EVALUATION OF A PRELIMINARY SAFETY CONCEPT FOR THE HPLWR

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Abstract

The main safety functions considered in the preliminary concept for the HPLWR have been evaluated by means of a comprehensive set of analyses, which have been performed using system and coupled codes. The investigated scenarios addressed a variety of initiating events, including anticipated transients as well as accidents. The simulations performed show that for each class of transients at least one of the computational tools used in this project is adequate for preliminary assessment of the safety concept of the HPLWR. The analyses have shown that the proposed systems can be expected to be capable to provide all the safety functions. The open issues that remain to be addressed in future projects are also discussed.

1. Introduction

The High Performance Light Water Reactor (HPLWR) is the European contribution to the supercritical water-cooled reactor concept. The results presented here summarise the main findings of the safety analyses performed within the project HPLWR 2 [1] of the Sixth European Research Framework Programme. The main characteristics of the current design have been presented in various publications (e.g. [1-3]), and will not be reported here. The specific three-pass core concept, however, is illustrated in Fig. 1, as this is a design feature that has a major effect on the system response and therefore on the safety analysis. The concept includes a thermal core in which supercritical water is heated up in three steps (evaporator, superheater 1 and superheater 2) with intermediate coolant mixing to minimize peak cladding temperatures of the fuel rods. The general safety concept, basic requirements and safety goals have been established, and a preliminary design for the safety systems has been proposed [4]. In this paper, the current configuration and the main safety functions will be briefly discussed. The main focus of the work is on the results of the safety analyses, which provide a first evaluation of the capability of the systems to provide the required safety functions.

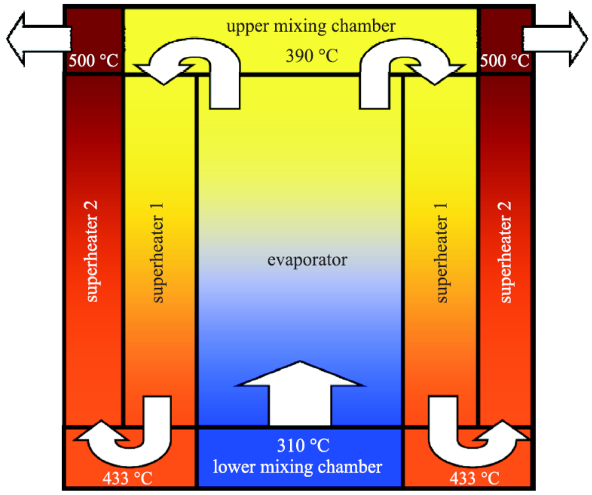

Figure 1 Three pass design concept of HPLWR core [3].

2. General safety systems configuration

In case of most of the transients as well as in the event of accidents, the following safety functions must be assured:

- Reactor scram
- Containment isolation
- RPV pressure relief and depressurization
- Heat removal from the RPV
- Reactor water makeup and control of core coolant inventory
- Heat removal from the containment

The general HPLWR safety systems configuration schematic is shown in Fig. 2. It includes:

- Safety Relief valves (SRV), which can be actuated to provide an automatic Depressurisation System (ADS)
- Residual Heat Removal (RHR) and Low Pressure Coolant Injection (LPCI)
- Containment condensers

The specific characteristics of the current HPLWR design which influence the safety of the reactor with respect to core cooling have been compared with those of typical PWR and BWR designs, and the issues related to the differences that have been identified gave important hints for design measures [4]. The specific characteristics to be considered for safety concept development are as follows:

- Water mass in the primary circuit and associated heat storage capacity
- Cooling of the core in the case of loss-of-flow conditions
- Heat capacity of the core
- Heat transfer mechanisms within the core after shutdown under specific conditions
- Challenges stemming from the supercritical single flow regime to subcritical two-phase flow conditions in case of LOCA

The safety analyses presented here will especially address the issues above.

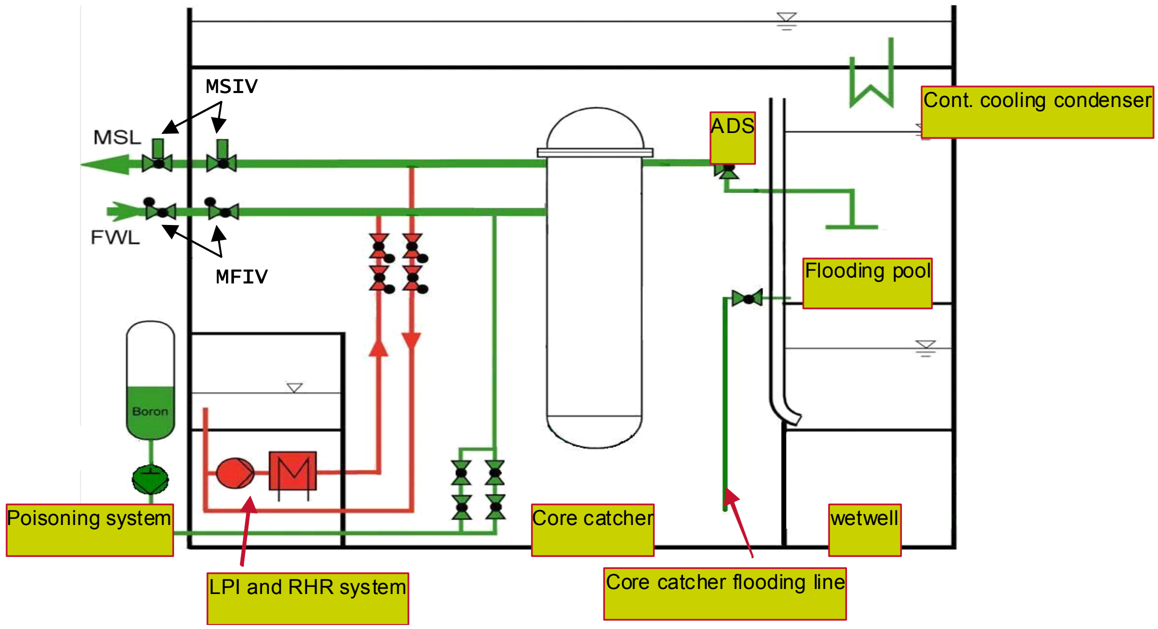

Figure 2 General configuration of the safety systems.

2.1 Analyses performed to dimension the safety systems

A preliminary dimensioning of the safety systems was performed using the code APROS developed by VTT Technical Research Centre of Finland and Fortum. The HPLWR whole steam cycle was modelled and analysed for part load, shut-down and start-up conditions [5]. The simulation of complete containment isolation with following depressurization through the ADS system showed that the reactor can be cooled efficiently. The ADS actuation pressure, the ADS valves flow area, the ADS valves driving time and the Main Feedwater Isolation Valves (MFIV) and Main Steam Isolation Valves (MSIV) driving time were varied. These studies indicated that an ADS actuation pressure of 26 MPa, a flow area of 0.09 m^2 and a driving time of 0.2 s are an optimal set of parameters. The simulation of the LPCI system showed no principle drawbacks. The reactor can be cooled efficiently after the system is depressurized through the spargers. After the LPCI system injects water at 6 MPa, the cladding temperature starts to rise again, since the reactor has to be filled with water. The simulation with a pump with 400 kg/s nominal mass flow rate prevents almost the rising of the cladding temperature, since the core was flooded. With this initial choice of systems and parameters, a preliminary safety assessment was conducted, which is presented in Section 3.

3. Safety Analyses

At the beginning of the project, the events to be analyzed have been classified according to the current practice into different categories and to these categories different acceptance criteria are applied [4]. The full list of events (104 events) to be considered for the HPLWR safety analyses has been provided at the beginning of the project by AREVA. The first analyses were intended to cover some events which would enable an assessment of the feasibility of the HPLWR. The range of events considered included Design Basis Conditions

(DBC) ranging from operational conditions (DBC 1) to very severe accidents (DBC 4). The events analysed can be subdivided in two groups: 1) transients controlled by the thermal-hydraulic response; 2) RIA's and ATWS. Since the results of the analyses of the second group of events have already been presented [6-8], the following sections will mainly present an overview of the results for the transients controlled by the thermal-hydraulic response. All analyses will be considered for drawing preliminary conclusions on the feasibility of the HPLWR.

3.1 Codes used for the analyses

Various codes have been used, which have been upgraded to the HPLWR conditions [6-14]. These are: RELAP5, CATHARE, APROS, KIKO3D-ATHLET and SMABRE/TRAB-3D. Some of the codes (though not all) have been proved to be capable to run through the critical point, and therefore to simulate depressurization transients. These codes were used to study LOCAs and transients leading to the intervention of the ADS system. Although within this work no code could be demonstrated to be capable to simulate all transients (including LOCAs and RIAs), the simulations performed show that for each class of transients at least one of the computational tools used in this project is adequate for preliminary safety analyses of the HPLWR. In absence of an adequate database, a certain confidence in the results could be gained from code-to-code comparison [15].

3.2 Methodology used for the safety analyses

One of the most important issues in the evaluation of the safety of a reactor is the maximum cladding temperature, which is reached in the hottest rod. Since the safety analyses were conducted in parallel with the development of the core design, which aimed, among others, at the reduction of the temperatures in the hot channel and of the cladding of the hottest pin, a methodology was selected to decouple the safety analysis from the optimisation of the design. Therefore, in the analyses of transients controlled by the thermal-hydraulic response a rather large value of the gap conductance was selected to satisfy the acceptability limits set to the fuel temperature under the reference operational conditions (steady-state at 100 % load). This permitted to study the response of the system to the various events starting from an acceptable initial state for the rod centreline temperature. A similar approach was also used in the coupled 3-D analysis of RIAs and ATWS, where the hot pin factors were adjusted on a case-by-case basis to match the temperature limits. In fact, the use of the same hot pin factors for all starting conditions would result in fuel centreline temperatures above the melting point. The analyses should be repeated for an optimized design.

3.3 Analyses of transients controlled by the thermal-hydraulic response

The safety analyses are fully reported in the project documents [15]. In this work, only a few results will be shown, selecting the analyses that especially address the impact of the special HPLWR characteristics and the three-pass core design on the safety of the reactor. The specific characteristics to be considered for safety concept development have been outlined in Section 2, and will be addressed separately in the following sections. The plant models used by the various codes for safety analyses are described in [15]. Here, only the main features of the models will be recalled, when these are needed to discuss the possible limitations of the analyses and for comparing results. A few sensitivity studies have been performed with respect to closure laws (e.g., [9]). The work, however, was focused on the understanding of the general response of the system for various transient scenarios, and therefore most studies were conducted using the models and correlations selected for supercritical water conditions. In consideration of this main objective, some simulations performed at different times during the project were not iteratively repeated using information obtained from simulations with other codes. Therefore, for some transients, simulations with different codes did not use the same approach with respect to certain model features (e.g., power history before scram, method to account for hot channel/hot pin factors).

3.3.1 Transients addressing heat storage capacity

Heat storage capacity within the coolant is considerably less compared to PWR and BWR, which is an indication for potential of faster pressure transients. In order to address this issue, the very severe turbine trip was investigated, to check if ADS could limit the pressure below design pressure. Two analyses were performed with two codes, using different models and assumptions:

- RELAP5: point kinetics was used. No scram was assumed, and the calculation showed that power was reduced to a new equilibrium value due to the negative reactivity effect. Since the code cannot simulate depressurisation below the critical pressure, the actuation of the ADS system was not simulated. Instead, the safety valves opened on high-pressure signal and cycled around the opening set point (26 MPa). Three different cases were investigated, using different assumptions on MSIV valve and SRV opening/closing times.
- SMABRE/TRAB-3D: coupled 3-D Thermal-Hydraulics and neutronics was simulated. With this code, also the effect of the intervention of the ADS could be studied. The ADS system was simulated to open when the pressure reaches 26 MPa; once opened, the valves remain open and the system depressurizes. Due to the numerical problems of SMABRE for fast depressurization transients, only the first few seconds could be simulated. The effect of the total valve flow area has also been studied.

Figure 3 shows the pressure calculated with the two codes. In both cases, the simulations show that a total flow area of 0.09 m^2 is sufficient to maintain the pressure below its design limit (28.75 MPa).

3.3.2 Transients addressing core cooling in case of loss-of-flow

The three pass core configuration does not permit any of the cooling modes of LWR. The consequence is that in all cases water has to be immediately supplied into the primary circuit to provide core cooling. Therefore, a severe test for the safety of the core design concept is provided by the analysis of loss-of-flow events. A variety of studies have been performed with various codes for sequences initiated by the failure of one or more pumps.

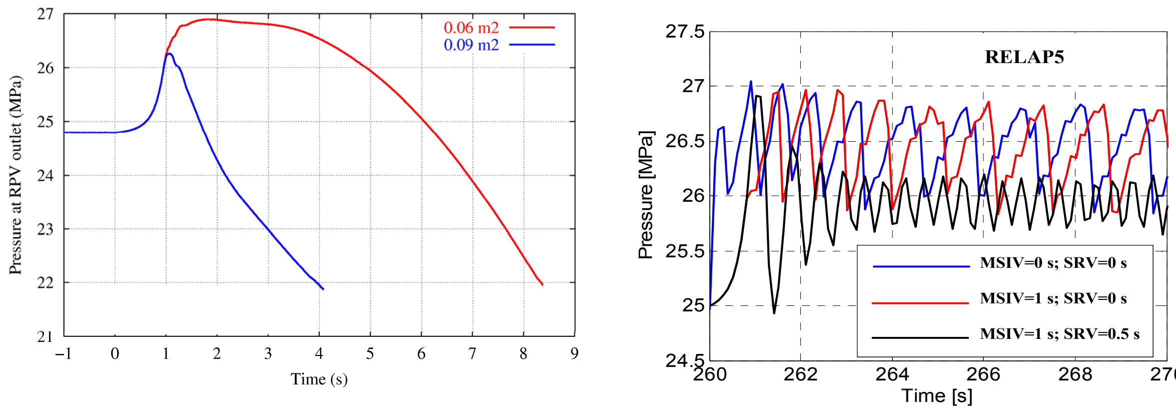

Figure 3 Pressure transient following a turbine trip calculated with the SMABRE/TRAB-3D (left) and RELAP5 (right) codes.

These simulations included the investigation of the effect of pump run-down times and of the parameters characterizing the intervention of the stand-by pump (delay and time for its actuation). In this paper, only two representative studies are recalled to illustrate the most important conclusions of those analyses.

For the first study, a partial LOFW, two of three pumps are assumed to fail, and the stand-by pump is assumed to intervene within a very short time. For this case, it was assumed that the failed pumps flow rate would run to zero in 10 s. This transient has been analysed with CATHARE, RELAP5 and SMABRE. The main features of the models are listed below:

- SMABRE: the core is represented with one channel (average) for each of the three core sections (evaporator, superheater 1 and superheater 2). Each fuel channel is connected to a moderator and a gap channel. The power distribution among the three regions of the core and its axial distribution are considered.
- RELAP5: similar representation as with SMABRE, but the peak power in the hot pin is represented by modelling a second fuel rod in the average hydraulic channel. The hot pin factors were also considered.
- CATHARE: the average assembly and the hot assembly are represented separately, resulting in the most detailed core model. Hot pin factors were not considered in this analysis.

For these studies, the assumption was made that the pressure remains constant. Representative results for the peak cladding temperature are shown in Fig. 4, where the calculation with the RELAP5 and CATHARE codes are shown. The cladding temperature excursion is rather small (around 50 K), due to the fast reduction in thermal power caused by the negative reactivity coefficient. These results have to be taken with some caution because the use of point kinetics could be inadequate for this transient [15], because the change in the flow distribution and even flow reversal in the gap channels results in variations of the moderator density distribution and therefore in modifications of the power distribution, both radial and axial. Nevertheless, the results show that the loss-of-flow event results in much milder transients than initially anticipated.

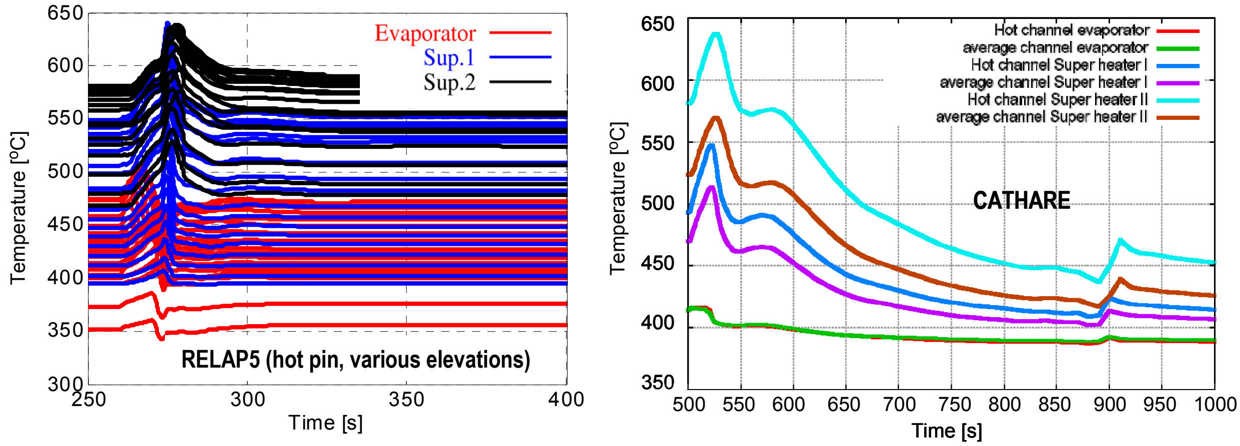

Figure 4 Peak cladding temperature calculated with the RELAP5 and CATHARE codes for a LOFW event, with failure of two pumps and delayed start-up of one stand-by pump.

In the second study, a total LOFW, the transient was investigated under different conditions with three codes:

- APROS: a model with simplified representation of the core (but which considers the hot channel) and a complete representation of the steam cycle was used. The pressure evolved according to the response of the system, which included actuation of the ADS system and the associated fast depressurization [5].
- SMABRE: a detailed model of the core was used, and a pressure boundary condition was prescribed at the vessel inlet. The total feedwater flow rate was ramped to zero within 10 s.
- RELAP5: the model had similar features as that which are used for SMABRE, the only important difference being that the hot pin was also represented.

For the simulation with APROS (which does not model reactivity feedback), a reasonable power time history after scram was imposed (details in [5]). Calculations with SMABRE and RELAP5, however, assumed that scram would not occur (the transient being therefore a very severe ATWS).

Moreover, the way to account for the hot pins in the APROS model is different from the method used in the simulation with RELAP. In fact, in the APROS model a hot channel is represented, where the fluid enthalpy rise is twice as high as that in the average channel. In the RELAP model, however, the hot pin is attached to the average hydraulic channel, but has a higher power than the average rod, the hot pin factor being taken from the results of subchannel analyses [15]. Due to the various differences in the models and the details of the transient represented, only the first few seconds until the cladding temperature peak could be compared to some extent.

The main findings of the analyses were:

- The calculated hot pin temperature calculated by RELAP is much lower than that calculated with APROS, although the thermal power decreased only as an effect of the negative reactivity coefficient.
- This peak temperature calculated by RELAP is lower than the average value calculated with SMABRE, although the same transient without scram was simulated. The differences in the results are related to a different time history of the thermal power [15].
- Cladding temperatures are below 1000 ^{o}C

Although, for the total LOFW event without scram (DBC 4 event) one could assume that the acceptable limit for cladding temperature would be 1200 ^{o}C, in consideration of the different results obtained with the different codes, the questionable application of point kinetics and the uncertainties in the heat transfer coefficient, the simulations performed within this project cannot exclude the risk that this limit could be exceeded. A more comprehensive study of this event using coupled 3-D codes is therefore recommended for a future phase of the project.

3.3.3 Transients addressing core heat capacity

The heat capacity of the core related to core thermal power is comparable to that for LWR. The consequence is that the challenges to be expected with high heat fluxes after scram are comparable to that of PWR and BWR plants. This was verified by investigating the event of an anticipated scram. For this simulation, the CATHARE code was used, which adopts a model which considers average and hot channel, the hot pin factor being taken from the results of subchannel analyses. The sequence is initiated by the SCRAM signal. When the control rods are fully inserted in the core 3.5 s later, the MSI valves are closed and the (turbine bypass) valves to the condenser are opened in 1 s. The feed-water temperature ramps down to 155 °C in 60 seconds after the closure of MSIV valves. The main result of the simulations is that no fuel cladding temperature excursion is encountered if the feed-water flow rate remains greater than 20 % of the nominal value. This transient response confirms that in relation to issues related the core heat capacity the HPLWR design is not different from other LWRs.

3.3.4 Transients addressing loss-of-offsite power

The heat transport capacity in case of loss of off-site power is influenced by the inertia of the pumps. Feedwater-pump-motor system of HPLWR is therefore to be decided. Various parametric studies of events initiated by pump(s) trip have been conducted, with the main goal to determine whether the run-down time (inertia) of the pump was critical for the cooling and measures should be taken to reduce the rate of flow reduction (e.g., by adding a fly-wheel). These analyses, mainly performed with the CATHARE code, are fully reported in [15]. In this paper, only the main results are summarised below. The transient considered is one initiated by the failure of one or two pumps, assuming that the pressure remains constant and the thermal power is only regulated by the neutronic feedback. The stand-by pump is assumed to intervene after a short delay time. Transients with run-down times between 3 and

20 s were simulated, assuming linear flow reduction. These analyses showed that the peak cladding temperature was somewhat affected by this parameter, but the relatively small difference in the results (about 20 K for the extreme cases) would not justify any measure to increase the pump inertia. Moreover, it should also be considered that inertia of the pumps would play a role only during anticipated transients of minor severity. In the case of an accident, such as a complete LOFW, the MFIV would be closed immediately after the closure of the MSIV, so that the feedwater flow ramp to zero is controlled by the valve closing time rather than by the pump inertia.

3.3.5 Transients addressing coolability of three-pass core in case of LOCA

In case of LOCA, special challenges stem from the transition from the supercritical single flow regime to subcritical two-phase flow conditions, and from the geometry (three-pass core). In fact, due to the strong density ratio between water and supercritical fluid, the core would empty very quickly in case of depressurisation and loss of mass through a break, and the complex geometry could hinder or delay the refilling of the entire core. In order to prove that the core could be cooled by means of safety injection systems under all circumstances, transients with breaks in both steam and feedwater lines have been analysed.

Small LOCA with break in one of the main steam lines

The accident investigated would be initiated by a 2.4% break in one of the main steam lines. This transient has been investigated using the SMABRE code [11, 15], which can calculate the transition from supercritical to subcritical conditions, although only for small depressurisation rates. In the simulation, the ADS is not activated and water is injected at high pressure after 25 seconds by means of a diesel-powered auxiliary feedwater pump, capable to deliver the same flow rate as one of the main pumps. Although this system is not part of the current safety concept, this analysis showed what flow rate would be needed to avoid core overheating and keep the fuel in a safe state. It was found that with an injection of 250 kg/s the reactor remains safe during the transient. Whether such an injection at high pressure is available or ADS has to be started is a design issue.

Large LOCA with break in one of the main steam lines

For this accident, it is postulated that a 100% break occurs in one of the main steam lines, just before the MSIV. This type of event is not expected to be very penalising for the HPLWR because the rupture of the steam pipe increases the flow rate inside the core and enhance the heat transfer between the fluid and the fuel cladding. The accident is initiated by the full rupture of one steam line inside containments in 1 ms. LPCI is activated automatically when pressure drops below 6 MPa, and injects water from the pressure suppression pool at a temperature of 40 °C and constant flow rate of 250 kg/s until the end of the transient. This event has been simulated with two codes, CATHARE and APROS, which are able to calculate the transient starting from the initial supercritical fluid conditions, and to run through the critical pressure to subcritical conditions. These codes use different models, and partly different boundary conditions and assumptions:

- CATHARE: the simulation is performed representing the hot channel (and considering the hot pin factors [15]) and considering the neutronic feedback. When steam flow reverses at one of the intact steam lines, the MSI valves are closed in 1 s, the SCRAM signal occurs (the duration of the control rod insertion is 3.5s) with a delay of 0.6 s. When the control rods are fully inserted, it is assumed that the overall reactivity of the core remains negative, even during the refilling phase.
- APROS: only the average channel is represented and no neutronic feedback has been considered [16]. Low pressure signal, sent 0.5 s after the pressure drops below 20.0 MPa, initiates the reactor SCRAM sequence and closure of the MSI valves. Reactor power decreases according to a decay heat curve. The MSIVs close within 4 s from the break. Two cases have been considered, assuming the feedwater pumps running or not.

For all simulations, it has been assumed that containment isolation occurred, because a common mode failure of both internal and external MSI valves can be excluded by appropriate design measures. Just after the rupture of the steam line, the power starts immediately to decrease because of the density reactivity effect. The increase of the pressure gradient inside the core makes the flow rate increasing inside the core, which enhances the heat transfer between the fluid and the fuel cladding. The fuel cladding temperature starts to decrease immediately, and keep on decreasing after the SCRAM signal. After LPCI activation, the simulations with the two codes show only minor differences. In the following, the results with CATHARE will be taken as reference, because CATHARE used the most detailed model for the core, including both average and hot channel. In Figure 5 the essential results obtained with the CATHARE code are shown. The time history of the void fraction in superheater1 (which is the last region to be refilled) shows that the entire core can be refilled within 1200 s with a LPCI injection rate of 250 kg/s. Long periods of low flow and flow reversal are calculated to occur in the hot or average channel and the cladding temperatures reflect these variations. In fact, small increases for long periods of time in the average channel and short duration peaks in the hot channel can be observed in the time history of the maximum cladding temperature in the evaporator and superheater 1 (Fig. 5). The initial phase of the transient, when the fluid was still supercritical, also exhibits mild cladding temperature excursions. In fact, as shown in Fig. 6, a moderate increase was calculated in the evaporator and oscillations of small amplitude were observed in superheater 1. In superheater 2, where the initial cladding temperature was the highest, a monotonic temperature decrease was calculated.

Qualitatively similar results have been obtained with the APROS code, with minor differences between the cases with and without continuing operation of the feedwater pumps [16]. The only important difference is that for the case with pumps running a very high void fraction is still prevailing in the upper region of superheater 2 at the end of the calculation. A parametric study showed, however, that increasing the LPCI flow rate to 400 kg/s would lead to complete refilling of the core within 1000 s. It could be therefore concluded that in case of MSLB the LPCI is capable to maintain the core in a safe status. A flow rate of 250 kg/s can be expected to be sufficient to refill the entire core within a reasonably short time, and 400 kg/s provides a comfortable safety margin.

Large LOCA with break in one of the feedwater lines

This type of event has already been identified as the most penalising in the first phase of the HPLWR project [17] as it induced backflows in the core and quasi-adiabatic core heat up after the automatic closure of the main steam isolation valves. This scenario suggested implementing an automatic depressurisation system to restore the flow circulation inside the core and keep on cooling the fuel cladding during depressurisation. The depressurisation during this accident is very fast, with the system going from supercritical to subcritical range within a very short time. This event has been analysed with the CATHARE code, which can run through the critical pressure. The hot pin factors have also been considered. The accident is initiated by the full rupture of one feed-water line inside the containment in 1 ms. LPCI is activated automatically when pressure drops below 6 MPa, and injects water from the pressure suppression pool at a temperature of 40°C and constant flow rate of 250 kg/s until the end of the transient. Just after the rupture of the water supply line, the power starts immediately to decrease because of the negative density reactivity effect.

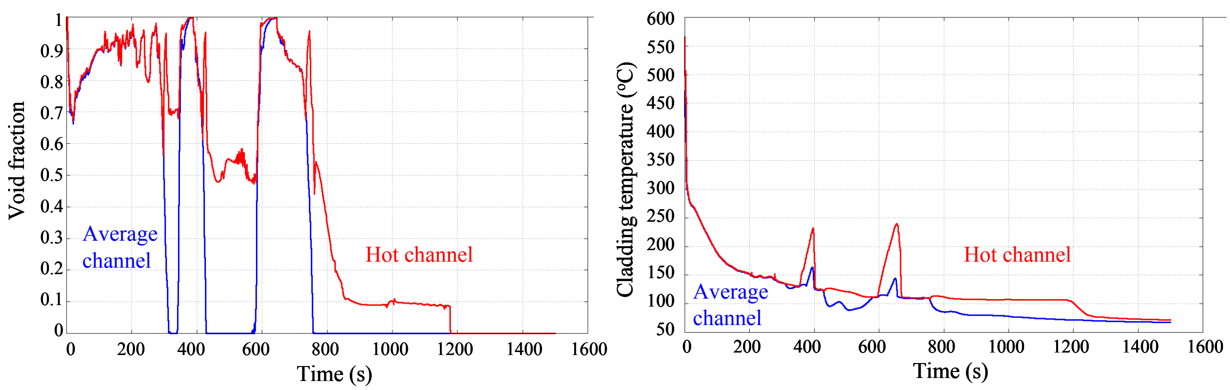

Figure 5 Average void fraction (left) and maximum cladding temperature (right) in superheater 1 calculated by CATHARE for a MSLB.

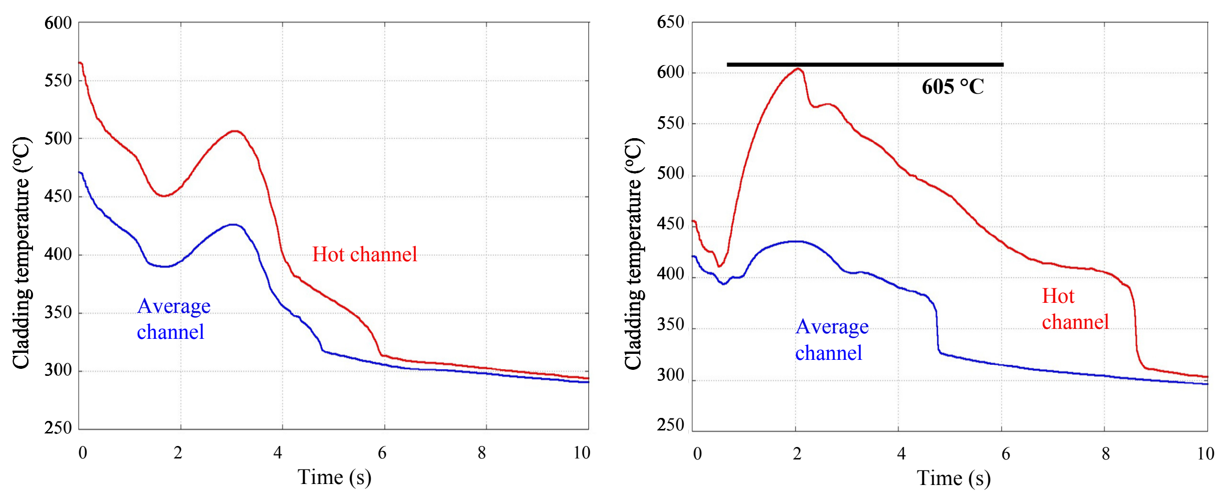

Figure 6 Peak cladding temperature in superheater 1 (left) and in the evaporator (right) during the first seconds of a MSLB calculated by CATHARE.

The period of increasing cladding temperature lasts a few seconds until the control rods are fully inserted because of the SCRAM signal and the ADS valves are fully opened. Then the flow rate inside the core becomes sufficient to cool again the core and the cladding temperatures remains very low whilst the depressurization process is going on. After LPCI activation, when pressure gets very low, a flow reversal in the hot or average channel may occur and the cladding temperatures increase a little for a short period of time. At the end of the simulation, the core refilling is complete. Figure 7 shows that superheater 1 (the last region of the core to refill) is completely filled with water at the end of the calculation. Figure 7 shows the time history of the maximum cladding temperature in superheater1 for the hot and average channel. The peak temperature during the refilling phase is certainly acceptable. These results have to be taken with some caution, because no model for counter-current flow limitation (CCFL) has been applied, due to the lack of specific information relevant for the complex geometry of the mixing chambers. However, in consideration of the flow reversals and the fact that the superheater 1 is the region of the core that remains filled with steam for the longest time, especially CCFL in the upper mixing chamber above superheater 1 could hinder or delay the downwards flow of water and therefore the refilling of the core. In conclusion, the results of the analyses show that the use of ADS system seems to be sufficient to limit the cladding temperature excursion to a reasonable value after the pressure has dropped below the critical value. However, for this transient, the first few seconds have also to be looked at. In fact, the first temperature excursion occurs during the time when the pressure is still supercritical.

Figure 8 shows that peak cladding temperature in the hot channel reaches about 850^{o}C, which is far below the acceptable limit (1200^{o}C) for class 4 accidents. These results have to be taken with some caution because of two reasons:

1. underestimate the initial temperature excursion. In fact, Fig. 8 shows the ratio between the heat flux and the value that according to the widely used Yamagata criterion for heat transfer deterioration would lead to this condition. It can be observed that due to the low flow rates, during the first second heat transfer deterioration is likely to occur.
2. The modeling of the transition from supercritical to subcritical pressure has never been assessed against experimental results, due to the lack of a database. It is therefore still uncertain whether the correlations used for physical processes (especially interfacial heat/mass transfer) are valid during the transition, or would lead to substantial overestimate of the cooling effectives of the forming two-phase mixture.

In conclusion, the issue of the possible cladding overheating during the first seconds of an accident initiated by a large break in one of the feedwater lines is still open, and only experiments can clarify whether the model implemented in the codes are adequate for simulating transient heat transfer for supercritical conditions and during the transition from supercritical to subcritical region.

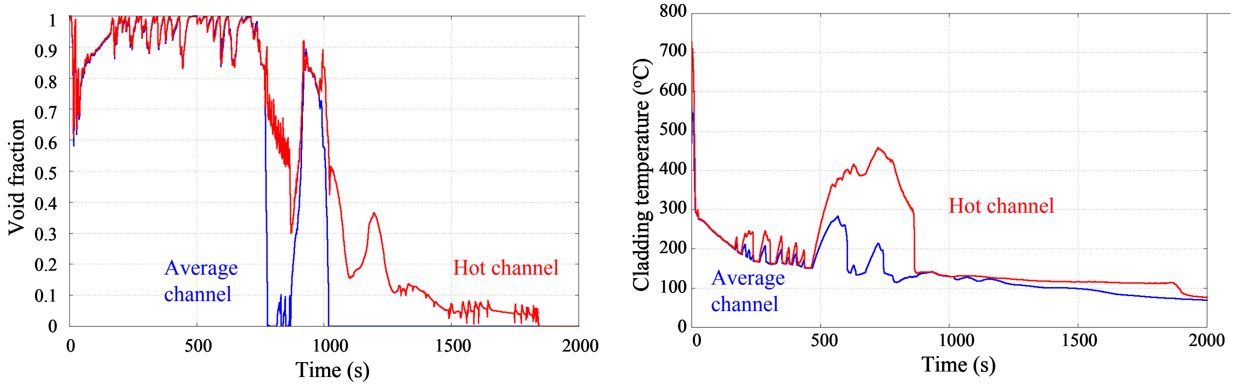

Figure 7 Average void fraction (right) and peak cladding temperature (right) in superheater 1 calculated by CATHARE for a large break in the feedwater line.

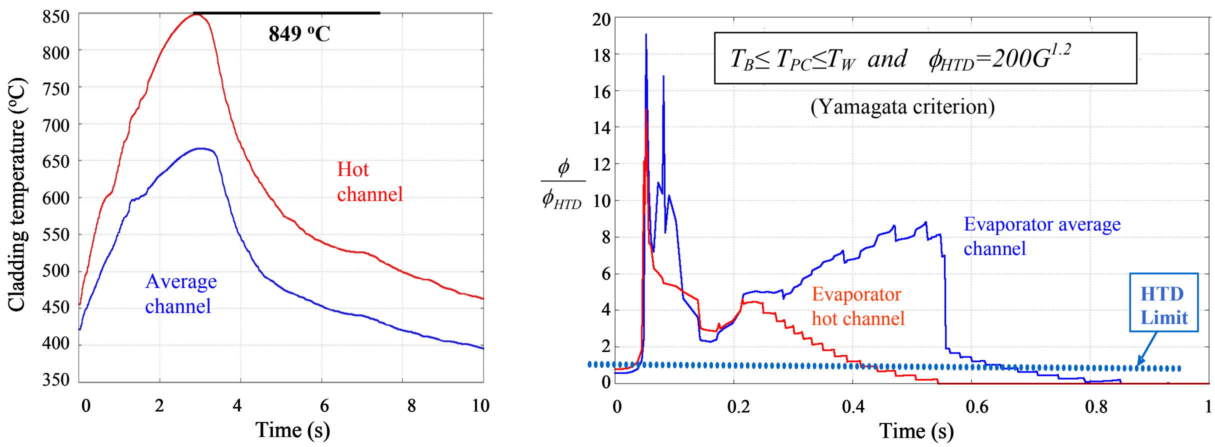

Figure 8 Peak cladding temperature in the evaporator calculated by CATHARE for a large LOCA in the feedwater line during the first seconds (left) and occurrence of heat transfer deterioration (right).

3.4 Safety analyses for selected RIAs and ATWS transients

Various RIAs and ATWS have been analysed. As mentioned for the other transients, the events selected for the analyses have been extracted from a complete list, and included events of classes 2 to 4. Here only the results for the reference transient, the control rod ejection, and the summary of the results for the other transients are reported.

3.4.1 Control Rod Ejection (CRE)

The accident is defined as the mechanical failure of a control rod mechanism housing such that the reactor pressure ejects a control rod up to its uppermost position. The consequence of this failure is a fast and large positive reactivity insertion which results in a core a power excursion with a large localized relative power increase. It is necessary to analyze the accident to determine if there is any fuel damage or damage to the reactor coolant system pressure boundary as a result of the power excursion. This event has been analyzed with two coupled codes, ATHLET-KIKO3D and SMABRE/TRAB-3D. In the ATHLET-KIKO3D model, the primary circuit of the HPLWR is modeled by 88 thermo-fluid and 59 heat conduction objects. In the KIKO3D core model, the 156 clusters are individually represented, and each cluster consisting of 9 assemblies is divided into 20 axial nodes, both for neutronics and thermal-hydraulics. The input for the 3D core analysis code TRAB-3D includes the detailed description of the core with fuel rods and an individual flow channel attached to each fuel assembly. Three different core nodalizations were created for SMABRE for testing: using 3 hydraulic channels (1/core section), 156 hydraulic channels (1/cluster) and 1404 hydraulic channels (1/fuel assembly). The model with 1404 hydraulic channels proved to be the most stable and therefore it is used in all calculations presented here. Except for the core region, the same SMABRE model is used for RPV and circuit as in stand-alone SMABRE simulations. The ejected control rod is situated in the superheater 1 close to the evaporator which is an asymmetric position. The rod is ejected from 0 cm to 420 cm within one second in Case A and in 0.1 second in Case B.

Since the thermal-hydraulic model is very different in the two codes, fluid, cladding and fuel centreline temperatures are difficult to compare. Therefore, the comparison is limited to the global variables that are directly affected by the 3-D neutronics, namely reactivity and power. Figure 9 shows the comparison of the power calculated by the two codes for the case with faster ejection (0.1 s). Consistently with the reactivity, the peak in the power calculated with SMABRE/TRAB-3D is higher than that calculated with ATHLET-KIKO3D. In general, however, the results are in excellent agreement with each other and provide sufficient confidence that the results obtained with the ATHLET-KIKO3D code for other transients (see below) are representative of the behavior of the HPLWR.

The results obtained with ATHLET-KIKO3D, where the hot channels are represented, permit to evaluate whether the acceptance criteria are fulfilled for this transient. Cladding and fuel centerline temperatures were examined in the hot channels of three assemblies. Figure 10 shows the maximum cladding and fuel centerline temperatures in the three regions of the core for the case with faster ejection time. The maximum cladding temperatures (860^{o}C) doesn't exceed maximum allowed values in any cases. The maximum centerline fuel temperatures (2600^{o}C) also remains below the allowed limit (2800^{o}C), although is not very far from it.

3.4.2 Other events

The analyses of the other events investigated with ATHLET-KIKO3D are reported in [7-8] and summarized below. The analyses showed that the acceptance criteria are fulfilled in all but one case. However, in spite of the relatively strong feed back, in some cases the hot channel temperatures are not far from the limits, which points out the necessity of the RIA analyses. The reasons are the strong sensitivity of the neutronic characteristics on the moderator density, changing in a wide range, and as a consequence, the significant perturbation of power peaking factors during the transients. In the case of uncontrolled withdrawal of an absorber from the bottom position without SCRAM fuel melting occurs, which can be avoided by limiting the allowed position of control rods or by applying lower worth regulating rods compared to shutdown rods. Some further study is needed to introduce appropriate preventing measures.

4. Conclusions

The simulations performed within the HPLWR 2 project provides valuable information on the system response to a variety of events, which could be used for the development of a preliminary safety concept. In order to verify its feasibility and to verify that the specific characteristics of the HPLWR, including the three-pass core design, would not pose safety concerns difficult to address with the available technology and considering realistic economic constraints, safety analyses have been performed using a variety of codes.

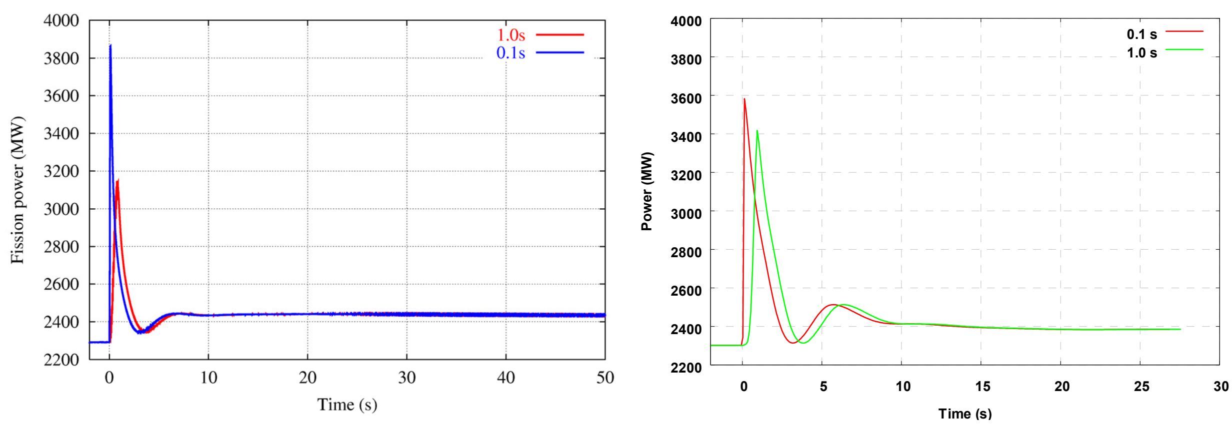

Figure 9 Comparison of the power calculated for a CRE event with SMABRE/TRAB-3D (left) and ATHLET-KIKO3D (right) for the two cases with 0.1 s and 1 s ejection time, respectively.

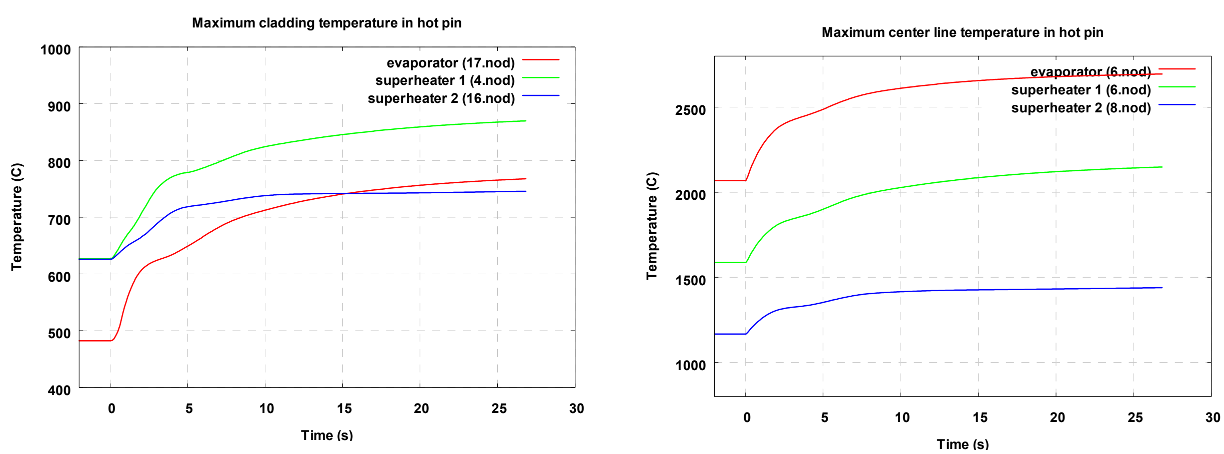

Figure 10 Peak cladding (left) and fuel (right) temperatures I the there regions of the core calculated by ATHLET-KIKO3D for a CRE with 0.1 s ejection time.

Upgraded systems codes and coupled codes with three-dimensional representation of thermal-hydraulics and neutronic feedback have been used for the simulations. In all codes supercritical water properties and appropriate correlations for heat transfer and pressure drops have been implemented. Numerical methods and physical models for the transition from supercritical to subcritical region have been developed. However, only two codes, APROS and CATHARE, have been demonstrated to be able to simulate fast depressurization events. This has to be regarded as one of the most valuable result of the project, as these codes are now ready for being used for any further development of the HPLWR concept. In fact, although within this work no code could be demonstrated to be capable to simulate all transients (including LOCAs and RIAs), the simulations performed show that for each class of transients at least one of the computational tools used in this project is adequate for preliminary assessment of the safety concept of the HPLWR

It should be considered, however, that heat transfer in the supercritical region and the physical basis of the methodologies adopted to simulate the transition from supercritical to subcritical conditions should be assessed against experimental data. Moreover, in the course of the analyses, it was realized that the use of codes using point kinetics is questionable for many transients implying core flow reduction, because variations of the water density distribution caused by modifications in the flow rate and direction in gap and moderator channels can cause substantial modification in both vertical and transversal power distribution. The further use of these codes (fast to run and thus suitable for sensitivity studies) is therefore to be recommended only for transients where the comparison with results obtained with coupled codes is available for reference conditions. Due to these power distribution modifications in a broad variety of transients, coupled codes will have a prominent role in the analyses of the HPLWR in future development work.

Within the limitations imposed by the methodology used to decouple safety analyses from design issues, uncertainties in the validity of certain models and the caution necessary to take the results of analyses with point kinetics, the main conclusion of the safety analyses is that the preliminary safety concept is probably adequate to match the requirements, although a number of open issues, discussed below, remain to be addressed in future projects.

A number of open issues have also been identified, which indicate the future work needed. The most prominent are listed below:

- Code capabilities should be assessed using experimental data. Especially urgent are blow-down experiments from supercritical conditions, where data for heat transfer during the depressurization can be generated
- Model for heat transfer deterioration is necessary for evaluating the cladding temperature excursion during the first seconds of various transients and accidents
- First assessment should be confirmed for an optimized core
- Coupled codes should be upgraded to consider separately the three regions of moderator (gap, moderator channel and fuel channel) and/or demonstrated to be capable to provide reliable results for the special three-pass core design of the HPLWR for a broad range of conditions, including transients with fast depressurization or flow reversal in gap and moderator channels

- Some further study is needed to introduce appropriate measures to prevent fuel melt in the case of uncontrolled withdrawal of an absorber from the bottom position without SCRAM. Limitations on the allowed control rod worth and positions should be investigated. Optimization of the core should aim to increase the safety margins also in case of other RIAs
- Events with total LOFW should be further investigated with special attention, using coupled codes

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