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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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Principal Working Group No. 5 - Risk Assessment

INTEGRATED ASSESSMENT OF LEVEL 1 AND LEVEL 2 PSA RESULTS FOR INTERNAL AND EXTERNAL EVENTS

English text only

63737

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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
- to contribute to the expansion of world trade on a multilateral. non-discriminatory basis in accordance with international obligations.

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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 Zealant 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 cooperates 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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ABSTRACT

Level-1 and level-2 PSA results for internal and external hazards are compiled in order to identify and evaluate vulnerabilities of specific plant designs to the various event classes and to develop a ranking of the contributions of the event classes to core damage frequency , frequencies of large release containment failure modes, and frequencies of exceeding 1% and 10% releases. The results are compared with PSA results for the classical internal events. Through the integrated assessment the report also provides a contribution to the issue of completeness of PSA results.

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 etc. (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 identifies vulnerabilities of specific plant designs to the various event classes and develops rankings of contributions of event classes to core damage frequency (CDF), frequencies of large release containment failure (LRCF) modes and frequencies of exceeding 1% and 10% releases.

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:

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TABLE OF CONTENTS

1. INTRODUCTION

In this report, level-1 and level-2 PSA results for internal and external hazards /x1-36/ are compiled and evaluated for selected plants. These results are compared to the results for internal events obtained for the same plants in /y1-yx/. To a large part, the presented material is based on work performed for the Swedish Nuclear Power Inspectorate (SKI) /37/. Material from other supporting PSA literature /38-41/ was also used in the preparation of this report.

The purpose of the evaluation is to identify vulnerabilities of specific plant designs to the various event classes and to develop a ranking of the contributions of the event classes to core damage frequency (CDF), frequencies of large release containment failure (LRCF) modes and frequencies of exceeding 1% and 10% releases. In this way, the report can also be seen as a contribution to the issue of completeness of PSA results.

2. PROBABILISTIC ASPECTS OF FIRE RELATED FAILURES

Plant specific probabilistic aspects of fire events are presented for the PWR plants Surry, Biblis-B, Konvoi, Beznau and the BWR plants Peach Bottom and LaSalle. In the concluding section, generic probabilistic aspects of fire events are discussed.

2.1 Surry

2.1.1 Level-1 Aspects

The fire frequencies for the examined critical locations are in the range 4 E-5/a to 3 E-4/a.

The conditional probabilities of hazard states with potential for core damage developing from fires are high (> E-1) for all examined fire areas. This is due to insufficient separation and fire protection of power and control cables for the HPI and CCW pumps. For the majority of the examined events, failure of the CCW pumps leads to a pump seal LOCA which, due to the inoperability of the HPI pumps, would lead to core damage if no corrective actions were taken.

At the plant, recovery procedures are in place that use the existing cross-connections to the HPI and CCW systems of the other unit for backing up the systems lost due to the fire event. Incorporating the recovery actions, the conditional probabilities for core damage, given a fire in one of the examined locations, are in the range 8 E-3 to 5 E-2. Because HPI at unit 1 is lost together with CCW (the reason for the seal LOCA), this figure is about a factor 100 higher than the mean safety function unavailability, given an internal event seal LOCA.

The core damage frequencies for the examined fire events are all in the 1 E-6/a to 6 E-6/a range, which is significantly above the range for internally initiated VSLOCA and SLOCA events.

In sum, fire events contribute 14.5 % to the total core damage frequency for internal and external events at power generation. Numerical details are provided in tables 2.1-1 and 5-1.

2.1.2 Level-2 Aspects

The fires lead to core damage accidents by destroying the electric cables or switchgear necessary to power or control the ECCS. The coolant loss from the RCS is due to RCP seal failures in three of the four examined cases, and to stuck-open PORVs in the fourth case. The destruction caused by the fire is considered not to be reparable in the time frame of interest, so there is no possibility of arresting the core degradation process and vessel failure is inevitable for the fire PDS.

Three of the 10 most probable accident progression bins (APBs) involve basemat melt-through; the other seven have no containment failure. Of the five most probable APBs involving early containment failure (before or at vessel breach), two have containment failure due to gross bottom head failure at intermediate pressure, two have containment failure due to an alpha mode failure of both vessel and containment, and one has containment failure due to melt ejection with the RCS at high pressure.

The mean conditional probability of early containment failure, given a fire induced core damage scenario, is 0.008. This is practically the same as for internally initiated core damage events. The mean conditional probability of basemat melt-through, given a fire induced core damage scenario, is 0.26, the mean conditional probability of late overpressure failure of the containment is 0.03, and the mean conditional probability of the containment remaining intact is 0.7, see also tables 2.6-2, 5-1, 5-2, 5-6 and 5-8.

As there are no fire induced containment bypass scenarios (which are the dominant contribution to releases for the overall result), the only significant contribution to large releases is from early containment failure. Fire events contribute 6% to the total exceedance frequency of 10% caesium release, whereas fire events contribute 14.5% to the total CDF, see also tables 2.6-2, 5-1 and 5-8.

2.2 Biblis-B

2.2.1 Level-1 Aspects

The fire frequency for the only critical area is fairly high (2.5 E-3/a), but due to strict separation of safety trains and their power and control cables, the conditional probability that a fire disables vital safety functions is low: the conditional probability that such a fire develops to a safety relevant fire is 1.6 E-3. Thus, frequency of a fire induced scenario with potential for core damage is 4.2 E-6/a. The scenario involves loss of steam generator feeding. Recovery is possible by cross-connecting to the emergency feedwater system of the other unit. Failure of this operator action (conditional probability 4.2 E-2) leads to a high pressure hazard state with frequency 1.7 E-7/a. By accident management action, which is assigned a 0.99 success probability, this can be converted to a low pressure core damage scenario with practically the same frequency, plus a high pressure core damage scenario with frequency 1.7 E-9/a, see tables 2.6-1 and 5-1. Fire induced low pressure core damage scenarios represent 5.3 % of the low pressure scenarios from all events, and high pressure scenarios represent 0.3 % of the high pressure scenarios from all events.

2.2.2 Level-2 Aspects:

Level-2 results are not available, but the small contribution of fire induced events to core damage scenarios, and the fact that no new scenarios are introduced by such events suggests that these events do not provide significant level-2 contributions.

2.3 Beznau

2.3.1 Level-1 Aspects

The fire frequencies for the examined critical locations are in the range 4 E-5/a to 5 E-5/a. In contrast to most other PSAs the fire frequencies include failed fire suppression actions (manual or automatic).

The conditional probability of hazard states with potential for core damage developing from fires is 1 for the significant fire scenarios, because the initiator frequency includes failure of fire suppression actions. The fires disable with certainty electrical equipment that is vital to the operation of safety systems

At the plant, recovery procedures are in place for locally restarting the diesel generators in the event of control room fire. A failure probability of 1.1 E-1 is assigned to these actions. The backup use of the NANO system is credited with a success probability of 0.76.

For switchgear room fires, the use of the independent and diverse NANO system is credited with a success probability of 0.99, see also tables 2.6-1 and 5-1.

2.3.2 Level-2 Aspects

Fire induced core damage is caused by transients resulting from fire damage to vital support and control equipment. The associated accident progression event sequences inside containment are similar to those caused by internal events resulting from failure of support systems or electrical and control equipment. As containment bypass sequences concurrent with fire events are not considered, the relative contribution from fire events to containment failure modes inducive to LRCF modes, (~1.8%), and the relative contribution to large releases (~2.7%), is much smaller than to core damage (18%), see tables 2.6-2, 5-1 and 5-8.

2.4 Peach Bottom

2.4.1 Level-1 Aspects

The frequencies of fires that can develop to significant fire scenarios are ~ 2.4 E-3/a. The conditional probabilities of fire induced damage to vital safety equipment or its control and power cabling is high for the significant scenarios:

- In the event of a control room fire, a general transient occurs, and forces abandonment of the control room due to heavy smoke (conditional probability 4.2 E-2). Failure of the operators to control the plant from the remote shutdown room (conditional probability 6.4 E-2) leads to core damage with frequency 6.2 E-6/a.

- In the two most significant switchgear room scenarios, fire damage fails both offsite power trains (conditional probability 0.38). In addition,

  - random failure of the emergency service water system (conditional probability 1.4 E-2) causes loss of cooling to the emergency diesel generators, leading to station blackout. In the long term, HPCI fails due to loss of room cooling or battery depletion, or

  - fire damage fails power to the emergency cooling water pump or the emergency service water pump. Coupled with additional random failures (conditional probability 1.5 E-2), this leads to the loss of the emergency service water system and station blackout. In the long term, HPCI fails due to loss of room cooling or battery depletion.

- The core damage frequency of these two scenarios together is 1.3 E-5/a, and the frequency of all fire induced core damage scenarios is 2 E-5/a. This makes fire events the major contributors to core damage (71 %), see also tables 2.6-3 and 5-1.

2.4.2 Level-2 Aspects

For fire initiated core damage, the conditional probability of early containment failure is high (0.63) due to the nature of the fire events, most of which do not have AC power or injection. This leads to a high probability of drywell melt-through since the drywell will, at best, only have water in the reactor cavity sump. Early containment failure due to fire makes up 71 % of all early containment failure events. In relation to other initiators (internal and seismic), the fire events are dominated by scenarios (66%) that do not allow for the recovery of injection or containment heat removal and are similar to long-term station blackout sequences. The impossibility of recovering injection or containment heat removal, including containment venting, means that the containment failure probability is very high since the base pressure inside the containment can not be reduced before vessel breach and long term containment failure from overpressure can not be mitigated.

As the fire induced core damage scenarios are similar to internally initiated station blackout scenarios (which dominate internal events), the conditional probabilities for containment failure modes and exceedance of 1 % and 10 % releases are similar for fire events and internal events (see tables 2.6-4 and 5-1 to 5-8).

2.5 LaSalle

2.5.1 Level-1 Aspects

Fire induced core damage frequencies >5E-7/a were found for the:

- main control room,
- turbine building corridor,
- electrical equipment room,
- auxiliary equipment room,
- division 1 essential switchgear room,
- division 2 essential switchgear room,
- cable shaft area adjacent to division 2 essential switchgear room

This is to be seen in relation to the overall total (mean) core damage frequency 3.2E-5/a.

In the case of the control room, smoke-induced abandonment and subsequent transfer of control of the plant to the remote shutdown panel was found to be the dominant scenario. Installation of an independent and redundant remote shutdown panel significantly reduced the core damage frequency contribution from this area.

In all other areas, additional failures of equipment and/or operators are required (conditional probabilities 3 E-3 to 1 E-2) to lead to core damage. Adequate separation of equipment and/or cabling between redundant functions had the effect of reducing core damage frequency for these areas by preventing fire induced failures of redundant system trains.

Control, power, and actuation circuitry/cabling for most of the plant safety functions could be damaged by fires in cable "pinchpoint" areas.

Fires in two locations contribute 85 % to the total fire induced core damage frequency:

- Control room fires, with frequency 1.3 E-5/a. The greatest potential for reducing this contribution lies in the improvement of operator actions to control the plant from the remote shutdown panel, and in the sealing of the tops of control room cabinets.

- Fires in the essential switchgear area with frequency 1.4 E-5/a. All fire scenarios in this area involve fire damage to one of the two injection trains and/or residual heat removal trains and random failure of the other residual heat removal train, see also table 2.6-3.

2.5.2 Level-2 Aspects

In the level-2 part of the LaSalle PSA, numbers different from the level-1 part are presented: In NUREG/CR-4832, SAND92-0537, Vol. 9 (Level-1), the core damage contribution from fire events is 3.2 E-5/a, and in NUREG/CR-5305, SAND90-2756, Vol. 1 (Level-2), the core damage contribution from these events is 5.5 E-5/a. In the following, the frequencies per year are scaled to the values presented in the level-1 analysis. Conditional probabilities, given fire events, are not affected by the differences of frequencies.

The following summary accident progression bins were examined in detail:

Core Damage Arrest, No Reactor Vessel Failure

- Given core damage caused by fire events, the mean value for the conditional probability of core damage arrest is 0.014. The reason for this low value is that only in one of the six PDSs that form the summary fire PDS can injection be recovered. This PDS contributes 17% of the fire group’s conditional probability of core damage.

- For all accident sequences together (internal and external events) the mean conditional probability of core damage arrest is 0.17.

Early Containment Failure

The following events are considered as possible causes for early containment failure:

- Before vessel breach: pressurisation of the containment caused by the generation of steam from a boiling suppression pool during accidents in which containment heat removal is lost or inadequate.

- At the time of vessel breach:

  - large in-vessel steam explosion events.

- fast pressurisation of the containment caused by the loads attending vessel breach. These loads include contributions from direct containment heating (DCH), ex-vessel steam explosions, and reactor vessel blowdown.

- failure of the reactor pedestal caused either by quasi-static pressurisation loads accompanying vessel breach or by dynamic loads associated with ex-vessel steam explosions. It is postulated that failure of the reactor support will place a large amount of stress on the piping penetrations, leading to failure of penetrations and of the drywell wall.

- Ex-vessel steam explosion that fails the cavity drain pipe beyond the containment wall.

The mean conditional probability of early containment failure, given core damage caused by fire, is approximately 0.30; nearly all of these failures occur before core damage. The mean conditional probability of early containment venting is 0.61. Therefore, the mean conditional probability that the containment integrity is lost early in the accident is 0.91.

For all internal and external events, the mean conditional probability of early containment failure is 0.25, and the mean conditional probability for early containment venting is 0.30.

Loads Resulting from Core-Concrete- Interaction:

- Late in the accident, the events that result in containment failure include the slow pressurisation of the containment from steam and non-condensable gases generated during core-concrete-interaction (CCI), and failure of the reactor pedestal caused by concrete erosion during CCI. In the analysis it is assumed that dry CCI always leads to containment failure

- The conditional mean probability for dry CCI, given core damage caused by fire events, is very high (0.98) because of the lack of recoverable injection systems .

- For all internal and external events, the mean conditional probability for dry CCI is 0.40, and the mean conditional probability for CCI with injection available is 0.28.

The frequencies of exceeding 1%, respectively, 10% releases is 1.4 E-5/a (about 48% of the releases for all accident sequences), respectively, 8 E-6/a (about 66% of the releases for all accident sequences). (see tables 2.6-4 and 5-1 to 5-8).

2.6 General Observations

Defences on four levels protect nuclear power plants against the impact from fires:

1. Keeping low the amount of combustible material in areas containing vital safety equipment or its cabling, and using fire resistant insulation material for electrical equipment.

2. Providing equipment for fire detection and for manual and automatic fire fighting. The purpose of these provisions is to prevent small fires from propagating to large, safety relevant fires.

3. separation of redundant or diverse equipment, and by fire resistant barriers. The purpose of separation is to prevent fire damage to multiple safety system trains and to preserve the availability of the redundancies needed for coping with fire induced events.

4. Implementation of recovery and accident management actions to mitigate the impact of fires by providing backup to safety equipment that has been disabled by fire events, and to preserve the retention capability of the containment.

The failures most frequently identified as causes for fire induced core damage are:

- loss of plant control due to fire induced damage to electrical cabling or switching equipment, or to failed transfer of control to the emergency control center,

- fire induced failure of component cooling water,

- fire induced failure of offsite power trains, combined with random failure of service water system,

- fire induced failure of 1v2 injection trains or 1v2 residual heat removal trains, combined with random failure of the second residual heat removal train.

To improve the plant's defences against fires, numerous modifications and backfits were implemented:

- At Surry: in response to the requirements in appendix R, procedures were put in place for cross-connecting to the HPI and CCW system of the other unit in the event of irrecoverable damage to power or control cables of the HPI and CCW system. The possibility for cross-connecting to these systems at the other unit had already been implemented to improve the plant response to certain internal events. No modifications and backfits were made to reduce the vulnerability to fires in the switchgear room by improving the separation of cables in this room.

- At LaSalle: installation of an independent remote shutdown panel for controlling the plant in the event of forced abandonment of the main control room.

- Putting in place of procedures for controlling the plant from the remote shutdown panel in the event of

  - fire induced failure of control cabling in the control room or forced abandonment of the control room due to smoke (Surry, Peach Bottom, LaSalle)

  - fire induced failure of control cabling in the cable spreading room (Peach Bottom)

- Putting in place of procedures for local operation of valves and pumps in areas not affected by the fire in the event of disabled cabling. (Surry)

- Installation of improved fire prevention and fire fighting equipment:

  - Computer based fire alarm system with alarm identification (Biblis-B)

  - Additional fire extinguisher trains (5 additional fire hydrants) (Biblis-B)

  - Water sprinklers, e.g. for cable ducts (Biblis-B)
  - Fire protection surveillance of activated carbon filters by CO measurement (Biblis-B)
  - CO_2 fire extinguishers for primary system, oil supply of main coolant pumps, turbine and switch gear (Biblis-B)
  - Provision of mobile equipment for the foam fire extinguishing system (Biblis-B).
  - Replacement of oil cooled transformers by air cooled transformers (Beznau).
  - Installation of an emergency connection to a third battery to provide redundancy to the main DC power supply (Beznau).
  - Improvement of fire resistance of piping and cable penetrations (Beznau).
  - Installation of metal cable sleeves for fire protection of cables inside containment (Beznau).
  - Replacement of cables with combustible insulation by cables with slow-burning insulation (Beznau).
  - Improvements to fire detectors and fire fighting equipment (Beznau).

- Improving separation between redundant equipment:
  - Installation of a fire proof door between the rooms for the two trains of the secured 220 V DC power supply: (Biblis-B).
  - Self-locking of the door in the closed position (Biblis-B).
  - Triggering of automatic closure of the door by signals from fire detectors, should the door be open , for example, during maintenance activities (Biblis-B).
  - Installation of a fire wall in the turbine building that separates the equipment for house load electrical power from the remaining turbine building (Beznau).
  - Improved separation of equipment and/or cabling between redundant functions (LaSalle).

- Putting in place of procedures for primary side bleed to prevent high pressure core damage scenarios, including re-establishing 10 kV and 380 V AC power (Biblis-B).

- Installation of the NANO system, consisting of diverse, independent safety systems for emergency core cooling and emergency feeding of steam generators. In the external event protected NANO building, fire areas are strictly separated. Since most fire scenarios do not affect the NANO systems, these systems can be used for plant shut down in the event of fire from the emergency control room in the NANO building (Beznau).

  Remark: Improving the defences against fires was not the only reason for the installation of the NANO system.

- Provision of mobile diesel generators for station blackout scenarios and putting in place of procedures for their use (Peach Bottom).

In probabilistic terms, the effectiveness of defences at the four levels described above can be measured by:

- The fire frequency for critical areas. In general, this figure is in the range E-3/a to E-2/a. Variation of the numerical values among different plants is due the variation of the size of critical areas and of the number of fire sources in critical areas (for example, insulation material of cabling). At some plants, this figure already includes fire suppression measures and is therefore lower

- The conditional probability for a hazard state with potential for core damage developing from a fire in a certain area. Inspection of table 2.6-1 shows a clear advantage of plants with systematic separation of safety system trains (for example Biblis-B), over plants that do not have this feature.

- Several plants have only two trains for injection, residual heat removal and vital support functions. With the frequencies of fire induced damage to one of the trains, as developed in the examined studies, and random failures of the respective other trains, with conditional probabilities in the range ~3 E-3 to ~1 E-2, core damage contributions of the order E-5/a result for plants with such features. Without significant design changes, reduction of the usually dominant core damage contributions would require improved fire suppression and fire protection to reduce the initiator frequency for the critical areas.

- At Beznau, the addition of the independent NANO emergency system significantly reduced the high contribution from fire events determined for the original design.

- The conditional failure probabilities for recovery and accident management actions are in the range ~1 E-2 to ~E-1 and generally show good agreement among the examined studies for comparable boundary conditions.

- In plants with high redundancy and systematic functional and physical separation of safety system trains, the contribution to core damage from fire events is insignificant. On the contrary, plants with only 2 safety trains and vital support system trains tend to have dominant contributions from fire events.

- In none of the examined studies do fire events lead to severe accident loading of the containment that would not also attend internally initiated accident sequences.

- Conversely, concurrent fire and containment bypass events are not considered for reasons of low frequency. Therefore, the relative contribution of fire induced events to LRCF modes and to large releases at PWR plants is generally smaller than the relative contribution of fire induced events to core damage (see tables 2.6-2, 5-1 and 5-8).

In the BWR studies, risk dominant fire induced event sequences and risk dominant internally induced event sequences are generally similar. With the exception of the LaSalle study, the relative contribution of fire induced events to early containment failure modes and to large releases differ little from the relative contribution of fire induced events to core damage (see tables 2.6-4, 5-1, 5-2, 5-6 and 5-8).

- At LaSalle, two issues contribute disproportional to large releases:

    1. the high likelihood of opening up a release path through the venting line early in an accident

    2. the very high conditional probability of dry CCI, which is due to the almost guaranteed absence of means of injection.

The importance of the frequencies of fire induced core damage, LRCF modes and exceedance of 10% Cs release, relative to these frequencies, given any event, is illustrated by figure 2.6-1

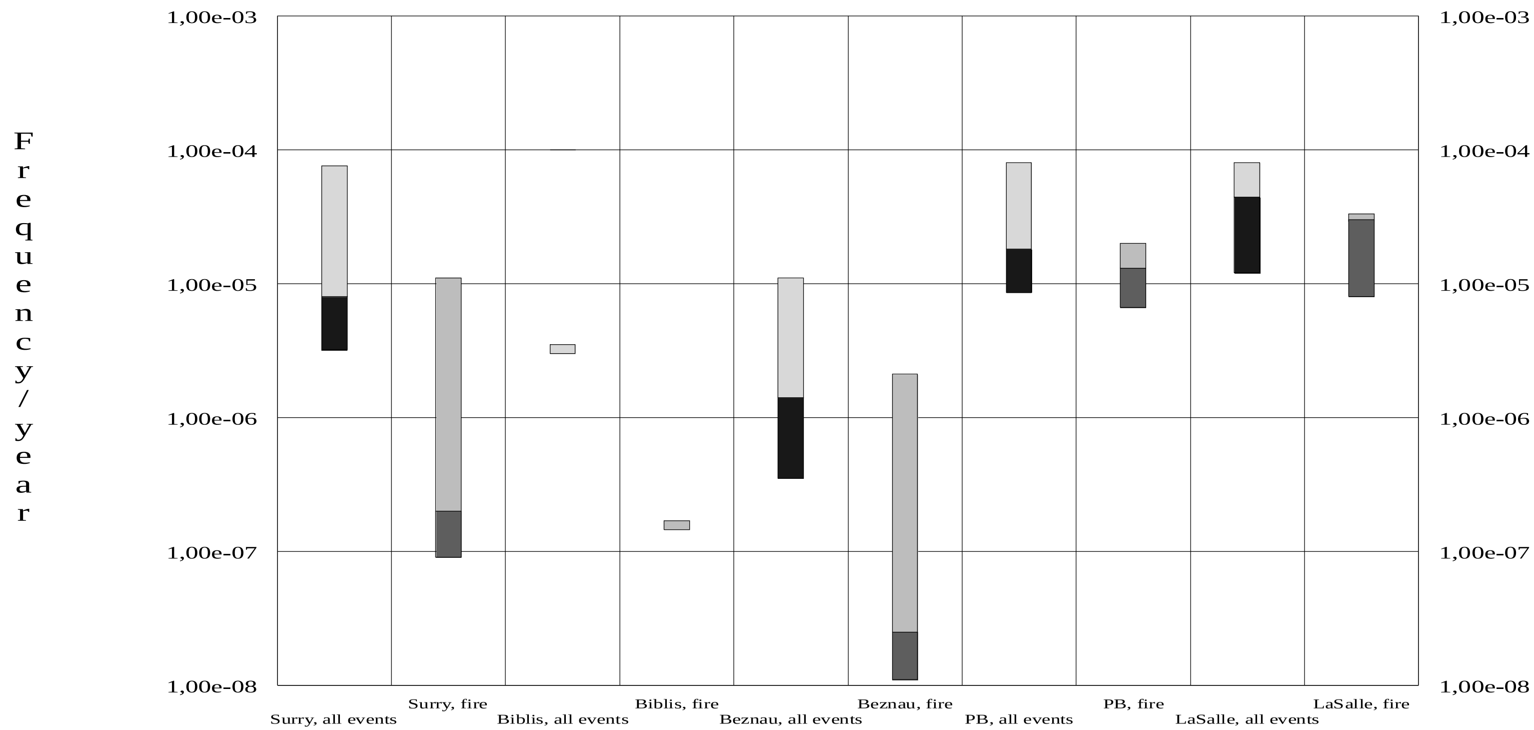

Figure 2.6-1. Frequencies of core damage, LRCF mode and exceedance of 10% Cs release for all events and fire events. Top of light grey bars: CDF, all events, bottom of light grey bar: frequency of LRCF mode, all events, bottom of *solid* black bars: exceedance frequency of 10% Cs release, all events. Top of orange bars: CDF, fire events, bottom of orange bar: frequency of LRCF mode, fire events, bottom of red bars: exceedance frequency of 10% Cs release, fire events. Length of grey and orange bars: capability of the containment to prevent LRCF mode, given CD. Length of black and red bars: measure of the capability of containment to mitigate large releases, given LRCF mode.
(For Biblis, level 2 results are not available.)

Table 2.6-1. Initiator Frequencies/a (Mean), Conditional Probabilities for Fire Propagation, Conditional Probabilities of Core Damage, Given a Safety Relevant Fire in Sensitive Areas, Core Damage Contributions from Fires for Power Generation States, PWRs.

	Surry-1	Biblis-B	Beznau
Fire frequency/a for:			
Control room	1.8 E-3		1.5 E-3
Cable spreading room/vault/tunnel	7.5 E-3		
Switchgear room	8.0 E-3	7.5 E-3	2.2 E-3
Auxiliary building	6.6 E-2		
Total	8.7 E-2		
Fire frequency/a for critical area(s) in:			including fire suppression
Control room	Benchboard 1-1 1.8 E-4		Panel TP9 4.8 E-5
Cable spreading room/vault/tunnel	1.8 E-4		
Switchgear room	3 E-4	2.5 E-3	4.2 E-5
Auxiliary building	4 E-5		
Total	7 E-4		
Conditional probability of a hazard state with potential for core damage (safety relevant fire) developing from a fire in the:			excluding fire suppression
Control room	1.2 E-1		1
Cable spreading room/vault/tunnel	1.8 E-1		
Switchgear room	4.6 E-1	1.6 E-3	1
Auxiliary building	2.1 E-1		
Conditional probability of failure of recovery, given a safety relevant fire in the:			includes recovery using NANO
Control room	7.4 E-2		2.6 E-2
Cable spreading room/vault/tunnel	4.4 E-2		
Switchgear room	4.4 E-2	4.2 E-2	1 E-2
Auxiliary building	2.6 E-1		
Total conditional probability of core damage, given a fire in critical area(s) of the:			excluding fire suppression
Control room	8.8 E-3		2.6 E-2
Cable spreading room/vault/tunnel	7.9 E-3		
Switchgear room	2 E-2	6.7 E-5	1 E-2
Auxiliary building	5.5 E-2		
CDF contribution from fire in the			
Control room	1.6 E-6		1.2 E-6
Cable spreading room/vault/tunnel	1.5 E-6		
Switchgear room	6.1 E-6	1.7 E-7	4.2 E-7
Auxiliary building	2.2 E-6		
Total Percent of internal + external events	1.1 E-5 14.5%	1.7 E-7 4.7%	2.1E-6 18%

Table 2.6-2. Core Damage Contributions from Fires, Conditional Probabilities for Important Containment Failure Modes, Conditional Probabilities for Significant and Large Releases, Given Core Damage Induced by Fire and by any Event, Power Generation States at PWRs.

	Surry-1	Biblis-B	Beznau
CDF contribution from fire in the			
Control room	1.6 E-6		
Cable spreading room/vault/tunnel	1.5 E-6		
Switchgear room	6.1 E-6		
Auxiliary building	2,2 E-6		
Total CDF fire contribution Percent of internal+external events	1.1 E-5 14.5 %	1.7 E-7 4.7 %	2.1 E-6 18 %
Conditional probabilities for containment failure modes and *major contributions*			
Early containment failure given a fire induced core damage scenario given any core damage scenario	 0.008 *DCH* 0.03 *DCH*		 0.012 *DCH* 0.013 *DCH*
Late containment failure given a fire induced core damage scenario given any core damage scenario	 0.27 *BMT* 0.16 *BMT*		 0.2 *vent fail.* 0.18 *vent fail.*
Containment bypass + Isolation failure given a fire induced core damage scenario given any core damage scenario	 - 0.06 *SGTR*		 - 0.11 *ISF*
Containment intact given a fire induced core damage scenario given any core damage scenario	 0.72 0.73		incl.. venting 0.76 0.7
Frequencies and conditional probabilities for releases			
Exceedance frequency/a for fire induced 1% Cs release percent of internal+external events Conditional probability for exceeding 1% Cs release, given a fire induced core damage scenario given any core damage scenario	7.7 E-7 9,7 % 0.07 0.1		3.9 E-7 20 % 0.18 0.2
Exceedance frequency/a for fire induced 10% Cs release, percent of internal+external events Conditional probability for exceeding 10% Cs release, given a fire induced core damage scenario given any core damage scenario	2 E-7 6.2 % 0.02 0.04		1.1 E-8 2.7 % 0.005 0.03

Table 2.6-3. Initiator Frequencies, Conditional Probabilities for Fire Propagation, Conditional Probabilities CDF, Given a Safety Relevant Fire in Sensitive Areas, Core Damage Contributions from Fires for Power Generation States, BWRs

	Peach Bottom-2	LaSalle-2	Barsebäck-1	Barsebäck-2
Fire frequency/a for:				
Control room	2.3 E-3	1.8 E-3		
Cable spreading room/vault/tunnel	3.5 E-3	6.5 E-3	1.3 E-4 (room A)	1.3 E-4(room A)
Switchgear room	2.7 E-3	8 E-3		
Auxiliary equipment room		4.9 E-2	1.3 E-3	1.3 E-4
Total	8.5 E-3	6.5 E-2	7.4 E-3	6.6 E-3
Fire frequency/a for critical area(s) in:				
Control room	2.3 E-3	1.8 E-3		
Cable spreading room/vault/tunnel	1.3 E-4	1 E-3	1.3 E-4 (room A)	1.3 E-4(room A)
Switchgear room	2.4 E-3	8 E-3		
Auxiliary equipment room		2.7 E-3	1.3 E-3	1.3 E-4
Total	4.7 E-3	1.3 E-2	7.4 E-3	6.6 E-3
Conditional probability of a hazard state with potential for core damage (safety relevant fire) developing from a fire in critical area(s) of the:				
Control room	4.2 E-2	1 E-1		
Cable spreading room/vault/tunnel	8 E-2	1.6 E-4, incl. random failures, 1.4 E-2	1.1 E-2	1.1 E-2
Switchgear room	5.4 E-3 (2 dominant sequences, both including random failures of ESW, or, DG, 1.4 E-2)	1.7 E-3 (2 dominant sequences, both including random failure of 1 RHR train, ~5 E-3)		
Auxiliary equipment room		8.1 E-4 (including random failures, 3 E-3)	1 E-2	1 E-2

Table 2.6-3. Initiator Frequencies, Conditional Probabilities for Fire Propagation, Conditional Probabilities CDF, Given a Safety Relevant Fire in Sensitive Areas, Core Damage Contributions from Fires for Power Generation States, BWRs

Conditional probability of failure of recovery, given a safety relevant fire in the:				
Control room	6.4 E-2	7.2 E-2		
Cable spreading room/vault/tunnel	6.4 E-2	1	1	1
Switchgear room	1	1		
Auxiliary equipment room		1	1	1
Total conditional probability of core damage, given a fire in the				
Control room	2.7 E-3	7.2 E-3		
Cable spreading room/vault/tunnel	5.1 E-3	1.6 E-4		
Switchgear room	5,4 E-3	1.7 E-3		
Auxiliary equipment room		8.1 E-4		
CDF contribution from fire in the				
Control room	6.2 E-6	1.3 E-5		
Cable spreading room/vault/tunnel	6.7 E-7	1.6 E-7	1.4 E-6	1.4 E-6
Switchgear room	1.3 E-5	1.4 E-5		
Auxiliary equipment room		2.2 E-6	1.4 E-5	1.4 E-6
Total fire Percent of internal + external events	2.0 E-5 71.5%	3.2 E-5 40 %	7.6 E-5 95%	6.8 E-5 94%

Table 2.6-4. Core Damage Contributions from Fires, Conditional Probabilities for Important Containment Failure Modes, Conditional Probabilities for Significant and Large Releases, Given Core Damage Induced by Fire, and by any Event, for Power Generation, BWRs

	Peach Bottom	LaSalle	Barsebäck-1+2
CDF contribution from fire in the:			
Control room	6.2 E-6	1.4 E-5	
Cable spreading room/vault/tunnel	6.7 E-7	1.6 E-7	
Switchgear room	1.3 E-5	1.4 E-5	
Auxiliary equipment room		2.3 E-6	
Total CDF fire contributions Percent of internal + external events	2.0 E-5 71.5%	3.2 E-5 40%	
Conditional probabilities for containment failure modes and *major contributions*			
Early containment failure: - given a fire induced core damage scenario - given any core damage scenario	0.63 *DWF* 0.64 *DWF*	incl. early vent 0.91 0.55	
Late containment failure: - given a fire induced core damage scenario - given any core damage scenario	incl. venting 0.12 *OPR* 0.16 *OPR*	 0.06 0.2	
Containment bypass + Isolation failure: - given a fire induced core damage scenario - given any core damage scenario	 - -	 - -	
Containment intact: - given a fire induced core damage scenario - given any core damage scenario	 0.24 0.21	 0.03 0.25	
Frequencies and conditional probabilities for releases			
Exceedance frequency for fire induced 1% Cs release percent of internal+external events Conditional probability for exceeding 1% Cs release: - given a fire induced core damage scenario - given any core damage scenario	1.1 E-5 67% 0.55 0.57	1.4 E-5 48% 0.44 0.36	
Exceedance frequency for fire induced 10% Cs release percent of internal+external events Conditional probability for exceeding 10% Cs release: - given a fire induced core damage scenario - given any core damage scenario	6.6 E-6 77% 0.33 0.3	8 E-6 66% 0.25 0.15	

3. PROBABILISTIC ASPECTS OF FLOODING RELATED FAILURES

Plant specific probabilistic aspects of flooding events are presented for the PWR plants Biblis-B, Konvoi, Beznau, Robinson and the BWR plants Peach Bottom and LaSalle. In the concluding section, generic probabilistic aspects of flooding events are discussed.

3.1 Surry

Core damage scenarios are examined but not further analysed for reasons of low occurrence frequency of flooding events. However, the cut-off frequency of 1 E-6/a is higher than in many other PSAs. It is to be expected that some of the eliminated sequences would be included in other PSAs.

3.2 Biblis-B

3.2.1 Level-1 Aspects

Biblis-B has a steel shell containment surrounded by a concrete structure. Vital systems are located in the annulus. Therefore, flooding of the annulus is a critical scenario.

Flooding of the annulus with water level between 70 cm and 90 cm

- The occurrence frequency for flooding events is 4 E-3/a. This figure includes breaks in the piping, as well as leaks caused by maintenance errors.
- The conditional probability for the initiating event to develop to a safety relevant flooding scenario because of failure to reduce or terminate the flow into the annulus before exceeding 70 cm water level is ~ 2 E-4, and the annual frequency of the corresponding flooding scenario is 9 E -7/a.
- The failure probability for recovering disabled system functions or using alternate systems is estimated to be less than 0.1, leading to < E-7/a for the frequency of a hazard state with potential for core damage.

Flooding of the annulus with water level exceeding 90 cm

- The occurrence frequency for flooding events is 4 E-3/a. This figure includes breaks in the piping, as well as leaks caused by maintenance errors.
- The conditional probability for the initiating event to develop to a safety relevant flooding scenario because of failure to reduce or terminate the flow into the annulus before exceeding 90 cm water level is ~ 1 E-3, and the annual frequency of the corresponding flooding scenario is 4 E-6/a.
- The failure probability for aligning emergency feedwater to the steam generators for heat removal is estimated to be less than 5 E-2, leading to < 2 E-7/a for the frequency of a hazard state with potential for core damage.

- In the analysis, no credit was given to accident management actions to utilise alternate steam generator feeding for avoiding core damage. Thus, the core damage frequency is the same as the hazard state frequency, it represents 5.5% of the total annual core damage frequency.
- For preventing high pressure core melt by depressurisation of the primary system, a success probability of 0.97 was estimated, see tables 3.10-1 and 5-1.

3.2.2 Level-2 Aspects

Level-2 results are not available.

3.3 Konvoi

The Konvoi plants have a steel shell containment surrounded by a concrete structure. Vital systems are located in the annulus. Therefore, flooding of the annulus is a critical scenario.

3.3.1 Level-1 Aspects

Scenarios involving flooding of the annulus were analysed. Even without AM measures, the contributions to CDF by flooding scenarios is very low (6 10^{8}).

3.3.2 Level-2 Aspects

Level-2 results are not available.

3.4 Beznau

3.4.1 Level-1 Aspects

The flooding frequencies are caused by pipe breaks:

- anywhere in the turbine hall, with annual frequency 1.2 E-3/a, and
- in the switchgear room in the turbine hall, with annual frequency 6.8 E-5/a.

The difference of the numerical values is due to different length of piping in the two areas.

The conditional probabilities of hazard states with potential for core damage developing from flooding of the pump yard is 5 E-2, reflecting the failure probability of operator actions to identify and isolate the broken piping in a timely fashion. The steam flood in the switchgear room fails electrical equipment with certainty. As the NANO systems are not affected by the two flooding events, they can be used for keeping the plant in a safe state. For both scenarios, a conditional probability of 0.99 was estimated for the success of the required actions.

For the switchgear room steam flood, a core damage contribution of 6.7 E-7/a was obtained, for the pump yard flooding the number is 6.5 E-7/a. Together with several other events with frequencies below 1 E-8/a, the total contribution from flooding to core damage is 1.5 E-6/a, which represents 12.6% of the total core damage frequency due to internal and external events, see tables 3.10-1 and 5-1.

3.4.2 Level-2 Aspects

Flooding induced core damage is caused by transients resulting from flood damage to vital support and control equipment. The associated accident progression event sequences inside containment are similar to those caused by internal events resulting from failure of support systems or electrical and control equipment. Containment bypass sequences concurrent with flooding events are not considered, therefore, the relative contribution from flooding events to containment failure modes inducive to large releases, LRCF, (~1.5%), and the relative contribution to large releases (~2.5%), is much smaller than to core damage (12.6%); see tables 3.10-2 and 5-1 to 5-8.

3.5 Robinson

3.5.1 Level-1 Aspects

The dominating event is a large service water (SW) flood in the auxiliary building area due to maintenance errors or component rupture. The frequency of the initiating event is 3.4 E-4/a. The response time for mitigative actions preventing critical flooding is only 3 min, therefore, no credit is given to them. The conditional probability for core damage, given this initiating event, is 0.16. The associated core damage frequency is 5.6 E-5/a, representing 17% of the total core damage frequency due to internal and flooding events.

The next important event is a maintenance induced service water system flood, also in the auxiliary building. The frequency of the initiating event is 4.1 E-4/a. The response time for operator actions to prevent critical flooding is 20 min. The conditional probability of failure to prevent the flooding to rise to critical level is 0.05, leading to a frequency of 2.1 E-5/a for critical flooding. Critical flooding leads to the loss of main feedwater and both motor-driven auxiliary feedwater pumps. The operations staff fails to control the auxiliary feedwater steam-driven pump (conditional probability 0.34) causing a loss of all secondary-side cooling. The flood also fails both residual heat removal pumps which, in turn, fails recirculation after successful feed-and-bleed cooling. The associated core damage frequency is 7.2 E-6/a, representing 2% of the total core damage frequency due to internal and flooding events.

All other flooding events contribute less than 1% to the total core damage frequency. The total contribution to core damage from internal flooding events is 22%, see tables 3.10-1 and 5-1.

3.5.2 Level-2 Aspects

The dominating flooding events cause irrecoverable damage to vital safety systems, leaving no chance for arresting progression of core damage and avoiding reactor pressure vessel failure.

Flooding sequences concurrent with containment bypass scenarios are not considered. Therefore, the relative contribution from flooding events to containment failure modes that are inducive to large releases is below the corresponding figures for internal events. Flooding events contribute 22% to core damage, but only 10% to LRCF modes, see tables 3.10-2 and 5-1 to 5-8.

3.6 Peach Bottom

Core damage scenarios were examined but not further analysed for reasons of low occurrence frequency of flooding events. However, the cut-off frequency of 1 E-6/a is higher than in many other PSAs. It is to be expected that some of the eliminated sequences would be included in other PSAs.

3.7 Browns Ferry

3.7.1 Level-1 Aspects

The initiator frequency for the only significant flood induced core damage scenarios in the turbine building is 5 E-2/a. Flood induced damage to safety relevant equipment does not directly lead to core damage. Additional random failures are required for core damage to occur:

- HPCI, RCIC and I+C relating to depressurisation of the RPV, or

- 250V DC battery boards for control power of HPCI, RCIC and I+C relating to depressurisation of the RPV.

The conditional probability for the flooding events to lead to core damage is 9.4 E-5.

The flood induced core damage frequency is 4.7 E-6/a, which is ~10% of the total core damage frequency, see tables 3.10-3 and 5-1.

3.7.2 Level-2 Aspects

About 70% of the flood induced core damage scenarios are associated with a “no containment failure” release category, and thus are attended by very small releases. The remaining ~30% are associated with a release category that is responsible for early containment failure and large releases (exceedance frequency of 10% Cs release: due to this group is 5 E-6/a). It contributes ~5% to the annual frequency of this group. More detailed information is not available, but it can be supposed that the relative contributions of flood induced early containment failure and large releases to the totality of such events are lower than 10% (which is the relative contribution of flood induced core damage to total core damage, see tables 3.10-4 and 5-1 to 5-8.

3.8 Perry

3.8.1 Level-1 Aspects

- Flooding of Lowest Level of the Control Complex:

  The flooding frequency for the zone is 2.6 E-3/a.
  The contributions from medium flood and small floods are 29% and 18%, respectively, of the total flooding induced core damage frequency. The frequency of a severe flood is low so the total contribution from this flood is approximately 10%.
  The mean conditional probability for core damage, given a flooding event in the zone, is 3.4 E-4.
  The core damage frequency from flood in this area is 8.8 E-7/a which is 57 percent of the total contribution from flooding.

- Flooding of the Second Level of the Control Complex

  The flooding frequency for the zone is 1.2 E-3/a.
  The contribution from moderate, small and severe floods are 11%, 6% and 4%, respectively, of the total flooding induced core damage frequency
  The mean conditional probability for core damage, given a flooding event in the zone, is 2.7 E-4.

- Flooding of the Turbine Building

  The flooding frequency for the zone is 6 E-3/a.
  The contribution to core damage comes from floods as the result of a small expansion joint failure, it is 11%, followed by floods in either the turbine power complex west or east rooms (3% each)
  The mean conditional probability for core damage, given a flooding event in the zone, is 4.7 E-5.
  The core damage frequency from floods in the turbine building and turbine power complex is 2.8 E-7/a which is 18% of the total contribution from flooding.

In sum, the flooding induced core damage frequency is 1.5 E-6/a, which accounts for ~12% of the total core damage frequency, see tables 3.10-3 and 5-1.

3.8.2 Level-2 Aspects

Almost all flooding induced core damage scenarios are grouped into a plant damage state characterised by containment intact and successful containment venting. Flooding events contribute 44% to this plant damage state. 98% of this plant damage state contribute to a source term category with very low Cs release fraction.

No detailed information is available on the contribution from flooding events to source term category with high releases, but it can be supposed that the frequency of exceeding 10% Cs release induced by flooding events is ~5 E-8/a, and that the frequency of exceeding 1% Cs release fraction induced by flooding events is ~5 E-7/a, see tables 3.10-4 and 5-1 to 5-8.

3.9 LaSalle

3.9.1 Level-1 Aspects

There are two significant initiators to flooding events, both being pipe breaks in the service water system in the reactor building. The sum of their frequencies is 3.6 E-5/a. They develop to safety relevant flooding events if the operators fail to isolate the leak within 7 minutes. A conditional failure probability of 9.1 E-2 is assigned to the respective actions. The flooding event fails the power conversion system, main feed water and the condensate system with certainty. Safety systems needed for coping with the ensuing transient are also failed with certainty, making core damage inevitable. Its frequency is 3.4 E-6/a, representing 4.2 % of the total core damage frequency from internal and external events, see tables 3.10-3 and 5-1.

3.9.2 Level-2 Aspects.

In the level-2 part of the LaSalle PSA, numbers different from the level-1 part are presented: In NUREG/CR-4832, SAND92-0537, Vol. 9 (Level-1), the core damage contribution from fire events is 3.2 E-5/a, and in NUREG/CR-5305, SAND90-2756, Vol. 1 (Level-2), the core damage contribution from fire events is 5.5 E-5/a. In the following section, the frequencies per year are scaled to the values presented in the level-1 analysis. Conditional probabilities, given flooding events, are not affected by the differences in frequencies.

The following summary accident progression scenarios were examined in detail:

Core Damage Arrest, no Reactor Vessel Failure

- Given core damage caused by flooding events, the mean value for the conditional probability of core damage arrest is 0.27. For the flooding PDS groups, the likelihood of core damage arrest is relatively high. For flooding events, the diesel driven firewater pumps are available. Therefore, it is likely that the operators will depressurise the RPV to allow firewater to be injected in the core. Furthermore, there is a high likelihood that the containment pressure will remain low either through the use of the containment heat removal systems or by venting. Thus, it is highly likely that the firewater system can be used to inject water into the core during core damage.

- For all accident sequences together (internal and external events) the mean conditional probability of core damage arrest is 0.17.

Early Containment Failure

- The following events are considered as possible causes for early containment failure:

  - Before vessel breach: pressurisation of the containment caused by the generation of steam from a boiling suppression pool during accidents in which containment heat removal is lost or inadequate.

  - At the time of vessel breach:

    - large in-vessel steam explosion events (that cause the vessel breach)

  - fast pressurisation of the containment caused by the loads attending vessel breach.

  - failure of the reactor pedestal caused either by quasi-static pressurisation loads accompanying vessel breach by dynamic loads associated with ex-vessel steam explosions. It is postulated that failure of the reactor support and subsequent gross motion of the RPV will place a large amount of stress on the piping penetrations, leading to failure of penetrations and drywell wall.

  - Ex-vessel steam explosion that fails the cavity drain pipe beyond the containment wall.

- Most flooding events involve accidents in which all injection is lost at the beginning of the accident, resulting in a subcooled suppression pool and a low containment pressure during the early phase of the accident. Thus, neither containment failure nor venting occurs before vessel breach for these PDSs. The mean conditional probability of early containment failure, given core damage caused by flooding, is approximately 0.20; all of these failures occur at the time of vessel breach. The mean conditional probability of late containment venting is 0.78. This high likelihood of late containment venting can be attributed to the fact that for flooding events AC power is always available and the vent valves are operable. Therefore, it is very likely (conditional probability 0.88) that the containment integrity will be lost either due to containment failure or late containment venting.

- For all internal and external events the mean conditional probability of early containment failure is 0.25, and the mean conditional probability for early containment venting is 0.30.

Loads Resulting from Core-Concrete- Interaction

- Late in the accident, the events that result in containment failure include the slow pressurisation of the containment from steam and non condensable gases generated during core-concrete-interaction (CCI), and failure of the reactor pedestal caused by concrete erosion during CCI.

- The conditional mean probability for dry CCI, given core damage caused by flooding events, is relatively low (0.09) because of the frequently recoverable injection systems. However, CCI is still likely to occur for accidents proceeding to vessel breach (the conditional probability, given a flooding event, is 0.53), but with an overlying pool of water.

For all internal and external events, the mean conditional probability for dry CCI is 0.45, and the mean conditional probability for CCI with injection available is 0.33.

The frequencies of exceeding 1%, respectively, 10% releases is 7.2 E-7/a (about 2.4% of the releases for all accident sequences), respectively, 1.3 E-7/a (about 1% of the releases for all accident sequences). Flooding events contribute about 4% to core damage. The relative contribution to large releases is significantly lower than the relative contribution to core damage because of the lower than average likelihoods for early containment failure and dry CCI, see tables 3.10-4 and 5-1 to 5-8.

3.10 General Observations

Defences on three levels protect nuclear power plants against the impact from flooding:

1. Providing equipment for flooding detection, procedures and hardware for isolation of flooding sources and drainage of water accumulation. The purpose of such provisions is to prevent flood induced water levels to reach critical heights in rooms containing safety relevant equipment.

2. Protecting vital cabling and safety equipment by suitable local arrangement, and by flood tight barriers. The purpose of separation is to prevent flooding damage to multiple safety system trains and to preserve the availability of the redundancies needed for coping with flood induced events.

3. Provision of recovery and accident management actions to mitigate the impact of floods by providing backup to safety equipment that has been disabled by flood events, and to preserve the retention capability of the containment.

The failures most frequently identified as causes for flood induced core damage are:

- Loss of SG feeding (PWRs)

- Loss of AC-power

- Flood induced failure of support systems (service water, instrument air) in conjunction with causal or random failures of one or several of the systems

  - MFW, AFW (PWRs)

  - HPCI, RCIC, LPCI, condensate system, depressurisation of RPV, DC-power (BWRs).

To improve the plant’s defences against floods, numerous modifications and backfits were implemented:

- Installation of a system and putting in place of procedures for the detection of leaks and identification of the leaking service water train inside the annulus (Biblis-B).

- Installation of additional water level measurements in the annulus (Biblis-B).

- Installation of additional flood barriers inside the annulus (Biblis-B).

- Installation of a system for automatic switching of service water pumps (Biblis-B).

- Installation of walls to separate the switchyard area from the area containing main and auxiliary feedwater equipment (Beznau).

- Installation of the NANO system, consisting of diverse, independent safety systems for emergency core cooling and emergency feeding of steam generators. Since most flooding scenarios do not affect the NANO systems, these systems can be used for plant shut down in the event of flood from the emergency control room in the NANO building (Beznau).

  Remark: Improving the defence against flooding was not the only reason for the installation of the NANO system.

- In conjunction with upgrading the battery capacity, the two battery banks will be separated, eliminating the potential for common cause failure due to water spray from the eyewash water supply pipe (Robinson).

- New procedures will be put in place for coping with the dominating flooding events in the auxiliary building level 226 area and in the relay room. The procedures will assist the operators in

  - identifying the location of the flood source

  - isolating the water supply before the flood reaches critical levels

  - opening of doors that would preclude the accumulation of water to critical depth. (Robinson).

- Improved drainage capacity in the emergency diesel generator buildings. The drainage is adequate to mitigate even a header break in the EECW system, which supplies the diesel unit coolers (Browns Ferry).

- Improved design of the CST header ring in the reactor building (Browns Ferry).

- Putting in place of enhanced alarm response instructions for flooding events (Perry).

In probabilistic terms, the effectiveness of defences at the three levels described above can be measured by:

- The flooding frequency for critical areas. In general, this figure is in the E-3/a to E-2/a range. Variation of the numerical values among different plants is due the variation of the size of critical areas and of the number and capacity of flood sources in critical areas. At some plants, this figure already includes flood identification and isolation measures and is therefore lower than at other plants.

- The conditional probability for a hazard state with potential for core damage developing from a flood. Inspection of table 3.10-1 shows a clear advantage of plants with systematic separation of safety system trains (for example Biblis-B), over plants without this feature.

- Several plants have only two trains for injection, residual heat removal and support functions. With the frequencies of flood induced damage to one of the trains, as developed in the examined studies, and random failures of the respective other trains,-with conditional probabilities in the range ~3 E-3 to ~1 E-2 - core damage contributions up to the order 10^{-5}/a can result for plants with such features.

- At Beznau, the addition of the independent NANO emergency system significantly reduced the high contribution from flood events found for the original design.

- The conditional probabilities of failure of recovery and accident management actions are in the range $\sim 10^{-2}$ to $\sim 10^{-1}$ and generally show good agreement among the examined studies for similar boundary conditions.

- In plants with high redundancy and systematic functional and physical separation of safety system trains, the contribution to core damage from flood events is not significant. On the contrary, plants with only 2 safety trains and support system trains tend to have significant contributions from flood events.

- In none of the examined studies do flood events lead to severe accident loading of the containment that would not also attend internally initiated accident sequences.

- In the PWR studies, concurrent flood and containment bypass events, (for example, steam generator tube rupture), are not considered for reasons of low frequency. Therefore, the relative contribution of flood induced events to early containment failure modes and to large releases is generally smaller than the relative contribution of flood induced events to core damage, see tables 5-1, 5-2, 5-6 and 5-8.

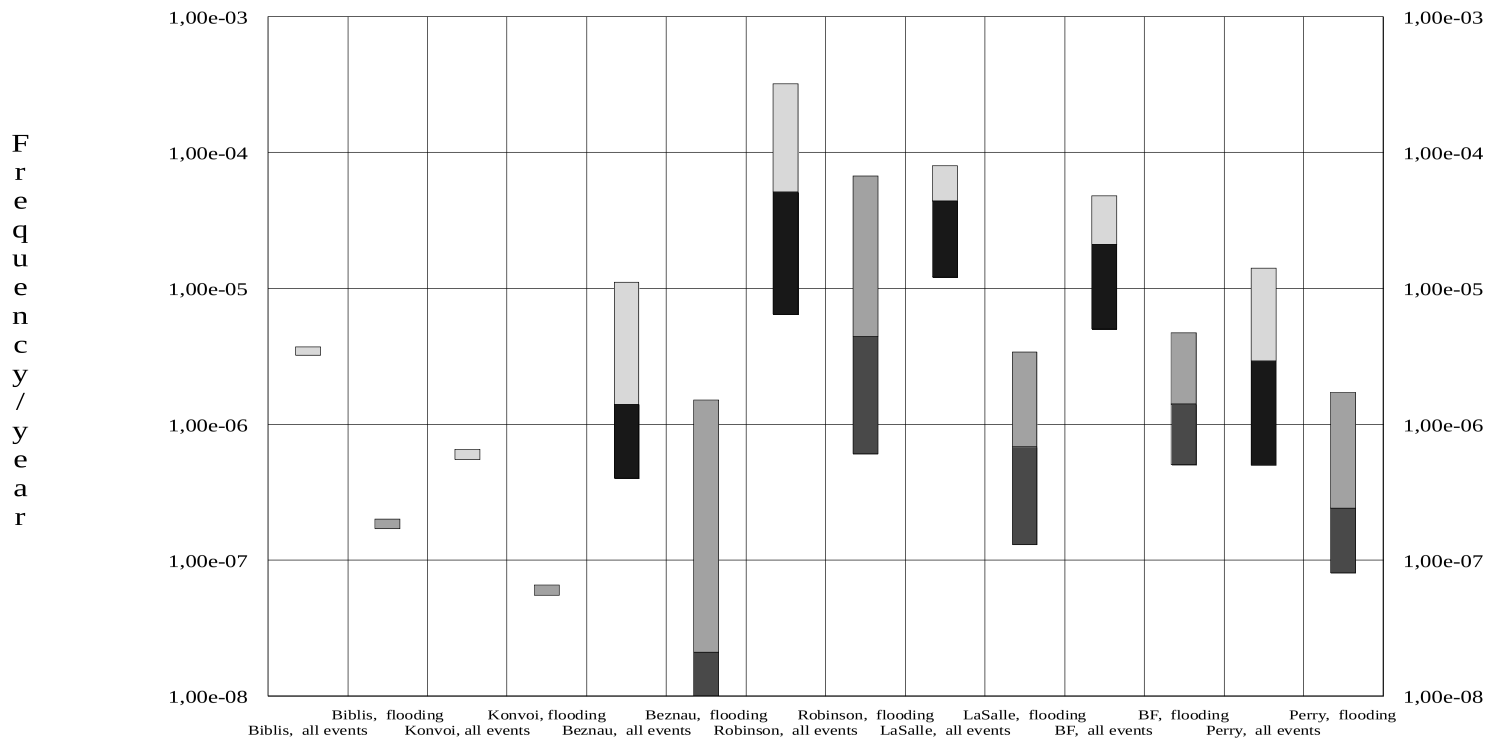

Figure 3.10-1. Frequencies of core damage, LRCF mode and exceedance of 10% Cs release for all events and flooding events. Top of light grey bars: CDF, all events, bottom of light grey bar: frequency of LRCF mode, all events, bottom of solid black bars: exceedance frequency of 10% Cs release, all events. Top of light blue bars: CDF, flooding events, bottom of light blue bar: frequency of LRCF mode, flooding events, bottom of dark blue bars: exceedance frequency of 10% Cs release, flooding events. Length of grey and light blue bars: capability of the containment to prevent LRCF mode, given CD. Length of black and dark blue bars: measure of the capability of containment to mitigate large releases, given LRCF mode. (For Biblis, level 2 results are not available.)

Table 3.10-1. Initiator Frequencies/a for Flooding Events, Conditional Probabilities for Safety Relevant Flooding, Conditional Probabilities of CD, CDF Contributions from Floods for Power Generation States, PWRs

	Surry-1	Robinson	Biblis-B	Beznau
Flooding frequency/a for:				
Auxiliary building		7.5 E-4		
Annulus			4 E-3	
Turbine building, switchgear room				6.8 E-5
Turbine building, pump yard				1.2 E-3
Total				
Flooding frequency/a for critical area(s) in:				
Auxiliary building		3.6 E-4		
Annulus			4 E-3	
Turbine building, switchgear room				6.8 E-5
Turbine building, pump yard				1.2 E-3
Total				
Conditional probability of a hazard state with potential for core damage (safety relevant flooding) developing from flooding in the:				
Auxiliary building		1		
Annulus Flooding between 70 cm and 90 cm Flooding above 90 cm			 2 E-4 1 E-3	
Turbine building, switchgear room				1
Turbine building, pump yard				5 E-2
Conditional probability of failure of recovery, given safety relevant flooding in the:				
Auxiliary building		0.17		
Annulus Flooding between 70 cm and 90 cm Flooding above 90 cm l			 < 0.1 5 E-2	
Turbine building, switchgear room				1 E-2
Turbine building, pump yard				1 E-2
Total conditional probability of core damage, given flooding in critical area(s) of the:				
Auxiliary building		0.17		
Annulus Flooding between 70 cm and 90 cm, Flooding above 90 cm			 <2.5 E-5 < 5 E-5	
Turbine building, switchgear room				1 E-2
Turbine building, pump yard				5.4 E-4
CDF contribution from flooding in the				
Auxiliary building		6.3 E-5		
Annulus Flooding between 70 cm and 90 cm, not further analysed Flooding above 90 cm	 sum		 << 1 E-7	
Instrument room	<< 1 E-6/			
Turbine building, switchgear room				6.7 E-7
Turbine building, pump yard				6.5 E-7
Total Percent of internal + external events	< 1 E-6/a	6.7 E-5 22%	<< 1 E-7 < 5.5%	E-6 12.6%

Table 3.10-2. Core Damage Contributions from Flooding, Conditional Probabilities for Important Containment Failure Modes, Conditional Probabilities Significant and Large Releases, Given Core Damage Induced by Flooding and by any Event, for Power Generation States at PWRs.

	Surry-1	Robinson IPE	Biblis-B	Beznau
CDF contribution from flooding in the:				
Auxiliary building		6.3 E-5		
Annulus			< 1 E-7	
Instrument room	all << 1 E-6/a			
Turbine building, switchgear room				6.7 E-7
Turbine building, pump yard				6.5 E-7
Total CDF flooding contribution Percent of internal + external events	< 1 E-6/a < 1.3 %	6.7 E-5 22 %	< 1 E-7 < 5.5 %	1.5 E-6 12.6 %
Conditional probabilities for containment failure modes and major contributions				
Early containment failure given a flooding induced core damage scenario given any core damage scenario		 0.007 0.012		 0.014 DCH 0.013 DCH
Late containment failure given a flooding induced core damage scenario given any core damage scenario		 0.27 0.1		 0.22 vent.fail 0.18 vent. fail
Containment bypass + Isolation failure given a flooding induced core damage scenario given any core damage scenario		 0.06 0.14		 - 0.11 ISF
Containment intact given a flooding induced core damage scenario given any core damage scenario		 0.8 0.75		incl. venting 0.76 0.7
Frequencies and conditional probabilities of releases				
Exceedance frequency/a for flood induced 1% Cs release percent of internal+external events Conditional probability for exceeding 1% Cs release, given a flooding induced core damage scenario given any core damage scenario		1.6 E-6 14% 0.024 0.044		2.8 E-7 14% 0.18 0.2
Exceedance frequency/a for flood induced 10Cs release percent of internal+external events Conditional probability for exceeding 10% Cs release, given a flooding induced core damage scenario given any core damage scenario		6 E-7 10% 0.01 0.02		9 E-9 2.5% 0.006 0.03

Table 3.10-3. Initiator Frequencies/a) for Flooding Events, Conditional Probabilities for Safety Relevant Flooding, Conditional Probabilities of Core Damage, Given Safety Relevant Flooding in Sensitive Areas, Core Damage Contributions from Floods for Power Generation States, BWRs

	Peach Bottom-2	LaSalle-2	Browns Ferry IPE	Perry IPE
Flooding frequency/a for:				
Control Complex				3.8 E-3
Turbine Building	all << 1 E-6		5 E-2	6 E-3
Reactor Building		3.6 E-5		
Other Buildings				
Total				
Flooding frequency/a for critical area(s) in:				
Control Complex				3.8 E-3
Turbine Building			5 E-2	6 E-3
Reactor Building		3.6 E-5		
Other Buildings				
Total				
Conditional probability of a hazard state with potential for core damage (safety relevant flooding) developing from flooding in critical area(s) of the:				
Control Complex				
Turbine Building				
Reactor Building		9.1 E-2		
Other Buildings				
Conditional probability of failure of recovery, given safety relevant flooding in the				
Control Complex				
Turbine Building				
Reactor Building		1		
Other Buildings				
Total conditional probability of core damage, given flooding in the				
Control Complex				3.2 E-4
Turbine Building			9.4 E-5 including additional random failures	4.7 E-5
Reactor Building		9.1 E-2		
Other Buildings				
CDF contribution from flooding in the				
Control Complex				1.2 E-6
Turbine Building	all << 1 E-6		4.7 E-6	2.8 E-7
Reactor Building		3.4 E-6		
Other Buildings				
Total flooding Percent of internal + external events	< 1 E-6 < 3.5%	3.4 E-6 4.2%	4.7 E-6 10%	1.5 E-6 12%

Table 3.10-4. Core Damage Contributions from Floods, Conditional Probabilities for Important Containment Failure Modes, Conditional Probabilities for Significant and Large Releases, Given Core Damage Induced by Floods, and by any Event, for Power Generation, BWRs

	Peach Bottom	LaSalle	Browns Ferry IPE	Perry IPE
CDF contribution from flooding in the				
Control Complex				1.2 E-6
Turbine Building			4.7 E-6	2.8 E-7
Reactor Building		3.4 E-6		
Other Buildings				
Total CDF flooding contributions Percent of internal + external events		3.4 E-6 4.2%	4.7 E-6 10%	1.5 E-6 12%
Conditional probabilities for containment failure modes and *major contributions*				
Early containment failure - given a flooding induced core damage scenario - given any core damage scenario		 0.2 0.55	 0.3, Ovp 0.46. liner failure	 0.16 0.23, H_2 burn
Late containment failure - given a flooding induced core damage scenario - given any core damage scenario		 0.52 0.2		 n.a. 0.07
Containment intact - given a flooding induced core damage scenario - given any core damage scenario		 0.28 0.25	 0.7 0.28	incl. venting 0.78 0.7
Frequencies and conditional probabilities for releases				
Exceedance frequency for flooding induced 1% Cs release percent of internal+external events Conditional probability for exceeding 1% Cs release - given a flooding induced core damage scenario - given any core damage scenario		7.2 E-7 2.4% 0.21 0.36	 n.a. n.a. 0.25	 3.5 E-7 0.23 *0.33*
Exceedance frequency for flooding induced 10% Cs release percent of internal+external events Conditional probability for exceeding 10% Cs release - given a flooding induced core damage scenario - given any core damage scenario		1.3 E-7 1% 0.04 0.15	 n.a. < 0.1 0.1	 8 E-8 0.05 0.04

4. PROBABILISTIC ASPECTS OF SEISMICALLY INDUCED FAILURES

4.1 Introduction

Nuclear power plants are designed to withstand the loads of a "safe shutdown earthquake" (SSE). The SSE is defined either directly in terms of peak ground acceleration, usually in the range of 0.1g to 0.2g, or indirectly in units of the earthquake occurrence frequency, typically 10^{-5}/a to 10^{-4}/a. In most countries, this frequency range corresponds to the peak ground acceleration range 0.1g to 0.2g. In some countries, there is a deterministic requirement that the containment survives earthquakes exceeding the SSE up to a peak ground acceleration range of~0.4g (which is associated with a occurrence frequency of~10^{-7}/a).

In view of the large uncertainties involved, the design against earthquake loads is largely based on conservative assumptions.

The steps of a seismic events PSA involves the probabilistic quantification of:

- earthquake occurrence frequencies/magnitudes,
- ground motion at the site,
- response of buildings and safety systems,
- frequencies of core damage,
- modes and frequencies of containment failure.

4.2 Frequencies of Occurrence of Seismic Events

In all examined PSAs hazard curves are provided that express frequencies of seismic events as function of the earthquake size - usually measured by the peak ground acceleration - of the events. Figure 4.2-1 presents frequencies versus ranges for peak ground acceleration. In this presentation, extrapolation to values beyond those shown in the hazard curves was necessary for some of the low frequency/high acceleration values.

The highest ground acceleration values are observed for the LLNL hazard curves for the Peach Bottom and Surry plants and for the LaSalle plants, at which the determination of the seismic hazard is based on LLNL curves, with some plant specific corrections.

Using the EPRI hazard curves for Peach Bottom and Surry, the peak ground acceleration values are found to be a factor 2 to 3 lower for identical frequency levels.

For the three European plants included in this evaluation, the provided peak ground accelerations are much lower. All three plants are located in an area with similar moderate seismic activity. Even for the lowest considered frequency level (10^{-7}), the peak ground acceleration remains below 0.6 g.

4.3 Free Field Ground Motion

Besides the hazard curves characteristic for the site, the seismic loads depend on the soil characteristics at the site, as they can significantly influence the acceleration response spectra. For example, Surry has lower peak ground acceleration than Peach Bottom, but experiences higher ground motion input to its structures because it is sited on loose soil, as opposed to sound rock at the Peach Bottom site. Loose and cohesionless soil tends to amplify the ground motion in the frequency range where building response spectra typically have their maximum.

4.4 Building and Component Fragilities

The median fragilities of critical components are generally in the two to three times SSE level.

Among the most vulnerable components are:

- ceramic insulators (explicitly included for the US plants, and implicitly included in “offsite power” components in the Swiss studies),
- large vertical tanks with flat bottom, for example, condensate storage tanks and refuelling water storage tanks,
- component cooling water heat exchangers and motive control centers.

Many of the seismic induced component failures result from loss of anchoring. At most plants, numerous modifications have been implemented which are directed at extending the strength of anchoring of components to higher acceleration ranges. However, a probabilistic assessment of the effect of such modifications is not available.

Among the most vulnerable structures and buildings are:

- service building,
- safeguards building,
- auxiliary building.

4.5 Core Damage Scenarios

Figure 4.5-1 depicts annual frequencies for the acceleration ranges considered at the examined PWR and BWR plants, the conditional probabilities for core damage, given earthquakes in the respective acceleration ranges, and the seismic CDF contributions for the acceleration ranges.

When comparing the results in these figures, it is important to note that the acceleration ranges are different for the various plants.

Acceleration ranges below or including the SSE level are marked by shading, whereas acceleration ranges beyond the SSE level are marked by solid black.

An obvious observation is that, with one exception, the maximum core damage contribution is reached for acceleration ranges twice or three times the SSE level. The reason is that for acceleration values above 0.3 g, extensive damage is to be expected to vital components and to buildings/structures, leading to significant degradation of the safety function's availability. The decrease of core damage contributions with higher acceleration ranges results from the strong decrease of the occurrence frequencies for these acceleration ranges, in conjunction with the fact that the safety function unavailabilities asymptotically approach values close to 1 for the higher acceleration ranges.

The only exception to this observation are the results for the LaSalle plant, which show a monotone decrease of the core damage contribution with increasing acceleration. In this analysis it is assumed that station power is lost with certainty already for the lowest acceleration range. The occurrence of random failures of equipment needed for coping with the accident then leads to core damage. The only other seismically induced failure is that of the condensate storage tank. Gross failure of the tank fails the HPCS pumps due to loss of suction. Besides this, there is no seismically induced degradation of safety functions: the safety functions unavailability increases only by the factor ~ 3.5 from the lowest to the highest acceleration range. This singular result of the LaSalle analysis appears to be an artefact of the analysis, more than a real deviation from the pattern of result obtained for the other plants/PSAs.

The contribution to core damage frequency by seismic events varies from significant to dominant among the examined plants, with the exception of LaSalle, at which it is insignificant.

Figure 4.5-2 shows mean safety function unavailabilities for dominant seismically induced accident sequences. For comparison, the safety function unavailabilities for the same type of accident sequences, caused by internal events, are also shown (marked by light shading). As in Figure 4.5-1, the mean safety function unavailabilities for seismic events with acceleration below the SSE level are marked by shading, and the unavailabilities for seismic events above the SSE by solid black.

For Surry and Peach Bottom, Figure 4.5-2 shows a moderate increase of the unavailabilities from that for internal events to those for low seismic acceleration, and a drastic increase for the higher acceleration values for which significant damage to safety relevant equipment and buildings is to be expected. For acceleration values above 3 SSE, the unavailability is practically 1.

For LaSalle, the biggest increase is from the unavailability for internally caused LOSP events to those caused by the lowest acceleration seismic events, and a relatively moderate increase for the higher acceleration values. Most remarkable, the unavailability does not exceed E-2. As for the reason for this singular pattern, see the above discussion.

For Biblis-B, the biggest increase of the safety function unavailability is also from the internal event to the lowest acceleration seismic events. The reason is the following: the assessment of the safety function unavailability for internally caused loss of feedwater events includes credit for cross-connecting of the emergency feedwater system to the emergency feedwater system of the neighbouring unit, as well as credit for secondary side bleed/feed as backup to failed emergency feed and cross-connecting. Primary side bleed/feed is also credited. For seismic events, no credit is given for secondary side accident management actions, and only little credit is given to the actuation of primary side bleed/feed. The analysis includes only seismic events up to acceleration 0.46 g, for which the safety function unavailability is ~ E-2. Higher acceleration values for which the safety function unavailability would approach 1 were not considered for reasons of low frequency.

In Figure 4.5-3 those acceleration ranges are extracted that lead to the dominant seismically induced core damage frequencies. As can be seen, the safety function unavailabilities are almost identical and high (about 0.3) for the US plants Surry and Peach Bottom. For the European plants they are much lower (1.2 E-3 to 4 E-2). The LaSalle PSA is not included in this comparison, because the typical degradation of safety functions due to seismic events is not seen in this analysis.

4.6 Containment Failure Modes and Releases

In Figure 4.5-4, the following information is provided for the PWRs, respectively, BWRs included in this compilation:

- total core damage frequency contributions from earthquakes and their share of overall core damage frequency,
- frequencies and conditional probabilities, given core damage, of large release containment failure (LRCF) modes,
- frequencies and conditional probabilities, given core damage, of exceeding 1% and 10 % caesium releases.

In this figure, the top of the dark grey shaded portion of a bar represents the core damage frequency due to seismic events, the bottom of the dark grey shaded portion represents the frequency of LRCF modes for seismic events. Thus, the length of the dark grey shaded portion represents the conditional probability of LRCF modes, i.e. it measures the resilience of the containment to dynamic loads early in the accident, and its capability to prevent open release paths. The bottom of the light grey shaded portion represents the exceedance frequency of 10% caesium release for seismic events. Thus, the length of the dark and light grey shaded portions together represents the conditional probability of LRCF modes, given core damage. The length of the light grey shaded portion alone represents the conditional probability of exceeding 10% caesium release, given LRCF modes.

For comparison, analogous information is provided on internal events, with solid black bars instead of dark grey shaded bars, and transparent bars instead of light grey shaded bars.

Inspection of Figure 4.5-4 shows the following:

4.6.1 Pressurised Water Reactors

At Surry the conditional probability of LRCF modes, given core damage, is similar for seismic and internal events. The dominant contribution for seismic events is early containment failure caused by failure of the supports of the steam generator and reactor coolant pumps . For internal events the dominant contribution is steam generator tube rupture with failure to isolate the faulted steam generator.

At Beznau, there is a high conditional probability of seismic induced failure of containment isolation.

At Surry and Beznau, the capability of the containment to mitigate large releases from the damaged containment is similar for seismic events, but relatively limited (conditional probability of exceeding 10% caesium release, given early containment failure or containment bypass is above 0.5). In the original HSK/ERI analysis for Beznau, the retention capability of the containment for internal events was also similar to Surry. However, procedures and hardware have since been put in place (at Beznau and some other plants) which permit to fill up a faulted steam generator with water in order to mitigate releases by scrubbing of fission product in the water column. Available calculations indicate a significant reduction of releases by this strategy, leaving seismic induced releases as the dominant contribution to large releases.

4.6.2 Boiling Water Reactors

For BWRs, the capability of the containment to absorb the loads attending vessel breach is generally smaller than for the PWRs. The only seeming exception - the low conditional probability of early containment failure shown for seismic events at Mühleberg - does not contradict this statement, because it only reflects the fact that there are no seismic induced accident sequences at Mühleberg, that could cause early containment failure.

The mitigation capability of the BWR containment, given early containment failure, is similar for seismic and internal events, but limited (above 0.5) with the exception of Mühleberg. At this plant, failure of the containment by CCI can be prevented by injection of water to the lower part of the containment. Furthermore, there is a hardened secondary containment and an outer torus around the secondary containment for fission product scrubbing.

At LaSalle, the mitigation capability of the containment for seismic events is particularly poor, because for the dominant LOSP accident sequence, there is always dry corium/concrete interaction due to the inability of water injection to the drywell (because power can not be recovered in due time). Thus, early containment failure with suppression pool bypass is almost certain.

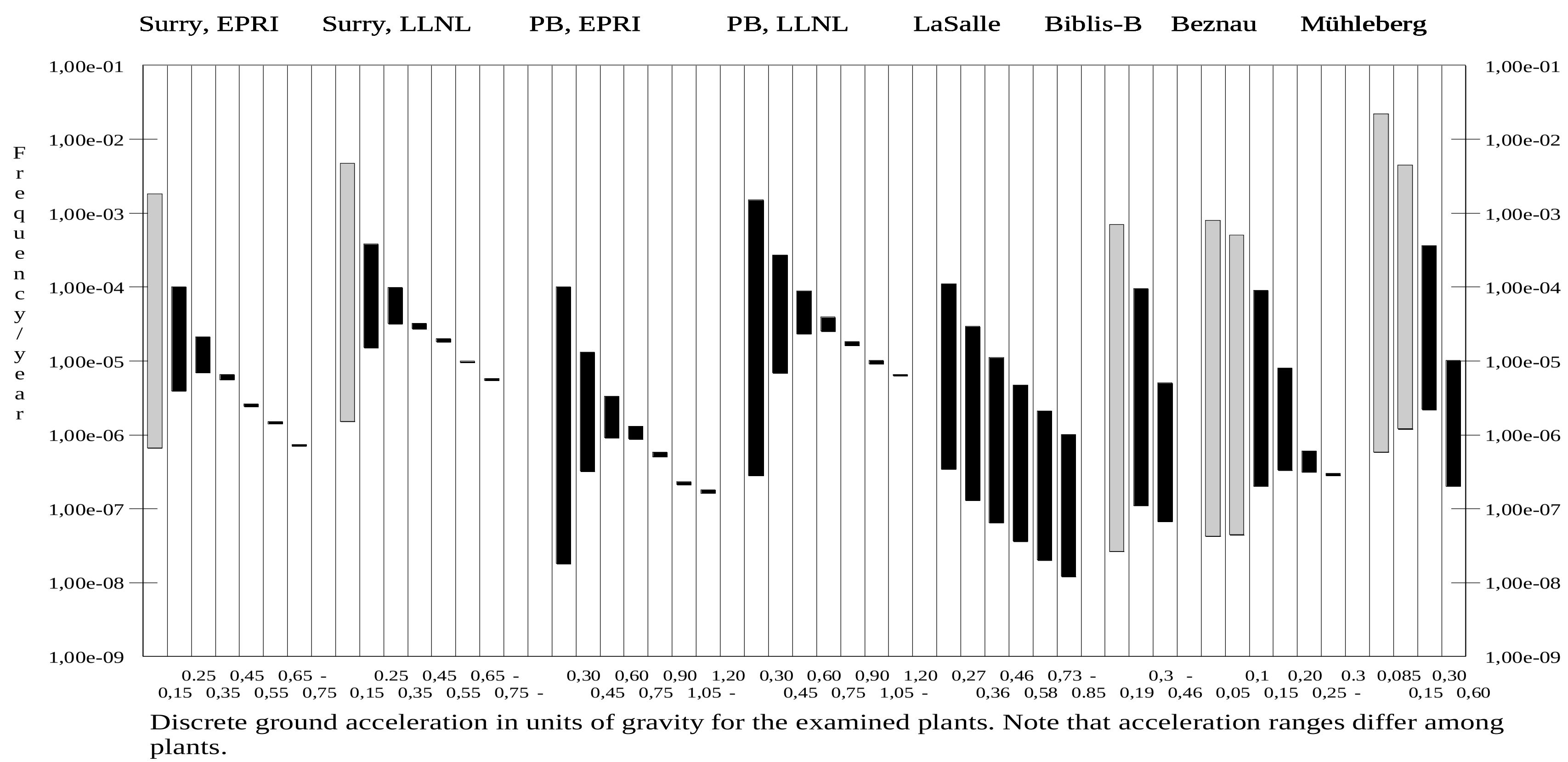

Figure 4.5-1. Average occurrence frequencies for discrete ground acceleration ranges (top of bar), core damage frequencies, given ground acceleration in the discrete range (bottom of bar). Acceleration ranges below SSE level are identified by grey shading., above SSE level by solid black

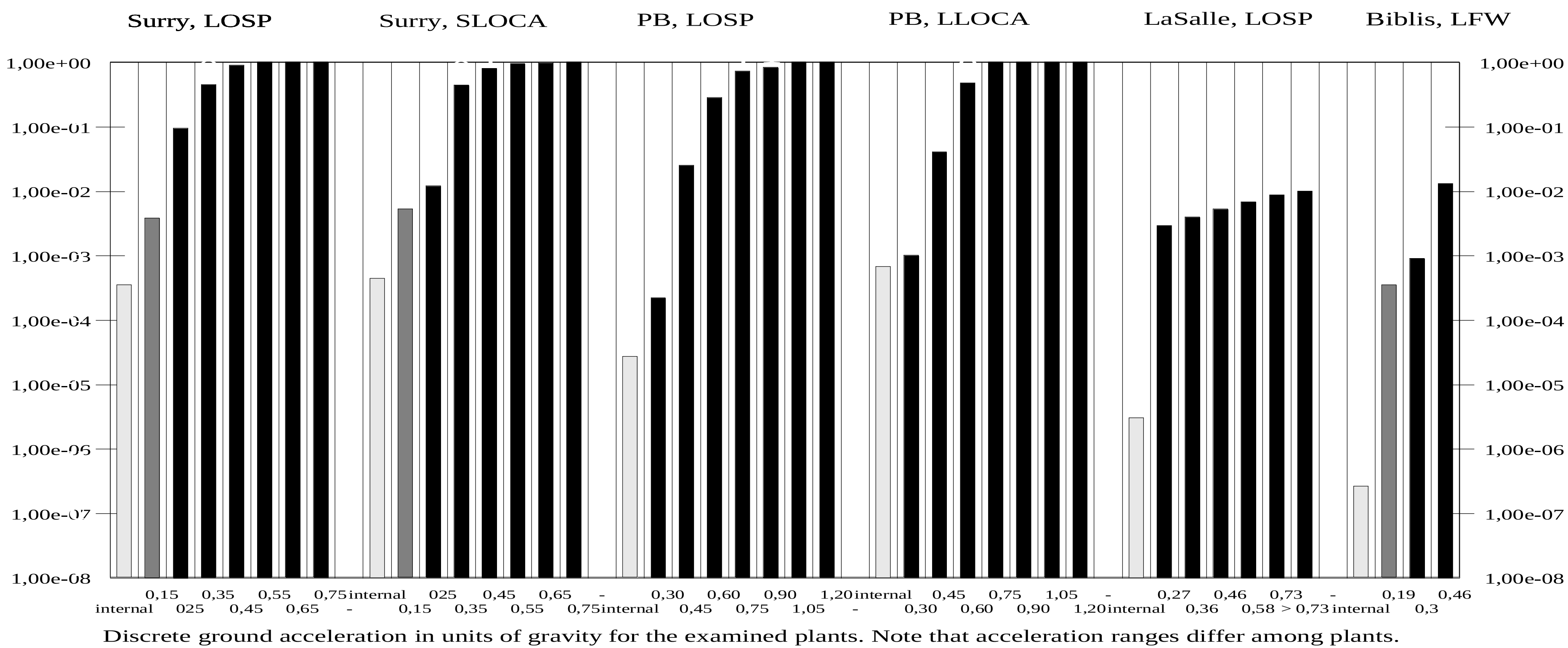

Figure 4.5-2. Mean unavailabilities of safety functions (vertical axis): for important seismically induced initiators and similar internal initiators. Acceleration ranges below SSE level are identified by dark grey shading, above SSE level by solid black. The unavailabilities for similar internal events are identified by light grey shading.

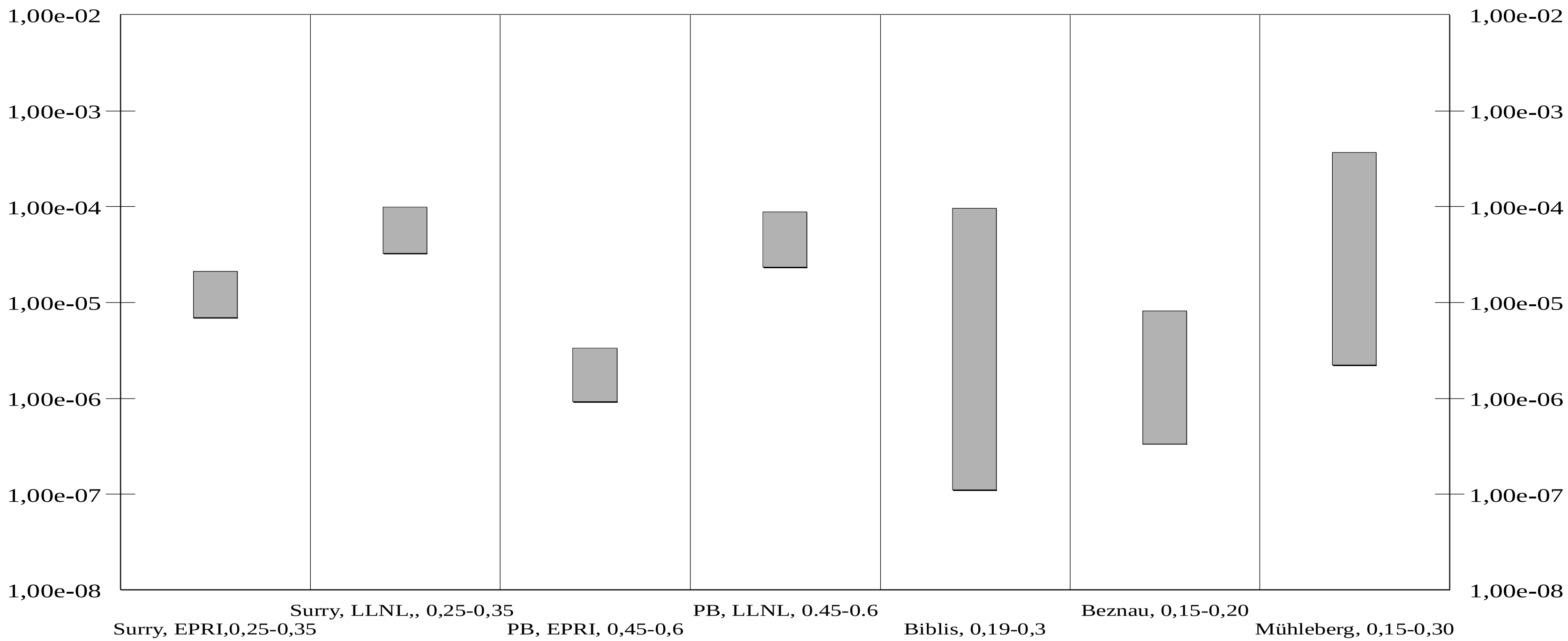

Figure 4.5-3. Dominant seismically induced core damage sequences and corresponding acceleration ranges. Note that the acceleration ranges differ among the plants. The top of bar marks the occurrence frequencies of the acceleration range responsible for the dominant sequence, the bottom marks the associated core damage frequency.

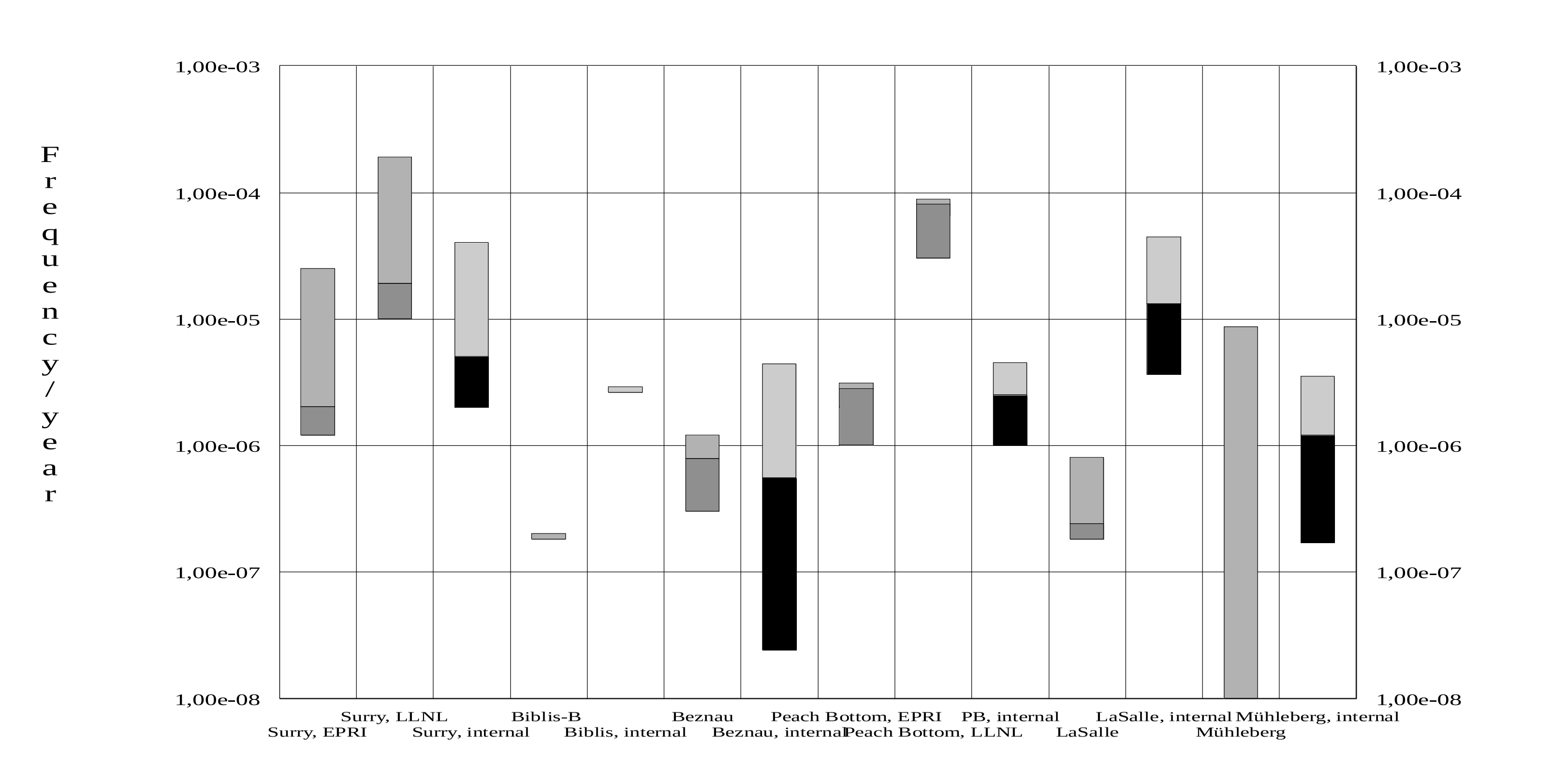

Figure 4.5-4. Frequencies of core damage, LRCF mode and exceedance of 10% Cs release for all events and seismic events. Top of light grey bar: CDF, all events, bottom of light grey bar: frequency of LRCF mode, all events, bottom of solid black bar: exceedance frequency of 10% Cs release, all events. Top of light green bar: CDF, seismic events, bottom of light green bar: frequency of LRCF mode, seismic events, bottom of dark green bar: exceedance frequency of 10% Cs release, seismic events. Length of grey and light green bar: capability of the containment to prevent LRCF mode, given CD. Length of black and dark green bars: measure of the capability of containment to mitigate large releases, given LRCF mode.
(For Biblis, level 2 results are not available.)

5. INTEGRATED EVALUATION

5.1 Level-1 Aspects

The important level 1 results for the examined plants are compiled in table 5-1. The level 1 findings described in the preceding sections can be summarised as follows:

- For all PWR plants included in the assessment, the internal events are the dominant contributors to the total core damage frequency.

- For BWR plants, dominant contributions are observed from internal, fire and seismic events

- Plants with only two trains of vital safety and/or support systems are vulnerable to internal fire and flooding events. The contribution of such events to core damage is significant. Such contributions result from situations with fire or flooding induced damage to one vital train (frequency range 10^{-3}/a to 10^{-2}/a), combined with random failure of the other respective train (conditional probability range 10^{-4} to 10^{-2}). At several plants, this leads to core damage contribution exceeding 10^{5}/a.

  Given such a situation, substantial reduction of core damage frequency can only be achieved by:

  - enhanced fire detection/fire suppression and /or flood detection/flood mitigation capabilities

  - addition of diverse back-up to the respective trains.

- On the other hand, plants with high redundancy and strict spatial and functional separation or diverse means for backing up vital system functions show low core damage contributions from fire and flooding events. Therefore, modern plants with highly redundant and/or diverse systems provide very effective defences against the impact of fire and flooding events. As for defending against fire and flooding events, such plants show advantages over other designs which are not apparent from the traditional internal events analyses.

- The contribution to core damage frequency from seismic events typically varies from significant to dominant among the examined plants. The maximum core damage contribution is typically reached for acceleration ranges two to three times the SSE level. The reason for this is that for acceleration values above 0.3 g, extensive damage to vital components and to buildings/structures housing such components is to be expected, leading to significant degradation of the safety function’s availability. The decrease of core damage contributions with higher acceleration ranges results from the strong decrease of the occurrence frequencies for these acceleration ranges, in conjunction with the fact that the safety function unavailabilities asymptotically approach values close to 1 for the higher acceleration ranges.

5.2 Level-2 Aspects

The important level 2 results are compiled in tables 5-2 to 5-8 and are illustrated by figures 5-1 and 5-2. The level 2 findings described in the preceding sections can be summarised as follows:

- In general, the challenges of the containments of PWR and BWR plants resulting from fire and flood induced core damage sequences do not significantly differ from the challenges due to other internal events

- For PWRs, the relative contribution to large releases from fire and flooding events is generally lower than the relative contribution to core damage from fire and flooding events. This is due to the fact that containment bypass sequences, which drive the releases in the traditional internal events analyses, are insignificant for fire and flooding events. The exception to this observation is the Beznau result. There, the contribution to large releases from internal events was drastically reduced by the implementation of hardware and procedures for filling up a damaged unisolated steam generator. Without such provisions, SGTR events with unisolated steam generator are typically responsible for the dominant contribution to large releases caused by internal events.

- For BWRs, the relative contribution to large releases from fire and flooding events differs little from the relative contribution to core damage from fire and flooding events. The only exception is the result for LaSalle. At this plant the contribution from fire events to large releases is disproportional high, due to the venting strategy at this plant and the very high conditional probability for dry core concrete interaction. On the other hand, the contribution from flooding events is disproportionally low, due to high conditional probability for recovery of injection. As LaSalle is the only plant with a Mark II containment included in this evaluation it is not possible to say whether or not this result is typical for plants with Mark II containment design.

- For PWR plants, the capability of a damaged or bypassed containment to mitigate large releases is, in general, similar for internal events and for seismic events, but relatively limited (conditional probability of exceeding 10% caesium release, given early containment failure v containment bypass is above 0.5). However, a different situation exists at plants where procedures and hardware have been put in place which permit to fill up a faulted steam generator with water in order to mitigate releases by scrubbing of fission product in the water column (for example, at Beznau, Borssele and the Ringhals PWRs)). Available calculations indicate a significant reduction of releases by this strategy, leaving seismic induced releases as the dominant contribution to large releases.

– The mitigation capability of the BWR containments, given early containment failure, is similar for seismic and internal events, but limited (conditional probability of large release above 0.5), with the exception of Mühleberg. At this plant, most earthquake induced damage states contribute to a release bin characterised by

  – delayed time of containment failure

  – filtered containment venting

  – suppression pool scrubbing effective to time of venting

  – mitigation of releases by secondary building and outer torus.

  – Therefore, most fission products can be retained inside the containment.

5.3 Conclusions

By the examination of a sizeable number of PSAs, typical vulnerabilities of the most common reactor system/containment designs to internal and external hazards could be identified and a ranking of the relative importances of these events - in relation to internal events - could be performed. The results provide background information to the report "Level 2 Methodology and Severe Accident Management", Task 94-2, of the CSNI/Principal Working Group No. 5. In particular, the results provide guidance on the issue of completeness of PSAs, in that they indicate for which plant designs particular hazards should be analysed..

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Table 5-1. CDF Contributions from Internal and External Events, Power Generation States

Event	Surry-1	Biblis-B	Konvoi	Beznau	Robin-son IPE	Sizewell	Borssele PSA 97	Ring-hals 2	Peach Bottom	LaSalle	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse-bäck 1	Fors-mark 3	Oskars-ham 1
Total of internal events	4.0 E 5 52%	2.9 E-6 80.5%	5 E-7 90%	4.4 E-6 40%	2.5 E-4 78%	2.2E-5 96%	1.7 E-6	1.5 E-5 75%	4.5 E-6 16%	4.4 E-5 55%	4.3 E-5 89.5%	1.2 E-5 85.7%	3.5 E-6 27%	3.9 E-6	7.2 E-6	2.3 E-5 65%
Fire	1.1E-5 14.5%	1.7 E-7 4.7%		2.1 E-6 18%		4.7E-7 2%		4 E-6 20%	2.0 E-5 71.5%	3.2 E-5 40%			7.2 E-7 5.5%			4.2 E-6 12%
Internal flooding	-	< 2.0 E-7 <5.5%	6 E-8 (without AM) 10%	1.5 E-6 12.6%	6.7 E-5 22%		3.6 E-7	1 E-6 5%	-	3.4 E-6 4.2%	4.7 E-6 10.5%	1.7 E-6 14.3%	2.0 E-7 1.5%			8 E-6 23%
External flooding	-	-		2.1 E-6 18%			3.4 E-8	?	-				2.1 E-7 1.6%			
Earthquake	2.5E-5 (EPRI hazard curve) 33.5%) (1.9 E-4)	<2.0 E-7 <5.5%		1.2 E-6 10.9%		E-7 2%	vapour cloud explo-sion 1.2 E-7	?	3.1 E-6 (8.8 E-5) (EPRI hazard curve) 12.5%	8.0 E-7 0.8%			8.6 E-6 66% (HSK/ ERI)			
Aircraft impact	-	<1.0 E-7 <1%		1.6 E-7 1.5%					-				1.3 E-7 1%			
Total of external events	**3.6 E-5** **48%**	**<7 E-7** **19.5%**	**6 E-8** **10%**	**6.6 E-6** **60%**	**6.7 E-5** **22%**	**9.3 E-7** **4%**	**6 E-7**	**5 E-6** **25%**	**2.3 E-5** **84%**	**3.6 E-5** **45%**	**4.7 E-6** **10.5%**	**1.7 E-6** **14.3%**	**9.9 E-6** **73%**			**1.3 E-5** **35%**
Total, internal + external events	**7.6 E-5 (2.4 E-4)**	**3.5 E-6**	**5.6 E-7**	**1.1 E-5**	**3.2 E-4**	**2.3 E-5**	**2.3 E-6**	**2 E-5**	**2.8 E-5 (1.1 E-4)**	**8.0 E-5**	**4.8 E-5**	**1.4 E-5**	**1.3 E-5**	**3.9 E-6**	**7.2E-6**	**3.5 E-5**

Table 5-2. ECF Contributions from Internal and External Events, Power Generation States

Event	Surry	Biblis-B	Beznau	Robinson IPE	Sizewell	Borssele	Ring-hals 2	Peach Bottom	LaSalle (*)	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse-bäck 1	Fors-mark 3	Oskars-ham 1
Internal events	2.8 E-7 9 %		7 E-8 46%	3.4 E-6 87%	E-8 97%	1.7 E-8	< 2 E-7	2.5 E-6 13.6%	1.3 E-5 30%	2 E-5 94%	2.9 E-6 91%	9.1 E-7 98%	3.9 E-7	2.1 E-8	
Fire	9 E-8 3.9 %		2.5 E-8 16%	n.a.	7 E-10 2%			1.3 E-5 71 %	3 E-5 68%						
Internal flooding	-		2.1 E-8 14%	5 E-7 13%				-	6.8 E-7 1.5%	1.4 E-6 6%	2.4 E-7 9%				
External flooding	-		2.8 E-8 19%					-							
Earthquake	2.8 E-6 (EPRI hazard curve) 87.3 %		8 E-9 5%		4 E-10 1%			2.8 E-6 (EPRI hazard curve) 15.3 %	1.8 E-7 0.4%						
Aircraft impact	-							-				2.2 E-8 2%			
Total of external events	**2.1 E-6 91.2 %**		**8 E-8 54%**	**5 E-7 13%**	**E-9 3%**			**1.6 E-5 86%**	**3.1 E-5 70%**	**1.4 E-6 6%**	**2.4 E-7 9%**	**2.2 E-8 2%**			
Total, internal + external events	**3.2 E-6**		**1.5 E-7**	**3.9 E-6**	**3.3 E-8**	**1.7 E-8**	**< 2 E-7**	**1.8 E-5**	**4.4 E-5**	**2.1 E-5**	**2.7 E-6**	**9.3 E-7**	**3.9 E-7**	**2.1 E-8**	

Table 5-3. LCF Contributions from Internal and External Events, Power Generation States

Event	Surry	Biblis-B	Beznau	Robinson IPE	Sizewell	Borssele	Ring hals2	Peach Bottom	LaSalle (*)	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse bäck 1	Fors-mark 3	Oskars-ham 1
Internal events	2.4 E-6 19%		9 E-7 43%	1.4 E-5 44%	4.2 E-6 98%	8.5 E-8	2 E-6	2.2 E-7 5 %	1.25 E-5 78%	1.2 E-5 >99%	8.4 E-7 99%	1.9 E-7 21%	< 3 E-8	9.1 E-10	
Fire	3 E-6 24%		4.2 E-7 20%	n.a.	E-8			2.4 E-6 54%	1.3 E-6 8%						
Internal flooding	-		3.3 E-7 15%	1.8 E-5 56%					1.8 E-6 11%	< 1 E-7 < 1%	< 1 E-8				
External flooding	-		4.6 E-7 22%												
Earthquake	7.2 E-6 (EPRI hazard curve) 57.6%		-		5.7 E-8			1.8 E-6 (EPRI hazard curve) 41 %	4.2 E-7 <1%			6 E-7 67%			
Aircraft impact	-											1 E-7 12%			
Total of external events	**1.02 E-5 81.6%**		**1.2 E-6 57%**	**1.8 E-5 56%**	**1.3 E-7 2%**			**4.2 E-6 95 %**	**3.5 E-6 22%**	**< 1 E-7 < 1%**	**< 1 E-8**	**7 E-7 79%**			
Total, internal + external events	**1.26 E-5**		**2.1 E-6**	**3.2 E-5**	**4.3 E-6**	**8.5 E-8**	**2 E-6**	**4.4 E-6**	**1.6 E-5**	**1.2 E-5**	**8.4 E-7**	**8.9 E-7**	**< 3 E-8**	**9.1 E-10**	

Table 5-4. NoCF, including successful venting. Contributions from Internal and External Events, Power Generation States

Event	Surry	Biblis-B	Beznau	Robinson IPE	Sizewell	Borssele	Ring-hals 2	Peach Bottom	LaSalle	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse-bäck 1	Fors-mark 3	Oskars-ham 1
Internal events	3.25 E-5 57.8%		3.5 E-6 43%	1.9 E-4 79%	1.5 E-5 94%	1.6 E-6	1.6 E-5	1.2E-6 26 %	1.8 E-5	9.7 E-6 75%	7.2 E-6 85%	1 E-6 10%	3.3 E-6	7.2 E-6	
Fire	7.9 E-6 13.5%		1.6 E-6 20%		4.6 E-7			4.8 E-6 72 %	1.1 E-6						
Internal flooding			1.15 E-6 14.2%	5.3 E-5 21%					1.0 E-6	3.3 E-6 25%	1.2 E-6 15%				
External flooding			1.6 E-6 20%												
Earthquake	1.6 E-5 (EPRI hazard curve) 28.4%		3.8 E-7 5%		1.6 E-7			1.2 E-7 (EPRI hazard curve) 2 %				9 E-6 90%			
Aircraft impact															
Total of external events	**2.37 E-5 42%**		**4.6 E-6 57%**	**5.3 E-5 218%**	**E-7 6%**			**4.9 E-6 74 %**	**2.1 E-6**	**3.3 E-6 25%**	**1.2 E-6 15%**	**9 E-6 90%**			
Total, internal + external events	**5.6 E-5**		**8.1 E-6**	**2.4 E-4**	**1.6 E-5**	**1.6 E-6**	**1.6 E-5**	**6.1 E-6**	**2 E-5**	**1.3 E-5**	**8.4 E-6**	**1 E-5**	**3.3 E-6**	**7.2 E-6**	

Table 5-5 Containment Bypass + Isolation Failure. Contributions from Internal and External Events, Power Generation States

Event	Surry	Biblis-B	Beznau	Robinson IPE	Sizewell	Borssele	Ring hals 2	Peach Bottom	LaSalle	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse bäck 1	Fors-mark 3	Oskars-ham 1
Internal events	4,8 E-6 99.5%		4.8 E-7 38%	4.3 E-5 92%	2 E-6 95%	1 E-7	1.8 E-6	2.5 E-6 13.6%	1.3 E-5 30%	2 E-5 94%	2.9 E-6 91%	2.7 E-7 100%	2 E-7	2.6 E-8	
Fire	-		-		3.8 E-8			1.3 E-5 71 %	3 E-5 68%						
Internal flooding	-		-	3.9 E-6 8%				-	6.8 E-7 1.5%	1.4 E-6 6%	2.4 E-7 9%				
External flooding	-		-					-							
Earthquake	2.8 E-8 (EPRI hazard curve) 0.5%		7.7 E-7 62% (mostly isolation failure)		3.5 E-8			2.8 E-6 (EPRI hazard curve) 15.3 %	1.8 E-7 0.4%						
Aircraft impact	-		-					-							
Total of external events	**2.8 E-8 0.5%**		**7.7 E-7 62%**	**3.9 E-6 8%**	**E-8 5%**			**1.6 E-5 86%**	**3.1 E-5 70%**	**1.4 E-6 6%**	**2.4 E-7 9%**				
Total, internal + external events	**4.8 E-6**		**1.25 E-6**	**4.7 E-5**	**2.1 E-6**	**1 E-7**	**1.8 E-6**	**1.8 E-5**	**4.4 E-5**	**2.1 E-5**	**2.7 E-6**	**2.7 E-7**	**6 E-7**	**2.6 E-8**	

Table 5-6. ECF + Isolation Failure + Bypass (LRCF). Contributions from Internal and External Events, Power Generation States

Event	Surry	Biblis-B	Beznau	Robinson IPE	Sizewell	Borssele	Ring hals 2	Peach Bottom	LaSalle	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse bäck 1	Fors-mark 3	Oskars-ham 1
Internal events	5.1 E-6 64 %		5.5 E-7 40%	4.6 E-5 90%	2 E-6 95%	1.2 E-7	2 E-6	2.5 E-6 13.6%	1.3 E-5 30%	2 E-5 94%	2.9 E-6 91%	1.2 E-6 >98%	6 E-7	2.6 E-8	
Fire	9 E-8 <1%		2.5 E-8 1.8%	-	3.8 E-8	-		1.3 E-5 71 %	3 E-5 68%						
Internal flooding	-		2.1 E-8 1.5%	4.4 E-6 10%		-		-	6.8 E-7 1.5%	1.4 E-6 6%	2.4 E-7 9%				
External flooding	-		2.8 E-8 2%			-		-							
Earthquake	2.8 E-6 (EPRI hazard curve) 35 %		7.8 E-7 56%		3.5 E-8			2.8 E-6 (EPRI hazard curve) 15.3 %	1.8 E-7 0.4%						
Aircraft impact	-			-								2.2 E-8 < 2%			
Total of external events	**E-6 36 %**		**E-7 61%**	**4.4 E-6 10%**	**E-8 5%**			**E-5 86 %**	**3.1 E-5 70%**	**1.4 E-6 6%**	**2.4 E-7 9%**	**2.2 E-8 < 2%**			
Total, internal + external events	**8 E-6**		**1.4 E-6**	**5.1 E-5**	**2.1 E-6**	**1.2 E-7**	**2 E-6**	**1.8 E-5**	**4.4 E-5**	**2.1 E-5**	**2.7 E-6**	**1.2 E-6**	**6 E-7**	**2.6 E-8**	

Table 5-7. 1 % Cs Release Exceedance Frequency. Contributions from Internal and External Events, Power Generation States

Event	Surry	Biblis-B	Beznau	Robinson IPE	Sizewell	Borssele	Ring hals 2	Peach Bottom	LaSalle	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse bäck 1	Fors-mark 3	Oskars-ham 1
Internal events	6.0 E-6 75%		1.5E-7 7.7%	1.2 E-5 86%	8 E-6 98%	1.5 E-7	2 E-7	2 E-6 12 %	1.4 E-5 48%	1.1 E-5 90%	3.6 E-6 90%	3.7 E-7 >99%	5.4 E-7	2.7 E-8	
Fire	7.7 E-7 9.7%		3.9 E-7 20%		1.3 E-7			1.1 E-5 67 %	1.4 E-5 48%						
Internal flooding	-		2.8 E-7 14%	1.6 E-6 14%				-	7.2 E-7 2.4%	1.2 E-6 10%	3.5 E-7 10%				
External flooding	-		4.6 E-7 23.5%					-							
Earthquake	1.15 E-6 (EPRI hazard curve) 14.5%		6.7 E-7 34.5% (mostly isolation failure)		1.2 E-7			3.5 E-6 (EPRI hazard curve) 21 %	4 E-7 1.3%						
Aircraft impact	-		-					-							
Total of external events	**1.9 E-6 24%**		**1,8 E-6 92%**	**1.6 E-6 14%**	**E-7 2%**			**1.4 E-5 88 %**	**1.5 E-5 52%**	**1.2 E-6 10%**	**3.5 E-7 10%**				
Total, internal + external events	**7.9 E-6**		**2.0 E-6**	**1.4 E-5**	**8.2 E-6**	**1.5 E-7**	**2 E-7**	**1.6 E-5**	**2.9 E-5**	**1.2 E-5**	**4 E-6**	**3.7 E-7**	**5.4 E-7**	**2.7 E-8**	

Table 5-8. 10 % Cs Release Exceedance Frequency. Contributions from Internal and External Events, Power Generation States

Event	Surry	Biblis-B	Beznau	Robinson IPE	Sizewell	Borssele	Ringhals 2	Peach Bottom	LaSalle	Browns Ferry IPE	Perry IPE	Mühle-berg	Barse bäck 1	Fors-mark 3	Oskars-ham 1
Internal events	2 E-6 62%		3.5E-8 9%	5.8 E-6 90%	5 E-6 96%	1 E-7	5 E-8	1 E-6 11.6 %	3.6 E-6 30%	4.5 E-6 90%	4.2 E-7 84%	1.2 E-7 >99%	1.4 E-7	5.E-9	
Fire	2 E-7 6%		1.1 E-8 2.7%		9 E-8			6.6 E-6 76.8 %	8 E-6 66%						
Internal flooding	-		9 E-9 2.5%	6 E-7 10%					1.3 E-7 1%	5 E-7 ?10%	8 E-8 16%				
External flooding	-		1.3 E-8 3.2%												
Earthquake	1 E-6 (EPRI hazard curve) 32%		3.3 E-7 (mostly isolation failure) 82%		9 E-8			1 E-6 (EPRI hazard curve) 11.6 %	2.7 E-7 2.2%						
Aircraft impact	-														
Total of external events	**1.2 E-6 38%**		**3.6 E-7 91%**	**6 E-7 10%**	**1.8 E-7 4%**			**7.6 E-6 88.4 %**	**8.4 E-6 70%**		**8 E-8 16%**				
Total, internal + external events	**3.2 E-6**		**4 E-7**	**6.4 E-6**	**5.2 E-6**	**1 E-7**	**5 E-8**	**8.6 E-6**	**1.2 E-5**	**5 E-6**	**5 E-7**	**1.2 E-7**	**1.4 E-7**	**5 E-9**	

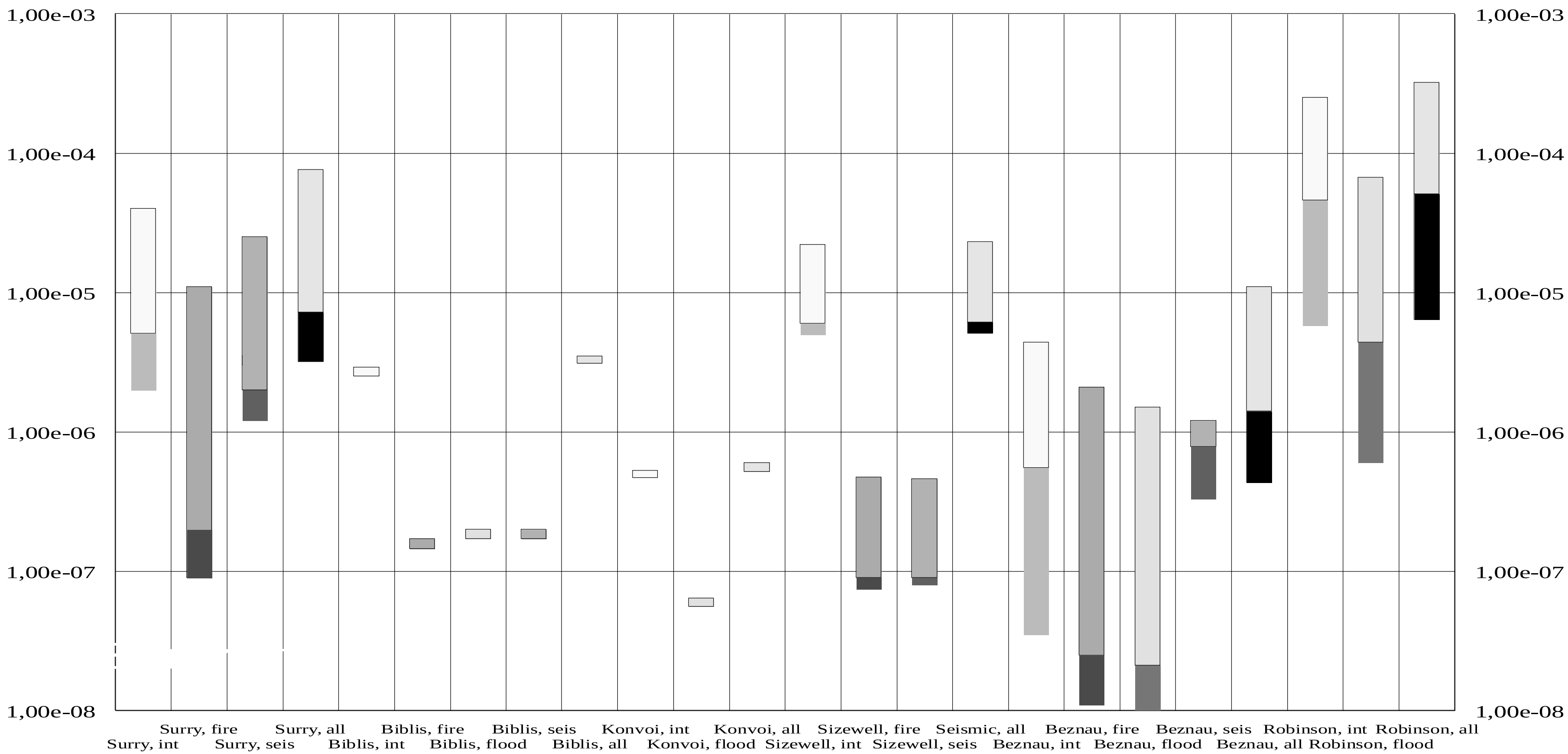

Figure 5-1. Frequencies of core damage, LRCF mode and exceedance of 10% Cs release for internal, fire, flooding, seismic and all events at PWR plants. Top of upper bars: CDF, bottom of upper bars: frequency of LRCF mode, bottom of lower bars: exceedance frequency of 10% Cs release. Light and dark yellow: internal events, light and dark red: fire events, light and dark blue: flooding events, light and dark green: seismic events, grey and black: all events. Length of upper bar: measure of the capability of the containment to prevent LRCF mode, given CD. Length of lower bar: measure of the capability of containment to mitigate large releases, given LRCF mode. (For Biblis and Konvoi level 2 results are not available.)

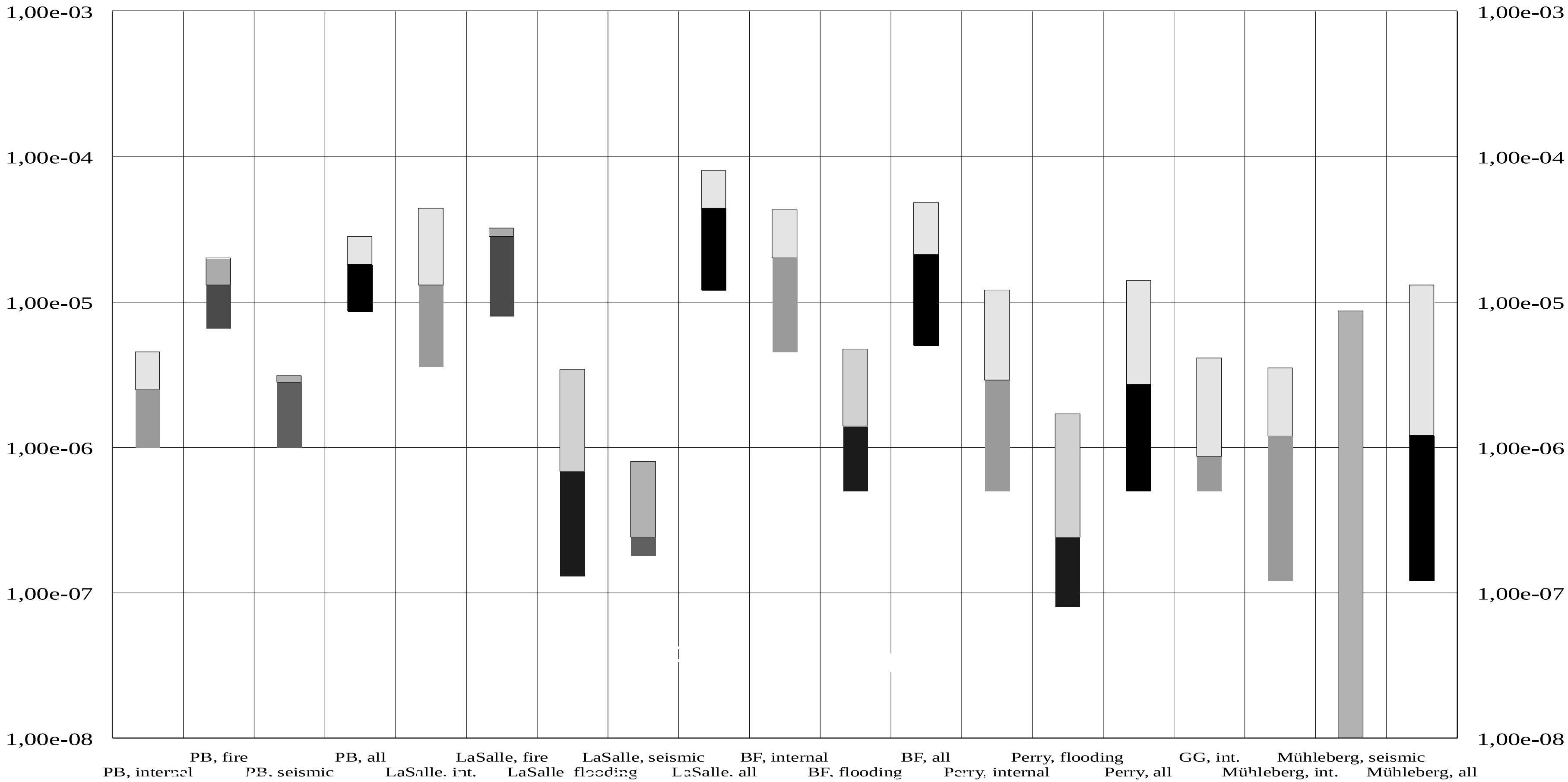

Figure 5-2. Frequencies of core damage, LRCF mode and exceedance of 10% Cs release for internal, fire, flooding, seismic and all events at BWR plants. Top of upper bars: CDF, bottom of upper bars: frequency of LRCF mode, bottom of lower bars: exceedance frequency of 10% Cs release. Light and dark yellow: internal events, light and dark red: fire events, light and dark blue: flooding events, light and dark green: seismic events, grey and black: all events. Length of upper bar: measure of the capability of the containment to prevent LRCF mode, given CD. Length of lower bar: measure of the capability of containment to mitigate large releases, given LRCF mode.