

ASTEC SIMULATION OF A SBO WITH RE-FLOOD IN A GERMAN KONVOI NPP

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ABSTRACT

This work describes the ASTEC simulation of a station black-out (SBO) scenario in a German 1300 MW_{el} KONVOI NPP with subsequent re-flood of a partly degraded core performed in the frame of the European research project CESAM in the 7th framework program. During the assumed SBO sequence no electrical supply is available on the main voltage levels including the failure of all electrical emergency systems.

The simulation results of ASTEC show the capability of the system code to depict the occurring phenomena and the accident scenario reasonably. Uncertainties can be seen in the simulation of degradation phenomena, especially depicting the re-flooding phase, whereas the general behavior of the calculated NPP is kept. Furthermore, the severe accident management (SAM) measure, the restore of electricity and high pressure injection (HPI), is capable to transfer the system into stable conditions after the desired accident.

KEYWORDS

Plant Application, KONVOI, SBO, SAM, ASTEC, CESAM

1. INTRODUCTION

Since the accident in the nuclear power plant (NPP) TMI-2 occurred 1979 in Harrisburg much emphasis was put in the nuclear safety research activities to understand the phenomena during severe accident progressions. Experiments facing these different phenomena were carried out in several international research programs where new knowledge in the degradation phase during an accident was gained and applied in the development of simulation codes. As the severe accident in the Fukushima-Daiichi NPP occurred in March 2011 this gave again an impact on the nuclear safety research resulting in national stress-tests and international projects. In the European project CESAM (Code for European Severe Accident Management) activities are performed to investigate phenomenological behaviors and assumed accident progressions in NPPs in Europe applying and developing the European Severe Accident Analysis Code ASTEC.

This work describes the ASTEC simulation of a station black-out (SBO) scenario in a German 1300 MW_{el} KONVOI NPP with subsequent re-flood of a partly degraded core performed in the frame of CESAM. During the assumed SBO sequence no electrical supply is available on the main voltage levels including the failure of all electrical emergency systems. The remaining water in the steam generators (SG) and battery capacities capable to keep open the SG relieve valves for around 2 h obtain a heat removal off the primary system not enough to expel the decay heat. The pressurizer safety valves limit the primary pressure increase to around 169 bar. As the pressure cannot be decreased by any other system this initiates loss of coolant into the containment through these valves. Therefore, the loss of coolant cannot be stopped heading towards an accelerated evaporation and core uncover. The core heat-up leads to severe

degradation with relocation and melt formation. In the simulation it is assumed that the electric supply is restored when a degraded core mass of 30 t is reached. Due to the reactivated systems, e. g. auxiliary feedwater, heat removal off the primary system is restored and the primary sided high pressure injection (HPI) starts to inject into the primary system.

2. SEVERE ACCIDENT ANALYSIS CODE ASTEC

The severe accident analysis code ASTEC (Accident Source Term Evolution Code) is a code system to simulate the behavior of nuclear power plants in stable operation conditions as well as during design basis and beyond design basis accidents. It is jointly developed by the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) and the German Gesellschaft für Anlagen- und Reaktorsicherheit (GRS). Different models are applied calculating the thermal hydraulic conditions in the circuits and for the detailed depiction of the reactor and the occurring degradation phenomena up to the reactor pressure vessel failure, following molten core concrete interactions (MCCI) and the behavior in the containment (CPA) as well as possible radioactive releases to the environment. Purpose is furthermore the investigation of the hydrogen (H_2) source term. ASTEC is therefore a highly modularized code with different modules accounting the phenomena described possible to occur up to severe accident conditions. Fig. 1 gives an overview of the ASTEC module structure.

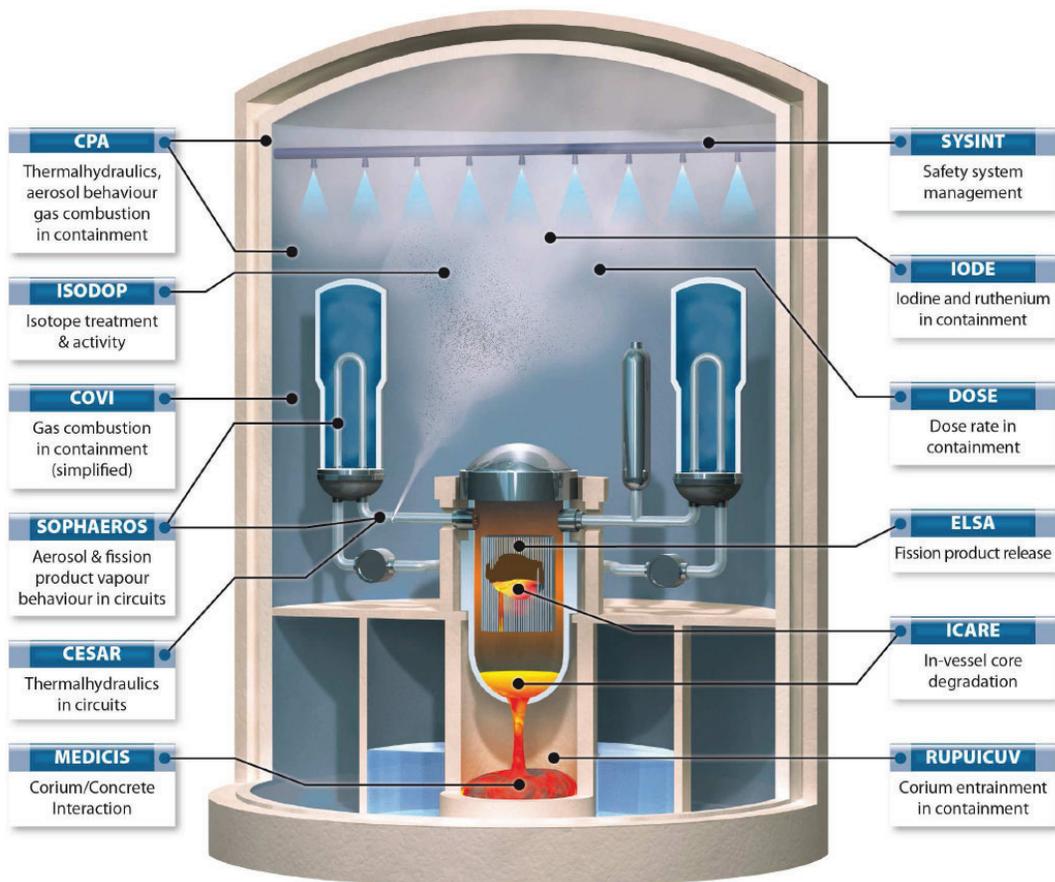


Figure 1 ASTEC Overview [1]

The analysis described in this paper focuses the in-vessel behavior of the KONVOI NPP. Therefore, a more detailed module description is given for the modules CESAR and ICARE. The module CESAR simulates the thermal hydraulic behavior in the NPP circuits based on a five equations approach. In ASTEC ICARE calculates the core behavior when set conditions for an automatic switch to a coupled CESAR-ICARE simulation are reached. Before, CESAR uses a simplified model to simulate the core. ICARE depicts the core with a two dimensional meshing, taking into account degradation phenomena of the former intact geometry. In further improved versions of ASTEC, this coupling is performed from the beginning of the simulation. One important change in the coupling is that CESAR will calculate the thermal hydraulics in the reactor pressure vessel instead of ICARE.

3. MODELLING IN ASTEC

Basis for the plant simulations is a German KONVOI 1300 MW_{el} input deck distributed by the German technical safety organization (TSO) GRS. With this data set a full plant simulation can be performed meaning almost all ASTEC modules are activated for the calculations. The applied models are CESAR, ICARE, CPA, SOPHAEROS, CORIUM, IODE, RUPUICUV, MEDICIS and ISODOP (Fig. 1 assigns the respective purpose).

3.1. Reactor Cooling System (RCS)

A KONVOI NPP has a 4-loop RCS where at one hot leg the pressurizer is connected via a surge-line. In ASTEC this geometry is considered by modelling two loops A and B with weighting loop A three times as they are symmetric and loop B once as the pressurizer is connected to this leg. Fig. 2 gives an overview of the representative modelling of the primary and secondary circuit. The loop A consists of three times ten volumes, the loop B of 13 volumes. The entire primary RCS therefore is represented by 43 volumes and additional nine volumes automatically created by CESAR giving a simple depiction of the reactor core as long as ICARE is not started. These nine core volumes represent the core rings, i.e. the core channels, bypass and downcomer, and the lower plenum mesh defined in ICARE. The hot leg (HL) is represented by one volume which connects the upper core volumes, the Upper Plenum (UPPL) and the Upper Head (UPH), to the steam generator inlet (SGIN). Following the SG is modelled by one volume which is separated into 15 cells representing the SG U-tubes (SGUT1 - 15). The SG outlet (SGOU) collects the down-cooled coolant and builds the beginning of the cold leg (CL) which consists of three volumes (CLS1, CLSS, CLL1). Between CLSS and CLL1 the main coolant pump (MCP) is located. The volume DC_T collects the coolant and transfers it to the downcomer volume. To the hot leg of loop B the pressurizer is connected via the surge-line which is subdivided into two volumes (SURG1, SURG2). The pressurizer is modelled as one volume which type is set to calculate a swollen water level inside the volume.

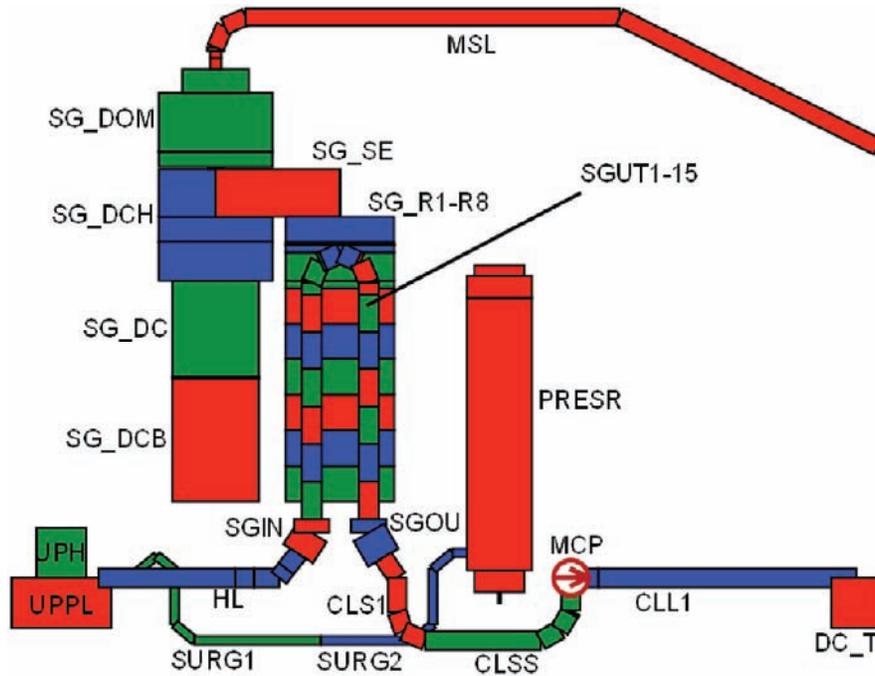


Figure 2 RCS Modelling in ASTEC [2]

The secondary circuit is modelled analogue to the primary side, meaning loop A is weighted three times and loop B once, where the modelling of the feed water tank, the connection pipes as well as the turbine and condenser is realized by connections feeding into the SG or releasing steam. Therefore, the secondary circuit is represented mainly by the SG and main steam line (MSL). The SG consists of three parts, the downcomer, riser and the SG dome with MSL connected. Furthermore, the downcomer is separated into three volumes for the upper part (SG_DCH), the mid part (SG_DC) and the lower part (SG_DCB). In comparison to the U-tubes modelling the riser consists of one volume which is subdivided into eight cells, i.e. per riser cell two U-tube cells transfer heat from the primary to secondary side. At top of the SG riser the SG separator (SG_SE) is located calculating a swollen water level inside the volume meaning the liquid phase is retransferred to the downcomer and the steam to the SG dome which further leads it to the main steam line. All other volumes, except the pressurizer and the SG separator as mentioned above, simulate a homogeneous water-steam distribution.

3.2. Reactor Pressure Vessel

The reactor core is modelled consisting of eight radial distributed core channels by the ICARE module. In the core six channels represent the streaming paths between the fuel element structures inside the baffle. The sept core channel is located between the baffle and barrel, the eighth channel, the downcomer, is situated between barrel and the reactor presser vessel wall. In each core ring (RING1 – RING6) representative rods (ROD1 – ROD12) for the fuel and absorber rods are located. The fuel rods consist of UO₂ pellets (FUEL) and a surrounding Zircalloy-4 (Zry-4) cladding (CLAD). The absorber rods are consisting of the silver-indium-cadmium (AIC) absorber (CROD), a stainless-steel (SS) cladding (GAIN) and a guiding tube (GTUBE) made of Zry-4. Furthermore, Zry-4 spacer grids (GRID) are regularly axially distributed in each core channel. Further modelled core structures are the barrel, baffle and the reactor pressure vessel wall as well as its insulation.

The reactor core is axial divided into 12 nodes. Over the active length of 3.90 m the discretization is done by ten nodes. Above until the core top at 4.34 m one node represents a distribution area before the

CESAR volume UPPL. Below until -0.525 m another node is described mainly containing the lower core plate. The lower plenum is represented by a simplified model creating one mesh enclosed by a discretized reactor pressure vessel wall which is subdivided into eight azimuthal distributed parts and three layers. During the first phase of the transient the module ICARE is not started as conditions leading to core degradation are not fulfilled. To obtain a coupled CESAR-ICARE calculation conditions, i.e. exceeding of threshold values, for the automatic start of ICARE have to be reached in certain CESAR volumes. In particular, the void fraction in the legs has to be larger than or equal to zero ($ALFACD = 0.0$), the percentage of accumulator liquid level must be smaller than or equal to one ($ACCUCD = 1.0$), the temperature in the Upper Plenum volume has to reach 573.15 K ($TEMPCD = 573.15$) and the percentage of the swollen water level in the reactor must be dropped to 0.925 ($SWELCD = 0.925$).

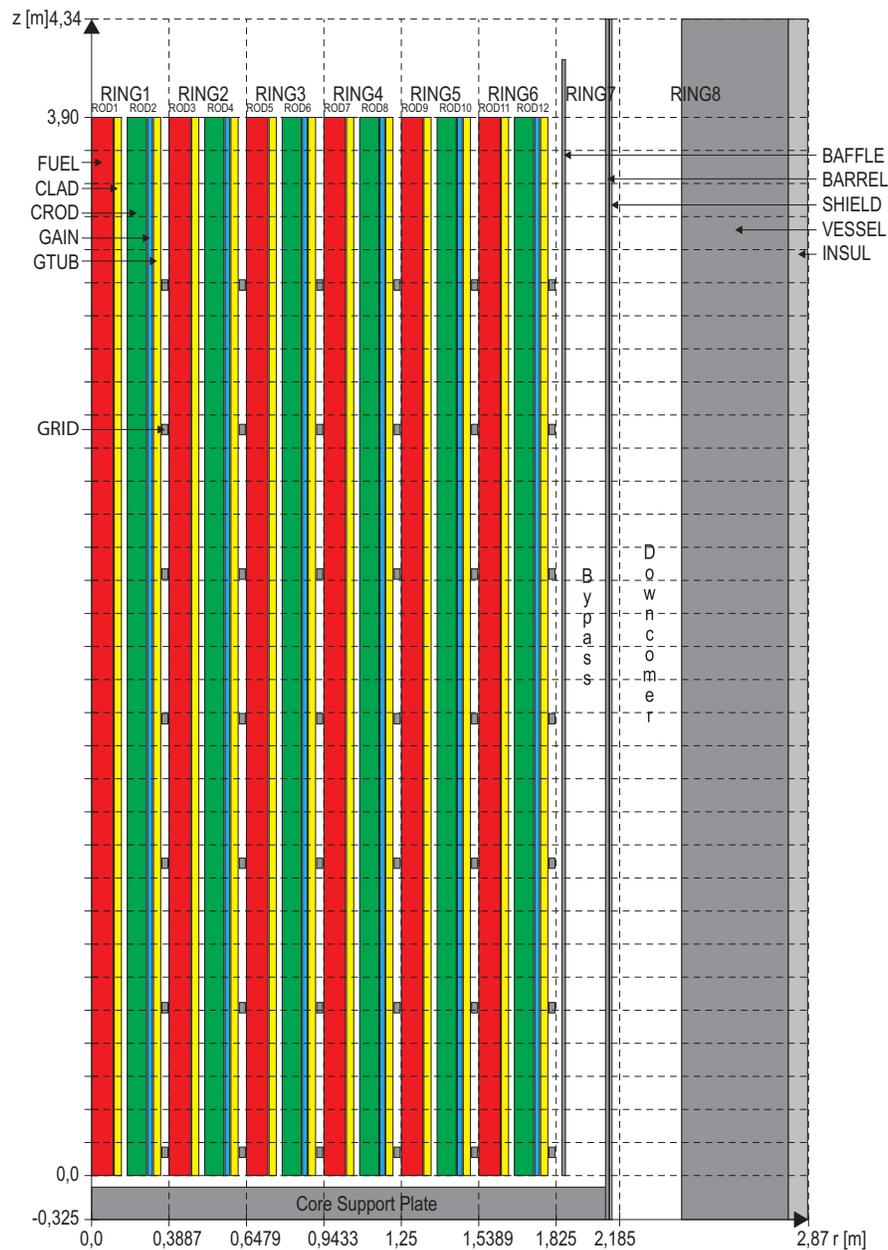


Figure 3 Reactor Core Modelling in ASTEC

Furthermore, ICARE simulates core degradation phenomena occurring during the early and the late phase of an accident. To account the zirconia (Zr) oxidation, which has one of the strongest impacts on the hydrogen generation during an accident, the URBANIC correlation is used. The loss of integrity is taken into account by setting criteria transferring core components into a degraded status once these are reached. For the spacer grids the guiding tubes and the claddings a temperature given performs the transfer, i.e. TGTUB \geq 1730 K, TGRID \geq 2700 K and TCLAD \geq 2700 K. For the cladding another criterion is set, which depends on the zirconia-dioxide (ZrO₂) layer thickness on the outer cladding surface and the temperature, i.e. TCLAD \geq 2600 K and δ ZrO₂ \geq 0.25 mm. Once one of the loss of integrity criteria for the cladding is fulfilled the component starts to degrade and can be relocated. For the fuel pellets the fusion temperature of UO₂ is set for this transition. The simulation of material relocation is done by the MAGMA model. This model uses a porous media approach calculating two-dimensional material movements. Furthermore, the formation and oxidation of debris beds and molten material accumulations can be taken into account by the DEBRIS and POOL model.

3.3. SBO Scenario

The chosen postulated transient scenario is a SBO with failure of all electrical systems except battery capacities which last for two hours based on investigations described in [3]. Therefore, the MCP loose there torque so that they run out, the feedwater system fails as well and all regulations are switched off. SCRAM occurs as the electro magnets holding up the control rods are not anymore electrically supplied. Furthermore, the turbine valves are closed leading the produced steam to be evacuated through safety valves. After a mass of degraded materials in the core and lower plenum of 30 t is reached it is assumed that electricity is restored, meaning the electrical systems restart. When a primary pressure of 110 bar is reached the HPI systems injects water into the primary circuit with a pressure depending mass flow rate. The maximum mass flow rate is three times 77 kg/s through three connections. The re-flood will be started by the heat removal off the primary system via the SG and enhanced by the safety injection.

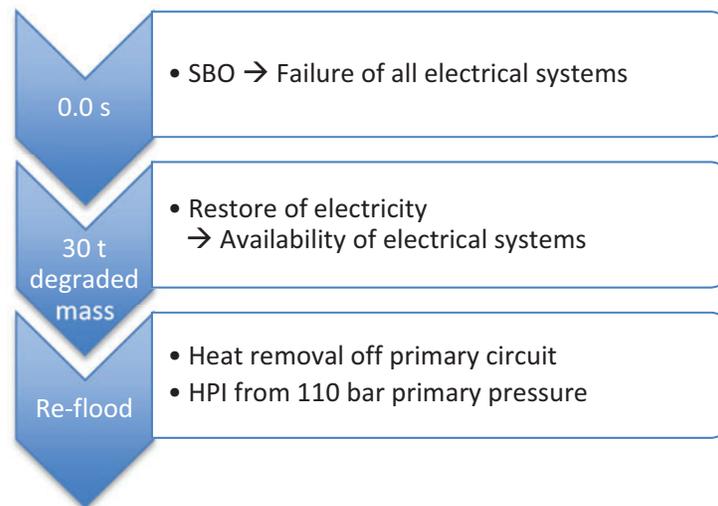


Figure 4 SBO Scenario Scheme

4. SIMULATION RESULTS

In the following the main simulation results are presented and discussed looking at the coolability of a degraded core. Therefore, in the beginning the calculated sequence of events by ASTEC is shortly described, listing the most important instants. Following, the in-vessel behavior, in particular the thermal hydraulics and the core state at different specific times are discussed.

Table I lists important instants with its events calculated by ASTEC. The LOCA is initiated after around 2 h from the beginning of the transient. During further 48 min the coolant mass reduces so far that the conditions for the switch to the coupled simulation are fulfilled. Close after ICARE starts core degradation begins, meaning the material oxidation generates H₂. At ~4:30 h after accident initiation, around 1:30 h after the beginning of core degradation, the electricity is restored as the 30 t of core materials are degraded. After further 1 h the pressure is reduced to 110 bar, so that the HPI is capable to inject into the primary circuit. The given scenario is calculated in a reasonable way with on each other reliable depending sequence of events.

Table I. SBO sequence of events by ASTEC

Time	Event
0 h (0 s)	Sequence initiation: Station Black-Out <ul style="list-style-type: none"> • SCRAM • Switch off SG regulation • Switch off pressurizer regulation • Loss of motor torque for MCPs → Pumps run out • FW cut off • Turbine valves closed • Containment, cavity and RR ventilation stopped
~2:00 h (7096 s) (7201 s)	<ul style="list-style-type: none"> • Pressurizer safety relieve valves open: 169 bar • Closing CSAM100 due to depletion of battery capacity
~2:48 h (10076 s)	Start of ICARE - automatic start
~2:57 h (10621 s)	<ul style="list-style-type: none"> • Beginning of oxidation • Start of structural material release
~3:30 h (12631 s)	<ul style="list-style-type: none"> • First cladding perforation by grid • Start of FPs release from fuel pellets
~4:30 h (16051 s) (16281 s) (16299 s)	Restore of electricity <ul style="list-style-type: none"> • Restart of FW system • Turbine valves opened • Restart of containment, cavity and RR ventilation First total core uncover First material slump in lower plenum
~4:37 h (16411 s)	Start of extra borating system
~5:36 h (20176 s) (20411 s)	Primary pressure decreased to 110 bar → Emergency cooling criterion <ul style="list-style-type: none"> • HPI start Extra borating system stop

In the following selected simulations results are presented focused on the in-vessel behavior. First the thermal hydraulics are described which course lead to core degradation which is explained afterwards.

4.1. Thermal Hydraulics

In Fig. 5 the primary (PRZP) and secondary (SGADOMP) system pressures and the cumulated liquid (VLVltotQM) and steam (VLVstotQM) mass flow rates through the three pressurizer valves (ARV, SV1, SV2) as well as the liquid mass flow rate through the HPI connections are plotted versus time. The primary pressure decreases in the first phase of the transient as there is water in the SG on secondary side which keeps a certain heat removal due to its evaporation (s. SGAL in Fig. 6).

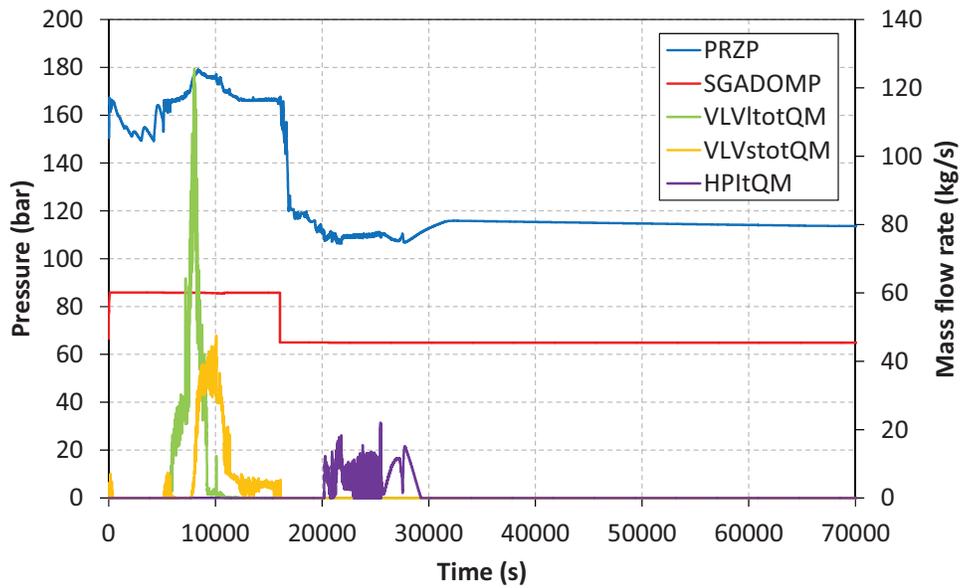


Figure 5 System Pressures and Mass Flow Rates through Pressurizer Valves and HPI

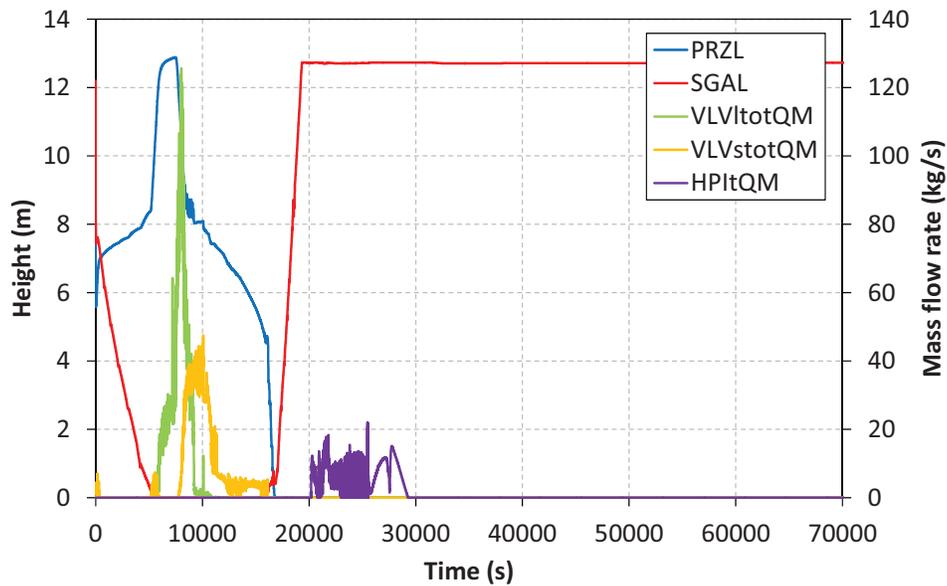


Figure 6 Pressurizer and Steam Generator Level and Mass Flow Rates through Pressurizer Valves and HPI

Once the entire feedwater in the SG evaporated the primary pressure increase is limited by the pressurizer valves ARV, SV1 and SV2 initiating LOCA at 7096 s. First, steam flows out of the valves for a short time as there is a steam cushion in the pressurizer leading to an increasing water level in it (s. PRZL in Fig. 6). Following the mass flow rate through the valves is mainly liquid with loss rates more than 120 kg/s at 8000 s. This peak is caused by the further primary pressure increase due to reaching saturation conditions of the coolant leading to evaporation in the primary circuit. The mass flow through the valves therefore becomes gaseous as phase separation takes place which is indicated as well by the falling pressurizer level (PRZL) in Fig. 6. Following, the valves are capable to limit the primary pressure until the restore of electricity which is triggered by reaching 30 t of degraded materials (s. 4.2) at 16051 s. Caused by the restart of electrical systems, in particular by feeding the secondary side, which can be seen at the SGAL in Fig. 6, the primary pressure is reduced quickly from ~170 bar to ~120 bar due to condensation of coolant. Until 20176 s the pressure is reduced more moderate by heat removal via the SG. From this instant the HPI injects coolant when the pressure is below 110 bar leading to stable conditions. The secondary pressure is given by the SG safety valves during the SBO to ~85 bar and after the restore of electricity by opening valves simulating the steam bypass-station.

4.2. Core Behavior

In Fig. 7 the percentage of the core water level (CORE_L) and the mass of degraded materials in the core (MAGMCORM) and in the lower plenum (METALLPM) can be seen. The core dry-out starts when the coolant reaches saturation state and phase separation takes place in the primary circuit (s. 4.1). Core degradation takes place during this phase until the restore of electricity is triggered by reaching a degraded mass of 30 t. This initiates the re-flood of the core before the HPI, stopping the core degradation after a short escalation through increased oxidation, leading to metallic material relocation into the lower plenum. The core level rises during the condensation in the primary system and is stabilized after the oscillating HPI. The graph for CORE_L shows a value of ~65 % which is caused by the not fully filled downcomer and by taking the average over all core channels in the plot procedure. The core water level

stays above the upper core edge once stable conditions are reached (s. Fig. 11). In total ~31 t of materials are degraded and ~0.5 t slumped into the lower plenum.

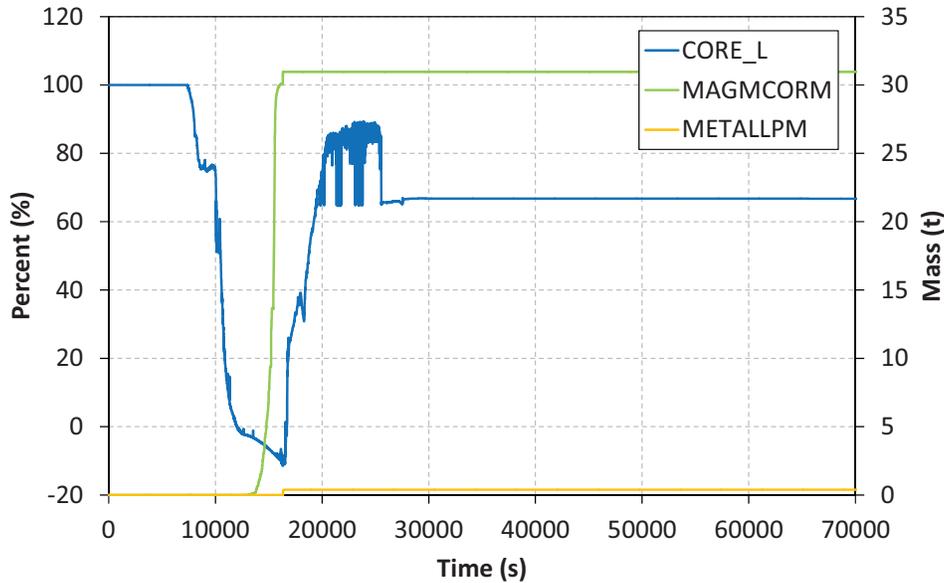


Figure 7 Percentage of Core Water Level and degraded Core Materials

Fig. 8 to Fig. 11 depict the core state at the given instants where important core degradation processes occur (temperature scale in K). The core is nearly fully dried-out at 14401 s and starts to heat-up from the central upper core region leading to melting and relocation in this region. Metallic melting and chemical material interactions forming eutectica cause lower fusion temperatures as given for the rare materials themselves. Looking at the following depicted times it can be seen, that the highest temperatures and therefore the material relocation shift to the outer radii of the reactor core. Meshes with an accumulation of degraded materials can particularly found in RING5 and RING6 with partly blockages. The higher steam availability due to evaporation caused by the re-flood shortly leads to stronger oxidation resulting in the higher magma saturation depicted in Fig. 10 in the central core region and the relocation of metallic material into the lower plenum. The re-flood occurs from the core bottom and top. As the models in ASTEC to simulate these phenomena are complex the calculation time is strongly increased especially during the phase with rising RPV water level. The quench front and the following calculated component temperatures appear reasonable and it can be seen that the temperatures above the water level are reduced compared to the instant before. Fig. 11 shows the core state which is kept until the end of the transient. The locations with higher magma saturation appear more difficult to cool but show a negative gradient over time whereas the rest is cooled down to a stable core behavior. Hydrogen produced due to these degradation processes accumulates to 685 kg.

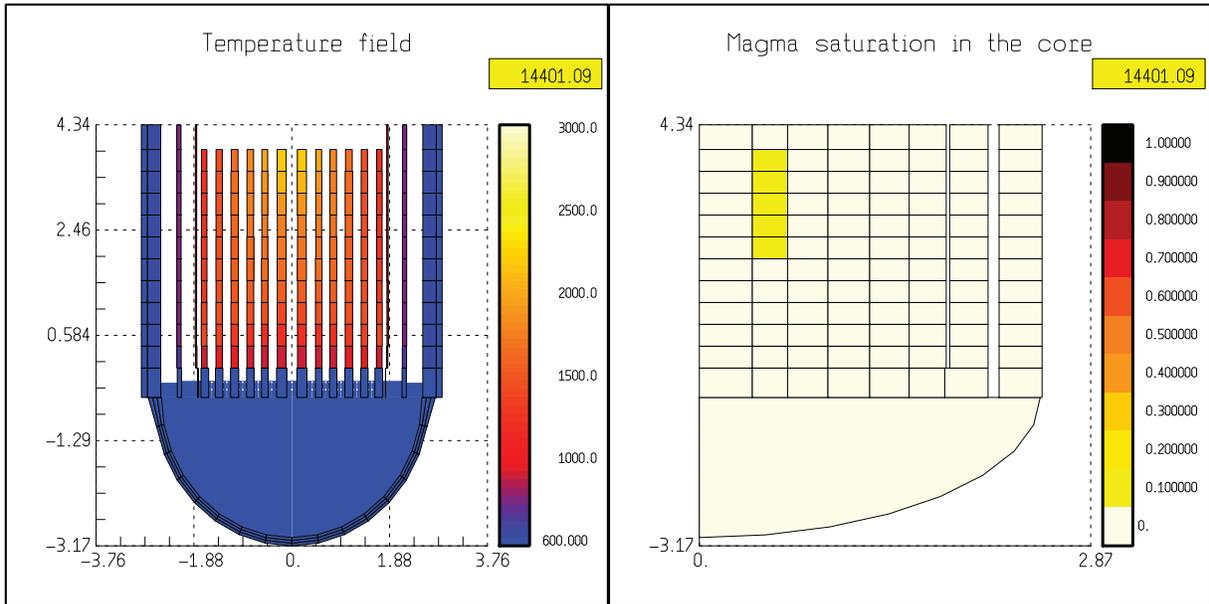


Figure 8 Core State at 14401 s

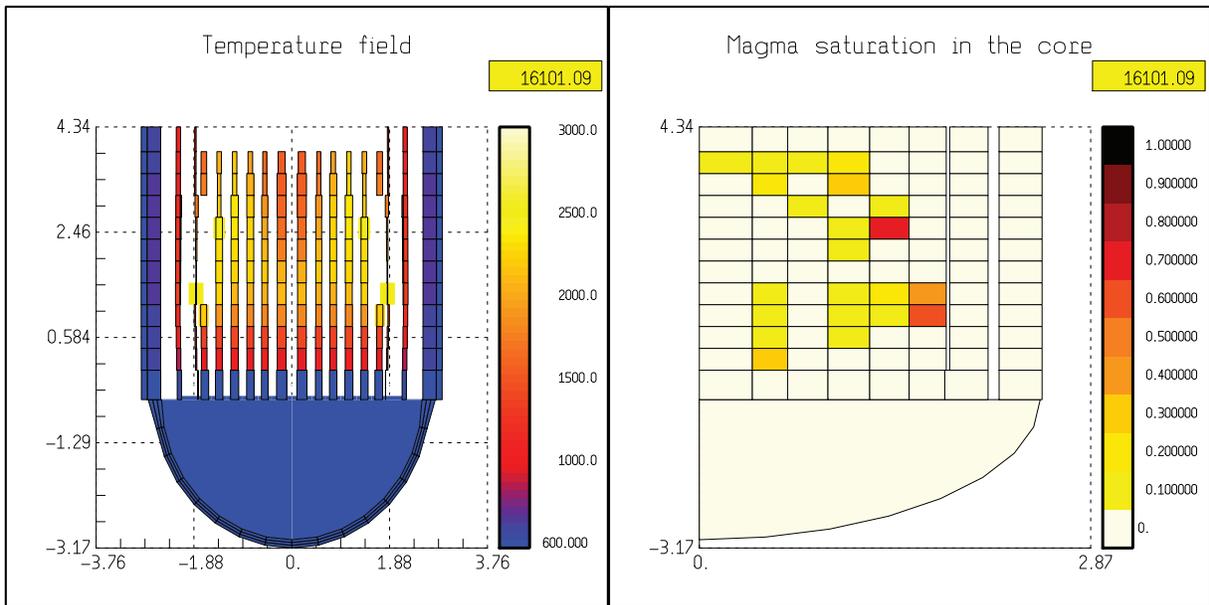


Figure 9 Core State at 16101 s

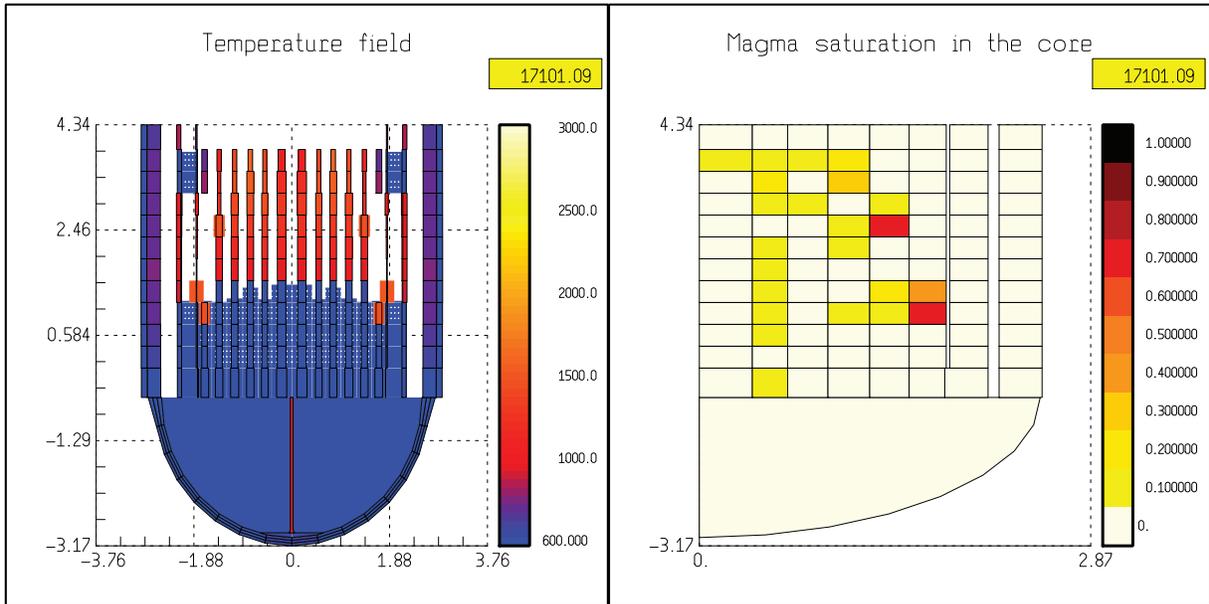


Figure 10 Core State at 17101 s

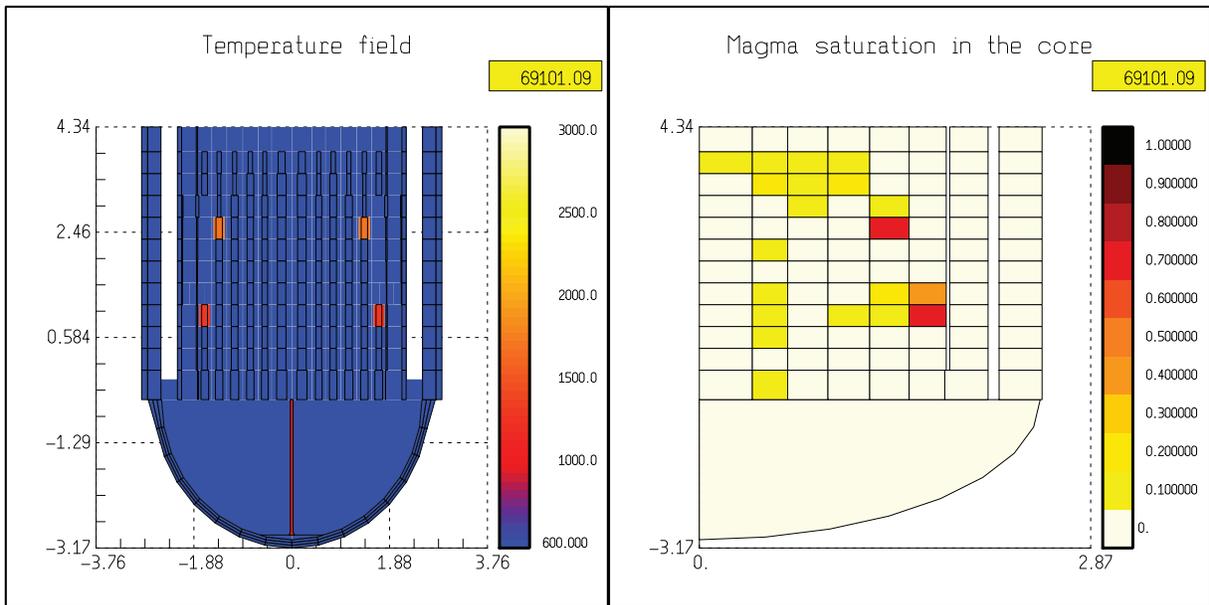


Figure 11 Core State at 69101 s

5. CONCLUSIONS

The given scenario is calculated with an on each other reasonably depending sequence of events. Especially the thermal hydraulic behavior appears consistent in the given context of the postulated scenario with exceptions inside the reactor pressure vessel as water remains on upper elevations in the core not flowing down during the re-flood (s. Fig. 10). Uncertainties can be seen in the simulation of the beginning accident late phase when core degradation starts. The reactor core starts to heat-up from the central upper region with a reliable heat distribution decreasing to outer radii. During the further heat-up the hottest core region shifts outwards and to lower elevations resulting in an uneven distribution. The respective material degradation and relocation follows this shift. This behavior appears unlikely in the given configuration and will be part of further investigations, in particular accounting heat transfer mechanisms in the core, like the radiation modelling. Furthermore, the cool-down of accumulated degraded materials appears to be simulated relatively slow. This will be part of further ongoing studies of the coolability of degraded structures with further improved versions of ASTEC. Overall the assumed SAM measure, the restore of electricity and HPI, is capable to cool-down widest parts of the reactor core and to transfer the NPP into stable conditions, stopping the generation of hydrogen and the emission of radioactive fission products.

ACKNOWLEDGMENTS

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