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APPLICATION OF ASTEC V2.0 TO SEVERE ACCIDENT ANALYSES FOR GERMAN KONVOI TYPE REACTORS

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Abstract

The integral code ASTEC [1] is jointly developed by IRSN (Institut de Radioprotection et de Sûreté Nucléaire, France) and GRS (Germany). Its main objective is to simulate severe accident scenarios in PWRs from the initiating event up to the release of radioactive material into the environment. This paper describes the ASTEC modeling approach and the nodalisation of a KONVOI type PWR as an application example. Results from an integral severe accident study are presented and shortcomings as well as advantages are outlined. As a conclusion, the applicability of ASTEC V2.0 for deterministic severe accident analyses used for PSA level 2 and Severe Accident Management studies will be assessed.

1. Introduction

Within the Severe Accident Research Network (SARNET2) of the European Commission's 7th Framework Program ASTEC (Accident Source Term Evaluation Code) plays a key role as it is designated to become the future European reference integral code. ASTEC covers the same scope of application as the integral code MELCOR, developed by Sandia NL in the US on behalf of the US NRC, which was used for deterministic severe accident analyses at GRS within PSA level 2 studies in the past and is still applied today.

In order to analyse complete severe accident scenarios in a reasonable time frame with the simulation of all important phenomena in the reactor core, the reactor cooling system and the containment, not all physical models used in integral codes can be as detailed as in mechanistic codes. For the purpose of detailed or "best estimate" simulations of severe accidents GRS is also developing the mechanistic codes ATHLET-CD [2] and COCOSYS [3]. In consequence of the different levels of detail of the ASTEC models also different nodalisation rules are valid for ASTEC datasets compared to detailed codes.

The current version ASTEC V2.0, revision 1, is used to simulate different severe accident scenarios for German PWRs of the KONVOI type. In particular comparisons with MELCOR 1.8.6 show still remaining differences in the prediction of several physical and chemical parameters such as thermal hydraulics in the reactor cooling system and the containment, but also in the prediction of the radioactive source term.

2. ASTEC code system

ASTEC is a highly modularized code system. Figure 1 gives an overview of the available ASTEC models. It is common practice to use all of these modules for integrated analyses of severe accidents or only single modules for special purposes like single effect test calculations or particular accident phases. The work presented here aims at the complete simulation of severe accident scenarios using all relevant modules of the ASTEC integral code. In the following the main modules are briefly discussed. CESAR is used for the calculation of primary and secondary circuit thermal hydraulics in the reactor cooling system. The core degradation is modelled by the module ICARE which is not started at the beginning of the scenario. It is started automatically if conditions are reached under which considerable core heat-up followed by core degradation is expected. These conditions concern for example the void fraction or the steam temperature in the core. The release of fission products from the fuel rods modelled in ICARE is determined by the module ELSA, whereas SOPHAREOS is responsible for the transport of fission products, aerosols and noble gases through the reactor cooling system i. e. the CESAR volumes into the containment. The thermal hydraulic behaviour inside the containment is modelled by CPA (Containment Part of ASTEC). The corium entrainment into the reactor cavity of the containment after failure of the reactor pressure vessel is calculated by the module RUPUICUV. During the ex-vessel phase MEDICIS calculates the corium-concrete interaction in the reactor cavity and the release of gases from the concrete as well as the release of fission products from the molten corium into the containment. Inside the containment the module IODE is available for the calculation of the specific behaviour of iodine species and the module DOSE for the calculation of dose rates in the different compartments. The module ISODOP determines the isotope distribution and radioactive decay processes. SYSINT is needed to model systems like pumps, valves, spray systems etc. For dealing with hydrogen deflagration the module COVI and a more detailed model (FRONT) are available.

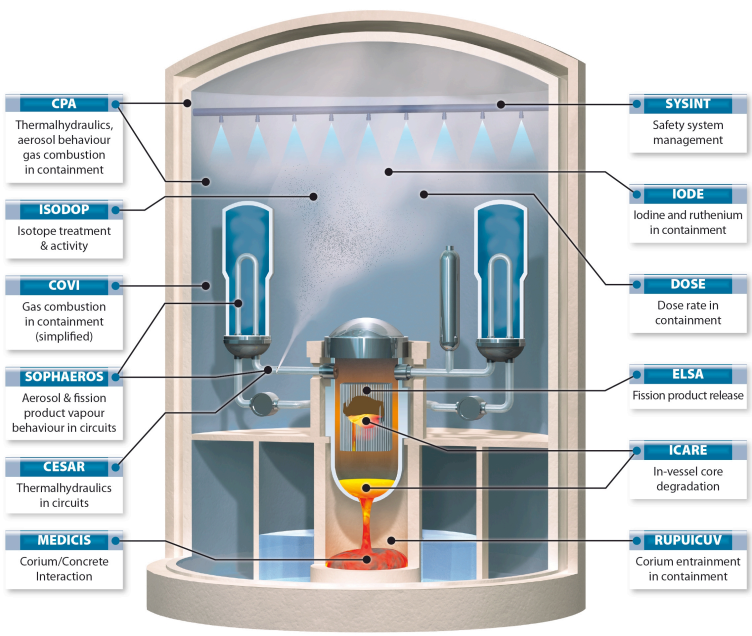

Figure 1 Modular program structure of ASTEC

3. ASTEC dataset of KONVOI PWR

All modules except the hydrogen deflagration module COVI are used in the KONVOI dataset. The different nodalisations for the core, the reactor cooling system and the containment of a generic KONVOI type reactor are briefly described in the following.

3.1 Core and reactor pressure vessel nodalisation

A scheme of the ICARE mesh of the reactor core region is given in Figure 2. The radial subdivision consists of 6 non-identical rings. The red volumes indicate the active core region. Additionally a bypass flow between the baffle and the barrel (blue) and a down comer flow path between the barrel and the reactor pressure vessel wall (white) are simulated. In vertical direction the core is subdivided into 10 zones. Additionally one inlet and one outlet zone (both blue) below and above the active core region is simulated, modelling the inactive part. The outlet zone is connected to the upper plenum zone of the reactor cooling system modelled by CESAR. The inlet zone is connected to the lower plenum volume (blue). The ICARE down comer volume is connected to the down comer top volume modelled in CESAR. Overall the core is modelled by 8 * 12 zones plus 1 lower plenum volume. Thus the ICARE mesh consists of 97 volumes. The nodalisation resembles the core model used in the MELCOR in order to have comparable nodalisations.
As long as no core degradation is expected the core region thermal hydraulics is modelled by CESAR. During this phase CESAR divides the core into 6 core channels, a bypass channel and down comer volume. Additionally a separated lower plenum volume is considered by CESAR. The radio nuclide inventory used in the core is the same as in the MELCOR simulation. The used inventory is representative for a burnup of 40 GWd/MT.

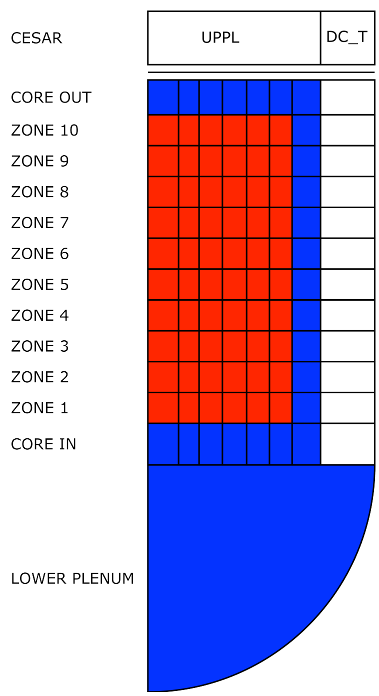

Figure 2 ICARE-nodalisation of the reactor core

3.2 Reactor cooling system nodalisation

Figure 3 shows the primary and secondary side nodalisation of the reactor cooling system loop with pressurizer for the ASTEC module CESAR. The different volumes are indicated by the colours red, green and blue. The dataset uses two loops; one containing the pressurizer and another one representing the other three identical loops. The volumes of the triple loop are internally scaled with a weight factor of 3 in order to model only one of three identical loops and to minimize the calculation effort. In addition the two volumes in the upper part of the reactor pressure vessel that represent the upper plenum (UPPL) and the upper head (UPH) are shown in Figure 3. The hot leg is represented by a single volume (HL1) that ends in the steam generator inlet volume (SGIN). The U-Tubes are modelled by 15 volumes on primary side coupled to 8 volumes representing the riser on secondary side. On the primary side the outlet volume (SGOU) connects the steam generator with the cold leg that is divided into three volumes (CLS1, CLSS, CLL1). The main coolant pump is between CLSS and CLL1. On the steam generator secondary side the down comer is divided into three volumes; the head (SG_DCH), the middle volume (SG_DC) and the bottom (SG_DCB). The riser is modelled by 8 volumes SG_R1 to SG_R8. The steam produced in the riser is separated from the rest of liquid water in the separator volume (SG_SE). The liquid part is distributed to SG_DCH and the steam is distributed to the steam generator dome (SG_DOM). The main steam line is connected at the top. In the loop with pressurizer the surge line is connected to HL1 and is divided into two volumes (SURG1 and SURG2). The pressurizer itself is modelled by a single volume. The relief and safety valves at the pressuriser are connected to the relief tank by a tube, whereas the relief tank is modelled as a part of the containment. The nodalisation was made first in a similar way as the reactor coolant system model in the MELCOR input deck used for comparison. Meanwhile the nodalisation of the steam generator primary and secondary side is more detailed than in the MELCOR deck, while the pressurizer model is simpler.

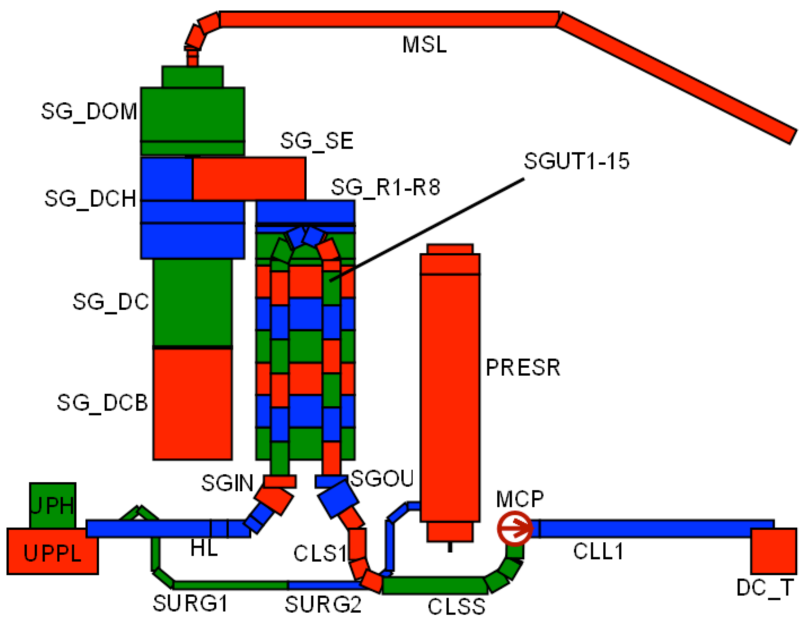

Figure 3 Nodalisation of the reactor cooling system

3.3 Containment nodalisation

The nodalisation for the CPA module (Containment Part of ASTEC) of the containment and the reactor building annulus is displayed in Figure 4. In principle the containment consists of two symmetric halves ‘A’ and ‘B’. The loop with pressurizer and 1/3 of the triple loop are assigned to part ‘B’ of the containment. This becomes obvious by zone DHHKPB that also contains the pressurizer. 2/3 of the triple loop is assigned to part ‘A’. Heat losses from the reactor cooling system to the containment are considered by connections between the CESAR module and CPA according to the assignment to the containment halves. In the figure the eleven equipment rooms are marked by grey colour. During normal operation they are separated by rupture disks (marked by red arrows in Figure 4) from the other nine accessible service compartments.

The annulus is horizontally separated into three volumes. Finally the environment is modelled by a very large zone ENVIRON.
Unlike MELCOR, CPA distinguishes between atmospheric flow paths and drainage junctions. Drainages are indicated by blue colour and atmospheric junctions by black colour in Figure 4. A design leakage is considered from OPERB into RRMITTE. Filtered venting is possible by a connection from OPERB into the ENVIRON zone. Passive autocatalytic recombiners inside the containment are modelled by available built-in correlations. Hydrogen deflagration is not taken into account.
As for the reactor coolant system, the nodalisation was made in a similar way as the containment and annulus model in the MELCOR input deck used for comparison. As requested by CPA the drainage junctions had to be added.

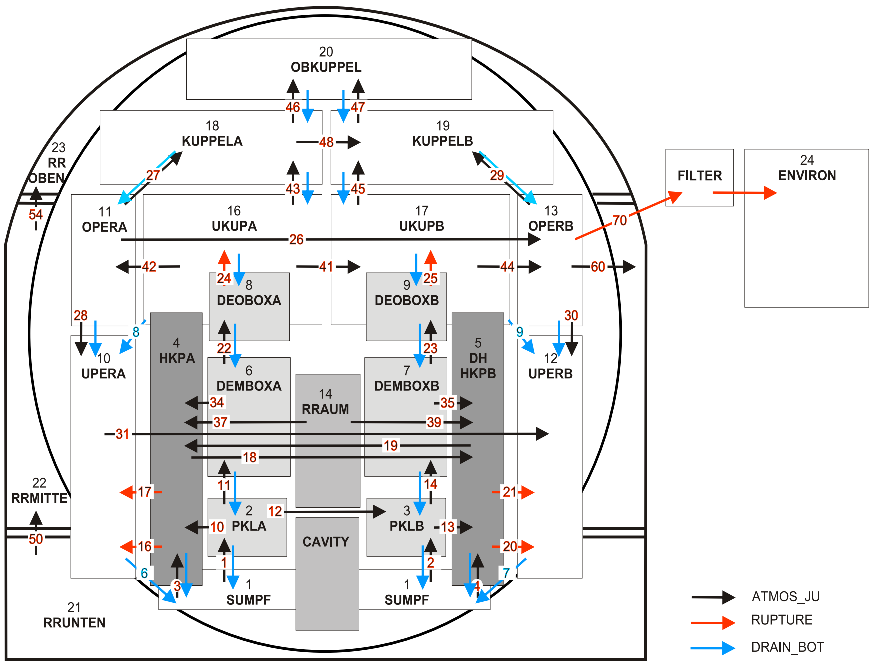

Figure 4   Nodalisation of KONVOI containment

4. ASTEC – MELCOR code to code comparison

4.1 General information

It is common practice to perform both validation calculations for experiments and comparison calculations with detailed codes as well as with other integral codes in order to assure that the partly simpler modelling approach doesn’t lead to incorrect results in the simulation of integral accident scenarios. The focus of this work is a comparison of the reactor behaviour throughout the entire accident sequence. The experience from this work shows that it is very hard to compare core degradation phenomena and the ex-vessel phase after failure of the reactor pressure vessel between two different codes, since small differences already occurring at the beginning of the scenario lead to a diverging sequence. For instance, small differences in the break flow rates as well as differences in the calculated coolant masses of core cooling systems can lead to different conditions at the beginning of core degradation. Another problem is the different modelling approaches especially concerning the fission product behaviour and thereby the determination of the overall radioactive source term. While MELCOR uses 17 fission product classes based on elements with similar chemical properties, ASTEC

uses a more detailed calculation for separate elements and isotopes. This leads to non-consistent comparisons between the MELCOR and the ASTEC inventory.
The work presented here deals with the application of the latest version ASTEC V2.0 revision 1 on the above mentioned scenarios and a comparison to the same MELCOR 1.8.6 YU results for the both scenarios. Details on the MELCOR model used can be found in [4].

4.2 Scenario description

For the comparison calculations a loss of feed water scenario with the simulation of pressure regulation systems and availability of emergency core cooling pumps has been chosen to compare the sequence up to reactor pressure vessel failure. For a comparison of long-term behaviour after vessel failure a simpler small break loss of coolant scenario without consideration of active emergency core cooling pumps has been chosen.

4.2.1 Loss of feed water (LOFW)

The loss of feed water scenario begins with a total loss of feed water supply to all 4 steam generators at 0 s. Reactor scram and turbine trip are initiated (simplified assumption) if the secondary side steam generator water level decreases to 9 m. From this time on no neutron power is considered and the heat source in the core originates only from decay heat. The decay heat can still be removed from the primary circuit over the steam generators. The steam generator safety valves open if the onset pressure of 8.6 MPa is reached. Then a partly shutdown to 7.5 MPa is initiated with 100 K/h. After decrease of the steam generator level below 4 m the main coolant pumps are switched off in order to avoid additional energy input from the main coolant pumps power and extend the dry-out time of the steam generators. When the steam generators are totally depleted the primary pressure increases due to the loss of secondary side heat sink. In a first phase the pressure is regulated by spraying the pressurizer, later by opening and closing of the pressure relief valve with a set point of 16.62 MPa. When the water level in the reactor pressure vessel falls below the so called “min3” criterion (i.e. water level in the RPV at lower edge of coolant line) the pressurizer relief valve and both safety valves are completely opened in order to start a primary bleed and feed procedure – the implemented accident management measure. For the emergency core cooling system it was assumed that only 3 of 4 high and low pressure pumps are available. Thus, only 3 of 4 flooding tanks are used. Besides active emergency core cooling systems hot side accumulator discharge is considered. Cold side accumulators are switched off 500 s after emergency core cooling signal. In order to reach conditions with severe core damages it is assumed that the switch to sump suction of the ECCS system fails and no core cooling system is available from this time on. In consequence core heat-up starts and leads to core degradation later on. After severe core damage molten corium falls into the lower plenum and finally causes a failure of the reactor pressure vessel bottom head and is released into the dry cavity, where molten corium-concrete interaction may occur.

4.2.2 Small break loss of coolant accident (LOCA50CL)

After an opening of a 50 cm² leak in the cold leg downstream the main coolant pump at 0 s reactor scram is triggered if the containment pressure raises about 3000 Pa and the primary pressure decreases below 13.2 MPa. The break flow is not sufficient to remove the decay heat from the primary circuit and steam generators are needed to remove the energy. The automatic shutdown of the steam generators on secondary side is considered with 100 K/h. In this scenario only passive emergency core cooling systems i. e. accumulators are available. Accumulator injection starts later than 500 s after emergency cooling signal. Therefore cold sided accumulator injection is avoided. Due to the failure of active emergency core cooling systems core degradation conditions are reached early. After reactor pressure vessel failure, the corium slumps into the cavity, where molten corium-concrete interaction is calculated by the MEDICIS module in ASTEC.

5. Results of code to code comparison

5.1 Loss of feed water (LOFW)

The pressure in the primary and secondary reactor cooling system is presented in Figure 5. A comparison of MELCOR with ASTEC results is given. After reactor scram the secondary side pressure rises and regulation to 7.5 MPa is initiated. Until the steam generators are empty the primary pressure is regulated by pressurizer spray system. After the total dry-out of steam generators heat removal is no longer possible and the pressure is regulated by opening and closing of the pressurizer relief valve. After reaching the “min3” criterion the primary pressure is decreased by the opening of the pressure relief valve as well as both safety valves. Up to this point MELCOR and ASTEC results are in good agreement. After depressurisation of the primary circuit ASTEC calculates in contrast to MELCOR only a weak cooling of the secondary side steam generator. This is due to a not sufficient circulation of coolant in the primary circuit. In consequence ASTEC calculates only a slow decrease of secondary pressure after 7000 s. A comparison of the dry-out behaviour of the steam generator levels is given in Figure 6. The initial decrease of the water level directly after reactor scram in ASTEC is greater than in MELCOR. In consequence the dry-out in ASTEC is experienced earlier. The level decrease behaviour up to 2000 s is similar. In the ASTEC calculation the steam generators are empty 500 s earlier than in the MELCOR calculation. This behaviour shows that already small differences at the beginning of a scenario can have a large influence on later results.

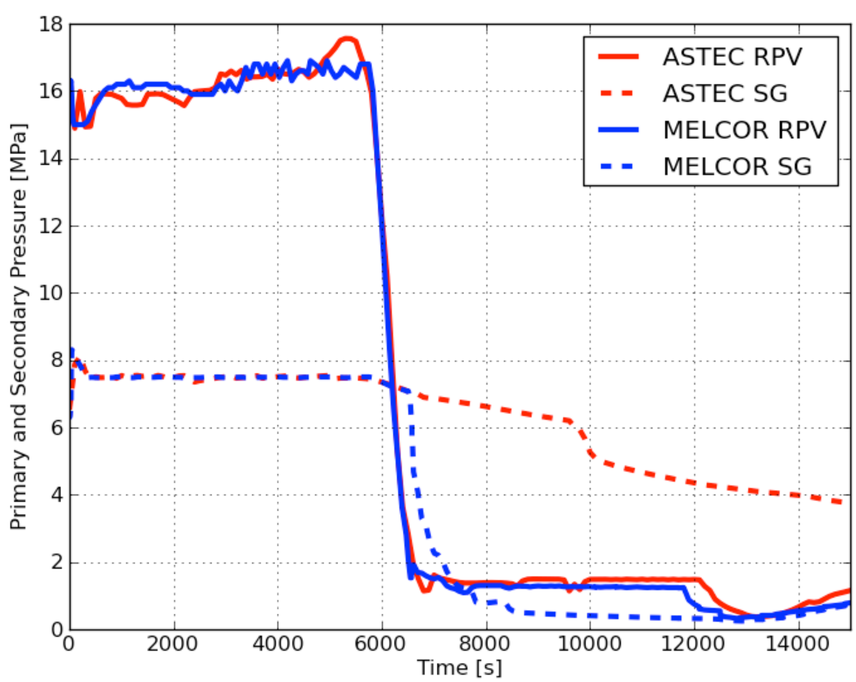

Figure 5 LOFW primary and secondary pressure

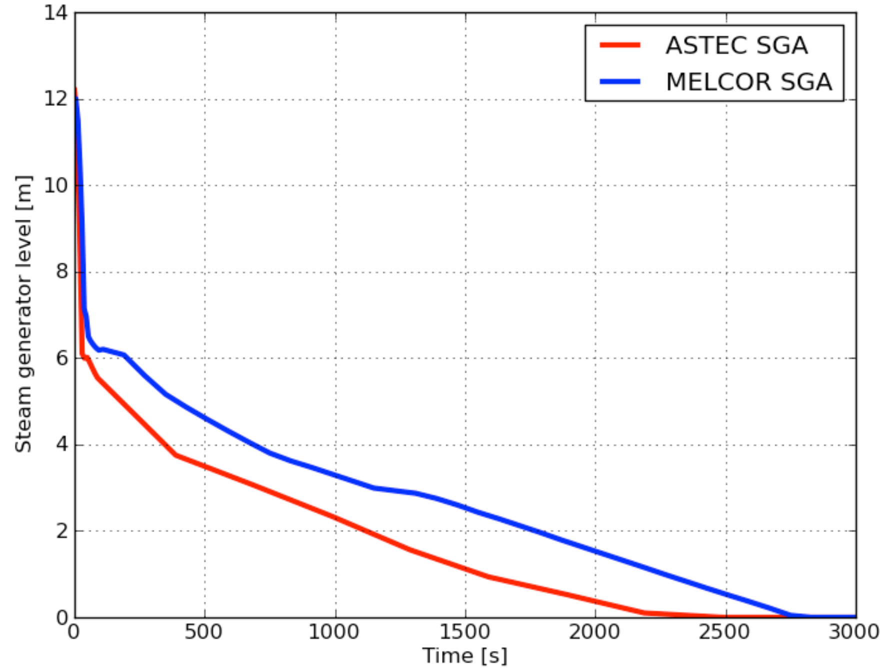

Figure 6 LOFW steam generator water level

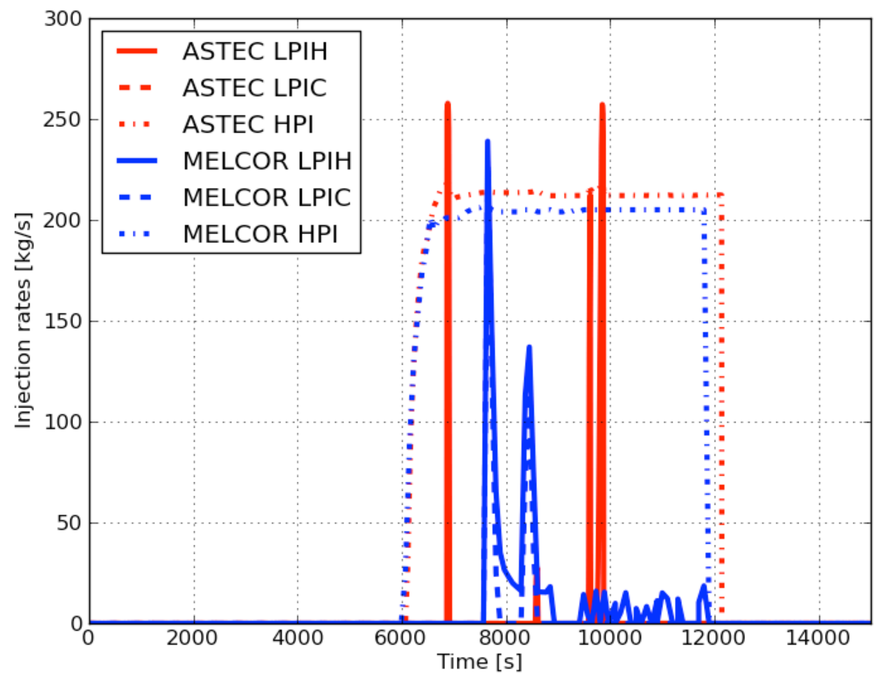

Figure 7 LOFW high and low pressure injection rates

Figure 7 shows the injection rates of the high and low pressure injection pumps. In both calculations 3 of 4 pumps are available for high and low pressure injection from 3 of 4 flooding tanks available. For simplicity reason the ASTEC low pressure injection is only to the hot leg side, while the MELCOR low pressure injection is split to the hot and the cold leg. Both codes predict a high pressure injection with about 205 – 210 kg/s over a period of 6000 s. The low pressure injection is only partly active, because the high pressure injection is sufficient to reach the delivery head of the low pressure injection pumps. In MELCOR the low pressure injection is longer active than in ASTEC. Therefore, the high pressure injection in ASTEC is slightly stronger compared to MELCOR. Overall, both codes are in good accordance concerning the injected water inventory.

5.2 Small break loss of coolant accident (LOCA50CL)

The pressure behaviour in the primary and secondary circuits during the loss of coolant scenario is given in Figure 8. After break opening the depressurization of the primary is fast and reaches the secondary pressure during the first 100 s. The secondary pressure decreases due to the shutdown with 100 K/h. The thermal coupling with the primary system leads to an equal pressure on primary and secondary side. Overall the pressure behaviour in ASTEC and MELCOR is in good accordance during the first 10000 s. The mass of coolant released over the break is displayed in Figure 9. The total mass is very similar in ASTEC and MELCOR during the whole accident sequence. But the distribution in steam and water is different. While MELCOR releases only 240000 kg of water, ASTEC releases 290000 kg. The released steam mass is the opposite. MELCOR releases 160000 kg steam and ASTEC only 110000 kg. This also leads to a higher energy loss over the break in the MELCOR calculation.

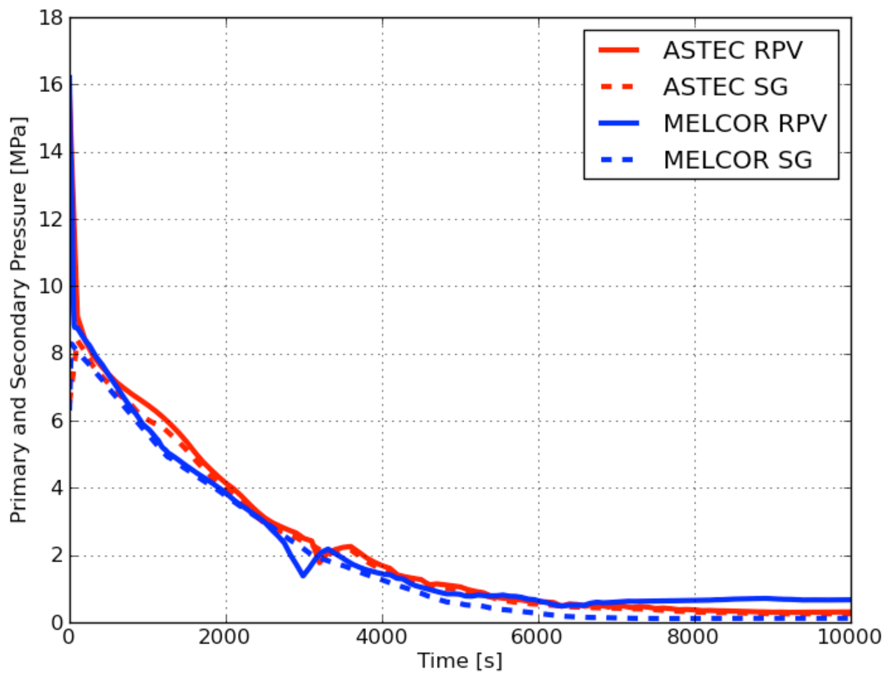

Figure 8 LOCA50CL Reactor cooling system pressure

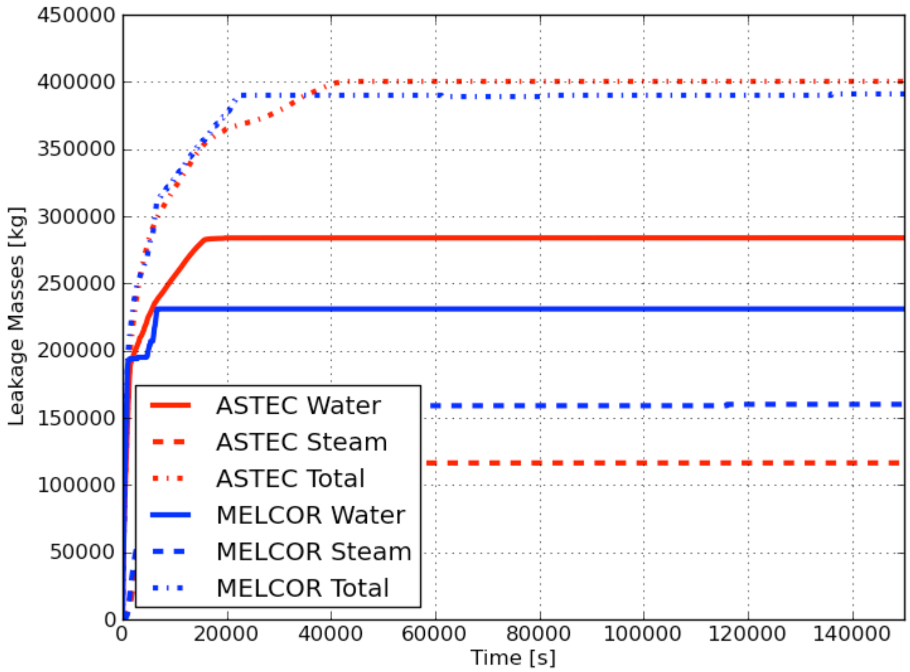

Figure 9 LOCA50CL Fluid loss over break

In the following some results for the ex-vessel behaviour after failure of the reactor pressure vessel of the accident scenario are given. There is a large difference, when the failure of the reactor pressure vessel occurs. ASTEC calculates 42571 s and MELCOR calculates a much earlier failure at 24360 s. Figure 10 shows the hydrogen masses released to the containment. MELCOR calculates generation of hydrogen already 2350 s after opening of the leakage, while ASTEC predicts hydrogen release only at 16000 s. This is explained by a different start of the core degradation phase. As the ICARE module is not able to calculate re-flooding of an already degraded core, the core degradation phase is disabled until accumulator injection stops. This also explains the slower accident progression calculated by ASTEC. While the MELCOR calculation predicts only 314 kg of hydrogen during the much shorter in-vessel phase, ASTEC releases 665 kg in-vessel. During the molten corium-concrete interaction more hydrogen is produced. Up to 150000 s ASTEC predicts nearly 3500 kg while MELCOR calculates only about 2700 kg. Up to that point 1800 kg for ASTEC and 1900 kg for MELCOR are recombined by the passive autocatalytic recombiners installed inside the containment. For both calculations the catalytic process stops at 110000 s due to the lack of oxygen available in the containment atmosphere.
Figure 11 displays the erosion in the cavity caused by molten corium-concrete interaction. Due to the different times the reactor pressure vessel failure occurs the onset of the erosion is different. The vertical erosion in MELCOR starts directly after vessel failure. At the beginning of the ex-vessel phase ASTEC calculates only weak erosion. Stronger erosion only occurs at about 50000 s that is 8000 s after vessel failure. This is explained by the process of corium relocation from the vessel into the cavity. Right after vessel failure ASTEC predicts only a partial slump of corium. Ongoing relocation occurs during the following period of time until enough corium is inside the cavity that leads to higher temperatures and intense erosion at 50000 s. After 150000 s ASTEC calculates about 2.2 m vertical erosion while MELCOR only expects 1.5 m in vertical direction. The predicted radial erosion of both codes is 1.5 m at 150000 s.

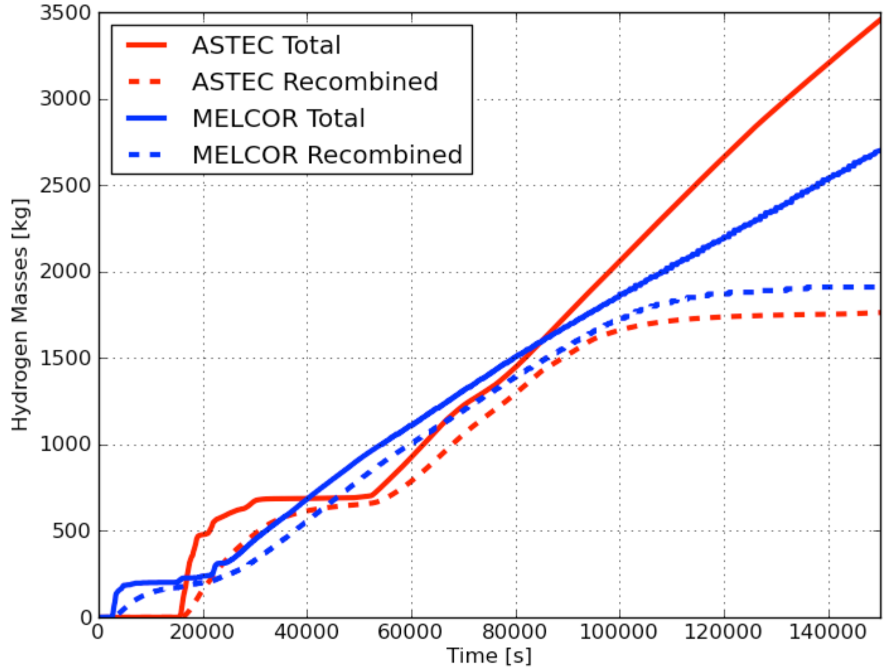

Figure 10 LOCA50CL Containment hydrogen

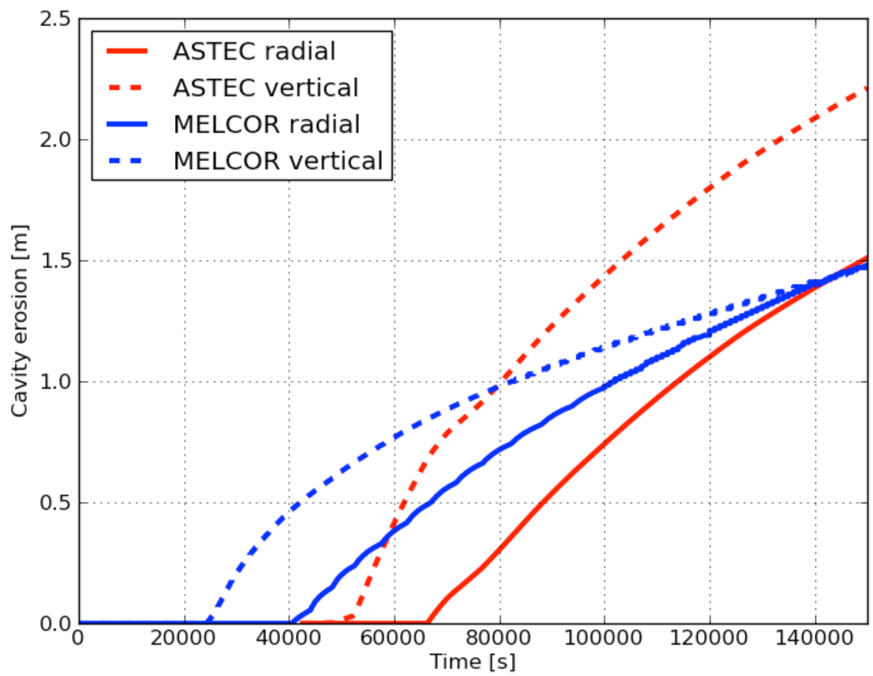

Figure 11 LOCA50CL Cavity erosion

6. Conclusions

Until 2001, the MELCOR code was used at GRS for the calculation of deterministic accident scenarios in the frame of probabilistic safety analyses level 2 for PWR. Since 1994, GRS is developing together with IRSN the European integral code ASTEC to be used as integral tool for probabilistic safety analysis at GRS.

In order to assess the ability of the current version ASTEC 2.0, revision 1 to be used in such studies, a comparison between MELCOR and ASTEC has been performed for two different accident scenarios in a PWR. A transient scenario with the consideration of regulation devices and passive and active emergency core cooling systems has been chosen for the comparison of the in-vessel phase. The long term comparison also considering the ex-vessel phase after failure of the reactor pressure vessel has been conducted on a small break loss of coolant scenario without any active emergency core cooling systems.

Both calculations show the general applicability of ASTEC to the calculation of complete accident scenarios from the initiating event up to the release of radioactive material into the environment including the phase of core degradation, calculation of reactor pressure vessel failure and also the phenomena during corium-concrete interaction in the reactor cavity. A weak point of ASTEC still is the inadequate simulation of re-flooding during core degradation. This will be solved in the next version 2.1 of ASTEC by a restructured coupling of the CESAR and ICARE modules.

However, the comparisons show strong deviations already at the beginning of an accident scenario. Thus, it is hard to compare and assess results in the core degradation or in the ex-vessel phase between two different integral codes if already the initial conditions vary for such models.

Concerning the behaviour of radioactive material, ASTEC uses a detailed elemental description for the release of fission products, while MELCOR uses 17 different classes for a set of similar elements. In addition to the uncertain initial conditions for the fission product behaviour also different model approaches complicate the comparison of both codes.

To overcome these problems GRS is planning to perform comparison calculations especially for the in-vessel phase with the detailed code ATHLET-CD coupled to COCOSYS. This will help to extend the user experience with the ASTEC code and to minimize or at least assess the model uncertainties during the in-vessel phase. This way ASTEC will become an integral severe accident simulation tool used in probability safety analysis at GRS.

7. References

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