Forschungszentrum Karlsruhe

in der Helmholtz-Gemeinschaft

Wissenschaftliche Berichte

FZKA 6427

Analysis of the OECD/NEA PWR Main Steam Line Break (MSLB) Benchmark Exercise 1 using the RELAP5 Code with the Point Kinetics Option

V. H. Sánchez-Espinoza, W. Hering and A. Knoll*

Institut für Reaktorsicherheit

Programm Nukleare Sicherheitsforschung

*Framatome ANP Erlangen

Forschungszentrum Karlsruhe GmbH, Karlsruhe

2002

Impressum der Print-Ausgabe:

Als Manuskript gedruckt

Forschungszentrum Karlsruhe GmbH
Postfach 3640, 76021 Karlsruhe

Mitglied der Hermann von Helmholtz-Gemeinschaft
Deutscher Forschungszentren (HGF)

ISSN 0947-8620

ABSTRACT

The main purpose of the computational OECD/NEA Pressurized Water Reactor Main Steam Line Break (PWR MSLB) Benchmark is the evaluation of the prediction capability of advanced code systems by means of a code-to-code comparison.

The postulated MSLB-transient is characterized by a strong non-symmetrical core thermal behaviour due to the feedback between neutron kinetics and plant thermal hydraulics. The analysis of such transients with pronounced spatial power distortion represents a considerable challenge for advanced code systems.

The transient is initiated by a double-ended break of one main steam line when the reactor TMI-1 is operated at nominal power. The high heat removal through the break leads to a strong cooldown of the primary coolant. Under such conditions a power increase and a re-criticality of the core despite scram can not be excluded due to the negative reactivity coefficients.

The MSLB-Benchmark enfolds three exercises as follows: Exercise 1: integral plant simulation with best-estimate codes using point kinetics, Exercise 2: multidimensional simulation of the core for given initial and boundary conditions, and Exercise 3: integral plant simulation with coupled, best-estimate codes using 3D-neutron kinetics models.

Das Forschungszentrum Karlsruhe (FZK) and Framatome Advanced Nuclear Power/Erlangen (former Siemens/KWU) participated on the MSLB-Benchmark with the code system RELAP5/MOD3.2 for the Exercise 1: In this report, the integral plant model developed for this Exercise 1 together with the calculated results will be presented and discussed.

Untersuchung der Phase 1 des OECD/NEA DWR Benchmarks zum Frischdampfleitungsbruch mit dem RELAP5-Code unter Verwendung eines Punktkinetikmodells

Zusammenfassung

Ziel des theoretischen OECD/NEA MSLB PWR Benchmarks ist es, die Vorhersagbarkeit fortgeschrittener Codesysteme durch einen Code-zu-Code-Vergleich zu bewerten.

Die postulierte Main Steam Line Break (MSLB) Transiente ist durch ein stark unsymmetrisches thermisches Kernverhalten gekennzeichnet, welches sich aus der Rückkopplung zwischen der Neutronenkinetik und der Kreislauf-Thermohydraulik ergibt. Die Analyse solcher Transienten mit erheblicher Verzerrung der räumlichen Leistungsverteilung stellt hohe Anforderungen an fortgeschrittene Codesysteme.

Es wird angenommen, dass die Transiente durch einen doppelendigen Bruch der Frischdampfleitung bei nominaler Leistung am Zyklusende (EOC) ausgelöst wird. Die starke Wärmeabfuhr über die Leckstelle führt zu einer signifikanten Abkühlung des Primärkreislaufes. Unter solchen Bedingungen ist eine Leistungssteigerung und Rekritikalität des Reaktors trotz Reaktorschnellabschaltung (RESA) nicht auszuschließen.

Der MSLB-Benchmark umfasst drei Phasen wie folgt: Phase 1: Simulation der gesamten Anlage mit der Punktkinetik, Phase 2: dreidimensionale Kernsimulation für festgelegte Anfangs- und Randbedingungen und Phase 3: Integrale Anlagensimulation mit gekoppelten Programmsystemen unter Verwendung von 3D-Neutronenkinetik-Modellen.

Das Forschungszentrum Karlsruhe (FZK) und Framatome Advanced Nuclear Power/Erlangen (früher SIEMENS/KWU) beteiligten sich gemeinsam an dem Benchmark. Die Phase 1 wurde mit dem Code RELAP5/MOD3.2 untersucht. Das dafür entwickelte Modell und die erzielten Ergebnisse werden in diesem Bericht vorgestellt und diskutiert.

TABLE OF CONTENT

LIST OF TABLES

LIST OF FIGURES

LIST OF ABBREVIATIONS

ACC	Accumulator
ANP	Advanced Nuclear Power
CR	Control Rod
DTC	Doppler Temperature Coefficient
EOC	End of Cycle
ECC	Emergency Core Cooling
FA	Fuel Assembly
FZK	Forschungszentrum Karlsruhe GmbH
FR	Fuel Rod
HFP	Hot Full Power
HPI	High Pressure Injection
GT	Guide Tube
INEEL	Idaho National Engineering and Environmental Laboratory, USA
IRS	Institut für Reaktorsicherheit,
KWU	Kraftwerk Union, Erlangen, part of Siemens, Germany,
LWR	Light Water Reactor
LOCA	Loss Of Coolant Accident
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
MSLB	Main Steam Line Break
MTC	Modeartor Temperature Coefficient
NEA	Nuclear Energy Agency
NSC	Nuclear Science Committee
NEM	Nodal Expansion Method
NK	Neutron Kinetics
OTSG	Once-Through Steam Generator
OECD	Organization for Economical Co-operation and Development
PSU	Pennsylvania State University
PSF	Projekt Nukleare Sicherheitsforschung
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
RELAP	Reactor Leak and Analysis Program
RPV	Reactor Pressure Vessel
SRV	Safety Relief Valve
SG	Steam Generator
TRW	Tripped Rod Worth
TMI-1	Three Mile Island Unit 1
TH	Thermal Hydraulics
USNRC	United States Nuclear Regulatory Commission

1 Introduction

FZK participated on the MSLB PWR Benchmark organized by the OECD/NEA to investigate the prediction capability of best-estimate coupled codes in analyzing complex transients with large spatial power distortion and large feedbacks between neutron kinetics and plant thermal hydraulics. This Benchmark is a continuation of former OECD-Benchmarks devoted to 3D-reactor physics problems in both PWR and BWR [Finne92], [Fraik97].

The main goal of this computational Benchmark is to assess both the code's prediction capability and the different coupling strategies implemented in advanced code systems now available for plant applications. These codes are characterized by a direct coupling of plant thermal hydraulics with multi-dimensional neutron kinetics models. Hence its application range and prediction capability were largely extended and improved. To this category of codes belong among others ATHLET-QUABOX/CUBBOX [Lange99], ATHLET-DYN3D-R [Grund95a], RELAP5/PANBOX [Knoll98], RELAP5/PARCS [Miller98], TRAC-M/PARCS [Miller99], TRAC/PF1/NEM [Ivan96], CATHARE/CRONOS/FLICA4 [Royer00].

The postulated MSLB-transient of the TMI-1 plant leads to an overcooling of the primary circuit and to the insertion of large amounts of positive reactivity into the core. Under such conditions, power increase and re-criticality after reactor shutdown may be feasible. How much the maximal values of such global parameters depends on the neutron kinetics model used for its predictions is one of the most challenging questions to be clarified by this Benchmark.

The MSLB-Benchmark was defined by the Pennsylvania State University (PSU) and it enfolds the following exercises:

- **Exercise 1**: Integral 1D plant simulation using the neutron point kinetics.
- **Exercise 2**: 3D simulation of the core neutronics response for given thermal hydraulic boundary conditions at core inlet and outlet.
- **Exercise 3**: Integral 1D plant simulation with a 3D core representation using best-estimate codes.

The main concern analysing the MSLB is the question whether a power increase after reactor scram is predicted or not by the codes due to the considerable positive reactivity insertion throughout the transient.

In this report the investigations performed with RELAP5/MOD3.2 including point kinetics will be presented. A short description of the TMI-1 plant and the developed plant model are also given.

2 Description of the TMI-1 plant

The TMI-1 plant is a two loop PWR with vertical once-through steam generator (OTSG) designed by Babcock&Wilcox (B&W). A detailed description of this plant is given in [Ivan99a] and in [nrcd97].

Figure 2-1 and Figure 2-2 show a vertical and horizontal arrangement of the primary system. A simplified scheme of the secondary coolant system including the relevant steam lines, feed-water system, turbine isolation valves, cross-connection line, steam relief valves, and the location of the breaks is given in Figure 2-3. The break of a steam line in a two-loop reactor like the TMI-1 causes a more pronounced non-symmetrical behaviour than the one in a four-loop plant. Hence the TMI-1reactor was chosen as a reference plant for the MSLB-Benchmark.

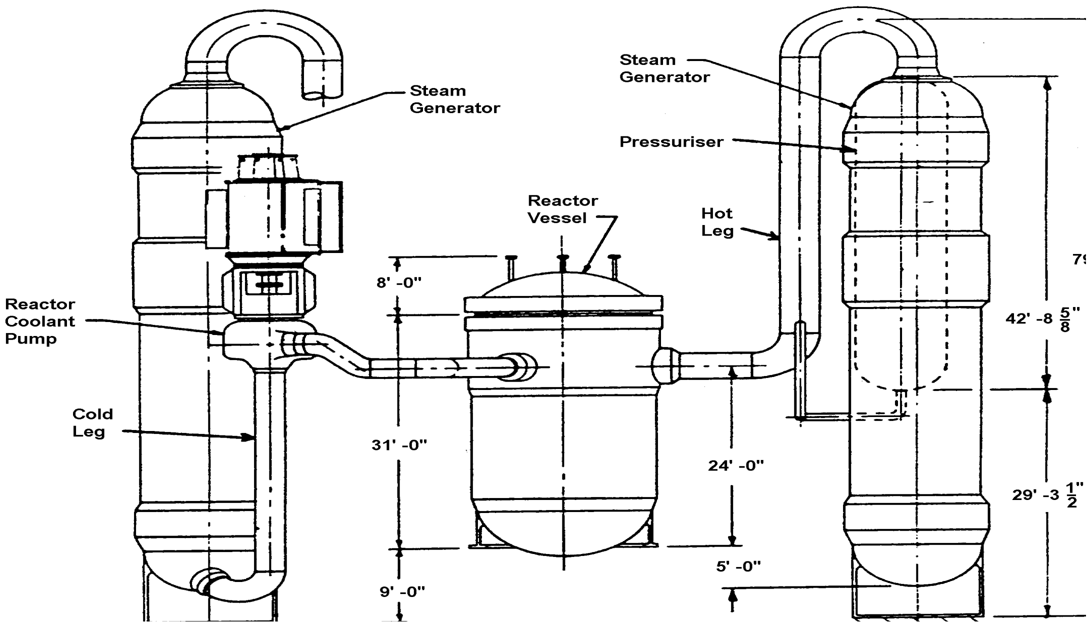

Figure 2-1 Vertical arrangement of the primary reactor coolant system

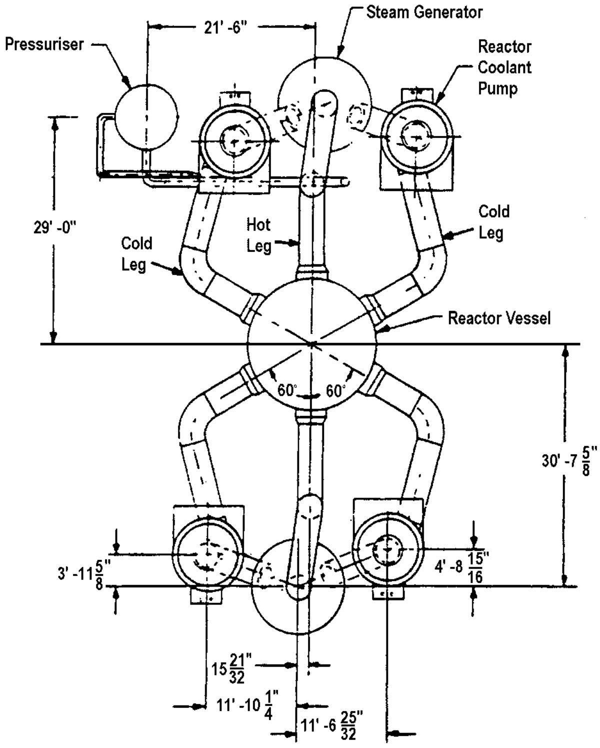

Figure 2-2 Horizontal arrangement of the primary coolant system

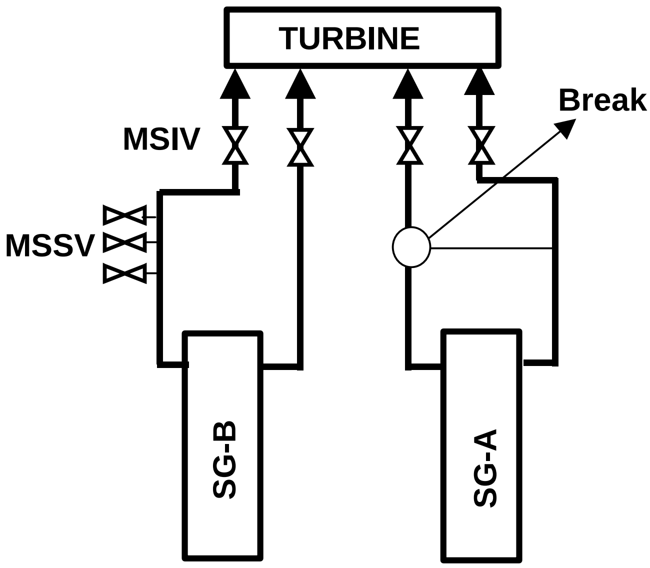

Figure 2-3 Scheme of the main steam system with break position

In Tab. 2-1 main dimensions of relevant core components and of the RPV are presented. The Tab. 2-2 gives a list of important parameters for the nominal operation conditions of the TMI-1 plant for both primary and secondary system.

Tab. 2-1 Dimension of the TMI-1 core

Core Components	**Parameter**	**Value**
Fuel rod (FR)	Pellet diameter (mm) Clad outside diameter (mm) Clad thickness (mm) Fuel rod pitch (mm)	9.391 10.928 0.673 14.427
Control rod (CR)	AIC/SS-clad outside diameter (mm) Number of CR s in core	0.78/0.87 2193
Guide tube (GT)	Inside/outside diameter (mm)	12.65/13.46
Fuel assembly (FA)	Number of FAs in core (15x15) Number of FRs per FA Number of CRs per FA Number of GTs per FA	177 208 16 16
Reactor pressure vessel (RPV)	Active height (mm) inner diameter (mm)	3657.6 4340

Tab. 2-2 Main parameters of TMI-1 at hot full power (EOC)

Primary circuit parameters	**Value**
Total core power (MW_{th})	2772
Total reactor coolant system (RCS) mass flow (kg/s)	17602
Core mass flow (kg/s)	16052
Bypass mass flow (kg/s)	1550
Coolant temperature at core inlet (K)	563.76
Coolant temperature at core outlet (K)	591.4
Pressure drop over the core (MPa)	0.20
RCS pressure drop (MPa)	0.738
RCS-hot leg pressure (MPa)	14.9
Pressure at the lower plenum (MPa)	15.36
Pressure at the upper plenum (MPa)	15.17
Pressurizer water level (m)	5.588
Secondary circuit parameters	
Secondary side net volume (m^3)	96.62
Steam generator outlet pressure (MPa)	6.42
Steam mass flow per loop (kg/s)	760
Main feed water flow per loop (kg/s)	760
Steam superheat at SG-outlet (K)	19.67

3 MSLB-scenario and assumptions

The initial event is the double-ended break of one main steam line at the tie-line with the cross-connection line. It occurs when the reactor is operated at nominal power at EOC. The break location implies a small and a large break. The area of the large break amounts 0.2465 m² (100 % of steam line) while the one small break is only 0.0324 m² (100 % of cross-connection line, see Figure 2-3).

After break opening large heat removal over the broken SG takes place due to the fast de-pressurization and consequently boil-off of the fluid inventory in the broken SG-secondary so that the core undergoes a large non-symmetrical overcooling. Since both moderator and Doppler reactivity coefficients are negative, positive reactivity is added to the core even though the reactor is shut down a few seconds after break opening.

Several assumptions were made for the analysis of the MSLB-transient according to the US-licensing criteria in order to maximize the heat removal through the breaks. Some selected assumptions extracted from the Benchmark specifications [Ivan99a] follows:

- Maximal fluid inventory on the secondary SG-side of about 26 tons.

- The break is located at the cross-connect position, upstream of the MSIV (maximization of the break outflow).

- As worst single failure the stuck-open of the feed water regulation valve was assumed. Hence feed water from the intact SG crosses over to the broken SG through the common header. Feed water flow to the broken SG ends 30 s after transient initiation by closure of the feed water block valve.

- About 16 tons of feed water collected in the feed water pipes enters into the broken SG down-comer during the accident progression.

- The feed water injection to the intact SG is kept constant until scram, then it is ramped to zero within 10 s.

- The turbine stop valves of the broken loop-A close just after break opening while those of the intact loop-B are closed after scram.

- All primary pumps remain in operation during the whole transient.

- Only two high pressure injection pumps are available for injection to the corresponding cold legs with 25 seconds delay. No other ECCS-action is expected during the 100 s transient time.

- The boron negative reactivity contribution is considered negligible due to assumed EOC conditions.

  - No emergency feed water system is expected to actuate.

  - Coolant mixing in the lower and upper plenum is defined based on measured data from the Oconee NPP [Ivan99a] as follows:

    - lower plenum mixing: 20 %, and

    - upper plenum mixing : 80 %.

- The most effective control rod group is assumed to remain out of the core when scram is activated (stuck-rod assumption).

- The EOC is characterized by a low boron concentration of about 5 ppm and an equilibrium concentration of Xenon and Samarium.

A complete list of initial and boundary conditions for this scenario is documented in [Ivan99a].

4 Model of the TMI-1 plant

An elaborated integral TMI-1 plant model was based on an input deck developed by INEEL for SCDAP/RELAP5 to simulate the TMI-2 accident. This model was extensively modified and extended for the MSLB-Benchmark analysis. The developed integral plant model is exhibited in Figure 4-1. A short description of the plant model follows.

4.1 Reactor pressure vessel and core

The plant behaviour in the course of the MSLB-transient demands a splitting of RPV and core to correctly simulate the mixing of coolant in the lower and upper plenum since the fluid temperature of the broken cold leg will considerably diverges from those of the intact cold leg during accident progression. Hence the RPV and the core were entirely split into two halves, one connected to the intact and the other one to the affected loop.

Two main coolant streams are modelled within the RPV. Each stream consists of downcomer, lower plenum, core, bypass, and upper plenum, see Figure 4-2. The main flow path in the upper plenum is determined by constructive peculiarities of TMI-1 plant, such as core support shield, baffle with large holes at the baffle top and a support plate that separates the upper plenum from the upper head.

The baffle and the core support shield form two annular regions. In addition a dead-end region is formed between the core support shield and the RPV-wall. Guide tubes are also considered as pipes in the model connecting the upper head with the core exit.

A core half consists of two flow channels with a flow area of 1.141 m^2 . Each flow channel has 10 volumes each with a length of 0.36576 m. Both flow channels of a core half are connected by cross-flow junctions at each axial node but there is no cross connection between the two core halves. Each core flow channel is coupled with a heat structure representing 25 % of the core fuel rods.

Other in-vessel structures like core barrel, thermal shield, core support shield as well as the RPV-wall are included in the model as RELAP5 heat structures.

4.2 Coolant mixing

The coolant mixing within the RPV is of great importance for the prediction of main parameters in case of the MSLB-transient. Hence the mixing was prescribed in the specifications by means of the definition of a conservative coolant mixing factor of 0.5, see Equation (4.1). This represents a 20 % coolant mixing in the lower and 80 % coolant mixing in the upper plenum. These values are based on experimental data obtained for the Oconne NPP which is of the same type as the TMI-1. In the equation (4.1) the mixing factor is defined by the ratio of the coolant temperature difference of hot legs to the coolant temperature difference of cold legs. This was considered in the model by the introduction of two junctions in both plena allowing a coolant mixing, so that R=0.5 is attained during the whole transient.

The definition of appropriate flow areas and friction loss coefficients for these junctions were important to achieve the equation (4.1).

$$R = \frac{[T_{hot}(itc) - T_{hot}(brk)]}{[T_{cold}(itc) - T_{cold}(brk)]} = 0.5 \qquad (4.1)$$

With ***itc*** for intact and ***brk*** for broken loop.

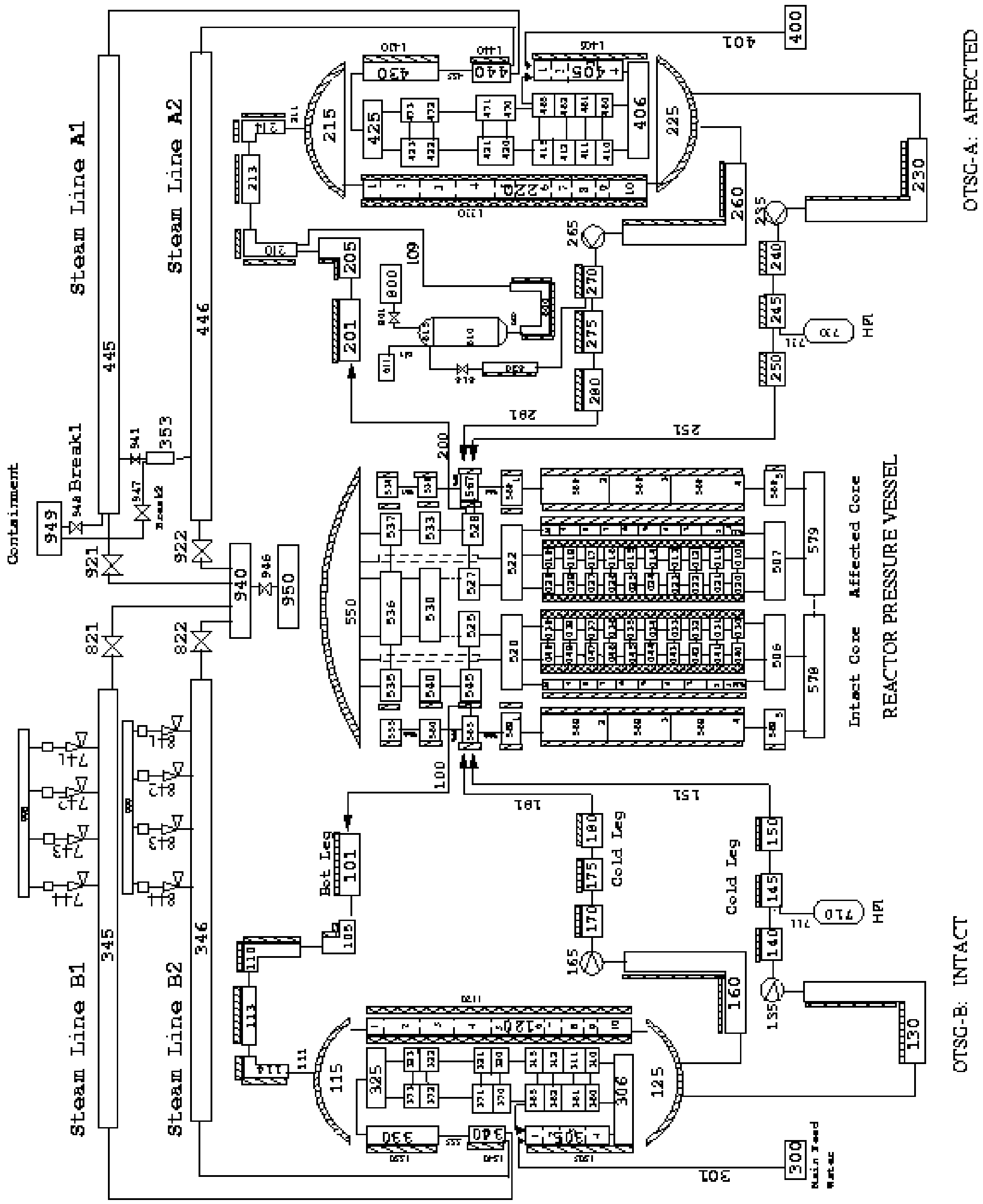

Figure 4-1 TMI-1 integral plant model for RELAP5/MOD3.2

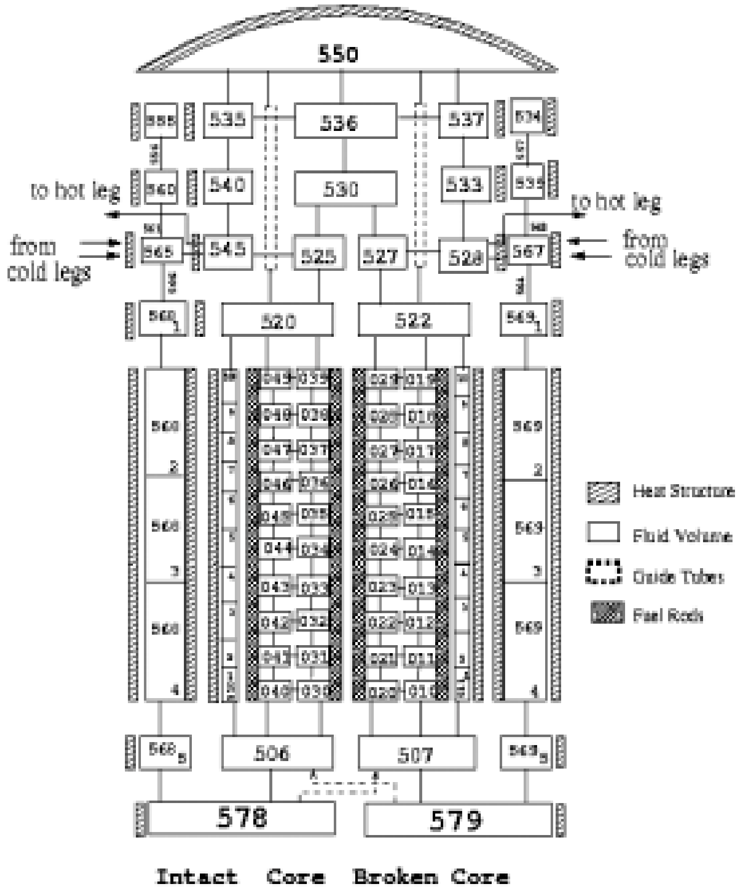

Figure 4-2 The RPV-splitting model

4.3 Fuel rod model

The 36816 fuel rods are modelled as four representative heat structures, each one representing a quarter of the core (9204 fuel rods). The representative heat structure consists of 10 axial nodes and 5 radial zones i.e. two zones (pellet), one zone (gap), and two zones (cladding) as sketched Figure 4-3. With the constant gap conductance of 11356 W/(m²K) prescribed in the specifications an effective gas gap conductivity of 1.0632 W/(m K) was used in RELAP5 to predict the heat transfer across the gap.

The control rods and guide tubes are also modelled as heat structures similarly to fuel rods. The representative heat structures for fuel rods, absorber, and guide tubes corresponding to a quarter of the core are coupled with the core thermal hydraulic channels by convective boundary conditions.

The axial power profile at EOC, Figure 4-4 is fixed in the fuel rod/ point kinetics model. It is considered to be constant during the whole transient.

No radial power peaking factor was considered in this model since no hot channel analysis was performed in the frame of this work.

An especial definition of an artificial “Doppler” temperature, T_f, was introduced in [Ivan99a] to calculate the Doppler feedback as follows:

$$T_f = 0.3 \cdot T_{f,c} + 0.7 \cdot T_{f,s} \quad , \tag{4.2}$$

where $T_{f,s}$ is the pellet surface temperature and $T_{f.c}$ is the pellet centre line temperature.

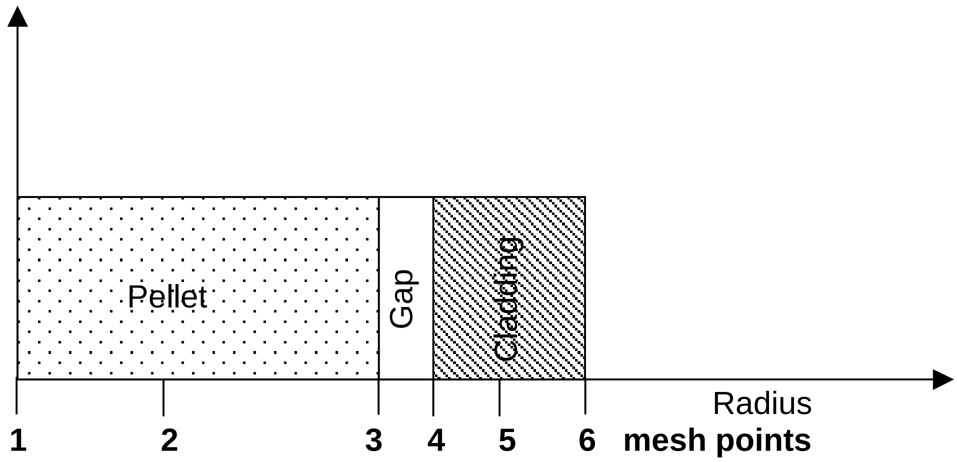

Figure 4-3 Radial discretization of the fuel rod pin

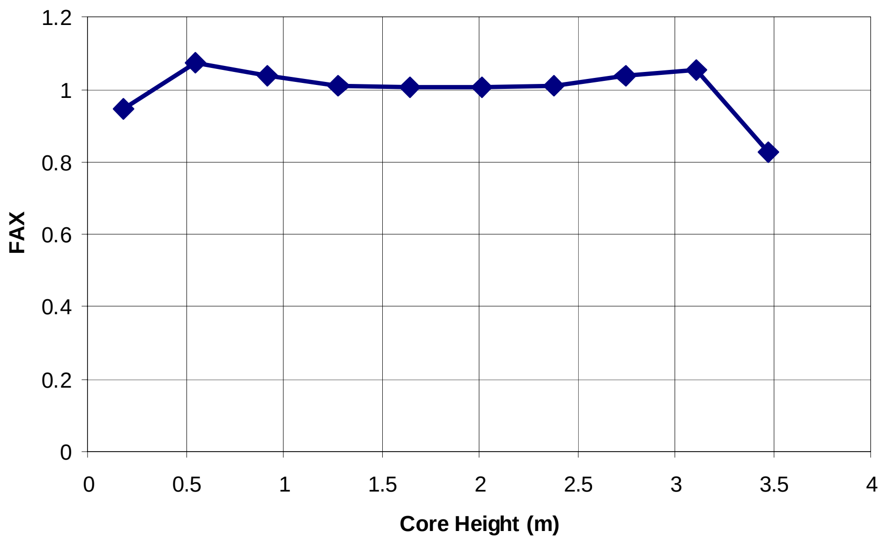

Figure 4-4 Relative axial power profile at EOC

4.4 Point kinetics model

A point kinetics model was developed based on the data from the specification [Ivan99a]. In Tab. 4-1 and Tab. 4-2. relevant global neutron kinetics parameter are listed. The moderator and Doppler reactivity coefficients were calculated with the RELAP5-separable model using the core averaged, relative axial power distribution for axial weighting.

Tab. 4-1 Global neutron kinetics parameters for the TMI-1 core

Parameters	Value
Moderator temperature coefficient ($/K)	-0.011967
Doppler temperature coefficient ($/K)	-0.004933
EOC tripped rod worth ($)	8.686
Total fraction of delayed neutron, $ß_{eff}$ (-)	0.005211
Prompt neutron lifetime (s)	0.0000184

Tab. 4-2 Decay constants and fraction of delay neutrons

Group (i)	Decay constants (s^{-1})	Relative fraction of delay neutrons (%)
1	0.012818	0.0153
2	0.031430	0.1086
3	0.125062	0.0965
4	0.329776	0.2019
5	1.414748	0.0791
6	3.822362	0.0197

4.5 Steam generator

The steam generator model is exhibited in Figure 4-5. On the right side the steam generator tubes are drown from the primary circuit (120). The secondary side was represented in some detail. It contains the down-comer (305), lower plenum (306), boiler and outlet-volume (325). The inclusion of the aspirator in this model is important to correctly predicting the fluid inventory. The major function of the aspirator is to suck superheated steam from the boiler (at 9.73 m elevation) and to re-inject it into the down-comer top so that the feed water temperature at the boiler entrance attains a defined value.

The 15500 Inconel tubes of 15.6 m height were grouped in two representative heat structures (13913/1587 tubes) and linked to both the primary and the secondary side (two boilers), respectively. Each representative heat structure consists of 10 axial and 5 radial nodes. In addition the steam generator wall was modelled.

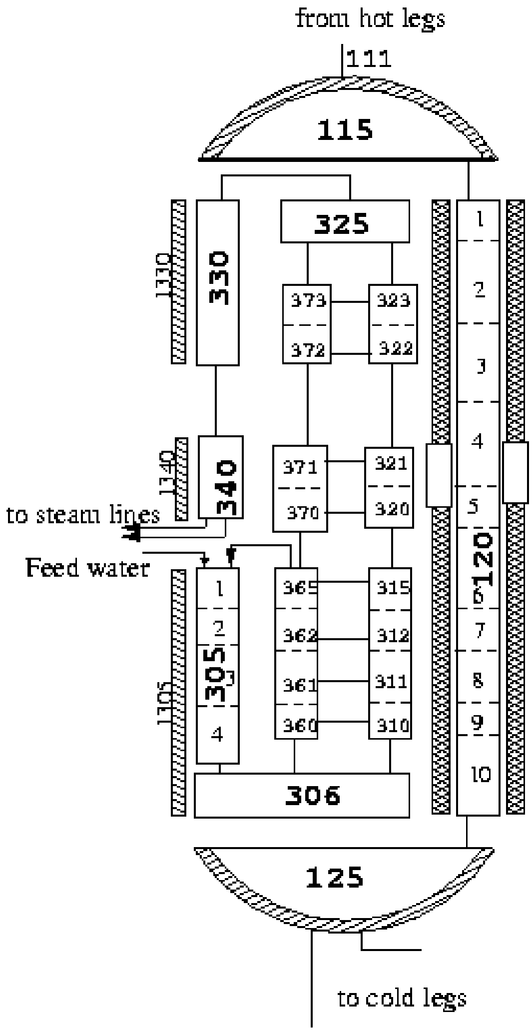

Figure 4-5 Nodalization of the once-through steam generator

4.6 Main feedwater and steam system

A time dependent volume and junction represent the feedwater tank and the injection line into the steam generator downcomer in a simplified manner. On the contrary the steam system including steam lines, cross-connection line, main steam safety valve (MSSV), main steam isolation valve (MSIV), turbine stop valves, common header and turbine are modelled in more detail. The two steam lines of the intact loop-B were merged into one neglecting the cross-connection line. But all MSSV are included in the model with their corresponding set-points for opening and closure. While the MSIV were modelled as simple trip valve, the MSSV were represented as motor valves. Moreover a common header and a simple turbine model are part of the integral plant model. For steady state calculations the secondary side pressure boundary conditions were fixed in the turbine volume.

4.7 Emergency core cooling system

Two HPI-systems are included in the model as a time dependent volume and junction, see Figure 4-1. The pressure dependent injection rate of each HPI-pump is given in [Ivan99a]. The condition for cold water injection by the HPI-pump is fulfilled when the system pressure falls below 11.34 MPa. Then injection starts with a delay time of 25 s.

4.8 Breaks and containment

The break location at the cross-over point between the main steam line and the cross-connection line implies the consideration of two breaks in the model. Consequently a small break in the cross-connecting line and a large break with an area of 0.032 m^2 and 0.246 m^2 were modelled. A trip valve was chosen to represent the breaks. The Ramson&Trapp critical flow model (default option in RELAP5/MOD3.2), homogeneous model i.e. single-velocity momentum equations and the default values for the discharge coefficients were chosen to model the chocked flow. The containment is represented by a large time dependent volume at atmospheric conditions. Pressure feedback was neglected.

4.9 Reactor control system

Several trips are considered in the model to simulate the prescribed actions during the transient, among others 1) reactor scram, 2) control rod insertion, 3) HPI-pump injection, 4) SG-isolation, 5) closure/opening of MSSV, 6) closure of MSIV, and 7) opening of the breaks.

5 Steady-state calculations

The main goal of steady-state calculations was to predict steady-state plant parameters as close as possible to the ones given in the final specifications. A crucial issue for RELAP5 and other the codes in the exercise was the prediction of the prescribed fluid inventory on steam generator secondary side. This value was artificially increased up to 26 tons to maximize overcooling of the primary system during the transient. Consequently, additional model adjustments e.g. reduction of aspirator flow area, increase of the boiler heated diameter were necessary for a successful steady-state calculation. In Tab. 5-1 a comparison of the parameters predicted by RELAP5 with parameters defined in the specifications is listed. It can be seen that the RELAP5-predictions are very close to values given in the specification.

Tab. 5-1 Comparison of RELAP5-predictions with Specification

Steady state parameters	**Specification** **(Referenz)**	**RELAP5** **(Steady state)**	**Deviation (%)**
Core power, MW_{th}	2772	2772	0
Coolant temperature of RCS cold leg, K	563.76	563.72	0.0071
Coolant temperature of RCS hot leg, K	593.43	591.1	0.3926
Lower plenum pressure, Mpa	15.36	15.40	- 0.2604
Upper plenum pressure, Mpa	15.17	15.22	- 0.3296
RCS pressure, Mpa	14.96	14.97	- 0.0668
Total RCS flow rate, kg/s	17692	17768	- 0.4296
Core flow rate, kg/s	16052	16160	- 0.6728
Bypass flow rate, kg/s	1549	1494	3.55
OTSG outlet pressure, Mpa	6.41	6.39	0.312
OTSG outlet superheating, K	19.67	18.9	0.3915
Initial OTSG-fluid inventory, kg	26000	26491	- 1.88

6 Transient calculations

6.1 Boundary conditions

An important boundary condition for the MSLB-transient is the feedwater injection into broken and intact steam generator after break opening. The FW injection rate into the broken SG is presented in Figure 6-1. It was assumed that the feedwater regulating valve fails in open position and that the closure of the FW block valve terminates the feeding of the broken SG after 30 s. Hence in Figure 6-1 can be seen that the injection rate rapidly increases during the first 10 s and then goes down until 30 s. Later on a constant FW injection follows lasting for 12 s and finishing around 45 s. This corresponds to the assumption that all FW mass (16 tons) collected between the FW-isolation valve and the downcomer of the broken SG is fed into the broken SG, maximizing the overcooling. On the other hand the FW flow into the intact SG is held at nominal values until scram and then it is ramped to zero in 10 s.

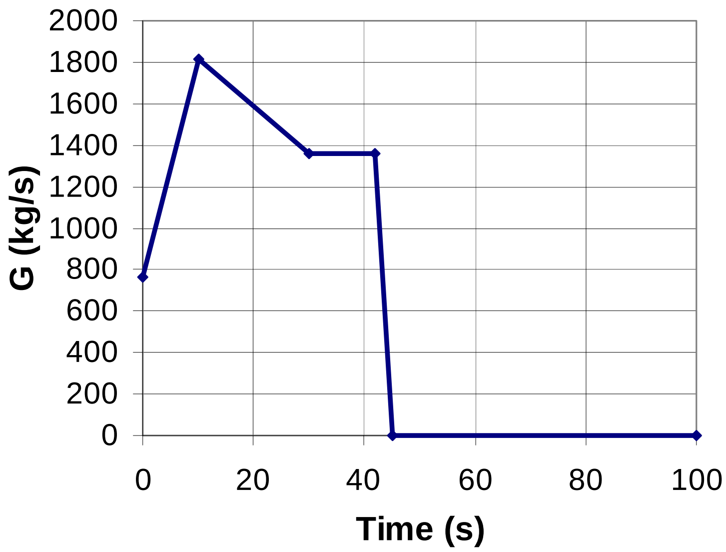

Figure 6-1 Feeding of broken SG considering additional feed water

6.2 Discussion of results

The main events predicted with RELAP5 using the point kinetics model is given Tab. 6-1. The transient is initiated at time zero followed by the break opening. The broken SG is immediately isolated by closing of the MSIV while the MSIV of the intact loop-B remains open until 6.6 s.

Tab. 6-1 Predicted sequence of main events

Sequence of events	**RELAP5/MOD3.2**
Break initiation	0.00 s
Broken loop-A turbine trip (MSIV close)	0.5 s
Overpower TRIP (P>114 %)	5.68 s
Control rods insertion	6.08 s
Intact loop-B turbine trip Intact loop-B Turbine valve (MSIV) close	6.08 s 6.58 s
MSSV-Bank 1 (open/close) MSSV-Bank 2 (open/close) MSSV-Bank 3 (open/close) MSSV-Bank 4 (open/close)	7.63 / 36.86 s 7.97 / 33.86 s 7.97 / 34 s 8.2 / 31.8 s
HPI-system trip (p<11.34 MPa) HPI-system injection (25 s delay time)	11.69 s 36.7 s
PZR-empty	31 s
Re-criticality reached	No (- 0.0176 $ at 71 s)
Max. Power after SCRAM	20.5 % of nominal
End of transient calculation	100 s

A sudden de-pressurization of the secondary side of the broken loop-A takes place after break opening, Figure 6-2. The flushing process leads to large losses of fluid through both breaks, see Figure 6-3, and Figure 6-4. Initially large outflow rates are predicted, which decrease continuously in time reaching very low values around 77s when the SG-secondary side becomes voided. At about 30 s a pronounced increase of the break outflow rate begins, reaching a maximum at around 45 s. This is due to the additional injection of about 16 tons of feedwater according to the assumed boundary condition, see Figure 6-1. This considerably affects the amount of reactivity feedback and the power peak after scram.

Due to the intensive evaporation, the heat removal over the broken SG is very effective and much larger than that over the intact SG. The heat power removed over both SGs is compared in Figure 6-5. Note that heat removal over the intact SG-B remains for few seconds constant; later on it goes rapidly down becoming negative at about 30 s. A reverse heat transfer occurs for about 50 s stabilizing at the end of the transient around very small heat transfer rates. This non-symmetrical cooldown of the primary circuit led to non-uniform coolant temperature of both cold and hot loops despite consideration of coolant mixing in both plena, see Figure 6-6. The different cooldown trends of the intact and broken core halves are also reflected in the volume average fuel temperature, see Figure 6-7.

As consequence of the significant overcooling of the primary circuit a large amount of positive reactivity is added to the core after reactor scram, Figure 6-8, since both moderator and Doppler reactivity coefficients are negative. At around 70 s a core reactivity of about -0.017 $ is reached which corresponds to the minimal coolant temperature reached during the transient. A maximum power peak of about 560 MW_{th} (20 % of nominal power) is predicted.

In Figure 6-9 the total power is expressed as sum of fission power and decay heat.

Later on, since the SG-A becomes empty (Figure 6-10 and Figure 6-11), no effective heat removal takes place and consequently the coolant temperature begins to increase. Hence the positive reactivity feedback stops and power starts to decrease, too. At the end of the transient the reactor achieved stable sub-critical conditions (Figure 6-8).

The fluid inventory of both SGs is presented in Figure 6-11. It can be observed that the inventory of the intact SG remains constant until the respective turbine is isolated. Then it increases due to the decreasing FW-injection within 10 s after reactor trip. Later on the inventory is slightly reduced by opening of the MSSVs.

The pressure in the intact SG increases after closing of MSIV after reactor scram, see Figure 6-12. After reaching a maximum value, it decreases due to the opening of the MSSVs. The predicted outflow rates of the MSSVs are shown in Figure 6-13.

Due to the very efficient heat removal through the broken SG during the transient progression, the primary coolant temperature reduces and thereby the system pressure goes continuously down, Figure 6-14. The PZR-pressure also follows this trend. At about 11 s the primary system pressure falls below the set point for activation of the HPI-pumps initiating with delay of 25 s, see Figure 6-16. In the course of the transient the PZR-pressure follows qualitatively the pressure of the primary circuit with a reduced gradient until it becomes empty at around 31 s, see Figure 6-15.

In Figure 6-17 the coolant mixing ratio predicted by RELAP5 is plotted. It attains an average value close to 0.5 during the transient. It greatly affects the heat removal and thus the power peak and the maximal reactivity insertion into the core.

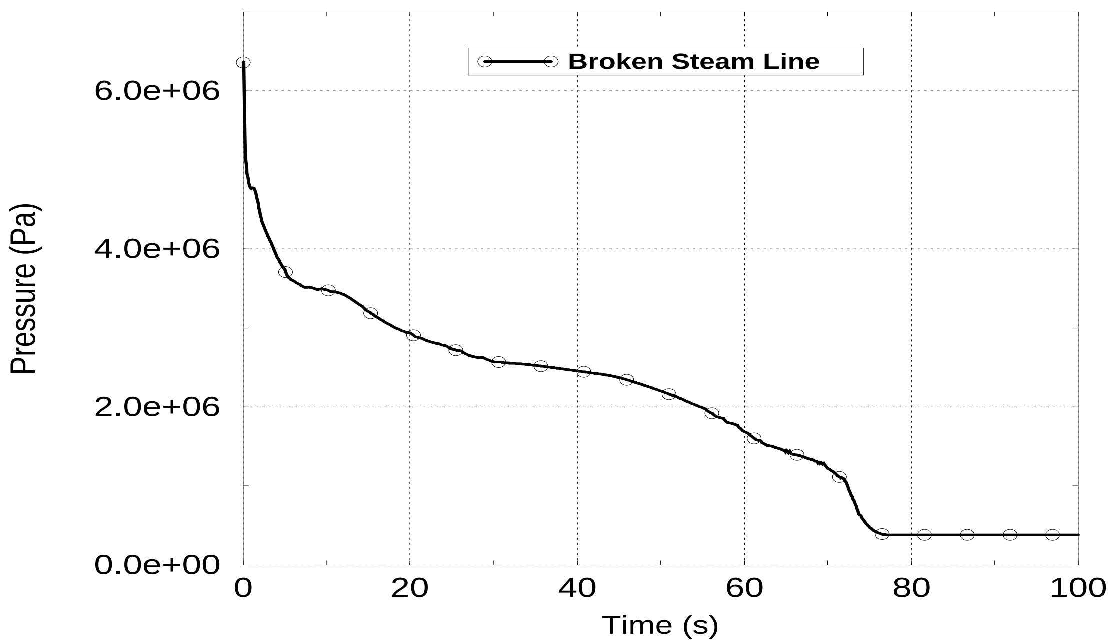

Figure 6-2 Secondary side pressure of broken loop-a

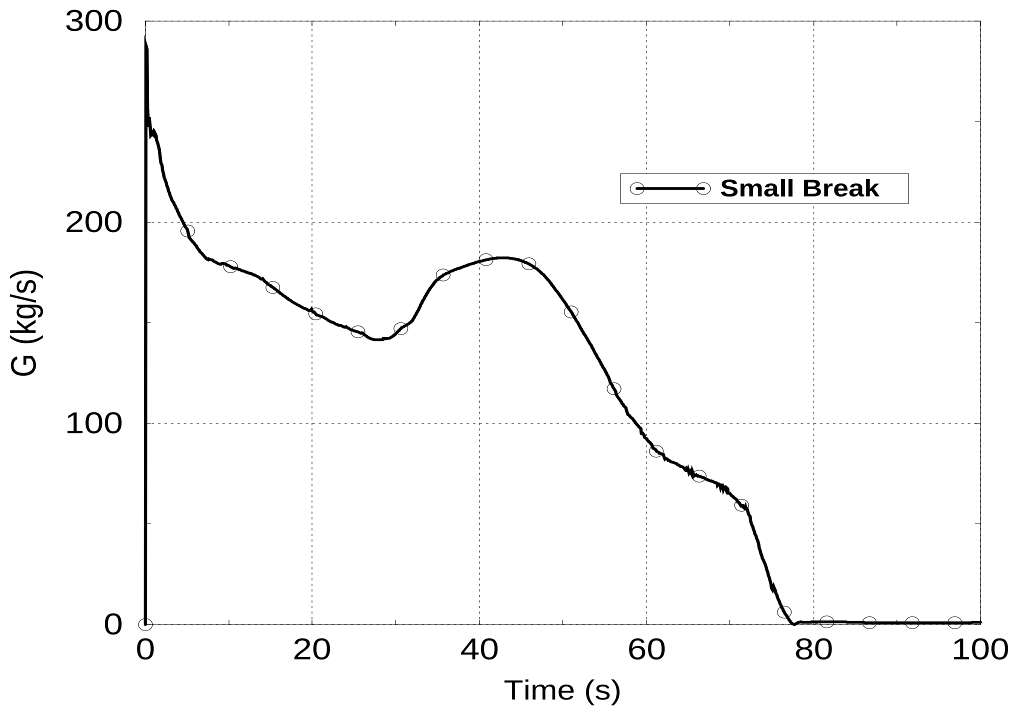

Figure 6-3 Fluid outflow through the small break (cross-connection line)

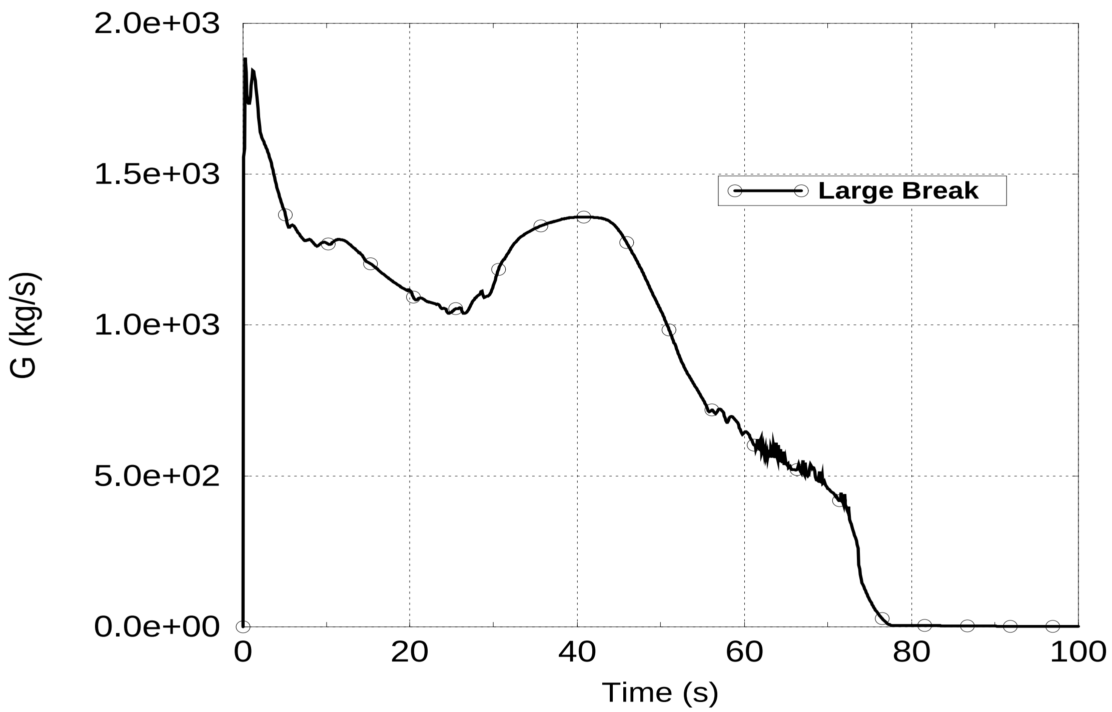

Figure 6-4 Fluid outflow through the large break (main steam line)

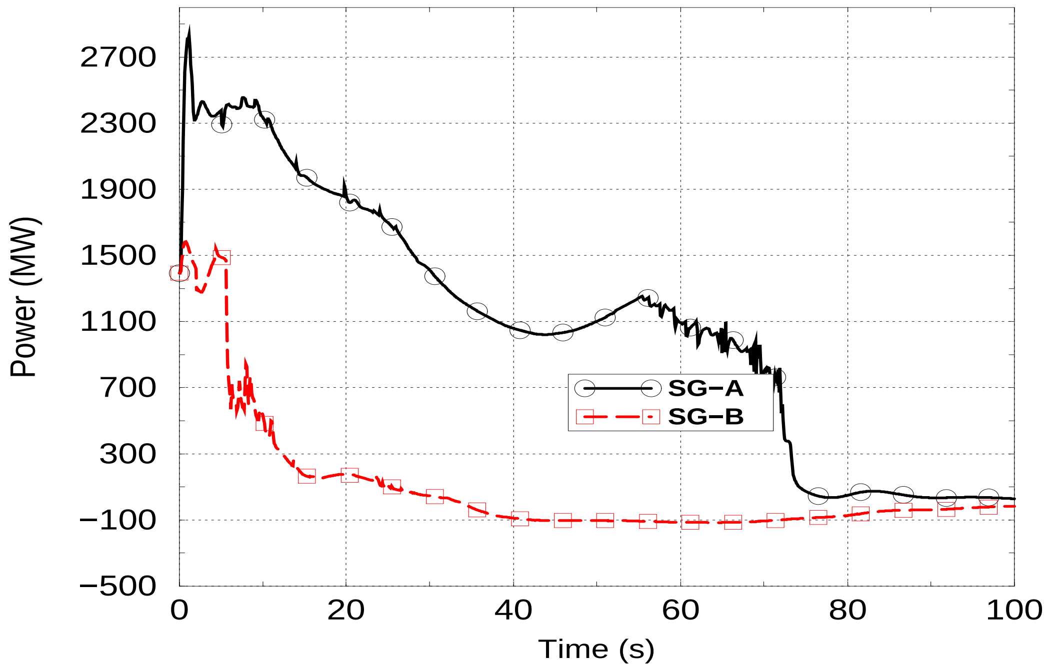

Figure 6-5 Total power transferred to secondary system for both loops

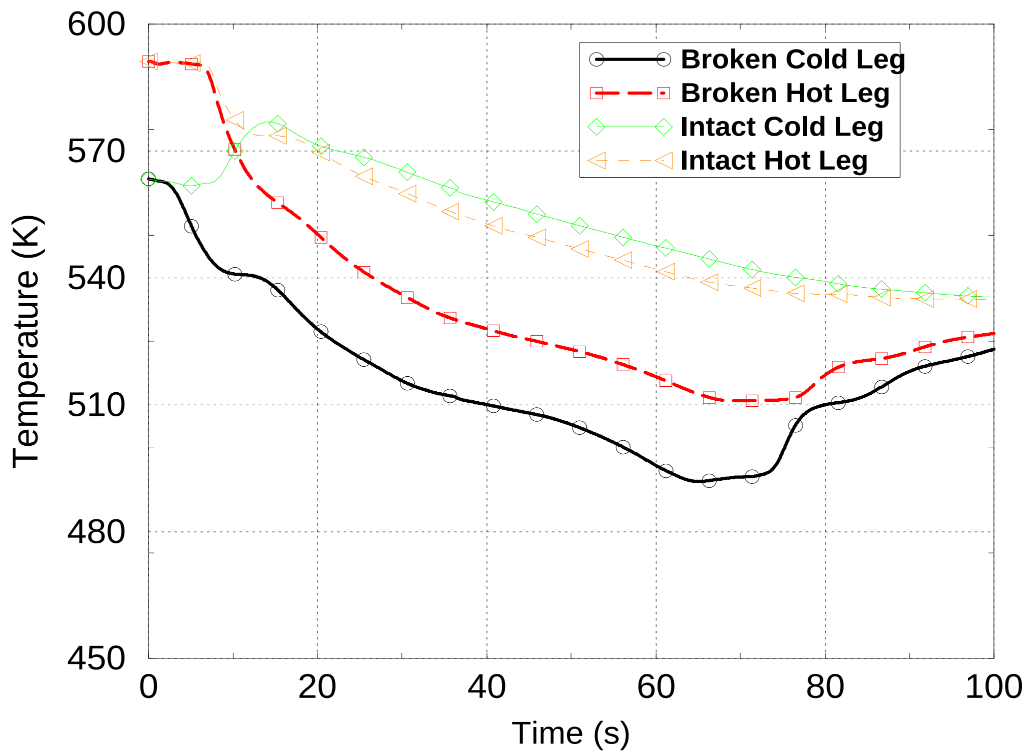

Figure 6-6 Coolant temperature of cold/hot legs of both loops

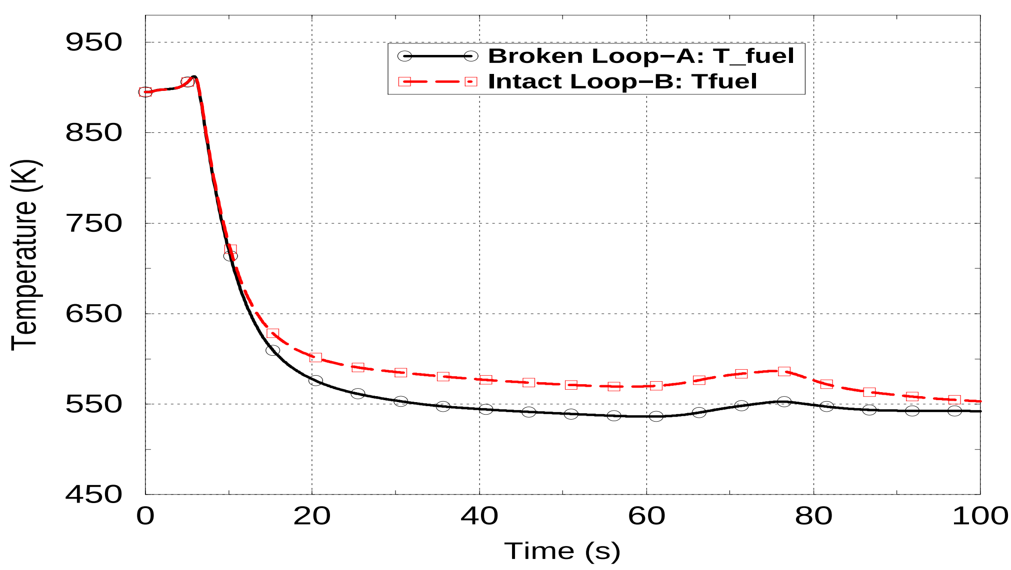

Figure 6-7 Volume average fuel temperature of both representative fuel rods

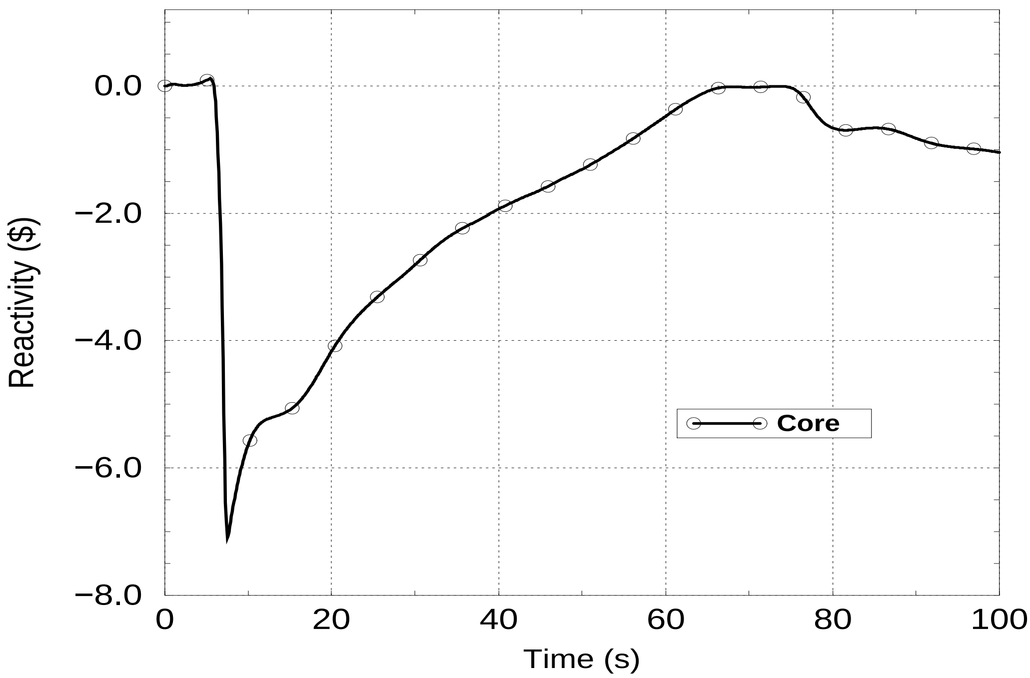

Figure 6-8 Total core reactivity

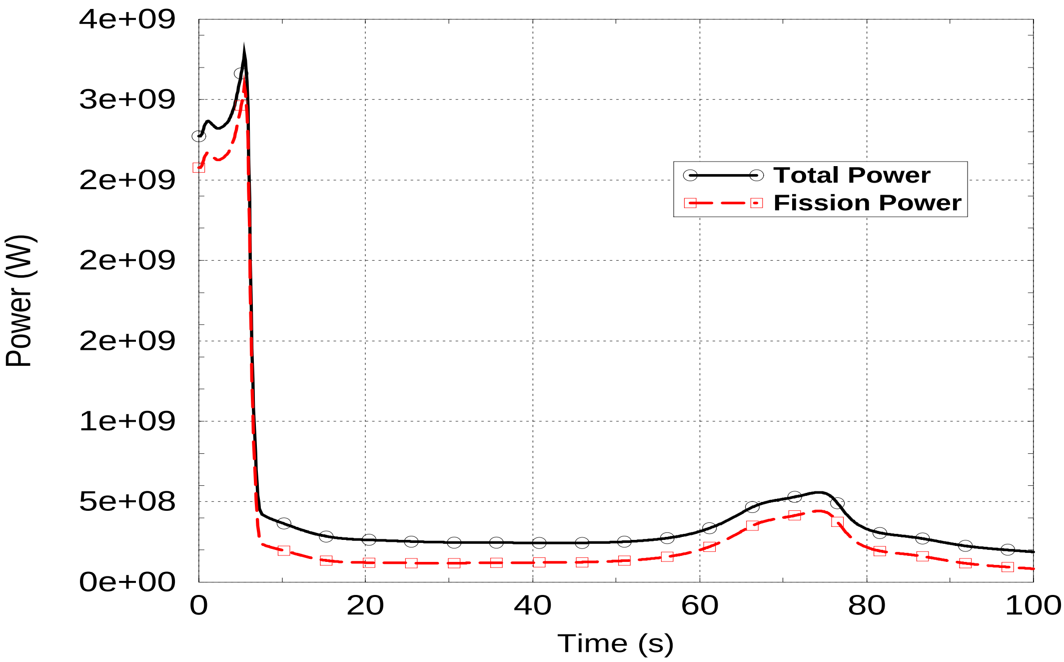

Figure 6-9 Total and fission power

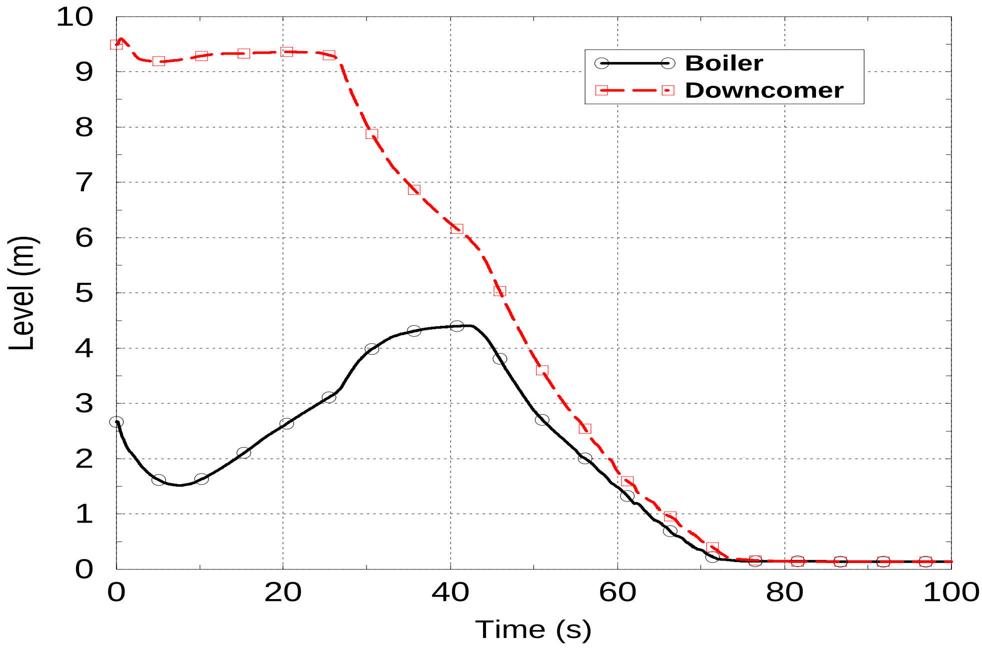

Figure 6-10 Water level of the broken steam generator secondary side

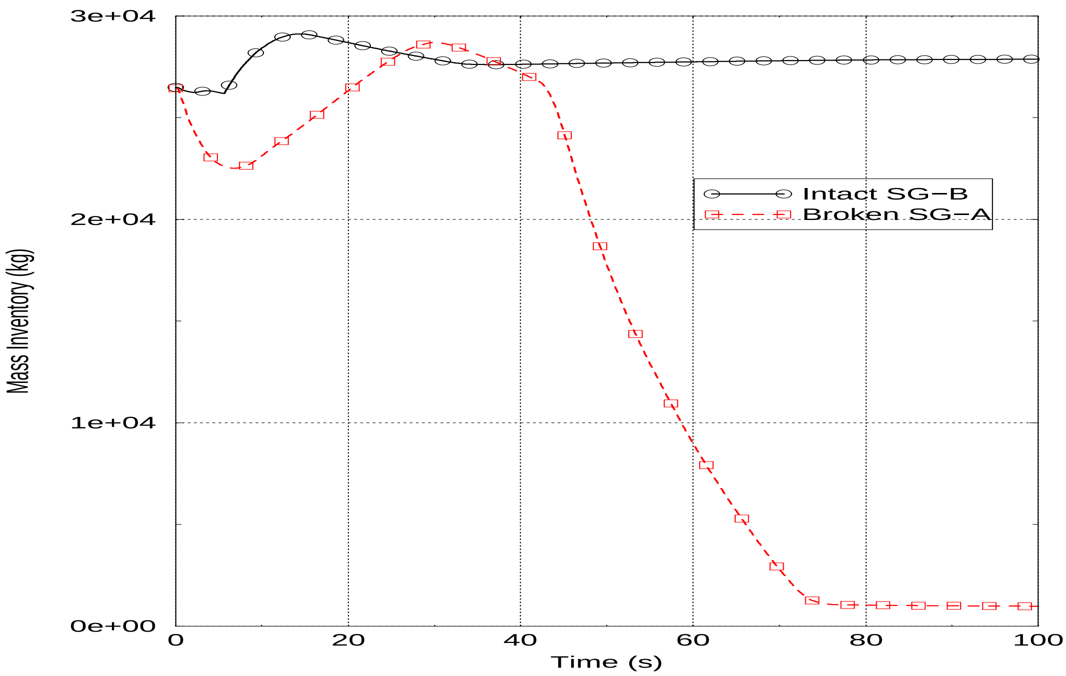

Figure 6-11 Fluid inventory of both steam generators on the secondary side

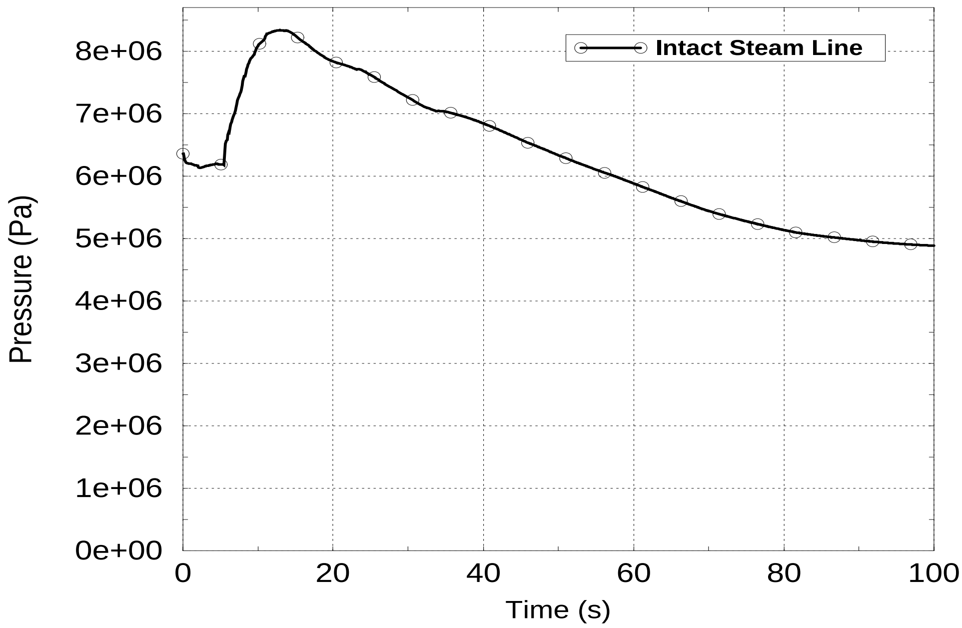

Figure 6-12 Secondary side pressure of intact steam generator

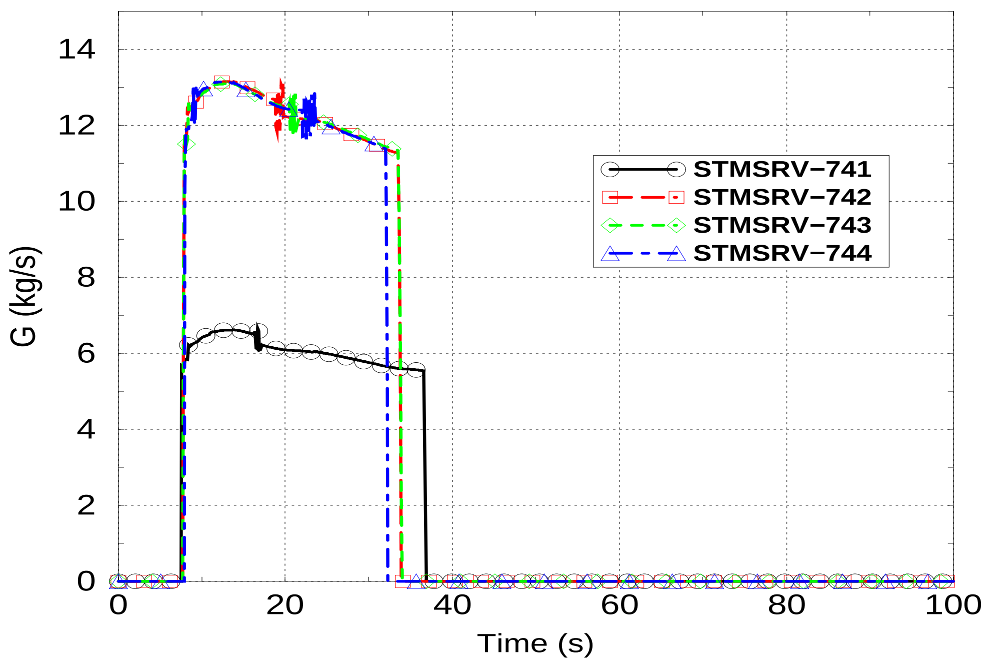

Figure 6-13 Outflow rates of MSSVs of the intact loop-B

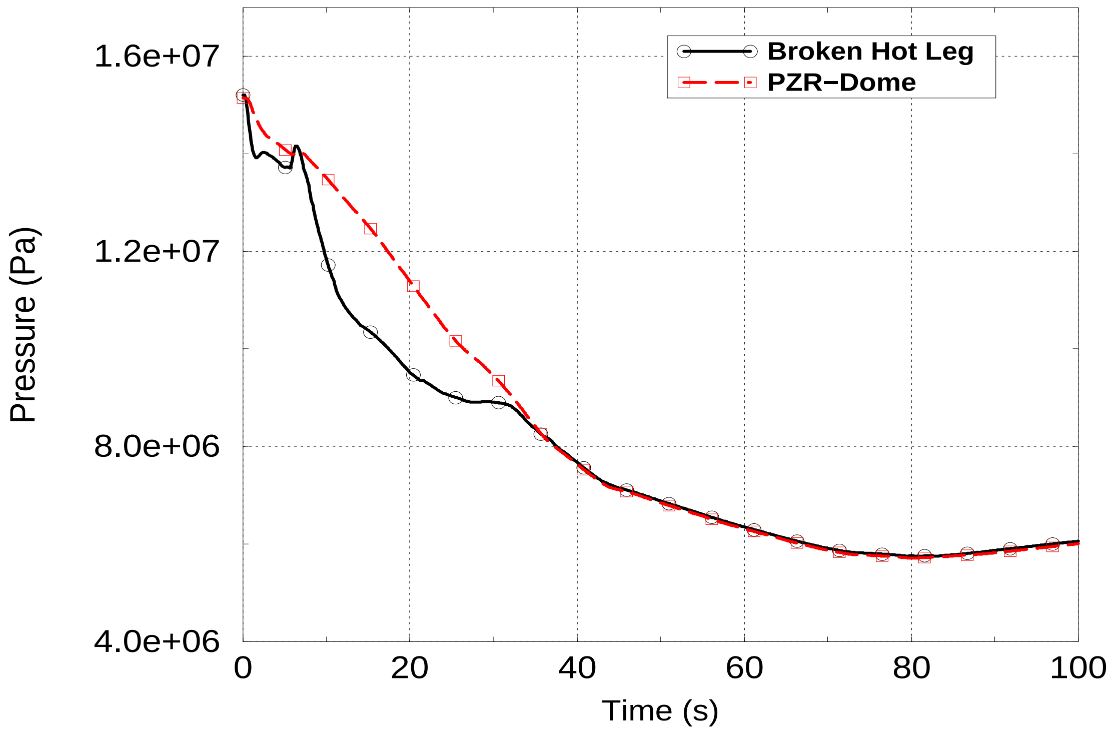

Figure 6-14 Primary system pressure at hot leg and PZR-dome

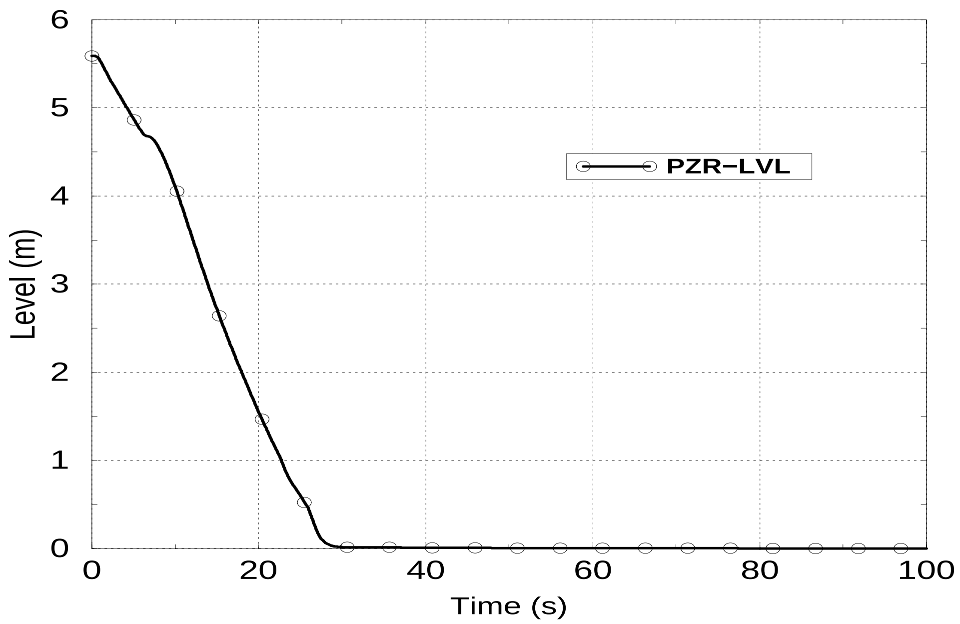

Figure 6-15 Water level of pressurizer

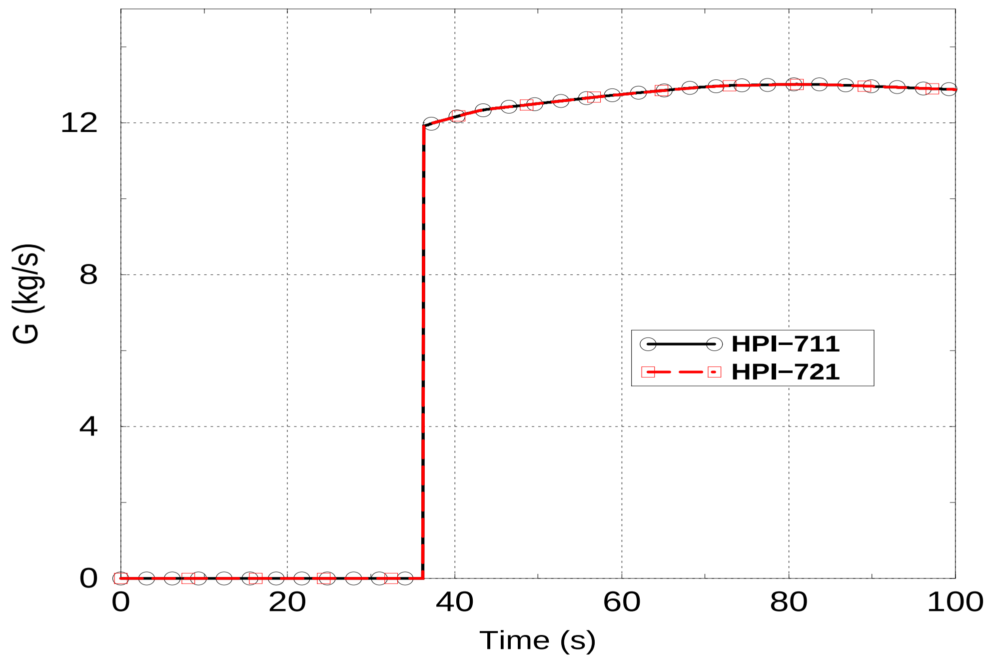

Figure 6-16 Predicted time for HPI-injection

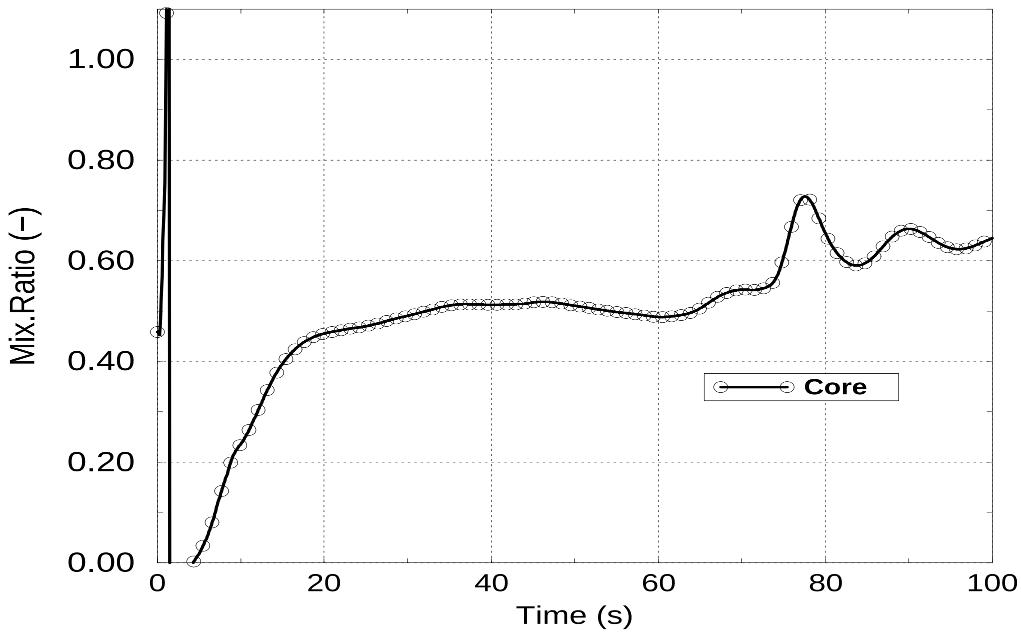

Figure 6-17 Coolant mixing factor as predicted by RELAP5

7 Sensitivity study

It is well known that the predicted plant behaviour depends not only of the physical models implemented in the codes but also on other issues like the overall plant model approach, the plant nodalisation, and the treatment of initial and boundary conditions. Based on the RELAP5-experience analyzing the MSLB-Benchmark, the influence of the subsequent topics were investigated:

- Coolant mixing: no mixing at all (No Mixing) and perfect mixing in the lower plenum (LP Full Mixing),
- Modelling of additional FW-feeding into the broken SG (FeedW) during the transient,
- Initial fluid inventory (19 and 26 tons) in the steam generator secondary side (SGINV), and
- Use of different critical flow models (CFM) for break model.

Sensitivity calculations were performed to evaluate the effect of these parameters on time history of selected global parameters like total power and reactivity for the MSLB-transient. It has to be noted that the RELAP5-results obtained according to the specifications is referred to as “Base Case” in the frame of sensitivity study.

1) Coolant mixing within the RPV

If no coolant mixing is considered the cooldown of the broken loop-A becomes more pronounced than that of the intact loop-B compared to the Base Case, see Figure 7-1 (upper part). It can be seen that the cold leg temperature of the intact loop-B begins to increase after the turbine isolation and it becomes equal to the hot leg temperature for the remaining transient time. But in general the coolant temperature (hot/cold leg) of the intact loop-B experiences a smoother decrease compared to the one of the Base Case, see Figure 7-1 (lower part). Consequently the amount of positive reactivity added to the core as well as the power peak after scram are not so high as in the Base Case, see Figure 7-2.

If perfect coolant mixing in the lower plenum is assumed, the coolant temperature of cold and hot legs of the broken loop-A undergoes a less decrease that those of the Base Case, see Figure 7-3 (upper part). But the coolant temperature of both legs of the intact loop-A, experiences a larger reduction in opposite to the one of the Base Case, see , Figure 7-3 (lower part). Hence the predicted power peak and the total reactivity added to the core are significantly higher than in the Base Case, see Figure 7-4.

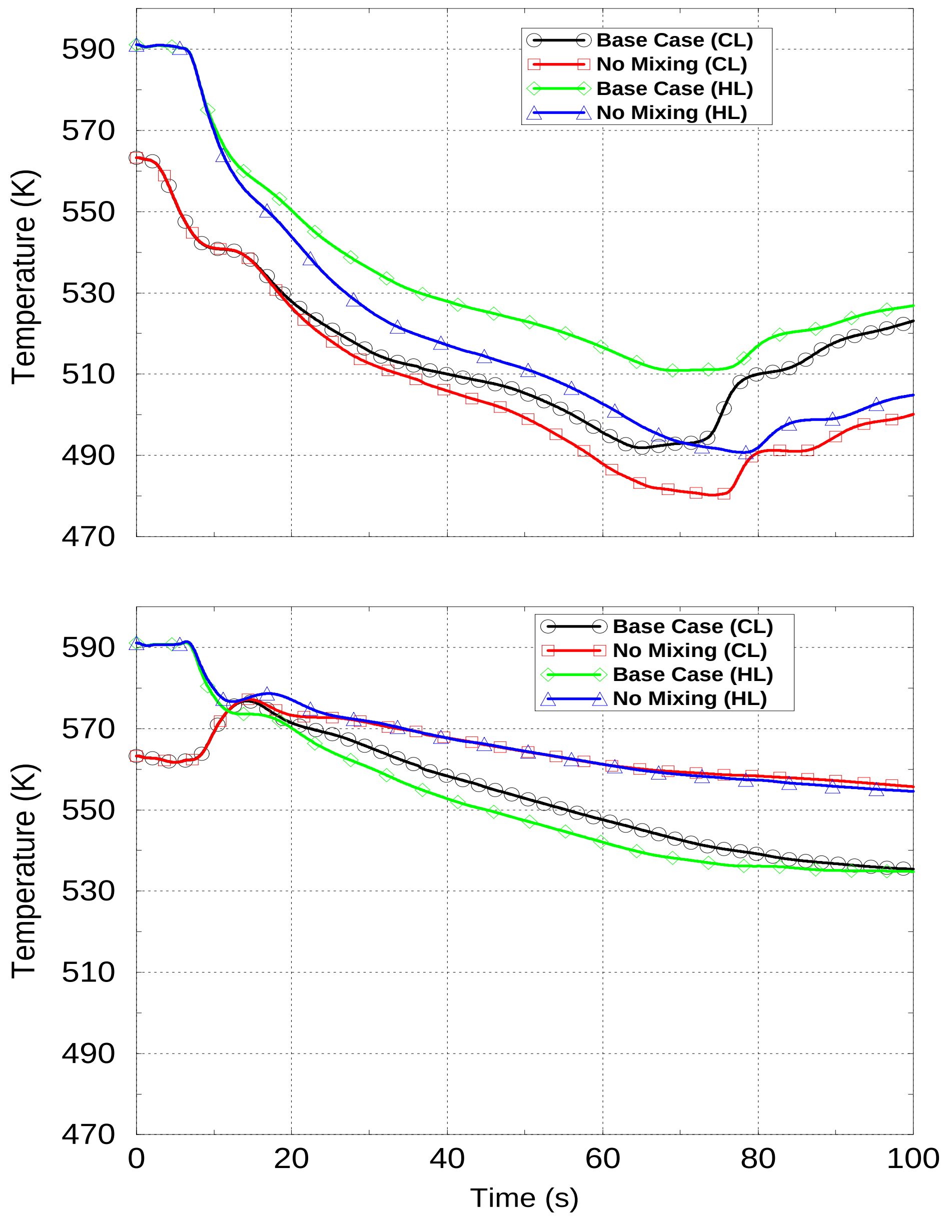

Figure 7-1 Coolant temperature of broken loop-A (upper) and intact loop-B(lower)

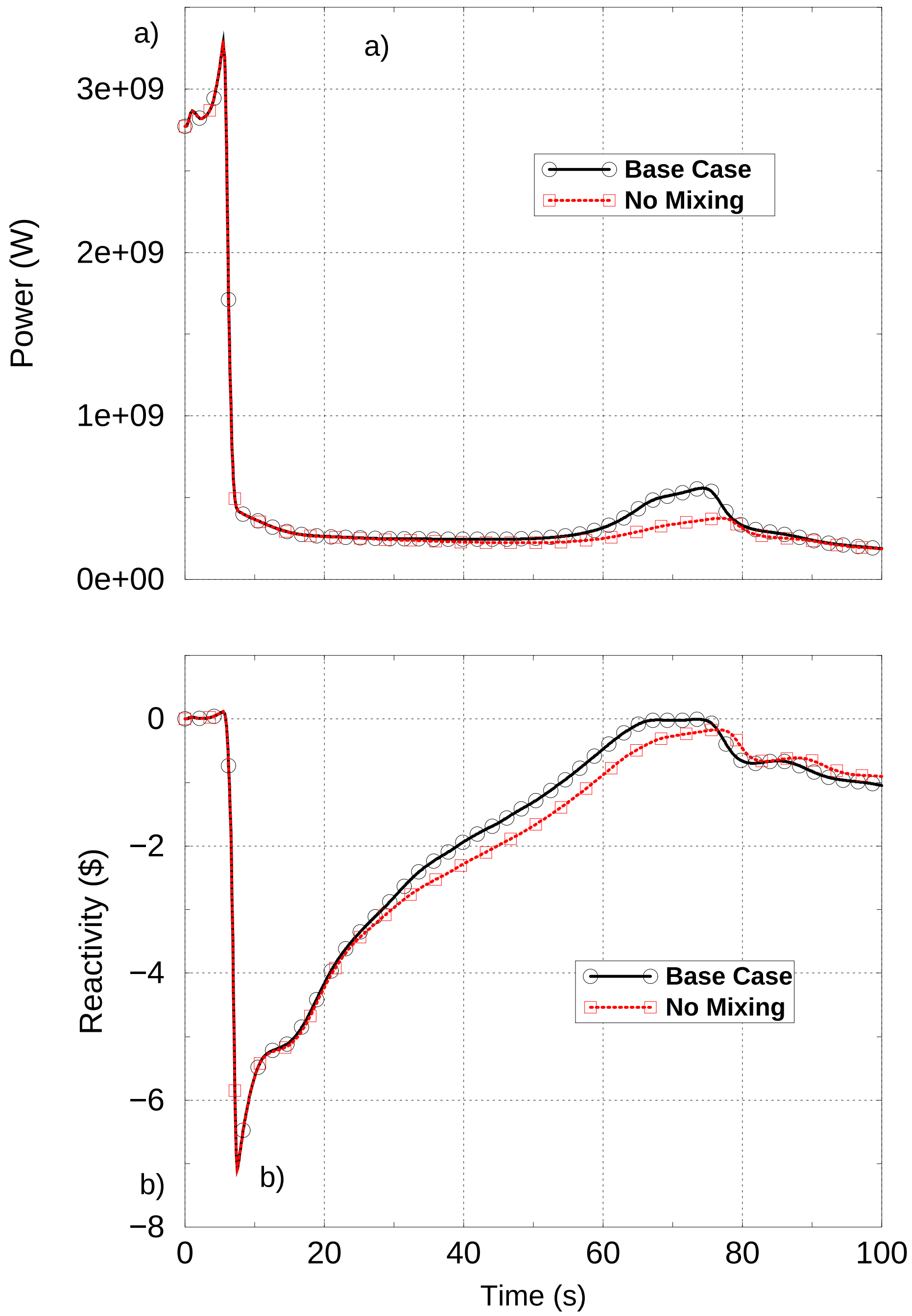

Figure 7-2 Total power and reactivity (no lower plenum mixing)

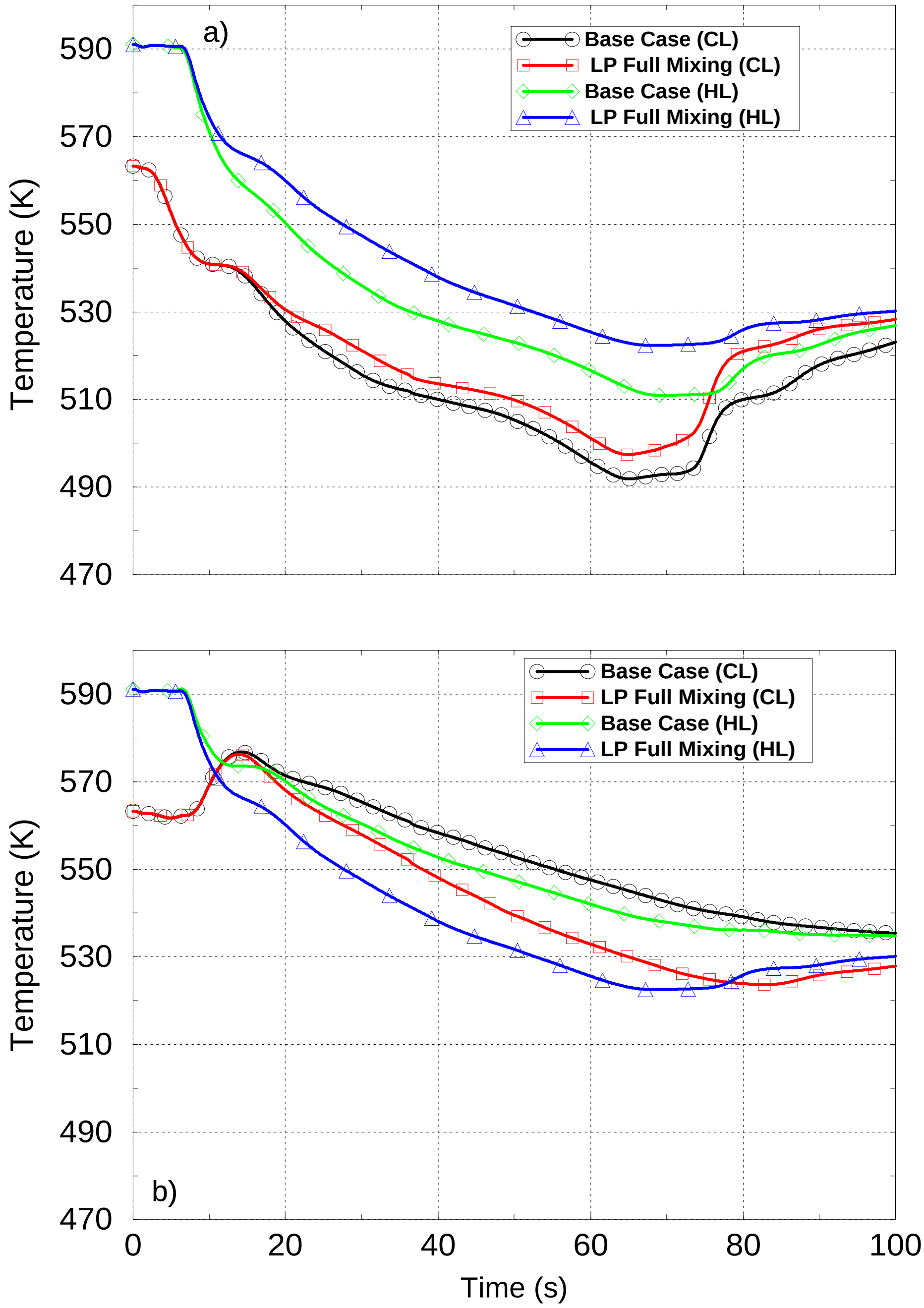

Figure 7-3 Coolant temperature of broken/intact loop-A (upper) and loop-B (lower)

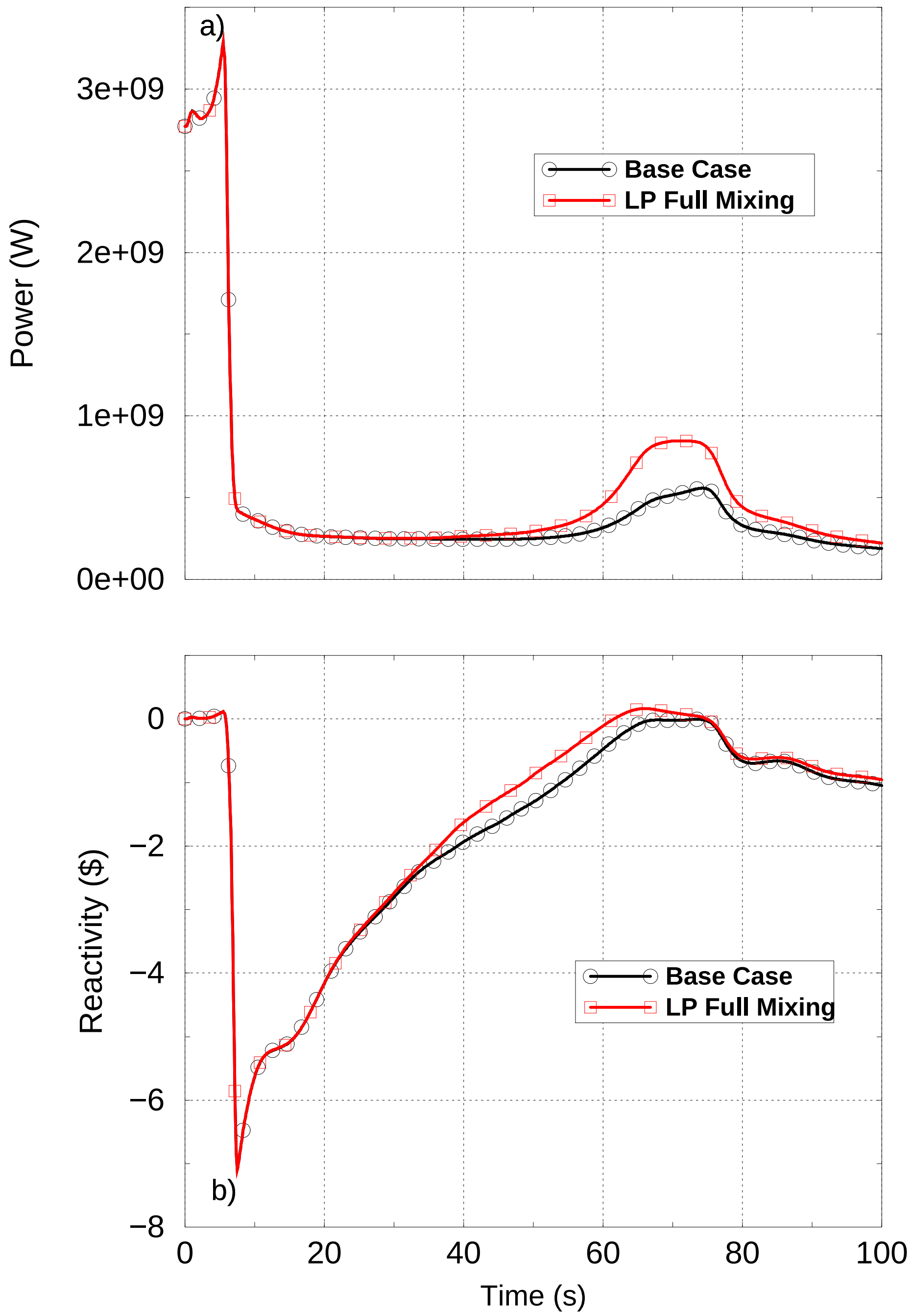

Figure 7-4 Total power and reactivity (lower plenum full mixing)

2) Model of feeding the broken SG with additional feedwater (FeedW)

The feeding of the additional FW collected in the FW-lines into the downcomer can be modelled in two ways:

- Injection starts when the pressure in the downcomer drops below the pressure of the feed water system (**FeedW**). In this case the injection time is not known a priori since it is a result of the accident progression.
- Inclusion of the additional FW-injection in the boundary condition as given in Figure 6-1 in order to minimize the model differences among the participants (**Reference**).

The injection rate as predicted by RELAP5 for the case FeedW is given in Figure 7-5. Even though the amount of injected feedwater is the same for both cases, its time history differs form each other. A comparison of the power and reactivity trends predicted for both cases showed large differences in the power peak and maximal reactivity, see Figure 7-6. It demonstrates that the prediction of the plant response is strongly dependent of how boundary conditions are treated in the plant model.

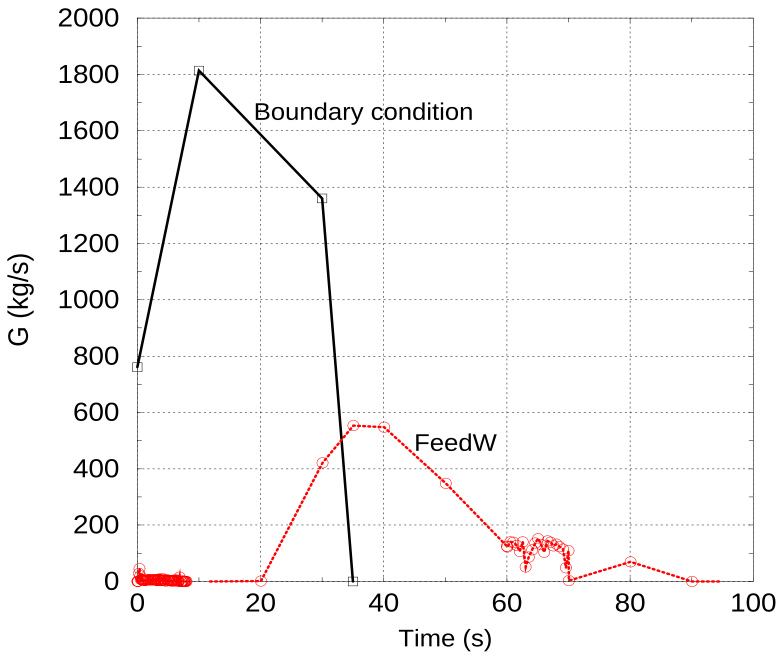

Figure 7-5 Feeding of broken SG (boundary condition + additional feed water)

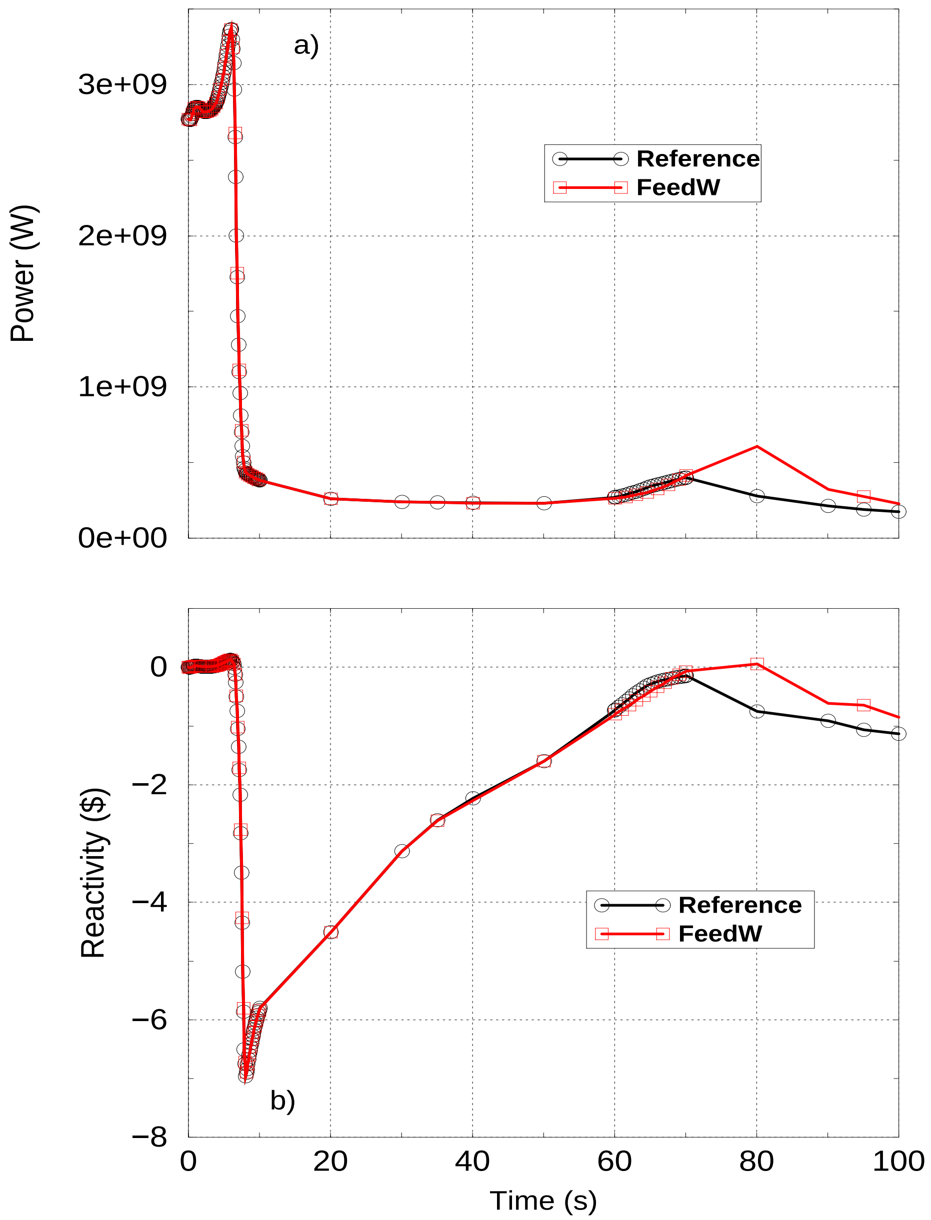

Figure 7-6 Total power and reactivity (feed water injection)

3) Fluid inventory on the steam generator secondary side

A peculiarity of the OTSG is that its fluid inventory on the secondary side increases with increasing power level. Since a maximal fluid inventory represents the worst case for the overcooling MSLB-Transient, it was artificially increased from about 19 tons to 26 tons to make a power increase despite scram feasible.

Some calculations were performed with RELAP5 to investigate how the secondary fluid inventory affects the global results like power and reactivity trends. In Figure 7-7 a comparison of the total power and reactivity for both cases (19 and 26 tons) is presented. The results confirm that less fluid inventory leads to a faster boil-off of the secondary side leading to higher cooldown rates compared to the case with 26 tons fluid inventory. Since the rate of positive reactivity insertion is higher for 19 tons than for 26 tons, the power increase starts earlier in time and achieves higher gradient. Even though the reactivity insertion rate for 26 tons is smaller (slightly lower power peak) than for 19 tons, the core is much closer to re-criticality since primary-to-secondary heat removal lasts for longer time.

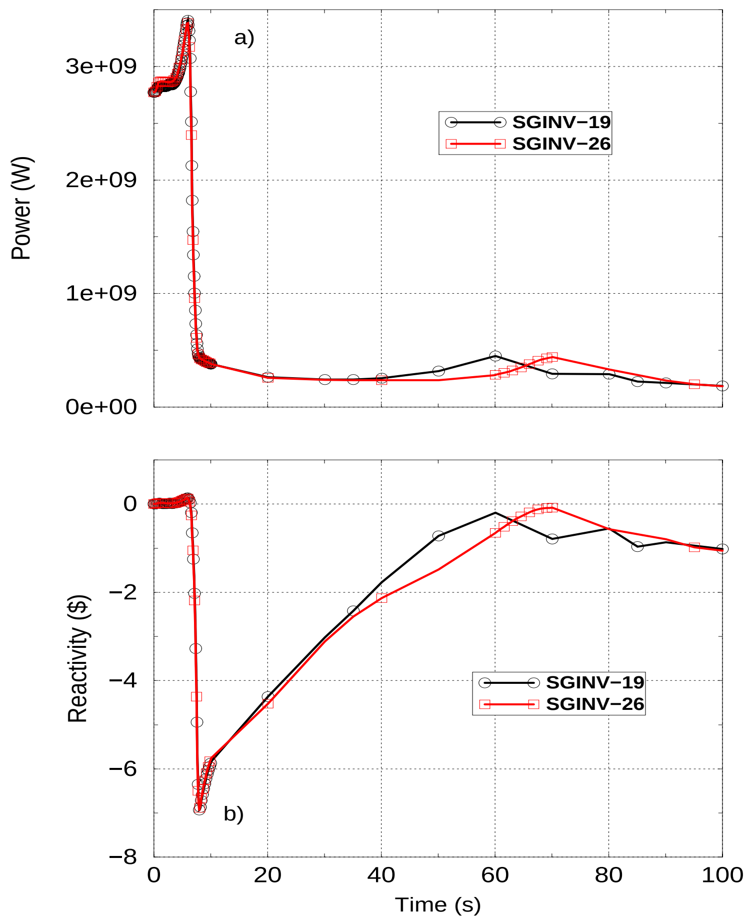

Figure 7-7 Total power (upper figure) and total core reactivity (lower figure)

4) Critical flow model

The Benchmark-specifications recommended the use of the Moody critical flow model, but this model is not available in RELAP5. Hence calculations were performed using two different RELAP5-models, namely the Trapp&Ramson (Reference) and the Henry-Fauske (CFM), to check its influence on predicted total power and reactivity. In Figure 7-8 (upper part) the calculated power and reactivity histories are given. The power curves shows marginal difference due to small differences in the outflow rates, see Figure 7-9. But the differences in the outflow rates are well reflected in the reactivity trends as exhibited in Figure 7-8 (lower part).

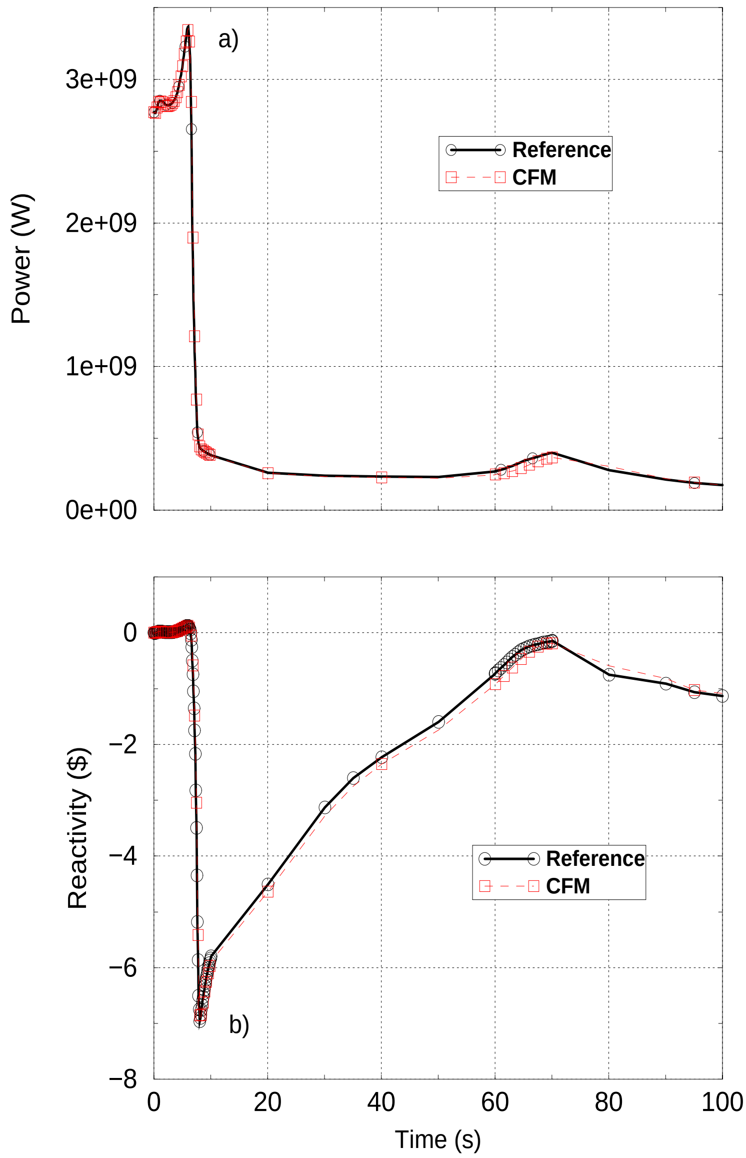

Figure 7-8 Total power and reactivity for two critical flow models

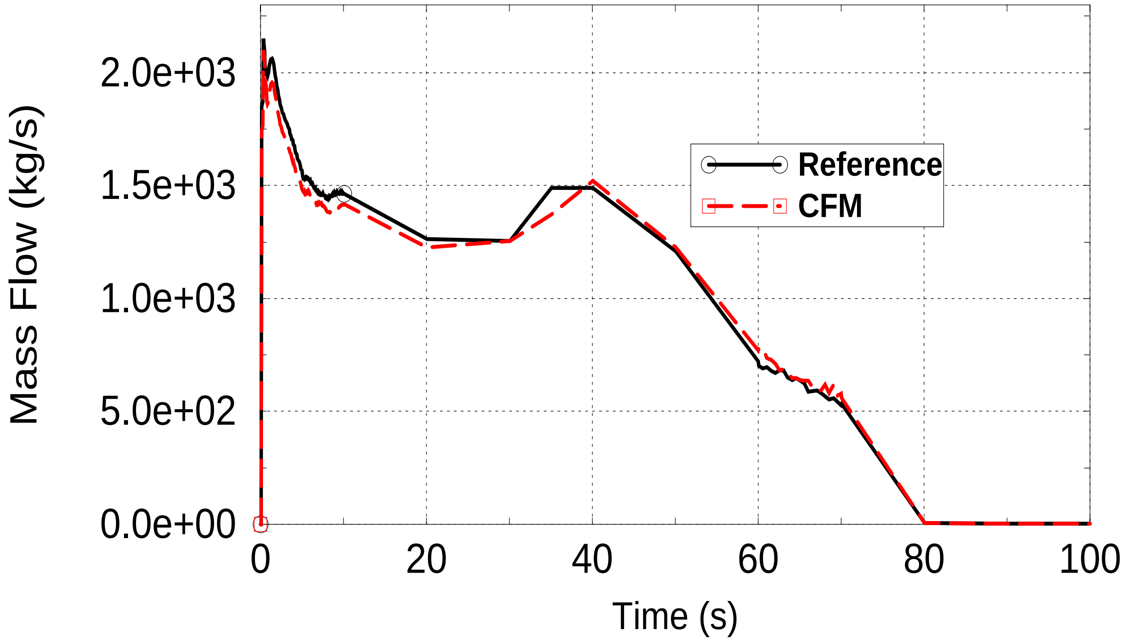

Figure 7-9 Total break outflow rates for two critical flow models

8 Code-to-code comparison

The participants` results obtained for the phase 1 with different code systems were evaluated by the PSU. This code-to-code comparison is extensively summarized in [Ivan00a]. It can be stated that the results of FZK/Siemens lie close to the statistical mean value of all submitted results. From [Ivan00a] the time history of the total power, core reactivity and break outflow predicted with RELAP5 by different institutions (Purdue, British Energy and FZK/SKWU) and with TRAC-PF1 (PSU) were extracted and presented in Figure 8-1, Figure 8-2 and Figure 8-3. The differences among the RELAP5-predictions (power and reactivity) are determined by the predicted break outflow, see Figure 8-3. British Energy calculated the highest outflow rates from the very beginning. Thus the corresponding reactivity gradient and the power peak is the highest compared to the other RELAP5-results. One of the reasons for the different break outflow rates predicted by the RELAP5-codes may be the break model itself as well as the SG-model. Nevertheless the overall trends predicted with RELAP5 by different participants are similar and qualitatively in good agreement.

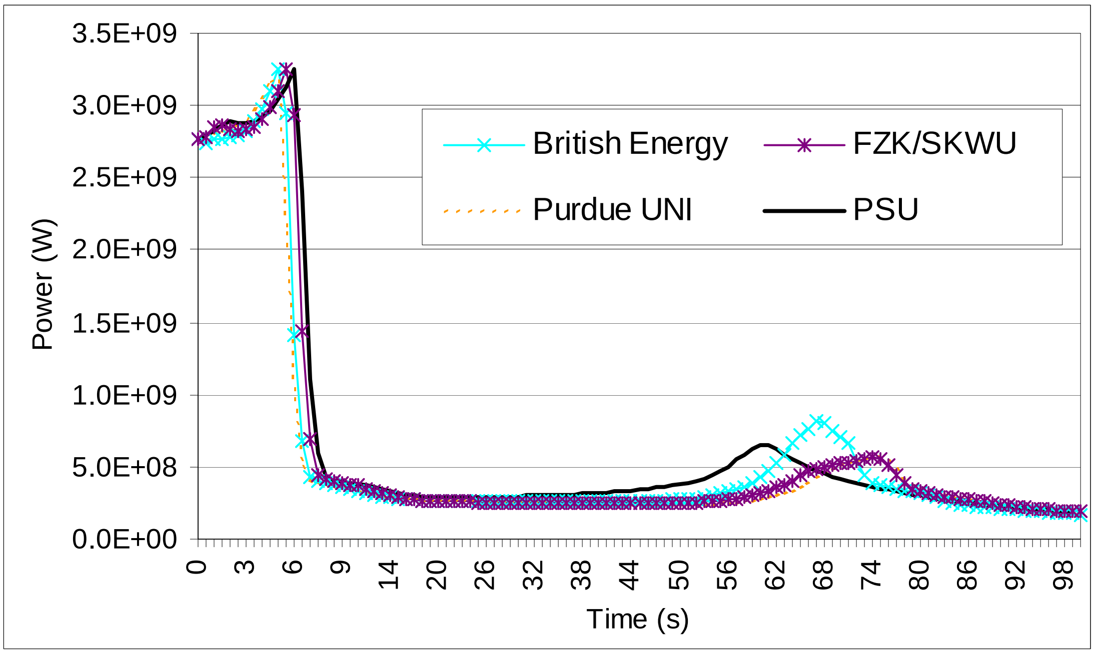

Figure 8-1 Total reactor power predicted by different codes

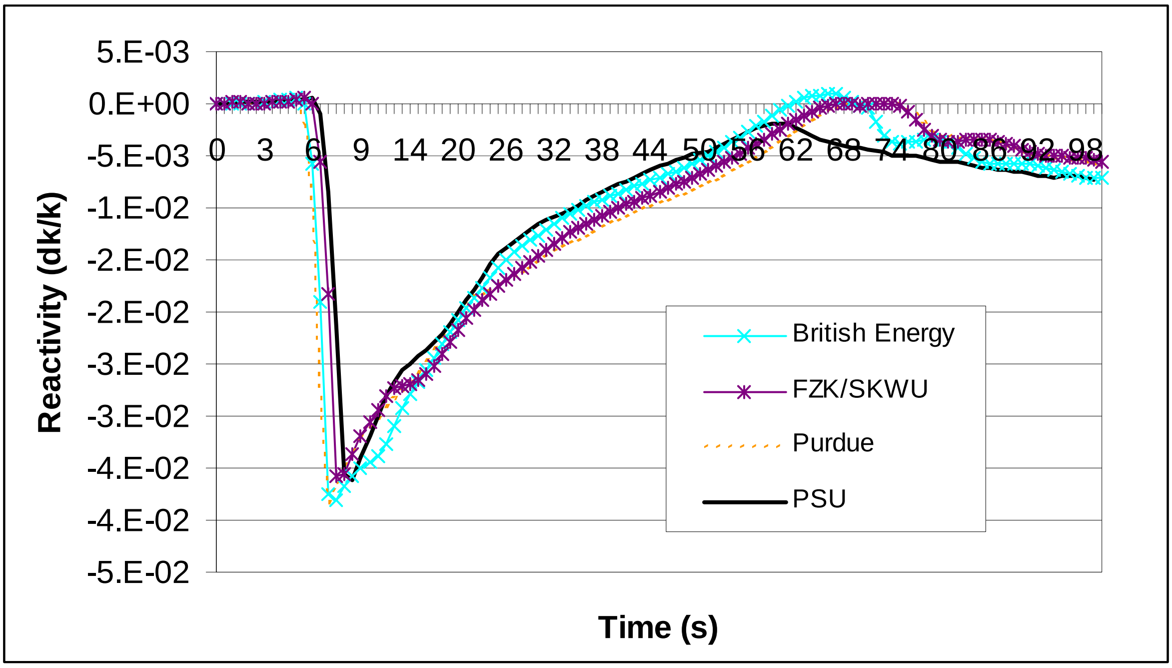

Figure 8-2 Total core reactivity predicted by different codes

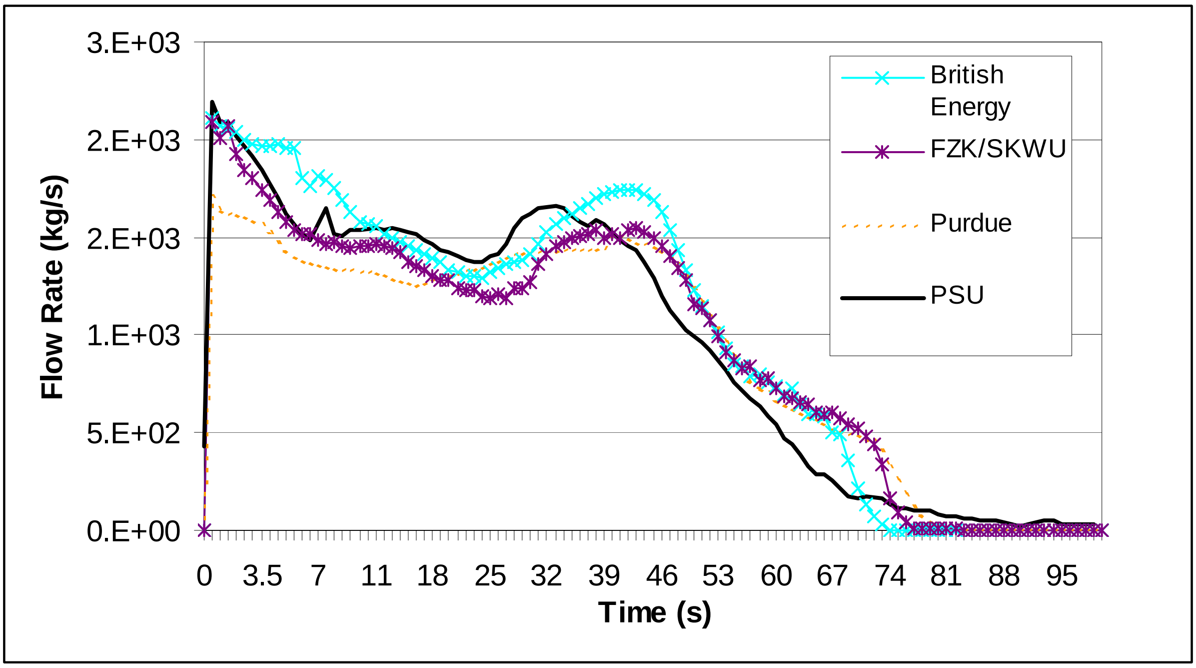

Figure 8-3 Total break outflow predicted by different codes

9 Summary and conclusions

As part of the OECD/NEA MSLB PWR Benchmark the Exercise 1 was investigated with the RELAP5/MOD3.2 code using the point kinetics approach. For this purpose a plant model was elaborated for the two-loop TMI-1 plant. The development of a RPV-splitting model was essential to properly simulate the MSLB-Transient, especially the non-symmetrical thermal behaviour of the core. Also the modelling of the coolant mixing in the lower and upper plenum was very important for the successful MSLB-prediction.

The steady-state plant calculation demonstrated the appropriateness of the developed plant model for the investigation of the MSLB-Transient phase. A good agreement between RELAP5-predictions and the data given in the specifications was achieved.

The results obtained for the transient phase have shown that RELAP5 is able to simulate the important phenomena characterizing the MSLB-transient in an acceptable manner. The code-to-code comparison has also demonstrated that the predicted time histories of most neutronic and thermal hydraulic parameters are close to most results of Benchmark-participants. It can be concluded that this Exercise 1 was very appropriate to develop a consistent and well-balanced plant model for further investigations using 3D-neutron kinetics models.

Due to intrinsic limitations of the point kinetics model and the use of fixed core averaged axial power profile for the whole transient, the reactivity feedback coefficients are overestimated by this approach so that the predicted parameters e.g. core power, reactivity, etc. are conservative.

Moreover the investigations have shown that the prediction of the TMI-1 plant response is in general sensitive to issues such as degree of detail of the overall plant model, initial and boundary conditions treatment, nodalization, and assumptions. The influence of the coolant mixing, feed-water injection, and SG-fluid inventory on global parameters like reactor power and reactivity is not negligible.

To conclude, it can be stated that the developed integral plant model is a good basis for the simulation of the subsequent Exercise 3, where a 3D-core model will be coupled with the 1D-plant model.

10 Literature

[Grund95a] U. Grundmann, D. Lucas, U. Rohde; Coupling of the Thermohydraulic Code ATHLET with the Neutron Kinetic Core Model DYN3D. proc. Int. Conf. on Mathematics and Computation, Reactor Physics, and Environment Analyses. April 30 –May 4, 1995. American Nuclear Society, Portland, Oregon, USA. Vol.1, p257.

[Ivan96] K. N. Ivanov et al., Recent Improvements to the CoupledTRAC-PF1/NEM Methodology. TANASO 75, 329. 1996.

[Lange99] S. Langenbuch, K.-D. Schmidt, K. Velkov, "The Coupled Code System ATHLET-QUABOX/CUBBOX, Model Features and Results for the Core Transients of the OECD PWR MSLB Benchmark", Int. Conf. on Mathematics and Computation, Reactor Physics and Environmental Analyses in Nuclear Applications, Vol.1, pp351-358. Sep 27-30, 1999. Madrid. Senda Editorial.

[Hoho94] J. Hohorst, S. Polkingome, L. J. Siefken, C. M. Allison, and C. A. Dobbe. TMI-2 Analysis using SCDAP/RELAP5/MOD3.1. INEL-94/0157. Lockeed Idaho Technologies Co. Idaho Falls, ID (USA). November 1994.

[Ivan99a] K. Ivanov, T. Beam, A. Baratta, A. Irani, N. Trikouros; PWR MSLB Benchmark. Volume1 Final Specifications. NEA/NSC/DOC(99)8. USNRC/OECD, PSU. April 1999.

[nrcd97] US NRC Reactor Safety Data Bank. Public. October 1997
http://www.nrc.gov/RES/databank/databank.htm

[R5DT96] RELAP5 Development Team. RELAP5/MOD3 Users Guide.
Idaho National Laboratory. July 1996

[Knoll98] A. Knoll, R. Böer, H. Finnemann, and A. Van de Velde; Coupled Neutronics-Thermal hydraulics Code System RELAP5/PANBOX and RELAP5/HEXTIME for Integrated Safety Analysis. Physics of Nuclear Science and Technology. Int. Conf. Oct. 5-8, 1998. Vol.2, pp 1429-1436. American Nuclear Society, La Grange Park, Ill., USA.

[Miller98] R. M. Miller, T. Downar, W. Wang, "Final Completion Report for the Coupled RELAP5/PARCS Code. PU/NE-98-31. Purdue University 1998.

[Miller99] R. M. Miller, T. Downar, "Completion Report for the Coupled TRAC-M/PARCS Code", PU/NE-99-20. Purdue University, 1999.

[Royer00] E. Royer, D. Caruge, and E. Raimond; IPSN&CEA/DRN Contribution for the MSLB-Benchmark Exercise 3 using CATHARE, CRONOS2, FLICA4 and ISAS codes. OECD MSLB Benchmark. 24-25 January. Paris, 2000.

[Ivan99b] K. Ivanov, et.al.; Features and Performance of Coupled Three-Dimensional Thermal-Hydraulic/Kinetics TRAC-PF1/NEM PWR Analysis Code. Ann. Nuclear Energy, 26, 1407(1999).

[Finne92] H. Finnemann, A. Galati,"NEACRP 3D-LWR Core Transient benchmark, Final Specifications", NEACRP-L-335 (rev. 1). January 1992.

[Fraik97] R. Fraikin, " PWR Benchmark on Uncontrolled Rods Withdrawal at Zero Power", Final Report. NEA/NSC/DOC(96)20. September 1997.

[Ivan00a] K. Ivanov, T. Beam, B. Taylor and A. Baratta; Pressurized Water Reactor Main Steam Line Break (MSLB) Benchmark. Volume II: Results of Phase 1 on Point Kinetics. NEA/NSC/DOC(2000)21. December 2000.

Appendix A: Codes used in MSLB-Exercise 1

Tab. 10-1 List of participants in the first phase of the PWR MSLB Benchmark

Participant Number	Company Name	Country	Code
1	VTT-1	Finland	SMABRE
2	GRS	Germany	ATHLET
3	FZR	Germany	ATHLET
4	GPUN/CSA/EPRI	USA	RETRAN-3D
5	Universities of Pisa and Za-greb	Italy/Croatia	RELAP5/MOD 3.2
6	British Energy	United Kingdom	RELAP5
7	IPSN/CEA	France	CATHARE 2
8	FZK/SKWU	Germany	RELAP5/MOD3.2
9	Netcorp	USA	DNP/3D
10	IBERDROLA	Spain	RETRAN-3D
11	UPM	Spain	TRAC-PF1/MOD3
12	VTT-2	Finland	APROS
13	Purdue University/NRC	USA	RELAP5
14	PSU	USA	TRAC –PF1/MOD2