CSNI Report No. 161
General Distribution

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THERMOHYDRAULICS OF EMERGENCY CORE COOLING IN LIGHT WATER REACTORS

October 1989

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
OECD NUCLEAR ENERGY AGENCY
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THERMOHYDRAULICS OF EMERGENCY CORE COOLING IN LIGHT WATER REACTORS

A STATE-OF-THE-ART REPORT BY A
GROUP OF EXPERTS OF THE NEA COMMITTEE
ON THE SAFETY OF NUCLEAR INSTALLATIONS

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Committee on the Safety of Nuclear Installations
OECD Nuclear Energy Agency
38, blvd. Suchet, 75016 Paris, France

NEA

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA, established on 20th December 1957) on the adhesion of Japan as a full member.

NEA now groups all European Member countries of OECD and Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress.

This is achieved by:

- encouraging harmonization of governments' regulatory policies and practices in the nuclear field, with particular reference to the safety of nuclear installations, protection of man against ionizing radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;

- keeping under review the technical and economic characteristics of nuclear power growth and of the nuclear fuel cycle, and assessing demand and supply for the different phases of the nuclear fuel cycle and the potential future contribution of nuclear power to overall energy demand;

- developing exchanges of scientific and technical information on nuclear energy, particularly through participation in common services;

- setting up international research and development programmes and undertakings jointly organised and operated by OECD countries.

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

CSNI

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. The Committee was set up in 1973 to develop and co-ordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.

The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done in a number of ways. Full use is made of the traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences. Some of these arrangements are of immediate benefit to Member Countries, for example by improving the data base available to national regulatory authorities and to the scientific community at large. Other questions may be taken up by the Committee itself with the aim of achieving an international consensus wherever possible. The traditional approach to co-operation is reinforced by the creating of co-operative (international) research projects, such as PISC and LOFT, and by a novel form of collaboration known as the international standard problem exercise, for testing the performance of computer codes, test methods, etc. used in safety assessments. These exercises are now being conducted in most sectors of the nuclear safety programme.

The greater part of the CSNI co-operative programme is concerned with safety technology for water reactors. The principal areas covered are operating experience and the human factor, reactor system response during abnormal transients, various aspects of primary circuit integrity, the phenomenology of radioactive releases in reactor accidents, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

The Sub-Committee on Licensing, consisting of the CSNI Delegates who have responsibilities for the licensing of nuclear installations, examines a variety of nuclear regulatory problems and provides a forum for the review of regulatory questions, the aim being to develop consensus positions in specific areas.

ABSTRACT

This report, by a group of experts of the OECD-NEA Committee on the Safety of Nuclear Installations, reviews the current state-of-knowledge in the field of emergency core cooling (ECC) for design-basis, loss-of-coolant accidents (LOCA) and core uncovery transients in pressurized- and boiling-water reactors. An overview of the LOCA scenarios and ECC phenomenology is provided for each type of reactor, together with a brief description of their ECC systems (Chapters 1 and 2). Separate-effects and integral-test facilities, which contribute to understanding and assessing the phenomenology, are reviewed together with similarity and scaling compromises (Chapter 3). All relevant LOCA phenomena are then brought together in the form of tables (Chapter 4). Each phenomenon is weighted, by expert judgement, in terms of its importance to the course of a LOCA, and appraised for the adequacy of its data base and analytical modelling. This qualitative procedure focusses attention on the modelling requirements of dominant LOCA phenomena and the current capabilities of the two-fluid models in two-phase flows (Chapter 5). The preceding topics lead into the key issue with ECC at the present time: quantitative code assessment and the application of system codes to predict with a well defined uncertainty the behaviour of a nuclear power plant. This issue, the methodologies being developed for code assessment and the question of how good is good enough are discussed in detail (Chapter 6). Some general conclusions and recommendations for future research activities are provided.

CONTENTS

EXECUTIVE SUMMARY

This report, by a group of experts of the OECD-NEA Committee on the Safety of Nuclear Installations, reviews the current state-of-knowledge in the field of emergency core cooling (ECC) for design-basis, loss-of-coolant accidents (LOCA) and core uncovery transients in pressurized- and boiling-water reactors.

A key issue concerned with ECC at the present time is quantitative code assessment and the application of best-estimate system codes to predict nuclear power plant behaviour with a well defined uncertainty. Code assessment is intimately connected with LOCA phenomenology, the experimental foundations on which it is based and the prediction capabilities of analytical models. The report is structured to discuss these topics before the important issue of code assessment itself is reviewed.

In order to place the detailed discussion of LOCA phenomena into perspective, the principal physical characteristics of the various commercial types of pressurized-(PWR) and boiling-water reactors (BWR), together with their ECC systems, are first presented in Chapter 1. In Chapter 2, LOCA categorization and typical LOCA scenarios for a PWR with U-tube steam generators and cold-leg injection are described in detail. Scenario modifications related to other PWR designs are then addressed. A similar procedure is adopted for BWR types and BWR scenarios.

Chapter 3 provides a critical appraisal of integral-test facilities and of those separate-effects facilities considered to have contributed significant data and information on emergency core cooling. Since scaling is a major consideration this is discussed first. Attention is then focussed on the special features of test-rigs, on scaling compromises and lessons learned from the tests.

It is concluded from Chapters 2 and 3 that PWR- and BWR-LOCA scenarios and core uncovery transients are, on the whole, very well understood from a phenomenological point of view and are strongly supported by experiments. The drive for additional tests or even new facilities is then not the need to confirm the cooling ability or effectiveness of ECC, but rather to quantify this ability more accurately. There must, however, be a feedback process between the degree of accuracy required, the degree achieved by prediction methods and the need for experimental and/or better analytical models.

A phenomena weighting and appraisal process is described in Chapter 4, which puts all of these points into a consistent and transparent, qualitative, framework. Each phenomenon discussed in Chapter 2 is weighted in terms of its importance to the course of a LOCA, and then appraised for the adequacy of its data base and analytical modelling. The process is largely based on expert judgement and the results presented in the form of tables. Based on these tables, recommendations for improvements to the modelling of particular LOCA phenomena, for example, are made.

In Chapter 5, a brief summary of the system codes used in LOCA analyses together with their theoretical framework is given. Their capabilities and limitations to model the above identified phenomena are then critically assessed. It is concluded that two-fluid models provide the means for a more realistic/mechanistic and more precise analysis of many dominant or controlling ECC phenomena, which cannot be analysed by mixture-models other than fully empirically. The price of this precision is the additional empirical information required for closure-laws at the two-fluid level. Much remains to be done in this area.

Despite large modelling and numerical differences at the closure-law level, however, studies have shown that engineering parameters important to safety and licensing can be predicted with acceptable precision by both mixture and two-fluid system codes. Here by acceptable is meant: may be used for design-basis accident evaluations in which conservative margins are applied to cover uncertainties and model deficiencies. Thus, the direct benefits to the ECC licensing basis of more accurate predictions are probably small at the present time.

For other purposes, such as improvements in plant operational flexibility or core-damage evaluations, direct benefits may on the contrary be significant.

LOCA phenomenology, test-rig data and the analytic modelling of LOCA phenomena are the foundations on which code assessment stands. In Chapter 6, both qualitative and quantitative code assessment are discussed. Three independent approaches to quantitative code asssessment - the UKAEA, GRS and NRC-CSAU methods - are reviewed in detail and their merits and weaknesses compared. Although the NRC-CSAU method has been applied to a large-breach LOCA on a PWR, all the methods are still undergoing evaluation. From the review of these methods it is concluded that best-estimate system codes and quantitative assessment are valuable, but complex and expensive tools for assessing LOCA. Any precision claimed for prediction accuracy should be treated with reserve, because expert judgement is still very much in evidence in all of them.

The following recommendations are offered:

1. A feedback is needed in terms of desired accuracy for a given problem or set of problems, achievable accuracy as defined by a quantitative assessment and, the research and development required to match the two.

2. A vast amount of data on ECC phenomena has been gathered. These data need to be scrutinized before embarking on additional experimental programmes. Specific areas requiring an expanded data base have been identified together with areas in which a more thorough examination of existing data should be undertaken.

3. In order to provide guidance on where model improvements are really needed, efforts should be made to quantify the impacts of modelling deficiencies on key safety parameters.

4. The three methods for quantitative code assessment should be completed and tested. Comparisons between the methods will illuminate strengths, weaknesses and costs involved.

5. Finally, the general question how good is good enough should be changed to how good need it be for a particular application.

ACRONYMS

Facility acronyms, such as LOFT, SSTF, etc. are listed in Tables 3.3 to 3.7 and 6.1.

ADS	=	Automatic Depressurization System
AHP	=	Analytical Hierarchical Process
ATHLET	=	Code Name
BWR	=	Boiling Water Reactor
CATHARE	=	Code Name
CCFL	=	Counter Current Flow Limiting
CEC	=	Commission of the European Communities
CHF	=	Critical Heat Flux
COBRA	=	Code Name
CRD	=	Control Rod Drive
CSAU	=	Code Scaling, Applicability and Uncertainty
CSNI	=	Committee on the Safety of Nuclear Installations
CVCS	=	Chemical and Volume Control System
DNB	=	Departure from Nucleate Boiling
DRUFAN	=	Code Name
ECC(S)	=	Emergency Core Cooling/Coolant (System)
EPRI	=	Electric Power Research Institute
FLUT	=	Code Name
GE	=	General Electric
GRS	=	Gesellschaft für Reaktorsicherheit
HPCI	=	High Pressure Coolant Injection
HPCS	=	High Pressure Core Spray
HPIS	=	High Pressure Injection System
IAEA	=	International Atomic Energy Agency
ICAP	=	International Code Assessment Program
ISP	=	International Standard Problem
IT	=	Integral Test

LOCA	=	Loss of Coolant Accident
LPCI	=	Low Pressure Coolant Injection
LPCS	=	Low Pressure Core Spray
LPIS	=	Low Pressure Injection System
NEA	=	Nuclear Energy Agency
NPP	=	Nuclear Power Plant
NRC	=	Nuclear Regulatory Commission
PIRT	=	Process Identification and Ranking Table
PWR	=	Pressurized Water Reactor
RCIC	=	Reactor Core Isolation Cooling
RELAP	=	Code Name
RHR	=	Residual Heat Removal
RSK	=	Reaktorsicherheitskommission
SET	=	Separate Effects Test
SG	=	Steam Generator
SMABRE	=	Code Name
SOAR	=	State-of-the-Art Report
T	=	Temperature
THYDE	=	Code Name
TMI	=	Three Mile Island
TRAC	=	Code Name
UKAEA	=	United Kingdom Atomic Energy Authority

TERMINOLOGY

Accuracy is a measure, usually expressed statistically, of the difference between measured and predicted quantities taking into account uncertainties and biases in both.

Assessment is a procedure for specifying, qualitatively or quantitatively, the accuracy of code predictions.

Best-estimate refers to the modelling of a reactor system or test facility on a realistic basis. It is not used here in a statistical sense.

Bias is a measure, usually expressed statistically, of the systematic difference between a true mean value and a predicted or measured mean.

Breach is a term used to cover the complete spectrum of pipe-ruptures, pipe breaks, pipe splits and leaks.

Evaluation model refers here to a system code, which is based on analyses and boundary conditions conservatively chosen to maximize or minimize selected predicted parameters.

Integral Effects are the combined influence of multiple phenomena on total system behaviour.

Qualification, see Validation or Verification

Scaling is the reproduction or representation in scaled facilities of the important or dominant phenomena found at full-scale. It is also the means for extrapolating scaled data to full scale.

Separate Effects are the local phenomena influencing the behaviour of a system.

System Code is an analytical simulation, in the form of a computer program for modelling the nuclear and thermohydraulic behaviour of a power plant or test facility including their control and safety systems.

Uncertainty is a measure of the scatter in experimental or predicted data.

User Qualification is the means whereby a code user demonstrates an ability to select suitable models, suitable noding and suitable boundary conditions for a system code, when analysing a LOCA or plant transient.

Validation is the procedure by which models and submodels in system codes are shown to be reasonable or acceptable representations of modelled phenomena. Validation is performed by comparisons of code predictions with measured or other validated data. The term validation is often used synonymously with qualification and verification.

Validation Matrix is a set of tests selected, on the basis of expert judgement, as suitable for system code validation, verification and assessment.

Verification is the procedure by which a complete system code is shown to be a reasonable or acceptable representation of a system's behaviour. The term verification is often used synonymously with qualification and validation.

FOREWORD

Worldwide, a massive programme of research into the thermohydraulics of emergency core cooling (ECC) in light water reactors has been undertaken over the past 20 years. At the beginning this was driven by two needs:

- to provide empirical information on loss of coolant accident (LOCA) and ECC phenomena, and

- to confirm the assumed conservatism of licensing or evaluation models.

These models were used, and in most cases still are being used, for the licensing assessments of design basis accident, large breach LOCA, scenarios.

Towards the end of the 1970's, and particularly after the accident at Three Mile Island, a change in emphasis occurred. Much more weight was given to realistic analyses of the full spectrum of pipe breaks, and to those transients having a potential for core uncovery. For these purposes research efforts were focussed on the development, validation and assessment of new, more advanced system codes. In these codes ECC phenomena are modelled on a realistic or best-estimate basis. The approach is commonly referred to as "best-estimate", and will continue to be so, to differentiate it from the overall conservatism of the evaluation models.

As time went by many of the earlier experimental programmes of a confirmatory nature, such as LOFT, became safety-research programmes supporting the development and assessment of these best-estimate codes. New experiments, such as PKL-3, BETHSY, LSTF, SPES and the 2D/3D programme, were introduced to resolve important issues, such as scaling effects and to provide essential empirical information, lacking in the earlier programmes and needed for the new codes.

To share costs and to avoid duplication many of these programmes were internationalised through joint agreements (e.g. ICAP) or through agencies such as the NEA-CSNI, IAEA and the CEC.

Licensing authorities were firmly of the opinion that the new codes would provide the means for a quantitative assessment of power-plant safety margins over the spectrum of LOCA and transient events. Such codes would be more appropriate for evaluating experimental and plant data, they would furnish a firmer foundation for operator procedures and, at the same time, they would supply the means for confirming the large conservatisms in the evaluation models, which indeed they have. They would also provide more realistic boundary conditions for fuel behaviour during design basis accidents and beyond.

The sum total of these efforts is a mature understanding of LOCA and core uncovery transients, of emergency core cooling, of the best-estimate codes for their analysis and of the relevant margins involved with ECC licensing procedures. This maturity rests on a vast amount of separate-effects and integral-test data, which is still being augmented by results from new experiments. Much of this information has been or will be published. But, at this moment (1989), a critical review at an international level is missing of what has been achieved, what limitations exist, what is known or understood and what remains to be done in this field. It is the intent of this state-of-the-art report (SOAR) to provide expert opinions on these matters. Particular emphasis is placed on quantitative assessment of the advanced codes, since the conservatism of the evaluation models and the conservative application of best-estimate methodologies for design basis, safety analysis purposes are both no longer in any doubt.

At the CSNI Principal Working Group 2 meeting in October 1985 a proposal was made for an SOAR on ECC. This proposal was referred to the Task Group on Status and Assessment of Codes for Transients and ECCS. At various meetings throughout 1986 the Task Group discussed the scope of such a report, recommended that it be limited to the thermohydraulics of emergency core cooling and drafted a list of contents. Firm support for the modified proposal was given by the Principal Working Group 2 and the CSNI at their meetings in Autumn 1986. This led to the establishment of a Writing Group, which comprised the following members:

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Members of the Writing Group met in July 1987 and May 1988, and a first draft of the report was distributed for comments in May 1989. A final editorial meeting was held in August 1989 to incorporate the comments received.

It is worth stressing that this state-of-the-art report is not meant to be an extensive encyclopedia or compendium, which cites every single experimental and theoretical result obtained in the area of emergency core cooling. Instead, the report sets forth what is considered to be known and understood, along with some judgement of the importance of remaining uncertainties. In particular, the authors have tried to clarify the technical and scientific background needed or available to understand governing phenomena and to extrapolate and transfer experimental observations into design and licensing decisions for nuclear power plants.

1. INTRODUCTION

1.1 Report Structure

The subject of this SOAR is the description and modelling of the thermohydraulic phenomena controlling the fuel-cladding's temperature excursion during a plant transient or loss-of-coolant-accident in a light water reactor. Such an excursion is initiated by a boiling crisis, if this occurs prior to core uncovery, or by the uncovery of the core itself. The temperature excursion is terminated by core rewetting, spray cooling if available and/or by core reflooding and recovery following the injection of emergency core coolant.

The purpose of this SOAR is to provide expert opinions on what is known or understood about the phenomena involved, what has been achieved with modelling, what has been achieved with specifying the uncertainties in reactor calculations, what limitations exist and what remains to be done in these areas.

In order to stay within reasonable bounds, the SOAR is restricted to the discussion of core uncovery either during plant transients or as a result of design basis loss-of-coolant-accident (LOCA) and emergency core cooling (ECC) scenarios. That means, to those scenarios which are assessed to comply with national regulatory design criteria concerned with ECC, such as the USNRC 10 CFR 50.46 or the German RSK Guidelines.

Depressurization loads on primary system internals, hydrogen generation, long-term cooling and decay heat removal subsequent to core recovery are not considered here. Degraded core or severe accident phenomena and modelling are also outside the scope of this report.

Although fuel cladding deformation, swelling, oxidation, embrittlement and bursting are important for determining radioactive releases during core uncovery, their influence is localized, so that their direct influence on the overall ECC thermohydraulic phenomena during design basis events is for the most part minor. This fuel behaviour during LOCA is the subject of a separate SOAR /1.1/.

Despite the apparent dissimilarity between a pressurized water reactor (PWR) and a boiling water reactor (BWR), the thermohydraulic phenomena involved with their transient, LOCA and ECC behaviour have much in common. For this reason it is considered appropriate to include both types of light water reactor in this report. The LOCA and ECC behaviour for each type is discussed separately in Chapter 2 where emphasis is placed on LOCA scenarios and the thermohydraulic phenomena involved.

A very large number of small- and large-scale experiments, /1.2/ and /1.3/ for example, have been performed to simulate LOCA and ECC phenomena, to provide correlations, to adjust codes, to support both code assessment and the understanding of the scenarios of Chapter 2. A summary and appraisal of these experiments is given in Chapter 3, together with the lessons learned and some opinions on the need for new facilities and new instrumentation.

Expert judgements of the relative importance of individual LOCA phenomena, how well they are understood and how adequately they are modelled are collected and evaluated in Chapter 4. This screening process enables the dominant phenomena, which need additional test data and those which still pose modelling difficulties, to be identified. The modelling difficulties are particularly important when a best-estimate assessment of ECC effectiveness is required. An assessment is also important for licensing purposes, but then model uncertainties may be compensated by conservatism in the models themselves and/or in boundary conditions.

Reactor system codes for assessing ECC effectiveness are based on two-phase flow and heat-transfer analyses, which either have a mixture formulation of the conservation laws, a two-fluid formulation or a combination of both. It is thus very important that the capabilities and limitations of two-phase flow analyses are clearly understood and these are the central topics of Chapter 5. However, this is a very broad task /1.4/ so much so that, to remain within reasonable bounds, modelling limitations are discussed in Chapter 5 solely with regard to the dominant ECC phenomena identified in Chapter 4.

Finally, the key issue involved with ECC is quantitative code assessment and the application of best-estimate codes to predict nuclear power plant LOCA behaviour with a defined uncertainty. The first five chapters of this SOAR give the background necessary to appreciate the difficulties involved with code assessment. Quantitative assessment of code predictions has been somewhat like a search for the "Holy Grail" over the last few years and is still a controversial issue /1.5/ which centres around the topic of how good is good enough. A review of this assessment is provided in Chapter 6.

Conclusions and recommendations from this SOAR are brought together in Chapter 7.

Since the task of writing this SOAR was undertaken, several very important reports on emergency core cooling have been published. No report on the phenomena involved with ECC would be complete without reference to the NRC Compendium of ECCS Research /1.6/, to the NRC tables for identifying and ranking LOCA phenomena /1.7/ and to the EPRI report on ECCS Evaluation

Methodology /1.8/. At the same time any review of system code capabilities must take into consideration two ECC reports, one on code comparisons /1.9/, the other on code assessment /1.10/. This SOAR draws heavily on material presented in all these documents.

To provide a framework for the later chapters, the remainder of this introduction discusses in a general manner:

- fuel-cladding integrity, the boiling crisis and core uncovery, Section 1.2, and
- PWR and BWR characteristic features which have an influence on core uncovery scenarios, Section 1.3.

ECC systems, transients and accidents involving a net loss of coolant from the primary system, the LOCA breach spectrum and the subject of main-coolant-pump or recirculation-pump trip are thereby briefly introduced.

1.2 Cladding Integrity, the Boiling Crisis and Core Uncovery

Maintaining the integrity of fuel cladding, the barrier for retaining fission products in the fuel, is a prime safety concern in all light water reactors. A main threat to the integrity of this barrier is impaired core cooling. This arises when coolant no longer covers the core, or when changes in core power, coolant flow, coolant pressure or coolant temperature lead to a boiling crisis at the cladding-coolant interface /1.11/ and /1.12/. If core cooling is impaired the cladding heats-up and may deform, swell, oxidize and eventually burst as a result of the temperatures and stresses imposed upon it. Part of the core's radioactive inventory is then released into the primary circuit or, if this is breached, into the containment.

A first line-of-defense against such releases is the prevention of abnormal occurrences which could lead to impaired core cooling. A second line-of-defence is to provide ECC or other coolant make-up systems to cope with impaired cooling should it occur. For this purpose ECC systems are conservatively sized and reactor operating limits are conservatively imposed to provide <u>assurance that cladding temperatures do not exceed those at which substantial geometric changes or exothermic chemical reactions occur</u>. By this means the core retains a coolable geometry and radioactive releases remain within allowable limits. Design basis limits for cladding temperatures, chemical reactions and radioactive releases are defined in the aforementioned national, regulatory design-criteria, and the effectiveness of the measures is assessed against these limits using reactor system codes.

All abnormal occurrences involving a net loss of coolant from the primary system have a potential for core uncovery. Any ensuing cladding temperature excursions, their rate and history, are a function of the balance of energy sources and sinks acting on the fuel and cladding.

These energy sources comprise:

- core power, decay heat and stored energy in the fuel, and
- chemical energy produced by metal-water reactions.

Energy sinks depend primarily on the thermohydraulic processes, which control the convective heat-transfer from the cladding. These processes influence or are influenced by:

- heat transfer between fuel and cladding (gap conductivity),

- fuel and cladding conductivities,

- the possible occurrence of a boiling crisis, or even a series of boiling crises and rewetting transients prior to or sometimes during core uncovery,

- the loss-of-coolant and the gain from ECC or other make-up systems until the core is recovered or reflooded from below,

- spray cooling and/or coolant injection above the core,

- energy stored in the reactor system structures,

- primary system geometry, breach size and location, (note that throughout this report the general term "breach" will be used to cover the complete spectrum of pipe-ruptures, double-ended pipe breaks, pipe splits and leaks),

- system depressurization through a breach, through safety or relief valves, as a result of ECC injection and/or through external means of cooling the primary system (e.g. steam generators, isolation condensors or residual heat removal systems), and

- forced- or natural-circulation conditions in the primary system.

1.3 PWR and BWR Characteristic Features Relevant to Core Uncovery, LOCA and ECC

1.3.1 Principal Physical Characteristics

1.3.1.1 PWR (Fig. 1.1)

Most commercial PWR's are very alike having a pressure vessel and several primary system loops comprising piping, pumps and steam generators. It is only to be expected then that core uncovery, LOCA and ECC phenomena are similar for all PWR's. An important feature is that all loop connections to the vessel are above the core making it possible to reflood the core for any pipe breach in the primary system. The following rather minor differences between PWR types have an influence on core uncovery behaviour:

- Elevations of loop seals relative to the core, main coolant pumps and steam generators vary from reactor to reactor. These influence small breach LOCA behaviour.

- ECC system injection points and injection pressures vary between plants. All have cold-leg injection, some have additional hot-leg and/or upper plenum, upper head or downcomer injection. High-, intermediate- and low-pressure injection systems are to be found. Most PWR-ECC systems include "passive" accumulators for injecting coolant under the action of compressed nitrogen. All commercial PWR-ECC systems inject borated water into the core to avoid reactor recriticality. ECC systems for PWR's are described in more detail in Appendix A1.1 and a schematic of a typical system is shown in Fig. 1.2.

- Once-through steam generators introduce uncooled high-points into the reactor coolant piping. These tend to impair natural circulation flow more quickly than in the more common U-tube generators, which have their highest point inside the steam generator itself. This leads to a somewhat different small-breach LOCA behaviour. Plants with once-through steam generators also have vent valves in the upper plenum, connecting the cold-leg loop to the hot-leg. During a LOCA these vent steam to the cold-leg mitigating the effects of steam-binding on core reflooding.

Plants with horizontal steam generators are not discussed in this report.

1.3.1.2 BWR (Fig. 1.3)

There are three types of commercial BWR, whose physical characteristics strongly influence core uncovery and reflooding following a LOCA:

a) Very early reactors had recirculation pumps external to the vessel driving the total core flow. If a large breach in the recirculation line occurs below core level, it is not possible to reflood the core. Spray cooling from above is then the only immediate means to keep the core cool.

b) Later reactors have two external pumps, which drive multiple jet-pumps inside the reactor vessel. Jet-pumps enable the core to be reflooded to at least 2/3 core height (collapsed liquid level) for any size breach in the recirculation lines.

c) The latest generation of BWR's has no external pumps, but rather multiple internal recirculation pumps. Since there are no pipe connections below the top of the core in these reactors, the ability to reflood the core completely is insensitive to the location of pipe breaches.

Depending on the reactor type and vendor, a wide spectrum of BWR-ECC systems are commonly found, although not all on one reactor:

- low (LPCS) and high pressure core sprays (HPCS) above the core in the upper plenum,

- low pressure coolant injection (LPCI) into the core bypass region inside the core shroud,

- low (LPCI) and high pressure coolant injection (HPCI) into the annulus between the core shroud and vessel wall, either directly, through feedwater lines or into recirculation lines.

- depressurization systems, such as auxiliary condensers and an automatic depressurization system (ADS), to lower reactor pressure enabling low pressure ECC systems to inject.

ECC systems for BWR's are described in Appendix A1.2 and a schematic of a typical system is shown in Fig. 1.4.

In addition to the ECC systems, normal vessel make-up systems such as feedwater, reactor core isolation cooling (RCIC) and control-rod-drive (CRD) cooling, have a certain availability and capability to mitigate BWR core uncovery events. ECC and make-up systems all inject unborated water into the vessel.

1.3.2 Transients and Accidents Involving a Net Loss of Coolant

1.3.2.1 Transients

A PWR, except for its chemical and volume control system (CVCS), has a closed primary system. Thus, but for CVCS malfunctions, no plant transient can lead to a net loss of coolant from the primary system unless primary system relief or safety valves are actuated. Such transients have a potential for core uncovery only after an extended period of time (ca. 1 hour) and may be classed with small breach LOCA's.

BWR's have a direct cycle primary system in which the continuous and very large loss of coolant from the reactor vessel through the steam lines during normal operation is balanced by feedwater. Any transient upsetting this balance, in particular a loss of all feedwater, has a potential for core uncovery within about 30 minutes.

Risk studies show that transients, which are precursors to core uncovery, have a far greater probability of occurrence than have small or large breaches in the primary system.

1.3.2.2 LOCA's

Breaches in the primary system of all light water reactors create the potential for core uncovery. Whether uncovery actually occurs or not depends on the breach size and location, on available ECC or other make-up systems, and on the reactor type. On a best-estimate basis, the following behaviour is representative; a complete description of LOCA scenarios for PWR's and BWR's is provided in Chapters 2.1 and 2.2, respectively:

A. Large breaches

These are characterised by rapid system depressurization with more or less homogeneous, two-phase, conditions in the primary system during the depressurization.

In PWR's core uncovery always occurs for large pipe breaks and, for cold-leg breaks, is preceded by a very rapid boiling crisis as core flow stagnates and reverses. Core recovery follows ECC injection.

Whether a boiling crisis prior to core uncovery occurs in a BWR, depends on breach location and reactor type:

- Core uncovery preceded by a boiling crisis occurs for recirculation line breaks in BWR's having external recirculation pumps. Steamline and feedwater line breaks can also lead to core uncovery, but without the earlier boiling crisis.

- Neither core uncovery nor boiling transition occur for any line breach in BWR's with motor-driven internal recirculation pumps.

B. Small breaches

These are characterised by a loss of inventory with little or no primary system depressurization. Where the high-pressure coolant injection supply is limited, or absent, this net loss can be compensated by reducing primary system pressure to a level at which sufficient make-up can be injected and recirculation cooling can be established. Thus,

- for PWR's maintaining the functional capability of the steam generators to cool the primary system and lower its pressure below the shut-off head of the intermediate-pressure injection pumps is a primary concern. This is reflected in the multiple auxiliary and emergency feedwater systems to be found in modern PWR's. Some PWR designs rely directly on their high-pressure injection system without the need for steam generator cooling.

- in BWR's, if the main condenser is not available, the primary system may be depressurized through auxiliary condensors or through one or more relief valves. A very rapid depressurization through multiple relief valves occurs automatically, if core uncovery is imminent. This is controlled by the automatic depressurization system (ADS), which effectively turns a small breach into a large breach LOCA, and can itself lead to core uncovery.

C. Intermediate Breaches

An intermediate rupture or leak completes the spectrum of pipe breaches postulated for either a BWR or PWR. Such breaches introduce no new phenomenology, rather they retain some of the characteristics of both large and small breaches. For example, at some intermediate liquid-line breach a boiling crisis may not occur prior to core uncovery. At the same time primary system depressurization through the breach may be much slower than for a large breach, so delaying ECC injection.

D. Leaks in the Pressure Vessel

Normally, because of the measures taken to assure that no significant defects exist in a reactor pressure vessel, no vessel leaks are postulated to occur. In some countries assessment of ECC effectiveness is required for postulated, hypothetical, small breaches in the pressure vessel's bottom head. From the viewpoint of ECC this is the most critical location, because breach flow will remain liquid and depressurization will be correspondingly slow. For PWR's such postulated leaks are the only ones below core height. These leaks are classed as small breaches and, other than the need for continuous coolant make-up to compensate loss through the breach, they introduce no new phenomenology into the LOCA.

1.3.2.3 Main Coolant Pump and Recirculation Pump Trips

Tripping or not tripping the main coolant pumps on PWR's influences strongly the course of small breach LOCA events:

- If they are not tripped, the pumps will tend to maintain a high core flow and a homogeneous two-phase circulation in the primary system. This is beneficial for immediate core-cooling requirements, but generally detrimental to system mass inventory, since homogeneous flow will have a tendency to increase the mass loss through a breach and to delay depressurization even further.

- If they are tripped early, phase-separation occurs as mass inventory falls, and steam is released when the coolant level falls to below the breach level. Inventory loss is thereby reduced and depressurization increased.

- If they are tripped too late, the low mass inventory and the phase separation that occurs can uncover the core. This was particularly evident at Three Mile Island.

For these reasons most PWR's either have operating procedures requiring a manual pump trip early in an accident, or have an automatic pump trip simultaneously with the start of the ECC systems. Following pump trip the modes of heat-transfer from the core to the steam generators change accordingly from forced- to natural-circulation or even to reflux-condensation.

For large breaches, tripping the pumps may influence the blowdown and the early quenching processes. Otherwise it does not have a significant influence either on large- or intermediate-breach PWR phenomenology.

Tripping recirculation pumps in BWR's has very little influence on core uncovery events. It gives a more rapid decrease in core flow during LOCA's involving a large breach in a recirculation line. Recirculation pumps are usually tripped automatically from vessel low water level for purposes other than LOCA mitigation.

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APPENDICES FOR CHAPTER 1

Appendix A1.1

PWR Emergency Core Cooling (ECC) Systems

A1.1.1 General Description

High-, intermediate- and low-pressure systems for injecting coolant into the primary system are found on pressurized water reactors. They all inject borated water into the primary system to avoid reactor recriticality.

a) High-head (ca. 180 bar) charging systems

These are systems capable of injecting coolant into the primary system against normal reactor pressure. Their prime function is chemical and volume control or make-up during normal reactor operation, and in this mode, they have a limited capacity (ca. 10 l/s) to prevent rapid overfilling of the primary system. On some reactors, during small-breach LOCA's, they also have an ECC function to inject coolant automatically and at higher than normal flow rates.

b) High-pressure (ca. 100 bar) injection systems, HPIS

These are really intermediate-pressure systems which, by means of motor-driven pumps, inject coolant into the primary system once its pressure has fallen to, or below, the saturated conditions (ca. 100 bar) corresponding to normal reactor operating temperature (ca. 300 °C). This injection pressure is low enough to remove any potential for inadvertently overfilling the primary system, but high enough to cover the saturated conditions, which are rapidly attained during all but the smaller breaches of the LOCA breach spectrum.

Each HPIS takes suction from an ECC dedicated borated water storage tank. In many reactors the HPIS is tripped and isolated from the storage tank as soon as the tank is nearly empty. On some reactors the HPIS can also be manually aligned to a residual heat removal system, which takes suction from the containment sump for long-term cooling purposes.

c) Accumulator injection system (25 - 50 bar)

Nearly all PWR-ECC systems include accumulators, which are tanks containing borated water, pressurized by a cushion of nitrogen to between 25 and 50 bar. Without any electrical supplies these accumulators provide an essentially passive injection capability, thereby helping to refill the reactor-vessel's lower plenum and downcomer after blowdown of the vessel during a large-breach LOCA.

d) Low-pressure (10 - 20 bar) injection systems (LPIS)

These are active systems similar to HPIS, which help to reflood the core by pumping borated water into the depressurized primary system from ECC dedicated borated water storage tanks. Depressurization of the primary system occurs either as a result of mass and energy lost directly through the breach, through ECC induced condensation, or from energy transfer to the steam generators.

When the storage tanks are nearly empty, suction is switched automatically to the primary containment sump, which is the intermediate heat-sink for most of the energy released from the pressure vessel and reactor during and following large- and intermediate-breach LOCA's (see Chapter 2.1). From then on, the LPIS maintains the core in a flooded condition by re-injecting into the primary system the coolant lost through the breach to the containment. Where heat-exchangers are installed in the LPIS circuit, residual-heat may be removed from the containment sump to an ultimate heat sink, thereby providing long-term cooling.

Several variations on this basic LPIS arrangement may be found. For example, LPIS can be aligned as a normal residual heat removal system by taking suction directly from the primary system at low pressures. This may also serve as a long-term cooling strategy for small breach LOCA's. Some reactors have containment heat removal systems, which are independent of the LPIS. Others use the LPIS as a booster pump for the HPIS, thereby providing a residual heat removal capability at intermediate reactor pressures. Most of these re-alignments from the basic LPIS function are normally performed manually.

e) <u>Primary system depressurization systems</u>

For small-breach LOCA's the steam generators are the designated means for cooling-down and thereby depressurizing the primary system to a level where HPIS or LPIS can perform their make-up, respectively, residual heat removal functions. Energy is removed from the secondary-side through the power-operated relief valves, or through the main condensor if available. A design-basis rate-of-cooldown is usually established manually, although the procedure is automated on some reactors.

As far as this cooldown and depressurization function is concerned, the steam generators, their relief valves as well as the auxiliary and emergency feedwater supplies can fulfil safety related functions.

Some reactors have power operated relief valves on the primary system, through which the primary system can be depressurized by blowing-down to a tank and, eventually, to the containment. The ensuing loss of inventory through the relief valves as well as through the breach is replaced by means of coolant injected by the available ECC injection systems. This procedure, known as "bleed and feed", is not part of the current design bases for ECC.

Various ECC injection locations may be found on PWR primary systems (Fig. 1.1). All reactors have injection into the cold-legs of the primary system loops, since this provides coolant almost directly to the downcomer for reflooding the core from below. Some have direct downcomer injection and others have hot-leg, upper-plenum or upper-head injection.

Finally, the number, capacity and type of ECC systems chosen for any particular PWR depends on the design criteria used for system redundancy, diversity and assumed unavailabilities, as well as upon the methodology, employed for confirming, against the ECC criteria, the effectiveness of the ECC systems as a whole.

A1.1.2 <u>A Typical PWR-ECC System, LPIS</u>

A complex of auxiliary systems and components are used to ensure that even a basically simple ECC system starts up automatically on demand and functions as required. High quality in design, construction, maintenance and testing ensure a high level of availability and relability on demand. The LPIS, shown schematically in Fig. 1.2, is typical and has the following auxiliaries and actuation logic:

- Signals from redundant reactor protection instrumentation are needed to start (on low pressurizer water level or low pressurizer pressure or high containment pressure or a combination of these) automatically the LPIS motor.

- These signals are also used to align valves 1 to 7 in Fig. 1.2, if necessary, to their normal position as indicated on this figure.

- In the event of a loss of auxiliary power the LPIS motor is coupled automatically to its emergency diesel/generator, which is started automatically by the LPIS initiation signals.

- On depletion of the borated water tank the three-way valve 2 automatically switches to draw coolant from the containment sump.

- Various relief-valves 8 are installed to prevent line-overpressure from coolant thermal expansion. Check-valves are installed to prevent back-flow or to act as containment isolation devices. Some of the check-valves may be motor-operated, remote-manually, to enable them to be aligned for residual heat-removal or test purposes. One check-valve 5 is aligned by system pressure differences.

Typically, an LPIS system starts-up and delivers its design flow rate within ca. 10 s. Maximum pump heads of about 15 bar and maximum pump flows of ca. 200 l/s are common.

Appendix A1.2

BWR Emergency Core Cooling (ECC) Systems

A1.2.1 General Description

The following dedicated ECC systems (Fig. 1.3) are commonly found on boiling water reactors, although not necessarily all on one reactor:

a) Low-pressure (ca. 20 bar) core sprays (LPCS) and high-pressure (ca. 80 bar) core sprays (HPCS)

These distribute coolant fairly uniformly into the upper plenum above the core by means of multiple nozzles mounted on distribution headers inside the upper plenum. Sprays enable the core to be cooled even when an uncovered core cannot be reflooded from below. Confirming their effectiveness is not straightforward, however, because the distribution of the spray controls the amount of coolant arriving at the top of any particular fuel channel, and counter current flow limiting (CCFL) can influence the amount of coolant that actually enters a channel.

b) Low-pressure (ca. 20 bar) coolant injection (LPCI) into the core bypass region inside the core shroud

Coolant so injected rapidly distributes itself under gravity fairly uniformly across the whole of the core bypass. It directly cools the outside of the fuel channels, which can then act as a radiant heat sink for any uncovered fuel cladding with a view of the channels. Coolant drains out of the bypass region into the control rod guide tubes, into the lower plenum and into the bottom of the fuel channel, thereby refilling the lower plenum (if this is voided) and reflooding the uncovered core from below. If the core is uncovered, the drainage of coolant from the bypass region into the bottom of the fuel channels may lead to some collapse of the swollen water level in the channels or even to drainage of coolant from the fuel channels into the lower plenum.

If sufficient coolant is injected by one or more LPCI systems, the bypass region can fill up and overflow into the upper plenum. From there it can overflow into the fuel channels helping to cool the core from above.

c) Low-pressure (ca. 20 bar) coolant injection (LPCI) and high-pressure (ca. 80 bar) coolant injection (HPCI) <u>into the annulus between the core shroud and vessel wall</u>

Coolant may be injected directly into the annulus or indirectly into feedwater or recirculation lines. Provided the injected coolant does not bypass the core by escaping through the LOCA breach, and provided the breach location does not prevent the core from being reflooded, then this form of injection is a very simple and effective means for avoiding core uncovery, or for refilling and reflooding the core after uncovery has occurred.

d) <u>Depressurization systems</u>

Auxiliary condensers and/or an automatic depressurization system (ADS) are used to lower reactor pressure enabling the low pressure ECC systems to inject into the vessel. Auxiliary condensers may be separate from other ECC systems. An ADS typically employs about half of the pressure relief valves mounted on the steam supply system.

High pressure injection systems, which have the capability to inject coolant during normal reactor operation, either inadvertently or as a result of an expected operational occurrence, usually draw reactor-quality coolant from a condensate storage tank. When this is nearly empty they switch to drawing coolant from the pressure suppression pool. Low pressure ECC systems always draw their coolant from the suppression pool. All ECC coolant supplies are unborated.

The suppression pool is the intermediate heat sink for most of the energy released from the pressure vessel and reactor during and following a LOCA. To transfer this energy to an ultimate heat sink, heat exchangers are installed either in some LPCI systems, or in separate suppression pool cooling systems. By means of simple valve re-alignments, some LPCI systems can also act as normal residual heat removal systems by drawing coolant directly from the primary system, cooling it through the heat-exchanger and returning it to the reactor vessel.

The number, capacity and type of ECC systems chosen for any particular BWR depends on the design criteria used for system redundancy, diversity and assumed unavailabilities, as well as upon the methodology employed for confirming, against ECC criteria, the effectiveness of the ECC systems as a whole.

A1.2.2 A Typical BWR-ECC System, HPCS

Even the simplest of ECC systems uses a complex of auxiliary systems and components to ensure that it starts up automatically on demand and functions as required. High quality in design, construction maintenance and testing ensure a high level of availability and reliability on demand. An HPCS system, shown schematically in Fig. 1.4, is a typical example having the following auxiliaries and actuation logic:

- A small jockey pump is used to maintain the injection lines full of water, prior to starting the HPCS pump, in order to avoid any water-hammer when the pump is started. This is necessary on BWR's, because the reactor vessel is located high above the ECC pumps.

- Signals from redundant reactor-protection or engineered-safeguards instrumentation are needed to start (on low reactor water level or high drywell pressure) the HPCS pump or close (on high reactor water level) automatically the HPCS injection valve 1.

- These signals are also used to align valves correctly and automatically; on demand of the HPCS, injection-valve 1 is opened, test-valves 2 and 3 are closed (if open), suction-valve 4 is opened (if closed) and 5 is closed (if open).

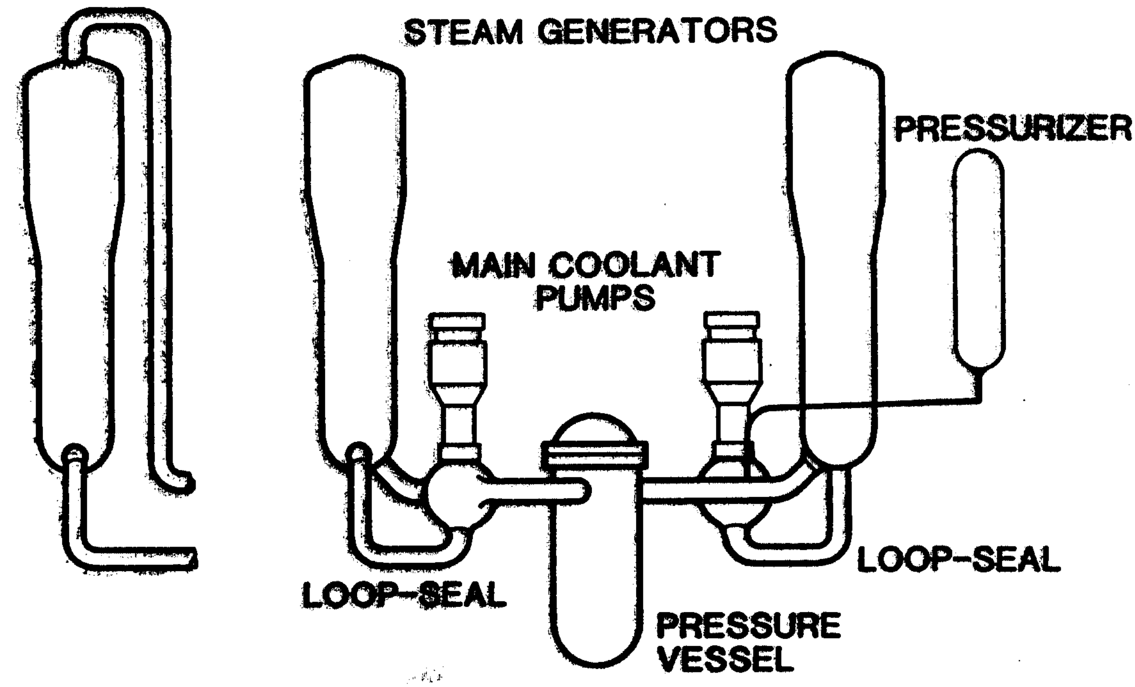
  The minimum flow valve 6 opens automatically at low coolant flow rates and closes again at high rates. On depletion of the condensate storage tank the suction valve 4 must be closed and 5 opened automatically.

- In the event of a loss of auxialiary power the HPCS motor is coupled automatically to its emergency diesel/generator, which is started automatically by the HPCS initiation signals.

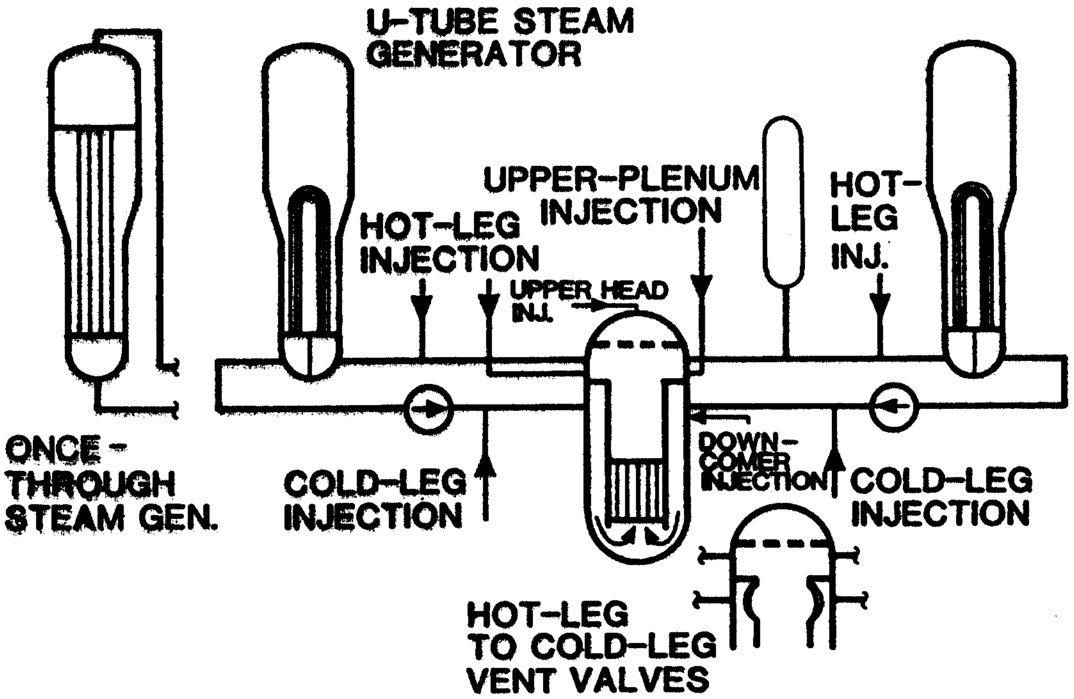
- The HPCS pump-seals, motor and emergency diesel all require cooling water to ensure their long-term performance.

- Relief valves 7 are installed to protect the HPCS piping from overpressure, and a check valve 8 acts as a containment isolation valve following loss of integrity of the HPCS system outside the containment.

Typically, an HPCS system starts-up and delivers its design flow rate within ca. 20 s. Pump flow-rates vary almost linearly with pump-head. Maximum pump heads of about 80 bar and maximum pump flows of 300 - 400 l/s are common.



A. TYPICAL PHYSICAL LAYOUT OF A TWO-LOOP PLANT



B. SCHEMATIC OF A TWO-LOOP PRIMARY SYSTEM AND VARIOUS ECC INJECTION POSSIBILITIES (SEE APPENDIX A1.1)

FIG. 1.1 TYPICAL PWR PHYSICAL CHARACTERISTICS

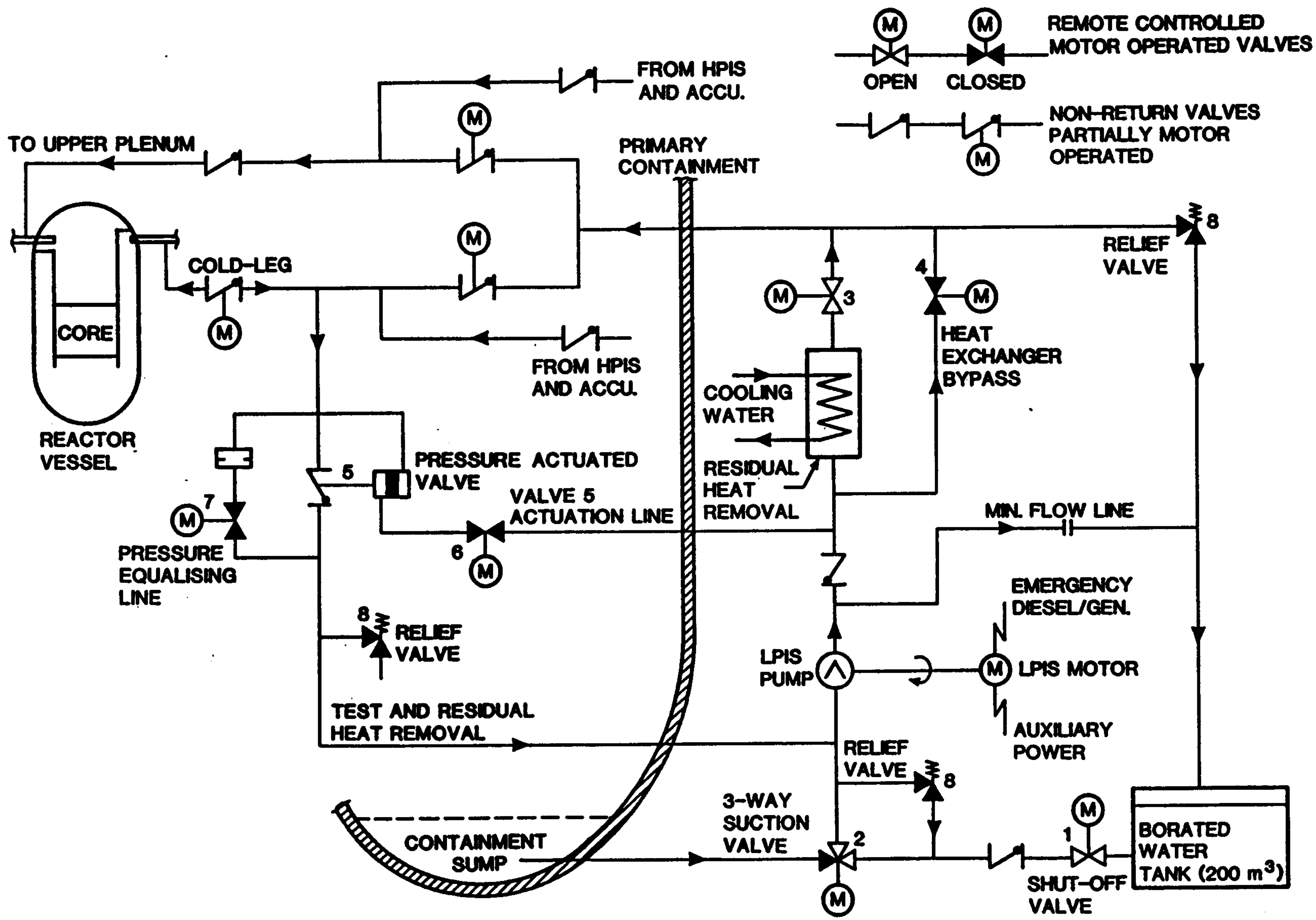
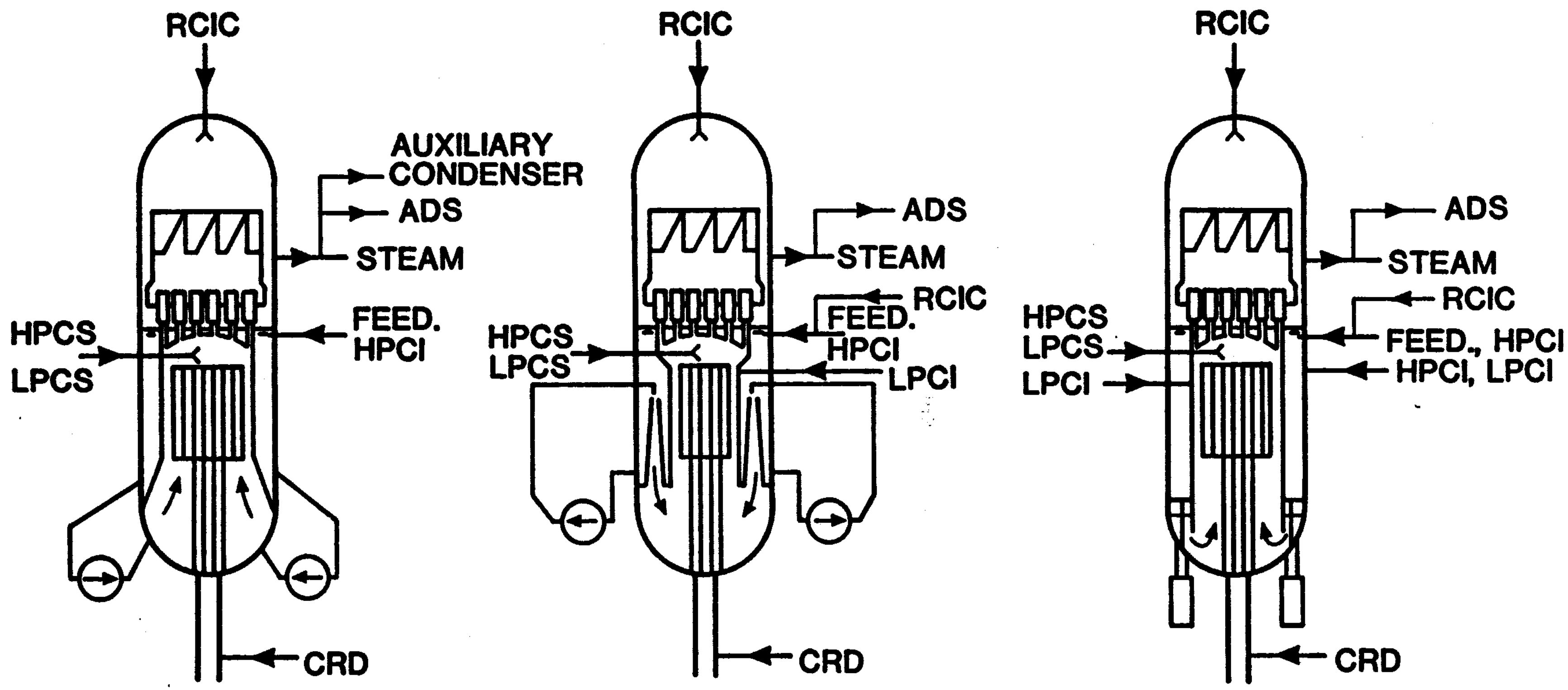


FIG. 1.2 SCHEMATIC OF A TYPICAL LOW PRESSURE INJECTION SYSTEM (LPIS) FOR PWR



A. EXTERNAL PUMPS

B. EXTERNAL PUMPS
INTERNAL JET-PUMPS

C. INTERNAL PUMPS

NOTE: THE COMBINATIONS OF ECCS INJECTIONS SHOWN ARE NOT ALL FOUND ON ONE REACTOR (SEE APPENDIX A1.2)

FIG. 1.3 SCHEMATIC OF THE THREE TYPES OF RECIRCULATION SYSTEMS ON COMMERCIAL BWR'S AND VARIOUS ECC INJECTION POSSIBILITIES

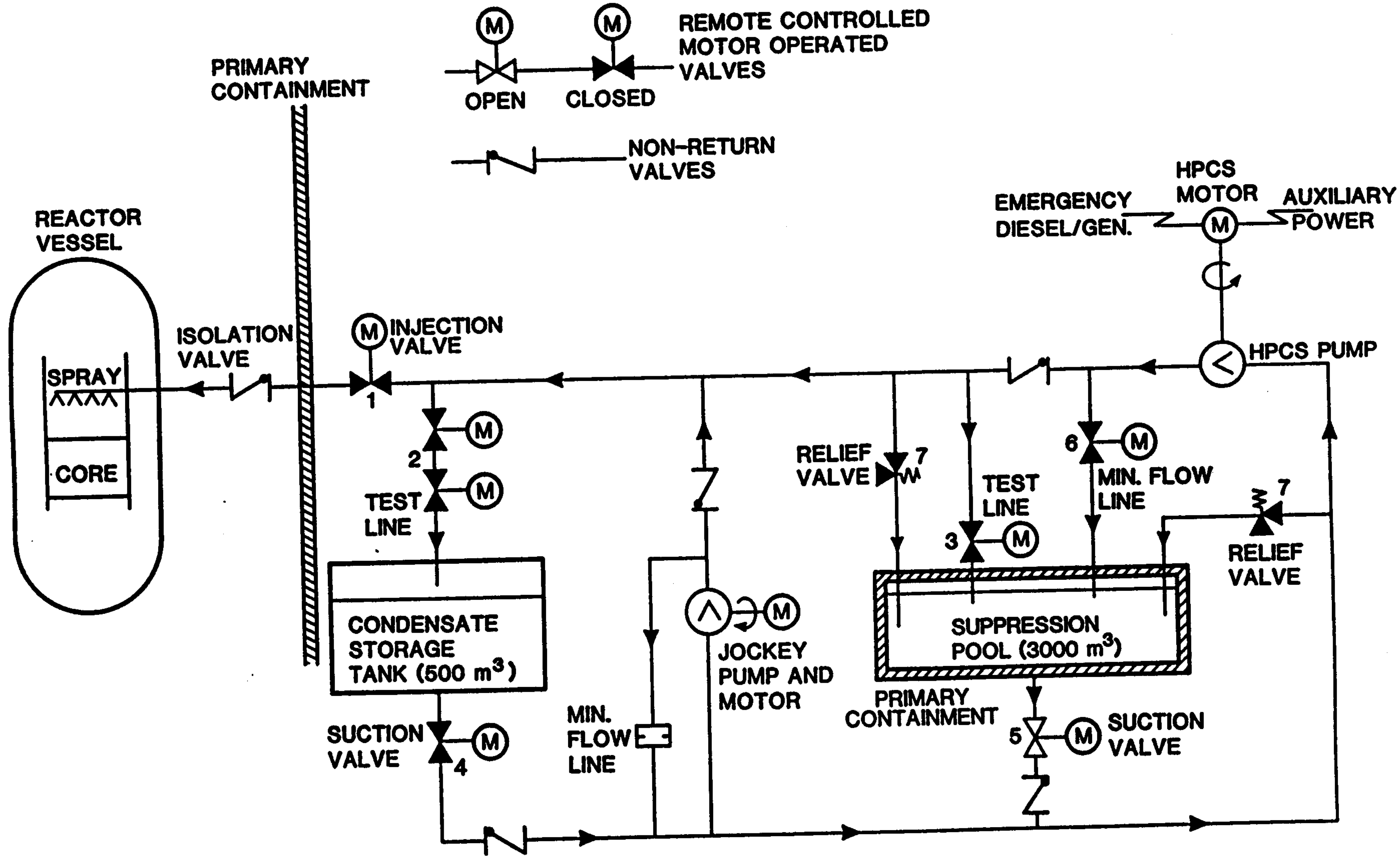


FIG. 1.4 SCHEMATIC OF A TYPICAL HIGH PRESSURE CORE SPRAY (HPCS) EMERGENCY CORE COOLING SYSTEM FOR BWR

2. LOCA SCENARIOS AND ECC PHENOMENOLOGY

The basic features of light water reactors relevant to LOCA and ECC are discussed in Chapter 1. In this chapter a global classification of LOCA scenarios in terms of breach size and location is provided. This is followed by detailed description of scenarios and phenomenology for each LOCA class and for typical reactor types. Section 2.1 deals with pressurized water reactors (PWR's) and Section 2.2 with boiling water reactors (BWR's). Phenomena identified in this chapter are collected in Chapter 4, where they are rated and appraised for their relative importance.

The PWR-LOCA scenarios were put together mainly from information directly provided by the Babcock and Wilcox, Framatome and Kraftwerk Union companies. BWR scenarios were synthesised from realistic LOCA descriptions and analyses provided by the General Electric, Kraftwerk Union and the Studsvik Energiteknik companies. A great deal of information was gathered from the US-NRC Compendium of ECCS Research, NUREG 1230, the NRC-PIRT, NUREG/CR-5047 and an EPRI report on Realistic ECCS Evaluation Methodology, NSAC-86, Refs. /1.6/, /1.7/ and /1.8/ of Chapter 1, respectively. Several figures have also been reproduced from the NRC report. A multiplicity of experimental support for the scenarios and the detailed phenomenology involved has been published. Much of it is described in the NRC Compendium and a critical discussion of particular aspects is provided in Chapter 3.

Although large pipe rupture LOCA's are discussed in detail, a great deal of attention in this chapter is also focussed on small-breach LOCA's. This is because in terms of frequency of occurrence a small breach is estimated (WASH-1400 and German Risk Study) to be several orders of magnitude more likely than a large pipe rupture. This is not at all surprising considering: the higher probability of a leak before a rupture of a pipe, the large number of small pipes (typically 50 or more) and valves attached to the primary system, as well as the several thousand tubes in the steam generators of PWR's. Furthermore, Three Mile Island 2 (TMI) demonstrated that small breach LOCA scenarios are sensitive both to mitigation system availabilities and to operator error, and can have consequences of a safety and economic nature just as severe as those of large breaches. It is stressed, however, that continued efforts to utilize reactor operating experience to reduce challenges to safety systems, to better understand ECC behaviour and to improve operator training have resulted in a steady reduction in the overall probability of small breach LOCA's with severe consequences.

2.1 PWR-LOCA SCENARIOS

A PWR core sits low-down inside a large pressure vessel, Fig. 1.1. Under normal operating conditions subcooled, borated, water at about 150 bar and 290 °C flows into the vessel from the so-called "cold-leg" piping, down the annulus between the core-shroud and vessel walls (the downcomer), into the lower plenum and through the core. The coolant's temperature increases by about 30 K as it flows, typically at 3000 kg/m^2 s, through the core into the upper plenum and out into the hot-leg piping. The hot- and cold-leg piping or loops connect the reactor vessel to the plena of the steam generators. The steam generators are very tall (ca. 20 m) vessels enclosing the multiple, primary - to - secondary side, heat-exchanger tubes. In flowing through the steam-generator tubes the primary coolant's temperature is reduced to the cold-leg or core inlet temperature. Coolant flows in the primary system are driven by a main coolant pump in each cold-leg. The number of pumps, loops and steam generators varies between plant designs. A pressurizer with a steam cushion allows the coolant to expand and contract freely to accommodate temperature changes.

From a LOCA point of view several primary system characteristics are worthy of note:

- Because the primary system's coolant is everywhere subcooled, except in the pressurizer, extremely rapid decompression to saturation (ca. 50 bar) occurs following an almost instantaneous rupture of the primary system piping. The subsequent motion of decompression waves through the piping and vessels induces severe dynamic loads on internals. Core internals have to be designed to withstand these dynamic loads without severe deformation in order to ensure that a coolable geometry is retained during a large breach LOCA. Severe deformation could also prevent the control-rods from entering the core. This is not essential during a large-breach LOCA on a PWR, because coolant voids and the boron injected with the ECC will anyway make the reactor subcritical. It is nevertheless good engineering practice and essential for small breaches to ensure that the reactor is scrammed. The decompression wave loads are not discussed further in this SOAR.

- The massive outflow of coolant from a large-breach in a cold-leg pipe not only decompresses the primary system coolant, but halts then reverses the normal flow of coolant through the core, Fig. 2.1.1. Almost immediately a boiling-crisis or departure from nucleate boiling (DNB) occurs followed by a cladding temperature excursion which, within a few seconds, depending on stored energy, peaks at typically 800 °C. For other breach locations and sizes DNB may also occur as a result of decompression and the automatic trip (in some plants) of all main coolant pumps simultaneously with the LOCA, but then the initial cladding temperature excursion tends to be milder. The subsequent cladding temperature history depends on the coolant dynamics in the primary system during blowdown; the cladding may or may not be cooled significantly or it may even be rewetted prior to core uncovery. For this reason the DNB induced cladding temperature excursion for a PWR becomes a very important initial condition for any subsequent core-uncovery induced excursions. This contrasts with the BWR initial dry-out excursion, which is always rewetted prior to core uncovery.

- All PWR coolant piping connections to the reactor vessel are above core height, which makes it possible to reflood an uncovered core for any breach in the primary system piping. During all LOCA situations this core reflooding or recovery process is driven by the relatively small hydrostatic head of the coolant in the downcomer annulus. Breach size and location, steam generation and condensation, the injection of emergency coolant, and the distribution of coolant throughout the primary loops can all work against or with this hydrostatic head and so influence the reflooding process. In other words, the total primary system behaviour governs the LOCA history in a PWR, in contrast to a BWR where phenomena in the core and pressure vessel are mainly controlling.

2.1.1 PWR-LOCA Global Classification (Table 2.1.1)

The spectrum of possible breach sizes in a PWR ranges from a double-ended rupture of a main coolant pipe to a small leak anywhere in the primary system. LOCA scenarios vary with breach size, breach location, reactor type, ECC availability and main coolant pump status. Nevertheless, it is common practice to classify PWR-LOCA's globally in terms of breach-size and to discuss separately the influence of breach location and other parameters. Such a global classification is illustrated in Table 2.1.1.

2.1.1.1 PWR Large-Breach LOCA

The upper end of the large-breach LOCA spectrum, the double-ended pipe rupture, is clearly defined. Such a large breach involves a very rapid depressurization, a highly turbulent, almost homogeneous blowdown and almost complete emptying of the primary system. Blowdown and core uncovery take tens of seconds and the course of events is largely governed by the critical flow discharge from the breach. Reactor shutdown or scram occurs automatically and almost immediately following the rupture. Core cooling departs from nucleate boiling during the first second or so of blowdown and this is followed soon after by core uncovery. The fuel cladding thereupon undergoes rapid temperature excursions, which bear a good likeness to the classical, double-peak, characteristic (see Fig. 2.1.8).

When the primary system pressure falls below the injection pressures of the various ECC systems (see Appendix A1.1), borated coolant enters the primary system and flows through the available paths to refill the lower-plenum and then to reflood and finally recover the core. Available paths depend on the break and ECCS injection locations. At the end of the reflood period, which is typically of several minutes duration, the low pressure injection systems continue to operate to dissipate residual heat. There is neither time nor necessity for operator intervention prior to core reflood.

The large-breach LOCA is divided classically into more or less separate phenomenological windows or periods; blowdown, refill and reflood. With normal availability of the ECCS there may be some overlap of the blowdown and refill periods. In plants where emergency core coolant is injected into both hot- and cold-legs, coolant reaches the upper-plenum almost immediately. The reflooding process then becomes bi-directional and somewhat more complicated, see Section 2.1.3.1. Nevertheless, the classical division is a fairly realistic one for all PWR's.

2.1.1.2 PWR Very-Small-Breach LOCA

At the other end of the spectrum are the very small breaches. These are defined by a leak discharge which can be completely compensated by the volume- or chemical-control systems. The upper bound for very small breaches is clearly defined by the capacity and availability of these systems, which are normally classed as operational rather than safety systems. If they are functioning normally, trip-levels for the actuation of ECCS are not reached, and the leak is detected by the imbalance in volume control system injection and let-down flows. The primary system continues to operate under single-phase, forced convection, conditions with the core always covered. On detection of the leak the operator initiates a reactor shutdown, removing heat through the secondary-side with a cooldown rate no greater than ca. 55 K/h. Since it would take several hours for the storage tanks supplying the volume control systems to empty, the operator has ample time to detect the leak and shut-down the reactor. Should the leak remain undetected, or the chemical and volume control systems fail, the very small leak becomes a small-breach LOCA, which is mitigated by the ECCS.

2.1.1.3 PWR Small-Breach LOCA

Small-breach LOCA's or leaks are characterized by an extended period (this can be tens of minutes to several hours at the lower end of the spectrum) after the occurrence of the breach, during which the primary system remains at a relatively high pressure and the core remains covered. As soon as the pumps are tripped, either automatically or manually, gravity controlled phase separation occurs and the flow and distribution of coolant inside the primary system are dominated by gravitational forces. The subsequent sequence of events, whether or not the core uncovers and is recovered or reflooded, depends not only on the location, shape and size of the breach, but also on the over-all behaviour of the primary and secondary systems. This behaviour is strongly influenced by both automatic and operator initiated mitigation measures.

The small-breach LOCA spectrum may be divided into two subcategories, Table 2.1.1:

a) Those pipe breaches at the lower end of the spectrum that require the secondary side as a heat sink, in order to reduce pressure to where ECCS can inject.

   For this subcategory of small breaches, the primary system is cooled down through the steam generator at between 50 and 100 K/h, either automatically or by the operators following procedures. The ECC systems automatically maintain primary system inventory and no core uncovery occurs. After the operator has tripped the main coolant pumps, the largely single-phase conditions in the primary system will change from forced circulation to natural circulation. The operator may also have to control the high-pressure injection systems to avoid overfilling the primary system.

Small leaks through inadvertently-open or stuck-open primary-system safety- or relief-valves, a breach in a steam-generator tube or a postulated (hypothetical) leak in the lower-plenum of the reactor pressure vessel all fall into the small breach LOCA category. The normal mitigation measures for small breaches, combined with some additional procedures, assure that the core is adequately cooled during these events, which are discussed in Section 2.1.5.1.

b) Those pipe breaches for which the energy flow through the breach is large enough by itself to depressurize the primary system.

Depending on available ECCS, the breach size and location, and the primary system's geometry, core uncovery and recovery may occur more than once as inventory is redistributed in the primary system or is lost through the breach.

The classical phenomenological windows of blowdown, refill and reflood are not really applicable at the small-breach end of the spectrum. System depressurization is relatively mild and complete uncovery of the core does not necessarily occur. Core uncovery and reflood are by themselves appropriate windows.

2.1.1.4 PWR Intermediate-Breach LOCA

Small breach LOCAs are bounded by the very small leaks, which can be compensated by operational systems, and by the large-breaches, which are not dominated by phase separation during the blowdown period. The upper bound of a small-breach and the lower bound of a large-breach cannot be defined exactly, because characteristics of both small- and large-breach LOCA's may be evident for some breach sizes. Nominally, the bound is defined as a breach area ca. 0.25 of the maximum single pipe rupture area. This is an anachronism from the time when separate codes were used to analyse small- or large breaches. Since modern codes have the capability to analyse the complete breach spectrum, and since the overlapping breach-size is very plant specific, the overlap is loosely and simply defined as the intermediate breach-LOCA part of the PWR breach spectrum (Table 2.1.1), which needs no further discussion from a phenomenological viewpoint.

2.1.2 Dominant Phenomena for Large-Breach LOCA's in a PWR

A typical LOCA scenario for the most common PWR, which has U-tube steam generators and cold-leg ECC injection, is first described. Scenario modifications related to other PWR designs, such as those with:

- hot-leg and cold-leg (double) ECC injection, and
- once through steam generators with upper-plenum to annulus vent-valves,

are then addressed in Sections 2.1.3.1 and 2.1.3.2, respectively.

The course of events and the phenomena involved are described for a guillotine rupture of a main coolant pipe between the pump and the reactor pressure vessel, Fig. 2.1.1. This cold-leg rupture is a design-basis case which is commonly analysed, because it tends to give the highest cladding temperatures within the spectrum of large breaches.

Flows in the primary system develop somewhat differently for other breach locations, such as a hot-leg rupture, but nearly all of the phenomena involved remain the same. Particular aspects for hot-leg breaches are discussed in Section 2.1.2.4. As breach sizes get smaller, the sequence of events is delayed, but no new phenomena are encountered until phase separation begins to occur at intermediate to small breach sizes.

2.1.2.1 PWR, Large Cold-Leg Breach, Blowdown

Immediately following breach initiation, the pressure in the primary system falls to saturation pressure at a rate dictated by breach opening time and the transit time of rarefaction waves through the primary system, typically ca. 100 ms. Voids form almost immediately in the core reducing reactor power to decay heat levels, independently of control-rod scram, through the corresponding decrease in moderator density and core reactivity. Reactor protection signals (such as pressurizer level-low, pressurizer pressure-low, containment pressure-high) initiate a reactor scram, start-up of ECC systems, containment isolation, trip of the main coolant pumps on some plants and trip of the main turbine.

Critical flow discharge rates during the initial depressurization are extremely high due to subcooled water near the breach. In the broken loop the flow reverses from the vessel to the breach. Core flow may reverse, or stagnation conditions may arise for a short period with upflow at the core outlet and downflow at the core inlet, as indicated in Fig. 2.1.1. Departure from nucleate boiling (DNB) occurs on the hottest rods in the core within the first second and then spreads radially and axially to the whole core. In spite of the very rapid reduction in core power, an abrupt cladding temperature increase occurs. The ensuing rise in cladding temperature and its extent depends on the energy stored in the fuel (i.e. its temperature profile prior to the accident) and the time when each part of the cladding in the core experiences DNB.

After the pressure has fallen to saturation, fluid in the primary system flashes and a two-phase mixture is discharged from the breach. Since the critical discharge rate is thereby reduced, the depressurization slows down. Void propagation from flashing first extends to the hotter regions of the primary system; the core, the vessel upper plenum and upper-head, and the hot-legs, and then to the colder regions. The discharge rate from the vessel-side of the breach tends initially to be about twice that from the steam generator side. Flow through the core in an upward direction may be briefly re-established when the broken-loop, cold-leg, breach-flow becomes two-phase and coolant is still being supplied by the pumps coasting down. Upflow may be reinforced by flashing of the coolant in the lower plenum. The upflow is followed by a sustained period of downwards core flow that lasts almost to the end of blowdown.

The downwards core flow is fed by coolant flowing back through the hot-legs from the intact-loops and their steam-generators. Flashing and draining of coolant in the vessel's upper head contribute to this flow. The intact-loops' steam-generators also drain through their cold-legs into the lower-plenum of the vessel and out through the breach. Coast-down of the main coolant pumps in the intact loops permits this draining process.

The broken loop steam-generator drains directly to the breach, the through-flow rapidly accelerating the main coolant pump rotor. The steam-generator also drains partly via the hot-leg and core. Water is discharged from the pressurizer for the first 10 s or so and this contributes to the core downflow until steam is discharged as the pressurizer empties.

High-pressure injection systems start to inject into the cold-legs. Hot-leg injection is discussed in Section 2.1.3.1. When the primary system's pressure and temperature fall below those of the secondary-side, reverse steam-generator heat-transfer occurs. At lower pressures the accumulators start to inject into the cold-legs. At first, emergency coolant accumulates in the cold-legs. It is entrained and carried round the downcomer annulus and out to the breach by the counter-current-flow of steam from the lower-plenum, as indicated in Fig. 2.1.2. Towards the end of the blowdown period, this so-called "downcomer bypass" process stops and the injected coolant "penetrates" the downcomer. At the end of blowdown most of the primary system is filled with steam, except for the vessel's upper head and lower plenum regions, and those regions collecting injected coolant.

During the blowdown period, which lasts typically ca. 20 s, both the upward and downward flows of coolant through the core have a dominating influence on the DNB induced clad-temperature excursion peak and the post-peak core cooling. This cooling continues until the core flow subsides then cladding temperatures, now driven by decay heat, rise again.

Throughout the blowdown period flows in the primary system are on a local scale one-dimensional. Multi-dimensional effects are very evident though during accumulator-bypass and downcomer-penetration.

2.1.2.2 PWR, Large Cold-Leg Breach, Refill

Since there is a period of overlap between the blowdown and refill windows, the coolant inventory in the pressure vessel reaches a minimum before the blowdown has ended, i.e. when the primary system pressure has equalized with the containment pressure. This minimum does not necessarily correspond to the ECC injection rate exceeding the breach-discharge rate, which solely indicates an increase in primary system inventory. Injected coolant is distributed throughout the primary system and the high flow rates to the breach also have a sweep-out effect.

The refilling of the lower-plenum is largely governed by how quickly the accumulator injected ECC can penetrate the downcomer annulus and reach the lower plenum. Steam flows limiting this injection are generated not only by the flashing of coolant during the overlapping blowdown window, but also by the energy stored in the downcomer annulus walls, which is transferred to ECC coolant as annulus penetration begins. As the blowdown comes to an end, the counter-current flow of steam is reduced. By this time massive quantities of ECC are being injected. Both high and low pressure injection systems are operational but their contributions to ECC are initially small compared with those from the accumulators. Multi-dimensional effects and steam condensation break-down the remaining steam-flow barrier allowing injected coolant to flow down the downcomer and fill the lower plenum progressively. The low-quality mixture first cuts off the steam flow path around the lower edge of the core barrel. Shortly afterwards, it reaches the bottom of the core (see Fig. 2.1.3). This signifies the end of the refill period and the beginning of core reflood.

The refilling process lasts typically ca. 20 s and during this period the core experiences nearly adiabatic heating. Some steam cooling is present and, if temperatures are high enough, zircaloy-steam reactions begin. Finally, a period of reverse breach flow may occur as the ECC condenses steam in the primary system. This can cause the primary system pressure to fall below that of the containment.

2.1.2.3 PWR, Large Cold-Leg Breach, Reflood

As soon as the ECC reaches the hot-fuel rods at the bottom of the core a quench front is formed on the fuel rods and large amounts of steam are generated by the energy released from the rods at a high temperature. This steam produces a back-pressure opposing the driving head of coolant in the annulus and thereby slowing or even reversing the water-level rise in the core. Thus, reflooding of the core proceeds in an almost quasi-steady manner with level oscillations occurring in both the core and downcomer driven by the varying back-pressure in the core.

Steam-binding, as this back-pressure is called, is a complex phenomena largely controlled by the pressure-drop of the steam and entrained droplet flows leaving the vessel and escaping to the breach through the primary system loops, Fig. 2.1.4. The pressure-drop is augmented by:

i) additional steam generated outside the core by heat-transfer from the secondary side or by stored energy in hot-pipes or vessel walls. Entrained liquid at the vessel outlet, injected coolant or water remaining in the loops at the end of blowdown, or even leakage through steam generator tubes provide the sources for this steam production.

ii) low reactor system pressures, which increase the specific volume of the steam in the loops. The system pressure tends to follow the containment pressure.

iii) nitrogen injected into the cold-legs after the coolant has been driven from the accumulators. Gas injection can completely change the flow and heat-transfer characteristics of the loops. On the one hand it can enhance the liquid flow into the core and thereby increase the core reflooding rate. On the other hand it can interfere with heat-transfer and steam condensation.

Emergency core coolant injected into the cold-legs not only condenses steam, thereby mitigating the loop pressure-drop, but also increases the head of water in the annulus, which is driving the reflood process. (Hot-leg injection is discussed in Section 2.1.3.1).

Once the accumulators are empty, or isolated, the low pressure injection systems supply the coolant necessary to continue and complete the reflooding process. This process becomes somewhat complex, Fig. 2.1.4, in that part of the water entering the core quenches the bottom of the fuel rods, bringing the cladding temperature down to saturation, whilst the rest is driven upwards through the core as a mixture of steam and entrained droplets. The mixture provides some cooling at upper core elevations, where the maximum cladding temperatures prior to the final quench are reached. De-entrainment of liquid may occur at the upper core tie-plate and on the forest of structures in the upper-plenum. Liquid films on these structures form. Droplets may be entrained by the steam-flow to the hot-legs or may fall downwards and back into the core. These droplets and films can lead to the formation of a pool of water in the upper plenum and/or a quench front which propagates downwards into the core (top-down quenching).

During this reflooding process the core flows and quench fronts are two-dimensional. The hottest core regions may experience upward flows and the cooler peripheral regions may experience liquid fall-back and early quenching. Some cross-flow will occur between these regions. In the hotter regions of the core, clad swelling and some bursting may occur during the post-DNB excursion. Swelling may introduce very localized flow blackages, but these do not significantly interfere with core cooling.

Fuel rod cooling differs between core regions, depending on the local flow situation. In the unquenched portions of the core the rods experience inverted-annular or dispersed-flow film-boiling heat-transfer soon after the beginning of the reflood window. In the dispersed-flow regime the heat-transfer effectiveness is largely governed by the density of entrained droplets. Sufficient heat-transfer is provided by these regimes to turn-around the cladding temperature excursion prior to quenching. Flow oscillations and the presence of spacer grids both have a heat-transfer enhancing effect. In the vicinity of the quench fronts a very steep vertical temperature gradient is apparent. Fuel elements of lower power will quench well before those in higher power regions. Radiation transport will only make a significant contribution if cladding temperatures exceed about 800 °C.

The end of the reflood period, which lasts typically ca. 150 s, is signified by complete quenching of the core. Thereafter, the water inventory in the primary system increases rapidly until the coolant lost through the breach balances that injected. Continued injection of coolant into the primary system enables decay heat to be transported to the containment from where it is transferred to an ultimate heat-sink (see Appendix A1.1) for long-term cooling purposes. This denotes the end of the large-breach LOCA condition.

2.1.2.4 PWR, Large Hot-Leg Breach

As far as the phenomenology is concerned the blowdown, refill and reflood scenarios for a hot-leg breach are very similar to those described above for a cold-leg breach. They differ in the following aspects, which make the hot-leg breach less severe from an emergency core cooling point of view:

- there is no core flow reversal during blowdown, a generally upwards flow through the core to the breach is experienced throughout.

- steam produced in the core can readily escape to the breach, so that coolant injected into the cold-legs does not bypass the core. Indeed, downcomer penetration, refilling and reflooding occur more rapidly because steam-binding effects are minimal.

Flows to the hot-leg breach will tend to maximize the steam upflow and entrainment during the reflood period, which will enhance core-cooling.

2.1.3 Large-Breach LOCA Phenomena Related to Particular PWR Plant Designs

2.1.3.1 Simultaneous Hot-Leg and Cold-Leg Injection

Injection of emergency core coolant directly into the hot-legs at some distance from the upper plenum (c.f. Fig. 1.1) was incorporated into some early PWR designs and then later removed. For these early designs it was feared that steam produced at the steam generator tubes would increase the pressure drop in the loops, which would enhance steam-binding back-pressure and significantly delay core reflooding. This fear has since proven to be unfounded for modern plants with combined injection.

Some modern reactors have hot-leg injection, which is directed towards the upper plenum (Fig. 1.1) and has sufficient capacity to condense all the steam produced during the reflood window. This mode of injection, combined with normal cold-leg injection, is generally beneficial from an emergency core cooling point of view. It does introduce additional phenomena into the large, cold-leg, breach scenario:

a) Towards the end of the blowdown period, coolant injected into the hot-legs flows via the upper plenum, downwards, through the core, then upwards through the annulus and out to the breach. Part of the core is quenched by this downflow.

b) Downflow continues during the refill-window until the flow-path to the annulus is cut-off by the rising water level in the lower-plenum. This downflow improves core cooling, reduces core temperatures and leads to early quenching of parts of the core towards the end of the blowdown and refill windows.

c) The most significant effects of hot-leg injection into the upper plenum are noticed during the reflood window. As the lower regions of the core are quenched, the steam so generated flows counter-current to the ECC water. Steam condensation and the momentum of the injected fluid lead to the immediate local breakdown of any counter-current flow-limiting condition at the upper tie-plate adjacent to the hot-leg injection locations. Water then penetrates the core from above and two, essentially vertical, flow regions are established in the core:

   - one region where the water penetrates the core. Fuel rods are rapidly quenched and excess water contributes to the bottom-up flooding rate.

- an adjacent region with a two phase upflow exhibiting bottom-up flooding characteristics.

The bottom flooding rate under these conditions is essentially determined by steam condensation in the upper plenum and in the hot-legs. This condensation acts as an effective steam sink for the additional steam produced by the more rapid quench process. Such additional steam augments the cooling and contributes to precursory cooling of the unquenched core regions.

The double mode of ECC injection enhances core cooling and increases the core flooding rate above that for cold-leg injection alone. This reduces peak cladding temperatures and provides an earlier quench of the entire core. Its efficiency relies on the hot-leg injection rate being large enough to condense the steam produced in the core and internals. The improved core-cooling is at the expense of increased complexity during reflood. Phenomena are more complicated than those identified in Fig. 2.1.4; multi-dimensional condensation occurs in the upper-plenum and hot-legs, and multi-dimensional flow regions occur in the core itself.

Note that for a <u>large breach in the hot-leg</u>, steam upflow through the core to the breach during the reflood period will be a maximum. In spite of this, much of the coolant injected into the upper plenum reaches the core.

2.1.3.2 <u>Vent Valves on Plants with Once Through Steam Generator</u>

For the most part the large-breach LOCA behaviour of a PWR with once through steam generators (Fig. 1.1), whether for a cold-leg breach or a hot-leg breach, is very similar to that for corresponding breaches in plants with U-tube steam generators. The <u>cold-leg breach</u> behaviour differs due to flows through multiple vent or check valves installed on the core barrel between the upper plenum and the downcomer annulus, just above the inlet nozzles.

The main purpose of these valves is to provide a low-resistance path from the upper plenum to the downcomer annulus and cold-leg breach, thereby bypassing the potentially much greater flow resistance of the very tall steam generators. By venting steam during reflood, the valves minimize steam binding and increase the core reflooding rate.

During normal reactor operation the valves are kept shut by the discharge pressure of the main coolant pumps. After the occurrence of a cold-leg breach the valves will open intermittently during blowdown and refill, venting a two-phase mixture to the breach. If ECC is being injected into the cold-legs and/or downcomer during refill and reflood, any steam vented by the valves will either be condensed in the annulus or escape to the breach.

2.1.3.3 Upper Plenum and Upper Head Injection

The phenomenology associated with upper plenum injection is similar to that described for the hot-leg injection directed towards the core, Section 2.1.3.1. Upper plenum injection from low-pressure accumulators is aimed at core cooling during the refill and reflood phases of a large-breach LOCA.

This contrasts with upper-head injection which, by means of high pressure accumulators discharging into the reactor vessel head, is designed to provide additional coolant flow through the core during the blowdown phase of a large, cold-leg breach LOCA. ECC from upper-head injection flows through the guide tubes and special upper-head injection columns directly on to the top of the core. During the downward core flow period, this ECC improves core cooling so that the reflooding phase begins with less core stored energy and lower fuel and cladding temperatures. Upper head injection accumulator valves close automatically before complete discharge of accumulator water occurs.

2.1.4 Dominant Phenomena for Small-Breach LOCA's in a PWR

A typical small-breach LOCA scenario for a PWR with U-tube steam generators and cold-leg ECC injection is first described. Scenario modifications related to PWR designs with:

- automatic cool-down through the secondary-side,
- hot-leg and cold-leg ECC injection, and
- once through steam generators with vent valves,

are then addressed in Sections 2.1.5.3 to 2.1.5.5, respectively.

By describing small cold-leg breach behaviour in some detail, an overview of the characteristic phenomena concerned with all small-breaches is provided, even though some of the phenomena may not actually occur following breaches at other locations. Within the limits of the small-breach categorization, discussed in Sections 2.1.1.3 and 2.1.1.4, the phenomena involved do not change with breach size, but amplitudes and durations certainly do.

Although core uncovery and reflood are appropriate windows for small-breach LOCA's, a full description of the small-breach scenario is best achieved by dividing the sequence of events into the following phases:

- initial depressurization
- void formation, phase separation, single- and two-phase natural circulation, and reflux condensation
- first core uncovery and quench, driven by loop-seals
- second core uncovery, driven by inventory loss
- accumulator injection and core reflood

2.1.4.1 PWR, Small-Breach, Initial Depressurization

A single-phase blowdown is initiated by the breach. Depressurization occurs and continues, at a rate dictated by breach size, until saturated conditions are reached in the hottest regions of the primary system. During this depressurization, which may take several 10 s at the lower end of the breach spectrum but just a few seconds for intermediate breaches, the pressurizer empties and reactor protection signals (pressurizer level-low, pressurizer pressure-low, containment pressure-high) initiate a reactor scram, start-up of ECC systems, containment isolation and trip of the main turbine. If the depressurization proceeds very slowly, the operator may initiate these signals manually. At saturation, boiling occurs in the upper regions of the core and voids are formed in the upper-plenum, in the hot-legs and in the upper-head region, see Fig. 2.1.5. The resulting steam production drastically reduces the rate of primary system depressurization.

Whilst the main condensor remains available, the secondary-side pressure follows the regulator set-point. If the main steam lines are isolated, the secondary side pressure increases to the set-point of the steam-generator safety-valves and is held roughly constant by valve actuations. In either situation both the primary and secondary system pressures and temperatures approach one another and stabilize, signifying the end of the initial depressurization period. Steam generator water levels are maintained by main, auxiliary or emergency feedwater systems.

2.1.4.2 PWR, Small-Breach, Void Formation, Phase Separation and Natural Circulation

The subsequent, global, pressure variation of the saturated primary and secondary systems, whether it remains on a plateau or sinks, depends on whether or not the breach size falls into the following subcategories, which were introduced in Section 2.1.1.3 and Table 2.1.1:

Subcategory a)
If the breach flow is not large enough to depressurize the primary system by itself, then, until ECC injection becomes effective, secondary-side cool-down, through secondary-side pressure control, governs the energy transfer and cool-down rate of the primary system. The subsequent LOCA behaviour then depends very strongly on those automatic or manual actions controlling the secondary-side cool-down rate. Until substantial ECC injection occurs the secondary-side pressure remains, for the most part, below the primary-side pressure throughout the event. On the other hand, if no action is taken, primary system pressure remains constant despite the steady loss of inventory through the breach.

Subcategory b)
If the breach flow is large enough by itself to depressurize the primary system, the primary system pressure falls below that of the secondary-side and heat is transferred back from the secondary-side to the primary. In this situation the primary side controls the cool-down rate of the secondary-side steam generators.

Main coolant pumps are tripped automatically, or by hand on the basis of operator procedures. After pump coast-down, single phase and then two-phase natural circulation develop in the primary loops and voids form in the upper regions of the steam generators. The high-pressure injection systems are not able to compensate the breach flow at this stage, so that the voids increase steadily in volume. When natural circulation flow over the top of the U-tubes cannot be sustained further, complete phase-separation occurs. The core experiences pool-boiling and countercurrent flow is set-up in the hot-legs with reflux condensation in the ascending parts of the U-tubes. This situation is shown schematically in Fig. 2.1.6. The actual behaviour is somewhat more complicated in that:

- bypass passages from the upper-plenum to the upper-head and to the cold-leg annulus influence the appearance and growth of voids in these regions,

- voiding in the steam generator tubes is not uniform and can be oscillatory in nature, particularly during and after the transition to countercurrent flow,

- the appearance and rate of growth of voids, and therefore the natural circulation and reflux condensation behaviour, may be different in each steam generator, which may lead to intermittent behaviour in loop flows.

If the separated-flow boundary approaches the breach, the critical flow will change from single-phase liquid, to two-phase low-quality. The actual sequence clearly depends on breach-location, size and orientation. Vapour and liquid pull-through further complicate the breach-flow conditions.

Since the core remains covered during this period any early, DNB induced, small temperature excursion, resulting from pump coastdown, is rapidly rewetted. Cladding temperatures remain very close to the saturated temperature of the coolant in the vessel, decay heat being transferred to the steam generators by the boiling and reflux condensation processes.

The duration of this void formation period may be 10's of seconds to 10's of minutes, it depends on breach size.

2.1.4.3 PWR, Small-Breach, First Core Uncovery and Quench, Driven by Loop-Seals

As primary system inventory loss and boiling at a high saturation pressure continue the voids in the pressure vessel and descending portions of the steam-generator tubes grow. The water level in the core is slowly driven downwards by the hydrostatic head of the two-phase mixture in the ascending portion of the U-tubes. Core uncovery occurs and the situation becomes as depicted in Fig. 2.1.7. How far core uncovery progresses depends on the depth of the loop-seal in the intermediate portion of the cold-leg between the main coolant pump and the steam-generator. How quickly and for how long the core is uncovered, again tens of seconds to minutes, depends on breach size and location. For example, a loop seal has no influence on small, hot-leg breach LOCA behaviour, because steam is discharged directly through the breach. As soon as vapour, either from the descending U-tubes, or from the uncovered cold-legs, can escape round the bottom of the loop-seal, it relieves the pressure-difference between the loops and water from the downcomer fully or partially refloods the core. This pressure difference is also influenced by the steam flow through the upper-plenum to downcomer bypasses.

For a cold-leg breach the breach-flow will change from single-phase to high-quality two-phase as the cold leg is voided. Clearly, breach uncovery and flow transition at a hot-leg breach will occur much earlier. Steam flow at the breach increases the depressurization rate of the primary system. For a subcategory a) breach (Section 2.1.1.3), if depressurization is now faster than the controlled depressurization through the secondary side, the breach subcategory changes from a) to b) and reverse heat-transfer from the steam generator to the affected loop occurs.

Cladding temperatures increase in the uncovered portion of the core, which are exposed to steam cooling. Covered portions are cooled by pool-boiling.

Those regions of the core reflooded from below after loop-seal clearing are rapidly quenched. Those portions not reflooded, and this is governed by breach size and loop-seal elevation, experience a continuing temperature excursion.

The above behaviour is further complicated in that:

- the depressed coolant level in the core may oscillate as a result of the dynamic condensation processes in the loops and as the flows to and from the cold-legs, as well as to the breach, vary.

- it is unlikely that all the loop seals will clear together. Rather, one will clear first relieving the pressure-difference across all the loop-seals. This will influence the subsequent coolant mass distributions in the loops.

- reverse heat-transfer in a steam generator may reverse the natural circulation flow in the affected steam generator, leading to temperature stratification and condensation of steam in the dome of this steam generator.

2.1.4.4 PWR, Small-Breach, Second Core Uncovery Driven by Inventory Loss

Whilst the first core uncovery is caused by a maldistribution of coolant in the loops, the second core uncovery is driven by loss of primary system inventory alone; the core is simply boiling dry. Depending on the breach-size and the primary-system's geometry, in particular the depth of the loop-seal, the second core uncovery may overlap with the first, or may follow the first after loop-seal clearing.

Core uncovery will continue until the ECC injected into the primary system reaches the core and compensates the loss through the breach. Except for breaches on the vessel itself, the breach flow will by this time be steam, which aids in the depressurization of the primary system. The duration of core uncovery, seconds to minutes, now depends on breach size, primary system cooldown, ECC availability and ECC effectiveness. In general, there is no refill window for small-breaches in PWR's.

Cladding temperatures in uncovered portions of the core increase and superheated steam may appear in the core, in the upper plenum and the hot-legs. Condensation of steam continues in the steam generators at a rate dictated by secondary-side cooldown.

2.1.4.5 PWR, Small-Breach, Accumulator Injection and Core Reflood

Coolant will be injected by the available ECC at a time and rate governed by primary system pressure. The high-pressure systems will inject first and, for breaches at the lower end of the spectrum, these may arrest or retard the core uncovery process. Significant quantities of ECC will be injected when the system pressure falls below the accumulator set-point. Condensation of steam by ECC and boiling of the injected coolant lead to a complex system behaviour under these conditions:

- when the accumulator injects into the cold-legs, condensation occurs at or near the injection points, which reduces the primary system's pressure. Injected coolant flows down the annulus mixing with the hot water there and transferring heat from the annulus walls.

- the increase in hydrostatic head in the annulus drives the core reflooding process, which begins to quench the uncovered portion of the core from below, thereby producing large quantities of steam.

- steam produced slows then reverses the pressure decrease of the system. At the same time the accumulator driving pressure decreases as the driver gas expands. Gas pressure in the accumulator is a complex function of the gas volume and the heat transfer processes occurring there. Eventually, primary system and accumulator pressure equalize, and accumulator injection stops.

This cycle may repeat itself several times as primary system pressure continues to fall due to flow through the breach and/or secondary side cool-down. Core reflooding may thus occur continuously or intermittently with considerable oscillatory behaviour appearing in the vessel and loop flows.

The complex phenomena involved with reflooding of the core during a small-breach LOCA, such as bottom-up quenching, entrainment and de-entrainment of coolant, and top-down quenching, are similar to those already described in Section 2.1.2.3 for large breaches. Core heat-up, quench and cool-down are also similar. The main difference is that reflooding takes place at somewhat higher pressure (several bar) and may progress much more slowly.

During the reflood phase or at its end, the primary system pressure falls below the set-point of the low-pressure injection systems. The accumulators are isolated automatically or by hand to prevent nitrogen entering the system and reducing heat transfer effectiveness. The primary system is then refilled by ECC, which may lead to a small pressure increase as steam is compressed in the pressurizer. Normal residual heat removal is initiated for long term cooling purposes. Normal make-up water, or suction from the containment sump, maintains system inventory. From a core cooling viewpoint the small-breach LOCA is ended.

2.1.5 The Influence of Operator Procedures and Particular PWR Plant Designs on Small-Breach LOCA Scenarios

2.1.5.1 Inadvertenly Open Valves, Steam-Generator Tube Ruptures and Pressure Vessel Leaks

TMI has shown that leaks through valves mounted on pressurizers can lead to hold-up of water in the pressurizer and, in consequence, to a false sense of security with regard to water levels in the reactor vessel. Since TMI, direct monitors of pressurizer valve positions have been installed to enable the operator to recognize the stuck-open valve condition more readily. Direct indications of reactor vessel liquid level and/or of primary system subcooling tell the operator whether or not the pressurizer level readings are trustworthy. The operator must also take specific procedural measures; to scram the reactor, to try to isolate the leaking valve, to cool down the primary system via the secondary-side and to start-up the ECC systems, if these do not occur automatically. Critical flow through valves is the one new phenomenon introduced by a stuck open valve.

After the occurrence of a steam-generator tube rupture primary system coolant flows into the affected steam generator and mixes with secondary-side coolant. N16 release to the secondary side is the first indication of a tube leak or rupture. The operator, following procedures, takes action to avoid any direct release of active steam to the environment from the affected steam generator through its safety valves. Typically, the reactor is shut-down automatically, or by hand, and energy transfer to the affected steam generator is minimized by, for example, tripping the main coolant pump for this loop. By depressurizing the primary system, it is possible to equalize the primary and secondary system pressures, before the actuation pressure of the secondary-side safety-valve is reached. As the primary system cools down further, some flow and energy transfer occurs from the secondary-side to the primary system through the rupture.

Vessel leaks introduced no new phenomena into the small-breach LOCA scenario.

In summary, other than additional procedures, critical flow through valves, a possible false indication of vessel water level and back-flow into the primary system, <u>none of these events</u> introduce anything new or unexpected into the small-breach LOCA phenomenology.

2.1.5.2 Cooldown of the Secondary Side by the Operator

After the small-breach event has been identified as such, through primary system pressure and levels and/or containment pressures and activity, the operator following procedures must manually initiate cooldown of the secondary side. The cooldown rate is chosen to reduce primary system temperatures at a rate between ca. 50 and 100 K/h. For subcategory a) breaches discussed in Section 2.1.1.3, the operator has ample time, typically half-an-hour or more, to do this without fear of cladding temperatures during core uncovery exceeding allowable limits.

Nevertheless, the earlier the operator begins the cooldown the less severe the core uncovery and core temperature excursion will be. The pressure plateau following initial depressurization is shortened, ECC injects earlier and the injection rate increases more rapidly. Indeed, depending on the leak-size and the time to initiate cooldown, core uncovery may be avoided altogether.

Only if operator action to cool down the primary system is delayed and high pressure injection systems are not available, will the core be uncovered as inventory is lost through the breach. It is then reflooded by the ECCS as the primary system's pressure continues to fall under the action of secondary side energy transfer, breach flow and cold ECC injection. If cooldown is not performed at all, and the borated water storage tanks are empty, the high-pressure recirculation of sump water may be manually aligned to inject water and avoid core uncovery. Not all plants have this capability, which requires series alignment of the low- and high-pressure injection pumps. These plants rely solely on timely cooldown through the steam generators.

If for any reason the steam-generators boil dry and primary system pressure increases above the shut-off head of the ECCS pumps, the operator may be able to bleed the system by opening a safety/relief valve on the pressurizer. This effectively changes the subcategory a) breach to subcategory b). By depressurizing the system and thereby allowing ECC to feed or inject coolant, the operator may be able to avoid or at least limit core uncovery. Since this "bleed and feed" procedure is for beyond design-basis situations it is not discussed further here.

As breach sizes increase they change to subcategory b) and the influence of operator actions on core uncovery scenarios diminishes, eventually becoming negligible.

2.1.5.3 Automatic Cooldown of the Secondary Side

Some plants have automated cooldown of the secondary side, which reduces primary system temperatures at a rate not exceeding ca. 100 K/h. Typically, two out of the following three criteria are used for cooldown initiation signals:

- primary system pressure-low
- pressurizer level-low
- primary containment sub-compartment pressure-high

The early initiation and the high rate of cooling down ensure a rapid depressurization of the primary system. So much so that, even at the lower end of the small-breach spectrum, the pressure plateau following initial depressurization no longer exists. ECC is injected very early on and the whole time sequence of events is shortened. Any core uncovery and temperature excursions characteristic of subcategory a) events are thereby minimized. Otherwise the basic LOCA scenario and phenomenology remain unchanged.

Plants with automated cooldown of the secondary-side usually have automatic isolation of accumulators also, to prevent nitrogen from getting into the loops.

2.1.5.4 Simultaneous Hot-Leg and Cold-Leg Injection during a Small-Breach LOCA

Some reactors have ECC injection into the hot-leg/upper-plenum, see Fig. 1.1, in addition to the normal cold-leg injection. This combined injection enhances the condensation of steam in the neighbourhood of the injection points and so increases the subsequent rate of depressurization of the primary system. Lower pressures lead to higher ECC injection rates and generally larger water inventories in the primary system and core. This is particularly beneficial for small, cold-leg, breaches because the depressurization rate from the inventory flow through the breach itself is least. On some plants high pressure coolant injection is connected solely to the hot-legs to avoid any risk of pressurized thermal shock.

ECC injected into the hot-legs/upper-plenum rapidly attains saturation. In a fashion similar to that described for large-breach reflood in Section 2.1.3.1 c, ECC flows through the upper-tie plate and down through the core in the regions adjacent to the injection locations. It adds to the coolant injected into the cold-legs which reaches the lower-plenum, thereby increasing the reflooding rate of the core.

Regardless of small-breach size and location, the LOCA scenario and phenomenology involved with combined injection, but for the condensation in the upper plenum and the downflow of coolant through the core, tend to be similar to that for cold-leg injection alone. Combined ECC injection does increase post-injection depressurization and cooldown rates of both primary and secondary systems. It improves the inventory balance in the primary system and enhances the reflooding rate of the core.

2.1.5.5 Small-Breach LOCA Behaviour in PWR's with Once Through Steam Generators and Vent Valves, Fig. 1.1

Reactor protection and safety systems on these reactors are similar to those on PWR's equipped with U-tube steam generators. LOCA-signals for the initiation of safety systems are also similar but, as an addition, an automatic increase in steam-generator coolant-levels is also initiated by these signals for reasons discussed in paragraph ii) below.

In general, the small-breach LOCA behaviour on these plants does not differ markedly from that already described in Section 2.1.4. Three specific characteristics do, however, influence particular scenarios.

i) Interruption of Natural Circulation Flow

The hot-leg outlet nozzles are at the top of the steam generators, see Fig. 1.1. From there coolant normally flows to the core by means of 180 ° bends and long fall-pipes: the well-known "Candy-Canes". The high elevation of these bends is no barrier to single-phase natural convection, which follows the automatic trip and coastdown of the main coolant pumps. Heating of the core-region fluid and the simultaneous cooling of the primary-fluid in the steam generators ensure a positive and substantial circulation.

The situation changes during a small-breach LOCA as soon as a net coolant inventory loss depressurizes the system to saturation conditions. Voids appear at the highest points in the hot-legs, the top of the Candy-Canes, driving the coolant levels down, which quickly interrupts the two-phase natural circulation flow. This is no different to the U-tube steam-generator behaviour described in Section 2.1.4.2, except that it occurs earlier. However, the subsequent behaviour during the system depressurization phase can be markedly different for subcategory a) small-breaches (Table 2.1.1). This is discussed in ii) below.

Interruption of natural circulation flow occurs for all but the smaller breach sizes, because breach-flow will initially exceed even the high-pressure ECC injection rate. Subsequent events, such as the appearance of steam voids at the vessel inlets to the hot-legs, may regenerate loop-flow intermittently, but this does not make a significant contribution to primary system depressurization.

ii) Boiler-Condenser Mode of Depressurizing the Primary System and its Interruption

For the small-breaches of subcategory a) the interruption of natural circulation flow will not hinder primary-side depressurization through secondary-side cooldown, provided steam-generator secondary-side water-levels are high enough to condense the steam in the short-leg of the Candy-Canes. As boiling continues in the core, steam so generated flows through the hot-legs to the top of the Candy-Canes and down into the steam-generators where it is condensed. To improve condensation efficiency the auxiliary feedwater, on receiving a LOCA-signal, automatically increases secondary-side water levels to the top of the tubes. Steam is thereby condensed on the primary-side as soon as the water-level there falls below the top of the tubes.

This very important and highly effective energy transport process is termed the "boiler-condenser" mode to distinguish it from the reflux-condensation mode described in Section 2.1.4.2. These modes differ in that:

- the boiler-condenser mode is more readily interrupted by a fall in secondary-side water-level; TMI provided a good example of this. To mitigate this condition auxiliary feedwater is actually sprayed into the tube bundle. This provides cooling, enhances natural circulation and improves primary side condensation under low secondary-side coolant inventory conditions. To interrupt reflux-condensation the secondary-side of a U-tube generator must first boil almost dry.

- the boiler-condenser mode is readily interrupted by an increase in primary-side water-level above that of the secondary side. In a U-tube steam-generator this can only occur with a simultaneous return to single-phase natural-convection over the top of the U-tubes. The Candy-Cane bend forestalls such a return in once-through-steam-generators until the steam blocking the bend is condensed or vented.

Because of this interruption in the boiler-condensor mode, the depressurization of the primary-system for a limited range of breaches in subcategory a) may occur intermittently; as depressurization begins and inventory is depleted, the levels in the Candy-Cane fall. Eventually ECC injection compensates the losses and levels increase until the boiler-condenser mode is interrupted. Pressure then increases and ECC injection is reduced. If the water levels in the Candy-Cane have not reached the top of the bend at the time when the ECC no longer balances the breach flow, the primary system's depressurization will be stalled. This condition is no threat to core-cooling and remains until the levels in the Candy-Cane are again depressed by voids to the steam-generator tubes, thereby allowing the boiler-condensor process to begin once more. Each cycle may take many minutes. Eventually, as decay heat decreases, either the steam blocking the Candy-Cane bends is condensed by conduction losses and/or sufficient ECC is injected to re-establish natural circulation over the bends. To avoid this intermittent depressurization, small-capacity vent-valves are installed on top of the Candy-Canes on some reactors. These valves may be manually opened to vent the steam which is blocking natural circulation.

iii) Reactor Vessel Vent Valves and Steam Condensation

The vent-valves, discussed in Section 2.1.3.2 for large-breaches, also play an important role during small-breach LOCA's. They provide a low resistance path from the upper-plenum to the downcomer-annulus and cold-leg, bypassing the much greater flow resistance of the very tall steam generators.

Towards the end of the pump coast-down period, the balance of flow forces on the vent-valves is such that the valves open, venting a two-phase mixture to the downcomer annulus. Steam so vented is either condensed by the relatively cool water in the annulus, or is collected at the top of the annulus lowering the water level there. Initially, several parallel two-phase natural circulation flows are established, some in the loops and others through the vents. The flow rates set up in each parallel path will depend on the path flow resistances. The flow continues through the vents even when natural circulation is interrupted in the loops by voids in the Candy-Cane bends. As primary system level is depleted the two-phase flow through the vents may change to steam flow.

As soon as the downcomer level has fallen to a level where steam comes into contact with ECC water injected into the cold-legs, the steam is condensed. Under these conditions a type of boiler-condenser mode is established between steam produced in the core and steam condensed by ECC in the annulus or cold-legs. This mode is independent of steam-generator conditions and can contribute significantly to the rate of depressurization of the vessel and the reflooding rate of the core. An additional benefit of steam flow through the vents, even without condensation, is the earlier change of a cold-leg breach flow to a two-phase discharge. This enhances depressurization, which enables ECC to be injected earlier to condense steam and to reverse vessel inventory depletion.

2.1.6 Concluding Remarks on PWR-LOCA Phenomenology and its Influence on Core Cooling

The phenomena having an important influence on the cladding temperature excursions during design-basis PWR-LOCA realistic scenarios have been identified in the previous sections, together with when and where they occur. Not all of them are important or even relevant in all situations, it depends largely on breach size, breach location and the location of ECC injection.

- Large-breach LOCA's involve a very early departure from nucleate boiling (DNB), which induces a very rapid cladding temperature excursion. This excursion is limited, but not necessarily quenched, by the flow dynamics in the core and loops during blowdown. It acts as a very important initial condition for subsequent core-uncovery driven temperature excursions.

  Breaking down the large-breach LOCA scenario into blowdown, refill and reflood windows or periods is a realistic procedure. Blowdown is characterized by a very rapid emptying of the primary system with core cooling controlled by core flow and the flow distributions in all the loops, and hence also by the location of the breach. Complete core uncovery occurs for all large breaches.

  The refill and early reflood windows are dominated by ECC accumulator injection and steam condensation processes. Condensation of steam allows the ECC to penetrate the downcomer during refill. Steam production and condensation throughout the primary system influence steam-binding back-pressures, which may hinder core reflooding. Core reflood and quench are very complex processes involving bottom-up and top-down quench fronts, steam generation, droplet entrainment and droplet de-entrainment above the core and in the loops. Typical, predicted, cladding temperature variations for a large-breach are shown in Fig. 2.1.8 a.

- Small-breach LOCA cladding temperature excursions are driven by core uncovery and recovery, the early DNB excursion either does not occur or is very rapidly rewetted. After coastdown of the main coolant pumps, core uncovery is influenced not just by the inventory losses and gains of the primary system, but also by phase separation and the coolant distribution in the loops; the latter manifesting itself through loop-seal and loop-seal clearing.

  Primary system inventory is a balance between coolant lost through the breach and ECC injected. Since both of these are strong functions of system pressure, primary system depressurization rate is a key parameter controlling the depth and duration of core uncovery, if it occurs at all. Two subcategories a) and b) of small-breaches were identified. For subcategory a) the breach flow, by itself, does not depressurize the system rapidly. Automatic or manually actuated cool-down of the primary system through the secondary-side steam generators assist system depressurization. The earlier this is done the better from a core-cooling viewpoint. For subcategory b) the breach flow itself is large enough to depressurize the system.

Small-breach LOCA behaviour is dominated by void formation throughout the primary system, which interrupts natural circulation flow and controls reflux-condensation. When ECC is injected, condensation in the loops as well as in the pressure vessel has a major influence on the distribution of coolant in the loops and core, and therefore a major influence on core cooling. The complex core reflood and quench processes for small-breaches are similar, but not identical, to those for large-breaches. Typical, predicted, cladding temperature variations for a small breach are shown in Fig. 2.1.8 c and for an intermediate breach in Fig. 2.1.8 b.

The relative importance of the phenomena involved in both large- and small-breach PWR-LOCA's is summarized in Chapter 4. Although many of these phenomena are globally one-dimensional in character, three-dimensional effects have to be considered. Mixing, boiling, entrainment and condensation of ECC injected coolant at the points of injection in the loops, in the upper plenum in the downcomer and in the core itself may all exhibit multi-dimensional behaviour. The sometimes unsteady, intermittent, nature of the loop flows, and of the core uncovery and recovery processes, must also be borne in mind. Analytic modelling of multi-dimensional, unsteady, non-equilibrium phenomena, is discussed in Chapter 5. Where a suitable modelling capability is lacking, experiments provide the necessary local or global correlations.

Finally, radiation heat-transfer and chemical energy production are not significant, if cladding temperatures remain below about 800 °C, which is the case for most realistic design-basis LOCA situations. Under these conditions fuel rod ballooning and bursting are localized, if they occur at all, and do not influence core cooling.

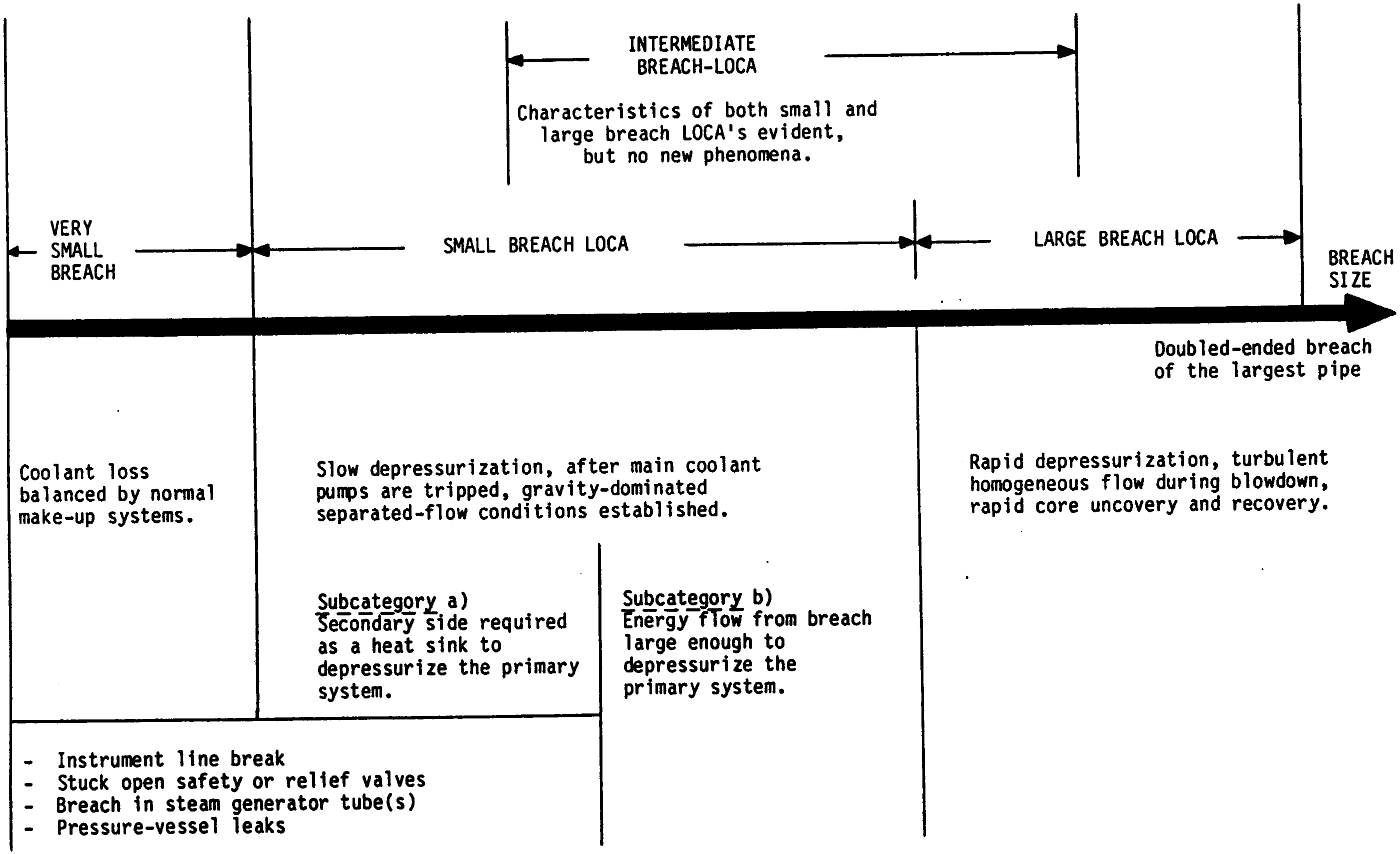


TABLE 2.1.1 GLOBAL CLASSIFICATION OF PWR LOCA-SCENARIOS

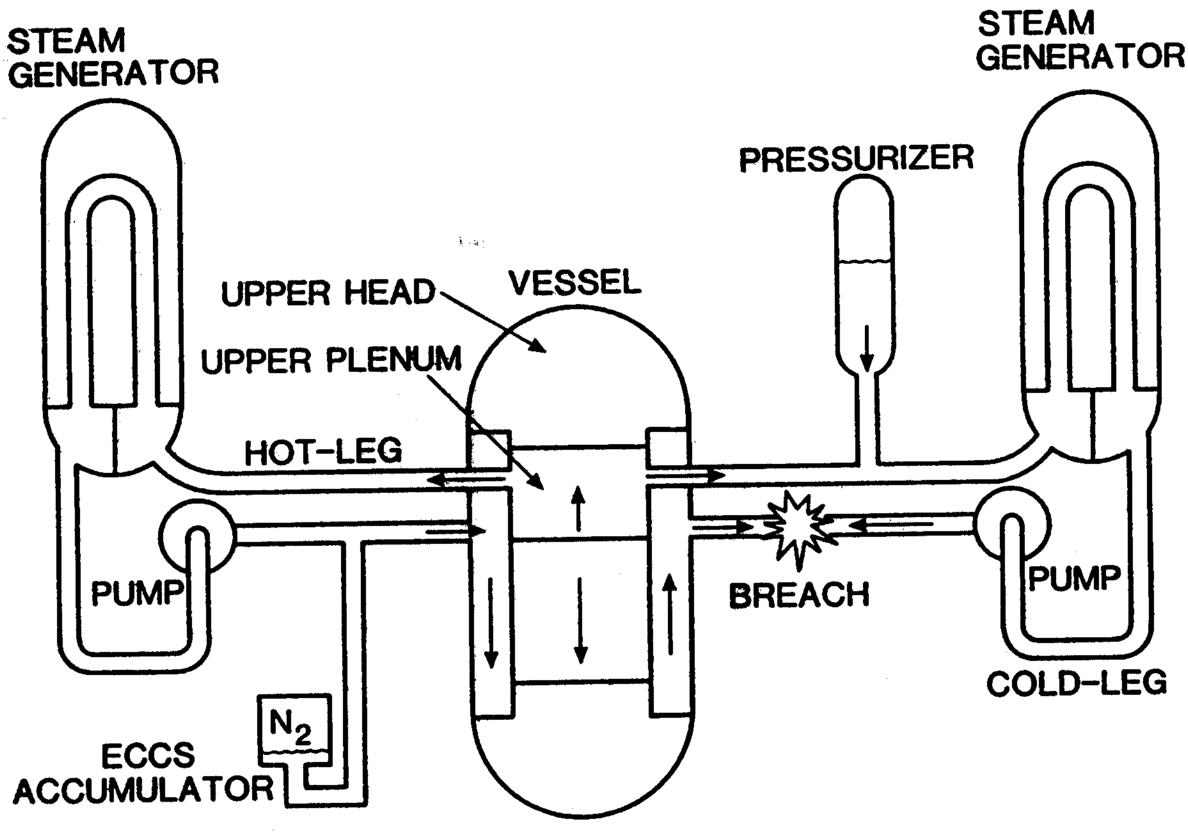


FIG. 2.1.1 FLOWS IN THE PRIMARY SYSTEM IMMEDIATELY AFTER A LARGE, COLD-LEG, BREACH IN A PWR

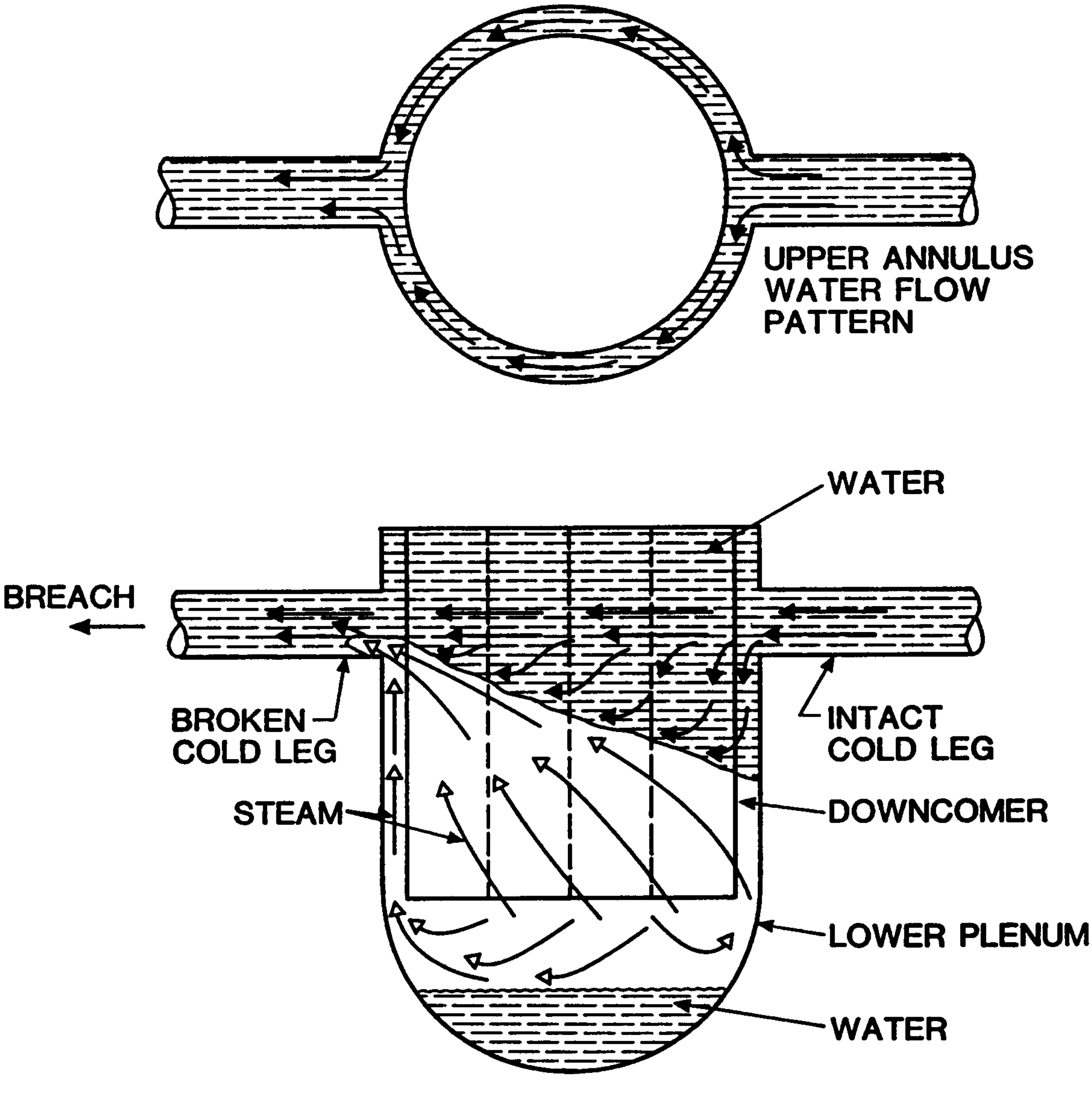


FIG. 2.1.2 SKETCH OF ECC BYPASS DURING BLOW-DOWN THROUGH A LARGE COLD-LEG BREACH IN A PWR

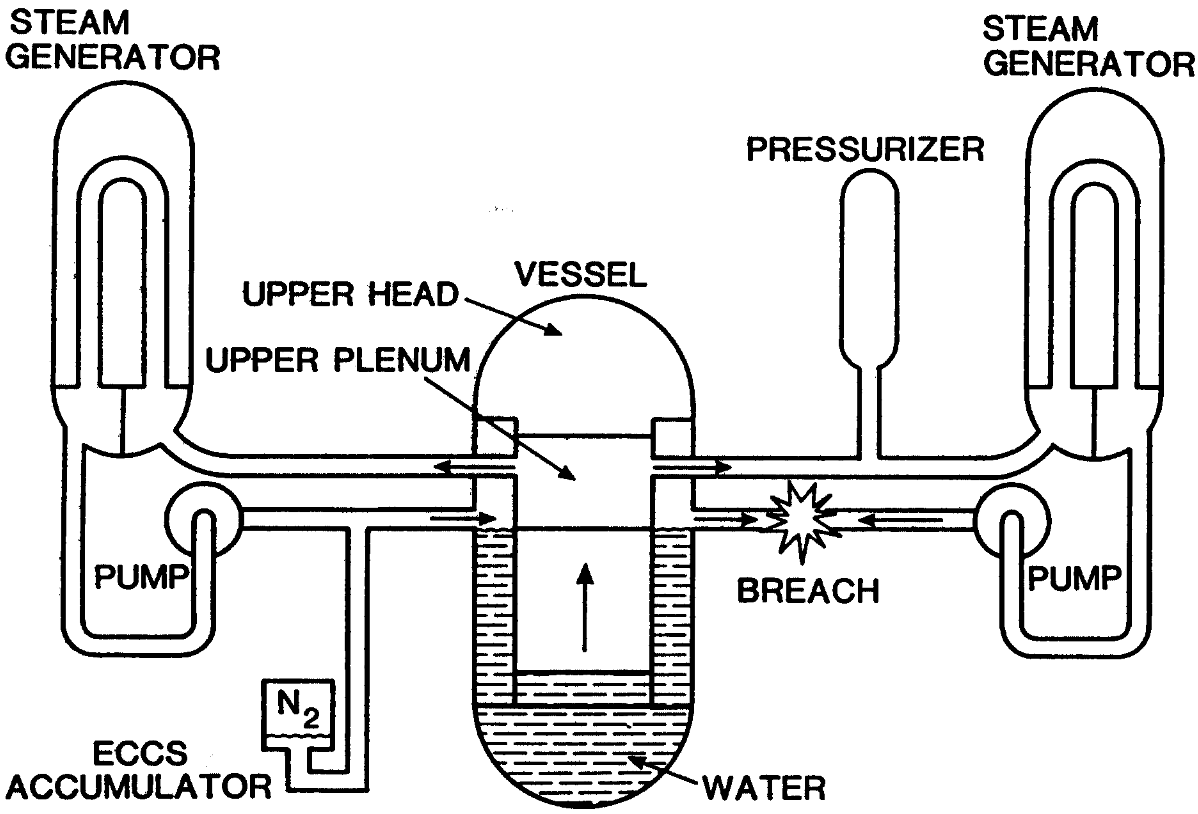


FIG. 2.1.3 FLOWS AND LEVELS IN THE PRIMARY SYSTEM AT THE END OF REFILL AND BEGINNING OF REFLOOD, PWR LARGE, COLD-LEG, BREACH

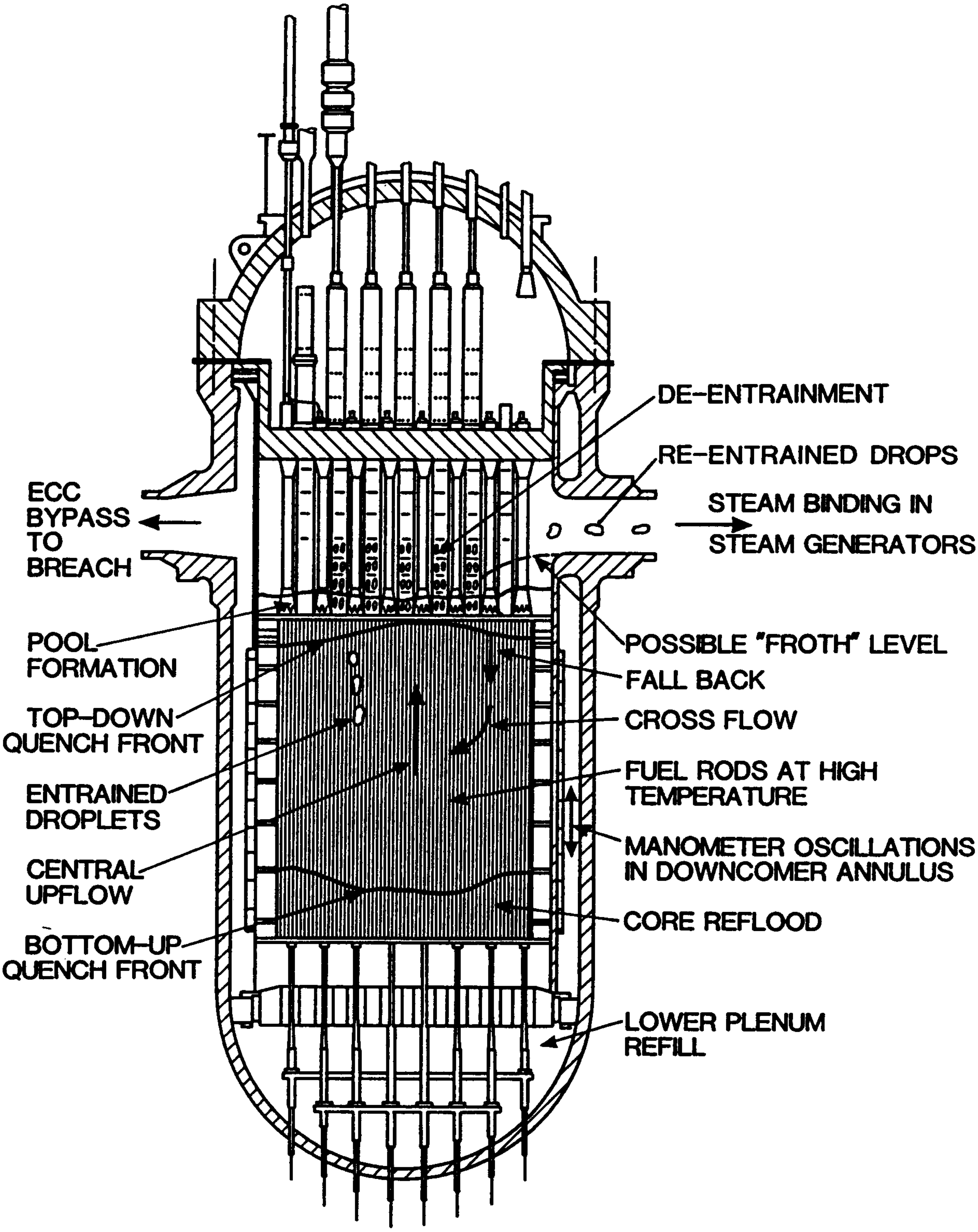


FIG. 2.1.4 REFILL/REFLOOD PHENOMENA, PWR LARGE, COLD-LEG, BREACH

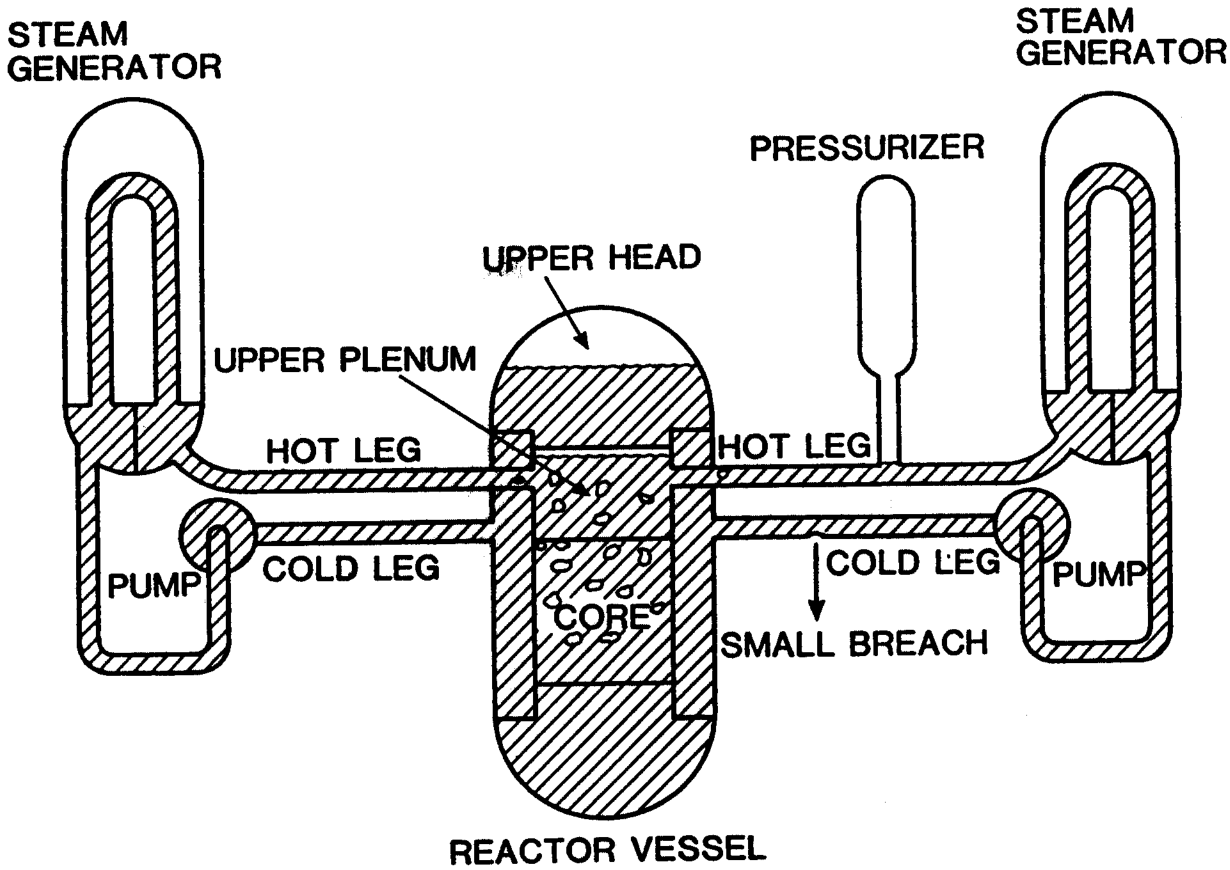


FIG. 2.1.5 INITIAL STAGE OF VOID FORMATION IN THE PWR CORE AND REACTOR VESSEL DURING A SMALL-BREACH LOCA

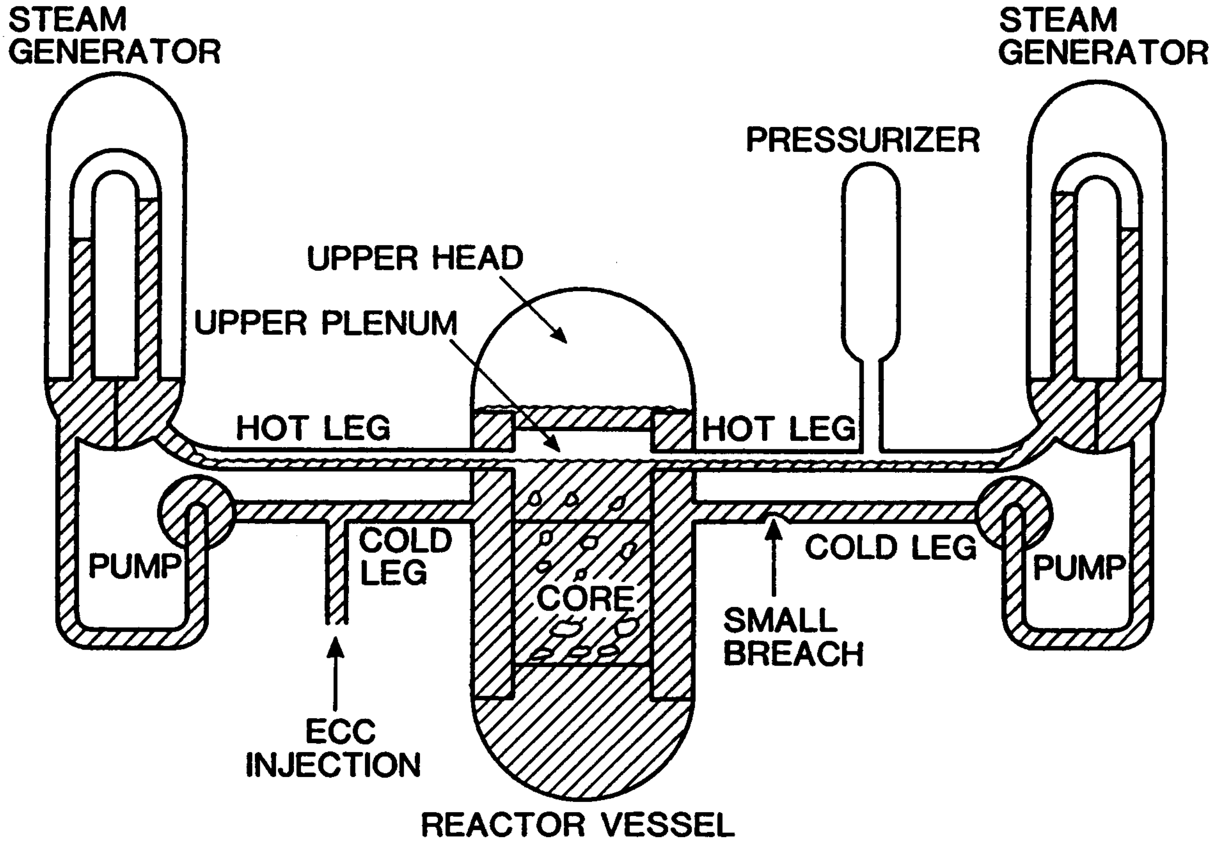


FIG. 2.1.6 FORMATION OF A CONTINUOUS VAPOR PHASE AT THE TOP OF THE STEAM GENERATOR U-TUBE AND VESSEL DURING A PWR SMALL-BREACH LOCA

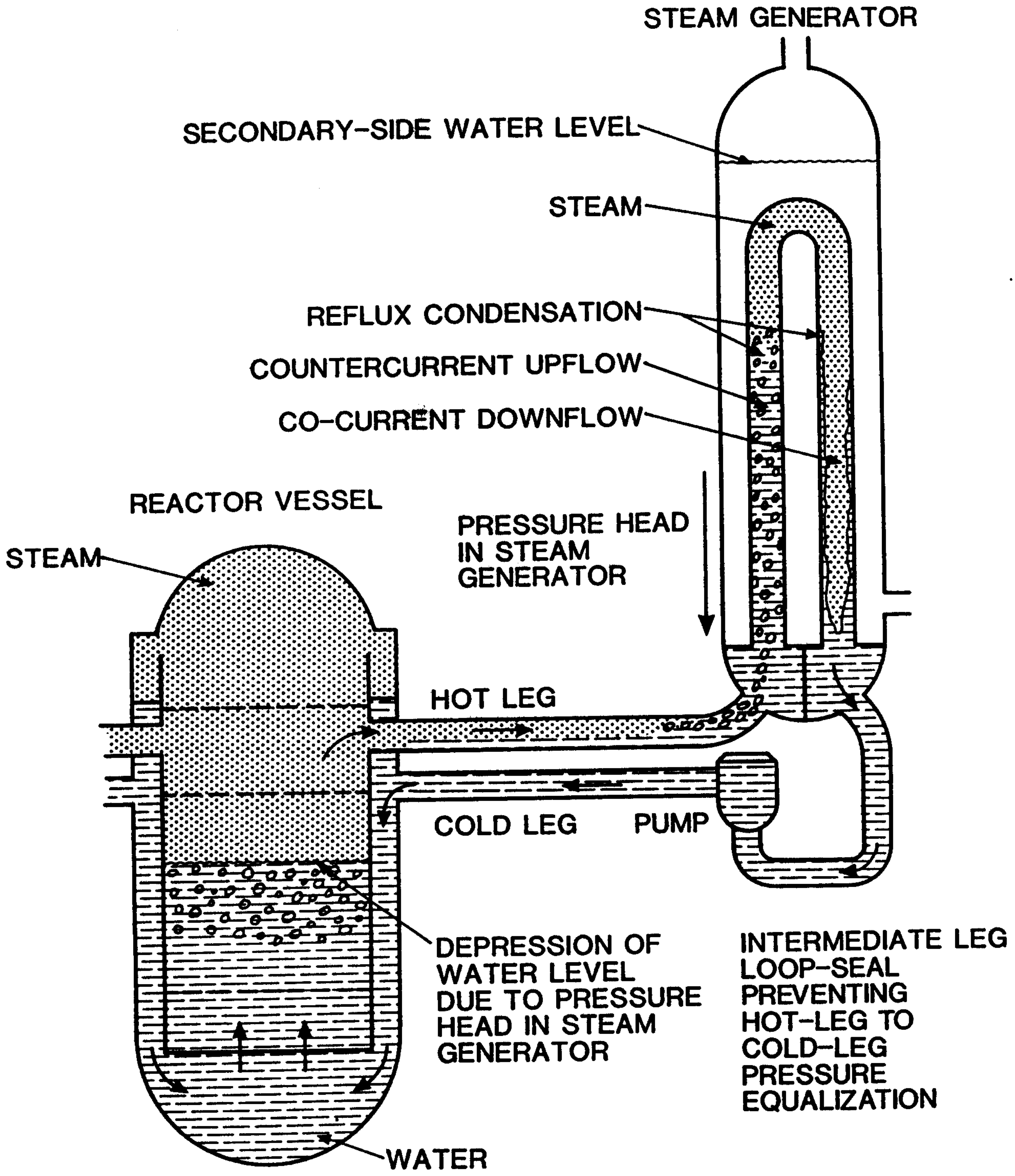


FIG. 2.1.7 DEPRESSION OF THE CORE WATER-LEVEL DUE TO THE LOOP-SEAL IN THE INTER MEDIATE LEG, PWR SMALL-BREACH LOCA

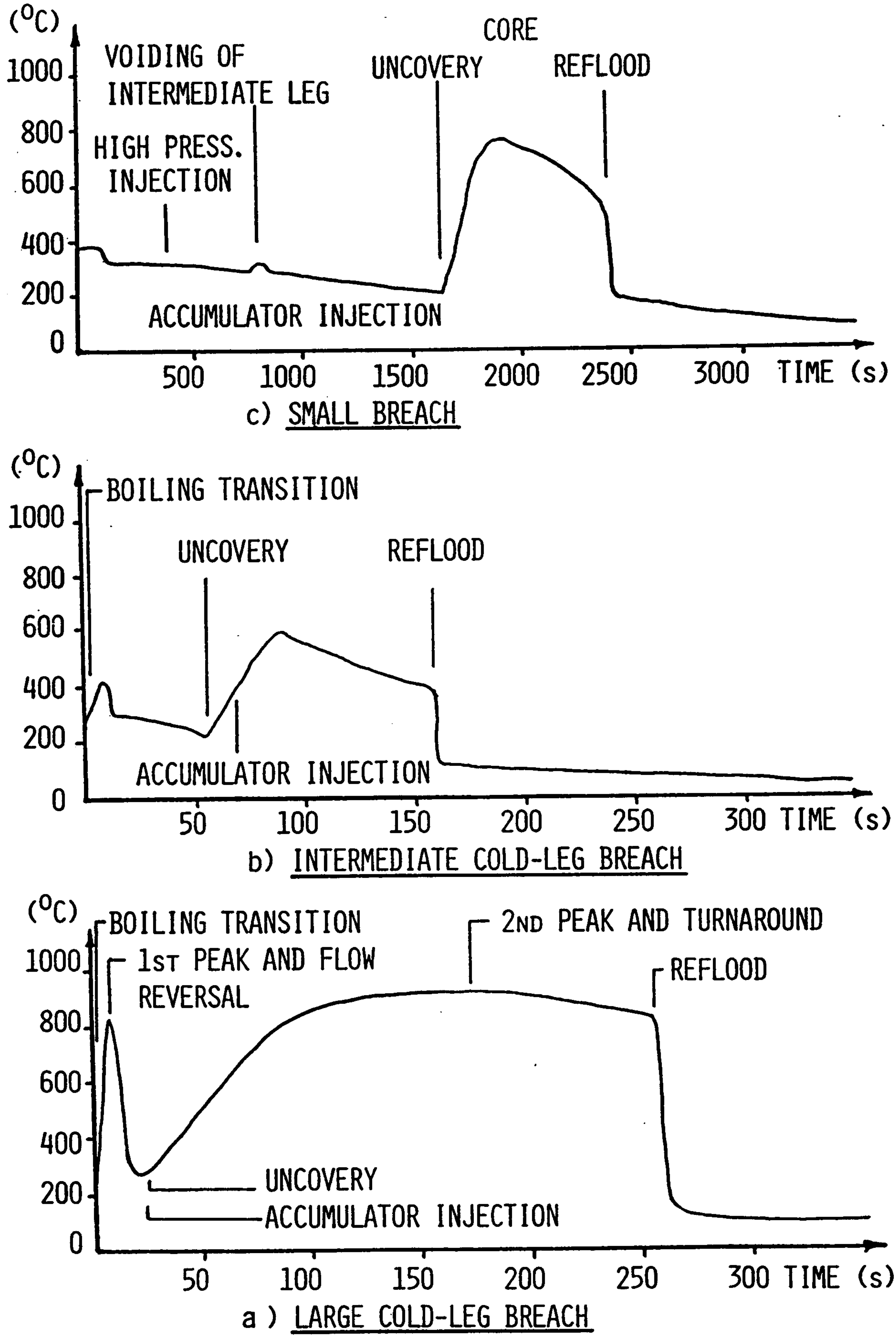


FIG.2.1.8 TYPICAL PEAK CLADDING TEMPERATURES CALCULATED FOR LOCA's IN PWR's

2.2 BWR-LOCA SCENARIOS

A BWR core sits low-down inside a very large pressure vessel. Under normal operating conditions, subcooled water at about 70 bar and 270 °C flows in the lower regions of the vessel and core, a two-phase saturated mixture flows in the upper regions of the core and the central regions of the vessel, and almost dry saturated steam at about 280 °C flows in the steam dome and out to the turbine. The coolant flow through the core, typically 1500 kg/m^2 s, is driven by recirculation pumps, which may be internal or external, the latter with or without jet-pumps. Fig. 1.3 illustrates the three BWR types and Fig. 2.2.1 shows typical details of a reactor vessel with jet-pumps.

Five points are worthy of note:

- the BWR vessel constantly loses coolant through its steamlines, the loss being balanced by feedwater. The turbine and feedwater control systems, and the main steam isolation system, can thus influence the realistic course of any LOCA. For design basis purposes it is usual to assume loss-of-auxiliary-power so that recirculation pumps are tripped, the vessel is isolated and normal feedwater becomes unavailable at the start of a LOCA. In reality, although recirculation pumps are anyway tripped automatically on low annulus water level signals, the main feedwater pumps remain operational during a LOCA for a period of time, which depends on breach size, feedwater system characteristics and operator procedures. For example, feedwater pumps are tripped from low feedwater/condensate tank level signals, main steam isolation will trip steam-driven feedwater pumps, or operators may be required by procedures to trip feedwater pumps to avoid high suppression pool water levels. Nevertheless, the extra time that feedwater pumps remain available may enable the operator to mitigate, or even to avoid, core uncovery for some LOCA situations.

- an inventory inbalance caused by a complete loss-of-feedwater, for example, is rapidly compensated by reactor scram, turbine trip and eventually main steamline isolation from the falling water level in the vessel. The subsequent release of steam from the isolated vessel through the relief valves may be looked upon as an intermittent small or intermediate breach LOCA. Under these conditions vessel inventory is readily maintained by the normally available high pressure injection systems such as emergency feedwater, RCIC, HPCS or HPCI (see Appendix A1.2). By analogy, a stuck open relief valve may also be looked upon as a small or intermediate breach LOCA in a steam line.

- the presence of saturated water and steam in the vessel implies that the global pressure response to any breach will be strongly damped by the expansion of the steam present in the vessel dome and the flashing of coolant in the vessel. Extremely rapid (0.1 s) decompression of subcooled regions occurs only locally in a BWR and does not in itself influence core cooling.

- from a global point of view, core uncovery and core cooling depend primarily on total vessel water inventory and its two-phase swollen level. Collapsed water level is a measure of actual vessel inventory, whereas the two-phase swollen level is more appropriate for core cooling. The draining of individual fuel bundles, and the finer details of bundle water levels and bundle cooling during a LOCA, depend on the complex, three-dimensional, distribution of coolant in the core and in the local regions surrounding the core.

- control-rod scram, to shut-down the nuclear fission process, is essential during a BWR-LOCA. Although coolant void generation and reduced moderation will do this without control-rod insertion, a return to criticality would occur on reflooding the core with unborated emergency core coolant.

Since a BWR core remains essentially cooled by forced or natural convection boiling until the core is uncovered, it is appropriate to take core uncovery as the central theme of BWR-LOCA phenomenology. For the purposes of classification, which forms the first part of this chapter, a simple vessel water inventory concept suffices to illustrate core uncovery scenarios. Transient boiling transition, which occurs during some large-breach, recirculation-line, LOCA's may be treated as a separate effect, as indicated in Table 2.2.1. It provides slightly different initial conditions for the potentially much more severe cladding temperature excursion during core uncovery. Clearly, if the uncovery is minimal, the boiling transition excursion may itself provide the highest LOCA cladding temperature. Details of transient boiling transition, coolant distribution, vessel local inventories and the injection of ECC are discussed in Section 2.2.2. Such details are important for any accurate assessment of core cooling but are not relevant to a global classification scheme.

2.2.1 Global Classification of BWR-LOCA in Terms of Breach Location and Size (Fig. 2.2.2)

Whether a BWR core uncovers or not during a LOCA depends not just on:

- breach size,

but also on:

- the type of recirculation system; internal pumps, or external pumps with or without jet-pumps,
- the breach location,
- available coolant make-up from feedwater - or other systems,
- available ECC and its mode of injection,
- configuration and operation of the automatic depressurization system (ADS),
- operator intervention.

Operators following procedures are expected to influence positively only those events, which are proceeding relatively slowly (10 minutes), such as small-breach LOCA's or somewhat larger breaches with normal feedwater maintaining vessel level.

LOCA's are usually classed in terms of large-, intermediate- or small-breaches of particular lines attached to a reactor pressure vessel. This is adequate, but a more general approach, applicable to all BWR types, is first,

- to class breaches in lines in terms of equivalent breach locations at the pressure vessel. These classes indicate the potential for core uncovery. Secondly,

- to use breach size and the performance of available coolant injection or spray systems as a gauge for whether core uncovery actually occurs and, if so, for how long.

2.2.1.1 Equivalent Breach Locations for Different BWR Types

The pressure vessel's global thermodynamic behaviour, and thereby its water inventory, is controlled not by the actual breach location on a particular line, which might be far removed from the vessel, but rather by the location of the nozzle on the vessel through which coolant is escaping to the breach. The location controls the upstream conditions for the critical flow at the breach and is defined as the equivalent breach location. The concept is schematically illustrated in Fig. 2.2.2, where four breach classes A, B, C and D are identified.

The table on Fig. 2.2.2 shows how the various breaches, which may be postulated for each reactor type, fall into one of the four classes A, B, C or D. These classes are identified by two features:

1) Equivalent breach location relative to the initial water level in the vessel

For any postulated breach size, the relative location of the equivalent breach at the vessel to the initial water level controls:

- the mass flow or inventory loss through the breach, i.e. whether it is steam, a two-phase mixture or water.
- the rate of depressurization, and thus the rate at which coolant can be injected.

Two extremes may be cited for illustrative purposes. A large breach in a steamline (class A) leads to rapid vessel depressurization without a massive loss of water inventory. Conversely, a large breach in the recirculation line of an external-pump reactor (class C) gives a massive water inventory loss until the equivalent breach level, in this case the suction line, is uncovered. Only then does rapid depressurization occur as steam escapes through the breach.

2) Equivalent breach location relative to the top and bottom of the core

As a general rule, and ignoring for the moment coolant injection, the vessel water level during a LOCA from breach class B, C or D first falls to the equivalent breach location where significant depressurization occurs. Its final level below the equivalent breach location is then controlled by the inventory loss during the subsequent depressurization. For class A breaches the level fall below the initial water level is just the inventory loss during depressurization.

The simple vessel inventory concept enables some general conclusions, which have been largely supported by calculations, to be drawn from Fig. 2.2.2:

Internal Recirculation Pump Reactors

With the exception of breaches in some small lines, equivalent breach locations are of class A or B, i.e. above the core, for all postulated pipe breaches. Thus, for any breach size the potential for core uncovery is small for this type of reactor. Unless compensated by injected coolant the core will simply tend to boil dry, a relatively straightforward, one-dimensional, phenomenological process.

Vessel leaks or breaks in very small lines at or below core height have a potential for complete core uncovery. The ADS and coolant injection capacities are usually sized to handle low probability vessel leaks and breaks in small lines without core uncovery.

External Pump Reactors with Jet-Pumps

Large breaches in the recirculation suction line (class C) have a great potential for core uncovery but, since the equivalent breach location is at core height, the core may not be fully uncovered and may not drain completely. Partial core uncovery will involve the complex distribution of coolant in the volumes surrounding the core. This distribution is strongly influenced by flashing in the lower plenum and guide-tubes, by the rates of coolant injection and where in the vessel this injection occurs. Spray cooling can play a very significant role in those situations where complete core uncovery and drain occurs. Whatever the class A, B or C or the breach size, the core can eventually be covered or reflooded at the least to the height of the jet-pump inlets.

The ECC phenomenology involved with jet-pump reactors is thus very complex, and requires very sophisticated models for its accurate assessment.

Vessel leaks or small line breaks below core height (class D) have a potential for complete core uncovery and are handled in a manner similar to internal recirculation pump reactors.

If the jet-pumps or their flanges did not remain intact during a class C breach, the height to which the core could be recovered would be reduced, i.e. the LOCA behaviour would tend to be that for breach class D. Such a loss of integrity is highly unlikely.

External Pump Reactors without Jet-Pumps

A large breach in the recirculation suction line (class D) has the potential to drain the core rapidly and completely, and also to prevent its recovery by flooding from below. Spray cooling of the drained core becomes an essential ECC feature of this type of reactor. Since partial core uncovery can also occur for smaller breaches, the ECC phenomenology and assessment will be similar to that in the jet-pump reactors but with most emphasis on spray cooling.

Clearly vessel leaks are covered by the spectrum of breach sizes usually postulated for the recirculation suction line in this type of reactor.

2.2.1.2 Global Classification of BWR Breach Size

A classification of breach size can now be made in terms of equivalent breach locations A, B, C and D, without reference to a particular BWR reactor type. The approach is illustrated in Fig. 2.2.3. Typical vessel global water levels and pressures are shown for each breach class and for three breach sizes: large, intermediate and small. Typical maximum injection pressures for various make-up and ECC systems are also indicated. A more complete description of these systems with their performance characteristics is provided in Appendix A1.2.

With reference to Fig. 2.2.3 and without taking credit for normal feedwater the characteristics and bounds of breach size classification may be summarized as follows:

BWR Large Breaches

The upper bound of a large breach is the double-ended break, or a split-breach having the same break area, in the largest line attached to the pressure vessel. Main steam, feedwater and recirculation lines fall into this category. Unless steam is vented through the breach, a rapid fall in water level occurs, accompanied by a slow drop in vessel pressure. Steam venting leads to a transient level swell as steam bubbles in the two-phase regions expand under the decreasing pressure. This is followed by a fall in level. Flashing of water in the downcomer annulus, in the lower-plenum and in the guide-tubes also leads to a brief rapid swelling of the vessel water level and to redistribution of water inside the vessel. Vessel water level swell can thereby influence the quality of the breach flow. Breach location, breach flow, recirculation system performance and flashing control core flow, and hence core-cooling, during the early stages of depressurization or blowdown.

Breach locations A, B, C and D, respectively, provide ever more severe core uncovery conditions because they tend to give a greater fall in water level, and hence core uncovery, whilst at the same time delaying depressurization and thus coolant injection.

Although the water level at which ADS is initiated is rapidly reached (for some plants, a high drywell pressure signal is also required) the actual opening of ADS relief valves is usually delayed automatically for up to 100 s. ADS thus has no or very little influence on the large breach LOCA behaviour.

Global core uncovery trends can be estimated for large-breach LOCA's by taking the information on Fig. 2.2.3 and using the performance of typical low-pressure and high-pressure ECCS pumps to estimate core-recovery times. This is indicated in Fig. 2.2.4. Ignoring those spray cooling situations for breach locations D, when core recovery does not occur, it is seen that for large breaches the core can be uncovered for ca. 50 to 100 s depending on ECC availability and breach location. These uncovery periods are rapidly reduced for breach locations above the core. Note that actual core draining and cooling may not correspond exactly to these global estimates, because of the complicated processes occurring in the core itself, discussed in Section 2.2.2.

BWR Intermediate Breaches

As breach size is reduced vessel water level and pressure fall less rapidly, level swell from steam venting and lower plenum flashing become less violent. Core flow and core cooling during the early stages now tend to depend mainly on recirculation system performance. The global behaviour is still very similar to that occurring with large breaches, but stretched out in time, as shown in Fig. 2.2.3.

Higher vessel pressures tend to delay further coolant injection so that although the core uncovery is stretched out in time, so is global core recovery, as indicated in Fig. 2.2.4. Clearly, for some breach locations and depending on assumed ECCS performance, core uncovery can be longer for intermediate than for large breaches. Depressurization through ADS may also be essential to avoid very long periods of core uncovery. These arguments indicate that the most severe cladding temperature excursions may not always occur for the largest breach size in a class.

The bound between intermediate and large breaches is not, and indeed need not be, well-defined. One conventional definition is the breach-size for which no transient boiling transition precedes core uncovery, but this is not appropriate for a plant with internal recirculation pumps, which does not experience transition over the complete break spectrum. Another definition is just 50 % of the largest breach area, which is not appropriate for a plant with external recirculation pumps.

BWR Small Breaches

Small breaches are characterized, Fig. 2.2.3, by a slow fall in vessel water level with very little or no depressurization. Core flow during the first period is dominated by the recirculation system's performance. If the vessel is isolated the pressure rises and is limited by safety/relief valves. The pressure history oscillates between the opening and closing set-points of the valves with the lowest set-points. If high pressure injection systems are not available, ADS or operator action to reduce pressure is then essential if complete core uncovery is to be avoided.

For all breach classes the fluid level first falls to the vessel level slightly above the core at which ADS is initiated. ADS is initiated automatically after a time delay of typically 100 s. Vessel level after an initial swell then falls to the level dictated by the inventory loss during ADS blowdown. Thus, from a global core uncovery viewpoint, small-breaches through ADS behave very much like intermediate-or large-breach LOCA's for class B breaches. Accordingly, core uncovery, if it occurs, will be for very short periods, as indicated in Fig. 2.2.4. The corresponding core temperature excursion will be less severe than for an actual class B breach with the same ADS breach area, since core uncovery for the small breach occurs much later in time.

The bound between small breaches and intermediate/large breaches may be defined as the breach size, which is large enough to avoid pressure limiting actuation of vessel relief-valves. Since such actuation has no direct influence on core uncovery, it hardly appears necessary to distinguish between small and intermediate breaches.

The lower bound for a small breach is governed by:

- its detection through a measureable leak, so that reactor scram and start-up of ECCS proceed automatically, and

- the capability of high pressure injection or spray systems to prevent an automatic depressurization by maintaining vessel level above the ADS actuation level.

Since the fall in water level is gradual, the threat of core uncovery is not urgent. Such a situation is easily recognised by the operator through high drywell pressure and high drywell activity, and falling vessel water level. If the operator cannot maintain level with available high-pressure injection, he is required by procedures to depressurize the reactor manually. He can do this using main or auxiliary condensors, if available, or by opening relief valves. Lowering vessel pressure before ADS occurs leads to an early coolant injection and can avoid core uncovery altogether. Put simply, the operator action tends to change the small leak from a class B to a class A breach. As Fig. 2.2.4 implies, this is beneficial from a core uncovery viewpoint.

2.2.1.3 Phenomenological Windows for BWR-LOCA: Blowdown, Refill, Reflood, Spray Cooling

The classical division of the LOCA behaviour into more or less separate phenomenological windows; blowdown, refill and reflood, arose as a result of the simplistic, but conservative, approach to PWR- and BWR-LOCA analyses then in vogue. Separate codes for each window, each with conservative but sometimes inconsistent assumptions and boundary conditions, were widely employed with the intent of conservatively predicting the core uncovery, recovery and/or cooling processes in order to maximise predicted cladding temperatures.

A glance at Figs. 2.2.3 and 2.2.4 indicates that realistically the classical division is an oversimplification in the sense that a large overlap in these windows is apparent. ECC injection, particularly with the high pressure systems, occurs well before the vessel is depressurized. For some equivalent breach locations and sizes the global core uncovery and then recovery processes are also complete before the vessel is fully depressurized. For others the vessel drains and the core is cooled only by sprays, that is, no refilling of the lower-plenum or reflooding of the core actually takes place. An added complication is the hold-up of water in the core by counter-current flow effects at the core inlet, even when a water level may be identified in the lower plenum. This hold-up, discussed in Section 2.2.2, can prevent complete draining of the core, which means that the refill and reflood windows in this situation overlap completely.

That blowdown, refill, reflood and spray cooling are not separable in time makes the identification and description of phenomena and processes occurring somewhat difficult. From a modelling point of view such separation is not of great significance, since nearly all the modern codes treat the LOCA history in a unified manner.

A more rational division of BWR-LOCA phenomenology is to consider just the two conditions which can occur namely:

i) the core remains covered throughout the LOCA
ii) the core experiences partial or full-uncovery.

Within these two scenarios two windows may be identified; blowdown and coolant injection. In condition i), where the core remains covered, blowdown and coolant injection may be combined. For condition ii) they can be separated. Then, within the coolant injection window, the complex distribution of coolant within and around the core may be identified by the terms refill, reflood and spray cooling. This division of BWR phenomenology is employed in the tables of Chapter 4.

2.2.1.4 Summary Remarks on BWR-LOCA Global Classification

Before proceeding to the details of core uncovery phenomena, it is well worthwhile to summarize the preceding discussions on BWR-LOCA global classification:

- Four equivalent breach location classes A, B, C and D have been identified, in terms of the location on the vessel through which coolant is escaping, relative to the core and to the initial vessel water level. These classes cover all possible pipe breaches and vessel leaks postulated to occur in the three BWR reactor types.
- Large, intermediate and small breaches differ somewhat in terms of the time-scale of events and magnitude of phenomena but, basically, the phenomenology involved is very similar. There is thus no real necessity to distinguish between breach sizes.
- The classical phenomenological windows; blowdown, refill and reflood are not separable in time and do not adequately represent the divisions in BWR-LOCA phenomenology.
- For all breach classes and sizes a more appropriate division is simply "core not-uncovered" and "core uncovered". Blowdown and coolant injection may be identified as pertinent phenomenological windows. Refill, reflood and spray-cooling may be treated as phases during the coolant injection window of a core uncovery scenario.

2.2.2 Dominant BWR-LOCA Phenomena

The global vessel inventory and water level considerations discussed in Section 2.2.1 do not take into account the complexity of the internal regions and the coolant flow paths in the core and vessel. These are illustrated for a typical jet-pump reactor in Fig. 2.2.1. The vessel blowdown, which typically takes 100 to 300 s, may be divided into three periods:

1. an extremely short, tens of milliseconds, decompression of water in subcooled regions. Decompression waves traverse vessel flow paths producing transient pressure differences across internal structures. Loads induced by these waves are evaluated to ensure structural integrity is maintained.

2. an early dynamic period in which transient boiling transition, core uncovery and a first cladding temperature excursion may occur. Whether these actually occur depends on breach class. If they do, the early temperature excursion is quenched by flashing in the lower portions of the fuel bundles, in the lower plenum and guide tubes or by core flow reversal. This dynamic period lasts for tens of seconds.

3. a quiescent period in which the liquid distributed in the vessel by flow to the breach and by lower-plenum and guide-tube flashing, redistributes itself by draining down through the core to the lower plenum. Depending on breach class, this may lead to a second core uncovery with a second cladding temperature excursion lasting up to several hundred seconds.

The quiescent period overlaps with coolant injection which, because this occurs at different localities on a vessel, leads to different vessel regions filling at different rates. Core cooling depends on the multiplicity of local phenomena experienced by individual bundles throughout these periods. One fundamentally important phenomenon is counter-current flow limiting. This not only holds-up water above the core but, more importantly, also supports water in the fuel channels to cool the core when, on a global vessel inventory basis, the core has apparently drained.

2.2.2.1 BWR-LOCA Blowdown

2.2.2.1.1 Core Flow Reduction and Transient Boiling Transition

Two processes, recirculation pump-trip and breach-flow, perturb or interfere with the normal coolant circulation in the vessel. Core flow is rapidly reduced by pump trip and coastdown. This is assumed to occur at the start of a LOCA. External motor driven pumps coast down in about 30 s, internal pumps with their lower inertia coast down in ca. 10 s. This reduction in core flow by itself causes no transient boiling transition, thereby posing no threat to core cooling, because it is a postulated operational occurrence. A boiling transition occurs within the first few seconds, when this coastdown is augmented by a large core-inlet flow diversion to a breach. Void reactivity feedback and reactor scram time can also influence the time to boiling transition.

For breach class C LOCA scenarios, core inlet flow is rapidly reduced as liquid in the lower plenum is diverted to the breach. For plants with external recirculation loops this reduction is initially compensated to some extent by flow from the intact loop into the lower plenum, until this loop becomes ineffective. Core inlet flow then approaches zero and may even reverse for a short period. Core flow reversal occurs very quickly (seconds) in large-breach class D scenarios. Both of these breach classes are thus characterized by a rapid boiling transition and a sharp initial increase in cladding temperature. Core uncovery follows soon after as liquid levels are driven down to the breach location by the expanding steam cushion in the top portion of the vessel.

Transient boiling transition does not occur during breach class A (steamline) or B (e.g. feedwater line) LOCA scenarios. This is because the breach flow tends to augment the core flow, as voids in the saturated regions expand, and flashing of liquid in the lower plenum drive a two-phase mixture through the core.

2.2.2.1.2 Void Expansion and Coolant Flashing

As soon as steam is free to escape through a breach, rapid vessel depressurization ensues. Expansion of voids and the flashing of coolant in saturated regions occur, followed by the flashing of liquid in originally subcooled regions. Flashing in the lower plenum drives a two-phase mixture through the core, either re-establishing a two-phase core flow or augmenting the existing flow. Flashing in the guide-tubes drives a two-phase mixture up through the core-bypass to the upper-plenum. These flows last for a few seconds, then decrease as the rate of steam production falls.

Lower plenum flashing has a very significant influence on mitigating the first cladding temperature excursion, which results from the transient boiling transition and early core uncovery during a breach class C LOCA. The cladding is quenched before significant temperatures are reached, see Table 2.2.1. At the same time stored energy is removed so that excursions from subsequent core uncovery can be less severe.

Lower plenum flashing has little influence on breach class D LOCA scenarios whereas guide-tube flashing does. The early transient boiling transition is quenched by the almost immediate reversal of core flow. Core uncovery and complete drain follow directly. Lower plenum flashing occurs when the liquid becomes saturated. This normally occurs when its level falls to the breach level in the lower plenum itself, but then most of the coolant in the lower plenum flashes directly through the breach. Coolant in the guide-tubes, however, flashes through the bypass to the upper-plenum then down through the core to the breach, contributing significantly to core cooling.

For breach classes A, B and C lower-plenum and guide-tube flashing redistributes liquid from the lower plenum to other parts of the vessel. This strongly influences subsequent core uncovery behaviour for these breach classes.

Flashing of pre-heated feedwater, either in the feedwater lines or as it is injected into the vessel, can influence on the rate of vessel depressurization through small- and intermediate-size breaches.

2.2.2.1.3 Core Uncovery After Lower-Plenum and Guide-Tube Flashing

Towards the end of the blowdown period the rapid core-flow and flashing transients cease. Steam continues to be produced from the core decay heat and stored energy in the reactor internals. The amount of liquid distributed in the various vessel volumes depends on the mass lost through the breach, the flow dynamics and any coolant injected up to this time. The liquid thrown up redistributes itself by draining down under gravity. This draining is strongly influenced by the available flow paths, and by counter current flows of steam or two-phase mixtures. Three scenarios may be identified, with corresponding, approximate, time scales for any core uncovery given by Fig. 2.2.4:

1) The core remains covered with the swollen level inside the core shroud above the core channels

If very little water has been lost during the blowdown period, liquid drains rapidly down the downcomer annulus refilling the lower plenum. Boiling continues in the lower plenum and core, and a two-phase mixture extends to well above the core. Natural circulation flow essentially stops between the upper plenum and the downcomer annulus, which now have a liquid connection solely through the lower plenum. This is a "U-tube" or manometer configuration in which the water boiling away in the core is replaced by liquid from the downcomer. Some natural circulation flow is established up through the hottest bundle and down through the bypass or cooler peripheral bundles, but essentially the core is in a pool-boiling cooling mode. The situation is illustrated for a jet-pump plant in Fig. 2.2.5. As water boils away the swollen water level slowly falls.

This scenario is typical of most class A breaches and those class B breaches, such as feedwater line breaks, which are high in the vessel. Coolant injection can lead to rapid void collapse if steam is condensed and a brief period (seconds) of core uncovery for large breaches on some reactors. As a general rule, most class A breaches do not lead to a significant core uncovery and cladding temperature excursion.

2) The core is uncovered but not completely

As soon as the core is uncovered the cladding undergoes another temperature excursion, which is terminated by reflooding the core or by spray cooling. Two different scenarios must be examined:

a) No separate water level in the lower plenum

This is similar to the situation in 1) above, but more water has been lost through the breach. Sufficient liquid and a flow path are available in the outside leg of the "U-tube" to fill the lower plenum, but not enough liquid to recover the core. The situation is illustrated in Fig. 2.2.6. Each fuel channel is now separately in a pool-boiling mode.

Most class B breaches fall into this category with the depth of core uncovery depending on how low in the vessel the breach is. Thus, for example, breaks in those ECCS lines which penetrate the core shroud can all lead to core uncovery. ADS from a water level just above the core can also lead to a core uncovery. On the other hand, feedwater line breaks do not.

b) A separate water level in the lower plenum

If the lower plenum cannot be filled by liquid from the outer leg of the "U-tube" (the downcomer annulus) a separate water level can form in the lower plenum. This condition will arise if the annulus contains insufficient liquid after blowdown, or if no path is available for the liquid there to flow directly to the lower plenum, or if liquid there simply drains away through a breach.

Coolant inside the core shroud will try to drain down to the lower plenum through the multiple paths available. Fig. 2.2.7 shows how complex these drain paths can be. Water will be held-up in the upper plenum by the counter current flow of steam, which is either produced in the core or flowing through the core from the lower plenum. Coolant will also be held-up in the core channels by the counter current flow of steam at the channel entrance restrictions. A schematic of the situation is provided by Fig. 2.2.8 for a jet-pump plant. In reality the phenomenology will be more complicated than suggested by this figure. Counter current flows can break down locally, allowing channels to fill from the upper plenum or drain completely to the lower plenum. Multi-channel behaviour with three characteristic patterns, indicated in Fig. 2.2.9, emerge. These are discussed in more detail in Section 2.2.2.2.

How long portions of the core remain uncovered, whether some channels drain completely and how quickly each channel is reflooded depends on ECC injection, the complex flow interchanges between the upper and lower plenums, and the possible flow paths from the lower plenum to the downcomer annulus. It is stressed that the hold-up of liquid in the channels by counter current flow limiting (CCFL) at the core inlet plays a very important role in cooling the core under these conditions.

The complex scenario depicted above is typical for class C breaches. It may not occur at all for class A and B breaches, except for the CCFL effects at the top of the core, if the core is briefly uncovered.

3) The core drains completely

Unless the rate of coolant injection is large enough to compensate for breach flow, complete drainage of the core occurs for class D breaches. The liquid level is pushed down to below the core by the expanding steam cushion before steam can escape to the breach. With limited coolant injection, complete drainage can also occur for those class C breaches when the level in the lower plenum falls so low that steam is vented directly from the lower plenum to the downcomer annulus. This diversion of steam round the bottom of the "U-tube" reduces CCFL at the core inlet allowing liquid in the fuel channels to drain away. Core spray cooling plays a significant role in both these situations. Note that a fully drained core and a water level in the lower plenum corresponds to the classical LOCA, end-of-blowdown, situation.

A broad categorization of the above scenarios for early boiling transition, and core uncovery prior to and following lower plenum flashing, is provided by Table 2.2.2.

2.2.2.2 BWR-LOCA Coolant Injection

2.2.2.2.1 Refill, Draining, Core-Reflood

Injection of coolant influences BWR-LOCA phenomenology in several ways. It affects:

- the refilling of local vessel and core volumes,
- the draining of these volumes to the lower plenum or to a breach, either directly or through the core,
- the condensation of steam, steam generation and local subcooling in the various volumes, and thereby
- the counter current flow limiting (CCFL) at various points on the drainage paths between volumes, and
- core reflooding and spray cooling.

Whether high-pressure, or low-pressure injection systems are employed is not significant from a phenomenological standpoint. High-pressure systems by injecting early during the blowdown window may prevent core uncovery or may shorten the time for which a core is uncovered, c.f. Fig. 2.2.4. What is most important is where in the vessel coolant is injected:

Injection into the Downcomer Annulus

This injection, into the outside leg of the annulus/core-shroud "U-tube", is typical of feedwater, emergency feedwater, RCIC, HPCI and LPCI on some plants. If the core is covered at the time of injection, Fig. 2.2.5, and injection exceeds mass loss through the breach, the core remains covered. If the core uncovers or is already uncovered, as in Figs. 2.2.6 or 2.2.8, the filling of the annulus leads to coolant flowing from the annulus through the jet-pumps into the lower plenum and reflooding the core from below. Reflooding from below does not cause a void collapse in the two-phase mixture in the core.

The head of water in the annulus, the relatively large flow area between the annulus and lower plenum, and the multiple flow paths available, ensure that CCFL does not prevent draining of the annulus to the lower plenum. Thus, provided the loss of fluid from the annulus to a breach is much less than that injected, coolant injection into the annulus is a very rapid, efficient and uncomplicated means of cooling the core.

Injection into the annulus is very effective for class A and B breaches, it ensures in many cases that the core remains covered throughout. It can be completely ineffective for breach classes C and D, if all of the injected fluid is lost through the breach. For intermediate situations core uncovery time depends solely on the relative rates of coolant injection and coolant loss through a breach.

Injection into the Lower Plenum

On some jet-pump reactors the LPCI is connected to the external recirculation loop. Coolant injected into the loop, provided this loop remains intact, reaches the lower plenum directly through the jet-pumps. The phenomenology is identical to annulus injection, except that intact-loop injection can still be effective for a recirculation line breach (class C).

Injection Inside the Core Shroud

HPCS and LPCS systems inject highly-subcooled water through spray nozzles into the upper plenum above the core. If the swollen water level is below the nozzle sparger, as in Figs. 2.2.5 or 2.2.6, the injected fluid is more or less uniformally sprayed onto the pool of coolant in the upper plenum, respectively, the top of the core. If the water level is above the sparger, as in Fig. 2.2.8, the injected coolant is mixed with the coolant in the upper plenum. Subcooled water from LPCI is injected into the bypass region, which fills rapidly and uniformly, and overflows into the upper plenum.

At first sight, injection of water inside the core shroud appears to be a more direct means of cooling the core for all breach classes. Indeed, core cooling is provided even when measured water levels in the vessel's downcomer annulus indicate zero. In practice core cooling is effective, but many complex phenomena are involved in the process:

- Water drains from the bypass into the control-rod-drive guide tubes, and into the lower plenum provided this is not full. Although leakage paths from the guide-tubes to the lower plenum are small, this will help to refill the lower plenum. It will also help to reflood the core from below if the lower plenum is full and a head of water can be maintained in the annulus region, the outer leg of the "U-tube".

- Water drains from the bypass into the bottom of the fuel channels. This can fill the channels and reflood the core, or drain into the lower plenum, if this is not full.

- Water in the fuel channels drains into the lower plenum, again if this is not full.

- Water drains into the bypass and fuel channels from the upper plenum, either directly from the sprays or from the pool of water in the upper plenum above the core. Drainage into the fuel channels can help to fill and reflood the channels or quench the channels from above.

- Water in the peripheral regions of the upper plenum can drain directly through peripheral bundles to the lower plenum, provided this coolant is subcooled.

How much water builds up in the various regions and how much water drains to or from the various regions, indeed whether any drainage occurs at all, depends on local counter current flow limiting (CCFL) at the restrictions in the drainage paths.

2.2.2.2.2 CCFL, CCFL Breakdown, Multichannel Effects

Steam generated within the vessel as a result of depressurization, decay heat and stored energy will escape to the breach either directly or by flowing through the saturated water in the various vessel regions. In so doing, it will maintain a swollen water level in the two-phase regions, and may also prevent by CCFL draining of liquid through restrictions. If steam flows are very high as, for example, during the early period of depressurization, water in the vessel as well as injected coolant can be carried out through a breach. Note that the rate of steam production in any region, particularly in the lower plenum, depends very strongly on the distribution of subcooled water in that region. Generally, the steam is saturated but superheated steam can appear in the core.

CCFL occurs at many locations, but only three are dominant. Consider the most complex situation of an uncovered core and a definite level in the lower plenum, Fig. 2.2.8:

1) CCFL at the upper tie plate near the core exit limits or prevents draining of spray droplets or water from the upper plenum into the fuel channels.

2) CCFL at the bypass to guide tube (CRD) restriction limits or prevents draining of water from the bypass to the guide-tube and lower plenum.

3) CCFL at the channel entry orifice holds up the coolant in the channels, maintaining core cooling even when the lower plenum is not full. This is extremely beneficial to core cooling.

Because of the multiple paths available to vent steam, CCFL does not prevent drainage of liquid from the upper plenum into the bypass. Leakage from the bypass into the bottom of the fuel-channels, which is possible for some fuel channel designs, can also significantly influence the rate of filling of individual channels.

Breakdown of CCFL, which enables a relatively large volume of water to drain through a restriction, occurs when subcooled water reaches a CCFL restriction and condenses steam there.

HPCS and LPCS spray droplets are rapidly brought to saturation as they condense steam, so that CCFL maintains a level in the upper plenum until the spray nozzles are covered. Condensation efficiency then falls and subcooling builds up in the regions near the nozzles. In those plants with a ring-header spray-nozzle configuration, this subcooling leads to CCFL breakdown at and near the peripheral bundles. CCFL breakdown drains the upper plenum into the peripheral bundles, filling and quenching them, and re-exposing the spray nozzles. The process repeats itself in a cyclic manner with the water level in the upper plenum oscillating about the nozzle level. Plants with the spray nozzles distributed on a support structure above the upper tie-plate, experience more uniform mixing in the upper-plenum, which may delay CCFL breakdown above the low-power bundles.

If the water at the bottom of a channel is subcooled, CCFL breaks down at the channel inlet orifice and coolant drains rapidly to the lower plenum. This enables full peripheral bundles to drain to the lower plenum.

Subcooled leakage from the bypass to the bottom of a channel can also lead to the channel draining, if CCFL breaks down at the channel inlet. It can also lower briefly the swollen water level in a channel by condensing the steam voids there. In practice, CCFL breakdown at channel inlet orifices tends to be limited by a redistribution of steam flow between the lower plenum and core inlet in such a way as to maintain a uniform core pressure drop.

The global, almost one-dimensional, pattern of Fig. 2.2.8 is disturbed by multi-channel effects. The channels tend to fill slowly from the bypass leakage and the CCFL limited drainage from the upper plenum. Hotter bundles generate higher swollen water levels in their channels and, when these are full, they can no longer match the increasing core pressure drop imposed by the average or peripheral channels. The two-phase mixture is then expelled from a limited number of hot channels, which switch to a two-phase up-flow to maintain equal differential pressures. A multi-channel flow situation, as depicted in Fig. 2.2.9, is established:

- the peripheral channels operate in a downflow mode, draining liquid from the upper plenum to the lower plenum,
- most of the inner channels have a level still maintained by CCFL at the channel inlet. They fill slowly from drainage and leakage.
- a few hotter channels operate in a two-phase upflow, venting steam from the lower plenum. The steam entrains water droplets, which play a significant role in cooling the bundles in the hot channels.

As soon as the lower plenum is full, the channels fill and reflood the bundles at rates dictated by the remaining leakage paths and by the head of water maintained in the downcomer annulus. The core uncovery cladding temperature excursion is thereby ended.

If the lower plenum water level sinks so that steam can vent around the bottom of the "U-tube" to the downcomer annulus, CCFL above and below the core is strongly reduced. Under these circumstances the core and upper plenum can drain completely. Core spray distribution is then very important for core cooling with top-down quenching eventually ending the cladding temperature excursion. Whilst the core draining and the vessel depressurizing raises the plenum water level, entrainment of water with the venting steam tends to lower it, so that the level remains at or around the steam-venting location. This situation is characteristic of class D breaches and of those class C breaches, which have limited or delayed coolant injection. The hydraulic conditions in a fuel channel with spray cooling and a top-down quench of individual fuel rods is shown schematically in Fig. 2.2.10.

2.2.3 Concluding Remarks on BWR-LOCA Phenomenology and its Influence on Core Cooling

The phenomena having an important influence on the cladding temperature excursions during design-basis BWR-LOCA realistic scenarios have been identified in the previous sections, together with when and where they occur. Not all of them are important or even relevant in all situations, it depends on breach class:

- Breach classes A and B normally involve no early transient boiling transition and no or very little core uncovery. Dominant phenomena are those which control global vessel response, inventory and swollen water levels. Whether or not counter current flow limiting is considered under these circumstances is of no great importance. Typical, predicted, cladding temperature responses for these classes are shown in Fig. 2.2.11 a. All small breaches with ADS are classed as breach class B.

- Breach class C normally involves both early transient boiling and an early core uncovery. The core is almost immediately quenched by the core flow induced from lower plenum flashing so that the cladding temperature excursion is minor. A quasi-steady core uncovery follows. This means that all the phenomena discussed have to be considered for this breach class. The quench from lower plenum flashing restores the core to a cooled condition, so that the direct influence of the initial cladding temperature excursion on the one following the quench is small. However, the re-distribution of fluid in and around the core, as a result of lower plenum flashing, is very important for subsequent regional draining, CCFL effects and their influence on core cooling. Typical, predicted, cladding temperatures are shown in Fig. 2.2.11 b.

- In breach class D scenarios the core is drained after undergoing a rapid boiling transition and quench from core flow reversal. The initial temperature excursion is significant. Guide-tube flashing contributes to core cooling during the blowdown window. Spray cooling from above the core and top-down quenching terminates the second cladding temperature excursion, which is also significant, see Fig. 2.2.11 c. Since steam generated in the core and lower plenum is readily vented to the breach, CCFL has a small influence on the spray cooling process. Spray distribution is very important under these circumstances, because it controls the droplet flow which is cooling and quenching the individual rods, in the individual channels, as indicated in Fig. 2.2.10. Complete draining of the core can also occur for some breach class C scenarios with limited coolant injection.

The relative importance of the LOCA phenomena is discussed in Chapter 4.

Three-dimensional influences have to be considered in those vessel regions, which are not tightly bounded. Subcooling and spray distribution are three-dimensional in the upper plenum, as is subcooling in the lower plenum. Three-dimensional flows occur in the steam dome, influencing liquid carryover, and in the downcomer annulus. The parallel channel flows in the core and bypass may be considered as essentially one-dimensional. Some three-dimensional effects are important for rod-cooling in a bundle, for example local cooling under core-spray conditions and rod-quenching by flooding.

Thermal non-equilibrium between coolant phases influences condensation, flashing, boiling and steam superheating. These in turn influence swollen water levels, CCFL, entrainment and spray cooling.

Radiation heat transfer and exothermic chemical reactions need to be considered, if cladding temperatures exceed about 800 °C for any significant period of time. Such temperatures may occur for class C and D breaches, when only core sprays are available. Even under these conditions, fuel rod ballooning and bursting are either not significant or are localized and do not influence core cooling.

As far as analytical modelling is concerned, it is important to identify the dominant phenomena and to know when and where they occur. Models have to consider all relevant local regions and be able to analyse the history of each region in order to provide the necessary "when and where" conditions. Where a suitable modelling capability is lacking, experiments provide the required local or global correlations. These problems are considered further in Chapter 5.

REACTOR TYPE	BREACH LOCATION (CLASS) Fig. 2.2.2	MAXIMUM CLADDING TEMPERATURE (°C) DURING EARLY BOILING TRANSITION EXCURSION	TEMPERATURE EXCURSION TERMINATED BY (Note 2)
INTERNAL RECIRCULATION PUMPS	FEEDWATER (B)	NO BOILING TRANSITION	—
BWR-4 JET-PUMPS	RECIRC. LINE (C)	ca. 500 (Note 1)	LOWER PLENUM FLASHING
BWR-6 JET-PUMPS	RECIRC. LINE (C)	ca. 140 (Note 1)	LOWER PLENUM FLASHING
EXTERNAL RECIRCULATION, NO JET-PUMPS	RECIRC. LINE (D)	ca. 900	REVERSAL OF CORE FLOW AND GUIDE TUBE FLASHING

Notes: 1. This does depend on initial critical power ratio and the rate of pump-coastdown assumed, but an excursion would in any case be very mild.
2. In all cases cladding is quenched to local saturation temperature, see Fig. 2.2.11

TABLE 2.2.1 TYPICAL CLADDING TEMPERATURE EXCURSIONS FOLLOWING EARLY BOILING TRANSITION DURING LARGE BREACH LOCA IN A BWR

BREACH CLASS Fig. 2.2.2	EARLY BOILING TRANSITION	CORE UNCOVERY PRIOR TO LOWER PLENUM FLASHING	CORE UNCOVERY AFTER LOWER PLENUM FLASHING	COMMENTS
A	NO	NO	NO	Very brief periods of core uncovery from void collapse for large breaches on some reactors.
B	NO	NO	YES/NO	Class B breaches low down in the vessel, and ADS for small breach LOCA's, can give core uncovery.
C	YES	YES	YES	Typically two cladding temperature excursions.
D	YES	YES	YES	Typically two cladding temperature excursions. Vessel empties very rapidly.

TABLE 2.2.2 <u>BROAD CATEGORIZATION OF EARLY BOILING TRANSITION AND CORE UNCOVERY IN A BWR</u>

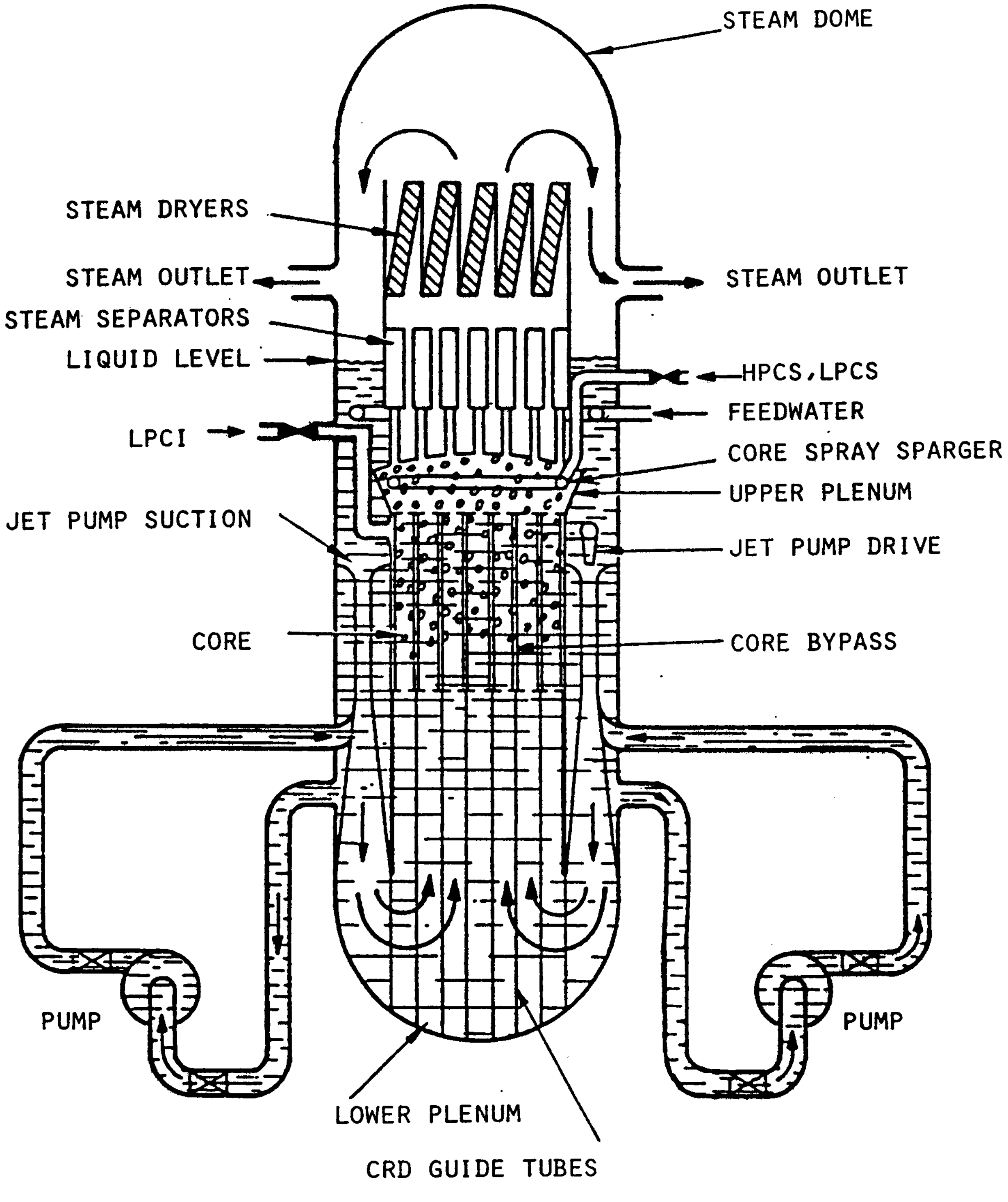
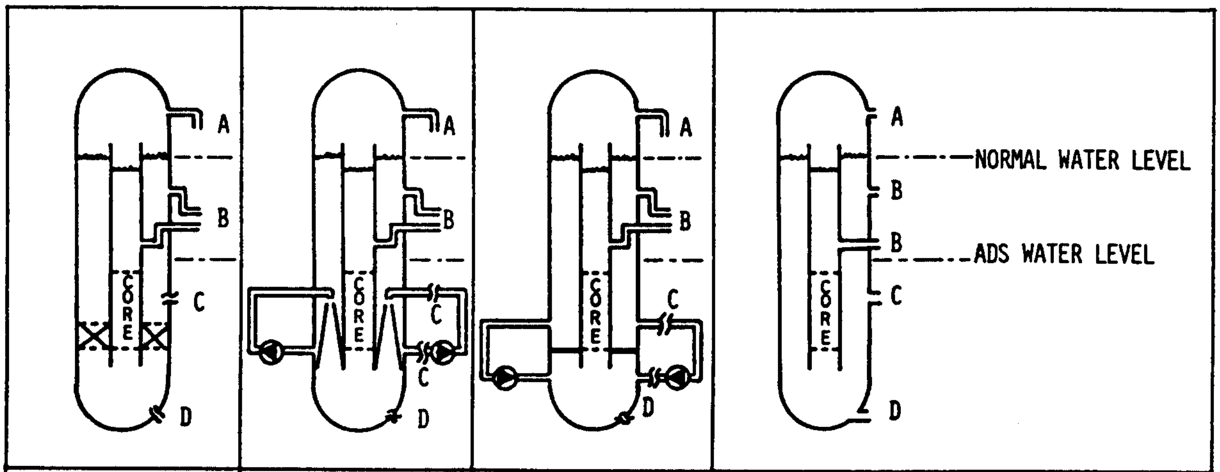


FIG.2.2.1 SCHEMATIC OF JET-PUMP BWR UNDER NORMAL OPERATING CONDITIONS



INTERNAL RECIRC. PUMP REACTOR	JET-PUMP REACTOR	EXTERNAL RECIRC. PUMP REACTOR	EQUIVALENT BREACH LOCATION	GENERAL CHARACTERISTICS
-STEAMLINE -RELIEF VALVE -RCIC STEAMLINE	-STEAMLINE -RELIEF LINE -RCIC STEAMLINE	-STEAMLINE -RELIEF LINE -AUX.COND.LINE	CLASS A	-ABOVE NORMAL COOLANT LEVEL -ABOVE CORE
-ECCS LINE -FEEDWATER LINE -RCIC LINE	-ECCS LINE -FEEDWATER LINE -RCIC LINE	-ECCS LINE -FEEDWATER LINE -RCIC LINE	CLASS B	-BELOW NORMAL COOLANT LEVEL -ABOVE CORE
-VESSEL BREACH ABOVE RECIRC. PUMPS	-RECIRC. LINE -VESSEL BREACH NEAR JET PUMPS	-RECIRC.SUCTION -VESSEL BREACH IN ANNULUS	CLASS C	-BELOW NORMAL COOLANT LEVEL -BETWEEN TOP AND BOTTOM OF CORE
-LARGE VESSEL BREACH IN LOWER PLENUM	-LARGE VESSEL BREACH IN LOWER PLENUM	-RECIRC.DISCHARGE -BREACH IN LOWER PLENUM	CLASS D	-BELOW NORMAL COOLANT LEVEL -BELOW CORE
-INSTRUMENT LINE OR LEAK IN LOWER PLEN.	-INSTRUMENT LINE OR LEAK IN LOWER PLEN.	-INSTRUMENT LINE OR LEAK IN LOWER PLEN.	NOMINALLY D BECOMES B AFTER ADS	

FIG.2.2.2 GLOBAL CHARACTERISTICS OF BREACH LOCATIONS IN BWR'S

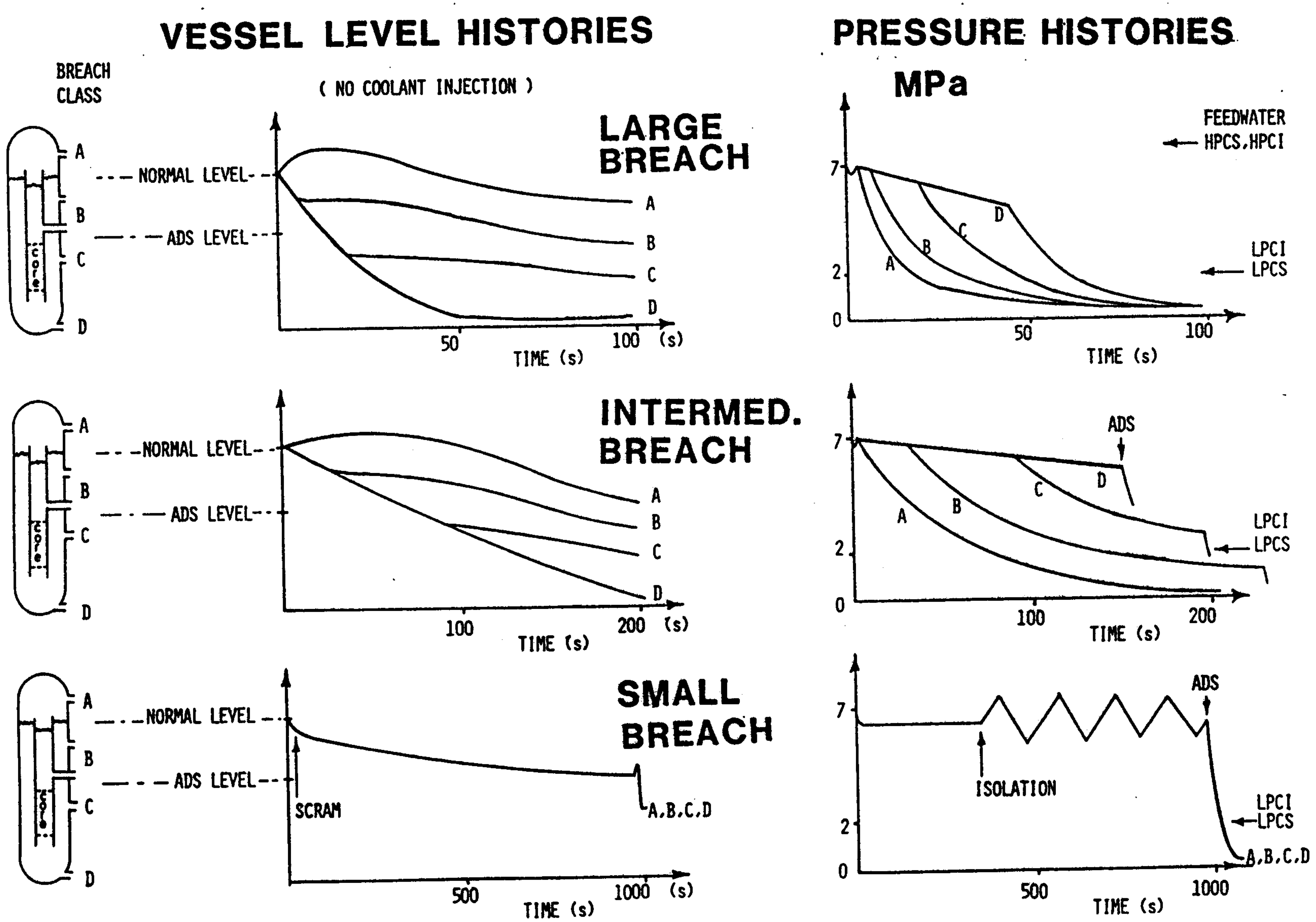


FIG.2.2.3 BWR GLOBAL VESSEL BEHAVIOUR FOR EQUIVALENT BREACH LOCATION (BREACH CLASS) AND TYPICAL BREACH SIZES

NOTE: - NORMAL FEEDWATER ASSUMED UNAVAILABLE

- EARLY BOILING TRANSITION, LOWER PLENUM FLASHING, RAPID CORE FLOW REVERSAL AND THE FILLING AND DRAINING OF LOCAL VESSEL REGIONS ARE NOT CONSIDERED HERE.

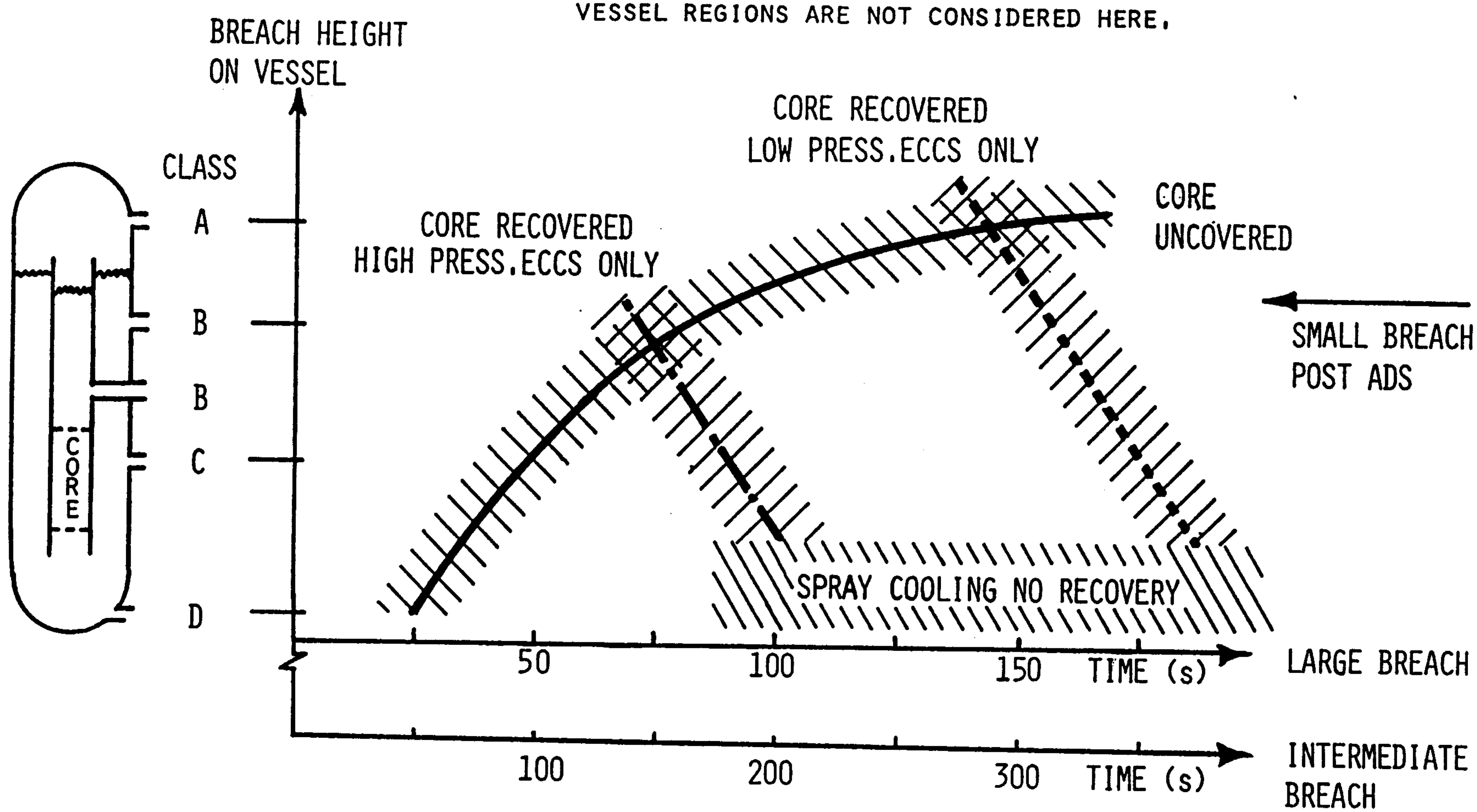


FIG. 2.2.4 BWR GLOBAL CORE UNCOVERY AND RECOVERY AS A FUNCTION OF BREACH CLASS AND BREACH SIZE

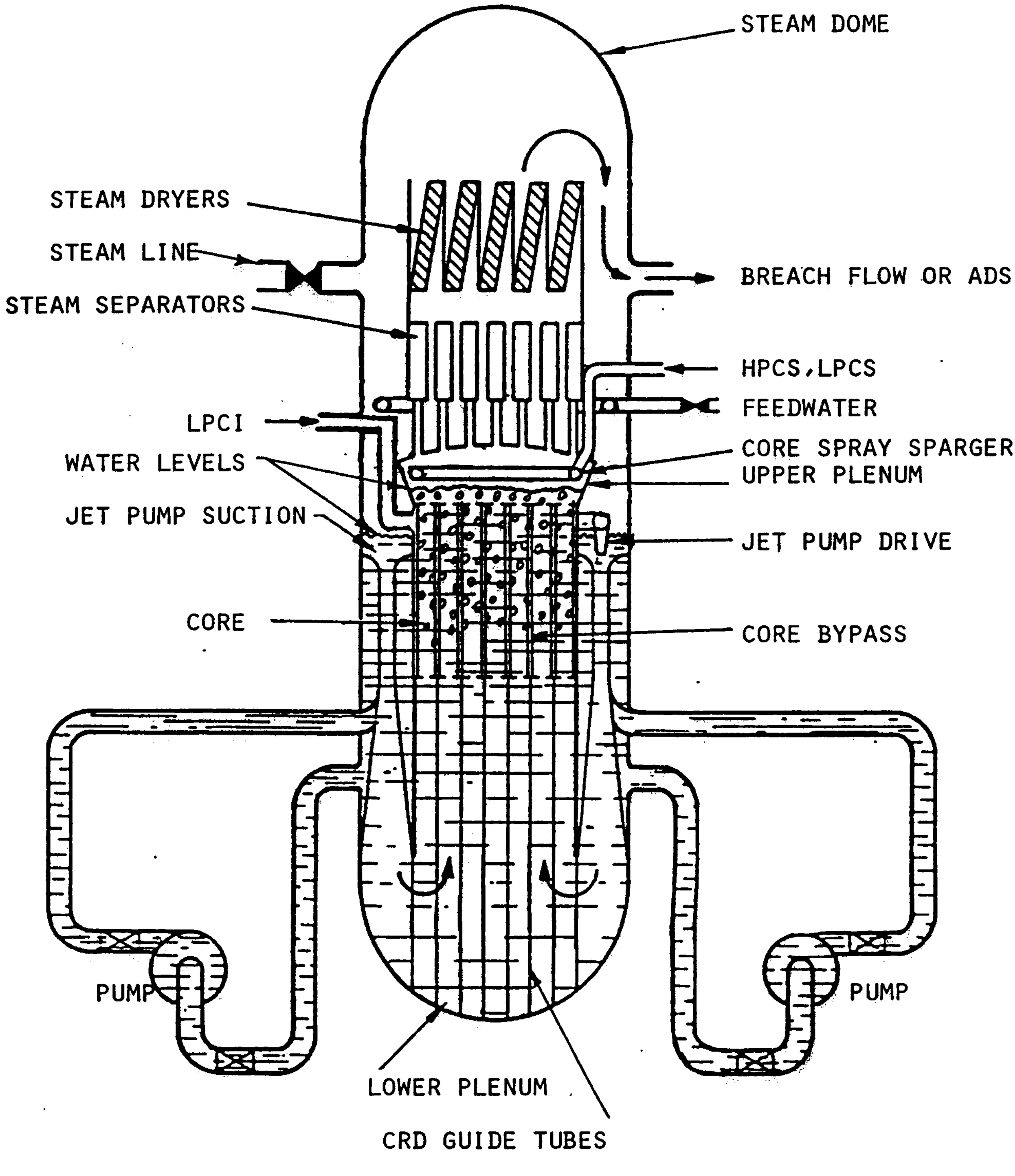


FIG.2.2.5 SCHEMATIC OF SWOLLEN WATER LEVELS FOR CLASS A AND B BREACHES NEAR THE END OF BLOWDOWN - CORE NOT UNCOVERED

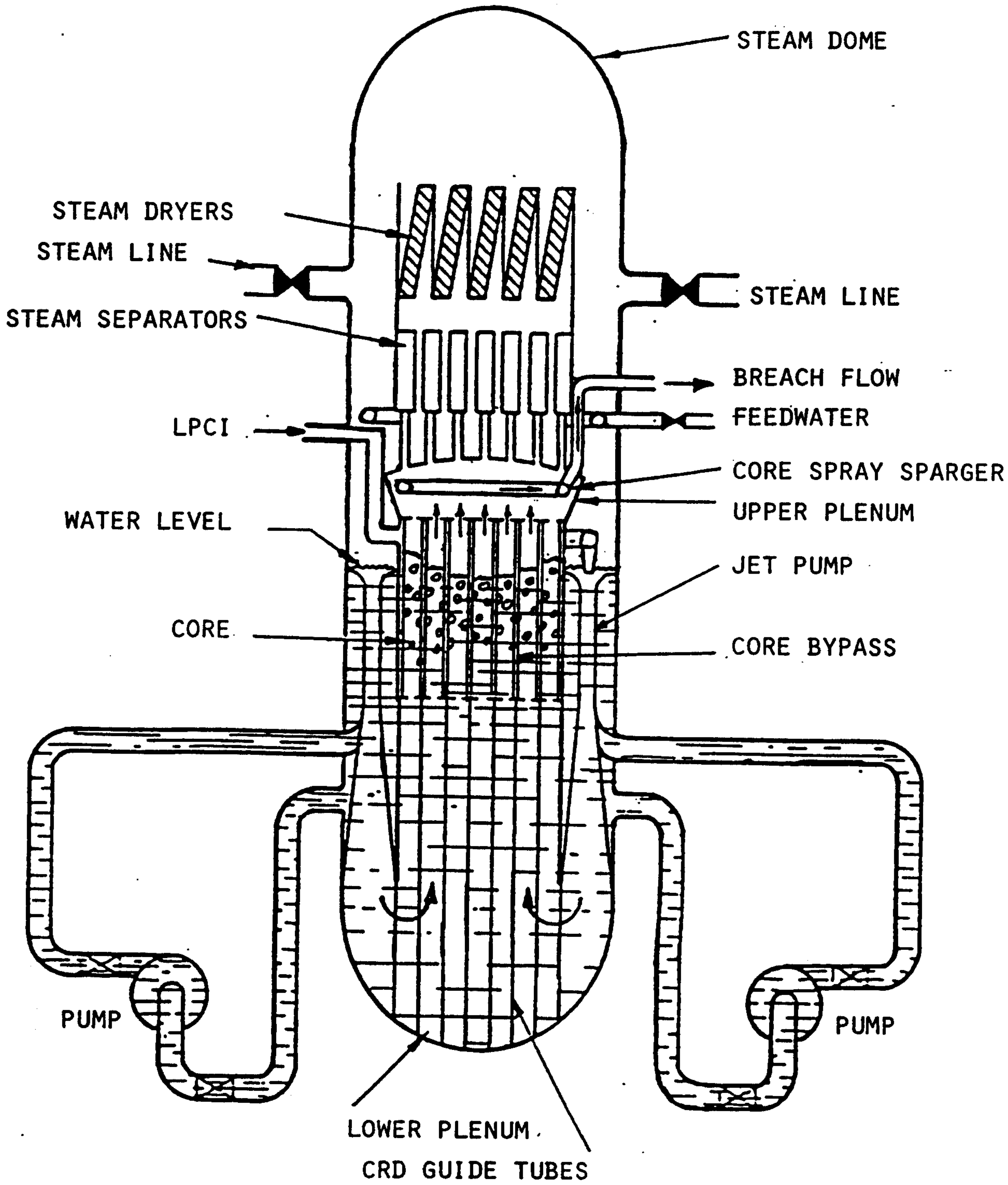


FIG.2.2.6 SCHEMATIC OF SWOLLEN WATER LEVELS FOR BWR CLASS B BREACH NEAR THE END OF BLOWDOWN - CORE UNCOVERED

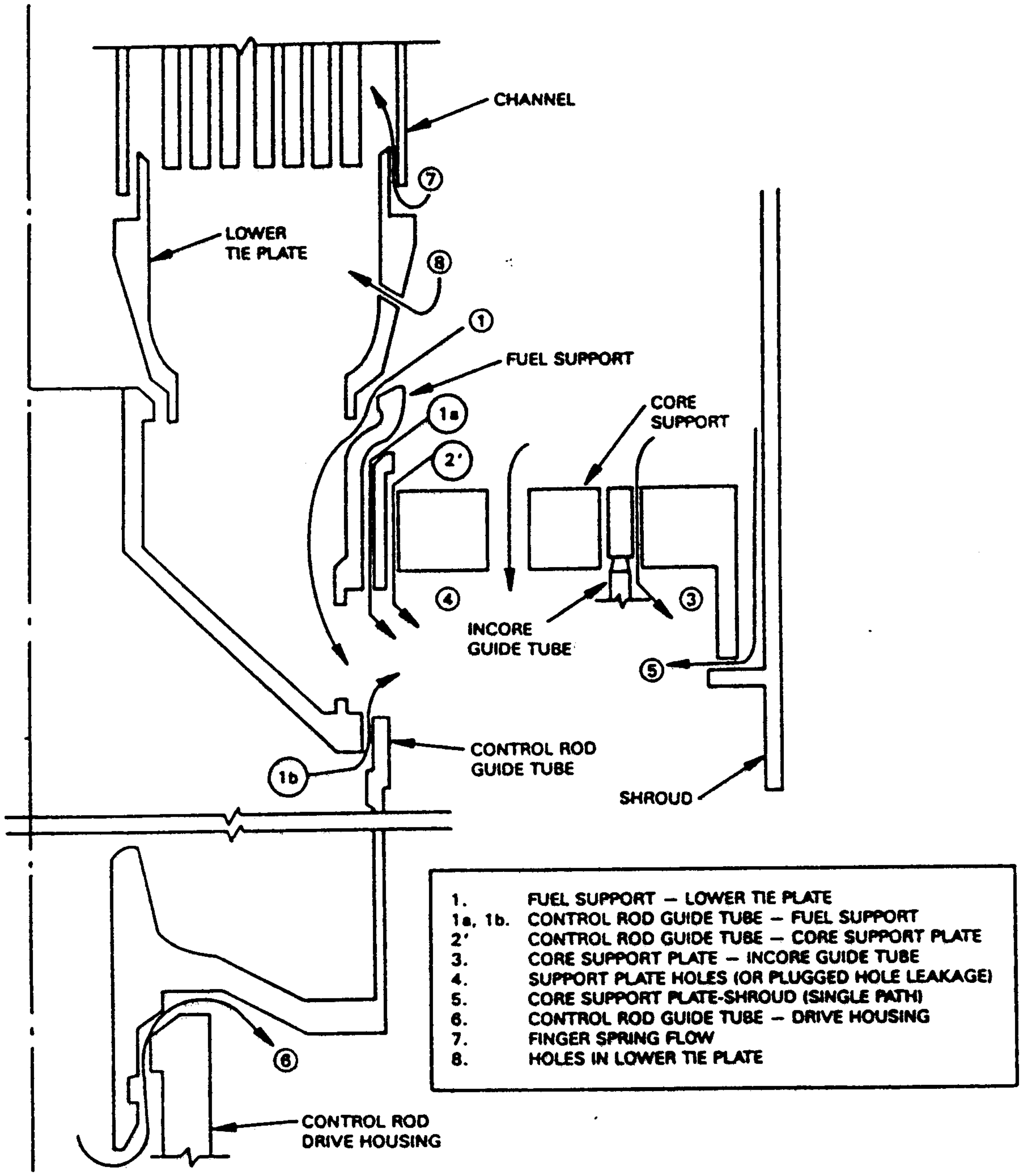


FIG. 2.2.7 TYPICAL LEAKAGE PATHS FROM THE CORE BYPASS AND FUEL CHANNELS TO THE CRD GUIDE TUBES AND LOWER PLENUM (BWR-6)

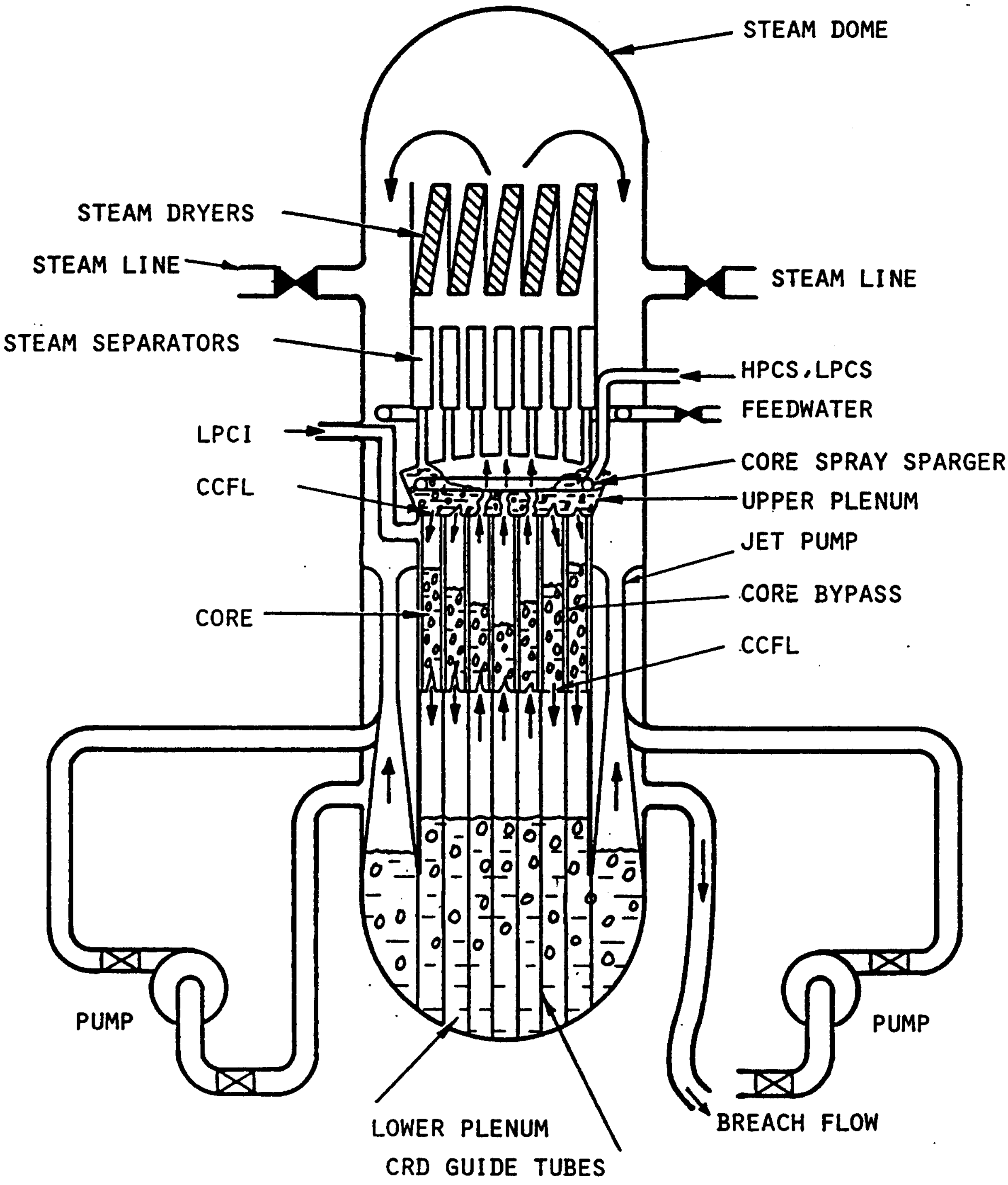


FIG.2.2.8 SCHEMATIC OF COOLANT DISTRIBUTION FOR CLASS C BREACH NEAR THE END OF BLOWDOWN - CORE UNCOVERED, LOWER PLENUM NOT FULL

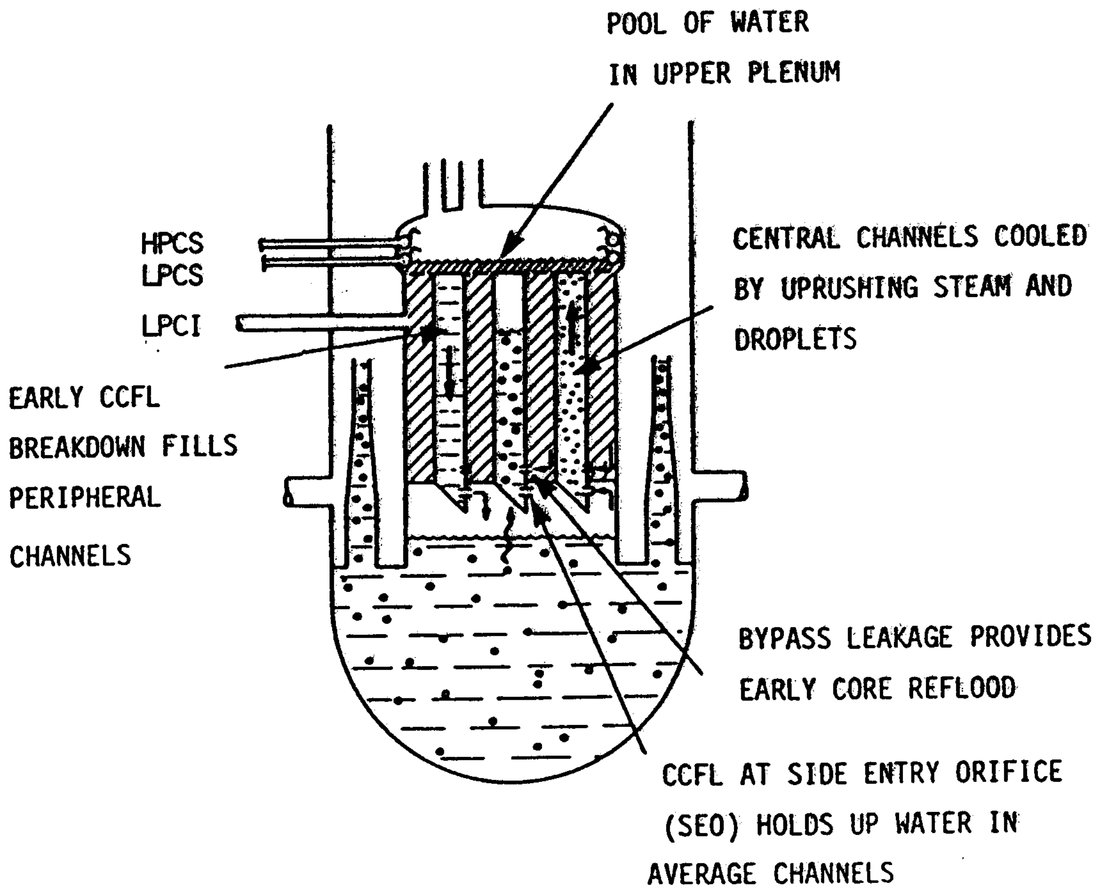


FIG.2.2.9 SCHEMATIC OF PARALLEL CHANNEL FLOWS IN A BWR CORE AFTER HOT CHANNELS FILL AND SWITCH TO A TWO-PHASE UPFLOW

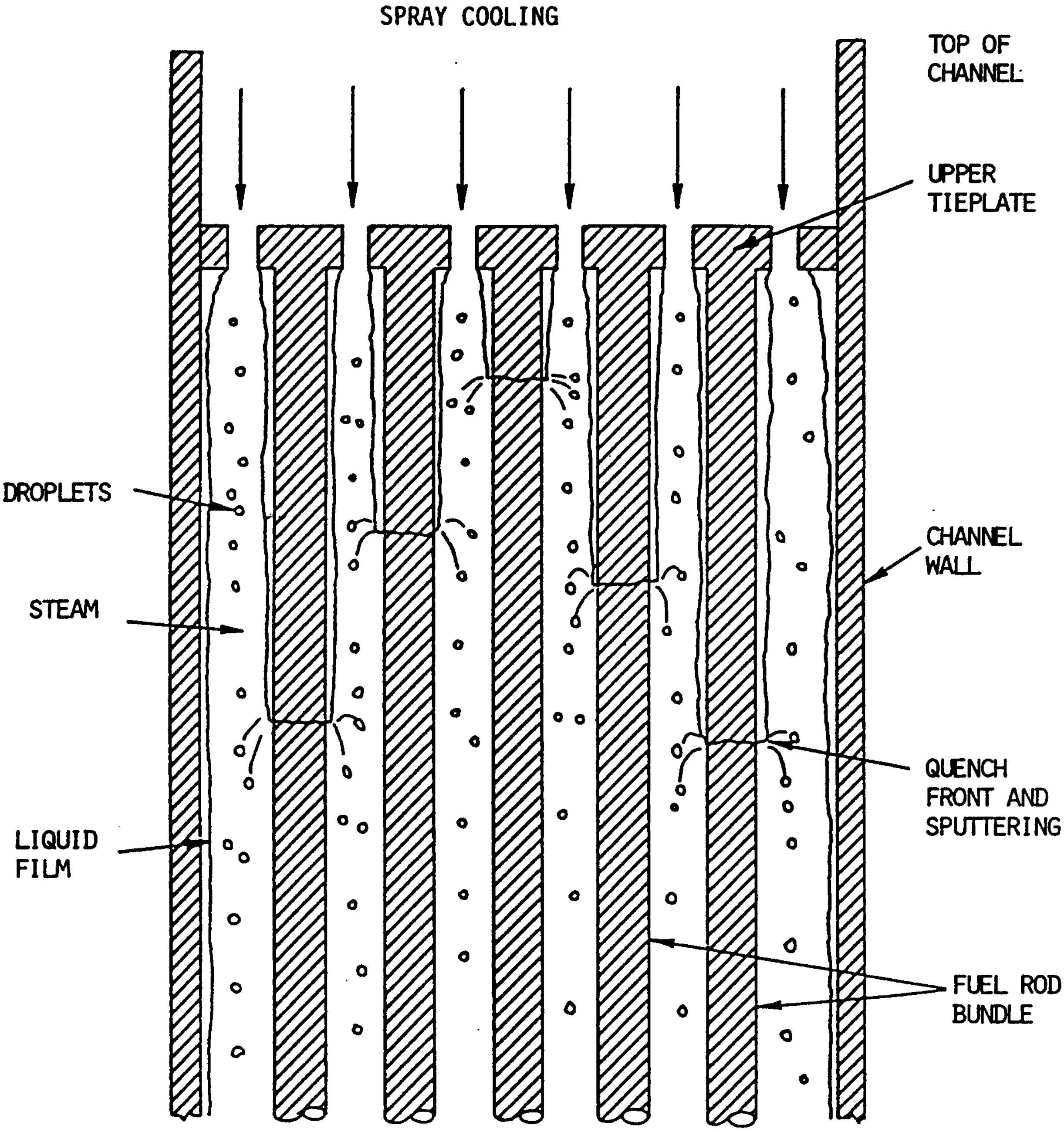


FIG.2.2.10 SCHEMATIC OF HYDRAULIC CONDITIONS IN A FUEL CHANNEL DURING SPRAY COOLING IN A BWR

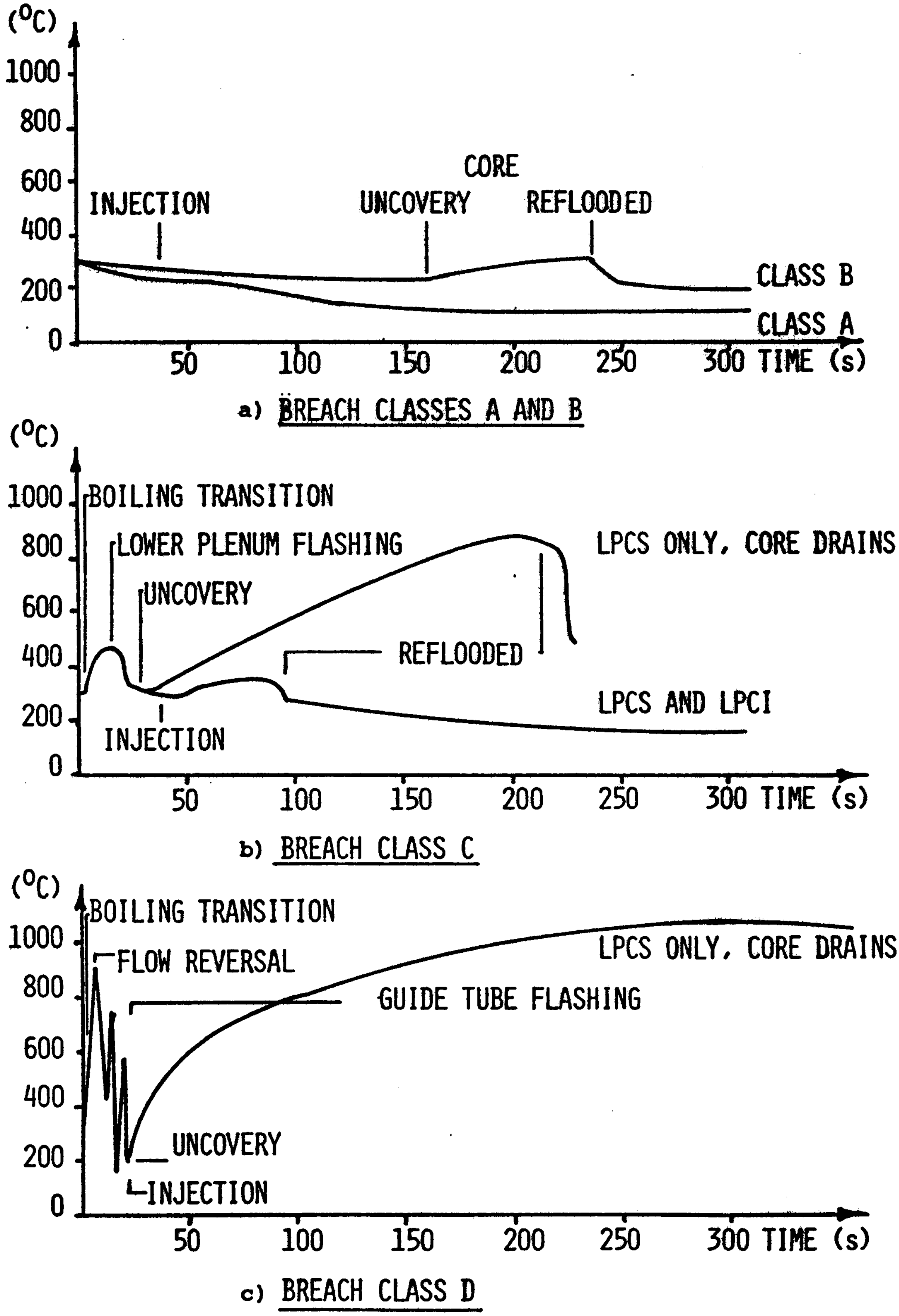


FIG.2.2.11 TYPICAL PEAK CLADDING TEMPERATURES CALCULATED FOR LARGE BREACH LOCA'S IN BWR'S

3. EXPERIMENTAL PROGRAMMES IN EMERGENCY CORE COOLING

3.1 The Role of Experiments

Three broad classes of ECC experiments may be identified:

a) Basic Two-Phase Flow Experiments

These are not necessarily aimed at any particular ECC phenomenon, but rather at investigating the fundamental processes involved with two-phase flow mass-, momentum- and energy-transport. Hundreds of facilities have been used or are still in use for these purposes. Most of them have simple geometries and very small scales when compared to reactor systems. Tests in these facilities provide basic information on flow-regimes, closure relations and correlations for pressure-drop, heat- and mass-transfer, void-quality and interphase-transport. Recourse may be made to such experiments when modelling the individual ECC phenomena, as discussed in Chapter 5. Although they also contribute to code development and validation, discussed in Chapter 6, they are not discussed further here.

b) Separate Effects Tests

Such tests are employed to investigate individual or localized phenomenon expected to occur in a reactor system during a LOCA, or to characterize the behaviour of a single component such as a jet pump. Interactions with other phenomena or components are either imposed as external boundary conditions on the test, or are simply neglected; either because a truly separate-effect is desired or because it may be too difficult to provide or even to define appropriate boundary conditions. Typical examples of these tests include investigations of fuel-bundle heat-transfer, of countercurrent flows at upper tie-plates or in the downcomer annulus, and pump performance tests. Over a hundred facilities have been used or still exist for LOCA separate effects tests /3.1/ to /3.3/. A noteworthy point is that many of these facilities are locally at full-scale, or very near to full-scale, which minimizes the concerns about dynamic similarity and scale effects to those associated solely with the interactions of phenomena, or the lack thereof, at the external boundaries. Such tests provide valuable data for assessing date prediction uncertainties at or near full-scale.

c) Integral Tests

Since it is not a practicable proposition to carry out LOCA tests on full-scale plants, scaled integral tests are used to simulate the overall behaviour of a reactor system during a LOCA. Not only the individual phenomena are important but also the interactions of phenomena and components throughout the entire system. About 20 integral test facilities have been built /3.2/ to /3.4/, one of the largest (LSTF, /3.23/) being volumetrically scaled to ca. 1:50 of a full-size plant, see Fig. 3.1. The size, complexity and cost of these facilities make compromises with respect to dynamic similarity and scaling inevitable. Transient tests and accidents which have occurred in nuclear power plants, may also be classed with integral tests, but data relevant to LOCA scenarios tend to be limited. Integral tests provide valuable guidance on the modelling and noding employed in system codes.

It is the separate-effects and integral tests which are the principal topics of this chapter. They provide the main support for the LOCA phenomenology and scenarios described in Chapter 2; see for example the matrices 1 to 6 of /3.3/ from which Table 3.1 is representative. They buttress the weighting and appraisal procedures in Chapter 4, c.f. Table 3.2 and Table 4.1.1. Finally, as discussed in Chapter 6, they play a central role in assessing the accuracy of reactor system code predictions of plant LOCA behaviour, particularly with regard to uncertainties and scale-effects.

The purpose of this chapter is to provide a critical appraisal of the integral-test facilities and of those separate-effects facilities considered to have contributed significant data and information on emergency core cooling. Since scaling is a major consideration, both as regards design and operation of a test-facility /3.5/ to /3.14/, this is discussed in some detail first. Attention is then focussed on the special features of test-rigs with emphasis on scaling compromises and lessons learned from the tests. These enable recommendations for additional tests to be made.

The drive for additional tests or even new facilities is not the need to confirm the cooling ability or effectiveness of ECC, but rather to quantify this ability more exactly. This becomes an endless task, unless the degree of accuracy required is dictated by its importance to safety or by the economic gains to be made from improved operational flexibility within existing, proven, safety limits. The implication is then that there must be a feedback process between the degree of accuracy required, the degree of accuracy achieved by prediction methods and the need for new experiments and/or better analytic models.

No detailed descriptions of the facilities are provided here, such descriptions are readily available in the literature referenced in Tables 3.3 to 3.7. These tables provide a basic summary of facility characteristics. Summary reports of most test data may be found as references in the separate effects matrix report /3.1/, in the integral test matrix report /3.3/ and in the NRC compendium /3.4/.

Lastly, a review of the progress with test-rig instrumentation is given. What is actually happening in a test is indirectly interpreted from point or line-integral measurements. The instrumentation in thus a very critical link between the phenomena "known" or assumed to occur and the transient scenarios, for example modelled by system codes. Instrumentation accuracy can contribute significantly to the uncertainty in reactor behaviour predicted by such codes.

3.2 General Remarks on Similarity, Scaling and the Design and Operation of Experimental Facilities

3.2.1 Similarity and Scaling Laws

By similarity is meant the correct reproduction or representation in small-scale facilities of the important or dominant phenomena found at full-scale. It is not just a question of ensuring that all the relevant phenomena occur, but that their relative weighting is also correct. Scaling-laws are the tools for achieving similarity, and these laws play a major role both in the conception, design and operation of test-rigs and in the extrapolation to full-scale of LOCA scenarios and data measured at smaller scales. In principle, they define the ideal size and geometry, rig component arrangement and test boundary conditions required to achieve similarity. In practice, poor geometric similarity, incompleteness in the scaling laws, and the inability to meet the requirements even of the incomplete laws, all introduce scale distortions /3.5/ and /3.6/ which have to be taken into account when assessing or predicting plant behaviour.

3.2.2 Scaling Laws

A major problem with dynamic similarity in two-phase flows with heat-transfer is to establish complete scaling-laws. That means to establish a set of laws which are not only necessary but are also sufficient to guarantee similarity.

In general, two different approaches are employed to define such scaling laws: dimensional analysis and system codes used as scaling tools.

The dimensional analysis approach takes either a Buckingham-II theorem, or the governing conservation equations and closure laws for two-phase flows, to provide scaling parameters: Froude, Reynolds, Nusselt, etc. numbers. An attempt at completeness with this method leads in most practical ECC situations to an unmanageable number of scaling parameters and inconsistent scaling criteria /3.7/ to /3.11/. To achieve a practical set of criteria for the design of test-rigs or performance of experiments, the number of scaling parameters has to be drastically reduced. The sufficiency or adequacy of the simplified practical-set is difficult to prove and leads to distortions in scaled quantities, which are not easy to evaluate. Nevertheless, two examples of simplified scaling methods, based on a homogeneous formulation of the two-phase flow equations are commonly used for designing test rigs:

i) Time-Reduced Scaling

The rigorous reduction of linear dimensions of a test-rig results in a proportional reduction in time scales. This is appropriate for simulating pressure wave phenomena; for example, when body forces due to gravity are small compared to local pressure differences. Geometric similarity is retained, but at the cost of severe distortion in heat-transfer processes.

ii) Time-Preserved or Power/Volume Scaling

This is based on a scaled reduction of the volume of the rig, combined with a proportional scaling of energy sources and sinks, such that the ratio power/volume remains constant. Within the scaled volume the relative distribution of component volumes is also preserved. This procedure is essentially an energy scaling of the test rig, thereby preserving the time history of energy controlled events. At the same time, gravity effects can be independently scaled by maintaining geodetic elevations at or near full-scale, and momentum effects by placing orifices in the flow paths. By maintaining elevations in a volume scaled system, geometric similarity is distorted; it is tacitly assumed that the reduced section behaves identically with or without the rest of the system. Scaling of structural heat sources and sinks is the main source of difficulty with this procedure.

An alternative to the dimensional analysis approach, and one considered to be more rigorous, is the direct use of established system codes as scaling tools. This is known as:

iii) Idealized Time-Preserved Scaling through System Modelling

Here it is assumed that all (i.e. complete) scaling-law information is contained in the conservation-equations and closure-laws for two-phase flows. System codes, which are based on these equations may then, in principle, be used:

a) to define the ideal rig, its boundary and operating conditions,
b) to assess and correct for the scale distortions of rigs built and operated on the basis of simpler scaling-laws, such as ii), and
c) to extrapolate test-rig data to full-scale.

Scaling through system modelling using established codes, or analytical simulation models, has been very successful with regard to the design a) and operation b) of integral test rigs. However, the problem of completeness or sufficiency of the closure laws raises questions with regard to scale distortions arising from their use. This problem is compounded by the fact that flow-regime maps, heat-transfer maps and correlations for the closure laws, associated with either the mixture or separated-flow formulations described in Chapter 5, are largely based on small-scale test-data. Their use at much larger scales has by no means been generally confirmed. Integral tests at different scales, specifically counterpart tests, can help to resolve this issue but, the problem of imperfectly scaled facilites must then be considered, see Section 3.2.3 below. Much attention has been focussed on these difficulties by organisations involved with the assessment of uncertainties in model predictions of full-scale plant behaviour c), which is fully discussed in Chapter 6.

3.2.3 Scaled Facilities

Even when given practical scaling laws, which may themselves involve scaling distortions, it is for the most part impossible to fulfil all the requirements imposed by these laws when designing and operating a test-rig. Compromises are made with geometry, with component shape and arrangement, with energy sources (electrical) and sinks (wall effects), with boundary conditions for operation of the rig (power history) and so on. All these introduce additional scaling distortions, which have to be evaluated. It is particularly important to watch out for scaling compromises which influence the relative weighting of phenomena, and even more so if they control the appearance or non-appearance of dominant phenomena.

The wide variety of scaling compromises employed in the various integral- and separate-effects tests, some of which are discussed in Sections 3.3.1 and 3.3.2, as well as the complexity of the phenomena involved with ECC, make a direct quantitative evaluation of scaling distortions resulting from rig compromises almost impossible. It has long been recognised that a synthesis through analytical models is the only practical means to do this. But, it then becomes difficult to distinguish between the distortions arising through the scaling compromises and those, discussed in Section 3.2.2 above, due to the incompleteness of the analytical simulation models themselves.

3.2.4 Counterpart Tests

Counterpart tests have been proposed /3.12/ and /3.13/ as a possible means to separate the above issues. In principle, counterpart tests are integral tests at different scales, selected on the basis that scale-distortions from rig operating procedures or rig imperfections are either minimal or well understood for both rigs. Tests so selected are analysed using established system codes in which all measures are taken to avoid analysis distortions from code user choices; particularly through the chosen nodalization and numerical solution procedures. These are formidable requirements, which may not readily be met and which need close cooperation of all concerned.

Interpretation of counterpart test data is by no means straightforward. A typical proposal is shown in Fig. 3.3, where the ratios of measured to predicted values of three different parameters 1, 2 and 3 are shown schematically as functions of volumetric scale ratios A, B and C, which could represent for example, SEMISCALE, LOBI and ROSA IV. Upper and lower bounds provide allowances for measurement and model uncertainties.

Curve 1 indicates that predictions of parameter 1 at full-scale may be made with some confidence provided:

- code user choice of noding concept and solution procedures at full-scale are identical to those used at small-scale, and

- dominant phenomena do not appear or disappear as a result of the extrapolation. This is difficult to prove other than through separate effects tests. In the absence of such tests reliance has to be placed on engineering judgement.

Curve 3 indicates that predictions of parameters 3 would be very dubious at full-scale, which is a positive although unwelcome result. Interpretation of curve 2 is ambiguous.

In summary then, there is no simple means to account for scale effects in the thermohydraulics of emergency core cooling. A judicious mixture of simplified scaling laws, integral and separate-effects tests, and perhaps some counterpart tests, all supported by an established or proven analytical simulation model, is the scaling methodology that has to be, and is (for example /6.5/), employed. Because of the limitations of the closure laws in two-phase flow equations, the question of completeness or adequacy of any scaling methodology for most two-phase flows cannot be fully answered. An element of engineering judgement always remains. These points are taken further in Chapter 6.

3.3 Separate Effects and Integral Test Facilities

3.3.1 Separate Effects Test Facilities (Tables 3.2 and 3.3)

3.3.1.1 Scaling Considerations in Separate Effects Tests

For separate effects tests the main scaling problems lie not so much with size, since many tests have been performed at or near full scale, but rather with boundary conditions. Distortions introduced by boundary conditions are assessed, where possible, by judicious comparisons of test data at various scales or from parameter studies using established analytical models. Chapter 6 of the NRC Compendium of ECCS research /3.4/ illustrates this rather well. Clearly, the largest scale separate effects tests have immense value for establishing code prediction uncertainties at or near full-scale. Typical examples of problems encountered with separate effects tests are:

- Blowdown
  Blowdown of a subcooled fluid depends on the thermal non-equilibrium processes influencing the nucleation delay of vapour-bubbles or dissolved-gases in the fluid. Low-quality critical breach-flow depends on thermal non-equilibrium and the exchange processes occurring between the phases. Blowdown depends quite strongly on specific breach geometry, throat length-to-diameter ratios, flow velocity profiles and upstream flow conditions. These are difficult to scale consistently, which demonstrates the value of the MARVIKEN full-scale tests.

- Mixing
  Fluid-to-fluid mixing and fluid-to-vapour mixing commonly occur following ECC injection during a LOCA. Both have been extensively investigated in separate effects test-rigs. With fluid-to-fluid mixing, scaling considerations are governed by hardware dimensions and a few dimensionless groups, e.g. Froude and Reynolds numbers, which are relatively easy to conserve. Consistent boundary conditions on the injected coolant are also straightforward to achieve.

  The contrary is true of fluid-to-vapour mixing in which phenomena such as condensation, entrainment, counter-flows and de-entrainment may play a role. Overall dynamic similarity is governed by local geometry and the thermodynamic properties and global flows of the steam and water involved. But, the time history, distribution and subcooling of incoming emergency or other core coolant, the rewetting of structures and the heat-transfer to and from test rig structures are particularly sensitive to test rig size and layout, in particular the location of ECC injection. It is thus very difficult to define, scale and reproduce consistent boundary conditions for all the phenomena concerned. These difficulties are compounded when phenomena, such as ECC bypass, downcomer penetration and counter-current flow-limiting in the loops or above the core are investigated, because they may occur at locations far removed from the ECC injection locations. Boundary conditions at the mixing locations of interest are then controlled by integral effects from the injection point to the mixing location. Under these circumstances a commonly applicable and accurate correlation of test data is difficult to achieve owing to the uncertainties in the boundary conditions. These problems were the main driving forces behind the 2D/3D programme for PWR's (UPTF, CCTF and SCTF) and other full-scale separate-effects tests, such as the SSTF for a BWR.

- Flow Stratification

  Flow stratification strongly influences small breach PWR-LOCA scenarios particularly with regard to the breakdown of natural-circulation flow, to reflux condensation and to loop seal clearing. Although simple scaling rules can be established for stratification in horizontal pipes, see Section 3.3.2.1, these rules no longer apply when entrance or transient effects become dominant. This occurs at scales, representative of full-size plants, when the length/diameter ratios of the horizontal pipes are just not large enough to allow flow-regimes to become fully-developed. Under these conditions perturbing flows from the vessel or steam generator plena control the formation and breakdown of flow-stratification in the loops. Such effects, which were evident in the LSTF integral test facility, are very difficult to scale or to simulate consistently. Experiments in the large diameter horizontal pipes of TPTF provide useful information on the formation and breakdown of flow stratification /3.24/.

- Fuel Bundle Heat Transfer

  Because most heat-transfer tests have been carried out with full-scale electrically heated bundles, the problems of simulation and scaling are reduced to:

  1) The specification of representative bundle boundary conditions. This is easier for a BWR bundle in its enclosed channel: bundle flow

phenomena are largely one-dimensional, there is no cross-flow between bundles, bundle top and bottom boundary conditions are governed by conditions in the upper and lower plena. Plena boundary conditions can vary enormously, particularly across the top of the core. Different boundary conditions and bundle resistances can lead to upflows, downflows and parallel flows occurring in neighbouring bundles under some ECC conditions. These were good reasons for performing full-scale tests in the SSTF.

Similar arguments about boundary conditions apply to PWR bundles, but then the situation is made more complex by the possible large-scale counter-flows and cross-flows that can occur in and between bundles. Hence, a scaling concern with PWR bundles is also:

2) The size or number of bundles, which can be considered as representative. This concern has driven experiments in the direction of more and more parallel bundles (SCTF).

3) The influence of simulating fuel rod heat-transfer and stored-energy in BWR's and PWR's by electrically heated rods. The nuclear core in LOFT, despite its ca. half-length, provided a good comparison base for separate effects test data. The influence of electrically heated rods is adequately taken into account in system code analytic models of the core, at least until massive, core-wide, cladding deformation or damage well beyond a design basis situation occurs.

3.3.1.2 Phenomena Investigated in Separate Effects Test Facilities

An overview of the phenomena investigated in the separate effects test facilities, which are or were available in the OECD member states, is provided by Tables 3.2 and 3.3. These tables clearly illustrate the vast scope of tests in support of the detailed LOCA phenomenology discussed in Chapter 2. It is neither necessary nor practicable to detail here the merits of all the separate effects tests listed in these tables. A synopsis of each test and references are given in a review report /3.1/ being prepared by the OECD-NEA-CSNI.

Separate effects tests have provided the primary experimental data bases for the correlations, flow-regime maps and heat-transfer maps used in the two-phase mixture or two-fluid formulations in system codes. The adequacy of these data bases is the topic of Chapter 4. Scale effects are evident in many tests, which have led to modifications at the basic correlation level as well as in the way boundary conditions or phenomena at the boundaries are handled. The tests at or near full-scale have also clearly brought out the importance of two- and three-dimensional effects, which influence the weighting of different LOCA phenomena and thereby alter the progression of a LOCA scenario. These points are discussed further in Chapter 5. Some typical lessons learned from the tests include:

- Critical flow in large diameter pipes (MARVIKEN, SUPER MOBY-DICK). Only the thermal non-equilibrium critical-flow formulations perform well at large scales.

- Phase separation in horizontal and vertical flows. It is difficult to distinguish between scale effects and two- or three-dimensional influences (TPTF, THL, CANON).

- Entrainment and de-entrainment. Interfacial drag correlations are continually being changed to accommodate measured parameters (NEPTUN, PERICLES, UPTF).

- Liquid and vapour mixing with condensation. The importance of realistic distributions of subcooled water in the mixing regions of the plena above and below the core (SSTF, UPTF, ESTA) in the core itself (SCFT) and in the neighbourhood of ECC injection locations (COSI, UPTF) has been amply demonstrated.

- Pressure losses at different scales (MOBY-DICK, TPTF).

- Behaviour of pumps, jet-pumps, separators and dryers at different scales (KARLSTEIN, EPOPEE, LTSF).

- Two- and three-dimensional effects in the core during refill and reflood (SCTF, PERICLES), counter-current flow-limiting and its breakdown (SSTF, UPTF) spray distribution and cooling (SSTF, SNTF, SHTF).

3.3.2 Integral Test Facilities (Tables 3.4 to 3.7)

3.3.2.1 Scaling Considerations in PWR Integral Test Facilities

Time-preserved scaling (power to volume) with separate preservation of geodetic elevations, momentum losses and gains is most commonly used for the integral PWR test facilities (Tables 3.4 and 3.5). The reduction in primary system volume is largely achieved by an equivalent reduction in vertical flow cross-sections, as illustrated in Fig. 3.1. Various scaling compromises are evident with different facilities:

- Not all the cross-sections of the piping exhibit consistent scaling ratios, see Fig. 3.2. Annulus cross-sections are more often than not larger than scale. Such compromises are selected to improve pressure drop scaling or to avoid amplifying three-dimensional effects, particularly in the annulus.

- Preservation of geodetic elevations is very important for all test rigs. OTIS and MIST which simulate reactors with once-through-steam-generators are, for example, 29 m in height. The half-length core and steam generators in LOFT introduce scale distortions for small-breach simulations, but then LOFT was designed originally for investigating large-breach phenomena.

- Except for PKL and CCTF, all the facilities operate at full reference pressure. The lower pressures at PKL distort the small-breach LOCA behaviour. CCTF was designed in any case for investigating large-breach reflood (i.e. low-pressure) phenomena. All the facilities have ECC systems.

- Momentum losses are scaled by compromises in the volume scaling, in the scaling and simulation of primary system components and by orificing. Momentum addition is scaled through the control of coolant pumps.

Scaling of core power is straightforward with electrically heated simulators, heat addition being controlled by the energy supply to the simulators. Electrically heated cores introduce distortions due to their much smaller thermal capacity; LOFT is the one test rig with a nuclear core. Rod linear power differs between facilities. It is not possible to simulate full-power on the LSTF or BETHSY electrical simulators, but specific operating procedures are established to account for this during the LOCA simulation. Secondary-side heat deposition rates are very important for small-breach LOCA investigations and are generally regulated by steam generators. Most of these power simulations are pre-programmed or manually controlled. Special attention was paid to obtaining good secondary-side scaling and control on the OTIS, LOBI and BETHSY rigs.

A major problem with all integral tests is the distortion of structural heat-transfer, and hence the magnitude and timing of the two-phase flow LOCA phenomena, by the larger area-to-volume ratio of heat-sources and sinks in the scaled confining structures. Various rigs are designed to compensate, at least partially, for these effects. For example, heaters and insulation were employed on OTIS, MIST, BETHSY and SEMISCALE. Structural heat-transfer is readily handled in system codes.

The above scaling procedures faithfully reproduce global single-phase forces and natural convection, as well as the almost homogeneous two-phase flow patterns experienced during a large-breach blowdown /3.8/. They do not provide a scaling methodology, for example, for the position and geometric shapes of downcomers and core bypass passages, for leakage flows between upper-plena, upper-head and downcomers, for the number and configuration of loops or for the shape and location of the pressurizer and surgeline. Their influence is assessed by recourse to analytical models or is handled pragmatically.

The combined effects of several loops on a PWR may strongly influence the early rewet of the core during a large breach LOCA, or the small-breach core uncovery processes driven by loop-seal clearing and steam-binding. This has led to a steady increase in the number of loops being used on PWR integral test facilities, ranging from the early rigs with 1 combined-intact-loop plus 1 breached loop (SEMISCALE, LOFT, LOBI) to the later rigs with completely separate loops (SPES, PKL, BETHSY), Table. 3.5

Finally, reductions in flow cross-sections have a pronounced influence on the phase-separation processes as well as on the separated-flow patterns evident during a small-breach LOCA. These patterns can have a strong influence on breach-flow, energy transport and loop-seal behaviour, all of which affect core uncovery behaviour. To reduce the dependency of these phenomena on test rig scale, new facilities such as SPES, BETHSY and the LSTF were designed to preserve flow stratification phenomena in the horizontal sections of the loop piping. This requires /3.5/ and /3.13/ operating the test rigs under identical full-scale boundary conditions and conserving the ratio length/diameter $^{1/2}$ in the horizontal pipes. The ratio is based on a modified Froude number, similarity requirements for flow regime transition and time-preserved scaling of the gravity dominated, local phase-separation process.

3.3.2.2 Lesson Learned from PWR Integral Test Facilities

In general the tests have supported the adequacy of ECC systems and operator actions to maintain cladding temperatures well within LOCA design criteria for all design-basis and, indeed, well beyond design-basis LOCA situations. System code predictions of reactor behaviour have confirmed this. All tests have shown that large-breach LOCA's exhibit a fairly straightforward global behaviour; the primary system blows down, is refilled by the ECC systems and the core is reflooded. Local details can be very complex, but they do not greatly influence this global behaviour. The contrary is true of small-breach LOCA's for which the both the primary- and secondary-system's global behaviour can dominate the core uncovery process.

1) Large-Breach LOCA's

The smaller, tighter, rigs SEMISCALE, LOBI, and PKL tended to amplify the ECC bypass and poor downcomer penetration during blowdown, which illustrates clearly the difficulties involved with extrapolating the LOCA phenomenology directly to full scale. Downcomer penetration occurred much sooner in the larger rigs (LOFT, CCTF) because of expected three-dimensional effects and the reduced influence of hot walls.

Early DNB is evident in most tests, but is followed by sufficient post-CHF cooling during the blowdown phase to limit the cladding temperature rise or first-peak to about 800 °C. Tests in LOFT and LOBI provide strong evidence for early rewetting of the core during blowdown, but this may be influenced by scale.

Cladding temperatures during core uncovery reached peaks which, in general, were not larger than the post-DNB peak, unless ECC injection was excessively reduced. Steam and two-phase cooling of the core control these second peaks and turn cladding temperatures down well before the core is actually reflooded. All tests provided support for the refilling and reflooding processes being straightforward processes on a global basis.

Various ECC injection modes were investigated. Cold-leg injection has been shown generally to be effective in cooling the core by reflooding from below. PKL and CCTF tests showed that hot-leg (upper-plenum) injection can be very effective in cooling the core from above, as well as by reflooding the core from below, once coolant break-through or flooding has occurred. Coolant injected into the hot-legs and upper plenum do, however, produce complicated flow-patterns in and above the core: pool formation in the upper plenum, counter-current flow limiting and breakdown, top-down quenching, counter-current flows and cross-flows in the core. Some of these effects are also noticed in small-breach integral tests and have been intensively studied in separate effects facilities.

Most tests have provided information on heat-transfer correlations during the core reflood and quench processes. They have indicated some shortcomings (LOFT) in code heat-transfer packages, which are still being investigated.

2) Small-Breach LOCA's

Tests in LOFT have shown how important it is to trip the primary system pumps at an early stage in a small-breach LOCA, in order to avoid an unnecessarily large loss of primary inventory through the breach, and hence, the potential for core uncovery. LOFT also confirmed the water hold-up in the pressurizer during a stuck open relief valve transient. These were typical post TMI-2 investigations.

Key transitions in natural circulation are driven by momentum balances round the loops, which are themselves influenced by two-phase flow rates, flow transitions, loop-seal clearing and hydrostatic heads. These can vary between differently scaled facilities.

After an early trip of primary system pumps and during the time period when the secondary-side heat sink was effective, all small-breach tests in which inventory was not maintained by ECC systems, demonstrated a distinct progression from single-phase natural-circulation in the primary system, through to two-phase natural-circulation at a mass inventory of about 85 % (BETHSY, LSTF, LOBI, PKL) to reflux condensation at about 60 % (BETHSY, LSTF, LOBI, PKL). During this progression non-uniform behaviour was evident in the steam generators of the LSTF loops /3.25/. Flow-reversal and emptying of different U-tubes occurred at different times. Counter-current flow effects were also noticed in the LSTF steam-generators and this tended to hold-up liquid in the U-tubes. Such hold-up was not evident in BETHSY tests. Hold-up in the hot-leg pipework was measured in most LOBI experiments.

During reflux condensation the core in the test rigs remained covered and adequately cooled until the primary system inventory fell to about 30 % (BETHSY, LSTF, LOBI, PKL). Reflux condensation appears to be shared almost equally between the up-flow and down-flow sides of the steam-generators (BETHSY, LOBI, PKL, SEMISCALE).

All facilities gave valuable information on the categories of small-breaches requiring secondary-side cooldown. The adequacy of procedures, or of automatic measures, to achieve sufficient secondary-side cooldown (ca. 50 K/h) has been demonstrated many times. Tests performed in several facilities (e.g. LOBI, BETHSY) showed that one steam generator suffices for cooldown purposes, but operating with isolated steam generators produces unbalanced convection and inverse heat-transfer in the isolated loops. This is to be expected and, although it complicates the flow processes, it does not interfere with core cooling.

PKL tests demonstrated that the U-tube steam-generator heat-transfer effectiveness during natural circulation and reflux condensation is not too sensitive to secondary-side water levels even down to very low levels. This differs strongly from the situation with once-through steam-generators, see below. PKL and SEMISCALE tests further showed that non-condensable (accumulator) gases in the loops interfere with reflux condensation, reducing primary-to-secondary side heat-transfer.

Core water level depression, as a result of steam-binding, loop back-pressures and an uncleared loop-seal, was evident in many tests. Asymmetric loop-seal clearing occurred in the multi-loop rigs BETHSY and LSTF. The first loop that cleared prevented the remaining loops from clearing, but did not hinder post loop-seal-clearing core recovery. Loop-seal clearing was in general not strongly dependent on cold-leg breach location. A hot-leg breach tended to be beneficial in terms of inventory loss (PKL) and for relieving steam-binding back-pressure, until ECC injection into the cold-legs condensed steam there. The vacuum so produced sucked down the core water level (LSTF /3.27/).

Core-reflood phenomenology for small-breaches proved to be very similar to that for large-breaches. The major exception is that any steam produced during the reflood phase tends to repressurize the primary system, rather than flowing rapidly to a breach. Thus, counter-current flow effects in and above the core are much reduced.

3) Small-Breach LOCA's in Plants with Once Through Steam Generators

Tests in both of the rigs OTIS and MIST were oriented towards investigating the small-breach natural circulation behaviour of primary systems equipped with once-through steam-generators. They demonstrated the effectiveness of one- and two-phase natural circulation and of the boiler-condensor mode of heat-transfer to the secondary side. They confirmed the breaking-down of reflux condensation when the water level in the primary-side of the steam-generator is above that of the secondary-side, showing that maintaining a high secondary-side water level is crucial to primary system depressurization during the boiler-condensor mode. On the other hand, a ready return to natural circulation was demonstrated (OTIS) by refilling the loop. This required sufficient high-pressure injection to compensate for the breach flow, together with the venting of steam in the top of the loop or Candy Cane.

Reactor vessel vent valves allow steam to flow from the upper plenum to the cold-legs, thereby reducing the back-pressure which is depressing the water level in the core. Vented steam is condensed in the cold-legs by ECC injected there.

But for the anticipated characteristics of the once-through steam-generators and the reactor vessel vent valves, no new or unexpected differences in small-breach LOCA behaviour were apparent from the OTIS and MIST tests.

4) Intermediate-Breach LOCA's

LOFT tests, in particular, showed that there are no new or adverse phenomena to be expected from intermediate-breach LOCA's. Elements of both small and large-breach behaviour were noted and ECC proved effective.

3.3.2.3 Scaling Considerations in BWR-Integral Test Facilities

Scaling procedures and compromises similar to those for PWR rigs are employed in BWR integral test facilities (Tables 3.6 and 3.7); power-to-volume scaling for large-breach simulations, separate preservation of momentum losses and gains and of geodetic elevations for small-breaches. The chosen reduction in vertical flow cross-section is largely controlled by the desired number of representative rod-bundles in the test rig:

- PIPER-I, FIX-II and FIST have full-length, single-bundles, although only FIST has a full quota (64) of rods.

- TBL and ROSA III have 2, respectively, 4 full-bundles, but half-length rods are employed in ROSA III to achieve the desired volume scaling with the extra bundles. To compensate in ROSA III for the reduced steam production, which strongly influences counter-current flow effects, flow areas of the inlet orifices and upper tie-plates are halved /3.26/. Both these rigs provide information on bundle interactions at the inlet and exit plena.

All the rigs incorporate relevant vessel internal components and operate at full reference pressure. All but PIPER-I have steam and feedwater control systems, all but FIX-II have ECC systems. Various solutions are employed for momentum losses and gains. External recirculation loops driving area scaled-jet pumps are employed on FIST, ROSA III and TBL. The jet pumps on FIST and ROSA III are mounted inside an external pipe representing the downcomer annulus, whereas those in TBL are mounted inside a scaled annulus. FIX-II simulates reactors without internal jet-pumps and so has only two external pumps. PIPER-I is designed for investigating natural-circulation in the core and downcomer during small-breach LOCA's and thus has no external loops.

All the rigs use electrically heated rods, which are controlled to simulate full-power (FIST, TBL, FIX-II), reduced power (ROSA-III) or decay heat (PIPER-I) histories. A major problem in all the rigs is the distortion of structural heat-transfer through scaling. This has a dominant effect on steam production, amplifying the influence of counter-current flow-limiting effects. In FIST and TBL this is partially compensated by increased flow areas. To reduce the influence of excess structural heat on natural circulation, external cooling is employed on PIPER-I.

3.3.2.4 Lessons Learned from BWR Integral Test Facilities

1) Large- and Intermediate Breach LOCA's

Tests in FIST, ROSA III, TBL and FIX-II have demonstrated the basic large- and intermediate-breach LOCA behaviour; an early boiling-transition and core uncovery, quenched by lower-plenum and/or guide-tube (FIX-II) flashing, which re-distributes coolant within the pressure vessel; a subsequent core uncovery, which depends on breach size, location and ECC injection. In general tests have supported the adequacy of ECC systems to maintain cladding temperatures well within LOCA design criteria. System code predictions of reactor behaviour have confirmed this.

The most important lesson learned from the integral tests is the dominant influence of counter-current flow-limiting on core cooling during core uncovery situations (FIST, ROSA-III and TBL). Counter-current flows at the upper tie-plate do not maintain a pool of water in the upper-plenum for an indefinite period. Indeed, subcooled ECC injected into the upper-plenum by core-sprays, or overflowing from the bypass, rapidly breaks down the limiting counter-flow condition enabling ECC to penetrate the heated-bundles. At the same time, counter-current flows at the core inlet orifices hold-up coolant in the core, maintaining a good degree of core cooling even when a water level is apparent in the lower plenum. Counter-flow limiting conditions at inlet orifices can break-down too as subcooled ECC leaks into the channels from the core bypass.

Care has to be taken when extrapolating the integral-test counter-current flow data to full-scale, because steam production in the test-rigs is exaggerated by stored energy, particularly in FIST. Corrections for stored energy were introduced in ROSA-III and TBL. The three-dimensional influences of spray-cooling and mixing on counter-current flow phenomena were intensively investigated at full-scale in the SSTF, separate effects test facility. These tests demonstrated non-uniform behaviour of different channels, early break-down of counter-current flow-limiting at the cooler peripheral bundles and rapid refill and reflood of the core following this breakdown. TBL data also provided evidence for neighbouring bundles exhibiting different uncovery/recovery histories due to counter-flow effects and an imbalance of coolant between channels. Both top-down quenching from spray cooling and bottom-up quenching from reflood were evident in FIST tests.

The worst cladding temperature excursions during core uncovery tended to occur for recirculation line breaches at the larger end of the breach spectrum (FIST), but not always for the largest breach (ROSA-III, TBL). For large breaches the effective breach size is reduced by the available flow areas in the jet-pumps and recirculation pumps. The higher the breach location on the pressure vessel the less severe is the core uncovery excursion. Eventually, for an equivalent steamline breach, no uncovery occurs at all (FIST). Loss of high pressure ECC proved always to be the worst single failure (FIST, ROSA-III, TBL). Under these conditions flashing of feedwater in the feedwater line tended to delay vessel depressurization and low-pressure injection (ROSA-III /3.26/).

2) Small-Breach LOCA's

Since automatic depressurization essentially changes the small-breach BWR-LOCA into a large-breach condition, small-breach testing in integral facilities has focussed upon natural circulation behaviour prior to automatic depressurization (FIST, PIPER-I) and upon the influence of depressurization on coolant inventory distribution (FIST) within the vessel.

The natural circulation tests demonstrated the two basic modes of circulation in a BWR; core-to-downcomer and core-to-bypass. In both situations the core was adequately cooled and no boiling transition occurred. Early recirculation pump trip had no influence on the evolution of the small-breach scenarios, but cooling of test-rig structures did (PIPER-I).

Automatic depressurization led to flashing of the coolant and swelling of coolant levels in the pressure vessel (FIST). As soon as the depressurization rate decreased, the flashing subsided, the swollen mixture level fell back and the core uncovered. This behaviour is no different to that experienced for simulated steamline breaches, except that core uncovery occurs because the automatic depressurization started with water levels just above the core.

3.4 Status of Test-Rig Instrumentation

Several hundred measurements are by no means exceptional on integral and large separate-effects facilities. Standard instrumentation found on most rigs include: fast-response transducers for the measurement of hydrostatic head, flow static pressures and pressure differences; shielded or cooled thermocouples, sometimes mounted on rakes for measuring fluid temperatures; surface mounted and buried thermocouples for measuring structural or cladding temperatures; conductivity probes for indicating mixture levels;

transducers for valve positions, pump speeds, injected coolant flow rates, energy supplies and so on. All of these involve fairly straightforward techniques for the measurements themselves and the data processing.

The contrary is true for two-phase global parameters such as void-fractions, mass-flows and velocities. Nevertheless, much progress has been made with their measurement on ECC test-rigs, Appendix C of /3.4/:

- Mechanical means to measure two-phase momentum-flux or mass-flow, such as drag-discs, screens or turbine-meters are used in many facilities (e.g. SEMISCALE, LOFT, PKL, BETHSY, ROSA III). These instruments have first to be independently calibrated, by itself a difficult task, they interfere with the flow field in which they are placed and they provide a time-averaged or integral measurement across the face of the instrument exposed to the flow. To minimize flow interference and to obtain a local measurement many of the probes are miniaturised (bi-directional microturbines of 12 mm diameter in BETHSY). How small a probe can be, typically several mm, is limited by the structural design required to withstand the large forces and severe environment to which it is exposed. To reduce calibration errors probes are sometimes used in conjunction with a gamma-ray densitometer (MIST, BETHSY). Alternatively, the combination may be used to extract flow velocities.

  A bulk measure of mass flow rates may be obtained by collecting or sampling fluid portions of interest. SEMISCALE made use of condensor tanks and MIST has a non-condensable gas collector system. A condensate sink and venturi meter is being used on BETHSY to measure reflux condensate with some success.

- Gamma-ray densitometers, which do not interfere with the flow field, provide a line-integral of the density or void fraction in the flow. They are used extensively for ECC testing (SEMISCALE, MIST, ROSA III, TPTF, TBL, BETHSY), and have undergone steady improvements to provide more information (dual-beams in OTIS and MIST, triple-beams in BETHSY, traversing beam in TBL) more localized information (horizontal and vertical traversing beams in TPTF) or better sensitivity (soft gamma-rays in TPTF). Since traversing time must be shorter than the timescale of any gobal transient in the flow, traversing beams find application in more or less steady-state situations only. X-rays have also been used to measure line densities, but shielding and collimation then pose more difficulties than with gamma-rays.

- Various approaches are used to measure flow velocities. Global values may be found from orifice or venturi pressure-drop measurements. Local values have been measured with a cooled thermocouple probe (MIST), and some attempts have been made to use an ultrasonic flow meter based on the Doppler effect (OTIS). None of these methods are very accurate. Much better results are obtained with pitot-static devices (ROSA III, TPTF, TBL) which are purged or cooled to prevent void formation in the sensing lines.

- Although flow regime transitions may be indirectly monitored through changes in signals from probes immersed in the flow (thermocouple rakes in MIST), flow visualization provides a direct indication. A video camera viewing a cross-section of the flow field is found on MIST. Small video probes, which can be inserted into the flow field of interest are employed on TPTF, TBL and LSTF. In the latter rig the width of a cooling film was measured with this probe.

None of the more fundamental two-phase flow quantities: interfacial area, interfacial-mass, momentum and energy transport, for example, are actually measured on ECC rigs. This is not surprising since they have not been measured in basic two-phase flow experiments either. In the absence of direct measurements, interpretation by calculation is the only available means to provide information, and this is the method used with ECC-tests. The approach is to assume a general validity of the relationships for fundamental quantities, such as interfacial drag, which are then employed in system codes to compare predicted global parameters with those measured on the rigs. These relationships, which are assumed to be the sole cause of discrepancies between measured and predicted parameters, are adjusted to remove or minimize discrepancies. Such assumptions are easy to make, but difficult to prove. Their use has led to correlations with wide-ranging values, since many phenomena in complex integral and separate-effects tests can cause identical discrepancies, see Chapter 5 and part II of reference /5.1/ in particular.

The only way out of this "blind-alley" is to develop techniques to measure fundamental quantities directly, or to restrict the indirect calculations to very simple experiments. Direct measurements will be very difficult to make and to interpret. Simple experiments have already been used during the system code development period, but they do not seem to have reduced the range of the resulting correlations, which then raises the question of the generality of the fundamental relationships so developed. On the other hand, even wide-ranging correlations at the fundamental level can provide predictions of parameters important to safety, such as peak cladding temperatures, with acceptable accuracy. These points are addressed further in Chapters 5 and 6.

3.5 Concluding Remarks, Recommendations for Additional Tests, New Facilities, New Instrumentation

In the interest of remaining within reasonable bounds the LOCA phenomenology, the problems with regard to scaling and the lessons learned from light water reactor LOCA experiments have simply been highlighted here. A more complete picture of the phenomenology, based on the same experiments plus analytical predictions of plant behaviour, is provided in Chapter 2. The central problem of scaling is tackled by a combination of separate-effects tests for dominant phenomena at or near full-scale, integral-tests with power-to-volume scaling, the use of analytical models in system codes to tie it all together and engineering judgement. How well the system codes perform in predicting full-scale plant behaviour is the topic of Chapter 6.

Throughout the many years of testing, steady improvements in understanding the details of LOCA behaviour have been made. Several surprises have turned up: steam-binding, ECC-bypass, downcomer penetration, counter-current flow limiting and its breakdown. Some of these were found to interfere with core-cooling, others to assist in the process. Scaling and multi-dimensional effects influenced the weighting of the phenomena. But, none of these details have changed the basic contention that ECC systems, provided they are not prevented from performing their safety function, are more than capable of maintaining cladding temperatures below those at which substantial clad bursting, geometric changes or exothermic chemical reactions occur in the core.

The drive then for additional tests or even new facilities and/or new instrumentation is not the need to confirm the cooling ability or effectiveness of ECC, but rather to quantify this ability more exactly. This becomes an endless task, unless the degree of accuracy required is dictated by its importance to safety or by the economic gains to be made from improved operational flexibility within existing, proven, safety limits. The implication is then that there must be a feedback process between the degree of accuracy required, the degree of accuracy achieved by prediction methods and the need for new experiments and/or better analytic models.

Accuracy requirements/achievements are central issues of code assessment discussed in Chapter 6. With so many parameters involved, an assessment becomes unmanageable without first eliminating those LOCA phenomena, which do not have a dominating influence on parameters important to safety. This led to a rating and appraisal procedure, which uses expert opinion supported by the above described LOCA experiments and, in some cases, parameter studies:
dies:

- to define dominant LOCA phenomena, and
- to appraise the physical understanding, scale-dependence, data-base and analytic modelling of the dominant phenomena.

Because of its central importance to this SOAR, this procedure is carried out separately and is fully described in Chapter 4. Until code assessment is completed, the rating and appraisal procedure itself lays firm foundations for judicious recommendations with regard to additional testing. Typical recommendations are given in Tables 4.2 and 4.3. As is to be expected, most of them concern phenomena involving multi-dimensional, fluid-to-vapour mixing. Other than experiments concerned solely with phenomena, a case can also be made for counterpart tests to aid in resolving the issue of full-scale predictions. They cannot resolve the issue directly but need support from separate effects tests at full-scale. Whether new test-facilities for these purposes are needed, or whether tests can be performed on existing facilities, requires a closer study of the above recommendations as well as the capabilities of existing test rigs.

Recommendations for modelling improvements of dominant LOCA phenomena are also listed in Table 4.2 and are discussed in Chapter 5. With many of the analytic models no real progress can be made until measurements of fundamental two-phase flow quantities at the interface level are carried out. This is easier said than done, since neither instruments nor rigs exist to investigate the dominant phenomena so closely. As far as global measurements of two-phase flow parameters are concerned, there appears to be no urgent need for new instrumentation, but see Appendix C of /3.4/ for further discussion on this point.

It is stressed that the above recommendations are solely concerned with design-basis LOCA conditions expected in the current generation of light water reactors. New or advanced reactors, expecially those with tighter core lattices, different internals, different loop geometries or different ECC systems, will require new, specific, correlations for parts of the system codes chosen to assess ECC effectiveness. Reliance on existing correlations, despite the claims made for the advanced nature of the two-fluid codes, would be a questionable procedure indeed, see Chapter 5. Another reason for additional testing is expansion of the data base beyond the design-basis to bridge the gap between design-basis and severe accidents.

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Simulation Tests"
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/3.27/ K. Tasaka et al.
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CROSS REFERENCE MATRIX

- Phenomenon versus test type
  - ● simulated
  - ○ partially simulated
- test facility versus phenomena
  - ● suitable for code assessment
  - ○ limited suitability
  - x expected to be suitable
- test type versus test facility
  - ● already performed or planned until 12/85
  - ○ performed or planned until 12/85, but of limited use

		Test Type							Test Facility: System Tests								Test Facility: Separate Effects Tests						
		Stationary test adressing energy transport on prim. side	Stationary test adressing energy transp. on sec. side	Small leak overfeed by HPIS, secondary side necessary	Small leak w/o HPIS overfeeding, secondary side necessary	Intermediate leak, sec. side not necessary	Pressurizer leak	U-tube rupture	PWR 1:1 (1)	LOFT 1:50	LSTF 1:50	BETHSY 1:100	PKL-I 1:134	SPES 1:430	LOBI-II 1:712	SEMISCALE 1:1600	UPTF 1:1	THL 1:15	GEST GEN 1:50	Patricia GV-1	PatriciaGV-2/GEN 3x3	G-2 or Pericles	Pressurizer Test (CISE)
Phenomena (4)	Natural circulation in 1-phase flow, primary side	●	●	●	○	–	●	●	●	●			●		●	●		–			–	–	–
	Natural circulation in 2-phase flow, primary side	●	–	○	●	●	○	–	–	●			●		●	●		–			–	–	–
	Reflux condenser mode and CCFL	●	–	–	●	●	–	–	–	○			○		○	○		–		●	–	–	–
	Asymetric loop behaviour	–	–	●	●	–	○	●	–	–	X	X	–	X	○	○		–			–	–	–
	Leak flow	–	–	●	●	●	●	●	–	●			○		●	●		●			–	–	–
	Phase separation without mixture level formation	●	–	○	●	●	●	○	–	○			○		●	○		○			–	–	–
	Mixture level and entrainment in vertic. comp. SG(2)	–	●	●	●	●	●	●	–	–			○		○	–		–	X		●	–	–
	Mixture level and entrainment in the core	●	–	–	●	●	●	–	–	○			○		○	○		–			–	●	–
	Stratification in horizontal pipes	●	–	–	●	●	–	–	–	●	X	X	○		○	○		●			–	–	–
	ECC-mixing and condensation	–	–	○	●	●	●	●	–	○			○		○	○		–			–	–	–
	Loop seal clearance	–	–	–	●	●	○	–	–	●			○		●	●		–			–	–	–
	Pool formation in UP/CCFL(UCSP)	●	–	–	○	●	●	–	–	○	X		○		–	○	X	–			–	–	–
	Core wide void and flow distribution	●	–	–	○	●	●	–	–	○	X		–		–	–		–			–	–	–
	Heat transfer in covered core	●	●	●	●	●	●	●	○	●			●		●	●		–			–	–	–
	Heat transfer in partially uncovered core	●	–	–	○	●	–	–	–	●			○		○	○		–			–	●	–
	Heat transfer SG primary side	●	○	○	●	●	○	○	–	○			○		●	○		–	X	●	–	–	–
	Heat transfer in SG secondary side	○	●	●	●	●	●	●	–	○			○		●	○		–	X		●	–	–
	Pressurizer thermohydraulics	○	–	○	○	●	●	●	○	○			–		○	–		–			–	–	●
	Surgeline hydraulics	○	–	–	○	●	●	○	–	○			–		○	○		–			–	–	●
	1-and-2-phase pump behaviour	–	–	–	○	●	–	–	○	○			–		●	●		–			–	–	–
	Structural heat and heat losses(3)	●	–	○	●	●	○	○	–	○			○		○	○		–			–	–	–
	Noncondensible gas effects	●	–	–	–	–	–	–	–	–			○		●	○		–			–	–	–
	Phase separ. in T-junct. and effect on Leakflow	–	–	–	●	●	–	–	–	○			–		○	–		●					
Test Facility System Tests	PWR	–	–	○	–	–	●	●															
	LOFT	–	–	●	●	●	●	–															
	LSTF																						
	BETHSY																						
	PKL-I	●	●	–	●	●	●	–															
	SPES																						
	LOBI-II	●	●	●	●	●	●	●															
	SEMISCALE	○	○	●	●	●	●	●															

(1) volumetric scaling

(2) secondary side

(3) problem for scaled test facilities

(4) for intermediate leaks phenomena included in large break reference matrix may be also important

TABLE 3.1 CROSS REFERENCE MATRIX OF PHENOMENA AND TEST FACILITIES FOR SMALL AND INTERMEDIATE BREACHES IN PWRs /3.3/

0 BASIC PHENOMENA	- Evaporation due to Depressurization - Evaporation due to Heat Input - Condensation due to Pressure - Condensation due to Heat Removal - Interfac. Frict. Vertic. Flow - Interfac. Frict. Horiz. Flow - Wall to Fluid Friction - Press. Drops at Geom. Discontinuities
1 CRITICAL FLOW	- Breaks - Valves - Pipes
2 PHASE SEPARATION/VERTICAL FLOW WITH AND WITHOUT MIXTURE LEVEL	- Pipes/Plena - Core - Downcomer
3 STRATIFICATION HORIZ. FLOW	- Pipes
4 PHASE SEPAR. AT BRANCHES	- Branches
5 ENTRAINM./DEENTRAINMENT	- Core - Upper Plenum - Downcomer - Steam Generator Tube - Steam Generator Mix. Chamb (PWR) - Hot Leg with ECCI (PWR)
6 LIQUID-VAPOUR MIXING WITH CONDENSATION	- Core - Downcomer - Upper Plenum - Lower Plenum - Steam Generator Mix. Chamb (PWR) - ECCI in Hot and Cold Leg (PWR)
7 CONDENSATION IN STRATIFIED CONDITIONS	- Pressurizer (PWR) - Steam Generator Primary Side (PWR) - Steam Generator Second. Side (PWR) - Horizontal Pipes
8 SPRAY EFFECTS	- Core (BWR) - Pressurizer (PWR) - Once Through Steam Gen. Sec. Side (PWR)
9 CCF/CCFL	- Upper Tie Plate / Spacers - Channel Inlet Orifices (BWR) - Hot and Cold Leg - Steam Generator Tube (PWR) - Downcomer - Surgeline (PWR)
10 GLOBAL MULTIDIMENSIONAL FLUID TEMPERATURE, VOID AND FLOW DISTRIBUTION	- Upper Plenum - Core - Downcomer - Steam Gen. Secondary Side
11 HEAT TRANSFER	
NATUR./FORCED CONV.	- Core Steam Gen. Structures
SUBC. NUCL. BOIL.	- Core Steam Gen. Structures
DNB/DRYOUT	- Core Steam Gen. Structures
POST CHF HT	- Core Steam Gen. Structures
RADIATION	- Core
CONDENSATION	- Core Steam Gen. Structures
12 QUENCH FRONT PROPAGATION/REWET	- Fuel Rods - Channel Walls and Water Rods (BWR)
13 LOWER PLENUM FLASHING	
14 GUIDE TUBE FLASHING (BWR)	
15 ONE AND TWO PHASE IMPELLER-PUMP BEHAVIOUR	
16 ONE AND TWO PHASE JET-PUMP BEHAVIOUR (BWR)	
17 SEPARATOR BEHAVIOUR	
18 STEAM DRYER BEHAVIOUR	
19 ACCUMULATOR BEHAVIOUR (PWR)	
20 LOOP SEAL FILLING AND CLEARANCE (PWR)	
21 ECC BYPASS/DOWNCOMER PENETRATION	
22 PARALLEL CHANNEL EFFECTS INSTABILITIES (BWR)	
23 BORON MIXING AND TRANSPORT	
24 NONCONDENSING GAS EFFECT (PWR)	

TABLE 3.2 CROSS REFERENCE MATRIX OF PHENOMENA AND SEPARATE EFFECTS TEST FACILITIES /3.1/

FACILITY NAME	LOCATION	MAIN FOCUS OF SIMULATION
MOBY-DICK	(F)	Blowdown, critical flow
S. MOBY-DICK	(F)	
Horiz. CANON	(F)	
Vert. CANON	(F)	
Super CANON	(F)	
Tapioca	(F)	
REBECA	(F)	
OMEGA	(F)	
Safety Valve	(F)	
FOG	(I)	
JF	(I)	
H. Pressure Rig	(UK)	
Blowdown Rig	(UK)	
Valve Blowdown	(UK)	
MARVIKEN	(S)	
HDR	(FRG)	
Battelle	(FRG)	
Pressuriz. Valve	(FRG)	
Steam/Water	(FRG)	
Subcool. Water	(FRG)	
BCL	(US)	
WL, MSSTF, CE	(US)	
MISC	(US)	
UCSB	(US)	
UCB	(US)	
EW PRTF	(US)	
TPFL	(US)	
1:10 PWR Refill	(UK)	Bypass, downcomer penetration
UPTF	(FRG)	Global multidimensional effects in upper-plenum and downcomer, ECC bypass, downcomer penetration, mixing and condensation, CCFL
Karlstein	(FRG)n	Upper tie plate CCFL
SCTF	(J)	Upper tie plate CCFL, reflood
HTL	(US)	Multidimensional effects in core
SCTF/UCLA	(US)	Multidimensional effects in core
PHSE	(US)	Multidimensional effects in downcomer
ESTMTF	(US)	Mixing in downcomer and cold legs

TABLE 3.3.1 SEPARATE EFFECTS TEST-FACILITIES USED TO SIMULATE LOCAL PHENOMENON IN PWR-LOCAs /3.1/

FACILITY NAME	LOCATION	MAIN FOCUS OF SIMULATION
CREARE	(US)	Downcomer penetration, CCF
BCL	(US)	Downcomer penetration, CCF
Pericles	(F)	Separation, entrainment, CCF
SEROPS		Separation, entrainment, CCF
CCFL	(I)	CCFL
ESTA-KP	(J)	Condensation, entrainment, CCFL
ECTHOR	(F)	Stratification, CCF
Horiz. CCFL	(UK)	CCF
Hot Leg CCF	(UK)	CCF
TPTF	(J)	Separation, entrainment
COSI	(F)	Condensation, mixing
T. Branch	(J)	Separation, entrainment
UP-BBR	(FRG)	Condensation, entrainment
KFK-Branch	(FRG)	Separation, entrainment
TUBE, PLENUM DART	(US)	Separation, CCFL
TUBE, CHANNEL DART	(US)	Entrainment
ESTF	(US)	Hot-Leg flow regions, separation
MIT	(US)	Condensation, pressurizer response
HOUSTON	(US)	CCF
UCB	(US)	Entrainment, CCF
OMEGA	(F)	Blowdown, DNB, post-CHF, reflood, heat-transfer
ERSEC	(F)	
Dadine	(F)	
Pericles	(F)	
120 Bar Loop	(S)	
JETI-4	(I)	
TPTF	(J)	
FLECHT	(US)	
ACHILLES	(UK)	Post-CHF, reflood, heat-transfer
THETIS	(UK)	
REFLEX	(UK)	
Multipin Cluster	(UK)	

TABLE 3.3.1 (CONT.) SEPARATE EFFECTS TEST-FACILITIES USED TO SIMULATE LOCAL PHENOMENON IN PWR-LOCAs /3.1/

FACILITY NAME	LOCATION	MAIN FOCUS OF SIMULATION
REWET	(S)	
SRTF	(J)	
HICOF		Post-CHF, reflood,
FLECHT	(US)	heat-transfer
THL	(US)	
THTF	(US)	
NEPTUN	(CH)	
BDHT	(FRG)	
Heat Transfer	(FRG)	
Trans. Boiling	(FRG)	
Rewet	(FRG)	
SUNYAB	(US)	
CHF	(I)	CHF
DNB	(FRG)	DNB
Post Dryout	(UK)	Post-CHF
UCHT	(US)	Post CHF
CRBTF	(US)	Heat transfer, DNB
ILTF	(US)	Heat transfer, DNB
UCSB	(US)	Heat transfer, condensation
UCB	(US)	Heat transfer, reflux condensation
FLNCF	(US)	Heat transfer, natural convection
UTSGTLTF	(US)	Heat transfer, natural convection
THTF	(US)	Heat transfer, radiation
SSCALE	(US)	Heat transfer
ANL	(US)	Heat transfer, nucleate boiling
NCBTF	(US)	Heat transfer, forced and natural convection
G2	(US)	Heat transfer
THEF	(US)	Heat transfer
UCLA	(US)	Heat transfer, natural convection

TABLE 3.3.1 (CONT.) SEPARATE EFFECTS TEST-FACILITIES USED TO SIMULATE LOCAL PHENOMENON IN PWR-LOCAs /3.1/

FACILITY NAME	LOCATION	MAIN FOCUS OF SIMULATION
GEN 3x3 GEST PATRICIA MB-2	(I) (I) (F) (US)	Steam generator behaviour
ECTHOR Loop Seal REWET	(F) (UK) (S)	Loop seal clearing and filling
Patricia ACHILLES REWET FLNCF UTSGTLTF	(F) (UK) (S) (US) (US)	Non-condensible gas effects
EPOPEE EVA Pumps Pump Behaviour CE MIT INEL AWTF	(F) (F) (C) (FRG) (US) (US) (US) (US)	Impeller pump behaviour

TABLE 3.3.1 (CONT.) SEPARATE EFFECTS TEST-FACILITIES USED TO SIMULATE LOCAL PHENOMENON IN PWR-LOCAs /3.1/

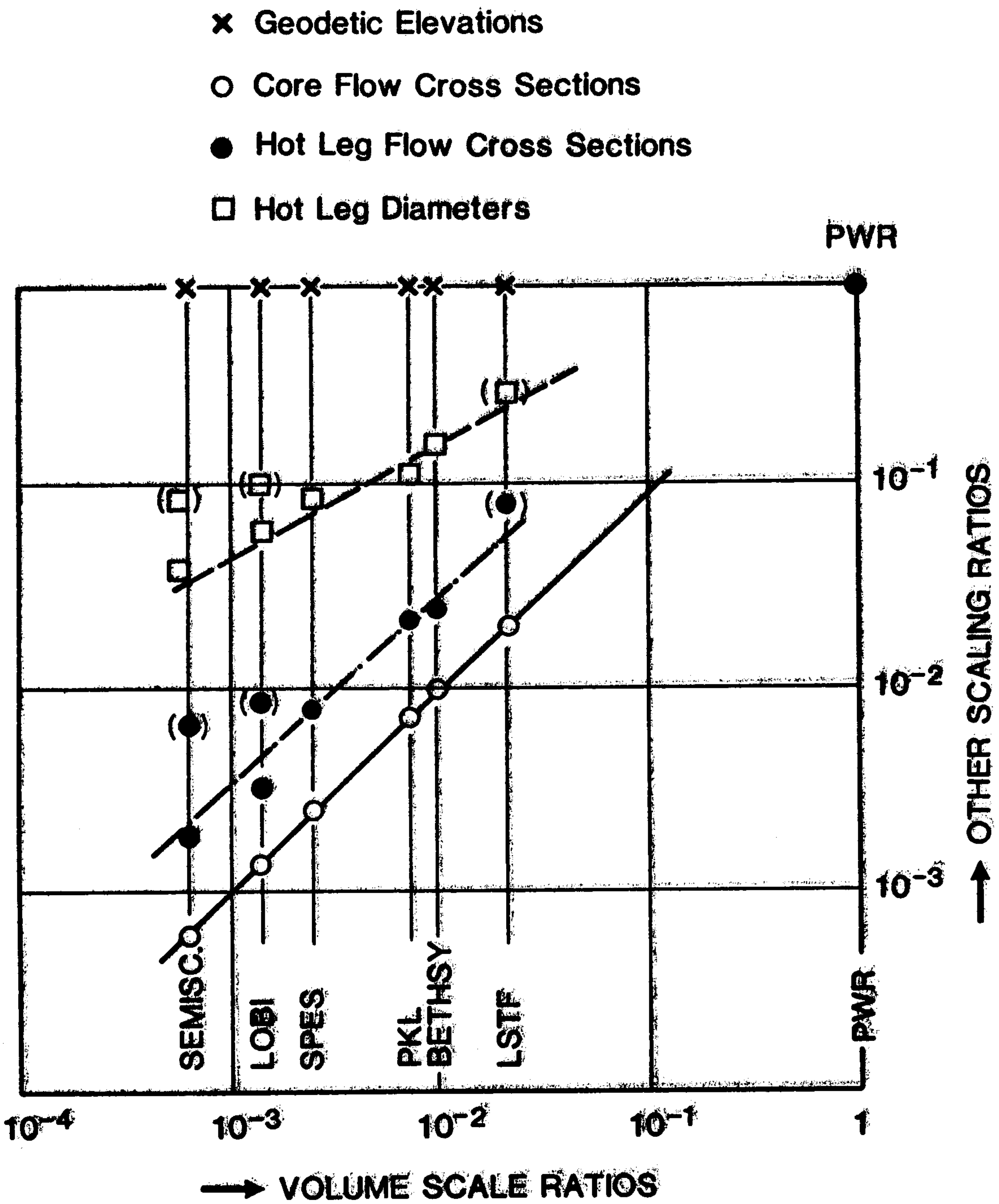


FIG. 3.2 HARDWARE SCALING OF VARIOUS INTEGRAL TEST RIGS WHICH WERE VOLUME SCALED /3.5/

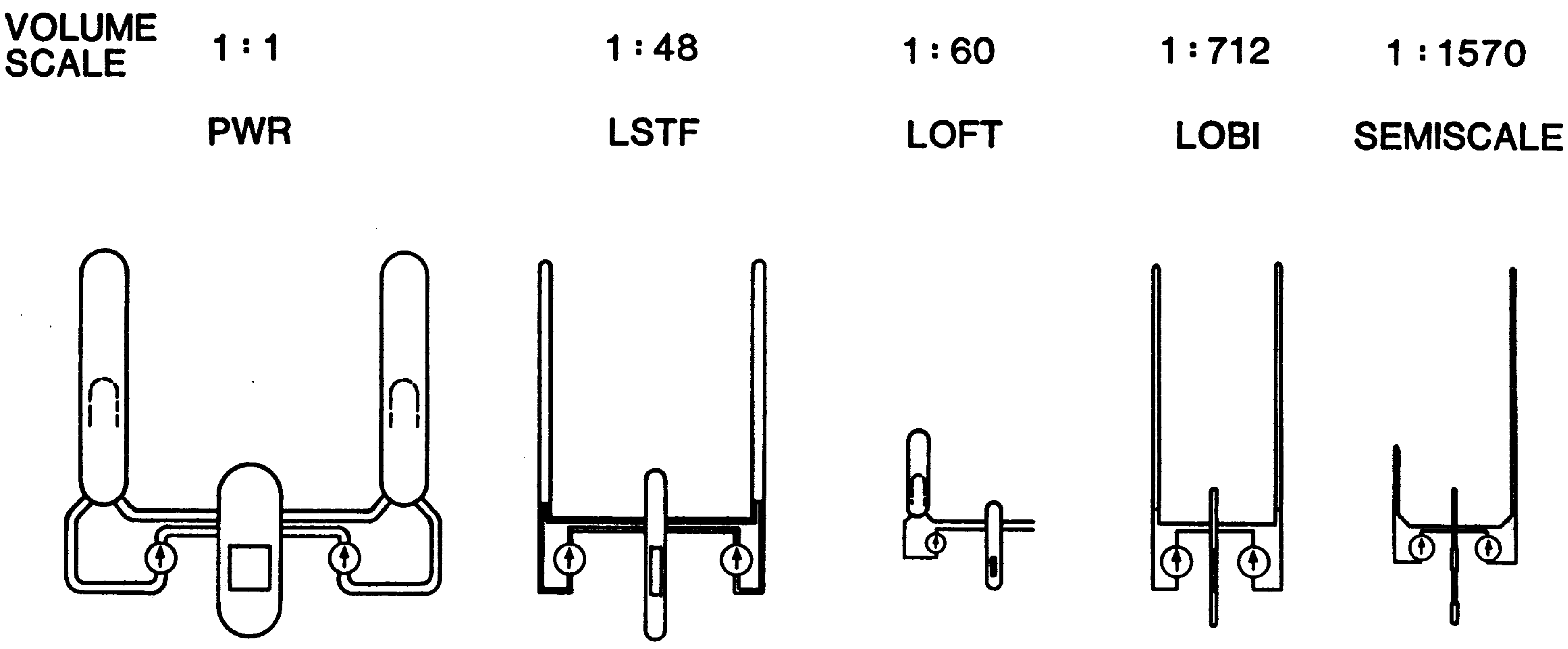


FIG. 3.1 SCHEMATIC OF SCALED CROSS–SECTIONS OF VARIOUS INTEGRAL TEST LOOPS /3.5/

FACILITY NAME	LOCATION	MAIN FOCUS OF SIMULATION
PSTF	(US)	Blowdown, transient void fraction
Battelle	(FRG)	Blowdown, critical flow
MARVIKEN	(S)	Blowdown, critical flow
ATLAS	(US)	Single bundle, critical power behaviour
CISE	(I)	Critical power, bundle heat transfer
THTF	(US)	Single bundle, interfacial heat transfer
BDHT	(US)	Single bundle, blowdown heat transfer
FRIGG	(S)	36 rod-bundle, steady state void
INEL-Test	(US)	Jet pump performance
LTSF	(US)	Jet pump behaviour
CSHT	(US)	Single bundle, core-spray heat transfer
CSHT	(J)	Single bundle, core-spray heat transfer
SHTF	(J)	Single bundle, core-spray heat transfer, CCFL
CCFL	(J)	Guide tube flashing

TABLE 3.3.2 SEPARATE-EFFECTS TEST-FACILITIES USED TO SIMULATE LOCAL PHENOMENON IN BWR-LOCAs /3.1/

FACILITY NAME	LOCATION	MAIN FOCUS OF SIMULATION
GÖTA	(S)	Single bundle, CCFL, spray-cooling, reflood
KARLSTEIN	(FRG)	Pump behaviour
CCFL	(US)	Single bundle CCFL
SIV	(S)	
SEPA	(S)	Separator behaviour
GEST	(I)	Separator behaviour
Aramis	(I)	Separator behaviour
USCB	(US)	Boron mixing in lower plenum
SNTF	(US)	Spray effects
HST	(US)	16° Sector, horizontal spray distribution
SSTF	(US)	30° Sector, spray distribution, CCFL, CCFL breakdown, refill
ESTA	(J)	18° Sector, spray distribution CCFL, CCFL breakdown
RRTF	(J)	Refill, reflood, CCFL
BWR-FLECHT	(US)	Post CHF
LEHIGH	(US)	Post CHF, spray effects

TABLE 3.3.2 (CONT.) SEPARATE-EFFECTS TEST-FACILITIES USED TO SIMULATE LOCAL PHENOMENON IN BWR-LOCAs /3.1/

FACILITY NAME	LOCATION	DESIGN BASIS	SCALING BASIS	SCALE	STATUS	EXPERIMENTAL FOCUS
LOFT	INEL (US)	W-PWR	Power/volume, half-height, full-power, full-pressure, nuclear core	1:60	Decommiss.	Complete breach, spectrum, all LOCA windows
SEMISCALE MOD-1 to 3	INEL (US)	LOFT (MOD-1) W-PWR	Power/volume, full-height in MOD-2 and 3, full-power, full-pressure	1:1570 (MOD-1) 1:1705	Decommiss. Decommiss.	Large-breach, small-breach, all LOCA windows, steam gen. tube rupture, secondary side effects, various ECC models, natural circulation
LOBI	JRC (I)	KWU-PWR	Power/volume, full-height, full-power, full-pressure	1:712	Operating	Large-breach, small-breach, all LOCA windows
PKL	Erlangen (FRG)	KWU-PWR	Power/volume, full-height, decay-power, reduced pressure	1:134	Operating	Large-breach, small-breach, refill and reflood, loop asymmetries, natural circulation
LSTF (ROSA-IV)	JAERI (J)	W-PWR	Power/volume, full-height, decay-power, full-pressure	1:48	Operating	Small-breach, all windows
CCTF	JAERI (J)	W-PWR	Power/volume, full-height, decay-power, reduced pressure	1:21.4	Operating	Large-breach, reflood

TABLE 3.4 DESIGN BASIS, SCALING AND EXPERIMENTAL FOCUS OF PWR INTEGRAL TEST FACILITES

FACILITY NAME	LOCATION	DESIGN BASIS	SCALING BASIS	SCALE	STATUS	EXPERIMENTAL FOCUS
BETHSY	CENG (F)	Framatome-PWR	Power/volume, full-height, decay-power, full-pressure	1:100	Operating	Small-breach, all windows, natural circulation, large-breach reflood, loop asymmetries
SPES	SIET (I)	W-PWR	Power/volume, full-height, full-power, full-pressure	1:427	Operating	Small-breach, all windows, loop asymmetries natural circulation
OTIS	B&W (US)	B&W-PWR Raised-Loop	Power/volume, full-height, decay-power, full-pressure	1:1686	Decommiss.	Small-breach, natural circulation
MIST	B&W (US)	B&W-PWR Lowered-Loop	Power/volume, full-height, decay-power, full-pressure	1:819	Operating	Small-breach, natural circulation, phase separation boiler-condenser mode
UMCP	B&W (US)	B&W-PWR Lowered-Loop	Power/volume, 1/3-height, core, full-height steam gen., decay-power, low-pressure	1:500	Operating	Small-breach, natural circulation, phase separation boiler-condenser mode

TABLE 3.4 (CONT.) DESIGN BASIS, SCALING AND EXPERIMENTAL FOCUS OF PWR INTEGRAL TEST FACILITES

FACILITY NAME /Description/	PRIMARY SYSTEM			LOOPS			CORE ROD GEOMETRY				
	VOLUME (m^3)	PRESSURE (MPa)	CORE FLOW AREA (m^2)	NO.	HORIZ. PIPE DIA. (mm)	SEAL DEPTH (m)	NO.	DIA. (mm)	PITCH (mm)	LENGTH (mm)	HEATING METHOD
LOFT /3.4/	7.63	15.5	0.165	1+(2)	130, (350)	1.27	1300	10.7	14.3	1680	Nuclear
SEMISCALE /3.4/	0.2	15.0	0.0028	1+(3)	34, (66)	2.7	25	10.7	14.3	3660	Indirect
LOBI /3.4/	0.82	15.5	0.0081	1+(3)	46, (73)	2.1, (2.5)	64	10.7	14.3	3660	Skin
PKL /3.4/	2.9	4.0	0.047	1+1+(2)	81, (113)	3.5	340	10.7	14.3	3900	Indirect
LSTF /3.23/ (ROSA IV)	7.2	16.0	0.1134	(2)+(2)	-, (207)	3.4	1168	9.5	12.6	3660	Indirect
CCTF /3.22/	16	0.6	0.260	1+1+1+1	155	3.4	2048	10.7	14.3	3660	Indirect
BETHSY /3.18/	2.9	17.2	0.043	1+1+1	118	2.2	428	9.5	12.6	3660	Indirect
SPES /3.17/	0.63	20.0	0.0096	1+1+1	67	2.9	97	9.5	12.6	3660	Skin
OTIS /3.4/				1		9.4					
MIST /3.4/	0.56	15.5	0.0063	2H+4C	54H, 34C	9.4	45	10.9	14.4	3660	Indirect
UMCP /3.4/	0.91	2.0	0.030	2H+4C	89H, 76C	9.4	16	25.4	ca. 80	1245	Indirect
PWR (typical)	350	16.0	4.75	1+1+1+1	737	3.4	51000	9.5	12.6	3660	Nuclear

Notes (-) refers to combined intact loop geometry, H = hot-leg, C = cold leg

TABLE 3.5 TYPICAL HARDWARE CHARACTERISTICS OF PWR INTEGRAL TEST FACILITIES

FACILITY NAME	PRIMARY SYSTEM PUMPS				STEAM GENERATOR CHARACTERISTICS						
	NO.	FLUID	SPEC. SPEED DIN-24260	COASTDOWN	NO.	TUBES	PRESSURE (MPa)	INSIDE DIAMETER (mm)	OUTSIDE DIAMETER (mm)	PITCH (mm)	HYDRAULIC DIAMETER (mm)
LOFT	2	1-PHASE	Resist., (64)	Inertia	1	-, (1845)	6.0	10.2	12.7	19	
SEMISCALE (MOD-2)	2	2-PHASE	16, (22)	Inertia	2	2, (6)	6.0	19.7			
LOBI	2	2-PHASE	29	Programmed	2	8, (24)	10	19.7	22	various	16.4 (24.39)
PKL	0	-	Resistance	-	3	30, (60)	5.6	19	22	26.4	
LSTF (ROSA-IV)	2	1-PHASE	74.3	Controlled	2	-, (141)	7.3	19.6	25.4	32.5	25.6
CCTF	0	-	Resistance	-	2	-, (158)	5.2	19.6	25.4	32.5	
BETHSY	3	2-PHASE	28.1	Controlled	3	34	8.0	19.7	22	32.5	35
SPES	3	1-PHASE	93.2	Programmed	3	13	10	15.4	17.5	24.9	23
OTIS	0	-	Resistance	-	1	19	8	14.1			
MIST	4	2-PHASE	110	Programmed	2	19	8	14.1			
UMCP	0	-	Resistance	-	2	28	0.3	30.0	31.7	50.8	
PWR (typical)	4	1-PHASE	101	Inertia	4	3382	6.2	19.6	22.2	32.5	

Note (-) refers to combined intact loop geometry

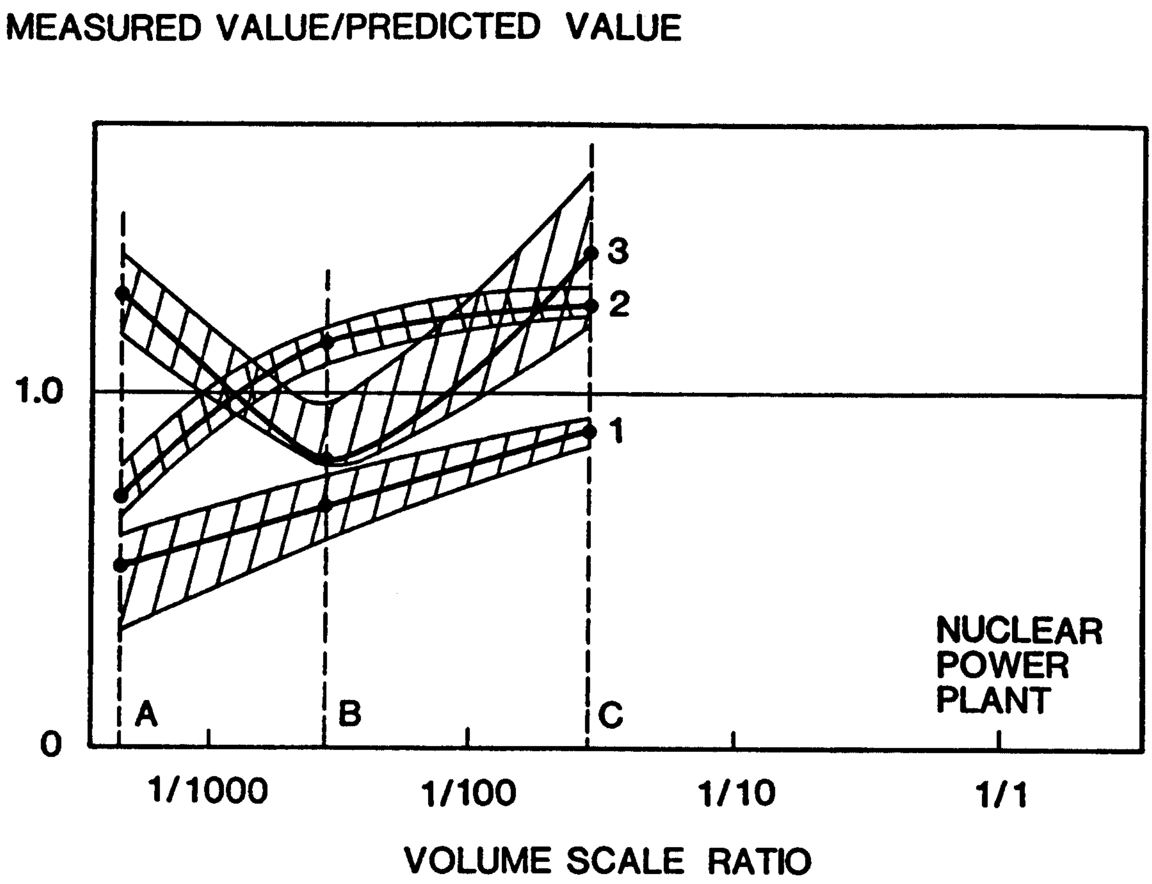
TABLE 3.5 (CONT.) TYPICAL HARDWARE CHARACTERISTICS OF PWR INTEGRAL TEST FACILITIES

FACILITY NAME	LOCATION	DESIGN BASIS	SCALING BASIS	SCALE	STATUS	EXPERIMENTAL FOCUS
TLTA	GE (US)	GE-BWR/4 GE-BWR/6	Power/volume, full-height for core only, full-power, full-pressure	1:624	Modified to FIST	Large-breach, small-breach, all LOCA windows
FIST	GE (US)	GE-BWR/6	Power/volume, full-height, full-power, full-pressure	1:624	Mothballed	Large-breach, small-breach all LOCA windows, transients
ROSA-III	JAERI (J)	GE-BWR/6	Power/volume, half-height, reduced-power, full-pressure	1:424	Decommiss.	Large-breach, small-breach all LOCA windows, natural circulation
TBL	Hitachi (J)	GE-BWR/5	Power/volume, full-height, full-power, full-pressure	1:350	Mothballed	Large-breach, small-breach all LOCA windows
FIX-II	Studsvik (S)	AA-BWR External pumps, no jet-pumps	Power/volume, full-height full-power, full-pressure	1:777	Mothballed	Large-breach, small-breach, blowdown, transients
PIPER-I	Pisa (I)	GE-BWR Jet-pumps	Power/volume, full-height, decay-power, full-pressure	1:2200	Operating	Small-breach, natural circulation

TABLE 3.6 DESIGN BASIS, SCALING AND EXPERIMENTAL FOCUS OF BWR INTEGRAL TEST FACILITES

	PRIMARY SYSTEM					CORE ROD GEOMETRY				
FACILTIY NAME /Description/	PRIMARY VOLUME (m^2)	PRESSURE (MPa)	JET PUMPS	RECIRC. LOOPS	CORE FLOW AREA (m^2)	NO.	DIA (mm)	PITCH (mm)	LENGTH (mm)	HEATING METHOD
TLTA /3.4/	0.93	7.4	2	2	0.0097	64	12.3	16.2	4140	Skin
FIST /3.4/	0.67	7.4	2	2	0.0116	64	12.3	16.2	4140	Skin
ROSA-III /3.26/	1.42	7.2	4	2	0.0392	284	12.3	16.2	1880	Indirect
TBL /3.21/	1.60	7.2	2	2	0.0232	128	12.5	16.2	3700	Indirect
FIX-II /3.20/	0.46	7.4	Ext. Pumps	2	0.0060	36	12.3	16.3	3700	Skin
PIPER-I /3.19/	0.19	7.4	1	None	0.0028	16	12.3	16.2	4300	Indirect
BWR (typical)	620	7.8	24	2	8.6	52576	12.3	16.2	3705	Nuclear

TABLE 3.7 TYPICAL HARDWARE CHARACTERISTICS OF BWR INTEGRAL TEST FACILITIES



1,2,3 : Values for the same variables in different calculations or different variables in the same calculation

A,B,C : Different facilities

ZZZZ Uncertainty bands in measurements

FIG. 3.3 SCHEMATIC OF POSSIBLE TRENDS RESULTING FROM ANALYSIS OF COUNTERPART TESTS IN DIFFERENT FACILITIES /3.13/

4. THE RELATIVE IMPORTANCE OF INDIVIDUAL LOCA PHENOMENA

Whatever the interest, whether it be additional testing to improve the LOCA data base, or best-estimate code improvement for licensing or code-assessment purposes, or code development for plant analysers and simulators, it is essential to identify each relevant LOCA phenomenon, to rate or weight it in terms of its importance for influencing the course of a LOCA and then to appraise the adequacy of its data base and analytical modelling. Such an appraisal not only focusses attention on the "right" phenomena but also provides a state-of-the-art for the judicious allocation of resources for research and development.

For the purposes of this report the weighting and appraisal process followed a similar procedure adopted by the USNRC /1.7/, being divided into three steps:

i) the LOCA phenomena identified in Chapters 2 and 3 were listed (Table 4.1.1 to 4.1.3) for both PWR- and BWR-LOCA's. Expert opinion was then used,

ii) to weigh the importance of each phenomenon on the list for its relative impact on the LOCA and/or safety parameter(s) of interest, and

iii) to appraise the physical understanding, scale-dependence, data-base and analytic-modelling of each phenomenon.

Here, in contrast to the NRC's approach /1.7/, the weighting and appraisal procedure rested solely on expert opinion. No analyses were performed to support or confirm the experts' judgement, existing studies were used where available.

Expert opinion may be a subjective, somewhat elastic and even contentious means to assess LOCA-phenomena. Nevertheless, it is a valuable qualitative instrument, representing a vast amount of experience based on firm experimental (Chapter 3) and theoretical foundations. The expert group comprised members of the OECD-NEA/CSNI-PWG 2 Task Group on Code Assessment and experts from industry and research establishments involved worldwide with LOCA experiments and analyses. Opinions of the experts reflected best-estimate plant safety and did not include consideration of country specific licensing criteria.

From the information provided in the Table 4.1 matrices, it is a straightforward task to identify:

- which dominant phenomena require improved analytical modelling or an expanded data base, Table 4.2, and

- which dominant phenomena require a more intensive evaluation of existing experimental data, Table 4.3

These tables played a key-role in the development of Chapter 5, which deals with the modelling of particular ECC phenomena. They also serve to support the discussions in Chapter 6 on code assessment.

4.1 Weighting and Appraising the LOCA Phenomena, Tables 4.1

Phenomena identified in Chapter 2 were first ordered in Tables 4.1 in terms of "where" and "when" they occur during a LOCA. The category "where" was divided into three sub-groups:

Table 4.1.1 Basic phenomena associated with several components of the reactor coolant system,

Table 4.1.2 Phenomena originating from the interaction of several components, and

Table 4.1.3 Component integral behaviour.

To define broadly "when" each phenomenon appears, the global classification schemes of Section 2.1.1 for a PWR and Section 2.2.1 for a BWR were employed:

- For pressurized water reactors (PWR) a LOCA was classified as a large-breach (L), intermediate-breach (I) or small-breach (S). Phenomenological windows blowdown (BL) and refill/reflood (R) then provided pertinent subdivisions for the large-breach class.

- For boiling water reactors (BWR) the LOCA classification was simply covered-core (CC) or uncovered-core (UC). Pertinent subdivisions for the uncovered-core class were then blowdown (BL) and refill/reflood/spray cooling (RS).

At each location "where" it occurs and for each period "when" it appears, a weighting A, B or C was given to each phenomenon to reflect its:

W. Influence on the course of a LOCA
- A: very important or dominant
- B: intermediate importance
- C: not important

Similarly, for each phenomenon, appraisals in terms of ratings A, B or C were given for:

P. Physical understanding
- A: well understood
- B: partially understood
- C: poorly understood

S. Scale dependence
- A: strong dependence
- B: intermediate dependence
- C: no dependence

D. Data base
- A: adequate and sufficient
- B: limited by quality or applicability
- C: no or inadequate

M. Analytic modelling
- A: adequate and sufficient
- B: improvement recommended, low priority
- C: improvement recommended, high priority

The rating scale A, B and C is general and is therefore used for both weighting and appraisal procedures to minimize nomenclature. Where opinions differed amongst the experts, intermediate ratings A/B and B/C were given. Finally, the notations:

- \- not occurring
- / not applicable to this plant type

were used where relevant.

4.2 Recommendations from the Weighting and Appraisal Procedure, Tables 4.2 and 4.3

Within the limitations of the above subjective weighting and appraisal procedure, a variety of criteria can be applied to Tables 4.1 to generate recommendations on research needs. To illustrate the possibilities, two examples are provided.

1. A search was made in Tables 4.1 for those phenomena, which were:

W - very important or dominant, rating A, and
M - need modelling improvements, rating B/C or C, and
D - need an improved data base, rating B or C, and
S - need tests at larger scale, rating A or B

Phenomena meeting the above criteria are listed in Table 4.2 with the recommendations for model improvements (M), for an expanded data base (D) and for better scaled experiments (S) weighted in terms of:

● high priority
O intermediate priority
\- adequate data base available.

2. Another search was made in Table 4.1 for those phenomena, which were:

W - very important or dominant, rating A, and
M - need modelling improvements, rating C, and
P - not fully understood, rating B or C

but which have:

D - an adequate data base, rating A.

Phenomena meeting these criteria are listed in Table 4.3 with the recommendation for a closer examination of the existing data base.

This weighting and appraisal procedure compresses the detailed information on the phenomena into a single category. The resulting recommendations may thus be too general for specific plant types or particular breach sizes. A certain amount of care and engineering judgement is necessary to ensure that the recommendations are correctly interpreted for such cases.

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
When it occurs / Where it occurs	L		I	S	CC	UC					
	BL	R				BL	RS				
1. CRITICAL FLOW											
– Breaks	A	B	A	B	A	A	B	A/B	B	A	B
– Valves	–	–	A	A[1]	A	A	–	B	A/B	B	B
– Surgeline	B	C	–	–	/	/	/	A/B	B	A/B	B
– Recirculation Pumps	A	–	–	–	–	C[2]	–	B	A/B	B	B
– Jet Pumps	/	/	/	/	B	A	C	A/B	B	A/B	A/B
2. PHASE SEPARATION/VERTICAL FLOW WITH AND WITHOUT MIXTURE LEVEL/GRAVITY DOMINATED PRIMARY SIDE											
– Upper Head (PWR)	B	B	B	A	A	A	C	A/B	B	A/B	A/B
– Steam Dome[3] (BWR)	B	B	B	A	A	A	C	A/B	B	A/B	A/B
– Upper Plenum	B	A	A	A	A	A	A	A/B	B	A	B
– Core	C	A	A	A	A	A	A	A/B	A/B	A	B/C
– Lower Plenum	C	A	B	C	C	B	B	A/B	B	A	A/B
– Downcomer	C	A	B	B	B	B	B	A/B	A/B	A/B	B
– Loop Seal	C	B	A	A	/	/	/	B	A/B	A/B	B
– Candy Cane (Once Through Steam Generator)	C	C	A	A	/	/	/	B	A/B	B	B
– Recirculation Line	/	/	/	/	C	C	C	A/B	B	A/B	A/B
– Steam Generator Plena	C	A	A	A	/	/	/	B	B	B	B
– Steam Generator Tubes	C	A	A	A	/	/	/	A/B	B	A	A/B
– Pressurizer	C	C	B	B	/	/	/	A	B	A/B	A
– Pressurizer Surge Line	C	C	C	C	/	/	/	A/B	B	A/B	A/B
– Downcomer	C	C	C	B	/	/	/	A/B	B	A/B	A/B
– Riser	C	C	C	B[4]	/	/	/	A/B	B	A/B	A/B

1) Only for primary side valves
2) Only for external pumps
3) Steam dome including dryer and separator
4) "A" for steam generator tube ruptures

Nomenclature: see next page

TABLE 4.1.1 BASIC PHENOMENA ASSOCIATED WITH SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
Where it occurs / When it occurs	L		I	S	CC	UC					
	BL	R				BL	RS				
3. PHASE SEPARATION/STRATIFICATION HORIZONTAL FLOW/ GRAVITY DOMINATED PRIMARY SIDE											
– Hot Leg	C	A	A	A	/	/	/	A/B	A	A/B	C
– Cold Leg	C	B	A	A	/	/	/	A/B	A	A/B	C
– Loop Seal	C	B	A	A	/	/	/	A/B	A	B	B/C
– Candy Cane (Once Through Steam Generator)	C	C	A	A	/	/	/	B	A	B	B/C
– Pressurizer Surgeline	C	C	C	C	/	/	/	B	A/B	B	B/C
– Recirculation Line	/	/	/	/	–	C	C	A/B	A	B	B/C
SECONDARY SIDE											
– Steam Line (Steam Generator Tube Rupture)	C	C	C	C	/	/	/	A/B	A/B	B	B
– Feedwater Line	/	/	/	/	C	B	C	B	B	C	B/C
4. PHASE SEPARATION/INERTIA DOMINATED PRIMARY SIDE											
– Lower Plenum (Flow Split)	B	C	C	C	–	C	B	A/B	A/B	B	B
– Candy Cane (Once Through Steam Generator)	C	C	C	C	/	/	/	A/B	A	B	B
– Steam Generator Tubes	C	B 5)	B	C	/	/	/	A/B	B	A/B	B
– Recirculation Line	/	/	/	/	C	C	C	A/B	A	B	B
PRIMARY/SECONDARY SIDE											
– Branches	C	C	A	A	C	C	C	A/B	A	B	B/C

5) "A" for plants with combined ECC injection

Table Nomenclature

L = Large breach
I = Intermediate breach
S = Small breach
CC = Covered core
UC = Uncovered core
BL = Blowdown
R = Refill/reflood
RS = Refill/reflood/spray

Appraisal

P = Physical understanding
S = Scale dependence
D = Data base
M = Analytic modelling

Ratings

A = well, strong, adequate
B = partial, intermediate, limited
C = poor, no, inadequate
– = not occurring
/ = not applicable

TABLE 4.1.1 (CONT.) BASIC PHENOMENA ASSOCIATED WITH SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
When it occurs / Where it occurs	L		I	S	CC	UC					
	BL	R				BL	RS				
5. ENTRAINMENT/DEENTRAINMENT											
– Core	C	A	A	A	C	A	A	B	A	A/B	B/C
– Upper Plenum	B	A [8]	A [8]	A [8]	C	B	B	B	A	A/B	B/C
– Hot Leg	B	B	B	B	/	/	/	B	A	A/B	B/C
– Cold Leg	C	C	C	C	/	/	/	B	A	A/B	B/C
– Steam Generator Plena	C	A	A	B	/	/	/	B	A	B	B/C
– Steam Generator Tubes	C	A	A	B	/	/	/	B	A	A/B	B
– Downcomer	C	B	B	C	C	C	C	B	A	A/B	C
– Jet Pump	/	/	/	/	–	C [6]	C [6]	B	A/B	B	B
6. LIQUID-VAPOUR MIXING WITH CONDENSATION											
PRIMARY SIDE											
– Upper Head/Steam Dome	C [7]	–	C [7]	B [7]	C	C	C	B	A/B	B	B
– Upper Plenum	C [7]	C [7]	C [7]	C [7]	A	A	A	B	A	A/B	C
– Core	C	C [7]	C	C	C	A	A	B	A/B	B	B/C
– Lower Plenum	B	A	A	C	–	C	B	B	A/B	A/B	B
– Downcomer	B	A	A	B	B	B	B	B	A	A/B	C
– Pressurizer	–	– [8]	B [8]	B [9]	/	/	/	B	A	B	B
– Steam Generator Plena	C	C	C	C	/	/	/	B	A	B	B
– Hot Leg without ECC Injection	C	C	C	C	/	/	/	A/B	A	A/B	A/B
– Cold Leg without ECC Injection	C	C	C	C	/	/	/	A/B	A	A/B	A/B
– Hot Leg with ECC Injection	A	A	A	A	/	/	/	B/C	A	B	C
– Cold Leg with ECC Injection	A	A	A	A	/	/	/	B/C	A	B	C
– Near ECC Injection Port	A	A	A	A	B	B	B	B/C	A	B	C
SECONDARY SIDE											
– Downcomer	C	C	B	B	/	/	/	B	A	B	B
– Riser	C	C	B	B	/	/	/	B	A	B	B

6) "A" for plants with limited ECCS
7) "A" for plants with combined injection, upper plenum or upper-head injection
8) "A" for plants with combined injection
9) "B" for plants with combined injection

Table Nomenclature

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TABLE 4.1.1 (CONT.) BASIC PHENOMENA ASSOCIATED WITH SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
When it occurs / Where it occurs	L BL	L R	I	S	CC	UC BL	UC RS				
7. CCF/CCFL											
– Upper Tie Plate	C	A	A	B	C	A	A	B	A	A/B	B/C
– Channel Inlet Orifice	/	/	/	/	–	A	A	B	A	A/B	B
– Hot Leg without ECC–Injection	C	B	A	A	/	/	/	A/B	A	A/B	B/C
– Cold Leg without ECC–Injection	C	C	B	B	/	/	/	A/B	A	A/B	B
– Hot Leg with ECC–Injection	C	A	A	A	/	/	/	B	A	B	C
– Cold Leg with ECC–Injection	C	B	B	B	/	/	/	B	A	B	B
– Steam Generator Tube	C	A	A	A	/	/	/	A/B	A/B	A/B	B
– Downcomer	A	A	A	B	C	C	C	B	A	A/B	C
– Jet Pump	/	/	/	/	C	C	C	B	A/B	B	B
– Surgeline	C	C	B	A	/	/	/	A/B	B	A/B	B/C
– Core	–	B	B	B	C	C	B	B	A/B	B	B/C
8. GLOBAL MULTIDIMENSIONAL TEMPERATURE, VOID AND FLOW DISTRIBUTION											
– Upper Plenum without ECC–Inj.	B	C	C	C	C	B	A	B	A/B	A/B	B/C
– Upper Plenum with ECC–Inj.	A	A	A	A	/	/	/	B	A	B	C
– Core without ECC–Injection	B	A	B	B	B	A	A	A/B	A/B	B	B
– Core with Hot–Leg, Upper–Plenum or Upper–Head Injection	A	A	A	A	/	/	/	B	A	B	C
– Lower Plenum	C	C	C	C	C	B	B	A/B	A/B	B	A/B
– Downcomer	B	A	A	C	C	B	B	B	A	A/B	B/C
– Steam Generator Second. Side	C	C	C	C	/	/	/	A/B	B	B	A/B

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TABLE 4.1.1 (CONT.) BASIC PHENOMENA ASSOCIATED WITH SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
When it occurs / Where it occurs	L BL	L R	I	S	CC	UC BL	UC RS				
9. SPRAY EFFECTS											
– Core Spray	/	/	/	/	B	A	A	A/B	A	A	A/B
– Pressurizer Spray	C	C	C	B	/	/	/	A/B	B	B	A/B
– Once Through Steam Generator Secondary Side	C	C	C	B	/	/	/	B	B/C	B	B
10. HEAT TRANSFER, NATURAL FORCED CONVECTION											
– Core	A	A	A	A	A	A	A	A	B	A	A/B
– Steam Generator Primary Side	B	A	A	A	/	/	/	A	B	A	A/B
– Steam Generator Second. Side	C	C	B	B	/	/	/	A	B	A	A/B
– Structures	C	C	C	C	C	C	C	A	C	A	A/B
11. SUBCOOLED/NUCLEATE BOILING											
– Core	B	B	A	A	A	A	A	A	B	A	A/B
– Steam Generator Primary Side	C	B	B	C	/	/	/	A	B	A	A/B
– Steam Generator Second. Side	C	C	B	B	/	/	/	A	B	A/B	A/B
– Structures	C	C	C	C	C	C	C	A	B	A/B	A/B
12. DNB/DRYOUT											
– Core	A	C	A	A	A	A	A	A/B	B	A/B	B/C
– Steam Generator Primary Side	C	C	C	C	/	/	/	A/B	B	A/B	B
– Steam Generator Second. Side	C	C	C	B	/	/	/	A/B	B	B	B
13. FILM BOILING DISPERSED FLOW (Post Critical Heat Flux)											
– Core	A	A	A	A	A	A	A	B	B	B	B/C
– Steam Generator Primary Side	C	C	C	C	/	/	/	A/B	B	B	B
– Steam Generator Second. Side	C	C	C	B	/	/	/	A/B	B	B	B

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TABLE 4.1.1 (CONT.) BASIC PHENOMENA ASSOCIATED WITH SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
	L		I	S	CC	UC					
Where it occurs / When it occurs	BL	R				BL	RS				
14. RADIATION											
– Core	C	B	B	B	C	B	A	A	B	A/B	A/B
15. CONDENSATION HEAT TRANSFER											
– Steam Generator Primary Side	–	–	C	B	/	/	/	A/B	B	B	B
– Steam Generator Second. Side	–	–	C	C	/	/	/	A/B	B	B	B
16. SUDDEN REWETTING DURING BLOWDOWN											
– Fuel Rods	A	–	A	–	–	A	–	C	B	C	C
– Channel Walls	/	/	/	/	–	C	–	C	B	C	B
– Water Rods	/	/	/	/	–	C	–	C	B	C	B
17. QUENCH FRONT PROPAGATION/ REWET DURING REFILL/ REFLOOD											
– Fuel Rods	/	A	A	A	/	/	A	B	B	A	B/C
– Channel Walls	/	/	/	/	–	/	A	B	B/C	B	B
– Water Rods	/	/	/	/	–	/	A	B	B/C	B	B
– Steam Generator Primary Side	/	B	C	C	/	/	/	A/B	B	A/B	A/B
– Steam Generator Second. Side	/	C	C	B	/	/	/	A/B	B	A/B	A/B
– Structures	/	C	C	C	/	/	C	A/B	B/C	A/B	A/B
18. HOT WALL EFFECT											
– Downcomer	C	C	C	C	C	C	C	A/B	B	A/B	A/B
– Lower Plenum	C	C	C	C	C	C	C	A/B	B	A/B	A/B

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TABLE 4.1.1 (CONT.) BASIC PHENOMENA ASSOCIATED WITH SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
When it occurs	L		I	S	CC	UC					
Where it occurs	BL	R				BL	RS				
19. PRESSURE LOSSES-PWR											
– Vessel Internals	A	B	B	B	–	–	–	A/B	A/B	A/B	B
– Piping	C	C	C	C	–	–	–	A/B	A/B	A	A/B
– Steam-Generator	B	B	B	B	–	–	–	A/B	A/B	A	A/B
– Pump	A	B	A	A	–	–	–	B	A	B	B
– Pressurizer Surge Line	B	C	C	C	/	/	/	A/B	A/B	A	A/B
20. PRESSURE LOSSES-BWR											
– Pump, Core, Separators, Dryers	/	/	/	/	C	B	C	A/B	A/B	A/B	B
21. PISTON EFFECT (PWR)											
– Pressurizer	–	–	C	A	–	–	–	A/B	B	A/B	A/B
– Steam Generator Second. Side	–	C	C	B	–	–	–	A/B	B	A/B	A/B
– Steam Generator Primary Side	–	–	C	A	–	–	–	A/B	B	A/B	A/B
22. FLASHING											
– Lower Plenum	B	C	B	C	C	A	C	A	B	A/B	A/B
– Guide Tubes	/	/	/	/	C	A	C	A/B	B	A/B	A/B

Table Nomenclature

L = Large breach
I = Intermediate breach
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D = Data base
M = Analytic modelling

Ratings

A = well, strong, adequate
B = partial, intermediate, limited
C = poor, no, inadequate
– = not occurring
/ = not applicable

TABLE 4.1.1 (CONT.) BASIC PHENOMENA ASSOCIATED WITH SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
Where it occurs / When it occurs	L		I	S	CC	UC					
	BL	R				BL	RS				
1. NATURAL CIRCULATION											
PRIMARY SIDE											
– Core/Loop/Downcomer	–	–	B	A	/	/	/	A	B	A/B	A/B
– Core/Vent Valve/Downcomer	–	B	B	B	/	/	/	A/B	B	A/B	B
– Core/Separator/Recirc. Line	/	/	/	/	A	A	B	A/B	B	A/B	A/B
– Core Bypass/Bundles	/	/	/	/	B	B	C	A/B	B	A/B	A/B
– Hot and Cold Bundles	–	B	B	B	C	C	C	B	A/B	B	B
SECONDARY SIDE											
– Riser/Separator/Downcomer	C	C	B	B	/	/	/	A	A/B	B	A/B
2. INTERMITTENT TWO PHASE NATURAL CIRCULATION (ONCE-THROUGH STEAM GENERATOR)	–	C	A	A	/	/	/	B	B	B	B
3. BYPASS FLOW PATHS BEHAVIOUR											
– Upper Plenum/Downcomer	C	C	B	B	/	/	/	A	A/B	A/B	A/B
– Upper Plenum/Upper Head	C	C	C	B	/	/	/	A	A/B	B	A/B
– PWR Core Bypass	C	C	C	C	/	/	/	A	A/B	B	A/B
4. REFLUX CONDENSOR MODE (U-TUBE STEAM GENERATOR)	–	–	B	A	/	/	/	A/B	B	A/B	B
5. BOILER CONDENSOR MODE (ONCE-THROUGH STEAM GENERATOR)	–	–	B	A	/	/	/	B	B	B	B

Table Nomenclature

L = Large breach
I = Intermediate breach
S = Small breach
CC = Covered core
UC = Uncovered core
BL = Blowdown
R = Refill/reflood
RS = Refill/reflood/spray

Appraisal

P = Physical understanding
S = Scale dependence
D = Data base
M = Analytic modelling

Ratings

A = well, strong, adequate
B = partial, intermediate, limited
C = poor, no, inadequate
– = not occurring
/ = not applicable

TABLE 4.1.2 PHENOMENA ORIGINATING FROM THE INTERACTION OF SEVERAL COMPONENTS

PHENOMENON	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
When it occurs / Where it occurs	L		I	S	CC	UC					
	BL	R				BL	RS				
6. LIQUID HOLD-UP IN STEAM GENERATOR	–	–	A	A	/	/	/	A/B	A/B	B	B
7. LOOP SEAL CLEARING AND FILLING	C	C	A	A	/	/	/	A/B	A/B	B	B/C
8. STEAM BINDING	C	A	B	C	/	/	/	A/B	B	B	B/C
9. ECC BYPASS/DOWNCOMER PENETRATION	A	A	B	C	/	/	/	B	A	B	C
10. BORON MIXING AND TRANSPORT	C	C	C	B	–	–	–	B	A/B	B	B
11. NONCONDENSIBLE GAS EFFECT	–	B	B	B	/	/	/	B	B	B	B

Table Nomenclature

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CC = Covered core
UC = Uncovered core
BL = Blowdown
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Appraisal

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TABLE 4.1.2 (CONT.) PHENOMENA ORIGINATING FROM THE INTERACTION OF SEVERAL COMPONENTS

COMPONENT BEHAVIOUR	IMPORTANCE WEIGHTING W							APPRAISAL			
	PWR				BWR			P	S	D	M
Where it occurs / When it occurs	L		I	S	CC	UC					
	BL	R				BL	RS				
1. ONE AND TWO PHASE PUMP IMPELLER BEHAVIOUR	B	C	A	A	A	A	C	B	A	B	B/C
2. ONE AND TWO PHASE JET-PUMP BEHAVIOUR	/	/	/	/	A	A	C	B	A	A/B	A/B
3. BWR SEPARATOR BEHAVIOUR	/	/	/	/	C	B	C	A/B	A	B	B
4. BWR STEAM DRYER BEHAVIOUR	/	/	/	/	C	C	C	A/B	A	A/B	A/B
5. ACCUMULATOR BEHAVIOUR	B	A	A	A	/	/	/	A/B	B	A	A/B

Table Nomenclature

L = Large breach
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CC = Covered core
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Appraisal

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M = Analytic modelling

Ratings

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C = poor, no, inadequate
– = not occurring
/ = not applicable

TABLE 4.1.3 COMPONENT INTEGRAL BEHAVIOUR

PHENOMENON	COMPONENT	RECOMMENDATIONS M	D	S
1. Phase Separation/Vertical Flow with and without Mixture Level/ Gravity Dominated	- Core	0	-	-
2. Phase Separation/Stratification Horizontal Flow/Gravity Dominated	- Hot-Leg	●	-	-
	- Cold-Leg	●	-	-
	- Loop-Seal	0	0	0
	- Candy Cane	0	0	0
3. Phase Separation/Inertia Dominated	- Branches	0	0	0
4. Entrainment/Deentrainment	- Core	0	-	-
	- Upper Plenum	0	-	-
	- Hot-Leg	0	-	-
	- Steam Generator Mixing Chamber	0	0	0
	- Downcomer	0	-	-

Nomenclature

M = need Model improvements
D = need more Data
S = need tests at larger Scale

● = high priority
0 = intermediate priority
- = adequate data base

TABLE 4.2 RECOMMENDATIONS WITH RESPECT TO THE IMPROVEMENT OF ANALYTICAL MODELS AND TO THE ESTABLISHMENT OF AN ADEQUATE DATA BASE

PHENOMENON	COMPONENT	RECOMMENDATIONS M	D	S
5. Liquid-Vapour Mixing with Condensation	- Upper Plenum	●	-	-
	- Core	0	0	0
	- Downcomer	●	-	-
	- Hot-Leg with ECC Injection	●	●	●
	- Cold-Leg with ECC Injection	●	●	●
	- Near ECC Injection Point	●	●	●
6. CCF/CCFL	- Upper Tie Plate	0	-	-
	- Hot-Leg without ECC Injection	0	-	-
	- Hot-Leg with ECC Injection	●	●	●
	- Downcomer	●	-	-
	- Surgeline	0	-	-
7. Global Multidimensional Fluid temperature, Void and Flow Distribution	- Upper Plenum without ECC Injection	0	-	-
	- Upper Plenum with ECC Injection	●	●	●
	- Core with Hot-Leg Upper Plenum or Upper-Head Inj.	●	●	●
	- Downcomer	0	-	-

<u>Nomenclature</u>

M = need Model improvements
D = need more Data
S = need tests at larger Scale

● = high priority
0 = intermediate priority
- = adequate data base

TABLE 4.2 (CONT.) RECOMMENDATIONS WITH RESPECT TO THE IMPROVEMENT OF ANALYTICAL MODELS AND TO THE ESTABLISHMENT OF AN ADEQUATE DATA BASE

PHENOMENON	COMPONENT	RECOMMENDATIONS M	D	S
8. DNB/Dryout	- Core	0	-	-
9. Post CHF-Heat Transfer	- Core	0	0	-
10. Sudden Rewetting During Blowdown	- Fuel rods	●	●	●
11. Quench Front Propagation During Refill/Reflood	- Fuel rods	0	-	-
12. Single and Two Phase Impeller-Pump Behaviour		0	0	0
13. Loop Seal Filling and Clearance		0	0	0
14. Steam Binding		0	-	-
15. ECC Bypass/Downcomer Penetration		●	●	●

Nomenclature

M = need Model improvements
D = need more Data
S = need tests at larger Scale

● = high priority
0 = intermediate priority
- = adequate data base

TABLE 4.2 (CONT.) RECOMMENDATIONS WITH RESPECT TO THE IMPROVEMENT OF ANALYTICAL MODELS AND TO THE ESTABLISHMENT OF AN ADEQUATE DATA BASE

PHENOMENON	COMPONENT
1. Phase Separation/Vertical Flow with and without Mixture Level/Gravity Dominated	- Core
2. Phase Separation/Stratification Horizontal Flow/Gravity Dominated	- Hot Leg - Cold Leg
3. Entrainment/Deentrainment	- Core - Upper Plenum - Hot Leg
4. Liquid-Vapour Mixing with Condensation	- Upper Plenum - Downcomer
5. CCF/CCFL	- Upper Tie Plate - Hot Leg without ECC Injection - Downcomer - Surgeline
6. Global Multidimensional Fluid Temperature, Void and Flow Distribution	- Upper Plenum without ECC Injection - Downcomer
7. DNB/Dryout	- Core
8. Quench Front Propagation/Rewet	- Fuel Rods

TABLE 4.3 RECOMMENDATIONS FOR A MORE INTENSIVE EVALUATION OF THE EXISTING DATA BASE

5. MODELLING OF PARTICULAR ECC PHENOMENA (TABLE 4.2)

5.1 Background and Introductory Remarks

Currently a variety of codes exist for ECC analysis purposes, /5.1/, Chapter 5 of /5.2/ and /5.3/ for example. These codes have varying degrees or levels of capability for analysing two-phase flows, ranging from the homogeneous 3-equation model to the two-fluid, 6-equation approach, see Table 5.1. Codes used in the early seventies for safety analyses and licensing were based either on the homogeneous model, or on extensions to it known as separate-flow models. These extensions take into account the differences in average velocity between the phases either by using correlations relating the void fraction to the local flow quality or by means of a drift-flux concept. Sometimes, thermal non-equilibrium is considered by employing a relaxation equation. In all these mixture models the behaviour of the mixture is prescribed a priori as a function of local parameters, such as mass flux and flow quality.

Modern codes for analysing transients and LOCA's on a best-estimate or realistic basis often use a two-fluid approach with up to 6 equations for the formulation of the conservation equations. Conservation equations are written separately for each phase, and the phases are allowed to evolve on their own, governed by the exchanges of mass, momentum and energy between the phases and across the boundaries of the flow. In this approach, it is the constitutive laws or, more generally, the closure laws describing the exchange processes, which are prescribed as a function of local flow parameters.

In Chapter 4, Table 4.2, the dominant LOCA phenomena that pose modelling difficulties have been identified. The purpose of this chapter is to expand on these difficulties, to note the capabilities and limitations of current mixture and two-fluid approaches and, in particular, to focus on when and where a two-fluid description is indispensable, if an accurate and general prediction is required. Deficiencies identified for a 6-equation model will also apply to 5- or 4-equation models. For each phenomenon or physical situation, an attempt is made to answer the following questions:

- Is the framework of the two-phase mixture or two-fluid model capable of analysing the situation in principle?

- Are the codes capable of analysing the situation in practice?

- What are the difficulties involved with fitting available state-of-the-art models of particular phenomena into the logical and numerical framework of mixture or two-fluid formulations?

- Since the codes are themselves used as scaling tools (see Chapters 3 and 6), are there any scaling deficiencies in the mixture and two-fluid formulations?

- How relevant to safety is the correct modelling of the phenomena?

This last question does need some clarification to avoid misunderstanding the, at times, harsh criticism of the models appearing in later sections. Although it is generally agreed that the relatively sophisticated two-fluid formulations are essential for the correct modelling of a number of phenomena and situations arising during a LOCA, this is in no way meant to imply that mixture or two-fluid models are inadequate for ECC licensing purposes /5.4/ when applied on a conservative basis.

The approach to design-basis safety-limits and plant operating-limits is founded on engineering judgement, a vast experimental data base and very conservative assumptions. To compensate for the many modelling deficiencies, as well as any other uncertainties, large conservative biases are incorporated in the system codes and/or in the way they are applied. By imposing restrictions on plant operating flexibility, this overall conservative approach has, however, led to economic penalties on some plants.

The better, in the sense of more realistic/mechanistic and more precise, capabilities of the two-fluid models have amply confirmed the conservatism of the design-basis approach. The two-fluid models have provided the means to quantify the conservatisms in the margins and thereby, in some cases, to remove economic penalties or even to improve performance. Furthermore, they have provided a much deeper understanding of LOCA phenomenology, which is of course very relevant to safety.

It is stressed right from the beginning that nearly all the LOCA analysis codes are based on a one-dimensional, transient, formulation of the conservation equations. Exceptions do incorporate two- (CATHARE) and three-dimensional (TRAC) formulations for particular problem areas, for example, the reactor pressure vessel, but these formulations create additional difficulties with regard to suitable correlations at the closure equation level. This means that in many situations an essentially two- or three-dimensional problem can only be attacked by one-dimensional tools, an inherent weakness that introduces additional uncertainties which have to be evaluated by recourse to experiment.

Finally, the material in this chapter is a condensed version of the original Chapter 5, which has been issued as a separate report /5.5/. In this way the balance between chapters in this SOAR has been maintained without losing the extra details, which are furnished by the separate report.

5.2 Selection of Correlations, Flow Patterns, Flow Regime Maps

Two-phase flow behaviour, and history effects in particular, are very strongly coupled to the distribution of phases or flow patterns in the components of a system under study and the way the patterns and distribution change with time. To what degree these flow patterns are recognised varies between the two-phase flow models. At one end of the spectrum, in the mixture models for example, no explicit recognition of particular patterns is made. A correlation, say, for heat-transfer or pressure-drop, is provided by an experiment, which as near as is possible simulates the physical situation being modelled. It is tacitly assumed that the experiment reproduces the flow patterns and interfacial exchanges, albeit unknown, of interest. This approach is satisfactory from an engineering - as well as a safety - viewpoint provided:

- enough suitable experiments are performed, which explains the multitude of representative separate-effects tests carried out.

- sufficient conservatism is introduced to cover model and other uncertainties, but without imposing a severe economic operating penalty.

The mixture-model approach lacks generality, because model predictions are very much tied to the particular experiments on which the correlations are based and these experiments may or may not be at full-scale.

At the other end of the spectrum the appearance and disappearance of particular flow patterns is usually taken into account in the two-fluid formulation, which is its strength. Identification of a particular flow pattern is assumed to be governed by parameters, such as local heat-flux and void fraction. Once a flow pattern is identified, suitable correlations for interfacial and other exchanges are selected to close the solution of the 6 (or fewer) conservation equations. This approach retains a wider or more general applicability, because the correlations are tied to the expected phenomenology, and this at the more fundamental closure-equation level. In practice, most two-fluid codes also make use of mixture type correlations for some phenomena because the necessary correlations at the interfacial level are not yet available.

Identification of a particular flow pattern may proceed explicitly by means of flow regime maps, examples of which are given in Fig. 5.1, or implicitly through a direct choice of closure equations, or through a combination of both. Examples of the wide range of correlations used in mixture models, and the flow maps and correlations used in two-fluid models, may be found in /5.1/. These maps, and most of the closure relations, are based on steady-flow experiments in vertical or horizontal pipes having limited diameters. Questions naturally arise as to their applicability:

- in heated bundles, near bends and at pipe entrances,
- in rapidly changing flow pattern situations,
- at larger scales (typically the co-ordinates of the maps are not dimensionless),
- in truly three dimensional flow situations.

It can be difficult to answer these questions, because the complexity of code assessment (Chapter 6), and the common practice of parameter fitting, more often than not prevent a separate evaluation of such effects, see Part 2 of /5.1/ for example.

Another important point is that flow regime maps are often used independently of heat-transfer regime maps or correlations, Fig. 5.2 for example, whereas, in reality, they are strongly connected. A complete and consistent relationship between flow patterns or regimes and the closure equations is lacking at the present time. In particular, uncertainties exist in the numerous transition regions between the basic flow regimes. The use of abrupt transition conditions can be a source of numerical instabilities.

Improvements will come from a better mechanistic understanding of flow patterns and analytic criteria that will result from mechanistic models. Details of the mechanistic processes involved and the accuracy of the models on which they are based are the objects of continued research.

Uncertainties created by model or code imperfections are determined to some extent during code assessment procedures. It must however be pointed out that comparisons /5.1/ have shown that enormous differences (order of magnitude) can appear in interface transport terms, for example, predicted by some best-estimate codes. Yet, the same codes predict very similar peak cladding temperatures, probably because of some code tuning, a balance or compensation of errors, or maybe weak coupling between heat-transfer at the wall and inter-phase transport.

Thus, although much experimental work needs to be done to tie closure equations more closely and consistently to two-phase flow patterns or regimes to improve best-estimate modelling, the direct benefits of accurate prediction of parameters to licensing and safety may in the end be small. This statement applies to many of the models discussed below. A broader discussion of code assessment issues is provided in Chapter 6.

5.3 Phase Separation

Phase separation with or without a specific mixture or fluid level is a common phenomenon in LOCA scenarios. Several mechanisms such as gravity, inertia or drag may control or dominate the separation process:

a) When inertial effects are minimal, which means when flow velocities and accelerations are low, gravity dominates leading to separation or stratification of the liquid below the vapour in vertical or horizontal flows. The situation arises in the reactor pressure vessel, when the dynamics of the blowdown phase during a large- or intermediate-breach LOCA have ended. It also occurs in the primary system piping during a small-breach LOCA, when the main coolant pumps are tripped. How rapidly the phases actually separate out will depend on vapour bubble-size, geometry and drag (form, interfacial, wall), which control the velocity of rising bubbles in the liquid and the stability of the separated flow interface.

b) In accelerating flows, particularly at pipe bends or branches, the much greater inertia of the liquid may cause it to separate out from the vapour phase. Under these circumstances inertial acceleration takes the place of gravity, the separation process being opposed by drag.

c) De-entrainment is a term for phase-separation reserved for the removal of liquid droplets from a two-phase flow, either by gravity, by inertial acceleration or by impingement on structures situated within the flow field. De-entrainment influences, for example, precursor cooling during core reflood, ECC distribution above the core and conditions in the steam-generator plenum. Interfacial drag plays a key role in determining the rate of droplet de-entrainment.

Mixture models are in principle not capable of treating phase separation, unless they are artificially forced to do so, for example by specifying a strong discontinuity in the void-fraction/quality relationship at a certain value of the quality. Indeed, the proper description of phase separation was one of the main incentives for developing codes based on the two-fluid formulation, and it is in this area that the formulation has had much success.

5.3.1 Modelling of Gravity and Inertial Separation

Most two-fluid codes /5.1/ employ a global condition for the occurrence of vertical or horizontal stratification. Typically this may be a characteristic void fraction for vertical flow /5.5/ or a characteristic vapour velocity for horizontal flow /5.6/. When the condition is predicted, vertical/ horizontal stratification, respectively, is assumed to have occurred. Closure laws are then specified from a flow regime map for stratified-flow. Some codes /5.7/ use specific interfacial friction laws to take into account the effects of spacer-grid induced mixing and bundle-geometry on the phase-separation processes in the core. Two-fluid models automatically assign inertia to each phase separately.

The existence or non-existence of a separated-flow in a region or computational cell can strongly influence code predictions in the neighbourhood of the cell. At branches in piping, for example, the distribution of the phases in the downstream paths are simply governed by conditions predicted in the donor cell. If a separated flow exists upstream of a branch or junction, vapour or liquid pull-through may occur. Under these conditions use of the donor void fraction is no longer appropriate. Most codes have empirical correlations for specifying when pull-through occurs and the effect of this entrainment on the junction void fraction.

The use of global criteria and flow regime maps with their associated correlations for resolving gravity controlled phase separation in two-fluid models is governed more by computer-time limitations than necessity. The same applies to the modelling of inertial separation. Numerical schemes used in the two-fluid codes have the natural tendency to modify discontinuities, including void fraction discontinuities. Moreover, properties such as void fractions are not defined as point quantities, but rather as volume or node averages. Nevertheless, by using a sufficiently fine two- or three-dimensional noding, stable predictions representative of the detailed separation phenomena are basically possible. In practice, because this is very costly in computer time, the fine node approach is used solely when it is indispensible, for example, for the definition of vertical mixture levels. Some codes, for example ATHLETE and TRAC-BF, have special models with a course noding to describe mixture level tracking. Two- or three-dimensional fine-node numerical solutions in particular geometries, such as the hot- and cold-legs in the primary system of a PWR, or in flow path branches, may need to be researched to provide either an efficient direct modelling capability or to investigate limitations of the flow regime map approach.

5.3.2 Modelling of De-entrainment

To model fully the de-entrainment of droplets with a two-fluid code would require a multi-field droplet model with groups of droplets having different diameters within each group. The complexity introduced by such a multi-field approach may not be justified in practical situations. In any case, the additional closure relations are not available. A typical example of the difficulties encountered with modelling de-entrainment is provided in Section 5.9.3, where dispersed-flow film-boiling is discussed.

5.4 Entrainment

Entrainment of a fluid into a flowing vapour phase, either as droplets or in bulk, plays an important role in ECC. It influences, for example: precursor cooling prior to quenching, liquid carryover during core reflood, ECC bypass above the core during upper-plenum injection, the stability of the separated flows in the hot-legs, liquid pull-through and downcomer penetration.

Entrainment is intimitely connected with the relative velocity of the phases and the interfacial drag. Thus, mixture models can handle entrainment only fully empirically. Two-fluid codes are better placed, but typically employ an entrainment correlation /5.8/ based on a critical vapour velocity and a Reynolds number for the liquid phase /5.9/. In other words this type of modelling provides an average entrainment effect based on correlations at the interface level, not on a mechanistic model of entrainment. The development of mechanistic models involves inherent difficulties, not least those associated with the essentially three-dimensional characteristics of the entrainment process and the required, corresponding, closure relations.

5.5 Liquid-Vapour Mixing with Condensation

Injection of subcooled ECC into a relatively hot primary system, where it comes into intimate contact with steam or a saturated two-phase mixture, can lead to the occurrence of violent pressure oscillations as a result of the very rapid condensation of steam. Such oscillations have been observed experimentally in the neighbourhood of ECC injection points, for example.

Mixture models typically incorporate a condensation relaxation time to smooth out these oscillations or to avoid them altogether. This artificially delays the attainment of thermal equilibrium or, in effect, artificially reduces condensation heat and mass transfer. Mixture type codes thus include some average mixing and condensation effects without considering any physics or mechanisms of the processes involved.

Two-fluid models go a step further when handling mixing and condensation in that inter-phase heat- and mass-transfer rates are empirically correlated to the flow-regimes assumed to occur /5.1/. An interfacial area then needs to be specified, which is straightforward for a well-defined, gravity-separated flow condition, but rarely so otherwise.

Several fundamental problems are encountered in the two-fluid approach:

- Some of the complex flow patterns occurring during ECC injection in the cold- or hot-legs, above the core or in the downcomer, bear no real relationship or similarity to the assumed flow-regimes from which closure relationships are drawn. Thus, in nearly all codes, mixing and condensation models must be tied to tests which are very much representative of the actual problem of interest.

- Any condensation model is as accurate as the interfacial area it employs. In a three-dimensional, highly unsteady, two-phase mixing situation, interfacial area can change very rapidly by orders of magnitude. Two-fluid codes employ an average interfacial area over a chosen numerical grid which, in practice, may be orders of magnitude larger than the characteristic mixing dimensions of the flow. It is not surprising then that predictions of mixing and condensation are often very dependent on the discretization chosen for a problem. For any particular problem the discretization may have to differ significantly from the commonly used volume/junction geometry of the codes in order to obtain reasonable correlation with test data.

- Two-fluid codes employ very high heat transfer coefficients for interfacial heat exchange during condensation. Thus, violent condensation-induced pressure oscillations can also be produced by the codes. At the same time all the codes have an inherent tendency to unstable behaviour. Artificial diffusion is used to overcome the subsequent numerical instabilities. Unfortunately this also tends to damp amplitudes predicted for real physical oscillations. The instability picture is particularly sensitive to non-physical steam generation and accompanying velocity spikes /5.10/ or by pressure spikes from artificial water packing /5.11/. In summary, when large oscillations in parameters are predicted, it is often very difficult to decide whether they are truly physical or numerical. Whether large oscillations occur may also depend on the numerical grid size chosen for the problem.

- Two-phase fluid models, which have only one average liquid temperature per node, cannot properly prescribe the heat exchanged between the phases in a stratified-flow situation when, for example, large transverse temperature gradients are present in the liquid phase.

None of the ECC codes has a satisfactory model for mixing and condensation in terms of the physics involved. Orders of magnitude differences in condensation rates are found between codes /5.1/, even for a relatively simple, horizontally-stratified, flow situation. Progress in this area will only be made once better mechanistic models for the mixing process are developed. Even on a global scale, the inherently three-dimensional nature of the mixing phenomena of interest during a LOCA makes this a daunting task, see Section 5.7.

5.6 Counter-Current Flow and Counter-Current Flow Limiting (CCFL) Condition

Counter-current flows play a very important role in the distribution of coolant in and around the core and primary system during a LOCA. Such flows are found in PWR's at the upper tie-plate of the core during hot-leg injection, during hot-leg ECC bypass and cold-leg ECC penetration of the downcomer, in the steam generator tubes, in the steam generator plenum and in the pressurizer surge line. In BWR's, they limit the draining of coolant from the upper plenum into the core at the upper tie-plate and the draining of coolant from the core to the lower plenum at the core inlet orifices.

In counter-current flow a limiting condition appears first when the downward flow of liquid is influenced by the upwards flow of vapour. Any increase in vapour flux then leads to a reduction in the liquid counter-flow until, eventually, this becomes zero. CCFL tends to occur at flow restrictions, such as orifices, where the relative velocity between the phases and hence the interfacial drag is a maximum. Mixture models are clearly not capable of modelling the mechanical and thermal non-equilibrium conditions found in the counter-current flow of the phases. In the early mixture-based ECC analyses, CCFL was not modelled. Instead, a conservative criterion for the effectiveness of coolant injected into the plenum above a core was used. At that time, CCFL at the inlet orifices of the bundles was given no credit for holding up coolant in the core itself, which has proved to be extraordinarily conservative for most jet-pump BWR's.

Two-fluid models in ECC codes /5.1/ use the void-quality relationship from a drift-flux model to obtain CCFL correlations of the Wallis/Kutateladze type. These CCFL correlations /5.12/ have been very successful in a wide range of experiments and under a wide range of conditions, although the values of the constants used are geometry specific. By tying the drift-velocity to the flow-regime parameters predicted by the two-fluid formulation, a coupling is achieved between this formulation and the well-established drift-flux concept.

Breakdown of CCFL occurs, when subcooled liquid condenses sufficient counter-flowing steam. Since condensation is modelled directly in the two-fluid formulation, CCFL breakdown can in principle be modelled; steam condensation reduces the vapour-flux, allowing a greater liquid flux, which increases condensation. This process is unstable so that counter-current flow rapidly collapses altogether. Since breakdown of CCFL is a very local phenomenon, the accuracy of such an approach depends very much on the mesh size chosen for the noding in the CCFL vicinity.

To move away from the Wallis type of correlation, mechanistic models, which take into account wave instabilities and interfacial exchanges are needed, /4.12/ for example. These may not bring worthwhile improvements to the well-defined CCFL situations occurring, for example, in and above BWR cores, but may substantially improve the understanding of the much more complex CCFL situation occurring above a PWR core /5.13/, in the downcomer and in the plena of the steam generators.

5.7 Global Multidimensional Effects

Nearly all two-phase flow phenomena are inherently three-dimensional and unsteady on a fluid-element scale. This multi-dimensional noise is ignored in practical problems, and space-time averaging of the conservation equations filters it out. Only the global three-dimensional, instationary, characteristics of the flow over a scale dictated by the integration process are retained.

In general the rather restricted or tight nature of LWR primary-system geometries tends to force many LOCA phenomena to be nearly one-dimensional on a local integral scale representative of a flow channel. Many two- and three-dimensional phenomena in the flow channels have a small influence and, if not explicitly modelled, appear as uncertainties or as a spread in the one-dimensional correlations. This is the reason why one-dimensional two-fluid codes such as RELAP5 /5.14/, and even the mixture-models, do quite well with LOCA analyses, for example, for conservative licensing purposes.

On the other hand, three-dimensional phenomena are sometimes very evident, even dominating the local thermal hydraulics, in those regions of the primary system not restricted by walls or channels, such as the upper plenum in the reactor vessel, the core and the downcomer in a PWR. Mixing, condensation and bypass of liquid and vapour during ECC injection are typical examples which give rise to three-dimensional velocity, temperature and void distributions in these regions. One-dimensional mixture and one-dimensional two-fluid models can only average these three-dimensional effects into their one-dimensional framework, which penalizes the scaling and generality of their prediction capabilities. Thus, uncertainties introduced by the one-dimensional approximation have to be assessed from near full-scale experiments.

Can the three-dimensional, two-fluid, codes such as TRAC, Table 5.1, do any better? When averaging the two-fluid conservation equations the terms for the production, convection and transport of fluid are all specified in a mathematically rigorous form. However, the averaging process generates unknown distribution coefficients on the convective terms and a multitude of unknown coefficients for transport within the phases, across the phases and to the walls. Distribution coefficients are usually taken to be unity. Transport or closure relationships are more often than not assumed to be algebraic relationships involving one-dimensional parameters such as friction factors, drag coefficients and heat-transfer coefficients, which all have to be established empirically.

Notwithstanding the mathematical implication of specifying algebraic relationships for what could be derivatives of dependent variables /5.15/, the simplifications imply:

i) that none of the transport coefficients generally employed are applicable with certainty to truly multi-dimensional flows, since they have nearly always been established from measurements in one-dimensional facilities under stationary conditions. The simple algebraic approach may give good engineering answers in situations where multi-dimensional effects are small. An example here is the cross-flow modelling in the sub-channel analysis code COBRA of the dominantly axial flows in rod-bundles. However, a more rigorous treatment of the transport terms is limited by the difficulties involved with their physical interpretation and their actual measurement.

ii) that only the convective and pressure gradient terms in the equations retain any semblance to a rigorous approximation of global, instationary, three-dimensional flows.

For some LOCA phenomena, it is the pressure gradient and convective terms which have a controlling influence on three-dimensional effects. A typical example is the distribution of subcooled liquids or voids in pressure vessel plena. TRAC-BWR gives good, global, three-dimensional, results for this situation /5.16/. It must be stressed that "global" in this context means averaged over a length scale of 0,5 m or so, which is a typical node size chosen for a plenum in a large BWR vessel. Such enormous node sizes are necessary to limit demands on computer memory and computational time, but they seriously limit the three-dimensional detail which can be handled by a code.

Where transport terms do play an important role, for example, in controlling the ECC spray distribution in the condensing/evaporating environment of a BWR upper plenum /5.17/ or in the downcomer flows in PWR's during ECC injection /5.18/, recourse still has to be made to near full-scale experiments if reliable predictions are required. This was one of the driving forces for the international 2D/3D experimental programme. Until progress is made with the modelling and measurement of the three-dimensional transport coefficients, this situation is unlikely to change in the near future.

5.8 Boiling Crisis, Departure from Nucleate Boiling, Dryout

Under forced convection boiling conditions two distinct mechanisms /5.19/ and /5.20/ govern the rapid deterioration of heat-transfer from the surface of a rod-bundle to the coolant, known as a boiling crisis:

a) A thermally limited boiling crisis can occur in a nucleate boiling situation when the bubble layer adjacent to the heated surface prevents liquid reaching the surface. This crisis is referred to as departure from nucleate boiling or DNB. Several different phenomena can initiate DNB, for example:
   - at very low void fractions or high subcooling, a high heat flux prevents the heated surface under the departing bubbles from rewetting, or
   - at high mass flow rates and high heat fluxes, bubbles near the surface coalesce to form a vapour blanket.

b) A liquid-limited or liquid deficient boiling crisis can occur in certain flows with a high void fraction, when the liquid film on the heated surface dries up. This crisis is referred to as dryout. It can arise, for example:
   - at low flow rates, in the slug flow regime, when the liquid film surrounding a large steam bubble dries out, or
   - at high flow rates, in the annular flow regime, when the liquid film thickness falls to zero or dries out.

DNB is associated with the rapid reduction in core flow occurring during the first few seconds of blowdown in a PWR large-breach LOCA scenario. Dryout is associated with the rapid reduction in core flow occurring during large recirculation line breach LOCA's in BWR's, see Chapter 2. Core uncovery in a PWR or BWR may also be considered as a core-wide liquid deficient boiling crisis.

Both mixture and two-fluid LOCA codes employ fully empirical correlations for predicting the occurrence of a boiling crisis /5.1/. The correlations are based on mixture model parameters and stationary-flow measurements in full-scale test-rigs. To avoid problems with the limited range and applicability of correlations, look-up tables with large sets of data are found in some codes.

Typically for PWR's the heat-flux just prior to a boiling crisis, designated the critical heat flux (CHF), is correlated with bundle mass-flow, inlet subcooling, vessel-pressure, exit-quality and a length scale. Influences of upstream flow and heat-flux profiles, called history effects, as well as the local effects of rod spacers, are incorporated by additional empirical correction factors. In BWR's, history effects are included directly in the so-called integral formulation in which critical power is correlated with the above parameters, but with critical quality instead of exit-quality. Spacer effects are introduced as additional correction terms. The critical power or integral formulation can be shown to be a limiting case of the CHF correlation at high quality conditions /5.21/.

These correlations are used to define when a boiling crisis occurs, which then governs the stored energy available for the first heat-up of the core. For those reactors having peak cladding temperatures which are sensitive to the boiling crisis, a case can be made for model improvements. Two-fluid models are clearly the best tools available but, no CHF correlations for the two-fluid models used in the codes are currently known to the authors. Once a boiling crisis and the time of its occurrence are established, models for post-CHF or post-dryout heat-transfer are employed.

5.9 Post-CHF or Post-Dryout Heat Transfer

In the wake of a boiling crisis, low heat-transfer coefficients are apparent and result from the low thermal conductivity of the vapour at the heated surface compared to that of the liquid. Boiling heat transfer beyond the crisis is called post-CHF and this term is commonly used for both post-DNB and post-dryout situations. Generically it is also referred to as film-boiling. Before reaching the stable film-boiling regime, an intermediate and unstable regime of short length or duration, known as transition boiling, is encountered. This is characterized by the heat-transfer surface being only partly or intermittently wetted. The above regimes are illustrated by the typical boiling curve of Fig. 5.2. Depending on whether the situation is stationary or transient, the regimes on the boiling curve may be distributed along a heated length, or a series of regimes may succeed one another at a fixed location.

Because these regimes play a very important role in the thermal response of the core during a LOCA, either when following a boiling crisis prior to core uncovery, during the core uncovery itself or during reflooding of the core, much attention has been focussed on them /5.22/ to /5.26/. Here, the main focus will be on transition and film boiling during core reflooding from below. Coolant injected above the core from spray-cooling, from hot-leg or from upper-plenum injection can lead to top-down quenching and increased reflooding rates. These effects have to be analysed in parallel with the reflooding model, but their direct influence on the reflooding process itself is small, see also Section 5.10.

5.9.1 Transition Boiling

Various physical mechanisms involving different flow regimes are met during transition boiling:

- in subcooled or low-quality flows a bubbly or intermittent flow regime (inverse-slug) appears subsequent to DNB. Nucleate boiling and film-boiling alternate on the hot-surface.
- in high-quality saturated flows the hot-surface dries out and is periodically re-wetted either by liquid slugs or by small rivulets of fluid left on the wall downstream of the dryout point. These meander over the hot surface and lead to intermittent wetting.

Transition boiling is modelled in safety analysis codes /5.1/ either directly, as a correlation, or by means of a superposition which combines contributions from nucleate boiling and film boiling. In two-fluid codes the heat-flux predicted from the global correlation is then redistributed to the liquid and vapour phases. None of the correlations do too well at fitting the data /5.27/. The superposition method does have the advantage of providing a smooth transition between regimes /5.23/. It requires, however, that the critical heat flux and minimum film boiling end-points of the superposition procedure be correctly defined. It is quite possible that in some flow boiling situations these end points may not be unique. This is a controversial issue at present, aggravated by the fact that upstream effects /5.28/ to /5.30/, such as enhanced turbulence in the flow, can have a strong influence on the end points.

Iloeje's /5.31/ mechanistic model of transition and film boiling phenomena has laid the foundations for model improvements in this area. He considers explicitly: heat transfer to droplets hitting the wall, convective heat transfer to steam and the disturbance of the steam boundary layer by droplets approaching the wall. Modelling of these complex interactions provides various components of heat-flux which may be added to obtain the total heat flux. This approach has reproduced the trends in transition boiling. It may lend itself to the modelling capabilities of the two-fluid equations but, since transition boiling is a short-lived regime, most emphasis has anyway been placed on film boiling.

5.9.2 Inverted-Annular Film Boiling

Under reflooding conditions, low-quality flow is usually associated with the inverted annular film boiling regime. It exists only as long as the flow at the quench front remains essentially subcooled /5.32/. Otherwise, sufficient voids are created to change the regime to dispersed flow, see Section 5.9.3.

The inverted annular regime consists of a long column of liquid separated from the wall by a vapour film. Vapour bubbles may exist in the liquid core. Rapid steam generation accelerates the vapour in the film more easily than the denser core and produces a high steam velocity. At a certain value of velocity, the liquid core becomes unstable and breaks up into large fragments, the flow regime then changes to dispersed flow film boiling.

Experimental observations show that the heat transfer coefficient in the inverted annular regime increases rapidly with liquid subcooling. Higher subcooling promotes heat transfer to the liquid in the core and reduces vapour generation, which reduces the film thickness and thus enhances heat transfer. At high flow rates a strong increase in the heat transfer coefficient with mass flux is observed. At low flow rates this effect may disappear.

Neither the classical Bromley-type nor the modified Bromley /5.33/ analyses have satisfactorily reproduced the experimental observations. Modelling of the inverted annular regime depends critically upon the heat-transfer between the superheated vapour and the subcooled liquid core. Although mixture type correlations are typically used in ECC analyses /5.1/, two-fluid models are necessary for a correct description, /5.34/ to /5.36/, but then difficulties arise when trying to select the interfacial and momentum exchange correlations. Turbulence in the film, sudden vaporization at the quench front and liquid contacts with the wall, for example, are mechanisms influencing the interfacial exchanges. Vapour generation at the quench front is a particularly complex phenomenon, taking place violently under highly non-equilibrium conditions. In general, there are too many adjustable parameters and too many difficulties in measuring flow parameters to validate existing models properly.

Although the two-fluid type of safety analysis codes have the necessary framework for modelling this regime, there are also limitations regarding the size of the nodes that can be used; the entire length of the inverted annular film boiling may be 10 to 30 cm, which is comparable to, or smaller than, the node size commonly used for core analysis.

5.9.3 Dispersed-Flow Film Boiling

At high void fractions post-dryout heat transfer takes place with the liquid phase distributed as droplets entrained in the vapour phase. This is the dispersed-flow film boiling regime, which can be encountered during the blow-down phase of a LOCA as well as in the form of precursory cooling during the reflood phase.

The dispersed flow regime can exist with a variety of droplet sizes. Energy is convected from the wall to the superheated vapour, from the vapour to the droplets and directly to the droplets. The first mechanism is the most important contributor with respect to heat transfer from the wall. Radiation heat transfer at high surface temperatures can also be an important transport mechanism.

Experimental work shows that significant phase velocity differences and departures from thermal equilibrium are to be expected in dispersed flows. Mixture models cannot handle these mechanisms directly, whereas the two-fluid models can, at least in principle. In practice, difficulties are encountered:

- Heat transfer from the superheated vapour to the droplets can be considered as being proportional to the driving temperature difference and to the interfacial area. This area is controlled by interfacial shear, droplet generation, break-up, coagulation and evaporation histories. It is the most difficult parameter to model, but it holds the key to correct prediction of wall temperatures /5.37/.

- Interfacial area transport is not modelled in the two-fluid ECC codes, a local-conditions model is employed instead. Typically, current ECC models employ a Weber number to provide an average droplet diameter, which is then used to calculate an interfacial area per unit volume for interface transport purposes. Correlation by a Weber number may occasionally result in non-physical predictions, for example, increasing droplet diameters above a dryout point.

- The spectrum of droplet sizes may vary from situation to situation, making it difficult to formulate a universally valid closure law.

- It is difficult to measure important parameters such as interfacial drag coefficients.

- Direct wall-to-droplet heat transfer is modelled empirically.

- The presence of spacer grids modifies the droplet population spectrum. They may also act as cooling fins. De-entrainment at spacers, or the influence of spacers or other obstacles on downstream droplet distributions, is either ignored or modelled empirically.

5.10 Rewetting Phenomena and Quench Front Propagation

Rewetting is the mechanism for re-establishing nucleate boiling or annular-flow boiling, together with the large heat-transfer coefficients normally associated with them. This transition from a post-boiling-crisis flow regime to a regime in which the core is again adequately cooled may occur during a LOCA:

a) in the blowdown phase, if coolant in the reactor vessel, or in the primary system, is forced through the core by the depressurization process. One example of this is lower plenum flashing during a large recirculation line breach LOCA in a BWR. Another example is the early rewetting of a PWR core, as coolant from the intact loops is swept into the core by the primary system's hydraulic behaviour during blowdown. This highly dynamic or sudden rewetting by a large mass of coolant may not involve progression of a quench front and may be followed soon after by another boiling crisis.

b) as a result of spray cooling above the core, or coolant injection into the hot-leg or upper plenum. This is known as top-down quenching. It is usually characterized by a quench front progressing slowly downwards from the top of the core, preceded by droplets falling through the core, Figs. 5.3.a and 2.2.10.

c) as a result of reflooding the core from below. This is known as bottom-up quenching. Here a quench front propagates slowly upwards from the bottom of the core. In inverted-annular flow it is preceded by a central column of liquid, which is itself prevented from wetting the surface by a vapour film, Figs. 5.3.b and 2.1.4. In dispersed flow there is precursory cooling of the hot surface by droplets entrained with the steam.

5.10.1 Sudden Rewetting During Blowdown

It is not at all clear what mechanisms are involved with sudden rewetting or quenching during blowdown. In BWR's, lower plenum flashing was early recognized as a real effect which quenches the core. But, for PWR's, sudden rewetting was an unexpected occurrence first observed in a LOFT large-breach LOCA simulation. A great deal of controversy ensued, and still remains, as to whether the rewetting was real or just a false indication or effect created by the external thermocouples on the LOFT fuel rods /5.38/. Subsequent experiments in the LOFT test support facility suggest that rapid quenching of the fuel-rods did indeed occur.

Mechanistic models of sudden rewetting are not yet available. Safety codes currently apply an empirical heat transfer coefficient, based on transition to nucleate-boiling, as soon as a low-quality, high-velocity, upward core flow is predicted to occur. Code calculations /5.39/ of the LOFT experiment clearly indicate that this is not a very satisfactory procedure.

Since a rapid-quench in a PWR is a very important initial condition for subsequent core-uncovery cladding temperature excursions, a reliable prediction of whether it occurs or not and, if it does occur, the mechanistic processes and heat-transfer coefficients involved, are high priority items warranting further investigation.

5.10.2 Top-Down and Bottom-Up Quenching

At least three types of convective rewetting have been identified /5.40/: axial conduction controlled, impulse cooling and dispersed flow rewetting. Of these, axial conduction controlled rewetting is the type relevant to top-down and bottom-up quenching.

In codes, which analyse the reflooding of the core, the propagation of the quench front is modelled either:

a) by defining the quench front as the point where cladding temperature has dropped below a certain characteristic rewetting temperature, or

b) by calculating a quench front velocity based on an axial-conduction model, and using it to update the position of the quench front.

In a system with an already wetted upstream surface, the displacement of the transition zone between film-boiling and nucleate boiling - the quench front - is indeed governed by axial wall conduction, as indicated in Fig. 5.3.c. Conduction transfers the heat from the film-boiling side to the nucleate boiling side, causing the surface temperature to fall. As the temperature at a given point drops below the rewetting temperature the quench front propagates.

Various phenomena have been proposed for specifying the conditions under which rewetting occurs. These embrace hydrodynamic mechanisms, such as vapour film instabilities /5.41/, and thermodynamic mechanisms, which involve a characteristic rewetting temperature. Examples of the latter include surface temperatures below a maximum liquid temperature /5.42/, surface wetability expressed by a liquid-solid contact angle and its dependence on temperature /5.43/ and, the temperature which allows the formation of a monolayer of liquid molecules on the surface /5.44/.

Clearly, characteristic temperatures play an important role in axial conduction-controlled rewetting models. Numerous models of this type exist /5.45/ and these have predicted rewetting, or quench-front velocities with varying degrees of success. Precursor cooling from droplets in front of the quench front is an additional complication for modelling. The major disadvantage of these models is the arbitrariness introduced by the choice of parameters required to fit experimental data /5.46/, and the fine meshes or noding required in numerical solutions to model the sharp discontinuities at the quench front. Such submillimeter meshes are not compatible with the usual noding of safety codes.

ECC codes vary widely in their treatment of rewetting /5.1/. Mixture-models can handle axial-conduction rewetting analyses. At the simplest level an empirical or conservative quench-front velocity is specified together with suitable heat-transfer coefficients for the unwetted and rewetted surfaces. Some two-fluid codes have an axial-conduction controlled rewetting model, which uses a moving fine-mesh for quench-front propagation. Even this mesh size is still not fine enough in certain codes to handle the flow details at the quench front.

The TRAC-BD code /5.47/ provides an example of the way top-down quenching as a result of core-spray coolant injected above a BWR core (Fig. 2.2.10) is modelled. The code takes the steam, water-droplets and the falling film as separate entities, and treats the individual modes of heat-transfer between them, including radiation heat-transfer. The falling-film quench front is tracked by a moving-mesh for each rod-group and for the channel walls. Radial and axial heat conduction in the fuel and cladding within each node of the mesh are analysed with the film front propagation calculated from a conduction limited sputtering model based on a characteristic rewetting temperature.

For bottom-up quenching, certain two-fluid codes employ characteristic rewetting temperatures to determine the quench front's location as a function of time, Fig. 5.3.b. CATHARE, for example, employs a burn-out temperature, determined from the intersection of a nucleate boiling heat-flux and a critical heat-flux. Other codes use the quench temperature from the intersection of the transition-boiling and film boiling regimes. CATHARE can also track quench front propagation using an axial-conduction-controlled rewetting model, which performs two-dimensional conduction calculations separately and returns the corresponding quench front velocity.

In summary, although two-fluid codes may be capable of handling the detailed processes involved with quench front propagation, the extremely fine (sub-millimeter) mesh numerical solutions required at the quench front are expensive consumers of computing time. They also require special programming techniques in the safety codes. At the present time, all the codes rely on empirical correlations in one way or another, which are adjusted to fit full-scale bundle test data.

5.11 Single and Two-Phase Impeller-Pump Behaviour

Pump performance under LOCA conditions has an important influence on the distribution of liquid in the primary system. During large breach LOCA's, it affects the coolant flow through the core and therefore the occurrence of a boiling crisis. For small breaches in PWR's, it can control phase separation and mass loss through the breach.

In the safety analysis codes pumps are treated as separate components, which are modelled through their integral performance characteristics, such as pressure-difference across the pump and torque, as a function of volumetric flow rate and pump speed. Even for single-phase flow, difficulty is experienced in obtaining accurate full-scale pump performance data to support the integral models, because of the company proprietary nature of these data. For two-phase flow the data are scarce indeed.

Typically, to model pump degradation under two-phase flow conditions, two-phase multipliers or interpolators for pump head are introduced. These are assumed to be a function of void fraction and are tied mostly to small-scale experimental tests, such as Semiscale. This simple mixture-model approach lacks generality and has been criticized /5.48/ to /5.50/ on the grounds that pressure level and flow regimes also influence the degraded performance of a pump.

Because of the mixing effect of the pump itself, two-phase flows exiting a pump will tend to be homogeneous. This may not be the situation within and upstream of the pump, both of which can influence the pump's performance considerably. More recent models of pump behaviour, /5.51/ and /5.52/, have attempted to overcome the homogeneous model deficiencies by introducing a drift-flux term. Furuya /5.51/ found that head degradation is caused mainly by the higher acceleration of the liquid phase and deceleration of the gas phase than is the case with single phase flow. He obtained a good correlation with data from pumps tested by Babcock and Wilcox at 1/3 scale and by Creare at 1/20 scale. Sami /5.52/ considered three-dimensional effects and the influence of different flow regimes. He notes that the coolant flow may choke in the pump, depending on the size and location of the breach.

The Furuya /5.51/ type of analysis could readily be extended to a full two-fluid model for the flow through a pump. De Crécy /5.53/ has developed a full two-phase pump model with two pressures to allow for mechanical non-equilibrium between the phases. This model is being validated against data for several types of pumps, at different speeds, and in single- and two-phase flow conditions for eventual incorporation in the CATHARE 2 code. Correlations required at the two-fluid level are, however, lacking. Notwithstanding the inherent difficulties of measuring flow parameters in pumps, the two-fluid data required will not be any easier to assemble than current integral pump data.

5.12 Loop-Seal Clearing and Filling

Clearing of the intermediate- or crossover-leg between the steam generator and the main coolant pump (Fig. 2.1.7) can strongly influence core uncovery during a small-breach, cold-leg LOCA in a PWR. The back-pressure of steam (steam-binding) driving the core uncovery process can only be relieved once the water-seal in the intermediate-leg is cleared of water.

Modelling of loop-seal clearing may be divided into three coupled parts:

a) the hydrostatic heads of the "collapsed" columns of water in the loops, in the downcomer annulus and in the core, which form or influence the loop-seal behaviour,

b) the transient steam-binding pressures acting on the steam/water interfaces of the columns, and

c) the behaviour of the loop-seal interface, just as the seal is clearing.

Hydrostatic head modelling is straightforward and one-dimensional, single-phase, energy-balances /5.54/ are adequate for this purpose. Steam-binding pressures are an integral of all the steam-production, entrainment, de-entrainment, steam condensation and counter current flow process occurring in the uncovered portions of the loops. The accuracy of loop-seal clearing prediction thus depends on how well the mixture or two-fluid models handle all of these phenomena, which have been the subjects of discussion in Sections 5.3 to 5.6. It is here that large uncertainties in core uncovery behaviour may arise, because the coolant mixture-level in the core can be very sensitive to small pressure differences between the core and the annulus.

Just as the loop-seal begins to clear, the one-dimensional steam/water interface at the seal can break up. The two-phase flow through the loop-seal can then be very complex indeed, involving density wave propagation, plugs of water being expelled and oscillatory behaviour with the loop seal clearing and refilling several times /5.55/. Such detailed phenomena cannot be analysed properly by codes which employ closure relations based on steady-state correlations. Modifications to the CATHARE code /5.56/, involving added mass and discretization of the liquid momentum equation, have been introduced, which enable the average behaviour of the loop-seal clearing process to be computed in terms of the void fraction or liquid mass remaining in the columns /5.57/. Other codes ignore these complications.

The apparent simplicity of loop-seal clearing is deceptive. It poses very severe challenges to two-fluid code capabilities, not only with regard to steam-binding and the seal-clearing process itself but also because of the oscillatory behaviour that can arise. For particular breach sizes, such oscillatory behaviour can lead to random clearing of different loop-seals or to more than one loop-seal clearing simultaneously /5.54/ and /5.59/. The loop-seal clearing process is a phenomenon that warrants further investigation.

5.13 Concluding Remarks on Modelling ECC Phenomena

Two-fluid models provide the means for a more realistic/mechanistic and more precise analysis of many dominant or controlling ECC phenomena, which cannot be analysed by mixture models other than fully empirically. Such phenomena include: phase separation, entrainment, mixing, condensation, boiling, counter-current flows, rewetting, quench-front behaviour and loop-seal clearing. The price of this precision is the additional empirical information required for closure laws at the two-fluid level. Much remains to be done in this area.

Comparisons /5.1/ of codes based on a two-fluid formulation have shown that, despite large modelling and numerical differences at the closure-law level, engineering parameters important to safety and licensing are predicted with acceptable precision. Here by acceptable is meant: may be used for design-basis accident evaluations in which conservative margins are applied to cover uncertainties. Analogous conclusions may be drawn for similar evaluations based on mixture models, where an additional layer of conservatism may be necessary to cover mixture model deficiencies. The direct benefits to the ECC licensing basis of more accurate predictions of parameters are probably small at the present time. Economic benefits can be gained, however, from improved operational flexibility.

This situation changes when best-estimate codes are required to support the development of plant analysers or simulators, to help assess operator procedures or for investigating situations beyond the design basis. Critical remarks on two-fluid models, which have been discussed in this chapter and elsewhere /5.58/ and which are summarized below, are then relevant to research and development efforts:

- The current algebraic formulations of the closure laws may be too crude an approximation for the general application of two-fluid models to a wide variety of geometries and scales. Interfacial relationships need to be improved, particularly where geometry effects have a strong influence.

- Closure-law formulations in the codes are selected either directly or on the basis of flow-regime maps. All of these are established from steady-state experiments, often in simple one-dimensional geometries at very small scales, and over a limited parameter range. Application of such formulations to full-scale conditions outside their tested range is still a questionable procedure.

- Closure-laws for identical physical phenomena are often based on flow-regime maps which are different for momentum- and heat-transfer.

- Many of the two-fluid models still rely heavily on global, mixture-type, correlations or conditions for predicting when or where a particular phenomenon occurs. This applies, for example, to boiling-transitions, phase-separation, entrainment, vapour and liquid pull-through, bypass-flows and loop-seal clearing.

- The partitioning of energy and momentum between phases and structures involves the specification of interfacial areas, which is a very difficult task since such areas are almost impossible to measure. Their choice in the codes is largely arbitrary, yet they have a very strong influence, for example, on mixing, condensation and counter-current flow phenomena.

- Space nodalisation and time-steps in the codes are controlled by the code user. The choice can seriously influence: the ability to model some very local phenomena, the appearance of real or numerical instabilities, the ability to achieve convergence of solutions and the applicability of predictions to large scales. The basically ill-posed nature of the closed equation system adds to these difficulties.

Many of the shortcomings listed above have not changed over the past several years or so. Without a dedicated programme of analyses, a typical example is /5.1/, it is difficult to quantify their impacts on the key safety parameters relevant to design basis or beyond design basis scenarios. Quantitative assessment of design basis predictions, discussed in the next chapter, illustrates the scope of such a programme.

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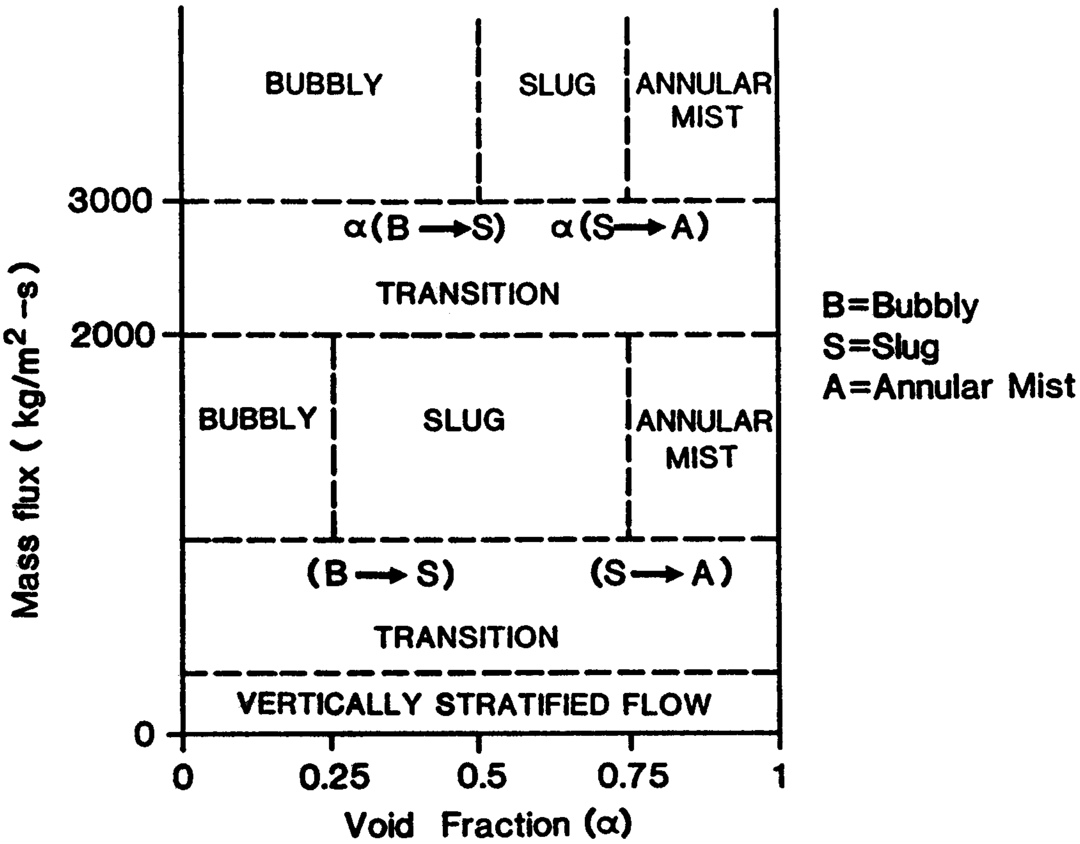
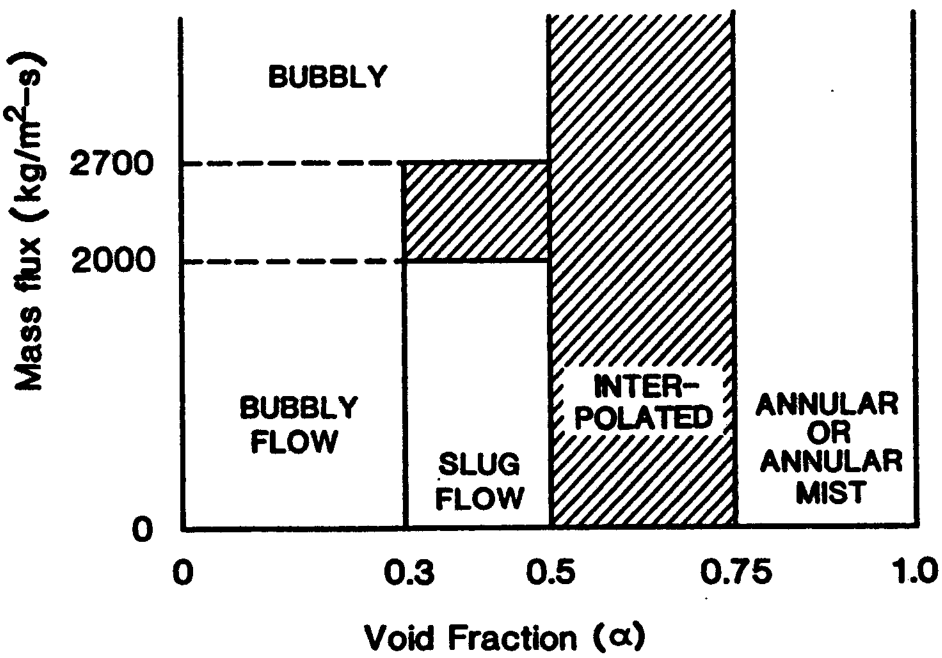
Field Equations / Dimensions	3 Equations	4 Equations	5 Equations	6 Equations
1 Dimension	RELAP 4/MOD 3 to MOD 6 THYDE-P 2	DRUFAN * SMABRE	RELAP 5/MOD 1	FLUT * TRAC PF 1/MOD 2 CATHARE 1 and 2 RELAP 5/MOD 3 ATHLET
2 Dimensions				CATHARE 2 (DOWNCOMER ONLY) **
3 Dimensions				TRAC-BD TRAC PD 2 TRAC PF 1 COBRA/TRAC *** (VESSEL ONLY)

* DRUFAN/FLUT is an integrated code system
** Under development for downcomer only
*** COBRA/TRAC has 8 field equations

TABLE 5.1 ECC CODES, NUMBER OF FIELD EQUATIONS AND DIMENSIONS /5.1/ AND /5.3/



Vertical flow regime map in RELAP 5/MOD. 3



Flow-regime map for three-dimensional hydrodynamics,used in TRAC-PF1 (Cross-hatched regions are transition zones).

FIG. 5.1 TYPICAL FLOW-REGIME MAPS

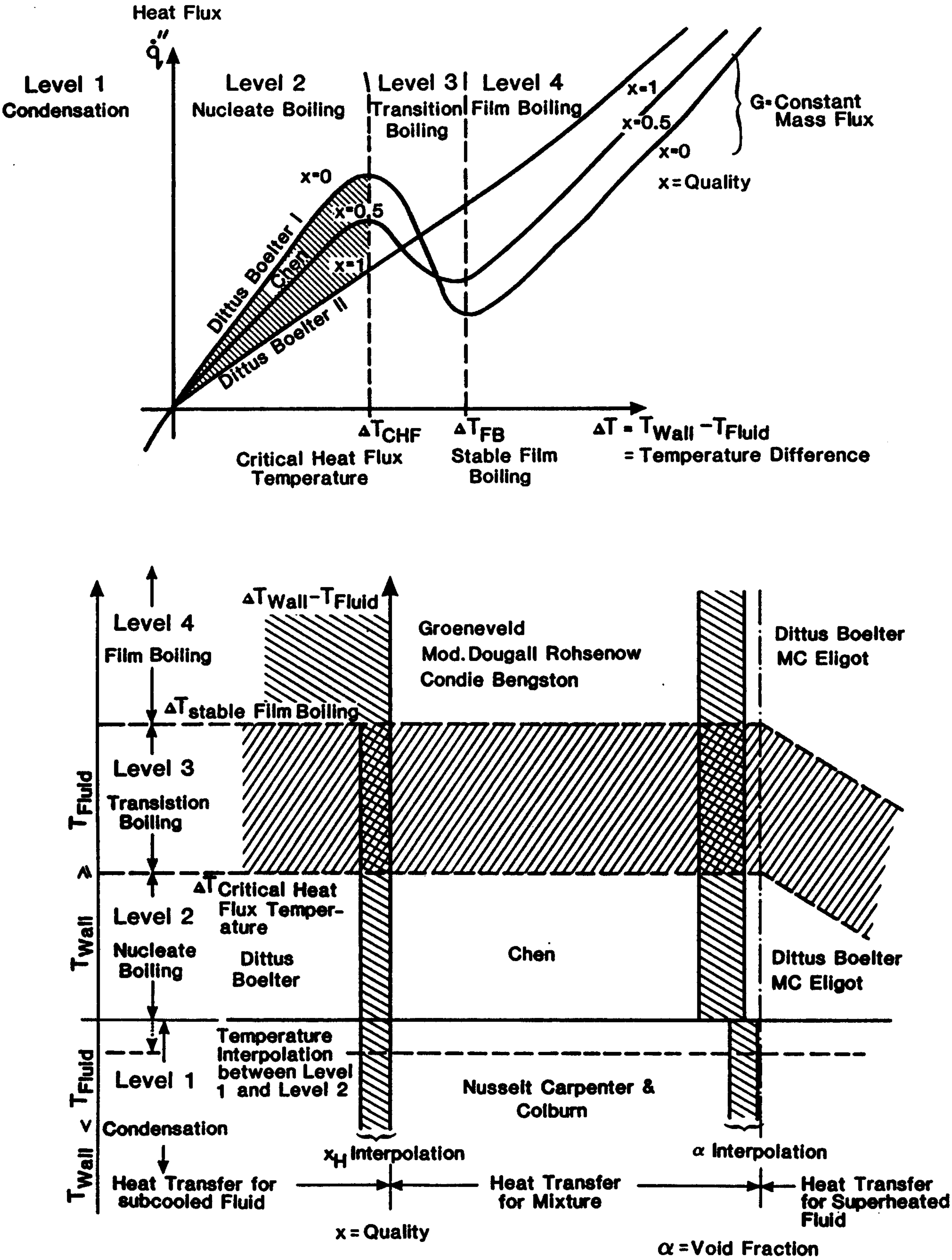
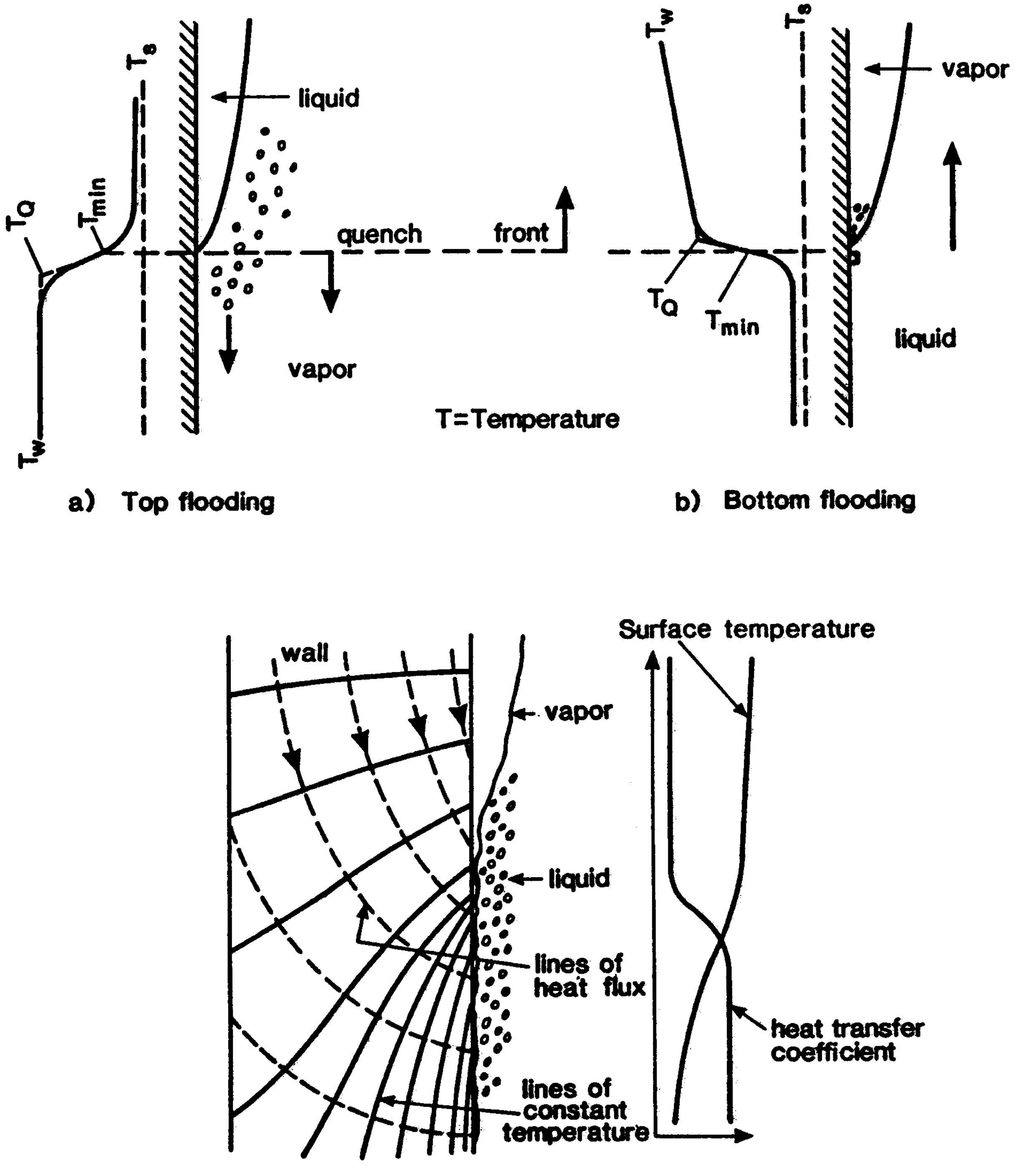


FIG. 5.2 HEAT TRANSFER REGIMES AND CORRESPONDING CORRELATIONS IN DRUFAN /5.1/.



c) Temperature and heat flux fields in a wall during axial-conduction-controlled rewetting of a system undergoing bottom flooding

FIG. 5.3 QUENCHING PHENOMENA DURING TOP AND BOTTOM FLOODING

6. CODE ASSESSMENT

6.1 The Scope of Code Assessment

The key issues involved with ECC are the application of system codes to predict nuclear power plant LOCA behaviour, and the level of understanding required to specify from these predictions quantitative safety margins and uncertainties. Best-estimate system codes, such as those in Table 5.1, are the practical means for resolving these issues. They must do this reliably and accurately with support for the accuracy claimed. The purpose of code assessment is to provide this support, which is not an easy task because the span of the LOCA problem from a phenomenological viewpoint is very broad indeed.

Test-rigs, scaling problems and measurement difficulties are discussed in Chapter 3, and analytic modelling of LOCA phenomena is reviewed in Chapter 5. Code assessment, the subject of this chapter, brings both of these topics together with the aim of drawing conclusions about the accuracy of code predictions.

Put very simply, code accuracy is the difference between measured and predicted quantities taking into account uncertainties and biases in both. The terms are illustrated in Fig. 6.1. Usually uncertainties and biases are simply added to obtain a value for code prediction accuracy. Note that since no measured data from a LOCA in a plant exist for code comparison purposes, (other than TMI), the term "uncertainty" should strictly be used when applying code predictions to plant LOCA behaviour.

Although largely concerned with differences and uncertainties in measurements and predictions, an assessment must take into account the goals of the assessment itself and other sources of uncertainty:

1) Correct specification of the problem and its goal

THE ACCURACY REQUIRED IN AN ASSESSMENT MUST BE COMMENSURATE WITH THE INTENDED USE OF THE PREDICTED INFORMATION. This statement, which is expanded on in the following examples, needs to be emphasised, because it provides an important boundary condition to the question "How good is good enough".

A qualitative assessment of ECC effectiveness is one in which no claim for accuracy is given. If very large, conservative, allowances or margins are provided to cover all uncertainties, predictions can be "good enough" from a design-basis safety and licensing point of view. They may not be good enough, if such large margins prevent economic plant operation. This approach, which commonly relies on conservative evaluation models, depends heavily on engineering judgement and has been used from the very beginning for licensing plant ECC. But for one or two exceptions (/6.1/, for example), it is still in use today although, as quantitative assessment matures, these exceptions may become the rule.

Quantitative assessment takes the question of accuracy a stage further. For example, if peak cladding temperatures are required to be less than 1200 °C, a quantitative assessment giving an uncertainty of ± 200 °C with predicted temperatures less than 1000 °C is good enough, but not so otherwise. This quantitative approach is somewhat more complicated in that:

- it only makes sense to make such assessments with best-estimate codes.

- the required accuracy may be governed by a subjectively imposed confidence statement, which brings probabilistic and statistical consideration into the picture. In a multi-faceted problem, such as a LOCA, these introduce enormous complexity into an assessment, as will be seen in Section 6.3.

- predicted temperatures will depend on plant operating boundary conditions, in particular, fuel linear heat generation rates, and these may or may not be controlling the plants economic performance. If not, there is no grounds for a very accurate assessment. This conclusion may change if the assessment is required for other purposes.

These illustrations show that there is not a single, magic, formula or definition for qualitative or quantitative accuracy, when looking for an answer to the question of how good is good enough. Required accuracy in any assessment is a function of the specific problem to be solved for a specific plant.

2) Uncertainties associated with code assessment

System codes are based on analytical models, which comprise selections and simplifications of real physical processes. In formulating a model the number and form of the conservation equations, correlations, special component models, etc. are chosen by the code developer. The code user has to put together an input deck for the power plant of interest, provide boundary conditions and select suitable model options. All of these can be major sources of errors (e.g. wrong choice of option or simply typing errors) and undefined uncertainties. To quantify or, at the least, minimize such errors and uncertainties, complete and correct documentation of codes and code user guidelines together with code comparison studies, such as the standard problem exercises, are indispensable.

A computational model consists of a code based on equations and a geometric idealization or nodalization of the system being modelled. Programming errors may exist, the numerical integration scheme for the equations may not be stable, numerical diffusion and truncation errors may arise. Errors may be disguised during code development by the large uncertainty bounds in some model parameters and the practice of adjusting specific predictions to particular experiments. The use of benchmark exercises during code assessment is one tool to avoid the errors or to assess the uncertainties arising from numerical solution procedures.

A unified nodalization scheme is particularly important when using codes to predict the performance of full-scale plants. Large uncertainties may be introduced by employing large nodes for the plant, when very detailed noding was used to assess the accuracy of code predictions from small-scale test data.

Traditionally, code assessment has been divided into developmental and independent assessment. The basic idea is that the code developers, by comparing and sometimes adjusting code predictions to selected experiments, assess the separate models or parts of the code. Developers also check that all the parts function together as intended. The code is then issued as a frozen version to be assessed, by groups working independently of the code developers, against data from separate-effects tests, integral-tests or even against predictions from other, already assessed, codes. At this stage, the independent groups can use the data to make a qualitative or quantitative assessment of the accuracy of code predictions. At this stage too, the influence of scaling on predictions of full-size plant behaviour has to be considered.

To summarize, the major element of any assessment is the comparison of code predicted quantities with test data or with calculations from already assessed codes. In addition, an assessment has to consider the rather ill-defined uncertainties introduced by the code user as well as those arising from code mathematics. Last, but by no means least, it has to consider the uncertainties introduced by scaling. Whatever the assessment, it has to take into account the goals or purposes for which the assessment is to be used. How these elements are addressed by different groups is discussed in the following sections.

6.2 Qualitative Assessment

6.2.1 Assessment Programmes and Comparison Exercises

With qualitative assessment attention is focussed on the correct simulation of all the relevant phenomena and the overall impression of a code's capabilities. The code is compared against a wide variety of experiments, mainly integral, to check that the trends in the data are reasonably well predicted. The comparison is based on engineering or expert judgement with little or no attempt to quantify accuracy or uncertainty other than very loosely. Clearly, this approach can be used with conservative evaluation models or with best-estimate codes.

The effort invested in qualitative assessment has been enormous, particularly in the main code developing countries such as the US. This effort is reflected in the two international programmes, which have dominated both qualitative and quantitative code assessment in the last decade:

- the International Code Assessment and Applications Programs (ICAP) organized by the USNRC /6.2/, and

- the International Standard Problem (ISP) Program, Table 6.1 and /6.3/, organized by the OECD-NEA/CSNI, mainly through its Principal Working Group 2 Task Group on Status and Assessment of Codes for Transients and ECCS.

In both of these programmes, experiments or actual plant transients are calculated as exercises by participating organizations using a range of computer codes, nearly all of them best-estimate. Typically, test boundary and initial conditions, together with a complete description of a facility or test-rig, are communicated to all the participants. Calculations are performed as:

- an open exercise, in which the experimental data to be calculated is made available to the participants when performing the calculations, or as

- a blind exercise, in which the data is not released until the calculations are completed, or as

- a double-blind exercise, in which the data is not released and the facility has not been analysed previously. This means there is no prior code user experience for modelling the facility.

Each exercise is controlled by a set of rules (/6.3/, for example). Calculated results from different codes are compared against measured data as well as against each other. Workshops are then organized to discuss the comparison report, which may eventually be published.

Experience from these exercises which, for each one listed in Table 6.1, is or will be documented in OECD comparison reports, has proven beneficial to all participants. Code developers have been able to make steady improvements to their codes. Code users have improved their skills and gained confidence. They have learned about code limitations, about the shortcomings of measured data and that code comparisons are difficult. Safety authorities have also followed the exercises closely. The findings from this work have produced no major surprises, or new phenomena, which have had a negative influence on safety. They show a trend towards lower predicted cladding temperatures, for example, and generally confirm the bounding conservative assumptions used in ECC licensing application.

6.2.2 A Procedure for Qualitative Code Assessment

Performing comparison exercises alone is not sufficient to justify a code and its user as properly assessed, even from a qualitative standpoint. The CSNI Task Group on Code Assessment reviewed the whole assessment process and formulated the following procedure, which is recommended for qualitative code assessment:

1) Transient and LOCA categories for which the code is to be assessed are defined. These may include: transients involving one- or two-phase flows, small-breach LOCA's, large-breach LOCA's, etc.

2) Test cases are selected. To bring some order into the selection process the Task Group generated Validation Matrices /6.4/, which are being updated and supplemented on a continuous basis. These matrices include many of the tests used for ICAP and ISP exercises.

3) LOCA phases, or phenomenological windows such as blowdown, refill or reflood, are specified. These define time spans in which particular phenomena and parameters are dominant.

4) Important phenomena are identified. The Validation Matrix, or a procedure such as that outlined in Chapter 4, may be adopted for this purpose.

5) Parameters, which address the identified, dominant phenomena, are selected. These may be, for example: characteristic temperatures for core-quenching, core fluid dynamics, flow-regime maps, etc.

6) The selected test cases are calculated.

7) Calculated and experimental data are compared. This must include deliberations on:

   . the reasonable simulation of the important phenomena,
   . the correct coupling of phenomena and any compensating errors that may arise from this coupling,
   . the reasonable time-scales of predicted phenomena,
   . the influence of threshold phenomena; for example, after dryout or rewet there are drastic changes in cladding temperatures,
   . the occurrence of real or mathematical instabilities.

8) Sensitivity studies are performed to investigate special problems, such as: the coupling of phenomena, the influence of threshold phenomena on predicted results, the influence of noding or mesh-size, etc.

9) Taking all of the above factors into account, particularly the scale dependence of dominant phenomena (Chapters 3 and 4), a first engineering judgement is made on the suitability of the code for full-scale analyses of the transient and LOCA categories defined in 1).

The above procedure lays firm foundations for a consistent approach to both qualitative and quantitative assessment. Indeed, it may help to clarify whether a quantitative assessment is at all necessary. If the code accuracy is clearly unacceptable, or very large margins from safety limits are demonstrated, a quantitative assessment may be superfluous.

6.2.3 Lessons Learned from Qualitative Assessment

The assessment exercises have shown that there is no such thing as an ideal set of experimental data. Although a validation matrix may provide a good selection of available data, much of the data base is limited both in quantity and quality. This has been expanded upon in Chapters 3 and 4. Even when an experiment appears to be ideal, a closer look may reveal deficiencies in instrumentation, failures of instrumentation, and/or uncertainties in the data. Uncertainty bands may be either too large to be useful or are unknown. Experimental data reports are often incomplete: poor facility descriptions, unclear test boundary conditions, out-of-date information and so on.

A code selected for assessment should, in principle, be frozen or kept unchanged. All users of the same code should have identical code versions and corresponding documentation, so that assessment results and user experience can be compared on a like basis. In practice, a code is never completely frozen. Minor error corrections, or the addition of features convenient to the user, are allowed during an assessment, major changes are not. If serious deficiencies are found, the continuation of an extensive assessment is a waste of time and resources. To avoid reaching this situation when well into an assessment programme, a procedure such as that illustrated by Fig. 6.2 is recommended. A small number of carefully chosen tests may first be analysed with the aim of revealing any major deficiencies. If no major deficiencies are revealed, the complete and much larger set of tests are then analysed.

On the other hand, fast and well-documented feedback between code users, assessors and code developers is extremely important during code assessment. This can greatly help to minimize the uncertainties or errors arising from the user or from code mathematics. In practice, communication channels between developers and users have often proved to be unsatisfactory. Sometimes no contact person is nominated or no time is reserved for the work.

Code "tuning" is the adjustment of input parameters, user options, nodalization, or even changes in the models themselves, to arrive at the best possible agreement with particular test data. If a new set of code optimum parameters has to be defined for every new test case, without a sound technical justification, code assessors can be misled. Code tuning during an independent assessment is therefore unacceptable. The object of an assessment is not to get the best possible results, but to obtain objective information on the code when used according to the normal user guidelines.

Finally, when should qualitative assessment end and quantitative assessment begin? When should code development stop? When should a code be considered as acceptable and what criteria should be used? Answers to these questions are really concerned with the question of how good is good enough. As argued in Section 6.1, there are no patent answers, it all depends on the use for which the code predicted information is intended /5.4/:

a) Qualitative assessment of ECC effectiveness with evaluation models or with best-estimate codes can be acceptable for plants in which substantial and conservative margins to safety limits are assured. This assurance can be provided, for example by sensitivity studies.

b) Quantitative assessments may be needed to improve operating performance, to define more accurately core damage during postulated design basis accidents or beyond. The accuracy required for each of those applications may be quite different.

c) Quantitative assessments may be needed to define more accurately the boundary conditions at which severe core degradation begins.

Other examples come to mind but, at the present time, continuing code development and quantitative assessment are certainly needed to attack b) and c) with confidence.

6.3 Quantitative Assessment

Quantitative assessment of a best-estimate system code's capability to predict any coupled two-phase flow and heat-transfer problem is an extremely complex task, even more so when the problem concerns LOCA. Quantitative assessment aims at expressing the uncertainties in predicted and measured parameters, and the accuracy of the prediction (as defined in Fig. 6.1), as numbers. By the very nature of the problem, these numbers will involve statistical distributions, which means that some sort of confidence statement or interval must be defined for the accuracy required or achieved. These numbers cannot just be tied to a set of experiments, they must also reflect the code's ability to predict full-scale plant behaviour.

Three independent approaches to quantitative code assessment have been developed, one by the UKAEA at Winfrith, another by the GRS in Munich and the third by the USNRC with the assistance of subcontractors and consultants. All of these methods are undergoing continuous development and, at the time of writing this report (1989), only the NRC methodology has actually produced a quantitative assessment of a large-breach, PWR-LOCA. Nevertheless, a brief description of each method, followed by a comparison of the methodologies, provides an insight into the difficulties involved. Another method has been developed separately by General Electric (GE) for application to BWR's. This is not exactly comparable with the above three methods, since it aims at predicting an upper-bound peak cladding temperature with a given probability. Nevertheless, it provides valuable insights and is also briefly described.

6.3.1 The UKAEA, Winfrith Method

The aim with this method is to extend the qualitative approach, described in Section 6.2.2, to include some meaningful quantitative assessment of uncertainties. Mechanical applications of statistical techniques are claimed not to be justified by the nature of the problem, by the data available or by the expense involved. The method relies heavily on expert judgement and experience, and attempts to use both in a logical way. It is designed to generate useful information with the minimum of effort. The main idea is the use of parameter sensitivity studies to generate code uncertainty bands for the key output parameters. The choice of the model parameters, the choice of boundary conditions, as well as the required number of sensitivity calculation, are based on expert judgement. Comparisons to the measured data of selected integral tests serve to check if the code accuracy is acceptable; that is, uncertainty bands bracket the data and are not so large that the results are useless. The same procedure is applied to both integral tests and plant calculations. It is argued that the generated uncertainty bands for the latter are meaningful if the procedure has been applied successfully to the former.

Fig. 6.3 is a flowchart depicting the Winfrith Method. This figure represents the whole process of code assessment from code development to final code application. The first three steps are consistent with current practices. There are, however, some requirements that a code must fulfil before the assessment method, from step 4 on, can be applied, e.g.:

- numerically stable, reliable
- absence of non-physical discontinuities
- reasonably fast-running
- all relevant physical phenomena addressed
- constituent models can be biased to predict the phenomena with a reasonable degree of accuracy.

The last requirement addresses the need for means to carry out sensitivity studies with the codes. If parameter variations are not possible through input data, the code will have to be modified, re-compiled and relinked for each sensitivity study, which increases the amount of work involved and the scope for error.

Uncertainty bands for integral experiments and plant LOCA's are derived in an identical manner. In both cases it is important to understand the transient as well as the code under scrutiny to ensure that all important phenomena and their corresponding code models are identified. The next step is a selection of the model parameters to be varied, and an estimate of a reasonable uncertainty range for each parameter. These ranges are based on experimental data from separate effects tests.

A set of calculations or sensitivity studies is then performed to identify reasonable bounds on chosen output variables or key parameters. This does not necessarily require a separate calculation for each input parameter to be varied. Where the expected effects of varying several parameters is in the same direction, these may be combined in a single run. The resulting bounding curves of the output parameters vs time may be determined most simply by taking the maximum and minimum of the calculated values at any time, although in some cases alternative approaches may be desirable.

Comparisons with measured data from integral tests serve to check whether the method and the generated uncertainty bands are valid or not. If the data are not bracketed by the bands, it means a failure of the approach, which then has to be amended in a justifiable way. This may mean corrections or improvements to the code. Furthermore, it is not sufficient just to bracket the data. The bands must be narrow enough to be useful. Blind standard problems are especially valuable for testing the method. Only if the method has been shown to be successful in the case of integral tests, can it be applied to plant calculations with confidence.

The first feasibility study of the Winfrith methodology is under way using the LOBI small break LOCA test BL-02 as the test case. The main objectives of the study are the following:

- to find out the number of sensitivity studies required,
- to unvestigate the usability and adequacy of the separate effects data base for determining the required model parameter uncertainty ranges,
- to study the difficulties in performing the model sensitivity studies in the absence of a means to access them through code inputs,
- to check the adequacy of the experts' initial judgement on the key phenomena and parameter uncertainty ranges,
- to examine the special problems of computational model implementation and noding uncertainties, and
- to find out the overall effort required to apply the method.

6.3.2 The GRS, Munich Method

The aim with the GRS method is to reduce the influence of expert judgement to a practical minimum. The basic idea is to use probability distributions to express the uncertainties in the computational model parameters and boundary conditions, and to propagate them through the code. By performing a number of calculations the probability distributions of desired output parameters are generated. Analyses following this procedure are first performed on integral tests to validate the procedure. An identical procedure is then followed for the plant transient or LOCA of interest.

Central elements of this methodology are shown on Fig. 6.4. Briefly:

1) Methodology Assessment through Integral Tests

- Transient categories, phenomenological windows and corresponding integral tests are selected (Steps 1 to 3).

- Decisions on the dominant phenomena and corresponding analytic models are made (Steps 4 to 5).

- All model parameters and boundary conditions, which potentially contribute to the uncertainty in code predictions for a chosen integral test, are listed (Steps 6 to 7).

- A maximum conceivable uncertainty range is specified for each identified parameter. Expert judgement is employed to select probability distributions or probability density functions consistent with the existing level of understanding of each parameter (Steps 8 and 9).

- Dependencies between model parameters are taken into account by introducing suitable restrictions, by quoting appropriate conditional probability density functions or by specifying suitable measures of association.

- A joint probability density function is set-up over the combined uncertainty range of the parameters.

- A probability density function for key output parameters is determined by propagation of the input density functions through the model. This is done by random sampling of input parameters and by performing a set number (see below) of calculations for the chosen integral tests (Steps 10 and 11).

- Quantitative statements concerning the combined influence of the quantified uncertainties on code predictions are made.

- Parameters are ranked according to their contributions to the overall uncertainty.

- Predictions are compared with measured integral test data to see whether the overall code prediction uncertainties bracket the data.

- The above steps are repeated for several integral tests to check that measured data are consistently bracketed.

2) Plant Transient or LOCA Predictions

The above procedure, but for the comparisons with measured data, is followed again with the plant transient or LOCA. Full-scale separate-effects tests and available plant data are used as much as possible to support the selection of probability density functions and to account for scale effects. Additional uncertainties may be imposed at any stage, if they are thought warranted.

Some idea of the costliness of this approach may be gained from the fact that 59 runs per integral experiment or plant transient are needed, regardless of the number of input parameters, to obtain a satisfactory confidence level (95:95) for predicted parameters. Blind standard problem exercises are considered to be particularly valuable for testing the methodology.

A first feasibility study of the GM method is currently under way. The main objective of this study is to gain experience with what is essentially a standard method applied to the area of thermohydraulics and ECC. The study serves, in particular:

- to clarify the difficulties involved with specifying probability distributions for model parameters and boundary conditions,
- to identify and investigate any specific dependencies between uncertain parameters,
- to evaluate the number of runs required to propagate the probability density functions through the model, and
- to determine the overall effort required to apply the method.

6.3.3 The USNRC Method (CSAU) /6.5/ and /1.7/

The USNRC's approach is to combine quantitative analyses and expert opinion to arrive at a computed value of the uncertainty to be expected in code predictions of any specific plant transient or LOCA. Code predictions are compared with a selected set of integral and separate-effects tests or assessment cases. The methodology addresses all the aspects of code assessment; qualitative and quantitative assessment, full-scale predictions and applicability to particular LOCA scenarios. For this reason it is called the Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology.

Stated objectives of the method are:

i) to provide a technical basis for quantifying uncertainties within the context of the revision to the ECCS Acceptance Criteria in the USA. Defining the context is very important, because it provides guidance as to how precise the assessment needs to be.

ii) to provide an auditable, traceable and practicable method for combining quantitative analyses and expert opinion to arrive at the computed values of uncertainty.

iii) to provide a systematic and comprehensive approach to:
   a) defining scenario phenomena
   b) evaluating code applicability
   c) assessing code scale-up capabilities, and
   d) quantifying code uncertainties concerned with:
      . code and experimental uncertainties
      . code scale-up capabilities
      . plant state and operating conditions.

Both the second and third objectives are laudable, and essential if reliable and consistent assessments are desired, but the scope of the problem becomes enormous. To make the method more practical attention is focussed on the important or dominant phenomena, and expert judgement is employed wherever feasible.

A flow chart of the CSAU methodology is shown in Fig. 6.5.1 It comprises three basic elements:

<u>Element 1)</u> <u>Requirements and Code Capabilities</u> (Step 1 to 6)

1. A transient or LOCA scenario is specified together with the set of relevant parameters to be considered.

2. A nuclear power plant (NPP) is selected and the set of scenario parameters augmented.

3. If necessary, the scenario is divided into characteristic time periods. The phenomena or processes appropriate to these periods are identified and ranked. From this appraisal the dominant parameters, which are to be examined in detail are selected. A process identification and ranking table (PIRT) combined with an analytical hierarchical process (AHP) is one way /1.7/ to do this. A similar approach is described in Chapter 4.

4. A code version is selected and "frozen" to ensure that no changes, which can impact the conclusions of the assessment, are introduced into the code.

5. Complete code documentation is mandatory. This includes the code manual, user guidelines, a quality evaluation of code models and correlations, and other supporting documents. The quality evaluation documents provide detailed information on equations, closure relations, data-bases, sources, uncertainties, scale-up capabilities and so on.

6. As a first check on whether code capabilities are adequate to model the identified phenomena, the requirements deduced from steps 1 to 3 are compared with the code capabilities of step 5.

Element 2) Code Assessment Against Tests and Establishing a Range for Each Parameter (Steps 7 to 10)

7. A test matrix, for example /6.4/, for assessing the accuracy of the code against test data is drawn up. Both separate-effects and integral-tests are selected for the matrix on the basis that the dominant processes and phenomena identified in step 3 actually appear in these tests. Tests may also be selected to address any code inadequacies identified in step 6.

8. The nodalization of the code to represent the nuclear power plant is chosen to be detailed enough to capture the important phenomena (step 3) and design characteristics of the plant. The same nodalization is used for both plant and code assessment calculations.

9. The tests in the matrix are analysed by the code and the accuracy of code calculations is determined. This procedure quantifies the bias and uncertainties between code predictions and test data, taking into account the uncertainties in the experimental data, c.f. Fig. 6.1. Integral-test data is used predominantly to evaluate the overall code accuracy. From these matrix analyses the range of variations in each parameter needed for sensitivity studies is established.

10. Test data are used to evaluate the scale-up capabilities of the code, as well as the codes ability to handle the dominant phenomena. By quantifying the differences between measured and code predicted parameters at different scales and, where necessary using expert judgement, an overall statement of uncertainties involved with scale effects is made. This step may be limited to those parameters known or judged to be potentially scale dependent.

The detailed CSAU procedure to evaluate a code's ability to scale plant behaviour is illustrated in Fig. 6.5.2. Very briefly, it consists of ten steps, grouped into two key sub-elements, which are applied to a specific LOCA scenario:

Sub-Element i Requirements for Scaling and Scaling Capabilities of Tests (Steps 10-1 and 10-2) and Codes (Steps 4 and 5 of CSAU)

10-1 The test facilities which supplied the data used for the correlations or closure relations in the code, or the data used to assess the code, are specified.

10-2 For each test, documentation is provided which discusses in detail: the important or dominant processes investigated, similarity criteria employed, similarity criteria to be preserved for scaling purposes and, scaling distortions.

A similar procedure for evaluating the requirements and the scaling capabilities of the codes is already included in Steps 4 and 5 of the CSAU method.

Sub-Element ii Evaluation of Scale Distortions and Specification of Corresponding Biases (Steps 10-3 to 10-10)

10-3 Based on Step 10-2 and the CSAU Step 3, scale distortions introduced by components or processes important to particular phases of a transient are identified.

10-4 Biases to cover these distortions are specified and justified.

10-5 For important processes not affected by scale distortions, the data base range is evaluated for its applicability to power plant conditions.

10-6 If the data base does not cover plant conditions, a bias is specified and justified.

10-7 to 10-10 The scale-up capability of code correlations or closure relations, and the range of application of the correlations, are evaluated in a fashion similar to steps 10-3 to 10-6. Where necessary, biases are also specified and justified.

The results of these scaling evaluations are a number of biases, which are implemented into element 3) of the CSAU method, where they are combined with other biases and uncertainties.

Element 3) Bias and Uncertainties in Code Predictions, Sensitivity Studies and Code Prediction Accuracy (Steps 11 to 14)

In this third element the bias and uncertainties in code predictions for the nuclear power plant are established by sensitivity studies and combined with the bias and uncertainties derived above in element 2), sub-elements i and ii.

11. Uncertainties in the reactor state and operating conditions at the initiation of the transient or LOCA are specified. Realistic variations are determined using statistical and experimental data supported by analytical studies.

12. Nuclear power plant sensitivity studies of the transient or LOCA are performed to provide the bias and uncertainty in code predictions of the dominant output parameters. Input parameter variabilities for these sensitivity studies are taken from the parameter ranging of element 2) and from step 11.

13. All the biases and uncertainties from elements two and three are combined in a statement of total uncertainty or code prediction accuracy. The way they are combined should be justified. Additional margins can be added to reflect, for example, limitations in the data base. Total mean and 95 % probability values of appropriate parameters are calculated.

14. A statement is made about the codes ability to predict a prime safety parameter; such as the peak cladding temperature during a large-breach LOCA. This statement may be in the form of an uncertainty band around a best-estimate value, or an upper or lower bounding value. The uncertainty band or bounding value is stated in terms of the probability, commonly 95 %, of being within the band, or remaining below or above the bounding value.

The CSAU methodology has already been applied to a large, cold-leg, breach in a PWR. Indeed, it was during this application that the methodology was developed to its current form. Several different approaches were explored to handle the details, some of these proved fruitful and others did not. This first application focussed on determining the uncertainty of peak-cladding temperatures calculated during the blowdown and reflood periods.

Results for the mean and 95 th percent probability level are shown in Table 6.2.

Peak-cladding temperatures are single-valued parameters. Extension of the CSAU method to small-breach LOCA and other transients requires development of the methodology to evaluate the uncertainties in continuous-valued parameters; such as water inventories and mass flows.

6.3.4 The GE Method /6.1/

The GE approach contains all the rudiments of the above three methods, but only the variabilites in plant operating and plant specific parameters are treated statistically. To account for modelling uncertainties, experimental uncertainties and scale effects, conservative bounding biases are introduced without any statistical weighting. By combining the biases and statistics GE aims at predicting upper-bound peak cladding temperatures, which will not be exceeded at a defined probability level; typically the 95th percentile. This means that the GE methodology lies somewhere between the qualitative, but conservative, evaluation approaches and the above fully-statistical, quantitative methods, which treat both biases and uncertainties statistically. Nevertheless, it is described briefly here, because it has proven to be relatively easy to use and it provides valuable insight into biases and uncertainties.

The GE method may be divided into three parts, Table 6.3:

1) For a selected plant the TRAC-B02 code is used to calculate best-estimate, nominal, peak cladding temperatures for the first and second peaks over the complete breach spectrum. This provides maximum first and second peak cladding temperatures, which are defined as Term 1, for the limiting breach case or cases. TRAC-B02 is a validated and verified code for BWR-LOCA calculation and GE is a well-qualified user of this code.

2) Comparisions of TRAC-B02 predictions with a wide variety of experiments under representative LOCA conditions enable bounding biases to be specified for:

   - uncertainties in TRAC due to modelling,
   - uncertainties in TRAC due to instrumentation/measurement uncertainties, and
   - uncertainties in TRAC due to scale-effects.

   Full-scale, separate-effects test data are used to support the relatively small bias on TRAC predictions at full-scale. The sum of these bounding biases is designated Term 2.

3) To account for uncertainties in plant operational parameters, such as linear heat generation rate, and to accommodate parameters not directly included in Term 2, such as decay-heat and stored-energy, a Term 3 is defined. The candidate plant parameters are selected by engineering judgement and sensitivity studies of these parameters are performed for the limiting breach cases. A response surface is fitted to the predictions. Random sampling of the parameter values using a Monte Carlo analysis provides a statistical distribution of peak cladding temperatures from the response surface. A bias on the mean and the variance are then readily determined.

   GE uses the SAFER/GESTR codes to develop the response surface. These are fast-running codes valided and verified for BWR-LOCA applications, and benchmarked against TRAC-B02. Their use greatly reduces the economic effort involved with the statistics, when compared with TRAC.

Finally, an upper-bound peak cladding temperature for the limiting breach case is calculated simply by summing the Terms 1, 2 and 3. Typical results are shown in Table 6.3. Although not strictly comparable, it is interesting to see how similar the data in Tables 6.2 and 6.3 turn out to be.

6.3.5 A Comparison of Quantitative Assessment Methods

1) An Overall Comparison

In order to compare the major points of the first three methods discussed above which, after all, do have the same objectives but use somewhat different means, a summary flow chart is provided in Fig. 6.6. This shows that the European GRS and Winfrith methods are very similar, differing only in the extent to which expert judgement is employed.

With the Winfrith approach the choice of model parameters as well as the choice of boundary conditions to be varied in the sensitivity analyses are dictated solely by expert judgement. The resulting uncertainty bands are determined simply by taking the maximum and minimum of the output values at any time. On the other hand the GRS uses probability distributions for uncertainties both in model parameters and boundary conditions, wherever this is feasible. These propagate through the assessment to give probability distributions for output parameters directly.

The American CSAU method is distinctly different from the other two. Whereas sensitivity studies are essential to the European methods and comparisons with test-data serve only to check that generated uncertainty bands are reasonable, comparisons of code predictions with test-data are central to the CSAU approach. With the CSAU the comparisons of key predicted parameters with data from the chosen test matrices provides the most important contributions to overall code uncertainties.

The CSAU method is in advance of the European approaches in that it has already been demonstrated to be practical, auditable and traceable in its application to a large-breach, PWR-LOCA. Although this methodology has been accepted by the research branch of the NRC, it remains to be exposed to a regulatory process. It has not yet been tested for application to small breaches, nor has it yet been applied to continuous-valued parameters.

2) Scale Effects

Codes are being assessed mainly against data from relatively small-scale test-rigs. The number of full-scale or near full-scale rigs is small and all of them are separate-effects facilities. These facilities and their scale distortions are discussed in Chapter 3. Plant data are scarce, and find only a limited application to code assessments for LOCA predictions. Except for the CSAU method, the quantitative code assessment methodologies are still under development. The methods for establishing the scaling capability of the codes are still under scrutiny.

Both of the European methods emphasise the use of large-scale, separate-effects test data to address scaling but, only for those phenomena considered not well enough understood to be handled on a theoretical basis. Indeed, three scaling categories are defined in the GRS method:

i) Full-scale separate-effects test data are available.
ii) Only small-scale separate-effects test data are available, but the effects of scale are known well enough to be handled by current theoretical means.
iii) Only small-scale separate effects test data are available, but no or inadequate theoretical means exist to extrapolate to full-scale.

Even with full-scale separate-effects data available, the scaling of boundary conditions for each separate-effect remains unclear. These boundary conditions are included in the integral tests, which tie all the separate effects together, but there are no integral-facilities at full-scale, other than the plants themselves.

In its first application to a large-breach LOCA, the CSAU methodology addressed the scaling problem directly and in some detail. But, ranking of processes and phenomena is exercised throughout this assessment, so that scaling is actually addressed only where it is considered important on the basis of expert judgement. Where the data-base is considered inadequate for assessing scale effects resort is made to bounding conservative assumptions.

To repeat what was stated at the end of Section 3.2.4, there is no simple means to account for scale. A judicious mixture of separate-effects tests, integral-tests, perhaps some counterpart tests, all supported by proven simulation models, is the scaling methodology that has to be, and is, employed in the three assessment methods.

6.4 Conclusions and Recommendations

The points raised in Section 6.2.3 for qualitative assessments are also pertinent to quantitative assessments:

- the need to define for what purposes the assessment is being carried out. It may be more reasonable to change the question of "how good is good enough" to "how good need it be"?
- the discipline required during code development and assessment to establish and retain acceptable frozen codes,
- the necessity to assess uncertainties introduced by the code user and from code mathematics,
- the lack of ideal experiments, ideal instrumentation and incomplete data bases, and
- the difficulties with scaling.

Even with these points satisfactorily resolved, many problems and difficulties remain with applying quantitative methodologies. For example, all of the methods to quantify uncertainty involve comparisons of calculated results and measured data. This is fairly straightforward for point-value functions, such as peak cladding temperature, but not so easy for continuous-valued parameters. For these, it may be very difficult to develop a defendable and simple mathematical method for deriving quantitative code accuracy statements, which are consistent with expert judgement. Several examples, which illustrate this point, are provided in Figs. 6.7 to 6.10. Methods to resolve the problem of continuous-valued parameters have been studied at the Sandia National Laboratory /6.6/ and the Idaho National Engineering Laboratory /6.7/ in the US, as well as at Pisa University /6.8/ in Italy. Their applications remain to be tested.

Additional issues, which are being addressed as the methodologies develop and are tested, include:

Winfrith Method

- Does the overriding use of expert judgement lead to opaque quantitative uncertainty statements which lack meaning?

- How are the criteria for acceptable uncertainty bands defined?

- Are all the sources of uncertainty addressed adequately? How is the uncertainty from scaling determined?

- What happens if the comparisons with test data are unacceptable, or if the bands on the final accuracy assessment are too wide to be useful?

- Frozen codes have to be modified so that model parameters can be altered through code input instructions.

GRS Method

The last three points in the Winfrith Method are also relevant to the GRS Method. Additional issues include:

- How many assessment cases (not to be confused with the 59 runs for statistical analysis of one power-plant transient) need to be calculated to assure that measured data are satisfactorily bracketed by generated uncertainties? How are validation matrices used?

- What is the selection basis for the parameters to be varied and how many need to be selected? On what basis are the probability distributions of model parameters generated?

- What sampling technique should be used for selecting parameter values?

In aiming to reduce reliance on expert judgement to a minimum by employing statistical methods throughout, the GRS Method exposes itself to a very large number of computer runs for each transient or LOCA to be analysed. A major question with this method is whether or not the total effort required is excessive.

<u>CSAU Method</u>

- How many calculations are needed to produce meaningful or useful results?

- What is the correct method for summing the various contributions to the total uncertainty?

- Can the method be tested; for example, by means of a blind standard problem exercise.

In the end any method is of value only if it can produce useful results with a reasonable and economic effort. An entire spectrum of breach sizes often has to be analysed in licensing calculations for a plant and the breach, which produces the highest peak cladding temperature, may not be known beforehand. Thus the calculational effort involved with an assessment must be practical for more than one breach size. The value and applicability of the quantitative methods described will be determined within the next few years as current programmes are completed. Then will be the time to decide whether additional, less-complex, more cost-effective methodologies need to be developed.

References for Chapter 6

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SAFER/GESTR Application Methodology"
June 1984

/6.2/ NUREG-1270
"International Code Assessment and Applications Program"
Vol. 1, March 1987

/6.3/ CSNI Report No. 17, Revision 2
"CSNI Standard Problems Procedures"
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/6.4/ CSNI Report No. 132
"CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR
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OECD/NEA/CSNI, March 1987

/6.5/ NUREG/CR-5249
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to a Large Breach LOCA"
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/6.6/ L.N. Kmetyk et al.
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Quantification Methodology"
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/6.7/ G.E. Wilson, G.S. Case
"Uncertainty Development and Application"
Thirteenth Water Reactor Safety Research Information Meeting,
October 22 - 25, 1985

/6.8/ F. D'Auria et al.
"Advancements in the Assessment of System Codes used for Safety
Analyses of Nuclear Power Plants"
Technical Committee/Workshop "Computer Aided Safety Analysis"
Warsaw, Poland, May 25 - 29, 1987

No.	Facility	Country	Type	Year	Subject
1	Edward's Pipe	UK	open	1975	Discharge, pressure waves
2	Semiscale	USA	blind	1975	Large break LOCA blowdown
3	CISE	Canada	open	1976	Pipe discharge
4	Semiscale	USA	blind	1976	6 % small large break LOCA
5	LOFT	USA	blind	1977	Isothermal large break LOCA
6	Battelle	FRG	open	1978	Discharge from top of vertical vessel
7	ERSEC	France	blind	1979	Reflood of heated tube
8	Semiscale	USA	open	1979	Large break LOCA
9	LOFT	USA	blind	1980	2.5 % small break LOCA
10	PKL-I	FRG	open	1980	Large break LOCA reflood
11	LOFT	USA	open	1981	2.5 % small break LOCA
12	ROSA-III	Japan	open	1982	5 % small break in BWR
13	LOFT	USA	blind	1983	Large break LOCA
14	REBEKA	FRG	blind	1984	Behaviour of fuel rod simulator bundle
15	FIX-II	Sweden	blind	1983	31 % break in BWR
16	HDR	FRG	blind	1984	Containment behaviour during large break LOCA
17	Marviken	Sweden	open	1984	BWR containment behaviour
18	LOBI-MOD2	Italy	blind	1985	1 % small break LOCA
19	PHEBUS	France	open	1986	Behaviour of fuel rod bundle
20	DOEL-2-PWR	Belgium	open	1987	Steam generator tube rupture
21	PIPER-ONE	Italy	blind	1988	1.6 %/2.8 % small break in BWR
22	SPES	Italy	blind	1988	Loss of feed water
23	HDR	FRG	blind	1988	Containment behaviour, including hydrogen injection
24	Sandia/SURC	USA	blind	1988	Core-concrete interaction
25	Achilles	UK	blind	1988	Effect of accumulator gas during LOCA reflood
26	ROSA-IV	Japan	open	1989	5 % small break LOCA

Separate Containment Analysis Standard Problems (CASP)

1	Battelle	FRG	open	1979	Steam discharge (1:4 Scale)
2	Battelle	FRG	open	1980	Water discharge (1:4 Scale)
3	AAEC	Australia	open	1982	Large breach LOCA (Small scale)

TABLE 6.1 INTERNATIONAL STANDARD PROBLEMS (ISP) OF THE OECD/NEA/CSNI

	PEAK CLADDING TEMP. (°C)		
	BLOWDOWN	REFLOOD	
		1st Peak	2nd Peak*
MEAN PEAK CLADDING TEMP. (Combined Uncertainty by Probability Distribution Function)	628	525	403
95 % PROBABILITY (Combined Uncertainty by Probability Distribution Function)	786	759	724
MEAN SEPARATE BIASES ADDED			
Hot-Channel Effects	35	14	-8
Dissolved Nitrogen	N/A	10	10
Nonconservative Implementation of Forsland-Rohsenow Correlation	26	47	89
Full-Scale Steam Binding Effects	N/A	-5	59
Full-Scale ECC Bypass Effects	N/A	-19	-19
COMBINED MEAN BIASES	61	47	131
TOTAL MEAN PEAK CLADDING TEMP. (including combined biases)	689	571	534
TOTAL 95 % PROBABILITY TEMP. (including combined biases)	847	806	855

N/A = Not Applicable

* TRAC did predict a double-humped reflood peak cladding temperature history, which may be a true representation of heat-balances on the core or a code artifact. In any case, the differences in the peaks are swamped by the uncertainty bands.

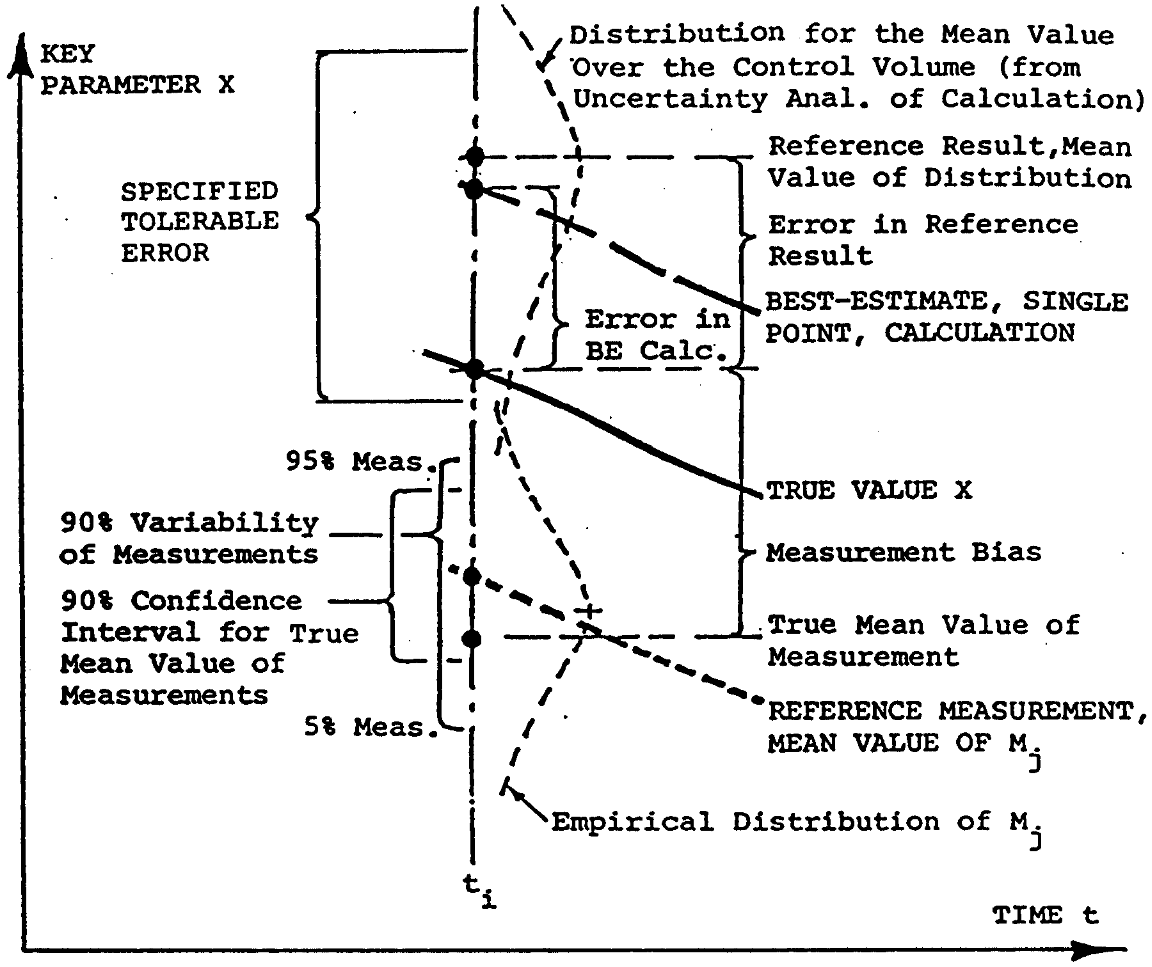
Application to a four-loop Westinghouse PWR for a cold-leg LBLOCA and using TRAC-PF1/MOD1, Version 14.3

TABLE 6.2 RESULTS FROM AN APPLICATION OF THE CSAU METHOD TO A COLD-LEG, LARGE-BREACH, PWR-LOCA

	PEAK CLADDING TEMP. (°C)	
	BLOWDOWN	REFLOOD
	1st Peak	2nd Peak
Term 1 NOMINAL PEAK CLADDING TEMP. (Calculated using TRAC-B02 for the limiting breach condition)	315	224
Term 2 BOUNDING BIASES FOR MODELLING UNCERTAINTIES		
- Modelling and instrumentation measurement uncertainties in TRAC-B02 for representative conditions	100	136
- Scaling bias in TRAC-B02 for representative conditions	28	28
COMBINED BOUNDING BIASES	128	164
Term 3 SENSITIVITY STUDY OF CANDIDATE PLANT VARIABLES (Calculated using SAFER/GESTR for limiting breach conditions)		
- Bias in the mean	15	3
- Two standard deviations	89	117
UPPER BOUND TEMP. (Term 1 + Term 2 + Term 3) at 95 % probability	547	508

Application to a large, recirculation-breach, LOCA on a GE-BWR/6-218 reactor using the codes TRAC-B02 and SAFER/GESTR.

TABLE 6.3 TYPICAL UPPER BOUND PEAK CLADDING TEMPERATURES CALCULATED BY THE GE METHOD



<u>The Situation at Time t_i</u>: A single TRUE VALUE for X; A single BEST-ESTIMATE prediction for X plus values from uncertainty analysis parameter variations; Several MEASUREMENTS M_i of X.

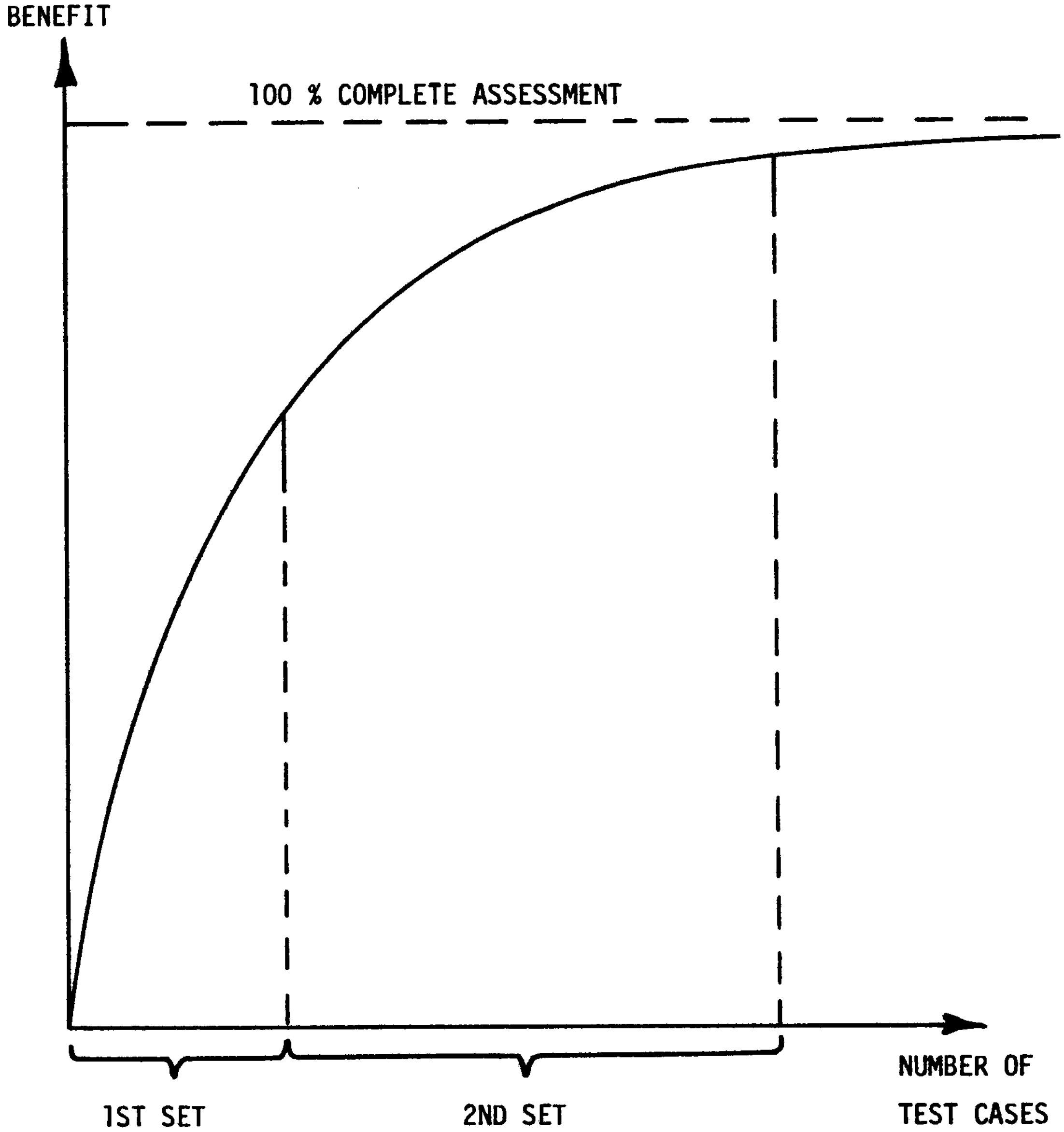
The measurements are either subject to measurement error only, or to measurement error <u>and</u> variability within the control volume used in the code. Computed and true values of X are to be the mean value over this control volume.

<u>Accuracy Statement</u>:

This requires a specification of the tolerable error. The code prediction X is "accurate enough" at a confidence level of 90 % if the interval about X, defined by the tolerable error, contains a 90 % confidence interval for the true value. In the above situation, the true value is the mean value over the control volume. A confidence interval is derived from a random sample of measurements (corrected for bias) from this control volume. Note that 90 % is here used for illustration, usually confidence levels of 95 % or larger are employed.

If expert judgement is used to account for imperfections in experiments and for measurement error contributions to the uncertainty, confidence levels and intervals are to be called "subjective".

FIG. 6.1 DEFINITIONS OF ACCURACY AND UNCERTAINTY



Completion of a small number of the best test cases (1st set) may reveal major deficiences in the code and thereby reduce the risk of having to repeat a large number of calculations.

FIG. 6.2 PROCEDURE TO MINIMIZE THE IMPACT OF CODE DEFICIENCES ON AN ASSESSMENT PROGRAMME

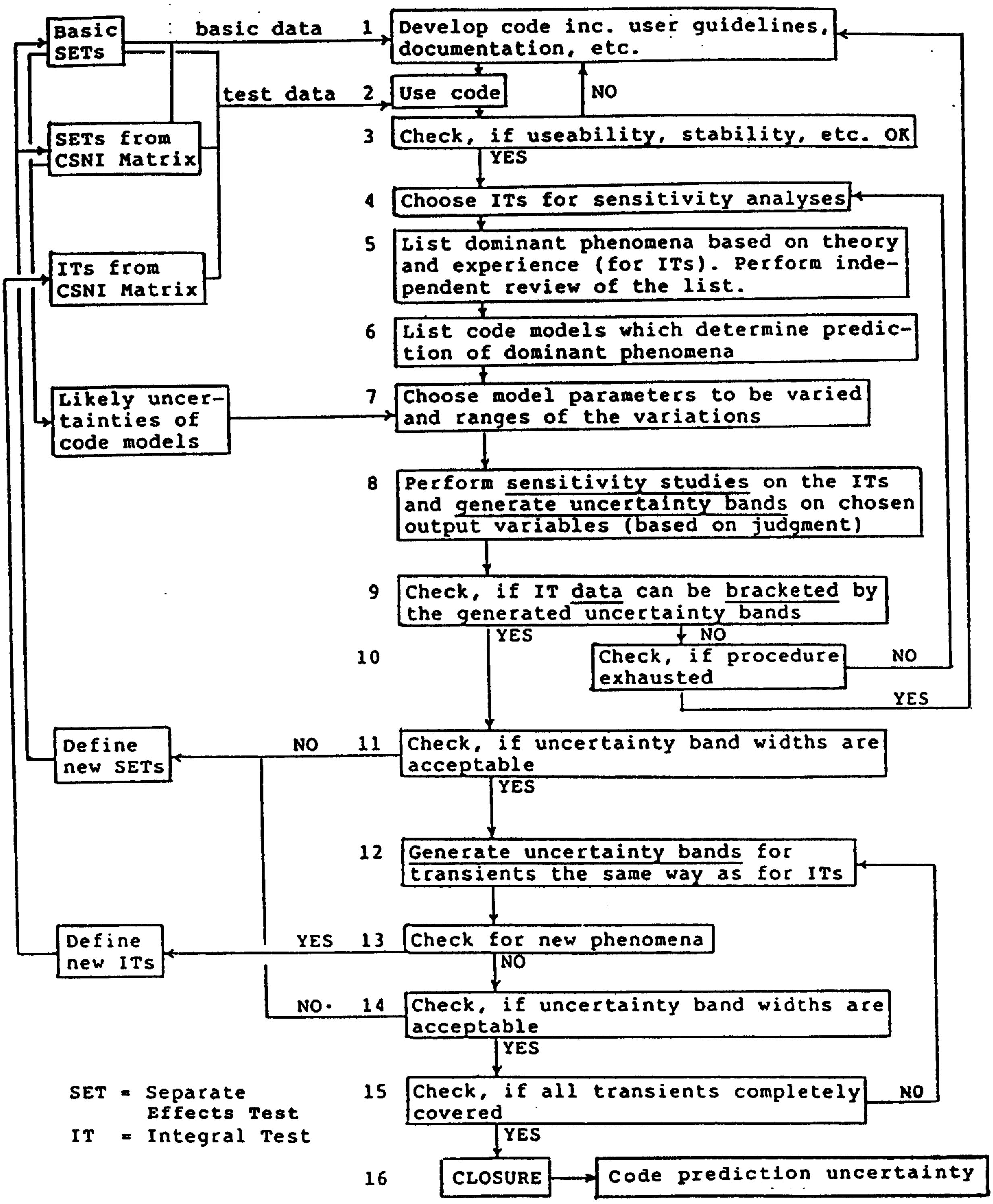


FIG. 6.3 FLOWCHART OF THE UKAEA, WINFRITH CODE ASSESSMENT METHOD

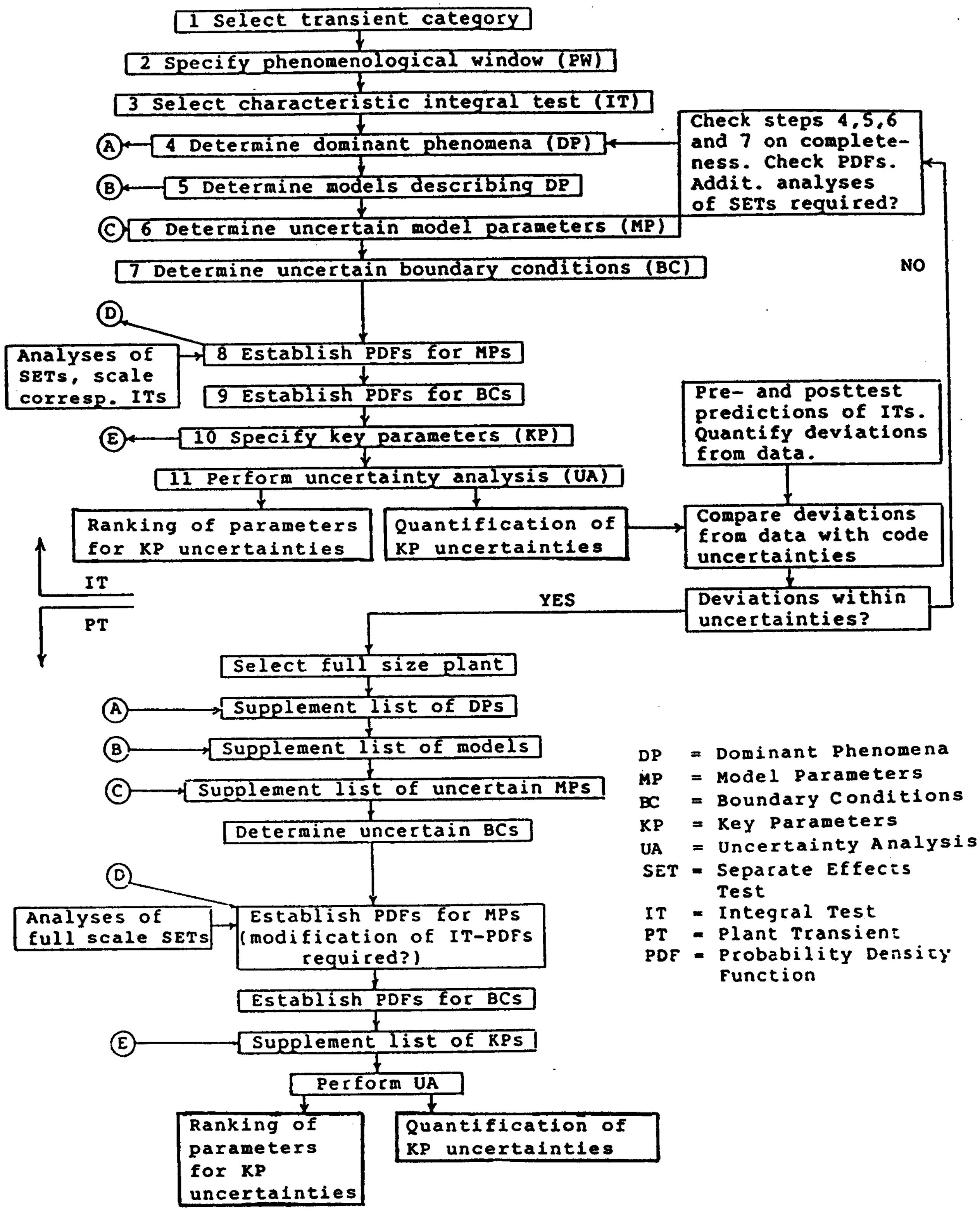


FIG. 6.4 FLOWCHART OF THE GRS, MUNICH CODE ASSESSMENT METHOD

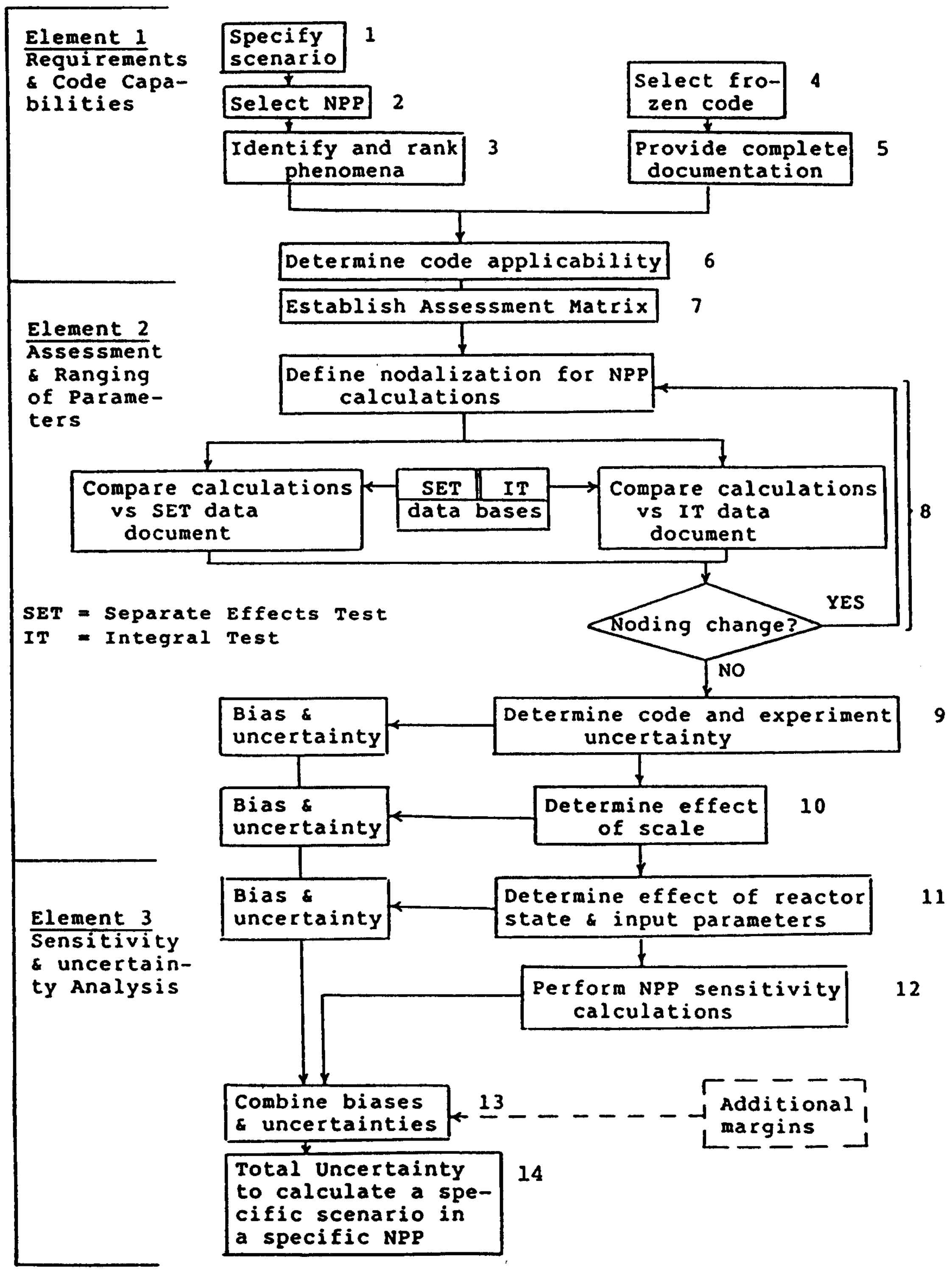
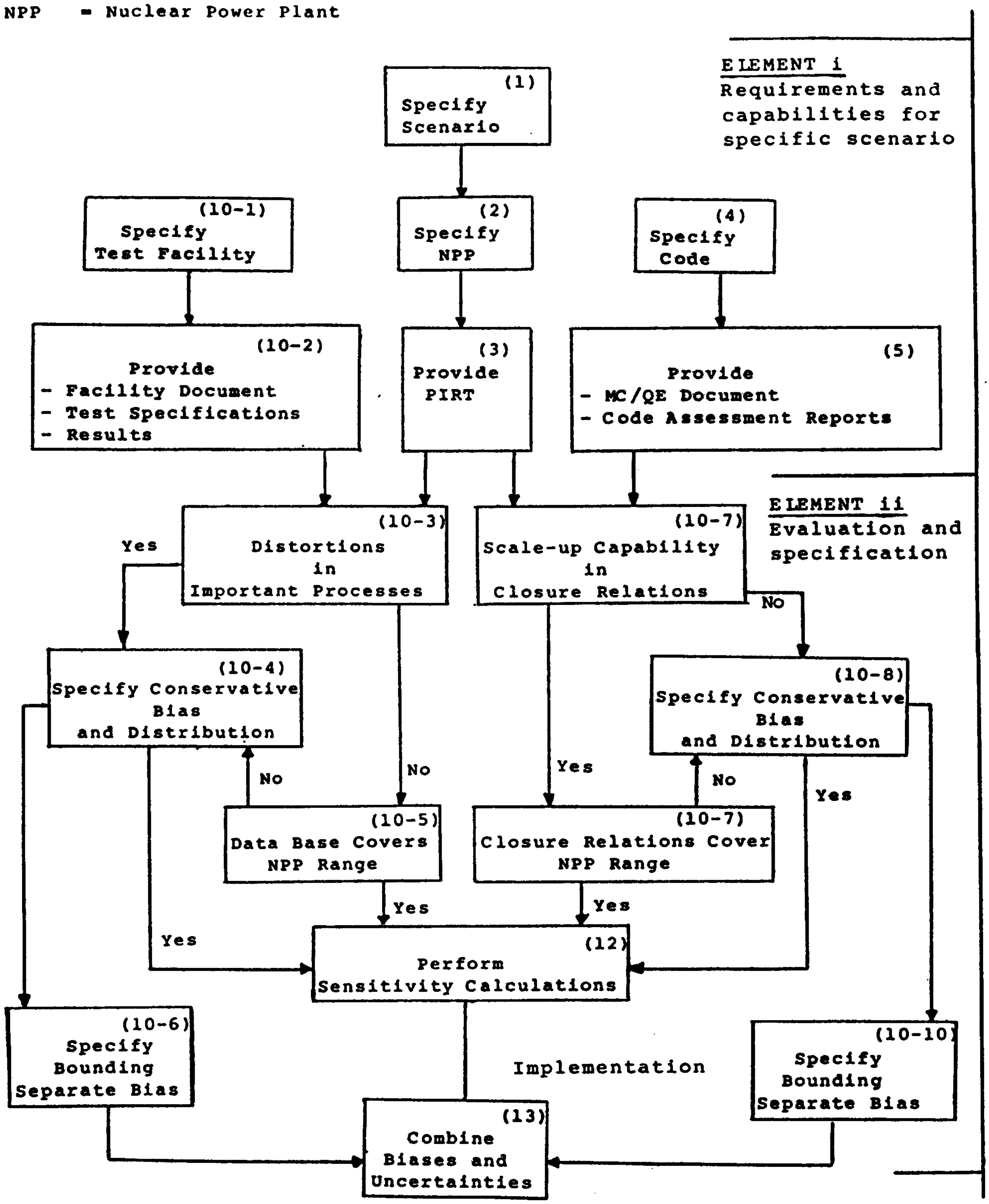


FIG. 6.5.1 FLOWCHART OF THE USNRC's CODE SCALING, APPLICABILITY AND UNCERTAINTY (CSAU) EVALUATION METHODOLOGY

MC/QE = Model Correlations / Quality Evaluation
NPP = Nuclear Power Plant



Notes

- Steps (1) to (5) and (12) to (13) are from the CSAU procedure.
- Steps (10-1) to (10-10) are the procedures used to evaluate code scale-up cabability.

FIG. 6.5.2 GENERIC PROCEDURE TO EVALUATE CODE SCALE-UP CAPABILITY IN THE CSAU METHOD

American Method (CSAU)

Specify scenario (windows), power power plant (NPP), code

↓

Rank processes, phenomena (using e.g. Analytical Hierarcy Process)

↓

Perform diagnostic analysis (requirements vs code capabilities including scale up, based on documents)

↓

Establish Assessment Matrix (AM)

Define NPP nodalisation

↓

Calculate the SET's & IT's of the AM

↓

Perform qualitative assessment (Consistency, threshold phenomena, scaling distortions?)

Perform quantitative assessment (Determine code accuracy - bias and uncertainty - by comparing test data against code results for key parameters)

Determine effect of scale and effect of reactor parameters

↓

Perform NPP sensitivity calculations

↓

Combine biases and uncertainties
Produce TOTAL UNCERTAINTIES

European Method (GRS & Winfrith)

Specify transient, window, import. phenomena, models, model parameters

↓

Calculate SET's (Separate Effects Tests)
Produce uncertainties of model parameters (MPs)

↓

Determine uncertainties in boundary conditions (BCs)

↓

Perform calculations on selected IT's varying MPs and BCs
Produce uncertainty bands for most important output parameters

↓

Examine, if test data are bracketed by the produced uncertainties
Determine, whether the uncertainties are small enough

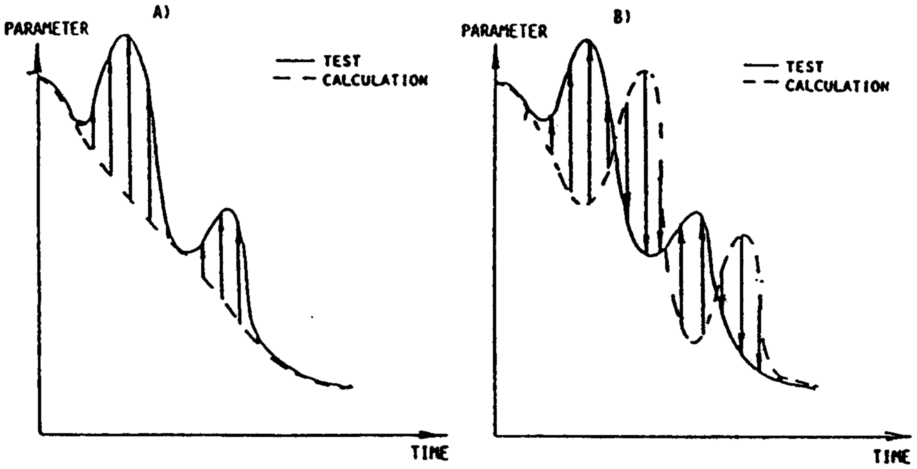
Perform NPP calculations varying MPs and BCs
Produce uncertainties for important parameters

↓

Determine, whether the uncertainty bands are narrow enough
Produce TOTAL UNCERTAINTIES

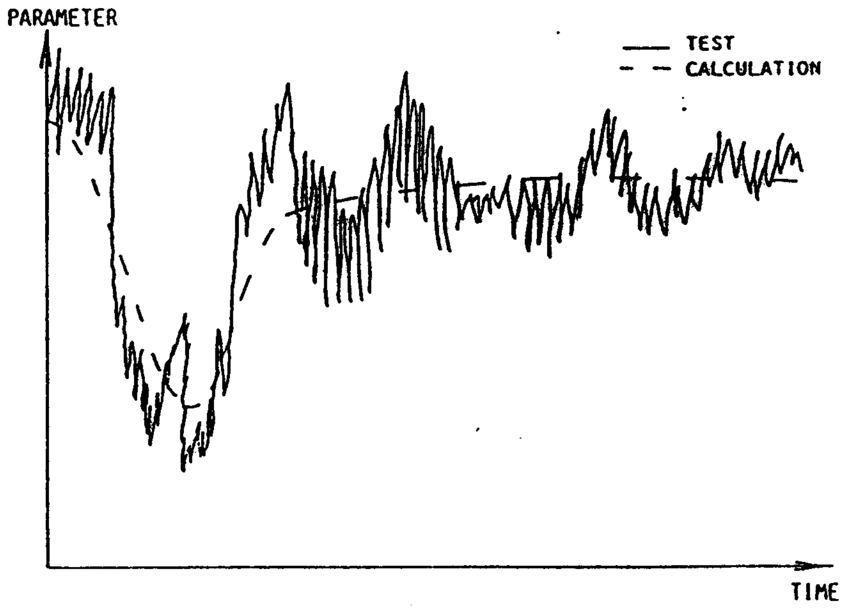
AM = Assessment Matrix
MP = Model Parameters
BC = Boundary Condition
NPP = Nuclear Power Plant
IT = Integral Test
SET = Separate Effects Test

Fig. 6.6 A COMPARISON OF CODE ASSESSMENT METHODOLOGIES



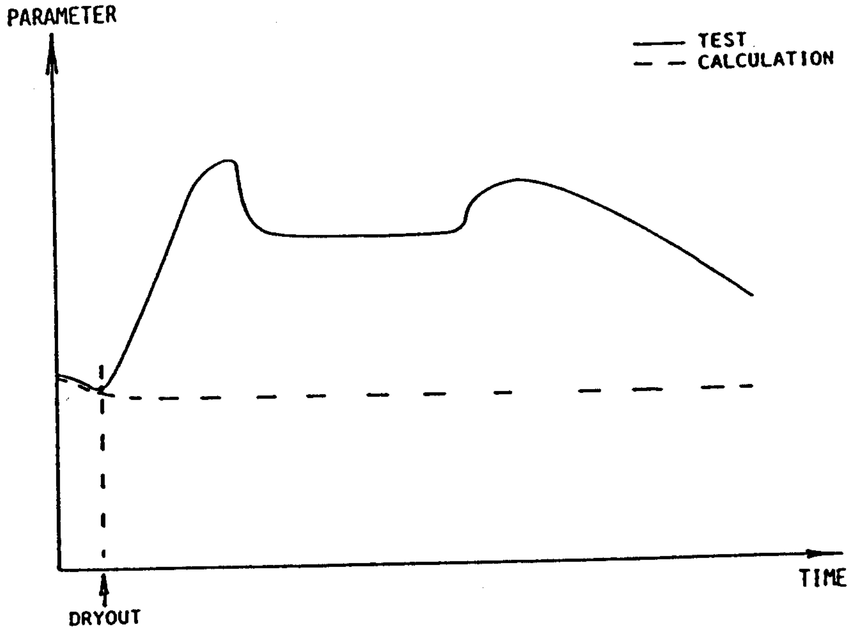
If the timing of the phenomena in the test and in the calculation is different, a quantitative comparison, based on a mechanical integration of absolute differences, for example, may give results (A better than B) which contradict expert judgement (B better than A).

FIG. 6.7 THE INFLUENCE OF TIMING ON QUANTITATIVE DATA COMPARISON



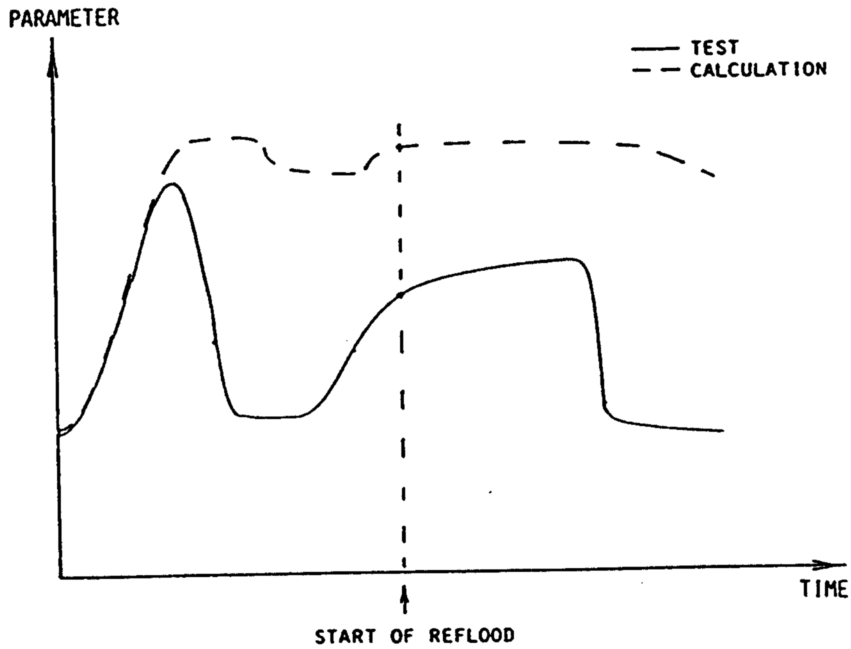
Filtering may be needed when comparing oscillating parameters.

FIG. 6.8 THE INFLUENCE OF OSCILLATIONS ON QUANTITATIVE DATA COMPARISON



Dryout is an example of a threshold phenomenon. After dryout the test and calculation may deviate very much, even if the agreement before and at the threshold point was very good.

FIG. 6.9 THRESHOLD PHENOMENA AND DATA COMPARISON



It may be meaningless to make code comparisons with data during a particular time period, the reflood period in this example, if there is a large discrepancy already at the beginning of the period.

FIG. 6.10 HISTORY EFFECTS AND DATA COMPARISON

7. CONCLUDING REMARKS AND RECOMMENDATIONS

7.1 General Conclusions

So where do we stand today with our understanding of the thermohydraulics of emergency core cooling for design basis LOCA and transient scenarios in light water reactors? How good are our analytical tools for assessing qualitative or quantitative margins to safety limits whilst maintaining economic plant operation and, how accurate need these tools be?

As demonstrated in Chapters 2 and 3, PWR- and BWR-LOCA scenarios and phenomenology are, on the whole, very well understood and strongly supported by experiments. Because of their complexity, not all of the phenomena are readily amenable to quantitative analysis, even with modern two-fluid analytical tools. Where a suitable modelling capability is lacking, experiments provide the required local or global empirical correlations. This is important because the level of conservatism or margin that needs to be placed on a qualitative assessment, or the level of confidence that can be placed on a quantitative assessment, depends not just on the understanding of the phenomena, but also on the level of empiricism involved in their analyses, the adequacy of the data base employed and the scaling distortions involved.

The weighting and appraisal process described in Chapter 4 puts all of these points into a consistent and transparent framework. In this respect the weighting and appraisal tables so developed are perhaps the most important part of this SOAR. They play a key role in connecting our understanding of LOCA phenomena with the requirements for model developments discussed in Chapter 5, with the need for additional experiments and with the developments in code assessment described in Chapter 6.

A point worthy of note is that the tables are largely based on expert opinion or judgement, which is supported by the experiments described in Chapter 3 as well as by parameter studies. In an ideal world there would be direct feedback from quantitative code assessment defining the research and development requirements. But, as indicated in Chapter 6, quantitative assessment is complex and all the methods described there rely to some degree or another, but still quite extensively, on expert judgement too. Thus, as already noted by the ACRS /1.5/, care must be taken not to give too much significance or weight to exact confidence statements based on rigorous statistical relationships, when all that can really be claimed is a reasonably high confidence level. For most practical, design basis, ECC problems, and particularly those involved with safety analyses, this is perfectly adequate; confidence levels can always be raised by introducing more conservatism or by adding margins.

There is of course a continuous feedback from both quantitative and qualitative assessments, particularly during the code validation exercises, pointing to modelling areas which could do with improvement. These areas were expanded upon in some detail in Chapter 5. Deficiencies in the two-fluid model closure laws, particularly with regard to interfacial relationships, were noted for many phenomena. In some cases, an example is quench front propagation, even where significant progress has been made in modelling phenomenological details, the mesh or noding size required to achieve this is impractical for use in a system code analysing a full-scale plant. Model developments are also closely interwoven with the limitations of existing instrumentation to measure parameters at the interface level in complex two-phase flows, On the other hand, comparisons of code predictions /5.1/ have shown that orders of magnitude differences in interface transport terms may appear between codes, and yet lead to very similar predictions of safety relevant parameters such as peak cladding temperatures.

Then do we really need to sharpen our analytical tools any further? As pointed out in Chapter 6, the general question of how good is good enough should be changed to how good need it be for a particular application. Only after this question has been considered from a qualitative and/or quantitative viewpoint can a judicious allocation of resources be made for further developments in the areas noted in Tables 4.2 and 4.3.

Most assessments of design-basis ECC effectiveness were, and still are, qualitative. This is acceptable provided:

- all relevant phenomena are identified,
- the domain of applicability of correlations is adequate,
- separate effects are correctly combined in space and time, and
- sufficient allowances, conservatisms or margins are provided for uncertainties.

The qualitative approach rests heavily on engineering judgement, and in it much attention is focussed on conservatism to ensure large, but often unspecified, margins to safety limits. But, as we have seen, the quantitative approach also rests heavily on expert judgement and really only differs from the qualitative approach in its treatment of the last item.

The first applications of quantitative assessment to typical, large-breach, PWR-LOCA scenarios give peak-cladding temperatures of about 600 °C with an uncertainty of about 250 K at a claimed 95 % probability, Table 6.2. These very expensive applications indicate that qualitative approaches are too conservative by typically 300 K, which may represent a large economic penalty for some plants, but not necessarily for others. If peak-cladding temperatures of 800 °C were predicted, an uncertainty of 250 K would be unacceptable, for example, for a quantitative assessment of cladding oxidation and rupture. This may not be a problem if temperatures are anyway so low, Table 6.3, that no ruptures occur but, as economic penalties are removed and predicted temperatures are increased, such an uncertainty may no longer be acceptable. Note that it could be improved or worsened simply by changing the chosen (by expert judgement) probability limit itself! The situation worsens as conditions beyond the design basis are investigated; most system codes have not been validated for beyond design basis conditions and uncertainties then will tend to be much larger.

Similar arguments may be put forward when using system codes, say, to benchmark full-scope simulators or to support operator procedures. Within the LOCA design-basis, operator procedures are anyway fairly straightforward and hardly need quantitative support with a high degree of precision. This changes when beyond design-basis scenarios are considered, but then code prediction uncertainties increase rapidly.

An overall conclusion then is that best-estimate system codes and quantitative assessments are valuable, but complex and expensive tools for assessing design-basis transient and LOCA margins to safety limits. For small analysis groups with limited resources, it may be practical to perform a quantitative assessment only for a limited number of plants and cases. From a safety point of view qualitative assessment may do just as well. On the other hand, best-estimate system codes do offer a much more realistic picture of plant transient and LOCA behaviour, which is beneficial to all aspects of safety. They provide a good vehicle for evaluating measured data, they give margins to safety limits with a high degree of confidence and are thus very suitable for evaluating any unnecessary economic penalties. An attempt at a quantitative assessment is in itself a very good and useful exercise which, at the least, will throw light upon the scope of the problem and bring discipline to the code user. Any precision claimed for code prediction accuracy should, however, be treated with care, despite the sophisticated statistical techniques employed, because expert judgement, an unquantified parameter, is still very much in evidence.

7.2 Recommendations

Most of the important recommendations with regard to ECC experiments, analytical models and data base improvements are gathered in Sections 3.5, Tables 4.2 and 4.3, and Section 5.13. It is clear from these recommendations that a very wide range of theoretical and experimental programmes could be proposed. But these recommendations should not be treated as open-ended research projects, although in the two-phase flow area this may be the case anyway.

On completion of this report members of the Writing Group offer the following recommendations:

1. A feedback is needed in terms of desired accuracy for a given problem or set of problems, achievable accuracy as defined by a quantitative assessment and, the research and development required to match the two. This may be easier said than done, but even an attempt at it can only improve the management of limited research and development resources. In this respect, international cooperation should continue and be encouraged to make the widest possible use of these resources.

2. It is clear from Chapter 3 that a vast amount of data on ECC phenomena has been gathered. These data need to be scrutinized before embarking on additional experimental programmes. Recommendations to expand data bases in specific areas are given in Table 4.2. In other areas there is a need for a more thorough examination of the existing data, some specific recommendations are made in Table 4.3. This examination may also help to resolve the following issue.

3. A great deal of attention is focussed in Chapter 5 on the deficiences in modelling some ECC phenomena, but their impact on key safety parameters is not quantified. It may indeed be small for the design-basis licensing of ECC, but not necessarily so for all situations or for beyond the design basis. It is recommended that efforts be made to quantify these impacts to provide guidance on where model improvements are really needed.

4. As far as quantitative assessment is concerned, we are still in the learning phase. The problems noted and questions posed in Section 6.4 need to be addressed. It is strongly recommended that the uncertainty assessment methodologies developed by Winfrith and GRS be completed and tested in a manner similar to the CSAU and that other approaches, if and when available, do the same. Comparisons between these methods and their predictions will illuminate the strengths, weaknesses and costs involved. Another recommendation here is that means be considered to reduce the rather ill-defined uncertainities introduced by the system-code user. For example, guide-lines need to be established defining how a code user is in fact qualified.

5. Finally, we recommend that the general question how good is good enough be changed to how good need it be <u>for a particular application</u>. The uncertainties calculated by any of the methodologies described in Chapter 6, apply strictly to those applications actually used in the LOCA studies. Key parameters and accuracy requirements may be quite different for other applications.