Correlation of initiating events with the PSA Level-2 results

H. Löffler, M. Sonnenkalb

Gesellschaft für Reaktor- und Anlagensicherheit (GRS) mbH, 50667 Köln, Germany

Abstract: A large event tree was set up to perform a probabilistic analysis of core melt accidents (PSA Level-2) for a German Konvoi PWR. Uncertain data and assumptions were taken into account in Monte Carlo simulations. The core damage states which are the starting point of the event tree were transferred consistently from a PSA Level-1. The study is limited to internal initiating events at full power. The most severe accident progression is a failure of the reactor vessel under high pressure in about 3% of all core melt accidents. The main contributions come from steam generator leaks and transients with loss of power. Other significant consequences are bypass through steam generators (7%), or late leak in the containment (7%). In about 80% of all core melt accidents the release to the environment is limited by filters. ‚Classical' containment threats (steam explosion, hydrogen burns, direct containment heating) do not contribute significantly to containment failures. The frequencies of release categories for radionuclides are calculated, including an uncertainty analysis. There still remain significant uncertainties for the evaluation of severe accidents.

1. INTRODUCTION

Since the German Risk Study for nuclear power plants [1] additional and new knowledge has become available for many aspects of the progression of core melt accidents. Better thermodynamic and structural mechanic codes have been developed and the progress in computing allows more simulations in greater detail. In addition, a certain international state of the art has evolved with regard to the probabilistic evaluation of significant accident phenomena. This progress now enables to perform a well founded accident progression event tree analysis (Level-2 PSA), although a considerable need for further investigation of specific questions still remains.

For the first time in Germany GRS has evaluated the methodology of a Level-2 PSA, including a consistent transfer of results from a Level-1 PSA, development of a large event tree and binning the results into plant damage states. The plant being investigated is the pressurized water reactor GKN-II of the Konvoi type. The active support of the plant personnel to this work is thankfully acknowledged.

This study is limited to internal initiating events at full power, partly due to a lack of adequate knowledge and methodology beyond that scope, and partly due to time and budget restraints. The work was performed within 3 years, including a great amount of computer code analysis of accident progressions. It was funded by the federal ministry for environment, nature conservation and reactor safety (BMU).

2. METHODOLOGY FOR LEVEL-2 PSA

The methodology for the event tree analysis which is basically applied for this study is similar to the methodology developed within the US for the NUREG-1150 report [2]. It consists of a large event tree (more than 80 branching points), including several simple computer models for the evaluation of certain branching probabilities. For more than 100 uncertain input data of the event tree the distributions of the uncertain values were defined and taken into account in 500 Monte Carlo simulations. The probabilities, including their uncertainties, for different kinds of results (e.g. final state of the containment or release of radionuclides) were evaluated.
The state of the probabilistic methodology is sufficient to create meaningful results which cannot be

obtained otherwise. In particular, the representation of the complete accident progression including uncertain phenomena and human actions is far beyond the capabilities of other present-day computer codes. It allows to identify dependencies and quantitative comparisons of the influence of different phenomena.

Considerable effort was devoted to develop and apply a methodology for a consistent transfer of the Level-1 results to the Level-2 analysis. The link between these levels is accomplished by a set of different core damage states. These core damage states contain all the information which is necessary to perform the accident progression analysis. In addition, each core damage state can be characterized by additional information, e.g. about those system failures which produce the respective event. This additional information can be used to trace relevant Level-2 results back to their underlying causes in the Level-1 domain.

There are inherent drawbacks of the event tree analysis which have stimulated a research project at GRS, integrating probabilistic aspects into deterministic computer models. But these developments are still far from being a practicable substitute for the event tree analysis.

An adequate setup of the event tree and a well founded probabilistic quantification of phenomena can only be done when sufficient knowledge is available about the accident progression. The main source of information are calculations with the US computer code MELCOR 1.8.4 [3]. Investigations in more detail have been done for specific questions, e.g. for the hydrogen issue in the containment or the structural mechanics of structures of the primary system.

3. DEFINITION OF INITIATING EVENTS AND CORE DAMAGE STATES

A companion paper at EUROSAFE 2000 [4] informs about methods and results of the Level-1 PSA, which has generated a set of 66 different core damage states for the Level-2 analysis. Each core damage state is defined by about 10 different properties, including information about the primary system (e.g. pressure and availability of emergency core cooling systems), about the secondary system (availability of steam generators) and the containment system (status of containment isolation and availability of ventilation systems in the reactor building annulus). Further information is given about the probability to depressurize the primary system in high pressure events after the beginning of the core melt process and before the failure of the reactor pressure vessel.

The point value of the total frequency of all core damage states, delivered by the Level-1 analysis is 2.2E-6/a. For each of the core damage states the point value of its frequency was transferred to the Level-2 analysis. These data are not fully consistent with [4], because the Level-1 analysis has been continued after transferring the data to Level-2 To take into account uncertainties, transfer of frequency distributions is desirable and feasible in principle, but this could not be achieved in due time.

To identify meaningful correlations between Level-1 and Level-2, the initiating events are binned into six groups. The numbers indicate their fraction of all core damage states:

1.	L<25: Leak with less than 25 cm^2 at a main coolant line	56%
2.	L>25: Leak with 25-200 cm^2 at a main coolant line	5%
3.	LPR: Leak at the pressurizer	15%
4.	LSG: Leak at a steam generator tube	9%
5.	TLP: Transient after loss of offsite power	11%
6.	TWP: Transient with offsite power available	4%

Tables 1 to 3 show the distribution of three important core damage state properties for each of the groups of initiating events.

Tab. 1 Time period between initiating event and core damage state

Time period between initiating event and core damage state	Fractions of the core damage states						
	Initiating events and fractions						
	L<25 0.56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04	All 1.0
Less than 2 hours	0.08	0.01	0.05	-	0.01	0.015	0.165
2 to 4 hours	0.04	0.035	0.10	0.01	0.10	0.025	0.315
4 to 12 hours	0.43	0.0	-	0.04	-	-	0.465
More than 12 hours	0.01	0.005	-	0.04	-	-	0.055

Tab. 2 Availability of emergency core cooling systems at the core damage state

Availability of emergency core cooling systems at the core damage state	Fractions of the core damage states						
	Initiating events and fractions						
	L<25 0.56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04	All 1.0
No high pressure and no low pressure systems available	0.39	0.02	0.10	0.08	0.10	-	0.69
Low pressure systems only available	0.10	0.03	0.05	0.01	-	-	0.19
High pressure systems available long-term. LP systems irrelevant	0.07	-	-	-	0.01	0.04	0.12

Tab. 3 Pressure in the primary system at the core damage state

Pressure in the primary system at the core damage state	Fractions of the core damage states						
	Initiating events and fractions						
	L<25 0.56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04	All 1.0
High pressure(>8 MPa)	0.07	-	-	-	0.11	0.04	0.22
Medium pressure (2 to 8 MPa)	0.09	0.03	0.05	0.09		-	0.26
Low pressure (<2 MPa)	0.40	0.02	0.10	-	-	-	0.52

The following general conclusions can be drawn:

1. Transients almost always have early core damage states at high pressure. If there is no loss of offsite power, emergency core cooling systems are available.
2. Small leaks are the most frequent initiating event for core damage states. Mostly the pressure is low, and emergency core cooling systems are not available.
3. Only after significant time periods steam generator tube leaks will lead to a core damage state. There is medium pressure, emergency core cooling systems are not available in most cases.

4. ACCIDENT PROGRESSION ANALYSIS

Many accident progression calculations for various initiating events and with different assumptions about uncertain phenomena were performed with MELCOR 1.8.4. Some of these calculations are summarized in [5]. To give an impression of the variety of results, two accident progressions are presented in the following sections:

1. slow core melt accident after a core damage state with low pressure
2. fast core melt accident after a core damage state with high pressure

4.1 Slow accident progression after low pressure core damage state

The initiating event is a small (10cm²) leak in the hot leg. High pressure emergency core cooling systems will feed the primary system until the flooding tanks are empty (at about 5 h). Then the low pressure systems are assumed to fail, and gradually the coolant level falls, until core melt begins at about 22 h.
Fig. 1 shows the evolution of the pressure in the containment. The first pressure peak due to the leak is not high (about 0.18 MPa at 1 h). Condensation of steam reduces the pressure, until the beginning of core melt at 22 h. The hot gases from the core, including hydrogen, increase the pressure up to about 2 MPa at 26 h, when the reactor pressure vessel fails. The pressure remains limited until the core melt reaches sump water in the ventilation ducts below the cavity at about 34 h, producing a sudden increase to 0.24 MPa. Thereafter a continuous pressure increase follows. The calculation was terminated at about 60 h. It can be assumed that the pressure would continue to rise, probably reaching the setpoint for depressurization of the containment.
Although hydrogen recombiners are taken into account, there are some hydrogen burns before and at reactor pressure vessel failure. But these burns are limited to the equipment rooms so that there is no significant contribution to the pressure.

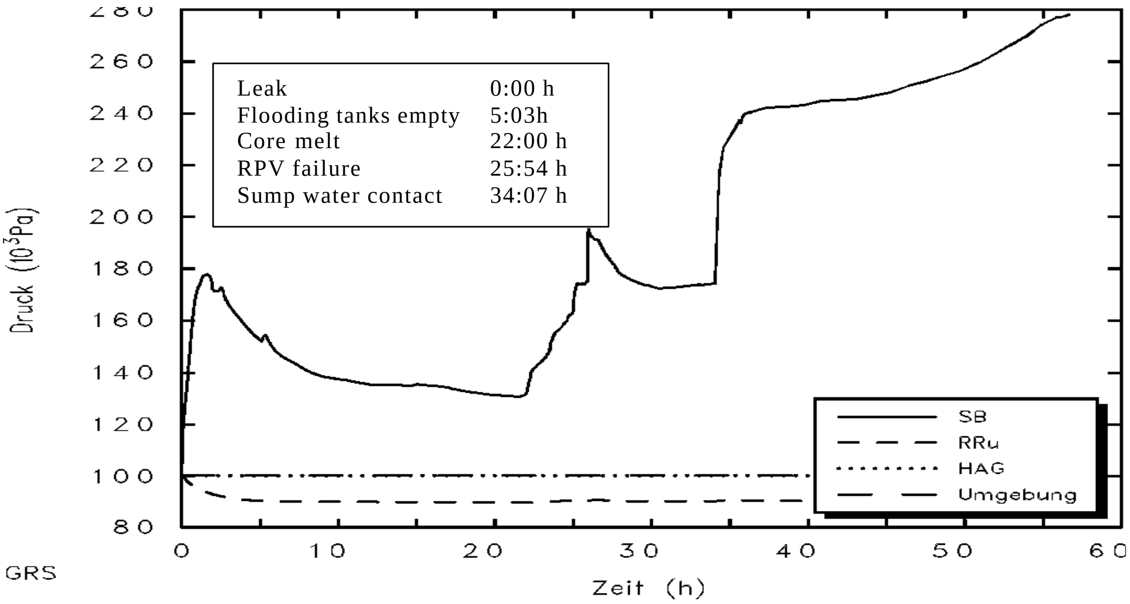

Fig. 1 Containment pressure for core melt accident initiated by small leak

4.2 Fast accident progression after high pressure core damage state

The initiating event is a station black out. The steam generators loose their inventory due to vaporization within about 1 h. Shortly after, the primary side pressure rises, limited by the valves on the pressurizer. At about 2:35 h core melting begins, releasing very hot gases through the hot leg and the surge line. At 2:55h the surge line reaches 800°C. At this temperature it is assumed to fail.

When this calculation was done, only a preliminary estimate was available for the failure of hot pipes. Meanwhile detailed structural mechanic analysis has been done, indicating failure temperatures of the main coolant line at 16 MPa between 820°C and 845°C [6]. The surge line fails at higher temperatures. These new data would not significantly change the calculated accident progression.

The pressure falls immediately after the pipe failure, the accumulators inject their water inventory and delay the further core degradation. At about 5 h the core relocates into the lower plenum, and the reactor pressure vessel fails at about 6:22 h.

Fig. 2 shows the evolution of the pressure in the containment. The first pressure peak due to coolant release to the containment, followed by the second at the failure of the surge line produce 0.42 MPa at 2:55 h. Then a decrease begins, interrupted by another small spike at reactor pressure vessel failure at 6:22 h. The pressure continues to decrease until the core melt reaches sump water in the ventilation ducts below the cavity at about 10 h, producing a sudden increase to 0.3 MPa. Thereafter a continuous pressure rise follows. At 38 h the increase is accelerated because at this time the spent fuel pool, which has no cooling due to the power loss, begins to boil. At 44:15 h the manual depressurization of the containment is assumed to begin. As can be seen from the pressure curve, the assumed capacity of the venting system (3 kg/s) is almost not sufficient to limit the pressure under these conditions.

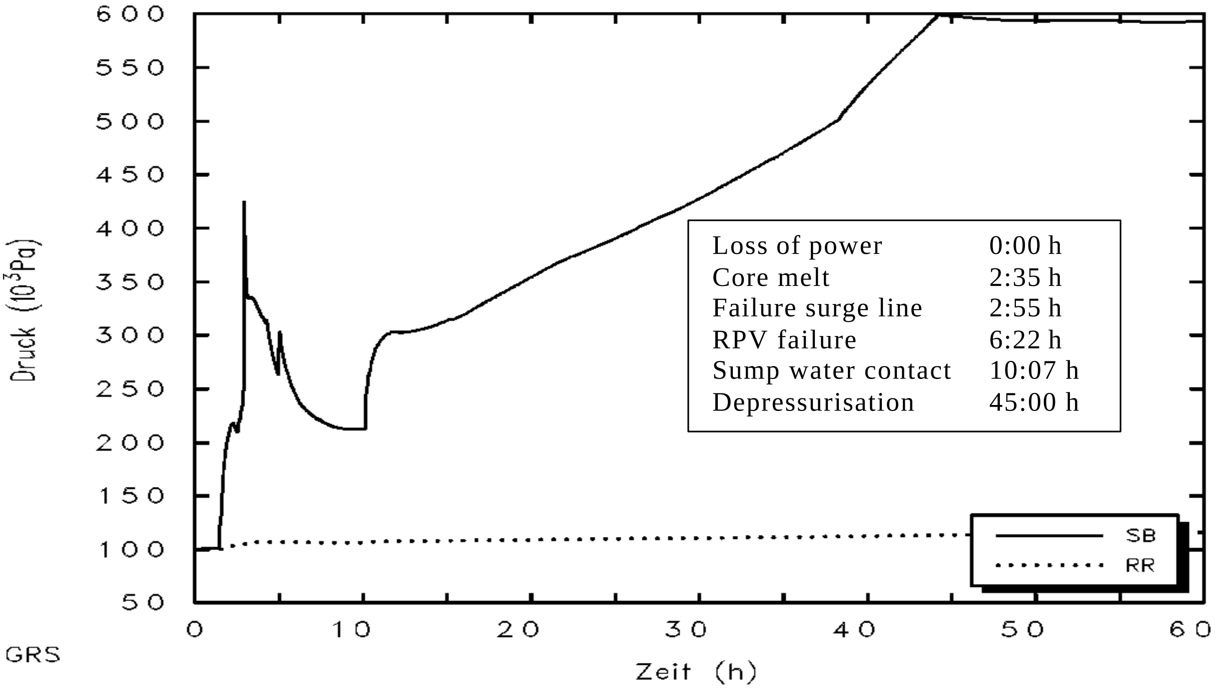

Fig. 2 Containment pressure for core melt accident initiated by station black out

5 EVALUATION OF CONTAINMENT THREATS

5.1 Containment performance

The containment has been tested against a pressure of 0.774 MPa. Structural mechanics assessment produced the following logarithmic distributions for the failure pressure of the containment:

For a slow pressure rise, the pressure will be limited by an increasing leak at the joint between equipment air lock and steel shell. The lowest failure pressure is 0.774 MPa, 5%-fractile is 1.04 MPa, and 50%-fractile is 1.53 MPa.

For a fast pressure rise, the leak size at the joint is not sufficient. In addition, a limited crack will develop at a weld between the plates of the containment shell. For this failure mode, the lowest failure pressure is 0.774 MPa, the 5%-fractile is 1.12 MPa, the 50%-fractile is 1.70 MPa.

5.2 Steam explosion inside the reactor pressure vessel

The mechanical energy release of a steam explosion was estimated by a simple computer model, which is implemented into the event tree. This model has input parameters which depend on the accident progression (e.g. rate of molten fuel entering the lower plenum) and which are directly transferred to the model from previous branching points in the event tree. There are other parameters which characterize the interaction (e.g. particle size, conversion factor) which were estimated with the support of an expert from Forschungszentrum Karlsruhe.

The performance of the reactor pressure vessel head was derived from experiments [7]. The mechanical behavior of the vessel bottom was evaluated at GRS.

Although considerable uncertainties had been assigned to almost all input parameters of the model, there was not a single Monte Carlo simulation with failure of the vessel head (alpha mode failure). Consequently zero probability was assigned to this event. There was, however, a small probability for a failure of the vessel bottom. But this failure mode was not assumed to change significantly the accident progression, since the vessel bottom would fail due to melt attack anyway.

5.3 Hydrogen inside the containment

Hydrogen is produced during the core melt process due to the oxidation of zirconium, and later during the core-concrete interaction, when various metals are oxidized. In addition, depending on the concrete composition, carbon monoxide is produced, which is ignitable as well.

GKN-2 is equipped with a set of passive autocatalytic recombiners which are installed in most of the compartments inside the containment. They recombine hydrogen or carbon monoxide with oxygen.

Under the conditions prevailing inside the containment, there is a short period of time during the core melt process, when the capacity of the recombiners cannot cope with the rate of hydrogen production. Therefore, some hydrogen burns are possible even when recombiners are present. Late in the accident, all the oxygen is used up, thus preventing any ignition, while there still is a considerable rate of hydrogen and carbon monoxide production.

The recombiners have been simulated by a computer model in the MELCOR-calculations, and a simplified model was integrated into the event tree analysis. This model considers the uncertainty of the hydrogen production rate, and the distribution of the hydrogen into different containment volumes. It calculates the composition of the atmosphere, including hydrogen, oxygen, nitrogen and steam. There is a check whether the mixture is ignitable, and probabilities for the ignition are assumed.

The pressure of a hydrogen burn is calculated, and compared to the containment performance. If a detonation develops, an effective pressure is estimated according to [8].

The evaluation of the event tree shows that in about 50% of all core melt accidents at least one hydrogen burn develops until failure of the reactor pressure vessel. Most burns occur in the equipment rooms, even detonations must be expected there. But these rooms are located far from the containment shell, so that the burn cannot directly threaten the containment. The service compartments near the containment shell have a significant probability for deflagrations as well, but practically no detonations have been calculated there.

The influence of hydrogen burns on the containment integrity is negligible. There is a very small probably (less than 1% of all core melt accidents, but with considerable uncertainty) that detonations produce a limited containment leak due to indirect mechanical impact at cable penetrations. It can be concluded that the threat due to hydrogen is insignificant for this plant which is equipped with recombiners.

5.4 Direct containment heating

If the reactor pressure vessel bottom fails at an internal pressure of more than 8 MPa, an immediate damage to the containment is assumed to occur due to the upward movement of the vessel [1]. For failure pressures below 8 MPa a simple model has been set up in the event tree to calculate the containment pressure, when hot core material, hydrogen and water is released from the reactor pressure vessel bottom. This model calculates that in a very limited number of cases (about 1% of all core melt accidents) pressures above 1 MPa develop. If these pressures are combined with a comparatively poor and unlikely containment performance, containment failure is calculated in less than 1% of all core melt

accidents.
The uncertainty of this value is considerable. Therefore, additional work is underway, including analysis with advanced computer codes.

5.5 Melt attack at the bottom of the containment

The spread of the melt after failure of the reactor pressure vessel bottom is shown in Fig. 3. At the bottom of the biological support shield there are 8 flaps (‚Druckausgleichsklappen‘ in Fig. 3) which can be assumed to be open at least partly after the core melt has left the reactor pressure vessel. Sump water flows through these flaps, flooding ventilation ducts (‚Lüftungskanäle‘ in Fig. 3) below the reactor cavity. When the core melt has penetrated about 0.7 m of concrete, it enters the ventilation ducts. Analysis of the melt properties and the geometrical situation led to the conclusion that the melt will spread further and cover the bottom of the sump almost completely. The melt will come into contact with the sump suction lines (‚Sumpfansaugrohre‘ in fig. 3) which lead through the containment shell into the reactor building annulus. The melt level will, however, not be sufficient to flow into the sump suction through their top openings.

Titel:
0009401x.p65
Erstellt von:
PageMaker 6.5
Vorschau:
Diese EPS-Grafik wurde nicht gespeichert
mit einer enthaltenen Vorschau.
Kommentar:
Diese EPS-Grafik wird an einen
PostScript-Drucker gedruckt, aber nicht
an andere Druckertypen.

Fig. 3 Spread of core melt at the bottom of the containment

Two mechanisms could contribute to a specific containment failure mode:

1. The melt penetrates the lower part of the sump suction line
2. Corium particles which are continuously produced by melt-water-interactions settle inside the sump suction line.

In both cases corium enters the line, settling inside the horizontal part of the line outside the containment. It is estimated that such a corium slug would not be coolable and that eventually the sump suction line would fail outside the containment. Then corium, water and the containment atmosphere would be released into the reactor building annulus.
It was assumed that there is a probability between 0% and 10% for such a containment failure mode if melt is released from the reactor pressure vessel.

5.6 Threats connected with containment depressurization

There will be a continuous pressure rise in the containment due to the core-concrete-water interaction, until the depressurization is necessary. The mean unavailability of the depressurization procedure was estimated to be about 4%. Human errors as well as equipment failure contribute to this unavailability in comparable fractions. In these cases the event tree analysis assumes a late containment failure with a

limited leak size.
The depressurization will be actuated late in the accident, when all oxygen has been used up inside the containment by the recombination of hydrogen and carbon monoxide. Typically the containment atmosphere volume consists of about 75% steam and more than 10% of hydrogen and carbon monoxide at the time of depressurization. The gases from the containment are lead through a water pool, where the steam condenses, increasing the volume fraction of ignitable gases to more than 40%. This gas then is discharged into a ventilation duct on the roof of the auxiliary building. The mixture with air almost inevitably produces an ignitable atmosphere. Large uncertainties exist regarding ignition sources in those ventilation ducts and the potential damage due to burns.
The event tree analysis showed that in about 9% of all core melt accidents the venting filter would be damaged, so that the containment depressurization would essentially be unfiltered. 12 % of all core melt accidents have burns which produce leaks at the venting ducts without affecting the filter. In these cases the filtered release is at roof height instead at stack level.

6. CORRELATION OF INITIATING EVENTS WITH PLANT DAMAGE STATES

The plant damage state describes the consequences of the core melt accident for the containment and for the release of radionuclides. As with core damage states, plant damage states are defined by a set of different properties. For the evaluation of results, the following properties have been selected:

1. Final location of core material
2. Pressure of primary system immediately before failure of the reactor pressure vessel
3. Final condition of the containment
4. Release paths and release fractions of significant radionuclides to the environment (release categories)

Tables 4 to 7 show the distribution of these properties for each of the groups of initiating events.

6.1 Final location of core material

If emergency core cooling systems begin to inject coolant into the core early, the core material can be cooled and retained in the core area or in the lower plenum of the rector pressure vessel without failure of the vessel. This sequence is typical for transients with electrical power available and high pressure in the primary system. In these cases the pressure decreases due to a leak in the hot leg, and the pumps begin to inject as soon as the pressure is low enough. In most other initiating events the emergency core cooling systems are not available, or the pressure is not high enough to cause a leakage.
The conditions under which a partly molten core can be retained within the reactor pressure vessel have been derived from MELCOR-results and from interpretations of the accident at TMI. Considerable uncertainty is involved in these evaluations.

Tab. 4 Probabilities for the final location of core material

Final location of core material	Fractions of the core damage states (CDS)						Mean over all CDS 1.0
	Initiating events and fractions						
	L<25 0.56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04	
Retention in core region, no failure of reactor pressure vessel	0.07	0.015	0.03	0.005	0.01	0.036	0.166
Retention in lower plenum, no failure of reactor pressure vessel	0.05	0.001	0.02	0.004	-	<<	0.075

Final location of core material	Fractions of the core damage states (CDS)						Mean over all CDS 1.0
	Initiating events and fractions						
	L<25 0.56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04	
Failure of reactor pressure vessel, core concrete interaction	0.44	0.024	0.10	0.081	0.10	0.004	0.759

6.2 Pressure in primary system before reactor pressure vessel failure

Pressure relief before the failure of the reactor pressure vessel is possible due to actively initiated depressurization by the operators, due to leaks at hot coolant lines, or due to stuck open safety valves. The most significant depressurization process is the failure of a hot line in transients, before the relocation of core material into the lower plenum. Important as well is the active depressurization and the stuck open failure of valves during and after the core relocation into the lower plenum.

The plenum still contains water at this time so that the pressure increases due to the vaporization, demanding action of the safety valves. If the safety valves act properly and close again, the pressure may be elevated for a certain time. If there is only a small leak in the primary system (initiating events L<25 and LSG) the pressure could remain high until failure of the reactor pressure vessel. In these cases there is low or medium pressure at the core damage state, but high pressure short before and at the reactor pressure vessel failure.

The failure temperature of hot coolant lines was investigated by appropriate structural mechanics analysis [6]. The pressure history inside the primary system as well as the temperatures of the components and the potential actions of operators under these circumstances are very uncertain. Since the consequences of a high pressure failure are severe, future investigations into these phenomena are recommended.

Tab. 5 Probabilities for the pressure inside the primary system short before reactor pressure vessel failure

Pressure inside the primary system short before reactor pressure vessel failure	Fractions of the core damage states (CDS)						Mean over all CDS 1.0
	Initiating events and fractions						
	L<25 0.56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04	
High pressure (>8 MPa)	0.005	-	-	0.015	0.01	<<	0.03
Medium pressure (2 to 8 MPa)	0.345	0.003	<<	0.027	-	-	0.37
Low pressure (< 2 MPa)	0.21	0.047	0.15	0.048	0.10	0.04	0.60

6.3 Final condition of the containment

If there is a high pressure failure of the reactor pressure vessel (3% of all core melt accidents, highest contribution from steam generator tube leaks and transients after loss of offsite power), then the containment will be damaged as well immediately, because of the upward relocation of the vessel. This judgement has been taken from the German risk Study [1].

System analysis shows that failure to close the containment isolation is insignificant. Containment failure due to pressure increase at reactor pressure vessel failure (direct containment heating, DCH) was evaluated by a simple model within the event tree analysis. Additional work is underway in this field. Presently, the analysis identifies DCH as an insignificant contribution to containment failure (0.4% of all core melt accidents).

Melt through of the sump suction line (3.8% of all core melt accidents) contributes to containment failure equally in all cases with reactor pressure vessel failure.

Failure to depressurize the containment takes into account wrong or missing human actions and failure of the necessary equipment. If depressurization fails, the containment pressure will continue to rise due to the core concrete interaction, and finally the containment develops a leak (3.4% of all core melt accidents). The failure mode is most probably a limited leak at the joint between the equipment air lock and the containment shell. According to structural mechanics analysis, a catastrophic failure can be ruled out.

In most core melt accidents (67.9%) the containment is depressurized according to the accident management directives.

All cases up to this point (77.2% of all core melt accidents) have a failure of the reactor vessel and a core concrete interaction. There is no evidence that this interaction could come to rest, not even under water. Therefore in the long run (one week or more) the core melt will penetrate the bottom concrete of the containment and reach the underground.

In the remaining 22.8% of all core melt accidents, the partly molten core can be retained within the reactor pressure vessel. There is only a minor containment load. Containment leaks into the reactor building annulus are directed to the filtered ventilation system.

Steam explosions and hydrogen burns do not contribute to containment failures. Consequently these phenomena are not included in table 6.

Tab. 6 Containment failure modes and probabilities for final states of the containment

Final state of the containment Note: in all cases except the last one, the melt penetrates the concrete foundation and reaches the underground	Fractions of the core damage states (CDS)						Mean over all CDS 1.0
	Initiating events and fractions						
	L<25 .56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04	
Damage due to high pressure failure of reactor pressure vessel	0.005	-	-	0.014	0.009	0.002	0.030
Failure to isolate containment ventilation	<<	<<	<<	<<	<<	<<	<<
Failure due to overpressure at reactor pressure vessel failure (DCH)	0.003	<<	<<	0.001	-	<<	0.004
Meltthrough of sump suction line	0.024	0.001	0.005	0.004	0.004	<<	0.038
Leak due to overpressure after failure to depressurize	0.021	0.001	0.005	0.003	0.004	<<	0.034
Intact with depressurization	0.434	0.020	0.095	0.050	0.078	0.002	0.679
Intact without depressurization (no reactor pressure vessel failure)	0.073	0.028	0.045	0.032	0.014	0.036	0.228

6.4 Release paths and release fractions of radionuclides to the atmosphere

In addition to the final condition of the containment, the further release path to the atmosphere determines the magnitude of the radioactive release as well. The following discussion concentrates on releases of Cs and J. The evaluation of the behavior and the final location of the radionuclides is based on an interpretation and extrapolation of MELCOR results. The fractions of core inventory released to the atmosphere which are given in table 7, are uncertain to a considerable extent.

When the reactor pressure vessel fails under high pressure, some damage at the containment and the reactor building annulus as well has to be assumed. Therefore the release from the containment to the atmosphere is almost direct. This release is called category FKA (3% of core melt accidents).

If the containment is not isolated properly or if there is an early failure due to overpressure, no credit has been given to mitigating mechanisms in the annulus. The release is called FKB (0.5% of core melt accidents).

If the release bypasses the containment and the annulus through a steam generator tube leak, there is some retention of radionuclides in the primary system and in the water of the steam generators. The quantity of this retention is very uncertain. The release is called FKC (7% of core melt accidents).

When there is a non energetic containment failure (slow overpressurization or meltthrough of sump suction line) the release enters the annulus. Aerosol particles settle there, while the pressure slowly rises. Under increasing pressure in the annulus, the flaps of the ventilation system will fail first, giving way to an unfiltered release through the ventilation ducts. This is release category FKE (6% of core melt accidents).

If the containment is depressurized according to emergency procedures, the venting filters may be destroyed due to hydrogen burns. Nevertheless, the long time available for aerosol removal processes inside the containment leads to limited releases (category FKF, 9% of core melt accidents).

If a hydrogen burn in the venting system does not affect the filter, but only results in leaks at ventilation ducts, then the release is filtered at roof level (category FKH, 12 % of core melt accidents)

If the depressurization of the containment proceeds as specified, the filtered release enters the atmosphere at stack height (category RCI, 38% of core melt accidents).

If there is no need for a depressurization, there is only an insignificant release through minor leaks and through the filtered emergency ventilation system of the annulus (category RCJ, 24% of core melt accidents).

Tab. 7 Correlation of initiating events with release categories

Definition of release category			Fractions of the core damage states					
Name; path to atmosphere	Fractions released		Initiating events and fractions					
(all release categories except FKJ have additional melt penetration of foundation)	Cs	J	L<25 0.56	L>25 0.05	LPR 0.15	LSG 0.09	TLP 0.11	TWP 0.04
FKA: Containment-> damaged annulus ->atmosphere	>0.5	>0.5	.005	0.0	0.0	.016	.009	<<
FKB: Cont.->ventilation->atmosphere *or* Cont.->annulus early->ventilation ->atmosphere	0.13... 0.24	0.14... 0.23	.004	<<	<<	.001	<<	<<
FKC: Bypass through steam generator ->steam relief valves->atmosphere	0.025.. 0.15	.025... 0.15	-	-	-	.07		
FKE: Cont.->annulus late->ventilation ->atmosphere	2e-4... 6e-3	.055	.04	<<	.01	-	.008	<<
FKF: Cont. depressurization->filter failure -> release at roof	6e-6... 1.2e-4	.0275	.012	<<	<<	-	.074	<<
FKH: Cont. depressurization->filter OK -> release at roof height	2e-7... 1e-5	.0001	.092	.004	.024	-	<<	<<
FKI: Cont. depressurization->filter OK ->release at stack height	2e-7... 1e-5	.0001	.29	.014	.076	-	<<	<<
FKJ: Small cont. leak->annulus -> filtered ventilation	3e-10... 2e-8	.0001	.12	.026	.045	-	.013	.033

The summary values for the release categories which are given above, are broken down in table 7 to the six initiating events. The following general tendencies can be observed:

1. Small and medium size leaks at the primary system (56% + 5% + 15% of core damage states) have a particularly large fraction of cases with a depressurization of the containment, including releases with and without damage at the venting system.
2. Leaks in steam generator tubes (9% of core damage states) always have a large release through the secondary system. A significant fraction shows an even increased release due to high pressure failure of the reactor pressure vessel.
3. Transients with loss of offsite power will end up in a depressurization of the containment. But due to loss of power the conditions in the venting system favor hydrogen burns damaging the venting filter.
4. Transients with offsite power available lead to benign consequences, because the core material can almost always be contained inside the reactor pressure vessel.

6.5 Uncertainty analysis

The distribution of about 120 uncertain input parameters for the event tree analysis was estimated. 500 Monte Carlo simulations have been done, and each individual simulation produces the frequencies of the various plant damage states. The aggregated result of the complete set of simulations can be transferred into a graph, where the y-axis is the frequency, and the x-axis is the number of simulations. Fig. 4 shows this type of visualization for the release categories. The lines indicate for each release category that fraction of Monte Carlo simulations, for which the respective frequency is exceeded. For example in 40% of all Monte Carlo simulations release category FKI has a frequency of less then 7.5E-7/a. This result can as well be interpreted as a degree of confidence: There is a confidence level of 40% that the frequency of category FKI will not exceed 7.5E-7/a.
For the severe release categories FKA and FKB the figure shows that FKA (or FBK) do not occur at all (frequency zero) with confidence levels of 70% (or 95%). If, however, confidence levels above 95% are taken into account, the frequencies of these categories exceed ~~0.~~5E-7/a. The main reasons for the uncertainties were evaluated by a sensitivity analysis. For the particularly important category FKA the main contribution comes from the uncertain temperature of the hot main coolant lines. These temperatures determine the probability for a failure of the pipes prior to the reactor pressure vessel failure.

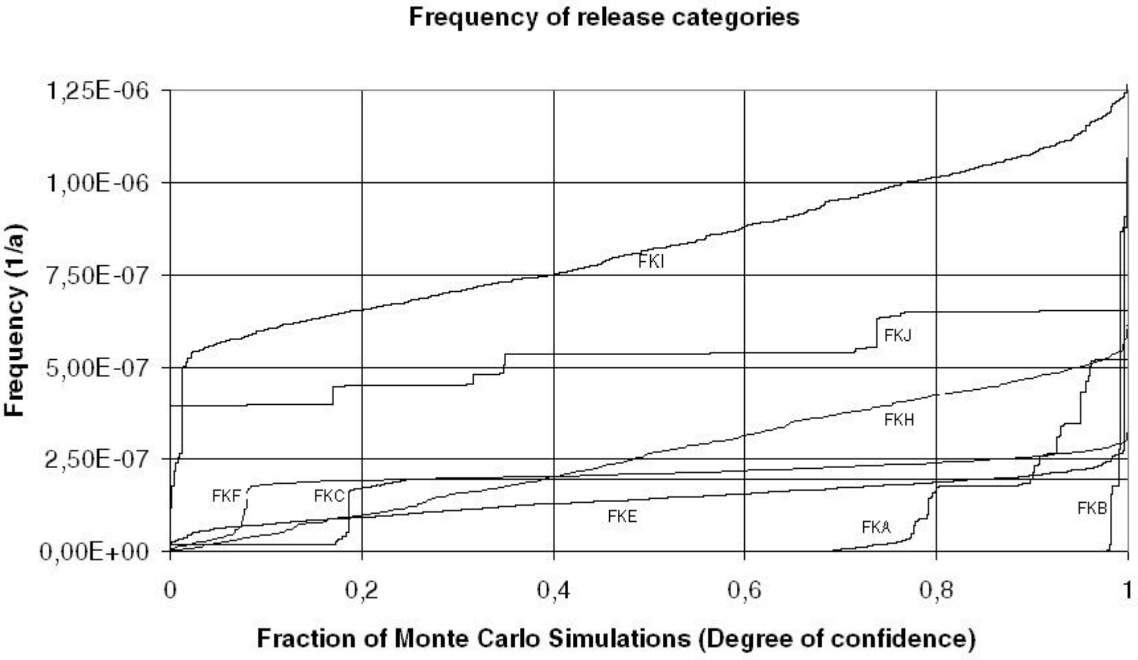

Fig 4 Uncertainty of frequencies for release categories

7. SUMMARY

The **methodology** for the event tree analysis which is applied for this study basically is similar to the methodology developed within the US for the NUREG-1150 report. It consists of a large event tree (more than 80 branching points), including several simple computer models for the evaluation of certain branching probabilities. Distributions of uncertain input values were defined and taken into account in 500 Monte Carlo simulations.
The state of the probabilistic methodology is sufficient to create meaningful results which cannot be obtained otherwise. In particular, the representation of the complete accident progression including uncertain phenomena and human actions is far beyond the capabilities of other present-day computer codes. It allows to identify dependencies and quantitative comparisons of the influence of different phenomena.

A Level-1 PSA has generated a set of 66 different **core damage states** for the Level-2 analysis. Each core damage state is defined by about 10 different properties. The point value of the total frequency of all core damage states, delivered by the Level-1 analysis is 2.2E-6/a. For each of the core damage states the point value of its frequency was transferred to the Level-2 analysis. There are six groups of initiating events. The following general conclusions can be drawn:

1. Transients almost always have early core damage states at high pressure. If there is no loss of offsite power, emergency core cooling system are available.
2. Small leaks are the most frequent initiating event for core damage states. The pressure mostly is low, emergency core cooling systems are not available in general.
3. Steam generator tube leaks develop core damage states only after significant time periods. There is medium pressure, emergency core cooling systems are not available in most cases.

Containment threats were evaluated as follows:

If there is a failure of the reactor vessel bottom at pressures above 8 MPa, a subsequent damage at the containment and the annulus of the reactor building is assumed.

A parametric model for *steam explosions inside reactor pressure* vessel taking into account large uncertainties shows no failure of the reactor pressure vessel head (alpha mode failure). There is a small probability for pressure vessel bottom failure.

Passive autocatalytic recombiners limit *hydrogen volume fractions in the containment* atmosphere. Hydrogen burns are frequent, but almost never causing failures. The late containment atmosphere is almost free of oxygen, but with large quantities of hydrogen and carbon monoxide.

A parametric model calculates less than 1% of core melt accidents with containment failure due to *direct containment heating* at reactor pressure vessel failure. Uncertainty is large, however, and additional work is in progress.

After penetration of 0.7 m of concrete, *melt released from the reactor vessel bottom* enters ventilation ducts. Eventually a bed of frozen core material particles can develop inside the sump suction line, beyond the containment boundary. About 4% of core melt accidents have containment failure due to meltthrough of sump suction line.

During containment depressurization the gas from the containment enters ventilation ducts on the top of the service building. Assumptions about ignition sources and effects of fire lead to defects at the venting filter (9% of accidents) or at the ventilation ducts (12% of accidents).

The **final results** of the PSA Level-2 can be summarized as follows (numbers indicate mean probabilities for all plant damage states):

The most likely final location of core material is determined by the core-concrete interaction which eventually will reach the underground below the foundation. A probability of about 23% is assigned to the retention inside the reactor vessel.

Significant final containment modes are as follows:
3% containment damage due to high pressure failure of reactor vessel
4% meltthrough of sump suction line
3 % overpressure failure due to unavailability of containment depressurization
67% depressurization, partly with failures at the venting system due to hydrogen burns
23% insignificant leak of containment, reactor vessel intact.

The amount of radionuclides released to the atmosphere has been evaluated, taking into account the final containment states and the release path to the environment. The releases have been grouped into a set of release categories.

Uncertainties of the results have been evaluated. If a high degree of confidence is desired, the probabilities given above for severe damage increase significantly.

As a general summary one can conclude that the method of event tree analysis has been applied successfully. Meaningful results have been obtained which could not be acquired by other methods. The PSA level-2 event tree analysis is a valuable tool for safety evaluation.

8. REFERENCES

[1] Gesellschaft für Reaktorsicherheit (GRS) mbH
Deutsche Risikostudie Kernkraftwerke Phase B
ISBN: 3-88585-809-6, 1990

[2] U.S. Nuclear Regulatory Commission
Severe Accident Risks:
An Assessment for five U.S. Nuclear Power Plants
NUREG-1150, December 1990

[3] Nuclear Reactor Commission
"MELCOR 1.8.4 - Reference Manual",
USA, Washington, July 1997

[4] Köberlein, K.
Comprehensive technical assessment of an advanced German PWR by PSA – Objectives and main results
Eurosafe, November 6-7 2000, Köln, Germany

[5] Sonnenkalb, M. et al.
Untersuchungen von Maßnahmen des anlageninternen Notfallschutzes zur Schadensbegrenzung für LWR - Zwischenbericht zum Vorhaben SR 2306
GRS-A-2735, November 1999

[6] Grebner. H.
Grenzbelastbarkeit einer DWR-Kühlkreislaufschleife bei einem Kernschmelzszenario mit hoher Temperatur und hohem Druck
GRS, Technische Notiz, August 2000

[7] Krieg, R. et al.
Reactor Pressure Vessel Head Loaded by a Corium Slug
Vortrag, eingereicht zur 15. Int. Conf. on structural mechanics in reactor technology (SMiRT), Seoul, August 1999

[8] Breitung, W., Redlinger, R.
A Model for Structural Response to Hydrogen Combustion Loads in Severe Accidents
Nuclear technology, Vol 111, Sep. 1995, pp 420-425