WASA-BOSS: ATHLET-CD Model for Severe Accident Analysis for a Generic KONVOI Reactor

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1. Introduction

Within the scope of the ongoing joint research project WASA-BOSS (Weiterentwicklung und Anwendung von Severe Accident Codes – Bewertung und Optimierung von Störfallmaßnahmen [1]) an ATHLET-CD model for investigation of severe accident scenarios has been developed. The model represents a generic pressurized water reactor (PWR) of type KONVOI. It has been applied for analyzing selected hypothetical core degradation scenarios, considering application of countermeasures and accident management measures, during the early phase of an accident, as well as the late in-vessel phase, when the core degradation process has already begun. Possible accident management measures for loss of coolant (LOCA) and station blackout (SBO) scenarios are discussed. This paper focuses on the ATHLET-CD model development and results from selected simulations for a SBO scenario without and with application of countermeasures.

2. Accident progression and countermeasures in SBO and LOCA events

The accident management measures, discussed in this “present” paper, focus on hypothetical severe accidents, prior to the reactor pressure vessel failure, and aim at the mitigation of the accident consequences. Figure 1 shows a generic scheme of the application of different measures in small and medium sized loss of coolant accidents (SBLOCA, MBLOCA) as well as in SBO. More detailed information can be found in [2, 3].

In MBLOCA/SBLOCA events at first the secondary cool-down procedure will be activated to remove the residual heat from the reactor via the steam generators (SG) and to reduce both primary and secondary side pressure and temperature. During the secondary cool-down the mass loss on the steam generators’ secondary side will be compensated by the auxiliary feedwater supply. With the emergency core cooling preparation signal, and after sufficient primary pressure reduction, the active and passive emergency core cooling systems will start to inject water into the primary system. After depletion of the flooding tanks the active emergency core cooling systems will switch to sump injection mode. During the early phase of SBO events the residual heat will be removed via the steam generators and the secondary side pressure regulation. Due to the loss of feedwater...
supply the steam generators’ levels decrease and after their depletion the heat removal breaks down. As a consequence the primary pressure and the temperatures start to increase. Later on the primary pressure is limited by the operation of the pressurizer relief and safety valves. The continuous mass loss through the pressurizer valves leads to core uncoveru and heat-up at high primary pressure. Two basic accident management strategies can be applied to reduce the primary pressure and to activate the low pressure safety injection systems (hydro-accumulators), or to reduce the secondary pressure and to feed the secondary side of the steam generators with water stored in the feedwater system or with a mobile pump; ‘primary bleed and feed’ and ‘secondary bleed and feed’, Figure 1. The secondary bleed and feed strategy would be an efficient way to reduce the secondary and primary pressure, and to re-establish the primary to secondary heat transfer (heat sink). If secondary side depressurization (SSD) fails or the strategy is not successful, then primary side depressurization (PSD) can be applied.

In SBLOCA events with failure of the secondary cool-down, for a certain time the primary pressure can remain high, and injection from the low pressure emergency core cooling systems might be difficult to start. In such cases, for reduction of the primary pressure to levels below the set-points of the low pressure injection systems depressurization of the primary side as accident management measure could be applied. In case of multiple failures of essential safety systems in both accident categories an early core heat-up can occur. After stop of the available emergency core cooling systems (due to hydro-accumulators and/or flooding tank depletion or if the switch to sump injection mode fails), the core uncoveru and core damage can only be stopped by an alternative water injection via a mobile pump. For that purpose the primary pressure must be reduced below the pump head of the corresponding pump. A restoration of an initially failed emergency core cooling system with remaining or alternative water sources can also help to mitigate the consequences of the accident, even if the injection starts during the core degradation phase [4].

3. Modeling of the plant
In the following chapter the severe accident code ATHLET-CD is described and the modeling of the plant is given.

3.1 ATHLET-CD code description
The system code ATHLET-CD (Analysis of THERmal-hydraulics of LEaks and Transients with Core Degradation, [5, 6, 7]) designed to describe the response of the reactor, the reactor coolant system, as well as the dynamic behavior of all relevant operational, auxiliary and safety systems during severe accidents, including core damage progression as well as fission product and aerosol behavior, to calculate the source term for containment analyses, and to evaluate accident management measures. It is being developed by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) in cooperation with the Institut für Kernenergetik und Energetiksysteme (IKE), University of Stuttgart. ATHLET-CD includes also the aerosol and fission product transport code SOPHAEROS, which is being developed by the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN).

The ATHLET-CD structure is highly modular in order to include a manifold spectrum of models and to offer an optimum basis for further development (Figure 2). ATHLET-CD contains the original ATHLET models for comprehensive simulation of the thermofluid-dynamics in the coolant loops and in the core. The ATHLET code comprises a thermofluid-dynamic module, a heat transfer and heat conduction module, a neutron kinetics module, a general control simulation module, and a general-purpose solver of differential equation systems called FEBE. The thermofluid-dynamic mo-dule is based on a six-equation model, with fully separated balance equations for liquid and vapor, complemented by mass conservation equations for up to 5 different non-condensable gases and by a boron tracking model. Alternatively, a five-equation model, with a mixture momentum equation and a full-range drift-flux formulation for the calculation of the relative velocity between phases is also available. Specific models for pumps, valves, separators, mixture level tracking, critical flow etc. are also included in ATHLET.

The rod module ECORE consists of models for fuel rods, absorber rods (AIC and B4C) and for the fuel assemblies including BWR-canisters and -absorbers. The module describes the mechanical rod behavior (ballooning), zirconium and boron carbide oxidation (Arrhenius-type rate equations), Zr-UO2 dissolution as well as melting of metallic and ceramic components. The melt relocation (candling model) is simulated by rivulets with constant velocity and cross section, starting from the node of rod failure. The models allow oxidation, freezing, re-melting, re-freezing and melt accumulation due to blockage formation. The feedback to the thermal-hydraulics considers steam starvation and blockage formation. Besides the convective heat transfer, energy can also be exchanged by radiation between fuel rods and to surrounding core structures.

The release of fission products is modeled by rate equations within the module FIPREM, mostly based on temperature functions taking into account the partial pressure of the material gases. The transport and retention of aerosols and fission
products in the coolant system are simulated by the code SOPHAEROS. The nuclide inventories are calculated by a pre-processor (OREST) as a function of power history, fuel enrichment and initial reactor conditions. The release and the transport of nuclides consider the decay heat (α, β, γ) and further decay by means of mother-daughter chains calculated within the module FIPISO.

For the simulation of debris beds a specific model MEWA was developed with its own thermal-hydraulic equation system, coupled to the ATHLET-thermo-fluid-dynamics on the outer boundaries of the debris bed. The transition of the simulation of the core zones from ECORE to MEWA depends on the degree of degradation in the zone. The current code version comprises also late phase models for core slumping, melt pool behavior within the vessel lower head (new module AIDA) and vessel failure. The ATHLET-CD code can be coupled to the containment code COCOSYS, and it is the main process model within the nuclear plant analyzer ATLAS. The ATLAS environment allows not only a graphical visualization of the calculated results but also an interactive control of data processing.

The code validation is based on integral tests and separate effect tests, proposed by the CSNI validation matrices, and covers thermal-hydraulic, bundle degradation as well as release and transport of fission products and aerosols.

For the simulations discussed in this paper the ATHLET-CD 3.0A version was used. For the thermal-hydraulic representation the two-fluid model (6-equation model) was used. For the core degradation phase the ECORE, OREST/FIPISO and QUENCH modules were applied.

3.2 ATHLET-CD model for the generic PWR
A common ATHLET-CD model for simulation of SBO and LOCA scenarios has been developed. All relevant components and systems for performing severe accident analyses have been modeled. The thermal-fluid-dynamic representation of the primary and secondary circuits is a two loop model of a generic KONVOI PWR. The pressurizer and the leak model are connected to the single loop (broken loop); the triple loop represents all intact loops. The leak model can be connected either to the hot leg or the cold leg of loop No 2. In the reactor core bypass is modeled by a separate module FIPISO.

The nodalization scheme of the generic KONVOI reactor is depicted in Figure 3. The input models the emergency core cooling systems – high and low pressure injection systems (HPIS and LPIS) for active, and the hydro-accumulators (HA) for passive safety injection. Each active emergency core cooling system is connected to a flooding tank (time dependent water inventory is calculated by a GCMSM signal). The availability of the emergency core cooling systems can be varied according to the investigated scenario. Additionally, a fire pump has been modeled, which can be connected to either the cold or the hot legs. The secondary cool-down procedure (100 K/h cool-down) and the auxiliary feedwater supply are modeled by FILL objects and a GCMSM control. The reactor protection signals for reactor SCRAM, activation of the safety systems and initiation of accident management measures are modeled by GCMSM signals.

In severe accident scenarios the initial event in combination with multiple failures of essential safety systems finally leads to core heat-up and melt down, either at high system pressure or at low system pressure. An appropriate representation of the early and late phase accident progression and the corresponding phenomena, including countermeasures and accident management measures to be applied, was an essential requirement for the model development, Figure 4.

With the current state of the developed ATHLET-CD model the following scenarios and accident management measures can be simulated:
- Station blackout and total loss of feedwater supply
- SBOCA/MBLOCA with and without secondary cool-down
- PSD and passive feeding on primary side (hydro-accumulators)
- PSD and fire pump injection on primary side (hot leg, cold leg or combined injection).

4. Simulations for a SBO scenario
With the ATHLET-CD model simulations for a SBO without and with
application of accident management measures have been performed. For SBO events the time margins for operators to initiate appropriate countermeasures and the time margins until core heat-up and core damage are the main parameters which have to be assessed by performing accident simulations. For the simulations the following assumptions have been made:

- Initial event: total loss of AC power supply (loss of the offsite electric power supply concurrent with a turbine trip and unavailability of the emergency power supply)
- Unavailability of all active safety systems and total loss of steam generators' feedwater supply
- Only passive safety systems and systems powered by batteries are available
- Pressurizer valves and secondary side pressure regulation are available
- Secondary bleed and feed is not considered
- Criterion for initiation of PSD: $T_{\text{core-outlet}} > 400 \, ^\circ\text{C}$
- Start of fire pump injection into cold leg: after PSD and sufficient pressure reduction
- Fire pump head = 20 bar with a maximal flow rate of 25 kg/s

- Time for transient simulation: 6 hours (for this time period it is assumed that secondary side pressure regulation remains available and that the pressurizer valves can be opened by the operators to depressurize the primary system)
- Simulations for three basic cases have been performed:
  - Case1: SBO without accident management measures
  - Case2: SBO with PSD
  - Case3: SBO with PSD and fire pump injection.

The results of the simulations are shown in Figures 5 to 8. The transient simulation starts from nominal plant conditions. The initial event leads to a turbine trip and stop of the main coolant and feedwater pumps. The reactor scram is initiated by the main coolant pumps’ coast down signal. After that the reactor power drops to the level of decay heat and the residual heat is removed by natural circulation on primary side and via the steam generators by steam dump to atmosphere (partial cool-down and secondary pressure regulation, see Figure 5). Caused by the steam dump to atmosphere and due to the unavailability of the emergency feedwater supply the water on the secondary side of the steam generators continuously evaporates and the steam generators’ levels decrease (Figure 6). After the depletion of the steam generators the primary to secondary side heat transfer breaks down and the primary pressure rises to the set-point of the pressurizer relief valve. After that the primary pressure is kept within the limits of the relief valve’s operation (Figure 5). Due to the rising pressurizer level also two-phase mixture and later on water is released through the valve. For a certain time the released steam is condensed in the pressure relief tank, which is connected to the pressurizer valves (Figure 3). The pressure in the relief tank rises as well. Approximately 4,670 s after the initial event the burst membrane breaks and the steam is released directly to the containment. The continuous mass loss through the pressurizer relief valve leads to core uncovering and core heat-up (Figures 6 and 7). At approximately 9,850 s the cladding temperatures exceed the critical limit of 1,200 °C. Without accident management measures the core degradation process cannot be stopped (Figure 8, Case1).

For Case2 PSD is applied as an accident management measure. At
approximately 8050 s the core outlet temperature exceeds 400 °C and from that criterion PSD by full opening of all pressurizer relief and safety valves is initiated. As a consequence the primary pressure drops rapidly (Figure 5) and after reaching the set-point of the passive emergency core cooling system the hydro-accumulators start to inject water into the primary system. The core recovery and core heat-up can be temporarily stopped (Figures 6 and 7). After the depletion of the hydro-accumulators the mass loss through the pressurizer valves again leads to core recovery and core heat-up. Without additional measures the final core degradation cannot be avoided. Compared to the simulation without accident management measures, the final core heat-up cannot be delayed.

The delayed core heat-up would give the operators more time for additional countermeasures, like restoration of power supply for injection with an active emergency core cooling system or for connecting a fire pump to the reactor cooling system.

For Case 3 a fire pump has been connected to the cold leg of loop No 2 (Figure 3). The injection starts after PSD and as soon as the primary pressure drops below the pump head (generic pump model, the injection starts below 20 bar with a maximal flow rate of 25 kg/s). The results show that in this case the pump is able to compensate the mass loss through the pressurizer valves at the given system pressure. Consequently further core recovery and core heat-up can be avoided (Figures 6 and 7).

To check the ATHLET-CD model for simulations with fire pump injection during the early and late in-vessel phase of the severe accident progression, additional simulations with variation of the fire pump activation signal have been performed. Figure 9 shows the time dependent behavior of the core outlet temperature for the SBO simulation with PSD, but without fire pump injection (Case 2). The dashed lines indicate the time for initiation of fire pump injection, based on temperature criteria: \( T_{\text{core-outlet}} > 400/650/1200/2300 \) °C. With the first two criteria pump injection starts before core degradation. The criteria \( T_{\text{core-outlet}} > 1200 \) °C and \( T_{\text{core-outlet}} > 2300 \) °C indicate pump injection during the core degradation phase. The simulation results with varied pump criteria are depicted in Figures 10 to 12. As a reference the results from Case 1 (SBO) and Case 2 (SBO with PSD) are included. To avoid core heat-up and core degradation, the water injection by the fire pump must be sufficient to compensate the mass loss through the pressurizer relief and safety valves. The pump flow rate depends on the primary pressure (pump characteristics). From Figure 10 it can be seen that immediately after PSD the primary pressure drops very fast. In all simulations after stop of the hydro-accumulators injection for a certain time the primary pressure increases again. As a result the fire pump injection can be reduced or temporarily interrupted (Figure 11). One reason for the rising pressure can
be an imbalance between the heat generated in the core and the heat which can be released through the pressurizer valves. Another reason could be an accelerated, exothermic cladding oxidation at higher temperatures when cold water from the hydro-accumulator and fire pump injection enters the core. Also a cold water injection into a partly melted core could lead to significant steam production and consequently primary pressure increase. As it can be seen from Figures 10 and 11, especially for late phase water injection the primary pressure peak is higher and consequently the fire pump flow rate is reduced. The effectiveness of the fire pump injection has a direct impact on the core degradation process (Figure 12). Compared to the simulations without pump injection (Cases 1 and 2) the results for all fire pump criteria show that the core degradation process can be mitigated. The results indicate that with earlier activation of the fire pump the injection is most efficient. But also with pump activation during the core degradation phase the water injection is still effective. The results of the presented simulations are preliminary. The effectiveness of late phase water injection depends not only on the phenomena and mechanisms which have been described. Also the injection mode (hot or cold leg injection) and phenomena like mixing in the cold leg or countercurrent flow during hot leg injection as well as the complex processes during re-flooding of a partial melted reactor core from bottom or top should be more detailed investigated.

Regarding the calculation time for a transient of 6 hours the calculation time was in the time span between 3 hours and 31 hours on a Xeon@3.30GHz, most CPU time needed for the case with fire pump injection.

5. Conclusion and outlook

The current paper gives an overview on the development of an ATHLET-CD computer model for a generic pressurized water reactor type KONVOI, performed within the frames of the joint research project WASA-BOSS. On the basis of this model preliminary simulations for a SBO scenario without and with application of accident management measures have been performed. The simulation results indicate the applicability of the model for investigation of core degradation scenarios.

As next steps the feasibility and the effectiveness of the primary bleed and feed strategy will be investigated for both SBO and SBLOCA scenarios. Different criteria for PSD initiation and for starting a fire pump injection, before and during the core degradation process, as well as different injection modes (hot or cold leg and combined injection) will be investigated. Additionally, late water injection into a degraded core will also be investigated.

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**Nomenclature**

<table>
<thead>
<tr>
<th>AMM</th>
<th>Accident management measure</th>
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<tbody>
<tr>
<td>ATHLET-CD</td>
<td>Analysis of Thermal-hydraulics of Leaks and Transients (-Core Degradation)</td>
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<tr>
<td>ECC(S)</td>
<td>Emergency Core Cooling (System)</td>
</tr>
<tr>
<td>ECORE</td>
<td>Core Degradation Module</td>
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<tr>
<td>FILL</td>
<td>Special object for simulation of mass-energy sources and sinks</td>
</tr>
<tr>
<td>FIPISO</td>
<td>Transient Nuclide Package</td>
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<tr>
<td>FIPREM</td>
<td>Fission products Release Module</td>
</tr>
<tr>
<td>GC/SM</td>
<td>General Control and Simulation Module</td>
</tr>
<tr>
<td>GRS</td>
<td>Gesellschaft für Anlagen- und Reaktorsicherheit mbH</td>
</tr>
<tr>
<td>HA</td>
<td>Hydro-accumulator</td>
</tr>
<tr>
<td>HECU</td>
<td>Heat Conduction and Heat transfer Module</td>
</tr>
<tr>
<td>HPIS</td>
<td>High Pressure Injection System</td>
</tr>
<tr>
<td>(SB/MB)LOCA</td>
<td>Small-Break/Middle-Break Loss of Coolant Accident</td>
</tr>
<tr>
<td>LPIS</td>
<td>Low Pressure Injection System</td>
</tr>
<tr>
<td>OREST</td>
<td>Stationary Nuclide Package</td>
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<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
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<tr>
<td>PSD</td>
<td>Primary Side Depressurization</td>
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<tr>
<td>RPV</td>
<td>Reactor Pressure Vessel</td>
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<tr>
<td>SA</td>
<td>Severe Accident</td>
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<tr>
<td>SBO</td>
<td>Station Blackout</td>
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<tr>
<td>SG</td>
<td>Steam Generator</td>
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<tr>
<td>SSD</td>
<td>Secondary Side Depressurization</td>
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**References**


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**Codes Developed and Used at GRS.**


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