

SIMULATION OF LOCA WITHIN A GERMAN BWR CONTAINMENT WITH THE COUPLED VERSION OF ATHLET-COCOSYS

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Abstract

In this paper the effectiveness of the safety system in a generic Boiling Water Reactor (BWR) is investigated by simulating a loss-of-coolant accident (LOCA). To this end, a two-sided large break in a feed water line within the containment is simulated. Since the LOCA is characterized by significant interactions between the reactor cooling circuit and the containment, the simulation is performed using the coupled version of the thermo-hydraulic code ATHLET and the containment code COCOSYS. The analysis focuses on the calculated plant behavior from the accident initiation to the depressurization phase. The results show that the emergency core cooling system with only two operational pumps succeeds in cooling the fuel elements throughout the entire accident sequence. A comparison of the simulation results with a separate ATHLET simulation indicates slight deviations when the pressure in the reactor cooling circuit approaches the drywell pressure.

Introduction

Safe operation of Light Water Reactors (LWR) requires an adequate safety system to prevent core damage during postulated accident scenarios. Among the most challenging accident scenarios for safety systems are loss-of-coolant accidents (LOCA). Typically, leakages of different sizes in the reactor cooling circuit cause LOCAs in LWRs. In contrast to small breaks, large breaks (LB) are believed to cause a much faster and larger loss-of-coolant from the reactor cooling circuit and are therefore considered to be the relevant accidents safety systems in LWRs have to be designed for. Licensing for operation of LWRs comprises detailed analysis of the effectiveness of the emergency core cooling system (ECCS) during defined LB-LOCAs which are also referred to as design basis accidents. In this study the procedure to demonstrate the effectiveness of ECCS for a generic Boiling Water Reactor (BWR) is presented by simulating the plant behaviour during a LB-LOCA.

The general aim is to demonstrate that the ECCS can provide a sufficient coolant injection in order to cool the fuel elements so to prevent core temperatures from rising. Basically, LB-LOCAs in BWRs are characterised according to the location of the large break. Design basis accidents cover LOCAs within the containment caused by large breaks in the steam line or in the feed water line as well as a leak of 80 cm² in the bottom of the Reactor Pressure Vessel (RPV). In addition, it is assumed that not all emergency core cooling systems are operational taking account of the possible failure of single components as well as of pumps which are unavailable due to repairing.

With these assumptions, a large break in the feed water line within the containment is believed to lead to the largest amount of coolant lost and was therefore chosen for this study.

For the analysis of the behaviour of the generic BWR during a LB-LOCA the coupled version of the thermo-hydraulic code ATHLET (Analysis of Thermo hydraulics of Leaks and Transients) and the containment code COCOSYS (Containment Code System) is used. In contrast to a separate ATHLET simulation, the coupling of a thermo-hydraulic code to simulate the reactor cooling circuit and a containment code is clearly beneficial for this analysis since interactions between reactor cooling circuit and containment are significant and can directly be taken into account during the simulation.

In the first section, a brief description of the generic BWR including the affected safety systems is given. Next, the accident scenario is described and the relevant boundary conditions are provided. The used codes to perform the analysis are introduced in the second section including a detailed description of the ATHLET-COCOSYS coupling. Afterwards, the results of the analysis regarding the thermo-hydraulic behaviour in the reactor cooling circuit and the resulting containment response are analysed and interpreted in section 3. Last, the main outcomes of this investigation are summarized and some conclusions concerning the application of the coupled version of ATHLET-COCOSYS are given.

1. Description of the generic BWR and accident scenario

In this analysis a generic four-loop BWR with a nominal power of 800 MW_{el} is considered. The main components of the containment described at first in this section are designed and constructed according to the so-called building line 69 by the German Kraftwerk Union. Afterwards, the safety systems as well as the analysed accident scenario are presented.

1.1 Containment

Figure 1 shows a sketch of the containment of the generic BWR. The spherical containment consists of the drywell with the RPV located in the centre. The RPV is surrounded by a toroid pressure suppression chamber which contains a pool of about 2200 m³ water and a gas-filled upper part with a volume of app. 2150 m³ is. Beneath the RPV, the lower part of the drywell encloses the control rod room.

Drywell and pressure suppression pool are connected by venting pipes. They serve to avoid a possible pressure build-up within the drywell in case of LOCA by directing the released steam into the pressure suppression pool for condensation.

1.2 Safety systems

In the following the main features of the safety system of the generic BWR considered in this analysis are given. Since the most efficient coolant injection is provided by the low-pressure coolant injection system (LPCI) which operates at pressures below app. 14 bar, the depressurization system is an essential part of the safety of the BWR. The automatic depressurization system (ADS) is triggered in case of a LOCA.

The LOCA criterion triggered is defined by a pressure difference larger than 0.25 bar between the pressure in the drywell compared to the one outside the containment which is assumed to be 1 bar, i. e. when detecting a pressure in the drywell larger than 1.25 bar. Then the safety/relief valves (SRV) located in pipes connected to the steam lines will automatically open in order to depressurize the reactor cooling circuit by discharging steam from the RPV into the pressure suppression pool.

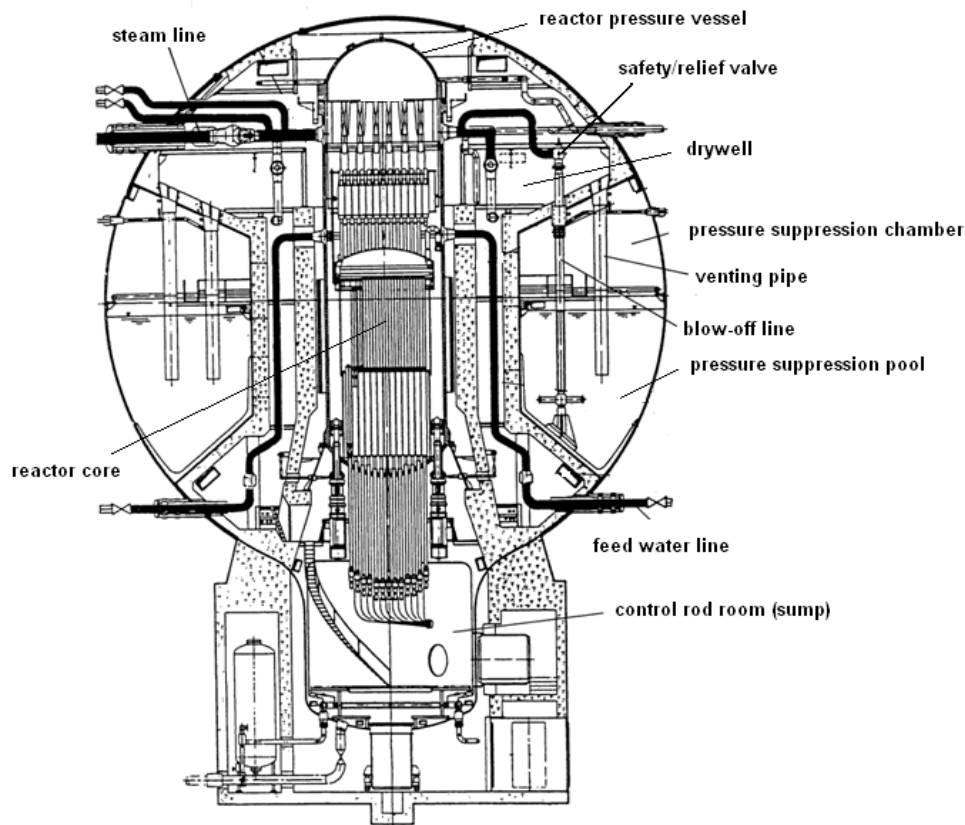


Figure 1 Containment of a generic BWR [1].

The LPCI consists of four pumps for the injection of coolant taken from the pressure suppression pool into the RPV. For this purpose each pump is connected to a feed water line. Besides the function of providing coolant into the RPV (emergency cooling phase I, $t \leq 600$ s), the LPCI also serves to cool the suppression pool beginning from 600 s after activation of the coolant injection (emergency cooling phase II, $t > 600$ s). Then the water taken from the suppression pool is cooled in external heat exchangers and returned to the suppression pool instead of being injected into the RPV. An additional injection of coolant is provided by an independent low-pressure coolant injection system with two pumps run by diesel engines situated outside the reactor building. Coolant provided by this independent system enters the RPV via two separate nozzles. Like the LPCI, it is activated by the same LOCA criterion and it also takes water from the suppression pool.

In contrast to the LPCI it does not contribute to the suppression pool cooling during emergency cooling phase II ($t > 600$ s) but it continues to inject coolant into the RPV.

The contribution of the two high-pressure coolant injection systems (HPCI) can be neglected for the analysis of LB-LOCA since the pressure in the reactor cooling circuit decreases rapidly due to the ADS so that the HPCI would only inject within a relative short time span.

1.3 Analysed LOCA scenario

The accident scenario chosen for this investigation is a LOCA caused by a two-sided break in one of the main feedwater lines within the containment. The location of the break is assumed to be between the RPV and the inner isolation valve of the feed water line so that this broken line cannot be isolated. Moreover, since the LPCI location lies outside the containment coolant injected by the LPCI pump connected to the broken feed water line is completely lost and will not get into the RPV. In addition, another pump of the LPCI is supposed to be unavailable, so that the core is flooded during emergency cooling phase I ($t > 600$ s) by two out of four pumps only.

The two pumps of the independent LPCI cannot inject into the RPV due to the assumed failure of the common isolation valve so to take account of the single failure concept. These assumptions are made to consider a most unfavourable failure combination.

By the time the LOCA occurs the reactor is assumed to operate at full-power (2,292 MW_{th}) and nominal steam flow rate (1,240 kg/s). All further relevant thermo-hydraulic parameters at accident initiation are chosen corresponding to the nominal operation of the reactor: pressure in the reactor cooling system is 69.6 bar, steam temperature is 285 °C, water level in the RPV is 13.86 m and the water temperature in the pressure suppression pool is 38 °C.

2. Code description

In the following a short description of the thermo-hydraulic code ATHLET and the containment code COCOSYS used in this investigation which are both under development by the German GRS (Gesellschaft für Anlagen- und Reaktorsicherheit) is given. The coupling concept of the two codes for the simulation is explained. Moreover, the advantages of the application of the coupling of a thermo-hydraulic with a containment code to simulate the plant behaviour during such accident scenarios are pointed out at the end of this section.

2.1 Thermo-hydraulic code ATHLET

The thermal-hydraulic computer code ATHLET is a mechanistic Lumped-Parameter (LP)-code for the analysis of anticipated and abnormal plant transients, small and intermediate leaks as well as large breaks in light water reactors [2]. The one-dimensional, two-phase fluiddynamic models are based on six conservation equations supplemented by a full-range drift-flux model, including a dynamic mixture-level tracking capability. The heat conduction and heat transfer module allows a flexible simulation of fuel rods and structures.

The nuclear heat generation can be calculated either by a 1D-kinetics or by a point-kinetics model. A general control simulation module is provided for a flexible modeling of Balance-of-Plant (BOP) and auxiliary plant systems [2].

The input deck for the simulation models the entire reactor cooling circuit. In particular, all four feed water lines which are connected to the downcomer of the RPV as well as all four steam lines connected to the steam dome of the RPV are modeled. The modeling of the RPV consists of various flow channels connected by junctions to simulate the flow path through the RPV. For the simulation of the decay heat generation the point-kinetics model is applied since this simplified model is sufficient for LOCA analysis. Furthermore, all relevant accident management measures as well as control systems are simulated in the input deck.

2.2 Containment code COCOSYS

The containment code system COCOSYS is a LP-code based on mechanistic models to describe the containment processes during design basis accidents [3]. The input deck used for this analysis is basically divided into different compartments to model the drywell, the pressure suppression pool and the sump. Moreover, the injected mass flow rates depending on the pressure in the reactor cooling system and the cooling of the water inventory in the pressure suppression pool by the LPCI system are simulated as well.

2.3 Coupling ATHLET-COCOSYS

The simulation tool used in this analysis is the coupled version of ATHLET and COCOSYS. As mentioned above, ATHLET simulates the thermo-hydraulic behaviour of the reactor cooling circuit including the discharged mass flow rates through the break as well as the core temperatures and decay heat generation. The resulting containment response in terms of pressure and temperature in the drywell as well as in the suppression pool is calculated by COCOSYS.

Each code runs parallel and the calculated data are exchanged on defined locations, so-called interfaces. Defined interfaces between the reactor cooling circuit and the containment are the break in the feed water line coupled to the drywell, the SRV coupled to the pressure suppression pool as well as the coolant injection of the ECCS calculated based on the corresponding pressure in the reactor cooling circuit.

By applying the coupled version long iteration calculations can be avoided which otherwise are needed to define the boundary conditions, for instance to perform separate ATHLET analysis. The main benefit of applying the coupled version is that interactions between the reactor cooling circuit and the containment can directly be taken into account during the calculation of the accident sequence. Therefore, the discharged mass and enthalpy flows from the reactor cooling circuit calculated by ATHLET affect the simulation of the temperature and pressure in the drywell. In turn, the pressure in the drywell simulated by COCOSYS influences the calculated discharged mass. In the same way, simulated temperature and pressure response in the pressure suppression pool are determined by the calculated released mass flow rate via the SRVs. Finally, the simulated coolant injection rate calculated by COCOSYS depends on the corresponding pressure in the reactor cooling circuit.

3. Results

The analysis of the calculated plant behaviour during a LOCA caused by a large break in one of the feed water lines is discussed in the following. Calculations were performed simulating the first 6000 s after opening of the large break in the feed water line.

Figure 2 illustrates the calculated **mass flow rates** through the break coming from the RPV-side and the lost mass flow rate from the feed water pump.

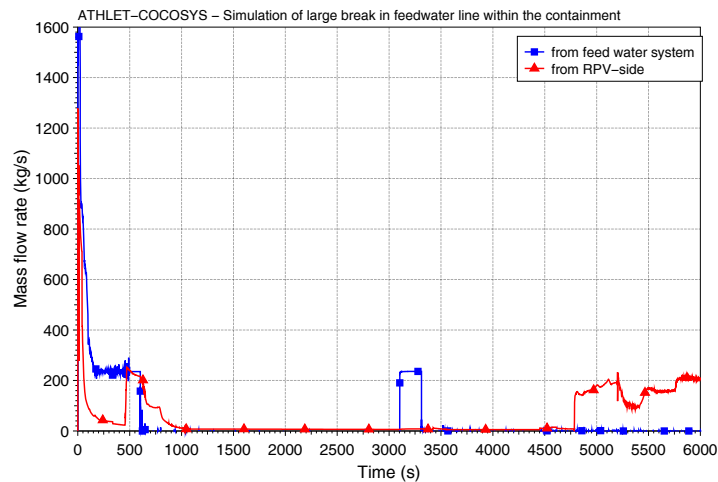


Figure 2: Leak mass flow rates (discharged from feed water line and RPV-side)

Figure 3 shows that the discharged mass flow rate causes a sharp **pressure increase in the drywell** to app. 2.4 bar.

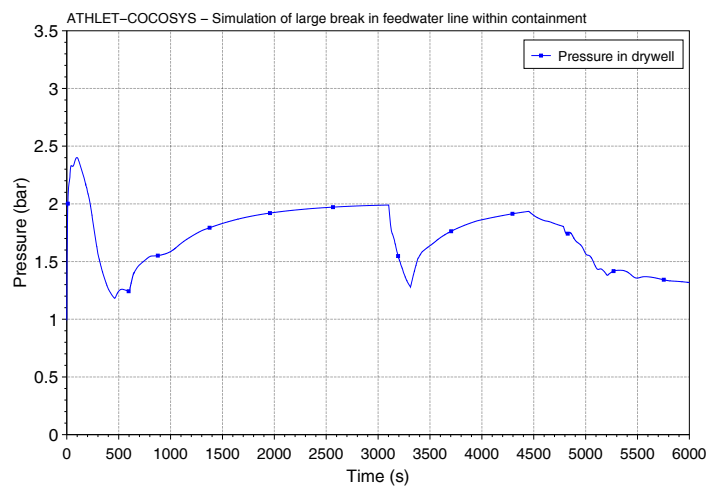


Figure 3: Pressure in drywell

This leads to the activation of all relevant safety measures by triggering the LOCA criterion (pressure difference larger than 0.25 bar between the drywell and the reactor building where the pressure is assumed to be 1 bar). Relevant actions taken are reactor scram, turbine shut-down, closure of the isolation valves, start of the ADS, trip of the feed water pumps and start of the LPCI system.

Figure 4 depicts that the **pressure in the reactor cooling circuit** decreases rapidly due to the released mass flow through the break and via the SRVs. This shows that an effective depressurization of the reactor cooling circuit by the ADS is provided. At first, four of seven available SRVs are opened for depressurization and the remaining three SRVs do not open until the pressure falls below 10 bar in order to provide a more efficient depressurization when the pressure difference between RPV and drywell is already small.

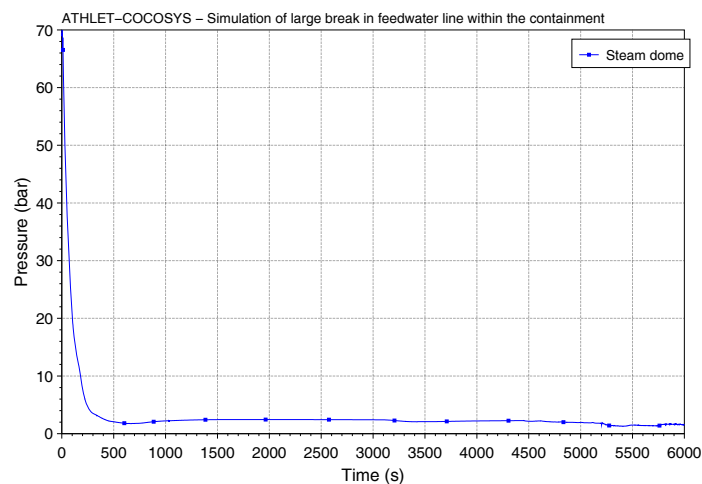


Figure 4 Pressure in reactor cooling system

The calculated **mass flow through SRVs** is shown in **Figure 5** and **Figure 6** illustrates the subsequent **temperature increase in the pressure suppression pool** due to the condensation of the discharged steam.

The **collapsed water level** within the RPV during the LOCA is shown in **Figure 7**. It can be seen that the collapsed level rapidly decreases shortly after the opening of the break reaching a minimum of app. 8.9 m. When LPCI starts delivering coolant, an effective refilling of the RPV is calculated so that the collapsed level rises to about 13 m.

Figure 8 gives the **flow rate of the fed-in coolant into the RPV** provided by the LPCI system. Injection starts when the pressure in the reactor cooling circuit falls below 14 bar (see Figure 4) after app. 210 s. An exception is the pump connected to the broken feed water line. This pump injects right after the break opens due to the fact that the pressure within the broken line falls almost immediately to the drywell pressure (see Figure 3). However, the injected water is lost through the break and accumulates in the sump.

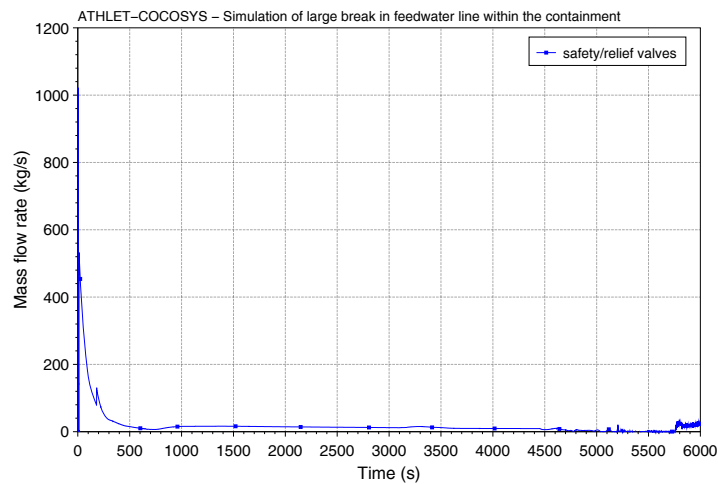


Figure 5 Total mass flow rate through safety/relief valves

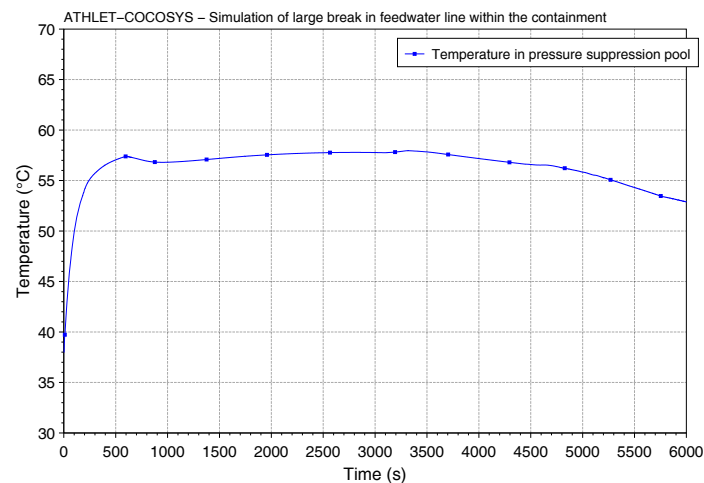


Figure 6 Temperature in pressure suppression pool

Coolant injection into the RPV is stopped after 600 s and the LPCI system is switched to the suppression pool cooling mode. Subsequently, the collapsed level in the RPV decreases slightly to the height of the nozzles of the feed water lines at app. 12 m due to the release of water via the break. The subsequent evaporation caused by the decay heat takes about 2,200 s for the water level in the RPV to decrease to 10 m.

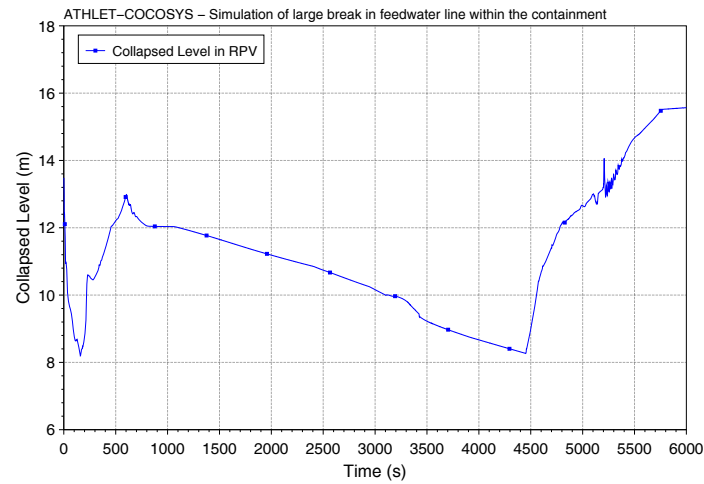


Figure 7 Collapsed level in RPV

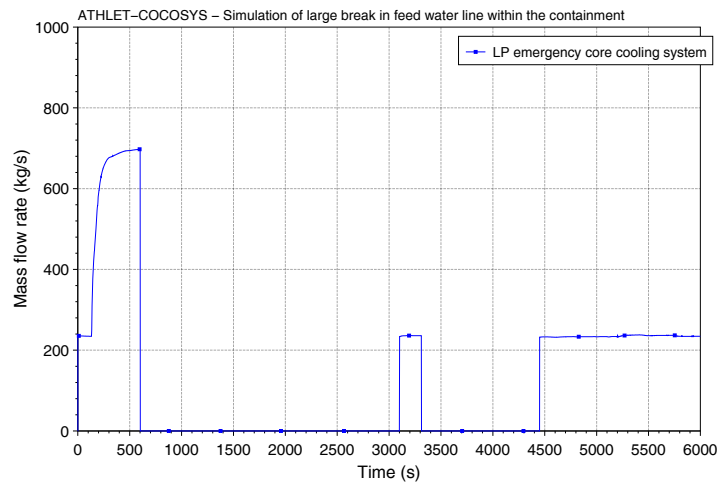


Figure 8 Mass flow rate provided by low-pressure coolant injection system

When the water level falls below 10 m, the LPCI is triggered to provide coolant into the RPV. According to the concept of the coolant injection into the RPV in the emergency cooling phase II ($t > 600$ s), the leg of the LPCI which is first switched back to the RPV flooding mode is the one that injects to the broken feed water line. Therefore, the collapsed level in the RPV continues to decrease since the injection via this leg of the LPCI is lost through the break (see also Figures 2 and 8). After 210 s in operation this leg is switched back to the suppression pool cooling mode since the water level in the RPV does not rise.

Subsequently, a second leg of the LPCI would be switched to inject into the RPV. This leg, however, is the one assumed to be unavailable. This constitutes a further conservative

assumption concerning the core cooling in the emergency cooling phase II ($t > 600$ s) since an effective refilling of the RPV is only provided with a maximum delay of 1.350 s after the activation of the LPCI to flood the RPV (water level below 10 m) in the emergency cooling phase II ($t > 600$ s). When the third LPCI leg is switched back to flood the RPV (see Figure 8) the collapsed level has fallen to about 8.5 m. Subsequently, an almost immediate increase in the collapsed level in the RPV is calculated.

Figure 9 indicates that during the entire accident sequence a sufficient cooling of the core is provided since the **temperatures of the fuel rod claddings** are kept at about 130 °C.

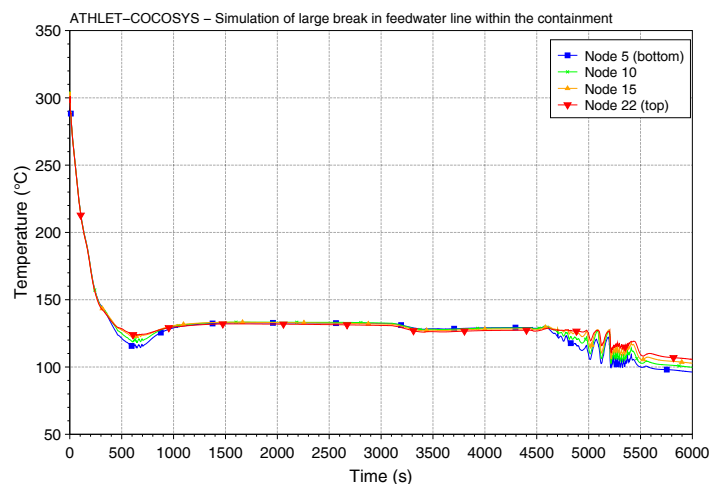


Figure 9 Temperature fuel rod cladding

The water level in the pressure suppression pool drops due to the injection of coolant into the core by the LPCI. Injected water taken from the suppression pool and lost via the break accumulates in the sump (control rod room of the containment) so that the water level in the sump rises. This process is of significance for the safety of the plant as the coolant inventory in the suppression pool available for core coolant injection sinks. Therefore accident management measures to assure a sufficient coolant inventory in the suppression pool are included in the design of the safety system.

The accident management measure taken for this purpose is the recirculation from the sump into the suppression pool which is triggered when the water level in the suppression pool falls below 17.63 m and the water level in the sump amounts to more than 4.8 m. Then one of the legs of the LPCI will be actuated to operate in a recirculating mode. The recirculation of coolant accumulated in the sump back into the pressure suppression pool is the third function of the LPCI besides the core coolant injection and the cooling of the water inventory of the pressure suppression pool. In this analysis the activation of the recirculation from the sump to the suppression pool starts shortly before the end of the considered accident sequence of 6,000 s and is for this reason not shown here.

3.1 Simulation results compared to separate ATHLET simulation

To assess the influence of the coupled version of ATHLET-COCOSYS used in this analysis, the simulation results are compared to a simulation with ATHLET in a stand-alone version. The containment response cannot be simulated by the separate ATHLET calculation but pressure and temperature in the drywell and suppression pool are input as boundary conditions.

Figure 10 shows the **collapsed level in the RPV** calculated by both codes. In the first phase, hardly any differences can be noticed. This demonstrates that important thermo-hydraulic parameters that determine the collapsed level in the RPV like the discharged coolant via the break and the SRVs as well as the injection rate of the LPCI are the same.

However, when the reactor cooling circuit is already depressurised and the pressure difference to the containment is relatively small compared to the one at accident initiation, the influence of the pressure in the drywell becomes noticeable since it directly affects the mass flow through the break. The discharged mass flow rate decreases due to a higher pressure in drywell simulated with ATHLET-COCOSYS. As a consequence, the collapsed level in the RPV rises faster compared to the separate ATHLET calculation.

So, it can be concluded that the coupled version of ATHLET-COCOSYS is in particular beneficial for the analysis of LOCA when small pressure differences between the reactor cooling circuit and the containment mainly determine the plant behaviour.

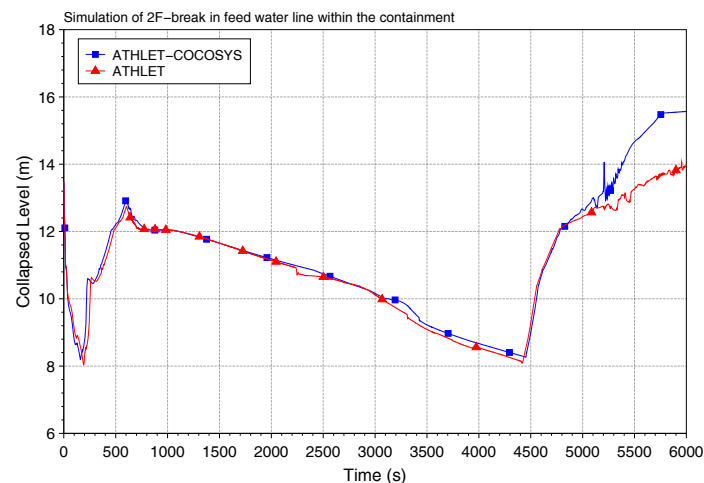


Figure 10 Comparison collapsed level in RPV

4. Conclusion

The plant behaviour of a generic BWR during a LB-LOCA is analysed in this paper to demonstrate the effectiveness of the emergency core cooling system by a simulation with the coupled version of ATHLET-COCOSYS. The use of the coupled version is beneficial for this task since interactions between the reactor cooling circuit and the containment are directly taken into account during the simulation. The accident scenario is a large break in a feed water line within the containment which is believed to be one of the most challenging design basis accidents for emergency core cooling systems. In addition, the number of effective coolant injection pumps is restricted to two in order to assess the emergency core cooling system under a most unfavourable failure combination. The aim of the analysis is to show that a sufficient coolant injection is provided to prevent core damage.

The results of the simulation focus on the most relevant thermo-hydraulic parameters regarding the functions of the ECCS. The released water inventory of the reactor cooling circuit due to the opening of the break and the depressurization can almost immediately be recovered when the low-pressure coolant injection system is actuated. During the entire accident sequence the removal of decay heat is achieved and a sufficient coolant supply is provided to prevent core temperatures from rising.

Comparing the calculation results with those obtained by a separate ATHLET simulation, a substantial agreement especially in the first phase of the accident scenario can be found. Deviations between the calculation results can be seen when the reactor cooling circuit is already depressurized so that the drywell pressure becomes more significant for the calculation of the behaviour in the reactor cooling circuit.

5. References

- [1] Kraftwerks Union AG, “Atomwirtschaft Atomtechnik”, Sonderdruck Siedewasserreaktoren, December 1984.
- [2] G. Lerchl, H. Austregesilo, “ATHLET Mod 2.2 – Cycle A – User’s Manual”, GRS-P-1/ Vol. 1 Rev. 5, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, July 2009.
- [3] H.-J. Allelein, S. Arndt, W. Klein-Heßling, “COCOSYS: Status of development and validation of the German containment code system”, Nuclear Engineering and Design, Vol. 238, pp. 872-889 (2008).