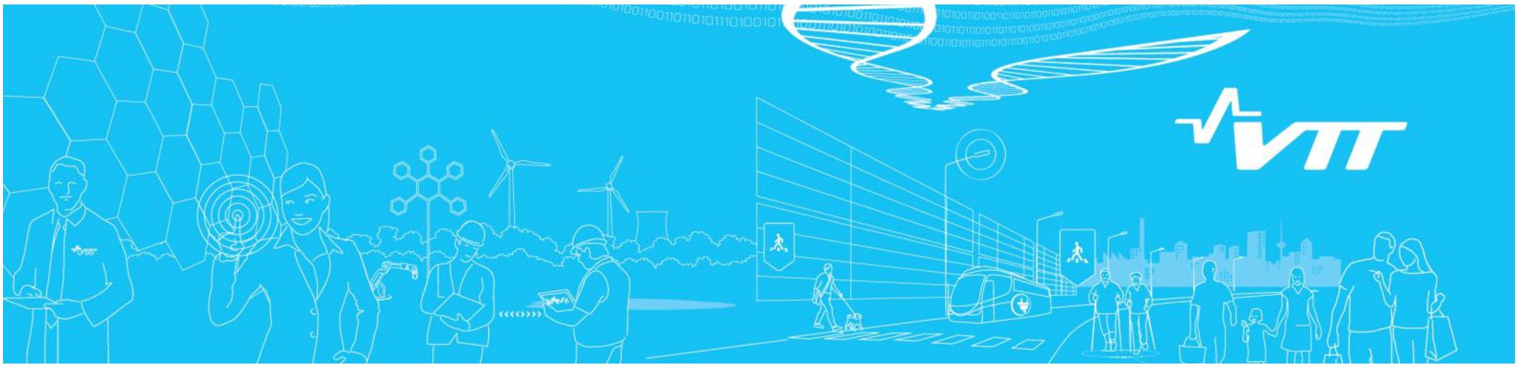

RESEARCH REPORT

VTT-R-00354-17

Level 2 PRA studies – Source term characteristics and hydrogen explosions

Authors: Tero Tyrväinen, Ilkka Karanta

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Summary

This document studies probabilistic modelling of severe nuclear power plant accidents. Factors that affect the height and temperature of radioactive releases are identified, and hydrogen explosions in boiling water reactor (BWR) nuclear power plant are studied. A containment event tree model is developed for a BWR plant based on a model developed in a previous study. Uncertainty analysis is implemented for release probabilities, and release heights and temperatures are included in the model.

The height and temperature of radioactive releases have typically not been included in level 2 probabilistic risk analyses (PRA), even though they are needed as inputs for level 3 PRA. Release height is usually the height of the location where the reactor building leaks (which depends on the containment failure mode), or the height of the chimney if the release is controlled. In an uncontrolled accident case, both the containment failure location and the flow path of radionuclides in the reactor building need to be analysed to determine the release height. The temperature of release from containment is in most cases close to 100°C, but the temperature of radionuclides can potentially change during their migration in the reactor building. Higher temperatures can be caused by fires and explosions. There are computer codes that can be used to analyse radionuclide flows in reactor building and determine the release heights and temperatures.

Hydrogen explosions can occur inside BWR containment if the containment is not inert. Therefore, the risk of hydrogen explosions comes mainly from reactor start-up, shut-down and refuelling. Hydrogen explosions outside containment have usually not been modelled in PRA, even though they can affect the releases significantly.

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Written by	Reviewed by	Accepted by
Tero Tyrväinen Research Scientist	Kim Björkman Research Scientist	Eila Lehmus Research Team Leader

VTT's contact address

VTT, PL 1000, 02044 VTT

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Contents

1. Introduction

Level 2 probabilistic risk analysis (PRA) studies how a nuclear power plant accident progresses after core damage and how frequent and large radioactive releases are [1]. Severe accident phenomena, e.g. hydrogen explosions, and timings of events, such as cooling system recovery, play an important role in such analyses. Information on severe accident progression provided by deterministic analyses is crucial to the construction of proper level 2 PRA. Integrated deterministic and probabilistic safety analysis (IDPSA) aims to bring the two types of analysis closer and improve their co-operation.

The height and temperature of radioactive releases have typically not been included in level 2 probabilistic risk analyses (PRA), even though they are needed as inputs for level 3 PRA [2]. This report identifies which factors affect the height and the temperature.

Hydrogen explosions can occur in nuclear power plant containment if heated zirconium reacts with steam producing significant amount of hydrogen during severe accidents. These explosions have the potential to be powerful enough to break the containment and cause a large early release. Hydrogen explosions in a pressurized water reactor plant were studied in [3]. Hydrogen explosions in boiling water reactor (BWR) plants have been studied less because inerting of containment - filling the containment with a non-flammable gas (usually nitrogen) - prevents explosions with certainty. Inerting of containment is often used in BWR’s, for example in the Olkiluoto 1 and 2 reactors for which the model presented in this report was built. However, an accident can also occur when the containment is not inert, e.g. during start-up or shut-down. Therefore, it is relevant to consider hydrogen explosions in BWR plants too.

This report continues the development of a level 2 containment event tree model for a BWR plant started in [4]. In [4], the focus of the research was on steam explosion modelling, and other phenomena were modelled in a very simple manner. In this report, uncertainty analysis is implemented for release probabilities. In addition, release heights and temperatures are included in the model.

2. Containment performance and failure modes

The purpose of a containment in a nuclear power plant is to isolate radioisotopes from the public (and also plant personnel and rescue workers) in case of a severe accident. Containment failure is necessary for an uncontrolled release. In PRA, it is also assumed that containment failure is sufficient for a release, although there are usually other structures (such as the outer walls) in a typical nuclear power plant unit that act as partial release barriers and may to a great extent affect the timing, radionuclide content and energy of the release to the atmosphere.

Uncertainty and variability are associated with many variables describing a structure’s performance; this can be accounted for by the use of probability theory and structural reliability theory. As PRA is explicitly concerned with uncertainty and variability, containment performance is best analysed in its context using the methods of structural reliability analysis [5].

A containment may fail in different ways, called containment failure modes. The failure mode greatly influences release height and energy. Some general factors that affect what failure modes a containment might have and how vulnerable the containment is to that failure mode are [6]:

- General configuration

- Construction materials. For concrete, information on rebar, stiffeners, aggregate. For steel, type(s) and tension.
- The location and size of penetrations, including electrical penetrations and major openings
- Major discontinuities, such as the transition from the spherical top head to the cylindrical shell in many PWR's, and from the cylindrical cell to the basemat.
- Weld locations
- Layout and anchorage of the reinforcement
- Liner walls and anchoring
- Interactions with surrounding structures at large deformations

Failure modes - or the vulnerabilities of individual containment - and their significance vary widely due to different designs. However, some generic containment failure modes can be identified that apply to a large variety of designs. Some of those are listed in Table 1.

Table 1. Some generic containment failure modes for BWRs and PWRs.

Failure mode	Description	Location	Causes and influencing factors	Consequences	Notes
Isolation failure or bypass	Containment leaks through a penetration, door, manhole or such	Depends on the geometry of the containment. Usually close to the bottom.	Explosions, static overpressure, hot temperature	Radionuclides enter the reactor building	The height of the release depends on which penetration is lost. Usually the penetrations are close to the bottom of the containment.
Over-pressure	Over-pressure develops slowly in the containment, until a part of the containment yields	Most probably a penetration, door, manhole or such.	Boiling of coolant water	Radionuclides enter the reactor building	
Corium-concrete interaction	Hot corium melts its way through the bottom of the	Basemat	Insufficient cooling of corium	Radionuclides, mixed with water, enter the ground below	Occurs at late phase of accident

	containment			the plant.	
Containment rupture	Containment fails due to a pressure impulse from an explosion or rapid fire	Anywhere except the basemat, most probably through a door or such	Steam explosions, hydrogen explosions	Radionuclides enter the reactor building	Failure of containment structure is unlikely; it is more likely that a penetration, door or such fails
Steam generator tube rupture	The tube from the containment to the steam generator is broken, volatile radionuclides may enter the tube (PWR)	Steam generator tube, usually close to the top of the containment.	Heat, pressure	Radionuclides may enter the turbine hall if the pipeline is not closed.	

3. Release height

The height of a radioactive release affects the spreading of the radioactive substances in the atmosphere which in turn affects the consequences of a nuclear power plant accident. Therefore, the height of the release is important information for atmospheric dispersion analyses and level 3 PRA [2]. Release height can be defined either based on the release from containment or release from reactor building. Before entering open environment, radionuclides are released from the containment to the reactor building. Even though PRA and safety analyses focus mostly on the events occurring inside the containment, the reactor building can affect significantly to the migration path of the radionuclides. This is a challenge for the determination of the release height.

To estimate release height accurately, some fluid dynamics analyses would be needed for radionuclides considering the architecture and conditions of the reactor building. Radionuclide flows in reactor building can be modelled using some deterministic computer code, such as OpenFOAM [7], FDS [8], MELCOR [9] or RADTRAD-NAI [10]. The fluid dynamics model could likely be quite simple because only rough estimates for the height are needed and no complex fluid phenomena are relevant for this problem. It also needs to be noticed that radionuclides can escape only from known holes in the reactor building or holes that are created by severe accident phenomena, such as hydrogen explosions. Possibly, in some cases, the correct hole can be identified based on reactor building architecture without any calculations.

Roughly speaking, there are three different types of release cases with regard to the release height:

- The release height is the height of the location where the reactor building leaks after containment failure.

- The release height is the height of the chimney if filtered venting is performed.
- An explosion throws the releases in the air above/surrounding the containment and reactor building.

Literature search gives very little about the release heights directly. There are some estimates for the heights of the releases of Fukushima Daiichi accident [11]. In containment leakage cases, the height was 20 meters. In the case of filtered venting, the height was 120 meters. In the two hydrogen explosion cases that occurred outside the containment, the maximum heights were 100 and 300 meters, and the centre heights were 50 and 150 meters. It is also given that the releases spread up to 100 meters away from the plant in different horizontal directions due to explosions. Reference [12] gives some slightly different numbers.

In most cases, the location of the containment failure is the basis for the analysis of the release height even though the reactor building has to also be taken into account. Structural reliability analysis [5] can be used to determine the location where the containment fails. Usually, penetrations and doors fail more likely than the concrete wall. PRA containment failure modes for which some structural analysis could be performed include

- Isolation failure, containment not leak-tight
- Containment over-pressurisation
- Containment rupture (e.g. due to hydrogen explosion or steam explosion)

The location of an explosion affects the location of the containment failure. Hydrogen explosions occur most likely in the upper part of the containment, and therefore, some weak point in the upper part of the containment would most likely be broken. Ex-vessel steam explosions occur in the lower part of the containment, whereas in-vessel steam explosions occur in the lower part of the pressure vessel, which is typically somewhere in the middle part of the containment. These locations, of course, depend on the reactor type and plant design. Previous statements are not necessarily applicable to all the nuclear power plants in the world. BWR plant of Olkiluoto is used as the main reference in this report.

For explosions, the determination of the release height is not straightforward. The release height can depend on the volume of the explosion. Explosion cases may require their own release height analyses using some proper deterministic explosion model. It is also noteworthy that an explosion can spread the release to a wider area in both horizontal and vertical directions, which could require special handling in atmospheric dispersion and level 3 PRA analyses, as well as in level 2 PRA model. However, explosions that occur inside the containment would likely not throw the radionuclides as far as hydrogen explosions outside the containment did at Fukushima.

For some containment failure modes, such as failure of containment penetration or basemat melt-through, the location of the containment failure is clear. If there are penetrations at different heights, a probability distribution covering different penetrations can be used in PRA. In these cases, the final height of the release from the reactor building may however be different than the height of the release from the containment.

It is difficult to determine accurate release heights for all accidents, because there are different release paths, and instead of single height, the release can be distributed at different heights in continuous or discontinuous manner. But for PRA, simplifications are justified because the release height has small importance with regard to level 3 results compared to the amounts of radionuclides. Mainly the magnitude of the release height is important.

4. Release temperature

Temperatures of releases from containment are in most cases close to 100°C because radioactive substances are usually released among steam. It is possible to calculate accurate temperatures using deterministic severe accident analysis software such as MELCOR [9].

The temperature of radionuclides can change during their migration through reactor building to environment. Reactor building structures can cool down radionuclides if they are in contact for a sufficient time. If radionuclides leak from a small crack, the temperature of the radionuclides will be the same as the temperature of the wall because the flow rate is so slow that the radionuclides have enough time to cool down. On the other hand, if there is a fire in the reactor building, e.g. due to hydrogen explosion, the temperature of the radionuclides can rise. The release temperature could be calculated, taking fire into account, using some deterministic fluid dynamics software, such as OpenFOAM [7] or FDS [8]. It would be possible to determine both the release height and temperature in the same analysis.

An explosion can be a special case with regard to release temperature too. During an explosion, the temperature is very high, but the release cools down fast. Some special explosion analysis would be required to determine the temperature of a radioactive release after an explosion.

5. Hydrogen explosions

Hydrogen produced inside the containment during severe accident can lead to an explosion that jeopardizes the integrity of the containment. Potential hydrogen sources are metallic materials inside and outside the vessel. Significant amount of hydrogen can be produced when hot zirconium reacts with steam. In addition to the hydrogen concentration, hydrogen combustions can occur only with particular steam and air concentrations. Hydrogen combustion can be mild deflagration, detonation or something in between. Explosions can be prevented using pre-inertization, igniter or passive autocatalytic recombiners. Igniter can burn off hydrogen before an explosive concentration is reached. Passive autocatalytic recombiners reduce hydrogen concentration in the atmosphere by spontaneous reactions.

Hydrogen explosions in pressurized water reactor plant were studied in [3]. Deterministic MELCOR [9] analyses were performed to find out if atmospheric conditions of hydrogen explosion are met in different scenarios. Particularly, the capability of passive autocatalytic recombiners to prevent explosions was studied. Probability of containment failure due to hydrogen explosion was calculated based on deterministic analyses and reliability analysis of passive autocatalytic recombiners. The probability was highly dependent on whether the containment was reflooded in critical time window.

Hydrogen management in BWR plants is significantly different from pressurized water reactor plants. Typically in BWR plants, the containment is inert during operation which prevents the hydrogen explosions with certainty. However, the containment is not inert during start-up and shut-down, and accidents occurring at those times can lead to hydrogen explosions. Also, it is possible that the inerting system fails.

Hydrogen explosions are typically modelled in level 2 PRAs of BWRs in very simple ways. For example, in the model presented in the next section, probability 0.3 is given for containment not being inert, and if the containment is inert, hydrogen explosion breaks the containment with probability 0.5. These probabilities have been assumed to be conservative. The model could be made more accurate by modelling the dependence of the hydrogen explosion probability on different events, such as cooling system recovery and depressurisation. In addition, it is possible to include physical variables, such as hydrogen concentration, in the model, and that way model physical dependencies explicitly. On the other hand, it is good practise to avoid unnecessary complexity in PRA models, and therefore, complex modelling should be well justified.

In practice, there are three probabilities that need to be determined for a given accident scenario:

1. the probability that the containment is not inert
2. the probability that an explosion occurs if the containment is not inert
3. the probability that the containment is broken if an explosion occurs.

Often, two latter probabilities are merged into one, like in the example model of this report. PRA model could be improved by determining these probabilities more accurately because the current probability estimates are very rough.

5.1 Containment inerting

The probability that the containment is not inert during an accident is the sum of the following probabilities of mutually exclusive events:

- the probability that reactor start-up is going on
- the probability that reactor shut-down is going on
- the probability that the containment is not inert due to the failure of the inerting system during operation
- the probability that the reactor is being refuelled.

The probabilities of start-up, shut-down and refuelling can be taken from level 1 results. Significant portion of transient and reactor over-pressurisation accidents occur during these phases. The probabilities must be calculated separately for each plant damage state.

The probability that the inerting system is failed during accident is likely small compared to start-up and shut-down accident probabilities However, to assess the significance of inerting system failure, reliability analysis should be performed for the system. It is also important to determine how long the containment is non-inert if the system fails.

It seems that inerting system failure and its consequences have not received much attention in research literature, and there are some potentially important research questions to which more detailed studies could shed light on, such as:

- If the inerting is incomplete during accident, what is the sufficient level of inerting to prevent an accident?
- How does the probability of hydrogen explosion behave as the function of the level of inerting? It seems that hydrogen flammability has been studied mainly as a function of oxygen and steam content of the containment atmosphere, and the proportion of inerting gas(es) have not been taken into account.
- How long does the inerting take? This information is obviously available to NPP operators. A useful piece of information would be the proportion of oxygen (or more generally, non-inert gases) in the containment atmosphere, tabulated as a function of time from the start of inerting.

- How long does the deinerting take? A reasonable guess is that it takes significantly shorter time than inerting, but again the proportion of oxygen tabulated as a function of time would be useful.

A report from 1980 [13] states that the containment is inerted in 24 hours after start-up and deinerted 24 hours before shutdown for the reference plant, but today's plants might have different time frames.

Inerting could possibly be analysed using MELCOR [9] or based on measurement data of a BWR plant.

It is also worth considering whether the probability that the reactor start-up or shut-down is on is exactly the probability from level 1 PRA. It is possible that a small portion of accidents counted as normal operation accidents occur before inerting is completed after start-up or after the inerting has been removed before shut-down.

5.2 Analysis of explosions

The probability that an explosion occurs if the containment is not inert, can be analysed based on deterministic analyses in the same way as in [3], i.e. determining hydrogen and steam volumes in different accident scenarios. Deterministic analyses could be performed using MELCOR. However, different models are needed for start-up, shut-down and refuelling than for normal operation. VTT does not have such models at the moment, so they should be developed first.

The conditional probability that the containment is broken given that an explosion occurs can be analysed based on pressure impulses and the strength of the containment. Deterministic computer codes exist for the computation of pressure impulses. For example, in [14], pressure impulses were calculated using DETO software for hydrogen explosions in reactor building. The containment failure probabilities could be calculated using load vs. strength approach in the same way as for steam explosions in [4]. Also some literature in the response of structures to impulse loads exists (e.g. [15]).

5.3 Hydrogen explosions outside containment

Hydrogen explosion can also occur outside the containment if hydrogen leaks from the containment to the reactor building. This kind of hydrogen explosions occurred at Fukushima causing significantly larger releases than what had occurred before that [11] because the roofs were destroyed. Such explosions have not usually been modelled in PRA. It is theoretically possible that a hydrogen explosion could break the containment from the outside (causing larger containment leak than from which the hydrogen entered to the reactor building), but it has not been considered likely.

Hydrogen explosion outside containment has to be related to some containment failure mode, e.g. failure of containment penetrations, because otherwise hydrogen could not leak to the reactor building. If it is assumed in PRA that such explosion can possibly break the containment, a new containment failure mode needs to be introduced, e.g. "failure of penetrations with ex-containment hydrogen explosion." The probability that the leak leads to an explosion and the containment failure probability need to be estimated. Load vs. strength approach could be applied to the latter. Hydrogen and steam volumes outside the containment and pressure impulses can be calculated using deterministic software. The distribution of the time of the explosion should also be estimated, and the effects of the explosion on the releases should be included in the source term model.

A more realistic scenario is that an ex-containment hydrogen explosion does not break the containment but breaks the reactor building. In that case, the containment failure mode is the cause of the leak. The difference to the previous case is that the reactor building failure probability needs to be estimated, and source term modelling is slightly different.

Source term modelling could be performed so that the amount of radionuclides accumulated in the reactor building would be calculated, like in [3], and when an explosion occurs, all those radionuclides could be assumed to be released. Furthermore, after the explosion, all releases from the containment can be assumed to be released to environment. More detailed modelling of the reactor building could possibly be performed too. If the explosion breaks the containment, the releases from the containment are larger (than with the leak that led to the hydrogen explosion).

6. Containment event tree model

A containment event tree (CET) model for a boiling water reactor plant was developed in [4]. This CET represented a station blackout scenario, and the plant damage state covered both low and high pressure transients. The CET was built originally using SPSA software [16] and later using FinPSA software [17]. In this study, FinPSA is used.

6.1 Dynamic containment event tree analysis

Software tools FinPSA [17] and SPSA [16] offer dynamic containment event tree approach that supports IDPSA. FinPSA is an updated version of SPSA (currently developed and maintained 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 CETL 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 a CET also contains an initial conditions section, where some probability and process variable values are defined. In addition, the model contains a global “common section”, where some global variables and functions can be defined. CETL programming is very flexible. At any branch, new value can be set or calculated for any global variable, and that way accident progression can be modelled dynamically. Binning rules can also 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.

6.2 Overview of the model

The CET model is presented in two parts in Figures 1 and 2. The CET structure is otherwise similar to the CET presented in [4] except that sections have been added for lower drywell (LDW) flooding and filtered venting. In the model, core cooling systems are assumed unavailable until the possible AC power recovery. The sections of the CET, except LDW flooding and filtered venting, as well as source term computation are gone through in detail in [4]. Containment failure modes are presented in Table 2.

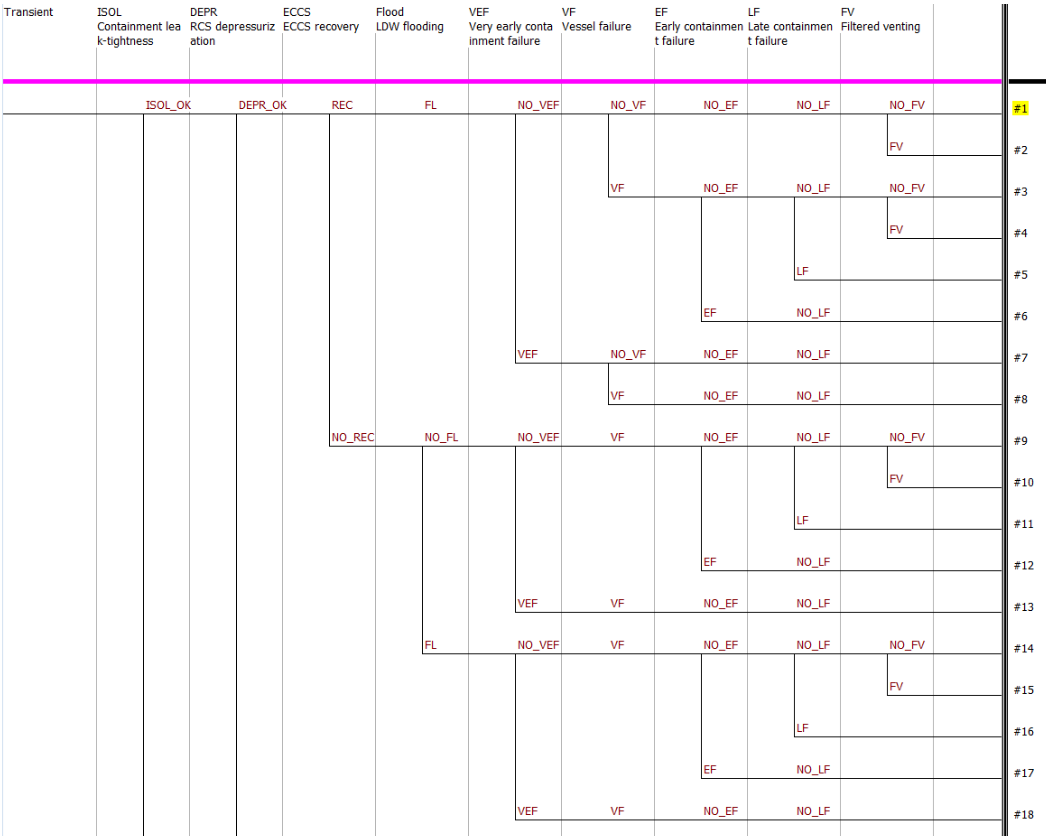

Figure 1: The CET model (part 1).

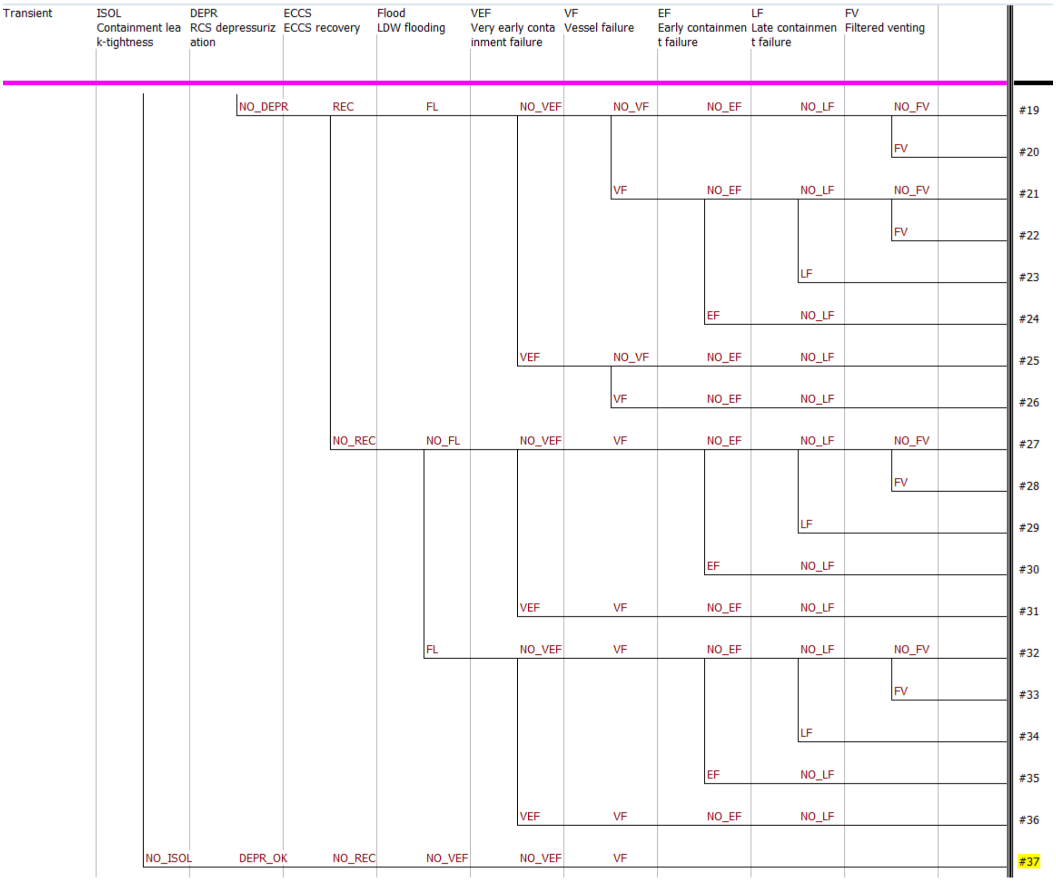

Figure 2: The CET model (part 2).

Table 2: 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)

6.3 Release heights and temperatures

A release height variable and temperature variable were added to the model. Their values are set always after the containment failure mode is defined. Table 3 presents the values that are used in the model. The values are however not based on any real data because the authors had no possibility to perform any deterministic calculations and had no access to any reactor building architecture document from which potential release heights could have been identified. It was assumed that hydrogen explosions and alpha mode steam explosions lead to higher release heights than the other failure modes because they occur high in the containment. Basemat release was assumed to occur at ground level, and the height of the stack was assumed to be 110 meters. It was assumed that release temperatures are higher than 100 °C if explosions occur. Without explosions the temperatures were assumed to be 100 °C with exception of basemat melt-through. In that case, it was assumed that the radionuclides are cooled down to a lower temperature by building structures. These values were chosen just for the sake of having an example and they should not be adapted to any real PRA.

Table 3: Release heights and temperatures for different containment failure modes.

Containment failure mode	Release height	Release temperature (°C)
Containment not leak-tight	5 m with prob. 0.5, 20 m with prob. 0.5	100
Containment over-pressurization	5 m with prob. 0.5, 20 m with prob. 0.5	100
Hydrogen deflagration/detonation	30 m with prob. 0.5, 50 m with prob. 0.5	Lognormal distribution with mean 500 and error factor 1.5
Alpha-mode failure	20 m with prob. 0.5, 30 m with prob. 0.5	Lognormal distribution with mean 200 and error factor 1.5
Ex-vessel steam explosion	5 m with prob. 0.5, 20 m with prob. 0.5	Lognormal distribution with mean 500 and error factor 1.5
Failure of containment penetrations	5 m with prob. 0.5, 20 m with prob. 0.5	100
Non-coolable ex-vessel debris causes basemat melt-through	0 m	Lognormal distribution with mean 50 and error factor 1.5
Filtered venting	110 m	100

6.4 Uncertainty analysis

In the original model [4], uncertainty analysis was not performed for release probabilities. Instead, only one sequence (or in some cases two sequences) was realised in a single simulation cycle. For this study, the model was changed so that results were calculated for all sequences in all simulation cycles, and uncertainty distributions were assigned to probability parameters. The parameters are presented in Table A-1 in Appendix A. The distributions were chosen so that the mean values remained same as in the original model, and the model was kept as close to the original as possible, though many parts required significant modifications.

A significant change is that previously some physical parameters, e.g. ECCS recovery time and core release fraction, determined which sequence is realised, but in the new model, the values of those parameters depend on the accident sequence. Hence, the logic is kind of reversed and emphasises probabilistic analysis, while the physical modelling is more on background. But still the source term results remain approximately the same.

A good and simple example of modifications that were made is the emergency core system recovery. The failure probability of recovery is now drawn from a lognormal distribution, which differs depending on whether the depressurisation is successful. Previously, the recovery time was drawn from a distribution and compared to time available which was also drawn from a distribution. On each simulation cycle, either the recovery occurred in time or not, but the updated version of the model covers both scenarios at the same time. The mean values of failure probability distributions (of high and low pressure cases) were taken from the results of the previous model. Also, in the updated model, for relevant accident sequences, the recovery time is drawn from the distribution as many times as it takes for the recovery time to be smaller than the time available for the recovery.

A new section was added for the LDW flooding so that both flooded and unflooded scenarios can be calculated in same simulation cycle. Previously, flooding was handled in ECCS recovery section.

In the old model, very early containment failure either occurred or not in a simulation cycle, but the new model calculates a probability for it as a sum of the probabilities of different failure modes. Still, only one failure mode is realised in one simulation cycle. The failure mode is drawn based on the fractions of the failure mode probabilities.

In the old model, the vessel failed if the core release fraction was larger than a given limit value. Both the core release fraction and the limit value were drawn from a distribution. In the new model, a probability is drawn for the vessel failure instead. In addition, a large value is given for the core release fraction if the vessel fails, and a small value is given if the vessel does not fail so that the source term calculation will work in the same way as in the old model.

Again, in the old model, early containment failure either occurred or not in a simulation cycle, but the new model calculates a probability for it as a sum of the probabilities of different failure modes. Only one failure mode is realised in one simulation cycle. The failure mode is drawn based on the fractions of the failure mode probabilities. In the old model, ex-vessel steam explosion induced containment failure was evaluated by drawing LDW strength and pressure impulses from distributions, but in the new model, failure probabilities taken from [4] are used directly as the mean values of the probability distributions.

The new model also includes a probability parameter for high amount of melt being ejected to the LDW. The mean probability was calculated from the results of the old model. Also, in the case of ex-vessel steam explosion and ECCS working, the core release fraction and debris fraction are updated because high amounts of melt are more likely. The distribution for these fractions was estimated from the results of the old model, but it is quite rough.

In the old model, late containment failure either occurred or not, but the new model calculates a probability for it. The probability is the product of the probability that debris exists, the probability that the debris is not coolable and the probability that the basemat melt-through occurs if the debris is not coolable. In the old model, debris coolability fraction needed to be small enough, and the debris needed to exist so that the basemat melt-through could occur. Probabilities for these conditions were estimated from the results and are used in the new model. A small value is also given for debris coolability fraction if basemat melt-through occurs, and it is ensured that the core release fraction and debris fraction are larger than 0.

A new section was also added for filtered venting. In the old model, the possibility of filtered venting was considered in NO_VEF, NO_EF and NO_LF functions of VEF, EF and LF sections. Because of this, there were sequences that could lead either to release category OK or FV depending on the drawn input parameters. The new filtered venting section was added so that these release categories can be analysed in different sequences in the same simulation cycle. This way the uncertainty distributions are calculated correctly. Very early, early and late filtered venting are all considered in this same section. The venting probability is calculated based on the probabilities of these venting modes as

$$p_{VEFV} + (1 - p_{VEFV})(p_{EFV} + (1 - p_{EFV})p_{LFV}).$$

One of these venting modes is realised in a simulation cycle. The venting mode is drawn based on the fractions of the probabilities.

6.5 Results

Table 4 presents the results including the conditional probabilities and release fractions for all release categories. The four values in the cells of the table are mean, 5th percentile, median and 95th percentile. The Prob. column shows the percentile values of the new uncertainty distributions. When uncertainty analysis is correct, FinPSA also produces a set of proper graphs presenting the uncertainty distributions, e.g. those in Figures 3-6.

Table 4: The summary of results.

Bin	Prob.	S_Xe	S_Cs	S_Ru
OK	2.24E-01 1.09E-01 2.25E-01 3.39E-01	7.58E-11 1.76E-11 9.97E-11 1.00E-10	0.00E+00 0.00E+00 0.00E+00 0.00E+00	0.00E+00 0.00E+00 0.00E+00 0.00E+00
ISOL	9.98E-03 4.54E-03 9.19E-03 1.82E-02	8.41E-01 4.50E-01 9.98E-01 1.00E+00	2.55E-01 6.46E-02 2.41E-01 4.75E-01	8.90E-03 0.00E+00 3.50E-03 3.40E-02
VEF	1.86E-01 9.34E-02 1.71E-01 3.27E-01	7.58E-01 1.76E-01 9.97E-01 1.00E+00	1.13E-01 9.01E-03 8.52E-02 3.15E-01	3.77E-03 2.10E-06 7.51E-04 1.75E-02
EF	3.09E-02 8.49E-03 2.43E-02 7.58E-02	7.80E-01 2.28E-01 9.97E-01 1.00E+00	1.57E-01 2.16E-02 1.39E-01 3.71E-01	5.49E-03 2.00E-06 1.71E-03 2.31E-02
LF	1.99E-02 6.66E-03 1.74E-02 4.11E-02	7.90E-01 2.76E-01 9.97E-01 1.00E+00	1.54E-01 2.36E-02 1.36E-01 3.54E-01	4.49E-03 0.00E+00 1.01E-03 1.99E-02
FV	5.29E-01 3.97E-01 5.28E-01 6.67E-01	7.57E-01 1.74E-01 9.97E-01 1.00E+00	9.82E-04 2.62E-06 6.57E-05 3.58E-03	2.90E-05 0.00E+00 3.19E-07 6.25E-05
Weighted Total	1.00E+00 1.00E+00 1.00E+00 1.00E+00	5.90E-01 1.41E-01 6.90E-01 8.67E-01	3.18E-02 4.69E-03 2.40E-02 8.58E-02	1.06E-03 5.01E-07 3.03E-04 4.46E-03

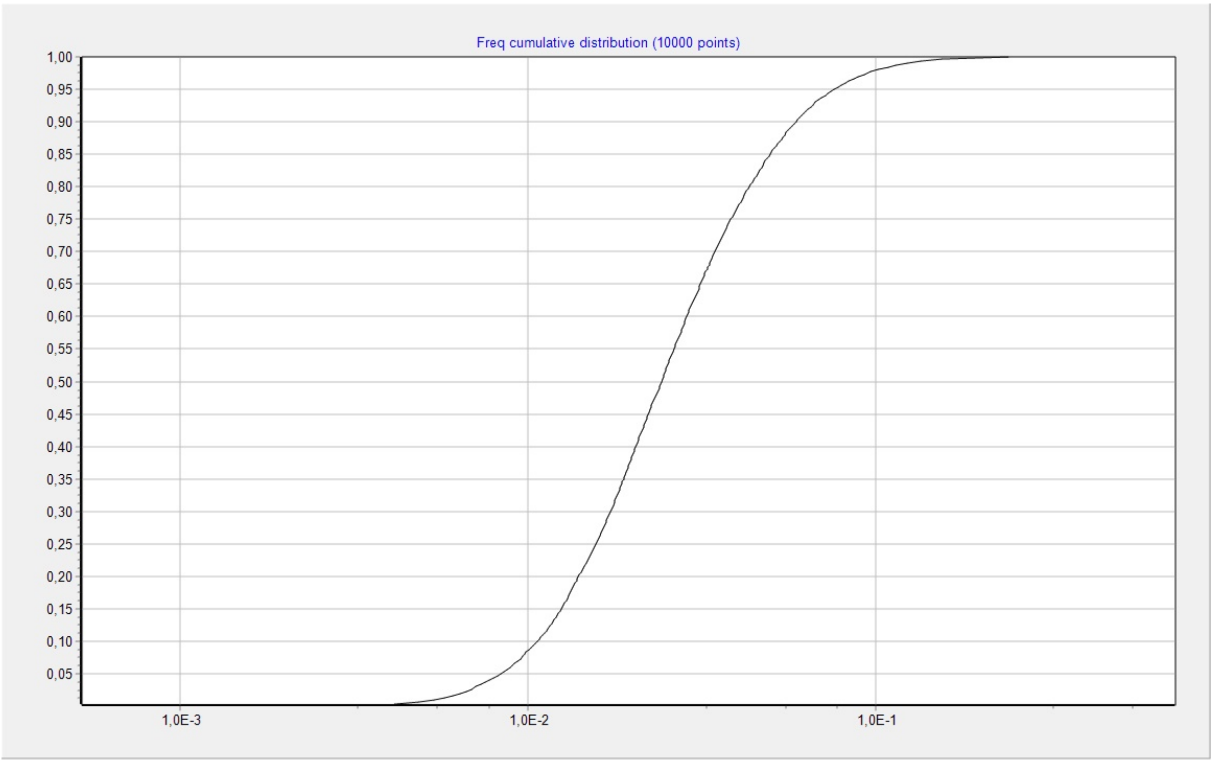

Figure 3: Cumulative distribution of the conditional probability of early containment failure.

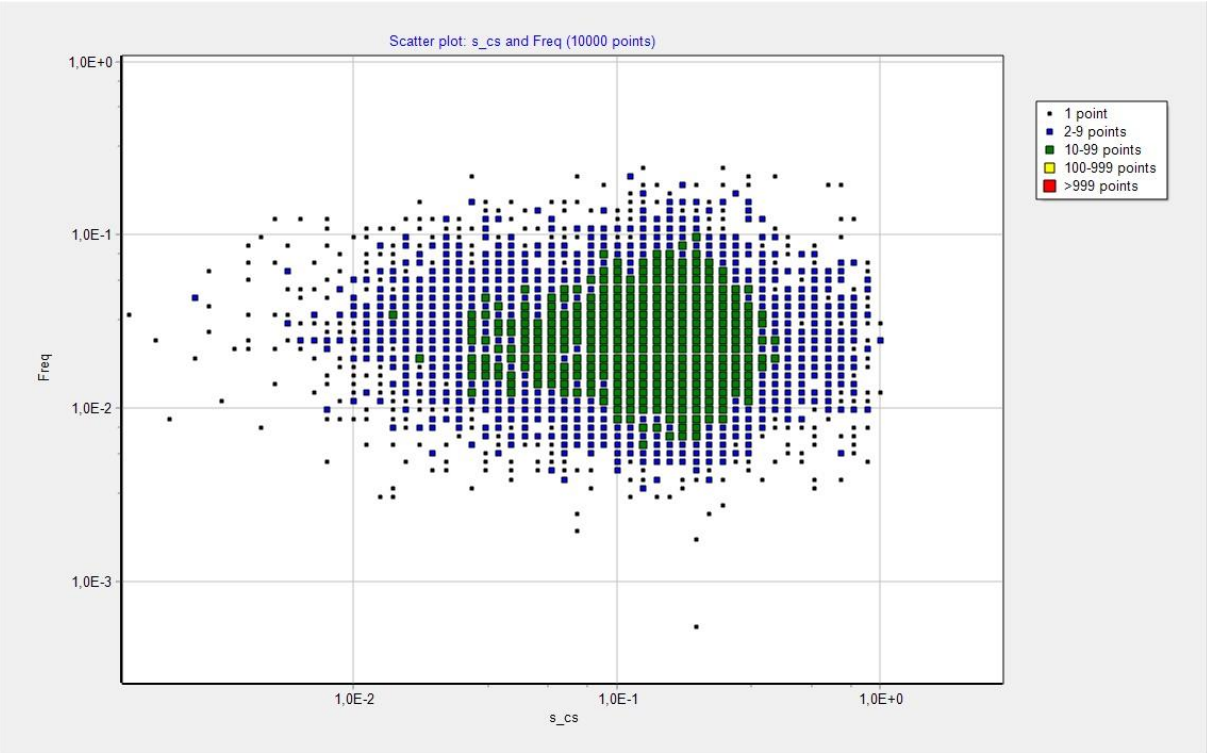

Figure 4: Scatter plot between conditional probability and cesium release fraction in early containment failure based on weighted point values.

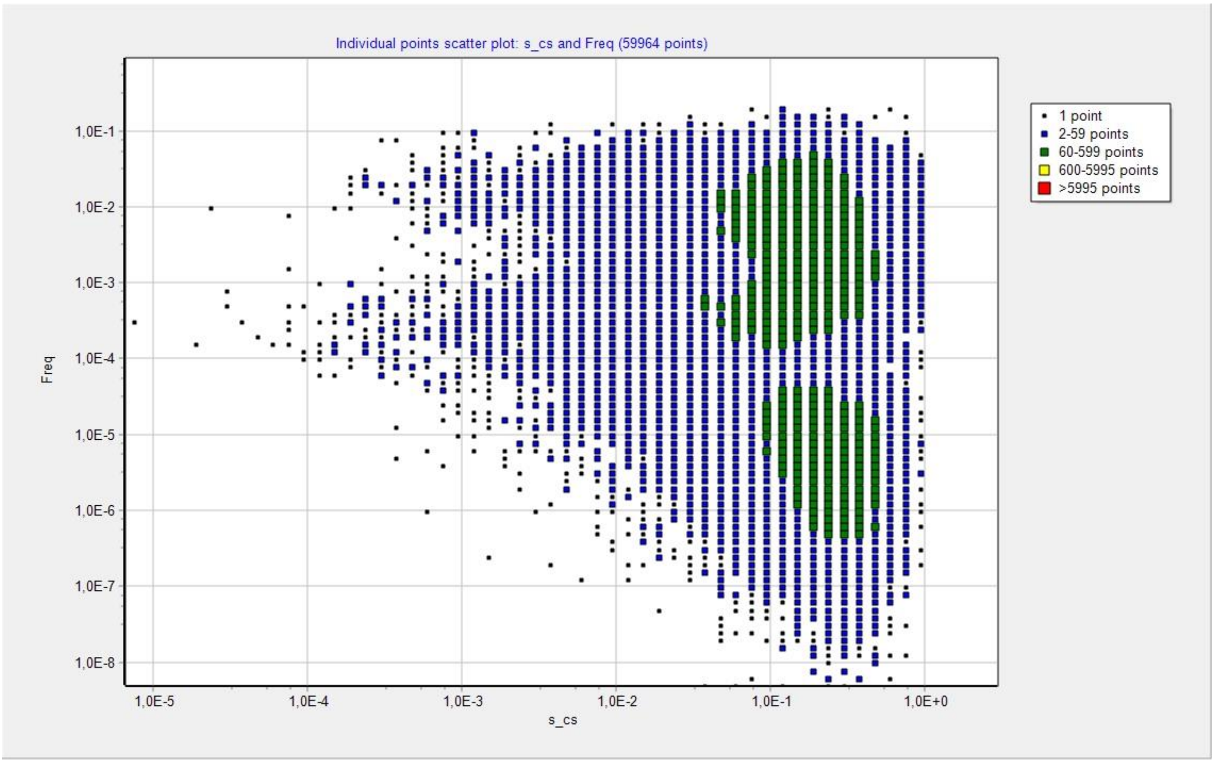

Figure 5: Scatter plot between conditional probability and cesium release fraction in early containment failure based on individual simulation points.

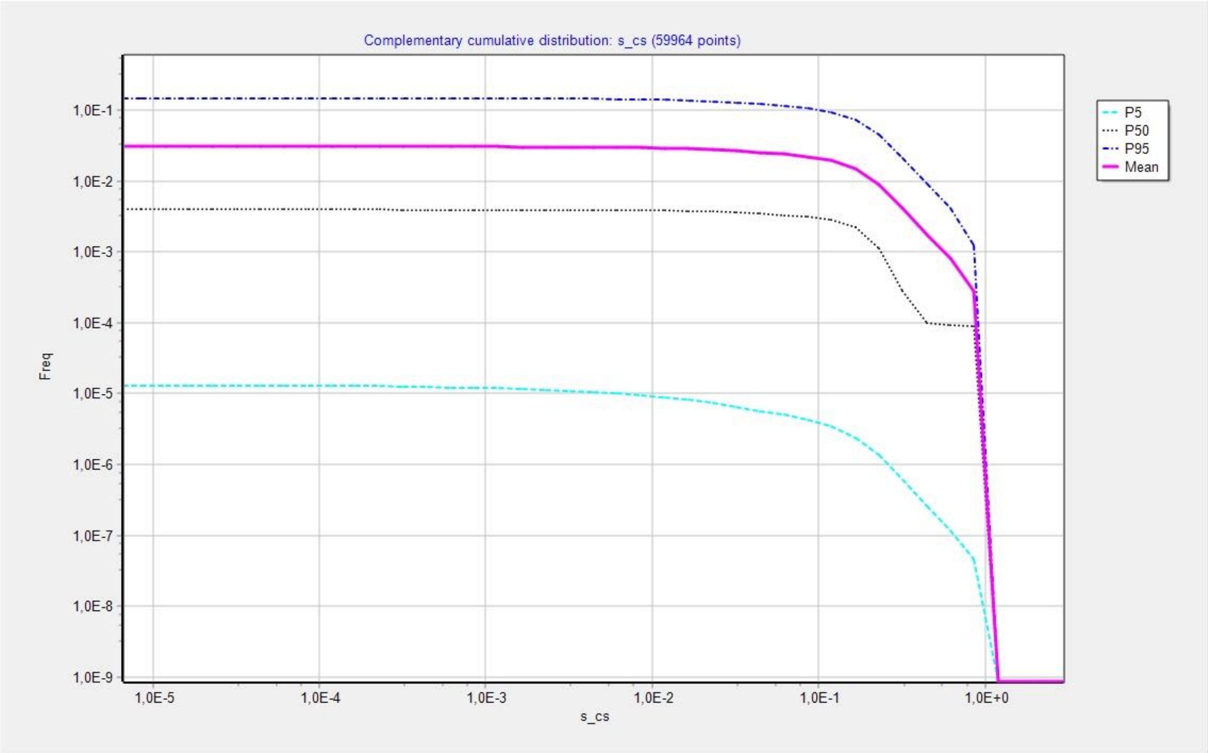

Figure 6: Complementary cumulative distribution of cesium release fraction in early containment failure.

7. Conclusions

This report studied nuclear power plant accident source term characteristics and hydrogen explosions from the PRA point of view. Release height and temperature were considered for different accident scenarios based on general knowledge, literature and discussions with deterministic safety analysis experts. Release height is usually the height of the location where the reactor building leaks (which depends on the containment failure mode), or the height of the chimney if the release is controlled. In an uncontrolled accident case, both the containment failure location and the flow path of radionuclides in the reactor building need to be analysed to determine the release height. The determination of the accurate release height distribution is very difficult in some cases, especially when an explosion occurs. The temperature of release from containment is in most cases close to 100°C, but the temperature of radionuclides can potentially change during their migration in reactor building. Higher temperatures can be caused by fires and explosions. There are computer codes that can be used to analyse radionuclide flows in reactor building and determine the release heights and temperatures. Both release heights and temperatures can be included in PRA models easily if their probability distributions are known.

Hydrogen explosions in BWR were considered. Different areas of the analysis concerning both explosions inside and outside containment were discussed. Hydrogen explosions can occur in BWR containment even if it is normally inerted, if accident occurs during start-up, maintenance or shut-down, or if the inerting system fails. Hydrogen explosions outside containment have not usually been modelled in PRA, even though they can affect the releases significantly.

Complete uncertainty analysis was developed for an existing simplified BWR containment event tree model. This required significant changes in the model. The original model was heavily based on computation with physical parameters, whereas in the new version, the focus is on probabilistic modelling. The mean values of the probability distributions used in the new model were derived from the original model and its results. If there had not been the original model, other supporting analyses to estimate those mean probabilities would have been needed instead. It can actually be a good idea to perform the modelling in two phases like this: first focusing on physical modelling to obtain preliminary results and implementing the uncertainty analysis later.

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Appendix A: Uncertainty distributions of parameters

Table A-1: Uncertainty distributions of probability parameters.

Parameter	Distribution	Mean	Error factor (P95/P50)
Failure probability of ECCS recovery in pressurised case	Lognormal	0.024	2.0
Failure probability of ECCS recovery in depressurised case	Lognormal	0.078	2.0
Recriticality probability	Lognormal	0.5	1.5
Failure probability of LDW flooding if ECCS does not work	Lognormal	0.05	3.0
Probability that containment is not inert	Lognormal	0.3	2.0
Probability that over-pressurisation breaks the containment if the core is recritical	Lognormal	0.1	2.0
Probability that hydrogen explosion breaks the containment if it is not inert	Lognormal	0.5	1.5
Probability that an alpha mode steam explosion breaks the containment	Lognormal	0.0001	10
Probability of very early filtered venting in depressurised case if the core is recritical	Lognormal	0.25	2.0
Probability of very early filtered venting in pressurised case if the core is recritical	Lognormal	0.5	1.5
Probability of no vessel failure	Lognormal	0.5	1.5
Failure probability of containment penetrations in pressurised case if LDW has been flooded and vessel has failed	Lognormal (multiplied by 2)	0.05	3.0
Probability that only little melt is ejected to the containment if ECCS works and vessel has failed	Lognormal	0.45	1.5
Failure probability of containment penetrations if LDW has not been flooded and vessel has failed	Lognormal	0.5	1.5
Probability of no early filtered venting in pressurised case if ECCS works and the core is recritical	Lognormal	0.01	5.0
Probability of early filtered venting in pressurised case if ECCS works and the core is not recritical	Lognormal	0.5	1.5

Probability of early filtered venting in depressurised case if ECCS works and the core is recritical	Lognormal	0.5	1.5
Probability of early filtered venting in pressurised case if ECCS recovery has failed and the core is recritical	Lognormal	0.5	1.5
Probability of early filtered venting in depressurised case if ECCS works and the core is not recritical	Lognormal	0.25	2.0
Probability of early filtered venting in depressurised case if ECCS recovery has failed and the core is recritical	Lognormal	0.25	2.0
Containment failure probability due to ex-vessel steam explosion in pressurised case if much melt is ejected	Lognormal	0.091	3.0
Containment failure probability due to ex-vessel steam explosion in pressurised case if little melt is ejected	Lognormal	0.003	3.0
Containment failure probability due to ex-vessel steam explosion in depressurised case if much melt is ejected	Lognormal	0.207	3.0
Containment failure probability due to ex-vessel steam explosion in depressurised case if little melt is ejected	Lognormal	0.021	3.0
Probability of no ex-vessel steam explosion triggered if melt is ejected in high pressure	Lognormal	0.01	10
Probability of ex-vessel steam explosion triggered if melt is ejected in low pressure	Lognormal	0.5	1.5
Probability of basemat melt-through if the debris is not coolable	Lognormal	0.1	2.0
Probability of no late filtered venting in pressurised case	Lognormal	0.1	2.0
Probability of late filtered venting in depressurised case	Lognormal	0.5	1.5
Probability of no debris if ECCS recovery failed and vessel has failed	Lognormal	0.006	5.0
Probability that the debris is not coolable if LDW is flooded and the debris exists	Lognormal	0.5	1.5