IAEA-TECDOC-1351

Incorporation of advanced accident analysis methodology into safety analysis reports

INTERNATIONAL ATOMIC ENERGY AGENCY

May 2003

The originating Section of this publication in the IAEA was:

Safety Assessment Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100
A-1400 Vienna, Austria

INCORPORATION OF ADVANCED ACCIDENT ANALYSIS METHODOLOGY
INTO SAFETY ANALYSIS REPORTS
IAEA, VIENNA, 2003
IAEA-TECDOC-1351
ISBN 92–0–103803–8
ISSN 1011–4289

Printed by the IAEA in Austria
May 2003

FOREWORD

Safety analysis reports (SARs) represent the most comprehensive compilation of specific information relevant to the safety of a nuclear power plant (NPP). They serve as the basis for assessment of the overall NPP safety status, as well as for modifications and improvements of plant hardware, procedures and operation. For the regulatory body the SAR is the essential document for the licensing process.

Deterministic safety analysis (frequently referred to as accident analysis) is an essential part of any SAR. Due to the close interrelation between accident analysis and safety, an analysis that lacks consistency, is incomplete, or of poor quality, is considered a safety issue for a given NPP. Developing IAEA guidance for accident analysis is thus an important step towards resolving this issue.

Requirements and guidance pertaining to the scope and content of accident analysis have, in the past, been partially described in various IAEA technical publications. Accident analysis is also covered in several IAEA Safety Standards, for example in the Safety Requirements on Design and in the Safety Guide on Safety Assessment and Verification. In consistence with these publications, the IAEA has developed a detailed Safety Report on Accident Analysis for Nuclear Power Plants, which is aimed at providing practical guidance for performing accident analysis. The report covers all the steps required for accident analyses, i.e. selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data and presentation of the calculation results. Advanced best estimate computer codes are suggested in all IAEA guidance publications to be used for safety analysis.

The current TECDOC complements the Safety Report on Accident Analysis for Nuclear Power Plants and presents additional guidance for specific application of advanced computer codes in accident analysis needed for the SARs. It provides an overview of presently available advanced computer codes for various technical disciplines relevant for reactor accidents and discusses the application of such codes for various categories of accidents. The report is intended for use primarily by the developers or reviewers of deterministic safety analysis for the SAR of a NPP, both on the utility as well as on the regulatory side.

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CONTENTS

1. INTRODUCTION

1.1. Background

The IAEA Safety Guide on Safety Assessment and Verification [1] defines that "the aim of the safety analysis should be by means of appropriate analytical tools to establish and confirm the design basis for the items important to safety, and to ensure that the overall plant design is capable of meeting the prescribed and acceptable limits for radiation doses and releases for each plant condition category". Practical guidance on how to perform accident analyses of nuclear power plants (NPPs) is provided by the IAEA Safety Report on Accident Analysis for Nuclear Power Plants [2].

The safety analyses are performed both in the form of deterministic and probabilistic analyses for NPPs. It is customary to refer to deterministic safety analyses as accident analyses. This report discusses the aspects of using the advanced accident analysis methods to carry out accident analyses in order to introduce them into the Safety Analysis Reports (SARs).

In relation to the SAR, purposes of deterministic safety analysis can be further specified as

(1) to demonstrate compliance with specific regulatory acceptance criteria;

(2) to complement other analyses and evaluations in defining a complete set of design and operating requirements;

(3) to identify and quantify limiting safety system set points and limiting conditions for operation to be used in the NPP limits and conditions;

(4) to justify appropriateness of the technical solutions employed in the fulfillment of predetermined safety requirements.

The essential parts of accident analyses are performed by applying sophisticated computer code packages, which have been specifically developed for this purpose. These code packages include mainly thermal-hydraulic system codes and reactor dynamics codes meant for the transient and accident analyses. There are also specific codes such as those for the containment thermal-hydraulics, for the radiological consequences and for severe accident analyses. In some cases, codes of a more general nature such as structural analysis codes and computational fluid dynamics codes (CFD) are applied.

The initial code development took place in the sixties and seventies and resulted in a set of quite conservative codes for the reactor dynamics, thermal-hydraulics and containment analysis. The most important limitations of these codes came from insufficient knowledge of the physical phenomena and of the limited computer memory and speed.

Very significant advances have been made in the development of the code systems during the last twenty years in all of the above areas. If the data for the physical models of the code are sufficiently well established and allow quite a realistic analysis, these newer versions are called advanced codes.

The assumptions used in the deterministic safety analysis vary from very pessimistic to realistic assumptions. In the accident analysis terminology, it is customary to call the pessimistic assumptions ‘conservative’ and the realistic assumptions ‘best estimate’. The assumptions can refer to the selection of physical models, the introduction of these models into the code, and the initial and boundary conditions including the performance and failures of the equipment and human action.

The advanced methodology in the present report means application of advanced codes (or best estimate codes), which sometimes represent a combination of various advanced codes for separate stages of the analysis, and in some cases in combination with experiments.

The Safety Analysis Reports are required to be available before and during the operation of the plant in most countries. The contents, scope and stages of the SAR vary among the countries. The guide applied in the USA, i.e. the Regulatory Guide 1.70 [3] is representative for the way in which the SARs are made in many countries. During the design phase, a preliminary safety analysis report (PSAR) is requested in many countries and the final safety analysis report (FSAR) is required for the operating licence. There is also a need to update the FSAR periodically (UFSAR) for holders of an operating licence and the corresponding guidance is being developed, see Ref. [4].

1.2. Objectives and scope

The first objective of this report is to give a short overview of the advanced codes that are available and are currently used for accident analyses of NPPs. The main tools for the accident analyses are thermal-hydraulic system codes. The other code types used for various purposes will be also discussed briefly.

The second objective is to discuss the application of such codes for the analyses to be presented in the SAR of an individual plant.

The report is applicable to the advanced codes to be used in the analysis of the plants that are mainly based on light water technology and to a certain extent to the pressurized heavy water reactor designs (CANDU). The report is generally applicable to existing plants as well as to new reactor power plants. It is noted, however, that most of the examples discussed here are connected to the pressurized water reactor (PWR) technology.

The report can be considered as a complementary publication to the IAEA Safety Report on Accident Analysis for Nuclear Power Plants [2], describing in more detail the use of computer codes for specific applications needed for the SAR.

1.3. Structure

Section 2 of this report gives an overview of the existing codes for thermal-hydraulics, reactor dynamics, containment analysis, severe accident analysis and other areas included in the scope of analyses and computation to support the SARs.

Section 3 describes the use of advanced methods for various transient and accident analyses to be included in the SARs. The special emphasis is on describing the methods which are used and how to achieve a reliable and conservative evaluation of safety margins.

Section 4 discusses the use of advanced methods to support the justification of safety in the various design-related applications included in the SARs.

In Section 5, the basic requirements for successful application of the best estimate methods and how the associated uncertainties can be quantified and managed in case of using advanced methodology are discussed.

In addition, a short guidance is presented regarding preparatory steps for SAR elaboration in Appendix I. This guidance is based on experience gained while preparing SARs for completion and modernization of PWRs. The process of the preparation of nodalization for computational analysis by means of thermal-hydraulic codes is presented in Appendix II. An example is given in Appendix III on the data transfer and the interface management between various stages of a complex analysis of pressurized thermal shock. The complications concerning the definition of the limiting cases for transient analyses are addressed in Appendix IV. Finally, Appendix V lists as an example the parts of the final safety analysis report, where advanced analysis methodology can be applied to justify the basic design solutions in accordance with the US Regulatory Guide 1.70 [3].

2. DESCRIPTION OF COMPUTER CODES

2.1. Thermal-hydraulic system codes

The thermal-hydraulic system codes have been developed and used for analysis of loss of coolant accidents (LOCAs) since thirty years. At the end of the sixties and beginning of the seventies the capabilities of the codes were strongly limited by lack of experimental data, details of modelling and the capacity of the computers. Since then, the situation has improved significantly on all these areas. New generations of the system codes have been developed in many countries.

Thermal-hydraulic system codes of the current generation are based on solving mixed hyperbolic-elliptic system of six conservation equations (conservation of mass, energy and momentum for the vapour and liquid phases). Experimentally based correlations that are often called ‘constitutive laws’ are incorporated to describe the needed boundary conditions for each of the phases, such as friction between the phases and the wall. The codes are typically developed for one dimensional modelling. However, there have been many efforts to introduce simulation of multidimensional behaviour and effects into these two-fluid codes. Such strategies as well as their needs and benefits were recently discussed by Bestion and Morel [5].

Extensive experimental databases for validation of these codes do exist. These databases comprise a large number of separate effects and integral tests carried out in many countries both in the national and international programmes. The databases have been defined as an OECD effort into the code validation matrices [6]. Many features which were considered problematic in the previous code generations have now been introduced, such as application of real two-fluid models and developing of the constitutive laws, resulting in better modelling of interfacial friction and of the various counter-current flow limitation (CCFL) situations. The development and application of best estimate accident analysis codes was recently reviewed by Reocreux [7]. Several examples of the currently used system codes are shortly introduced in Table I.

TABLE I. EXAMPLES OF CURRENTLY USED THERMAL-HYDRAULIC SYSTEM CODES

Code	*Model*	*Country*	*Numerical method*	*Specific features*
APROS [8]	**6-eq. model** (fast-running version with 5-eq. model and drift flux)	Finland	semi-implicit	Modular structure, developed for the plant analyzer use. Development and validation in line with the PACTEL integral experiments, special efforts for WWER validation. Includes options for 1D and 3D reactor dynamics models
ATHLET [9]	**5-eq. model with drift flux** (6-eq. model for reflood calculations)	Germany	fully implicit	Applicability to the combined ECCS injection to the both cold and hot leg
CATHARE [10]	**6-eq. model** (boron concentration, 4 gas concentrations)	France	1D: fully implicit 3D: semi-implicit	Development supported by large number of own separate effect experiments and in-line with the BETHSY integral experiments. 0D, 1D, 2D, 3D, modular structure
CATHENA [11]	**6-eq. model**	Canada	one-step semi-implicit	Developed for CANDU applications
RELAP5 [12]	**6-eq. model** (boron concentration, 5 gas components, reflood model (axial fine-mesh))	USA	semi-implicit or two-step nearly-implicit	Special process models. Widest international acceptance D, 2D (Framatome), 3D (INEEL)
TRAC [13]	**6-eq. model** (1D, 3D)	USA	SETS	3D features, specific versions for PWR and BWR

For simulation purposes, the graphic presentations have turned out to be very helpful for the code user in order to ensure quality of the input and to obtain better understanding of the situation analysed. Therefore, the current analysis codes also include advanced graphical presentation features.

The development and validation of the current codes have proceeded to a level that they can be regarded as reliable analysis tools for most of the anticipated operational occurrences and postulated accidents. In particular, they have made it possible to transfer the analyses from the very conservative evaluation models (EM) to more realistic approaches that are often called best estimate (BE) methods.

A precondition for successful application of these advanced methods is that the crucial phenomena during the accident to be analysed can be governed by the models included in the codes. That typically means that there is not a direct need for detailed neutronic modelling, the mixing within the phases in the single volume can be assumed to be perfect (or governed by special modelling as in the pressurizer volumes), and the 3-D effects can be sufficiently governed by the included models or appropriate nodalization.

In spite of the development made in the advanced codes, they are not yet fully capable of a realistic description of the physical phenomena that occur during the accidents. The complex phenomena due to their multidimensional and micro scale features cannot be modeled in detail. Many different condensation processes are an example of a situation, where the accuracy may be problematic. The common approach to overcome this difficulty is to apply conservative assumptions, whenever they can be specified.

Perhaps the most serious limitations of the system codes are related to those situations, where the density or temperature stratification inside any of the phases becomes important, i.e. the phases are not well mixed. Examples, where such phenomena are crucial, are the turbulent mixing and stratification phenomena important for the pressurized thermal shock (PTS) thermal-hydraulics (thermal-hydraulic analyses supporting the pressurized thermal shock studies), inhomogeneous boron dilution transients, steam line break accidents and anticipated transients without scrams (ATWS) accidents. The difficulties in those cases are related on one hand to the disability of introducing stratified layers. On the other hand, there is no modelling for turbulent diffusion, but at the same time the numerical diffusion can cause by far too excessive mixing. Therefore, specific methods are often applied to those phenomena: the boundary conditions for the accident situation are taken from the system code calculations.

There are modelling limitations that are related to the solving procedures of the six equations. There has been an effort to take a different approach by developing the separation of the flow according to the velocity (SFAV) method [14], which considers two regions of the channel cross-section, each possibly containing a mixture of vapour and liquid phases instead of considering complete separation of the phases. Within each of the regions, both components in the mixture are considered to have the same velocity. As a result, a hyperbolic equation system is obtained that can be solved without introducing such artificial concepts as virtual mass. In addition, the numerical diffusion can be avoided.

2.2. Reactor dynamics codes including coupled codes

The transients and accidents, which are featured by the criticality concerns, need a detailed presentation of the core neutronics. Such events have been traditionally analysed with the reactor dynamics codes. Because of the computer capacity limitations, these codes often did not include the detailed thermal-hydraulics for the reactor circuit itself. In most analysed transients, the situation is considered satisfactory as long as there is a well-mixed single-phase flow in the primary circuit.

The analysis needs for more complicated thermal-hydraulic progressions with the criticality concerns gave an incentive to couple the 1-D and 3-D reactor dynamics models with an advanced thermal-hydraulic modelling. The first usage of the coupled codes was to study the consequences of the recriticality during the steam line break (SLB) accidents, and the feedback effects during the ATWS accidents under two-phase flow conditions. The studies of inhomogeneous boron dilution events have also benefited of the availability of the coupled codes.

HEXTRAN [15] was the first coupled code being taken into production use in the beginning of the 1990s. The coupling of the whole thermal-hydraulic model SMABRE [16], and the core model HEXBU [17], has been done in the core inlet.

In general, the incorporation of full 3-D modelling of the reactor core into system transient codes allows ‘best estimate’ simulations of interactions between reactor core behaviour and plant dynamics. Recent progress in computer technology makes the development of such coupled code systems feasible. During the last years, several coupled code systems have been established in numerous countries and are currently used for licensing purposes. Examples of such code systems are HEXTRAN/SMABRE [18], DYN3D/ATHLET [19], KIKO3D/ATHLET [20], RELAP5/PANBOX [21], RELAP5-PANTHER [22], RELAP5/PARCS [23] and RELAP5-3D® [24] (with the NESTLE neutron kinetics model).

2.3. Containment thermal-hydraulics codes

The containment thermal-hydraulic codes have been traditionally developed as the lumped parameter codes that model thermal-hydraulic behaviour of the containment compartments and heat structures. The term 'lumped parameter' is used to describe the modelling techniques: the containment is divided into compartment volumes and heat structures. Each volume has one temperature (or two, for thermodynamic non-equilibrium model) and pressure value, and the flows between the volumes are governed by the pressure difference. When the noding is denser than the natural division of the compartments by walls, special efforts are needed to limit the excessive flows and instability of the calculation. Widely applied lumped parameter codes are CONTAIN [25], WAVCO [26] and RALOC [27] (and its recent update COCOSYS [28]).

The limitations of the lumped parameter codes have initiated many attempts to develop a more detailed code system for the fluid flows inside the containment. The improvement in the computer capacity and speed has made it feasible to apply a specific type of CFD methodology to the containment calculations such as GOTHIC [29].

The application area of the containment codes is:

- maximum pressure and temperature analysis during the LOCA and SLB accidents,
- minimum back pressure analysis during large break (LB)-LOCA as a boundary condition for core reflood and for loading of internal liner,
- analysis of differential pressures on the containment internal structures during LOCA and SLB as a basis for estimating loads on containment internals,
- long-term containment pressure and temperature behaviour during a design basis accident (DBA) as a basis for calculation of radioactivity releases,
- containment pressure and temperature behaviour during severe accidents (containment overpressure, long-term cooling, hydrogen distribution, hydrogen burns, aerosol behaviour, i.e. ex-vessel analysis).

Thus the application of the containment codes has been extended to the severe accident domain, where they account for the phenomenology related to the temperature and pressure, hydrogen and fission product (FP) behaviour. In those cases the boundary conditions are needed e.g. from the integrated code calculations (see Section 2.4.1) or the mechanistic severe accident analysis code calculations (see Section 2.4.2) i.e. in-vessel analysis.

Examples of the containment thermal-hydraulic codes are shown in Table II.

2.4. Severe accident analysis codes

In some countries, the severe accident analyses are also included in the SAR. Analysis of the severe accidents is normally done by starting with analysing the sequences, that are selected based on the PSA results. In order to get a basic overview of the various expected phenomena and timing during the accident sequences, integrated severe accident analysis codes can be used. For detailed modelling of physical phenomena, mechanistic severe accident analysis codes are applicable. The strategy of the severe accident code development has been already from the early phases to strive for as realistic modelling as possible utilizing the existing knowledge of physical phenomenology.

TABLE II. EXAMPLES OF MOST COMMONLY USED CONTAINMENT ANALYSIS CODES

Code	*Type*	*Model*	*Country*
CONTAIN	lumped parameter	thermal-hydraulics, hydrogen burning, aerosol models	USA
COCOSYS	lumped parameter	thermal-hydraulics, hydrogen burning, aerosol models	Germany Replacing currently used RALOC and FIPLOC codes
GOTHIC	lumped parameter and 3D CFD versions	thermal-hydraulics, hydrogen distribution and reduction	USA/Germany
WAVCO	lumped parameter	thermal-hydraulics, pressure differences	Germany Used for NPPs in Central Europe and in South America

Development of the severe accident management strategies is not typically considered as a part of SAR, but a description of the severe accident behaviour and the selected approach to develop the severe accident management strategies can be added to the SAR for information at a general level.

2.4.1. Integrated severe accident analysis codes

The idea behind the integrated severe accident codes is to have a tool that can be used to model the whole sequence of the accident. These integrated codes are useful in the initial severe accident analyses, since they give an overall picture of the sequence of phenomena and the timing of events as a result. Since they often include many parametric models, their application range is limited and interpretations have to be made with special care, where profound understanding and modelling of the phenomena is required.

The initial development for an integrated code was STCP [30], a source term code package, where many separate codes were included. The widely applied and up-to-date integrated codes for severe accident analyses are MAAP4 [31] and MELCOR [32], which have been developed and maintained under contracts with EPRI and NRC in the USA, respectively. These codes have been developed as an integral code system from the beginning. There are also European and Japanese integrated codes such as ESCADRE [33], ASTEC [34] and THALES [35].

2.4.2. Mechanistic severe accident analysis codes

The integrated codes apply parametric models to many physical phenomena, but they are not necessarily suitable for cases, where more profound modelling is required. Therefore, a number of more mechanistic codes and separate effect methods have been developed. They are mechanistic in the sense that they aim at physically based modelling of each phenomenon that is in question. In many cases, the degree of physical accuracy or knowledge is not yet fully sufficient to permit that the models to be called mechanistic (i.e. to be considered as realistic).

For the initial phases of the accident as long as the reactor pressure vessel (RPV) has maintained its integrity, the development of the SCDAP/RELAP5 [36] package, that started already over 15 years ago, aims at full modelling of the in-vessel phenomena. However, some special cases, such as the in-vessel steam explosions, have not been included.

There are other corresponding codes that have been developed based on the thermal-hydraulic system codes such as TRAC/MELPROG [37], ICARE [38] based on CATHARE [10] and ATHLET-CD [39]. The common feature of these codes is that they have more accurate models and do not differ from the respective thermal-hydraulic system code as long as there is no significant core degradation. In the core degradation phase and later, when the corium gathers in the core area and relocates to the lower plenum, the accuracy of the predictions are bound with the complexity of the problem. Thus, to a large extent, the results cannot be considered strictly mechanistic, but they are rather bound by the inaccuracy of the initial and boundary conditions.

Severe accident research also involves extensive experimental work. Based on this experimental work, separate methods and models are often developed that can be used for specific, separate calculations. In many applications the CFD calculations are being applied.

2.4.3. Separate computational tools for severe accident phenomena

In many cases, the studies of specific phenomena issues necessitate separate models. The most efficient way of using such methods have been in the separate issue resolution context for in-vessel steam explosion, Mark-I liner failure and direct containment heating (DCH) for existing US plants and for in-vessel retention of corium by external cooling as demonstrated for the Loviisa WWER-440 and for AP-600 like plant [40].

2.5. Other computer codes

2.5.1. Fuel behaviour codes

The transient fuel behaviour codes include models, correlations and properties for various phenomena of interest during the transients and accidents such as cladding plastic behaviour, phase changes, and large cladding deformations (ballooning) as well as fission gas release. Modern transient codes for fuel behaviour are such as FALCON/FREY [41], FRAPTRAN [42] and SCANAIR [43] and they are used for analyzing fuel response during transients, reactivity induced accidents (RIAs) and LOCAs .

Special attention should be paid to the advanced fuel code development and application for the higher burnups and for using the mixed oxides (MOX) fuel.

All the issues related to application of existing fuel behaviour codes and including modelling of new reactor fuel for WWER reactors have been described in detail in Ref. [44].

2.5.2. Fluid dynamic and structural analysis codes

There are many typically commercial structural analysis codes that are capable of advanced structural computations. They are usually based on the finite element method (FEM) and in many cases very accurate analyses, due to increased computer speed and capacity, are possible.

The safety analysis application range of the structural analysis codes is wide. The codes are applied for analysis of the strength of the pipes, components and pressure vessels as well as of the containment structures. The fuel integrity calculations also belong to an application range for these codes. The state of development has made meaningful calculations possible in the fracture mechanics with application to the RPV PTS analyses. These advanced methods are very useful in determining the real safety margins for the structural integrity.

For purpose of the structural analysis, thermal-hydraulic loadings are needed as boundary conditions for specific parts of the RPV and the steam generator (SG) and its internals. Breaks or leaks in a pipe result in initial pressure waves, which leads to forces on specific parts of RPV and SG. The loads are dependent on the range of the pressure wave generated by the break and entering the vessel after propagation. With the initial thermodynamic conditions (mainly pressure and temperature) the incident wave range can change a lot and lead to a modification of the hydrodynamic forces on the vessel internals. A three dimensional fluid structure interaction analysis of a loss-of-coolant accident is also performed to ensure the integrity of the core support columns and the operability of a sufficient number of control rod guide structures in the upper plenum.

When a break or leak occurs the sudden discharge of water can generate a strong, local depressurization of the pipe fluid, which initiates rarefaction waves that propagate through the pipe fluid in both upstream and downstream directions. As the rarefaction wave enters the RPV upper plenum, the internal structural components such as control rod guide tubes, support columns and plenum cylinder are subjected to large pressure differential loading. This horizontal loading is time dependent and vertically non-uniform.

Relief and safety valves are employed as part of overpressure protection systems. Essential features of the dynamic behaviour are mainly observed with fast operating valves, with operating times in the order of magnitude of the period of the propagating pressure pulses in the piping system. Here the geometric design of connecting piping and vessels have an essential impact on the valve stem motion as it influences the fluid forces. The blow down is initiated by opening at first the relief valve and afterwards the safety valve under steam, water and steam-water two-phase mixture conditions which cause load transients on the relief, safety and pilot valve lines. When the safety valve opens, steam is first discharged causing a pressure drop that indicates the acceleration of the steam water interface to steam velocity in the safety line towards the safety valve orifice. If the mixture front hits the valve orifice, a high pressure peak (water hammer) is caused by the mixture front .

For design verification of piping systems to so-called pressure surge load cases such as pump failure or pipe breaks a realistic determination of the fluid loads is particularly important. As the fluid forces depend strongly on the pressure gradients and time history, the need for a tool, which determines the realistic physical effects, is very high.

To analyse the containment integrity, the steam line break accidents and LB-LOCA are investigated for a spectrum of breaks. The study is performed to confirm the containment integrity against the internal pressure resulting from the mass and energy release. The time history of the forces on the various components is calculated and will be handed over to the structure dynamics to perform the stress analysis.

2.5.3. Computational fluid dynamics (CFD) codes

In recent years, the rapid development of computer capacity and speed has made it possible to apply such detailed spatial grids for the CFD calculations and provide results, which are physically accurate enough for many safety and design related applications.

Henceforth, advancements have been made in developing new code systems that are versatile and flexible, thus reducing the manpower efforts needed for the calculations.

Unfortunately, the state-of-the-art of the CFD codes is only sufficient for single-phase calculations. However, many application areas have been referenced in recent years for design and operation safety support calculations:

- Mixing and transport of the diluted boron slugs. The CFD codes have been widely applied to mixing calculations to study the most important mitigating mechanism during inhomogeneous boron dilution transients. The main issues from the code application viewpoint are to find sufficient solutions to the applied turbulence models and to avoid influence of numerical diffusion.
- PTS mixing calculations. The CFD methods have been validated to some extent based on the extensive experiments in the 1980s. The application has many similarities to the inhomogeneous boron dilution.
- Hydrogen distribution and mixing in the containment during severe accident sequences. Many CFD codes have been developed and applied by various institutions.
- Flow mixing in the fuel assemblies, in particular the subchannel flow distribution. The CFD has been successfully applied in combination with the experiments to show that there is a larger margin to bulk boiling than expected based on the previous assumptions. This has led to improved fuel economy without any compromise with the real fuel safety margins [45].
- Molten corium pool convection. There have been many efforts to apply CFD to the thermal analysis of convective corium pool that is of interest for the in-vessel retention by external cooling.
- There is a large number of various advanced calculation efforts for the premixing and wave propagation phases of the steam explosions.
- A number of the CFD applications to the containment and turbine hall fire analysis have been reported. In the fire analysis CFD methods are referred to as the 'field models'.

The use and validation of the CFD codes differs from that of the thermal-hydraulic system codes as they can rather be seen as mathematical solvers of the applied Navier-Stokes equations. The physical models to be used are defined by the user and the spatial mesh can be generated in two or three dimensions with the help of the included mesh generation programme. Examples of the widely used commercial codes are PHOENICS [46], CFX [47] and FLUENT [48]. There is a number of in-house CFD tools that are developed and applied on a national scale in connection with the reactor safety or design studies such as TRIO [49] by the Commissariat à l'énergie atomique in France, N3S [50] by Electricité de France and FINFLO [51], developed in Finland. All these codes have also been used for computations in the area of reactor safety.

2.5.4. Fire analysis codes

The practical fire risk assessment employs three different types of numerical simulation: (1) application of experimental correlations for rough estimates of the fire development, (2) compartment model (zonal and system models) for calculation of spreading of fire and gases, and (3) CFD codes (field models) are needed for complicated compartments.

General purpose CFD codes have been applied successfully, and there is also development work going on for CFD codes tailored specifically to fire application. A CEC standard problem exercise for a cable fire experiment has been applied in the German Heissdampfreaktor (HDR) facility [52].

2.5.5. Radiological analysis codes

The radiological analyses include calculations performed for various issues. The occupational doses and radiation protection during normal operation have to be analysed to optimize the efforts according to the as low as reasonably achievable (ALARA) principle.

The FP behaviour and releases during the accident conditions to the containment (the so-called containment source term), to the environment and the radiological consequences in the environment require various methods. For example, the radiation levels in the containment should be known in order to estimate the releases through the containment leakage, to evaluate radiation conditions on-site (direct radiation, scattered radiation) and to calculate the radiation loading for the instrumentation & control (I&C) qualification to the accident conditions.

Initially, the dose calculations during DBA sequences were based on a very conservative set of assumptions. These assumptions concerned the number of fuel rods damaged (typically, all of them[1]), the releases of the FPs to the containment atmosphere, the leakage to the environment and transport to the affected individual person. Consequently, the requirements for the calculation methods were not very demanding. More realistic calculations become necessary, if the dose criteria are set so low that the conservative assumptions will automatically result in higher doses above the criteria. Reference [53] proposes a realistic method for calculating the radiological consequences of a successfully terminated LB-LOCA, and it includes all the issues from the core inventory to the summation of doses. Similarly, Ref. [54] proposes a realistic method for calculating the release following SG tube rupture faults. The recommendations for carrying out the realistic calculations have been based on the best supportable methods and data.

In case of a severe accident analyses, all phenomena starting from the release from the fuel to the releases to the environment are included in the integrated severe accident analysis codes. Since the most important part of the FPs would be released in the form of aerosols, a number of detailed codes to predict the aerosol behaviour in the primary circuit and containment have been developed.

2.5.6. Separate effect methods

In many cases, it may turn out to be more effective or reliable to apply separate effect calculation methods instead of more integrated methods. As mentioned above in Section 2.4.3, it is often done in severe accident assessments. In particular, in the issue resolution context such applications have turned out to be effective and often are the only recommended way.

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[1] In Germany less than 10% of all fuel rods.

3. ACCIDENT ANALYSES

3.1. Application of advanced methodology

The overview in the previous section demonstrates that the progress in understanding and physical modelling of various thermal-hydraulic, neutronics and other phenomena have led to the development of many advanced codes. These codes are often called BE codes, since they are capable of much more detailed modelling of the studied physical situation than the originally applied EM codes. The EM codes included pessimistic and simplified assumptions to replace or to complement the insufficient physical knowledge.

The concept of conservatism was introduced to the analyses, when the capabilities (physical knowledge and modelling, experimental database and the computer capacity) were insufficient for the realistic calculations. The other reasons were to account for the statistical character of the plant data and to account for equipment failures. By using pessimistic assumptions and simplifications concerning the initial conditions, boundary conditions and the physical models, it was believed that the limiting results could be obtained for the chosen bounding (enveloping) cases. One of the basic difficulties of that approach is that too many and too coarse assumptions may result in very unrealistic sequences, which in the worst cases may turn out not to be conservative, since the real progression might lead to more limiting results.

The key question when applying BE methods is, how to assess the associated uncertainties. The uncertainties will be discussed later in Section 5 in more detail.

The accident analysis chapter of the SAR should reflect all the pertinent transients and accidents for the given plant. The aspects of carrying out the analyses with applying the advanced methods are discussed in the following section.

3.2. Transients and accidents to be analysed and acceptance criteria

3.2.1. Transients and accidents

The preparation of the accident analysis for the SAR should start with the definition of the postulated initiating events (PIE) to be included. All the events within the design basis are subdivided into anticipated operational occurrences (often called transients for simplicity) and postulated accidents. The subdivision is done based mainly on the frequency: the transients are more frequent (e.g. frequency of the PIE occurrence higher than 10^{-2} per reactor year) and the accidents are much less frequent. In many countries the beyond design basis accidents (BDBA), such as ATWS, and in some countries even the severe accident analyses are included in the SAR.

The list of initiating events that should be considered in the SARs has been given in the USNRC Standard Format [3], and in the IAEA Guidelines for the WWER accident analysis [55]. The classes of transients and accidents are, according to Ref. [3]:

- Increase in heat removal by the secondary system;
- Decrease in heat removal by the secondary system;
- Decrease in reactor coolant flow rate;
- Reactivity and power distribution anomalies;
- Increase in reactor coolant inventory;

- Decrease in reactor coolant inventory;
- Radioactive release from a subsystem or component;
- ATWS sequences.

The IAEA guideline for the WWER accident analysis [55], recommends a corresponding list to be used especially for the WWER-440 plants. The fuel handling events are mentioned in addition to these. A similar list is also provided for the PWRs in a special volume of the Safety Report on accident analysis [2].

The FSAR for the Loviisa NPP in Finland, as updated in line with the plant modernization and power uprating project, lists the following particular accidents that additionally have to be included in the chapter on accident analyses [56]:

- Inhomogeneous boron dilution case is added to the reactivity transients;
- Overcooling transient and accidents;
- Shutdown states: hot zero power, cold shutdown and refueling outage conditions;
- Severe accidents;
- Failure of fuel pool cooling.

3.2.2. Acceptance criteria and conservatism in analyses

The safety requirements regarding conservatism in analyses are usually given in the national legislative documents. The fulfillment of the safety requirements is to be demonstrated by the proper analytical and experimental tools. For practical reasons, when appropriate, the safety requirements are quantified using acceptance criteria established or approved by the national regulatory body. More stringent acceptance criteria are applied for events with a higher probability of occurrence. Meeting of the acceptance criteria shows that the safety requirements have been fulfilled.

Furthermore, the methodology applied for the event analysis depends also on the relevant acceptance criteria. For example, let us consider the case that the most conservative assumptions concerning the system availability in the analysis of a transient lead to a difficulty in meeting criteria prescribed for transients. Then additional analyses with more realistic assumptions may be applied. This more realistic approach can assume operation of non-safety systems. In this case, the analysis of events can be further extended assuming additional failures in non-safety systems. If the probability of such combined events is substantially less than 10^{-2} per year (i.e. below the frequency level for an event to be classified as an accident), the relaxed criteria for accident analysis can be applied (see Appendix IV).

According to the IAEA safety report [2], the conservative approach should lead to pessimistic results in relation to specified acceptance criteria. The level of conservatism may be different depending on the analysed event type (transient, DBA, BDBA). It is reasonable to accept the level of conservatism adequate to the severity of the event. For example, a relaxed level of conservatism can be acceptable for the analysis of transients concerning the acceptance criteria that aim at prevention of the transient from escalating to an accident. This is in conformance with level 2 of the defence in depth concept stating that the plant control systems (not typically considered as safety grade systems) should prevent disturbances from escalating to a more severe event. Should the plant control systems fail then there is level 3 of the defence in depth present that provides the safety systems to cope with the situation.

3.3. Application of codes to various accident classes

The list of transients and accidents to be analysed is very extensive. In this section, a selected set of transient and accident classes will be discussed related to the purposes of this report; i.e. to consider application of advanced codes to different accident classes.

3.3.1. Anticipated operational occurrences

Analysis of anticipated operational occurrences requires in many cases only a single phase modelling capability, or in case of a two-phase flow, often even the equilibrium codes would be sufficient. The real challenge for the best estimate transient analysis is that the level of the plant system modelling, in particular I&C and protection systems, should be quite complete. It is therefore recommendable to strive for an as realistic analysis as possible. The current level of computer capability including the simulator development makes it possible to use advanced methods with the full-scope modelling of the plant. Thus, the plant analyzers equipped with advanced modelling features can represent a good choice for the methodology.

3.3.2. The spectrum of the LOCA sequences

The main driving force and objective of development of the advanced thermal-hydraulic system codes discussed in Section 2.1 has been their application to LOCA accidents. Therefore, these codes are well suited for analysing the full spectrum of LOCAs.

The initial approach in most countries was to apply the EM approach to show compliance with the acceptance criteria by applying the fully conservative approach according to so-called ‘Appendix K’ criteria of the US regulation [57]. The current rules in many countries allow application of more realistic methods. In that case, the quantification of the uncertainties is usually required. The relevant methods will be briefly discussed in Section 5. In case that the full uncertainty analysis cannot be performed due to limited resources, it is advisable to apply conservative assumptions to the items that involve a statistical character or a pronounced uncertainty.

The clear benefit of the advanced models is that it becomes possible to quantify the margins for the acceptance criteria. It has proven to be valuable in case of power uprating and the emergency core cooling system (ECCS) parameter optimization studies. For instance, in case of the Loviisa WWER-440 power uprating project, the margins were found to be sufficient to ensure the high safety level and possibility for improved fuel economy at uprated power [58].

3.3.3. Steam line breaks

The steam line breaks result in a rapid blowdown of the affected SG. The main emphasis is typically on studying recriticality due to low temperature in the core or in a sector of the core. It depends on the national requirements, whether recriticality is accepted during such an accident or not. The steam line breaks are also overcooling initiators for the PTS studies. They are also often enveloping cases for the containment temperature analysis. The integrity of the SG tubing should also be studied, since the accident should not proceed to a significant primary-to-secondary leakage. All these aspects should be investigated separately, since selection of the conservative initial and boundary conditions depends on the case.

The issues during steam line break accidents requiring advanced methodology are:

- mixing of the colder loop flow in the downcomer,
- entrainment of the droplets from the SG to the break,
- potential water hammers in the steam lines,
- reactor power generation in case of recriticality.

Mixing in the downcomer is crucial for defining the coolant temperature in the core inlet for criticality calculations and for the vessel wall temperature in the PTS analyses. There has been limited experimental work on defining the primary side flow distribution and mixing for such cases. The thermal-hydraulic system codes are not efficient to analyze the issue and separate engineering methods or CFD application should be considered.

In cases for which the maximum heat transfer is conservative, the droplet entrainment should be minimized (recriticality analysis, PTS analysis and degree of superheat of the steam for the containment analysis). On the other hand when the steam line integrity or loss of secondary coolant is studied, the entrainment should be estimated to be maximum.

Because of the assumption of the stuck control rod, and in particular when the inlet temperature distribution is non-uniform, it may become necessary to apply the 3D reactor dynamics model to study the recriticality and power transient caused by the lower temperature coolant entering the core. If the thermal-hydraulic feedback mechanisms of the circuit are significant, it may be necessary to apply coupled codes.

3.3.4. Primary to secondary leakage accidents (PRISE)

The advanced thermal-hydraulic system codes are readily suitable for analysing the PRISE accidents. The selection of the initial and boundary conditions depends on which aspect of PRISE accidents is studied. The different aspects to be studied are:

- effectiveness of core cooling during the accident,
- limitation of the environmental releases and doses in the DBA domain,
- prevention of the PRISE accident from developing into a severe accident due to releasing the available ECCS water out from the containment,
- prevention of the PTS to the reactor vessel,
- prevention of inherent boron dilution due to backflow of pure water from the SGs.

The conservative assumptions for various cases may differ significantly. It is also worthwhile to note that the progression of the PRISE accidents is very dependent on the operator actions, which should be clearly described in the procedures. There is a specific IAEA guide for the PRISE analysis of the WWERs [59].

3.3.5. Reactivity initiated accidents (RIA)

The RIAs can be studied in a detailed way by applying modern reactor dynamics methods with 1D and 3D kinetics models. In case that the thermal-hydraulic feedback from the reactor circuit behaviour becomes essential, the coupled thermal-hydraulic and reactor dynamics codes should be used. The most demanding cases are normally the control rod ejection accidents.

ATWS accidents and inhomogeneous boron dilution accidents, which represent important classes of reactivity accidents, are discussed in separate items below.

3.3.6. Anticipated transients without scram (ATWS)

In most countries the ATWS accidents have not been considered in the DBA framework, since the frequency is estimated to be very low. Therefore from the early phases, the realistic assumptions and methods have been applied to analysis of ATWS. Due to the complex nature of these accidents, the realistic analysis methods should have preference even when including these accidents to the DBA range.

The ATWS cases are very demanding for the applied analysis methodology as long as the reactor remains critical or near to critical state. In such a situation there is a strong coupling between the reactor power, the loop thermal-hydraulics and possible different boron concentration in the loops. When the coupling is not properly modelled, it is extremely complicated to define the relevant conservative assumptions for the analysis. Therefore the methods used should be as realistic as possible including the need for use of the coupled codes. If the results are sensitive to small differences in assumptions, a significant number of runs may turn out to be necessary.

The ATWS with the inhomogeneous boron dilution represent a special case, since even small differences in concentration have a pronounced influence on the critical system. In case that the calculations show high sensitivity of the plant behaviour to the assumptions or even non-predictable trends in behaviour in the predicted range, a possible solution is to consider more robust design measures to ensure subcriticality e.g. by reliable boron injection.

There is a specific IAEA guide for the ATWS analysis of WWER plants [60].

3.3.7. Inhomogeneous boron dilution

Inhomogeneous boron dilution events may occur under different operational conditions in the PWRs. Partially diluted or completely unborated slugs can be created in the primary system under stagnated loop flow conditions. The slugs could be transported into the core after restart of a RCP or re-establishment of natural circulation in the loop. This might lead to severe core degradation without any additional failure. Coolant mixing provides an inherent safety mechanism, which is the most important feature against diluted slugs.

A clear distinction exists between external and inherent boron dilution events. In an external dilution, the diluted or pure water slug is created by injection from the outside. These could also be called system-related inhomogeneous boron dilution events. Steam generators, chemical and volume control systems, diluted accumulator or diluted refueling water storage tanks and diluted containment sumps are considered as potential sources of diluted water. Dilution may occur during power operation, shutdown or accident conditions.

An inherent dilution mechanism is connected to a number of accident classes, where dilution could take place as an inherent phenomenon during the accident such as boiling-condensing heat transfer mode inside the primary system, or backflow from the secondary system in case of primary-to-secondary leakage accidents. A boiling/condensing heat transfer mode may be established between the core and SGs during such accidents as SB-LOCAs, ATWS and PRISE. During this mode, pure (practically zero concentration of boric acid) condensate is collected in the loop seal and a nonborated slug is created. This slug can be transported to the core after re-establishing the natural circulation or starting a reactor coolant pump (RCP) in the primary circuit. Such an inherent dilution mode could also take place during the transients leading to severe accidents.

The advanced methodologies needed for analysis of various phases of the accident are the following:

- Advanced thermal-hydraulic system codes are an efficient tool to study the slug formation mechanisms during inherent sequences; such a study can be supported by validation based on the integral experiments being performed in PKL [61], and PACTEL [62], test facilities.
- 3-D CFD codes provide an effective tool for mixing calculations. The applied turbulent mixing models have to be assessed by relevant experiments in sufficiently large scale. Although a number of small and large scale tests have been performed in existing facilities, the current status of CFD codes assessment is not considered as complete.
- Coupled thermal-hydraulic and reactor dynamics codes can be applied to study the core response by using the information from the CFD calculations concerning the boron concentration distribution in the core inlet as a boundary condition.
- Initial estimates for the maximum tolerable slug sizes and minimum boron concentrations can also be obtained by applying 3D reactor physics methods.

There was an OECD specialists meeting in 1995 [63], and an EU research project [64], organized focusing on inhomogeneous boron dilution.

3.3.8. Pressurized thermal shock (PTS)

The phases of the PTS study of the plant include the transient selection, its thermal-hydraulic analysis, stress field and fracture mechanics analysis and integration of results. The advanced thermal-hydraulic system codes are efficient for the transient and accident analyses. For situations with flow stagnation, it is necessary to apply separate methods for thermal mixing of the injected ECCS water. Both, engineering type methods and CFD codes, are available for such analyses. An extensive database was established in the 1980s for the thermal and fluid mixing studies starting from low temperature medium scale experiments up to the full parameter HDR experiments [65] and full-scale Upper Plenum Test Facility experiments [66].

The conservative assumptions are quite different for overcooling transients as compared to the analyses of the core cooling capability.

The stress field and structural analyses can be made in detail by applying advanced structural analysis methods, and the fracture mechanics results can be obtained by applying them. There is a specific IAEA guide for the PTS analysis of WWER plants, see Ref. [67].

Appendix III discusses the specific features of the PTS analysis in more detail including the data transfer between the individual steps of analysis, the relation between deterministic and probabilistic studies and the integration of results.

3.3.9. Severe accidents

There are many applications of results of severe accident analysis, but only to a limited extent are they relevant for the development of SAR. The closest one is probably the use of analysis in development of a severe accident management programmeme, but such analysis is still not typically included in the SARs.

The computer codes for the severe accident analyses were partly discussed in Section 2.4. A specific aspect is management of the uncertainties which is a crucial and difficult task for the phenomenology involved.

There are specific IAEA guidance documents for accident management programme development including the supporting analyses, Refs [68, 69], and for computational analysis of in-vessel phenomena during severe accidents [70].

3.3.10. Containment analyses

The temperature and pressure analysis of the containment during various accidents requires time dependencies of mass and energy release (blowdown curves) from the analysis of process systems as input. Another option is to use coupled computer codes covering both process systems as well as the containment. The maximum containment pressure typically occurs after a LOCA and the maximum temperature of the containment atmosphere after steam line breaks. In the latter case, advanced methodology is needed to study the degree of superheating in the compartments, where instrumentation is located and its qualification limits might be exceeded during the accident.

The containment minimum back pressure analysis for LB-LOCA conditions, needed to specify conditions for the reflooding phase or for loading of the containment liner in case of subatmospheric pressure, requires naturally quite opposite conservative assumptions in comparison to the maximum pressure analyses. Usually, the data is transferred between the codes by the user. In Germany, a coupled system of the thermal-hydraulic RELAP5 code and the COCO containment code is available to study the impact of the containment back-pressure.

The variety of pressure suppression containment designs requires specific models to take into account the effects of pressure suppression in various phases of the accident. The designs include BWR suppression pools, ice condensers and bubble condensers. For these types of containments, timing of various physical phenomena is quite important to predict containment response and it is difficult to establish a set of conservative assumptions for the analysis.

Severe accident containment phenomena are also very complex regarding the methodology to be applied. These phenomena include hydrogen distribution and combustion, aerosol behaviour (FPs), containment overpressurization and cooling as well as corium interactions with the coolant and structures. However, severe accident analysis usually does not form an integral part of a SAR.

The requirements for the containment analyses and related experiments were the subject of the future experimental containment facilities in the European Union (EUCOFA) research action [71].

3.3.11. Shutdown state analyses

The shutdown state analyses start with the definition of those plant operation states to be considered. The list of initiating events and the availability of systems and equipment differ from those for the power operation states. The acceptance criteria also need modification, since e.g. the ECCS and containment isolation may not be available.

The analyses may pose a problem for thermal-hydraulic methods, since the coolant properties and validity of correlations in the codes may not be extended to sufficiently low coolant parameters (pressure, temperature, mass flow rate).

Among the accidents that have received most attention in the shutdown states are for PWRs:

- inadequate conditions during the mid-loop operation,
- inhomogeneous boron dilution,
- PTS sequences.

The two latter ones were already discussed above. The mid-loop operation has been also studied by applying the advanced accident analysis methods.

There is a specific IAEA guide for the shutdown state accident analysis of WWER plants [72], where the operational states, initiating events, acceptance criteria, assumptions and available methods have been discussed.

3.3.12. Power instability of BWRs

For high power/low flow operating conditions if associated with unfavourable core power distributions, boiling water reactor (BWR) operation requires attention with respect to potential power and flow oscillations. The most likely modes of BWR instabilities are the core-wide or global mode with in-phase oscillations and the regional mode, where the power in one half of the core oscillates out-of-phase with the power in the other half. In the latter mode, local flux signal oscillations tend to cancel out in the summation, so the local amplitudes may grow before they are detected by the average power range monitors.

The large number of measurements at different plants under various conditions gave information about the physical and operational aspects of BWR instabilities. They have been the basis for reliable calculations of the locations of regional and core-wide stability boundaries based on linear and non-linear methodologies.

Up to now all known instabilities have been suppressed before thermal limits like the minimum critical power ratio (MCPR) were exceeded. Theoretical analyses have shown that in particular during regional oscillations with high neutron flux amplitudes, which could be a result of a pump trip transient, there is a certain probability of exceeding MCPR.

To maintain a high level of availability for BWRs and to minimize safety concerns related to severe transients, advanced reactor dynamics codes are extensively qualified based on stability measurements, which were carried out in cooperation with various utilities in the past. Furthermore, automatic modules for core stability monitoring were developed which support the operators to survey the plant stability behaviour.

3.4. Conservative analyses with best estimate methods

3.4.1. Best estimate methodology

The IAEA safety standards, Refs [1, 73], state that the conservative rules and criteria should be used in order to incorporate safety margins to the NPP design. INSAG-10 [74] also defines that an appropriate conservatism is one of three conditions, together with the quality assurance (QA) and safety culture, for an effective implementation of defence in depth. INSAG-10 also states that conservative assumptions are to be made in all steps of the deterministic analyses in order to show that the safety criteria are met with adequate margins. INSAG-10 also explains, consistently with the IAEA Draft Safety Guide [1], how to apply conservatism in the analyses at the individual levels of the defense-in-depth. Based on the relevant IAEA publications, the conclusions drawn in Ref. [75] are that

"Conservatism should be fully applied at the first three levels of defense-in-depth, with the most evident degree, rigour and formality at Level 3. At Level 4 and 5, best estimate considerations are increasingly important and recommended, and only when this is not possible, reasonably conservative assumptions should be made which take into account the uncertainties in the understanding of the physical processes. Conservative assumptions should be applied at the first three levels of defense-in-depth in all phases of the design, and also in the review of modifications, assessment of ageing effects, periodic safety re-assessment as well as in regulatory review and subsequent licensing decisions. Best estimate analysis should be applied in the elaboration of accident management measures and emergency plans, and also in the validation of emergency operating procedures (EOPs), plant simulators and in all probabilistic safety related (PSA) related analyses."

The discussion in this chapter aims at pointing out the difficulties and complications of a straightforward application of both conservative and best estimate approaches. Since there are plenty of advanced best estimate methods available, those best estimate analyses need to be chosen, which may lead to a more profound understanding of the transient and accident progression, related phenomena and consequences. On the other hand, strict application of the best estimate methodology to the licensing may turn out to be too laborious and thus would create a very heavy burden for the licensees.

Application of the best estimate codes with best estimate assumptions can be applied in the safety studies, if there is a possibility to carry out a complete analysis of uncertainties in order to obtain the distribution for the critical parameters. Then the distribution can be compared with the criterion to obtain the safety margin. From the physical viewpoint it would presuppose, and also reveal, the best understanding of the situation. For example, in evaluation of the departure from nuclear boiling (DNB) margins for the fuel rods, the statistical approach has been applied for a very long time, since there is a large number of individual rods and the experimental database is quite extensive for the fuel bundles. Unfortunately, in some complex cases the complete uncertainty analysis can easily become impractical.

The new acceptance criteria for ECCSs for light water reactors in the USA allow the use of the best estimate methodology [57]. In order to test applicability of the approach, the USNRC organized an extensive programme to develop and apply the code scaling, applicability and uncertainty (CSAU) methodology for investigating the maximum peak cladding temperature during the reflooding phase of a LB-LOCA in a large PWR [76]. The baseline of the work was to carry out a comprehensive analysis of the accident sequence accounting for all identified and ranked parameters in order to obtain the distribution of the critical parameter — in this case, the peak cladding temperature during re-flooding. A large number of runs were carried out using the TRAC code. The distribution includes information of the uncertainties. The application has been extended to other accident sequences, which will be discussed below in Section 5.2 covering uncertainty studies.

3.4.2. Conservative aspects

The concept of conservatism has been introduced to the safety analyses in order to ensure limiting and enveloping assumptions for the cases, where knowledge of the physical phenomena is not sufficient or it is intended to cover statistical data and behaviour. The aim has also been to restrict the number of needed calculations. However, there are some difficulties in using the conservative assumptions that may still lead to the need of many runs of the analysis.

Firstly, the transient or accident sequence is often very complicated and it becomes nearly impossible to define, what is conservative from the viewpoint of the overall sequence. An assumption that is conservative for the initial phase may lead the event progression to a path, where the safety parameters are not so critical in the later phase. Therefore a reasonable number of variations should be calculated in order to obtain understanding of the possible sequences.

Secondly, there may be various relevant safety issues involved, as discussed in Section 3.3.4 for the SLB and PRISE accidents. The conservative assumptions depend strongly on the safety aspect which is investigated. This leads to the situation that separate analyses with specific conservative assumptions are needed for studying each safety issue.

The third complication arises, when the safety issues, which are studied for the same transient or accident class, require opposite directions of conservatism. Examples of such cases are (1) the minimum and maximum pressure analysis of the containment during a LB-LOCA and (2) the increase of the ECCS temperature to limit the extent of PTS during small break (SB)-LOCA versus the need to cool the core. If the assumptions are too conservative, it may become impossible to satisfy both of the safety criteria.

Another concern is that the concept of the enveloping (bounding) case can be misunderstood. When the LB-LOCA was postulated and selected as the design basis for the ECCS, fuel and containment, it led in many cases to good safety solutions: full pressure containment and the ECCS capacity, which were really enveloping such accidents. In other cases, such as specific design of some pressure suppression containments and addition of massive pipe whip, restraints only based on the LB-LOCA loads sometimes resulted in questionable solutions.

Excessive conservatism in the assumptions may lead the analyses to such unrealistic results that the measures undertaken based on such analyses are not any more relevant for real safety concerns. In the past, a number of design alterations have been implemented based on concerns, which were the outcome of a too conservative methodology, such as introducing an upper head injection system to some PWRs.

3.4.3. Performing conservative analyses

In spite of all concerns, the conservative analysis will still be needed in the future, since it is impractical and costly to strive for a full application of the best estimate analyses. Methods to avoid the adverse effects of conservative assumptions have to be developed and applied. In the following a short survey will be given on the means available.

One has to be aware that any reduction of conservatism necessarily leads to an increase of the number of analyses. Therefore the task is to reduce the degree of conservatism to the optimum in order to obtain a balance between the quantification of the available safety margins needed and the cost and effort of the analyses. Another important issue is that it is intended to reduce only the computational but not the real safety margins. Certainly, the evaluation of this issue is not easy.

It is necessary to ensure that the investigated safety issues are really relevant. The reduction of conservatism can be done in both physical modelling and the assumptions concerning initial and boundary conditions. Since advanced accident analysis methods are available that apply the best available knowledge of the physical parameters, they should be applied in order to obtain a better understanding of the sequences analysed. For the initial and boundary conditions, overlapping and superposing of the assumptions should be avoided (such as mixing of data from the beginning and the end of fuel campaign), in particular since they are often inconsistent. Unfortunately, this often means that more variation calculations have to be performed.

The only way to study different safety aspects of the same accident class is to carry out tailored analyses with specific assumptions for investigation of each aspect separately.

3.4.4. The licensing analyses of the anticipated operational occurrences

The analysis of anticipated operational occurrences may assume the operation of non-safety systems. In case that their operation is crucial for meeting the acceptance criteria, the situation is considered in different ways in various countries. The assumptions and practices are discussed in more detail in Appendix IV by giving examples of some approaches and interpretations.

3.5. Relation to risk assessment

3.5.1. The role of risk assessment

The relationship between risk assessment and deterministic safety analysis has an extended history. From the time the decision was taken in 1954 to initiate commercial nuclear power generation, relative risk was considered. The first study was issued by the Atomic Energy Commission in 1957 as WASH 740 [77]. Work continued thereafter and includes major milestones such as WASH 1400 [78] and NUREG 1150 [79]. All these studies were conducted with considerable difficulties and were subject to extensive review.

There is a tendency to categorize the approach to safety assessment to be deterministic and probabilistic. In fact, the two views are not really divergent. Implicit in most deterministic rules is some consideration of risk. However, such consideration is generally not well documented and quantified. As more information is produced, the ability to evaluate and quantify risk is generally improved. One example is the LOCA rule, 10 CFR 50.46 and Appendix K [57]. In this case, the majority of the research was performed after the initial rule was adopted. However, this was recognized at the time and every effort was made to revise the licensing criteria when the research was performed and led to a revision to allow realistic analysis in 1988. Other important examples exist as well. Currently the risk associated with pressurized thermal shock is being revisited. There are very large margins to be gained in terms of risk evaluation based on improved knowledge of flaw size, flaw density, and flaw distribution. There have also been some improvements in thermal-hydraulic modelling, some of which can be attributed to the very large improvements in computing capability and cost reductions.

Part of the advancement in risk assessment is attributable simply to experience. From the start of the first NPP, thousands of years of operating experience have been gained. The reporting and analysis of operating experience has been codified and fed back to improve operating procedures. The ability to quantify margins is inherently linked to risk assessment. Other examples of issues that have been extensively studied and discussed are DCH and steam explosion. Typically, resolution of these issues has relied to a large extent on engineering judgment and expert elicitation. Risk assessment is gaining more widespread acceptance as it is generally recognized that zero risk is never possible and that decisions involve some consideration of relative risk and absolute risk.

3.5.2. The role of risk in deterministic analysis

As noted, relative risk must be a consideration in the design, operation, and regulation of nuclear installations. This is not different to any other engineering activity such as considering wind loads when designing a bridge or maximum probable flood in the case of a dam. The purpose of establishing a PSA approach is to try to develop a unified treatment of risk. Risk assessment methodology has to a large extent been driven by nuclear safety, but is now being extended to other endeavours.

The use of risk assessment in safety analysis has acquired the nomenclature of 'risk informed regulation'. This means that criteria should reflect an understanding of risk in order to optimize the allocation of limited resources and, consequently, optimize safety. To accomplish this goal, it must be clear what is the basis for the criteria. More effort is required in this direction as the pioneers of nuclear power technology pass from the scene. Now some 45 years have passed since the development of the initial regulations directed at nuclear power. In hindsight, there are many examples of uneven treatment of risk and some effort is required to rationalize the body of regulations.

The interplay of the deterministic and probabilistic analyses is described in Appendix III for the complex analysis of pressurized thermal shock. The comparison of both safety analysis methods provides an improved understanding of the issue in question.

3.6. Transfer of data between codes and coupling of codes

In many analysis tasks it is necessary to use separate codes for the various stages of a transient or accident. The input data for the next stage have to be specified from the result of the previous stage and transferred properly to the consecutive stage. The transfer can be done manually by the analyst or the codes can be coupled together. The first procedure is sufficient in case there is only little impact on the next stage. Coupling of the codes is often quite a laborious task, but it is necessary when the feedback between the stages is essential. An example of essential feedback is the coolant temperature and density impact on the reactor power during such cases as the boron dilution reactivity accident and ATWS. Therefore a number of coupled neutronics and thermal-hydraulic system codes have been developed as discussed in Section 2.2.

When transferring data from one stage to another, the analyst has to evaluate whether there is a need to adjust the data in order to ensure that bounding results will be obtained on the next stage. The adjustment may turn out to be necessary, e.g. in cases where there are strong feedback impacts from the later stage to the previous one. The lack of feedback modelling may be compensated by adding some conservatism. An example is the adjustment of minimum containment back pressure for the LB-LOCA analyses. Adjustment of results in a conservative way can be, however, in conflict with the need for consistency of data. In such a case the parameter with the strongest influence should be selected conservatively and other parameters adjusted accordingly. A similar procedure can be used when applying measured plant data as input to the analysis.

The data transfer between various stages of the integrated PTS analyses are discussed in more detail in Appendix III in order to demonstrate the interplay of various disciplines of the analyses and the management of their interfaces.

There is a growing tendency to couple computer codes together in order to provide for a more integrated analysis. This approach is not new. For example, it used to be common to apply separate codes for blowdown and reflood in a LB-LOCA. One of the achievements of the TRAC code development was to integrate the analysis into one code. Later, the thermal-hydraulic system codes were extended from their original intent of LB-LOCAs to SB-LOCAs and any other transients.

Recent examples for coupling of codes, that were originally separate, include integrating reactor physics codes with thermal-hydraulic codes such as PARCS/RELAP5 and PARCS/TRAC. Another recent example of coupled analysis is the linking of the containment code CONTAIN with RELAP. This was however not successful because of the separate time step control in each of the codes. In principle, there is no inherent limitation in combining codes together. In fact, individual codes then perform like subroutines of a coupled code. Coupling of codes can be used in conjunction with parallel processor computers or networked computers to perform calculations simultaneously.

3.7. Consideration of different stages of SAR

The design information and analyses provided in SARs should reflect the most advanced state of design or construction at the time of SAR submission.

The most essential results of analyses are presented in the PSAR and FSAR. The scope of analysis and format of both, the PSARs and the FSARs, are similar. The changes of evaluation of effectiveness of the plant safety features, based on actual information obtained during plant construction and commissioning, should be identified in the FSAR. The reasons for and the safety significance of each change should be explained and analysed.

For a new plant, preliminary licensing analyses for PSAR are performed prior to, and as a condition of, awarding the construction licence. Final licensing analyses for FSAR are performed prior to, and as a condition of, awarding the operating license.

In general, the PSAR should describe the preliminary design of the plant in sufficient detail to enable an independent evaluation of the safety of the design by the regulatory authority. At the construction licence stage, the plant data should be relatively well defined. Some details, e.g. of the control system design, may be missing and addressed by using ‘reasonable’ values, which must be confirmed at the operating licence stage. Computer codes should be validated in their areas of application.

Normally, the final analyses in the FSAR are submitted in advance of commissioning. Potential changes to safety analysis assumptions or data arising during commissioning must be assessed. If necessary, the safety analysis is revised using updated input data with parameters, which can be measured directly in the plant, e.g. measured characteristics of safety systems with measurement uncertainties.

There are no essential differences, apart from the knowledge of data, in the required methodology for the accident analyses performed on the level of preliminary and final SARs.

4. BASIC DESIGN ANALYSES

Besides the chapter on accident analysis (typically Chapter 15 in accordance with US NRC Regulatory Guide 1.70), there are various other chapters in the SARs, where analyses to support the basic design solutions are presented. The methods applied are mostly the same as discussed in the previous sections of this report. The way of application may vary significantly depending on the aims of the design calculations. Appendix V lists the selected parts from the USNRC Regulatory Guide 1.70, where advanced calculational and computational methods are employed to substantiate safety.

The specific disciplines, where advanced methods are applied for the design support, are:

- system design and in particular ECCS, automation, reactor protection and containment systems (i.e. evaluation of set points for I&C),
- structural analysis for various components and structures, e.g. RPV and SG internals,
- radiation protection, and
- fuel design and management.

The benefits of advanced methods are twofold. First, they can give a more realistic understanding of the real situation. This understanding may be crucial in avoiding erroneous design decisions and it can assist to take the more complex dependencies between the systems, components and structures into consideration. On the other hand, they can be used to reduce the excessive calculation margins that are not necessary for the plant safety. They may also help in reducing the requirements of limits and conditions (technical specifications) that may be beneficial to safety, operational flexibility and costs.

For existing plants, the needs for reanalysis or even redesign of the plant systems and structures may be connected to:

- plant back-fittings to improve the safety,
- periodic safety reviews,
- advanced fuel and advanced fuel management (MOX fuel, fuel cycle length),
- plant modernization and power uprating projects,
- plant modifications in support of the severe accident management approach development (in some countries), and
- plant life management and extension studies.

Naturally the same issues may make an extensive re-analysis necessary for the accident analysis chapter of the SAR.

5. MANAGEMENT OF UNCERTAINTIES

5.1. Application of best estimate analyses

The BE codes are best estimate in the sense that the physical modelling applied uses the best knowledge of the phenomena available. In such areas as severe accident analysis, the BE methods have been striven for from the beginning. The CFD methods employ BE methods by definition, since they aim at very accurate modelling per se.

The basic elements of successful application of the BE methods, as discussed by Kirmse [80], are that:

(1) the codes are carefully validated against the existing database,
(2) the code users are well educated,
(3) the associated uncertainties can be quantified or are at least qualitatively understood and managed.

The methodologies for sufficient validation are readily available. As an example, the validation matrices produced by the Principal Working Group 2 of the OECD/CSNI can be employed for the case, see Ref. [6]. The matrices have been established to identify the phenomena assumed to occur in the LWR plants during the accident conditions and the tests suitable for code validation.

The code users should be experienced in performing complex thermal-hydraulic system analysis. The validation calculations against the experimental database from various facilities and successful simulation of the plant experience from the real transients and accidents are efficient means for user training.

The preparation of nodalization and its application to different cases is essential for a good quality of the results. The code users should apply their experience and justify the nodalization by validation analyses. The aspects related to preparation of nodalization for the real plant analyses are presented in Appendix II.

The quantification of the associated uncertainties is a challenging task. Methods of uncertainty analyses have been developed extensively in recent years and a short overview is given in the following section.

5.2. Uncertainty analyses

The purpose of the uncertainty analysis used in licensing is to confirm that the combined code and application uncertainties are less than the design margin for the safety parameter of interest, as described in Ref. [81]. The sources of uncertainty are uncertainties in theoretical models, uncertainties due to scaling of the models and uncertainties due to plant nodalization and solution techniques. All these sources should be taken into consideration. The required level of the uncertainty analysis may vary depending on the case.

Extensive work has been carried out to study and develop methods for the statistical uncertainty analysis. The following methods have been developed in recent years [82]:

- CSAU developed by the USNRC;
- AEAW method developed by AEA-Winfrith;

- SUSA (Software System for Uncertainty and Sensitivity Analyses) method developed by GRS;
- UMAE (Uncertainty Methodology Based on Accuracy Extrapolation) developed by the University of Pisa;
- DRM developed by EdF and Framatome.

Even in sequences that have apparently quite straightforward progression such as the LB-LOCA, the complete uncertainty analysis for the peak cladding temperature during re-flood is very laborious. This was demonstrated by the first application of the CSAU methodology by its developers [76]. Other published applications of the CSAU methodology deal with e.g. SB-LOCA, Refs [83, 84]. The benchmark of the methods has been carried out in an exercise of calculating the uncertainties in a 5% cold leg SB-LOCA experiment in the large scale testing facility at JAERI [85].

The methods, which are statistical or are based on direct comparisons of various parameters become very difficult and laborious for complicated event sequences. For the purposes of the SAR it is not practical to include a complete uncertainty analysis for all cases. On the other hand, it is not recommended to strive for the full application of the EM type calculations, as there are advanced methods available and the concept of conservatism is not appropriate for most complicated transient and accident sequences. The above methodologies apply in the first place only to the thermal-hydraulic system codes. When expanding the methods to other situations, the differences might become very large.

The quantification of associated uncertainties often turns to be extremely difficult and in some cases uncertainties are not even quantifiable. In the latter cases, it is very important that the nature of the uncertainties is understood qualitatively and that appropriate measures are taken to manage the uncertainties. For such situations, in particular if the phenomenon is of a chaotic nature or if the progress of the phenomenon arrives at a bifurcation point, either a large number of runs by varying the parameters could be applied, or the full application of the risk oriented accident analysis methodology (ROAAM) has to be carried out [40].

Appendix I

PREPARATORY STEPS FOR PERFORMING SAFETY ANALYSES

The purpose of this Appendix is to give an explanation that reflects the experience gained on accident analysis for different PWRs. The basic steps of the procedure are basically the same for the analyses to be included in a Safety Analysis Report for a new plant as well as for safety upgrading or safety re-assessment of existing plants. Corresponding activities will be referred to as analytical projects in the following text.

Details of the approach may differ from country to country, due to different conditions in:

- legislative background (licensing procedure),
- availability of design basis information from the vendor and quality of plant specific data,
- support of the vendor or architect/engineer organization in the country,
- age of the NPP,
- communication problems between contractor and other parties involved in the project.

These aspects influence the timing and success of the analytical projects quite significantly. Poorly prepared projects, late clarification of requirements, criteria, etc. can cause the project to come to a stalemate situation. The situation is quite specific in countries operating WWER reactors, although there are differences in safety assessment and upgrading programmes in individual countries. Typical problems that have to be overcome are:

- architect/engineer role shared between domestic and foreign organizations,
- unavailability of certain design basis information,
- missing national regulations/standards and need to adopt international (IAEA) or foreign rules.

In the following, basic steps are briefly described which are required for the development of SARs:

1. Set up of the detailed database:

- develop list of necessary data/information/drawings regarding e.g. reactor physics, primary and secondary side systems, safety systems, I&C, etc.
- distribute the tasks regarding collection of data and set up the schedule
- fill in the list of data
- identify missing data
- adopt QA procedure.

If it is not possible to get plant specific data, the missing data have to be estimated by engineering judgment before the QA procedure can be adopted.

2. **Specification of the legal framework applicable to development of SAR:** regulations, standards, other rules, requirements, licensing criteria, acceptance criteria, etc.

3. **Specification of responsibilities of the partners involved, typically:**

- the utility:
  - to provide database,
- the contractor:
  - to discuss and agree on the content of the documentation of the computational results,
  - to perform analyses and develop related reports of the calculation results,
  - to support the utility in communication with the regulatory body,
  - to prepare the SAR (if applicable),
- the regulatory body:
  - to discuss/agree on the required scope of analyses, use of the codes and the initial and boundary conditions which have been proposed by the licensee or are required in the individual countries.

4. **Definition of content and scope of the SAR:** this may vary in different countries and typically depends on the stage of the SAR (PSAR, FSAR).

5. **Development/upgrading of existing methodologies for various applications,** such as for:

- reactor physics analyses,
- LOCA analyses,
- transient analyses,
- containment analyses,
- severe accident analyses,
- analyses of events under shutdown conditions.

These methodologies typically cover:

- definition of the scope of the analyses (initiating events),
- categorization of safety and operational (non-safety) systems,
- definition of the plant initial states, signals for reactor protection system, handling of the single-failure criterion and concept for considering operator actions,
- code selection, including justification of its applicability, e.g.
  - availability of the sufficient code manual
  - availability of the sufficient code validation report
  - sufficient user experience.

6. **Preparation of the Engineering Handbook for safety analysis,** for each of the codes to be used.

7. **Preparation of the code validation reports;** additional validation if necessary, for various applications, as stated in item 5 above.

8. **Performing the analyses as needed.**

9. **Development the SAR.**

10. **Presentation of the SAR to the utility for comments.**

11. **Presentation of the SAR by the utility to the regulatory body for comments and/or acceptance.**

The following example describes some of the practices adopted by the Bohunice NPP in Slovakia in the process of developing safety analyses for the WWER-440 units.

(A) Development of a plant-specific database for design basis and severe accident analyses of processes in plant systems as well as in the containment.

This database is maintained and controlled by dedicated plant personnel, has a well-defined QA programme and is handled like a living document, considered as part of the FSAR. The database is continually upgraded by including all pertinent information gained from operation, experiments or from analyses. The procedures used in database development are described in detail in Appendix I of the Safety Report on Accident Analysis for NPPs [2]. The database is provided to contractors as input information. It is maintained as living document by introducing all changes. The NPP staff has the right to audit whether the QA procedures of the contractors provide for proper handling of the Bohunice plant data and models. There were several reasons for in-house development of the database: lack of some data or absence of reliable original data, need for systematic preparation of the plant experiments or evaluation of data which can be collected only in specific operational regimes of the plant, duplication of efforts in the past with data collection for different analyses, loss of information etc.

(B) Development of methodology reports for all basic types of accident analyses.

Development of a set of in-house methodology reports covering all types of accident analyses included in Chapter 15 of FSAR and also in the areas where extension of FSAR can be expected in the near future, like severe accidents and accidents in shutdown operational states. These methodology reports are based on available guidelines issued by the Slovak regulatory body, IAEA guidelines, US NRC guidelines etc. and are being elaborated to the level of detail necessary to clarify all legal aspects of analyses (rules, requirements, success criteria). It is considered useful to start discussion with the regulatory body early enough in the preparatory phase of large projects, before the actual analytical work starts. Methodology reports will be continually upgraded and will reflect state-of-the-art requirements on safety assessment of WWER reactors.

These methodology reports generally do not contain requirements regarding the use of computer codes for analyses. Aspects related to code usage, specific requirements on modelling and nodalization, which are not covered in the methodology reports, have to be agreed upon during the preparatory phases of the individual projects. The general requirement is that the computer codes and models, used by the contractor performing the analyses, have to be validated. For the purposes of validation of BE models of Bohunice NPP, potential contractors will have access to data from real operational events and the tests (start-up tests) which will be performed. Bohunice NPP expects that the results of the analyses will be presented and can also be used as additional source of information for in-depth analyses of operational events and identification of direct and root causes.

Appendix II

CHANGES OF NODALIZATION

Selection of nodalization for modelling of a NPP by means of a computer code has a significant impact on the results of the analysis. Basic requirements on the nodalization are given in Ref. [2]. Accordingly, the development of the nodalization takes the following steps: using of the database, development of the engineering handbook and development of the master model (master input deck). The master model is then subject to an extensive verification and validation process.

Specific analytical tasks often create situations where changes to the basic nodalization scheme are necessary. Typical reasons include:

- there is a possibility for more detailed models due to advancing developments in computers and programmes;
- there are general trends towards use of BE methods to reduce unnecessary conservative assumptions in physical modelling;
- the existing master model is not sufficiently detailed in some aspects, e.g does not give adequate information on system behaviour or does not allow verification of certain criteria;
- code should provide output data to be used as input data for another code, which strictly specifies the required data and their format;
- time to be analyzed is too long or calculational time step is too short and it is necessary for practical reasons to simplify the noding without too strong impact on modelling of the physical behaviour of the system;
- nodalization and boundary conditions of the analysis must reflect layout of the plant systems, i.e. nodalization of the pipelines with respect to the containment nodalization should allow realistic modelling of the break locations.

The changes to the master model (master input deck) mentioned above require updating of the engineering handbook and documentation of the change. This requirement should be covered by QA procedure. The record of the nodalization change should include the following information:

- the purpose of the change and an explanation of the relevant physical phenomena to be modelled;
- justification of the nodalization change;
- changes in the engineering handbook;
- assessment of the adequacy of the nodalization change;
- assessment of the impact of the nodalization change on other aspects of the model (nodalization change should not deteriorate the overall quality of the model);
- assessment if the nodalization change does not cause application of the model outside the validated area.

Basic requirements that the analyst should comply with while making changes to the nodalization include:

- the calculational stability should not be influenced negatively by the nodalization change;
- minor changes to nodalization should not cause major changes in the model behaviour (model must be stable with respect to nodalization) — this should be justified by a sensivity study;
- the nodalization change should be validated to the extent necessary, e.g. a start-up test calculation may partially validate the applicability of the nodalization chosen for the relevant plant analysis.

A specific problem is the need for validation of the modified model in the area where the master model was successfully validated. This problem needs to be solved on a case by case basis.

Appendix III

DATA TRANSFER AND INTERFACE MANAGEMENT OF THE PRESSURIZED THERMAL SHOCK ANALYSIS

III.1. TRANSFER OF DATA

The pressurized thermal shock (PTS) study for the reactor pressure vessel (RPV) is a multidisciplinary process that can be done on a deterministic basis, but there are also probabilistic approaches. This Appendix discusses the structure of modelling and data transfer from one step in the analysis to the next in more detail. Various steps that represent different disciplines will be discussed with a special emphasis on managing the interfaces between them. More detailed discussion can be found in Refs [67, 86].

Figure III.1 illustrates the typical inputs for the various steps and how the data transfer occurs from one step to the next. In addition, an active interaction between the experts from the different disciplines is required to ensure correct transfer of the information.

The first step is to select the set of overcooling sequences for the whole PTS analysis. The selection is an interactive process, and feedback is most likely needed from the thermal-hydraulic analyses and even from the stress and fracture mechanics analyses. The feedback takes place through communication among the experts.

The task for the thermal-hydraulic analyses is to calculate the coolant temperature and heat transfer coefficient distribution in the downcomer as well as the primary circuit pressure as a function of time. These data are then transferred to the temperature and stress field calculations for the RPV wall. Appropriate RPV material properties of the base metal and welds are needed as input values for the temperature and stress field calculations.

The calculated stress fields serve as an input to the fracture mechanics. Additionally, the actual state of the RPV materials with regard to the degraded toughness properties is given as input. For the fracture mechanics, one has to postulate the defects, i.e. size, form, location and number (density) of cracks, in accordance with the national requirements and the selected approach. The fracture mechanics analysis results in estimation of the unstable crack initiation (deterministic analysis) and the conditional through-wall crack propagation probability (probabilistic analysis).

The results obtained from the various steps are then integrated into the final statement about the residual RPV lifetime with regard to the radiation embrittlement. The integration for the deterministic study means that the results are compared to the PTS criteria, e.g. avoiding unstable crack initiation, and that available margins are demonstrated with reasonably bounding input values. The integration of the results in the probabilistic study means that the calculated conditional probability of a through-wall crack is multiplied with the sequence frequency to yield the through-wall crack propagation frequency for each selected sequence. Sensitivity analyses and definition of the uncertainty band of the obtained point estimates are an integral and essential part of the integration.

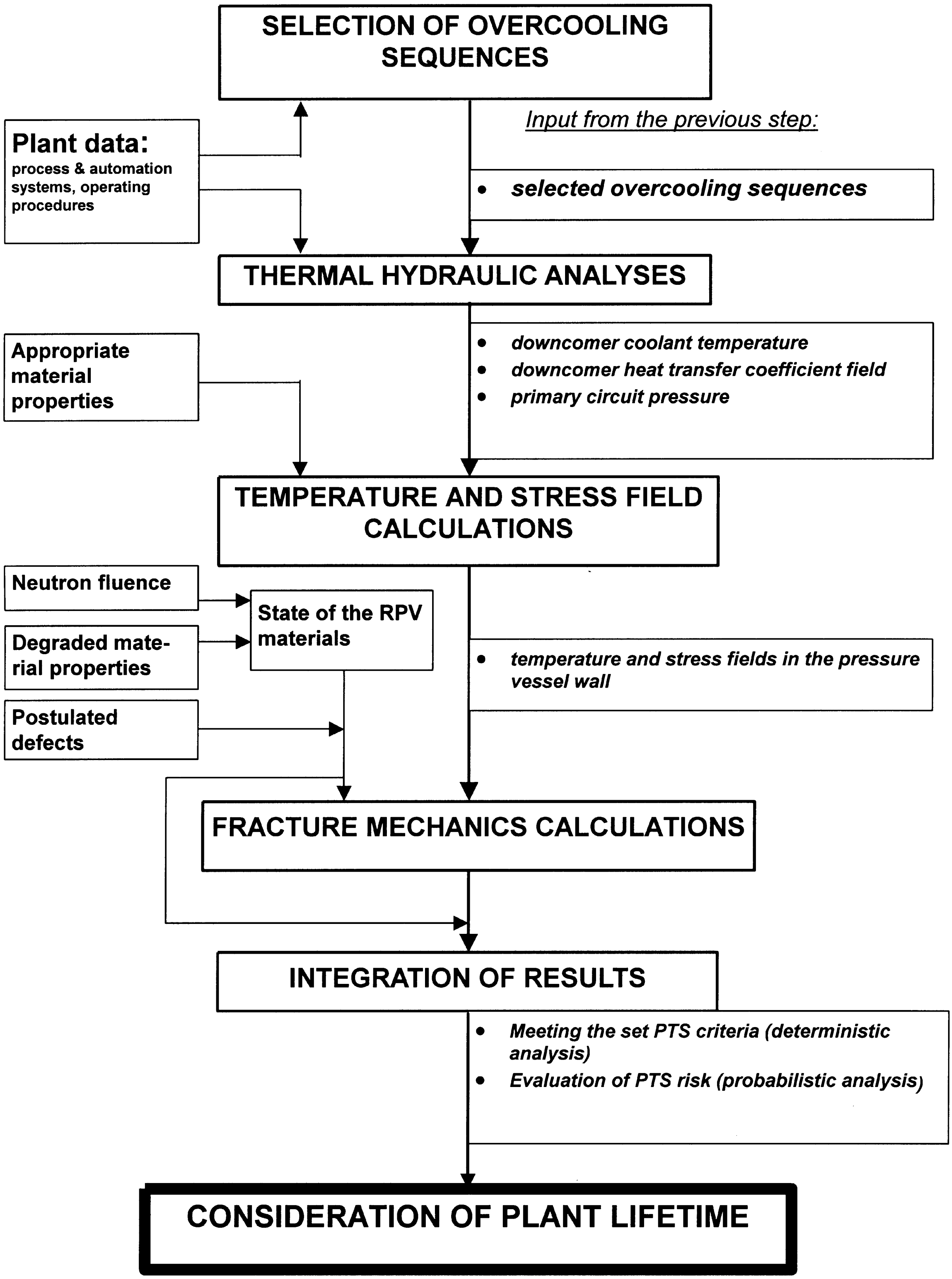

FIG. III.1. Structure of the PTS modelling and data transfer between the steps.

The deterministic selection of transients aims at finding the limiting overcooling sequences by choosing bounding conditions for the degree of overcooling. At least all the transients that are included in the FSAR of the plant should be covered. If the plant has been built to earlier standards, the scope of the anticipated operational occurrences and postulated accidents that have been taken into account during the design phase may be limited. Then, one also needs to consider sequences, which were deemed to be outside the original design basis, e.g. large leaks from the primary circuit, large primary to secondary leakage accidents for WWER-440.

The probabilistic selection of transients starts by identifying the systems and important operator actions associated with potential overcooling sequences. The selection process covers such initiating events that either directly or through consequential failures lead to downcomer temperature decrease. Usually, the relevant plant systems and operator responses are determined for each overcooling initiator by an event tree analysis. In principle, the probabilistic transient selection includes all credible PTS sequences. Most likely, their number is so large, that it has to be reduced by defining a screening limit for frequency and by extensive binning of sequences.

In the case of flow stagnation, separate methods have to be applied for thermal mixing studies. They provide realistic cooldown rates as well as temperature and heat transfer coefficient fields in the downcomer.

Since the final results of the PTS analysis have shown to be very sensitive to thermal-hydraulic parameters, particularly to the final cooldown temperature, the thermal-hydraulic analysts should be able to produce reasonably bounding values for the deterministic analysis and reliable uncertainty bounds for the probabilistic analysis.

As in the case for thermal-hydraulic calculations, the stress calculations are deterministic by nature in both approaches (except the crack size distribution in the probabilistic calculation for stress intensity in the crack tip) and the main difference results from the number of the cases studied. The temperature and stress distribution calculations should be as detailed as practical and they should account correctly for the global stresses.

Integration of the deterministic results is done to allow comparison with the PTS criteria set-up in the national requirements.

Sensitivity analyses and definition of the uncertainty band of the obtained point estimates make an essential part of the integration for the probabilistic study. In case of the deterministic study, it is considered that the uncertainties are governed by the bounding input values and the required margins.

Finally, the plant lifetime in relation to the RPV is directly obtained from the application of the results to the degraded material properties at the end of the desired lifetime. If the criteria are not met, corrective measures should be taken to obtain the desired RPV lifetime.

In case of planning corrective measures, it may turn out beneficial to carry out both, deterministic and probabilistic studies. As the above comparison shows, it is possible to obtain different valuable insights from both studies: the probabilistic study aims at defining, what are the most important overcooling sequences and what is the overall risk from the plant viewpoint. On the other hand, the deterministic study aims at defining the available safety margins under bounding conditions with as accurate methods as practical.

III.2. MANAGEMENT OF INTERFACES

Management of the interfaces between different technical disciplines is of crucial importance to ensure satisfactory treatment of the PTS issue. Obviously, it is important to ensure correct information transfer from one step of analysis to another, but in many issues there is a strong need of feedback from the other disciplines as well.

The selection of transients for the deterministic study requires first of all experience in thermal-hydraulic analysis. Knowledge about the process and automation design and about NPP operation is also required. In addition, influence of various thermal-hydraulic conditions on the RPV stress and material behaviour should be considered.

In the probabilistic study the event tree analysis carried out by reliability and process design experts aims at a more formal way to carry out selection of transients. Thermal-hydraulic knowledge plays again a key role in pruning the event tree branches and binning the sequences. Because of the special requirements of the PTS transient analysis, the thermal-hydraulic analyst needs strong support from the process design experts. All the relevant process and automation systems should be properly modelled.

The importance of the cooling rate, the temperature differences and the final cool-down temperature should be understood. The influence of non-uniformities in the downcomer fluid temperature and the wall-to-coolant heat transfer coefficient on the stress distribution is particularly complex, but they should be properly accounted for in transient selection as well as thermal-hydraulic analyses. In case of shallow cracks, very fast and deep cooling might plasticize the crack tip and thus lead to more relaxed conditions. Also, it should be noted that the change in heat transfer coefficient distribution might be even more important in some transients than the fluid temperature itself.

The neutron flux varies as function of the time and location on the RPV wall. Flux reduction measures that lead to a flux distribution should be taken into account, in line with the temperature and heat transfer distribution non-uniformities in the fracture mechanics evaluations.

Finally, the decision to choose between the deterministic and probabilistic approach (or to apply the two methods in a complementary/confirmatory sense) requires a proper understanding of the uncertainties and limitations of these methods.

As described above, there are many disciplines involved:

- plant operation,
- reliability analysis: plant data and human errors,
- process and automation design,
- reactor physics,
- thermal-hydraulics,

- structural analysis,
- material sciences,
- flaw detection through non-destructive testing (NDT).

These disciplines can only be covered by a group of experts, who are knowledgeable in their own area and capable to co-operate. In order to ensure the success, the group should have experts who are able to communicate well with the other disciplines.

III.3. PLANT LIFETIME MANAGEMENT

Basically, the plant lifetime is derived from the above described procedure, i.e. from the PTS analysis performed for RPV. There are various ways to increase the RPV lifetime. It is possible to attempt to reduce the uncertainties involved in the studies or to define and implement corrective and mitigative measures.

Uncertainties can be reduced by extensive research programmes on the following topics:

- to analyse the flux distribution more accurately,
- to improve the flow stagnation and associated final cooldown temperature predictions,
- to measure flaw sizes and distributions with improved testing techniques,
- to apply extensive surveillance programmes for the available, plant specific RPV base and weld materials,
- to take samples from the RPV base and weld metals to define the real material contents and properties.

A large variety of corrective measures has been proposed and implemented in the PWR plants to add margins to brittle behaviour and to reduce the PTS risk. As also reflected, these measures can be subdivided as follows:

- Reduction of the vessel loading:
  - reduction of overcooling transient frequencies,
  - reduction of the degree of overcooling.
- Improvement of the material properties:
  - reduction of neutron flux,
  - restoration of the toughness of the material by thermal treatment (annealing).

Various plant design and operating procedure modifications, aimed at reduction of the vessel loading, have been proposed and applied to PWR plants. Plant design modifications include heating of the ECCS water, lowering of the high pressure injection (HPI) capacity and shut-off head, qualification of the safety valves for the two-phase flow conditions, modification of the steam line and feedwater isolation criteria, implementation of the cold overpressure protection and others.

All measures taken to reduce the vessel loading have to be based on plant specific studies. Here it should be remembered that the principal aim in reactor safety is to ensure sufficient cooling of the reactor core during all transients and accidents considered. When ensuring sufficient core cooling, it is necessary to take into account that excessive cooling of the RPV wall must be avoided.

Improvement of the material properties can be made by reducing the integral high energy neutron flux on the RPV wall and by restoring the material properties by thermal treatment of the vessel. The flux reduction by low-leakage loading patterns has to be integrated to the fuel management of the plant. Thus, it cannot be made in isolation.

Thermal annealing of the WWER-440 RPV critical weld has been successfully performed for a number of plants. The final effect on the re-embrittlement rate and end-of-life material properties has to be still confirmed with extensive surveillance programmes.

Appendix IV

THE ASSUMPTIONS AND PRACTICES IN THE LICENSING ANALYSES OF THE ANTICIPATED OPERATIONAL OCCURRENCES

The deterministic analyses of the anticipated operational occurrences represent one category of events included in the SARs. The analysis should identify the limiting conditions and demonstrate the adequate margins to corresponding acceptance criteria.

The main safety goal of the transient analysis is to prove that a transient is arrested without any damage to fuel and does not evolve into an accident without additional failures (beyond the single-failure assumption in the safety systems). The criteria applied to transients are concerned with the fuel cladding integrity and, more generally, with the integrity of technological circuits.

If all plant control systems are assumed operable for accident analysis, one could strive for the situation that these systems could cope with the anticipated operational occurrence, without initiation of safety systems operation (with the exception of a reactor trip). However, for this principle to be applicable it is necessary that it has been the basis for the plant systems' design and classification as well. For example, such principle cannot be directly applied to those WWER units, where all primary and secondary pressure-limiting valves — except turbine and bypass valves — are classified as safety systems.

In many cases the transient progression is very complex. In these cases, advanced methods are important in order to obtain a realistic picture of the transient progression. Sometimes it is necessary to perform a number of variation runs. In one of these analyses it should be assumed that the non-safety systems operate in their most probable manner under given conditions.

In case that operation of a non-safety system is important for meeting the acceptance criteria for the barrier integrity, its operation should not be relied upon in the licensing analysis. Several variation runs are usually necessary to study the impact of non-safety systems. These variation runs can be presented in the SAR when deemed important and informative. It should be noted that these analyses are not used to check compliance with the safety criteria but to provide the analyst and the licensing authority with additional information on the behaviour of the system. This will enable the analyst to define the limiting case.

The limiting case to be presented in the SAR (also called the licensing analysis), needs to apply the most limiting assumptions concerning the operation of the non-safety systems and applying the single-failure criterion to the safety or safety-related systems. The national approaches differ in the conservatism applied to non-safety systems as well as in the level of detail, i.e. how the rules for analyses are defined in the national regulatory documents. The level of conservatism of a licensing analysis in individual countries can range from realistic analysis assuming operability of all systems of normal operation (e.g. Finnish approach for primary pressure control during transients) to analyses using the same approach in transient as well as in accident analysis. For example, according to the IAEA Guidelines for WWER reactors, systems of normal operation are considered only if they have a detrimental effect on the safety parameter being verified. The acceptance criteria have to be defined with proper consideration of the required level of conservatism applied to the licensing case. There is a possibility that the specific acceptance criteria for the anticipated operational occurrences cannot be met without taking operation of safety systems into account, when assuming that some or all non-safety systems are not operable. If the transient leads to the need for a safety system to operate, the safety system is assumed to work with its minimum or maximum capacity (whichever is the conservative assumption for the specific transient), taking the criteria for safety system availability into account, e.g. the single failure assumed in the most penalizing manner.

It should be noted that when speaking about anticipated operational occurrences (events of frequency higher than 10^{-2} per year) the different terminology used in individual countries could cause communication problems. In the IAEA Safety Report on Accident Analysis for NPPs several examples of terms belonging to this group of events are given: anticipated transients, transients, frequent faults, incidents of moderate frequency, upset conditions, abnormal conditions. These terms are historically established and reflect the evolution of the national standards. Therefore, definitions and requirements derived from these terms may have different meanings in different countries. For example, an abnormal event used in Russian terminology (violation of normal operating condition) is defined as a specific initiating event (of frequency higher than 10^{-2} per year), which is analysed without any further failures of non-safety systems. Anticipated operational occurrence in IAEA terminology maybe the same initiating event, but analysed with assumed failures of all non-safety systems. It is obviously not possible to consider both these events on an equal basis. Obviously, the acceptance criteria applied to both of these events must reflect the different approach in performing the analysis or level of conservatism. Consequently, if adopting requirements from another regulatory system, a check has to be performed first on applicability of definitions, national safety requirements and analytical approaches.

For the Finnish approach the following example can be given: a decrease of heat removal to the secondary side can be analysed assuming the availability of non-safety systems; the criterion of pressurizer safety valves not opening during the transient should then be fulfilled. Such analysis would not be acceptable in a regulatory system, where consideration of failures in all non-safety systems and in addition one single failure in the safety-related system are required. For example, the IAEA Guidelines for WWER reactors require for the analysis of anticipated overpressure transients to assume failure of the pressurizer relief valve, which then usually leads to opening of the pressurizer safety valves.

In advanced analyses of the transients, elements of probabilistic assessment can be applied to define scenarios (initial and boundary conditions). If the probability of combination of initiating event plus additional failures is substantially below 10^{-2} per year (e.g. by two orders of magnitude) and is therefore unequivocally in the frequency range of postulated accidents, it may be justified to exclude such low probability transients from consideration as transients. It may be then justified to apply acceptance criteria which are valid for the postulated accidents for these low probability scenarios. This is in compliance with the safety principle that for infrequent cases less stringent criteria (i.e. with lower margins to damage of the barriers) are to be applied than for more frequent occurrences. A similar approach is applied for the transient analyses that are performed for the pressure control and overpressure protection for WWER in Finland. The outlined probabilistic approach is less conservative than the line followed in the IAEA Guidelines for WWER reactors [55], which do not provide for a compensating (for super-imposing aggravating conditions) change of the category of the ‘transient’ scenarios given in the ‘List of Initiating Events to be Considered’ (Appendix I of these Guidelines).

In general, the level of conservatism from the operator performance point of view can be lower in transients than in accidents. Consideration of operator actions in transient analysis should be consistent with the approach applied to the non-safety systems availability. That means that in an analysis where the non-safety systems are not considered, the operator actions should also be treated more conservatively than in realistic analyses. The justification of this recommendation is similar to the probabilistic approach given above for non-safety system availability. The probability of operator failure can be treated similarly as non-system failure and depends on procedures, training and instrumentation available to operators. The probabilistic method used in PSA can be useful in assessing operator performance during the transient. Definition of scenarios therefore requires a good knowledge of operational procedures and EOPs.

Appendix V

EXAMPLE OF STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS BASED ON CHAPTERS OF US NRC REGULATORY GUIDE 1.70 (STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS, REVISION 3, NOVEMBER 1978) WHERE ADVANCED METHODOLOGY COULD BE APPLIED

Chapter of SAR	*Description*
Chapter 3	
3.5	Missiles
	The modern structural analysis methodology is available to fracture mechanics calculations, and to the dynamic response and integrity of the concrete structures subjected to the missile impact (such as ANACAP)
3.5.1.2	Internally generated missiles
3.6	Protection against dynamics effects associated with the postulated rupture of piping
3.7	Seismic design
	Advanced structural analysis codes are available for calculating the dynamic response of structures including the concrete structures
3.8	Design of category I structures
	Loadings are obtained from the DBA, SLB and containment code calculations. Structural analysis methods to be used.
3.9	Mechanical systems and components
	The design transients and the computer programmes used in analyses are required in 3.9.1
Chapter 4	
4.2	Fuel system design
	Fuel behaviour during transients and accidents should be analysed.
4.3	Nuclear design
4.4	Thermal and hydraulic design
Chapter 5	
5.2	Integrity of reactor coolant pressure boundary
5.2.2.2	Design evaluation of overpressurization system
5.3	Reactor vessels
5.3.3.6	Operating conditions for RPV integrity evaluation
	Provide a basis for considering that vessel integrity will be maintained during the most severe postulated transients, or refer other appropriate SAR sections.

Even the version of Reg. Guide 1.70 was written before the rector vessel PTS evaluations were completely extended to comprehensive studies of all the possible overcooling transients and accidents, the requirement has been given for SAR. The methodology for the PTS analyses includes application of the advanced thermal-hydraulic system codes, separate thermal mixing analyses, advanced structural analysis including fracture mechanics.

5.4.7 Residual heat removal system: Design basis

Chapter 6

6. Engineered Safety Features (ESF)

Design basis and design evaluation of containment system (especially mass and energy release into containment), ECCS (design basis and performance evaluation), habitability systems and FP removal and control systems (radiological calculation methods). E.g. 6.3.3 specifies the performance evaluation for the ECCS through the safety of a spectrum of postulated accidents.

Chapter 7

7. Instrumentation and control

The design of actuation criteria for reactor trip and ESF systems.

Chapter 9

9.1 Fuel storage and handling

9.5.1 Fire protection system

Chapter 13

13.2 Training

Although it has not been explicitly addressed in this Chapter, the application of advanced accident analysis methodology to the training simulators increases the accuracy of simulators. The EOP validation could also benefit of the advanced methodology.

Chapter 15

15. Accident analysis

Chapter 16

16. Technical specifications

It may be beneficial to apply advanced methods for justification of Technical Specifications in order to eliminate excessive margins.

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ABBREVIATIONS

ALARA	as low as reasonably achievable
ATWS	anticipated transient without scram
BDBA	beyond design basis accident
BE	best estimate
BWR	boiling water reactor
CANDU	Canadian Deuterium Uranium reactor
CCFL	counter current flow limitation
CEC	Commission of European Communities
CFD	computational fluid dynamics
CSAU	code scaling, applicability and uncertainty evaluation
CSNI	Committee for Safety of Nuclear Installations
DBA	design basis accident
DNB	departure from nucleate boiling
ECCS	emergency core cooling system
EM	evaluation model
EOP	emergency operating procedure
ESF	engineered safety features
EUCOFA	future experimental containment facilities in the European Union
FEM	finite element method
FSAR	final safety analysis report
HDR	Heissdampfreaktor
HPI	high pressure injection
INSAG	International Nuclear Safety Advisory Group
I&C	instrumentation and control
LB	large break

LB-LOCA	large break loss of coolant accident
LOCA	loss of coolant accident
MOX	mixed oxides
MCPR	minimum critical power ratio
NDT	non-destructive testing
PIE	postulated initiating event
PRISE	primary to secondary leakage accident
PSA	probabilistic safety analysis
PSAR	preliminary safety analysis report
PTS	pressurized thermal shock
QA	quality assurance
RCP	reactor coolant pump
RIA	reactivity initiated accident
ROAAM	risk oriented accident analysis methodology
RPV	reactor pressure vessel
SAM	severe accident management
SAR	safety analysis report
SB	small break
SB-LOCA	small break loss of coolant accident
SETS	numerical method of the TRAC code
SFAV	separation of the flow according to the velocity
SG	steam generator
SLB	steam line break
UFSAR	updated final safety analysis report

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Consultants Meeting

Vienna, Austria: 6–10 March 2000