Analyses of Severe Accident Sequences During Shutdown and Caused by External Hazards

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Abstract: According to the German regulations for periodic safety reviews it is obligatory for each nuclear power plant to perform a Level 1 PSA for full power and shutdown operating conditions and for events caused by plant-external hazards. In contrary, a Level 2 PSA has to be performed only for full power operating conditions. The German regulatory body therefore supports a project with the objective of closing this gap of knowledge. First, a limited set of scenarios covering most of the relevant scenarios with respect to the time scale of the physical effects to be expected, the pressure buildup in the containment and the source term has to be identified. In order to calculate the set of scenarios by the computer code MELCOR the plant has been modeled in a plant-specific input deck and some scenario-specific settings need to be defined. Then the scenarios will be calculated by MELCOR and analyzed accurately regarding the relevant physical effects including core melting and the release of radionuclides. The results of the deterministic analyses will support development of a probabilistic event tree approach and recommendations for prevention and mitigation of such accidents.

Keywords: Level 2 PSA, shutdown modes, MELCOR, severe accident analyses

1. INTRODUCTION

The present obligation in Germany to perform Level 2 PSA with the only focus on full power operation modes is based on the assumption that pressure and decay heat are quite low at shutdown operational conditions. Despite these facts pertinent analyses have shown that shutdown modes represent a significant contribution to the overall core damage frequency, e.g. due to limited availability of safety systems. Furthermore, it is obvious that external hazards can cause damage of the relevant barriers.

Since the German PSA Guide [1] does not require performing Level 2 PSA for shutdown modes and external hazards, research and development (R&D) activities recently performed by GRS and depicted in this paper can be subdivided into the following five major parts:

1. Identification of the state of the art considering already performed studies in respect of shutdown operational modes;
2. Identification of relevant sequences covering all other sequences that lead to similar core damage states and determination of the initial and boundary conditions;
3. Analyses of the relevant accident sequences (PWR / BWR) using the integral code MELCOR;
4. Conclusions based on the above mentioned analyses concerning:
    a. Phenomena during accidents caused by external hazard or during shutdown operational modes;
    b. Behavior and release of fission products;
    c. Designated and possibly additional emergency procedures;
5. Quantitative assessment of the significance of accident sequences during low power and shutdown operational modes in comparison to sequences during full power operation and plant internal initial events including the influence of uncertainties on the results.

This paper will give an overview of the work done so far.

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2. IDENTIFICATION OF RELEVANT SEQUENCES

2.1. Shutdown Operational Modes

To identify relevant sequences of accidents for Level 2 PSA it is self-evident to use the results of an appropriate Level 1 PSA that depicts sequences leading to so-called system damage states, which are defined as states that lead to core damage if preventive measures do not succeed.

There is a pre-defined interface between PSA Level 1 and Level 2 characterizing the physical and technical state of the facility. Whilst using the information delivered by this interface it is possible to continue these states to the scope of Level 2 PSA. This interface is defined in the technical document on PSA methods [2] supplementary to the German PSA Guide [1] intended to be used for full power operation. Hence before this interface can be used it has to be extended according to the characteristics of low power and shutdown operational modes or external hazards. The extensions, necessary in this project, concern conditions such as an open RPV (reactor pressure vessel), time after shutdown, state of the reactor protection system or the water level in the refueling cavity. This extended interface has been applied to Level 1 PSA [3] that had been performed for a PWR of KONVOI type, also in the context of a research project to evaluate sequences leading to core damage if no preventive measures are executed. Using this Level 1 PSA, it has been possible to systematically assign the entire system damage states to newly defined core damage states. The transition from system damage states to core damage states requires some assumptions for failures such as a not-initiated primary depressurization. If the primary pressure release is available, some further unavailabilities as those of the residual heat removal (RHR) systems has to be assumed because this system, if it is intact, it could inject or remove the decay heat in case of a successful primary pressure release. So the system damage states with the additional assumptions are summarized to a certain set of core damage states.

The next step is the choice of a set of relevant sequences leading to core damage, which is intended to be calculated and analyzed. The claim that has to be met by this set of sequences is to cover all other relevant sequences in respect of their frequency but also in respect of the expected severity of their consequences. This process is mainly based on the quantification of the system damage states, the obvious extrapolation of the sequences in consideration of the state of the facility and on expert judgment. Furthermore, system damage states considering phenomena of deborated primary coolant and sequences in the spent fuel pool are not considered within this selection. The reason for the exclusion of the deborating events is the absence of expected fuel element damages, which is described in [4]. The events related to the spent fuel pool are subject of another R&D project that GRS is working on.

In the case of BWR-type nuclear power plants the KWU-type BWR72 is chosen to be the reference object because it is the only facility representing a BWR in Germany that is in service. Screening the given documents (the most relevant is [5]) a set of four relevant initiators has emerged:

1. Loss of the modified heat removal during cool down,
2. Incorrect injection into RPV,
3. Leakage at the flood compensator,
4. Leakage at the bottom of the RPV due to dismounting a circulation pump.

The sequences leading to a set of six core damage states are selected according to the most evident system-technical states. The chosen sequences also cover a broad range of severe accident progression due to very different initial states including e.g. a filled and a dry RPV which may affect the access of atmospheric oxygen. Furthermore, the time since shutdown ranges between $16\,\mathrm{h} \leq \Delta t_{\mathrm{scram}}^{\mathrm{shutdown\ mode}} \leq 200\,\mathrm{h}$.

2.2. External Hazards

In the case of external hazards, literature research provided scenarios that comprise LOCAs and transients which should be controlled by the design features of the facility. Even in the case of aircraft crashes, earthquakes, floods or blasts caused by explosions no differences between these scenarios and known ones with internal initiators and induced additional unavailabilities have been identified. That means that no further phenomena emerged that should be studied. According to the objectives of this project to extend the knowledge in the context of shutdown operational modes and external hazards, nevertheless some scenarios are created disregarding the corresponding probability. This is the reason that the basic scenario of a station blackout is chosen because it is well known and thus comparable to former analyses using another input deck for MELCOR. Additional to this station blackout scenario, further scenarios are defined with some additional damages or unavailabilities respectively. The station blackout is as far as possible derived from the external hazard as well as the additional damage. The selected sequences comprise earthquakes and aircraft crashes onto the reactor building and the reactor auxiliary building. Damages are assumed at the primary circuit (LOCA) as well as at the reactor building and the containment respectively or reclusive at the venting system in case of a PWR. The scenarios related to BWR consider only earthquakes with additional postulated damages at a feed water line within the containment and low level leakage at the suppression chamber due to a rupture in the residual heat removal system.

3. DESCRIPTION OF THE MELCOR INPUT DECK

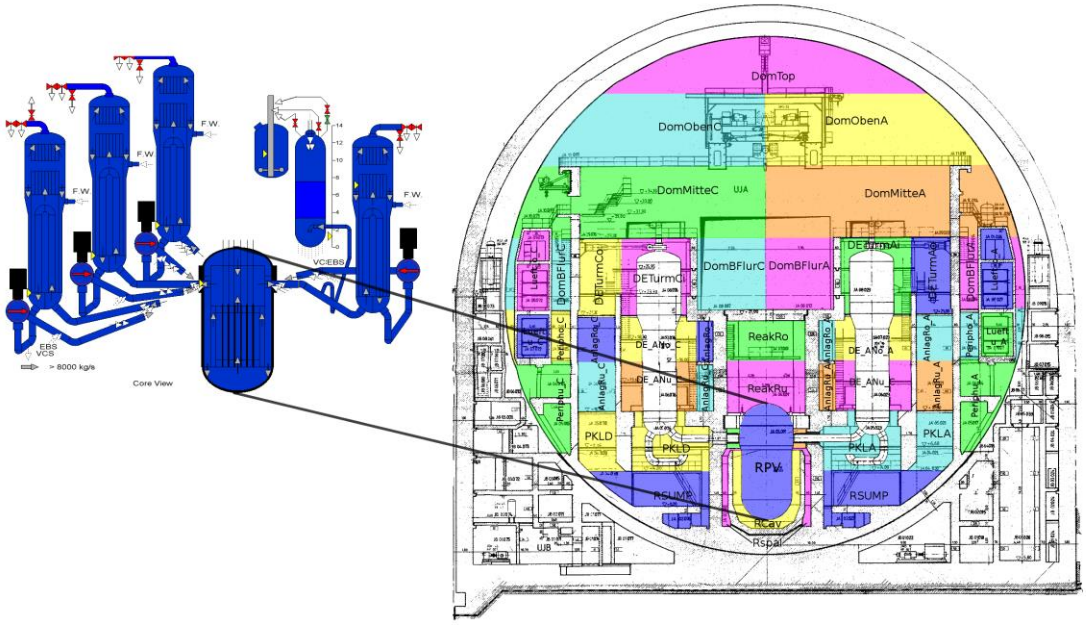
To perform severe accident analyses it is necessary to model the plant in an appropriate manner considering certain accuracy on the one hand and a certain simplification in order to limit the numerical effort on the other hand. Both input decks (PWR / BWR) will be presented in the following.

3.1. Pressurized Water Reactor (PWR)

The modelling of the reactor coolant system (RCS) is shown on the left side of Figure 1. The four loops of the real power plant are modelled using two loops. One of them comprises three real loops and the other one is a single loop with the pressurizer and the relief tank. The modelling consists of certain numbers of control volumes, heat structures and flow paths listed in Table 1. The systems attached to the primary side of the reactor coolant system incorporate reactor coolant pumps, safety injection pumps, residual heat removal system, accumulators, volume control system, and the extra borating system. The RPV itself is thermo-dynamically modelled (CVH package in MELCOR) by using only one control volume. Within the COR package it is modelled by using 5 core rings and 15 axial meshes whereof 12 are related to the active core region. Moreover, the two pressure relief valves and the blow-off control valve are modelled including the corresponding control from the reactor protection system.

At the secondary side, the steam generators are modelled including separator, main steam line and also the blow-off valve, the safety relief valve including the corresponding control by the reactor protection system, which implies the shutdown, the runback and the safety relief function. The conventional part of the nuclear power plant including the turbine, main condenser and the feed water heater line are modelled using one time-independent control volume for each the turbine and the feed water station. The nodalization of the containment is shown on the right side of Figure 1. The flow paths between the zones of the containment are partially equipped with doors that are assumed to be closed initially in most of the scenarios. The control of these doors implies the possibility of being opened by damage at the lock or the frame of the door depending from the direction of the pressure gradient. In the first case, the door can be re-closed again; in the latter case this is impossible. In addition to the doors, some rupture discs are modelled, too. Furthermore, some control volumes of the containment accommodate passive autocatalytic recombiners (PAR) – as realized in the reference plant – which keep the hydrogen concentration low in order to avoid large-scale hydrogen combustions. The hydrogen originates mainly from the reaction of zircaloy and steam but also from the reaction of steel and the corresponding alloy additions with steam. The areas in which the molten core concrete interaction (MCCI) takes place are modelled by MELCOR cavities. In this input deck, three cavities are defined, which stand for the reactor cavity, the zone between the biological shield and the support shield, in the following called “gap volume”, and the reactor sump. The biological shield itself is quite thin ($\Delta r_{\text{biol. shield}} = 0.55$ m) and can be penetrated quickly by radial ablation.

Figure 1: MELCOR modelling of the reactor cooling system and the containment



In addition, the ventilation channels below the surface of the reactor cavity bottom are considered as well. Thus, two modes of cavity rupture are possible to transfer molten material into the next cavity (“gap volume”). Between the latter one and the sump cavity there are dampers within the support shield whose lower edges are located very low above the bottom of this cavity. So there is only a certain small amount of corium necessary to trigger this rupture mode of the second cavity.

Table 1: Objects for the MELCOR modelling in the case of the PWR (pri: primary side of the RCS, sec: secondary side of the RCS, int: internal connections within area / object, ext: connections between the area / object and its environment)

Area / Object	n^{CV}	$n^{\mathrm{HS}}_{\mathrm{int/pri}}$	$n^{\mathrm{HS}}_{\mathrm{ext/sec}}$	$n^{\mathrm{FL}}_{\mathrm{internal}}$	$n^{\mathrm{FL}}_{\mathrm{external}}$
single loop (primary/secondary)	5/6	20	10	9	31
triple loop (primary/secondary)	5/6	36	15	9	10
RPV	6	30	10	10	7
containment	77	218	10	256	7
annulus	12	21	22	19	7
burst elements (door / disc)	82/56				
n^{CV} equipped with recombiners	37				

3.2. Boiling Water Reactor (BWR)

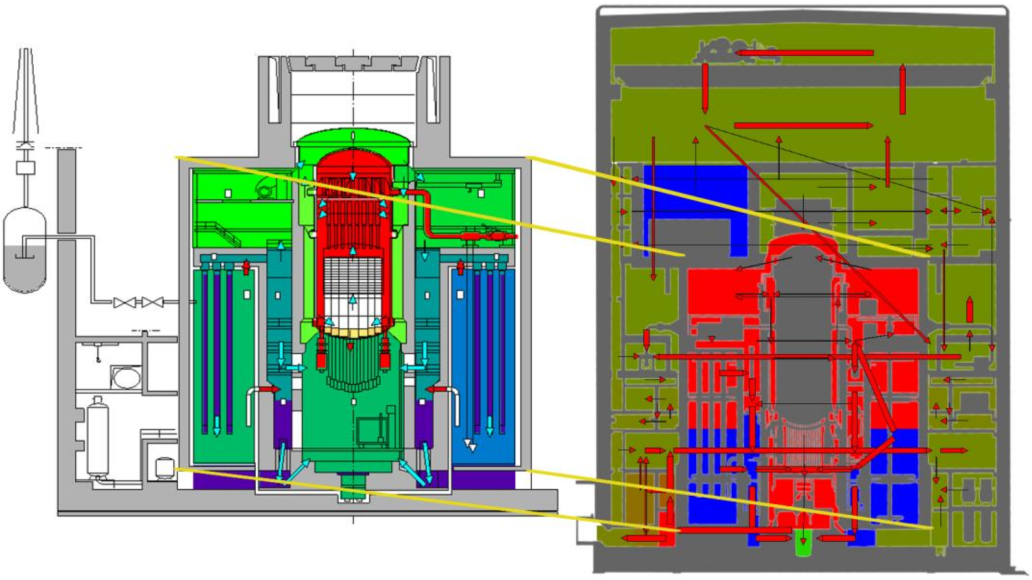
The modelling of the reactor cooling system and the containment is shown in Figure 2 and consists of a set of control volumes, heat structures and flow paths whose numbers are given in Table 2. In the case of a BWR, the reactor cooling system mainly consists of the RPV and the main steam lines that conduct the saturated steam which leaves the separator towards the turbine. The RPV of the BWR is nodalized in the CVH package using only one control volume. In the COR package it is divided into 5 rings and 19 axial meshes whereof 12 meshes belong to the scope of the active core region, the remaining ones are assigned to the lower plenum. The systems that are attached to the reactor cooling system comprise the circulation pumps, the residual heat removal system, the purging system, the safety relief valves and the corresponding control by the reactor protection system. The control of these valves implies the safety relief function to limit the pressure in the RPV and the automatic depressurization to decrease the pressure in the RPV in order to make the low pressure injection available. In reality there are 11 safety relief valves so there is a high level of redundancy but the valves are of the same kind. To handle common cause failures the facility provides 3 diverse pressure limiting valves which open at lower pressures in order to conserve the safety relief valves. In the case of the residual heat removal systems also several operational modes are possible. The system has high and low pressure pumps for the different pressure in the RPV. The high pressure pumps are intended to stabilize the water level in the RPV in the case of loss of feed water. The low pressure pumps are intended to flood the RPV in the case of LOCA events and remove the decay heat out of the RPV or the suppression chamber. The conventional part of the nuclear power plant is modelled like that of the PWR (see par. 3.1).

Table 2: Objects for the MELCOR modelling in the case of the PWR (pri: primary side of the RCS, sec: secondary side of the RCS, int: internal connections of the area / object, ext: connections between the area / object and its environment)

Area / Object	n^{CV}	$n^{HS}_{int/pri}$	$n^{HS}_{ext/sec}$	$n^{FL}_{internal}$	$n^{FL}_{external}$
RCS	14	32	14	18	3
containment	22	43	69	38	5
reactor building	196	859	212	602	21
burst elements (door / disc)	240/27				
n^{CV} equipped with recombiners	13				

In contrary to the modelling of the PWR an arrangement of only two cavities is considered here. The first one is related to the room where the control rod drives are located. There is also a cylinder symmetric wall on which an assembly machine is supported. The space that is surrounded by this wall represents the first cavity. Due to the purpose of this wall to support a device and not to retain a molten pool it has only a thickness of about 0.48 m, thus the radial ablation will quickly penetrate the wall. Then a part of the molten pool has access to an area of the basement of the containment which is separated from the reactor building by steel plates that are not able to cope with an attack of molten corium so they will immediately rupture in such a case. The molten corium will then flow into the basement of the reactor building. This area represents the second cavity but it is only that part of the basement that may be expected to cause the most severe consequence. This assumption is based on the circumstance that the boundary of this area provides a door which directs to the environment.

Figure 2: Reactor cooling system and the containment on the left side and the reactor building including the containment on the right side respectively (BWR72 type)



BWRs in Germany are also equipped with autocatalytic passive recombiners (PAR) which are located within the containment. To adapt the input values to the real devices in the facility, experimental data of the international THAI project [6] has been examined. So the values related to the PAR start concentration, dead time, relaxation time and the parameters for the simple dependence of the volume flow rate from the hydrogen concentration could be obtained. In the real facility there are 59 recombiners within the containment whereas 13 control volumes in the input deck are equipped with a PAR input. The doors and rupture disks are treated like that in the PWR (see par. 3.1).

4. EXEMPLARY ANALYSIS

One exemplary sequence related to a PWR in shutdown mode will be described in the following.

4.1. Initial Conditions

The PWR plant is in the shutdown mode called 1B2, in which the water level of the RCS is decreasing down to mid-loop (that means ¾ of the height of the reactor coolant line). Two trains of the residual heat removal system (RHR) assure the transportation of decay heat. One train is in maintenance and the last one is in standby. Under these initial conditions a postulated leakage occurs in one operating train of the RHR, for example due to thermal stress. This leak is located between the residual heat removal pump and the plunger check valve. It is assumed that the first shutoff valve and the plunger check valve (i.e. first and second isolation of the residual heat removal from the primary circuit) or the shutoff valve in the bypass line for the plunger check valve fail to close. The result is a permanent bypass from the RCS to the reactor building annulus. The residual heat removal is then lost by the drop of the water level below the suction point of the corresponding pumps. Due to reaching the threshold in the minimum flow line of these pumps they are shut down. According to the plant operating manual the accumulators inject to fill up the RCS. Moreover, some inventory of coolant is present in the flooding tanks that could be injected. This measure may delay core damage but it will not prevent it. This measure is not considered in the basic calculation shown here.

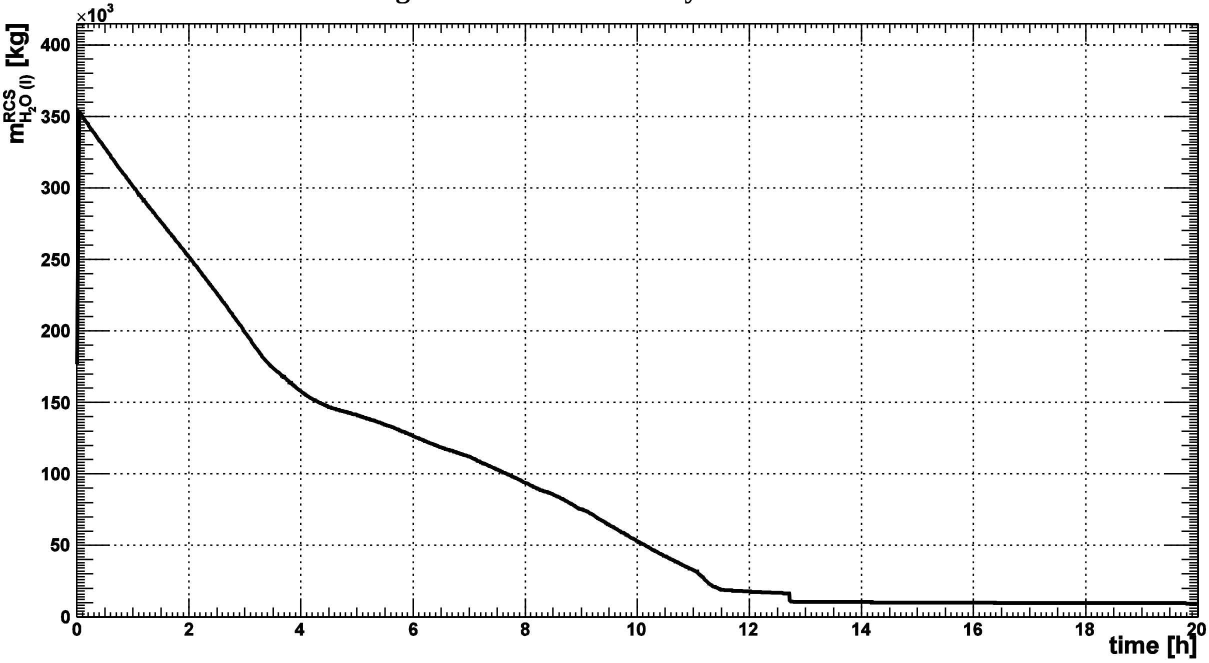
Table 3: Main events / phenomena during the sequence

Event / Phenomenon	**Time**	
shutdown	-23:00 h	
loss of all cooling systems due to leak in RHR system	0:00 h	
reach of the boiling point in the RPV (core related control volume)	0:33:20 h	
begin of the core uncovering at $L_{\mathrm{RPV}} \leq 6.63$ m	6:09:18 h	$\Delta t_{\mathrm{uncovering}}$
end of the core uncovering at $L_{\mathrm{RPV}} \leq 2.73$ m	8:09:20 h	$= 2:00:02$ h
begin oft the production of hydrogen	6:21:40 h	
gap release (begin; core ring 2)	6:47:33 h	
begin of the core melt process (first relocation of core material)	**7:02:33 h**	
rupture of the lower core grid, core drop, quenching	**11:04:00 h**	
dry-out of the lower plenum	11:35:50 h	
rupture of the RPV, begin of relocation of the molten pool into the cavity	**12:42:22 h**	
contact of the molten pool with the ventilation channels (dry)	17:23:54 h	
reaching the design temperature of the containment ($T_{\mathrm{design}}^{\mathrm{containment}} = 418.15$ K)	58:32:50 h	
rupture of the burst disc of the relief tank	76:59:51 h	
maximum pressure in the containment	76:59:51 h ($= t_{\mathrm{burst}}^{\mathrm{relief\ tank}}$); $p_{\mathrm{max}}^{\mathrm{cont}} = 0.16$ MPa	
large combustion in the reactor building annulus	07:54:29 h	
end of the calculation	336:33:31 h	

4.2. Conditions in the RCS

At the beginning of the scenario at 00:00:00 hours, the leak occurs and at the same time (simplification) the accumulator injection takes place. This increases the pressure and decreases the temperature for a short time period. The further progression is characterized by local pressure maxima and minima that are based on the leakage rate and relocation of material of the uncovered part of the core which falls into the water pool and provides an increase of the evaporation for a certain time which increases also the pressure. The leakage rate at the beginning is about 13.88 kg/s for an interval of 03:18:00 hours. After that the leak becomes uncovered. From this time on only steam is discharged which means a significant decrease of the leak mass rate. This behavior can also be seen in Figure 3, where the whole mass of liquid water in the RCS is shown. At the time of the rupture of the RPV at 12:42:22 hours, only the water in the pump suctions resides in the RCS.

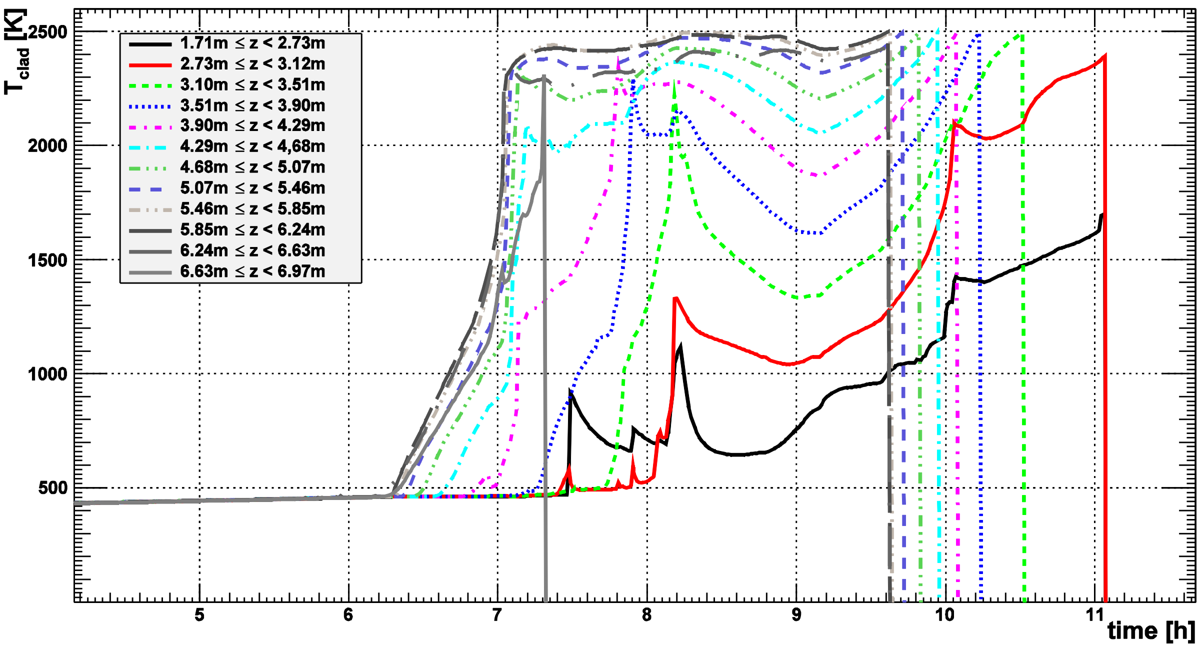
Figure 3: Water inventory of the RCS



In Figure 4 the temperatures of the cladding of the inner ring of the MELCOR modelling can be seen. A significant increase from the boiling temperature ($T_{\text{boiling}}^{\text{RCS}} = 462.0\,\text{K}$) can be realized at 06:09:10 hours. This time correlates with the beginning of uncovering the active region of the core at 6.63 m. The slope of the increase of the cladding temperatures rises at 06:50:00 hours significantly due to the beginning of the zircaloy steam reaction. This reaction ends if a certain oxide layer in the cladding is achieved. Thus the temperatures decrease through the loss of thermal radiation and conduction. Then the temperature increases once more due to the main heat up of the RCS and ends with culmination at $2500.0\,\text{K}$ which is assumed as the melting point of the interacting ZrO_2 and UO_2. At 07:02:33 hours this temperature is achieved for the first time in the uppermost cell of the inner ring. The gap release of the five MELCOR rings occurs quite earlier at a temperature of $1173.0\,\text{K}$ at the following times for the different MELCOR rings: $(06{:}47{:}37\,\text{h}, 06{:}47{:}33\,\text{h}, 06{:}48{:}39\,\text{h}, 06{:}47{:}39\,\text{h}, 07{:}07{:}43\,\text{h})$.

The relocation of core material to the lower plenum begins at 11:04:00 hours and ends 00:09:20 hours later. This also means that the steel temperature of the lower head increases so it strains. Considering the stress and strain of the steel MELCOR assumes a rupture at 12:42:22 hours which results in an ejection of debris. This process ends in essence 00:26:39 hours later with a mass of $2.0797 \cdot 10^5$ kg which is ejected into the reactor cavity (cavity #1).

Figure 4: Cladding temperatures during the uncovering of the reactor core

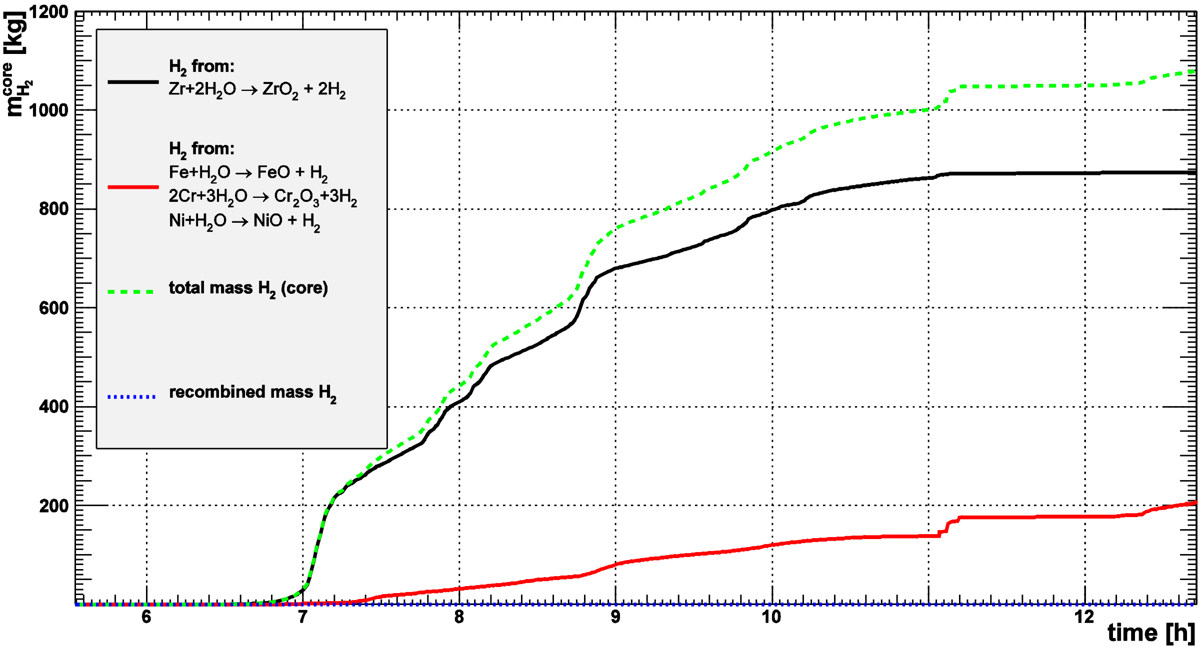


4.3. Conditions in the Containment and the Annulus

Due to the leak in the RHR connecting the RCS with the reactor building annulus, the essential barrier of the containment is bypassed, so no significant pressure built up is expected. There are local maxima ($p_{\text{cont.}} < 0.13$ MPa) of pressure within the containment due to effects like the rupture of the lower head and the rupture of the first cavity triggered by the radial rupture criterion at 17:23:54 hours which results in a discharge of molten pool into the second ("gap volume") and immediately into the third (reactor sump) cavity. This discharge means an increase of the production rate of hydrogen, which causes several combustions. Before these phenomena also a large combustion within the annulus occurs at 07:54:29 hours which results in a short increase of pressure and temperature which opens burst discs connecting the annulus with the environment directly and enlarges the existing connection through the reactor auxiliary building which has opened before due to the pressure built up resulting from the leakage. Hence no further significant pressure build-up is possible. The absolute maxima achieved in the containment correlates with the rupture of the burst discs of the relief tank at 76:59:51 hours, which is heated up by the containment atmosphere. The resulting maximum pressure is 0.16 MPa, which is not conserved due to condensation of the steam and the leak rate. This pressure is significantly below the design pressure of the containment, which is 0.63 MPa. In contrary, the design temperature, which is 418.15 K, is achieved at 58:32:50 hours and is increasing monotonically during the further devolution of the accident achieving a maximum value of 592.45 K at the end of the calculation. The conditions in the annulus also exceed the design in respect to the temperature, which reaches a value of 495.85 K at the end of the calculation. The design temperature of important safety devices in the case of a residual heat removal pump for example is 473.15 K according to [7].

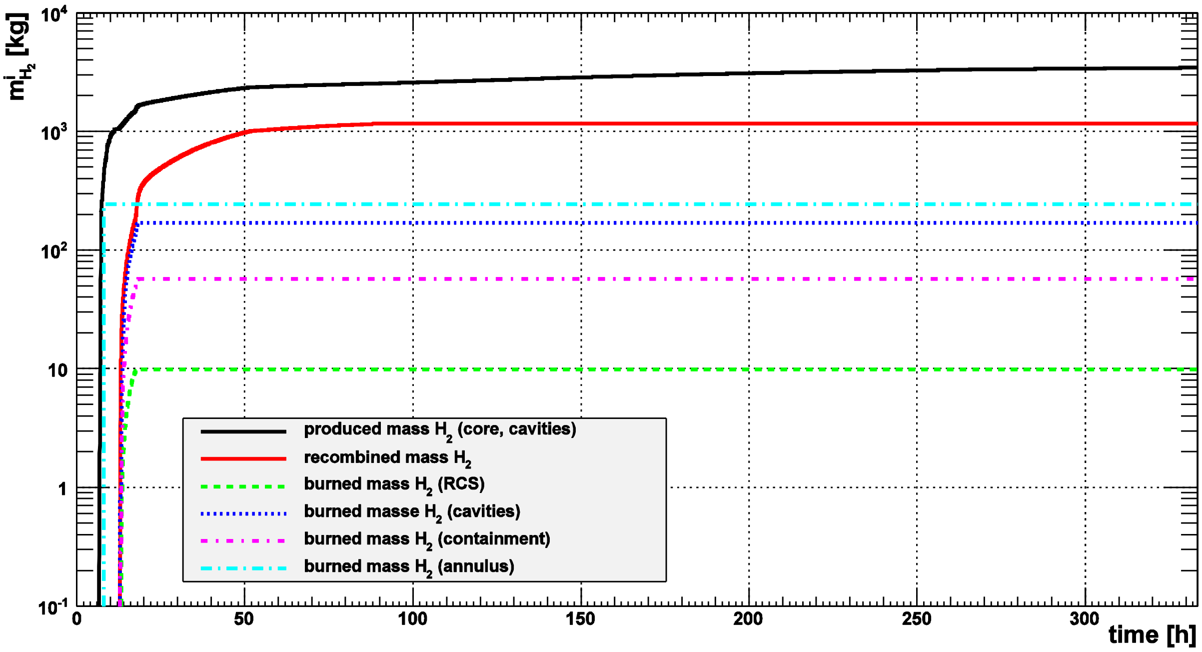
In Figure 5, the production of hydrogen based on several chemical reactions during the in-vessel phase is shown. This phase begins at 06:21:40 hours, slightly before a significant increase of the cladding temperature can be observed. It ends with the rupture of the lower head of the RPV. The hydrogen production within the core continues until 18:22:31 hours with a generated total mass of 1082.4 kg of hydrogen. The main contribution with a mass of 874.0 kg is delivered by the reaction of zircaloy with steam. The remaining reactions with steam considering the steel and its alloy additions (chrome and nickel) deliver a mass of 208.4 kg of hydrogen. The hydrogen mass that is recombined by the passive autocatalytic recombiners which is also given in Figure 5 is exactly 0.0 kg during the in-vessel phase, because no hydrogen is delivered into the containment.

Figure 5: Hydrogen production during the in-vessel phase



The whole hydrogen balance is shown in Figure 6 where the produced mass, the recombined mass and the consumed mass due to the combustions in the different areas of the reactor building are depicted. The total hydrogen mass produced in the core and the three cavities at the end of the calculation is 3460.9 kg, in which $\left(m^{H_2}_{cav00}, m^{H_2}_{cav01,\,sat}, m^{H_2}_{cav02,\,sat}\right)\big|_{t=t^{calculation}_{end}} = (2167.4\,\text{kg}, 5.3\,\text{kg}, 133.4\,\text{kg})$ is related to the corresponding cavity. But only the first cavity produces hydrogen up to the end of the calculation. The productions originating from the other cavities go into saturations due to the cooling down of the thin pools. This is the case of the second cavity ("gap volume"), because the connecting path between the second and the third cavity (rector sump) is near to the floor of this cavity. Thus, the amount of the remaining molten pool mass is very limited. In the case of the third cavity the reason is based on the large area which is covered by the molten pool. In both cases the pool is cooled down below 1420 K which represents the solidus line of the concrete. Hence, no further liquefaction is possible and the gas production stops. As already mentioned the recombination of the PARs starts when the rupture of the RPV occurs at 12:42:22 hours. The total amount of catalytic recombined hydrogen is 1173.1 kg which also goes into saturation due to oxygen starvation within the containment at a molar fraction of $\frac{n_{O_2}}{\sum_i n_i} \approx 0.28\,\%$. The combustions are not dependent from such concise events such as the RPV rupture because the hydrogen is discharged continuously into the annulus since the zircaloy steam reaction takes place in a significant manner at about 06:21:40 hours. So here the first large scale combustion appears at 07:54:29 hours. Overall a mass of 485.03 kg recombine till the end of the calculation through combustions in which 246.07 kg are related to the reactor building annulus. An amount of 647.97 kg remains in the containment, RCS and annulus.

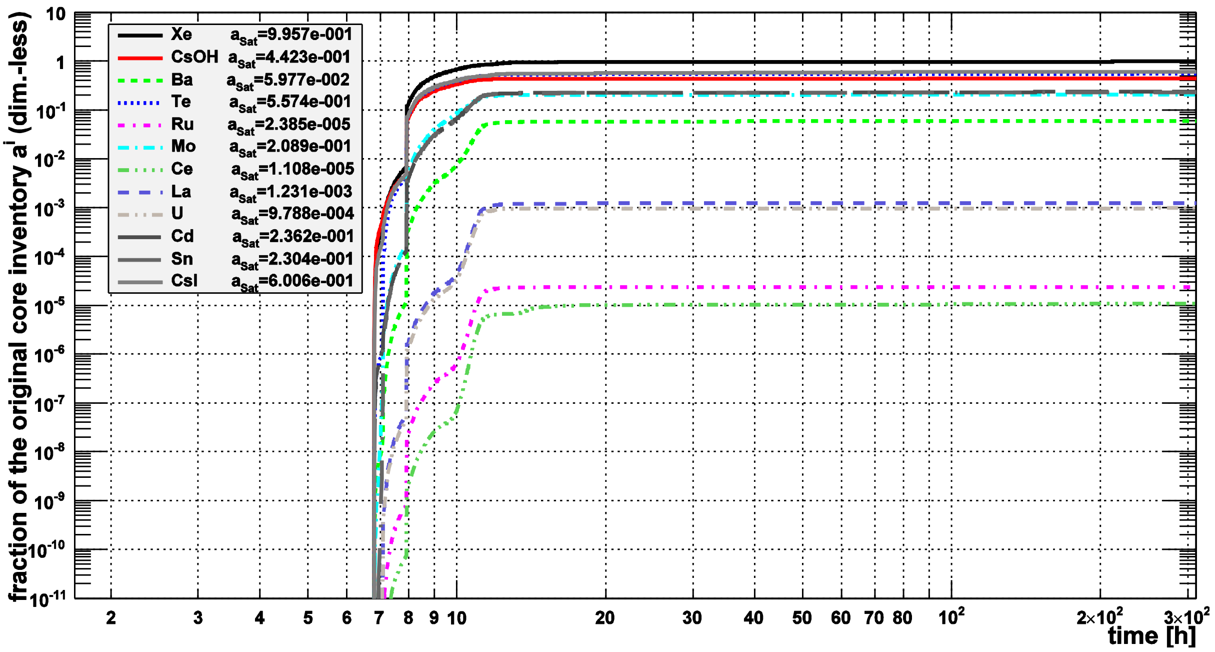
Figure 6: Balance of the hydrogen production and consumption during the accident devolution



4.4. Release of Radionuclides

Figure 7 depicts the release into the environment of the plant. It begins with the burst of the fuel rod cladding at 06:47:33 hours (gap release). There is an early significant increase at 07:54:29 hours which is based on the hydrogen deflagration in the annulus that causes the activation of the release path into the environment of the facility by opening the door into the reactor auxiliary building.

Figure 7: Released fraction of the original core inventory of the MELCOR element classes



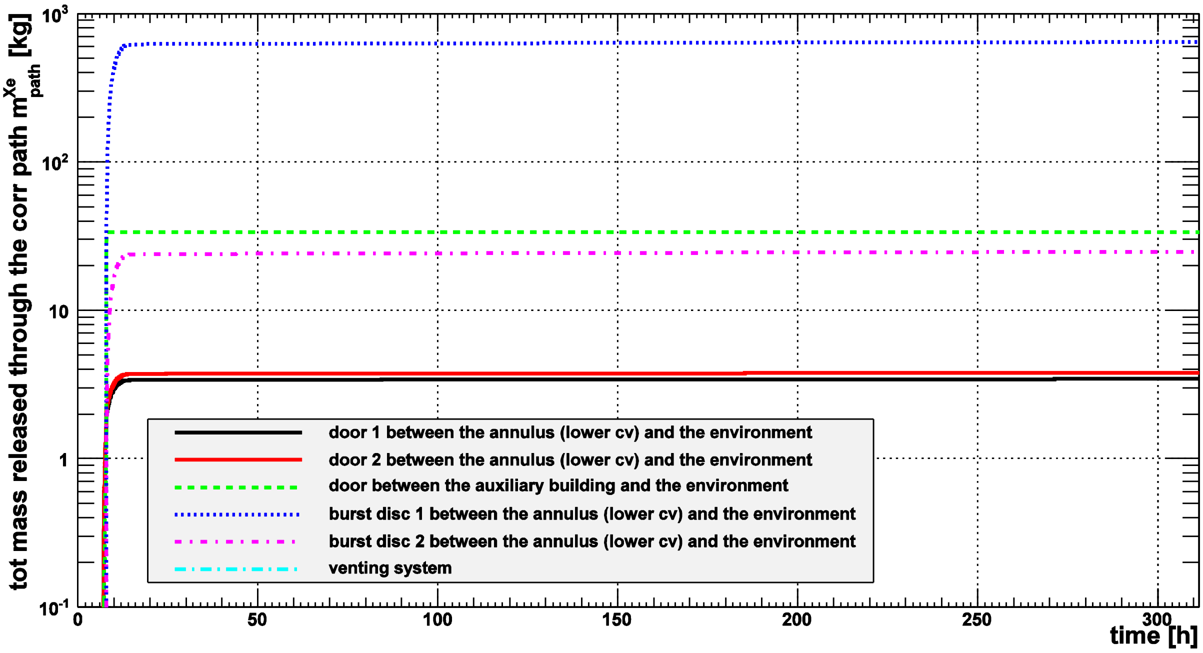
According to the explanation above the significant release into the environment starts with the large-scale deflagration in the annulus. The values at the end of the calculation are given in Table 4. For example, the fraction of the most volatile MELCOR group from the original core inventory is $\frac{m^{\text{Xe}}_{\text{released}}}{m^{\text{Xe}}_{\text{core inventory}}} = 99.57\ \%$.

Table 4: Released fractions of the element classes from the original core inventories. The numbers in brackets indicate a hierarchy in relation on the volatility (fraction, total mass).

MELCOR element classes	Release fraction into the environment:	Original core inventory:	Released mass into the environment:
Xe (1,1)	$9.957 \cdot 10^{-1}$	$7.111 \cdot 10^{+2}$ kg	$7.081 \cdot 10^{+2}$ kg
CsOH (4,2)	$4.423 \cdot 10^{-1}$	$4.050 \cdot 10^{+2}$ kg	$1.791 \cdot 10^{+2}$ kg
Ba (8,7)	$5.978 \cdot 10^{-2}$	$2.969 \cdot 10^{+2}$ kg	$1.775 \cdot 10^{+1}$ kg
Te (3,5)	$5.574 \cdot 10^{-1}$	$6.501 \cdot 10^{+1}$ kg	$3.624 \cdot 10^{+1}$ kg
Ru (11,12)	$2.385 \cdot 10^{-5}$	$5.327 \cdot 10^{+2}$ kg	$1.271 \cdot 10^{-2}$ kg
Mo (7,3)	$2.089 \cdot 10^{-1}$	$5.070 \cdot 10^{+2}$ kg	$1.059 \cdot 10^{+2}$ kg
Ce (12,11)	$1.107 \cdot 10^{-5}$	$2.095 \cdot 10^{+3}$ kg	$2.320 \cdot 10^{-2}$ kg
La (9,10)	$1.232 \cdot 10^{-3}$	$1.009 \cdot 10^{+3}$ kg	$1.243 \cdot 10^{+0}$ kg
U (10,4)	$9.788 \cdot 10^{-4}$	$9.750 \cdot 10^{+4}$ kg	$9.544 \cdot 10^{+1}$ kg
Cd (5,9)	$2.362 \cdot 10^{-1}$	$1.310 \cdot 10^{+1}$ kg	$3.093 \cdot 10^{+0}$ kg
Sn (6,8)	$2.304 \cdot 10^{-1}$	$1.764 \cdot 10^{+1}$ kg	$4.064 \cdot 10^{+0}$ kg
CsI (2,6)	$6.006 \cdot 10^{-1}$	$5.806 \cdot 10^{+1}$ kg	$3.487 \cdot 10^{+1}$ kg

Figure 8 depicts the total masses of xenon as a representative class for the release through the corresponding release paths is shown. Most of the gas mass ($m^{\mathrm{Xe}}_{\mathrm{release,\ BD}} = 647.04$ kg) is released through a burst disc which connects the annulus (lower cv) and the atmosphere. This disc is opened at 07:54:29 hours when the pressure in the annulus reaches 20.0 kPa for a short time due to the hydrogen burn. This pressure peak opens also other release paths.

Figure 8: Total mass released through different paths into the environment



5. CONCLUSIONS

As described above, the selection of relevant scenarios with respect to shutdown operational modes and external hazards in order to deepen the knowledge in this context have been accomplished for PWR and BWR. Furthermore, several severe accident sequences have already been modelled in the input deck of MELCOR and calculated. This shows that MELCOR is capable to be applied also for sequences different from full power operation.

The exemplary analysis shown in this paper is initiated by a leak in the RHR system at mid-loop operation of a PWR, bypassing the containment. It shows a hydrogen mass of 1082.4 kg during the in-vessel phase lasting up to about 12:42:22 hours until the RPV rupture occurs. In comparison to the full power operation analyses (in [8] it is about ~600 kg ... 800 kg) that means a quite high amount of hydrogen mass during a relatively long in-vessel phase.

Beside this, the analysis has shown that the calculated temperature in the containment as well as in the annulus significantly exceed the design temperature whereas no significant pressure built up occurs. Furthermore, early hydrogen combustion in the reactor building before RPV failure opens direct release paths to the environment. No noteworthy retention inside the plant is possible; therefore a very large release into the environment will occur.

Further conclusions will be drawn in the next steps of this ongoing R&D project. In those steps the remaining calculations will be performed and analyzed in respect of discovering weak points in the design of the facility and they will contribute to a deepened comprehensive assessment of such kinds of sequences. Furthermore, they will be assessed in respect to their probabilities and considered in an event tree in order to evaluate the relevance of such severe accident sequences also in comparison to full power operation.

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