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Validation of the Reflood Model of the RELAP5/MOD3.2.2Gamma using Experimental Data from the Integral Facility LOFT LP-LB-1

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Abstract

In the framework of the Code Assessment and Maintenance Program (CAMP) of the US Nuclear Regulatory Commission (NRC) the RELAP5 code system is being validated at Forschungszentrum Karlsruhe (FZK). The validation work is focused on the assessment of the RELAP5-reflood model.

The data obtained from the integral test LOFT-LP-LB-1 is used to validate the RELAP5-reflood model implemented in the version RELAP5/MOD3.2.2Gamma (322Gamma). The test LP-LB-1 simulates a double-ended, off-set shear of the cold leg of a Pressurized Water Reactor (PWR) coincident with the loss of off-site power.

A post-test calculation of the LOFT-test was performed using a model developed at Paul Scherrer Institut (PSI) for RELAP5. Results of these investigations are presented and discussed in this report.

Based on the predictions with the original version 322Gamma it can be stated that the overall system behaviour and the core thermal response are reasonable predicted by RELAP5. The reflooding process is qualitatively well predicted by this code version. The cladding temperature in several bundle elevation are closer to measured data compared to the ones of earlier versions. But PSI-model tends to under-predict the rewetting temperature due to the use of the empirical Weisman correlation to determine the transition boiling heat transfer.

Therefore the FZK-transition boiling correlation was implemented in the original version instead of the Weisman model (code version 322Gamma+FZK). Recalculations of the LP-LB-1 test with the version 322Gamma+FZK showed that the rewetting temperature in all axial elevations better fits with experimental data now.

Qualifizierung des Flutmodells im RELAP5/MOD3.2.2Gamma unter Verwendung experimenteller Daten aus der Integralanlage LOFT LP-LB-1

Zusammenfassung

In Rahmen des internationalen Code Assessment and Maintenance (CAMP) Programms der US Nuclear Regulatory Commission (NRC) wird das Thermohydraulik-Programm RELAP5 am Forschungszentrum Karlsruhe (FZK) umfassend qualifiziert. Diese Arbeiten konzentrieren sich auf die Qualifizierung des RELAP5-Flutmodells.

Die an der integralen LOFT-Versuchsanlage, Test LP-LB-1 gewonnenen Daten werden zur Validierung des in RELAP5/MOD3.2.2Gamma (322Gamma) implementierten Flutmodells herangezogen. Dieser LP-LB-1-Versuch simuliert einen 2F-Bruch im kalten Strang der Hauptkühlmittelleitung eines Druckwasserreaktors (DWR) bei gleichzeitigem Ausfall der Eigenenergieversorgung.

Zur Nachrechnung des LOFT-Versuches wurde ein am Paul Scherrer Institut (PSI) für RELAP5 entwickeltes Modell verwendet. Ergebnisse dieser Untersuchungen werden in diesem Bericht vorgestellt und diskutiert.

Aufgrund der mit der ursprünglichen Version 322Gamma durchgeführten Nachrechnung kann festgestellt werden, dass das System- und Kernverhalten in angemessener Weise simuliert wurde. Der Flutprozess wird qualitativ gut gegenüber früheren Codeversionen beschrieben. Die berechnete Hüllrohrtemperatur stimmt in mehreren Bündelhöhen gut mit den Messdaten überein. Dennoch tendiert das PSI-Flutmodell, bedingt durch die Verwendung der empirischen Weisman-Korrelation für die Bestimmung des Wärmeübergangs im Übergangssiedensbereich, zur Unterschätzung der Wiederbenetzungstemperatur.

Daher wurde das FZK-Übergangssiedemodell anstelle der Weisman-Korrelation in die ursprüngliche RELAP5-Version implementiert (neue Version 322Gamma+FZK). Die erneute Nachrechung des LP-LB-1-Versuches mit der 322Gamma+FZK-Version zeigte, dass die berechnete Wiederbenetzungstemperatur in allen axialen Maschen besser mit den experimentellen Daten übereinstimmt.

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LIST OF ABBREVIATIONS

ACC	**ACC**umulator
CHF	**C**ritical **h**eat **f**lux
CAMP	**C**ode **A**ssessment and **M**aintanance **P**rogram
DWR	**D**ruck**W**asser**R**eaktor
EOC	**E**nd of **C**ycle
ECC	**E**mergency **C**ore **C**ooling
FA	**F**uel **A**ssembly
FZK	**F**orschungs**Z**entrum **K**arlsruhe, Technik und Umwelt
FB	**F**ilm **B**oiling
INEEL	**I**daho **N**ational **E**ngineering and **E**nvironmenta**l** Laboratory, USA
IRS	**I**nstitut für **R**eaktor**S**icherheit,
ISL	**I**nformation **S**ystems **L**aboratory
KWU	**K**raft**W**erk **U**nion/Siemens
KAERI	**K**orean **A**tomic **E**nergy **R**esearch **I**nstitute
LWR	**L**ight **W**ater **R**eactor
LOFT	**L**oss **O**f **F**luid **T**est
LOCA	**L**oss **O**f **C**oolant **A**ccident
LP	**L**ow **P**ressure
LB	**L**arge **B**reak
LPIS	**L**ow **P**ressure **I**njection **S**ystem
NEA	**N**uclear **E**nergy **A**gency
OECD	**O**rganization for **E**conomical **C**ooperation and **D**evelopment
PCT	**P**eak **Cl**adding **T**emperature
PSI	**P**aul **S**cherrer **I**nstitut
PWR	**P**ressurized **W**ater **R**eactor
PZR	**P**ressurizer
RCS	**R**eactor **C**oolant **S**ystem
RPV	**R**eactor **P**ressure **V**essel
RELAP	**Re**actor **L**eak and **A**nalysis **P**rogram for LWR transients and SBLOCA
RPV	**R**eactor **P**ressure **V**essel
RBHT	**R**od **B**undle **H**eat **T**ransfer
TB	**T**ransition **B**oiling
US NRC	**U**nited **S**tates **N**uclear **R**egulatory **C**ommission

1 Introduction

As in-kind contribution within the CAMP-Program the test LOFT LB-LP-1 was analysed to validate the reflood model of RELAP5/MOD322Gamma. The main objective of the LOFT-LP-LB-1 test [NRC87], carried out by INEEL in 1984, was the investigation of the blow-down, refill and reflood phase of a postulated LB-LOCA accident in a commercial four-loop Westinghouse PWR. Investigations were focused on complex two-phase flow and heat transfer mechanisms which may be encountered during the reflood phase of a LOCA. Hence the data of the test LP-LB-1 are very useful to qualify the physical models implemented in best-estimate thermal hydraulic codes like TRAC [Liles84], CATHARE [Barre90], RELAP5 [R5M32], and ATHLET [Burw89].

At FZK investigations were performed to validate the reflood model implemented in several RELAP5-versions. The work started with RELAP5/MOD3.1[Sanc97]. A considerable validation work has been performed worldwide to assess the reflood model of RELAP5 [Anal96], [Chung96], [HCNo98], [Ban99].

The reflood model of the 322Gamma-version was developed at PSI [Anal96]. It distinguishes two heat transfer packages, one for reflood and the other one for non-reflood conditions, compared to the unique heat transfer package of earlier versions. A new characteristic of this PSI-model is the introduction of the dependency of the heat transfer coefficient from the distance from the quench front in all post-CHF flow regimes. In this approach the transition boiling heat transfer is predicted by the empirical Weisman correlation.

Preliminary investigations showed that the Weisman correlation tends to over-predict the heat transfer coefficient close to the quench front. Therefore this correlation was replaced by the FZK-transition boiling model which is based on a semi-mechanistic approach of the Chen formalism [Elias98]. The modified code version is called RELAP5/MOD3.2.2Gamma+FZK (322Gamma+FZK). The FZK-model represents an extension of the phenomenological formulation of Chen that uses only local state variables predicted by the code itself and does not require other history parameters e.g. quench position, CHF or minimum film boiling temperature. According to this model the total transition boiling heat flux q_l'' is calculated as an average heat flux during the short period of contact between the liquid and the superheated wall. Hence a three-step process was postulated to describe the mechanisms of heat removal by a liquid film from the wall: 1) conduction heating of the liquid film, 2) nucleation and bubble growth within the liquid layer and 3) evaporation of a residual water film at the clad surface. In Appendix A and B both the Weisman and the FZK-transition boiling are summarized.

In this report the post-test calculations performed with both code versions, i.e. with the original version 322Gamma and with the modified version 322Gamma+FZK are presented and discussed. Furthermore, the model of the LOFT-LP-LB-1 test developed for the RELAP5-investigations is described.

2 The integral facility LOFT

2.1 Test LP-LB-1 description

The LOFT facility was designed at INEEL to simulate the behaviour of commercial PWR during a Loss Of Coolant Accident (LOCA). A scheme of the LOFT-facility with its major components is shown in Figure 2-1. The test facility was constructed using a power-to-volume scaling factor of ¼. The power scale is 1:63. The nuclear core consists of five square and four triangular fuel assemblies, in total 1300 fuel rods with an active length of 1.67 m and an outside diameter of 10.72 m. The fuel pins were un-pressurized. The peak linear heat rate was 51.7 kW/m. All nine fuel assemblies are located inside the reactor core (Figure 2-2, [NRC87]).

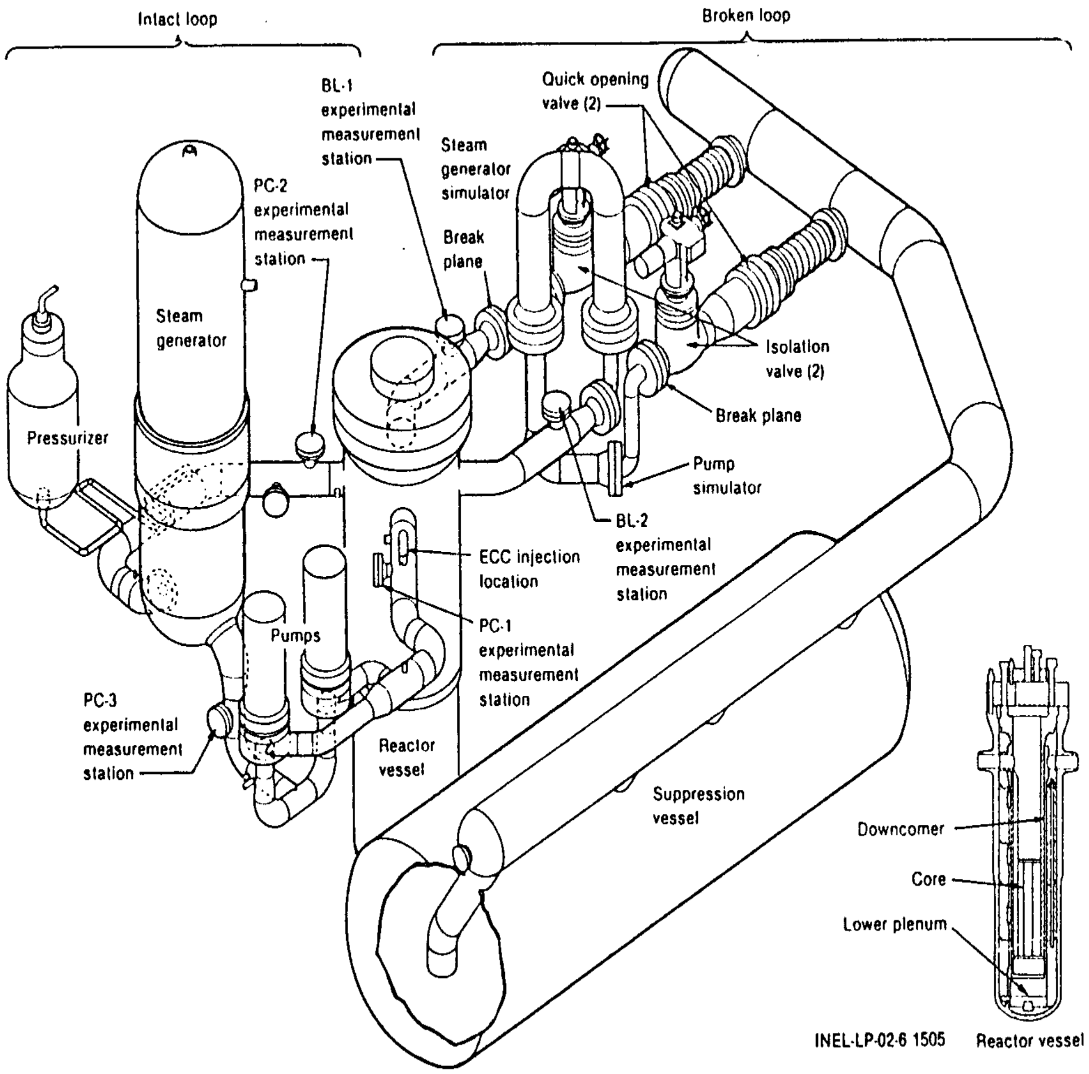

Figure 2-1 The LOFT test facility with major components

Two primary coolant loops, the intact loop and the so-called “broken loop”, are simulated in the LOFT-facility. The intact loop represents three loops of a four-loop PWR. It contains an active steam generator and two active main coolant pumps. The pressurizer is connected to the intact loop. The broken loop represents a single loop of a four-loop PWR and contains an inactive steam generator and pump simulators. An orifice device was added to this loop up-stream of the fast opening valves to simulate different break sizes. Hence the quick-opening blow-down valves simulate the primary system pipe rupture. The primary coolant outflow from the ruptured pipe is collected in a large suppression vessel. The high and low pressure safety injection system (HPIS, LPIS), and accumulator (ACC) are also part of the LOFT-facility.

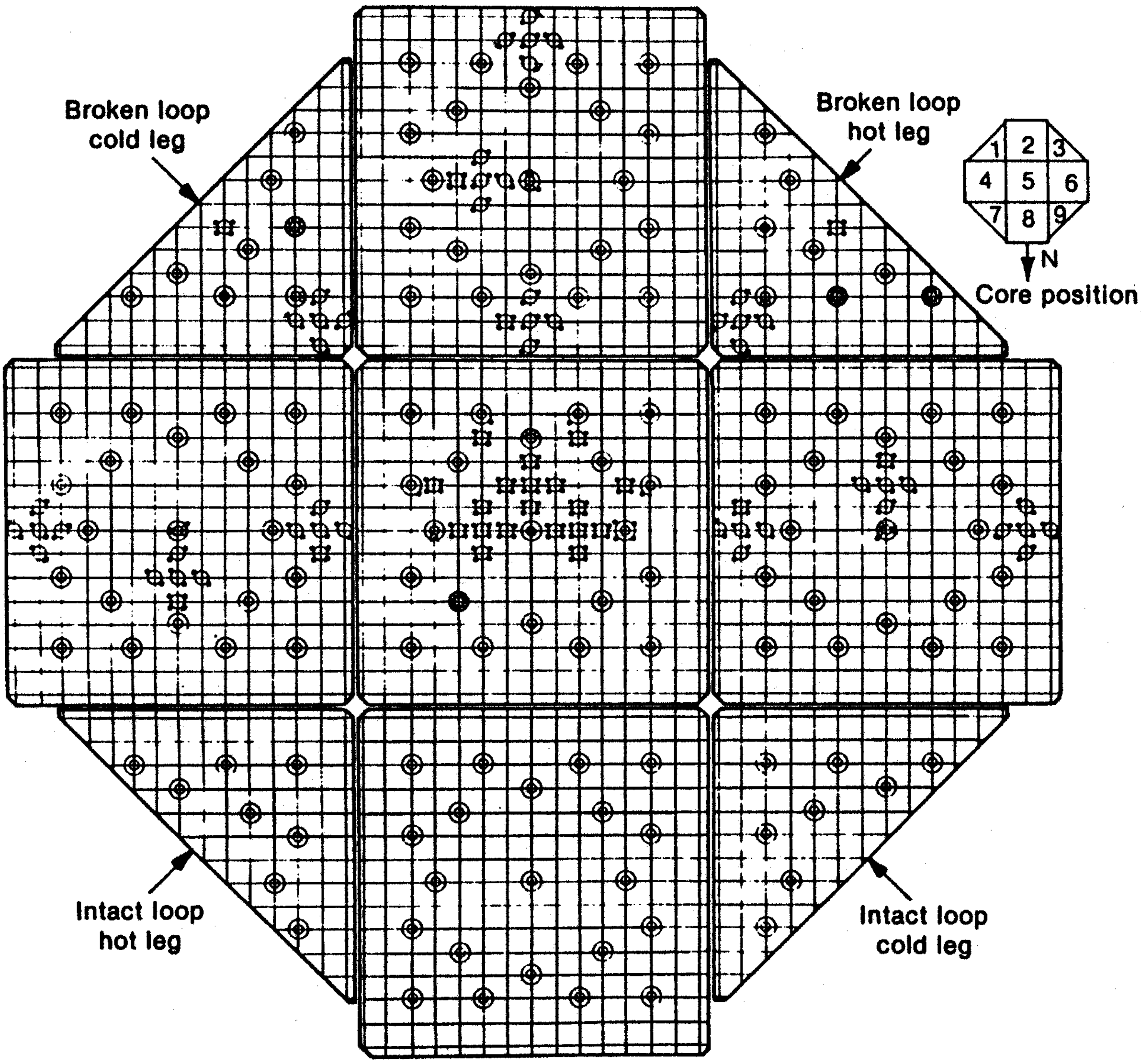

Figure 2-2 Radial section of the LOFT-core

The LOFT-tests are well instrumented so that a large number of thermal hydraulic parameters were measured during the tests, e.g. mass flow rates, pressure, coolant temperature, cladding, and fuel temperature.

A large number of thermocouples were located at different axial elevations of the fuel rods within the different fuel assemblies to measure the axial and radial temperature distribution. At each axial level, several thermocouples were distributed within the central and peripheral fuel assemblies.

2.2 Test conduction, initial and boundary conditions

The LOFT-LP-LB-1 test simulates a double-ended break in the cold leg of a four-loop Westinghouse PWR. The test initial conditions are representative for the United Kingdom (UK) licensing limits, where a loss of off-site power coincident with the LB-LOCA is assumed. The emergency core system injection rates were also determined according to the UK-safeguards assumption [Britt90]. Most relevant thermal hydraulic system processes e.g. blow-down, refill, and reflood are covered by this test so that the obtained data are useful for the validation of best-estimate codes. The LOFT-conditions just before the test initiation are listed in Tab. 2-1. For the LB-LP-1 test it is assume that only ACC and LPIS are available during the accident progression.

Tab. 2-1 Measured steady-state parameters of LOFT-LP-LB-1 Test

Parameters	**Unit**	**Measured value**
Thermal power	MW	49.3 ± 1.2
Maximum linear heat	KW	51.7 ±3.6
Reactor coolant system hot leg pressure	MPa	14.9±0.08
Reactor coolant system mass flow rate	Kg/s	305.9±2.6
Intact cold leg fluid temperature	K	556±1
Intact hot leg fluid temperature	K	585±1
Broken cold leg fluid temperature	K	552±0.6
Broken hot leg fluid temperature	K	581±0.6
Core heat-up	K	29.8±1.4
Pressurizer liquid level	m	1.04±0.4
Pressurizer pressure	MPa	14.9±0.1
Pressurizer water temperature	K	615±6.8
Accumulator liquid level	K	2.36±0.01
Accumulator standpipe position from the bottom	m	2.11± 0.03
ECC-accumulator pressure	Mpa	4.21±0.06
ECC-accumulator liquid temperature	K	302±6.1
Low pressure injection system liquid temperature	K	305±7.0

2.3 Thermal hydraulic phenomena during the test conduct

The transient is initiated at time t=0 s by opening the break valves (quick-opening valves), when the reactor is operated at nominal power of about 50 MW_{th}. The reactor scram occurs when the hot leg pressure of the intact loop drops below 14.5 MPa. The primary coolant pumps were tripped manually and decoupled from their flywheels within the first second causing a rapid coast-down. Thus the up-flow quench phase during the blow-down was prevented. Above the core the flow stagnated immediately leading to an overall temperature increase of the core.

Cladding temperatures above 1100 K were measured above 0.6 m elevation in the central core bundle. A maximum cladding temperature of 1261 K was recorded during the blowdown and of 1257 K during the reflood phase. The core-wide temperature escalation continued until a partial core top-down quench started at about 13 s affecting the top part of the core. It is assumed that this top-down quenching was caused by the liquid fallback from the upper plenum induced by gravity.

By actuation of the low pressure injection system the core was finally quenched at about 72 s from the bottom. No fuel rod ballooning or cladding rupture was detected during this experiment due to the non-pressurized fuel rods.

3 Model of the test LOFT LP-LB-1

The basic LOFT-model for the post-test calculations with RELAP5 was developed and extensively validated by PSI, [Lübb91]. Few modifications were made to adapt the model to the new RELAP5-code version. In the LOFT-model both primary and secondary plant systems as well as the emergency core cooling systems are included, Figure 3-1.

The *intact loop* consists of hot leg, pressurizer, steam generator, cold leg with two pump lines. The *steam generator* consists of *8* volumes on the primary and 5 volumes on the secondary side. On the secondary side, the steam generator model includes feed water, backflow, steam separator, and steam flow control valve. The *SG-tubes* are modelled as RELAP5-heat structures. The PZR-main vessel, the dome and the surge line are represented with pipe-components. The LPIS and an the ACC are connected to the intact cold leg between the pumps and the *reactor pressure vessel* (RPV) only.

The *broken loop* is represented by two individual lines (hot and cold legs) using pipe and branch components. The hot leg comprises the piping, the SG-simulator, break-valve and the containment. The cold leg is represented by the piping, the pumps, the break-valve and the corresponding *containment*.

Two down-comer volumes, each one linked to one cold leg, and to the lower plenum are modelled. Hot legs are connected to the upper plenum. Further volumes representing the core, the bypass, the core outlet, the upper plenum, and the vessel dome correspond to the reactor pressure vessel.

The *core section* is modelled by an average and a hot channel with 5 and 13 axial nodes chosen in such a manner that the corresponding thermocouple position are always located near to the axial centre of the axial node. The central bundle corresponds to the hot channel while the five peripheral bundles to the average channel. The total core mass flow rate is distributed between hot and average channel according to its flow areas. The 1300 fuel pins are modelled by two heat structures, one with 219 (hot channel) and the other one with 1081 pins (average channel). Additional RELAP5-heat structures were used to model the primary and secondary pipe walls.

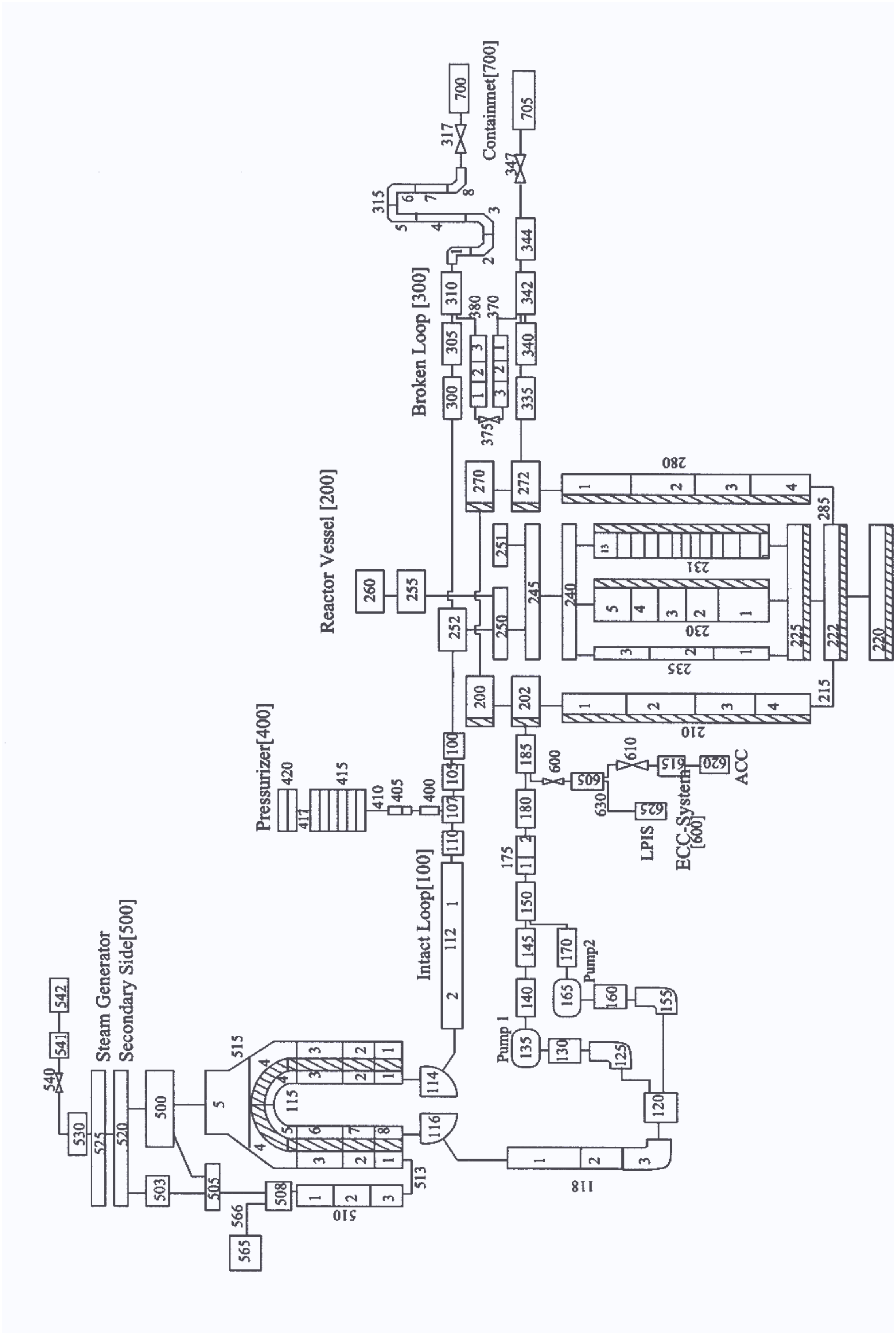

Figure 3-1 LOFT-LP-LB-1 Nodalisation for RELAP5/MOD3.2.2 Gamma

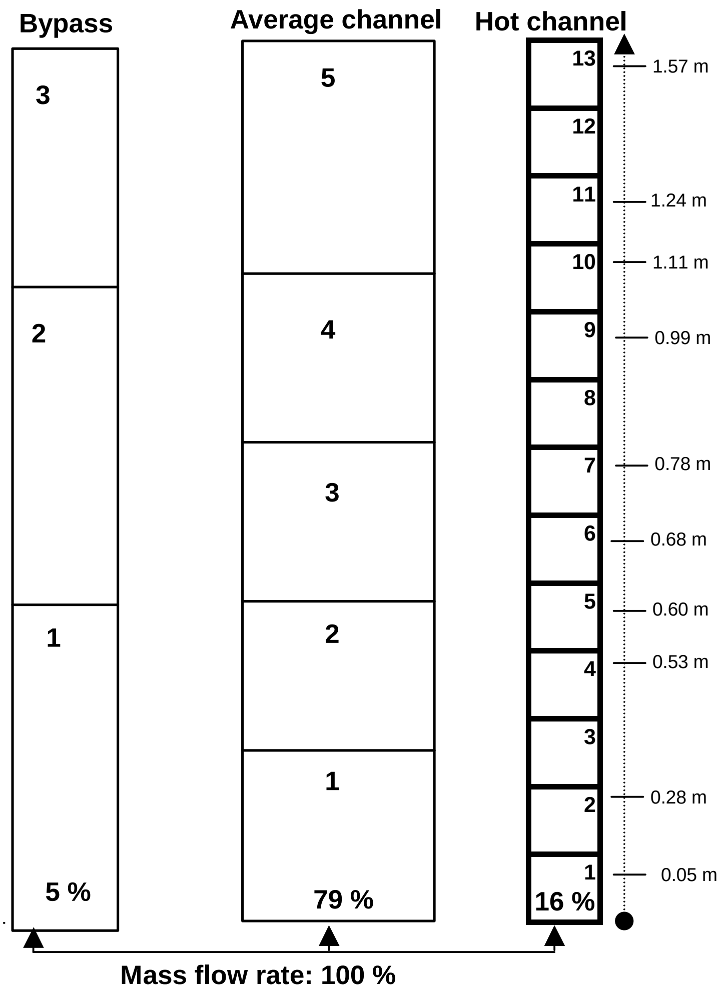

Figure 3-2 Core mass flow rate redistribution among the hot, average and bypass channels

For the reflood simulation, each axial node (Figure 3-2) of both heat structures (average and hot core) is sub-divided into maximal 64 fine nodes (fine mesh rezoning scheme) by the code depending on local thermal conditions. Hence the axial node elevations may vary from 0.7 mm up to 5 mm. During the reflood calculation, the two dimensional heat conduction for the core heat structures is activated to appropriately catch the strong axial cladding temperature variations near the quench front.

4 Comparison of code predictions with data

4.1 RELAP5/MOD3.2.2Gamma

In Tab. 4-1 the official code versions delivered by the US NRC to CAMP-members are listed indicating the status of the reflood model. Since the reflood model in the 322Beta-version exhibited some coding errors [Anal99], post-test calculations were performed with the version 322 Gamma.

Tab. 4-1 Delivered RELAP5-versions with different reflood models

Code version	Reflood model	Delivery date
RELAP5/MOD3.1	INEEL, usable	1994
RELAP5/MOD3.2	INEEL , not usable	1995
RELAP5/MOD3.2.1.2	INEEL, not usable	1996
RELAP5/MOD3.2.2Beta	PSI, usable but with a lot of errors	1998
RELAP5/MOD3.2.2Gamma	PSI, usable	1999

4.1.1 System response

The sequence of main events calculated by RELAP5 is listed in Tab. 4-2. It can be seen that the break is opening at transient begin. Immediately the RCPs are coasted down due to the assumption of loss of off-site power coincident with break opening. The primary system de-pressurises very fast so that the pressurizer becomes empty early in the transient. Once the primary system pressure is low enough, accumulators begin to add water into the core. Later on the primary circuit pressure continues decreasing and around 32 s the LPIS starts with cold water injection that leads to the core reflooding. In Figure 4-2 the predicted hot leg pressure is compared with the data. It can be seen that both trends are similar even though RELAP5 calculates a faster de-pressurization of the RCS.

The calculated PZR–behaviour is in good agreement with experimental data (Figure 4-1). Since the RCS-pumps are tripped after scram, the RCS-flow drops rapidly leading to a reverse flow in all intact legs and in the broken cold leg as exhibited in Figure 4-3 and Figure 4-4. The predicted trends are qualitatively in good agreement with measured ones. However RELAP5 tends to under-predict the mass flow rates in the intact loop (hot and cold legs). On the contrary the mass flow rate though the broken cold leg is slightly over-predicted during the first *15 s*.

Consequently the total break outflow calculated by RELAP5 is over-predicted for the first *10 s*, and it is slightly under-predicted afterwards, Figure 4-5. The net mass flow rate though the core is given in Figure 4-6. It can be seen that reverse flow prevails until accumulators begin to inject cold water into the intact cold leg. The corresponding water levels in the hot and average channel are represented in Figure 4-7. The core uncovering begins with the blow-down phase reaching its minimum level at about 25 s. Later on the water level increases due to the LPIS-activation.

Tab. 4-2 Main events predicted by RELAP5

Events	**Time (s)**
Break opening	0.0
Scram	0.13
MCP-coastdown	0.6
Pressurizer empty	15.5
ACC-injection (begin/end)	17.5/37
Core uncovered (from/to)	20/25
LPIS-injection (begin/end)	32/100
End of calculation	100

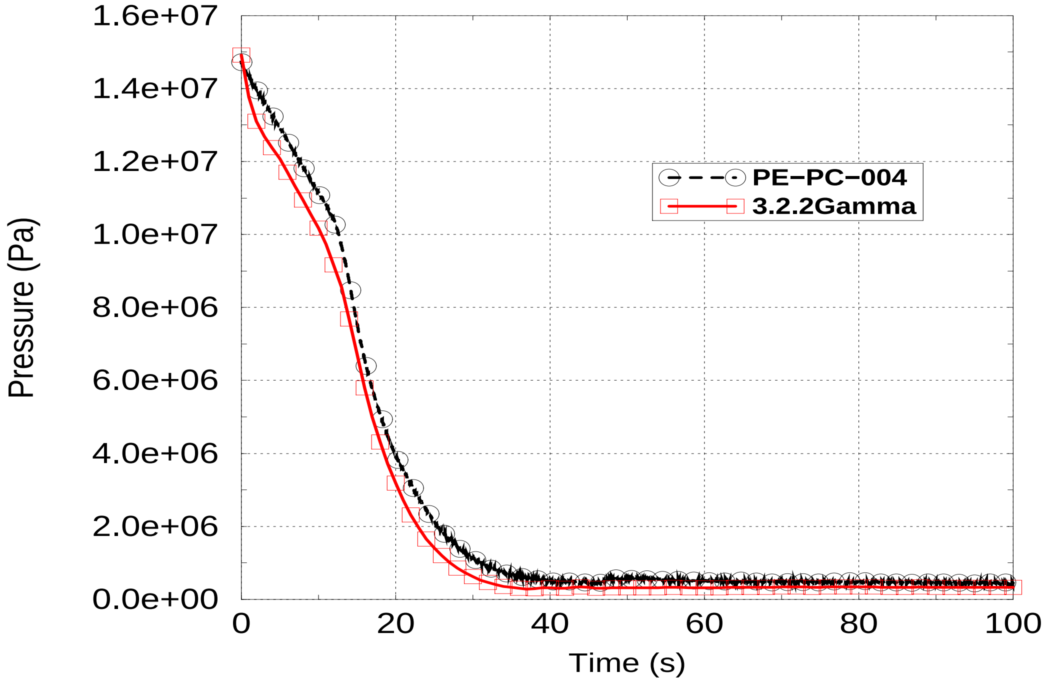

Figure 4-1 Calculated and measured pressurizer pressure

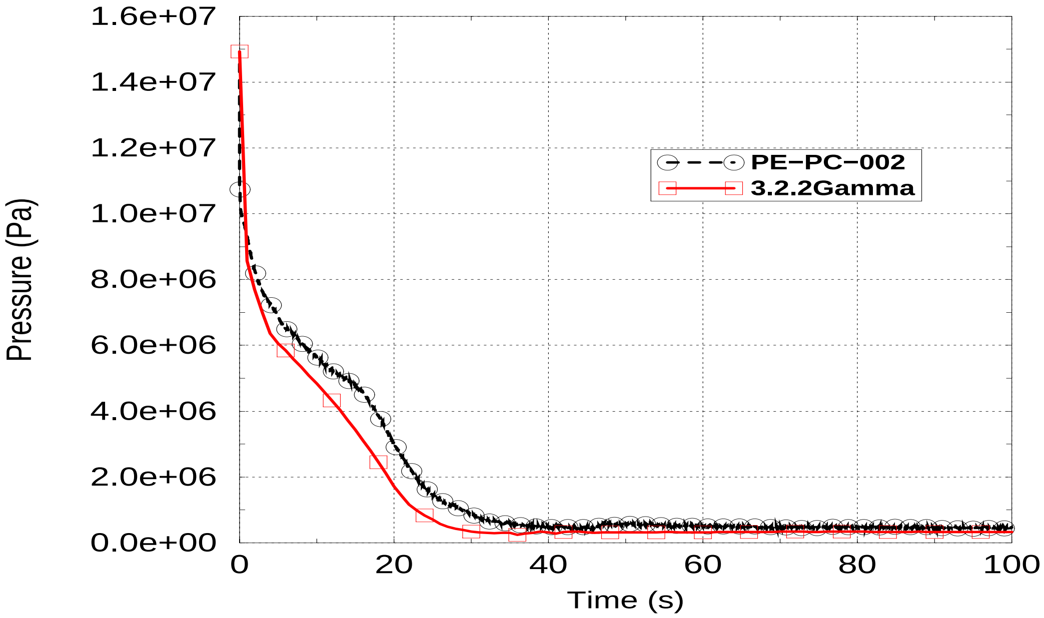

Figure 4-2 Predicted and measured hot leg pressure

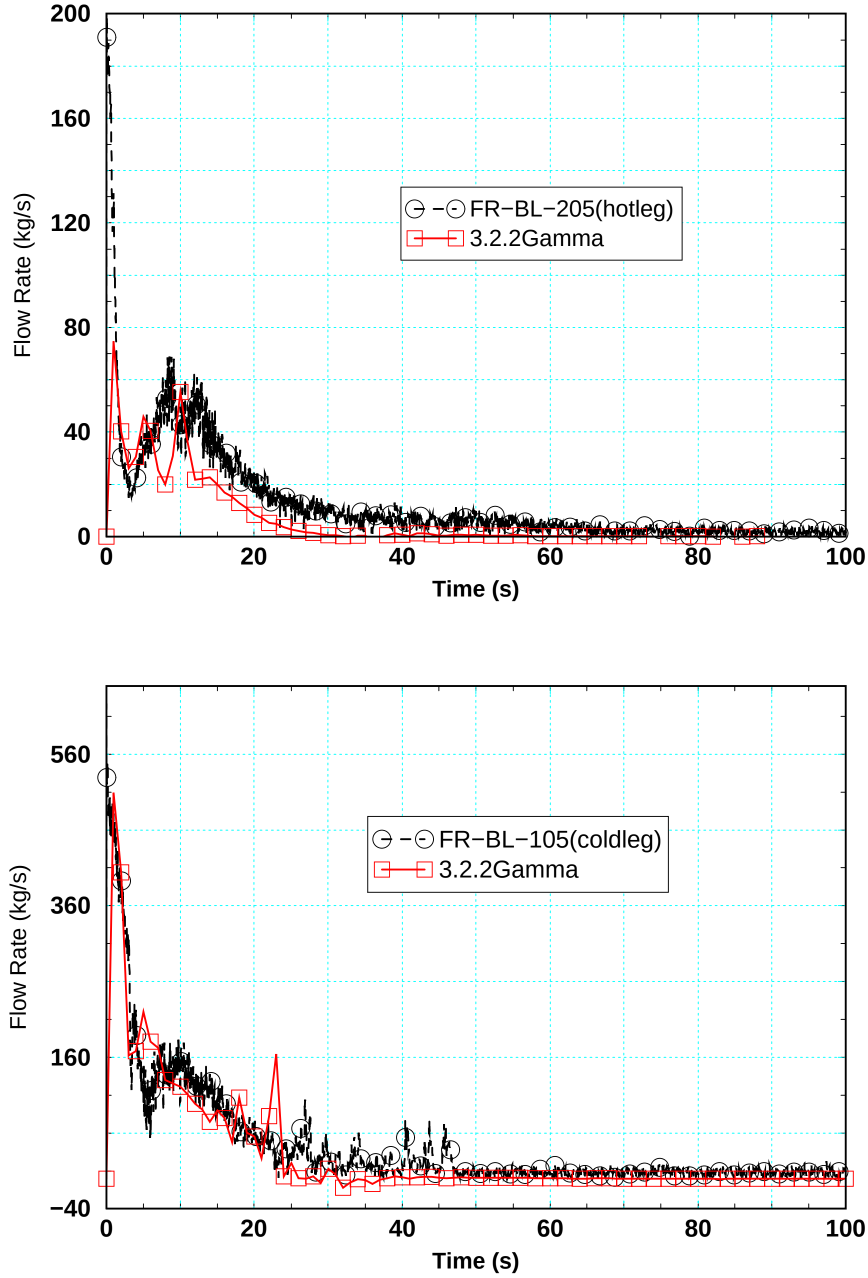

Figure 4-3 Calculated and measured mass flow rate through the broken hot/cold legs

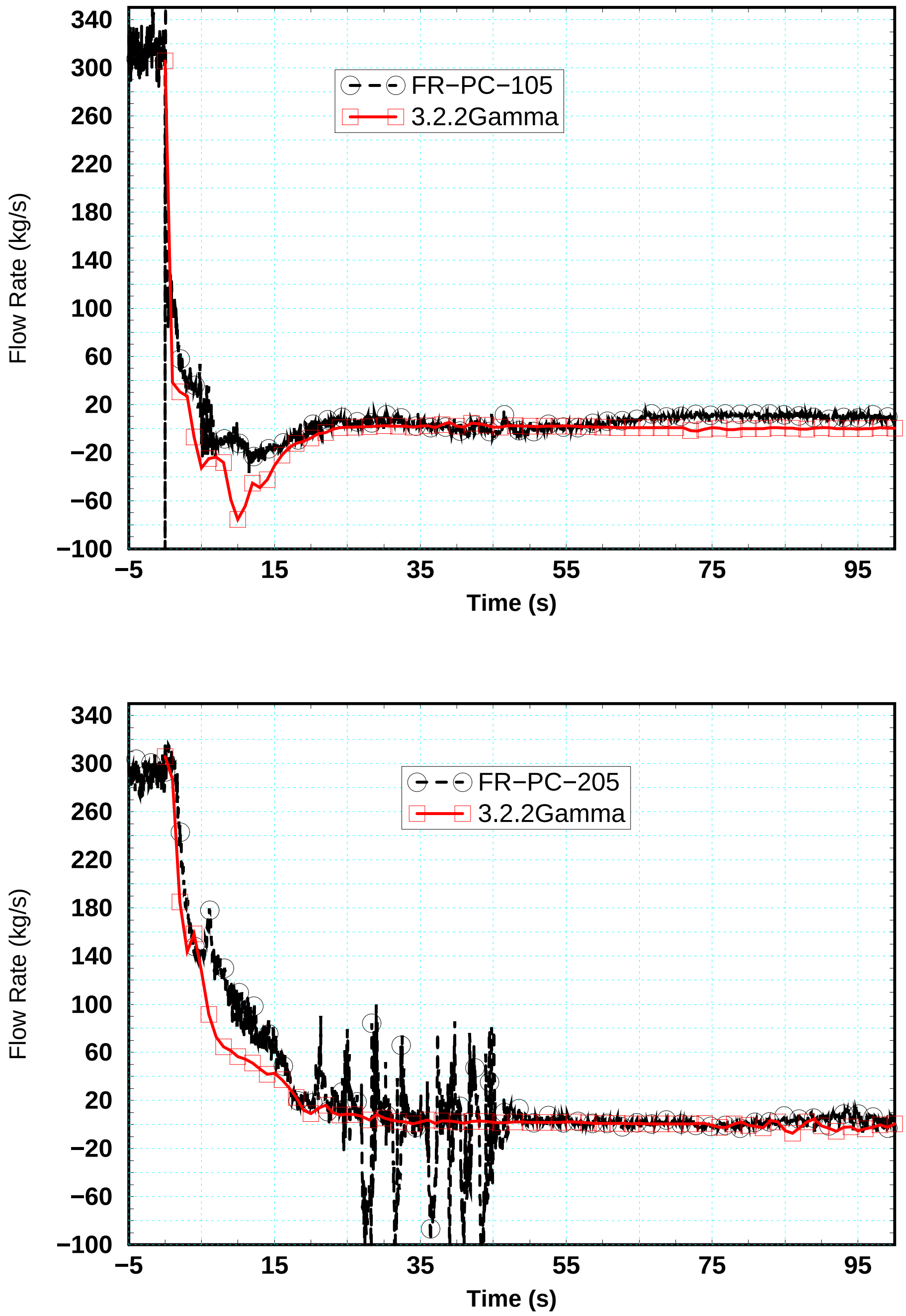

Figure 4-4 Calculated and measured mass flow rate through the intact hot/cold legs

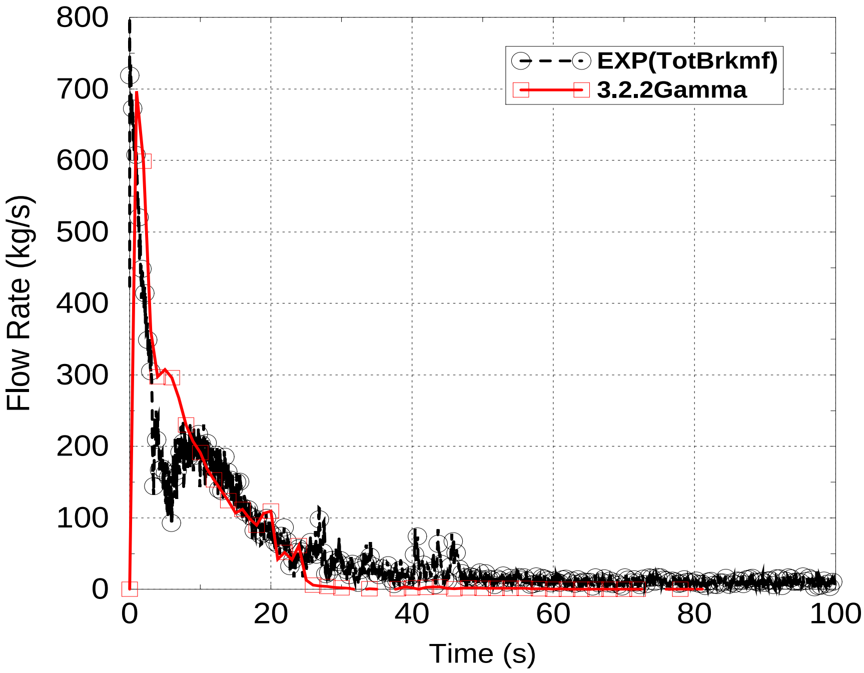

Figure 4-5 Calculated and measured total outflow through both breaks

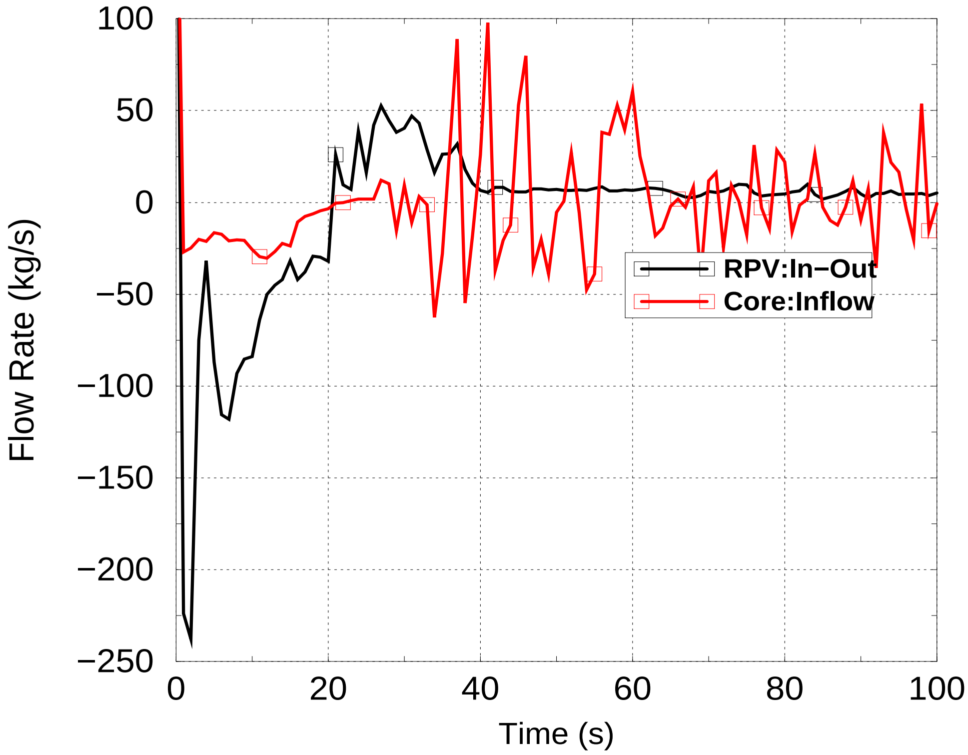

Figure 4-6 Calcualted net mass flow rate through the core

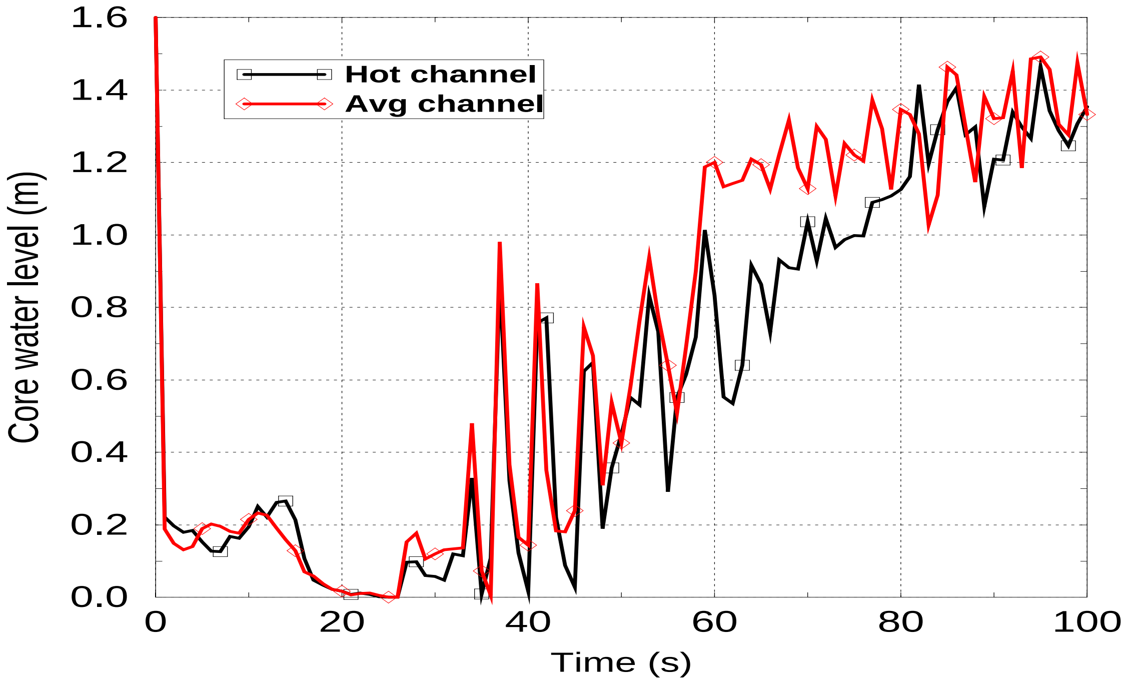

Figure 4-7 Calculated water level in hot and average core channel

A comparison of calculated and measured accumulator and LPIS flow rate injected into the RCS is showed in Figure 4-8 and Figure 4-9.

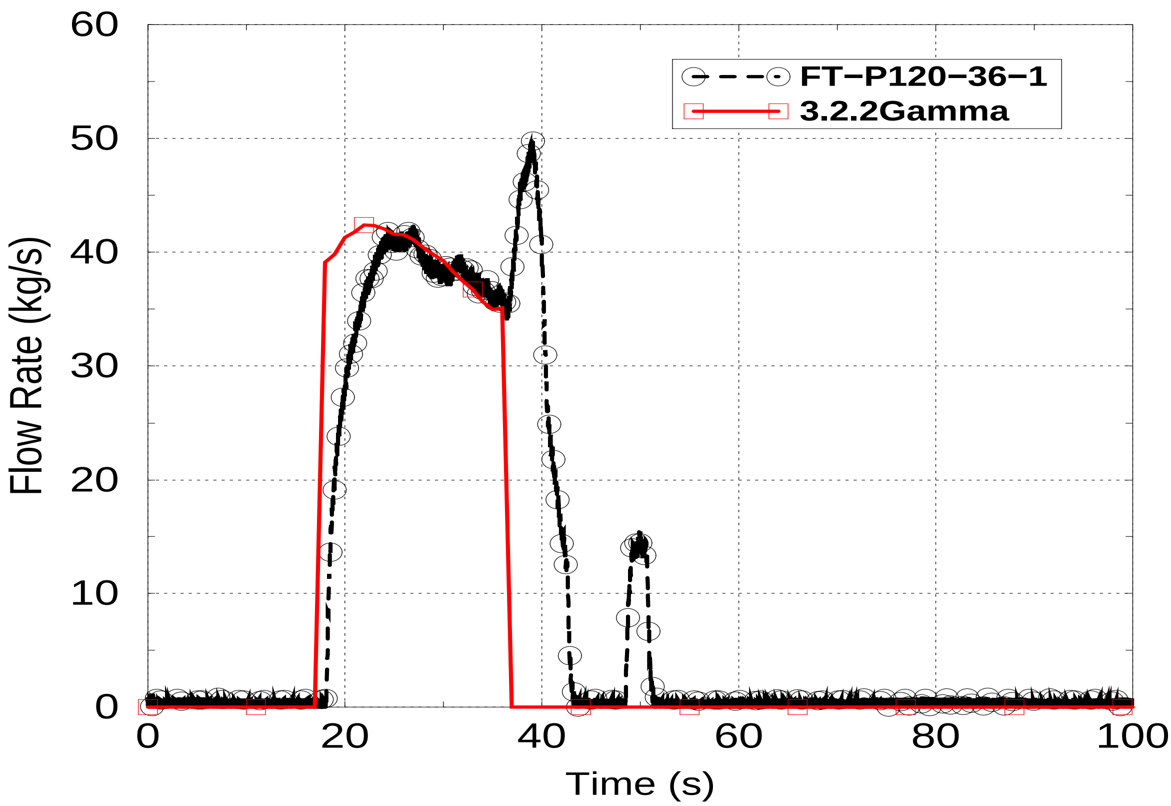

Figure 4-8 Calculated and measured accumulator mass flow rate

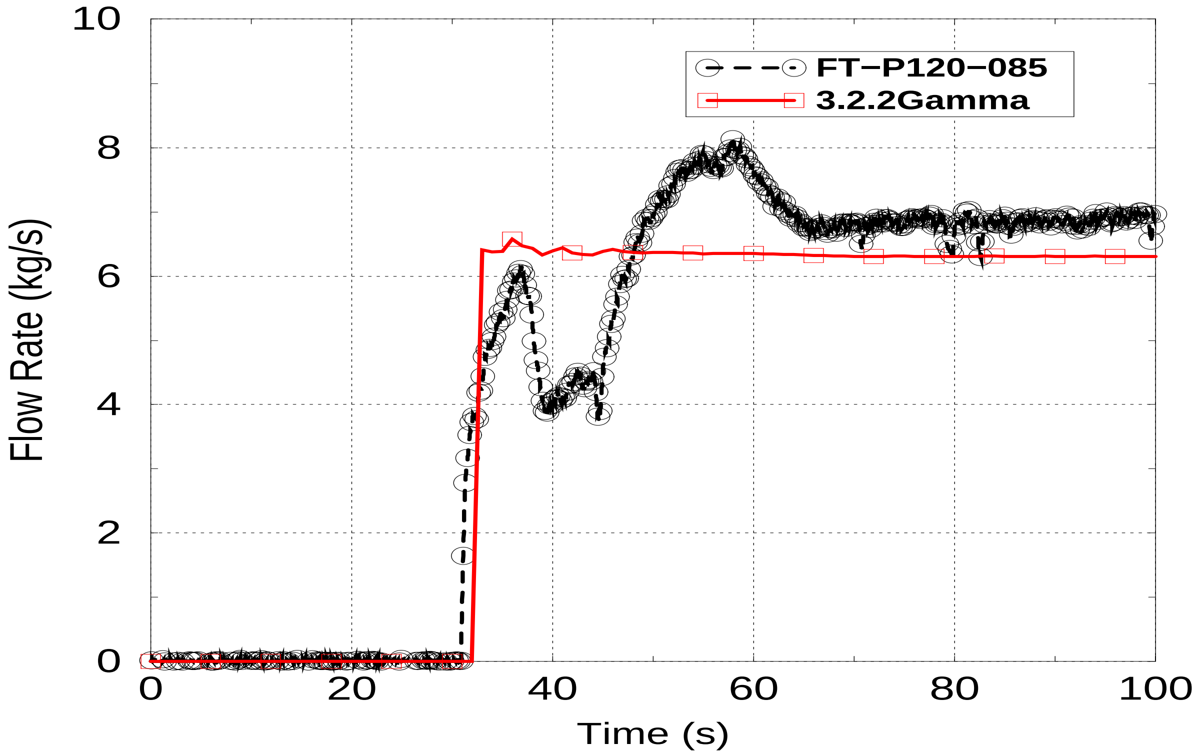

Figure 4-9 Calculated and measured LPIS-mass flow rate

4.1.2 Core behaviour

Since the major objective of large break LOCA investigations is the determination of the core thermal response, a comparison of calculated and measured fuel rod temperatures at different axial elevations is required. The main issue of interest is the cladding temperature history and the time of core rewetting, since the latter marks the onset of long-term core cooling.

For comparison purposes an average temperature is calculated at each axial elevation taking into account all thermocouples signals that are distributed within the central bundle. As illustration Figure 4-10 shows the average temperature (calculated) compared to the thermocouple signals for the same elevation (0.68 m).

The overall core heat-up during the accident progression is illustrated in the next figures, where a comparison of experimental data with results of calculations is given, Figure 4-11 to Figure 4-15. It can be seen that during the early blow-down phase all fuel rod segments show a rapid temperature escalation due to the loss of convective heat transfer caused by loss of flow. Also the reverse flow core flow conditions established after transient begin contribute to the heat-up.

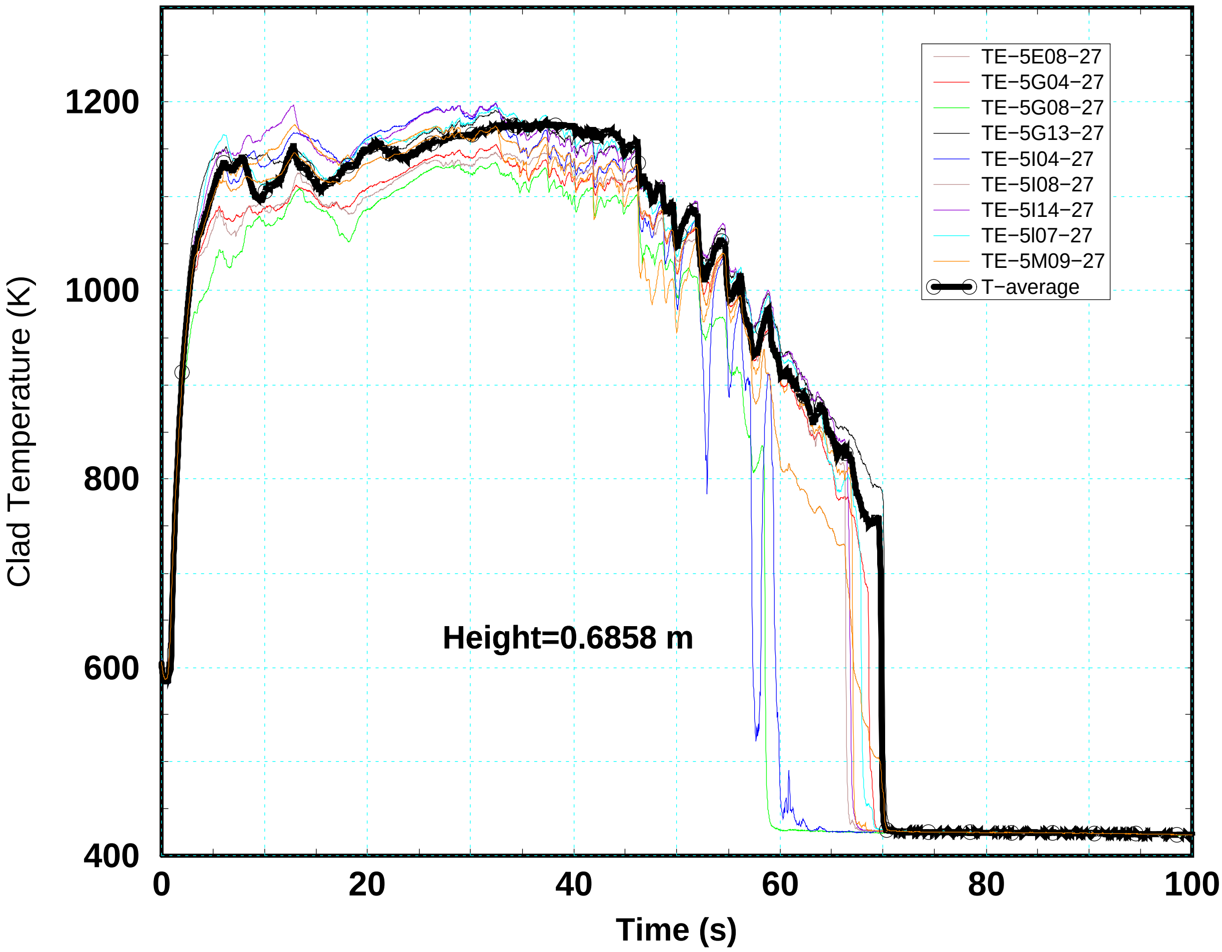

Figure 4-10 Measured cladding temperature at elevation 0.6858 m

Generally, core heat-up trends obtained from RELAP5 and from the experimental data are similar (Figure 4-11 to Figure 4-15) in nearly all elevations except for the lowest and highest segments. In the lowest segment the code predicts a higher peak cladding temperature. One of the reasons for this over-prediction may be the use of the Chen-correlation in RELAP5/MOD3.2.2Gamma during the pre-quenching phase (first 20 s) where the reflood model is not yet activated. It was shown in [Sanc97] that the original Chen correlation is problematic since it was developed for high pressure conditions and it did not take into account for quality effects. The discrepancy between experiment and calculation becomes smaller in the middle part of the core height. In the uppermost elevations the peak cladding temperature is strongly under-predicted (elevation 1.57 m) especially during the first 12 s. At about 13 s after break opening the cladding temperature in the upper third of the bundle experiences a rapid decrease. It is attributed to the so called top-bottom quenching where liquid from the upper plenum, formed by entrained droplets and steam condensation, falls back into the core driven by gravity. The refill phase begins at about *18 s* with the injection of the accumulators. The reflood model, i.e. the two dimensional heat conduction and the reflood heat transfer package, is activated at around 22 s by a pressure initiated trip (P<1,2 MPa and the void> 0.1).

In RELAP5 the reflood heat transfer mode is given by 40 plus the normal heat transfer mode of non-reflood situations. In Figure 4-16 the heat transfer mode 48 (htmode) indicates that the actual heat transfer mode is saturated film boiling i.e. htmode equal to 48.

In the corresponding Figure 4-17 the reduction of the heat transfer coefficient when the heat transfer mode changes from convective heat transfer to liquid (high HTC) to convective heat transfer to vapour (low HTC).

The bottom-top quench front starts to move slowly, driven by the LPIS-injection. The resulting collapsed water level shows oscillations but increases continuously (Figure 4-7). After reflood begin the fuel rods are mostly exposed to sub-cooled and saturated film boiling for a long time (htmode 47 and 48), which is characterized by low heat transfer coefficient (Figure 4-17). The core rewetting due to the upwards moving quench front along the core height is characterized by the sudden and strong cladding temperature reduction. It can be observed in almost all axial segments (Figure 4-11 and Figure 4-15). Such considerable temperature decrease is caused by the rapid flow regime change from film/transition boiling to nucleate boiling (htmode 43 or 44). The latter one is characterized by high heat transfer coefficients. This behaviour can be clearly observed in heat transfer coefficient development (e.g. Figure 4-17). This behaviour is representative for all fuel rod segments.

In Figure 4-18 a comparison of the calculated and measured quench temperature for the LOFT LP-LB-1 test is given. It can be observed that the quench temperature is always underpredicted by the RELAP5/MOD3.2.2Gamma in all axial elevations. The reason for it is the use of the empirical Weisman-correlation in the PSI-reflood model to describe the transition heat transfer.

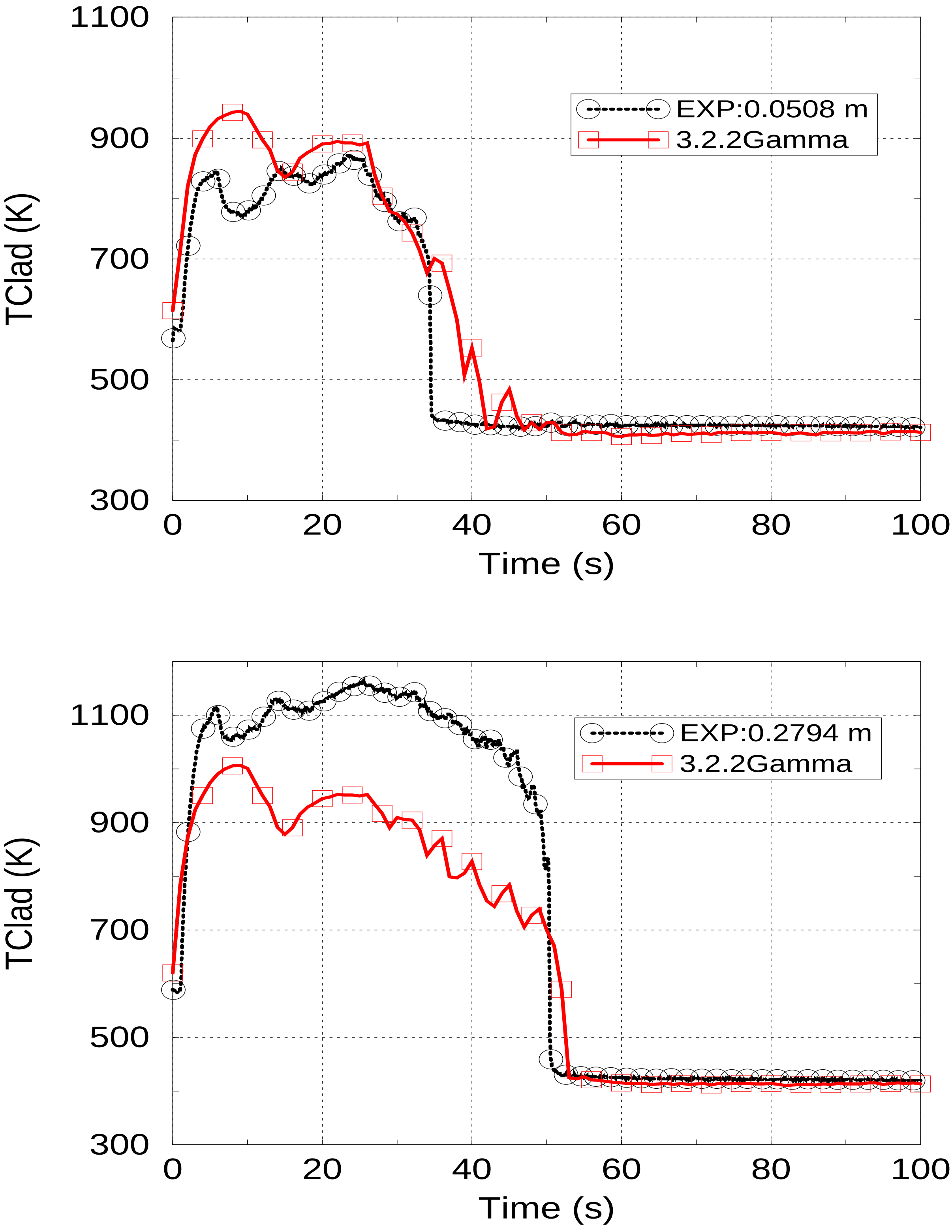

Figure 4-11 Predicted and measured cladding temperature at 0.05 and 0.27 m (hot channel)

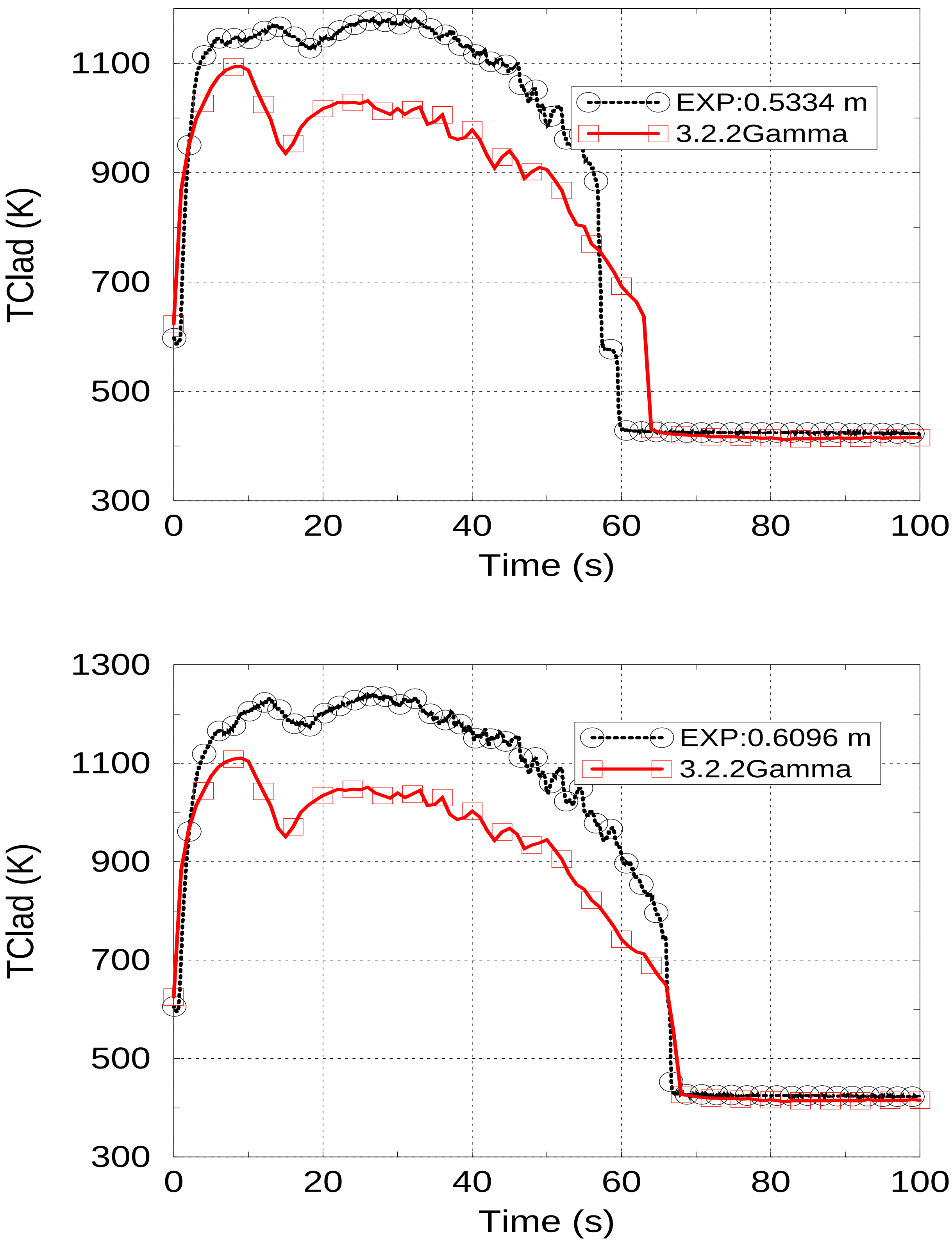

Figure 4-12 Predicted and measured cladding temperature at 0.53 and 0.6 m (hot channel)

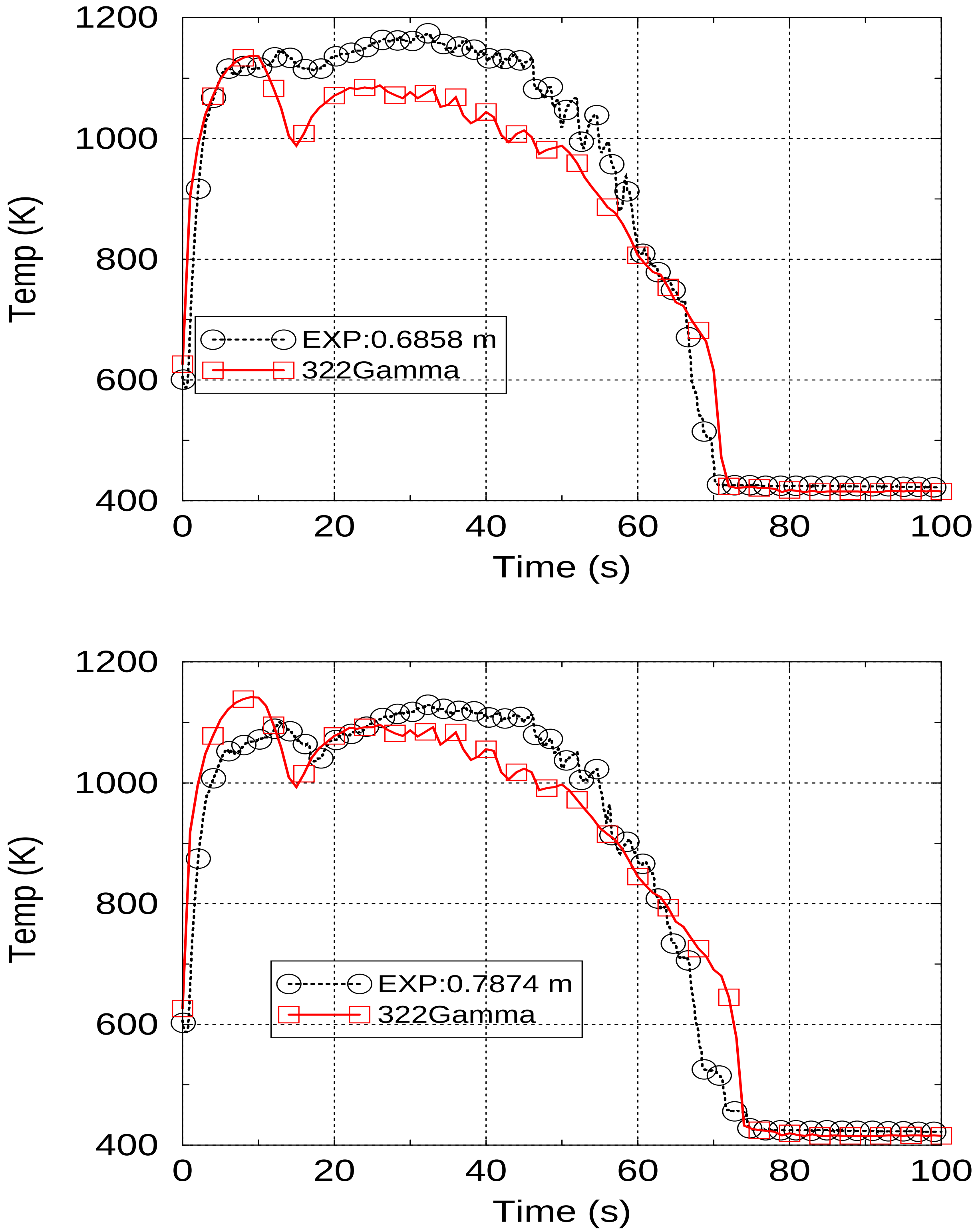

Figure 4-13 Predicted and measured cladding temperature at 0.68 and 0.78 m (hot channel)

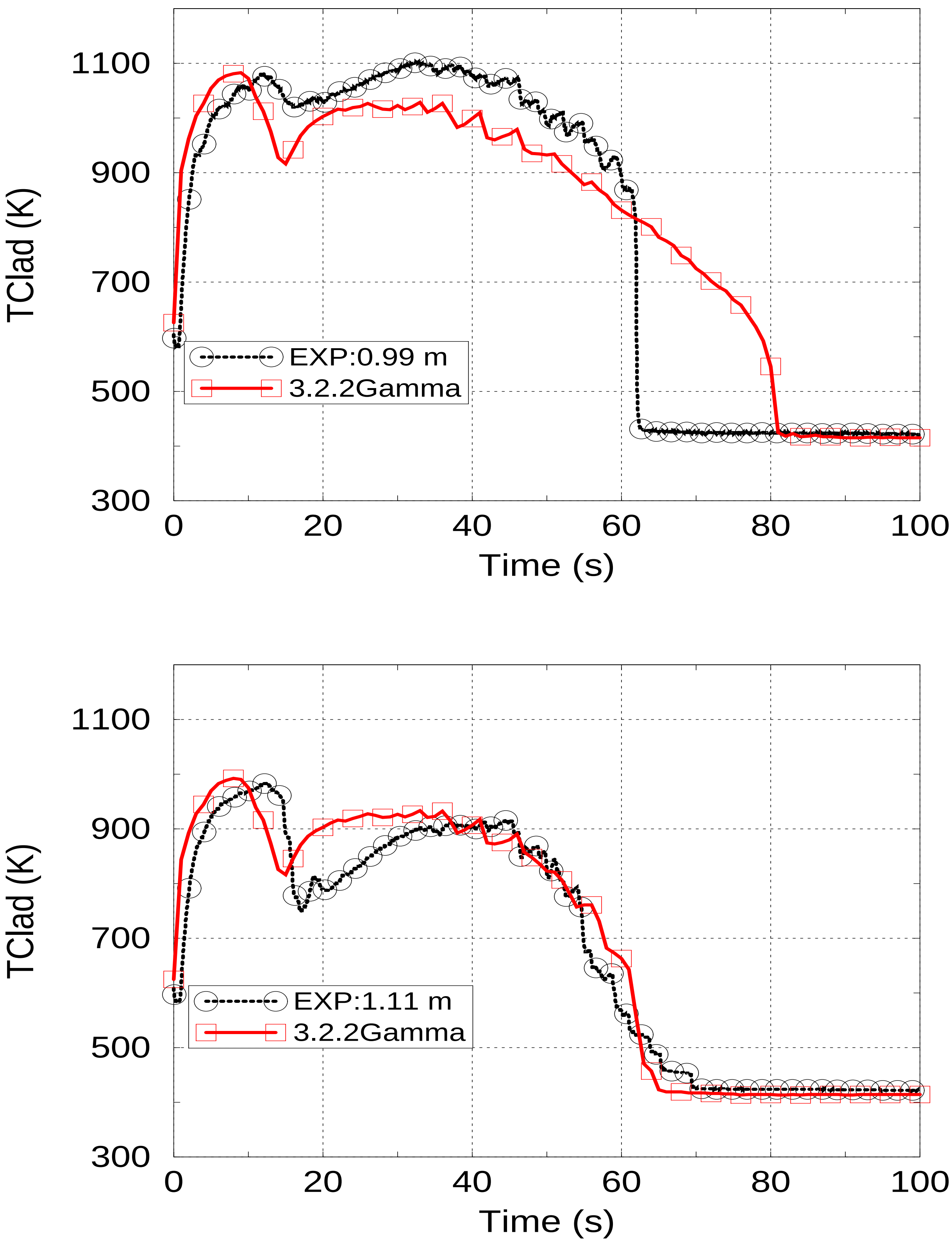

Figure 4-14 Predicted and measured cladding temperature at 0.99 and 1.11 m (hot channel)

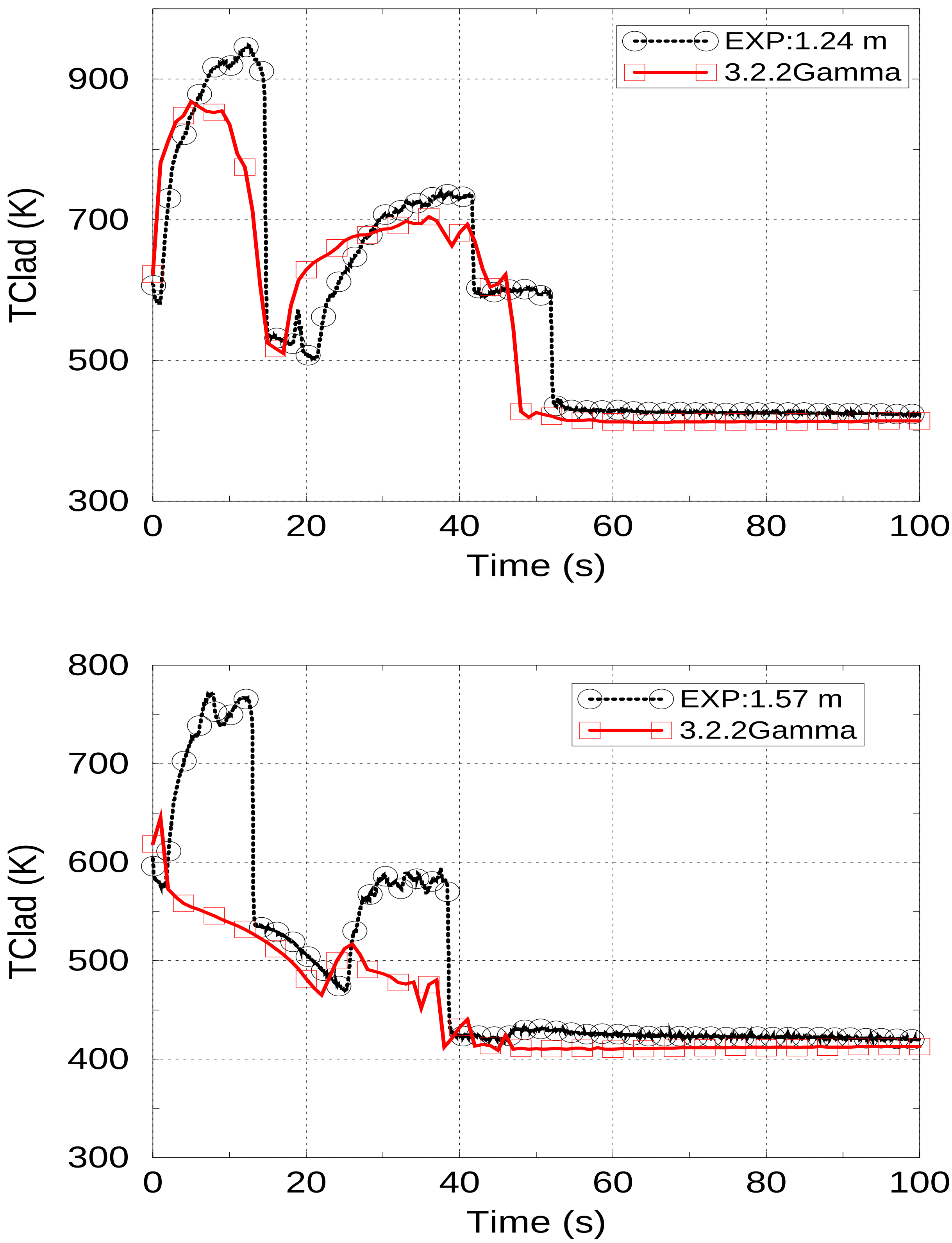

Figure 4-15 Predicted and measured cladding temperature at 1.24 and 1.57 m (hot channel)

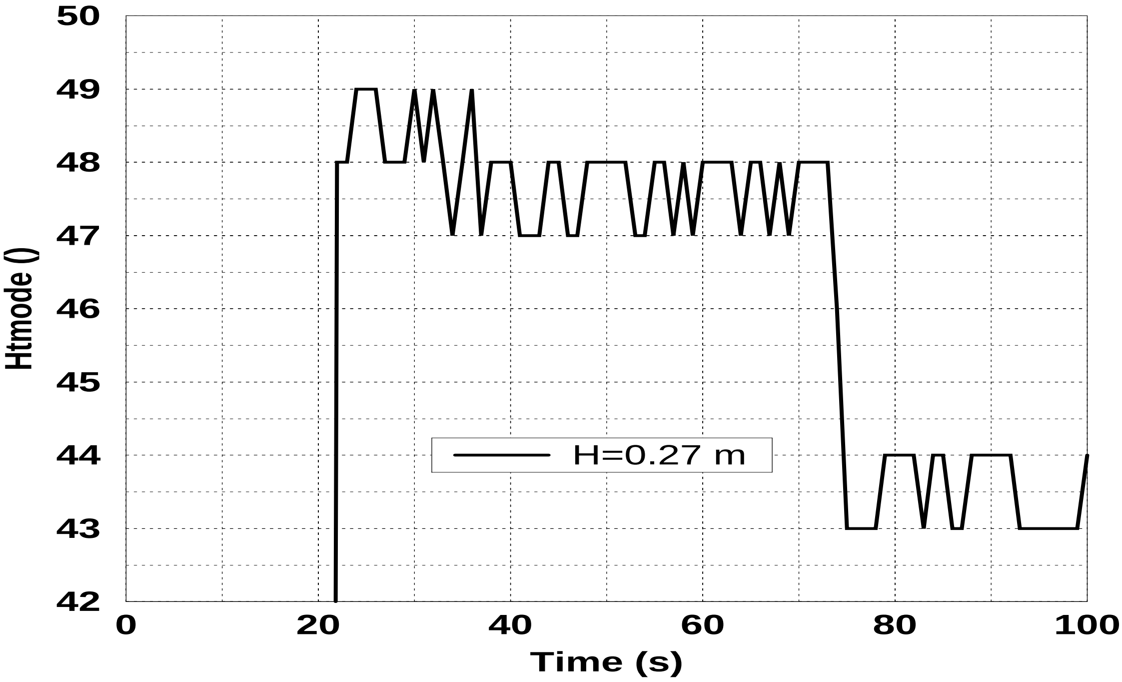

Figure 4-16 Predicted heat transfer modes at 0.27 m elevation

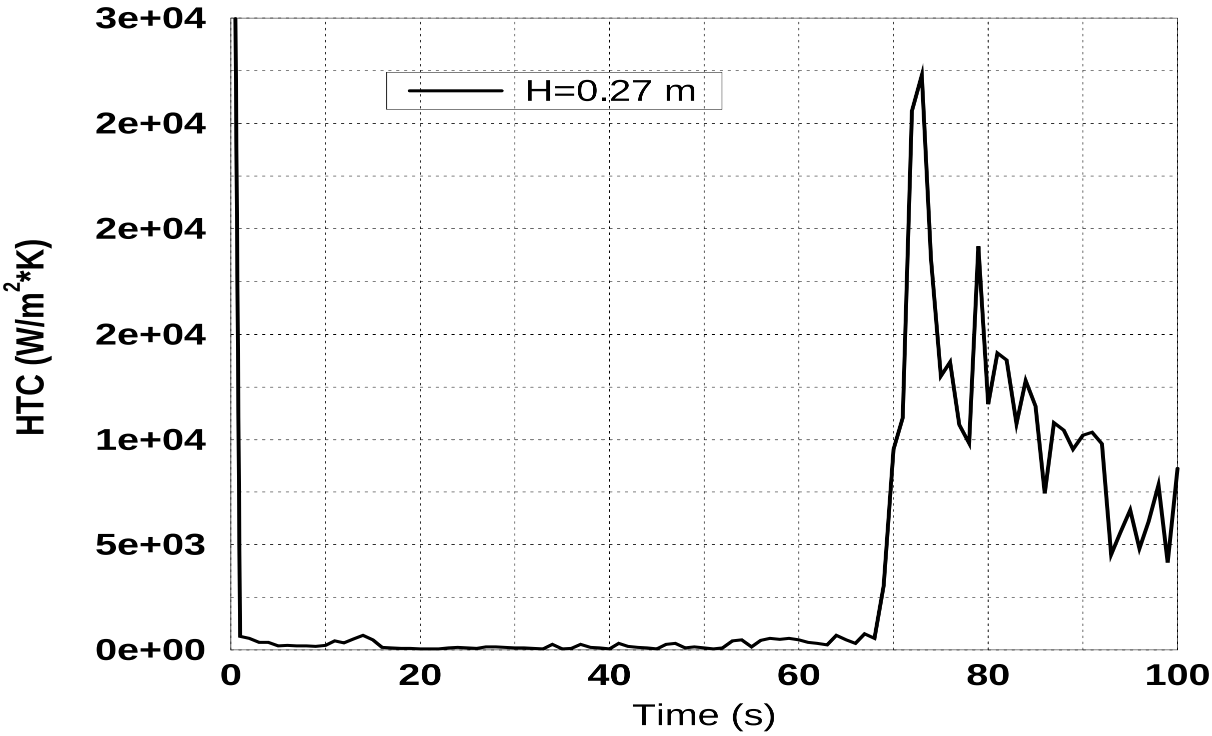

Figure 4-17 Predicted heat transfer coefficient at 0.27 m elevation

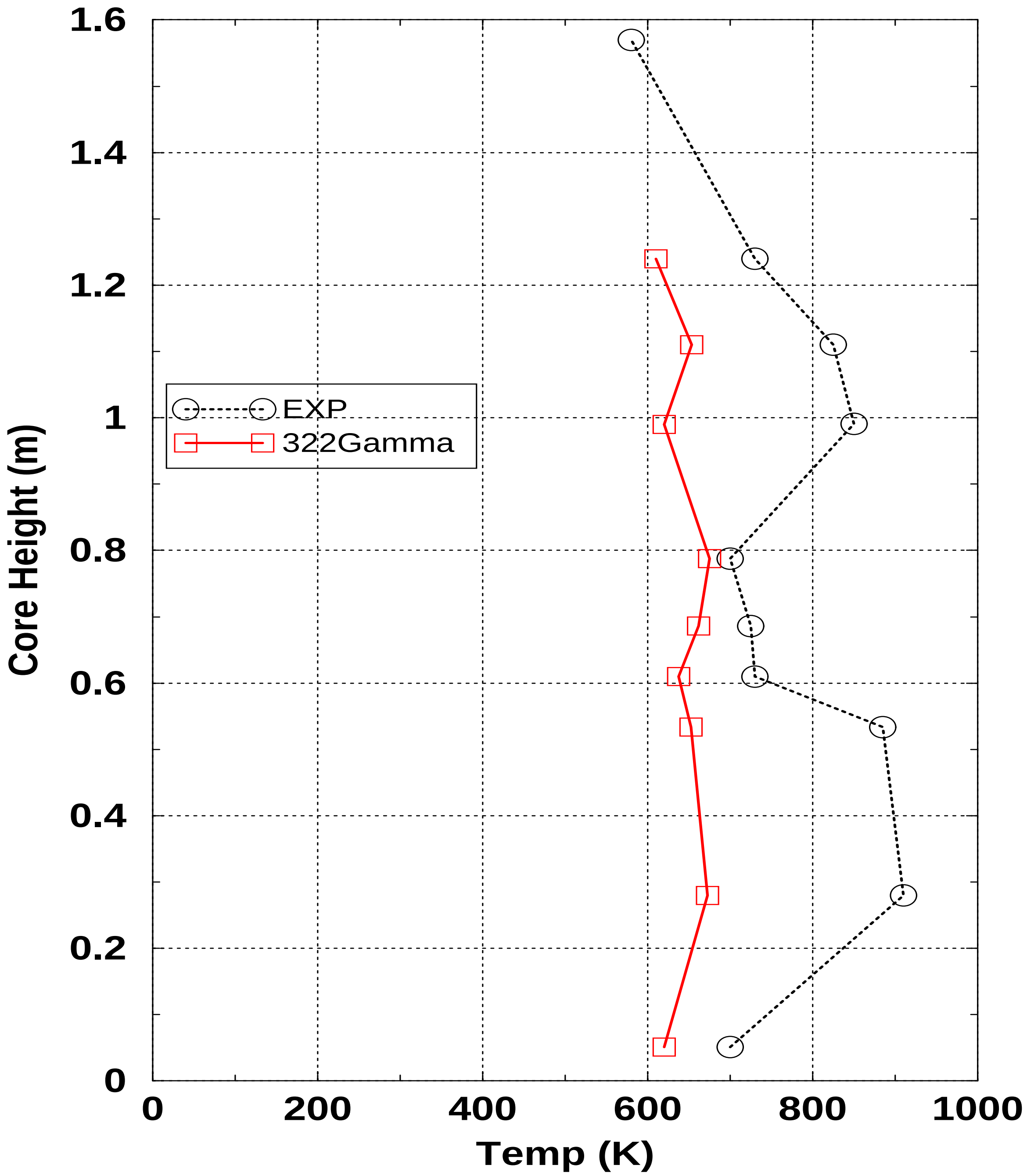

Figure 4-18 Predicted and measured quench temperature along the core height

4.2 RELAP5/MOD3.2.2Gamma+FZK

4.2.1 Transition boiling heat transfer improvements

One of the major shortcomings of the PSI-reflood model was the use of the empirical Weisman correlation to predict the transition boiling heat transfer. It was shown in the previous section, that the Weisman correlation over-predicts the heat transfer coefficient resulting in quench temperatures which are always below the measured data. Hence the FZK-transition boiling model that represents an extension of the semi-mechanistic Chen-approach, was implemented in 322Gamma instead of the Weisman-correlation. This code version was named to 322Gamma+FZK.

Post-test calculations of the LOFT-LB-LP-1 were carried out with the improved version 322Gamma+FZK using the same input. Since the modifications are related to the transition boiling heat transfer, comparison of results with data are focused on core thermal behaviour.

Since the system behaviour predicted by both code versions is similar, the discussion hereafter is focused on the fuel rod thermal behaviour during the transient progression.

4.2.2 Core behaviour

In Figure 4-19 to Figure 4-23 the cladding temperature calculated by both the original (322Gamma) and the modified (322Gamma+FZK) code versions are compared to each other including experimental data. One can see that results of both predictions are identical for the pre-quench phase in all axial elevations. With the beginning of the reflood phase predictions of both versions start to diverge as a result of the different transition boiling correlations. These discrepancies become larger close to the quench front and especially in the transition boiling flow regime.

One can state that the results of the code version 322Gamma**+**FZK are closer to the measured data compared to the ones of the original version in almost all axial nodes. The measured cladding temperature trend for the upper core elevations, i.e. 1.57 m, is strongly affected by the top-down quenching at 16 s.

In all plots it can be noted that the improved code version 322Gamma+FZK predicted the rapid decrease of the cladding temperature with the “knee” marking the sudden change from film boiling to nucleated boiling, while the original version calculates a smoother decrease of the cladding temperature.

In Figure 4-24 a comparison of the measured quench temperature along the core height with the prediction of both code versions is given. One can see that the results were considerably improved using the FZK-transition boiling model instead of the Weisman-correlation.

The sudden decrease of the fuel rod temperature during the rewetting period can be observed also in Figure 4-25, where the measured fuel centreline temperature is compared to the predictions of both code versions at two axial elevations.

Calculated temperature with the improved version shows the steep decrease and is closer to experimental data than the results of the original version.

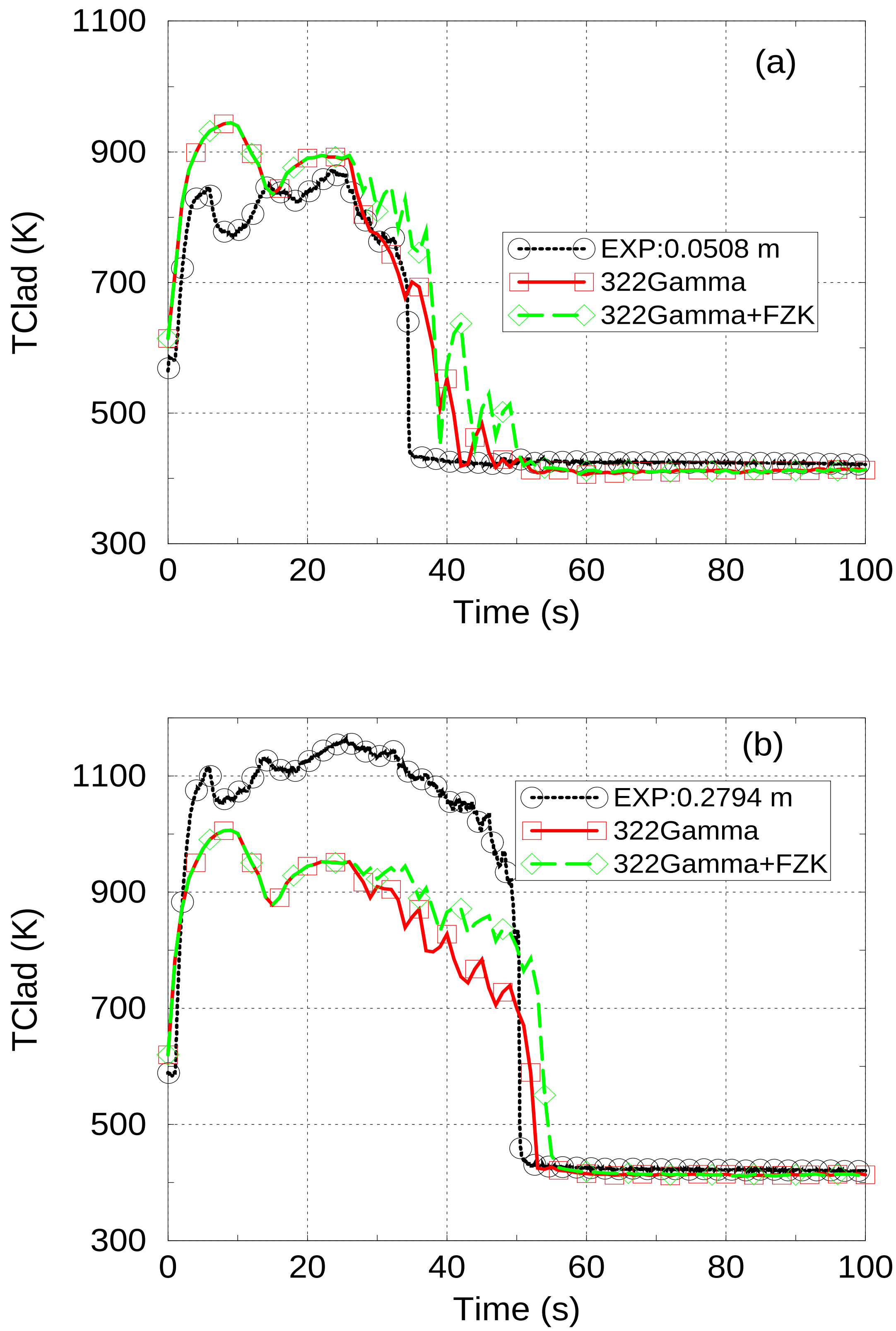

Figure 4-19 Comparison of data with predictions of cladding temperature

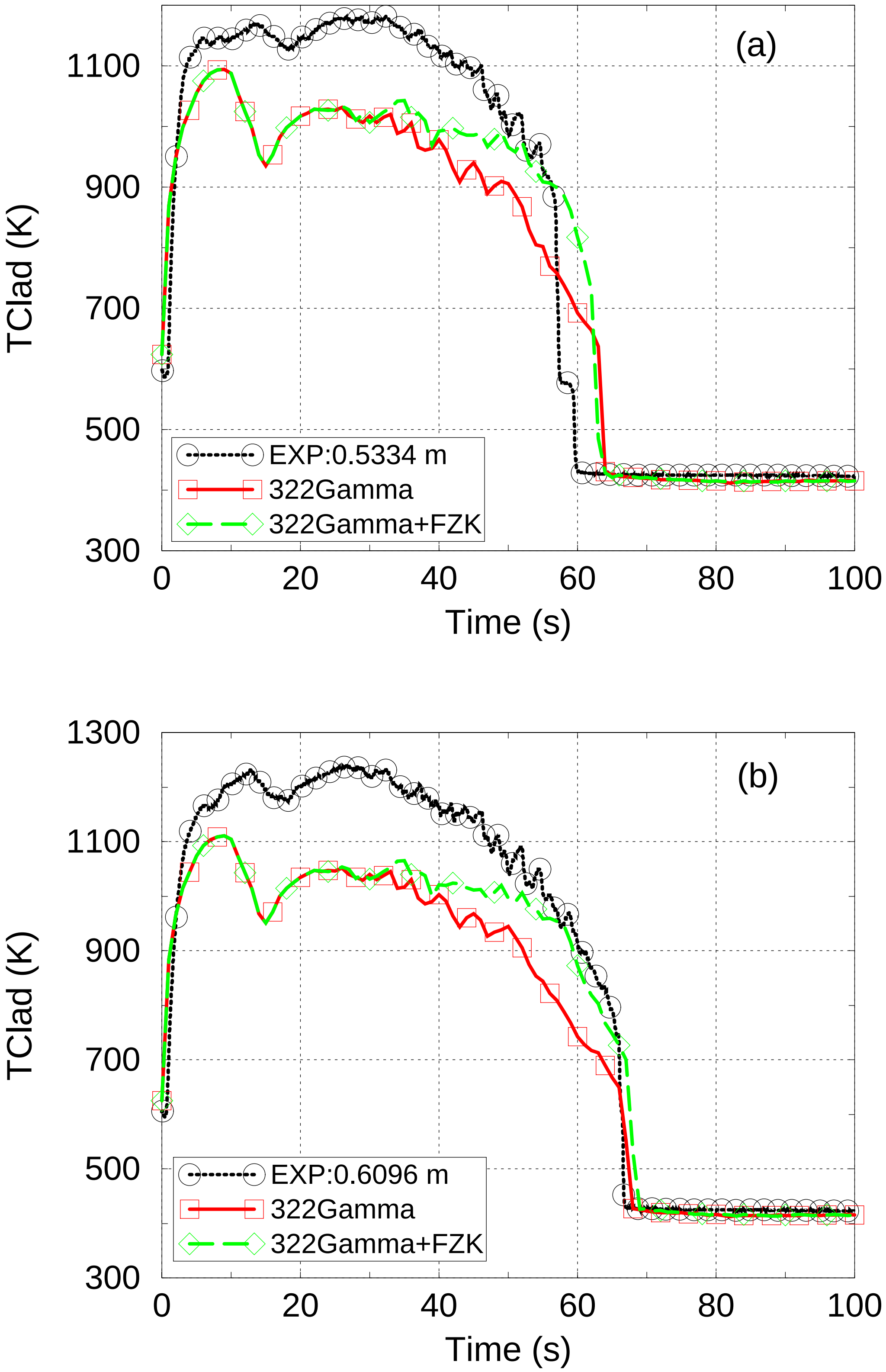

Figure 4-20 Comparison of data with predictions of cladding temperature

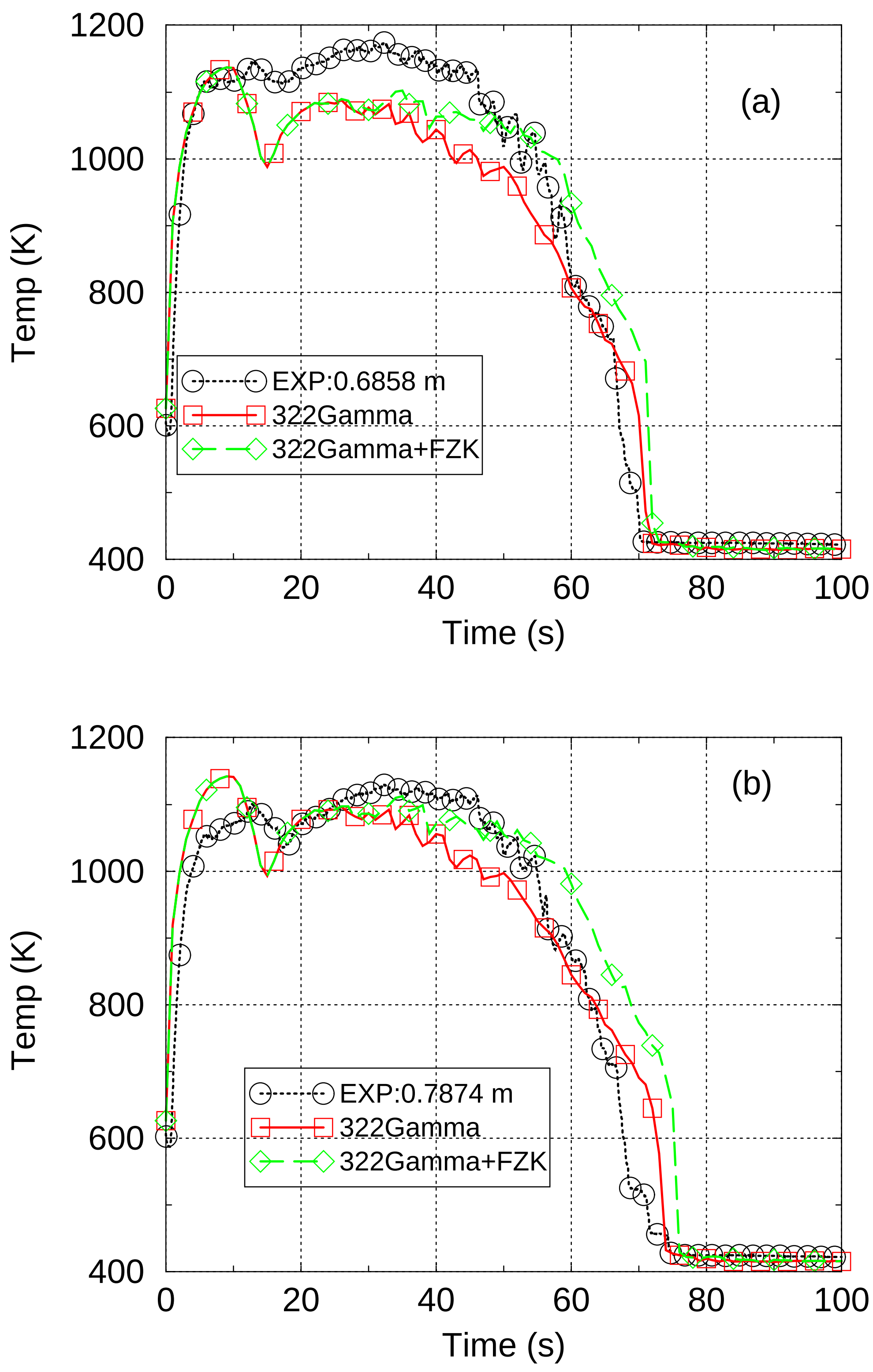

Figure 4-21 Comparison of data with predictions of cladding temperature

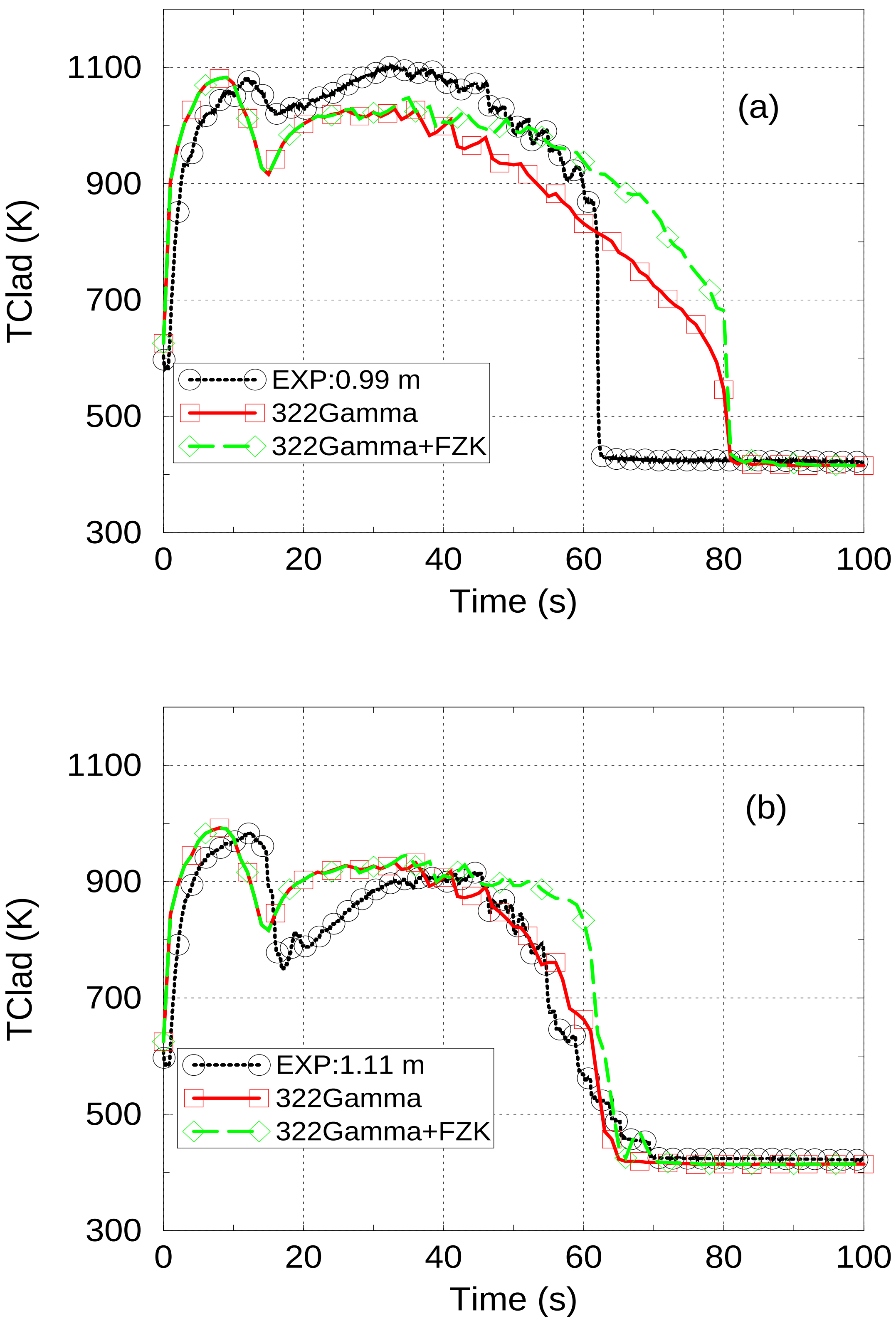

Figure 4-22 Comparison of data with predictions of cladding temperature

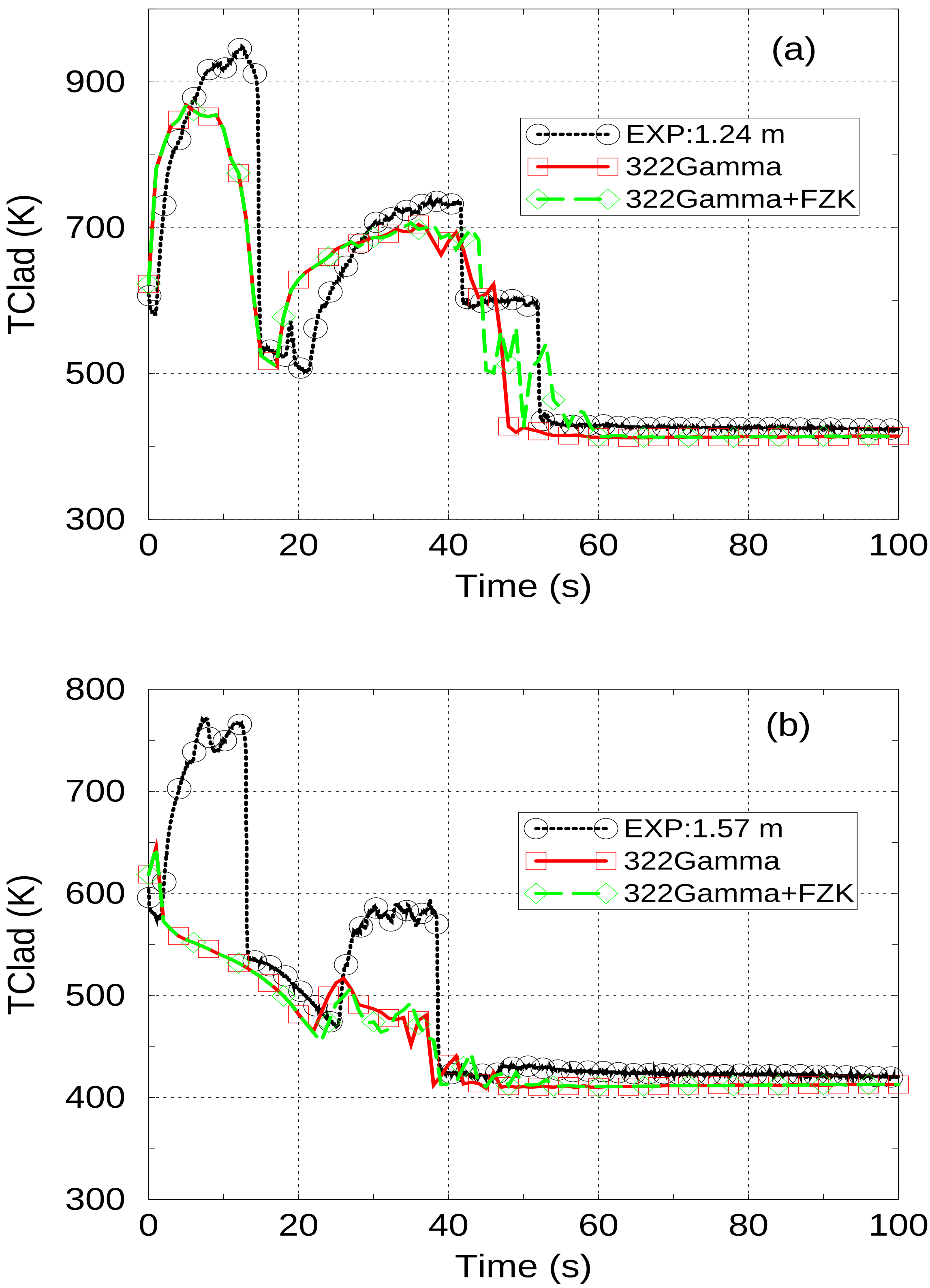

Figure 4-23 Comparison of data with predictions of cladding temperature

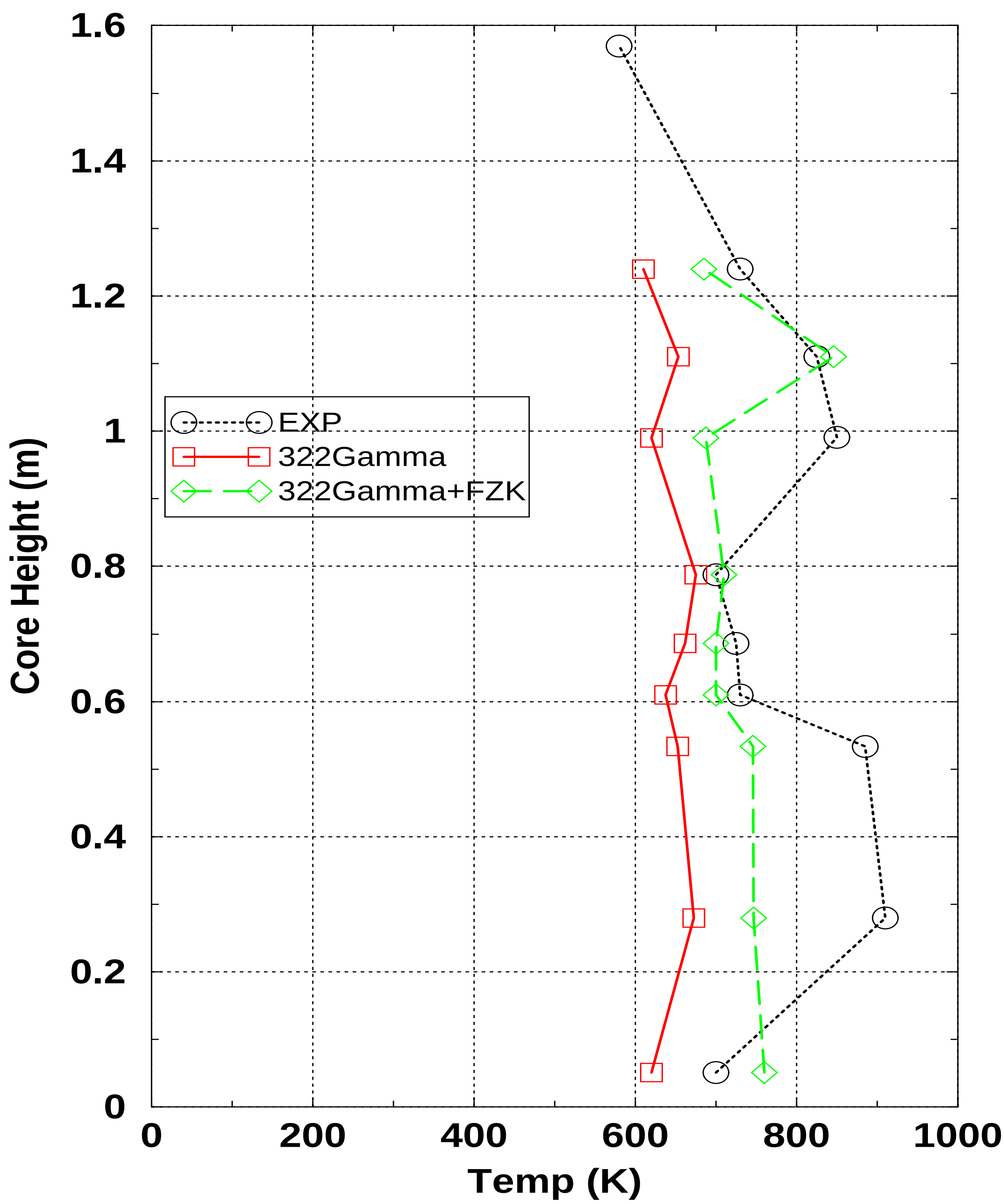

Figure 4-24 Comparison of predicted quench temperature versus data

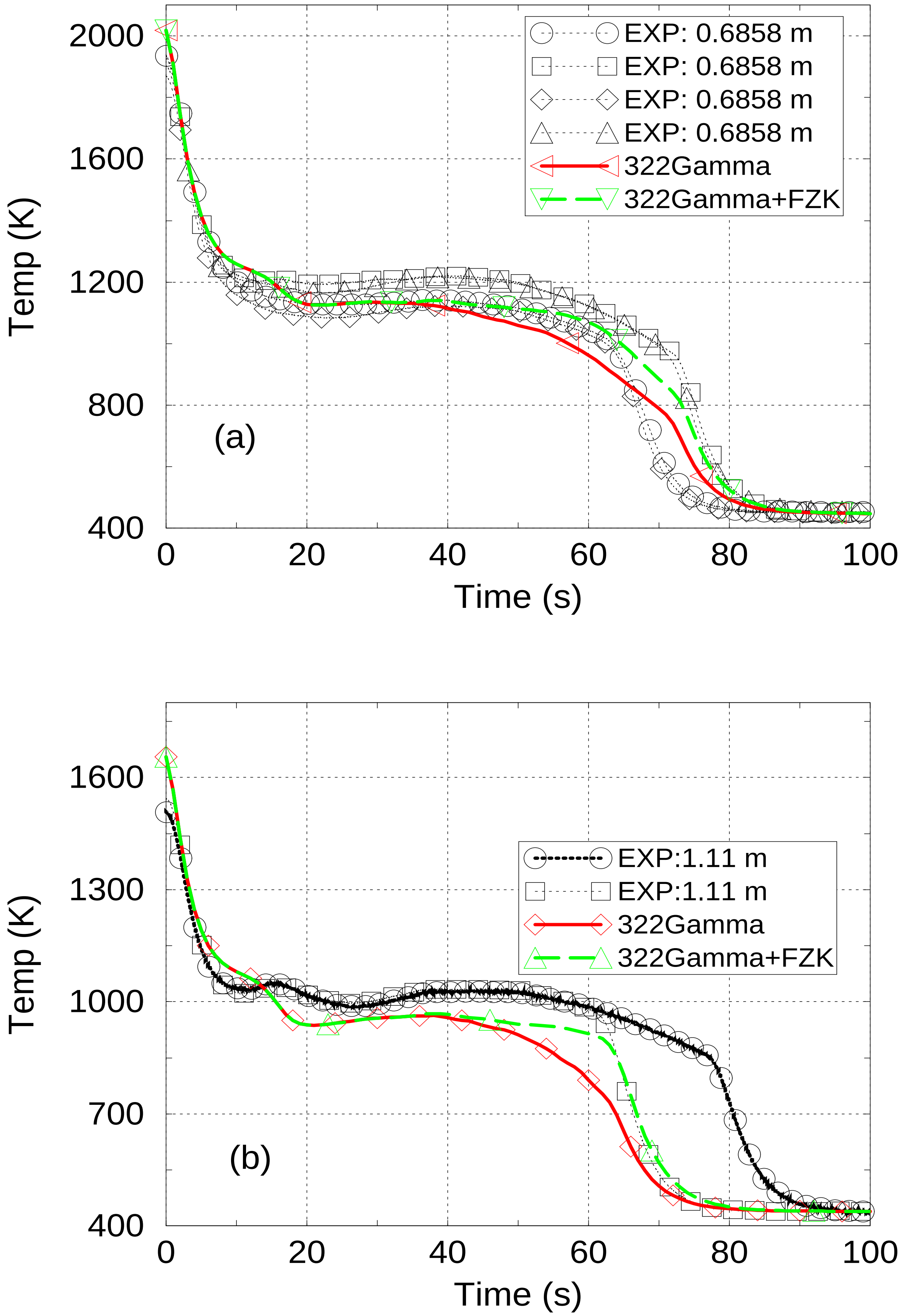

Figure 4-25 Comparison of data with predictions of fuel centre line temperature

5 Summary and conclusions

The PSI-reflood model of the RELAP5/MOD3.2.2Gamma-version was validated against the LOFT LP-LB-1 test data. The post-test calculation with RELAP5 was performed using a LOFT-model developed by PSI. Based on the results obtained with 322Gamma it can be stated that the PSI-model is capable to qualitatively predict the overall trends of the reflood heat transfer mechanisms. It represents a great progress compared to the reflood prediction capability of earlier code versions. The predicted cladding temperature in the majority of axial segments is in acceptable agreement with experimental data except for the uppermost level. This agreement is quite good in the centre part of the core. But the original RELAP5-version tends to under-predict the cladding temperature, especially the rewetting temperature which is always below the measured data. This is attributed to the empirical Weisman correlation which is used to determine the transition boiling heat transfer coefficient.

To improve the transition boiling heat transfer prediction, the Weisman-correlation was replaced by the FZK-Chen transition boiling model which was especially developed for reflood conditions under low pressure and low mass flux. A new code version was generated, named RELAP5/MOD3.2.2Gamma+FZK.

The re-evaluation of LOFT LP-LB-1 with 322Gamma+FZK showed a much better agreement of the predicted rewetting temperatures with measured data. In addition it was demonstrated that the PSI-reflood with the FZK-model leads to an overall improvement of the RELAP5-reflood simulation for the LOFT-LP-LB-1 test.

Concluding can be stated that the quality of reflood-simulations is greatly influenced by two phase flow mechanisms e.g. droplets behaviour and inter-phase heat transfer. Hence current lumped parameter models have inherent limitations to realistically describe dispersed film boiling heat transfer. The experimental program on the Rod Bundle Heat Transfer Test Facility (RBHT) may contribute to get a better understanding of the basic reflood heat transfer mechanisms [Rose99].

6 Future work

Additional investigations are necessary to further qualify the overall reflood model of RELAP5/MOD3.3 e.g. use of additional experimental data from test programs like RBHT, PKL, and FZK-QUENCH.

7 Literature

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Appendix A Weisman Transition Boiling Correlation

Remarks	**Equations**
Heat transfer coefficient depends on distance from the quench front $h_{\mathrm{wl}}(\mathrm{Z}_{\mathrm{QF}})$	$h_{\mathrm{wl}}(\mathrm{Z}_{\mathrm{QF}}) = \begin{cases} h'_{wl}(TB) \;:\; \text{for } \mathrm{Z}_{\mathrm{QF}} \leq 0.1\mathrm{m} \\ 0.0001 : \text{ for } Z_{\mathrm{QF} \geq 0.2\,\mathrm{m}} \\ \text{interpolation} : 0.1\,\mathrm{m} < \mathrm{Z}_{\mathrm{QF}} < 0.2\,\mathrm{m} \end{cases}$
Weisman correlation $h'_{\mathrm{wl}}(TB)$ that depends on mass flux *(G),* critical heat flux (q''_{CHF}), wall temperature (T_w) and CHF-temperature (T_{wchf})	$h'_{\mathrm{wl}}(TB) = \{h_{\max} - 4500(\frac{G}{G_R})^{0.2}\} e^{-0.02\Delta T_{\mathrm{wchf}}} + 4500(\frac{G}{G_R})^{0.2} \cdot e^{-0.012\Delta T_{\mathrm{wchf}}}$.
With following definitions:	$h_{\max} = \frac{0.5 \cdot q''_{CHF}}{\Delta T_{CHF}}$, $\Delta T_{CHF} = T_w - T_{wchf}$, $\Delta T_{CHF} = T_w - T_{wchf}$, and $G_R = 67.8 \frac{kg}{m^2 s}$

Appendix B FZK-Transition Boiling Model

Comments	Equations
Total transition boiling heat flux consists of contribution from heat flux to the liquid and to the vapour phase during transition boiling	$q'' = (1 - f_l \cdot M_{\mathrm{stf}}) q''_{\mathrm{v}} \cdot M_{\alpha} + f_l \cdot q''_l \cdot M_{\mathrm{stf}}$
Transition boiling heat flux to the liquid q''_l	$q''_l = q''_l(p, \Delta T_{\mathrm{s}})$
q''_l represents an average heat flux during the short period of contact between the liquid and the superheated wall. A three-step process was postulated to describe the mechanisms of heat removal by a liquid film from the wall: 1) conduction heating of the liquid film, 2) nucleation and bubble growth within the liquid layer and 3) evaporation of a residual liquid film at the wall.	$q''_l = \dfrac{\phi_1 + \phi_{1-2} + \phi_2}{t_1 + t_{1-2} + t_2}$
Heat flux to the vapour phase	$q''_{\mathrm{v}} = h_{\mathrm{v}}(T_{\mathrm{w}} - T_{\mathrm{v}})$
Extension of CHEN-approach to take into account low pressure (function ψ) and quality (function n) reflood situations, derived base on experimental data.	$f_1 = e^{[-\sqrt{1.8} \cdot a(\alpha) \cdot f(G) \cdot \Delta T_s{}^n]} \cdot \psi(p, \Delta T_s)$ $n = 0.6 + 0.12 \cdot e^{(p/10^5)} - 0.24 \cdot x$ $\psi = 1 + 3 \cdot e^{[-0.42 \cdot p_{red}{}^{3/2} \cdot \Delta T_s^2]}$
Void fraction dependency of wetted area fraction *(f)*	$a(\alpha) = \dfrac{0.005}{(1 - \alpha^{40})} + 0.0075 \cdot \alpha$
Mass flux dependency of wetted area fraction *(f)*	$f(G) = max(f_1, f_2)$
Mass flux dependency of wetted area fraction *(f)*	$f_1 = 20 - 0.6 \dfrac{G}{135.6}$
Mass flux dependency of wetted area fraction *(f)*	$f_2 = 0.2 \dfrac{G}{135.6}$
Limitations	$x \leq 1, \quad \mathrm{f}_1 \cdot q''_l \leq q''_{CHF}$