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Feasibility Study for Connection Between IDPSA and conventional PSA Approach to Analysis of Nordic type BWR's

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Abstract

This report summarizes the experience achieved within the NKS-DPSA project during 2013. The project is motivated by the discussions at the Workshop on Integrated Deterministic-Probabilistic Safety Analysis (ID-PSA-2012). The aim of the project has been to: summarize the state of the art review of the probabilistic, deterministic and IDPSA analysis; and to carry out a feasibility study. The following areas are covered in this study: mapping, information collection and identification of areas of certain interest based on existing PSA; results of analysis of core relocation scenarios taking into account timing of PSA Level 1 events and possible recovery actions on the melt conditions in the lower head; results of feasibility study on connection between conventional PSA, DSA and IDPSA methods. There are three topics for the feasibility studies identified: a transient with complete or partial failure of the hydraulic scram; station black-out with varying degrees of safety system recovery; steam explosion. Experiences from performed studies are summarized in the report as well as suggestions of areas which need further investigations. The results from the ID-PSA show that an increased number of thermohydraulic calculations, performed according to an intelligent algorithm, can improve the understanding of the sequences and therefore input to the PSA or to the deterministic safety analyses. There is a good potential for development of a mathematical model to represent the IDPSA results in form of a decision tree as input for the quantification of the PSA Level 2 structure. Steam explosion analyses exploiting IDPSA methodology would necessitate more detailed approach than VTT's contribution presented here, with use of dedicated analysis codes for FCI phenomena and structural response of the containment. Nevertheless, joint use of MELCOR and SPSA for steam explosion analysis provided a good basis that can easily be refined further.

Key words

IDPSA, PSA, SPSA, DSA, BWR, Severe accident, MELCOR, core degradation, steam explosion

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IDPSA: Integrated Dynamic Probabilistic Safety Assessment

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Final Report from the NKS-R DPSA activity

(Contract: NKS_R_2013_107)

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Chapter 1. **Introduction**

This reports summarizes the experience achieved within the project DSA/PSA Dynamic Probabilistic Methodology during 2013. The project is motivated by the discussions at the workshop "Proceeding of the IDPSA-2012 Integrated Deterministic-Probabilistic Safety Analysis Workshop"[1]. The aim of the project has been to:

- Summarize the state of the art review of the probabilistic, deterministic and IDPSA analysis
- Feasibility study. The following areas are covered in this study;
    - Mapping, information collection and identification of areas of certain interest based on existing PSA
    - Results of analysis of core relocation scenarios taking into account timing of PSA Level 1 events and possible recovery actions on the melt conditions in the lower head
    - Results of feasibility study on connection between conventional PSA, DSA and IDPSA methods

There are three feasibility studies identified:

- A transient with complete or partial failure of the hydraulic scram (system 354)
- Station black-out with varying degrees of safety system recovery
- Steam explosion

Experiences from performed studies are summarized in the report as well as suggestions of areas which need further investigations.

Chapter 2. State of the art review of the probabilistic, deterministic and combined DPSA analysis.

Over the past decades, methods of Probabilistic Safety/Risk Analysis (PSA/PRA) have emerged as important tools to examine safety of complex, potentially hazardous, engineered systems such as Nuclear Power Plants (NPP). As safety requirements become increasingly stringent, requirements for quality and completeness of PSA models also increase. However further increase of the PSA models complexity is not necessarily an effective way to increase accuracy of PSA methods.

Deterministic analyses are the basis for construction of a nuclear power plant. The analyses are based on the single failure criterion and a number of conservative assumptions such as loss of offsite power and no credit for non-safety systems. The initiating events considered in the analyses are divided in different event categories, ranging from likely events (once or several times/year) down to residual risks. The more likely the event category is the higher the margins (conservatisms) against core damage must be. Thus, there are already some probabilistic considerations in the deterministic analyses.

The core damage frequency for both existing and advanced future plants is calculated to be in the range from 10^{-5}/reactor year to 10^{-8}/reactor year. However, the plant operation is sometimes hit by “improbable” (defined in PSA as very low probability) events, which can surprise, revealing a potential vulnerability in the complex plant system. We recognize that state of the art PSA methods provide numbers to quantify probability of what is already known as an “issue”, but are not capable of revealing what, and to what extent, is not known (i.e. scenarios that are not prescribed in the PSA input). PSA is based on a set of assumptions about possible accident scenarios believed to be conservative. Such “decomposition” of a complex problem into a set of pre-defined sequences can be prone to false conservatism in the PSA or deterministic analysis, rendering possibility of potentially dangerous scenarios being missed or underestimated.

Standard PSA and deterministic approach has fundamental problems with resolving the dynamic nature of mutual interactions between (i) stochastic disturbances (e.g. failures of the equipment), (ii) deterministic response of the plant (i.e. transients), (iii) control logic and (iv) operator actions. Passive safety systems, severe accident and containment phenomena are examples of the cases when such dependencies of the accident progression on timing and order of events are especially important.

Since the late eighties, realistic deterministic-dynamic models, commonly referred to as best-estimate methods, received recognition as safety analysis tools. However, the best estimate codes are still used in a largely decoupled manner from the PSA. That hinders their application to risk analysis and identification of plant vulnerabilities.

In making predictions regarding the response of a system to disturbances, both the uncertainties arising from the stochastic nature of events (aleatory uncertainties) as well as those arising from lack of knowledge about the processes relevant to the system (epistemic uncertainties) have to be taken into account. Often, it is difficult to distinguish between epistemic and aleatory uncertainties [2]. Dynamic PSA methodologies allow a unified framework to account for the joint effects of both types of uncertainties simultaneously in predicting the distribution of risk associated with the system response.

Dynamic PSA methodologies can be divided into three main categories [3]: (i) continuous-time methods, (ii) discrete-time methods, and (iii) methods with graphical interfaces. While the methods with graphical interfaces are also either continuous or discrete time methods, they are listed as a separate category because the availability of a graphical interface is usually regarded as rendering them more user friendly. The inputs for all dynamic methodologies are:

- a time-dependent system model (such as RELAP5 [4] or MELCOR [5] codes),
- possible normal and abnormal system configurations which may need to be determined using a failure-modes-and-effects (FMAE) analysis, and
- transition probabilities (or rates) among these configurations.

2.1 IDPSA Workshop 2012 Discussions

In November 2012 a large workshop was held in Stockholm with the objective to:

- Discuss the needs within industry and how they can be solved with IDPSA
- Discuss development of IDPSA methods and applications
- Plan joint research activities on deployment of IDPSA in the framework of research agenda

About 50 nuclear safety analysis experts from Europe and USA participated in the workshop. The conclusions drawn are presented in sections 2.1.1 to 2.1.8.

It could, at the workshop, be stated that IDPSA tools have been developed for decades and that there has been considerable experiences gained during these years.

2.1.1 End user needs (Industry)

Post Fukushima stress tests led to suggestions for improvements regarding the safety in nuclear power plants (NPP's) around the world. Issues of interest to the industry (for investigation) concerns: Filtered containment venting, Loss of ultimate heat sink, Flood protection etc. However, changes due to some of the suggestions could render both positive and negative effects. Furthermore, the impact on the risks involved due to the changes suggested vary depending on reactor system design (which is generally plant specific).

Concerns for IDPSA from an industry point of view

A concern from the industry's point of view is the risk that advanced computational software may produce a lot of data instead of knowledge. The need of consistency, completeness and credibility (in terms of uncertainties) were therefore emphasized. Furthermore, experimental support and validation of results remain very important and issue resolution is regarded as "the measure of success" (not just successful running of a code).

2.1.2 End user needs (Regulators)

There is a need for verification of the current process in decision making and its criteria. Defense in depth and consistency between PSA and DSA need to be verified. IDPSA is seen as a way of (almost) "assumption free" approach to determine what is conservative in terms of both PSA and DSA.

The question *"Why every time there is a new issue at NPP's, we go back to classic*

PSA/DSA?" was raised during the meeting.

The traditional approach was considered problematic in terms of licensing of new reactors since Gen III+ Gen IV reactors does not show the same kind, nor level, of safety issues as present reactor designs.

Concerns for IDPSA from a regulator's point of view

"PSA standards are in the process of "harmonization". The motivation for harmonization is to establish a confidence in PSA results."

IDPSA must therefore acquire best practice guide lines in order to establish the above mentioned confidence in results. There are evolutionary changes towards a risk-informed approach, but this is not a mainstream approach yet.

2.1.3 The "Catch-22" situation

The end users (both industry and regulators) are likely to benefit from a risk-informed approach for assessments and decision making. However, they are generally lacking in awareness and knowledge about the benefits with IDPSA. This leads to the following situation:

- "Risk assessment tools (IDPSA) are not adopted by industry/regulator because Risk informed decision making is not a common practice.
- Risk informed decision making is not a common practice because Risk assessment tools (IDPSA) are not adopted yet."

2.1.4 Opportunities and challenges for IDPSA deployment

"There is a consensus that to break the "22 loop" successful demonstrations of cases are necessary."

"Consistent, complete and credible assessment of selected problems (e.g. stress test suggestions) in order to "resolve the issues" can be a way for IDPSA to break the "22 loop"".

Safety margin assessment requires both an integrated Probabilistic and Deterministic assessment which IDPSA provides. However there are concerns regarding benefits versus costs considering IDPSA for risk-informed decision making. Therefore, clear communication about costs and benefits of risk-informed decision making is crucial for a successful deployment.

During the meeting the following questions were raised:

- How would decision makers like to make decisions?
    - Answering to the wrong questions "precisely” and quickly"?
    - Answering the right questions with explicit quantification of uncertainties, through a time consuming effort?
    - Is there even a choice?

2.1.5 IDPSA methods and the decision making process

It was established during the meeting that the readiness of a tool is difficult to determine if there are no clear criteria for success or goal for the analyses. In terms of decision making, quantification of consequences into figures of merit is necessary (i.e. to establish safety goals and success criteria).

IDPSA methods are capable of quantifying aleatory uncertainties in time dependent scenarios. It was emphasized (during the meeting) that this mostly had an effect within the context of academia whereas it did not do much for deployment into the industry. Therefore, focus must be directed towards what the decision makers need and what they regard as important.

- Is it more important to the decision makers by which tool the analysis is provided, or what kind of information they are provided with?

Credibility, uncertainty quantification (robustness of decision), comprehensiveness (risk profile instead of one number) and understanding were outlined as important factors in terms of what kind of data to be provided for the decision makers. Consistency was also emphasized as important since different kinds of decisions (e.g. for industry or regulators) put different requirements on the data provided.

An example of a decision making approach is Risk Oriented Accident Analysis Methodology (ROAAM). The focus of ROAAM is upon reducing the uncertainty to the extent that a defense-in-depth is considered as achieved. The idea is that when the whole community of experts in a given problem area is convinced that the demonstration is effected and regarded successful the problem may be considered solved (in a robust and final way).

"Eventually the complete reaching of all experts is effected by publication in the technical literature, with additional iterations thereof if necessary."

ROAAM integrates risk assessment and risk management in an effective manner.

"ROAAM provides guidelines for development of framework for bounding of epistemic (modelling) and aleatory (scenario) uncertainties in a transparent and verifiable manner that enables convergence of experts opinions on the outcome of the analysis (not necessarily on the uncertainties in the input information)."

2.1.6 Needs for methodology development to facilitate deployment of IDPSA

During the meeting it was established that depending on the level of completeness, consistency and credibility, IDPSA may increase as well as decrease the uncertainty. Furthermore, completeness must be evaluated against computational costs and consistency against assumptions and simplifications. Constant awareness and documentation are important aspects to consider within methodology development. Finally, uncertainties must be traced and understood to obtain transparency and credibility.

"There is a general consensus that any source of uncertainty (including domains of applicability/validation for deterministic codes) should be subject to sensitivity and, if necessary, uncertainty quantification studies with appropriate (existing) approaches."

2.1.7 Connections between IDPSA and PSA, DSA

"Main potential benefit from PSA/DSA application in combination with IDPSA is establishment of consistent conservatism in PSA and DSA analyses:

- *PSA conservatism: clarification of consistency in assumptions.*
    - *IDPSA can provide information about phenomenology to the PSA experts when there is large uncertainty about it.*

    - *Epistemic uncertainty concerning model uncertainties (not parameter uncertainties) is not taken into account in the probabilistic part of PSA, but should be.*
  - *Even if PSA is on "average conservative" but locally it can be "non-conservative".*
    - *Example: an effort in USA on IDPSA analysis for Feed and Bleed to increase NRC's confidence in PSA.*
  - *Misunderstanding of the true sources of safety margins can lead to decisions which might increase the risk.*
- *DSA conservatism: clarification of conservatism in assumptions about the transients to be run as "enveloping" or maximum credible accident.*
- *Identification of major uncertainty sources in deterministic modelling.*
  - *Basis for code improvement and*
  - *Validation.*
- *Eventually, aiming at consistency between success criteria in PSA and DSA analysis.*

Other issues related to connection between PSA, DSA and IDPSA:

- *It is important to identify where (in which scenarios, for which purposes) IDPSA is needed.*
- *There can be a "cliff edge" effect. How to determine how far away are we from the cliff?*
- *Plant modifications*
  - *Power uprate changes the success criteria. Re-evaluation of all success criteria is a huge effort.*
  - *Changing of the PSA model is a huge effort when the plant changes.*
  - *How to change the IDPSA model when the plant changes?*
    - *A model consists also of the assumptions made. These assumptions should be verified to see which ones are no longer valid and what new assumptions are to be made."*

2.1.8 Post meeting agenda for deployment of IDPSA as complimentary to the PSA and DSA methods

It was concluded during the meeting that effort should be directed towards revising the EU-IDPSA proposal in order to prepare the IDPSA project for NUGENIA. It was emphasized that in order to succeed with the deployment of IDPSA, focus must be directed towards aspects that are relevant and important to the end users (industry and regulators).

Two successive meetings (to the meeting summarized within this chapter) were held during the year 2013 with the aim of increasing the awareness of IDPSA. Those meetings were:

- NURETH-15 The International Topical Meeting on Nuclear Reactor Thermal Hydraulics 12-17 May, 2013, Pisa, Italy, Workshop/Seminar W1 "Combining deterministic and probabilistic methods for comprehensive safety margin assessment"

- Nordic PSA Conference – Castle Meeting 2013, 10th -12th April, Stockholm, Sweden.

2.2 An overview of DPSA methodologies

The ADAPT (Analysis of Dynamic Accident Progression Trees) methodology developed at Ohio State University (OSU) with support from the Sandia National Laboratory (SNL). In conjunction with a computer code describing the dynamic system behavior, ADAPT uses dynamic event trees for the systematic and mechanized quantification of the joint contribution of the impact of aleatory and epistemic uncertainties on the consequences of possible event sequences and their likelihoods. The methodology is implemented using massively parallel processing using the ADAPT software. The output of ADAPT consists of possible event sequence or scenarios, as well as their frequencies, originating from a user specified initiating events and based on user specified branching rules. ADAPT has various graphical capabilities for the display of the results. It has been linked to MELCOR and RELAP5 codes and implemented for the analysis of various initiating events, including station blackout (SBO) with power recovery in two PWRs and aircraft crash on the towers of the reactor vessel auxiliary heat removal system of an example sodium cooled fast breeder [6, 8].

The MCDET methodology developed by GRS combines Monte Carlo (MC) simulation with Dynamic Event Tree (DET) analysis. MCDET is capable of accounting for any discrete and continuous aleatory and epistemic uncertainties. Any probabilistic model may be applied without need for simplifications or for focusing on specific probability distributions. Discrete aleatory uncertainties are treated by the DET approach. It keeps track of all combinations of potential alternatives for the discrete uncertainties. More than two alternatives for a discrete uncertainty can be considered. MC simulation is applied in combination with the DET approach to consider continuous aleatory uncertainties. The values obtained from MC simulation are successively supplied as input to the further calculation of a DET. The output of MCDET consists of a huge amount of event sequences and (conditional) probability distributions. The final probabilistic assessments are derived from the mean probability distributions over all DETs which are given together with confidence intervals. So far, MCDET has been linked to the MELCOR code and has been implemented for various type of initiating events in a Konvoi type PWR. The implemented MCDET module system can in principal be coupled with any deterministic dynamic code [8, 9].

The Genetic Algorithm IDPSA (GA-IDPSA) approach jointly developed by KTH and Moscow Power Engineering Institute uses global optimum search method (genetic algorithm) to increase computational efficiency in exploration of the uncertainty and plant accident scenarios space. In the exploration process the GA method imposes no restrictions on consideration of different types of (i) uncertain variables (continuous and discrete), (ii) uncertainties (aleatory and epistemic), and (iii) branching (binary and non-binary). This method is best suited for identification of failure domains including worst case scenarios (maybe rare but high consequence hazards). Stochastic properties of the GA are used for estimation of the probabilities. DET can be constructed based on post-processing of the GA-IDPSA search data. The methodology is implemented using massively parallel calculations implemented in the GA-NPO software. It has been linked to RELAP5 code and implemented for the analysis of various initiating events of the

WWER-1000 type reactor and model of the typical PWR. It can be adapted to any other deterministic code [8, 10].

The modeling strategy of ADS (Accident Dynamic Simulator) developed at University of Maryland(UM) is based on breaking down the accident analysis model into different parts according to the nature of the processes involved, simplifying each part while retaining its essential features, and developing integration rules for full scale application. Whenever a hardware system state transition point or an operator interaction point is reached, the accident scheduler chooses one path to follow. After the simulation process reaches an end point, the scheduler directs the simulation back to the previous branch point, reinitializes every simulation module back to this time point, and follows the other branch point path. In the multiprocessor version of ADS, the simulations are distributed among multiple client computers. A central server is responsible for managing assignment of simulation tasks to individual clients and post-simulation reassembly of the simulation results. ADS has been linked to the RELAP5 code and is mostly used for human reliability analysis [8, 11].

The Dynamic Flow graph Methodology (DFM) is a digraph-based technique. A process variable is represented by a node discretized into a finite number of states. The system dynamics is represented by a cause-and-effect relationship between these states which can be obtained from a system code, but could be qualitative relations as well. Instead of minimal cut sets, the DFM yields the prime implicants for the system. A prime implicant is any monomial (conjunction of primary events) that is sufficient to cause the top event, but does not contain any shorter conjunction of the same events that is sufficient to cause the top event. DFM has been implemented for the reliability analysis and PSA of control systems, human behavior and software [8, 12].

The GO-FLOW methodology is a success-oriented system analysis technique, capable of evaluating system reliability and availability. The modeling technique produces the GO-FLOW chart, which consists of signal lines and operators. The operators model function or failure of the physical equipment, a logical gate, and a signal generator. Signals represent some physical quantity or information. The system model is assembled from the available hardware models (e.g. valves, pumps) in the GO-FLOW library through a graphical user interface. The analysis is performed from the upstream to the downstream signal lines, and is completed when the intensities of the final signals at all time points are obtained. GO-FLOW output includes time dependent system reliability/availability, cut sets, common cause failure analysis and, uncertainty analysis [8, 13, 14].

The ISA (Integrated Safety Analysis Methodology), developed at the Modeling and Simulation Department (MOSI-CSN) of the Spanish nuclear regulatory agency CSN, has been implemented in a computer code cluster (SCAIS, Code system for integrated safety analysis), by MOSI-CSN together with several Madrid Polytechnic University departments (UPM) and INDIZEN software company. Universite Libre de Bruxelles (ULB) and Lietuvos Energetikos Institutas (LEI) have also cooperated. ISA-SCAIS distinctive feature is its regulatory orientation towards the development of diagnostic tools able to check the fulfillment of essential regulatory aspects in specific safety assessments made by the industry in defending their safety cases [8, 15].

One particular point that becomes an important practical issue in all IDPSA developments is the consistency between the assumptions made in state-of-the-art static PSAs and its dynamic counterparts, in order to ensure that they do not contradict each other. The large industry and regulatory engineering effort that underlies the present use of DSA transient analysis codes and PSA FT/ET plant models, imposes as a strong requirement that the new methods and codes will add-to, but not replace the existing ones. A consistent link between IDPSA, DSA and PSA approaches is necessary to solve well recognized deficiencies of stand-alone or weakly linked present tools and methods.

Chapter 3. Mapping, information collection and identification of areas of certain interest

Severe accident management (SAM) in Nordic boiling water reactors (BWRs) relies on ex-vessel core debris coolability. In the case of core meltdown and vessel failure, melt is poured into a deep pool of water located under the reactor. The melt is expected to fragment, quench, and form a debris bed that is coolable by natural circulation of water. Success of the strategy is contingent upon melt release conditions from the vessel which determine (i) properties of the debris bed and thus if the bed is coolable or not, and (ii) potential for energetic interactions (steam explosion) between hot liquid melt and volatile coolant. Both non-coolable debris bed and steam explosion pose credible threats to containment integrity.

While conceptually simple, this strategy (i) involves extremely complex and often tightly coupled physical phenomena and processes, which are also (ii) sensitive to the conditions of transient accident scenarios. For instance, late recovery actions might affect core degradation and relocation processes, which can change formation of the in-vessel debris bed, reheating and re-melting of multi-component corium debris, thermo-mechanical interactions between melt and vessel structures and penetrations, vessel failure, melt release and jet fragmentation, debris solidification, energetic melt-coolant interactions, two-phase flow in porous media, spreading of debris in the pool, spreading of particulate debris bed, etc. (Figure 3.1). These phenomena have been a subject of extensive investigations in a large-scale research program on Melt-Structure-Water Interactions (MSWI) at the Royal Institute of Technology (KTH) over the past few decades.

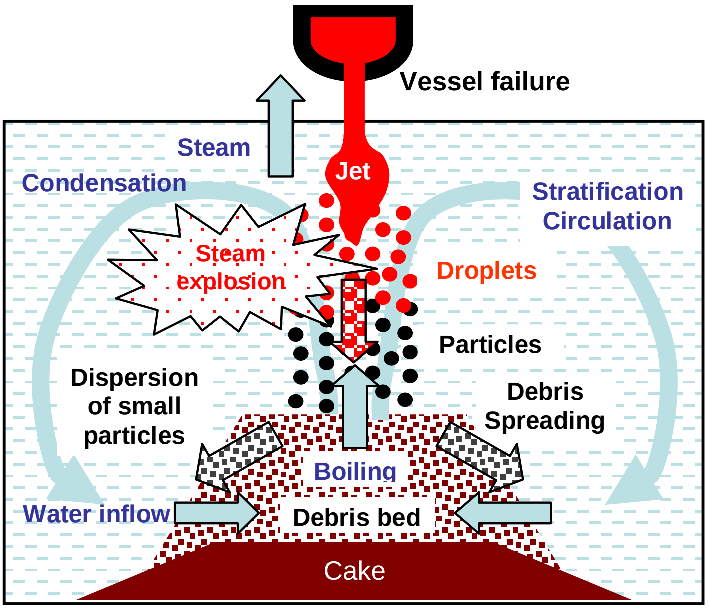


Figure 3.1. Severe accident phenomena in Nordic BWR.

While a significant progress has been made in understanding and predicting MSWI physical phenomena, complex interactions and feedbacks between (i) scenarios of accident progression, and (ii) phenomenological processes, have hampered a comprehensive assessment of SAM in the Nordic BWRs. Presently, the issues of ex-vessel debris coolability and steam explosion are considered as intractable by only probabilistic or only deterministic approaches.

Typical phenomenological stages of severe accident progression in Nordic BWR are shown in Figure 3.2.

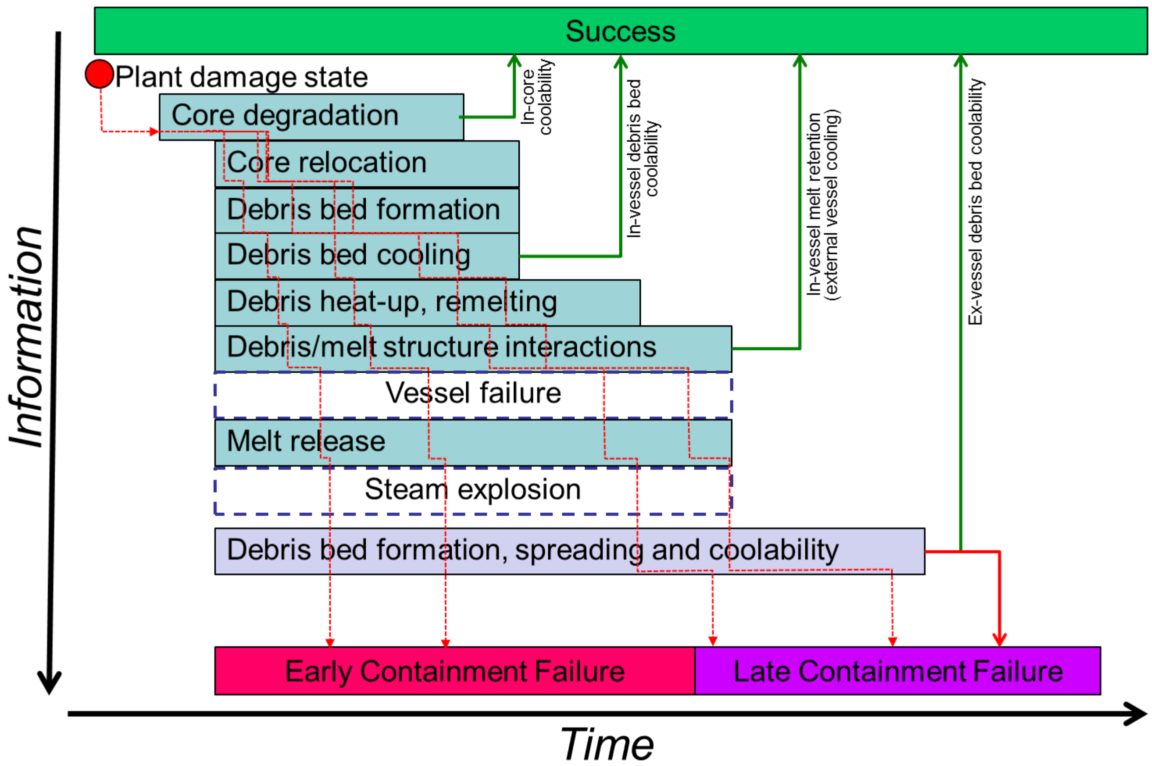


Figure 3.2. Severe accident progression in Nordic BWR.

The multistage path from the initial plant damage state to the containment threats is an important source of complexity and uncertainty. Phenomena and scenarios including operator actions are tightly coupled in their mutual interactions and eventual impact on the possibility of different containment failure modes. Conditions created at the earlier stages can significantly affect configurations and problem statements at later stages. For instance, if there is no activation of lower drywell flooding, then steam explosion risk is eliminated, but hot corium melt will attack cable penetrations in the containment floor leading to almost immediate containment failure.

Timing of transition between different stages is also important. Different time-dependent trajectories of the accident scenarios with the same logical sequence of the stages can result in different outcomes. For instance, decay heat is decreasing with time providing much better chances for coolability of the debris bed if melt is released from the vessel later [16]. However, if melt is released from the vessel later, it will have higher temperature, which could increase the risk of debris agglomeration [1719] hindering coolability of the debris bed [20], and creating a potential for an energetic steam explosion which can threaten containment integrity.

Combination of (at least) two threats (non-coolable debris and steam explosion) is another source of uncertainty. For instance, even a mild steam explosion might lead to degradation of debris bed cooling function, e.g. by destroying protective covers for cable penetrations in the containment floor and exposing them to hot debris, or by creating a leak of coolant from the lower drywell, or by activating filtered containment venting, releasing fraction of nitrogen which can potentially lead to drop of containment pressure below atmospheric level, etc.

Apparent major challenge for application of classical PSA and DSA approaches to Nordic BWR is complexity of tightly coupled transient phenomena and scenarios which limit effectiveness of heuristic approaches in a priori judgment about importance and impact of such coupling on the accident progression and outcome. Integrated deterministic-probabilistic approach is required in order to address these issues. Information about the initiating events and plant damage states is necessary input information for the IDPSA analysis and it can be provided from the PSA-L1. In this section we provide an overview of PSA-L1 results which are used in this work.

Timing of events such as failure and recovery of safety systems determines in-vessel accident progression, core relocation process and properties of the debris in the lower head. The properties and configuration of the debris determine initial conditions for corium-structure interactions, vessel failure and melt release conditions. Therefore core degradation and relocation scenarios have significant impact on the ex-vessel accident progression and risks. For addressing the effect of timing of the events on the in-vessel and ex-vessel accident progression a set of initial plant damage states and possible further failures and recovery actions has to be provided. Such information is available from the PSA-L1. In the following section a discussion of the basic information from PSA-L1 which is necessary for IDPSA analysis and identification of specific topics of interest are provided.

3.1 Assumptions and Limitations

Probabilistic Safety Assessment (PSA) is used to systematically identify, evaluate and rank the sequence of events that can lead to core damage and radioactive release to the environment. Identification and hence opportunities for improvement in risk dominant feature of the facility is one of the overall objectives. The analysis is probabilistic, i.e. it is based on probability and reliability calculations and the result is an estimate of the frequency of detected events.

Some key assumptions and limitations in PSA L1 analysis are:

- Implemented deterministic analyses are correct.
- Blow-down paths and building structures can withstand emerging loads at rupture.
- Studied transient time is normally 1 day, i.e., objective function is required during this time (Level 1 analysis includes 24 hours from initiating event, sequences that have not led to the core overheating within this time are not considered as core damage sequences and excluded from Level 2 analysis).
- Aggravating manual interventions are not considered.
- Restricted modeling of manual interventions during transients (only when clear instructions are provided and there is sufficient time available).
- System requirements should be established either via thermal-hydraulic calculations or through references in the SAR.

3.2 Phases during severe accidents

The first phase of an accident is studied in PSA L1 and the result is a number of sequences ending with either success or core damage.

For those sequences ending with core damage the following accident progression is studied in PSA L2. The accident progression is normally divided in:

- **In-vessel** – Describes the heatup and meltdown of the core
- **Vessel melt through**– Describes the phenomena occurring at vessel melt through
- **Ex-vessel** – Describes the long term progression of the plant after melt through

There are interesting phenomena to study with deterministic methods both in PSA L1 and in the different phases of PSA L2.

3.2.1 Core Damage States in PSA L1

The simplest form of core damage states in PSA L1 is to just differ between core damage and success. Normally the core damage states are separated into different categories with respect to the cause of the core damage. Possible reasons to core damage can be:

- HS1: Failure to shut down the reactor.
- HS2: Failure to make up water to the reactor.
- HS3: Loss of residual heat removal.
- HS4: Overpressure of the primary system.
- Overpressure of the containment

Typically loss of core cooling or failure of residual heat removal give the major contributions to core damage, but this varies from plant to plant.

Failure to shut down the reactor normally gives a low contribution to the total core damage frequency. Reactivity control is a very complex process to model since an incomplete or delayed shutdown puts higher demands on the other functions such as higher demands for core cooling, increasing pressure in the primary system etc. It may therefore be interesting to study this in more detail since the core damage frequency due to failure of shutdown may be underestimated in the existing PSA studies. This is further described in section 3.3.1 below.

3.2.2 Plant Damage States Classification in PSA L2

In PSA L1 for Nordic BWR reference plant design the core damage states are grouped into 4 categories: HS1 (ATWS), HS2 (Loss of core cooling), HS3 (Failure to remove decay heat) and HS4 (Primary system overpressure). The categories (HS1, HS2, HS4) correspond to early core damage scenarios, HS3 – corresponds to late core damage.

In addressing ex-vessel behavior and consequences the following physical phenomena can challenge containment integrity: direct containment heating (DCH), ex-vessel steam explosions (EVE) and ex-vessel debris coolability (DECO).

A quantitative perspective on these matters should be derived from the Level 1 PSA. DCH scenario corresponds to high pressure (HP) accident scenario, steam explosion in the containment (EVE) corresponds to low pressure (LP) scenario and, finally, both consequences will lead to large amounts of core debris relocated to the lower drywell and it can challenge lower drywell floor and penetrations integrity, so the question of ex-vessel debris bed coolability is an all-pervasive issue.

Initial conditions and correspondent frequencies that will lead to different core degradation, in-vessel debris bed formation, vessel failure scenarios can be identified from PSA L1 data.

The core damage sequences, thus, can be grouped together based on the aforementioned challenges to the containment integrity as follows:

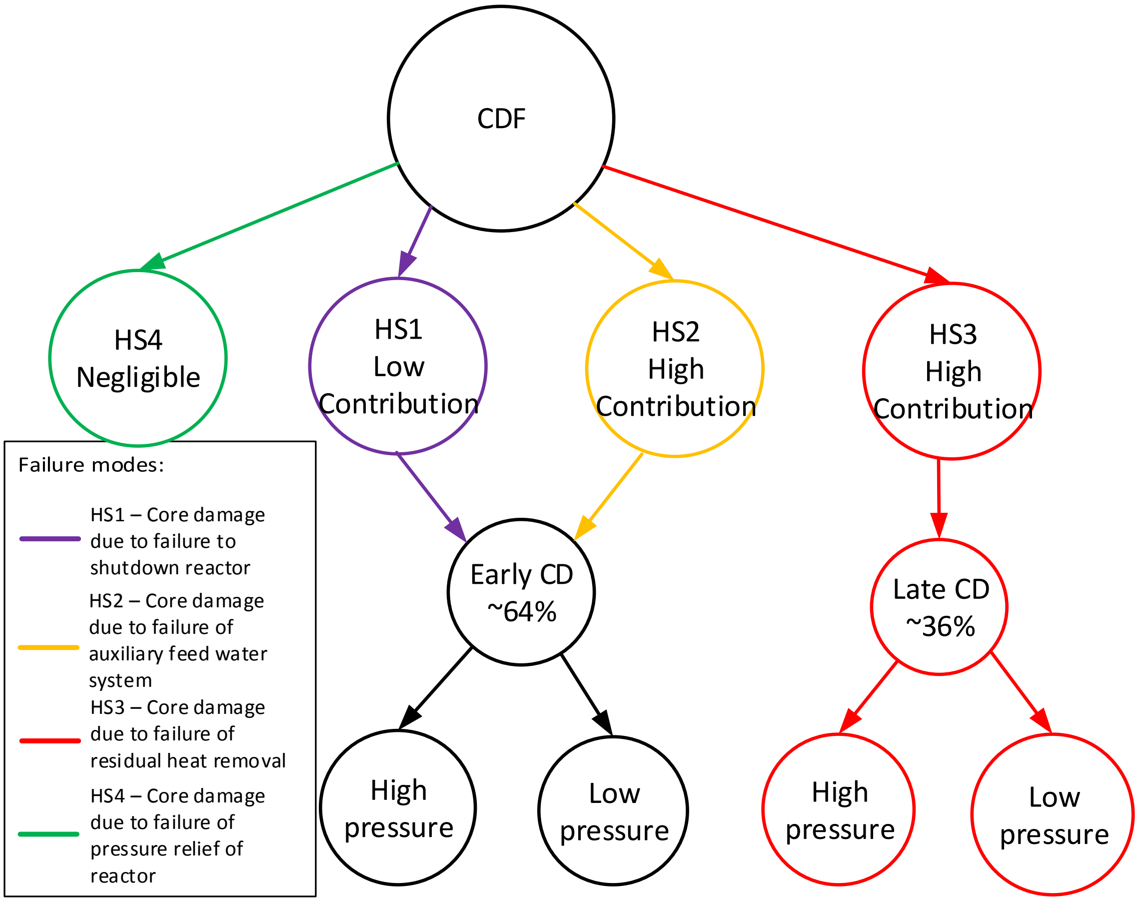


Figure 3.3. Core Damage States Classification

The phenomena during core meltdown and vessel melt through belong to PSA L2. The link between PSA L1 and L2 is the plant damage states. The plant damage states describe not only the core damage state but also the conditions in the primary system and the containment. This is further described in section 3.3.2.

3.2.3 Phenomena at vessel melt through

One of the most important phenomena at reactor vessel melt through is steam explosion when molten corium enters the water pool below the reactor vessel. This is further described in section 3.3.3.

3.3 Interesting areas to study

3.3.1 Reactivity control

An area of interest in PSA is reactor shutdown. The modelling of the reactor shutdown is normally very crude (claimed conservative) in PSAs.

There have been some events relating to the reactor pressure vessel over a period of time, e.g. the reactor instability event 1998 and the HTG event in 2003, both for Oskarshamn 3. Even though there is not a link between the events and the reactor shutdown as analysed in PSA – it was considered that the complexity in the shutdown could be of interest for IDPSA.

The current modelling for the reactor shutdown could be as follows (also illustrated by the figure below):

- If the reactor scram is working properly (there are not two adjacent control rods failing after scram), then the reactor is considered to be in a safe state.
- If the reactor scram fails, then both electrical insertion and auto-boron is considered as alternatives.
- If more than 4 adjacent control rods fail to be inserted by the hydraulic scram it is necessary with automatic limitation of the main circulation flow.
- Auto-boron is considered as a completely diversified way to shut down the reactor. The boron injection system can be used for all initiating events except large and medium pipe breaks. (In a large/medium LOCA case there is a continuous loss of water from the RPV and the boron is diluted.) Automatic start of boron injection is only credited if more than 5 adjacent control rods fail. In other cases only manual start is credited.
- To be able to take credit for the boron injection system the main circulation pumps must be running at minimum speed. Otherwise the boron mixing in the RPV is insufficient. This is a conservative assumption. To be able to use the boron injection system it is also necessary that the normal pressure relief system works. If the water blowing valves open the situation is similar to a LOCA and there is a risk for boron dilution.
- It takes a long time to shut down the reactor with the boron injection system (compared to SCRAM or CRD). The demands on the core cooling and residual heat removal systems are therefore higher during this period. In the PSA model however, there is a higher demand on these system for the whole sequence.

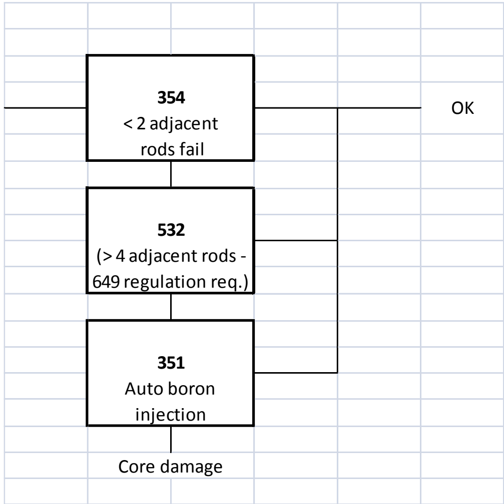


Figure 3.4. Shut down of NPP in PSA

The requirements on the other functions, e.g. core cooling, pressure control and residual heat removal from the containment – is different in different potential scenarios. However, the approach in the PSA is to be reasonably conservative to avoid modelling of too many scenarios and therefore the representation is simplified – but should be conservative.

A simplified modelling approach is fully satisfactory if it is conservative and the core damage frequency is well below the target value (<1E-8).

An analysis of reactivity control could aims at:

- Investigate if there are some parameters in the shutdown process that should be dealt with in a refined manner in the PSA, e.g. circulation pumps running or not, in configuration with SCRAM, electrical shutdown or auto-boron.
- Investigate if some parts of the shutdown model in the PSA should benefit from an increased level of detail.

The proposed sequence for further study was:

- A transient (not loss of offsite power since this means that the main circulation pumps stop) with complete or partial failure to shutdown with the hydraulic scram (system 354).

The following results should be studied in the proposed sequence:

- How many control rods are allowed to fail before automatic limitation of the main circulation flow is necessary?
- Is it necessary that the main circulation pumps are running at minimum speed for boron mixing?
- What are the requirements on the core cooling and residual heat removal systems in case of shutdown of the reactor with only the boron injection system available? (i.e. complete failure of scram and control rod drive system)

At present it was decided not to continue with the study of reactivity control.

3.3.2 Link between Level 1 and Level 2 PSA

In a standard PSA, the output of PSA Level 1 is typically core damage (possibly separated in a few sub-categories). These core damage sequences are then divided into a number of sub-categories based on attributes, which shall be representing the important features for the Level 2 progression.

There is normally around 20-40 Plant Damage States (PDS) defined in the interface between Level 1 and 2. This interface is therefore reasonably crude.

For the plant studied there are 27 PDSs for power operation and low power operating modes. The attributes that are considered relevant to characterize the core melt for the continued process are:

- Core damage state (failure of shutdown, core cooling or residual heat removal)
- Initiating event (Transient or LOCA)
- Time point of the core melt (early, late)
- Reactor pressure (low, high)
- Containment atmosphere (inert, air)
- Can cooling with containment spray system be taken into account (Failed, Yes)?
- Activated containment pressure relief, 361 (activated, not activated)
- Activated filtered release, 362 (activated, not yet activated, failed)
- Bypass of containment (bypass, intact)
- Warm suppression pool (warm if pool cooling fails, else cool)

The events that are represented in a PSA Level 2 are the events that change the conditions for retaining of releases within the RPV or within the containment. Hence, if the coolability in the RPV is different in different scenarios – then this is vital information. If the sequences are affecting the phenomena that can occur, then this is also vital information.

For each of the PDS there is a containment event tree. The containment event tree (CET) defines the accident progression as analyzed in the PSA. The sequences in the CET end at the release categories (RC), and there are normally around 15-40 of such. The RCs can be defined in different ways, for example release size or defined by type of sequence. The normal approach is to use "by type of sequence", because then only a limited amount of verifying deterministic calculations are considered to be required. For the "by type of sequence" approach the characterization is based on, for example;

- Release path (containment bypass, containment rupture, filtered release, leakage)
- Timing of release (early, late)
- Initiator (pipe rupture, transient)
- Sprinkling of containment established (yes/no)

The feasibility study is aiming at studying, in a greater level of detail, the attributes that are of interest for the core relocation – and further on melt through of the reactor pressure vessel, RPV, and the following effects on phenomena.

The type of phenomena that are usually accounted for in a PSA are:

- Re-criticality (in the core, in lower plenum, in containment)
- Hydrogen burn (deflagration and detonation)
- In-vessel steam explosion
- Ex-vessel steam explosion
- Direct containment heating
- Rocket mode
- Melt concrete interaction (basemat penetration)
- Steam generator tube rupture (only for PWR)

The effect of the phenomena can be:

- Containment rupture
- Different type of bypass
- Activation of filter

The consequence most focused on is of course containment rupture.

From the initial sequences in the PSA Level 1, all events that are leading to a certain PDS are then treated in the same manner in the continued sequence (however, dependencies are treated logically correct if the failure should affect systems in PSA Level 2). It is however obvious that it will be different scenarios from a deterministic stand point if there is an initial loss of offsite power and no start of the diesels, compared to a scenario where the diesels would stop after some hours.

The purpose with the improved integrated link between the PSA and deterministic analyses is hence to be able to judge if, for example, these scenarios need to be treated differently in the PSA context.

The approach chosen was to identify some sequences from the PSA Level 1 and to use the DPSA methodology to evaluate the progress of these sequences. The first phase of the project shall focus on the mechanisms for core relocation, since this is an important factor for the continued sequence. By identifying the mechanisms that contribute to the core relocation, the analysis is also expected to provide information on efficient measures, non-efficient measure or contra-productive measures to avoid core relocation. Eventually, it is expected that the IDPSA integration will give a clearer answers to:

- Can the core melt process/relocation be stopped and how?
- Are there actions that must not be taken during a sequence?
- How will the conditions for phenomena be affected, and thereby give guidance on how to treat these in the PSA?
- Based on the above information, it will also feedback requirements on definition of plant damage states.

The PSA scenarios chosen for analysis of recovery were:

- High pressure loss of core cooling scenarios

- Recovery of core cooling in different combinations
    - Different time points for depressurization of the RPV
    - Different time points for start of the ECCS
    - Different mass flows from the ECCS

3.3.3 Steam explosion case study

VTT's contribution to DPSA methodology study was implemented by a steam explosion case study dealing with Olkiluoto NPP and its units 1&2. Full report has been published as a VTT report [21]. The methodological approach was Integrated Deterministic and Probabilistic Safety Assessment (IDPSA), and the weight was on the probabilistic side of analysis. Ex-vessel steam explosion can occur when the core melt is released from the reactor vessel into a pool of water. This is the case for Nordic type BWR designs with lower drywell filled by water as a severe accident management measure to provide ex-vessel coolability of corium debris.

Steam explosions are an excellent severe accident phenomenon for developing and exploring the capabilities of IDPSA methods, because of relatively significant phenomenological uncertainty and intractability of steam explosion phenomenology for a standalone PSA approach. In general, IDPSA approach suits well to scenarios where timing and deterministic response of the plant play a crucial role. Occurrence and consequences of a steam explosion are highly dependent on preceding accident progression, especially on core degradation processes along with vessel failure mode and timing. Thus important information can be obtained by using both integral and dedicated deterministic codes, and this information can be refined and used for assessment of inherent uncertainties to achieve a sufficient depiction of the event for Level 2 PSA purposes.

Figure 3.5 is a simple diagram of how a steam explosion modelling process for Level 2 PSA purposes might proceed and which tools can be used to support it. One can use MELCOR or some similar tool to obtain e.g. the amount of core material involved in explosion and also values for some other explosion-relevant input parameters. MELCOR simulations are useful for the Level 2 PSA modelling phase as well. Also results from Level 1 PSA analysis can prove to be useful in this phase, and FinPSA is an example of a tool used for that purpose. The analysis tool for Fuel-Coolant Interaction (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 PSA analysis is ultimately up to the analyst and the scope and objective of the study. SPSA software was used in this study for Level 2 PSA analyses at VTT. There exist codes for structural analysis of containment integrity as well, but at this phase they were not used.

In the case study performed at VTT, only the first and the last box in Figure 3.5 were focused on and no FCI or structural analysis tools were used. Deterministic accident progression simulations were performed with MELCOR and the knowledge obtained from them was then implemented into a probabilistic containment event tree model of the plant constructed with SPSA. Ex-vessel steam explosions and source term were modeled more rigorously than many other parts of the model. Results from the analysis are expressed mainly in terms of radionuclide releases to the environment, but instead of accurate numerical values, the successful illustration of the methodological approach is of higher importance.

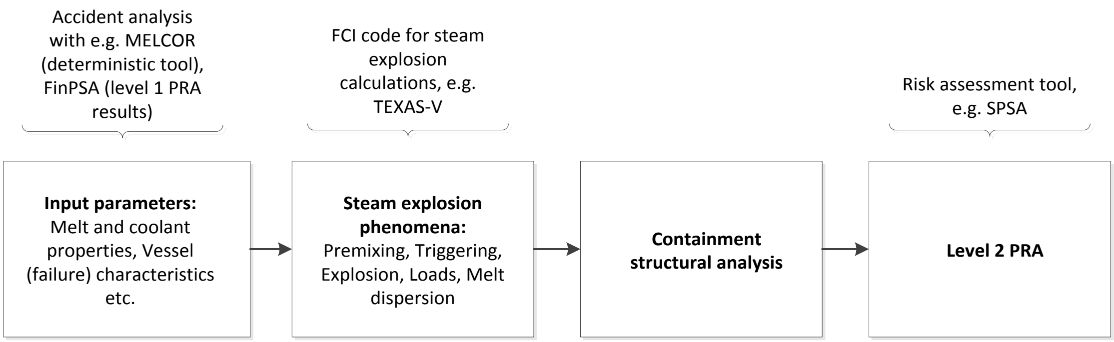


Figure 3.5. An example of phases of steam explosion analysis and its integration into a Level 2 PSA. Several computational tools can be used to support the analysis.

Chapter 4. **Results of deterministic analyses of identified areas**

4.1 Application of GA-IDPSA methods for Analysis of Nordic BWR Severe Accident Scenarios

4.1.1 Introduction

A station black-out transient with varying degrees of safety system recovery is addressed in this work using MELCOR analysis. Special attention was directed towards the melt mass and composition in the lower head. This information is a starting point for detailed studies of the vessel breaching and melt relocation to the containment. Melt relocation mode is determinative factor for the ex-vessel risk (such as debris bed coolability, steam explosion, etc.) for the containment integrity of the plant.

4.1.2 Description of the simulated Transient

In the cases considered, we assume the following:

- **Complete station blackout**
  This includes battery-powered half-passive systems like the ADS-Valves (System 314) and the Water-Valves (System VX105). The realism of this assumption is open to discussion, but should not be considered entirely out of the realm of possibility considering the Forsmark 2006 Incident[16], in which an overvoltage incident in the 400 kV switchyard caused the failure of 2 (out of 4) of the so called Uninterruptable Power Supply (UPS), including their 220 V batteries. The failure of all 4 was a distinct possibility.

- **Overpressure protection system still active**
  The spring-operated part of the overpressure protection system is still operating to specifications and will open valves stepwise, starting at slightly above 70 bar and opening completely at 75 bar, to discharge steam to the containment and protect the Reactor Pressure Vessel (RPV) from explosive failure.

- **Failure of the de-pressurization Systems for the RPV(Systems 314 and VX105) with eventual recovery**
  The depressurization valves, while primarily steam powered, depend on battery power to divert a steam flow into opening the valves, and are therefore regarded inoperable for a varying amount of time.

  Full function will be recovered after a time-period imposed by the GA-NPO software [10]. The exact control logic activated upon power recovery will depend on the iteration of the input file.

- **Complete failure of the Auxiliary Feedwater System (System 327)**
  The high pressure injection system activates upon low water level in the downcomer ($L \leq 3.3$ m over top of the core) and could deliver 2 times 45 kg/s of coolant, independent on the internal pressure of the reactor vessel. The coolant will be supplied until the water level rises over 4.6 m over the active part of the core, when the system will turn itself off.

  In the current transient the system is considered non-functional and will not come back on-line, even when power is restored to other systems.

- **Complete failure of various not accredited Safety Systems**
  System like the Control Rod Guide Tube (CRGT) Cooling or the Residual Heat Removal (RHR) Systems, which might be employed in the typical Nordic BWR are considered non-functional in the current transient and will not come back on-line, even when power is restored to other systems.

- **Failure of the Emergency Core Cooling System (System 323) with eventual partial recovery**
  The Emergency Core Cooling System (ECCS) discharges to the downcomer with a theoretical maximal mass flow of four times 366 kg/s. In the current implementation the ECCS activates upon low pressure difference between the water source (wet well pool) and the RPV. Mass flow begins at pressure difference of 12.5 bars and will reach its maximal value at a difference of 2 bars. ECCS is power dependent and therefore regarded inoperable for a varying amount of time.

  Function will be recovered after a time-period imposed by the GA-NPO Algorithm. Compound maximal mass flow will vary between 20 and 50 kg/s (the equivalent area of ECCS is from 0.008 to 0.02 m^2) for most iterations of the input file and will be another variable imposed by the GA-NPO Algorithm.

The control logic of the various active safety systems have been subjected to a continuous development and will be explained in detail for each considered iteration of the input file.

We employ the GA-NPO software to search the space of scenario parameters (such as timing of recovery actions) to find combinations which provide desired values of fitness function (see: 4.1.5). For the current simulation, we use the following parameter set:

1. Activation time delay of the depressurization of the Reactor Pressure Vessel (affected: Systems 314 and VX105) from 0 to 15 000 seconds
2. Activation time delay of the low pressure coolant injection (affected: System 323) from 0 to 15 000 seconds
3. Maximal possible mass flow delivered by System 323, when working, between 20 and 50 kg/s

4.1.3 The MELCOR Input

The MELCOR Input file was configured to simulate a typical Nordic BWR. The current model is of a BWR with a thermal power output of 3900 MW thermal. We assume 700 fuel assemblies, all of type SVEA-96 Optima2 with an active length of 3.68 meters and Uranium Dioxide density of 10460 kg/m^3.

The original input file is split in 14 cells in the z-direction, 1 to5 for the lower plenum, 6 and 7 for the core support plate and the inactive inlet zone, 8 to 13 for the active core zone and 14 for the core exit below the core grid. In radial direction the input file has 5 cells for an inner radius of 2.4075 meters for the RPV. The outermost radius also contains the downcomer.

It employs the 'ORIGEN' data set for decay heat power, which assumes a 3578 MW thermal General Electric BWR, 5 types of assemblies with an initial Uranium-235 enrichment from 2.66 to 2.86 percent and a burn-up of 3 to 4 years [5, 16].

4.1.4 Data Post-Processing and Automatic Time Step Control

We employ the GA-NPO algorithm to search the parameter space for either a minimum or a maximum of the fitness function (see: 4.1.5). The value of the fitness function can be any parameter or combination of parameters, which is producible using a Post Processing-Script, usually a bash-file, and which is available in the MELCOR Plot File. The result has to be a single value which is written into an external goal function, where it is read by GA-NPO and used to select the parameters of the next simulation.

MELCOR adjusts the time step of the simulation dynamically to reach convergence of the simulation. There is a limiting ratio between two consecutive time steps, resulting in a maximal allowable change between two time steps. The user input on this feature is essentially limited on the largest and smallest allowable time step. When several discontinuities happen in a relatively short time frame and the code "crashes" at this critical point in time, the times step can often not be automatically adjusted downward fast enough resulting in a premature crash of the simulation.

GA-NPO will automatically assign the parameter combination leading to a crash a very large negative value, causing future parameter selections to drift away from the "crash side". This is on the one hand logical as we want complete simulations for our analysis but can lead to a distortion of results.

To fix this problem the secondary purpose of the Post Processing-Script was introduced. In case of a crash, e.g. the actual end time is significantly smaller than the expected end time, the Post Processing-Script falls back to the closest MELCOR safe point, which are regularly distributed over the length of the simulation run, and restarts the MELCOR simulation from this point with a dynamically adjusted maximal time step. If the simulation fails again, the procedure is repeated with ever-smaller maximal time step. GA-NPO is paused during this process.

If successful the external goal function is computed as usual and the simulation is continued.

4.1.5 Fitness Function Selection

For the time being the work-horse of the GA-NPO software suite is the 1.8.5 MELCOR code. While further development in later versions of the MELCOR –code are underway, work will continue with the current 1.8.5 MELCOR version with which the proof of concept and the first practical experiments were conducted, part of the reasoning are legacy issues. Transformation of older input decks into newer versions are notoriously time intensive and error prone. Another part are lingering issues with 2.x MELCOR, which has been a great conceptual leap forward, but has been plagued with ongoing, non-obvious error concerns common to many software projects.

The fitness function can be chosen as any plot-able MELCOR variable or any combination of plot-able MELCOR variables that can be implemented using a bash script or any programming language (e.g. perl, AWK, or similar) installed on the Linux machine and callable from a bash script. The bash script is called the post-processing script and, if activated, will be called after the MELCOR run has been completed. It has to write its intended results in an external text-file, called the external goal function (or ext.goal) in the current implementation. The name of the post-processing script and the external goal function have to be specified in the GA-NPO input file config.npo.mel.
Note that the criterion specified need not be identical to the input used to generate the external goal function. The former will override the later.

In the current implementation we use aptplot-script aptbatch to read out data vectors from the MELCOR-Outputfiles, called MELC.PTF. The data vectors are than further processed by a combination of bash programming and AWK, an interpreted programming language designed for text processing, typically used for data extraction, and reporting.

The current fitness function is simply returning the absolute difference between the relocated mass to the lower plenum, defined as all cells below the lower core plate, and a user defined constant. This is useful, if we want to avoid unwanted concentration of the algorithm on the safe zone and are more interested in sounding out the critical edges of the “target area”.

The external goal function is rather flexible and allows the superposition of various values, as long as we can combine it in final return value, weighted to user specifications, for the fitness function, that will either be maximized or minimized.

For the current crop of simulations, we are interested in the coolability of debris in the lower plenum, depending on the reactivation time of various safety systems. For this reason we selected a relocation mass of 10 tons, which is about the largest amount of debris which various specialized simulations yielded as still coolable.

4.2 GA-NPO Results

Figure below represents the summary of the results obtained with GA-NPO [10] MELCOR analysis of core relocation phenomena (see Appendix A for more details).

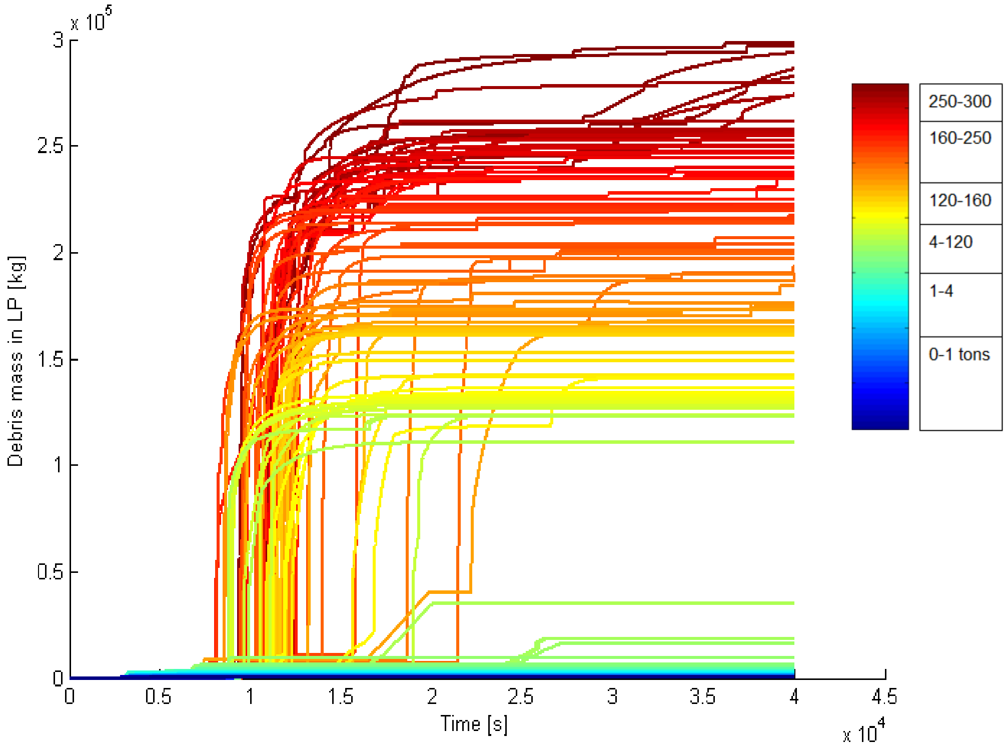


Figure 4.1. Iteration-3; Relocation Mass over Time

Maximal relocation mass in the above plots is a bit less than 301 tons. The activation of the ECC-System is usually a better indicator of the amount of core damage than the ADS-Activation. This is unsurprising as the ECCS-Activation will always occur after the ADS-Activation and provides the first influx of new coolant. Large time delays between depressurization and new coolant influx will nearly always produce large relocation masses.

It should be noted that the purpose of these calculations was to focus on core relocation phenomena and on finding the maximum relocation debris mass, so the vessel breach condition was bypassed and there was no vessel breach and melt ejection in any case.

GA-NPO is self-learning and should "discover" this effect. The amount of points with differentiating position between the x-z- and y-z-projection of a scatterplot can be taken as unconsidered influences (inaccurate modeling or unknown parameters, influencing the outcome) superimposed on the learning speed of the algorithm.

The methodology is capable of finding boundaries of domains with small or large debris relocation. Result from the latest analysis iteration (Figure 4.10) shows that if recovery times of ADS and ECCS are kept under ca. 5000 s, total mass of relocated debris to the lower plenum will be smaller than ca. 20 tons, which is very likely coolable. If ADS and ECCS recover at later than ca. 10000s, the amount of debris relocated to the lower plenum will be very large (> 200 tons) which will be more difficult for in-vessel coolability and retention. Further analysis is needed to verify exact boundaries of these domains.

4.2.1 Analysis of the Results

We can notice a certain set of trends analyzing the above scatterplots:

1. There is a linear connection between ADS-Opening time and the earliest possible ECCS mass flow. This is not particularly surprising considering that the ECCS mass flow is pressure dependent and there will be no pressure relief until the ADS opens.
2. The further we progress along this line into the direction of greater time-values, the more dominating become the large relocations to the lower plenum. This is caused by the constant steam discharges from the pressure protection valves and the gradual heat-up of the system that will eventually lead to core uncovery and damage setting in, even when the ECCS is activated as quickly as possible.
3. Moving along the ΔT-ECCS axis there will generally be an area in which large discharge mass flows from the ECCS can compensate for the later activation of the ECCS, but with limitation. Early activation of the ADS-depressurization gives additional breathing room.
4. The third scatterplot, and as far as we can notice also the first, has generally less “Data noise” than the second, in the sense that they offer a much more clear partition in “target area” and “non-target-area” in the parameter space.
5. There is a noticeable step function in the amount of debris mass relocated to the lower plenum. Relocations tend to be either between 0 and 10 tons or 120 tons and above.

The last phenomena especially is deserving of closer study. Producing plots of the actual debris mass will show that the accumulated core damage is nearly the same for small and large relocations.

Yet, in the case of small relocation nearly all the debris is located above the lower core plate at the end of the transient. We can compare total debris mass in the core and lower plenum region for small and large relocations in Figure 4.2 and Figure B.6 (see Appendix B for more details), respectively.

The reader should be aware that the lower core plate is located at level 9, in the following graphs. The lower plenum extends from level 14 to 10. For small relocations we can recognize the retention of debris mass over the core plate with icicle like structures hanging from the core plate.

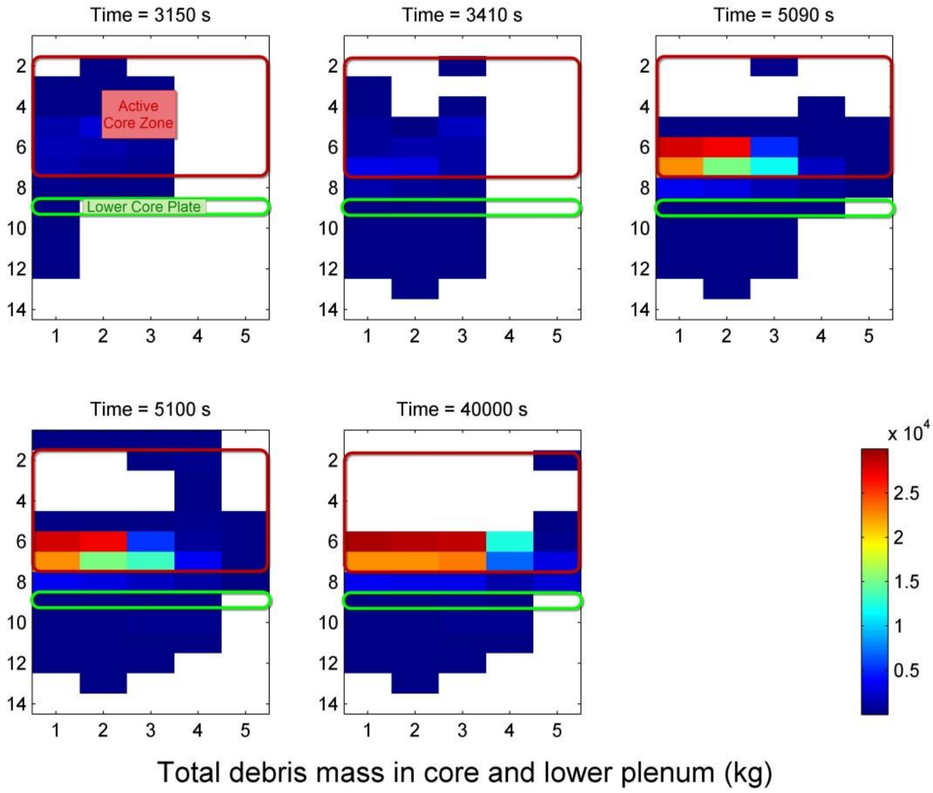


Figure 4.2. Small relocation to the lower plenum

4.2.2 Clustering and classification analysis of the results

Clustering and classification approach overview

One of the major issues in using dynamic methodologies is a large number of scenarios that can be produced by a single initiating event. For decision making, it is often insufficient to merely calculate a quantitative value for the risk and its associated uncertainties, the extraction of useful information which can be appreciated and handled by a decision maker is a challenge. The development of risk insights that can improve system safety and performance requires the interpretation of scenario evolutions and the principal characteristics of the events that contribute to the risk.

The approach used to resolve this problem is based on decision tree built using clustering results data, to explain cluster structure attending to the values of uncertain parameters.

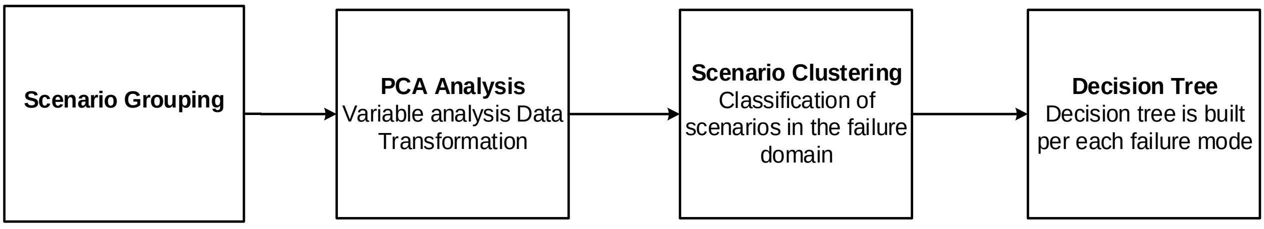


Figure 4.3.Grouping and Classification approach

The main steps of this approach are briefly explained below. Firstly, the scenario grouping is performed. The main idea of this step is to focus the analysis on the sequences intractable in classical PSA. Thus scenarios where the order and timing of events are not important are grouped first an excluded from further considerations as directly amenable to PSA analysis.

Next, a Principal Component Analysis (PCA) is carried out. PCA is a technique for revealing the relationships between variables in a data set by identifying and quantifying a group of principal components which have the largest influence on the system response [7]. Then, based on the PCA results the clustering analysis is performed using Adaptive Mesh Refinement (AMR) method. In the final step a decision tree is built for each failure mode using clustering results data [8]. Decision tree is used for data representation that explains failure domain-cluster structure. Decision tree classification algorithm performs orthogonal partitioning of the search space using data impurity measure as a splitting criterion [7, 9]. The main purpose of decision tree application is to present data in easy to interpret and transparent way to a decision maker to support decision making process. Finally, information of the leaf nodes is used for failure domain probability calculation.

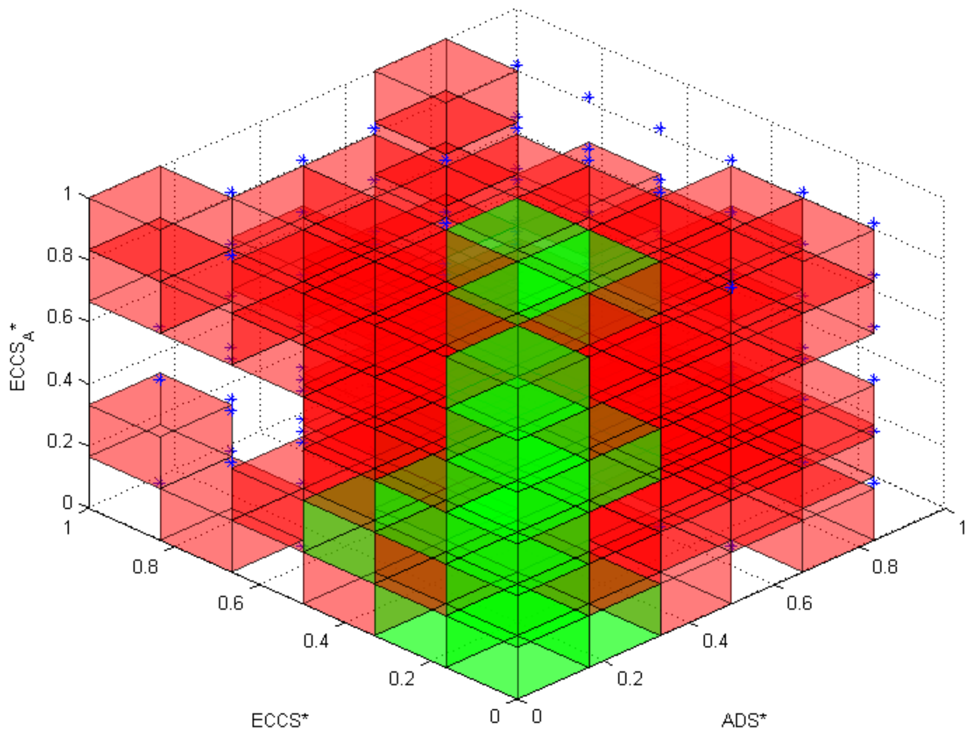


Figure 4.4. Failure (red) and safe (green) domain for the sequence [ADS ECCS] with flow area $ECCS_A$, initial step = 1/6. Blue points - scenarios to evaluate.

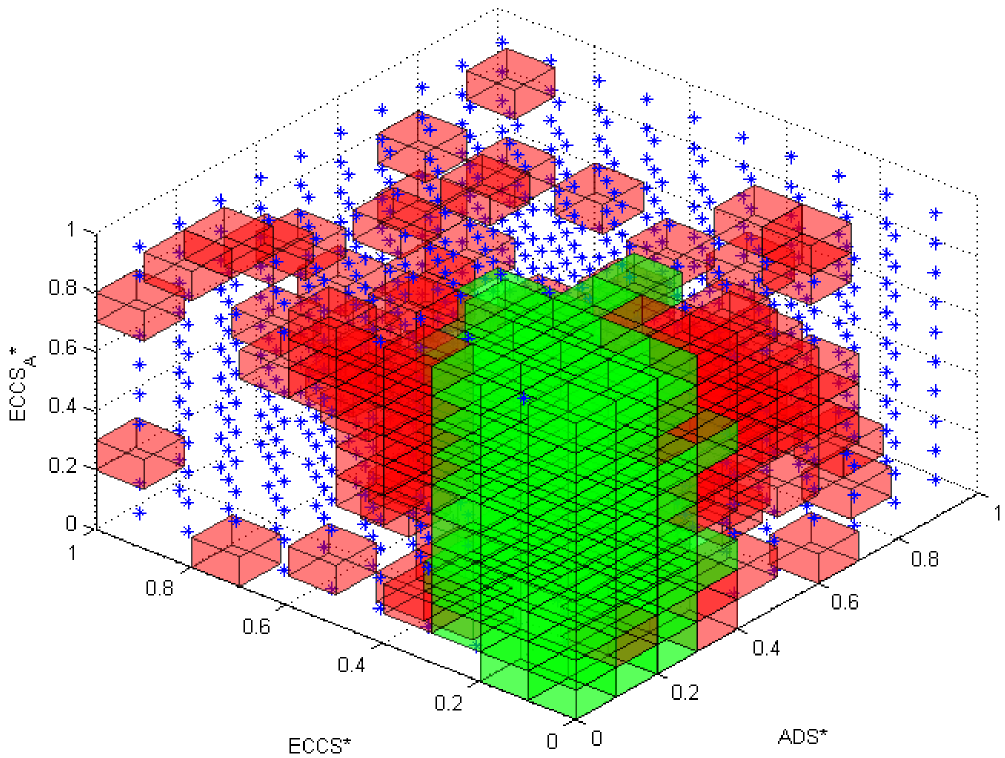


Figure 4.5. Failure (red) and safe (green) domain for the sequence [ADS ECCS] with flow area $ECCS_A$, initial step = 0.1. Blue points - scenarios to evaluate.

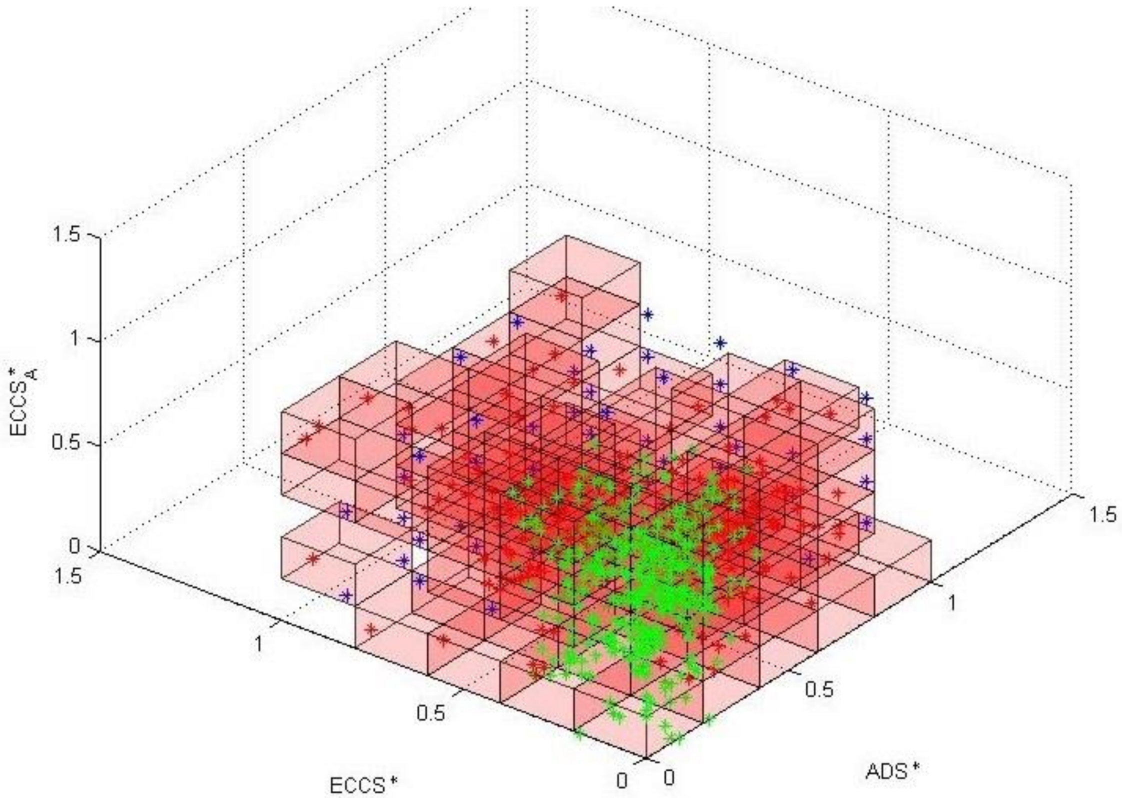


Figure 4.6. Failure domain for the sequence [ADS ECCS] with flow area $ECCS_A$, initial grid step = 0.2, no refinement; with failure scenarios – red; safe scenarios – green;scenarios to evaluate – blue points.

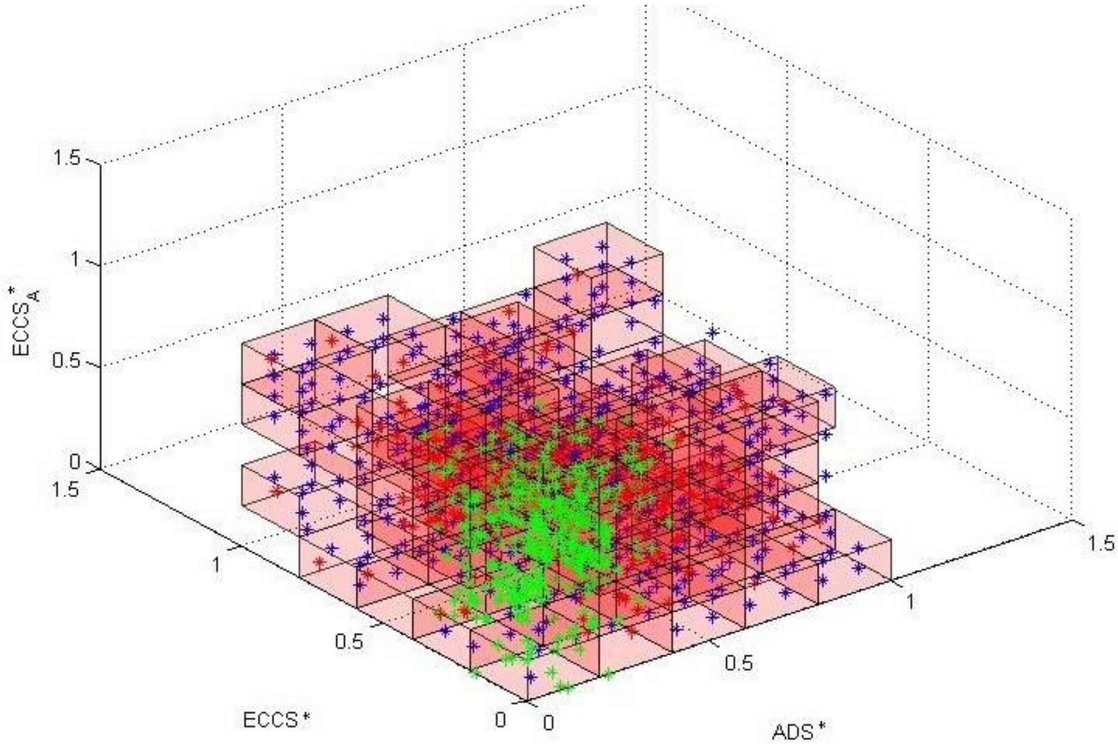


Figure 4.7. Failure domain for the sequence [ADS ECCS] with flow area $ECCS_A$, initial grid step = 0.2, 1st step refinement; with failure scenarios – red; safe scenarios –green; scenarios to evaluate – blue points**.**

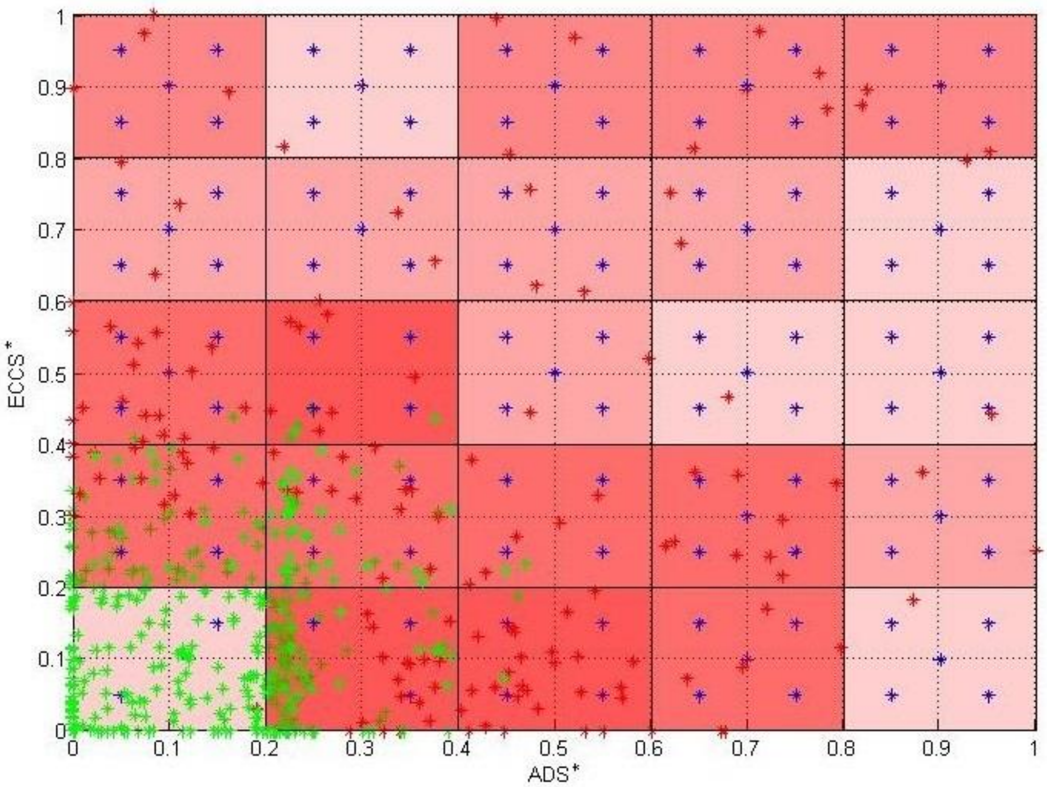


Figure 4.8. Failure domain for the sequence [ADS ECCS] with flow area $ECCS_A$, initial grid step = 0.2, 1st step refinement; with failure scenarios – red; safe scenarios –green; scenarios to evaluate – blue points. (ADS-ECCS axes)

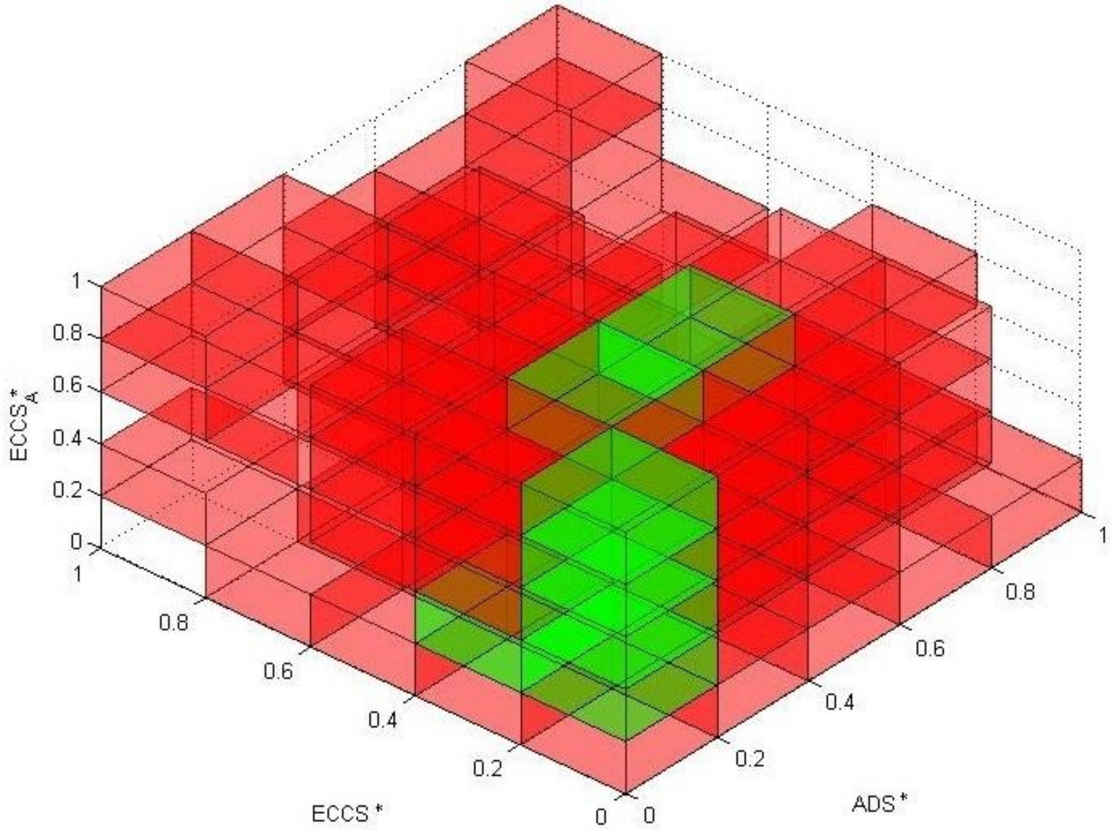


Figure 4.9. Failure domain for the sequence [ADS ECCS] with flow area $ECCS_A$, initial grid step = 0.2, 1st step refinement.

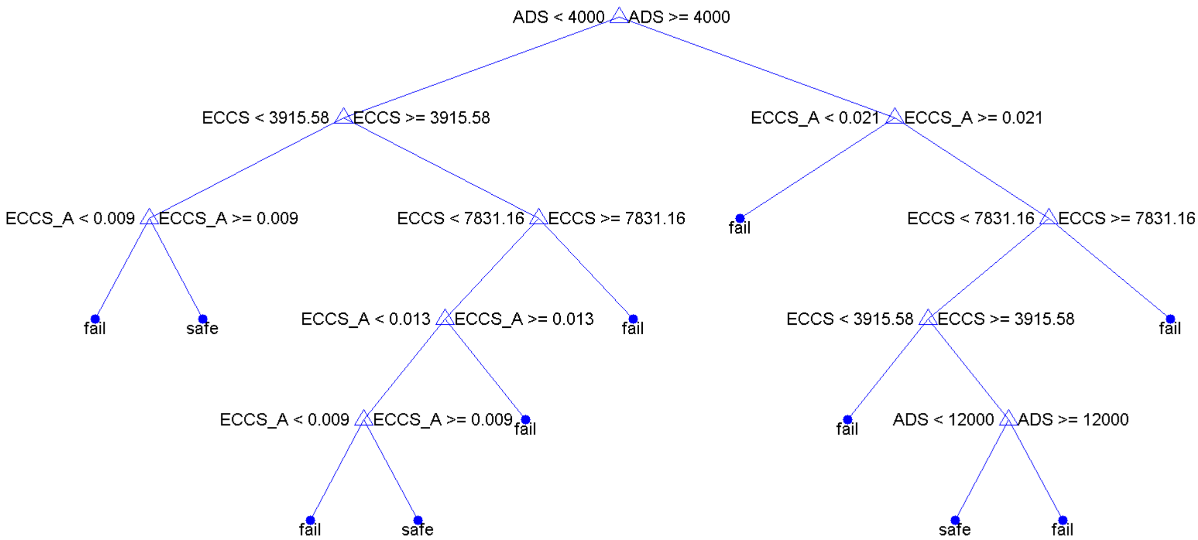


Figure 4.10. Decision tree for the sequence [ADS ECCS] with flow area $ECCS_A$, initial grid step = 0.2, no refinement.

Analyzing clustering results and decision trees built for the sequence [ADS, ECCS] with flow area $ECCS_A$, we can conclude that the safety domain (relocation below 10000kg) is majorly located in the corner with time delay for ADS and ECCS less than 4000 sec and ECCS flow area > 0.01. Higher ECCS flow area may increase limits of time delays for both ADS and ECCS that will not cause massive relocation.

Decision tree also indicates that there are might be an effect of high water injection rate to damaged core, and it may cause additional reconfiguration and relocation of core debris, another possible explanation is that there might be a time window when the core is without absorber material while the fuel is still in intact geometry, so reflooding in this time window may cause recriticality, further analysis and better sampling over failure domain border are necessary to clarify this issue.

While performing clustering and classification analysis we faced a problem of sparse data coverage in the failure domain. Since the algorithm is very sensitive to initial grid resolution (see figures 4.4, 4.5 and 4.9), higher initial resolution gives better representation of the failure domain, but on the other hand requires higher data set quality without artificial "gaps". The algorithm identified additional scenarios to be simulated to get complete uncertainty space coverage for initial grid step and different refinement steps (see figures below).

To get better quality and precision on the results it is possible to improve refinement algorithm by:

- forcing refinement algorithm to concentrate on the area where there're high percentage of "safe" scenarios in the failure domain(through the failure domain border) (see Figure 4.7 and 4.8, failure cells around safe domain).
- adjusting refinement strategy in these regions (possibly by using smaller grid step, force split and analysis of resulting sub-cells even if refinement criterion is not met, use another refinement criterion in general/for these type of cells).
- make algorithm less sensitive to initial mesh refinement.

4.3 Steam explosion

A case study performed at VTT dealt with Olkiluoto nuclear power plant and ex-vessel steam explosions at units 1 and 2. The study demonstrated the applicability of Integrated Deterministic and Probabilistic Safety Assessment (IDPSA) to support risk-informed decision making in plant safety considerations. At the Olkiluoto plant, the lower drywell (LDW) is flooded as an accident management measure when a possible vessel breach occurs, and the hot fuel ejected into water pool can vaporize water so rapidly that the resulting sudden pressure peak meets all the characteristics of an explosion. To produce a sensible steam explosion analysis, the governing physical phenomena of steam explosions had to be studied from literature. Thus a good understanding of fuel coolant interactions, and parameters it is sensitive to, was obtained. A brief introduction to steam explosion phenomenology is given next.

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 vapor film between melt and water (premixture). At this stage, the melt fragments are typically of cm scale. Due to the vapor film, heat transfer from corium to water is relatively low. For a steam explosion to occur, a triggering event is required. One example of a trigger is when the melt front hits the bottom of the cavity or the vessel. The trigger induces vapor film around melt fragments to collapse locally, and melt-water contact takes place resulting in fine fragmentation (typically $< 100\,\mu$m) 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.

The steam explosion phenomena can be divided into four steps:

- Premixing
  - The premixing phase includes melt fragmentation, void (gases in mixture) generation and melt solidification.
  - The premixing process is named so because it precedes a possible steam explosion, which is a different type of mixing process
  - During the premixing process most of the corium must remain in molten state for a steam explosion to occur
- Triggering
  - Collapse of vapor film around melt fragments may initiate fine fragmentation and rapid heat transfer. The initiating event is called the trigger.
  - A spontaneous trigger is at issue when the trigger is generated by the system itself
  - The triggering of a steam explosion still appears as a stochastic event and it is practically impossible to predict it in real situations
- Propagation
  - The explosion trigger causes the vapor film around the neighbor particles to collapse, inducing rapid fine fragmentation of the melt drops
  - In the propagation phase the thermal energy of the melt is converted into thermal energy of the coolant
- Expansion and energy release
  - Thermal energy of the coolant converts into mechanical energy

    - When a detonation wave progresses, the region behind detonation front is an expansion zone where high pressures are reached

Parameters that are known to influence the occurrence and consequences of a steam explosion include:

- Contact mode of the molten core and water
    - Melt either enters water or water is poured onto melt
    - Pressurized vs. gravity-driven melt ejection
- Melt temperature, mass and composition
    - It has been observed in experiments that there is a threshold in melt mass to make a steam explosion possible
    - A strong explosion requires a large premixture
    - The presence of metallic components in the corium melt tends to favor suppression of the explosion
- Coolant temperature and amount of coolant
    - It has been observed that explosions are suppressed when coolant is at saturation temperature (in ex-vessel scenarios coolant is subcooled)
    - A sufficient water depth is needed so that there is enough water available for mixing
- Ambient pressure
    - With higher ambient pressures, the occurrence of explosions is suppressed thanks to increased vapor film stability supported by the elevated pressure
- Void fraction and non-condensable gases during premixing phase
    - Has a strong effect on the pressure pulse propagation
    - Affects fragmentation and heat transfer between melt and coolant
    - High void fraction interferes with thermal interaction between melt and coolant

Deterministic accident progression simulations were conducted by using MELCOR input file developed for Olkiluoto BWR units 1&2. 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. Thereafter the reactor can be depressurized by discharging steam into the wetwell by use of automatic depressurization system (ADS) and its relief valves (314 valves), and this action is initiated by very low water level in the reactor. Depressurization allows the use of low-pressure core spray (system 323) in order to provide core cooling and thus core uncovery that could eventually lead to a vessel melt-through can hopefully be avoided. Also, a high-pressure melt ejection is an even more undesired event than a low-pressure melt ejection because it is less predictable. At 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 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 depends on AC power whereas 314 valves operate with batteries. The termination time of the simulations is set to 10 hours, i.e. 36000 seconds.

Six different cases were selected in order to get an idea how e.g. 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, i.e. in this case the recovery of external power supply, was the main parameter varied – for both high and low reactor pressure vessel (RPV) pressure scenarios. After evaluating this kind of extreme situations, one can use expert judgment to interpolate to less drastic scenarios and avoid performing too high number of simulations, which could prove quite an impractical approach.

All the cases analyzed are shown in Table 4.1. 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. Other but the first two rows in the table are thus determined purely by the model without user interference. In cases 1 and 2 core cooling is provided by system 323 and in cases 3 and 4 by system 327. All ECCS pumps are assumed to work, i.e. both systems 323 and 327 operate with full capacity. LDW flooding was carried out in all of the cases at 1800 s, and is therefore not listed in the table. Different depressurization times (cases 1&2 vs. case 5) originate from the physical criteria which trigger the starting of the system. The table also gives time points for e.g. when the fuel cladding temperature exceeds oxidation threshold for Zirconium (1100 K) and when the core melt relocation starts. In cases 1 and 3 an ex-vessel steam explosion is not possible, because in both cases the accident progression is halted by successful safety actions.

Table 4.1: MELCOR simulation results from the six cases investigated including e.g. information on the major events affecting the possible steam explosions

	Case #					
	1	2	3	4	5	6
ECCS availability?	Recovery at 3000s	Recovery at 4000s	Recovery at 18000s	Recovery at19000s	No	No
Depressurization 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

The instant of vessel failure is the most crucial point in time regarding potential ex-vessel steam explosions. Therefore values for LDW pool subcooling, partial pressure of water and the amount of melt ejected into water when the vessel fails are listed in Table 4.1. Also the timing of filtered containment venting through system 362 is given in the table, even though it would probably take place after RPV breach, given that a steam explosion has not induced a containment failure. As discussed earlier, high subcooling and low ambient pressures favour explosions, and the more melt, the bigger the premixture and the explosion potential. The model used does not accommodate e.g. high pressure melt ejections with high melt jet velocities but with MELCOR at least some important accident sequence-specific information can be obtained. Steam explosion phenomena starting with premixing phase could then be calculated in detail with specialized software.

For illustration purposes, time development of some important quantities is shown in the following figures. In Figure 4.11 is displayed the partial pressure of liquid water in the LDW. In cases 4 & 6 the pressure is high when the RPV breaches, but decreases rapidly due to drywell rupture disk venting (system 362) which is designed to prevent containment over-pressurization. In other cases the pressure increases at quite a steady rate, although pressure spikes can be seen at the instant of vessel breach. In general, the LDW pressure and subcooling curves resemble one another in terms of how the different cases analyzed differ from each other. This is natural, because the two quantities are interconnected.

Figure 4.12 shows the cumulative melt amounts ejected from the RPV into the lower drywell, and in cases 1 and 3 there is no vessel failure and thus no melt ejection. In case 4 the vessel fails but apparently the recovered ECCS can prevent any major vessel penetration and no melt is ejected into the LDW. In cases 2, 5 and 6 high amounts of melt end up in the LDW. The melt amounts lie at about 185 tons in cases without ECCS recovery (5 & 6), and in case 2 the ECCS seems to be able to restrict the mass of ejected melt. Also the melt exits RPV in a more gradual manner. In case 6 the RPV is still pressurized when the vessel breach occurs, but that does not appear to have any significant effect on melt amount. Theoretically, a high pressure melt ejection could result in bigger melt amounts in the LDW, or at least higher momentary melt ejection rates, but in the light of evidence from Figure 4.12, that conclusion cannot be drawn.

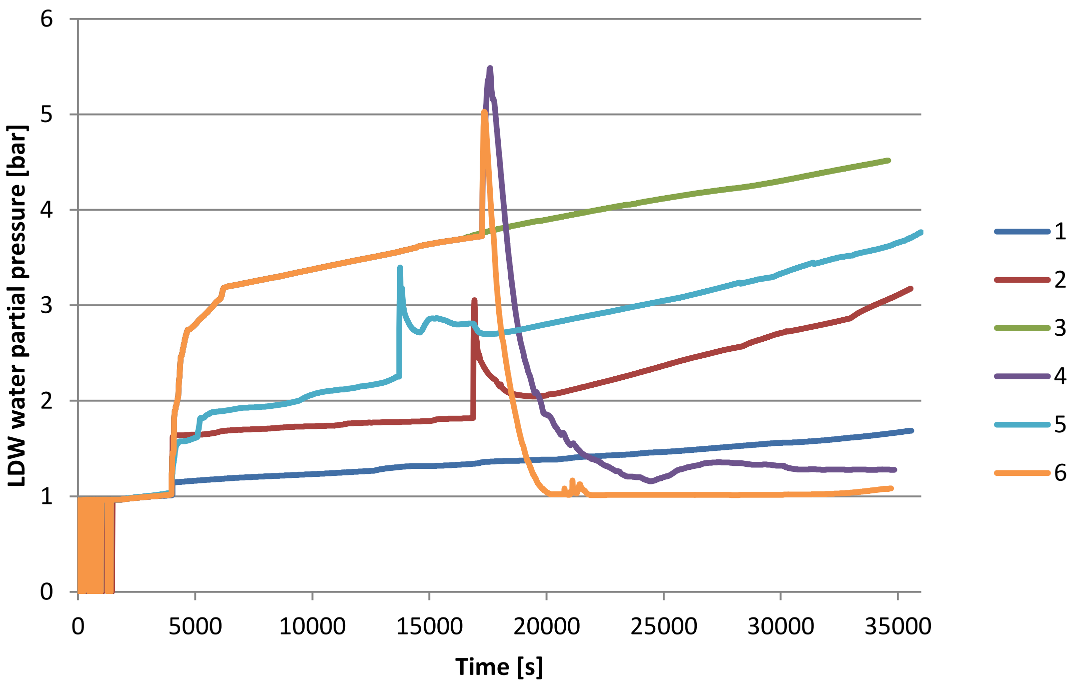


Figure 4.11: Partial pressure of liquid water in the lower drywell. High ambient pressures suppress the occurrence of explosions.

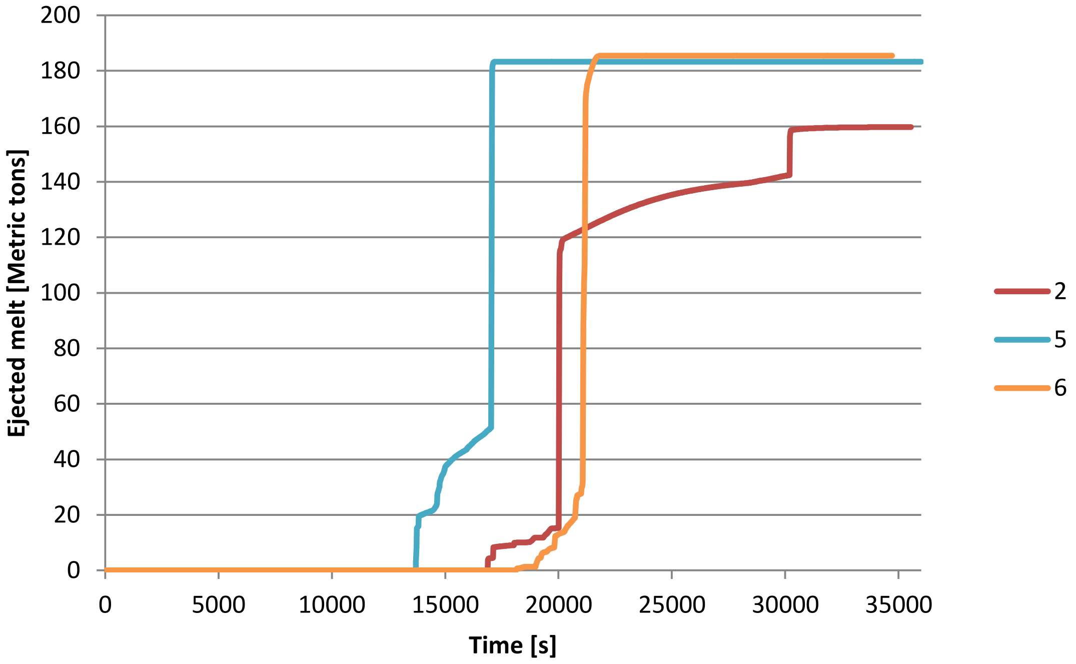


Figure 4.12: Melt ejected from the reactor pressure vessel after the vessel breach. In cases 1 & 3 there is no vessel failure and in case 4 there is a vessel breach but no melt ejection into the LDW.

The MELCOR results shown here mainly relate to core melting and relocation phenomena that can have an effect on ex-vessel steam explosions. Thus the simulations do not practically go much beyond vessel breach and the possible ex-vessel steam explosion. The results would be best exploited in molten fuel-coolant interaction FCI analysis in order to get estimates of pressure loads the containment is expected to withstand. These pressure estimates would then be implemented into a Level 2 PSA model together with MELCOR results. In this study, no FCI code is used and the results are taken into a CET model without further refinement. This is also a workable approach, but more expert judgment is needed. In practice one has to refer to literature with respect to e.g. explicit pressure values or explosion probabilities.

In light of the simulations, it is not straightforward to deduce which scenario has the highest risk regarding explosive FCI phenomena. When RPV depressurization is not successful, high water subcooling favors explosions whereas at the same time high ambient pressure functions the other way around. Failure in depressurization can also lead to high melt jet velocity which is likely to have an influence on the premixing phase. When depressurization is successful, subcooling and ambient pressure values are smaller. Maybe most importantly, depressurization affects the timing of vessel failure. When the RPV is depressurized, vessel breach can occur significantly earlier than without depressurization. Therefore the time window for ECCS recovery was observed to be a lot wider for pressurized cases, although depressurization allows the use of low-pressure core injections with system 323.

Without sensitivity studies with some FCI code it is difficult to determine which parameter has the biggest impact on the severity of a potential explosion. Even so, the results shown in this section are utilized in the development of a Level 2 PSA model, which is implemented conventionally as a containment event tree.

Chapter 5. Results of feasibility study on connection between conventional PSA, DSA and DPSA methods

5.1 Results of analysis of core relocation scenarios

IDPSA analysis show that the relocation process, amount and composition of relocated debris in the lower plenum strongly depend on the timing of recovery of safety systems and their capacity. The results of IDPSA analysis provide relocated debris properties (i.e. timing of big relocation, debris composition, etc.) for analysis of vessel failure conditions in ROAAM framework [24]. Different in-vessel accident progression scenarios will result in both different melt-ejection modes and containment conditions (e.g. depressurization history and ECCS will affect wet-well water temperature, which in turn affects water pool temperature in the cavity, and, therefore, affects probability of ex-vessel steam explosion). Thus, based on IDPSA analysis it is possible to provide actual containment failure probabilities for CET in PSA L2. Furthermore, based on the ROAAM analysis it is possible to select the most conservative scenarios to consider in DSA.

Currently, the results from the DSA analysis show that there are some thresholds that change the conditions for the sequence. Therefore, it is very likely that these thresholds will be interesting for the PSA.

The output of the core relocation scenarios is not yet in a position where it is fully useable in the PSA. There is a need for further investigation to make sure that the correct conclusions are drawn.

The indications from the IDPSA evaluations for the reference Nordic BWR plant design are that, studying the results generally, there seems to be a threshold where less than 20 tons will be relocated to the lower plenum. If less than 20 tons is relocated then it is very likely to be coolable and the melt can be arrested in the RPV. The analyses indicate that if feedwater is recovered within 5 000 seconds, then less than 20 tons is relocated. The analyses also show that a recovery later than 10 000 seconds will likely result in relocation larger than 200 tons, which is probably not coolable. In between 5 000 and 10 000 seconds it is hard to conclude whether or not the melt is coolable in the RPV.

The decision tree approach refines the information, and there are some very interesting results that will require further evaluation. If figure 4.10 is studied, it states that in a situation where depressurization is performed within 4000 seconds and ECCS is started within 3900 seconds, the more water you inject with ECCS the better. If ECCS is started later than 4000, but within 7800 seconds, it is necessary to balance the water injection rate to be in safe state (which in this approach is defined as less than 10 tons relocation). If more, or less, water is injected the core relocation will be greater than 10 tons.

A similar situation is if depressurization is actuated after more than 4000 seconds. If ECCS is activated it needs to provide more flow area than represented by 0.021 to have the potential to be in safe state, but ECCS cannot be activated before 3900 seconds.

The decision tree approach shows that the general results can be refined, to better define the exact conditions that are relevant. But, there are also results that will need further investigation.

Methodological enhancement – DSA and PSA integration

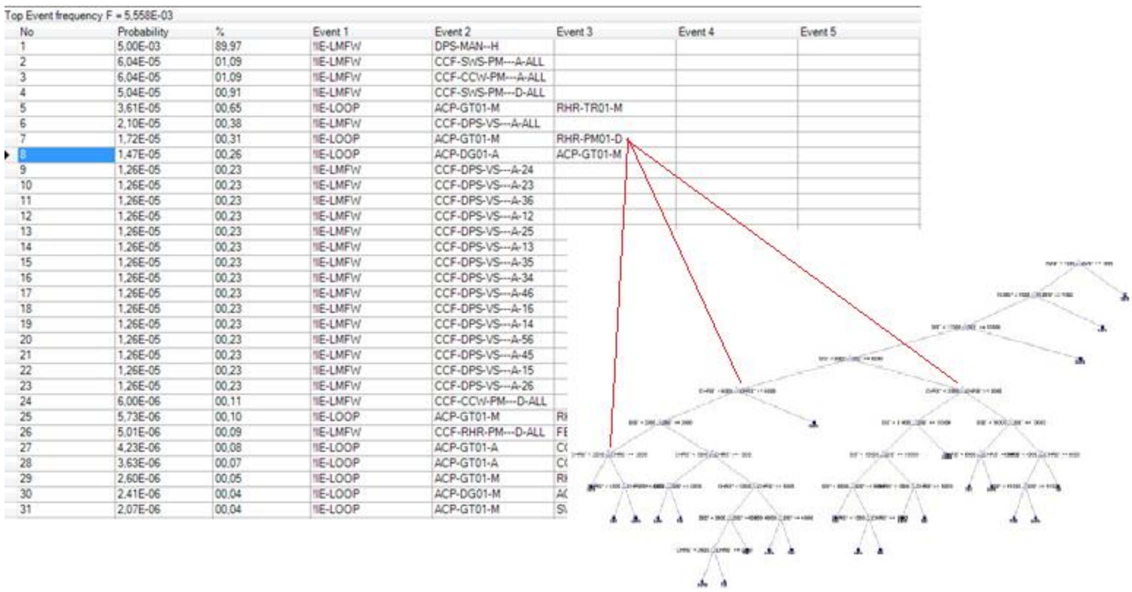
The results from the study are also very interesting from a methodological point of view. It has been shown that a decision tree is a viable way of presenting the output of the DSA analysis. The question is; how can this data be used in a PSA context?

From the PSA, cut set lists are produced (or rather minimal cut set lists). In most PSA tools, like RiskSpectrum® PSA, the same approach is used also for PSA Level 2. There are other methods to perform the PSA Level 2 calculations, which e.g. is the situation in FinPSA. A short discussion on advantages and disadvantages with the concepts are presented below.

The discussion in this section is focusing on the MCS approach, as used in RiskSpectrum PSA, and how this can be improved by including information from the decision trees developed within the DSA.

Let us assume that we have an MCS list. This list will include basic events representing phenomena (as well as component failures and human actions – but these are not of interest in this context). These phenomena are treated as individual events – and there is no information on timing. Now, let us assume that we have the decision tree as presented in section 4.2.2.

The combination of the MCS list, and the information in the decision tree could be merged. Figure 5.1 is presenting the conceptual idea, assuming that the event in the MCS list can have different probability (different mission time) in different sequences driven by the decision tree.



Top Event frequency F = 5,558E-03

No	Probability	%	Event 1	Event 2	Event 3	Event 4	Event 5
1	5,00E-03	89,97	!IE-LMFW	DPS-MAN--H			
2	6,04E-05	01,09	!IE-LMFW	CCF-SWS-PM---A-ALL			
3	6,04E-05	01,09	!IE-LMFW	CCF-CCW-PM---A-ALL			
4	5,04E-05	00,91	!IE-LMFW	CCF-SWS-PM---D-ALL			
5	3,61E-05	00,65	!IE-LOOP	ACP-GT01-M	RHR-TR01-M		
6	2,10E-05	00,38	!IE-LMFW	CCF-DPS-VS---A-ALL			
7	1,72E-05	00,31	!IE-LOOP	ACP-GT01-M	RHR-PM01-D		
8	1,47E-05	00,26	!IE-LOOP	ACP-DG01-A	ACP-GT01-M		
9	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-24			
10	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-23			
11	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-36			
12	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-12			
13	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-25			
14	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-13			
15	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-35			
16	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-34			
17	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-46			
18	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-16			
19	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-14			
20	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-56			
21	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-45			
22	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-15			
23	1,26E-05	00,23	!IE-LMFW	CCF-DPS-VS---A-26			
24	6,00E-06	00,11	!IE-LMFW	CCF-CCW-PM---D-ALL			
25	5,73E-06	00,10	!IE-LOOP	ACP-GT01-M	R		
26	5,01E-06	00,09	!IE-LMFW	CCF-RHR-PM---D-ALL	F		
27	4,23E-06	00,08	!IE-LOOP	ACP-GT01-A	C		
28	3,63E-06	00,07	!IE-LOOP	ACP-GT01-A	C		
29	2,60E-06	00,05	!IE-LOOP	ACP-GT01-M	R		
30	2,41E-06	00,04	!IE-LOOP	ACP-DG01-M	A		
31	2,07E-06	00,04	!IE-LOOP	ACP-GT01-M	S		

Figure 5.1. The conceptual idea of having the decision tree as input for the quantification of an MCS list. The figure is intended to illustrate that one event may have different failure probability in different cases.

This would mean that each individual combination of failure events, leading to a particular release, could be quantified for the possible alternatives of combinations in the decision tree.

Why is the approach with a decision tree interesting, compared to the alternatives?

There are several possible methods for including timing and correlation between phenomena into a risk calculation:

- Integrated PSA L1 and L2, but probabilities for phenomena are determined outside the PSA tool
- Integrated PSA L1 and L2, but conditional release probabilities for the different PDSs are calculated outside the PSA tool

Integration of PSA Level 1 and Level 2 is the most common method for modeling the interface between PSA Level 1 and Level 2 (the first case above). The advantage with the method is that there will be an automatic treatment of dependencies between Level 1 and Level 2, failures that are considered will fail both systems in Level 1 and Level 2. The amount of plant damage states may due to this be limited. The disadvantage with the approach is that a strict probabilistic approach does not include time as a parameter and there is not an easy way to represent other type of dependencies between phenomena.

The second alternative means that the results are computed from PSA Level 1, and transferred to PSA Level 2 as frequencies for each PDS. This means that the treatment of dependencies will have to be established by defining more plant damage states (then also including system states). It is obvious that the full set of dependencies cannot be represented. This method, on the other hand, is very powerful because an equation system can be defined for the different events that can happen, and time is a possible parameter.

The idea behind the merge of the decision tree and the MCS list would be to capture the good features of the MCS approach, but also to, in a condensed way, utilize all the relevant information from the deterministic calculations – including timing. An obvious strength of the concept would be to not simplify the very complicated scenarios by equations, but instead use the thresholds that are of interest.

The approach to run a lot of calculations to define the relevant scenarios could be compared to CFD calculations that are performed for example for explosion risk analysis within the oil and gas industry. Initially a limited amount of deterministic explosion calculations is performed and this information is used to make judgments on all scenarios, but today the trend is surely towards using a very massive amount of CFD calculations.

Quantification and representation

Given that the information in the MCS list is available, and given that we have a decision tree capturing all relevant information, the quantification will not be a trivial task since additional information will be required. Timing is a very important part, and it is also expected that some events in the MCS list could be obsolete in some scenarios. Obsolete events should preferably, from PSA stand point, be represented separately in the PSA logic – and thereby be represented by separate MCSs.

To some extent the timing information could be considered being already part of the basic events, through the reliability model "mission time". This reliability model is represented by following formula:

$$Q(t) = q + (1-q)\left(1 - e^{-\lambda Tm}\right) = 1 - (1-q)e^{-\lambda Tm}$$

Where:

Q(t) is the failure probability of the component q is mission time independent failure probability for the component

λ is failure rate, mission time dependent failure probability

Tm is the mission time

This reliability model is used for most components in the PSA model, which have a mission time. However, this mission time is not referring to a specific time point, but rather specifying the time the component is required to operate. There is no way, in the PSA tool today, to specify an origo, for example, at the time core melt begins. In many cases the same events are used in the sequences prior to core damage, and hence they cannot be set zero in the PSA Level 2. This issue needs to be resolved, for example by assuming that the mission time in PSA Level 2 is in addition to the specified mission time (which would then be for PSA Level 1).

Other type of events that could be affected by different timing is for example phenomena and operator actions that could be dependent on available response time. The type of data for the operator response would need to have some time dependent model, like THERP, or to have some other method for inclusion of such data. It could also be a very simple reliability model, in which you specify the failure probability for some time points. Example is given in the graph below.

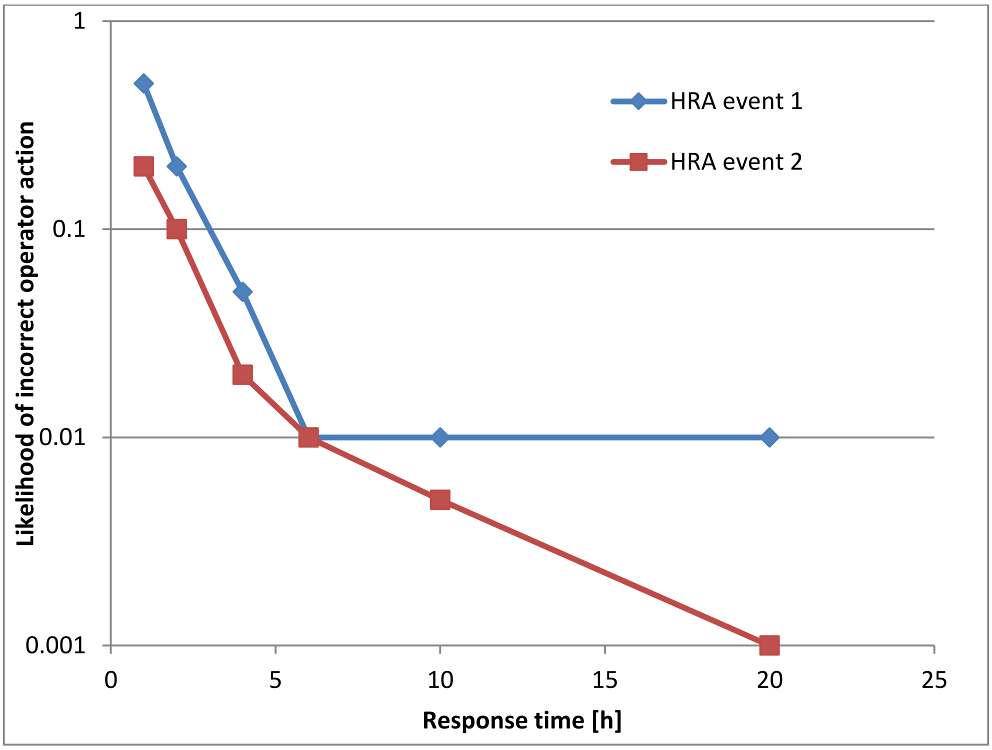


Figure 5.2. An example of two HRA events depending on required response

Regarding phenomena and their likelihood; the idea would be to identify the type of scenarios that could result in a specific type of phenomena, and scenarios where the phenomena could be ruled out. Therefore, a first hypothesis could be to have a few probabilities for phenomena – and select the most representative one for each scenario.

A challenging part will be to identify which events that should be referred to as affected. The decision tree will be on a high level – but the results for the MCS are on a very detailed level. It should be reasonably easy to, for example, allocate basic events to systems. But, there are support systems that affect more than one front line system, for example diesels, and it will be a challenge to define the appropriate timing for these events.

Assuming that these data are available and that we have been able to connect all basic events with their relevant node in the decision tree, we could calculate the MCS conditional the decision tree. It is likely that there may be several combinations of timing of events that may cause the MCS. Some sort of convolution approach will then be needed to encompass all scenarios.

It should also be mentioned that how the quantification of the MCS list shall actually be performed is also a subject for research. Some possible methods could be:

- BBN, Bayesian belief networks, to be able to quantify events depending on other events
- Automata, to be able to include states and time into the analysis

5.2 Results of analysis of steam explosion scenarios

In this section, the results of probabilistic part of VTT's work are presented. The containment event tree model tries to exploit deterministic simulations to as great extent as reasonably possible. Because deterministic accident progression analysis was carried out by using an input for a station blackout (SBO) scenario, also the probabilistic modeling part deals with this type of plant state. Figure 5.3 shows the CET developed in this study. The plant damage state at issue is best described as high or low pressure transient/melting depending on the result from CET heading concerning reactor coolant system (RCS) depressurization. Core cooling systems are assumed unavailable until the possible AC power recovery.

The branch functions are also shown in the figure and there are 21 sequences in the tree. The sections/headings of the tree include:

- ECCS recovery (ECCS)
  - Are core cooling systems (both 323 & 327 recovered in time to avoid full core melt?
- Very early containment failure (VEF)
  - Recriticality induced containment over-pressurization
  - Hydrogen deflagration/detonation induced containment failure
  - In-vessel steam explosion causing containment alpha-mode failure
- Vessel Failure (VF)
  - Vessel lower head failure caused by melt-through
- Early containment failure (EF)
  - Ex-vessel steam explosion
  - LDW penetration failure
- Late containment failure (LF)
  - Containment failure occurring hours after RPV failure

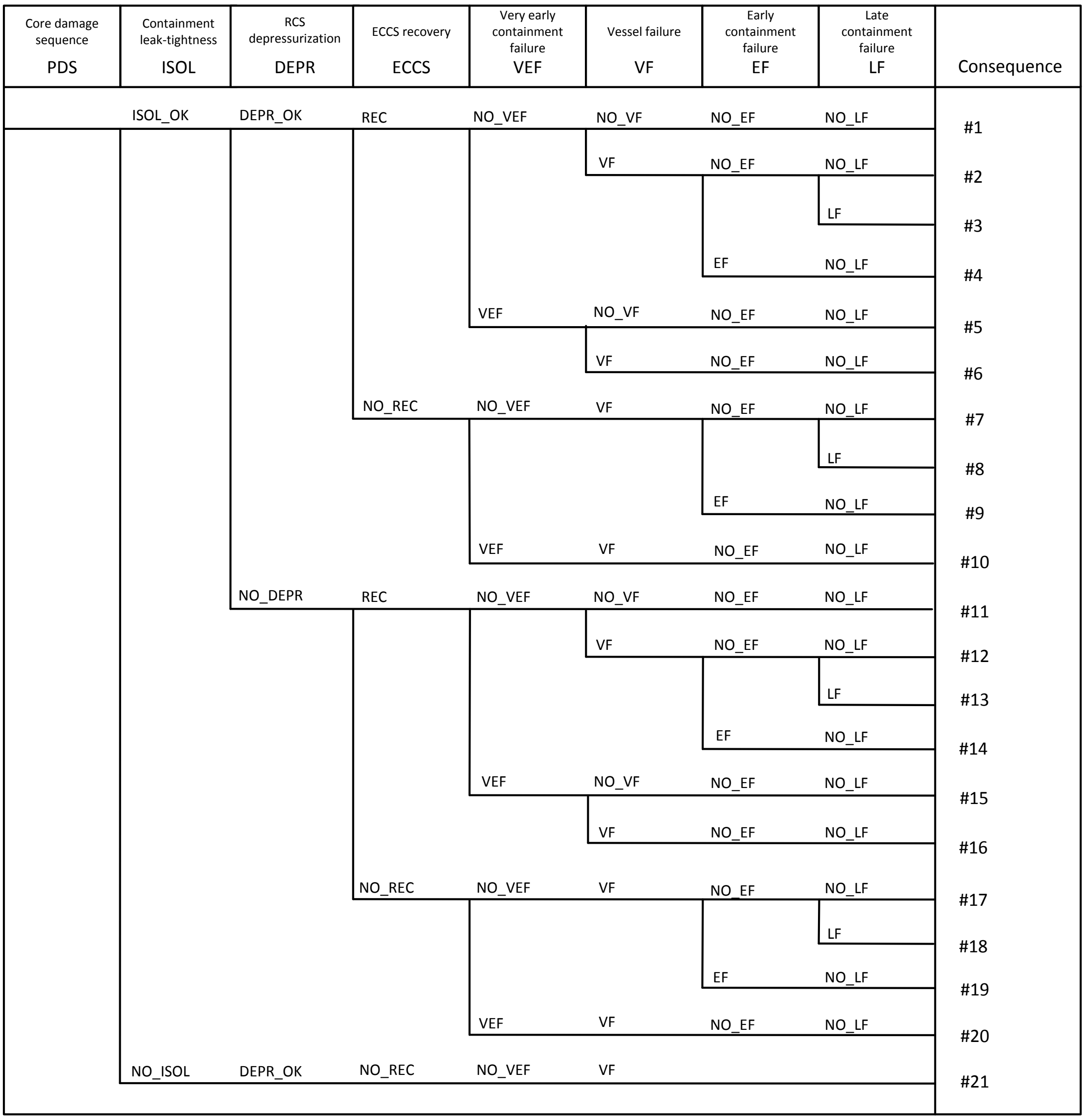


Figure 5.3 The containment event tree developed for the case study. The upwards branches represent successes in safety functions whereas downwards branches result in unwanted consequences. Names of the branch functions are also given in the figure.

Table 5.2 summarizes all the failure categories and failure modes containment is assumed to be prone to in this study. The same categorization is used in the source term model with the exception that early containment failure is decomposed into two failure categories according to the failure mode that has induced the release of fission products out of the containment. Thus the results from source term analysis regarding steam explosions can be directly observed.

The modeling of steam explosions in the CET is mainly based on the general phenomenology and knowledge obtained from the literature. Without possibility to conduct steam explosion analysis, a less detailed and somewhat heuristic approach had to be adopted. There are results about the shape and order of magnitude of explosion loads available in literature. The basic idea is that LDW has a certain resistance to pressure impulses caused by steam explosions and containment failure occurs if this capacity is exceeded by the explosion impulse. Both structural capacity and impulse load are expressed as distributions in order to introduce probabilistic elements into the analysis.

Table 5.2. Containment failure categories and the corresponding failure modes used in the CET model. The same categories are used to bin accident sequences in source term analysis.

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

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 $\sigma = 0.421$. Also for explosion impulses, lognormal distributions are used and an explosion impulse in the CET model depends primarily on three things:

- LDW flooding
- Containment debris fraction
- RCS pressure

LDW flooding is a requirement for any FCI phenomenon to take place, and it is assumed that LDW flooding has to be halfway through at the instant of vessel failure (melt ejection) so that the emergence of 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 big, 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, thus reflecting the possibility of really massive explosion impulses.

RCS depressurization is modeled to have an effect on the triggering of an explosion and also on the magnitude of it. If primary circuit is pressurized when the vessel breaches, the triggering of an explosion is assumed to take place with 100 % probability, whereas low-pressure melt ejection triggers explosive FCI phenomenon with 50 % probability. On the other hand, explosions associated with high-pressure melt ejection are estimated milder.

Figure 5.4 illustrates the load distributions used and also shows the distribution employed for LDW strength. Table 5.3 shows the conditional probabilities of impulse load exceeding LDW strength. For instance if RCS is depressurized and there is much melt involved in the FCI (case LP1), the probability of containment failure in a steam explosion is over 20 %. Being conditional probabilities, the values in Table 5.3 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.

Along with steam explosions, more modeling effort has been put to source term evaluation. The model developed is based on the so called XSOR method. The model here concerns only three radionuclide groups, namely noble gases (source term variable S_Xe), cesium (S_Cs) and rutherium (S_Ru). The three groups have been chosen so that the behavior of radionuclides belonging to different groups would deviate significantly from one another. Source term modeling contains considerable uncertainties, especially regarding fission product transport. Therefore the model aims to give order of magnitude estimates instead of trying to predict accurate values.

Fission product model applies to atmospheric, i.e. gaseous releases. All releases, except noble gases, are assumed to be in aerosol form, which effectively 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 eventually released from the containment as well. Almost all parameters used in the source term model are treated probabilistically as distributions because of the uncertainties involved. Starting point for the calculation of releases is the core release, from which three different release mechanisms are derived. The mechanisms are early and late release from RCS and ex-vessel debris release. The timings of different release mechanisms are shown in Figure 5.5.

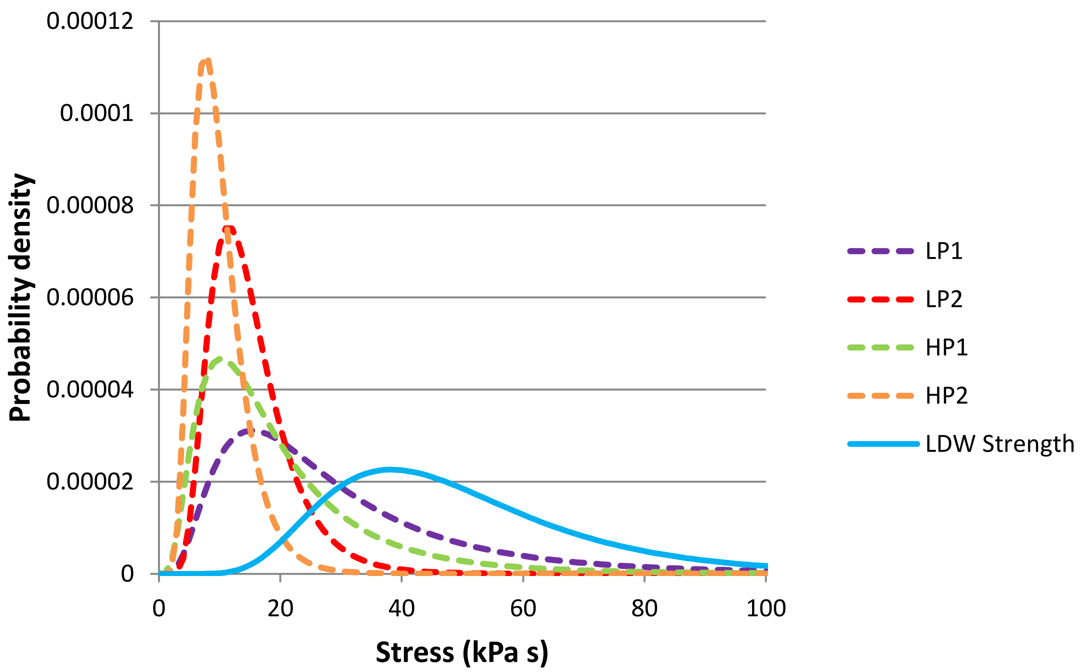


Figure 5.4. Distributions used to determine whether LDW fails due to pressure impulse caused by ex-vessel steam explosion.

Table 5.3. 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

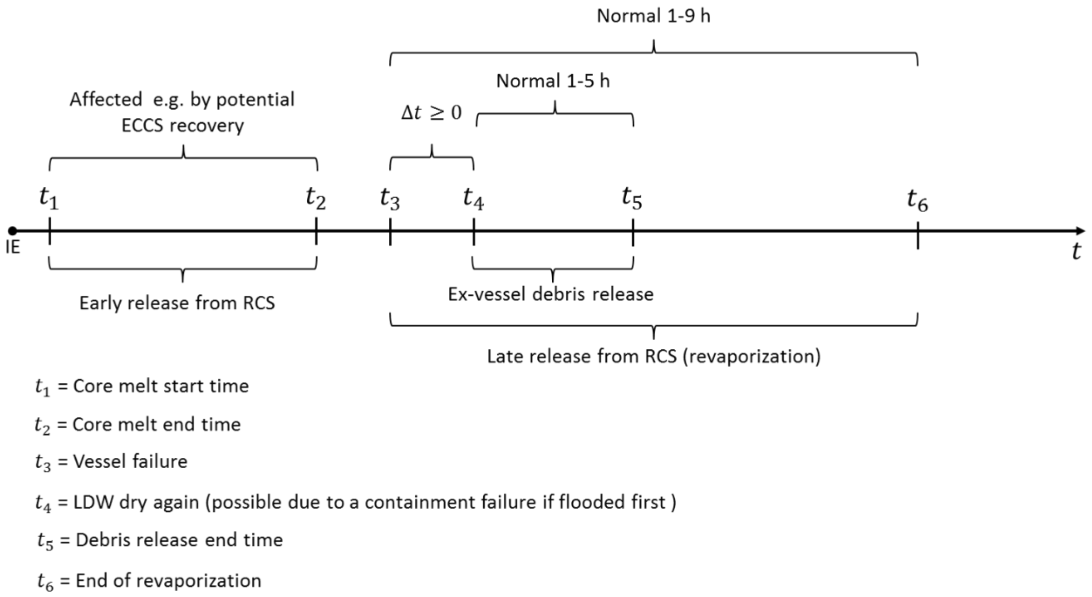


Figure 5.5: The timings of the different release mechanisms.

The results shown in this section are merely results from source term analysis which is a well-established way to communicate the findings of a Level 2 PSA study. Release categories are given probabilities and the associated releases are evaluated with respect to the three radionuclide groups selected for the analysis. The results can be seen in Table 5.4. Release categories used in source term analysis are more or less consistent with containment failure categories used in the CET. From the table it can be observed that filtered venting (FV) and OK bins are dominant with a combined share of 77% in mean probability. Very early failure (VEF) is also quite probable with 19% of the simulations ending up in this bin. Early containment failure caused by a steam explosion (EF_STEAM) has 2.6% mean probability and together with penetration failure (EF_PENE) it constitutes the early containment failure categories. Along with the mean conditional probability, the 5th and 95th percentiles are shown in the table so that one could get an idea of the shape of probability distributions associated with each bin.

Source term for noble gases (S_Xe) is generally very high because there is neither decontamination nor deposition considered for them and release fraction parameters for noble gases used in calculations are the highest for every release mechanism. Even for OK bin there is a minor leakage of noble gases. Cesium release (S_Cs) is also quite high for all release categories where containment fails whereas filtered releases are more moderate. Rutherium (S_Ru) release behaves similar to cesium release but the release is a couple of orders of magnitude smaller. Of all failure categories, releases are biggest for late containment failure (LF), but the difference to other failure categories is due to probabilistic approach used in calculations rather than actual dissimilarities regarding the severity of CET sequence outcomes.

Table 5.4: Probability means of each bin along with 5% and 95% percentiles calculated by SPSA. Bin EF is divided into two in order to separate failure modes. Source term means are expressed as fractions of core inventory.

Bin	Conditional probability			S_Xe mean	S_Cs mean	S_Ru mean
	Mean	P(5)	P(95)			
OK	2.3 E-1	1.9 E-15	9.7 E-1	8.4 E-11	-	-
ISOL	1.0 E-2	4.6 E-3	1.8 E-2	8.7 E-1	2.0 E-1	6.0 E-3
VEF	1.9 E-1	9.8 E-6	9.9 E-1	7.8 E-1	1.3 E-1	4.3 E-3
EF_PENE	1.4 E-3	3.2 E-14	6.8 E-4	8.3 E-1	2.0 E-1	7.2 E-3
EF_STEAM	2.6 E-2	1.7 E-11	1.7 E-1	8.6 E-1	2.8 E-1	8.9 E-3
LF	1.0 E-2	5.3 E12	5.3 E-2	9.9 E-1	4.1 E-1	1.4 E-2
FV	5.4 E-1	8.6 E-8	9.7 E-1	7.8 E-1	7.8 E-4	2.3 E-5
Weighted average	1.0 E00	-	-	6.6 E-1	6.1 E-2	1.9 E-3

The purpose of the case study performed at VTT was to look into phenomenology of ex-vessel steam explosions in the context of probabilistic analysis of severe nuclear accidents. The ultimate goal was to implement steam explosion consideration into a Level 2 PSA model utilizing IDPSA methodology, which takes both stochastic disturbances and deterministic response of the plant, and especially their mutual interactions, into account in safety justifications.

Many aspects of accident progression or plant itself were not modelled at all and were left out of the containment event tree. Examples of such events are boration after core recriticality and containment water filling. Also many possible containment failure modes were left out. Taking such short cuts in modelling tasks was nonetheless well warranted because a Level 2 PSA model was intended to serve only as a framework or platform for the study of IDPSA methodology and steam explosion analysis. Therefore accurate results of e.g. individual accident sequence frequencies were of secondary importance, although as realistic results as possible were obviously pursued.

Performing IDPSA analysis is always a challenging task for instance because analysis tools do not necessarily communicate with each other very well and therefore a lot of work and interpretation has to be done by the analyst in terms of fitting the results obtained from different analyses into a meaningful synthesis. In this study, deterministic simulations (MELCOR) were used to support probabilistic modelling (SPSA) of a severe accident progression in reactor containment. To analyze a certain event or phenomenon, such as ex-vessel steam explosions, calls for e.g. thorough planning of the research approach including the selection of analysis tools and how to define the scope of the study and which restrictions and limitations are necessary.

VTT's case study demonstrated successfully the applicability of IDPSA methodology for steam explosion analysis and also in general. A major drawback was the lack of FCI analysis to provide case-specific estimates for explosion loads. In practice, it would be imperative to produce plant-specific estimates to be utilized in PSA studies. Also structural capacity analysis for containment would be important for the credibility of the method used, and this part of analysis was also missed out. In spite of simplifications and compromises made in modeling work, this study clarifies well how to put IDPSA methodology into action and demonstrates the use of it with an example case focusing on steam explosion phenomenology.

Chapter 6. **Conclusions and Suggestions**

The results from the IDPSA show that an increased number of thermohydraulic calculations, performed according to an intelligent algorithm, can improve the understanding of the sequences and therefore input to the PSA or to the deterministic safety analyses.

There seems to be thresholds that can be defined regarding core relocation, which could better define how recovery of feedwater can be considered. There are however some results that need further evaluation to understand if they are correct or they are a result of the numerical modeling uncertainty.

An important step is also to provide a cross code comparison for some sequences predicted with the MELCOR model and with the MAAP model to assess and to reduce possible influence of the epistemic modeling uncertainty.

Currently the IDPSA analysis is performed assuming that feedwater is stopped at time=0. It would also be very interesting to study how the conclusions differ if feedwater would be operating for some time and then stop.

From the PSA stand point, the vision would be to continue the sequences past the vessel melt through to study the impact on phenomena that could potentially challenge the containment. For each phenomena the key factors of importance should be identified, for example timing of vessel melt through, pressure in vessel at melt through, containment pressure at melt through etc. This would allow for an improved representation of phenomena, and also to represent the uncertainties in phenomena with their contributing factors.

There is a good potential for development of a mathematical model to represent the IDPSA results in form of a decision tree as input for the quantification of the PSA Level 2 structure. This approach shall be further evaluated in the next phase of the project.

Steam explosion analyses exploiting IDPSA methodology would necessitate more detailed approach than VTT’s contribution presented here, with use of dedicated analysis codes for FCI phenomena and structural response of the containment. Nevertheless, joint use of MELCOR and SPSA for steam explosion analysis provided a good basis that can easily be refined further.

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Disclaimer

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Appendix A. GA-NPO Analysis of Core Relocation Phenomena

The first iteration of the MELCOR-Input file employed no post processing and simply opened the ADS-Valves in the steam dome completely upon receiving the depressurization signal, with no signal for re-closure.

The target function for the earliest version was to minimize the relocation mass to the lower plenum.

The following two plots show the same graph, once from above as x-y-projection, once as Scatterplot. We have the time delay, in [s], for the ADS-System on the x-axis, the time delay, in [s], for the ECC-System on the y-axis. Both run from 0 to 15 000 seconds. On the z-axis we got the relocation mass from, in [kg], from 0 (dark blue) to 300 000 (bright red).

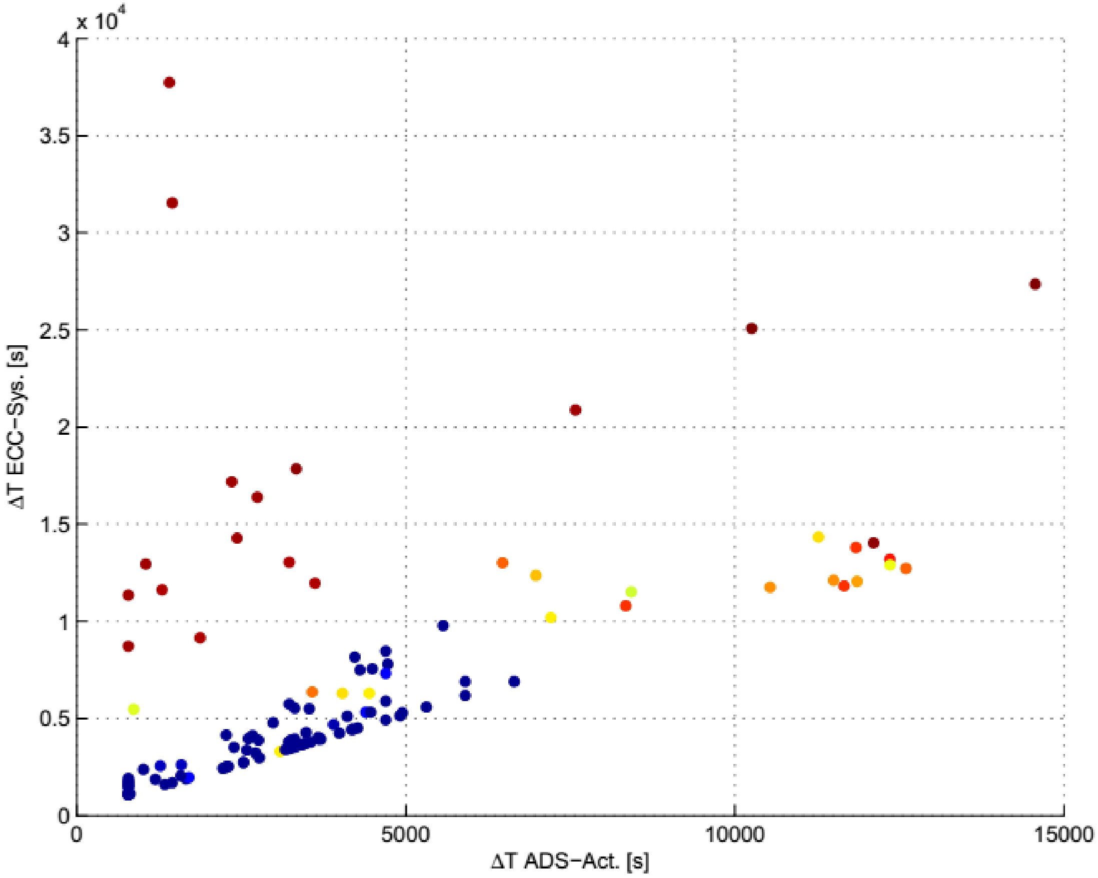

Figure A.1. Iteration-1; Relocation Mass over ΔT for ADS and ECCS; x-y-Projection

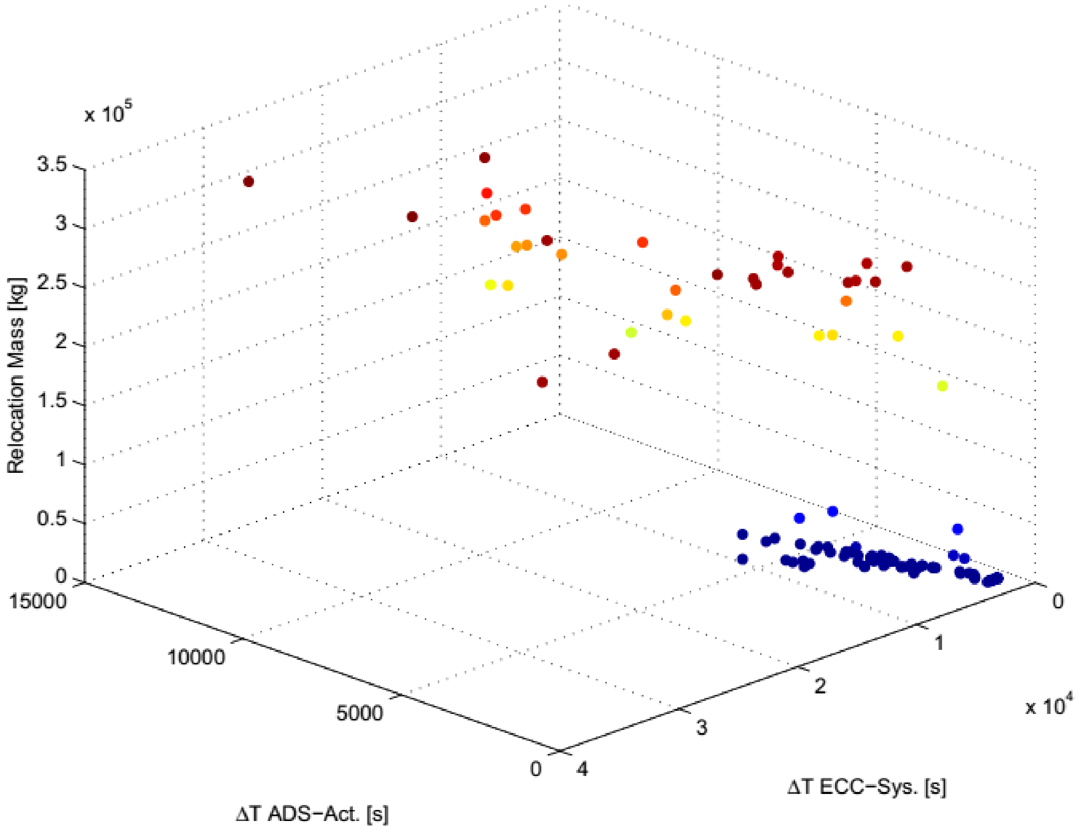

Figure A.2. Iteration-1; Relocation Mass over ΔT for ADS and ECCS; 3D-Scatter-Plot

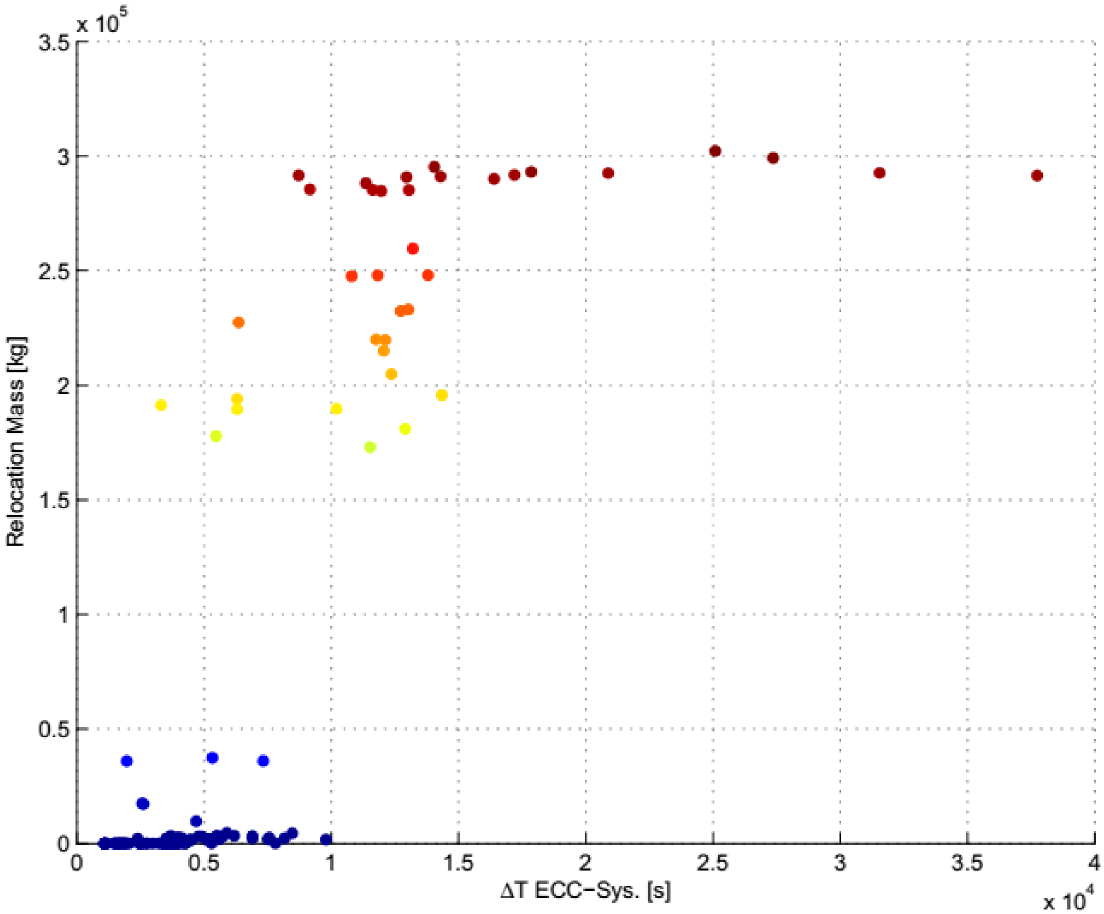

Figure A.3. Iteration-1; Relocation Mass over ΔT for ADS and ECCS; Projection on y-z-Plane

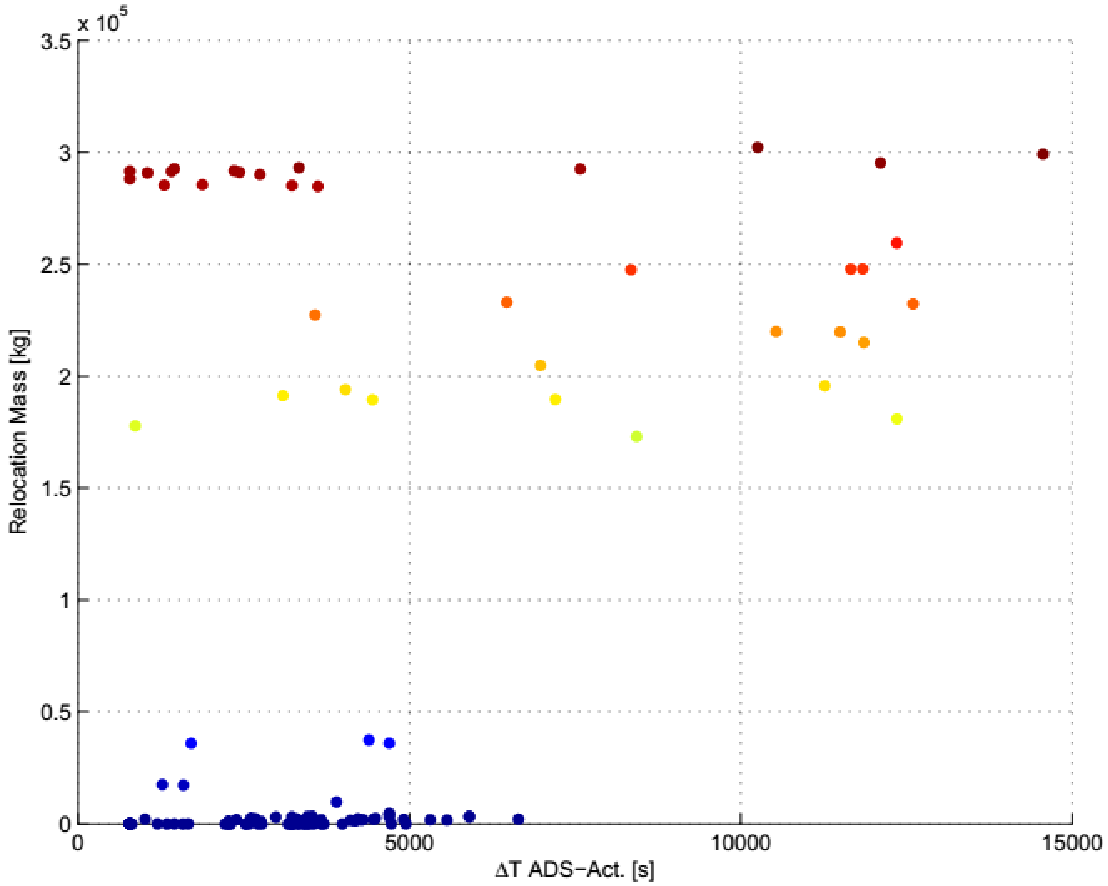

Figure A.4. Iteration-1; Relocation Mass over ΔT for ADS and ECCS; Projection on x-z-Plane

For the second iteration of the GA-NPO run the control logic of the ADS-Valves was modified in an effort to get a more realistic core uncovery. Specifically the ADS-Valve were modified to close again due to their own weight, when the internal pressure in the RPV drops below 4 bar and be entirely closed at 3 bar. Due to original misinterpretation of the description of the control logic, the valves were assumed to stay closed until an internal pressure of 70 bar was reached again, yielding the typical graph in Figure A.5.

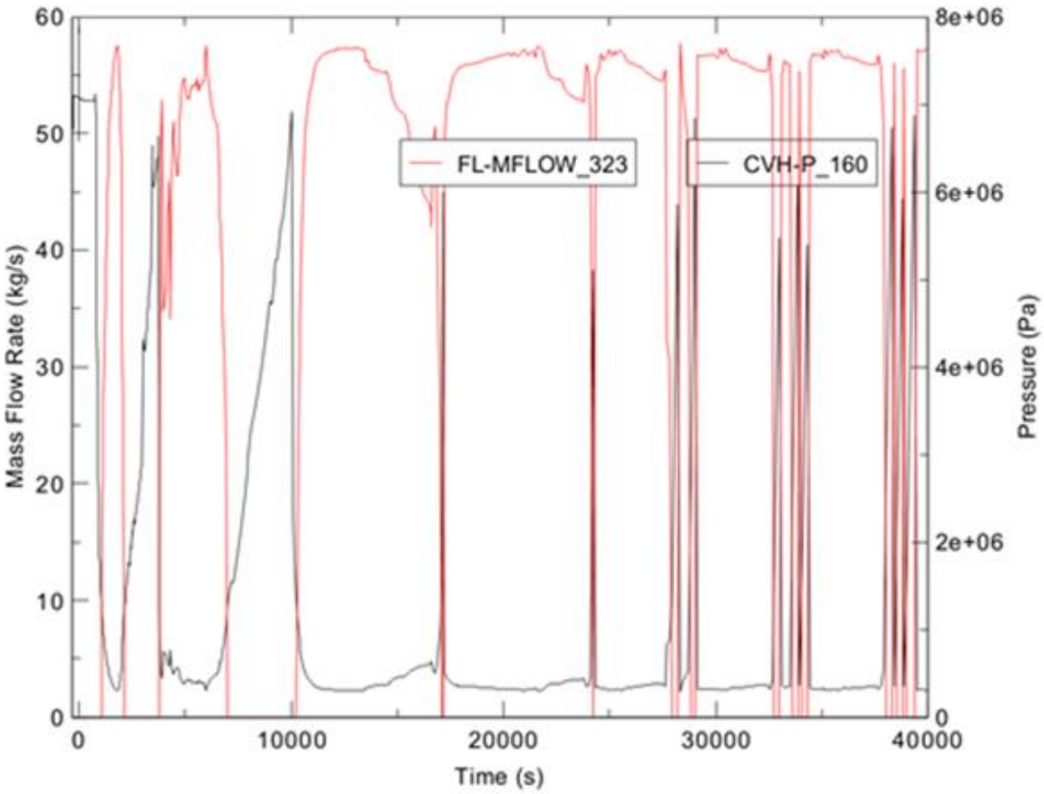

Figure A.5. ECCS Mass Flow Rate on the left y-axis; Steam Dome Pressure on the right y-axis

The mass-flow from the ECCS will proceed either way, as long as the pressure dependency of the pump curve allows (massflow goes to 0 for a pressure of 12.5 bar), rising the pressure, together with the decay heat vaporization towards 70 bars, until the ADS-Valves open again, resulting in the oscillation form in (Figure A.5).

Pressure – difference [MPa]	**Fraction - Open**
1.25	0.000
1.15	0.2571
1.0	0.4288
0.75	0.57159
0.5	0.82991
0.2	0.98578
0.1	1.0
0.0	0.0

Appendix B. Hypothesis about the cliff-edge effect in core relocation

There are several working hypothesis, as to what might contribute to the large discontinuity in the relocation masses of the scatterplots. To eliminate as many factors as possible we compared instances of the code, which were as close as possible in the chosen parameter values, but differed substantially in the relocation masses.

The large difference is likely caused by the continued integrity of the lower core plate plus refrozen melt (or a similar plug anywhere above the lower plenum). The discontinuity is likely caused by the binary (yes/no) behavior of the plug which is either holding or failing.

Melting of Support Structures

This mechanism for melt relocation occurs when the reflooding does not provide sufficient cooling to support structures and significant amount of high temperature particulate debris directly contacts and melts lower core support plate or support structures in the lower plenum. The difference in cooling power between large and small relocations is caused by the amount of water circulating through a given cell, which in turn is tied to the amount of debris blocking the water circulation.

For this Mechanism we will compare 2 cases:

- Large Relocation Mass: 182 tons
  - ADS-Valve Opening Time: 1170[s]
  - ECC-System Opening Time: 3940 [s]
  - ECC-System Flow Area: 0.024 [m^2]
- Small Relocation Mass: 1.2 tons
  - ADS-Valve Opening Time: 1250[s]
  - ECC-System Opening Time: 4260 [s]
  - ECC-System Flow Area: 0.020 [m^2]

In both the small and large debris relocation cases (see Figure B.1, Figure B.2 at beginning of the transient, temperature profiles of lower core support plate rings are similar. At ca. 4500s, temperature of the middle ring sharply increased. In the small relocation case, the middle ring was then cooled down, while in the large relocation case, the middle ring temperature continued to increase until the ring failed. Figure B.3 and Figure B.4 show that in the large relocation case, particulate debris contacted directly the lower core support plate until its failure, while in the small relocation case after ca. 5000s there was no direct contact.

MELCOR simulation of the core plate uses the PLATEB-Model and failure of a ring means the immediate relocation of all debris mass supported by that ring and eventually part of the debris on neighboring rings by debris settling.

Since capacity of low pressure water injection system is higher in the large core relocation case, it is very likely that molten and refrozen debris that is relocated to the lower plenum will plug the channels for water to go up and cool down the middle ring of the lower core support plate.

In summary the difference is in placing and heat production/heat conduction of the relocated debris (several tons are not unusual even for “small” relocations), which stops sufficient cooling and causes the supporting structures to collapse.

Places especially vulnerable, to this mode of attack, are the lower core support plate and supporting structures immediately beneath it, which have to deal with conducted heat, as well as “icicle” debris-structures hanging from the lower core plate.

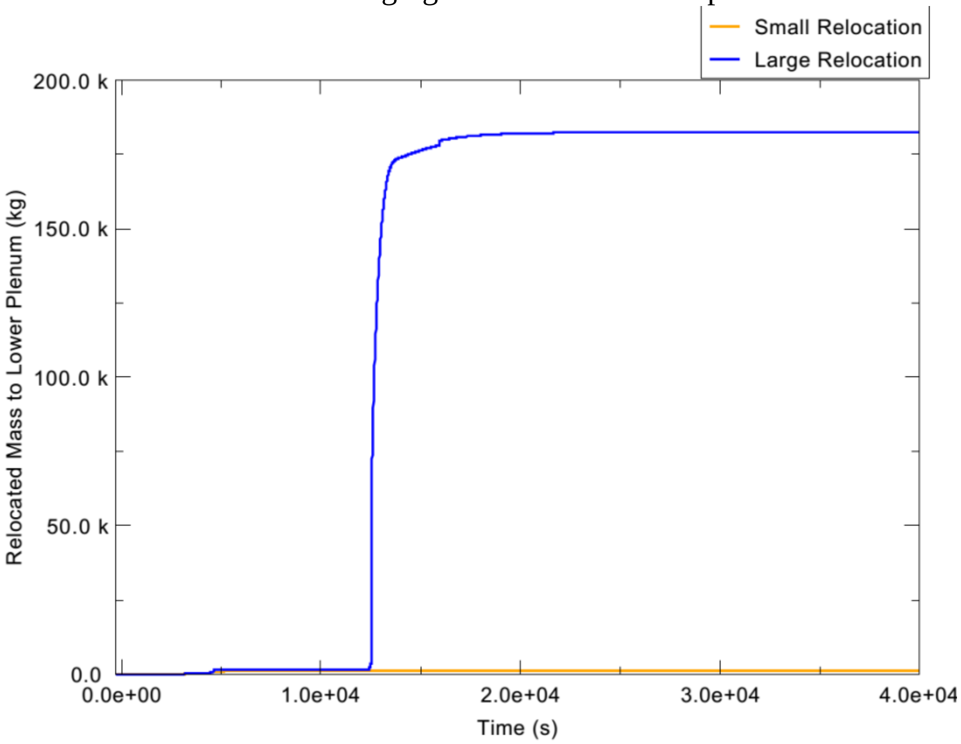

Figure B.1. Relocation Mass to Lower Plenum

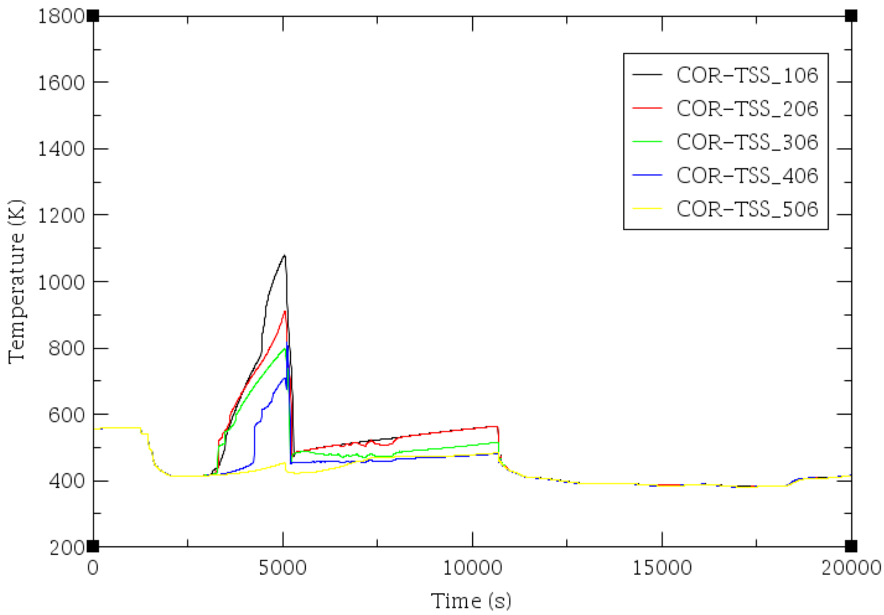

Figure B.2.Temperature of lower core support plate rings - Small debris relocation to the lower plenum

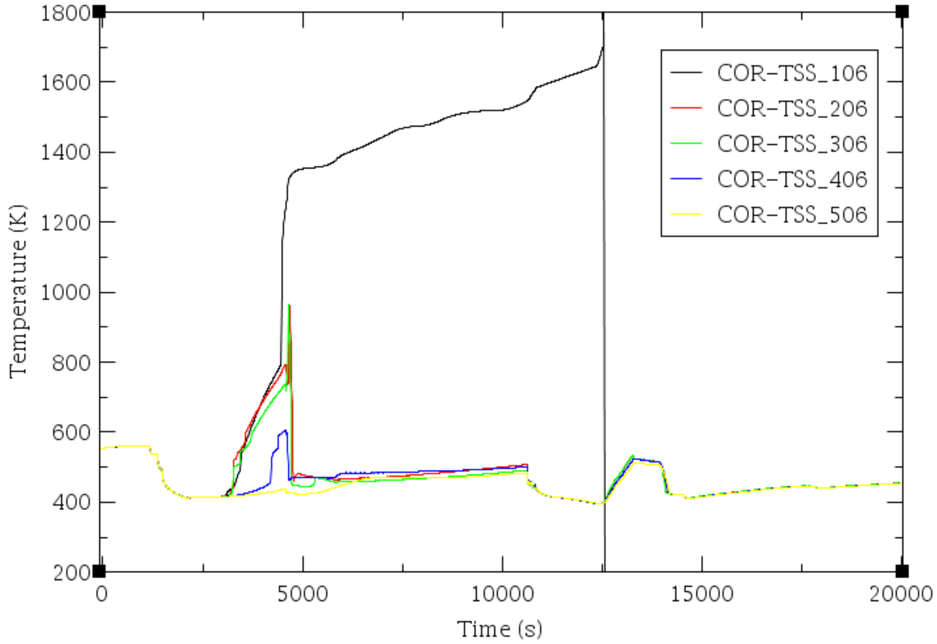

Figure B.3. Temperature of lower core support plate rings - Large debris relocation to the lower plenum

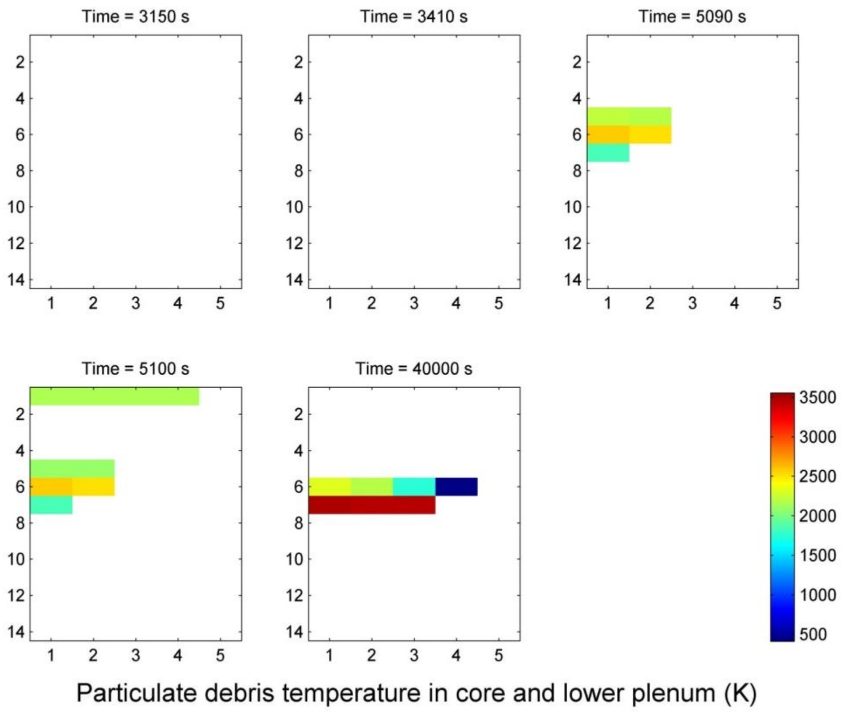

Figure B.4. Small debris relocation to the lower plenum

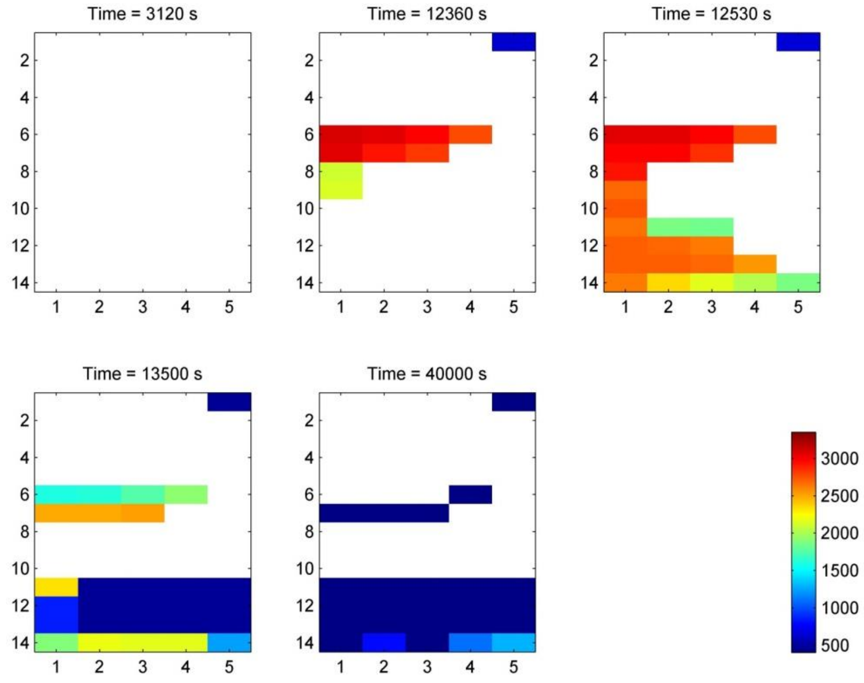

Figure B.5. Large debris relocation to the lower plenum

However, neither mechanism can explain the entirety of all differentiating cases. Numerical effects of the code must also be considered as a possible source. Research in this matter is ongoing.

The code-to-code comparison of MELCOR model with MAAP results will be done for PSA reference scenarios (e.g. HS2, HS3 high pressure/low pressure modes) and, also, scenarios in the "cliff-edge" area (the area with large discontinuity in the relocation masses) to find out if MAAP model produce similar results.

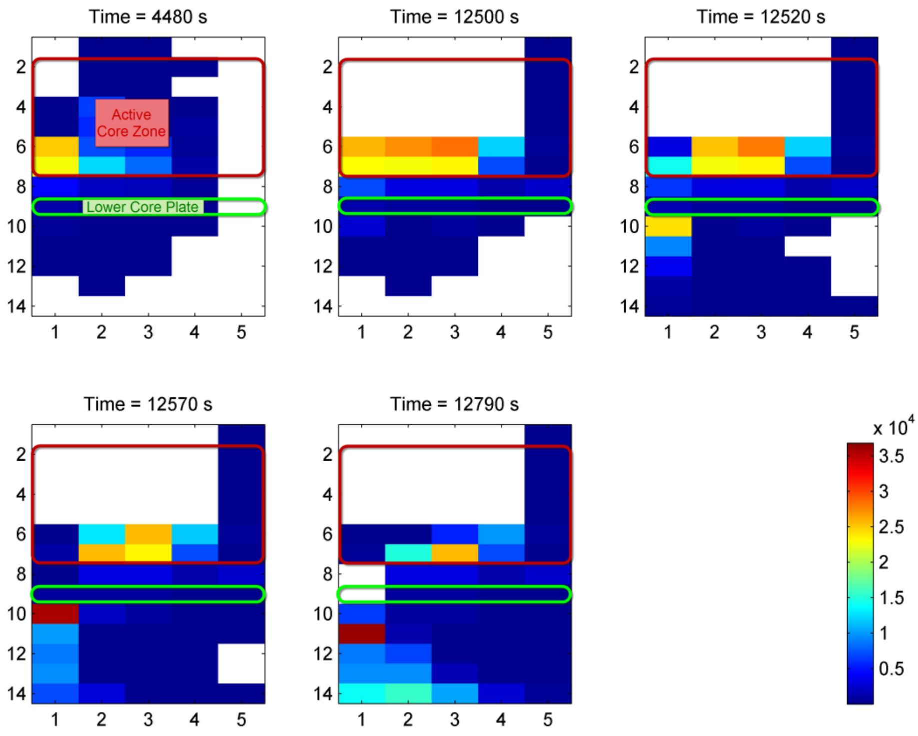

Figure B.6. Large relocation to the lower plenum

Bibliographic Data Sheet	**NKS-315**
Title	Feasibility Study for Connection Between IDPSA and conventional PSA Approach to Analysis of Nordic type BWR's
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Abstract max. 2000 characters	This report summarizes the experience achieved within the NKS-DPSA project during 2013. The project is motivated by the discussions at the Workshop on Integrated Deterministic-Probabilistic Safety Analysis (IDPSA-2012). The aim of the project has been to: summarize the state of the art review of the probabilistic, deterministic and IDPSA analysis; and to carry out a feasibility study. The following areas are covered in this study: mapping, information collection and identification of areas of certain interest based on existing PSA; results of analysis of core relocation scenarios taking into account timing of PSA Level 1 events and possible recovery actions on the melt conditions in the lower head; results of feasibility study on connection between conventional PSA, DSA and IDPSA methods. There are three topics for the feasibility studies identified: a transient with complete or partial failure of the hydraulic scram; station black-out with varying degrees of safety system recovery; steam explosion. Experiences from performed studies are summarized in the report as well as suggestions of areas which need further investigations. The results from the IDPSA show that an increased number of thermohydraulic calculations, performed according to an intelligent algorithm, can improve the understanding of the sequences and therefore input to the PSA or to the deterministic safety analyses. There is a good potential for development of a mathematical model to represent the IDPSA results in form of a decision tree as input for the quantification of the PSA Level 2 structure. Steam explosion analyses exploiting IDPSA methodology would necessitate more detailed approach than VTT's contribution presented here, with use of dedicated analysis codes for FCI phenomena and structural response of the containment. Nevertheless, joint use of MELCOR and SPSA for steam explosion analysis provided a good basis that can easily be refined further.
Key words	IDPSA, PSA, SPSA, DSA, BWR, Severe accident, MELCOR, core degradation, steam explosion.

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