Investigations of a BWR SB-LOCA Severe Accident Scenario including SAM to prevent the Reactor Pressure Vessel Failure using the German Code ATHLET-CD

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Summary –This paper describes the investigations performed by KIT in the frame of the German WASA-BOSS projected funded by the Federal Ministry of Economics and Technology to improve severe accident codes and to analyse options to prevent or delay the failure of the main safety barriers e.g. reactor pressure vessel of PWR and BWR. The KIT contribution was focused on the study of the potential accident management to delay or prevent the failure of the reactor pressure vessel (RPV) of a BWR using ATHLET-CD. For a generic BWR, different severe accident sequences without and with different SAM-measures were extensively investigated. The obtained results have shown that SAM-measures can be effective to mitigate the accident progression if they start before material relocation in the lower plenum. This paper will discuss the SAM-measures to avoid or delay the RPV-failure by performing reflooding of the core in certain time windows and under which conditions it will fail. The modelling issues, assumptions and results will be presented and discussed.

1. INTRODUCTION

The Fukushima accident lead to a re-evaluation of severe accident management (SAM) measures for nuclear power plants worldwide. The German WASA-BOSS projected was initiated aiming at improvement of severe accident codes and the evaluation of severe accident measures for both PWR and BWR. Universities and research centres including KIT participated in this German project. The KIT-contribution was focused on the study of the potential accident management to delay or prevent the failure of the reactor pressure vessel (RPV) of a BWR using ATHLET-CD. For this purpose, an accident scenario consisting in a small break of the steam line inside the containment and assuming the multiple failures of the safety systems was selected according to the PSA-analysis. Crucial for the RPV integrity, according to the analysis, is the coolability of a partly-damaged core and the time of core reflooding as part of a SAM-measure due to the recovery of safety systems. The obtained results have shown that SAM-measures can be effective to mitigate the accident progression if they start before material relocation in the lower plenum. But the opposite is the case if the water injection is done when material relocation in the lower plenum is about to start or has already led to significant amount of corium formation. This paper will discuss the SAM-measures to avoid or delay the RPV-failure by performing reflooding of the core in certain time windows and under which conditions it will fail. The modelling issues, assumptions and results will be discussed and an outlook will be given.

2. CODES, PLANT AND MODELS

2.1. *The ATHLET-CD simulation tool*

ATHLET (Analysis of Thermal-Hydraulics of Leaks and Transient) is a the German system thermal hydrualic code developed by the GRS to describe the reactor coolant system thermal-hydraulic response during normal and off-normal operating conditions. It has a highly modular structure in order to include a large spectrum of models and to offer a flexible basis for further development. In this frame, the simulation of in-vessel processes is carried out by different coupled modules which constitute the code version ATHLET-CD (Core Degradation). These modules allow the user to reproduce the core damage progression (ECORE module), the debris bed behaviour (MEWA), the lower plenum behaviour (AIDA or LHEAD) fission products and aerosol behaviour (FIPREM) during severe accidents, to calculate the source term for containment analyses, and to evaluate accident management measures (Austregesilo, et al., 2013). The core damage progression described by the ECORE module takes into account the mechanical fuel rod behaviour, oxidation of zirconium and boron carbide, melting of metallic and ceramic components, freezing, re-melting and refreezing, formation and dissolution of blockages. The mechanical rod model consists in a simplified one-dimensional approach aimed at the calculation of the fuel and cladding deformation as a consequence of both thermal expansion and creep. Cladding creep velocity is computed by means of specific correlations which take into account α and β phase transition of Zircaloy cladding, as well as oxygen and hydrogen effects in order to estimate the amount of clad ballooning and the time-to-failure which is established by different criteria available in the code. The code is widely validated using experimental and plant data for LWR applications.

2.2. Short description of the German BWR

The generic BWR consists of 784 fuel assemblies with a thermal power of 3840 MW (Pointner, 1984). Fuel assemblies contain between 80 and 96 fuel rods arranged, respectively, in a 9 x 9 or 10 x 10 grid according to the different designs. The fuel rod claddings are made of Zircaloy with an outer diameter of 10.75 mm and are filled with uranium dioxide or MOX pellets over an active length of 3710 mm. The reactivity control is carried out by means of 193 cross-shaped control rods which are inserted from the core bottom. The containment has a cylindrical shaped structure made from pre-stressed concrete with steel liner. The pressure suppression pool located in the lower part and all the systems dealing with reactor coolant are included in the containment, allowing easy accessibility of most of the drywell areas. As for the safety-relevant systems, the most important features of the NPP are described as follows. The reactor scram system provides control rod injection at any reactor pressure by means of a nitrogen cushion and can also be diversely driven with an electric motor. The nuclear residual heat removal system (RHR) consists of three trains, having completely separated and independent process, electric, instrumentation and control redundancies (3 x 100%). The RHR system allows both high and low-pressure injections, and pressure suppression pool cooling in order to guarantee long-term core cooling at any reactor pressure. Moreover, an additional, independent residual heat removal system is implemented in this plant which works with a diverse heat sink by means of cooling towers and is equipped with a diverse emergency power diesel. The automatic pressure limitation system is able to handle undesirable pressure build-up in the reactor in case of isolation of the main steam line. The excess of steam produced by residual heat is discharged into the pressure suppression pool via eleven safety and relief valves (SRV) or three diverse pressure-relief valves. The reactor protection system monitors all safety-relevant data and on reaching limit values, it actuates reactor-protection signals that initiate automatic protective actions. The emergency-power system can supply the safety-related components and systems in case of loss of main power by means of an automatic switch to the 110 kV stand-by supply or to the emergency diesels.

2.3. Integral plant model of the German BWR

The integral input deck of ATHLET developed for BWR plant to simulate the small break LOCA at the steam line is described in detail in (Di Marcello, Imke, & Sanchez, 2016). It is based on a basic model provided by the GRS to the German WASA-BOSS project. As described in (Di Marcello et al., 2014), the plant model was extensively modified in order to apply the ECORE module for the simulation of the core degradation behaviour and to improve the numerical stability during core degradation transients. The plant model simulates the thermal-hydraulic behaviour of the reactor pressure vessel, the feed water line and the steam line systems. In addition the emergency and residual heat removal systems, the boric acid control system, the reactor cleaning system, control rod drive cooling system, and the seal water pump are taken into account. The nodalization of the reactor pressure vessel and the steam supply system is shown in Figure 1. The primary loop thermal hydraulics behaviour is described by the following thermo-fluid objects: the downcomer represented by 4 TFOs (TFYD0nRR00 with n = 1 to 4) with the relative re-circulation pumps (only one is shown in the picture); the lower plenum (TFYD00UP01/02/03); 5 reactor core rings (TFYM0nKE00 with n = 0 to 4) with their respective bottom part (TFYM0nKEBO); the core inlet (TFYM00KEIN); the control rod guide assembly (TFYD00SS00); the core bypass (TFYD0nBY02 with n = 0 to 4 and the inlet TFYD00BY01); the upper plenum (TFYD00OP00); the riser (TFYD00SR00); the separator (TFYD00SEI0/IN/ AU/A0/A1/01); the dryer (TFYD00TRIN/AU) and the steam dome (TFYD00SD01/02/03). The four main steam lines are simulated by means of the TFRAn1Z001 object (n = 1 to 4) whereas the turbine bypass lines are represented by the TFRAn3Z001 objects. The 11 SRVs are merged together and represented by the 4 valves in the objects TFTKn1Z211. The 3 pressure relief valves are simulated at the 2nd, 3rd and 4th steam lines by the objects TFTKn1Z001 (n = 2, 3, 4). The steam flow to the turbine, the turbine bypass and the auxiliary steam line supply to the feed water tank are determined in GCSM module and in the thermo-fluid system by the fill components specifically implemented. The feedwater system is also represented in the model consisting of the feedwater tank, the pre-heaters and the four feedwater lines. All the reactor main structures are considered for the analysis of the heat transfer and heat losses as well as all major reactor protection systems, reactor control systems, safety systems and auxiliary systems. As far as the ECORE implementation is concerned, the core is subdivided in 5 concentric rings according to the thermal-hydraulics scheme. Each ring contains either 176 or 174 fuel assemblies (FA) except for the central ring which has 84 FAs. The power radial distribution across the core is prescribed based on data obtained with neutronics calculations performed by means of TRACE-PARCS (Hartmann, 2016). The axial power distribution is assumed to be constant in time and uniform across the core rings. As far as the fuel assembly is concerned, the 9 x 9 configuration is assumed in the present work. FAs consist of 80 fuel rods with one dummy rod filled with water. No distinction is made in ATHLET-CD between different fuel compositions, and all the rods are made of standard UO_2 fuel. The 193 control rods are cross-shaped and are inserted from the bottom of the core via the control rod drive system between 4 fuel assemblies. Each arm of the external stainless steel sheath contains 18 absorber rods made of boron carbide (B_4C) powders surrounded by a stainless steel cladding. As far as the lower plenum modelling is concerned, the new model LHEAD has been adopted in the present work, being more adequate for BWR conditions. LHEAD (Austregesilo, 2015) takes into account the heat conduction in the corium region (liquid melt and crust) derived from an enthalpy conservation equation in a two-dimensional calculation grid in cylindrical coordinates. The convective heat transfer within the liquid melt pool is taken into account by means of an additional heat conductivity term based on correlations for axial and radial Nusselt numbers. The most important advantage of LHEAD is that it is implemented within the ECORE module, thus allowing a full coupling with the thermal-hydraulics and the heat structures describing also BWR internals. Concerning the RPV failure criterion, the simple temperature model is adopted in the present analysis, which establishes the vessel failure when the average temperature exceeds 1600K.

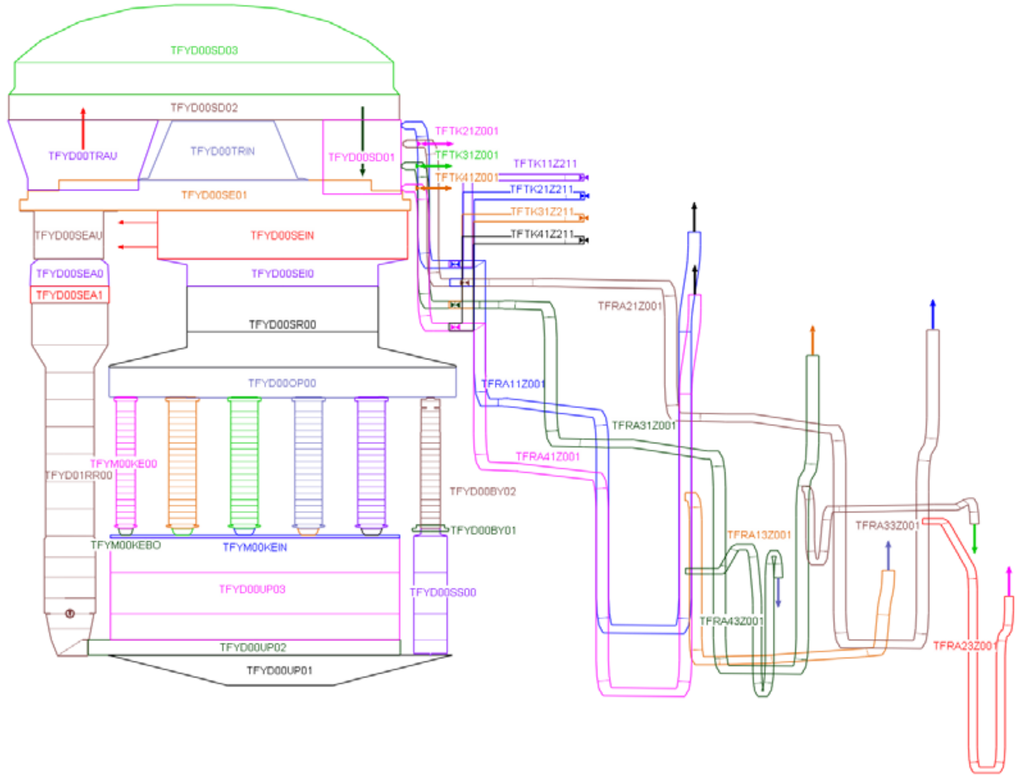

Figure 1 Vertical cut through the German BWR reactor building

3. SELECTED SEVERE ACCIDENT SCENARIO

The definition of the reference scenario for the considered BWR plant is based on PSA carried out by GRS in 1993 (GRS, 1993). These studies have been reviewed in the frame of the WASA-BOSS project (Pohlner, et al., 2014) for the study of possible accident measures to mitigate the consequences of reactor accidents with severe core damage. In the GRS analyses, two different initiating events have a relevant probability to occur: the small break LOCA at the steam line inside the containment which is analysed and discussed in this paper; and the station blackout, which is investigated by other partners within the WASA-BOSS project.

The reference scenario is characterized by a lower frequency of the so-called plant hazard state compared to the station blackout, but it leads to a very short time from the break opening to core uncovering according to the GRS analysis. This hazard state may only occur if a multiple failure of the safety systems is assumed. In particular, all the emergency core cooling injections including both high pressure (HP) and low pressure (LP) systems are assumed as not available as well as the additional heat removal system and all not safety classified and emergency means to feed the RPV. According to the event tree calculated by probabilistic safety analysis and described in (GRS, 1993), after the break opening, SCRAM is actuated by the reactor protection system and the automatic pressure control system is efficient to deal with RPV pressure variations. In the scenario, the failure of the HP injections is assumed at this point. These failures are not enough to lead the plant to any hazard state because RPV feeding can be still provided via the low pressure systems. To this purpose, RPV depressurization is automatically actuated to bring the pressure below 1.4 MPa. Nevertheless, the failure of all LP injection systems, of all not-safety-classified systems and emergency measures is assumed compromising the RPV feeding. These conditions lead to core uncovering and temperature escalation leading to melting of structural materials if no external action takes place. During the transient, one out of three heat removal systems is assumed to be operational in order to deal with the containment overpressure occurring as a consequence of the LOCA by being able to cool-down the suppression pool water inventory via the residual heat removal systems (RHR). To find out which is the most appropriate SAM-measures to prevent or delay the degradation of the RPV, the water injection in the core assuming that one of the low pressure emergency core cooling systems (max. feeding capacity is 600 kg/s) recovers its functionality at different degrees of core damage is investigated. In (Di Marcello, Imke, & Sanchez, 2016), different reflooding cases were defined and investigated with ATHLET-CD:

a) Case 1: water injection is actuated at 10 tons of molten material in the core (at 5500 s) representing a limited core damage state where partial control rods and Zr melting has occurred

b) Case 2: water injection is actuated at 20 tons of molten material (6000 s). The core damage is extended to most of the control rods, canister and cladding material with significant core channel blockage

c) Case 3: water injection is actuated at the onset of material relocation in the lower plenum (40 tons of molten material). The core is in a severe damage state with partial fuel melting and most of the structural components relocated at the bottom of the core and

d) Case 4: water injection is actuated at 70 tons of molten material. In this situation most of the core is molten and already relocated in the lower plenum with extensive fuel damage and formation of ceramic melt.

The cold water used to reflood the degraded core is taken directly from the suppression pool assuming a temperature of around 40 °C. For these investigations, the ATHLET-CD integral model was extended to include the actions and systems needed to realize the SAM-measure mentioned above. With the extended integral plant model, the simulations with the different SAM-cases were performed. The main results will be discussed in the next chapter.

4. MAIN RESULTS OF SA-SEQUENCES WITH ACCIDENT MANAGEMENT MEASURES

A summary of the main results for the studied cases are given in **Table 1**. Based on these results it can be stated that the SAM measure of water injection into damaged BWR-core is effective as accident mitigation only if the reflooding is initiated before or close to the onset of molten material relocation in the lower plenum. Consequently, no RPV-failure is predicted in Case 1 to 3 while for Case-4 a RPV-failure at around 10251 s is predicted. In **Table 1**, you can also see that if no reflooding at all is performed, the RPV will fail. On the other hand, we can observe that the core melting process is stopped by the injection of large amount of water leading to a water level increase and to significant temperature decrease within the core. The water injections will also influence the amount and dynamics of the clad oxidation and hydrogen production during accident progression under reflooding. This effect is more pronounced in the reflooding at 20 tons (Case 2) because the core temperature is much higher than that of Case 1. But for the Case 3 and 4, few additional hydrogen is produced as a consequence of water injection since most of the cladding material is already molten and relocated in the lower plenum and therefore not available for oxidation.

Table 1 main events from the start of core melting for the different core water injection cases.

Event	Time				
	No reflood	Case 1	Case 2	Case 3	Case 4
Start of core melting	4856.5 s	4856.5 s	4856.5 s	4856.5 s	4856.5 s
First occurrence of molten Zr	5180.8 s	5180.8 s	5180.8 s	5180.8 s	5180.8 s
First occurrence of metallic melt	5597.9 s	5600.1 s	5597.9 s	5597.9 s	5597.9 s
Start of melt relocation in the lower plenum	6275.0 s	-	-	6275.0 s	6275.0 s
First occurrence of ceramic melt	7165.9 s	-	-	-	7165.9 s
Vessel failure	10234.0 s	-	-	-	10251.0 s

In **Figure 2**, the predicted maximal core temperature, the hydrogen production and the masses of molten material in the core and lower plenum is shown. Due to the cold water injection, the core temperature is reduced to below 1000 °C in almost all cases except, Case 2. In this case, although the injected water can reach the core also from the bypass flow available thanks to canister box failure, the maximum core temperature remains at about 1700°C due to extended core channel blockages which prevents the cool-down of these regions. On the other hand, for the reflooding in Case 3 and 4, the material relocation in the lower plenum is favourable, because it allows removing part of the existing blockages of the core regions. For the Case 4, the injection of cold water in the core is not sufficient to prevent vessel failure although the progression of core degradation is mitigated leading to core temperature decrease. It is worth to mention that in this case, the reflooding is started when there is already a significant amount of corium relocated to the lower plenum. The lower head and the upper zones of the plenum are completely filled with melt preventing any possibility of temperature mitigation and leading to temperature increase of the vessel and ultimately to its failure. Detailed analysis of these sequences is given in (Di Marcello, Imke, & Sanchez, 2016).

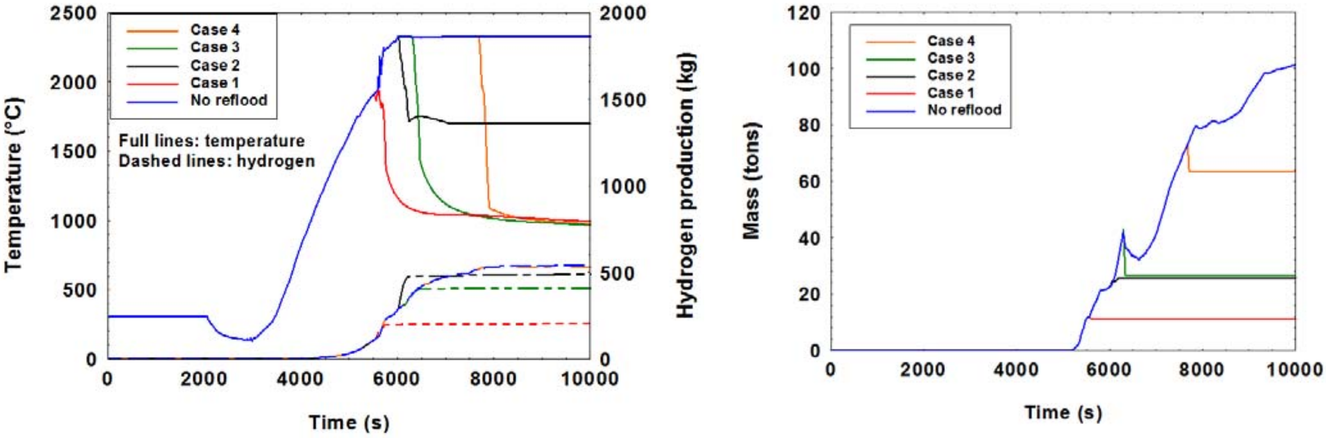

Figure 2: Predicted maximum core temperature and mass of hydrogen production (right side) as well as molten material relocated in the core and in the lower plenum.

5. CONCLUSIONS

The investigations discussed in this paper are mainly focused on in-vessel phenomena that determine the RPV-integrity since an ATHLET-CD version without models for detailed ex-vessel and containment phenomena was used here. Despite of it, the results have demonstrated the capability of the code to simulate severe accident sequences including accident management measures such as reflooding of the core. The authors are aware about the uncertainties in the modelling of the complex in-vessel phenomena including the lower plena behaviour of the molten material and also about the accuracy and the validity of the results presented here. After Fukushima accident a re-focus on severe accident research (modelling, experiments) is going on worldwide in order to solve open issues and to improve the modelling capability and accuracy of the codes. For example, some more work is needed to improve the description of the degradation of fuel rods, control rods, canister box and their interaction, the behaviour of the core support, the relocation of molten material in the lower plenum, etc. These topics will be tackled in the future considering newest experimental results that gradually contribute to an improvement of the current understanding of severe accident phenomena and their implementation in severe accident codes which are used to develop and optimize accident management measures.

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