SAFETY SERIES No. 50-P-8

SAFETY PRACTICES

safety series

Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2)

Accident Progression, Containment Analysis and Estimation of Accident Source Terms

A PUBLICATION WITHIN THE NUSS PROGRAMME

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1995

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PROCEDURES FOR CONDUCTING PROBABILISTIC SAFETY ASSESSMENTS OF NUCLEAR POWER PLANTS (LEVEL 2)

Accident progression, containment analysis and estimation of accident source terms

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

Printed by the IAEA in Austria
May 1995
STI/PUB/969

SAFETY SERIES No. 50-P-8

PROCEDURES FOR CONDUCTING PROBABILISTIC SAFETY ASSESSMENTS OF NUCLEAR POWER PLANTS (LEVEL 2)

Accident progression, containment analysis and estimation of accident source terms

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1995

VIC Library Cataloguing in Publication Data

Procedures for conducting probabilistic safety assessments of nuclear power plants (Level 2) : accident progression, containment analysis and estimation of accident source terms. — Vienna : International Atomic Energy Agency, 1995.

p. ; 24 cm. — (Safety series, ISSN 0074-1892 ; 50-P-8)
STI/PUB/969
ISBN 92-0-102195-X
Includes bibliographical references.

1. Nuclear power plants—Risk assessment. 2. Nuclear reactors—Safety measures. 3. Nuclear power plants—Reliability. I. International Atomic Energy Agency. II. Series.

VICL 95-00120

FOREWORD

Probabilistic safety assessment (PSA) is increasingly important in the safe design and operation of nuclear power plants. The activities of the International Atomic Energy Agency in this area are focused on facilitating the use of PSA by reviewing the techniques developed in Member States, assisting in the formulation of procedures and helping Member States to apply such procedures to enhance the safety of nuclear power plants.

In this context a set of publications is being prepared to establish a consistent framework for conducting a PSA and forms of documentation that would facilitate the review and utilization of the results. Since December 1986 several Advisory Group meetings, Technical Committee meetings and Consultants meetings have been convened by the IAEA in order to prepare the publications.

The lead publication for this set establishes the role of PSA and probabilistic safety criteria in nuclear power plant safety. Other publications present procedures for the conduct of PSA in nuclear power plants and recognized practices for specific areas of PSA, such as the analysis of common cause failures, human errors and external hazards and collection and analysis of reliability data.

The publications are intended to assist technical persons performing or managing PSAs. They often refer to the existing PSA literature, which should be consulted for more specific information on the modelling details. Therefore, only those technical areas deemed to be less well documented in the literature have been expanded upon. The publications do not prescribe particular methods but they describe the advantages and limitations of various methods and indicate the ones most widely used to date. However, they are not intended to discourage the use of new or alternative methods; in fact the advancement of all methods to achieve the objectives of PSA is encouraged.

The present publication on Level 2 PSA is based on a compilation and review of practices in various Member States. It complements Safety Series No. 50-P-4, issued in 1992, on Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1).

The IAEA wishes to convey its thanks to all those who participated in the drafting and review of the publication, in particular M. Khatib-Rahbar, who was the principal contributor.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Probabilistic safety assessments (PSAs) for nuclear power plants are conducted to yield insights into the design and performance of the plants and their potential environmental effects. This includes the identification of dominant risk contributors, determination of the vulnerabilities of plant and containment systems, and comparison of options for risk reduction. Information on the design and operation of the plant, component reliability, human–machine interaction, the physical progression of events, and potential health and environmental impacts is processed with analytical models to estimate plant safety.

In order to promote the use of PSA techniques by Member States, the IAEA is producing a comprehensive set of publications on procedures for conducting PSAs of nuclear power plants. Procedures for conducting Level 1 PSAs, in which a set of possible accident sequences in response to various initiating events is developed, are given in IAEA Safety Series No. 50-P-4 [1].

This Safety Practice presents procedures for conducting Level 2 PSAs. A Level 2 PSA covers events occurring in accidents that generate thermal and mechanical loads on the containment boundaries with the potential for causing structural failure and consequent release of radioactive material to the environment. Other published procedures and guidance for Level 2 PSAs are listed in Refs [2–6].

In a Level 3 PSA, off-site consequences are assessed and total risk integration is performed. A complementary publication on procedures for Level 3 PSAs and other publications on specific topics in PSA and the treatment of external hazards will follow.

Methods that have been either used or proposed for Level 2 analysis include: (1) event trees and/or fault trees [2, 3, 6]; (2) Markov methods [7]; and (3) direct uncertainty propagation methods [8–9].

An integrated model can be used in which ranges and probability distributions are assigned to uncertain parameters and issues and direct simulation techniques are used to propagate the uncertainties to deduce the uncertainties in outcomes [8]. Although such techniques have been used in several studies for a particular phenomenon or a narrow phase of the accident, they have not generally been applied to accident progression studies owing to incompleteness in the modelling in the available codes and considerations of computing time. The other methods have the advantage that they allow the analyst to make full use of the available codes and to include expert judgement where required. By far the most common approach in current Level 2 PSAs is to use event trees and/or fault trees. This Safety Practice follows this approach.

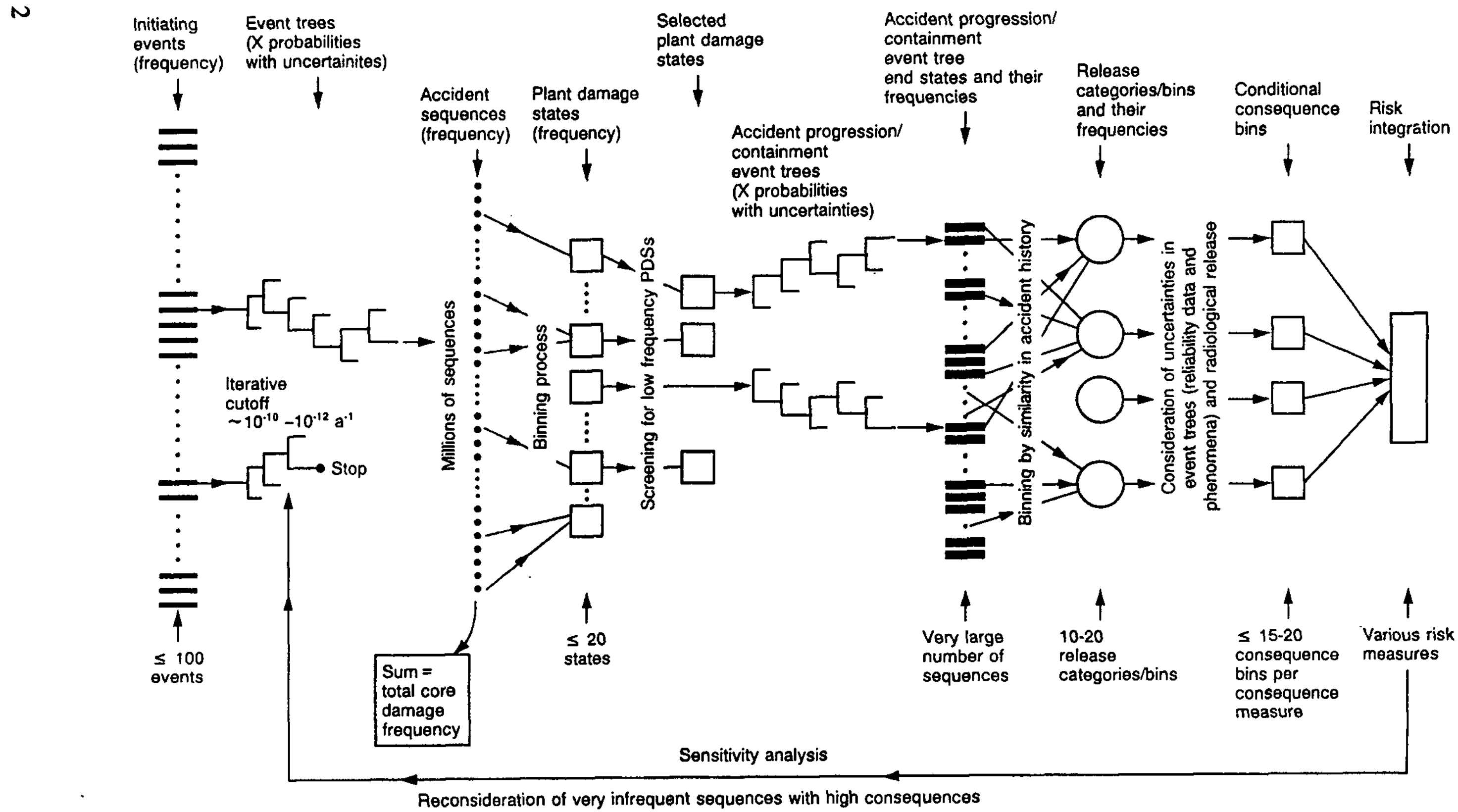

FIG. 1. Method for probabilistic safety assessment.

Figure 1 depicts the major analytical elements of a PSA [10, 11]. The starting point for Level 2 PSA is the grouping of a large number of accident sequences, derived in a Level 1 PSA, into a smaller number of plant damage states (PDSs) in accordance with accident characteristics and containment response characteristics for various accident sequences [10, 12, 13]. After any screening of low frequency PDSs, the progression of accidents and impacts on containment behaviour are examined probabilistically with event trees. The various end states of the event trees are grouped to a more manageable set of release categories for which distinct source terms are estimated. These distinct release categories define the conditions for estimation (in a Level 3 PSA) of conditional consequences. The product of each release category frequency and its conditional consequence, summed over all possible release categories, defines the risk of reactor accidents. It is important to note that the elements depicted in Fig. 1 are not unique and that they depend strongly on the approach selected to the Level 2 PSA.

1.2. OBJECTIVE

This Safety Practice is intended to assist technical persons managing or performing Level 2 PSAs. A particular aim is to promote a standard framework, terms and set of documents for PSAs to facilitate external peer review of their results. The procedures presented here are internationally recognized practices; however, it is not intended to pre-empt the use of new or alternative methods. On the contrary, the use of any method that achieves the objectives of PSA is encouraged.

The details of methods of analysis are subject to change with better understanding of severe accident phenomena. However, the framework outlined here is expected to apply for the foreseeable future.

1.3. SCOPE

This Safety Practice presents procedures for conducting Level 2 PSAs; that is, PSAs concerned with accident progression and phenomena leading to potential containment failure. The emphasis is on procedural steps of the PSA rather than on details of the modelling methods, since modelling is considered to be well documented in the relevant literature. Methods for determining the likelihood of containment failure and of the release of radionuclides to the atmosphere are also given.

Information in certain areas on non-procedural aspects of Level 2 PSAs is included as background information. The non-procedural aspects relate to light water reactors (LWRs). Differences between LWRs and other reactor types are not addressed.

1.4. STRUCTURE

Sections 2 to 7 correspond to the six major procedural steps for a Level 2 PSA (see Fig. 2). Section 2 is based on the procedures discussed in Safety Series No. 50-P-4 on Level 1 PSAs [1] and discusses the organization and management of a Level 2 PSA. This includes definition of the scope and objectives of a Level 2 PSA, project management, selection and organization of the PSA team, project scheduling, and procedures for quality assurance and peer review.

Section 3 briefly discusses various aspects of the plant and the containment that are important to progression and the mitigation of the consequences of severe accidents and releases of radionuclides.

The various tasks in developing the interface between Level 1 and Level 2 PSAs and grouping accident sequences are discussed in Section 4. Guidance is provided on the development of PDS bins, in consideration of the scope of Level 1 and Level 2 studies.

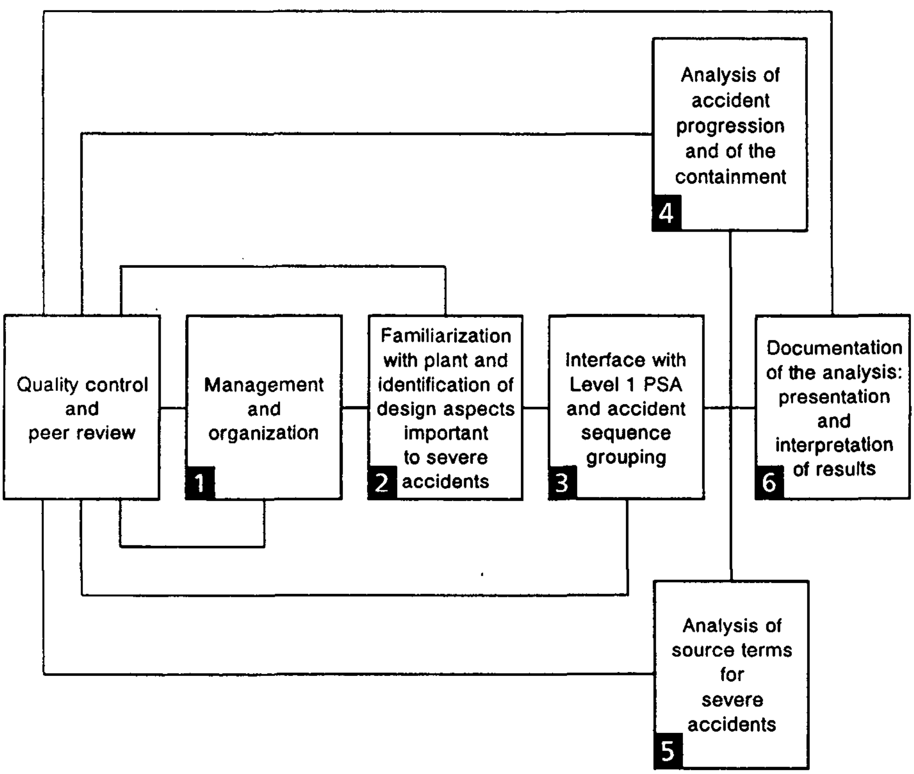

FIG. 2. Major procedural steps for a Level 2 PSA.

Section 5 discusses the procedural tasks in the performance of accident progression analysis and containment performance analysis. Probabilistic methods for the characterization of containment failure potentials and releases of radionuclides are included. Various uncertainty and sensitivity issues and methods for uncertainty or sensitivity analysis are also discussed.

Section 6 discusses the tasks in the evaluation of releases of radionuclides and transport attributes leading to the estimation of environmental source terms. The interface between Level 2 and Level 3 PSAs and accident progression groupings (source term bins) are also discussed. In addition, examples of various issues of uncertainties in source terms are given with procedures for their quantification and propagation through PSA models.

Information on the format and contents of a Level 2 PSA report is provided in Section 7. This is a modification for Level 2 PSAs of the documentation described in Ref. [1]. A sample outline of a Level 2 PSA document is also provided.

Appendix I gives an example of a typical schedule for a level 2 PSA. Various computer codes available for severe accident and PSA studies are discussed in Appendix II. Physical processes governing core melt progression, release of radionuclides, containment loading, and key uncertainty areas are discussed in Appendix III.

Two annexes give examples of PSA sequences.

2. MANAGEMENT AND ORGANIZATION

Detailed management and organizational aspects of PSAs, described in IAEA Safety Series No. 50-P-4 [1], are also applicable to the conduct of a Level 2 PSA and will not be repeated here. In Section 2, only those aspects that are more applicable to the conduct of a Level 2 PSA are discussed.

2.1. TASK 1: DEFINITION OF THE OBJECTIVES OF THE LEVEL 2 PSA

The general objectives, stages of the plant life-cycle, and scope of PSAs are described in detail in Safety Series No. 50-P-4 [1]. The scope of the Level 2 PSA is also determined by its intended use. Although the basic framework and methods of Level 2 PSA have been well established, the analysis in Level 2 PSA demands high levels of expertise and technical resources. Even when these high levels of resources are utilized, analyses of containment and radiological source terms are subject to large uncertainties in phenomena [8–11]. Therefore, it is important that PSAs be structured and conducted to support the intended end use.

Differing end uses place differing emphases and requirements on the various inputs and components of a Level 2 PSA. The proponent of a Level 2 PSA must therefore set out the requirements fully and must ensure that the user/recipient both understands these requirements and believes them to be realizable.

Some typical uses of Level 2 PSAs are:

— To gain insights into the progression of severe accidents and containment performance.
— To identify plant specific vulnerabilities of the containment to severe accidents.
— To provide a basis for the resolution of specific regulatory concerns.
— To provide a basis for the demonstration of conformance with quantitative safety criteria.
— To identify major containment failure modes and to estimate the corresponding releases of radionuclides.
— To provide a basis for the evaluation of off-site emergency planning strategies.
— To evaluate the impacts of various uncertainties, including assumptions relating to phenomena, systems and modelling.
— To provide a basis for the development of plant specific accident management strategies.
— To provide a basis for plant specific backfit analysis and evaluation of risk reduction options.
— To provide a basis for the prioritization of research activities for minimization of risk significant uncertainties.
— To provide a basis for a Level 3 PSA consistent with the PSA objectives.

Each of these examples would place differing emphasis on one or another aspect of the Level 2 PSA. Nevertheless, under all conditions, the PSA model needs to be as realistic as possible. Appropriate attention needs to be paid to the significance of governing uncertainties in phenomena. Care must be taken to avoid distorting the conclusions of the PSA through models and assumptions that are conservative and unrealistic.

It is likely that the Level 2 PSA will follow the completion of the Level 1 PSA. If the Level 1 PSA is not sufficiently comprehensive, complete and consistent, the deficiencies transmitted to a Level 2 PSA may lead to questionable conclusions.

2.2. TASK 2: DEFINITION OF THE SCOPE OF THE LEVEL 2 PSA

In undertaking a Level 2 PSA, there are two types of situation likely to be encountered. In the first, the Level 2 portion is part of an integrated full scope PSA analysis. In the second, the Level 2 PSA is seeking to extend an existing Level 1 PSA. In the former case, the requirements of the Level 2 analysis need to be fed into the Level 1 analysis so that all plant related features that are important to the analysis of the containment response and source terms are considered where possible in the Level 1 PSA. The PDSs which form the interface between the two analyses can then be defined according to the requirement of the Level 2 analysis.

If the starting point is an existing Level 1 analysis, then the output may not explicitly cover all the features that need to be accounted for in the Level 2 analysis. For instance, if the objective was the quantification of core damage frequency, then the status of the containment and the containment safeguard systems may not have been directly addressed and so these will have to be treated as part of the Level 2 analysis. In particular, if the scope of the PSA includes external hazards (e.g. earthquakes), then it may be necessary to consider dependent failures affecting containment function as part of the Level 2 analysis, if they have not been previously accounted for in the Level 1 output.

Finally, in determining the scope of a Level 2 PSA, the input requirements for a Level 3 PSA must be accommodated if one is contemplated. The ultimate product of a Level 2 PSA, then, is a description of a number of challenges to the containment, a description of the possible containment responses and their estimated probabilities, and an assessment of the consequent releases to the environment. This description will include the inventory of material released, its physical and chemical characteristics, and information on the time, energy, duration and location of the releases.

2.3. TASK 3: PROJECT MANAGEMENT

Information on the decisions that the PSA project managers must take and on the supervision, co-ordination and implementation of various tasks is provided in Ref. [1]. This information is also applicable to the Level 2 PSA and will not be repeated here.

The Level 2 PSA also requires coherent management to ensure that the PSA does indeed represent the actual plant under consideration and realistic operating practices. Also, and more importantly, it must be ensured that the insights gained are properly understood by the plant operating staff and management.

It must be the objective of the overall management to ensure a high level of interaction between the analysts and to ensure that, as insights are developed, the approaches to the different technical areas are modified as necessary. In this way a reasonable balance of effort across all topics can be achieved. The need to sustain good communication between the analysts during the entire PSA cannot be overemphasized.

2.4. TASK 4: TEAM SELECTION AND ORGANIZATION

The selection and organization of the Level 2 PSA team need to satisfy three requirements, namely: (1) knowledge of the design and operation of the plant;

(2) knowledge of severe accident phenomena and containment challenges; and (3) knowledge of PSA techniques. The team's expertise can vary in depth depending on the scope of the PSA, but the extensive participation of the plant engineers, utility personnel, and analysts of phenomena and probabilistic safety analysts is essential.

Ideally, the team comprises:

— *Systems analysts*: persons familiar with the Level 1 systems analysis part of the PSA, and the design of reactor coolant and containment systems, operational aspects and plant layout. Intimate familiarity with the Level 1 core damage sequences modelled, the success criteria, and interactions between containment and plant systems is required.
— *Operators and operational analysts*: persons familiar with design and operation of the plant and key containment systems. This includes intimate familiarity with the emergency operating procedures.
— *Specialists in phenomena*: persons familiar with severe accident phenomena, containment performance, severe accident uncertainty issues, chemical and physical processes governing accident progression, containment loads, releases of radionuclides and computer codes for the analysis of severe accidents.
— *Structural specialists*: persons familiar with containment structural design, capacity and failure modes.
— *PSA specialists*: persons familiar with event tree analysis, fault tree analysis, uncertainty analysis, statistical methods and PSA computer codes.

The organizational, training, funding and scheduling aspects are similar to those outlined in Ref. [1]. However, the conduct of the Level 2 PSA is more focused on the physical processes and probabilistic quantification of uncertainties in the progression of severe accidents. Operational expertise and systems analysis requirements are therefore mostly important for ensuring the correct interface with the Level 1 PSA and crediting the potential accident recovery measures.

The resources required for a Level 2 PSA are shown in Table I, which is based on recent experience. The lower estimate is representative of an experienced PSA team, while the upper estimate is for a relatively inexperienced PSA team, perhaps performing a PSA for the first time. Even for an inexperienced PSA team, some familiarity with PSA methods and severe accident issues is assumed. The ultimate resource needs will depend not only on the expertise of the team but also on the availability of the necessary information, methods and data, in particular if the PSA is for a different reactor design. The need for computing resources must also be taken into account.

The level of effort indicated in Table I for the performance of Level 2 PSAs is considerably lower than that suggested in Ref. [2]. This reflects an improved understanding of the various phenomena associated with core melt accidents, and a knowledge base expanded by the results of a large number of Level 2 PSA studies.

TABLE I. ESTIMATES OF HUMAN RESOURCES REQUIRED TO PERFORM A LEVEL 2 PSA[a]

Major steps in a Level 2 PSA	Human resources (person-months)
1. Management and organization	3–6
2. Plant familiarization and identification of design aspects important to severe accidents	2
3. Interface to Level 1 PSA and sequence grouping	4–6
4. Accident progression and containment analyses	10–24
5. Severe accident source terms	7–14
6. Documentation	4–8
Total resource requirements	30–60

[a] The scope is limited to the analysis of the reactor core for a full power operating state of the plant.

Also, extensive research on severe accidents has yielded experimental data and permitted computer code simulations of severe accident sequences and radiological source terms. PSA was primitive when Ref. [1] was published, and the level of effort suggested was based on the labour intensive WASH-1400 study [12].

The time required for the performance of the Level 2 PSA depends on the availability of Level 2 PSA tools (i.e. event tree codes, severe accident and source term codes, uncertainty analysis codes), the knowledge of the PSA team and the scope of the PSA study. For a given scope, even with unlimited personnel resources, some tasks have to be performed sequentially. Thus, there is a lower limit to the time necessary to complete the study. Table II provides lower and upper estimates of the time necessary to complete a Level 2 PSA, together with the corresponding resource composition.

The scheduling of the entire Level 2 PSA study is of paramount importance. The schedule covers:

— All tasks integral to the project (generally not broken down below tasks of one week in duration);
— The identification of individual(s) responsible for each task;
— The recognition of dependences between tasks, including the definition of interfaces and inputs/outputs among the tasks; and
— The expected duration of all tasks.

TABLE II. ESTIMATE OF TEAM COMPOSITIONS REQUIRED

Team member	Personnel required[a] (persons)	
	Short schedule (1½ years)	Long schedule (3 years)
Team leader	1	1
Systems analyst	1–2	1
Specialist in design and operations	2	1
Specialists in phenomena and severe accidents	3–4	3
Structural specialists	2	1–2
PSA methodologists/quantification specialists	2–3	2

[a] Some of these personnel may not be required for the duration of the study or may be involved in more than one task.

The schedule needs to be monitored and updated at monthly intervals. A typical schedule for a short schedule Level 2 PSA is given in Appendix I.

2.5. TASK 5: ESTABLISHMENT OF A QUALITY ASSURANCE PROGRAMME AND INTERACTIVE PEER REVIEW

As part of the management and organization of a Level 2 PSA, it is essential to establish a quality assurance programme and an interactive peer review process (see for example Ref. [14]). Detailed information on the establishment of a quality assurance programme and an interactive peer review procedure is provided in Safety Series No. 50-P-4 [1].

3. FAMILIARIZATION WITH THE PLANT AND IDENTIFICATION OF DESIGN ASPECTS IMPORTANT TO SEVERE ACCIDENTS

3.1. TASK 6: FAMILIARIZATION WITH THE PLANT

In this task the PSA team must identify and highlight component data, system data and operational data that may be of significance in assessing the progression of

TABLE III. EXAMPLES OF KEY PLANT AND/OR CONTAINMENT DESIGN FEATURES

Key plant and/or containment design feature	Comment
Reactor type	BWR/PWR/other
Power level	Actual thermal power
Fuel/cladding type and mix	Oxide, mixed oxide/Zr, etc.
Reactor coolant and moderator type	Water, heavy water, others
RCS coolant/moderator volume	As designed and fabricated
Accumulator volume and pressure set point	Actual operational values
Containment free volume	As built
Containment design pressure/temperature	As designed
Containment structure	Steel, concrete
Operating pressure/temperature	Actual operational values
Hydrogen control mechanisms	Inerted, ignitors, others
Mass of fuel	Actual operational values
Mass of cladding material	Actual operational values
Control rod type and mass	Actual operational values
RCS depressurization devices/procedures	Specify set point/procedures
Pressure relief capacity	Actual operational value
Suppression pool volume	Water and atmosphere volumes
Containment cooler capacity and setpoints	Actual operational values
Concrete aggregate	Specify chemical content
Cavity/keyway, pedestal design	Dispersive, non-dispersive
Flooding potential of cavity/pedestal	Flooded, dry
Sump(s), volume and location(s)	Specify details
Proximity of containment boundaries	Relative to reactor vessel
Venting procedure and vent location	Specify location/procedures
Containment geometry	Compartmentalization
External events impact	Seismic, flooding and impact
Potential for bypass	Penetrations/interfaces

severe accidents and the containment response. Plant features important to arresting the progression of severe accidents include systems such as fan coolers, containment sprays and suppression pools. This description is extended to the reactor and/or auxiliary buildings and secondary containments or other relevant structures and buildings. For existing plants, this may include a plant walk-through. Participation of operating staff and engineers is recommended.

3.2. TASK 7: IDENTIFICATION OF DESIGN ASPECTS IMPORTANT TO SEVERE ACCIDENTS

It is important to include all relevant design features as part of the Level 2 analysis. Unique plant features that can conceivably be used to circumvent potential severe accident issues that give rise to uncertainties also need to be clearly described. For instance, a sump under the reactor pressure vessel (RPV) could potentially affect the spreading behaviour of a mixture of molten fuel, cladding and structural material onto a containment floor, and thereby its ultimate thermal state. The dispersal of core debris following a high pressure melt-through of the RPV is a strong function of the dispersion path, geometry, degree of compartmentalization and the intervening structures. Similarly, mixing and distribution of combustible gases inside the containment atmosphere is a strong function of release location, containment geometry and compartmentalization.

TABLE IV. SAMPLE COMPARISON OF PLANT/CONTAINMENT DESIGN CHARACTERISTICS

Parameter and design feature	Significance/comparability
Reactor power/RCS volume ratio	Accident progression times, time for recovery actions
Reactor power/containment volume ratio	Scaling of containment loads
Zr mass/containment free volume ratio	Potential for combustion and scaling of containment loads
Under vessel to containment pathways	Potential for dispersion and high pressure melt ejection
Fuel and Zr mass/containment volume ratio	Scaling of containment loads
Concrete aggregate	Non-condensable gas generation and fission product release during molten core–containment interaction

Examples of key design features of the plant and containment that are significant to the progression and mitigation of severe accidents are listed in Table III. In addition to plant features, relevant operating procedures are also considered.

A comparison of key design features with those of plants similar in design and configuration, and for which a Level 2 PSA has been performed, is of great value in identifying the potential for similar vulnerabilities. Table IV lists examples of design features of the plant and containment for comparison with those of other plants.

4. INTERFACE WITH LEVEL 1 PSA: GROUPING OF SEQUENCES

The Level 1 PSA identifies a very large number of accident sequences (i.e. cut sets) which lead to potential core damage. It is neither practical nor necessary to treat each one of these individually when assessing accident progression, containment response and fission product release. The sequences are grouped together into PDSs so that all accidents within a given PDS can be treated as a group for the purposes of Level 2 assessment. Ideally this requirement will have been fed into the specification of the Level 1 PSA. It is recommended in Ref. [6] that this be done even if the extension of the Level 1 analysis to Level 2 is not planned at the time the Level 1 analysis is carried out. However, this will not always be the case. Section 4 outlines the approach to the definition of PDSs for PSAs that consider internal initiators at power. It then discusses the combining of these with Level 1 PSAs that have not defined PDSs for other initiators and other power states.

Examples of the definition of PDSs can be found in Ref. [12] and several more recent documents [2–4, 10]. Currently, there is no universal nomenclature for PDS descriptions, though most are derived in part from those used in the WASH-1400 study [12].

4.1. TASK 8: PLANT DAMAGE STATES FOR INTERNAL INITIATORS AT POWER

PDS group sequences that would be expected to have similar effects on containment response and fission product source terms. It is therefore important to identify those attributes of an accident progression that will influence either the containment response or the release of fission products to the environment. Broadly, PDSs can be grouped into two main classes: those in which radioactive materials are initially released to the containment and those in which the containment is either

TABLE V. EXAMPLE OF PLANT DAMAGE STATE ATTRIBUTES

Initiator type:

- Large LOCA
- Small LOCA
- Transients
- Bypass events:
  - Interfacing LOCAs
  - Steam generator tube rupture

RCS pressure at core damage:

- High
- Low

Status of the emergency core cooling system (timing of core damage):

- Fails in injection mode (early core damage)
- Fails in recirculation mode (late core damage)

Status of the containment's engineered safety features:

- Sprays (if any) operate at all times
- Sprays fail early (injection)/late (recirculation)
- Suppression pool (if any) effective at all times
- Suppression pool bypassed (early or late)
- Suppression pool ineffective
- Fan coolers (if any) operate at all times
- Fan coolers fail (early or late)
- Vented

Containment status:

- Isolated at core damage
- Not isolated at core damage
- Failed at core damage[a]

Status of reactor building and/or secondary containment:

- Isolated and effective
- Not isolated and ineffective
- Failed and ineffective[a]

[a] This includes any external events that may violate containment integrity.

bypassed or ineffective. Thus, the PDSs identify the containment status (e.g. intact and isolated, intact and not isolated, failed or bypassed) and, for bypass, the type and size of the bypass (e.g. interfacing system loss of coolant accident (LOCA), steam generator tube rupture). If the reactor building/secondary containment is likely to have a major influence on the source term, then the status of this is defined by the PDS. For PDSs in which the containment is intact, an accident progression event tree (APET) and/or containment event tree (CET) analysis will need to be performed. For other PDSs, only source term analysis is required. The following subsections give examples of the attributes that may need to be taken into account in defining these two classes of PDSs. Examples of these are given in Table V.

Plant damage states not initiated by bypass

In defining PDSs that are not initiated by bypass, account must be taken of the plant failures defined in the Level 1 analysis that could influence either the containment challenge or the release of fission products. This will include the type of initiator (large break LOCA, small break LOCA or transient), since this will affect the rate of discharge of fluid to the containment and the timing of the release of fission products. The timing of core melt will be affected by the mode of failure of the emergency core cooling system, since failure in the injection mode will lead to an early melt whereas failure during recirculation will lead to a late melt. The circuit pressure at vessel failure may influence the mode of vessel discharge and could challenge the containment if, for instance, high pressure melt ejection (HPME) and direct containment heating (DCH) ensue. The pressure at vessel failure will be influenced by the size of the initial breach in the circuit (i.e. the initiator type) as well as by the functionality of any depressurization system.

The status of the containment's engineered safety features is of vital importance in determining containment response. Consider, for example, a containment fitted with sprays and fan coolers. The status of the spray system and the timing of failure (if it occurs) will affect containment cooling, the removal of fission products and the availability of water for debris cooling. Similarly, the availability of fan coolers will be important from the point of view of containment cooling, the mixing of the combustible gases present and, to a lesser extent, the removal of fission products.

Other PDS attributes may be important in some applications of PSA. For instance, if the PSA is being used to help identify accident management measures, then it may be useful to identify electrical power status, since this information may be required for some later actions.

Plant damage states with containment bypass

For PDSs with containment bypass, the main consideration will be the identification of those attributes that influence the fission product source term. This will include the initiator type, the status of the emergency core cooling system (including failure time) and whether the leak is isolable after a period of time or if under water. In addition, for leaks into the auxiliary building, the status of emergency exhaust filtration systems with heating, ventilation and air conditioning and whether or not the leak is submerged could be significant. For sequences with steam generator tube rupture it may be necessary to distinguish between those sequences for which the affected steam generator dries out and those for which it still contains water when the main release of fission products occurs.

Final selection of plant damage states

If all possible combinations of parameters that affect the Level 2 analysis are combined, there are a large number of potential PDSs. In practice, these may have to be reduced to a manageable number. Two approaches can be used.

The first is to combine similar PDSs. For instance, sequences with the availability of sprays and fans might be grouped into a separate PDS from ones with sprays only. However, to first order they may be grouped together since the two main attributes of the sequences, which are containment cooling and fission product removal by sprays, are the same. The degree of cooling may be different, but that is a second order factor. Such groupings will allow a more manageable number of PDSs to be defined, but with some loss of detail. If the grouping is carried out by bounding one PDS with another PDS that can lead to higher consequences, this will introduce additional conservatism into the process.

The second approach is to use a frequency cut-off as a means of screening out less important PDSs. A careful screening is required prior to introducing a frequency cut-off criterion at the PDS level. This is especially the case when dealing with PDSs that could potentially involve large and early releases of radionuclides to the environment (for a large power reactor, releases of volatile species of iodine, and of caesium in excess of 5% of the core inventory, are typically considered significant for off-site consequences). Cliff edge effects must be avoided by carrying through the Level 2 analysis those PDSs with mean frequencies near the cut-off criterion, especially if PDSs could potentially lead to large releases of radionuclides to the environment.

The final stage is to define a representative sequence to characterize the PDS for the purpose of the Level 2 analysis. If the PDS grouping has been done correctly then the exact choice of sequence will not be critical. However, depending on the objectives of the PSA, one may choose to select a sequence which largely bounds the PDS group or one may select the highest frequency contributor to the group. In either case, it needs to be recognized that this grouping will introduce a degree of variability and hence some uncertainty into the analysis.

4.2. TASK 9: PLANT DAMAGE STATES FOR AN EXISTING LEVEL 1 PSA

A pre-existing Level 1 PSA may not explicitly present all the information required for Level 2 analysis to proceed. For instance, if the Level 1 objective was to estimate the core damage frequency, then the containment systems may not have been modelled. It is generally considered good practice to model plant related failures in the Level 1 rather than the Level 2 analysis.

In principle the additional plant items can be included by adding additional nodes or gates to the cut sets. The quantification will need to take account of dependences that may be carried through from the cut sets. For instance, for a core damage sequence due to random failure of the emergency core cooling system in the injection phase, the containment system's reliability may be independent of this failure. However, if the core melt was due to failures caused by loss of electrical power, then the containment systems would fail if they were dependent on the same power sources. Similarly if the emergency core cooling system failed on switchover to recirculation, it is likely that the sprays would also fail.

In principle, by systematically reviewing the dependences, the required PDSs can be constructed, but in practice the dependence information may not be readily available and so judgements may be needed.

4.3. TASK 10: EXTENSION TO OTHER INITIATORS

In principle the extension to other initiators, particularly internal and external hazards, could lead to the definition of a new set of distinct PDSs. In practice this may not be necessary. In many cases, the hazards simply cause dependent failures of plant items and so are treated using the same plant models as are used for internal initiators. They will therefore yield the same PDSs.

The area where there may be differences relates to direct containment damage. Events such as earthquakes or external missiles may lead to containment failure as well as core damage. It may be necessary to create additional PDSs to cover these, but it may be possible to use existing PDSs which represent isolation failure. It is not clear that a failure of the containment will be materially different from a major failure of containment isolation from the point of view of its effect on the release of fission products. However, it is up to the analyst to ensure that there are sufficient states defined to cope with the various degrees of leakage that need to be taken into account.

The only additional information that may need to be carried through is that which would affect actions to be modelled in either the Level 2 or Level 3 analysis. In particular, for sequences initiated by either seismic events or extreme weather, the modelling of countermeasures or dispersion in Level 3 may be affected.

4.4. TASK 11: EXTENSION TO OTHER POWER STATES

The extension of a PSA to states other than full power is a relatively recent development. This extension introduces requirements to determine whether additional PDSs may be required. Changes needed to a Level 1 analysis result from differences in the level of protection that are available at low power and shutdown, but for a Level 2 analysis the significant differences occur primarily as a result of differences in inventory, primary circuit and containment state.

Although accidents initiated at power states down to cold shutdown may develop more slowly because of reductions in decay heat or differences in initial pressure, these considerations are likely to be of secondary importance. The use of PDSs defined for full power may therefore be possible but the representation may be pessimistic. If, however, there are significant differences that are judged likely to have a major impact on behaviour in severe accidents, it may become necessary to define additional PDSs. Some examples include operation at mid-loop when the primary circuit inventory is low, or cases in which the primary circuit is open (e.g. during head removal or during refuelling) or the containment is not isolated (e.g. during some refuelling operations). Additional PDSs may therefore be required to allow specific analysis of such states.

5. ACCIDENT PROGRESSION AND CONTAINMENT ANALYSIS

5.1. TASK 12: CONTAINMENT PERFORMANCE

First it is necessary to provide a detailed description of the structural design of the containment, including containment type, design temperature, design pressure and an identification of containment penetrations (see Table VI). This is an important step towards determination of the structural performance of the containment under potential severe accident conditions. In this context the concern is with the performance of the containment as a leaktight barrier and so the analysis must include consideration of the liner and penetrations. For certain PDSs, the containment may already have failed or been bypassed. For instance, external hazards such as earthquakes may lead to containment failure, whilst interfacing system LOCAs or steam generator tube ruptures may lead to containment bypass. In these cases, in general, only accident progression and source term analysis are needed to determine the final outcome. This final outcome needs to take due account of the extent of damage to the containment, particularly if this has not been explicitly covered in the Level 1 analysis.

TABLE VI. STRUCTURAL PERFORMANCE AND DESIGN CONSIDERATIONS

Containment type	Steel
	Concrete:
	  Prestressed
	  Post-tensioned
	  Reinforced

Containment penetrations	Equipment hatch(es) Personnel hatch(es) Piping penetrations Electrical penetrations Purge line(s) Vent line(s)
Others	Transition from cylindrical shell to top head and basemat Layout and anchorage Liner walls and anchorage Interactions with other surrounding structures Shape effects (cylindrical, spherical, etc.)

This task is best performed by carrying out plant specific structural calculations; however, depending on the scope of the Level 2 study, use can be made of calculations for plants having similar containment designs that have been performed and published in the literature. In this case, the PSA must document a thorough justification of similarities of the designs and a demonstration of the applicability of the existing structural response analyses to the plant under consideration.

Two basic models have been used in PSA studies to characterize the loss of containment integrity; namely, the 'threshold model' and the 'leak before break model'. The threshold model defines a threshold pressure, with some associated uncertainties, at which the containment is expected to fail, with a large rupture, with the potential for significant and rapid blowdown of the containment atmosphere to the environment.

In the leak before break model, pertinent to liner tear and penetration failure, containment leakage is expected to precede major rupture. In general, leakage begins at pressures below the ultimate capability pressure and progressively increases up to the ultimate capability pressure, at which point a larger failure is expected to occur. Furthermore, if the rate of addition of mass and energy to the containment atmosphere is smaller than or equal to the out-leakage rate, containment pressurization is not expected and massive failure could be averted.

The containment integrity tests performed at Sandia National Laboratories, United States of America, using 1/32 scale and 1/8 scale steel and 1/6 scale reinforced concrete containment models for plants in the USA indicate that containment function is maintained well above the design basis limits for the containment structure established as part of the design basis accident envelope in the USA. The large scale steel model containment experienced a bursting type of failure at approximately five times its design pressure (tested with nitrogen at ambient temperature). However, the failure behaviour of the concrete containment model was observed to be by the development of large cracks in the concrete structure.

In general, if the containment's ultimate capability limit is reached very late into an accident sequence, the off-site consequences may not be appreciably different whether the containment fails by leakage mode or via a rupture mechanism. On the other hand, the mode of early containment failure could have a significant effect on ensuing consequences, and is therefore more important.

Containment performance analyses are based on validated structural models supported by data and reasonable failure criteria in order to assess the magnitudes of various loads to fail the containment, e.g. static loads, localized heat loads and localized dynamic pressure loads. The supporting analyses provides an engineering basis for containment failure mode, location, size and ultimate pressure/temperature capabilities.

While internal pressure loading is the principal determinant of potential containment failure, consideration is also given to the possible effects of temperature on structural performance of containment. Containment temperature could affect the strength characteristics of the structural materials as well as cause degradation of penetration seal materials. The potential temperature effects depend on the containment design. In general, large volume concrete containments are less susceptible to temperature effects than small volume steel structures.

The containment structural performance must also take account of uncertainties associated with estimating the structural capacities for withstanding extremes of pressure and/or temperature. These uncertainties can be determined by a sophisticated technique for uncertainty quantification and propagation, as part of the structural capacity assessment. Alternatively, expert judgement supported by simple analysis could be used to establish the failure pressure/temperature distribution for various credible failure modes (leaks and ruptures).

5.2. TASK 13: ANALYSIS OF THE PROGRESSION OF SEVERE ACCIDENTS

In this task, plant specific analyses of the progression of severe accidents are performed using any of the appropriate computer codes (see Appendix II). In addition, generic studies of severe accidents and containment loading reported in the

TABLE VII. EXAMPLES OF UNCERTAINTY AREAS RELEVANT TO ACCIDENT PROGRESSION

Severe accident issues	Related phenomena
In-vessel hydrogen generation	Blockage formation Cladding ballooning Recovery and water addition Molten fuel relocation
In-vessel natural circulation	Recirculating flows Hot leg, surge line and steam generator heat-up
In-vessel fuel–coolant interactions (energetic and non-energetic)	Recovery and water injection Recriticality Potential for energetics Radiological releases
RCS failure mechanisms	Location and mode of failure Local failures of head Gross failures of RPV
High pressure melt ejection/ direct containment heating	Debris trapping Zr oxidation/hydrogen generation Debris transport Hydrogen combustion Radiological release
Ex-vessel fuel–coolant interactions	Steam spike Steam explosion Radiological release
Core debris coolability and core concrete interactions	Steam pressurization Incondensable gas generation Debris spreading and potential interactions with containment shell Radiological release
Hydrogen combustion	Mixing/stratification Detonation Deflagration–detonation transition Deflagration Pressurization loads

literature for similar plants and containments could also be used as a basis for establishing an adequate framework for the APET/CET quantification.

In general, the uncertainties in the progression of severe accidents typically outweigh the differences that may be caused by plant specific design aspects. In most cases, plant specific design differences could be easily accommodated by appropriate scaling of reference plant analyses, on the basis of key design attributes of the type summarized under Task 7 for use in Level 2 PSA studies.

Known areas of uncertainty in phenomena are included, even though no general consensus may exist of their impact on pressure and temperature loads under various accident conditions. Examples of issues with potential implications for severe accident progression are listed in Table VII. These issues are discussed in detail in Appendix III.

Deterministic accident progression analyses could be performed for dominant (with respect to frequency) PDSs. In addition, deterministic calculations could also be performed for those PDSs that involve either direct containment bypass or early failure of the primary and/or secondary containments (i.e. large early releases). Ideally, the PDSs selected for deterministic analysis will reveal insights into a wide range of issues for phenomena in severe accidents.

The extent of code application depends on the objective of the PSA. The requirements for meaningful code use include that:

(1) most of the events and phenomena that may appear in the course of the accident be modelled;
(2) interactions between various physiochemical processes be correctly considered;
(3) computing time and resource requirements be reasonable.

The user must be aware of the limitations and weaknesses of the codes. Sensitivity analyses are performed to study the effects of changes in code parameters.

The key process variables (such as pressure, temperature, combustible gas generation, or the timing of major events) must be assessed and documented for use in the quantification of APETs or CETs. These process variables are displayed as a function of time for various positions of interest. The displayed results are also briefly described and any peculiarities clearly discussed. Sensitivity calculations are also performed for use in event tree quantification. These sensitivity analyses address both the modelling limitations of the code as well as the range of outcomes in terms of phenomena.

5.3. TASK 14: DEVELOPMENT AND QUANTIFICATION OF ACCIDENT PROGRESSION EVENT TREES OR CONTAINMENT EVENT TREES

In general in PSAs, the accident pathways that contribute to risk are described by two types of event trees. System event trees are used to define the spectrum of accident sequences (i.e., combination of accident initiators and subsequent system failures) that can lead to core damage. APETs or CETs are used to characterize the progression of severe accidents and containment failure modes that lead to releases of fission products beyond the containment boundary. They provide a structured approach for systematic evaluation of the capability of the containment to withstand severe accidents.

The term CET is adopted in most Level 2 PSAs, while APET was adopted in the NUREG-1150 study [10], made to provide information in considerable detail on the progression of accidents from initiation of damage to failure of the containment. These terms are used interchangeably in the present text.

In developing the APETs or CETs, important guidelines as set out in the following are followed.

APET/CET structure and nodal questions

These questions must address all of the relevant issues important to the progression of severe accidents, containment response, failure and source terms. The APET/CET structure must be logical, open to scrutiny, complete, consistent and to an appropriate level of detail, as mandated by the objectives of the Level 2 PSA. The APET/CET nodal questions must determine the likelihood of whether the containment is isolated, bypassed, failed, vented or intact. The APET/CET nodal questions are strongly specific to plant type. It is useful to divide the tree into time frames, delineated by the major events in the accident progression, e.g.:

(1) in-vessel processes during the early phase of damage progression;
(2) in-vessel processes during the late phase of damage progression;
(3) ex-vessel processes at or soon after vessel breach;
(4) long term processes following vessel breach.

In addition, to the extent possible, it is desirable to keep the number of nodal questions reasonably small and at a level consistent with the current understanding of severe accident phenomena. Very large and detailed event trees are difficult to quantify and scrutinize. Examples of a typical APET/CET structure and nodal questions for a typical pressurized water reactor (PWR) with a large, dry containment are provided in Table VIII.

Accident recovery/management actions

These actions must remain consistent between the Level 1 PSA and the APET/CET analyses. All recovery actions prior to initiation of core damage must only be credited in the Level 1 PSA, while any recovery actions beyond the initiation

TABLE VIII. EXAMPLES OF NODAL QUESTIONS FOR APETs/CETs FOR A PWR

	Top event question	Prior dependences	Question type
Very early time frame (early phase of damage progression)			
1.	Is containment isolated?	None	Based on PDS
2.	Fraction of PDS with AC power available?	None	Based on PDS
3.	What is the mechanical status of sprays in very early time frame?	None	Based on PDS
4.	What is the mechanical status of fans in very early time frame?	None	Based on PDS
5.	Does RCS depressurize manually in very early time frame?	2	Based on emergency operating procedures
6.	Does temperature induced hot leg failure occur in very early time frame?	5	Accident progression
7.	Does temperature induced steam generator tube rupture occur in very early time frame?	5, 6	Accident progression
8.	Is AC power restored or maintained in very early time frame?	2	Based on PDS
9.	Are sprays actuated in very early time frame?	3, 6, 8	Accident progression
10.	Does hydrogen combustion occur in very early time frame?	4, 5, 6, 8, 9	Accident progression
11.	Does containment fail in very early time frame?	1, 10	Accident progression
12.	Is containment isolation recovered in very early time frame?	1, 8	Based on PDS
13.	Is filtered vent system actuated in very early time frame?	1, 10, 11	Accident progression
Early time frame (late phase of damage progression including vessel breach)			
14.	Is core damage arrested in-vessel preventing vessel breach?	5, 6, 7, 8	Accident progression

TABLE VIII (cont.)

15.	Does energetic fuel coolant interaction occur and fail RPV and containment?	5, 6, 7, 14	Accident progression
16.	What is the mode of vessel breach and the core debris ejection process?	5, 6, 7, 14, 15	Accident progression
17.	Does vessel rocketing occur and fail containment?	16	Accident progression
18.	Is under-vessel region flooded or dry at vessel breach?	None	PDS and design
19.	What is the mode of under-vessel fuel-coolant interaction following vessel breach?	16, 18	Accident progression
20.	Does hydrogen combustion occur at vessel breach?	4, 8, 9, 10, 14, 16	Accident progression
21.	Does containment fail at vessel breach?	1, 11, 13, 15, 16, 19, 20	Accident progression
22.	Does filtered vent system actuate at vessel breach?	1, 11, 13, 15, 16, 19, 20, 21	Accident progression
Late time frame (long after vessel breach)			
23.	Is AC power restored or maintained in late time frame?	8	Based on PDS
24.	Do sprays actuate or continue to operate in late time frame?	23, 9	PDS/accident progression
25.	Do fan coolers actuate or continue to operate in late time frame?	4, 8	Based on PDS
26.	What is the status of fans and sprays in late time frame?	24, 25	Summary type question
27.	Is core debris in a coolable configuration ex-vessel?	16, 18, 19, 15, 17	Accident progression
28.	Does hydrogen combustion occur in late time frame?	10, 20, 26	Accident progression
29.	Does containment failure occur in late time frame?	1, 10, 11, 13, 15, 21, 26, 20, 28, 19	Accident progression

TABLE VIII. (cont.)

	Top event question	Prior dependences	Question type
30.	Does filter vent system actuate in late time frame?	1, 10, 11, 13, 15, 19, 20, 21, 26, 28, 27	Accident progression
31.	Is containment basemat integrity maintained?	11, 12, 21, 22, 27, 29, 31	Accident progression
32.	What is the mode of containment failure?	11, 21, 29	Accident progression

of core damage (accident management actions post core damage) could be credited as part of the APETs/CETs, provided that the following guidelines are followed:

(1) The recovery actions are included as part of emergency operating procedures for the plant under consideration. The APET/CET quantification is based on a realistic human reliability analysis, thus providing adequate bases for selection of the branch probability estimates.
(2) The effect of the environmental conditions resulting from a severe accident on the survivability of active components must also be considered. For instance, recovery of power does not necessarily ensure the recovery of pumps, even though the initiating event may have been caused by loss of power. This is because the pumps in question could have been rendered inoperable as a result of flooding, excessive aerosol loading and/or the effects of a severe radiation environment beyond the equipment qualification limits in the original design basis.
(3) Potential adverse effects of recovery must also be considered as part of the event tree quantification. For instance, injection of water into a degraded core may be able to arrest the progression of a severe accident; however, there is also the potential for an energetic fuel-coolant interaction, fuel shattering and additional releases of steam, hydrogen and fission products.

APET/CET quantification process

The quantification of the conditional probabilities for the branch point must be supported by documented analyses and recent data, including considerations of uncertainty issues for severe accidents.

The assessment of an issue can sometimes be made more traceable by decomposing the problem into a number of subissues according to the governing phenomena [9–11, 13]. An example is late hydrogen combustion. In this case, the phenomenon may be decomposed into the following subissues:

— the extent of in-vessel oxidation of Zr;
— the extent of ex-vessel oxidation of Zr;
— prior combustion events;
— concentration of combustible hydrogen to support hydrogen deflagration or detonation;
— the extent of mixing in the containment atmosphere; and
— the presence of an ignition source.

Another example is a complex in-vessel steam explosion giving rise to a potential for alpha mode containment failure. The quantification in Ref. [9] is via an extensive logic tree derived from issue decomposition. Other examples of decomposition are provided in the supporting documents for NUREG-1150 [10, 15]. Generally,

TABLE IX. TYPICAL CONTAINMENT FAILURE MODES AND MECHANISMS

Mode of failure	Mechanism for failure
Direct bypass	Interfacing systems LOCA Steam generator tube ruptures Externally initiated
Isolation failure	System failure Operating mode
Vapour explosion	Rapid pressurization Blast loads Missile generation
Overpressurization	Steam spike Gradual boil-off Incondensable gases Direct energy transfer
Underpressure	Inappropriate recovery of isolation failure Inappropriate operation of filtered vent
Overtemperature	Core–concrete interactions Direct contact by core debris Thermal attack of penetrations
Combustion	Detonation Deflagration to detonation transition Deflagration
Concrete penetration	Basemat penetration Pedestal/support failure
Other	Vessel thrust forces Pipe whip Random failure of RPV

decomposition is not carried out explicitly in an APET/CET for practical reasons. However, this can be achieved by using nodal event trees, fault trees [6] or other methods. In addition, the APET/CET end states appropriately distinguish the temporal dependence of containment failure modes.

Bins for releases of radionuclides

These will be consistent with the plant and containment system design under consideration, and they include relevant containment failure modes and mechanisms (see Table IX). The release bins (also referred to as release categories or source term bins) must clearly distinguish the key release characteristics, consistent with the PDS definitions. These include the impact of recovery actions, accident management and emergency operating actions (as these may affect the releases of radionuclides). Task 15 provides a detailed description of the APET/CET end state (source term) binning requirements.

Containment performance is characterized for each of the APET/CET end states on the basis of assessment of plant and containment response, carried out under Task 12. The containment boundary is defined as any interface with a possible pathway to the environment (e.g., primary to secondary coolant systems in PWRs, steel containment in boiling water reactors (BWRs) and some PWRs).

In a Level 1 PSA, systems analysis techniques are used to define accident sequences and to arrive at their frequencies. Accident sequences are defined in terms of initiating events and a series of system functional failures. Frequencies of accident sequences are quantified by multiplying the estimate of initiating event frequency by algebraic expressions consisting of combinations of system or component unavailabilities and human error probabilities, which are effectively physical probabilities.

In a Level 2 PSA, given a core damage scenario, APETs/CETs are used in modelling the severe accident progression and containment response. Each branch point corresponds either to the availability of some containment system function or to the likelihood of occurrence of some physical phenomenon. A containment/accident progression event tree can therefore be used to define sequences, such as those in the Level 1 PSA, and serves to link core damage sequences with radiological source terms.

However, lack of knowledge about issues that are invariant with respect to repeated occurrences of the core damage sequences (or, for simplicity, are taken to be invariant) necessitates the consideration of alternative event trees. These may differ only in the numerical values of their split fractions or may even have a different structure [16]. Numerical values of the analysts' degrees of belief (subjective probabilities) are associated with the alternative split fraction values and the event tree structures.

The determination of conditional probabilities is based on deterministic analyses (if the phenomenon at hand lends itself to some form of engineering analysis) and expert judgement. The quality of this expert judgement is dependent on the analyst's current state of knowledge pertinent to a particular issue. Ideally, key sources of up to date information would be available to support the nodal probability assignments and, typically, these include:

— deterministic analyses using the principal severe accident codes or basic principles;
— other PSA studies for similar plants, such as the NUREG-1150 study [10];
— relevant experiments;
— reviews and analyses.

The point to be made here is that there is generally available a good body of information, from recent studies of various severe accidents, to supplement the analyst's own analyses. Incorporation of such information would enhance the quality of the technical assessment. There is at present no standard protocol for the use of expert judgement in a PSA process. The general approach of most PSAs is to rely almost entirely on the engineering judgement of the analysts, perhaps with some limited interaction with recognized experts. An interesting aspect of the NUREG-1150 [10] study was the elicitation of expert opinion on issues for which the uncertainty was judged to be greatest [10]. This formed an integral part of the uncertainty analysis for NUREG-1150. The potential use of expert judgement in a Level 2 PSA is generally recognized, but it is an area needing further development in its practical implementation.

Assignment of subjective probabilities of occurrence to various events and phenomena is itself controversial; nevertheless, in order to assess the uncertainties in the progression of severe accidents, some reliance on numerical assignment of subjective probabilities is indispensable. The following provides an example of probability values that could be assigned to various subjective descriptors:

Subjective descriptor	*Probability*
Certain	1.0
Likely to very likely	>0.5–<1.0
Indeterminate, ambiguous	0.5
Very unlikely (edge of spectrum) to unlikely	0.01–<0.5
Extremely unlikely (physically unreasonable)	0.001
Impossible	0.0

Threshold approach

In the threshold approach the threshold to failure is reached when the pressure, temperature, etc. reach and/or exceed the failure limits. The failure probability is, therefore, a function of how close the parameter is to the failure threshold.

Example: Failure of the reactor coolant system induced by natural circulation

Step 1. Perform calculations for a high pressure accident sequence using a code or model that treats this phenomenon (MAAP, SCDAP/RELAP5, etc.). Determine the structural temperature distributions around the reactor coolant system (RCS) prior to bottom head failure of the RPV.
Step 2. On the basis of these calculations, and the body of evidence that exists in the literature, determine the likelihood of creep rupture failure by comparing the temperatures at the hot leg nozzle, surge line and steam generator tube ends, near the tube sheets with the yield point. Allow for potential degradation in steam generator tubes during normal operation.
Step 3. If calculated structural temperature is much less than the yield point at a given pressure, then the probability of failure at that location is closer to the edge of the probability spectrum; thus, a value close to 0.01 could be assigned (the actual numerical assignment is a 'judgement call'). If the calculated structural temperature is near the yield point, then the assignment of probability reflects the indeterminate nature of the outcome; thus a probability value of 0.50 is more appropriate. If the structural temperature is greater than the yield point, then the probability of failure is closer to certainty; thus a probability value of ≈ 0.9 is appropriate.

The assignment of numerical values is, therefore, indicative of the analyst's judgement of and belief in the acceptability of the deterministic predictions of uncertain phenomena. Furthermore, by constructing APETs/CETs that are not excessively detailed with regard to the finest aspects of phenomena in severe accidents, the need for a high degree of precision in quantification of the probabilities can be reduced.

Integral approach

In the integral approach, both the quantity of interest (pressure, temperature, etc.) and the failure criteria (failure pressure, failure temperature, etc.) are treated as uncertain parameters. Probability density functions representing uncertainty distributions are arrived at on the basis of deterministic analyses and expert judgement, and the overlap/interference of these two uncertainty distributions determines the degree of belief (subjective probability) for failure.

Example: Containment failure induced by direct containment heating due to high pressure melt ejection

Step 1. Perform parametric HPME and/or DCH calculations to arrive at containment pressure at vessel breach.

Step 2. Determine the reasonable upper bound pressurization (based on parametric assumptions) and assume that it corresponds to the 90th or 95th percentile of the distribution.

Step 3. Determine the reasonable lower bound pressurization and assume that it corresponds to the 5th or 10th percentile of the distribution.

Step 4. Construct an uncertainty distribution for pressurization loads inside containment due to HPME and/or DCH. Note that the distributional shape and/or type is also judgemental and depends strongly on the extent of information available to the analysts. However, if needed, the sensitivity of the calculated results to the shape and/or type of distributions can also be assessed.

Step 5. On the basis of analyses of the structural performance of the containment, determine the uncertainty distribution for containment structural failure pressure (Task 12).

Step 6. These two uncertainty distributions (see Fig. 3) determine the degree of belief (subjective probability) for failure under the conditions for which the 'state of knowledge' uncertainties are constructed. Quantitatively this is given by the following stress strength interference integral:

$$\text{Conditional probability of failure} = \int_0^{\infty} P_r(P_c = p) \left\{ \int_0^{p} P_r(P_f = p')dp' \right\} dp \quad (1)$$

where P_r is the probability density function, P_c is the containment peak pressure and P_f is the containment failure pressure. The expression inside the curly brackets is the cumulative probability distribution $C_f(P_f < p)$ for total containment failure. A detailed discussion of the formalism of addressing uncertainties is provided in Ref. [17].

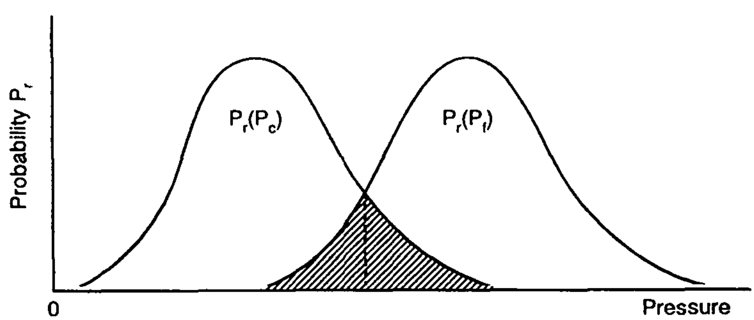

FIG. 3. Probability density functions for containment peak failure P_c and containment failure pressure P_f (failure indicated by overlap).

5.4. TASK 15: BINNING OF EVENT TREE END STATES INTO RELEASE CATEGORIES/BINS

The APETs/CETs will provide the conditional probability that a containment failure mode can be realized, given a PDS. The APETs/CETs produce a large number of end states, some of which are either identical or similar, in terms of key release (and ex-plant consequence) attributes. These end states are often grouped together. In some studies (e.g. NUREG-1150 [10]) this has been carried out using a two stage process in that a number of intermediate bins have been defined for source term analysis. These source term groups are then further grouped to define a smaller number of release categories.

These groupings are often referred to as release categories, release bins or source term bins. The release bins group APET/CET end states that would be expected to have similar radiological characteristics and potential off-site consequences. Although the same result could be obtained by evaluating off-site consequences for each APET/CET end state, the release bins reflect explicit consideration of APET/CET end states that could produce unique effects on the magnitudes of off-site consequences.

It is important that the source term/release groups or bins are defined on the basis of appropriate attributes that affect fission product releases and accident consequences. These attributes are specific to the plant and containment type, and there is no unique way to perform this task; however, Table X provides a list of important binning attributes for PWRs and BWRs. Examples of binning schemes can be found in several recent PSA studies, including those performed as part of NUREG-1150 [10] in the USA.

In defining the attributes of the release categories, attention is paid to the requirements of Level 3 PSA if the analysis is ever to be extended to this. The interface between Level 2 PSA and Level 3 PSA and the exchange of information will determine to a large extent the quality of the Level 3 PSA results and will therefore be important.

The source term information that the Level 3 PSA requires for each release category covers:

(1) *The radionuclides* (see also Section 6.1, Task 18) including also the chemical forms of each radionuclide.
(2) *The frequency* (or frequency distribution) of each release category.
(3) *The amount of radionuclides* released as a function of time, expressed as fractions of the initial core inventory for each group of radionuclides having similar physical and chemical characteristics, i.e. usually having similar volatility. The duration of the release is assigned such that no significant additional release will occur after the considered time interval. If the release fractions need to be extrapolated beyond the calculated time window, the basis for such

TABLE X. EXAMPLE OF BINNING ATTRIBUTES FOR APET/CET END STATES

Release attributes	Variations
Timing of release	Very early (containment failure prior to core damage) Early Intermediate Late
Containment bypass/isolation	Interfacing LOCAs Steam generator tube ruptures Other initiators
Mode/mechanisms of release	Design basis accident leakage Beyond design basis accident leakage Rupture Basemat penetration
Active fission product removal mechanisms	Sprays Fan coolers Suppression pools Overlying water pools Ice beds Filtered vents Others
Passive fission product removal mechanisms (release pathways)	Secondary containments Reactor buildings Tortuous pathways
Location of release	Ground level Elevated
Energy of release	Low High and energetic
Duration of release	Rapid Protracted

an extrapolation needs to be justified. In addition, any major change in the release fractions needs to be explained.

(4) *The time of the release* which is relative to reactor shutdown.

(5) *The warning time* for implementation of appropriate countermeasures, defined as time from accident initiation to the actual occurrence of a release.

(6) *The location of the release* relative to ground level (ground level release or elevated release).

(7) *The energy content of the release* which is a function of containment temperature and pressure prior to failure. This is important in determination of the potential for plume rise.

(8) *The particle size distribution* of released aerosols, which will affect the deposition during plume transport.

In principle, the constituent radionuclides of the release, its energy content and the particle size distribution are also given as a function of time for the duration of the release. Note that the information given by items (3) to (5) are reflected in the binning attributes shown in Table X. On the basis of the attributes 'containment bypass/isolation', 'mode/mechanisms of release', 'active fission product removal mechanisms', and 'passive fission product removal mechanisms', different magnitudes of release can be established, e.g. small, intermediate and large releases. Generally, a more detailed and precise distinction will be necessary, depending on the requirements determined by the Level 3 PSA analysis. This applies also to the attributes 'timing of release' and 'duration of release'. In addition, some attributes are interdependent, e.g. an early large release will most probably result from containment failure due to rapid pressurization and is then highly energetic.

Since the grouping of the APET/CET end states into release categories could affect the results of a subsequent Level 3 PSA, the rationale for a particular grouping scheme must be thoroughly discussed and documented. In addition, it must be shown that the selected release categories adequately represent the full spectrum of possible releases. In some cases, iterations between the Level 2 and Level 3 analyses may become necessary to ensure that important attributes are correctly selected.

5.5. TASK 16: TREATMENT OF UNCERTAINTIES IN ACCIDENT PROGRESSION

Uncertainties or variability are going to arise in the Level 2 PSA analysis as a result of several factors, including:

(1) *Completeness*. The main thrust of the PSA model is to assess the possible scenarios (sequences of events) that can lead to releases of radionuclides. However, there is no guarantee that this process can ever be complete and that all possible scenarios have been identified and properly assessed. This lack of

completeness introduces an uncertainty in the results and conclusions of the analysis that is difficult to assess or quantify [18]. However, extensive peer review can reduce this type of uncertainty.

(2) *Modelling adequacy.* Within the Level 2 analysis, uncertainties in the modelling are due either to incomplete knowledge of the phenomena or to inadequacies or simplifications introduced into the modelling of the phenomena. These uncertainties are discussed as part of the uncertainty treatment in the Level 2 PSA (see Tasks 16 and 23).

(3) *Uncertainties in input parameters.* The parameters of the various models used in the PSA are not exactly known because of scarcity or lack of data, and the models will include assumptions made by experts. The grouping of level 1 sequences into PDSs as the input to the Level 2 will introduce uncertainties in addition to those associated with the specific Level 2 PSA input parameters.

Level 2 analysis is in some sense entirely concerned with the treatment of uncertainties since the analyst, in putting probabilities on the various possible outcomes of an accident progression, is stating a degree of belief in the possible outcomes given the uncertainties involved. In the extreme, given no uncertainty there may be only one outcome.

The task that the analyst faces is to make use of all the information available in a structured fashion to arrive at a conclusion about both the relative probabilities of accident progression sequences and the radiological source terms associated with them. Inevitably this will require the use of expert judgement and the methods selected must allow this to be included in an auditable fashion. Note that besides the explicit use of expert opinion for treating uncertainties, much expert judgement is implicitly present in the rest of the analysis, e.g. when analysts make decisions about the problem definitions, boundary conditions and screening criteria.

Examples of areas of uncertainty relevant to accident progression are presented in Table VII and in Table XXV (in Appendix III). These uncertainties primarily result from incomplete knowledge of the governing phenomena. In addition, Table XXV in Appendix III provides a qualitative appreciation of the levels of uncertainty and sensitivity for the identified areas.

There is no universally accepted approach to uncertainty analysis. A general framework for uncertainty analysis is given in Ref. [11]. Other very similar but more elaborate approaches are given in Refs [8–10]. In general, uncertainty analysis can be divided into the following three principal steps:

(1) *Definition of the scope of the uncertainty analysis.* In deciding the method of uncertainty analysis for the Level 2 PSA, the nature of uncertainties in the accident progression, containment and source term analysis must be considered. In addition, depending on the requirements for the uncertainty analysis in the overall PSA, the choice of method will also depend on the need to achieve compatibility with the other components of a PSA. Because the number of

uncertainties is large and the resources may be limited, it is important to make a selection of the issues to be included [11].

(2) *Characterization/evaluation of each uncertainty issue.* For the APET/CET uncertainty analysis, the issues may be derived from those which have been identified as a result of some review process undertaken by experts. Some suggestions are provided in Refs [3, 10, 11]. For the accident progression analysis, containment analysis and source term analysis, the selection of issues is mainly achieved by sensitivity analysis, but also by the analyst's judgement.

The purpose of sensitivity analysis is to determine the degree of change in the key figures of merit due to uncertain data or parameters, and to address those modelling assumptions of potential significance for the results. Sometimes issues may also be selected from recommendations given in the computer code users' manuals.

The format and range of the uncertain parameters of each issue must either be defined by probability distributions or, more simply, represented by sensitivity bounds. The format to be adopted is largely dictated by the nature of the subsequent uncertainty treatment. The use of probability distributions [8–10], representing a mathematically more rigorous approach, will facilitate the propagation of uncertainties in the PSA models. Whichever format is adopted, the judgement in the formulation of uncertainties/sensitivities are supported by data, analyses and consideration of the published literature. In addition, it is necessary for the uncertainty distributions to be thoroughly peer reviewed as part of the PSA study.

The propagation of uncertainties may be done by an appropriate technique suitable for the overall PSA model. Examples of available propagation techniques include: (a) use of discrete probability distributions; (b) direct simulation methods based on either simple (Monte Carlo) random sampling or stratified (Latin hypercube) sampling procedures. Additional details can be found in Refs [8–10].

(3) *Display and interpretation of the results.* The results of the uncertainty analysis task are carefully displayed to strengthen the Level 2 conclusions. In recent PSAs that have included uncertainties, the results have been displayed using histograms, probability density functions, cumulative distribution functions and tabular formats showing the various quantiles of the calculated uncertainties, together with the distributional mean and median estimates [8–11]. In contrast, if a sensitivity analysis approach is used in assessing the variability in the Level 2 PSA results, then the sensitivity of key accident progression signatures to alternatives in numerical assignments is displayed. The displayed uncertainties and/or sensitivities are discussed in sufficient detail to enable the end user easily to interpret the resulting insights and conclusions.

5.6. TASK 17: SUMMARY AND INTERPRETATION OF CONTAINMENT PERFORMANCE RESULTS

In this task the insights gained from APET/CET quantification are summarized and discussed. Tabulation of the APETs/CETs in a form of a containment matrix (C-matrix) is very useful in showing the main failure modes and/or release categories for each key PDS. The C-matrix shown in Table XI provides the conditional probability C(m, n) that a release bin n can be realized, given a PDS m. It is of the same nature as, for instance, the S-matrix for Level 1 PSA, which provides the conditional probability S(k, m) that PDS m can be realized, given initiating event k. Uncertainty analysis leads to alternative values of the elements of the C-matrix just as it does for the elements of the S-matrix for Level 1 PSA. Subjective probabilities are associated with alternative values.

A discussion is provided that identifies the major contributors to early (including bypass and unisolated containment events) and late containment failure modes.

By combining the frequencies of PDSs and their associated uncertainties resulting from the Level 1 PSA with the conditional probabilities of various failure/release modes (and their associated uncertainties resulting from APET/CET quantification), the frequencies and uncertainties of each release bin/failure mode can be determined. In other words, the release frequency for each release bin is given by:

$$R(n) = \sum_{m=1}^{M} F(m)\ C(m,n) \tag{2}$$

TABLE XI. CONTAINMENT MATRIX ELEMENTS (C-MATRIX)

PDS	Release bin							PDS frequency
	1	2	..	n	..	..	N	(F)
1	C(1,1)	C(1,2)	..	C(1,n)	..	..	C(1,N)	F(1)
2	C(2,1)	C(2,2)	..	C(2,n)	..	..	..	F(2)
3	C(3,1)	C(3,2)	..	C(3,n)	..	..	..	F(3)
..	..	..	..	..	..	..	..	..
m	C(m,1)	C(m,2)	..	C(m,n)	..	..	C(m,N)	F(m)
..	..	..	..	..	..	..	..	..
M	C(M,1)	C(M,2)	..	C(M,n)	..	..	C(M,N)	F(M)
Bin frequency	R(1)	R(2)	..	R(n)	..	..	R(N)	

where:

R(n) is the release frequency for bin n (per reactor year);
F(m) is the frequency of PDS m (per reactor year);
C(m,n) is the conditional probability of release bin n, given PDS m.

The contribution of each release bin to total release frequency

$$X(n) = \frac{R(n)}{\sum_{n=1}^{N} R(n)} \tag{3}$$

is also tabulated for identification of major contributors to total release frequency.

Generally, for each of the selected release categories, one representative PDS is selected for which a source term is estimated based on the available results from other PSAs, or plant specific calculations using an appropriate computer code for source terms for severe accidents (e.g. MELCOR, STCP, MAAP) as discussed in Section 6 and Appendix II. The selection of the representative PDS is governed by its frequency and consequence dominance within the release category. Alternatively, source terms can be estimated for each and every PDS contributing to a particular release category/bin. In addition, for those release categories that result from potentially uncertain mechanisms (e.g. steam explosion, DCH) for which trustworthy models are not readily available, code calculations could be augmented by simple analyses and expert judgement.

6. SOURCE TERMS FOR SEVERE ACCIDENTS

Many characteristics and phenomena of plant systems and containment systems have been shown to influence the magnitude and characteristics of source terms for severe accidents, as discussed in the previous section. These include fuel and control assembly design; core power density and distribution; metallic contents; RCS geometry and plant configuration; RCS pressure; availability of cooling water; depth of ex-vessel core debris, composition, initial temperature and concrete aggregate; design of containment heat removal system (suppression pool, fans, sprays, ice condensers, etc.) and geometric configuration and leak and/or rupture pathways for the containment and secondary system.

In general, it is recommended to make several plant specific source term calculations with any appropriate computer code (at least for high frequency release categories and those categories expected to include relatively large releases, i.e.

early containment failure and bypass sequences) (see Appendix II). Nevertheless, examples of recent analyses performed for typical LWRs designed in the USA are also provided. These could be used in the Level 2 studies, provided that Task 7 has not revealed major design and operational differences which could potentially alter the release and evolution of radionuclides during severe accidents. Here again, uncertainties in the calculated source terms will most likely overshadow any variations resulting from design specific and plant specific differences.

TABLE XII. THERMODYNAMIC PROPERTIES OF SOME FISSION PRODUCTS

Element/compound	Melting point (°C)	Boiling point (°C)
Noble gases		
Xe	−111.6	
Kr	−156.7	
Volatiles		
I	113	185
HI	−51	−35
Cs	29	690
CsI	626	1280
Te	450	990
Semivolatiles		
Sr	800	1384
Ba	850	1638
Sb	630	1380
Refractories		
SrO	2430	3249
La	920	3469
Ru	2250	4150
Fuel		
UO_2 [a]	2840	3293

[a] Oxidized Zr forms a eutectic with UO_2 at about 1900°C.

6.1. TASK 18: GROUPING OF FISSION PRODUCTS

The thermodynamic properties listed in Table XII provide some indication of the relative volatility of various core materials [4, 19]. Therefore, fission products are grouped in accordance with their common chemical and physical characteristics. Various group structures have been proposed to date, ranging from very coarse to very fine groupings. A reasonable group structure for fission products might include:

Group	*Species*
Xe	Xe, Kr
I	I, Br
Cs	Cs, Rb
Te	Te, Sb, Se
Ba	Ba, Sr
Ru	Ru, Rh, Pd, Mo, Tc
La	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
Ce	Ce, Pu, Np

In addition, the following volatile chemical forms are also suggested:

Species	*Chemical forms*
I(g)	I_2, CH_3I, HI
I(aerosol)	CsI
Cs	CsOH, CsI

Other chemical forms can also be postulated during severe accidents; however, for the source terms it suffices to recognize the lack of retention of the gaseous components of some of the radionuclides, most notably the radioiodines.

6.2. TASK 19: RELEASE OF FISSION PRODUCTS FROM FUEL DURING THE IN-VESSEL PHASE

Small quantities of fission products created inside the fuel matrix are released from the fuel pellets during normal operation. These fission products will reside in the gap between the fuel pellets and the cladding. Table XIII lists examples of conservative estimates of gap inventories during normal operation of oxide fuelled LWRs [20].

In severe accidents, additional fission products are released by vaporization or some other thermally activated process resulting from the heating up of fuel and of control and structural material inside the reactor core.

TABLE XIII. EXAMPLES OF GAP INVENTORIES OF FISSION PRODUCTS IN LWRs

Species	Percentage of initial core inventory
Xe, Kr	3–10
Cs, I	2–5
Others	Insignificant

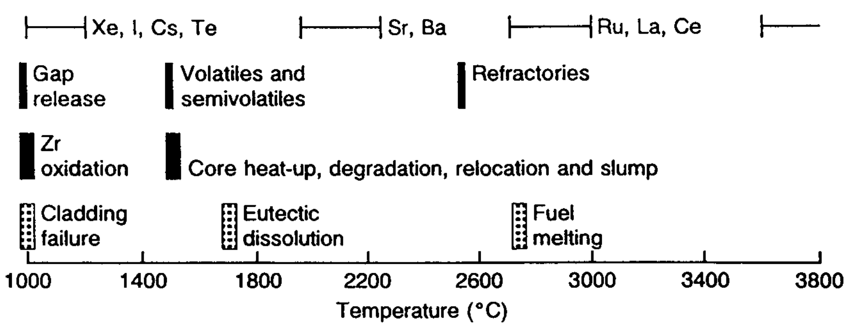

FIG. 4. Qualitative sequence of events in melt progression and release of fission products.

Figure 4 provides a qualitative perspective for sequence of events during core heat-up, meltdown and fission product release in LWRs.

During fuel heat-up, degradation and meltdown, several factors will influence the release of volatile fission products. The most important factors are the maximum temperature reached in the fuel and the time at which that temperature is reached. A third factor is the composition and flow rate of the steam/hydrogen mixture through the fuel matrix. The ratio of steam (an oxidizing agent) and hydrogen (a reducing agent) governs the effective oxidation potential which in turn can alter the chemical forms of the released species. The partial pressure of steam will also affect the volatility of some material.

The Zr cladding is also significant in the release of iodine, caesium and tellurium. Removal of Zr, either through oxidation or through reactions with the UO_2 fuel, can increase the release of volatiles, in particular Te.

TABLE XIV. EXAMPLES OF FRACTIONAL RELEASES OF RADIONUCLIDES FROM FUEL DUE TO CORE DEGRADATION LEADING TO VESSEL BREACH

Group	Low system pressure	High system pressure
Xe	0.95	0.95
I, Cs	0.95	0.95
Te	0.60[a]	0.40[b]
Sr	0.001	0.001
Ba	0.02	0.02
Ru–La	≤0.001	≤0.001

[a] High Zr oxidation fractions.
[b] Low Zr oxidation fractions.

The release of fission products cannot be modelled by first principles, and empirical approaches are usually relied on for PSA applications. A number of computer codes are available that model the progression of severe accident source terms (see Appendix II for details).

Extensive calculations have been performed in support of several PSA studies in the USA and elsewhere. Differences, sometimes substantial ones, have been observed in the prediction of in-vessel releases of radionuclides, even for nearly identical accident sequences [20–22]; nevertheless, for application to PSAs, some general trends in estimation of releases are emerging. Calculations performed [22] using the source term code package (STCP) and the newer MELCOR code can be put into an approximate and general framework.

Table XIV shows examples of fractional releases of fission products from fuel (in-vessel) for LWRs. These release fractions show a near complete release of I and Cs, while release of Te is strongly controlled by the extent of Zr oxidation.

As mentioned earlier, to the extent possible, plant specific source term calculations are recommended. Considerable insights are gained from these plant specific analyses; however, these calculations require substantial technical expertise and resources.

6.3. TASK 20: RETENTION OF FISSION PRODUCTS WITHIN REACTOR COOLANT SYSTEMS

On the basis of the discussion of Section 6.2, fission products, following their release from fuel, are carried along with the flow of steam and hydrogen, both as

TABLE XV. EXAMPLES OF FRACTIONAL RETENTION OF RADIONUCLIDES INSIDE THE REACTOR COOLANT SYSTEM PRIOR TO VESSEL BREACH

Fission product group	PWR, system pressure low	PWR, system pressure high	BWR, system pressure low	BWR, system pressure high
Xe	0.0	0.0	0.0	0.0
I, Cs	0.15	0.70	0.15	0.70
Te	0.20	0.50	0.50	0.60
Others	0.20	0.50	0.50	0.60

vapours and as aerosols. Fission product vapours can condense on cooler surfaces as well as on other aerosol particles during their passage through the RCS into the containment. Fission product aerosols can agglomerate with other radioactive and non-radioactive (i.e. inert structural material) aerosols to form larger particles which can in turn settle on structural surfaces or water pools.

Chemical interactions between fission product vapours or aerosols and metallic surfaces lead to a slow heating up of structural surfaces (due to the decay heat content of deposits) that can be expected to increase the surface temperatures beyond those required for the revaporization of chemically unbound volatile fission products previously deposited.

Treatment of fission product vapours and aerosols is similar in most of the available computer codes. However, weaknesses exist in terms of present treatment of the fission product gas phase and aqueous chemistry as well as the effects of condensation processes.

In general, for those accident sequences with attendant conditions that are conducive to relatively high retention of fission products within the RCS boundaries, revaporization and mechanical resuspension of material previously deposited can sometimes play a dominant role in the overall environmental release, and therefore require careful analysis.

Retention fractions for fission products for RCSs, derived on the basis of calculations by STCP and MELCOR [22] for PWRs and BWRs designed in the USA, are listed in Table XV.

6.4. TASK 21: RELEASE OF FISSION PRODUCTS DURING THE EX-VESSEL PHASE

Only a partial release of fission products can occur during the in-vessel phase of an accident. If energetic events such as a steam explosion or HPMEs occur, additional releases of fission products due to oxidation of finely dispersed fuel fragments could be of significance. However, models are not currently available in the literature to treat the potential volatilization of fission products during energetic dispersal. After debris ejection (energetic events) or relocation (low pressure scenarios) onto the containment floor, if a coolable core debris configuration is not maintained, high core debris temperatures could be sustained as the melt interacts with the concrete basemat, with potential for the release of more fission products into the containment. A brief discussion of the fission product release mechanism during core–concrete interactions is given in Appendix III.

In general, most of the core inventory of volatile fission products is released in-vessel; nevertheless, the remaining volatiles (most notably Te) and some of the

TABLE XVI. PROPERTIES OF VARIOUS CONCRETE AGGREGATES [4]

Property	Basaltic	Limestone/common sand	Limestone
Solidus temperature (K)	1350	1420	1690
Liquidus temperature (K)	1650	1670	1875
Decomposition temperature (K)	1450	1500	1750
Water content (%)	2.0	4.7	6.0
Carbon dioxide content (%)	1.5	21	36

TABLE XVII. TYPICAL ESTIMATES OF FRACTIONAL RELEASES RESULTING FROM EXTENDED CORE–CONCRETE INTERACTIONS

Fission product group	PWR, basaltic concrete	PWR, limestone concrete	BWR, basaltic concrete	BWR, limestone concrete
Xe	Complete	Complete	Complete	Complete
I, Cs	Complete	Complete	Complete	Complete
Te	Complete	Complete	Complete	Complete
Sr	0.20	0.40	0.35	0.75
Ba	0.10	0.20	0.30	0.60
Ru	$\ll 0.001$	$\ll 0.001$	$\ll 0.001$	$\ll 0.001$
La	0.01	0.01	0.02	0.05

refractory fission products (lanthanides and actinides) will also be released during interactions with concrete. The quantity of fission products released during the ex-vessel phase of a severe accident is a function of core debris temperature and pool temperature, content of Zr, the chemical activity of various species and compounds, and the gaseous content of the decomposing concrete (see Table XVI).

Special attention must be paid in 'conserving' Zr content between the in-vessel and ex-vessel phases of severe accidents. Increasing the Zr content of core debris increases the addition rate of chemical energy to the melt due to exothermic oxidation of Zr. This has the potential of increasing the melt temperature and, subsequently, generation of fission product aerosols.

Table XVII lists typical estimates of fractional releases resulting from extended core–concrete interactions. Plant specific calculations are strongly recommended.

6.5. TASK 22: RETENTION OF FISSION PRODUCTS INSIDE THE CONTAINMENT

Deposition of fission products inside pressure suppression pool water (for BWRs or other reactors with a similar concept) depends strongly on, among other things, particle size, water temperature, gas stream velocity, injection path and water depth. The calculated decontamination factors have been shown to vary between

TABLE XVIII. TYPICAL DECONTAMINATION FACTORS DUE TO ACTIONS OF ENGINEERED SAFETY FEATURES, WATER POOLS AND NATURAL REMOVAL PROCESSES

Active or natural removal process	Decontamination factor
Water pools	
Shallow depth	2–5
Deep	10–20
Pressure suppression pools	100–1000
Containment sprays	100–1000
Natural removal processes	
Early containment failure	5–10
Late containment failure	50–100

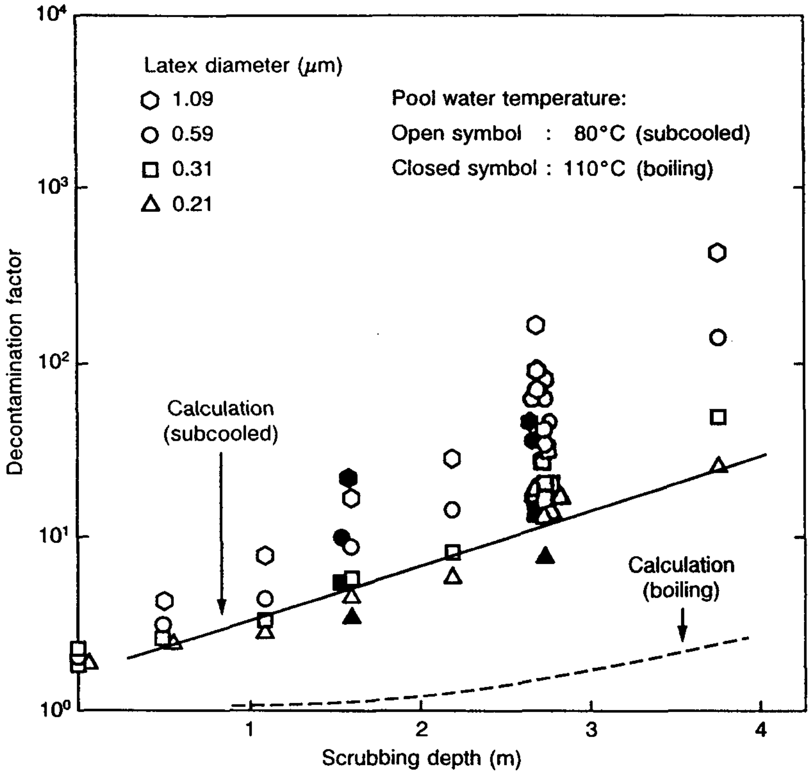

FIG. 5. Effect of scrubbing depth and pool temperature on pool decontamination factor [23].

about 10 to 100 000 [8, 20–23]; however, even at the lower estimates of decontamination factors, substantial retention could take place. Capabilities of overlying water pools (i.e. a flooded core debris pool during core concrete interactions) to retain fission products are also significant. Table XVIII lists some typical decontamination factors. Figure 5 shows a comparison of calculated decontamination factors and experimental data as reported recently [23]. Fission product removal by other engineered safety systems (e.g. sprays, fans) inside the containment is best assessed on the basis of the plant, the containment and the accident sequences. Order of magnitude estimates are also provided in Table XVIII.

Even in the absence of active mechanisms for the removal of fission products, substantial reductions in airborne activity of fission product aerosols can be realized, provided that the containment building remains intact for some time following initiation of an accident. Here again, the amount of retention due to natural processes of agglomeration and deposition is a strong function of the initial particle size, steam condensation rates, natural convection effects and containment configuration. The decontamination factors listed in Table XVIII are only order of magnitude estimates; they need not represent the particular conditions encountered under specific accident conditions in a specific plant or containment configuration and reactor type.

Other factors influencing environmental releases include potential retention of fission products inside secondary buildings and engineered filtered vent systems. These impacts must be included on a case by case and plant specific basis.

Additional contributions to environmental releases resulting from revaporization, resuspension, chemical decomposition and other mechanisms are also taken into account. These mechanisms are highly uncertain, as attested to in Ref. [10]. Nevertheless, late revaporization of previously deposited volatiles has been found to be of significance to accident sequences that involve late failure of the primary containment.

6.6. TASK 23: TREATMENT OF UNCERTAINTIES IN THE ESTIMATED SOURCE TERMS

In Level 2 PSAs, the uncertainties in the estimates of source terms are significant and the PSA analysts need to be careful in using the estimated source terms.

Past research [8, 10, 19–22] and studies have identified the main contributors to the uncertainty in severe accident source terms, which are listed in Table XIX. Table VII addresses the uncertainty in accident progression, which also propagates through the source term estimates.

TABLE XIX. ISSUES GIVING RISE TO UNCERTAINTIES IN SOURCE TERMS

Chemical forms of iodine
Chemical processes during early and later core degradation
Transport and retention of fission products and aerosols in RCS
Chemical processes during molten core–concrete interaction
Release of fission products and aerosols during molten core–concrete interactions
Interaction between hydrogen burn/radicals in flame front fission products
Deposition of fission products in containment
Decontamination of fission products in BWR suppression pool, PWR pressurizer and PWR ice bed
Revaporization of fission products
Accident progression (see Table VII)

A major cause of uncertainty is modelling in the codes. Modelling is difficult for some phenomena owing to lack of knowledge. Some models are inadequate due to oversimplifications, wrong assumptions and mere programming errors. All these affect the uncertainty in the calculated source terms. Although systematic use of computer codes will give substantial insights into the characteristics of severe accident progression, fission product release and transport behaviour, the user of the codes needs to know that there may be significant uncertainties in the calculated source terms. Experiments and code comparisons and the results of benchmark exercises are very useful for an understanding of these uncertainties [24].

The general treatment of sensitivities and uncertainties has already been discussed under Task 16, which is also applicable to uncertainties in estimated source terms. However, it is important to note that, in quantifying the uncertainties in source terms for severe accidents, use can be made of simple parametric models, such as those developed for the NUREG-1150 study and other recent studies [10, 11]. In applying these simple parametric approximations, physically realistic parametric values that are based on more detailed calculations need to be used.

6.7. TASK 24: PRESENTATION AND INTERPRETATION OF THE ESTIMATED SOURCE TERMS

The environmental release quantities associated with each release bin are a direct function of the bin attributes. For instance, for a severe accident resulting from a large break LOCA in a PWR (low RCS pressure at vessel breach), failure of the containment cooling system (fans and/or sprays) may lead to containment failure if hydrogen combustion occurs immediately after vessel breach (early). The source term example listed for bin 1 of Table XX can be estimated using the numerical estimates provided in Tables XIV to XVIII (i.e., for low RCS pressure, basaltic concrete, dry cavity and a containment decontamination factor of 10). In Table XX, early releases are defined as those occurring at vessel breach, and late releases result from extended core–concrete interactions on the containment floor. Of course, similar source terms can be calculated using any of the available computer codes.

Tabulation of the radiological source term estimates for each release category are performed as exemplified in Table XX. The insights gained from quantification of releases of radionuclides and the Level 2 PSA also need to be summarized and discussed.

In addition, the results of Level 2 PSA need to display the uncertainties, possibly in a form of complementary cumulative distribution functions. Specifically, for each fission product group, the frequency of exceeding a given quantity of release is provided. The results can clearly show the statistical significance of each complementary cumulative distribution function curve (e.g., mean, median, 95 percentile, etc.).

TABLE XX. SUMMARY OF RADIOLOGICAL SOURCE TERM ESTIMATES

Release			Release group (fraction)							
Bin	Frequency	Phase	Xe	I	Cs	Te	Ba	Sr	Ru	La
1	R(1)	Early	0.95	0.08	0.08	0.5	0.002	8.0×10^5	—	—
		Late	0.05	0.005	0.005	0.04	0.01	0.02	—	0.001
		Total	1.00	0.085	0.085	0.09	0.012	0.02	—	0.001
2	R(2)	Early	..	..	..	..	..	..	..	..
		Late	..	..	..	..	..	..	..	..
		Total	..	..	..	..	..	..	..	..
..	..	Early	..	..	..	..	..	..	..	..
		Late	..	..	..	..	..	..	..	..
		Total	..	..	..	..	..	..	..	..
N	R(N)	Early	..	..	..	..	..	..	..	..
		Late	..	..	..	..	..	..	..	..
		Total	..	..	..	..	..	..	..	..

Finally, the report clearly documents important findings of the PSA, including vulnerabilities identified, key operator actions, the potential benefits of various engineered safety systems, and areas for possible improvements in operations or hardware for the plant and/or the containment. At this stage the results of the PSA may be compared with Level 2 probabilistic safety criteria which are discussed in Ref. [25]. Criteria may be expressed as a point estimate frequency, usually less than 10^{-6} per year of a large, early release, or as curves for comparison with the complementary cumulative distribution functions discussed in the previous paragraph. Particular questions that may be addressed in these assessments are:

- The definition of a large release;[1]
- Whether the probabilistic safety criteria reflect the design target on the threshold of intolerability; and
- The level of detail required for the estimates of risk in the Level 3 PSA (e.g. time and duration of release, energy content, height).

[1] The definition of a large early release is a matter of great controversy; however, for the present purpose, a large release may be defined as a release corresponding in terms of activity to 5–10% of the core inventory of radioiodines. The activity associated with iodine corresponding to 10% of the core inventory of a large power reactor is an assumed threshold for prompt fatalities.

7. DOCUMENTATION OF THE ANALYSIS: PRESENTATION AND INTERPRETATION OF RESULTS

The details of rationale and analyses for a Level 2 PSA are reported in a way which presents information on the methods used, the PSA process, and the insights and conclusions drawn in a logical development. The report itself needs to be amenable to peer review and to provide a structured entry route to detailed supporting material.

A comprehensive guidance for documentation requirements, objectives, organization and preparation has been developed as part of the IAEA Safety Series No. 50-P-4 [1], and is equally applicable to the Level 2 PSA. Therefore, the main objective of Section 7 is to provide specific guidelines for the Level 2 PSA.

7.1. TASK 25: OBJECTIVES AND PRINCIPLES OF DOCUMENTATION

The documentation for a Level 2 PSA must reflect the objectives of the study and the needs of end users, and must meet requirements for subsequent upgrading and refinement in the light of technical advances.

The potential users of a Level 2 PSA include:

— utilities (management, operating personnel);
— designers and reactor vendors;
— peer reviewers;
— regulatory authorities;
— other government bodies; and
— the general public.

The main applications/users of a Level 2 PSA are identified in Section 2.1 in conjunction with the definition of the PSA objectives.

The documentation needs to be well structured, clear, concise, succinct and open to scrutiny by the readers and peer reviewers. In addition, the Level 2 PSA documentation must be easily upgradable as new research results and models become available. It also needs to allow for easy broadening of the scope of the PSA in question and use for alternative applications. Explicit presentation of the underlying assumptions, exclusions, limitations and features are integral elements of the documentation for a Level 2 PSA.

Finally, it is recommended that:

— conclusions be distinct and reflect not only the main general results but also the contributory analyses;
— emphasis be given to the analysis of uncertainties in phenomena, models and the database; and

TABLE XXI. SAMPLE OUTLINE OF EXTERNAL DOCUMENTATION FOR A LEVEL 2 PSA STUDY

S. Summary report

- S1. Introduction
- S2. Overview of the objectives and motivation for the study
- S3. Overview of the approach
- S4. Results of containment failure modes and likelihoods
- S5. Radiological source terms and their frequencies (complementary cumulative distribution functions)
- S6. Summary of plant vulnerabilities to severe accidents, and interpretation of results, conclusions, and recommendations
- S7. Potential risk reduction measures
- S8. Organization of the main report

M. Main report

- M1. Introduction
  - M1.1. Background
  - M1.2. Objectives and motivations for the study
  - M1.3. Scope of the study
  - M1.4. Project organization and management
  - M1.5. Composition of the study team
  - M1.6. Overview of the approach
  - M1.7. Report organization
- M2. Description of the design of the plant and the containment
  - M2.1. Plant and containment design features affecting severe accidents
  - M2.2. Operational characteristics
  - M2.3. Description of plant and containment system modifications (if any)
- M3. Interface to Level 1 PSA
  - M3.1. Grouping of sequences and definition of attributes
  - M3.2. PDSs for internal initiators and uncertainties
  - M3.3. PDSs for external initiators and uncertainties
  - M3.4. PDSs for other power states and uncertainties
- M4. Analysis of the containment's structural performance
  - M4.1. Description of the structural design and failure modes of the containment
  - M4.2. Approach for structural analysis
  - M4.3. Structural response and fragility results
  - M4.4. Summary of uncertainties and/or fragility curves for containment performance
  - M4.5. Impact of external events

TABLE XXI. (cont.)

- M5. Accident progression and containment analysis
  - M5.1. Severe accident progression analysis
    - M5.1.1. Scope of the analysis
    - M5.1.2. Method of analysis (codes, models, etc.)
    - M5.1.3. Summary of point estimate results for analysed PDSs
  - M5.2. Accident progression event trees/containment event trees
    - M5.2.1. APET/CET structure
    - M5.2.2. Operating procedures and recovery
    - M5.2.3. APET/CET quantification process
    - M5.2.4. Binning of APET/CET end states
    - M5.2.5. Treatment of uncertainties
    - M5.2.6. Results
      - M5.2.6.1. Point estimate C-matrix
      - M5.2.6.2. Uncertainties in failure probabilities
      - M5.2.6.3. Interpretation of results
- M6. Accident source terms
  - M6.1. Grouping of fission products
  - M6.2. Method of analysis (codes, models, etc.)
  - M6.3. Summary of point estimate results for analysed PDSs
  - M6.4. Treatment of uncertainties
  - M6.5. Results
    - M6.5.1. Point estimate source term characteristics
    - M6.5.2. Uncertainties in source term characteristics
    - M6.5.3. Interpretation of results
- M7. Sensitivity and importance analyses
  - M7.1. Identification of sensitivity issues
  - M7.2. Results of sensitivity analysis
  - M7.3. Importance ranking of issues
- M8. Conclusions
  - M8.1. Key insights on severe accidents and containment response characteristics
  - M8.2. Design features and inherent mitigatory benefits
  - M8.3. Conclusions relative to PSA objectives
  - M8.4. Recommendations for future work

A. Appendices
- A1. Basis for containment structural fragilities
- A2. Basis for APET/CET quantification
- A3. Results of deterministic severe accident analyses
  - A3.1. Containment loads
  - A3.2. Accident source terms
- A4. Basis for uncertainty/sensitivity distributions/ranges
- A5. Detailed results of uncertainty/sensitivity analyses

— the impact of underlying assumptions, uncertainties and conservatisms in the analyses and methods be demonstrated through the presentation of sensitivity results.

7.2. TASK 26: ORGANIZATION OF THE DOCUMENTATION FOR A LEVEL 2 PSA

The nature and amount of information for the external documentation compared with that intended for in-house support documentation is established by the PSA team and reviewed by the PSA management.

The Level 2 PSA documentation contains all of the detailed information that would be needed to reconstruct the PSA study. To the extent possible, all of the intermediate analyses, rationales for probabilistic estimates and supporting calculations are documented, either as appendices or as internal reports. All working papers and computer code inputs and outputs need to be retained in a traceable format, outside the formal documentation.

The documentation principles established as part of the IAEA Level 1 PSA guidelines [1] also apply here. The Level 2 PSA documentation is also divided into three major parts, namely:

— summary report
— main report
— appendices to the main report.

The summary report provides an overview of the entire effort, including objectives, scope, approach, results, conclusions and potential impacts on plant design, operation and maintenance. The summary document is aimed at a wide audience of reactor safety specialists and peer reviewers. Other aspects of the summary report are discussed in Ref. [1].

A 'map' for the main report is also provided, to guide reviewers to sections where additional details and supporting analyses are included. The summary document needs to be prepared by an individual who has an excellent overview of the entire PSA study. It is prepared after the entire documentation has been completed and reviewed by individual task leaders/analysts for consistency.

The main report gives a transparent, traceable, step by step presentation of the complete study, including clear statements of all assumptions, rationale and plant specific aspects affecting the results. The report is intended for specialized PSA analysts and peer reviewers. It also includes sufficient information to show that the conclusions are completely supported by the main report and all of the appendices (without the need for the supporting informal calculational notes, computer outputs, etc.).

A possible outline for the external documentation for a Level 2 PSA is given in Table XXI.

Appendix I

EXAMPLE OF A TYPICAL SCHEDULE FOR A LEVEL 2 PSA

Table XXII shows a simplified schedule for a short schedule Level 2 PSA based on the tasks defined in this report. The durations shown are indicative (for example, the containment structural task assumes the use of reference plant analyses) and the iterative nature of some of the tasks is not shown. The tasks need to be split into more than one phase so that some of them can be repeated when the results of other tasks are available.

TABLE XXII. EXAMPLE OF A SCHEDULE FOR A LEVEL 2 PSA

	Month																	
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
Management and organization																		
1. Definition of objectives																		
2. Definition of scope																		
3. Project management																		
4. Selection of approach and codes and establishment of procedures																		
5. Team organization																		
6. Team training																		
7. Funding and scheduling																		
8. Setting of quality assurance procedures																		
9. Selection of internal peer reviewers																		
Study performance																		
10. Plant familiarization and identification of important design features																		
11. Interface to Level 1 PSA and sequence grouping																		
12. Accident progression and containment analyses:																		
a. Containment performance analysis																		
b. Severe accident progression analysis																		
c. Development and quantification of APETs/CETs																		
d. Binning of end-states into release categories																		
e. Treatment of accident progression uncertainties																		
f. Summary and interpretation of accident progression and results of containment analysis																		
13. Source terms for severe accidents:																		
a. Grouping of fission products																		
b. Point estimate source terms																		
c. Treatment of uncertainties in source terms																		
d. Summary and interpretation of results																		

	Month																	
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
Documentation of PSA: display and interpretation of results																		
14. Integration, interpretation and documentation of results																		
a. Objectives and principles of documentation																		
b. Organization of documentation																		
c. Preparation of documentation																		
Quality control and review																		

Appendix II

COMPUTER CODES FOR SEVERE ACCIDENTS

II.1. INTRODUCTION

Severe accident phenomena are complex with many interactions and they can be realistically modelled only with large computer codes. Appendix II lists some of the codes commonly used in Level 2 PSAs with a brief description of their areas of application. General principles for the use of codes of phenomena in PSAs are outlined in the following.

II.2. GENERAL DESCRIPTION OF COMPUTER CODES

II.2.1. Types of code

The codes that model the phenomena of severe accidents can be divided into three types according to their capabilities and intended use. Mechanistic codes attempt to model the phenomena in as much detail as possible, without regard for how long the code takes to run (although if the run times begin to exceed a few days then the utility of the code for any purpose begins to diminish dramatically). These codes are used typically in severe accident research, and also to provide benchmarks for simpler codes (comparing the results of the two codes for a few key sequences and perhaps providing information to quantify certain parameters in the simpler codes to improve the representation of phenomena).

Among the mechanistic codes are also those describing the mechanical response of the containment building under internal loads resulting from LOCAs, steam explosions, hydrogen burn or internal missiles. Generally these codes are multipurpose finite element codes for structural dynamics and/or statics. The codes are able to deal with non-linearities, both geometrical (i.e. large displacement) and material (i.e. plasticity). The code architecture allows easy implementation of various material models and failure criteria into the code. The codes can usually model both local (e.g. crack propagation, penetration) and global responses of the loaded structure. There are many codes available to analyse containment response, e.g. DYNA3D, ABAQUS, NASTRAN, HONDO, NEPTUNE or WHAMS. In general it is not possible to use these structural mechanics codes as 'black boxes'. Deep understanding of both the code used and the problem analysed is necessary.

In contrast, the PSA codes, intended for routine application in PSA, are designed to run fast, so that they can calculate many sequences (and a number of times for a single sequence if uncertainty analyses are required). In order to achieve these shorter run times, the modelling has to be simpler than in the mechanistic codes. As an example of the sort of simplification used, consider aerosol modelling. In the mechanistic codes a numerical solution is found for the integral–differential equation for aerosol agglomeration and deposition, giving the aerosol size distribution at each time step. In contrast, the MAAP code sponsored by the Electric Power Research Institute [26], developed for PSA application, uses a correlational approach for aerosol behaviour. This was developed on the basis of dimensional analysis, with parameters specified by experimental measurements of aerosol behaviour. If for a sequence there is concern that the correlation may be outside its range of applicability, the information could be reinforced with further calculations using a mechanistic aerosol code.

There are also the so-called simple parametric codes, intended for specific PSA applications, such as source term estimation, where more runs are needed than can be reasonably handled even by the PSA codes. These are based on simple parametric models which interpolate between fixed points, for which calculations with a more complicated code have already been performed, to determine the values of the parameters. The use of such codes is reasonable in generating uncertainty values, but it is important to take into account that the parameters used in the codes as well as the results produced by them have to be calibrated by more detailed calculations or experimental data.

In order to cover the whole range of phenomena needed to determine the threats to the containment and the fission product source terms, typically a single monolithic code need not be used. Separate codes, each dealing with a particular phase or aspect of severe accident behaviour, are coupled in a suite, with some interfacing facility for the transfer of information between the codes. For routine PSA application it is desirable to have an automatic transfer of information between the elements of a code suite. Manual transfer is slow and can also lead to the introduction of errors. However, it does have the advantage that the analyst can examine the intermediate outputs of the codes to check that the results are meaningful. A more integrated and modular approach tends to be adopted in the newer generation of severe accident codes.

II.2.2. Validation status of a code

Verification and validation of computer codes are crucial mechanisms in a process to enhance confidence in their application. It is useful to distinguish between the different stages of testing a code using the terms 'verification' and 'validation'. In the most common usage, verification of a code means testing by performing the calculations for which it is intended. A code that solves a differential equation might be tested on a known analytic solution of the equation to confirm that it is indeed giving solutions to an acceptable level of accuracy. However complex the phenomena may be, the laws of conservation of mass and energy must apply. Checking that the code predictions obey the conservation laws would be another simple verification test. Validation, on the other hand, is a process that a code must undergo to see whether it provides a sufficiently accurate representation of the reality of the severe accident phenomena that it models.

Achieving a state with severe accident codes that could reasonably be called validation is very difficult. The extreme conditions that occur in a severe accident and the scale of the physical geometry are difficult to realize in an experiment. The process of validation, in general, comprises a validation matrix involving many simulations. These may range from comparison with experiments on separate effects in examining the more fundamental aspects of the phenomena to larger scale integral experiments. Typically, the experiments, designed on the basis of scaling arguments, are conducted in smaller scale facilities using some representative materials for simulations.

Care needs to be taken with those code validations which have been achieved by varying the values of user supplied parameters until a reasonable fit to experimental data is achieved. At best this is an indirect experimental measurement of the parameter values, and not an independent validation of the code. A true validation includes the accurate prediction of many more data points than there are adjustable parameters within the code.

II.2.3. Use of the codes

By definition, PSA codes are designed so that a Level 2 PSA analyst with a good degree of familiarity with general accident phenomena can run them reliably without the same detailed knowledge as a specialist using a mechanistic code dealing with a particular phenomenon or a phase of a severe accident. It is also essential that the analyst must have a good working knowledge of the reactor systems. In order for the code calculations to be meaningfully incorporated into the framework of a Level 2 PSA, it is essential that the analyst has a reasonable knowledge of the following:

— the phenomena addressed in a code and their modelling approach and limitations;
— the meaning of the input variables;
— the meaning of the output variables.

The point to be emphasized here is that, given the complexity of these issues, the code must not be simply treated as a black box.

Many codes ask the user to specify the time steps for the differential equation solver within the code. A choice of too small a time step will make the run time unacceptably long, while a choice of too large a time step will make the solution inaccurate. Numerical instabilities can also occur with either. The analyst checks the sensitivity of the predictions to the choice of time step and looks for convergence of the results as a function of decreasing time step. If, for a given application, convergence cannot be achieved without going to impracticably long run times, the code may be inappropriate for this application.

If the analyst intends using a mechanistic code, especially one that is research oriented, the learning curve may be considerable and the help of a code specialist will be essential for correct interpretation of results.

II.3. EXAMPLES OF CODES FOR SEVERE ACCIDENTS

The general form of severe accident codes is shown diagrammatically in Table XXIII. This section provides a brief description of some specific codes, currently in use for Level 2 PSAs, which deal with most or all of the phenomena shown in Fig. 6. A list of major mechanistic codes is also included.

II.3.1. Source term code package

General

The source term code package (STCP) [35] developed by Battelle Columbus Division is a suite of codes that calculates a spectrum of phenomena leading to radioactive releases to the environment. The STCP is the United States Nuclear Regulatory Commission's (USNRC's) first generation integrated severe accident code for use in PSA applications. It has been used to analyse a number of accident sequences in the NUREG-1150 study [10]. It is now being superseded by the more advanced integrated code MELCOR [36], which addresses the weaknesses of the STCP, such as: inadequate or inconsistent modelling of important phenomena or plant features; lack of facility to address the uncertainties and sensitivities; a structure lacking in flexibility for incorporation of improved or alternative models of accident phenomena; and interfaces that are not well matched and do not take account of feedback effects.

Accidents analysed with the STCP include: small break and large break LOCAs, anticipated transients without scram, transients with loss of AC power, and loss of heat removal, makeup water and emergency core cooling.

The STCP code is a linked set of modules comprising essentially existing codes. There are four main elements:

(1) MARCH3 is a combination of MARCH2, CORSOR-M and CORCON/MOD2, includes the thermal-hydraulics and core melt progression (MARCH for general thermal-hydraulics, CORCON/Mod2 for core–concrete interactions), and in-vessel fission product release from fuel (CORSOR-M).
(2) VANESA calculates the ex-vessel radionuclide release and aerosol generation from the core–concrete interactions.

TABLE XXIII. MECHANISTIC CODES AND IN-VESSEL SEVERE ACCIDENT PHENOMENA MODELLED

Country	Computer codes	In-vessel phenomena				
		Thermal-hydraulics	Core melt progression	Release from fuel	Transport in RCS	Vessel failure
USA	RELAP5 [27]	✓				
	TRAC [28–30]	✓				
	SCDAP-RELAP5 [31]	✓	✓	✓	✓	✓
	MELPROG-TRAC [32]	✓	✓	✓	✓	✓
	VICTORIA [33]			✓	✓	
Germany	ATHLET-SA [34]	✓	✓	✓	✓	
France	CATHARE-ICARE [34]	✓	✓	✓	✓	

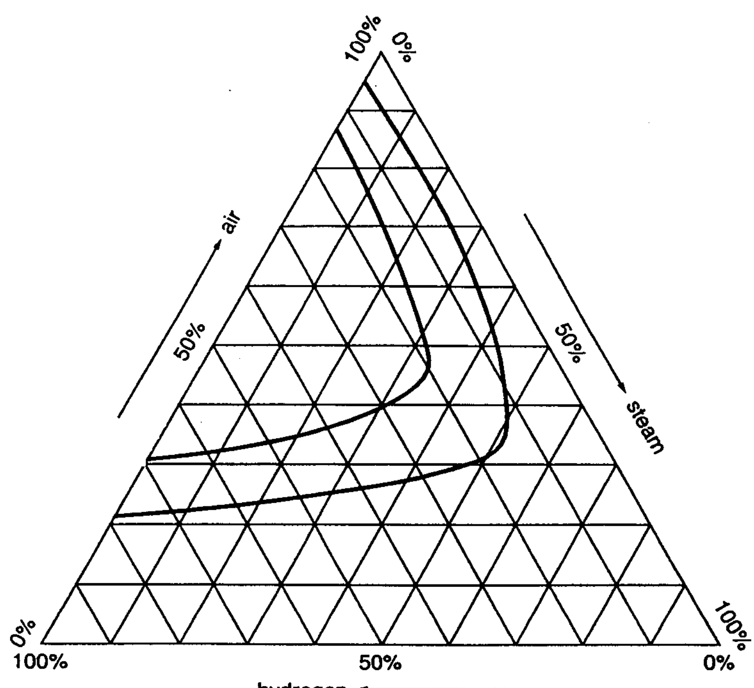

FIG. 6. General form of severe accident codes for LWRs.

(3) TRAP-MELT3 is a combination of the MERGE code that provides more detailed flow rates and temperature with the TRAP-MELT2 code that calculates the fission product transport and deposition in the RCS.

(4) NAUA/SPARC/ICEDF is three separate codes treating fission product retention in the containment. Interactions between thermal-hydraulics and fission product transport are only taken into account by obtaining conditions from MARCH3.

Thermal-hydraulic modelling

Compared with other detailed thermal-hydraulic codes, the STCP code uses a very simple control volume scheme (e.g. one single control volume to model the RCS). The core–concrete interactions and the in-vessel fission product release from the fuel are treated with the CORCON/Mod2 and CORSOR-M codes, respectively, as subroutines.

MERGE subdivides the control volumes in MARCH into as many as ten subvolumes and provides more detailed flow rates and temperatures in a manner suitable for TRAP-MELT2.

Core geometry and core melt modelling

The reactor core is divided into annular rings, called nodes, arranged in radial and axial segments. Radionuclide generation and heating in each of these nodes are based on the power profile for normal operation. MARCH uses a single, lumped parameter core slumping model.

Other physical processes

The pressure and temperature effects on the containment due to mass and energy additions and the burning of hydrogen and carbon monoxide are taken into account. For phenomena that are not well understood, there are a number of user specified options to analyse various modelling assumptions.

Radionuclide behaviour

TRAP-MELT2 calculates the fission product transport and deposition in the RCS. The VANESA code calculates the ex-vessel radionuclide and aerosol generation from the core–concrete interactions. NAUA, developed by Kernforschungszentrum Karlsruhe in Germany, calculates the agglomeration and deposition of aerosol particles within the containment. The code has a provision to interact with the SPARC and ICEDF codes for the aerosol removal effect of suppression pools and ice condensers, respectively.

II.3.2. Modular accident analysis program: MAAP 3.0B

General

The Electric Power Research Institute's Molecular Accident Analysis Program (MAAP 3.0B) [26], developed as a PSA tool, is a fully integrated code that couples thermal-hydraulics with fission product release and transport. It has been used for many PSAs, especially for most of the US Individual Plant Examination (IPE) programme. It analyses the accident progression from a set of initiating events either to a safe, stable and coolable state, or to structural failure of the containment and radioactive release to the environment.

Accidents analysed include a variety of transients, including bypass, mid-loop operation and shutdown sequences.

The design intent for this code for PSA application results in major differences in modelling assumptions, in comparison with the mechanistic codes. An example is provided by the configuration of the debris pool in modelling the core–concrete interaction. This is modelled as a homogeneous molten debris pool, in contrast with the mechanistic code representation of a stratified pool which requires more complex modelling of underlayer heat transfer. The code has been subjected to independent design review and is currently being reviewed by the USNRC. The extent of validation is indicated in Ref. [26]. Also, it has been compared with other codes on some aspects of severe accident phenomena (e.g. creep rupture failure of RCS piping induced by natural circulation (SCDAP/RELAP5) [37]; core melt progression and source term estimates (STCP) [38]; analytical models for in-vessel and ex-vessel accident progression and estimates for selected sequences (STCP) [24]).

Over a period of time, the code has also been modified to be used as a tool to evaluate accident management actions. Separate versions for PWRs and BWRs are available.

Thermal–hydraulic modelling

MAAP uses a control volume and flow path approach in which the geometry of the control volumes (called regions) is prespecified and different for a PWR and a BWR. The primary system is divided into regions: upper and lower plenum, reactor core and downcomer; and for PWRs, cold and hot legs and steam generator loops. Separate mass and energy conservation equations are solved for each of the regions. The PWR containment is divided into regions: upper and lower compartment, cavity, annular compartment, pressurizer relief tank, pressurizer, possibly two extra compartments for an ice condenser, and primary system. The BWR containment is divided into two regions: reactor pedestal cavity, dry well, wet well, possibly an upper and a lower containment compartment, and primary system. Flows consist of steam, water, hydrogen and other incondensable gases, and molten core material.

Flow paths can be, for example, pipes, surge lines, penetrations and relief valves. Separate mass and energy conservation equations are solved for each region. The equations are lumped parameter, non-linear, first order, coupled and ordinary differential equations.

Core geometry and core melt modelling

The core is divided into concentric radial rings (up to 7) and axial segments (up to 10). MAAP uses a single core relocation model. Features are included in the code such that limited sensitivity studies can be performed on the core melt behaviour and hydrogen generation. MAAP assumes a reduction in steam supply, and hence in hydrogen generation, due to channel blockage in the relocated core.

Other physical processes

Concerning hydrogen combustion, MAAP does not distinguish between flame ignition and flame propagation. The incomplete burning model is one dimensional.

Radionuclide behaviour

MAAP models the transport and retention of fission products. The materials released from the core are divided into six groups. The fission product states modelled are: vapour, aerosol, and deposited and contained in-core or molten core material. Revaporization is included as transfer between the states. The retention rate is calculated using a correlation which is a function of the aerosol concentration. Agglomeration of aerosols is calculated using a correlation derived from experiments.

II.3.3. Modular accident analysis programme: MAAP 4

MAAP 4 [39] has recently been released. Apart from general modelling enhancement, it is designed to evaluate potential accident actions and also for application in ALWR studies. Three major differences compared with MAAP 3.0B are:

— *Core melt progression model and creep rupture failure of the reactor coolant system induced by natural circulation.* Prior to core uncovering, the RCS response is not substantially different from that calculated by the MAAP 3.0B code. Once the core is uncovered and overheated sufficiently to result in rapid oxidation of the Zircaloy cladding, the first major difference is apparent. In MAAP 4, when the melting point of the control rod material is calculated, it can relocate away from the fuel. In addition, the MAAP 4 models include the process of dissolving the uranium dioxide fuel with molten zirconium and the relocation of the material of lower melting point. This is substantially different from the lumped fuel behaviour in the MAAP 3.0B code. As the core melting progresses, the potential for natural circulation flows, particularly for the open lattice PWR core design, is evaluated together with the potential for creep rupture of the hot leg piping, the pressurizer surge line and the steam generator tubes.

— *Modelling of the reflooding process, external vessel cooling and vessel creep rupture.* If the accident sequence being considered results in reflooding of the reactor core once core degradation has occurred, the MAAP 4 models address this reflooding process and the potential for quenching of the core debris, both within the original core boundaries and in the lower plenum of the RPV. The first of these is different from the MAAP 3.0B code, while the second is a set of phenomena that could be represented in the MAAP 3.0B codes. However, if water is available on the exterior of the RPV, the influence of external cooling in removing energy from the vessel wall and in preventing the potential creep rupture of the vessel due to thermal attack by core debris on the vessel lower head is modelled. Both external cooling and creep rupture of the vessel are phenomena not included in previous MAAP versions.

— *Containment model:* For the containment analyses, the containment model has been enhanced to provide a generalized description of the containment such that the nodalization can be specified by the user. The containment model for many of the advanced plants has been set up to include those features typical of the ALWR designs.

II.3.4. MELCOR

General

The USNRC's MELCOR code [36, 40–42], developed by Sandia National Laboratories, is an integrated code that replaces the STCP. MELCOR is designed to be a fully integrated, relatively fast running code with the flexibility to model a wide spectrum of severe accident progression phenomena. It includes many modelling features and concepts of the other USNRC codes, such as for example MELPROG [32] and CONTAIN [43]. The use of parametric models is, in general, limited to areas with great uncertainties where there is no consensus concerning an acceptable mechanistic approach.

MELCOR calculations have been performed for the NUREG-1150 study [41]. MELCOR is a generic code applicable for both PWRs and BWRs.

Thermal-hydraulic modelling

MELCOR uses a control volume and flow path approach in which the geometry of the control volumes is specified by the user, and an arbitrary number of flow paths may be used to connect pairs of control volumes. Each volume may contain a single phase or a pool of water and an overlying gaseous atmosphere. The flow of pool and atmosphere through flow paths is determined by simplified two-fluid momentum equations. There is no formal distinction between the RCS and the containment thermal-hydraulics; the same models and algorithms are used for both. Components such as steam generators are built up from general control volumes, flow paths and heat structures. A general interface for mass and energy sources and sinks is used to couple the thermal-hydraulics to models for heat structures, the core, the reactor cavity and decay heat generation. The use of multiple volumes allows natural circulation loops to form within the vessel.

Core geometry and core melt modelling

The core and lower plenum regions of the RPV are typically divided into concentric radial rings and axial segments, which define individual core cells that may be significantly smaller than the control volumes. Each cell may contain one or more types of component, including intact fuel, cladding canister walls (for BWRs), components such as control rods or guide tubes, and particulate debris, which may each contain several materials (e.g. UO_2, Zircaloy, ZrO_2).

Oxidation and heat transfer by radiation, conduction and convection are calculated separately for each component. A simple candling model treats the downward flow and refreezing of molten core materials, thereby forming layers of solidified debris on lower cell components, which may lead to flow blockages and molten pools.

Failure of the core structures such as the core plate, as well as lower head heat-up and failure followed by debris ejection, are treated by simple parametric models.

Other physical processes

In addition to the processes already mentioned, MELCOR includes models for: the forming of incondensable gases, combustion of gases (using the HECTR models [44]), the thermal–hydraulic part of core–concrete interactions (using the CORCON-Mod2 model [45]), and DCH (using a parametric model).

Radionuclide behaviour

The release of aerosols and vapours from the core materials is treated by the CORSOR correlations with a dynamic surface to volume multiplier. Releases from core–concrete interactions are treated by the VANESA models. Aerosol agglomeration and deposition are calculated by the TRAP-MELT models. Transport of aerosols and vapours between control volumes occurs with the bulk fluids, gases or water, with zero slip, and aerosols can be removed as they pass through water pools, based on a modified pool scrubbing model from the SPARC code. User specified chemical reactions can be treated, on the basis of the results of more detailed codes or on experiments [46].

II.3.5. The Japanese THALES/ART suite of codes

General

THALES/ART [24], developed by the Japan Atomic Energy Research Institute, consists of THALES for severe accident thermal-hydraulics and ART for fission product release and transport. This is a fully integrated and fast running code package that analyses the accident progression from a set of initiating events to the ultimate containment failure and radioactive release to the environment.

The code package has been used within JAERI for accident progression analyses, sensitivity analyses on source terms, accident mitigation analysis and Level 2 PSAs. In order to validate the codes, some experiments were analysed and comparison with the RETRAN code was also carried out.

In addition, a benchmark study was carried out for THALES/ART, the STCP and MAAP codes for some accident sequences. Separate versions for PWRs and BWRs are available.

Thermal-hydraulic modelling

THALES uses a control volume and flow path approach. The geometry of the control volumes in the RCS is prespecified and different for a PWR and BWR. For example, the RCS in a BWR is divided into reactor core, upper plenum, steam alone (including downcomer above the feedwater sparger) downcomer (below the sparger), lower plenum (including jet pump differences) and two recirculation loops which are connected with junctions. In each of the control volumes a mixture level is then considered which separates the control volume into a gas region and a liquid region with void. For junctions, the countercurrent flow model can be applied when specified by input. The containment can be divided into several compartments as a user option. For example, BWR Mark 1 containment can be divided into dry well, reactor cavity and suppression pool.

TABLE XXIV. MECHANISTIC CODES AND EX-VESSEL SEVERE ACCIDENT PHENOMENA MODELLED

Country	Computer codes	Ex-vessel phenomena					
		High pressure melt ejection	Core–concrete interaction	Fission product release from debris	Fission product transport in containment	Hydrogen combustion	Containment performance
USA	CONTAIN]43]	✓	✓	✓	✓	✓	✓
	CORCON [44]		✓				
	VANESA [45]			✓			
	MAEROS [46]				✓		
	HECTR [47]					✓	
Germany	WECHSL [48]			✓			
	WAVCO [49]					✓	✓

Core geometry and core melt modelling

The reactor core is axially and radially divided into many nodes. When the calculated node temperature reaches a level specified by the user, the node is assumed to be molten and begins to relocate downwards. Just after the melting of the node, the material relocates according to the relocation model chosen by the user.

In its hydrogen combustion modelling, THALES does not distinguish between flame ignition and flame propagation. Burning occurs only in the compartments where ignition conditions are reached and global burning is assumed to occur.

The THALES core-concrete interaction model is one dimensional.

Radionuclide behaviour

ART uses the same control volumes as the THALES code and models the transport and retention of fission products. Fission products can exist as gaseous material or aerosols, and can deposit on structure walls and floors and solution in water. The code solves the governing equations for multicomponent aerosols, taking into account the growth by agglomeration, and condensation/vaporization of steam and deposition of volatile materials on the aerosol.

THALES 2

THALES 2, also developed by JAERI, is the second version of the THALES/ART code package. It is also a fast running and fully integrated code. In addition to the capabilities of the THALES/ART code package, THALES 2 takes into account the feedback effect of fission product behaviour on thermal-hydraulics. The code has been used for analysing fission product revaporization phenomena.

II.3.6. Mechanistic codes

Examples of some key mechanistic codes that have been used in recent severe accident studies are listed in Tables XXIII and XXIV. The phenomena addressed are indicated in the tables.

II.4. PROBABILISTIC CODES

Fault tree, event tree and other simulation codes typically used for Level 1 PSAs are also required for a Level 2 PSA. A compilation of computer codes for Level 1 PSA is provided in a recent IAEA document [50]. In addition, a number of computer codes that have been developed in support of the NUREG-1150 [10] study are available. These include the EVNTRE code, a general and versatile event tree analysis program; and the LHS77, a stratified Monte Carlo sampling code; both developed by Sandia National Laboratories under USNRC sponsorship.

Appendix III

SEVERE ACCIDENT PHENOMENA

III.1. INTRODUCTION

An important part of a Level 2 PSA is modelling the phenomena that might occur between the onset of core damage and either the safe termination of the accident or the failure of the containment and leaking of radioactive materials to the environment. Such an understanding of what happens as the accident progresses is needed to quantify the probability and magnitude of the threats to the containment, and the magnitude and characteristics of the source term for the potential release to the environment.

The aim of Appendix III is to give an overview of possible phenomena during a severe reactor accident. The level of detail is sufficient for the purposes of a Level 2 PSA practitioner. In particular, the aim is to give the reader an appreciation of the full range of the phenomena that might occur, so that important possibilities are not omitted without consideration, and to indicate the level of uncertainty in the modelling at each step. Some processes are well modelled in a deterministic way in the Level 2 computer codes. Others are uncertain because of lack of knowledge. Still others may intrinsically be unpredictable; even with a perfect understanding, we would have to model them in a probabilistic manner.

The full list of phenomena can be very long and complex. Therefore, before producing this list, the general principles are discussed of why there are challenges to the containment and what governs the magnitude of fission product release. Then the phenomena that can occur in severe accidents at LWRs are listed in approximate chronological order of their occurrence. For a radically different reactor type it would be necessary to construct an analogous list, in consideration of the general principles of severe accident phenomenology. To illustrate how the phenomena occur in specific instances, the sequence of events for two hypothetical accidents — a high pressure PWR sequence and a low pressure BWR sequence are given in the two annexes.

III.2. GENERAL DESCRIPTION OF CONTAINMENT CHALLENGES

The function of a containment in a severe accident is to contain fission products. What makes this difficult is the presence of large heat sources. Increasing temperature makes the fission products more mobile and threatens the integrity of the containment. The basic goal of reactor safety is therefore to remove heat while retaining fission products. To do this there must be either some way of removing the heat from the containment, at some point with conduction through a solid barrier to the movement of fission products, or a large enough heat sink within the containment to contain the heat without undue rises in temperature and pressure.

Temperature increases threaten the containment because of the associated increase in the pressure of the containment atmosphere. Elevated temperatures can themselves threaten the integrity of the containment (e.g. by degrading the seals around containment penetrations). If the containment is dry the pressure increases linearly with the temperature (according to the ideal gas law). If much water is present, as will usually be the case following an LWR accident, the pressure rise is much steeper (approximated by the water vapour saturation curve if the temperature rise is slow enough) and so pressure rather than temperature is likely to be the primary threat to the containment.

An important part of any analysis of severe accident phenomena is therefore knowing what are the heat sources and where the heat resides, both as functions of time. The most obvious heat source is the decay heat; if this were not present, there would be no severe accident problem from shut down reactors. It is important to know the decay power as a function of time (the integrated decay heat release is also a useful measure of the threat to containment for use in scoping calculations). The decay power is also broken down into contributions from the volatile and less volatile fission products. As the accident proceeds, the more volatile fission products will leave the core, and their decay will no longer contribute to the heating of the core debris. But their decay heat must not be ignored; it will still be contributing to the heating of the containment atmosphere and structures.

Another potential source of heat is provided by the possibility of achieving recriticality during severe accidents. This is especially a concern for BWRs, where the control material (B_4C) has a lower melting point and eutectic formation temperature than the fuel rods (UO_2). The fuel rods can therefore remain standing while the control rods relocate downwards. If in this rather unlikely case there is reflooding

TABLE XXV. SPECIFIC ENTHALPIES OF REACTION [26]

Reaction	Specific enthalpy of reaction (kJ/kg)
Steam–Zircaloy	6 430
Steam–iron	645
Hydrogen–oxygen	121 000
Carbon monoxide–oxygen	10 107

of the RPV with non-borated water, the possibility of recriticality arises. The reaction will nevertheless be self-limiting, and so it is not expected to be a major source of heat.

A less obvious source of heat is the inventory of chemical energy stored within the containment. In a sodium cooled fast reactor there is the heat that can be released by burning sodium in the containment atmosphere. In LWRs the coolant is not flammable, but the fuel rods are typically clad in Zircaloy; the oxidation of zirconium releases an amount of heat of the same order of magnitude as the burning of the same mass of sodium.

This heat can be released in one step in the reaction $Zr + O_2 \rightarrow ZrO_2$.

This could occur if core debris still containing zirconium is brought directly into contact with the containment oxygen; for example, following an HPME. It is more likely, though, for the heat to be released in two steps:

steam-Zircaloy reaction: $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$
hydrogen combustion: $2H_2 + O_2 \rightarrow 2H_2O$

The first reaction accelerates the core degradation; the second is potentially dangerous because it can occur explosively. A further source of chemical energy is the oxidation of steel. As with zirconium, this can proceed in two steps: reaction first with steam to produce hydrogen and the later combustion of hydrogen, or in one step if steel comes into contact with oxygen at high temperature. Unlike for zirconium, the steel–steam reaction is not strongly exothermic and so it is not a driving force for core degradation. Specific enthalpies of reaction for the exothermic reactions important in severe accident analysis are listed in Table XXV.

Once heat has been released within the containment, it can reside either within the containment atmosphere, where it contributes to the pressure threat, or within the solid structures, where it does not. An important part of the analysis of severe accident phenomena will consist of an analysis of heat transfer: from the gas to the structures by convection and radiation, and by conduction into the inner parts of the heat sinks. (In some reactor designs, large pools of liquid might also be heat sinks.)

The containment pressure can be increased by heating, but also by increasing the mass of gas. This gas can be water vapour (that released from the concrete in a core–concrete interaction as well as the vaporized coolant), hydrogen (from the zirconium–steam reaction) and other incondensable gases produced in the core–concrete reaction (carbon dioxide and carbon monoxide, if the concrete contains carbonates, e.g. limestone).

In consideration of these general principles, the challenges to the containment can be classified under the following headings:

Slow overpressure: steady buildup of heat and gases in the containment atmosphere;
Rapid overpressure: steam explosions, hydrogen burns, DCH;

Underpressure: condensing of steam in absence of incondensable gases;
Overpressure of annulus: in some containment types overpressurizing of the area between the primary containment and the outer containment/shield building by a pipe break in the event of a core degradation accident and isolation failure;
Overtemperature: degradation of containment systems and structures by elevated temperatures;
Basemat penetration: core debris melting through the basemat of the reactor building;
Missile generation: especially from in-vessel steam explosion or catastrophic vessel failure.

Basemat penetration is generally considered to be a less severe mode of containment failure, because the fission products have to pass through the subsoil before they can reach the external atmosphere. The most likely pathway into the environment is via groundwater contamination. However, for some reactor designs (e.g. Mark I BWR), core debris can attack the containment wall as well as the basemat, leading to a leak path direct to the external atmosphere. This is then a much more severe failure mode.

As well as these containment failures, account must also be taken of the possibility of containment *bypass* and *failure to isolate*. Bypass sequences are those in which fission products find a leak path to the outside without entering the containment. Although the containment is supposed to be a surface completely enclosing the reactor, there is still a need for the routine transfer of heat and matter across the containment boundaries, and the associated pathways can become fission product leak paths. An obvious example is the need to extract heat from the reactor. In a PWR the parts of the containment surface across which heat is conducted are the steam generator tubes. Therefore one of the possible bypass accident sequences is steam generator tube rupture, either as an initiator or as a consequence of the accident progression. Similarly in a BWR a main steam line break outside the containment automatically creates a leak path which bypasses the containment.

Another containment bypass scenario, in both PWRs and BWRs, is the so-called interfacing systems LOCA (V-sequence in WASH-1400 [12]). In this case, the barriers fail between the high pressure RCS and connected low pressure systems with some components outside the primary containment. Some precursors of this type of event have been experienced in the past. Whatever the assessed frequency of these events may be, their consequence would be potentially large releases of radionuclides because they provide a direct path for release of fission products to the environment.

Even if the fission products go into the containment building atmosphere, there will be little retention if there has been a failure to isolate the containment in the first place. This can happen if one of the containment penetrations has inadvertently been left open. Some external event initiators, such as earthquakes and aircraft crashes, may destroy containment integrity at the start of the accident. In general, bypass and failure-to-isolate sequences have source terms at the high end of the range, and for this reason they must not be ignored in a Level 2 PSA.

For existing nuclear power plants a possible objective of a Level 2 PSA is support of accident management. That means that the PSA is used to indicate possibilities for prevention of accidents beyond the design basis and mitigation of their consequences. Accident management would include taking full opportunity to use existing plant capabilities, going if necessary beyond the originally intended functions of some systems and using some temporary or ad hoc systems to achieve this goal.

It must be realized, however, that severe accident phenomena are often highly uncertain or coupled to each other in non-linear ways. An accident management measure designed to mitigate a particular phenomenon might cause or make worse other undesirable phenomena. It is necessary therefore to assess the consequences of each accident management measure. Each proposed action may have disadvantages. For example, the depressurization action to prevent HPME may increase the probability of an in-vessel steam explosion. The operation of a filtered containment venting system might induce underpressure containment failure.

III.3. GENERAL DESCRIPTION OF THE TRANSPORT OF FISSION PRODUCTS

The amount of radioactive fission products present in the fuel is dependent upon reactor power and burnup of the fuel. As an example of typical inventory levels, a 1000 MW reactor contains 100 t of fuel (around 3.2% enriched). For an average burnup of 16.5 MW·d/kgU (11 MW·d/kgU at the beginning of the cycle and 22 MW·d/kgU at the end of the cycle), the total mass of fission products in the core is approximately 1300 kg. Of this, about 250 kg is radioactive, and it gives rise to a total activity of 2.5×10^{20} Bq:

— 2.5 kg of noble gases (Kr, Xe) with a total activity of 1.6×10^{19} Bq;
— 1 kg of halogens (mainly iodines) with a total activity of 3.3×10^{19} Bq;
— 65 kg of alkali metals (mainly Cs) with a total activity of 1.8×10^{19} Bq;
— 0.75 kg of chalcogens (mainly Te) with a total activity of 1.9×10^{19} Bq;
— 2.5 kg of the alkaline earth Ba with an activity of 1.4×10^{19} Bq;
— 33.5 kg of the alkaline earth Sr with an activity of 1.5×10^{19} Bq;
— 13 kg of noble metals (Mo, Tc, Ru, Pd) with an activity of 3×10^{19} Bq;
— 57 kg of the rare earth metals (Y, La, Sm, Pr, Pm, Eu) with an activity of 7.3×10^{19} Bq; and
— 50 kg of Zr and Nb with a total activity of 3.2×10^{19} Bq.

To a first approximation, the fission products constitute a hazard while they are part of the gas phase (fission products in liquid water can also move through the environment, but the dispersal by wind is rather swift and more difficult to protect against in comparison with dispersal through water pathways). Fission products are therefore an obvious concern while they remain as vapours. But they can continue to travel with the gas even after condensing to a liquid or solid form. This happens if they condense as an aerosol of very small particles suspended in a gas.

Fission products, and often substantial amounts of non-active materials, can be released as vapours from an overheated core or core debris. As the vapours are transported into cooler regions they will condense, either onto the walls, or directly as aerosols, or onto aerosols that have already been formed. The actual temperature at which a given fission product element will condense depends strongly on its chemical form. For example, iodine in the form I_2 remains a vapour almost down to room temperature, whereas combined with caesium in the form CsI it will condense at a much higher temperature. Fission product vapours can also react chemically with structures or aerosols. Again this depends strongly on chemical form.

Once they reach the containment, all fission products still airborne will be in the aerosol form, apart from the noble gases, which remain gases under all circumstances, and iodine, whose chemistry determines its partition between vapour and aerosol. The deposition rate of aerosols is a sensitive function of particle size, a fact which explains why so much of fission product codes is concerned with aerosol physics.

Fission products may not be the only aerosols present in the containment atmosphere. The core or its debris can release components of structural materials and the concrete as aerosols. The coolant likewise can give rise to aerosols: in water cooled reactors steam can condense while in sodium cooled reactors the combustion of the coolant would give rise to a concentrated aerosol of sodium oxides. The inactive materials can dominate the total aerosol mass and their presence is important for fission product transport. The fission products can become attached to the inactive particles and share their fate (deposition or leakage).

The dominant mitigating effect for aerosols in the containment is the settling of the particles under gravity. The settling rate is strongly size dependent and so particle growth mechanisms are important. Agglomeration growth (sticking together of particles) is faster when more aerosol material is present. This has two important consequences:

— the presence of inactive aerosol can enhance fission product deposition; and
— larger releases earlier in the sequence can be mitigated by a faster deposition in the containment.

The latter means that earlier uncertainties in fission product release can be damped by containment processes. It is even possible that a large release to the containment leads to a smaller release from it.

Whereas to a first approximation fission products which have been deposited on solid surfaces or in liquid pools within the containment surface have been safely prevented from escaping to the environment, it is possible in the longer term for these fission products to re-enter the gas phase. Resuspension of the aerosol is possible (as dry dust or liquid films being blown off surfaces, or water droplets ejected from boiling pools). Since fission product deposits may still be producing significant amounts of heat, they may become revolatilized. If claims are made for extensive deposition of fission products, care is needed to check that they are not released to the environment at some later time (for example after a very late containment failure).

III.4. OUTLINE OF PHENOMENA

The progression of a severe reactor accident beyond core melt involves a highly complex set of physical and chemical phenomena. The purpose of this section is not to describe fully this complexity, but to place the phenomena within an overall structure, and to use this structure to locate the areas of greatest uncertainty.

To begin with the phenomena can be divided into three areas:

(1) core and core debris behaviour;
(2) fluid flows and heat transfer; and
(3) fission product transport.

The analysis has to follow the behaviour of the core and the core debris until it reaches its final coolable configuration. The heat and mass transfer processes set up by core overheating have to be followed until the heat is into its ultimate heat sink and the gases are in the containment atmosphere or are leaked or vented (or the water vapour has condensed). Finally the fission products released from the core have to be followed until they either leak to the environment or are retained within the reactor building in a coolable configuration.

The phenomena in areas (1) and (2) are strongly coupled. Fission product transport is strongly influenced by areas (1) and (2), but generally has little influence on them in turn. The main exception is the transport of decay heat contributions away from the core.

The area (2) calculations provide input also for the structural response calculations (these latter are generally regarded as being separate from the phenomenology because they are based on standard engineering methods, whereas the phenomena require knowledge of chemistry and physics, often highly specialized). Once the containment has begun to leak significantly, the calculations of structural response and containment thermal-hydraulics will be in principle strongly coupled: the pressure determines the leak size and the leak size influences the pressure.

These phenomena are now considered in more detail, in the approximate order in which they occur during the course of a severe accident. The phenomena are here classified according to another threefold division, based on the location of the events, which is particularly useful for LWR accident sequences:

— phenomena within the RPV and RCS;
— phenomena within the reactor cavity/pedestal; and
— phenomena within the containment building.

The description of each phenomenon is very general, attempting to cover all the things which might occur in a severe LWR accident. An individual sequence will be made up of a subset of these phenomena, as illustrated in Sections I.4.1 and I.4.2.

III.4.1. Phenomena within the reactor pressure vessel and reactor coolant system

(a) *Core heat-up and degradation up to loss of geometry*

When it loses the means of removing decay heat, the core heats up, and the material cladding of the fuel pins can soften. The pressure of the gas can then cause the cladding to swell, an effect called clad ballooning. This can block channels, decreasing heat removal but also hydrogen production. Eventually the cladding will fail, releasing the gaseous contents of the pin. At higher temperatures the cladding material will melt. It can also interact with the fuel, forming low melting point alloys (eutectics), so that degradation of the fuel may begin well below the UO_2 melting point. Unmelted fuel can slump simply because it is not supported by a barrier. If the temperature gets high enough, the fuel can simply melt. Note that in some reactor designs, accidents are possible in which there is a prompt core disruptive event directly after the initiator. These cases go straight to the loss of geometry condition.

(b) *In-vessel heat transfer and fluid flow*

As the core heats up, convective heat transfer to the surrounding coolant vapour will occur. Natural circulation will be set up in the RPV, and possibly around the RCS. Experiments have shown that steam and hydrogen convection velocities in the intact core geometry are about an order of magnitude greater than the steam boil-off velocities. In-vessel natural circulation produces more uniform core temperatures and also transfers more of the core heat to the structure and walls of the upper plenum, the hot leg nozzles and even, by countercurrent flow in the hot leg, the steam generator tubes. There is some indication that this wall heating may cause early failure of the primary system pressure boundary and depressurization of the RCS before melt-through of the RPV in high pressure sequences, e.g. failing of the pressurizer surge line, of one or more steam generator tubes, or of a pump seal. Radiative heat transfer will also be important. Heat will be transferred to the walls of the vessel, especially to the upper head. The prediction of natural circulation in particular is associated with large uncertainties. Codes used in Level 2 PSAs have only a simplified representation of this phenomenon.

(c) *Zircaloy oxidation/hydrogen production*

As the core heats up, the zirconium in Zircaloy cladding begins to react with the surrounding steam to form zirconium oxide and hydrogen. The reaction is highly exothermic, and once it becomes established can be a larger source of heat than the decay heat, becoming the dominant factor governing the time-scale of core melt. Calculations of the reaction rate are complicated by the effect of blockages, reducing steam access, by the formation of an oxide layer, and by the reduction in steam concentration in higher regions of the core due to the reaction in the lower regions. The amount of hydrogen generated during a severe accident is also dependent on the accident scenario and operator actions. Reflooding of a melting core tends to increase the hydrogen generation substantially.

(d) *Coolant circuit heat transfer and fluid flow*

The hot gases leaving the core then flow through the RCS. In low pressure sequences (e.g. LOCAs) they flow to the breach in the circuit. In high pressure sequences (e.g. transients) a natural circulation may be set up (which means more uncertainty in the modelling) with the initiation of the leak (at least in PWRs) via the pressurizer relief valve. Heat transfer to structures along the leak path has to be considered. The carrier gas may be almost all hydrogen, which has different heat transfer characteristics from steam. Radiative heat transfer along the axis of the gas flow can also be important. In sequences with very low flows of coolant derived gases (e.g. the steam–hydrogen mixture in LWRs), it is possible that the bulk of the flow comes from fission product gases and vapours. These will have different thermodynamic and transport properties from the gases normally treated in RCS thermal-hydraulic calculations. The fission products are also a source of heat as they move; deposition of fission products could cause the failure of vulnerable parts of the RCS.

(e) *Fission product release from fuel in the vessel*

When the cladding fails, the fission products that have already moved into the gap between fuel and cladding are released. This so-called 'gap release' is important when the core is recovered before melt, but in severe accidents it is small compared with the subsequent 'melt release'. As the fuel heats up towards its melting point the noble gases and the more volatile fission products will be driven off. The release can be modelled either using experimentally determined rates (functions of temperature) or mechanistically: modelling the chemical forms of the fission products in fuel and their diffusion through the solid matrix. The important volatile fission products are iodine, caesium and tellurium. Since there is typically ten times more caesium than iodine, usually all of the iodine is released as caesium iodide and the rest of the caesium as caesium hydroxide. Tellurium is released in the elemental form, but it can react with the tin component of the Zircaloy, thus being retained in the core debris. As well as fission product release, the vaporization of structural and control rod material also needs to be considered, because these can form the bulk of the aerosol which subsequently carries the fission product. CsI is thermodynamically stable up to at least 2000°C in the system UO_2–Zircaloy–steam. The remainder of the Cs of the inventory would be released from the fuel mainly in elemental form and then could react with steam to form very stable CsOH gaseous molecules. Above 2000°C, CsI will react with steam to form CsOH + HI. There is also strong experimental evidence that CsI reacts with boric acid. Boric acid (an additive to coolant for reactivity control) is present in the coolant and emergency cooling water, or is produced by the decomposition of B_4C control rods. If this reduction occurs to a significant extent in the RCS, HI and caesium borates would be produced. Both CsOH and HI are more volatile than CsI. HI is also chemically reactive, and would interact with circuit surfaces and aerosols.

(f) *Fission product transport in reactor coolant circuit*

The noble gas fission products will move with the other gases through the RCS (unless in very low flow cases they can form a stratified layer). The other fission products can either be deposited in the RCS or be transported into the containment (or elsewhere, in bypass sequences). As the carrier gas cools going away from the core, the least volatile substances, usually structural or control rod materials, will condense as an aerosol (the process by which aerosols are formed directly from the vapour without pre-existing particles is called homogeneous nucleation). The uncertainties here are high, in the modelling of both homogeneous nucleation rates and also the steep thermal gradients above the core. PSA codes are likely to require the user to specify an initial aerosol mass and particle size distribution. It may be necessary to test the sensitivity of the final outcome to the choice of these parameters. The fission product vapours may react chemically with either the wall or with aerosols or, when the temperature is low enough, condense on walls or aerosols. Both reaction and condensation depend on the chemical form of the fission products. Simple PSA codes may have only a few fixed species: e.g. CsI, CsOH and Te. More mechanistic codes may have more species and model reactions, for example the reaction between CsI and boric acid to form caesium borate and hydrogen iodide, the latter being a more volatile and reactive form of iodine than caesium iodide. A code dealing with fission product transport in the RCS also needs to model aerosol agglomeration and deposition (discussed in more detail under item (u)).

(g) *Core degradation and loss of geometry*

During the heat-up process, the first failures in the core typically occur in the control rods. For silver–indium–cadmium control rods for a PWR, failure occurs near the 1700 K melting point of the stainless steel cladding of the control rods. The cadmium rapidly vaporizes at rod failure and condenses into an aerosol when cooled outside the core. The molten silver and indium relocate downwards with no interaction with the stainless steel guide tubes of the control rods. Eventually they will either freeze in the colder regions of the core or fall into the water of the lower plenum. If they fall into the water, more steam is produced, which may temporarily overcome steam starvation. The molten stainless steel interacts strongly with Zircaloy and Inconel (rod spacer grids) to form eutectics at about 1500 K. The Al_2O_3 in the Zircaloy clad and burnable poison rods forms eutectics with Zircaloy at 1750 K and with both ZrO_2 and UO_2 at about 2200 K. For BWRs, eutectics occur between B_4C and stainless steel even at temperatures as low as 1500 K. The stainless steel cladding of the control blades liquefies, with subsequent relocation and possible formation of blockages by the liquefied material.

The fuel rods normally fail when molten unoxidized metallic Zircaloy breaks the ZrO_2 surface sheet produced by oxidation of the cladding. The molten metallic Zircaloy then relocates downwards along the individual rods in a 'candling' process. This process removes the supply of metallic Zircaloy for oxidation from the high temperature region of the core where oxidation can occur, effectively limiting the rapid temperature rise and the rapid hydrogen generation from autocatalytic oxidation of the initially intact fuel rods. This relocation of the molten unoxidized metallic Zircaloy is the first of three significant and distinct material relocation processes that occur during in-vessel core melt progression.

Near its 2100 K melting point, molten metallic zirconium can dissolve up to 10% of its mass of solid UO_2. Near the 2700 K liquid monotectic point this rises to 20%, and above this temperature the dissolved mass fraction goes up to 80%. This 'liquefied' fuel relocates downward and freezes on colder portions of the fuel rods and rod spacer grids. As water boils off and core melt progression proceeds, this solidified material may remelt and relocate downward again in a repetitive process. This process was responsible in the accident at Three Mile Island Unit 2 for the formation of the tough 'hard pan' across the mid-region of the core.

After the initial autocatalytic oxidation transient and relocation of the molten metallic Zircaloy (and dissolved UO_2), free standing columns of declad, stacked, cracked ceramic (UO_2, ZrO_2) fuel pellets in essentially the original rod geometry remain. The later collapse of the ceramic pellet columns is the second major material relocation process in core melt progression. This collapse forms a rubble bed on top of the layer of frozen relocated Zircaloy and liquefied fuel and substantially changes the thermal characteristics of the debris, including its flow resistance. The natural circulation flow from the upper plenum into the damaged core is virtually eliminated by this collapse.

As steam boil-off continues, the debris region, which consists of frozen relocated Zircaloy and liquefied fuel in the fuel rod stubs at the bottom and mostly ceramic particulate rubble above, is heated by fission product decay and probably by some continued oxidation of the relocated Zircaloy. Because of the surface heat removal, melting starts near the centre of the debris region, and increasing loads are imposed upon the lower crust and the core support structure. The third major material relocation comes with failure of the lower support crust, or possibly first the core support plate, with slumping of the molten core material into the lower plenum and quenching of the surface of the melt mass by the lower plenum water. This picture of the meltdown events has emerged from examination of the core at Three Mile Island Unit 2, which has provided very useful insights into what can happen.

(h) *In-vessel core-coolant interaction*

The slumping of the core into the lower plenum of the RPV might be instantaneous or more gradual by pouring of molten fuel. During the quenching process, copious quantities of steam are generated, producing a steam pressure spike, and oxidation of the molten unoxidized Zircaloy can generate considerable additional amounts of hydrogen. The speed of this quenching process might range from slow, via moderately fast giving rise to a pressure spike, up to extremely rapid quenching with an explosive character (steam explosion). It is well known from accidents in foundries that pouring large quantities of melt into water pools can lead to an explosion. A steam explosion might occur when the molten core mass slumps into the lower plenum water in lower pressure melt sequences (the higher the pressure, the lower the probability that a steam explosion will occur). For this to happen the melt has first to fragment. The particles will form a layer of steam around themselves which reduces heat transfer. A small shock can trigger the disruption of this semistable state. During a few milliseconds a very high heat transfer rate from the fuel to the surrounding water occurs, resulting in an explosive phase transition from water into steam. As the growing shock wave moves through the system, it strips away the steam layer and further fragments the melt, and the consequently greatly increased heat transfer will amplify the shock wave further.

Because detonation requires a highly specific set of circumstances, its probability is generally considered to be low. However, the possibility must not be ignored completely. An energetic steam explosion can deliver significant shock loads, possibly failing the vessel lower head, and can significantly redistribute the core debris. It is also possible that the shock wave may accelerate the debris still left on the core support plate. These internal missiles, together with the shock wave, might fail the RPV head which may be further accelerated. In the worst case such a missile generated by an energetic in-vessel steam explosion might fail the containment (the so-called alpha mode of containment failure).

(i) *Vessel melt-through*

Once the debris in the lower head becomes uncoolable (either by boiling off the water there or by forming an uncoolable layer under water), it will begin to attack the structure of the lower head. If the PWR design has the core instrumentation cabling coming through the lower head, the instrument penetrations may be attacked first. If the debris and/or melt is stratified, say with ceramic fuel material beneath and silver metal from the control rods above, the attack may come at the level of the silver layer, because of the high thermal conductivity of the metal. In BWRs local melt-through may occur, owing to the numerous control rod drive and instrument penetrations, while most of the debris is still solid, leading to depressurization before debris relocation. Modelling these processes is important in determining the timing of vessel failure and also the way in which the core debris/melt enters the reactor cavity.

(j) *Vessel lift-off*

For some reactor types another high pressure core melt scenario may be important. This is the so called 'reactor vessel launch' (rocket) scenario. In this case the RPV fails catastrophically during a high pressure core melt scenario prior to vessel melt-through, because of a sudden and complete failure of the lower circumferential welding. The melt will heat the inner surface of the RPV, and the consequential high temperatures may after a longer period (2000 s) lead to plastic deformation of the material by creep (if there are penetrations in the lower plenum of the RPV, such as for instrument lines or control rods, then vessel failure by failure of the penetrations could occur before elastoplastic deformations can take place). High pressures may then lead to a sudden rupture of this lower welding. If the cavity is small, the sudden release of steam and ejected molten core material will cause the RPV to experience high upward forces due to the back pressure. If the pressures of the primary circuit before vessel failure are greater than 3 MPa (30 bar), both the anchoring of the RPV on the pedestal and the anchoring due to the connected piping of the primary loop could fail. At pressures in the range of 8 to 10 MPa, the launched upper part of the RPV could cause the containment to fail.

III.4.2. Phenomena within the reactor cavity/pedestal

(k) *Debris ejection from vessel (gravity drop or high pressure melt ejection)*

The ejection rate of the melt and solid debris into the reactor cavity upon vessel failure is dependent upon the mode of vessel failure (pressurized, via instrument-line nozzles in the lower head, sudden total failure of the lower circumferential weld, etc). It also depends on the pressure in the RCS prior to failure. If the pressure is low, the melt will fall under gravity into the reactor cavity. On the other hand, if the melt is sprayed out under pressure, it can be distributed more widely through the cavity and even out into the containment. As the extent of core debris dispersion increases, the risk of core-concrete attack decreases. However there are a number of highly undesirable effects of debris dispersion. A very finely dispersed melt can add a large amount of heat rapidly into the containment atmosphere, creating an overpressure and posing a serious threat to the containment. This phenomenon, direct containment heating, is discussed in the following.

In recent years there have been many studies of HPME to see how much of the melt would be retained in the cavity rather than being transported into the containment. The answers are very sensitive to details of cavity geometry. Apart from the rapid pressure increase in the containment, HPME might have other disadvantages compared with melt-through at low pressure. Dispersal of the melt out of the cavity may damage penetrations and might also have a negative impact on the recirculation system for cooling the containment. Because of the risks of DCH, considerable thought has been given to adopting deliberate RCS depressurization prior to vessel failure as an action to mitigate the effects of high pressure accident sequences. However, one must consider also the fact that lowering the RCS pressure may increase the likelihood of in-vessel steam explosion.

Dispersing core debris can induce other hazards. If hydrogen is present in the containment atmosphere, dispersion of hot debris particles could serve as a catalyst to promote recombination of the hydrogen with free oxygen even though the H_2 concentration may be below the conventional flammable limit. Hydrogen recombination will generate more energy to raise the pressure and the temperature in the containment. The issue would be further complicated if the reactor cavity is filled with water at the time of the RPV failure. The pressurized stream of molten core materials might cause a steam explosion that might contribute to debris fragmentation and promote dispersion, at the same time causing dynamic loading of the containment.

(l) *Direct containment heating*

If HPME does result in core melt particles being injected into the cavity or containment atmospheres, the rate of heat transfer from these particles must be considered in order to estimate the threat to the containment. In addition, the exposure of the core melt to the oxygen of the containment atmosphere may result in oxidation reactions, which generate still more heat. Any zirconium that has not been oxidized by steam in-vessel may now react, as may structural steel within the melt. If the containment atmosphere contains a flammable mixture of hydrogen, the HPME may ignite it, with the coincidence of DCH and hydrogen burn producing an even bigger threat to the containment.

(m) *Ex-vessel core-coolant interaction*

If the melt falls or is ejected down into the cavity under the vessel, it may in certain sequences and reactor types encounter water there. There is then a possibility again of a steam explosion. The pressure wave generated by the explosion could be a threat to the containment. In the scenario with water in the cavity at HPME, it is also possible that some of the water is driven out as a slug from the cavity. Water on cavity at melt-through causes a pressure spike in the cavity due to rapid evaporation of water. The pressure peak in the cavity is in this case much higher than in the containment.

(n) *Fission product release in steam explosion or high pressure melt ejection*

If core debris is finely fragmented in the containment atmosphere, either by steam explosion or HPME, an additional release of fission products may occur. One mechanism can be oxidation by the oxygen or steam in the atmosphere; this release is sometimes called the 'oxidation release'. Ruthenium could be particularly susceptible to this release, because although most of its chemical forms are quite refractory, the oxide RuO_4 is much more volatile.

(o) *Core-concrete interaction and gas production*

If the core melt and/or debris has fallen onto the concrete at the bottom of the cavity in an uncoolable configuration, then it will begin to deposit its heat into the concrete. As the concrete heats up it will begin to break up physically and decompose chemically. The chemical reaction will produce gases: water vapour and, if the concrete is limestone based, carbon monoxide and carbon dioxide. The extent of the ablation of the concrete and rates of gas production are strongly dependent on the composition of the concrete aggregate. Basemat penetration is generally considered to be a less severe mode of containment failure, because the fission products have to pass through the subsoil before they can reach the external atmosphere. The most likely pathway to the environment is via groundwater contamination. After penetration to a few metres into the subsoil, the melt will be in thermal equilibrium with its direct environment, and will remain so. If, however, there is a pathway to the external atmosphere just outside the cavity, then cavity penetration by core–concrete interaction can lead to containment failure.

(p) *Release of fission products from debris in core–concrete interactions*

As the gases produced in the interaction between the molten core debris and the concrete bubble up through the melt, they can take fission product vapours and aerosols with them. The volatilities of fission products depend on the chemical conditions in the melt. The melt may stratify into an oxidic layer, in which the conditions are oxidizing, and a metallic layer, in which conditions are reducing. To begin with, the oxidic layer may consist largely of uranium oxide, and would therefore be heavier than the metallic layer and lie underneath it. But as concrete decomposition proceeds, this oxidic layer may get lighter and lighter as the concentrations of calcium and silicon oxides increase, and the order of the layers may reverse. Alternatively, the passage of the gas through the melt may act to mix the oxidic and metallic components, producing more uniform chemical conditions throughout the melt. In addition to fission product volatilization, the core–concrete interaction may produce considerable quantities of non-active aerosol, which may influence fission product transport at later times. If there is a pool of water overlying the melt, bubble scrubbing may act to reduce the release of fission product and aerosol to the containment (see item (*v*) for a discussion of bubble scrubbing).

(q) *Debris quenching*

After the core–concrete interaction has begun, water may find its way into the cavity, or it may be introduced there deliberately in an attempt to cool the debris and stop the interaction. To predict whether the water does cool the debris, one must calculate heat transfer through the debris and into the water layer. In particular, the possibility must be examined that an insulating crust might form on top of the debris and prevent cooling.

III.4.3. Phenomena within the containment building

(r) *Containment thermal-hydraulics: steady pressurization*

In the absence of events causing sudden pressure increases, the pressure in the containment atmosphere is governed by a balance between the addition of heat and the injection of gases into the atmosphere and the transfer of heat to, and condensation of steam onto, the walls. If a core–concrete interaction occurs, and without the intervention of engineered safety features or venting, the pressure will almost certainly continue to rise towards the containment failure pressure; heat conduction through concrete walls cannot compensate for the heat sources in the core–concrete interaction. If the containment leaks before breaking, then the associated loss of gases will modify the thermal-hydraulics for the containment. The transfers of heat to the walls are strongly influenced by heat conduction through the walls (the only route out of a closed containment to the ultimate heat sink, the external atmosphere). A thermal–hydraulics code for the containment needs to be able to model these transfers and calculate the temperature and composition of the atmosphere as a function of time. In the containment, newly injected hot gases will tend to rise, and the gases in contact with the walls will tend to sink as they lose heat and steam. These effects will tend to mix the containment atmosphere, but there may be cases in which they are not strong enough to do this and the atmosphere will become poorly mixed (especially for complicated containment geometries). This may in turn change the heat transfer and condensation rates to the walls, and thus indirectly affect the pressure. Thermal–hydraulics codes for the containment that divide its atmosphere into control volumes and can treat the circulation patterns (or lack of them) are available. However, in using them it must be ensured that the patterns predicted are not sensitive to the number and locations of control volumes chosen.

(s) *Hydrogen combustion*

During core melt accidents, Zircaloy as well as the other in-core metallic materials react at high temperatures with water or steam. Consequently large amounts of hydrogen are produced. For example, 880 kg of hydrogen would be produced if the total Zr mass (around 20 000 kg) of a 1000 MW(e) PWR were oxidized, and 2800 kg of hydrogen would be produced if there were total oxidation of the Zircaloy in a 1000 MW(e) BWR core (65 000 kg Zr). In the in-vessel stage of the accident, between 20% and 80% of the Zircaloy may be oxidized and, upon the failure of the primary circuit, released into the containment. This may occur gradually if the primary loop were to fail prior to total core melt, or suddenly at vessel rupture. The rest of the Zircaloy will be oxidized during the core–concrete interactions (concrete contains around 6.5% water). Large amounts of hydrogen will also be produced if the melting process is arrested in the RPV owing to reflooding of the core.

Regarding ignition of the hydrogen, three different rates of combustion can be recognized: local burning by diffusion flames, deflagration and detonation. Deflagration is a form of combustion in which the flame moves at subsonic speed relative to the unburned gas. Unburned gas is heated to reaction temperature by thermal conduction and mass diffusion from the hot burned gas. Local burning as well as deflagration may cause static or quasistatic pressure loads on the containment owing to the extra heating of the containment atmosphere. Hydrogen detonations involve the reaction of hydrogen through the supersonic propagation of a burning zone or combustion wave. The pressure loads developed are essentially dynamic

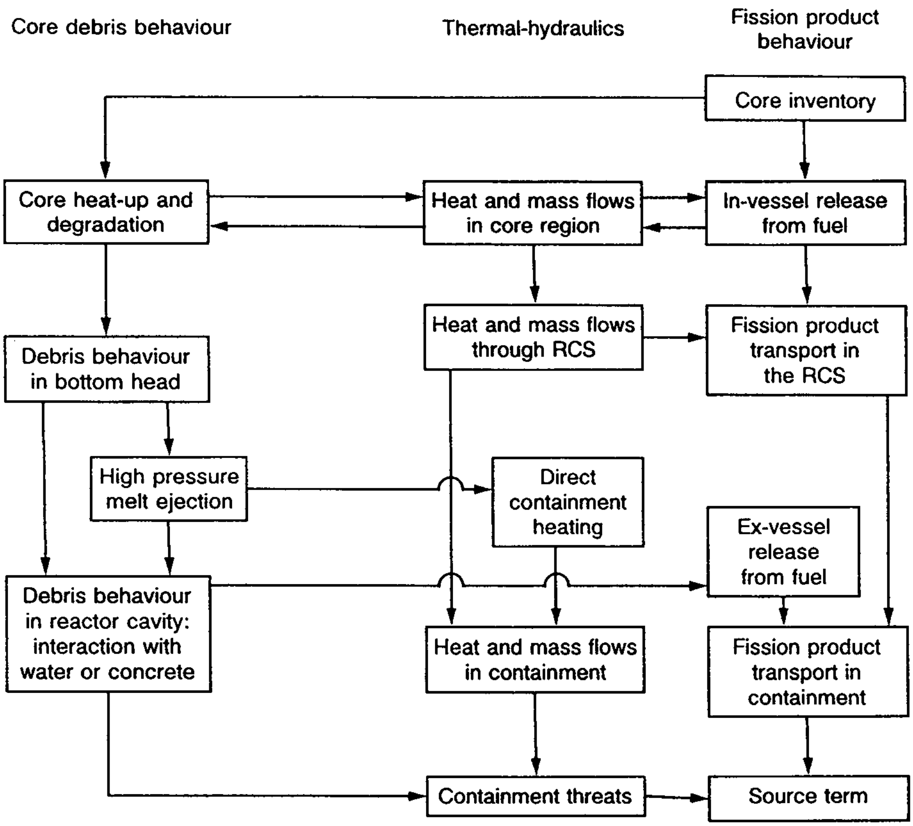

FIG. 7. Concentration limits for hydrogen–steam–air mixtures.

loads. These dynamic pressure loads due to detonations may cause a breach of the containment or damage to important safety related equipment.

The areas of different combustion rates are primarily dependent on the concentration of hydrogen, as well as on the concentration of steam and other gases such as CO or CO_2. The initial temperature and pressure of the gas are also important factors. In Fig. 7 the concentration limits are depicted for hydrogen–air–steam mixtures.

If a sufficiently strong energy source is available and the gas mixture is detonable, a global detonation may occur. It is, however, very unlikely that such an ignition source will exist, and therefore a global detonation has a low probability.

In some cases transitions from deflagration to detonation can occur. The potential of a transition from deflagration to detonation depends on the composition of the gas and the containment geometry. The presence of obstacles or other sources of turbulence in long confined passageways will promote acceleration of the flame front.

In some cases these accelerations may eventually lead to supersonic speeds, thus providing a transition from deflagration to detonation.

The presence of other diluent gases, such as steam or CO_2, however, reduces the likelihood of detonations. This effect can be used as a possible accident management strategy. Filling the containment with another diluent and inert gas (both pre- and post-accident inerting may be considered) can prevent detonation. A disadvantage of preaccident inerting is the inaccessibility of the containment during normal operations for maintenance activities, refuelling preparations and so on. Other possibilities for preventing or mitigating the effects of hydrogen detonations and deflagrations is deliberately to ignite or recombine the hydrogen before dangerous concentrations can be reached.

Chemical effects and aerosol generation are also possible consequences of hydrogen combustion (or a steam explosion). Of all the fission products, iodine is expected to be most affected chemically by hydrogen burn. Both hydrogen combustion and steam explosions will have an impact on the aerosols in the containment. The energy deposition in the gas may promote turbulent agglomeration and may also lead to changes in aerosol sizes. Steam explosions are not likely to lead to substantial aerosol generation but may in fact lead to phenomena that reduce existing aerosol concentrations. The phenomenon of HPME could lead to significant aerosol generation.

Hydrogen combustion, in addition to generating very high temperatures at the moving flame front (1000–2200°C), also generates large transient concentrations of reactive radicals. These radicals may react with radionuclides in either airborne molecular or particulate form, for example:

$$CsI + OH^- \rightarrow CsOH + I^-$$

where the I atoms subsequently form I_2 or HI. However, the airborne lifetime of the I_2 or HI will be quite short. I_2 molecules will be subject to plateout on the aerosol particles, which are present after the H_2 burn. Also HI is expected to react rapidly with aerosol Ag to form AgI. The iodine on new aerosol particles will be subject to the same aerosol depletion mechanisms as the original aerosol borne CsI and only a fraction of the I_2 or HI will persist for a longer time as airborne iodine.

(t) *Engineered safety features*

Some containment systems have engineered systems to remove heat and steam from the containment atmosphere: water sprays, fan coolers, ice beds. Their influence has to be included in the containment thermal-hydraulics calculations for the sequences in which they operate. The problem of hydrogen deinerting by removal of steam has been mentioned earlier. Another potential problem connecting hydrogen burn and engineered safety features is flame acceleration. If a hydrogen flame encounters a region containing obstructions, the turbulence induced by the obstructions can increase the burning rate and even cause a transition to detonation. Fan coolers, which not only induce turbulence in the gas but also remove steam from the atmosphere at the same time, may be of concern when operating in the presence of hydrogen at elevated levels.

(u) *Transport of fission products in the containment: natural*

In the containment atmosphere the noble gases krypton and xenon will mix with the other gases. All the other fission products except iodine, if airborne, will be in the form of aerosols. For these the important factor is how fast these particles deposit onto surfaces. The deposition rate depends strongly on particle size. Particles with a radius of 0.5 to 1 μm can remain suspended for many hours. The rate at which they are deposited depends on the rate at which they can grow larger. One growth mechanism is agglomeration, by which particles collide and stick together. If the conditions are right, steam can condense onto aerosol particles, which is an effective means of washing them out of the atmosphere. However, this condensation process is very difficult to model; unless used with great care, existing models for the process could lead to unrealistically large deposition rates, and therefore overoptimistic source terms. Iodine has to be treated as a special case. In the form of CsI it will be part of the aerosol, but if it reacts (for example under the influence of radioactivity) to form volatile species such as HI, I_2 and possibly IOH, it will be partitioned between vapour, aqueous aerosol and deposited water.

(v) *Retention of fission products in liquid pools: bubble scrubbing*

In some sequences for some reactor designs, the gas laden with fission products may be bubbled through water pools (e.g. BWR suppression pools). There is then the possibility of bubble scrubbing, the transfer of fission products from the gas into the pools. Codes modelling this effect consider bubble rise and breakup, heat and steam transfer to pool, and aerosol and vapour removal to the pool. These models are quite well established for low flow rates and therefore isolated bubbles. Uncertainties grow at higher flow rates, when the bubbles form swarms, and ultimately merge to produce churn turbulent flows. As well as being sensitive to the nature of the bubbles, the aerosol decontamination factor depends strongly on the size of the aerosol particles within the bubbles.

(w) *Transport of fission products in the containment: effects of engineered safety features*

Although the main purpose of the engineered safety features within the containment is to remove heat and steam, their operation can also accelerate the removal of fission products from the containment atmosphere. Water spray droplets can absorb iodine vapours and capture aerosol particles. When heat and steam are transferred to a surface, aerosol deposition on that surface is accelerated. Ice condensers and fan coolers can therefore enhance aerosol removal. However, if such aerosol deposition is claimed, or even suspected, it must be considered whether the functioning of the device is threatened by the deposit. Can fans continue to operate and are coolers still efficient? Can the spray recirculation system tolerate insoluble aerosol particles suspended in the water?

(x) *Retention of fission products in the leak path*

As the pressure within the containment increases, leak paths may open up (this is termed 'leak before break'). The final failure of containment, that is, the one which prevents a further rise in pressure and may even cause the pressure to fall to atmospheric pressure, may still involve comparatively narrow and tortuous leak paths (e.g. paths around failed penetration seals) or may open a large hole. In the latter case the fission products are released to the environment with no further retention, but in the other cases there can be deposition of aerosols in leak paths. Not only does this mean that some of the aerosol is prevented from reaching the environment, but also there is the possibility that the aerosol may plug the leak path, preventing all further aerosol and maybe even gas leakage. There is good evidence that aerosols are good at plugging narrow capillaries and slits. The problem, however, in claiming leak path retention and plugging in a PSA is the uncertainty in the nature of the leak paths.

(y) *Containment venting*

Because of the uncertainties in the timing and mode of containment failures, for a number of plants the decision has been made to install a vent as the ultimate protection system. This trades off an increased probability of a smaller release of fission products (short of having a cryogenic device, noble gases cannot be filtered out) against a decreased probability of a larger uncontrolled release. To include the effect of the vent on the containment thermal-hydraulics, the vent rate must be known as a function of containment pressure. Also the adiabatic cooling of atmosphere by the sudden expansion and the boiling of water pools that were in equilibrium with the containment atmosphere prior to venting and are superheated relative to the depressurized atmosphere must be taken into consideration.

If the vent is provided with a filter, the calculation of the source term requires a knowledge of the efficiency of the filter. It must also be considered whether a reduction in flow rate and degradation of filter efficiency can occur after large quantities of aerosol have been deposited. In addition, combustion processes within the filter path also need to be considered. For instance, hydrogen concentration in vent

TABLE XXVI. LEVELS OF UNCERTAINTY AND SENSITIVITY IN SEVERE ACCIDENT PHENOMENA

	Severe accident phenomenon	Intrinsic uncertainty	Sensitivity		
			Accident management	Containment challenge	Source term
(a)	Core heat-up degradation	M (e)	✓	✓	
(b)	In-vessel thermal-hydraulics	M	✓		
(c)	Hydrogen production	M	✓	✓	
(d)	RCS thermal-hydraulics	L	✓	✓	
(e)	In-vessel release of fission products	M			✓
(f)	RCS fission product transport	M			✓
(g)	Core loss of geometry	H (e)	✓	✓	✓
(h)	In-vessel core coolant interaction	H (e)		✓	
(i)	Vessel melt-through	H (e)	✓	✓	
(j)	Vessel lift-off	L		✓	
(k)	Debris ejection from vessel	H	✓	✓	
(l)	Direct containment heating	H (e)	✓	✓	
(m)	Ex-vessel core–coolant interaction	H (e)		✓	
(n)	Release of fission products in HPME	M			✓
(o)	Core concrete interaction	M	✓	✓	✓
(p)	Ex-vessel release of fission products	H			✓

TABLE XXVI. (cont.)

Severe accident phenomenon	Intrinsic uncertainty	Sensitivity		
		Accident management	Containment challenge	Source term
(q) Debris quenching	M	✓		
(r) Containment thermal-hydraulics	L		✓	
(s) Hydrogen combustion	H		✓	
(t) Engineered safety features	L		✓	
(u) Transport of fission products in containment	M			✓
(v) Pool scrubbing	M			✓
(w) Effects of engineered safety features on fission products	L			✓
(x) Leak path retention	H			✓
(y) Containment venting:				
unfiltered	L	✓		
filtered	L			✓
(z) Resuspension	M			✓

Notes: H: high; M: medium; L: low. Areas in which expert judgement is most likely to be necessary are indicated by (e).

paths could reach levels at which hydrogen deflagration and even detonation can be of concern. In other cases, heating up of the filter as a result of radionuclide deposition may produce conditions that can produce delayed releases of fission products owing to revaporization processes. If charcoal filters are employed, charcoal surface temperatures reaching autoignition levels can cause charcoal fires.

(z) *Revolatilization and resuspension*

Underlying the source term calculations is the assumption that suspended fission products are still a hazard to the environment, while those deposited within the reactor building somewhere are not. However, the fission products still have their own heat source and cannot be ignored until they are in a configuration which is coolable in the long term. This is particularly true if there has been heavy deposition in a restricted location. For example, in turbulent flows aerosol deposition may be concentrated at pipe bends. Such deposits could reheat and release vapours, or even melt their way through the pipe. Deposits of fission products in the containment may be concentrated in water pools. Continuing release of decay heat may eventually evaporate these pools, and then the fission products left behind may revolatilize. This may occur a long time after a very late containment failure that appeared to be relatively innocuous because of the low level of fission products in the atmosphere at the time. Filters are designed to concentrate fission products; what happens in the filter at later times needs to be considered.

III.5. UNCERTAINTIES AND SENSITIVITIES

Severe accident phenomena are highly complex, and so there are significant uncertainties in the prediction of what will happen. Often knowledge is far from perfect. Some phenomena may be quasi-stochastic, so that there would remain objective uncertainties even if knowledge were perfect.

As well as stating what is uncertain, it is also necessary to indicate which uncertainties are important to the final result. This is the question of sensitivity: how much does the final answer change if parameters are varied across their uncertainty range? The most important thing to know is the region of overlap between the greatest uncertainty and the greatest sensitivity.

Table XXVI shows a subjective appreciation of the levels of uncertainty and sensitivity in the phenomena identified in Section I.4. No great weight is to be given to individual judgements on the table; more important is the overall pattern that emerges. The uncertainties are graded high (H), medium (M) or low (L). These are relative judgements: in some cases where an M mark has been given, the uncertainty may be high, but the judgement is that this uncertainty is not as important as some of the other issues.

A distinction is made between intrinsic and consequential uncertainties. When one phenomenon depends on initial or boundary conditions set by an earlier phenomenon, it will inherit the uncertainties from that earlier phenomenon. Here it has been attempted to judge intrinsic uncertainties: those that would persist even if the initial conditions were known exactly. In addition those areas in which appeal to expert judgement is most likely to be necessary are indicated with '(e)'.

In discussing sensitivity, a distinction has to be drawn between different outputs and end uses of the calculation. In Table XXVI we consider three: assessing accident management procedures, assessing challenges to the containment and calculating source terms.

Annex I

TC1 SEQUENCE FOR THE PEACH BOTTOM NUCLEAR POWER PLANT

Annex I describes a low pressure accident sequence in a BWR. The example given here is the prediction of the course of events for the TC1 accident sequence for the Mark I BWR at Peach Bottom in the USA. The example is taken from document BMI-2139 [I-1], and is based on the STCP [I-2] calculations performed by staff of Battelle Columbus Laboratories, Ohio, USA. In these summaries all the quantities have been converted to SI units; the times of the events are retained in minutes. For the present purposes the actual numbers are not of prime interest; the objective here is to show the sort of results to be expected from a calculation of accident phenomena.

The sequence of events is predicted to be as follows.

t = 0: initiator

The TC1 accident sequence is initiated by a failure to achieve reactor shutdown, the closure of the main steam line isolation valve and the depressurization of the RCS. The reactor power remains at 21% of nominal power. Because of the venting of the RCS into the primary containment, the pressure in the primary containment builds up, eventually causing a primary containment failure.

t = 85 min: primary containment failure

At this point the RCS pressure is 7.8 MPa and the primary containment pressure and temperature are 0.89 MPa and 435 K respectively.
Once the primary containment fails, the suppression pool flashes, causing failure of the emergency core cooling pumps.

t = 86.7 min: failure of emergency core cooling pumps

With the emergency core cooling pumps gone, it is only a short time until core uncovering.

t = 93.8 min: core uncovering

At this point the average core temperature is 667 K, and the primary and secondary containment pressures are 0.67 MPa and 0.109 MPa respectively.

In spite of the failure of the primary containment, its pressure is kept high by the continued boiling of the suppression pools. The pressure in the secondary containment (the reactor building) may already be sufficient to compromise the integrity of the refuelling building.
As the core is uncovered, the power falls rapidly to decay heat levels. The primary containment pressure falls rapidly owing to leakage through the pressure relief valve. The core heats up until it begins to melt.

t = 134 min: start of core melt

The primary containment pressure has fallen to 0.163 MPa but is still not down to the secondary containment value. The average and peak core temperatures are 1475 K and 2550 K respectively.
In this analysis the meltdown is assumed to proceed according to the MARCH Model A, that is, by gradual or regional slumping. Molten fuel is allowed to fall out of the core region when the lowest node in a radial region is fully molten. All molten nodes in that radial region are then assumed to slump onto the lower core support structure. On the controversial question of whether melting blocks the flow of steam and thereby stops hydrogen production, the analysts chose to allow the metal steam reactions in the molten nodes. According to this model core relocation begins at core slump.

t = 166.8 min: core slump

By this time the average core temperature is up to 2123 K while peak temperatures are still 2550 K. 30% of the core has melted and 18% of the clad has reacted with steam to form hydrogen.
The meltdown model used allows the entire core, including unmelted portions, to collapse when 75% of the core is molten, or when the lower core support structures reach their melting point. The progressive meltdown of the core continues until the core collapses.

t = 172 min: core collapse

At this time 39% of the core has melted and 26% of the clad has reacted. The collapse is due to the overheating of the support structures. By this time the primary containment pressure has come down to 0.110 MPa, almost the same as in the reactor building.
For low pressure sequences like this one there may be little driving force for expulsion of the debris from the primary containment and a large fraction of the core support structures may overheat and be absorbed into the debris before bottom head failure. As the debris falls into the bottom head it interacts with

the water there, being partially cooled in the process, until the bottom head dries out.

t = 201.9 min: bottom head dryout

At this time the debris temperature has fallen to 1770 K. The debris then heats up again, until the bottom head fails.

t = 230 min: bottom head failure

The debris temperature is then back to 2017 K.
Because the primary containment fails at low pressure, there is little dispersal of debris in the containment atmosphere. Instead the debris falls onto the dry well floor and a core–concrete interaction begins. During this interaction, the analysis predicts that the melt layers invert.

t = 400.4 min: melt layers invert

The initially heavier oxidic layer becomes progressively less dense until the melt inverts, with the metallic layer sinking to the bottom.
During the period of core concrete interaction, the secondary containment is predicted to suffer hydrogen burns.

t = 383 min to 510 min: 11 hydrogen burns

Even without deliberate hydrogen ignition, it is unlikely that very high concentrations of hydrogen could be developed. The recommended value of 8 vol.% of hydrogen was used as the ignition value.
Even if the earlier overpressure events have not already compromised the integrity of the refuelling building, these hydrogen burns will do so. Pressure peaks of up to 0.15 MPa are predicted even with a 30 m^2 opening in the building. Each hydrogen burn will cause a volumetric leak from the secondary containment equal to the volume of the building, effectively ejecting its entire contents.
For the source term calculation this history divides into two periods:

t = 93.8 min to 230.5 min: in-vessel release

The fission products and other materials released as vapours or aerosols from the core or core debris take the following leak path to the environment:

RCS → suppression pool → wet well → dry well → reactor building → environment

TABLE I-I. FRACTIONAL DISTRIBUTION OF FISSION PRODUCTS BETWEEN LOCATIONS FOR THE TC1 SEQUENCE FOR PEACH BOTTOM NUCLEAR POWER PLANT

Species	Melt	RCS	Pool	Dry well	Reactor building	Environment
I	—	0.15	0.78	0.01	0.03	0.03
Cs	—	0.16	0.76	0.01	0.04	0.03
Te	0.31	0.25	0.14	—	0.04	0.26
Sr	0.25	—	0.12	0.01	0.13	0.49
Ru	1.00	—	—	—	—	—
La	0.98	—	—	—	0.01	0.01
Ce	0.96	—	0.01	—	0.01	0.02
Ba	0.42	—	0.09	0.01	0.09	0.39

t = 230.5 min onwards: ex-vessel release

The materials released in the core–concrete interaction then have a much shorter path, and they also have the driving force of the hydrogen burns behind them:

debris → dry well → reactor building → environment

Many of the phenomena in the transport of fission products in TC1 can be illustrated by the table showing the fractional locations of the fission products, divided into their usual groups, at the end of the accident. Table I-I gives these fractions, above a cutoff value of 1%. The wet well as a location is excluded from the table because none of the fractions is above this value.

In the initial core melt phase almost all of the iodine and caesium and some of the tellurium are released. Some of the material is deposited in the RCS, and almost all of what remains airborne is scrubbed by the suppression pool. Because of the predicted efficiency of scrubbing in the pool, the final source terms are insensitive to RCS phenomena. To secure the predictions of low source terms for the volatile fission products, consideration is given primarily to validating the models for bubble scrubbing. The efficiency of the scrubbing accounts for the somewhat peculiar pattern of the fractions released to the environment in this sequence, with higher fractions for less volatile elements.

This is because they are not released until the suppression pool is bypassed and the containment is being swept by repeated hydrogen burns. The source terms for these elements are dominated by the predictions of release during the core–concrete interactions.

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Annex II

TMLU SEQUENCE FOR THE ZION NUCLEAR POWER PLANT

Annex II describes as a second example from BMI-2139 [II-1] a high pressure accident sequence, the TMLU sequence, for the Zion PWR in the USA, which has a large dry containment. The predicted sequence of events is the following.

t = 0: initiator

The TMLU accident sequence is initiated by a transient and is accompanied by the failures of the power conversion, auxiliary feedwater and emergency core cooling systems. Both the containment coolers and sprays are available during this sequence. The coolers are assumed to be switched on at the start of the accident.
While the steam generators are providing an effective heat sink, the primary system pressure is reduced to around 14 MPa. Then as the steam generators begin to dry out, the primary pressure rises again until the primary coolant begins to be expelled.

t = 65 min: beginning of expulsion of primary coolant

The pressurizer relief valve set point (16.4 MPa) is reached and liquid primary coolant begins to be expelled.
The expulsion of primary and the evaporation of secondary coolant continue until the steam generator dries out.

t = 93 min: steam generator dry

At around this time the primary coolant becomes saturated and the rate of expulsion increases.
During this process the containment pressure rises to 0.18 MPa, at which time the containment spray injection is turned on.

t = 101.8 min: containment spray injection on

Soon after this the pressurizer surge line becomes uncovered, changing the leak flow from liquid to vapour. Thereafter the primary coolant loss continues at a reduced rate until the core is uncovered.

t = 124.6 min: core uncovering

By this time the core temperature is around 627 K. The action of the containment sprays has reduced the containment pressure to 0.13 MPa. As the core progressively uncovers it heats up until it begins to melt.

t = 148.4 min: start of core melt

At this time the average core temperature is 1340 K and the peak temperature is 2550 K.
During the core melt phase, the containment spray injection system runs out of water.

t = 151.6 min: spray injection off and recirculation on

By this time the sprays have brought the containment pressure down to 0.11 MPa. After this it begins to rise again.
During the core melting, the core exit gas temperatures are predicted to rise to around 2400 K, but the temperature of the gas leaked to containment rises only to around 650 K. Most of the heating of the primary circuit structures is predicted to be concentrated in the first upper plenum structure, though the analysts warn that the MARCH code has no means of modelling the recirculation of hot gases, which is likely to be an important feature of such high pressure sequences. At around 173 min, with about 36% of the core melted and 15% of clad reacted, there is a sudden increase in the melting and reaction rate, leading rapidly to core slumping.

t = 178.2 min: core slump

At this time the core temperature is up to 2730 K. 57% of the core has melted and 33% of the clad has reacted.
The core is now melting very rapidly, leading almost immediately to its collapse.

t = 179.4 min: core collapse

At the collapse of the core, 86% of the core has melted and 50% of the clad has reacted. At this time the containment pressure has crept back up to 0.13 MPa.
The remaining water in the lower head boils away, the core debris attacks the bottom head, leading to failure of the bottom head.

t = 189.6 min: bottom head failure

Within the space of a minute the following sequence of events is predicted to occur. With the full primary circuit pressure (16.4 MPa) behind it, the core debris is ejected into the containment atmosphere, causing DCH and hydrogen burn. As the primary circuit pressure is relieved, the accumulators discharge. The containment pressure rapidly increases to 1.0 MPa, at which point the containment fails. The containment coolers and sprays are assumed to fail at the same time. Over the next 10 min the pressure inside the containment falls to atmospheric pressure, leaking 1.1×10^5 m^3 of gas in the process (compared with a containment free volume of 7.7×10^4 m^3).

The Source Term Code Package used for this analysis did not have the capability to model DCH as such. The effect was simulated by stipulating a very small particle size in the debris–water interaction and removing the steam inerting inhibition to hydrogen–oxygen recombination in the containment atmosphere. Because the containment sprays were in operation, there was a large quantity of water potentially available for interaction with the ejected core debris. In the simulation of DCH, the water interacting with the debris was restricted to that discharged from the accumulators. Subsequently the water in the reactor cavity is continually replenished by overflow from the containment sump. After a period during which the debris is quenched, it begins to reheat, causing concrete to be attacked.

TABLE II-I. FRACTIONAL DISTRIBUTION OF FISSION PRODUCTS BETWEEN LOCATIONS FOR THE TMLU SEQUENCE FOR ZION NUCLEAR POWER PLANT

Species	Melt	RCS	Cavity	Containment	Environment
I	—	0.77	—	0.22	5.7×10^{-3}
Cs	—	0.81	—	0.19	6.4×10^{-3}
Te	0.30	0.28	0.15	0.23	4.0×10^{-2}
Sr	0.99	—	—	—	9.4×10^{-5}
Ru	1.00	—	—	—	4.3×10^{-7}
La	1.00	—	—	—	3.9×10^{-7}
Ce	1.00	—	—	—	2.0×10^{-8}
Ba	0.98	0.02	—	—	1.8×10^{-3}

t = 253 min: start of concrete attack

In this calculation core–concrete interactions are assumed to occur at the same time as the boiling off of cavity water. Alternative scenarios would be that the dispersal of debris by the HPME would have made concrete attack impossible, or that the debris would form a coolable bed, quenched as long as water is available on the containment floor. During this process it is predicted that there is inversion of the melt layers.

t = 303 min: melt layers invert

The initially heavier oxidic layer becomes progressively less dense until the melt inverts, with the metallic layer sinking to the bottom.
At around this time the failed containment begins to leak significantly once more. The core–concrete attack and containment leakage continue steadily until the end of the calculation.

t = 861 min: end calculation

By this time a further 1.1×10^5 m^3 of gas has leaked from the containment.

The locations of the various fission products at the end of the calculation, expressed as fractions of the original inventory, are shown on Table II-I. Because the fractions released to the environment in this accident sequence are much lower than those in the Peach Bottom TC1 scenario, the restriction in Table I-I to fractions not less than 0.01 is relaxed here for the 'environment' column.

The fission products released from the degrading core in this sequence encounter high pressures and low flow rates on their way out of the primary circuit (via the pressurizer) to the containment. Under these circumstances the retention in the primary circuit is predicted to be high. The fraction of the material that enters the containment then encounters the action of the sprays and coolers. The rapid removal brought about by these engineered safety features means that there is time for most of the fission products to be removed before the containment fails at the time of the RPV failure and DCH.

Any fission products released as a result of the core–concrete interaction will pass into a failed containment with a substantial amount of gas flowing through it and out through the breach. There will therefore be little scope for retention in the containment. The fact that only small amounts of the less volatile elements are released reflects the prediction that the releases from the core debris attacking the concrete are low. The authors of BMI-2139 [II-1] do not comment on why this is so.

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LIST OF ABBREVIATIONS

APET:	Accident progression event tree
BWR:	Boiling water reactor
CET:	Containment event tree
DCH:	Direct containment heating
HPME:	High pressure melt ejection
LOCA:	Loss of coolant accident
LWR:	Light water reactor
PDS:	Plant damage state
PSA:	Probabilistic safety assessment
PWR:	Pressurized water reactor
RCS:	Reactor coolant system
RPV:	Reactor pressure vessel

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ISBN 92-0-102195-X
ISSN 0074-1892