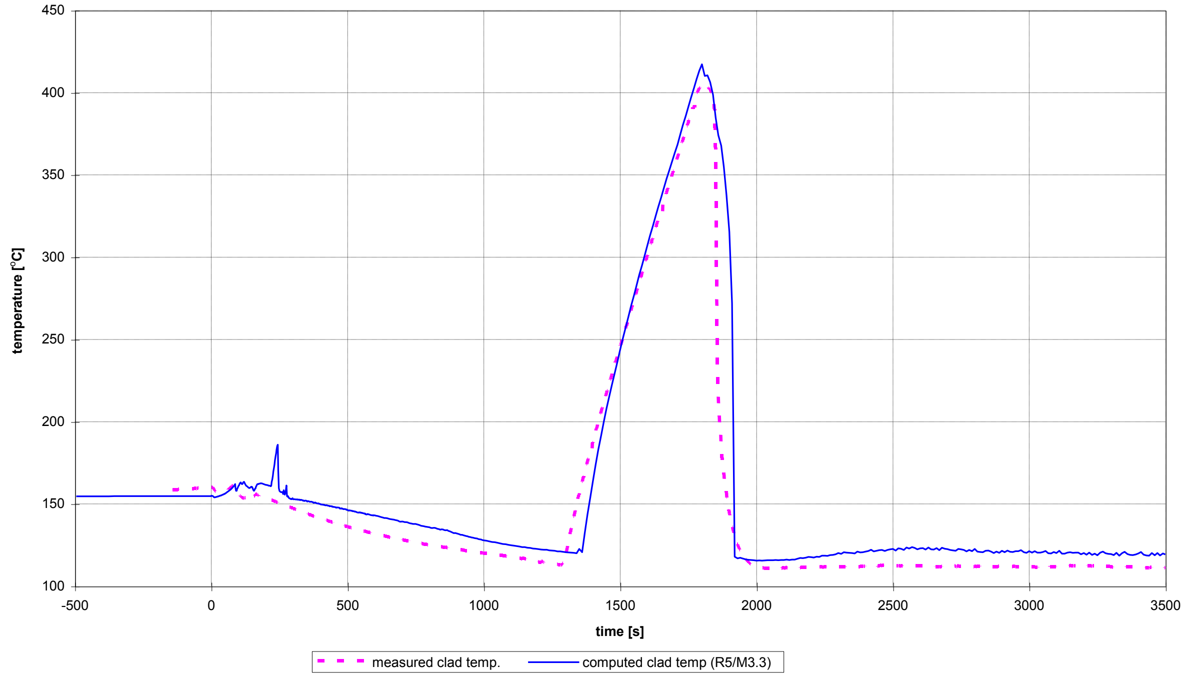

Fig.A-5 Cladding temperature (T3.1)

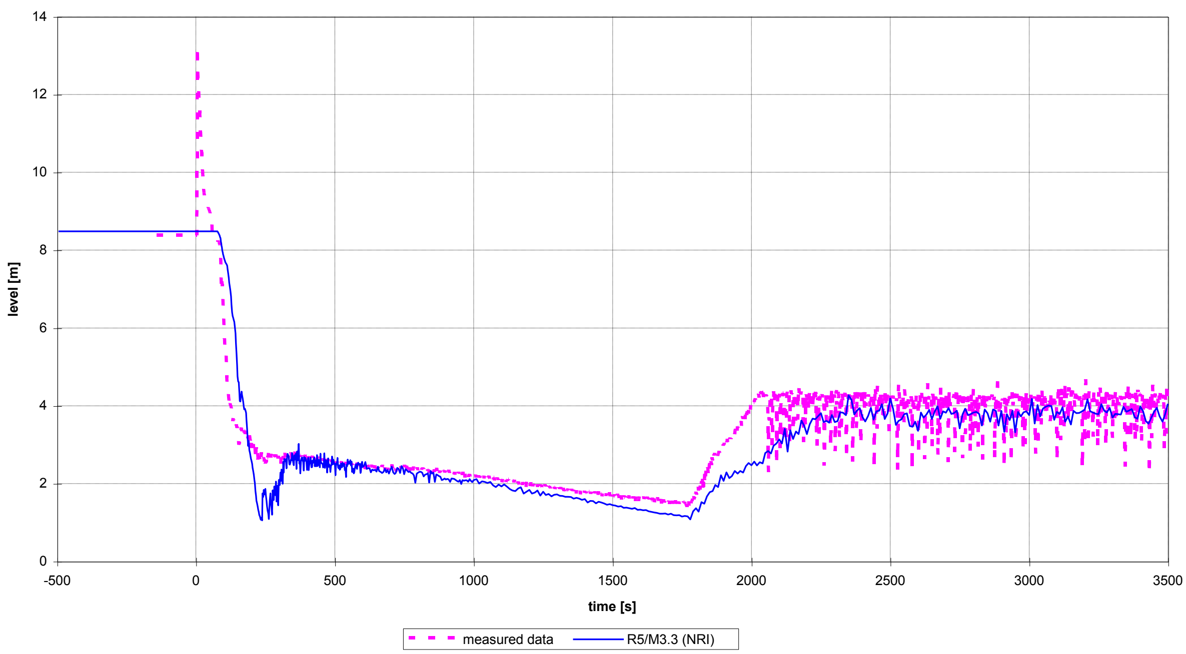

Fig.A-6 Collapsed level in reactor (T3.1)

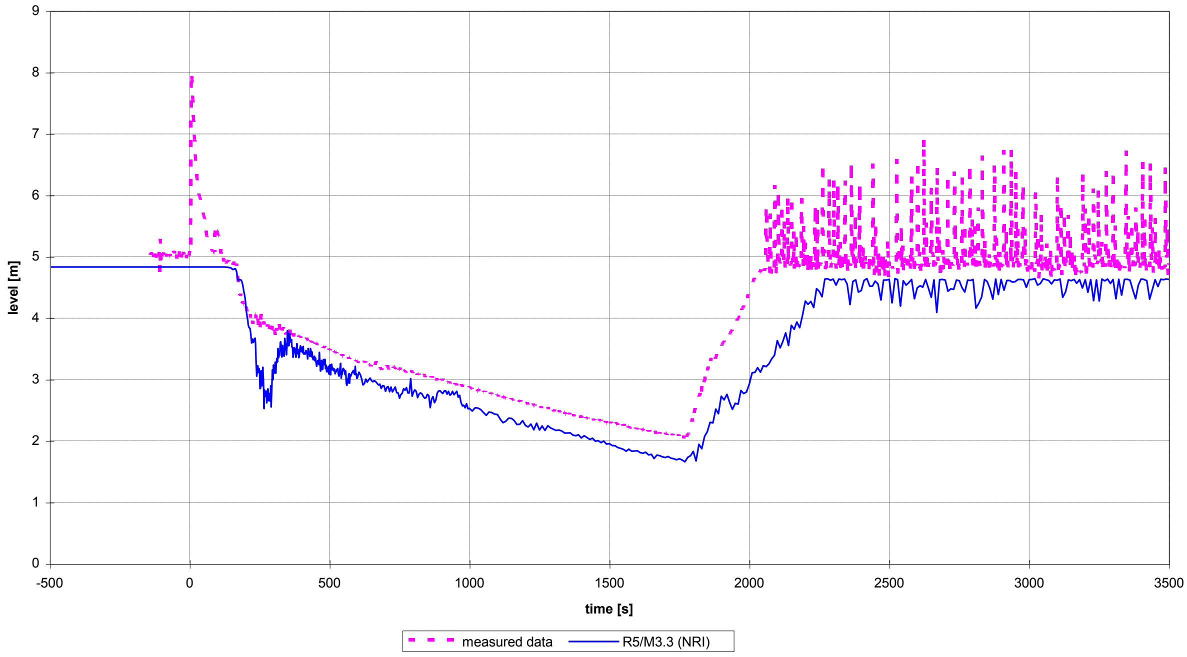

Fig.A-7 Collapsed level in reactor downcomer (T3.1)

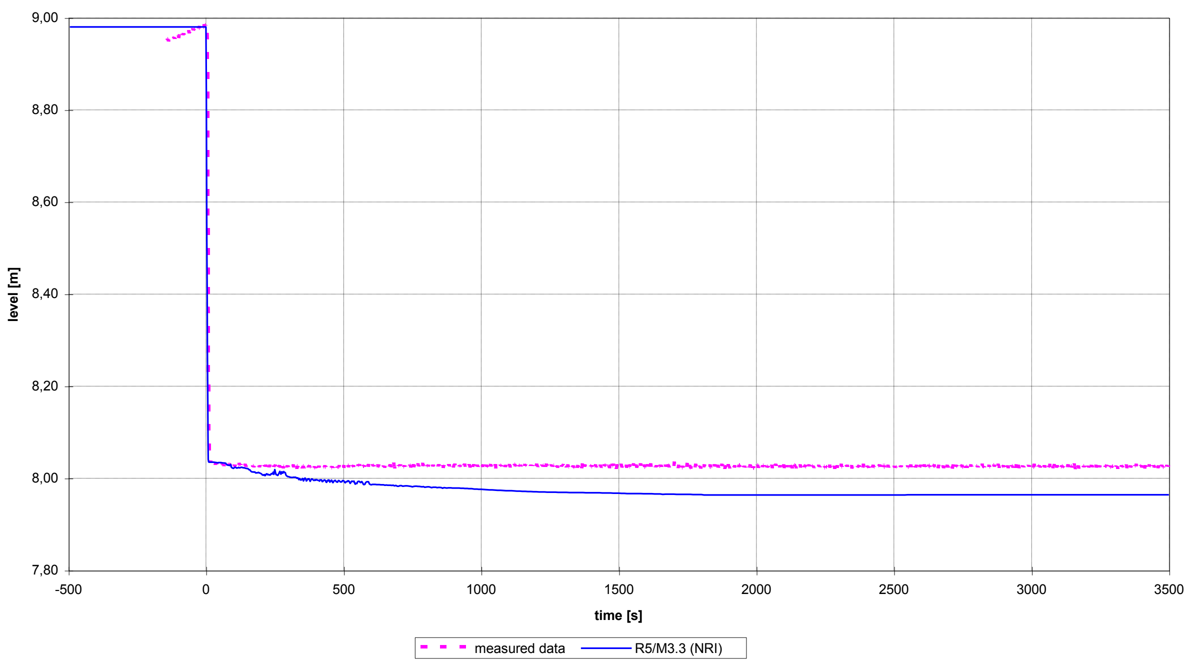

Fig.A-8 Collapsed level in PRZ (T3.1)

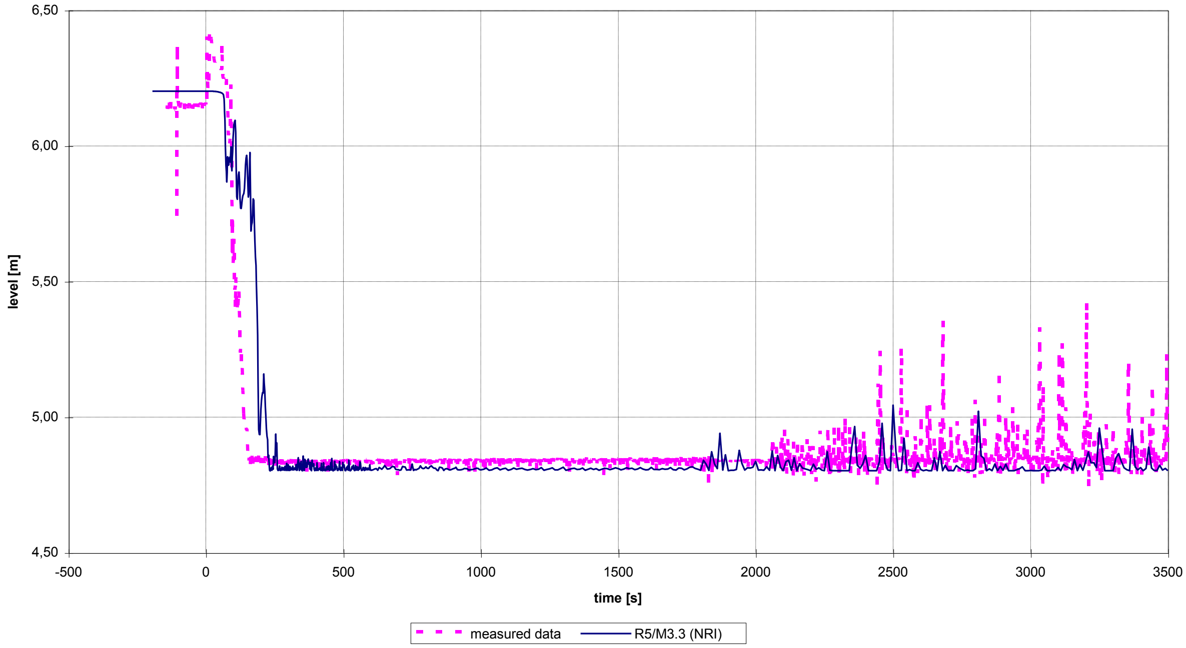

Fig.A-9 Collapsed level in hot leg loop seal – reactor side (T3.1)

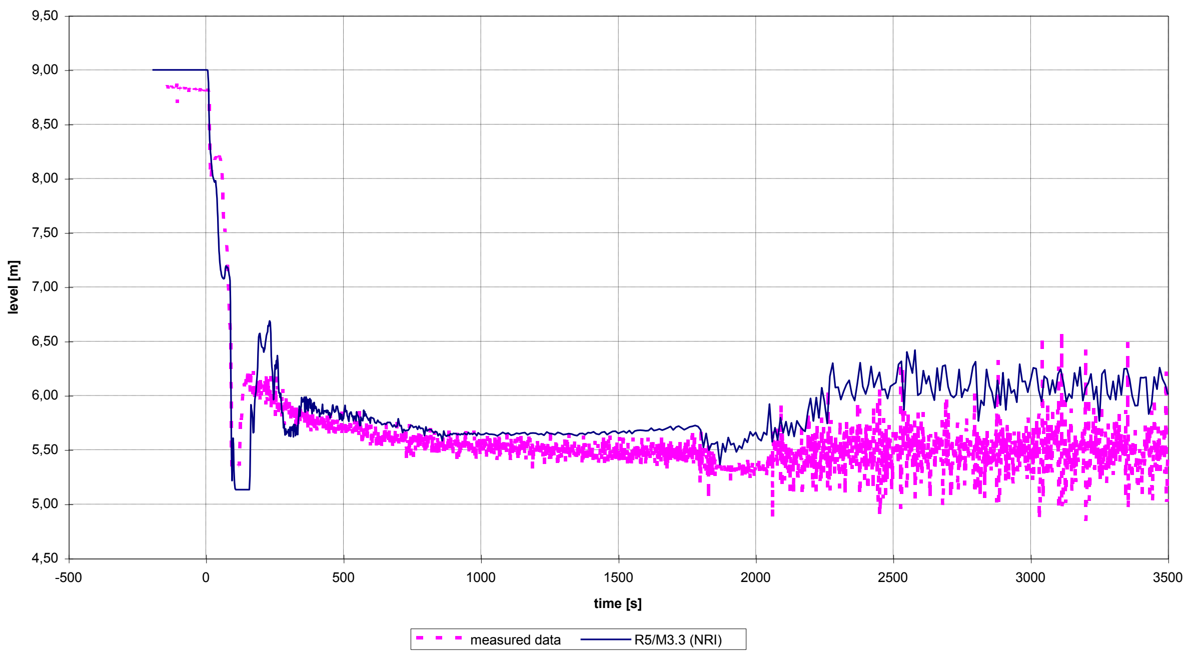

Fig.A-10 Collapsed level in hot leg loop seal – SG side (T3.1)

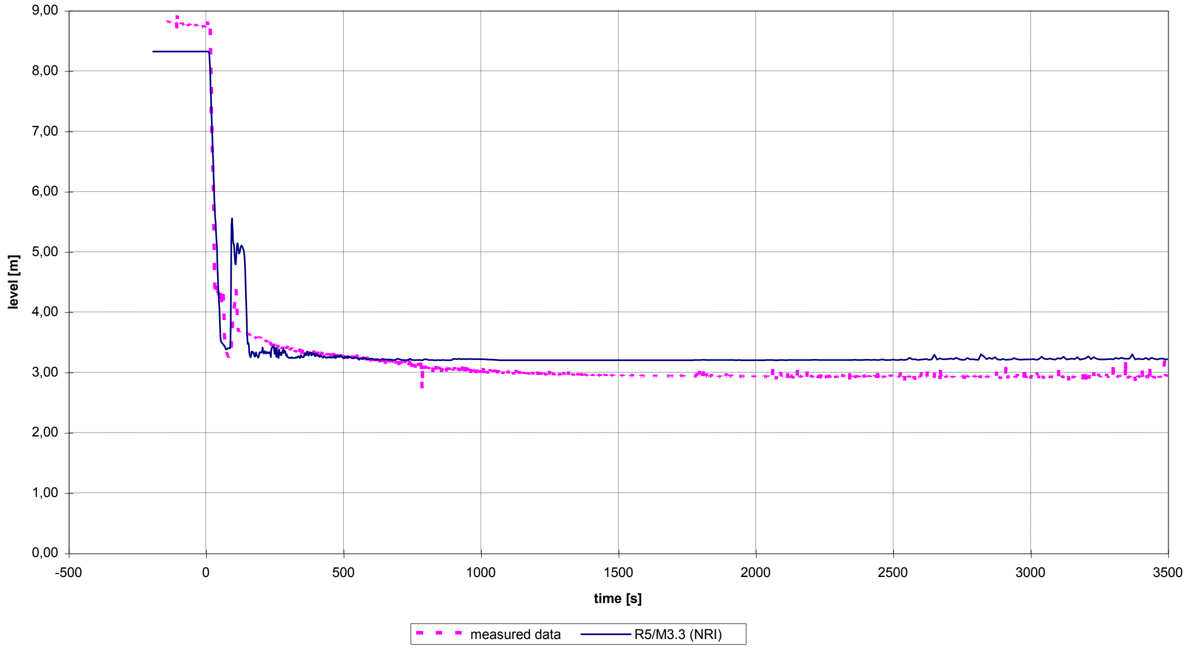

Fig.A-11 Collapsed level in cold leg loop seal – SG side (T3.1)

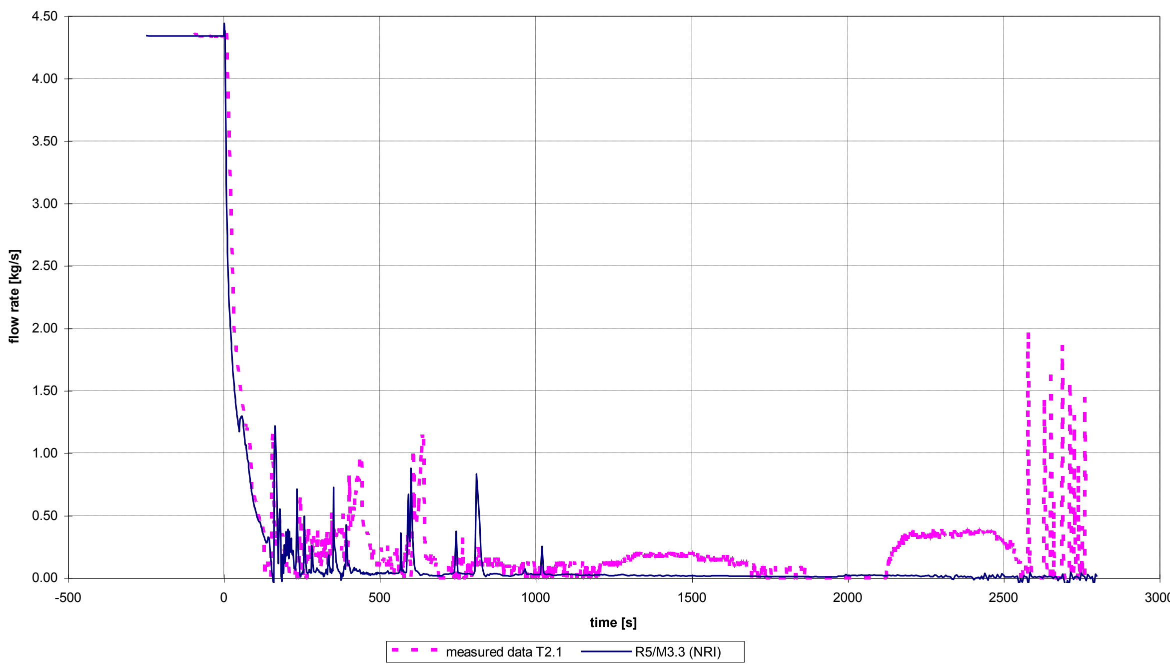

Fig.A-12 Loop mass flow rate (T3.1)

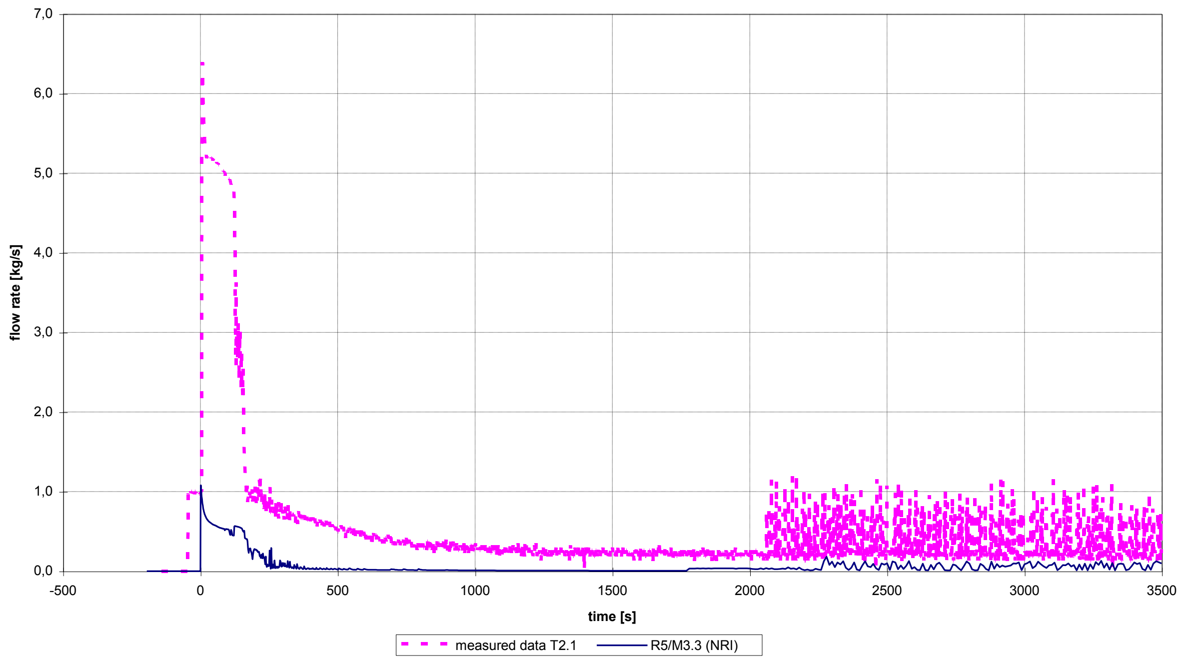

Fig.A-13 Break mass flow rate (T3.1)

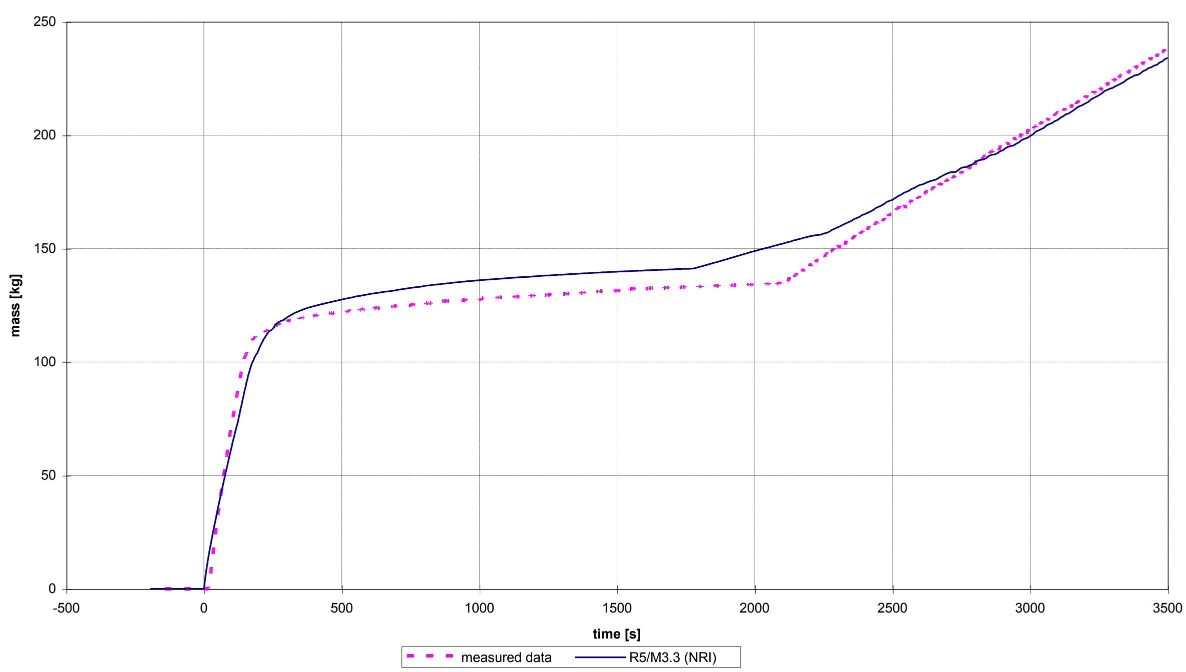

Fig.A-14 Integrated break mass flow rate (T3.1)

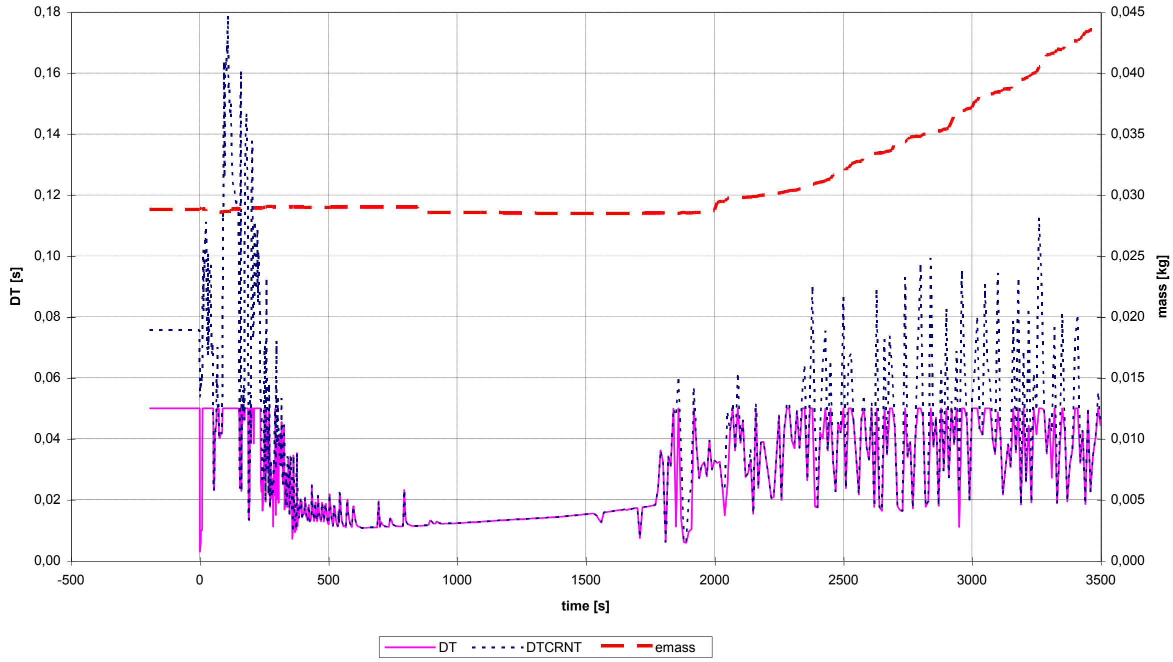

Fig.A-15 Parameters of calculation (T3.1)