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Investigations of the Appropriateness of RELAP5/MOD3 for the Safety Evaluation of an Innovative Reactor Operating at Thermodynamically Supercritical Conditions

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Abstract

The high performance light water reactor (HPLWR) project is being carried out within the 5. European Framework Programme (Contract N° FIKI-CT-2000-00033). The main objective of this project is to evaluate the technological merits and economics of a high efficiency light water reactor (LWR) operating at thermodynamically supercritical conditions. One of the concepts developed by the Tokyo University, the high temperature supercritical light water reactor (SCLWR-H), was chosen as the reference plant for the HPLWR-project.

A major activity of the HPLWR-project is the assessment of the appropriateness of RELAP5 - developed for light water reactors- to perform steady-state and safety analysis of the reference plant.

The investigation of such reactor concepts whose operating conditions are far beyond the operation range of current light water reactors, is very challenging for codes like RELAP5.

Since RELAP5 was not developed for supercritical water conditions and therefore not validated in this domain, the investigations to check the appropriateness of RELAP5 are focused on the following areas:

- Thermo-physical properties of water in the supercritical region.
- Heat transfer mechanisms for wall/supercritical water and corresponding correlations.
- Development of a simplified plant model for steady state and transient analysis of the reference plant.
- Exploratory analysis of selected postulated transients and accidents.

Despite the preliminary character of this work, the investigations performed for the reference design have demonstrated that RELAP5 is capable to qualitatively predict the plant behavior under both normal operation and different accidental conditions. But there are still problems in the prediction of the thermo-physical properties around the critical point in case of depressurization transients.

In this report, the results obtained in each area will be presented and discussed in detail. The additional work needed for both further code improvement and assessment of the final HPLWR-plant design is listed.

Untersuchungen zur Eignung von RELAP5/MOD3 für die Sicherheitsbewertung eines unter thermodynamisch überkritischen Bedingungen arbeitenden innovativen Reaktors

Zusammenfassung

Im High Performance Light Water Reactor (HPLWR) Projekt des 5. Europäischen Rahmenprogramms (Contract N° FIKI-CT-2000-00033) werden die technischen und ökonomischen Vorzüge des mit thermodynamisch überkritischen Bedingungen und mit hohem Wirkungsgrad arbeitenden Leichtwasserreaktors untersucht. Von den vielen Konzepten, welche die Tokio Universität entwickelt hat, wurde der überkritische Leichtwasserreaktor mit hoher Kühlmittelaustrittstemperatur (SCLWR-H) als Referenzanlage im HPLWR-Projekt ausgewählt.

Einer der Schwerpunkte innerhalb des HPLWR-Projekts ist es, die Eignung des Thermohydraulik-Programms RELAP5, welches für herkömmliche Leichtwasserreaktoren entwickelt wurde, als Analysewerkzeug zur Sicherheitsbeurteilung der Referenzanlage zu untersuchen. Der Referenzreaktor wird mit thermodynamisch überkritischem Wasser bei einem Betriebsdruck von 25 MPa gekühlt und moderiert.

Die Analyse solcher Reaktorkonzepte, deren Betriebsparameter sich sehr von denen herkömmlicher Leichtwasserreaktoren unterscheiden, stellt besondere Herausforderungen für LWR-codes dar. Da RELAP5 nicht für die Simulation von mit thermodynamisch überkritischen Bedingungen arbeitenden Reaktoranlagen entwickelt wurde, konzentrieren sich diese RELAP5-Untersuchungen auf folgende Themenkreise:

- Thermo-physikalische Eigenschaften von überkritischem Wasser.
- Wärmeübergangsmechanismen für Wand/überkritisches Wasser und deren Korrelationen.
- Entwicklung eines vereinfachten Modells der Referenzanlage für die Analyse des stationären Betriebszustandes, von Transienten und Störfällen.
- Vorbereitende Analysen ausgewählter, postulierter Transienten und Störfälle.

Trotz des vorläufigen Charakters dieser Untersuchungen konnte gezeigt werden, dass das Verhalten der Referenzanlage sowohl im Normalbetrieb als auch unter Störfallbedingungen von RELAP5 qualitativ gut beschrieben werden kann.

Dennoch traten erhebliche Probleme bei der Simulation von Transienten mit Druckentlastung wie z. B. Kühlmittelverluststörfälle auf, welche auf die Berechnung der thermo-physikalischen Eigenschaften von überkritischem Wasser in der Nähe des kritischen Punktes zurückzuführen sind.

In diesem Bericht werden die erzielten Ergebnisse vorgestellt und in Detail diskutiert. Die dabei gewonnenen Erkenntnisse werden zusammengefasst und Schlussfolgerungen für die weiterführenden Arbeiten gezogen.

TABLE OF CONTENT

LIST OF TABLES

LIST OF FIGURES

LIST OF ABBREVIATIONS

ABWR	Advanced boiling water reactor
ADS	Automatic de-pressurization system
AFS	Auxiliary feedwater system
ACC	Accumulator
ASME	American society of mechanical engineering
BOC	Beginning of cycle
3D	Three dimensional
DBA	Design Basis Accident
DOE	Department of Energy
DNB	Departure from nucleate boiling
EOC	End of cycle
ECC	Emergency core cooling
FW	Feedwater
FA	Fuel assembly
HTC	Heat transfer coefficient
HPLWR	High performance light water reactor
HT	High temperature
ISL	Information Systems Laboratory
KARBUS/KAPROS	Karlsruhe modular neutronic program system
LB	Large break
LOCA	Loss of coolant accident
LOFA	Loss of flow accident
LWR	Light water reactor
LPCI	Low pressure coolant injection
PARCS	Purdue analysis reactor core simulator
RPV	Reactor pressure vessel
SB	Small break
TCV	Turbine control valves
TSV	Turbine stop valve

LIST OF SYMBOLS

C_p	Heat capacity
D	Diameter
G	Mass flux
h	Enthalpy
L	Length
Nu	Nusselt number
Pr	Prandtl number
P	Pitch
p	Pressure
q“	Heat flux
Re	Reynolds number
T	Temperature
U	Internal energy

1 Introduction

Different advanced reactor concepts are being studied worldwide for both energy production and actinide burning. Among others, the category of reactors cooled and moderated by supercritical water seems to be very promising. The University of Tokyo proposed several concepts, including both fast and thermal reactors with hexagonal and quadratic fuel arrangements [Doba98a], [Mukjo99a]. In Canada, different concepts are also being studied e.g. CANDU-X Mark 1, CANDU-X NC and CANDUal-X [Bush00] that are based on the CANDU-technology. In the USA, a reactor for both actinide burning and energy production is being investigated [MacD01]. Common to all these concepts is the use of supercritical water as coolant (fast reactor, actinide burner) and as coolant and moderator (thermal reactor) that may lead to a substantial improvement of the thermodynamic efficiency.

Within the 5th EU-Framework Program, the HPLWR-project is being investigated by several European institutions and the University of Tokyo. The main goal of this project is the evaluation of the technical and economical merits of the “high temperature supercritical light water reactor (SCLWR-H)” -one of the several concepts proposed by the Tokyo University- taking into account both the European utility requirements (EUR) and the Generation IV (US DOE) requirements [Jevr94], [Okan94], [Doba98a], [Oka00b], and [Yama01].

The SCLWR-H concept was developed by Oka and co-workers [Doba98a], [Oka00b]. The attractiveness of this concept is its high thermal efficiency of about 44 % and system simplicity compared to current light water reactors (LWR). Since the reactor is operated at a pressure of 25 MPa, the coolant is in single phase supercritical state that permits large core outlet coolant temperature and enthalpy. Hence the coolant heat-up over the core is large i.e. over 228 K. Since the coolant inlet temperature into the down-comer is around 553 K, the RPV-walls remain at acceptable low temperatures.

A peculiarity of this concept is that use of supercritical water in two flow streams to cool the core (upwards flow) and to improve the neutron moderation (downward flow of the moderator in so-called moderator tubes), see Figure 2-8. During normal operation, coolant and moderator are in single phase supercritical state, whereby the coolant temperature along the core changes from subcritical to supercritical and the critical point is located in the lower core half. In the vicinity of the critical point several thermo-physical properties of the coolant vary sharply e.g. density, heat capacity, thermal conductivity, etc. [Okan96], [Doba97a].

Under transients conditions between two different scenarios can be distinguished. To the first one belong transients where the coolant circuit remains intact and therefore the system pressure remains above the critical pressure. On the contrary, to the second scenario belong the DBA-accidents where the integrity of the piping system fails and hence a sudden depressurization takes place. In such cases the system pressure reduces far below the critical pressure i.e. down to atmospheric pressure. The coolant thermodynamic state varies from single phase supercritical to two phase subcritical flow. These situations are challenging for system codes like RELAP5.

An additional peculiarity of the reference plant is determined by the location of the critical temperature within the core at stationary operation conditions resulting in a unique density and temperature profile of the coolant along the core height. Each perturbation of the coolant flow rate, coolant temperature, system pressure, heat source, etc. leads to a shift of the position where the coolant temperature reaches the critical one and thus to a strong perturbation of the initially established trends of major thermal hydraulics core parameters with the respective feedback to the core neutronics behavior.

Moreover, thermal hydraulic sub-channel studies, [Muko00], [Cheng01], [Berg01a] indicated, that the proposed core design of the reference plant is characterized by a pronounced non-uniform radial distribution of the pressure loss, even within a fuel assembly, resulting in non-uniform distribution of the coolant density and temperature. On the other hand neutronic design studies [Broed02] have demonstrated that the axial power profile is very sensitive to the axial coolant density and fuel temperature as well as to the stagnant water properties surrounding a water rod tube. Based on this, it must be pointed out that the core thermal hydraulics is strongly linked to the core neutronics, much stronger as in BWR cases.

Hence a more reliable simulation of the core behavior under normal and accidental conditions, requires the use of multidimensional tools for both design and safety studies.

Another safety issue is the thermal hydraulic and coupled (thermal hydraulic-neutronic) stability that may occur at nominal or partial-power operation conditions of the reference plant, [MacD01], [Ji01]. These issues are not addressed in this project but it has to be investigated in follow-up activities.

In the frame of this project, the steam tables implemented in RELAP5 were assessed to assure that the thermo-physical properties of supercritical water are predicted in a consistent manner in the whole pressure and temperature range of interest. This is a precondition for a reliable prediction of the steady state parameters and safety margins of the reference design.

Another important issue is the modeling of the wall/supercritical-water heat transfer mechanisms in tight fuel rod arrangements for both steady state and accidental conditions. Since the heat transfer correlations used in RELAP5 were developed and validated using experimental data obtained from for LWR-specific tests i.e. for pressure ranges far below the critical pressure, a re-evaluation of these models with respect to their application in design studies and safety evaluations of the reference plant is necessary.

An integral model for the reference plant was developed to demonstrate that the RELAP5 code is capable of predict the plant behavior under steady state and accidental situations in an acceptable manner, too.

To evaluate the appropriateness of the code, representative transients and accidents belonging to the two categories mentioned above were selected. The plant model was extended to consider the specific-scenario conditions and the actuation of ECC-systems within numerical model. A general description of the reference plant emphasizing its constructive and operational peculiarities is anticipated to these investigations.

2 Peculiarities of the reference plant

2.1 Thermodynamic characteristics

The reference plant is characterized by very challenging operation conditions i.e. system pressure of 25 MPa far above the critical pressure (22.07 MPa) and the coolant temperature at core inlet and outlet of 553 K and 781 K, respectively. The critical temperature (647.096 K) lies just in between. Consequently the core heat-up is very large e.g. 228 K compared to that in LWRs, see Figure 2-1.

Around the critical temperature, large changes of most thermo-physical properties of the fluid occur e.g. density (Figure 2-1), enthalpy (Figure 3-2), heat capacity (Figure 3-3), viscosity (Figure 2-2), and thermal conductivity (Figure 2-3) for the operational pressure of 25 MPa. On the other hand the thermal diffusivity is very small around the critical point and it becomes zero theoretically at the critical point [Masu02].

At the operation conditions of the reference plant, the supercritical water is in single phase. It is called pseudo-critical liquid when its temperature is below the critical one and pseudo-critical vapor when the temperature is above the critical one. Due to the high coolant outlet temperature and outlet enthalpy, a high thermal efficiency can be achieved. It amounts about 44% compared to 33 % of current LWRs. The high temperature of the coolant represents considerable thermal loads for the in-vessel core structures, for the reactor pressure vessel itself and for the steam line piping system and turbine.

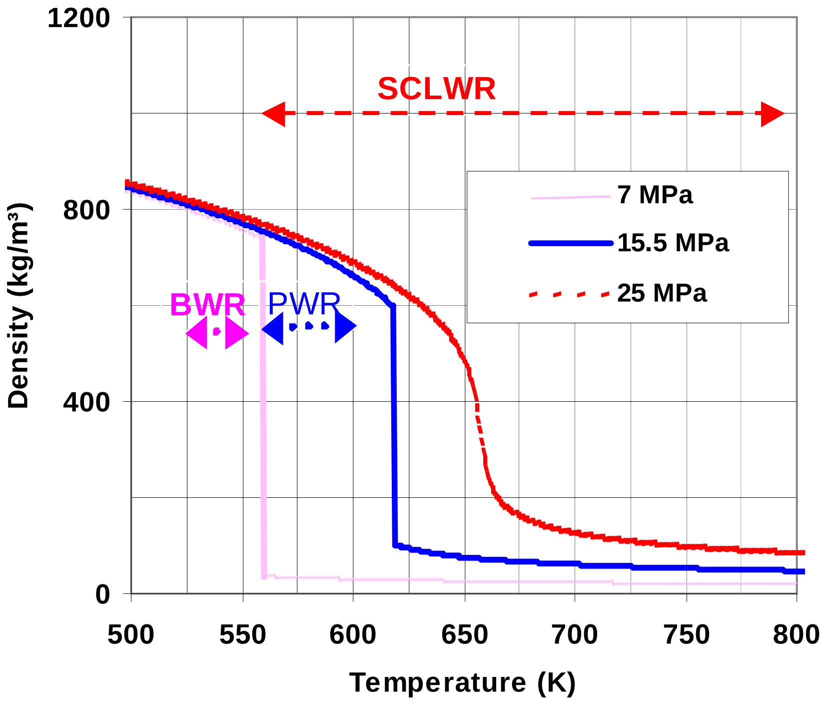

Figure 2-1 Coolant density as function of the coolant temperature

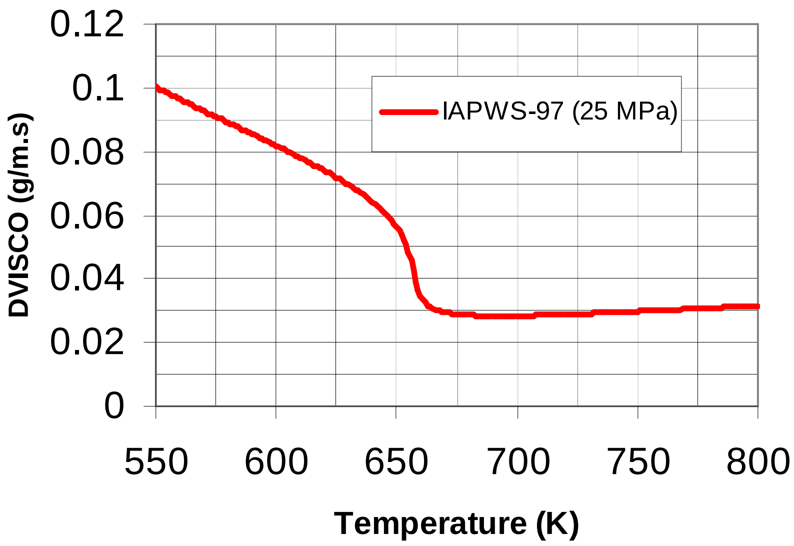

Figure 2-2 Dynamic viscosity as function of the coolant temperature for 25 MPa [Wagn97]

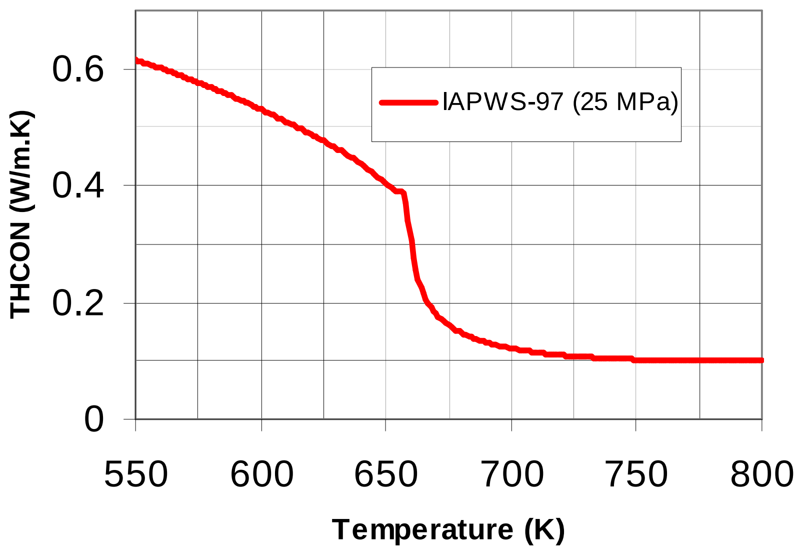

Figure 2-3 Thermal conductivity as function of the coolant temperature for 25 MPa [Wagn97]

2.2 Core design features

The core of the reference plant consists of 211 hexagonal fuel assemblies with very tight triangular pin arrangement (P/D = 9.5/8 = 1.187). In Figure 2-4 a cut through the RPV of the reference plant is given. The fuel assemblies (FA) are arranged in a hexagon surrounded by radial reflector. The inter-fuel assembly gap is 0,002 m wide. The face-to-face length of a hexagonal FA is 0.21 m. Each FA consists of 258 fuel rods with an outer diameter of 0.008 m and a length of 4.2 m. The fuel pellet is enriched UO_2 and the cladding Ni-based alloy. The pellet diameter is 0.007 m while the cladding thickness amounts 0.0004 m. In the central position, an instrumentation rod is positioned. In addition, 30 water tubes with a diameter of 0.022 m are distributed within a fuel assembly, see Figure 2-5. The water rod tubes are made of Ni-based alloy with a thickness of 0.0002 m. In nine of them, guide tubes with a diameter of 0.0117 m for the absorber rods are foreseen. Through the water rods, moderator water with a temperature equal to the coolant temperature at the core inlet will flow downwards. These so called “moderator rods” were conceived to improve the moderation of the core, since the coolant density decreases very much along the core height. Without moderator flow, a thermal reactor is not feasible from the neutronic point of view.

In Figure 2-6 one-twelve of a fuel assembly is represented with the numbering of the fuel pins. Three enrichment zones are distributed in radial direction within a fuel assembly. But also axially three enrichment zones are defined to compensate the strong coolant density reduction in the upper core part. To keep the moderator water cold enough for good neutron moderation, a complicated design of the water rod tubes was proposed [Oka00b]. The water tubes are thermally isolated by stagnant water that is contained in narrow channels surrounding the water rod. The heat transfer from the hot coolant to the moderator should thereby be minimized or even avoided so that the moderator inside the moderator tubes remains as cold as possible.

In Figure 2-7 the layout of the moderator rod is exhibited with the stagnant water and surrounding pins.

Within a fuel assembly up to 5 thermal hydraulic sub-channel types can be identified that are characterized by different hydraulic diameter, flow area, pressure drop, etc.. In such core design, an uniform lateral temperature distribution within a fuel assembly may not be possible. This is crucial for the fuel rod performance and safety margins.

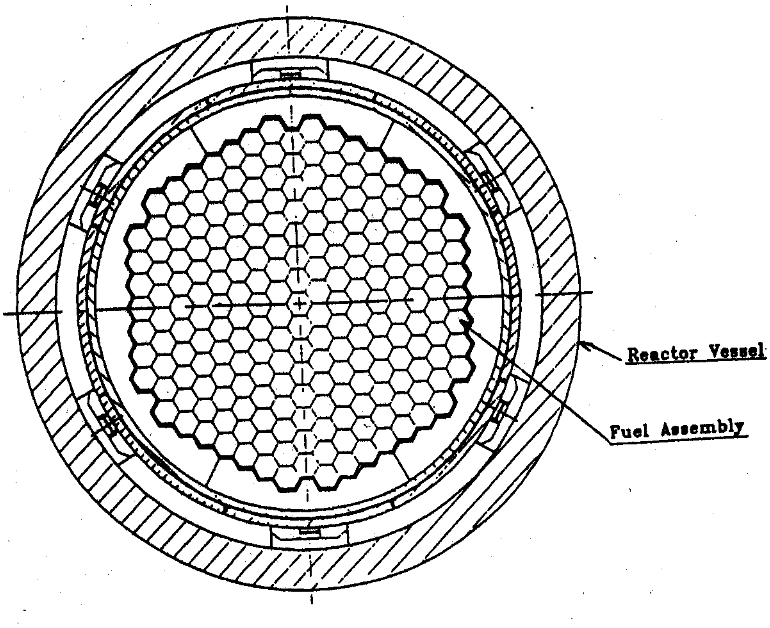

Figure 2-4 Section through the RPV of the reference plant

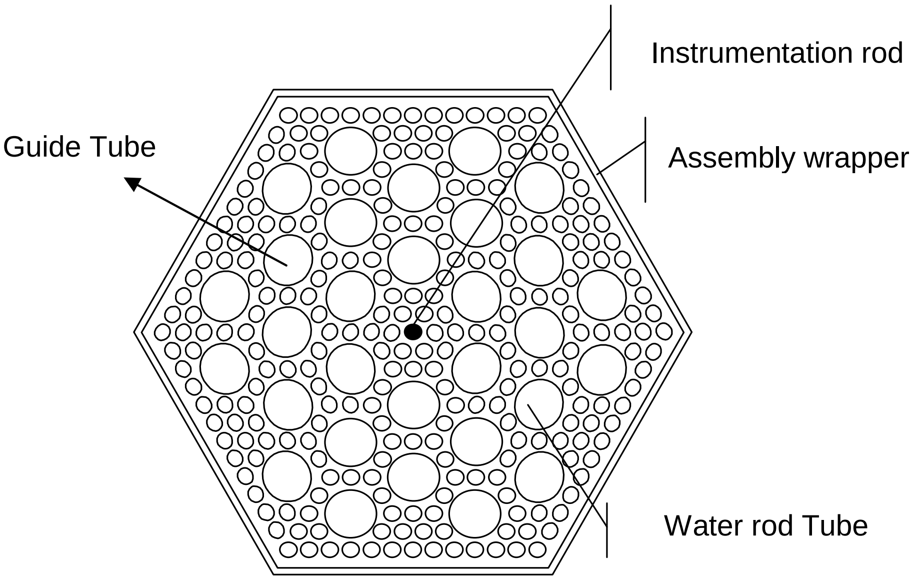

Figure 2-5 Section through the fuel assembly of the reference plant

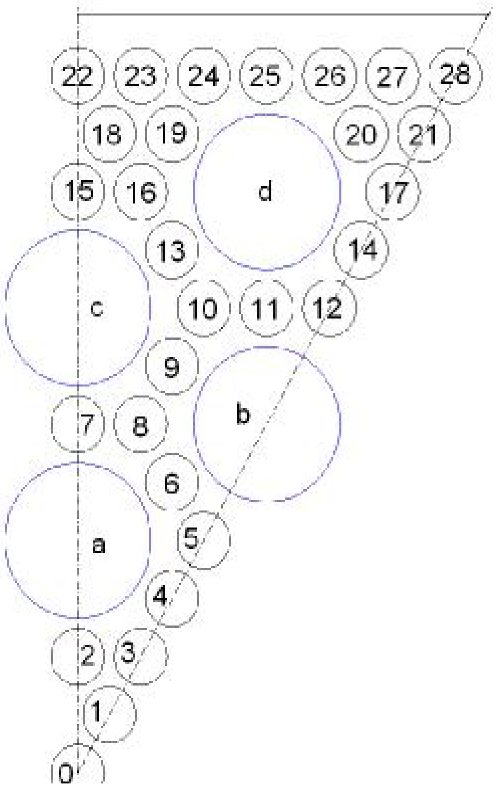

Figure 2-6 One-twelve of a fuel assembly (reference plant)

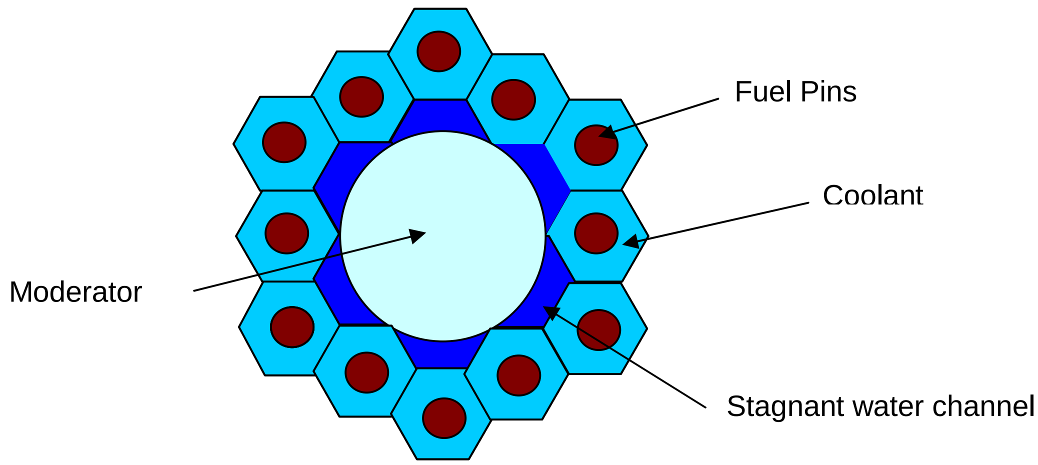

Figure 2-7 Design of a moderator rod with isolation (stagnant water) with surrounding pins.

2.3 Design criteria

In [Doba98a] the core design criteria for the reference plant are defined as follows:

- o Maximum fuel centerline temperature below 2203 K to limit fission product release. This corresponds to a maximum linear heat generation rate of 40 kW/m.

- o Maximum cladding temperature (Ni-based alloy) of 893 K in order to limit oxidation and corrosion.

- o Keep positive coolant density coefficient (negative void). This is related to the ratio of atomic hydrogen to heavy metal (H/HM) that determines the degree of moderation. For the reference design H/HM amounts 4.3 at the core inlet and 3.5 as core average.

2.4 Operational aspects

The HPLWR is operated at supercritical pressure and cooled and moderated by single phase supercritical water. The critical coolant temperature is located in the lower third of the core height. A novel feature is the counter-current flow of the coolant and the moderator that is mixed in the lower plenum before entering into the core, see Figure 2-8. Above the core, a so called hot box (region between the core and upper plenum) with a height of about 1.65 m is foreseen to hinder that the overheated coolant comes into contact the with the upper RPV-structures that may cause undesirable thermal loads.

Approximately 7 % of the total coolant flows though the water rods at BOC [Oka00b] and it changes during the cycle up to 50 % at EOC. The coolant flow rate is very low compared to that of a BWR. It amounts only 1816 kg/s. In Tab. 2-1, a comparison of the main operational parameters of different reactor designs including the supercritical reference plant is given.

The fluid inventory in the core and hot box is very low (about 5 tons) since the coolant temperature is high and hence the coolant density low, especially in the upper half of the core. Therefore the colder water in the down-comer and in the water rods represents the main heat sink in case of accidents. Consequently the RPV-fluid inventory of the HPLWR is far below that of a BWR (50 tons compared to 300 tons). This is a large disadvantage of the original OKA-concept especially in case of accidents, where large fluid inventory is needed to safely remove the decay heat over a short and long time period.

The coolant system consists of two lines, each one with a capacity of 50 % of the nominal mass flow rate. The main feed-water (MFW) pumps are driven by steam turbines. It injects coolant to the core at 25 MPa. This pressure is kept constant by the turbine control valves (TCV).

There is no re-circulation flow like in the BWR to control the reactor power, see Figure 2-10. Hence the control of the reference reactor is based on feedwater flow.

Therefore, it is required that the feed-water inflow (feedwater lines) and the steam outflow from the core may be assure under all circumstances.

Safety and auxiliary systems are foreseen to assure that the feed-water flow and the steam flow are guaranteed in any abnormal and accidental conditions.

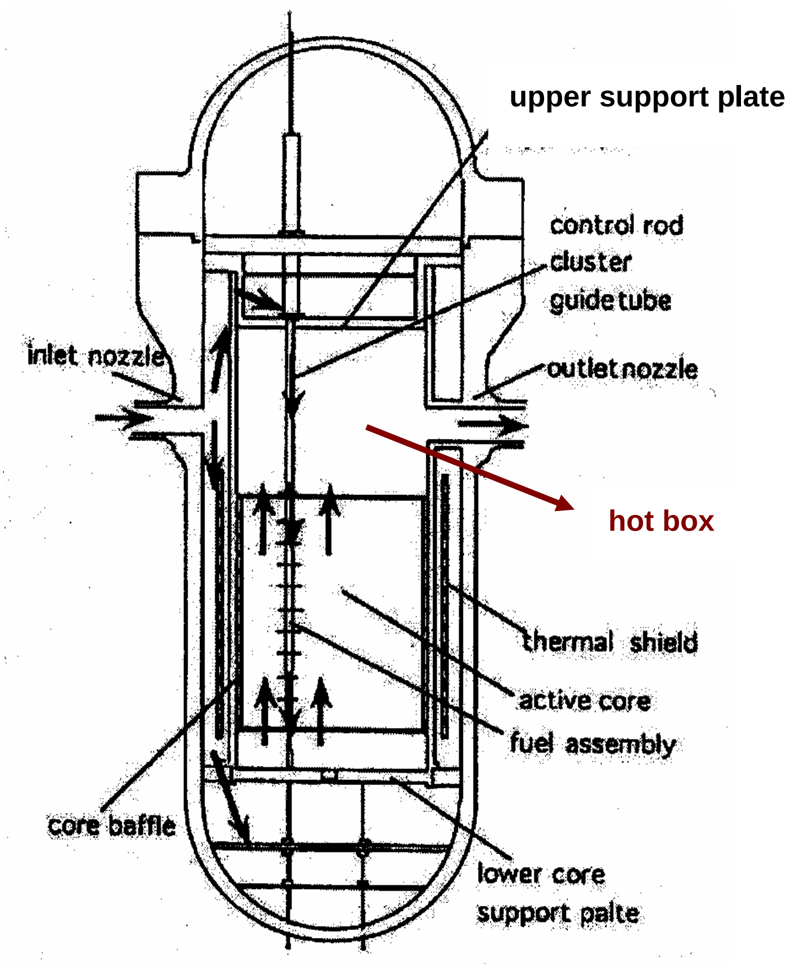

Figure 2-8 Schematic representation of the coolant and moderator flow through the core of the reference plant

Tab. 2-1 Comparison of main operational parameters of different reactor types

Parameter	**SWR-1000**	**ABWR**	**PWR**	**SC-FPP (Fossile)**	**SCLWR-H (Reference)**
Cycle type	Direct, re-circulation	Direct, recircula-tion	indirect	Direct, once-through,	Direct, once-through
Th. efficiency (%)	35.2	34.5	34.4	41.8	44
System pressure (MPa)	7	7.2	15.5	24.1	25
Coolant tempera-ture Inlet/outlet (K)	483/680	551/560	562/598	562/811	553/781
ΔT-core (K)	77	9	36	249	228
Mass flow rate (kg/s)	12000	14400	16700	821	1816
Power (MWe)	1000	1340	1150	1000	1570

2.5 Safety criteria and safety systems

2.5.1 Safety criteria

The general safety objectives for the reference plant are oriented to avoid fuel rod and/or large core damage under postulated accidents. Moreover the reestablishment of normal plant conditions under postulated transients has to be guaranteed.

The plant safety will be maintained, if the following requirements are satisfied: 1) no fuel rod damage, 2) no RPV-damage and 3) no UO_2-pellet melting. In accordance with these objectives, the following safety criteria were defined for the reference plant [Oka00b]:

- Accidents:
    - Maximum cladding temperature below 1533 K.
    - System pressure not higher than 110% of 27.5 MPa (maximal operation pressure) i.e. 30.3 MPa

- Maximum fuel enthalpy not higher than 230 cal/g (963 J/kg, applied only for reactivity abnormality)

- Transients:

    - Maximum cladding temperature not higher than 1113 K (Ni-based alloy)

    - Maximum cladding plastic deformation not higher than 1%

    - Maximum fuel enthalpy not higher than 65 cal/g (272 J/kg, applied only for reactivity abnormality)

    - System pressure not higher than 105% 27.5 MPa (maximal operation pressure) i.e. 28.9 MPa.

2.5.2 Safety systems

The safety systems of the reference plant are mainly based on active systems. In Figure 2-9 the layout of the reference plant with the safety systems is represented [Koshi94a], [Lee98a]. It consists of the automatic de-pressurization system (ADS), the low pressure coolant injection system (LPCI), the turbine-driven auxiliary feed-water system (AFS), steam dump valves, etc. The suppression pool is located within the containment vessel like in the BWR-design.

This safety philosophy is not in line with that applied for other advanced reactor like AP-600, SWR-1000, and the Generation IV reactors since mainly active systems are foreseen. Within the HPLWR-project, some options to replace the originally proposed active safety features by passive ones have been studied. But in the RELAP5-analyis the safety systems of the reference plant as proposed by Oka and co-workers will be taken into account.

A basic safety requirement for this reactor concept as for the ABWR is to maintain the core flow in all situations so that the short and long term core cooling is assured. Hence the coolant flow into the RPV (feedwater line) and out of the RPV (steam line) must be guaranteed.

Under accidental conditions, the actuation of emergency safety systems (ADS, LPCI) and auxiliary feed-water system is needed according to the abnormality level. This is determined measuring the coolant flow rate and system pressure. Consequently, the coolant in- and outflow into/from the RPV and not the core water level as is the case for BWRs must be monitored.

Depending on coolant flow rate disturbance, the following actions will be taken:

- Level 1 (feedwater mass flow rate < 90 % of nominal value): **è** reactor scram

- Level 2: (feedwater mass flow rate <20 % of nominal value) **è** activation of auxiliary feedwater system (AFS)

- Level 3: activation of automatic de-pressurization system (ADS) and LPCI.

In Tab. 2-2 the foreseen safety systems of the reference design are listed. The main characteristics and some design parameters are also mentioned there.

Tab. 2-2 ECCS of the reference reactor design[Oka00b], [Koshi94a], [Lee98a]

System	Activation	Comments
ADS ([Koshi94a], [Lee98a])	• LOCA mitigation • Feedwater line breaks with a break size below 100 %	• Eight units with a length of 13 m • ADS-valve opens when the pressure rises over 4% over nominal value • Total ADS-area for 8 units: 0.19 m² (steam line area) • ADS-valves open with a delay time of 30 s.
High pressure Turbine-driven AFS	• Mass flow rate reduces 20 % below nominal values • Trip of MFW-pumps	• Short term mitigation • Three lines with 160 kg/s/unit to mitigate LOFA and SB-LOCA • Delay time 5 s (3 s signal processing and 2 s for coolant inertia)
Motor-driven AFS		• MD-AFS actuated by emergency D/G • 16 kg/s/unit (0.8% of nominal G) • Delay time 30 s
Low pressure LPCI ([Koshi94a][Lee98a])	• LOCA mitigation • Mass flow rate decreases 10 % below nominal value • When P < 1 MPa	• Four units with a capacity of 805 kg/s per unit. • Two trains connected with the feedwater lines and • Two trains with the down-comer. • Only 2 trains available for injection • Typical delay time for injection is 30 s.
Containment isolation (CI)	• Steam line or feedwater break events	• Isolation of intact feed-water and steam lines immediately after break opening. It is assumed that CI starts 1.8 s after break and ends at 3 s (BWR data).
Turbine control valves (TCV)	• Maintain system pressure	• Pressure changes below 0.8 %
Turbine bypass valves (TBV)	• Rapid closure of turbine control valves (TCV)	• Actuates between 0.8 and 4 % pressure changes • Delay time for signal processing: 0.1 s
Safety relief valves (SRV)	• Open/close: 26.5/25.5 MPa (2) • Open/close: 26.7/25.7 MPa (3) • Open/close: 26.9/25.9 MPa (10) • Open/close: 27.1/26.1 MPa (10)	• Actuates when pressure changes over 4% (1MPa) • It consists of four banks

The coolant circuit consists of two feedwater lines of an inner diameter of 34 cm and two steam lines with an inner diameter of 54 cm.

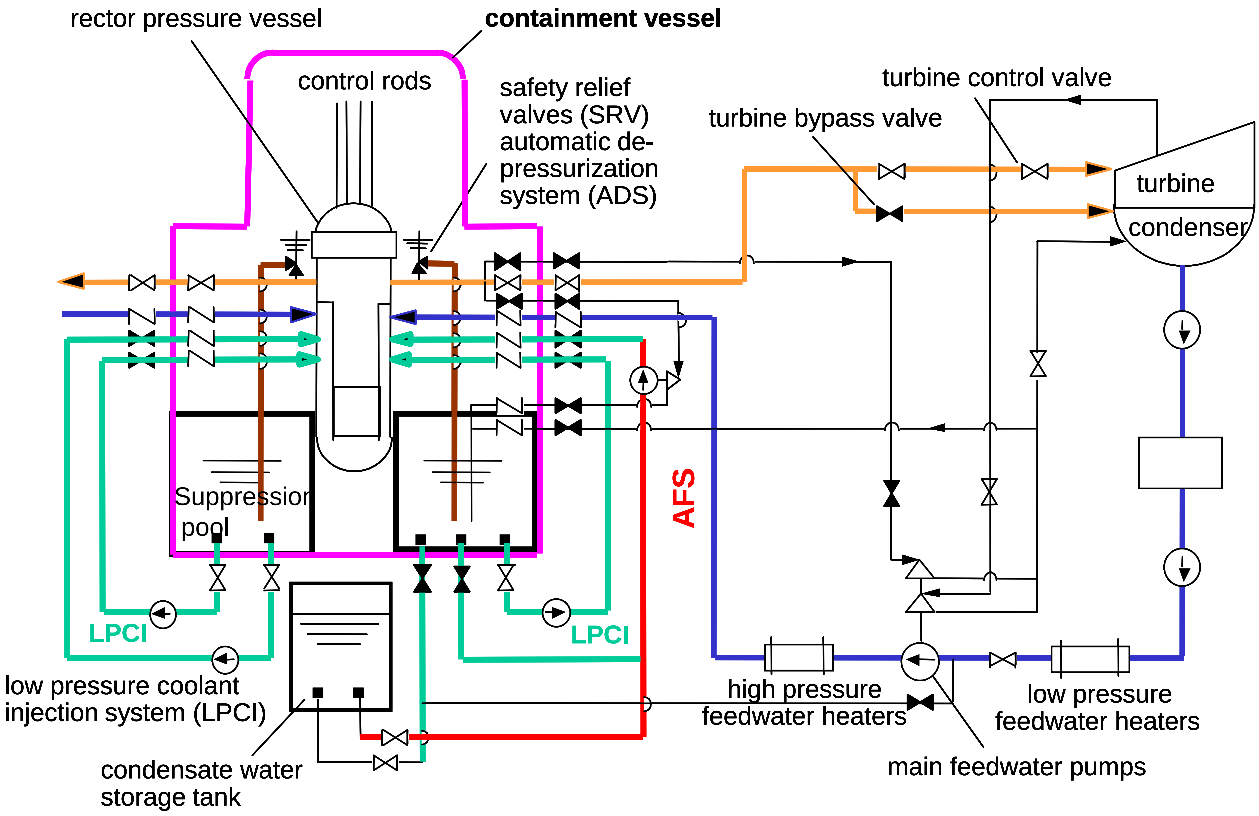

Figure 2-9 Schematic representation of the safety features of the reference plant

The containment of the reference plant is compared to that of an ABWR in Figure 2-10. It can be seen that the height of the reference plant is much smaller than that of the ABWR. Since no re-circulation pumps are needed, and the control rods can be inserted from the top, a small containment seems to be feasible for the reference plant.

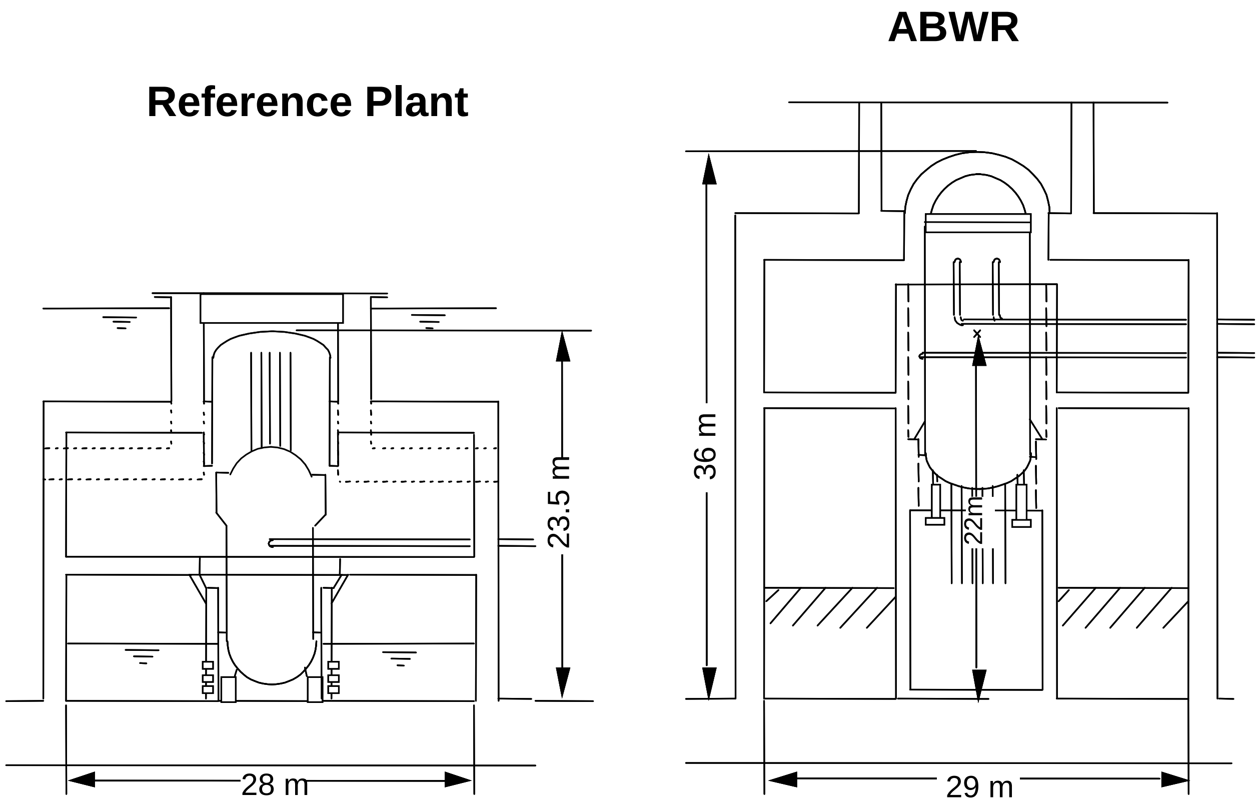

Figure 2-10 Comparison of the containment layout of the reference plant versus ABWR

3 Review of RELAP5

The RELAP5 thermal hydraulics code system is a six equations (mass, momentum, energy of liquid and vapor phase) two-phase flow code that allows the simulation of a wide range of operational transients and accidents for light water reactors [R5M32].

The analysis of the reference plant that operates at thermodynamically supercritical conditions represents a challenge for the so called best-estimate thermal hydraulic codes like TRAC [Liles84], CATHARE [Barre90], RELAP5 [R5M32], and ATHLET [Burw89].

Several RELAP5-versions are installed at FZK/IRS (see Tab. 3-1), that were distributed within the US NRC CAMP-program, on which FZK is participating.

Tab. 3-1 Available RELAP5-versions with different steam tables

Code version	**Institution**	**Steam Tables**	**Date**
RELAP5/MOD3.1	INEEL/USNRC	ASME IFC-67	1994
RELAP5/MOD3.2	INEEL/USNRC	ASME IFC-67	1995
RELAP5/MOD3.2.1.2	INEEL/USNRC	ASME IFC-67	1996
RELAP5/MOD3.2.2 β	ISL/USNRC	ASME IFC-67	1998
RELAP5/MOD3.2.2φ	ISL/USNRC	ASME IFC-67	1999
RELAP5/MOD3.3 β	ISL/USNRC	IAPWS-95 (default) (ASME-IFC-67 as option)	2001
RELAP5/MOD3.3	ISL/USNRC	IAPWS-95 (default) (ASME-IFC-67 as option)	2002

RELAP5 is coupled with the Purdue Advanced Reactor Core Simulator (PARCS) via message-passing protocols in the Parallel Virtual Machine (PVM) package, [Barb98]. The PARCS code is a multidimensional system to predict the global and local neutronic response of LWRs in steady state and transient conditions by solving the multigroup time-dependent neutron diffusion equation [Joo98]. Within RELAP5, a point kinetics model is also available.

3.1 RELAP5-steam tables

The review of the validity range of steam tables implemented in thermal-hydraulic design and safety analysis codes, like COBRA and RELAP5 is very important within this project.

In RELAP5 both the ASME-IF-67 steam table [ASM67] and the scientific formulation of the IAPWS-97, called IAPWS-95 [Wagn97], steam table are implemented.

Selected thermo-physical properties of water of the ASME-IF-67 and IAPWS-97 steam tables are compared in Figure 3-1, Figure 3-2, and Figure 3-3 for the range of pressure and temperature of interest. The system pressure and the coolant temperature at core inlet and outlet are covered in the plots. From these evaluations can be concluded that both steam tables are capable to predict most relevant thermo-physical properties of supercritical water that are needed in simulation tools like RELAP5. A numerical problem may be the steep change of e.g. the heat capacity around the critical point in some accidental situations.

It can be stated that both steam tables implemented in RELAP5 gives similar results in the supercritical region of interest, except for the heat capacity. Here discrepancies are encountered around the critical point, i.e. between 650 and 675 K.

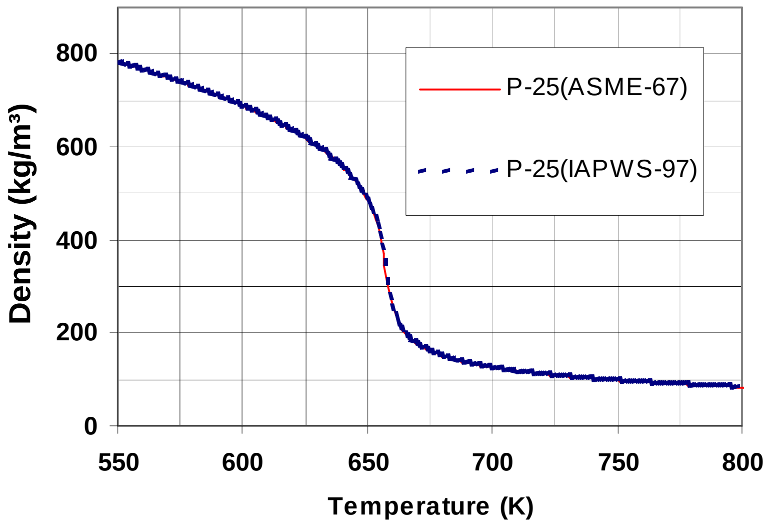

Figure 3-1 Comparison of the density predicted by steam tables ASME-67 and IAPWS-97

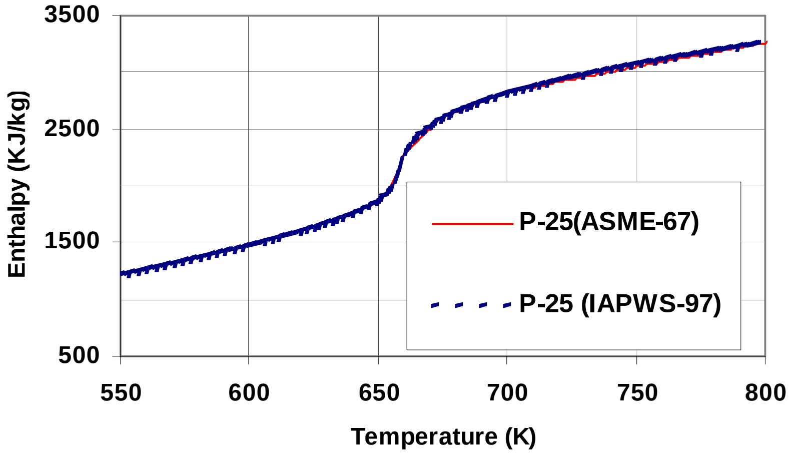

Figure 3-2 Comparison of the enthalpy predicted by steam tables ASME-67 and IAPWS-97

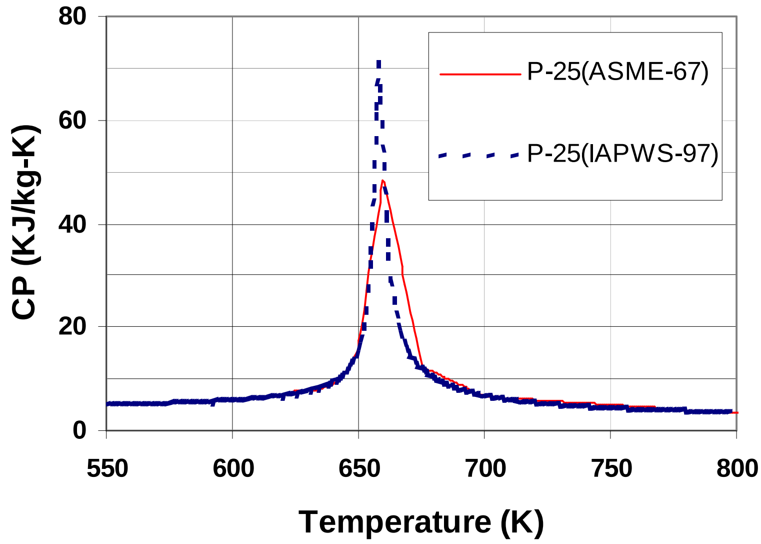

Figure 3-3 Comparison of the heat capacity as predicted by two different steam tables

3.2 Wall-supercritical water heat transfer

In RELAP5 the well-known Dittus-Boelter correlation is used to describe the wall/single-phase (liquid/vapor) heat transfer. It is given by the relation:

$$Nu = 0.023 \cdot \mathrm{Re}^{0.8} \cdot \mathrm{Pr}^{0.33} \tag{3.1}$$

Where Re is the Reynolds and Pr the Prandlt number.

The literature review performed in [Cheng01] has shown that several wall/supercritical water heat transfer correlations have been developed based on experimental investigations and numerical evaluations. The importance of the available correlations and data regarding their applicability in design and safety studies for the reference plant were also discussed. It was found out that the Bishop-correlation is the most appropriate one for the condition of the reference plant (sub-channel geometry, hydraulic diameter, working fluid). This correlation is defined as follows:

$$Nu = 0.069 \cdot \mathrm{Re}_B^{0.90} \cdot \widetilde{\mathrm{Pr}}^{0.66} \cdot \left(\frac{\rho_W}{\rho_B}\right)^{0.43} \cdot \left(1 + 2.4\frac{D}{L}\right) \tag{3.2}$$

With:

$$\widetilde{\mathrm{Pr}}_B = \left(\widetilde{C}_p \cdot {}^{\mu_B}\!/_{\lambda_B}\right) \tag{3.3}$$

and

$$\widetilde{C}_p = \frac{h_w - h_b}{T_w - T_b}. \tag{3.4}$$

The parameter range for this correlation is given in Tab. 3-2.

Tab. 3-2 Parameter range of the Bishop correlation

Parameter	**Range of parameter**	**Reference plant parameter**
Pressure (MPa)	22.6 - 27.5	25
Mass flux (Mg/m²s)	0.68 – 3.6	1.038 (a=1.74m²)
Heat flux (MW/m²)	0.31 - 3.5	-
Hydraulic diameter (mm)	2.5 - 5.1	4.1689 (average)
Length-to-diameter ratio (L/D)	30 – 565	525 (=4.2/0.008)
Bulk temperature (K)	567 – 798	553 – 781
Wall superheat (K)	16 – 216	-

In Figure 3-4 the wall temperature predicted by several correlations for a typical sub-channel type of the reference plant design is given. It can be seen that the Dittus-Boelter correlation overestimates the heat transfer compared to the Bishop one for coolant temperatures above the pseudo critical temperature, i.e. 657 K for 25 MPa.

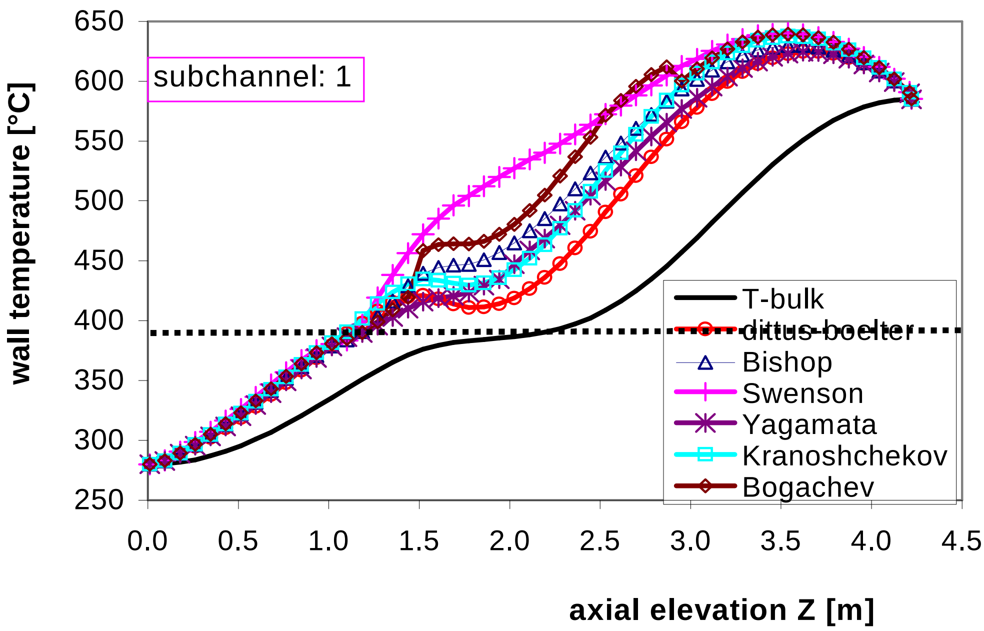

Figure 3-4 Comparison of the wall temperature versus the core height predicted by different correlations

Another peculiar heat transfer mechanisms of the reference plant is the so called heat flux deterioration, [Okan96], [Doba98a], etc. It is characterized by a strong reduction of the heat transfer coefficient under high heat flux and low mass flux conditions. The resulting wall temperature increase is however milder than under DNB-conditions.

Since the heat flux deterioration is smooth, it is not so easy to define a point for the onset of heat flux deterioration. According to Yamagata [Yama72] the onset of heat flux deterioration can be described by the following correlation that developed based on experimental data:

$$q'' = 200 \cdot G^{1.2} \tag{3.5}$$

The mass flux G is used here as a dimensionless number.

For the exploring studies within the HPLWR-project, it was agreed to use the Dittus-Boelter correlation. In a next stage improved correlations, probably the Bishof correlation, have to be implemented in RELAP5. Furthermore, additional experimental investigations of the heat transfer mechanisms for supercritical water are absolutely necessary to confirm and further validate the available models.

4 Development of a RELAP5-model for the reference plant

A simplified plant model for RELAP5 was developed to check the code's appropriateness to simulate the steady-state and transient conditions of the reference reactor (SCLWR-H). This model is based on the data given in [Doba98a]. In the first step, the main thermal-hydraulic core parameters for the nominal reactor operation conditions will be calculated. Then selected transient and accidental situations will be investigated.

4.1 The simplified steady-state plant model

The accurate prediction of steady-state conditions is prerequisite for subsequent analysis of transients and postulated accidents using RELAP5. For this purpose, a simplified plant model for the reference reactor including major in-vessel components, the most significant parts of the primary coolant system such as feed-water and steam system lines, was elaborated. In addition, a point kinetics model was developed to account for feedback mechanisms in case of transient and accidental situations. In Figure 4-1 the simplified plant model is shown. The down-comer, lower plenum, core, and hot box are modeled by hydrodynamic volumes considering the reactor geometry and flow areas, hydraulic diameter, etc.. These volumes represent the main coolant path which enters into the down-comer and leaves the RPV via steam lines to the turbine. The second fluid stream, called moderator flow, consists of the upper head, water tubes that connect the upper plenum with the lower plenum, where the moderator and the coolant get mixed before entering the core.

The core region is represented by an average coolant channel with the total core flow area. It consists of 20 axial nodes of 0.21 m elevation. An additional node is considered below the core representing the grid plate holes.

The fuel rods are grouped into one representative heat structure with 20 axial nodes, which correspond to the core flow channel nodes. In radial direction, three material zones i.e. fuel, gap, and clad are considered, each one subdivided in 6 (fuel), 1(gap), and 4 (cladding) nodes. A cosine-shaped axial power profile is assumed according to [Doba98a] for the simulation.

The water rod tubes are modeled by an average channel with 23 axial elevations i.e. one below the core, 20 in the core region, and 2 in the hotbox region (above the core). An additional heat structure was also added to represent the water tubes. Since the water rods in the reference design are considered to be completely isolated, adiabatic boundary conditions were applied for the water tubes heat structure i.e. no heat transfer from the hot coolant to the water tubes is allowed.

The coolant and moderator are injected into the down-comer and upper plenum by two independent junctions. The system pressure is fixed by a boundary condition in the turbine volume. The fraction of moderator flow rate to total primary circuit flow rate amounts around 7 % (130 kg/s) for the BOC-conditions of the reference plant.

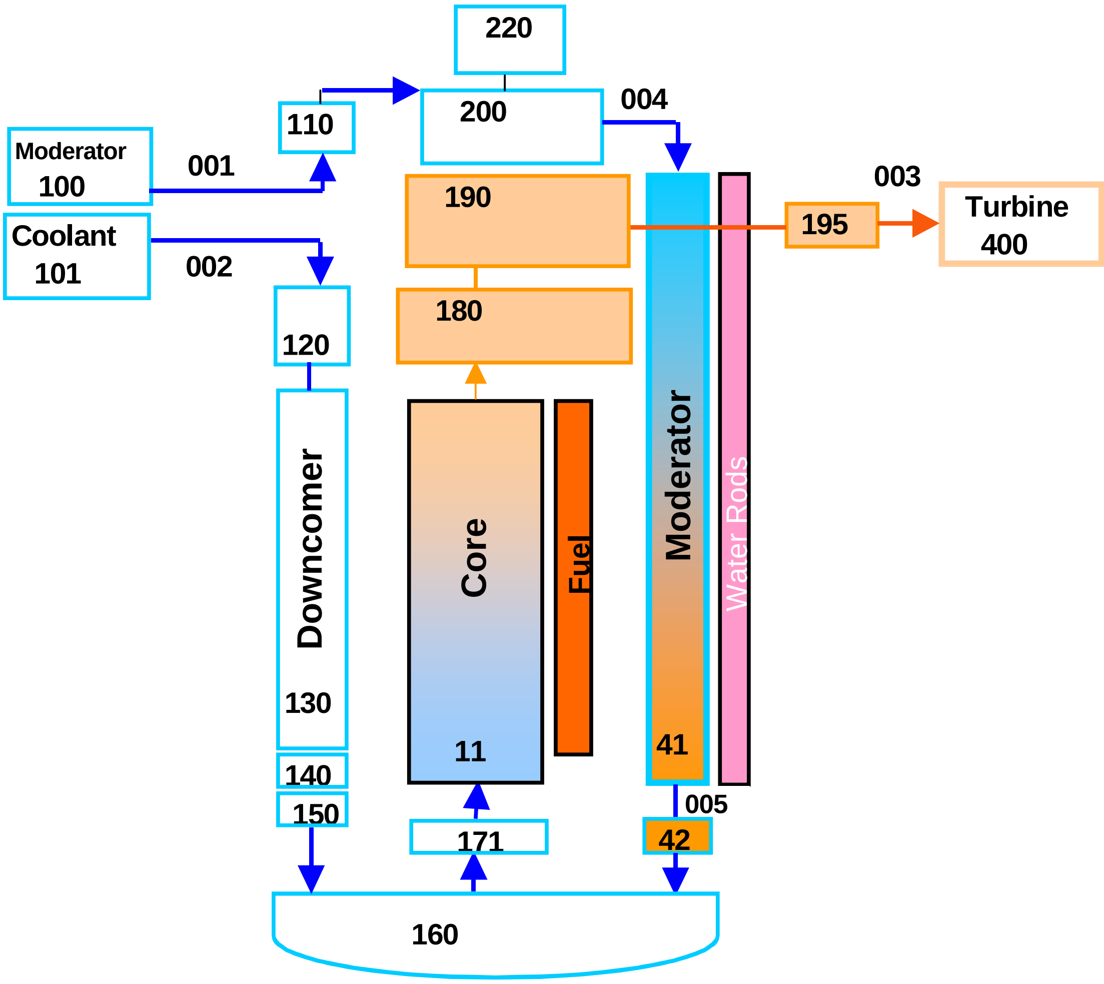

Figure 4-1 Simplified model of the reference plant for the RELAP5-simulation

4.2 Predicted steady state parameters

The investigations started with the code version RELAP5/MOD3.2.2φ. But this code version failed to reach steady state conditions. It stopped due to negative sound speed and very small time steps after few seconds. One of the reasons was the very coarse grid points for the coolant temperature and system pressure used to generate the look-up table for the steam properties within RELAP5, especially in the supercritical region and near the critical point.

In addition, large mass errors were predicted by this version of RELAP5 due to wrong use of the drift-flux model in the supercritical region, when the coolant temperature becomes greater than the critical one. In this case, the void fraction switches from 0 to 1 per definition.

After the refinement of the temperature/pressure grid points, reasonable steady-state parameters were predicted by modified version of RELAP5 without code failure.

Moreover the mass error of about 18 % in the original version was reduced to 1% by modifying the subroutine “statep”, where the actual void fraction ***voidg(i)*** was set equal to the void fraction of the former time step ***voidgo(i)*** *instead to the actual quality* ***qual(ix)*** [Mort02a]**.**

These modifications in RELAP5/MOD3.2.2φ were implemented in the following RELAP5-versions released by ISL/USNRC.

A recalculation with RELAP5/MOD3.3β led to results similar to those of RELAP5/MOD3.2.2φ. All follow-up investigations are performed with RELAP5/MOD3.3β. Selected parameters for the reference plant, predicted by RELAP5, are compared in Tab. 4-1 with the data of the reference design. It can be seen, that the RELAP5-predictios are in good agreement with the values given in [Oka00b].

Tab. 4-1 Predicted steady-state parameters compared to reference design

Parameters	**Reference Design [Oka00b]**	**RELAP5/MOD3.3**b
Core inlet temperature (K)	553	553.14
Core outlet temperature (K)	781	782.6
Core pressure drop (MPa)	--	0.175
Core mass flow rate (kg/s)	1816	1816
Moderator mass flow rate (kg/s)	130	130
Power (MW)	3568	3568

Moreover the axial distribution of important core parameters as predicted by RELAP5 are unique and novel compared to those of current LWRs. In Figure 4-2 the density of the coolant and of the moderator along the core and the hot box together with the volume-weighted average density are given. Note that due to the axial trend of the core-averaged density, the moderation along the core height is not uniform.

In Figure 4-3 the temperature of the coolant and of the moderator along the core and hot box is presented, too. It can be seen in this calculation that the pseudo-critical temperature is located around the 1.8 m elevation. In the hot box region the coolant temperature is almost constant and high enough to enable large efficiency of the reference pant.

The axial temperature distribution of the pellet center, the pellet and cladding surface, and the coolant is given in Figure 4-4. The pellet surface and cladding temperature follow the bulk temperature trend. The predicted heat transfer coefficient along the core height is shown in Figure 4-5. The maximum value is located around the core middle.

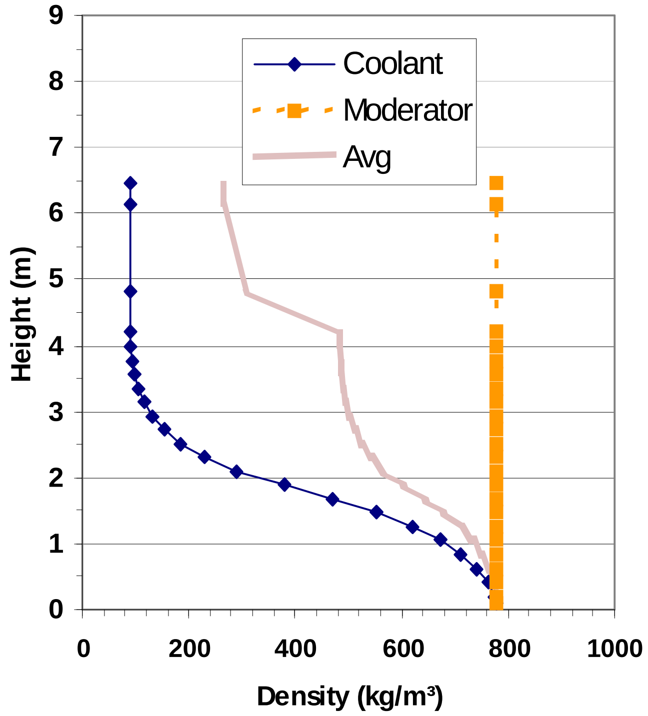

Figure 4-2 Density distribution along the core and hot box

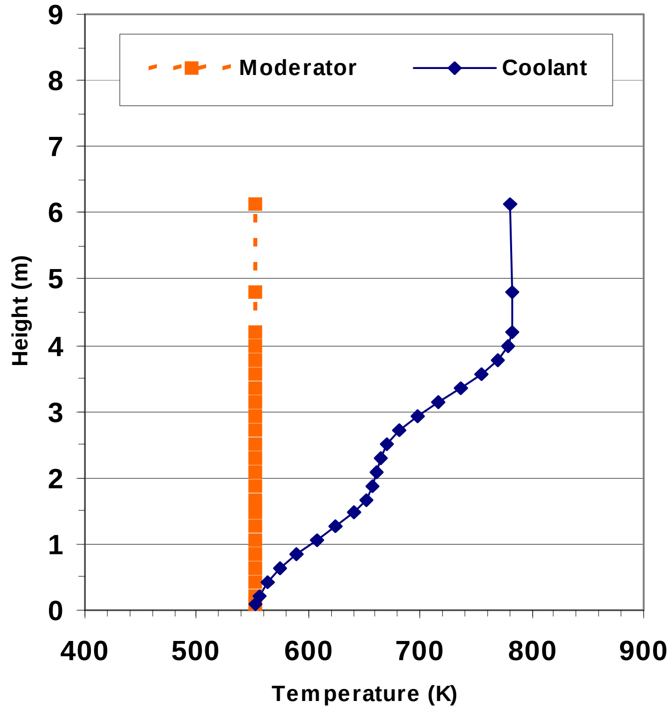

Figure 4-3 Temperature distribution along the core and hotbox

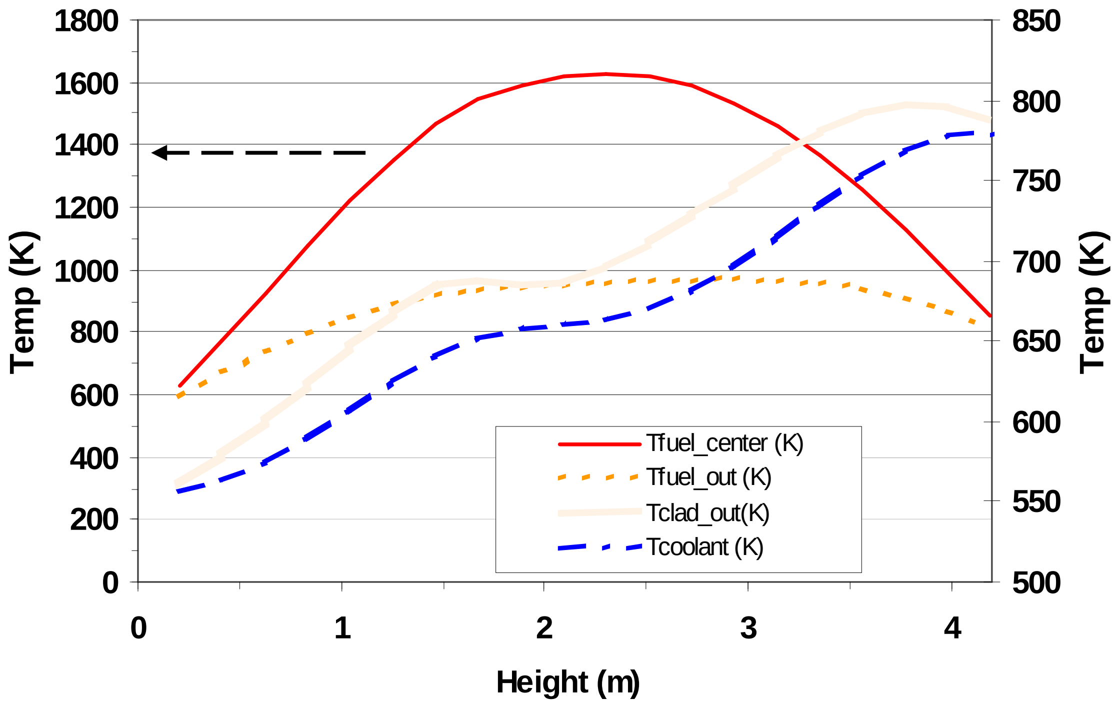

Figure 4-4 Axial distribution of the fuel, cladding and coolant temperature in the core

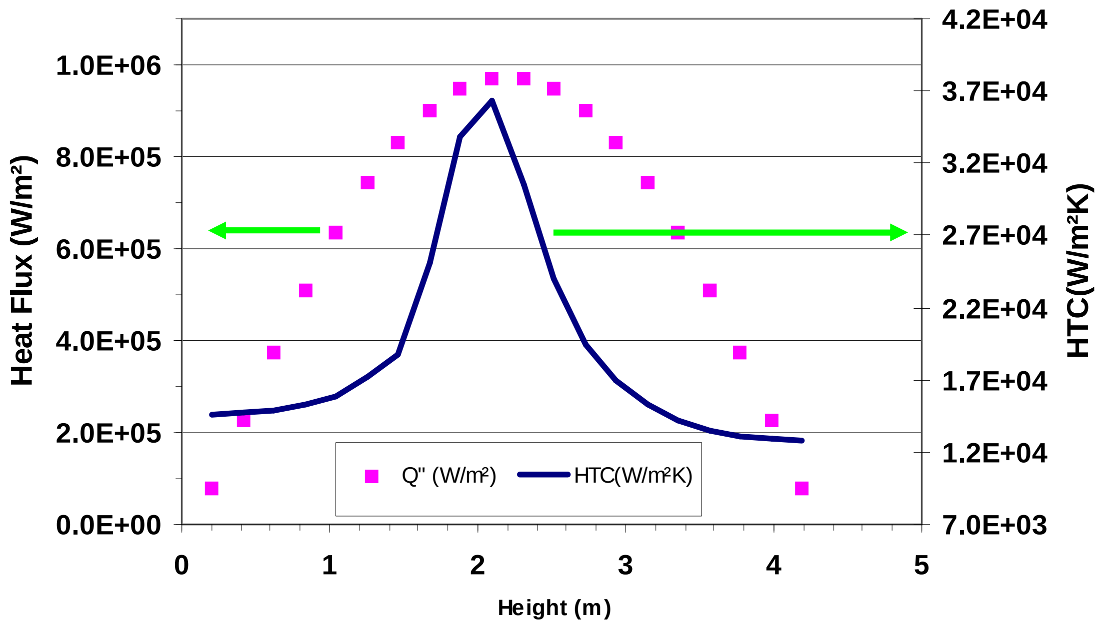

Figure 4-5 Axial heat flux and heat transfer coefficient distribution in the core

The water inventory within the RPV for the reference plant at nominal operation conditions is distributed as indicated in Tab. 4-2. It is very low compared to the 300-400 tons of current LWRs, due to low density and the smaller dimensions.

Tab. 4-2 Fluid inventory distribution within the RPV

RPV-part	Fluid inventory (Tons)
Down-comer	19.9
Lower plenum	19.2
Core	2.9
Hot Box	1.43
Total fluid inventory within the RPV	43.4

4.3 Analysis of different moderator concepts

4.3.1 Investigated alternative moderator concepts

In the reference design [Doba98a] a core with descending moderator flow through 6330 water rods is proposed. The water rod tubes are thermally isolated by a complicated construction containing stagnant water so that the moderator density remains high (Reference case) during the reactor operation. Hence, in [Meet27401] it was recommended to investigate alternative moderator concepts aiming to achieve similar moderation characteristics as the reference design. In Tab. 4-3 some moderator concept proposals are listed. RELAP5-investigated were performed to find out how these concepts may affect the core-average density of supercritical water (coolant and moderator).

Tab. 4-3 Alternative moderator concepts

Cases	Description
Reference case	Ideal thermal isolation of water rod tubes
Case 1	Water rod tubes with heat conducting walls, no stagnant water considered
Case 2	Replacement of 70 % (4431) of the water rods by solid moderator
Case 3	Isolation of remaining water rod tubes (1899) in the hot box region

4.3.2 RELAP5-Predictions

The original plant model was slightly modified to take into account the different moderator concepts. In all calculations the moderator mass flow rate amounted 7% of the nominal coolant mass flow rate corresponding to the BOC-conditions [Koshi00a], i.e. a moderator flow of 130 kg/s.

The volume-weighted average core density (coolant and moderator) is used to compare the cases. It is calculated by the following relation:

$$\rho_{avg} = \frac{V_{mod}}{V_{tot}} \cdot \rho_{mod} + \frac{V_{cool}}{V_{tot}} \cdot \rho_{cool} \tag{4.1}$$

Where: V_{tot} is given by

$$V_{tot} = V_{mod} + V_{cool} \ . \tag{4.2}$$

In Figure 4-6 the predicted density and temperature trends of the coolant, moderator and cladding along the core height are represented. If heat transfer from the coolant to the water rod tubes is considered, a considerable heat-up of the moderator water is calculated, as can be seen in Figure 4-7.

A comparison of the predicted core-averaged density of moderator and coolant is shown in Figure 4-8 for all cases. Based on these results, the following conclusion can be drawn:

- The reference case is the one with the best moderator conditions but it is the most complicated one.

- The removal of the complicated isolation (Case 1) leads to a considerable heat-up of the moderator and thereby to a reduction of the core average density along the core height. It has to be checked by neutronic calculations whether this low core average density is sufficient for a thermal reactor design.

- The use of solid moderator (ZrH-slabs) instead of 70% of the water rod tubes contributes to a remarkable increase of the core average (moderator+coolant) density (Case 2) compared to case 1. But the axial density distribution is still below that of the reference case.

- An additional improvement of the core moderation can be reached when the water rod tubes of case 2 are thermally isolated in the hot box region above the core (Case 3).

There are many possibilities to improve the core design from constructive, mechanical, neutronics, thermal-hydraulics, and last but not least from the safety point of view. Extensive optimization studies will be necessary for the final core design.

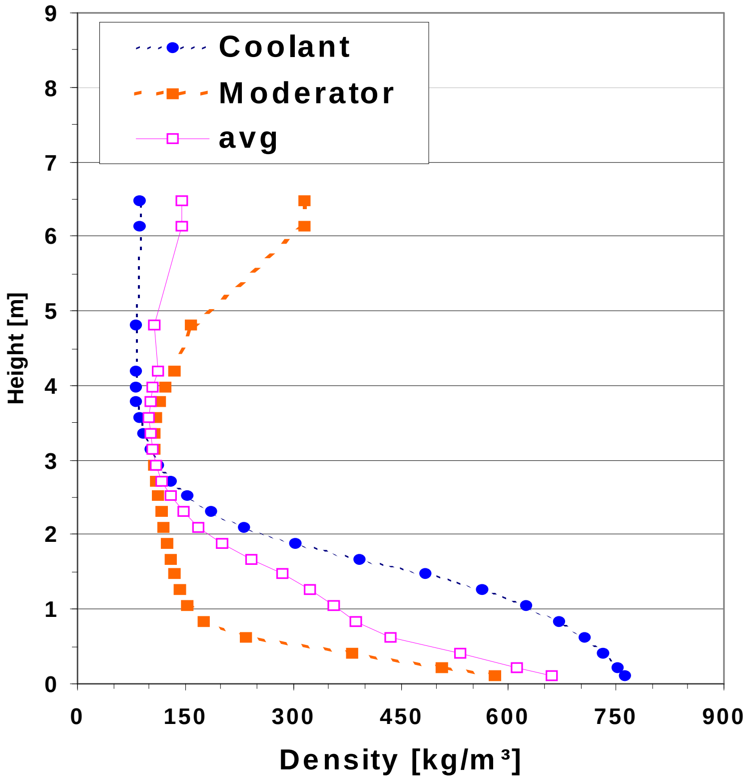

Figure 4-6 Axial distribution of the coolant and moderator density (Case 1)

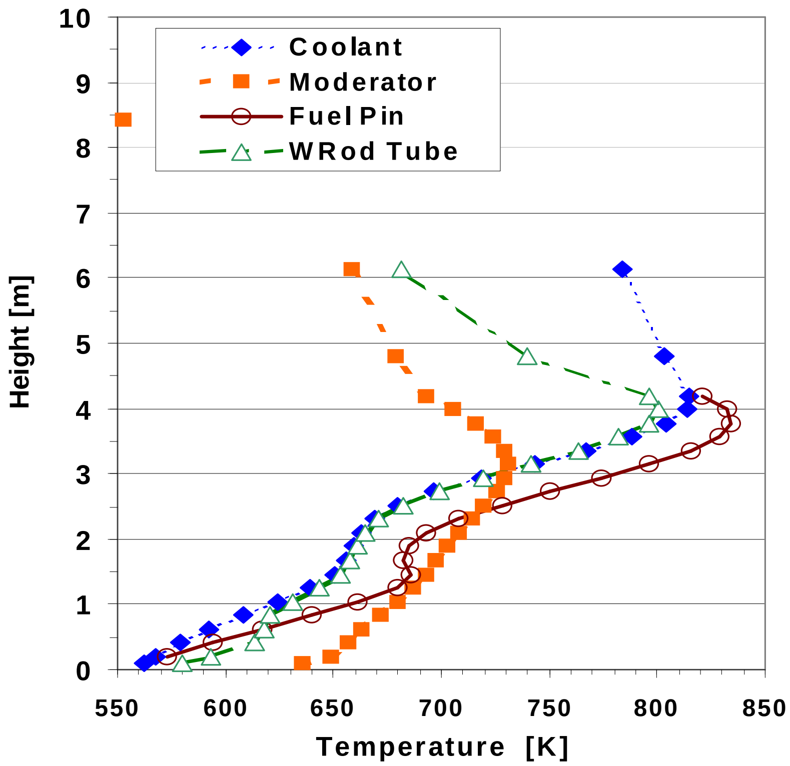

Figure 4-7 Axial temperature of coolant, moderator, fuel pin, and WR-tube (Case 1)

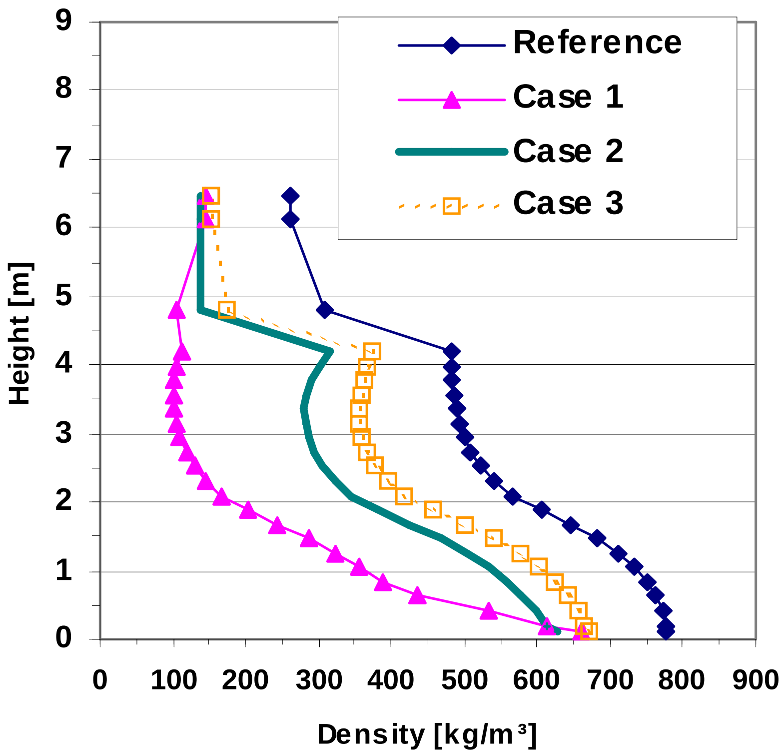

Figure 4-8 Comparison of the averaged core density for cases 1 to 3

Furthermore, it was found out that the moderator flow rate has an important impact on the moderator density if conducting walls are assumed. Low mass flow rate through the water rods means increased heat-up and hence reduction of the moderator density. In Figure 4-9 the predicted moderator density for two mass flow rates is given. It can be seen that a reduction of the flow rate from 130 to *30 kg/s* leads to a non-negligible density decrease.

The optimal mass flow rate of the moderator depends not only on thermal considerations but also on reactor control philosophy during the fuel cycle. For the reference design, a variation of the ratio of moderator to total core flow rate from 7 % for BOC up to about 50 % for EOC conditions is foreseen to compensate e.g. reactivity changes due to burn-up.

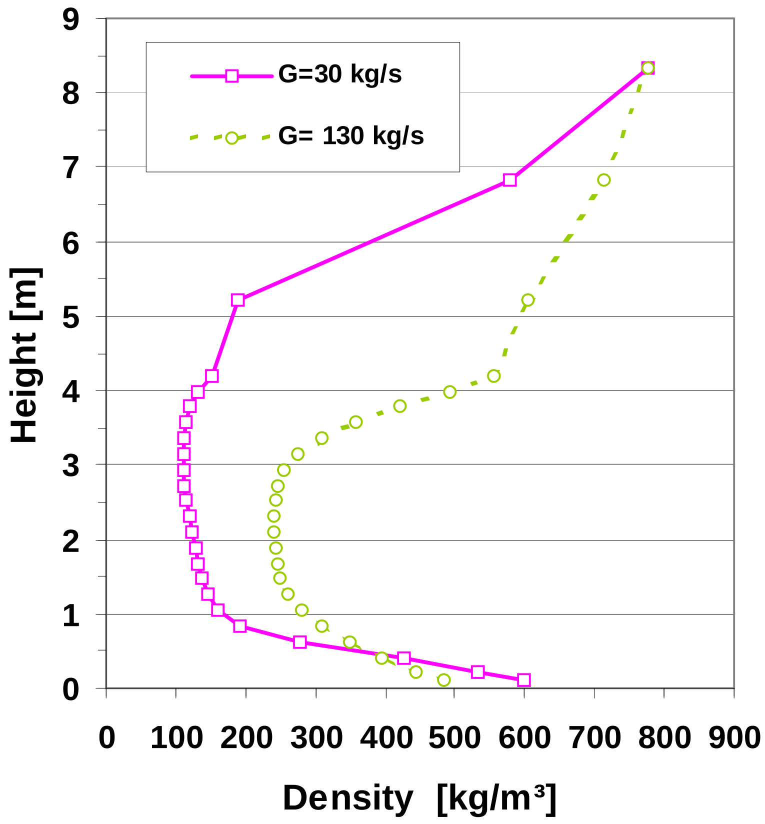

Figure 4-9 Comparison of the moderator density in water rod for different flow rates

5 Effects of axial power on thermal parameters

5.1 RELAP5/KARBUS prediction of axial power profile

In the reference design (SCLWR-H) a cosine-shaped axial power profile [Doba98a] was assumed. Neutronic investigations performed in the frame of the HPLWR-project have shown that the core neutronics and the thermal hydraulics are strongly linked [Broed02].

Since both the core design and the coolant/moderator conditions of the reference design are not similar to that of current LWRs, the validation of the deterministic neutronic tools is mandatory. In this connection Monte Carlo simulations are of fundamental importance since no experimental data is available for the HPLWR. Hence at FZK the neutronic design system KARBUS/KAPROS [Broed01] was coupled with the best-estimate thermal hydraulic code RELAP5 to analyze the reference reactor. The KARBUS/RELAP5 solutions were benchmarked with the Monte Carlo calculations performed also at FZK for the reference core to assure that the deterministic results are reliable [Broed02].

The coupled system KARBUS/RELAP5 was used iteratively to determine the axial power profile taking into account the actual axial thermal hydraulic conditions (coolant and moderator density, fuel temperature) for the cross sections.

In Figure 5-1 a comparison of the predicted cosine-shaped with axial power profile is given. For the BOC-conditions of the reference design, the predicted power distribution differs very much from the cosine assumption. A maximum is predicted in the lower core part while in the upper part the values are relative small. This curve follows mainly the axial coolant density trend.

The investigations have also shown that the power profile is very sensitive not only to the number of energy groups but also to the thermal hydraulics conditions i.e. the coolant and moderator density, the fuel temperature, etc., [Broed02].

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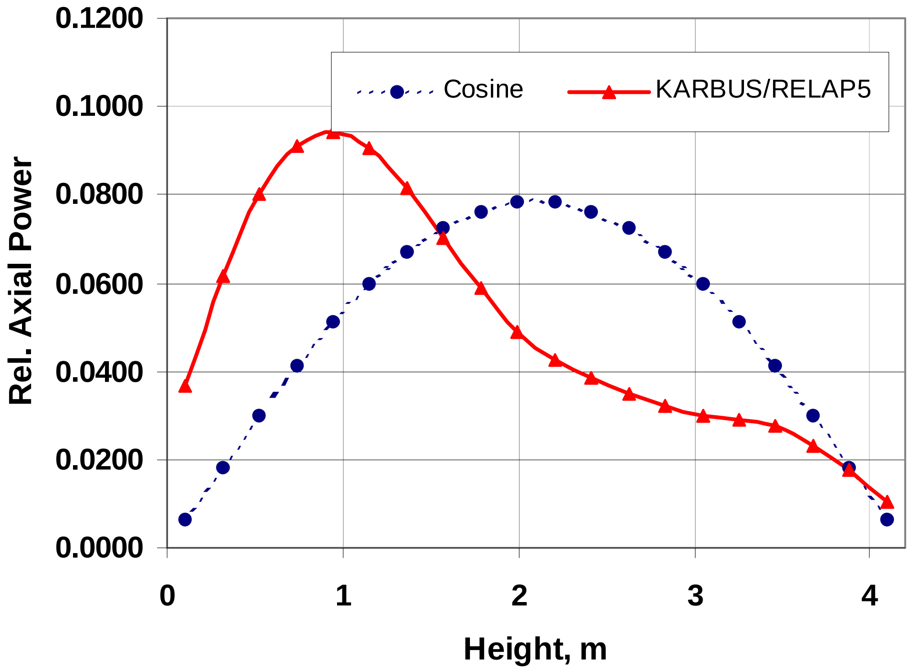

Figure 5-1 Comparison of assumed cosine-shaped and predicted power profile

5.2 Steady-state prediction for new axial power profile

The steady-state parameters of the reference reactor were recalculated using the axial power profile iterated by KARBUS/RELAP5. In Figure 5-2 the coolant density along the core height for the estimated and assumed power profile is presented. The discrepancy is quite large specially in the lower and middle part of the active core. The pseudo-critical point (658 K) moved downwards due to the higher coolant temperature, Figure 5-3.

The fuel and cladding temperature for both cases are shown in Figure 5-4 and Figure 5-5. In Figure 5-6 the heat transfer coefficient (HTC) is compared. The HTC-peak corresponds with the pseudo-critical coolant temperature. Both trends are comparable but the new HTC-maximum is shifted into lower position. The HTC-peak is determined by the thermo-physical properties of the coolant e.g. heat capacity, thermal conductivity, viscosity which undergo large variations around the pseudo-critical point.

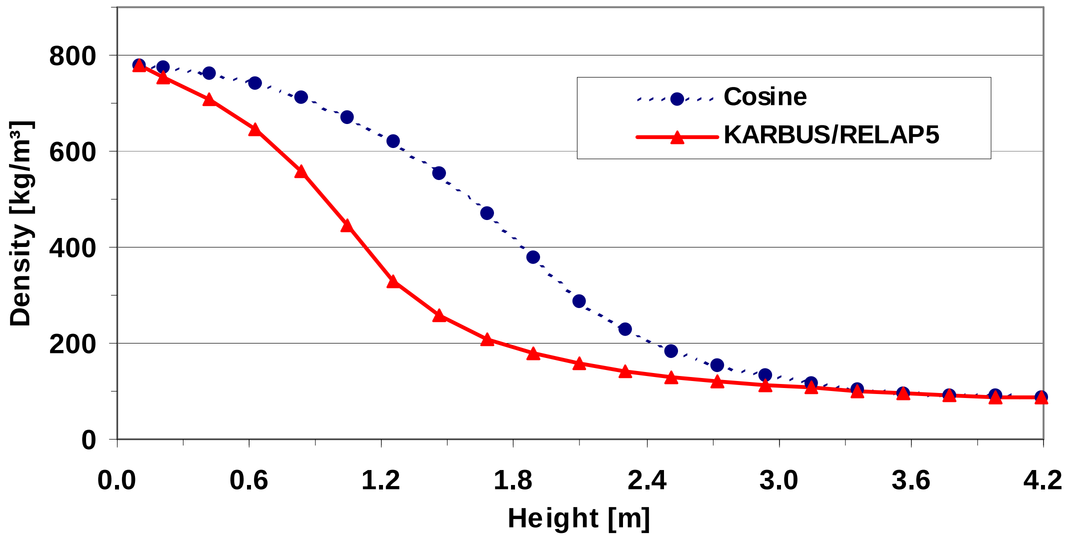

Figure 5-2 Predicted coolant density distribution for the reference case

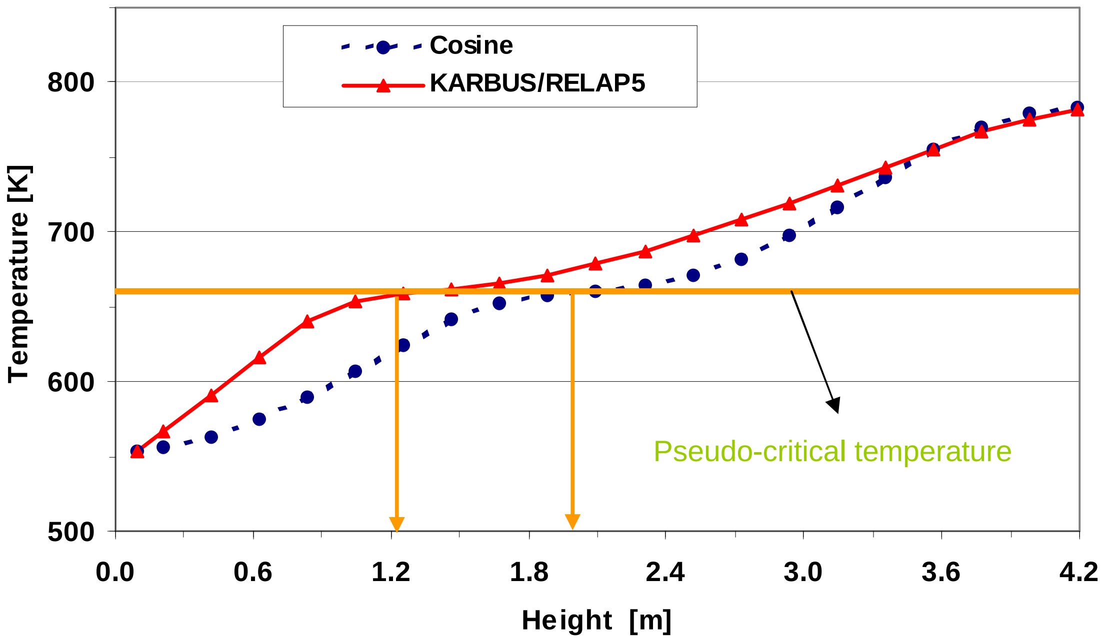

Figure 5-3 Predicted coolant temperature distribution for the reference case

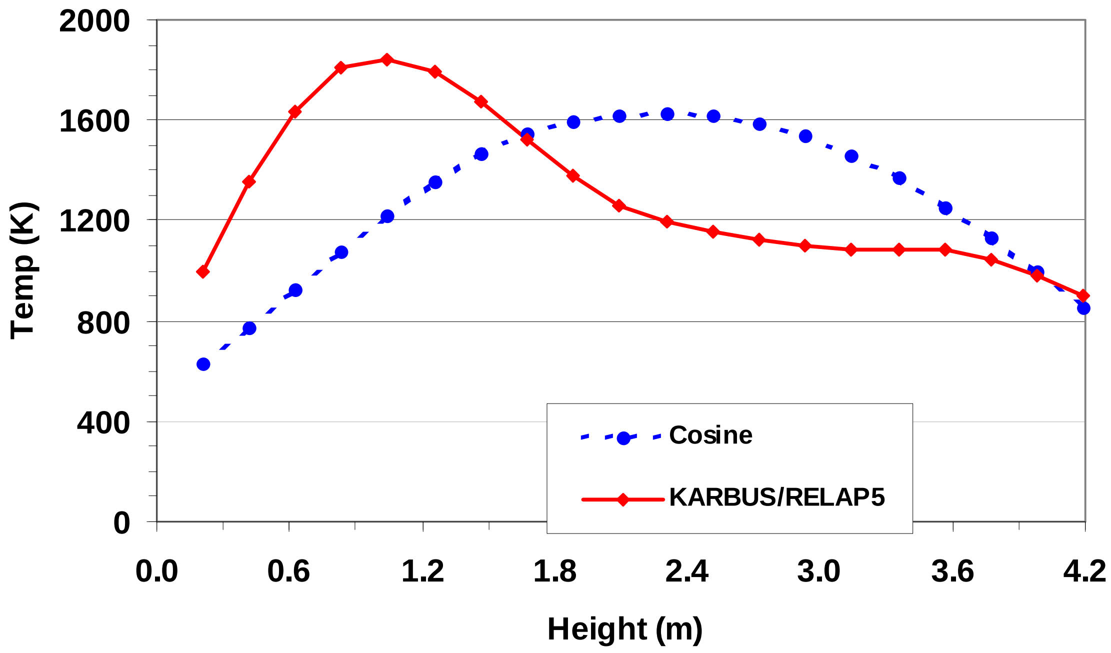

Figure 5-4 Fuel center line temperature distribution along the active core

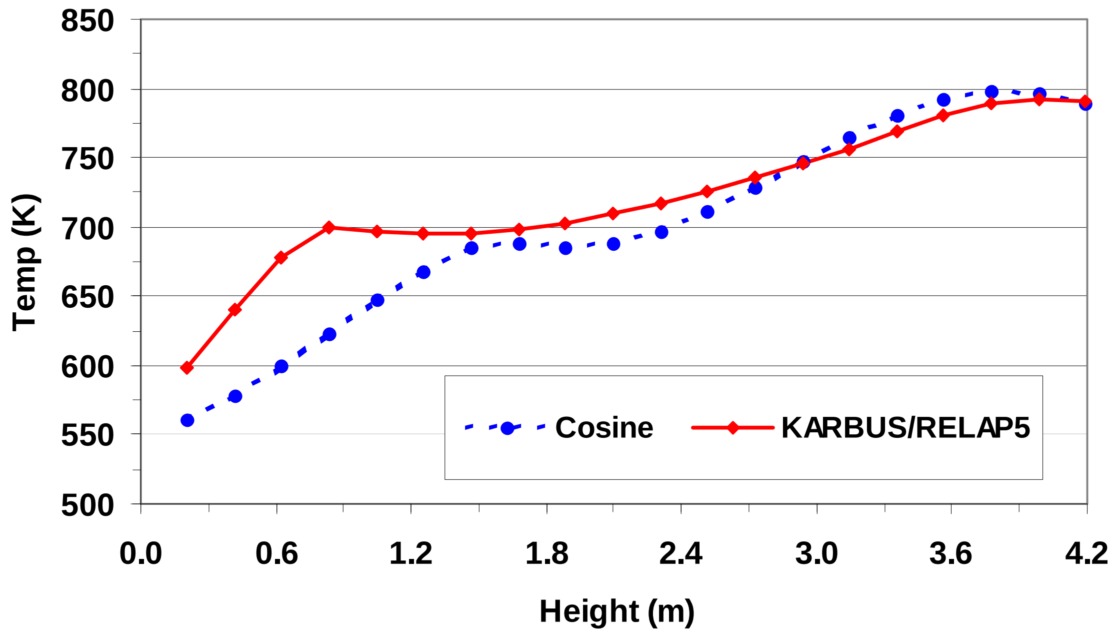

Figure 5-5 Cladding temperature distribution along active core

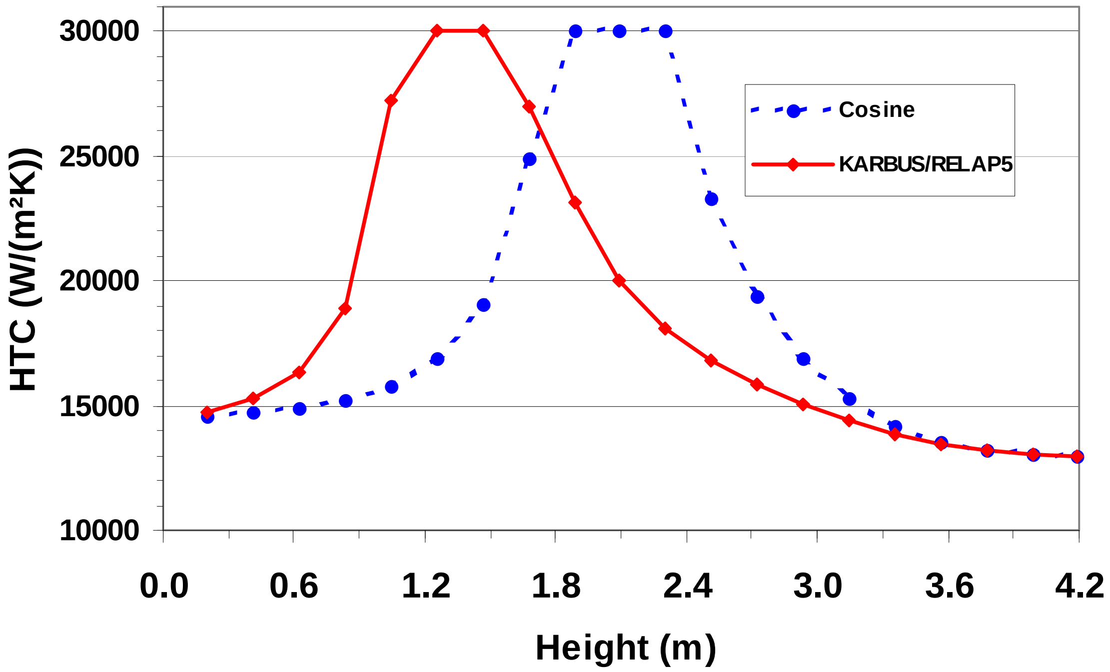

Figure 5-6 Comparison of the heat transfer coefficient for two axial power profiles

6 Simulation of selected transients

The main aim of the computational exercise is to check the applicability of the RELAP5-code for safety investigations of a new reactor operated at supercritical pressure and cooled and moderated by supercritical water. For this purpose, some postulated transients were selected to be analyzed with RELAP5. The simulation results will show how the reference plant may behave under transient conditions. Several transients were investigated by Oka and co-workers using a code especially developed for such type of reactor, [Kitoh97a], and [Kitoh99a]. All RELAP5-simulations presented here are performed with code versions using the point kinetics model taking into account given reactivity coefficients. The following transients were selected: a) Loss of feed water heating, b) Reduction of coolant flow and c) Loss of off-site power.

6.1 Loss of feedwater heating

It is assumed that one stage of the feedwater heaters fails [Kitoh97a], [Kitoh99a] leading to a sudden decrease of the feedwater temperature from 553 to 498 K within 1 s. For this simulation the integral plant model developed for the steady state prediction was extended with the point kinetics model. Additional modifications were necessary e.g. to consider the feedwater temperature reduction. The interaction of the plant thermal hydraulics with the core neutronics is taken into account in the point kinetics model by the moderator density and Doppler feedback coefficients. In this study, the reactivity coefficients given in [Doba98a] are applied, see Tab. 6-1.

Tab. 6-1 Reactivity coefficients used for transient analysis

[Doba98a]	Cold Zero Power	Hot Full Power
Density reactivity coefficient (dk/k(g/cm³)	**0.13**	**0.40**
Doppler reactivity coefficient (pcm/K	**-4.4**	**-2.6**

The feedwater heaters fail at time zero when the reactor is operated at nominal power e.g. 3568 MWth. As consequence of the feedwater temperature reduction, see Figure 6-1, the coolant density at the core inlet and thus the core average coolant density increases for a short period of time, see Figure 6-2. Thus the coolant mass flow rate oscillates and reduces as long as the feedwater temperature is sinking, see Figure 6-1. In this short period the cladding temperature of the hottest axial position increases, Figure 6-3, since the core power increases due to the better moderation and the heat transfer coefficient between cladding and supercritical water decreases.

Later on, the competing contribution of the density and Doppler reactivity coefficients, Figure 6-4, leads to a power decrease as long as the resulting total reactivity is decreasing. It reaches a minimum at about 1.7 s. The further evolution of the density, and Doppler reactivity is determined by the history of the core average coolant density and core average fuel temperature which show the same qualitative trends as **reacm** (moderator reactivity coefficient) and **reactf** (Doppler reactivity coefficient), see Figure 6-4. It is important to note that the cladding temperature of the hottest axial core node is continuously reducing and its highest value is below the temperature criteria 1113 K (for cladding material of Ni-based alloy). The reactor power remains below the scram condition of 112% of the nominal power at the end of the transient.

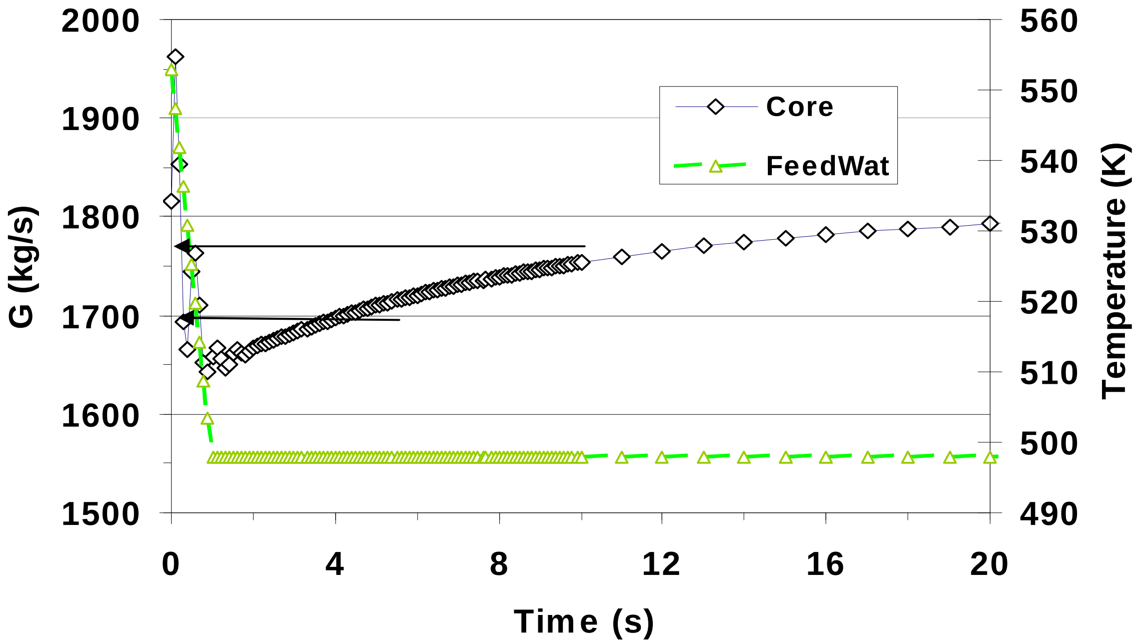

Figure 6-1 Core flow rate and feedwater temperature

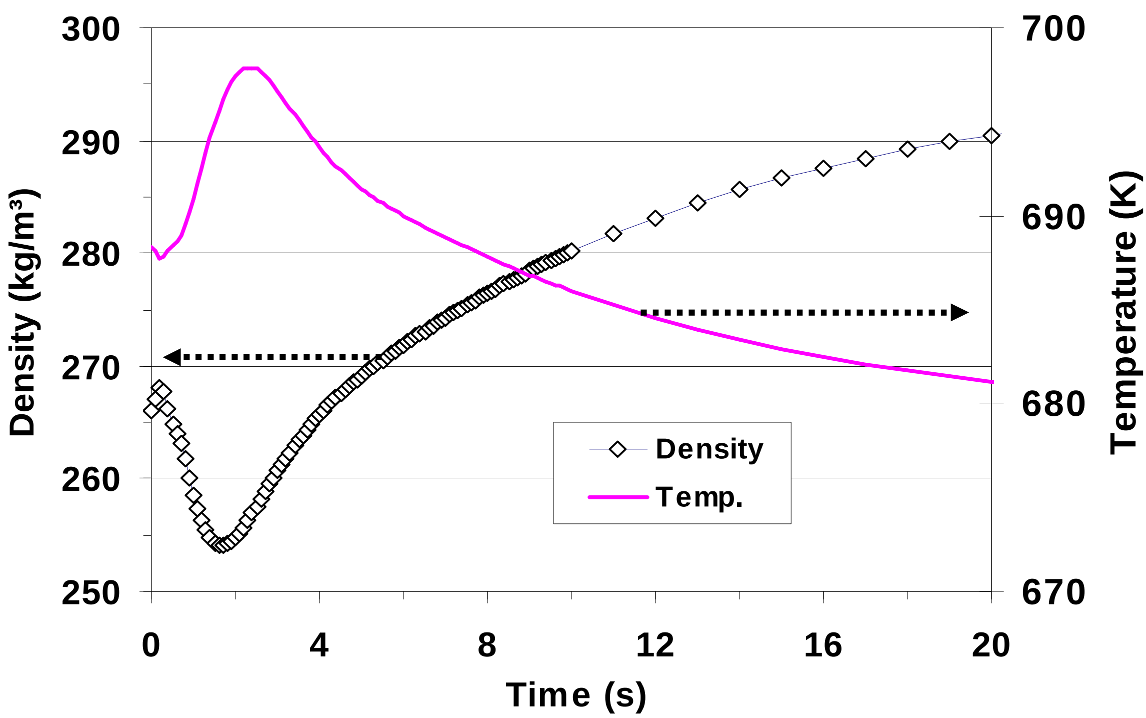

Figure 6-2 Core average coolant density and coolant temperature evolution

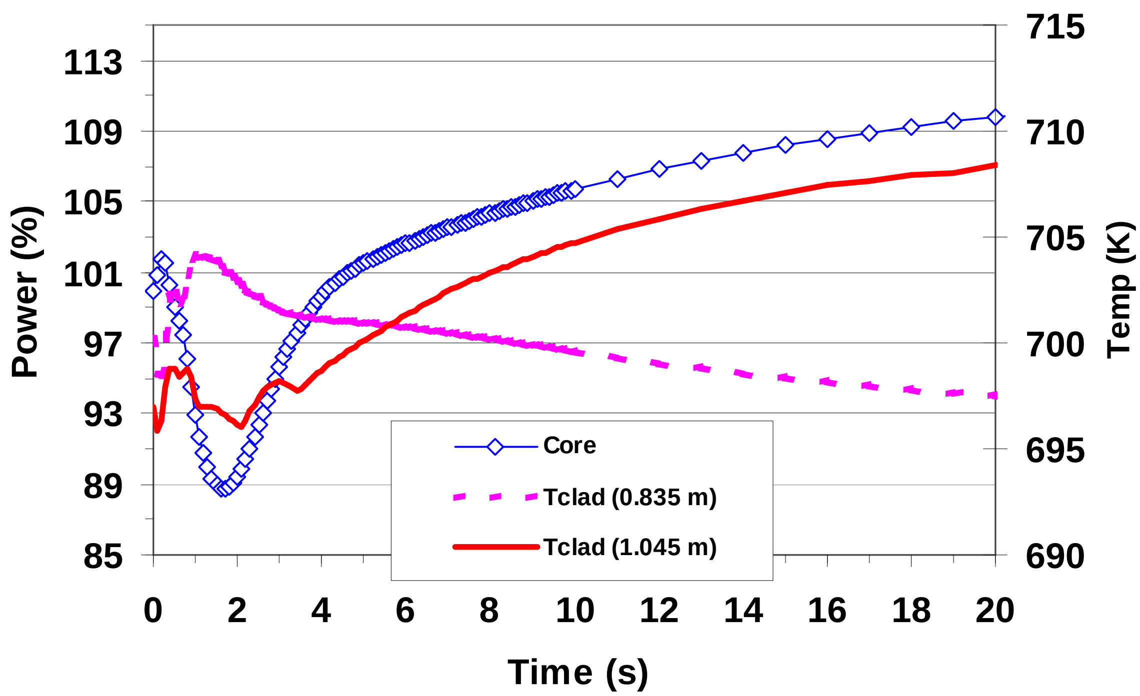

Figure 6-3 Relative thermal power and maximal cladding temperature at 0.835 m height

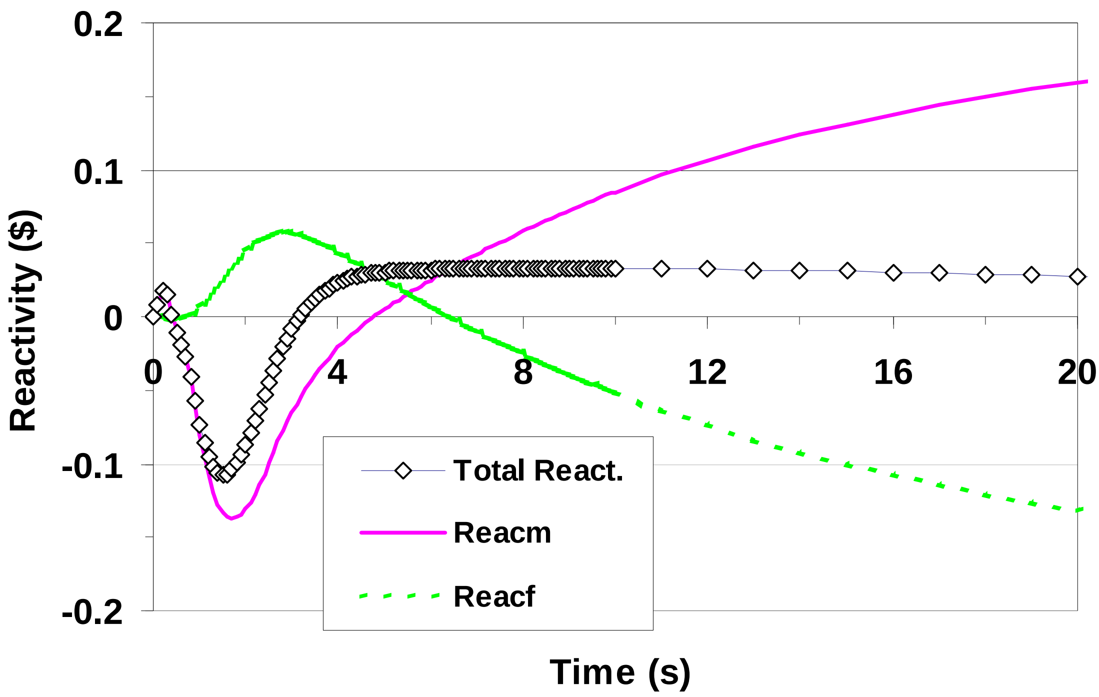

Figure 6-4 Evolution of the total reactivity compared to the contributions due to moderator (reacm) and fuel temperature (reactf)

As can be seen in Figure 6-3, the cladding temperature at different core elevations shows different trends underlining the strong variation of key core parameters along the core height. In this context, the poorness of the point kinetics approximation could be disastrous and not enough to catch the physics of the reference reactor.

6.2 Reduction of coolant flow

The postulated coolant flow reduction transient, [Leppän01], was investigated with RELAP5 using the same point kinetics model developed for the analysis of the loss of feedwater heater transient. It is assumed that the feedwater flow reduces very quickly from the nominal value of 1816 to 908 kg/s within 1 s. Corresponding transient assumptions were taken into account in the model.

At time zero, the feedwater flow decreases to 908 kg/s within 1 second. Consequently the core average coolant density and core average fuel temperature immediately start to decrease, see Figure 6-5, reaching a minimum value at about 2.5 s and 5 s, respectively. In this time period a considerable negative reactivity is added to the core by means of the moderator reactivity coefficient while the Doppler reactivity coefficient causes a positive reactivity insertion, see Figure 6-6. As a result of both competing processes the core becomes subcritical and the reactor power decreases as long as the resulting total reactivity is decreasing, Figure 6-7.

Later on, the average coolant density begins to increase reaching an asymptotic value at about 10 s in the transient. The core average fuel temperature shows a similar trend than the core average coolant density, but with some time delay. The core reactivity stabilizes at a subcritical level of -0.1 $ while the reactor power ends up at about 64% of the nominal power.

Typical for this reactor concept is a strong axial variation of important thermal hydraulic parameters. In Figure 6-8 the predicted heat transfer coefficient of two different axial elevations of the core is compared to each other. It reflects the local thermal hydraulic conditions prevailing at each elevation. These local conditions strongly vary during the transient progression.

The resulting fuel center line temperature at these elevations is given in Figure 6-9. In the upper core half a positive reactivity is inserted via the Doppler feedback while in the lower core half a negative reactivity is added to the core. The cladding temperature behaves similarly as can be observed in Figure 6-10. A maximum cladding temperature value of 975 K was predicted at the core height of 0.4095 m. This value lies below the maximal cladding temperature of 1113 K fixed by the transient criteria for Ni-alloy cladding.

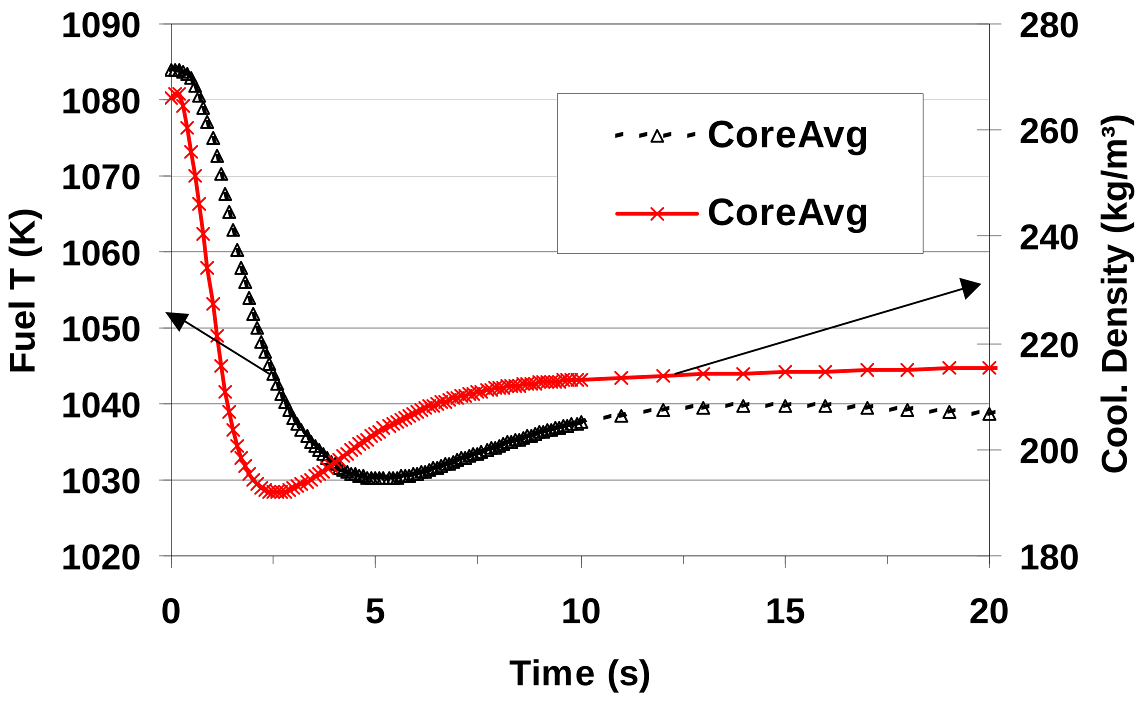

Figure 6-5 Evolution of the core averaged coolant density and fuel rod temperature

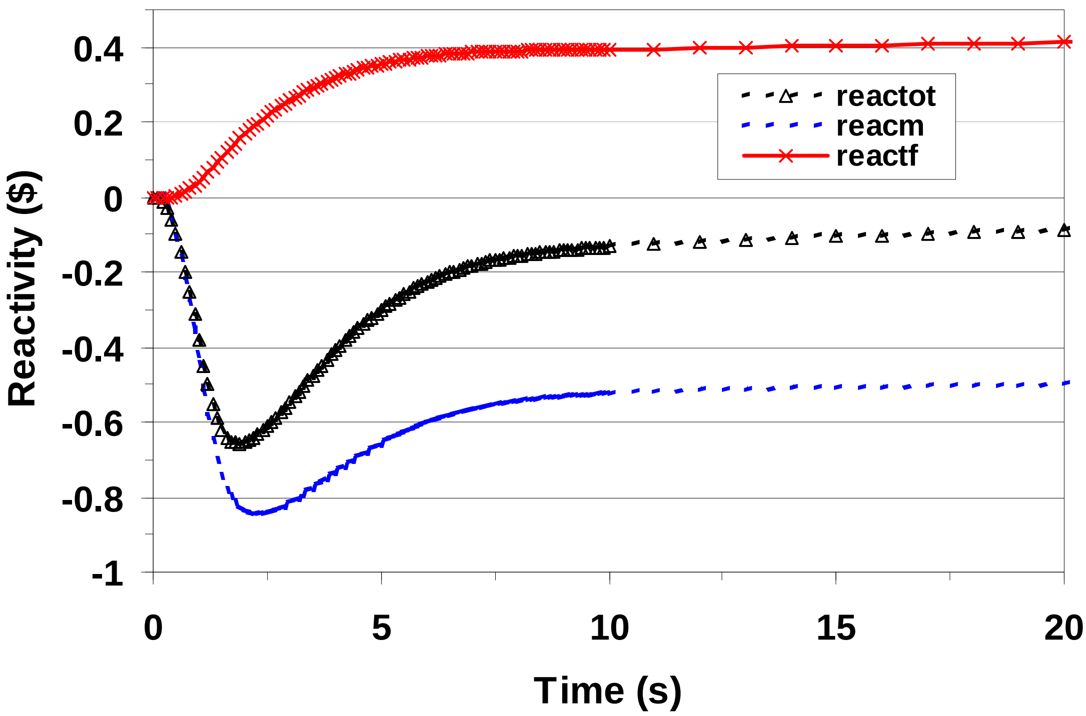

Figure 6-6 Evolution of the total reactivity with its contribution from the moderator and Doppler effect

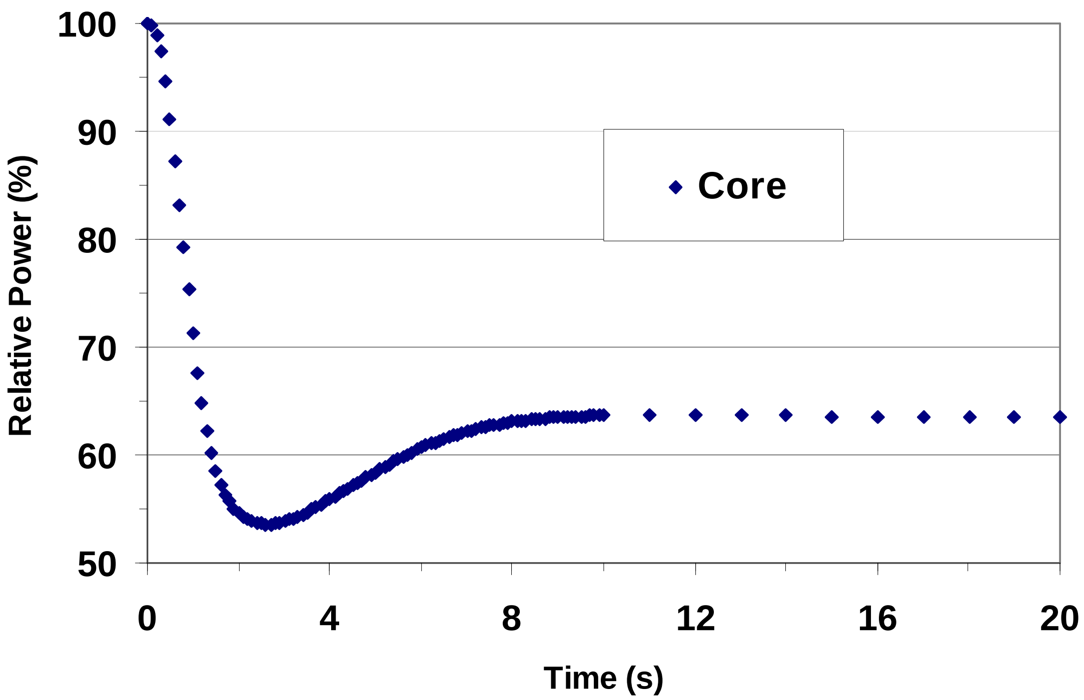

Figure 6-7 Predicted trend of the relative core power

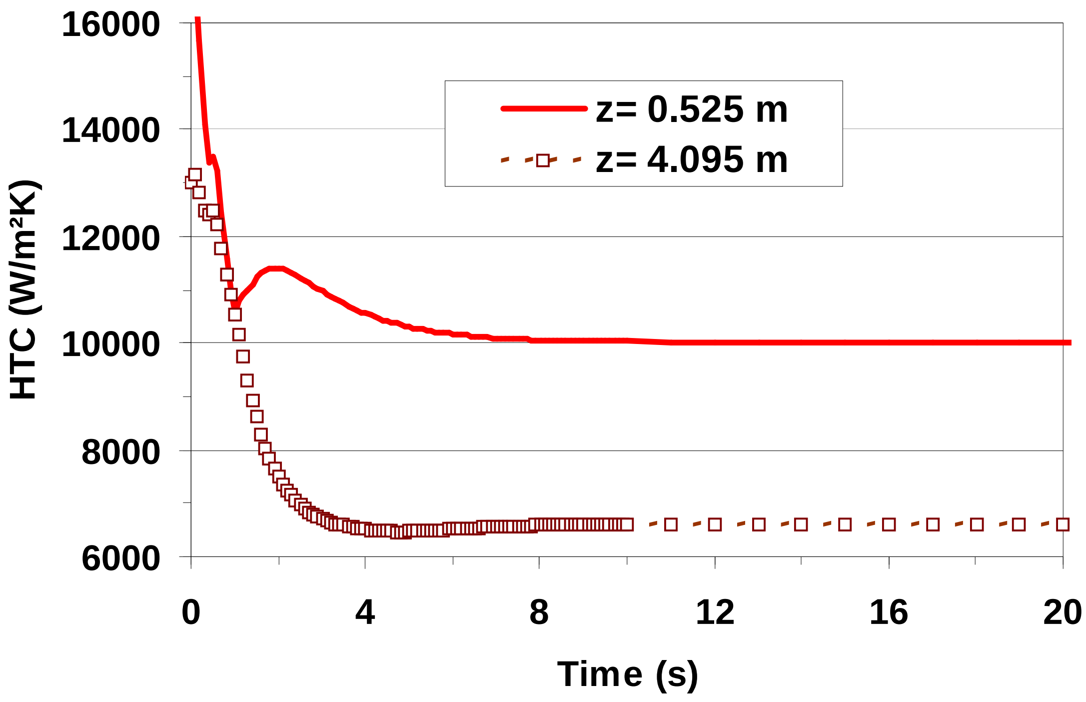

Figure 6-8 Predicted heat transfer coefficient for two core elevations

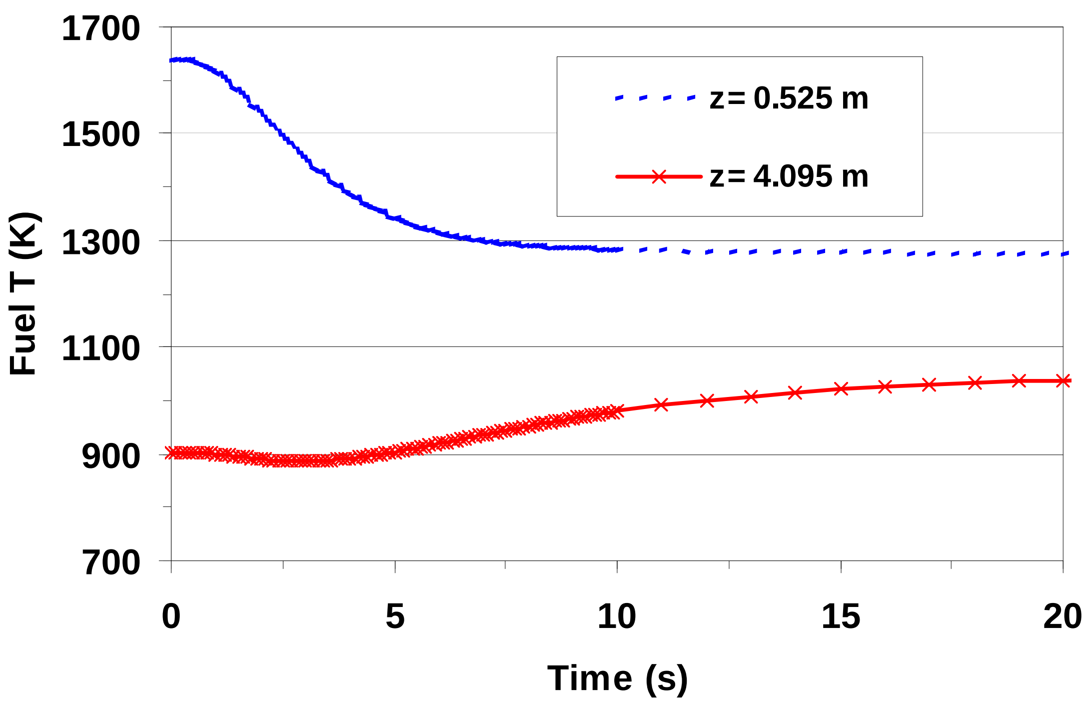

Figure 6-9 Predicted fuel center line temperature at two core elevations

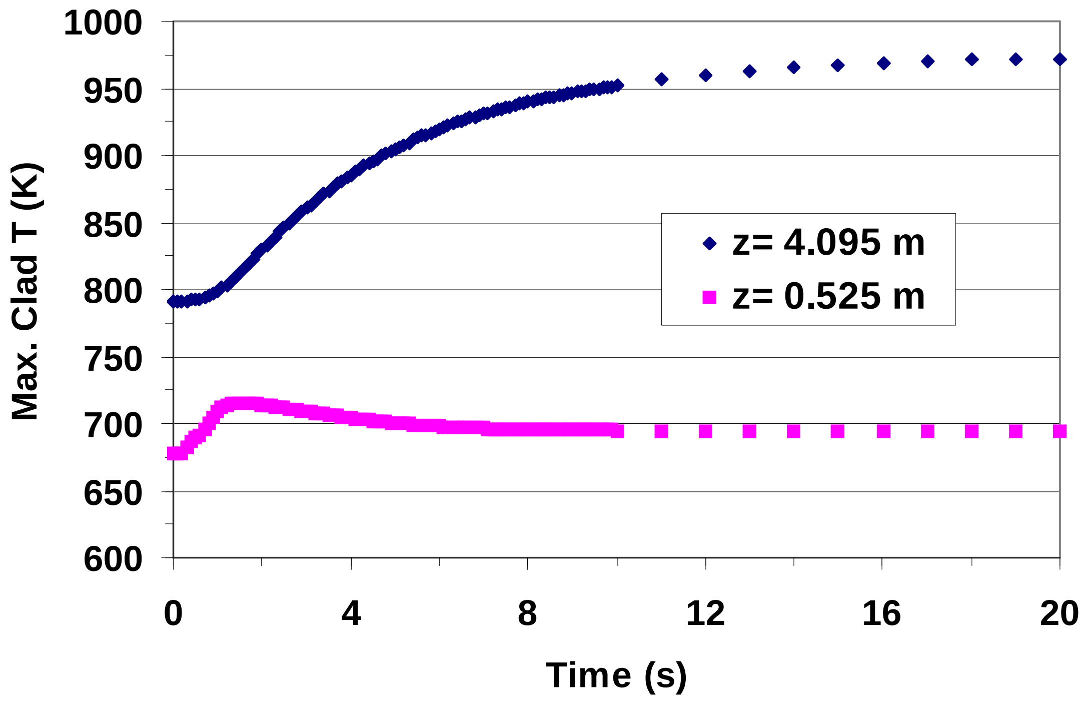

Figure 6-10 Predicted maximal cladding temperature

6.3 Loss of off-site power

The transient is initiated by the postulated loss of off-site power leading to shutdown of the motor-driven main feedwater pumps. It is assumed that the pump trip signal becomes active 5 s after transient begin [Kitoh99a]. Afterwards the mass flow rate is linearly ramped within 5 s to a level of approximately 5 % of the nominal value. The scram signal is assumed to occur at transient initiation. The control rod banks are fully inserted after 2.3 s introducing an assumed reactivity of –8.7 $.

The coolant flow was considered in the integral plant model for RELAP5 as boundary condition, see Figure 6-11. Since the total power starts to decrease from the very beginning, Figure 6-12, the coolant temperature reduces until about 5 s. In this period the RPV-pressure decreases as a consequence of the coolant contraction during the cooling down, Figure 6-13. Later on the reduced amount of coolant entering the core heats up even though the power level is very low. The cladding temperature follows in some way the coolant temperature trends. RELAP5 predicted the maximum cladding temperature at the elevation of 3.46 m, see Figure 6-14. It can be stated that this maximum temperature is below the maximal cladding temperature criteria for transients i.e. below the 1113 K for Ni-alloy cladding.

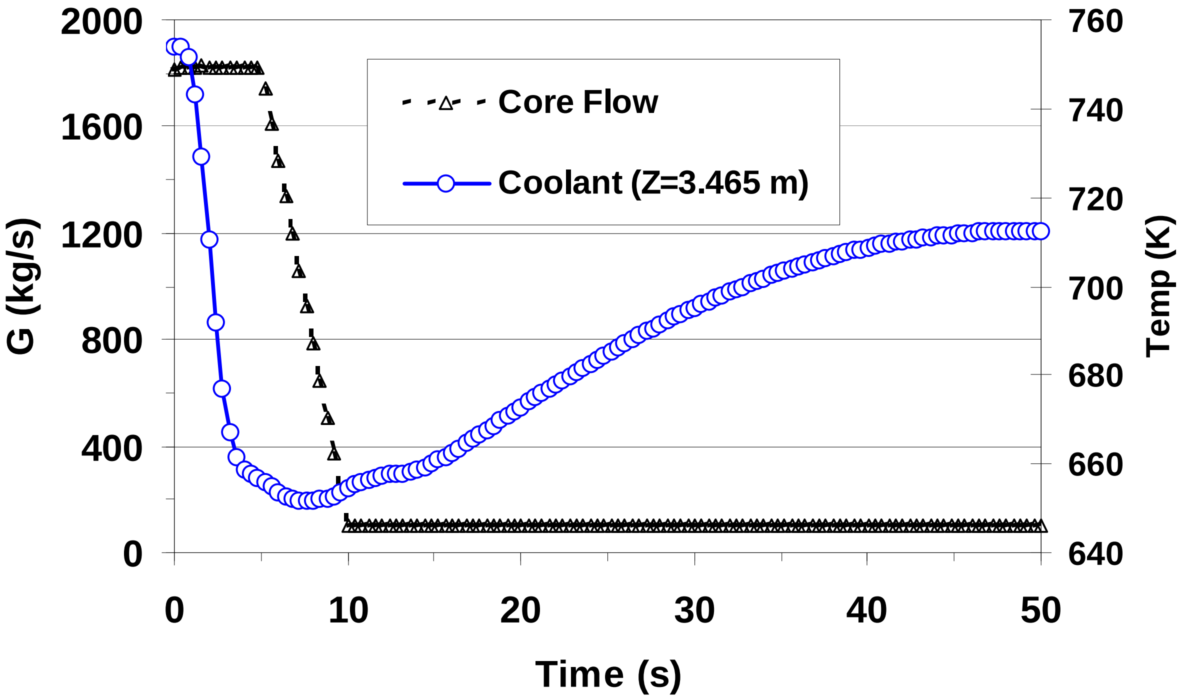

Figure 6-11 Coolant mass flow rate and predicted coolant temperature at height 3.465 m

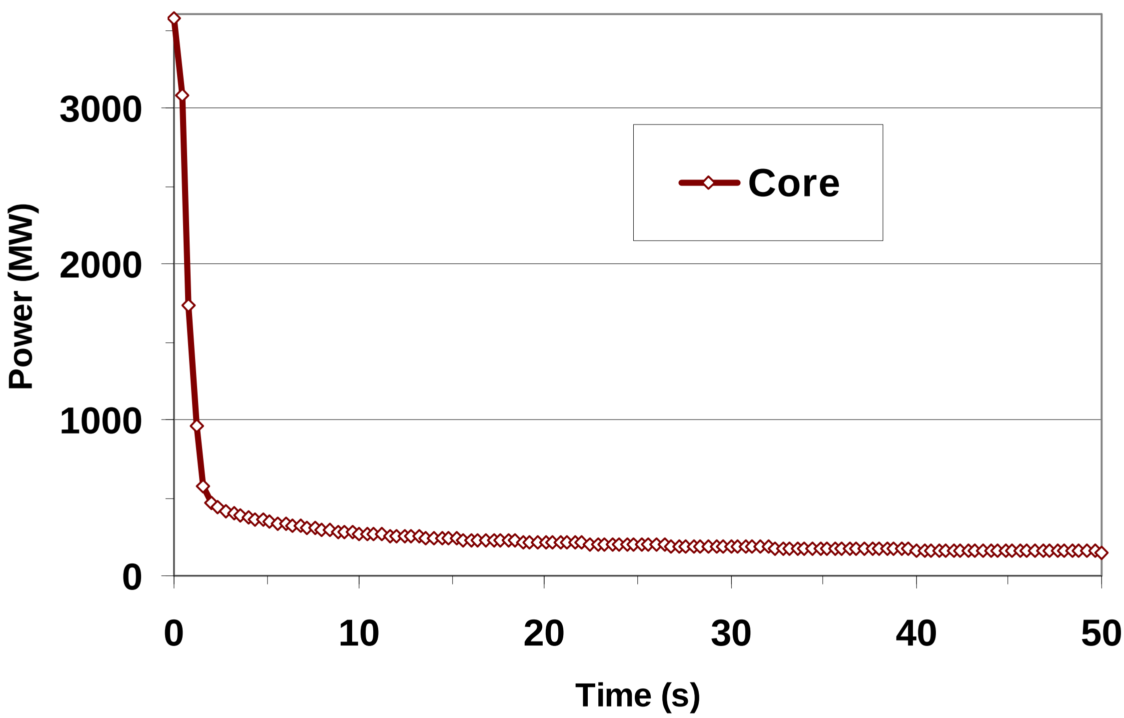

Figure 6-12 Thermal power reduction after scram

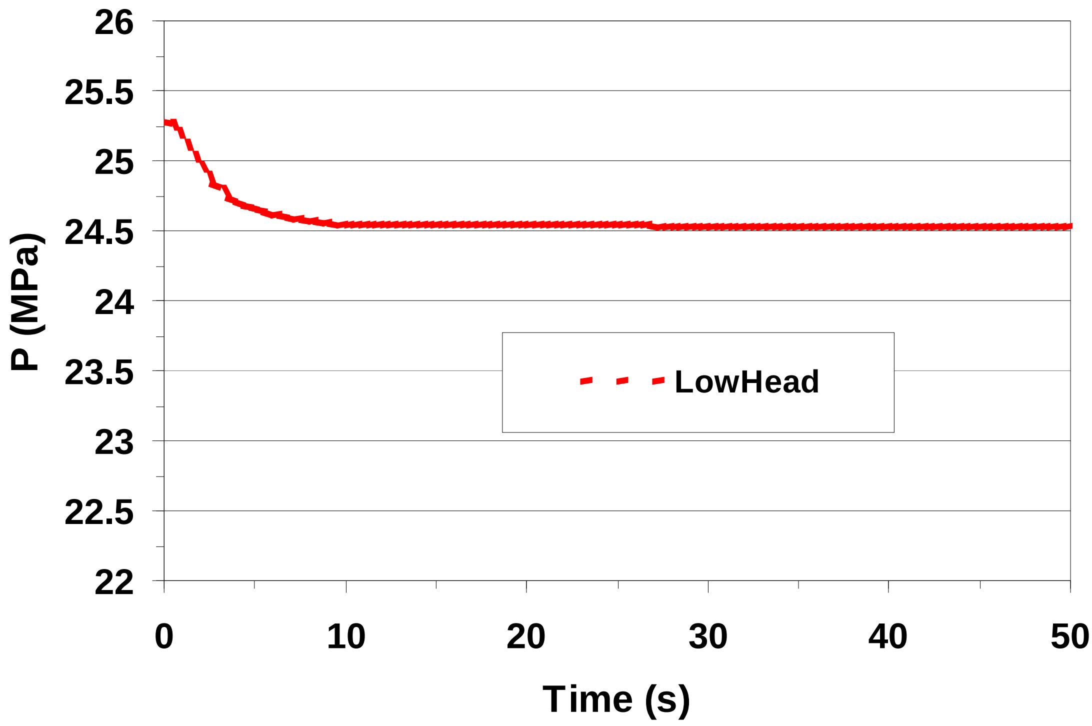

Figure 6-13 Pressure evolution of the RPV predicted by RELAP5

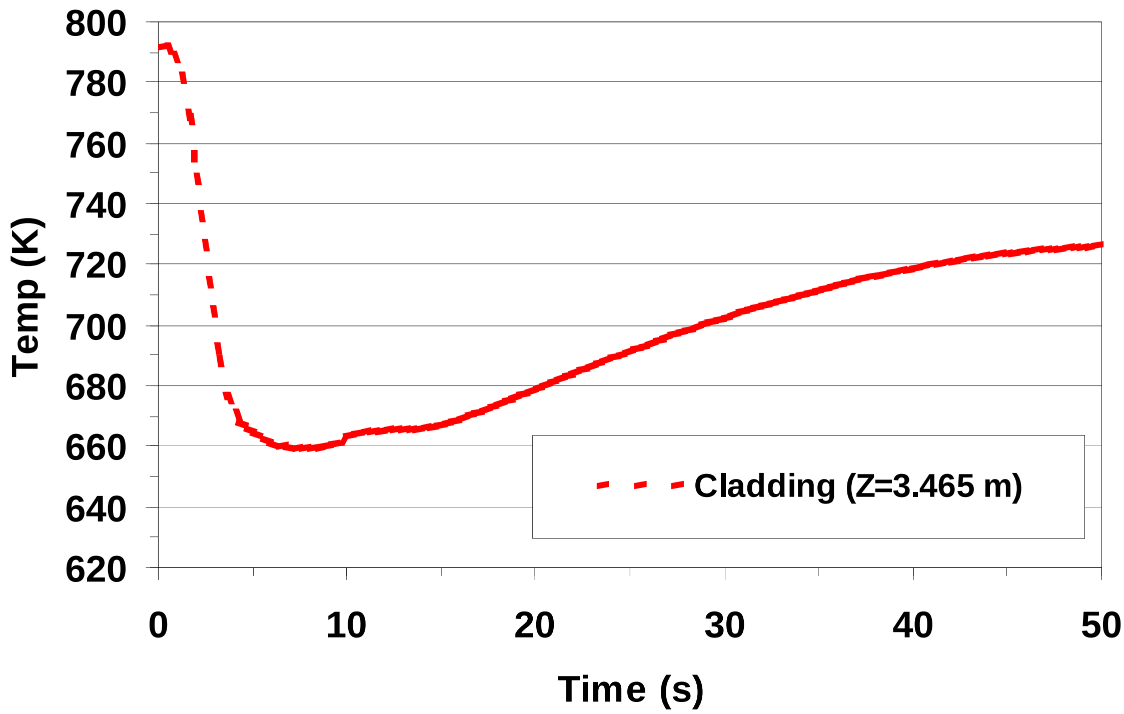

Figure 6-14 Maximal cladding temperature during the transient

7 Simulation of loss of coolant accidents

To investigate the code's capabilities to predict LOCAs of thermodynamically supercritical reactors, several LOCA-scenarios beginning with the blow down phase are considered. In all calculations, a point kinetics model is applied.

7.1 Blow-down phase

The blow-down phase of a large break located in the steam line (0.22 m^2) close to the RPV is analyzed with RELAP5. The following assumptions were considered according to [Koshi94a], [Lee94a], [Lee98a]:

- No actuation of emergency core cooling systems,
- Feed-water injection stops after break opening (containment isolation),
- Turbine stop valves (TSV) close after break opening (containment isolation),
- Control rod banks insertion (- 8.7 $) in 2.3 s,
- Control rod bank signal delay amounts 0.3 s.

Scram is caused once the feedwater mass flow rate reduces below 90 % of its nominal value.

Preliminary investigations have shown that the blow-down starting at nominal pressure, i.e. 25 MPa, is very challenging for two-phase codes like RELAP5.

It has to be noted that at nominal conditions, the reference reactor is characterized by an axial coolant temperature varying from subcritical to supercritical, as can be seen in Figure 7-1 (scenario A).

During the blowdown phase, both pressure and coolant temperature suddenly decreases after break opening. Once the system pressure and the coolant temperature simultaneously pass the critical point downwards, RELAP5 fails due to problems in the interpolation scheme, especially near the critical point, when the fluid temperature goes below the critical temperature. Under such situations, the derivatives of some quantities like the isobar (β), isothermal (κ) compressibility go to infinity. An alternative method is here necessary, [Mort02a], to be able to predict all thermo-physical properties under de-pressurization transients.

7.2 LOCA-Analysis starting from artificial steady state conditions

An artificial scenario B was defined to check whether RELAP5 is able to simulate the blow-down starting at nominal pressure but with an axial coolant temperature distribution below the critical temperature.

The main differences between scenario A (normal steady state) and B (artificial steady state) are given in Tab. 7-1 and Figure 7-2.

Tab. 7-2 Differences between the inventory in scenario A and B at steady state conditions

Cases	**Scenario-A**	**Scenario-B**
Down-comer (kg)	19900	19900
Lower plenum (kg)	19200	19200
Core (kg)	2900	4500
Hot box (kg)	1400	12300
RPV (kg)	43500	5600

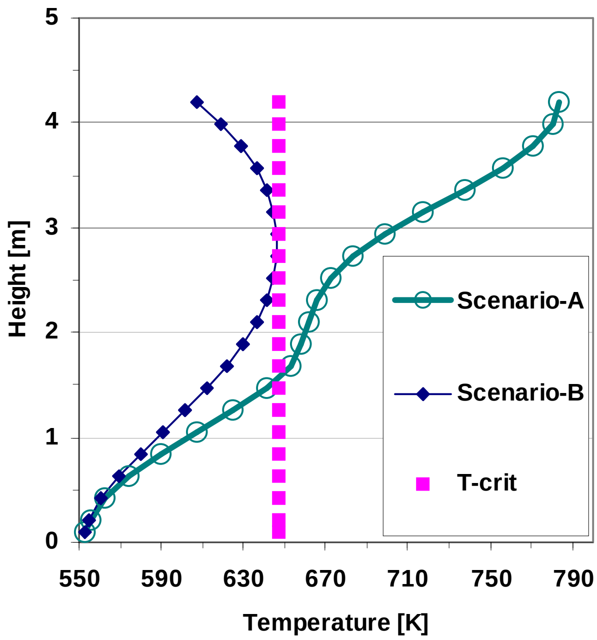

Figure 7-1 Axial coolant temperature distribution at steady state for scenario A and B

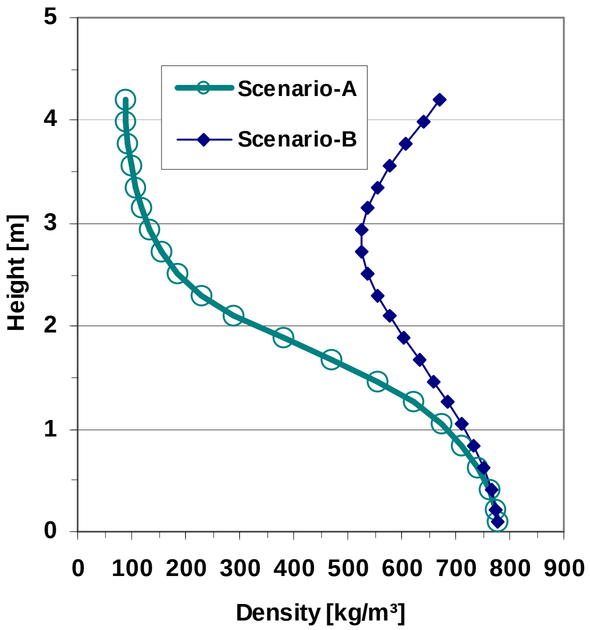

Figure 7-2 Axial density distribution at steady state conditions for scenario A and B

The results presented hereafter were obtained with RELAP5 for the scenario B. This scenario was simulated by RELAP5 without difficulties. Although the initial steady state conditions of scenario B do not match the conditions of scenario A, the parameter trends obtained by RELAP5 for the LOCA analysis under different conditions e.g. break size, actuation of ECCS (accumulators, low pressure coolant injection) and the location of ECC-injection give qualitatively the behavior of the reference plant that may be encountered under the postulated accidental conditions.

7.2.1 Influence of the break size in the steam line for a LB-LOCA

Three different break sizes were investigated for the LB-LOCA in the steam line assuming no actuation of the emergency core cooling systems. In Figure 7-3 break outflow calculated for three break sizes are compared. The subsequent system pressure reduction due to boil-off of the primary coolant in the RPV is shown in Figure 7-4. As expected, the fastest depressurization is encountered for the largest break size (0.22 m²). For this case, the highest mass flow rate is predicted by RELAP5.

It must be pointed out that the de-pressurization from the supercritical region (25 MPa) to the subcritical region is very fast and it takes less than one second. The resulting evolution of the maximal cladding temperature is depicted in Figure 7-5. The highest values of the cladding temperature were predicted for the break size 0.073 and 0.11 m². It corresponds to the axial height of 1.365 m and 1.785 m, respectively. Later on a second outflow peak for all three cases is predicted, Figure 7-3. The largest spike belongs to the 0.22 m² break size. The other peaks are not so high but appear delayed in time. Hence a better cooling of the core is determined for the smallest break size in the time period investigated here.

For the other break sizes, the maximal cladding temperature begins to increase after about 25 s and 40 s since the remaining coolant is not enough to remove the decay heat when no LPCI is activated.

For all three break sizes, the maximal cladding temperature does not exceed the 1473 K temperature criteria for DBA in case of LOCAs. As can be seen in Figure 7-3, the total RPV-fluid inventory of about 56 tons is rapidly dumped out after 30 s and 45 s in case of the break size of 0.22 m² and 0.11 m², respectively. The water inventory within the RPV of the reference plant is too small to prevent the core overcooling and to safely remove the decay heat from the core. In current LWR, an amount of around 300 tons is available within the RPV.

Consequently the maximal cladding temperature starts to increase very early in the transient, i.e. after 24 s, and 42 s for the break sizes of 0.22 and 0.11 m², see Figure 7-5. The same trend with some time delay is expected for the break size of 0.073 m².

Based on these results it can be stated that the time window for the actuation of the emergency core cooling systems (accumulators, LPCI) seems to be very short and that the core overheating can not be prevented unless respective ECC-systems are foreseen. The key question to be answered in the next project stage is when, where and how much coolant may be injected into the RPV to guarantee a long term decay heat removal without serious fuel rod integrity loss.

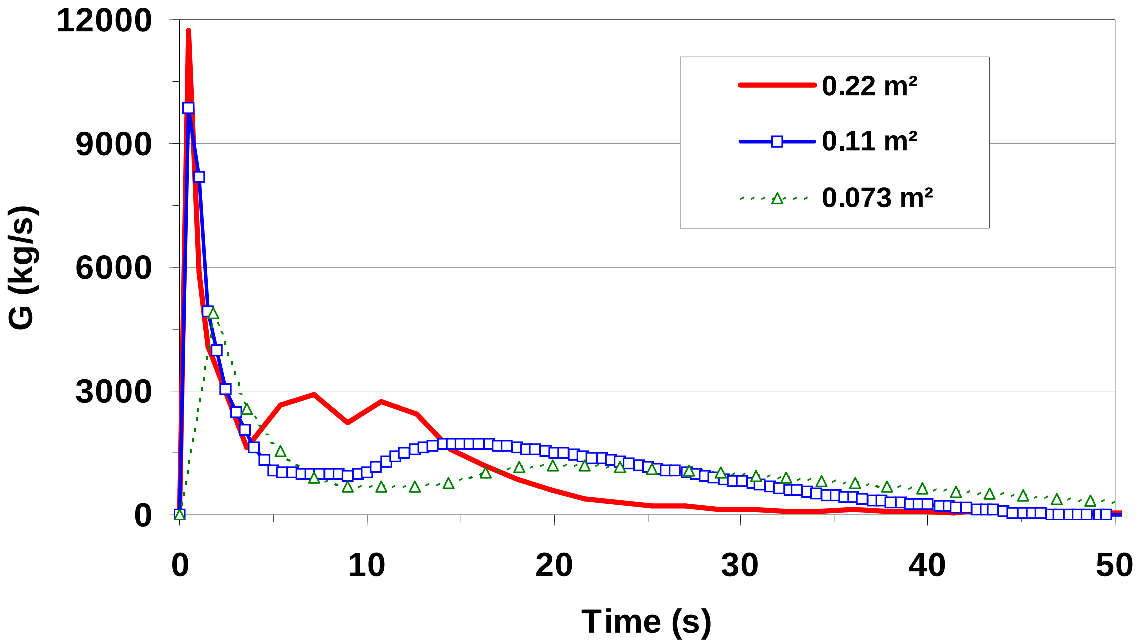

Figure 7-3 Break outflow during the blowdown phase of LB-LOCA

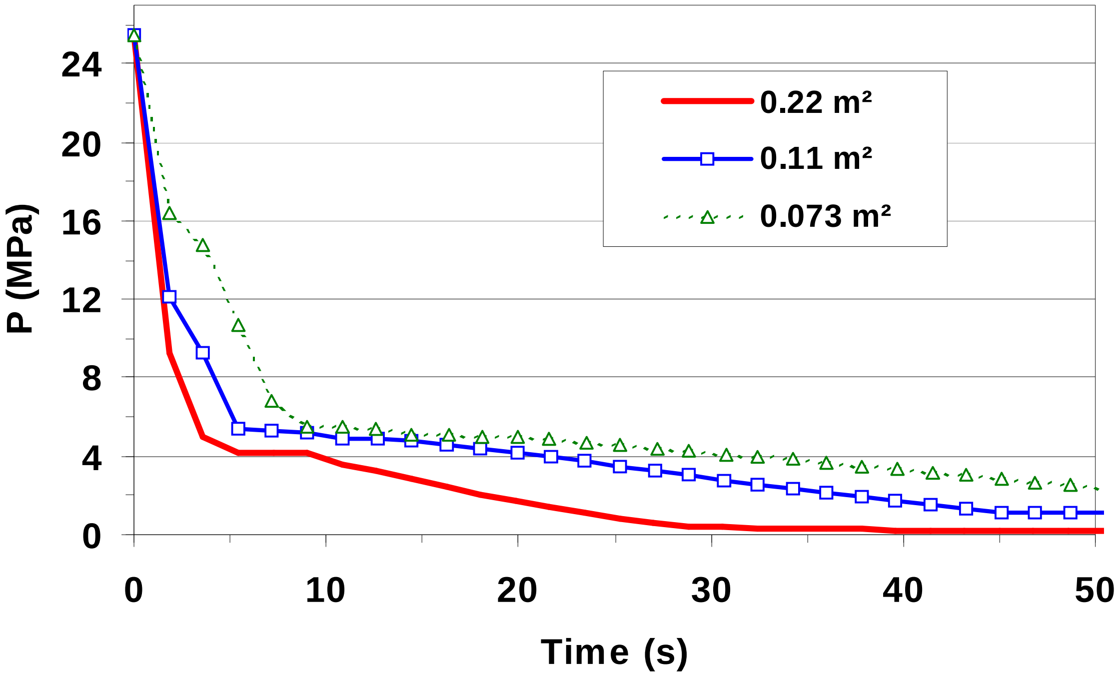

Figure 7-4 RCS-Pressure evolution during the Blowdown phase of LB-LOCA

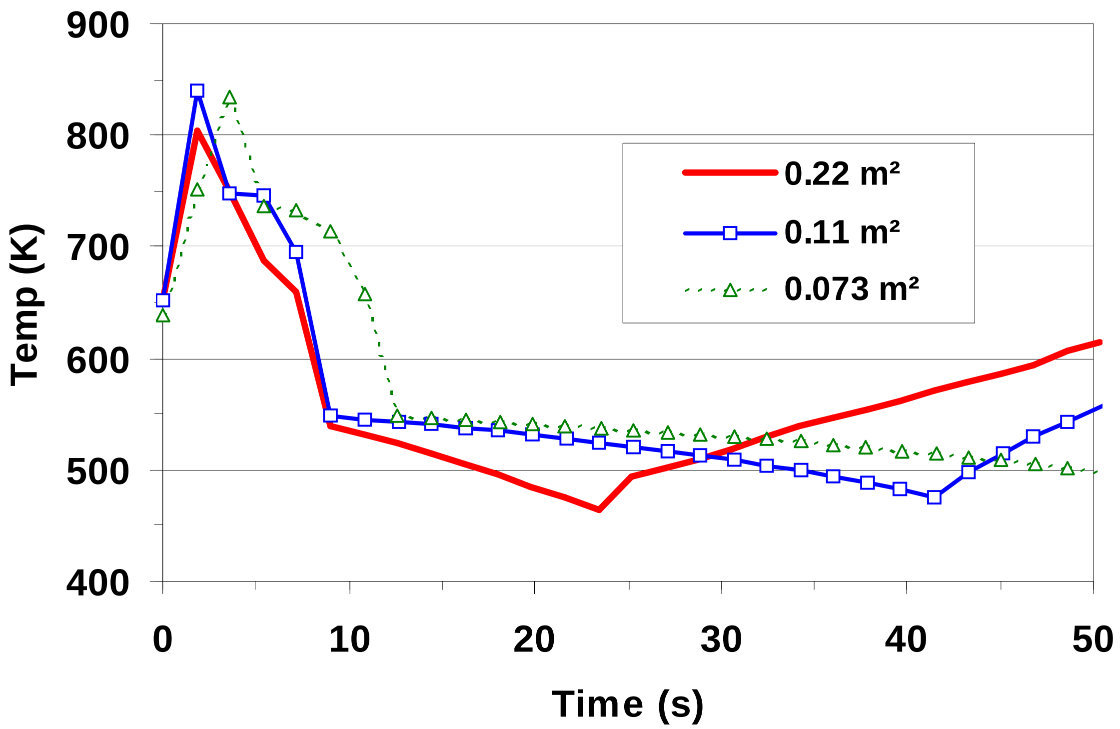

Figure 7-5 Cladding temperature evolution during the blowdown phase of LB-LOCA

7.2.2 Influence of the injection rate of the emergency core coolant system in case of LB-LOCA in the steam line

For the large break LOCA in the steam line break the following assumptions were investigated:

- No LPCI actuation.
- LPCI actuation with an injection rate of 1000 kg/s.
- LPCI actuation with an injection rate of 1600 kg/s.

The LPCI signal is activated when the system pressure falls below 4 MPa. The cold water injection into the intact feedwater line starts only 25 s later. In Figure 7-6 the predicted outflow through the break for these cases is exhibited. If no LPCI is activated the RPV becomes in few seconds empty and the maximal cladding temperature steadily increases due to the insufficient decay heat removal, see Figure 7-7. The injection of 1000 and 1600 kg/s cold water, leads to a turnaround of the cladding temperature escalation. The higher the injection rate, the more effective is the core cooling. Later on the cladding temperature begins again to increase, when the break outflow goes again to almost zero (around 70 s). These results clearly show that the injection rate of about 850 kg/s per unit proposed for the reference plant are insufficient to maintain a safe long term core cooling. This problem may be tackled by the consideration of additional passive safety systems e.g. accumulators or by the increase of the injection rate of the LPCI, etc..

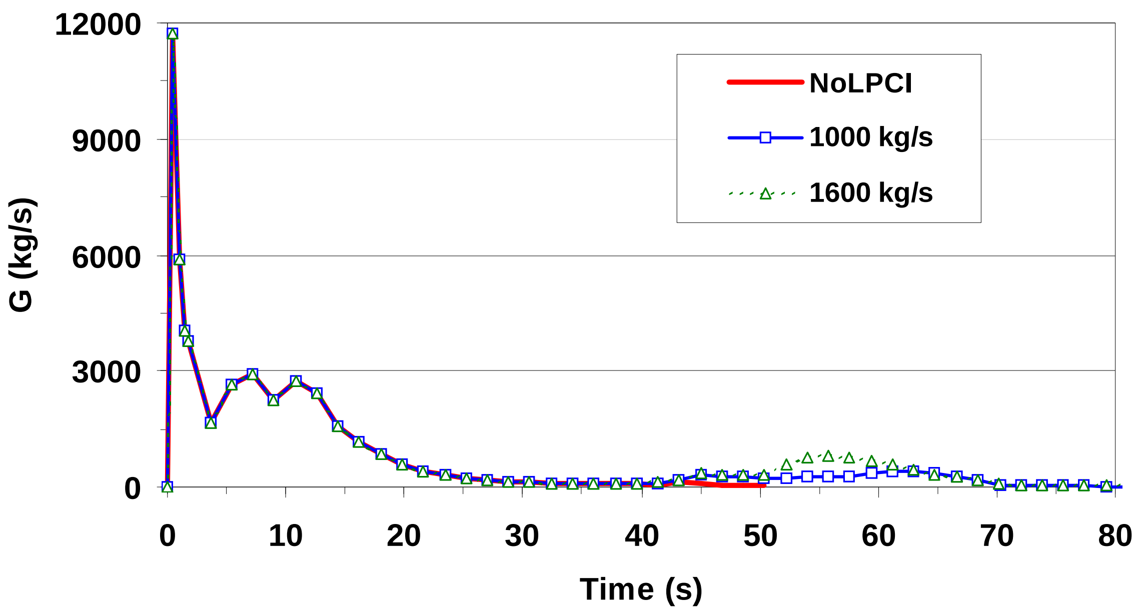

Figure 7-6 Predicted break outflow for the investigated cases

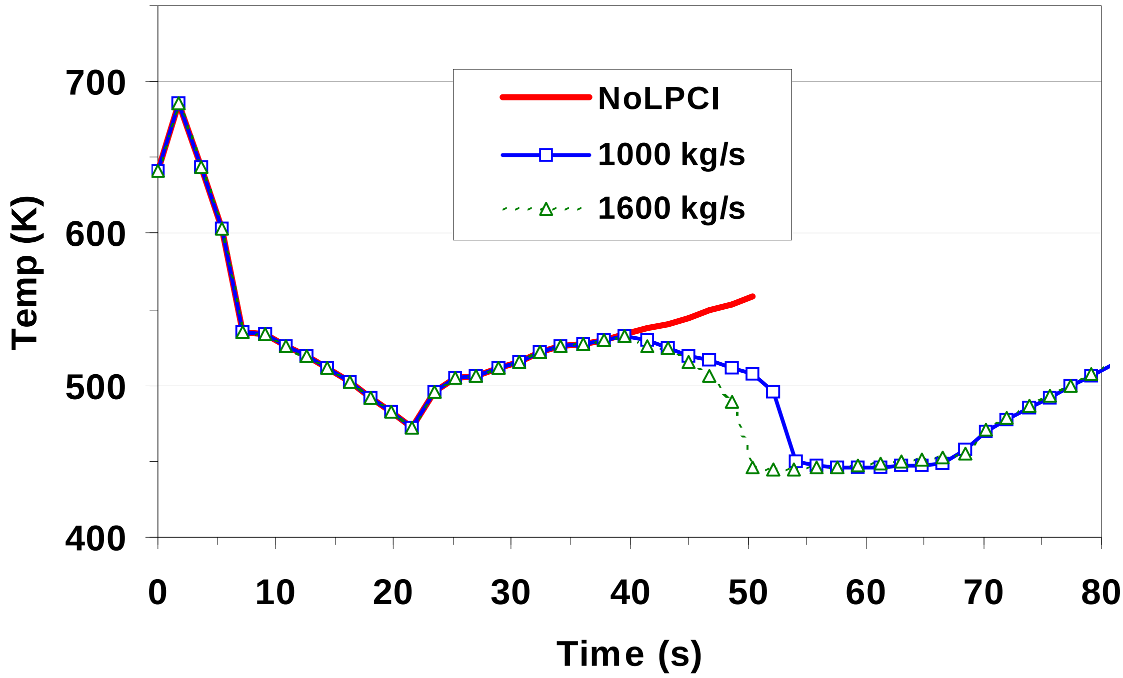

Figure 7-7 Comparison of predicted cladding temperature for the LB-LOCA

7.2.3 Influence of the accumulators in case of a large break in the steam line

To explore the efficiency of different ECC-systems, it is assumed that both the passive accumulators and the low pressure coolant injection system are available. Ones the system pressure falls below 13 and 4 MPa, the ACC and LPCI are activated, respectively. The injection rate of the accumulators and of the LPCI was assumed to be 2000 kg/s and 1000 kg/s. A delay time of 25 s was considered for the LPCI-system. In Figure 7-8 the break outflow rate for this scenario is compared to the cases without any ECC-injection and with only LPCI. Initially the break outflow rate is similar for all cases. Later on, the outflow trends develop differently as consequence of the ECC-actuation. The injection rate and time of actuation of the accumulator and LPCI are represented in Figure 7-9. The resulting maximal cladding temperature for the different cases is compared in Figure 7-10. It can be stated that the actuation of the accumulators in connection with the LPCI is very efficient to cooldown the fuel rods and bring them to low, safe temperatures.

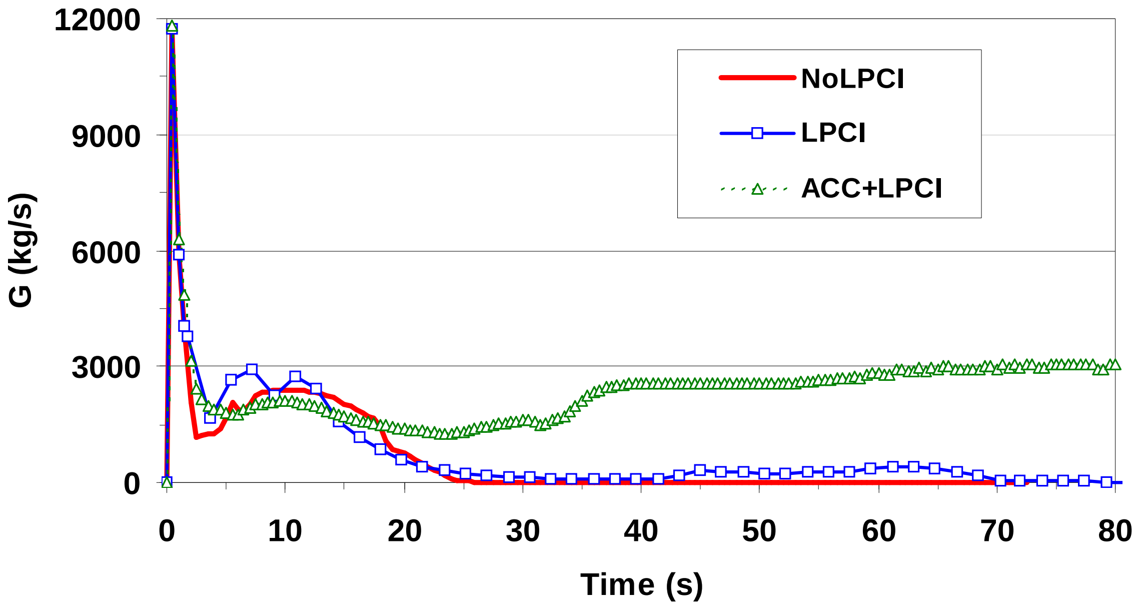

Figure 7-8 Predicted break outflow for three different scenarios

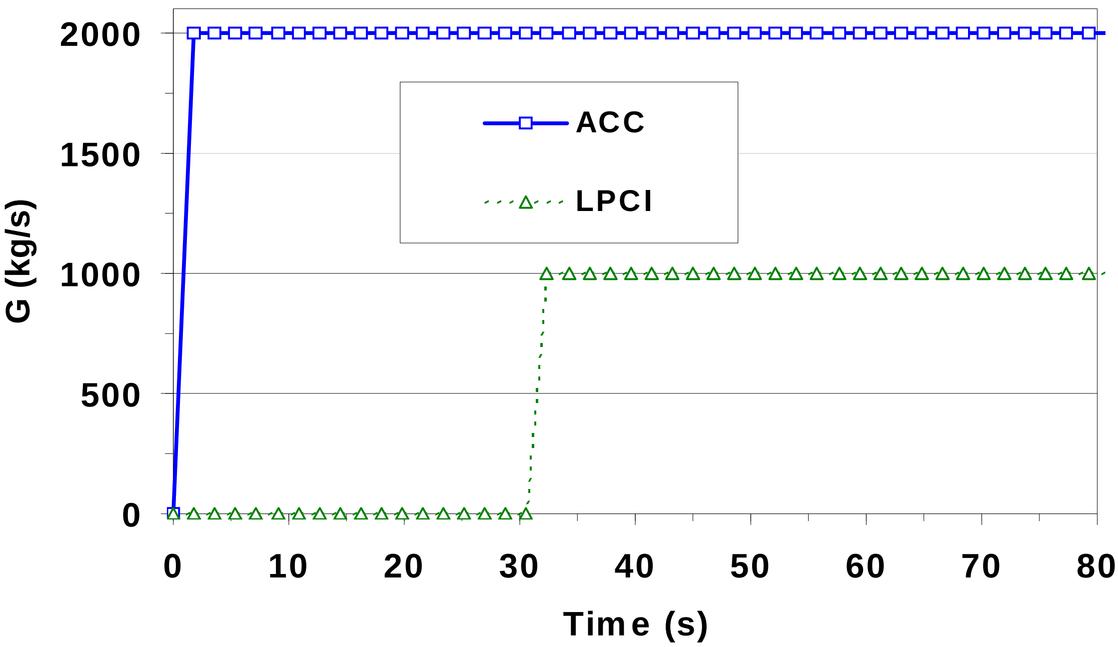

Figure 7-9 Injection rate of the accumulators and of the LPCI

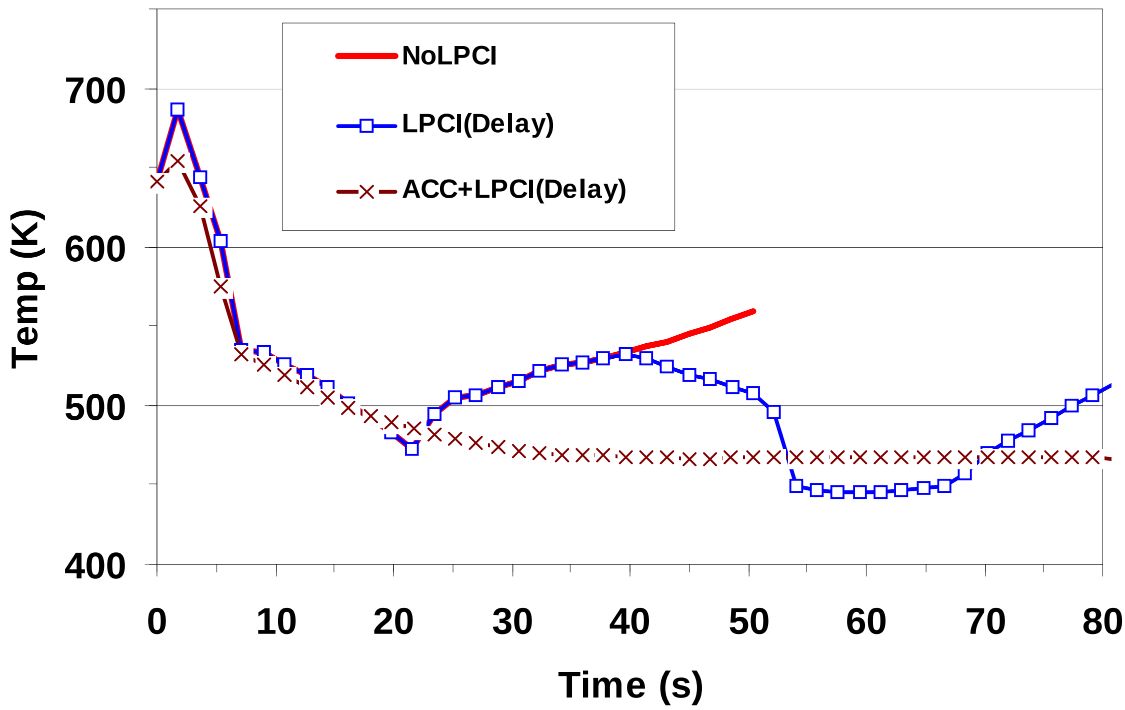

Figure 7-10 Comparison of the predicted maximal cladding temperature at 3.045 m

7.2.4 Large break loss-of-coolant accident: steam line versus feedwater line break

According to [Lee98a] the feedwater line break is the most severe one for the reference reactor. Hence a large break LOCA in the feedwater line is analyzed and compared with the corresponding steam line break (LB-LOCA) results. The feedwater line break size amounts 0.11 m^2. The predicted break outflows are shown in Figure 7-11. Even though the first peak of the steam flow through the steam line break is larger than the one of the feedwater line break, the latter remains higher compared to steam outflow during the first 6.5 s. Since the rather cold water (high density water) located in the downcomer and in lower plenum leaves the RPV through the break, the core cooling condition during a feedwater break are worse compared to that of the steam line break. Consequently a pronounced core heatup is predicted by RELAP5 which is in good agreement with results obtained by other codes [Lee98a].

Although these results were obtained with RELAP5 for the “artificial steady state”, they are in acceptable agreement with the trends and order of magnitude predicted by the SCRELA-code, see figure 7 and 8 in [Lee98a]. The feedwater LB-LOCA-outflow peak predicted by RELAP5 is slightly higher than the peak of about 10000 kg/s predicted by the SCRELA-code [Lee98a].

In case of the hot LB-LOCA, the break flow curves predicted by RELAP5 are in qualitative good agreement with the time history predicted by SCRELA. The first peak of the break outflow can only be plotted if a sufficient small time interval for plotting is defined. Otherwise this first peaks may not be seen in the calculations. Later on in the calculation, RELAP5 predicts a lower break outflow (about 3800 kg/s) than the SCRELA-code (about 4500 kg/s). This is also valid for the system pressure curves predicted by RELAP5 as can be seen in Figure 7-12. Initially both pressure trends are similar. With transient progression the pressure reduction is slower for the feedwater line break than for the steam line break, since the break size is smaller (diameter of the feedwater and steam line is different). A comparison of the cladding temperature trend at the same elevation (3.045 m) predicted for the feedwater and steam line break without actuation of the LPCI is given in Figure 7-13. In addition, the maximum cladding temperatures predicted for the steam line and feedwater line break is depicted in Figure 7-14. It can be seen that for the steam line break, the maximum is located at 3.045m while for the feedwater line break the maximum temperature is located at 1.995m elevation. These temperature trends reconfirm qualitatively the results obtained in [Lee98a], where it was demonstrated that the feedwater line large break LOCA is the most severe one from the safety point of view.

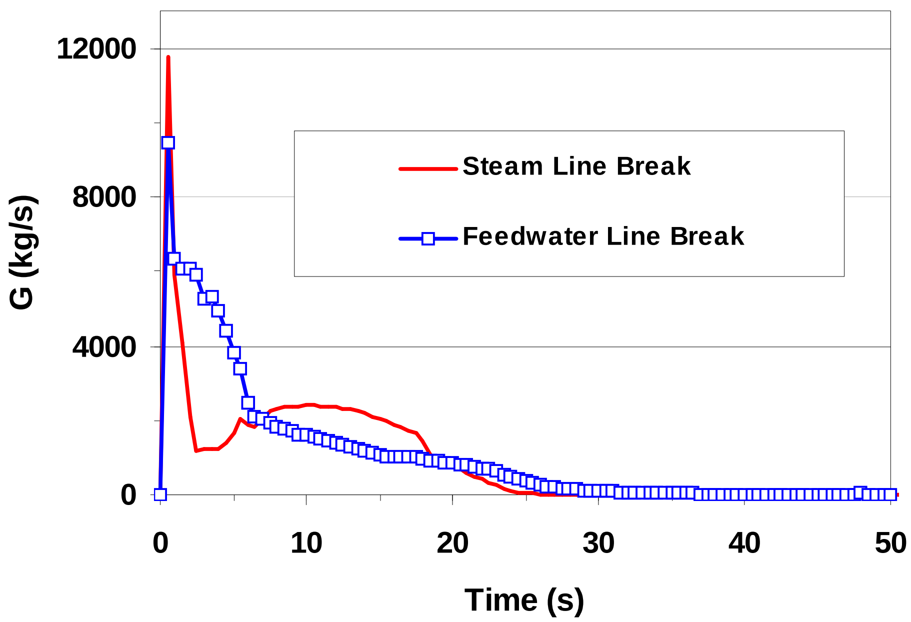

Figure 7-11 Predicted break outflow rate for steam and feedwater line break

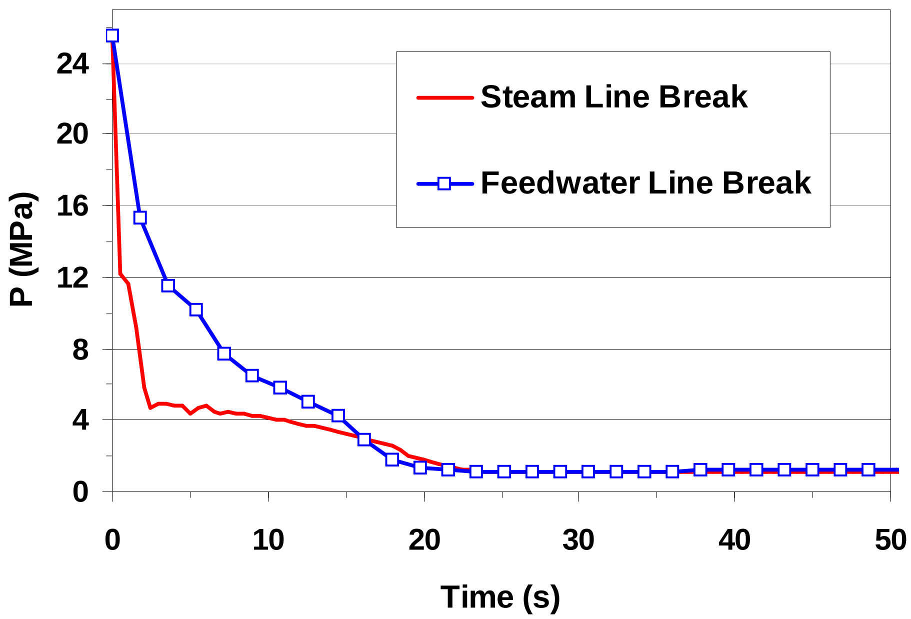

Figure 7-12 Predicted system pressure for steam line and feedwater line break

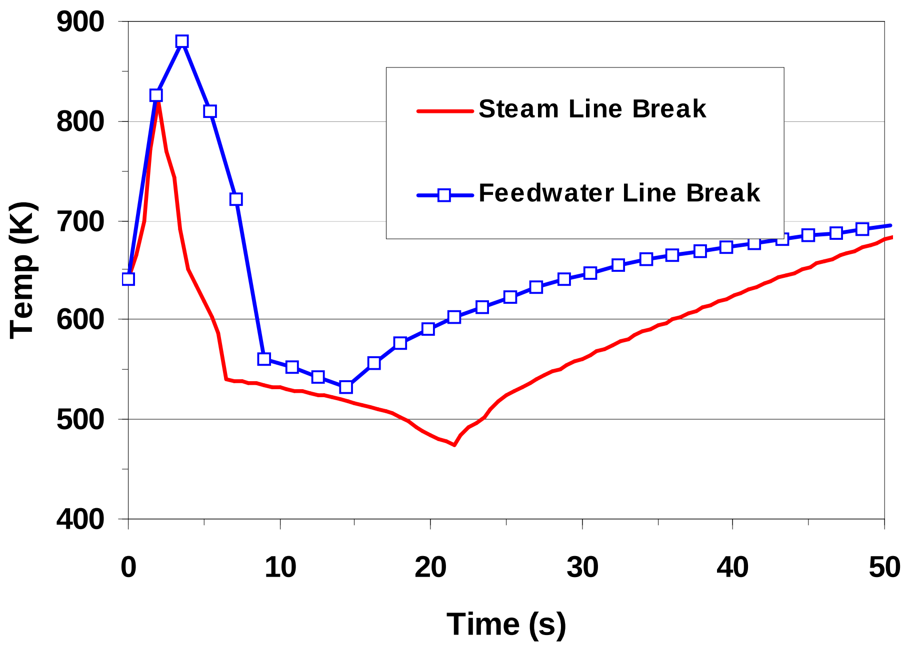

Figure 7-13 Predicted cladding temperature at 3.045 m height for the feedwater and steam line break

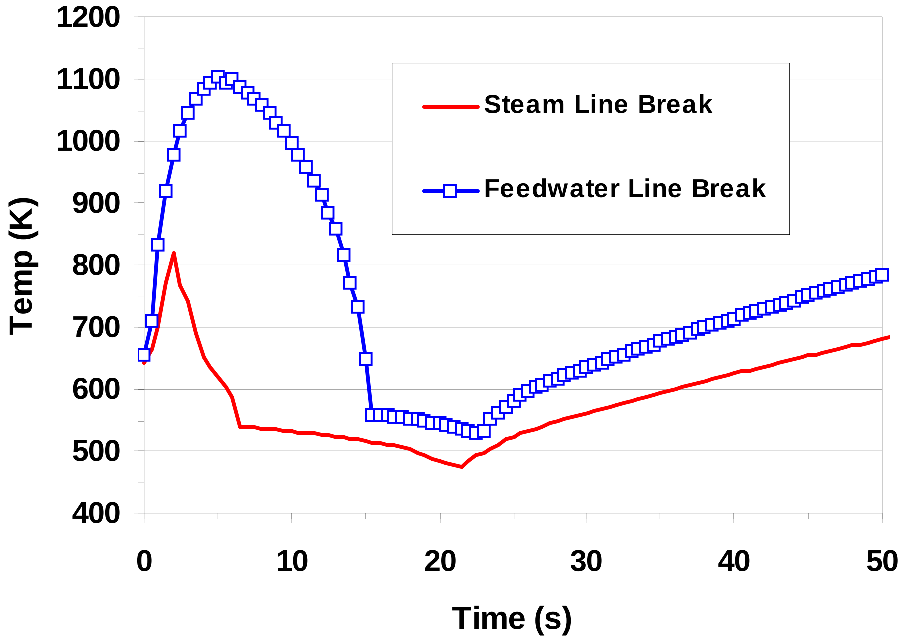

Figure 7-14 Predicted maximal cladding temperature for steam line (3.045 m) and feedwater line (1.995 m)

7.2.5 Parameter studies for feedwater line break

The effectiveness of the emergency core cooling system is determined by several parameters and assumptions e.g. injection rate, location of injection, temperature of injected water, availability and combination of ECCS. To investigate the influence of such parameters on the cooling phase after a large break in the feedwater line of the reference plant, the following cases were investigated in addition to the pure blow-down without activation of any ECC-systems:

- LPCI-injection directly into the down-comer with an injection rate of 2000 kg/s.
- LPCI-injection directly into the hot box with an injection rate of 2000 kg/s.
- Actuation of accumulators (ACC) and LPCI-injection that inject emergency water into the hot box assuming an injection rate of 2000 kg/s and 1600 kg/s, respectively.

In Figure 7-15 the outflow rates through the break predicted by RELAP5 are compared. As can be seen in Figure 7-16, almost all fluid inventory of the reactor pressure vessel is dumped out through the break in a short period of time i.e. about 30 s. The LPCI-signal is activated when the system pressure reduces below 4 MPa. It starts to inject water with 25 s delay.

In Figure 7-17, the predicted maximum cladding temperature for these cases is given. It is obvious that the no activation of ECCS might lead to a core overheating (Case NO LPCI). If the LPCI-injection into the down-comer is considered, the maximum cladding temperature continues increasing despite the injection of 2000 kg/s, since most of the injected water does not pass through the core. On the contrary a major part of this water leaves the RPV since the break is located close to the down-comer.

If the LPCI-system injects the same amount of water directly into the hot box instead of into the down-comer, the picture changes completely. In this case, the injected water is forced to pass through the core before leaving through the break. Consequently core cooling is more efficient and the cladding temperature increase can be considerable reduced to an acceptable level, see Figure 7-17.

Since the temperature increase during the first 10 seconds is considerable, the activation of accumulators, in addition to the LPCI, is considered. Both ACC and LPCI feed the water into the hot box. In Figure 7-19 the time and injection rate of the ACC and LPCI is given. In this case, it is possible to stop the initial temperature rise very early in the transient and hence the core seems to be cooled to safely conditions, Figure 7-17.

These investigations, although very preliminary, underline the importance of systematic investigations related to design optimization of the safety systems.

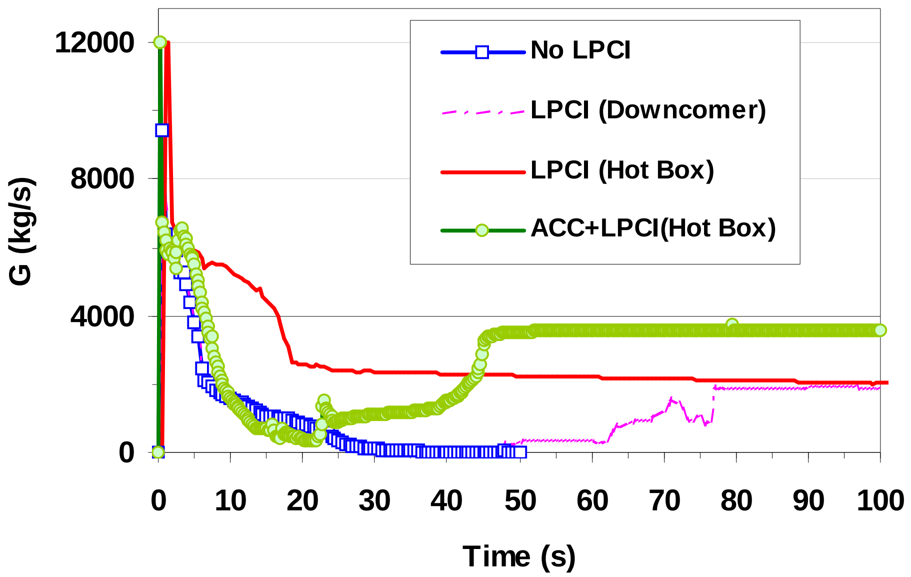

Figure 7-15 Predicted break outflow for different option of LPCI-injection

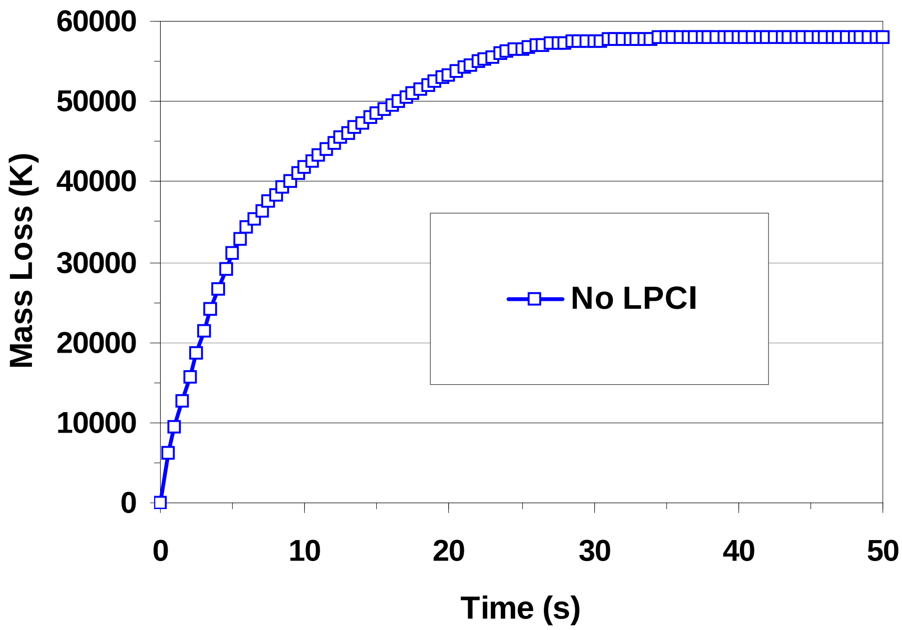

Figure 7-16 Predicted integrated fluid inventory that leaves the reactor pressure vessel

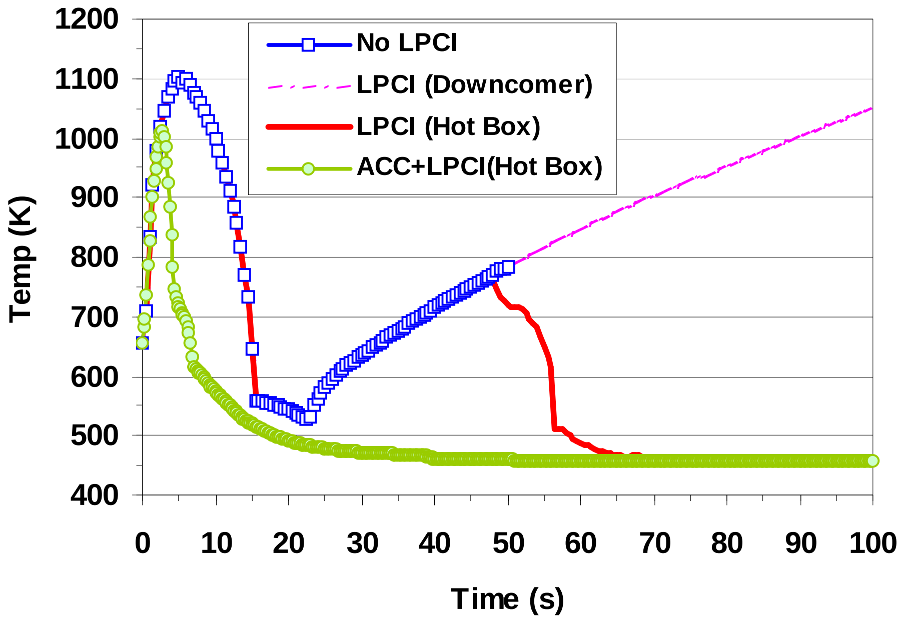

Figure 7-17 Predicted maximal cladding temperature for the studied cases

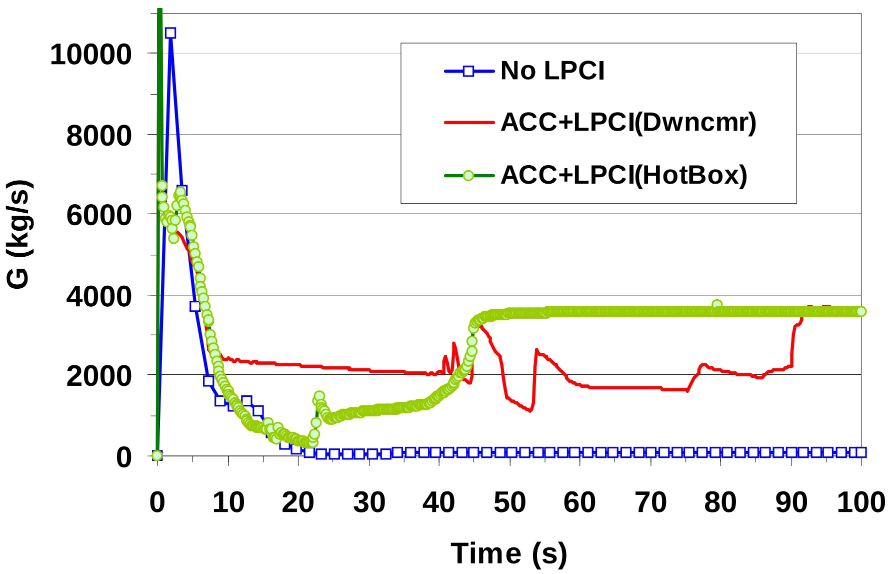

Figure 7-18 Predicted break outflow rate for studied cases

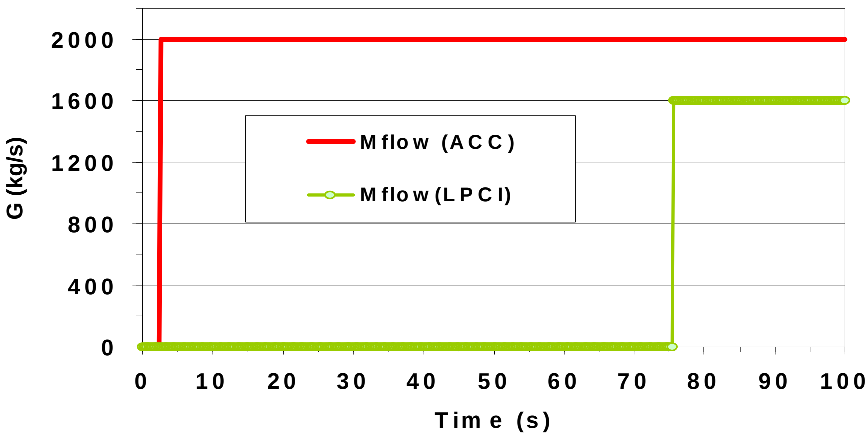

Figure 7-19 Predicted time for injection of ACC and LPCI

Finally, since the scenario B (artificially steady-state) is characterized by a high fluid inventory in the in the hot box compared to that of the scenario A (see Tab. 4-2), the volumes of the hot box were reduced in the RELAP5-model so that the fluid inventory is approximately 1.4 tons. Then, the following transients were rerun with the modified model:

- Blowdown without actuation of any safety ECC-system, and

- Blowdown phase with actuation of LPCI-system (2000 kg/s injection at P= 4MPa).

The predicted maximal cladding temperature for these scenarios are compared with the corresponding ones, where the fluid inventory in the hot box amounts about 12.3 tons

The fluid inventory within the hot box largely affects the core cooling during the Blowdown phase in case of the large feedwater line break, see Figure 7-20. If a small amount of water is available in the hot box (as is the case for the normal steady state of the reference plant) the core heat up may not be prevented without ECC-systems. The assumption of about 1.4 tons fluid inventory within the hot box is very close to the conditions of the reference plant.

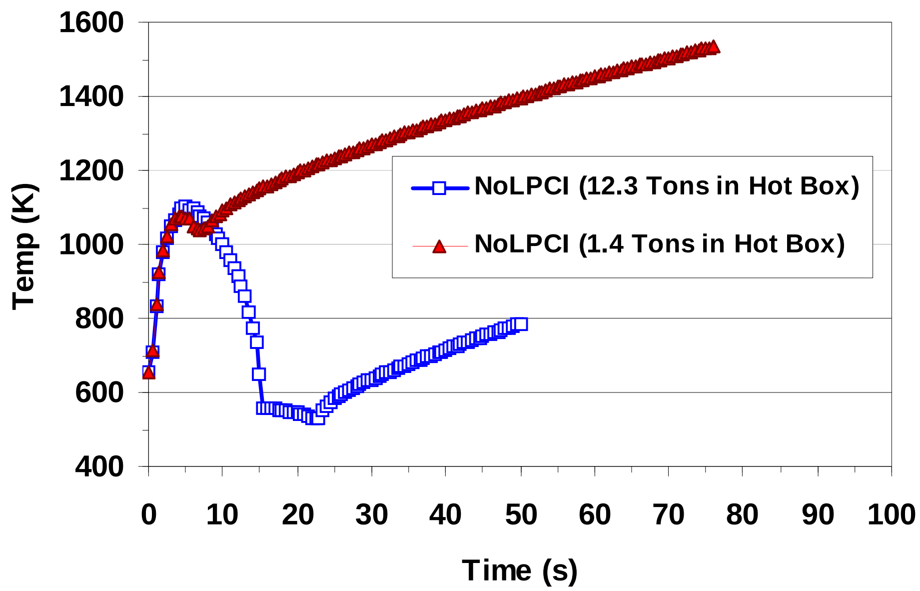

Figure 7-20 Predicted maximal cladding temperature for different fluid inventory in the hot box

Furthermore, the actuation of the LPCI-system which injects 2000 kg/s coolant into the hot box was considered in this exercise. In Figure 7-21, a comparison of the predicted cladding temperature for the two cases, one with only 1.4 tons and the other with 12.3 tons fluid inventory within the hot box, is shown. It can be noted that the LPCI-system is very effective to successfully remove the decay heat from the core. The cladding temperature increase was stopped within the first 60 s.

It must be point out that there are several possibilities to assure the core cooling during LB-LOCA accidents of the reference plant by appropriate design. Hence further studies are necessary to come up with an optimized ECC-system configuration for the final plant design.

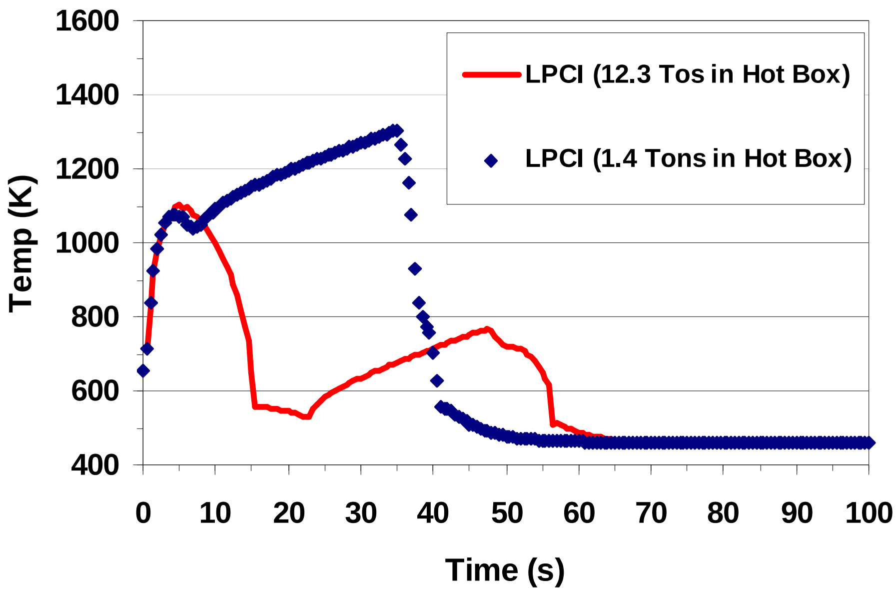

Figure 7-21 Predicted maximal cladding temperature for different fluid inventory in the hot box

8 Summary and Conclusions

Within the 5th EU-Framework Program several European institutions and the Tokyo University investigated several aspects of the proposed reference within the High Performance Light Water Reactor (HPLWR) Project.

The main goal of this project was to study the technological and economical merits of the high-temperature supercritical light water reactor (SCLWR-H) -one of several reactor concepts developed by the Tokyo University- taking into account both European and Generation IV safety requirements.

In this report the investigations performed to assess the appropriateness of the best-estimate code system RELAP5 for the safety-related evaluations of the reference plant design is described.

Based on the investigations presented here the following conclusions can be drawn:

- The old (ASME-IFC-67) and new (IAWSP-95) steam tables implemented in RELAP5 cover the whole range of normal operation and of postulated transients and accidents. These steam tables allow the accurate prediction of the thermo-physical properties of steam and water that are needed by the RELAP5-code to solve the equation of state of the fluid-dynamic system for the reference design.

- In the frame of the investigations for the HPLWR-project, appropriate heat transfer correlations for supercritical water were identified, but is was agreed that the Dittus-Boelter correlation is sufficient for the purpose of checking the overall prediction capability of RELAP5.

- After code improvements, RELAP5 is capable to predict the nominal operational conditions of the reference reactor with acceptable low mass error and reasonable time steps. This was achieved by slightly modification of the interpolation scheme and by the redefinition of the void fraction and quality in the supercritical region (version MOD3.2.2φ). These improvements were introduced in all subsequent code versions.

- Based on the good RELAP5-results for the steady state parameters of the reference plant, RELAP5 was used to assess different moderator concepts. These studies revealed different options how to improve the core moderation characteristics of the reference plant e.g. by use of solid moderator and isolation of the water tubes in the hot box region.

- The investigations of selected transients without de-pressurization e. g. loss of offsite power, reduction of the coolant inlet temperature, feed water flow reduction, etc.. have shown that RELAP5 with the point kinetics option is in principle capable of qualitatively predict the plant behavior under transient conditions. The obtained results are similar to the analysis performed by the University of Tokyo.

- The transient evaluations with a rather coarse core model, however, also indicate that many of the core parameters strongly vary along the core height e.g. coolant density, fuel temperature, etc.. In addition, during the course of most transients, there are axial core regions with pronounced different thermal behavior. Hence the applicability of a point kinetics approach may not be sufficient to analyze the reference plant design. The analysis of postulated loss-of-coolant accidents, where a strong de-pressurization of the primary system occurs, has demonstrated the limitations of RELAP5. Due to its interpolation scheme implemented for the calculation of the thermo-physical properties in both sub- and supercritical region, the code has serious difficulties to calculate the supercritical water properties around the critical point when the coolant temperature decreases below the critical temperature. Due to the sharp changes of the water properties at the critical point, some intermediate variables needed in RELAP5 go to infinity and hence the code fails. The present interpolation scheme of RELAP5 has to be revised and accordingly modified to guarantee a reliable crossover of the critical point in both upward and downward directions.

- Despite these limitations, exploratory investigations of different LOCA-scenarios were performed with RELAP5 starting from an “artificial steady state”. These investigations have shown the peculiar behavior of the reference plant under LOCA-conditions. Following very preliminary conclusions based on the qualitative trends predicted by RELAP5 can be drawn:

    - The small RPV-fluid inventory of about 60 tons is dumped out through the break within 30-40 s with the consequence that the core overheating may not be prevented if no appropriate ECC-systems are activated.

    - It was confirmed that the feedwater line break is worse compared to the steam line break for the reference plant.

    - In case of the feedwater line break, if a realistic fluid inventory in the hot box is considered, the cladding temperature increase is considerable even though a conservative axial cladding temperature distribution which is lower than that of the steady state of the reference plant. But the actuation of the LPCI-system is able to prevent the core heat up also for the case of a low fluid inventory within the hot box.

    - The injection of emergency water from the LPCI-system into the downcomer in case of a feedwater line break appears to be inefficient in terms of decay heat removal. It may be better to inject cold water into the hot box.

    - The core overheating of the reference plant may only be prevented if the emergency water is injected directly into the hot box region.

    - It was noted that during the first 10 seconds of a feedwater line break a very strong cladding temperature rise takes place. This trend of the cladding temperature may be prevented if the actuation of passive emergency core cooling systems e.g. accumulators is considered.

- All RELAP5-simulations are preliminary in nature and mainly focused on the assessment of the code's capability rather than in quantifying the plant response under transient conditions.

- A thorough evaluation of the HPLWR-safety features will be possible after the implementation of the relevant physical models in RELAP5-code and the improvements of the code's numerics regarding the thermo-physical properties calculation.

- Finally, these numerical exercises have demonstrated how RELAP5 can be applied to optimize the ECCS-systems regarding e.g. mass flow rate, location of injection, pressure level for activation of ECC-systems, etc.. Such work is highly recommended to be performed with RELAP5 after its improvement for any plant and safety system design proposal.

In general it can be concluded that the code RELAP5 has the potential to be used as a reliable safety analysis tool for the assessment of the reference plant. However additional code improvement and qualification are necessary to fully cover the analyses of postulated transients and accidents and to qualify the heat transfer correlations used in the code.

9 Future work

The reference plant (SCLWR-H) was evaluated with the RELAP5 code system. Although the investigations are preliminary and exploratory in nature for the code's capabilities and the reactor design, important insights on the plant behavior under abnormal and accidental situations were gained.

After the careful review of major technical, safety and economical features of the reference plant, essential improvements of the plant design and safety system were proposed in the different work groups of the HPLWR-project to meet the economic and safety requirements from the European point of view. On the other hand, it became evident, that several challenging fundamental issues -peculiar to the SCLWR-H- must still be experimentally and theoretically studied in detail within a technological research program.

In this context, the research activities related to the qualification and validation of thermal hydraulic and safety analysis tools that may be used to evaluate the safety features of the high performance light water reactor (HPLWR) may be concentrated on the following areas:

1. Fundamental heat transfer mechanisms between the wall and supercritical water under normal and accidental conditions

2. Further validation of available correlations for supercritical water against new experimental data

3. Numerics of system codes to predict supercritical water thermo-physical properties under LOCA-conditions

4. Optimization of different moderator rod concepts

5. Supporting activities to optimize the safety systems design

6. Coupling of thermal hydraulic codes with neutronic design tools (KARBUS/RELAP5) and subsequent qualification

7. Safety analysis of the final plant design with best-estimate codes using point kinetics and/or best-estimate codes coupled with one dimensional or three-dimensional kinetics e.g. RELAP5/PARCS.

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