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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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DOCUMENTATION OF THE USE OF SEVERE ACCIDENT COMPUTER CODES IN SELECTED LEVEL-2 PSAs FOR NUCLEAR POWER PLANTS

62813

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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

The use of severe accident computer codes in published PSAs for PWR and BWR plants is documented. The report examines whether correlations exist between quantitative results obtained at the various levels of the analysis and the use of particular computer codes.

FOREWORD

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

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

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

The information contained within this report reflects (along with three other reports) supplemental material which was prepared in conjunction with the main report. This specific report looks at the use of severe accident computer codes in selected Level 2 PSAs.

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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1. INTRODUCTION

In this report the use of severe accident computer codes in the examined PSAs is discussed. This includes discussions whether correlations exist between the quantitative results at the various levels of the analysis and the use of certain severe accident computer codes.

In this report the use of severe accident computer codes in published level-2 PSAs is examined. The discussions cover severe accident phenomena that were found important to level-2 results for PWR and BWR plants, as listed below:

- Analysis of containment loads from in-vessel phenomena, like arrest of core melt progression, temperature induced hot leg/surge line/SGT rupture (only PWRs), in-vessel hydrogen generation, in-vessel steam explosion and bottom head failure.

- Analysis of containment loads from ex-vessel phenomena, like loads at vessel breach, ex-vessel steam explosion, ex-vessel generation of non-condensable gases, loads from combustion of hydrogen and carbon monoxide, molten corium/containment structure interaction, containment structural response.

- Analysis of source term issues, like in-vessel fission product release, transport and retention, scrubbing of fission products in water filled steam generator (only PWRs) or in water pool, fission product release, transport and retention inside containment, releases to the environment.

Particular emphasis is on whether correlations exist between the quantitative results at the various levels of the analysis and the use of certain severe accident computer codes. The magnitude of uncertainties and their influence on the results is also examined.

2. SEVERE ACCIDENT COMPUTER CODES USED IN THE EXAMINED PSAS

In tables 1 - 6 the severe accident computer codes used in the examined PSAs are compiled.

The tables are organised by PSA relevant issues and phenomena; listed are the computer codes applied to the various items in the examined PSAs.

In the following, qualitative and quantitative aspects of the use of the codes are discussed. For each item a short discussion of the phenomenological context is provided.

2.1 Analysis of containment loads from in-vessel phenomena at PWR plants

2.1.1 Arrest of core melt progression, temperature induced hot leg/surge line/SGT rupture.

Core melt progression can be arrested if injection to the RPV can be re-established. Besides recovery of injection by operator action, relevant passive recovery scenarios involve:

- failed steam generator feeding, high pressure in the reactor system resulting from operator failure to depressurise, inability to inject to the RPV, beginning core heat-up. Superheated steam flow is from the core through hot leg, surge line, pressuriser, out of the power operated relief valve (scenario 1).

- depressurisation by temperature induced passive failure of hot leg, surge line or steam generator heating tubes. Once the system is sufficiently depressurised, injection may be recovered. Whether or not RPV integrity can be maintained depends on the timing of the depressurisation.

Three different assessment bases exist in the examined PSAs:

1. In the NUREG-1150 analyses, the quantification of probabilities of passive depressurisation was based on an expert opinion elicitation process. The experts based their quantification on results of calculations with the computer codes MELPROG, TRAC/MELPROG, CORMLT/PSAAC, RELAP5/SCDAP and MAAP, as well as on evaluation of pertinent experiments.

   For scenario (1) a conditional probability of temperature induced hot leg failure of about 0.99 is obtained from the aggregation of the expert's quantifications. If in scenario (1) a seal LOCA is induced (scenario 2) the probability of hot leg failure is much lower, i.e. about 0.15. The figures are taken for all PWRs included in the NUREG-1150 analyses. The underlying probability distribution functions generated by the expert team for the cases without and with seal LOCA are shown in Figures 2.1.1-1 and 2.1.1-2.

2. In all other examined studies but Beznau HSK/ERI, the assessment is based on calculations with the MAAP code. For a scenario similar to the one described above, the following quantifications are made:

   - Robinson IPE: ~0.9 (point value)
   - Maine Yankee IPE: ~0.75 (point value)
   - Beznau PLG: ~0.99 (same data as in NUREG-1150)
   - Ringhals 2: ~0.8
   - Borssele: ~0.73

3. In the Beznau HSK/ERI study the assessment is based on MELCOR results, plant specific calculations with RELAP5/SCDAP and assessment of the TMI accident.

   The conditional probability for hot leg failure in scenario 1 was estimated to be 0.75, and 0.0 in scenario 2

The estimates of the conditional probabilities of temperature induced depressurisation have significant impact on the fraction of core damage sequences remaining at high pressure at time of RPV bottom head failure. An exact correlation can not be established because of the differing shares of relevant sequences and scenarios.

In the examined PSAs, the following percentages of high pressure core melt sequences are reported:

Plant/PSA	Percentage of high pressure core melt sequences
Surry	3%
Zion	2%
Robinson	22%
Maine Yankee	16%
Beznau HSK/ERI	10%
Ringhals 2	12%
Borssele PSA-3	6%
Borssele PSA-97	?

2.1.2 In-vessel hydrogen generation.

The amount of hydrogen generated in the in-vessel phase of core degradation and meltdown is proportional to the fraction of zirconium oxidised. The oxidation process is the result of complex interactions of thermo-hydraulic and chemical phenomena.

Basis for the assessment in the examined PSAs are:

- in the NUREG -1150 studies:

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

  Experts who had experience with several of the computer codes rated MAAP and MARCH lower than the others, because MAAP was considered to underestimate zirconium oxidation, and MARCH to overestimate it.

  The available information was assessed by a formalised expert opinion elicitation process. Subjective probability distribution functions for the amount of oxidised zirconium have been aggregated to one distribution function, which then was used in the quantification process.

  For the investigated cases, the median values of the aggregated distribution functions are between 30% and 50% zirconium oxidation.

- In the IPE-studies:

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

  In the Robinson study, point values are being used, which are in good agreement with the median values in NUREG-1150. In the Maine Yankee study the point values are in the in the upper range of the distribution functions of NUREG 1150. They are generally higher that in the other PWR studies.

  In the HSK/ERI analysis of the Beznau plant the assessment is based on MELCOR calculations. The range for the fraction of oxidised zirconium is 40 - 50%. As point value, 44% is used.

  In the Ringhals 2 and Borssele analyses the assessment is based on MAAP calculations. The range for the fraction of oxidised zirconium is 30% - 52 %.

The uncertainties in the modelling of in-vessel zirconium oxidation are large. However, a significant influence on early containment failure can only be identified in the Maine Yankee analysis. For this plant, with the fuel loaded at the time of the analysis, the ratio "amount of zirconium in the core/ containment volume" is much larger than for the other plants, thus making the containment vulnerable to hydrogen generation. At the other plants examined in this study, the threat from hydrogen burn is insignificant.

2.1.3 In-vessel steam explosion.

In all studies, the assessment of the impact of in-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/. In all examined studies, the potential of in-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. The quantified conditional probabilities for containment failure due to in-vessel steam explosions, given core melt, are in the range 10^{-3} to 10^{-2} for low pressure sequences, and in the range 10^{-4} to 10^{-3} for high pressure sequences.

2.1.4 Bottom head failure.

Important questions are: mode of bottom head failure (HPME, pour or dump); temperature, mass and fraction of metal in the ejected material.

- In the NUREG-1150 analyses the assessment is based on expert judgement. Input to the expert judgement are calculations with the codes MELPROG and MAAP and evaluations of the TMI accident. The investigations covered three cases:

1. high pressure in the reactor system, no accumulator discharge, no upper head injection,

2. intermediate pressure in the reactor system, no accumulator discharge, only partial upper head injection,

3. low pressure in the reactor system, partial accumulator discharge, upper head injection functioning.

The aggregated answers of the experts are shown in the table below.

Mode of bottom head failure				
		Failure mode (fraction)		
Case	RCS pressure (psia)	HPME	Pour	Dump
1	2500	79%	8%	13%
2	2000	60%	27%	13%
3	200-1200	60%	27%	13%

Other studies: Comparable information on this issue is not provided in the examined studies.

2.2 Analysis of containment loads from ex-vessel phenomena at PWR plants

2.2.1 Loads at vessel breach.

According to the quantifications made in the examined PSAs, the loads at vessel breach primarily result from „direct containment heating“ (DCH), which is a superposition of several physical phenomena, most notably:

- blowdown of steam and hydrogen,
- combustion of hydrogen,
- interaction of core debris with water on the containment floor and in the cavity,
- transfer of heat from dispersed debris to the containment atmosphere.

The parameters most important to DCH loads are:

- pressure in the reactor system at time of vessel breach (see the discussion in section 2.1.1),
- amount of unoxidised metal in the core (see the discussion in section 2.1.2),
- amount of ejected core debris,
- size of hole in the RPV,
- depth of water pool in the cavity,
- availability of containment spray.

The following code systems are used for the quantification of containment loads resulting from DCH:

- In NUREG-1150: CONTAIN, MAAP, HMC.
- In the IPE studies and in the studies for Beznau PLG, Ringhals-2, Borssele: MAAP.
- In the Beznau HSK/ERI study: SCDAP/RELAP5, CONTAIN, MAAP.

In the table below conditional probabilities related to DCH are compiled.

Conditional probabilities of DCH			
PSA	Conditional probability of high pressure in reactor system, given core damage	Conditional probability of containment failure due to DCH, given HPME	Conditional probability of containment failure due to DCH, given core damage
NUREG-1150	0.02-0.03	~ 0.2	0.004-0.006
IPE, Ringhals-2, Borssele	0.1-0.22	~ 0.1	0.01-0.025
Beznau HSK/ERI	< 0.1	0.13	0.013

The conditional probabilities are in good agreement for all large dry containments. For most plants DCH is the main contribution to early containment failure.

All presented results on DCH loads are accompanied by large uncertainties. but the impact of the uncertainties on the failure probabilities of the large dry containments is small.

2.2.2 Ex-vessel steam explosion.

In all studies, the assessment of the impact of ex-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/. In all examined studies, the potential of ex-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. The quantified conditional probabilities for containment failure due to ex-vessel steam explosions, given core melt, are in the range 10^{-4} to 10^{-3}.

2.2.3 Ex-vessel generation of non-condensable gases.

Non-condensable gases generated in the ex-vessel phase are:

- hydrogen resulting from unoxidised core debris reacting with water,
- hydrogen and carbon monoxide resulting from core debris/concrete interaction.

A parameter critical to the estimation of the amount of hydrogen generated from unoxidised core debris is the amount of zirconium in the core.

In the examined studies the following codes were used for the prediction of the amount of combustible gases generated:

- in NUREG-1150: CORCON,
- in the Beznau PLG study: MAAP, COMPACT,
- in the Beznau HSK/ERI study: MELCOR, COBURN,
- in all other studies: MAAP.

In the MAAP calculations it is assumed that core debris/concrete interaction is suppressed if the cavity is filled with water. This assumption is not made in other computer codes. Therefore, for situations with the cavity being filled with water, the ex-vessel generation of non-condensable gases is significantly lower for MAAP calculations than for other codes. Otherwise, predictions of the total amount of non-condensable gases generated - scaled to the amount of zirconium in the core - agree well among the various codes. However, significant uncertainties exist on the time history of generation of combustible gases.

2.2.4 Loads from combustion of hydrogen and carbon monoxide.

Distinction is made between loads early in the accident that contribute to early containment failure, and loads late in the accident that contribute to late containment failure or - if applicable - to venting failure.
Codes used for the quantification of containment loads are MAAP, HCTOR, MELCOR and ERPRA-BURN.

Loads relevant to early containment failure depend on the amount of zirconium generated in the in-vessel phase (section 0). For all examined plants but Maine Yankee, the containment loads resulting from combustion of hydrogen in the early phase stay clearly below containment capacities, see typical examples, Figures 0-1 and 0-2. Therefore, the conditional probability, given core damage, of early containment failure due to combustion of gases is insignificant relative to other containment failure modes.

The majority of combustible gases is produced in the late accident phase. Thus, higher loads than in the early phase are seen in this phase, see a typical example, Figure 0-3, which indicates a high likelihood of containment failure due to combustion of gases at plants without venting capabilities.

2.2.5 Molten corium/containment structure interaction.

Molten corium/containment structure interaction can lead to penetration by the core debris of the containment basemat. MAAP, CORCON and MELCOR are used for the analysis of this phenomenon. Basemat penetration is a significant contribution to containment failure at most plants, but its contribution to releases is generally low.

2.2.6 Containment structural response.

Several well established code systems are available for quantification of containment load capacity, for example, NASTRAN; ABAQUS, DYNA3D, NEPTUNE.

In all studies, containment failure is described by cumulative probability functions. Probability of failure begins to rise from practically zero at about 5 bar. For the Surry and Ringhals-2 containments, probability of failure approaches 1 in the range 13-14 bar. For Zion NUREG-1150 and the IPE studies, probability of failure approaches 1 in the range 18-23 bar. In the examined IPE studies, higher pressures than in the other studies are required to fail the containment.

For illustration, see Figures 2.2.6-1 and 2.2.6-3.

For the steel containment of the Beznau plant, probability of failure approaches 1 in the range 9-10 bar, see Figure 2.2.6-4.

2.3 Analysis of source term issues for PWR plants

2.3.1 In-vessel fission product release, transport and retention.

A large number of different code systems is used for predicting fission product release and transport inside the reactor system and to the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Beznau HSK/ERI, MELCOR is used, and MAAP for all other plants.

The agreement among predictions of releases and retention inside the reactor system is reasonably good for noble gases and volatile fission products. For the refractory aerosols, there is more disagreement of predictions.

2.3.2 Scrubbing of fission products in water filled steam generator or in water pool.

For mitigating releases from an unisolated defective steam generator, severe accident management procedures have been put in place at several plants. Essentially, the procedures are aimed at filling up a defective steam generator with a water column in which fission products are retained. For the quantification of the scrubbing effect The MAAP code system is used in the analyses for Ringhals-2 and Borssele, and the MELCOR code system in the Beznau HSK/ERI analysis.

The releases are predicted to be reduced by factors in the range 10 - 100. However, more clarification of this important issue is needed

In the Robinson and Maine Yankee IPEs, reductions of Cs releases by a factor 20-100 are reported for situations in which core debris on the containment floor is covered by an overlying water pool. However, this is to be attributed to two effects:

1. suppression of the core debris/concrete interaction (an assumptions that is made in MAAP, but not in the other codes),

2. fission product scrubbing by the water pool.

The information provided does not permit to differentiate between the two effects.

2.3.3 Fission product release, transport and retention inside containment.

These issues are controlled by several phenomena which are not well understood, most notably thermophoresis, Brownian diffusion, aerosol agglomeration, aerosol plate out on surfaces, settling under influence of gravity. Most of these processes are governed by the aerosol particle size distribution which is not well known. Another important factor influencing deposition and plate out is the time history of the convection processes.

A large number of different code systems is in use for predicting fission product release and transport inside the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Beznau HSK/ERI, MELCOR is used, and MAAP for all other plants.

Differences among predictions are large and difficult to compare and interpret.

2.3.4 Releases to the environment.

For the calculation of source terms the XSOR suite of codes is used in the NUREG-1150 analyses, MELCOR and ERPRA are used in the Beznau HSK/ERI analysis, and MAAP in all other examined studies.

Suitable measures for comparing releases at different plants are the

- conditional probability of exceeding 10% Cs release, given one of the containment failure modes „early containment failure“ (ECF), „containment bypass“ (Bypass) or „isolation failure“ (ISF).

- conditional probability of exceeding 1% Cs release, given core damage.

In the table inserted below, these conditional probabilities are compiled for the plants examined in this section.

Conditional probabilities for Cs releases		
PSA	Conditional probability of exceeding 1% Cs release, given core damage	Conditional probability of exceeding 10% Cs release, given ECF or bypass or ISF
Surry, NUREG-1150	0.15	0.39
Zion, NUREG 1150 update	0.08	0.66
Maine Yankee IPE	0.06	0.19
Robinson IPE	0.1	0.23
Beznau HSK/ERI	0.03	0.05
Ringhals-2	0.01	0.03

The calculated conditional probabilities reflect the combined effect of all issues discussed above, including the associated uncertainties. In view of the described differences, the agreement among plants with comparable retention capabilities is satisfactory.

Among the US PSAs, the IPE studies calculate lower large releases than the NUREG-1150 studies, but the source of the discrepancy is difficult to identify.

The Beznau and Ringhals-2 plants have implemented severe accident management procedures for filling up a defective steam generator with water. The retention capability of the water column is reflected by the significant reduction, relative to the other plants, of the conditional probability of exceeding 10% Cs release, given ECF or bypass or ISF.

Also, at the two plants high capacity filtered containment venting is available and severe accident management equipment and procedures are in place that permit to flood the containment using external water sources. This feature is reflected by the reduction, relative to the other plants, of the conditional probability of exceeding 1% Cs release, given core damage.

2.4 Analysis of containment loads from in-vessel phenomena at BWR plants

2.4.1 Arrest of core melt progression.

Core melt progression can be arrested if injection to the RPV can be re-established. The relevant scenarios involve

- recovery of AC power if the accident was initiated by loss of AC power,
- operator actions to depressurise the reactor and align low pressure injection systems in situations with failed high pressure injection and failed automatic depressurisation.

2.4.2 In-vessel hydrogen generation.

The amount of hydrogen generated in the in-vessel phase of core degradation and meltdown is proportional to the fraction of zirconium oxidised. The zirconium oxidation is the result of complex thermo-hydraulic and chemical interactions. Basis for the assessment in the examined PSA s are:

- In the NUREG -1150 studies:

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

  Experts who had experience with several of the computer codes rated MAAP and MARCH lower than the others: MAAP was considered to underestimate zirconium oxidation, and MARCH to overestimate it.

  The available information was assessed by a formalised expert opinion elicitation process. Subjective probability distribution functions for the amount of oxidised zirconium have been aggregated to one distribution function, which then was used in the quantification process.

  For the investigated cases, the median values of the aggregated distribution functions are between 10% and 25% zirconium oxidation.

- In the IPE-studies:

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

  In the Browns Ferry and Perry studies, nominal values are being used, which are in good agreement with the median values of the distribution functions of NUREG-1150.

  In the HSK/ERI analysis of the Mühleberg plant the assessment is based on MELCOR calculations. The range for the fraction of oxidised zirconium is 21% - 25%.

2.4.3 In-vessel steam explosion.

In all studies, the assessment of the impact of in-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/. In all examined studies, the potential of in-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. The quantified conditional probabilities for containment failure due to in-vessel steam explosions, given core melt, are below 10^{-3}.

2.4.4 Bottom head failure.

Important questions are: mode of bottom head failure (HPME, pour or dump); temperature, mass and fraction of metal in the ejected material.

- In the NUREG-1150 analyses the assessment is based on expert judgement. Input to the expert judgement are calculations with the codes BWRSAR and MAAP and evaluations of the TMI accident. Of the investigated cases, three are presented here:

  1. high pressure in the reactor system, no injection,

  2. low pressure in the reactor system, no injection

  3. low pressure in the reactor system, LPI injection restored, no recriticality after LPI restoration.

The aggregated answers of the experts are shown in the table below.

Mode of bottom head failure				
		Failure mode (fraction)		
Case	RCS pressure	HPME	Pour	Dump
1	high	80%	20%	
2	low		75%	25%
3	low		74%	26%

Other studies: Comparable information on this issue is not provided in the examined studies.

2.5 Analysis of containment loads from ex-vessel phenomena at BWR plants

2.5.1 Loads at vessel-breach.

The pressure rise at vessel breach primarily results a superposition of several physical phenomena, most notably:

- blowdown of steam and hydrogen,
- combustion of hydrogen,
- interaction of core debris with water in the pedestal area,
- transfer of heat from dispersed debris to the containment atmosphere,
- impulse loads.

The parameters most important to DCH loads are:

- pressure in the reactor system at time of vessel breach,
- amount of unoxidised metal in the core,
- amount of ejected core debris,
- size of hole in the RPV,
- depth of water pool in the pedestal area,
- availability of containment spray.

The following code systems are used for the quantification of the pressure rise at vessel breach:

- In NUREG-1150: CONTAIN, MAAP, HMC.
- In the IPE studies and in the studies for Barsebäck, Forsmark 3: MAAP.
- In the Mühleberg PLG study: BWRSAR/CONTAIN.
- In the Mühleberg HSK/ERI study: MELCOR.

Loads due to vessel breach are among the dominant containment failure modes in all examined PSAs.

2.5.2 Ex-vessel steam explosion.

In all studies, the assessment of the impact of ex-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/. In all examined studies, the potential of ex-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. In all examined studies the quantified conditional probabilities for containment failure due to ex-vessel steam explosions, given core melt, are below 10^{-3}.

2.5.3 Ex-vessel generation of non-condensable gases.

Non-condensable gases generated in the ex-vessel phase are:

- hydrogen resulting from unoxidised core debris reacting with water,
- hydrogen and carbon monoxide resulting from core debris/concrete interaction.

A parameter critical to the estimation of the amount of hydrogen generated from unoxidised core debris is the amount of zirconium in the core.

In the examined studies the following codes were used for the prediction of the amount of combustible gases generated:

- in NUREG-1150: CORCON,
- in the Mühleberg PLG study: MAAP,
- in the Mühleberg HSK/ERI study: MELCOR,
- in all other studies: MAAP.

In the MAAP calculations it is assumed that core debris/concrete interaction is suppressed if the debris is covered by a overlying pool of water. This assumption is not made in other computer codes. Therefore, for situations with the cavity being filled with water, the ex-vessel generation of non-condensable gases is significantly lower for MAAP calculations than for other codes. Otherwise, predictions of the total amount of non-condensable gases generated - scaled to the amount of zirconium in the core - agree well among the various codes. However, significant uncertainties exist on the time history of generation of combustible gases.

2.5.4 Combustion of hydrogen and carbon monoxide.

Distinction is made between loads early in the accident that contribute to early containment failure, and loads late in the accident that contribute to late containment failure or - if applicable - to venting failure.
Codes used for the quantification of containment loads are MAAP, HCTOR, MELCOR and BWRSAR/CONTAIN.

Loads relevant to early containment failure depend on the amount of zirconium generated in the in-vessel phase (section 2.4.2). For examined plants with Mark III containments, which are not inerted, hydrogen combustion is a dominant contribution to containment failure. At all plants with inerted containment, the conditional probability, given core damage, of early containment failure due to combustion of gases is practically zero.

2.5.5 Molten corium/containment interaction.

Molten corium exiting the reactor pressure vessel can erode the pedestal structure. In many Mark I containments the in-pedestal sump volume is too small to accomodate the molten core. Therefore, molten corium may spill over to the drywell floor and lead to drywell shell meltthrough and subsequent erosion of the concrete drywell structure.

- In the NUREG-1150 studies, expert elicitation was performed for both pedestal erosion and drywell attack. The results provided by the experts differed widely. Research results that have since become available have removed some of the discrepancies, as they indicate that concrete erosion and drywell shell attack can be reduced by the presence of water. This issue was controversial at the time of the expert elicitation.

  Input to the predictions by the experts were calculations with CORCON.

- In the Mühleberg PLG study, the CONTAIN is used, and in the Mühleberg HSK/ERI the TEXAS code is used.

- In all other studies, predictions are based on MAAP calculations.

For US plants with Mark I containments there still may be significant contributions to early containment failure from molten corium/containment interaction. For all other plants, only insignificant contributions are reported.

2.5.6 Containment structural response.

Several well established code systems are available for quantification of containment load capacity, for example, NASTRAN; ABAQUS, DYNA3D, NEPTUNE.

In all studies, containment failure is described by cumulative probability functions. For plants with Mark III containments, probability of failure begins to rise from practically zero at about 4-5 bar, and probability of failure approaches 1 in the range 7-8 bar. For plants with Mark I containments, probability of failure begins to rise from practically zero at about 7-8 bar, and probability of failure approaches 1 in the range 18-20 bar.

For illustration, see Figures 2.5.6-1 and 2.5.6-2.

2.6 Analysis of source term issues for BWR plants

2.6.1 In-vessel fission product release, transport and retention.

A large number of different code systems is used for predicting fission product release and transport inside the reactor system and to the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Mühleberg PLG, BWRSAR(RMA), CORSOR-M, and for Mühleberg HSK/ERI, MELCOR is used, For all other plants, MAAP is used.

The agreement among predictions of releases and retention inside the reactor system is reasonable for noble gases, CsI and CsOH. For Te and the refractory aerosols, there is considerable disagreement of predictions.

2.6.2 Scrubbing of fission products in water pool.

In most accident sequences, gas mixtures containing aerosol particle pass through the pressure suppression pool where very effective scrubbing of fission products takes place. For the quantification of the scrubbing effect STCP with the SPARC module is used in NUREG-1150, MELCOR is used for Mühleberg HSK/ERI, and MAAP for the others. Reported fission product reduction factors typically are in excess of 1000.

2.6.3 Fission product release, transport and retention inside containment.

These issues are controlled by several phenomena which are not well understood, most notably thermophoresis, Brownian diffusion, aerosol agglomeration, aerosol plate out on surfaces, settling under influence of gravity. Most of these processes are governed by the aerosol particle size distribution which is not well known. Another important factor influencing deposition and plate out is the time history of the convection processes.

A large number of different code systems is in use for predicting fission product release and transport inside the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Mühleberg PLG, CONTAIN is used, for Mühleberg HSK/ERI, MELCOR and ERPRA are used, and MAAP for all other plants.

Differences among predictions are large and difficult to compare and interpret.

2.6.4 Releases to the environment.

For the calculation of source terms the XSOR suite of codes is used in the NUREG-1150 analyses, MELCOR and ERPRA are used in the Mühleberg HSK/ERI analysis, and MAAP in all other examined studies.

Suitable measures for comparing releases at different plants are the:

- conditional probability of exceeding 10% Cs release, given one of the containment failure modes „early containment failure“ (ECF), „containment bypass“ (Bypass) or „isolation failure“ (ISF),

- conditional probability of exceeding 1% Cs release, given core damage.

In the table inserted below, these conditional probabilities are compiled for the plants examined in this section.

Conditional probabilities for Cs releases		
PSA	Conditional probability of exceeding 1% Cs release, given core damage	Conditional probability of exceeding 10% Cs release, given ECF or bypass or ISF
Peach Bottom, NUREG-1150	0.46	0.54
Grand Gulf, NUREG 1150	0.36	0.58
Browns Ferry IPE	0.25	0.22
Perry IPE	0.33	0.26
Mühleberg HSK/ERI	0.1	0.13
Barsebäck-2	0.13	0.36

The calculated conditional probabilities reflect the combined effect of all issues discussed above, including the associated uncertainties. In view of significant differences in the quantifications in the various analysis steps, the agreement among plants with comparable retention capabilities is satisfactory.

Among the PSAs for US plants, the IPE studies calculate lower large releases than the NUREG-1150 studies, but the source for this discrepancy can only be speculated on. One reason may be the different treatment of core debris covered by water by the MAAP code in which core concrete interaction is practically suppressed in such situations, see section 2.5.3.

At Mühleberg and Barsebäck, high capacity filtered containment venting is available and severe accident management equipment and procedures are in place that permit to flood the containment using external water sources. This feature is reflected by the reduction, relative to the other plants, of the conditional probability of exceeding 1% Cs release, given core damage.

Mühleberg has the lowest conditional probability of exceeding 10% Cs release. In relation to US plants, this can be explained by the much larger in-pedestal sump volume which can easily accomodate the whole molten core. This practically eliminates the drywell attack problems seen in US plants with Mark I containment.

In relation to Barsebäck which does not have the drywell attack problem, the low exceedance frequency can be explained by additional retention in the strong reactor building which acts as a secondary containment and from which the release path for sequences bypassing the filter is through an outer water filled torus.

FIGURES

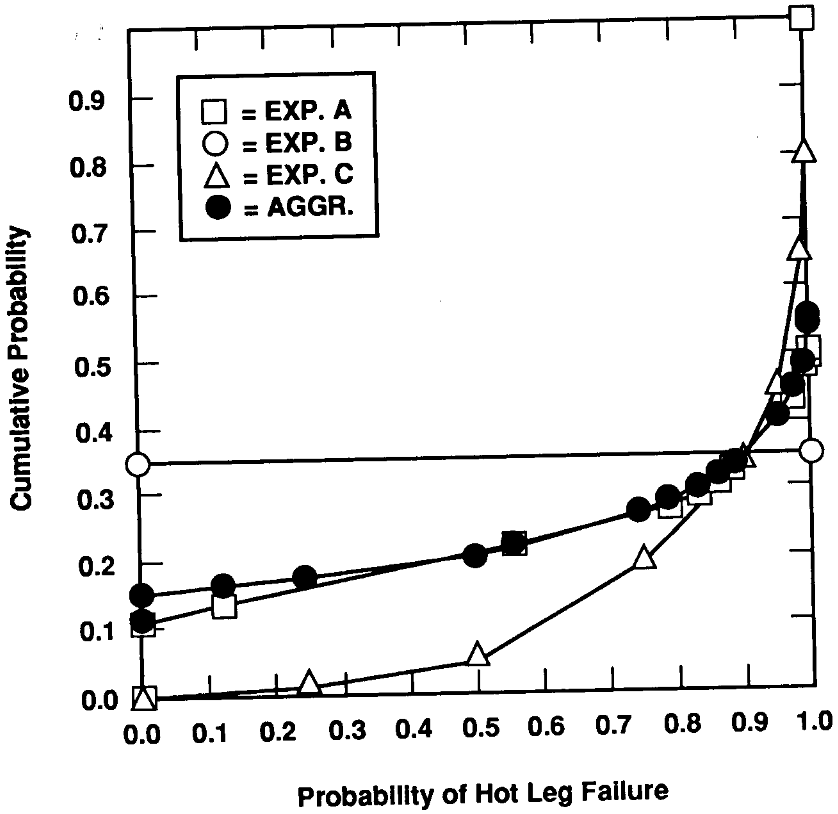

Figure 2.1.1-1. Induced hot leg failure in PWRs, scenario 1 (from NUREG/CR-4551)

Figure 1-1. Case 1: Induced Hot Leg Failure in PWRs.

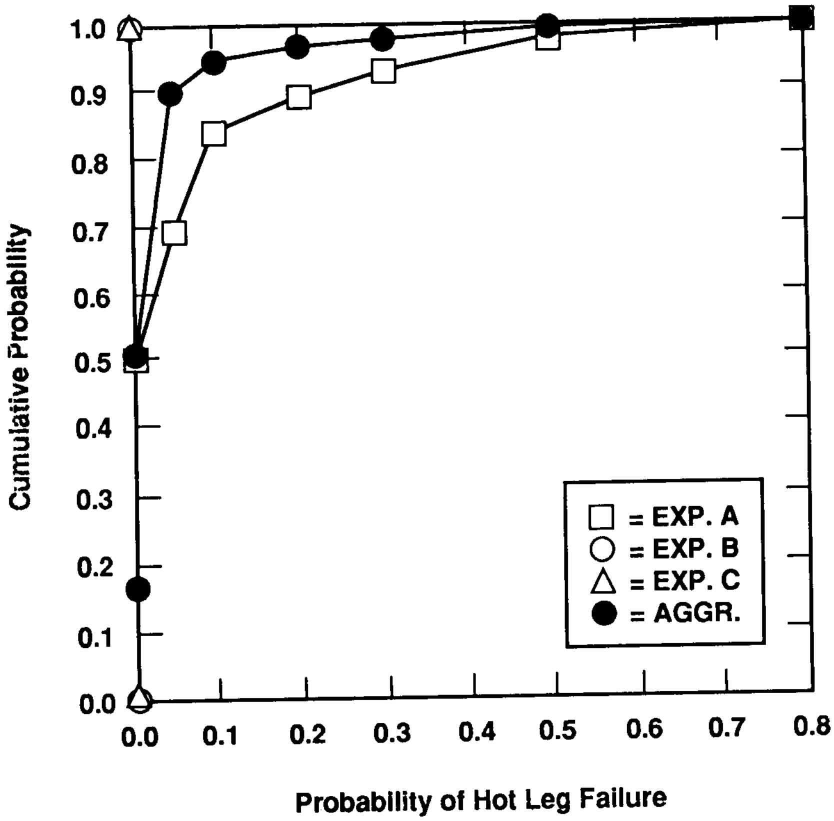

Figure 2.1.1-2. Induced hot leg failure in PWRs, scenario 2 (from NUREG/CR-4551)

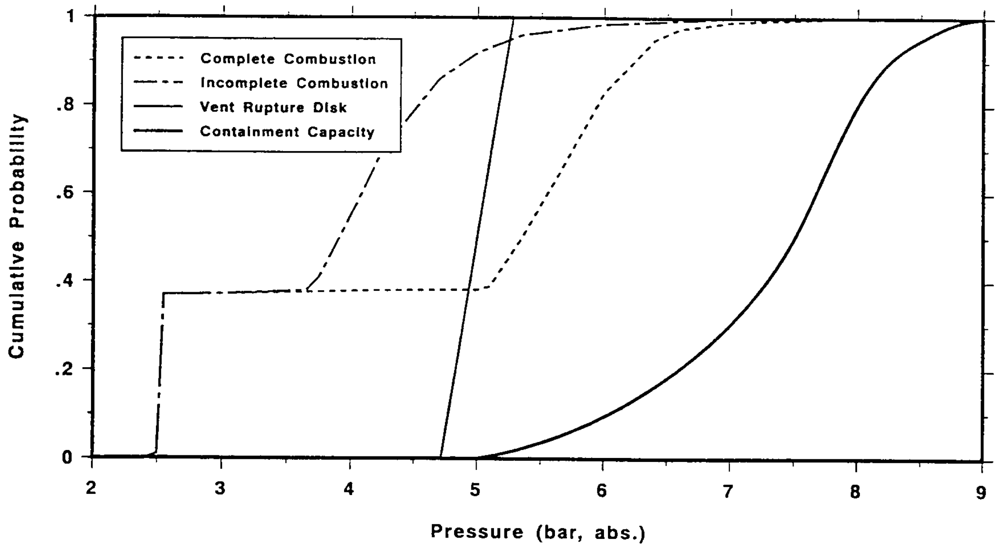

Figure 2.2.4-1. Distribution of loads due to hydrogen combustion at vessel breach for a high pressure scenario at Beznau (from ERI/HSK 94-301, Vol. 2)

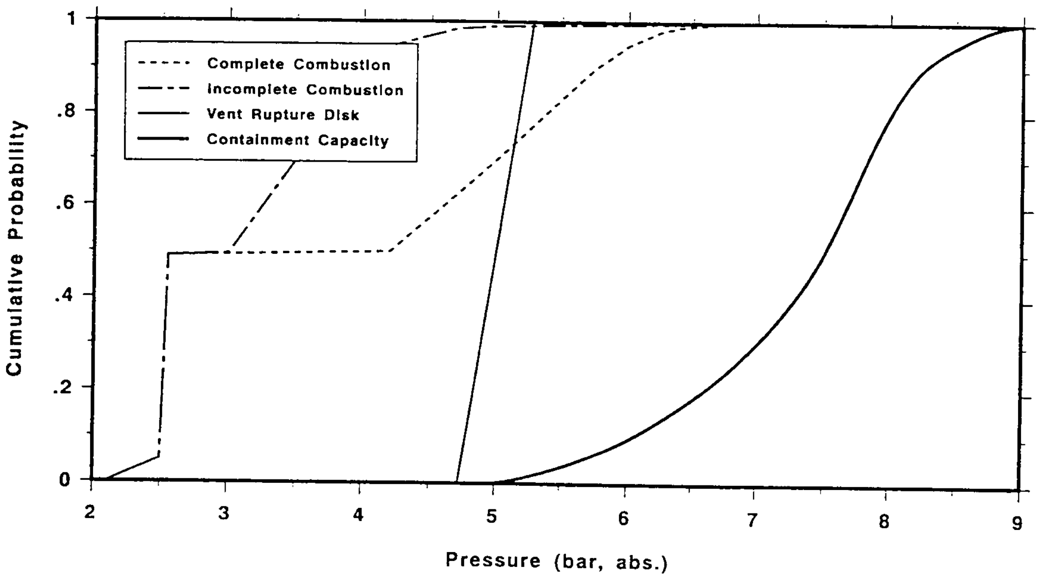

Figure 2.2.4-2. Distribution of loads due to hydrogen combustion at vessel breach for a low pressure scenario at Beznau (from ERI/HSK 94-301, Vol. 2)

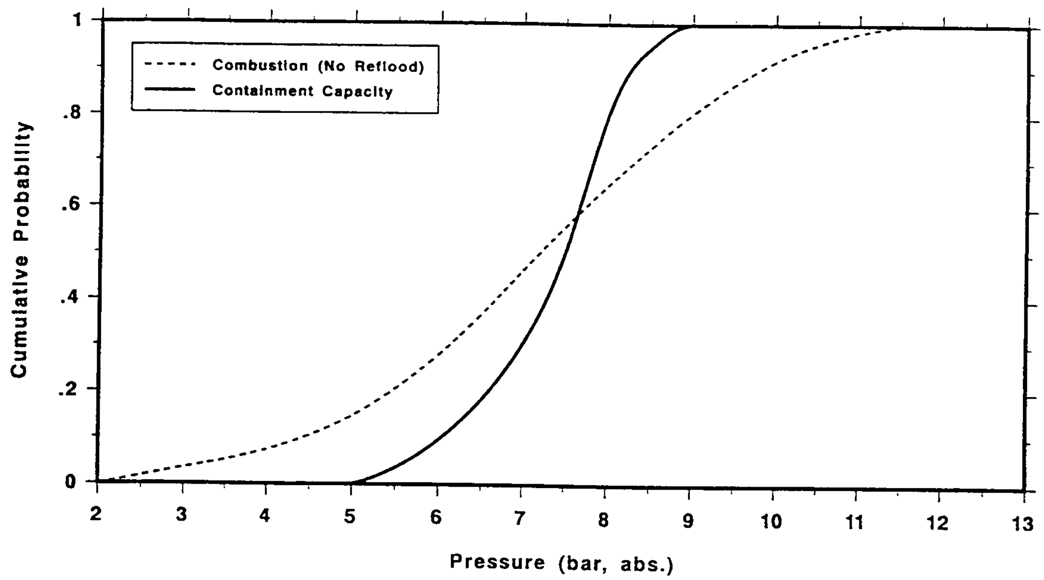

Figure 2.2.4-3. Distribution of loads due to hydrogen and carbon monoxide in the late stages of an accident at Beznau (from ERI/HSK 94-301, Vol. 2)

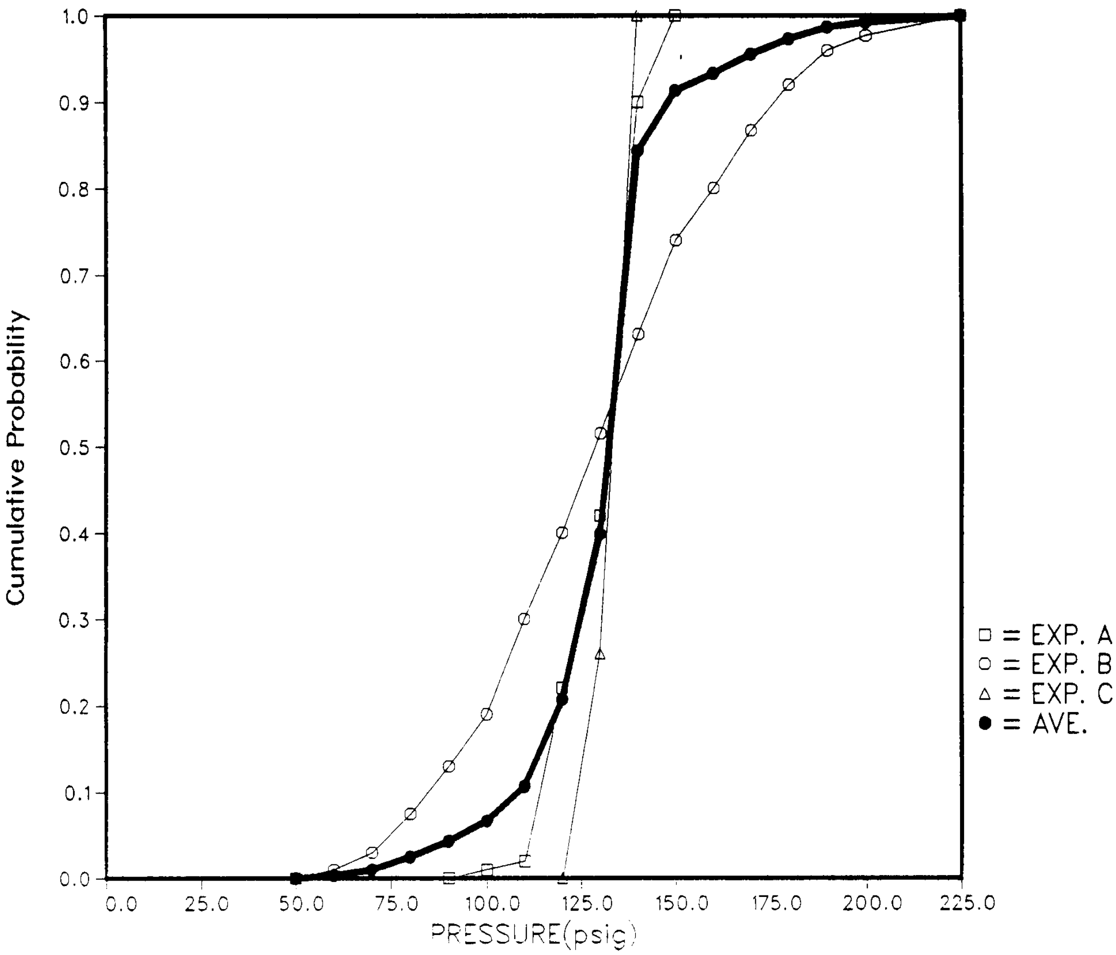

Figure 2.2.6-1. Probability distribution of the failure pressure for the Zion containment (from NUREG/CR-4551)

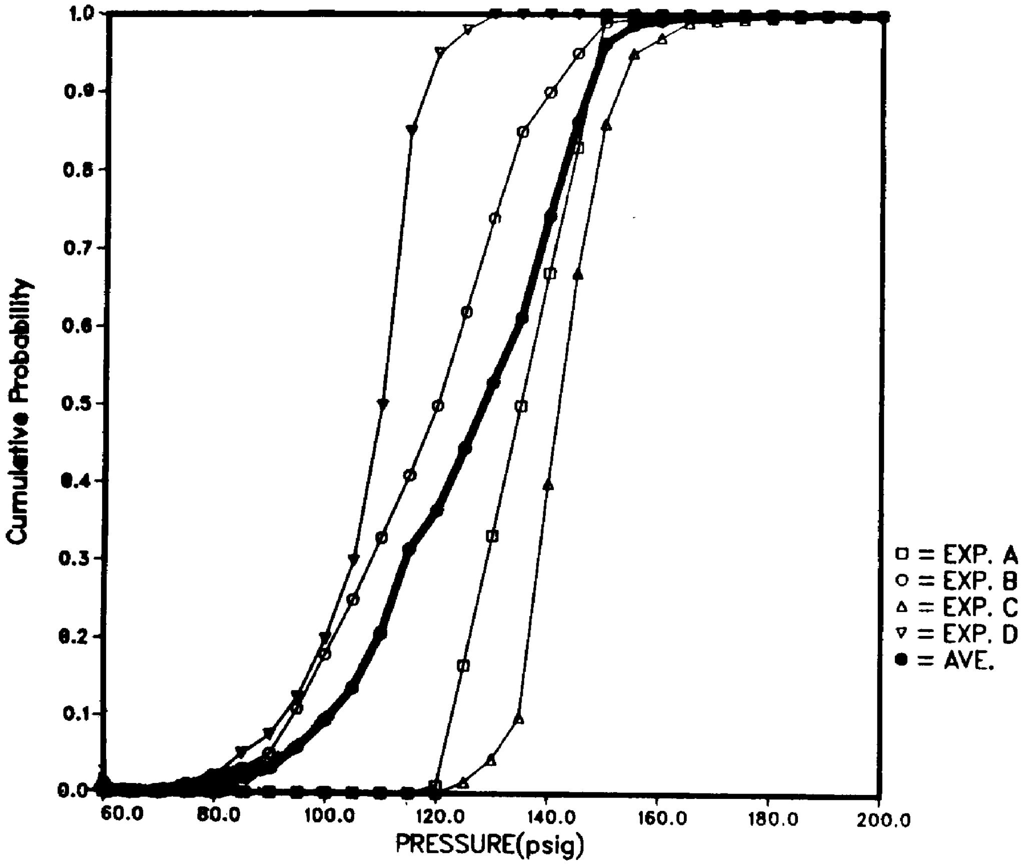

Figure 2.2.6-2. Probability distribution of the failure pressure for the Surry containment (from NUREG/CR-4551)

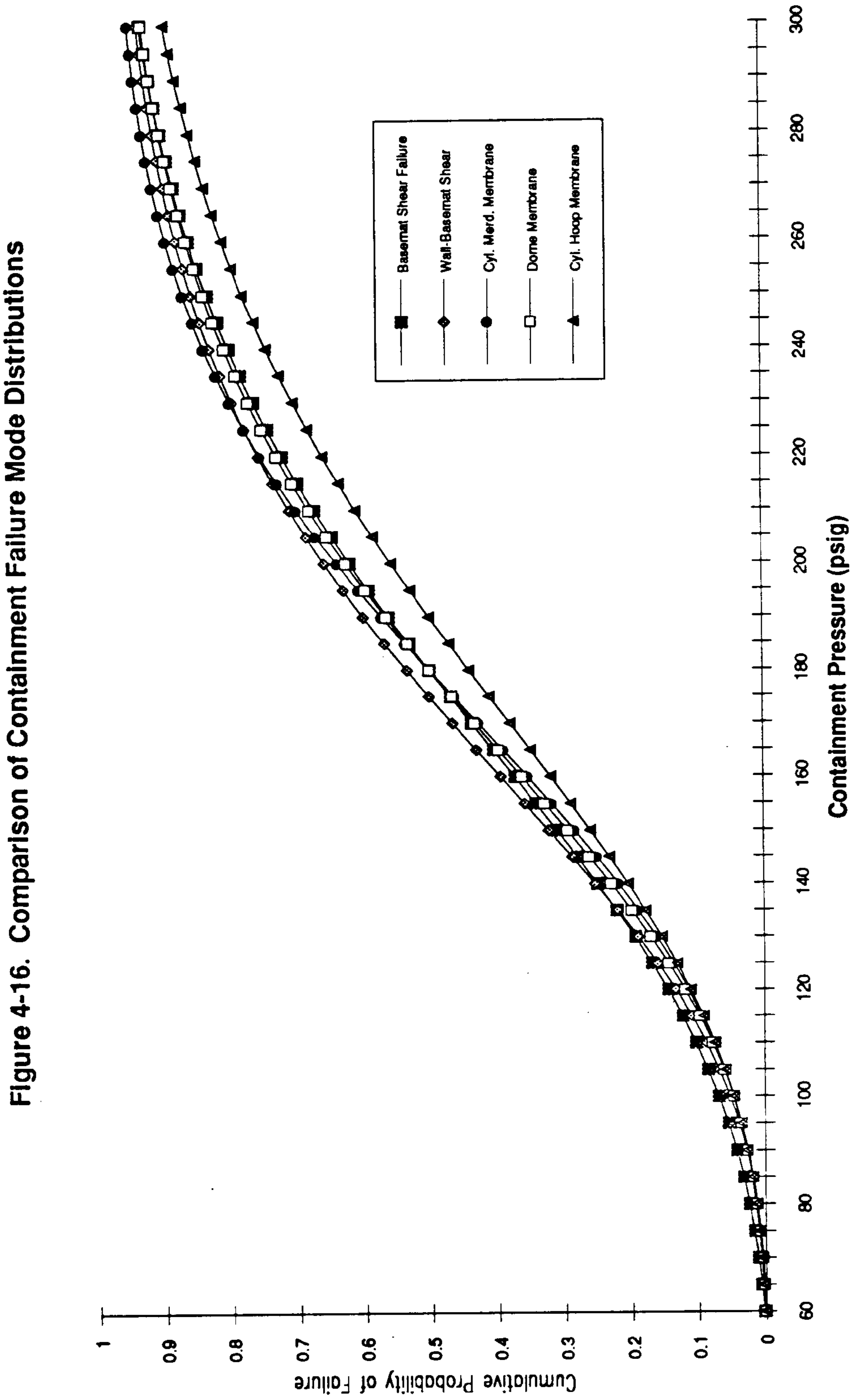

Figure 2.2.6-3. Probability distribution of the failure pressure for the Robinson (HRB2) containment (from Robinson IPE)

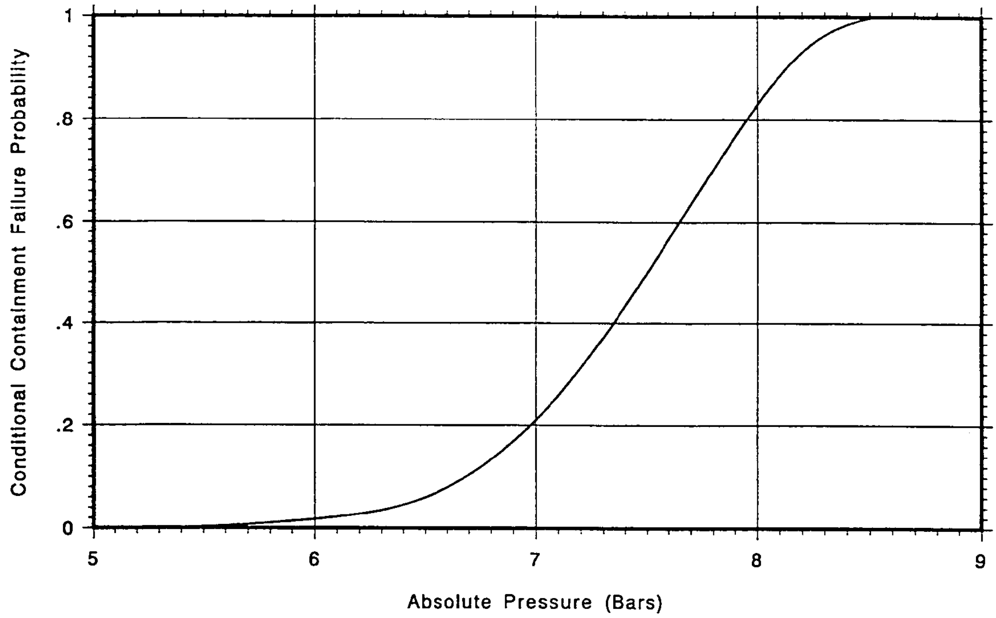

Figure 2.2.6-4. Probability distribution of the failure pressure for the Beznau containment (from ERI/HSK 94-301)

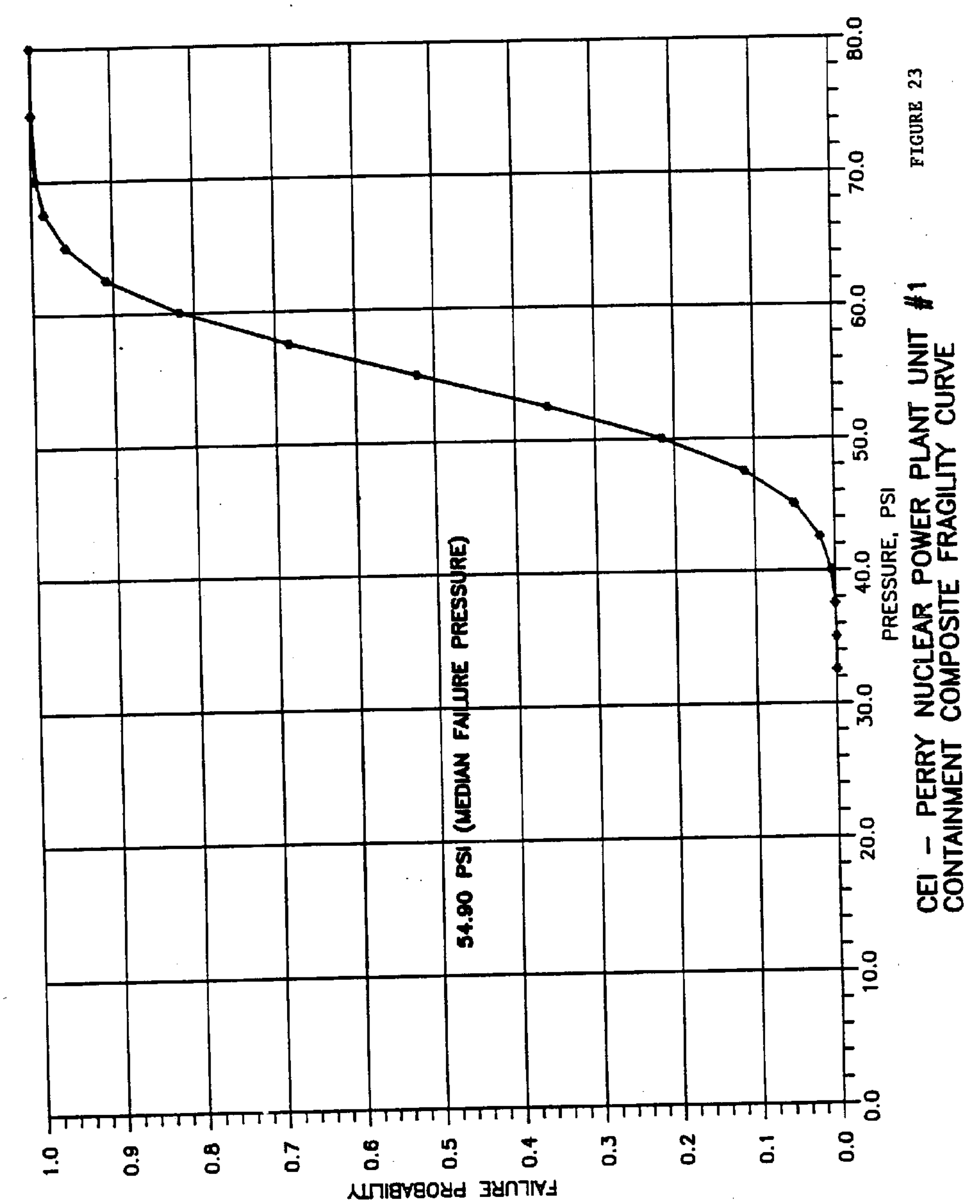

Figure 2.5.6-1. Probability distribution of the failure pressure for the Perry containment (from Perry IPE)

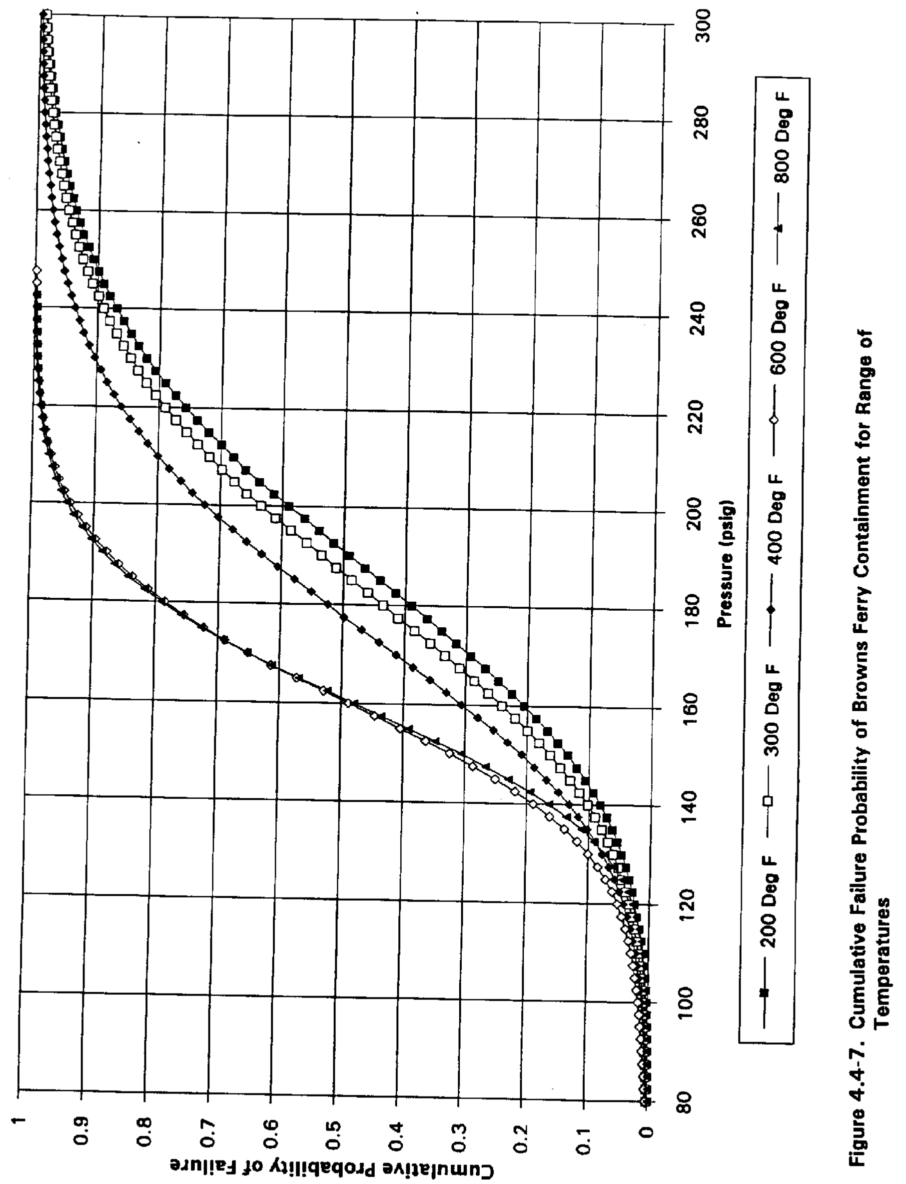

Figure 2.5.6-2. Probability distribution of the failure pressure for the Browns Ferry containment (from Browns Ferry IPE)

Table 1. Computer codes used in the examined PSAs for analysis of containment loads from in-vessel phenomena in PWRs

Phenomena PSA	Arrest of core melt progression	Temperature induced hot leg/surge line/SGT rupture	In vessel hydrogen generation	In-vessel steam explosion	Bottom head failure
NUREG-1150 Surry	Depends on Level-1 systems analysis parameters Likelihood of passive depressurisation mechanisms (column 2) Rate of accident progression (MELCOR calculations)	Hot leg/surge line rupture MELPROG, TRAC/MELPROG, CORMLT/PSAAC, RELAP5/SCDAP, MAAP used by expert panel members	MELPROG SCDAP CORMLT MARCH MAAP used by expert panel members	Expert judgement, based on USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/	MELPROG, MAAP analysis of TMI accident, used by expert panel members
NUREG-1150 Zion	Depends on Level-1 systems analysis parameters Likelihood of passive depressurisation mechanisms (column 2) Rate of accident progression (MELCOR calculations	Hot leg/surge line rupture MELPROG, TRAC/MELPROG, CORMLT/PSAAC, RELAP5/SCDAP, MAAP used by expert panel members	MELPROG SCDAP CORMLT MARCH MAAP used by expert panel members	Expert judgement based on USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/	MELPROG, MAAP analysis of TMI accident, used by expert panel members
Robinson IPE	Depends on Level-1 systems analysis parameters Likelihood of passive depressurisation mechanisms (column 2)	MAAP	MAAP	Expert judgement	MAAP
Maine Yankee IPE	Depends on Level-1 systems analysis parameters Likelihood of passive depressurisation mechanisms (column 2)	MAAP	MAAP	Expert judgement	MAAP
Bezznau PLG	MAAP	MAAP	MAAP	Expert judgement	MAAP

Table 1. Computer codes used in the examined PSAs for analysis of containment loads from in-vessel phenomena in PWRs

Phenomena PSA	Arrest of core melt progression	Temperature induced hot leg/surge line/SGT rupture	In vessel hydrogen generation	In-vessel steam explosion	Bottom head failure
Beznau HSK/ERI	MELCOR	SCDAP/RELAP5, NUREG-1150 results, TMI evaluation	MELCOR	Expert judgement, based on work by Theofanus /r31/ Corradini/x2/, and HSK sponsored analyses.	MELCOR
Ringhals 2	MAAP	MAAP	MAAP	Expert judgement, based on work by Theofanus /r3/ Corradini (r2)	MAAP
Borssele	MAAP	MAAP	MAAP	Expert judgement	MAAP

Table 2. Computer codes used in the examined PSAs for analysis of containment loads resulting from ex-vessel phenomena, PWRs.

Phenomena PSA	Loads at vessel breach	Ex-vessel steam explosion	Ex-vessel generation of non-condensable gases	Combustion of hydrogen and carbon monoxide	Molten corium /containment structure interaction	Containment structural response to pressurisation
NUREG-1150 Surry	CONTAIN, MAAP, HMC	Expert judgement, based on NUREG 1116 /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/	CORCON	Expert judgement, HECTR	CORCON	
NUREG-1150 Zion	CONTAIN, MAAP	Expert judgement based NUREG 1116 /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/		Expert judgement, HECTR		Structural analysis codes
Robinson IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Maine Yankee IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Beznau PLG	MAAP	Expert judgement	MAAP	COMPACT, MAAP		Structural analysis codes
Beznau HSK/ERI	SCDAP/RELAP5, CONTAIN, MAAP	Expert judgement, based on work by Theofanus /r3/ Corradini /r2/, and HSK sponsored analyses.	MELCOR	MELCOR, ERPRA-BURN	MELCOR	Structural analysis codes
Ringhals 2	MAAP	Expert judgemen based on NUREG 1116 /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/	MAAP	MAAP	MAAP	Structural analysis codes
Borssele	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes

Table 3. Computer codes used in the examined PSAs for analysis of source term issues, PWRs.

Phenomena PSA	In-vessel fission product release, transport and retention	Scrubbing in water filled steam generator or in water pool	Fission product release, transport and retention inside containment	Environmental release
NUREG-1150 Surry	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, experiments,		STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, experiments,	SURSOR
NUREG-1150 Zion	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, experiments,		STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, experiments,	ZISOR
Robinson IPE	MAAP	MAAP	MAAP	MAAP
Maine Yankee IPE	MAAP		MAAP	MAAP
Beznau PLG	MAAP		MAAP	MAAP
Beznau HSK/ERI	MELCOR	MELCOR	MELCOR	MELCOR, ERPRA
Ringhals 2	MAAP	MAAP	MAAP	MAAP
Borssele	MAAP	MAAP	MAAP	MAAP

Table 4. Computer codes used in the examined PSAs for analysis of containment loads from in-vessel phenomena, BWRs.

Phenomena PSA	Arrest of core melt progression	In vessel hydrogen generation	In-vessel steam explosion	Bottom head failure
NUREG-1150 Peach Bottom		MELPROG, SCDAP, CORMLT, MAAP, MARCH, BWRSAR and APRIL	Expert judgement based on USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/	BWRSAR Expert judgement
NUREG-1150 Grand Gulf		MELPROG, SCDAP, CORMLT, MAAP, MARCH, BWRSAR and APRIL	Expert judgement based on USNRC Steam Explosion Review Group (NUREG 1116) /r1/, Corradini /r2/, Theofanus /r3/, Turland et al. /r4/	BWRSAR Expert judgement
Browns Ferry IPE		MAAP	Expert judgement	MAAP
Perry IPE		MAAP	Expert judgement	MAAP Expert judgement,
Mühleberg PLG	BWRSAR	BWRSAR/CONTAIN	Expert judgement	MAAP
Mühleberg HSK/ERI		MELCOR	Expert judgement	Expert judgement
Forsmark 3		MAAP	Expert judgement	MAAP
Barsebäck 1/2		MAAP	Expert judgement	MAAP

Table 5. Computer codes used in the examined PSAs for analysis of containment loads resulting from ex-vessel phenomena, BWRs

Phenomena PSA	Loads at vessel breach	Ex-vessel steam explosion	Ex-vessel generation of non-condensable gases	Combustion of hydrogen and carbon monoxide	Molten corium/containment interaction	Containment structural response
NUREG-1150 Peach Bottom	CONTAIN, MAAP, HMC	Expert judgement	CORCON	Expert judgement, HECTR, MELCOR	CORCON	Structural analysis codes
NUREG-1150 Grand Gulf	CONTAIN, MAAP, HMC	Expert judgement		Expert judgement, HECTR, MARCH-HECTR, MELCOR		Structural analysis codes
Browns Ferry IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Perry IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Mühleberg PLG	BWRSAR/CONTAIN	Expert judgement	MAAP	BWRSAR/CONTAIN	CONTAIN	Structural analysis codes
Mühleberg HSK/ERI	MELCOR	TEXAS, expert judgement	MELCOR	MELCOR	MELCOR, TEXAS	Structural analysis codes
Forsmark 3	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Barsebäck 1/2	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes

Table 6. Computer codes used in the examined PSAs for analysis of source term issues, BWRs.

Phenomena PSA	In-vessel fission product release and retention	Scrubbing in suppression pool	Ex-vessel fission product release, transport and depletion inside containment	Environmental release
NUREG-1150 Peach Bottom	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2/VANESSA, NAUA, experiments,	STCP	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2/VANESSA, NAUA, experiments,	PBSOR
NUREG-1150 Grand Gulf	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2/VANESSA, NAUA, experiments,	STCP	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2/VANESSA, NAUA, experiments,	GGSOR
Browns Ferry IPE	MAAP	MAAP	MAAP	MAAP
Perry IPE	MAAP	MAAP	MAAP	MAAP
Mühleberg PLG	BWRSAR (RMA); CORSOR-M	BWRSAR	CONTAIN	MAAP
Mühleberg HSK/ERI	MELCOR	MELCOR, ERPRA	MELCOR,ERPRA	MELCOR, ERPRA
Forsmark 3	MAAP	MAAP	MAAP	MAAP
Barsebäck 1/2	MAAP	MAAP	MAAP	MAAP