Safety Reports Series

No.30

Accident Analysis for Nuclear Power Plants with Pressurized Water Reactors

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ACCIDENT ANALYSIS FOR NUCLEAR POWER PLANTS WITH PRESSURIZED WATER REACTORS

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FOREWORD

Deterministic safety analysis (frequently referred to as accident analysis) is an important tool for confirming the adequacy and efficiency of provisions within the defence in depth concept for the safety of nuclear power plants (NPPs). Owing to the close interrelationship between accident analysis and safety, an analysis that lacks consistency, is incomplete or of poor quality is considered to be a safety issue for a given NPP. Developing IAEA guidance publications 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 reports. Several guidance documents relevant to water moderated, water cooled power reactors (WWERs) and high power boiling reactors with pressurized channels of Russian design known as RBMKs have been developed within the IAEA's Extra-budgetary Programme on the Safety of WWER and RBMK Nuclear Power Plants. To a certain extent, accident analysis is also covered in several publications of the revised NUSS Series, for example, in the Safety Requirements on Safety of Nuclear Power Plants: Design (NS-R-1) and in the Safety Guide on Safety Assessment and Verification for Nuclear Power Plants (NS-G-1.2). For consistency with these publications, the IAEA has developed a series of Safety Reports on Accident Analysis for Nuclear Power Plants. Many experts have contributed to the development of these Safety Reports. In addition to several consultants' meetings, comments were collected from more than fifty selected organizations. These reports were also reviewed at the IAEA Technical Committee Meeting on Accident Analysis held in Vienna from 30 August to 3 September 1999.

These Safety Reports aim to provide practical guidance for performing accident analysis. The guidance is based on present good practice worldwide. The reports cover 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. The reports also discuss the various aspects that need to be considered to ensure that an accident analysis is of acceptable quality.

The first volume of the series is intended to be as generally applicable as possible to all reactor types. The specific features of individual reactor types are taken into account in the subsequent reports in the series. The reactor types to be covered include pressurized water reactors (PWRs), boiling water reactors (BWRs), pressurized heavy water reactors (PHWRs) or more specifically PHWRs of Canadian design known as CANDU, and RBMKs. The present report is devoted to specific guidance for PWRs.

The report is intended for use primarily by analysts co-ordinating, performing or reviewing accident analysis for NPPs, on both the utility and the regulatory sides. The report will also be used as a background publication for relevant IAEA activities, such as training courses and workshops.

The IAEA staff member responsible for this publication was J. Mišák of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Consistent with the revised Safety Standards Series, in particular with the Safety Requirements on Safety of Nuclear Power Plants: Design [1] and the Safety Guide on Safety Assessment and Verification [2], the IAEA has developed a Safety Report on Accident Analysis of Nuclear Power Plants containing a comprehensive description of the general methodology for accident analysis [3]. The objective of that Safety Report is to establish a set of practical suggestions based on the best practice worldwide for performing accident analysis for nuclear power plants (NPPs). The following items are covered in that report:

(1) Classification of initiating events and acceptance criteria;
(2) Methodology used for the analysis;
(3) Types of accident analysis;
(4) Computer codes;
(5) User effects on the analysis;
(6) Input data preparation;
(7) Presentation and evaluation of results;
(8) Quality assurance.

Annexes to Ref. [3] provide examples of the practical application of accident analysis; they specify and characterize the main steps in performing accident analysis, provide more discussion on and examples of uncertainty analysis, give a practical example of the preparation of input data for analysis and of the production of the corresponding documents. The annexes also contain references to the typical computer codes, provide an extensive list of codes for accident analysis and include explanations of technical terms used in the main report and its appendices. That Safety Report [3] is general in nature and does not focus exclusively on any single reactor type.

More specific guidance on performing accident analysis depends on the characteristics of the NPP in question and can only be developed for specific reactor designs or, in a more general manner, for a group of reactor designs. Such design specific guidance documents are being developed as separate Safety Reports. The reactor types to be covered include pressurized water reactors (PWRs), boiling water reactors (BWRs), pressurized heavy water reactors (PHWRs) or, more specifically, PHWRs of Canadian design known as

CANDUs, and high power boiling reactors with pressurized channels (of Russian design) known as RBMKs.

1.2. OBJECTIVE

The objective of the present report is to provide specific guidance for accident analysis for NPPs with PWRs, taking into account the specific design features of these reactors. This guidance contains a detailed list of initiating events with their direct causes and an overview of the safety aspects of the event leading to potential degradation of the barriers against release of the radioactive material. Licensing type safety analyses, aimed at demonstration of sufficient safety margins, are mainly addressed. The analysis methodology for the various events differs mainly in the selection of the acceptance criteria and in the level of conservatism required in the analysis. Examples of acceptance criteria are also provided. Conservative assumptions (i.e. those leading to more pessimistic results) for typical initial parameters and boundary conditions are indicated. Suggestions on the selection of acceptance criteria as well as initial conditions and boundary conditions are provided. Specific methodological instructions on how to perform the analysis of individual events are given. Lists of output parameters to be presented for various events are suggested.

This report is mostly based on the experience of the experts involved, with considerable use of other relevant IAEA documents [4–9] and national documents [10–16].

1.3. SCOPE

In a similar way to the first volume of this series, Ref. [3], this PWR specific volume also deals only with 'internal' events originating in the reactor or in its associated process systems. It does not cover originating events affecting broad areas of the plant (often called internal and external hazards), such as fires, floods (internal and external), earthquakes and aircraft crashes. However, analysis of the consequences of these events from a thermohydraulic point of view is partially covered by the present guidance. The emphasis in this guidance is on the transient behaviour of the reactor and its systems, including the containment and/or confinement.

The main focus of the analyses is towards the thermohydraulic aspects of the transients considered. The neutronic and radiological aspects are also covered to some extent. Limited consideration is given to structural (mechanical) aspects.

This report deals with both best estimate and conservative accident analyses. Although originally all the NPPs were licensed using a fully conservative approach (conservative code and conservative input data), such an approach can be misleading in a number of applications. Thus the use of best estimate computer codes and tools is encouraged, but always in such a way that sufficient safety margins are ensured for all applications aimed at the demonstration of plant safety.

The report covers both design basis accident (DBA) and beyond design basis accident (BDBA) situations; however, BDBA events are covered to a much lesser extent, since specific guidance documents are envisaged to cover the complex area of severe accidents (SAs) with significant core damage.

The information provided by this report is intended primarily for code users performing accident analysis. Regulatory bodies are encouraged to use this report as the basis for their national requirements. It may also be used by analysts as a reference for contacts with the national regulatory body or for the formulation of detailed company procedures for analysts.

This report in the series, along with all the other design specific reports, is intended to be, to a large extent, a self-standing publication. Nevertheless, to avoid repetition of suggestions included already in the general guidance, it is advisable for users to read the first volume in the series, Ref. [3], before using any of the design specific guidance reports.

1.4. STRUCTURE

The structure of the present report is consistent with that of the first Safety Report in the series, Ref. [3]. Naturally, the content of the sections and the level of detail take into account the design specific features of PWRs. Whenever considered to be important, reference is made to the general guidance given in Ref. [3]. In addition to this introductory section, there are 18 sections in this report.

Section 2 deals with the proper selection and categorization of initiating events. The characteristic physical phenomena are the basis for the selection and classification of the events. According to the frequency of their occurrence, initiating events are put into one of two categories, either anticipated operational occurrences (AOOs), also known as transients (Ts) or accidents (As). Accidents are further subdivided into DBAs, BDBAs and SAs. An appropriate grouping of events, for which a similar approach can be used, is provided.

In Section 3 the acceptance criteria for accident analysis applicable to PWRs are presented. Different sets of criteria for different categories of events are given. Since setting up acceptance criteria rests with national regulatory authorities, the criteria provided in this report can only be understood as examples.

Section 4 provides a short description of ways to ensure sufficient safety margins by proper selection of the initial conditions and some boundary conditions for the accident analysis. Suggestions are provided in relation to how the various conditions should be transferred to the code input decks, in particular for the conservative specification of initial and boundary conditions.

More detailed suggestions for analysis of different initiating events are described in Sections 5–18. These suggestions are formulated for groups of events with similar phenomena occurring during the course of an event. The level of detail is chosen in such a way as to avoid entering into features specific to individual plants. For each of the groups, initiating events and their safety aspects are identified and specific suggestions for analysis are formulated.

Section 19 provides a list of the calculated parameters to be reported that are specific to the different accident scenarios.

2. INITIATING EVENTS AND THEIR CATEGORIZATION

According to their frequency of occurrence [3], initiating events can either be considered as AOOs, also called transients (Ts), or accidents (As). Accidents can be further subdivided into DBAs, BDBAs and SAs. The methodology of analysis for various events differs mainly in the use of diverging acceptance criteria and also in the level of conservatism in the analysis assumptions.

A typical categorization for groups of initiating events [3] can be proposed, consistently with Ref. [4], as follows:

(a) Reactivity induced accidents (RIAs):
   — Control rod (CR) ejection (A);
   — CR withdrawal (T);
   — CR malfunction (T);
   — Incorrect connection of an isolated reactor coolant system (RCS) loop, if applicable (A);
   — Boron dilution due to a chemical and volume control system (CVCS) malfunction (T);
   — Inadvertent loading of a fuel assembly into an improper position (A).

(b) Decrease of reactor coolant flow:

— Single or multiple reactor coolant pump (RCP) trips (T);
— Inadvertent closure of a main isolation valve (MIV) in an RCS loop, if applicable (T);
— Seizure of one RCP (A);
— Shaft break for one RCP (A);
— Coolant flow blockage in the fuel assembly.

(c) Increase of reactor coolant inventory:
— Inadvertent actuation of the emergency core cooling system (ECCS) (T);
— Malfunction of CVCS leading to reactor coolant inventory increase (T).

(d) Increase of heat removal by the secondary side:
— Steam line breaks (A);
— Inadvertent opening of steam relief valves (T);
— Secondary pressure control malfunction with increase of steam flow rate (T);
— Feedwater system malfunction leading to increase of heat removal (T).

(e) Decrease of heat removal by the secondary side:
— Feedwater line break (A);
— Feedwater pump trips (T);
— Reduction of the steam flow from the steam generator (SG) (T).

(f) Decrease of reactor coolant inventory:
— Inadvertent opening of the primary system isolating valves (A);
— Spectrum of postulated pipe breaks — loss of coolant accidents (LOCAs) (A);
— Leaks from the primary to the secondary side of the SG (A);

(g) Anticipated transients without SCRAM (ATWS).

During the course of a transient or an accident, a number of different effects may challenge the integrity of the barriers against uncontrolled release of radioactivity, by affecting the three fundamental safety functions (control of reactivity, removal of heat from the fuel and confinement of radioactive materials). These effects are also called 'safety aspects' of the accident. Some potential safety aspects are listed below, following the sequence of the four successive barriers, covering the full spectrum of consequences, from transients to DBAs to SAs:

— Reactor power excursions due to reactivity insertion;
— Reactor re-criticality (local or global) after shutdown;
— Fuel enthalpy and temperature rise;
— Local fuel melting;

— Reduction of the departure from nucleate boiling ratio (DNBR) due to reduced coolant flow or due to increased temperature or decreased pressure;
— Boiling crisis due to loss of coolant inventory;
— Overheating of fuel cladding;
— Zirconium–water reaction of the cladding;
— Deformation of and/or damage to the fuel cladding;
— Hydrogen production;
— Major fuel melting and core degradation;
— Primary or secondary system pressurization;
— Pressure waves formed inside the reactor system acting on reactor internals;
— Pressurized thermal shock;
— Reactor vessel melt-through;
— Mechanical impact of the escaping coolant jet and the corresponding reaction forces on plant components and systems;
— Environmental impact of the escaping coolant on system and component qualification requirements (humidity, temperature and radiation);
— Direct coolant radioactivity releases due to containment bypass;
— Containment pressurization;
— Radioactivity releases from the containment;
— Containment basemat melt-through.

Any initiating event may be analysed with respect to these safety aspects. Acceptance criteria, as discussed in Section 3, are basically related to these safety aspects. More details on safety aspects are also provided in Sections 5–18.

As already discussed in the first Safety Report in this series, Ref. [3], different safety aspects may be related to a single internal initiating event. For example, various LOCAs lead to the degradation of core cooling and possibly to fuel overheating. At the same time, LOCAs lead to containment pressure buildup, radioactivity transport and possibly environmental releases. Under certain conditions, safety injection may cause a significant temperature reduction in the downcomer region of the reactor vessel, thus leading to a potential pressurized thermal shock (PTS). The boiling condensing regime during a LOCA may result in boron dilution by creating a slug of coolant of low boron concentration in the primary circuit. Such a case would have the potential for a reactivity accident. In some countries, complete failure of reactor scram in the case of LOCAs is required to be studied as an ATWS sequence. For primary to secondary leakage accidents, which are a special case of LOCAs, radioactive fission products can be directly transported to the environment through the secondary side steam relief paths, thus leading to a containment bypass. Each of these safety aspects requires different assumptions for the analysis, at least a partially different methodology and often even a different computer code.

For practical reasons, safety aspects and dominant phenomena are used as a basis to group the suggestions about methodologies in this report. Consequently, for each group it is possible to use the same computer code, to select similar acceptance criteria and/or similar initial conditions, applying similar methodologies with the results being presented in similar form.

The groups selected for this guidance are as follows:

— Analysis of core cooling and system pressure for various events;
— Analysis for core cooling for LOCAs;
— Analysis of containment bypass due to primary to secondary side leakage accidents;
— Analysis of ATWS;
— Analysis of boron dilution;
— Analysis of PTS;
— Analysis of residual heat removal (RHR) degradation during shutdown operational modes;
— Analysis of pressure–temperature transients in containments;
— Analysis of radioactivity transport during DBAs;
— Analysis of SAs; in the present report, only very short guidance is provided since separate guidance documents will be issued by the IAEA for SAs.

For each group of events listed above, a more detailed list of initiating events is given in Sections 5–18 with their direct causes, categorized according to their frequency of occurrence. Safety aspects of the event leading to a potential degradation of barriers are summarized, and suggestions on the selection of acceptance criteria as well as of initial conditions are provided. Finally, some specific methodological instructions on how to perform the analysis of individual events are provided.

Some suggestions, in particular the conservative selection of initial and other conditions for accident analysis, are mainly applicable to conservative analysis, which is typical for design or licensing analysis. The majority of the other suggestions is also applicable to best estimate analysis.

It is emphasized that most of the information given in this report is meant as examples of good practices, based on experience from several countries. The guidance may not automatically apply to all the individual reactor designs, computer codes and selected approaches used in accident analysis.

3. ACCEPTANCE CRITERIA

The applicable acceptance criteria used in accident analysis typically depend on the objective of the analysis (e.g. design or licensing), on the frequency of the initiating events, on the related safety aspects and on the existing high level limits for accident consequences. Thus, different criteria exist for AOOs, for DBAs and for BDBAs or SAs. Acceptance criteria for accidents are further differentiated for LOCAs, for ATWS and for other types of accident. Typical examples of acceptance criteria are given below.

3.1. ACCEPTANCE CRITERIA FOR TRANSIENTS

For transients it has to be demonstrated that the intrinsic features of the design and the systems automatically actuated by the instrumentation, particularly the reactor trip system, are sufficiently effective to ensure that:

(1) The probability of a boiling crisis anywhere in the core is low. This criterion is typically expressed by the requirement that there is a 95% probability at the 95% confidence level that the fuel rod does not experience a departure from nucleate boiling (DNB). The DNB correlation used in the analysis needs to be based on experimental data that are relevant to the particular core cooling conditions and fuel design.
(2) The pressure in the reactor coolant and main steam systems is maintained below a prescribed value (typically 110% of the design pressure).
(3) There is no fuel melting anywhere in the core.

3.2. ACCEPTANCE CRITERIA FOR DESIGN BASIS ACCIDENTS

For DBAs it has to be demonstrated that the design specific engineered safety features are sufficiently effective to ensure that:

(4) The radially averaged fuel pellet enthalpy does not exceed the prescribed values (the values differ significantly among different reactor designs and depend also on fuel burnup) at any axial location of any fuel rod. This criterion ensures that fuel integrity is maintained and energetic fuel dispersion into the coolant will not occur (specific to RIAs).
(5) The fuel rod cladding temperature does not exceed a prescribed value (typically 1480°C). This criterion ensures that melting and embrittlement of the cladding are avoided.

(6) Fuel melting at any axial location of any fuel rod is limited (typically, no fuel melt is allowed or a maximum 10% melt of the fuel volume at the hot spot is accepted). This criterion ensures that substantial volumetric changes of fuel and a release of radioactive elements will not occur.
(7) The pressure in the reactor coolant and in the main steam system is maintained below a prescribed value (typically 135% of the design value for ATWSs and 110% for other DBAs). This criterion ensures that the structural integrity of the reactor coolant boundary is maintained.
(8) Calculated doses are below the limits for DBAs, assuming an event generated iodine spike and an equilibrium iodine concentration for continued power operation, and considering actual operational limits and conditions for the primary and secondary coolant activity.

In addition to criteria (4)–(8), particularly for design basis LOCAs, short term and long term core coolability should be ensured by fulfilling the following five criteria:

(9) The fuel rod cladding temperature should not exceed a prescribed value (typically 1200°C); the value is limiting from the point of view of cladding integrity following its quenching and is also important for avoiding a strong cladding–steam reaction, thus replacing criterion (5) which is valid for other accidents.
(10) The maximum local cladding oxidation should not exceed a prescribed value (typically 17–18% of the initial cladding thickness before oxidation).
(11) The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam should not exceed a prescribed value (typically 1% of the hypothetical amount that would be generated if all the cladding in the core were to react).
(12) Calculated changes in core geometry have to be limited in such a way that the core remains amenable to long term cooling, and the CRs need to remain movable.
(13) There should be sufficient coolant inventory for long term cooling.

3.3. ACCEPTANCE CRITERIA FOR ALL ACCIDENTS LEADING TO CONTAINMENT PRESSURIZATION

In addition to the relevant criteria given above, the following criteria apply:

(14) The calculated peak containment pressure needs to be lower than the containment design pressure and the calculated minimum containment pressure needs to be higher than the corresponding acceptable value.
(15) Differential pressures, acting on containment internal structures important for containment integrity, have to be maintained at acceptable values.

3.4. ACCEPTANCE CRITERIA FOR PRESSURIZED THERMAL SHOCK ANALYSIS OF ACCIDENTS

Specific acceptance criteria for PTS analysis should apply, as follows:

(16) There will be no initiation of a brittle fracture or ductile failure from a postulated defect of the reactor pressure vessel (RPV) during the plant design life for the whole set of anticipated transients and postulated accidents.

3.5. ACCEPTANCE CRITERIA FOR ACCIDENTS OCCURRING DURING SHUTDOWN

The operational modes considered have several barriers partially degraded (reactor pressure vessel closed or open, containment closed or open). Besides generally applicable criteria, such as (8), the following specific (more stringent in the case of degraded barriers) criteria have to apply:

(17) If both the reactor and the containment are closed, the fuel cladding temperature and oxidation have to be limited to the same values as those for a LOCA.
(18) If one of the barriers (either reactor or containment) is open while the other is closed, uncovery of the fuel in the reactor needs to be avoided.
(19) If both barriers (reactor and containment) are open, both coolant boiling in the core and fuel uncovery need to be avoided.

3.6. ACCEPTANCE CRITERIA FOR SEVERE ACCIDENTS

Acceptance criteria for SAs are less prescriptive than the criteria for DBAs. Typically, the criterion is considered in relation to the very low probability associated with an SA.

Examples of more specific criteria applicable for SAs with non-negligible probability are as follows:

(20) There should be no failure of the containment because of pressure and temperature loads.
(21) There should be no immediate health effects on the population.
(22) For long term effects the ^{137}Cs release limit needs to be below the prescribed value (e.g. 100 TBq), and all the other nuclides together are not to cause a larger danger after the time period specified (e.g. three months).

4. SELECTION OF INITIAL AND BOUNDARY CONDITIONS

There are a number of input parameters for analysis, corresponding to the NPP status prior to the accident, which have a major influence on the results; they are often called key parameters. Among them, plant initial conditions and possible bounding values for some process variables play an important role. Initial and boundary conditions have to be selected, either realistically or conservatively (depending on the approach selected), from a range depending on the plant operational mode and on the parameter uncertainties. These uncertainties include technically acceptable tolerances, calculation uncertainties or measurement inaccuracies.

A typical set of initial conditions, to be specified for thermohydraulic and neutronic calculations, is shown in Table I. For each parameter, the conservative direction (maximum/minimum) is indicated for two basic types of analysis: for conservative analysis of core cooling (leading to a low value of the DNBR) and for conservative analysis of pressure in the primary system (leading to a high value of the system pressure).

The results of an analysis depend also on the neutronic characteristics of the core, which determine the reactor power behaviour during the course of accidents. The conservative approach typically aims to overestimate the reactor thermal power, which is strongly dependent on reactivity feedback coefficients and on the change of core parameters. Reactivity feedback depends on the direction of the change (increase or decrease) of the parameter under consideration. The direction may change during the course of the accident, and therefore the influence of feedback coefficients may also vary during the process.

A conservative selection of the reactivity feedback coefficients for a typical change of parameters is shown in Table II. For more complicated

TABLE I. TYPICAL INITIAL CONDITIONS FOR PLANT ACCIDENT ANALYSIS

Parameter	Conservative direction	
	Core cooling	System pressure
Reactor power	Max.	Max.
Reactor residual heat	Max.	Max.
Reactor coolant flow	Min.	Min.
Reactor core bypass	Max.	Min.
Reactor coolant temperature	Max.	Min.
Reactor coolant pressure	Min.[a]	Max.
Steam generator level	Min.	Min.
Steam pressure	Max.	Max.
Feedwater flow	N/A (consistent with power)	N/A (consistent with power)
Pressurizer level	Min.	Max.
Power peaking factor	Max.	Max.
CR worth available for reactor scram	Min.	Min.

[a] For LOCA analysis, a maximum value should be selected. For ATWS and SA analysis, best estimate plant initial conditions are typically acceptable, even for design and licensing type analyses.

processes it is suggested that parametric analyses to evaluate the influence of the coefficients be performed. In selecting the coefficients, there should be, however, no intermixing of values belonging to different operational states (beginning of fuel cycle (BOC); end of fuel cycle (EOC)).

In Table II, ‘weak’ means minimum absolute value of a feedback coefficient and ‘strong’ means maximum absolute value of a feedback coefficient. Table II is only illustrative. The selected parameters need to be checked carefully for their influence on the results of the analysis, case by case before each application.

A significant influence on reactor power comes also from the assumptions associated with scram reactivity. The negative reactivity insertion following a reactor trip is a time function of the rod position itself and of the variation in rod worth with position in the core. For accident analysis, the key parameter is typically the time of insertion up to ‘dashpot entry’ (around 85% of CR assembly travel). The selected value needs to be conservatively higher than the measured values for any of the fuel loadings. Conservative values also have to be selected for the integral value for scram reactivity.

TABLE II. CONSERVATIVE SELECTION OF NEUTRONIC PARAMETERS LEADING TO OVERESTIMATION OF REACTOR POWER

Parameter change	Reactivity feedback			Fraction of delayed neutrons	Prompt neutron lifetime
	Fuel temperature coefficient (FTC)	MTC[a] + void	Boron concentration coefficient (BCC)		
Increase of coolant temperature	Strong	Weak	Weak	Max.	Max.
Decrease of coolant temperature	Weak	Strong	Weak	Min.	Min.
Reactivity increase by CRs	Weak	Weak	Weak	Min.	Min.
Reactivity decrease by CRs	Strong	Strong	Weak	Max.	Max.
Void fraction in the core during LOCA	Strong	Weak	Strong	Max.	Max.
Boron dilution	Weak	Weak	Strong	Min.	Min.

[a] MTC: moderator temperature coefficient.

For containment pressure–temperature analysis, the basic initial conditions of the containment atmosphere are shown in Table III. The conservative selection of parameters, leading to an overestimation of the maximum containment pressure, is also shown.

Other boundary conditions that affect reactor power are the various instrumentation processing time delays associated with each trip function. These time delays typically include delays in signal actuation, in opening the trip breakers and in the release of the rods by the magnetic mechanisms. Accident analysis needs to assume conservative values for the total time delay from reaching the trip conditions up to the beginning of rod insertion.

For a conservative PTS calculation from the point of view of the RPV, conditions maximizing pressurization of the RCS and cooling down of the downcomer region have to be selected, and the selection of RPV material properties also has to be done conservatively. Conservative RPV material properties need to maximize the defect size in the RPV that is non-detectable by non-destructive testing (NDT), maximize the material brittle fracture temperature and minimize the reactor material fracture toughness.

For conservative radioactivity transport calculations, radioactivity accumulated in the primary coolant as well as radioactivity contained/released

TABLE III. INITIAL CONDITIONS FOR CONTAINMENT PRESSURIZATION ANALYSIS

Parameter	Conservative selection
Containment initial pressure	Max.
Containment initial temperature	Min.
Spray water temperature	Max.
Containment leak rate	Min.

from the fuel rods has to be maximized. For primary to secondary leakage (PRISE) accidents, the leakage from the primary to the secondary system has to be maximized.

5. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURE FOR REACTIVITY INDUCED ACCIDENTS

5.1. CONTROL ROD EJECTION

5.1.1. Initiating events and safety aspects

An RIA may be initiated by the loss of integrity of the CR drive housing with rapid expulsion of a control assembly from the core due to the differential pressure between primary coolant and the containment. This event leads to a rapid reactivity insertion, causing an RIA, and to a small break LOCA (SB-LOCA) at the same time.

Safety aspects leading to the challenging of acceptance criteria are as follows:

(a) Rapid reactor power increase resulting in a fuel temperature rise and in a reduction of DNBR; hence a reduction of heat removal and potential for consequential fuel rod damage and radioactivity release.

(b) Primary coolant pressure increase as a consequence of the power increase as well as of a turbine trip. This depends on the actual break size, on whether the pressure increase will in fact occur and also on whether there will be a need for actuation of the ECCS due to a loss of primary coolant.

(c) Containment pressure and differential pressures increase, leading to pressure loading of the containment walls; owing to the smaller break size, this aspect is usually much less important.
(d) Radiological consequences due to a loss of primary coolant, potentially also due to a loss of cladding integrity or fuel disintegration.

The suggestions provided in the following are relevant for the first safety aspect mentioned above; other aspects need to be considered separately (Sections 10, 16 and 17). Relevant acceptance criteria are Nos (4)–(8) in Section 3.

5.1.2. Specific suggestions for analysis

Several cases need to be considered in the analysis. These include BOC and EOC, hot full power (HFP) and hot zero power (HZP), as well as intermediate power levels. The BOC case is the minimum feedback case since it typically has the least negative moderator temperature coefficient (MTC). The EOC case has a slightly smaller Doppler coefficient but a much larger moderator feedback effect. This causes the EOC transient to be less severe than the BOC transient for the same reactivity excursion (measured in US dollars). However, for EOC the reactivity in US dollars may be larger due to the reduction of the fraction of delayed neutrons with increasing core lifetime. Both these cases therefore have to be analysed to sufficiently cover the range of expected conditions. Sometimes intermediate stages between BOC and EOC have also to be considered to ensure conservatism of results.

For the purpose of the analysis, the accident may be simulated by linearly introducing reactivity up to the ejected rod worth within a sufficiently short time span (e.g. 0.1 s). This linear reactivity addition is a calculation convenience rather than an attempt to simulate the actual ejection of a rod. Such an assumption is acceptable since the 0.1 s ejection time is rapid enough compared with the non-linear feedback effects, so that the calculation is not sensitive to the expected S shape of the reactivity versus distance curve.

The value of the reactivity to be inserted during the accident is to be found by means of three dimensional (3-D) steady state neutronics calculations that incorporate all possible reactor states and all possible CR positions allowed by the operational limits and conditions of the NPP.

It is also preferable that a transient 3-D neutronic analysis be performed. Optionally, a combination of a 3-D steady state power distribution analysis with a transient point kinetics calculation of the total reactor power is acceptable. For the latter case, the steady state power distribution has to be calculated at least for two operational states: for the initial reactor state before the accident and for the state after CR ejection (at the same power level). For the transient analysis with a fixed power distribution, the distribution with a higher peaking factor needs to be chosen. The neutron flux power peaking factor can optionally be linearly increased from its initial steady state value to the final value within the time for the rod to be ejected.

Core inlet coolant temperature, core flow and coolant pressure may be kept constant for the analysis of the reactor power increase due to the fact that changes of these parameters are not substantial during the short time interval analysed.

Conservative values of reactor trip reactivity (conservative time delay and reactivity versus CR position dependence) are used, typically assuming a stuck CR in addition to the ejected rod. The stuck rod selected is the highest worth rod at HZP with all rods inserted except the ejected rod, and will usually be a rod adjacent to the ejected rod. This assumption is made to account for the possibility of the ejected rod causing damage to an adjacent rod drive housing and preventing that rod from tripping.

A weighting factor is applied to the calculated Doppler feedback to account for the expected increase in feedback as a result of the skewed power distribution after the CR has been ejected. The Doppler weighting has to be conservatively calculated.

In applying the single failure criterion, considering that the first signal to actuate the reactor trip system has to come from an excessive increase rate of the neutron flux, a potential failure in a reactor power measurement channel also needs to be taken into consideration.

5.2. CONTROL ROD WITHDRAWAL

5.2.1. Initiating events and safety aspects

For analysis of CR withdrawal, the uncontrolled withdrawal of a bank of CRs either at subcritical/zero power conditions or at full power is considered. The accident is caused either by a control system failure or by an erroneous operator action.

Control rod withdrawal results in a power excursion accident with a reduction in DNBR. The core thermal limit, i.e. the DNB, can be exceeded. The power excursion is limited by temperature feedback effects and/or by reactor trip. The excursion also leads to an increase in both the primary and the secondary system pressure. Owing to the fact that CRs are withdrawn at the normal operational velocity, the reactivity insertion rate, and consequently the power excursion, is significantly slower than in the rod ejection case.

Relevant acceptance criteria are Nos (1)–(3) in Section 3.

5.2.2. Specific suggestions for analysis

Both zero power and full power conditions have to be considered to verify the effectiveness of feedback effects and the reactor trip system in limiting the power excursion. Other initial conditions will be selected similarly, as for the CR ejection accident.

The maximum reactivity insertion rate may be calculated by a 1-D or a 3-D code. Other conditions are the same as for a CR ejection accident. For a conservative calculation, conservative values of the time when reactor trip occurs (from which parameter the reactor scram is activated) and of the reactivity of withdrawn CRs have to be used.

Selection of a single failure criterion needs to be made depending on the acceptance criterion considered. For example, a failure in the pressurizer relief valve, which is conservative for the primary system pressurization, is not necessarily conservative from the point of view of minimum DNBR.

5.3. CONTROL ROD MALFUNCTION

5.3.1. Initiating events and safety aspects

Three possible events are typically considered in this group:

— Drop of one CR,
— Withdrawal of one CR,
— Misalignment of one CR.

As a consequence of any of these events, there is a distortion in the core power distribution with potential reduction of DNBR. For a CR withdrawal, there is also a global reactor power increase, which is reduced later by the reactor power control.

A potentially relevant safety aspect comes from the case when a rod drops into the core and the CR system is in automatic mode. In this case, the rods will be moved out to compensate for the sudden power decrease. Before achieving a new equilibrium power, a transient overshoot on nuclear power can be expected, coincident with a significant distortion in radial power distribution caused by the dropped rod. High local peaking factors together with an overshoot in power may violate the limits on fuel power density.

Applicable criteria for these events are Nos (1)–(3) in Section 3, since those cases are categorized as AOOs. Optionally, a withdrawal of one CR can be also categorized as an accident, with relevant acceptance criteria to be used.

5.3.2. Specific suggestions for analysis

A dropped CR or withdrawal of one CR has to be analysed as a transient problem. Misalignment of a CR can be analysed as a steady state problem. A change in core power distribution during an accident needs to be adequately dealt with (see Section 5.1.2 for the case of ejection of one CR).

For the dropped rod case, a sensitivity study has to be performed to search for different possible combinations of rods that drop and various single failures in reactor trip actuation. A 3-D neutronic calculation has to be performed to estimate the effect on the ex-core neutron detectors used to actuate the reactor trip system, or a conservative approach needs to consider the case in which no reactor trip is directly caused by the rate of change in neutron flux. These cases will maximize the combined effect of local flux distortion and total nuclear power. Additionally, for transient cases with the control system in automatic mode, adequate modelling of such a system has to be considered.

Loss of off-site power is usually not considered in combination with this initiating event.

5.4. INCORRECT CONNECTION OF AN INACTIVE REACTOR COOLANT SYSTEM LOOP

5.4.1. Initiating events and safety aspects

Such an event is typical, although not exclusively, of WWER-440 reactors with an MIV installed in both the hot and cold legs of all circulation loops. The reactor, in accordance with operational limits and conditions, may be in operation at reduced power, with one of the loops isolated by an MIV. Coolant temperature and boron concentration can be reduced in the isolated loop. The reduced values are limited by the plant operational limits and conditions.

The event may be initiated by:

— Startup of the main circulation pump (MCP) in an isolated loop, followed by opening of MIVs;
— Startup of the MCPs in an inoperable non-isolated loop, either not equipped with an MIV or with the MIV remaining in the open position;
— Inadvertent opening of an MIV.

Both causes result from an erroneous operator action. The accident is developing into an RIA due to feedback through the MTC, either due to lower core inlet temperature or due to higher core mass flow rate. Consequently, reactivity insertion leads to a power excursion and a reduction in DNBR.

The event is classified as an accident and therefore acceptance criteria Nos (4)–(7) apply, although a DNB (applicable for an AOO) can usually be avoided. Criterion No. (7) (maximum primary pressure) is not challenged in this case.

5.4.2. Specific suggestions for analysis

The course of the accident is mostly influenced by the MTC. For conservative evaluation, the highest possible negative value of the MTC has to be selected, which is typical for the EOC. At the same time, weak fuel temperature feedback needs to be considered. The high power peaking factor is another key parameter for the analysis.

As the accident may lead to an asymmetry in the core power distribution, coolant mixing in the lower reactor plenum needs to be based either on experiments or on operational data. Low coolant mixing in the downcomer and in the lower plenum is a conservative assumption. Adequate consideration of timing for the MIV opening and MCP startup has to be made in connection with mixing and the time development of the process. If mixing is not proven to be sufficient at the core inlet, the 3-D core neutron kinetics model and the 3-D thermohydraulic model are suggested for the analysis.

Selection of initial parameters, such as reactor power, coolant temperature, number of operating loops and boron concentration in the isolated loop, has to be made in accordance with plant operational limits and conditions.

It is acceptable not to consider loss of off-site power for the analysis. A single failure needs to be typically considered in the systems influencing the core power limitation.

5.5. BORON DILUTION

Boron dilution events are featured by an inadvertent decrease of the boron concentration in the primary coolant.

Homogeneous dilution may occur due to CVCS malfunction when operation of primary pumps or natural circulation of coolant is sufficient for uniform mixing in the whole primary circuit. The process is slow and the operator usually has enough time to take corrective measures. Homogeneous dilution is usually classified as an AOO.

Inhomogeneous or local boron dilution takes place if a slug of non-borated water or water with low boron concentration is formed in the primary loop. There is a potential for rapid core power increase if the slug is transported without sufficient mixing to the reactor core, due to pump startup or for example to re-establishment of natural circulation. The slug could be formed by non-borated water injection by the CVCS to stagnant loops or due to leakages into the primary system from other systems. These sequences are called external dilution cases and they are classified as accidents.

During accidents with decreased reactor coolant inventory, such as SB-LOCA, PRISE and ATWS, there is a possibility for an inherent dilution when the reactor enters into boiling–condensing mode. During this mode, the steam produced in the core condenses in the SGs and the non-borated condensed water is collected in the loop seal, forming a diluted water slug. Inherent dilution is one of the safety aspects to be analysed during the corresponding accidents. These aspects are further discussed in Section 13.

5.6. INADVERTENT LOADING OF A FUEL ASSEMBLY INTO AN IMPROPER POSITION

5.6.1. Initiating events and safety aspects

Such a situation may be caused by inadvertent loading of one or more fresh fuel assemblies into an improper position or incorrect assembly rotation in the core due to human error during the core reloading procedure. The core power distribution can be consequently distorted, leading to high power peaking factors with potential violation of the core thermal limits, which means a DNBR.

The event is classified as an accident and acceptance criteria Nos (4)–(7) should apply. Criterion No. (7) (high primary pressure) is not particularly challenged.

5.6.2. Specific suggestions for analysis

The analysis needs to be performed for BOC conditions, corresponding to the HFP state of the reactor. A combination with other failures is not relevant. Representative combinations of core loading mistakes have to be considered, including wrong positioning of fresh and partially burned fuel assemblies with different burnups.

Owing to the nature of the event, a steady state calculation is adequate for the analysis. The analysis should prove that either distortion of the core power distribution is detected by the in-core instrumentation or that the change of the reactor power distribution is acceptable for long term reactor operation (i.e. it is within acceptable values for power peaking factors).

6. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURE FOR A DECREASE OF REACTOR COOLANT FLOW

6.1. SINGLE OR MULTIPLE MCP TRIP

6.1.1. Initiating events and safety aspects

An MCP trip due to an interruption of the power supply or failure of the control system is the most typical cause for a reduction of primary coolant flow. Reduction of the primary flow leads to an imbalance between the heat produced by the fuel and the heat removed from the core, potentially exceeding core thermal limits, which means a DNBR. Thermal imbalance also leads to an overall pressure–temperature transient, typically resulting in a short term pressurization of both the primary and the secondary circuit.

MCP trips are typically categorized as anticipated operational occurrences, therefore acceptance criteria Nos (1)–(3) of Section 3 apply.

6.1.2. Specific suggestions for analysis

The flow distribution between the core flow and the bypass flow needs to be conservatively considered. Selection of an adequate DNBR correlation is essential. A statistical analysis (validation combined with a definition of uncertainty against experimental data) is required.

Three dimensional effects of coolant mixing in the reactor downcomer and the RPV lower plenum are required to be taken into account or conservative assumptions have to be adopted, except for the case when all the MCPs are tripped simultaneously. For an analysis of individual pump trips, four quadrant pump characteristic curves are needed. A combination of the event with loss of power supply is not required.

The single failure criterion typically considers a failure to open either a secondary steam bypass station or the pressurizer relief valves.

For reactors equipped with a special power control system designed to cope with MCP trips, consideration of this control system is acceptable. Optionally, a failure of this system might be analysed.

6.2. INADVERTENT CLOSURE OF A MAIN ISOLATION VALVE IN A REACTOR COOLANT SYSTEM LOOP

6.2.1. Initiating events and safety aspects

Such an event is possible for reactor designs with MIVs (gate valves) installed in circulating loops. The MIV closure time is typically several tenths of a second, resulting in a relatively slow reduction of the core flow. The event can be initiated by an erroneous manual operator action. Safety aspects and applicable criteria are the same as in the case of an MCP trip.

6.2.2. Specific suggestions for analysis

Events can be modelled by introducing an input function for the hydraulic resistance of the MIV in the corresponding primary loop. Other suggestions are the same as for MCP trips (Section 6.1.2).

6.3. SEIZURE OF ONE MAIN CIRCULATION PUMP OR SHAFT BREAK OF ONE MAIN CIRCULATION PUMP

6.3.1. Initiating events and safety aspects

The cause of such an event can be mechanical damage. The event itself leads to a very sharp reduction of the flow in the corresponding loop, typically within a time shorter than 1 s. In the case of MCP seizure, it is assumed that the support components are designed to withstand transient mechanical loads without the failure of primary piping.

The event is categorized as an accident, and therefore acceptance criteria Nos (4)–(8) apply.

6.3.2. Specific suggestions for analysis

The same additional suggestions are valid as provided for MCP trips (Section 6.1.2).

6.4. COOLANT FLOW BLOCKAGE IN A FUEL ASSEMBLY

6.4.1. Initiating events and safety aspects

Such an event is typically caused by the presence of debris in the primary circuit following a refuelling/maintenance period. They lead to total or partial coolant flow reduction and sudden disproportion between heat generation in the fuel assembly and heat removal from the fuel assembly. Owing to these facts the local coolant temperature increases, the cladding temperature also increases and, consequently, the DNBR decreases. This two phase flow typically takes place in the fuel assembly. However, the impact on the overall plant behaviour can still be insignificant.

The event is categorized as an accident, and therefore acceptance criteria Nos (4)–(8) apply.

6.4.2. Specific suggestions for analysis

A subchannel or 3-D thermohydraulic computer code is required for this analysis. Parameters in the lower plenum and the upper plenum of the reactor can be specified as boundary conditions for detailed core analysis.

7. ANALYSIS OF SYSTEM PRESSURE FOR INCREASE OF REACTOR COOLANT INVENTORY

7.1. INADVERTENT ACTUATION OF THE EMERGENCY CORE COOLING SYSTEM

7.1.1. Initiating events and safety aspects

Such an event can be initiated either by control system malfunction or by operator error, often during testing of the ECCS.

The safety aspects of the event are as follows:

(a) There is a pressurization of the primary system due to an increase of the coolant inventory.
(b) Emergency injection of cold coolant may lead to non-symmetrical reduction of the coolant temperature at the RPV inlet, potentially affecting vessel integrity.

(c) In the case of overfilling of the pressurizer, the accident could lead to the pressurizer safety valve becoming stuck in the open position and a LOCA developing.
(d) In the case of loss of integrity of the pressurizer relief tank, containment pressurization will take place.

For aspect (a), which is mainly considered in this section, the applicable acceptance criterion is No. (2) (Section 3). Aspect (b) needs to be analysed by the methodology for PTS (Section 14). Aspect (c) has to be analysed in accordance with Section 10 and aspect (d) in accordance with Section 16.

7.1.2. Specific suggestions for analysis

Application of a single failure criterion has to take a failure in the pressurizer relief (safety) valves into consideration. The integrity of the pressurizer relief tank also needs to be analysed. Such events are typically not combined with a loss of off-site power. The maximum capacity of the ECCS has to be used as a conservative assumption in the analysis.

7.2. MALFUNCTIONS IN THE CHEMICAL AND VOLUME CONTROL SYSTEMS

7.2.1. Initiating events and safety aspects

The event may be initiated by a failure in the control system of the CVCS, which leads to an increase of the reactor coolant inventory. Safety aspects are the same as for the startup of the ECCS. Moreover, there is a potential for injecting non-borated water, which would cause an RIA (Sections 5 and 13). For other aspects, the relevant acceptance criterion is No. (2) in Section 3.

7.2.2. Specific suggestions for analysis

The same suggestions as provided for the actuation of the ECCS apply (Section 7.1.2).

8. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURE FOR INCREASE OF HEAT REMOVAL BY THE SECONDARY SIDE

8.1. STEAM LINE BREAKS

8.1.1. Initiating events and safety aspects

Such accidents can be initiated by a partial or full steam line rupture, which may occur either inside or outside the containment. The limiting size for a break (typically located outside the containment) is rupture of the main steam header (if relevant) up to its full size break.

The accident leads simultaneously to depressurization (cooling down) of the secondary circuit and loss of the secondary coolant, leading also to overcooling of the RCS. Typically, the effect of cooling down dominates, leading to a non-symmetrical cooling of the RPV wall, to a positive reactivity insertion, to a potential recriticality and to a reactor power increase in some situations regardless of scram.

Safety aspects of this accident, with possible violation of acceptance criteria, are as follows:

(a) Possible reactor power increase or reactor recriticality after its shutdown due to a substantial decrease of the core inlet temperature, typically in one section of the core adjacent to the affected loop. In the case of a coincident loss of power supply (MCP coastdown), the DNBR is reduced and a boiling crisis can occur in the reactor core.
(b) Loss of secondary coolant due to steam outflow; this is usually less significant than the cooling effect of depressurization.
(c) Significant and rapid non-symmetrical reduction of coolant temperature at the RPV inlet followed by high pressure emergency coolant injection, potentially affecting the vessel integrity.
(d) Containment pressure and sub-compartment differential pressure increase, leading to pressure loading of the containment structure (in the case of a break inside the containment).
(e) Containment temperature loading (usually higher than during primary circuit LOCAs).
(f) Loss of secondary coolant can lead to SG tube bundle uncovery, primary circuit temperature increase and high pressure injection system (HPIS) activation. This situation can lead to an increase of primary pressure, filling of the pressurizer and overpressurization of the primary circuit.

Aspects (b)–(e) above have to be analysed by a special methodology (Sections 9, 14 and 16). Therefore, the rest of this section is applicable only to aspects (a) and (f). Such events belong to the category of accidents, and therefore acceptance criteria Nos (4)–(8) in Section 3 apply. For some designs it is, however, possible to comply even with the more stringent criteria, Nos (1)–(3), for this type of accident.

8.1.2. Specific suggestions for analysis

Typically, the analysis needs to be performed for both HFP and HZP operational states, with and without MCPs running. Reactivity feedback coefficients (in absolute values) are to be selected for the EOC with the aim of maximizing the possibility of returning to criticality.

The break location and flow model for steam release have to be selected in such a way that the break outflow is maximized. Consideration of the flow measuring nozzles and valves installed in the steam lines is, however, important for the calculation of the break outflow. The most severe case for a steam line break is typically a break upstream of the main steam isolation valve (MSIV). Breaks downstream of the MSIV will be stopped when the valve closes (if not damaged).

It should be considered that at the reactor trip the most effective CR was stuck in its upper position. It has to be assumed that the stuck rod is located in the cold section of the reactor core adjacent to the affected loop.

The influence of HPIS needs to be analysed parametrically with respect to the cooldown rate and the change in primary circuit boron concentration. Different numbers of ECCS pumps, pump characteristics and possible non-uniform injection into individual main coolant loops have to be considered in this analysis.

In the case of a recriticality, in particular when a point kinetics model is used, maximum power peaking factors possibly occurring during the whole process have to be considered for the DNBR calculations. Xenon decay after reactor scram may be neglected for processes with durations of several minutes. Boron addition is allowed to be considered during the accident, but its time delay needs to be considered conservatively high and its mixing has to be considered conservatively low.

DNBR calculations should be performed by a subchannel approach. For simplicity, steady state calculations are acceptable at time points chosen conservatively along the transient.

Specific requirements for the computer code to be used are as follows:

(a) A complex computer code capable of adequately simulating the behaviour of both the primary and secondary systems as well as neutron kinetics is needed.
(b) A 3-D dynamic neutronics model is preferred. If this is not available, it is necessary to combine a conservative 3-D steady state neutronics model with a 1-D thermohydraulics model for integral calculations.
(c) A model of non-ideal coolant mixing in the reactor plenum is preferred. If such a model is not available, conservative assumptions on coolant mixing need to be used.
(d) The code used has to be able to consider water carry-over by steam flowing out of the secondary circuit; otherwise, a conservative approach is to be chosen to maximize the loss of secondary coolant, also considering, however, the next requirement.
(e) The SG secondary side mixture level has either to be calculated by the best estimate approach, or a conservative approach needs to be taken to maximize the cooldown rate.

For the single failure criterion, the following failures are typically considered:

— CR stuck in the upper position;
— Failure of one of the steam leak isolating valves.

The failure of one of the steam leak isolating valves, however, may also occur as a dependent consequential failure due to forces induced by the steam line break.

8.2. INADVERTENT OPENING OF STEAM RELIEF VALVES

8.2.1. Initiating events and safety aspects

Such an event can be initiated by a control system failure, by an instrumentation failure (e.g. sensor failure) or by an operator error, leading to inadvertent opening of or failure to close the valves, and a subsequent release of steam from the secondary circuit (SG safety valves, steam dump to the atmosphere or steam bypass to the condenser).

The safety aspects of the event are similar to those for the steam line break, except for the containment pressurization. In addition, owing to considerably lower steam outflow, the change of all the parameters is slower. The event is categorized as an AOO, therefore acceptance criteria Nos (1)–(3) (Section 3) apply.

8.2.2. Specific suggestions for analysis

The suggestions made in Section 8.1.2 are generally valid for these events as well, but less stringent requirements on the computer code may apply. Typically, the 1-D neutronics model, ideal coolant mixing in the reactor plenums, no special model for water-carry over by steam outflow and a conservative approach for the SG mixture level model are acceptable.

8.3. SECONDARY PRESSURE CONTROL MALFUNCTION WITH AN INCREASE OF STEAM FLOW RATE

Such events are initiated by a secondary pressure controller malfunction. Safety aspects, acceptance criteria and specific suggestions for analysis are the same as provided for inadvertent opening of the steam relief valves (Section 8.2).

8.4. FEEDWATER SYSTEM MALFUNCTION

8.4.1. Initiating events and safety aspects

Two types of event are considered:

— Malfunctions that increase feedwater flow (a failure of a feedwater control valve);
— Malfunctions that decrease feedwater temperature (a failure of a feedwater preheater).

Both events lead to an increase of heat removal by the secondary side, thus reducing primary coolant temperature and increasing the reactor power due to reactivity feedback. Reduction of the DNBR is the main issue, and acceptance criterion No. 1 (Section 3) applies.

8.4.2. Specific suggestions for analysis

The analysis needs to be performed for both HFP and HZP. The event is typically not combined with a loss of off-site power.

9. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURE FOR DECREASE OF HEAT REMOVAL BY THE SECONDARY SIDE

9.1. FEEDWATER LINE BREAK

9.1.1. Initiating events and safety aspects

Feedwater line breaks are typically caused by a material degradation of the secondary circuit piping, leading to a partial or full rupture of a pipe in the feedwater system. Corresponding to the break location, the feedwater supply can be interrupted in one or several SGs. In the case of a main feedwater header rupture, the feedwater supply will be interrupted in all SGs.

The accident differs from steam line breaks due to the fact that the water outflow leads to a rapid decrease of the affected SG secondary side water level. Thus, the secondary side heat removal capability is reduced, while cooling, due to energy outflow, is not so high.

Safety aspects that challenge the acceptance criteria are as follows:

(a) Reduction or loss of secondary side heat removal leads to overheating of the primary coolant, with its corresponding expansion and pressurization of the primary circuit.
(b) For the longer term, heat removal from the core may also be threatened, leading to fuel rod overheating.
(c) In the case of a long term loss of secondary side heat removal, the pressurizer relief or safety valves may stay opened. The consequences would be similar to those for the corresponding (in size) primary circuit rupture, a LOCA, resulting in the release of radioactivity into the containment.
(d) Outflow of the secondary coolant, and possibly also the primary coolant, leads to containment pressurization.
(e) Injection of cold water on previously uncovered SG tubes by recovered feedwater pumps may threaten the structural integrity of the boundary between the primary and secondary circuits.

This section deals only with the first two aspects of the accident; other aspects are partially dealt with in Sections 10, 16 and 17. The event belongs to the category of DBAs, therefore acceptance criteria Nos (4)–(8) (Section 3) apply.

9.1.2. Specific suggestions for analysis

The position of the break has to be selected so as to maximize the loss of secondary coolant. The secondary coolant outflow model has to maximize the break flow. The accident is typically analysed for both cases, with and without loss of external power supply. The single failure criterion is typically applied to emergency feedwater pumps and pressurizer relief or safety valves.

9.2. FEEDWATER PUMP TRIP

9.2.1. Initiating events and safety aspects

Feedwater pump trip, typically due to an interruption in electric power supply, leads to loss of secondary coolant inventory and, consequently, to a reduction of secondary side heat removal. Owing to the imbalance between heat production and its removal by the SG secondary side, the temperature and consequently also the pressure increase in both the primary and the secondary circuits. These are the dominant effects for such events. The margin to the limiting value of the DNBR will also be reduced due to the temperature rise of the reactor coolant and, for some events, also due to primary coolant flow reduction, although a bulk boiling crisis is typically not expected in the core.

Auxiliary and/or emergency feedwater pumps are used to supply water to the SG after the normal feedwater pump trip.

The event is categorized as an AOO; acceptance criteria Nos (1)–(3) (Section 3) apply.

9.2.2. Specific suggestions for analysis

Loss of off-site power is usually not considered for this type of event. Conservative conditions for reactivity feedback typically occur at the BOC.

Depending on the selected acceptance criterion (short term or long term cooling objectives), application of a single failure criterion needs to take into account potential failures in auxiliary feedwater pumps, pressurizer relief or safety valves, SG safety valves and steam dump stations.

9.3. REDUCTION OF STEAM FLOW FROM STEAM GENERATOR FOR VARIOUS REASONS

9.3.1. Initiating events and safety aspects

Reduction of the secondary side heat removal capability can also be caused by an increase of the SG secondary side pressure and temperature as a consequence of steam flow reduction. The steam flow reduction and/or loss of a heat sink may be caused by one of the following events:

- Reduction of the steam flow extracted from the SGs due to control system malfunction;
- MSIV closure;
- Turbine trip, with corresponding admission valve closure, due to many different possible failures;
- Loss of external electrical load;
- Loss of condenser vacuum.

The safety aspects of these events are similar to those for a feedwater pump trip. All these events are categorized as AOOs and acceptance criteria Nos (1)–(3) apply.

Typically, a simultaneous complete loss of electric supply to the MCPs is postulated, creating additional concern about the achieved minimum DNBR in the reactor core during the transient.

9.3.2. Specific suggestions for analysis

All the suggestions made in Section 9.2.2 are also valid for these events.

10. ANALYSIS OF CORE COOLING FOR LOCAs

10.1. INITIATING EVENTS AND SAFETY ASPECTS

A LOCA is caused by loss of integrity of the primary circuit or its associated pipes and devices. The direct cause of such an accident is a material defect, material fatigue, an external impact (internal missiles or heavy loads) or a device failure during the operation of the plant. Inadvertent opening of the pressurizer valves or any other isolation valve on the primary system boundary can also be analysed as a LOCA. The spectrum of postulated leakage sizes within the reactor coolant pressure boundary has been divided in various ways depending on the selection of the acceptance criteria. In this report, the subdivision is made into two groups, namely large break LOCAs (LB-LOCAs) and small break LOCAs (SB-LOCAs).

LB-LOCAs mainly include a full or partial rupture of the main circulation line, typically with break areas higher than 25% of the cross-section of the main circulation line. Ruptures of the major pipes connected to the primary circuit, such as the pressurizer surge line or the accumulator discharge lines, can also be considered as LB-LOCAs. In the case of an LB-LOCA, the loss of primary coolant cannot be compensated for by the ECCS prior to substantial depressurization and loss of coolant inventory from the primary circuit.

SB-LOCAs include such breaks, smaller in size in comparison with LB-LOCAs, which cannot, however, be compensated for by the make-up system and thus require activation of the ECCS.

All LOCA events are classified as accidents.

Loss of primary coolant into the secondary circuit, for example due to the rupture of the SG tubes, requires special consideration and is discussed separately in Section 11.

LOCAs have various safety aspects leading to potential violation of the acceptance criteria, as follows:

(a) The high velocity stream of the escaping primary coolant generates jet forces and reaction forces (leading to pipe whip) that endanger other systems close to the ruptured pipe and containment internals. Similarly, mechanical damage can be caused by the MCP rotor overspeed induced by a very high primary coolant flow directed to the break.
(b) Pressure wave propagation in the primary circuit at the very initial stage of the accident leads to pressure differences across the reactor internals with large forces acting on the internals.
(c) Loss of coolant resulting in core dry-out leads to loss of coolability of the core in spite of reactor shutdown; fuel rods are heated, cladding mechanical properties are degraded and the integrity of the cladding can be lost due to internal fission gas overpressure or thermally induced stresses.
(d) Cladding ballooning and geometrical distortions of the fuel assemblies may endanger the long term coolability of the reactor core.
(e) At high temperatures, the cladding material reacts with the steam in an exothermic reaction, with hydrogen as a by-product. This reaction represents an additional heat source for the cladding and can cause further

degradation of the cladding material due to oxidation, and the potential for hydrogen burning or explosions inside the containment.

(f) High energy coolant outflow into the containment leads to pressurization of the containment.

(g) Containment pressurization together with high radioactivity in the containment atmosphere (due to fuel damage) leads to leakages into the environment with potential radiological consequences.

This section deals mainly with safety aspect (c), and partially with aspects (d) and (e).

Aspects (a) and (b) require special methodologies not covered by this report. Aspects (f) and partially (g) are dealt with in Sections 16 and 17, respectively.

As already emphasized several times, the selection of acceptance criteria and consequently also the selection of conservative conditions differs considerably depending on the safety aspect under consideration. Generally, different calculations have to be performed for an analysis of different aspects of the accident.

The relevant acceptance criteria typically corresponding to the safety aspects under consideration are Nos (6), (8) and (9–13) (Section 3).

10.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

The break location varies along the primary circuit and the break size has to be selected to maximize core heat-up; maximum core heat-up is usually obtained for a break size smaller than a full size break.

All sources of generated and stored energy in the RCS and in the secondary side have to be adequately considered, with a realistic timing for each of the sources. Selection of a conservative power distribution in the core depends on the break size. For an LB-LOCA the power distribution shape is not so important and it is mainly the power peaking factor that plays a crucial role. For an SB-LOCA with core uncovery, the top skewed power distribution is usually conservative, which is typical for EOC. Conservative conditions for an SB-LOCA may be found by parametric calculations taking into account core power shape changes and fuel element properties. All metallic structures have to be adequately modelled to account for the accumulated heat.

Conservatively low characteristics (low efficiency, low capacity and a long delay before action is taken) need to be used for reactor scram, and for both high and low pressure ECCS pumps. A sensitivity study with respect to the hydraulic resistance of the hydro-accumulator discharge lines needs to be performed to maximize core heat-up. The flow distribution into the parallel core channels, particularly distribution of the ECCS coolant from the upper plenum into the fuel channels, and the flow intake from both ends of the fuel channel have to be considered. Three dimensional effects in the RPV, especially in the reactor upper plenum and the downcomer, need to be adequately modelled; the applicability of 1-D models with an appropriate (preferably experimentally based) selection of nodalization and pressure loss coefficients is not, however, excluded for this purpose.

Attention has to be paid to proper modelling of the MCP characteristics at a very high flow rate in the affected loop; the pump characteristics for the turbine flow regime are needed.

LOCAs are typically analysed with the assumption of loss of off-site power. Consequently, failure of a diesel generator is a typical assumption for the single failure to minimize safety injection. For LB-LOCAs, the loss of off-site power is usually assumed to be coincident with the break, but for SB-LOCAs parametric studies may be necessary to find the bounding case.

The long term behaviour of LOCAs, such as the loop seal effect (blocking of steam outflow by water plugs in the circulation loops), may cause overheating of the core: such cases can be analysed with sufficiently detailed models.

The tolerance of the core to the residual insulation material and other debris in the ECCS water during recirculation needs to be studied. Possible boron precipitation problems in the reactor during long term boiling also have to be studied.

Adequate selection of nodalization is crucial for the quality of the analysis. Therefore, the nodalization needs to be carefully verified and validated [3]. Sufficiently detailed nodalization has to be selected particularly in the vicinity of the break. This should adequately account for removal of ECCS water out of the break (vessel bypass). Consideration has to be given to detailed modelling of dead ends (e.g. the reactor upper head and CR guide tubes), mainly regarding coolant temperature stratification. These ends may act as pressure maintaining components during the accident. For SB-LOCA analysis, sufficiently detailed nodalization needs to be applied for loop seals and SGs in order to adequately consider steam binding and primary to secondary heat transfer, which strongly influence the course of accidents and core heat-up.

Generally, the analysis for all LOCAs has to show how a plant is brought into a safe controlled state, for example into cold shutdown conditions.

11. ANALYSIS OF PRIMARY TO SECONDARY SYSTEM LEAKAGE

11.1. INITIATING EVENTS AND SAFETY ASPECTS

Primary to secondary system (PRISE) leakages are caused by a single or multiple SG tube rupture or, in addition, for WWER designs with horizontal SGs, by an SG collector break or an SG primary collector cover lifting off. The accident may be initiated by fatigue failure of the SG tube wall, material defects produced during manufacture or, as a special case, as an induced failure from another initiating event (e.g. steam line break or injection of cold feedwater on the hot SG tubes).

Owing to the large number of SG tubes, the probability of SG tube rupture is relatively high. This accident type is therefore normally considered in the design. The likelihood of a large PRISE leakage due to a collector break itself was not included in the original design of WWER reactors. However, operating experience and in-service inspections have demonstrated the potential for large SG leaks for this design.

The safety aspects associated with a PRISE accident are as follows:

— Direct radioactivity release to the environment, bypassing the containment via steam dumps and/or via the SG safety valves;
— Reduced core cooling (short and medium term);
— PTS of the RPV due to cooldown and subsequent re-pressurization following break isolation on the primary side (typical for WWER-440s);
— Boron dilution and reactivity insertion due to reverse flow (from secondary to primary side) through the break;
— Potential for an SA initiator because of long term loss of ECCS water inventory.

The main feature of PRISE leakage is containment bypass resulting in a direct radioactivity release to the environment and an irreversible loss of primary coolant. The primary water becomes radioactive during reactor operation due to the activation of corrosion products and the presence of fission products. The water in the secondary system may also contain some radioactive material.

Typically, the short term core cooling during PRISE accidents is assured even with a minimum design ECCS configuration. However, in the case of late leak isolation, the ECCS tanks can be completely depleted and long term core heat-up may occur.

The safety aspects associated with PTS, boron dilution and core cooling (LOCAs) are described in other sections (10, 13 and 14) and are therefore not reflected here. Only the radioactivity release and long term loss of ECCS water are considered in the following text.

PRISE leakages are categorized as accidents. The main acceptance criterion relevant for these events is No. (8) (Section 3). It is also required that the ECCS water resources be sufficient to cope with the accident without early endangerment of the core cooling, assuming maximum PRISE leakage and late operator interventions.

11.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

The radioactivity release to the environment depends on the following factors:

— Quantity of coolant released to the atmosphere;
— Primary coolant activity;
— Secondary coolant activity;
— Spiking of the iodine isotopes and the noble gases;
— Degradation of fuel rods due to high temperatures;
— Position of the tube rupture in the broken SG and the decontamination factor;
— Initial power level of the plant.

The radiological consequences of PRISE leakages are usually analysed in two sequential steps:

(1) System analysis using system thermohydraulic codes;
(2) Analysis of radiological consequences using the results of the thermohydraulic calculation as boundary conditions.

The process of radioactivity transport is closely related to the two phase outflow of the primary coolant. Some computer codes offer the possibility to calculate the transport of radioisotopes (including radioactive decay) in both water and steam phases of the coolant in addition to the two phase flow modelling. If needed for separation of the processes of radioactivity transport and coolant transport, a conservative approach has to be used (e.g. not considering the dilution of the active primary coolant due to ECCS injection or the mixing of the primary coolant with the secondary coolant in the SG affected).

The maximum primary to secondary side leak and the maximum activity of the primary coolant are conservative assumptions. To assume of the full configuration of the ECCS and the make-up system may lead to a higher loss of coolant. In addition, the loss of off-site power and the application of the single failure criterion can make the process more severe as only limited means are available for mitigation of the accident.

The primary coolant activity used in the analysis needs to be estimated according to plant operational limits and conditions. The spiking effect caused by the increased release of water soluble products from micro-defects in the fuel cladding into the coolant has to be considered in the estimation of the coolant activity. This effect varies significantly and quite irregularly for the same reactor. The spiking effect has been noticed most clearly for ^{131}I. A typical upper estimate for the spiking factor (SF) for iodine isotopes is SF = 100 (activity increased by a factor of 100 during transient conditions as opposed to the iodine activity during a steady state operation). The average value during the first half hour is typically SF = 50. The most conservative value for the spiking factor is 500. The spiking of noble gases is comparatively small and has a completely different time trend. The spiking factor proposed in the literature for noble gases is SF = 10, Ref. [10].

The possibility that secondary side SG safety valve(s) or steam dump(s) to the atmosphere are stuck open after their actuation has to be considered, in particular if these devices are not qualified for two phase flow. Generally, the systems not qualified for relevant accident conditions need to be considered to fail, particularly if such a failure leads to a worsening of the consequences (especially failure of isolation devices to close).

A single failure in safety systems, for example failure to close one steam dump valve to the atmosphere or an SG safety valve in the affected SG, has to be considered. However, if the SG safety valves are not qualified for two phase flow or water flow operation, their failure to close under such conditions shall be considered. The analysis is to be performed until the final safe condition of the plant is reached, for example with the primary pressure below the set points of the steam dumps to the atmosphere and the SG safety valves in the case of their proper operation, or until the primary system is depressurized to atmospheric pressure.

In the case when the accumulator pressure is higher than the set points of the secondary side safety relief valves, the isolation of the accumulators needs to be considered only in the case where this is clearly stated in existing emergency operating procedures.

Operator interventions are essential for the termination of a PRISE leak and mitigation of the radiological consequences. These interventions are to be considered after a sufficient time, which is typically 30 min but in any case not shorter than 10 min after the initiating event. It should be demonstrated that this time is sufficient for reliable identification of the affected SG by plant operators.

National limits placed on releases of radioactivity, individual doses and collective doses shall be used in the analysis.

Final safe conditions of the plant can also be assumed after reaching conditions for operation of the RHR system.

12. ANALYSIS OF ANTICIPATED TRANSIENTS WITHOUT SCRAM

12.1. INITIATING EVENTS AND SAFETY ASPECTS

The initiating event for ATWS is an AOO combined with a reactor scram failure when required to actuate. It is typically assumed that the scram signal fails or the CRs do not move for mechanical reasons. Typical initiating events to be analysed are to be selected from a group of initiating events with higher probability (except the probability of reactor scram) and may be summarized as follows:

(a) Failure of main heat sink:
- loss of condenser vacuum;
- closure of MSIVs;
- turbine trip.

(b) Increase of steam flow:
- opening of turbine bypass valves;
- opening of steam line safety valves.

(c) Loss of main feedwater supply.

(d) Loss of alternate circuit (AC) power.

(e) Inadvertent CR withdrawal:
- at a particular power level;
- at subcritical or low power startup conditions.

(f) Loss of load.

Usually, generic studies are designed to identify which of these initiating events are the limiting cases. Very often, loss of the main feedwater gives the most adverse consequences.

The safety aspects of these accidents, leading to challenging of acceptance criteria, are as follows:

(a) Primary coolant temperature rise as a consequence of inadequate heat transfer to the ultimate heat sink;
(b) Primary coolant pressure rise as a consequence of primary coolant expansion;
(c) Possible rise of pressurizer water level to the valve nozzles, with 'solid water' being discharged to the pressurizer relief tank;
(d) Fuel cladding temperature rise as a consequence of possible DNB;
(e) Formation of a low boron concentration water plug in loop seals as a consequence of a possible primary coolant boiling–condensing cooling mode due to degradation of the primary coolant level below the hot leg elevation;
(f) Radiological consequences due to loss of primary coolant via the pressurizer safety valves in conjunction with a loss of fuel cladding integrity.

This section deals mainly with the first four aspects of the list above. The remaining aspects are dealt with separately (Sections 13 and 17). Acceptance criteria Nos (4)–(8) apply for this type of accident. Of course, these criteria apply only if ATWS are included among the DBAs. However, ATWS are typically considered as BDBAs, which are either excluded from the analysis or, if not, a best estimate approach is accepted for the analysis. It is therefore suggested that detailed conditions for the analysis are agreed upon between the analyst and the user of the results.

12.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

The level of conservatism for this type of accident is typically subject to specific national requirements, but usually a best estimate analysis is acceptable. Owing to the very low probability of coincidence of the initiating event and a reactor scram failure, it is acceptable to consider that other plant systems, including the plant control systems, will operate in accordance with the design requirements if they are qualified for post-accident conditions. This specifically means that all trains of safety systems can be considered operable, i.e. the single failure criterion may not be adopted. Loss of power is also not considered. There are, however, other approaches which assume a similar choice of calculation parameters and of failures in safety systems as for conservative analyses of other accidents [12].

It is acceptable to use realistic data for the Doppler reactivity and CR effectiveness if they are derived from experiments or adequately validated codes. It is, however, recommended to use reasonably conservative values for other parameters including CR position, fuel temperature, hot channel peaking factor and local linear heat rate.

As CRs are not considered to stop the generation of fission power, the boron injection system is an essential means for shutting down the reactor (in addition to coolant boiling). The minimum injection flow of this system needs to be considered. For analysis, transport of boron has to be assumed for the liquid phase only.

If reactor coolant pumps are stopped following the event, natural circulation is needed, together with the boron injection system, to transport boron into the reactor and to shut it down. The code used to analyse ATWS events must be capable of calculating natural circulation under the corresponding operating conditions. This includes both one phase and two phase natural circulation models.

Steam generator modelling has to be such that the heat transfer from the primary circuit to the secondary circuit is calculated as realistically as possible. This means that when the heat transfer tubes are uncovered as the level drops, there is a resultant decrease in the heat transfer rate to the secondary side.

If the boiling–condensing cooling mode takes place during the accident as a result of excessive loss of coolant inventory, the boron concentration increases in the core area. However, during this phase a water plug with very low boron concentration may be formed in the cold legs, especially for designs with the potential for formation of loop seals. If the water plug for any reason departs from the loop, there is a chance that it may enter the core without any major mixing during its transport to the core and cause a recriticality of the core. The diluted slugs formed during the boiling–condensing mode can cause large reactivity changes. ATWS events should also be analysed taking into account this phenomenon.

Attention has to be paid to an adequate modelling of the pressurizer to account for pressurization of the primary circuit, for the possibility of completely filling the pressurizer with water and for water discharge through the pressurizer valves.

Preferably a 3-D reactor kinetics calculation capability, and at least a 1-D capability, is required in most ATWS event analyses. The capability for hot channel analysis needs to be available. A detailed nodalization needs to be selected in particular for the core, both the reactor plenums and the SGs on both the primary and secondary sides.

13. ANALYSIS OF BORON DILUTION ACCIDENTS

13.1. INITIATING EVENTS AND SAFETY ASPECTS

Boron dilution events are caused by an inadvertent decrease of the boron concentration of the primary coolant. The boron dilution process can be homogeneous or inhomogeneous.

Homogeneous dilution may occur due to CVCS maloperation when the primary pumps or natural circulation are sufficient for uniform mixing of the whole primary circuit. The process is slow and the operator usually has sufficient time for corrective measures. It is usually classified as an AOO.

Inhomogeneous or local dilution takes place if a slug of non-borated water or low boron concentration water is formed in the primary loop. There is a potential for rapid core power increase if the slug can be transported without sufficient mixing to the reactor core by pump startup or by re-establishing natural circulation. The slug could be formed by non-borated water injection of CVCS into stagnant loops or through leakages from other systems. The process of slug formation could appear during different modes of operation. These sequences are called external dilution cases and they are classified as accidents.

During accidents with decreased reactor coolant inventory such as SB-LOCAs, PRISE leaks and ATWS there is a possibility for an inherent dilution when the boiling–condensing cooling mode occurs. During this mode, the steam produced in the core condenses in the SG and the non-borated condensate collects in the loop seal, forming a diluted slug.

In SB-LOCA and ATWS events the liquid level in the RPV may decrease below the hot leg elevation. As a consequence, steam begins to flow to the SG and condenses there. Because the steam carries practically no boron, the boron concentration in the cold leg loop seals begins to decrease. If for some reason this water plug with low boron concentration begins to flow towards the core and enters there without any major mixing with the borated coolant, the event results in a positive reactivity insertion.

During a PRISE leakage the primary pressure during the accident may decrease below the affected SG pressure, reversing the break flow. Water with a low boron concentration flows from the affected SG to the primary circuit and further to the core. The situation becomes worse if feedwater isolation has failed.

Generally, the course of boron dilution accidents strongly depends on:

— The formation of a slug with reduced boron concentration;

— Mixing of the slug with the primary coolant during its transport to the core;
— Reactor power excursion due to propagation of the slug into the core.

Inadvertent positive reactivity insertion to the core may lead to recriticality and a power increase in the core. The course of such an accident is similar to those of other types of reactivity accidents, either slow (i.e. CR withdrawal) or rapid (i.e. CR ejection). While for slow accidents there is typically a sufficient time margin for a timely isolation of the source of non-borated water from the primary circuit, for rapid accidents the power increase and its consequences can be serious, even if reactor scram occurs.

If boron dilution occurs during the course of another initiating event (e.g. a LOCA or ATWS) then the safety aspects of this particular event are also relevant.

For accidents with pure water injection from an external source, the relevant acceptance criteria may be as follows:

(a) For inhomogeneous external boron dilution events with the MCP startup accelerating the slug into the core, the acceptance criterion could be that no recriticality of the core (after reactor scram) is allowed.
(b) For inherent boron dilution events, where the slug is typically transported by re-establishment of natural circulation, the acceptance criteria have to be combined with the acceptance criteria of the initial accident concerning the fuel heat-up.

13.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

For a conservative analysis, the initial boron concentration and mass flow of injected non-borated water has to be maximized, while the circulating volume of primary coolant, as well as the estimated mixing of non-borated water with primary coolant, has to be minimized.

For slow processes with sufficiently high flow rates in the primary circuit, a simplified approach with uniform coolant mixing may be used.

For a sudden reactivity insertion the process is very complex and has still not been fully investigated. Modelling needs to cover the following aspects of the accident:

— Overall system response;

— Process of diluted boron mixing in the RPV downcomer and in the lower plenum, by application of a 3-D thermohydraulic computer code with turbulent mixing;
— 3-D neutron kinetics;
— Mechanical response of the fuel.

Various aspects of the accident may be analysed in several separate steps using different computer codes.

14. ANALYSIS OF PRESSURIZED THERMAL SHOCKS

14.1. INITIATING EVENTS AND SAFETY ASPECTS

PTS represents one aspect of a variety of accident scenarios, characterized by overcooling of the RPV wall (typically non-uniform), occurring under high pressure in the primary circuit. PTS results in high temperature gradient induced stresses, threatening the integrity of the RPV wall. Conditions could be worsened by RPV repressurization. PTS aspects have to be considered for all accidents with ECCS cold water injection and for accidents leading to cooldown of the downcomer, such as rapid depressurization of the secondary circuit.

Consequently, the following groups of initiating events need to be considered:

— LOCAs, especially small and medium breaks;
— Stuck open pressurizer safety or relief valves;
— PRISE leaks;
— Large secondary circuit leaks;
— Inadvertent actuation of high pressure injection or make-up systems;
— Accidents resulting in cooling down of the RPV from the outside; this reflects flooding of the reactor cavity, for example by the ECCS or spray system.

Rapid cooldown, typically non-uniform, of the RPV wall is caused by a cold water plume in the reactor downcomer or by external flooding of the RPV. The conditions are further worsened if the primary circuit pressure is high. The Resulting thermal stresses in the RPV wall may potentially threaten its integrity.

Other aspects of relevant accident scenarios, such as loss of primary or secondary coolant inventory or of containment pressurization, are of secondary importance from the reactor integrity point of view and they are not considered here. The relevant acceptance criterion for PTS is No. (16) (Section 3).

14.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

Owing to the complexity of the PTS, the compliance with criterion No. (16) is typically analysed in the following consecutive steps:

(1) Neutron fluence calculation for the period of the reactor operation;
(2) System thermohydraulic calculation;
(3) Fluid–fluid mixing calculation;
(4) Structural analysis.

This guidance does not deal with the neutron fluence calculation, which requires a special methodology, usually not considered as corresponding to the term 'accident analysis'. Other steps of the analysis are interrelated as follows:

(a) The system thermohydraulic calculation should provide the following parameters as a function of time for the RPV wall temperature and stress calculations:
   — Downcomer coolant temperature distribution;
   — Coolant to wall heat transfer coefficients in the downcomer;
   — Primary circuit pressure.
(b) The objective of the structural analysis is to calculate the temperature and stress fields in the RPV and to evaluate the stress intensity factors for postulated cracks loaded by thermohydraulic transients. The results of the analysis for the structure are typically as follows:
   — Temperature distribution;
   — Stress–strain distribution;
   — Loading of postulated defects in terms of stress intensity factor.

The sources of uncertainties in the analysis are associated mainly with the following aspects:

— Thermohydraulic transient analysis (prediction of flow stagnation, cold coolant injection rate, cooldown rate and pressure behaviour);
— Fluid temperature distribution and coolant to wall heat transfer coefficients distribution in the RPV downcomer;

— Assumptions of the structural analysis model including boundary (the coolant temperature distribution) and initial conditions;
— RPV material properties including fracture toughness in initial as well as end of life states;
— Neutron fluence;
— Method of calculation of the stress intensity factors;
— Crack geometry and size with respect to NDT effectiveness;
— Operator action.

Therefore, careful consideration needs to be given to the aspects above and corresponding sensitivity studies have to be carried out.

In the analysis, attention is to be paid to scenarios leading to flow stagnation, which causes a higher cooldown rate and formation of cold plumes in the downcomer. Reliable prediction of the flow stagnation is very complicated but essential for a PTS analysis. Various phenomena influence the onset of flow stagnation, such as coolant discharge, ECCS injection, SG heat transfer and coolant mixing in the downcomer. Nodalization of SGs and of the downcomer plays an important role in predicting the occurrence of flow stagnation. Typically, earlier prediction of the flow stagnation represents a conservative direction. Particularly for the time period close to the primary flow stagnation, the existence of downcomer cold plumes, causing the temperature and heat transfer coefficient distributions to be non-uniform and asymmetric, should be taken into account.

For cases with subcooled water in the whole primary circuit, modelling of the discharge flow (selection of the critical flow model and specification of the hydraulic resistance of the discharge tube) may have a significant effect on the results. For conservative analysis, the ECCS injection flow rate and the secondary system cooling capacity have to be maximized, while the ECCS water temperature and thermal mixing of the ECCS water with primary coolant have to be minimized.

Modelling of the natural circulation is typically validated for the steady state conditions during normal operation of an NPP. In many cases the conditions during accidents differ significantly from normal operation, and the validity of modelling needs to be carefully considered. The same consideration also applies for the discharge flow through the safety valves, for which the equivalent size is defined for the single phase steam outflow. For complex accident situations with water and a two phase discharge, the flow also depends on the model of the critical outflow applied.

In PTS analysis, the various aspects to be analysed (overall cooldown of the RPV and asymmetric plumes along the RPV wall) may require different sets of conservative assumptions. The final judgment about the conservatism may often not be made on the basis of thermohydraulic analysis alone, since structural analysis is needed for such a judgment as well.

Failure of the normal operating system needs only to be considered if it leads to a more severe PTS loading.

In addition to modelling of the internal coolant behaviour, a detailed modelling of the coolant thermal stratification (the fluid–fluid mixing calculation) in the primary loops is also required.

The stresses due to internal pressure, temperature gradients (as well as the residual stresses for both), cladding and welds have to be taken into account. Plasticity effects should preferably be considered. Temperature dependent material properties for base/weld materials and cladding, and changes of the material properties due to neutron irradiation also have to be included. For the selection of the location of the postulated surface or subsurface cracks (including their orientation) in the limiting areas of the vessel, consideration needs to be given to the stress level, to the material degradation and to the results of NDT.

15. ANALYSIS OF RESIDUAL HEAT REMOVAL DEGRADATION DURING SHUTDOWN OPERATIONAL MODES

15.1. INITIATING EVENTS AND SAFETY ASPECTS

The loss of RHR during shutdown operational modes can typically occur due to:

— Reduced coolant inventory;
— Degradation of primary coolant circulation;
— Failures of RHR equipment;
— Failures of support systems.

The main events which may cause loss of core cooling due to the degradation of primary coolant circulation, resulting from manipulations during shutdown modes, are the following:

— Excessive RCS draining, for example due to erroneous opening of drain valves or by a malfunction of the letdown lines;

— Erroneous nitrogen injection into the RCS from the nitrogen supply system or from the isolated loops through leaky MIVs (for reactors equipped with such a feature);
— Primary depressurization following a rapid cooldown, leading to bubble formation in the RCS.

The main events which cause loss of core cooling due to support system and RHR equipment failure are:

— Loss of power supply during reactor shutdown conditions, which could be more likely because of the reduced number of available power sources under such conditions;
— Loss of component cooling water, which could lead to failures of some safety system components that perform the core cooling function, such as low pressure ECCS pumps;
— Loss of essential service water, which leads to the loss of the heat exchanger cooling capability;
— Cavitation due to failure of RHR pumps or malfunction of control valves and/or orifices; these events are more probable with the primary water level under the hot leg elevation.

For a loss of RHR, there is a potential for heating the coolant in the core to its boiling point. Once boiling begins, the RPV water level decreases, leading to a possibility for uncovering the fuel assemblies. Finally, fuel damage may occur if the situation continues. If the containment function is also degraded, radioactivity can be released to the environment. For these types of event, acceptance criteria Nos (8), (12), (17), (18) and (19) apply (Section 3).

It should be noted that boiling in the primary circuit does not always lead to a dangerous decrease of the core water level. If RCS integrity is maintained and the SGs are available for heat removal, the core may remain covered and the residual heat is extracted by natural circulation or in reflux condensing heat transfer mode. If the primary circuit is open, core uncovery takes place only if there is no automatic action or no operator action in a sufficiently short time.

15.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

The computer codes used for system thermohydraulics have to include special features required for adequate modelling of relevant physical phenomena, in particular:

— Heat transfer in the natural circulation mode.
— Adequate modelling of SGs in the natural circulation mode.
— Adequate modelling of heat losses from both primary and secondary circuits.
— Proper consideration of changes in pressure loss coefficients due to possible changes in coolant flow directions and flow regimes.
— Internal coolant circulation in the reactor through the fuel channel and core bypass.
— Water carryover by steam at low primary system pressure.
— Interfacial friction correlations between steam and water; these are supposed to be well known for high pressure states in different two phase flow regimes; their range of validity needs to be checked and/or extended to low pressure.
— Modelling of water entrainment; it can strongly influence the time to uncover the core, depending on the size of the openings in the primary circuit during some shutdown states.
— Accumulation of non-condensable gases in some parts of the primary circuit.
— Qualification of SG water reflux, as a condensing mode of cooling, particularly when the non-condensable gas inventory is large enough to limit heat removal by the SG.

In cold shutdown conditions, special models of the primary circuit may be needed due to the fact that openings (access holes) can be intentionally opened in the pressurizer or in SGs and that a free water level exists when the RPV head is removed. An intermediate water level in the hot leg may exist under shutdown conditions for some plants. For such states, detailed modelling is needed. This could possibly be done by existing codes with a sufficiently modular structure. The sensitivity to the nodalization needs to be checked.

Attention should be paid to the input data with regard to pressure differences developing in the primary circuit during the initial states of shutdown conditions. Pressure differences are of great importance in determining flow rates and water levels in the loops.

The presence of non-condensable gases influences the physical process during shutdown modes, particularly when the primary circuit is open through the venting lines. Nitrogen can accumulate in some parts of the primary circuit and leads to unexpected re-pressurization. Thus, system computer codes need to be capable of taking non-condensable gases into account.

For an analysis of events occurring during operation at reactor power, the required level of conservatism is further achieved by adoption of certain modelling assumptions, for example by neglecting some physical processes and by selecting bounding rather than realistic correlations. An overconservative combination of assumptions can, on the one hand, assure sufficient safety margins regarding acceptance criteria, but, on the other hand, may lead to a completely unrealistic view of the course of an accident. The above mentioned overconservatism can be particularly counterproductive in an analysis of accidents during shutdown modes, where the correct operator action plays a decisive role and the time margin to perform such action often represents the main result of this analysis. It is therefore important that the computer codes used, as well as the various code options, be of the best estimate type. A best estimate analysis is recommended whenever a calculation is used to provide the basis for the specification of adequate operator action. A reasonable uncertainty is afterwards added to provide a time margin in which to perform the required action.

During shutdown modes, several protective barriers against radioactivity releases are partly or fully degraded. On the other hand, the time available until loss of safety functions is usually sufficiently long to perform recovery actions, using redundant, alternate or diverse systems, components and methods. External equipment (e.g. power sources) can also be considered as an option for temporary replacement of a standard component, provided that measures are taken for the possibility of its quick installation.

For recovery actions, such systems and components can be considered which are designed to operate under shutdown modes and which, in accordance with procedures, either are in service or can be placed in service (including repair) by manual or automatic actuation within an acceptable time interval. This time interval has to consist of both diagnosis and response components. Any consideration of manual action is to be based on carefully checked and verified procedures. A system under maintenance can be considered as an option for recovery only if the maintenance time is short enough in comparison with the duration of the particular plant operational mode.

In the analysis, consideration needs to be given to the fact that some of the instruments may not be operable or may produce misleading information. This is particularly important if operator diagnosis and response are based on instrument readings. Unavailability of instruments can be caused either by their disconnection in accordance with operating procedures or by damage caused by an erroneous manipulation. Another reason for the unavailability of instruments can be an interruption of power supply to the instruments. Misleading or confusing information can easily be obtained if an instrument is working outside its measurement range or under unexpected operating conditions. An example is the temperature difference measurement in the loop with negative natural circulation flow or with stagnant fluid. Special attention has to be devoted to such devices, which are essential to monitor RHR, such as instruments that monitor primary coolant temperature, natural circulation flow and RPV level.

16. ANALYSIS OF PRESSURE–TEMPERATURE TRANSIENTS IN THE CONTAINMENT

16.1. SHORT TERM CONTAINMENT PRESSURIZATION

16.1.1. Initiating events and safety aspects

Pressure and temperature loadings of the containment structure are one of the consequences of a LOCA, a steam line break or a feedwater line break within the containment boundary. ‘Short term’ refers here to the initial period of the accident encompassing the largest release rate of the primary/secondary coolant, during which the maximum containment pressure is reached. The containment pressurization in its short term phase results mostly from the release of the stored energy of the RCS.

Owing to the post-accident pressure buildup in the containment, the release of radioactivity occurs with a potential radiological impact on the environment. The integrity of the containment as the last barrier against radioactivity releases is endangered by:

— Local effects of the accident, such as jet impingement, pipe whip and internal missiles;
— Combined pressure and temperature loads acting on the internal structures of the containment and on the containment steel wall itself.

Only the combined pressure and temperature loads will be dealt with in this section. Other aspects of primary or secondary leaks, such as core cooling or PTS, are partly covered by other sections of this report.

The relevant acceptance criteria for the containment analysis are Nos (14) and (15) (Section 3). It should be noted here that conservative conditions relevant to these acceptance criteria differ substantially from the conditions relevant to criteria for other safety aspects (see Section 10 on LOCA analysis).

16.1.2. Specific suggestions for analysis

The initial atmospheric conditions within the containment and its subcompartments need to be selected so as to maximize the peak containment pressure. The allowable ranges of the temperature, pressure and humidity prior to the accident have to be considered.

The mass and energy released to the containment (including their time variation) need to be considered conservatively high, including all sources of stored and generated energy such as heat removed by accumulators of the ECCS and water spilled into the containment, heat transfer from the secondary circuit, residual heat and heat generated from metal–water reactions. For all heat sources and heat sinks, a realistic timing has to be taken into account in order to avoid too unrealistic results. The effect of timing differs considerably for different types of containment. Specific attention needs to be given to modelling metal structures in the containment, since they may represent a significant heat sink during short term pressurization.

Pressure propagation at the very initial stage of the accident is to be investigated taking into account post-accident water–steam–air mixture inertia, to confirm that pressure oscillations in the containment internal volume and/or its compartments do not endanger the containment integrity. Uncertainties in the prediction of pressure and pressure differences need to be evaluated, for example, by parametric analysis of the following factors which have a strong influence on the results:

— Nodalization;
— Flow resistances (often represented by flow contraction coefficients) for the openings between subcompartments;
— Heat exchange with solid heat absorbing/releasing structures;
— Water carry-over through openings.

Subcompartment nodalization schemes have to be chosen in such a way that there is no substantial pressure gradient within a node. The nodalization scheme needs to be verified by a sensitivity study that includes increasing the number of nodes until the calculated peak pressures converge to small resultant changes (nodalization sensitivity study). However, this approach is not applicable for consideration of local phenomena, for example vortexes.

If flow paths (flow channels, corridors, flaps and openings) are present which are not open at the time of pipe rupture, then the flow area and flow resistance, as a function of time after the break, have to be based on adequate analysis.

Pressure suppression containments require additional code capabilities to model the effectiveness of the pressure suppression function.

For conservative calculation of the maximum pressure in the compartments, the calculation of the flow rate through all the flow paths within the model has to be based on a conservative assumption regarding water entrainment. If not proven otherwise by experiments, a homogenous mixture in thermal equilibrium, with 100% water entrainment, needs to be assumed.

Appropriate safety margins have to be considered if thermal non-equilibrium and non-uniformities in the containment atmosphere composition are not sufficiently modelled.

16.2. LONG TERM PRESSURE–TEMPERATURE TRANSIENTS

16.2.1. Initiating event and safety aspects

Long term pressure–temperature transients cover the continuation in time of the short term containment pressurization following a loss of coolant, as described in Section 16.1. Long term behaviour is typically analysed for a time period of several hours until the containment internal overpressure is sufficiently minimized and radioactive releases substantially reduced. During this period, the containment pressure is typically decreasing, with few temporary exceptions.

The pressure–temperature behaviour of the post-accident containment atmosphere results from an imbalance between heat sources and heat sinks, namely:

(a) Heat sources:
- Reactor residual heat released typically by evaporation of the coolant;
- Accumulated heat in structures of both primary and secondary circuits;
- Heat released by potential hydrogen burning or explosions (more typical for SAs).

(b) Heat sinks:
- Heat absorption in the containment walls and other heat absorbing structures;
- Spray system operation;
- Fan and vent cooler operation;
- Pressure suppression heat sinks (water pools, water trays and ice baskets);

— ECCS operation; the ECCS strongly influences the ways of removing reactor residual heat and often also affects the containment spray water temperature.

The safety aspects of the accidents are as follows:

— Internal containment overpressure together with radioactivity in the containment leads to radioactivity releases through containment leakages.
— Hydrogen explosions inside the containment may endanger its integrity.
— Containment integrity or its leaktightness may also be affected by the containment underpressure resulting from the heat sinks.

Relevant acceptance criteria are Nos (8), (14) and (15) (Section 3).

16.2.2. Specific suggestions for analysis

For long term transients with a strong non-homogeneous energy and mass distribution in the containment, the use of a best estimate code is very important. This is due to the strong coupling between the main parameters and the occurrence of various local effects. In such cases, a conservative approach could lead to an overestimation of the influence of one parameter in relation to the others, thus providing misleading results.

The selection of the degree of nodalization and the selection of nodes have to be made depending on the phenomena that are expected to occur (gas plumes, stratified conditions and a non-homogeneously or only homogeneously mixed atmosphere).

A realistic timing of the heat sources and heat sinks needs to be taken into account to avoid misleading results. The containment walls and internals have to be investigated carefully for their effect on containment pressure reduction. Thermal resistances and capacities, including various heat transfer modes and low conductivity coatings, need to be properly addressed. The cooling capacity of the spray system has to be minimized for a conservative analysis. For such an analysis, heat exchange with the containment walls has also to be minimized, whereas the containment leakages have to be maximized (for the calculation of releases). The capacity of the vent and fan coolers has to be minimized. An adequate assessment of the ECCS performance has to be made in order to determine the containment heat sources resulting from the residual heat and additional boiling of the primary coolant.

In the determination of the containment spray system (both active and passive) heat removal capability, the following factors need to be taken into consideration:

- Location of the spray headers relative to the internal structures;
- Spray patterns due to the arrangement of nozzles;
- Spray drop size spectrum as a function of the differential pressure across the nozzle;
- Effect of the drop residence time and drop size on the heat removal effectiveness.

The consideration of hydrogen sources for the containment atmosphere needs to include:

- Release of hydrogen from the escaping coolant;
- Radiolysis of the coolant in the sump;
- Metal–water reaction in the core;
- Chemical reactions of spray water with materials in the containment.

Thermal non-equilibrium, especially for compartments with high liquid content, needs to be adequately accounted for. Superheating of the containment atmosphere during stem line breaks inside the containment has to be accounted for when analysing temperature loads.

For the analysis of long term processes, which consequently determines the measures needed for SA management, a very detailed nodalization is required.

The equipment needed during an accident has to be qualified to withstand the conditions during the accident; thus, a reasonable equipment capability needs to be assumed in the analysis.

The results of a best estimate analysis require as a measure of their credibility specification of their uncertainties.

17. ANALYSIS OF RADIOACTIVITY TRANSPORT DURING DESIGN BASIS ACCIDENTS

17.1. INITIATING EVENTS AND SAFETY ASPECTS

Radioactivity release is one of the safety aspects for several DBA scenarios. The following cases may be examples of such events:

— LOCAs with radionuclides escaping into the containment and releases into the environment through containment leakages;
— Accidents with leaks from the primary circuit bypassing the containment, such as instrument line rupture and PRISE leaks;
— Leaks from the primary circuit during maintenance, refuelling or other outages.

This report does not deal with transport of radioactivity originating outside the RCS, such as that due to damage to a system containing radioactive gases or liquids.

With all the other safety aspects described in the corresponding sections, this group of accidents, moreover, leads to radioactivity releases from the fuel and the reactor coolant, their transport through the primary circuit, their release and transport to the containment and to the radiological source term released into the environment. The relevant acceptance criterion is No. (8) (Section 3).

17.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

At the time when the reactor power starts to change significantly (to decrease or increase), increases in the iodine, caesium and other radioactive substance concentrations need to be assumed as they are the most extensive increases in connection with power changes that have been observed with the type of plant in question (see also Section 11).

The primary coolant leak rate and the time until the potential isolation of the leak takes place have to be estimated as high, i.e. conservatively. If actions affecting the isolation of a leak or the dispersion of radioactive substances are automatic and designed to cope with a single failure, these actions can be assumed to be effective.

Radioactivity releases which are caused by the liquid part of the leaking coolant and those which are caused by the gaseous part need to be examined separately. It can be assumed that the concentration of radioactive materials in the gaseous part is lower than that in the liquid coolant in the immediate vicinity of the leak. The dispersion coefficient has to be justified by reference to practical observations or test results. As an exception to this, it will be assumed, however, that all the noble gases in the leaking coolant are always fully discharged into the environment.

If the leak occurs with a direct outflow into the environment and the coolant is in liquid form just in front of the leak, all the radioactive substances in the escaping coolant have to be taken into account when calculating off-site doses.

Steam and its radioactive contents which have leaked into the plant systems should be assumed to be transferred into the environment in a way which corresponds to the normal functioning of the ventilation systems.

Part of the iodine which has mixed with the steam needs to be assumed to be gaseous. The distribution of iodine into gas and aerosol portions has to be justified.

If filters are placed in the ventilation systems, the retention factors of the filters need to be selected conservatively.

The time of failure of fuel rods and the number of failed rods have to be selected conservatively, taking into account the results of analyses related to plant behaviour. The number of fuel rods having failed during a LOCA can be obtained from the relevant analysis. However, a conservative assumption of 100% fuel failures for LB-LOCAs is often prescribed by national requirements. The amount of radioactivity released from the failed rods also has to be chosen conservatively. A certain share of the radioactive substances released from failed rods to the primary coolant enters the containment atmosphere directly. The air–steam and liquid volume distribution pattern needs to be justified.

An additional release of radioactive substances from the failed rods has to be assumed later when cooling water enters the rods and carries fission products out from the fuel. This share of radioactive substances, which initially ends up in the water, needs to be justified by experiments, or the assumptions have to be conservative.

Assumptions on the transport of radioactive substances within the containment can be based on experiments if the results are applicable to the situation in question. Alternatively, a conservative model may be used which gives a slower normal release of radioactive substances from the gaseous volume of the containment.

If air is discharged from the containment during normal plant operation, the mixing of radioactive substances with the discharged air needs to be estimated conservatively. The isolation of ventilation systems may be assumed to take place according to the design of the plant system so that any changes in the parameters, used as protection limits during accidents, are assessed conservatively. Before isolation, ventilation has to be assumed to function in the normal way. After isolation, radioactive substances have to be assumed to mix with the gaseous volume of the whole containment. The containment leak rate has to be selected taking into account the tightness requirement set up for the containment and the containment overpressure calculated during the analysis of postulated accidents. Appropriate safety margins need to be employed for the selection.

The halogens which have leaked from the containment should be assumed to be partly in inorganic compounds and partly in organic compounds. The distribution into the various kinds of compound needs to be justified.

The releases caused by the leaks and the potential malfunctions of the ECCS and the leaks of the containment cooling systems outside the containment boundary have to be taken into account conservatively.

The ventilation of the space surrounding the containment has to be assumed to function in the way designed for accident conditions, and the releases arising from a containment leak need to be calculated accordingly. If the ventilation system is used in the normal way with the filters bypassed, the time spent in the possible switch over to the filters has to be justified.

18. ANALYSIS OF SEVERE ACCIDENTS

18.1. INITIATING EVENTS AND SAFETY ASPECTS

Severe accidents may be caused by any of the accident initiators discussed in the preceding sections in the case of extraordinary equipment failures or human errors, such as initiating events not considered in the design or multiple equipment failure beyond the single failure criterion. Severe accidents are typically identified through probabilistic safety assessments performed for each plant.

Examples of accident initiators include:

— Complete loss of power supply for an extended period of time;
— Complete loss of feedwater for an extended period of time;
— LOCA combined with failure of the ECCS;
— LOCA combined with failures of long term RHR or recirculation water systems.

External events which result in one or more of the preceding initiators are also considered in the probabilistic safety assessments. As indicated above, initiating events involving power excursions or overpower accidents which result in fuel melting or cladding failure are not normally considered, since the probability of such accidents has been shown to be relatively low compared with that of initiators involving loss of cooling. The notable exception may be consideration of the influence of a recriticality of the core associated with the liquefaction of core control materials in combination with the addition of unborated or insufficiently borated water.

Severe accidents can lead to a failure of the barriers for fission product release to the environment, including containment overpressure due to the combustion of flammable gases in the containment, direct heating of the containment atmosphere due to the high pressure ejection of molten corium from the vessel, or direct thermal attack of the containment wall by molten corium. Severe accidents can also include the release of radioactive materials into the environment through the failure of system components bypassing the containment boundaries.

The acceptance criteria for SAs are typically less prescriptive than the criteria developed for DBAs. In general terms, the probability and/or the consequences of SAs must be shown to be extremely small. In some cases, SA policy statements define the overall goals for core melt and the risk to the population in probabilistic terms. Directly measurable criteria, similar to the peak cladding temperature limits defined for LOCAs, are basically not included. In other cases, the goals are defined in more absolute terms, although probabilistic assessments are still considered. As an example, acceptance criteria Nos (20)–(22) (Section 3) can be used for the analysis.

18.2. SPECIFIC SUGGESTIONS FOR ANALYSIS

The influence of plant initial conditions is less significant for SA conditions since these accidents generally occur over a relatively long period of time compared with DBAs. It is typically assumed that the core operates for a long time at full power and under equilibrium conditions to maximize the decay heat and fission product inventory. In some cases, for example in the assessment of the response of a passive reactor design to SA conditions, a conservative estimate for decay power is required. However, owing to the non-linear nature of the response of the plant, assigning a maximum core decay heat may not result in the most conservative values for the consequences of an accident.

The design of the plant and, in particular, of the reactor core is much more important for SA conditions. For example, chemical interactions between different materials in the core can have a significant influence on the progression of core damage and on the release of hydrogen and radionuclides into the containment. Chemical interactions between fuel and cladding, between grid spacers and cladding, and in CR/blade structures can change the liquefaction and relocation of core materials over a wide range of temperatures and conditions. The rate of core heating, which is influenced by the decay heat in the core and the thermohydraulics of the system, is also much more important than in other accident sequences and can have a significant impact on the progression of damage, the chemical form of the radionuclides released into the containment, and the amount of hydrogen produced. As in the preceding example of decay heat, it is not possible to establish conservative design or physical parameters, due to the non-linearity of the plant response.

Best estimate approaches are used almost exclusively to describe the overall response of the plant during SA conditions. In some cases, conservative assumptions are used for some portion of the analysis to evaluate a specific response of the plant. For example, in evaluations of direct containment heating of US reactor designs, conservative assumptions were used in combination with detailed best estimate calculations to establish that direct containment heating was not a significant contributor to the risk of containment failure under loss of plant electric power. However, the same assumptions resulted in non-conservative estimates of the hydrogen release to the containment and thus could not be used to evaluate the risk of containment failure due to hydrogen combustion.

The analysis of SAs requires consideration of a much wider range of physical phenomena than that of DBAs. In addition to the assessment of the thermohydraulic response of the system required for DBAs, it is necessary to analyse phenomena including the heating and melting of the core, steam explosions, melt–concrete interactions, hydrogen and other non-condensable gas combustion, and fission product behaviour.

The analysis of SAs typically uses a multi-tiered approach with multiple codes, including the detailed system and containment analysis codes, more simplified risk assessment codes and other ‘separate effects’ codes. In most cases, detailed multidimensional system calculations are required to describe the response of the reactor coolant system, including multi-dimensional calculations to account for natural circulation flow patterns in the vessel and reactor coolant system piping, the initial failure and relocation of fuel rod and CR/blade materials, the growth of molten pools in the core and lower plenum regions, and the heat-up and failure of the reactor vessel.

Owing to the complexity of SAs, special guidance documents are being prepared for their analysis.

19. SUGGESTIONS ON REPORTING OF RESULTS

Detailed suggestions for evaluation and presentation of results have been presented in Ref. [3]. Only specific suggestions concerning selection of the parameters to be presented are included in this report. The presentation of results of an analysis for any individual event should include a set of key parameters, which are plotted as a function of time. Selection of the set needs to reflect the nature and the complexity of the process analysed. The basic requirements for selection of the set include:

— Good understanding and interpretation of the course of the accident;
— Possibility for checking of each acceptance criterion;
— Coverage of the key phenomena expected to occur during the course of the accident;
— Coverage of the process until stabilization of the safe condition.

The final selection of plots has to be made taking the applicability of the above suggestions into account.

For scenarios with a single phase coolant in the primary circuit, typical sets of parameters to be plotted are:

— Core thermal power;
— Total core reactivity, as well as partial contributions from Doppler reactivity, moderator temperature and voids;
— Pressurizer pressure;
— SG secondary side pressures;
— Core entrance and exit coolant temperatures;
— Pressurizer water level;
— Boron concentration in the core;
— Core mass flow rate;
— Mass flow rate in individual loops;
— Heat flux to the coolant in the core;
— Heat transfer rate to the secondary side;
— Primary to secondary leak mass flow rate;
— ECCS injection mass flow rate;
— Make-up system injection mass flow rate;

— Boron injection system mass flow rate;
— Break flow and enthalpy; if applicable (LOCAs, breaks of feedwater or steam lines);
— Main and auxiliary/emergency feedwater injection mass flow rate;
— Steam mass flow rate to turbine, through turbine bypass, and through secondary side relief and safety valves;
— SG mixture and liquid levels;
— MCP speed;
— Maximum fuel temperature on average and in the hot fuel rod;
— Maximum cladding temperature;
— Minimum DNB margin;
— Peaking factor above which rods experience DNB, if it occurs;
— Maximum radially averaged fuel pellet enthalpy;
— Total primary circuit coolant inventory;
— Mass flow rate through pressurizer safety and relief valves;
— Pressure in the pressurizer relief tank;
— Coolant outflow and enthalpy from the pressurizer relief tank (in the case of loss of tank integrity).

For scenarios with a two phase coolant in the primary circuit, besides all the plots already indicated above, plots of the following parameters are to be added:

— Pressure in the reactor upper plenum, near the break, in the SG and in the main steam header;
— Coolant temperature at the outlet from all the loops to the reactor, in the downcomer, in the lower plenum, in the upper plenum, and the saturation temperature corresponding to the pressurizer pressure;
— Total liquid volume in the primary circuit;
— Mixture and collapsed levels in the pressurizer, downcomer, lower plenum, core, upper plenum, hot and cold leg loop seals in individual loops and SGs;
— Total mass of coolant in the primary and secondary circuits, and total mass of coolant loss from the primary circuit;
— Maximum thickness of oxide layer on the cladding;
— Total amount of hydrogen produced;
— Number of fuel rods damaged;
— Fuel rod deformations;
— Cladding temperature at different axial positions of hot fuel rods during reflooding;
— Internal fuel rod gas pressure.

For analysis of the containment pressurization, the following parameters should be typically reported in addition:

— Mass flow rate and enthalpy of the coolant outflow;
— Maximum containment pressure and temperature;
— Differential pressures between compartments;
— Post-accident temperatures in various compartments;
— Mixture flows through selected openings;
— Thermal power released from various energy sources into the containment;
— Containment temperature and composition of the atmosphere (air and hydrogen content) in various compartments;
— Leakages/releases from the containment into the environment;
— Sump water temperature;
— Thermal power removed by the spray system, heat absorption and other heat sinks.

If radioactivity transport is involved in the analysis, the list of reported parameters needs to further include:

— Concentrations of radionuclides in the fuel, the coolant and their deposits on the RCS structures;
— Concentrations of radionuclides in the containment atmosphere and in deposits on the containment structures;
— Quantity, composition and time period for the discharge of radioactive materials to the environment.

For specialized PTS analysis, besides all the parameters relevant for the overall thermohydraulic behaviour of the systems, the following additional parameters need to be reported:

— Coolant temperature in the reactor inlet nozzle as a function of time;
— Coolant temperature in the downcomer as a function of position and time;
— Coolant to wall heat transfer coefficient field;
— Temperature distribution in the RPV wall in analysed areas as a function of time;
— Stress field through the RPV wall in analysed areas as a function of time;
— Postulated defect characteristics;
— Stress intensity factors for postulated defects with respect to crack front location;

— Crack depth and shape as functions of temperature and time.

Finally, for analysis of SAs, additional parameters reflecting degradation of the core and potentially also of the containment need to be reported:

— RCS temperatures, flow rates and pressures; including natural circulation flow patterns in the vessel and system piping;
— Core, vessel and RCS structure temperatures;
— Changes in core geometry, including the melting and relocation of core structures, the formation of debris beds associated with collapse of the core or core reflood, and the formation, growth and collapse of molten core materials;
— Oxidation of core structures, including the production of hydrogen;
— Concentration of radionuclides in the fuel, coolant and deposits on RCS structures;
— Containment atmosphere temperatures, flow rates and pressures;
— Concentrations of hydrogen and other non-condensable gases in the containment atmosphere;
— Concentrations of radionuclides in the containment atmosphere and deposits on the containment surfaces;
— Changes in molten core/containment floor geometry due to core–concrete interactions.

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CONTRIBUTORS TO DRAFTING AND REVIEW

Allison, C.	Innovative Systems Software, United States of America
Balabanov, E.	ENPRO CONSULT Ltd, Bulgaria
D'Auria, F.	University of Pisa, Italy
Jankowski, M.	International Atomic Energy Agency
Mišák, J.	International Atomic Energy Agency
Munhoz-Camargo, C.	International Atomic Energy Agency
Salvatores, S.	Electricité de France, France
Snell, V.	Atomic Energy of Canada Limited, Canada

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