IAEA-EBP-WWER-08 (Rev. 1)

GUIDELINES ON PRESSURIZED THERMAL SHOCK ANALYSIS FOR WWER NUCLEAR POWER PLANTS

Revision 1

A PUBLICATION OF THE EXTRABUDGETARY PROGRAMME ON THE SAFETY OF WWER AND RBMK NUCLEAR POWER PLANTS

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FOREWORD

The IAEA initiated in 1990 a programme to assist the countries of central and eastern Europe and the former Soviet Union in evaluating the safety of their first generation WWER-440/230 nuclear power plants. The main objectives of the Programme were: to identify major design and operational safety issues; to establish international consensus on priorities for safety improvements; and to provide assistance in the review of the completeness and adequacy of safety improvement programmes.

The scope of the Programme was extended in 1992 to include RBMK, WWER-440/213 and WWER-1000 plants in operation and under construction. The Programme is complemented by national and regional technical cooperation projects.

The Programme is pursued by means of plant specific safety review missions to assess the adequacy of design and operational practices; Assessment of Safety Significant Events Team (ASSET) reviews of operational performance; reviews of plant design, including seismic safety studies; and topical meetings on generic safety issues. Other components are: follow-up safety missions to nuclear plants to check the status of implementation of IAEA recommendations; assessments of safety improvements implemented or proposed; peer reviews of safety studies, and training workshops. The IAEA is also maintaining a database on the technical safety issues identified for each plant and the status of implementation of safety improvements. An additional important element is the provision of assistance by the IAEA to strengthen regulatory authorities.

The Programme implementation depends on voluntary extrabudgetary contributions from IAEA Member States and on financial support from the IAEA Regular Budget and the Technical Cooperation Fund.

For the extrabudgetary part, a Steering Committee provides co-ordination and guidance to the IAEA on technical matters and serves as a forum for exchange of information with the European Commission and with other international and financial organizations. The general scope and results of the Programme are reviewed at relevant Technical Cooperation and Advisory Group meetings.

The Programme, which takes into account the results of other relevant national, bilateral and multilateral activities, provides a forum to establish international consensus on the technical basis for upgrading the safety of WWER and RBMK nuclear power plants.

The IAEA further provides technical advice in the co-ordination structure established by the Group of 24 OECD countries through the European Commission to provide technical assistance on nuclear safety matters to the countries of central and eastern Europe and the former Soviet Union.

Results, recommendations and conclusions resulting from the IAEA Programme are intended only to assist national decision makers who have the sole responsibilities for the regulation and safe operation of their nuclear power plants. Moreover, they do not replace a comprehensive safety assessment which needs to be performed in the frame of the national licensing process.

CONTENTS

SUMMARY

The need for detailed guidance in the performance and review of pressurized thermal shock analysis for WWER nuclear power plants has been identified as a high priority within the IAEA Extrabudgetary Programme on the Safety of WWER and RBMK NPPs.

The integrity of the reactor pressure vessel has to be maintained throughout the plant life since there are no feasible provisions which would mitigate a catastrophic vessel failure. Adequate approach to the reactor pressure vessel integrity assessment provides a basis for safe operation and for timely implementation of preventive and corrective measures if necessary.

These guidelines deal with the reactor pressure vessel (RPV) pressurized thermal shock (PTS) analysis required to justify RPV integrity for nuclear power plants with WWER type reactors. The guidelines provide advice on individual elements of the PTS analysis, such as acceptance criteria, selection and categorization of initiating events to be considered, thermal hydraulic analysis, structural analysis including fracture mechanics assessment, evaluation of material properties and neutron field calculations.

The objective of the guidelines is to establish a set of recommendations to guide the RPV PTS analysis. The purpose of the PTS analysis is to provide a reasonably bounding plant specific demonstration of the RPV integrity by using realistic modeling methods for the individual elements of the analysis with conservative assumptions, initial and boundary conditions and appropriate safety factors in the assessment of the results. Deterministic approach is used in the guidelines. The demonstration of the RPV integrity is performed in terms of the safety margin between maximum allowable value of critical brittle fracture temperature and its actual RPV material specific value. The recommendations for computer codes used in the PTS analysis are provided as well as requirements on quality assurance for the whole assessment.

The guidelines are complemented with a series of Appendices, containing lists of initiating events used for PTS analysis of WWER plants and other plants as well as examples of some national practices. Example of PTS evaluation performed for Loviisa plant is also provided.

Relevant IAEA documents have been considered in the elaboration of these guidelines. These guidelines complement other guidelines prepared for WWER plants within the IAEA Extrabudgetary Programme.

This report was prepared in the frame of the IAEA Technical Cooperation Project RER/9/035 and of the Extrabudgetary Programme on the safety of WWER and RBMK NPPs.

After publication in 1997, the guidelines were applied in the IAEA PTS benchmark exercise as well as extensively used in Member States operating WWER NPPs. The experience and results accumulated in the period 1997–2001 led to a proposal to revise the guidelines. The revision of the report was performed in the frame of the IAEA Programme on Safety Analysis and Accident Management in the period November 2001–March 2002.

The document “Unified Procedure for Lifetime Assessment of Components and Piping in WWER Nuclear Power Plants – VERLIFE ” was prepared within the frame of the VERLIFE project of the EU 5th framework programme in the period 2001 – 2003. The subject of the VERLIFE procedure is much broader than that of these Guidelines. Several sections and appendices of VERLIFE document deal with integrity of RPV and PTS assessment. The preparation of the Revision 1 of the Guidelines was scheduled to facilitate harmonisation of both documents.

1. INTRODUCTION

The integrity of the reactor pressure vessel has to be maintained throughout the plant life since there are no feasible provisions which would mitigate a catastrophic vessel failure. Adequate approach to the reactor pressure vessel integrity assessment provides a basis for safe operation and for timely implementation of preventive and corrective measures if necessary.

The reactor pressure vessel integrity is ensured by a margin between its load bearing capacity, given by vessel design and material properties and the acting loads, which could occur during the plant operation. The material properties are subject to degradation during operation by neutron irradiation, fatigue, thermal ageing and other mechanisms, which reduce the resistance of the vessel against brittle fracture. The loads to be considered in the vessel integrity assessment are mainly related to plant states leading to a pressurized thermal shock (PTS) events, characterized by rapid cooldown in the primary coolant system usually with high level of primary system pressure. Such events depend strongly on the actual plant status, configuration, systems operation and operator actions.

The need for detailed guidance for the PTS analysis for WWER plants has been identified through the IAEA activities. In addition to the design deficiencies identified, such as high degree of embrittlement, lack of baseline information, surveillance programme weaknesses, and incompleteness of the analyses carried out, the impact of operator actions on RPV integrity was neither systematically estimated nor reflected in the operational procedures. High priority to isolation of breaks was originally given without consideration to potential PTS aspects. Procedural guidance available in control room to cope with excessive heat removal from the secondary side was weak. Last years the situation in many of WWER units was improved significantly. New symptom based operating procedures were developed and implemented in the plants. In these procedures, the issue of RPV integrity during accidents is treated in a systematic way.

The RPV PTS analysis is complementary to other kind of accident analyses, such as analysis of core cooling, analysis of system pressurization or containment integrity analysis. However, the assumptions, initial and boundary conditions used in the RPV PTS analysis could differ significantly from those used in the most of other analyses (especially from those used in core cooling analysis). The other analyses are dealt with in Refs [1, 2].

An important support to RPV integrity is provided by the non-destructive testing (NDT) for in-service inspection (ISI). Guidelines for qualification of NDT are dealt with in another IAEA report [3].

1.1. OBJECTIVES

These guidelines deal with the reactor pressure vessel (RPV) pressurized thermal shock (PTS) analysis required to justify RPV integrity for nuclear power plants with WWER type reactors. The guidelines provide advice on the individual elements of the PTS analysis, such as acceptance criteria, analysis methods, computer codes, and assumptions to be used as well as on quality assurance.

It should be pointed out, that PTS analysis is a multidisciplinary effort and involves selection and categorization of initiating events to be considered, thermal hydraulic analysis, structural analysis including fracture mechanics assessment, evaluation of material properties and neutron field calculations.

The objective of the guidelines is to establish a set of recommendations for RPV PTS analysis, considering related recommendations of the IAEA Safety Standards Series. The recommendations of this guidelines are based on state of the art practices, operational experience and results of research and development effort in Member States. The application of this guidelines is subject to final judgement of national authorities.

The PTS analysis outlined in the guidelines covers transients and accidents to be considered in the reactor design according to Ref. [4]. The purpose of the PTS analysis is to provide a reasonably bounding plant specific demonstration of the RPV integrity by using realistic modelling methods for the individual elements of the analysis but with conservative assumptions, initial and boundary conditions and appropriate safety factors in the assessment of the results. Deterministic approach is used in the guidelines by analysis of limiting transients from each group of events. Limiting in this sense is understood as limiting from the point of view of RPV integrity.

The demonstration of the RPV integrity can be performed in terms of the safety margin between maximum allowable value of critical brittle fracture temperature and its actual RPV material specific value.

In some cases, where the scope of the considered transients and accidents cannot be directly reduced, the material data involve large uncertainties or the RPV material embrittlement tends to be high, a probabilistic approach [5] would provide important complementary information. It should be noted, however, that the probabilistic approach requires a substantial amount of data to be plausible.

1.2. BACKGROUND

According to Ref. [6], it is required that the primary circuit pressure boundary including reactor pressure vessel shall be designed to withstand the static and dynamic loads anticipated during all operational states and accident conditions. Further, the design and conditions of the primary pressure boundary should be such as to avoid brittle behavior. Reference [7] suggests to set a probability target for pressure vessel failure in the range 10^{-6} to 10^{-7} per reactor year and the Russian General Regulations for Nuclear Power Plant Safety, OPB-88 [8], requires the demonstration of the vessel failure probability being less than 10^{-7} per reactor year.

Reference [9] further specifies that the regulatory body shall review and assess the design of an NPP to confirm that it can meet acceptable safety requirements.

More detailed national standards related to RPV integrity and in particular to PTS have been developed in several countries. These standards, especially those from the Russian Federation [10] and the USA [5, 11], have been considered for the elaboration of the present guidelines, when appropriate.

It is important to note that in particular in cases of decisions on plant modifications or their licensing and of periodical reassessment, it is necessary to consider the PTS related aspects and if needed, perform the PTS analysis.

The development of these guidelines was initiated by a small group of experts who met on September 5–7, 1995. A preliminary draft of an extended table of contents was developed and the experts from UJV (Czech Republic), Fortum (former IVO, Finland), EDF (France), GRS (Germany), Paks NPP (Hungary), OKB Gidropress (Russia), and VUJE (Slovak Republic) prepared the first draft of the document accordingly.

Consultants meeting was convened in March 1996 to review the first draft which was then circulated for comments to the Member States concerned. A third consultants meeting, November 6–8, 1996 incorporated the comments, and finalized the guidelines. Relevant IAEA documents have been considered in the elaboration of the report.

The original guidelines were prepared in the frame of the IAEA Technical Cooperation Project RER/9/035 and of the Extrabudgetary Programme on the safety of WWER and RBMK NPPs.

After publication in 1997, the guidelines were applied in the IAEA PTS benchmark exercise ([29], [30]) as well as extensively used in Member States operating WWER NPPs. The experience and results accumulated in the period 1997 till 2001 led to a proposal to revise the guidelines. The revision of the report was performed in the frame of the IAEA Programme on Safety Analysis and Accident Management in the period November 2001 to March 2002. The revision of the guidelines was drafted by a group of experts during a meeting held 27–29 November, 2001. The draft was then circulated for comments to experts who participated in drafting and review of the original report as well as to participants of the PTS benchmark exercise. The document "Unified Procedure for Lifetime Assessment of Components and Piping in WWER Nuclear Power Plants – VERLIFE " was prepared within the frame of the VERLIFE project of the EU 5th framework programme in the period 2001 – 2003. The subject of the VERLIFE procedure is much broader than that of these Guidelines. Several sections and appendices of VERLIFE document deal with integrity of RPV and PTS assessment. The preparation of the Revision 1 of the Guidelines was scheduled to facilitate harmonisation of both documents.

The most important changes introduced are related to:

- emphasis of the relevance of nonuniform temperature and heat transfer coefficient fields (section 4.2);
- consideration of important influence factors on structural analyses like zero-stress-temperature thermal expansion coefficients (section 4.3);
- inclusion of temperature dependent properties of WWER RPV materials (section 4.3);
- emphasis of residual stresses (section 6.1);
- reduction of safety factors especially concerning postulated defects smaller than 1/4 of wall thickness (section 7);
- irradiation embrittlement prediction formulae for RPV materials (section 8);
- recent development in the field of EOPs was taken into account;
- postulated defects size and shape;
- use of safety factors;
- introduction of the "Master Curve" approach.

1.3. PTS ANALYSIS PROCEDURE

A PTS analysis is a complex task which puts significant requirements on the experts performing it. These requirements include knowledge of dominant physical phenomena and associated computer codes, knowledge of the plant analysed and knowledge of the relevant codes and standards relevant to the RPV integrity assessment.

The PTS analysis is performed in several consequential steps. The flowchart in Fig. 1 illustrates these steps and the necessary input data. The procedure starts with definition of the PTS sequences. Thermal hydraulic analyses of these sequences provide necessary pressure, temperature and heat transfer data to be used in temperature and stress field calculations for the RPV. The fracture mechanics analysis and material data provide input for integrity assessment.

1.4. STRUCTURE

Section 2 of these guidelines discusses selection of the overcooling transients and accidents to be analyzed. Section 3 explains the acceptance criteria to be used for RPV PTS analysis.

Section 4 outlines the assumptions to be made in order to ensure that the analysis is reasonably conservative, i.e. will lead to a result which will sufficiently and reliably demonstrate the integrity of the RPV to justify safety of operation. Adequate selection of assumptions on functioning of systems and operator actions are discussed.

Section 5 provides information on the objectives and requirements of the thermal hydraulic analysis. Section 6 gives details on structural analysis including fracture mechanics assessment and postulation of defects.

In Section 7 the assessment of results is discussed, including the application of safety factors. Section 8 includes the assumptions concerning material properties used for fracture mechanics assessment. Section 9 provides comments related to corrective actions addressing both material properties and load reduction.

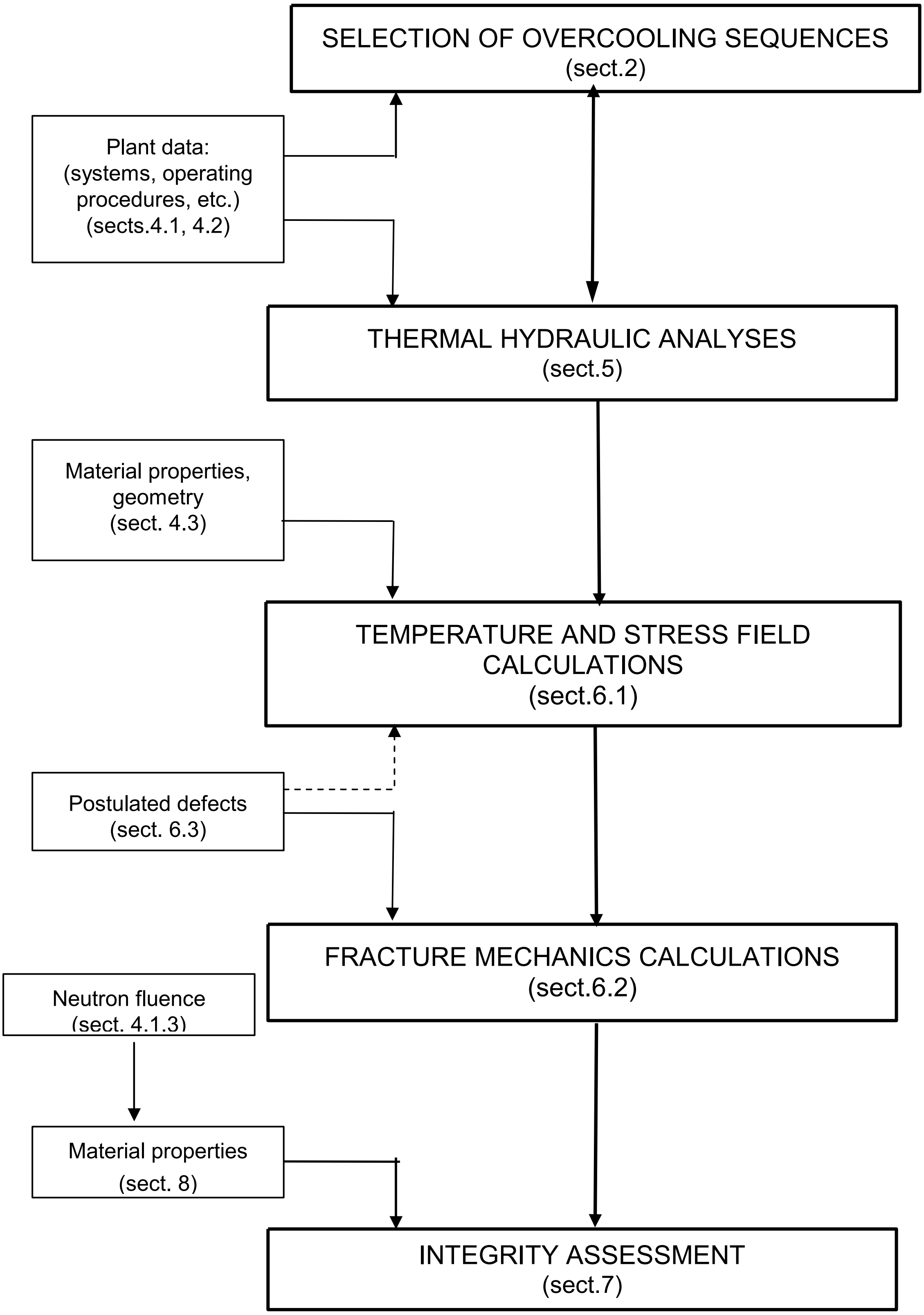

FIG. 1. Integrity assessment related to the PTS.

Recommendations for the computer codes used in the PTS analysis are discussed in Section 10 with special attention devoted to the code validation. Section 11 summarizes both general and specific recommendations on quality assurance.

The guidelines are complemented with a series of Appendices, containing lists of initiating events used for PTS analysis of WWER plants and other types of plants as well as examples of some national practices. Example of PTS evaluation performed for Loviisa plant is also provided.

2. SEQUENCES TO BE CONSIDERED

2.1. GENERAL CONSIDERATIONS

The selection of PTS transients should be performed in a comprehensive way taking into account various accident sequences including the impact of equipment malfunctions and/or operator actions. The main goal is to select initiating events which by themselves are PTS events or along with other consequences can lead to a PTS event. The sequences to be considered in the PTS analysis are unit specific and all relevant and meaningful plant features should be taken into account.

Independent events beyond the application of the single failure criteria [12] need not be considered to occur simultaneously. Where individual initiating events could credibly lead to consequential failures, they should be considered in the analysis (for instance, a main steam line break with a failure of the main steam isolation valves on the neighboring main steam line because of the lack of fixed points or separation walls in the steam line layout). The impact of the application of the single failure criteria [12] in PTS analysis is not straightforward and should be carefully evaluated. Attention should be paid mainly to the differences as compared to the accident analysis performed with respect to the core cooling.

Selection of the transients for deterministic analysis can be based on engineering judgement using the design basis accident analysis approach, see also [4], combined with the operational experience accumulated at WWER plants.

The most comprehensive and effective approach to the selection of transients is the probabilistic event tree methodology. This methodology can help in identifying those specific transient scenarios that contribute most significantly to the total PTS risk. In this case a broad risk assessment is performed that utilizes various lower resolution (simplified) techniques to assess the PTS risk of several cooldown transients.

The probabilistic PTS analysis is considered complementary to the deterministic analysis of the limiting scenarios.

When performing the deterministic selection of transients, it is important to consider several factors determining thermal and mechanical loading mechanisms in the downcomer during the overcooling events. These factors are:

- the final temperature in the downcomer;
- the temperature decrease rate;
- nonuniform cooling of the RPV, characterized by cold plumes and their interaction and by the nonuniformity of the coolant-to-wall heat transfer coefficient in the downcomer;
- the level of primary pressure;
- width of cold plume,
- initial temperature in downcomer,
- stratification or stagnation of flow in cold leg.

The possibility of outside cooling of the RPV needs also to be considered at WWER plants. In some situations the reactor cavity can be flooded from the ECCS or spray system. The cold water contained in the biological shield tank can be the other source for external vessel cooling for WWER-440/230 units. In the case of some WWER-440/213 equipped with pressure suppression system, cold water from bubble condenser trays is in some accident scenarios (e.g. medium and large break LOCAs) spilled out on the floor of SG boxes and hence, after reaching overflow level, the reactor cavity can be flooded. Similarly, in the case of Loviisa NPP equipped with ice condenser containment, the reactor cavity is flooded after melting ice condenser in most of the scenarios with high energy coolant release into the containment. Intentional cavity flooding was adopted as severe accident management measure for in-vessel retention of corium at Loviisa NPP, where necessary plant modification were performed. Implementation of similar modifications is also under consideration for standard WWER-440/V213 units.

Special cases of external flooding should be also considered (if applicable):

- direct water injection in case of near-RPV break,
- unintentional actuation of a cavity flooding system (system installed in some plants for severe accident mitigation , e.g. in-vessel corium retention).

These special cases can be more challenging in small-break LOCA or non LOCA accidents, because relatively small amount of cold water is sufficient to flood the RPV while primary pressure and temperature are high.

Based on the above loading mechanisms, the accident sequences to be considered in the PTS analyses can be selected.

2.2. INITIATING EVENTS GROUPS

The aim for setting a list of initiating events is to assure a complete analysis of the RPV response to postulated disturbances which may threaten its integrity. The analysis should determine the consequences and evaluate the capability built into the plant to withstand such loadings.

The sequences should be considered for various plant operating conditions: full power, hot zero power, heat up, cool down and cold shutdown [13].

The complexity of many interacting systems and operator actions makes it sometimes very difficult to choose the limiting transients. At least the following groups of initiating events should be taken into account.

Compilation of the list of initiating events corresponding to each of the following groups is usually based on engineering judgment while assisted with probabilistic consideration, taking into account the design features and implemented modification of the given nuclear plant.

Loss of coolant accidents

Different sizes of both cold and hot leg loss of coolant accidents (LOCA) which are characterized by rapid cooldown should be considered. Attention should be paid on the scenarios leading to flow stagnation which causes faster cooldown rate and cold plumes in the downcomer. Attention should be given to breaks sizes corresponding to existing pipes connected to primary system. Cold repressurization of the reactor vessel is usually prohibited in principle, but the possibility of isolating the leak and the subsequent repressurization have to be considered.

Stuck open pressurizer safety or relief valve

After an overcooling transient caused by a stuck open pressurizer safety or relief valve, possible reclosure can cause a severe repressurization. Even without the valve reclosing, the system pressure can remain high after having reached the final temperature. The low decay power may further lead to the main loop flow stagnation. In addition, the "feed&bleed" method of mitigation for loss of feedwater should be assessed.

Primary to secondary leakage accidents

Different sizes for both single and multiple steam generator tube ruptures up to the full steam generator collector cover opening should be considered. The risk of repressurization should be taken into account, if the relevant EOP contains a requirement to isolate the affected SG by closing of MGVs.

Large secondary leaks

Transients with secondary side depressurization caused either by the loss of integrity of the secondary circuit or by the inadvertent opening of a steam dump valve can cause significant cooldown of the primary side. Consequently, start of high pressure injection due to low primary pressure (and/or low pressurizer level or directly due to low secondary circuit parameters), which leads to re-pressurization, can be expected. The degree of secondary side depressurization is strongly dependent on the plant configurations (mainly presence of fast acting main steam isolation valves and the criteria for steam line isolation). Possible sources of secondary side depressurization are as follows:

- steam line break;
- main steam header break;
- spurious opening and sticking open of the turbine bypass valve (BRU-K), atmospheric dump valve(BRU-A) and steam generator safety valve(s);
- feedwater line break.

After the leaking steam generator(s) is (are) empty, the temperature increase in the primary circuit can lead to an increase in primary pressure (this pressurization is very fast, especially in the case when the primary circuit is completely filled by fluid due to previous ECCS injection). During this process, the opening of the pressurizer relief or safety valve can occur and the valve can stick open under fluid flow conditions. The resulting PTS effects should also be considered.

Inadvertent actuation of high pressure injection or make-up systems

This kind of accident can result in a rapid pressure increase in primary system. Cold, hot, and cooldown initial conditions should be considered.

Accidents resulting in cooling of the RPV from outside

Break of the biological shield tank or some other possible sources of reactor cavity flooding (ECCS or containment spray system, loss of coolant from primary or secondary circuit, intentional cavity flooding) should be considered in this group of accidents.

2.3. INITIATING EVENTS CATEGORIZATION

The complexity of many interacting systems and operator actions makes it very difficult to determine which are the limiting PTS sequences and what is their significance. An integrated probabilistic PTS study should be used to reveal the probability of individual events. Potential risk from all credible overcooling events might be higher than from postulated limiting events, even though each event individually is less severe than the limiting one. Therefore for events with high probability of occurrence, more stringent requirements need to be applied to assure RPV integrity. Based on the frequency of occurrence the initiating events may be categorized into two broad groups:

Anticipated transients

which are defined as relatively frequent deviations (frequency of occurrence higher than 10^{-2} per reactor year) from normal operating conditions which are caused by malfunction of a component or operator error. These transients should not have safety related consequences to RPV integrity, which would prevent the continued plant operation.

Postulated accidents

which are defined as rare deviations from normal operation which are not expected to occur (less than 10^{-2} per reactor year globally) but are considered in the original design or in the design of plant upgrading or are based on plant safety reassessment, see also Ref. [1]. For these events, immediate resumption of operation may not be possible.

3. ACCEPTANCE CRITERIA

The objective of the RPV PTS analysis is to demonstrate by a conservative deterministic analysis that there will be no initiation of a brittle fracture from the postulated defect during the plant design life for the whole set of anticipated transients and postulated accidents which has been selected according to principles described above in Section 2. The assessment is based on the static fracture toughness K_{IC}. The RPV material degradation during operation should be taken into account. Different safety factors are recommended to be used for anticipated transients and postulated accidents.

It should be noted that other complementary approaches could be used provided they are properly justified and validated, such as crack arrest approach for postulated accidents. In such cases, specific acceptance criteria may need to be defined.

The PTS should not constitute a significant contribution to the core damage frequency obtained in level 1 PSA [14]. If some PTS sequence classes are clearly dominating the PTS risk, efforts should be taken to smooth the risk profile.

4. ASSUMPTIONS FOR PTS ANALYSIS

4.1. PLANT DATA

4.1.1. Systems pertinent to PTS

The systems to be taken into consideration in PTS analysis are usually:

- reactor cooling system;
- main gate valves;
- reactor protection system;
- pressurizer and pressure control system;
- emergency core cooling system;
- chemical and volume control system;
- main steam system;
- feedwater system;
- support systems;
- systems pertinent to reactor cavity flooding;
- containment sump;
- containment (confinement);
- ECCs heat exchanger.

According to the selected transient sequences the design and operational characteristics of the systems to be considered in the PTS analysis should be determined.

4.1.2. Reactor pressure vessel

The data of the reactor vessel to be taken into consideration in PTS analysis are usually:

- material characteristics of base metal, weld metal, heat affected zones and cladding including chemical composition and mechanical properties;
- corrective measures implemented, such as annealing;
- geometrical characteristics describing the location of nozzles and of welds;
- characteristics of the manufacturing technology influencing the fracture mechanics characteristics of the vessel;
- weld geometry and other aspects (e.g. surface quality) which might affect the capability of NDT;
- characteristics of RPV thermal insulation;
- results of pre-service inspection.

4.1.3. Fluence

The past, current and predicted neutron fluence on the vessel should be obtained using validated computers codes and information gained from the neutron flux surveillance dosimetry (if available).

A fluence map should be determined taking into account such important characteristics as position of welds, inlet nozzles and core configuration, etc.

4.1.4. In-service inspections results

The quality, effectiveness and results of the NDT for ISI of the vessel should be considered on a plant specific basis in the case when smaller defects than ¼ of the RPV wall thickness are postulated (see sect. 6.3).

4.1.5. Plant operating experience

Overcooling transients that have occurred at a given plant and in WWER plants generally should be summarized, including lessons learned from these transients and from other transients. Actions taken to prevent recurrence or to minimize severity of overcooling transients should be indicated.

4.2. ASSUMPTIONS FOR THERMAL HYDRAULIC ANALYSIS

4.2.1. Assumptions of system operation

Concerning the parameters of the normal operation and control systems, the expected values based on the operational experience should be assumed as they usually tend to lead to more serious overcooling. Failure of components of these systems (when it is not a direct consequence of the initial event) should be considered only in cases that lead to more severe PTS loading.

The loss of the external power supply has to be taken into consideration as an additional failure if it will further aggravate the analysis results.

The availability of the emergency core cooling systems should be taken into consideration in such a way as to produce the most intensive overall cooling or the most unsymmetrical cooling. In some cases, the action of 1/3 of the safety injection systems is more conservative while in other cases it is the action of 3/3 of these systems.. It is assumed that the systems operate on maximum installed capacity (with corresponding head value taken according to maximum pump characteristics) and that they inject the lowest possible temperature cooling water to the primary circuit. Time variation of injected water temperature should also be conservatively evaluated (e.g. automatic switching from heated high to non heated low pressure tanks) along with considering a relevant single failure.

The stuck open safety valve should be considered as a consequential failure if the valve is not qualified for the discharged coolant (liquid or steam-water mixture) or if there is a demand for a large number of successive cycles.

The possible later reclosure of the opened and stuck open safety valve should be taken into account. The reclosure can lead to the repressurisation by the normal operating make-up or safety injection pumps or, in case of the water solid primary system (completely filled by water), through thermal expansion of coolant volume. The time of the safety valve reclosure should be selected conservatively from the PTS severity point of view.

In the case that operation of the secondary circuit steam and feedwater systems results in cooling and depressurization of the primary circuit, then those systems have to be taken into account.

4.2.2. Operator actions

Prior to the analysis, those operator’s activities that are to be carried out in the case of a given overcooling transient should be determined. The estimated time of the operator’s intervention is to be evaluated separately.

Two different groups of operator actions are considered that can have an important impact on PTS transients. The first group is where operator actions may turn the ongoing accident sequence into a PTS transient. Such adverse actions should be identified and removed from the operating procedures when possible. The second group includes actions that have a possible impact by mitigating the severity of an ongoing PTS transient.

When the operator takes action, it is acceptable to assume that the operator takes the correct action according to related procedures. In the cases where operator action has favourable impact on PTS transient (e.g. switch off high pressure injection pumps) it should be demonstrated that the operators have sufficient time and appropriate training for such action. It should be noted that the timing of the operator action is a very important aspect.

If according to the operational procedures the isolation of a potential break is prescribed for the operator, then this action has to be taken into account in the analysis, including addition of the estimated time necessary for the preparation of the intervention, (or this time can be conservatively reduced).

The decrease of safety injection flow rate by the operator may be taken into consideration only in case that the circumstances are unambiguously defined by the procedures.

The PTS relevant operator interventions might be:

- trip and restart of the reactor pumps;
- stopping of the ECCS injection;
- stopping or starting of make-up injection;
- isolation of a break (primary or secondary, including safety or relief valve reclosure);
- starting the secondary side cooldown;
- primary feed and bleed;
- primary system depressurization.

4.2.3. Plant operating conditions

The initial power of the reactor has to be set always to the most conservative value determined by the conditions of the overcooling transient selected according to Section 2 of these Guidelines. Full power, hot zero power , heat-up, cooldown and cold shutdown regimes should be analysed.

The value of the residual heat should be the lowest possible one, defined on the basis of the initial power level. For this reason the analyses are to be performed for the initial period of the fuel cycle (after longest planned outage). The estimated error of the residual heat calculation is to be taken into consideration with negative value. The determination of the residual heat might be based on actual operational measurement information except for cases of low power operation.

Other initial conditions such as reactor coolant flow rate, temperature as well as pressure and SG water level should be chosen conservatively.

4.2.4. Thermal hydraulic conditions

The cooling down processes should be calculated up to the stabilized primary circuit parameters. In many cases this means that the temperature of the primary circuit reaches the temperature of the water stored in the tanks of the Emergency Core Cooling System, the containment sump, or ECCS heat exchanger outlet temperature.

The cooling down rate has to be determined by taking into consideration various aspects. As far as the forced or intensive natural circulation is maintained, cooling down of the whole primary circuit can be assumed (except of pressurizer and reactor upper head). If the flow stagnation occurs in the primary system, the cooling process has to be investigated in a significantly smaller volume. In such cases it has to be taken into account that in the downcomer colder plumes will exist causing the temperature and heat transfer coefficient distribution to be nonuniform and asymmetric.

There are separate assumptions for flow stagnation cases, e.g.:

- in case of a compensated LOCA, when the reactor coolant pumps are tripped and the decay heat level is very low, the flow stagnation takes place when loop flow rate is about the same as the injection rate;
- for a non-compensated LOCA the onset of the flow stagnation appears when steam enters hot legs.

The nonuniform temperature and heat transfer coefficient field is created by cold plumes or cold sectors or cold stripes. Both cold plume and cold sector mean cold water input into the downcomer that is full of hot water. Cold plume presents nonuniformity in downcomer coolant temperature in both radial and azimuthal direction while cold sector shows nonuniformity only in azimuthal direction. In contrast, cold stripe is characterized by the input of cold water into the downcomer containing steam. The phenomena result from safety injection into the cold legs (high pressure injection or part of low pressure injection) or directly to the downcomer (accumulators or low pressure injection).

The nonuniformities in the temperature and velocity fields in the downcomer can affect flow rate in individual loops and time of beginning the flow stagnation in the individual loops. Therefore, usage of 2D modelling of the reactor downcomer already incorporated in the system thermal hydraulic analyses is recommended.

The effects of those temperature nonuniformities are to be taken into account in the analysis in case of loop flow stagnation or asymmetric secondary side cooling. Since even in case of a uniform temperature field significant flow rate differences might occur in the downcomer. The nonuniformity of the heat transfer coefficient field has to be investigated in addition to the temperature distribution. For various aspects of PTS scenarios (overall cool down, asymmetric plumes) different sets of conservative assumptions may be required.

The changes in the primary circuit pressure are to be determined in accordance with the initial event and the system parameters. The possible increase ofprimary pressure has to be evaluated in every case when the leak might be compensated or isolated, or when overcooling is caused by a secondary side anomaly.

4.3. ASSUMPTIONS FOR STRUCTURAL ANALYSIS

4.3.1. RPV materials

In calculations the thermal (λ, α, ρ, c_p,) and elastic (E, ν) properties of base metal, weld metal and cladding equal to their initial values should be considered. The appropriate interpretation of thermal expansion coefficient corresponding to zero-stress-temperature used in the calculation should be taken into account (the modified values of thermal expansion coefficients in the following tables are denoted by α_0).

Suggested values are provided in Tables I and II.

TABLE I. THERMAL AND ELASTIC PROPERTIES OF WWER–440 RPV MATERIALS

Material	***T***	***E***	α	α_0	ν	λ	c_p	ρ
	[°C]	[10^3 MPa]	[10^{-6} K^{-1}]	[10^{-6} K^{-1}]	[1]	[$Wm^{-1}K^{-1}$]	[$Jkg^{-1}K^{-1}$]	[kgm^{-3}]
Base material or weld	20	210	-	12.9	0.3	35.9	445	7821
	100	205	11.9	13.3	0.3	37.3	477	7799
	200	200	12.5	13.9	0.3	38.1	520	7771
	300	195	13.1	14.5	0.3	37.3	562	7740
Cladding	20	165	-	15.9	0.3	15.1	461	7900
	100	160	14.6	16.5	0.3	16.3	494	7868
	200	153	15.7	16.5	0.3	17.6	515	7830
	300	146	16.0	16.8	0.3	18.8	536	7790

*Stress free temperature (equal to the normal operational temperature) for recalculation of α_0 is 267°C.

TABLE II. THERMAL AND ELASTIC PROPERTIES OF WWER–1000 RPV MATERIALS

Material	***T***	***E***	α	α_0	ν	λ	c_p	ρ
	[°C]	[10^3 MPa]	[10^{-6} K^{-1}]	[10^{-6} K^{-1}]	[1]	[$Wm^{-1}K^{-1}$]	[$Jkg^{-1}K^{-1}$]	[kgm^{-3}]
Base material or weld	20	208		12.5	0.3	35.0	446.9	7830
	50				0.3	35.5	458.9	7822
	100	201	11.6	12.9	0.3	36.1	478.8	7809
	150				0.3	36.6	499.7	7795
	200	193	12.0	13.6	0.3	36.8	520.4	7780
	250				0.3	36.6	541.2	7765
	300	183	12.6	14.2	0.3	36.2	562.0	7750
	350	177.5			0.3	35.6	584.6	7733
Cladding	20	165		16.6	0.3	13.2	448.9	7900
	50				0.3	13.5	460.4	7889
	100	160	15.7	17.0	0.3	14.4	479.6	7870
	150				0.3	15.3	499.6	7851
	200	153	16.1	17.6	0.3	16.4	519.2	7830
	250				0.3	17.5	538.7	7809
	300	146	16.7	18.2	0.3	18.4	558.5	7788
	350	142			0.3	19.6	579.2	7766

* Stress free temperature (equal to the normal operational temperature) for recalculation of α_0 is 290°C.

Tensile properties (R_p, R_m, A_m) and fracture mechanics parameters for base metal, weld metal and cladding should be determined for analysed state of the RPV (fluence and time) taking into account all possible ageing mechanisms (shift in fracture toughness of ferritic steels is assumed equal to shift in critical brittle fracture temperature if there are no direct fracture toughness data). Individual properties or their conservative assessment for each assessed weld metal, base metal and cladding should be used in calculations.

4.3.2. Simplified vs. detailed fracture mechanics calculations

Detailed fracture mechanics calculations using finite elements method (FEM) should be used.

Simplified fracture mechanics calculations based on formulas such as given in [10] can be used in cases when linear elastic fracture mechanics can be applied.

5. THERMAL HYDRAULIC ANALYSIS

5.1. OBJECTIVES

There are two main objectives of thermal hydraulic analyses: to support the transient selection process and to produce necessary input data for structural analyses.

Thermal hydraulic calculations should give the following parameters as a function of time during the overcooling event, these parameters are used as input data for the subsequent temperature and stress fields calculations for the RPV wall:

- downcomer temperature field;
- local coolant-to-wall heat transfer coefficients in the downcomer;
- primary circuit pressure.

(Some thermal hydraulic codes give directly the inner surface temperatures of the RPV wall. In these cases, the coolant temperatures and local heat transfer coefficients are not necessary.)

5.2. THERMAL HYDRAULIC ANALYSIS TO SUPPORT TRANSIENT SELECTION

The overcooling transients are usually very complex. It is often not possible to define in advance conservative or limiting conditions for all system parameters. Engineering judgement might not be sufficient to decide whether an accident under consideration is, by itself, a PTS event or along with other consequences can lead to a PTS event that may potentially threaten RPV integrity. Therefore thermal hydraulic analyses are often necessary for choosing, from a number of accidents, those initiating events and scenarios that can be identified as limiting cases within the considered group of events.

5.3. SEQUENCE ANALYSIS PLAN

The overall progression of accidents is calculated with advanced thermal-hydraulic system codes which are used for the system thermal hydraulic analysis.

The output from this analysis is primarily the time variation of primary pressure, coolant temperature and velocity in cold legs, and further temperature and velocity of medium injected by emergency systems into the primary circuit.

In case of non-symmetric cooldown and/or flow stagnation in the primary circuit, when buoyancy induced forces dominate the fluid flow behavior in cold legs and the downcomer, the system code results are no longer reliable for temperature field calculation. In order to calculate the thermal stratification and mixing effect in these cases, separate methods, so called thermal mixing calculations, shall be applied.

In flow stagnation cases, the role of the thermal hydraulic system codes is, in addition to the inner pressure calculations, to estimate the initiation of the stagnation, and to give the initial and boundary conditions for thermal mixing calculations.

The calculation period of a transient should be long enough to reach stabilized conditions or at least to overreach the critical time from the point of view of RPV integrity or to reach the termination of the PTS regime by operator action.

5.4. REQUIREMENTS FOR THERMAL HYDRAULIC METHODS

The calculation methods, employed for the PTS analysis, should be validated for this purpose.

Thermal hydraulic analyses of overcooling sequences include many features that are different from those in accident analyses performed with respect to core cooling.

The utilized methods must be capable of modelling the normal operation systems, such as control systems, main feedwater system and make-up system because the proper operation of these systems usually leads to more severe overcooling.

Heat losses from the systems should be modelled in system thermal hydraulic analyses.

Direct ECCS injection into reactor vessel (esp. to the downcomer) should be modelled. Plus, if flow baffles exist in the neighborhood of HA lines connections to the reactor downcomer, then these baffles must be modelled in the system TH and mixing calculation, as they can deteriorate a course of LOCA from the PTS point of view.

The pressurizer modelling used in the code must be capable of calculating increasing pressure which can occur after the repressurization of the primary circuit.

Nonuniform cooldown should be analysed with appropriate fluid mixing codes that are capable of taking into account thermal stratification of high pressure injection water in the cold leg. They should be able to determine the azimuthal, axial, and in some cases also radial fluid temperature distribution in the downcomer and the azimuthal and axial distribution of the heat transfer coefficient to the RPV wall. Current quasi 3D methods applied in mixing codes based on engineering models or on the regional mixing model, allow sufficiently accurate calculation of the extent of the thermal stratification integrated into the overall system response.

A promising tool for the proper prediction of the turbulent mixing of fluids with different temperatures and velocities in a complex geometry is a three-dimensional general-purpose computational fluid dynamic (CFD) code applying a finite element or finite difference technique.

The exponential decay of the temperature in the mixing volume (mixing cup model) gives very simple presentation for transient cooldown. This approach can also be used when the mixing volume is properly defined and the heat transfer from the RPV wall is also added.

Conservative assumption for the system TH analyses of external flooding cases should be selected in such a way that the following general criteria are met:

- maximum coolant outflow to hermetic confinement/containment from primary system in case of LOCA, or from secondary system in case of feed water line break and from ECCS tanks (including trays of bubble condenser) to reach overflow of water to reactor cavity,
- minimum temperature of water in hermetic confinement/containment (esp. as for the water in reactor cavity),

- maximum temperature in reactor coolant system (esp. in reactor downcomer),
- maximum primary pressure.

6. STRUCTURAL ANALYSIS

The objective of the structural analysis is to evaluate stress intensity factors K_I for postulated cracks loaded by thermal hydraulic transients. For it temperature and stress field calculations as well as fracture mechanics calculations are necessary.

6.1. TEMPERATURE AND STRESS FIELD CALCULATIONS

Temperature and stress fields should be calculated for all chosen PTS sequences as well as cold overpressurization regimes.

The stresses due to internal pressure, temperature gradients, and residual stresses (for both cladding and welds) should be taken into account including the beneficial effect of the first hydrotest if deemed useful. Plasticity effects should be also considered. The value of zero-stress-temperature in the calculation should be chosen equal to normal operation coolant temperature in the downcomer. The residual stress in weld could be taken as

$$\sigma = 60 \cdot \cos\left(\frac{2\pi x}{s}\right) \quad [MPa],$$

where

σ residual stress in weld (the same value for both axial and circumferential stress component),

x radial coordinate in weld (with its origin in cladding/weld interface),

s weld thickness.

Stress fields should be calculated for different time steps, which must be chosen properly to catch all local maxima/minima as well as situation until stabilised conditions, or until the time important for the determination of maximum allowable critical brittle fracture temperature T_k^a (see section 7.3).

Stress fields are calculated on the basis of temperature dependent material properties for base and/or weld materials and cladding. Changes of material properties due to neutron irradiation should be taken into account in the stress field calculation.

The stresses may be evaluated by numerical methods as well as analytical ones. The use of FEM is generally recommended for stress calculations. Analytical methods can be used only in special justified cases.

6.2. FRACTURE MECHANICS ANALYSIS

Reactor pressure vessel integrity assessment with respect to brittle failure is carried out using fracture mechanics approach based on static fracture toughness. In most cases, linear elastic fracture mechanics using stress intensity factor K_I is fully acceptable (but this should be demonstrated). For more severe conditions characterized by significant plasticity and especially for vessels with cladding, elastic-plastic fracture mechanics based on the J-integral should be used.

To calculate the K_I values, simple engineering methods as well as numerical methods based on FEM with cracks included in the meshes may be used. The use of analytical methods is acceptable for vessels without cladding and in specified cases, characterized by small scale plasticity. For such analysis, a proper approximation of the non-linear stress distribution in the RPV wall is required.

In case of a postulated defect under or going through the cladding, the numerical methods based on FEM should be used to calculate K_I values. Alternatively, simplified analytical methods could be developed which take into account discontinuity in stresses at the boundary between cladding and base or weld metal respectively.

The use of FEM with postulated crack integrated in the FEM mesh is recommended.

The stress intensity factor K_I should be evaluated for the crack front point with the highest crack loading and subsequently compared with the material fracture toughness K_{IC}, see also Sections 7 and 8. Usually, it is sufficient to evaluate K_I for the deepest point of the crack front and for the point of intersection of the crack front with the free surface (for uncladded RPV), or for the point close to intersection of the crack front with the boundary between cladding and base or weld metal (for cladded RPV).

The analysis for the point of intersection of the crack front with the free surface could be also performed by other more sophisticated methods, which take into account the absence of the plane strain conditions in this point. The method used should be properly justified.

The stress intensity factor can be calculated from the value of J-integral according to one of the following formulae:
for plane stress condition (surface point of a crack)

$$K_I = \sqrt{J \cdot E}$$

for plane strain condition (other points)

$$K_I = \sqrt{\frac{J \cdot E}{1-\nu^2}} .$$

In the assessment of the results, adequate safety margins have to be assumed (see Section 7.1).

6.3. POSTULATION OF DEFECTS AND NDT REQUIREMENTS

For the purpose of the RPV PTS analysis, defects are postulated in the RPV with the objective to demonstrate by fracture mechanics analysis that the acceptance criteria are met for these postulated defects, see also Section 3 of this report.

The postulated defects are surface or subsurface cracks, located in the limiting areas of the vessel. In selection of the limiting areas of the vessel, consideration should be given to the stress level, to the temperature of material, to the material degradation and to the results of the non-destructive testing. The orientation of the postulated defect should be considered axial and circumferential depending on the direction of the maximal principal stress. The external cooling of the vessel should be also taken into consideration when selecting limiting areas. In this case, the crack located near the outer surface is to be considered.

The postulated defect should be defined in the following way:

- For uncladded vessels the postulated defect is a surface semi-elliptical crack with depth up to ¼ of the RPV wall thickness and with aspect ratio a/c of 0.3 and 0.7 (Fig. 2).

- For cladded vessels, cladding integrity of which is verified by redundant non-destructive testing and its mechanical properties are known, the postulated defects are underclad semi-elliptical or, if applicable, elliptical cracks with depth up to ¼ of the RPV wall thickness, and with aspect ratio a/c resp. 2a/c of 0.3 and 0.7 (Figs. 3 and 4).

- For cladded vessels, where limited or no information on cladding exists, the postulated defect is surface through cladding semi-elliptical crack with depth up to ¼ of the RPV wall thickness and with aspect ratio a/c of 0.3 and 0.7 (Fig. 5).

In the case when the whole crack front is being assessed, postulation of the crack with only the maximum value of crack depth is sufficient.

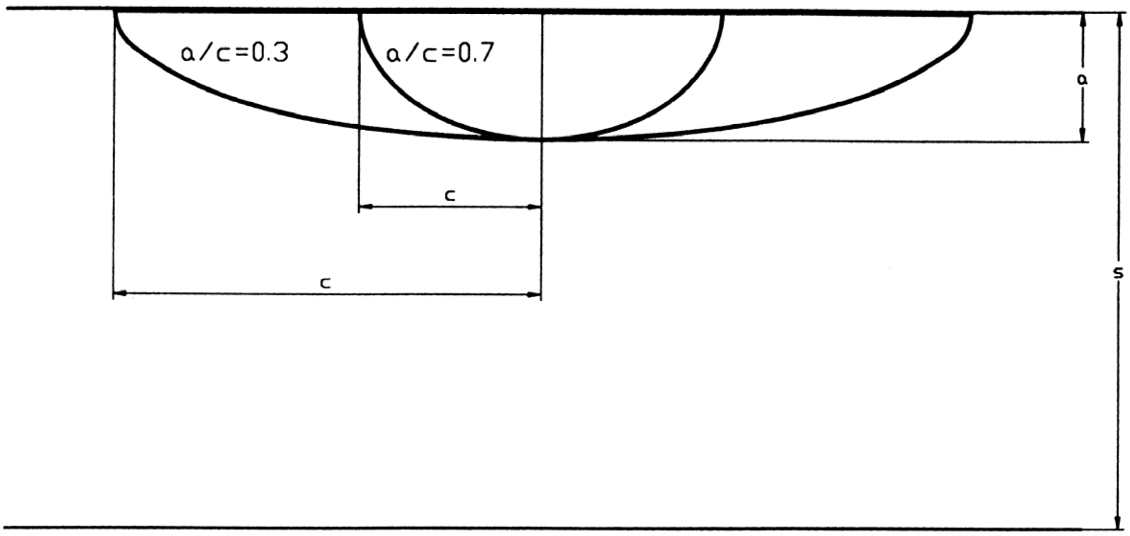

FIG. 2. Semielliptical surface crack, uncladded vessel.

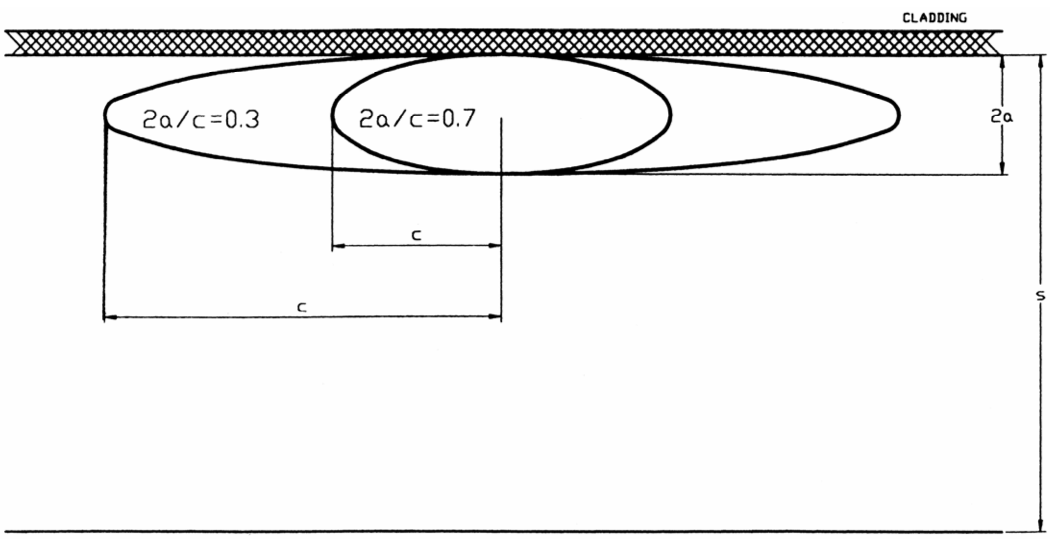

FIG. 3. Elliptical undercladding crack, cladded vessel.

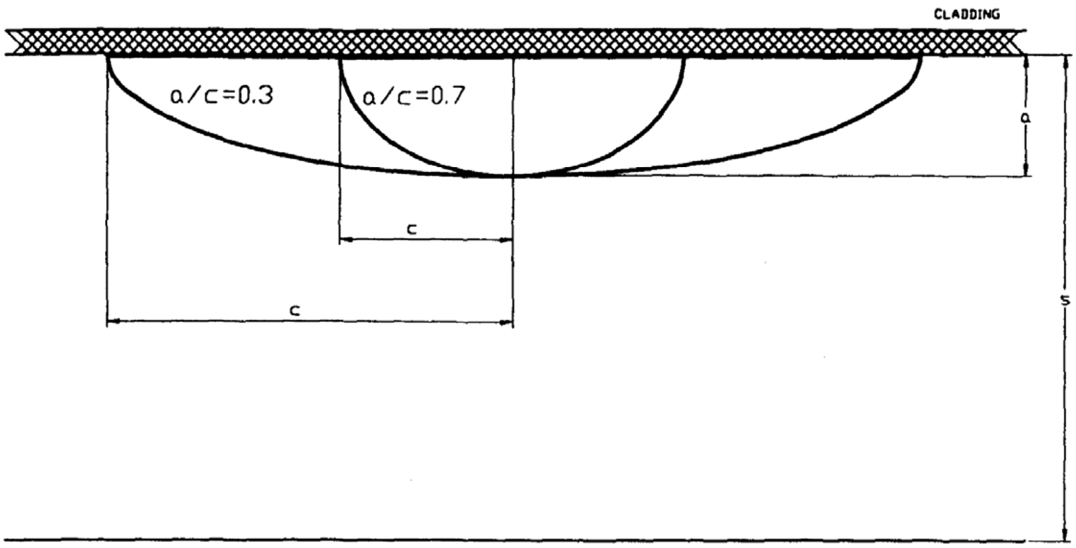

FIG. 4 Semielliptical undercladding crack, cladded vessel.

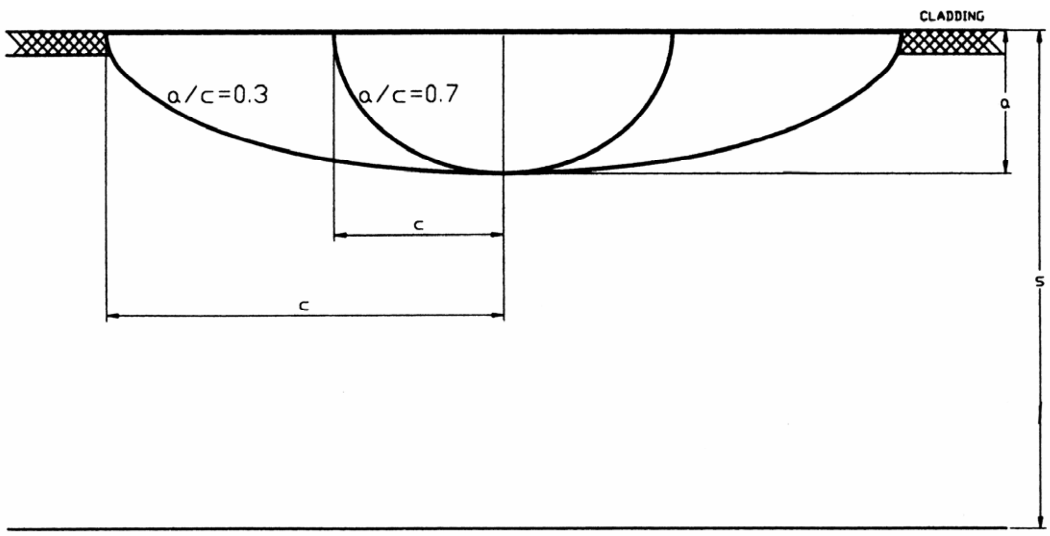

FIG. 5. Semielliptical surface crack, cladded vessel.

Defect sizes smaller than ¼ of the wall thickness could be used for the RPV integrity assessment under PTS loading if it is possible to demonstrate the required non-destructive testing reliability and if permitted by the national regulatory requirements. In such cases, the shape and the size of the postulated defect should be conservatively postulated with respect to the plant specific non-destructive testing qualification criteria. The national standards for NDT and related standards for schematization of detected defects should be taken into account. The size of the postulated defect could be selected with respect to the size of a realistic manufacturing defect probable to exist in the vessel. Determination of postulated defects' sizes should take into account international practices, i.e. application of safety factor n_a=2 to "high confidence of detection" crack depth from the NDE qualification results.

There should be a high confidence in the NDT detection, location and sizing capability of a crack-like defect postulated for the PTS analysis. Such capability and effectiveness of the NDT methods should be demonstrated through qualification of the in-service inspection systems used (NDT procedure, equipment and personnel) in line with respective national regulatory requirements, and with Ref. [3]. The actual quality and status of qualification of NDT for ISI at the particular plant for which the PTS analysis is carried out should be conservatively taken into account.

A probable fatigue propagation of postulated defects with depth smaller than ¼ of the wall thickness within the analysed time period should be conservatively evaluated and taken into account considering the required inspection frequency for the particular area where the defect is postulated.

7. INTEGRITY ASSESSMENT

7.1. EVALUATION OF RESULTS AND SAFETY FACTORS

To demonstrate the RPV integrity for a specified transient loading, two following conditions must be met simultaneously for the postulated crack with depth a:

$$n_k K_I (T, a) \leq [K_{IC} (T)]$$

$$K_I (T, a) \leq [K_{IC} (T\text{-}\Delta T)]$$

where K_I is the crack loading in terms of stress intensity factor, see Section 6.2 and $[K_{IC}]$ is the allowable stress intensity factor value. It should be obtained as the 5% lower tolerance bound of experimental fracture toughness data.

The following temperature dependence of $[K_{IC}]$ can be used for RPV PTS analysis:

$$[K_{IC}] = 26 + 36 \cdot \exp [0.02 (T\text{-}T_K)];\ [K_{IC}] \leq 200 \text{ MPa m}^{1/2}$$

This dependence corresponds to specimen thickness of 150mm and $P_f = 0.05$.

Other allowable stress intensity curves could be used if properly justified

The parameters n_K and ΔT are safety factors with respect to the origin of uncertainties in the overall PTS analyses. Therefore n_K represents uncertainties with respect to the assumptions on the loading and ΔT reflects uncertainties in the fracture toughness.

In Table III, the recommended values of safety factors to be used when carrying out PTS analysis for postulated defects are given, see Section 6.3. The values in Table III are based on Russian standards [10] and [27]. Other values could be also used, if justified.

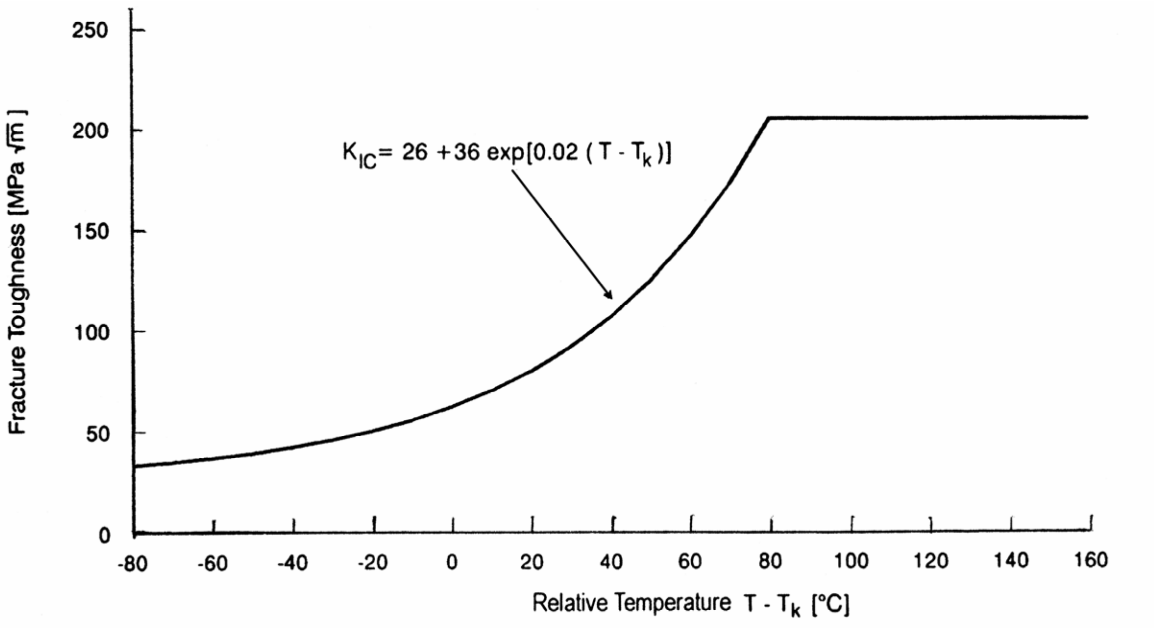

FIG. 6. Allowable stress intensity factors for WWER RPV steels.

TABLE III. SAFETY FACTORS

Safety factor	**Anticipated transients**	**Postulated accidents**
$\mathbf{n_K}$	$\sqrt{2}$	1
$\mathbf{\Delta T^oC}$	30	0

7.2. PRESENTATION OF RESULTS

Results for each analyzed event should be presented at least in a diagram form and archived as a data file. The following parameters should be indicated:
- primary circuit pressure as a function of time,
- coolant temperature in RPV inlet nozzle as a function of time,
- coolant temperature in the downcomer as a function of time in relevant positions,
- coolant-to-wall heat transfer coefficient as a function of time in relevant positions (when used for temperature fields calculations),
- postulated defect characteristics,
- stress intensity factors for postulated defects with respect to crack front location, different crack depth and shape as a function of temperature.

For the determination of safety margins, the following characteristics are also necessary:

- time dependence of neutron fluence for analyzed RPV areas as a function of operation time,
- critical brittle fracture temperature T_K as a function of time,
- RPV allowable stress intensity factor as a function of temperature.

7.3. ASSESSMENT OF RESULTS

For each individual analyzed sequence (i), the material behaviour in terms of allowable stress intensity factor [K_{IC}], and the crack loading path in terms of stress intensity factor K_I, are considered as a function of temperature and should be presented in a stress intensity factor respectively allowable stress intensity factor vs. temperature diagram, taking into account also the safety factors as indicated in Section 7.1. A schematic description of the assessment is provided in Fig 7.

The allowable critical brittle fracture temperature for analyzed sequence (i), $T_k^a(i)$, corresponds to the allowable stress intensity factor curve shifted horizontally up to the point where it becomes a tangent to the crack loading path of the sequence (i). The vessel maximum allowable critical brittle fracture temperature T_k^a is equal to the minimum value of the set of obtained $T_k^a(i)$ values for all sequences analyzed.

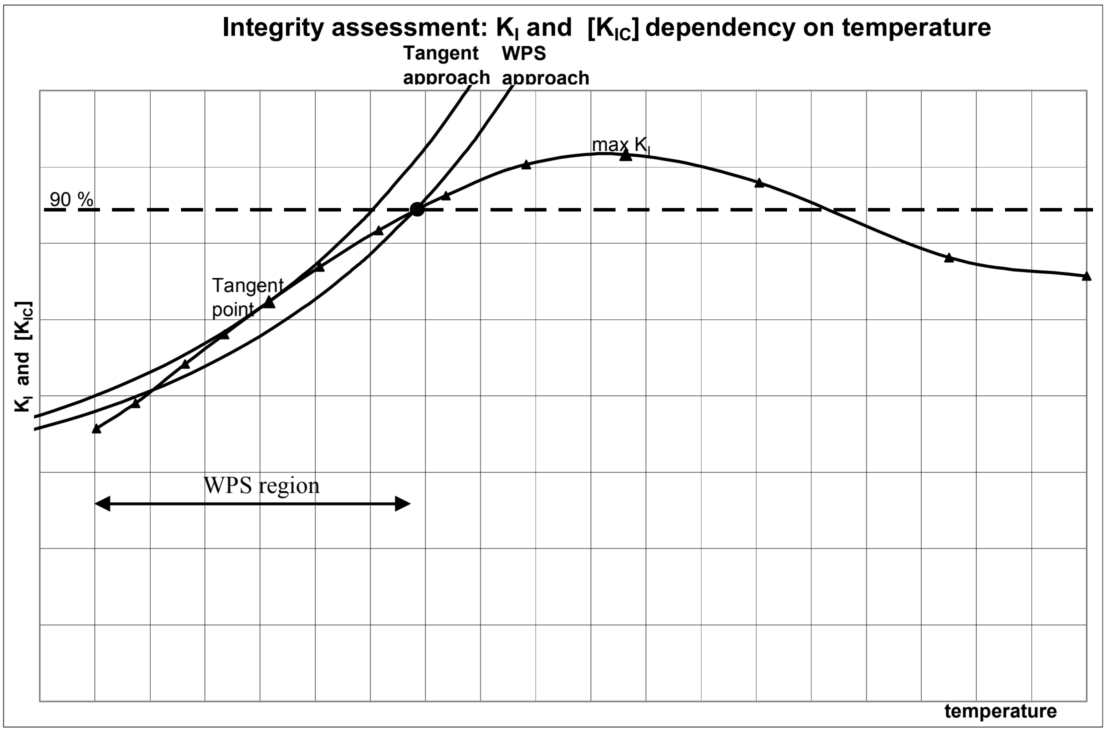

FIG. 7. Integrity assessment.

In the assessment, warm prestress(WPS) could be credited for loads below 0.9 of peak stress intensity factor in the continuously (monotonically) decreasing crack loading path, see also Fig.7, utilizing the assumption, demonstrated by large scale testing, that crack initiation does not occur in the decreasing crack loading path [17].

In the case with reloading, i.e. when the path of temperature dependence of K_I is not monotonically decreasing, allowable critical brittle fracture temperature, $T_k^a(i)$, can be determined using the most conservative value from all 90% of local maxima of stress intensity factor K_I.

When credit is being given to warm prestress, its applicability in particular for materials with higher embrittlement rate should be carefully considered since it may not be fully applicable in the highly embrittled materials. The national regulatory requirements may not allow to use this approach directly and further justification may be needed.

The difference between the vessel maximum allowable critical brittle fracture temperature and the vessel material critical brittle fracture temperature (for its determination see sect. 8) determines the safety margin.

The value of this safety margin should be larger than or equal to zero depending on the national regulatory requirements and considering the reliability of individual input data, such as material properties and effectiveness of NDT.

If crack initiation for a postulated defect cannot be excluded for an accident sequence with sufficient margin, crack arrest approach could be used to demonstrate, that the initiated crack will arrest within the vessel wall thickness (related material data are needed). The crack arrest assessment should be performed according to national standards.

7.4. UNCERTAINTY OF RESULTS

In the analysis, the source of uncertainties could be associated with the following aspects:

– material properties including fracture toughness in initial as well as end-of-life states;
– neutron fluence;
– transient description (gradient, final temperature, pressure);
– fluid temperature and heat transfer coefficients;
– assumptions of the structural analysis model including boundary and initial conditions;
– method of calculation of stress intensity factors;
– crack geometry and size with respect to NDT effectiveness;
– operator action.

Therefore, careful consideration should be given to these aspects and if necessary, sensitivity studies should be carried out. Further, computer codes used should comply with the specific requirements discussed in Section 10.

8. MATERIAL PROPERTIES

8.1. GENERAL INFORMATION

The principal characteristic of the RPV ferritic material used in the assessment is the fracture toughness K_{IC} as a function of temperature.

The allowable stress intensity factor values for a particular material are based on a statistical evaluation of the fracture toughness database and are given as a function of relative temperature $T\text{-}T_k$, and for this particular material are adjusted to an indexing temperature T_k.

Critical brittle fracture temperature T_k, obtained using Charpy impact testing, is used as the indexing temperature. The shift of the K_{IC} curve is assumed to be equal to the shift of the critical brittle fracture temperature due to the following degradation mechanisms: radiation embrittlement, thermal ageing, and fatigue damage.

Critical brittle fracture temperature T_k is evaluated by the formula [10]:

$$T_k = T_{k0} + \Delta T_T + \Delta T_N + \Delta T_F,$$

where	T_{k0}	initial critical brittle fracture temperature
	ΔT_T	shift in T_k due to thermal ageing
	ΔT_N	shift in T_k due to fatigue damage
	ΔT_F	shift in T_k due to neutron irradiation

Fracture toughness of cladding should be based on a lower bound estimate.

An alternative method based on "Master Curve" concept can be also used. More detailed information is given in Appendix IX.

8.2. INITIAL CRITICAL BRITTLE FRACTURE TEMPERATURE

A conservative value of the initial critical brittle fracture temperature T_{k0} should be used in the assessment, which corresponds to the highest value of T_{k0} obtained by acceptance testing (material certificate) for the particular material or from technical specification if acceptance test data are not available.

When no reliable experimental data exists, a conservative estimate can be used provided that its conservativeness is adequately and thoroughly justified, and if allowed by the national regulatory requirements.

8.3. CHEMICAL COMPOSITION

Chemical composition of RPV materials used in the assessment should be based on conservative values of measurements results obtained on actual production test coupon welded by identical procedure and materials (e.g. welding equipment, team of welders, same heat of material) as the weldments of the vessel and provided in the respective material certificate.

If no reliable data on chemical composition exists, a conservative estimate of chemical composition supported by reliable vessel sampling should be used.

The sampling method must ensure by adequate means, such as thorough chemical analysis, that representative weld material will be tested. In particular the zone diluted by the carbon electrode cover weld at the surface must be avoided.

At the same time it should also be taken into account that the acceptance tests at the manufacturer, at least for base metal, are performed with specimens cut out from ¼ of the thickness.

8.4. IRRADIATION EMBRITTLEMENT

Embrittlement

The following formula is used to evaluate the shift in T_k due to neutron irradiation ΔT_F [10]:

$$\Delta T_F = A_F^T \, (F_n/F_o)^n$$

where:

A_F^T	irradiation embrittlement factor for irradiation at temperature T, °C;
n	exponent
F_n	neutron fluence, E ≥ 0.5 MeV, neutron/m^2;
F_o	10^{22} neutron/m^2.

This formula is valid for neutron fluences in the range $1 \times 10^{22} < F_n < 4 \times 10^{24}$ neutron/m^2, E ≥ 0.5 MeV for WWER-440 steels, and in the range $1 \times 10^{22} < F_n < 6 \times 10^{23}$ neutron/m^2, E ≥ 0.5 MeV for WWER-1000 steels and within the limits of chemical composition for a given material provided in Table IV.

TABLE IV. VALIDITY LIMITS FOR EMBRITTLEMENT INDUCED SHIFT PREDICTION

Material	% P	% Cu	% Ni
15Ch2MFA base metal 15Ch2MFA weld metal	<0.025 <0.042	- -	- -
15Ch2MFAA base metal 15Ch2MFAA weld metal	<0.012 <0.012	<0.10 <0.10	- -
15Ch2NMFA base metal	<0.020	<0.20	*
15Ch2NMFAA base metal 15Ch2NMFAA weld metal	<0.010 <0.010	<0.10 <0.08	* *

* The data available at present suggest that the A_F values given above are applicable for 15Ch2NMFAA and 15Ch2NMFA steels for Ni contents up to 1.3 wt.%. For higher Ni contents the irradiation embrittlement should be assessed case by case.

The neutron fluence and its profile across the vessel wall thickness including cladding interface needs to be precisely characterized in order to evaluate the irradiation induced changes in the reactor pressure vessel wall.

For different types of materials used for RPVs of WWER reactors the following values of irradiation embrittlement prediction formulae should be used (while other formulae appropriately justified are also applicable) [33, 10]:

WWER-440

Weld metal*

$$\Delta T_F = [884 \times P + 51.3 \times Cu] \times (F_n / F_o)^{0.29} = 800 \times (1.11 \times P + 0.064 \times Cu) \times (F_n / F_o)^{0.29}$$

σ=22.6 °C

Base metal*

$\Delta T_F = 8.37 \times (F_n/F_o)^{0.43}$

σ = 21.7 °C

** valid for neutron fluences in the range $10^{22} < F < 4 \times 10^{24} m^{-2}$*

Where ΔT_F is a mean prediction value and σ is a standard deviation.

WWER-1000

A_F	29 for 15Ch2NMFA base metal (T_{irr} = 290°C)
A_F	23 for 15Ch2NMFAA base metal (T_{irr} = 290°C)
A_F	20 for 15Ch2NMFAA welds (T_{irr} = 290°C)
n	0.333

Provided that the above indicated conditions are satisfied, the formula gives an upper bound of the critical brittle fracture temperature shift due to neutron irradiation.

Plant specific surveillance data should be analyzed and compared with values obtained from the formula. The irradiation embrittlement trend for a given unit to be used in lifetime assessment can be generated using reliable statistical method.

For cases outside of limits given in Table IV, the irradiation embrittlement trend prediction should be based on reliable and representative surveillance data interpolation increased by 1σ margin. Case by case justification is required to demonstrate that such a prediction provides a conservative bounding estimate of irradiation embrittlement for a given vessel.

Annealing and re-embrittlement

The material properties of annealed vessel are determined by the residual shift in critical brittle fracture temperature, $\Delta T_{F, res}$, and the re-embrittlement behaviour, see also [14].

The value of the residual shift should be based on a conservative assessment of available data and, if feasible, supported by experimental verification by mechanical testing of samples directly cut from the vessel.

The re-embrittlement behaviour can be described by using the vertical, lateral or conservative shift of transition temperature. The use of a particular approach to derive the material properties to be used in the vessel assessment should be based on a conservative analysis of data available and its conservativeness adequately justified using experimental data wherever possible.

8.5. FATIGUE AND THERMAL AGEING

The shift of the critical brittle fracture temperature T_k due to fatigue damage, ΔT_N, is assumed to be equal 0°C for the beltline zone.

The shift of the critical brittle fracture temperature T_k due to thermal ageing, ΔT_T, is assumed to be equal 0°C for base and weld metal of WWER-440 reactor pressure vessels.

The shift of the critical brittle fracture temperature T_k due to thermal ageing, ΔT_T, for base and weld metal of WWER-1000 reactor pressure vessels could be larger than 0^oC and should be taken into account in the assessment.

9. CORRECTIVE ACTIONS

Corrective actions are required, if it is not possible to demonstrate the vessel integrity. These could address either material properties or loads or both. In addition, reliable in-service inspection could provide a basis for removal of some over-conservativeness included in the original vessel assessment practices, if properly justified through qualification of the in-service inspection systems used at each particular plant.

Corrective actions need to be properly justified on a plant specific and case by case basis if they are to be credited in the PTS analysis. In all cases, respective national requirements should be taken into account and complied with.

9.1. NEUTRON FLUX AND MATERIAL PROPERTIES

Corrective actions addressing material properties could either lead to reduction of the rate of embrittlement by measures reducing neutron flux on the RPV wall or to restoration of material properties by thermal annealing.

Flux reduction measures could include the use of low leakage core loading pattern, use of partial shielding assemblies with outer fuel pins replaced by steel pins, insertion of dummy shielding assemblies and use of fuel with poison at the core periphery. The benefits of flux reduction depend on the time of implementation, original flux level and chemical composition of the vessel material. The implementation of some flux reduction measures may result in plant power derating.

The selection of the corrective action to be used has to be carefully considered on a plant specific basis. Plant specific verification by measurements is required to confirm the desired effects are being achieved.

Mechanical properties of embrittled vessel could be restored by thermal annealing, i.e. heating up of the critical vessel section to temperatures higher than the irradiation temperature. Two PWR RPVs were annealed using the "wet" low temperature annealing at 345°C, resulting in relatively low recovery of the material properties. The "dry" high temperature annealing at 460°C to 475°C has been applied to WWER-440 reactors and produced substantial or even almost complete recovery of the embrittled vessel weld.

The degree of recovery and reembrittlement behavior with respect to transition temperature shift have to be evaluated on plant specific basis. Further consideration of upper shelf drop and of other impacts on the vessel itself and on other structures of the plant may be also necessary.

9.2. LOADS

The reduction of vessel loading could be achieved by plant design modifications or by modification of operational procedures.

Plant design modification include for example heating up of the ECCS water tanks or sumps, operator or automatic control of ECCS heat exchanger,, modification of high pressure injection capacity and shut-off head, redirection of the safety injection in the downcomer to improve mixing, implementation of low temperature overpressure protection system, modification of steamline isolation criteria, installation of fast acting isolation valves in the secondary circuit, reduction of hydroaccumulator pressure, especially for the hydroaccumulators connected to downcomer, and others.

The modification of operational procedures involves operator training and establishing operational actions based on integrated systematic evaluation of PTS events. These include instructions to avoid isolation of breaks, high pressure injection pumps throttling etc. System responses need to be thoroughly understood by operators and detailed guidelines for operator actions developed. Such steps could reduce the probability of severe overcoolings or reduce the overcooling severity itself (in terms of thermal stresses).

These modifications need to be based on a detailed plant specific PTS analysis; the impact of modifications proposed on the core cooling in general has to be also evaluated.

10. COMPUTER CODES

10.1. GENERAL REQUIREMENTS

Computer codes play a crucial role in various stages of the PTS analysis. Extensive code application is needed for the following tasks:

Fluence calculations

- neutron fluence and its profile across the RPV wall for the considered period of operation.

Thermal hydraulic calculation

- thermal hydraulic system behaviour;
- thermal mixing in the downcomer.

Structural analysis

- temperature field in the RPV wall;
- structural response;
- fracture mechanics parameters for postulated defects.

The general requirements from [1, 5] have been included in this Section. For details and further references [1, 5] should be consulted.

The confidence in the results depends strongly on the capability of the code to model the pertinent physical phenomena and on the judicious preparation of input data for the calculations.

In order to ensure consistency of results and permit independent review of the analysis, a comprehensive documentation of the code(s) used should be developed.

All models and correlations used in the code should be explained. The applicability range of correlations must be stated and they should not be used outside that range.

It should be ensured, that the numerical scheme is suitable and convergent. Truncation errors and numerical diffusion should be kept within an acceptable level.

Since the user of codes may not have been involved in the code development and testing, code documentation should include detailed user guidelines. These should include, among others, guidance on how to make a good nodalization scheme (examples), and detailed guidance in the use of each specific component, node type or separated model. Further measures to reduce the user effect should be applied.

Advances in safety research usually lead to better understanding of physical phenomena and consequent improvements in the computer models. Each version of the code should be clearly identified and all correction and changes included should be documented.

Based on the experience from benchmark exercises it is recommended to perform uncertainty and sensitivity analyses to identify the main influence factors on the relevant results to support the assumptions for PTS analysis.

To ensure consistency of the analysis results of the computer codes used for integrity assessment, a comprehensive documentation of the methods and user guidelines including simple verification examples should be available. Furthermore the users of the codes should be well-trained, which can be achieved by participation in round robin exercises or by using benchmark results for training purposes. Due to the common practice that structural and thermo-hydraulic calculations are performed separately, the adequacy of the coupling should be assessed for each specific application.

10.1.1. Fluence calculations

Because of high neutron fluence attenuation between the core and RPV the calculated RPV fluence is strongly sensitive to the physical model of the core and RPV internals as well as to the mathematical model of the neutron transport calculations. The accurate determination of the RPV fluence is difficult and comparisons of measured and calculated data show varying degrees of agreement for different WWER designs and different core loading schemes [18].

In spite of the progress in both deterministic and stochastic transport calculation and revision of basic cross section data and systematic benchmarking [19], models and methodologies should be validated in forehand on the basis of benchmarks. There is a difference between PWR and WWER type reactors in the design i.e. in the calculational modeling (geometry, material compositions, etc.) and the calculational methodology should be qualified in benchmarks and mock-ups including the corresponding group cross section libraries. The calculational uncertainty should be evaluated.

The source (core power) distribution is to be provided by standard codes with detailed pin to pin calculation of the distribution in the outer (WWER-1000) or two outer (WWER-440) rows of the fuel assemblies. The WWER-440 control assemblies consist of absorbing and fuel parts and any change of their position can cause remarkable changes in the source distribution during the cycle. The local flux distributions have to be taken into account [20].

The absolute fluence calculation, rather than any calculational extrapolation of fluence measurement, should be used for the RPV neutron fluence determination.

According to existing standard procedure the integral fast neutron fluence above 0.5 MeV should be reported. The presentation of the fluences above 0.1, 0.5, 1.0 MeV and the neutron spectra, calculated and measured, are recommended. The report on RPV fluence evaluation should also contain the details of calculation and experimental methods including the results of their qualification.

In particular attention should be given to:

- the selection of validated database for fast neutron cross sections (including possible corrections) and their treatment (grouping or not);
- the adequate fission spectrum;
- the detailed geometry and its modeling of the core, barrel, water annulus and pressure vessel;
- the chemical composition of the various materials (considering conservative assumption according to the specification or the acceptance data);
- the temperature of all individual parts (metal and fluid);
- the computer code to be used;
- neutron flux spectrum above 0.5Mev in the core, at the core boundary and inner pressure vessel wall.

The uncertainty of the evaluated integral fluences should be less than 20% (one sigma) and this uncertainty should be considered in the fluence determination.

The international cooperation in benchmarking and methods qualification is recommended.

10.1.2. Thermal hydraulic calculations

Thermal hydraulic system codes

A basic requirement is the adequacy of the physical model being used to represent plant behaviour realistically. The choice of the model also depends on the accident being evaluated.

The models should include an accurate presentation of the pertinent part of the primary and secondary systems. Particular attention should be given to the modelling of control systems.

The thermal hydraulic models should be capable of predicting system behaviour and critical flow in single and two-phase flow conditions. The models should be capable of predicting plant behaviour for LOCAs, steam line breaks, primary-to-secondary leakage accidents, and various overcooling transients. In general, a one dimensional lumped parameters code is suitable for most overcooling sequence calculations (except for thermal stratification as discussed below). In case it is likely that the non-uniform temperature and velocity fields in reactor downcomer can influence overall system behaviour (especially circulation rates in individual loops), the application of system TH codes with 2D/3D capabilities is more appropriate than a simple 1D system TH calculation.

The models should be capable of predicting condensation at all steam-structure and steam-water interfaces in the primary system, especially in the pressurizer during the repressurization phase of an overcooling event or during refilling of the primary system with safety injection water. The effects of noncondensable gases (if present) on system pressure and temperature calculations should be included.

In special cases the thermal hydraulic models should be coupled with appropriate neutronic models that have the capability to analyse pressure surges resulting from sequences involving recriticality.

Thermal mixing calculations

An important feature of some PTS transients is flow stagnation in the primary circuit. It occurs when the flow distribution is governed by buoyancy forces (i.e. thermal stratification and mixing of cold high pressure injection water in the cold legs and the downcomer become the dominant effects). These phenomena can also be influenced by the loop seals behaviour. These phenomena are not predicted correctly with the existing thermal hydraulic system codes. Therefore specific fluid-fluid mixing calculations are needed as discussed above in Section 5.

10.1.3. Structural calculations

For the RPV integrity assessment a three step structural analysis is necessary:

- structure temperature field;
- structure stress-strain fields;
- loading of postulated defects in terms of stress intensity factor.

For the determination of temperature distribution and the structural response the numerical FEM method is normally used. Crack loading can be calculated with computer codes based on simplified engineering methods or by FEM analysis with postulated cracks generated in the FEM mesh. The choice between linear elastic or elastic plastic 2D or 3D FEM analysis with or without postulated cracks generated in the FEM mesh depends on several factors such as: the degree of accuracy required in the results, the complexity and severity of the loading conditions, the existence of RPV inner surface cladding, and the computation time. The confidence in the results depends on the capability of the codes to model the related physical phenomena. Therefore a basic requirement is the adequacy of the methods to represent the structural and fracture behavior of components like the RPV based on plant specific geometry and loading conditions.

Generally the 3D elastic plastic FEM approach is capable of providing full information on vessel behavior during overcooling transients with asymmetric loading assumptions. The fracture assessment of postulated cracks generated in the FEM mesh based on the calculation of the J-integral or the energy release rate G converted into the stress intensity factor allows to quantify safety margins and estimate the degree of conservativeness of simplified engineering methods. The 3D FEM method with J-integral is available in several recognized code packages such as ABAQUS, ADINA, ANSYS, BERSAFE, SYSTUS, CORPUS, MARC, RASTR-SIGMA, etc. The treatment of problems in the frame of RPV integrity assessment with such complex codes should be stepwise and after a sufficient check of the analysis technique by analyses of experiments with fracture problems.

10.2. CODE VALIDATION

The quantification of the status of validation can be expressed in terms of the accuracy of the code predictive capabilities for specific output quantities, which can be derived by theoretical formula or measured in the frame of experiments or plant monitoring. In practical sense the validation process includes the comparison between experimental and analysis results which could effect (if necessary) a reformulation of the analytical model. Sometimes a code can predict a set of parameters with high degree of accuracy and still be inaccurate for other ones. This has led to the need to develop a validation matrix with respect to different types of experimental facilities and different sets of conditions in each facility. In that sense experiments influence the code development and vice versa.

Furthermore round robin calculations have been performed for a WWER reactor pressure vessel loaded under overcooling transients based on thermal hydraulic calculations under the auspices of IAEA [29, 30] and for a western type RPV in the frame of the CSNI project RPV PTS ICAS [31]. The validation process is an ongoing effort. It is proposed to improve the assessment of the predictive capabilities of computer codes by further calculations and by round robin calculations.

10.2.1. Fluence calculations

Due to uncertainties in physical parameters (cross section, etc.) and difficulties in accurate modelling of neutron transport in the reactor, the experimental verification of fluence calculation should be performed. The fluence calculation for the RPV inner surface should be verified by results from a surveillance programme using mock-up transfer data or by direct measurements using scrap method. The fluence on the outer RPV surface should be validated by cavity measurements. These data should serve also for validation of fluence on inner RPV surface retrospectively using transfer coefficients based on mock-up experiment.

Therefore, for fluence calculations a two-step validation is required:

- validation of individual physical data by adequate experimental benchmarking;
- industrial or mock-up benchmarking including comparison with available data, such as retrospective dosimetry, ex-vessel measurements, surveillance data.
-

The calculations should be performed by higher order theory than diffusion theory.

10.2.2. Thermal hydraulic calculations

The principal requirement is that the phenomena of interest have to be described to a sufficient degree of accuracy. Usually the choice of the mechanisms to be described and the method to combine those selected mechanisms are based on various assumptions. The validation process must therefore be concerned with the following aspects: modelling of individual mechanisms, the way of combining them, the simplifying assumptions and the possible lack of inclusion of some of the important mechanisms.

The applied thermal hydraulic system code and fluid flow mixing code are required to provide input for the structural analysis in terms of the downcomer temperature field, heat transfer coefficient field, and the primary pressure during selected transients and accidents.

The validation process relates to the confidence on the accuracy of the predicted values.

The principles of validation process and the recommendations for practical validation against test data, plant data and standard problems data have been discussed in more detail in Ref. [1]. These principles are generally valid for the purposes of the PTS thermal hydraulics. In particular adequate modelling of natural circulation and validation for such regime is important.

Fluid flow mixing codes should be able to describe the phenomena like mixing near the injection location, stratification in the cold leg and mixing in the downcomer. Post-test assessment calculations of available experiments should confirm that the applied fluid flow mixing code is valid.

An extensive data base exists for thermal fluid mixing that is relevant to PTS issue [21]. The applied fluid flow mixing method should be validated against this data base.

10.2.3. Structural analysis

The integrity assessment of nuclear components relies strongly on the validity of the computer codes used for structural and fracture mechanics analysis. The development of the codes is guided by the principle that the phenomena of interest can be described to a sufficient level of accuracy. A predictive structural code is typically a compilation of individual mechanisms (e.g. material behavior, deformation, heat conduction, crack behavior) which are combined into analytical models. Usually the choice of the model to describe the mechanisms is based on a variety of assumptions. Therefore the validation of a code must take into account the procedures to model the individual mechanisms including simplifying assumptions.[a]

In the frame of RPV integrity assessment due to PTS loading, large scale thermal shock tests have been performed at several experimental facilities managed by organizations like EDF (France), MPA Stuttgart and HDR (Germany), JAPEIC (Japan), Prometey (Russia), AEA-Technology (UK), ORNL (USA). Fracture analyses have been performed with different analysis methods and by different organizations. In the frame of the CSNI project FALSIRE [22, 23] the results of up to now 84 analyses for 13 tests have been compiled into an electronic database for comparative assessment concerning the predictive capabilities of fracture mechanics analysis methods. The database is available on request at GRS, Germany, Department of Barrier Effectiveness.

11. QUALITY ASSURANCE

11.1. GENERAL REQUIREMENTS

The overall PTS analysis including its individual parts on:

- transient selection,
- thermal-hydraulic analysis,
- fluence calculations,
- temperature field, stress and fracture mechanics calculations,
- material properties assessment,
- assessment of results

should be subject to formally established quality assurance procedures in line with applicable national codes and standards and Ref. [26]. Such procedures should consider the following general principles:

(a) The responsibility of any individual working in the organization involved in the analyses should be clearly specified.

(b) The qualification of experts should be sufficiently high and adequately documented.

(c) Calculational notes and results should be documented the extent needed to allow their independent checking by qualified reviewers.

(d) Only validated methods and tools should be used.

(e) Procedures and results should be independently reviewed both from a technical as well as a procedural point of view.

(f) All differences found during the review should be resolved before the final use of the results.

11.2. SPECIFIC REQUIREMENTS

The general principles given above may be expressed more specifically as follows:

[a] Assessing the status of validation of any computer code is difficult because there has not been a formal consensus on what constitutes a validated code. In some countries (USA, UK) procedures to certify code capabilities have been developed.

(a) Selection of initial data and boundary conditions, computer codes and users, influence the quality of results, therefore all of them should be subject to quality assurance procedures.

(b) Any activity should be performed only by qualified personnel. A record documenting the qualification should be maintained.

(c) The origin and version of computer codes used should be clearly documented and must be referenced in order to allow a meaningful evaluation of a specific accident analysis. Computer codes should be verified and validated for the relevant area of their application; verification and validation should be documented.

(d) All sources of primary plant data should be clearly referenced. Derivation of input data for the analysis from primary information should be documented in such way, which permit adequate control, review, check and verification. A form should be used which is suitable for reproduction, filing and retrieval. Notes should be sufficiently detailed such that a person technically qualified in the subject can review, understand and verify the results.

(e) It is advantageous to have one "master" input deck. All calculations should be done introducing the necessary changes (e.g. initial conditions, functioning of safety systems) with respect to this "master". All such changes should be documented so that it can be traced to the date in which improvements/error corrections have been done. Inputs should be designated in a way that permits later checking. Data permitting reconstruction of calculated results must be archived (including relevant parametric studies).

(f) For each case analyzed a sufficient description of input data, basic assumptions and process and control system operational features should be provided giving a possibility of a unique interpretation and reproducibility of the results. It is recommended to follow the same format for all cases analyzed.

(g) "User effects" should be reduced to minimum. This implies that guidelines should be developed at the institution performing the analyses, permitting each member of the staff to benefit from the experience accumulated in applying a given computer code. For the same reason, data transfer between computer codes should either be automatic or it must be assured that they are defined in an unequivocal way.

(h) Results should be presented in such quality and detail to allow the reviewer to check the fulfillment of all acceptance criteria and to understand properly all elements and in particular the interdisciplinary aspects (interfaces) of the PTS analysis. The same format for presentation of results for all cases under consideration is recommended.

Results of analysis should be archived for a sufficiently long period of time.

(i) All calculations and analyses should be checked by a competent individual other than the author. The following methods may be used for checking the adequacy and correctness of calculations:

- Independent review and checking of calculations,
- Comparison of the results with results of other methods, such as:
  (i) simplified calculations
  (ii) alternate calculational methodology.

Other appropriate methods may be also used.

The review process and all comments as well as deficiencies found by the reviewer should be adequately documented. Specifically it must be documented which parts of calculations and results have been checked and which methods have been adopted.

In response, the author should properly address all comments and remove all deficiencies to the satisfaction of the reviewer.

(j) All input data for structural analysis (like RPV geometry, material properties etc.) should be documented according to the QA manual prepared for the PTS analysis.

Appendix I

LIST OF INITIATING EVENTS FOR WWER NPPs

This appendix contains the list of initiating events to be considered for WWER NPPs as recommended in Ref. [1]. Some of these events were not the original design accidents for existing WWERs, especially for old models (i.e. WWER–440/230). This fact should be taken into account when using recommendation of this document, concerning conservative assumptions, boundary conditions and acceptance criteria.

The events listed are referred to category T (anticipated transients) or A (postulated accidents) to enable the correct application of acceptance criteria which depend upon the category of event.

1.	Reactivity and power distribution anomalies	
1.1	Uncontrolled withdrawal of a control rod group during startup	T
1.2	Uncontrolled withdrawal of a control rod group during power operation	T
1.3	Control rod maloperation	
	– drop of one CR	T
	– withdrawal one CR from a control group	A
	– statical misalignment of one CR in a control group	T
1.4	Incorrect connection of an inactive RCS loop	A
1.5	Control rod ejection	A
1.6	Decrease of the boron concentration in the reactor coolant due to chemical and volume control system malfunction	T
1.7	Inadvertent loading and operation of a fuel assembly in an improper position	A
2.	Decrease in reactor coolant flow rate (LOFA)	
2.1	Inadvertent closure of one main isolation valve in a RCS loop	T
2.2	Seizure of one main circulation pump (MCP)	A
2.3	Break of the shaft of one MCP	A
2.4	Single and multiple MCP trips	T
3.	Decrease in reactor coolant inventory (LOCA)	
3.1	Spectrum of postulated piping break within the reactor coolant pressure boundary	A
3.2	Rupture of the line connecting the pressurizer and a pressurizer safety valve	A
3.3	Inadvertent opening of one pressurizer safety valve	A

3.4	Leaks from the primary to the secondary side of the steam generator – SG tube rupture – Primary collector leaks up to cover lift-up	A
3.5	Rupture of I&C line or other lines from reactor coolant pressure boundary that penetrate containment	A
3.6	Inadvertent opening of one check or isolation valve separating reactor coolant boundary and low pressure part of the system.	A
4.	Increase in reactor coolant inventory	
4.1	Inadvertent actuation of ECCS during power operation	T
4.2	Chemical and volume control system malfunction that increases reactor coolant inventory	T
5.	Increase in heat removal by the secondary side	
5.1	Feedwater system malfunctions that decrease feedwater temperature	T
5.2	Feedwater system malfunctions that increase feedwater flow rate	T
5.3	Secondary pressure regulator malfunctions that increase steam flowrate	T
5.4	Inadvertent opening of one steam generator safety or relief valve or turbine bypass valve	T
5.5	Spectrum of steam system piping break inside and outside of containment	A
6.	Decrease in heat removal by the secondary side	
6.1	Control system malfunction that decreases steam flowrate	T
6.2	Loss of external electric load	T
6.3	Turbine(s) stop valve closure	T
6.4	Main steam isolation valve(s) closure	T
6.5	Loss of condenser vacuum	T
6.6	Main feedwater pump trips	T
6.7	Loss of on-site and off-site power to the station	T
6.8	Feedwater piping break	A
7.	Radioactive release from subsystem or component	
7.1	Radioactive gas waste system leak or failure	A
7.2	Radioactive liquid water system leak or failure	A

8.	Fuel handling events	
8.1	Fuel assembly drop during refuelling	A
8.2	Fresh and spent fuel cask drop	A
9.	Anticipated transient without scram[b]	
9.1	Uncontrolled withdrawal of a control rod group during startup or power operation	A
9.2	Loss of main feedwater flow	A
9.3	Loss of on-site and off-site power to the station	A
9.4	Loss of condenser vacuum	A
9.5	Turbine trip	A
9.6	Loss of electrical load	A
9.7	Closure of main steamline isolation valves	A
9.8	Inadvertent opening of one steam generator safety or relief valve or turbine bypass valve	A

[b] ATWS events are usually analyzed with relaxed assumptions and specific acceptance criteria.

Appendix II

WWER-440/230: LIST OF PTS INITIATING EVENTS

The following list of PTS initiating events is recommended to be considered for WWER-440 PTS analysis by the plant's designer at present:

1. Spectrum of postulated piping break within the reactor coolant pressure boundary.

2. Rupture of the line connecting the pressurizer and a pressurizer safety valve.

3. Inadvertent opening of one pressurizer safety valve.

4. Leaks from the primary to the secondary side of the steam generator:

   - SG tube rupture
   - Primary collector leaks up to cover lift-up.

5. Inadvertent opening of one check or isolation valve separating reactor coolant boundary and low pressure part of the system.

6. Inadvertent actuation of ECCS during power operation

7. Chemical and volume control system malfunction that increases reactor coolant inventory

8. Inadvertent opening of one steam generator safety or relief valve or turbine bypass valve.

9. Spectrum of steam system piping break inside and outside of containment.

10. Feedwater piping break.

11. RPV cooling from outside.

The safety concept of the WWER-440/230 was based on design rules and standards at the time of design and construction of these plants in former USSR. The primary piping was made exclusively from austenitic stainless steel. On the basis of this provision, primary circuit failures which would result in core damage were not taken into consideration. Therefore, the plant design does not include any special provisions to protect against large failure of the primary circuit. The safety concept also required that essential primary circuit equipment and its auxiliary systems should have high reliability during their lifetime. Therefore, the original design basis accident was the loss of integrity of the primary cooling circuit equivalent to a break of 32 mm diameter. The emergency core cooling system of limited capacity was used in original design. The two trains of the safety injection system, each fed by three pumps with capacity of 50 m^3/h and head of 12.5 MPa, are connected to primary circuit on the cold legs, at the suction and discharge of the MCP (total of 12 injection points using the connections of the primary purification system). The safety injection pumps (and the spray pumps) take suction from the 800 m^3 safety injection tank. Borated water in the tank is preheated to between 55°C and 59°C to avoid thermal shock loading of the RPV. At newer units 3 and 4 of Kozloduy NPP an additional low pressure injection system was installed. This safety injection system consists of three trains with one centrifugal pump each. The pumps discharge to the isolable part of the cold legs independently of the high pressure injection system.

In general the original basic design was found to be significantly conservative due to extended built-in safety margin. However, compared with current practice for most other plant types, the original maximum DBA is very limited. Significant effort was devoted therefore to increase DBA of existing units. This is in particular case of two units of Bohunice V-1 NPP, where complex modernization was performed.

Besides number of other measures, existing high pressure safety injection lines were separated from purification system. New low pressure injection system with high capacity and relatively high head was installed. The system is discharging into hot legs of two loops. Maximum DBA was increased to break with equivalent diameter 200 mm (double ended break when using “best estimate” approach). The impact of upgraded ECCS on RPV integrity was considered carefully in parallel with core cooling aspects. Besides other sequences, extended spectrum of LOCAs ranging from small breaks to maximum size breaks for which core cooling is assumed, was taken into account. Symptom oriented emergency operating procedures with complex treatment of RPV integrity aspects are in final stage of the development.

Appendix III

WWER-440/213 (LOVIISA): LIST OF PTS INITIATING EVENTS

Downcomer overcooling

The sequences for the deterministic analysis for licensing purposes have been selected based on the integrated PTS study by accounting for all the sequences with a higher through-wall crack frequency than 10^{-8} [24]. After deterministic screening (e.g. elimination of those where excessive failures were assumed) and combination of the sequences, the following six transients have been selected:

- steam generator collector break (90 cm^2) where operator isolates the break after twenty minutes from the onset of the accident;
- large break LOCA with no isolation;
- stuck-open pressurizer safety valve (and later reclosing) after repressurization of the primary circuit following small steam line break;
- small break LOCA with very low decay heat, which leads to the flow stagnation;
- medium size isolatable LOCA in hot standby conditions; and
- inadvertent pressurization in the cold state.

Since the fracture-mechanics calculations are made with the 3D methods, the nonuniformity effects are included. The temperature decay in the HPI plume and the transient downcomer cooldown is calculated with the REMIX code. The heat transfer coefficient 5000 W/Km^2 is applied for the plume region where flow velocities are higher. In the ambient downcomer fluid the heat transfer coefficient value of 2000 W/Km^2 is applied. The values are consistent with the values reduced from the various thermal mixing experiments.

External cooling of the pressure vessel

During most of the accident sequences, the Loviisa reactor cavity will be flooded with the water. This water comes from the primary circuit leakage, from the ECCS and spray systems and from the melting ice condenser. Flooding of the reactor cavity might take place also during a spurious operation of the spray system, even though such sequences can be shown to be very unlikely.

The external cooling of the pressure vessel may lead to the integrity problem, if the crack is assumed to be located on the outer surface of the vessel. Fortunately, embrittlement due the neutron fluence is not so severe near the outer surface.

The most limiting conditions are obtained for the medium size LOCA case, where the cavity flooding is added to the sequence. New difficulties and uncertainties appear when trying to estimate the effect of the external cooling. The system code calculations cannot be utilized, because the happenings outside the coolant systems are not modelled. Temperature of coolant flooding the reactor cavity is not easily estimated for the accident situations. The flow paths and heat transfer behaviour in the gap between the thermal insulation and the reactor vessel have to be estimated. Additionally, the assumptions concerning the crack sizes have to be reconsidered, because the inspection work for the reactor vessel outer surface are different.

In addition to the deterministic PTS assessment, a probabilistic study has been performed. The initiator classes are shown in Table V (for the original study in 1986). The table also defines how many sequences were selected from each initiator class for further investigations.

:

TABLE V. INITIATOR CLASSES

CLASS	TITLE	SEQUENCES
	LOSS-OF-COOLANT ACCIDENTS	
11	Small break LOCA, FP	5
12	Medium size LOCA, FP	9
13	Large break LOCA, FP and HZP	3
14	Small break LOCA, HZP	5
15	Medium size LOCA, HZP	1
16	SG collector break, FP	1
17	SG collector break, HZP	1
18	SG tube rupture, FP	1
19	SG tube rupture, HZP	1
	STEAM LINE BREAKS	
21	MSLB inside containment, FP	9
22	SSLB before MSIV outside containment, FP	26
23	MSLB inside containment, HZP	8
24	SSLB before MSIV outside containment, HZP	15
25	Reactor scram	
26	Loss of off-site power, FP	
27	Loss of off-site power, HZP	
	OTHER CLASSES	
31	Inadvertent spraying of PRZ	
32	PRZ level control failure	
33	Inadvertent operation of make-up piston pumps	
34	Loss of condenser vacuum	
35	Inadvertent MSIV closure	
36	Inadvertent operation of high capacity make-up pumps	
37	PRZ pressure control failure	

INITIATORS DURING HEAT UP AND COOLDOWN		
51	Small break LOCA, H&C	3
52	Medium size LOCA, H&C	7
53	Large break LOCA, H&C	3
54	SG collector break, H&C	1
55	SG tube rupture, H&C	1
61	MSLB inside containment, H&C	7
62	SSLB before MSIV outside containment, H&C	10
71	Inadvertent operation of HPI, H&C	
72	Inadvertent operation of make-up piston pumps, H&C	
73	PRZ pressure control failure, H&C	
74	Isolation of primary system letdown, H&C	
75	Inadvertent operation of PRZ spray, H&C	
76	Inadvertent pressurization in cold state	2
77	Inadvertent operation of PRZ heaters	

FP full power
HZP hot standby
H&C Heatup and cooldown
SG steam generator
PRZ pressurizer
MSIV main steam isolation valve
MSLB main steam line break
SSLB small steam line break

Appendix IV

WWER-1000/320: LIST OF PTS INITIATING EVENTS

The following list of initiating events is recommended to be considered for WWER-1000 PTS analysis by the plant's designer at present:

1. Spectrum of postulated piping break within the reactor coolant pressure boundary.

2. Rupture of the line connecting the pressurizer and a pressurizer safety valve.

3. Inadvertent opening of one pressurizer safety valve.

4. Leaks from the primary to the secondary side of the steam generator:

   - SG tube rupture
   - Primary collector leaks up to cover lift-up.

5. Inadvertent opening of one check or isolation valve separating reactor coolant boundary and low pressure part of the system.

6. Inadvertent actuation of ECCS during power operation.

7. Chemical and volume control system malfunction that increases reactor coolant inventory.

8. Inadvertent opening of one steam generator safety or relief valve or turbine bypass valve.

9. Spectrum of steam system piping break inside and outside of containment.

10. Feedwater piping break.

Appendix V

FRENCH PWRs: LIST OF PTS INITIATING EVENTS

The list of design conditions has been analyzed and extended for PTS assessment using a multilevel selection process.

Particular attention is given to the conditions which induce a risk of high pressure and a significant thermal shock with low final temperature at the RPV inlet nozzles. The definition of the transient includes conservative assumptions for some variables (temperature of safety injection water, heat exchange coefficient, decay heat level, etc.) as well as realistic assumptions when justified (physical phenomenon, operating procedures, etc.).

Primary system breaks and main steam line breaks have been evaluated in detail. This part of the study has provided additional detailed transients for

- large primary circuit breaks (LOCA),
- small (∅ = 3", 2" and 1") primary leaks,
- large main steam line breaks before the main steam line isolation valves,
- intermediate (800 cm^2) and small (107 cm^2) main steam line breaks before and after the main steam line isolation valves.

In addition to those, a series of specific conditions, derived from degraded normal upset conditions by supplementary failures, which would increase the severity of the transients are defined. For that purpose, the different states of the reactor (operating, hot, intermediate and cold standby) are considered to define the envelope of transients. The corresponding frequency evaluation allows to classify the conditions in higher ranks than the initiating ones. This part of the study has provided the following complementary transients:

- cooling down and fuel removal,
- normal and incidental stop of the last primary coolant pump with normal pressure,
- inadvertent safety injection,
- very small, small and large leaks (approx. 25 mm^2) in the decay heat removal or the makeup system,
- inadvertent steam bleeding from the secondary circuit and consequent activation of the safety injection system,
- incidental opening of a residual heat removal system control valve,
- cooling and depressurization of the primary circuit by means of the pressurizer and consequent activation of the safety injection system,
- worse operation of the main feedwater system and consequent activation of the safety injection system.

By means of preliminary screening criteria, based on dominant parameters (difference between final temperature of the transient and the RT_{NDT} reference temperature, severity of the thermal shock and pressure) the envelope of transients has first been selected.

A preliminary study is conducted to select a limited set of leading transients by evaluating the critical defect size of surface and underclad defects from the preliminary list of pertinent transients, Table VI. The assessment of the reference defects is based on a reduced set of leading transients (at least one for each condition category), Table VII.

TABLE VI. PRELIMINARY LIST OF PTS TRANSIENT

Category	Conditions
2	Normal stop of the last primary coolant pump
	Cooling down and fuel refuelling
3	Inadvertent safety injection during single phase conditions - decay heat removal system connected
	Small leak on the main steam line after the main steam isolation valve - Case A
	Small leak on the main steam line before the main steam isolation valve - Case D - affected leg
	Small leak on the main steam line before the main steam isolation valve - Case E - non affected leg
	Small leak on the main steam line (design condition)
	Intermediate leak on the main steam line after the main steam isolation valve - Case A
	Intermediate leak on the main steam line before the main steam isolation valve - Case C - non affected leg
	Intermediate leak on the main steam line before the main steam isolation valve - Cases C & D - affected leg
	Small primary leak (3") - maximum decay heat - Case 1A
	Small primary leak (3") - minimum decay heat - Case 1B
	Small primary leak (2") - maximum decay heat - Case 2A
	Small primary leak (2") - minimum decay heat - Case 2B
	Small primary leak (1") - nominal power - maximum decay heat - with operator action - Case 3A
	Small primary leak (1") - nominal power - minimum decay heat - with operator action - Case 3A
4	Small leak in the decay heat removal or the makeup system - normal application of the operator procedure (Ti < 150°C)
	Main steam line break before main steam isolation valve without loss of off-site power (design condition)
	Main steam line break before main steam isolation valve with loss of off-site power (design condition)
	Main steam line break before main steam isolation valve - Cases A & B (affected leg)
	Main steam line break before main steam isolation valve - Case A (non affected leg)
	Intermediate leak on the main steam line after the main steam isolation valve - Case A
	Intermediate leak on the main steam line before the main steam isolation valve - Case D - non affected leg
	Small leak on the main steam line after the main steam isolation valve - Case B
	Small leak on the main steam line after the main steam isolation valve - Case C
	Small leak on the main steam line before the main steam isolation valve - Case E (non affected leg)
	Large primary circuit break (design condition)

Category	Conditions
	Large primary circuit break - Cases 1, 2, 9 & 10
	Large primary circuit break - Cases 3, 4, 11 & 12
	Large primary circuit break - Cases 5 & 6
	Large primary circuit break - Cases 7 & 8
	Small primary leak (1") hot standby - decay heat: 0.33% of nominal power - Case 3A
	Small primary leak (1") hot standby - decay heat: 0% of nominal power - maximum safety injection flow rate - Case 3D
BD	Large leak in the decay removal or makeup system - normal application of the A 10 procedure - Ti < 150°C
	Main steam line break before main steam isolation valve - Case B (non affected leg)
	Cooling down and depressurization of the primary circuit by the pressurizer - intermediate shutdown condition - natural circulation

TABLE VII. REDUCED SET OF LEADING TRANSIENTS

Category	Conditions
2	Cooling down and fuel refuelling (*)
3	Small primary leak (3") - minimum decay heat - Case 1B (*)
	Small primary leak (2") - maximum decay heat - Case 2A (*)
	Intermediate leak on the main steam line before the main steam isolation valve - Cases C & D - affected leg
4	Large primary circuit break - Case 1 (*)
	Main steam line break before main steam isolation valve with loss of-site power (design condition)
	Main steam line break before main steam isolation valve - Cases A & B (affected leg) (*)
	Intermediate leak on the main steam line before the main steam isolation valve - Case D - non affected leg

(*) reduced set of leading transients

Appendix VI

H.B. ROBINSON 2: LIST OF CONSIDERED EVENTS AND SELECTED SEQUENCES IN THE PTS EVALUATION

An evaluation of the PTS risk to the H.B. Robinson Unit 2, a Westinghouse PWR with three reactor coolant loops, was carried out by Oak Ridge National Laboratory with the assistance of several other organizations, [25]. This study was part of a NRC program to evaluate the PTS risk to three nuclear plants. The specific objectives of the program were to (1) to provide a best estimate of the frequency of a through-wall crack in the rpv together with the uncertainty in the estimated frequency and its sensitivity to the variables used in the evaluation; (2) determine the dominant overcooling sequences contributing to the estimated frequency and (3) evaluate the effectiveness of potential corrective measures. In the study, thousands of hypothetical overcooling events were constructed and quantified. All scenarios with a frequency greater than 10^{-7} per reactor year were considered explicitly, and those having lower frequencies were assigned to 12 "residual" groups to ensure inclusion of their contribution in the study.

The initiating events identified as potentially leading to one of the overcooling mechanisms were defined into eight classes as follows:

(1) Events causing a decrease in the charging water enthalpy.

(2) Events causing an excess steam flow from the steam generators.

(3) Events causing a decrease in the feedwater enthalpy.

(4) Events causing feedwater overfeed.

(5) Inadvertent safety injection (SI) events.

(6) Loss-of-coolant accidents (LOCAs).

(7) Events consisting of pressurizer pressure control failures.

(8) Events leading to steam generator tube ruptures.

Based on the further examination of these classes, 11 potential initiating events for overcooling were identified:

(1) A large steam-line break at hot 0% power.

(2) A small steam-line break at hot 0% power.

(3) A large steam-line break at full power.

(4) A small steam-line break at full power.

(5) A reactor trip from full power.

(6) Loss of main feedwater.

(7) A small-break LOCA at full power.

(8) A medium-break LOCA at full power.

(9) A small-break LOCA at hot 0% power.

(10) A medium-break LOCA at hot 0% power.

(11) Steam generator tube rupture.

Event trees were developed for each of these initiating events. The procedure to quantify and collapse the event tree sequences produced 209 sequences for which thermal-hydraulic and fracture-mechanics analyses were performed. Summary of the event tree sequence collapse is shown in Table VIII.

TABLE VIII. SUMMARY OF EVENT TREE SEQUENCE COLLAPSE

Sequence Series No	Initiator (Event Tree)	Number of Sequences			
				Grouped with Other Sequences[a]	
		To Be Analysed	In Event Tree	Above 10^{-7}/yr	Below 10^{-7}/yr
1, 11	Small-break LOCA at full power	22	6938	8	27
2	Medium-break LOCA at full power	12	6824	2	32
3, 12	Small-break LOCA at hot 0% power	5	158	4	4
4	Medium-break LOCA at hot 0% power	2	124	1	3
5	Small steam-line break at full power	29	923	6	28
6	Large steam-line break at full power	15	1763	10	41
7	Small steam-line break at hot 0% power	16	292	4	7
8	Large steam-line breat at hot 0% power	9	508	0	4
9	Reactor trip	90	9773	54	174
10	Tube rupture	5	NA	NA	NA
13	Loss of feedwater	6	NA	NA	NA
14	Support system failure	3	NA	NA	NA

[a] A screening frequency of 10^{-7}/yr was used to initially identify sequences which should be analysed on an individual basis.

NA: Not applicable.

For the estimations of pressure, temperature and heat transfer coefficient profiles, the overcooling sequences were regrouped on the basis of the controlling thermal-hydraulic phenomena. Altogether, the sequences were reduced to 10 groups (A-J).

TABLE IX. REGROUPING OF SEQUENCES BY CONTROLLING CONDITION OR PHENOMENON

Group	Controlling Condition or Phenomenon	Sequences
A	Secondary-side break, one affected SG	5.1, 6.1–6.9 7.1–7.8 8.1–8.6 9.25–9.32
B	Secondary-side break, three symmetrically affected SGs	7.12, 9.2–9.23 9.41–9.47
C	Secondary-side breaks with two affected SGs or three symmetrically affected SGs	5.4, 5.15 5.17–5.20 7.9–7.11 9.33–9.40
D	Reactor trip from full power, no primary- or secondary-side breaks	9.1, 9.49–9.55 13.1–13.6 14.1–14.2
E	MFW overfill	9.56
F	Primary-side breaks	1.1–1.4 2.1–2.4 3.1, 3.2, and 4.1
G	Primary-side breaks combined with symmetric secondary-side breaks	1.5–1.8 2.5–2.8 3.3
H	Primary-side breaks combined with asymmetric secondary-side breaks	1.9–1.12 2.9–2.11
I	Isolatable primary-side breaks	11.1–11.4 12.1–12.4
J	SG tube ruptures	10.1–10.5

MFW: main feedwater

Appendix VII

PTS EVALUATION OF LOVIISA NUCLEAR POWER PLANT

This appendix presents the integrated probabilistic PTS study that was carried out by Fortum (former Imatran Voima Oy) for the Loviisa WWER-440 plant in 1984–1986. The extensive PTS analysis was performed because it became apparent that the Finnish regulatory authority STUK would require a renewed Loviisa RPV brittle fracture analysis taking into account consequences of PTS type phenomena. It included a comprehensive process of identifying and selecting the overcooling transients, performing thermal hydraulic sequence analyses and probabilistic fracture mechanics calculations. A number of further plant modifications were made based on the results. The role of the probabilistic approach was to give an overview of the severity of all different PTS sequences, and give a quantitative estimate of the importance of the PTS issue in relation to the overall safety of the plant. The deterministic licensing calculations were made resulting from the integrated probabilistic PTS study. The integrated PTS study has been updated several times to account for plant modifications and RPV external cooling.

EVENT DESCRIPTION

The first step was development of the overcooling sequences. The work started by identifying the systems affecting overcooling transients, and by identifying the important operator actions associated with potential overcooling sequences. The transients selected as initiating events include those that either directly or through consequential failures lead to downcomer temperature decrease. The plant system response was determined for each initiator employing an event tree analysis. Event trees were established and quantified always when these were available. Operator actions associated with initiators were included. To reduce the number of sequences the screening frequency limit of 10^{-7}/reactor year was defined. The Loviisa training simulator was used extensively to give feedback on the process technology, operator actions and thermal-hydraulics.

The development of overcooling sequences resulted in definition of 21 transient classes, and the total number of selected sequences that had to be analyzed was 121. Thermal hydraulic analyses were performed for 55 sequences out of the selected 121 sequences. The selected sequence “Steam generator collector break (90 cm^2) starting from hot full-power conditions” is considered in the PTS analysis in this Appendix. It is assumed that the operator isolates the leaking steam generator after 20 minutes from the accident beginning.

The horizontal steam generators have two vertical primary collectors, the inlet and outlet collector, which are connected through horizontal tubes. A break in these collectors has been included in the scope of this PTS study. The upper limit of the break size has been estimated at 90 cm^2, which covers a ductile fracture of the collector and a break of the collector cover flange. In the Loviisa reactors, hot and cold leg nozzles are on different elevations. There are two loop seals in each of the six loops, and the pressurizer is connected to two hot legs. Reactor coolant pump suction takes place from the side and discharge is downward. There are gate valves in the hot and cold legs of each loop.

There are also some minor differences from a typical western reactor, including a narrower downcomer gap and no thermal shield. Each hexagonal fuel assembly is surrounded by a shroud. In the lower plenum, there is a perforated flow distributor plate to stabilize the coolant flow before it enters the core. High-pressure injection (HPI) water is injected into the RCP suction side and into the RCP discharge side in three loops. Two accumulators inject into the downcomer and two into the upper plenum through surge lines separately connected to the pressure vessel.

THERMAL HYDRAULIC ANALYSIS

The main task of the thermal hydraulic calculations was to provide the following parameters during the selected overcooling event for the fracture mechanics calculations:

- downcomer temperature field;
- coolant-to-wall heat transfer coefficients in the downcomer;
- primary circuit pressure.

Thermal hydraulic response of the plant to the selected overcooling sequence has to be analyzed using reliable and realistic calculation methods. The assumed transient time was two hours. Hence, the computer codes utilized had to be fast running in order to keep the computing time within reasonable limits. The demands of speed and reliability were satisfied using the advanced thermal hydraulic system code RELAP5/MOD2 (cycle 36.02), which was assessed against the plant transient data. The primary flow stagnation situations require special calculational tools.

In the case of zero or very low loop flow rates, the cold high-pressure injection water has a strong tendency to stratify in the cold leg bottom. When the cold stream enters the downcomer, a buoyant vertical plume or jet is formed. The plume temperature is of great importance for fracture mechanics calculations. When considering thermal stratification of single-phase liquid and plume formation, the basic difficulty is that they are not represented in the system code RELAP5/MOD2 utilized to simulate overcooling transients. Therefore, separate calculations for stagnant conditions were performed using the REMIX computer program, which was assessed against the experimental data.

A special attention should be given for the transients where the primary flow stagnation is possible. The decision of the stagnating flow conditions is more difficult since the used system code RELAP5/MOD2 tends to predict an oscillating single-phase natural circulation flow.

In the flow stagnation cases the role of RELAP5/MOD2 calculations is to estimate the initiation of the stagnation, and give the initial temperature conditions and HPI flow rates as boundary conditions for the REMIX calculations.

No probabilities are associated to the thermal-hydraulics, since it is not feasible to run statistically meaningful number of runs for a sequence. Instead, one tries to make realistical assumptions to obtain a best-estimate history of the sequence. Thus thermal hydraulic part of the probabilistic study differed significantly from the other parts of the PTS study, when sensitivities, uncertainties and conservatisms were estimated.

Traditionally only internal cooling of the downcomer has been accounted for. This is consistent with assuming the crack on the inner surface of the vessel wall. The inner surface cracks have been subject to a greater interest, since embrittlement is higher in that region.

RELAP5/MOD2 analysis

RELAP5/MOD2 is a PWR system transient analysis code which can be used for simulation of a wide variety of PWR system transients of interest with regard to the reactor safety. The code has a full nonequilibrium six equation two-fluid model. The code has been developed at the Idaho National Engineering Laboratory under the sponsorship of the Nuclear Regulatory Commission. The version used in this study is RELAP5/MOD2 Cycle 36.02.

RELAP5/MOD2 input model description

The input model included 191 volumes, which were connected to 203 junctions. There were 131 heat structures. The main features of this detailed model included a description of all major flow paths for both primary and secondary circuits. The six loops were described with three calculational loops. There was a single loop, a double loop (the two loops connected to the pressurizer), and a triple loop (the loops where HPI into the RCP discharge side took place). The model included a description of all four HPI pumps (using time-dependent volumes and junctions), all makeup pumps (three piston pumps and one high-capacity centrifugal pump), and all four accumulators. Also, the pressurizer heaters and spray systems, as well as the power-operated relief valve and two pressurizer safety valves, were modelled.

Special attention was paid to the pressurizer, loop seals, and the downcomer. These components were modelled using many hydraulic volumes since they were expected to have a significant influence on the results.

In the secondary side, steam generators and steam lines were modelled to the extent allowed by the computer (all available memory of CDC Cyber 180-840 was used). The model included a description of steam line isolation valves, secondary-side relief and safety valves, and turbine bypass valves. Main and emergency feedwater injections as well as turbines were modelled as boundary conditions.

RELAP5/MOD2 input model verification

The input model was first checked by a steady-state run. The comparison of achieved steady state with plant operational data showed good agreement. Next, input was checked against data from the plant startup natural circulation experiment with 2 % thermal power. Good agreement with measured data was also found in this case. The third validation run analyzed the stuck-open turbine bypass valve incident at Loviisa Unit 2 in 1981. This incident can be considered an overcooling transient since during the incident, coolant temperature in the downcomer decreased from 265 to 215°C in 15 minutes. The main results of the RELAP5/MOD2 calculation match the measured data qualitatively well, and quantitative agreement is also satisfactory.

Selected sequence analyzed using RELAP5/MOD2

The RELAP5/MOD2 runs for the selected sequence “Steam generator collector break (90 cm^2) starting from hot full-power conditions” give the initiation of the stagnation, the initial temperature conditions and HPI flow rates as boundary conditions for the REMIX calculations. The RELAP5/MOD2 results are illustrated in Figs 8, 9 and 10.

REMIX analysis

REMIX/NEWMIX code description

The REMIX/NEWMIX computer program was developed by Prof. T.G. Theofanous et al of Purdue University (currently of University of California, Santa Barbara) starting from the Regional Mixing Model developed by the same authors. This model provides a phenomenologically-based analytical description of the stratified flow and temperature fields resulting from HPI into the cold legs in the loop of a pressurized water reactor.

The REMIX version was initially developed for low Froude number top injections with $Fr_{HPI} < 3$. The computation proceeds at two levels: The global level establishing mean system response, referred to as “ambient” and the local level partitioning mass and energy into the cold and hot streams. At the global level of the computation, the whole system is assumed to be well mixed, and the computation proceeds from the initial conditions in time steps. The local computation provides a detailed picture of the flow and temperature distributions at arbitrarily selected times.

The heat transfer rate from the walls is computed from co-currently run transient conduction calculations at the global level of computation. Turbulent mixing in plume mixing regions (HPI plume in the cold leg and assumed planar plume in the downcomer) is calculated according to the k-ε-θ′ turbulence model predictions.

In the REMIX code only one cold leg is modeled. The downcomer and lower plenum volumes are partitioned equally among the loops.

The NEWMIX version has been developed for high Froude number top injections ($Fr_{HPI} > 10$). Forceful jets provide sufficient local mixing in the region associated with the injection stream to reach the maximum level of entrainment allowed by countercurrent limitations. The computational procedure of NEWMIX is similar to that of REMIX except slight differences.

Imatran Voima Oy carried first out an extensive experimental research program to investigate thermal mixing in the Loviisa specific geometry. In order to achieve a better prediction capability for the bottom injection of the HPI water, a fully empirical cold stream calculation technique was introduced into the REMIX/NEWMIX program. The thermal mixing experiments of Imatran Voima Oy were utilized to give the correlation for the hot stream flow rate and for the cold stream height in the loop. Based on these results, the REMIX/NEWMIX thermal mixing program was modified for the Loviisa analyses. This modified version is called REMIX-LOVIISA.

REMIX/NEWMIX code validation

The extensive validation of the REMIX/NEWMIX program against the data from different thermal and fluid mixing experiments (Creare 1/5-scale, Purdue 1/2-scale, Creare 1/2-scale, IVO 2/5-scale, HDR and UPTF full-scale tests) has been made.

Selected sequence analyzed using REMIX-LOVIISA

A single loop corresponding to the original system is formed and the REMIX-LOVIISA computer program is applied to a sixth of the reactor pressure vessel with a single loop. The heat transfer coefficient 3000 W/Km^2 is applied for the downcomer region. The temperature response of the Loviisa plant under stagnant loop conditions for the selected sequence “Steam generator collector break (90 cm^2) starting from hot full-power conditions” is shown in Fig.11. In the transient temperature responses “3737 mm (weld)” is the downcomer plume temperature in the circumferential weld area located at the core height at a distance of 3737 mm below the cold leg centre line and “cold stream” is the cold stream temperature in the cold leg. The resulting curve for the fracture mechanics calculation is given in Fig.12.

FRACTURE MECHANICS CALCULATION

Probabilistic OCA-P structural analysis

The probabilistic fracture mechanics calculations were carried out by applying probabilistic code OCA-P that was developed at the Oak Ridge National Laboratory under the sponsorship of the Nuclear Regulatory Commission.

Because of the long run times of this Monte Carlo based method, the selected 55 sequences were further grouped and stylized so that finally 26 different transient cases were calculated. The used calculation method OCA-P assumes uniform temperature and heat transfer field in the downcomer and an infinite 2D crack. Therefore the temperature and heat transfer nonuniformities were handled with limiting point value assumptions. As the heat transfer coefficient is so large in most cases that it does not affect the through-wall crack probability, it was assumed to have a constant value of 5000 W/Km^2 throughout all transients.

The probabilistic OCA-P fracture mechanics calculations for the selected sequence “Steam generator collector break (90 cm^2) starting from hot full-power conditions” give the conditional through-wall crack probability for this transient.

For the deterministic licensing calculations the selection of deterministic sequences for downcomer overcooling was made by looking for all the sequences resulting from the integrated PTS study with a higher than 10^{-8} through-wall crack frequency. After deterministic screening (e.g. elimination of those where excessive failures were assumed) and combination of the sequences, the sequence "Steam generator collector break (90 cm^2) starting from hot full-power conditions" was one of the selected transients. Deterministic structural integrity assessment is carried out with a detailed 3D fracture mechanics calculations using assumptions of nonuniform temperature and heat transfer fields in the downcomer during the PTS transient.

3D finite element structural analysis

Elastic-plastic brittle fracture calculations have been done with 3D finite element methods both by IVO International Ltd and Technical Research Centre of Finland (VTT). IVO uses the BERSAFE code with its pre- and post-processing modules, and VTT uses the ADINA and PATRAN codes. Thus, generating the element models and running the codes have been done with two independent calculation systems.

The main aim of IVO calculations has been to predict the crack initiation. For the crack initiation calculations IVO has applied various surface and subsurface crack sizes: 5 × 10; 13.5 × 30; 15 × 30 and 15 × 50 mm, and infinitely long cracks (360 º) of depth 5 mm. These cracks have been assumed to locate mostly in weld 5/6 (i.e. the circumferential weld on the lower core region). The orientation of these cracks has been assumed to be along the weld, i.e. circumferential cracks. Thus they are perpendicular to the downcomer plumes in the RPV wall.

Since the fracture mechanics calculations are made with the 3D methods, the nonuniformity effects are included. The heat transfer coefficient 5000 W/Km^2 is applied for the downcomer plume region where flow velocities are higher. Outside of the downcomer plume the heat transfer coefficient value 2000 W/Km^2 is applied. The thermal mixing experiments of Imatran Voima Oy support the choice. The values are also consistent with the values reduced from the various thermal mixing experiments.

In addition to the thermal and mechanical loading stresses, the calculations account for stresses caused by cladding residual stresses and weld seam stresses.

The used finite element models apply rotation and axial symmetry with rigid boundary conditions. The cracks are modelled with a very fine element structure.

The calculated J integrals are converted into the equal K_J values, and these values are compared with the material critical K_{IC} curves obtained with calculational rules and with the critical curves developed at VTT in Finland. The calculated J integral in the cladding layer was compared with the J-R curves of irradiated cladding. These curves were also developed at VTT. Crack initiation is not allowed during the calculated PTS cases.

In Fig.13 the K_J values are compared with the weld material K_{IC} curve for the selected sequence "Steam generator collector break (90 cm^2) starting from hot full-power conditions" with the assumption of the 15 x 50 mm circumferential surface crack. Node A is in the deepest point of the crack front, and Node B is in the interface of the cladding and the weld.

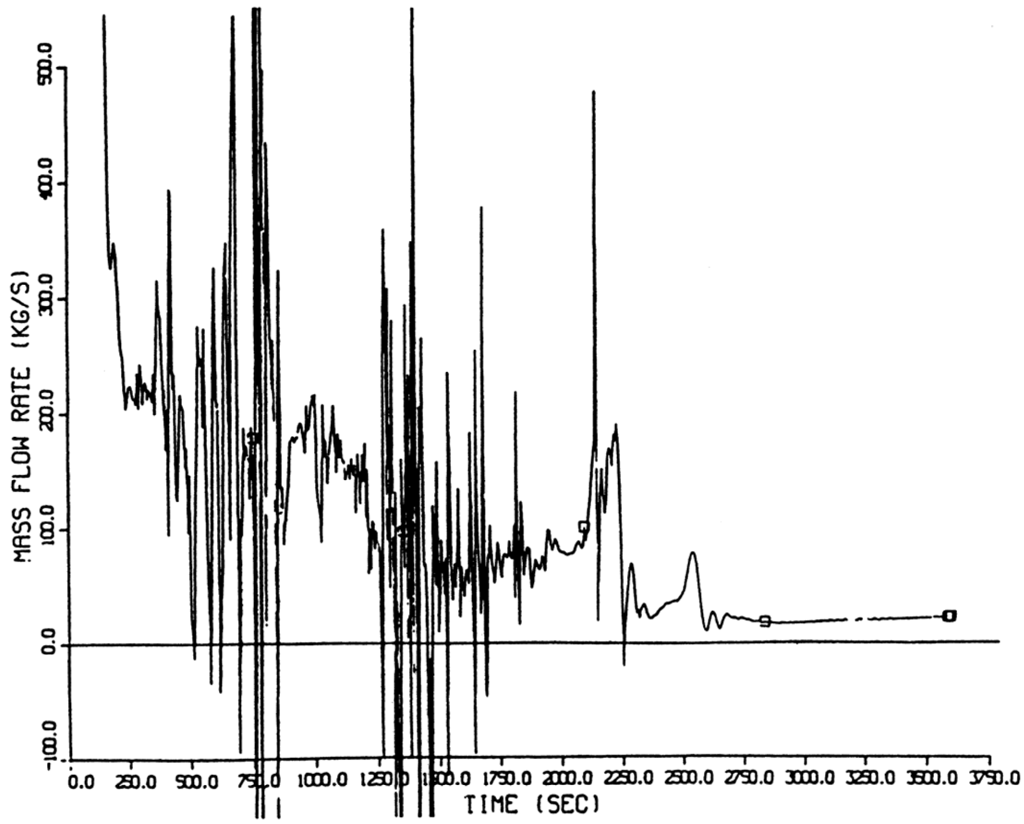

FIG. 8. Mass flow rate in the downcomer in RELA P5/MOD2 calculation for sequence "Steam generator collector break (90 cm^2) starting from hot full power conditions.

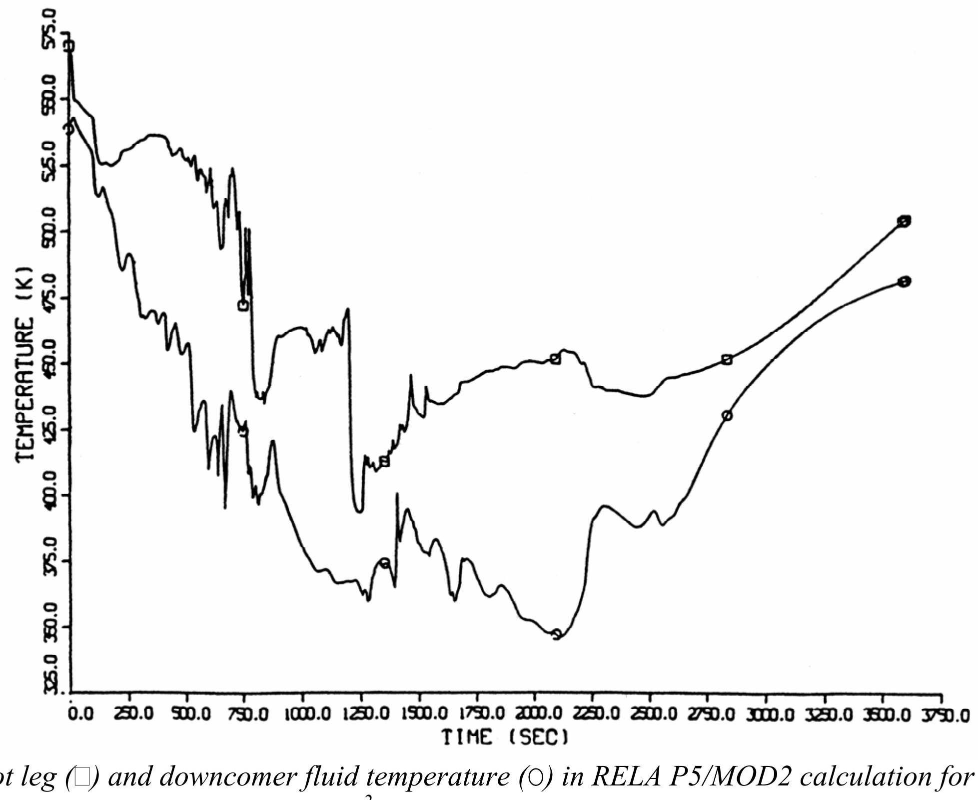

FIG. 9. Hot leg (□) and downcomer fluid temperature (○) in RELA P5/MOD2 calculation for sequence "Steam generator collector break (90 cm^2) starting from hot full power conditions".

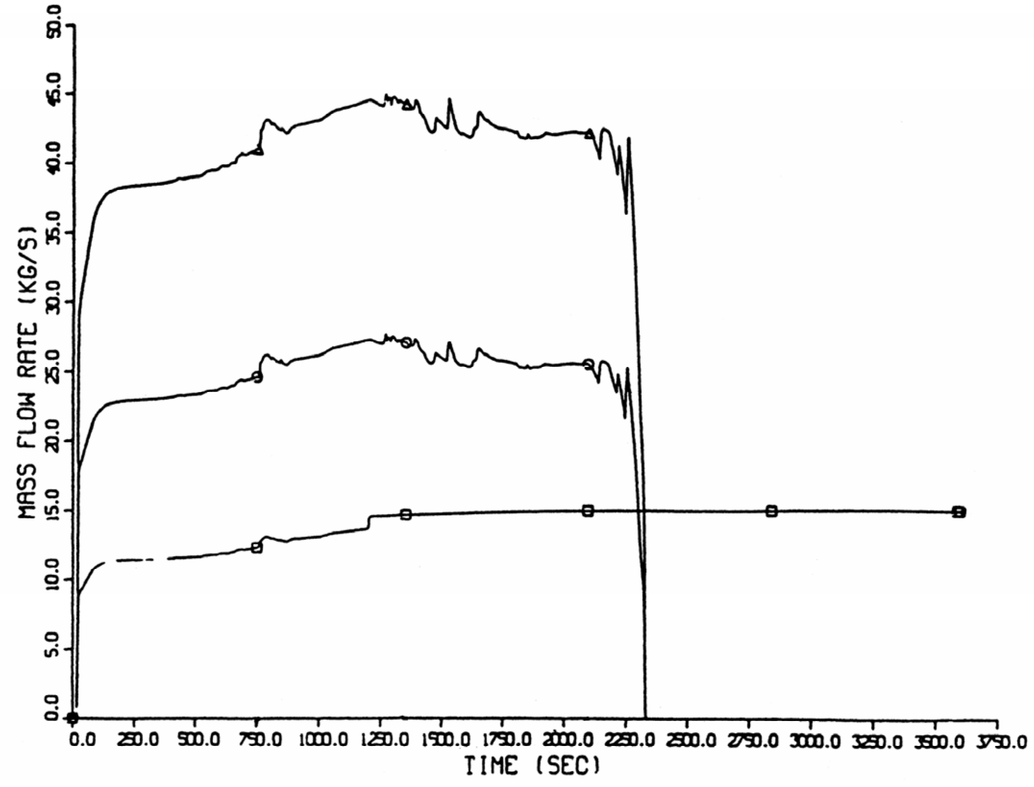

FIG. 10. HPI flow rates into the single (□), double (○) and triple (Δ) loop in RELA P5/MOD2 calculation for sequence "Steam generator collector break (90 cm²) starting from hot full power conditions".

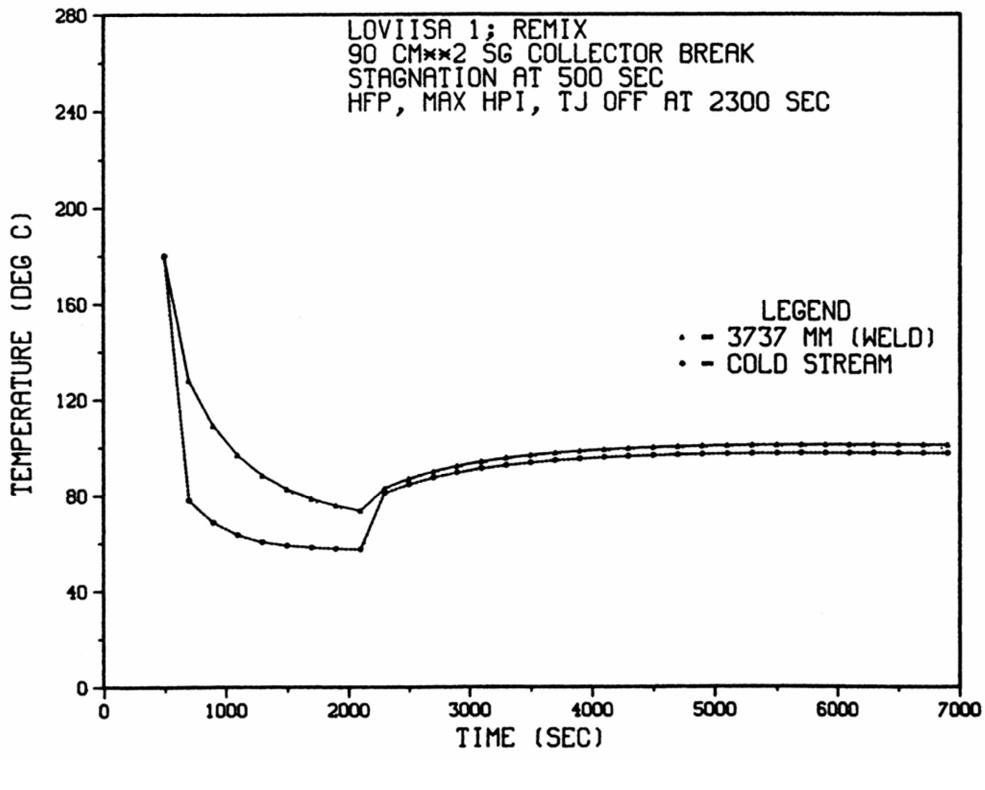

FIG. 11. REMIX calculation for sequence "Steam generator collector break (90 cm²) starting from hot full power conditions". TJ is high pressure injection.

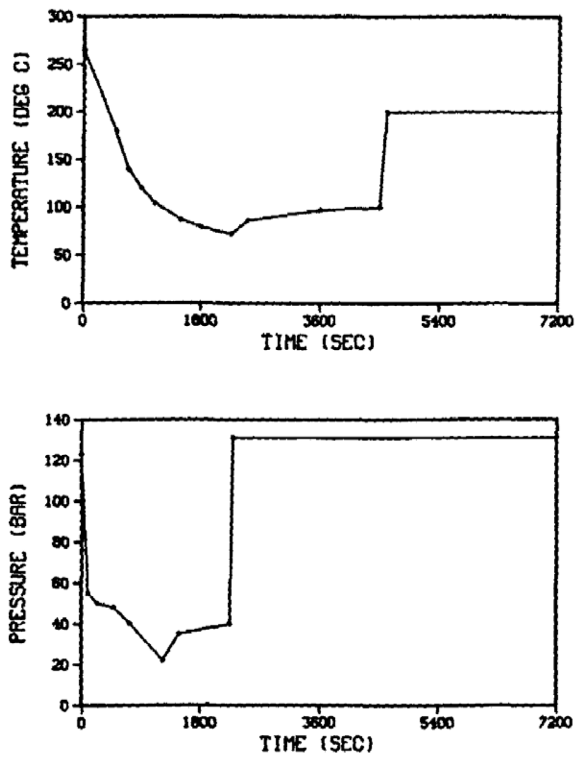

FIG. 12. Downcomer temperature and primary pressure for sequence "Steam generator collector break (90 cm^2) starting from hot full power conditions", 15 x 50 mm surface crack. T_k = 135°C.

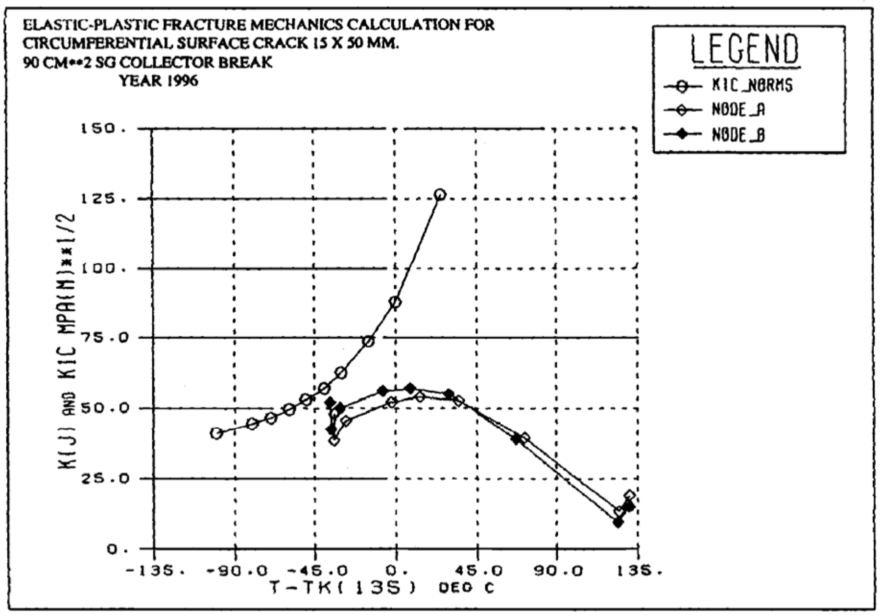

FIG. 13. Comparison of the K_j curve to the weld material K_{IC} curve for sequence "Steam generator collector break (90 cm^2) starting from hot full-power conditions", 15 x 50 mm surface crak. T_k = 135°C.

Appendix VIII

DETERMINATION OF FRACTURE MECHANICS PARAMETERS

One of the following two loading parameters has to be evaluated: either the elastic stress intensity factor K_I or the elastoplastic stress intensity factor K_J corresponding to an elastoplastic evaluation of J integral. In order to estimate these parameters, different refinements in the methodology based on FEM can be used depending on the different simplified assumptions:

– 1D, 2D or 3D models;
– intact or cracked models;
– elastic or elastoplastic approaches;
– with or without cladding and corresponding residual stresses;
– FEM or analytical models for stress field evaluation;
– stress intensity factor evaluated either analytically or directly from FEM using J-integral or energy release rate.

Three examples of usable methods follow:

1. Temperature distribution and elastic stress evaluation on a 1D uncracked model and K_I evaluation using influence functions for infinite or elliptical cracks (using mean, linear and parabolic part of the stress distribution through the thickness): it is a very simple engineering method to select the limiting transients but the thermal loadings have to be conservatively simplified (uniform on the RPV internal surface); a plastic zone correction factor (IRWIN type) has to be added to the elastic K_I evaluation to compare to the corresponding K_{IC}.

2. Temperature distribution and elastic-plastic stress evaluation on a 2D cracked model: it remains a simple engineering method that can take variation of heat exchange coefficient in one direction but the defect is conservatively assumed to be of infinite length and the method gives an upper bound of the plastic zone correction by comparison with the elastic stress evaluation.

3. Temperature distribution and elastoplastic stress evaluation on a 3D cracked model: it is more realistic model and a reference method but more complex to perform than the two previous ones.

All these methods are well adapted for uncladded vessel. For cladded vessels, specific limitations have to be considered:

- for underclad (or sub-surface or partly through the interface) cracks for which influence function are not available, method 1 is not usable without complementary development;
- the plastic zone correction (IRWIN type) can be non-conservative for large plasticity situation or for intersection point (clad/base metal) and consequently in these cases, method 1 is not sufficiently accurate.

Appendix IX

"Master Curve" application to RPV integrity assessment

RPV integrity assessment can be also performed using the "Master Curve" approach [34] In such a case, allowable stress intensity factor values are determined with the use of reference temperature T_0 (based on testing static fracture toughness of surveillance specimens and/or specimens from template cut from RPV wall) instead of critical brittle fracture temperature T_k (from Charpy V -notch impact specimens). Neutron fluence of these specimens should be close to the analysed state of the RPV; in this case no initial values of any transition temperature (neither T_{k0} nor T_0^{ini}) of tested material are necessary. Transition temperature T_0 for the analysed state of the RPV is determined using single or multiple temperature method in accordance with the ASTM standard E 1921 [28].

Allowable stress intensity factors are then given as a 5% lower tolerance bound by the equation:

$$[K_{IC}]_{25mm} = 25.2 + 36.6 \cdot \exp [0.019 (T-T_0)] \qquad (1)$$

which is valid for the specimen thickness/crack length equal to 25 mm.

For cases when crack front length B_i is larger than 25 mm, the following re-evaluation of the aforementioned dependence is recommended (B_i is in mm):

$$[K_{IC}]_{Bi} = (25/B_i)^{1/4} \cdot ([K_{IC}]_{25mm} - K_{min}) + K_{min} \qquad (2)$$

where K_{min} is a minimum value of fracture toughness of the material and is usually taken equal to 20 $MPa.m^{0.5}$.

Crack front length B_i of postulated defects can be calculated using the following equations:

- semielliptical surface crack:

$$B_i = 2c \cdot [1 + 4.6(a/2c)^{1.65}]^{0.5} \qquad (3)$$

- elliptical subsurface/underclad crack:

$$B_i = 4c \cdot [1 + 4.6(a/2c)^{1.65}]^{0.5} \qquad (4)$$

This correction for a postulated crack front length is performed only for relations in the range of 25 mm $\leq B_i \leq$ 150 mm. For values $B_i >$ 150 mm, the value of B_i = 150 mm is taken, for values of $B_i <$ 25 mm the value of B_i = 25 mm is taken and no correction to the equation (1) is required.

The RPV integrity is assured if the following equation is fulfilled:

$$K_I (T, a) < [K_{IC} (T)]_{Bi} \qquad (5)$$

for all values of K_I larger than 0.9 K_{max} (see Section 7.3).

In the case of a nonuniform distribution of K_I and K_{IC} along the crack front of the postulated defect, it is sufficient that the following relation is fulfilled [32] (instead of (5)):

$$\frac{1}{25}\int_{B_i}\frac{(K_I(b)-K_{\min})^4}{([K_{IC}(b)]_{25mm}-K_{\min})^4}db<1 \qquad (6)$$

K_{min} is usually taken equal to 20 MPa.m.$^{0.5}$.

Those parts of the crack front where K_I is below 0.9 K_i^{max} and is continuously (monotonically) decreasing (see Section 7.3) should be excluded from the integration. Safety margins according to section 7 of this report should be used in such analysis.

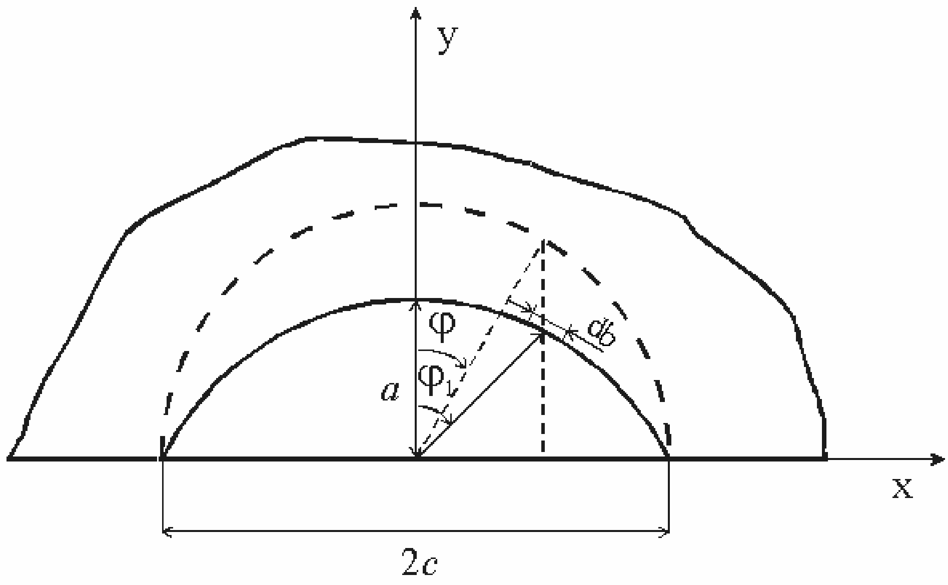

Fig. 14. Integration Path for Formula (6) Evaluation.

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ABBREVIATIONS

a, c	minor resp. major semiaxes of the postulated defect
a/c, 2a/c	postulated semielliptical resp.elliptical defect aspect ratio
A_F^T	irradiation embrittlement factor at irradiation temperature T
B_i	crack front length
BRU-A	steam generator safety valve, atmospheric relief
BRU-K	steam generator safety valve, turbine bypass
CFD	computational fluid dynamics
c_p	specific heat
DBA	design basis accident
E	Young's modulus
ECCS	emergency core cooling system
FEM	finite element method
F_n	neutron fluence
HAZ	heat affected zone
HPI	high pressure injection
HTC	heat transfer coefficient
ISI	in-service inspection
K_I	stress intensity factor
K_{IC}	fracture toughness
$[K_{IC}]$	allowable stress intensity factor
LOCA	loss of coolant accident
LOFA	loss of flow accident
MCP	main circulation pump
MGV	main gate valve
MSLB	main steam line break
n_a	safety factor on crack size
NDT	non-destructive examination
n_K	safety factor on stress intensity
NPP	nuclear power plant
NUSS	IAEA Nuclear Safety Standards
OPB	Russian safety standards
PSA	probabilistic safety assessment
PTS	pressurized thermal shock
RCS	reactor coolant system
RPV	reactor pressure vessel
s	RPV wall thickness
T	temperature
T_k	critical brittle fracture temperature
T_k^a	maximum allowable critical brittle fracture temperature
T_{ko}	initial value of critical brittle fracture temperature
T_R	reference temperature
ΔT	safety factor on fracture toughness temperature
ΔT_F	shift in T_k due to irradiation
$\Delta T_{F, res}$	residual shift in T_k after annealing
T_{irr}	irradiation temperature
ΔT_T	shift in T_k due to thermal ageing
ΔT_n	shift in T_k due to fatigue damage
T_0	transition temperature
T_0^{ini}	initial value of transition temperature
WPS	warm prestress

λ	thermal conductivity
α	coefficient of linear thermal expansion
ρ	density
ν	Poisson's ratio
σ	standard deviation

CONTRIBUTORS TO DRAFTING AND REVIEW

Brumovský, M.	Nuclear Research Institute, Czech Republic
Čepček, Š.	Nuclear Regulatory Authority, Slovakia
Dragunov, Y.G.	OKB Gidropress, Russian Federation
Elter, J.	Paks Nuclear Power Plant, Hungary
Faidy, C.	Electricité de France, France
Havel, R.	International Atomic Energy Agency
Hrazsky, M.	VUJE, Slovakia
Kovbassenko, S.	Goscomatom, Ukraine
Kovyrshin, V.G.	Ministry for Environmental Protection and Nuclear Safety, Ukraine
Král, P.	Nuclear Research Institute, Czech Republic
Liška, P.	VUJE, Slovakia
Matejovič, P.	IVS Trnava, Slovakia
Miannay, D.	Institute de Protection et de Sûreté Nucléaire, France
Miteva, R.	Committee on the Peaceful Use of Atomic Energy, Bulgaria
Parduba, Z.	Nuclear Research Institute, Czech Republic
Petőfi, G.	Hungarian Atomic Energy Authority, Hungary
Piminov, V.	OKB Gidropress, Russian Federation
Pištora, V.	Nuclear Research Institute, Czech Republic
Rantala, R.	Finnish Centre for Radiation and Nuclear Safety, Finland
Rieg, C.Y.	JRC Petten, Netherlands
Sievers, J.	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Germany
Tendera, P.	State Office for Nuclear Safety, Czech Republic
Tuomisto, H.	FORTUM, Finland
Wang, L.M.	International Atomic Energy Agency

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