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Author(s)	Tyrväinen, Tero; Silvonen, Taneli
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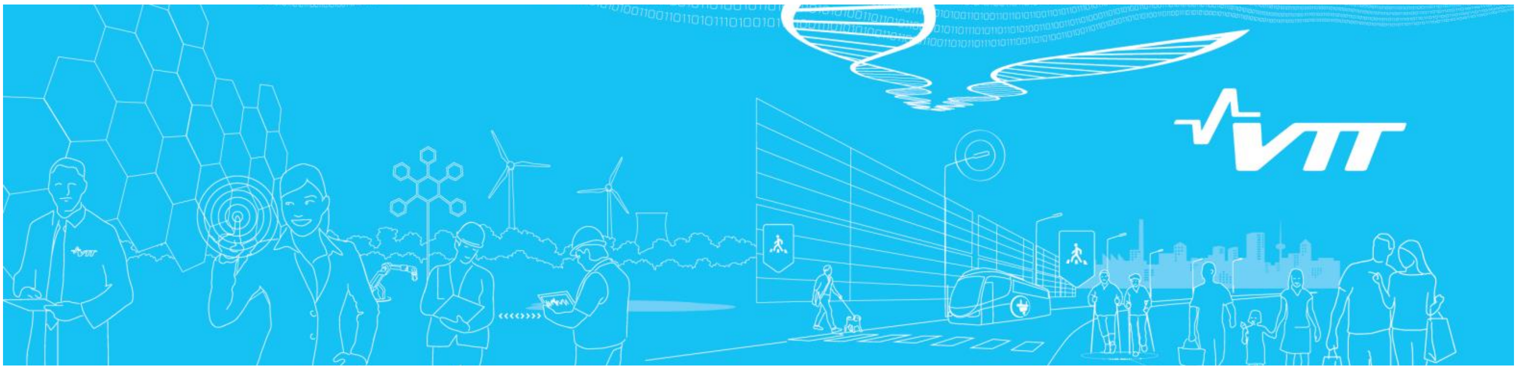

RESEARCH REPORT VTT-R-04473-15

Summary on integrated deterministic and probabilistic safety assessment development and case studies

Authors: Tero Tyrväinen

Based on work by Taneli Silvonen

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Summary

Integrated deterministic and probabilistic safety assessment (IDPSA) combines two methodologies to support risk-informed decision making in nuclear power plant safety considerations. In IDPSA applications, information on physical phenomena is obtained from deterministic analyses, and probabilistic risk models are constructed based on that information. This report concerns the application of IDPSA to the analysis of severe nuclear power plant accidents and level 2 probabilistic risk assessment (PRA). Level 2 PRA analyses how large and probable are radioactive releases after core damage. Severe accidents involve complex physical phenomena of which information can be best gathered by performing deterministic analyses.

This report summarises case studies conducted on passive reactor cooling systems, hydrogen recombinators and explosions, steam explosions, and the coolability of ex-vessel debris. These systems and phenomena, except passive reactor cooling system, were modelled in level 2 PRA based on supporting accident progression simulations. The studies provided useful insight on the phenomena and their modelling in level 2 PRA.

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Espoo 4.12.2015 **Written by** Tero Tyrväinen Research Scientist	**Reviewed by** Ilkka Karanta Senior Scientist	**Accepted by** Eila Lehmus Head of Research Area
VTT's contact address VTT, PL 1000, 02044 VTT		
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Contents

1. Introduction

Integrated deterministic and probabilistic safety assessment (IDPSA) combines two methodologies to support risk-informed decision making. IDPSA takes both stochastic disturbances and deterministic response of a nuclear power plant, and especially their mutual interactions, into account in safety justifications. The methodology can also reveal new plant weaknesses and reduce conservatism in the analysis.

This report concerns the application of IDPSA to the analysis of severe nuclear power plant accidents and level 2 probabilistic risk assessment (PRA). Level 2 PRA analyses how large and probable are radioactive releases after core damage. Severe accidents involve complex physical phenomena of which information can be best gathered by performing deterministic analyses.

This report summarises work performed in five IDPSA related reports [1-5], and most of the texts of this report are more or less taken from them. The first report [1] studied reliability analysis of a passive containment cooling system. In the second report [2], reliability analysis of passive autocatalytic hydrogen recombiners was performed. The third report [3] analysed steam explosions, and the last reports [4, 5] continued the study with more advanced steam explosion simulations and ex-vessel debris coolability analysis.

Chapter 2 provides an introduction to level 2 PRA. The analysis of passive systems is addressed in Chapter 3. Chapter 4 presents the steam explosion research, and Chapter 5 concludes the IDPSA studies.

2. Level 2 probabilistic risk analysis

Level 2 PRA analyses severe nuclear reactor accidents after core damage has occurred and calculates the amounts and frequencies/probabilities of radioactive releases. Level 1 PRA provides plant damage states as inputs for level 2 PRA, if the PRA levels are integrated. A plant damage state consists of accident scenarios with similar plant conditions and accident progression on level 2. Typically, a containment event tree (CET) is developed for each plant damage state in level 2. CETs model the progression of severe accidents from core damage to releases. At the end points of a CET, source term calculation is performed to obtain the level 2 consequence results. Figure 1 shows the general main steps for performing a level 2 PRA and how it relates to other levels of PRA [6].

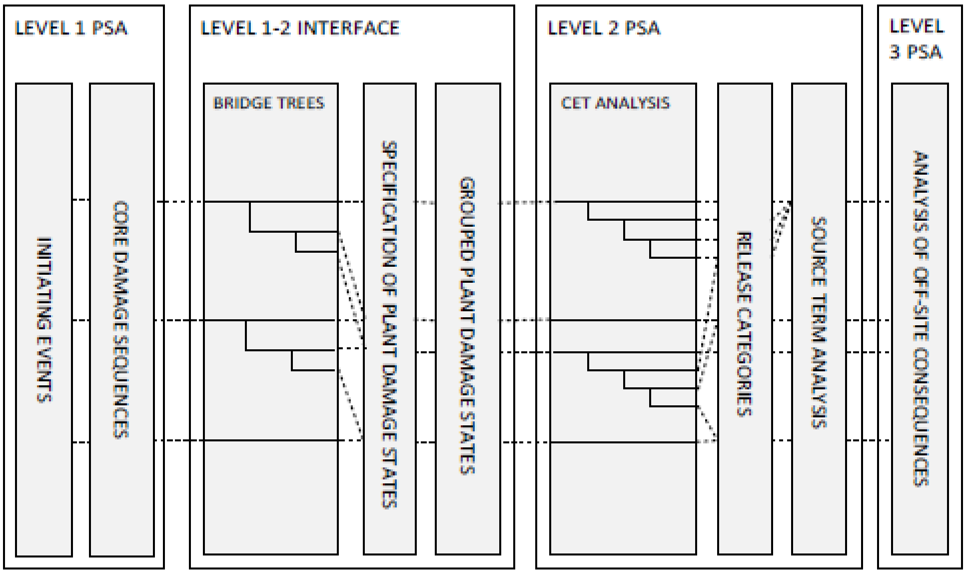

Figure 1: Overview of the development of a typical level 2 PRA and its connection to level 1 and level 3 PRA.

2.1 Accident progression analysis

General tasks in developing event trees include the definition of safety functions and event tree headings, and determination of system success criteria. The top events or nodal questions in a CET should address the events and physical processes that govern accident chronology and plant response to severe conditions. For example, whether the containment is isolated or not, and does an early hydrogen combustion occur are typical nodal questions.

The analysis of the progression of severe accidents can be performed using computer codes for severe accident simulation. Such codes should be capable of modelling most of the events and phenomena that may appear in the course of the accident. Key variables, such as pressures and temperatures, are used in the quantification of a containment event tree. [7]

The objective of an assessment of containment performance is to develop a realistic characterization of the modes of the containment failure as well as to develop failure criteria under severe conditions [6]. Therefore, detailed information on the structural design of the containment should be collected. One approach in containment performance analysis is to define a threshold pressure at which the containment is expected to fail.

2.2 Quantification of the containment event tree

The assignment of conditional probabilities to branches of the containment event tree is significant to the results of the analysis. Relevant information sources to support the assignment of probabilities are, for instance, deterministic computer codes to model accident progression, experimental measurements, studies of similar plants and expert judgement.

Level 2 PRA calculates the source terms associated with the end states of a CET. A source term mainly determines the quantity of radioactive material released from the plant to the environment, but can also contain other attributes, such as the time and height of the release. The end states of the CET are grouped into specified release categories, and source terms are calculated for the release categories. [6]

2.3 Dynamic containment event tree analysis

Software tools FinPSA [8] and SPSA [9] offer dynamic containment event tree approach that supports IDPSA. The PRA modelling in the IDPSA reports [2, 3] was performed using SPSA. FinPSA is an updated version of SPSA developed by VTT.

The level 2 modelling in FinPSA and SPSA is based on dynamic containment event trees and containment event tree programming language (CETL). The CET language is used to define functions to calculate conditional probabilities of event tree branches, timings of the accident progression and amounts of releases. A CETL function is defined for each branch of a dynamic containment event tree, and the model also contains an initial conditions section, where some probability and process variable values are defined, and a global “common section”, where some global variables and functions can be defined. CETL programming is very flexible. At any branch, new values can be calculated for any variables that are defined in earlier sections, and that way accident progression can be modelled realistically. In the initial conditions, binning rules can be defined to divide the end points of the CET into release categories.

To account for uncertainties related to variable values, it is possible to define value distributions and perform Monte Carlo simulations. At each simulation run, a value is sampled from each defined distribution, and based on that, conditional probabilities are calculated for all the branches, and values are calculated for all variables at each end point of the CET. After the simulations, statistical analyses are performed to calculate frequency and variable value distributions for each end point and release category among other statistical results and correlation analyses. It is also possible to just calculate point values of the CET based on the mean values of distributions.

3. Passive systems

Passive safety systems differ from active systems so that they do not need any external input to operate. They depend on natural phenomena like gravity. Safety systems of a modern nuclear plant are implemented so that they combine both passive and active safety features. In future plant designs, passive systems are used increasingly because they are regarded more reliable and simpler than their active counterparts. It is also advantageous that the need for human interaction and external signals is reduced. The reliability analysis of passive systems requires more research, and passivity makes it necessary to consider the definition of a system failure from a somewhat unconventional point of view.

This chapter is divided into two parts. First, hydrogen management with passive autocatalytic recombiners is studied. Second, the reliability analysis of a passive containment cooling system is summarised.

3.1 Hydrogen management

3.1.1 Hydrogen hazards

Hazards relating to hydrogen production during the progression of an accident sequence are quite well studied and taken into account in safety strategies of nuclear power plants. The combustion of hydrogen can create pressure or detonation forces that may exceed the strength of the containment structure and lead to an early containment failure. During an accident progression, highly combustible hydrogen is primarily produced as a result of heated zirconium reacting with steam. Pre-inertization with nitrogen, igniters and passive autocatalytic recombiners (PAR) are often used means for hydrogen management. [10]

Hydrogen production, distribution and combustion are very complex and highly plant and accident scenario specific phenomena. Hydrogen combustion can take place in a variety of forms: mild deflagration, fast or accelerated flames, deflagration to detonation transition and detonation. Hydrogen release rate and total amount of hydrogen in the containment are examples of important factors to be studied in hydrogen hazard analysis. Local and global hydrogen concentrations are determined by the distribution of the released hydrogen, and they have an impact on the combustion modes. An understanding of these phenomena is crucial for planning and implementing effective hydrogen management measures, such as use of recombiners.

The major risk from hydrogen combustion is the risk for loss of containment integrity and failure of safety systems. Pressure loads can cause structural damage and thermal loads can cause damage to cables and components. In order to evaluate the actual risk from these loads, the structural response of the containment must be analysed. Hydrogen risk can be analysed in the PSA with help of computational tools capable of calculating production, distribution and other hydrogen-significant factors.

Combustion of hydrogen takes place if the hydrogen-steam-air gaseous mixture is flammable and an ignition source is present. Autoignition temperature makes it possible for combustible gas mixture to spontaneously ignite. The required ignition energy is dependent on hydrogen concentration. For flammable gases, the flammability limits are the experimentally determined minimum concentration of fuel and oxidant required for self-sustaining flame propagation at a given pressure and temperature. Flammability limits also depend on parameters such as turbulence and temperatures of gas mixtures. Indicative flammability limits of hydrogen-air-steam mixtures in terms of volume fractions are often displayed in triangular diagrams, called Shapiro diagrams. Figure 2 presents a Shapiro diagram at standard atmospheric pressure for 100% relative humidity (after [11]). The German guidelines on PSA [12] provide a simplified necessary condition for estimating whether a gas mixture is combustible:

$$H_2O < 100 - 37.3 \cdot exp^{-0:007 \cdot H_2} - 518 \cdot exp^{-0:488 \cdot H_2};$$

where H_2O and H_2 are volume fractions for steam and hydrogen. The curve determined by the equation is shown in *Figure 3*.

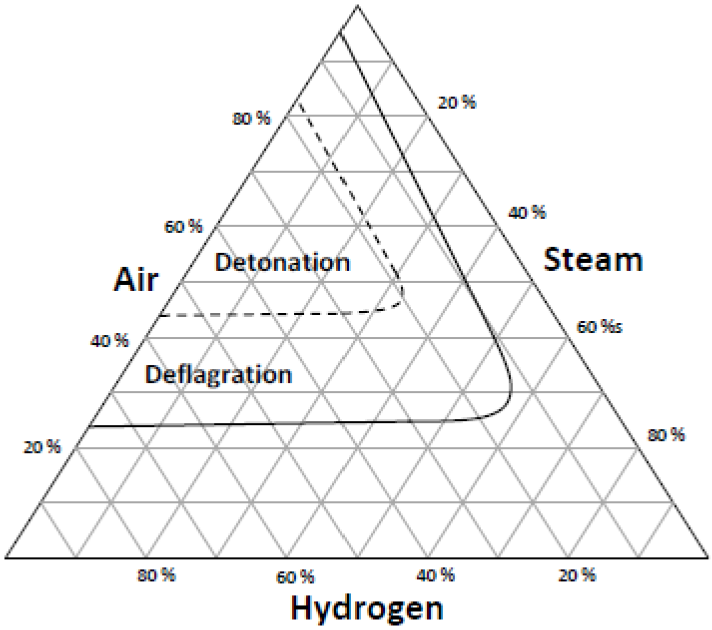

Figure 2: Shapiro diagram for the flammability limits of hydrogen-air-steam mixtures at standard atmospheric pressure for 100% relative humidity. Percentages refer to volume fractions.

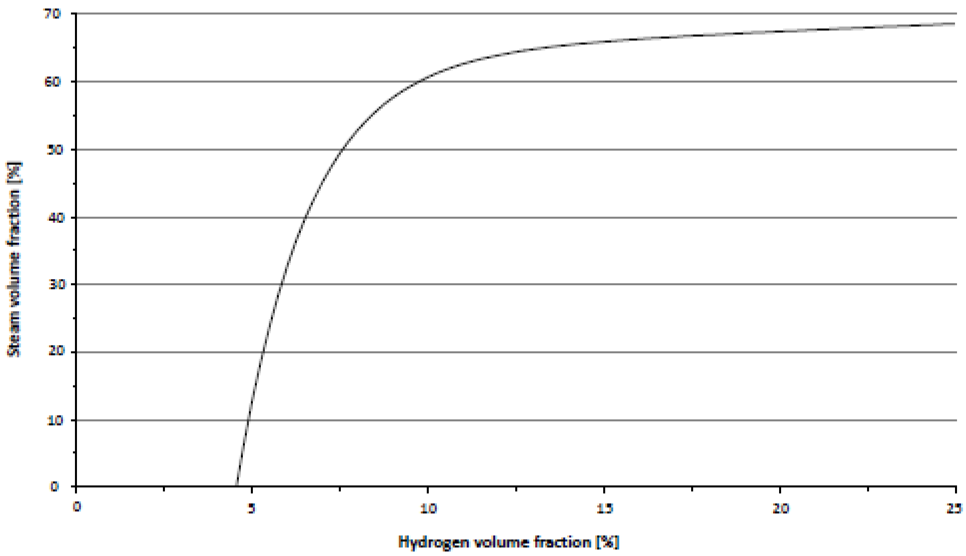

Figure 3: Simplified necessary condition for the combustibility of a gas mixture. Combustion is possible in the area below the curve.

Hydrogen hazard analysis can be planned as a systematic stepwise process as outlined in Figure 4 and discussed in [10]. Hydrogen production is calculated for all relevant accident scenarios, and a suitable computer code is used to calculate hydrogen and temperature distributions. Whether there are hydrogen mitigation measures present or not must be included in the calculations. After potential ignition, the combustion mode is determined using appropriate criteria, and thereafter the pressure loads are calculated to determine the structural response of the containment. In Figure 4, DDT means deflagration to detonation transition.

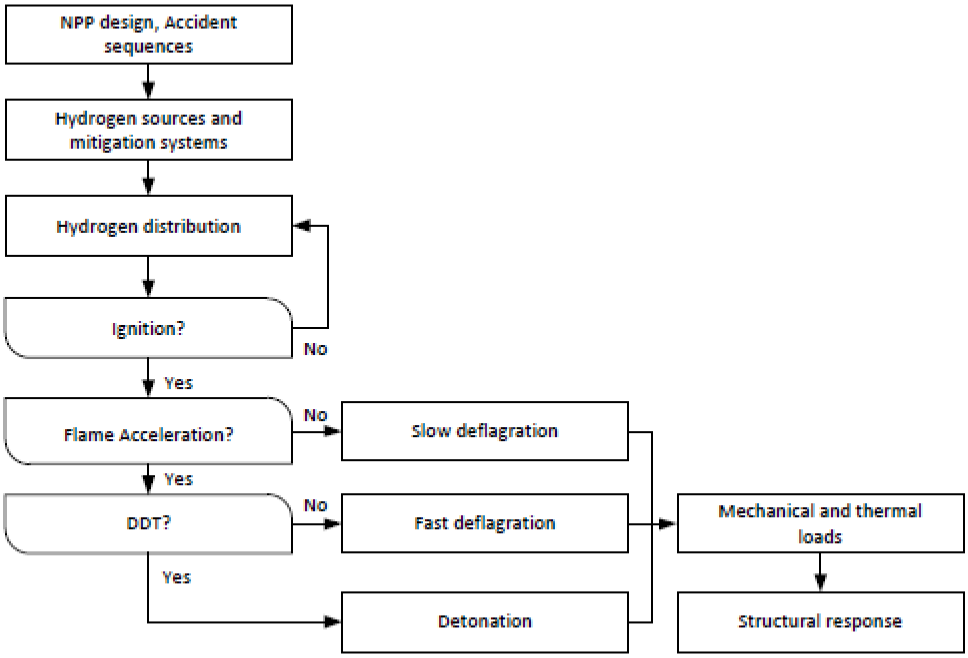

Figure 4: Roadmap for hydrogen hazard analysis under severe accident conditions.

3.1.2 Autocatalytic recombiners

Passive autocatalytic recombiners (PAR) are rather simple devices consisting of catalyst surfaces arranged in an open ended enclosure as shown in Figure 5, drawn after [13]. In the presence of hydrogen, a catalytic reaction involving hydrogen and oxygen, and producing steam occurs spontaneously at the catalyst surface. The physical phenomena taking place in the flow channel between two catalyst sheets is also illustrated in Figure 5 [14, 15]. The heat of the reaction produces natural convection through the enclosure, ensuring continuous gas supply. PARs do not need external power or operator action.

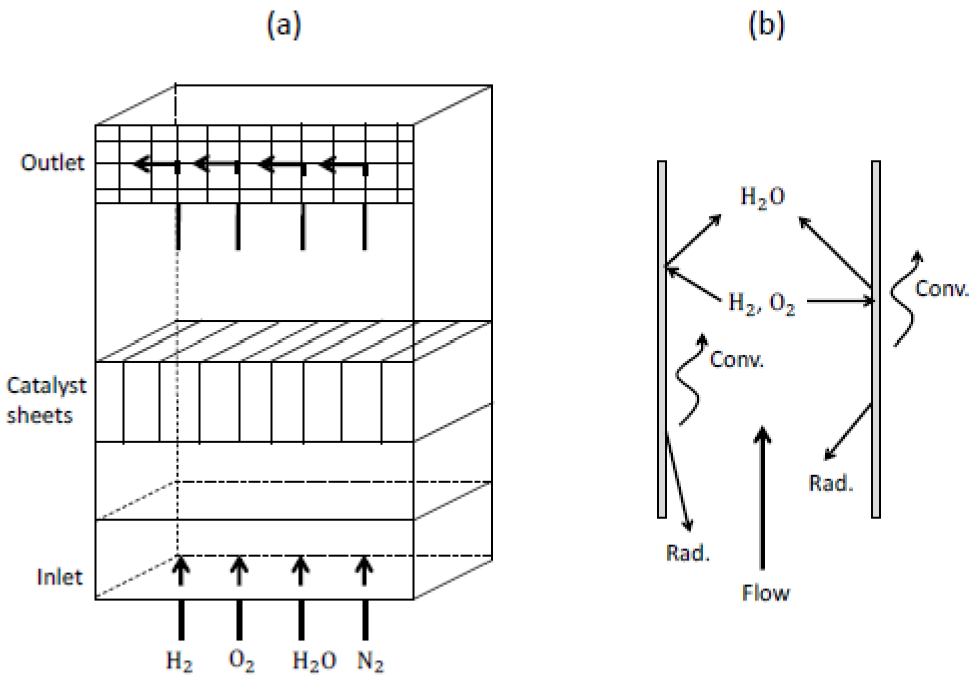

Figure 5: (a): Concept of a passive autocatalytic recombiner. (b): Physics phenomena in the catalyst flow channel.

Catalytic recombiners use catalysts to oxidize the hydrogen according to the reaction in the following equation and are operable outside the limits of flammability.

$$H_2 + \frac{1}{2}O_2 \rightarrow H_2O + 238\ \frac{kJ}{mole}$$

The reaction starts only after overcoming high required activation energy, which can be significantly reduced by the use of catalysts, such as platinum or palladium. In experimental tests, a PAR comes into action spontaneously as soon as the hydrogen concentration reaches 1-2 vol.% depending on the containment atmosphere temperature and humidity [11].

Table 1 presents a simple outline of failure modes and effects analysis (FMEA) for a PAR unit. Failures are divided into catalyst sheet related and natural phenomena related failures. Failures of PARs are often not observed until the consequences emerge, depending on the failure cause, which can be separated into standby (“S” in Table 1) and on-demand (“D” in Table 1) causes.

Table 1: A simplified failure modes and effects analysis for a PAR system.

Component or function	Failure mode	Failure cause	Consequence	Identification in advance
Catalyst sheets	Degraded capacity	Poisoning (S, D)	Less H_2 processed	Performance tests
	Sheets destroyed or unavailable	Unintended ignition (D), External cause (S, D)	Unit not operational	Visual inspections
Natural phenomena	Degraded capacity	Oxygen starvation (D), Impaired gas inlet (S, D)	Less H_2 processed	Performance tests

3.1.3 Probabilistic modelling in a case study

In [2], a case study on the PAR system of Loviisa 1&2 was performed. The chosen case was a small break loss of coolant accident, which is considered important with respect to hydrogen build-up. A simplified containment event tree model was built for the accident. Only hydrogen related phenomena were modelled more in detail. Reliability estimates for PAR system were based on common cause failure models and accident progression simulations performed using MELCOR software [16].

There are two kinds of hydrogen recombiners in Loviisa containment, with and without a chimney. Little different PAR parameters are used for the two types of recombiners. In total, there are 154 PAR units in the containment.

The simulations of hydrogen phenomena during accident progression were conducted using MELCOR model created in 2007 at VTT for Loviisa 1&2 power plant [17]. The aim was to evaluate whether hydrogen combustion conditions are met during the accident, using the two criteria introduced in Section 3.1.1 and Figures 2 and 3. Besides, the total amount of hydrogen removed by PARs was monitored. The upper containment dome was chosen to represent the whole containment regarding hydrogen conditions, although the conditions could be more severe in other parts of the containment. However, big differences in atmospheric compositions are considered unlikely, because the natural circulation keeps the containment well mixed.

According to the simulation results, PAR system seems to be an effective method to remove hydrogen from the containment atmosphere. Even with 50% degradation in performance, flammable gas mixtures are avoided for a great part of the accident time. On the other hand, without PARs, combustible conditions are inevitable, and if there was an ignition source, a burn would occur and challenge the containment integrity.

Figure 6 shows how the maximum hydrogen volume concentration in upper containment dome during the accident depends on PAR performance. According to the simulation results regarding combustibility, 5.5% hydrogen concentration was selected to represent a critical value for hydrogen concentration. At least 70% PAR performance is required to avoid higher hydrogen concentrations throughout the accident progression. However, because the choice of the critical limit was somewhat arbitrary and the MELCOR results surely contain uncertainties, this conclusion should be regarded only as indicative.

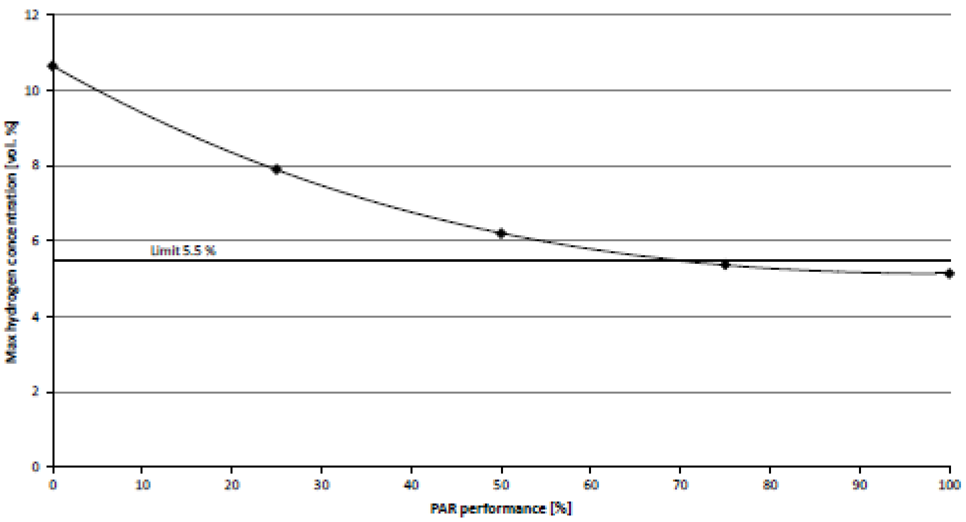

Figure 6: The dependence of maximum hydrogen volume concentration on PAR system performance.

The PAR system’s on-demand reliability was estimated with two different common cause failure models: Alpha-factor [18] and Extended Common Load [19] models, and the latter one turned out to be more convenient. For Alpha-factor modelling, the PAR system was considered as an entity constituted by 5 groups, and for Extended Common Load model, the PAR system was considered as an entity constituted by 10 groups. A failure of a single group was not interpreted as a complete failure of all units in that group, but instead as a 10/20% decrease in the performance of the whole PAR system. System failure rate was estimated using several parameter sets and failure criteria for both models to study the sensitivity of the estimate. However, there was no reliability data to help with parameter estimation which undermines the credibility of the reliability analysis to an extent.

The reliability estimate of PARs was combined with other case-significant information in a containment event tree model. The model calculated releases to environment as fractions of the initial core inventory for three radionuclide groups. Most of the nodal questions in the CET were modelled without much detail by giving simple distributions for the failure probabilities. More effort was put in the modelling of issues important with regard to hydrogen management.

Intense hydrogen source can be caused by core reflood in critical time window. This is possible when there is no core cooling available, the core is exposed and overheated and if the core has maintained its geometry at least to some extent. The emergency core cooling system is initially unavailable, but is repaired and refloods the core at a critical time causing much more rapid hydrogen formation than normally. It is assumed that the critical time window starts some time after core damage depending on core decay heat (see details in [2]). The critical time window is assumed to shut when the core has lost its geometry and it is collapsed to the bottom of the pressure vessel. The probability of this phenomenon was derived with help of the level 2 PRA made by the utility [20].

The containment hydrogen concentration was modelled to be dependent on the state of the PAR system and whether the core is reflooded in the critical time window or not. The effect of hydrogen igniters was not explicitly taken into account, but it can be thought to be included in the rather conservative hydrogen management failure probability estimates. The maximum hydrogen concentrations emerging during accident progression were divided into three categories: low, medium and high. The probabilities obtained using the common cause failure models for PAR performance decrease and the MELCOR simulations were used to evaluate different hydrogen concentration likelihoods.

If the core is reflooded in the critical time window, low hydrogen concentrations are regarded unachievable and maximal PAR performance is required for avoiding high hydrogen amounts. In the case of no critical reflood, high hydrogen amounts are assumed if the PAR system is completely unavailable. Medium concentrations take place when the performance is 10-80%. The probability ranges of hydrogen concentrations are presented in Table 2.

Table 2: Probability ranges of different hydrogen concentration maxima.

Reflood in critical time window	Probability intervals for maximum hydrogen concentrations		
	Low (0-5%)	Medium (5-10%)	High (over 10%)
Yes	0.00	0.80-0.90	0.10-0.20
No	0.97-0.99	0.01-0.02	0.00-0.01

Hydrogen concentration alone does not determine whether the potentially flammable gas mixture causes concerns with regard to containment integrity. However, with higher hydrogen concentrations, flame propagations are more likely, and even detonations are possible. Low concentration burns, on the other hand, do not usually cause high pressure loads which would risk the containment integrity. Thus, in the CET model, it is assumed that the probabilities of hydrogen management failure leading to early containment failure are 0.1%, 5% and 50% for low, medium and high hydrogen concentrations, respectively. Consequently, the expected value for the probability of hydrogen management failure is approximately 0.095 given core reflood in critical time window and 0.0042 otherwise.

The results of CET analysis show the importance of the hydrogen management because hydrogen failures constitute 5% of the early release frequency. Although only hydrogen management related issues and source term calculation were modelled in a bit more detail, the results in general appear credible.

3.2 Containment cooling system

Passive containment cooling system (PCCS) is designed to provide steam suppression in the drywell in the event of a loss of coolant accident (LOCA). PCCS relates generally to protection systems for shutting down a boiling water reactor (BWR) and maintaining it in a safe condition in the event of a system transient. PCCS does not contain power actuated valves or any other components that must function actively. It is a thermal-hydraulic (T-H) system, which contains moving working fluids and relies on natural circulation.

PCCS's operation is initiated by the difference in pressure between the drywell and the wetwell. The PCCS condenser receives a steam-gas mixture supply directly from the drywell. The gases are cooled in heat exchangers (HEX), and some or all of the steam vapour is condensed. The condensed steam is drained to the gravity driven cooling system (GDCS) pool and the non-condensable gases are vented through the vent line, which is submerged in the pressure suppression pool. The vent line functions whenever the drywell-wetwell pressure differential is sufficient to clear the water from the vent line terminus within the pressure suppression pool. A schematic picture of the PCCS is shown in Figure 7. [21]

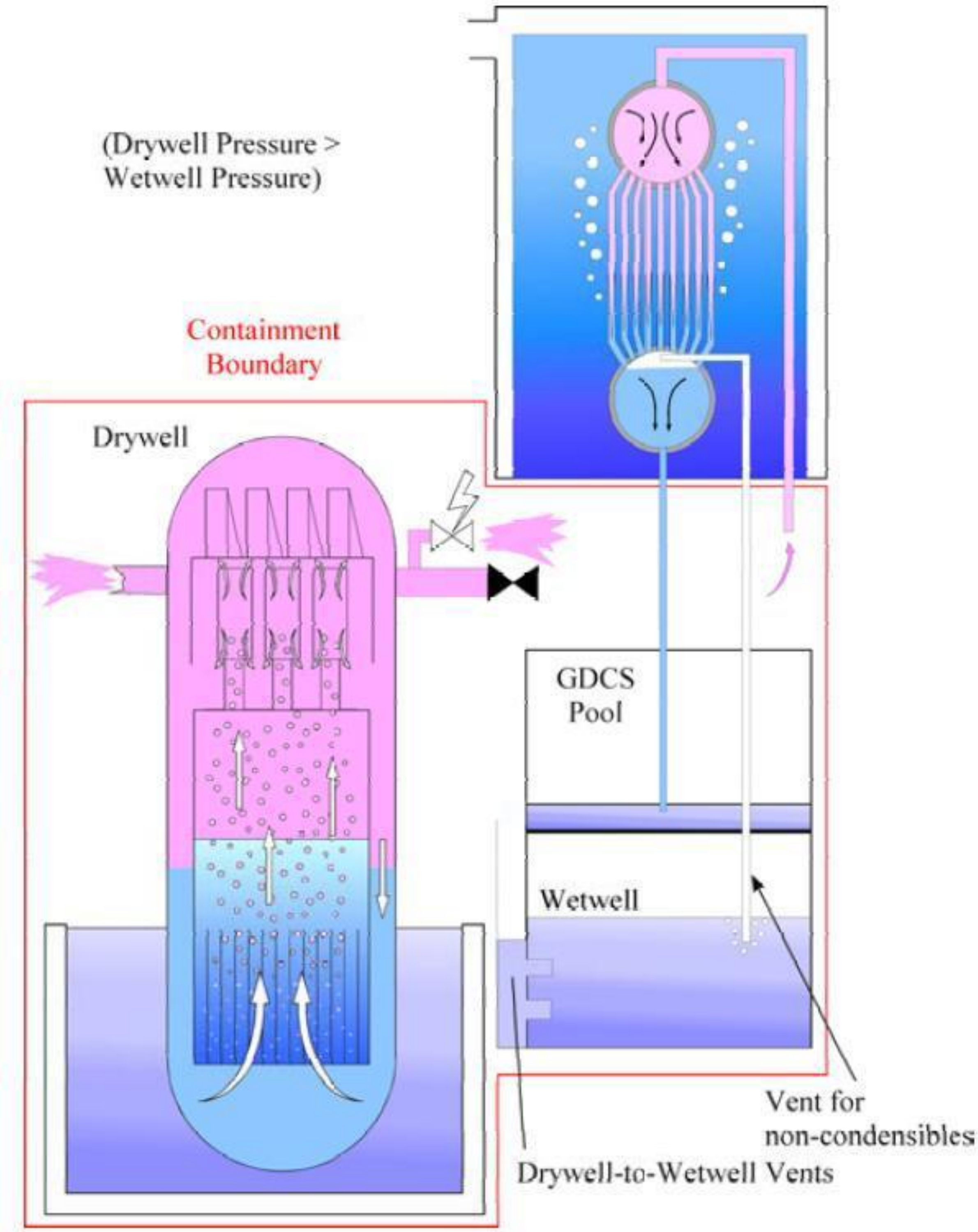

Figure 7: A schematic picture of the passive containment cooling system. [22]

The reliability assessment of the PCCS, or any other passive thermal-hydraulic system, necessarily differs from classical component reliability based approach. The significant uncertainty related to thermal-hydraulic system operation is complicating reliability analysis. However, also methods which apply classical reliability assessment techniques have been developed. Many other techniques rely on computer programs, which simulate the thermal-hydraulic physical phenomena related to the system under investigation. When the failure criteria have been defined, it is possible to provide a reliability estimate for the passive safety system according to the simulations. The assessment may also be purely qualitative without giving any numerical values to evaluate the reliability.

The reliability analysis of a PCCS was performed in [1]. Table 3 presents failure modes and effects analysis (FMEA) for PCCS. Failure rates of heat exchangers, piping and natural circulation were obtained from [23] and mission time 72 was used. The failure probability of the system was estimated by summing the failure probabilities of different components and it was $1.066 \cdot 10^{-5}$. 83.76% of it came from the failure of natural circulation.

Table 3: Simplified FMEA for PCCS.

Component/function	Failure mode	Failure cause	Consequence	Identification
Heat exchanger	1. Pipe rupture 2. Pipe plugging		System not operational	
Piping	Pipe rupture		System not operational	
Natural circulation	Insufficient heat transfer	1. Insufficient water in PCCS pool 2. Pipe fouling	Decreased heat transfer capability	Water level indicators
	Envelope failure	Piping rupture	System not operational	
	High concentration of non-condensable gases	Vent line fouling	No vent line flow	

The PCCS was also investigated with help of MELCOR [16] simulations. The model represented a BWR nuclear power plant in Olkiluoto. A PCCS was not originally in the model and was added for this study. When running simulations using a condenser package provided by MELCOR, the PCCS did not work as efficiently as it should have worked. The drywell pressure and temperature did not stay in the control limits even though the PCCS is designed to secure tolerable atmospheric conditions during a transition phase. Also, the energy amounts transferred to the PCCS pool in the station blackout accident scenario (SBO) should presumably have been considerably bigger. The main reason for the rather poor performance is that the required pressure differential between the drywell and the wetwell did not exist for the most of the time. Without the needed conditions, the natural circulation is dead and no heat transfer takes place.

Some better results regarding system performance were obtained when the accident scenario was changed to represent a main steam line type of LOCA. The changes improved the PCCS operation and the energy amounts transferred to the PCCS pool were manifold compared to the SBO case. Despite the slight improvements in the development of the drywell atmospheric conditions, pressure and temperature rose too high, and the PCCS did not work well enough in the LOCA simulations either.

The problems encountered were not necessarily only modelling related. Systems based on natural circulation can be very sensitive and they must be designed thoroughly and carefully, requiring accurate calculations. Obviously, in Olkiluoto designs, the addition of such a system is not taken into account e.g. in the dimensions of the containment. It is possible, perhaps even probable, that also in reality an afterwards installed PCCS would not function desirably in the given setting. In this respect, it is understandable that the results turned out to be somewhat modest. Also, there are not many safety systems operating in the scenarios examined. Thus, the workload for PCCS is too big and the system alone cannot cope with the extreme conditions emerging.

The effects of the number of PCCS units were also examined. As the PCCS did not improve the drywell conditions much, the quantities of interest were chosen to be the energy amount transferred to the PCCS pool and the vent line flow of non-condensable gases from drywell to the suppression pool in wetwell. The results were as anticipated. The more heat exchanger units available, the better the system performance is. Also, a higher water level enhanced heat transfer capability.

4. Steam explosions and debris coolability

A severe nuclear accident that leads to a core meltdown can escalate (among other undesired events) into a steam explosion which can take place if molten fuel gets in contact with water and vaporizes it rapidly. More generally such processes are called fuel coolant interactions (FCI). Steam explosions are considered plausible both in the reactor pressure vessel (RPV) and underneath it in the lower drywell (LDW) of containment. An in-vessel steam explosion can lead to a so called alpha-mode containment failure, but the probability of such event is at present considered almost negligible. Ex-vessel explosions are regarded more hazardous if vessel melt-through occurs and melt is ejected into flooded LDW. This report concentrates more on ex-vessel explosions.

Steam explosions concern all light water reactor (LWR) plants, i.e. both BWRs and pressurized water reactors (PWR) are prone to such events. General phenomenology and the potential threat to the containment integrity posed by steam explosions are more or less the same for all light water reactor types but some differences exist nonetheless. Structures in the lower plenum of the RPV are different (e.g. guide tubes in BWRs and flow plates in PWRs) which affects the premixing of melt and water and the failure mode of the vessel. Melt contains usually more metallic parts in BWRs which can affect explosion characteristics. Here, steam explosions are primarily discussed from BWR perspective, because the case study of [3] deals with Olkiluoto 1 & 2 BWR units.

There are certain events and phenomena that need to precede a steam explosion. A core meltdown can basically be due to overpower or undercooling conditions, which typically relate to reactivity transients and LOCA, respectively. After the core is uncovered, the fuel temperature increases and several oxidation processes creating more heat and hydrogen begin. Meltdown itself starts when heat production rate in the core exceeds heat removal rate. When temperatures raise high enough, core relocation processes begin, starting from the relocation of molten fuel cladding materials. Eventually, the core collapses, and when the core support plates fail, the molten corium slumps into the lower plenum of the RPV. If there is water in the lower plenum, an in-vessel steam explosion can occur.

When the molten corium has been relocated to the lower head of the vessel, the core can be cooled from inside with water injections to keep the melt in the vessel. But the RPV lower head may fail despite cooling efforts, although probably later than without in-vessel water injection. In that case, if there is enough water in the cavity below the vessel, an ex-vessel steam explosion may take place when the molten corium jet reaches the water.

The steam explosion phenomena can be divided into four steps which are:

- Premixing
- Triggering
- Propagation
- Expansion and energy release

Large-scale steam explosions associated with melt pour entering water can be presented conceptually as in Figure 8 (after [24]). When the corium melt penetrates and breaks up into water in a film boiling regime, it creates a melt-water-steam mixture, i.e. there is an insulating vapour film between melt and water (premixture). At this stage, the melt fragments are typically of cm scale. Due to the vapour film, heat transfer from corium to water is relatively low. For a steam explosion to occur, a triggering event is required. For example, the melt hitting the bottom of the LDW can be a trigger. The trigger induces vapour film around melt fragments to collapse locally, and melt-water contact takes place resulting in fine fragmentation of the melt. Rapid heat transfer and high pressurization follow. The process escalates and propagates to all the premixture, and the propagation is a self-sustained process that can reach supersonic velocities. The propagation front leaves behind it a high pressure region inducing dynamic loading of the surrounding structure.

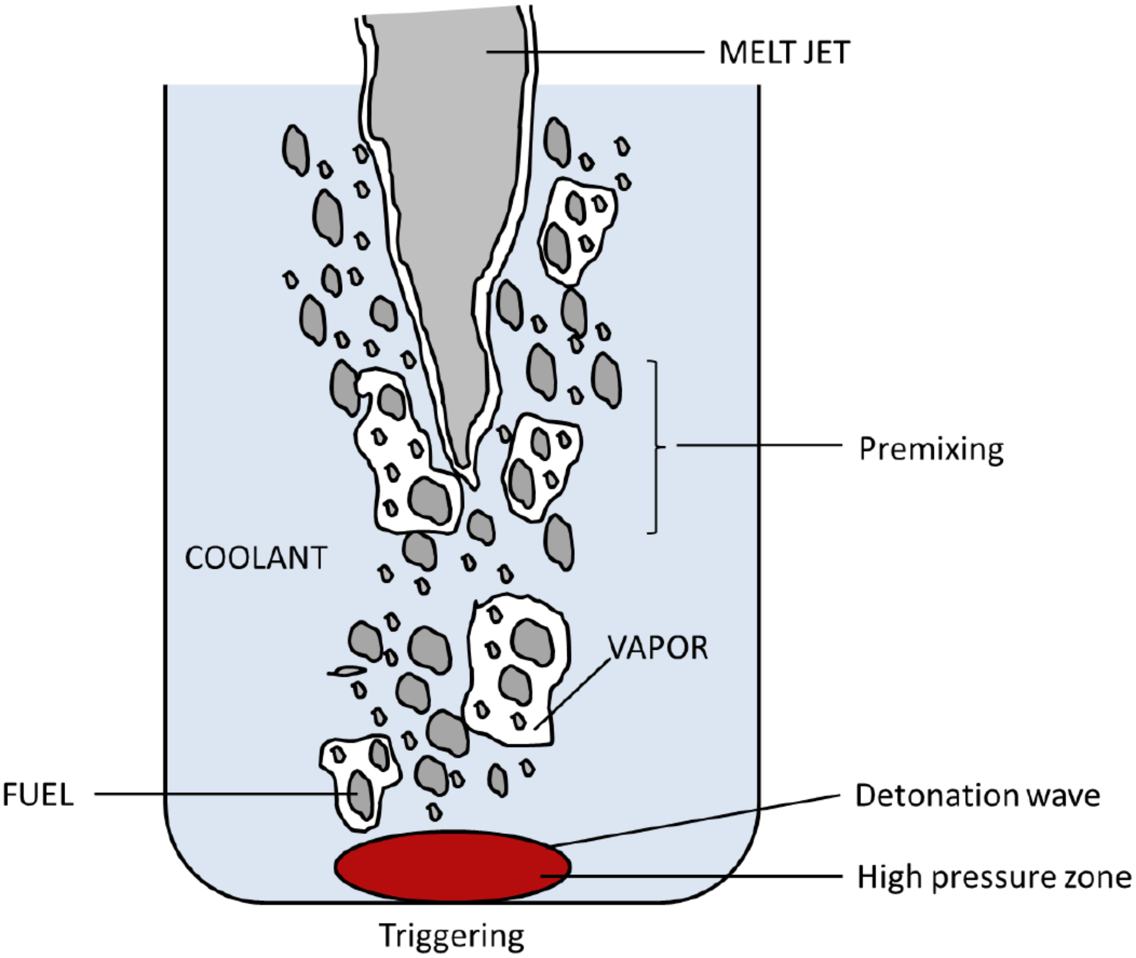

Figure 8: Conceptual picture of a steam explosion associated with melt pour.

4.1 Modelling

Steam explosions are very complex events with heat and mass transfer, and it is demanding to describe them accurately. The deterministic FCI codes describe all the relevant processes by using approximations, but still huge equation systems must be solved. Complex multidimensional fluid dynamics tools are used in order to overcome the lack of experimental data, and they give best-estimate evaluations. Some simplified tools are also available, but they contain even larger uncertainties [25].

All the explosion phases must be modelled to achieve as accurate results as possible. The premixing phase gives the initial state of the medium through which the explosion propagates and must therefore be calculated properly. The triggering time and position are chosen by the user in most codes, because of their stochastic nature. In the explosion phase, there are currently two approaches, namely micro-interaction model and non-equilibrium model. In the former, the mixture components are assumed in thermal equilibrium, whereas in the latter, part of the heat is used to produce vapour.

Figure 9 presents a simple diagram of how a steam explosion modelling process for level 2 PRA purposes could proceed and which tools could be used to support it. MELCOR or some similar tool can be used to obtain the amount of core material involved in the explosion and also values for some other explosion-relevant input parameters. MELCOR simulations are useful for the level 2 PRA modelling phase as well. In addition, results from level 1 PRA can prove useful in this phase. The analysis tool for FCI evaluation has probably the most crucial role, because it determines the actual threat in terms of loads posed to the structural integrity of the containment. How the information obtained from the previous steps is used in PRA analysis is ultimately up to the analyst and the scope and objective of the study. There exist codes for structural analysis as well, but they were not considered in this study.

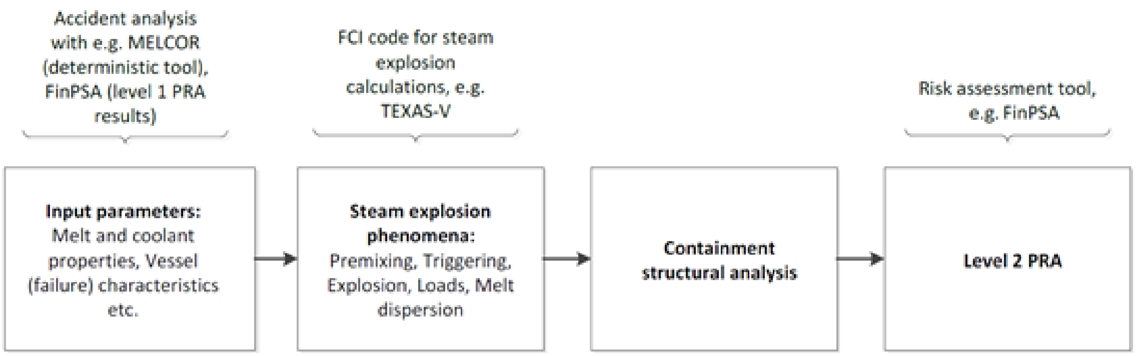

Figure 9: An example of phases of steam explosion analysis and its integration into level 2 PRA.

4.2 Deterministic analyses

Steam explosion case studies [3-5] were based on Olkiluoto 1&2 units, which are BWR type of reactors. First, information on accident progression was gathered in deterministic simulations using MELCOR [16]. This information included timings of events and initial conditions for the fuel coolant interaction phenomena. Second, more advance analyses were conducted with MC3D code [26], which is a multidimensional numerical tool devoted to analysis of FCI phenomena [4]. Third, ex-vessel coolability was studied more closely with help of MELCOR [5].

4.2.1 Timings and initial conditions

MELCOR simulations were conducted by using input file developed for Olkiluoto BWR units 1&2 at VTT [27]. The accident scenario begins with a loss of all AC power and the reactor is scrammed. In the beginning of accident progression, the reactor pressure is kept at 70 bars by safety relief valves. Then, the reactor can be depressurized by discharging steam into the wetwell by use of automatic depressurization system (ADS) and its relief valves (314 valves). This action is initiated by very low water level in the reactor. Depressurization allows the use of low-pressure core spray (system 323) to provide core cooling to avoid core uncovery that could eventually lead to a vessel melt-through. Also, a high-pressure melt ejection is even more undesired event than a low-pressure melt ejection because it is less predictable. After 30 minutes, the LDW is flooded from the wetwell to cool the ejected melt in case of a vessel breach to protect LDW penetrations for e.g. piping and to delay radioactive release. Flooding is initiated manually by operators who follow SAM-guides, but in the MELCOR model the flooding time is fixed at 30 minutes. The emergency core cooling system (ECCS), i.e. core spray 323 and high-pressure injection (system 327), is also implemented in the model so that the significance of recovery AC power can be examined. ECCS is dependent on AC power whereas 314 valves operate with batteries.

Six different cases were selected to find out how the availability of ADS and the recovery time of ECCS affect accident sequence progression. The analysis was mostly performed in a bounding sense, i.e. for example ADS either functioned or not, and for instance, the sensitivity of results to ADS valve capacity was not investigated. Delay in ECCS actuation was the main parameter varied – for both high and low RPV pressure scenarios. After evaluating this kind of extreme situations, expert judgment can be used to interpolate to less drastic scenarios and avoid performing too high a number of simulations.

The analysis cases are shown in Table 4. The recovery time and the availability of the ECCS and ADS were varied, and practically, these two parameters determine the others in the table, i.e. how the accident progresses. In other words, all other values come from the results of MELCOR analyses.

Table 4: MELCOR simulation results including information on the major events affecting the possible steam explosions.

	Case #					
	1	2	3	4	5	6
ECCS availability	Recovery at 4000s	Recovery at 4000s	Recovery at 18000s	Recovery at 19000s	No	No
Depressurisation through ADS [s]	1821	1821	-	-	1805	-
Core dry for the first time [s]	2510	2510	4650	4650	2510	4650
Zr oxidation starts [s]	2620	2620	3080	3080	2620	3080
Core support structures start to fail [s]	-	5678	7534	7534	5093	7534
Vessel breach (VB) [s]	-	17447	-	19021	13706	19018
Filtered venting (System 362) [s]	-	-	-	19078	-	19087
LDW water subcooling at VB [K]	-	65.53	-	95.84	73.61	95.84
LDW water partial pressure at VB [bar]	-	1.82	-	3.72	2.25	3.72
Melt ejected [ton]	-	159.7	-	-	183.3	185.5

4.2.2 Fuel coolant interaction

More detailed analyses on ex-vessel steam explosion loads posed to containment structure were conducted in [4] using MC3D code [26]. MC3D is a multidimensional numerical tool devoted to study multiphase and multi-constituent flows. It is developed especially to analyze FCI phenomena but it can calculate very different situations including direct containment heating and even debris coolability. For FCI applications, MC3D is practically divided into two separate parts, the first of which calculates the premixing phase whereas the latter proceeds from premixing to calculation of explosions themselves.

Only cases 2, 5 and 6 from Table 4 were analysed using MC3D because others do not end up with melt ejection into LDW and are therefore not interesting for steam explosion considerations. Each case was studied using two different fragmentation models and two different vessel failure modes (single large jet and multiple smaller jets). Explicit differences in FCI sense between the cases remained vague because explosion phase of case 6 with pressurized melt ejection could not be simulated and the other two cases produced quite similar and even a little inconsistent results. Same melt properties were used for all cases.

Throughout the analyses there were difficulties to trigger explosions. For a particular premixing calculation, there may have been only one or two if any possible points in time for initiation of the explosion. Time of trigger can affect explosion strength significantly, and thus, it is difficult and even dubious to compare cases that have very different triggering times. Because of inconsistent triggering times the analysis results were quite scattered, but otherwise the results were characterized by unexpectedly high pressure loads. The maximum pressure in the whole simulation domain was as much as ~300 MPa, pressures along the LDW side wall exceeded 30 MPa and the impulse maxima reached values over 300 kPa·s. However, most simulations produced remarkably weaker explosions.

Results showed generally quite large pressures and impulses in comparison to results from OECD’s SERENA program [28]. For multiple melt jets, the pressure loads were higher than for a single jet.

4.2.3 Ex-vessel debris coolability

If containment remains intact in melt ejection phase and debris bed forms on the LDW floor, the presence of a coolant is a necessary although not sufficient condition for avoiding containment threatening interactions between molten corium and concrete. Debris coolability issue in a flooded LDW was studied in [5] by conducting further MELCOR analyses. Again, analysis included only cases 2, 5 and 6 from Table 4.

Probability of coolability was estimated with load vs. capacity concept, i.e. probability distributions were assigned to debris bed heat flux (load variable) and dryout heat flux (DHF) (capacity variable). Heat flux was estimated by using equations found in literature (Equations (3)-(5) in [5]) and Monte Carlo method, i.e. parameter values used to calculate heat flux were sampled from probability distributions assigned to them. Dryout heat flux is the maximum heat flux that can be removed from the bed through its upper surface. Distribution for DHF was obtained principally from experimental results [29, 30], but uncertainty was incorporated by formulating normal probability distribution for DHF.

Two scenarios were considered: depressurized and pressurized melt ejection modes. Results suggested that pressurized case produces coolable debris with much higher probability (0.995) than depressurized case (0.910), mainly because of less threatening debris bed configuration and elevated ambient pressure. However, this conclusion should not be taken as a guideline for severe accident management (whether to depressurize primary system or not) because accident progression in its whole complexity necessitates many other things to be taken into account as well. Nonetheless, many new insights into problems involved in ex-vessel phenomenology of a severe accident were achieved.

4.3 Containment event tree model

A containment event tree for the station blackout scenario is presented in Figure 10. The plant damage state can be described as high or low pressure transient/melting depending on the result from the CET heading concerning reactor coolant system (RCS) depressurization. Core cooling systems are assumed unavailable until the possible AC power recovery. Different sections of the CET are gone through in detail in [3]. Containment failure modes are presented in Table 5.

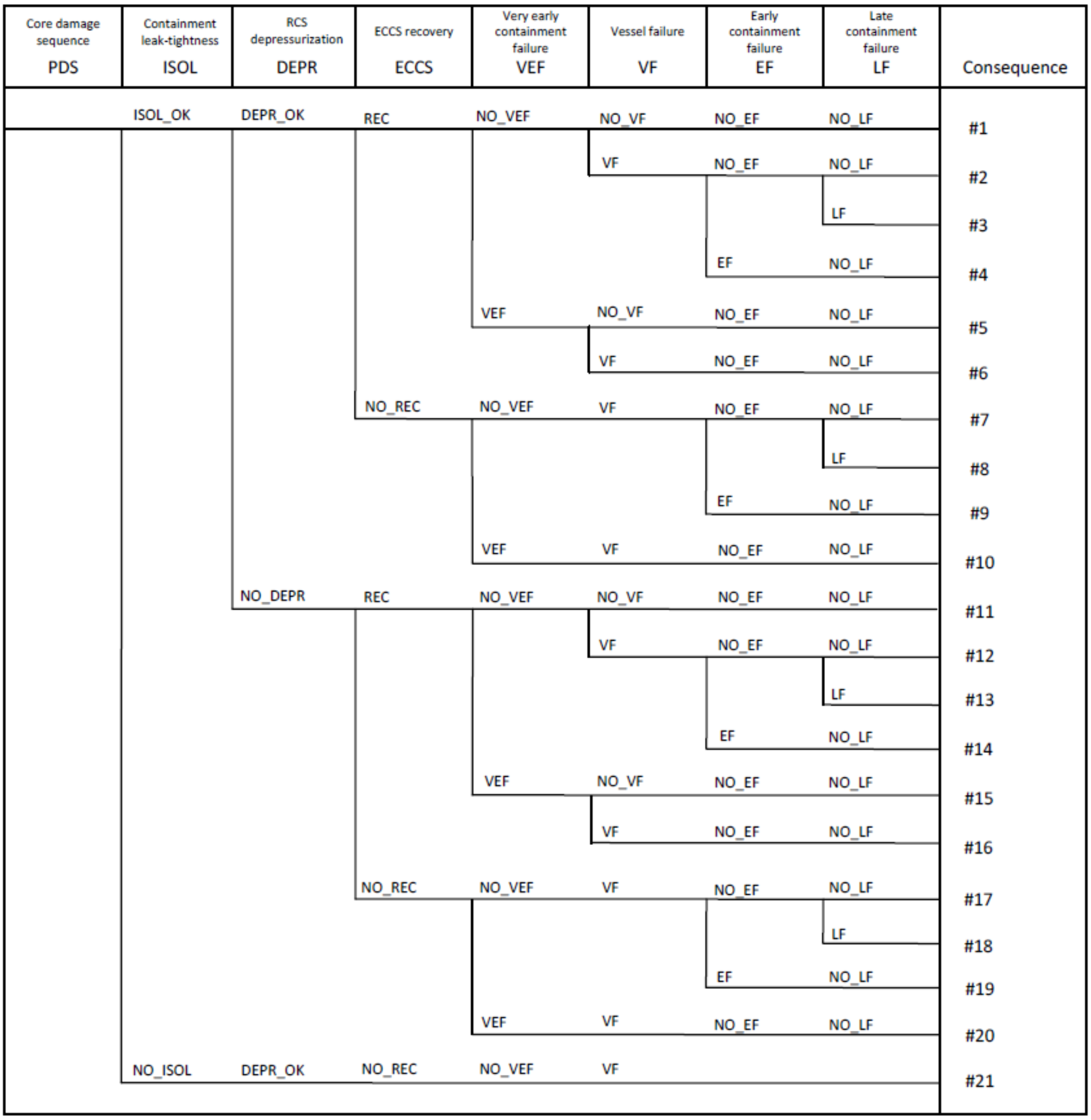

Figure 10: The containment event tree developed for the case study.

Table 5: Containment failure categories and the corresponding failure modes used in the CET model.

Release category	Containment failure/vent mode
No containment failure of filtered venting (OK)	-
Isolation failure (ISOL)	1. Containment not leak-tight (ISOL)
Very early containment failure (VEF)	1. Containment over-pressurization (COP) 2. Hydrogen deflagration/detonation (H2) 3. Alpha-mode failure (ALPHA)
Early containment failure (EF)	1. Ex-vessel steam explosion (STEAM) 2. Failure of containment penetrations (PENE)
Late containment failure (LF)	1. Non-coolable ex-vessel debris causes basemat melt-through (BASE)
Filtered venting (FV)	1. Very early venting (VEFV) 2. Early venting (EFV) 3. Late venting (LFV)

A containment over-pressurization is taken into account only if recriticality has been indicated in ECCS recovery section. Given reactor recriticality, a containment failure caused by over-pressurization is assumed to occur with probability 0.1. Hydrogen induced failure modes are possible only if containment is not inert, which is assumed to occur with probability 0.3. This value may be very conservative considering that non-inert containment occurs practically only during reactor start-up or shutdown.

The containment fails in a detonation/deflagration with probability 0.5. Even though many studies in the past have deduced alpha-mode containment failure to be unfeasible (see e.g. [31]), it is not considered entirely impossible in this study, and it is given a probability of 0.0001. The value, even though small, represents an upper bound and thus reflects the improbability of the event. Alpha-mode causes the failure of the RPV as well.

Early containment failure occurs at the time of vessel breach or shortly after it. The modelling of ex-vessel steam explosions – the main focus of the study – is presented in its own section 4.3.1.

Containment penetrations can be damaged if the core melt is not quenched in the LDW water pool. The flooding of the LDW is of utmost importance regarding possible damage to LDW penetrations. A lognormal base probability of 0.05 is given for penetration melt-trough, and a multiplicative factor of 2.0 is applied if flooding percentage is less than 50%. Without LDW flooding, the probability is set straight to 0.5. High-pressure melt ejections are assumed to contribute to this failure mode as well, and failure probability is doubled if RCS remains pressurized until vessel breach. High pressure melt-ejection could lead to violent ex-vessel phenomena that result in containment failure. However, it has to be pointed out that the relationship between core quenching and high-pressure melt ejection phenomena is quite vague. Therefore, the LDW penetration failure mode should be understood as a coupled treatment of these two distinct failure modes. Penetration failure is also assumed to lead to a fast LDW dryout with a doubled rate compared to the flooding rate. It has not been taken into account that structures protecting LDW penetrations could be damaged in a mild steam explosion that does not threaten containment integrity, thus making the penetrations much more vulnerable to melt attacks.

Only one failure mode is considered for late containment failure: a basemat melt-through caused by non-coolable ex-vessel debris. Prerequisite for basemat melt-through is that vessel breach has occurred and ex-vessel debris exists. Only fully coolable debris can prevent melt-through for sure, and debris bed is fully coolable in half of the cases when flooding is successful. In other cases, a 0.1 probability of basemat melt-through is assumed. This failure mode causes a dryout of flooded LDW similar to the dryout due to penetration failure in the EF-section.

4.3.1 Ex-vessel steam explosions

The weakest point in the LDW is typically the LDW door and the LDW strength refers to the strength of the door structure. In this study, a lognormal distribution with mean value of 50 kPa·s and error factor of 2.0 is used. In terms of scale and shape parameters μ and σ, the distribution is characterized by $\mu = 10.731$ and $\sigma = 0.421$. For explosion impulses, lognormal distributions are also used, and an explosion impulse in the CET model depends primarily on three things:

- LDW flooding
- Containment debris fraction
- RCS pressure

It is assumed that LDW flooding has to be halfway through at the time of vessel failure (melt ejection) so that a containment threatening pressure impulse could be considered plausible.

Containment debris fraction is set to equal core meltdown fraction if vessel breach occurs. If containment debris fraction exceeds 50%, melt amount engaged in FCI is considered large, while otherwise the interpretation is that there is only a little melt involved. The melt amount is assumed to have influence on both the mean value and the shape of the impulse distribution. High melt amount both shifts the distribution to the right and lengthens its tail, reflecting the possibility of really massive explosion impulses.

RCS depressurization is modelled to have an effect on the triggering of an explosion, and also on its magnitude. If primary circuit is pressurized when the vessel breaches, the triggering of an explosion is assumed to take place with certainty, whereas low-pressure melt ejection triggers explosive FCI phenomenon with probability 0.5. On the other hand, explosions associated with high-pressure melt ejection are estimated milder.

Table 6 presents the explosion impulse distributions. They are tabulated so that the effect of depressurization and amount of melt involved can easily be read. The most severe case is the one with low pressure and much melt. Figure 11 illustrates the load distributions and also shows the distribution used for LDW strength.

Table 6: Parameter values for log-normal pressure impulse distributions due to ex-vessel steam explosions in four different cases.

	Much melt ejected (case 1, late or no ECCS recovery)	Little melt ejected (case 2, early ECCS recovery)
RCS depressurized (case LP)	Mean = 30 kPa·s, ErrF = 3.0 μ = 10.086, σ = 0.668	Mean = 15 kPa·s, ErrF = 2.0 μ = 9.527, σ = 0.421
RCS not depressurized (case HP)	Mean = 20 kPa·s, ErrF = 3.0 μ = 9.680, σ = 0.668	Mean = 10 kPa·s, ErrF = 2.0 μ = 9.122, σ = 0.421

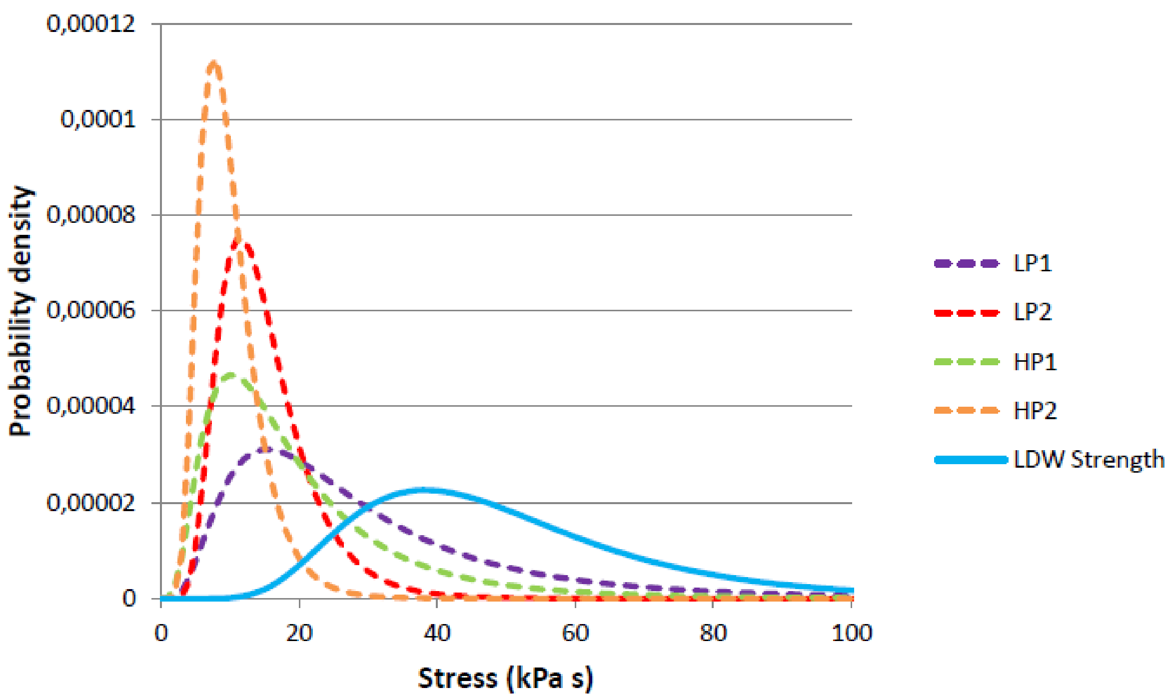

Figure 11: Distributions used to determine whether LDW fails due to pressure impulse caused by ex-vessel steam explosion.

Table 7 presents the conditional probabilities of impulse load exceeding LDW strength. Being conditional probabilities, the values in Table 7 do not take into account that for high-pressure cases the explosion is always assumed to trigger, whereas for low-pressure sequences there is a 50 % chance of trigger. Thus the difference between high- and low-pressure cases is really not as big as it appears at first sight. Melt amount engaged in FCI phenomena plays a more significant role.

Table 7: Conditional probability of explosion impulse exceeding strength of LDW walls given vessel failure, explosion trigger and enough water in LDW.

	Much melt ejected (case 1, late or no ECCS recovery)	Little melt ejected (case 2, early ECCS recovery)
RCS depressurized (case LP)	0.207	0.021
RCS not depressurized (case HP)	0.091	0.003

4.3.2 Source term model

The source term model is based on the model described in [32], which in turn has been influenced by the so called XSOR method [33]. The model here concerns only three radionuclide groups: noble gases (source term variable S_Xe), cesium (S_Cs) and rutherium (S_Ru). The three groups were chosen so that the behaviour of radionuclides belonging to different groups deviates significantly from each another. Iodine releases are also often considered in release models but not here. Anyway, the behaviour of iodine releases would be closer to the behaviour of Cs than the behaviour of Xe. Source term modelling contains considerable uncertainties, especially with regard to fission product transport. Therefore, the model aims to give order of magnitude estimates instead of trying to predict accurate values.

Fission product model applies for atmospheric, i.e. gaseous releases. All releases, except noble gases, are assumed to be in aerosol form, which means that they follow the same diffusion, deposition and decontamination mechanisms and rules throughout the simulations. For noble gases, there is no such release decreasing phenomenon, and all noble gases released from the core are released from the containment as well. The basic idea of the source term model is presented in Figure 12. The meanings of parameter/variable names can be checked from Table 8. Almost all parameters used in the source term model are treated probabilistically as distributions because of the uncertainties involved. The starting point for the calculation of releases is the core release. After the core release, there are three different release mechanisms: early and late release from reactor coolant system (RCS), and ex-vessel debris release.

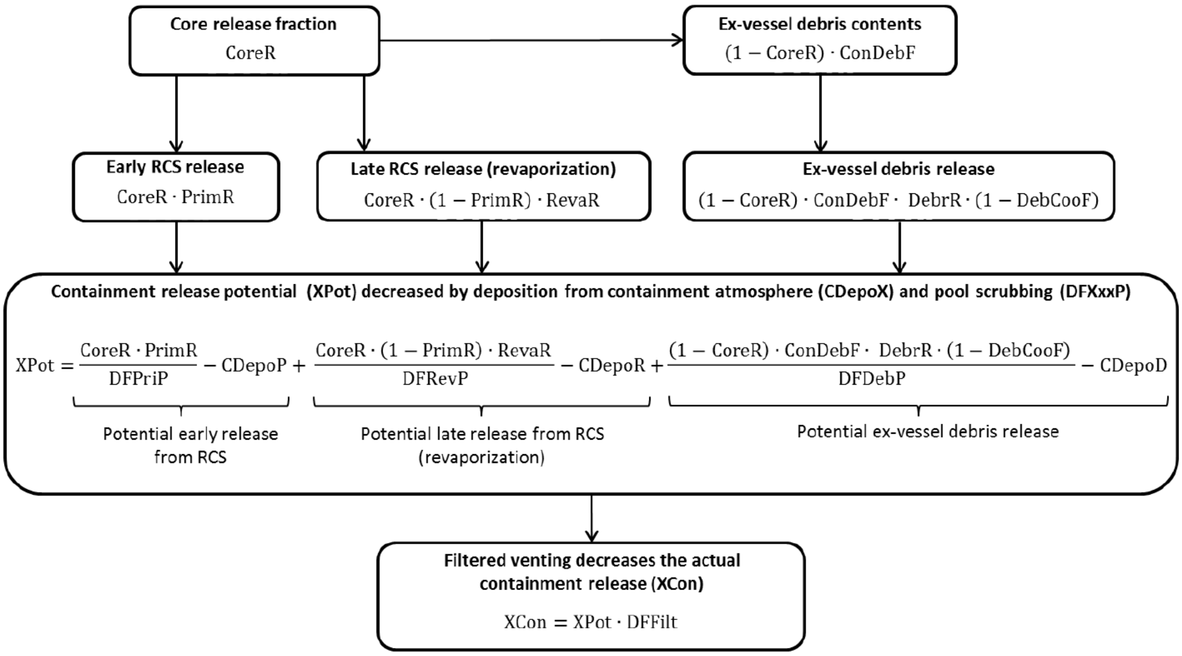

Figure 12: Source term model that calculates releases out of the containment.

Table 8: The meanings of the model's variables.

Variable	Meaning
CoreR	Core release fraction
ConDebF	Fraction of potential ex-vessel debris actually formed
PrimR	Primary release fraction (Fission products from core penetrating RCS)
RevaR	Release fraction due to revaporization
DebrR	Debris release fraction
DebCooF	Debris coolability fraction
XPot	Potential release from the containment
DFPriP	Pool decontamination factor for primary release
CDepoP	Deposition from containment atmosphere for primary releases
DFRevP	Pool decontamination factor for revaporization release
CDepoR	Deposition from containment atmosphere for revaporization releases
DFDebP	Pool decontamination factor for debris release
CDepoD	Deposition from containment atmosphere for debris releases
XCon	Actual release from the containment
DFFilt	Decontamination factor for filters

Core release corresponds to severe core degradation and it is sampled for each radionuclide group from a distribution assigned to it. The core release fraction remains the same throughout the simulation. The actual release mechanisms are dependent on the core release fraction.

Early RCS release refers to fission products that are first released from the core and then penetrate the RCS. Release fraction from the primary circuit is sampled from a distribution which depends on whether RCS is pressurized or not. The expected value for early RCS releases is higher for low-pressure case because the residence time of fission products in the RCS is longer. The time interval for early releases is from the start of core melting to the end of melting. The melting can end due to the recovery of ECCS or core being fully molten.

Late RCS release is formed from the fission products that do not penetrate the RCS but are deposited on the RCS structures. Late in-vessel release occurs when these deposits are revaporized. The revaporization fraction for each radionuclide group also comes from a distribution. If the ECCS is recovered in time, the revaporization fraction is set to zero. The length of the release interval for late revaporization release is assumed to be normally distributed between 1 and 9 hours, and the release starts when the RPV breaches.

Ex-vessel debris release consists of core contents that are not involved in releases from the RCS, and it can occur only if the vessel fails. Source term variable specific debris release fraction is sampled from a distribution assigned to it, and only the non-coolable part of debris is assumed to contribute to the release. A debris coolability fraction is used, and it has a non-zero value if there is water in the LDW. The length of ex-vessel debris release time interval is assumed normally distributed between 1 and 5 hours (truncated distribution). The release begins when the vessel breaches. If the flooded LDW dries out because of a containment failure, a dryout delay is taken into account in the start time.

Figure 13 summarises release intervals for all release mechanisms and places them along major events that occur during the accident progression.

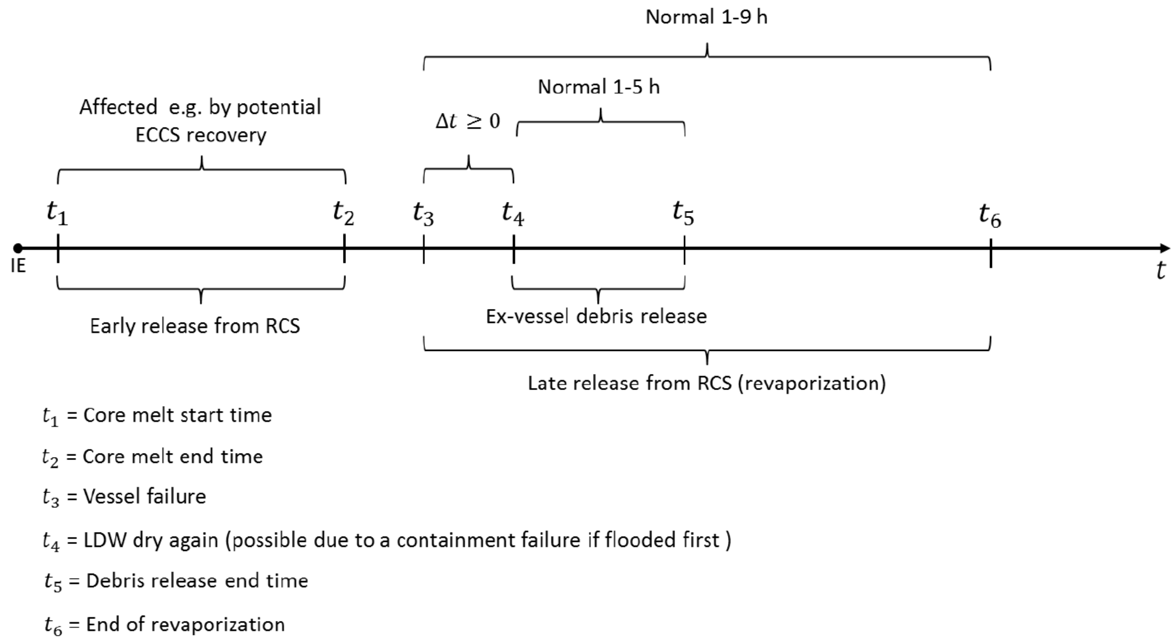

Figure 13: The timings of the different release mechanisms.

Pool scrubbing occurs when fission product releases are driven through a water pool, and then, the release is decontaminated before it enters the containment atmosphere. The effect of pool scrubbing is implemented by using decontamination factors, which are defined as the ratio of radionuclide flow into pool and radionuclide penetration through the pool. Each release mechanism and radionuclide group pair has its own decontamination factor. For late RCS revaporization release, the factors are only nominal and do not have big effect, because any release is assumed to flow out of the RCS directly into the containment atmosphere practically without pool decontamination. For debris release, pool scrubbing occurs only if the LDW is flooded and does not dry out completely when the vessel breaches. A decontamination factor can then be decreased by a gradual LDW dryout. For noble gases, there is no pool scrubbing related decontamination at all.

When the fission products are in the containment atmosphere, they can get gradually deposited on containment structures and in water pools. The approach taken to model **containment deposition** is based on defining a deposition rate for the fission products. The deposition rate is affected by the containment area available for deposition which in turn depends on LDW flooding. Also, the free gaseous volume in the containment influences the process. From the release rate, the actual amount of deposited fission products is derived using release time intervals defined for each release mechanism. The effect of containment spray is included in the containment deposition consideration, and the value assigned to the spray impact is also treated probabilistically. The spray is assumed to operate if the ECCS recovery is successful. Containment deposition is calculated similarly using the same parameter values for all releases, but some parameters can have different values because release timings are not the same.

Release from the containment is calculated as a sum of releases associated with all different release mechanisms, as suggested by Figure 12, taking into account containment deposition and pool scrubbing decontamination. The sum represents the containment release potential. The actual release is calculated by applying decontamination factors related to filtered containment venting, which can occur if there is no major containment failure. Filters do not have any effect on noble gases.

Source term calculations are performed at each end point of the CET at the end of the analysis on each simulation run. The release time interval for each release mechanism is divided into ten, and a discrete point release is calculated at the centre of each tenth and added to total release. The accuracy can be increased by dividing the release interval into larger number of subintervals, but ten is considered sufficient.

After all the simulations (10 000 runs) have been performed, the results are binned into release categories. The categorization can be seen in Table 5 with the exception that early containment failure modes have separate release categories. Forming a specific release category for ex-vessel steam explosions alone is generally not reasonable or recommended, but in this study, steam explosions needed to be separated to facilitate the analysis of results.

4.3.3 Results

Table 9 presents the results including the conditional probability of release and the mean fractions of release for each release category. Filtered venting (FV) and OK bins are dominant with a combined share of 77% of the probability. Very early failure (VEF) is also quite probable with a share of 19%. Early containment failure caused by a steam explosion (EF_STEAM) constituted 2.6% of the mean probability. Hence, the contribution of steam explosions to early containment failure is significant. The results are discussed more extensively in [3].

Table 9: Release category results for the steam explosion case study.

Bin	Conditional probability	S_Xe mean	S_Cs mean	S_Ru mean
OK	0.23	8.4 E-11	-	-
ISOL	0.01	0.87	0.20	0.006
VEF	0.19	0.78	0.13	0.0043
EF_PENE	0.0014	0.83	0.20	0.0072
EF_STEAM	0.026	0.86	0.28	0.0089
LF	0.01	0.99	0.41	0.14
FV	0.54	0.78	7.8 E-4	2.3 E-5
Weighted average	1	0.66	0.061	0.0019

By reviewing uncertainties in results, it can be seen that probability distributions are not realistic. For example, Figure 14 presents the probability distribution of OK category (this figure was taken from recalculated results, not the original results from [3]). In this distribution, e.g. 50th percentile is circa 1E-7, while OK category should clearly have high probability in reality. The reason for this is the way the probabilities have been treated in the model. On most of the simulation runs, one accident sequence has probability close to 1, and other probabilities are close to 0. On each simulation run, a sequence must have a realistic probability or probability 0 so that the probability distributions can be calculated correctly. In this example case, no sequence can realistically have a probability that is very close to 0 or 1. The approach that is used here produces only the mean values of probabilities correctly. This conclusion did not appear in the original report [3].

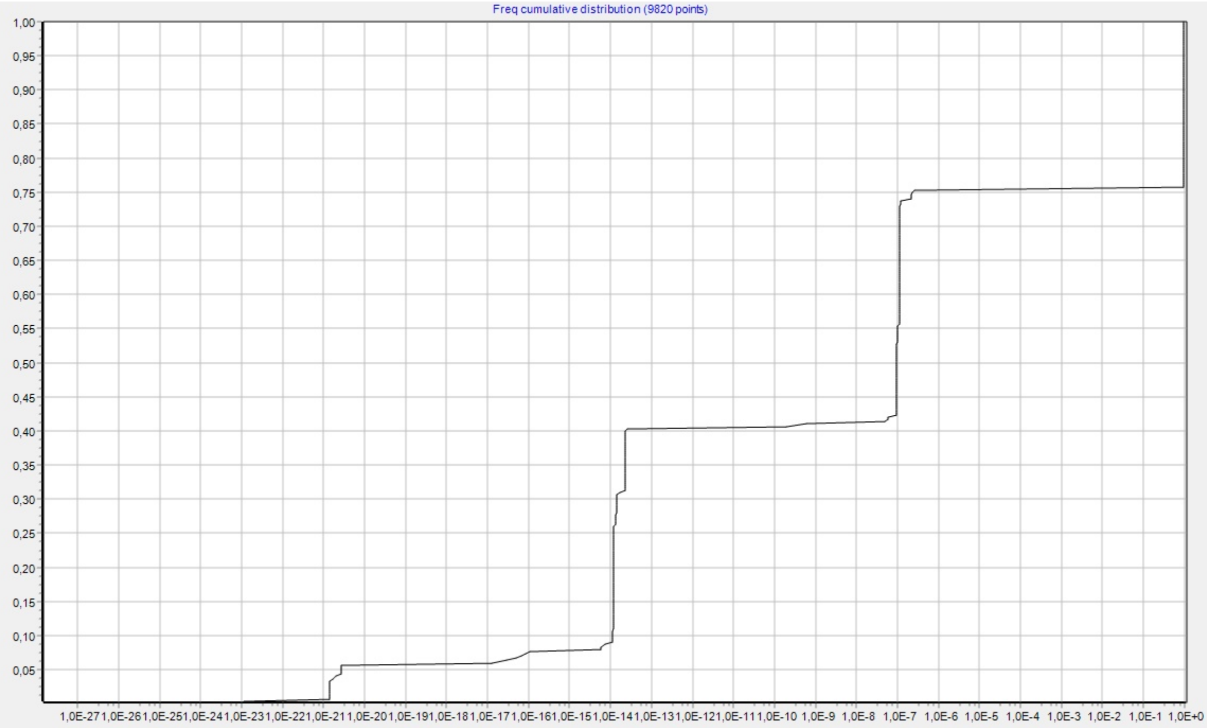

Figure 14: The cumulative distribution of the conditional probability of OK category.

5. Conclusions

The applicability of IDPSA methodology to level 2 PRA was successfully demonstrated in case studies concerning hydrogen management [2] and steam explosions [3]. The reliability analysis of a passive reactor cooling system (PCCS) was examined in [1]. In addition, detailed supporting analyses on ex-vessel steam explosions [4] and debris coolability [5] were performed.

The reliability analysis of passive nuclear safety systems was studied in [1, 2]. Both physical components and natural phenomena were considered, and uncertainties were taken into account in the analyses.

The reliability of a passive autocatalytic recombiner (PAR) system was estimated using common cause failure models. Accident progression simulations were performed using MELCOR to assess whether the PAR system is able to prevent combustible circumstances in the containment atmosphere. The reliability estimate of the PAR system was combined with other case-significant information in a containment event tree model constructed for Loviisa PWR design. The results showed the importance of the hydrogen management. [2]

Failure modes and effects analysis (FMEA) and fault tree analysis were used to conduct reliability assessment for a PCCS. The PCCS was also investigated with help of MELCOR simulations. In the case study, the PCCS was added to Olkiluoto BWR design. The simulations indicated that the PCCS could not provide adequate core cooling for the Olkiluoto units. [1]

Steam explosions were modelled in a level 2 PRA model utilising IDPSA methodology in [3]. MELCOR simulations were performed to obtain knowledge on physical parameters affecting steam explosions. The analysis showed clear differences between high and low pressure cases with regard to important steam explosion parameters such as ambient pressure and LDW water pool subcooling. Particularly, useful knowledge was acquired about the timing of events and about the time available for ECCS recovery. Steam explosions were modelled in the CET by using load vs. strength approach. Depending on accident progression, four different pressure load impulse distributions were used for sampling in CET simulations, and whenever load exceeded LDW strength, a containment failure was induced.

By reviewing the results, it was noticed that the way the probabilities are treated in the CET model is problematic from the uncertainty analysis point of view. The resulting probability distributions are not calculated correctly. Only the mean values of probabilities can be considered meaningful. The model could be developed further so that the probabilistic assessment would be performed in correct manner and would produce "correct" probability distributions.

Ex-vessel steam explosions were examined more closely using deterministic MC3D code in [4]. The resulting pressure loads were larger than expected and presented in literature. However, the level of detail of modelling could be improved, and throughout the analyses, there were difficulties to trigger explosions. Time of trigger affects explosion strength significantly, and thus, it is difficult and even dubious to compare cases that have very different triggering times. These problems degrade the applicability of the results in level 2 PRA.

Coolability of ex-vessel debris bed and its treatment in safety analyses in a risk-informed manner were studied in [5]. MELCOR simulations were performed and results from literature were utilised. Probability of coolability was estimated with load vs. capacity concept, i.e. probability distributions were assigned to debris bed heat flux (load variable) and dryout heat flux (capacity variable). Two scenarios were considered: depressurized and pressurized melt ejection modes. Results indicated that pressurized case produces coolable debris with much higher probability, mainly because of less threatening debris bed configuration and elevated ambient pressure. However, this conclusion should not be taken as a guideline for severe accident management (whether to depressurize primary system or not) because accident progression as a whole is a complex entity with many factors to be taken into account.

Even if simplifications and compromises were made in modelling work, the studies demonstrated successfully how to use IDPSA methodology in level 2 PRA. General knowledge on selected severe accident phenomena and modelling capabilities were also obtained. The focus was on particular safety systems and accident phenomena, but the suggested methods should be applicable for any safety device.

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