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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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RESULTS AND INSIGHTS FROM LEVEL-2 PSAs PERFORMED IN GERMANY, JAPAN, THE NETHERLANDS, SWEDEN, SWITZERLAND, THE UNITED KINGDOM AND THE UNITED STATES

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- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
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The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonisation of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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Requests for additional copies of this report should be addressed to:

Nuclear Safety Division
OECD Nuclear Energy Agency
Le Seine St-Germain
12 blvd. des Iles
92130 Issy-les-Moulineaux
France

ABSTRACT

Level-2 results of PSAs for 11 PWRs and 10 BWRs in 7 different Western countries are compared, with special emphasis on AM measures for mitigation of the consequences of severe accidents.

The PSA results of interest in this comparison generic or design specific insights in:

- plant design characteristics and provisions for severe accident management (SAM) found important to accident progression in the reactor/containment system and to fission product releases;
- structure of accident sequences;
- capabilities and capacities available to preserve containment integrity in severe accident sequences;
- the fission product retention capabilities of containment designs after loss of containment integrity.

As quantitative basis for the comparison, conditional probabilities of important containment failure modes and of significant and large caesium releases are presented. These permit to assess the containment response to loads typically attending severe accidents, and the containment retention capability, regardless of the absolute level of core damage frequencies.

FOREWORD

The NEA Committee on the Safety of Nuclear Installations (CSNI) believes that an essential factor in achieving their mandate is the continuing exchange and analysis of technical information. To facilitate this exchange CSNI has established various working group. To deal with technology and methods for identifying contributors to risk and assessing their importance, the Committee established Principal Working Group No. 5 - Risk Assessment in 1982. In 1987, “the Committee supported a suggestions that PWG5’s activity should for the moment be primarily focused on PSA Level 1 methods, uses and assessments ... (i.e., to consideration of PSA Level 2 issues where appropriate”.

Over the last 10 years the scope of PSA programmes increased progressively to where today, in many countries, a Level 2 PSA is considered the normal standard. Accordingly, with the advent of increasing use of PSAs, a proposal was made at the 1993 PWG5 Annual meeting for future work in the area of Level 2 PSA. The main objective of the proposed task was to perform a state-of-the-art review of the methods available for performing level 2 PSAs and severe accident/source term uncertainty analyses for use in the regulatory process and the evaluation/implementation of severe accident management strategies. This proposal was accepted by PWG5 and forwarded to the CSNI. The new task was endorsed by CSNI during its annual meeting in 1993.

The overall scope of the task included review current Level 2-PSA methodologies and practices and to investigate how Level 2-PSA can support severe accident management programmes, i.e. the development, implementation, training and optimisation of accident management strategies and measures. The final product is contained in CSNI Report OCDE/GD/(97)198 published in late 1997. For the most part, the presented material reflects the state-of-the-art in 1996.

The information contained within this report reflects (along with three other reports) supplemental material which was prepared in conjunction with the main report. This specific report compares results of Level 2 PSAs results in several Member countries.

Much appreciation and thanks go to the task group members listed below, who provided valuable time and considerable knowledge into this report. Special acknowledgement is given to Dr. Wolfgang Werner, who as an expert consultant provided much of the in-depth technical analysis provided throughout the report as well as many man-hours in editing and compiling the final report.

Task Group Member contributing to the report were:

Cojazzi, G.	Italy	Kersting, E.	Germany	Muramatsu.K.	Japan
Cozzoli, E.	Switzerland	Kim, T.W.	Korea	Murphy, J.A.	United States
Cunningham, M.	United States	Lantaron, J.	Spain	Otero, M.	Spain
Evrard, J.M.	France	Lee, C.J.	Korea	Seebregts, A.	Netherlands
Grant, A.	Canada	Liwang, B.	Sweden	Shepherd, C.	United Kingdom
Herttrich, P.M.	Germany	Meyer, P.	Switzerland	Versteeg, M.F.	Netherlands
Hirose, H.	Japan			Werner, W.	Germany

TABLE OF CONTENTS

1. INTRODUCTION

Level-2 results of PSAs for PWRs and BWRs from Germany /1, 31/, Japan /2,3/, the Netherlands /4,5/, Switzerland 6-9/, the United Kingdom /10.11/ the United States /12 - 27/ and Sweden /28-30/ are compared, with special emphasis on AM measures for mitigation of the consequences of severe accidents. Material from the articles /32-34/ was also used for the preparation of this report. Two sets of results are available for the Sizewell-B plant: the first set (pessimistic analysis) is used in the regulatory process; this analysis involves pessimistic assumptions with significant implications. Another set of results /9,10/ reflects the attempt to remove some of the pessimistic assumptions (indicative analysis). The results presented in this report are based on the pessimistic analysis. For the German plants Biblis-B and Brockdorf (a pre Konvoi plant), level-2 PSAs are not available. However, the containment bypass modes steam generator tube rupture (SGTR) and interfacing systems LOCA (V-sequence) are discussed in the Biblis-B study but without assessing source terms.

The PSA results that are of interest in this comparison are not the bottom line numbers presented in the various studies, but generic or design specific insights in:

- plant design characteristics and provisions for severe accident management (SAM) found important to the progression of severe accidents in the reactor/containment system and to fission product releases

- the structure of accident sequences,

- the capabilities and capacities available at the various plants to preserve containment integrity, given the loads attending significant accident sequences,

- the fission product retention capabilities of the different containment designs in event sequences with loss of containment integrity.

The quantitative basis for the comparison are conditional probabilities of important containment failure modes, and of significant and large caesium releases. This permits to assess the containment response to loads characteristically attending severe accidents, and the containment retention capability, regardless of the absolute level of core damage frequencies.

All statements and assessments in this document apply to the plants included in this comparison. Generalisations with regard to other plants require the consideration of the specific characteristics of these plants.

2. PLANT/CONTAINMENT DESIGN CHARACTERISTICS

Specific plant and containment design characteristics can substantially influence the importance of severe accident phenomena to a particular plant. Parameters such as reactor thermal power, containment free volume, mass of fuel and zircalloy and containment failure pressure, among others, can impact the severity of containment challenges resulting from phenomena such as hydrogen combustion, direct containment heating, gradual pressurisation, etc. A comparison of important plant and containment design characteristics is provided in Table 1 for PWRs and Table 5 for BWRs. Additionally, information is presented in these tables on provisions for carrying out SAM measures for mitigation of accident consequences. Remarks on important design characteristics and provisions for accident consequence mitigation are provided below.

2.1 RCS Water Volume/Power

The ratio of RCS water volume to reactor thermal power influences containment loads and the time available to implement corrective actions during the accident. Because of its being practically constant (between 0.11 and 0.12 across all plants included in the comparison, the parameter is not included in Tables 1 and 5.

2.2 Reactor Power/Containment Volume

The ratio of reactor power to containment volume provides a measure of the likelihood of the containment to be significantly challenged by severe accident loading. The plants with higher ratios should experience higher containment loading for similar severe accident situations. The ratio is lowest for PWRs with large dry containments (typically below 0.05). For BWRs with Mark III containments, this ratio is typically in the range 0.08 - 0.1. For the BWRs with Mark I containments, the ratio typically is about 0.5. Exception among plants with Mark I containments are Mühleberg and Dodewaard, which have ratios 0.21, respectively, 0.24.

2.3 Fuel Mass to Containment Volume

The ratio of fuel mass to containment volume provides a measure of the potential for direct containment heating effects. Due to their small containment volume, the US-BWRs with Mark I containments have high ratios (~ 19). Among plants with Mark I containments, Mühleberg is an exception with ratio 9.5, and Dodewaard is also in the lower range with 14. For BWRs with Mark III containments the ratio is about 4, and for PWRs with large dry containments the ratio is generally below 2.

2.4 Average Hydrogen Concentration at 30oC, Dry Atmosphere, Given 100% Zirconium Oxidation

The average hydrogen concentration provides a measure of the potential for and the loads resulting from hydrogen combustion events, including the contribution from hydrogen combustion to direct containment heating. For the PWRs the values are in the range 10% to 20%, with the highest values for the Combustion Engineering and Siemens KWU plants which had relatively more zirconium in the core at the time the studies were done than the other plants. For cores reloaded with other fuel, the percentages may be lower or higher than those reported in Table 1.

The percentage is substantially higher in BWR containments. Since these containments are inerted, they are not susceptible to hydrogen combustion events, but the pressurisation due to the generation of non-condensable gases is still a concern for plants with Mark I containments.

The assumption of 100% zirconium oxidation is made to have a point of reference. According to present understanding, oxidation of 50% - 60% of the zirconium has to be expected during the in-vessel phase of core degradation. 100% oxidation would only be reached in the ex-vessel phase, if - after failure of the RPV - the core debris is not coolable.

2.5 Estimated Pressure due to Quasi-static Hydrogen Combustion, Given Dry Atmosphere and 100% Zirconium Oxidation

The pressure due to a complete quasi-static combustion of the hydrogen, generated by 100% zirconium oxidation in a dry containment, is one of several parameters and design characteristics that, together, determine the potential for containment failure due to hydrogen combustion. Other important parameters are, for example, the steam content in the containment atmosphere and the availability of provisions for the controlled removal from the containment atmosphere of hydrogen. The estimated pressure due to hydrogen combustion is below, or just at the estimated mean failure pressure for all plants but Biblis-B and Maine Yankee. For these two plants, the value of the parameter is above the estimated mean failure pressure.

At Maine Yankee the containment spray is operating in ~ 85% of the core damage sequences, leading to increased likelihood of hydrogen combustion. Of all the plants for which containment failure modes are quantified, Maine Yankee shows the highest likelihood of failure due to hydrogen combustion.

At Biblis-B there is steam inertisation in most severe accident situations. This substantially reduces the likelihood of significant hydrogen combustion. Also at Biblis-B, like at all German PWRs, hydrogen control by a combination of igniters and catalytic recombiners is foreseen. As for assuming 100% zirconium oxidation, see the discussion in the preceding paragraph.

2.6 Estimated Mean Failure Pressure of the Containment

The estimated mean failure pressure provides a measure of the ability of the containment to survive severe accident challenges. PWRs with large dry prestressed concrete containments and BWRs with Mark I containments have the highest failure pressure (~ 10 bar or higher). The failure pressures for the PWRs with steel containments is around 8 bar, and for BWRs with Mark III containments the failure pressure is about 5 bar. For the Dodewaard plant with a Humbold Bay containment (pre-Mark I), the failure pressure is about 7 bar.

2.7 Power to Suppression Pool Water Volume at BWR Plants

The ratio of reactor power to suppression pool water volume provides a measure of the likelihood of significant challenges by severe accident loads to BWR containments. Plants with low ratios have a greater capability to remove decay heat and, therefore, are expected to experience lower containment loads than plants with high ratios (for similar severe accident situations). This ratio is typically ~1, with the exception of Mühleberg and Dodewaard, for which it is ~ 0.5.

3. PROVISIONS FOR MITIGATION OF SEVERE ACCIDENT CONSEQUENCES

3.1 Pressurised Water Reactors

3.1.1 Containment Spray

Heat removal by containment spray is available at all plants but the Siemens/KWU plants. Its operation in the event of severe accidents may have positive and negative effects:

- Positive: Removal of decay heat from the containment and removal from the containment atmosphere and deposition in the containment of fission products released to the containment during severe accidents.

- Negative: Removal of steam from the containment atmosphere by condensation. This reduces the steam inertisation of the containment, thereby increasing the likelihood of hydrogen combustion. With the fuel loaded at the time of the studies, this is a significant concern only at the Combustion Engineering plant Maine Yankee which is vulnerable - due to its high amount of zirconium in the core - to hydrogen combustion. At the other plants it could become a concern if reloaded fuel had thicker fuel rod cladding.

At Borssele, containment spray is only used for fission product depletion in severe accidents

Additional water injection from the internal fire water system or externally from fire trucks can be used for backing up the water supply to the containment spray system at Beznau, Borssele, Sizewell B and at all Swedish PWRs.

3.1.2 Hydrogen Control

Hydrogen control by a combination of igniters and catalytic recombiners is foreseen at Biblis-B (and other German PWRs). Presently, the design, localisation and composition of these devices is being optimised. At the Swedish PWRs, catalytic recombiners that are qualified for severe accident environment are available, although these plants are not particularly vulnerable to hydrogen combustion. At Beznau, early containment venting for removal from the containment atmosphere of hydrogen and oxygen will be implemented. At Borssele early venting is under study, as well as combinations of recombiners with igniters, and post- accident inertisation. For Sizewell-B, hydrogen control is achieved by mixing the hydrogen produced in the whole of the containment atmosphere in the short term and using hydrogen recombiners in the longer term. The hydrogen mixing is carried out by mixing fans assisted by the containment sprays and coolers. The hydrogen recombiners are only designed for the post LOCA duty. Recombiners for controlling the hydrogen generated in design basis accidents are available at all plants.

3.1.3 Additional Water Injection to the Containment

Additional water injection to the containment can be used for prevention of core damage and for mitigation of the consequences of core damage.

- Prevention: Providing backup water sources for

    - low pressure injection/recirculation (Swedish PWRs, using fire trucks (CWIS))

    - containment spray (Beznau and Swedish PWRs, with injection from fire trucks, Sizewell B with injection from the fire water system)

    - cooling of containment fan coolers (Beznau, using river water and mobile pumps)

- Mitigation: Water supply for flooding of the containment when core damage and possible RPV failure is imminent. By having a deep water pool underneath the reactor vessel, the extent of basemat attack by molten core debris can be reduced or basemat attack may even be prevented, thus reducing or eliminating the production of combustible gases, as well as the likelihood of basemat penetration. To avoid late overpressurisation failure of the containment due to steam production, this strategy is likely to require the availability of high capacity filtered containment venting.

Procedures and hardware for implementing containment flooding and filtered containment venting are available at Beznau and at the Swedish PWRs. At Sizewell B, the fire water system can be used for flooding the containment.

On the negative side of the containment flooding strategy is the possibility, although very remote according to present understanding, of containment failure due to massive ex-vessel steam explosions resulting from the contact of the molten core debris with the water.

3.1.4 Depressurisation of the RCS for Prevention of High Pressure Melt Ejection

Depressurisation of the RCS for prevention of high pressure melt ejection (HPME) is available at the plants for at least some sequences involving loss of steam generator feeding; at some plants for nearly all such sequences. By the application of the depressurisation procedure it is intended to prevent phenomena accompanying RPV failure that could threaten containment integrity.

3.1.5 Filtered Containment Venting

In the event of pressure build-up due to ex-vessel production of steam and non-condensable gases, or combustion of flammable gases, the failure pressure of the containment may be exceeded in the late phase of an accident. By filtered containment venting prior to critical pressure build-up, catastrophic failure of the containment can be avoided. For conducting filtered containment venting, provisions have to be made for avoiding hydrogen detonations in the filter and its connecting lines. The likelihood of such events could become significant due to condensation phenomena that reduce steam inerting. At Beznau, design modifications of the venting system have been implemented that are believed to reduce this likelihood to insignificant**.** Filtered containment venting is implemented at all German and Swedish PWRs, Beznau and Borssele. It is very beneficial in combination with the strategies for having large quantities of water available for debris quenching (available at Beznau and the Swedish plants). For Sizewell B, provision was made in the design to include a filtered containment venting system and the PSA was used to consider the benefit in term of risk reduction from the system. It was not incorporated since it was concluded that it would not be cost-effective.

3.1.6 Use of Primary Side Bleed/Feed in the Event of Steam Generator Tube Rupture

In the event of core damage involving steam generator tube rupture with unisolated steam generator, the release to the environment of the volatile fission products, including noble gases, can be significantly reduced if primary side bleed/feed (PB/F) is applied: through the split-up of the mass flow between the broken steam generator tube (few cm^2) and the open pressuriser valves (40 to 60 cm^2), most of the fission products released from the core are directed to the containment. Calculations performed for phase B of the German Risk Study (DRS-B) have shown the potential for significant reduction of the releases. Further analysis are needed for the scenario after vessel failure.

The strategy is in place at Biblis-B, the Swedish PWRs, Beznau, Borssele, Sizewell B and many US PWRs.

3.1.7 Filling with Water of an Unisolated Steam Generator in the Event of Steam Generator Tube Rupture

The releases from a ruptured tube in an unisolated steam generator can be drastically reduced by scrubbing of gases in a column of water in the defective steam generator. This option is available as accident management action at Beznau, Borssele, Sizewell-B and at the Swedish PWRs. At Beznau, Borssele and Sizewell-B, fire water can be used to fill up the defective steam generator. At the Swedish plants, the optimal strategy is still under investigation. In the Sizewell B analysis, no credit is taken in the PSA for this action.

The scrubbing effect strongly depends on the height of the water column above the break. For U-tube steam generators, analyses consistently show that the likelihood of leaks is highest in the lower part of the steam generator.

In the Ringhals-2 analysis, the overall reduction of caesium releases for events with unisolated steam generator amounts to a factor about 100. In the Beznau analysis a reduction by the factor 10 - 100 of iodine and caesium releases is assumed for sequences with unisolated SG. In DRS-B, the achievable reduction is estimated to be in the same range.

3.2 Boiling Water Reactors

3.2.1 Containment Pressure Relief System

The high capacity containment pressure relief system without filtering is used for alternate heat removal in sequences that include failure of the pressure suppression system, but operability of the normal ECCS systems. As the request for pressure relief will occur when the core is not yet damaged, filtering is not required. The system is implemented at most Swedish and US BWRs plants. In addition to using this system for alternate heat removal, it is also used for overpressure protection.

3.2.2 Water Pool Underneath the Core Without Supply from External Source

To avoid attack by core debris of the drywell liner or of the concrete containment structure that could ultimately lead to the penetration of the containment barrier, the availability of a water pool of sufficient depth under the RPV would be beneficial in the event of RPV failure. Provisions have been made at many Swedish BWRs for utilisation of the freshwater reservoir and/or the firewater system, or the water volume of the condensation pool as water source for flooding of the reactor containment in the event of a severe accident.

At the Mühleberg plant a very large in-pedestal sump volume is available by design (about 5 times as large as in US plants with Mark I containment), which can accommodate twice the debris volume. Additional provisions for flooding of the containment are also available at this plant.

At Forsmark 3, at least two systems are available that can be used for flooding the lower drywell: (1) a dedicated system (system 363), and (2) the containment vessel spray system. In addition, water can be added to the drywell via coolant systems, should they be available after vessel breach.

3.2.3 External Water Injection System

External water supply for safety systems and for accident mitigation is available at the Swedish BWRs and at Mühleberg. It can be used to provide additional suction sources for:

- high pressure auxiliary feedwater (Ringhals 1)
- backing up the water supply to the containment spray system (all Swedish BWRs, Mühleberg)
- accident mitigation by flooding of the containment to the upper core level, ensuring stable terminal cooling of core material (all Swedish BWRs, Mühleberg)

3.2.4 Filtered Containment Venting

To avoid a breach of containment integrity due to a slow pressure increase following an accident, systems for filtered containment venting have been installed at all Swedish and German BWRs and at Mühleberg. The filtered containment venting system is used for alternate heat removal (preventive AM) and for overpressure protection of the containment (mitigative AM).

3.2.5 Combination of Containment Flooding and Filtered Containment Venting

The combination of:

- having or making available large quantities of water underneath the core in the event of a severe accident (to avoid attack of the drywell liner or other containment structures) and
- having available high capacity filtered containment venting

appears to be most beneficial for avoiding large releases to the environment, see Table 8. This table shows that Mühleberg, Forsmark 3 and Barsebäck (which are equipped with this feature) have the lowest conditional probabilities for Cs releases of all examined plants (at Mühleberg, this is also due to the large in-pedestal sump volume which practically eliminates drywell liner attack in the event of debris pour).

4. TREATMENT OF PHENOMENOLOGICAL PROBLEMS IN THE MODELLING OF CORE DAMAGE PROGRESSION AND OF FAILURE MODES OF THE REACTOR COOLANT SYSTEM AND CONTAINMENT

With a plant damage state as starting point, the progression of the accident in the reactor/containment system is traced to end states which permit to estimate the consequential release of radioactive substances. The individual steps treat the process of core degradation inside the reactor pressure vessel, the destruction of the reactor pressure vessel or other parts of the reactor coolant system, as well as the processes outside the reactor pressure vessel in the containment. In various steps and varying degree of detail, the impact of the events on the containment structure, as well as the possible containment failure modes are examined. The events take place in the period between the onset of the core damage and either the termination of the accident progression without significant offsite consequences, or the failure of the containment.

This section surveys the treatment of important phenomenological issues in the examined studies. The codes used in the examined studies for the analysis of the various phenomena are compiled in reference /35/.

The primary cause of the threat to the containment is the nuclear decay heat. With the progression of the accident, the consideration of the partitioning of the decay heat into volatile and less volatile fission products becomes important. The first are released from the melting corium and can henceforth only contribute to the heat-up of the containment atmosphere, whereas the second heat up the corium. From the melting corium the heat can be transferred through the evaporating coolant or, after failure of the reactor pressure vessel, directly to the containment atmosphere.

An additional source of heat is the possible recriticality after reflooding of a partly destroyed core with unborated water.

Further major sources of heat are the zirconium-steam reaction $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$ of the fuel rod cladding, and the possible hydrogen deflagration $2H_2 + O_2 \rightarrow 2H_2O$. The first reaction sets in with increasing core heat-up at about 1000 ° C and becomes strongly exothermic beyond 1200° C, releasing at least the same amount of heat as the nuclear decay, thus accelerating core destruction. The second reaction takes place in the containment atmosphere and heats up the atmosphere and structures of the containment in the event of a hydrogen deflagration. The loads resulting therefrom can be magnified by transition to detonation.

The pressure in the containment can also rise through increase of the gas mass. Steam from evaporated coolant and from the corium-concrete interaction, hydrogen from the zirconium-steam reaction and other non-condensable gases from the core-concrete interaction all can contribute to the pressure build-up.

In the studies, the following containment failure modes are considered:

- slow overpressure failure caused by steady increase of temperature and gas masses in the containment atmosphere
- rapid overpressure failure caused by steam explosion, hydrogen deflagration, direct containment heating
- underpressure failure caused by condensation of steam
- temperature induced failure of containment systems and structures
- basemat erosion caused by core-concrete interaction
- mechanical damage through missiles

Besides these and possible further specific containment failure modes, accident sequences with containment bypass and with defective containment isolation are examined in all studies. These are:

- steam generator tube rupture in PWRs
- streamline breaks outside the containment in BWRs
- breaks and leaks between the high pressure reactor cooling system and connecting low pressure systems with components outside the containment (V-sequence).

4.1 Generation of Hydrogen during the In-Vessel Phase of the Core Destruction

With increasing core heat-up, the zirconium of the fuel rod cladding begins to react exothermally with vapour. The percentage of the zirconium being oxidised depends on several interrelated factors which are difficult to model, for example:

- blockage of flow paths through the core due to temperature induced loss of geometry,
- formation of oxide layers on the fuel rod surface,
- reduction of vapour flow by reactions at upstream locations,
- availability of water, for example, as a result of reflooding of the core.

Per 1000 kg oxidised zirconium, 44 kg hydrogen are generated and released: i.e., with 100% oxidation in a 1000 MWe PWR with 20 000 kg zirconium, 880 kg hydrogen, and in a 1000 MWe BWR with 65 000 kg zirconium, 2 800 kg hydrogen.

Basis for the assessment are in NUREG -1150:

- Calculations with the program systems MELPROG, SCDAP, CORMLT, MAAP, MARCH, BWRSAR and APRIL, as well as evaluations of experiments and of the TMI-accident. For PWRs and BWRs a number of typical cases have been defined, characterised by various pressure ranges and time scales, with or without flooding of the core.

- For the assessment of the available information, formalised expert opinion elicitation has been performed. Subjective probability distributions for the percentage of zirconium oxidation have been generated for each defined case by the experts, based on their knowledge and experience with calculations and experiments. The individual distribution functions have been aggregated with equal weights for each case to one distribution function, which then was used in the quantification process.

- As far as the experts had experience with several of the above listed program systems, MAAP and MARCH were rated lower than the others, because MAAP was considered to underestimate the zirconium oxidation, and MARCH to overestimate it.

- For the various PWR cases, the median values of the aggregated distribution functions are between 30% and 50%, the 95% percentile between 65% and 95% zirconium oxidation. The differences among the evaluations by the 5 experts are large. For the case with the largest differences of opinion, the smallest median value is 20%, the largest 70%; in the best case, the smallest mean value is 20% and the largest 35% zirconium oxidation.

- For the various BWR cases, the median values of the aggregated distribution functions are between 10% and 25%, the 95% percentiles are between 35% and 50% zirconium oxidation. The differences among the evaluations by the four experts are large as well. For the case with the largest differences of opinion, the smallest mean value is 0%, the largest 35%, in the best case the smallest mean value is 18% and the largest 30% zircon oxidation.

In the IPE-studies:

- Results of calculations with the program MAAP, were adapted to the special circumstances at the plant, evaluation of separate effect tests and of the TMI-accident.

- In the IPE for the H.B. Robinson PWR nominal values are being used, which correspond to the median values in NUREG-1150, in the Maine Yankee PWR-IPE the nominal values are in the in the upper range of the distribution functions of NUREG 1150. They are generally higher that in the other PWR studies.

- In the Browns Ferry BWR IPE and the Perry BWR-IPE, nominal values are being used, which correspond approximately to the median values of the distribution functions of NUREG-1150.

- In the studies for the Swiss plants results from MELCOR calculations were used

Obviously, the uncertainties attending the modelling of in-vessel zirconium oxidation are large. The extent to which such uncertainties affect the assessment of the containment integrity, depends on the:

- ratio containment volume/amount of zirconium in the core,
- containment failure pressure,
- vapour content in the containment atmosphere, and
- availability of provisions for prevention of hydrogen combustion.

4.2 Arrest of Core Melt Progression

If the progression of core destruction can be arrested before the reactor pressure vessel is breached, loads that can threaten the integrity of the containment can result only from the combustion of hydrogen being generated in the zirconium-steam reaction of the cladding material of the fuel rods. Whether this severely threatens the containment integrity depends on specific design features of the plant.

The progression of core destruction can be terminated, if emergency cooling functions can be restored in time. This can be achieved by restoring the electrical energy supply in the event of station blackout sequences, or by depressurisation as consequence of passive failure of parts of the primary coolant system, and following activation of emergency core cooling.

If the progression of core destruction can be arrested, the accident sequence ends in an accident progression bin characterised by "no penetration of the reactor pressure vessel". For such categories the probability of containment failure is essentially determined by the possibility of hydrogen combustion. According to the available documentation, the ability to model the core destruction process is not satisfactory. Therefore, not too much credit should be given to the conditional probabilities of the accident progression bins "no penetration of the reactor pressure vessel".

4.3 Temperature Induced Structural Failure of the Reactor Coolant System at Locations other than the Bottom Head of the Reactor Pressure Vessel in High Pressure Accident Sequences in PWRs

It is expected that the failure of the reactor coolant system at locations other than the bottom of the reactor pressure vessel leads to less severe loads of the containment, than the failure at the bottom of the RPV. Possible mechanisms that can lead to depressurisation before bottom head failure of the reactor pressure vessel are:

- temperature induced failure of the hot leg of the primary coolant circuit,
- temperature induced failure of the surge line,
- temperature induced failure of steam generator heating tubes,
- temperature induced failure of the main coolant pump seals,
- stuck open pressuriser relief valves.

The following analytical and experimental basis for the modelling of possible depressurisation mechanisms are used:

- In NUREG-1150: Calculations with the programs MELPROG, SCDAP, CORMLT and MAAP, as well as the evaluation of EPRI natural convection experiments and of the TMI-accident. Significant uncertainties are associated with the available calculated results. A formalised expert opinion elicitation with three experts has been performed for the evaluation of the information. It can be concluded from the aggregated distribution functions for the conditional possibilities of occurrence of the four considered failure mechanisms, that in case of high pressure in the reactor coolant circuit, corresponding to the pressuriser valve's set point, failure of the hot leg is most likely (medium value of the failure probability > 0,95) followed by failure of the pressuriser valves and the main coolant pump seals. However, better corroboration of the quantitative results is desirable.

- In the IPE studies, calculations with the MAAP-program, the qualification of which has been based on the assessments performed in NUREG-1150. In the H.B. Robinson PWR-IPE a probability of 0.9 is used for the failure of the hot leg, and in the Maine Yankee PWR-IPE a probability of 0,75 is used for the aggregate of all failure mechanisms but bottom head failure.

- In the study for the Beznau plant, the conditional probability (mean) for hot leg/surge line rupture is 0.7 for transients and 0.03 for SLOCA events. The conditional failure probability for hot leg/surge line rupture in Beznau was derived using plant-specific calculations (using SCDAP/RELAP5 and MELCOR), and other evidence including TMI-2 and the utility MAAP analysis.

- In the Sizewell B and Borssele analyses, the MAAP code was used. The conditional failure probabilities for temperature induced hot leg/surge line failure are 0.9 for Sizewell B and 0.73 for Borssele.

4.4 Loads Attending Pressure Vessel Failure under High Pressure for PWRs

If the bottom of the of the reactor pressure vessel fails while the system is under high pressure, a large amount of molten core and structure materials, water, steam and hydrogen are being ejected into the containment.

- The attending pressure rise in the containment atmosphere results from the superposition of several effects:

- blow down of steam and hydrogen

- combustion of hydrogen

- interaction of the molten core material with water on the containment floor

- transfer of heat from finely dispersed corium to the containment atmosphere

- The following parameters are regarded as important for the analysis:

- pressure in the reactor pressure vessel

- amount of unoxidised metal in the core

- amount of ejected molten core material

- size of the hole in the reactor pressure vessel

- presence of water in the reactor cavity

- availability of the containment spray system

There are significant uncertainties about some of these parameters.

In NUREG-1150 the containment loads were estimated by expert opinion elicitation on the basis of engineering judgement. In the IPE-studies for PWRs and in the Ringhals 2, Sizewell B and Borssele analyses, calculations were performed with the MAAP program system. The qualification of the corresponding program parts takes into account the NUREG-1150 results.

The uncertainties about the containment loads appear to be relatively unimportant in the evaluation of the integrity of the strong prestressed concrete containments of the plants Surry, Zion, Maine Yankee, and H.B. Robinson that are expected to survive the loads from DCH with high conditional probability. For the Beznau plant, the conditional probability of surviving DCH loads is also high.

4.5 Erosion of the Containment Basemat in PWR Containments

The speed and extent of the concrete erosion by molten core material strongly depend on the split-up of the decay heat in the portion that is consumed by the destruction of concrete and the portion that is consumed by the evaporation of water and the heat-up of the containment atmosphere. Among the various computational models, there exist major differences regarding this split-up. However, there is agreement that the uncertainties about a possible penetration of the basemat do not significantly influence the accident consequences outside the plant.

4.6 Generation of Hydrogen Outside the Reactor Pressure Vessel in PWR Containments

The hydrogen generation rate in the ex-vessel phase in a core melt accident depends on the coolability of the molten material, respectively, on the interaction between molten material and the concrete of the reactor basemat, if the molten core is not coolable.

The coolability depends on the mode of reactor pressure vessel failure and the quantity of water present in the reactor cavity or in other parts of the containment. Significant concrete destruction by molten core material is unlikely, if:

- the molten core material is sufficiently spread out on the containment floor, and

- a sufficiently deep water pool is available at the time of failure of the reactor pressure vessel.

If there is no water available and, thus, the molten material not coolable, the hydrogen generation rate depends mainly on the speed and intensity of the interaction between the molten core material and concrete. As stated in the section dealing with the destruction of the basemat, there are major modelling uncertainties.

4.7 Containment Failure Modes under Static Overpressure

The response of the containment to possible static over pressure loads attending core damage is examined in all studies. Distinction is being made between:

- Slow pressure rise, i.e. due to long-term production of steam and non-condensable gases.

- "Fast" pressure rise, i.e. events which happen fast from the thermodynamic viewpoint, but are associated with quasi-static load changes from the structural mechanics viewpoint. Examples are direct heating of the containment atmosphere attending the failure of the reactor pressure vessel under high pressure, and hydrogen deflagration (as opposed to hydrogen detonation).

The distinction is important, because in the case of low pressure rise, which, in general, first leads to a leakage, a further pressure rise may be prevented by the developing leak. In the event of fast pressure rise, however, a further pressure rise is possible after the formation of a leak, eventually leading to larger breaks or even to catastrophic loss of integrity. Numerous investigations and calculations with structural mechanics computer codes form the basis for the evaluation of strength and of failure mechanisms. In all studies, these calculations are used to generate distribution functions expressing the probability of failure modes as function of pressure.

4.8 Melt-through of the Drywell-Shell in Mark I Containments of BWRs

An early failure of Mark I containments can result from the penetration by molten core material of the steel shell of the containment. In the analyses for the plants Peach Bottom and Browns Ferry (and many other BWRs with Mark I containment) it is the dominant contribution to large scale early containment failure. This failure mode can lead to severe consequences outside the plant. In NUREG-1150, the assessment of this failure mode is based on subjective probability distributions, which were generated by experts on basis of engineering judgement. The estimates of the 6 members of the experts group were nearly binary, i.e. 3 of the experts believed that drywell melt-through would always take place, and 3 believed that it would never occur. The individual assessment was almost independent of the initial and boundary conditions. The aggregated probability distribution depends mainly on the composition of the expert group. The mean conditional probabilities for melt-through developed in NUREG-1150 (ranging from 0.3 to 0.8) have been adopted in Browns Ferry and other IPEs. Because in such accident sequences, severe consequences outside the plant are combined with large uncertainties, investigations directed at improved understanding of the important phenomena were performed after the publication of the mentioned studies. Now available research results /28/ suggest that containment failure due to liner attack can be avoided by the presence of large quantities of water in the pedestal area. Severe accident mitigation strategies directed at having large quantities of water available in the area underneath the core have since been implemented at many plants.

5. METHODS FOR THE ANALYSIS OF CONTAINMENT RESPONSE TO SEVERE ACCIDENT LOADS

Level-2 PSAs examine the responses of the containment and of its engineered safety systems to the loads attending core damage accidents. Input to the analysis are the frequencies of occurrence of a set of defined failure states of the reactor system and the status of active containment systems (plant damage states, PDS), that are produced in the preceding level-1 PSA. Results of the level-2 PSA are containments failure states (accident progression bins, APB) and their frequencies of occurrence, as well as the attendant releases to the environment of radioactive substances.

5.1 Plant Damage States

Plant damage states are used to establish the level 1/level 2 interface between plant systems and containment analyses. PDS development involves the grouping of accident sequences into classes having similar plant and containment response characteristics, thus providing initial and boundary conditions for the containment analysis. As can be seen from Tables 2 and 6, among the PSAs included in this study, the number of PDSs with occurrence frequencies > 10^{-7}/a varies from 6 to 26.

Criterion for the grouping into PDSs of the level-1 accident sequences is the similarity of the failure states with regard to the

- further progression of the accident in the reactor system
- functionality of the active containment systems
- response of the containment structure to the attendant loads

The PDSs are described by vectors of characteristics (numerical values, respectively, attributes), the number of which differs greatly among the examined studies, see Tables 2 and 6. Mainly for the presentation of results, the PDSs are further condensed into summary PDS groups, SPDS, in some of the studies.

5.2 Containment Event Trees/Accident Progression Trees

The logical structure for the modelling of the progression of the accident in the reactor system and in the containment is provided by containment event trees (CET). In some studies these are termed "accident progression trees" (APET). In a structured approach, the interdependent physical-chemical processes are traced that are relevant to the integrity and retention capability of the containment. The questions asked at the branch points of the event paths are ordered chronologically; they characterise the various possibilities of accident progression inside the containment. The quantification of the branching probabilities addresses the availability of containment systems, as well as the physical phenomena. It provides a conditional probability for each accident path, originating from a PDS, and ending at an APB. The logical structure of the CETs is analogous to that of the system event trees in level-1 analyses.

Two main methodologies are employed for the development of the CETs:

- The large CET, which contains virtually all top event questions regarding the specifics of severe accident modelling; and

- The small CET method, which includes top event questions concerning the major severe accident phenomena, which are then supported by fault trees.

The NUREG-1150 studies employed the large CET method, whereas the IPEs and the Swiss and Swedish studies use the small CET method. In principal, neither method is more accurate or complete, but the small CET method is much more traceable, and considerably easier to review.

The number of branch points differs greatly in the examined studies, it varies between 9 and 145, see Tables 2 and 6. For the quantification of the branching probabilities, calculations and analyses of varying complexity are performed, using mechanistic computer codes, parametric codes and engineering judgement (for phenomenological questions), as well as systems analysis codes (for questions of availability of systems). The number of branch points, by itself, is not a measure for the depth and degree of detail of a level-2 investigation. With a compact CET having the essential questions of accident progression in the event tree, and associated fault tree-like analyses, the same analysis quality and completeness can be obtained, as with very complex and large CETs.

The questions asked at the branch points of the CETs often are of global nature, i. e. "amount of zirconium oxidised in the pressure vessel?", "amount of core material, that is released into the containment following failure of the reactor pressure vessel?", "is the molten material coolable?" There are many instances where such questions are not answered by using mechanistic computational models. Instead, engineering judgement is used to assign subjective probabilities to the branch points of the CETs. When mechanistic models are not directly used, the dependency of the results of the accident progression analysis on the underlying physical phenomena is often hidden.

The information that is available to model and quantify the progression of accidents consists of a variety of research results including numerous calculations with computer programs that model special important aspects of the accident progression, as well as experimental results.

The flexibility and the generality of the CET method makes it a powerful tool for conducting level-2 analyses. The possibility of defining questions at various analysis levels makes the CET efficient. At the same time, however, this possibility has to be used with caution, especially when it is applied to questions with a poor knowledge base.

5.3 Accident Progression Bins

The result of the CET analysis is a large number of different containment states. Analogous to the development of PDSs, accident progression bins, APB, are used to group CET end-states based on similarity in fission product release characteristics. Since the number of CET end-states typically is very large, this step is necessary in order to reduce the level of effort required for calculating representative source terms. Again from Tables 2 and 6, it is seen that, for the studies included here, the number of APBs ranges from 4 to 25.

Criterion for the binning into APBs of the containment event sequences is the similarity of the containment failure states with regard to the:

- possibilities of release of radioactive substances

- offsite consequences of releases of radioactive substances.

Like the PDSs, the APBs are described by vectors of characteristics that can, in general, take on several numerical values, respectively, attributes.

Following the development of the APBs, appropriate source term magnitudes must be obtained. The typical methods for calculating source term magnitudes are:

- to use plant-specific calculations with simplified parametric codes, e.g. the MAAP code or the ERPRA code , or

- to adapt source terms based on similar severe accident progression sequences from accepted level-2 PSA studies for similar plants (e.g. the NUREG-1150 studies).

Due to the large number of APBs, for those studies that involve plant-specific source term calculations, calculations can only be performed for a portion of the APBs. To the remaining APBs, source terms are assigned based on similarity to one of the APBs for which calculations have been performed. Another approach is to use simplified parametric codes that permit to trace each plant damage state explicitly through the in-vessel and ex-vessel phases of accident progression.

5.4 Coupling of the Analyses Steps

In the examined studies, the significance of the PDSs and APBs for the coupling of the analyses steps is different:

- In the NUREG-1150 and in the Swiss studies, each accident sequence is traced explicitly from the initiating event through the stations:
  - response of safety systems
  - plant damage states
  - accident progression in the reactor / containment system
  - accident progression bins
  - release categories
  - offsite consequences

  The coupling elements PDS and APB only serve as interfaces for passing on information to the subsequent analysis step, and they are used for a transparent and systematic presentation of results.

- In the IPE studies and in the Ringhals 2, Sizewell B and Borssele analyses, the PDSs and APBs are used to condense the numerous results of the preceding step into a few representative elements of the PDSs, respectively, APBs. Only these are used as initial and boundary conditions for the subsequent step. This greatly reduces the computational expenditure, however, at the loss of information.

6. RESULTS OF THE ACCIDENT PROGRESSION/CONTAINMENT ANALYSES

6.1 Assessment Basis

The essential information with relevance for the comparison of level-2 results explicitly provided in the studies, are:

- total core damage frequencies
- contributions of the important groups of initiating events to the total core damage frequencies
- containment failure modes and their frequencies and, in some studies, conditional probabilities of their occurrence
- description of the methodology.

Suitable parameters for the integral assessment of the retention capabilities of the containments are the conditional probabilities of various containment failure modes, given core damage.

6.2 Considered Containment Failure Modes

Based on the reactor and containment type, there are a number of different principal containment failure modes, each of which impacting the magnitude of the radiological releases attending severe accidents in different ways. Some containment failure modes, or failure mode attributes, are common among all reactor and containment types:

- Containment bypass events, which benefit only little or not at all from the mitigative effects of the containment or containment systems, generally result in large radiological releases. Typical bypass events include interfacing systems LOCAs for all reactor types, unisolated steam generator tube ruptures for PWRs, and breaks outside containment for BWRs. Particularly for PWRs, containment bypass events generally dominate the releases of radioactive substances from the plants.
- Early versus late containment failure also has a large impact on the releases attending the severe accident, because the later the containment fails, the more fission products are removed from the containment atmosphere by various physical and chemical processes. Also, emergency action plans can be more effective if the containment fails late.

A comparison of the conditional probabilities, given core damage, of the various containment failure modes, is provided in the following subsections, broken down by reactor and containment type. The following subsections also highlight significant differences between plants of similar reactor and containment type.

6.3 PWRs

Characteristics of the examined PWR plants important to the level-2 results are compiled in Table 1. Table 3 shows the mean conditional probabilities for:

- Early containment failure, ECF
- Late containment failure, LCF
- Containment bypass
- Isolation failure, ISF
- Filtered containment venting
- Containment intact,

given core damage. For the individual failure modes, the dominant phenomena responsible for the failure, and their relative contributions are also indicated.

6.3.1 Early Containment Failure

Two subcategories of ECF are distinguished:

- containment failure before RPV failure
- containment failure attending RPV failure

ISF may lead to consequences similar to those of ECF. If ISF frequencies are not negligible, ISF is treated as a separate category; otherwise ISF is included in ECF.

Table 3 shows a dichotomy of the conditional probabilities for early containment failure: very small values below 0,02 for the Surry, Zion, Robinson, Beznau, Sizewell-B and Borssele plants, but significantly larger values for the Maine Yankee plant. An analogous dichotomy exists for the characteristic parameter "average hydrogen concentration" in Table 1. For the Maine Yankee (and also the Biblis-B) plants, the value of this parameter is significantly higher than for the other PWR plants.

Two phenomena mainly contribute to early containment failure:

- Loads from direct containment heating (DCH):

  the probabilities of containment failure from the loads attending DCH, given core damage, are small for all examined PWRs, because:

  - the probabilities of occurrence of HPME, given core damage, are estimated to be small for all examined PWRs (0,03 - 0,1). However, the uncertainties associated with the issue are large.

  - the probabilities of containment failure, given HPME, are also small (0.1 - 0.25) for the large dry containments.

- The uncertainties of the load estimates are large. Yet, the impact of the uncertainties on the failure probabilities is low for the strong large dry containments. In all plants but Maine Yankee, DCH is the main cause for early containment failure.

– Loads from hydrogen combustion

Prior to RPV failure, large amounts of hydrogen are produced by the zirconium-steam reaction, and released to the containment atmosphere. Whether or not sufficient hydrogen is accumulated to form ignitable air/steam/hydrogen mixtures depends on the amount of zirconium in the core, on the zirconium oxidation rate, on the free volume of the containment, and on the steam concentration in the containment atmosphere. Whether or not the combustion of hydrogen threatens the containment atmosphere, depends on the average hydrogen concentration.

- With the assumed percentage of zirconium oxidation, the containment pressurisation stays 30% - 50% below the failure pressure at all plants but Maine Yankee; therefore, early containment failure due to hydrogen deflagration is considered negligible at these plants.

- At the Maine Yankee plant, the core contains significantly more zirconium; therefore the average hydrogen concentration is high enough to pressurise the containment, in the event of hydrogen combustion, to a value above its failure pressure. At this plant, ignitable mixtures exist in 55% of the PDS (in which the containment spray system is operating). Hydrogen combustion accounts for 70% of early containment failure at the Maine Yankee plant.

6.3.2 Isolation failure

For most plants, isolation failures are considered negligible (for internal events). For the Robinson plant, ISF, resulting from delayed station blackout sequences with failure of isolation of the containment spray discharge line, was a significant accident progression bin in the original IPE submittal. Subsequent plant modifications have reduced this failure mode to insignificant. At plants with very low frequencies of SGTR and interfacing systems LOCAs, ISF may contribute significantly to releases.

6.3.3 Containment Bypass

Accident scenarios with containment bypass for example, SGTR and V-sequences, can cause releases as massive as ECF. The occurrence frequencies are in the same range as those for ECF. In all examined studies, SGTR with unisolated steam generator and absence of mitigative actions is the dominant release scenario.

At Ringhals-2, Beznau (updated) and Borssele (updated), accounting for fission product scrubbing in the filled up steam generator results in much reduced releases.

At plants with very low frequencies of LRCF modes (order of 10^{-7} or lower), like Borssele PSA '97 and the Konvoi plants, V-sequences can be significant contributors to large releases because it is difficult to show that the frequency of V-sequences is below 10^{-7}/a.

6.3.4 Late Containment Failure

Production of steam and non condensable gases by the core-concrete interaction as well as late hydrogen combustion may lead to pressurisation of the containment atmosphere beyond containment failure pressure. For the Maine Yankee plant, hydrogen combustion is the main cause for LCF; for the other plants, the production of steam and gases is the dominant cause in situations with failed containment heat removal. At Beznau, Borssele and the Swedish plants (and also at German PWRs) late containment failure due to overpressurisation can be avoided by filtered containment venting.

The conditional probabilities of late containment failure due to overpressurisation leading to atmospheric release are high for Maine Yankee (0.47) and Sizewell B (0.19).

6.3.5 Filtered containment venting

Filtered containment venting for avoiding late overpressure failure is available at Biblis-B (and all other German PWRs), Beznau, Borssele (under preparation) and at the Swedish PWRs.

At Beznau, 54% of the severe accident sequences lead to the actuation of the venting function. In 22% of these cases, the filtering function is failed by hydrogen combustion in the vent line. By the design modifications mentioned in section 0, this failure mode is now assumed to be significantly reduced. Since the venting function is actuated before the containment failure pressure is reached, there is a down side to the use of the venting function, because of introducing the possibility of failure of the vent line in situations in which the containment might have remained otherwise intact.

According to preliminary results for the Borssele plant, venting will be actuated in 72% of the core damage sequences.

6.3.6 Containment intact

In this category, it is distinguished, whether the:

– RPV fails

– accident progression is arrested by passive mechanisms or active measures, thus avoiding RPV failure

Possible mechanisms, respectively, measures are:

– restoration of electrical power supply

– depressurisation of the primary side by open pressuriser valves or by pump seal leakage.

The uncertainties associated with the quantification of core damage arrest are large.

At Beznau, Ringhals 2 and Borssele, the conditional probabilities for the containment remaining intact are smaller than at most other plants, because the actuation pressure for the venting function is well below the containment failure pressure, thus invoking the venting function in situations in which the containment failure pressure might not have been reached. However, the sum of the conditional probabilities of "successful containment venting" and "containment intact" is higher than at the other plants, due to the availability of containment flooding and filtered venting.

6.4 BWRs

Characteristics of the examined BWRs plant that are important to the level-2 results are compiled in Table 5. Table 7 shows the mean conditional probabilities for:

– early containment failure , ECF,

- early containment and drywell failure without suppression pool bypass in MK III containments, ECF, DWF, NoSpBY,

- late containment failure, LCF,

- containment venting, Vent,

- containment intact,

given core damage. For the individual failure modes, the dominant phenomena responsible for the failure, and their relative contributions are also indicated.

6.4.1 Early Containment Failure

Two subcategories of the APBs are distinguished:

i) ECF

ii) ECF, DWF, NoSPBy

MK I and MK III containments are treated separately in the following.

- MK I containment (Peach Bottom, Browns Ferry and Mühleberg)

  - APB (i)

    At Peach Bottom and Browns Ferry, the main contribution to this APB is failure of the drywell shell due to attack by molten corium, following RPV failure. According to the analyses results, it is the dominant cause of ECF. However, the uncertainties associated with this failure are so large, that it is difficult to assess the significance of the scenario, see also the discussion in subsection 4.8. A sensitivity analysis performed in the Browns Ferry IPE and a recent study /28/ showed a significant reduction of the drywell shell failure mode, if sufficient water is available on the drywell floor for debris quenching. At Mühleberg, the liner melt-through scenario is practically eliminated by the large in-pedestal sump volume that can accommodate all core debris. Therefore, the conditional probability of early containment failure is much lower than at the US plants with Mark I containments. Early containment failure at the Mühleberg plant is dominated by overpressure failure in ATWS situations.

- MK III containment (Grand Gulf and Perry)

    - APB (ii)

      Containment failure combined with drywell failure provides a release path bypassing the suppression pool. Causes are hydrogen deflagration and loads attending RPV failure. In both analyses, the conditional probabilities for this APB are significant, with the main contribution from hydrogen burns. The contribution associated with RPV failure is smaller in the Perry IPE than in the Grand Gulf analysis, due to modelling assumptions concerning RPV failure in the MAAP code calculations performed for the Perry analysis.

      Due to the low failure pressure (relative to the large dry PWR containments), the conditional probabilities of containment failure, given hydrogen deflagration or RPV failure, are high. To reduce the threat from hydrogen deflagration, the MK III containments are equipped with ignition devices. In case of loss of AC power, the igniters are not available (now DC-powered at Grand Gulf).

      Pressurisation by production of steam and non condensable gases following RPV failure can lead to containment failure with the drywell remaining intact, and a release path through the suppression pool.

      In the Grand Gulf analysis, this scenario is significant, whereas, in the Perry analysis it is not, due to the modelling assumptions concerning RPV failure.

- Forsmark 3 containment

  The biggest potential contribution to early containment failure is attack by molten corium of cable penetrations located on the lower drywell floor. This event is strongly influenced by the availability of water in the lower drywell. At Forsmark 3, at least three systems are available that can be used for flooding the lower drywell: (1) a dedicated system (system 363), (2) the containment vessel spray system and (3) the external water injection system. In addition, water can be added to the drywell via coolant systems, should they be available after vessel breach. Therefore, the likelihood of drywell penetration by molten core debris is extremely small at Forsmark 3. Presence of water in the drywell also makes containment failure due to DCH unlikely, because injection of core debris into a deep water pool inhibits the dispersion of finely fragmented core debris into the containment atmosphere.

6.4.2 Late Containment Failure

This APB represents overpressure failure of the containment due to production of steam and non condensable gases in the core melt-water-concrete interaction following RPV failure, if containment venting is not effective. In the not inerted MK III containments, hydrogen combustion can also contribute to the APB. The relative contributions differ significantly among the examined studies, mainly due to differences in the availability of containment venting. In the Browns Ferry IPE, no information concerning venting is given; therefore, the difference to the Peach Bottom analysis is not traceable.

At the Swedish plants, where several possibilities exist for flooding the containment and where high capacity filtered venting is available, late containment failure is almost non-existent.

6.4.3 Containment venting

Most BWR plants are provided with a containment venting function, serving for

- diverse heat removal from the suppression pool
- avoidance of overpressure failure of the containment.

At the US plants venting is not effective if AC power is unavailable. The conditional probabilities for successful containment venting differ significantly between the Grand Gulf and Perry plants, mainly due to the difference in the relative contributions of LOSP. At the Swedish plants and at Mühleberg, filtered containment venting is available, its actuation is DC powered. The conditional probability that the containment venting function is invoked is much higher than at the US-plants with Mark I containment. This is mainly due to a different containment venting strategy (because of the availability of filtered venting).

6.4.4 Containment intact

In all but the Mühleberg analyses, the conditional probabilities for the containment remaining intact are significant. At the Mühleberg plant, the containment venting strategy invokes filtered venting in scenarios that might otherwise lead to the APB "containment intact". Contributions from situations with core damage arrest prior to RPV failure range from being substantial to dominant in the studies for the US plants. In the Mühleberg study no credit is given to accident recovery by core damage arrest.

Large uncertainties are associated with the quantification of core damage arrest prior to RPV failure. In the two examined IPE studies, the modelling in the MAAP-code calculations produces substantially larger conditional probabilities for the RPV remaining intact, than those estimated in the NUREG-1150 analyses: 0.25 for Browns Ferry in contrast to 0.09 for Peach Bottom, and 0.3 for Perry, in contrast to 0.17 for Grand Gulf.

At the Swedish plants, the sum of the conditional probabilities of "containment remaining intact" and "successful containment venting" is very high: i.e.: 0.84 for Barsebäck and 0.98 for Forsmark 3. This is due to several possibilities for flooding the containment and the availability of filtered venting.

7. CALCULATION OF SOURCE TERMS

7.1 Analysis of the Transport of Radioactive Substances

To calculate the source terms, the migration of the radioactive substances is traced from the nuclear fuel through the reactor coolant system and the containment to their release to the environment.

The fraction of the inventory of radioactive substances released to the atmosphere, the time history of the release, as well as further information needed for assessing the offsite consequences, are designated "source terms".

Calculating source terms with detailed mechanistic codes is a very complex and cost intensive task. Therefore, only a few accident sequences can be examined with such codes. If source terms have to be calculated for every accident sequence of the APET, as in NUREG-1150, simplified parametric computational models are required. In NUREG-1150, the XSOR suite of codes is used, and in the IPE studies, the MAAP code system. In the IPE studies, only one representative calculation is performed for each APB. In the studies for the Swiss plants, time dependent calculations with the parametric ERPRA code are used to arrive at source terms. In the Sizewell B and the Swedish studies, MAAP is used.

The releases are grouped into release categories (RC), according to their potential for causing offsite consequences. Associated with the RCs are the initial and boundary conditions for the calculation of offsite consequences. In NUREG-1150, the RCs are formed by combining the results of the source term calculations according to defined characteristics. In the IPE studies, the RCs are explicitly linked to the APBs. More details on the codes used are provided in reference /35/.

7.2 Results of Source Term Calculations

The magnitude of source terms varies substantially across the various containment failure modes. The estimates of the source terms are strongly affected by phenomenological uncertainties; they can vary by several orders of magnitude. The range of uncertainties for the source term associated with one accident sequence can be wider than the range of variation between different accident sequences, even between different plants. In the NUREG-1150 and in the Swiss studies the range of the uncertainties is explicitly presented together with the results. In the IPE studies, the Swedish studies and the Sizewell B study, only point values are presented. For the release categories examined in this report, the IPE results are within the 5%-95% percentile range of the NUREG-1150 results for the same or similar plants; mostly they are in the median / mean area.

Tables 4 and 8 present the frequencies of exceeding 1% and 10% Cs release fractions for the different reactor and containment types included in this comparison. It should be noted that a release fraction of 10% or more of Cs can be taken as a surrogate for a large release. The two tables also show conditional probabilities of exceeding 1%, respectively 10% releases.

7.2.1 PWRs

Review of tables 3, 4 and figures 1, 2 shows that

- the exceedance frequencies for 1% Cs release differ significantly.

  They are lowest at Beznau, Borssele and Ringhals 2, where the concept of combining external water injection to the containment with high capacity filtered venting, as well as the strategy for coping with SGTR events, leads to the lowest frequencies of exceeding 1%, Cs releases (see figure 2).

- regarding the conditional probabilities of exceeding 10% Cs release, given ECF or Bypass or ISF, (LRCF modes) three groups can be identified:

  - Plants without strategies for mitigating steam generator tube rupture events: In this group, the majority of large releases is explained by unisolated steam generator tube rupture events.

  - Plants with mitigation strategies for SGTR events (now being implemented at Beznau, Borssele, Ringhals-2, see also the discussion in subsection 0): At these plants the releases resulting from unisolated steam generator tube rupture events are significantly reduced. This is evident for Beznau and Ringhals, but not for Borssele, where the frequency of SGTR is reduced so much by preventive measures that the interfacing system LOCA (V-sequence) and isolation failure (ISF) become dominant. For these events, no mitigative action is foreseen. The three plants have by far the lowest frequencies of exceeding 10% Cs release (see figure 1).

  - The results for Sizewell B differ substantially from the other results in that large releases are dominated by late containment failure. As the analysis is "pessimistic", it is not clear whether this result is an artefact of the analysis.

A special situation exists for the analysis of the Japanese plant because accident management actions are not credited in the analysis.

7.2.2 BWRs

Review of tables 7, 8 and figures 1, 2 shows that:

- the exceedance frequencies for 1%, respectively, 10% Cs differ very significantly.

- regarding the conditional probabilities for exceeding 10% Cs release, given ECF or Bypass or ISF, two groups can be identified:

  - At plants without the capability for external water injection, large early releases are driven by structural failure of the drywell due to core melt attack.

  - At plants with external water injection capability, combined with high capacity filtered venting (in this comparison, Mühleberg, Forsmark 3 and Barsebäck), the frequency of the drywell attack failure mode is greatly reduced. At Mühleberg several other features contribute to the low release frequency relative to US plants with Mark I containment:

⇒ the much larger in-pedestal sump volume (which is also responsible for the lower conditional probability of early containment failure)

⇒ the large containment volume / power ratio, and

⇒ the strong reactor building (acting as secondary containment), from which the vent path is through an outer water filled torus.

Mühleberg, Forsmark 3 and Barsebäck also have much smaller conditional probabilities of exceeding 1% Cs release, given core damage, than the other plants. This reflects the possibility to flood the containment and the availability of filtered venting at these plants.

A special situation exists for the analysis of the Japanese plant because accident management actions are not credited in the analysis.

8. CONCLUSIONS

The results of the studies can be summarised as follows:

8.1 PWRs

- The sum of the conditional probabilities of the containment failure modes ECF, ISF and Bypass, given core damage, differ little in the examined studies and are about 0.1. (see the illustration in figure 1).

- The containment failure mode ISF is insignificant in all but the Borssele studies. Another exception in the Robinson IPE had resulted from isolation failure of the containment spray discharge line, given station blackout. This vulnerability has since been corrected.

- The largest releases generally result from containment bypass sequences, most notably steam generator tube rupture with unisolated steam generator. Therefore, the largest benefit in terms of reduction of offsite consequences is to be expected from severe accident management directed at mitigating the consequences of such accident sequences. Two strategies are reported in the examined PSA studies:

  - Application of primary side bleed/feed (PB/F) to steam generator tube rupture events. This is now implemented at most plants. The application of PB/F to steam generator tube rupture events reduces

    ⇒ the occurrence frequency of such events

    ⇒ the release to the environment from the ruptured steam generator, because, by the split-up of mass flow between open pressuriser valves and the ruptured steam generator heating tube, the majority of the fission products released from the core is directed to the containment. Due to the long time to RPV failure in such accident sequences, significant depletion of fission products can take place inside the containment.

  - Filling up of the ruptured steam generator with water. The effectiveness of fission product scrubbing in the water column was investigated in several studies. The reported calculations show significant reductions of fission product releases. The strategy, using fire water (severe accident management), is now implemented at the Beznau, Borssele and Sizewell B plants. However, no credit is taken in the Sizewell B PSA. At the Swedish plants a different strategy is being studied which aims at maintaining a high water level in the defective steam generator from the onset of the accident (except for accident sequences with extremely low frequencies of occurrence). On the negative side of the severe accident management strategy could be loss of steam generator integrity due to pressurised thermal shock. This possibility is discussed in the Robinson IPE.

Further investigations into the subject appear to be necessary.

Releases resulting from LOCAs at the high pressure system/low pressure system interface (V-sequence) have been made extremely unlikely at the plants examined in this comparison, due to improved redundancy/diversity of the high pressure system/low pressure system isolation, and improved surveillance, testing and maintenance strategies for the interface. In the Borssele studies, (and also for the Konvoi plants) this sequence has gained relative importance because the frequency of other LRCF modes is extremely low .

- Early containment failure which potentially can cause large releases, is very unlikely at most plants with large dry, prestressed concrete containments. Due to their robust design, they can absorb the majority of the loads attending the early phase of severe accidents. Containments with steel shell construction are less resilient to such loads because of their lower failure pressure. However, at the Beznau and Borssele plants, the lower failure pressure of the steel shell containment is compensated for by lower ratios of "(reactor power)/(containment volume)" and "(fuel mass)/(containment volume)" that lead to reduced severe accident loads.

  A parameter important to early containment failure is the amount of zirconium in the reactor core, which can vary considerably, depending on the fuel manufacturer. Among the investigated plants, there is substantially more zirconium in the core of the Maine Yankee and the Biblis-B reactor, than in the other PWRs. This leads to increased vulnerability to hydrogen combustion in the early (and late) phase of severe accident situations with operating containment sprays at Maine Yankee. This issue could also become important at other plants if fuel with thicker cladding is reloaded.

  As defence against the threat from hydrogen combustion a combination of igniters and catalytic recombiners will be implemented at Biblis-B (and other German PWRs). At Beznau, hydrogen and oxygen can be removed from the containment atmosphere through the venting line. Similar strategies are also studied for the Borssele plant.

- The conditional probabilities for exceeding a 10% release, given core damage, are mostly in the range 0.02 to 0.03. For Ringhals 2, Beznau and the earlier Borssele study, they are below 0.01. In the newest study for Borssele, this conditional probability is 0.08, reflecting the fact that V-sequence and ISF (which are not mitigated) are the dominant contributors.

- With the exception of Sizewell B, the releases from late containment failure are generally much lower than from the accident sequences described above. At Sizewell B, late containment failure is the main contributor to large releases. The occurrence of late containment failure can be made less likely if filtered containment venting is available (currently at the Swedish and German plants, Beznau and Borssele). A further reduction is possible by the addition of large quantities of water for debris quenching from internal or external sources. This is available at the Swedish plants, Beznau and Sizewell B. Most effective for mitigating releases in the late accident phase is the combination of water injection to the containment with filtered containment venting. Such accident mitigation strategy is implemented at Beznau and at the Swedish plants.

8.2 BWRs

- In BWRs with MK I containment, there is the potential for early containment failure due to melt-through of the drywell liner under attack by molten corium. In most studies, this early containment failure mode is the main source for offsite consequences. The uncertainties associated with that issue were very large at the time of conduction of the earlier studies included in this comparison. Recently, research results have become available, suggesting that the probability of liner melt-through could be significantly reduced if sufficient quantities of water were available on the drywell floor. At the Mühleberg plant, with its large in-pedestal sump volume, this scenario is de-facto eliminated.

- Strategies for making available large quantities of water. The purpose is:

  - to keep a damaged core inside the RPV by outside cooling, or - should this fail -

  - to protect the containment structures against attack by molten corium.

Such provisions are made at all Swedish BWRs, with water either from internal or from external sources (external containment water injection from fire trucks).

- Containment venting for avoiding late overpressure failure of the containment is available at all plants included in this comparison. At the Swiss and Swedish BWRs, venting is also possible through additional high capacity filter devices. In addition to severe accident mitigation, containment venting is also used for alternate heat removal in situations with failed suppression pool cooling.

- At Mühleberg and the Swedish plants, the combination of the:

  - possibility to flood the containment from internal or external sources and
  - overpressure protection by filtered containment venting

leads to low conditional probabilities of early containment failure and of significant and large releases. At Mühleberg, the large in-pedestal sump volume also contributes to improved severe accident behaviour, relative to other plants with Mark I containment. At the modern Swedish plants, several systems can be used for containment flooding.

- All Mark I containments are inerted, therefore, hydrogen combustion is of no concern at these plants.

- The load capacity of MK III containments is substantially lower than of MK I containments. In part, this drawback is compensated for by the large volume of the containment that encloses drywell and wetwell. Nevertheless, the conditional probabilities of early containment failure due to hydrogen combustion and production of steam are significant. Because of their large volume, the Mark III containments are not inerted. As defence against hydrogen combustion, igniters are provided. They are ineffective if AC power is unavailable. At several plants, control and operation by DC power is now implemented.

All conclusions presented here apply to the plants included in this comparison. Any inference made to other plants requires caution, even if the designs are similar, because plant specific features can significantly affect the response of a plant to severe accident loads.

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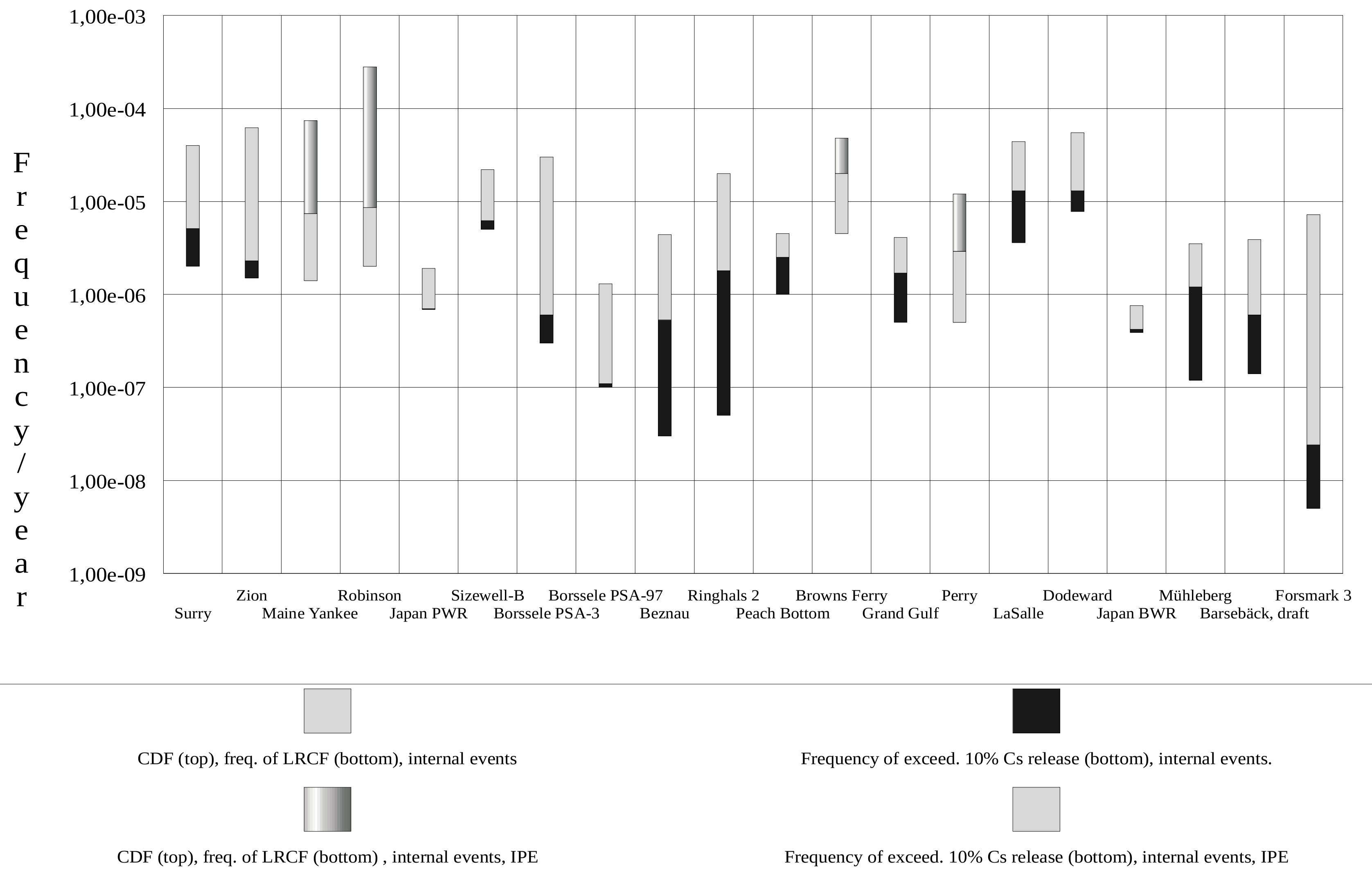

Figure 1. Frequencies of CD, LRCF, exceedance of 10 % Cs release, internal events

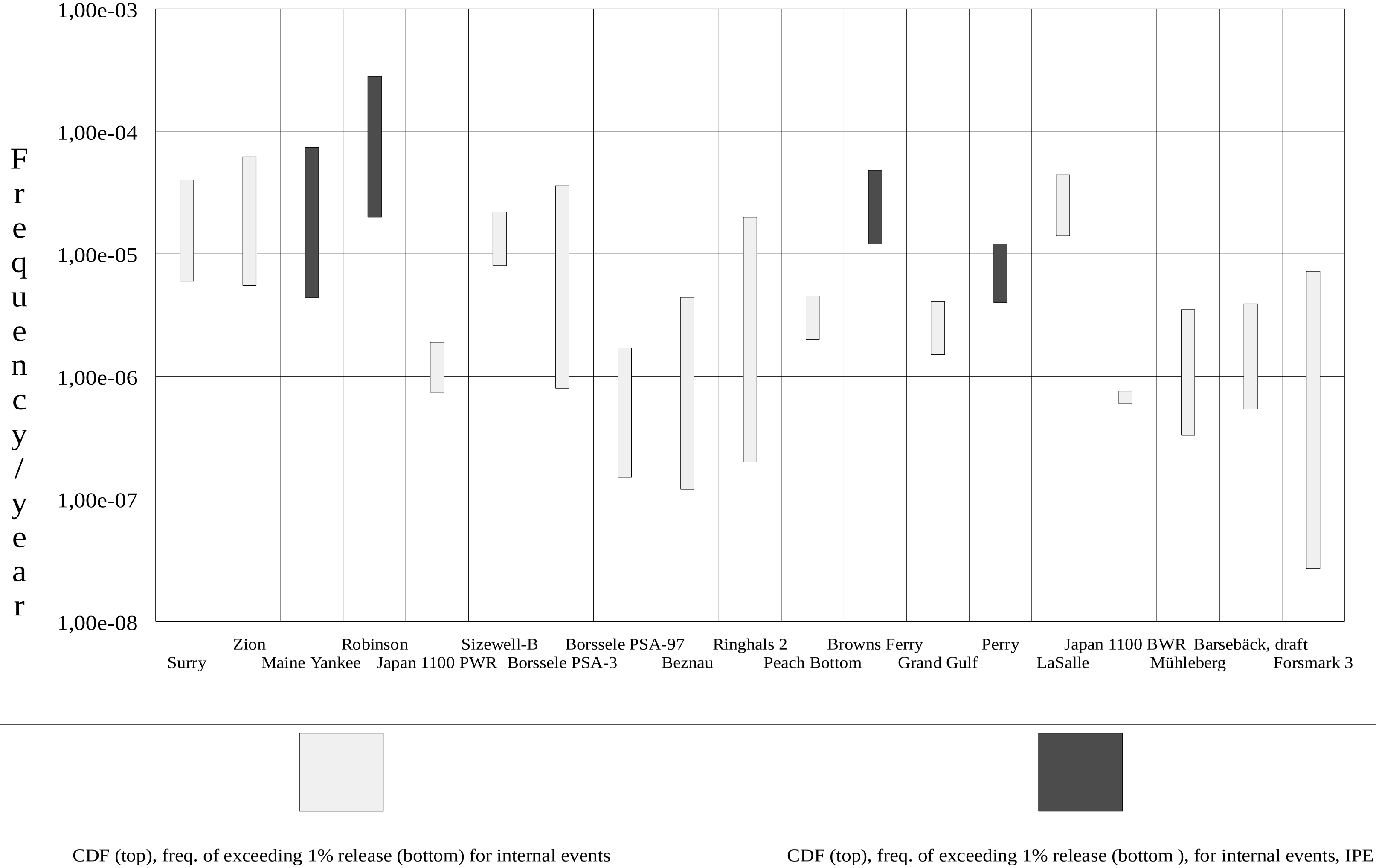

Figure 2. Frequencies of CDF, exceedance of 1% Cs release

Table 1. PWRs with large dry containments. Plant/containment design characteristics and provisions for severe accident management

Plant Characteristic/ relevant for	Surry	Zion	Maine Yankee	Robinson	Beznau	Biblis-B	Sizewell-B	Ringhals 2	Borssele PSA-3 (PSA97)	Japan 1100 Mwe PWR	Pickering A (CANDU)
Power, MWth	2441	3250	2630	2300	1130	3750	3411	2660	1365	3441	1760
Containment type / Load capacity, tightness	Concrete	Concrete	Concrete	Concrete	Steel	Steel	Double primary: prestressed. concrete secondary: concrete	Concrete	Double, inner. Steel outer: concrete	Prestressed concrete	Concrete
Containment volume (m^3)	51000	81000	52600	59400	36400	70000	85500	58000	37130	73300	81300 (accident unit + pressure relief duct) + 438700 (vacuum building + other units) (Total: 520000)
Power/Containment volume/ Containment loads, Time scale of accident Time budget for AM	0.048	0.040	0.05	0.039	0.031	0.054	0.04	0,045	0.037	0.046	0.0034
Fuel mass (kg)	79000	98000	80000	78900	43500		101.000	69000	42955	89500	101.300
Fuel mass / Containment volume/ Containment loads from DCH	1.5	1.2	1.5	1.3	1.2		1.18	1.2	1.16	1.2	0.19 (total volume)
Zirconium mass (kg)	16500	20050	24300	16335	12000	29750	19600	16400	9910	19500	9600 in fuel sheaths, 34500 in channels and rest
H_2 -mass (kg), with 100% Zr-oxidation.	780	880	1062	718	530	1350	866	770	435	855	648 (100% oxidation in fuel sheats, 15% other)

Table 1. PWRs with large dry containments. Plant/containment design characteristics and provisions for severe accident management

Plant Characteristic/ relevant for	Surry	Zion	Maine Yankee	Robinson	Beznau	Biblis-B	Sizewell-B	Ringhals 2	Borssele PSA-3 (PSA97)	Japan 1100 Mwe PWR	Pickering A (CANDU)
Average H_2-concentration (%), at 30^0C, dry, 100% Zr-oxidation/ Potential for H_2 burn Containment loads from H_2 burn	15	10.8	20.2	12.1	12.8	19	10	13.2	11.5	12.4	8.9% (with above oxidation)
Estimated pressure (bar) due to H_2 burn, 100% Zr-oxidation	9.4	6.7	12.4	7.6	7.9	11.7	6.3	8.2	7.1	6.7	not assessed
Plant Characteristic/ relevant for	Surry	Zion	Maine Yankee	Robinson	Beznau	Biblis-B	Sizewell-B	Ringhals 2	Borssele PSA-3 (PSA97)	Japan 1100 Mwe PWR	Pickering A (CANDU)
Estimated containment failure pressure (bar)	9.7	10.2	10.5	10.4	7.8	8	10.1	12	8		2.2
Containment spray / Potential for hydrogen burn, late containment failure by overpressurization	yes	yes	yes	yes	yes	no	yes	yes	yes	yes	yes, in vacuum building
Hydrogen control / Potential for hydrogen burn					early containment venting	Igniters/ recombiners	Hydrogen mixing in the short term Recombin- ers in the long term	Recombiners	under study: • early venting • combination of recom- biners/ igniters post-accident inertisa- tion/ recom- biners	combination of recombiners and inertisation of atmosphere	Igniters

Table 1. PWRs with large dry containments. Plant/containment design characteristics and provisions for severe accident management

Plant Characteristic/ relevant for	Surry	Zion	Maine Yankee	Robinson	Beznau	Biblis-B	Sizewell-B	Ringhals 2	Borssele PSA-3 (PSA97)	Japan 1100 Mwe PWR	Pickering A (CANDU)
Additional water injection to containment / Ex-vessel cooling of core debris, late containment failure by overpressurization and/or BMP					External from fire truck: • backup water source for containment spray • flooding of contain-ment External from river for cooling of fan coolers		Fire water system for debris quenching and cooling	External from fire truck (CWIS) for: • backup water source for containment spray flooding of contain-ment		under preparation: water injection from RWST and fire water system	-
Depressurisation of RCS for prevention of HPME ("primary side bleed") / Likelihood of DCH	Transients with loss of all FW	Most events with loss of all FW	Transients with loss of all FW	Transients with loss of all FW	Transients SLOCA with loss of all FW	Most events with loss of all FW	All events with high primary pressure	Most events with loss of all FW	Most (all) events with high primary pressure	Most events with loss of all FW	inherent due to failure of fuel channels, given core melt
Depressurisation of containment/ Late containment failure by overpressurization					filtered containment venting	filtered containment venting		filtered containment venting	filtered containment venting		filtered containment venting
Use of PB/F in the event of SGTR / Release attending SGTR		yes			yes	yes	yes	yes	yes	yes	yes
Filling of SG with water in the event of SGTR / Release attending SGTR					yes	under study	yes	yes	yes		-

Table 2. Methods used and level of detail of the logical models in the examined PSAs, PWRs with Large Dry Containment

Plant	Method	Number of PDS, $> 10^{-7}$/a	Number of Summary PDS	Number of CET/APET Nodes	Number of Source Term Bins
Surry, NUREG-1150	Large event tree	25	7	71	7
Zion, NUREG 1150	Large event tree	18	5	72	4
Maine Yankee, IPE	Small event tree	17	10	14	18
Robinson, IPE	Small event tree	21	-	12	8
Beznau, HSK/ERI	Small event tree	28	11	33	18
Sizewell-B	Small event tree	20	30	20	331/ (22 release categories)
Ringhals 2	Small event tree	23	23	12	16
Borssele PSA-3 (PSA-97)	Small event tree	25	111	51	13 ((16)
Japan 1100 Mwe PWR	Medium event tree	7	6	20	12
Pickering A	Small event tree	25 (irrespective of frequency	-	9	7

Table 3. PWRs with large dry containments. Conditional probabilities of containment failure modes, given core damage Dominant phenomena and their relative contributions

		Containment-Failure mode					
Plant	Total CDF	Early Containment Failure	Late Containment Failure	Containment-Bypass	Isolation Failure	Successful containment venting	Containment intact
Surry	4.0 E-5, 3% at high pressure	0.007, DCH >90%	0.06, BMP	0.12, SGTR ~60%	-		0.81, RPV intact 57%:
Zion	6.5 E-5, 2% at high pressure	0.005, DCH >90%	0.24, BMP	0.02, SGTR ~90%	-		0.73
Maine Yankee	7.4 E-5, 16% at high pressure	0.08, H_2 burn ~70%, DCH ~ 30%	0.47, Overpressure	0.02, SGTR ~70%	-		0.43, RPV intact :30%
Robinson	2.4 E-4, 22% at high pressure	0.016, DCH>90%	0.07, Overpressure	0.02, SGTR ~70%	0.13		0.77
Beznau	4.4 E-6, <10% at high pressure	0.016, DCH>80%	0.19, includes vent failure due to hydrogen burn: ~22%, lower after modification of vent line	0.11, SGTR >90%		0.54	0.15
Biblis-B	2.9 E-6, 9% at high pressure	n.a.	n.a.	< 0.04 , V-Seq. <80% (conservative estimate), SGTR,~20%			
Sizewell-B, conservative	2.2E-5	< 0.01	0.19 overpressure	0.09, SGTR ~92%	< 0.01	-	0.71
Ringhals 2	2.0 E-5, 12% at high pressure	< 0,01	0.11, BMP > 95 %	0.08, SGTR >90%	0.01	0.3	0.5
Borssele PSA-3	3.6 E-5	< 0.01	0.07	0.01, V-seq. 50%, SGTR 15%	< 0.01-	0.65	0.26
Borssele PSA-97	1.7 E-6	0.01	0.05	0.05, SGTR: 60%, V-seq.: 40%	< 0.01	0.72	0.21
Japan 1100 Mwe PWR	1.9 E-6, 18% at high pressure	0.01, DCH 50%, H_2 burn 40%	0.08	0.34, SGTR 80%	< 0.01	-	0.56

Table 4. PWRs with large dry containments. Frequencies and conditional probabilities of significant and large Cs releases. Dominant phenomena and their relative contribution.

Plant	Frequency/a of		Exceedance frequency/a for		Conditional probability of exceeding	
	Total CDF	ECF +Bypass+ISF	1% release	10% release	1% release, given core damage.	10% release, given ECF + Bypass + ISF
Surry	4.0 E-5,	5.1 E-6	6 E-6	2 E-6, SGTR >90%	0.15	0.39
Zion	6.5 E-5,	1.5 E-6	5,5 E-6	1 E-6, SGTR ~30%, DCH: ~70%	0.08	0.66
Maine Yankee	7.4 E-5,	7.4 E-6	4.4 E-6	1.4 E-6, SGTR: ~20%, H_2 burn: ~80%	0.06	0.19
Robinson	2.4 E-4,	8.6 E-6	2 E-5	2 E-6, SGTR ~50%, DCH: ~50%	0.1	0.23
Beznau	4.4 E-6,	5.3 E-7	1.2 E-7	3 E-8, SGTR ~45%, DCH ~55%	0.03	0.05
Biblis-B. Releases were quantified only for SGTR with low RCS pressure. Frequency of high pressure SGTR sequences: 1 10^{-8}	2.9 E-6,		< E-8, PB/F with scrubbing in SG >E-8 otherwise (only SGTR)	<<E-8, with scrubbing in SG > E-8 otherwise (only SGTR)		
Sizewell-B, conservative	2.2 E-5	2 E-6	8 E-6	5 E-6 late overpressurization: 80%	0.36	0.25, given LRCF mode 0.99, given late overpressurization 0.22, given core damage
Ringhals 2	2.0 E-5,	1.8 E-6	2E-7, ECF ca. 50%, isolation failure ca. 50%	5 E-8, ECF > 90%	0.01	0,03
Borssele PSA-3	3.6 E-5	8 E-7	8 E-7	3 E-7, V-seq.: 70%	0.02	0.37
Borssele PSA-97	1.7 E-6	1.1 E-7	1.5 E-7	1 E-7, V-seq.: 70%, ISF: 15%	0.08	0.6
Japan 1100 Mwe PWR	1.9 E-6,	7 E-7	7.4 E-7	6.9 E-7	0.39 (without credit to SAM)	0.36, given core damage (without credit to SAM)
Pickering A	1.3 E-4		< 1 E-7	< 1 E-8	< 8 E-4	?

Table 5. BWRs. Plant/containment design characteristics and provisions for accident management

Plant Characteristic / relevant for	Peach Bottom	Browns Ferry	Grand Gulf	Perry	Mühleberg	Barsebäck 1/2	Forsmark 3	Dodewaard	LaSalle	Japan 1100 Mwe BWR
Power, MWth	3.293	3.293	3.833	3.579	1.097	1.800	3.300	183	3293	
Containment type	MK I	MK I	MK III	MK III	MK I	ASEA II	ASEA IV	Humbold Bay (pre-MKI)	MK II	MK II
Estimated containment failure pressure, bar / Load capacity, tightness	10.7	10.7	6.5	6.5	9.4		11	7	13.4	9.4
Containment volume, m^3	8.230	8.100	40.300	40.800	5.000	5.000	8.600	754	10300 ???	10300
Power/Containment volume/ Containment loads, time scale for accident, time budget for AM	0.4	0.4	0.097	0.09	0.21	0.36	0.38	0.24	0.3 ???	0.3
Power/Suppression pool volume/ Likelihood of containment challenges	0.9	0.9	1.0	0.8	0.5	0.6	1.0	0.45	1.0 ???	1.0
Fuel mass, kg	159.400	155.600	166.200	156.000	48.500	89.000	126.000	10.550		
Fuel mass/containment volume/ Containment loads from DCH	19.4	19.2	4.2	3.9	9.5	17.8	14.6	14		
Zirconium mass, kg	65.500	55.000	80.000	75.000?	24.000			4.338		
H_2-mass, kg, 100% Zr-oxidation / Potential for H_2 burns	2.850	2.410	3.510	3.290	1.044			190		

Table 5. BWRs. Plant/containment design characteristics and provisions for accident management

Plant Characteristic / relevant for	Peach Bottom	Browns Ferry	Grand Gulf	Perry	Mühleberg	Barsebäck 1/2	Forsmark 3	Dodewaard	LaSalle	Japan 1100 Mwe BWR
Primary containment venting / Containment overpressure failure	yes	no hardened vent path	yes	yes	yes, filtered	yes, filtered	yes, filtered	yes, filtered is planned	yes	yes, hardened vent path planned
Flooding of lower drywell / Prevention of liner failure					yes, using fire water system	yes	yes	planned, by wetwell-drywell connection		
Additional water injection to containment / Containment protection					Fire water system, optionally with external water supply, for • back-up water source for containment spray flooding of containment	Fire truck (CWIS) for back-up water source for • containment spray • flooding of containment Fire water pool for flooding of containment	Fire truck (CWIS) for back-up water source for • containment spray • flooding of containment Fire water pool for flooding of containment	Fire protection system as backup for containment spray		Fire water injection planned

Table 6. Summary of Methods Used in the Examined PSAs, BWRs

Plant	Method	Number of PDS, > 10^{-7}/a	Number of Summary PDS	Number of CET/APET Nodes	Number of Source Term Bins
Peach Bottom, NUREG-1150	Large event tree	9	4	145	10
Browns Ferry, IPE	Large event tree	20	8	125	10
Grand Gulf, NUREG-1150	Large event tree	12	4	125	8
Perry, IPE	Medium event tree	12	4	68	25
Mühleberg, HSK/ERI	Small event tree	6	6	18	15
Forsmark 3	Small event tree	7	7	10	5
Dodewaard	Medium event tree	44	6	70	10
LaSalle	Large event tree	30 (including external events)	7 (including external events)	135	20
Japan 1100 Mwe BWR	Small event tree	4	8	21	46

Table 7 BWRs. Conditional probabilities of containment failure modes, given core damage state Dominant phenomena and their relative contributions

			Containment failure mode					
Plant	Total CDF	Containment type	Early containment failure	Containment bypass	Early containment drywell-failure without suppression pool bypass MK III	Late containment failure	Containment-venting	Containment intact
Peach Bottom	4.3 E-6	MK I	0.56, Liner failure ~60%, DCH ~5%%		-	0.05, Overpressure >90%	0.11	0.27, RPV-intact ~40%
Browns Ferry	4.8 E-5	MK I	0.46, Liner failure ~60%, Overpressure ~8%		-	0.26		0.28, RPV-intact ~90%
Grand Gulf	4.1 E-6	MK III	0.21, H_2-burn ~75%		0.22, Over pressure >90%	0.28	0.04	0.23, RPV-intact ~75%
Perry	1.2 E-5	MK III	0.16, H_2-burn >90%		0.07, 0ver-pressure >90%	0.07	0.31	0.39, RPV-intact 70%
Mühleberg	3.5e-6	MK I	0.26, Overpressure		-	0.07, Overpressure	0.66	-
LaSalle	4.4 E-5	MK II	0.33, Over-pressure, CCI			0.1	0.46	0.11
Barsebäck 1/2, (draft)	3.9 E-6	ASEA II	0.1 CCI, 40% Reactor over pressure, impact of vessel head failure, 40%	0.05, Isolation failure of main steam line with reactor at high pressure >90%		<0.01	0.84	
Forsmark 3	7.2 E-6	ASEA, IV	2 E-3	5 E-4		8 E-5	0.5	0.48
Dodewaard	5.5 E-5	Humbold Bay (pre-MKI)	0.25, Over-pressure: 95%, Ex-vessel steam explosion: 2.5%			0.36, CCI or liner attack 90%, thermal drywell failure 3%		0.38
Japan 1100 Mwe BWR	7.6 E-7	MK II	0.51, Over-pressure failure before core melt 70%	0.03, V-seq.	-	0.29, Over-pressure > 40%	-	0.16

Table 8. BWRs. Frequencies and conditional probabilities of significant and large Cs releases, dominant phenomena and their relative contribution

Plant/PSA	Frequency/a of		Exceedance frequency/a for		Conditional probability of exceeding	
	Totality of containment failure modes	ECF + bypass + ISF	1% release	10 % release	1% release, given core damage	10% release, given ECF + Bypass + ISF
Peach Bottom, NUREG-1150	4.3 E-6	2.4 E-6	2 E-6	1.3 E-6, liner failure	0.46	0.54
Browns Ferry, IPE	4.8E-5	2.2 E-5	1.2 E-5	5 E-6, liner failure	0.25	0.22
Mühleberg, HSK/ERI	3.5 E-6	9 E-7	3.3 E-7	1.2 E-7, early overpressure failure	0.1	0.13
LaSalle	4.4 E-5	1.5 E-5	1.4 E-5	3.6 E-6, overpressure, CCI	0.32	0.24
Grand Gulf, NUREG-1150	4.1 E-6	8.6 E-7	1.5 E-6	5 E-7, hydrogen burn	0.36	0.58
Perry, IPE	1.2 E-5	1.9 E-6	4 E-6	5 E-7, hydrogen burn	0.33	0.26
Barsebäck 1/2 (Draft)	3.9 E-6	3.9 E-7	5.4 E-7	1.4 E-7, steam line isolation failure, CCI, impact of vessel head failure	0.13	0.36
Forsmark 3	7.2 E-6	2.4 E-8	2.7 E-8	5 E-9 containment bypass	0.0038	0.2
Japan 1100 Mwe BWR	7.6 E-7	3 E-8	6.5 E-7	4.2 E-7	0.86 (without credit to SAM)	0.56, given core damage (without credit to SAM)