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Analysis of ISP-33 using RELAP5-3D

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1. Introduction

Natural circulation is an essential core cooling mechanism during regular as well as accidental plant operation. In accidental conditions with loss of coolant from primary system and with preserved secondary side heat sink, different modes of natural circulation may take place. Therefore the dependence of the natural circulation on the primary system inventory is of particular interest in a small-break LOCA scenarios, where core cooling at reduced coolant inventory should be ensured. In the particular case of VVER-440 geometry the natural circulation in primary system is influenced due to loop seals in both, hot and cold legs of the circulation loops. Furthermore, due to the horizontal construction of the steam generators the driving head for the natural circulation is rather small. The PArallel Channel TEST Loop (PACTEL) facility was designed and constructed in Finland to study experimentally thermal hydraulic characteristics of VVER-440 reactors. In 1992-1993 the OECD/CSNI standard problem ISP-33 was conducted on the PACTEL facility with the aim to study experimentally natural circulation in VVER-440 over a range of primary side inventory levels as well as to test the ability of thermal-hydraulic computer codes to analyse this kind of phenomena. In this paper the post-test calculation of the ISP-33 experiment with recent version of RELAP5-3D code (version 1999cb) is presented. Taking into account the character of PACTEL facility and ISP-33 experiment, only 1-D capabilities of RELAP5-3D were used.

2. Description of PACTEL facility

The PACTEL facility is a volumetric scaled model of the 6-loop VVER-440 reactor with three separate loops and full length, electrically heated fuel rod simulators. The reference reactor of the PACTEL is Loviisa NPP. The volumetric scaling factor of the facility is 1:305; elevations are preserved in full height to match the natural circulation gravitational heads in the reference system.

The facility was designed to simulate the major components of the primary system of VVER-440 reactors during SB and MB LOCAs, natural circulation and operational transients. The reactor vessel model consists of U-tube construction including downcomer, lower plenum, core and upper plenum. The rod bundle simulating the core consists of 144 fuel rod simulators with a chopped cosine axial power distribution and a maximum total power output of 1 MW (22% of scaled full power). The fuel rod pitch (12.2 mm) and diameter (9.1 mm) are identical to those of reference reactor. The rods are divided into three roughly triangular-shaped parallel channels representing the intersection of the corners of three hexagonal VVER rod bundles. Three loops with double capacity steam generators are used to model the six loops of reference plant. The horizontal U-tubes lengths and diameters of the PACTEL steam generators correspond to those of the full-scale models, but the overall height is smaller. All the major components are constructed of stainless steel. The facility is shown on Fig.1. Basic characteristics of the PACTEL facility and reference plant are summarised in Table I.

TABLE I. Characteristics of the PACTEL facility and their comparison with reference plant.

Parameter	PACTEL	VVER-440/V213 Loviisa NPP
Volumetric scaling ratio	1:305	-
Scaling factor of component heights and elevations	1:1	-
Number of primary loops	3	6
Maximum heating /thermal power	1MW	1375 MW ¹⁾
Number of fuel rods	144	39438 ²⁾
Outer diameter of fuel rod	9.1 mm	9.1 mm
Heated length of fuel rod simulators	2.42 m	2.42 m
Maximum operating primary pressure	8.0 MPa	12.3 MPa
Maximum operating primary temperature	300 °C	300 °C
Maximum operating secondary pressure	5.0 MPa	5.0 MPa
Maximum operating secondary temperature	260 °C	260 °C

Comments:

1) increased power 1500 MW after modernisation of the NPP

2) this number correspond to “reduced” core (outer fuel assemblies replaced by dummies)

3. Description of ISP-33 experiment

The main goal of ISP-33 and the corresponding experiment was to study natural circulation in a VVER-440 primary system including several single- and two-phase natural circulation modes. Contrary to typical SB LOCA event without ECCS injection with continuous transitions between different natural circulation modes, the primary coolant mass was reduced stepwise in the liquid form and the amount of drained water was given as boundary condition. The heat generated in the core representing decay heat was transported by coolant to the steam generators. The amount of

water drained in one step was about 9.5 % of initial coolant inventory. The draining periods were very short compared to the stabilising period between two subsequent drainings. That is why the different natural circulation mechanisms were clearly identified during the experiment. The expected periods during experiment were:

- Single-phase natural circulation.
- Two-phase natural circulation with continuous liquid flow.
- Natural circulation of reflux boiling type. Heat is transported by saturated steam to steam generators and condensed there. The resulting water is then returned to the reactor pressure vessel via cold legs and the downcomer.
- Cooling of the core with partially superheated steam. The steam generators are still capable of removing heat from primary system.

Before the experiment initiation the facility was heated until it reached the selected temperature and pressure, and steady state was established at these conditions. Constant core power level 165 kW (3.7% of the scaled nominal power) was selected for the test. This power was held constant during whole experiment. The PRZ heaters were switched off after first draining. The secondary side conditions in steam generators were held constant; secondary side pressure was controlled by control valve. When the core temperature began to rise and maximum cladding temperature reached 350 °C, secondary side was depressurised and the core heaters were switched off (termination of the experiment).

4. Modelling assumptions for RELAP5-3D

Due to the nature of the facility and ISP-33 experiment, only 1-D capabilities of RELAP-3D code were used. The nodalization consists of a pressure vessel and three-loop representation of the PACTEL facility in which each loop is connected to separate steam generator. The core region is modelled by one single hydraulic channel (instead of three). Nine heated volumes were used in axial direction in order to reproduce stepped cosine axial power profile of the fuel rod simulators. Geometry of the individual loops was modelled precisely in order to enable simulation of asymmetrical behaviour of the loops during experiment (e.g. loop seal clearing). The surge line, connecting the hot leg to the pressurizer, is located on loop No.1. Steam generators heat exchange tubes were lumped into three horizontal layers. A recirculation model of the SG [3] was used on the secondary side. FW injection was modelled as boundary condition. Taking into account relatively long time period during which the PRZ heaters were in operation, recirculation model of PRZ was used too in order to avoid unphysical thermal stratification of the water in pressurizer (upper flooded volumes colder due to heat losses than lower volumes heated by PRZ heaters). Having in mind the importance of heat capacities and heat losses in phenomenology of natural circulation in scaled facilities, all important heat structures in primary system (vessel internals and walls) were modelled. Default control flags were used preferably. Nodalization scheme is shown in Fig.2. Basic information about the nodalization is summarised in Table II.

TABLE II. RELAP5-3D nodalization scheme of the PACTEL facility

Number of control volumes	286
Number of junctions	308
Number of heat structures	317
Number of mesh points in heat structures	1921

During the development of the PACTEL nodalization, attention was paid to geometrical and material fidelity with the reference system. Developed model was tuned in order to adjust proper pressure head losses within primary system as well as heat losses from primary system to surroundings. Comparison of measured and calculated steady state values is given in Table III.

TABLE III. Comparison of measured and calculated steady state values (initial conditions)

Parameter	Experiment	Calculation
Pressurizer pressure [MPa]	7.39	7.39
Steam generator pressure [MPa]	4.18	4.17
Downcomer flow rate [kg/s]	1.47	1.46
Pressurizer level [m]	5.18	5.18
Power of PRZ heaters [kW]	2	2
Level in steam generator No.1 [m]	0.271	0.270
Level in steam generator No.2 [m]	0.272	0.271
Level in steam generator No.3 [m]	0.263	0.263
Feedwater temperature [°C]	32	32
Feedwater flow-rate per SG [kg/s]	0.018	0.018
Fluid temperature in lower plenum [°C]	~ 252	252
Fluid temperature at reactor outlet [°C]	~ 275	275.2

The most important boundary condition during transient calculation is the amount of water drained from the bottom of the reactor vessel in each step. This value together with maximum value of drainage flow was tuned according to experimental values. The results of the ISP-33 experiment clearly show that during three pressure spikes after second draining the PACTEL safety valve was leaking. According to [2], about 10 kg of coolant escaped through this valve located in upper plenum. However, this amount was not measured during the experiment. Since the parameters of the safety valve were not exactly known, sensitivity study was performed with the aim to tune the setpoints for its opening/closing as well as cross-flow area representing "leaking" valve. The correspondence of measured and calculated course of the primary pressure was used as criterion for this tuning. No artificial "tricks" were used in the analysis.

5. Results

Experimental results obtained in ISP-33 are described and illustrated e.g. in [1], [2]. Only brief description is therefore given here. The effect of first draining was rapid decrease of the primary pressure until saturation conditions were reached at the core outlet. The flow in the downcomer remained single-phase with almost constant flow rate.

During the second draining the amount of vapour in upper plenum increased rapidly. When the swell level in the upper plenum drops below the hot leg nozzles level, voiding also began in the hot legs. Consequently, the flow rate dropped and became stagnant. The hot leg loop seals prevented the vapour from upper plenum to reach the steam generators. Consequently, primary pressure rose sharply. When the pressure increased, water flowed back into pressurizer. After short flow stagnation, loop seal clearing occurred. This was repeated two additional times. During the highest pressure spike the setpoint for safety valve leaking was reached. According to [2], about 10 kg of water escaped through safety valve on the top of upper plenum. However, this value was not measured exactly. In the calculation, integrated mass-flow-rate through this valve was about 14.4 kg.

During the 3rd draining pressure dropped and saturation temperature was reached in upper plenum. The PRZ was emptied again, and primary mass flow rate increased. Relatively steady two-phase flow was established. The bulk of this flow took place through one loop, though any of the loops was totally stagnant. As the primary inventory was reduced further, the flow rates continued to decline, finally becoming nearly stagnant. The heat transfer mechanism from primary to secondary changed to boiler-condenser mode. Reflux condensation was not observed because of the SG and loop seals geometry.

Up to the 6th draining, the overall behaviour of the primary system was well predicted by RELAP5-3D calculation, including the effect of flow stagnation followed by multiple loop seal clearing after 2nd drainage. Since the 6th draining, the water level in reactor vessel predicted by code is more than 1 meter higher than experimental value. Consequently, in the experiment core heat-up started already during the 7th drainage whereas in the calculation was predicted about 890 s later during the 8th draining. As soon as the maximum cladding temperature reached 350 °C, the depressurisation of secondary side took place. Experiment was completed 140 seconds later to protect core heaters. Despite time delay, the drop of secondary pressure, primary pressure and water level in reactor was well predicted by code.

The measured downcomer mass flow rate (averaged value over 3 minutes just before next draining) as a function of primary system inventory is shown in Fig. 3. Experimental results and results obtained using RELAP5-3D are compared in Table III and in Figs. 4.1-4.12.

6. Conclusions

Recalculation of ISP-33 exercise was performed using RELAP5-3D code. All advantages of post-test calculation were used in the analysis. Thus the impact of the user effect should be reduced significantly. No artificial tricks were used in analysis. The following conclusions result from ISP-33 experiment and RELAP5-3D simulation:

- natural circulation in VVER-440 geometry as a function of primary coolant inventory (Fig.3) differs significantly from western PWRs ;
- RELAP5-3D is capable to predict with sufficient accuracy most important phenomena and overall transient behaviour, especially for coolant inventories higher than 50 % of nominal value;
- transition from single- to two-phase natural circulation is characterised by flow stagnation followed by loop seals clearing; overall unsymmetrical loops behaviour was well predicted although this is not the case of loops as individuals;
- the ability of horizontal SG tubes to retain significant amount of the condensate in boiler condenser mode could shorten elapsed time to core dry-out;
- thermal-hydraulic computer codes used in ISP-33 exercise as well as RELAP5-3D fail in predicting correctly this ability and for the primary water inventories lower than 50 % of nominal value provide non-conservative results (delayed prediction of start of core heat-up);
Comment: in accordance with ISP-33 specifications, analysis of ISP-33 was performed assuming “exactly” horizontal SGs tubes; an alternative possibility is, that the SG tubes of PACTEL facility were slightly declined between hot and cold SG collectors and thus the ability to retain coolant results from this geometrical feature;
- under the conditions of reduced primary mass inventory, severe depressurisation of the secondary side removes the coolant from the reactor vessel towards steam generators; thus, although the short-time effect of this accident measure on core cooling is positive, for the long-time core cooling it is necessary to combine this measure e.g. with ECCS injection.

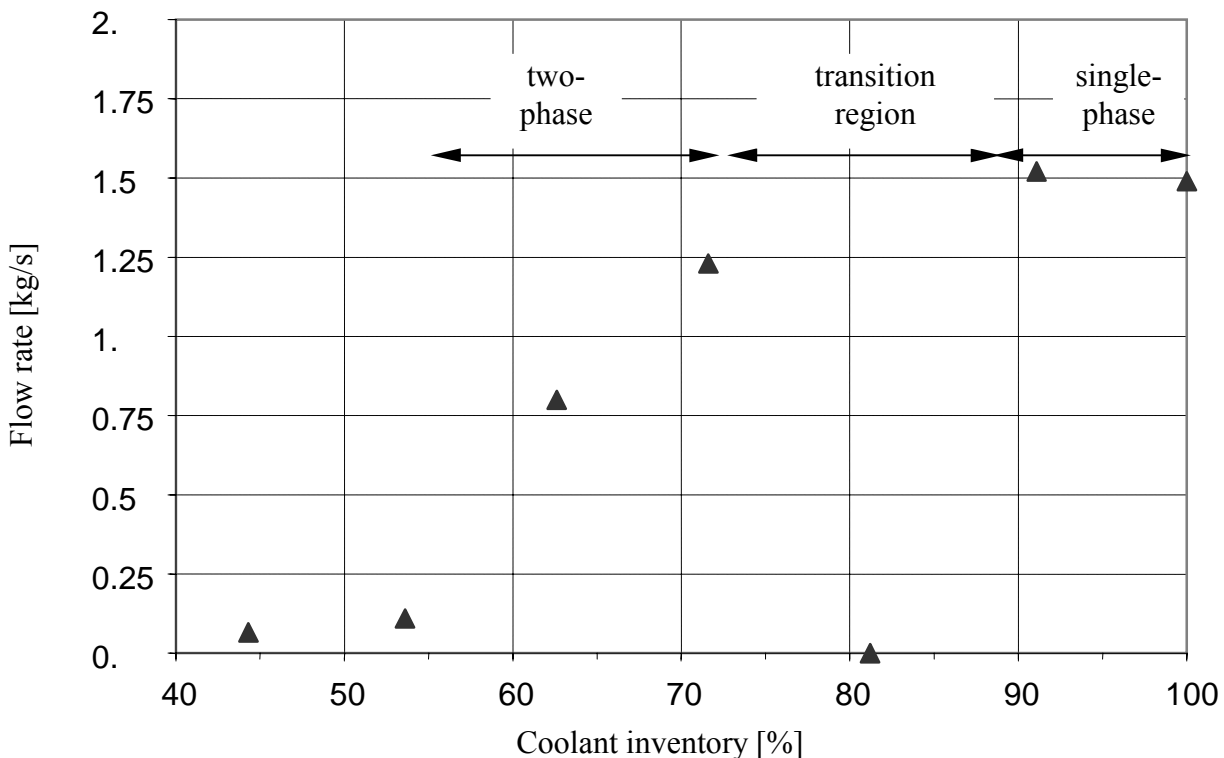


Fig. 3. Mass-flow-rate through downcomer versus primary inventory

TABLE III. Comparison of main measured and calculated events and phenomena during experiment

Time	Draining No.	Primary inventory [%]	Experiment	Calculation
0 s		100	$P_{PRZ} = 2 \text{ kW}$, $p_{prim} = 7.39 \text{ MPa}$	$P_{PRZ} = 2 \text{ kW}$, $p_{prim} = 7.39 \text{ MPa}$
420 s			$P_{PRZ} = 6 \text{ kW}$, $p_{prim} = 7.36 \text{ MPa}$	
430 s				$P_{PRZ} = 6 \text{ kW}$, $p_{prim} = 7.36 \text{ MPa}$
680 s			$P_{PRZ} = 2 \text{ kW}$, $p_{prim} = 7.52 \text{ MPa}$	
690 s				$P_{PRZ} = 2 \text{ kW}$, $p_{prim} = 7.53 \text{ MPa}$
1200 s	1 st	91.1	<ul style="list-style-type: none"> - PRZ almost empty - no significant voiding in upper plenum - saturation pressure reached 	<ul style="list-style-type: none"> - well predicted overall behaviour of the system - behaviour of individual not matched correctly
2100 s	2 nd	81.2*	<ul style="list-style-type: none"> - upper plenum voiding - swell level near hot legs - flow stagnation-oscillations - primary pressure rise - leaking of the safety valve - PRZ level rise 	<ul style="list-style-type: none"> - well predicted overall behaviour of the system - behaviour of individual not matched correctly - 14.4 kg of coolant escaped through safety valve
3000 s	3 rd	71.6	<ul style="list-style-type: none"> - pressure drop to saturation - PRZ empty - 2-phase flow trough 3rd loop - single-phase in cold legs 	<ul style="list-style-type: none"> - higher predicted total mass flow rate in downcomer
3900 s	4 th	62.6	- boiler-condenser mode	
4800 s	5 th	53.6	- boiler-condenser mode	
5700 s	6 th	44.3	- boiler-condenser mode	- overpredicted water level in reactor
6600 s	7 th	35.3	- start of draining	
6660 s			- $T_{cladding}$ starts to rise	
6700 s			- $T_{cladding} = 350 \text{ }^\circ\text{C}$ depressurisation of secondary	
7500 s	8 th	26.2		- start of draining
7550 s				- $T_{cladding}$ starts to rise
7586 s				- $T_{cladding} = 350 \text{ }^\circ\text{C}$ depressurisation of secondary

* About 10 kg of coolant (~ 1.5 % of initial inventory) escaped from primary system between 2nd and 3rd draining through safety valve [2]. Since the third draining the primary inventory given in Table III should be reduced by this value.

Abbreviations

ECCS	emergency core cooling system
FW	feedwater
ISP	international standard problem
LOCA	loss of coolant accident
MB	medium break
NPP	nuclear power plant
PRZ	pressurizer
PWR	pressurized water reactor
SB	small break
SG	steam generator

References

- [1] J. Kouhia et al: International Standard Problem ISP-33 at the PACTEL Facility for the Simulation of VVER-440 Type PWRs, European Two-Phase Flow Group Meeting, 7-10 June 1993 Hannover, Germany
- [2] OECD/NEA/CSNI International Standard Problem No.33, PACTEL Natural Circulation Stepwise Coolant Inventory Reduction Experiment, Comparison Report, Volume 1 and 2, NEA/CSNI/R(94)24, December 1994
- [3] P. Matejovič, L. Vranka, E. Václav: Application of the thermal-hydraulic codes in VVER-440 steam generators modelling, Third International Seminar on Horizontal Steam Generators, Lappeenranta, Finland, 1995

Fig. 1. PACTEL facility

Fig. 4.1. Pressure in pressurizer.

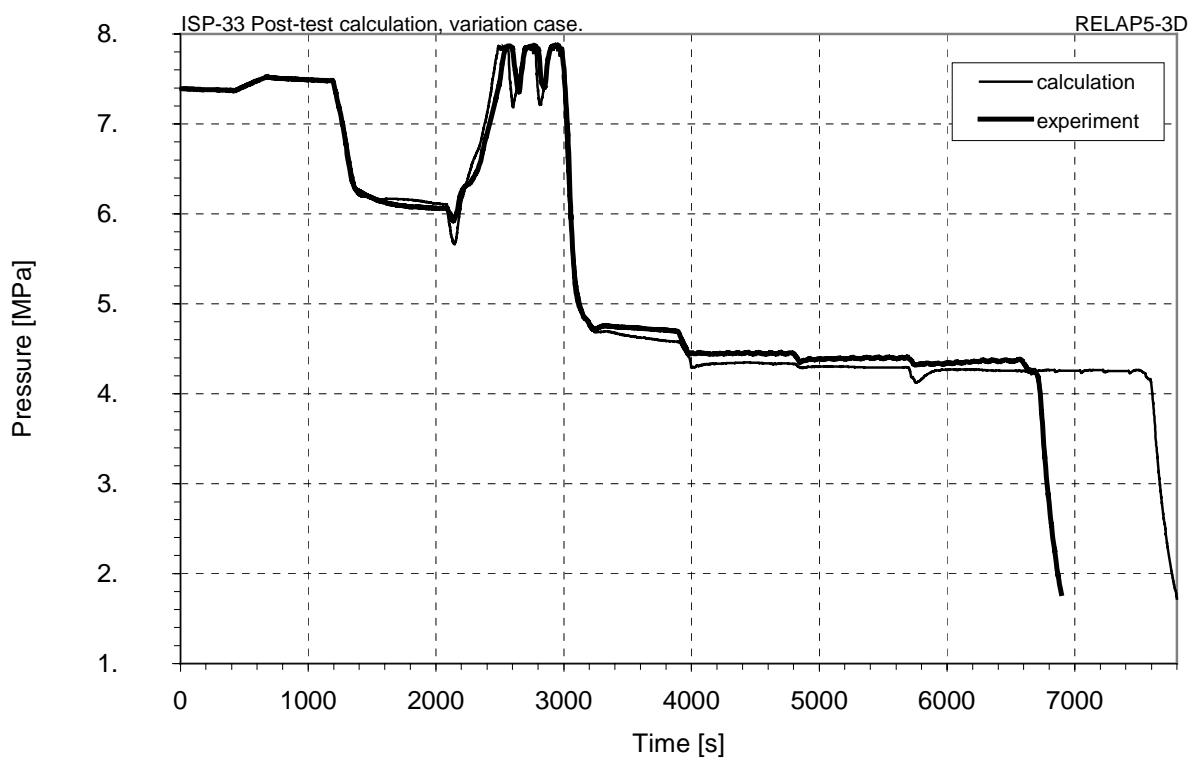


Fig. 4.2. Secondary pressure in SG1.

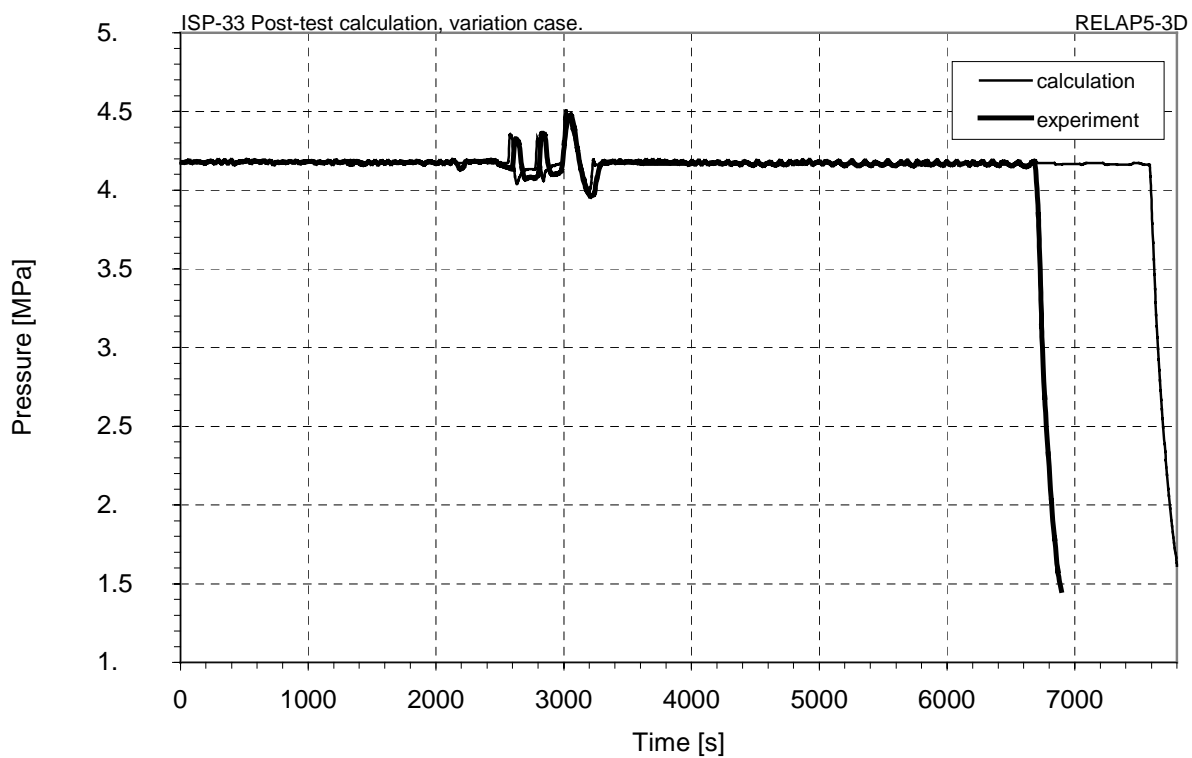


Fig. 4.3. Mass flow through the drainage valve.

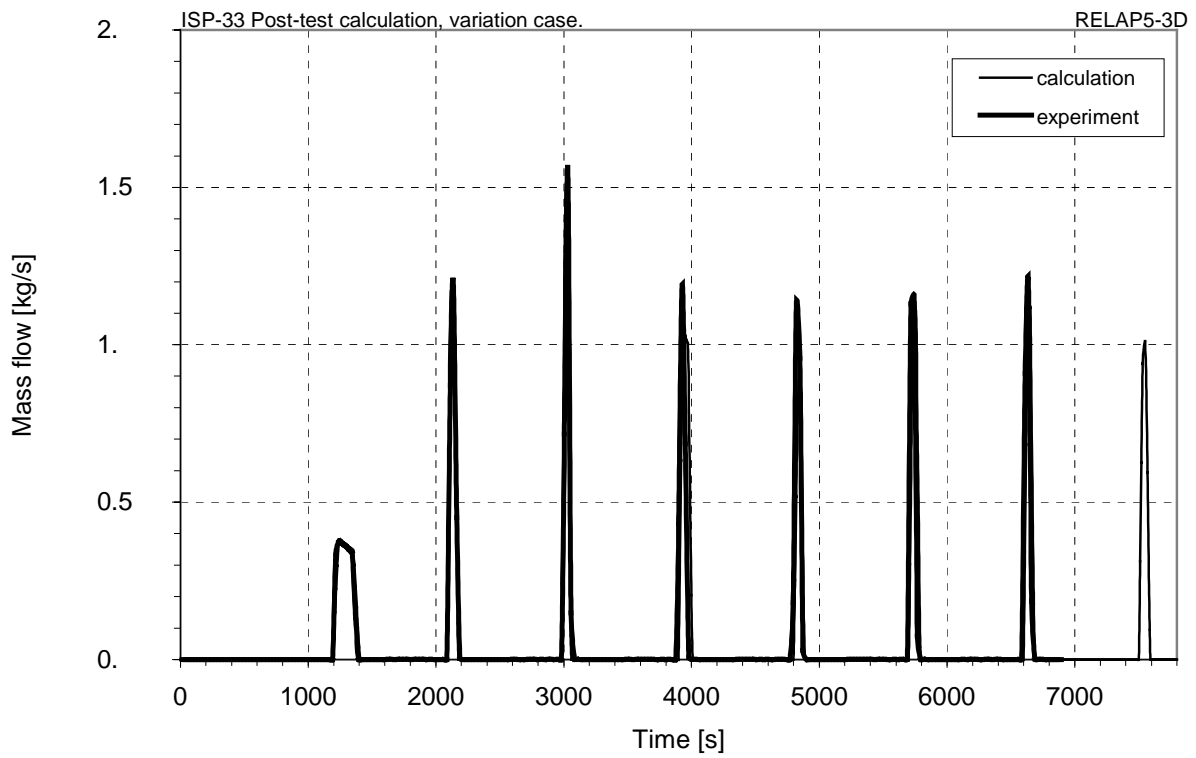


Fig. 4.4. Cladding temperature at elevation 1210 mm.

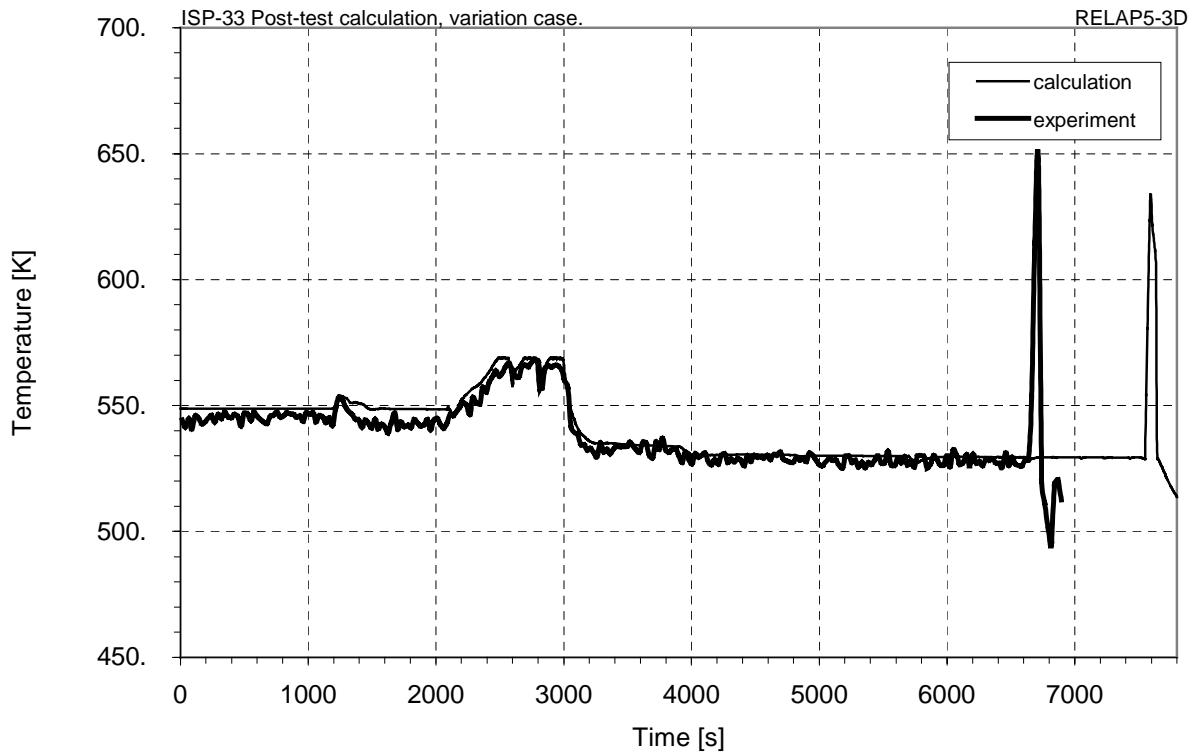


Fig. 4.5. Collapsed level in reactor vessel.

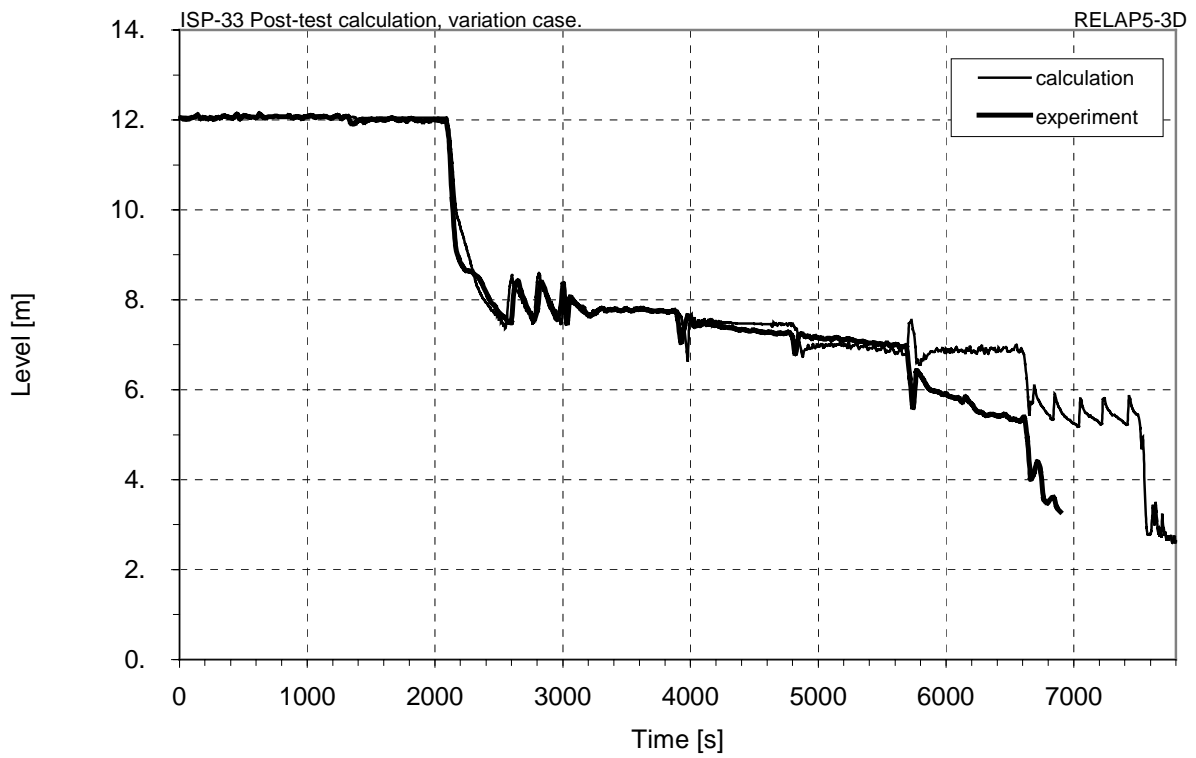


Fig. 4.6. Pressurizer collapsed level.

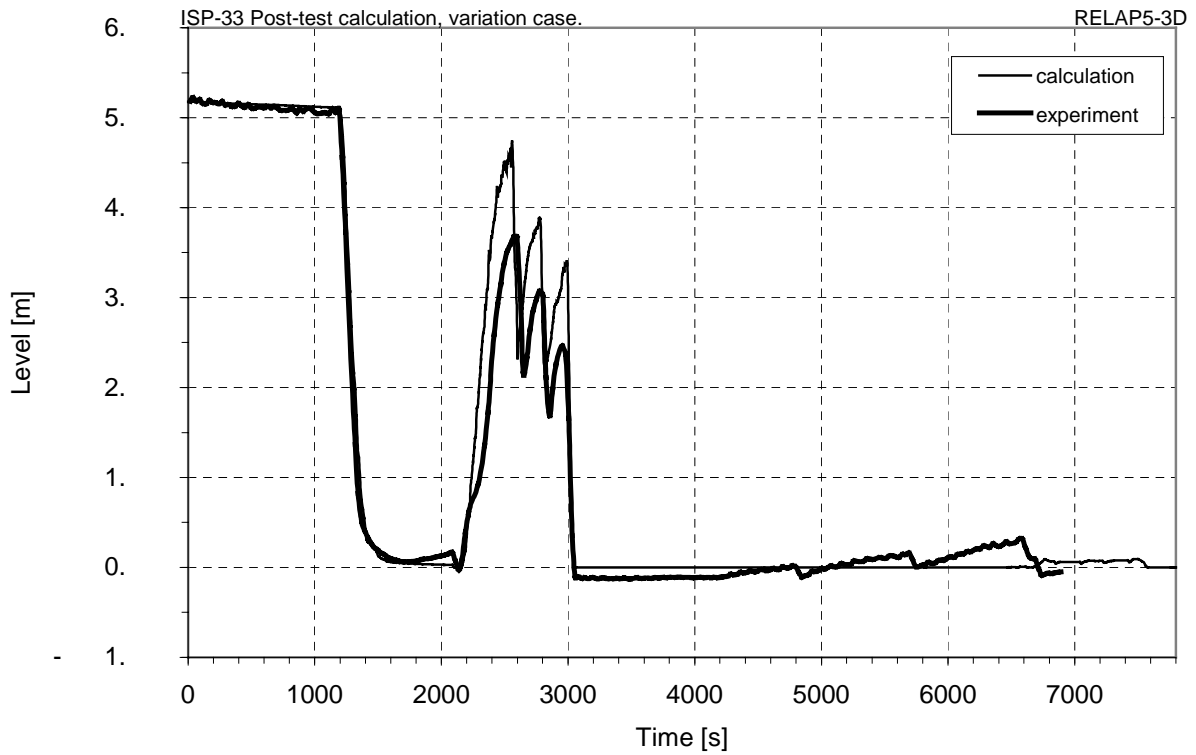


Fig. 4.7. Upper plenum collapsed level.

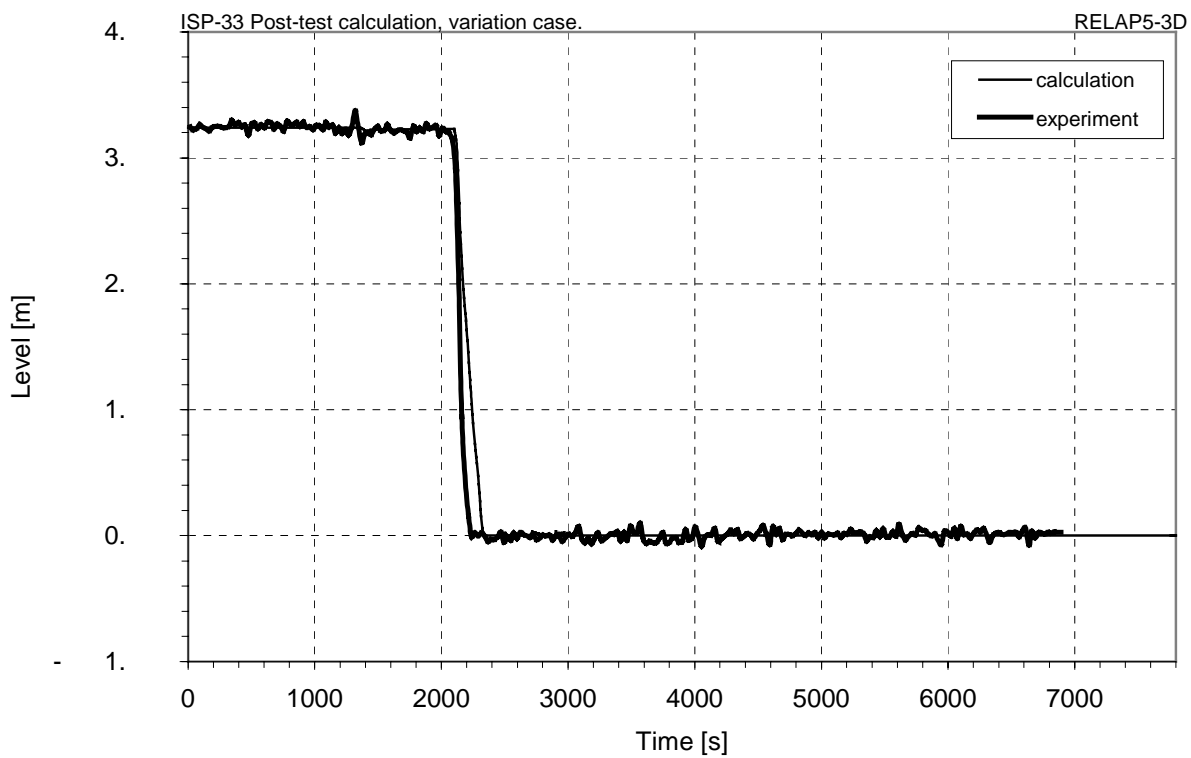


Fig. 4.8. SG1 secondary collapsed level.

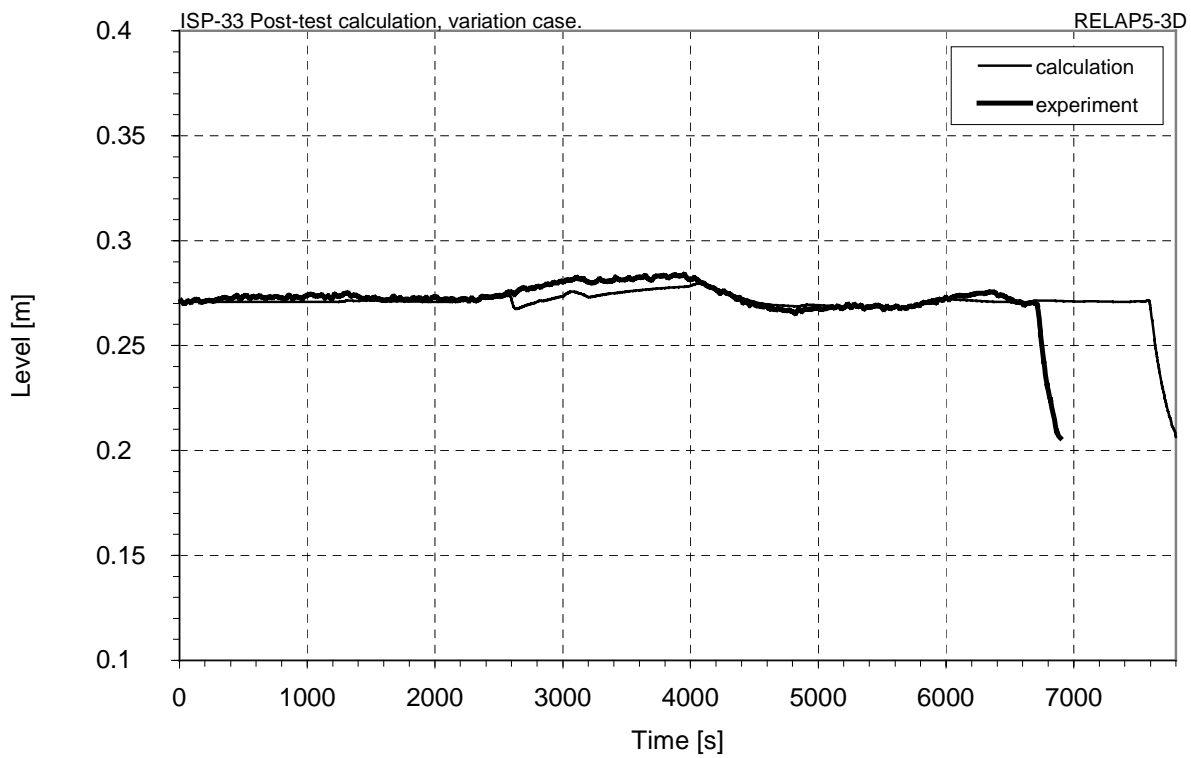


Fig. 4.9. Downcomer mass flow.

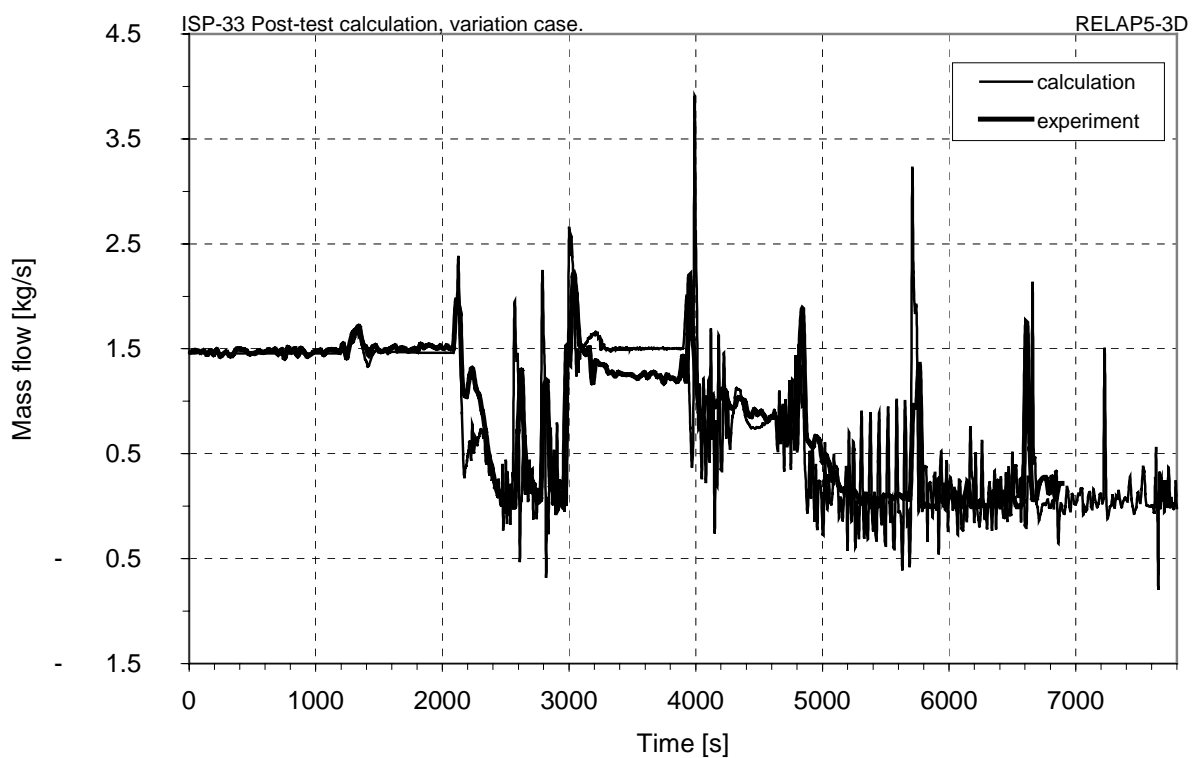


Fig. 4.10. Mass flow through cold leg 1.

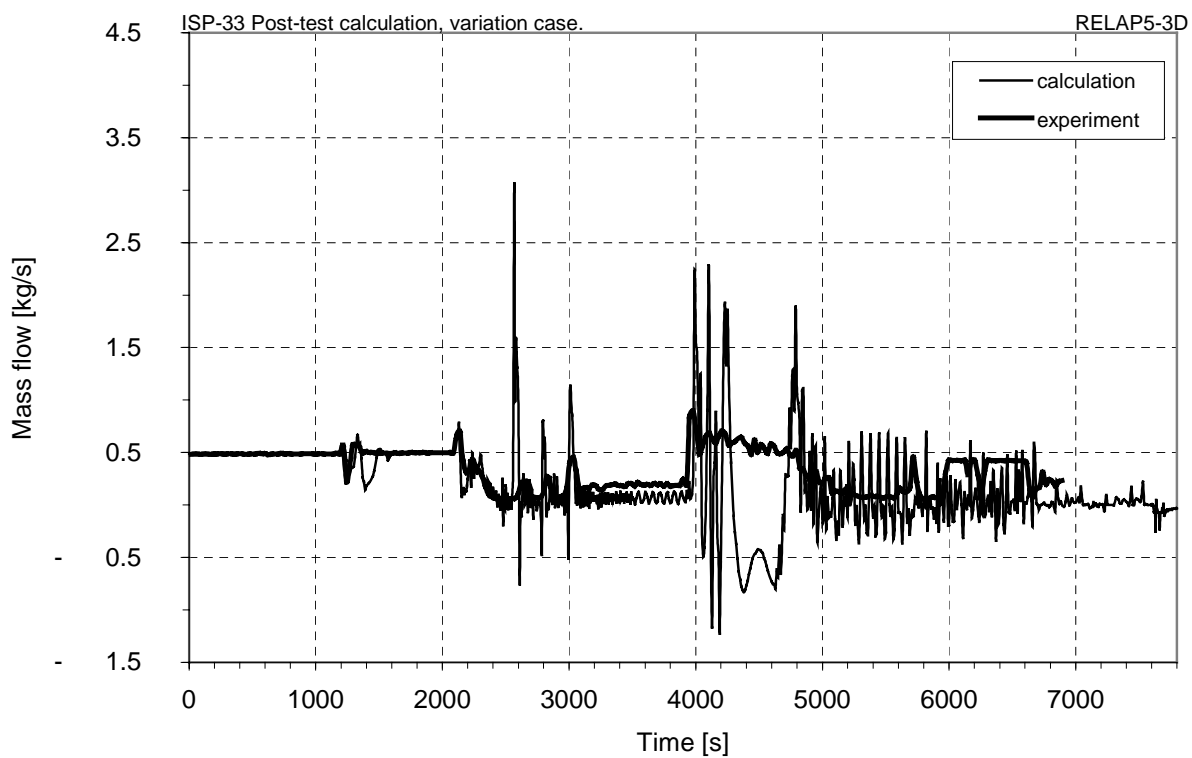


Fig. 4.11. Mass flow through cold leg 2.

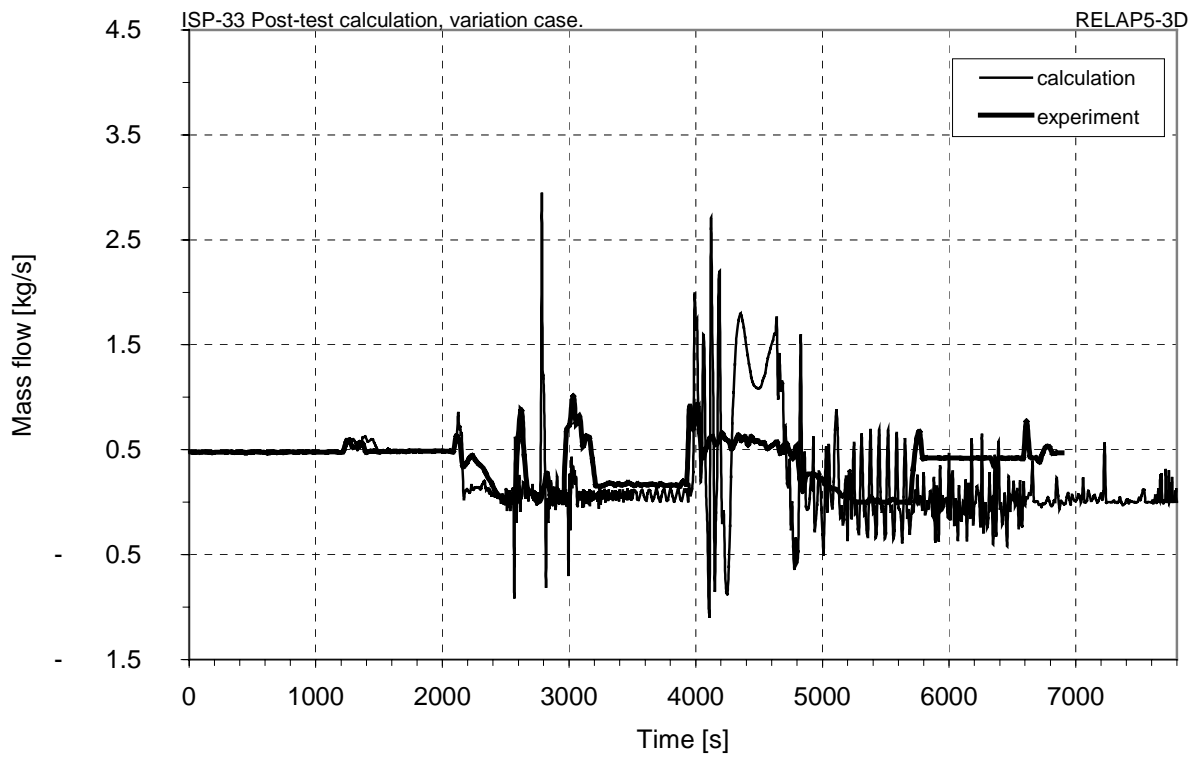


Fig. 4.12. Mass flow through cold leg 3.

