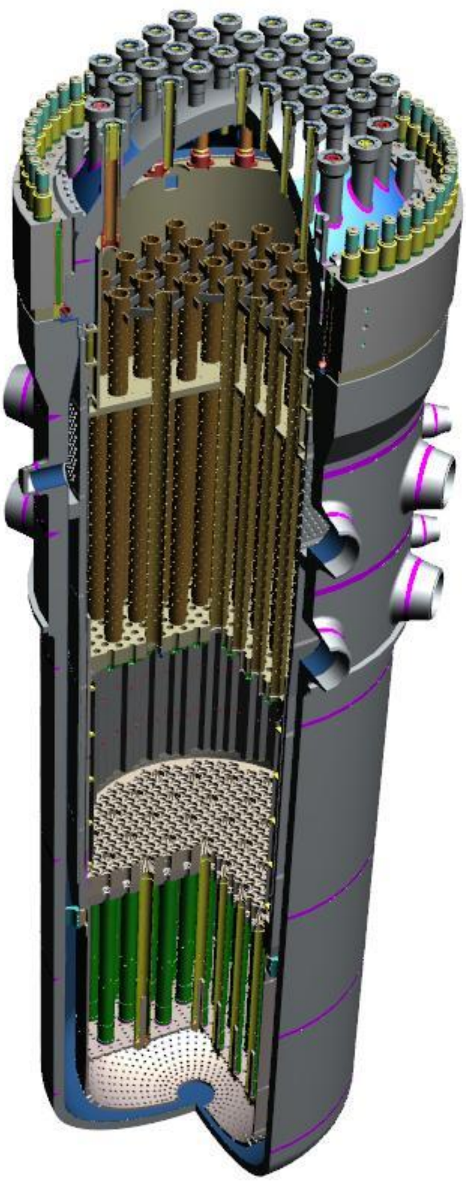
VERLIFE Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plants



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Contents

Abstract

The VERLIFE Guidelines are validated procedures for the calculation of the residual lifetime and integrity of components and piping of WWER–type reactors. They cover procedures for the calculation of the resistance against fast fracture, fatigue damage, corrosion-mechanical damage and acceptability of flaws found during in-service inspection. They also contain procedures for integrity evaluation of piping, in-service inspection qualifications, risk-informed in-service inspection as well as integrity and lifetime of reactor vessel internals and components supports. Numerous appendicies on related specific topics like e.g. degradation assessment of material properties by radiation damage and thermal ageing, thermal hydraulic regimes for pressurized thermal shock evaluations, etc. are part of the guidelines.

VERLIFE Guidelines are mainly based on former Soviet/Russian Rules and Codes for design and manufacturing of components and piping of WWER-type reactors. But they also incorporate operational experience of countries with running WWER-type reactors (i.e. Bulgaria, Czech Republic, Finland, Slovakia, Ukraine, …) and to some extent approaches used in codes and rules of pressurised water reactors (PWRs) to be consistent with current PWR codes and rules as much as possible and thus provide modern, applicable procedures for component integrity and residual lifetime assessment of WWER-type plant components and piping.

The VERLIFE Guidelines, in full or in parts, may be used for the preparation of official reports to support periodic safety review, licensing and plant life management programmes of WWER-type reactors with the approval of the national nuclear regulatory authority.

This publication contains the main chapters and Appendices I – XVIII of the VERLIFE Guidelines. Appendices A-F are published as a separate JRC report (JRC 138451).

Foreword

The VERLIFE Guidelines are the result of a collaborative effort over 15 years involving 68 experts in the field from 9 different countries with operating WWER-type reactors. Their efforts are highly appreciated and without their contribution these guidelines would have not been possible.

The authors also acknowledge all the efforts of the International Atomic Energy Agency (IAEA), specifically Dr. Ki-Sig Kang, former technical head of plant life management / long-term operation of the Nuclear Power Engineering Section within the IAEA Division of Nuclear Power, and his team for having organised 11 expert meetings to substantially progress and almost complete the VERLIFE Guidelines. Unfortunately, purely due to editorial reasons the IAEA could not publish the guidelines in the end.

Most appreciation goes to Dr. Milan Brumovsky, distinguished expert for integrity and lifetime assessment of reactor components having worked for 50+ years in the field. He was the main driver for issuing the VERLIFE Guidelines and followed their drafting all the years. Unfortunately, Dr. Brumovsky passed away in July 2023 and thus could not experience the VERLIFE Guidelines finally being published. However, his name will always be linked to the VERLIFE Guidelines.

The VERLIFE Guidelines are validated procedures for the calculation of the residual lifetime and integrity of components and piping of WWER–type reactors. They are supposed to be a living document and are revised when needed.

1 Introduction

1.1. Background

Lifetime assessment of individual components and piping in nuclear power plants (NPP) is a mandatory part of every Periodic Safety Review as well as it is necessary for component/ plant life management and potential plant life extension. In the same time, such assessment is also necessary for safe operation of components in NPPs. Till the Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plants (VERLIFE) preparation, no legal procedures or standard guidelines existed for lifetime/integrity assessment of components and piping in operating water cooled and water moderated energy reactor (WWER). Former Soviet rules and standards had been prepared and approved only for design and manufacturing stage of NPPs and those rules/standards need some modifications and extensions to be usable also for operating nuclear power plant components. Approaches used in WWER codes and standards are in some parts different than they are applied in pressurized water reactor (PWR) ones, thus a comparison of lifetime assessment using these two types of codes could be different and non-comparable.

1.2. Objectives

The guideline is intended to address all relevant aspects related to application of procedure and to introduce a collective guidance and best practices. The guideline will reflect the following focal points:

— Elements of lifetime evaluation and integrity assessment of major components and piping in WWER reactors, such as, reactor pressure vessel (RPV), primary piping, reactor vessel internals, primary piping supports etc.

— VERLIFE procedure implementation in the countries operating WWER reactors.

— Compile the operational experience in application of VERLIFE procedure.

These "Guidelines for Integrity and Lifetime Assessment of Components and Piping in Nuclear Power Plants with WWER operating Reactors - VERLIFE ("Guidelines") provide a methodology for lifetime and integrity assessment of components and piping in WWER operation reactors with focusing on failure caused by non-ductile and ductile fracture, fatigue, mechanical corrosion and corrosion-erosion damage under operational conditions. Additionally, it contains also procedures for integrity evaluation of piping, in-service inspection system qualifications as well as integrity and lifetime of reactor vessel internals and components supports.

These "Guidelines" are mainly based on former Soviet/Russian Rules and Codes applied during design and manufacturing of components and piping of WWER type reactors. It also incorporates some approaches used in PWR codes and rules in order to be as consistent with PWR codes and rules as possible.

These "Guidelines" are not intended to replace the national legislative documents. However, this document suggests modern, applicable procedures for component integrity assessment and remaining lifetime evaluation for WWER type plants. This "Guidelines" or parts of this "Guidelines" can be used for development of official reports (e.g., Periodic Safety Review Report, Licensing, Life Management Acceptance etc.) only with the acceptance of the cognizant national regulatory authority.

1.3. End users

These "Guidelines" can also be used at:

— Evaluation of indications found during in-service inspection in components and piping;

— Elaboration of Periodic Safety Review Reports during NPP operation including assessment of residual lifetime of components and piping;

— Preparation or modification of plant component for life management.

1.4. History

First version of the VERLIFE procedure was prepared in the Concerted Action within the 5th European Union Framework Programme (EU 5th FP) by the Nuclear Research Institute Rez (NRI), Czech Republic. This version was based on the international experts' meetings to share the status of guidelines for lifetime assessment of individual components and piping in WWER NPPs in members´ countries. The project started on 1 October 2001 and finished on 30 September 2003.

The Consortium was co-ordinated in such a way to include:

— Group of experts from technical support organizations that are incorporated in WWER component lifetime assessment in Finland, Czech Republic, Slovak Republic and Hungary – FORTUM Nuclear Services Ltd. in Finland, Nuclear Research Institute Rez and Institute of Applied Mechanics in Czech Republic, VUJE in Slovak Republic and AEKI Atomic Energy Research Centre in Hungary, and Institute of Metal Science in Bulgaria.

— Experts from nuclear regulatory bodies that are connected with evaluation of such assessments and/or their acceptance - State Office of Nuclear Safety of Czech Republic, Nuclear Regulatory Office of Slovak Republic.

— Specialists from nuclear power plants that are responsible for component lifetime assessment and/or plant life management – FORTUM Nuclear Services Ltd. for Loviisa NPPs in Finland, CEZ a.s. for Dukovany NPPs and for Temelin NPPs in Czech Republic, Slovenske elektrarne a.s. for Jaslovske Bohunice NPPs and Mochovce NPPs in Slovak Republic, and Paks NPPs through AEKI.

— Used experience from components design, stress analysis, lifetime evaluation and manufacturing experience – SKODA JS a.s. in Czech Republic (main manufacturer for WWER components for Czech Republic, Slovak Republic and Hungary, also former Germany Democratic Republic, Poland and Bulgaria).

The practical work was carried out, in principle, in meetings of task groups. A common kick-off meeting of all experts held to review the current status. In the kick-off meeting the structure of the "Guidelines" and expected results were proposed, discussed and agreed.

In preparation of the "Guidelines" the following principles and inputs were agreed:

— WWER components were designed and manufactured in accordance with former Soviet rules and standards;

— Approaches applied in PWR components integrity and lifetime evaluation;

— Last developments in fracture mechanics and their application to component integrity.

The structure of the "Guidelines" covers systems, structures and components (SSCs) all important parts of lifetime and integrity assessment as they are required for periodic safety reviews and plant life management programme. These "Guidelines" have been prepared for pressurized components of primary circuit of WWER-440 and WWER-1000 units, even though it shall be also used for safety related components of other circuits, too.

In 2005 a new project within the EU 6th FP was opened – COVERS –"WWER Safety Research" that was also co-ordinated by the Nuclear Research Institute (NRI) Rez, Czech Republic. In this project, work package 4 deals with the upgrading and updating of the VERLIFE procedure to assure that the experience obtained and also new developments will be appropriately included into the new version. Experts from nine member states took part in this project, i.e., not only from WWER operating countries – Czech Republic, Slovak Republic, Hungary, Finland, Spain, Netherlands, Germany, Russian, and Ukraine as well as from EC-JRC in Petten, NL and ISTC.

Within this project, some improvements and upgrading/updating have been prepared and discussed in regular technical meetings twice a year.

New version of the VERLIFE guidelines was finalized and accepted by the participants in March 2008, after that standard was accepted as Czech and Slovak Codes; partially acceptance was obtained in Hungary, Finland, Ukraine and China. Thus, this upgraded/updated "Guidelines" was in good agreement with the state-of-the-art of the knowledge in the field.

To assure that VERLIFE guidelines will remain a living document and that it will get more advanced level and more disseminated application, a new 3-year IAEA project was started in 2009 and first kick-off meeting was organized on 11–13 March 2009. Such project incorporates experts not only from European Union countries but also from other IAEA member states. Moreover, this international project was performed in a close co-operation with the "pilot project" - VERLIFE extension - approved within the 6th Framework Programme project of the European Union "NULIFE – Plant Life Management of NPPs". Further meetings of experts were organized during 2009–2011 with the final approval meeting on 28–30 June 2011.

Within this project, upgrading/updating of the VERLIFE guidelines was prepared together with the extension by some additional procedures for evaluation of main components integrity during operation not included in the design codes.

Preparation of the VERLIFE guidelines has been done with active co-operation of experts from WWER operating countries – Czech Republic, Slovak Republic, Hungary, Bulgaria, Finland, Ukraine, Russian Federation and also within the NULIFE "pilot project" including experts from PWR operating countries – Germany, France, Spain, UK, Sweden. In preparation of the final version, experts from China also took place in the programme.

1.5. Structure of Guidelines

Chapter 2 describes the field of application of these Guidelines - for assessment of integrity and lifetime of components and piping in WWER NPPs during operation with application to Periodic Safety Reports, and for a definition of conditions for further reactor operation beyond the design life. Then, list of main abbreviations and definitions is given.

Chapter 3 defines general requirements for calculation of residual lifetime. This calculation shall be performed taking into account main damaging mechanisms, mainly calculation of the resistance against fast fracture, against fatigue damage, corrosion-mechanical damage and acceptability of flaws found during in-service inspection. Then, content of required documentation is described as well as requirements for a system of quality assurance of calculations.

Chapter 4 describes general approach for calculation of residual lifetime and integrity of components during operation. Four types of calculations defined in Chapter 3 are supplemented by calculations of components and piping integrity during operation - leak-before-break concept, no-break zone concept, integrity of reactor internals, in-service qualification, risk-informed in-service inspection and integrity of supports.

Chapter 5 describes in detail the calculation of component resistance against fast fracture which is based either on reference temperature T_0 or critical temperature of brittleness T_k concepts. Definition of postulated defects and construction of design fracture toughness for different materials is given. Procedure to include residual stresses and warm pre-stress effect is described in detail. Apart from deterministic calculations, requirements for calculation of RPV failure probability are also included.

Chapter 6 deals with calculation of residual lifetime of components as a result of fatigue damage. Criteria for full and simple calculation are given.

Chapter 7 shows the procedure for assessment of the effect of corrosion-mechanical damage on initiation of cracks and definition of hypothetical initial crack.

Chapter 8 describes the procedure for assessment of acceptability of flaws found during in-service inspection.

Chapter 9 finally summarizes all previous calculations to obtain final residual lifetime of components.

Appendix I gives the structure of the report assessing residual lifetime of the component.

Appendix II gives the procedure for determination of neutron fluence in reactor pressure vessels and reactor vessel internals, both by calculation and measurements.

Appendix III describes the procedure for assessment of degradation of properties of materials by radiation damage and thermal ageing, either by prediction formulae or using results from surveillance specimen tests. Evaluation of the effect of annealing and further re-embrittlement of RPV materials is also given.

Appendix IV shows the methods for determination of stress intensity factor K_I of postulated or experimentally found defects.

Appendix V gives a method for determination of reference/design fracture toughness curve using "Master curve" approach for new materials.

Appendix VI gives requirements for selection and calculation of thermal hydraulic regimes for pressurized thermal shock evaluations.

Appendix VII describes the procedure for calculation of fatigue damage in components caused by real operating loading.

Appendix VIII summarizes requirements for temperature measurements in components and piping to be used for further fatigue calculations by real operating loading.

Appendix IX shows the procedure for assessment of corrosion-mechanical damage of materials - for initiation and defect propagation.

Appendix X describes the method for schematization of flaws found during in-service inspection into the crack form suitable for fracture mechanics calculation.

Appendix XI gives the tables of allowable sizes of indication found during in-service inspection in reactor pressure vessels, steam generators, pressurizes and piping in WWER-440 and WWER-1000 type reactors.

Appendix XII describes the procedure for evaluation of flaw acceptability in components for cases when flaws are larger than those allowed by Appendix XI.

Appendix XIII describes the procedure for evaluation of flaw acceptability in austenitic piping for cases when flaws are larger than those allowed by Appendix XI.

Appendix XIV describes the procedure for evaluation of flaw acceptability in ferritic steel piping for cases when flaws are larger than those allowed by Appendix XI.

Appendix XV summarizes unified material properties to be used in calculations of temperature and stress fields for evaluation of RPV resistance against fast fracture.

Appendix XVI describes in detail the procedure for probabilistic evaluation of RPV resistance against fast fracture - RPV failure probability.

Appendix XVII deals with flow-accelerated corrosion in piping of WWER NPPs - requirements for monitoring, evaluation of results and prediction of corrosion damage.

Appendix XVIII gives recommendations for taking into account water environment into calculation of fatigue damage in WWER components.

Appendix A describes in detail the procedure of application of leak-before-break concept to the selected high-energy piping in WWER type NPPs including requirements for calculation and the procedure itself.

Appendix B gives the bases for assessment of no-break zone to the piping of WWER NPPs.

Appendix C describes the procedure for integrity and lifetime evaluation of reactor vessel internals. This procedure includes assessment of fatigue crack nucleation, IASCC crack nucleation, limit of embrittlement, crack propagation and changes in component geometrical sizes.

Appendix D provides an overview of risk-informed in-service inspection and basic requirements for its application to WWER type piping.

Appendix E describes the procedure and requirements for non-destructive system qualification of In-Service systems to be used in WWER components.

Appendix F contains requirements for structure design, materials, operation, lifetime and life extension of supports and hangers for elements of WWER NPPs.

2 General statements, definitions, abbreviations

2.1. Field of application

These "Guidelines" can be used for evaluation of integrity and residual lifetime of components and piping of NPPs with operating WWER reactors designed and manufactured in accordance with former Soviet/Russian Rules and Codes

These "Guidelines" can be used for evaluation of integrity and residual lifetime of replacement components and piping of NPPs with operating WWER reactors manufactured in accordance with original design specifications.

These "Guidelines" can be used for development of Periodic Safety Reports (or similar type of documentation) to demonstrate operational safety and reliability of components and piping during reactor operation.

These "Guidelines" can be used for a definition of conditions for further reactor operation within or beyond the component or piping design lifetime/license validity.

These "Guidelines" are based on a philosophy of operation lifetime and integrity evaluation similar to that used worldwide in codes and standards for PWR type reactors, but it does not exclude further use of results of current research and developments. Thus, these "Guidelines" are harmonised with PWR Codes and rules as much as possible, taking into account original Soviet rules for design, manufacturing and inspection.

These "Guidelines" can be applied to technological parts of NPPs with WWER reactors:

— Metallic parts of pressure boundary components of safety related systems.

— Metallic parts of containments.

Figures and schemes in these "Guidelines" represent only the generic conceptual design of a component and are not intended to provide actual design details.

2.2. List of main symbols and abbreviations

a Depth (minor semi-axis) of a postulated defect, [m]

c Half length (major semi-axis) of a postulated defect, [m]

a_{calc} Depth of maximum postulated defect, [m]

a_{hyp} Depth of hypothetical starting crack, [m]

a_{arrest} Depth of finally arrested crack, [m]

E Young modulus, [MPa]

ν Poisson ratio

G Energy release rate, [J.m^{-2}]

D Fatigue usage factor

J_{1mm} Value of J-integral corresponding 1 mm ductile tearing, [kJm^{-2}]

K_I Stress intensity factor, [MPa.m$^{0.5}$]

K_{IC} Static fracture toughness under plane strain conditions, [MPa.m$^{0.5}$]

K_{JC} Static fracture toughness, [MPa.m$^{0.5}$]

$[K_{IC}]_i$ Allowable value of stress intensity factor, [MPa.m$^{0.5}$]

$[K_{IA}]_i$ Allowable value of stress intensity factor for crack arrest, [MPa.m$^{0.5}$]

n_k Safety factor

ΔT Temperature safety factor, [°C]

s Thickness of component wall, [m]

s_{cl} Thickness of cladding, [m]

p Pressure, [MPa]

T Temperature, [°C]

T_t Material transition temperature, [°C]

T_k Material critical temperature of brittleness, [°C]

T_0 Material reference temperature, [°C]

RT_k Material reference temperature for integrity evaluation based on T_k approach, [°C]

RT_0 Material reference temperature for integrity evaluation based on T_0 approach, [°C]

$[T_h]$ Allowable hydrotest temperature, [°C]

$[T_t]_j$ Maximum allowable transition temperature for regime j, [°C]

$[T_t]$ Maximum allowable transition temperature, [°C]

$[T_A]_j$ Maximum allowable transition temperature for crack arrest for regime j, [°C]

$[T_A]$ Maximum allowable transition temperature for crack arrest, [°C]

AOT Anticipated operational transients

EC Emergency conditions

HT Hydrotest or pressure test

I&C Instrumentation and control system (for technological data collection (temperature and pressure of the media, velocity and volume of the flowing media, amounts of chemical admixtures, steam humidity etc.) measured by sensors in accordance with the plant design)

ISI In-service inspection

MDS Monitoring and diagnostic system (for collection of supplementary data (temperature of the metal, stress and strain, frequency etc.) necessary for evaluation of the running damage of the material of the component, piping and their supports and for determination of their residual lifetime)

NOC Normal operating conditions

NPP Nuclear power plant

QI Quality instructions

QM Quality manual

QP Quality procedures

RI-ISI Risk-informed in-service inspection

RPV Reactor pressure vessel

SM Special measurements (for temporary collection of data of the same character as the data collected by the monitoring and diagnostic system. The special measurements are usually used to obtain supplemental information on the strain in areas without permanently located MDS sensors)

WWER Water cooled and water moderated energy reactor (PWR) of Russian design

Other symbols and abbreviations specific to each chapter are defined and used in the separate chapters. The appendices include their own lists of symbols and abbreviations if necessary.

2.3. General terms

Design Specification

Design Specification is a set of documents issued by the developer and specifying in details design of components, material requirements, quality control, and requirements for service conditions etc. to safe operation and to ensure that the design conforms to the demands of the Controlling and Regulatory Authorities with legal authority for the nuclear power plant.

Limit

Limit represents the allowable value of a mechanical or physical quantity of the material, media or the value of load. To assure the required level of safety and compliance with the nuclear safety criteria issued by the State supervising bodies, the limit must not be exceeded.

Operability of the component

Operability of the component means the ability of the component to function as designed and should be distinguished from the issue of integrity.

Authorised body

Authorised bod (institution, company) is the body holding the Authorisation of the professional qualification issued by relevant national organisation and/or the body employing workers who hold the Certificate of the professional qualification, in compliance with the relevant national legislation.

Certified/qualified computational code

Certified/qualified computational code is the code awarded the Certificate of the Authorised body (as defined in 2.3.4) for its use by the competent organisation, after verification its quality and function, by the appropriate commission of the State Regulatory Authority.

Holder of the authorisation

Holder of the authorisation is the natural or legal person in sense of national laws.

2.4. Terms related to assessment of residual lifetime

The terms used in these "Guidelines" for treating assessment or residual lifetime are defined in the following sections.

Reduced stress (stress intensity)

Reduced stress (stress intensity) is defined as two times the maximum shear stress. In another way, the reduced stress is the difference between the maximum principal stress (algebraic value) and the minimum principal stress (algebraic value) in the given point. The tensile stresses are to be taken as positive, the compressing stresses as negative.

This definition of the reduced stress has no significance for the definition of the reduced stress in the area of fracture mechanics.

Large construction discontinuity (change)

A large construction discontinuity is a change in geometry or material, which affects the stress or strain distribution through the entire wall thickness of the component loaded by the inner or outer pressure or the temperature field. Examples of areas where large construction discontinuities may exist include: the bottom head – shell wall interface; the shell flange; couplings with the shell; and transitions between shells of different radius or thickness.

Local construction discontinuity (change)

A local construction discontinuity is a geometry or material change that influences the stress or strain distribution in a small part of the wall thickness. The stress distribution connected with the local change causes only the localised deformation or reshape and has no significant influence on the deformation discontinuities of the shell. Examples include connections of pipes of small radius to large pipe or component, notches, small clamps, and welds with penetrations.

Normal stress

Normal stress is the stress component perpendicular to the given plane. The normal stress through thickness of the component is usually not uniform and is considered to consist of three parts: membrane stress, bending stress and peak stress. In some cases, also peak stress is considered.

Shear stress

Shear stress is the stress component parallel to the given cross-section area.

Membrane stress

Membrane stress is the component of the normal stress, which is uniform and equal to the mean value of stress through the thickness of the given section.

Bending stress

Bending stress is the variable component of the normal stress described in this Section. Bending stress is assumed to vary linearly through the thickness of the section under consideration.

Mechanical stress

Mechanical stresses are stresses initiated by pressure loads such as inner and outer pressure, inertial load, seismic impacts or gravitational effects. The magnitude of such loads will not change due to deformation.

Total stress

Total stress is the sum of stresses of all categories, taking into account also the stress concentration.

Operation cycle

Operation cycle is defined as initiation of new conditions, their progress and return to the conditions prevailing at the beginning of the cycle.

Stress cycle

Stress cycle is the state in which the stress varies from its initial values through its algebraically maximum and minimum values and returns again to its initial value. A simple operation cycle can be formed by several stress cycles.

Free shift

Free shift represents the relative movements between the rigid bound and the connected wire or piping, in the case that those elements would be separated and permitted to displace.

Deformation (strain)

Deformation of the part of the component represents the change of its form and sizes. Strain means relative change in shape or size implies that it is dimensionless and has no units.

Operational lifetime of the component

Operational lifetime of the component is the total period of operation, during which all prescribed limits are fulfilled in accordance with Section 2.3 (see the definition of "Limit") and operability of the component is ensured.

Design lifetime of the component

Design lifetime of the component is the period given by design of the component (nuclear power plant). All conditions for lifetime of the component based on Design specification of the component and on ensured properties of materials of the component, have to be fulfilled.

Residual lifetime of the component

Residual lifetime of the component is the period from executed assessment until the end of its lifetime.

Flaw

The flaw is an imperfection of the steel part of component. It can be an inclusion, lack of fusion, void or a crack. The flaw can be found during manufacturing, pre-service or in-service inspection by some non-destructive testing.

Crack

The crack is special case of flaw, which is characterized by sharp boundary (crack front) and which is of two-dimensional character (the third dimension is negligible). For the purpose of computational assessment, the flaw is usually postulated as a crack.

Nucleation of crack

The nucleation is a part of a process of a crack formation on a component surface. The process starts when a first penetration into material happens, continues during very slow growth and ends when the micro crack transits to propagation. Before a for the time period (called precursor) necessary conditions dominated on the component surface for the initiation to occur.

3 General requirements for calculation of residual lifetime

3.1. Operation conditions and limits

Components and systems in nuclear power plants of the WWER type are allowed to operate in service conditions given in the Design Specifications after the system safety has been checked and verified. Significance of the operation conditions can be different for different components and their parts. Temperatures, stresses, and mechanical loads can be treated as design, operation and test loads.

The suitable guidelines for selection of operation conditions that can be of some significance for selection of appropriate design, operation and test loads and limits of those loads can be derived from the safety criteria for the systems of nuclear power plants. The guidelines can be included into requirements of the Regulatory Authorities with the legal authority for the nuclear power plants.

The loads given in the Design Specifications are to be used for assessment of residual lifetime of the component, piping and their supports. The load time variation should be corrected in accordance with data measured by the sensors of the Instrumentation and Control System (I&C), Monitoring and Diagnostic System (MDS) and temporary Special Measurement (SM). In the case that measured data on loading obtained from I&C, MDS, and SM systems are available for the assessed period, utilisation of these data is preferred rather than data from the Design Specification.

For calculation of residual lifetime of the component, the limits established in the Design Specifications are to be used. If the necessary limits are not given in the Design Specifications, the limits given in the appropriate chapters of this Procedure shall govern.

3.2. Mechanisms of material damage

Components, piping and their supports of WWER type NPPs experience various degradation mechanisms during operation. The degradation mechanisms are:

— Fatigue of the material by cyclic loading;

— Corrosion (pitting corrosion, corrosion cracking under permanent load and corrosion fatigue under accidental cyclic loading, irradiation assisted corrosion etc.);

— Flow accelerated corrosion;

— Cyclic loading induced fatigue growth of flaws potentially induced by cyclic loading;

— Embrittlement and hardening due to radiation;

— Swelling;

— Irradiation assisted creep;

— Phase transformation.

Nuclear power plant components are typically subjected to several degradation mechanisms simultaneously. In the case that this Procedure does not give suitable guidelines for assessment of interacting damage mechanisms, it is necessary to evaluate each mechanism separately and then to assess their simultaneous effect (synergism) based on engineering judgement. The gradual degradation of material properties during the lifetime of the nuclear power plant is taken into account in calculation of the residual lifetime of the components and piping.

3.3. General provisions for calculation of residual lifetime

These "Guidelines" provide the methods for:

— Assessment of residual lifetime from the point of view of resistance against fast fracture;

— Assessment of residual lifetime from the point of view of resistance against fatigue damage;

— Assessment of residual lifetime from the point of view of resistance against corrosion-mechanical damage;

— Assessment of acceptability of flaws found during in-service inspections and assessment of residual lifetime of the component with those flaws;

— Final assessment of residual lifetime of the component.

These "Guidelines" do not offer the methods for assessment of all kinds of material failure. For assessment of material creep at elevated temperatures or material wear, the procedures used should be based on the latest scientific knowledge and accepted by the Regulatory Authorities.

The material characteristics of strength, plastic and brittle fracture resistance, referred to in these "Guidelines" or experimentally derived by the accredited or authorised body, are used for calculation of residual lifetime of the component.

For the lifetime calculation, increases in ultimate strength and yield strength and changes in non-conventional characteristics due to radiation are not taken into account, with the exception of elastic-plastic calculations for fast fracture assessment in accordance with par. 5.2. However, the reduction of plastic characteristics and characteristics of material resistance against fast fracture and fatigue failure has to be taken into account.

For assessment of gradual fatigue damage to cyclically loaded components, the procedures described in these "Guidelines" are to be used.

For calculation of residual lifetime of the component, the groups of stress categories are to be used, in accordance with these "Guidelines".

For calculation of stresses and strains, component nominal wall thickness s taken from the design documentation is to be used. The actual wall thickness can be used if it is known and has to be used if it is less than s.

For calculation of stress in the component with cladding or anti-corrosion layer, the temperature effects of the cladding or the anti-corrosion layer and also mechanical properties of the cladding and the anti-corrosion layer are to be taken into account.

The reduced stresses for static strength and fatigue analyses are to be derived in accordance with the maximum shear stress theory. The reduced stresses for the calculation of the resistance against fast fracture are to be derived in accordance with the maximum normal stress theory.

Calculation of the stress is carried out assuming elastic behaviour of the material in the whole loading range, unless the special cases are studied mentioned in the appropriate sections of these "Guidelines".

For calculation of residual lifetime of the component, the limits for allowable values of stress, displacement, and loading, postulated number of repeats of operation transient modes as well as the required operational lifetime are taken in accordance with the Design Specifications. The number of the actual repeats of the operations transient modes is to be taken in accordance with the measurements of the I&C and MDS systems. The number also can be extracted from the Design Specifications, in relation to the current time of operation. The number of repeats of the operation transient regimes during the residual lifetime is considered in accordance with the component Technical Specifications. This is proportional to the assumed period of the subsequent operation of the component.

The limits of allowable values are different in accordance with the type of structure, the character of load, working regime and categories of active stresses, taking into account also the level of importance of the component and effects of its possible failure.

Calculation of residual lifetime of the components and piping in accordance with this Procedure is to be carried out only by the authorised body (company) having the appropriate Authorisation; the employees, who are performing the work, have to be awarded the Certificate required by the relevant legislation. The authorised body is responsible for extraction of the proper input data for calculation of strength from the developer ´s Design Specifications and from the producer's technical specification and also for use of the proper data from the measurements of the I&C, MDS and SM systems. The authorised body is responsible also for the choice and proper use of the methods of calculation of temperature fields, strains and stresses, if specific methods have not been prescribed. Nevertheless, the used methods have to consider all computational operation loads for all computational modes and facilitate determination of all necessary groups of stress categories.

The authorised body performing calculation of strength of the component is also responsible for correctness of the results of calculation.

Computer codes certified in accordance with the relevant legislation may be used for calculation. If such certification cannot be obtained, then verification report for such code shall be available.

3.4. Documentation

Calculation of residual lifetime is to be carried out for every component and their structural details having lifetime limiting effects. Different types of structures can be included into a single report only if the structures are functionally interconnected.

The holder of the authorisation is authorised to adapt, in accordance with the relevant legal provisions, the required extent of calculation of residual lifetime by issuing the relevant requirements. Calculation of residual lifetime has to involve at least the following parts:

— Technical Terms of Reference;

— Justification of the computational model;

— Calculation of the temperature fields, strains and stresses;

— Calculation of residual lifetime;

— Conclusions and recommendation of measures.

The part "*Technical Terms of Reference*" shall provide a short description of the type of the component and its use, and the sources of inputs (projects, designs, plans etc.) must be identified. Geometry of the component, the marks of the used materials and the loads in the sites of their actions have to be displayed in figures and sketches. Specification of loads, operation modes and loading blocks including their specific properties, physical and strength characteristics necessary for strength calculation should be also given. Specification of design bases requirements and load combinations shall be taken into account, including loads due to external effects specified in the technical design (seismic events, external explosion, and fall of an airplane). The limits for reduction of stresses, strains, cumulative damage etc. have to be also displayed.

In the part "*Justification of computational model*", the choice of the areas in the component and acting loads is to be justified and the methods of schematisation of geometry and acting loads used for computational or experimental model are to be presented.

In the part "*Calculation of temperature fields, strains and stresses*", the methods used for calculation and for experimental measurements have to be justified. Use of the computing codes also must be justified, and the source of their verification (the verifying body, the date of verification, relevant legal provisions for evaluation of the compliance) must be identified. Results of calculations and experimental measurements shall be presented in a clear (graphic) format.

In the part "*Calculation of residual lifetime*", it has to be demonstrated that the calculated strains, stresses and temperatures and flaws, fatigue usage factor and wall thinning, if applicable, do not exceed limits established by this Procedure.

In the part "*Conclusions and recommendations of measures*", the results of calculation of residual lifetime of the component are to be generalised. If necessary, the required verification of in-service measurements and special measurements for a specified operation period are to be specified. If necessary, the storage sites of the surveillance samples for demonstration of changes of the material properties during operation of the component and piping are to be recommended. The storage sites of part of the surveillance samples must be (for the purpose of comparison) out of influence of any degradation process. If necessary, the areas of the component or piping, where non-destructive in-service tests have to be carried out, are to be determined or specified. The necessary frequency of non-destructive testing is to be determined. If needed, the period of exchange due to ageing of replaceable parts of the component is to be determined.

An example of the table of contents of the Residual Lifetime Assessment Report is given in Appendix I.

3.5. System of quality assurance

The System of the quality assurance, in accordance with the relevant legal provisions, has to be established in the body performing calculation of residual lifetime of the component.

The System of the quality assurance has to be demonstrated and documented by the Manual of the Quality Assurance (QM – Quality Manual), the Procedures of the Quality Assurance (QP – Quality Procedures) and the Instructions for Quality Assurance (Quality Instructions – QI). The Quality Assurance Procedures for development and use of computing codes and for performing the residual lifetime calculation of the component (the manual) shall be issued. The Procedures have to involve the declaration of responsibility for application of valid and verified computing codes for the defined computing technique. The requirements for professional qualification of workers who are responsible for activities or execute activities related to development and use of computing codes and performing the calculation of residual lifetime of the component have to be defined. The responsibilities for awarding, maintaining and control of the professional qualification of workers have to be presented.

The Quality Assurance Procedure for identification and recording of disagreements related to the development and use of computing codes and to performing the calculation of residual lifetime of the component and procedures for adoption and inspection of corrective measures shall be issued.

The List of controlled documents with nominal responsibilities for their elaboration and deposition shall be issued.

The Procedure for archiving of controlled documents with responsibilities for their storage, maintenance and access to those shall be issued.

A page containing name of the document, date of its issue, the issue number, the review number, indication if changes shall be sent to the holder and name and signature of the author of the document has to accompany the controlled document, in accordance with the established Quality Assurance System. Also names, signatures and date of signing of the persons who verified, approved and adopted (if applicable) the document have to be added.

The author of the certificates has to prove in the controlled way compliance with all requirements contained in the Design Specification issued by the owner of the component for which calculations of residual lifetime are performed.

The author of the certificates has to take into account all comments to the elaborated calculation of residual lifetime issued by the Inspection authority chosen by the holder of the permission.

The author of the residual lifetime calculation has to issue the List of approved suppliers of computing codes, single parts of residual lifetime calculation and material characteristics. Before enlisting them into the List of approved suppliers, the author has to check the application of their Quality Assurance Systems, ownership and validity of their authorisations and certificates for the required activities. The mentioned systems have to be regularly audited by the author. If the supplier of services has not established its own Quality Assurance System, then the responsibility for his activities is transferred to the author and the author has to prove the way of the quality assurance of the supplied services. If the supplier of services does not own the required Authorisation or Certificate, then the author will have to own the necessary Authorisation or Certificate for the supplied services, for example, for measurements of the material characteristics.

The author of calculation of residual lifetime has to own the valid Authorisation and to employ workers with valid Certificates for activities related to performing the residual lifetime calculation for the component of the nuclear power plants (NPP).

4 Procedure for assessment of residual lifetime and integrity of the component during operation

4.1. General Approach

4.1.1. The residual lifetime assessment of the components excepting reactor internals is usually carried out in four steps described in Sections 4.2–4.5 and in Chapters 5–8. The residual lifetime assessment of reactor internals is carried out according to specific methods and procedures presented in Appendix C. The integrity assessment of piping and supports are described in Sections 4.7–4.9 and Appendices A, B, D, F.

4.2. Assessment of residual lifetime of the component from the point of view of the resistance against fast fracture ("with the postulated defect")

4.2.1. During the assessment, all environmental effects and operational conditions should be taken into account, including design accidents conditions. Assessment represents a check of the project level lifetime assessment and is described in detail in Chapter 5 of this Procedure.

4.2.2. The calculation is based on a "postulated defect" – the surface or underclad semielliptical crack. The actual operation conditions are to be used for calculation, including the degradation processes in the material of the component.

4.2.3. The residual lifetime calculation is carried out on the basis of trends of material transition temperature changes established for the designed or assumed component lifetime.

4.2.4. The criterion of residual lifetime of the component from the point of view of resistance against fast fracture is the exclusion of fast fracture initiating from the "postulated defect" during all design bases operating conditions. Resistance against fast fracture is assured (for emergency conditions or anticipated operational transients) if the transition temperature of the component material is lower than its maximum allowable value [Tk] or [T0]. In case of reactor pressure vessels, this temperature is usually determined from the worst mode of pressurised thermal shock. For normal operation conditions or hydrotests, so-called [p]-[T] curves (dependence of allowed pressure on the primary coolant temperature), can represent the decisive mode. In such special cases, the allowed range of operation parameters can be so close to the saturation curve (shifted to lower temperatures with the necessary safety margin) that the safe operation of the component is not possible.

4.3. Assessment of "fatigue lifetime" of the component

4.3.1. Residual lifetime of the component from the point of view of resistance against fatigue damage is to be determined during the whole assumed technical lifetime of the component until initiation of the surface macro-crack of the conventional size equal to 2.0 mm that is conservatively expected for fatigue usage factor reaching value D = 1. In a special case, when a macro-crack larger than 2.0 mm is found on the component, the component residual lifetime may be determined according to Section 4.5.

4.3.2. The aim of assessment is to determine conditions for potential initiation of the macro-crack on the material originally without any flaw due to thermal-mechanical cyclic (fatigue) loading. This is assessment of residual lifetime of the component under the cyclic loading, which is described in Chapter 6 of these "Guidelines".

4.3.3. The actual operation conditions are taken into account in this calculation in accordance with Section 3.3, i.e., the actual operational thermal-stress cycles, their time variations, frequency and series.

4.3.4. The criterion of residual lifetime of the component from the point of view of resistance against fatigue damage is exclusion of any formation of the macro-crack of the size equal or above 2.0 mm due to the cyclic loading. During the period of the assumed technical lifetime of the component as mentioned, for example, in design, no existence of the macro-cracks is admissible. A possibility of the time-limited operation with the detected macro-crack, which arose during operation, has to be assessed in accordance with Section 4.5 of these "Guidelines".

4.4. Assessment of residual lifetime of the component from the point of view of resistance against corrosion-mechanical damage

4.4.1. Residual lifetime of the component from the point of view of resistance against corrosion-mechanical damage is defined as the time remaining until formation of a surface flaw due to corrosion-mechanical loading.

4.4.2. The aim of assessment is to determine the conditions for formation of the macro-crack sized 2.0 mm on the material originally without any flaw due to corrosion-mechanical loading. The procedure of assessment is described in Chapter 7 of these "Guidelines".

4.4.3. The actual operation conditions are taken into account in this calculation, i.e., the actual operation thermal-stress cycles, their time variations, frequencies and sequences and influence of fluid chemical additives or influence of adulterants in surface deposits.

4.4.4. The criterion of residual lifetime of the component from the point of view of resistance against corrosion-mechanical damage is exclusion of formation of any macro-crack of size equal to or above 2.0 mm due to a corrosion-mechanical damage. During the period of the assumed technical lifetime of the component as mentioned, for example, in design, no initiation of macro-flaw (pit, crack) is admissible. The possibility of a time-limited operation with the macro-flaw detected in service has to be assessed in accordance with Chapter 7 of these "Guidelines".

4.5. Assessment of residual lifetime of the component with flaws detected during in-service inspection

4.5.1. Residual lifetime of the component with flaws found by non-destructive tests during in-service inspections is to be calculated in accordance with the procedure described in Chapter 8 of these "Guidelines".

4.5.2. The aim of assessment is to determine the conditions for ensuring stable flaw behaviour, taking into account fatigue crack growth of the flaw detected.

4.5.3. For the calculation, the actual operation conditions are to be taken into account in accordance with Section 3.3. Actual operation thermal-stress cycles, their time variations, frequencies and sequences, actual temperature of the metal and also the actual radiation load, radiation damage and thermal ageing of the component material are to be considered. Possible growth of the flaws due to operation conditions during the assumed technical life of the component is to be derived for the flaws schematised in accordance with Appendix X of these "Guidelines".

4.5.4. The criterion of residual lifetime of the component with flaws determined by non-destructive tests carried out during outages and shutdowns is exclusion of growth of the flaws over the allowed value during the period of the assumed technical lifetime of the component as mentioned, for example, in design.

4.6. Assessment of residual lifetime of reactor vessel internals

4.6.1 The components of internals are undergone high neutron irradiation damage (dose may exceed 60 dpa for 40 years of operation and maximum irradiation temperature may reach 400 °C) and, hence, degradation mechanisms of reactor vessel internals have specific features. The residual lifetime assessment of internals is carried out according to specific methods and procedures presented in Appendix C.

4.7. Assessment of integrity of safety important piping

4.7.1. For high-energy or moderate-energy piping properly designed, manufactured, assembled and operated, the risk of its break for unknown and hard predictable reasons may be so high that it is necessary to adopt and carry out additional procedures and measures that enable either lowering the probability of possible ruptures, or reducing the effects of its potential breaks.

4.7.2. The required NPP protection against the risks resulting from potential break of high-energy piping for reasons not taken into account in the design may be ensured by one of the following approaches:

— Demonstration that at the event of potential high-energy piping break it is possible to ensure safe shut-down of the unit, its maintaining in stand-by state, and keeping possible radioactive releases in the allowable limits.

— Documentation and realization of additional measures that can reduce the probability of break of high-energy piping for reasons not taken into account in the design.

4.7.3. Demonstration that at the event of potential high-energy piping break it is possible to ensure safe shut-down of the unit, its maintaining in stand-by state, and keeping possible radioactive releases in the allowable limits may be realized based on one of the following procedures:

— Physical separation of the appropriate high-energy piping lines and components.

— Evaluation of the effects of high-energy piping breaks.

4.7.4. Documentation and realization of additional measures that can reduce the probability of break of high-energy piping for reasons not taken into account in the design (like physical separation, different type of whip restraints, additional supports, etc.) may be applied based on one of the following procedures:

— Application of approach "Reduction of Probability of Break" – the requirements are described in the Appendix B.

— Application of approach "Leak-Before-Break" – the requirements are described in the Appendix A.

4.8. Procedure for RI-ISI methodology for piping systems

4.8.1. Risk Informed In-Service Inspection (RI-ISI) methodology and basic requirements on RI-ISI applications with special emphasis on WWER type piping systems and other passive components applications are given in Appendix D.

4.9. Integrity of component and piping supports

4.9.1. Appendix F contains requirements, rules and recommendations for integrity assessment of component and piping supports used in NPPs with WWER type reactors during the whole lifetime including:

— Design;

— Manufacture, installation, repair and replacement;

— Test and operation;

— In-service inspection;

— Operational life time extension.

The provisions of this Appendix cover as well ageing management for supports and lifetime information management (data ware). Appendix F is applicable to supports of equipment components and pipelines of 1st, 2nd, 3rd and 4th safety classes.

5 Assessment of component resistance against fast fracture

5.1. General conditions

5.1.1. Assessment of ferritic steel component lifetime based on resistance against fast fracture should be performed in accordance with this Chapter. Assessment is based on stress intensity factor KI, computation of which is based on either linear-elastic or elastic-plastic fracture mechanics. Selection of components for the assessment is based on national Regulatory Authority requirements.

5.1.2. Assessment of component resistance against fast fracture is performed for all regimes of NOC, HT, AOT as well as EC.

5.1.3. Base materials characteristics for the calculations are static fracture toughness, K_{IC} (K_{JC}), and transition temperatures: reference temperature T_0 (based on Master curve approach) and/or critical temperature of brittleness, T_k. Material damage due to operating conditions is expressed in terms of a shift of temperature dependences of static fracture toughness (characterised by reference temperature, T_0) or impact notch toughness (characterised by critical temperature of brittleness, T_k) as a result of different operating stressors.

5.1.4. Resistance against fast fracture is assured if the following condition

$$K_I \leq [K_{IC}]_i, \qquad (5.1)$$

is fulfilled for a postulated crack-like defect, where $[K_{IC}]_i$ is the allowable value of stress intensity factor for a given type of operating condition. Index **i** indicates different operating conditions:

i = 1 - normal operating conditions (NOC),

i = 2 - anticipated operational transients and hydrotests (AOT and HT),

i = 3 - postulated accidents / emergency conditions (EC).

5.1.5. Values of K_I and $[K_{IC}]_i$ are compared at least for the deepest and surface or near interface points of the postulated defect. (The near interface point is the point just below the interface between cladding and base or weld material in the case of cladded components). Comparison of those values for all points of postulated crack front is recommended (it is usually possible only for a finite element solution on a model with crack included in the mesh).

5.1.6. This procedure may be applied for component integrity assessment and lifetime evaluation during NPP operation based on resistance against fast fracture. In the case of component assessment during operation, actual material characteristics may be used in calculations if their definition is accepted in advance by Regulatory Authorities.

5.2. Temperature and stress fields

5.2.1. Stress and temperature fields must be calculated for all normal operating regimes, anticipated operational transients and postulated accidents either represented by their design parameters or by parameters calculated from thermal-hydraulic analyses according to the Appendix VI.

5.2.2. Stress calculations must be performed taking into account internal pressure, dead weight and temperature gradients. If significant, also other mechanical stresses such as stresses due to pre-tighting of flanges or stresses due to loading from connected piping, etc., have to be taken into account. For all operating conditions, residual stresses (in welding joints and in cladding) must be considered.

5.2.3. Residual stresses in components without cladding other than RPV

— Residual stresses σ_R in welding joints of ferritic steels after annealing in components without cladding other than RPV in absence of measured values can be taken conservatively according to the following formula:

$$\sigma_R = 60 \cdot \cos\left(\frac{2\pi x}{s_w}\right), \quad [\mathrm{MPa}], \tag{5.2}$$

where

x is coordinate in weld thickness direction (with its origin in surface point for the case of uncladded component or in interface point for the case of cladded component),

s_w is weld thickness.

Formula (5.2) can be used only in the case when heat treatment of the weld joint was performed after welding.

— Residual stresses may be modelled by introducing initial strain field to provide required profile of residual stresses. The appropriate procedure must be previously accepted by national Regulatory Bodies.

5.2.4. Cladded reactor pressure vessels:

— Distribution of residual stresses in the circumferential weld

Distribution of residual stresses over the cross-section of a circumferential weld is calculated by the formula

$$\begin{cases} \sigma_{res}(N) = \sigma_{res}^{clad} \quad if \quad N \le S_{clad} \\ \sigma_{res}(N) = \sigma_{res}^{W} \quad if \quad S_{clad} \le N \le N_o \\ \sigma_{res}(N) = \sigma_{res}^{W} \cdot \cos(2\pi \cdot \dfrac{N - S_{clad}}{S}) \quad if \quad N > N_o \end{cases} \tag{5.3}$$

where N_o is calculated from equation:

$$\cos(2\pi \cdot \frac{N - S_{clad}}{S}) = 1 - \frac{2 \cdot (N - 2.5 \cdot S_{clad})}{S_{clad}} \tag{5.4}$$

In Equations 5.3 and 5.4: S is the weld thickness (wall thickness without cladding); S_{clad} is the thickness of cladding; N is the normal coordinate to surface of RPV element. Residual stresses σ^W_{res} and σ^{clad}_{res} are determined according to Figure 5.1.

— Distribution of residual welding stresses and strains outside the weld

The distribution of residual stress σ_{res} for WWER RPV after cladding and post-weld tempering for region outside of circumferencial weld may be presented according to the scheme in Figure 5.3. Value of residual stresses σ^{cl}_{res} and σ^{b}_{res} depending on duration of tempering is presented in Figure 5.2.

Axial residual stresses $(\sigma_{zz})_{res}$ in the cladding, as well as in the zone of base metal adjoining the cladding (zone size $2S_{cl}$), are accepted to be equal to circumferential stresses $(\sigma_{\theta\theta})_{res}$, that is,

$$(\sigma_{zz})^{cl}_{res}=(\sigma_{\theta\theta})^{cl}_{res}=\sigma^{cl}_{res},$$

$$(\sigma_{zz})^{b}_{res}=(\sigma_{\theta\theta})^{b}_{res}=\sigma^{b}_{res}, \tag{5.5}$$

where σ^{cl}_{res} and σ_{bres} – residual stresses in the cladding and base metal, respectively.

Value of compressive residual stresses (see Figure 5.3) is determined from the condition of self-balancing of the field of residual stresses in the considered section. With this, it is allowed to accept that the compressive residual stresses $(\sigma_{\theta\theta})^{com}_{res}$ and $(\sigma_{zz})^{com}_{res}$ are distributed uniformly over the remaining part of the section as:

$$(\boldsymbol{\sigma}_{\theta\theta})^{com}_{res} = -\frac{(\boldsymbol{\sigma}^{cl}_{res} + 1{,}75\boldsymbol{\sigma}^{b}_{res})S_{cl}}{R_2 - R_1 - 3S_{cl}},$$

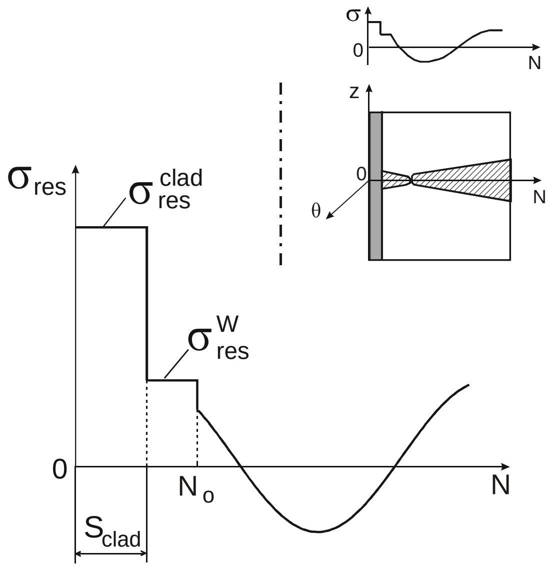
$$(\boldsymbol{\sigma}_{zz})^{com}_{res} = -\frac{\boldsymbol{\sigma}^{cl}_{res}(2R_1S_{cl} + S^2_{cl}) + \boldsymbol{\sigma}^{b}_{res}(3{,}5R_1S_{cl} + 6{,}58S^2_{cl})}{(R_2)^2 - (R_1 + 3S_{cl})^2} \tag{5.6}$$

— Distribution of residual stresses in the circumferential weld in uncladded RPV

Distribution of residual stresses over the cross-section of a circumferential weld is calculated by the formula:

$$\sigma_{res}(N) = \sigma^{W}_{res} \cdot \cos(2\pi\frac{N}{S}), \tag{5.7}$$

where S – wall thickness of the considered component of the RPV; N – coordinate at the normal to a surface of the considered component of the RPV (the considered coordinate system is given in Figure 5.4); stresses σ^{W}_{res} is determined by the dependence presented in Figure 5.2.



Z, X, N is the local coordinate system;
N is the normal to surface of RPV element;
Z is the tangent to surface of RPV element.

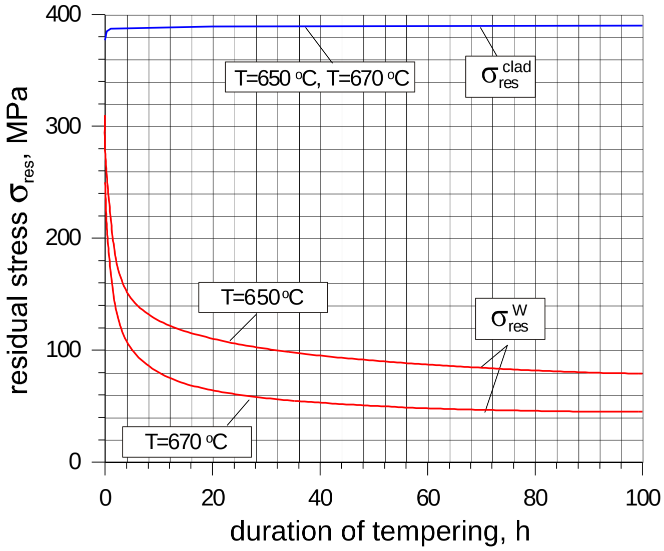


FIG. 5.1. Schematization of distribution of residual stresses σ_{clres} and σ_{Wres} on cross-section of the weld of RPV caused by cladding and post-weld tempering.

FIG. 5.2. The dependence of residual stress σ_{Wres} and $\sigma_{cladres}$ on duration and temperature of tempering.

5.2.5. Temperature and stress field must be calculated for different time steps which must be chosen in such a way to catch all local maxima/minima of stress as well as the transient course until stabilised conditions occur or until the time important for the determination of temperature [T_t] is reached (see 5.10.6).

5.2.6. Temperature and stress fields are calculated using temperature-dependent characteristics of base material, weld metal as well as cladding materials. Changes of these properties due to radiation damage may be taken into account if they are reliably known. Values of parameters of thermal-physical properties for materials of WWER 440 and WWER 1000 reactor pressure vessels recommended for calculations are presented in Appendix XV. Generic thermal-physical properties as prescribed in Appendix XV can be used for both layers of cladding. The tensile properties necessary for elastic-plastic calculations should be taken plant-specific, if such data do not exist, then generic properties can be taken.

5.2.7. Calculation of temperature and stress fields shall take into account also the existence of austenitic cladding and its plastic behaviour in the case of cladded components.

5.2.8. All formulae (5.2) – (5.7) of stress distribution profiles are valid for room temperature in the components after their heat treatment for stress relieving. Stress profiles in the components after hydrotest should have to be calculated prior calculation of pressurized thermal shock.

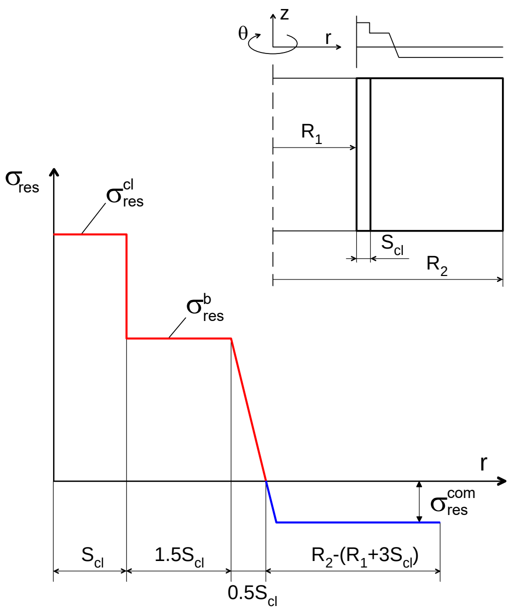
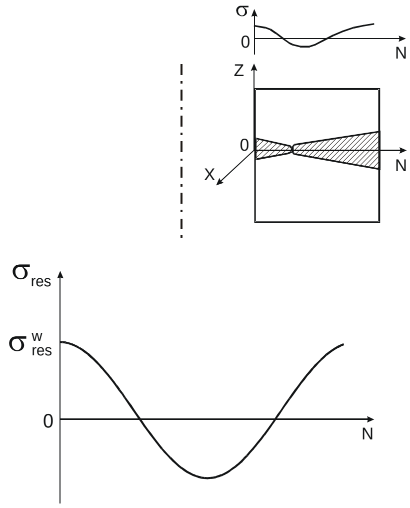


FIG. 5.3. Schematization of the distribution of residual stresses $\sigma^{cl}{}_{res}$ and $\sigma^{b}{}_{res}$ on cross-section of the wall of RPV caused by cladding and post-weld tempering.
$\sigma^{b}{}_{res} = \sigma^{w}{}_{res}$



(Z, X, N – local Cartesian coordinate system;
N – normal to surface of the considered component of the RPV;

Z – tangent to the generating line of the considered component of the RPV).

FIG. 5.4. Distribution of residual stresses over the cross-section of the RPV circumferential weld (for uncladded RPV).

5.2.9. Calculation of temperature and stress fields may be performed using numerical as well as analytical methods if they are accepted by Regulatory Authorities. Generally, Finite Element Method is recommended for temperature and stress fields calculations.

5.3. Stress intensity factor

5.3.1. Stress intensity factor for a chosen postulated defect may be calculated by numerical methods or by simplified engineering methods if they are accepted by Regulatory Authorities in advance.

5.3.2. The preferable way for determining the stress intensity factor for both surface and underclad cracks is using Finite Element Method (FEM) code on elastic-plastic model with crack included in the mesh. In that case the J-integral is usually calculated directly by the FEM code and subsequently the stress intensity factor can be calculated using the following formulae:

for surface points of the crack (plane stress condition):

$$K_I = \sqrt{J \cdot E} \tag{5.8}$$

for other points of the crack (plane strain condition):

$$K_I = \sqrt{\frac{J \cdot E}{1-\nu^2}} \tag{5.9}$$

where the Young modulus *E* is determined for the actual temperature of the relevant point on crack tip.

5.3.3. Stress intensity factor for surface cracks for all types of operating conditions may be also determined with the use of procedure given in Appendix IV, based on stresses computed on model without crack. In the case of cladded components, stress values extrapolated from the base/weld material to points located in cladding are used in the formulae from the Appendix IV instead of stresses computed directly in the cladding.

5.3.4. In the case of underclad cracks, specific formulae taking into account the effect of cladding are given in Appendix IV.

5.4. Transition temperatures of material

5.4.1. The following transition temperatures, T_t, may be used for characterisation of material state: reference temperature, T_0, determined from static fracture toughness tests using "Master Curve" approach, as well as critical temperature of brittleness, T_k, determined from Charpy impact tests. Material state of degradation should be determined according to Appendix III for all assessed points of the postulated crack, as defined in Section 5.7.

5.4.2. Reference temperature T_0

Reference temperature T_0, increasing during operation, is determined experimentally from surveillance specimens irradiated to required neutron fluence in location well representing operating conditions of reactor pressure vessel beltline. End-of-life design fluence should be taken as a basis for initial evaluations. Possible thermal and cyclic damage ageing should be also taken into account.

Determination of reference temperature T_0 is performed using "Master Curve" approach using multi-temperature approach preferably to the single-temperature one.

Reference temperature T_0 is defined from experimentally determined values of static fracture toughness, K_{JC}, adjusted to the thickness of 25 mm. Margin σ is added to cover the uncertainty in T_0 in accordance with Appendix III and for the assessment the value

$$RT_0 = T_0 + \sigma + \delta T_{type} \tag{5.10}$$

is used,

where δT_{type} is a correction to the specimen type:

$$\delta T_{type} = \begin{cases} 15^{o}\mathrm{C} & \text{if precracked Charpy specimens are used} \\ 0^{o}\mathrm{C} & \text{if CT specimens are used} \end{cases} \tag{5.11}$$

5.4.3. Critical temperature of brittleness T_k

[1] Critical temperature of brittleness during component operation is determined as:

$$T_k(t) = T_{k0} + \Delta T_F(t) + \Delta T_T(t) + \Delta T_N(t) \tag{5.12}$$

where

$T_k(t)$ - critical temperature of brittleness for time t, °C

T_{k0} - initial critical temperature of brittleness, °C

$\Delta T_F(t)$ - transition temperature shift due to radiation damage, °C

$\Delta T_T(t)$ - transition temperature shift due to thermal ageing, °C

$\Delta T_N(t)$ - transition temperature shift due to cyclic damage, °C

Values of T_{k0} are determined from Acceptance tests of materials. Values of $\Delta T_F(t)$, $\Delta T_T(t)$ and $\Delta T_N(t)$ are determined from surveillance specimens programme tests, material qualification tests or standards (see also Appendix III).

Trends in changes of critical temperature of brittleness are determined taking into account evaluated trends of component operation - from the point of view of number and types of operation regimes as well as neutron fluences and truly determined material properties.

— Reference initial critical temperature of brittleness RT_{k0} is calculated from the initial critical temperature of brittleness given in the Passport of the component as:

$$RT_{k0} = T_{k0} + 1.64\ \delta T_M \tag{5.13}$$

where margin δT_M is added to cover the uncertainty in T_{k0} in accordance with Appendix III.

5.5. Procedures for determination of neutron fluence in reactor pressure vessels

5.5.1. Determination of radiation loading, i.e. values of neutron fluences in different places of reactor pressure vessel, is necessary for precise determination of degradation trends in reactor pressure vessel materials as well as for prediction of reactor pressure vessel residual lifetime.

Determination of radiation loading of reactor pressure vessels is carried out by combining measurements and calculations. Neutron fluences in given reactor pressure vessel points and in surveillance specimens positions are determined by calculating the absolute values. The calculation results are compared with the measurement results in surveillance specimens positions and on outer reactor pressure vessel wall. All values which are necessary for calculations, interpretation of measurements and comparison of calculated and measured values shall be archived.

5.5.2. Inputs for reactor pressure vessel integrity evaluation must not be only neutron fluences reached during actual operation but also estimation of their trend changes until the reactor end-of-life. These trends must be prepared taking into account planned further operation and potential changes in reactor operation, such as changes in operational cycles, campaigns, loading pattern, uprating, types and enrichment of fuel elements, etc.

5.5.3. More detailed procedure of fluence determination is presented in Appendix II.

5.6. Allowable stress intensity factors

5.6.1. Allowable stress intensity factors, $[K_{IC}]_i$, and also allowable stress intensity factors for crack arrest, $[K_{IA}]_i$, depend on material temperature and operating conditions of the component. It is allowed to determine their temperature dependences with respect to the transition temperature according to the formulae in this section. More detailed description is presented in Appendix V.

5.6.2. Allowable stress intensity factors based on reference temperature T_0

— Temperature dependence $[K_{IC}]i$ based on reference temperature T_0 is defined as lower bound curve of two curves obtained from the 5% tolerance bound of "Master Curve"; one curve is calculated using a safety factor nk and the other one using a temperature safety factor ΔT, i.e. using the formula:

$$[K_{IC}]_i(T\text{-}RT_0) = \min \{ n_k^{-1} .\ K_{JC}^{5\%}(T\text{-}RT_0);\ K_{JC}^{5\%}(T\text{-}RT_0\text{-}\Delta T) \} \tag{5.14}$$

where $K_{JC}^{5\%}(T\text{-}RT_0)$ represents a 5 % tolerance bound of "Master curve" for temperature $(T\text{-}RT_0)$ etc. The denotation $[K_{IC}]_i$ is used here for simplification of the following text even though it is based on K_{JC}.

Safety factors are defined as follows:

for NOC (i=1) : $n_k = 2$; $\Delta T = + 30$ °C (5.15)

for HT and AOT (i=2) : $n_k = 1.5$; $\Delta T = + 30$ °C (5.16) for EC (i=3) : $n_k = 1$; $\Delta T = + 0$ °C (5.17)

For components made from 15Kh2MFA(A), 18Kh2MFA, 25Kh2MFA, 15Kh2NMFA(A), 22K and 10GN2MFA type steels and their welding joints, the following relation may be used:

$$K_{JC}^{5\%}(T) = \min\{25.2 + 36.6\cdot\exp[0.019\cdot(T-RT_0)]; 200\} \quad (5.18)$$

— Temperature dependence of allowable values of stress intensity factors for crack arrest $[K_{IA}]_3$ based on reference temperature T0 for materials used in WWER-440 and WWER 1000 reactor pressure vessels may be calculated from the following formula:

$$[K_{IA}]_3(T) = \min\{25.2 + 36.6\cdot\exp[0.019\cdot(T-RT_0-30)]; 200\} \quad (5.19)$$

— The formulae for $K_{JC}^{5\%}$ (for above mentioned materials used in WWER-440 and WWER 1000 NPPs the formula (5.18) may be used) and for $[K_{IA}]_3$ (5.19) are valid for length of crack front 25 mm. For general value of length of crack front for emergency conditions the size correction has to be applied as follows:

$$K_B = K_{min} + \left(K_{25} - K_{min}\right)\cdot\left(\frac{B_{25}}{B}\right)^{\frac{1}{4}} \quad (5.20)$$

where

K_B is the value of $K_{JC}^{5\%}$ corrected to length of crack front B,

K_{25} is the original value of $K_{JC}^{5\%}$ valid for 25 mm,

B is the length of crack front (in mm),

B_{25} is 25 mm,

K_{min} is 20 MPa·m$^{1/2}$.

For cracks postulated according to Chapter 5.7 the length of crack front used in size correction is given by the formula (5.21)

$$B = 2c\sqrt{1 + 4.6(a/2c)^{1,65}} \quad (5.21)$$

The maximum length of crack front used in the correction is the wall thickness.

The size correction is applied only to the "Master Curve part" of formulae (5.18) and (5.19).

— The fracture toughness value at the upper shelf equal to 200 MPa·m$^{1/2}$ remains unchanged for above mentioned type of steels with neutron fluence not larger than 10^{22} m^{-2}. For cases with higher neutron fluences, it is necessary to show actual values based on correct surveillance specimen testing data or the procedure in 5.6(2).

— Dependence of the fracture toughness value of 15Kh2MFA(A), and 15Kh2NMFA(A) type of steels at the upper shelf on neutron fluence and temperature $K_{JC}^{US}(T,F)$ of base and weld metal is calculated by the following formula:

$$K_{JC}^{US}(T,F) = \sqrt{\frac{J_C^{US}(T,F)\cdot E}{1-\nu^2}} \quad (5.22)$$

where $J_C^{US}(T,F) = J_C^{*}\cdot[1 - C\cdot\Delta T_K - B]\cdot\frac{R_{P0.2}(T,F)}{R_{P0.2}(20,F)}$ (5.23)

J_C^{*} – value of J_C when neutron fluence F=0 and temperature T=20 °C,

$R_{P0.2}(T,F)$ – yield strength of irradiated base or weld metal at temperature T.

When calculating zones RPV, where neutron fluence is:

$F<10^{22}$ 1/m^2 it is assumed that J_C^* =280 N/mm is 5 % lower bound value,

$F\geq 10^{22}$ 1/m^2 it is assumed that J_C^* =175 N/mm is 5 % lower bound value.

Coefficients C and B are assumed to be equal: $C=2.4\cdot 10^{-3}$ K^{-1}; B=0.14.

5.6.3. Allowable stress intensity factors based on critical temperature of brittleness T_k

— Temperature dependence [KIC]i based on critical temperature of brittleness Tk is defined as lower bound curve of two curves obtained from the temperature dependence of static fracture toughness, KIClower bound; one curve is calculated using a safety factor nk and the other one using a temperature safety factor ΔT, i.e. using the formula:

$$[K_{IC}]_i(T-RT_k) = \min \{ n_k^{-1}.K_{IC}^{lower\ bound}(T-RT_k);\ K_{IC}^{lower\ bound}(T-RT_k-\Delta T) \} \qquad (5.24)$$

where $K_{IC}^{lower\ bound}(T-RT_k)$ represents the ”eye-ball” lower bound curve as a function of $(T-RT_k)$ etc. of all experimental data of static fracture toughness of the materials obtained within the qualification and other type tests and values of safety factors n_k and ΔT are given in 5.6.2).

For components made from 15Kh2MFA(A), 18Kh2MFA, 25Kh2MFA, 15Kh2NMFA(A), 22K and 10GN2MFA type steels and their welding joints, the following temperature dependences of allowable stress intensity factors may be used:

$$[K_{IC}]_1(T-RT_k) = \min \{13 + 18\cdot\exp [0.020\cdot(T-RT_k)];\ K_{JC}^{US}(T,F)/2\} \qquad (5.25)$$

$$[K_{IC}]_2(T-RT_k) = \min \{17 + 24\cdot\exp [0.018\cdot(T-RT_k)];\ K_{JC}^{US}(T,F)/1.5\} \qquad (5.26)$$

$$[K_{IC}]_3(T-RT_k) = \min \{26 + 36\cdot\exp [0.020\cdot(T-RT_k)];\ K_{JC}^{US}(T,F)\} \qquad (5.27)$$

For steels 15Kh2MFA(A), 15Kh2NMFA(A) the following dependence of fracture toughness can be used

$$\overline{K}_{IC}(T-RT_k) \equiv \min\{23+48\exp[0.019(T-RT_k)];\ K_{JC}^{US}(T,F)\}, \qquad (5.28)$$

for reference thickness B=150 mm and for fracture probability $P_f=0.05$.

Dependence of allowable values of fracture toughness are given generally as

$$[K_{IC}(T)]_i=\min\{\overline{K}_{IC}(T)/n_k,\ \overline{K}_{IC}(T-\Delta T),\ K_{JC}^{US}(T,F)/n_k\}, \qquad (5.29)$$

where n_k is safety margin in accordance with (5.15)–(5.17).

— Temperature dependence of allowable values of stress intensity factors for crack arrest $[K_{IA}]_3$ based on critical temperature of brittleness Tk for materials used in WWER-440 and WWER 1000 reactor pressure vessels may be obtained from the following formula:

$$[K_{IA}]_3(T-RT_k) = \min \{26 + 36\cdot\exp [0.020\cdot(T-RT_k-30)];\ 200\} \qquad (5.30)$$

— No additional safety margin besides that defined in this paragraph should be applied to $[K_{IC}]_i$ and $[K_{IA}]_3$.

— Formulae for allowable stress intensity factors given in this paragraph can be used only if material testing and evaluation of the results is performed in accordance with the appropriate paragraphs or appendices of this procedure.

5.7. Postulated cracks

5.7.1. The integrity assessment has to be performed for a set of postulated cracks with depth values varying from the minimum value (see 5.7.4) up to the maximum postulated crack depth equal to a_{calc} defined in next paragraphs. Minimum depth of postulated crack for RPV wall to be assessed is 4 mm for uncladded RPVs or for underclad cracks, and thickness of cladding plus 4 mm for surface cracks in cladded RPVs.

5.7.2. Maximum postulated crack depth a_{calc} may be defined with the use of one of the following ways:

a) On the basis of the plant specific ISI non-destructive testing qualification criteria (according to Appendix E);

b) On the basis on manufacturing inspection and technological data;

c) Conservative approach – this approach is used when no information regarding the technological flaws and ISI cannot be carried out.

In cases b) and c), determination of a_{calc} shall be concentrated on the following provisions:

- Analyses of manufacturing technology and maximum size of possible technological flaws;
- Analyses of NDE methods used during manufacturing (sensibility, reliability, acceptance criteria);
- Analyses of experience of manufacturer;
- Calculation of fatigue propagation during analyzed operating time;
- If this approach is used, it is strongly recommended to qualify ISI procedures using a_{calc} as a bases for qualification criteria.

5.7.3. If in-service inspections (see 5.7.2) are performed with devices, procedures and personnel qualified according to requirements of Regulatory Authority using Appendix E, the maximum postulated crack depth a_{calc} may be defined on the basis of the plant specific non-destructive testing qualification criteria. In this case, value a_{calc} is taken equal to higher of the two following values:

(i) Doubled depth of crack detectable with high confidence (i.e. safety factor 2)

(ii) **Reference depth** of a **crack sizable with high confidence** (i.e. safety factor 1). Here, crack sizable with high confidence means a crack the size of which was determined (using UT) with high confidence. **Reference depth of crack sizable with high confidence** is then equal to the sum of depth of crack sizable with high confidence and maximum allowable error in crack depth sizing. The maximum allowable error in crack depth sizing means here the maximum allowable +/- tolerance in crack depth estimation determined as the difference of the measured crack depth by the above mentioned qualified UT method and the real "as built" crack depth.

For postulation of a_{calc} according to this paragraph, the site feed-back is required, i.e. results of qualified in-service inspections for the appropriate inspection areas of components have to be available, together with the appropriate UT qualification criteria and results. *Note: In terminology of non-destructive testing community, "crack depth" is termed as "crack height" or "crack through wall extent dimension".* The recommended value corresponding to application of advanced qualified non-destructive testing techniques is a_{calc} = 0.1 s.

5.7.4. If the conditions when there are not enough information according to 5.7.2 and no qualified non-destructive testing can be performed, the maximum postulated crack depth shall be defined as:

$$a_{calc} = 0.25\ s. \tag{5.31}$$

This size of the postulated defect also includes potential crack growth during operation.

5.7.5. The postulated defects are defined as semielliptical cracks with aspect ratios

$$a/c = 0.3 \text{ and } a/c = 0.7 \tag{5.32}$$

5.7.6. Two orientations of postulated crack shall be considered: perpendicular to direction of first (maximum) principal stress and perpendicular to direction of second principal stress, i.e. in the case of cylindrical vessel, axial orientation and circumferential orientation.

5.7.7. Postulated defect size shall be agreed with the national regulatory body.

5.7.8. For the assessment of postulated cracks in weld metal of weldments, all points of crack front are supposed to lie within the weld material (independently of weld and crack actual dimensions and of the crack orientation) taking into account the possible existence of cladding.

5.7.9. The position of the crack within the component wall is defined specifically for individual types of operating conditions in the following paragraphs.

5.7.10. The integrity assessment must be performed for a postulated cracks with depth values varying up to a_{calc} (or in the case when the whole crack front is assessed, it can be performed for only one crack depth a_{calc}). The crack is located in the component area with the most damaged material or in the region with the maximum thermal-mechanical impact of the regime evaluated. Postulated crack is defined in the following manner:

5.7.11. Uncladded component:

The postulated defect is defined as inner surface crack (see Figure 5.5). In this case, at least the deepest and surface points of the postulated defect must be assessed.

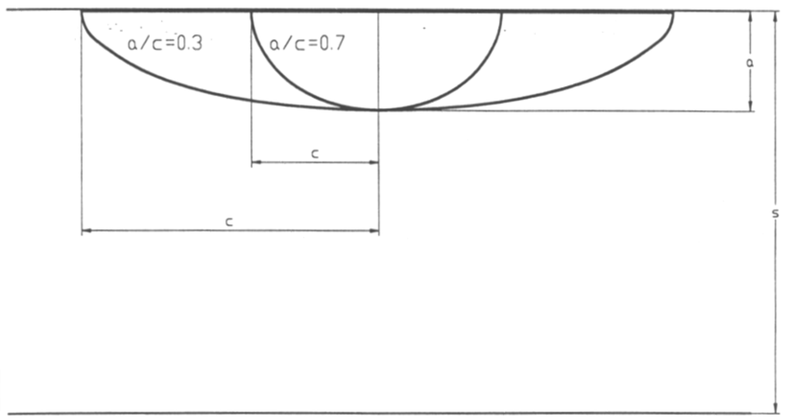


FIG. 5.5. Semielliptical surface crack, uncladded vessel.

5.7.12. Cladded component provided that the cladding integrity is not assured by qualified non-destructive inspections.

The postulated defect is defined as inner surface crack going through the austenitic cladding (see Figure 5.6). In this case, at least the deepest and near interface points of the postulated defect must be assessed.

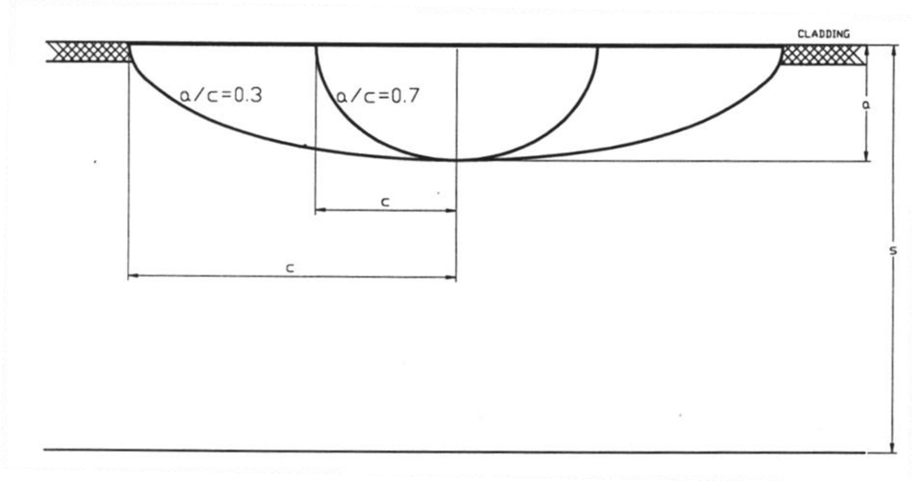


FIG. 5.6. Semielliptical surface crack, cladded vessel.

5.7.13. Cladded component provided that its integrity is assured by qualified non-destructive inspections.

— The crack may be postulated according to this paragraph when the following four conditions are met:

  (i) Non-destructive inspection of the cladding during operation is performed.

  (ii) Performance of the used NDE procedure meets some specified requirements (sensitivity, type and size of reliably detected flow).

  (iii) Any detected flaw with the size, exceeding allowable, has been repaired or its acceptability is justified by calculations.

  (iv) The minimum distance between any found allowable flaw of type specified in a) larger than 3 mm and any found underclad type flaw larger than 3 mm is higher than $4 \cdot s_{cl}$ (where s_{cl} is the thickness of cladding).

— If conditions (i)–(iii) are not met, surface crack as in 5.7.11 has to be postulated. If only the condition (iv) is not met, the detailed analysis of combination of found flaws can be performed. For postulation of the crack according to this paragraph, the site feedback is required, i.e. results of qualified in-service inspections for appropriate inspection areas of components have to be available.

— Assessment of effect of cladding is based on the use of its J-R curve (in the case of multi-layer cladding, J-R curve of its 1st layer). Generic J_{1mm} value of the cladding is recommended, if no real component specific values are available.

— The postulated underclad crack is conservatively defined as partially penetrating 1 mm into the cladding (see Figures 5.7 and 5.8). The extension of the crack to the cladding is supposed with the same length as the length of original major axis of the underclad postulated crack (i.e. the major axis of the semi-ellipse remains on the interface between cladding and base or weld material). Instead of this conservative approach for postulating the crack, a more detailed calculation of crack penetration into the cladding during the regime, based on J-R curve, is allowed, if properly validated.

  In this case, at least the deepest and near interface points of the postulated defect must be assessed with respect to the resistance of base or weld material against fast fracture.

— In this case, the integrity of cladding above the postulated defect during the whole AOT or EC regimes has to be verified. Assessment of ductile tearing for the part of the postulated crack front lying in the cladding shall be based on J-R curve approach. J-values for all time steps of the regime shall be calculated (it is sufficient to calculate J-values only for the middle point of crack front in cladding). These J-values have to be (for all assessed time steps) smaller than the end-of-life value of the appropriate J-R curve corresponding to 1 mm crack extension (i.e. J1mm value). The J1mm values for different RPVs are specified as follows:

a) If no RPV specific data are available, generic values of J_{1mm} are:

   (i) for non-irradiated materials:

   190 kJ/m^2 at 20 °C, 170 kJ/m^2 at 100 °C, and 150 kJ/m^2 at 200 °C,

   (ii) for irradiated materials:

   100 kJ/m^2 for WWER 440 RPV and 150 kJ/m^2 for WWER 1000 RPV.

b) If component specific data are available, then experimentally determined J_{1mm} divided by safety factor 2 shall be used.

c) In the case of other components than WWER 440 or WWER 1000 reactor pressure vessels for which no component specific J-R data are available, surface crack as in 5.7.12 has to be postulated.

— If any of the above conditions a)–c) on J is not fulfilled, surface crack as in 5.7.12 has to be postulated.

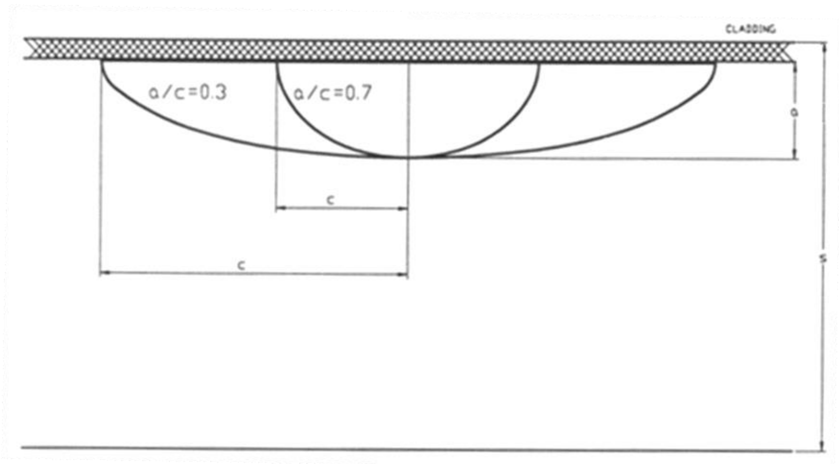


FIG. 5.7. Semi-elliptical underclad crack, cladded vessel.

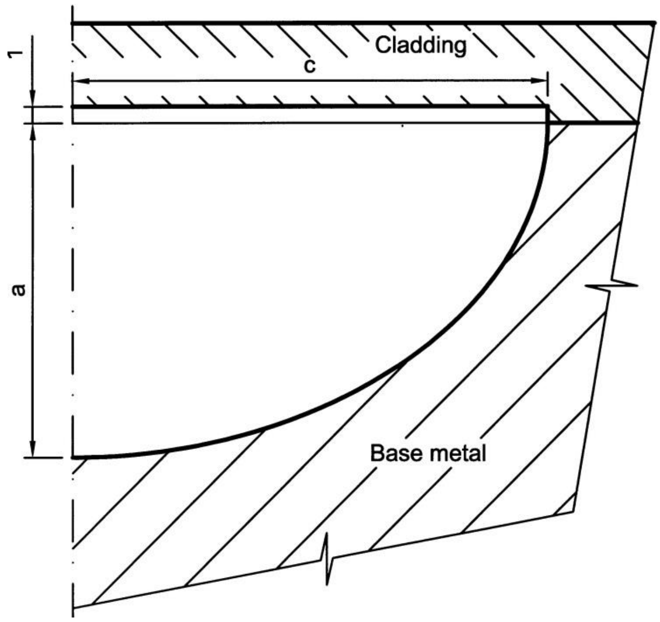


FIG. 5.8. *Semi-elliptical underclad crack partially penetrating into cladding, detail.*

5.7.14. Components for which cooling from the outer surface can occur (e.g., reactor pressure vessel for the accident with reactor cavity flooding): The crack is postulated both on the inner and outer surface.

5.8. Evaluation of normal operating conditions (NOC)

5.8.1. Resistance of the component against fast fracture under NOC is assured, if the following condition is fulfilled in points defined in 5.1.5:

$$K_I \leq [K_{IC}]_1 \qquad (5.33)$$

5.8.2. The postulated defect is defined as a surface crack with further specifications defined in Section 5.7.

5.8.3. Temperature dependence of allowable pressure, i.e., the [p] - [T] curve, can be determined for individual NOC regimes on the basis of condition (5.20) for given transition temperature. The [p] - [T] curve can be constructed in the following way:

For all time steps within the NOC regime, contributions due to thermal loading, K_I^T, to the total stress intensity factor K_I are computed. Contribution due to loading by unit inner pressure, K_I^p, is computed separately. Assuming linear fracture mechanics, the condition (5.19) can be rewritten according to the principle of superposition as follows:

$$p \cdot K_I^p + K_I^T + K_I^{res} \leq [K_{IC}]_1 \qquad (5.34)$$

where K_I^{res} is he stress intensity factor for the crack under residual stresses.

From the equality in this condition the maximum allowable pressure for the particular time step can be determined. The resulting curve is presented as dependence of maximum allowable pressure on the temperature of the system coolant, the time variation of which is known for the actual NOC regime.

5.9. Evaluation of allowable hydrotest temperatures

5.9.1. Resistance of the component against fast fracture during a hydrotest without fuel in the reactor is assured, if the following condition is fulfilled in the points of postulated defect as defined in 5.1.5:

$$K_I \leq [K_{IC}]_2 \tag{5.35}$$

This condition must be fulfilled during the heating-up of the component, pressure increase, holding at maximum pressure and during the following cooling. If the hydrotest if performed with fuel in the reactor, then resistance of the component against fast fracture shall be evaluated in accordance with Section 5.8.

5.9.2. The postulated defect is defined as a surface crack with further specifications defined in Section 5.7.

5.9.3. Allowable temperature during hydrotest is then determined from the equality in condition (5.22) for individual parts of the component for given transition temperature for the time of the hydrotest, where similarly as in 5.8.3 contributions due to thermal loading, K_I^T, and due to hydrotest pressure, $p_H \cdot K_I^p$, to the total stress intensity factor K_I are considered:

$$p_H \cdot K_I^p + K_I^T + K_I^{res} = [K_{IC}]_2 \tag{5.36}$$

Allowable hydrotest temperature, $[T_h]$, is the maximum value of all allowable temperatures determined for individual parts for the hydrotest. The temperature of the component during the hydrotest must be greater than or equal to $[T_h]$.

5.9.4. Temperature dependence of allowable pressure, i.e. the [p] - [T] curve, during the whole hydrotest can be determined from the condition (5.35) for given transition temperature in similar way as described for NOC in 5.8.3.

5.10. Evaluation of emergency conditions (EC) and anticipated operating transients (AOT)

5.10.1. Choice of proper emergency conditions as well as anticipated operational transients shall be performed in accordance with the design. Such evaluation is mandatory for reactor pressure vessels. Requirements for selection of the EC regimes of the PTS (pressurised thermal shock) type for reactor pressure vessel and for thermal-hydraulic calculations of the selected regimes are presented in Appendix VI.

5.10.2. Resistance of the component against fast fracture during EC and AOT is assured if the following condition is satisfied respectively

(1) for AOT: $$K_I \leq [K_{IC}]_2 \tag{5.37}$$

(2) for EC: $$K_I \leq [K_{IC}]_3 \tag{5.38}$$

for points of the postulated crack front (as defined in 5.1.5). The values of $[K_{IC}]_2$ and $[K_{IC}]_3$ may be based either on T_k or T_0 approach.

5.10.3. Calculation of stress intensity factors is performed for selected time intervals of selected EC and AOT regimes and for postulated defects of defined shapes and locations and for the assessed points of their fronts in accordance with 5.10.3. Time steps of calculation are chosen in such a way that values of K_I near the time critical for $[T_t]_j$ determination may be calculated in sufficient detail.

5.10.4. Stress intensity factors K_I are compared with allowable stress intensity factors, $[K_{IC}]_2$ or $[K_{IC}]_3$ for all time steps.

5.10.5. Maximum allowable transition temperature for static crack initiation, $[T_t]_j$ is determined for each of calculated regimes *j*. This temperature is determined from condition (5.24) or (5.25) in which equality is reached. This temperature is defined as the value of transition temperature T_t for which the curve of temperature dependence of allowable stress intensity factors $[K_{IC}]_2$ or $[K_{IC}]_3$ is tangent to the envelope curve of temperature dependences of stress intensity factors K_I for all calculated points of all postulated cracks, for regime *j*.

5.10.6. Warm pre-stressing (WPS) approach may be applied using the following procedure: Global maximum point on the K_I vs. T curve is determined (denoted by K_{Ij}^{max}), see Figure 5.9. Further, points corresponding to level of 0,9 K_{Ij}^{max} are determined on the K_I vs. T curve. From these points that one is selected that corresponds to the lowest temperature (denoted by A in Figure 5.9). The $[K_{IC}]_2$ vs. T curve (or $[K_{IC}]_3$ vs. T curve, if appropriate) is found that goes through the point A (instead of going through the tangent point according to 5.10.6, and the $[T_t]_j$ is accordingly determined. For the whole K_I vs. temperature curve starting from the beginning of the event until the point A, the condition (5.24) (or (5.25) if appropriate) must be met; in opposite case the WPS approach cannot be applied and value of $[T_t]_j$ must be determined based on 5.10.6.

5.10.7. For the K_I vs. T curve portion subsequent to the point A, the local minimum points are determined (denoted by $K_{Ij}^{min}{}_i$, where i means order number of the local minimum point lying behind the point A). For each i-th point and for maximum allowable transition temperature $[T_t]_j$ determined according to 5.10.7, the following temperature dependent curves are established:

(1) for AOT: $$[K_{IC}]_2^{WPS}{}_i = \sqrt{[K_{IC}]_2 . \left(K_{Ij}^{\max} - K_{Ij}^{\min}{}_i\right)} + K_{Ij}^{\min}{}_i \tag{5.39}$$

(2) for EC: $$[K_{IC}]_3^{WPS}{}_i = \sqrt{[K_{IC}]_3 \cdot (K_{Ij}^{\max} - K_{Ij}^{\min}{}_i)} + K_{Ij}^{\min}{}_i \tag{5.40}$$

For each i-th point, the following conditions must be met on the curve starting from the i-th point to the end of the event:

(1) for AOT: $$K_I \leq [K_{IC}]_2^{WPS}{}_i \tag{5.41}$$

(2) for EC: $$K_I \leq [K_{IC}]_3^{WPS}{}_i \tag{5.42}$$

If condition (5.41) (or (5.42), if appropriate) is not met, value of $[T_t]_j$ must be decreased, and the whole procedure must be repeated until the condition (5.41) (or (5.42), if appropriate) is satisfied. For determining the local minimum points on the K_I vs. T curve it is necessary that the corresponding thermal-hydraulic analyses are, from the appropriate viewpoint, conservative (regarding to maximum possible unloading levels, e.g. due to temporary switch-off or switch-over of high pressure ECCS, temporary opening of pressurizer safety or relief valve, for both intended and non-intended operator actions). If the conservativeness of the unloading cannot be ensured, maximum possible unloading must be assumed, i.e., it is necessary to consider total temperature and pressure unloading (only the K_I -values due to residual stresses according to 5.2.2 are assumed).

5.10.8. Maximum allowable transition temperature for static crack initiation, $[T_t]$ is determined from the minimum of all calculated values of $[T_t]_j$ for individual regimes as:

$$[T_t] = \min ([T_t]_j) \tag{5.43}$$

5.10.9. Crack arrest approach may be used in the integrity assessment for EC when probabilistic approach is used – see Section 5.11 and Appendix XVI.

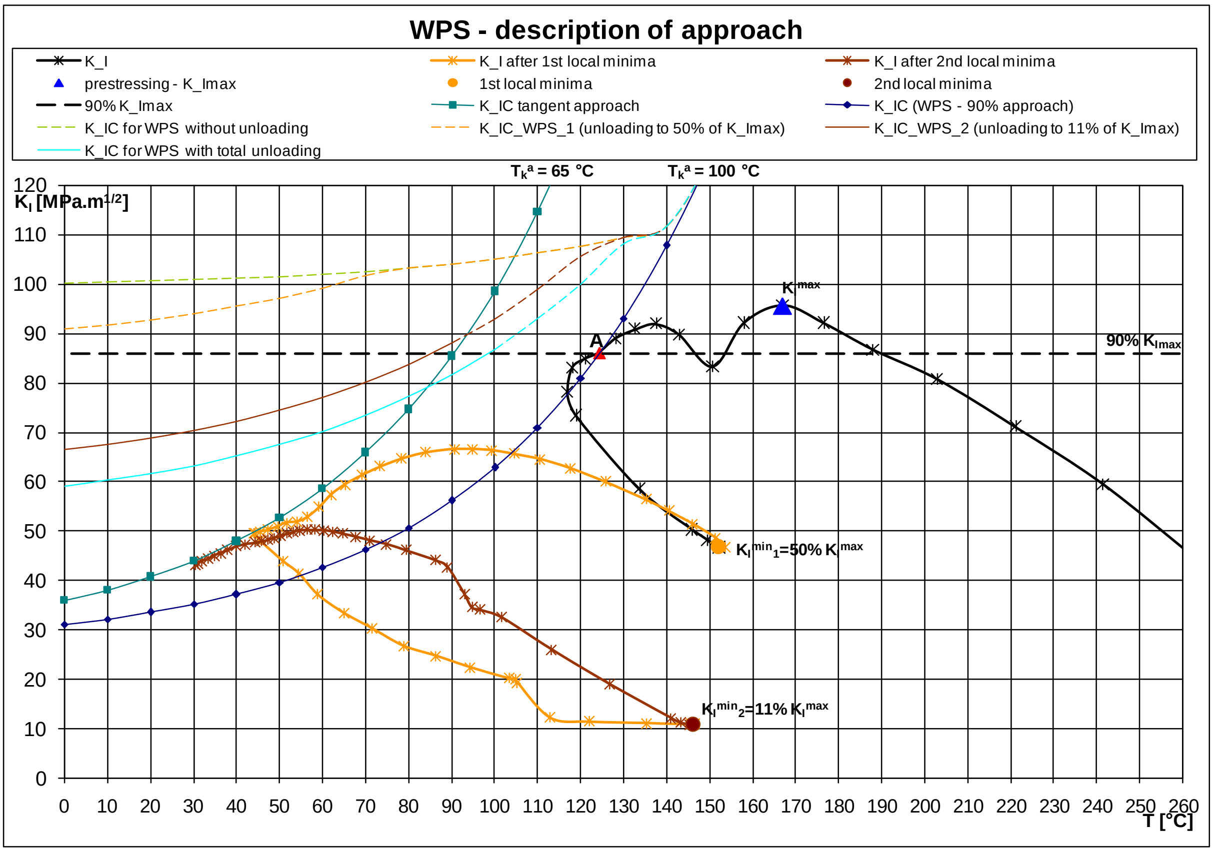


FIG. 5.9. Illustration of the procedure for WPS approach application according to 5.10.6 and 5.10.7.

5.10.10. Integral approach (applicable only to T_0 approach)

The strength conditions are considered as fulfilled if for any time point the following condition is satisfied for the postulated defect front located in base and/or weld metal:

$$\frac{1}{\overline{B}}\int_0^B ZdL < 1 \tag{5.44}$$

In the condition (5.44) the parameter Z is equal to the maximum value α for the whole loading period from 0 to the considered time point τ:

$$Z = \max_{(0,\tau)}\{\alpha\},$$

$$\text{where} \quad \alpha = \left(\frac{n_i K_I(L) - K_{\min}}{\overline{K}_{IC}(L) - K_{\min}}\right)^4 \tag{5.45}$$

excluding those time points when the following condition is fulfilled:

$$K_I < 0{,}9\ \Phi, \tag{5.46}$$

(1) for AOT: $\overline{K}_{IC}$ = $[K_{IC}(L)]_2$ for 5% lower boundary and B equal to $\overline{B}$

(2) for EC: $\overline{K}_{IC}$ = $[K_{IC}(L)]_3$ for 5% lower boundary and B equal to $\overline{B}$

$K_I(L)$ - the distribution of SIF K_I along crack front;

$\overline{K}_{IC}(L)$ - the distribution of $\overline{K}_{IC}$ along crack front that is caused by non-uniform distribution of temperature or fluence ($\overline{K}_{IC}$ is the reference temperature dependence of fracture toughness for the reference crack front length $\overline{B}$ and the fracture probability P_f=0.05 (see, for example, Equation 4.2 in remark 4 or Equation 5.11 in par. 5.6.2.1 of this document):

(1) for AOT: $\overline{K}_{IC}$ = $[K_{IC}(L)]_2$ (5.47)

(2) for EC : $\overline{K}_{IC}$ = $[K_{IC}(L)]_3$ (5.48)

L - curvilinear coordinate (see Figure 5.10); dL - part of the crack front;

B - crack front length; K_{min}=20 MPa√m;

Φ is function of time. For the time point τ, Φ(τ) is equal to the maximum K_I value for the time period from 0 to τ (see Figure 5.11), herein within the time range from 0 to $\tau_{max}^{(1)}$, Φ(τ)=0; $\tau_{max}^{(1)}$- a time point corresponding to the first maximum of K_I dependence on τ. As evident from Figure 5.11, within the range $0 \leq \tau < \tau_{max}^{(3)}$ $\max(K_I)=K^{(1)}_{max}$. Therefore, within the range $\tau_{max}^{(1)} \leq \tau < \tau_{max}^{(3)}$ $\Phi(\tau)=K^{(1)}_{max}$. Within the range $0 \leq \tau < \tau_{max}^{(4)}$ $\max(K_I)=K^{(3)}_{max}$. Therefore, within the range $\tau_{max}^{(3)} \leq \tau < \tau_{max}^{(4)}$ $\Phi(\tau)=K^{(3)}_{max}$.

When applying integral approach, maximum allowable reference temperature $[T_0]_j$ is determined as maximum value of RT_0, for which the component does not fail during the whole regime *j*, i.e. such value of RT_0 for which in conditions (5.47) or (5.48) the equality is achieved (for regime *j*).

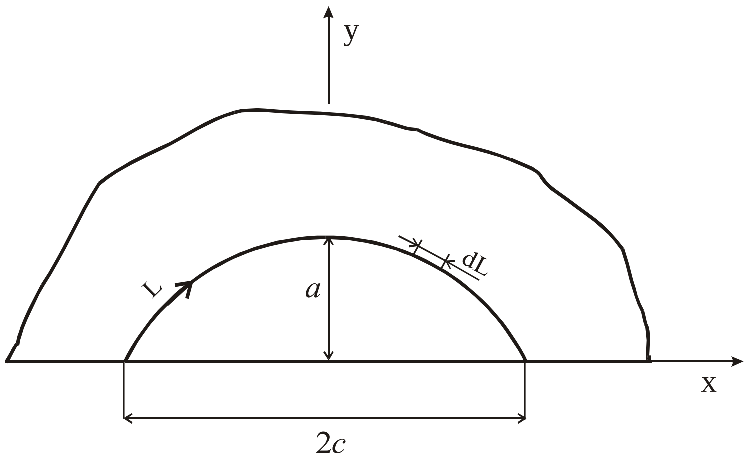


FIG.5.10. Curvilinear coordinate system for surface semielliptical crack.

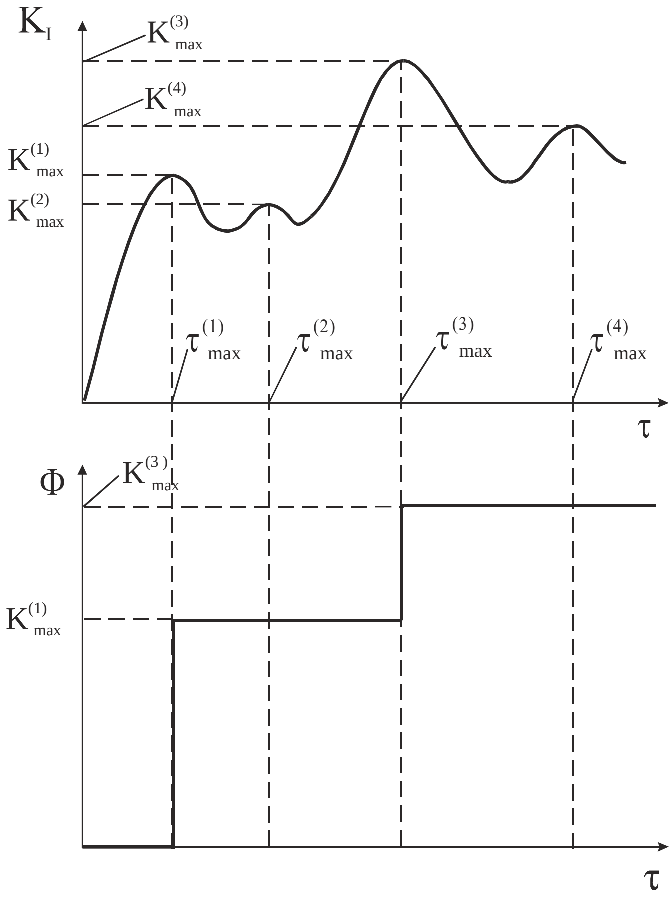


FIG. 5.11. Scheme illustrating the $\Phi(\tau)$ function determination based on a known dependence $K_I(\tau)$.

5.11. Probabilistic evaluation of reactor pressure vessel failure

5.11.1. Probabilistic evaluation of resistance of reactor pressure vessel (RPV) against fast fracture is performed based on a procedure that is described in detail further in the text. The output of the procedure is frequency of initiation of fast fracture, FI, or frequency of RPV failure, FF (the latter means occurrence of crack propagation through the RPV wall, with possibility of crack arrest taken into account). Procedures to determine the frequencies as well as their definitions are described in Appendix XVI.

5.11.2. In case that probabilistic evaluation of RPV resistance is performed based on frequency of failure of RPV, *FF*, the following relation is used as an acceptance criterion:

$$\overline{FF} < 1\cdot 10^{-7}/\text{reactor.year}, \quad (5.49)$$

or

$$FF_{95\%} < 1\cdot 10^{-6}/\text{reactor.year}, \quad (5.49a)$$

where $\overline{FF}$ is mean value of frequency FF of RPV failure, $FF_{95\%}$ is the 95% percentile of the statistical distribution of FF.

5.11.3. Conservative approach may be applied, if instead of determination of frequency of RPV failure, FF, frequency of initiation of fast fracture, FI, is calculated (FI is always higher than FF). In this case, the acceptance criterion has the following form:

$$\overline{FI} < 1 \cdot 10^{-7}/\text{reactor.year}, \tag{5.50}$$

or

$$FI_{95\%} < 1 \cdot 10^{-6}/\text{reactor.year}, \tag{5.50a}$$

where $\overline{FI}$ means mean value of frequency of initiation of fast fracture, $FI_{95\%}$ is the 95% percentile of the statistical distribution of FI.

5.12. Assessment of component residual lifetime with respect to resistance against fast fracture (with postulated defect)

5.12.1. Assessment of residual lifetime is performed periodically in intervals required by Regulatory Authority.

5.12.2. The following inputs must be prepared for this assessment:

— Trends in operation, i.e. number and sequences of different regimes;

— Trends in material changes (shifts of transition temperature) as a function of operational time.

5.12.3. Residual lifetime of the component for normal operational regimes is evaluated from the comparison of calculated temperature dependences of allowable pressure ([p]-[T] curves dependent on the transition temperature of the material - see Section 5.8) with limit dependences given by thermo-hydraulic conditions for safe operation (saturation curve shifted by some safety margin). Knowledge of trends in materials transition temperature as a function of operational time must be taken into account.

5.12.4. Conditions for hydrotests, i.e. allowable hydrotest temperature $[T_h]$ and subsequent dependences [p]-[T] are determined in accordance with Section 5.9. Residual lifetime assessment is based on comparison of allowable dependences [p]-[T] (dependent on material transition temperature) with their limit ones. Usually neither normal operating conditions nor hydrotests are limiting for the lifetime of a component.

5.12.5. Residual lifetime of the component based on emergency conditions and anticipated operating transients is usually limiting for the whole residual lifetime. For a reactor pressure vessel the most important part is the beltline region but other vessel regions, e.g. nozzles, must be also taken into account.

Evaluation must be performed for all design bases regimes of EC and AOT. Residual lifetime is determined on the basis of maximum allowable transition temperature for static crack initiation, $[T_t]$, and from the trend in the change of material transition temperature T_t and from neutron fluence increase during the operation.

6 Residual lifetime of the component from the point of view of resistance against fatigue damage

The residual lifetime assessment of the component from the point of view of fatigue damage shall be carried out according to standards accepted by national Regulatory Authorities.

6.1. The residual lifetime assessment of the component

6.1.1. From the point of view of fatigue damage shall be carried out in two steps. Until initiation of the macro-crack of size 2.0 mm occurs for the loading condition according to 4.3.4, i.e. as long as the criterion of $D \leq 1$ is fulfilled.

6.1.2. For the phase of potential growth of the "hypothetical crack" which could escape detection by non-destructive testing. The procedure described in Chapter 8 shall be used for this step. Usage factor D shall be calculated according to standards accepted by national Regulatory Authorities.

6.2. Selection of assessed areas of the component

6.2.1. Based on the calculations included in the supporting documentation supplied by the manufacturer of the component, or in accordance with the Pre-Operation Safety Analysis Report. The selection shall be specified prior to operation of the NPP and it should be supported by monitoring using I&C, MDS and SM systems. Assessment shall be executed at the areas of the component for which the following value of the design usage factor was determined (in the supporting documentation supplied by the manufacturer):

$$D \geq 0.3 \tag{6.1}$$

6.3. Use of actual loads and actual series of operation modes

6.3.1. From the point of view of resistance against fatigue damage, the actual loads and actual series of operation modes shall be taken into account, in accordance with 3.3.11 and 4.3.3.

6.3.2. For calculation of residual lifetime of component, the usage factor is determined as a sum of individual usage factors calculated for single load blocks created from appropriate operational transients. The usage factor calculated for the load block created from all transients shall not be used.

6.4. Use of design calculations

6.4.1. In fatigue assessment if the differences between actual and design parameters of operational modes are not higher than 10%. In case that numbers of applied or extrapolated (predicted) modes exceed the design numbers of modes, it is necessary to take this fact into consideration. In case that historical operational modes cannot be classified (in any case) as design operation modes, it is necessary to perform calculation of fatigue damage with these operational modes.

6.5. Simplifation of the calculation

6.5.1. Construction of a "qualified" trend in operation loading is admissible, including the sequences of operational modes and their occurrence, on the basis of the results of the monitoring by I&C, MDS and SM systems. Calculation for these operational loads shall be executed in accordance with Section 6.1.

— The trend in operation loading including sequences and frequencies of operational modes shall be considered up to next periodic evaluation (or up to the end of design lifetime).

— For determination of fatigue damage, conservative assessment from design documentation shall be used (considering actual number of modes).

6.6. High value of the usage factor

If the calculation performed during operation of the component (for the design lifetime of the component) demonstrates non-fulfilment of the condition

$$D \leq 0.8 \tag{6.2}$$

then the following procedure shall be applied:

6.6.1. Perform a conclusive non-destructive test of the mentioned area up to the time of reaching equality in the condition (6.2). If a flaw due to fatigue is identified, it shall be schematised and further assessed in accordance with the procedure described in Chapter 8. **If the calculation performed during operation of the component** (for the design lifetime of the component) demonstrates non-fulfilment of the condition

$$D \leq 1 \tag{6.3}$$

then the following procedure shall be applied:

6.6.2. Perform a conclusive non-destructive test of the mentioned area up to the time of reaching equality in the condition (6.2). If a flaw due to fatigue is identified, it shall be schematised and further assessed in accordance with the procedure described in Chapter 8.

6.6.3. If a non-destructive testing of certain area is unfeasible for determination of the actual size of the defect, then parameters of a hypothetical crack are as follows:

$$a_{hyp} = 0.1\ s,\ \ a/2\ c = 1/6 \tag{6.4}$$

or if no flaws were identified, the semielliptical "hypothetical starting crack" shall be defined, of the size, which can be with high probability by NDE .

Assessment of crack admissibility shall be carried out in accordance with Chapter 8, including assessment of crack growth due to repeated mechanical and corrosion-mechanical loading.

6.7. Assessment of residual lifetime of the component from the point of view of resistance against fatigue damage

6.7.1. The resistance against fatigue damage during the assumed technical lifetime of the component, as given for example in the component design, is reached when the condition (6.2) is fulfilled in any of assessed areas of the component.

6.7.2. If the condition (6.2) is not fulfilled during the whole technical lifetime of the component, then either an assessment of allowed growth of the "hypothetical crack" shall be carried out, or an assessment of allowed growth of the flaw detected by non-destructive testing shall be carried out. This assessment shall be elaborated in accordance with the procedure described in Chapter 8 of this procedure. If allowance of these flaws is proven, it is considered as an evidence of resistance of the component against fatigue damage. Appendix VII describes recommended good practices in residual lifetime fatigue evaluation. Appendix VIII describes general recommendations for piping and components temperature measurement.

7 Residual lifetime of the component from the point of view of resistance against corrosion-mechanical damage

— Assessment of residual lifetime from the point of view of resistance against corrosion-mechanical-damage is executed in accordance with the procedures of Appendix IX. The acceptability of time evolution of any flaw that could be formed, grow and propagate under constant load (stress corrosion cracking), variable loading (corrosion fatigue) and/or time dependent combination of both of them in the water coolant environment in the operation conditions is evaluated. The actual loads and influence of the actual chemical admixtures are to be taken into account for calculation.

— If the stress fields due to operating loadings do not differ more than 10 % from the stresses given in the supporting documentation supplied by the producer of the component (or the associated sub-suppliers), then the stress fields from the supporting documentation can be used.

— To simplify calculation it is admissible to construct a "qualified" trend of operational loading, including also the series of operation modes and their occurrences, based on results of monitoring by the I&C and MDS systems. Then the stress fields and their time changes are to be calculated for the above-mentioned operational loads.

— If the assessment demonstrates a possibility of initiation of the flaw capable of growth in the stress corrosion conditions already during the assumed technical lifetime of the components (which given for example in the project), it is necessary to perform the following:

— Perform non-destructive testing of the area where flaw initiation and growth due to stress corrosion cracking is possible.

— In the case that the non-destructive test of the area is not feasible or if no flaw was found during non-destructive testing, a "hypothetical initial crack" in the form of a semielliptical surface crack is to be assumed, with the following semi-axes:

$$a_{hyp} = 0.1\ s \qquad a/(2c) = 1/6 \qquad (7.1)$$

The limit value a_{hyp} is to be determined in accordance with the wall thickness s:

$$a_{hyp} = 5\ \text{mm for} \quad s \leq 50\ \text{mm},$$

$$a_{hyp} = 30\ \text{mm for} \quad s \geq 300\ \text{mm} \qquad (7.2)$$

Then assessment of acceptability is to be done for this assumed crack in conditions of stress corrosion in accordance with procedures given in Appendix IX and XII.

— For the austenitic piping, the "hypothetical initial crack" is to be taken as a semielliptical inner surface crack with semi-axes:

$$a/(2c) = 1/6 \qquad (7.3)$$

The ratios a/s for different values of wall thickness are given in Table 7.1.

Table 7.1 Allowable depths of defects in austenitic piping.

Wall thickness s [mm]	**a/s** [%]
10	50.0
25	20.0
50	10.0
75	10.0
100 to 300	10.0

7.1. Residual lifetime of the component from the point of view of resistance against corrosion-mechanical damage is demonstrated if, during the assumed technical lifetime period (for example, according to design) of the component, no macro-flaw capable of growth in stress corrosion conditions originates. The limit value of depth of such a macro-flaw is 2.0 mm.

7.2. If the requirement of the previous paragraph cannot be met, then assessment must be performed for a postulated crack in accordance with this chapter. The resistance of the construction against corrosion-mechanical damage is adequately demonstrated if it is shown that conditions necessary for growth of a macro-flaw are absent for the whole assumed technical lifetime of the component.

7.3. If the requirement of the previous paragraph is not fulfilled and growth of a macro-flaw in the stress corrosion conditions occurs, then only temporary operation of the component is to be allowed, only for the necessary period and only with the special permission of the supervising body after supplying a supporting justifiable assessment.

7.4. Assessment of residual lifetime from the point view of resistance against flow-accelerated corrosion is carried out in accordance with procedures in Appendix XVII.

8 Assessment of acceptability of flaws found during in-service inspections and of residual lifetime of component with flaws

Any flaw found during in-service inspections is to be schematised in accordance with the procedure shown in Appendix X.

The flaws in components, ferritic or austenitic piping schematised in this way are to be compared with the Tables of allowable sizes of flaws, which are given in Appendix XI. Flaws that do not exceed the schematised size requirements prescribed in the Tables are allowable, and it is not necessary to continue with their assessment.

The flaws that do not fulfil some of the requirements prescribed in the Tables must be assessed in accordance with the appropriate Appendix as follows: flaws in components – Appendix XII, flaws in austenitic piping – Appendix XIII, flaws in carbon steel piping – Appendix XIV.

The parameters of the realised operation modes are to be used in this assessment (pressure, temperature, water chemistry) including their sequences, and for EC also their design courses.

In calculation of fatigue and corrosion-mechanical growth of the cracks, it is necessary to re-calculate all previously performed (in the frame of the supporting documentation or the Pre-Operation Safety Report) computations of the temperature and stress fields by incorporating the following operation conditions:

— Actual temperature-pressure course of single operation modes, including the actual water regimes,
— Actual sequences of operation modes.

Calculation of possible growth of the flaws is to be performed in accordance with Appendix XII (components) or Appendix XIV (carbon steel piping).

If the stress fields of the actual operation modes do not differ more than 10 % from the computational fields given in the supporting documentation, then it is possible to base the assessment of acceptability of flaws on calculation results given in the supporting documentation. In the case when number of run-off or predicted modes exceeds the design number of modes, this fact has to be taken into account.

Based on past operation modes, their sequences, operation practices, and computational blocks of modes, the "qualified" trend of operational loading of the component during the whole period of design lifetime is to be prepared. Consequently, assessment in accordance with Appendix XII or Appendix XIV is to be carried out for this trend.

8.1. Assessment of residual lifetime of the component with flaws

8.1. 1.The residual lifetime of the component with the flaws detected during in-service inspections is ensured if the detected flaws are smaller than the flaws shown in the Tables of allowable sizes of the flaws.

8.1.2.If the condition given in the previous paragraph is not fulfilled, then it is necessary to use the procedures of the appropriate Appendix (XII, XIII or XIV).

8.1.3. From the point of view of acceptability of flaws detected during in-service inspections, the residual lifetime of the component is defined as the period for which the validity of 1) and 2) is ensured.

8.1.4.New evaluation of previously detected and evaluated flaw is not required, as long as its growth has not been detected and provided that the conditions of original evaluation have not been changed.

9 Complex assessment of residual lifetime

The assumed technical lifetime of a component, given for example by its design, is ensured by the successful completion of the assessments required by Chapters 5–8.

Residual lifetime of the component is to be based on the shortest residual lifetime determined by assessments executed in accordance with Chapters 5–8.

If the period is shorter than the period given, for example, by design, then it is necessary to take the appropriate measures for operation management and maintenance, in accordance with the "Programme for life management of the components of the nuclear power plant".

List of figures

List of tables

APPENDIX I

Structure of the report assessing residual lifetime of the component

The report on residual lifetime assessment for component has to contain all data (at least in an abbreviated form or, as appropriate, references to associated and other accessible documentation related to the component), which is necessary for the evaluation of procedures and results of the assessment. The quantity of the data presented must be sufficient for passing judgement on the obtained results.

1. An overview and analysis of the fate obtained from the construction, production, and assembly documentations

1.1. The design drawings of the component

1.2. The determined chemical composition of materials

1.3. The accessible mechanical properties of the materials

1.4. The results of strength calculation based on the associate documentation

1.5. The drawings of the actual component including its dimensions and deviations

1.6. The data on production history

1.7. The data on flaws determined during production of the component

1.8. The data on assembly

1.9. The data on all deviations and repairs during production and assembly

1.10. The data on welding and thermal processing

2. An analysis of pre-operation results and in-service inspections

2.1. The range of the inspected areas and description of the used methods

2.1.1. The range of inspection of welded joints

2.1.2. The range of inspection of the basic material

2.1.3. The range of inspection after repairs during production/assembly

2.2. The type, size, and location of the determined flaws

2.3. An analysis of the fulfilment of the requirements for the applied qualification procedures in regards to detection, determination of flaw dimensions, and inspection procedure

2.4. Supplementary determination of the chemical composition (if applicable)

2.5. Other information

3. An overview of history of operation

3.1. The total period of operation

3.2. The course of single operational modes (temperature, pressure)

3.3. The series of single operation modes

3.4. The radiation load of the reactor pressure vessel material

3.5. An overview of the chemical parameters of the primary circuit coolant

3.6. The records on the relevant operation problems of the component

4. Assessment of material degradation

4.1. Determine the level of degradation due to single ageing mechanisms

4.1.1. Radiation damage (the results of the surveillance samples or the standard data)

4.1.2. Thermal ageing (the results of the surveillance samples or data from the computing model)

4.1.3. Fatigue damage (calculated)

4.1.4. Corrosion-mechanical damage (computed using empirical data, etc…)

4.2. Determine the allowable values of the stress intensity factor (temperature dependence) including the safety coefficients

5. Assessment of the lifetime of the component according to resistance against brittle/sudden failure if applicable

5.1. Identify the size and location of postulated cracks

5.2. Select the transients to be used in calculation

5.3. Determine the load based on the selected transients

5.3. Execute the temperature and stress analysis

5.4. Determine the magnitude of the stress intensity factor KI (or J-integral) of the postulated crack

5.5. Determine the maximum allowable transition temperature

5.6. Determine the [p]-[T] curves, and the allowable hydrotest temperature

5.7. Compute the maximum safety operation period

6. Assessment of acceptability of the determined flaws of the determined flaws and the assessment of component integrity until its end of life/next inspection if applicable

6.1. Compare the allowed values of the stress intensity factor [KIC]i for single operation modes and the values of the stress intensity factor KI computed for the schematised flaw 6.1.1. Determine the safety margin

6.1.2. Compare the safety margin and the required value of the safety coefficient

6.2. Calculation of the maximum safety operation period with the given flaw of the initial size, including its prospective growth, and the assumed degradation of the material properties

6.2.1. Compute the flaw rate of growth

6.2.2. Compute the rate of changes of the material properties due to degradation mechanisms

6.2.3. Compare the growth trend of the stress intensity factor KI with the allowed values of the stress intensity factor [KIC]i

6.3. Determine the maximum safety operation period until the next inspection/decision on necessity of the flaw ´s repair

7. Assessment of fatigue lifetime if applicable

7.1 Select the operation modes for use in the assessment
7.2 Compute the size of fatigue damage in single areas of the component
7.3 Compare the computational values of fatigue damage with the allowed values, and identify locations where the condition is not fulfilled
7.4 Identify potential locations for the initiation of fatigue macro-cracks
7.5 Define the size and form of the fatigue macro-cracks
7.6 Compute fatigue growth of the defined macro-cracks
7.7 Compare the final sizes of these macro-cracks with the allowed values
7.8 Determine the maximum safety operation period with the flaws of the fatigue crack type. Take into account the possible growth of the crack during operation and the assumed degradations of the material properties

8. Assessment of the corrosion-mechanical lifetime if applicable

8.1 Select the operation modes to be used in the computation
8.2 Compute the size of the corrosion-mechanical damage in single areas of the component
8.3 Identify the size of corrosion-mechanical damage in single areas of the component – determine the period before crack initiation
8.4 Compare the computational values of corrosion-mechanical damage with the allowed value. Identify locations where this condition is not fulfilled
8.5 Identify locations with the potential for the initiation of the corrosion-mechanical macro-cracks
8.6 Define the size and form of the corrosion-mechanical cracks
8.7 Compute the growth of the defined macro-cracks under both static and repeated loading
8.8 Compare the final macro-crack sizes with the allowed ones
8.9 Determine the maximum period of safety operation with the flaw of corrosion-mechanical type. Include possible flaw growth during operation and the assumed degradation of the material properties

9. Recommendation for the next steps

9.1 Recommend continued in-service operation, if all conditions are fulfilled
9.2 Recommend the necessary repairs
9.3 Recommend the necessary exchanges

10. Recommendation for the modification of the next in-service inspection

10.1 Determine the required scope

10.1.1 Incorporation/expansion of the areas where the potential for flaw initiation exists and where a higher sensitivity of inspection is asked

10.1.2 Incorporation of the areas where repairs were executed

10.2 Recommend modification of the period between subsequent inspections

10.2.1 Use the results of the worst flaw determination (step 4) and determination of the critical flaw size

10.2.2 Estimate the residual lifetime

10.2.3 Compare the residual lifetime and the planned inspection intervals

APPENDIX II

Procedure for determination of neutron fluence in reactor pressure vessels and reactor vessel internals

Definitions, symbols and units

λ[s^{-1}] decay constant

$T_{1/2}$ [s] decay half-time, i.e. $T_{1/2} = ln2/\lambda$

$\sigma(E)$ [m^2] differential cross section or reaction probability

$\varphi(E)$ [s^{-1} m^{-2} MeV^{-1}] neutron spectrum (supposing time dependence separation)

RR [s^{-1}] reaction rate per one target nucleus, i.e.

$$RR = \int \varphi(E)\sigma(E)dE \tag{II.1}$$

A [s^{-1}] activity,

$A = \lambda N$, where N is number of radioactive nuclei (II.2)

APN [s^{-1}] activity in the time of the irradiation end and related to 1 nucleus of detection material.

$\Phi_{(E>0.5\ MeV)}$ [s^{-1} m^{-2}] neutron fluence rate for E>0.5 MeV. i.e.

$$\Phi_{(E>0.5MeV)} = \int_{0.5MeV}^{\infty} \varphi(E)dE \tag{II.3}$$

$F_{(E>0.5\ MeV)}$ [m^{-2}] neutron fluence for E>0.5 MeV over time period t, i.e.

$$F_{(E>0.5MeV)} = \int_{0}^{t} \Phi_{(E>0.5MeV)} d\tau \tag{II.4}$$

$\sigma_{(E>0.5\ MeV)}$ [m^2] effective reaction cross section for E>0.5 MeV, i.e.

$$\sigma_{(E>0.5MeV)} = \frac{\int_{0}^{\infty} \varphi(E)\sigma(E)dE}{\int_{0.5MeV}^{\infty} \varphi(E)dE} \tag{II.5}$$

or

$$\sigma(E>0.5\ MeV) = RR\ /\ \Phi(E>0.5\ MeV)\ . \tag{II.6}$$

1. General rules

1.1 Determination of neutron fluence in reactor pressure vessel and reactor vessel internals

An assessment of components significant for nuclear power plant safety is carried out periodically and forms the basis for a license for continued operation of the plant.

Determination of the fast neutron fluence exposure to the reactor pressure vessel (RPV) and reactor vessel internals (RVI) is fundamental to the estimation of RPV and RVI material properties degradation and, thus, to the RPV and RVI integrity assessment and residual life-time determination.

RPV and RI fast neutron fluence determination is a combination of analytical calculations and dosimetry measurements. Fast neutron fluence values in required RPV and RVI locations are determined using calculations. Calculation results are qualified with the use of measured results performed at the nuclear power installation and benchmark experiments. All necessary data are stored in a manner that would allow future re-evaluation with respect to methods development and more precise input data determinations.

Measurements are comprised of a series of limiting factors that follow both the nuclear-physical data of used monitors, and the technical possibilities of their installation at the nuclear reactor, mode of irradiation and the possibility of their retrieval. The calculation method, qualified with measurement results, makes possible the absolute neutron fluence value determinations in required locations for a given period of reactor operation. The calculation of RPV and RVI fast neutron fluence is a complicated task due to the complex problem of geometry, several orders decrease of fast neutron values from the reactor core to the RPV and RVI, input uncertainties in nuclear-physical data, etc... Experimental results (activity of monitors) are valuable because they provide independently determined values, serve as qualification benchmarks for calculation methods, and decrease the uncertainties associated with fluence calculations.

Technically it is almost impossible to measure in RVI besides cases of retrospective dosimetry of retrieved parts of internals. Therefore, all values of internals neutron fluence are determined with calculation methods.

1.2 RPV and RVI neutron fluence computational determination

The determination of the calculated neutron fluence in defined RPV and RVI locations includes 1) determination of geometry and material composition input data, 2) simulation of core cycles, 3) determination of neutron sources in the reactor core from space and energy point of view for the whole reactor fuel cycle/campaign for which RPV/RVI fluence calculation is carried out (taking into account fuel isotopic change due to its burn-up which causes changes of neutron spectrum and yields of neutron production and energy released per one fission), 4) neutron fluence and monitors activity calculation and 5) calculation procedure qualification which consists of calculation and experimental values comparison, eventual differences explanation and evaluation of calculation results uncertainties.

1.3 RPV neutron fluence experimental determination

The RPV neutron fluence experimental determination is carried out with the use of passive integral neutron fluence monitors situated in both the surveillance capsules and the reactor cavity. The experimental determination incorporates the monitors specification and use at the nuclear power plant, determination of monitor response, evaluation of results uncertainties, the reference values for calculation methods (activity calculated to the irradiation end), and the fast neutron fluence determination from the measured values.

1.4 Reporting, quality assurance and storage

Input data, methods and procedures used and obtained results of the RPV and RVI fluence (including uncertainties of used procedures) should be reported in an unambiguous manner and should reflect the real situation at which they were obtained. With respect to the long-term validity of these values, it is necessary to retain all documentation in proper storage during the whole reactor lifetime. Data are stored at two places in different regions in the form of reports and/or electronic media. All input data, procedures, and results of the evaluation are to be stored in a manner that makes it possible for their re-evaluation at any time.

2. RPV and RVI neutron flunce calculation method

Calculation of the RPV and RVI neutron fluence is carried out by the calculation of neutron fluence absolute values and consists of the following steps:

— Determination of geometry and material composition;

— Simulation of core cycles;

— Determination of neutron sources;

— Neutron fluence and monitors activity calculation;

— Calculation procedure qualification.

2.1 Geometry and material input data

For the determination of the neutron fluence rate decrease from the reactor core boundary to defined places, detail input data describing both particular geometry and their material composition are necessary. These data include individual parts description (reactor core, internals, RPV, reactor cavity, thermal shielding, concrete cladding, and biological shielding), their material composition, temperature distribution in regions, and geometry parameters of each region.

Geometry input data include dimensions and positions of the fuel assemblies, RVI, RPV cladding (including positions of welds), and surveillance capsules. For a comparison with activity measurements in the reactor cavity, data should also include cavity width, a description of monitors supporting equipment.

Input data are based on documented and verified data from the as-built installation. Material composition values for nuclides having an important role in the given described region are based on measured values. If these geometry data and/or composition measurements were not performed, it is possible to take the geometry data reported in the installation design and standard material composition. In that case, a sensitivity analysis is to be performed and results included in the calculation results uncertainties. Special attention is to be paid to the description of the regions containing the main isotopes which cause fast neutron fluence rate decrease (i.e., iron and water). Great attention is to be given to the detailed description of the capsules with surveillance specimens and their surroundings since data from the neutron fluence monitors situated in these capsules are used for the calculation procedure qualification. Water density corresponding to the temperature distribution for nominal reactor power is taken.

2.2 Neutron sources in reactor core determination

Determination of neutron sources in the reactor core includes the sources distribution from space, time, and energy point of view for the whole reactor fuel cycle for which fluence calculation is carried out.

2.2.1 Core cycle simulation

Core cycle simulation is carried out for every operated NPP unit cycle. Fixed neutron source for every operated cycle is determined using the results of the core cycle simulation calculation as a set of consequent steady-states used for simulation of the cycle power history. These results include steady-states intervals duration, relative power and burn-up values distribution in the core, for every steady-state step. Number of the steady-states and their durations has to be adequate for accurate determination of the fixed neutron source applied for calculation of RPV and RVI fluence monitors' activities. 3-D core criticality calculations are usually carried out at every steady-state in diffusion approximation for pin-wise, node-wise, and assembly-wise space description. More detailed core power and burn-up space description is preferable in the periphery fuel assemblies where significant radial gradient in power distribution is occurred. Number and size of axial segments used in criticality calculation have to provide adequate fixed neutron source description for fluence and monitors' activities calculation. Neutron cross section libraries used for core cycle simulation includes data describing dependence of fuel isotopic content, yields of neutron production and energy released per one fission on deepness of fuel burn-up. These data also is needed for fixed neutron source space distribution and spectrum calculation. Adequate transformation of the data from core cycle simulation calculation described for WWER in hexagonal geometry to geometry used for fixed source fluence and monitors activities calculation is also needed.

2.2.2 Neutron sources distribution in reactor core

Determination of neutron sources in the reactor core includes the sources distribution from space, time, and energy point of view for the whole reactor fuel cycle for which fluence calculation is carried out. The most frequent way is that a volumetric source distribution is calculated by a core design code. Another possible way can be to use the source on the surface of the core design boundary by excluding this inner region from the subsequent transport calculation. This latter method can be advised in case of an appropriate verification, which can be done - for example - by comparing the results to the more time-consuming volumetric source method.

Attention is paid to the peripheral region of the reactor core, which contributes mostly to the calculated fluence and in which the greatest radial gradient of neutron source occurs. Data describing the fuel and absorbers distribution should correspond with the real state of the operated core including axial profiling. The choice of time interval during the whole reactor cycle for individual calculation parts should reflect the influence of possible changes in this cycle significant for fluence and monitors' activity accurate calculation. Data measurement of real reactor power parameters and their distribution in this cycle should also be taken into account.

2.2.3 Neutron spectrum

The calculation should reflect the changes in fuel isotopic composition due to the fuel burn-up, which causes changes in the neutron fission spectrum, energy per fission, and changes in the yields of neutron production per one fission. This is important especially for a low leakage core, in which fuel assemblies with high burn-up are situated at the core outer layer.

2.3 Neutron fluence and neutron fluence monitors activity calculation

Deterministic neutron transport codes as well as codes based on the Monte Carlo transport method can be used for the neutron fluence calculation qualified for WWER. The approach to the calculations is based on the fixed source described in previous chapter.

It is possible to develop specific codes and libraries. The test of cross sections libraries, as well as the developed codes sensitivity analysis of input data influence on calculation results should precede their application.

2.3.1. Calculation approach

The calculation is carried out in 3-D geometry with a deterministic code or with codes based on a Monte-Carlo method. It is possible to use the approximation of obtaining a three-dimensional flux from combination of the one-dimensional neutron fluence rate calculation in R direction and the two-dimensional calculations in R-☐ and R-Z geometry [9].

$$\Phi(r,\theta,z,E) = \Phi(r,\theta,E)\frac{\Phi(r,z,E)}{\Phi(r,E)} \tag{II.7}$$

2.3.2. Cross sections libraries

The latest versions of the cross-sections libraries (for example ENDF/B, JENDL, JEF, etc.) should be used for the transport calculations. Multigroup cross-sections libraries are possible to obtain by the partial integration through neutron spectra close to real spectra, but before their use, it is necessary to verify them (e.g., by calculation comparison of benchmark tasks using the master and the group library).

2.3.3. Neutron fluence and neutron fluence monitors activity values calculation

The fluence calculation is carried out at the RPV cladding, inner surface of the, RPV, at the ¼ RPV thickness, and at the RPV outer surface. This is done in the maximum fluence and weld position in the axial direction, (and if necessary, at places where surveillance capsule specimens are situated). The positions in RVI and other positions in RPV should be specified by the VERLIFE fracture toughness procedure and correspond to requirements of regulators.

Neutron fluence monitors activity is calculated at the monitor position (surveillance capsules, ex vessel cavity, inner cladding scrape take-off positions by retrospective dosimetry, etc.).

Analogous methods for the RPV fluence calculation can be used inside and outside the reactor core for the region limited with bottom and top levels of the reactor core (or in their vicinity). Special attention has to be paid for the appropriate detailing of the geometry and for the validation if the fluence results are to be applied also in the regions in the axial elevation of the active core (upper and lower) boundaries (or farther from the core), where the axial gradient of the flux is significant.

2.4 Calculation procedure qualification

With respect to the complex calculation, a qualification of the method must be carried out by comparison with measured values in order to ensure a reliable and precise determination of RPV and/or RVI neutron fluence values. This calculation and measurement comparison must reveal eventual overestimation or underestimation of the calculation procedure results and must reliably evaluate calculation result uncertainties. If the data used are sufficient from both a quality and quantity point of view, they may be used for the evaluation of influence of used calculation model approximation on the calculation results, or for eventual evaluation of overestimation or underestimation of calculation results. These values are then used either for calculation model modification, or for calculation results adjustment, if needed. As an additional method of qualification, a sensitivity analysis of the used calculation procedure should be carried out with respect to important input parameters and simplifications used in the model.

Prior to the transport calculation the calculation methodology should be qualified/validated (with respect to technical possibilities and data availability). The qualification procedure consists of three steps:

— Analytical estimation of uncertainties;

— Comparison with measurement at power reactors and benchmark results;

— Total uncertainty estimation of calculated fluence values.

For determination of both neutron fluence and its uncertainty level it is possible to apply the least square adjustment procedure consisting in adjustment of the results using dosimetric measurements and their covariance data and uncertainties. This procedure (which is based on the generalised method of least squares) minimizes the output data variances, and its results are best estimates of the calculated parameters with uncertainties (e.g., Refs [36, 37]). This approach can incorporate integral data of a series of reference reactor dosimetry benchmark experiments, differential data used in the calculation of the flux distribution together with their associated uncertainties, and data on experimental surveillance dosimetry of the analysed reactor at the surveillance capsule or/and at ex-vessel cavity positions.

2.4.1. Analytical estimation of uncertainties

Analysis includes identification of parameters that are important for estimation of calculated values uncertainties, e.g.:

— Nuclear data;

— Geometry description;

— Isotopic composition;

— Neutron sources in the reactor core (precision of determination for periphery fuel assemblies, dependence on spatial and energetic distribution on fuel burn up);

— Method used (e.g. mesh density, angle dependence, convergence, macroscopic cross sections, spatial synthesis, influence of capsules with surveillance specimens, neutron "streaming" in reactor cavity).

Analysis of individual parameters influence is carried out by perturbation calculation and/or through performing series of calculations, in which the individual input data and approximation models are gradually changed, and numerical influence on fluence calculation values is determined.

2.4.2. Comparison with measurement at power reactors and benchmark results

Calculation methods should be qualified by comparing the calculated results with the experiment benchmark task results, e.g.:

— Tests data from the measurement with fluence monitors situated in surveillance capsules and in the reactor cavity;

— Tests results of benchmark experiments describing this reactor type;

— Benchmark test calculation tasks in which results calculated from different transport codes and developed codes are compared.

Significant differences between calculated and measured values (e.g., more than 20% related to measured values) should be explained.

2.4.3. Total uncertainty estimation of calculated fluence values

The total uncertainty estimation of the calculated value is carried out through a combination of analytic uncertainty estimation results and uncertainties evaluated from comparison with measurement results of benchmark experiments and power reactors measurements.

2.5 Results reporting, quality assurance and storage

The input data, the method used, the calculation procedure, the obtained calculation results, and a comparison with measured values data, uncertainties, and reference values availability should be reported in an unambiguous way.

2.5.1. Quality assurance

The calculation procedure of the RPV and RVI fluence determination should be in accordance with the required precision. The particular procedures used should be well reported so that the obtained results and the determined calculated result uncertainties reflect the real state at which the calculation was carried out. With respect to the long-term validity of these values, it is necessary to store all documentation for the whole reactor lifetime. To ensure data precision and completeness, and unambiguous access to them for the whole reactor lifetime, the following set of documents is elaborated for each calculation of the RPV/RVI fluence:

— Report on the geometry and the material data;

— Report on the neutron source distribution in the reactor core determination;

— Report on the used cross sections library;

— Report on the neutron fluence and monitors activity calculation;

— Report on the calculation procedure qualification;

— Final report on the calculation results;

— Description of the procedure for data access during their long-term storage.

It is not necessary to elaborate each report as an individual one. Reports may be combined and references to previous reports may be used for complex parts, which did not change.

2.5.2. Report on geometry and material data

This report describes the geometry and material data used as input data for the calculation together with a description of the sources from which they were obtained and a description of simplification against the real state.

2.5.3. Report on neutron source distribution in the reactor core

This report contains data on the computer code and cross section library used for core cycle simulation, and the way of receipt and qualification of the basic code version. Adaptation of the basic version from which this code was developed (e.g., for increase of precision of source distribution determination in fuel assemblies situated in the reactor core boundary layer) is described. A procedure for the method used is described including the connection with measured values obtained at the reactor during this cycle/campaign and the methods of receipt for these data. This report also contains a description of the input data simplification against the real core fuel load and reactor cycle/campaign course/history and the assessment of simplification influence on calculation results uncertainty. A methodology used for fixed neutron source determination from the core cycle simulation calculation results is presented. The data transformation from core cycle simulation calculation geometry to fluence/monitors activity fixed source calculation is described. Adequacy of the fixed neutron source course/history description for fluence monitors' activities calculation is demonstrated. Finally, the report contains an assessment of the reactor core fixed neutron source distribution symmetry validity, which is further used for the neutron fluence calculation.

2.5.4. Report on used cross section library

This report contains the source cross section library or the way of receiving the multigroup cross section library.

2.5.5. Report on fluence values and monitors activity calculation

This report contains description of the computer code, method of solution, description of the performed tests and their results (sensitivity analysis, benchmark tests, and qualification). The report contains a description of the neutron source results procedure use as input data for the calculation.

2.5.6. Report on calculation procedure qualification

This report contains data on the comparison of calculated values with values measured at this reactor for this cycle. Explanation and comment should be provided for significant differences (more than 20 %) between calculated and measured values.

2.5.7. Final report on calculation results

This report contains the calculation results and their uncertainties.

2.5.8. Description of data access during their long-term storage

This report describes the data storage description and the way to access the stored data.

2.5.9. Storage

Data are stored at two places in different regions in the form of reports or on electronic media. All input data, the procedures, and the results of evaluation are stored in a manner that permits their re-evaluation at any time. The following list should be taken only as the orientation one:

Stored data:

- Geometry and material power unit parameters;
- Fuel load chart with necessary material and physical data;
- Physical, material, and geometry input data;
- Used calculation method and way or source of its receipt;
- Used cross sections library and way or source of its receipt;
- Used procedure for determination of source distribution in the course of reactor core operation including sources values or input data for their calculation code;
- Comparison with measured data for given period;
- Evaluation results with uncertainty estimation.

3. RPV neutron fluence experimental determination

The RPV neutron fluence experimental determination (e.g., Refs [15, 16, 17]) is carried out with the use of passive integral neutron fluence monitors situated in the surveillance capsules and in the reactor cavity. The determination consists of the following steps:

- Monitors specification and use at nuclear power installation;
- Monitors response determination;
- Measurement results uncertainties determination;
- Reference values for calculation comparison;

— Fast neutron fluence determination.

3.1 Monitors specification and use at nuclear power installation

— The choice of fluence monitors and their use at the nuclear power installation (e.g. Ref. [18]) comes from the analysis of the following monitor materials properties and their irradiation, temperature, and environmental conditions:

— Irradiation places and irradiation frequency determination;

— Monitors choice on basis of their properties and availability;

— Technical procedure of monitors installation and retrieval from the unit after the end of irradiation.

3.1.1. Irradiation places and irradiation frequency determination

The monitors are installed in the surveillance capsules and the reactor cavity.

Surveillance capsule retrieval from the reactor is in accordance with the surveillance programme. It is recommended to install capsules with fluence monitors into places where surveillance capsules have been removed from (e.g., after the surveillance programme ends, when a long period between surveillance capsules retrieval occurs, or a significant change of reactor fuel load occurs).

The neutron fluence measurement in the reactor cavity is performed for all reactor cycles. At the end of a given cycle when a surveillance capsule with specimens is retrieved, it is recommended to perform a measurement in reactor cavity in the corresponding geometry sector in this cycle. Measurement of the azimuthal neutron fluence distribution should cover one segment of the reactor fuel load symmetry and is performed at the core midplane and/or at the level of the circumferential weld at the bottom core region. Vertical distribution should be measured over the whole attainable core height. The dosimetry sets should be placed at the azimuthal maximum and minimum as well, at levels of the weld of the core region.

Other conditions, which should be taken into account, include temperature and environmental parameters.

Samples of the cladding material (extracted during retrospective dosimetry) should be extracted in those axial positions of inner surface of RPV where the neutron fluence is monitored both in the reactor cavity (on the outer surface of RPV) and/or in the surveillance capsule.

3.1.2. Monitors choice on basis of their property and availability

Selection of monitor types is based on the detection material physical properties, the monitor material form availability, and the monitor materials mechanical properties, irradiation period and conditions.

3.1.3. Detection material specification

Activation or fission fast neutron fluence monitors are used with threshold energy of a given interaction. Monitors set determination is carried out in a manner that covers the whole energy interval. Fast neutron fluence monitor sets are complemented with monitors for the determination of thermal neutron fluence in order to determine the influence of thermal neutron interference reactions. The material choice also takes into account the relationship between arising radioisotope half-life, and the influence of possible changes in the neutron fluence rate in the place of fluence monitor position. A review of recommended materials of fast neutron fluence monitors and their basic physical data are presented in the Attachment A (possible references are [19–28]).

3.1.4. Chemical and isotopic composition

Monitors are used in the form of pure material, compound metals, or alloys. It is necessary to know the precise content of the detection isotope and to keep to a minimum the acceptable content of ingredient materials, in order that interaction with them would not significantly influence the monitor response.

3.1.5. Monitors encapsulation

The technical design of the monitor encapsulation and holder should minimize the possible influence on monitors response and the influence on the neutron spectrum. Sets of fast neutron fluence monitors are shielded against low-energy neutrons.

3.1.6. Realisation of technical procedure of monitors installation and retrieval from the unit after the end of irradiation

Monitors are situated in the surveillance capsules and in the reactor cavity. The precise position of each monitor should be known and documented.

1.1 Monitor response determination

Values obtained from the fluence monitors response are:

- Monitors activity per one nucleus of target isotopes calculated to the irradiation end;
- Mean reaction rate.

The monitor response determination consists of the following steps (possible references are [15], [16], [29]):

- Monitors activity measurement including corrections coming from used measurement spectrometric string;
- Application of corrections taking into account course of monitors irradiation and cooling;
- Determination of values for neutron fluence calculation;
- Determination of neutron fluence values from measurement with fluence monitors.

3.2.1. Monitor activity determination

A spectrometric string with a particular γ and/or X-ray radiation detector is used for monitor activity measurement. The string is calibrated by set of radioactive standards. The measurement precision should be up to 5 %.

When activity per 1 nucleus of detection monitor material from measured activity is determined the following parameters are taken into account:

- Measured count rate;
- Correction on detection efficiency;
- Correction on measured gamma/X-ray radiation yield in decay scheme;
- Correction on nuclei number in monitor;
- Correction on gamma/X-ray radiation self-absorption in monitor;
- Correction on cooling after the irradiation end;
- Correction on background and interference reactions;
- Correction on photo fission at fission monitors.

3.2.2. Irradiation period and course influence

For reaction rate evaluation the course of neutron fluence rate correction in the place of fluence monitors position should be incorporated.

3.3. Reference values for calculation comparison determination

3.3.1. Reference values

Primary reference value for comparison and qualification of the calculation method are activity per nucleus *APN* – activity in the time of the irradiation end and related to 1 nucleus of detection material. Experimental neutron fluence is determined either from APN or from reaction.

Under the supposition that the time course of the neutron fluence rate is possible to divide into I partially constant parts in intervals <t_{i-1} , t_i >, i = 1, …, I, and the spectral averaged cross section is also constant over all irradiation periods, then the reaction rate depends only on the magnitude of neutron flux. The relation between APN and RR can be written as

$$APN = \sum_{i=1}^{I} RR_i K_i \, , \qquad \text{(II.8)}$$

where

$$K_i = \{ 1- \exp[-\lambda (t_i - t_{i-1})]\} \exp [-\lambda (t_I - t_i)], \quad i = 1,\dots,I \qquad \text{(II.9)}$$

t_0 is the time of start of irradiation, t_I is the time of end of irradiation.

In case that a neutron flux Φ_s (corresponding to the reference reactor power) and corresponding reaction rate RR_s are introduced, the RR_i can be express according to the following expression

$$RR_i = \frac{\Phi_i}{\Phi_s} RR_s \, , \qquad \text{(II.10)}$$

where Φ_i is averaged neutron flux of the i-th irradiation interval which takes into account the local changes caused by power distribution changes in the core.

The relation between APN and RR_s is therefore as follows:

$$RR_s = \frac{(APN).\Phi_s}{\sum_{i=1}^{I} \Phi_i K_i} . \qquad \text{(II.11)}$$

3.3.2. Measurement results uncertainty determination

Variance and covariance of measured activities and reaction rates are determined with classic statistic methods on a basis of knowledge of partial variances and covariances [30] covering:

— Nuclear data;

— Nuclei number in monitor;

— Efficiency of gamma/X-ray radiation detection;

— Self-absorption of gamma/X-ray radiation in monitor;

— Interference reaction influence;

— Irradiation course for RR.

3.3.3. Neutron fluence determination from monitor measurements

Final determination of fast neutron fluence from RR is based on the combination of neutron spectrum and fluence monitors measurement results.

Neutron fluence values and variances and covariances are determined directly from the evaluated reaction rates by spectrum adjustment with the appropriate code (e.g. SANDBP [33], STAY'SL [34], BASACF [35], LSL-M2 [38]). Required integral values can be calculated from the resulted neutron spectrum. A spectrum received at benchmark measurement or calculated with a transport code is used as an input spectrum.

Reaction rates can be used for the estimation of neutron fluence rate $\Phi_{(E>0.5\ MeV}$ using effective cross sections according to the equation

$$\Phi_{(E>0.5MeV)} = \frac{RR_s}{\sigma_{(E>0.5MeV)}} \qquad \text{(II.12)}$$

where

RR_s mean reaction rate based on measured APN and including influence of neutron fluence rate course during irradiation in the place of fluence monitor position

$\sigma_{(E>0.5\ MeV)}$ [m^2] reaction effective cross section for E>0.5 MeV defined in (II.5).

For determination of effective cross section two following spectra can be used:

— Spectrum calculated with a transport code;

— Spectrum measured at benchmark experiment.

Experimental neutron fluence rate $\Phi_{(E>0.5\ MeV)}$ estimation is also carried out using only measured *APN* according to the equation

$$\Phi_{(E>0.5\ MeV)} = (APN_{exp}/APN_{cal})\ \Phi_{cal}(E>0.5\ MeV), \qquad \text{(II.13)}$$

where,

APN_{exp} measured APN,

APN_{cal} calculated APN,

$\Phi_{cal}(E>0.5\ MeV)$ calculated fluence rate.

The result obtained from Equation II.13 coincides with the result from (II.12) if calculated spectrum is used in (II.5). The right side of the expression (II.13) for the most important monitors with threshold close to 0.5 MeV is stable against variation of the spectrum shape. That means that when in (II.5) experimental spectrum is used a good consistency between neutron fluence rate assessment by (II.12) and (II.13) could be expected too.

For calculation at other RPV places than measurement places, coefficients determined at benchmark experiments, from retrospective dosimetry, or calculated are used.

3.3.4. Neutron fluence uncertainty values from fluence monitors measurement determination

Uncertainty of the measured fluence evaluated from RR is determined on the basis of knowledge of following parameters variances and covariances:

— Reaction rate / APN;

— Neutron input spectrum used for the evaluation;

— Differential cross sections;

— Attenuation coefficient.

3.4. Results reporting, quality assurance and storage

3.4.1. Reporting

Input data, methods and procedures used, fluence results for the RPV, and uncertainties should be reported in an unambiguous manner.

3.4.2. Quality assurance

All particular procedures should be well reported so that the obtained results and their uncertainties reflect the real situations at which they were obtained. With respect to the long-term validity of these values, it is necessary to store all documentation during the whole RPV lifetime. To ensure data precision, completeness, and unambiguous access to them during the whole RPV lifetime, the following document set is elaborated for each RPV fluence determination:

— Report on used detection material origin and purity;

— Report on monitors sets preparation;

— Report on monitors installation;

— Report on monitors irradiation;

— Report on reactor coarse;

— Report on procedure used for activity/reaction rates evaluation and results;

— Final report on measurement results (neutron fluence evaluation);

— Description of data access during their long-term storage.

3.4.3. Report on used detection material origin and purity

This report contains data on detection material origin, material and impurities content certificate and material form for neutron fluence monitors preparation.

3.4.4. Report on monitors set preparation

This report contains data on the form and shape of each monitor and their mass together with uncertainty of mass determination. The report also contains data on produced monitors sets, data on materials of shielding against low-energetic neutrons, and data on the material and way of monitor's encapsulation.

3.4.5. Report on monitors installation

This report contains data on the carrier (capsule) for the monitor's installation at the reactor and on the position and orientation of monitors in this carrier (capsule).

3.4.6. Report on monitor's irradiation

This report contains data on the installation of the carrier (capsule) with monitors at the reactor capsule or in the cavity and on irradiation conditions.

3.4.7. Report on reactor coarse

The period of the irradiation of fluence monitors is divided into several irradiation intervals. This report contains the mean calculated neutron fluence rates over the irradiation intervals. Each value takes into account the local changes caused by power distribution changes in the core during this interval.

3.4.8. Report on procedure used for activity/reaction rates evaluation and results

This report contains a description of the methods used for activity/reaction rates evaluation and the data starting from monitors (capsule) retrieval till the evaluation of monitors activity/reaction rates and uncertainties.

3.4.9. Final report on measurement results

This report contains the measurement results and their uncertainties together with all data used for their determination.

3.4.10. Description of data access during their long-term storage

The report describes the storage procedure and the way for accessibility of stored data. It is not necessary to elaborate each report as an individual one. Reports may be combined and references to previous reports may be used for complex parts, which did not change.

3.4.11. Storage

Data are stored at two places in different regions in the form of reports or on electronic media. All input data, procedures, and results of evaluation are stored in a manner that permits their re-evaluation at any time. The following list should be taken only as an example:

3.4.12. Stored data:

— Geometry and material power unit parameters;

— Fuel load chart with necessary material and physical data;

— Physical, material and geometry data on used monitors;

— Geometry and time data on measurement performed at the power unit;

— Irradiation time course;

— If the measurement in the reactor cavity does not cover the whole RPV perimeter, then data proving the power distribution in the reactor core symmetry;

— Time and technical description of measurement and monitors activity determination;

— Used correction and the procedure for their determination;

— Procedure and input data evaluation of monitors activity to neutron fluence values;

— Evaluation results with uncertainty estimation.

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APPENDIX III

Assessment of material properties degradation

Definitions, symbols and units

T_{k0}	Initial critical temperature of brittleness, [°C]
T_k	Critical temperature of brittleness, [°C]
T_0	Reference temperature from "Master Curve" approach, [°C]
ΔT_F	Shift of the critical temperature of brittleness due to irradiation, [°C]
ΔT_T	Shift of the critical temperature of brittleness due to thermal ageing, [°C]
ΔT_t	Shift of the critical temperature of brittleness due to thermal ageing under irradiation conditions, [°C]
ΔT_N	Shift of the critical temperature of brittleness due to cyclic damage, [°C]
A_F	Irradiation embrittlement coefficient (mean value), [°C]
n	Exponent of irradiation embrittlement
D	Fatigue usage (damage) factor
σ_1	Margin due to limited number of specimens, [°C]
δT_M	Margin due to scatter of material properties, [°C]
δT_m	Margin due to scatter of material properties and embrittlement effects, [°C]
σ	Resulting margin, [°C]
RT_0	Reference temperature (not to be confused with MC reference temperature T_0) based on T_0 approach, [°C]
RT_k	Reference temperature, based on T_k approach, [°C]
t	Time of ageing [hours]
KV	Impact notch energy, [J]
F	Fluence, [m^{-2}]
K_{Jc}	Static fracture toughness, [$MPa.m^{1/2}$]

1. Reference temperature T_0

1.1. Reference temperature T_0 is determined from static fracture toughness tests using a single- or multiple-temperature „Master Curve" approach in accordance with the standard ASTM E 1921-05. Then, a chosen lower tolerance bound (usually 5%) should be applied for determination of fracture toughness temperature dependence to be used in integrity/lifetime calculations.

1.2. In principle, transition temperature T_0 is usually determined for a fluence required for the RPV integrity assessment, i.e., for end-of-life fluence or for extended life fluence. In these cases, one set of specimens is sufficient for the determination of value T_0 for a required neutron fluence. In the case when required fluence is not available, interpolation between two fluences and their transition temperatures T_0 can be applied using fluence dependences as in formula (8).

1.3. Transition temperature T_0 for end-of-life (extended life) fluence can be also determined from its initial value by adding the appropriate shift of critical temperature of brittleness ΔT_F in accordance with Equation III.4.

1.4. Similarly, this temperature could be determined also for a given time of ageing, i.e., for characterisation of thermal ageing of materials.

1.5. Reference temperature, T_0, as determined in accordance with the standard ASTM E 1921-05 is increased by a margin σ, i.e. either in initial or for a given degradation state. Reference temperature T_0 is defined from experimentally determined values of static fracture toughness, KJC, adjusted to the thickness of 25 mm. Margin is added to cover the uncertainty in T_0 associated with using of only a few specimens to establish T_0. The standard deviation σ_1 of the estimate of T0 is given by:

$$\sigma_1 = \beta / N^{0.5},\ ^\circ\mathrm{C} \tag{III.1}$$

where N = total number of valid data-points used to establish the value of T_0,

$$\beta = +\ 18\ ^\circ\mathrm{C}.$$

For determination of the guaranteed value T_0 with 95% probability it is necessary to introduce

$$\sigma_2 = 1.64\frac{\beta}{\sqrt{N}},\ ^oC \tag{III.2}$$

To consider the scatter in the material properties (within the component), another margin denoted in what follows δT_M should be applied. If this value is not available, the application of the following values is suggested

$$\delta T_M = 16\ ^\circ\mathrm{C} \tag{III.3}$$

Other values of δT_M should be justified and agreed with the national Regulatory body.

The resulting margin is:

$$\sigma = (\sigma_2{}^2 + (1.64\delta T_M)^2)^{1/2} \tag{III.4}$$

Thus, reference temperature when used in integrity evaluation, RT_0, is defined as:

$$RT_0 = T_0 + \sigma + \delta T_{type} \tag{III.5}$$

$$\delta T_{type} = \begin{cases} 15^oC & if\ \ precracked\ Charpy\ specimens\ are\ used \\ 0^oC & if\ CT\ specimens\ are\ used \end{cases} \tag{III.6}$$

2. Critical temperature of brittleness T_k

To determine the temperature dependences of fracture toughness K_{IC}, K_{IA}, respectively, the critical temperature of brittleness T_k may be used.

2.1. The critical temperature of brittleness T_k is usually determined for the time corresponding to the designed end of life. If admissibility either of the assessed mode or of the flaw or the temperature $[T_t]$ is not reached, it is admissible to determine this temperature for a shorter time interval. For example, this temperature can be determined for the period until the next in-service inspection or until a certain time point between this inspection and the designed end of life.

2.2. The critical temperature of brittleness Tk is given by the following relationship:

$$T_k = T_{k0} + \Delta T_F + \Delta T_T \ , \qquad \text{(III.7)}$$

where

T_{k0} - initial critical temperature of brittleness, [°C]

ΔT_F - shift of the critical temperature of brittleness due to irradiation, [°C]

ΔT_T - shift of the critical temperature of brittleness due to thermal ageing, [°C]

2.3. In principle, the values of the initial critical temperature of brittleness T_{k0} equal to the values confirmed for the given material by the corresponding Technical Requirements for a given material $T_{k0}^{guaranteed}$ are used for assessment of residual life of the component, if no actual values from component material Acceptance Tests (based on component Passport) are available.

In this case

$$RT_{k0} = T_{k0}^{guaranteed}$$

In such case it is allowed to perform testing non-irradiated control surveillance specimens with the determination of critical temperature of brittleness T_{k0} by a methodology approved by the Regulatory body.

2.3.1 *If the experimentally determined values of the initial critical temperature of brittleness T_{k0}* from component Acceptance Tests are known (based on component Passport) or supplementary testing in accordance with Section 2.3, they can be used only in the case that the following temperature margin δT_M will be added; the margin has to take into account the scatter of the values of mechanical properties in the semi-product; δT_M

δT_M is the mean quadratic deviation of T_{k0} determined for the given semi-product in the frame of Qualification Tests or in the frame of a set of identical semi-products established during production of the component by the identical technology. If this value is not available the application of the following values is suggested

$$\delta T_M = 16\ °C$$

This resulting margin should be applied with $\sigma_1 = 0$, i.e.

$$RT_{k0} = T_{k0} + 1.64\ .\ \delta T_M \qquad \text{(III.8)}$$

When RT_{k0} is larger than $T_{k0}^{guaranteed}$, then

$$RT_{k0} = T_{k0}^{guaranteed}.$$

3. Determination of the effect of irradiation embrittlement

3.1. The shift of the critical temperature of brittleness due to irradiation can be determined by the two following procedures:

3.1.1. Results of tests of the surveillance programme for specimens of the material of the vessel are available (respectively, also results for other vessels containing identical materials - for example, identical heat of the welding wire and flux):

3.1.2. If material is not sensitive to thermal ageing, then shift of the critical temperature of brittleness is determined from the formula

$$\Delta T_F = T_{kF} - T_{ki} \qquad \text{(III.9)}$$

where T_{kF} is a value of transition temperature for a fluence F,

T_{ki} is a value of transition temperature for initial conditions (unirradiated).

If value of $\Delta T_F < 0$ °C, then corrected value $\Delta T_F = 0$ °C shall be used in all further calculations.

3.1.3. If material is sensitive to thermal ageing, then shift of the critical temperature of brittleness is determined from the formula

$$\Delta T_F = \Delta T_F + \Delta T_T \qquad \text{(III.10)}$$

where ΔT_T is the shift of temperature of brittleness due to thermal ageing. $T_T(t)$ may be determined from thermal set of specimens and recalculated on irradiation temperature by Holomon dependence.

3.1.4. In both cases, transition temperature shifts are determined from similar sets of specimens (minimum 10) using similar test equipment and procedure. The difference in fluence between specimens of one set should be smaller than 15% of the mean value, and the difference in irradiation temperatures of individual specimens should be within ±10 °C. Finally, the mean value of irradiation temperature should be no higher than +10 °C above the wall inner surface temperature of the reactor pressure vessel. If this irradiation temperature of surveillance specimens is higher than +10 °C above temperature of RPV inner surface in the beltline region, analysis and necessary correction shall be performed.

Obtained experimental values of KV (impact notch energy) are evaluated using the following equation

$$KV = A + B \tanh [(T-T_0)/C] \qquad \text{(III.11)}$$

where A, B, C and T_0 are constants derived by statistical evaluation. It is strongly recommended to set lower shelf energy A = 3 J to avoid incorrect fitting when a small number of specimens are tested in the lower shelf energy temperature region and also to define upper shelf energy, i.e. A+B value, based on results from testing at this temperature region (100% shear fracture is required for upper shelf energy determination).

Shift of the transition temperature is determined for the criterion

$$KV = 47 \text{ J} \qquad \text{(III.12)}$$

3.1.5. This procedure results in valid values of ΔT_F only when the upper shelf energy, derived from the formula (III.10) - i.e., sum of (A+B), - is greater than 68 J. If this condition is not fulfilled, specific procedure has to be developed and approved by national Regulatory Authority.

3.1.6. The results of determinations of the shift in the critical temperature of brittleness obtained at least for three different approximately uniformly distributed neutron fluences are to be evaluated by the least squares method using the relationship:

$$\Delta T_F^{\ mean} = A_F^{exp} \ . \ (F.10^{-22})^{n\ exp} \qquad \text{(III.13)}$$

where F is the fluence of fast neutrons with the energy higher than 0.5 MeV, A_F^{exp} and n^{exp} are empirical constants obtained by statistical evaluation. The fluence is to be derived in accordance with Appendix II.

This procedure can be used for materials that are not sensitive to thermal ageing. If they are sensitive to thermal ageing, then a procedures for determination of thermal ageing part shall be applied (similarly to part 4 of this Appendix).

3.1.7. If results from testing surveillance specimens for higher fluence (above approx. $3x10^{24}$ m^{-2} for 15Kh2MFA(A) type steel and $6.4x10^{23}$ m^{-2} for 15Kh2NMFA(A) type steel and their welding joints are no more consistent with the formula (III.12), then probably some new damaging mechanism can be present (e.g. late blooming phase) and appropriate well described formula shall be chosen based on mathematical analyses.

3.1.8. In parallel to evaluation of the shift of the critical temperature of brittleness, ΔT_F, evaluations of results of tests of the shift of temperature dependence of static fracture toughness ΔT_0 and of dynamic fracture toughness ΔT_{0d} , if available, are to be performed.

3.1.9. Shifts ΔT_0 and ΔT_{0d} are determined and evaluated only for the use as described in 3.1.10.

3.1.10. Determination of shifts ΔT_0 and ΔT_{0d} is based on unirradiated and irradiated test data obtained from the same testing equipment and using the identical methods for statistically processed curves. Both unirradiated and irradiated test specimens should be cut from the same part of the vessel wall/welding joint and with the same orientation as specimens for notch impact testing. For base metal, specimens may be cut from the central half of the thickness, i.e. between ¼ and ¾ of the wall thickness. Weld metal specimens can be cut from the weld metal not closer than 15 mm from the surface as well as from the weld root.

3.1.11. Using mathematical statistics methods, the mean trend line of shifts ΔT_k , ΔT_0 and ΔT_{0d} is determined from all shifts, using the formula of the following form:

$$\Delta T_F^{mean\ all} = A_F^{exp\ mean\ all} \ . \ (F.10^{-22})^{n\ exp\ all} \qquad \text{(III.14)}$$

This procedure can be used for materials that are not sensitive to thermal ageing. If they are sensitive to thermal ageing then a procedures for determination of thermal ageing part shall be applied (similarly to part 4 of this Appendix).

For this procedure, i.e. for incorporating the experimental data, the following requirement has to be fulfilled: The experimentally derived shifts of transition temperatures must be based on at least three different neutron fluences. In an opposite case, more conservative values have to be used in accordance with 3.1.15.

3.1.12. Using mathematical statistics methods, the mean trend line of absolute values of T_0 is determined from all experimental data, using the formula of the following form:

$$\Delta T_0^{mean} = A_0^{exp\ mean} . \ (F.10^{-22})^{n\ exp} \qquad \text{(III.15)}$$

For this procedure, i.e. for incorporating the experimental data, the following requirement has to be fulfilled: The experimentally derived values of reference temperatures T_0 must be based on at least three different approximately uniformly distributed neutron fluences.

3.1.13. Upper boundary trend of transition temperature shifts ΔT_F or absolute values of T_0 evaluated in accordance with the formulae (III.13), (III.14) or (III.15) is obtained by shifting the mean line by a margin δT_m, defined as

$$\delta T_m = \max\{1.64\ \sigma^{exp};\ 1.64\ \delta T_M\} \qquad \text{(III.16)}$$

where σ^{exp} is a standard deviation of the results in formulae (III.13), (III.14) or (III.15).

If some experimental data lie above this upper boundary, special analyses shall be carried out with the aim of its explanation. This upper boundary of the shifts shall be used in assessment of RPV resistance against fast fracture.

3.1.14. It is not allowed to extrapolate shifts of the transition temperatures to the fluences higher than 10% increased maximum fluence used for the experiment.

3.1.15. Reference critical temperature of brittleness as a result of radiation embrittlement, RT_k, is calculated as

$$RTk = T_{k0} + \Delta T_F^{mean} + \delta T_m \qquad \text{(III.17)}$$

3.1.16. Reference temperature as a result of radiation embrittlement, RT_{0F} is determined as:

$$RT_{0F} = T_{0F} + \delta T_m \qquad \text{(III.18)}$$

where T_{0F} is a reference temperature T_0 determined experimentally for a given fluence F.

3.1.17. If there are insufficient surveillance test results:

Shift in materials of WWER-440 type RPVs from 15Kh2MFA(A) type steel, components from 22K, 16GNMA and 18Kh2M type steels and their welding joints is determined in accordance with the following procedure:

In a such a case, the coefficients of irradiation embrittlement A_F have to be used in the following relationship for the pressurised reactor vessel materials in accordance to the formula (III.19):

$$\Delta T_F = A_F \cdot (F.10^{-22})^n \qquad \text{(III.19)}$$

The values of A_F, n and σ for specified temperature of irradiation are as follows:

Table III.1 Irradiation embrittlement parameters

Material	$T_{irradiation}$ [°C]	A_F [°C]	n [-]	σ [°C]
base 22 K	100-160	33	0.33	
base 22 K	285	21	0.33	
base 16GNMA	285	35	0.33	
base + weld 18Kh2M	150-240	30	0.33	
base + weld 18Kh2M	260-270	25	0.33	
base + weld 18Kh2M	290	18	0.33	
base 15Kh2MFA(A)	270	$0.651+358(0.046C_{Cu}+C_P-0.002)$ *, **	0.48	11
weld Sv-10KhMFT(U)	270	$6.4+610(C_P+0.07C_{Cu}-0.01)$ *,***	0.33	10
base 15Kh2MFA	270	8.37	0.43	21.7
weld Sv-10KhMFT	270	$800\cdot(1.11\cdot C_P+0.064\cdot C_{Cu})$ *	0.29	22.6

* C_P means mass % of phosphorus, C_{Cu} means mass % of copper.

** valid for: $F \leq 3\cdot10^{24}$ n/m^2; $C_P \leq 0.013\%$; $C_{Cu} \leq 0.11\%$.

*** valid for: $C_P \leq 0.013\%$, $C_{Cu} \leq 0.11\%$, if $C_P+0.07\cdot C_{Cu}<0.01\%$, then it is taken $C_P+0.07\cdot C_{Cu}=0.01\%$.

3.1.18. Formula (III.19) with values of A_F and n represents upper boundary of transition temperature shifts for the following materials: 22K, 16GNMA and 18Kh2M.

3.1.19. Upper boundary for materials 15Kh2MFA(A) and their welding joints can be obtained by vertically shifted upward the mean line, given by coefficients A_F and n, by the value of 1.64σ, where σ is a standard deviation of material properties scatter.

This upper boundary is to be used in assessment of RPV resistance against fast fracture. Formula (III.19) is valid for neutron fluences in the range

$$10^{22} < F < 3 \times 10^{24}\ m^{-2} \qquad \text{(III.20)}$$

3.1.20. Shift in materials of WWER-1000 type RPVs from 15Kh2NMFAA type steel and its welding joints are determined in accordance with the following procedure:

3.1.21. Procedure for determination of parameters of radiation embrittlement of WWER 1000 RPV base and weld metals, as described in this Appendix, should be applied only in the case that method of direct determination of radiation embrittlement parameters based on results of RPV-specific surveillance specimens tests cannot be used.

3.1.22. Parameters of functional dependence of radiation embrittlement of WWER 1000 RPV base and weld metals given in this Appendix may be used also in case of direct determination of radiation embrittlement parameters A_F and ΔT_t^{inf} based on results of surveillance specimens tests.

3.1.23. Fluence dependences for determination of radiation embrittlement of WWER 1000 RPV materials are valid for steels of the following types: 15Kh2NMFA, 15Kh2NMFAA, 15Kh2NMFAA class 1 and their welding joints (weld materials: wire Sv-12Kh2N2MAA/flux FC-16(A); Sv-10KhGNMAA/flux FC-16; Sv-10KhGNMAA/flux AN17M; wire Sv-08KhGNMTA/flux 48NF-18M) after irradiation at temperature 290 ± 10 °C to fast neutron fluence of $6.4 \cdot 10^{23}$ m^{-2}.

3.1.24. Current fluence dependences have to be used instead of corresponding dependences defined in PNAE-G-7-002-86.

3.1.25. Fluence dependences for WWER 1000 RPV materials are of the form

$$\Delta T_K(F,t) = \Delta T_t(t) + \Delta T_F(F) + 2\sigma\ , \qquad \text{(III.21)}$$

where ΔT_K is shift of critical temperature of brittleness due to thermal ageing and neutron irradiation;

ΔT_t is shift of critical temperature of brittleness due to thermal ageing, depending on time of material exposure to operational temperature, it is calculated according to formula (III.22);

ΔT_F is shift of critical temperature of brittleness due to neutron irradiation, depending on neutron fluence, it is calculated according to the formula (III.21).

$$\Delta T_t\ (t) = \left(\Delta T_t^{inf} + b_T \exp\left(\frac{t_T - t}{t_{OT}} \right) \right) \cdot \tanh\left(\frac{t}{t_{OT}} \right), \qquad \text{(III.22)}$$

where ΔT_t^{inf} is shift of critical temperature of brittleness at t=∞;

t_{OT}, t_T and b_T are material constants dependent on temperature of ageing.

$$\Delta T_F = A_F \left(\frac{F}{F_0} \right)^m , \qquad \text{(III.23)}$$

where A_F is coefficient of radiation embrittlement. Constants t_{OT}, t_T, b_T and parameter ΔT_t^{inf} in formula (III.22) are defined in accordance with Table III.2.

Table. III.2 Values of constants t_{OT}, t_T, b_T and parameter ΔT_t^{inf}

Material	b_T, ^{0}C	t_{OT}, hour	t_T, hour	ΔT_t^{inf}, °C
Base metal	26.2	32700	40700	2
Weld metal with Ni content ≤1,3%	26.2	32700	40700	2
Weld metal with Ni content >1,3%	10.1	23200	40900	18

3.1.26. In the formula (III.23), the following values of m and A_F should be taken:

- for base material

m = 0,8; (III.24)

A_F = 1,45 °C; (III.25)

- for weld metal

m = 0,8; (III.26)

$A_F = \alpha_1 \exp(\alpha_2 \cdot C_{eq})$, °C (III.27)

where
$$C_{eq} = \begin{cases} C_{Ni} + C_{Mn} - \alpha_3 C_{Si}, \; if \; C_{Ni} + C_{Mn} - \alpha_3 C_{Si} \geq 0 \\ 0, \; if \; C_{Ni} + C_{Mn} - \alpha_3 C_{Si} < 0 \end{cases} ; \qquad \text{(III.28)}$$

α_1 = 0,703; α_2 = 0,883; α_3 = 3,885;

and C_{Ni}, C_{Mn}, C_{Si} are contents of Ni, Mn, Si respectively, in weight percent.

Formulae (III.24) to (III.28), take place for contents of Cu and P lower than 0.10 % and 0.014 %, respectively.

Formulae (III.27) and (III.28) take place for the following contents of Ni, Mn, and Si:

$1,00 \leq C_{Ni} \leq 1,90$ %,

$0,40 \leq C_{Mn} \leq 1,10$ %,

$0,20 \leq C_{Si} \leq 0,40$ %.

3.1.27. For calculation of upper 95% curve of $\Delta T_K(F, t)$ according to (III.21), the following values of σ are used:

for base metal σ = 19 °C;

for weld metal σ = 10 °C.

For calculation of 50% (median) curve of $\Delta T_K(F, t)$ according to (III.21), value σ = 0 should be taken, both for base and weld metal. This type of dependence is used for direct determination of radiation embrittlement parameters based on results of RPV-specific surveillance specimen tests.

3.2. Reference temperature T_0 is determined usually only for a given level of material damage, i.e. for required time of operation directly. If this cannot be determined directly by fracture toughness testing, then the following relation (III.29) may be used for determination of critical temperature of brittleness during operation only in the case of probabilistic evaluation of reactor pressure vessel failure probability, i.e.

$$T_o^{operation} = T_o^{initial} + \Delta T_F + \Delta T_T + \Delta T_{type} \tag{III.29}$$

where ΔT_F is determined by the same process as is shown in 3.1., i.e. using Charpy impact specimen testing and/or prediction using formula (III.15),

ΔT_{type} is a correction to the loss of constraint equal to:

= 0 °C for CT type specimens

= + 15 °C for three-point bend specimens. (III.30)

In this case, margin characterizing the scatter of the material properties and uncertainty due to limited number of specimens, σ, determined in accordance with (III.2) and (III.3) of Section 1.5 should be applied for determination of $T_0^{initial}$.

3.3. Fluence of neutrons used in all these formulae means the mean value of calculated or measured neutron fluence if the accuracy of its determination is within the requirements of Appendix II. In the case that requirements of Appendix II are not fulfilled, special explanation must be given.

3.4. Results from the surveillance specimens testing shall be checked with the predictive formulae (III.17), (III.19). If their transition temperature shifts are above the 95 % upper boundary calculated in accordance with (III.13), (III.14), then detailed analysis of these outliers shall be given.

3.5. Reference temperature RT_{kF} is equal to:

(a) if experimentally determined T_{k0} during Acceptance Tests exists then

$$RT_{kF} = T_{k0} + \Delta T_F + \delta T_m \tag{III.31}$$

(b) if only $T_{k0}^{guaranteed}$ exists, then

$$RT_{kF} = T_{k0}^{guaranteed} + \Delta T_F + \delta T_m \tag{III.32}$$

3.6. Results from the surveillance specimen programme tests can be applied also for other materials of the RPV beltline region under the following conditions:

3.6.1. Materials of RPV WWER-440:

Irradiation embrittlement coefficient A_F of the surveillance material is larger than those coefficients calculated for other beltline components (separately for base metal and weld metal) according to 3.1.15.

3.6.2. Materials of RPV WWER-1000:

Irradiation embrittlement coefficient A_F of the surveillance material is larger than those coefficients calculated for other beltline components (separately for base metal and weld metal) according to 3.1.24.

4. Determination of the effect of thermal ageing

4.1 The shift of the critical temperature of brittleness due to thermal ageing without irradiation can be determined in two ways:

4.1.1 The results of testing of surveillance samples or results of special measurements on the component are available:

4.1.2 Shift of the critical temperature of brittleness is determined from the formula

$$\Delta T_T = T_{kT} - T_{ki} \qquad \text{(III.33)}$$

where T_{kT} is a value of transition temperature for a time T

T_{ki} is a value of transition temperature for non- aged material.

In both cases, these transition temperatures are determined from similar sets of specimens (minimum 10) using similar test equipment and procedure. Obtained experimental values of KV (impact notch energy) are evaluated using the following equation

$$KV = A + B \tanh [(T-T0)/C] \qquad \text{(III.34)}$$

where A, B, C and T0 are constants derived by statistical evaluation. It is strongly recommended to set lower shelf energy A = 3 J to avoid incorrect fitting when a small number of specimens are tested in the lower shelf energy temperature region.

Shift of the transition temperature is determined for the criterion:

$$KV = 47 \text{ J} \qquad \text{(III.35)}$$

Specimens for one set of testing should be held at a given temperature with a maximum difference ±5 °C

4.1.3 In the time dependence evaluation of transition temperature shifts, ΔT_T, only non-negative values are included. Thus in the case when ΔT_T is determined as a negative, its value is taken as equal to 0 °C.

4.1.4 The time dependence of transition temperature shift, ΔT_T, can be described by one of the following forms:

4.1.4.1 Time dependence has a monotonous tendency – in this case the following formula is applied for a statistical evaluation of the data:

$$\Delta T_T = \Delta T_T^{lim} [1 - \exp (- pt)] \qquad \text{(III.36)}$$

where ΔT_T^{lim} and p are empirical constants.

4.1.4.2 In cases where the time dependence of the transition temperatures has a monotonous trend without saturation, then using the Holomon formula for extrapolation of shifts from testing at higher temperatures is allowed:

$$Hp = (T+273)(k+\lg t) \cdot 10^{-3}, \qquad \text{(III.37)}$$

where t is time of ageing in hours, T is temperature of ageing in °C and *k* is an empirical value determined with the use of statistical evaluation.

4.1.4.3 In cases where the time dependence of the transition temperature has a local maximum for ageing times between 500 and 10,000 hours, the following formula can be used for the decreasing part of its time dependence:

$$\Delta \mathrm{T_T}(t) = \left(\Delta T_T^{\inf} + b_T \exp\left(\frac{t_T - t}{t_{OT}} \right) \right) \cdot \tanh\left(\frac{t}{t_{OT}} \right), \qquad \text{(III.38)}$$

where t_T, b_T, t_{OT} – the constants, ΔT_T^{inf} - shift of the ΔT_T for $t \to \infty$. This formula describes both the increasing part and the decreasing part of dependence $\Delta T_T(t)$.

4.1.5 There are no corresponding experimental data available: In such a case, the values from formula (III.21) are to be used, where the following data for reactor of the WWER type materials after 100,000 hours are displayed:

ΔT_T = + 10 °C for steel 15Kh2MFA(A) and its welding joints (III.39)

= + 30 °C for steel 22K and its welding joints

= + 10 °C for steel 10GN2MFA and their welded joints.

Transition temperature shift ΔT_T for steel 15Kh2NMFA(A) and its welding metals can be determined with the use of formula (III.38) with the following constants:

Table III.3. Thermal ageing coefficients

Material	b_T, °C	t_{OT}, hour	t_T, hour	$\Delta T_T^{\inf}$, °C
Base metal	26.2	32700	40700	2
Weld metal with Ni content ≤1,3%	26.2	32700	40700	2
Weld metal with Ni content >1,3%	10.1	23200	40900	18

4.2 Reference temperature RT_k is equal to:

(a) if experimentally determined T_{k0} during Acceptance tests exists and if

$[T_{k0}^{guaranteed} - (T_{k0} + 1.64\ \delta T_M)] \geq 1.64\ \sigma_3$ then

$$RT_k = T_{k0} + 1.64\ .\ \delta T_M + \Delta T_T + 1.64\ \sigma_3 \qquad \text{(III.40)}$$

(b) if experimentally determined T_{k0} during Acceptance tests exist, and if

$[T_{k0}^{guaranteed} - (T_{k0} + 1.64\ \delta T_M)] < 1.64\ \sigma_3$ then

$$RT_k = T_{k0}^{guaranteed} + \Delta T_T \qquad \text{(III.41)}$$

(c) if only $T_{k0}^{guaranteed}$ exists, then

$$RT_k = T_{k0}^{guaranteed} + \Delta T_T \qquad \text{(III.42)}$$

where σ_3 = 6 °C and represents the uncertainty in thermal ageing test data.

5. Determination of changes in tensile properties

5.1 In some cases (reactor pressure vessel surveillance specimens, primary piping materials), determination of the changes in tensile properties are also required. In such cases, comparison of initial properties with those determined after irradiation or time ageing is carried out.

5.2 The following rules should be maintained:

— Specimens from both conditions should be cut from the same depth and orientation;

— Specimens from both conditions should have comparable test diameter and measured length.

5.3 Changes in material properties are then determined for room and operating temperature as:

ΔR_p, ΔR_m, ΔA, ΔA_m, ΔZ

where ΔR_p is change in yield strength, [MPa]

ΔR_m is change in ultimate tensile strength, [MPa]

ΔA is change in elongation, [%]

ΔA_m is change in uniform elongation, [%]

ΔZ is change in reduction of area, [%]

5.4 Increase in yield strength (or ultimate tensile strength) of materials as an effect of radiation hardening can be expressed in the form

$$\Delta R_p = B_F^{exp} \,.\, (F.10^{-22})^{n\ exp} \quad \text{(III.43)}$$

$$\Delta R_m = C_F^{exp} \,.\, (F.10^{-22})^{n\ exp} \quad \text{(III.44)}$$

where coefficients B_F and n^{exp} do not depend on test temperature.

5.5 In all cases, the mean value from at least three tests is calculated as well as the mean standard deviation σ_4 for the given trend. The upper boundary can be obtained by shifting this mean trend by a value of 1.64 σ_4.

5.6 For fluence/time dependence of the changes in these properties, similar processes should be used as for the changes in transition temperatures as described in this Appendix for transition temperature shifts.

5.7 Non-destructive methods, if properly validated (like Automated Ball Indentation Test Method), are allowed for determination of tensile properties directly on components.

6. Annealing and re-embrittlement

6.1 Methods for determining the transition temperature shifts, ΔTF and/or ΔT_0, as described in Chapter 3 of this Appendix, can also be applied for determining the shifts obtained as a result of annealing of steel and weld joints of 15Kh2MFA type and the following re-irradiation.

6.2 The residual value of the transition temperature shift after annealing, ΔTR, is determined by:

$$\Delta T_R = T_R - T_{ki} \quad \text{(III.45)}$$

where T_R is a transition temperature after irradiation by a fluence equal to F_1 and subsequent annealing, and T_{ki} is an initial value of this transition temperature for a given material.

This value should be non-negative; if the calculated number is negative, it is set equal to 0 °C.

In cases where no experimental data exist, the following values should be taken for reactor pressure vessel integrity assessment:

ΔT_R = + 30 °C for welds with content P < 0.040 mass % (III.46)

ΔT_R = + 40 °C for welds with content P > 0.040 mass %

6.3 The transition temperature shift after re-irradiation is determined by the formula:

$$\Delta T_{FR} = T_{FR} - T_{ki} \quad \text{(III.47)}$$

where T_{FR} is a transition temperature after re-irradiation by a fluence equal to F_2 after annealing, and T_{ki} is an initial value of this transition temperature for a given material. This formula (III.26) can also be expressed in the form:

$$\Delta T_{FR} = \Delta T_R + \Delta\ T_{RI} \quad \text{(III.48)}$$

where $\Delta\ T_{RI}$ is a shift due to re-irradiation by an additional fluence of F_2 after annealing comparing with the residual value T_R.

6.4 Trend of transition temperatures shifts, ΔT_{FR} and $\Delta T_{RI,}$ can be assessed similarly as for the case of primary irradiation.

6.5 If experimental data exist in a required quantity and volume, then the trend curve is determined in accordance with 3.1.4.

Experimental data should be obtained in similar irradiation conditions as exist at the inner wall of reactor pressure vessel (material, irradiation temperature, neutron flux and neutron fluence).

6.6 Reactor pressure vessel operated after annealing shall contain surveillance specimens in sufficient quantity to monitor changes in mechanical properties during re-irradiation.

6.7 In the case where no archive materials exist, surrogate material specimens having chemical and mechanical properties closely representative of the reactor pressure vessel may be irradiated. Material should be chosen in such a way that the content of critical elements (i.e. phosphorus, copper, nickel, manganese and silicon) is conservative in comparison to the pressure vessel.

6.8 If experimental data do not exist, or are not of a required quality and quantity, then the trend curve is allowed to be described using the "lateral shift" model described by:

$$\Delta T_{FR} = A_F \, . \, [\, F_2/10^{22} + (\Delta T_R/A_F)^3 \,]^{1/3}, \qquad \text{(III.49)}$$

where A_F is the radiation embrittlement coefficient given in (III.19) and F_2 is a neutron fluence after annealing, i.e. during re-irradiation.

7. Requirements for plant life extension

7.1 Plant life extension of reactor pressure vessels beyond design lifetime shall be supported by experimental data from surveillance specimen programme of the assessed RPV. These data shall be obtained for fluences up to the expected neutron fluence with lead factor not larger than 5 for assessed extended lifetime.

7.2 In the case where no archive materials exist, surrogate material specimens having chemical and mechanical properties closely representative of the reactor pressure vessel may be irradiated. Material should be chosen in such a way that the content of critical elements (i.e. phosphorus, copper, nickel, manganese and silicon) is conservative in comparison to the pressure vessel material.

7.3 Analysis of all experimental data obtained shall be performed in such a way to show that no additional damaging mechanism to the initial one is developed for such high neutron fluences.

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APPENDIX IV

Determination of values of stress intensity factor KI

1. General principles

— The formulas attached in this Appendix can be used only for calculation of the stress intensity factor, KI, in the range of linear-elastic-fracture mechanics.

1.1 List of symbols

a	depth (minor semi-axis) of a crack, [m]
c	half length (major semi-axis) of a crack, [m]
s	wall thickness, [m]
r	cladding thickness, [m]
τ	time, [s]
b	distance of the point of the crack closest to the nearest surface from this surface (length of the ligament), [m]
$R_{p0.2}$	yield strength, [MPa]
E, E_1, E_2	Young modulus, [MPa]
σ_K, σ_{KA}, σ_{KB}	equivalent stresses, [MPa]
σ_A, σ_B, σ_C,	normal stresses in points A, B, C, respectively, [MPa]
σ_R	residual stress, [MPa]
Y, Y_A, Y_B	shape factor
K_I	stress intensity factor, [MPa.m$^{0.5}$]
A, B, C, D	points on crack front

2. Determination of values K_I for uncladded components

The stress intensity factor, K_I, should be derived from the relation

$$K_I = \sigma_K \cdot Y \cdot a^{1/2} \qquad \text{(IV.1)}$$

The relation (IV.1) is valid for σ_K smaller than the yield strength, $R_{p0.2}$. The stress σ_K includes all relevant stresses in accordance to Chapter 5.2 of the Main document. The stress σ_K should be derived according to Chapter 3 of this Appendix.

2.1 To calculate KI according to (IV.1), the following σK and Y should be taken into account for a surface crack (see Figure IV.1):

a) at point A

$$\sigma_K = \sigma_{KA}, \quad Y = Y_A \ , \qquad \text{(IV.2)}$$

where the shape factor, Y_A, is determined as follows:

$$Y_A = \frac{2 - 0.82.a/c}{\left\{1 - \left[0.89 - 0.57(a/c)^{1/2}\right]^3.(a/s)^{1.5}\right\}^{3.25}} \ ; \qquad \text{(IV.3)}$$

b) at point B

$$\sigma_K = \sigma_{KB}, \quad Y = Y_B \ , \qquad \text{(IV.4)}$$

and $Y_B = [1.1 + 0.35 (a/s)^2] \cdot (a/c)^{1/2} \cdot Y_A$.

2.2 For an inner (embedded) crack (see Figure IV.2), the following σ_K and Y values should be used for K_I calculation:

a) at point A

$$\sigma_K = \sigma_{KA} \ , \tag{IV.5}$$

$$Y = \frac{1.79 - 0.66.a/c}{[1 - \beta^{1.8}(1 - 0.4(a/c) - 0.8\gamma^{0.4})]^{0.54}} \ ,$$

where

$$\beta = \frac{a}{b+a}; \gamma = 0{,}5 - \frac{b+a}{s} \ , \tag{IV.6}$$

b is the distance of the point of the crack closest to the nearest surface from this surface (length of the ligament) - see Figure IV.2;

b) at point B

$$\sigma_K = \sigma_{KB} \ , \tag{IV.7}$$

$$Y = \frac{1.79 - 0.66.a/c}{[1 - \beta^{1.8}(1 - 0.4(a/c) - \gamma^2)]^{0.54}} \ .$$

c) at point D (the vortex on the main axis of the ellipse), the value of the stress intensity factor, KI, should be determined using its values at points A and B from the relationship

$$K_I(D) = \frac{K_I(A) + K_I(B)}{2} \tag{IV.8}$$

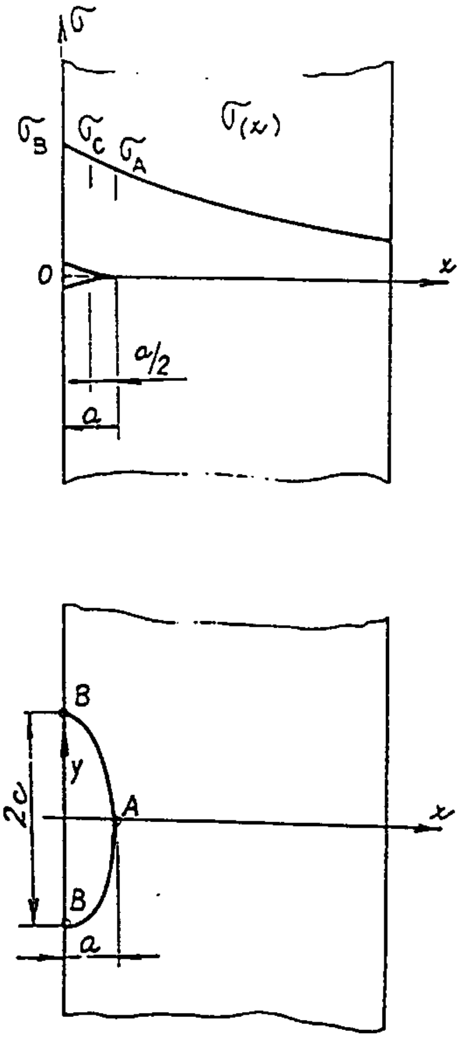


FIG. IV.1. Surface crack.

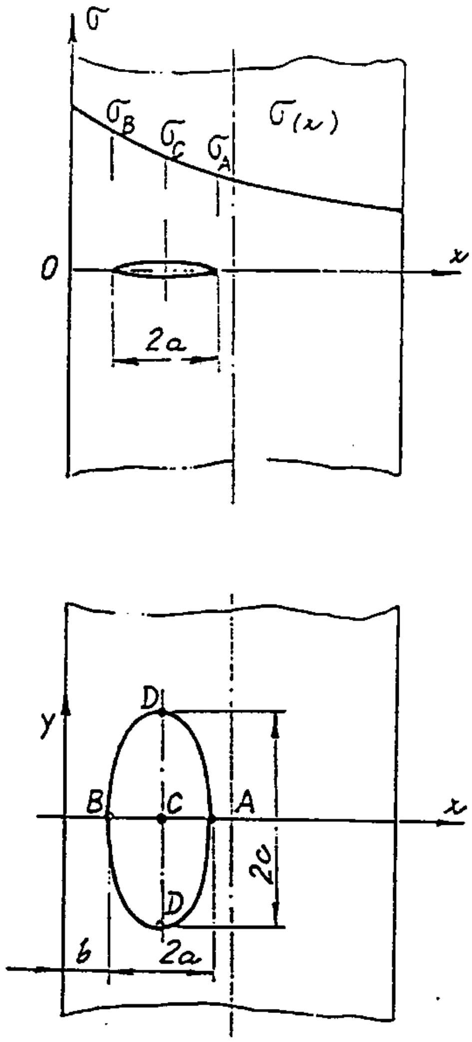


FIG. IV.2. Inner (embedded *crack).*

2.3 Cracks with the ratio of their semi-axes a/c > 1 are taken as circular ones or semi-circular with the radius *a.*

2.4 A crack is taken as an inner (embedded) crack if its distance from the closer of both surfaces (Figure IV.2) is given by the following relationship:

$$b > 0.4\,a \tag{IV.9}$$

But the type of crack has to be changed and the crack has to be treated as a surface one when a prospective growth of the crack is also being taken into account and due to it the criterion (IV.9) does not hold.

2.5 Calculation of equivalent stress

The equivalent stress, σ_K, needed for the calculation of the stress intensity factor is calculated from stresses $\sigma_A, \sigma_B, \sigma_C$ at points A, B, and C (see Figs. IV.1, IV.2) which are determined from the stress fields computed on the model of the component without the crack. The stresses $\sigma_A, \sigma_B, \sigma_C$ are normal stresses regarding the crack plane.

2.5.1. Determination of the equivalent stress for a surface crack (symbols see Figure IV.1):

2.5.1.1. If the stress is constant through the wall thickness, then

$$\sigma_{KA} = \sigma_{KB} = \sigma_A = \sigma_B \quad . \tag{IV.10}$$

2.5.1.2. If the stress is linearly dependent on the wall thickness, then

$$\sigma_{KA} = 0.61\sigma_A + 0.39\sigma_B + \left\{0.11 \cdot (a/c) - 0.28 \cdot (a/s) \cdot \left[1 - (a/c)^{1/2}\right]\right\} \cdot (\sigma_A - \sigma_B)\,, \tag{IV.11}$$

$$\sigma_{KB} = 0.18\,\sigma_A + 0.82\,\sigma_B\,. \tag{IV.12}$$

2.5.1.3. For a parabolic gradient of the stress through the wall thickness

$$\begin{aligned}\sigma_{KA} = 0.111 \cdot (3\sigma_A + \sigma_B + 5\sigma_C) + 0.4 \cdot (a/c) \cdot (0.38\sigma_A - 0.17\sigma_B - 0.21\sigma_C) - \\ - 0.28 \cdot (a/s) \cdot [1 - (a/c)^{1/2}] \cdot (\sigma_A - \sigma_B)\end{aligned} \tag{IV.13}$$

$$\sigma_{KB} = 0.64\sigma_B + 0.36\sigma_C \tag{IV.14}$$

2.5.2. Determination of the equivalent stress for an inner (embedded) crack (symbols see Figure IV.2):

2.5.2.1. If the stress is constant through the wall thickness, then

$$\sigma_{KA} = \sigma_{KB} = \sigma_A = \sigma_B \tag{IV.15}$$

2.5.2.2. If the stress is linearly dependent on the wall thickness, then

$$\sigma_{KA} = \frac{3\sigma_A + \sigma_B}{4} + \frac{a}{c} \cdot \frac{\sigma_A - \sigma_B}{12}\,, \tag{IV.16}$$

$$\sigma_{KB} = \frac{3\sigma_B + \sigma_A}{4} + \frac{a}{c} \cdot \frac{\sigma_B - \sigma_A}{12} \tag{IV.17}$$

2.5.2.3. For a parabolic approximation of the stress through the wall thickness

$$\sigma_{KA} = \frac{\sigma_A + \sigma_C}{2} + \frac{a}{c} \cdot \frac{4\sigma_A - 3\sigma_C - \sigma_B}{12} \tag{IV.18}$$

$$\sigma_{KB} = \frac{\sigma_B + \sigma_B}{2} + \frac{a}{c} \cdot \frac{4\sigma_B - 3\sigma_C - \sigma_A}{30} \tag{IV.19}$$

3. Determination of values K_I for cladded components

First calculation step is the nominal stress field determination: the stress field component perpendicular to the crack surface and calculated without considering the defect must be determined. This stress field can be determined either by finite element calculation or with using analytical stress solutions.

Main difficulty at this step is the representation of the stress discontinuity at the interface between the cladding and ferritic steel (discontinuity due to the difference of mechanical properties). The procedure describing the stress field is the following:

— The origin of the axis is taken on the inner surface (and not at the interface between the two materials). This choice ensures consistent description for both through and underclad defects. The axis is normalised by the total thickness (cladding + ferritic: r + s – see Figure IV.3).

— The stress field in the ferritic material is fitted by a 3rd or 4th degree polynomial form to obtain good description of the stress through the thickness – see Figure IV.4.

$$\sigma = \sigma_0 + \sigma_1 . \frac{x}{s+r} + \sigma_2 . \left(\frac{x}{s+r}\right)^2 + \sigma_3 . \left(\frac{x}{s+r}\right)^3 + \sigma_4 . \left(\frac{x}{s+r}\right)^4 \qquad \text{(IV.20)}$$

— Then, the supplement of stress field in the cladding is fitted by a linear fit – see Figure 4.

$$\sigma = \sigma_{0r} + \sigma_{r1} . \frac{x}{s+r} \qquad \text{(IV.21)}$$

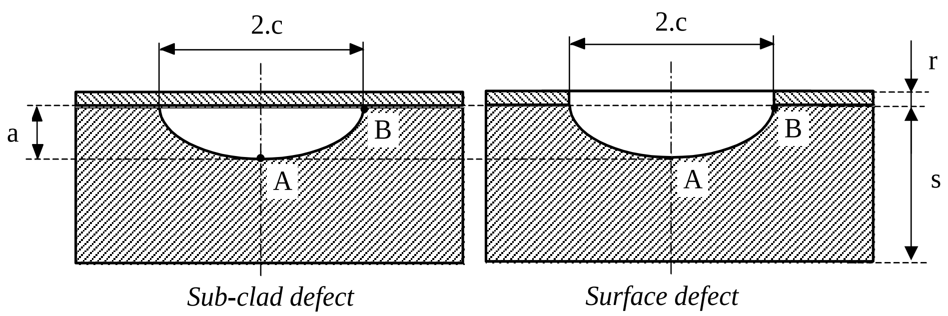


FIG. IV.3. Description of the defects considered.

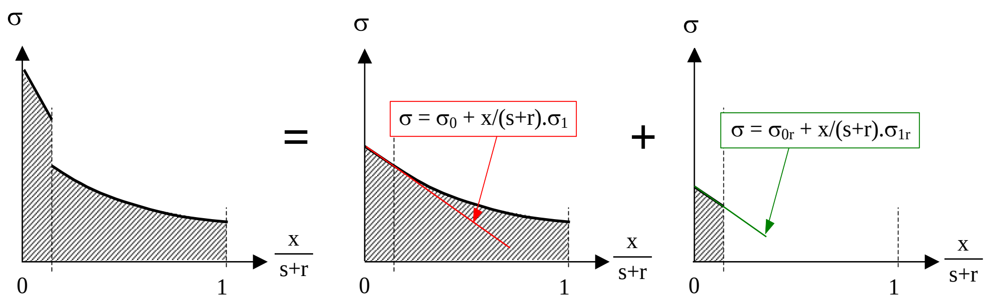


FIG. IV.4. Stress description.

Using these fits, 6 or 7 stress coefficients: σ_0 through σ_3 or σ_4, σ_{0r} and σ_{1r} are obtained. Knowing the stress coefficients, the K_I value is calculated using the influence function methodology. In this case, the influence functions were determined by precise finite element calculation, and are tabulated in the compendium.

The compendium for cladded components (given in tables IV.1–IV.11) is expressed as function of the non-dimensional parameters a/c, a/r and E_1/E_2, where:

— a is the crack depth in the ferritic material;

— r is the cladding thickness;

— E_1 and E_2 are the Young modules of cladding and ferritic material.

This compendium gives the possibility to calculate K_I at the deepest point of the crack (point A in figure 3) or the point at the interface of the two materials (point B in Figure IV.3).

K_I-value is then determined by the following formulae:

For a surface crack:

$$K_I = \left(\sigma_0 . i_0 + \sigma_1 . i_1 . \frac{a+r}{s+r} + \sigma_2 . i_2 . \left(\frac{a+r}{s+r} \right)^2 + \sigma_3 . i_3 . \left(\frac{a+r}{s+r} \right)^3 + \sigma_4 . i_4 . \left(\frac{a+r}{s+r} \right)^4 + \sigma_{0r} . i_{0r} + \sigma_{1r} . i_{1r} . \frac{a+r}{s+r} \right) . \sqrt{\pi . (a+r)}$$

(IV.22)

For the sub-clad crack:

$$K_I = \left(\sigma_0 . i_0 + \sigma_1 . i_1 . \frac{a+r}{s+r} + \sigma_2 . i_2 . \left(\frac{a+r}{s+r} \right)^2 + \sigma_3 . i_3 . \left(\frac{a+r}{s+r} \right)^3 + \sigma_4 . i_4 . \left(\frac{a+r}{s+r} \right)^4 \right) . \sqrt{\pi . a} \qquad \text{(IV.23)}$$

For values of a/r, a/c and E_1/E_2 that are not given in the compendium, a linear interpolation on the a/r, a/c and E_1/E_2 dimensions is recommended.

For sub-clad defects, the calculated K_I value must be corrected by the □ correction as follows:

$$K_J(\tau) = K_I(\tau) + (\square(u)-1).K_I(u) \qquad \text{(IV.24)}$$

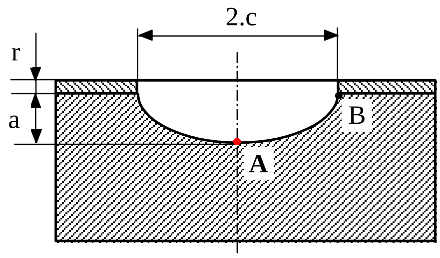
With: $u = \min(\tau, \tau_{max})$, where τ_{max} corresponds to the time at maximum K_I,

$$\beta^A = 1 + 0.5 \tanh[36 r_y^B / r], \qquad \text{(IV.25)}$$

$$\beta^B = 1 + 0.3 \tanh[36 r_y^B / r], \qquad \text{(IV.26)}$$

$$r_y^B = (K_I^B / R_{p0.2}^B)^2 / 6\pi \qquad \text{(IV.27)}$$

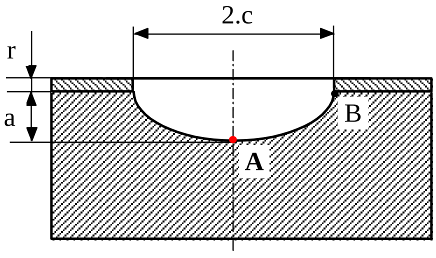
$R_{p0.2}^B$ is the yield stress (depending on temperature and thus on time) of the material at point B (i.e. of cladding).



Surface defect

Table IV.1. Influence functions for a through clad defect (E_1/E_2 = 1, point A)

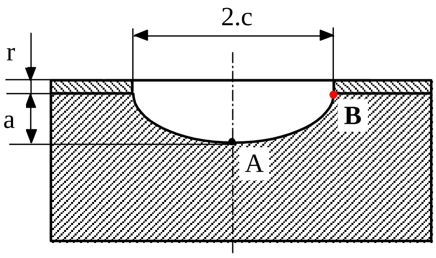
a / c	a / r	E_1 / E_2	i_0	i_1	i_2	i_3	i_4	i_{0r}	i_{1r}
1	**0.125**	**1**	0.229	0.215	0.205	0.196	0.189	3.51E-2	2.72E-2
1	**0.25**	**1**	0.308	0.277	0.256	0.239	0.226	4.80E-2	2.99E-2
1	**0.5**	**1**	0.399	0.337	0.3	0.275	0.255	6.17E-2	2.73E-2
1	**1**	**1**	0.488	0.385	0.333	0.299	0.274	6.95E-2	1.99E-2
1	**1.5**	**1**	0.532	0.407	0.347	0.309	0.282	6.76E-2	1.47E-2
1	**2**	**1**	0.558	0.419	0.355	0.314	0.286	6.30E-2	1.11E-2
1	**3**	**1**	0.588	0.432	0.363	0.320	0.290	5.36E-2	6.95E-3
1	**4**	**1**	0.604	0.439	0.367	0.323	0.292	4.59E-2	4.71E-3
0.5	**0.125**	**1**	0.339	0.302	0.278	0.260	0.245	0.100	7.15E-2
0.5	**0.25**	**1**	0.464	0.382	0.336	0.305	0.281	0.141	7.81E-2
0.5	**0.5**	**1**	0.602	0.451	0.381	0.337	0.306	0.179	7.01E-2
0.5	**1**	**1**	0.721	0.502	0.411	0.357	0.320	0.185	4.92E-2
0.5	**1.5**	**1**	0.771	0.522	0.422	0.364	0.325	0.168	3.48E-2
0.5	**2**	**1**	0.798	0.533	0.428	0.368	0.328	0.150	2.56E-2
0.5	**3**	**1**	0.827	0.544	0.434	0.372	0.331	0.121	1.53E-2
0.5	**4**	**1**	0.843	0.550	0.437	0.374	0.332	0.101	1.02E-2
0.25	**0.125**	**1**	0.466	0.381	0.337	0.307	0.284	0.203	0.127
0.25	**0.25**	**1**	0.649	0.477	0.400	0.353	0.320	0.289	0.140
0.25	**0.5**	**1**	0.819	0.550	0.443	0.382	0.340	0.338	0.122
0.25	**1**	**1**	0.927	0.592	0.466	0.396	0.350	0.307	7.85E-02
0.25	**1.5**	**1**	0.962	0.606	0.474	0.401	0.353	0.261	5.27E-02
0.25	**2**	**1**	0.979	0.612	0.477	0.403	0.355	0.224	3.76E-02
0.25	**3**	**1**	0.998	0.619	0.481	0.405	0.356	0.174	2.18E-02
0.25	**4**	**1**	1.010	0.624	0.483	0.407	0.357	0.143	1.43E-02
0.125	**0.125**	**1**	0.634	0.460	0.388	0.344	0.313	0.358	0.194
0.125	**0.25**	**1**	0.845	0.563	0.452	0.389	0.347	0.462	0.204
0.125	**0.5**	**1**	0.984	0.621	0.485	0.411	0.363	0.465	0.161
0.125	**1**	**1**	1.047	0.645	0.499	0.419	0.368	0.377	9.53E-02
0.125	**1.5**	**1**	1.064	0.651	0.502	0.421	0.369	0.309	6.20E-02
0.125	**2**	**1**	1.073	0.655	0.504	0.422	0.370	0.261	4.35E-02
0.125	**3**	**1**	1.085	0.659	0.506	0.423	0.370	0.200	2.50E-02
0.125	**4**	**1**	1.098	0.664	0.508	0.425	0.371	0.164	1.64E-02
0.0625	**0.125**	**1**	0.829	0.544	0.437	0.378	0.338	0.543	0.269
0.0625	**0.25**	**1**	0.983	0.620	0.486	0.413	0.364	0.584	0.248
0.0625	**0.5**	**1**	1.061	0.654	0.506	0.426	0.374	0.523	0.179
0.0625	**1**	**1**	1.095	0.667	0.513	0.430	0.376	0.404	0.102
0.0625	**1.5**	**1**	1.104	0.671	0.515	0.431	0.377	0.326	6.54E-02
0.0625	**2**	**1**	1.110	0.673	0.515	0.431	0.377	0.274	4.57E-02
0.0625	**3**	**1**	1.124	0.677	0.518	0.432	0.377	0.211	2.63E-02
0.0625	**4**	**1**	1.142	0.684	0.521	0.434	0.378	0.174	1.74E-02
0	*See infinitely long longitudinal and circumferential defects (Tables IV.5–IV. 7)*								



Surface defect

Table IV.2. Influence functions for a through clad defect (E_1/E_2 = 0.7, point A)

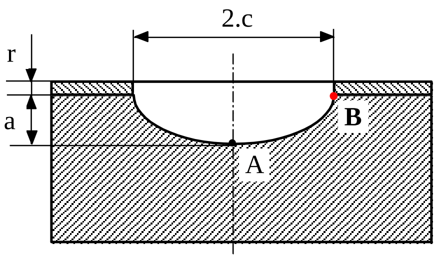
a / c	a / r	E_1 / E_2	i_0	i_1	i_2	i_3	i_4	i_{0r}	i_{1r}
1	**0.125**	**0.7**	0.238	0.223	0.211	0.202	0.193	4.09E-02	3.15E-02
1	**0.25**	**0.7**	0.321	0.286	0.262	0.244	0.230	5.62E-02	3.46E-02
1	**0.5**	**0.7**	0.416	0.345	0.306	0.278	0.258	7.24E-02	3.15E-02
1	**1**	**0.7**	0.506	0.393	0.337	0.302	0.276	8.06E-02	2.28E-02
1	**1.5**	**0.7**	0.549	0.413	0.350	0.311	0.283	7.71E-02	1.66E-02
1	**2**	**0.7**	0.574	0.424	0.357	0.316	0.287	7.09E-02	1.24E-02
1	**3**	**0.7**	0.600	0.436	0.365	0.321	0.291	5.88E-02	7.58E-03
1	**4**	**0.7**	0.615	0.443	0.369	0.324	0.293	4.95E-02	5.07E-03
0.5	**0.125**	**0.7**	0.362	0.320	0.293	0.272	0.255	0.117	8.30E-02
0.5	**0.25**	**0.7**	0.497	0.402	0.351	0.316	0.290	0.166	9.05E-02
0.5	**0.5**	**0.7**	0.644	0.471	0.393	0.345	0.312	0.209	8.08E-02
0.5	**1**	**0.7**	0.761	0.518	0.420	0.363	0.324	0.211	5.55E-02
0.5	**1.5**	**0.7**	0.805	0.535	0.429	0.369	0.328	0.188	3.85E-02
0.5	**2**	**0.7**	0.827	0.543	0.433	0.372	0.330	0.165	2.79E-02
0.5	**3**	**0.7**	0.849	0.552	0.438	0.375	0.332	0.130	1.64E-02
0.5	**4**	**0.7**	0.860	0.556	0.441	0.376	0.333	0.107	1.07E-02
0.25	**0.125**	**0.7**	0.506	0.408	0.358	0.324	0.299	0.235	0.147
0.25	**0.25**	**0.7**	0.706	0.508	0.421	0.368	0.331	0.334	0.161
0.25	**0.5**	**0.7**	0.881	0.578	0.460	0.393	0.348	0.385	0.137
0.25	**1**	**0.7**	0.977	0.612	0.477	0.403	0.355	0.339	8.63E-02
0.25	**1.5**	**0.7**	1.002	0.621	0.482	0.406	0.357	0.283	5.70E-02
0.25	**2**	**0.7**	1.012	0.624	0.484	0.407	0.358	0.240	4.01E-2
0.25	**3**	**0.7**	1.021	0.628	0.485	0.408	0.358	0.183	2.29E-2
0.25	**4**	**0.7**	1.029	0.631	0.487	0.409	0.359	0.149	1.49E-2
0.125	**0.125**	**0.7**	0.690	0.494	0.412	0.363	0.329	0.406	0.219
0.125	**0.25**	**0.7**	0.914	0.597	0.475	0.405	0.360	0.518	0.228
0.125	**0.5**	**0.7**	1.047	0.649	0.502	0.423	0.371	0.512	0.177
0.125	**1**	**0.7**	1.093	0.664	0.509	0.426	0.373	0.407	0.103
0.125	**1.5**	**0.7**	1.100	0.666	0.510	0.426	0.373	0.329	6.59E-02
0.125	**2**	**0.7**	1.103	0.666	0.510	0.426	0.373	0.275	4.59E-02
0.125	**3**	**0.7**	1.107	0.668	0.511	0.426	0.372	0.208	2.60E-02
0.125	**4**	**0.7**	1.115	0.670	0.512	0.427	0.373	0.169	1.69E-02
0.0625	**0.125**	**0.7**	0.888	0.577	0.461	0.396	0.353	0.594	0.294
0.0625	**0.25**	**0.7**	1.044	0.652	0.507	0.428	0.376	0.634	0.270
0.0625	**0.5**	**0.7**	1.117	0.680	0.522	0.436	0.382	0.564	0.193
0.0625	**1**	**0.7**	1.138	0.685	0.523	0.436	0.381	0.431	0.108
0.0625	**1.5**	**0.7**	1.138	0.684	0.522	0.435	0.380	0.345	6.91E-02
0.0625	**2**	**0.7**	1.138	0.683	0.521	0.435	0.379	0.288	4.79E-02
0.0625	**3**	**0.7**	1.144	0.685	0.522	0.435	0.379	0.219	2.73E-02
0.0625	**4**	**0.7**	1.158	0.690	0.524	0.436	0.380	0.180	1.79E-02
0	*See infinitely long longitudinal and circumferential defects (Tables IV.5–IV. 7)*								



Surface defect

Table IV.3. Influence functions for a through clad defect (E_1/E_2 = 1, point B)

a / c	**a / r**	**E_1 / E_2**	**i_0**	**i_1**	**i_2**	**i_3**	**i_4**	**i_{0r}**	**i_{1r}**
1	**0.125**	**1**	0.262	0.226	0.197	0.173	0.153	0.139	0.115
1	**0.25**	**1**	0.353	0.268	0.209	0.166	0.133	0.179	0.123
1	**0.5**	**1**	0.458	0.282	0.186	0.127	8.85E-2	0.224	0.113
1	**1**	**1**	0.565	0.263	0.139	7.89E-2	4.74E-2	0.262	8.73E-2
1	**1.5**	**1**	0.619	0.241	0.112	5.80E-2	3.31E-2	0.272	6.80E-2
1	**2**	**1**	0.652	0.225	9.61E-2	4.77E-2	2.69E-2	0.270	5.43E-2
1	**3**	**1**	0.689	0.202	7.92E-2	3.84E-2	2.20E-2	0.257	3.72E-2
1	**4**	**1**	0.709	0.188	7.07E-2	3.45E-2	2.01E-2	0.241	2.72E-2
0.5	**0.125**	**1**	0.292	0.242	0.207	0.179	0.156	0.175	0.136
0.5	**0.25**	**1**	0.397	0.284	0.216	0.168	0.133	0.230	0.145
0.5	**0.5**	**1**	0.515	0.297	0.190	0.128	8.79E-2	0.286	0.133
0.5	**1**	**1**	0.617	0.274	0.142	7.97E-2	4.73E-2	0.315	9.94E-2
0.5	**1.5**	**1**	0.661	0.249	0.114	5.84E-2	3.30E-2	0.312	7.54E-2
0.5	**2**	**1**	0.685	0.230	9.70E-2	4.76E-2	2.66E-2	0.301	5.91E-2
0.5	**3**	**1**	0.709	0.203	7.84E-2	3.75E-2	2.12E-2	0.277	3.95E-2
0.5	**4**	**1**	0.722	0.187	6.91E-2	3.31E-2	1.90E-2	0.256	2.86E-2
0.25	**0.125**	**1**	0.304	0.238	0.198	0.169	0.146	0.203	0.147
0.25	**0.25**	**1**	0.409	0.275	0.203	0.156	0.122	0.264	0.155
0.25	**0.5**	**1**	0.509	0.280	0.175	0.115	7.76E-2	0.308	0.137
0.25	**1**	**1**	0.581	0.249	0.125	6.75E-2	3.85E-2	0.317	9.88E-2
0.25	**1.5**	**1**	0.608	0.220	9.57E-2	4.64E-2	2.47E-2	0.305	7.36E-2
0.25	**2**	**1**	0.620	0.198	7.82E-2	3.57E-2	1.85E-2	0.290	5.72E-2
0.25	**3**	**1**	0.632	0.169	5.94E-2	2.57E-2	1.34E-2	0.264	3.79E-2
0.25	**4**	**1**	0.637	0.151	5.01E-2	2.15E-2	1.14E-2	0.243	2.74E-2
0.125	**0.125**	**1**	0.311	0.228	0.186	0.156	0.133	0.225	0.151
0.125	**0.25**	**1**	0.391	0.253	0.184	0.139	0.108	0.268	0.152
0.125	**0.5**	**1**	0.458	0.247	0.152	9.76E-2	6.46E-2	0.289	0.129
0.125	**1**	**1**	0.505	0.210	0.101	5.23E-2	2.82E-2	0.287	9.08E-2
0.125	**1.5**	**1**	0.520	0.180	7.36E-2	3.28E-2	1.58E-2	0.274	6.73E-2
0.125	**2**	**1**	0.526	0.158	5.72E-2	2.33E-2	1.06E-2	0.261	5.22E-2
0.125	**3**	**1**	0.530	0.129	4.00E-2	1.48E-2	6.57E-3	0.238	3.47E-2
0.125	**4**	**1**	0.532	0.113	3.18E-2	1.15E-2	5.26E-3	0.220	2.51E-2
0.0625	**0.125**	**1**	0.295	0.211	0.170	0.142	0.120	0.230	0.152
0.0625	**0.25**	**1**	0.349	0.224	0.162	0.121	9.31E-2	0.255	0.148
0.0625	**0.5**	**1**	0.396	0.211	0.128	8.12E-2	5.26E-2	0.269	0.124
0.0625	**1**	**1**	0.429	0.174	8.12E-2	4.00E-2	2.04E-2	0.266	8.77E-2
0.0625	**1.5**	**1**	0.439	0.146	5.61E-2	2.31E-2	1.01E-2	0.255	6.55E-2
0.0625	**2**	**1**	0.442	0.125	4.17E-2	1.52E-2	6.02E-3	0.243	5.12E-2
0.0625	**3**	**1**	0.442	9.88E-2	2.69E-2	8.52E-3	3.20E-3	0.224	3.44E-2
0.0625	**4**	**1**	0.441	8.33E-2	2.01E-2	6.11E-3	2.42E-3	0.209	2.51E-2



Surface defect

Table IV.4. Influence functions for a through clad defect (E_1/E_2 = 0.7, point B)

a / c	a / r	E_1 / E_2	i_0	i_1	i_2	i_3	i_4	i_{0r}	i_{1r}
1	**0.125**	**0.7**	0.308	0.265	0.231	0.202	0.178	0.173	0.142
1	**0.25**	**0.7**	0.412	0.310	0.241	0.191	0.152	0.220	0.150
1	**0.5**	**0.7**	0.532	0.323	0.212	0.144	0.100	0.275	0.138
1	**1**	**0.7**	0.651	0.298	0.156	8.82E-2	5.28E-2	0.320	0.105
1	**1.5**	**0.7**	0.707	0.271	0.125	6.43E-2	3.65E-2	0.328	8.13E-2
1	**2**	**0.7**	0.738	0.250	0.106	5.24E-2	2.94E-2	0.323	6.44E-2
1	**3**	**0.7**	0.770	0.222	8.60E-2	4.15E-2	2.37E-2	0.303	4.35E-2
1	**4**	**0.7**	0.784	0.204	7.60E-2	3.67E-2	2.14E-2	0.282	3.16E-2
0.5	**0.125**	**0.7**	0.344	0.284	0.242	0.209	0.182	0.215	0.167
0.5	**0.25**	**0.7**	0.466	0.329	0.249	0.193	0.153	0.282	0.177
0.5	**0.5**	**0.7**	0.599	0.340	0.216	0.144	9.91E-2	0.349	0.160
0.5	**1**	**0.7**	0.706	0.309	0.159	8.85E-2	5.24E-2	0.377	0.118
0.5	**1.5**	**0.7**	0.747	0.277	0.126	6.39E-2	3.59E-2	0.369	8.87E-2
0.5	**2**	**0.7**	0.767	0.253	0.106	5.15E-2	2.86E-2	0.353	6.91E-2
0.5	**3**	**0.7**	0.784	0.220	8.40E-2	3.97E-2	2.23E-2	0.321	4.57E-2
0.5	**4**	**0.7**	0.790	0.200	7.30E-2	3.46E-2	1.98E-2	0.294	3.28E-2
0.25	**0.125**	**0.7**	0.356	0.277	0.230	0.196	0.168	0.246	0.178
0.25	**0.25**	**0.7**	0.476	0.316	0.233	0.178	0.138	0.318	0.186
0.25	**0.5**	**0.7**	0.584	0.317	0.197	0.129	8.66E-2	0.366	0.163
0.25	**1**	**0.7**	0.656	0.277	0.138	7.38E-2	4.19E-2	0.371	0.116
0.25	**1.5**	**0.7**	0.679	0.241	0.104	4.99E-2	2.62E-2	0.355	8.54E-2
0.25	**2**	**0.7**	0.688	0.215	8.40E-2	3.78E-2	1.93E-2	0.336	6.61E-2
0.25	**3**	**0.7**	0.693	0.181	6.26E-2	2.67E-2	1.36E-2	0.304	4.35E-2
0.25	**4**	**0.7**	0.693	0.161	5.21E-2	2.20E-2	1.15E-2	0.278	3.12E-2
0.125	**0.125**	**0.7**	0.358	0.262	0.213	0.179	0.153	0.266	0.180
0.125	**0.25**	**0.7**	0.445	0.287	0.208	0.157	0.121	0.313	0.179
0.125	**0.5**	**0.7**	0.516	0.276	0.169	0.108	7.12E-2	0.337	0.151
0.125	**1**	**0.7**	0.562	0.231	0.111	5.65E-2	3.02E-2	0.332	0.105
0.125	**1.5**	**0.7**	0.575	0.196	7.92E-2	3.49E-2	1.66E-2	0.316	7.75E-2
0.125	**2**	**0.7**	0.579	0.171	6.09E-2	2.43E-2	1.09E-2	0.300	6.00E-2
0.125	**3**	**0.7**	0.579	0.138	4.19E-2	1.51E-2	6.55E-3	0.272	3.96E-2
0.125	**4**	**0.7**	0.576	0.119	3.29E-2	1.16E-2	5.15E-3	0.251	2.86E-2
0.0625	**0.125**	**0.7**	0.337	0.242	0.195	0.163	0.139	0.267	0.179
0.0625	**0.25**	**0.7**	0.396	0.254	0.183	0.138	0.106	0.296	0.173
0.0625	**0.5**	**0.7**	0.447	0.238	0.144	9.06E-2	5.86E-2	0.312	0.145
0.0625	**1**	**0.7**	0.480	0.193	8.95E-2	4.37E-2	2.21E-2	0.307	0.102
0.0625	**1.5**	**0.7**	0.488	0.160	6.10E-2	2.49E-2	1.07E-2	0.294	7.58E-2
0.0625	**2**	**0.7**	0.489	0.136	4.49E-2	1.61E-2	6.24E-3	0.281	5.91E-2
0.0625	**3**	**0.7**	0.486	0.107	2.85E-2	8.76E-3	3.18E-3	0.258	3.96E-2
0.0625	**4**	**0.7**	0.482	8.90E-2	2.09E-2	6.13E-3	2.34E-3	0.240	2.88E-2

Table IV.5. Shape factors (i_0 to i_4) for infinite circumferential through clad defect (E_1/E_2 = 1, 0.7)

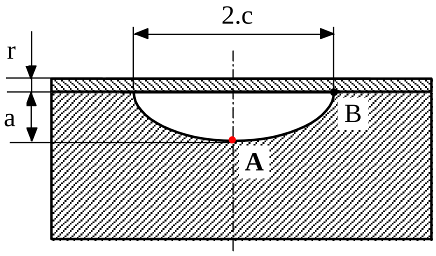
(a+r)/(s+r)	i_0	i_1	i_2	i_3	i_4
0.00	1.122	0.683	0.526	0.441	0.378
0.10	1.155	0.694	0.530	0.443	0.387
0.20	1.255	0.735	0.554	0.459	0.399
0.30	1.400	0.791	0.586	0.480	0.415
0.40	1.583	0.861	0.625	0.506	0.433
0.50	1.804	0.946	0.672	0.537	0.455
0.60	2.060	1.046	0.726	0.572	0.481
0.70	2.352	1.158	0.791	0.615	0.512
0.80	2.700	1.307	0.880	0.676	0.557

Table IV.6. Shape factors (i_0 to i_4) for infinite longitudinal through clad defect (E_1/E_2 = 1, 0.7)

(a+r)/(s+r)	i_0	i_1	i_2	i_3	i_4
0.00	1.122	0.683	0.526	0.441	0.387
0.10	1.176	0.702	0.535	0.446	0.390
0.20	1.338	0.767	0.572	0.471	0.408
0.30	1.592	0.865	0.627	0.507	0.434
0.40	1.959	1.004	0.704	0.558	0.470
0.50	2.481	1.197	0.810	0.626	0.519
0.60	3.222	1.467	0.955	0.719	0.584
0.70	4.253	1.837	1.152	0.844	0.672
0.80	5.535	2.297	1.397	0.999	0.781

Table IV.7. Shape factors (i_{0r} and i_{1r}) for infinite longitudinal or circumferential through clad defect (E_1/E_2 = 1, 0.7)

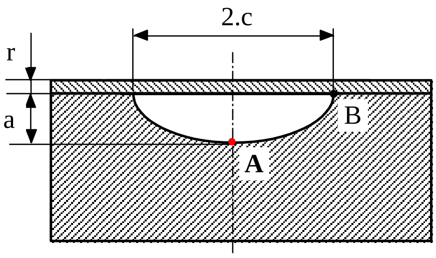
r/(a+r)	i_{0r}	i_{1r}
0.000	0.000	0.000
0.050	0.056	0.002
0.100	0.107	0.006
0.150	0.154	0.012
0.200	0.197	0.021
0.250	0.238	0.031
0.300	0.276	0.042
0.350	0.313	0.055
0.400	0.350	0.070
0.450	0.387	0.087
0.500	0.425	0.106
0.550	0.464	0.128
0.600	0.505	0.153
0.650	0.550	0.181
0.700	0.598	0.214
0.750	0.651	0.252
0.800	0.710	0.296
0.850	0.774	0.348
0.900	0.845	0.409



Sub-clad defect

Table IV.8. Influence functions for an underclad defect (E_1/E_2 = 1, point A)

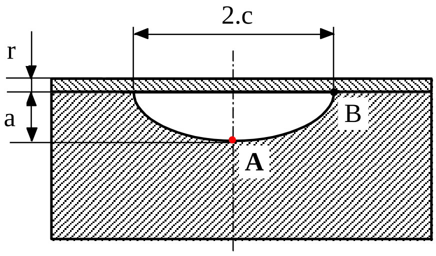
a / c	a / r	E_1 / E_2	i_0	i_1	i_2	i_3	i_4
1	**0**	**1**	0.550	0.550	0.550	0.550	0.550
1	**0.125**	**1**	0.550	0.537	0.524	0.512	0.500
1	**0.25**	**1**	0.551	0.526	0.504	0.484	0.466
1	**0.5**	**1**	0.551	0.511	0.477	0.448	0.423
1	**1**	**1**	0.554	0.493	0.445	0.409	0.379
1	**1.5**	**1**	0.557	0.482	0.429	0.389	0.357
1	**2**	**1**	0.560	0.476	0.418	0.376	0.345
1	**3**	**1**	0.565	0.468	0.406	0.363	0.331
1	**4**	**1**	0.570	0.464	0.399	0.355	0.323
0.5	**0**	**1**	0.642	0.642	0.642	0.642	0.642
0.5	**0.125**	**1**	0.643	0.626	0.610	0.594	0.579
0.5	**0.25**	**1**	0.645	0.613	0.585	0.559	0.536
0.5	**0.5**	**1**	0.647	0.594	0.550	0.513	0.481
0.5	**1**	**1**	0.653	0.572	0.511	0.464	0.426
0.5	**1.5**	**1**	0.660	0.560	0.490	0.439	0.400
0.5	**2**	**1**	0.666	0.553	0.478	0.425	0.385
0.5	**3**	**1**	0.677	0.545	0.463	0.408	0.368
0.5	**4**	**1**	0.686	0.541	0.456	0.399	0.359
0.25	**0**	**1**	0.677	0.677	0.677	0.677	0.677
0.25	**0.125**	**1**	0.680	0.661	0.643	0.626	0.610
0.25	**0.25**	**1**	0.682	0.648	0.617	0.590	0.564
0.25	**0.5**	**1**	0.687	0.629	0.581	0.540	0.505
0.25	**1**	**1**	0.696	0.607	0.540	0.488	0.448
0.25	**1.5**	**1**	0.706	0.595	0.518	0.462	0.420
0.25	**2**	**1**	0.714	0.588	0.505	0.447	0.404
0.25	**3**	**1**	0.729	0.581	0.491	0.430	0.386
0.25	**4**	**1**	0.741	0.578	0.483	0.421	0.377
0.125	**0**	**1**	0.689	0.689	0.689	0.689	0.689
0.125	**0.125**	**1**	0.692	0.673	0.654	0.637	0.621
0.125	**0.25**	**1**	0.696	0.660	0.629	0.600	0.574
0.125	**0.5**	**1**	0.701	0.642	0.592	0.550	0.515
0.125	**1**	**1**	0.713	0.620	0.551	0.498	0.456
0.125	**1.5**	**1**	0.723	0.609	0.529	0.471	0.427
0.125	**2**	**1**	0.732	0.602	0.516	0.456	0.411
0.125	**3**	**1**	0.749	0.596	0.502	0.439	0.393
0.125	**4**	**1**	0.764	0.594	0.495	0.430	0.384



Sub-clad defect

Table IV.9. Influence functions for an underclad defect (E_1/E_2 = 1, point A) *(continued)*

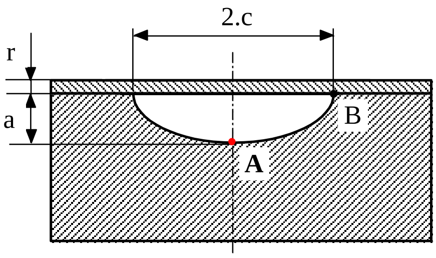
a / c	a / r	E1 / E2	i_0	i_1	i_2	i_3	i_4
0.0625	**0**	**1**	0.693	0.693	0.693	0.693	0.693
0.0625	**0.125**	**1**	0.696	0.677	0.658	0.641	0.624
0.0625	**0.25**	**1**	0.700	0.665	0.633	0.604	0.578
0.0625	**0.5**	**1**	0.706	0.646	0.596	0.554	0.518
0.0625	**1**	**1**	0.718	0.625	0.555	0.501	0.459
0.0625	**1.5**	**1**	0.729	0.613	0.533	0.475	0.430
0.0625	**2**	**1**	0.739	0.607	0.520	0.459	0.414
0.0625	**3**	**1**	0.757	0.601	0.506	0.442	0.396
0.0625	**4**	**1**	0.774	0.600	0.499	0.434	0.387
0	**0**	**1**	0.680	0.680	0.680	0.680	0.680
0	**0.125**	**1**	0.681	0.661	0.643	0.625	0.608
0	**0.25**	**1**	0.683	0.647	0.615	0.586	0.559
0	**0.5**	**1**	0.687	0.627	0.576	0.534	0.498
0	**1**	**1**	0.699	0.606	0.536	0.482	0.439
0	**1.5**	**1**	0.710	0.594	0.513	0.455	0.410
0	**2**	**1**	0.720	0.588	0.501	0.439	0.394
0	**3**	**1**	0.741	0.583	0.487	0.423	0.377
0	**4**	**1**	0.760	0.584	0.482	0.415	0.369



Sub-clad defect

Table IV.10. Influence functions for an under clad defect (E_1/E_2 = 0.7, point A)

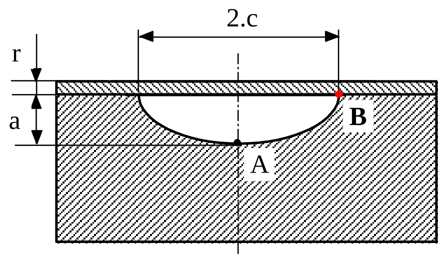
a / c	a / r	E_1 / E_2	i_0	i_1	i_2	i_3	i_4
1	**0**	**0.7**	0.558	0.558	0.558	0.558	0.558
1	**0.125**	**0.7**	0.558	0.544	0.531	0.518	0.506
1	**0.25**	**0.7**	0.559	0.533	0.511	0.490	0.471
1	**0.5**	**0.7**	0.560	0.517	0.482	0.452	0.426
1	**1**	**0.7**	0.562	0.498	0.449	0.411	0.381
1	**1.5**	**0.7**	0.565	0.487	0.432	0.391	0.359
1	**2**	**0.7**	0.568	0.480	0.421	0.378	0.346
1	**3**	**0.7**	0.573	0.472	0.408	0.364	0.332
1	**4**	**0.7**	0.578	0.468	0.401	0.357	0.324
0.5	**0**	**0.7**	0.657	0.657	0.657	0.657	0.657
0.5	**0.125**	**0.7**	0.658	0.640	0.622	0.606	0.590
0.5	**0.25**	**0.7**	0.660	0.626	0.596	0.569	0.545
0.5	**0.5**	**0.7**	0.662	0.606	0.560	0.521	0.487
0.5	**1**	**0.7**	0.669	0.583	0.519	0.470	0.431
0.5	**1.5**	**0.7**	0.675	0.570	0.497	0.444	0.403
0.5	**2**	**0.7**	0.681	0.562	0.484	0.428	0.388
0.5	**3**	**0.7**	0.692	0.553	0.468	0.411	0.370
0.5	**4**	**0.7**	0.701	0.549	0.460	0.402	0.361
0.25	**0**	**0.7**	0.695	0.695	0.695	0.695	0.695
0.25	**0.125**	**0.7**	0.698	0.678	0.659	0.642	0.625
0.25	**0.25**	**0.7**	0.701	0.665	0.632	0.603	0.576
0.25	**0.5**	**0.7**	0.706	0.645	0.593	0.550	0.514
0.25	**1**	**0.7**	0.716	0.621	0.550	0.496	0.453
0.25	**1.5**	**0.7**	0.726	0.608	0.527	0.469	0.424
0.25	**2**	**0.7**	0.734	0.601	0.513	0.452	0.408
0.25	**3**	**0.7**	0.749	0.592	0.497	0.434	0.389
0.25	**4**	**0.7**	0.761	0.589	0.489	0.425	0.380
0.125	**0**	**0.7**	0.709	0.709	0.709	0.709	0.709
0.125	**0.125**	**0.7**	0.712	0.691	0.672	0.654	0.636
0.125	**0.25**	**0.7**	0.716	0.679	0.645	0.615	0.587
0.125	**0.5**	**0.7**	0.723	0.659	0.606	0.562	0.524
0.125	**1**	**0.7**	0.734	0.636	0.562	0.506	0.462
0.125	**1.5**	**0.7**	0.745	0.623	0.539	0.478	0.433
0.125	**2**	**0.7**	0.755	0.616	0.525	0.462	0.416
0.125	**3**	**0.7**	0.772	0.608	0.510	0.444	0.397
0.125	**4**	**0.7**	0.786	0.605	0.502	0.435	0.388



Sub-clad defect

Table IV.11. Influence functions for an under clad defect (E_1/E_2 = 0.7, point A) *(continued)*

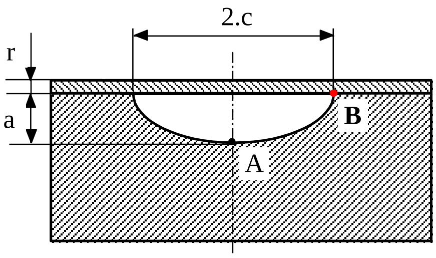
a / c	a / r	E1 / E2	i_0	i_1	i_2	i_3	i_4
0.0625	**0**	**0.7**	0.713	0.713	0.713	0.713	0.713
0.0625	**0.125**	**0.7**	0.717	0.696	0.676	0.658	0.640
0.0625	**0.25**	**0.7**	0.721	0.684	0.650	0.619	0.591
0.0625	**0.5**	**0.7**	0.728	0.664	0.611	0.566	0.528
0.0625	**1**	**0.7**	0.741	0.641	0.567	0.510	0.466
0.0625	**1.5**	**0.7**	0.752	0.628	0.543	0.482	0.436
0.0625	**2**	**0.7**	0.762	0.621	0.529	0.466	0.419
0.0625	**3**	**0.7**	0.781	0.615	0.514	0.448	0.400
0.0625	**4**	**0.7**	0.797	0.613	0.507	0.439	0.391
0	**0**	**0.7**	0.702	0.702	0.702	0.702	0.702
0	**0.125**	**0.7**	0.703	0.682	0.662	0.643	0.625
0	**0.25**	**0.7**	0.705	0.666	0.632	0.601	0.573
0	**0.5**	**0.7**	0.709	0.645	0.591	0.546	0.508
0	**1**	**0.7**	0.722	0.622	0.548	0.491	0.446
0	**1.5**	**0.7**	0.733	0.609	0.524	0.463	0.416
0	**2**	**0.7**	0.744	0.603	0.510	0.446	0.399
0	**3**	**0.7**	0.766	0.597	0.496	0.429	0.381
0	**4**	**0.7**	0.785	0.597	0.490	0.421	0.373



Sub-clad defect

Table IV.12. Influence functions for an underclad defect (E_1/E_2 = 1, point B)

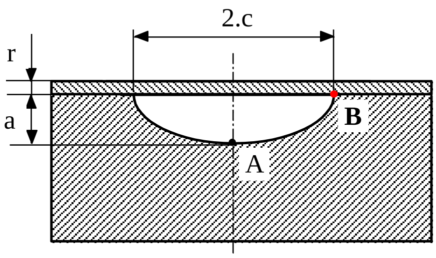
a / c	a / r	E_1 / E_2	i_0	i_1	i_2	i_3	i_4
1	**0**	**1**	0.263	0.263	0.263	0.263	0.263
1	**0.125**	**1**	0.264	0.238	0.214	0.193	0.174
1	**0.25**	**1**	0.264	0.217	0.178	0.147	0.121
1	**0.5**	**1**	0.265	0.186	0.131	9.27E-2	6.61E-2
1	**1**	**1**	0.267	0.148	8.32E-2	4.80E-2	2.85E-2
1	**1.5**	**1**	0.270	0.125	6.05E-2	3.08E-2	1.67E-2
1	**2**	**1**	0.272	0.111	4.79E-2	2.26E-2	1.19E-2
1	**3**	**1**	0.277	0.093	3.51E-2	1.56E-2	8.21E-3
1	**4**	**1**	0.281	0.083	2.90E-2	1.27E-2	6.91E-3
0.5	**0**	**1**	0.244	0.244	0.244	0.244	0.244
0.5	**0.125**	**1**	0.245	0.220	0.197	0.177	0.160
0.5	**0.25**	**1**	0.245	0.200	0.164	0.134	0.110
0.5	**0.5**	**1**	0.246	0.171	0.119	8.36E-2	5.89E-2
0.5	**1**	**1**	0.248	0.135	7.47E-2	4.20E-2	2.42E-2
0.5	**1.5**	**1**	0.251	0.114	5.34E-2	2.61E-2	1.35E-2
0.5	**2**	**1**	0.253	0.100	4.16E-2	1.86E-2	9.16E-3
0.5	**3**	**1**	0.257	0.083	2.95E-2	1.21E-2	5.91E-3
0.5	**4**	**1**	0.261	0.073	2.38E-2	9.58E-3	4.80E-3
0.25	**0**	**1**	0.215	0.215	0.215	0.215	0.215
0.25	**0.125**	**1**	0.216	0.193	0.173	0.155	0.139
0.25	**0.25**	**1**	0.216	0.176	0.143	0.116	9.45E-2
0.25	**0.5**	**1**	0.217	0.149	0.103	7.11E-2	4.92E-2
0.25	**1**	**1**	0.219	0.117	6.26E-2	3.40E-2	1.87E-2
0.25	**1.5**	**1**	0.220	9.70E-2	4.34E-2	2.00E-2	9.50E-3
0.25	**2**	**1**	0.222	8.40E-2	3.28E-2	1.34E-2	5.87E-3
0.25	**3**	**1**	0.224	6.80E-2	2.20E-2	7.86E-3	3.24E-3
0.25	**4**	**1**	0.227	5.87E-2	1.69E-2	5.71E-3	2.40E-3
0.125	**0**	**1**	0.185	0.185	0.185	0.185	0.185
0.125	**0.125**	**1**	0.185	0.165	0.147	0.132	0.118
0.125	**0.25**	**1**	0.185	0.150	0.121	9.79E-2	7.92E-2
0.125	**0.5**	**1**	0.185	0.126	8.61E-2	5.88E-2	4.02E-2
0.125	**1**	**1**	0.186	9.74E-2	5.11E-2	2.70E-2	1.43E-2
0.125	**1.5**	**1**	0.187	8.01E-2	3.46E-2	1.51E-2	6.70E-3
0.125	**2**	**1**	0.188	6.86E-2	2.54E-2	9.63E-3	3.78E-3
0.125	**3**	**1**	0.190	5.44E-2	1.61E-2	5.08E-3	1.76E-3
0.125	**4**	**1**	0.191	4.61E-2	1.18E-2	3.37E-3	1.16E-3



Sub-clad defect

Table IV.13. Influence functions for an underclad defect (E_1/E_2 = 1, point B) (continued)

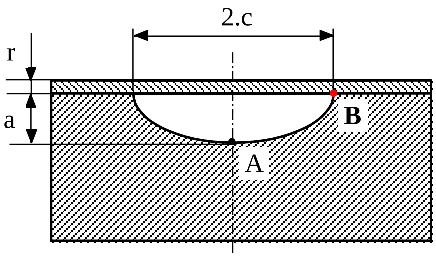
a / c	a / r	E1 / E2	i_0	i_1	i_2	i_3	i_4
0.0625	**0**	**1**	0.156	0.156	0.156	0.156	0.156
0.0625	**0.125**	**1**	0.156	0.139	0.124	0.111	9.91E-2
0.0625	**0.25**	**1**	0.156	0.126	0.102	8.20E-2	6.62E-2
0.0625	**0.5**	**1**	0.156	0.106	7.19E-2	4.87E-2	3.31E-2
0.0625	**1**	**1**	0.157	8.11E-2	4.20E-2	2.18E-2	1.13E-2
0.0625	**1.5**	**1**	0.158	6.62E-2	2.79E-2	1.18E-2	5.05E-3
0.0625	**2**	**1**	0.158	5.63E-2	2.02E-2	7.31E-3	2.70E-3
0.0625	**3**	**1**	0.159	4.40E-2	1.24E-2	3.60E-3	1.11E-3
0.0625	**4**	**1**	0.160	3.68E-2	8.74E-3	2.23E-3	6.65E-4



Sub-clad defect

Table IV.14. Influence functions for an underclad defect (E_1/E_2 = 0.7, point B)

a / c	a / r	E_1 / E_2	i_0	i_1	i_2	i_3	i_4
1	**0**	**0.7**	0.284	0.284	0.284	0.284	0.284
1	**0.125**	**0.7**	0.284	0.256	0.231	0.208	0.187
1	**0.25**	**0.7**	0.284	0.233	0.192	0.158	0.131
1	**0.5**	**0.7**	0.285	0.200	0.141	0.101	7.20E-2
1	**1**	**0.7**	0.287	0.160	9.04E-2	5.25E-2	3.14E-2
1	**1.5**	**0.7**	0.290	0.136	6.61E-2	3.39E-2	1.86E-2
1	**2**	**0.7**	0.293	0.120	5.25E-2	2.51E-2	1.33E-2
1	**3**	**0.7**	0.298	0.101	3.87E-2	1.74E-2	9.26E-3
1	**4**	**0.7**	0.302	8.99E-2	3.20E-2	1.43E-2	7.79E-3
0.5	**0**	**0.7**	0.262	0.262	0.262	0.262	0.262
0.5	**0.125**	**0.7**	0.262	0.236	0.212	0.191	0.171
0.5	**0.25**	**0.7**	0.263	0.215	0.176	0.144	0.118
0.5	**0.5**	**0.7**	0.264	0.184	0.129	9.04E-2	6.39E-2
0.5	**1**	**0.7**	0.266	0.146	8.09E-2	4.58E-2	2.66E-2
0.5	**1.5**	**0.7**	0.269	0.123	5.82E-2	2.87E-2	1.50E-2
0.5	**2**	**0.7**	0.271	0.108	4.55E-2	2.06E-2	1.03E-2
0.5	**3**	**0.7**	0.276	0.090	3.26E-2	1.36E-2	6.73E-3
0.5	**4**	**0.7**	0.280	0.079	2.64E-2	1.08E-2	5.48E-3
0.25	**0**	**0.7**	0.229	0.229	0.229	0.229	0.229
0.25	**0.125**	**0.7**	0.230	0.206	0.184	0.165	0.148
0.25	**0.25**	**0.7**	0.230	0.187	0.152	0.124	0.101
0.25	**0.5**	**0.7**	0.231	0.159	0.110	7.61E-2	5.28E-2
0.25	**1**	**0.7**	0.233	0.125	6.72E-2	3.66E-2	2.03E-2
0.25	**1.5**	**0.7**	0.235	1.04E-1	4.68E-2	2.17E-2	1.04E-2
0.25	**2**	**0.7**	0.236	9.00E-2	3.55E-2	1.47E-2	6.51E-3
0.25	**3**	**0.7**	0.239	7.32E-2	2.40E-2	8.71E-3	3.66E-3
0.25	**4**	**0.7**	0.242	6.33E-2	1.85E-2	6.39E-3	2.74E-3
0.125	**0**	**0.7**	0.195	0.195	0.195	0.195	0.195
0.125	**0.125**	**0.7**	0.195	0.174	0.155	0.139	0.124
0.125	**0.25**	**0.7**	0.195	0.158	0.128	0.103	8.37E-2
0.125	**0.5**	**0.7**	0.195	0.133	9.10E-2	6.22E-2	4.26E-2
0.125	**1**	**0.7**	0.196	0.103	5.41E-2	2.86E-2	1.52E-2
0.125	**1.5**	**0.7**	0.197	8.47E-2	3.67E-2	1.61E-2	7.19E-3
0.125	**2**	**0.7**	0.198	7.26E-2	2.70E-2	1.03E-2	4.09E-3
0.125	**3**	**0.7**	0.200	5.77E-2	1.73E-2	5.50E-3	1.94E-3
0.125	**4**	**0.7**	0.202	4.90E-2	1.27E-2	3.69E-3	1.30E-3



Sub-clad defect

Table IV.15. Influence functions for an underclad defect (E_1/E_2 = 0.7, point B) (continued)

a / c	a / r	E1 / E2	i_0	i_1	i_2	i_3	i_4
0.0625	**0**	**0.7**	0.163	0.163	0.163	0.163	0.163
0.0625	**0.125**	**0.7**	0.163	0.146	0.130	0.116	0.104
0.0625	**0.25**	**0.7**	0.163	0.132	0.106	8.57E-2	6.92E-2
0.0625	**0.5**	**0.7**	0.163	0.111	7.51E-2	5.10E-2	3.46E-2
0.0625	**1**	**0.7**	0.164	8.48E-2	4.39E-2	2.28E-2	1.19E-2
0.0625	**1.5**	**0.7**	0.165	6.93E-2	2.93E-2	1.24E-2	5.32E-3
0.0625	**2**	**0.7**	0.165	5.89E-2	2.12E-2	7.70E-3	2.85E-3
0.0625	**3**	**0.7**	0.166	4.61E-2	1.30E-2	3.82E-3	1.19E-3
0.0625	**4**	**0.7**	0.167	3.86E-2	9.23E-3	2.39E-3	7.23E-4

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APPENDIX V

Determination of reference/design fracture toughness curve including "master curve" approach

Content of this Appendix is an input information for calculation of resistance against fast fracture - Chapter 5 of the main document.

1. The temperature dependence of fracture toughness for a given material can be determined in two equivalent ways:

- Reference fracture toughness curve based on reference temperature T0 determined from direct fracture toughness testing.
- Design fracture toughness curve based on empirical correlation between critical temperature of brittleness Tk, and fracture toughness KIC.

2. The static fracture toughness reference curve is based on a reference temperature T_0 (Master Curve approach). In this case, a 5% tolerance lower bound can be applied in the integrity/ lifetime evaluation:

$$K_{Jc(0.05)} = 25.2 + 36.6 \cdot \exp\left[\,0.019\,(\,T - RT_0\,)\,\right] \qquad (V.1)$$

where the reference temperature RT_0 has been determined for the condition of component integrity evaluation in accordance with the ASTM standard E 1921-05 (including standard deviation).Any other value for the lower tolerance bound should be negotiated in advance with a national regulatory body.

2.1. The application of the "Master Curve" approach has been validated for steels and welding joints of 15Kh2MFA(A), 15Kh2NMFA(A), 22K and 10GN2MFA type steels. Application of this approach for other type of steels should be validated by a proper statistical number of material tests in conditions with representative and/or simulated operating conditions close to the integrity evaluation time.

3. The design static fracture toughness curves $[K_{IC}]_3$ for component integrity assessment and lifetime can be used in the following form:

$$[K_{IC}]_3 = \min\{26 + 36\,.\,\exp\,[0.02\,.\,(\,T - T_k\,)];\ 200\} \qquad (V.2)$$

where T_k is a critical temperature of brittleness for the evaluated time of operation. This formula is valid for steels and weld metals of 15Kh2MFA(A), 15Kh2NMFA(A), 10GN2MFA(A) and 22K types.

3.1. For steels 15Kh2MFA(A), 15Kh2NMFA(A) the following dependence of fracture toughness [R1,R2] can be used

$$\overline{K}_{IC}\,(T) = \min\{23+48\exp[0.019(T-T_k)];\ K_{JC}^{US}(T,F)\}, \qquad (V.3)$$

for reference thickness B=150 mm and for fracture probability P_f=0.05.

Dependence of the fracture toughness value of 15Kh2MFA(A), and 15Kh2NMFA(A) type of steels at the upper shelf on neutron fluence and temperature $K_{JC}^{US}(T,F)$ of base and weld metal is calculated by the following formula:

$$K_{JC}^{US}(T,F) = \sqrt{\frac{J_C^{US}(T,F)\cdot E}{1-\nu^2}}\,, \qquad (V.4)$$

where $$J_C^{US}(T,F) = J_C^* \cdot [1 - C \cdot \Delta T_K - B] \cdot \frac{R_{P0.2}(T,F)}{R_{P0.2}(20,F)},$$ (V.5)

J_C^* – value of J_C when neutron fluence F=0 and temperature T=20°C,

$R_{p0.2}(T,F)$ – yield strength of irradiated base or weld metal at temperature T.

When calculating zones of RPV, where neutron fluence is:

$F<10^{22}$ 1/m² it is assumed that J_C^* =280 N/mm,

$F\geq10^{22}$ 1/m² it is assumed that J_C^* =175 N/mm.

Coefficients C and B are assumed to be equal: $C=2.4\cdot10^{-3}$, $°C^{-1}$; B=0.14.

3.2. Use of these formulas for other types of steels and/or welding joints should have to be validated by Qualification Tests. These tests should contain semiproducts, in their final heat treatment, with dimensions/thicknesses equivalent to dimensions of the real components. Specimen thicknesses should be chosen in such a way that they should cover real component thicknesses. The number of tests should be sufficient for a proper statistical evaluation of experimental data.

4. The crack arrest fracture toughness reference curve can be expressed in the following way:

4.1. When crack arrest fracture toughness data are available, then a 5 % tolerance lower bound is applied (which is similar to static fracture toughness reference curve):

$$K_{IA(0.05)} = 25.2 + 36.6 \cdot \exp[0.019(T - RT_0^A)]$$ (V.6)

where K_{IA} is the arrest fracture toughness, RT_0^A is the reference temperature RT_0 for crack arrest fracture toughness tests.

4.2. If crack arrest fracture toughness data are not available, then the arrest reference temperature T_0^A in formula (3) can be taken as follows:

$$RT_0^A = RT_0 + 30\ °C$$ (V.7)

where RT_0 is defined in Appendix III, formula (V.5).

4.3. The crack arrest fracture toughness design curve can be expressed in the following way, based on critical temperature of brittleness, T_k, as follows:

$$[K_{IA}]_3 = \min\{26 + 36\cdot\exp[0.020.(T-RT_k-30)];\ 200\}$$ (V.8)

5. Use of another fracture toughness design/reference curve for any material should be validated by Qualification tests. These tests should contain semiproducts, in their final heat treatment, with dimensions/thicknesses equivalent to the dimensions of real components. Specimen thicknesses should be chosen in such a way that they should cover real component thicknesses. The number of tests should be sufficient for a proper statistical evaluation of experimental data.

5.1. The new/modified fracture toughness design curve (based on critical temperature of brittleness, T_k) could be determined with the use of a statistical analysis as a 99% lower boundary of all experimental data from tests of all test thicknesses up to real component thickness.

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APPENDIX VI

Requirements for pressurized thermal shocks (PTS) selection and thermal hydraulic calculations

List of abbreviations

BRU-A	Atmospheric dump valve
BRU-K	Turbine bypass valve
CFD	Computational fluid dynamic (code)
CVCS	Chemical and Volume Control System
ECCS	Emergency Core Cooling System
FW	Feed Water
FWLB	Feed Water Line Break
HPIS	High Pressure Safety Injection
LB LOCA	Large Break LOCA
LOCA	Loss-of-coolant accident
LTOP	Low Temperature Overpressure
MSL	Main Steam Line
MSLB	Main steam line break
MSS	Main Steam System
NPP	Nuclear power plant
PRZ	Pressurizer
PSA	Probabilistic Safety Analysis
PTS	Pressurized thermal shock
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
RPV	Reactor pressure vessel
SB LOCA	Small Break LOCA
SG	Steam generator
SI	Safety injection
TH	Thermal hydraulic
WWER	Nuclear power plant with water-water energy reactor

1. Sequences to be considered

1.1 General considerations

The loads to be considered in the reactor pressure vessel integrity assessment are mainly related to plant states leading to pressurized thermal shock (PTS) events, which are characterized by a rapid cooldown in the primary coolant system usually with a high primary system pressure. Such events depend strongly on the actual plant status, plant configuration, systems operation, and operator actions.

It is also necessary to perform the integrity assessment for Low Temperature Overpressure (LTOP) events when primary pressure exceeds its allowable value while the coolant temperature is low (it happens mainly during start-up or shutdown). The requirements for thermal hydraulic calculations for LTOP are the same as for PTS or lower (no need for detailed mixing calculation) and are therefore not distinguished as different events in the following text.

The selection of the 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 [2] 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 neighbouring 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 [2] in PTS analysis is not straightforward and should be carefully evaluated. Attention should be paid mainly to differences as compared to the accident analysis performed with respect to the core cooling.

The selection of the transients for deterministic analysis can be based on engineering judgement using the design basis accident analysis approach combined with the operational experience accumulated at WWER plants.

The most comprehensive and effective approach for 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.

Cooperation of TH and PSA experts is strongly recommended in this phase of PTS evaluation - to achieve a comprehensive list of PTS relevant initiating events and scenarios to be analysed in frame of PTS evaluation.

If the warm pre-stressing approach based on par. 5.10.7 and 5.10.8 of the main part is used for integrity evaluation, it is necessary, for conservative determining of the local minimum points on the K_I vs. T curve, that the corresponding thermal-hydraulic calculations are, from the appropriate viewpoint, conservative (regarding to maximum possible unloading levels, e.g. due to temporary switch-off or switch-over of high pressure ECCS, temporary opening of pressurizer safety or relief valve, for both intended and non-intended operator actions). If the conservativeness of the unloading cannot be ensured, maximum possible unloading must be assumed.

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 for determining thermal and mechanical loading mechanisms in the downcomer during the overcooling events. These factors are:

— Final temperature in the downcomer (a lower final temperature is more unfavourable);

— Temperature decrease rate (a higher temperature decrease rate is more unfavourable);

— Level of primary pressure (a higher pressure is more unfavourable);

— Non-uniform cooling of the RPV, characterized by the cold plumes and their interaction, and by the nonuniformity of the coolant-to-wall heat transfer coefficient in the downcomer (a higher nonuniformity is more unfavourable);

— Width of the cold plume (a narrower plume is more unfavourable);

— Initial temperature in the downcomer (a higher initial temperature is more unfavourable);

— Stratification or stagnation of flow in the cold leg (a lower flow rate in cold leg or total flow stagnation is more unfavourable).

The possibility of outside cooling of the RPV (PTS at outer surface of RPV) also needs to be considered at WWER plants. In some situations, the reactor cavity can be flooded from the ECCS or the spray system or by the secondary water (e.g. in FWLB) or by technical cooling water or other source of water with piping in hermetic zone. 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 WWER-440/213 (which is equipped with a pressure suppression system), some accident scenarios have cold water from barbotage trays spilled onto the floor of the SG boxes (e.g., medium and large break LOCAs) and hence, after reaching overflow level, the reactor cavity can be flooded. Similarly, in the case of Loviisa NPP (which is equipped with an ice condenser containment), the reactor cavity is flooded after melting the ice condenser in most of the scenarios with high energy coolant release into the containment. The flooding of reactor cavity by cold water in combination with high parameters in the reactor vessel (pressure and temperature) creates conditions for severe PTS at outer RPV surface.

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

1.2 Initiating events groups

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

Loss of coolant accidents

Different sizes of both cold and hot leg loss of coolant accidents (LOCA) should be considered. Attention should be paid to the scenarios leading to flow stagnation, which causes a faster cooldown rate, and cold plumes in the downcomer. Attention should be given to breaks sizes corresponding to existing piping connected to the 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. Both the full and zero reactor power should be analysed as the initial conditions. Also the variant analyses with different numbers of working ECCS trains should 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 main loop flow stagnation. In addition, the "feed&bleed" method of mitigation for initiating event ot type of e.g. 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. If the operator is requested to isolate the affected steam generator, the successful isolation may lead to repressurization.

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, the start of the high-pressure injection due to the low primary pressure (and/or low pressurizer level or directly due to low secondary circuit parameters), which leads to repressurization, can be expected. The degree of secondary side depressurization is strongly dependent on the plant configuration (mainly the presence of the 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), the atmospheric dump valve (BRU-A), and the steam generator safety valve(s);
- Feed water line break (usually bounded by steam line break analysis).

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 pressurisation 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.

There could be two different unsymmetrical cooldown issues in the MSLB event. The first one is a rapid cooldown from loop with affected and depressurized SG. It is connected with strong circulation in this loop and potentially "supported" by injection of safety system into this loop. With respect to strong circulation in the affected loop, there is no thermal stratification in the cold leg and it results in cold sector in reactor downcomer (see below the definition of "cold sector"). The second one is injection of cold SI water into an intact loop with flow stagnation, which leads to thermal stratification in the cold leg and cold plume formation in the relevant part of reactor downcomer.

Inadvertent actuation of the high-pressure injection or make-up systems

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

Accidents resulting in cooling of the RPV from outside

Cooling of the RPV from outside is possible only for WWER 440 NPPs. A 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) should be considered in this group of accidents. Special attention should be paid to the scenarios with flooding or reactor cavity by cold water while there is full pressure and temperature in the primary system (e.g., FW line break and containment spraying). In some WWER NPPs cooling of the RPV from outside is used as mitigation of several accidents, but this case is not part of PTS analyses.

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

The following sequences should be considered for various plant operating conditions: full power, hot zero power, heat up, cool down and cold shutdown. Only events with a frequency of occurrence higher than 10^{-6} per reactor year have to be assessed.

1.3 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 a 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

Anticipated transients 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 continued plant operation.

Postulated accidents

Postulated accidents 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, in the design of plant upgrading, or are based on plant safety reassessment [2]. For these events, immediate resumption of operation may not be possible.

2. Assumptions for PTS analyses

2.1 Systems pertinent to PTS

The systems to be taken into consideration in the PTS analyses are usually the following:

- Reactor Coolant System (RCS);
- Main gate valves;
- Reactor Protection System (RPS);
- Pressurizer (PRZ) and pressure control system;
- Emergency Core Cooling System (ECCS);
- Chemical and Volume Control System (CVCS);

— Main Steam System (MSS);
— Feedwater system;
— Support systems (important for PTS analyses);
— 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 analyses should be determined.

2.2 Assumptions of system operation

Concerning the normal operation and control systems parameters, 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 ECC 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. The systems should be assumed to operate on the maximum installed capacity corresponding head value according to maximum pump characteristics and to inject the lowest possible temperature cooling water to the primary circuit. Time variation of the injected water temperature should be conservatively evaluated (e.g., automatic switching from heated high to non-heated low pressure tanks) along with considering a relevant single failure.

If it deteriorates results of the PTS analysis, the stuck open safety valve should be considered as a consequential failure if the valve is not qualified for the discharged coolant (fluid or steam-water mixture) or if there is a demand for a large number of successive cycles.

Possibly having a 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 the 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 of the primary circuit and deterioration of PTS results, then those systems have to be taken into account.

2.3 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 the related procedures. In the cases where operator action has a favourable impact on the 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 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 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 cases where the circumstances are unambiguously defined by the procedures.

The PTS relevant operator interventions (which make the PTS variation better or worse) might be:

- Trip and restart of the reactor pumps;
- Stopping and restart of the ECCS pumps;
- 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.

2.4 Plant operating conditions

The initial power of the reactor has always to be set to the most conservative value determined by the conditions of the overcooling transient. Full power and hot zero power regimes should be analysed.

The value of the residual heat should be the lowest possible one and is 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 maximally long outage). The estimated error of the residual heat calculation is to be taken into consideration with a negative value. The determination of the residual heat can be based on actual operational measurement information except for cases of low power operation.

Other initial conditions such as reactor flow, initial temperature, initial primary pressure or SG level should be conservatively chosen.

2.5 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 PRZ 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 the 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 the 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 the hot legs;

— In the case of MSLB with trip of all RCPs, a strong natural circulation can be usually observed in the loop with affected and depressurized SG, whereas a flow stagnation can occur in loops with non-affected SG.

The non-uniform temperature and heat transfer coefficient field is created by cold plumes (or cold sectors or cold stripes). Cold plume means non-uniformity in downcomer coolant temperature in both radial and azimuthal direction (typical for SBLOCA with HPIS injection) while cold sector means non-uniformity only in azimuthal direction (typical for MSLB with cooldown of one downcomer sector under affected loop with depressurized SG and strong natural circulation). Both cold plume and cold sector mean cold water input to the downcomer that is full of hot water. On the other side, cold stripe means the input of cold water into the downcomer containing steam (typical for LBLOCA, bleed & feed with deep decrease of reactor levels etc.). They result from safety injection into the cold legs (high pressure injection or part of low-pressure injection) or directly to the downcomer (accumulators or part of low pressure injection).

The non-uniformities in the temperature and the velocity field in the downcomer can affect natural circulation flow rate in individual loops and beginning time for flow stagnation in the individual loops. Therefore, usage of 3D or at least 2D modelling of the reactor downcomer already in system thermal hydraulic analyses is recommended.

As the thermal stratification in the end part of cold leg with safety injection can affect flow in the loop (e.g. if a part of the cold water from SI flows backward to reactor coolant pump, flows over the pump and enters the loop seal, it can form a plug of heavier cold water here, resulting in earlier onset of flow stagnation), it is also recommended to apply 2D modelling of cold leg already at the stage of system TH calculation.

The effects of those temperature non-uniformities are to be taken into account in the analysis in the case of loop flow stagnation or asymmetric secondary side cooling. Even in the case of a uniform temperature field, significant flow rate differences might occur in the downcomer. The non-uniformity of the heat transfer coefficient field has to be investigated in addition to the temperature distribution. For various aspects of PTS scenarios (overall cooldown, 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 of primary circuit 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.

3. Thermal hydraulic analyses

3.1 Objectives of thermal hydraulic analyses

There are two main objectives of thermal hydraulic analyses: to support the transient selection process and to produce the 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;

— 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 heat transfer coefficients are not necessary.)

3.2 Thermal hydraulic analyses 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.

3.3 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 above all the time variation of primary pressure, coolant temperature and velocity in cold legs, and furthermore temperature and velocity of medium injected by emergency systems into the primary circuit.

In case of non-symmetric cooldown and/or flow stagnation of the primary circuit, when buoyancy induced forces dominate the fluid flow behaviour in cold legs and the downcomer, the system code results are not reliable for the 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 to estimate the initiation of the stagnation, and to give the initial and boundary conditions for thermal mixing calculations.

As shown in the latest US NRC project focused on PTS [5], an alternative approach to TH system and mixing analyses could be a single system TH analysis with 2D nodalization of reactor downcomer. When applying this more realistic approach (comparing to 2-stage analysis with system TH code followed by thermal mixing calculation, which can be in some cases very conservative), one should evaluate suitability of this methodology. See more details below in Chapter 3.4.

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.

3.4 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 the system thermal hydraulic analyses. The pressurizer modelling used in the code must be capable of calculating increasing pressure, which can occur after the repressurization of the primary circuit. Non-uniform 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 also be able to determine the azimuthal, axial, and in some cases radial fluid temperature distributions in the downcomer and the azimuthal and axial distributions of the coolant-to-wall heat transfer coefficient.

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 a 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.

An alternative approach to TH system and mixing analyses, suitable for selected types of PTS transients, is a single system TH analysis with (pseudo)2D or (pseudo)3D nodalization of reactor downcomer. This approach and methodology is supported by the latest US NRC project focused on PTS [5]. The experiments carried out on the APEX, LOFT, ROSA, UPTF facilities showed only very small temperature difference between the plume and ambient, and good agreement between these experiments and system TH calculations. It means that the (pseudo)2D or (pseudo)3D representation of reactor downcomer in frame of system TH calculation could be sufficient for prediction of asymmetries in cooldown of RPV. When applying this more realistic approach (comparing to 2-stage analysis with system TH analysis followed by mixing calculation, which can be in some cases very conservative), one should validate suitability of this methodology - to stay in compliance with experimental basis and to avoid non-conservatism of results. Validation of suitability of this methodology can be done e.g. using CFD calculations for selected representative PTS regimes, or at least critical time periods of them. This approach could be suitable e.g. in cases with full flow stagnation in loops, when temperature at reactor inlet becomes in a short time equal to temperature of cold water from Safety Injection (in full cross section of inlet nozzle), i.e. in cases where thermal stratification in cold leg does not play important role.

4. Computer codes

4.1 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. Detailed modelling of ECCS trains for both the injection and recirculation phase of safety injection is recommended, as it enables qualified prediction of both the ECCS performance and resulting temperature of injected water.

The thermal hydraulic models should be capable of predicting single and two-phase flow behaviour and critical flow. 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). If the non-uniform temperature and velocity fields in the reactor downcomer and end part of cold leg (with SI) can influence overall system behaviour - especially circulation rates in individual loops - then 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 the refilling of the primary system with safety injection water. The effects of non-condensable gases (if present) on system pressure and temperature calculations should be included.

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

4.2 Thermal mixing calculations

An important feature of some PTS transients is flow stagnation in the primary circuit. In such a case, 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, optimally, calculations using coupled system and CFD computer codes may be performed. In this case CFD code computes not only fluid flow in reactor, but also in cold leg, including loop seal.

4.3 Code validation

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 of combining them 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 give output for the structural analysis in the form 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 the validation process and the recommendations for practical validation against test data, plant data, and standard problems data have been discussed in more detail in [2]. 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 phenomena like mixing near the injection location, thermal 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.

References

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[4] Kral, P., *NRI Practice in TH Analysis for PTS Evaluation of WWER Reactors in Czech Republic, WWER COVERS Project*, UJV, Rez (2007).

[5] U.S. NUCLEAR REGULATORY COMMISSION, *Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule* (10 CFR 50.61): Summary Report, NUREG-1806, USNRC, Washington, DC (2006).

APPENDIX VII

Residual lifetime of the component damaged by fatigue due to operating loading

1. Basic principles

1.1 Definitions, symbols and units

b	Exponent of fatigue strength
c	Exponent of fatigue ductility
β_H	Notch coefficient
β_n	Coefficient expressing influence of surface finish, $\beta_n \leq 1$
β_σ	Coefficient expressing part size influence, $\beta_\sigma \leq 1$
D	Usage factor
E	Young's modules of elasticity at operating temperature, [MPa]
$\varepsilon_{ap,vr}$	Amplitude of reversible plastic strain
(ε_{at})	Amplitude of total (elastic plus plastic) strain intensity value
ε_f'	Coefficient of fatigue ductility
(ε_t)	Total (elastic plus plastic) strain intensity value
$(\varepsilon_t)_{max}$	Maximum value of total strain intensity value in time of technical lifetime of an component
m	Exponent in fatigue equation
N_C	Number of cycles at the fatigue limit
$[N_o]$	Allowed number of given cycle load type
n_N	Safety factor on number of cycles
n_σ	Safety factor on stress (strains)
φ_F	Strength reduction factor due to radiation
φ_W	Coefficient of strength reduction at fatigue by influence of welds and welded attachments
R_m	Guaranteed value of tensile strength for operating temperature, [MPa]
$R_{p0,2}$	Guaranteed value of yield strength for operating temperature, [MPa]
(σ)	Stress intensity value, [MPa]
σ_{AC}	Fatigue limit at repeated cycle of stress, [MPa]
$(\sigma_{a,nom})$	Amplitude of nominal stress intensity value, [MPa]
σ_C	Fatigue limit at symmetric alternate cycle of stress, [MPa]
σ_f'	Coefficient of fatigue strength, [MPa]
σ_H	Stress calculated in assumption of validity of Hooke´s law in whole load range, [MPa]
(σ_m)	Mean cycle value of stress intensity, [MPa]
$(\sigma_{m,nom})$	Mean cycle value of nominal stress intensity, [MPa]

TC Limit temperature over which creep occurs under elevated temperature, [K, °C].

1.1 The analysis of operating conditions under cyclic loading is performed for components working at temperatures below the creep range temperature, T_C.

1.2 The level and the trend of the material fatigue damage of components shall be determined at their most stressed areas.

1.3 Stresses and strains at the most stressed areas of components shall be calculated from data monitored by the Instrumentation and Control system (I&C), Monitoring and Diagnostic System (MDS) and Temporary Measurement system (TM).

1.4 The corrective measures for operation and maintenance management shall be taken when the given criterions shown in Section 6.3 are not met.

2. Monitoring of the operating parameters

Data used for the calculation of stresses and strains at the most stressed areas of components shall be monitored (e.g., by the I & C, MDS and TM systems).

2.1 Monitored data shall be verified and incorrect data shall be omitted.

2.2 Software used for the removal of incorrect data shall be verified in accordance with relevant quality assurance procedure.

3. Calculation of the stresses and strains

3.1 The calculation of stresses and strains shall be carried out one time during the given period. This calculation can be done automatically by the diagnostic system.

3.2 Influence functions from the neural network method can be used for determining the relationship between stresses and strains in the most loaded areas and between measured data.

3.3 The linear elastic stresses, σ_H, could be calculated under the assumption of elastic behaviour of the material in the whole load range. For elastic-plastic condition in the most loaded areas, the Neuber principle can be used for approximate calculation of stresses, σ, and total strains, ε_t.

3.4 Elastic-plastic analysis can be used for a more exact determination of the relationship between the stresses, σ, the total strains, ε_t, and the measured data.

3.5 The theory of maximum shear stress (Tresca hypothesis) shall be used for the calculation of stress and strain intensity values. The three values are identified as:

$$\begin{aligned}(\sigma)_{ij} &= \sigma_i - \sigma_j \\ (\sigma)_{jk} &= \sigma_j - \sigma_k \\ (\sigma)_{ik} &= \sigma_i - \sigma_k\end{aligned} \qquad \text{(VII.1)}$$

and

$$\begin{aligned}(\varepsilon_t)_{ij} &= \varepsilon_{t,i} - \varepsilon_{t,j} \\ (\varepsilon_t)_{jk} &= \varepsilon_{t,j} - \varepsilon_{t,k} \\ (\varepsilon_t)_{ik} &= \varepsilon_{t,i} - \varepsilon_{t,k}\end{aligned} \qquad \text{(VII.2)}$$

where i, j, k are indexes of principal stresses or strains.

4. Determination of the stress and strain cycles

4.1 The peaks, up to the present time of unclosed hysteresis loops, shall be given at the beginning of the stress intensity (σ) or the strain intensity (ε_t) patterns, Figure VII.1.

4.2 All hysteresis loops will be closed when stress intensity (σ) or strain intensity (ε_t) patterns (time response) are repeated, it means to duplicate the pattern, see Figure VII.2. For the assessment, cycles situated between the same peaks of maximum absolute values of the stress and the strains should be used (see peaks No 8 and 8z of the example on the Figure VII.2).

4.3 The rain flow method shall be used for determining strain cycles from strain patterns.

4.4 When the material is loaded in an elastic state only, stresses can be obtained from strains multiplied by Young's modulus, E.

4.5 When the component material is loaded in the elastic-plastic state, then stress peaks of the hysteresis loop shall be chosen from the stress pattern for the same time frame as the strain peaks of the given hysteresis loop, which is created by the rain flow method.

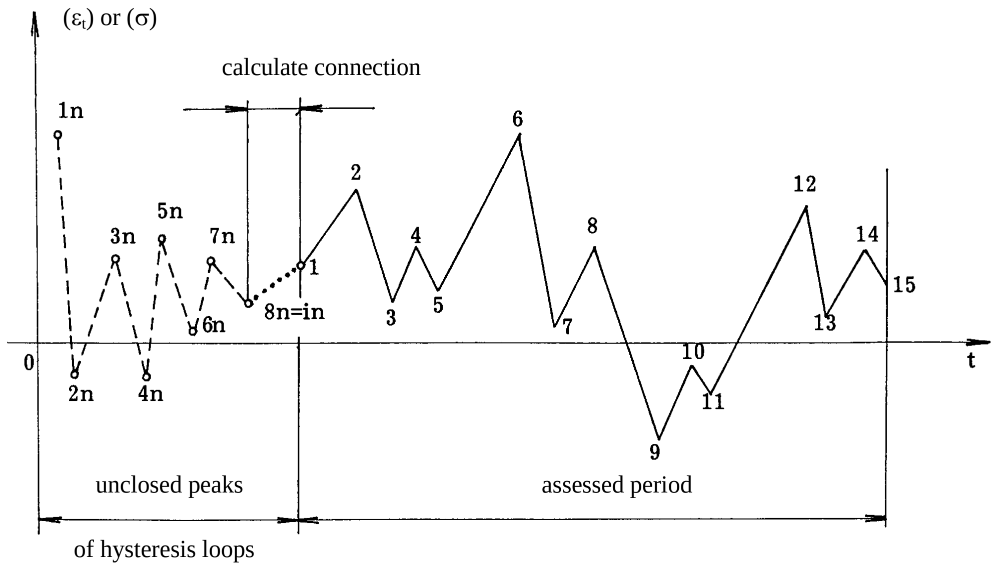

FIG. VII.1. Unclosed peaks put at beginning of strain or stress pattern.

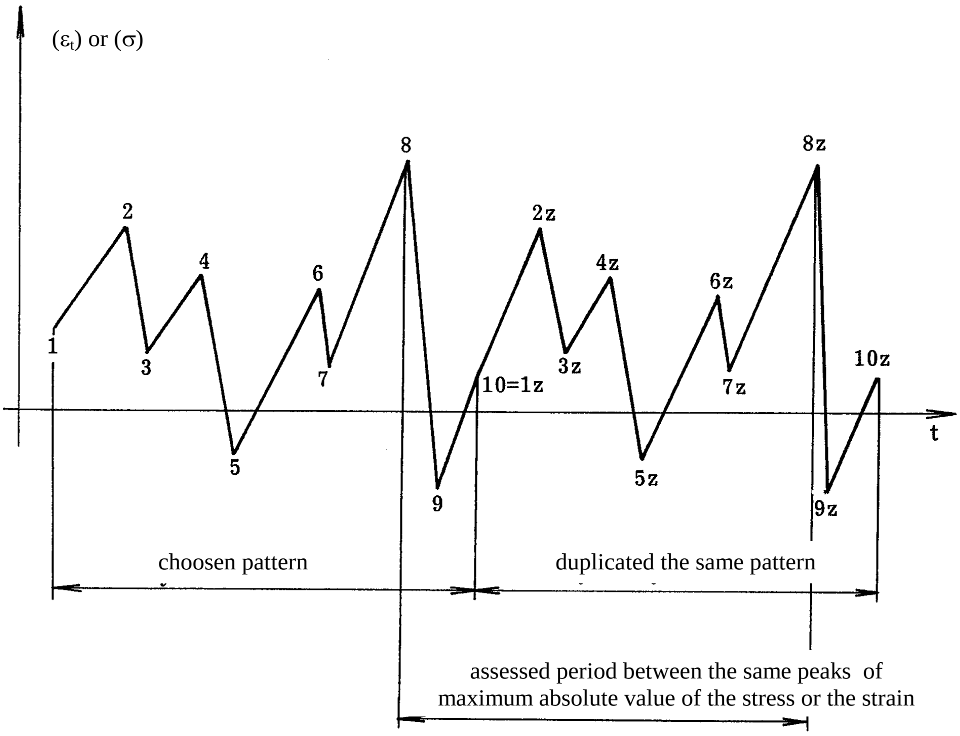

FIG. VII.2. Duplicated strain or stress pattern.

5. Calculation of usage factor

The allowed numbers of cycles $[N_o]$ for a given total strain amplitude should be determined from the design fatigue curve which are based on a fatigue test in air.

5.1 Low cycle fatigue conditions

5.1.1. The allowed number of cycles $[N_o]$ is equal to the lower value $[N_o]$ calculated from two relationships of the design fatigue curves (VII.3). These relationships (VII.3) shall be used for $100/n_N \leq [N_o] \leq 10^8$. If the value of plastic reversible strain, $\varepsilon_{ap,vr}$, is unknown, the relationship (VII.3) can be used up to $[N_o] \leq 10^6$ (and then $\beta_\sigma = \beta_n = 1$).

$$\frac{(\varepsilon_{at})}{\varphi_F \cdot \varphi_w} = \frac{\varepsilon'_f - 0{,}35\left[(\varepsilon_t)_{max} - R_{p0,2}/E\right]}{n_\sigma\,(2[N_o])^{-c}} + \frac{\varepsilon_{ap,vr}}{n_\sigma} + \beta_\sigma\,\beta_n\,\frac{\sigma'_f - (\sigma_m)}{n_\sigma\,E}\cdot\left[\frac{1}{(2[N_o])^{-c}} + \frac{\varepsilon_{ap,vr}}{\varepsilon'_f}\right]^{\frac{b}{c}}\;; \tag{VII.3}$$

$$\frac{(\varepsilon_{at})}{\varphi_F \cdot \varphi_w} = \frac{\varepsilon'_f - 0{,}35\left[(\varepsilon_t)_{max} - R_{p0,2}/E\right]}{(2\,n_N\,[N_o])^{-c}} + \varepsilon_{ap,vr} + \beta_\sigma\,\beta_n\,\frac{\sigma'_f - (\sigma_m)}{E}\cdot\left[\frac{1}{(2\,n_N\,[N_o])^{-c}} + \frac{\varepsilon_{ap,vr}}{\varepsilon'_f}\right]^{\frac{b}{c}}\;.$$

5.1.2. If the values of ε_f', σ_f', b, and c are not determined experimentally, then it is possible to use the values given in the Table VII.1.

5.2 High cycle fatigue conditions

5.2.1. The allowed number of cycles [No] is equal to the lesser of [No]-values calculated from the following two relationships of the design fatigue curves (VII.4). These relationships (VII.4) should be used at $2.10^5 \le [N_o] \le 10^8$. A stable strength can be assumed when $[N_o] > 10^8$ is calculated.

$$[N_o] = N_C \left[\frac{\beta_\sigma \, \beta_n \, \sigma_C}{n_\sigma \, \beta_H \, (\sigma_{a,nom})} \left(1 - (\sigma_{m,nom}) \frac{\sigma_C - \sigma_{AC}}{\sigma_C \cdot \sigma_{AC}} \right) \right]^m$$

$$[N_o] = \frac{N_C}{n_N} \left[\frac{\beta_\sigma \, \beta_n \, \sigma_C}{\beta_H \, (\sigma_{a,nom})} \left(1 - (\sigma_{m,nom}) \frac{\sigma_C - \sigma_{AC}}{\sigma_C \cdot \sigma_{AC}} \right) \right]^m \qquad \text{(VII.4)}$$

5.2.2. If the values of σ_C, σ_{AC}, N_C, and m are not determined experimentally, then these values can be determined as follows:

$\sigma_C = 0{,}4\ R_m$ for $R_m \le 700$ MPa (VII.5)

$\sigma_C = (0{,}54 - 0{,}0002\ R_m)\ R_m$ for $700 \text{ MPa} < R_m \le 1200$ MPa

$$\sigma_{AC} = \frac{\sigma_C}{1 + \psi_\sigma}, \qquad \text{(VII.6)}$$

where:

$\psi_\sigma = 0$ for $R_m \le 500$ MPa

$\psi_\sigma = 0{,}05$ for $R_m \le 700$ MPa

$\psi_\sigma = 0{,}10$ for $R_m \le 1000$ MPa

$\psi_\sigma = 0{,}2$ for $R_m \le 1200$ MPa

The number of cycles, N_C, at the fatigue limit can be taken equal to 10^7 and exponent m equal to 4.

5.3 The safety factor on stress, n_σ, is taken equal to 2 and safety factor on cycles is $n_N = 10$. In the analysis of components or their parts (e.g. thermal diaphragms) loaded only thermally or component parts of limited expansion loaded by thermal and mechanical loads (e.g. anticorrosion cladding), failure of which will not lead to a leak of the fluid through the boundary of the loaded parts, the safety factor on stress is taken as $n_\sigma = 1{,}5$ and on cycles as $n_N = 3$. The safety factors for bolts and stud bolts are taken $n_\sigma = 1{,}5$ and $n_N = 3$.

Table VII.1. Material characteristics of the design fatigue curve

Steel	T [°C]	σ_f' /E [-]	ε_f' [-]	b [-]	c [-]	$\varepsilon_{ap,vr}$ [-]
22K	20	0.0045	1.0414	-0.105	-0.632	-
forging	325	0.0042	1.0079	-0.081	-0.678	-
10GN2MFA, plate	20	0.00347	1.362	-0.057	-0.685	-
thickness ≤ 150 mm	350	0.0033	1.412	-0.048	-0.737	-
08Kh18N10T	20	0.0115	0.3158	-0.216	-0.528	-
N ≤ 2300	350	0.00743	0.327	-0.184	-0.489	-
08Kh18N10T	20	0.00496	0.0930	-0.110	-0.383	-
N ≥ 2300	350	0.0074	0.327	-0.184	-0.489	-

6. Criteria for assessment of material damage by fatigue due to fluctuating load

6.1 An example of the recommended procedure for the assessment of material damage due to fatigue is shown in the flowcharts from Figure VII.3. The following input parameters are assumed for the assessment:

a) design parameters,

b) monitored data during the operating loading.

6.2 Expected material usage factor, D_f, at the end of assumed component lifetime is used for comparison with given criterion of material damage. An assumed component lifetime is considered as the component design service life including its extension above this time.

6.3 The D_f criterion value of the flowchart Figure VII.3 shall be used as follows:

6.3.1. Criterion of $D_{fd} \leq 0.3$

If the D_{fd} value exceeds 0.3 at the end of assumed component operating lifetime, then the (total) usage factor, D_f, can be determined by multiplication of the partial usage factors (previously determined for individual design regimes) by number of actual occurrences of the design regimes during the assessed period, and subsequent summation over the design regimes. This can be implemented provided that no significant deviations of actual time variations of regime parameters (pressure, temperature and flow rate of fluid and their changes) from the design ones were not identified. If some deviations were identified, then their insignificance has to be demonstrated by calculation.

6.3.2. Criterion of $D_{fp} \leq 0.4$

This criterion is similar to $D_{fd} \leq 0.3$, but the partial usage factors D_{fdj} have to be calculated for typical parameter time variations for real operating regimes which are monitored by I&C and the diagnostic systems.

6.3.3. Criterion of $D_{fp} \leq 0.6$, ON-LINE assessment

If both above criteria of $D_{fd} \leq 0.3$ and $D_{fp} \leq 0.4$ cannot be used, then the $D_{fp} \leq 0.6$ criterion can be applied for any area of the component, using a quasi ON-LINE assessment, if the following conditions are met:

a) The input parameters for the assessment were monitored by I&C or diagnostic systems,

b) For the determination of the relationship between stress or strain and ON-LINE monitored parameters, the following procedures can be used:

   (i) Influence functions,

   (ii) Neural network method.

6.3.4. Criterion of $D_{fp} \leq 0.8$, OFF-LINE assessment

If the three criteria discussed above cannot be used, then the detailed OFF-LINE assessment of material damage under actual operating conditions shall be performed, with using the $D_{fp} \leq 0.8$ criterion. Input parameters for the assessment have to be monitored by I&C or diagnostic systems.

If the $D_{fp} \leq 0.8$ criterion is not met, then corrective measures in operation and maintenance management have to be taken with the desired result of decelerating the trend of material fatigue damage. Reconstruction of the component can be one of the tools utilized for this.

6.4 If the usage factor, D_{fp}, calculated for actual operating conditions and predicted for the assumed technical component lifetime does not fulfil the condition:

$$D_{fp} \leq 0.8 \qquad \text{(VII.7)}$$

then the following procedure shall be applied after usage factor reaches the value 0.8:

a) To perform a conclusive non-destructive testing of the mentioned area. If the flaw due to fatigue is identified in a certain area, it shall be schematised and further assessed in accordance with the procedure described in Chapter 8.

b) If a non-destructive test of a certain area is unfeasible or if no flaws were identified, the semielliptical "hypothetical starting crack" shall be defined, parameters of which are as follows:

$$a_{hyp} = 0.1\ s; \quad a/2\ c = 1/6 \qquad \text{(VII.8)}$$

and further assessment in accordance with the procedure described in Chapter 8 shall be performed.

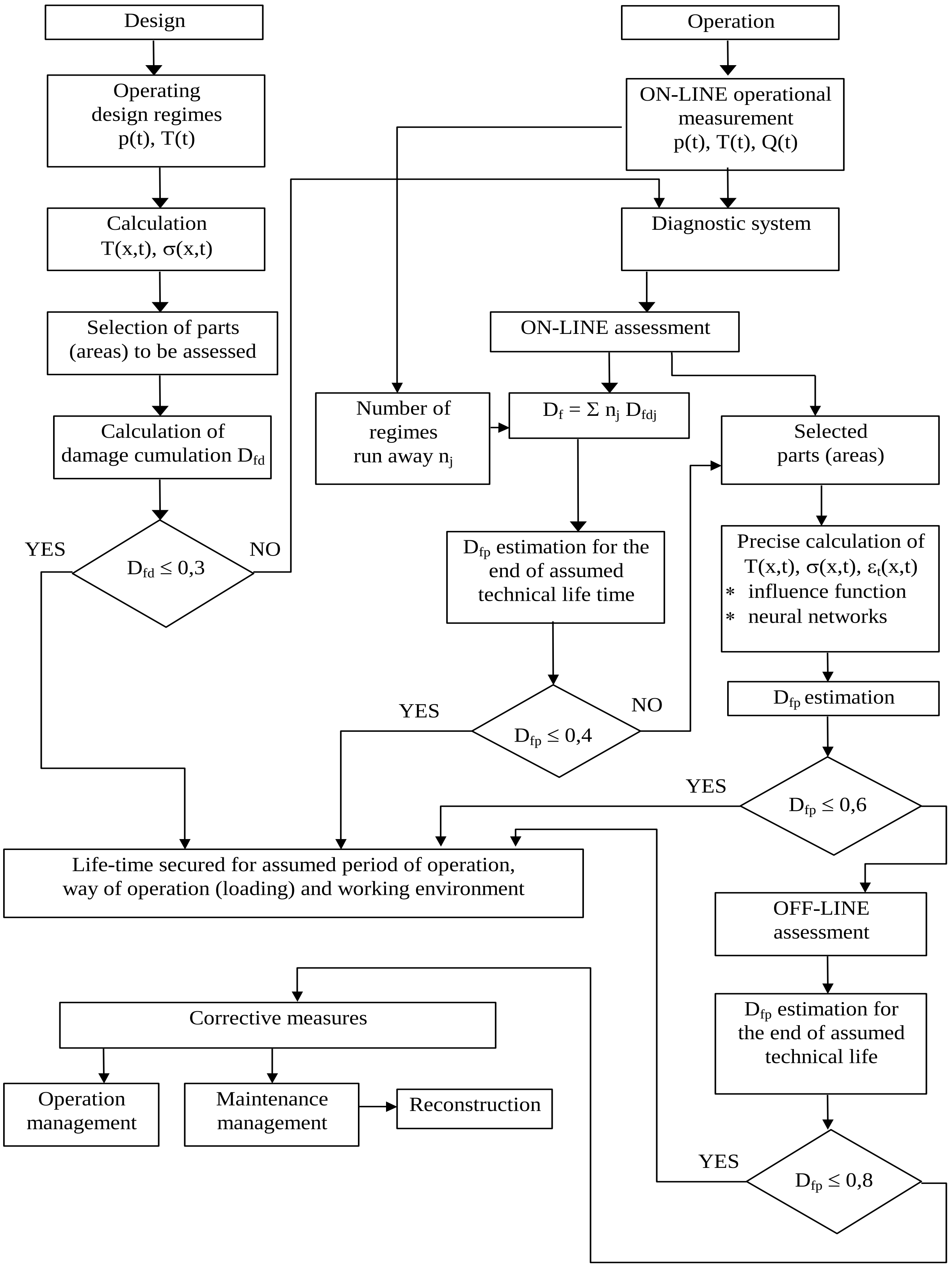

FIG. VII.3. Example of recommended assessment procedure based on component usage factor.

D_{fd} – usage factor for design condition (D_{fdj} – usage factor for the design regime)

D_{fp} – estimation of usage factor at the end of assumed technical lifetime of component

References

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APPENDIX VIII

General recommendations for piping and components temperature measurement

It is known that measured temperature and temperature changes on the outer surfaces of piping are not identical with the temperatures (or temperature changes) of the medium inside piping. The differences are dependent upon:

— Thickness and diameter of piping or piping components;

— Velocity and type of medium;

— Thermo-mechanical properties of the piping or component, quality of insulation used, etc.

In the case of un-insulated piping, the differences are also dependent on the environmental conditions (temperature, conditions of ventilation etc.) There can also be temperature changes in the medium, which are almost impossible to identify on the outer surface and for which the measured response is very different from the medium temperature changes inside. Consequently, in some cases, insufficient information is available for evaluation.

1. It is recommended to use temperature measurement for the assessment of fatigue lifetime in cases where lack of information about temperature distribution in piping and components is supposed. In particular, it is recommended to use temperature measurement in all cases when changes in the medium temperature do not clearly correspond to the pump or valve actions, or where the medium temperature is not measured directly.

2. For any temperature fields calculations, justified measured data should be used.

3. All temperature measurements used for lifetime assessment have to be verified according to Quality Assurance Procedures.

4. Verification of used data has to be done periodically.

5. All measured data used for the lifetime assessment have to be archived during the prescribed lifetime of the piping or component.

6. It is recommended to install measurement devices in those areas where temperature fields are changing quickly, with a high number of repeating, and where temperature changes do not directly depend on standard operation modes.

7. It should be clear for which type of thermal stress cycles the temperature measurement is being performed:

a) Thermal expansion cycles of the piping systems (Temperatures along the whole piping should be known)

b) Fast temperature transients of the medium (Short time steps should be managed for transient description)

c) General temperature distribution on the whole component (a sufficient number of thermocouples should be installed).

d) Special temperature distribution – stratification:

— Several thermocouples along the piping cross-section are recommended with equal vertical distance between thermocouples;

— It has to be proved that stratification (vertical temperature distribution) is identical along the whole horizontal part of piping;

— Measuring of stratification only in the middle of horizontal part of the piping is insufficient.

There are different demands on the number of thermocouples the time step archiving etc., for the types of thermal stress cycles a)–d) mentioned above. These different conditions have to be fulfilled before the measurement installation and the data saving for on-line or off-line evaluation are performed.

APPENDIX IX

Assessment of corrosion-mechanical damage of materials

1. Definitions, symbols and units

a	Depth of a defect (crack), [m]
a_i	Initial depth of a EAC defect (crack), [m]
a_L	Limit depth of a defect (crack), [m]
a_R	Depth of a defect capable of EAC propagation from threshold K, [m]
[a]	Allowable depth of a defect, [m]
Δa	Expected EAC increment of depth of a crack, [m]
da/dN	Crack growth rate in cyclic load, [m/cycle]
da/dt	Crack growth rate in constant load, [m/s]
C	Experimentally determined constant
CF	Corrosion fatigue
EAC	Environmentally assisted cracking
f	Frequency, [Hz]
HOR	Hide-out return;
K_{Iapp}	Applied stress intensity factor, [MPa·m$^{1/2}$]
$K_{I,max}$, $K_{I,min}$	Maximum and minimum values of the stress intensity factor, [MPa·m$^{1/2}$]
K_{ISCC}	Threshold value of stress intensity factor in constant loading, [MPa·m$^{1/2}$]
ΔK_{Iapp}	Applied range of stress intensity factor in a given cycle, [MPa·m$^{1/2}$]
$\Delta K_I = K_{I,max} - K_{I,min}$	Range of the stress intensity factor in a given cycle, [MPa·m$^{1/2}$]
ΔK_{th}	Threshold range of stress intensity factor in a given cycle, [MPa·m$^{1/2}$]
M	Exponent - material characteristics
pH_T	pH calculated at working temperature of the coolant
S	Wall thickness, [m]
SCC	Stress corrosion cracking
SG	Steam generator
β	Biaxiality factor, [$\beta = T^*(\pi.a)^{1/2}/K_I$],
σ	Stress, [MPa]
σ_i	Initiation stress, [MPa]
ε	Strain
ε_i	Strain for a crack initiation
t	Total time of EAC defect development equals: $t_P+t_i+t_R$, [hour]
t_B	Period of incubation of EAC defect, [hour]

t_i	Period until initiation of EAC defect of depth a_i, [hour]
t_P	Period of precursor of EAC defect, [hour]
t_R	Period of propagation of EAC defect, [hour]
t_{TL}	The assumed period of technological lifetime of the component, [hour]
Δt	Time expressing the safety allowance, [hour]
T	T stress, [MPa]
v_c	Crack growth rate, [m/s]
Y	Shape factor
WC	Water chemistry

2. Conditions for corrosion-mechanical damage

2.1 A general term used for the degradation mechanism is the Environmentally Assisted Cracking (EAC). Moreover, it is called the Stress Corrosion Cracking (SCC) under pure influence of a constant load and Corrosion Fatigue (CF) if only cyclic loading.

2.2 The corrosion-mechanical damage comes from simultaneous interaction of three requisite factors: the corroding medium, the material and the stress (Figure IX.1).

2.2.1. The material is defined by characteristic properties of the component from production and heat treatment processes, i.e. chemical composition, microstructure, mechanical and fracture properties, surface quality and others.

2.2.2. The stress means residual stress in the component from production processes (forming, heat treatment, welding, cold work, surface finish processes) and operation stress from temperature dilatations (low cycle fatigue), coolant flow (vibrations) and others.

2.2.3. The environment is characterised by temperature, pressure of the water coolant, the water composition and pH_T and other specifications.

2.3 The detection of EAC sensitivity is performed according test standards [1]. EAC sensitivity of a component depends on a real combination of the three requisite factors specific for each component and operation conditions. Two components made of one material could differ in the sensitivity to EAC.

2.4 EAC cracks form at the component surface and the surface quality primarily limit lifetime of the component. It is recommended to check and to document the quality of the component surface before operation. The component microstructure close to the surface in contact with water coolant during operation could have higher sensitivity to EAC than bulk material and high local residual stresses.

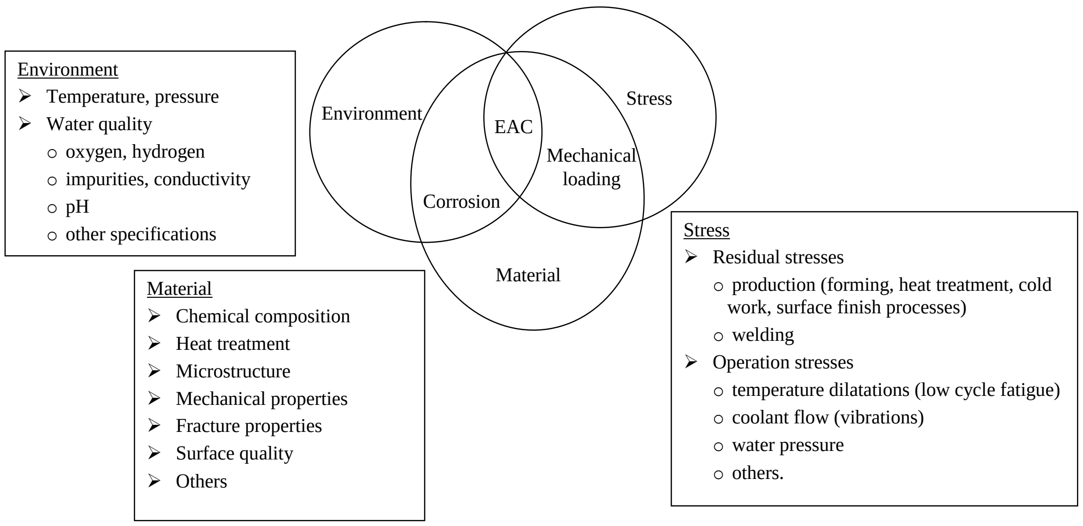


FIG. IX.1. EAC, the corrosion - mechanical degradation process, general scheme.

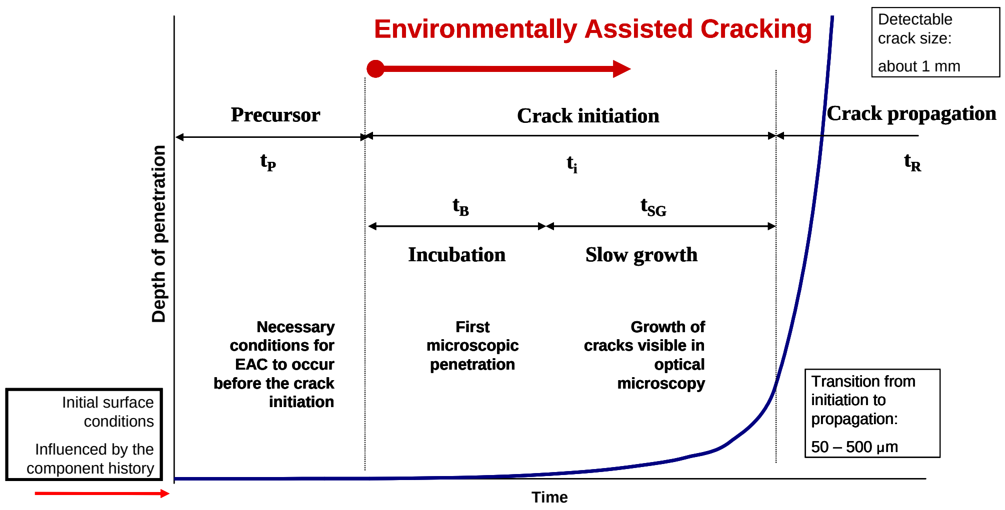


FIG. IX.2. Evolution of EAC defect.

2.5 For lifetime assessment the EAC process duration is determined as sum of periods of precursor, t_P, initiation, t_i, and propagation, t_R, (Figure IX.2).

2.5.1. After a first contact of the component with the environment the conditions for EAC to occur are developed on the component surface. The period t_P is time of the precursor.

2.5.2. The crack initiation starts after the precursor period. The period t_i is time to initiation of the EAC defect capable to propagate (the corresponding defect size is a_i). First penetration into the component surface occurs during incubation period, t_B. After the incubation the micro-crack growth begins. The crack length/depth of penetration into surface has usually about 50-500 μm after the initiation period.

2.5.3. The propagating defect size, a_i, can be shorter than a detectable limit of present NDE techniques.

2.5.4. The period t_R is the time of the EAC defect propagation.

2.6 In general, any period during which the component is exposed to the coolant medium is important for the evaluation of an environmental effect.

2.6.1. Crucial factors are the period duration and the acting operation conditions.

2.6.2. The period of reactor unit operation should be divided as follows:

- Start-up of the unit until reaching the reactor's nominal power, here abnormal straining and water chemistry regimes are expected;
- Operation at the reactor's full nominal power, here normal operation straining and water chemistry regimes are expected;
- Transient modes with starts and returns to the nominal power of the reactor and cold shutdowns of the unit, here straining and water chemistry regimes excursions are expected;
- Cold technological outage of the unit, here abnormal straining and water chemistry regimes are expected.

2.7 The following operation regimes are the most important for the component lifetime assessment in cases of corrosion-mechanical damage:

2.7.1. The nominal operation regime due to the longest duration

2.7.2. Over limits and conditions periods during which the content of chemical admixtures exceeds allowable limits specified in Operation limits and conditions,

2.7.3. For a steam generator (SG) EAC defect could be formed on tubes:

- If the secondary circuit coolant medium chemistry is over "Limits and conditions";
- In normal operation in local areas of changed water chemistry (i.e. in shielded volumes) with regard to non-homogenous distribution of impurities, deposits layer and filled crevices between tubes and collector or support system, deposits layer on the stud bolt tread holes in the primary collector flanges, etc.

3. Conditions for the initiation of a defect

3.1 Preconditions of the EAC process initiation are sustained dynamic straining of component surface and corrosion reaction / adsorption processes on the surface. Both are needed to further growth the initiated micro defects, too.

3.1.1. Conditions for EAC initiation: $\sigma > \sigma_i$ and $\varepsilon > \varepsilon_i$ and $d\varepsilon/dt > 0$, where σ_i, ε_i are characteristic of the material / environment interaction. Positive strain rate ranking from 10^{-9} to 10^{-6} s^{-1} for stationary stresses, or from 10^{-6} to 10^{-3} s^{-1} for cyclic stresses.

3.2 Period ($t_P + t_i$) until initiation of a defect of depth a_i capable to propagate depends on many factors and it is hard practically to predict and measure for most of materials from NPPs.

3.3 In the document the initiation time t_i is defined as the time to formation of the defect a_i, where a_i = min {10% of s; 0.5 mm}.

3.4 If an over limits and conditions period exceeds the period ($t_P + t_i$), one or more defects of a_i are likely present in a component of the coolant system.

3.5 For a steam generator:

3.5.1 Time associated with an accumulation of species on superheat surfaces is the t_P precursor period.

3.5.2 A probability of the a_i defect initiation depends on chemistry of the shielded volumes. The chemistry can be determined from Hide-out Returns (HOR) during SG shut downs or from sampling of the crevices.

3.5.3 If pH_T in the filled crevices indicates the acid condition, the probability of the a_i defect occurrence is very high and the period (t_P + t_i) can be measured in laboratory tests using the predicted acid solution.

3.5.4 If pH_T in the filled crevices indicates the neutral or alkaline conditions, the probability of the a_i defect initiation is very low and the initiation time is very long (more than 10 years).

4. Conditions for the EAC defect propagation

4.1 If suitable conditions for EAC are persisting on the component surface then EAC process continues to develop.

4.2 The depth of EAC crack capable to propagate, a_R, can be determined from threshold K as follows:

4.2.1. In constant load (SCC) the applied stress intensity factor is higher than the threshold

$$K_{Iapp}\ (a_R) \geq K_{ISCC} \qquad \text{(IX.1)}$$

The threshold stress K_{ISCC} is to determine experimentally for the corrosion system or optionally K_{ISCC} = 5 $MPam^{1/2}$; the stress intensity factor K_{Iapp} for the given defect and for thermal-mechanical loading to be determined in accordance with Appendix IV or by a procedure based on the finite elements method.

4.2.2. The defect with the depth, a_R, propagates due to cyclic (CF) or changing loading if:

$$\Delta K_{app}\ (a_R) \geq \Delta K_{th} \qquad \text{(IX.2)}$$

The threshold stress ΔK_{th} is to determine experimentally for the corrosion system or optionally ΔK_{th} = 4 MPa $m^{1/2}$.

4.2.3. Actual loads and their time changes monitored by I&C and MDS systems are used for the stress calculation. If such data are not available, the loads designated within the Project Specifications may be used.

4.3 Pure mechanical loading (fatigue) or only corrosion damage (pitting corrosion, intercrystalline attack, etc.) can also generate the defect of the depth a_R.

4.4 The propagation rate, v_c, of the EAC defect shall be calculated from empirical equations according type of loading with parameters defined in experiments.

4.5 For a constant load (an empirical disposition line):

$$v_c = C(K_I)^m = F(K_I, \beta, WC)\ [m/s;\ MPa.m^{1/2}] \qquad \text{(IX.3)}$$

4.5.1. Especially for ferritic low alloy steel 15Kh2MFA in nominal water of primary circuit WWER 440 (oxygen < 20 ppb), 290°C, β = 0.55, small scale yielding, constant load including partial unloading [2]:

$$v_c = 2.20*10^{-11} \qquad \text{for } K_{Iapp} < 27$$
$$v_c = 3.29*10^{-17} * K^{4.07} \qquad \text{for } K_{Iapp} \geq 27 [m/s;\ MPam^{1/2}] \qquad \text{(IX.4)}$$

4.5.2. Especially for austenitic steel 08Kh18N10T of reactor vessel internals of fluence lower than 1 dpa in nominal water of primary circuit (oxygen < 20 ppb), 320°C, β = 0.55, small scale yielding, constant load including partial unloading [3]:

$$v_c = 2.1 * 10^{xx} (K_I)^{2.161} \text{ [m/s; MPam}^{1/2}] \quad \text{(IX.5)}$$

4.5.3. Especially for austenitic steel 08Kh18N10T of SG tubes in the concentrated acid, neutral and caustic solutions of the crevices, 275°C, β = -0.4 to 0.2 [4]:

$$v_c = 5*10^{-8} \text{ [m/s]} \quad \text{(IX.6)}$$

4.6 For a cyclic load:

$$v_c = \frac{da}{dN} f = f \,.\, C(\Delta K_I)^m = F(\Delta K_I, K_{I,min}/K_{I,max}, f, WC) \quad [\text{MPa.m}^{1/2}; \text{m/s}] \quad \text{(IX.7)}$$

4.6.1. The constant, C, and the exponent, m, of ferritic and austenitic steels in water of primary circuit are determined in accordance with Appendix XII.

4.7 The values of stress intensity factor for different depths of the defect, a, shall be calculated by the finite elements method or by using the procedures described in Appendix IV.

5. Resistance against corrosion-mechanical damage

5.1 Two criteria for assessment of corrosion-mechanical damage resistance of the components are applied – the criterion of the defect propagation time and the criterion of the defect size.

5.2 Limit defect, a_L, ensuring safety and expected lifetime shall be determined for every component.

5.2.1. In option in components where an EAC defect is allowed it is recommended to apply for ferritic steels as upper limit of the defect size a_L = min {[a]; s}; for the component parts made of austenitic steels the growth of the defects can be limited by reaching the component wall surface, i.e. a_L = s.

5.2.2. In other cases, here is recommended to apply the limit defect size a_L = 1 mm.

5.3 The defect total time of EAC defect development is calculated from the relationship:

$$t = t_P + t_i + t_R \quad \text{(IX.8)}$$

5.4 The material component resistance is ensured if

$$t > t_{TL} + \Delta t \quad \text{(IX.9)}$$

where the recommended value for the allowance, Δt, is 87600 hours, i.e. 10 years;

and in other case, if the precursor and the initiation time are not known, when:

$$t_R > t_{TL} \quad \text{(IX.10)}$$

5.5 The propagation time of the defect, t_R, is calculated from known crack growth rate of the material as

$$t_R = \frac{a_L}{v_c} \quad \text{(IX.11)}$$

5.6 For the variable rate of the defect propagation, the period, t_R, can be determined as follows:

$$t_R = \int_{a_i}^{a_L} \frac{da}{v_c} \tag{IX.12}$$

5.7 For walls containing several different types of steels, the growth period, t_{Ri}, is to be calculated for each part. Then the total growth period of the defect through the wall is given by the sum:

$$t_R = \sum_{i=1}^{k} t_{Ri} \tag{IX.13}$$

where k represents the number of different layers of the wall, through which the defect can grow.

The defect size criterion is based on assessment of time needed to develop EAC defect of limit size, a_L. The time equals to $t = t_P + t_i + t_R$, where

$$t_R = \frac{a_L - a_i}{v_c} \tag{IX.14}$$

5.8 If during inspection the component had been identified a defect of the depth a, shorter than a_L, then it is recommended to perform:

5.8.1 to determine maximum allowable Δa, optionally $\Delta a = a_L - a$;

5.8.2 to calculate time of propagation of the defect using the material characteristic v_c according to

$$t_R = \frac{\Delta a}{v_c} \tag{IX.15}$$

5.8.3 if t_R is longer than two time periods of regular inspections of the component then to elaborate of a proposal for the component future operation.

5.9 The limited operation with the component containing the defect can be allowed if the calculated time t_R is longer than two time periods between regular inspections of the component and if a special procedure take into account the defect development and inspections has been approved.

5.10 The corrective measures that are to be accepted are in the field of operation and maintenance management. Reconstruction of the components and/or the change in chemical modes can be tools for these measures.

References

[1] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, *International standard: Corrosion of metal alloys - Stress corrosion testing*, ISO 7539, parts 1-9, ISO, Geneva (2012).

[2] Ernestová, M., Hojná, A., *Over Limit Oxygen Effect on Crack Propagation of WWER 440 RPV Steel in Primary Circuit Water, Proceedings of PVP2009*, ASME Pressure Vessels and Piping Division Conference, Prague (2009).

[3] U.S. NUCLEAR REGULATORY COMMISSION, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping* (ASME Code Class 1, 2 and 3 for pressure boundary piping and safe ends), NUREG-0313 (rev. 1), Washington, DC (1980).

[4] Brožová, A., Burda, J., Papp, L., Pochman, K., *Nature of Tube Degradation in Horizontal Steam Generators, Proceedings from International Symposium Fontevraud 5*, Fontevraud (2002).

APPENDIX X

Schematisation of flaws

List of symbols

a depth (minor semi-axis) of a schematized flaw, [m]

c half length (major semi-axis) of a schematized flaw, [m]

l length of flaw, [m]

v depth (TWE -through wall extent) of flaw, [m]
(final TWE value consisting of measured depth and depth tolerance obtained within NDE qualification)

b ligament, [m]

s wall thickness, [m]

F area of flaw, [m^2]

1. General principles

1.1 The basic principle valid for all computational models using data from non-destructive tests and for schematisation of flaws is the following:

For the minimum amount of available input data, a computational model has to be conservative from the point of view of assessment of resistance against a failure. In this case, the supplementary data on flaws detection and sizing from a non-destructive test would be supplied. Such information taking into account results of appropriate NDE qualification has to cause a decrease of conservatism of the computational model applied.

1.2 All technological types of manufacturing and service induced flaws (cracks, pores, inclusions, lacks of fusion etc.) are conservatively assumed, for needs of schematisation, as elliptical cracks with semi-axes *a* and *c*. From all possible choices, the most unsafe orientation is chosen from the point of view of assessment by the methods of fracture mechanics.

1.3 The protocol on the executed in-service inspection serves as a basis for characterisation of flaws. This has to contain the following necessary data:

— Information on the NDT method used for the inspection including references to the technical standards and approved inspection procedures;
— Information on the completed NDT qualification of the inspection procedures, NDT equipment applied, and additional (usually blind tests) qualification of the personnel (including references on NDT qualification certificate issued, NDE qualification criteria, applied NDT qualification methodology, and NDT qualification dosier issued if applicable);
— Characteristics of the testing equipment;
— A list of NDT inspectors from the NDT vendor company who performed the in-service inspection (without the auxiliary staff), including the numbers of their certificates and the NDT levels according to the national qualification scheme for the testing method applied;
— The method of calibration of the equipment and results achieved;
— The adjusted sensitivity of the equipment during the inspection;

- Description of the inspection area (where the inspection was executed) and the appropriate component;
- Description of flaws found:
- Detailed information on flaw detection;
- Coordinates of their location;
- Their orientation (tilt, skew);
- Assumed type (surface or embedded flaw, see for details Section 1.4);
- Assumed origin, damage mechanism, and topography;
- Sizing data (determined dimensions: length, height/TWE (through wall extent) or depth in terminology of calculations, ligament – distance to the nearest component surface).

1.4 For schematisation, embedded and surface flaws have to be distinguished. As a criterion to differentiate the mentioned types, the distance, *b,* of the nearest flaw point to the nearest wall surface of the component is to be used.

a) If the ligament – the shortest distance of the flaw contour to a free component surface fulfils the relationship

$$b < 0.4a \tag{X.1}$$

then the flaw is schematised and assessed as a surface flaw.

b) If the flaw ligament meets the condition

$$b \geq 0.4a\ , \tag{X.2}$$

then the flaw is taken as the embedded one.

1.5 The laminar flaws are schematised in accordance with the determined dimensions/lengths in the single directions of inspection. Laminar flaws are plane flaws, located in the plane parallel to the outer surface of the component.

2. Schematisation of embedded flaws

Schematisation can deal with single flaws or with a group of flaws.

2.1 Schematisation of a single embedded flaw

Schematisation of embedded flaws is executed taking into account the input data on the flaws´ size and location obtained in the frame of the executed in-service inspection.

The flaw is characterised by its maximum size in the direction perpendicular to the surface *v* and by its maximum length, *l.* In this case, the following relationships are to be used for calculation of semi-axes of the crack (see Figure X.1):

$$a = \frac{v}{2}, c = \frac{l}{2} \tag{X.3}$$

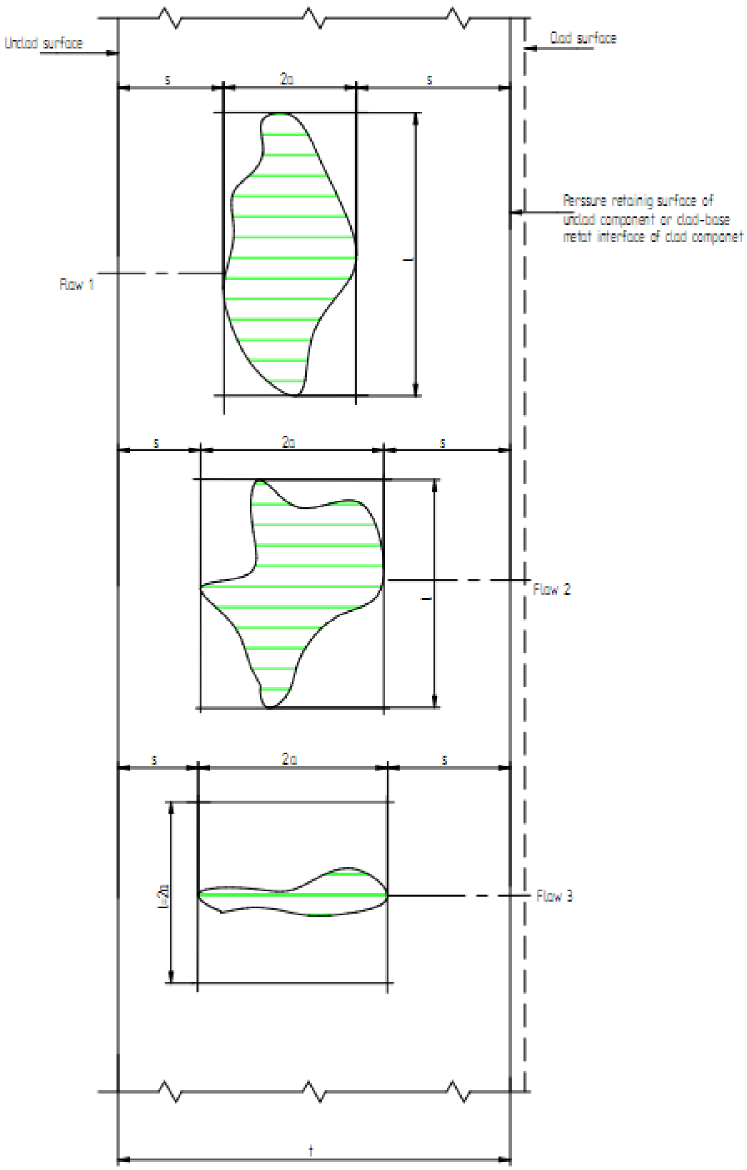


FIG. X.1 Embedded (subsurface) planar flaws oriented in plane normal to pressure retaining surface – Illustrative configurations and determination of dimensions 2a and l where s ≥ 0,4 a and where s = b.

2.2 Schematisation of a group of embedded flaws

Schematization of a group of embedded planar flaws oriented in plane normal to pressure retaining surface is performed according to Figure X.2, see flaw examples # 2 - # 4.

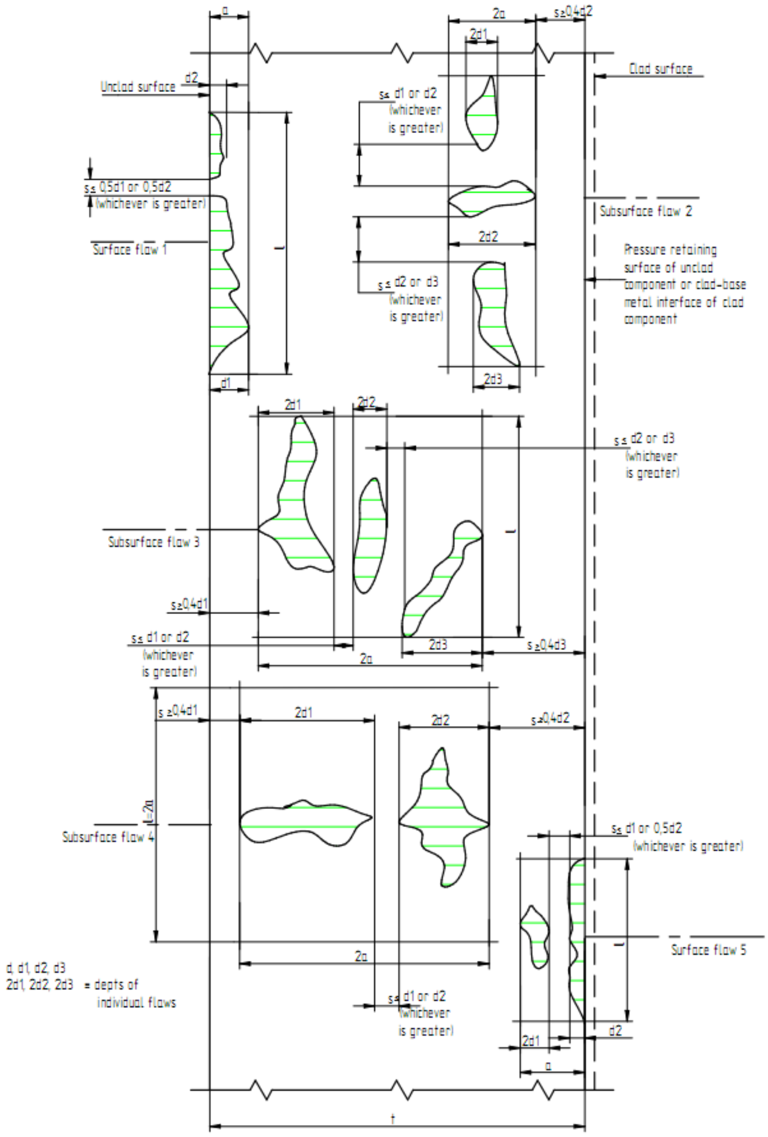


FIG. X.2 Embedded (subsurface) planar flaws oriented in plane normal to pressure retaining surface – Illustrative configurations and determination of dimensions 2a and l where s ≥ 0,4 a (see examples #2 - #4) and where s = b.

3. Schematisation of surface flaws

Schematisation deals either with a single flaw or with a group of flaws.

3.1 Schematisation of a single surface flaw

Schematisation of surface flaws is to be executed taking into account the input information on their size as obtained in the frame of the executed in-service inspection.

The surface flaw is to be schematised as the semi-elliptical crack with semi-axes

$$a = \delta \quad c = l/2 \tag{X.4}$$

The flaw is characterised by the length, *l,* in the direction to the free surface by the ligament, *b,* and its height (depth), *v.* To calculate the semi-axes of the substituting crack, the following relationships are to be used (see Figure 3):

$$a = b + v \qquad c = l/2 \tag{X.5}$$

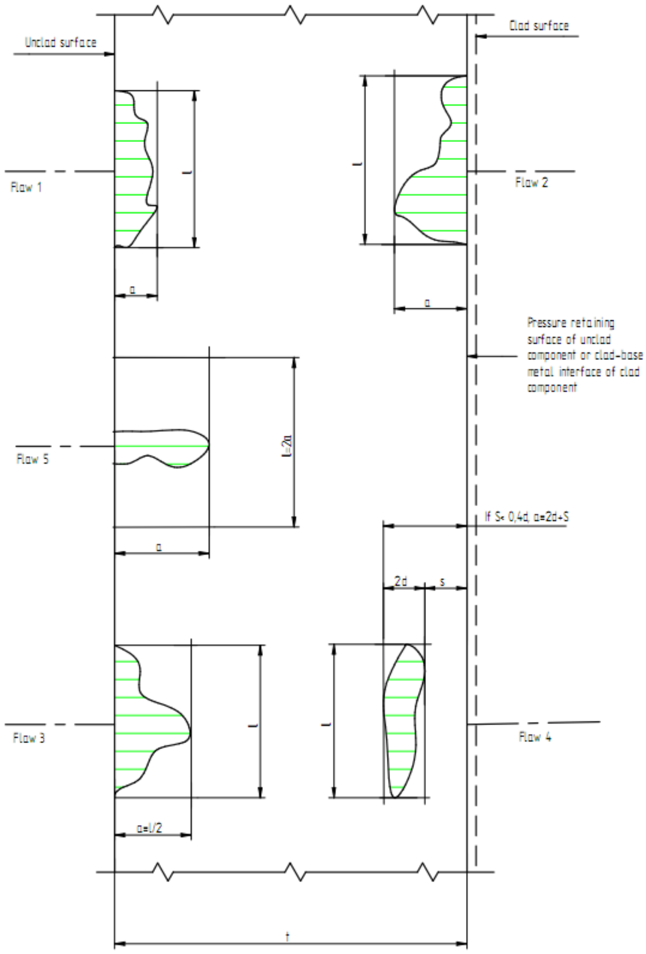


FIG. X.3 Surface planar flaws oriented in plane normal to pressure retaining surface – Illustrative configurations and determination of dimensions 2a and l where s < *0,4 a (see examples # 1 - # 5) and where s = b.*

3.2 Schematisation of a group of surface flaws

Schematization of a group of embedded planar flaws oriented in plane normal to pressure retaining surface is performed according to Figure.4, see flaw examples # 1 and # 5.

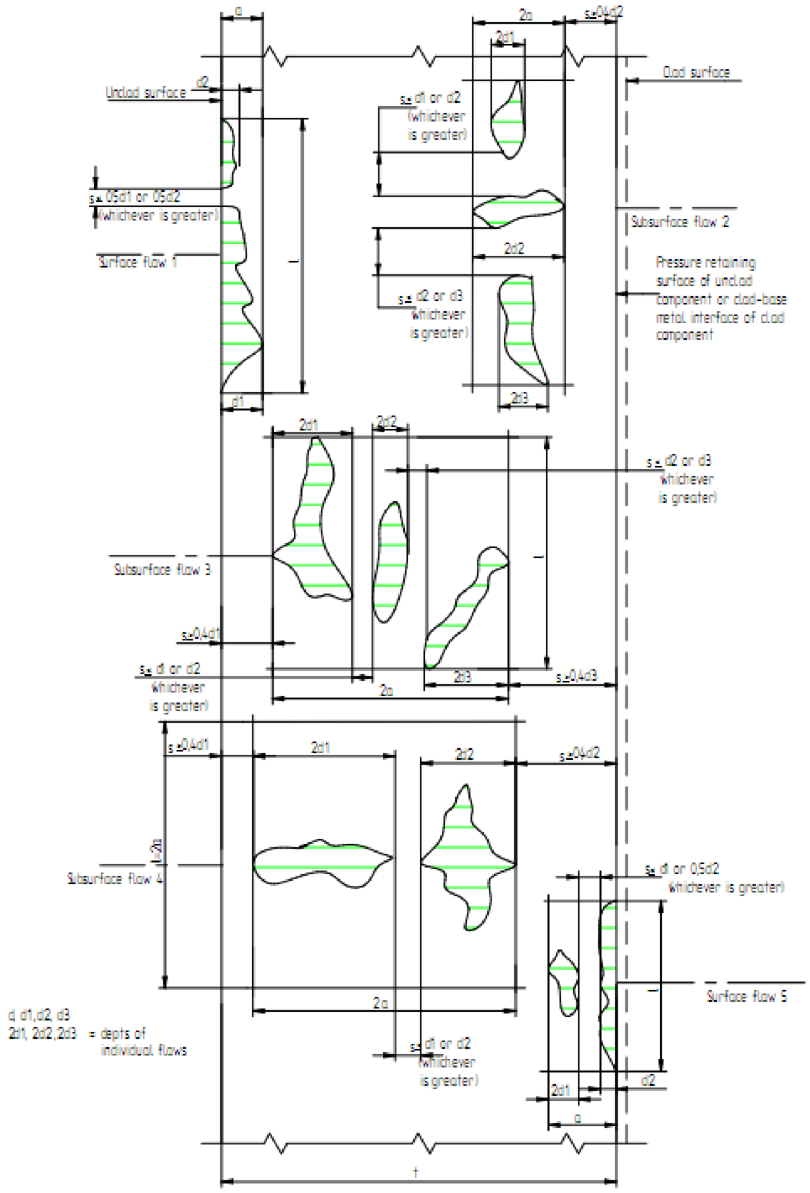


FIG. X.4 Surface planar flaws # 1 - # 5 oriented in plane normal to pressure retaining surface – Illustrative configurations and determination of dimensions 2a and l where s ≥ 0,4 a and where s =b.

APPENDIX XI

Tables of allowable sizes of indications found during in-service inspections

List of symbols

a depth (minor semi-axis) of a schematized flaw perpendicular to surface, [mm]

c half length (major semi-axis) of a schematized flaw perpendicular to surface, [mm]

l length of flaw, [mm]

F area of flaw, [mm^2]

[a] allowable depth (minor semi-axis) of a schematized flaw, [mm]

[c] allowable half length (major semi-axis) of a schematized flaw, [mm]

[l] allowable length of flaw, [mm]

[F] allowable area of flaw, [mm^2]

b ligament, [mm]

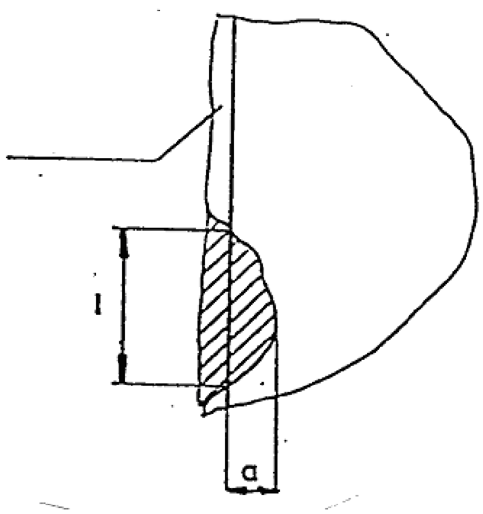
s wall thickness, [mm]

1. General rules

1.1 In this Appendix, allowable sizes of schematised flaws of linear, laminar, and planar types are given with respect to tested places of individual components.

1.2 In the case that schematised sizes of the indication are smaller than given in the tables, the indication is allowable and no further analysis is necessary.

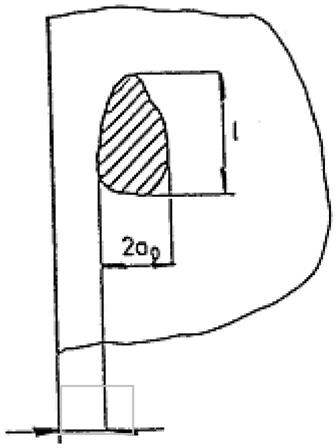
1.3 Generally, the following rules apply in the process of indication evaluation:



1.3.1. Part of the indication located in the austenitic cladding is not included in the flaw size

1.3.2. If the ligament, *b,* between the flaw and outer surface is smaller than $b < 0.4a$ then the flaw is taken as a surface one and

$$a = 2a_0 + b$$



2. Reactor pressure vessels

Indications found during in-service inspections in the RPV of WWER-440 are allowable if their schematised dimensions are smaller than values given in tables XI.1, XI.2 or XI.3.

TABLE XI.1 Allowable sizes of embedded flaws in RPV of WWER-440

RPV part	Cylindrical part	Other parts
[2a], [mm]	[2c], [mm]	[2c], [mm]
4	14.5	-
5	9.5	-
6	8.5	-
7	8	-
8	8	66
9	-	33
10	-	25
11	-	22
12	-	20
14	-	19
16	-	18
18	-	18

TABLE XI.2 Allowable sizes of surface flaws on outer surface in RPVs of WWER-440 or WWER-1000

[a], [mm]	[2c], [mm]
2.5	30
3.0	16
3.5	13
4.0	12
5.0	12
6.0	12

TABLE XI.3 Allowable sizes of laminar flaws in RPVs of WWER-440 and WWER-1000.

s, [mm]	[F], [mm2]
60	4500
100	7500
150	11250
200	15000
250	18750
300	22500
350	26250
≧400	30000

2.1.1. For intermediate thickness, *s*, linear interpolation of the area is permissible.

2.1.2. Laminar flaws in vessel nozzles should be evaluated as planar embedded flaws.

2.1.3. The flaw sizes shown in table XI.3 are independent of the reference/transition temperature values.

2.2 Indications found during in-service inspections in the RPV of WWER-1000 are allowable if their schematised dimensions are smaller than values given in tables XI.2, XI.3 or XI.4.

TABLE XI.4 The allowable sizes of embedded flaws in RPV of WWER-1000

RPV part	Cylindrical part	Other parts
[2a], [mm]	[2c], [mm]	[2c], [mm]
3	55	-
4	10	-
5	9	-
6	8	-
7	8	-
8	8	-
10	-	140
12	-	39
14	-	29
16	-	25
18	-	24
20	-	24
22	-	24
24	-	24

2.3 Table values (Tables XI.1. and XI.2.) are valid for the reactor pressure vessels of WWER-440 and WWER-1000 types only in cases when critical temperature of brittleness, T_k, or reference temperature, T_0, of the materials of the RPV (on boundary between austenitic cladding and base/weld materials) is not larger than + 100 °C. In the case that this temperature is higher, supplementary calculations should be performed using this Procedure.

2.4 Indications found during in-service inspections in the RPVs of WWER-440 are allowable for cases when reference temperature, T_0, and/or critical temperature of brittleness is higher than + 100 °C, but not larger than + 150 °C, if their schematised dimensions are smaller than values given in tables XI.2, XI.5 or XI.6.

TABLE XI.5 The allowable sizes of embedded flaws in RPV of WWER-440

RPV part	Cylindrical part	Other parts
[2a], [mm]	[2c], [mm]	[2c], [mm]
4	12.5	-
5	8	-
6	7	-
7	7	-
8	-	66
9	-	33
10	-	25
11	-	22
12	-	20
14	-	19
16	-	18
18	-	18

TABLE XI.6 The allowable sizes of surface flaws on the outer surface in RPVs of WWER-440 or WWER-1000

[a], [mm]	[2c], [mm]
2.5	26
3.0	14
3.5	11
4.0	10
5.0	10
6.0	10

2.5 In cases where the schematised flaws are larger than those given in tables XI.1, XI.2, XI.4, XI.5, or XI.6, supplementary calculations should be performed using this Procedure.

3. Steam generator and pressurizer

3.1 Indications found during the in-service inspections in the steam generator and the pressurizer of WWER-440 are allowable if their schematised flaws are smaller than the values given in tables XI.7 through XI.16.

TABLE XI.7 Cylindrical part of steam generator – wall thickness 135 mm

Allowable linear flaw sizes

Surface flaw length [2c] = [l] [mm]	**Embedded flaw** Length [2c] = [l] [mm]
14.0	20.5

Allowable laminar flaws

Flaw area [F] [mm^2]
10306

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.7	54.0	3.0	59.4
0.10	3.0	28.7	3.4	33.8
0.15	3.4	22.5	3.9	26.1
0.20	3.8	18.9	4.5	22.3
0.25	4.5	17.8	5.1	20.5
0.30	5.1	17.1	5.9	19.8
0.35	5.9	17.0	6.9	19.7
0.40	6.7	16.9	7.8	19.5
0.45	6.9	15.3	9.1	20.1
0.50	7.0	14.0	10.3	20.5

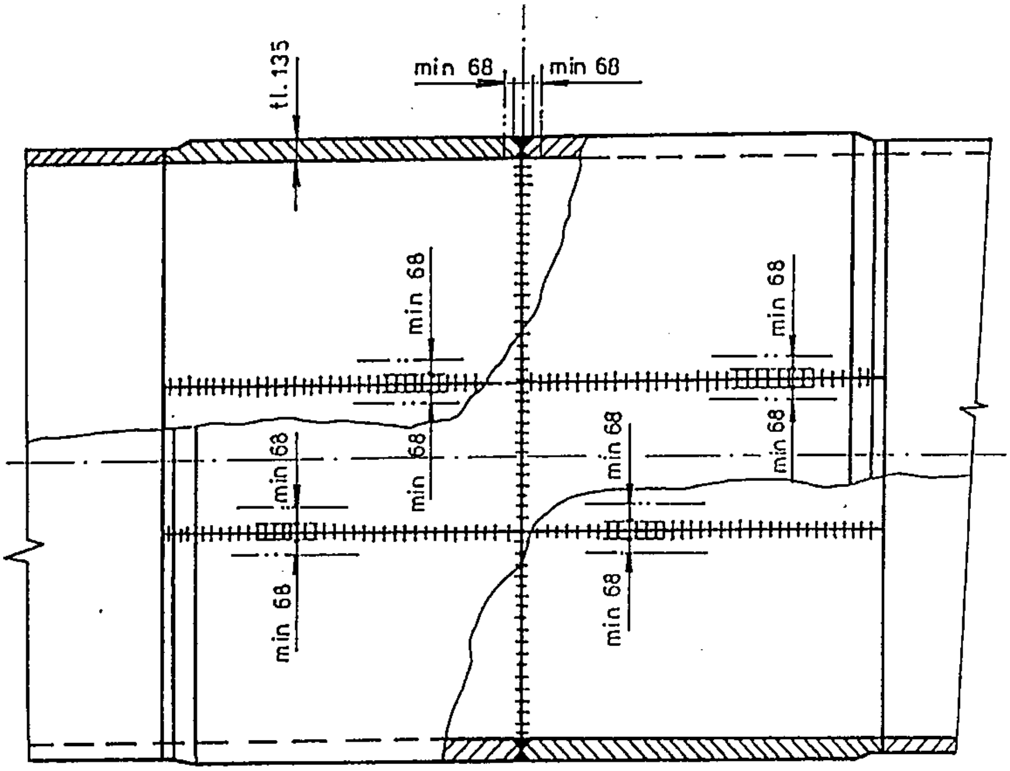


FIG. XI.1 Cylindrical part of steam generator – wall thickness 135 mm.

TABLE XI.8 Cylindrical part of steam generator – wall thickness 75 mm

Allowable linear flaw sizes

Surface flaw Length [2c] = [l] [mm]	**Embedded flaw** length [2c] = [l] [mm]
11.1	17.7

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
5715

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.2	43.6	2.4	48.0
0.10	2.4	23.9	2.8	28.3
0.15	2.7	18.1	3.2	21.5
0.20	3.1	15.5	3.7	18.7
0.25	3.6	14.5	4.3	17.2
0.30	4.2	14.0	5.1	16.9
0.35	4.9	13.9	5.9	16.8
0.40	5.5	13.7	6.8	17.0
0.45	5.6	12.4	8.0	17.7
0.50	5.7	11.5	9.2	18.4

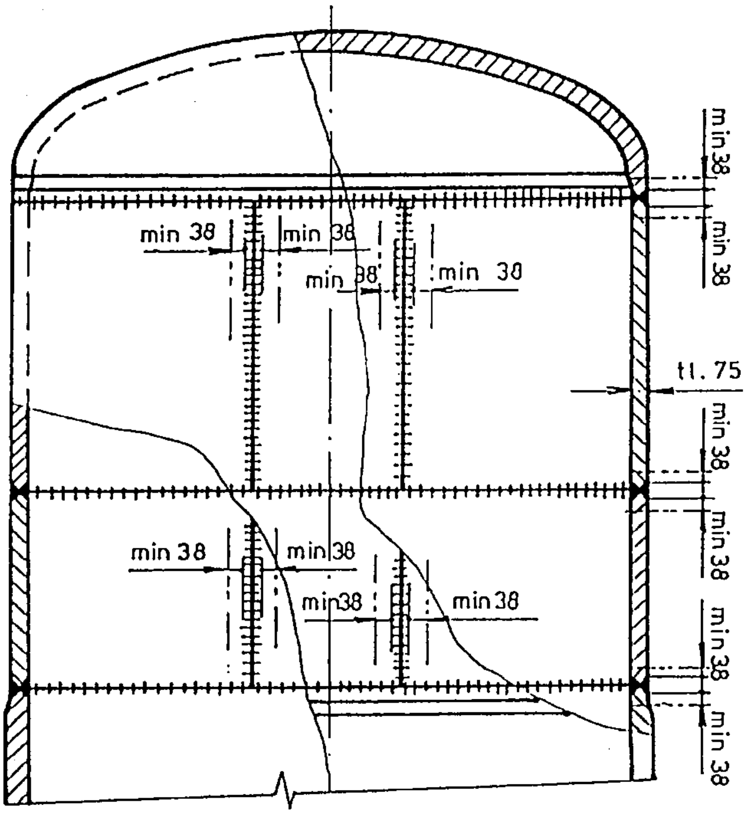


FIG. XI.2 Cylindrical part of steam generator – wall thickness 75 mm.

TABLE XI.9 Steam generator cover - wall thickness 84 mm – central part

Allowable linear flaw sizes

Surface flaw length [2c] = [l] [mm]	**Embedded flaw** Length [2c] = [l] [mm]
11.1	17.3

Allowable laminar flaw size

Flaw area [F] [mm^2]
6540

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.2	43.6	2.5	49.4
0.10	2.4	23.8	2.8	28.0
0.15	2.7	18.1	3.2	21.4
0.20	3.1	15.5	3.7	18.5
0.25	3.6	14.5	4.3	17.1
0.30	4.2	14.0	5.0	16.7
0.35	4.9	13.9	5.8	16.6
0.40	5.5	13.7	6.7	16.7
0.45	5.6	12.4	7.8	17.3
0.50	5.7	11.5	9.0	18.0

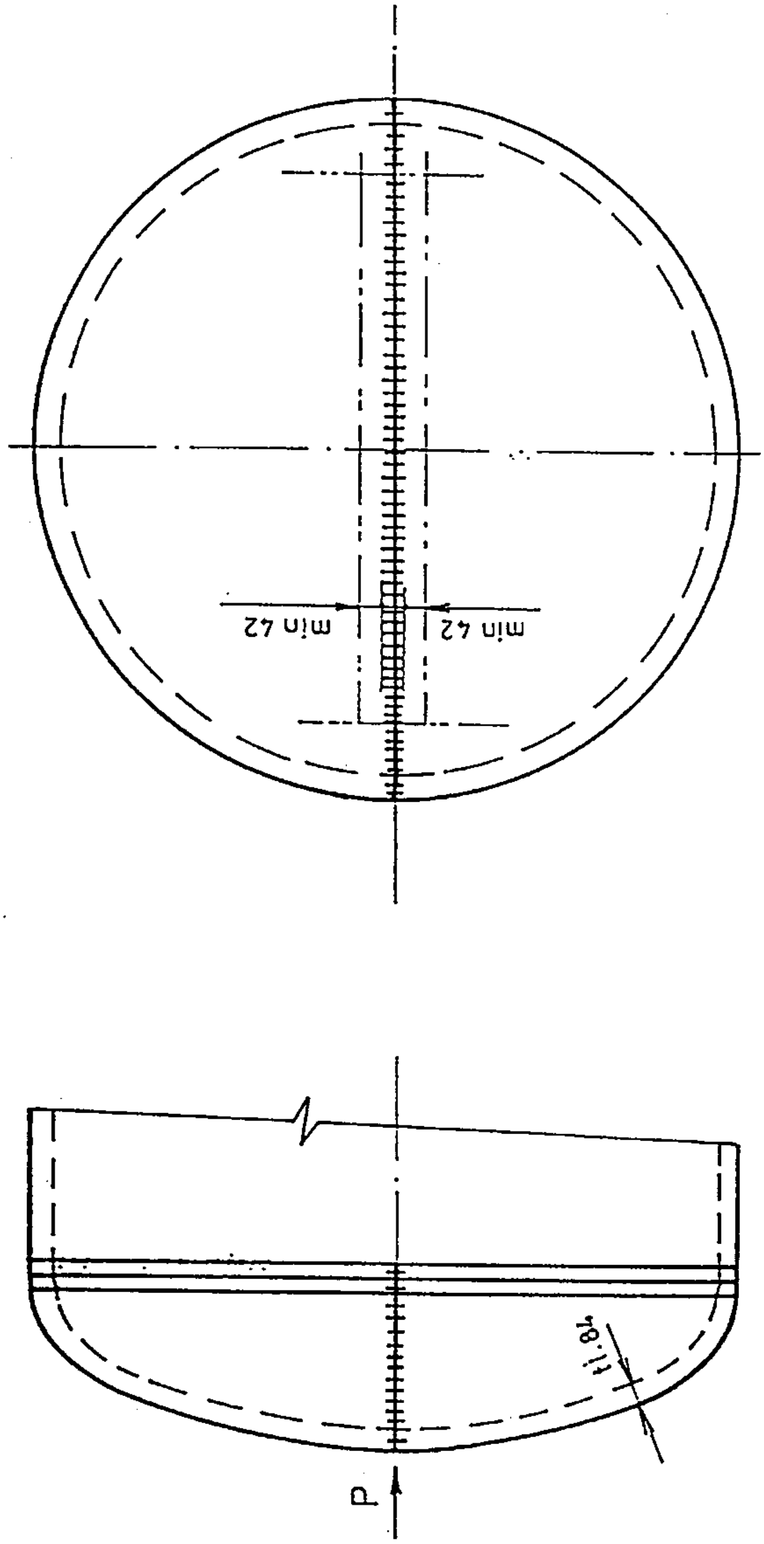


FIG. XI.3 Steam generator cover - wall thickness 84 mm – central part.

TABLE XI.10 Steam generator cover – wall thickness 84 mm – area with high curvature

Allowable linear flaw sizes

Surface flaw length [2c] = [l] [mm]	**Embedded flaw** length [2c] = [l] [mm]
11.1	17.3

Allowable laminar flaw size

Flaw area [F] [mm^2]
6540

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	1.6	33.0	2.5	49.4
0.10	1.9	19.5	2.8	28.0
0.15	2.3	15.3	3.2	21.4
0.20	2.8	13.8	3.7	18.5
0.25	3.3	13.2	4.3	17.1
0.30	3.8	12.7	5.0	16.7
0.35	4.9	13.9	5.8	16.6
0.40	5.5	13.7	6.7	16.7
0.45	5.6	12.4	7.8	17.3
0.50	5.7	11.5	9.0	18.0

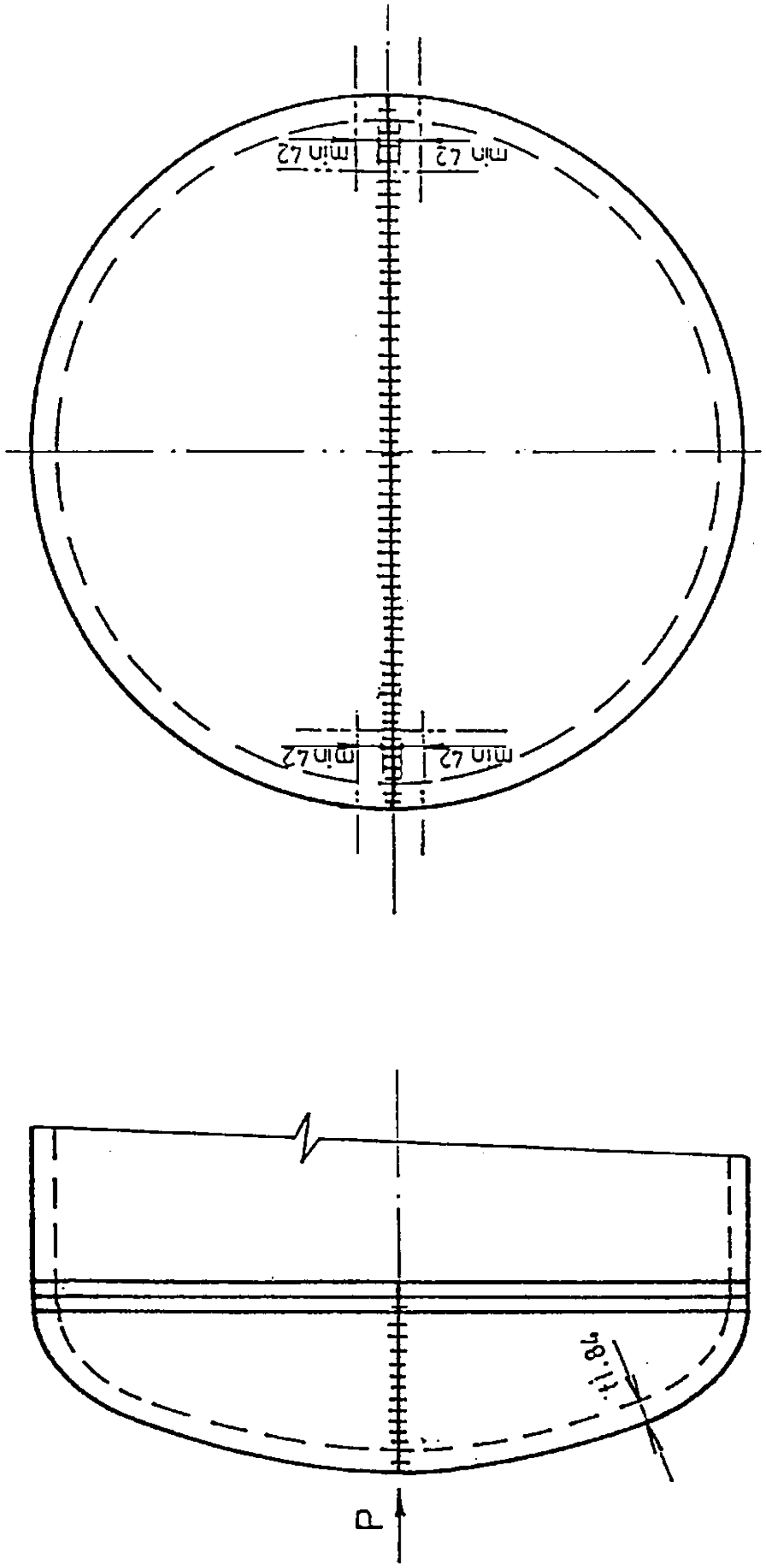


FIG. XI.4 Steam generator cover – wall thickness 84 mm – area with high curvature.

TABLE XI.11 Nozzle ID 700 on steam generator - wall thickness 55 mm

Allowable linear flaw sizes

No evaluation for nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
4190

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	1.8	36.3	2.1	41.8
0.10	2.0	19.8	2.4	23.6
0.15	2.2	15.0	2.7	18.0
0.20	2.6	13.0	3.1	15.7
0.25	3.0	12.1	3.6	14.5
0.30	3.5	11.7	4.3	14.3
0.35	4.1	11.6	4.9	14.1
0.40	4.6	11.4	5.8	14.4
0.45	4.7	10.4	6.8	15.0
0.50	4.8	9.6	7.9	15.7

In the area of inner corner of the nozzle it is necessary to take: a = 1.3 mm.

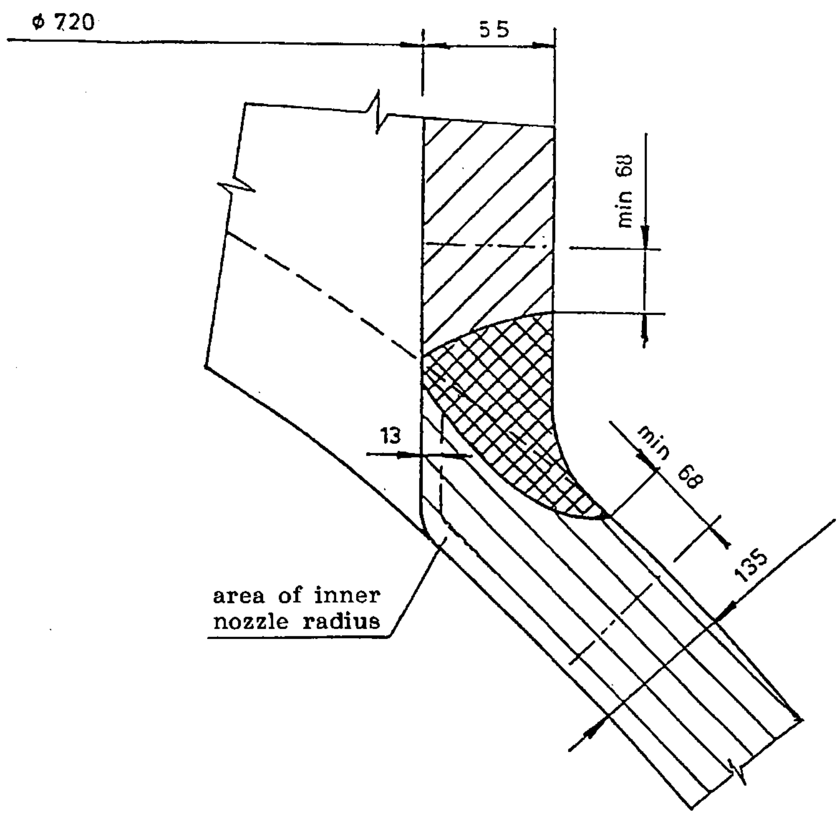


FIG. XI.5 Nozzle ID 700 on steam generator - wall thickness 55 mm.

TABLE XI.12 Nozzle 1100 on steam generator – wall thickness 77.5 mm

Allowable linear flaw sizes

No evaluation for nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
5905

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.1	41.8	2.4	48.0
0.10	2.2	22.5	2.7	27.1
0.15	2.6	17.6	3.1	20.7
0.20	2.9	14.7	3.6	17.8
0.25	3.5	13.9	4.1	16.4
0.30	4.1	13.4	5.0	16.8
0.35	4.6	13.3	5.6	15.9
0.40	5.3	13.2	6.5	16.3
0.45	5.4	12.0	7.6	16.9
0.50	5.5	11.0	8.7	17.5

In the area of inner corner of the nozzle it is necessary to take: a = 1.9 mm.

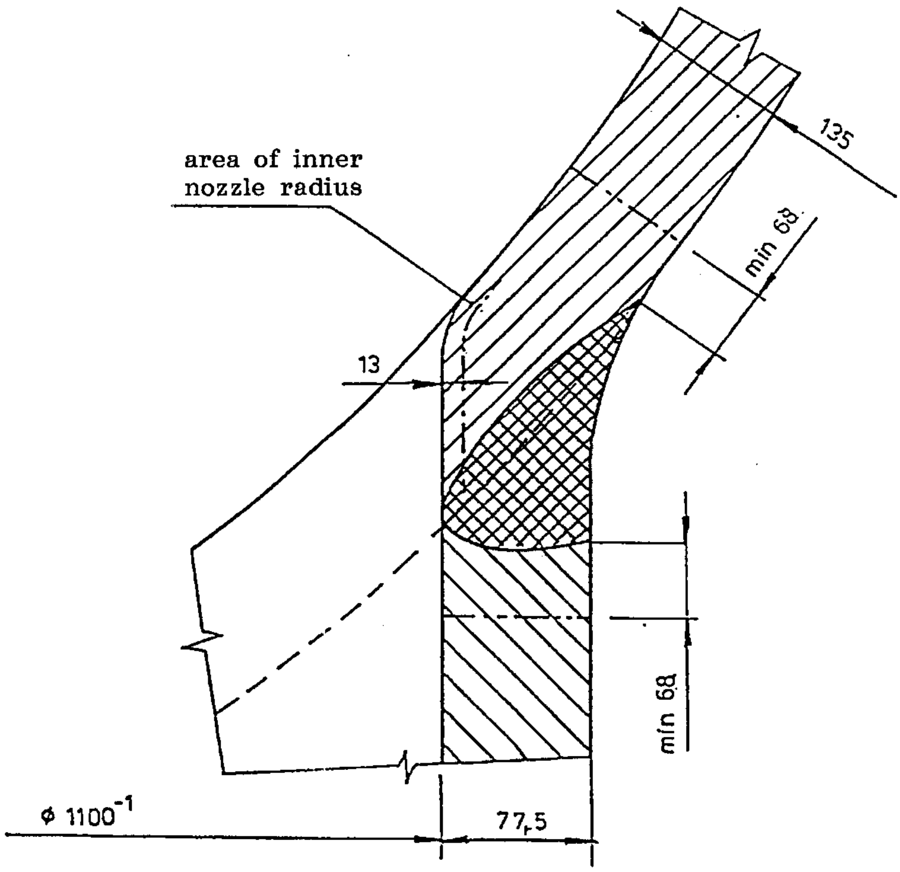


FIG. XI.6 Nozzle 1100 on steam generator – wall thickness 77.5 mm.

TABLE XI.13 Cylindrical part of pressurizer – wall thickness 145 mm

Allowable linear flaw sizes

Surface flaw Length [2c] = [l] [mm]	**Embedded flaw** Length [2c] = [l] [mm]
15.0	22.0

Allowable laminar flaw size

Flaw area [F] [mm^2]
11091

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.9	58	3.2	64
0.10	3.2	32	3.6	36
0.15	3.6	24	4.2	28
0.20	4.1	20	4.8	24
0.25	4.8	19	5.5	22
0.30	5.5	18	6.4	21
0.35	6.4	18	7.4	21
0.40	7.2	18	8.4	21
0.45	7.4	16	9.7	22
0.50	7.5	15	11.0	22

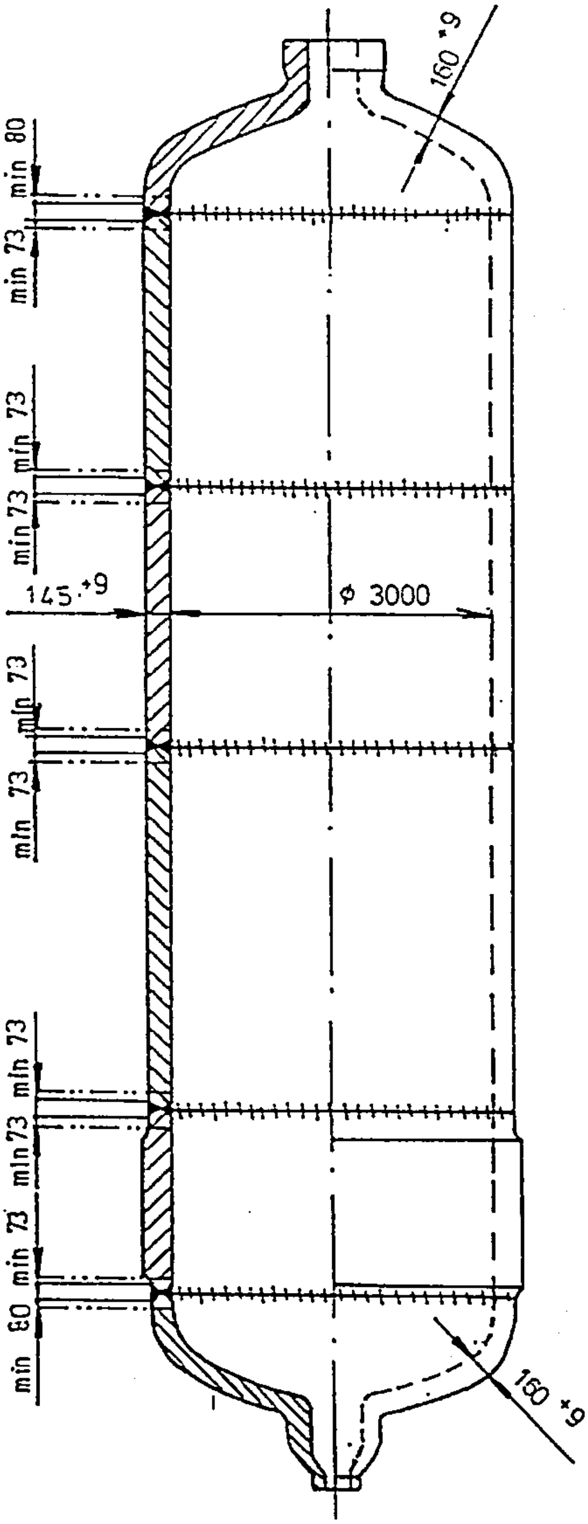


FIG. XI.7 Cylindrical part of pressurizer – wall thickness 145 mm.

TABLE XI.14 Nozzle of man-hole on upper cover of pressurizer – wall thickness 125 mm

Allowable linear flaw sizes

No evaluation for nozzles

Allowable laminar flaw sizes

a) In the bottom of the pressurizer – wall thickness 160 mm

Flaw area [F] [mm^2]
12247

b) The nozzle wall must be evaluated as planar flaws

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.5	50.0	2.7	55.0
0.10	2.7	27.0	3.1	31.0
0.15	3.1	21.0	3.6	24.0
0.20	3.5	17.0	4.1	20.0
0.25	4.1	16.0	4.7	19.0
0.30	4.7	16.0	5.5	18.0
0.35	5.5	16.0	6.4	18.0
0.40	6.2	16.0	7.2	18.0
0.45	6.4	14.0	8.4	18.0
0.50	6.5	13.0	9.5	19.0

In the area of inner corner of the nozzle it is necessary to take: a = 3.1 mm.

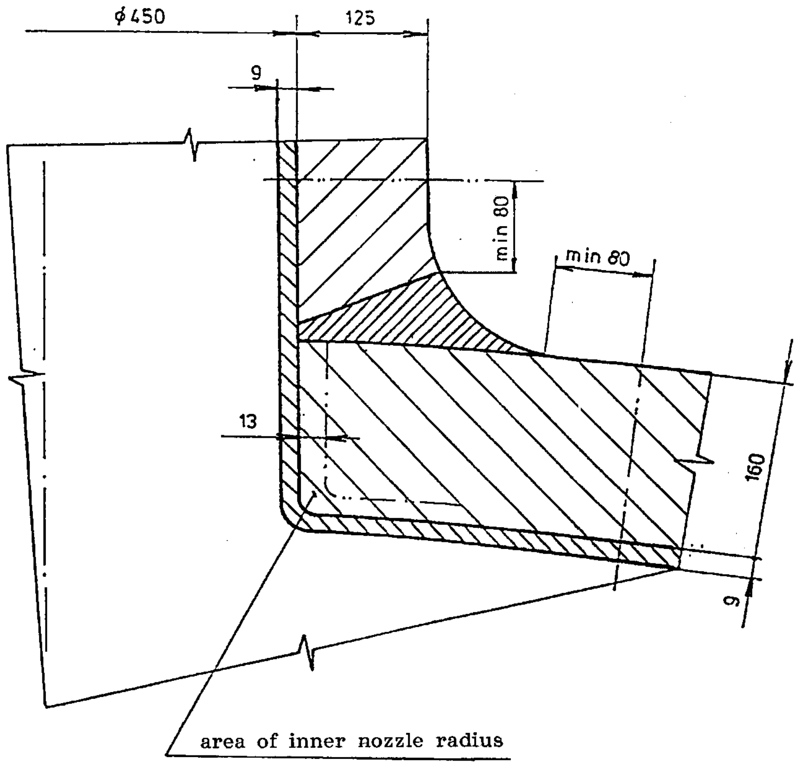


FIG. XI.8 Nozzle of man-hole on the upper cover of pressurizer – wall thickness 125 mm.

TABLE XI.15 Nozzle ID on lower bottom of pressurizer – wall thickness 75 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

a) In the bottom of the pressurizer – wall thickness 160 mm

Flaw area [F] [mm^2]
12247

b) The nozzle wall must be evaluated as planar flaws

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.1	42.0	2.4	48.0
0.10	2.2	22.0	2.7	27.0
0.15	2.6	17.0	3.1	20.0
0.20	2.9	15.0	3.6	18.0
0.25	3.4	14.0	4.1	17.0
0.30	4.0	13.0	5.0	17.0
0.35	4.6	13.0	5.6	16.0
0.40	5.2	13.0	6.5	16.0
0.45	5.4	12.0	7.6	17.0
0.50	5.5	11.0	8.8	17.0

In the area of inner corner of the nozzle it is necessary to take: a = 1.8 mm.

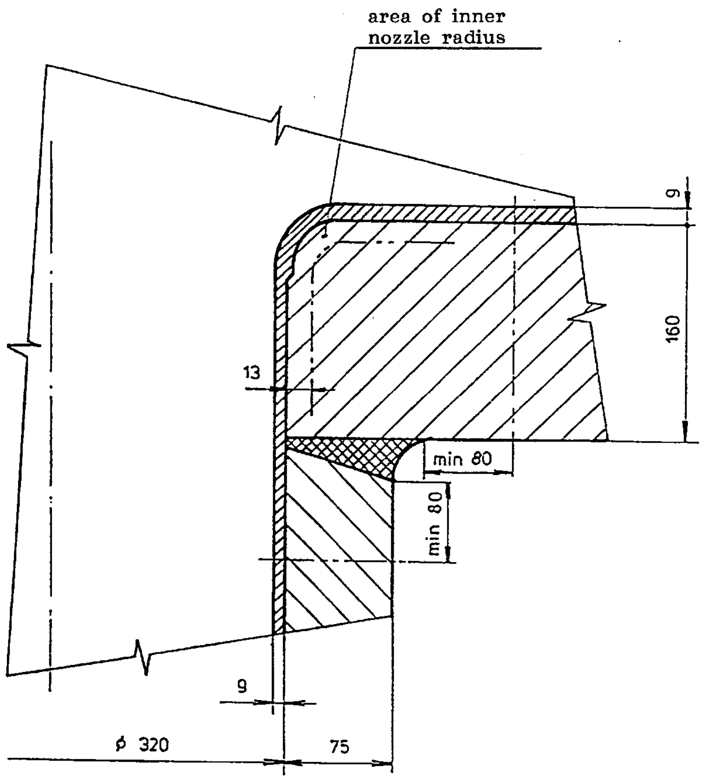


FIG. XI.9 Nozzle ID on lower bottom of pressurizer – wall thickness 75 mm.

TAB. XI.16 Lower bottom of pressurizer in the place of welded support – wall thickness 160 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
12247

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	1.9	38.0	3.2	64.0
0.10	2.1	21.0	4.0	40.0
0.15	2.6	17.0	4.6	31.0
0.20	3.1	16.0	5.3	26.0
0.25	3.7	15.0	6.1	24.0
0.30	4.4	15.0	7.0	23.0
0.35	5.1	15.0	8.2	23.0
0.40	5.9	15.0	9.3	23.0
0.45	6.7	15.0	10.7	24.0
0.50	7.6	15.0	12.1	24.0

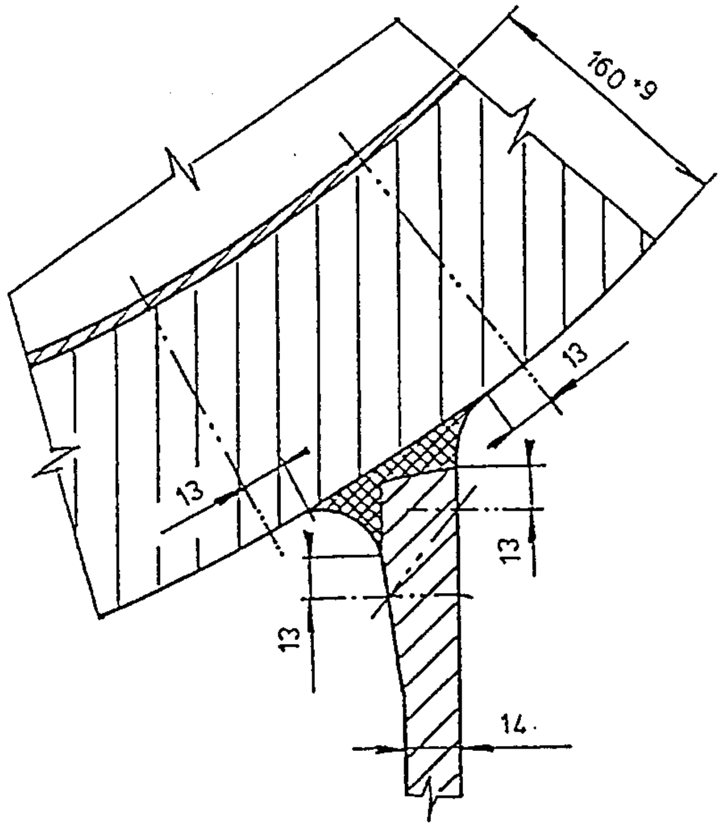


FIG. XI.10 Lower bottom of the pressurizer in the place of welded support – wall thickness 160 mm.

3.2 Indications found during in-service inspection in the SG and the PR of WWER 1000 are allowable if their schematised dimensions are smaller than values given in table XI.17 through XI.30.

TABLE XI.17 Cylindrical part of steam generator – wall thickness 145 mm

Allowable linear flaw sizes

Surface flaw length [2c] = [l] [mm]	**Embedded flaw** Length [2c] = [l] [mm]
15.1	22.5

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
11210

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.9	58.0	3.1	63.8
0.10	3.1	31.9	3.6	36.2
0.15	3.6	24.1	4.2	28.0
0.20	4.0	20.3	4.7	23.9
0.25	4.7	19.1	5.5	22.0
0.30	5.5	18.3	6.3	21.2
0.35	6.3	18.2	7.4	21.1
0.40	7.2	18.1	8.4	21.0
0.45	7.4	16.4	9.7	21.5
0.50	7.5	15.0	11.0	22.0

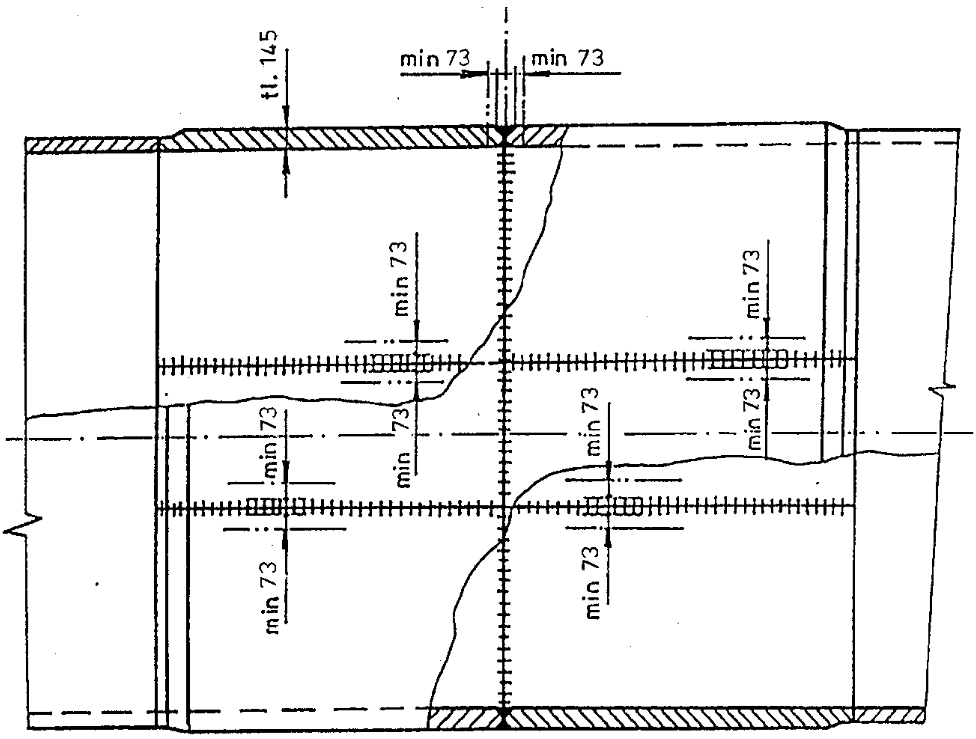


FIG. XI.11 Cylindrical part of steam generator – wall thickness 145 mm.

TABLE XI.18 Cylindrical part of the steam generator – wall thickness 105 mm

Allowable linear flaw sizes

Surface flaw Length [2c] = [l] [mm]	**Internal flaw** length [2c] = [l] [mm]
10.9	16.0

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
8090

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.1	42.0	2.3	46.2
0.10	2.3	23.1	2.6	26.2
0.15	2.6	17.5	3.0	20.3
0.20	2.9	14.7	3.4	17.3
0.25	3.4	13.8	3.9	15.9
0.30	4.0	13.3	4.6	15.4
0.35	4.6	13.2	5.3	15.3
0.40	5.2	13.1	6.0	15.2
0.45	5.3	11.9	7.0	15.6
0.50	5.4	10.9	7.9	15.9

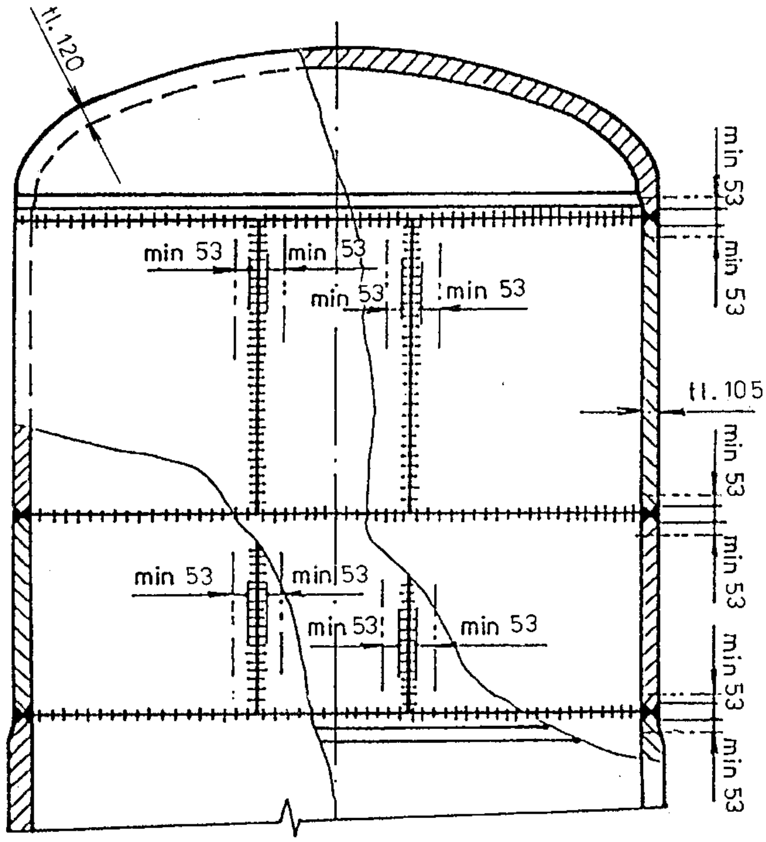


FIG. XI.12 Cylindrical part of steam generator – wall thickness 105 mm.

TABLE XI.19 Steam generator cover - wall thickness 120 mm – central part

Allowable linear flaw sizes

Surface flaw length [2c] = [l] [mm]	**Internal flaw** Length [2c] = [l] [mm]
12.5	18.2

Allowable laminar flaw size

Flaw area [F] [mm^2]
9260

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.4	48.0	2.6	52.8
0.10	2.6	26.4	3.0	30.0
0.15	3.0	20.0	3.4	23.2
0.20	3.3	16.8	3.9	19.8
0.25	3.9	15.8	4.5	18.2
0.30	4.5	15.2	5.2	17.6
0.35	5.2	15.0	6.1	17.4
0.40	6.0	15.0	6.9	17.3
0.45	6.1	13.6	8.0	17.8
0.50	6.2	12.4	9.1	18.2

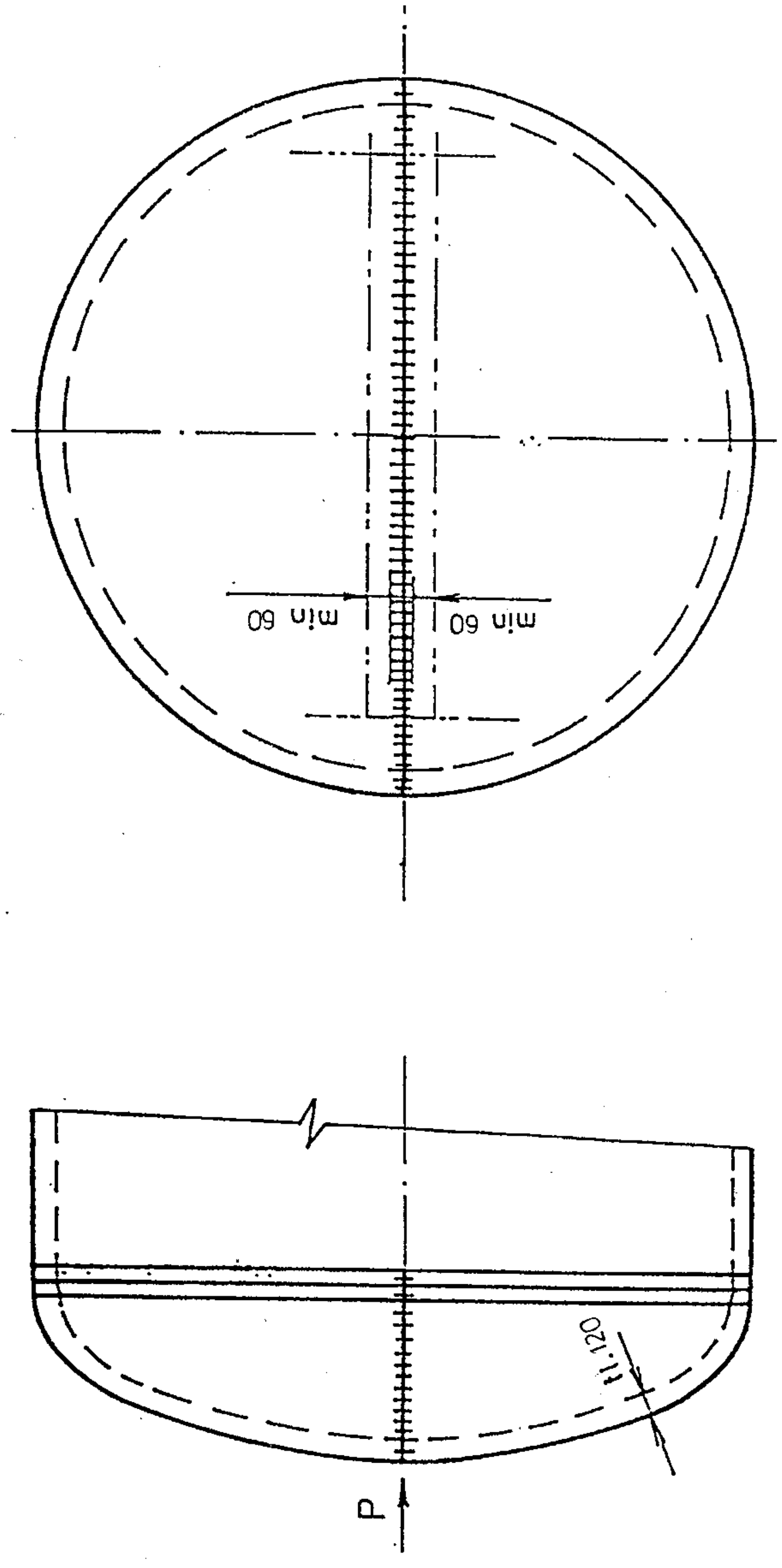


FIG. XI.13 Steam generator cover - wall thickness 120 mm – central part.

TABLE XI.20 Steam generator cover – wall thickness 120 mm – area with high curvature

Allowable linear flaw sizes

Surface flaw length [2c] = [l] [mm]	**Internal flaw** length [2c] = [l] [mm]
12.5	18.2

Allowable laminar flaw size

Flaw area [F] [mm^2]
9260

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	1.8	36.0	2.6	52.0
0.10	2.1	21.0	3.0	30.0
0.15	2.6	17.3	3.4	22.7
0.20	3.1	15.5	3.9	19.5
0.25	3.8	15.2	4.5	18.0
0.30	4.5	15.0	5.2	17.3
0.35	5.2	14.8	6.1	17.4
0.40	5.8	14.5	6.9	17.3
0.45	6.1	13.6	8.0	17.8
0.50	6.2	12.4	9.1	18.2

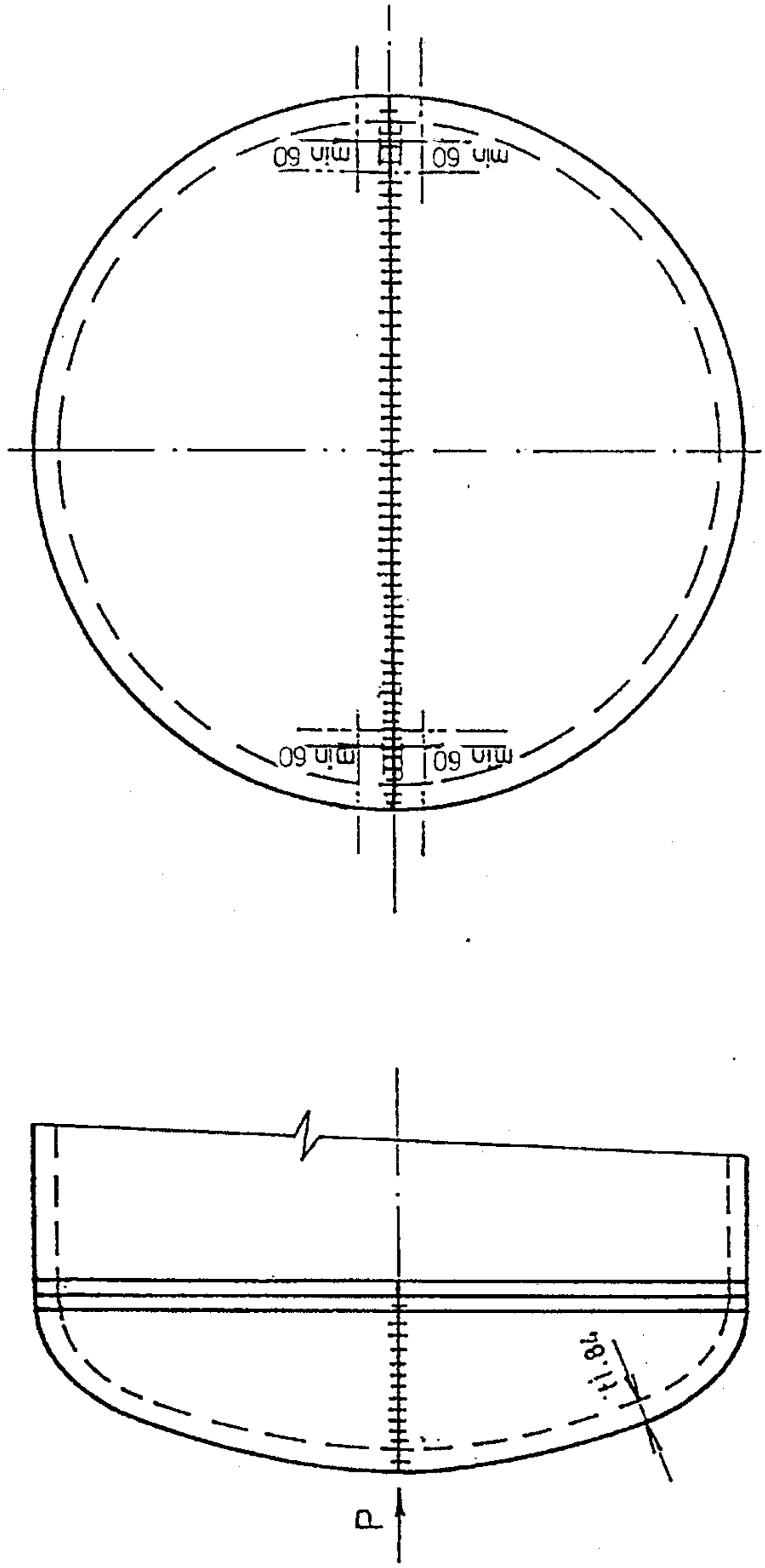


FIG. XI.14 Steam generator cover – wall thickness 120 mm – area with high curvature.

TABLE XI.21 Nozzle ID 800 on steam generator - wall thickness 127.5 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
11210

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.5	51.0	2.8	56.1
0.10	2.8	28.0	3.1	31.8
0.15	3.1	21.2	3.7	24.6
0.20	3.5	17.8	4.2	21.0
0.25	4.2	16.8	4.8	19.3
0.30	4.8	16.1	5.6	18.7
0.35	5.6	16.0	6.5	18.5
0.40	6.3	15.9	7.4	18.4
0.45	6.5	14.4	8.5	18.3
0.50	6.6	13.2	9.6	19.3

In the area of inner corner of the nozzle it is necessary to take: a = 3.19 mm.

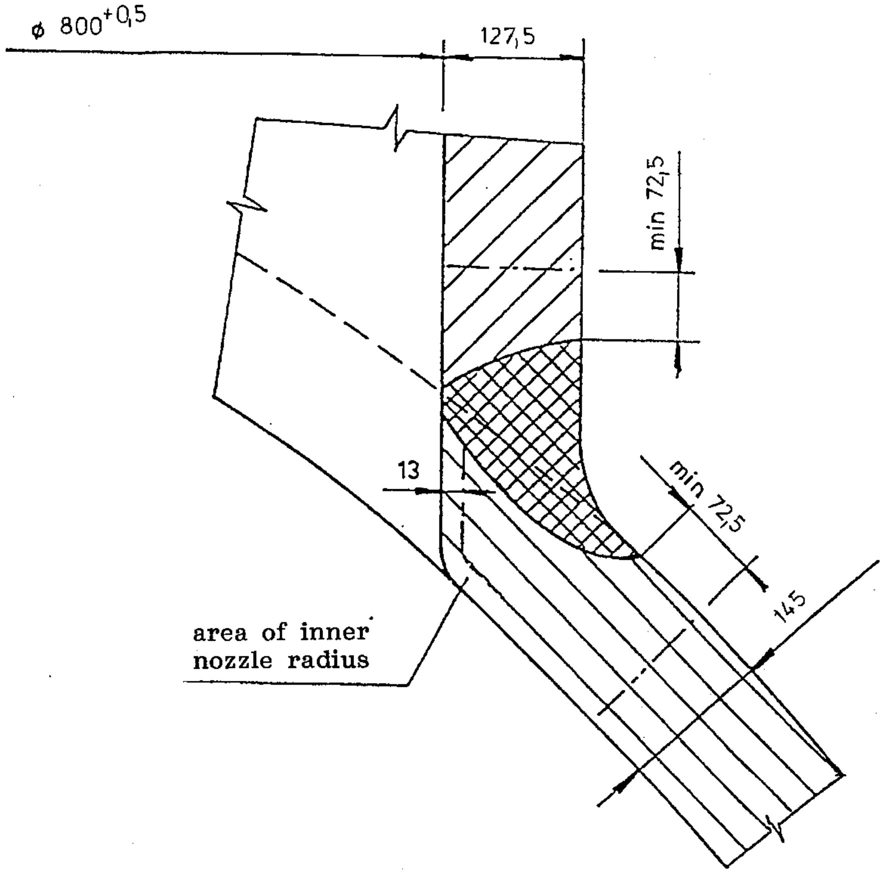


FIG. XI.15 Nozzle ID 800 on steam generator - wall thickness 127.5 mm.

TABLE XI.22 Nozzle 1190 on steam generator – wall thickness 122.5 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
11210

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.4	49.0	2.7	53.9
0.10	2.7	26.9	3.0	30.6
0.15	3.0	20.4	3.5	23.6
0.20	3.4	17.1	4.0	20.2
0.25	4.0	16.1	4.6	18.6
0.30	4.6	15.5	5.3	17.9
0.35	5.3	15.4	6.2	17.8
0.40	6.1	15.3	7.1	17.7
0.45	6.2	13.8	8.2	18.2
0.50	6.3	12.7	9.3	18.6

In the area of inner corner of the nozzle it is necessary to take: a = 3.06 mm.

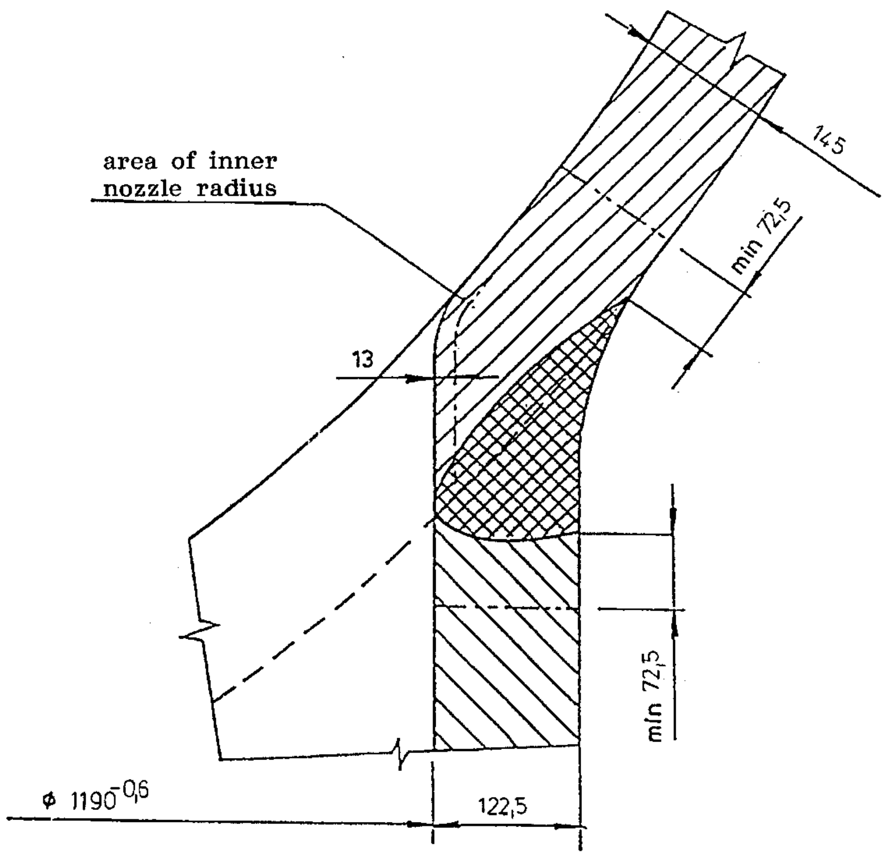


FIG. XI.16 Nozzle 1190 on steam generator – wall thickness 122.5 mm.

TABLE XI.23 Nozzle 506 on steam generator shell – wall thickness 62 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm²]
11210

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.0	40.9	2.3	47.1
0.10	2.2	22.3	2.6	26.6
0.15	2.5	16.9	3.0	20.2
0.20	2.9	14.5	3.5	17.6
0.25	3.4	13.6	4.0	16.3
0.30	3.9	13.2	4.8	16.1
0.35	4.5	13.1	5.5	15.9
0.40	5.1	12.8	6.5	16.2
0.45	5.2	11.7	7.6	16.9
0.50	5.3	10.7	8.8	17.7

In the area of inner corner of the nozzle it is necessary to take: a = 1.55 mm.

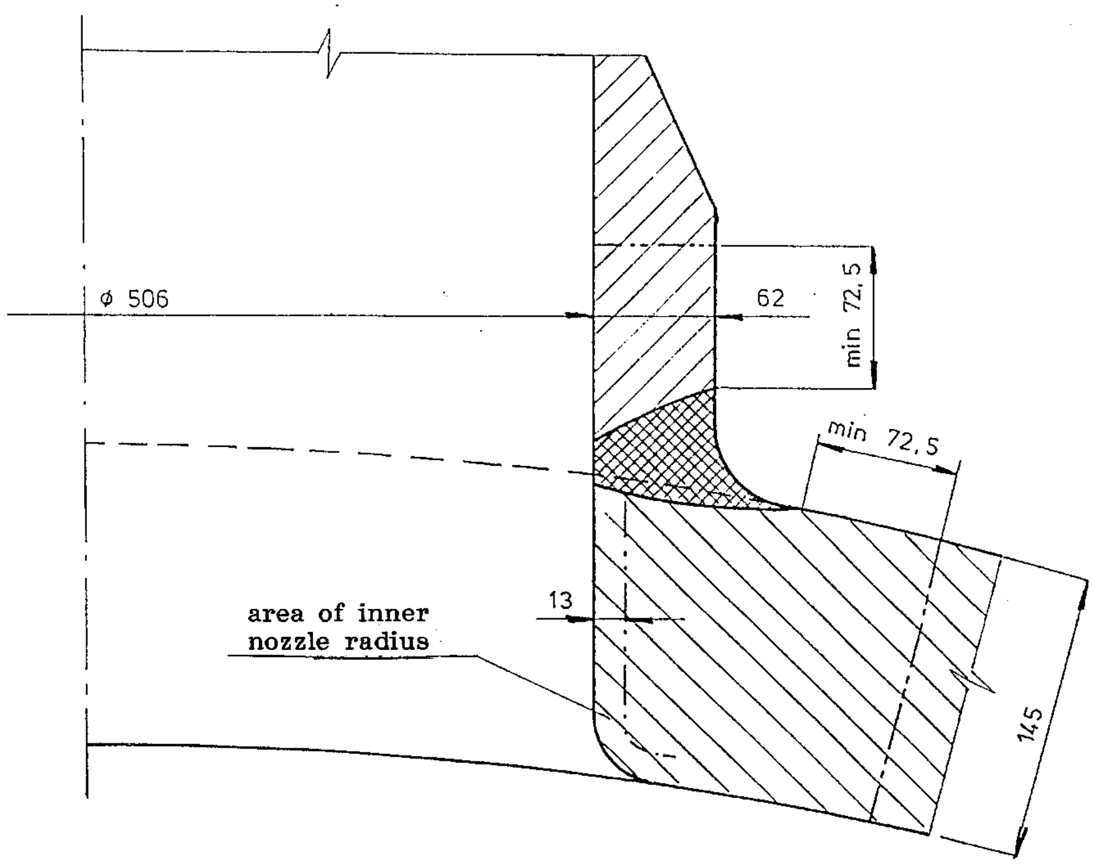


FIG. XI.17 Nozzle 506 on steam generator shell – wall thickness 62 mm.

TABLE XI.24 Nozzle 345 on steam generator shell – wall thickness 90 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm²]	
Shell wall thickness 145 mm	**Shell wall** thickness 105 mm
11210	8090

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.1	42.2	2.4	46.6
0.10	2.3	23.9	2.8	28.0
0.15	2.7	18.1	3.2	21.3
0.20	3.0	15.2	3.6	18.4
0.25	3.6	14.4	4.2	16.9
0.30	4.1	13.8	4.9	16.5
0.35	4.8	13.7	5.7	16.4
0.40	5.4	13.6	6.6	16.6
0.45	5.5	12.2	7.7	17.2
0.50	5.6	11.3	8.9	17.9

In the area of inner corner of the nozzle it is necessary to take: a = 2.25 mm.

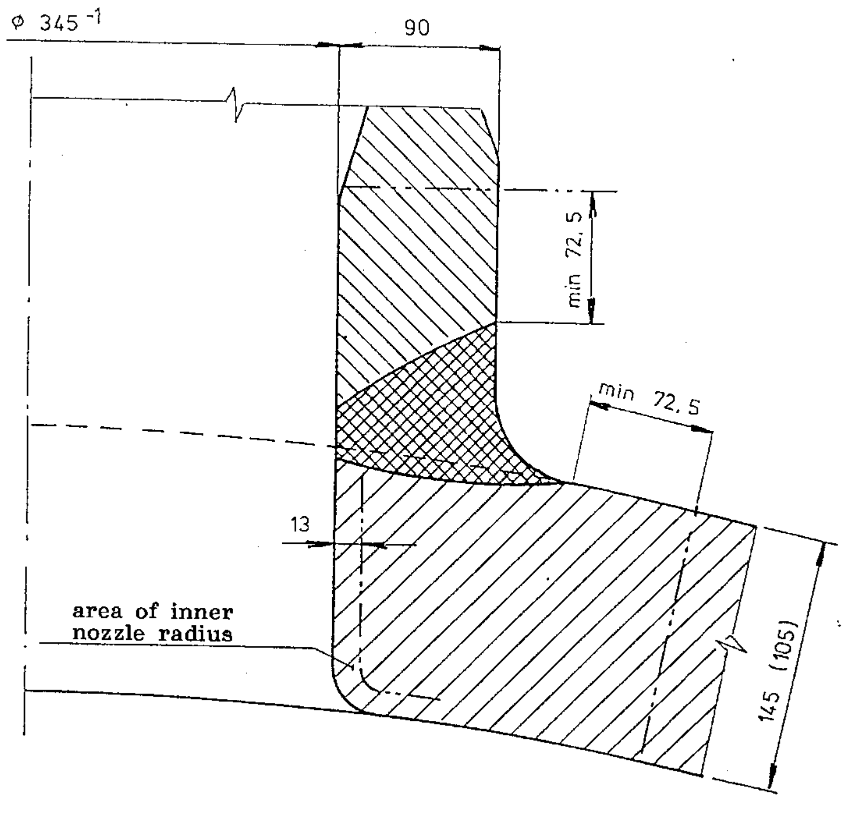


FIG. XI.18 Nozzle 345 on steam generator shell – wall thickness 90 mm.

TABLE XI.25 Nozzle 500 on steam generator cover – wall thickness 80 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
9260

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.1	43.2	2.3	46.8
0.10	2.3	23.4	2.7	27.0
0.15	2.6	17.4	3.0	20.4
0.20	2.9	14.8	3.5	17.5
0.25	3.5	14.0	4.0	16.2
0.30	4.0	13.5	4.7	15.9
0.35	4.6	13.3	5.4	15.6
0.40	5.3	13.2	6.3	15.9
0.45	5.4	12.0	7.3	16.4
0.50	5.4	10.9	8.4	16.9

In the area of inner corner of the nozzle it is necessary to take: a = 2.25 mm.

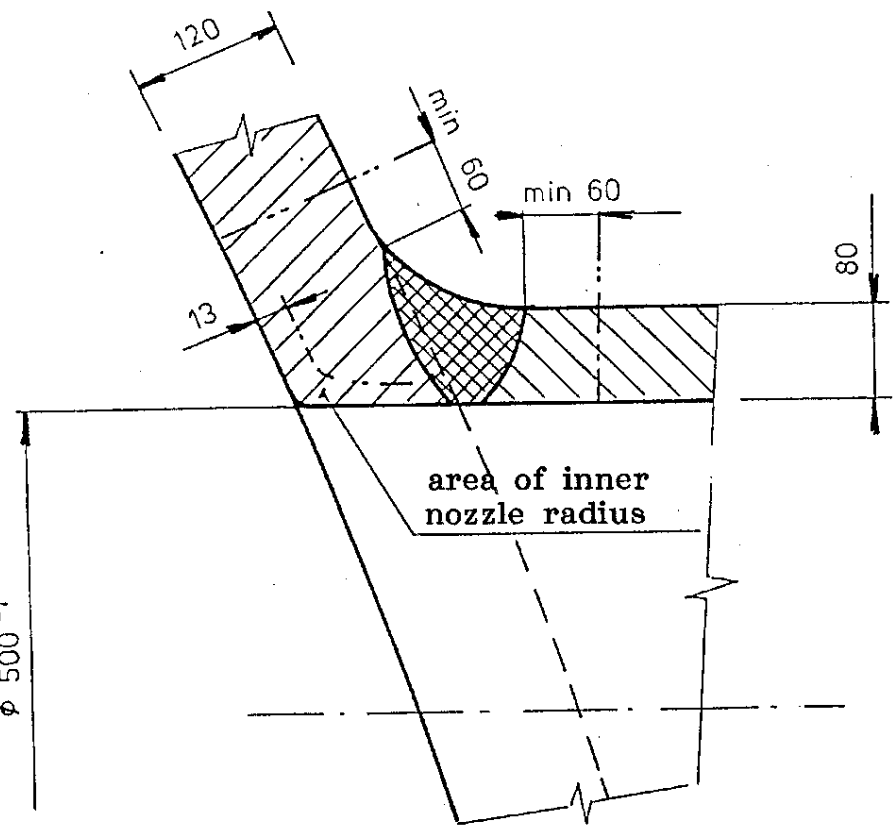


FIG. XI.19 Nozzle 500 on steam generator cover – wall thickness 80 mm.

TABLE XI.26 Cover of nozzle 800 of steam generator – cover wall thickness 26 mm

Allowable linear flaw sizes

Surface flaw Length [2c] = [l] [mm]	**Internal flaw** length [2c] = [l] [mm]
4.5	7.4

Allowable laminar flaw sizes

Flaw area [F] [mm^2]
2000

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	0.8	17.1	0.9	19.7
0.10	0.9	9.3	1.1	11.1
0.15	1.0	7.1	1.2	8.4
0.20	1.2	6.1	1.4	7.4
0.25	1.4	5.7	1.7	6.8
0.30	1.6	5.5	2.0	6.7
0.35	1.9	5.5	2.3	6.6
0.40	2.1	5.4	2.7	6.8
0.45	2.2	4.9	3.2	7.1
0.50	2.2	4.5	3.7	7.4

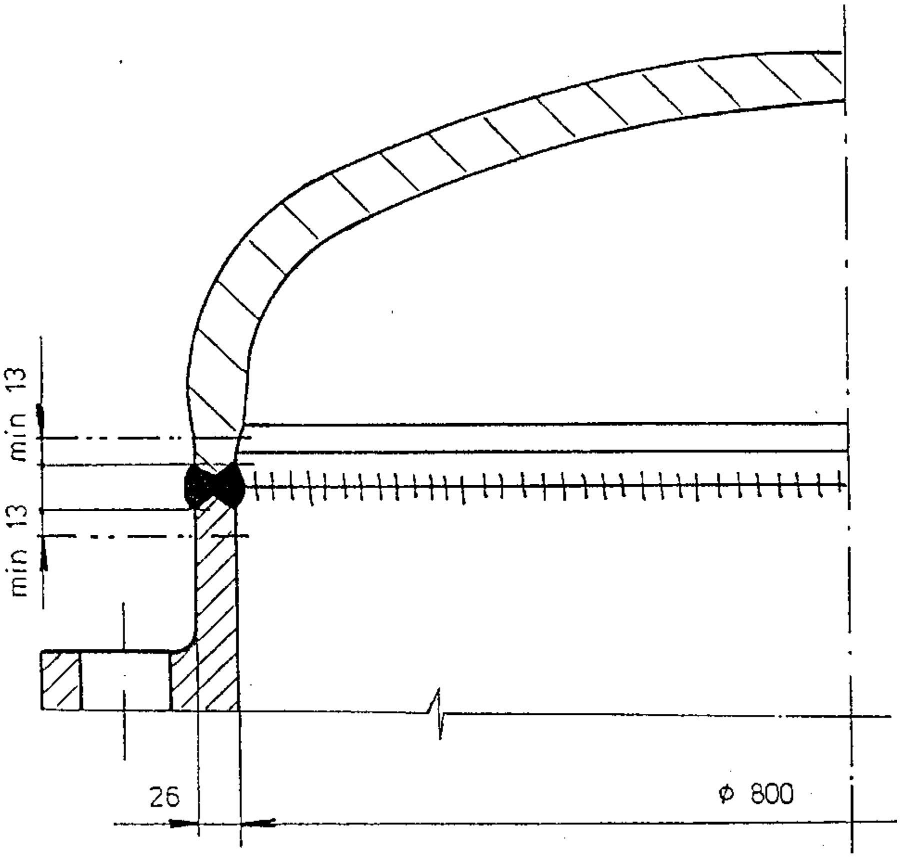


FIG. XI.20 Cover of nozzle 800 of steam generator – cover wall thickness 26 mm.

TABLE XI.27 Cylindrical part of pressurizer – wall thickness 157 mm

Allowable linear flaw sizes

Surface flaw Length [2c] = [l] [mm]	**Embedded flaw** Length [2c] = [l] [mm]
16.3	23.8

Allowable laminar flaw size

Flaw area [F] [mm^2]
12146

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	3.1	62.8	3.4	69.0
0.10	3.4	34.5	3.9	39.2
0.15	3.9	26.1	4.5	30.3
0.20	4.4	21.9	5.1	25.9
0.25	5.1	20.7	5.9	23.8
0.30	6.0	19.8	6.9	23.0
0.35	6.9	19.7	8.0	22.8
0.40	7.8	19.6	9.1	22.7
0.45	8.0	17.7	10.5	23.3
0.50	8.1	16.3	11.9	23.8

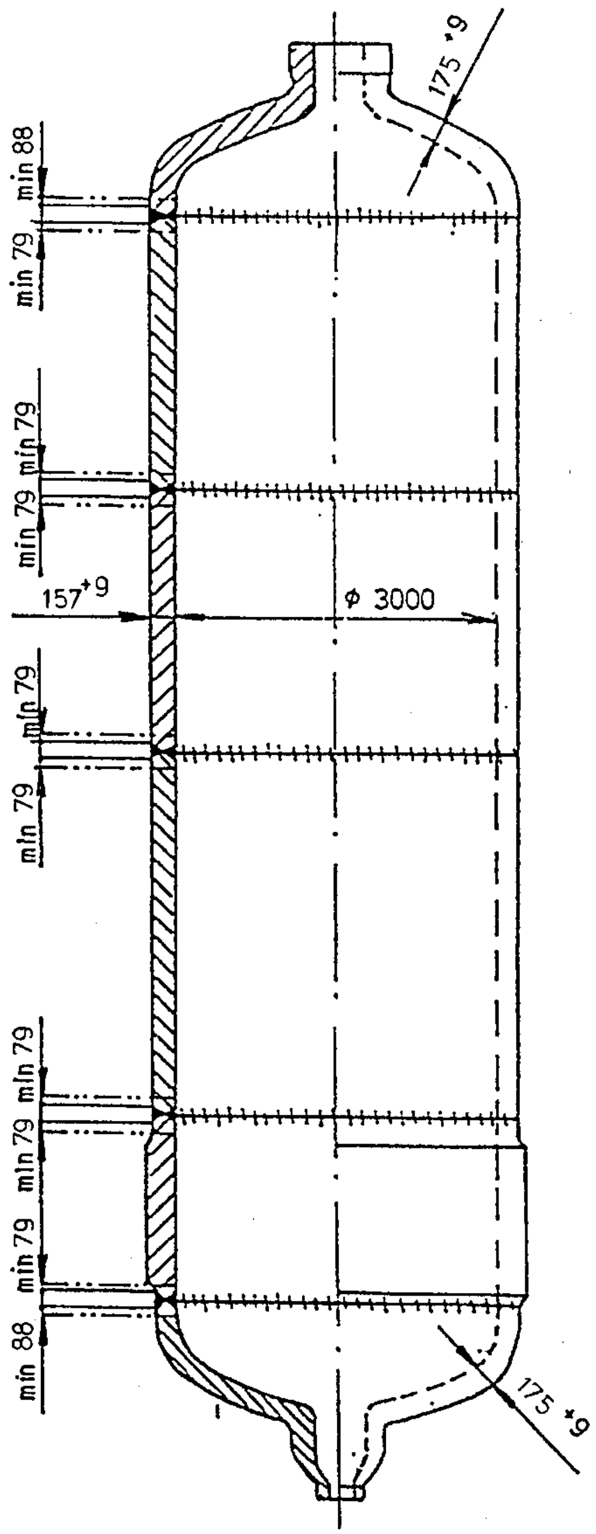


FIG. XI.21 Cylindrical part of pressurizer – wall thickness 157 mm.

TABLE XI.28 Nozzle 450 of manhole on upper cover of pressurizer – wall thickness 131 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

a) In the bottom (or cover?) of the pressurizer – wall thickness 175 mm

Flaw area [F] [mm^2]
13357

b) In the nozzle wall, flaws must be evaluated as planar flaws

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.6	52.4	2.9	57.6
0.10	2.9	28.8	3.3	32.8
0.15	3.3	21.8	3.8	25.3
0.20	3.7	18.3	4.3	21.6
0.25	4.3	17.3	4.9	19.9
0.30	5.0	16.6	5.8	19.2
0.35	5.8	16.5	6.7	19.1
0.40	6.6	16.4	7.6	19.0
0.45	6.7	14.8	8.8	19.5
0.50	6.8	13.6	10.0	19.9

In the area of inner corner of the nozzle it is necessary to take: a = 3.2 mm.

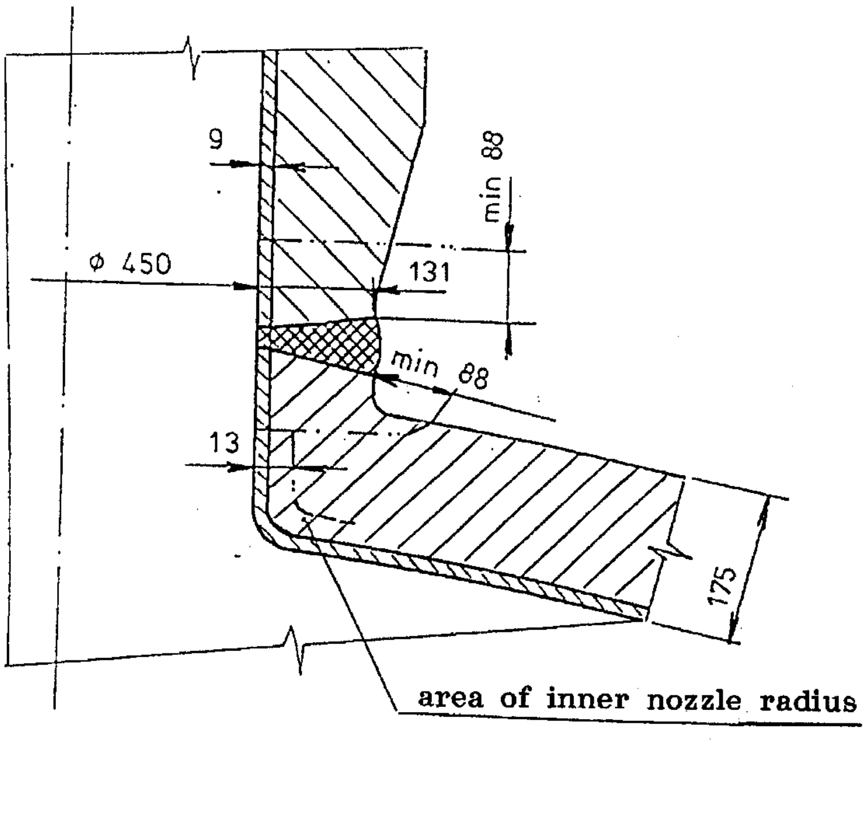


FIG. XI.22 Nozzle 450 of manhole on upper cover of pressurizer – wall thickness 131 mm.

TABLE XI.29 Nozzle 450 on lower bottom of pressurizer – wall thickness 131 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

a) In the bottom of the pressurizer – wall thickness 175 mm

Flaw area [F] [mm^2]
13357

b) In the nozzle wall, flaws must be evaluated as planar flaws

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	2.6	52.4	2.9	57.6
0.10	2.9	28.8	3.3	32.8
0.15	3.3	21.8	3.8	25.3
0.20	3.7	18.3	4.3	21.6
0.25	4.3	17.3	4.9	19.9
0.30	5.0	16.6	5.8	19.2
0.35	5.8	16.5	6.7	19.1
0.40	6.6	16.4	7.6	19.0
0.45	6.7	14.8	8.8	19.5
0.50	6.8	13.6	10.0	19.9

In the area of inner corner of the nozzle it is necessary to take: a = 3.2 mm.

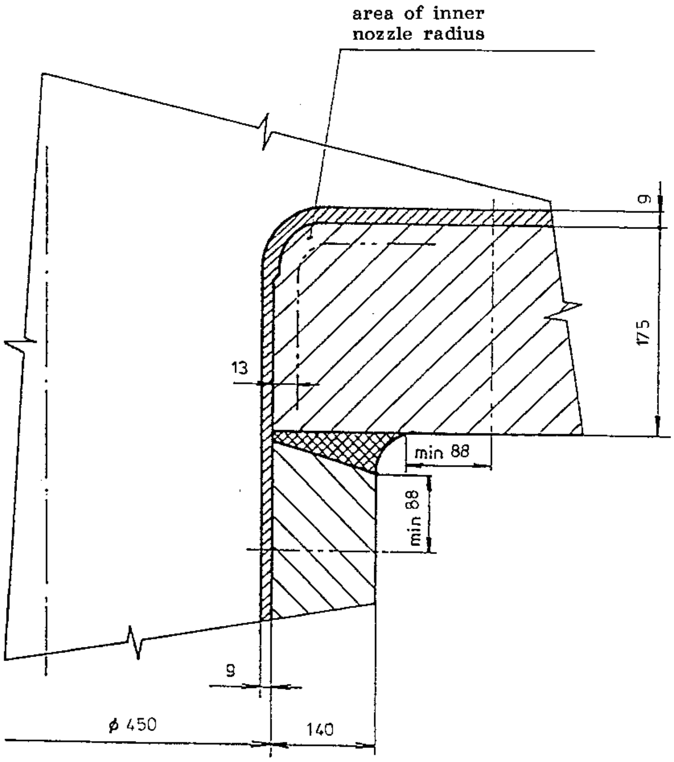


FIG. XI.23 Nozzle 450 on lower bottom of pressurizer – wall thickness 131 mm.

TABLE XI.30 Lower bottom of pressurizer in the place of welded support – wall thickness 175 mm

Allowable linear flaw sizes

No evaluation for the nozzles

Allowable laminar flaw sizes

Flaw area [F] [mm²]
13357

Allowable planar flaw sizes

	Surface flaw [mm]		Embedded flaw [mm]	
a/l	[a]	[l] = [2c]	[a]	[l] = [2c]
0.05	3.5	70.0	3.8	77.0
0.10	3.8	38.0	4.4	43.8
0.15	4.3	26.1	5.1	33.8
0.20	4.9	24.5	5.8	28.9
0.25	5.7	23.1	6.7	26.6
0.30	6.6	22.1	7.0	23.3
0.35	7.7	22.0	8.9	25.5
0.40	8.8	21.8	10.1	25.4
0.45	8.9	19.8	11.7	26.1
0.50	9.1	18.2	13.3	26.6

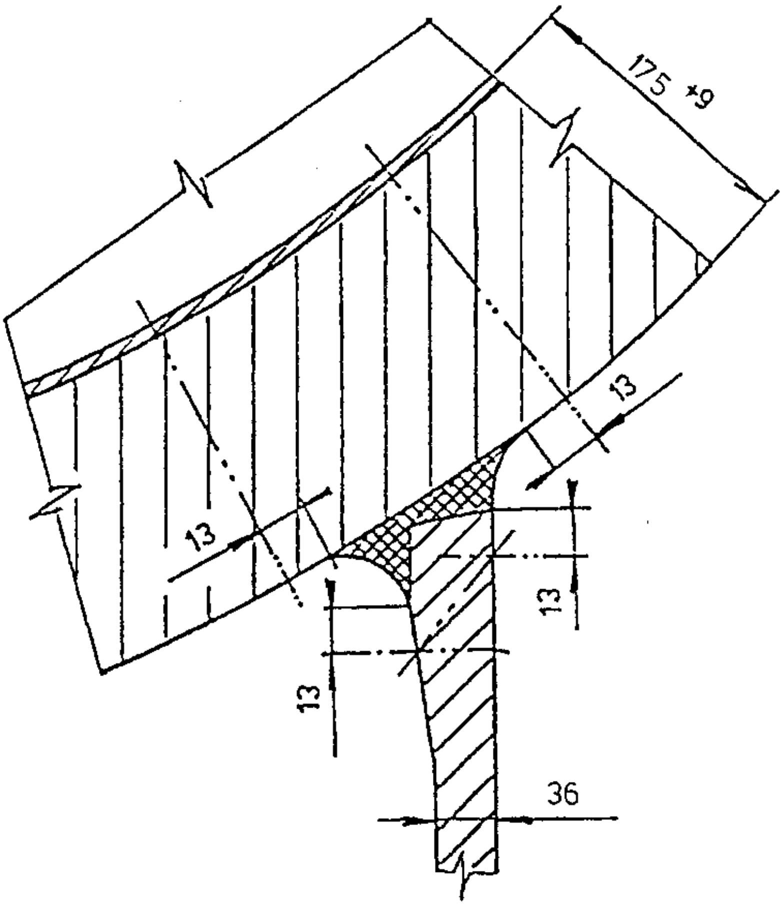


FIG. XI.24 Lower bottom of pressurizer in the place of welded support – wall thickness 175 mm.

4. Austenitic piping

TABLE XI.31 Allowable flaw sizes for austenitic piping with $R_{P0.2}$ < 300 MPa

	s [mm]			
a/l	10	20	50	100
	[a/s]			
0.00	0.103	0.105	0.107	0.108
0.05	0.103	0.105	0.107	0.108
0.10	0.105	0.106	0.108	0.110
0.15	0.106	0.107	0.110	0.112
0.20	0.107	0.109	0.111	0.113
0.25	0.109	0.111	0.113	0.115
0.30	0.110	0.113	0.115	0.118
0.35	0.113	0.115	0.118	0.120
0.40	0.116	0.117	0.120	0.123
0.45	0.119	0.119	0.123	0.125
0.50	0.123	0.124	0.126	0.128

5. Ferritic piping

TABLE XI.32. Allowable flaw sizes for ferritic piping

a/l	s(mm)			
	65 and less		100 through 300	
	Surface Flaw a/s,%	Embedded Flaw a/s,%	Surface Flaw a/s,%	Embedded Flaw a/s,%
0.00	3.1	3.4	1.9	2.0
0.05	3.3	3.8	2.0	2.2
0.10	3.6	4.3	2.2	2.5
0.15	4.1	4.9	2.5	2.9
0.20	4.7	5.7	2.8	3.3
0.25	5.5	6.6	3.3	3.8
0.30	6.4	7.8	3.8	4.4
0.35	7.4	9.0	4.4	5.1
0.40	8.3	10.5	5.0	5.8
0.45	8.5	12.3	5.1	6.7
0.50	8.7	14.3	5.2	7.6

NOTES:

(1) For intermediate flaw aspect ratios "*a/l*" and thickness "*s*", linear interpolation is permissible.

(2) The total depth of a Internal flaw is 2*a*.

(3) For embedded flaw, any portion of the flaw is not less than "*a*" from the surface of the component nearest the flaw. If the nearest surface of the component is cladding, any portion of the flaw is not less than "*a*" from the the clad-base metal interface of the component nearest the flaw.

References

[1] Vejvoda, S., *Proposal of the procedure for an assessment of found flaws*. Report of UAM Brno No. 2150/95, Brno (1995).

[2] Licka, A., *Allowed flaws database*. Report of UAM Brno No. 2193/95, Brno (1995).

[3] Smejc, V., *Procedure for determination of allowed flaws found at the service non-destructive examination of the pressurizer. Database of allowed flaws in chosen areas of the pressurizer,* Report of VITKOVICE department 603. No. Z-4-JL-000056, Vitkovice.

APPENDIX XII

Evaluation of flaw acceptability in components

1. General remarks

1.1 List of symbols

a	Depth (minor semi-axis) of a crack, [m]
c	Half length (major semi-axis) of a crack, [m]
s	Wall thickness, [m]
a_0	Depth of a crack increased by safety factor, [m]
c_0	Half-length of a crack increased by safety factor, [m]
a_f	End-of-life depth of a crack, [m]
c_f	End-of-life half-length of a crack, [m]
a_e	Estimated depth of a crack, [m]
c_e	Estimated half-length of a crack, [m]
n_a	Safety factor applied to the crack depth
da/dN	Crack growth rate, [m.cycle^{-1}]
K_I	Stress intensity factor, [MPa.m$^{0.5}$]
K_{Imin}	Minimum value of stress intensity factor (within its range), [MPa.m$^{0.5}$]
K_{Imax}	Maximum value of stress intensity factor (within its range), [MPa.m$^{0.5}$]
ΔK_I	Stress intensity factor range, [MPa.m$^{0.5}$]
R	Load ratio, (K_{Imin} / K_{Imax})
N	Number of cycles
n_i	Number of occurrences of stress intensity range of type *i*
p	Pressure, [MPa]
T	Temperature, [^{o}C]
[T_h]	Allowable hydrotest temperature, [°C]
[T_t]	Maximum allowable transition temperature, [°C]
[T_A]	Maximum allowable transition temperature for crack arrest, [°C]
A, B, D	Points on crack front
NDT	Non-destructive testing

1.2 This procedure will be applied for the evaluation of flaws found in components (ferritic pressure vessels etc.) by non-destructive testing (NDT) during in-service inspection. Flaws are characterised by necessary parameters and are schematised in accordance with Chapter 2 of this Appendix and Appendix X into elliptical or semi-elliptical, embedded, or surface cracks.

1.3 If a flaw was found during several in-service inspections or by several different methods or by testing on different surfaces (inner or outer), the results obtained by the most relevant NDT shall be taken for the evaluation of flaw acceptability. If results of qualified NDT are available, they shall be used for the evaluation. The relevance of different NDT methods (techniques, applied probes) is determined using qualification dossier (technical justification file), if it is available. In the case of equal level of relevance of several NDT examinations, the most conservative results (i.e. the largest one of determined depths and the largest one of determined lengths) are taken for the evaluation.

Note: In terminology of non-destructive testing community, "crack depth" is termed as "crack height" or "crack through wall extent dimension".

1.4 The allowable sizes of flaws are given in Appendix XI; smaller flaws need not be further evaluated.

1.5 If the schematised flaw dimensions do not meet the conditions prescribed in Appendix XI, a more detailed evaluation according to this Appendix is necessary.

1.6 A safety factor is applied to the schematised flaw according to Chapter 3 of this Appendix.

1.7 The crack growth calculation for the schematised flaw increased by the safety factor is then performed according to Chapter 4 of this Appendix.

1.8 The assessment of final crack resistance (obtained by the application of the previous steps) against fast fracture is performed according to Chapter 5 of this Appendix.

2. General remarks

Schematisation of flaws is performed in accordance with the following rules:

2.1 The qualitative and quantitative evaluation of flaws found by non-destructive testing serves for the determination of flaw parameters - sizes, orientation and location.

2.2 All flaws are conservatively schematised as cracks in accordance with Appendix X.

2.3 The schematised crack is projected to planes perpendicular to the directions of two main stresses (i.e., in the case of cylindrical vessel perpendicular to circumferential direction) as well as perpendicular to axial directions.

2.4 The criterion for distinguishing between surface and internal flaws (presented in Appendix X) depends on the flaw location in the vessel wall thickness.

2.5 Cladded vessel flaws which do not meet the criterion for an internal flaw are evaluated either as underclad cracks, if both cladding properties are well known and cladding integrity is assured by qualified NDE; otherwise, they are evaluated as surface cracks.

2.6 The detailed procedure of flaw schematisation is given in Appendix X.

2.7 The schematised flaw sizes are compared to the values prescribed in the table of allowable sizes of flaws given in Appendix XI. If it meets the criteria, the flaw need not be further evaluated. Otherwise, a more detailed evaluation in accordance with this Appendix is necessary.

3. Safety factor applied to schematised flaw

3.1 Safety factor, $\boldsymbol{n_a}$ **= 2,** is applied to the depth *a* of schematised flaw.

3.2 The aspect ratio *a/c* of the crack increased by the safety factor should preserve its original value.

3.3 The centre of the schematised crack increased by the safety factor should remain in its original position.

3.4 If the condition (XII.2) in Appendix X is not met after applying the safety factor on internal schematised crack, the crack has to be converted to a surface or underclad one.

4. Crack growth calculation

4.1 The schematised crack increased by the safety factor with its sizes denoted by a_o, c_o can grow as a result of cyclic loading during component operation. Crack sizes for the end-of-life are then denoted by a_f, c_f.

4.2 The crack sizes considered for end-of-life, a_f, c_f, are calculated from the relations

$$a_f = a_o + \sum_{i=1}^{k} n_i \, (da/dN)_i$$

$$c_f = c_o + \sum_{i=1}^{k} n_i \, (dc/dN)_i \qquad \text{(XII.1)}$$

where k is total number of different types of stress intensity factor ranges, described by their upper values, K_{Imax}, and their lower values, K_{Imin}, with a number of occurrences, n_i, for the required lifetime.

4.3 The crack growth rate, da/dN, is determined on the basis of linear elastic fracture mechanics using the following equations describing the crack growth rate dependence on stress intensity factor range, ΔK_I, load ratio, *R,* and the environment effect:

$$da/dN = C \cdot (\Delta K_I)^m \qquad \text{(XII.2)}$$

Formulae for crack growth rate, dc/dN, are identical with those for da/dN.

Material constants *C* and *m* dependent on load ratio, *R,* and the environment are given in 4.4,

The stress intensity factor range, ΔK_I, is given by the relation:

$$\Delta K_I = K_{Imax} - K_{Imin} \qquad \text{(XII.3)}$$

The load ratio of stress intensity factor, R, is given by the relation:

$$R = K_{Imin} / K_{Imax} \qquad \text{(XII.4)}$$

When calculating crack growth of semi-axis *a*, stress intensity factor range, ΔK_I, in point A is used for surface or underclad crack, and in points A and B for internal crack.

When calculating crack growth of semi-axis *c*, stress intensity factor range, ΔK_I, in point B is used for surface or underclad crack and in point D for internal crack.

Formulae for calculation of crack growth (see Section 4.4) differ with respect to load ratio, *R*.

4.4 The following material constants and subsequent crack growth rates for materials of WWER type components may be used in crack growth calculations:

4.4.1. Steels 15Kh2MFA, 15Kh2MFAA, 15Kh2NMFA, 15Kh2NMFAA and their welding joints:

(a) *air:*

$$\frac{da}{dN} = 2.8 \cdot 10^{-11} \left(\frac{\Delta K}{(1-R)^{0.25}} \right)^{2.7} \quad \text{(XII.5)}$$

(b) *water:*

$$\frac{da}{dN} = 2.1 \cdot 10^{-17} \left(\frac{\Delta K}{(1-R)^{0.5}} \right)^{7.2} \quad \text{for} \quad \frac{\Delta K}{(1-R)^{0.5}} < 31.8 \quad \text{(XII.6)}$$

$$\frac{da}{dN} = 1.08 \cdot 10^{-8} \left(\frac{\Delta K}{(1-R)^{0.5}} \right)^{1.4} \quad \text{for} \quad \frac{\Delta K}{(1-R)^{0.5}} \geq 31.8 \quad \text{(XII.7)}$$

4.4.2. Austenitic cladding:

(a) *air:*

$$\frac{da}{dN} = 5.2 \cdot 10^{-12} \left(\frac{\Delta K}{(1-R)^{0.25}} \right)^{3.3} \quad \text{(XII.8)}$$

(b) *water:*

$$\frac{da}{dN} = 1.04 \cdot 10^{-11} \left(\frac{\Delta K}{(1-R)^{0.25}} \right)^{3.3} \quad \text{(XII.9)}$$

4.4.3. Steel 22K and its welding joints:

(a) *air:*

$$\frac{da}{dN} = 1.5 \cdot 10^{-11} \left(\frac{\Delta K}{(1-R)^{0.25}} \right)^{3.1} \quad \text{(XII.10)}$$

(b) *water*:

$$\frac{da}{dN} = 2.1 \cdot 10^{-17} \left(\frac{\Delta K}{(1-R)^{0.5}} \right)^{7.2} \quad \text{for} \quad \frac{\Delta K}{(1-R)^{0.5}} < 31.8 \quad \text{(XII.11)}$$

$$\frac{da}{dN} = 1.08 \cdot 10^{-8} \left(\frac{\Delta K}{(1-R)^{0.5}} \right)^{1.4} \quad \text{for} \quad \frac{\Delta K}{(1-R)^{0.5}} \geq 31.8 \qquad \text{(XII.12)}$$

4.4.4. Austenitic steels of 08Kh18N10T type and their welding joints:

(a) *air:*

$$\frac{da}{dN} = 5.2 \cdot 10^{-12} \left(\frac{\Delta K}{(1-R)^{0.25}} \right)^{3.3} \qquad \text{(XII.13)}$$

(b) *water:*

$$\frac{da}{dN} = 1.04 \cdot 10^{-12} \left(\frac{\Delta K}{(1-R)^{0.25}} \right)^{3.3} \qquad \text{(XII.14)}$$

(c) water, steam water, steam with O_2:

$$\frac{da}{dN} = 5.2.10^{-11} \left(\frac{\Delta K}{(1-R)^{0.25}} \right)^{3.3} \qquad \text{(XII.15)}$$

4.5 If not applicable the previous formulae, experimentally determined dependences *da/dN* may be used for chosen materials, if sufficient number of representative materials (heats and welding joints of different chemical compositions and mechanical properties, all within allowed manufacturing tolerances) and test specimens have been tested. Tests must be performed in an environment similar to operating conditions in corresponding component of the WWER type, either in air/vacuum at temperatures between 20° and 325°C or in a pressurised water of coolant parameters. Such a programme must be negotiated in advance with the Regulatory Authority.

4.6 Changes in parameters C, *m* as a result of thermal ageing and radiation damage are not considered.

4.7 Retardation (acceleration) of crack growth rate as a result of overloading or as a result of crack penetration through transition zones of welding joints or cladding is not considered in calculations.

4.8 In the course of crack growth, the dependence of shape factors Y (Appendix IV) on a/c, a/s, β, γ is determined.

4.9 A simplified conservative procedure for crack growth calculation may be applied as follows:

4.9.1. First estimation of crack growth with estimated final sizes a_e, c_e should be made.

4.9.2. Stress intensity factor ranges, ΔK_I, corresponding to fixed crack sizes a_e, c_e, should be calculated for individual types of cycles.

4.9.3. The crack growth calculation for these ΔK_I should be performed according to Sections

4.2.–4.8. starting with crack sizes a_0, c_0, and ending with crack sizes a_f, c_f.

4.9.4. If $a_f < a_e$ and $c_f < c_e$, the resulting crack sizes a_f, c_f are conservatively calculated. If it is not the case, the first estimation sizes a_e, c_e must be increased and the procedure must be repeated.

4.10 An exact crack growth calculation may be performed using a cycle-by-cycle adjusting of ΔK_I.

4.11 If the sequence of cycles cannot be found, then cycles are put in order by dependence on the crack increment they induce, starting with the cycle causing the smallest crack increment and continuing with cycles causing larger increments.

5. Final assessment of the crack resistance against fast fracture

5.1 The schematised crack increased by the safety factor according to Chapter 3 of this Appendix, and due to crack growth according to Chapter 4 of this Appendix (with the resulting sizes a_f, c_f), is then assessed on its resistance against fast fracture using the same procedure as used for "postulated defect" in Chapter 5 of the main part of this Procedure.

5.2 The only difference is that instead of using postulated crack sizes according to Chapter 5.7 of the main part of this Procedure, the crack with sizes a_f, c_f and with the actual position of its centre is assessed.

5.3 The resistance against fast fracture of this crack should be assessed for normal operating conditions, hydrotests, anticipated operational transients, and emergency conditions as prescribed in Chapter 5 of the main part of the Procedure. The resulting [p]-[T] curves, hydrotest temperatures [T_h], and maximum allowable transition temperature [T_t] (or maximum allowable transition temperature for crack arrest [T_A]) are assessed similarly as in Chapter 5 to predict the vessel residual lifetime with the flaw found in its wall.

5.4 No new assessment is necessary if the schematised crack dimensions increased by both the safety factor and due to crack growth (with the resulting sizes a_f, c_f) are "covered" by dimensions of any crack postulated in the original vessel integrity assessment performed according to Chapter 5, (i.e. sizes of the postulated crack are larger than a_f, c_f and the aspect ratio a_f/c_f is within the range of aspect ratios used in the original assessment), provided that the integrity assessment result was satisfactory from the point of view of the component lifetime.

6. Criteria for decision on repair of flaw

6.1 The flaws not meeting the allowable sizes of the flaws in accordance with Appendix XI have to be assessed numerically using the procedure given in this Appendix.

6.2 As long as the dimensions of detected flaw are such that the flaw is admissible (taking into account its possible growth due to static, cyclic thermal-mechanical or corrosion-mechanical loading, incl. appropriate safety factors) for period until the end of design lifetime of the appropriate component, the component may be allowed for continued operation without any further limitations. All additional in-service inspections have to be adapted in a way what will include the flaw in all future inspection programmes. If the assessed growth of the flaw is bigger than the growth already included in the calculation of acceptability, then it is necessary to repeat the assessment in accordance with this Appendix.

6.3 As long as the dimensions of detected flaw are such that the flaw is admissible (taking into account its possible growth due to static, cyclic thermal-mechanical or corrosion-mechanical loading, incl. appropriate safety factors) for period shorter than period until the end of design lifetime of the appropriate component, the component may be further operated only if certain conditions are met. In this case, the period between two subsequent service inspections must be adjusted. The aim of the adjustment is to ensure that possibly omitting or underestimation of the flaw growth size will not have as a consequence exceeding its critical size a_c with the safety factor $n_c = 2$.

The maximum allowable period of operation with such conditionally allowable flaw is the time interval lasting until the in-service inspection that most closely precedes the end of period for which the flaw is predicted to be conditionally allowable. If it is proven during this in-service inspection that the flaw is allowable until the next in-service inspection, the analogous assessment procedure is to be performed for the next period (until the next inspection). However, if the trends in both the flaw growth and material degradation confirm the results of primarily executed assessment (i.e. that the flaw is not allowable for the whole design lifetime of the appropriate component), then it is necessary to adopt a decision on flaw repair, and to perform the flaw repair with sufficient time reserve (with respect to limited lifetime of the flawed component).

6.4 If a conditionally allowable flaw is detected, then an alternative procedure may be applied: This procedure consists in lowering the degradation rate of material properties by methods of ageing management (if possible) and, consequently, in prolongation of the lifetime of the assessed flawed component.

6.5 In the case that assessment indicates that the flaw is not allowable until the next planned in-service inspection, it is necessary to use the following procedure:

6.5.1 Shorten the period until the next in-service inspection and locate the next inspection at the time for which the flaw is still allowable.

6.5.2 Prepare the technology and repair procedure for the last shutdown for which the flaw is still allowable.

6.5.3 Carry out the assessment of influence of flaw repair on the component lifetime.

6.5.4 Include the repaired site into the programmes of all future in-service inspections.

6.6 To repair the inaccessible flaw, a special provision has to be elaborated that is based on the certified technology of the repair.

6.7 The technological procedure for the repair has to be agreed upon by the owner of the component and approved by the State authorised supervising body.

APPENDIX XIII

Assessment of acceptability of flaws in austenitic piping

1. List of symbols

ϑ	Angle crack length, [rad]
β	Angle of neutral axis, [rad]
σ_b	Primary bending stress, [Pa]
$\sigma_{b,lim}$	Reference limit load bending stress, [Pa]
σ_f	Flow stress, [Pa]
σ_m	Primary membrane stress, [Pa]
σ_o	Hoop stress, [Pa]
$\sigma_{o,lim}$	Reference limit load hoop stress, [Pa]
a	Flaw modelling ellipse minor axis, [m]
a_c	Computational flaw depth, [m]
c	Flaw modelling ellipse major axis, [m]
D_o	Pipe outer diameter, [mm]
M	Auxiliary parameter
n_σ	Safety margin
p	internal pressure, [Pa]
R	Pipe mean radius, [m]
R_e	Yield strength, [Pa]
R_m	Ultimate strength, [Pa]
s	Pipe wall thickness, [m]
Z	Load multiplier
ℓ	Axial crack length, [m]
ℓ_{crit}	Critical flaw length for stability of an axial through wall flaw, [m]

2. Introduction

The flaws located in the straight part of the austenitic piping can be assessed using the equations for flawed pipe plastic collapse evaluation. This procedure can be used without any adjustment for the materials with high resistance against flaw propagation (with high toughness); in the weldment, a correction factor shall be used.

In the plants of WWER design the flaw occurrence means that the flaw originates either from manufacture or some mechanism not considered in the design occurred during the operation. It shall be decided whether the flaw originates from manufacture or whether the mechanism of flaw creation occurred during the operation. The mechanism of flaw creation shall be identified and removed. Mechanisms can be thermal fatigue due to leaking valve, fatigue due to extra vibration etc. Any acceptable flaw shall be checked during the next outage to prove there is no increment. Flaw growing during the operation is not acceptable.

3. Schematisation of flaws

In principle, schematisation of flaws is to be carried out according to Appendix X. In all cases, it can be treated as a conversion to the surface flaw. If the elliptical flaw is located neither in the axial nor in the circumferential plane, it has to be converted by the corresponding projection into two flaws lying in those planes. The plane is taken as the axial one if the pipe axis lies in this plane. The circumferential plane is perpendicular to the pipe axis. The approximate flaw boundaries are indicated for the assessment needs dealt with in this Appendix in accordance with the following formulae

$$a_c = a,\ \vartheta = c/R \qquad \text{(XIII.1)}$$

for circumferential flaw,

$$a_c = a,\ \ell = 2c \qquad \text{(XIII.2)}$$

for axial flaw.

4. Determination of acceptermination of acceptability of circumferential flaw

The values of the simple membrane stress σ_m, the simple bending stress σ_b, and the thermal expansion stress determined in the location of the flaw are to enter into the assessment. The flow stress is:

$$\sigma_f = (R_m + R_e)/2 \qquad \text{(XIII.3)}$$

It is assumed for this assessment that the pipe failure mechanism is plastic collapse. The bending stress causing such a collapse is to be determined from the equation:

$$\sigma_{b,\lim} = \frac{2\sigma_f}{\pi}\left(2\sin\beta - \frac{a_c}{s}\sin\vartheta\right), \qquad \text{(XIII.4)}$$

where

$$\beta = \frac{1}{2}\left(\pi - \vartheta\frac{a_c}{s} - \pi\frac{\sigma_m}{\sigma_f}\right); \qquad \text{(XIII.5)}$$

if $\vartheta + \beta > \pi$, then

$$\sigma_{b,\lim} = \frac{2\sigma_f}{\pi}\left(2 - \frac{a_c}{s}\right)\cdot \sin\beta\,, \tag{XIII.6}$$

where

$$\beta = \pi \cdot \frac{1 - \frac{a_c}{s} - \frac{\sigma_m}{\sigma_f}}{2 - \frac{a_c}{s}} \tag{XIII.7}$$

To determine acceptability of the flaw located in the base austenitic material, the following criterion is to be used:

$$\sigma_{b,\lim} > n_\sigma(\sigma_b + \sigma_m) - \sigma_m \tag{XIII.8}$$

where $n_\sigma = 2.77$ for normal operating conditions without hydrotest and $n_\sigma = 1.39$ for hydrotest, emergency and faulted conditions.

To determine acceptability of the flaws located in a weld of the austenitic piping, the following criterion is to be used:

$$\sigma_{b,\lim} > Z \cdot n_\sigma\left(\sigma_b + \sigma_m + \frac{\sigma_t}{n_\sigma}\right) - \sigma_m \tag{XIII.9}$$

The values of the factor Z suitable for the assessment of an austenitic weld are to be determined for *D0* in mm from the following equations:

$$Z = 1.15 \cdot \left[1 + 0.013 \cdot \left(\frac{D_0}{25.4} - 4\right)\right] \text{for shielded arc welding,} \tag{XIII.10}$$

$$Z = 1.30 \cdot \left(1 + 0.010 \cdot \left(\frac{D_0}{25.4} - 4\right)\right) \text{for submerged arc welding.} \tag{XIII.11}$$

5. Assessment of acceptability of axial flaw

The value of the hoop stress σ_o enters into the assessment. The flow stress is evaluated according to (3).

The allowable depth of the axial flaw for given flaw length ℓ can be derived from the following formula:

$$\sigma_o = \frac{\sigma_f}{n_\sigma} \cdot \left(\frac{1 - \frac{a_c}{s}}{1 - \frac{a_c}{sM}}\right), \tag{XIII.12}$$

where

$$\sigma_o = \frac{pR}{s},$$

$$M = \sqrt{1 + \frac{1.61 \cdot \ell^2}{4 \cdot R \cdot s}} \tag{XIII.13}$$

and $n_\sigma = 3$ for normal operating conditions without hydrotest and $n_\sigma = 1.5$ for hydrotest, emergency and faulted conditions.

The criterion of applicability is given by the inequality $a_c < 0.75\ s$ and $\ell < \ell_{crit}$, where

$$\ell_{crit} = 1.58\sqrt{R \cdot s} \cdot \sqrt{\left(\frac{\sigma_f}{\sigma_o}\right)^2 - 1} \tag{XIII.14}$$

APPENDIX XIV

Computational assessment of acceptability of flaws in ferritic steel piping

1. List of symbols

α	$(a/s)(a/\ell)$
β	Angle to neutral axis of flawed pipe, [Rad]
$\sigma_{b,lim}$	Reference limit load bending stress, [MPa]
σ_{φ}	Hoop stress in pipe at the flaw, [MPa]
R_e	Yield strength, [MPa]
$\sigma_{\varphi,lim}$	Reference limit load hoop stress, [MPa]
n_{σ}	Safety coefficient
2ϑ	Final length of the flaw, [Rad]
a	Computational depth of the flaw, [m]
A	Auxiliary variable for calculation of the coefficient Z
D	outer diameter of the pipe, [m]
D_{nom}	Nominal diameter of the pipe, [mm]
E'	$E' = E / (1 - \mu)$, μ is Poisson constant, [MPa]
F	Auxiliary parameter for calculation of the stress intensity factor of axial flaw
F_b	Auxiliary parameter for calculation of K_{Ib} for the circumferential flaw
F_m	Auxiliary parameter for calculation of K_{Im} for the circumferential flaw
J_{Ic}	Material toughness, [J/m^2]
K_I	Mode I of the stress intensity factor, [MPa·m$^{0.5}$]
K_{Ib}	Mode I of the stress intensity factor intensity for the bending load, [MPa·m$^{0.5}$]
K_{Im}	Mode I of the stress intensity factor for the membrane load, [MPa·m$^{0.5}$]
K_{Ir}	Mode I of the intensity factor for the residual stress, [MPa·m$^{0.5}$]
K_r'	Numerator in the screening criterion for fracture mechanism
ℓ	Length of the computational flaw, [m]
ℓ_c	Critical flaw length for stability of an axial through wall flaw, [m]
M	Bending moment, [Nm]
M_2	Auxiliary parameter
n_{σ}	Safety margin
$F_{x,tot}$	Total axial force including the load caused by the internal pressure, [N]
p	Internal pressure, [MPa]
σ_b	Primary bending stress, [MPa]
$\sigma_{b,lim}$	Bending stress at limit load for any combination of primary and expansion stresses, [MPa]
σ_K	Expansion stress, [MPa]

σ_m	Primary membrane stress, [MPa]
Q	Shape correction factor
R	Mean radius of the pipe, [m]
$[\sigma_b]$	Allowable bending stress for a circumferentially flawed pipe, [MPa]
$[\sigma]$	Design stress intensity, [MPa]
S_r'	Screening criterion factor for determining analysis method
s	Wall thickness of the pipe
x	$x = a / s$
Z	Load multiplier

2. Introduction

In this Appendix, the methods of computational assessment of the flaws in ferritic steel piping are described. The computational assessment is carried out in order to determine acceptability of the flaw. The allowable flaw is the flaw enabling operation of a piping component containing the flaw during a certain period, usually until the next inspection.

In accordance with the assumed fracture mechanism, the evaluation methods used are based on:

— Assessment of the limit load for the plastic behaviour;

— Elastic-plastic fracture mechanics, if the flaw growth before reaching plastic collapse is to be assumed;

— Linear-elastic fracture mechanics for the brittle material.

The real flaws are usually of an irregular form. For assessment, the real flaws are to be substituted by computational flaws. The procedure transferring the real flaws into computational ones is called characterisation. Characterisation is dealt with in Appendix X. A flaw that lies entirely in the cladding need not be evaluated. If the flaw cannot be proved to be fully located in the cladding it should be postulated to be located in the basic material. Cladding is not considered in the flaw evaluation. Additionally in this appendix, the screening criteria for determination of fracture mechanism in dependence on temperature, loading, sizes of the flaws, and the material properties are to be defined. Assessment is based either on comparison of the flaw size with the allowable size of the flaw, or on comparison of the stress with the allowed stress for the flaw. Single computational methods are described in sections 6, 7, 8.

3. Evaluation of growth of flaws

In the plants of WWER design the flaw occurrence means that the flaw originates either from manufacture or due to some mechanism not considered in the design occurred during the operation. It shall be decided whether the flaw originates from manufacture or whether the mechanism of flaw creation occurred during the operation. The mechanism of flaw creation shall be identified and removed. Mechanisms can be thermal fatigue due to leaking valve, fatigue due to extra vibration etc. Any acceptable flaw shall be checked during the next outage to prove there is no increment. Flaw growing during the operation is not acceptable.

For some piping flow accelerated corrosion (FAC) is inherent mechanism leading to flaws in form of thinning. This is complex problem which is not solved in this Appendix.

4. Fracture toughness

Fracture toughness of the material, J_{Ic} , is necessary for a decision on the fracture mechanism identification. This should be determined experimentally for the given heat, or it can be substituted by the envelope for the given type of material.

5. Fracture mechanism screening criterion

The screening criterion on the fracture mechanism identification and on the use of the suitable computational method (plastic collapse, elastic-plastic fracture mechanics or linear-elastic fracture mechanics) is given in this subsection.

The screening criterion is to be defined by the fraction K_r'/S_r', where

$$K_r' = \sqrt{\frac{K_I^2}{E'J_{Ic}}},$$
$$E' = \frac{E}{1-\mu^2},$$
, (XIV.1)

and E is the Young modulus and μ is the Poisson number.

For the circumferential flaw, the S_r' is

$$S_r' = \frac{\sigma_b + \sigma_e}{\sigma_{b,\lim}}, \quad \text{(XIV.2)}$$

where σ_b is the primary bending stress in the pipe in the location of the flaw, and σ_e is the thermal expansion stress.

For the axial flaw

$$S_r' = \frac{pD}{2s\sigma_{\varphi,\lim}}, \quad \text{(XIV.3)}$$

where p is the internal pressure, D is the outer diameter of the pipe, s is the wall thickness of the pipe, and $\sigma_{b,lim}$ resp. $\sigma_{\varphi,lim}$ are the reference limit load bending stress resp. reference limit load hoop stress. The equations for calculation of the reference bending stresses and the equations for K_I will be given in the following Sections 5.1, 5.2.

If $K_r'/S_r' < 0.2$, then the approach is based on limit load criteria described in subsection 6 is to be used. If $0.2 \leq K_r'/S_r' < 1.8$, then the approach is based elastic-plastic fracture mechanics described in subsection 7 is to be used. If $K_r'/S_r' \geq 1.8$, then the approach is based on linear fracture mechanics described in subsection 8.

5.1 Circumferential flaw

5.1.1 Stress intensity factor

The stress intensity factor is to be determined from the equations

$$K_I = K_{\mathrm{Im}} + K_{Ib}, \tag{XIV.4}$$

where

$$K_{\mathrm{Im}} = \frac{F_{x,tot}}{2\pi Rs} \cdot \sqrt{\pi a} \cdot F_m, \tag{XIV.5}$$

$$K_{Ib} = \left(\frac{M}{\pi R^2 s} + \sigma_K\right) \cdot \sqrt{\pi a} \cdot F_b \tag{XIV.6}$$

where

$$F_m = 1{,}10 + x\left(0{,}15241 + 16{,}772\left(\frac{x\vartheta}{\pi}\right)^{0.855} - 14{,}944\left(\frac{x\vartheta}{\pi}\right)\right), \tag{XIV.7}$$

$$F_b = 1{,}10 + x\left(-0{,}09967 + 5{,}0057\left(\frac{x\vartheta}{\pi}\right)^{0.565} - 2{,}8329\left(\frac{x\vartheta}{\pi}\right)\right),$$

$$x = \frac{a}{s}, \frac{\vartheta}{\pi} \text{ is the ratio of crack depth to pipe inner circumference.}$$

$F_{x,tot}$ is the total axial force on the pipe including the pressure, M is
the applied moment, R is the mean radius of the pipe, and s is the wall thickness of the pipe.
The equations for F_m, F_b are valid under the following conditions:

$$\frac{\ell}{a} \geq 2$$

$$0{,}08 \leq x \leq 0{,}8$$

$$0{,}05 \leq \frac{\vartheta}{\pi} \leq 0{,}5; \tag{XIV.8}$$

$$\text{for} \quad 0.5 \leq \frac{\vartheta}{\pi} \leq 1.0 \quad \text{is} \quad \frac{\vartheta}{\pi} = 0.5.$$

5.1.2 Reference limit load bending stress

The reference bending stress can be determined from the following equations:

$$\sigma_{b,\lim} = \frac{2\sigma_y}{\pi}\left(2\sin\beta - \frac{a}{s}\sin\vartheta\right), \tag{XIV.9}$$

where

$$\beta = \tfrac{1}{2}\left(\pi - \frac{a}{s}\vartheta - \pi\frac{\sigma_m}{2{,}4[\sigma]}\right), \tag{XIV.10}$$

or for

$$(\vartheta + \beta) > \pi \tag{XIV.11}$$

$$\sigma_{b,\lim} = \frac{2R_e}{\pi}\left(2 - \frac{a}{s}\right)\sin\beta,$$

where

$$\beta = \frac{\pi\left(1 - \frac{a}{s} - \frac{\sigma_m}{2{,}4[\sigma]}\right)}{2 - \frac{a}{s}}. \tag{XIV.12}$$

5.2 Axial flaw

5.2.1 Stress intensity factor

The stress intensity factor is to be derived from the equations

$$K_I = \frac{pR}{st}\sqrt{\frac{\pi a}{Q}}\cdot F, \tag{XIV.13}$$

where

$$Q = 1 + 4.593\left(\frac{a}{\ell}\right)^{1.65},$$

$$F = 1.12 + 0.053\alpha + 0.0055\alpha^2 + \left(1.0 + 0.02\alpha + 0.0191\alpha^2\right)\frac{\left(20 - \frac{R}{s}\right)^2}{1400}, \tag{XIV.14}$$

$$\alpha = \frac{a}{s}\cdot\frac{a}{\ell}.$$

5.2.2 Reference limit load bending stress

The reference bending stress of the limit load can be derived from the following equations:

$$\sigma_{\varphi,\lim} = R_e \frac{1-x}{1-\dfrac{x}{M_2}}, \tag{XIV.15}$$

where

$$x = \frac{a}{s}, \qquad M_2 = \sqrt{1+\frac{1.61}{4Rs}\cdot \ell^2}. \tag{XIV.16}$$

6. Allowable flaw depth using limit load criteria

6.1 Circumferential flaw

The conditions for use of the approach described in this subsection are given in subsection 5. The allowable bending stress [σ_b] in the flawed pipe for a given end of evaluation period flaw size, for either the normal operating conditions without hydrotest and for hydrotest, emergency and faulted conditions, shall be determined using the formula below. These equations are valid for σ_b / $\sigma_m \geq 1.0$ and $\sigma_m \leq 0.5$ [σ] for normal operation conditions without hydrotest, and for $\sigma_m \leq$ [σ] for hydrotest and postulated accidents. For the circumferential flaw not penetrating into the compressive side of bend piping (such that $(\vartheta+\beta)\leq\pi$), the relation between the applied loads and flaw depth at incipient plastic collapse is given by:

$$\sigma_{b,\lim} = \frac{2\sigma_f}{\pi}\left(2\sin\beta - \frac{a}{s}\sin\vartheta\right), \tag{XIV.17}$$

where

$$\beta = \tfrac{1}{2}\left(\pi - \frac{a}{s}\vartheta - \pi\frac{\sigma_m}{\sigma_f}\right). \tag{XIV.18}$$

σ_f is the mean value of the yield point and ultimate strength, or, if those quantities are not available, σ_f = 2.4 [σ]. For longer flaws penetrating the compressive bending region, $(\vartheta+\beta)>\pi$, the relation between the applied loads and the flaw depth at incipient plastic collapse is given by:

$$\sigma_{b,\lim} = \frac{2\sigma_f}{\pi}\left(2-\frac{a}{s}\right)\sin\beta, \tag{XIV.19}$$

where

$$\beta = \frac{\pi}{2-\dfrac{a}{s}}\cdot\left(1-\frac{a}{s}-\frac{\sigma_m}{\sigma_f}\right). \tag{XIV.20}$$

The allowable bending stress [σ_b] is:

$$[\sigma_b] = \frac{\sigma_{b,\lim}}{n_\sigma} - \sigma_m\left(1 - \frac{1}{n_\sigma}\right), \tag{XIV.21}$$

where

n_σ is the safety coefficient, n_σ = 2.77 for normal operating condition without hydrotest, n_σ = 1.39 for hydrotest, emergency and faulted conditions. The limits of applicability of these equations is 0.75 ≥ a / s. These equations can be used to determine acceptability of the flaw.

6.2 Axial flaw

The allowable depth of the axial flaw for given flow length ℓ in the pipe can be derived from the following equation:

$$\sigma_\varphi = \frac{\sigma_f}{n_\sigma}\left(\frac{\frac{s}{a} - 1}{\frac{s}{a} - \frac{1}{M_2}}\right), \tag{XIV.22}$$

where

$$\sigma_f = 2.4[\sigma], \tag{XIV.23}$$

$$M_2 = \sqrt{1 + \frac{1.61 \cdot \ell^2}{4Rs}}, \tag{XIV.24}$$

$\sigma_\varphi = \frac{pR}{s}$ (25)

n_σ is the safety coefficient, n_σ = 3.0 for normal operating conditions without hydrotest, n_σ = 1.5 for hydrotest, emergency and faulted conditions.

The limits of applicability of this equation are 0.75 ≥ a / s and $\ell < \ell_c$, where

$$\ell_c = 1.58 \cdot \sqrt{Rs}\sqrt{\left(\frac{\sigma_f}{\sigma_\varphi}\right)^2 - 1} \tag{XIV.25}$$

7. The allowable depth of the flaw determined on the basis of elastic-plastic fracture mechanics

The procedure for determination of the critical flaw depth for flawed piping meeting the elastic plastic fracture mechanic criteria is described in this subsection. The conditions for the use of the procedure described in this subsection are displayed in subsection 5.

7.1 Circumferential flaw

The allowable bending stress [σ_b] in the flawed pipe shall be determined using:

$$[\sigma_b] = \frac{1}{n_\sigma}\left(\frac{\sigma_{b,\lim}}{Z} - \sigma_K\right) - \sigma_m\left(1 - \frac{1}{Z\,n_\sigma}\right), \tag{XIV.26}$$

where

$\sigma_{b,lim}$ is the bending stress of incipient plastic collapse determined according to Section 6.1, Z is the load multiplier, n_σ is the safety coefficient, n_σ = 2.77 for the normal operation conditions without hydrotest, n_σ = 1.39 for hydrotest, emergency and faulted conditions. Equation XIV.26 is valid assuming 0.75 ≥ a / s.

The values of factor Z are as follows:

$$Z = 1.35\left(1 + 0.0184 \cdot A \cdot \left(\frac{D_{nom}}{25.4} - 4\right)\right) \quad \text{(XIV.27)}$$

The parameter is to be determined from the equations:

$$A = \sqrt[4]{0{,}125\frac{R}{s} - 0{,}25} \quad \text{for} \quad 5 \le \frac{R}{s} \le 10 \quad \text{(XIV.28)}$$

(XIV.29)

and

$$A = \sqrt[4]{0{,}4\frac{R}{s} - 3{,}0} \quad \text{for} \quad 10 < \frac{R}{s} \le 20.$$

8. Allowable depth of the flaw determined on the basis of linear fracture mechanics

The conditions for use of the procedure described in this subchapter are given in subsection 5.The allowable depths of the flaws are to be derived solving the equation

$$K_I(a) = \sqrt{J_{Ic}E'} \quad \text{(XIV.30)}$$

The depths are dependent on the stress intensity factor K_I, what is solved for the circumferential flaw in Section 8.1 and for the axial flaw in Section 8.2.

Equation XIV.31 can be rewritten as equivalent criteria in terms of the stress intensity factor:

$$K_I(a) \le \sqrt{J_{Ic}E'} \quad \text{(XIV.31)}$$

8.1 Circumferential flaw

The stress intensity factor for the circumferential flaw is:

$$K_I = K_{\mathrm{Im}} + K_{Ib} + K_{Ir}, \quad \text{(XIV.32)}$$

where

$$K_{\mathrm{Im}} = n_\sigma \frac{F_{x,tot}}{2\pi Rt}\sqrt{\pi \cdot a} \cdot F_m, \quad \text{(XIV.33)}$$

$$K_{Ib} = \left(\frac{n_\sigma M}{\pi R^2 s} + \sigma_K\right)\sqrt{\pi \cdot a} \cdot F_b. \quad \text{(34)}$$

n_σ is the safety coefficient, n_σ = 2.77 pro the normal operating conditions without hydrotest, n_σ = 1.39 for hydrotest, emergency and faulted conditions. The influence of residual stresses intensity factor K_{Ir} has to be included in the assessment. The safety coefficient l is sufficient for the residual stresses. Other symbols are explained in the Section 1 of this Appendix.

8.2 Axial flaw

Stress intensity factor for an axial flaw is:

$$K_I = K_{\mathrm{Im}} + K_{Ir},$$

where (XIV.34)

$$K_{\mathrm{Im}} = n_\sigma \frac{pR}{t}\sqrt{\frac{\pi \cdot a}{Q}} \cdot F. \quad \text{(XIV.35)}$$

n_σ is the safety coefficient, n_σ = 3.0 for the normal operating condition without hydrotest, n_σ = 1.5 for hydrotest, emergency and faulted conditions. Other symbols are defined in 5.1.

APPENDIX XV

Material properties to be used for temperature and stress fields calculations within the assessment of reactor pressure vessel resistance against fast fracture

1. List of symbols

T_{sf}	Stress-free temperature, [°C]
T_{ref}	Reference temperature used for thermal expansion coefficient measurement, [°C]
T	Temperature, [°C]
E	Young modulus, [MPa]
ν	Poisson ratio
α_{ref}	Thermal expansion coefficient established for T_{ref}, [K^{-1}]
α_{ref}^{true}	Thermal expansion coefficient established for T_{ref} related to true strains, [K^{-1}]
α_{ref}^{eng}	Thermal expansion coefficient established for T_{ref} related to engineering strains, [K^{-1}]
α_0	Thermal expansion coefficient corrected to T_{sf}, [K^{-1}]
λ	Thermal conductivity, [$Wm^{-1}K^{-1}$]
c_p	Specific heat, [$Jkg^{-1}K^{-1}$]
ρ	Density, [kgm^{-3}]

2. Formulae for thermal expansion coefficient correction

One of the following formulae for thermal expansion coefficient correction should be used in the case when the FEM code used for mechanical calculations does not correct it automatically to stress-free-temperature T_{sf} (the stress-free-temperature T_{sf} may be different from reference temperature T_{ref} used for thermal expansion coefficient measurement):

$$\alpha_0(T) = \frac{\alpha_{ref}(T)\cdot(T - T_{ref}) - \alpha_{ref}(T_{sf})\cdot(T_{sf} - T_{ref})}{(T - T_{sf})\cdot[1 + \alpha_{ref}(T_{sf})\cdot(T_{sf} - T_{ref})]}\ , \qquad \text{(XV.1)}$$

$$\alpha_0(T) = \frac{\alpha_{ref}(T)\cdot(T - T_{ref}) - \alpha_{ref}(T_{sf})\cdot(T_{sf} - T_{ref})}{(T - T_{sf})\cdot} \qquad \text{(XV.2)}$$

The formula (XV.1) is valid, when engineering strains are used within the applied FEM code for thermal load treatment. It is usually when linear-elastic problem with small strains and displacements is solved. In this case, the input α_{ref} data shall be based on engineering strains.

The formula (XV.2) is valid, when true strains are used within the applied FEM code for thermal load treatment. It is usually when elastic-plastic problem with large strains and displacements is solved. In this case, the input α_{ref} data shall be based on true strains.

Since the measured α_{ref} data are usually based on the engineering strains, they must be first transformed into true strain data according to formula:

$$\alpha_{ref}^{true}(T) = \frac{ln(1+\alpha_{ref}^{eng}(T)(T-T_{ref}))}{(T-T_{ref})} \tag{XV.3}$$

The effect of using true vs. engineering strain is small in comparison to the effect of α_{ref} to α_0 correction.

3. Material properties for WWER 440 reactor pressure vessel

Base material: 15Kh2MFA (15Kh2MFAA in core region)

Weld metal: Sv-10KhMFT

Cladding 1st layer: Sv-07Kh25N13

Cladding 2nd layer (surface): Sv-08Kh19N10G2B

Stress-free temperature T_{sf}=267 ºC.

T_{ref} = 20 ºC

TABLE XV.1 Thermal-physical properties

Material	***T***	***E***	α_{ref}	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 (both layers)	20	165	-	15.9	0.3	15.1	449	7900
	100	160	14.6	16.5	0.3	16.3	480	7868
	200	153	15.7	16.5	0.3	17.6	519	7830
	300	146	16.0	16.8	0.3	18.8	559	7790

Base material: 15Kh2NMFA (15Kh2NMFAA in core region)

Weld metal: Sv-12Kh2N2MA (Sv-12Kh2N2MAA in core region)

Cladding 1st layer: Sv-07Kh25N13

Cladding 2nd layer (surface): Sv-04Kh20N10G2B

Stress-free temperature T_{sf}=290 ºC.

T_{ref} = 20 ºC

TABLE XV.2 Thermal-physical properties

Material	T	E	α_{ref}	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	447	7830
	50				0.3	35.5	459	7822
	100	201	11.6	12.9	0.3	36.1	479	7809
	150				0.3	36.6	500	7795
	200	193	12.0	13.6	0.3	36.8	520	7780
	250				0.3	36.6	541	7765
	300	183	12.6	14.2	0.3	36.2	562	7750
	350	177.5			0.3	35.6	585	7733
Cladding (both layers)	20	165		16.6	0.3	13.2	449	7900
	50				0.3	13.5	460	7889
	100	160	15.7	17.0	0.3	14.4	480	7870
	150				0.3	15.3	500	7851
	200	153	16.1	17.6	0.3	16.4	519	7830
	250				0.3	17.5	539	7809
	300	146	16.7	18.2	0.3	18.4	559	7788
	350	142			0.3	19.6	579	7766

APPENDIX XVI

Procedure for probabilistic evaluation of RPV resistance against fast fracture

1. List of abbreviations and symbols

FEM	Finite Element Method
MC	Master Curve
NPP	Nuclear Power Plant
PTS	Pressurized Thermal Shock
RPV	Reactor Pressure Vessel
TH	Thermal hydraulic calculation
CPI	Conditional probability of initiation (of fast fracture)
$cpi(\tau)$	Conditional probability of initiation of fast fracture at time τ
CPF	Conditional probability of RPV failure
FI	Frequency of initiation of fast fracture
FF	Frequency of RPV failure
fr_j	Frequency of occurrence of group of PTS scenarios
D	Crack depth (crack dimension in direction of wall thickness), [m]
$D_{rel,sh}$	Relative (in % of wall thickness) shifted crack depth
$R_{p0,2}$	Tensile yield strength, [MPa]
K_I	Stress intensity factor, [MPa.m^{0,5}]
K_{IC}	Fracture toughness, [MPa.m^{0,5}]
K_{Ia}	Fracture toughness for crack arrest, [MPa.m^{0,5}]
L	Crack length, [m]
M	Number of flaws found in a particular material volume (statistical data-point)
N	Number of flaws in a particular material volume (random variable), [m^{-3}]
N_f	Number of cases in which the examined phenomenon occurs, within Monte Carlo method
N_{sim}	Number of all cases (all simulations), within the Monte Carlo method
P_f	Probability of the phenomenon occurrence, within the Monte Carlo method
T_k	Critical temperature of brittleness, [°C]
T_0	Master Curve reference temperature, [°C]
T_{KIa}	Reference temperature for crack arrest, [°C]
τ	Time, [s]
pdf	Probabilistic distribution function
cdf	Cumulative distribution function, $0 \le cdf \le 1$
ccdf	Complementary cumulative distribution function, ccdf = 1 – cdf
$ccdf_{exp}(D; \beta_D)$	Complementary cumulative distribution function for exponential distribution of random variable D with parameter β_D
s	RPV wall thickness, incl. cladding, [m]
β	Parameter in the exponential distribution density function of a form $\beta\ exp(-\beta x)$
β_D	Parameter in the density function of the exponential distribution of flaw depths D (the density function is of the form $\beta_D\ exp(-\beta_D\ D)$)

2. Groups of scenarios for probabilistic evaluation of RPV resistance against fast fracture

Within **PSA** (Probabilistic Safety Assessment) , the event trees shall be developed for all initiating events potentially occurring in NPP that may lead to pressurized thermal shock (PTS) according to precisely defined criteria. It is necessary that the events included into these event trees cover at least the groups of scenarios summarized in Appendix VI.

The identified scenarios have to be aggregated into different **groups of similar variations of TH parameters**, and from each group a **representative** has to be selected, having the worst impacts on RPV integrity from point of view of PTS. The representative may be selected from scenarios analysed previously within the deterministic evaluation of RPV resistance against fast fracture, provided that the evaluation was performed and under condition that the conservativeness of the selected scenario in the frame of the group is assured.

Using procedures of PSA, for each group the **frequency of occurrence of individual PTS scenarios within the group** will be determined, and (as a sum) **the frequency of occurrence of the whole group, fr_j,** will be established, including uncertainties (statistical distributions). The entire process of producing and application of the PTS-PSA results is iterative in dependence on results of other disciplines (e.g. thermal hydraulic analyses or deterministic structural analyses).

For all representatives selected, the **conditional probabilities of fast fracture initiation, CPI_j,** are determined. For less conservative evaluation, **conditional probabilities of RPV failure**, **CPF_j,** may be determined (i.e. conditional probabilities of crack propagation through RPV wall), provided that through-wall crack propagation after crack initiation as well as possible arrest of the crack are considered. Under conditional probability of fast fracture initiation and RPV failure is understood probability of fast fracture initiation and RPV failure, respectively, under condition that the particular scenario occurred. Requirements for determination of CPI_j and CPF_j, respectively, are summarized in the following chapters.

Combining the frequencies of groups fr_j with corresponding conditional probabilities CPI_j and CPF_j, we obtain (unconditional) **frequencies of fast fracture initiation and RPV failure**, respectively, for **particular scenario groups FI_j and FF_j,** respectively. Summing them up, we obtain the final **frequencies of fast fracture initiation FI and RPV failure, FF**, respectively. Both group occurrence frequencies fr_j and conditional probabilities CPI_j and CPF_j, respectively, are statistical distributions, therefore the final frequencies of fast fracture initiation and RPV failure, respectively, are statistical distributions as well. For final evaluation (comparison with acceptance criterion, see Chapter 5.11 of the main part of this Guidelines) their mean values should be used.

A simplified basic scheme of the probabilistic evaluation is seen in Figure XVI.1.

3. Deterministic part of the evaluation

3.1 Thermal hydraulic analyses

Thermal hydraulic analyses for selected scenarios – representatives of the groups – are performed in compliance with Appendix VI. Results of thermal hydraulic calculations performed for deterministic evaluation of RPV resistance against fast fracture may be used, provided that the evaluation was performed and its results are available. For certain groups that were not included into deterministic analyses or, if they had to be represented by too conservative scenario analysed in deterministic analyses, it is necessary to select a scenario not analysed so far, and to perform new thermal hydraulic calculations for it.

In compliance with Appendix VI, **system thermal hydraulic analyses** for NPP as a whole are performed first, and then **detailed mixing analyses** for cold leg and reactor downcomer are conducted. The following data, resulting from thermal hydraulic analyses, serve as input data for temperature and stress fields calculation performed subsequently: pressure in downcomer, temperatures of medium in downcomer, and heat transfer coefficients between medium and RPV wall in downcomer (instead of two last quantities, temperature of the RPV inner wall surface may be calculated), in all cases time variations of the quantities are needed.

For the purpose of RPV probabilistic evaluation, it is possible to perform, as a simplified approach, the system thermal hydraulic analyses only, without the subsequent mixing analyses, i.e. without detailed modelling of cold plums (cold sectors, strips).

3.2 Calculations of temperature and stress fields

For data obtained from deterministic thermal hydraulic analyses performed for the selected scenario, the **temperature and stress fields in the RPV wall** are calculated in deterministic manner in compliance with Chapter 5.2 of Main Part of this Guidelines, i.e. taking also into account the residual stresses in both weld and cladding in accord with Section 5.2.2, and including the effect of austenitic cladding. Material properties have to be selected in compliance with Appendix XV. For the purpose of probabilistic RPV evaluation, solution of linear thermal problem is sufficient, with material properties independent of temperature. In that case, material properties have to be applied that correspond approximately to mean temperature of the medium at the transient process. The subsequent mechanical problem may be solved as a linear elastic problem. For calculations of temperature and stress fields, it is recommended to use FE method.

For the purpose of probabilistic RPV evaluation, temperature and stress field calculations may be performed by FE method with using **one-dimensional axi-symmetric elements.** In this case axi-symmetric cooling of downcomer is modelled (with both temperature and heat transfer coefficient constant along the downcomer circumference), i.e. effect of cold plums (cold sectors or cold strips) is not taken into account. If, in this case, data from thermal hydraulic calculations for more locations along the downcomer circumference are available (either from system TH calculations using 2D or pseudo-2D model of the downcomer, or from the mixing analyses), conservative values of both temperature and heat transfer coefficient should be used, i.e. minimum values of temperature and maximum values of heat transfer coefficient (along downcomer circumference).

3.3 Calculations of stress intensity factor

To determine stress intensity factors for randomly generated cracks (see Chapter 3.8), it is sufficient to use analytical methods based on stresses calculated on models without crack. For determination of stress intensity factors, it is possible to use Appendix IV. Detailed calculations on models with cracks included cannot be performed in majority cases due to high number of randomly generated cracks of different sizes, shapes and locations.

4. Probabilistic part of the evaluation

This chapter deals with those parameters entering the calculation that have to be considered as stochastic and are expressed via statistic distributions in the calculation.

Note. Also some parameters entering the deterministic part of evaluation are of stochastic character (e.g. some of the boundary conditions of thermal hydraulic calculations, or RPV material properties entering both temperature and stress fields calculations). However, using of their conservative values is sufficient for the purpose of deterministic part of the evaluation.

Within these distributions, a random selection of values of the particular parameters is performed, and using simulation method (e.g. Monte Carlo method), the probability of fast fracture initiation may be determined.

Instead of simulation methods also analytical methods like First Order Reliability Method (FORM) or Second Order Reliability Method (SORM) may be used for calculation of probability of fast fracture initiation.

4.1 Monte Carlo method

To determine conditional probability of fast fracture initiation or conditional probability of RPV failure, the **Monte Carlo method is recommended**. Using of other methods is possible.

In general, under Monte Carlo method an arbitrary technique using random generating of data is understood that serves for numerical determination of quantities that cannot be easily established analytically. The quantities that are of stochastic nature may take on different values under the same conditions and, moreover, these values are random and cannot be predicted precisely, and it is only possible to say that they obey certain law of probability distribution. Application of the method consists in numerical simulation of the problem, i.e. in generating a sequence of random values of the input random quantity vector, and in recording whether the examined phenomenon occurred or not. The probability of the phenomenon concerned may be estimated according to the formula

$$P_f = N_f / N_{sim}, \quad , \tag{XVI.1}$$

where N_f is number of cases in which the phenomenon occurred and N_{sim} is number of all cases (number of all simulations).

For the simulations, it is necessary to utilize a sufficiently reliable generator of random numbers with uniform distribution in interval (0,1).

The very determination of conditional probability of fast fracture initiation for a particular simulated reactor pressure vessel and a particular simulated crack may be performed analytically, with using Master Curve concept (see Chapter 3.9 of this Appendix).

4.2 Randomly generated RPV

Under the notion "randomly generated RPV" it is understood a set of randomly generated values of random quantities characterizing the RPV. The following quantities have to be generated as stochastic:

- **Crack depth (size);**
- **Neutron fluence;**
- **Chemical composition of RPV materials** (Cu, P contents – only for certain materials);
- **Initial value of MC reference temperature T0;**
- **Shift ΔT0 of MC reference temperature** in dependence on neutron fluence (and possibly in dependence on chemical composition of RPV material).

4.3 RPV areas entering the assessment

It is necessary to assess the entire **area adjacent to the reactor core,** where the embrittlement due to neutron irradiation is highest. The assessed area should exceed moderately the region of reactor core.

For reactors of **WWER 440** Type, it is necessary to include the following three **rings** into the assessment (these rings are manufactured from base metal and are positioned in area near to reactor core): short smooth ring (its upper part), long smooth ring (whole) and nozzle lower ring (its lower part; with respect to its position above the reactor core, it is possible to omit it from the assessment). Further, it is necessary to include also both **welds** between these rings into the assessment.

For reactors of **WWER 1000** Type, it is necessary to include the following three **rings** into the assessment (these rings are manufactured from base material and positioned in the area close to reactor core): lower smooth ring (its upper part), upper smooth ring (entire in reactor core) and support ring (its lower part). Further, it is necessary to include both **welds** between these rings into the assessment.

4.4 Neutron fluence

For the purpose of RPV probabilistic evaluation, it is possible to use **results of neutron fluence calculations performed for passed operation periods** of the unit and to extrapolate them until the end of design lifetime. Neutron fluence calculations have to be performed in accord with Appendix II. In case that these calculations are not available, the design values of fluences may be used. Neutron fluences of energy higher than 0.5 MeV are considered. Fluence values dependent on three spatial variables (r, z, θ) enter the RPV probabilistic evaluations. The calculated fluences are assumed to be stochastic quantities. Usually **normal distribution** is supposed. In this case the results of neutron fluence calculations are expressed in terms of mean value and standard deviation. Other statistical distribution can be used if it is more adequate from the point of view of approach applied to fluence calculations.

4.5 Chemical composition of RPV materials

Under **chemical composition** of RPV materials is, for the purpose of RPV probabilistic evaluation, understood the **content of mass % of phosphorus and copper (P, Cu, resp. Ni, Mn, Si),** on which mainly the embrittlement of the materials due to irradiation depends. Chemical composition is assumed to be a stochastic quantity with usually **normal distribution**. Other distribution can be applied if it is more adequate, with respect to the statistical evaluation of the input data. Chemical composition of material has to be generated separately for each of the areas defined in Chapter 3.3 (particular welds, rings). Chemical composition enters the calculation only in the case that for calculation of reference temperature shift ΔT_0 (see Chapter 3.7), the dependence of this shift on chemical composition is used.

The appropriate data on chemical composition are taken from the RPV passport. Data have to be statistically treated. Mean value is determined from values relevant for each ring or weld separately. The standard deviation may be obtained by statistical treatment of data points appropriate for more rings or welds of the same type and from the same manufacturer.

4.6 Initial value of MC reference temperature T0

Within RPV probabilistic evaluation, the main parameter for characterization of material state from point of view of RPV resistance against fast fracture is **Master Curve (MC) reference temperature T_0**. Under initial value of T_0 the value of T_0 appropriate for the unirradiated material in the beginning of NPP operation is understood. It is determined based on results of static fracture toughness tests, using a procedure established in the standard ASTM [1] and in compliance with Master Curve concept, see also Appendix III. It is necessary to use the material-specific values of initial T_0 for materials of particular rings and welds.

The initial value of reference temperature T_0 is assumed to be a **stochastic quantity** with **normal distribution**. Mean values of initial T_0 are taken from the RPV passport or from results of "zero state" of the Surveillance specimens programme. The standard deviation has to be determined in compliance with Appendix III, according to the following relation:

$$\sigma = \sqrt{(\sigma_1^2 + \Delta T_M^2)} \qquad \text{(XVI.2)}$$

where σ_1 is the standard deviation describing systematic uncertainty of T_0 which is due to limited number of test specimens n

$$\sigma_1 = 18/\sqrt{n}\ (^{o}C) \qquad \text{(XVI.3)}$$

and ΔT_M is standard deviation describing random uncertainty of T_0 due to scatter of material properties; the following values may be taken

$$\Delta T_M = 16\ ^{\circ}C \qquad \text{(XVI.4)}$$

Plant specific ΔT_M may be used, based on statistical treatment of initial T_0-data determined in the frame of Qualification Tests or in the frame of tests performed on specimens made from a set of identical semi-products established during production of the component by the identical technology.

In the case, when no information on T_0 for the assessed RPV is available, assessment based on correlation between T_k and T_0 can be applied, in the case when this correlation is well established. This correlation has to be taken into account as a statistical distribution.

4.7 Reference temperature shift ΔT_0

Material property degradation due to neutron irradiation is expressed through a shift of **Master Curve reference temperature *T_0***. The shift is usually determined in dependence on neutron fluence and chemical composition of the material. When determining the dependence, it is possible to use Surveillance specimens programmes results treated according to Appendix III.

If sufficient number of surveillance specimens tests is not available, formulae in chapter 3.1.8 of Appendix III (which are with regard to Chapter 3.2 applicable also to T_0) may be used.

In this case, the following relation should be applied:

$$\Delta T_0 = 1.1 \cdot \Delta T_k \qquad \text{(XVI.5)}$$

The shift ΔT_0 is a stochastic quantity. If its distribution is not determined based on statistical treatment of results of tests performed within the Surveillance specimens programmes, normal distribution is assumed and standard deviation values according to Appendix III may be used.

4.8 Randomly generated cracks

For a randomly generated RPV (see Section 3.2), it is necessary to **generate randomly the cracks**. This random generating is performed based on statistical distributions of particular crack parameters that have to be derived from statistical treatment of data obtained for flaws detected in the evaluated RPV. These data may be obtained from manufacturing, pre-operational or in-service inspections. It is necessary that the appropriate non-destructive methods used within the inspections were capable with sufficient accuracy to both indicate the flaws and identify their position and size. It is recommended that the appropriate non-destructive methods of the inspections were qualified.

To these parameters belongs mainly: Flaw density (i.e., number of cracks in a particular material volume), flaw depth (that is in correlation with flaw density), flaw shape (expressed through a ratio or difference of the semi-axes of the crack replacing ellipsis or semi-ellipsis), and flaw position (distribution along the wall thickness).

If for the evaluated RPV a sufficient amount of flaw data obtained from inspections is not available, a statistical treatment of flaws appropriate for another RPV (or RPVs) of the same type and of the same manufacturer may be performed.

4.8.1 Types of cracks entering the evaluation

For RPVs of WWER-440 and WWER-1000 Types, the following crack types have to be included into the RPV probabilistic evaluation:

(a) Underclad cracks

Underclad cracks are generated in the base metal (not weld) of RPV wall and are positioned precisely under the cladding. They are assumed to be only axial, i.e. oriented in the direction perpendicular to the direction of cladding procedure. The underclad cracks may be generated as semielliptical (with main axis in the base material - cladding interface) or elliptical (touching the cladding in the minor vertex of the ellipsis). In case of semielliptical crack both the deepest point of the crack and the point in the base material - cladding interface have to be evaluated. In case of elliptical crack, it is sufficient to evaluate the point in the base material - cladding interface.

(b) Embedded cracks

Inner (embedded) cracks are those that lie fully inside the RPV wall and do not touch either intersect cladding.

i. Embedded cracks in the weld

Embedded cracks in the weld are assumed to be **circumferential** only, due to the technology of welding. All are assumed to have an **elliptical** shape. The evaluation is performed for the point nearest to the inner surface of RPV (since in this point K_I is the highest and temperature is the lowest, in consequence of which the probability of fast fracture initiation is the highest).

ii. Embedded cracks in base metal

50 % of embedded cracks in base metal are assumed to be **circumferential** and **50 %** of them are assumed to be **axial**. All are assumed of elliptical shape. The evaluation is performed for point nearest to the inner surface of RPV.

4.8.2 Statistical distribution of crack parameters

Statistical distributions of crack parameters that are presented below are only **recommended ones.** Within statistical treatment of results of inspections also other distributions may be used, provided that it is demonstrated that they better describe the flaw data obtained.

(a) Crack density (number of cracks per volume)

Number of cracks N in a particular material volume V (for underclad cracks it is in fact number of cracks in a particular *area* V of inner RPV surface) is a stochastic quantity that is assumed to have Poisson distribution with mean ρV:

$$pdf_{Pois}(N \mid \rho V) = \frac{e^{-\rho V}(\rho V)^N}{N!} \tag{XVI.6}$$

The mean value of the parameter ρ is associated with the number of flaws M detected in a control volume V_0 through a relation $\rho = M/V_0$. In compliance with Bayes approach, the parameter ρ of the Poisson flaw density distribution is understood as a stochastic quantity and it is generated from the gamma distribution with parameters M and V_0

$$pdf_{Gamma}(\rho \mid M, V_0) = \frac{V_0^M \rho^{M-1} \exp(-V_0 M)}{(M-1)!} \tag{XVI.7}$$

The last equation is exactly valid only if it is certain that all flaws in the control volume V_0 were detected. In case of real non-destructive inspections, an appropriate correction shall be applied because the number M of flaws detected in the volume V_0 underestimates the true number of flaws in V_0.

(b) Crack depth

Depth (=through-wall dimension) of any particular crack is a stochastic quantity. It is assumed to have exponential distribution

$$pdf_{Exp}(D \mid \beta_D) = \beta_D e^{-\beta_D D} \tag{XVI.8}$$

where, in compliance with Bayes approach, the parameter β_D understood as a stochastic quantity with gamma distribution.

In an ideal case, when the non-destructive inspections can detect flaws with probability that is independent on their depth, the parameter β_D has distribution

$$pdf_{Gamma}(\beta_D \mid d, M) = \frac{d^M {\beta_D}^{M-1} \exp(-d\, M)}{(M-1)!} \tag{XVI.9}$$

where M is the number of all detected flaws, d_i are their depths and $d = \sum_{i=1}^{M} d_i$. In reality, flaws with small depths, i.e. flaws fulfilling $d_i < d_0$, are not detected. This can be taken into account by setting

$$d = \sum_{i=1}^{M} (d_i - d_0) \tag{XVI.10}$$

and increasing the generated number of flaws by replacing V_0 with $V_0\, exp(-d_0 \beta_D)$ in the gamma distribution for the crack densityρ.

(c) Crack shape

Crack shape is characterized through the semi-axis ratio L/D or the semi-axes difference $L - D$. The semi-axes difference is assumed to be a stochastic quantity with exponential distribution that is statistically independent of the crack depth D. In particular, this implies that for any simulated crack, its length L is always larger that its depth D. In compliance with Bayes approach, the parameter of the $L - D$distribution is generated from gamma distribution.

In more detail, the probability density for crack length L is

$$f(L|\beta_{LMD}, D) = \beta_{LMD}\, e^{-\beta_{LMD}(L-D)}, \quad \text{for L} \geq \text{D}$$

$$f(L|\beta_{LMD}, D) = 0, \quad \text{whenever L} < \text{D} \tag{XVI.11}$$

where

$$f(\beta_{LMD} \mid h, M) = \frac{h^M {\beta_D}^{M-1} \exp(-h\, M)}{(M-1)!} \tag{XVI.12}$$

where M is the number of detected flaws, l_i are the lengths of individual detected flaws, d_i are the depths of individual detected flaws, and

$$h = \sum_{i=1}^{M} (l_i - d_i). \qquad \text{(XVI.13)}$$

These formulae are valid provided that $L - D$ does not affect the probability of flaw detection during non-destructive testing.

(d) Flaw location in RPV

Distribution of embedded cracks (both in weld and base metal) **along RPV wall thickness** is assumed to be **uniform**.

Cracks whose nearest point to inner surface lies in a depth larger than 40 % of wall thickness, need not be considered. (These cracks have no influence on the calculated results, since they are located in a wall thickness area that is warmer during the PTS event, where, moreover, the thermal stresses are rather compressive than tensile, in consequence of which they do not possess the potential to initiate fast fracture.)

(e) If the above-mentioned statistical distributions cannot be constructed for the embedded cracks in the base metal due to low number of flaws detected in base metal, it is possible to use the same distributions as for embedded flaws detected in the weld, but with lower flaw density (decreased in a ratio corresponding to the ratio of number of flaws detected in unit volume of base metal and of weld).

4.9 Fracture toughness

Statistical distribution of fracture toughness depends on the temperature *T* of the material. For temperatures in the brittle region, $T \le T_{us}$, the fracture toughness is assumed to have Weibull distribution in accord with the Master Curve concept, see [1]. In the ductile (upper-shelf) region, i.e. for $T > T_{us}$, the fracture toughness (expressed in J-units) can be established based on [4] - [6] and is assumed to be normally distributed.

4.9.1 Fracture toughness in the brittle region

Master Curve approach is based on the assumptions that:

(1) Fracture toughness values at a particular temperature *T* have Weibull distribution with exponent 4; and

(2) The curve representing the dependence of fracture toughness on the difference of actual temperature *T* and the reference temperature T_0, *is* of universal shape (for given level of fracture probability).

Median of fracture toughness values (for specimens of thickness 1 inch = 25,4 mm) is assumed to be described through the following equation

$$K_{Jc}(\text{med}) = 30 + 70 \exp[0.019(T - RT_0)] \qquad \text{(XVI.14)}$$

Master Curve concept is applied in such a manner that for a randomly generated reference temperature T_0 and actual temperature of material *T* (dependent on time τ of the running PTS), value of Weibull distribution parameter K_0 is determined, corrected to crack length *L*:

$$K_0(\tau) = 20 + [11 + 77 \cdot \exp \cdot (0{,}019 \cdot (T(\tau) - RT_0))] \cdot [25/L]^{1/4} \qquad \text{(XVI.15)}$$

Based on time variation of K_I, the conditional probability of fast fracture initiation for time τ is determined (for crack *k*, vessel *l* and scenario *j*):

$$cpi_{jkl}(\tau) = 1 - \exp\left[-\left(\frac{K_I(\tau) - K_{\min}}{K_0(\tau) - K_{\min}} \right)^4 \right], \quad \text{(XVI.16)}$$

where K_{min} = 20 MPa.m$^{1/2}$.

Conditional probability of fast fracture initiation over the entire PTS event is determined as its maximum over the time interval considered:

$$CPI_{jkl} = \max_{\tau} cpi_{jkl}(\tau) \quad \text{(XVI.17)}$$

Determination of mean value of conditional probability of fast fracture initiation for the selected scenario is schematically shown in Figure XVI.2.

4.9.2 Fracture toughness in the ductile (upper-shelf) region

Median of the fracture toughness in the brittle region is given by the formula (XVI.14). After translation into J-units $[\mathrm{kJ/m^2}]$, we get

$$J_{Ic}^{tran}(T) = 1000 \frac{(1-\nu^2) K_{Jc}^2(\mathrm{med})}{E(T)}, \quad \text{(XVI.18)}$$

where *E(T)* denotes the Young modulus at temperature *T.* We define

$$T_{us} = 48.844 + 0.7985\, T_0 \quad \text{(XVI.19)}$$

$$f(T) = 1807.75\, e^{-(k_1 T + k_2)} \quad \text{(XVI.20)}$$

with k_1 = 0.01022698, k_2 = 2.793499

The mean fracture toughness in the ductile region in the J-units $[\mathrm{kJ/m^2}]$ is given by

$$J_{Ic}^{us}(T) = f(T) + J_{Ic}^{tran}(T_{us}) - f(T_{us}). \quad \text{(XVI.21)}$$

The fracture toughness in the ductile region is supposed to be normally distributed with the standard deviation

$$\sigma = 51.199\, e^{-0.0056T}. \quad \text{(XVI.22)}$$

Based on a time variation of the stress intensity factor $K_I(\tau)$ expressed as J-integral $J(\tau)$ the probability of fracture initiation for time τ, crack *k*, vessel *l* and scenario *j* is given by:

$$cpi_{jkl}(\tau) = \frac{1}{2}(1 + \mathrm{erf}\ \frac{J(\tau) - J_{Ic}^{us}(T(\tau))}{\sqrt{2}\sigma}). \quad \text{(XVI.23)}$$

Here erf(x) denotes the so-called „error function", which is defined by

$$\mathrm{erf}(x) = \frac{2}{\sqrt{\pi}} \int_0^x \exp(-t^2)\, dt. \quad \text{(XVI.24)}$$

Note: Fracture toughness need not be randomly generated within this particular part of the evaluation (as are the other stochastic quantities entering the evaluation) and either the probability of fracture need not be determined through the Monte Carlo method, but the cpi may be directly calculated according to formulae (XVI.16) or (XVI.23).

4.10 Fracture toughness for crack arrest

In cases that besides of conditional probability of fast fracture initiation also the conditional probability of RPV failure is evaluated (i.e. the possibility of crack arrest is taken into account), it is necessary to know **the fracture toughness for crack arrest K_{Ia}.** It is assumed that K_{Ia} is a stochastic quantity. Temperature dependence of K_{Ia} and its statistical distribution are assumed in accordance with [3] as follows:

— Mean value of K_{Ia} is described through the same type of exponential dependence on difference of actual temperature of the material and reference temperature, as is the median of fracture toughness (XVI.14) (Master Curve approach):

$$K_{Ia} = 30 + 70 \exp[0.019(T-RT_{KIa})] \qquad \text{(XVI.25)}$$

where T_{KIa} is reference temperature for crack arrest.

— Fracture toughness for crack arrest K_{Ia} has (at constant temperature) **lognormal distribution with relative standard deviation 18%.**

Reference temperature for crack arrest T_{KIa} is determined in relation to previously randomly sampled temperature T_0 in such a manner that it is assumed in accord with [3] that the difference T_{KIa} -T_0 *is* a random quantity with **lognormal distribution** and its mean value depends on T_0 (and on yield strength $R_{p0,2}$ as well) according to formula

$$T_{KIa} - T_0 = \exp\left(5 - \frac{T_0 + 273}{136{,}3\ °C} + \frac{R_{p0,2}}{683{,}3\ MPa}\right) \qquad \text{(XVI.26)}$$

The standard deviation is 19ºC.

4.11 Determination of conditional probability of RPV failure (taking into account the possibility of crack arrest)

Evaluation of crack arrest is performed analogously as in Section 5.10.10 of the Main part of this Guidelines with the exception of the fact that the **particular variables are generated randomly**. For each initiated flaw a specified number *NTEST* (which should be at least 20) of randomly generated "tests" for crack arrest are performed. These tests are performed essentially in a deterministic way.

After initiation, the crack is converted to surface breaking crack of infinite length. The eventual crack arrest is tested for the deepest point of the crack. The crack depth is step by step increased and the following condition is tested:

$$K_I < K_{Ia} \qquad \text{(XVI.27)}$$

When condition (27) is fulfilled, crack arrest is assumed. For the arrested crack the condition for re-initiation is tested ($K_I > K_{IC}$) at subsequent times of the scenario, etc. If the crack propagates into depth higher than 75% of wall thickness, it is considered as RPV failure. In other words, no stable crack arrest (i.e. crack arrest not followed by a re-initiation) occurs for any crack with depth lower than 75% of wall thickness.

The crack arrest, reinitiation, stable arrest and failure tests are carried out within each of *NTEST* crack propagation simulation cycles. The tests are performed only for those transient's times τ, for which the value of $cpi_{jkl}(\tau)$ increases, because only for these times the initiation of fast fracture can be expected. Increments $\Delta cpi_{jkl}(\tau)$ between those time steps are established.

The contribution $cpf_{jkl}(\tau)$ of time τ to the conditional probability of RPV failure is the product of $\Delta cpi_{jkl}(\tau)$ and the ratio of number of tests exhibiting RPV failure to total number of tests *NTEST*. The resulting conditional probability of RPV failure for *j*-th transient *k*-th flaw and *l*-th RPV simulation is then a sum of $cpf_{jkl}(\tau)$ over all contributing time steps.

4.12 Determination of the total conditional probability of fast fracture initiation and/or of total conditional probability of RPV failure

For determination of total conditional probability of fast fracture initiation CPI_{jl} for j-th scenario and l-th vessel with m randomly generated cracks the following relation is used:

$$CPI_{jl} = 1 - \prod_{k=1}^{m}(1 - CPI_{jkl}), \qquad \text{(XVI.28)}$$

where CPI_{jkl} is conditional probability of initiation of the k-th crack for j-th scenario and l-th vessel.

Determination of the total conditional probability of RPV failure is performed in an analogous manner.

4.13 Determination of (unconditional) frequency of occurrence of fast fracture initiation and/or RPV failure

Since **the frequencies of occurrence for scenarios groups fr_j are stochastic quantities as well** (their distributions are assumed to be known from the results of PSA), it is possible for each group of scenarios $j=1,...,p$ to generate randomly the frequency of occurrence of group fr_{jl} for each simulated RPV $l=1,...,n$.

(Unconditional) frequency of occurrence of fast fracture for the l-th simulated vessel is then determined as follows:

$$FI_l = \sum_{j=1}^{p} fr_{jl} \cdot CPI_{jl} \qquad \text{(XVI.29)}$$

In this manner we obtain **statistical distribution of frequency of occurrence of fast fracture initiation** (it is also a stochastic quantity). For the **final evaluation** (comparison with acceptance criteria, see Chapter 5.11 of the Main part of this Guidelines) its **mean value** or 95 % percentile is used. The mean value is calculated (analogously to (1)) as the mean of FI_l over all simulated vessels.

$$\overline{FI} = \frac{\sum_{l=1}^{n} FI_l}{n}. \qquad \text{(XVI.30)}$$

Determination of (unconditional) frequency of occurrence of RPV failure is performed in an analogous manner, using values CPF_{jl}.

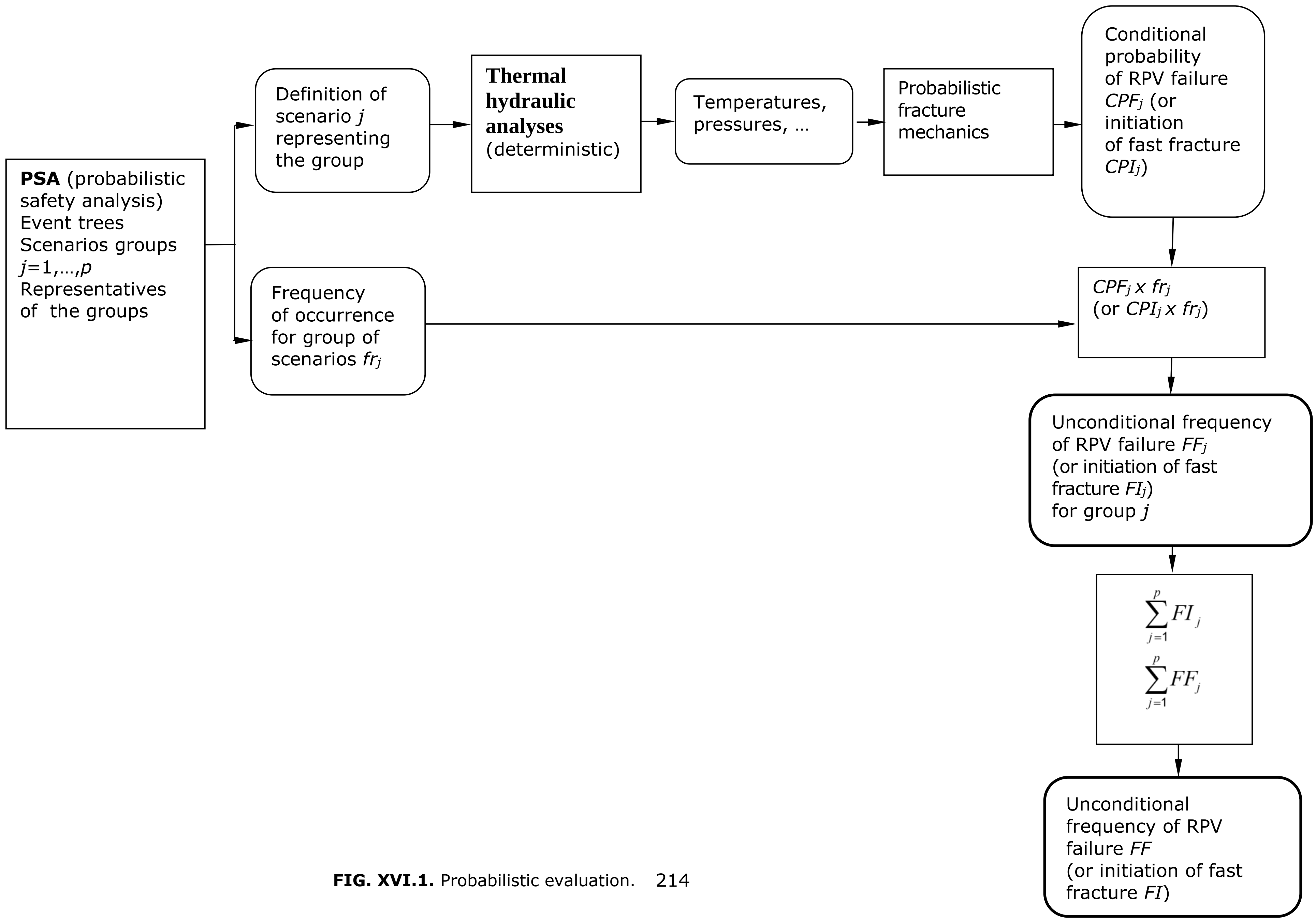
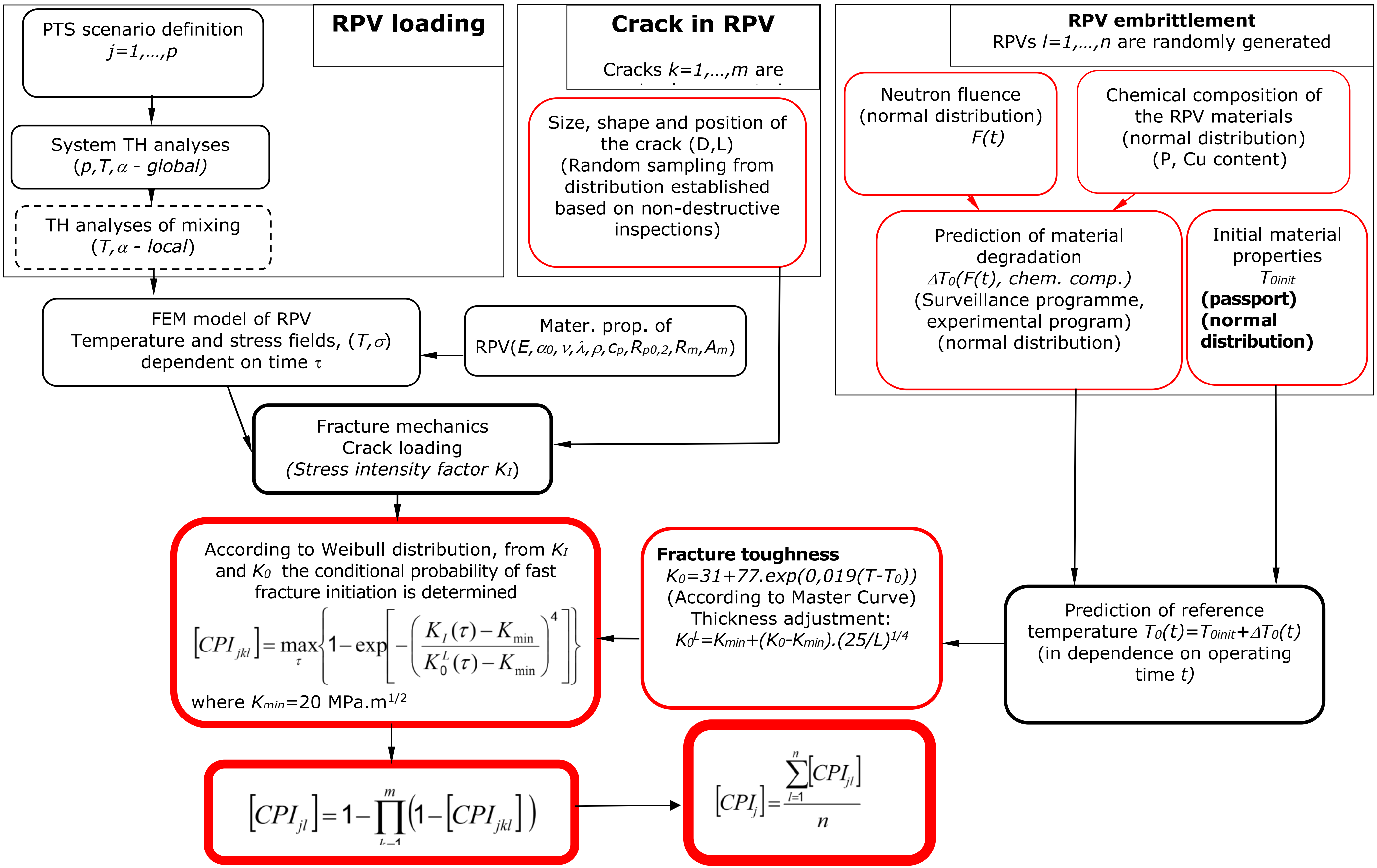


FIG. XVI.1. Probabilistic evaluation.



$$[CPI_{jkl}] = \max_{\tau}\left\{1-\exp\left[-\left(\frac{K_I(\tau)-K_{\min}}{K_0^L(\tau)-K_{\min}}\right)^4\right]\right\}$$

$$[CPI_{jl}] = 1-\prod_{k=1}^{m}\left(1-[CPI_{jkl}]\right)$$

$$[CPI_j] = \frac{\sum_{l=1}^{n}[CPI_{jl}]}{n}$$

FIG. XVI.2. Probabilistic evaluation – determination of mean value of conditional probability of fast fracture initiation for the *selected scenario*

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APPENDIX XVII

Monitoring, evaluation and prediction of flow-accelerated corrosion in nuclear power plants of WWER type

Annotation

Failures of the secondary circuits in the nuclear power plants of WWER type confirm that it is necessary to pay special attention to the main components of this part of the power plant. One of the significant degradation mechanisms with respect to the integrity and the residual lifetime of these components is the flow-accelerated corrosion (FAC). A suitable method for solution of the flow-accelerated corrosion is implementation and performance of the long-term monitoring programme.

The goal of this document is to provide an instruction and recommendation for performance of such long-term monitoring programme at pipelines potentially sensitive to flow-accelerated corrosion in WWER nuclear power plants. The aim of successful implementation of the programme is minimization of damage inflicted to important components of the secondary circuit and reduction of probability of degradation of the secondary circuit pressure pipeline systems.

1. Introduction

Failures of secondary circuits in nuclear power plants of WWER type confirm that it is necessary to pay particular attention to the main components of this part of the power plant. One of the significant degradation mechanisms with respect to the integrity and the residual lifetime of these components is the flow-accelerated corrosion (FAC). A suitable method for solution of the flow-accelerated corrosion is implementation and performance of the long-term monitoring programme.

The goal of this document is to provide an instruction and recommendation for performance of a long-term monitoring programme at pipelines potentially sensitive to flow-accelerated corrosion in nuclear power plant. The result of a successful implementation of the programme should include minimization of damage caused to important components of the secondary circuit and reduction of probability of degraded integrity of pressure pipeline systems of the secondary circuit including consequences following from this fact.

Corrosion of pressure pipeline systems internal walls is accelerated by flow of a medium – water or wet steam. This corrosion mechanism, which is called the flow-accelerated corrosion (sometimes also incorrectly called erosion corrosion), is one of the main degradation mechanisms of pipeline components of secondary circuits of nuclear power plants of pressurized-water type and depends on concurrence of several important parameters, such us chemistry of water, material composition, and hydrodynamics. Impacts of this type of damage have unfavourable safety and economic consequences. The effect of erosion corrosion mechanism on an operated pipeline or pipe component causes wear of material, which creates an area of worn wall of a pipe component in the range of units up to many tens of percents of the pipe component designed wall thickness.

In spite of the fact that this type of damage is taken into account at the construction and calculations of pipeline systems, the erosion corrosion damage issue cannot be considered to be solved yet. In most cases the damage is not immediate but is being incurred slowly and the resistance against loss of integrity is reducing. The loss of integrity appears only after several years of a flawless operation, often after a slight change of operating parameters. The flow-accelerated corrosion in the operating nuclear power plants caused number of damages to pipelines, which are important with respect to nuclear safety, such as steam and feed water pipelines.

The attempt for minimization of such cases, when flow-initiated corrosion appears on pipelines of nuclear power plants, uses combination of two approaches.

Firstly, to achieve such a combination of operating parameters that could exclude or significantly suppress occurrence of flow-accelerated corrosion. This is in particular possible at newly built power plants or in power plants, where system changes are in progress, which could allow change of these parameters. Such change may for example include replacement of material of heat transfer surfaces of heat exchangers or condensers, and the related change of the chemical mode of the coolant, or replacement of the pipeline taking into account requirements of prevention against flow-accelerated corrosion, and selection of a suitable material at the new pipeline.

Secondly, implementation of system measures consisting in systematic monitoring activities, reliable prediction of damage, and implementation of all attainable operating experience in assessment of state of selected pipelines. The goal of the second approach is to eliminate possible occurrence of defects, which could lead to damage of pipeline system pressure range, sufficiently in advance.

2. Basic issue of an effective programme for FAC monitoring

The main issues of an effective programme for FAC monitoring and predictions include:

— **Support to operator**, which means provision of necessary financial resources for all necessary tasks; determination of authorities, proper qualification by training of employees, securing of necessary communication and essential data and information sharing among respective departments; assurance of continuous monitoring of experience with FAC also outside of the power plant in question; development and implementation of long-term plans leading to reduction of damage caused by FAC; assurance of required quality, which consists also in preparation and documentation of procedures of necessary operations and realization of periodical and independent control assessment of all monitoring and prediction programmes for FAC; securing that all procedures, analyses, predictive models and documentations are being continuously updated and that all final reports from outages are being submitted in time.

— **Analyses**; In a typical nuclear power plant there are several thousand components, which are potentially susceptible to FAC. Without accurate and thorough FAC analyses, isometric drawings, pipe lines database, including inspections and history of replacements there is only one method, how to prevent media leakage and fractures of pipe lines – inspect every potentially susceptible component during each outage, which would mean a significantly demanding inspection programme as regards economy. The basic goal of FAC analyses is to identify the most susceptible components and thus reduce number of necessary inspections. A sample of components for inspections should be selected so that it contains the most susceptible components and secures sufficient fidelity of the model. It is necessary to take into account local conditions, age, and history of the power plant / block, wall thickness and dimensions of the pipe, material, length of the nuclear fuel cycle and existing industrial experience. For every pipe component an analytic method should estimate the degree of its wear, when the component should be inspected again, repaired or changed. The analytical model may be also utilized for further studies for costs reduction such as changes in chemical composition of water, changes of materials, increasing power, design changes etc. The analytical model may also create a powerful tool at the development of a long-term schedule of inspections, repairs, and replacements. The good and in practice tried and tested tools include e.g. CHECWORKS (EPRI), BRT-CICERO (EdF), COMSY (AREVA) computer codes. Currently, also the Russian Rosenergoatom is preparing their analytical model RAMEK.

— **Operating experience**; Research and integration of operating experience in the FAC monitoring programme is a very valuable complement of analyses and related inspections, in particular at the identification of problematic areas, for understanding of causes of differences of rate of damage at some similar components, understanding of FAC consequences at changes of chemical composition of water, increased power etc., sharing information on costs, materials, quality of suppliers, technology of repairs and replacements, inspection plans, new devices, etc. The very valuable contribution in this area is e.g. CHUG group at EPRI, which works at the FAC issue in the long-term and unites a large number of power plants operators. It is also possible to recommend FAC conferences organized by EdF or workshops prepared by IAEA.

— **Inspections**; Precision of inspections is the basis of an effective programme for FAC monitoring and predictions. Measurement of walls thicknesses determines the extent of components damage, provides data necessary for determination of the trend of FAC caused wear, and at the same time data leading to improvement of the prediction model. A thorough and precise inspection of several components is much useful than only a general check of a large number of components. The current practice does not recommend recording only one minimum thickness determined on a component, but it rather recommends a systematic data collection, which provides repeatability of measurements and gives a scope for determination of resulting trends.

— **Training and engineering judgment**; Regular trainings of respective employees create an inevitable precondition of a successful programme for FAC monitoring and predictions. Implementation of correct technical decisions, from modelling up to assessment of inspection data creates an inseparable part of programme for FAC monitoring and predictions. Therefore it is important so that workers, who participate in the programme, were familiar with operating experience, were properly qualified in their professions, trained in the FAC issue, and were able to process also necessary initial information from other departments, such as information on chemical composition of water, maintenance operation, thermal power etc. However, it is necessary to mention that even if the knowledge and the engineering judgment create an important part of a successful programme for FAC monitoring and predictions, they can't replace necessary analyses and inspections. All components are mutually interconnected and should be used together, not as a substitute of one or the other.

— **Long-term strategy**; Another item in the success of the FAC monitoring and prediction programme is creation and realization of a long-term strategic plan, which should be focused on FAC rate reduction and inspection in the most endangered localities. Monitoring of components is decisive for failure prevention, but without any effort to reduce the rate of wear due to FAC in time, with increasing number of service hours it is necessary to increase the number of inspections due to increased wear; it is the same as with increasing hours of operation it is necessary to increase number of repairs and replacements due to increasing probability of possible cracks and resulting media leakage.

3. Working progress and documentations

It is recommended to prepare a complex set of processes, which define implementation of programme for FAC monitoring and predictions in a power plant, determination of responsibilities, realization of inspections, assessments, and others. All these processes must be audited and documented.

3.1 Management documentation

The documentation should include specification of the entire programme and determination of individual responsibilities. It should also include:

— Determination to work at long-term aspects of erosion corrosion prevention and create environment for programme for FAC monitoring and predictions implementation including organizational and financial measures;

— Determination of tasks, which shall be performed (including operating procedures) and related duties;

— Definition of basic responsibilities for realization of erosion corrosion programme, authorities of workers responsible for realization of the programme and their organizational relation to maintenance workers and other departments, to which responsibilities to programme for FAC monitoring and predictions arise from prepared working processes;

- Requirements for quality assurance;
- Determination of long-term tasks and strategies leading to reduction of FAC rate.
- The documentation must be regularly revised and updated so that it includes:
- Organizational changes and changes of responsibilities;
- Changes of standards, legal regulations, and license requirements.

3.2 Operating processes

Operating working processes should be prepared for each concrete task within the FAC monitoring and prediction programme, however at least for the following scope of activities:

- Identification of systems susceptible to flow-accelerated corrosion;
- Execution of FAC analysis;
- Selection of components and preparation of inspections schedule;
- Preparation of inspections;
- Procedure for determination of content of alloying elements in the pipe line material, provided it is carried out;
- Preparation of inspection data;
- Assessment of damaged components;
- Repair and replacement of components or pipe line sections, provided it is necessary;
- Preparation (updating) of a schedule of further inspections in future outages.

Working processes and instructions must be also regularly revised and updated so that they reflect individual and organizational changes and responsibilities, changes of standards, legal regulations and license requirements, and development of new knowledge and technologies.

3.3 Other documentation

The input data must be available at least in the following extent:

- Axonometric drawings of modelled or measured pipe lines;
- Thermodynamic values describing state of media (pressure, temperature, enthalpy, steam quality, pipe roughness);
- Chemical state of media (concentration of constituents, pH);
- Geometry of components;
- Material parameters of pipe line (steel, alloy content, mechanical properties);
- Dimensional parameters of pipe line.

All decisions, results, and measurement reports should be archived. Control documentation and operating regulations should furthermore include also the susceptibility analysis, prediction model, and last but not least the appraising report from each operating outage, which contains a list of inspected components and reason for their selection (susceptibility to FAC, operating experience, engineering judgment, trends in wear etc.), results of inspections, appraisals, and recommendations for further operation of recommendations for repairs or replacements.

The susceptibility analysis should be periodically updated and should contain:

- Changes in secondary circuit systems including adjustment of valves;
- Design changes in pipe lines and changes of material in pipe components;
- Changes related to changes of power (pressure, temperature, flow, enthalpy);
- Changes related to possible untightness of valves and steam leakage;
- Gained operating experience.

Updating of the prediction model should be carried out after every operating outage and it should include:

- Results of inspections carried out during the last outage;
- Replacements of components;
- Chemical composition of water, operating conditions, changes in geometry (design) and changes in power.

It is also recommended to carry out an independent inspection of the susceptibility analysis, prediction model, selection of inspection localities, assessment of components, assessment report from the outage and all further documentations related to the assessment.

3.4 Records on replacements of components and pipeline sections

As regards assessment, the records on replacements of components, which were carried out in the past, are very important, because prediction of the wear rate and also residual lifetimes are related to the date, when the respective component was put into operation and to the number of operating hours. Information on those replacements should be included in the prediction model, and ideally also marked on isometric drawings used for the needs of programme for FAC monitoring and predictions.

4. Basic tasks at the fulfilment of the FAC monitoring and prediction programme

Some basic recommendations:

- The analyzed line means a set of pipe components in succession, which show identical or very similar operating conditions, such as chemical composition of medium, flow volume, temperature, amount of moisture (in case of a two-phase medium) etc.
- An inspection of five components, which show the wear rate over 0.8 mm (lower wear rate is already at the limit of the precision of measurement tools) for the period of their operation at least should be carried out on the analyzed pipe lines. The inspected components should include as wide variety of shapes as possible – elbows, direct parts, nozzles, reductions, expansions, T-pieces, and other related components in the flow direction.
- Together with inspection measurement of wall thickness of components it is recommended to carry out also measurement of content of alloying elements in the material, in particular chromium, copper, and molybdenum. Those elements have significant influence on the flow-accelerated corrosion, thus also on precision of the prediction.

The analysis carried out on a pipeline, where no inspection measurement of wall thicknesses was carried yet and the results are based only on operating data and theories, is called the **first predictive analysis.** The analysis carried out on a pipeline, where measurement of wall thicknesses was already carried out on selected components, is called the **improved predictive analysis.**

The prediction is an estimation/determination of components walls thickness wear rate due to flow accelerated corrosion and the total wear or thickness of walls of these components as of a certain date, which is carried out based on mutual correlation of various factors, such as geometric factors, composition of material, chemical composition of water, conditions of flow etc.

4.1 Identification of potentially susceptible systems and pipeline

4.1.1 Potentially susceptible systems

- Based on accepted criteria it is important to determine, which systems must be regarded as susceptible to flow accelerated corrosion and keep a list of these systems. Determination of these systems is based on the following information:
- Operator's experience supported with records. The operator keeps records of all events, which may relate to the flow-accelerated corrosion. These events should in particular include:
- All cases of media leakage due to FAC;
- All cases of damage of a component due to FAC;
- All measured wears of components, which show wall thickness less than 70 % of the nominal wall thickness of the component;
- Known experience with damage caused by FAC or significant wear from other WWER type power plants operated based on analogous conditions, in particular chemical composition of water in the secondary circuit and experience of other operators.
- As regards experience with occurrence of flow-accelerated corrosion on secondary circuit systems it is known that the potentially most susceptible systems are as follows:
- Feed water;
- Extraction lines of turbo generator, in particular from the high pressure part;
- Heating of steam separators;
- Low-temperature and high-temperature regeneration of condensate;
- Moisture separator and pre-heating of steam;
- Boiler drainage pipe line (FWH);
- Blow-downs of steam generators;
- Condensation water.
- As regards safety and consequences of possible damage to components integrity to their surrounding the systems in particular include:
- Feed water pipe line;
- Live steam pipe line;
- High pressure extraction lines.

4.1.2 Systems excluded from the assessment

Some systems or their parts may be excluded from further assessment because of their relatively low susceptibility to the flow-accelerated corrosion. Based on the quantity of laboratory data and operating experience we can exclude the following systems from the assessment:

— *Systems or their parts, which are made of stainless or low-alloy steel with minimum content 1.25 % of Cr.* However, it is possible to exclude only such lines from the assessment, in which all pipe components are made of such material highly resistant to FAC. In case a line includes some components, which are made of carbon steel, it is not possible to exclude such line from the assessment; on the contrary, those components are considered to be highly susceptible and must be analyzed. It means, there is a danger of the so called "entrance effect" when components made of less resistant material, which are installed behind the components made of highly resistant material in the direction of the flow, succumb to FAC influence with increased rate. It is also necessary to mention, that materials with high content of chromium, which are resistant to FAC, may be susceptible to other types of damage, such as cavitation or erosion caused by impact of liquid particles (liquid impingement erosion). So, if the mechanism of damage is not clearly identified, the replaced components should remain in the monitoring and inspection programme.

— *Systems, in which the share of a liquid component in the two-phase medium is less than 0.01 %, it means lines with pre-heated steam.* Nevertheless, the existing knowledge proves that on certain conditions - out of common operating conditions or at the decreased power - some systems with pre-heated steam may show increased amount of moisture, therefore a certain extent of monitoring should be maintained.

— *Systems or their parts with one-phase media flow, in which the temperature is less than 90°C*. However, it is necessary to point out that in those systems various types of damage, such as cavitation, may appear.

— *Systems or their parts without media flow or which work less than 2 % of the operating time of the power plant.*

With the exception of systems, which include materials with high content of chromium it is not possible to determine thresholds for exclusion of flow accelerated corrosion origination, but in most cases they contribute to significant decrease of its influence by mutual combination. The decision on exclusion of a system or its part from the inspection programme should therefore be based on a qualified assessment from all mentioned points of view and to be a part of the basic document for assessment of flow-accelerated corrosion in the power plant.

Systems or their parts should not be excluded from assessment only based on low pressure. Pressure has no influence on FAC impacts; it influences only the degree, when the damage could be caused. Damage in low-pressure systems may have significant consequences on the operation.

Systems or their parts, which were excluded from the assessment based on the above-described criteria, are less susceptible to FAC, however it is not possible to exclude other corrosion or degradation mechanisms, which include e.g. cavitation, liquid impingement erosion, stress corrosion cracking – SCC or solid particle erosion. But, these degradation mechanisms are not part of FAC monitoring and prediction programme and this report and must be assessed separately.

4.2 Predictive analysis

An effective programme of measures against the flow-accelerated corrosion must be based on the ability to predict wear in parts of the secondary circle for longer period of operation. Nuclear power plant operator must have a method available for estimation of FAC degradation development or estimation of the component lifetime. Based on this prediction the measures such as measurements, replacements or repairs of components over the lifetime and total strategy of the progress in relation to operating conditions shall be determined.

In order to have such prediction it is possible to create a calculating model for a concrete system of a power plant secondary circuit, with description of the method for realization of the analysis with determination of the scope of parameters necessary for the analysis, susceptibility of the model and method, and how the individual values of parameters were achieved. Further it should include a method for support of the model with experimental assessment with the parameters analogous to the operating parameters of components and materials of the secondary circuit, how the data from inspections are included in the model, and how these data improve the model and the method of testing of the model.

Another and probably more suitable method includes utilization of a tested computer programme for damage predictions, such as CHECWORKS, BRT-CICERO, or COMSY, which are used in many power plants in the world, and there is a good correlation between results of the prediction and measurements on concrete components.

4.3 Selection of components for inspections

Inspections include measurements of wall thicknesses of pipe components by means of a non-destructive method (e.g. ultrasound) on a grid of points on their surface.

Selection of inspection places should meet the following basic requirements:

- The selection must include components from all systems of the secondary circuit, which were determined as susceptible to erosion corrosion.
- The selection must be sufficiently objective. The source of information for selection of components is:
    - Modelling by a tested computer programme (CHECWORKS, BRT-CICERO, COMSY) or prediction according to a proper and independently tested model. Approximately 50 – 70 % of components are selected this way.
    - Operator’s experience with the system, in particular with already identified wear, media leakage, or fractures of components. It results from the documented experience, which is kept by the operator. Approximately 20 -30 % of components are selected this way.
    - Engineering judgment. Even if it is the least unbiased method, its inclusion is appropriate, because it takes into account all operating conditions. Not more than 10 % of components for inspections should be selected this way.
- All geometries of components, which appear in the assessed systems should be represented in the selection.
- The selection criteria should also include experience of other operators, in particular from analogous nuclear power plans. Further it is necessary to define the method, how to include this information in the assessment process.
- For selection of components it is necessary to prepare similar criteria, which take into account:
    - Necessary measurements of those components, which are installed in similar lines in places with already proved degradation caused by the flow accelerated corrosion.
    - The total number of components inspected within the outage. Operator shall give reasons for the number of selected components based on the state of the system, existence of initial data, experience, and recommendations.
    - The relation between the measurement on a damaged or replaced component respectively and its neighbouring components.

Number of measurements on one pipeline should not be lower than 25 % of all components, with at least 5 components for the following lines with:

- Two-phase media;
- Pressure above 3 MPa, provided predicted occurrence of FAC on components in this line was confirmed, in particular on feed water and live steam lines;
- Positive record of occurrence of integrity failure caused by FAC.

At all other pipelines selected for FMPP the minimum number of measurements per line should not be lower than 15 % of all components, with minimum of 4 components. The line means a part of the pipeline of the secondary circuit with identical thermodynamic conditions.

At the selection of components for inspections it is also necessary to take into account whether they are:

- Lines measured for the first time;
- Lines with repeated measurement;
- Lines with unknown or not exactly defined operating conditions.

4.4 Preparation and performance of inspections

The scope of measurement on a component should be defined so it is possible to:

- Accept a decision, whether the component is still serviceable, or how long it is serviceable until the end of its lifetime (achievement of acceptable thickness);
- Improve the prediction model either in form of a computer programme or other approved procedure.

Measurement of walls thicknesses is a routine procedure and measurement of a significant quantity of points in the grid may lead to mistakes expressed by large deviation, as current experience shows. If such deviations wouldn't be excluded, they could cause a gross mistake in the assessment and in the changed model as well, which could influence prediction of damage of measured components. Taking into account features of these mistakes, they mostly show higher degree of conservatism than necessary, which causes higher number of inspections carried out during further outages. Gross mistakes may be eliminated directly; small differences in measured values must be assessed by statistical methods.

Some types of components, such as T-pieces, elbows, and bends, have uneven thickness of walls caused by the manufacturing technology. The proposed procedures must therefore take into account also elimination of such differences so that they will not cause a significant degree of conservatism in the assessment.

4.4.1 Inspection techniques

Inspection of pipe components may be carried out using the ultrasound technology (UT), radiography (RT) or visual observation. The ultrasound technique and also the radiographic method may be utilized at the determination of damage degree, but the UT method provides more complex data at the assessment of the residual thickness of large bore pipelines. On the contrary, the radiographic method is more often used at the assessment of small-bore pipes (up to 50 mm diameter) and also welds, components with uneven surface, such as valves or flow nozzles and coupling sleeves. Another advantage of the RT method is its wide coverage and visual information on wear of component walls. Besides this, the RT method may be used also under working conditions without necessity to remove the insulation. Although, on one hand the RT method may reduce cost and time during the inspection, on the other hand it may increase those costs and extend the time of the outage due to complications with other tasks linked to radiation requirements and employees safety. Therefore, in majority of cases of inspections the UT method measurement connected with electronic data recording is used.

Visual inspections are most frequently used at inspections of valves and other fittings and also at inspections of very large bore pipelines (e.g. 900-1500 mm), such as steam pipe lines leading from high-pressure part of turbine of separators pre-heaters and from these pre-heaters to the low-pressure part of the turbine. Those visual inspections are carried out inside of the pipeline. Complementarily, at the suspicion of a larger wear of pipe wall thickness the ultrasound measurement is carried out, by means of which measurement may be carried out both outside and inside the pipeline.

The inspection process performed by the UT method lies in measurement of component wall thickness in points of intersection of the measuring grid, which is applied on the component. If the obtained data indicate significant wear, the measuring grid in the given locality should be reduced (shortened distance between the points of intersection of the grid) and the measurement should be carried out to identify extent and depth of the defect.

The obtained inspection data are further utilized in three ways:

— For determination, whether a wear occurs at the component, and for determination of the locality, where the wear occurs most of all;

— For determination of the scope and depth of the wear;

— For determination of the rate and trend of the wear.

4.4.2 Preparation of the measured place

Preparation of the measured place includes stripping the insulation from the component, its basic cleaning, application of the working grid, abrading surfaces of the working grid in the required quality and sufficient marking of measuring points.

The extent of the stripped insulation differs according to the shape of the given component. In case of long direct parts, it is necessary to strip a part with the length of two diameters of a pipe line at least; in case of short direct parts the insulation from the entire direct section must be stripped. In case a weld is located at the beginning of the direct part (for short direct parts also at the end of the component), the insulation in the surrounding of the weld must be stripped as well to provide suitable access from both sides. The rule on accessibility to welds applies also for other shapes of pipes (elbows, Tee etc.). Elbows and bends must be stripped from the beginning of the curve to its end. T-shaped pipes must be stripped in the extent of two diameters of the pipe to each side from the axis of the branching. The same applies for reductions and expansions. The surface of the measured components must be suitably cleaned from gross dirt, e.g. dust, paint, rust, glass fiber rests etc.

The working grid is applied on the clean surface for marking of measuring points. For measuring out the working grid special templates in the shape of a band with marked points in the distance of 1/12 of the circumference of the pipe is used. A writing tool, which is visible on the given surface, shall be used for marking. If the grid is applied on a part of the pipe without surface unevenness and dirt (e.g. new or replaced pipes) it is not necessary to apply mechanical cleaning on this section and the final measuring grid is marked directly instead of the working grid.

Quality of the surface of the inspected place is one of the most important factors influencing precision of thickness measurements. The marked intersection points of the orthogonal working grid (places marked with crosses) are mechanically cleaned so that unevenness caused by surface corrosion, sediments, or other dirt, or coat of paint as the case may be, are removed from area of approximately 2x2cm of every intersection point. It is not necessary to clean the surface to the layer of the non-oxidized (burnished) metal, but only to the state, which ensures sufficient acoustic coupling between the measuring probe and the ultrasonic tool, while no basic material of the pipe is removed unnecessarily. It is recommended to carry out cleaning of the surface using an electric grinder with lamellar grinding wheel.

4.4.3 Marking of the grid and measuring points

The final measuring grid is marked on the adjusted surface in the centre of grinded (cleaned) areas and individual measuring points have shapes of crosses. The grid is applied by means of a permanent heat-resistant and abrasion-resistant paint of a paint-marker. It is important to pay particular attention to visibility and legibility of marked crosses, because the marked grid should be maintained on the pipe also for future repeated inspections. For better orientation the beginning of the system of coordinates on the pipe – the node point A001 – is encircled in colour, and thus it represents the initial point of the component. In case of repeated measurement on the component it is necessary to use the same grid as regards dimensions, position, and orientation. At the same time, special attention must be paid to correct orientation of the grid, because slight twisting of the grid (from the beginning) makes use of the comparative point-to-point method impossible.

If measurement on the component is carried out repeatedly and quality of the surface of the place measured in past does not achieve the required parameters, the surface is cleaned to the required quality either by means of a steel brush, soft sandpaper, or electric grinder with buffing wheel. Final cleaning is carried out very gently and always only to the state, which ensures quality coupling between the probe and the thickness gauge to prevent further unfounded removing of the basic material of the pipeline. It is also necessary to pay attention to synchronization of the original damaged and the new repaired grid for assurance of comparability of the measured points.

Current experience proves that it is very difficult to forecast in which place on the component the most serious wear occurs. To ensure detection of the locality with the most serious degree of wear the measuring grid must fully cover the inspected shaped component (elbow, tee, expansion, reduction etc.). Fully covering grid is also a good basic element for future inspections and assessments of the component. At direct parts it is recommended to apply the measuring grid in the distance of three diameters of the pipe at least behind the shaped component or the weld.

Application of the working or the final measuring grid respectively is carried out in accordance with the following rules. The measuring is carried out in node points of the orthogonal grid. The grid consists of line segments, which are led in axial direction towards the surface of the component and from circles, which are led in the circumferential direction. The circles are marked with numbers 001 – *n*, in which *n* is number of circles in the three figures marking, e.g. at 15 circles the following marking will be used 001 – 015. Superficial line segments (axes) are marked with alphabet letters A – *n*, in which *n* is the last segment adjoining the A line segment from the left. At the standard marking – 12 points on the circumference the circles will be marked A - L. The beginning of the system of coordinates is selected always at the point of intersection of the first circumferential circle in the direction of the media flow with the first axis – standard marking of A001 point.

The order at the marking of axes is carried out clockwise (the so-called right-hand rule). The extent of the grid depends on size of the component or on diameter of the pipeline. Generally, the component is divided to 12 parts circumferentially, which is the recommended and tried-and-tested system. In case of need, e.g. for performance of a detailed measurement of thicknesses, it is possible to use a denser grid. The table 1 shows recommended spacing of points in the grid for measurement.

Valves and fittings, orifices, coupling sleeves, flanges, and similar components cannot be completely measured by the UT method due to their shapes and unevenness of their surface. It is recommended to apply the grid and to carry out measuring on the nearest component in the direction of the media flow in the distance of two up to three pipeline diameters from the connecting weld at least. If significant wear is found on that subsequent component, then other type of inspection, e.g. visual or radiographic, should be carried out on components with complicated shapes, e.g. on valves. However, this approach may be used only in case that the component with complicated shape is followed by a component made of the same material or of a material susceptible to FAC (with low content Cr). This method may be also used at replaced components.

TABLE XVII.1. Recommended spacing of grid points

Outside diameter of the component (mm)	Maximum dimension of the grid (mm)
< 90	25
90 - 135	36
136 - 160	42
161 - 210	55
211 - 270	70
271 - 324	85
325 - 367	96
368 - 420	110
421 - 460	120
461 - 500	130
501 - 535	140
536 - 635	166
636 - 690	180
691 - 820	214
> 820	250

4.4.4 Wall thickness measurements by means of UT method

Ultrasonic measurement is carried out by means of a device, which meets the following requirements:

— Parameters of the ultrasound device and accessories allows measuring with the minimum sensitivity of 3 % at measured thicknesses ranging between 5 and 20 mm and sensitivity better than 10 % at thicknesses less than 2 mm.

— It is calibrated for measurement. The calibration is carried out by means of a gradual gauge regularly before initiation and completion of measurement to prevent mistakes at measurement caused by incorrect setting of the device.

— It has the option of automatic data storing in the internal memory and the subsequent export of measured data in the proper format to a computer.

— It is available in two examples at least for the purpose of prevention against failure of data collection due to malfunction.

The measurement is carried out in defined and pre-prepared grid. Measurement starts in the point described as the "beginning of the system of coordinates" and continues on axes in the direction of flow and on circumferential circles clockwise, i.e. after all points of the axes are used, the measurement continues on the next axis in the mentioned direction. The figure 1 shows an example of measured grid on an elbow. Measurements are carried out with precision of two decimal places.

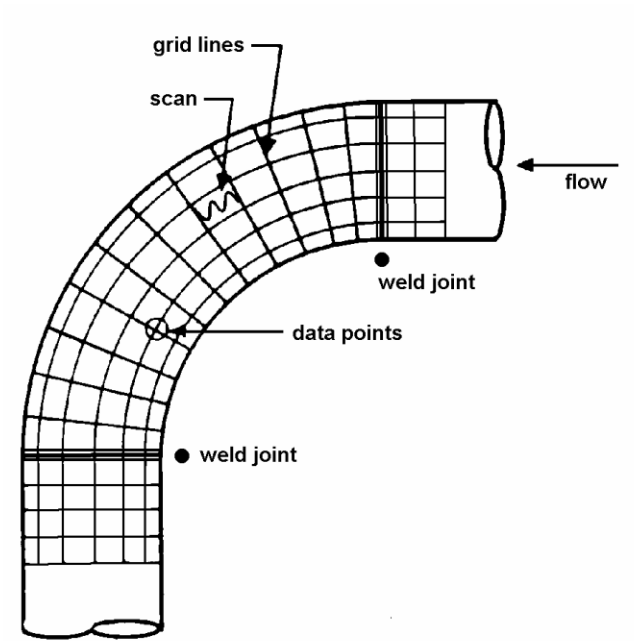

FIG. XVII.1. Measuring grid on elbow (taken from [2]).

The workers, who carry out measurement, have the so-called signal value of thickness available for each component. If the measured thickness is lower than the signal one (or it differs significantly) in one or several points, the respective responsible worker is notified. Such component must not be insulated again unless a decision on its future operation is accepted. By all manners of means, in case of occurrence of this circumstance it is necessary to start a detailed measurement of the surrounding of the point with thickness lower than the signal one. Such measurement is carried out by means of raster scanning of thickness. Data from this additional measurement are recorded in a special file. After every measurement the measured data are assessed for presence of significant deviations as regards higher or lower thicknesses as well. In case of suspicion of occurrence of a gross mistake in measurement, the measurement is repeated in the given place.

4.4.5 Measurement of chemical composition of material of components

Some chemical elements, in particular chromium, molybdenum, and copper, have positive impact on decreased degradation caused by flow-accelerated corrosion. Their content in a component is important information for assessment and damage prediction. Without knowledge of the particular composition, it is necessary to use the lowest content of those elements in the calculation, which is guaranteed for the given steel and may achieve also zero value. Then, this leads to considerably conservative results. Therefore, it is suitable to utilize the access to free surface of components during measurement of wall thickness and (at the first measuring on the component) to measure also its chemical composition. The most efficient is utilization of emission spectroscopy from de-sparked surface of the component or to use of handheld XRF analyzer.

The surface must be cleaned from any and all oxides and smears. For this an electric grinder may be used, but only necessary quantity of material may be grinded away. Places for measuring must be selected in a method, which ensures sufficient distance of those places from areas susceptible to wall thinning (e.g. material on the shortest axis of an elbow or in near proximity of welds should not be grinded off).

Before performance of the proper measuring, it is necessary to correctly set the equipment on check samples and in case of discrepancy recalibration must be carried out, which will secure achievement of correct results. The measurement is carried out on each part of the pipeline divided by welds, i.e. for example on segment elbows composition of all individual segments is ascertained. Measuring of each part consist of three samples, while the resulting content of individual elements is the average of those three samples. A draft and record on the carried-out measurement is made for the carried out measurement.

4.5 Assessment of inspections

Measuring of thicknesses on each component must be assessed in particular taking into account determination of:

- The minimum thickness and its comparison with the acceptable thickness. Based on this comparison it is necessary to determine the time till the next check or further operability of the component;
- Extent of defect (wear) for possible strength analysis.

Measuring of the component is assessed so that it is possible to determine the following factors for the given component:

- Flow-accelerated corrosion estimation;
- Estimation of the time until the acceptable thickness is achieved.

The assessment process is complicated by the following factors:

- Initial (pre-operational) thicknesses are not known or the so called "baseline" data were not determined;
- Variety of thicknesses along the axial axis and along the circumference as well;
- Inaccuracy at non-destructive (NDE) measurement;
- Possible constructional unevenness;
- Incorrectly recorded data or mistakes incurred at the transmission from the device to PC;
- Obstacles preventing performance of measurement in the entire grid (complicated access, insulation, spring hinges etc.);

Impact of these problem factors should be minimized by means of an engineering judgment and application of a suitable assessment method.

4.5.1 Methods of data evaluation from one outage only

Flow-accelerated corrosion wear assessment could be divided into two categories. The first category includes components, for which the pre-operational data are known and their assessment is carried out by the ***Point-to-Point*** method. The measurement is carried out in orthogonal system of cylindrical coordinates, which is created by superficial circumferential circles and axes. The axis is always perpendicular to the plane of a circle. Thickness wear is defined by the following relation:

$$\Delta t = \max \{t^0_{i,j} - t^1_{i,j}\}, \qquad \text{(XVII.1)}$$

where superscripts 0 or 1 relate to the original or the current measurement and subscripts go through the entire set of the measured points of the grid.

The second category includes components with unavailable pre-operational data. In this case, determination of the real component wall thickness wear is one of the most important tasks at the monitoring of the flow accelerated corrosion impact. It is complicated in particular by the fact that if pre-operational data are not available, it is not easy to determine the initial thicknesses. That is, even if data from two outages are available, even a slight wear, mistake in measurement, and some other factors make the FAC damage estimation difficult.

In these cases, four methods, which utilize UT data from inspections are usually used:

— Band Method;

— Averaged Band Method;

— Area Method;

— Moving Blanket Method.

All these methods are used also for estimation of the initial thickness of a component. All these methods are based on the theory that the wear caused by erosion corrosion appears in typical surfaces and areas.

The *Band method* is based on the assumption that the flow-accelerated corrosion damage is localized. The component is divided circumferentially to circles with the width of one square of the selected grid. These circles are in the plane perpendicular to the direction of media flow. The initial thickness of every circle is the higher of the values of the nominal or the maximum thickness. The wear is then calculated as the difference of the initial thickness determined this way and the measured minimum thickness. The maximum wear from all these circumferential circles is then understood as the wear of the component wall thickness. The initial thickness of component wall is then understood as the initial thickness of this circle, at which the maximum wear was determined.

The *Average band method* is an analogy to the Band method, but the difference is that the initial thickness of each circle is the highest of the values of the nominal thickness or the average of all the values. The wear of one circumferential circle is then understood as the difference between the initial thickness determined this way and the measured minimum thickness and the maximum wear from all of them is understood as the component wall thickness wear.

At the *Area method* the grid is divided to mutually disjoint subsets (areas) with the same size. The initial thickness of every such area is understood as the highest of values of the nominal thickness or the maximum thickness found in this area. Wear of the wall thickness in every area is then understood as the difference between this initial thickness and the found minimum thickness.

The *Moving blanket method* is the improved version of the Area method. The assessment process is performed so that in the area – the blanket - selected by us, the initial and the minimum thickness is ascertained with the same method as at the previous one. Then, the entire blanket moves by one field of the grid in the circumferential direction and the assessment is carried out again. This procedure continues the same way as long as all the points are included in the blanket with the same weight. This method allows eliminating impacts of ovality and local unevenness of wall thickness.

The mentioned methods and others are described in detail in the reports [2] or [3].

4.5.2 Assessment of inspection data by the Point-to-Point method

The *Point-to-Point method* is used if data from two or more outages are available, and they were measured in the same grid. If the wall thickness is measured during later outage and then it is subtracted from the thickness measured in the same point of the grid in the earlier outage, the difference shows the wear in the given point. The largest wear in the grid is then the wear of the component during the period between those two outages. This method is not used for estimation of the initial thickness of the component.

Combination of the Point-to-point method for data from more outages (with unknown pre-operational data) and one method for data with one outage only, enables to determine the so-called general damage of the component during the operation. This is carried out in three steps:

(1) By means of data acquired during the first inspection the wear is determined through the Band method, the Averaged band method, the Area method or the Moving blanket method.

(2) By means of the Point-to-point method the incremental wear between two outages is determined.

(3) The total damage of a component during its operation is determined as a summary of values of wear from steps 1) and 2).

TABLE XVII.2. Recommended assessment methods

Component type	Band Method	Averaged Band Method	Area Method	Moving Blanket method	Point-to-Point Method
Elbows, bends	NO	YES	NO	YES	YES
Tee	NO	NO	NO	NO	YES
Straight pipe	YES	YES	NO	YES	YES
Reduction or Expansion	YES	YES	NO	NO	YES
Flanges and Coupling sleeves	NO	NO	NO	NO	YES
Valves/Pumps	NO	NO	NO	NO	NO

Selection of the assessment method suitable for determination of the total damage of a component depends also on the type (shape) of the component. Suitability of individual assessment methods for various types of pipe components is given in Table 2.

4.6 Assessment of components and determination of residual lifetime

After collection and entry of initial data in the programme a model is created and the so called first predictive analysis showing in particular susceptibility to individual components to flow-accelerated corrosion damage is prepared. By periodical entry of data measured directly on components (wall thicknesses and chemical compositions) development of flow accelerated corrosion in time is monitored and precise predictive analyses are prepared. The main outputs of the predictive analyses are component wall thickness wear rate and time until its acceptable value is reached, i.e. the residual lifetime for inspected and for not inspected components as well.

4.6.1 Determination of critical (acceptable) thickness

Determination of critical (acceptable) thickness T_{crit} results from internal directives of a nuclear power plant, e.g. ASME Code Case 597 [4], and normative and technical documentation for “Evaluation of the piping and equipment strength in WWER NPPs” [5].

Generally, it is possible to determine criterial thicknesses of components within the monitoring and prediction programme at three levels:

<u>Level 1</u>

$$T_{crit-1} = 0.875T_{nom}, \tag{XVII.2}$$

where T_{nom} is the nominal thickness of the component wall. If the measured or the predicted current thickness is over $0.875T_{nom}$, the component is considered to be safe and no additional activities are necessary until this limit is reached. It is important to mention that the first criterional thickness means only the fact that the value, which is normatively permitted at the production of a pipe component, was achieved. It does not mean that integrity of pipeline is endangered somehow and there is a danger of media leakage.

Level 2

Periodical inspections are carried out on the component with thickness below 0.875 T_{nom}. They are carried out depending on the predicted erosion corrosion rate and lifetime. At the same time the so called second critical or acceptable thickness is determined. The acceptable thickness is determined

a) based on the strength calculation for the given type of steel and operating and dimensional parameters on the basis of normative and technical documentation for assessment of solidity of devices and pipelines of WWER type nuclear power plants and internal directives of the respective power plant. The basis for calculation of wall thickness is the following formula:

$$T_{crit-2} = \frac{p\,D}{2\varphi[\sigma] + yp} + c \qquad \text{(XVII.3)}$$

where

p = maximum service pressure [MPa]
D = external diameter of pipeline [mm]
[σ] = allowable stress
y = design factor = 0.4
c = allowance for corrosion and erosion

or

b) $T_{crit-2} = 0.3T_{nom}$, (XVII.4)

whereas higher of these two values is taken into account. When this criterial value is achieved the decision-making process should take place, during which it is determined, whether the component shall be repaired, replaced or a local measurement in more detailed grid will be carried out to determine precise thresholds of the defect. Based on localization of this defect the third local criterial thickness is determined then.

Level 3

If the measured minimum thickness on the component equals to the thickness T_{crit-2} or is even lower, then the component must be replaced or it is possible to carry out the local strength analysis depending on the type of the component, its location, locality and dimensions of the defect. By means of this analysis it is possible to determine by what value the determined critical acceptable value T_{crit-2} may be reduced for components with local damage in the given place without endangering the required functioning of the component. The output of the strength analysis of the damaged component is the *"Record of used calculating thicknesses"*. Wall thicknesses, which were used in the strength analysis, are considered for the evaluated component as critical values and walls may not be thinned below these values under no circumstances. After this critical thickness is achieved the component must be repaired or replaced. Local strength analyses may be used in exceptional cases, when extension of the time until replacement is desirable (e.g. due to lack of spare parts, outage duration, planned life time of nuclear power plant etc.)

However, in most cases it is sufficient to determine only the first the two critical (acceptable) thicknesses; determination of the third is usually not carried out and due to economic and safety reasons the respective component is repaired or replaced already after the second level is reached.

The allowance for corrosion and erosion C to the wall thickness makes up wear of wall thickness due to all types of corrosion and erosion for the time of the required technical lifetime of the component. Values of the allowance C are determined in accordance with the Suggested specifications. In case necessary data are not available, the values given in table 3 may be used as minimum; their values may be increased with respect to erosion or corrosion speed and operation time.

TABLE XVII.3. Values of allowance for corrosion and erosion C

Material and its welded joints	Operating conditions of the material in the nominal mode	Allowance C [mm]
Ferritic perlite steel	Water (40 up to 160°C) Water (160 up to 270°C) Water, to (350°C), pH = 10 Saturated steam up to (300°C) Overheated steam	0.3 1.2 1.0 1.0 0.5

4.6.2 Determination of the residual lifetime

The line correction factor (LCF) is auxiliary tool for the prediction of wear and residual lifetime of inspected and not inspected components; its principle is based on comparison of expected and measured wear rates. For each inspected component on the monitored line the ratio of the measured wear and the wear predicted by the programme is determined; median of values determined this way becomes the correction line correction factor.

Its further use for predictions is as follows:

1. Modification of FAC rate and wear:

$$FACR_2 = FACR_1 \cdot LCF$$

$$Wear_2 = Wear_1 \cdot LCF$$

2. The current estimated thickness is determined:

$T_{pred} = T_{init} - (FACR_2 \cdot Time_1)$ for non-inspected components

$T_{pred} = T_{meas} - (FACR_2 \cdot Time_2)$ for inspected components

3. The residual lifetime of component is determined

$$\text{Time to } T_{crit} = (T_{pred} - T_{crit}) / FACR_2$$

where

LCF = line correction factor

$FACR_1$ = estimated (theoretical) rate of FAC

$FACR_2$ = repaired rate of FAC

$Wear_1$= estimated (theoretical) wear of wall thickness

$Wear_2$ = repaired wear of component wall thickness

T_{pred} = predicted thickness

T_{init} = initial or nominal thickness

T_{meas} = minimum measured wall thickness

T_{crit} = criterial (acceptable) thickness

$Time_1$ = time of component operation

$Time_2$ = $time_1$ – time of last inspection

Time to T_{crit} = residual lifetime

5. Collection and archiving of input and output data

In the system of works supporting the flow-accelerated corrosion monitoring and prediction programme the data collection is the starting point for further activities, in particular:

— Creation of thermodynamic and chemical model of secondary circuit;

- Fundamental selection of lines, which will be included in the FAC prediction programme;
- Creation of predictive model;
- Assessment of acceptability of components for further operation.

Since using of the computing code puts demands in particular on quality of input data, it is necessary to define individual input parameters, method of their acquisition, verification, and requirements for their accuracy. In addition, the operator should secure a transparent archiving of input data, results of measuring and predictions in the power plant and at external cooperation as well, in particular with respect to creation of building the prediction models. The database of results serves for determination of a component for further operation, its putting out of operation, or it defines the time until the next measurement.

5.1. Data classification to groups

The system of division of input data to individual groups, which creates the basis also for their archiving, is based on principles, which various computer programmes for monitoring, assessment and prediction of accelerated-flow corrosion damage work with. The reason is that majority of data should meet requirements of these programmes, which create the tool for FAC damage modelling and prediction. The nuclear power plant secondary circuit system is divided in this respect to individual groups, which represent:

- Unit;
- Performance of reactor and thermodynamic data of the secondary circuit;
- Chemical mode of nuclear power plant for the monitored period;
- Operation and outage of power plant;
- Line, which is a dynamic logic subsystem of a secondary circuit given by connection weld between important devices of the secondary circuit (e.g. pipe line of one extraction line of HP part of turbine or feed water pipe line of steam generator). In specific cases (small number of components) this term may be merged with the term "segment";
- Segment, which includes a pipe line with homogenous thermodynamic and chemical features;
- Component, by means of which a part of segment is delimited, in particular by connecting joints, and is distinguished by the only geometric classification according to user manual of the respective computer programme;
- Inspection data;
- Results of modelling.

5.2. Method of data archiving

Input and output data are archived in a way that respects the mentioned classification. Data must be saved in two different methods at least, while one of them should always include archiving of "hard copies". The second method may be for example an archive kept in electronic form. In case data are saved in more ways, on various carriers, the primary source of data is always the "hard copies" archive. All data saved there must be verified by a worker responsible for performance of assessment of flow-accelerated corrosion in the power plant.

Besides the operator the data are also archived in the organization performing modelling and organizing of the measured thickness of components in the extent following from their share in the assessment.

6. Long-term strategy

The goal of the flow-accelerated corrosion monitoring and prediction should not cover only determination of the current state and performance of inspections, but also effort to decrease susceptibility of pipe components to this type of degradation mechanism and optimization of the inspections planning process. It is desirable to optimize the entire process so that the number of inspections is cut and probability of threat of bursts and loss of integrity is decreased (see Figure XVII.2). Therefore, preparation of a long-term strategy is recommended.

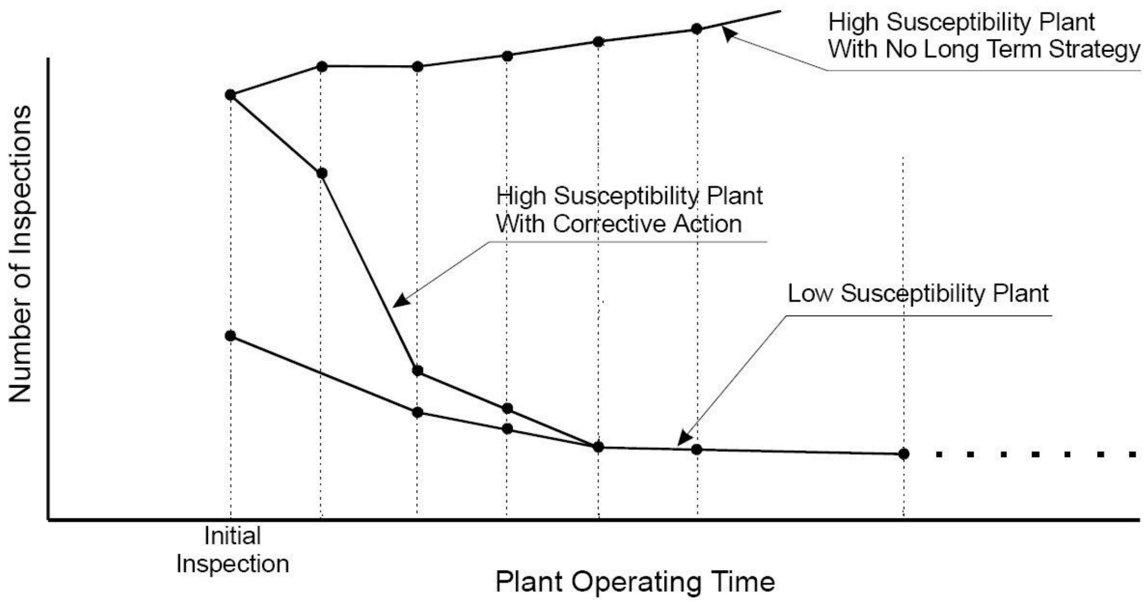

FIG. XVII.2. Estimated trend in number of inspections (taken from [2]).

One of possible approaches is replacement of only those components, which show the highest degree of wear. This approach is satisfactory, if the wear is precisely localized. It includes mostly localities behind valves or curtains. However, in most cases the wear is spread within the entire system, because the flow and chemical composition of water have influence on the entire line and it is only a matter of time, when it will be necessary to replace the individual pipe components and devices. "Replacement-to-replacement" approach may be less expensive in short-term bases, but in long-term its efficiency is decreasing significantly. Moreover, some current experience shows that unexpected failures may occur at components, whose replacement was planned on later date.

Therefore, in the interest of achievement of long-term goals – i.e. costs decrease and safety increase, it is recommended to adopt a strategy of systematic reduction of flow-accelerated corrosion in the secondary circuit systems of a power plant. There are three possibilities how to achieve it:

— Better, more resistant material;

— Better chemical water treatment;

— Local constructional changes.

Use of resistant material may reduce damage rate almost to zero, depending on locality and the given secondary circuit system the changes in chemical mode may reduce FAC rate even ten times, and in some areas also constructional changes may lead to improvement.

6.1 Resistant material

Many experimental works and operational experience have proved that the flow accelerated corrosion rate is influenced in particular by chromium, copper, and molybdenum, with the chromium content being. Its impact is favourably applied in particular on strengthening of the oxidic layer on a pipe surface, reduction of its porosity, and changes of the corrosion potential.

Replacement of carbon steel by chromium-molybdenum steel or by a material with a piece of stainless steel leads to a significant reduction of damage caused by FAC over the period of the power plant's lifetime. Another option is to ensure that all the replaced components made of carbon steel contain at least 0.1 % of Cr.

The table XVII.4 shows expected reduction of flow-accelerated corrosion rate for materials with higher content of alloying elements in comparison with carbon steel predicted by the computational code CHECWORKS based on the Ducreux's model [8] and current operating and laboratory data [9].

TABLE XVII.4. FAC rate decrease at the utilization of more resistant material

Material with content of alloying element	FAC rate $_{carbon}$ / FAC rate $_{alloy}$
0.10% Cr	10
1.25% Cr, 0.50% Mo	34
2.25% Cr, 1.00% Mo	65
18% Cr	> 250

Changes of material may be applied in entire systems, e.g. high pressure extraction lines or repairs may be carried out in areas which are particularly susceptible to damage. In the other case it is necessary to pay particular attention to threat of possible occurrence of the so called "entrance effect". It is also possible to draw the attention to the fact that replacement of material does not automatically mean reduction of damage rate, if degradation mechanisms are different from FAC. For example, if the damage is caused by cavitation.

6.2 Chemical composition of water

Generally, the flow-accelerated corrosion depends on many parameters. The parameters of the type such as flow rate, geometry of pipe (diameter, bends, elbows, Tees), temperatures and enthalpies of the flowing media, content of the liquid phase in the two-phase mixture water/steam etc. are given by the unit operating conditions; the project also sets material of pipe, which is also unchangeable with the exception of additional adjustments (replacement with higher-grade, cladding). The only parameters, which may be changed in the current device, and thus to reduce significantly the FAC, are chemical parameters of media.

Changes in chemical modes are attractive in particular because they may reduce the damage rate globally in the entire system of the secondary circle and they help to reduce rate of iron transfer, which also slows down plugging of the steam generators and lifetime of ion exchangers at the demineralization is extended. However, it is necessary to point out that adjustments of the chemical mode only slow down the damage rate and do not recover walls of the damaged pipeline. Therefore, the inspections must proceed.

Chemical values, which influence the flow accelerated corrosion rate are as follows:

— Content of oxygen (or redox potential)

— Alkalinity expressed as the cold pH - (measured or calculated value of pH at 25 °C)

— Alkalinity expressed as the hot pH(t) – (calculated value of pH at the temperature t °C).

6.2.1 Influence of pH on FAC

The pH value is the most important factor influencing FAC rate. In order to evaluate pH influence the pH is measured at 25 °C, which corresponds to the common measurement of the cool sample, similarly as in a laboratory. The negative logarithm of hydrogen ions as the pH is defined is in the range from 0 to 14 with neutral value at pH = 7. Also the high-temperature pH value, thereinafter marked as pH(t), is used very often. This value is not usually measurable and considering dissociation of water, which is dependent on the temperature, the scope of values is changing so as the value of neutral pH(t) depending on the temperature; e.g. at the often used temperature of 150 °C in SPP the neutral pH(t) = 5.82 , it means the scope of values of pH(150) is 0.0 – 11.6.

Figure XVII.3 shows general dependence of the flow-accelerated corrosion expressed as the wear of material per pH.

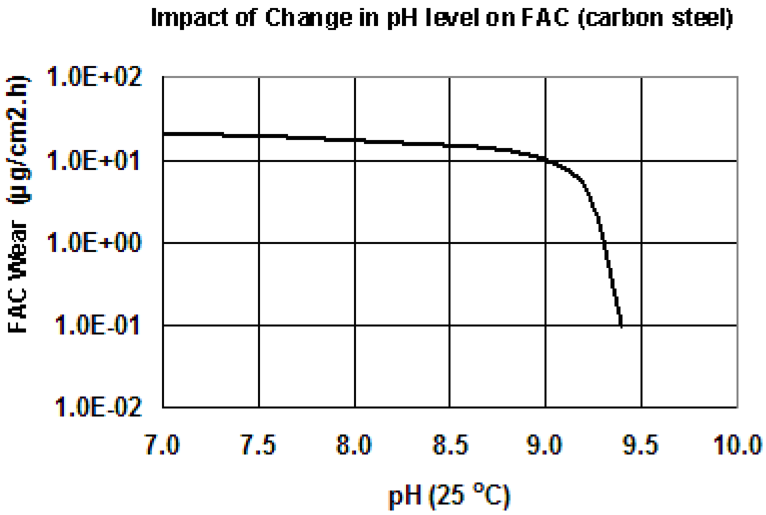

FIG. XVII.3. FAC dependence on pH.

The graph shows that the FAC significantly decreases with pH over 9 and the operator of the block should try to achieve approximately pH = 10, when the flow-accelerated corrosion is the lowest.

6.2.2 Influence of oxygen on FAC

Content of oxygen in a medium pushes the balance in the direction of creation of oxidic layer. This is utilized in the so-called oxygen mode. The cathodic function of oxygen as depolarizer at a suitable concentration of oxygen extent is suppressed and its anodic function is emphasized, i.e. creation of magnetite and oxidation of Fe^{2+} on the surface of the resulting protective oxidic layer. Oxygen in a flowing environment supports creation of a two-layer protective coat on Fe surface, which prevents further dissolution of Fe and rising and carrying corrosion products in the steam pipeline circle. In direct contact of water and Fe a lower topotactic layer of magnetite Fe_3O_4 arises on the Fe surface, which at the activity of oxygen is covered by the top topotactic layer of hematite Fe_2O_3, by means of which the surface is perfectly passivated.

— However, it is necessary to mention the following two facts:

— Oxygen content is explicitly harmful for austenitic materials; it causes pitting, corrosive cracking and other local forms of corrosion. Therefore, there is an effort to eliminate oxygen from steam pipe circuits of WWER blocks physically (showers and air extractors of condensers, thermic degassing) and chemically (hydrazine).

— Oxygen concentration in a block is divided between the steam and water according to the distribution coefficient, which reaches to the value of 104. It means that at boiling there is one part of oxygen in water and 104 parts of oxygen in the steam above this water!

6.2.3 Influence of hydrazine on FAC

Influence of hydrazine N_2H_4 on flow-accelerated corrosion is in different ways. According to various theories the hydrazine shows almost no influence on FAC at lower temperatures. At higher temperatures the influence is significant according to some data. At the concentration less than 150 ppb of hydrazine the FAC rate is decreasing, at higher concentrations above 150 ppb the FAC rate is increasing. However, these concentrations are high above common operating concentrations of hydrazine. Higher concentrations of N_2H_4 are used at wet outages (conservation), when the flow-accelerated corrosion is irrelevant.

6.2.4 Possibilities of FAC reduction in WWER type power plants

The above-described recommendations for reduction of flow-accelerated corrosion (high pH, oxygen concentration) have a lot of restrictions in real WWER block, which is also the second block of nuclear power plant, which are given by conception on one side and used materials of the secondary circle on the second side.

For WWER blocks with brass condensers the maximum long-term permissible value of pH < 9.2 in feed water is being presented. The reason is corrosion of brass. The progress of brass corrosion depends on pH and shows flat minimum with suitable pH interval between 8.5 and 9.2 (see Figure XVII.4).

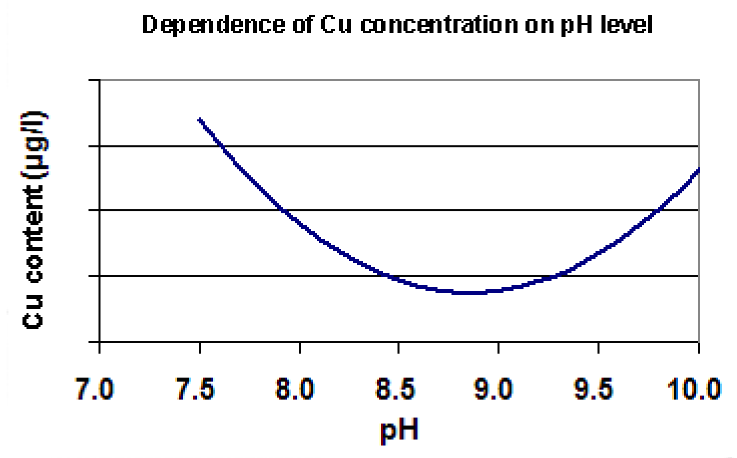

FIG. XVII.4. Copper concentration depending on pH.

In order to suppress the flow-accelerated corrosion not only the pH value is important, but also concentration of alkalinizing agent, which depends on the dissociation constant as the alkalinity rate and the distribution coefficient. The coefficient indicates the concentration of alkalinizing agent in steam or liquid. The most often used ammonium is intensely volatile and tends to remain in steam (90 %). The result is low protection (low pH) in the area of wet steam.

On the other hand, some amines have more suitable distribution coefficients, i.e. they alkalize the water phase more and they pass to steam less. The condensate pH then does not exceed the permitted value pH = 9.2 and it alkalizes the liquid phase more (separate etc.). These amines are often more alkaline than ammonium and less volatile, so a part of amines condenses in separator, in condenser of heating steam, and a large part bypasses the condensate treatment block (CTB). The result is then a suitable pH at low load of ion exchanges in CTB with lower frequency of regenerations.

The following table XVII.5 lists some used amines. Their concentration is selected so that the flow-accelerated corrosion of the most endangered part (SPP) has always high-temperature pH(150) = 6.82 in separate, it means higher by one unit than the neutral pH(t) at the temperature of 150°C. This value is being generally recommended for EPRI materials.

TABLE XVII.5. Concentration of amines and pH in nodes of WWER 440 block necessary for achievement of high-temperature pH(150) = 6,82

Amine	Feed water		Blow-down	Separate from SPP			Condensate	
	ppm	pH(25)	ppm	ppm	**pH(150)**	pH(25)	ppm	pH(25)
NH3	4.64	9.79	1.12	0.80	**6.82**	9.33	5.30	9.82
MPH	3.30	8.98	2.97	5.20	**6.82**	9.09	3.04	8.95
ETA	0.34	8.68	1.08	1.83	**6.82**	9.27	0.11	8.22
DAE	0.19	8.49	0.79	1.17	**6.82**	9.21	0.04	7.80
MPA	1.28	9.09	0.94	2.06	**6.82**	9.27	1.17	9.06
5AP	0.28	8.43	1.77	1.78	**6.82**	9.21	0.03	7.52

Abbreviations:

NH3 ammonium

MPH morpholine

ETA ethanolamine

DAE diaminoethane

MPA 3-methoxypropylamin

5AP 5-aminopentanol

Figure XVII.5 shows comparison of distribution of ammonium and ethanolamine (ETA) in some nodes of the WWER 440 circle at the achievement of the value of the recommended high-temperature pH(150) = 6.82.

The red line marks the marginal pH(25) = 9.2; higher pH(25) at units with brass condensers can't be used for longer period of time.

Table XVII.5 and Figure XVII.5 show that on blocks with brass condensers (copper-base alloy) in secondary circle it is not possible to achieve the recommended value of the high-temperature pH(150) = 6.82 with ammonium.

To be able to operate blocks with high ammoniated alkalinity, at which the flow-accelerated corrosion is significantly suppressed, the blocks are equipped with condensers with pipes without copper-base alloy, i.e. either by titanium or stainless pipes at the simultaneous replacement of technological condensers and removal of all copper-base alloys from the circuit (cladding, valve seat, slide bearing etc.). Then it is possible to choose pH = 10.0 in the feed water. The operation at this pH corresponds to approximately 10 mg/l of ammonium and the unit is operated without adjustment of the condensate.

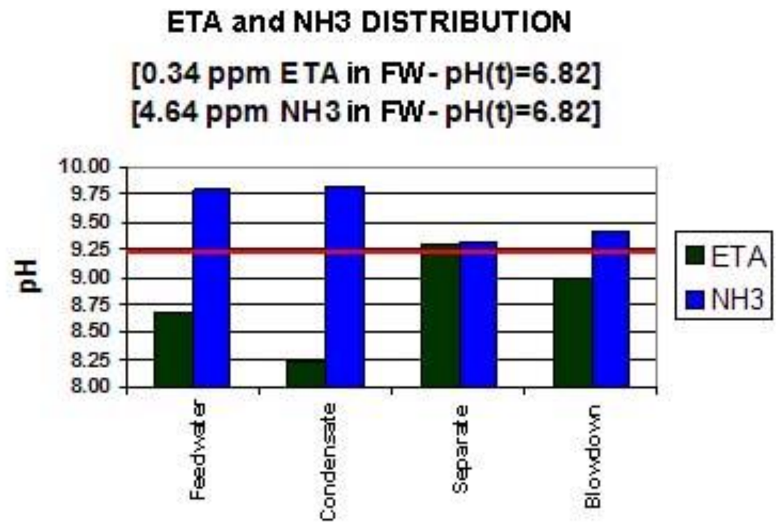

FIG. XVII.5. pH and concentration of ETA and NH3 in feed water necessary for achievement of pH(150) = 6.82.

6.3 Local construction changes

Generally, it is possible to state that influence of constructional changes on reduction of rate of damage caused by flow accelerated corrosion is lower than with material changes and adjustments of chemical mode. For example, if the diameter of the pipe is increased from 300 mm to 350 mm the FAC rate will be reduced only by approximately 20 %. But there are cases, when constructional changes may be more efficient:

— Increased pipe line diameter leads to reduction of rate at regulation valves. Those are usually designed so that at the flow through these valves the diameter of the main pipe line is usually reduced by 60 %, the result of which is increased flow rate. This locally increased flow rate often causes damage to the downstream pipeline. Reconstruction of the system of regulation valves, which leads to reduction of local rate of flow and to reduction of turbulences, thus may significantly reduce damage caused by flow-accelerated corrosion.

— In case of pipe lines with a two-phase medium (wet steam) the damage caused by FAC may be reduces by local decrease of moisture. This can be achieved partly by adjustments leading to better effectiveness of current moisture separators, which will achieve lower quantity of water drops hitting the wall of following pipe components and devices. This leads to a significant reduction of FAC damage, e.g. at pipes from steam separators to the low-pressure turbine or feed water heaters.

7. Quality assurance

All activities linked to implementation of these suggestions and recommendations must be carried out in accordance with a programme for quality assurance prepared in advance. Such programme must be in accordance with principles for quality assurance in a nuclear power plant. The main requirements are as follows:

— Assurance of working processes, check of carried out activities;

— Assurance of independent check of entering and evaluation of input data of all activities connected with the FAC monitoring and prediction programme;

— Creation of a system, which uncovers deficiencies in working processes and activities;

— Creation of suitable organizational structure for control assurance;

- Definition of relation of the quality assurance programme for monitoring and prediction programme to documents on quality assurance in the nuclear power plant;
- Assurance of regular audits for assessment of the quality control application in the FAC monitoring and prediction programme.

8. Conclusions

The goal of the Flow-accelerated corrosion monitoring and prediction programme in term of nuclear power plant operation safety is to ensure with acceptable probability that there will be no losses of integrity of the secondary circuit pressure pipeline important with respect to the nuclear safety during its operation.

For performance of this goal, it is necessary to create sufficient preconditions within the FAC monitoring and prediction programme implementation for realization of the following basic tasks:

- Identification of systems susceptible to flow-accelerated corrosion;
- Determination (prediction) of flow-accelerated corrosion on the given component;
- Identification of pipe components for inspections and use of suitable inspection methods;
- Analysis of measured data including their archiving;
- Provision of basic data before initiation of power plant operation; in case of lack of these data it is necessary to proceed in a substitute method lying for example in assurance of the real state and available operation information as the input data for the first – substitute assessment;
- Carry out assessment of pipe line integrity and its residual life time until the further measurement including recommendation for repairs or replacements of the degraded part of the pipe line based on the assessment criteria.

In the basic scheme of activities, which create an inseparable part of the FAC monitoring and prediction programme, the following actions must be performed:

- Definition of the monitoring and prediction programme for concrete conditions of every block of the nuclear power plant;
- Identification of sub-systems of the secondary circuit susceptible to damage due to flow accelerated corrosion;
- Qualified method of decision-making on the scope of inspections;
- Damage prediction;
- Performance of inspections in the scope necessary for confirmation of predictions and identification of damaged components;
- Method of measured data assessment;
- Decision criteria for further operation of the measured component;
- Controllable system of documentation.

The above listed works cover the minimum scope for implementation of an effective FAC monitoring and prediction programme. This programme may be further supplemented with other parts, which will streamline maintenance and operation. In particular by combination of better chemical modes with utilization of more resistant materials it is possible to reduce damage due to flow accelerated corrosion. The power plant operator should carefully assess these possibilities from technical and financial point of view as well and make the decision on the most suitable method for reduction of FAC impacts.

Responsibility for performance of the monitoring and prediction programme is held by the nuclear power plant operator. It is useful to create such organizational conditions, in which responsibilities and relations of workers performing such programme to further organizational departments or organizations are given unambiguously. It is also suitable to specify obligations of other departments in relation to the flow accelerated corrosion monitoring and prediction programme.

References

[1] Chexal, B., et al., *Flow-Accelerated Corrosion in Power Plants*, TR 106611, EPRI, Palo Alto, CA (1996).

[2] ELECTRIC POWER RESEARCH INSTITUTE, *Recommendations for an Effective Flow-Accelerated Corrosion Program* (NSAC-202L-R3), TR 1011838, EPRI, Palo Alto, CA (2006).

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[7] ELECTRIC POWER RESEARCH INSTITUTE, *Flow-Accelerated Corrosion Investigations of Trace Chromium*, TR 1008047, EPRI, Palo Alto, CA (2003).

Appendix 1: Scheme of the Flow-accelerated corrosion prediction programme

Initial activities before modeling and measurement

+ Working procedures

+ Procedures ensuring QA

Collection of secondary circuit data
(drawings, thermodynamic and chemical parameters, limiting criteria)

↓

Creation of thermodynamic and chemical model of the secondary circuit

↓

Selection of lines for assessment

→ FAC problems are not expected

→ FAC problems are possible

→ FAC is very probable

↓

First predictive analysis of selected lines

Activities before outage

Predicted analysis
provided data from measurement are available

↓

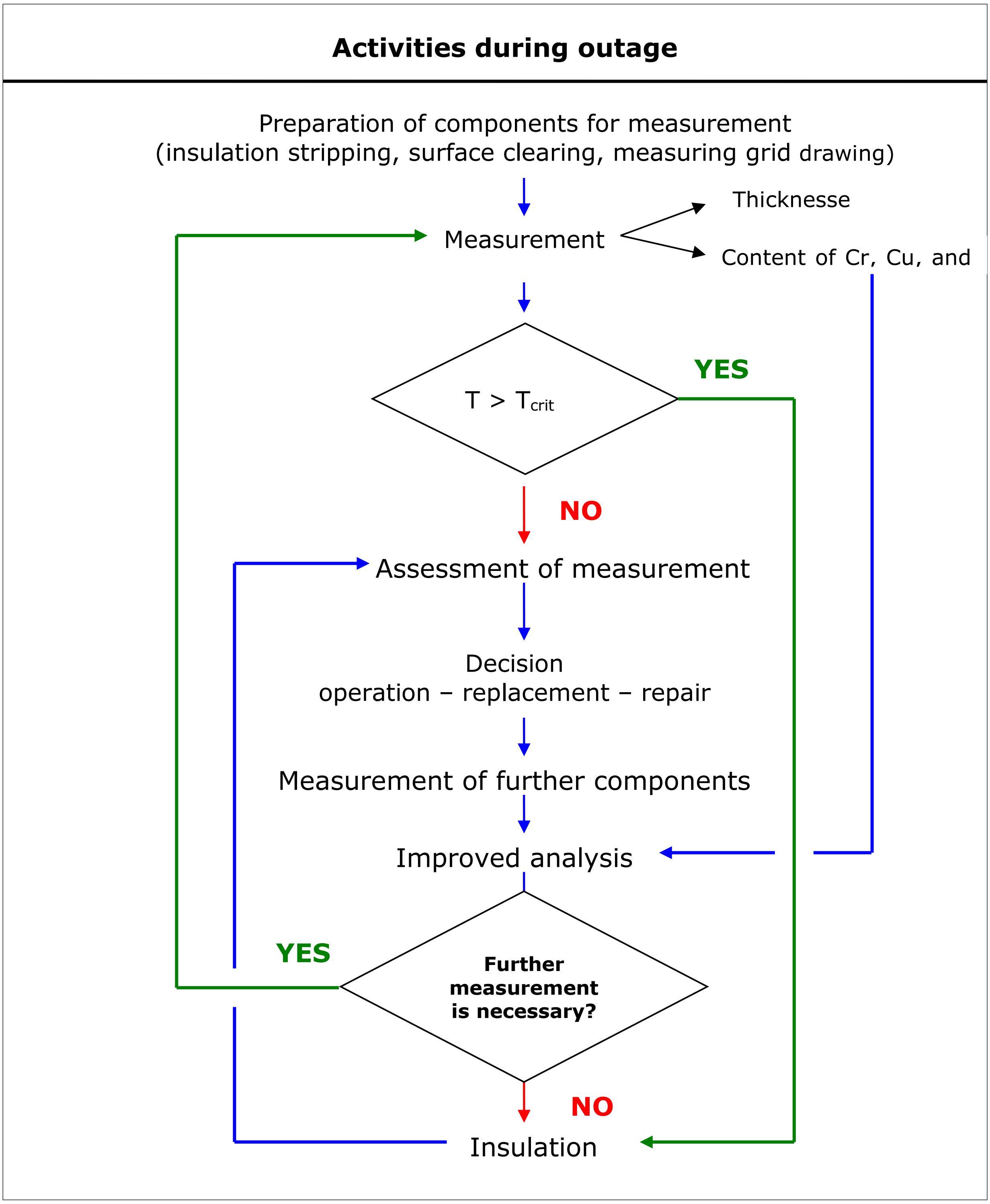
Selection of components for inspections

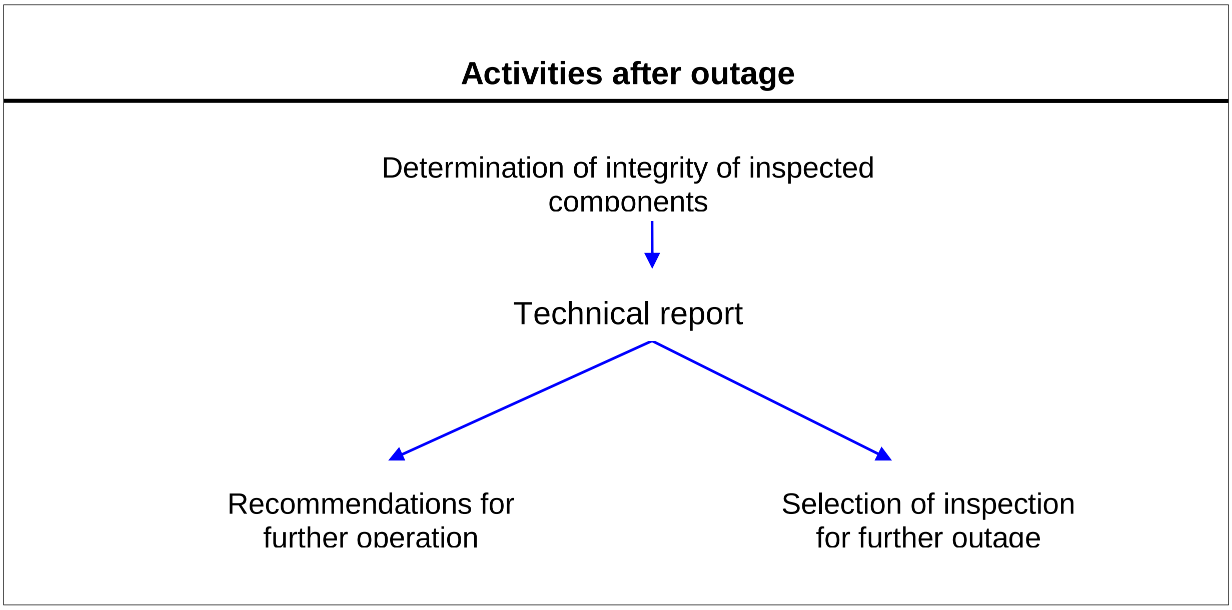
↓

T_{crit}, determination of criteria for further assessment

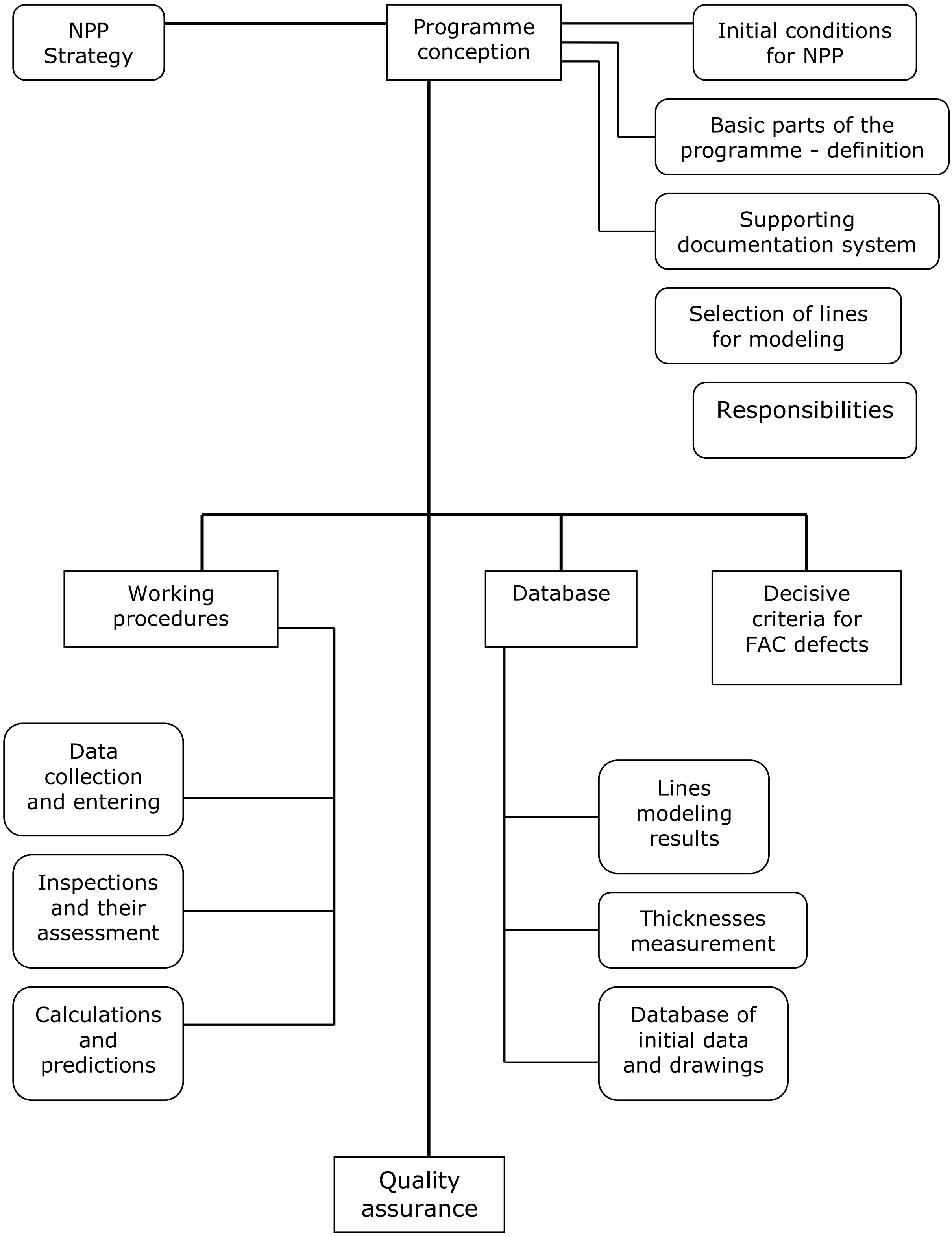
↓

Verification of selection of components


Activities during outage
Preparation of components for measurement
(insulation stripping, surface clearing, measuring grid drawing)
Measurement
Thicknesse
Content of Cr, Cu, and
T > Tcrit
YES
NO
Assessment of measurement
Decision
operation – replacement – repair
Measurement of further components
Improved analysis
YES
Further measurement is necessary?
NO
Insulation


Activities after outage
Determination of integrity of inspected components
Technical report
Recommendations for further operation
Selection of inspection for further outage

Appendix 2: Documentation system for FAC monitoring and prediction programme



APPENDIX XVIII

(Recommended)

Recommendations on calculation of cyclic strength, taking into account effect of water environment

1. General assumptions (specifications)

1.1. Recommendations concern components and piping made of perlite-type steels (carbon steels, low-alloy steels, and low-alloy Cr-Mo-V steels) as well as Cr-Ni corrosion-resistant steels of austenitic type with $R_m^{20} \leq 700$ MPa, which are subject to loading at temperatures below 350°C and are in contact with water environment of light water reactors.

1.2. To the factors affecting cyclic strength of carbon and low-alloy steels and their welding joints belong:

— Sulphur content S in the metal, temperature T of the cycle, strain rate $\dot{\varepsilon}$ in the tension half-cycle of stress intensity (reduced stress), and oxygen concentration (KO) in water environment.

To the factors affecting cyclic strength of Cr-Ni corrosion-resistant steels of austenitic type, and their welding joints belong:

— Temperature of the cycle, strain rate in the tension half-cycle of stress intensity (reduced stress), and oxygen concentration (KO) in water environment.

1.3. Calculation is performed according to the formulas (XVIII.1) and (XVIII.2), taking into account coefficient of cyclic strength reduction due to water environment $F_{pn}(F_{pn} \leq 1)$.

For loading conditions in contact with water environment, with different cyclic loads and loading by high-frequency stresses, application of recommendations presented in this Appendix is conditional and should be subject to experimental verification.

1.4. If during calculation (for a particular application) the conditions for cyclic strength are not met, but this finding is not in accord with data obtained from checking the calculated construction element, in the examined area of which no cracks were found, then, based on the results of calculation, this area of construction element is added to the set of construction element areas that are subject to periodic inspections during continuing operation. Scope and periodicity of the inspections shall be defined in accord with calculation of allowable sizes of the defects.

2. Calculation formulas

2.1. Allowable amplitude of conditional (Hook´s) elastic stress $[\sigma_{aF}]$ or allowable number of cycles $[N_0]$ for $[N_0] \leq 10^{12}$ are equal to minimum of values determined according to the formulas (XVIII.1) and (XVIII.2), as well as formulas (XVIII.6) and (XVIII.9):

$$[\sigma_{aF}] = E^T e_c^T \left(4n_N [N_0]\right)^{-m} + \left(R_c^T - [\sigma_{F\max}] i_\sigma\right)\left[\left(4n_N [N_0]\right)^{m_e} - i_\sigma\right]^{-1} \tag{XVIII.1}$$

where

$[\sigma_{F\max}] = R_p^T$ for $[\sigma_{F\max}] \geq R_p^T$

$i_\sigma = 0$ for $[\sigma_{aF}] \geq [\sigma_{F\max}]$ or $[\sigma_{aF}] \geq R_p^T$;

$i_\sigma = 1$ for $[\sigma_{aF}] < [\sigma_{F\max}] < R_p^T$;

$$[\sigma_{aF}] = \left\{E^T e_c^T \left(4[N_0]\right)^{-m} + \left(R_c^T - n_\sigma [\sigma_{F\max}] i_\sigma\right)\left[\left(4[N_0]\right)^{m_e} - i_\sigma\right]^{-1}\right\} n_\sigma^{-1} \tag{XVIII.2}$$

using:

$n_\sigma\left[\sigma_{F\max}\right]=R_p^T$ for $n_\sigma\left[\sigma_{F\max}\right]\geq R_p^T$

$i_\sigma=0$ for $\left[\sigma_{aF}\right]\geq\left[\sigma_{F\max}\right]$ or $n_\sigma\left[\sigma_{aF}\right]\geq R_p^T$;

$i_\sigma=1$ for $n_\sigma\left[\sigma_{aF}\right]<n_\sigma\left[\sigma_{F\max}\right]\leq R_p^T$;

n_σ, n_N – safety coefficient with respect to stress and number of cycles

m, m_e – metal characteristics;

R_c^T – strength characterization, equal to:

$$R_c^T=R_m^T\left(1+1,4\cdot 10^{-2}Z^T\right) \tag{XVIII.3}$$

e_c^T – plasticity characterization, depending on Z_c^T, determined as:

$$e_c^T=1,15\lg\frac{100}{100-Z_c^T}-\frac{\left|\left(\sigma_F^*\right)_{\max}\right|-R_{p0,2}^T}{2E^T} \tag{XVIII.4}$$

or when $\left(\sigma_F^*\right)_{\max}<R_{p0,2}^T$ then:

$$e_c^T=1,15\lg\frac{100}{100-Z_c^T} \tag{XVIII.5}$$

$$\left[\sigma_{aF}\right]=E^Te_C^{20}\left(4n_NF_{pn}\left[N_0\right]\right)^{-0,5}+R_{cF}^T\left[\left(4n_N\left[N_0\right]\right)^{m_{eF}}\right]^{-1}, \tag{XVIII.6}$$

where e_C^{20} is determined according to formulas (XVIII.7) and (XVIII.8), provided that guaranteed values $Z^T=Z^{20}$ are used (if $Z^{20}\geq 50\%$, then it is necessary to put $Z^{20}=50\%$).

$$e_c^T=1,15\lg\frac{100}{100-Z_c^T}-\frac{\left|\left(\sigma_F^*\right)_{\max}\right|-R_{p0,2}^T}{2E^T} \tag{XVIII.7}$$

or for $\left(\sigma_F^*\right)_{\max}<R_{p0,2}^T$ according to:

$$e_c^T=1,15\lg\frac{100}{100-Z_c^T} \tag{XVIII.8}$$

and

$$\left[\sigma_{aF}\right]=\left[E^Te_C^{20}\left(4F_{pn}\left[N_0\right]\right)^{-0,5}+R_{cF}^T\left(\left(4\left[N_0\right]\right)^{m_{eF}}\right)^{-1}\right]n_\sigma^{-1}, \tag{XVIII.9}$$

where n_σ, n_N are taken equal to 2 with respect to stress and 10 with respect to number of cycles.

Calculation in which maximum effect of mean cycle stress is taken into account, is performed according to formulas (XVIII.1) and (XVIII.2) with $[\sigma_{Fmax}]=R_p^T$, provided that conditions for i_σ are met, where value R_p^T is defined in accordance with ((XVIII.1) and ((XVIII.2).

2.2. Values of R_{cF}^T, m_{eF} in formulas (XVIII.6) and (XVIII.9) are defined, taking into account effect of water environment, according to the following formulas:

$$R_{cF}^T = R_m^T\left(1+0,014Z_F\right), \qquad \text{(XVIII.10)}$$

$$m_{eF} = 0,132\lg\left[2,5\left(1+0,014Z_F\right)\right], \qquad \text{(XVIII.11)}$$

where

$$Z_F = 100\left\{1-\left[\exp\left(2e_C^{20}F_{pn}^{-0,5}\right)\right]^{-1}\right\} \qquad \text{(XVIII.12)}$$

if guaranteed values Z^{20} are used;

$$Z_F = Z^{20}F_{pn}^{-0,5} \qquad \text{(XVIII.13)}$$

if actual (real) values Z^{20} are used.

2.3. Coefficient F_{pn} is defined according to formulas:
for carbon steels

$$F_{pn} = \exp\left(0,912-0,101S^*T^*O^*\varepsilon^*\right); \qquad \text{(XVIII.14)}$$

for low alloy steels and steels of Cr-Mo-V type

$$F_{pn} = \exp\left(1,031-0,101S^*T^*O^*\varepsilon^*\right), \qquad \text{(XVIII.15)}$$

where in formulas (14) and (15)

$S^* = 0,015$ – for $\mathrm{KO} > 1,0$ mg/kg;

$S^* = S\%$ – for $\mathrm{KO} \le 1,0$ mg/kg and $0 < S \le 0,015$ %;

$S^* = 0,015$ – for $\mathrm{KO} < 1,0$ mg/kg and $S > 0,015$ %;

$T^* = 0$ – for $T < 150$ °C;

$T^* = T-150$ – for $150 < T \le 350$ °C; (XVIII.16)

$O^* = 0$ – for $\mathrm{KO} \le 0,05$ mg/kg;

$O^* = ln\frac{\mathrm{KO}}{0,04}$ – for $0,05 < \mathrm{KO} \le 0,5$ mg/kg;

$O^* = \ln 12,5$ – for $\mathrm{KO} > 0,5$ mg/kg;

$\varepsilon^* = 0$ – for $\dot{\varepsilon} \ge 1$% s^{-1};

$\varepsilon^* = \ln\dot{\varepsilon}$ – for $10^{-3} < \dot{\varepsilon} < 1$% s^{-1};

$\varepsilon^* = \ln 0.001$ – for $\dot{\varepsilon} \le 0.001$ % s^{-1},

Cr-Ni corrosion-resistant steels of austenitic type

F_{pn} = exp (-T*O*ε*) (XVIII.17)

where

$T^* = 0$ – for $T < 150$ °C;

$T^* = \frac{T-150}{175}$ – for $150 \le T < 325$ °C;

$T^* = 1$ – for $T \ge 325$ °C;

O* = 0.395 for all values of KO; (XVIII.18)

$\varepsilon^* = 0$ – for $\dot{\varepsilon} \ge 0.4$% s^{-1};

$\varepsilon^* = \ln\left(\frac{\dot{\varepsilon}}{0.4}\right)$ – for $4.10^{-4} < \dot{\varepsilon} < 0.4\%$ s^{-1};

$\varepsilon^* = \ln 0.001$– for $\dot{\varepsilon} \leq 0.4\%$ s^{-1}.

Content of sulphur in steel of perlitic type is determined according to Passport or Technical documentation, temperature T is determined as equal to maximum temperature in the tension half-cycle of the interval where stress intensity varies, $[\sigma_{F\max}]$ equals to R_{-1} for asymmetrical cycle ($[\sigma_{Fmax}]]>[\sigma_{aF}]$), and $[\sigma_{aF}]$equals to R_{-1} for symmetrical cycle, strain rate $\dot{\varepsilon}$ is determined as equal to its minimum value in the tension half-cycle of stress intensity variation in the same interval as for determination of T, oxygen concentration KO is equal to its maximum value over regimes defining tension half-cycle for steels of perlitic type; and for Cr-Ni corrosion resistant steels of austenitic type, KO is equal to its minimum value in the same interval of stress intensity variation as for determination of T.

Method of taking into account variation of both T and $\dot{\varepsilon}$ during the cycle, permitting reducing conservatism of the calculation, has to be still developed.

2.4. If value $[\sigma_{aF}] = [\sigma_{aF}]_*$ for given $[N_0]$ is determined according to formulas (XVIII.6) and (XVIII.9), then for weld joint allowable stress amplitude is equal to

$$[\sigma_{aF}]_s = min\begin{Bmatrix}\phi_s[\sigma_{aF}] \\ [\sigma_{aF}]_*\end{Bmatrix}, \quad \text{(XVIII.19)}$$

where $[\sigma_{aF}]$ is allowable stress amplitude according to formulae (XVIII.1) and (XVIII.2)

Allowable number of cycles $[N_0]$ for given stress amplitude $[\sigma_{aF}]$

$$[N_0] = min\begin{Bmatrix}[N_0]for[\sigma_{aF}] = [\sigma_{aF}]/\phi_5 & \text{according to formulae (XVIII.1) and (XVIII.2)} \\ [N_0]for[\sigma_{aF}] & \text{according to formulae (XVIII.6) and (XVIII.9)}\end{Bmatrix}$$

3. Calculated fatigue curve with including effect of water environment

Calculated fatigue curves for carbon and low-alloy Cr-Mo-V type steels for temperature 300°C for steels of perlite type, and also $Z^{20} \geq 35\%$– carbon steels, $Z^{20} \geq 50\%$ – low-alloy Cr-Mo type steels, taking into account maximum effect of mean cycle stress and water environment in accord with conditions (XVIII.14) and (XVIII.16) are seen in Figures XVIII.1 to XVIII.4 where it is assumed that$S \geq 0{,}015\%$, $T = 300$°C, KO $\leq 0{,}05$mg/kg, $\dot{\varepsilon} \leq 0.001$ % s^{-1}.

Calculated curve for Cr-Ni corrosion-resistant steels of austenitic type for temperature 350°C for values $R_m^T \geq 350$ MPa; $R_{p0,2}^{20} = 200$ MPa, $\frac{R_{p0,2}^T}{R_m^T} \geq 0{,}4$; $Z^{20} \geq 40\%$ and $E^T \geq 175$ GPa, taking into account minimum effect of mean cycle stress and water environment in accord with conditions (XVIII.17) and (XVIII.18) is seen in Figure XVIII.5, where it is assumed that $T = 350$ °C, $\dot{\varepsilon} \leq 4.10^{-4}$ % s^{-1}.

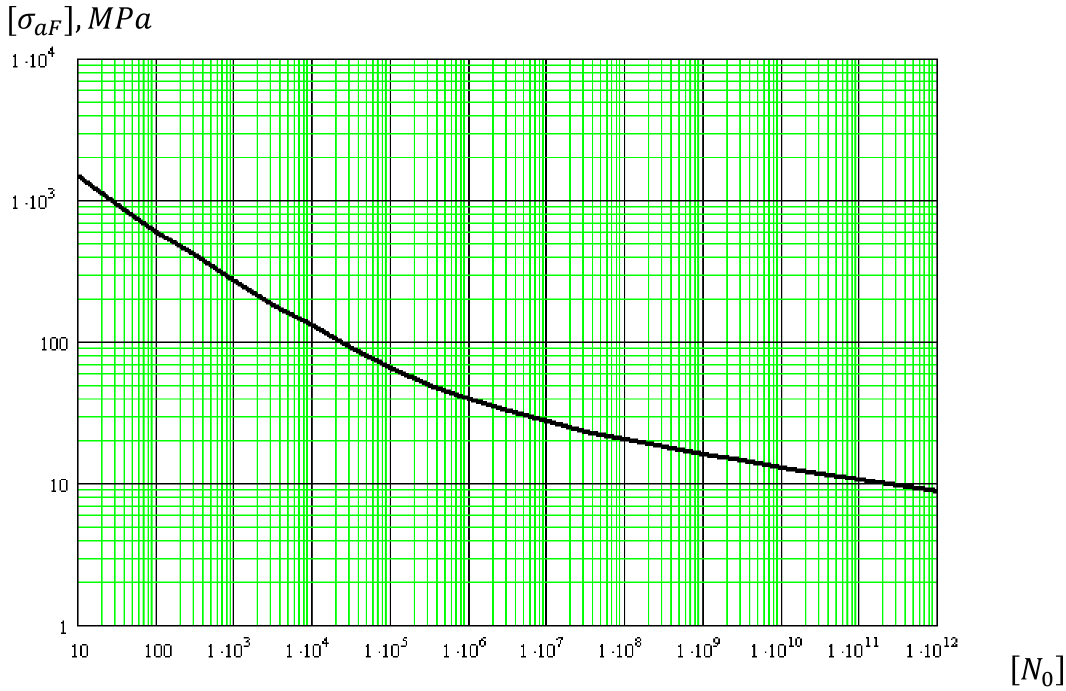


FIG. XVIII.1. Calculated fatigue curve for carbon steels with $R_m^T \geq 300$ MPa, $R_{p0,2}^{20} \leq 200$ MPa, $R_{p0,2}^T / R_m^T \geq 0,5$, $Z^T \geq 30\%$, $E^T \geq 175$ GPa, $Z^{20} \geq 35\%$, KO $\leq 0{,}05$mg/kg, $T \leq 300$°C.

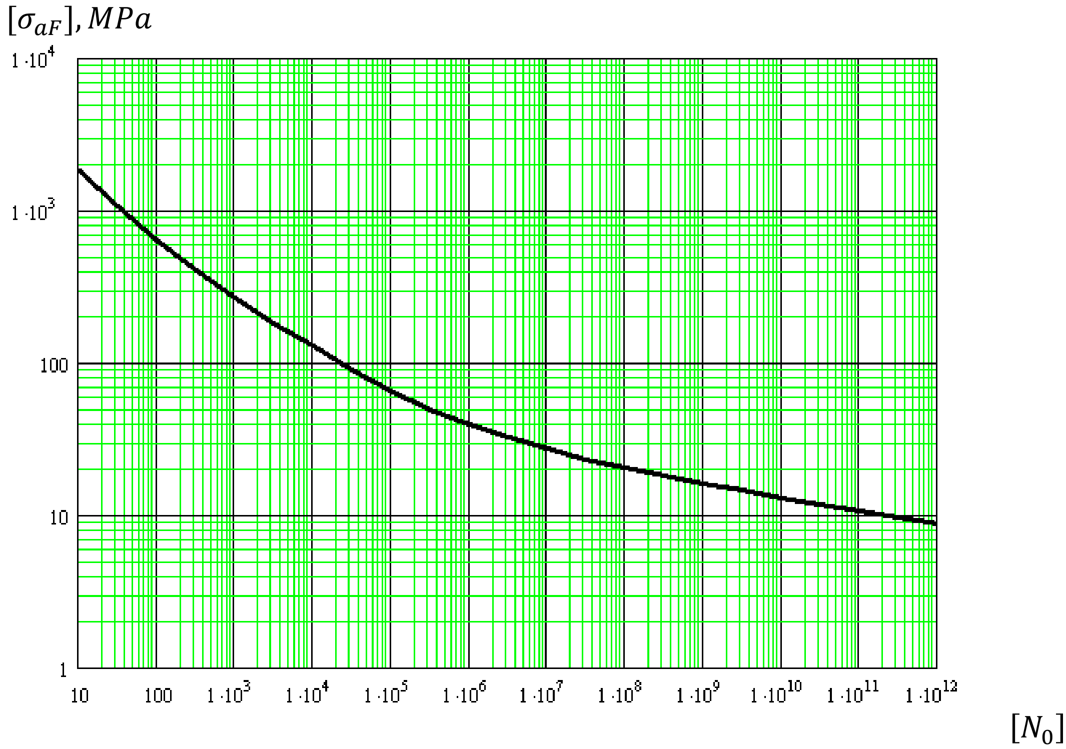


FIG. XVIII.2. Calculated fatigue curve for low-alloy Cr-Mo steels with $R_m^T \geq 300$ MPa, $R_{p0,2}^{20} \leq 200$ MPa, $R_{p0,2}^T / R_m^T \geq 0,5$, $Z^T \geq 30\%$, $E^T \geq 175$ GPa, $Z^{20} \geq 50\%$, KO $\leq 0{,}05$ mg/kg, $T \leq 300$ °C.

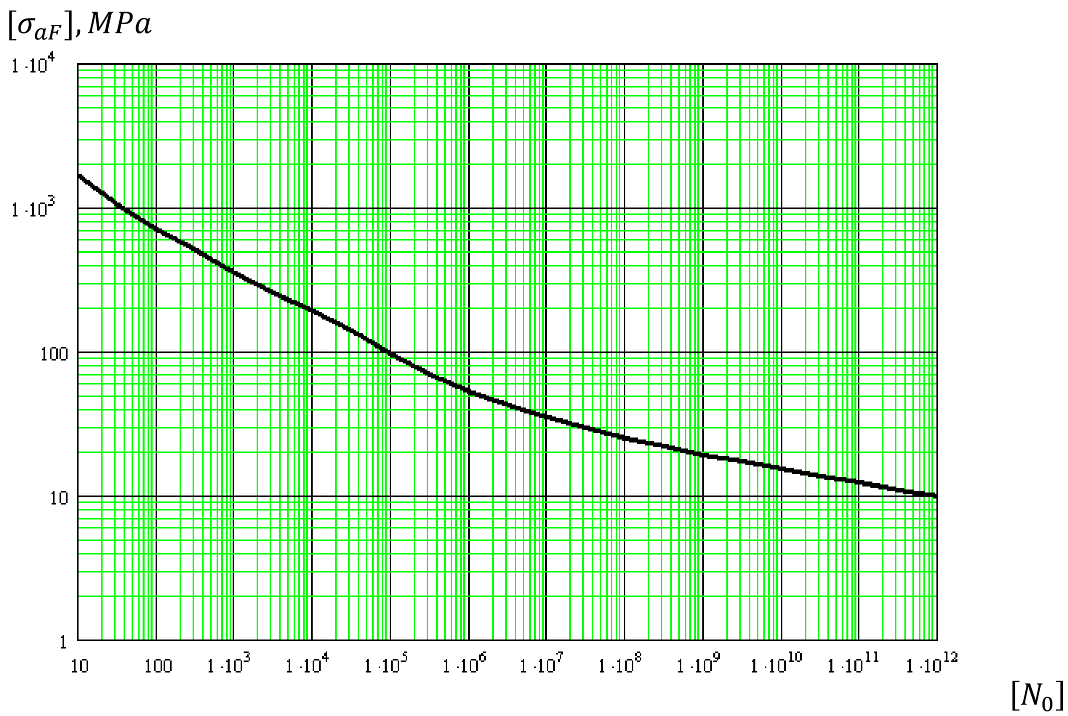


FIG. XVIII.3. Calculated fatigue curve for carbon steels with $R_m^T \geq 400$ MPa, $R_{p0,2}^{20} \leq 400$ MPa, $\frac{R_{p0,2}^T}{R_m^T} \geq 0{,}6$, $Z^T \geq 45\%$, $E^T \geq 190$ GPa, $Z^{20} \geq 35\%$, KO $\leq 0{,}05$ mg/kg, $T \leq 300°$C.

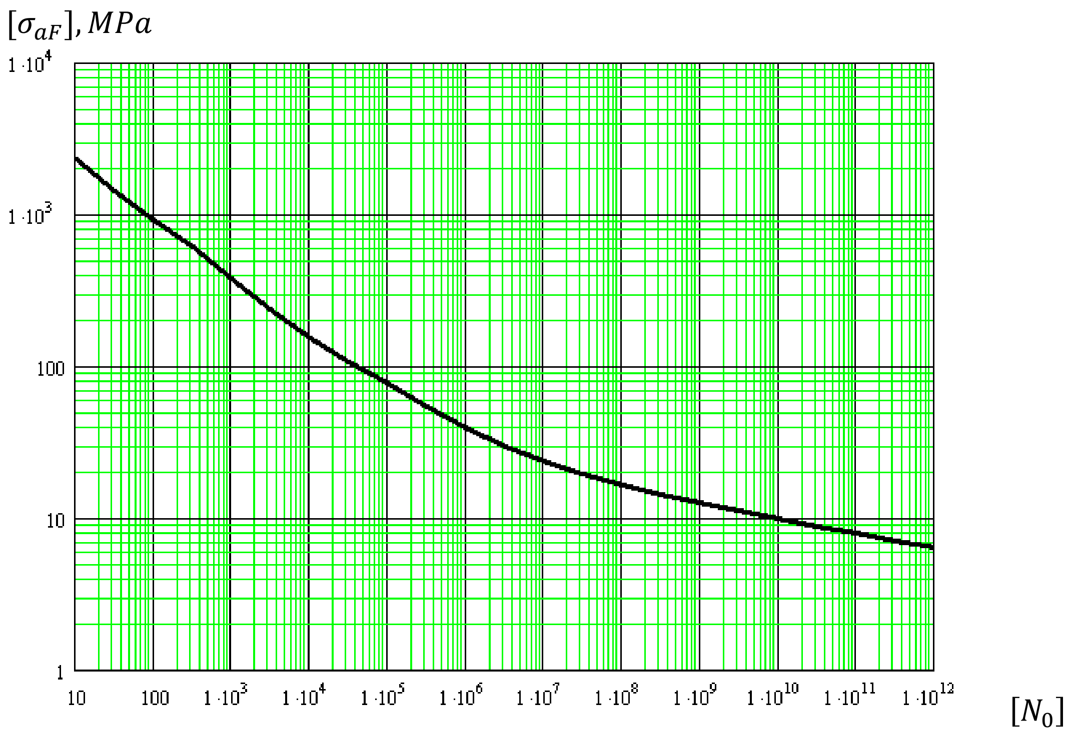


FIG. XVIII.4. Calculated fatigue curve for low-alloy Cr-Mo steels with $R_m^T \geq 400$ MPa, $\frac{R_{p0,2}^T}{R_m^T} \geq 0{,}6$, $Z^T \geq 45\%$, $E^T \geq 190$ GPa, $Z^{20} \geq 50\%$, KO $\leq 0{,}05$ mg/kg, $T \leq 300°$C

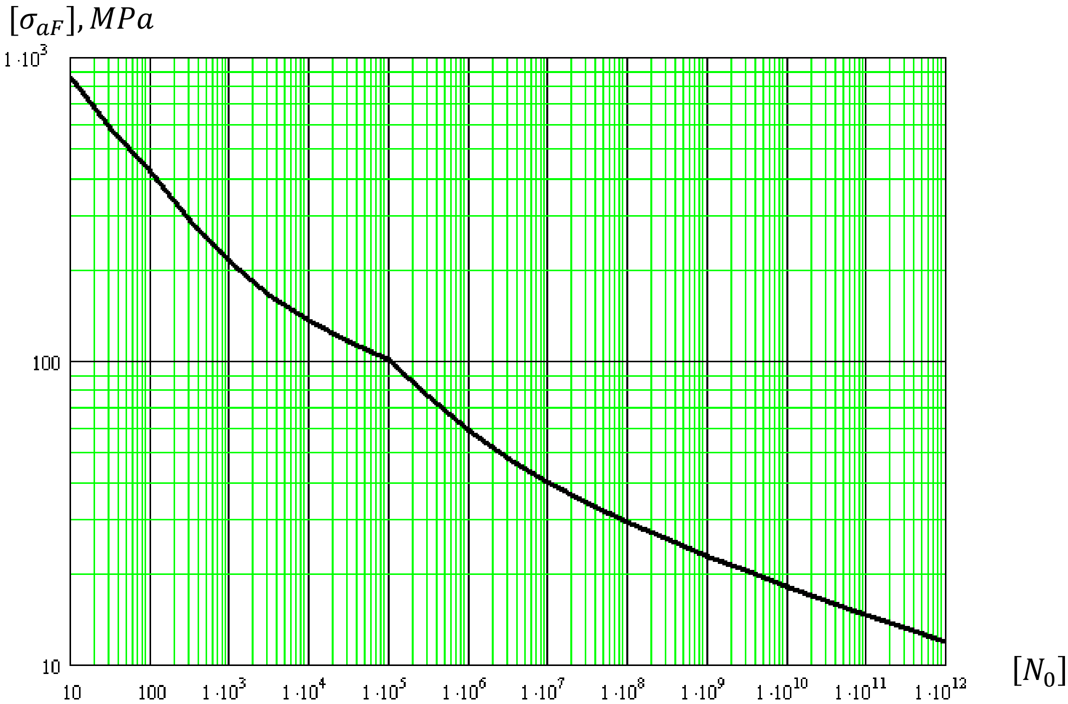


FIG. XVIII.5. Calculated fatigue curve for Cr-Ni corrosion-resistant steels of austenitic type with $R_m^T \geq 350$ MPa, $\frac{R_{p0,2}^T}{R_m^T} \geq 0{,}4$, $R_{p0,2}^{20} \leq 200$ MPa, $Z^T \geq 45\%$, $E^T \geq 175$ GPa, $Z^{20} \geq 40\%$, $T \leq 350$°C (in water environment).

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