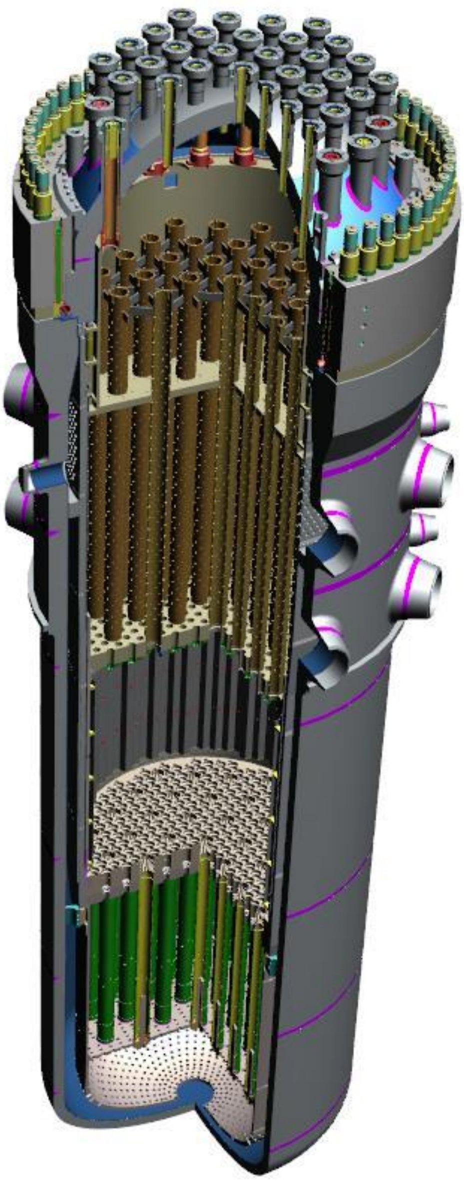
European Commission

VERLIFE Guidelines for Integrity and Lifetime Assessment of Components and Piping in WWER Nuclear Power Plants - Appendices A-F



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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 Appendices A-F of the VERLIFE Guidelines:

- Appendix A: Leak Before Break (LBB) Concept application to the Selected Piping systems of WWR-type NPPs
- Appendix B: No-Break-Area Assessment for WWER Piping
- Appendix C: Integrity and Lifetime Assessment Procedure of RPV Internals in WWER-type NPPs during Operation
- Appendix D: Risk-Informed In-Service Inspection
- Appendix E: NDE System Qualification
- Appendix F: Component and Piping Supports

This publication is linked to publication JRC 138449, which is the main document of the VERLIFE Guidelines. The latter contains the main chapters and all other appendices, Appendices I – XVIII, of the guidelines. It also contains a description of the structure of the VERLIFE Guidelines and a description of their drafting process.

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.

Introduction

VERLIFE - Guidelines for Integrity and Lifetime Assessment of Components and Piping in Nuclear Power Plants with WWER operating Reactors 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 much 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 respective national regulatory authority.

This publication contains Appendices A-F of the VERLIFE Guidelines

- Appendix A: Leak Before Break (LBB) Concept application to the Selected Piping systems of WWR-type NPPs
- Appendix B: No-Break-Area Assessment for WWER Piping
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APPENDIX A

LEAK BEFORE BREAK (LBB) CONCEPT APPLICATION TO THE SELECTED PIPING SYSTEMS OF WWER TYPE NPPS

(Proposals for the procedure of application of LBB concept to the high- energy piping of WWER-440 and WWER-1000 NPPs)

1. DEFINITIONS AND ABBREVIATIONS

1.1. DEFINITIONS

LBB concept

A design and operational principle in which available in-service inspection, maintenance and monitoring systems (including leak detection), give early and reliable warning of piping component defects so that necessary actions may be undertaken in a timely manner to to avoid component failures as a consequence of any design basis condition.

1.2. ABBREVIATIONS

ASME	American Society of Mechanical Engineers
IAEA	International Atomic Energy Agency
ISI	In-service inspection
LBB	Leak before break
MIV	Main isolation valve
MSH	Main steam header
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission
NRI	Nuclear Research Institute
RG	Regulatory guide (US NRC)
SRP	Standard review plan
SSC	Systems, structures and components
SSE	Safe shutdown earthquake
WWER	Water-water energetic reactor (Soviet designed PWR)

2. LBB APPLICATIONS PURPOSES

2.1 Systems, structures, and components (SSCs) important to safety should be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, anticipated transients and postulated accidents (including loss-of-coolant accidents). These SSCs should be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures or from events and conditions outside of the nuclear power unit. When an LBB analysis has been conducted and approved, the specific dynamic effects associated with such postulated pipe may be excluded from the design basis.

2.2 Modern safety requirements associated with the elimination of high-energy piping breaks may only have been partially addressed in the original design or older NPPs. Subsequent design modifications typically resulted in the installation of pipe whip restraints and barriers against effects of fluid jet impingement, primarily from broken main coolant piping. The selection of locations for installation of this equipment and their design were based on conservative approaches. At present there are two complex solutions available for the purpose reduction of probability of high-energy piping rupture. The first one, LBB analysis, is based on the confirmation of an extremely low probability of pipe rupture under conditions consistent with the design basis for the piping. A deterministic evaluation of the piping system that demonstrates sufficient margins against failure, including verified design, verified fabrication, and an adequate inservice inspection programme, can be assumed to satisfy the extremely low probability criterion.

2.3 Successfully applied leak-before-break (LBB) concept permits:

- To remove (or not install) protective hardware such as pipe whip restraints and jet impingement barriers;
- Redesigned pipe connected components, their supports and their internals;
- Other related changes in plants (e.g. steam generator and reactor coolant pump snubber requirements due to thrust loads).

2.4 Requirements for containment, environmental qualification of electrical and mechanical equipment, emergency core cooling system performance are not affected by LBB.

2.5 When applying LBB, dynamic loads to internals and supports of equipment due to relevant pipeline rupture need not to be taken into account.

3. REQUIREMENTS FOR THE APPLICATION OF LBB CONCEPT

3.1 LBB analyses should demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. A deterministic evaluation of the piping system that demonstrates sufficient margins against failure, including verified design, verified fabrication and an adequate inservice inspection programme, can be assumed to satisfy the extremely low probability criterion.

3.2 Specific guidance regarding the piping systems that are qualified to be considered for LBB application, fracture mechanics analyses of postulated pipe cracks, and leak detection system capability, are provided to ensure that the probability of pipe rupture is extremely low.

3.3 LBB evaluation uses design basis loads and is based on as-built piping configurations, as opposed to the design configuration. Correct location of supports and their characteristics (such as gaps) is verified, as are weights and locations of components such as valves. Particular attention is to be given to the reliability of snubbers, whose failure could invalidate the stresses used in fracture mechanics evaluations.

3.4 LBB should only be applied to high-energy piping that is designed and manufactured according to requirements for groups A and B [6] or for safety class 1 or 2 according to the Safety Analysis Report (or equivalent) for the relevant NPP. Applications to other high-energy piping will be considered based on an evaluation of the proposed design and in-service inspection requirements as compared to safety class 1 and 2 according to the Safety Analysis Report or to groups A and B according to „Pravila ustrojstva”. Approval of the elimination of dynamic effects from postulated pipe ruptures is obtained individually for particular piping systems at specific nuclear power units.

3.5 LBB is applicable only to an entire piping system or analyzable portion thereof. LBB cannot be applied to individual welded joints or other discrete locations. Analyzable portions are typically segments located between piping anchor points. In the case of partially pressurized piping the piping is modeled between anchor points as well, however leaking throughwall cracks are postulated only in the pressurized portion. A typical case of this is piping attached to the main circulation loop (MCL), especially ECCS, which are pressurized only as far as the check valve during normal operation of the unit.

3.6 It should be demonstrated that direct and indirect pipe failure mechanisms and other degradation sources could not challenge the integrity of piping considered for LBB application.

3.6.1. An assessment of potential indirect sources of pipe rupture is required to demonstrate that indirect failure mechanisms defined in the plant Final Safety Analysis Report (FSAR) will not cause pipe ruptures with very high probability. Indirect failure mechanisms include seismic events accidents resulting from human error, fires, or flooding, and failures of SSCs in close proximity to the applicable safety related piping. Compliance with snubber surveillance requirements in the technical specifications ensures that snubber failure probability is acceptably low.

3.6.2. Potential for degradation by erosion, erosion/corrosion, erosion/cavitation and other relevant damage mechanisms should be evaluated. Industry experience for specific piping systems plays an important role in evaluation of these degradation mechanisms. Additionally, an evaluation of wall thinning at elbows and other fittings is undertaken to ensure that requirements for minimum wall thickness are met. These evaluations should demonstrate that these mechanisms are not potential sources of pipe rupture.

3.6.3. Material susceptibility to corrosion, potential for high residual stresses, and environmental conditions that could lead to degradation by stress corrosion cracking should be evaluated. Industry experience for specific piping systems plays an important role in evaluation of this degradation mechanism.

3.6.4. Corrosion resistance of piping, which can be demonstrated by the frequency and degree of corrosion in the specific piping system, should be evaluated. Modifications to operating conditions (e.g., controlling water chemistry) or design changes (e.g., replacing piping material) are measures that can be taken to improve corrosion resistance in piping. It should be possible to recognize that remedial residual stress improvement treatments are effective in reducing susceptibility to stress corrosion cracking.

3.6.5. LBB is typically not applicable to piping susceptible to intergrannular-stress-corrosion cracking (IGSCC) or primary water stress corrosion cracking (PWSCC). If it can be demonstrated through analysis, data or operational experience, that effective mitigation measures are taken in place to counteract these mechanisms, then LBB may be considered.

3.6.6. LBB is not applicable to piping systems with a history of fatigue cracking or failure. An evaluation to ensure that potential pipe ruptures due to thermal and mechanical induced fatigue is unlikely should be performed. LBB may be applied if it can be demonstrated that (a) the effect of mixing of low and high temperature fluids (which can cause cyclic thermal stresses) has a low influence on fatigue usage factor and has been taken into account, and (b) there is no potential for vibration-induced fatigue cracking or failure.

3.6.7. LBB is not applicable to piping systems for which pipe rupture due to water hammer is likely. It should be demonstrated that the potential for water hammer in the candidate piping systems is very low. Water hammer is a generic term, which includes various unanticipated hydrodynamic events. Historical frequencies of water hammer events in specific piping systems, coupled with reviews of operating procedures and conditions, may be used to demonstrate that water hammer is not a significant contributor to pipe rupture. Alternatively, design changes such as the use of J-tubes, vacuum breakers, and jockey pumps coupled with improved operating procedures, can be used to reduce concerns related to water hammer. It should be possible to establish that any measures needed to abate water hammer frequency and magnitude will be effective for the life of the plant.

3.6.8. LBB is not applicable to piping systems subject to creep mechanisms. Operation below 350°C for ferritic steel piping and below 450°C for austenitic steel piping can exclude creep concerns.

3.6.9. LBB is not applicable to piping systems for which brittle fracture is possible. It should be demonstrated that piping material is not susceptible to brittle cleavage-type failure over the full range of system operating temperatures. The necessary condition for this is that the piping is operated at temperature corresponding to the upper shelf of the material transition curve.

3.6.10. The adequacy of leakage detection systems associated with the reactor coolant system should be evaluated. Determination of leakage from a piping system under pressure involves uncertainties, and therefore margins are needed. Sources of uncertainties include plugging of leakage cracks with particulate material over time, leakage prediction, measurement techniques, personnel, and frequency of inspections. Leakage detection systems are evaluated to determine whether they are sufficiently reliable, diverse and sensitive so that a sufficient margin for detection of unidentified leakage exists for through-wall flaws in support of deterministic fracture mechanics evaluations. Application of LBB to piping systems outside containment/confinement would require that the applicant demonstrate that leakage detection systems are available to provide required reliability, redundancy, and sensitivity.

3.6.11. Unless a detailed justification that accounts for the effects of these sources of uncertainty in leakage measurements can be presented, a multiplication margin of 10 on the predicted leakage rate will be required for determining the leakage size flaw.

4. STEPS IN AN ACCEPTABLE LBB ANALYSIS

4.1. Demonstrate Accuracy of Leak Rate and Fracture Mechanics Computational methods

4.1.1. Accuracy of leak rate and fracture mechanics computational methods used in analyses should be demonstrated.

4.1.2. Demonstration can be made by comparison with other acceptable computational programs or with experimental data, where available.

4.2. Identify Materials and Material Property Data

4.2.1. Types of material and material specifications should be identified for base metal, weld metal, nozzles and safe ends, for the piping system being analyzed.

4.2.2. Material property data including fracture toughness and tensile data and ageing effects on these properties should be specified.

4.2.3. Piping materials fracture toughness (J-R curves) or other fracture mechanics material properties relevant to ductile fracture analyses and tensile (stress-strain curves) properties should be determined at temperatures near the upper range of normal plant operation. Properties should be derived from actual material heat tests.

4.2.4. Materials tests preferably should be conducted using archival material for the pipe being evaluated. If archival material is not available, plant-specific or industry-wide generic material databases can be assembled and used to define the required material tensile and toughness properties. Test material should include base and weld metals.

4.2.5. If the J-R curve approach is used, the specimens used to generate the J-R curves should be large enough to provide crack extensions up to an amount consistent with the J/T condition determined by the analysis. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used as described in [2, 4].

4.2.6. The stress-strain curves should be obtained over the range from the proportional limit to maximum load.

4.2.7. To provide an acceptable level of reliability, plant-specific generic databases should show reasonable lower bounds for compatible sets of material tensile and fracture toughness properties associated with material at the plant. To ensure that the plant-specific generic database is adequate, a determination should be made to demonstrate that the generic database represents the range of plant materials to be evaluated. This determination is based on a comparison of plant material properties identified in subparagraph 4.2.4 above with those of the materials used to develop the generic database. The number of material heats and weld procedures tested should be adequate to cover the strength and toughness range of actual plant materials. Reasonable lower bound tensile and toughness properties from the plant-specific generic database are to be used for the stability analysis of individual materials, unless otherwise justified.

Industry generic data bases should provide a reasonable lower bound for the population of material tensile and toughness properties associated with any individual specification, material type or welding procedures.

If data is being developed from an archival heat of material, three stress-strain curves and three J-resistance curves from that one heat of material are sufficient. The tests should be conducted at temperatures near the upper range of normal plant operation. Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where a pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. The lower toughness should be used in the fracture mechanics evaluation. One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

4.3. Specify Types and Magnitude of Applied Loads

4.3.1. Specify the type and magnitude of the loads applied to the piping system (forces, bending and torsion moments), their sources (thermal, deadweight, seismic, and seismic anchor movement), and method of combination.

Typically for leak-rate analysis, applied loads are combined algebraically, while for crack stability analysis, applied loads are combined either absolutely or algebraically.

4.3.2. For each pipe size in the piping system, identify location(s) that have the least favourable combination of stress and material properties for base metal, and welds.

4.4. Postulate a Leaking Through-Wall Crack at Critical Assessment Locations

4.4.1. When LBB technology is applied, all potential pipe rupture locations are examined. Alternately at the critical assessment locations identified above (according to par. 4.3 and 4.2, worst case combination of stress and material data), a postulated leaking through-wall crack is assumed to exist. Such locations are usually welds. Critical locations are postulated only on piping, not on piping components such as valves, pumps etc.

4.4.2. The size of flaw should be large enough so that flaw leakage during nominal operation is 10 times greater than the minimum leakage the detection system is capable of sensing, unless a detailed justification has been performed according to par. 3.6.10.

4.4.3. Orientation of the postulated crack is circumferential. This may lead to a double ended guillotine break. Note that axial welds were used limitedly only on older type WWERs, and that components containing axial welds usually are forged and more massive than other piping. For longitudinal welds in piping components it should be possible to demonstrate that through wall cracks postulated in these welds meet LBB requirements. This recognition can be performed for one piping component, which should cover all other cases.

4.4.4. For leakage estimation, normal operating loads, deadweight, thermal expansion, and pressure at unit nominal conditions, are to be combined based on the algebraic sum of individual values and applied to the leakage flaw size.

4.4.5. Auxiliary leak detection systems, if relied upon, should be described.

4.5. Determine Critical Crack Size and Critical Crack Size Margin

4.5.1. Using fracture mechanics analysis or modified limit load analysis in Equation(1) [1], determine the critical crack size for the postulated throughwall crack using loads acting at unit nominal conditions plus safe shutdown earthquake (SSE).

4.5.2. Determine the crack size margin by comparing the selected leakage crack size calculated in accordance with paragraph 4.4, to the critical crack size. Demonstrate that the margin is at least 2.0 between leakage crack size and critical crack size.

4.6. Determine Applied Load Margin

4.6.1. Calculate the margin on flaw size in terms of applied loads by a crack stability analysis.

4.6.2. Demonstrate that the size of leaking cracks will not become unstable if 1.4 times the nominal plus Safe Shutdown Earthquake (SSE) loads are applied.

4.6.3. Demonstrate that crack growth is stable and that final crack size is limited such that a double-ended pipe break will not occur.

4.6.4. The 1.4 margin can be reduced to 1.0 if deadweight, thermal expansion, pressure, SSE (inertial), and seismic anchor motion (SAM) loads are combined based on individual absolute values as follows:

$$
\begin{aligned}
F_{\sup} &= \left|F_{deadweight}\right| + \left|F_{press.\,\mathrm{exp}ansion}\right| + \left|F_{therm.\,\mathrm{exp}ansion}\right| + \left|F_{SSE}\right| + \left|F_{SAM}\right| \\
M_{\sup} &= \sqrt{M_x^{\,2} + M_y^{\,2} + M_z^{\,2}} \\
M_{x,y,z} &= \left|M_{x,y,z,deadweight}\right| + \left|M_{x,y,z,pressure}\right| + \left|M_{x,y,z,therm.\,\mathrm{exp}ansion}\right| + \left|M_{x,y,z,SSE}\right| + \left|M_{x,y,z,SAM}\right|
\end{aligned}
\tag{1}
$$

4.6.5. Evaluation of seismic anchor motion loads at SSE conditions may be omitted when these are shown to be small at operating basis earthquake (OBE) conditions.

4.6.6. The same load combination method must be used to determine critical crack size and to determine applied load margin.

5. LEAK RATE EVALUATION

5.1. Calculations should be made under assumptions that leak detection capability of installed systems is at least 3.8 l/min and leak detection systems meet the criteria of RG 1.45 [5].

5.2. Leak detection systems should be installed with at least three independent different physical principles of detection.

5.3. Leak detection systems should have a response time of no greater than 1 hour.

5.4. Leakage monitoring systems should provide output and alarms in the main control room. Periodic calibration and testing of leakage monitoring systems should take place. Operating procedures covering the leakage alarms should be prepared.

6. IN SERVICE INSPECTION (ISI) PROGRAMME

6.1. Volumetric in-service weld examinations should be conducted in accordance to regulatory requirements. NDE systems should be qualified (see Appendix E). Risk-informed ISI is applicable in accordance to Appendix D.

REFERENCES

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[6] INTERNATIONAL ELECTROTECHNICAL COMMISSION, Nuclear Reactors – Instrumentation and Control Systems Important for Safety – Detection of Leakage in Coolant Systems, International Standard IEC 1250, IEC 61250 ed 1.0, CEI, Geneva (1994).

[7] FEDERAL NUCLEAR AND RADIATION SAFETY AUTHORITY OF RUSSIA, Calculation Standard for Strength of Equipment and Pipes of Nuclear Power Units, Energoatomizdat, PNAE-G-7-002-86, NGA-01-85-1, NIKIET, Moscow (1989) (in Russian).

[8] MINISTRY OF ATOMIC ENERGY, Leak-before-break application guide for NPP piping, R-LBB-01-99, RD 95 10547-99, Minatom, Moscow (1999) (in Russian).

Supplement to APPENDIX A

S-1. INPUT PLANT DATA

- Geometry of piping components, technical drawings, sizes , and clearances;
- Restraints, hangers, supports (including snubbers and dampers) locations, stiffnesses, measured displacements, and prestresses;
- Peculiarities of joints to RPV nozzles, to other equipment, to valves, to MCP etc.;
- Maps (drawings) of welds, with IDs, types and categorization (according to certificates, standards, literature etc.);
- List or table of base and welding materials used, and their material properties within the operating temperature range;
- Information, if available, for technology of manufacturing, welding, installation and control (for assessment of residual stresses and conditional defects);
- Plant data on operational conditions, all types of loadings for DC and NOC, including transients and their frequency;
- Service records for excluding water hammer, ISI results, corrosion data, etc.;
- Seismic data – design or tested floor response spectra for different elevations;
- Data for specific loadings – hydraulic and thermal shocks, vibrations etc.;
- Results from design strength analyses.

Discrepancies between piping as-built conditions and design documentation, as well as manufacturer's data (if available) should be taken into account.

S-2. MATERIAL DATABASE

Chemical composition:

- Physical and mechanical properties (Young modulus, Poisson coefficient, thermal expansion coefficient, relative elongation and reduction of area or deformation σ-ε diagram, etc.);
- Crack resistance properties (J-R curves, critical temperature of brittleness, Charpy energy, etc.);
- Kinetics diagrams of defect growth for fatigue and corrosion-fatigue mechanisms;
- Test data that shows that design characteristics of materials are not beyond design characteristics used.

APPENDIX B

NO-BREAK-AREA ASSESSMENT FOR WWER PIPING

1. ABBREVIATIONS

ASME	American Society of Mechanical Engineers
AOC	Abnormal operating conditions
IAEA	International Atomic Energy Agency
ISI	In-service inspection
LBB	Leak before break
NBA	No-break area
NDE	Non destructive examination
NOC	Normal operation conditions
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission
NRI	Nuclear Research Institute
OBE	Operating basis earthquake
RG	Regulatory guide (US NRC)
SRP	Standard review plan
SSE	Safe shutdown earthquake
WWER	Water-water energetic reactor (Soviet designed PWR)

2. NOMENCLATURE AND SYMBOLS

$R_{p0.2}$	Yield strength, [MPa]
R_m	Tensile strength, [MPa]
S_m	Allowable design stress intensity, [MPa] (ASME approach)
$[\sigma]$	Nominal allowable stress, [MPa] (PNAE G approach)
$(\sigma)_{RK}$	Range of equivalent stresses due to system's transient from one service load set to another: thermal expansions and thermal anchor movements, temperature gradient loads, change of service pressure, [MPa] (PNAE G approach)
$(\sigma)_2$	Group of equivalent stresses due to mechanical and seismic (OBE) loading. Defined as combination of membrane and total bending stresses, [MPa] (PNAE G approach)
$(\sigma)_a$	Allowable stress, according to stress category for load combination and level of service limits, [MPa] (PNAE G approach)

3. JUSTIFICATION OF PROPOSED ASSESSMENT

3.1. Pipe rupture is a rare event that may only occur under unanticipated conditions such as those that might be caused by design, construction, or operational errors, unanticipated loads, or unanticipated corrosive environments. Expert observation of actual piping failures indicates that failures generally occur at high stress and fatigue locations. Examples of these are terminal ends of piping systems at connections to component nozzles, or pipe anchors that act as rigid constraints to piping motion and thermal expansion.

3.2. Subject to certain limitations, dynamic effects associated with postulated pipe ruptures may be excluded from the design basis, where analyses reviewed and approved by the regulatory authority demonstrate that fluid system piping rupture probability is extremely low under design basis conditions. These analyses are commonly referred to as „leak-before-break" (LBB) analyses. The application of LBB to piping system design is reviewed in accordance with Appendix A of the VERLIFE procedure.

3.3. The goal of the No-Break Area (NBA) assessment proposed for evaluation of integrity of high and moderate-energy piping in NPPs of WWER-440 and WWER-1000 types is to establish conditions and requirements for piping under which it may be ensured, by performing additional analyses, provisions and procedures, that the probability of fluid system piping break and/or leakage crack occurrence for unknown and/or hard to predict reasons is lowered to an acceptable level.

The assessment utilizes available piping design information, postulates pipe ruptures at locations having relatively higher potential for failure, and demonstrates that an adequate and practical level of protection may be achieved.

4. SCOPE AND LIMITS OF APPLICABILITY

4.1. The NBA assessment as defined in this document is applicable to high- and moderate-energy piping designed and manufactured according to the requirements for quality Group B and C (PNAE-G - 01 - 011 - 97) or for safety Class 1, 2 and 3 according to Safety Analysis Report, or the equivalent.

4.2. NBA assessment is applicable only to piping manufactured from metal materials.

4.3. Applications to high-energy piping other than Group B and C / Class 1, 2 and 3 may be considered, based on an evaluation of proposed design and in-service inspection requirements as compared to Group B and/or C / Class 1, 2 and/or 3 requirements.

4.4. Design of pipelines, including the piping itself as well as related components such as valves, anchoring points, supports, and hangers, must be performed in accordance with the Technical Specifications. Manufacturing, welding, heat treatment and piping assembly must be performed in accordance with the Technical Specifications.

5. EXCLUSION OF PIPING BREAK AND LEAKAGE CRACK POSTULATION

Satisfying appropriate design and operating criteria, ISI requirements (defined below in chapter 4) and requirements defined below in chapters 3.1 and 3.2 ensures that for the portions of piping under consideration breaks and leakage cracks need not be postulated.

Two approaches can be applied for this assessment:

— ASME approach – using ASME Section III Subsection NB and Subsection NC;

— PNAE G approach.

In both approaches loads and conditions should be considered that correspond to NOC resp. AOC, as specified in system Technical Specifications (i.e. sustained loads, occasional loads, and thermal expansion), including an OBE event (if applicable).

Water hammer should be considered as follows:

(a) worst case for portion of piping under consideration should be analysed;

(b) selection of the worst case water hammer event can be based either on

  — Calculations, or

  — Engineering judgment and experience from similar piping system(s).

5.1. Breaks need not be postulated provided that:

5.1.1. ASME approach

ASME approach is based on suggestion of meeting stress requirements defined by individual ASME equations used in this approach.

5.1.1.1. For Class 1 high-energy piping:

(a) Maximum stress range calculated by equation (10) of ASME Section III Subsection NB (NB-3653) in Ref. [4] should not exceed $2.4\ [S_m]$.

  In calculations, maximum primary plus secondary stress range (between any two load sets and including the zero load set) of loading have to be considered.

(b) If maximum stress range as calculated using Equation (10) in Ref. [4] between any two load sets (including the zero load set) exceeds allowable stress intensity value $2.4\ [S_m]$ in any portion of the piping, then additional analyses have to be performed, demonstrating that the following two stress criteria are met for the piping portion concerned, under NOC:

  — Secondary thermal expansion and thermal anchor movements stress range between any two load sets (calculated from moments, using Equation (12) ASME NB-3653.6 in Ref. [4] should not exceed allowable stress intensity value $2.4\ [S_m]$.

  — Primary plus secondary membrane plus bending stress intensity should not exceed allowable stress intensity value $2.4\ [S_m]$ (calculated using Equation (13) ASME NB-3653.6 in Ref. [4]). The range of pressure and moments between any two load sets determined from design mechanical loads should be taken into account. Thermal bending and thermal expansion stresses are excluded.

(c) Attachments welded directly to the outer piping surface (serving as pipe supports or for other purposes), should be analyzed by performing detailed stress analysis, or tested to demonstrate they meet the requirements specified in 3.1.1.1 (a) and (b).

(d) Maximum primary stress as calculated by Equation (9) in ASME Code, NB-3652 in Ref. [4] under the loadings resulting from a postulated piping failure beyond these portions of piping should not exceed the minimum of $2.25\ [S_m]$ and $1.8\ R_{p0,2}$ at the temperature under consideration. The following loading should be included: loads of service weight and loading due to postulated pipe break, water hammer and dynamic mass flow effect of medium in the piping. Following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed, and that operability of valves and other safety components with such stresses is ensured.

(e) Cumulative usage factor should be less than 0.4, unless otherwise required by the competent regulatory body.

5.1.1.2. For Class 2 and 3 high-energy piping:

(a) The sum of stresses calculated by Equation (9) and (10) in NC/ND-3653 ASME code in Ref. [4] should not exceed 0.8 times the sum of stress limits given in NC/ND-3653.

(b) Attachments welded directly to the outer piping surface (serving as pipe supports or for other purposes), should be analyzed by performing detailed stress analysis, or tested to demonstrate they meet the requirement specified in 3.1.1.2(a) in Ref. [4]. If it is not possible to perform detailed analysis (for example due to insufficient input data), engineering judgement should be utilized.

(c) Maximum primary stress as calculated by Equation (9) in ASME Code NC-3653.1 in Ref. [4] under the loads resulting from a postulated piping failure beyond these portions of piping including loads of service weight and loading due to postulated pipe break, and water/steam hammer, and dynamic mass flow effect of medium in the piping, should not exceed the minimum of $2.25\ [S_h]$ and $1.8\ R_{p0,2}$ at the temperature under consideration. Following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed, and operability of valves and other safety components with such stresses is ensured.

5.1.1.3. Terminal ends

Breaks need not be postulated in terminal ends in the containment penetration area. In other terminal ends breaks should be postulated.

Containment penetration area is the portion of piping between the hermetic penetration and the first isolation valve mounted on this piping.

5.1.2. PNAE-G approach

5.1.2.1. For Group B and C high-energy piping:

(a) Maximum primary plus secondary range of equivalent stresses should not exceed:

$$(\sigma)_2^{NOC+OBE} + (\sigma)_{RK}^{max} \leq 0.8\ [(\sigma)_a^{NOC+OBE} + (\sigma)_a^{RK}] \tag{1}$$

(b) Attachments welded directly to the outer piping surface (serving as pipe supports or for other purposes), should be analyzed by performing detailed stress analysis, or tested to demonstrate they meet the requirements defined in 3.1.2.1 (a) in Ref. [4].

(c) Maximum primary equivalent stresses (from a combination of membrane and total bending stresses) under loads resulting from a postulated piping failure beyond these portions of piping should not exceed the $(\sigma)_2$ taken as of value $1.6[\sigma]$. The following loading should be included: loads of service weight and loading due to postulated pipe break, water hammer and dynamic mass flow effect of medium in the piping. Following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided that a plastic hinge is not formed and operability of valves and other safety components with such stresses is ensured.

(d) Cumulative usage factor should be less than 0.4, unless otherwise required by the competent regulatory body.

5.2. Leakage cracks need not be postulated in those portions of high and moderate-energy piping for which both appropriate design and operating criteria and the following stress criterion are met: half of the limiting value defined in 3.1.1.1(a), 3.1.1.2(a) or 3.1.2.1(a) in Ref. [4], based on approach applied and Class of piping under consideration, is not exceeded.

5.3. Leakage cracks need not be postulated in the containment penetration area.

6. ISI REQUIREMENTS

6.1. Portions of piping where breaks and leakage cracks can be excluded according to chapter 3, shall meet the following additional criteria:

6.1.1. A 100% volumetric in-service examination of all welds should be conducted during each inspection interval. The extent of examination of welds can be reduced when risk-informed ISI specific to No-Break Area is applied and accepted by the regulatory body. NDE system should be qualified (see Appendix E).

6.1.2. In case of excessive occurrence of degradation mechanisms or excessive pipe wall thinning, additional volumetric examinations should be carried out.

6.1.3. Piping shall be included into an ageing management programme.

REFERENCES

[1] U.S. NUCLEAR REGULATORY COMMISSION, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (Chapter 3, Section 3.6.2): Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, NUREG 0800 Chapter 3, Section 3.6.2 Rev2, US NRC, Washington, DC (2007).

[2] U.S. NUCLEAR REGULATORY COMMISSION, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (Chapter 3, Section 3.6.3): Leak-Before-Break Evaluation Procedures, NUREG 0800 Chapter 3, Section 3.6.3 Rev1, US NRC, Washington, DC (2007).

[3] U.S. NUCLEAR REGULATORY COMMISSION, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (Chapter 3, Branch Technical Position 3.4): Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment, NUREG 0800 Chapter 3, BTP 3-4 Rev2, US NRC, Washington, DC (2007).

[4] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, ASME Boiler and Pressure Vessel Code, Section III: Rules for Construction of Nuclear Facility Components, Division 1 - Subsections NB: Class 1 Components and Subsections NC: Class 2, ASME, New York, NY (2013).

[5] FEDERAL NUCLEAR AND RADIATION SAFETY AUTHORITY OF RUSSIA, Standard for Strength Calculations of Components and Piping in NPPs, Energoatomizdat, PNAE-G-7-002-86, NGA-01-85-1, NIKIET, Moscow (1989) (in Russian).

[6] FEDERAL NUCLEAR AND RADIATION SAFETY AUTHORITY OF RUSSIA, General Regulations on Ensuring safety of nuclear power plants, OPB-88/97, NP-001-97 (PNAE G-1 011-97), Gosatomnadzor, Moscow (1997).

[7] BERKOVSKY, A., KOSTAREV, V., STEVENSON, J.D., Adaptation of the Modern Approaches for Protection of Nuclear Power Plants against the Effects of Postulated Pipe Ruptures to the Russian National Guides. Problems and Experience, in Proc. of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17), Prague (2003).

APPENDIX C

INTEGRITY AND LIFETIME ASSESSMENT PROCEDURE OF RPV INTERNALS IN WWER-TYPE NPPS DURING OPERATION

1. SCOPE OF APPLICATION

This procedure covers reactor vessel internal (RVI) components of WWER water-moderated water-cooled power reactors. It is used to assess RVI lifetime according to the following criteria:

- Crack nucleation and growth by mechanisms of fatigue and irradiation assisted stress corrosion cracking (IASCC);
- Inadmissible change of RVI component geometrical sizes.

2. DESIGNATIONS AND ABBREVIATIONS

2.1. ABBREVIATIONS

AOO	Anticipated operational occurrences
LEA	Limit embrittlement area
FCGR	Fatigue crack growth rate
FEM	Finite element method
HAZ	Heat affected zone
MWF	Method of weight functions
NOC	Normal operating condition
RVI	Reactor vessel internals
NPP	Nuclear power plant
IASCC	Irradiation assisted stress corrosion cracking
SIF	Stress intensity factor
SSF	Stress-strain field
WWER	Water-moderated water-cooled power reactor

2.2. DESIGNATIONS OF GEOMETRICAL PARAMETERS

a, c	Minor and major semiaxes of semielliptical and one-quarter elliptic cracks
a_0, c_0	Initial sizes of minor and major semiaxes of semielliptical and one-quarter elliptic cracks
l_o	Maximum possible depth of a technological flaw
a_τ, c_τ	Semiaxes sizes of a postulated crack for the end of a considered time under IASCC
$a_{\tau N}$ $c_{\tau N}$	Semiaxes sizes of a postulated crack for the end of a considered time under fracture mode by IASCC and fatigue
S_t	Wall thickness of an RVI component

2.3. DESIGNATIONS OF LOADING PARAMETERS

P_p	Primary loads (pressure, force and weight loads)
P_s	Secondary loads due to physical effects resulting in self-balanced stress fields (residual welding stresses, temperature stresses, stresses caused by swelling)
F	Damage dose of neutron irradiation
Φ	Damage dose rate
T_{irr}	Irradiation temperature
T	Operation (test) temperature
t	Time
t_{life}	Lifetime
Δt	Time interval
f	Loading frequency
N	Cycle endured
N_f	Fatigue life
R	Stress ratio
ω^{env}	Scceleration factor of fatigue crack growth rate of irradiated metal in WWER water as compared with inert (air) environment

2.4. DESIGNATIONS OF TENSILE PROPERTIES

ν	Poisson ratio
E	Young’s modulus of elasticity
G	Shear modulus
σ_Y	Yield strength
$\Delta\sigma_Y$	Increment of yield strength under neutron irradiation
σ_{ul}	Ultimate tensile strength
$\Delta\sigma_{ul}$	Increment of ultimate tensile strength under neutron irradiation
σ_{-1}	Fatigue limit at symmetric alternate cycle of stress
C_f , n_f	Empirical constants of the Paris equation
ε_f	Fracture strain
S_0	Free swelling
$\dot{S}_0$	Rate of free swelling
S	Swelling at the given level of stress and plastic strain
$\dot{S}$	Rate of swelling at the given level of stress and plastic strain

2.5. DESIGNATIONS OF FRACTURE AND DAMAGE MECHANICS PARAMETERS

K or K_I	Mode I stress intensity factor (SIF)
K_{Jc}	Fracture toughness
J_c	Critical value of J-integral
J^e	Elastic part of J-integral
ΔK_a	SIF range in the point a of a postulated crack
ΔK_c	SIF range in the point c of a postulated crack
ΔK_{th}	Fatigue crack growth threshold
D_N	Fatigue damage
D_τ	IASCC damage
$\Delta\varepsilon$	Strain range
$\Delta\varepsilon_{n_\varepsilon}$	Strain range with regard to the strain range safety factor $n\varepsilon$
$\Delta\varepsilon_{n_N}$	Strain range with regard to the cycle number safety factor nN
F^*	Critical damage dose of neutron irradiation for IASCC initiation
t_{fat}	Time for fatigue crack nucleation
t_{IASCC}	Time for IASCC crack nucleation
t_{nuc}	Time for crack nucleation
t_{life}	Considered period of time, for which component strength is justified

2.6. DESIGNATIONS OF STRESS-STRAIN STATE PARAMETERS

σ_{ij}	Stress tensor components
ε_{ij}	Strain tensor components
δ_{ij}	Kronecker symbol
s_{ij}	Deviatoric stress tensor components
ρ_{ij}	Microstress tensor components
σ_1, σ_2, σ_3	Principal stresses: $\sigma1>\sigma2>\sigma3$
ε_1, ε_2, ε_3	Principal strains: $\varepsilon1>\varepsilon2>\varepsilon3$
σ_{eq}	Equivalent stress by Mises
ε_{eq}	Equivalent strain
σ_m	Hydrostatic stress
σ_{max}	Peak stress in a cycle
ξ_{ij}^c	Creep strain rates tensor components

ξ^{p}_{ij}	Plastic strain rates tensor components
ξ^{c}_{eq}	Equivalent creep strain rate
ξ^{p}_{eq}	Equivalent plastic strain rate
ξ^{ten}_{min}	Minimum equivalent strain rate in a tensile semi-cycle (in the region where $\sigma 1 > 0$)
$æ_c$	Accumulated creep strain
$æ_p$	Accumulated plastic strain
σ_{ref}	Reference stress
ξ_{ref}	Strain rate calculated from the creep equation with $\sigma eq=\sigma ref$

2.7. DESIGNATION OF SAFETY FACTORS

n_J	Safety factor used in calculation of crack critical size
n_ε	Strain range safety factor for construction of fatigue disposition lines
n_N	Cycle number safety factor for construction of fatigue disposition lines
φ_s	Coefficient of reduction of fatigue curve

3. TERMS AND DEFINITIONS

The following terms with corresponding definitions are used in this procedure:

Actual model of RVI operation

An operational model that reflects the actual operating conditions of RVI components over the investigated period of time.

Crack nucleation

Formation of engineering significant crack of defined size (nucleus crack).

Critical event „Fatigue crack nucleation”

Condition of a component when a crack is nucleated by the fatigue mechanism. Sizes of nucleus crack are presented in Section 6.

Critical event „Formation of a limit embrittlement area”

Condition of a component under which a zone phase $\gamma \rightarrow \alpha$ – transformation occurs due to the area reaching critical values of neutron dose damage and irradiation temperature. This results in formation of a limit embrittlement area.

Critical event „IASCC crack nucleation”

Condition of a component when a crack is nucleated by the mechanism of irradiation assisted stress corrosion cracking. Sizes of nucleus crack are presented in Section 6.

Critical event „Loss of carrying capacity”

Condition of a component when the primary load reaches its limit value based on the assumption $\sigma_{ul}=\sigma_{Y}$.

Critical event „Unstable crack propagation”

Condition of a component when the criterion of unstable crack propagation is met.

Critical events of a failure of a component

Condition of a component under which the one of following events occur. fatigue crack nucleation; IASCC crack nucleation; formation of limit embrittlement area; unstable crack propagation; loss of carrying capacity; inadmissible change of geometrical sizes of a component.

Elastic problem

A problem in which determination of fields of stress and strain in a structure is calculated on the basis of elasticity theory with given boundary and initial conditions.

Elastic-plastic problem

A problem in which determination of fields of stress and strain in a structure is calculated on the basis of the theories of elasticity and plasticity with given boundary and initial conditions

Irradiated condition

The condition of a material exposed to irradiation at which damage dose exceeds 1 dpa.

Limit condition „Inadmissible change of geometrical sizes of a structural component”

Condition of a component under which an inadmissible change in component size is reached.

Model of RVI operation

Schematization of changes in parameters that determine loading of RVI components during operation, including a combination of loads arising from seismic and other dynamic effects

Non-radial loading

The loading under which for each point of a component the relationship between deviatoric stress tensors is changed.

Postulated crack

A crack of a given depth, length and location postulated in an investigated component.

Predicted model of RVI operation

The assumed operational model of RVI components that is predicted for the expected lifetime.

Radial loading

The loading under which for each point of a component the relationship between deviatoric stress tensors is not changed.

RVI components

Reactor core barrel, core baffle, core basket, protective tube unit.

RVI material

Austenitic steels with basic composition 18Cr-10Ni-Ti and its weld. The materials are fabricated in compliance with regulations referenced in design.

Viscoelastoplastic problem

A problem in which determination of fields of stress and strain in a structure is calculated on the basis of the theories of elasticity, plasticity and creep with given boundary and initial conditions.

Weld joint

The region of an RVI component, including weld metal proper and the heat affected zone (HAZ), that is located in base metal at a distance of up to 5 mm from the fusion line of the weld metal and base metal.

4. GENERAL PRINCIPLES

4.1 This procedure is applicable to neutron irradiation of material that is characterized by the following ranges of neutron damage dose and irradiation temperature: $F \leq 70$ dpa; T_{irr}=270 to 450 °C

Strength of a component is considered from as resistance to loss of carrying capacity and resistance to unstable crack propagation. Analysis of unstable crack growth and loss of carrying capacity is carried out for components with an initial crack. Nucleation and crack growth may happen via IASCC, fatigue mechanisms, and/or due to formation of a LEA.

Refer to Section 5 for a logic diagram for calculation of component strength.

In addition to a strength calculation, a serviceability analysis is carried out to confirm any component geometrical size changes are within allowable limits. Component lifetime is recognized as the maximum time for which strength and serviceability of the applicable component is assured.

4.2 Critical events

4.2.1 The following component conditions are considered critical events with respect to component failure

4.2.1.1. Fatigue crack nucleation.

4.2.1.2. IASCC crack nucleation.

4.2.1.3. Formation of limit embrittlement area (LEA).

4.2.1.4. Unstable crack propagation.

4.2.1.5. Loss of component carrying capacity.

4.2.1.6. Inadmissible change of component geometrical sizes.

4.3. Analysis of strength of the investigated component

4.3.1. Strength assessment is carried out according to actual and (or) predicted models of RVI operation. When analyzing strength and wear of RVI during operation, vibration loading shall be taken into account (see Application 1).

4.3.2. The strength of the investigated component is recognized as provided over the investigated period of time if critical events 4.2.1.4 and 4.2.1.5 have not occurred.

4.3.3. The analysis of RVI component strength is carried out both for base metal and weld joints. When analyzing the strength of a base metal it is assumed that initial cracks are absent. When analyzing the strength of weld joints a postulated crack is assumed on the basis of the results of nondestructive in-service inspections.

4.3.3.1. If a nondestructive in-service inspection was not carried out in the investigated zone, it is assumed that from the beginning of operation there is a crack with sizes as indicated in Section 6.3 for a semi-elliptical surface flaw.

4.3.3.2. If a nondestructive in-service inspection is carried out in the investigated RVI component zone, and the revealed flaws are less than sizes allowable, then the sizes of initial cracks are determined according to 6.3 for a semi-elliptical surface flaw.

4.3.3.3. If nondestructive in-service inspection is carried out in the investigated RVI component zone and revealed flaws exceed the sizes allowable, then special justification of strength for this zone must be performed

4.3.4. If over the investigated period of time the critical events indicated in 4.2.1.1 – 4.2.1.3 have not occurred for the irradiated ($F > 1$ dpa) RVI component zone, then strength is recognized as provided as long as the critical events „Unstable crack growth“ and „Loss of load carrying capacity“ do not occur for the same period of time. In this case the critical events indicated in 4.2.1.4 and 4.2.1.5 are analyzed provided that in this zone there is a semi-elliptical crack with sizes according to 6.3 independent of the zone metallurgical features (base metal or weld metal).

4.3.5. If over the investigated period of time the critical events indicated in 4.2.1.1 – 4.2.1.3 have not occurred for an unirradiated ($F \leq 1$ dpa) RVI component zone, then its strength is recognized as provided without additional analysis of the critical events indicated in 4.2.1.4 and 4.2.1.5.

4.3.6. If possible, vibration loads should be taken into account when the critical events indicated in 4.2.1.1 and 4.2.1.4 are considered (see Application 1).

4.4. Analysis of serviceability of the investigated component.

The condition of serviceability of RVI component is provided, if the critical event indicated in 4.2.1.6 has not occurred over the investigated period of time.

4.5. Monitoring of flaws and geometrical sizes

4.5.1. If over the investigated period of time critical events in 4.2.1.4 – 4.2.1.5 have occurred for the considered RVI component zone, then the maximum period of time t_{life} in which the above events have not occurred is determined.

4.5.2. Operation of the considered components may continue following t_{life} if the following activities take place.

At $t=t_{life}$ it is necessary to perform at least one of the following procedures:

— Non-destructive TV examination;

— Non-destructive ultrasonic examination;

— Non-destructive vibration diagnostics including measurements of neutron noise;

These procedures should be qualified to be able to detect flaw sizes larger than 0.25 x the critical crack size in the tested component.

If the above procedures do not reveal flaws greater than 0.5 x the critical crack according to Section 10, then operation of the component may be continued until the next in-service inspection.

4.5.3. If the above procedures revealed flaws greater than 0.5 x the critical crack, then special justification of strength for this zone must be performed according to Section 10.

4.5.4 If over the investigated period of time a critical event as indicated in 4.2.1.6 has occurred for the considered RVI component zone, then maximum period of time t_{life} in which the above event has not occurred is determined.

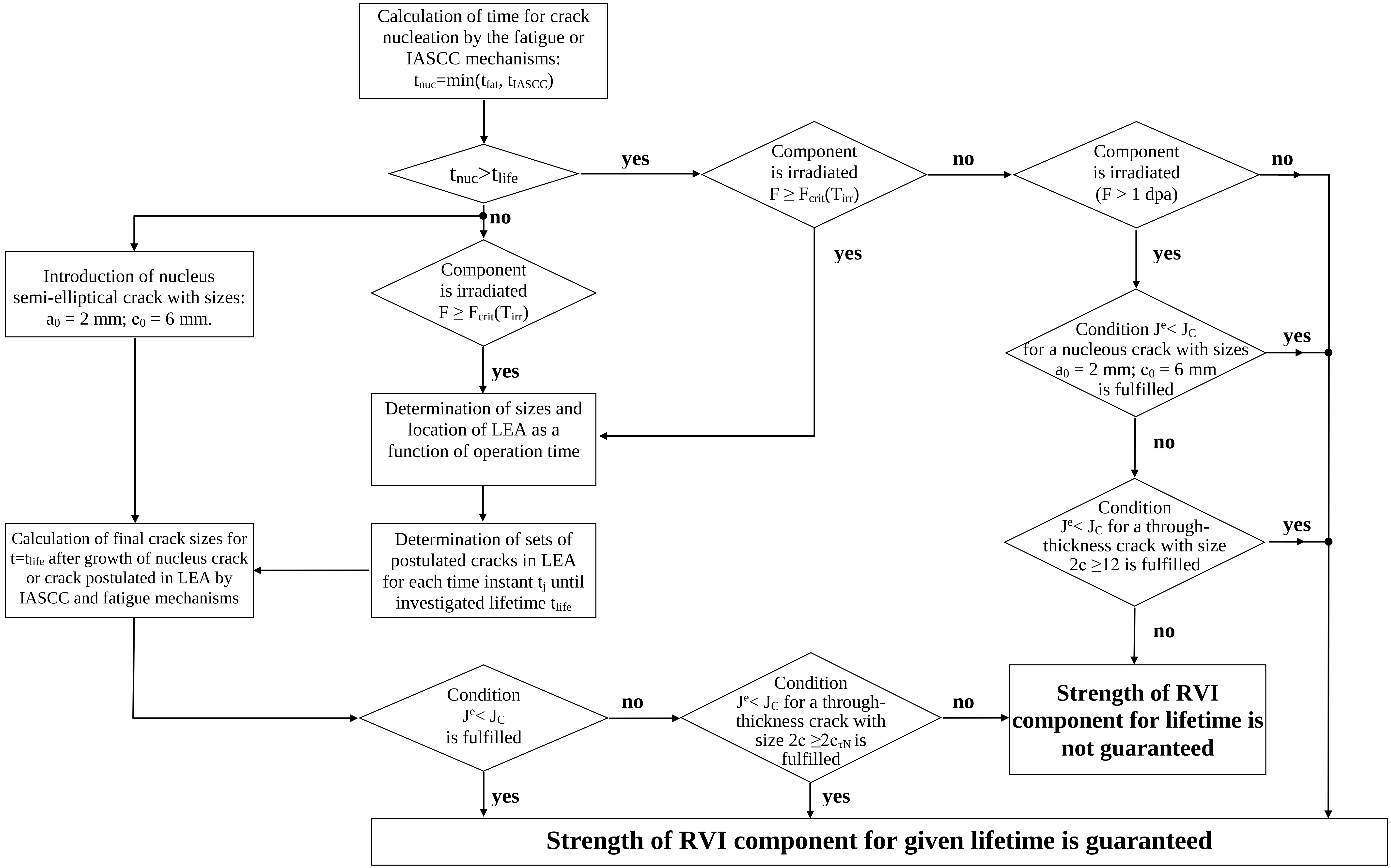
4.5.5. Operation of considered components may be continued beyond t_{life} if at t_{life} measurements of geometric sizes of the considered component are performed.

If the above measurements do not reveal allowable component size exceedances then operation of the component may continue up to its next in-service inspection.

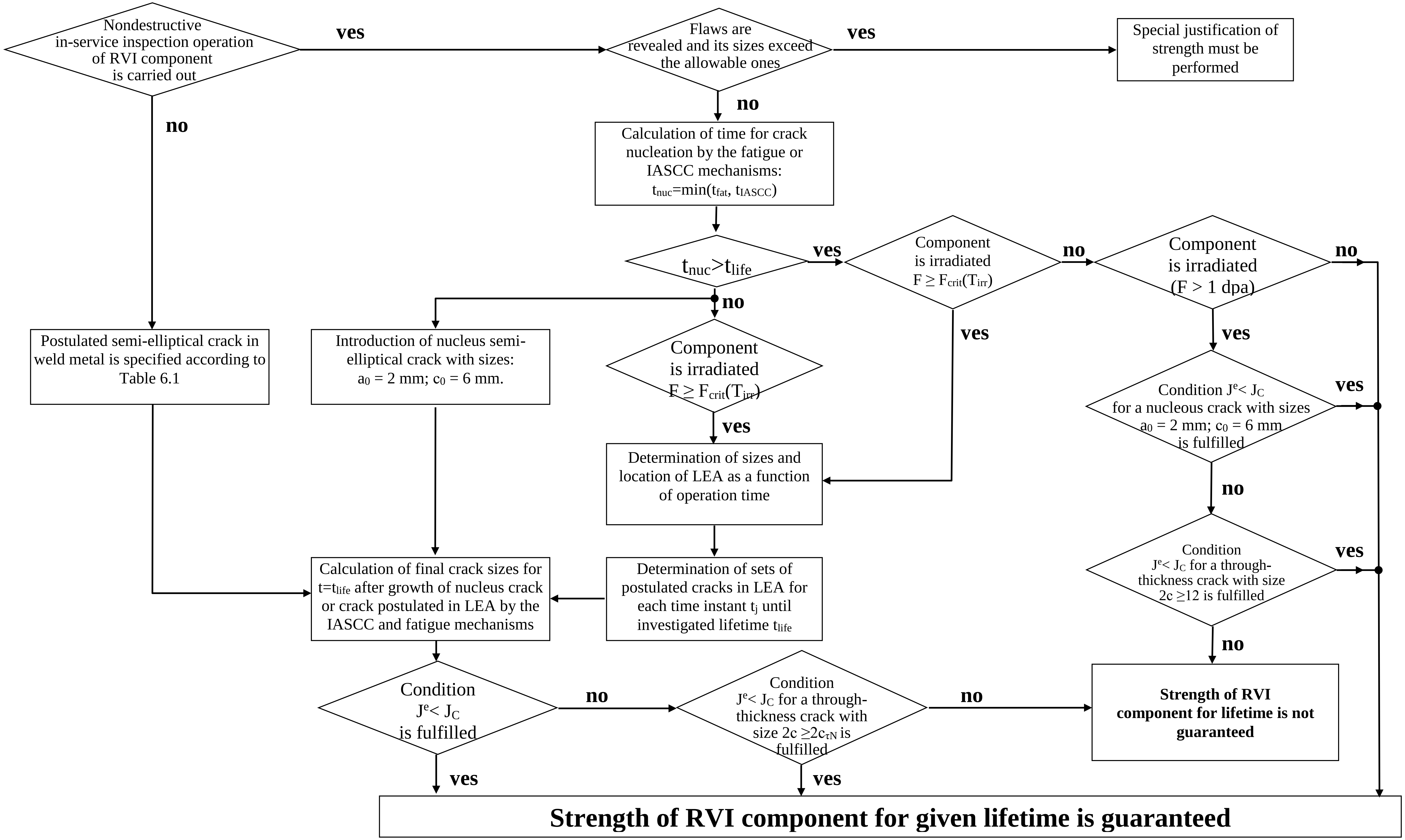
4.5.6. If the above measurements reveal component size exceedances, then special justification of serviceability for this zone must be performed.

5. LOGIC DIAGRAM FOR ANALYSIS OF RVI STRENGTH

5.1. BASE METAL



5.2. WELD METAL



6. POSTULATED CRACK

6.1. The surface semielliptical crack (Figure 1) or surface corner one-quarter elliptical crack (Figure 2) or internal elliptical crack, that are characterized by the sizes of their major semiaxis **c** and minor semiaxis **a,** are taken as a postulated crack. The initial ratio between the major semiaxis c_0 and minor semiaxis a_0 of a postulated crack is taken to be equal to $(c_0/a_0) = 3$.

6.2. In weld joints the initial depth of the semielliptical or one-quarter crack is determined by the equation

$$a_0 = l_0, \tag{1}$$

where l_0 is the maximum possible depth of the technological flaw.

The value of l_0 for weld joints is determined depending on the thickness S_t of welded components according to Table 1.

TABLE 1. INITIAL DEPTH OF THE POSTULATED CRACK IN WELD JOINTS.

S_T, MM	L_0, MM
Less than 10	2
10 - 20	3
21 - 30	4
More than 30	5

6.3. When the critical events indicated in 4.2.1.1 and 4.2.1.2 occur, the size of the nucleus crack (postulated crack) is taken to be in compliance with Table 1, with a_0 = 2 mm for semielliptical cracks and a_0 = 1 mm for elliptical cracks.

6.4. When the critical event indicated in 4.2.1.3 occurs, the size of the postulated crack is determined to be in compliance with Section 9 and Annex L.

6.5. The final sizes of the crack are determined in compliance with Section 10.

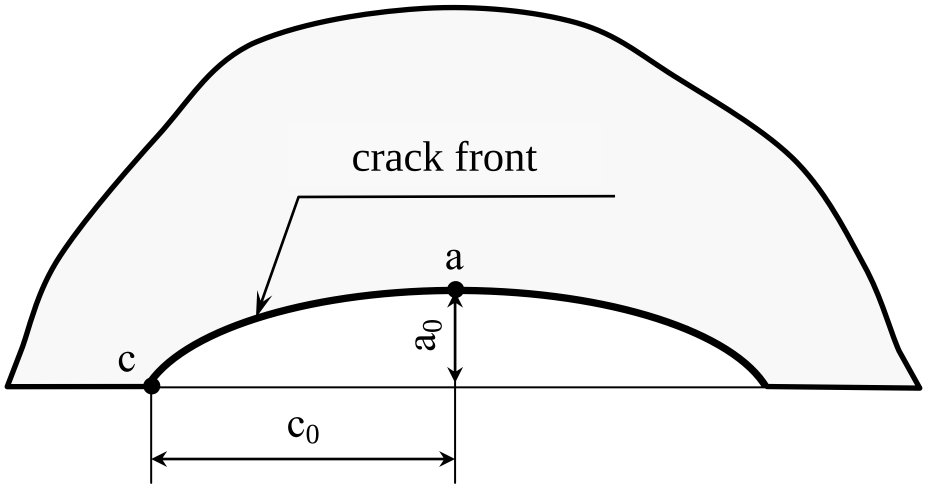


FIG. 1. Diagram of the postulated crack in basic RVI components. Surface semi-elliptical crack.

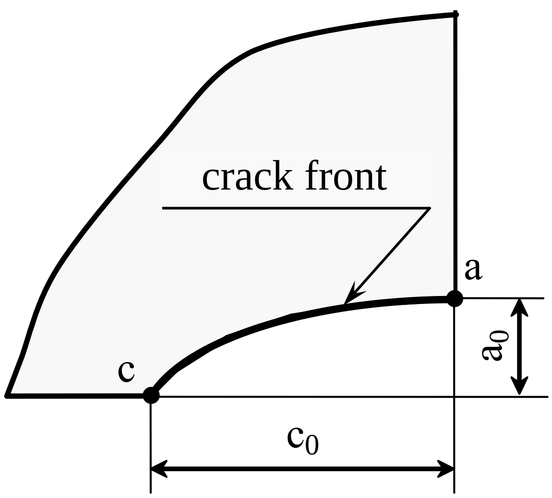


FIG. 2. Diagram of the postulated crack in basic RVI components. Surface corner one-quarter elliptic crack.

7. CRITICAL EVENT „FATIGUE CRACK NUCLEATION“

7.1. This calculation is carried out for the RVI component zones out of LEA.

7.2. The whole period of RVI operation is divided into time intervals Δt_j, so that $t_{j+1} = t_j + \Delta t_j$; $j = 1 \div L$, where L is a number of Δt_j intervals. For each time interval Δt_j the mechanical properties (yield strength, strain hardening parameters) and fatigue properties (fatigue curves depending on a damage dose of neutron irradiation; see Annex B) are taken to be invariant to damage dose change inside the interval.

In this case the following conditions are fulfilled in the time interval Δt_j:

— the mechanical properties for calculating SSF are appropriate to the damage dose of neutron irradiation F_j that corresponds to the instant in time t_j (the beginning of the time interval Δt_j);

— the fatigue curve is appropriate to the damage dose of neutron irradiation F_{j+1} that corresponds to the instant in time t_{j+1} (the end of the time interval Δt_j).

Mechanical properties of RVI material are presented in Annex A. Fatigue curves are shown in Annex B.

Note: The calculation is recommended to be carried out with a maximum possible number of time intervals L. The value of the interval Δt_j is chosen in such a way that the temperature and conditions of loading at the beginning and end of the interval are approximately the same. As the number of intervals L decreases, the calculated value of fatigue damage increases and calculation of damages becomes more conservative.

7.3. In order to determine strain ranges for each time interval Δt_j the elastic-plastic problem in an unsteady quasi-static non-isothermic statement is solved. The constitutive equations for solving the elastic-plastic problem are given in the Annex K.

Solution of the elastic-plastic problem is carried out by FEM by means of stepwise tracing of a loading history on the time interval from t_j to $t_j+\Delta t_j$. It is recommended to specify the step dimension $\Delta\tau_k^{FEM} << \Delta t_j$ where $\Delta\tau_k^{FEM}$ is a time step at which loading is close to radial.

Notes: Since calculation is made with the use of fatigue curves with a maximum possible stress ratio R ($R = \frac{\sigma_{min}}{\sigma_{max}}$, where $\sigma_{max}=\sigma_Y$; $\sigma_{min}=\sigma_Y - E\cdot\Delta\varepsilon$) then:

(a) When solving the elastic-plastic problem in order to define strain ranges, it is not necessary to take into account residual welding stress and stress by swelling, as these stresses affect stress ratio.

(b) It is allowed to perform independent solutions of elastic-plastic problems for each time interval Δt_j, that is, without consideration of the loading history at $t < t_j$.

7.4. The sequence of loading conditions over the investigated time interval Δt_j is determined by actual and predicted models of RVI operation.

7.5. The procedure of loading cycle formation under non-radial loading (determination of the number of cycles and strain ranges corresponding to each cycle) is shown in Annex I.

7.6. Damage D_{N_j} obtained for the time interval Δt_j is calculated by equation

$$D_{N_j} = \sum_{i=1}^{k} \frac{N_i}{N_{fi}}, \qquad (2)$$

where

N_i is cycles endured with the same strain ranges $\Delta\varepsilon_{eq}^{i}$;

k is the number of blocks with different strain ranges (different stress ratio) in the time interval Δt_j;

N_{fi} is fatigue life with strain ranges $\Delta\varepsilon_{eq}^{i}$.

Note: The value of N_{fi} is determined from the disposition line of fatigue (the calculation procedure of the disposition line of fatigue is shown in Annex B) that is appropriate to the parameter ξ_{min}^{ten} (the minimum equivalent strain rates in a tensile semi-cycle, in the region where $\sigma_1 > 0$) and damage dose of neutron irradiation F_{j+1} corresponding to the instant time t_{j+1}. The value of N_{fi} should be determined from the fatigue curves with the maximum stress ratio R.

7.7. Damage D_N for the whole investigated period of operation is calculated by

$$D_N = \sum_{j=1}^{L} D_{N_j}. \qquad (3)$$

7.8. The critical event „Fatigue crack nucleation" in the investigated period of time does not occur, if the following condition is fulfilled

$$D_N < 1. \qquad (4)$$

7.9. If by the end of the investigated period of operation the damage D_n>1, then the instant time t_{fat} at which D_N=1 is determined (counting off is performed from the beginning of RVI operation). If condition t_{fat}<t_{IASCC} is fulfilled (see Section 8), calculation of crack growth is performed since t=t_{fat} (Section 10).

8. CRITICAL EVENT „IASCC CRACK NUCLEATION"

8.1. Calculation is carried out only for the irradiated zones of RVI components that contact with coolant.

Note: Calculation of crack nucleation by IASCC may not carry out inside LEA.

8.2. The whole period of RVI operation is divided into time intervals Δt_j, so that $t_{j+1}=t_j+\Delta t_j$; $j=1\div L$, where L is a number of Δt_j intervals. For each time interval Δt_j a viscoelastic-plastic problem is solved in the unsteady quasi-static non-isothermic statement taking into account residual welding stresses, swelling, radiation creep and loading history over the time interval from

0 to t_j. The constitutive equations for solving a viscoelastic plastic problem are given in Annex K. The equations for calculation swelling and radiation creep are presented in Annexes G and H.

For each time interval Δt_j mechanical properties are taken to be corresponding to the damage dose of neutron irradiation F_{j+1} at time t_{j+1} (the end of the time interval Δt_j).

Solution of a viscoelastic plastic problem is performed by FEM by means of stepwise tracing of a loading history, the time steps $\Delta\tau_k^{FEM}$ << Δt_j where $\Delta\tau_k^{FEM}$ is a time step at which loading is close to a radial one.

The necessary information on mechanical properties and equations of creep is given in Annexes A and H, correspondingly.

Note: It is recommended to make the calculation with the maximum possible number of time intervals L. As the number of intervals L decreases, the calculated value of a damage increases and the calculation of damages becomes more conservative.

8.3. For the whole investigated period of loading the curves $\sigma_1(t)$, $\sigma_{eq}(t)$ and $\sigma_{th}^{IASCC}(t)$ are calculated taking into account $\sigma_{th}^{IASCC}(F)$ and $F = \int_0^t \Phi(t)\cdot dt$ (see Figure 3); the dependence $\sigma_{th}^{IASCC}(F)$ is given in Annexes C. Then the intervals Δt_k^{calc} are determined for which the condition $\sigma_1 > 0$, $\sigma_{eq} > \sigma_{th}^{IASCC}$ and $T \geq T_{th}^{IASCC}$ is fulfilled (see Figure 4), where k is interval number; T_{th}^{IASCC} – minimal temperature, at which IASCC happens; T_{th}^{IASCC}=200°C. Scheme of intervals Δt_k^{calc} determination is shown in Figure 4. For these intervals the calculated dependence $\sigma_{eq}(t)$ is constructed. After construction, $\sigma_{eq}(t)$ is approximated by the stepped function where in the subinterval $[t_{i-1}^{sub}, t_{i-1}^{sub} + \Delta t_i^{sub}]$ σ_{eq} and σ_{th}^{IASCC} are taken to be a constant values and denoted by σ_{eq}^{i} and $(\sigma_{th}^{IASCC})_i$ correspondingly (see Figure 5). The value of σ_{eq}^{i} is accepted as maximum value of σ_{eq} in subinterval $[t_{i-1}^{sub}, t_{i-1}^{sub} + \Delta t_i^{sub}]$: σ_{eq}^{i}=max{$\sigma_{eq}(t)$}, where t=$[t_{i-1}^{sub}, t_{i-1}^{sub} + \Delta t_i^{sub}]$. The value of $(\sigma_{th}^{IASCC})_i$ is accepted as minimum value of σ_{th}^{IASCC} in subinterval $[t_{i-1}^{sub}, t_{i-1}^{sub} + \Delta t_i^{sub}]$: $(\sigma_{th}^{IASCC})_i$=min{σ_{th}^{IASCC} (t)}, where t=$[t_{i-1}^{sub}, t_{i-1}^{sub} + \Delta t_i^{sub}]$.

Numeration of the subintervals Δt_i^{sub} in each interval Δt_k^{calc} begins from the start (see Figure 5).

8.4. For each Δt_i^{sub} in each interval Δt_k^{calc} the value of parameter $\Lambda_i = \frac{\sigma_{eq}^{i}}{\left(\sigma_{th}^{scc}\right)_i}$ and t_{f_i} for Λ_i is calculated; dependence $\Lambda(t_f)$ is given in Annex C

8.5. Cumulative damage over the time Δt_i^{sub} is calculated by

$$\Delta D_\tau = \frac{\Delta t_i^{sub}}{t_{f_i}}, \tag{5}$$

For each interval Δt_k^{calc} damage is calculated by

$$D_\tau^k = \sum_{i=1}^{m} \frac{\Delta t_i^{sub}}{t_{f_i}}, \tag{6}$$

where m is a number of subintervals Δt_i^{sub}, on which interval Δt_k^{calc} was divided.

Total damage is calculated by

$$D_\tau = \sum_{k=1}^{M} D_\tau^k, \tag{7}$$

where M is a number of intervals Δt_k^{calc}.

8.6. The critical event „IASCC crack nucleation" over the investigated operation period of time does not occur, if $D_\tau<1$.

8.7. If by the end of the investigated period of time $D_\tau>1$, then the instant in time t_{IASCC} at which $D_\tau=1$ is determined (counting off is performed from the beginning of RVI operation). If condition $t_{IASCC} < t_{fat}$ is fulfilled (see Section 7), calculation of crack growth is performed since $t= t_{IASCC}$ (Section 10).

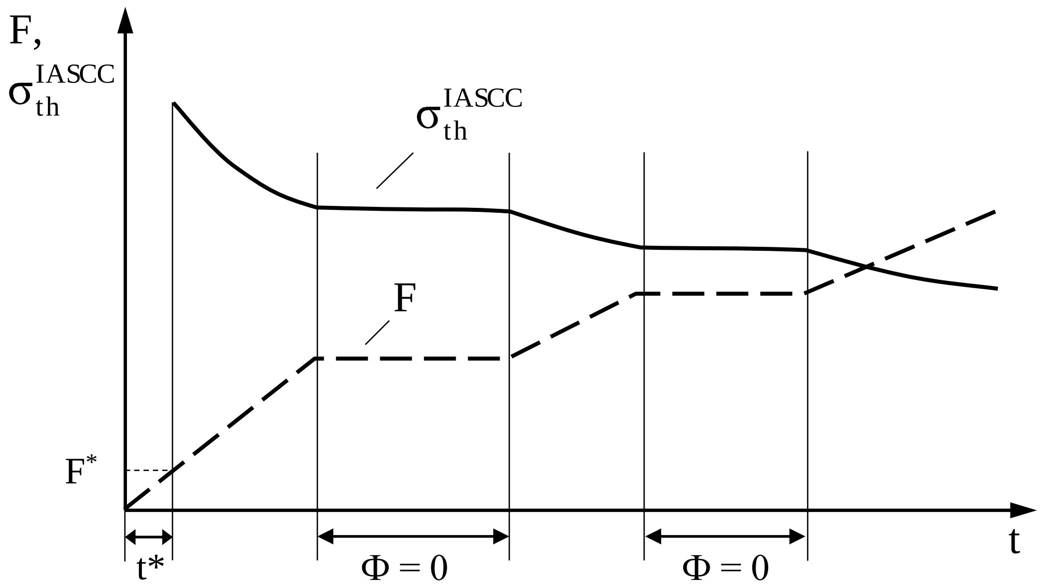


FIG. 3. Sample of construction of σ_{th}^{IASCC}*(t) on the basis of F(t) and* σ_{th}^{IASCC}*(F).*

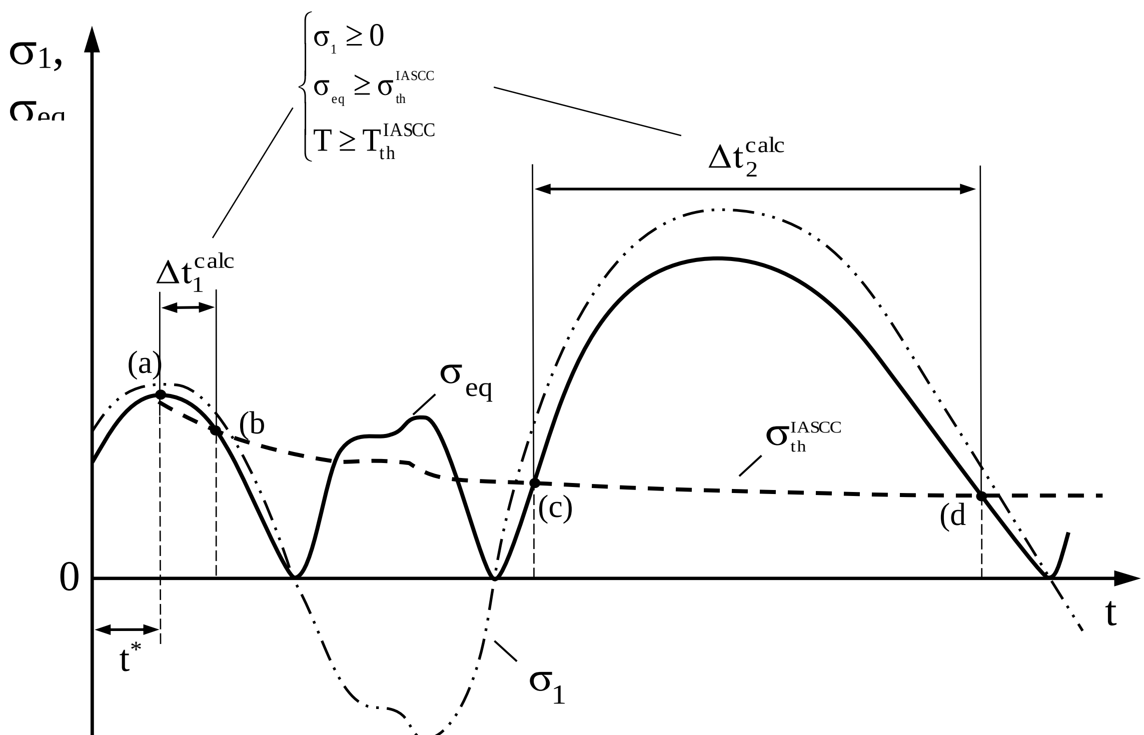


FIG. 4. Scheme of determination of intervals Δt_k^{calc} *for calculation of IASCC damages;*

——— σ_{eq}, -··-··-··- σ_1, - - - - - σ_{th}^{IASCC}.

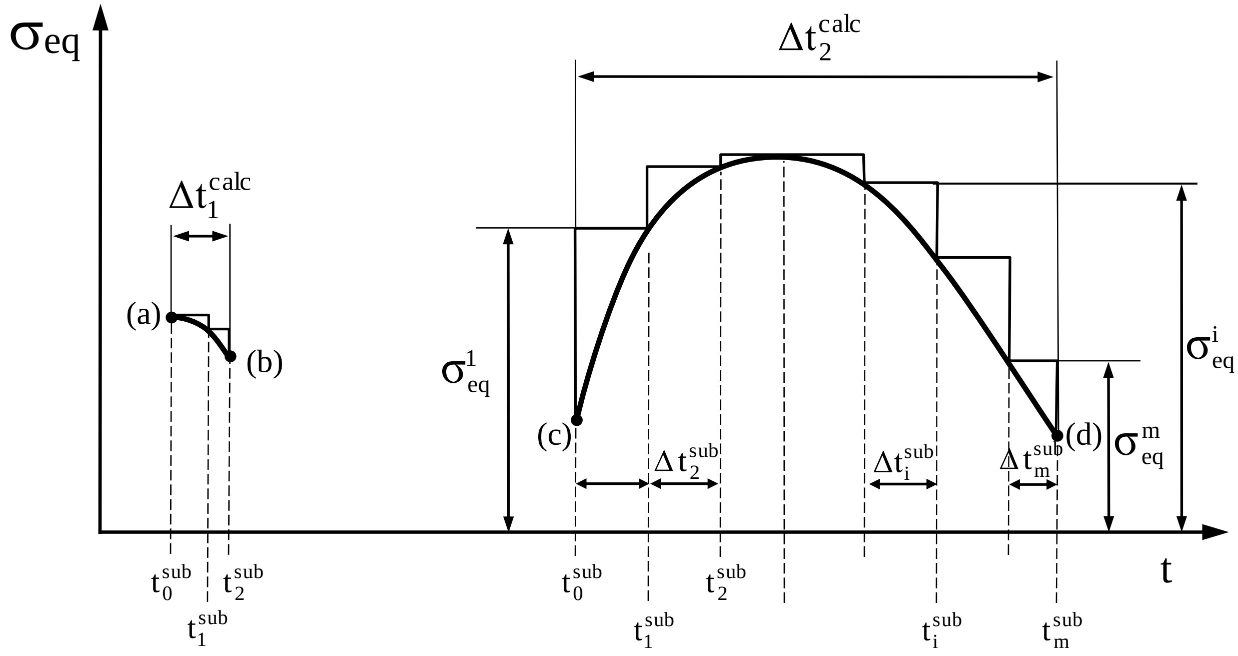


FIG. 5 .Scheme of dividing Δt_k^{calc} *intervals on subintervals* Δt_i^{sub}.

9. CRITICAL EVENT „FORMATION OF LIMIT EMBRITTLEMENT AREA“

9.1. The whole period of RVI operation is divided into time intervals Δt_j so that $t_{j+1} = t_j + \Delta t_j$, j = 1 to L, where L is a number of intervals. For each instant time t_j, the fields of neutron damage dose F and irradiation temperature T_{irr} are calculated for the considered component.

9.2. For each time instant t_j, for the considered internal component, a material area termed „the limit embrittlement area" (LEA) is determined which answers the following conditions.

This area is bordered by one or several closed surfaces, for each point on which the values of F and T_{irr} correspond to the $F_{crit}(T_{irr})$ curve. For all the inner points of LEA the values of F and T_{irr} lie above the $F_{crit}(T_{irr})$ curve. The $F_{crit}(T_{irr})$ curve is represented in Annex L.

A crack is assumed to exist in LEA, and the initial crack sizes a_0 and c_0 are determined according to Annex L.

9.3. For each crack postulated at instant in time t_j, its growth is calculated for time interval (t_{life}-t_j) in accordance with Section 10. This calculation is carried out depending on the type of postulated crack as follows.

— For an inner elliptical crack for which coolant does not penetrated into a crack tip, crack growth is calculated only by the fatigue mechanism for $\omega^{env} = 1$ (see Annex E).

— For surface semi-elliptic and quarter-elliptic cracks, crack growth is calculated both by the IASCC mechanism and by the fatigue mechanism allowing for the corrosive environment effect for $\omega^{env} > 1$ (see Annex E).

10. CRACK GROWTH AND CRITICAL EVENTS „UNSTABLE CRACK PROPAGATION“ AND „LOSS OF CARRYING CAPACITY“

10.1. For surface cracks growth calculation takes into account fatigue and IASCC mechanisms. For internal cracks growth calculation takes into account the fatigue mechanism only with ω^{env} = 1 (see Annex E).

10.2. For a crack postulated in compliance with 4.3.4, only analysis of unstable crack propagation in compliance with 10.11 and 10.12 is carried out. In this case the crack sizes $a_{\tau N}$ and $c_{\tau N}$ are accepted as sizes of nucleus cracks in compliance with 6.3. Calculations of crack growth by IASCC and fatigue mechanisms are not carried out.

10.3. Calculations of surface crack growth are carried out in two stages. In the first stage a calculation is carried out for crack growth by the IASCC mechanism over the whole investigated operational period. In the second stage the calculation is carried out over the same period for a crack growth by the fatigue mechanism. Crack sizes obtained at the end time of the first stage are used as the initial crack sizes for the second stage.

10.4. The sequence of loading conditions over the investigated period of time is determined according to actual and (or) predicted models of RVI operation.

In the case of crack growth in a weld joint when fulfilling 4.3.3.1 the investigated time period begins from the moment of the beginning of RVI operation.

In the case of the analysis of crack growth in a base metal and weld metal when fulfilling 4.3.3.2, the investigated time period begins from the time instant of crack nucleation. The time of crack nucleation is calculated as the minimum time of crack nucleation by the fatigue mechanism and the time of crack nucleation by the IASCC, i.e. $t_{nuc}=\min(t_{fat}, t_{IASCC})$. In the case of the analysis of crack growth in LEA, time of crack nucleation is determined to be in compliance with Section 9 and Annex L.

10.5. SSF of a structural component is calculated in two variants: by solving the elastic problem in an unsteady quasi-static non-isothermic statement, and by solving the viscoelastic-plastic problem in an unsteady quasi-static non-isothermic statement.

The elastic problem is solved to determine crack growth by the fatigue mechanism.

The viscoelastic plastic problem is solved to determine crack growth by the SSC and for analysis of unstable crack propagation.

10.6. For each loading condition SSF of a structural component is determined by means of solving an elastic problem. If swelling S_0 calculated by Equation (G.2) with $T=T_m$ (Annex G) does not exceed the value $1.0\cdot10^{-5}$, then SSF calculation is made without regard for swelling. With $S_0 > 1.0\cdot10^{-5}$ calculation of SSF is made with regard to swelling.

Notes:

(a) With $S_0 > 1.0 \times 10^{-5}$ it is allowable to calculate SSF without regard for swelling, however in doing so the stress ratio in Eqs. (12) and (13) is taken to be equal to 0.95.

(b) When solving the elastic problem with swelling it is allowable to disregard the influence of stress σ_{eff} on swelling (that is, it is allowable to assume in Equation (G.1) $f_1(\sigma_{eff})=1$).

10.7. If condition in 4.3.3.1 is fulfilled then based on the solution results of the elastic problem the most loaded welds of a structural component are determinated and it is assumed that the postulated crack is located in a weld joint. The shape and initial sizes of the postulated crack are taken according to Section 6. The postulated crack is oriented so that its growth over the investigated period of time is the greatest.

When analyzing a crack in a base metal or in weld joint, for which condition 4.3.3.2 is fulfilled, the location of a crack is determined by the location of crack nucleation according to the critical events 4.2.1.1-4.2.1.3. The diagram of the postulated crack (shape and initial sizes) is taken according to Section 6. For a postulated crack located in LEA the initial sizes are taken in compliance with Section 9 and Annex L. A postulated crack is oriented so that its growth over the investigated period of time is the greatest.

For analysis of strength condition in compliance with 4.3.4 the location and orientation of postulated crack is assumed so that its resistance to unstable crack propagation is minimal.

10.8. To solve the viscoelastic plastic problem the whole RVI operational period is divided into time intervals Δt_j, so that $t_{j+1} = t_j + \Delta t_j$; j=1 to L. For each time interval Δt_j the viscoelastic plastic problem is solved with regard to swelling and loading history proceeding in the time interval from 0 to t_j. Solution of the viscoelastic plastic problem is carried out by stepwise tracing of a loading history and the time steps for solving FEM of the indicated problem being $\Delta\tau_k^{FEM} << \Delta t_j$, where $\Delta\tau_k^{FEM}$ is a time step on solving FEM of the viscoelastic plastic problem at which loading is close to a radial one. The constitutive equations for solving the viscoelastic plastic problem are given in Annex K.

In each time interval Δt_j mechanical properties are appropriate to the damage dose of neutron irradiation F_{j+1} that corresponds to the instant time t_{j+1} (end of time interval Δt_j).

Note: To improve the accuracy and decrease the calculation conservatism it is recommended to increase the number of time intervals L.

10.9.. Calculation of a crack growth by the mechanism of IASCC is carried out.

10.9.1. The whole investigated loading period is divided into the time intervals $\Delta t_i << \Delta t_j$. A decrease of the number of intervals results in a more conservative estimation of crack growth on IASCC.

10.9.2. For the time interval Δt_i the values of SIF of K_a^i and K_c^i in points a and c are calculated (see Figs. 1 and 2), the values are taken to be constant and equal to their maximum values in the interval Δt_i. The values K_a^i and K_c^i are calculated on the basis of distribution of normal to crack stress σ_n that are determined by solving the viscoelastic plastic problem and crack sizes in the previous time interval a_{i-1} and c_{i-1}.

Note: It is recommended to calculate the values K_a^i and K_c^i by the method of weight functions (MWF).

10.9.3. If in the time interval Δt_i SIF of $K_a^i < 0$, it is taken that $K_a^i = 0$. If in the time interval Δt_i SIF of $K_c^i < 0$, it is taken that $K_c^i = 0$.

10.9.4. Based on the dependences given in Annex F the values of crack growth Δa_i and Δc_i are determined by the equations

$$\Delta a_i = A_\tau \left(K_a^i\right)^{n_\tau} \cdot \Delta t_i, \tag{8}$$

$$\Delta c_i = A_\tau \left(K_c^i\right)^{n_\tau} \cdot \Delta t_i, \tag{9}$$

where A_τ and n_τ are material constants, they are determined according to Annex F.

10.9.5. Current sizes of a crack are determined by the equations:

$$a_i = a_{i-1} + \Delta a_i, \tag{10}$$

$$c_i = c_{i-1} + \Delta c_i, \tag{11}$$

10.9.6. The calculations according to 10.9.2-10.9.5 are repeated until the end of tracing of the investigated loading period. As a result of such procedure, the sizes a_τ and c_τ of the postulated crack are determined for the end of the investigated loading period at the expense of the IASCC

10.10. Calculation of crack growth by the fatigue mechanism is carried out. The calculation is carried out only for the loading conditions for which the value ΔK exceeds ΔK_{th}

10.10.1. The sizes a_τ and c_τ obtained in 10.9.6 are considered as the initial crack sizes

10.10.2. The calculation of crack growth is based on the results of solution of the elastic problem according to 10.6 and the procedure described in 10.10.3 - 10.10.6

10.10.3. Based on the calculations performed according to 10.6 the determination is carried out of the values ΔK_a^i and R_a^i, ΔK_c^i and R_c^i for the i-th loading cycle where ΔK_a^i and ΔK_c^i are mode I SIF ranges in points a and c of a crack (see Figs. 1 and 2); R_a^i and R_c^i are the stress ratios in points a and c of a crack.

Notes:

(a) The values ΔK_a^i, R_a^i and ΔK_c^i, R_c^i can be determined directly from solution of the elastic problem by FEM or by means of MWF. The distribution of stresses in a crack plane when a crack is absent, as well as crack sizes at the end of the previous loading cycle a_{i-1} and c_{i-1} .are used as the initial information for MWF.

(b) If $\Delta K_a^i < \Delta K_{th}(R_a^i)$, then under the i-th loading condition the crack growth in the direction of a minor semiaxis is absent. If $\Delta K_c^i < \Delta K_{th}(R_c^i)$, then under the i-th loading condition the crack growth in the direction of a major semiaxis is absent. The equations for determination of ΔK_{th} are given in Annex E.

(c) It is assumed that:

— If R_a^i < 0, then R_a^i = 0;

— If R_a^i > 0.95, then R_a^i = 0.95;

— If $R_c^i < 0$, then $R_c^i = 0$;

— If $R_c^i > 0.95$, then $R_c^i = 0.95$.

10.10.4. Based on the dependences given in Annex E the values of crack propagation Δa_i and Δc_i per cycle are determined by:

$$\Delta a_i = \omega^{env} \cdot C_f \cdot \left[\frac{\Delta K_a^i}{(1 - R_a^i)^{0,25}} \right]^{n_f}, \quad (12)$$

$$\Delta c_i = \omega^{env} \cdot C_f \cdot \left[\frac{\Delta K_c^i}{(1 - R_c^i)^{0,25}} \right]^{n_f}. \quad (13)$$

where ω^{env} is acceleration factor of FCGR of irradiated metal in WWER water as compared with inert (air) environment; C_f and n_f are empirical constants of the Paris equation.

The values of factors ω^{env}, C_f and n_f are determined according to Annex E. The value of factor C_f is determined for the damage dose of neutron irradiation at the instant in time that corresponds to the end of the i-th cycle of loading.

Note: If it is not possible to recognize a loading cycle (SIF at the beginning and end of a cycle should be the same), then the characteristics of a loading cycle (ΔK, R) should be taken as a characteristic of a semi-cycle. In this case the number of cycles should be taken to be equal to the number of semi-cycles.

10.10.5. Current sizes of a crack are determined by the following equations:

$$a_i = a_{i-1} + \Delta a_i, \quad (14)$$

$$c_i = c_{i-1} + \Delta c_i. \quad (15)$$

10.10.6. The calculation is repeated in accordance with 10.10.3 - 10.10.5 until all loading cycles over the investigated operational period are taken into account. As a result the sizes $a_{\tau N}$ and $c_{\tau N}$ of the postulated crack at the end of the investigated operation period are determined.

10.11. The critical event „Unstable crack propagation“ does not occur, if under all operational conditions the following conditions are fulfilled

$$n_J \cdot J^e(a_{\tau N}, c_{\tau N}, P_p, P_s) < J_c(F, T, T_{irr}), \quad (16)$$

where

J^e is elastic part of J-integral depending on crack sizes $a_{\tau N}$ and $c_{\tau N}$, as well as on primary loads P_p and secondary loads P_s;

n_j is a safety factor, it is taken as n_j=1.2;

J_c is the critical value of a J-integral depending on a damage dose of neutron irradiation F, irradiation temperature T_{irr} and current operational temperature T; J_c is calculated according to Annex D at T_{irr} under NOC.

The critical event „Loss of carrying capacity“ does not occur, if under all operation conditions the following conditions are fulfilled

$$\sigma_{ref}(a_{\tau N}, c_{\tau N}, P_p) < \sigma_Y^0(T)\,, \tag{17}$$

where σ_{ref} is reference stress depending on crack sizes $a_{\tau N}$ and $c_{\tau N}$ and on stresses caused by primary loading P_p; σ_Y^0 is yield stress for unirradiated condition (F=0).

Condition Equation (16) is appropriate to the provision of strength by the criterion of initiation of unstable crack growth. Condition Equation (17) is appropriate to the provision of load-carrying capacity of a component with a crack.

In Equation (17) σ_{ref} is calculated according to Annex J; dependence $\sigma_Y^0(T)$ is calculated for a base metal and weld metal according to Annex A as for a base metal.

It should be additionally shown that condition Equation (16) is fulfilled for any crack sizes less than $a_{\tau N}$ and $c_{\tau N}$. In this case the loading conditions (F, T and T_{irr}) in Equation (16) should correspond to current crack sizes

10.12. If conditions in Eqs. (16) and (17) are not fulfilled, it is allowed to analyze the critical events for a through-thickness crack.

The critical event „Unstable crack propagation“ does not occur, if under all operating conditions the following conditions are fulfilled

$$n_J \cdot J^e(2c_{\tau N}, P_p, P_s) < J_c(F, T, T_{irr})\,, \tag{18}$$

The critical event „Loss of carrying capacity“ does not occur, if under all operation conditions the following conditions are fulfilled

$$\sigma_{ref}(2c_{\tau N}, P_p) < \sigma_Y^0(T)\,. \tag{19}$$

In Equation (18) J_c depends on a damage dose of neutron irradiation F, irradiation temperature T_{irr} and current operation temperature T; J_c is calculated according to Annex D at T_{irr} under NOC;

In Equation (19) σ_{ref} is calculated according to Annex J; dependence $\sigma_Y^0(T)$ is calculated for a base metal and weld metal according to Annex A as for a base metal in the unirradiated condition.

10.13. In sections 10.11 and 10.12 loading conditions for actual and (or) predicted models of RVI operation are considered.

10.14. The calculation of J^e in 10.9 and 10.11 may be carried out using the following equation:

$$J^e = (1-\upsilon^2)\cdot\frac{(K_I)^2}{E}\,, \tag{20}$$

where K_I – SIF, calculated by MWF using as initial information the distributions of stresses normal to a crack plane calculated when a crack is absent; this calculation is performed by means of FEM solution of the viscoelastic-plastic problem with regard for swelling. The constitutive equations for the viscoelastic-plastic problem are given in Annex K; ν is Poisson ratio; E is Young's modulus of elasticity, calculated by Equation (B.8) given in Annex B.

11. CRITICAL EVENT „INADMISSIBLE CHANGE OF GEOMETRICAL SIZES OF A COMPONENT“

11.1. The calculation according to this section is carried out for RVI components whose change of geometrical sizes is limited by some specified values. The excess of these specified values results in violation of normal operation of a structural component or RVI equipment components.

11.2. The calculation of the change of geometrical sizes of a component is carried out by means of solution of the viscoelastic-plastic problem taking into account swelling and radiative creep in a finite strain statement.

Note: It is allowable to make the calculation only for the NOC conditions

The whole investigated period of RVI operation is divided into the time intervals Δt_j, so that $t_{j+1} = t_j + \Delta t_j$; j=1 to L, where L is a number of Δt_j intervals.

For each time interval Δt_j the viscoelastic-plastic problem is solved taking into account a loading history proceeding in the time interval from 0 to t_j. In each time interval Δt_j the tensile properties are appropriate to the damage dose of neutron irradiation F_{j+1} that corresponds to the time t_{j+1} (end of the time interval Δt_j).

The solution of the viscoelastic-plastic problem is performed by means of stepwise tracing of a loading history, the time steps for FEM solution of the indicated problem being $\Delta\tau_k^{FEM} \ll \Delta t_j$ where $\Delta\tau_k^{FEM}$ is a time step on FEM solution of the viscoelastic-plastic problem and loading on this step is close to a radial one.

Notes:

(a) It is allowable in this calculation to use the data on tensile properties that are appropriate to the end of the investigated period of operation as the initial information.

(b) To improve the accuracy and decrease the calculation conservatism it is recommended to increase the number of time intervals L.

11.3. The critical event „Inadmissible change of geometrical sizes of a component" over the investigated period of time does not occur, if the following condition is fulfilled:

$$\Delta H_i < [\Delta H]_i, \tag{21}$$

where ΔH_i is the change of the geometrical size of the component in the i-th direction; $[\Delta H]_i$ is the allowable changes of geometrical sizes.

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1. LIST OF ANNEXES TO APPENDIX C

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ANNEX A TO APPENDIX C

MECHANICAL PROPERTIES AND STRESS-STRAIN CURVES FOR RVI MATERIALS

A-1. DEPENDENCE OF YIELD STRENGTH ON OPERATIONAL (TEST) TEMPERATURE, NEUTRON DAMAGE DOSE AND IRRADIATION TEMPERATURE

A-1.1. The temperature dependence of yield strength for a base metal and a weld metal in the initial (unirradiated, F=0) condition over the operational temperature range is described by the following equation:

$$\sigma_Y^0(T) = \sigma_{YG} + \sigma_{YS}\,,$$
$$\sigma_{YS} = \beta \cdot \exp\left(-h\left[T + 273\right]\right), \qquad (A.1)$$

where T is the operational (test) temperature in °C; σ_{YG}, σ_{YS} – temperature independent and temperature dependent components of yield strength, correspondingly, β and h – material constants independent of temperature.

To determine parameters β and σ_{YG} it is recommended to use the certified values $\sigma_Y^0(T)$ (average values) obtained at T=20°C and T=350°C. In this case the parameter h may be taken from Table A.1. When there is no possibility to determine β and σ_{YG} based on the certified values it is allowable to use the coefficients given in Table A.1.

TABLE A.1. THE VALUES OF CONSTANTS σ_{YG}, β AND H FOR RVI MATERIALS

Material	σ_{YG}, MPa	β, MPa	h, K^{-1}
Base metal	155	239	2.22×10^{-3}
Weld metal	255	420	2.22×10^{-3}

A-1.2. The dependence of yield strength for a base metal and a weld metal on a neutron damage dose F, operation (test) temperature T and irradiation temperature T_{irr} is described by the equation

$$\sigma_Y\left(T, F, T_{irr}\right) = \sigma_Y^{eff}\left(T, F\right) \cdot \left(1 - \overline{A}_v\left(F, T_{irr}\right)\right);\ T \le T_{irr} \qquad (A.2)$$

where σ_Y^{eff} – the effective yield strength (yield strength for material without swelling); $\overline{A}_v$ – the relative area of vacancy voids, generated under irradiation

$$\overline{A}_v = \left(\frac{S}{1+S}\right)^{2/3}, \qquad (A.3)$$

In Equation (A.3) S is a radiation swelling calculated according to Annex G.

A-1.3. The dependence of effective yield strength for a base metal on a neutron damage dose F and operation (test) temperature T is described by the equation

$$\sigma_Y^{eff}(T,F)=\begin{cases}\sigma_Y^0(T)+\Delta\sigma_Y^T(T,F) & \text{for } F<F_{stab}\\ \sigma_Y^*(T)+\Delta\sigma_Y^*(F) & \text{for } F\geq F_{stab}\end{cases};\ T\leq T_{irr} \tag{A.4}$$

where

$\sigma_Y^0(T)$	is the temperature dependence of yield strength in the initial (irradiated) condition;
$\sigma_Y^*(T)$	is the temperature dependence of yield strength for damage dose $F=F_{stab}$;
$\Delta\sigma_Y^T(T,F)$	is the dependence of yield strength increment on a damage dose of neutron irradiation and test temperature over the dose range $0\leq F<F_{stab}$;
$\Delta\sigma_Y^*(F)$	is the dependence of yield strength increment on a damage dose of neutron irradiation for damage doses $F\geq F_{stab}$;
F_{stab}	is the dose, after which yield strength increment does not depend on operation (test) temperature; F_{stab} = 7 dpa.

A-1.4. The temperature dependence of yield strength for neutron damage dose $F=F_{stab}$ is described by equation

$$\sigma_Y^*(T)=650+1405\cdot\exp\left(-5.9\cdot10^{-3}\cdot(T+273)\right),\ \text{MPa}. \tag{A.5}$$

A-1.5. The dependence of yield strength increment on a neutron damage dose and operation (test) temperature over the dose range $0\leq F<F_{stab}$ is described by the equation

$$\Delta\sigma_Y^T(T,F)=A_{\sigma_Y}^T\cdot\sqrt{1-\exp(-0.126\cdot F)}, \tag{A.6}$$

where

$$A_{\sigma_Y}^T=\frac{\sigma_Y^*(T)-\sigma_Y^0(T)}{\sqrt{1-\exp(-0.126\cdot F^*)}}, \tag{A.7}$$

A-1.6. The dependence of yield strength increment on a neutron damage dose for damage doses $F\geq F_{stab}$ is described by the equation

$$\Delta\sigma_Y^*(F)=621\cdot\left(\sqrt{1-\exp(-0.126\cdot F)}-\sqrt{1-\exp(-0.126\cdot F^*)}\right), \tag{A.8}$$

A-1.7. The dependence of effective yield strength for a weld metal on a neutron damage dose F and operational (test) temperature T is described by the equation

$$\sigma_Y^{eff}(T,F)=\sigma_Y^0(T)+\Delta\sigma_Y(F);\ T\leq T_{irr} \tag{A.9}$$

where $\Delta\sigma_Y(F)$ is the dependence of yield strength increment for a weld metal on a neutron damage dose

$$\Delta\sigma_Y(F)=498\cdot\sqrt{1-\exp(-0.3\cdot F)},\ \text{MPa} \tag{A.10}$$

Note: Equations (A.2), (A.4) and (A.9) are true at $T \leq T_{irr}$.

A-2. DEPENDENCE OF ULTIMATE STRENGTH ON OPERATION (TEST) TEMPERATURE, NEUTRON DAMAGE DOSE AND IRRADIATION TEMPERATURE

A-2.1. The temperature dependence of ultimate strength for a base metal and a weld metal in the initial condition is described by the equation

$$\sigma^0_{ul}(T) = \sigma_{UG} + \beta_U \cdot \exp(-h_U \cdot T), \text{ MPa, } 20°C \leq T \leq 450°C, \quad \text{(A.11)}$$

where T is the operational (test) temperature in ºC.

To determine parameters β_U and σ_{UG} it is recommended to use the certified values of ultimate strength obtained at T=20°C and T=350°C. In this case the parameter h_U may be taken from Table A.2. When there is no possibility to determine β_U and σ_{UG} based on the certified values it is allowable to use the coefficients given in Table A.2.

TABLE A.2. THE VALUES OF CONSTANTS σ_{UG}, β_U AND H_U FOR RVI MATERIALS

Material	σ_{UG}, MPa	β_U, MPa	h_U, C^{-1}
Base metal	350	247	$6.6 \cdot 10^{-3}$
Weld metal	439	222	$9.74 \cdot 10^{-3}$

A-2.2. The dependence of ultimate strength for a base metal and a weld metal on a neutron damage dose F, operational (test) temperature T and irradiation temperature T_{irr} is described by the equation:

$$\sigma_{ul}(T, F, T_{irr}) = \sigma^{eff}_{ul}(T, F) \cdot \left(1 - \overline{A}_v(F, T_{irr})\right); \; T \leq T_{irr} \quad \text{(A.12)}$$

where σ^{eff}_{ul} is the effective ultimate strength (ultimate strength for material without swelling); $\overline{A}_v$ is the relative area of vacancy voids, generated under irradiation calculated according to Equation (A.3).

A-2.3. The dependence of effective ultimate strength on the neutron damage dose F and operation (test) temperature T is described by the equation:

$$\sigma^{eff}_{ul}(T, F) = \sigma^0_{ul}(T) + \Delta\sigma_{ul}(F); \; T \leq T_{irr} \quad \text{(A.13)}$$

where $\Delta\sigma_{ul}(F)$ is the dependence of ultimate strength increment on a neutron damage dose.

A-2.4. The dependence of ultimate strength increment on a neutron damage dose is described by the equation:

$$\Delta\sigma_{ul}(F) = A_{\sigma_{ul}} \cdot \sqrt{1 - \exp\left(-C_{\sigma_{ul}} \cdot F\right)}, \text{ MPa} \quad \text{(A.14)}$$

where for a base metal $A_{\sigma_{ul}}$ = 483 MPa, $C_{\sigma_{ul}}$ = 0.11; for a weld metal $A_{\sigma_{ul}}$ = 440 MPa, $C_{\sigma_{ul}}$ = 0.25;

A-3. STRESS-STRAIN CURVE

A-3.1. The stress-strain curve for a base metal and a weld metal for an operational (test) temperature T and a neutron damage dose F with regard for swelling is approximated by the equation

$$\sigma_{eq} = \left(\sigma_Y^{eff}(F,T) + A(F,T)\cdot(æ_p)^{n(F,T)}\right)\cdot\left(1-\overline{A}_v\left(F,T_{irr}\right)\right), \tag{A.15}$$

where

$æ_p = \int d\varepsilon_{eq}^p$	;
$d\varepsilon_{eq}^p$	is the equivalent plastic strain increment;
σ_Y^{eff}	is the effective yield strength (yield strength for material without swelling);
A(F,T,) and n(F,T)	are material parameters that depend on a neutron damage dose F and operational (test) temperature T;
$\overline{A}_v$	is the relative area for vacancy voids, generated under irradiation calculated according (A.3)..

A-3.2. The parameter **A** of a stress-strain curve is calculated by the following equations:

— for a base metal

$$A(T,F) = \begin{cases} b_1 - b_2\cdot T - b_3\cdot\left(\sigma_Y^{eff}(T\ ,F) - \sigma_Y^0(T)\right) & \text{for } 20 < T\ \le T^* \\ b_1 - b_2\cdot T^* - b_3\cdot\left(\sigma_Y^{eff}(T^*\ ,F) - \sigma_Y^0(T^*)\right) & \text{for } T^* < T\ \le 450°C \end{cases}, \text{ MPa,} \tag{A.16}$$

where T^* =290°C, b_1=982 MPa, b_2=1.93 MPa/°C, b_3 = 0.158;

— for a weld metal

$$A(F,T) = b_1 - b_2\cdot T - b_3\cdot\left(\sigma_Y^{eff}(T\ ,F) - \sigma_Y^0(T)\right), \tag{A.17}$$

where b_1=734.4 MPa, b_2=0.77 MPa/°C, b_3 = 0.337, T is the current operational (test) temperature in ^{o}C;

A-3.3. The parameter **n** of a stress-strain curve is calculated from following combined equations:

$$n = \frac{\varepsilon_{ul}^{calc}}{1 - \frac{\sigma_Y^{eff}}{\sigma_{ul}^{eff} \cdot \exp\left(\varepsilon_{ul}^{calc}\right)}}, \tag{A.18}$$

$$A = \frac{\sigma_Y^{eff}}{\left(\varepsilon_{ul}^{calc}\right)^{n-1} \cdot \left(n - \varepsilon_{ul}^{calc}\right)}. \tag{A.19}$$

where

ε_{ul}^{calc} is the plastic deformation corresponding to plastic instability (beginning of neck formation) of cylindrical tensile specimen;

σ_Y^{eff} is the effective yield strength calculated according (A.4) for a base metal or (A.9) for a weld metal;

σ_{ul}^{eff} is the effective ultimate strength calculated according (A.13).

Note: When calculating SSF under stationary operational conditions, and when using dependences given in Annex A, it is necessary to take into account that a current operational (test) temperature T coincides with an irradiation temperature T_{irr}. Under transient conditions an irradiation temperature T_{irr} differs from a current operational temperature T. In this case it is assumed that T_{irr} corresponds to the temperature of the previous stationary operational conditions.

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ANNEX B TO APPENDIX C

CONSTRUCTION OF LOW-CYCLE FATIGUE CURVES FOR RVI MATERIALS

B-1. The design fatigue curve $\Delta\varepsilon(N_f)$ for a base metal for an operation temperature $T \leq 450°C$ is determined as the lower envelope of fatigue curves corresponding to the temperatures 350ºC and 450ºC:

$$\Delta\varepsilon(N_f) = \min\begin{cases} [\,\Delta\varepsilon(N_f)\,]_{T=350^oC} \\ \\ [\,\Delta\varepsilon(N_f)\,]_{T=450^oC} \end{cases}. \tag{B.1}$$

where the curve for 350ºC should be constructed with regard for the influence of corrosion environment and the curve for 450ºC must be constructed without regard for environment:

B-1.1. The fatigue curves for a base metal for temperatures 350ºC and 450ºC are determined with regard for the strain range safety factor n_ε and cycle number safety factor n_n:

$$[\Delta\varepsilon(N_f)]_T = \min\begin{cases} \Delta\varepsilon_{n_\varepsilon}(N_f) \\ \Delta\varepsilon_{n_N}(N_f) \end{cases}. \tag{B.2}$$

B-1.1.1. With the provision that $\sigma_{max} \leq \dfrac{E\Delta\varepsilon}{2}$ (σ_{max} is peak stress in a cycle) the fatigue curve of a base metal with regard for the strain range safety factor and fatigue curve of a base metal with regard for the cycle number safety factor are calculated with R=-1 in the following way:

$$\Delta\varepsilon_{n_\varepsilon} = \frac{\varepsilon_f\,(4N_f)^{-0.5}}{n_\varepsilon} + \frac{2R_c}{n_\varepsilon E(4N_f)^{m_e}}. \tag{B.3}$$

$$\Delta\varepsilon_{n_N} = \varepsilon_f\,(4n_N N_f)^{-0.5} + \frac{2R_c}{E(4n_N N_f)^{m_e}}. \tag{B.4}$$

B-1.1.2. With the provision that $\sigma_{max} > \dfrac{E\Delta\varepsilon}{2}$ or the unknown σ_{max} (unknown stress ratio) the fatigue curve of a base metal with regard for the strain range safety factor and fatigue curve of a base metal with regard for the cycle number safety factor are calculated with maximum stress ratio from solution of the following nonlinear equations:

$$\Delta\varepsilon_{n_\varepsilon} = \begin{cases} \dfrac{\varepsilon_f (4N_f)^{-0.5}}{n_\varepsilon} + \dfrac{2R_c}{n_\varepsilon E\left((4N_f)^{m_e} + \dfrac{2\sigma_Y - n_\varepsilon \Delta\varepsilon_{n_\varepsilon} E}{n_\varepsilon \Delta\varepsilon_{n_\varepsilon} E} \right)}, \text{ at } 2\sigma_Y - n_\varepsilon \Delta\varepsilon_{n_\varepsilon} E > 0 \\ \dfrac{\varepsilon_f (4N_f)^{-0.5}}{n_\varepsilon} + \dfrac{2R_c}{n_\varepsilon E (4N_f)^{m_e}}, \text{ at } 2\sigma_Y - n_\varepsilon \Delta\varepsilon_{n_\varepsilon} E \le 0 \end{cases} \tag{B.5}$$

$$\Delta\varepsilon_{n_N} = \begin{cases} \varepsilon_f (4n_N N_f)^{-0.5} + \dfrac{2R_c}{E\left((4n_N N_f)^{m_e} + \dfrac{2\sigma_Y - \Delta\varepsilon_{n_N} E}{\Delta\varepsilon_{n_N} E} \right)}, \text{ at } 2\sigma_Y - \Delta\varepsilon_{n_N} E > 0 \\ \varepsilon_f (4n_N N_f)^{-0.5} + \dfrac{2R_c}{E (4n_N N_f)^{m_e}}, \text{ at } 2\sigma_Y - \Delta\varepsilon_{n_N} E \le 0 \end{cases} \tag{B.6}$$

where in Eqs. (B.3), (B.4), (B.5) and (B.6)

$\Delta\varepsilon(N_f)$ is allowable strain range;
N_f is allowable number of cycles with strain ranges $\Delta\varepsilon$ (according to the limit condition 4.2.1.1);
ε_f is fracture strain with regard for effect of damage dose and with or without regard for effect of environment according to Eqs. (B.1) and (B.11)-(B.14);
n_ε strain range safety factor; it is assumed that $n_\varepsilon = 2$;
n_N is cycle number safety factor; it is assumed that $n_N = 10$;
R_c is true fracture stress obtained from standard tensile test of cylindrical smooth specimens;
E is Young's modulus of elasticity depending on temperature T;
m_e is material constant;
σ_Y is yield strength.

B-1.1.3. True fracture stress obtained from standard tensile test of cylindrical smooth specimens may be calculated by the equation:

$$R_c=\sigma_{ul} \cdot 1{,}4(1-\exp(-\varepsilon_f))], \tag{B.7}$$

where σ_{ul} is ultimate tensile strength; $\varepsilon_f = -\ln(1-Z/100\%)$; Z – reduction of area, %.

B-1.1.4 Young's modulus of elasticity (in MPa) is calculated by the equation:

$$E = \begin{cases} 196000 - 72.423 \cdot (T - 20), \quad \text{at} \quad 20°C \le T \le 375°C \\ 206000 - 95.94 \cdot T, \quad \text{at} \quad 375°C < T \le 450°C \end{cases}, \tag{B.8}$$

where T is temperature in ºC.

B-1.1.4. The parameter m_e is calculated by the equation:

$$m_e = 0.132 \cdot \log_{10}\left[\frac{\sigma_{ul}}{\sigma_{-1}}\left(1+1.4\cdot\left(1-\exp\left(-\varepsilon_f^0\right)\right)\right)\right], \tag{B.9}$$

where

σ_{-1} is fatigue limit at R=-1;

ε_f^0 is fracture strain of a material in the initial condition; ε_f^0 is recommended to determine on the basis of the certificate data (average values); If the value determined from the certificate is ε_f^0>0.69, then it is assumed that ε_f^0=0.69; In the absence of the certificate data it is assumed that ε_f^0=0.69.

B-1.1.5. Fatigue limit is determined from the equation:

$$\sigma_{-1} = \begin{cases} 0.4\cdot\sigma_{ul}, & \text{at} \quad \sigma_{ul} \le 700\ \text{MPa} \\ (0.54 - k\cdot\sigma_{ul})\cdot\sigma_{ul}, & \text{at} \quad \sigma_{ul} > 700\ \text{MPa} \end{cases}, \tag{B.10}$$

where $k = 2\cdot10^{-4}$ 1/MPa.

B-1.1.6. Calculation of yield strength σ_Y: The yield strength σ_Y is calculated according to Annex A to Appendix C.

B-1.1.7. Calculation of ultimate tensile strength σ_{ul}: The ultimate strength σ_{ul} is calculated according Annex A to Appendix C.

B-1.1.8. The calculation of the fracture strain obtained from standard tensile test of cylindrical smooth specimens with regard for neutron irradiation and environment effect may be calculated by the following equations:

$$\varepsilon_f(F,T_{irr}) = \varepsilon_f^0\cdot\left(1 - A_\varepsilon(T_{irr})\cdot\sqrt{\left(1-\exp\left[-0.08\cdot F\right]\right)}\right), \tag{B.11}$$

$$Z(F,T_{irr}) = 1 - \exp\left(-\varepsilon_f(F,T_{irr})\right) \tag{B.12}$$

$$Z^{env}(F,T_{irr}) = Z(F,T_{irr})\cdot\left(F_{env}\right)^{-0.5} \tag{B.13}$$

$$\varepsilon_f^{env}(F,T_{irr}) = -\ln\left(1 - Z^{env}(F,T_{irr})\right) \tag{B.14}$$

where

$Z^{env}(F,T_{irr})$ is reduction of area for given level of damage F and irradiation temperature T_{irr} with regard for environment effect;

ε_f^{env}(F, is fracture strain for given level of damage F and irradiation temperature T_{irr} with regard for environment effect.

T_{irr})

F_{env} is a coefficient taking into account an environment effect.

$$A_\varepsilon = \begin{cases} 0.4 \text{ at } 250^\circ C \le T_{irr} \le 300^\circ C \\ 0.4 + 0.003(T_{irr} - 300) \\ 0.70 \text{ at } 400^\circ C < T_{irr} \le 450^\circ C \end{cases}; \qquad \text{(B.15)}$$

B-1.1.9. If investigated component zone is in contact with coolant the calculation of the coefficient F_{env} is made by the equation:

$$F_{env} = \exp(0 - T^* \cdot k^* \cdot \varepsilon^*), \qquad \text{(B.16)}$$

where $k^* = 0.281$;

$$T^* = \begin{cases} 0 & \text{at } T < 150^o C \\ \dfrac{T - 150}{175} & \text{at } 150^o C \le T \le 325^o C; \\ 1 & \text{at } T > 325^o C \end{cases} \qquad \text{(B.17)}$$

$$\varepsilon^* = \begin{cases} 0, & \text{at } \xi \ge 0.004 s^{-1} \\ \ln\left(\dfrac{\xi_{min}^{ten}}{0.004}\right), & \text{at } 4 \cdot 10^{-6} s^{-1} < \xi < 0.004 s^{-1} \\ \ln(0.001), & \text{at } \xi \le 4 \cdot 10^{-6} s^{-1} \end{cases}, \qquad \text{(B.18)}$$

where ξ_{min}^{ten} is the minimum equivalent strain rates in a tensile semi-cycle (in the region where $\sigma_1 > 0$). In another cases the coefficient $F_{env} = 1$.

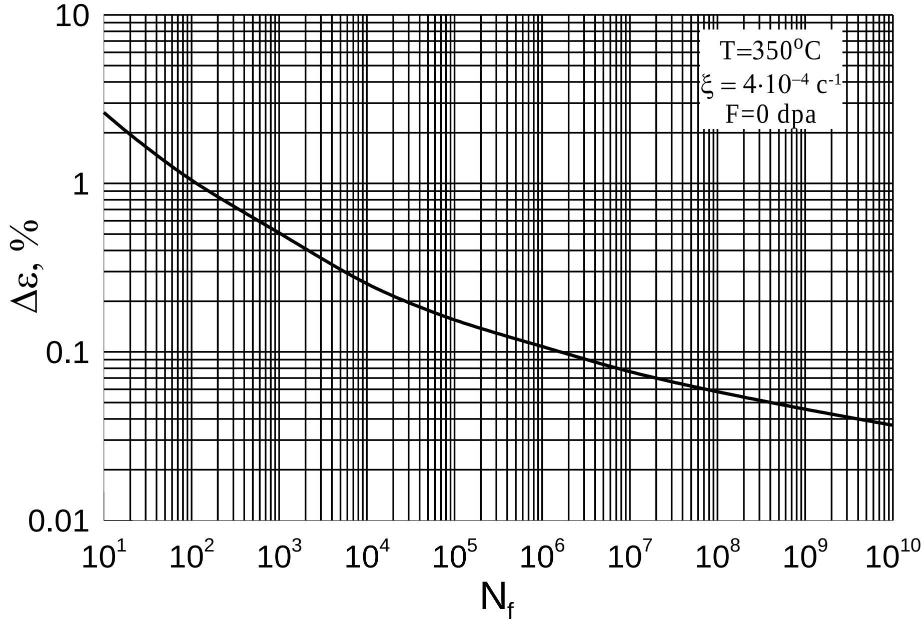
Note: The calculation of fatigue curves is made provided T_{irr}=T. To decrease the conservatism of estimations it is assumed to determine ε_f by means of slow strain rate tests of smooth cylindrical specimens instead of the calculation by Eqs. (B.11) - (B.14). The tests are recommended to carry out in the range of strain rates $10^{-8} \le \dot{\varepsilon} \le 2 \cdot 10^{-6}$ s^{-1}.

B-2. The standard fatigue curve $\Delta\varepsilon(n_f)$ for a weld metal at an operational temperature T ≤ 450°C is determined by the equation:

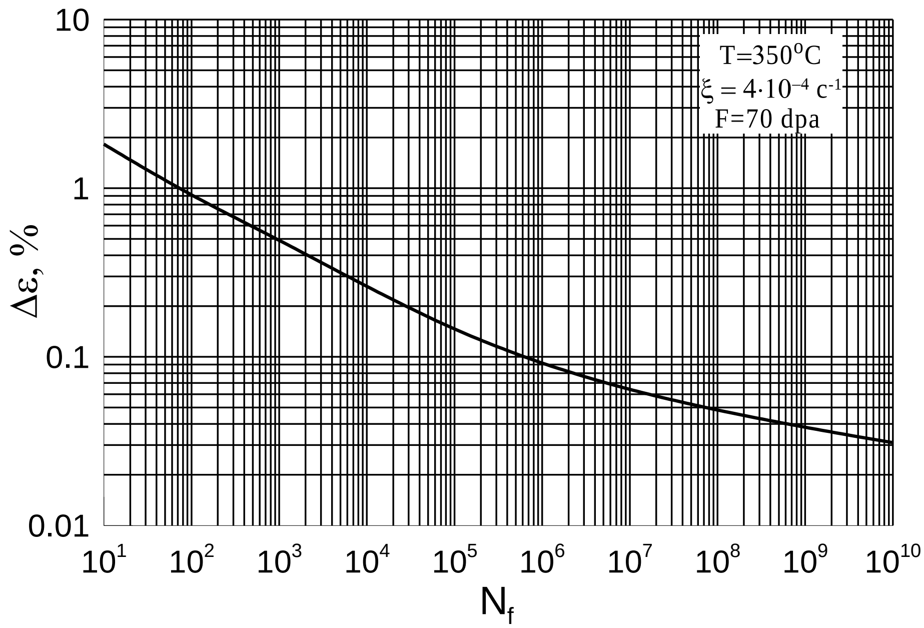
$$\Delta\varepsilon^w(N_f) = \varphi_s \cdot \Delta\varepsilon(N_f), \qquad \text{(B.16)}$$

where φ_s is a coefficient of reduction of fatigue curve, φ_s=0.7.

B-3. Examples of construction of design fatigue curves: The examples of construction of design fatigue curves for a base metal are shown in Figures B.1 and B.2. The fatigue curves are constructed by Eqs. (B.1), (B.2), (B.5) and (B.6) provided T=T_{irr} for the case of the maximum stress ratio and safety factors n_ε=2 and n_n=10 based on the indicated above dependences for E(T), σ_Y(F, T, T_{irr}), σ_{ul}(F, T, T_{irr}) and ε_f(F, T_{irr}). Figure B.1 shows the fatigue curves at T=T_{irr}=350°C obtained by Eqs. (B.2), (B.5) and (B.6). Figure B.2 shows the fatigue curves at T=T_{irr}=450°C obtained by Eqs. (B.2), (B.5) and (B.6).

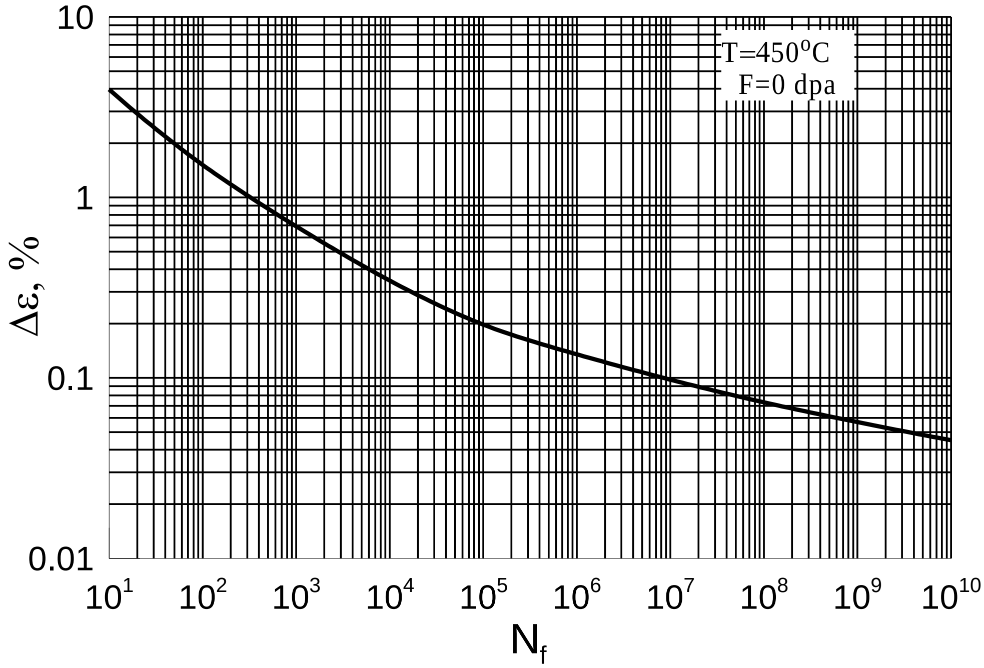


a)

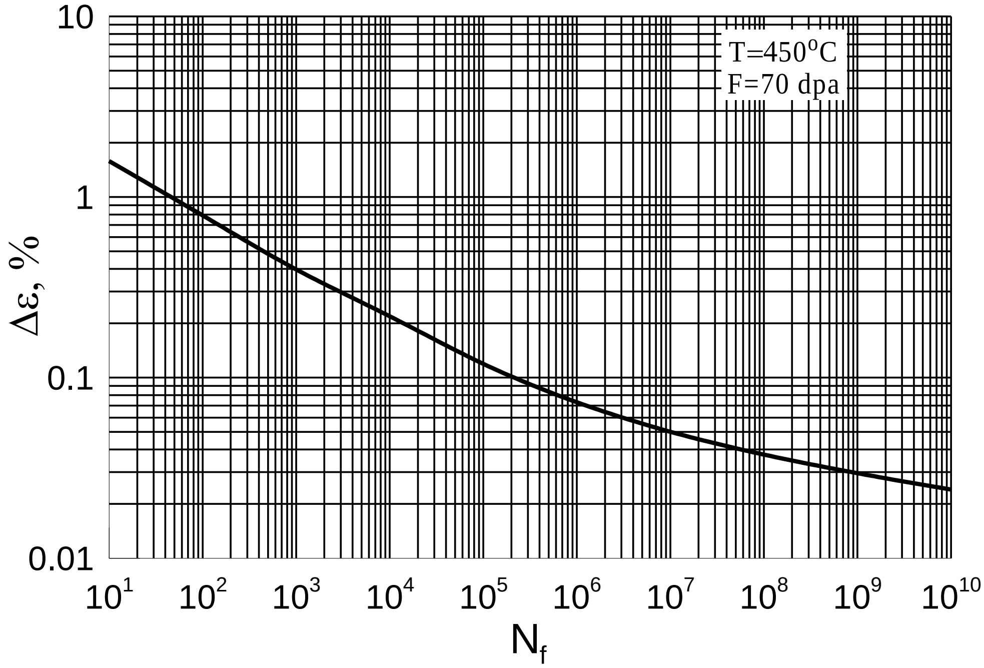


b)

FIG. B.1. Examples of construction of design fatigue curves (maximum stress ratio) for a base metal for damage doses F at a temperature T=T_{irr}=350°C, strain rate ξ^{ten}_{min} =4·10^{-4} s^{-1} for n_ε = 2 and n_N = 10;(a) F=0 dpa and (b) F=70 dpa.



a)



b)

FIG. B.2. Examples of construction of design fatigue curves (maximum stress ratio) for a base metal for damage doses F at a temperature $T=T_{irr}=450°C$, strain rate $\xi_{min}^{ten}=4\cdot10^{-4}\ s^{-1}$ for $n_\varepsilon = 2$ and $n_N = 10$;(a) F=0 dpa and (b) F=70 dpa.

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ANNEX C TO APPENDIX C

DOSE-TIME DEPENDENCE OF IASCC INITIATION

C-1. In Figures C.1 and C.2 dependences σ_{th}^{IASCC}(F), and $\Lambda(t_f)$, are schematically presented. In these dependencies σ_{th}^{IASCC} is the locus of threshold stresses below which IASCC is absent at any level F, $\sigma_{eq}(t_f)$ is equivalent stress; t_f is time to failure at constant stress, $\Lambda = \frac{\sigma_{eq}}{\sigma_{th}^{IASCC}}$.

C-2. When calculating of damage by the criterion „IASCC crack nucleation" the value σ_{th}^{IASCC}(F) may be calculated by the equation:

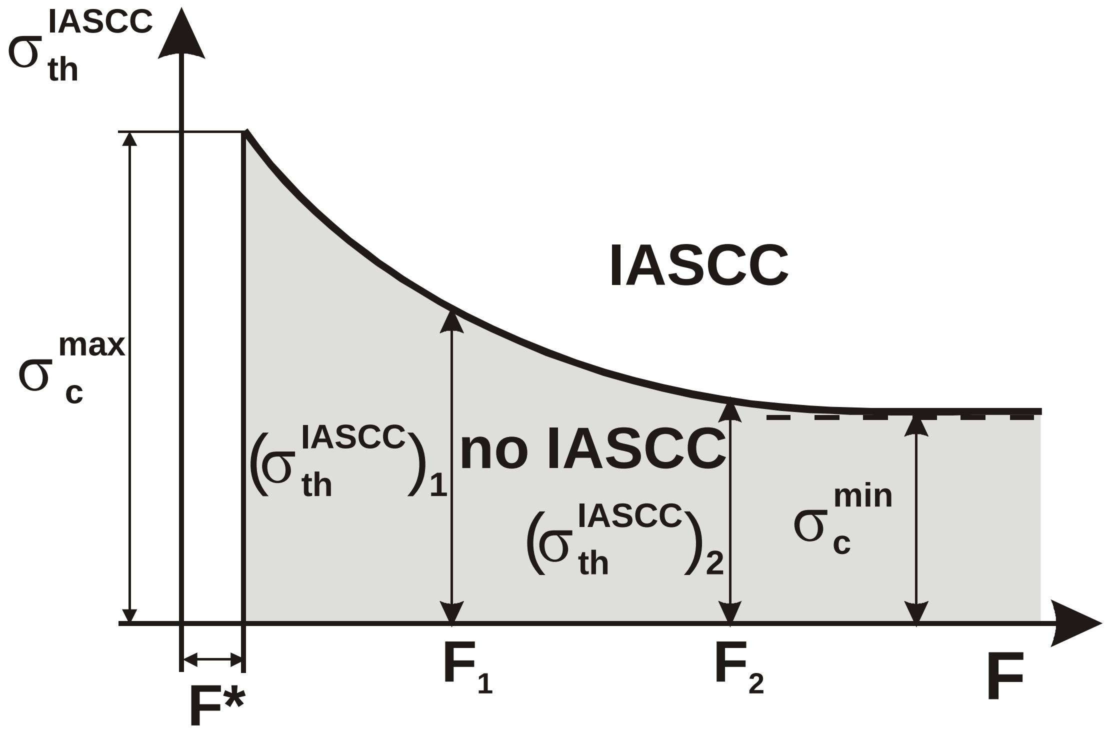
$$\sigma_{th}^{IASCC} = (\sigma_c^{max} - \sigma_c^{min})\cdot\exp[-b(F-F^*)] + \sigma_c^{min},\ F > F^*, \tag{C.1}$$

where

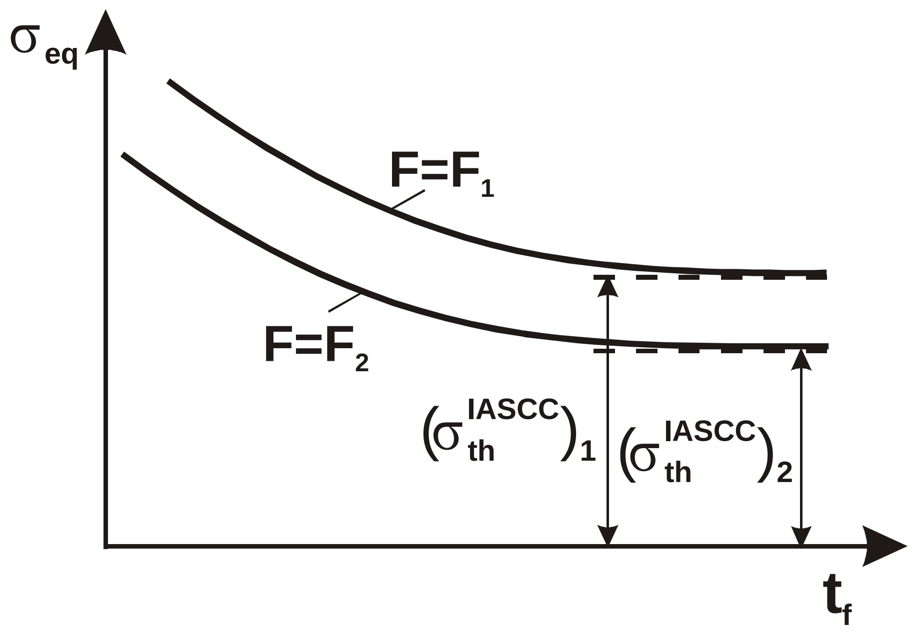
σ_c^{max} is the maximum stress, at which IASCC takes place at $F > F^*$;
σ_c^{min} is the minimal stress below which IASCC does not occur at any level F;
F* is a threshold dose below which IASCC does not occur in deoxygenating water environment of WWER primary coolant circuit.

In Equation C1 following values of parameters are accepted:

F* = 3 dpa, σ_c^{max} =660 MPa, σ_c^{min} = 217 MPa, b=$2.76\cdot10^{-2}$.



a)



b)

FIG. C.1. Dependences σ_{th}^{IASCC} *(F) (a) and $\sigma_{eq}(t_f)$ (b) for two levels of damage dose F_1 and F_2 under IASCC.*

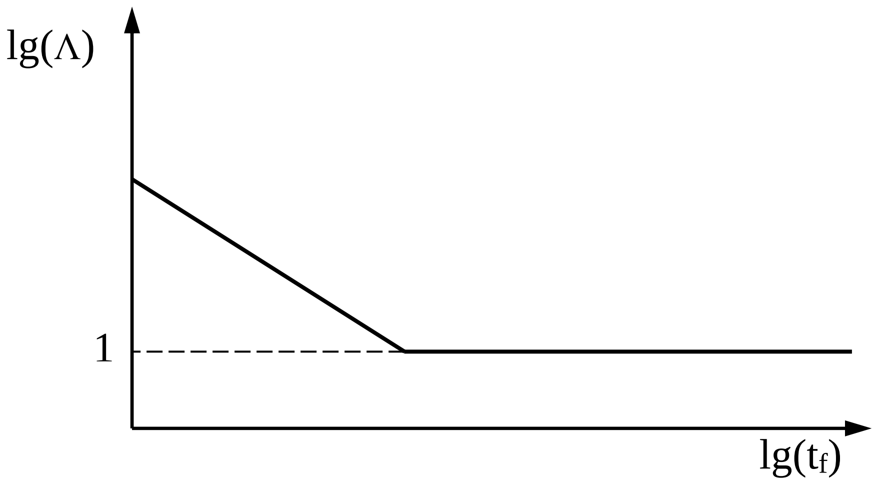


FIG. C.2. Plot $\Lambda = \frac{\sigma_{eq}}{\sigma_{th}^{IASCC}}$ *against* t_f *(plot* $\Lambda = \frac{\sigma_{eq}}{\sigma_{th}^{IASCC}}$ *as a function of* t_f*).*

The dependence $\Lambda(t_f)$ is given by the equation:

$$\Lambda = \begin{cases} \alpha(t_f)^{-\beta}, \text{ if } \alpha(t_f)^{-\beta} > 1 \\ 1, \text{ if } \alpha(t_f)^{-\beta} \le 1 \end{cases} \tag{C.2}$$

where α and β are parameters independent of level F. For austenitic steels α=1.55, β=-0.063.

Note: For decrease of conservatism dependencies σ_{th}^{IASCC}(F) and $\Lambda(t_f)$ may be determined experimentally for a concrete material of which RVI component is made. Test should be carried out at constant load in water environment simulated WWER coolant. Test duration should be not less than 2000 hours or before failure of a specimen. As design dependence it is necessary to use dependence σ_{th}^{IASCC}(F), above which experimental points are located with 95 % probability. Design dependence $\Lambda(t_f)$ is determined by means of experimental data processing by a method of least squares.

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ANNEX D TO APPENDIX C

FRACTURE TOUGHNESS OF RVI MATERIALS

This Annex may be applied to calculate fracture toughness in the terms J_c (N/mm) for given values of damage dose of neutron irradiation F over the range from 0 to 70 dpa, at irradiation temperature T_{irr} , and current operational temperature T over the range 250°C to 450°C.

D-1. For given values of F, T and T_{irr} , fracture toughness for materials under irradiation is calculated by following equation:

$$J_c(F,T,T_{irr}) = 2.5 \cdot 10^{-4} \cdot \sigma_Y(F,T,T_{irr}) \cdot \left[1 - A_{J(e)}\sqrt{1-\exp(-0.2\cdot F)}\right], \text{ N/m,} \tag{D.1}$$

where σ_Y – yield strength of a irradiated material calculated by equation (A.2) given in Annex A, $A_{J(e)}$ = 0.93.

Note: For a decrease in conservatism $J_c(F,T,T_{irr})$ may be determined experimentally using material from which the RVI component is made. As design dependence it is necessary to use dependence $J_c(F,T,T_{irr})$ above which experimental points are located with 95 % probability.

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ANNEX E TO APPENDIX C

FATIGUE CRACK GROWTH RATE FOR RVI MATERIALS

E-1. The FCGR in the air for RVI materials at the operational temperature $T \le 450°C$ is determined by the equation:

$$\left(\frac{dl}{dN}\right)_{air} = \begin{cases} C_f \cdot \left[\frac{\Delta K}{(1-R)^{0.25}}\right]^{n_f}, \text{at } \Delta K > \Delta K_{th}(R) \\ 0, \quad \text{at } \Delta K \le K_{th}(R) \end{cases}, \text{ m/cycle,} \tag{E.1}$$

where

$\Delta K = K_{max} - K_{min}$	is a SIF range, $MPa\sqrt{m}$ (the difference between a maximum and minimum SIF value);
R	is a stress ratio, R= K_{min}/K_{max} ;
ΔK_{th}	is a SIF threshold range;
n_f	is a Paris equation coefficient, it is taken to be equal to 3.3.

E-2. When determining ΔK it is taken that

If $K_{min} < 0$, then $K_{min} = 0$;

If $K_{max} < 0$ and $K_{min} < 0$, then $\Delta K = 0$.

E-3. When determining R it is taken that

If R > 0.95, then R=0.95;

If R < 0, then R = 0.

E-4. The coefficient C_f is taken to be equal:

— for $T \le 350°C$:

$$C_f = \begin{cases} 5.2 \cdot 10^{-12}, \text{for base metal at } F \le 70 \text{ dpa} \\ 5.2 \cdot 10^{-12}, \text{for weld metal at } F \le 4 \text{ dpa} \\ 1.56 \cdot 10^{-11}, \text{for weld metal at } F > 4 \text{ dpa} \end{cases}, \tag{E.2}$$

— for $350°C < T \le 450°C$:

$$C_f = \begin{cases} 5.2 \cdot 10^{-12} \cdot \exp(0.342 \cdot (T-350)^{0.356}), \text{for base metal at } F \le 70 \text{ dpa} \\ 5.2 \cdot 10^{-12} \cdot \exp(0.342 \cdot (T-350)^{0.356}), \text{for weld metal at } F \le 4 \text{ dpa} \\ 1.56 \cdot 10^{-11} \cdot \exp(0.342 \cdot (T-350)^{0.356}), \text{for weld metal at } F > 4 \text{ dpa} \end{cases}, \tag{E.3}$$

where F is a damage dose of neutron irradiation.

E-5. The dependence $\Delta K_{th}(R)$ is calculated by the equation:

$$\Delta K_{th} = \Delta K_{th}^{0} \cdot (1 - 0.7 \cdot R),\ \text{MPa}\sqrt{\text{m}}, \tag{E.4}$$

where $\Delta K_{th}^{0} = \Delta K_{th}(R = 0)$; $\Delta K_{th}^{0} = 6.5\ \text{MPa}\sqrt{\text{m}}$.

E-6. The FCGR in water environment of the WWER primary coolant circuit for RVI materials at the operational temperature $T \leq 350^{o}C$ are determined by the equation:

$$\left(\frac{dl}{dN}\right)_{env} = \omega^{env} \cdot \left(\frac{dl}{dN}\right)_{air}, \tag{E.5}$$

where

$$\omega^{env} = 1 + A \cdot \left[\frac{1}{t_r} \cdot \left(\frac{dl}{dN}\right)_{air}\right]^{-m}; \tag{E.6}$$

where

t_r is the rise time of the loading cycle (when crack tip is opening); if $t_r > 600$ s, then $t_r = 600$ s;

$\left(\frac{dl}{dN}\right)_{air}$ is the FCGR in the air calculated by Equation (E.1);

$A = 2.74 \cdot 10^{-5}$

$m = 0.5$

If value of ω^{env} calculated by Equation (E.6) exceeds 7.7, then it is taken that $\omega^{env} = 7.7$.

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ANNEX F TO APPENDIX C

TIME-DEPENDENT CORROSIVE CRACK GROWTH RATE FOR RVI MATERIALS IN WWER ENVIRONMENT

F-1. The crack growth rate in water environment of WWER primary coolant circuit for RVI materials under static loading at coolant temperatures T ≤ 350 °C is determined by the formula:

$$\frac{dl}{d\tau} = A_\tau \cdot (K)^{n_\tau}, \text{ m/s}, \tag{F.1}$$

where K is stress intensity factor, $\text{MPa}\sqrt{\text{m}}$.

F-2. The coefficients A_τ и n_τ in Equation (F.1) are equal

— for a base metal to:

$$\frac{dl}{d\tau} = \begin{cases} 7.04 \cdot 10^{-17} (K)^{3.0}, & \text{at } F < 3 \text{ dpa} \\ 7.0 \cdot 10^{-14} (K)^{2.16}, & \text{at } F \geq 3 \text{ dpa} \end{cases}; \tag{F.2}$$

— for a metal of weld joints to $\frac{dl}{d\tau} = 7.0 \cdot 10^{-14} \cdot (K)^{2.16}$.

F-3. In Equation (F.1) for the values K > 55 $\text{MPa}\sqrt{\text{m}}$ it is taken that K = 55 $\text{MPa}\sqrt{\text{m}}$.

Note: To decrease conservatism the dependence dl/dt(K) may be determined experimentally for a considered material of which RVI component is made. As design dependence it is necessary to use dependence $\frac{dl}{d\tau}$=f(K) which is an upper envelope of all experimental points at a 95 % confidence level.

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ANNEX G TO APPENDIX C

SWELLING OF RVI MATERIALS

G-1. Swelling at the given level of effective stress σ_{eff} and plastic strain is calculated by the equation:

$$S = S_0 \cdot f_1(\sigma_{eff}) \cdot f_2(æ_p) \qquad \text{(G.1)}$$

where

$$S_0 = c \cdot F^{n_v} \cdot \exp[-r \cdot (T_{irr} - T_m)^2]; \qquad \text{(G.2)}$$

$$\sigma_{eff} = (1-\eta_1) \cdot \sigma_m + \eta_1 \cdot \sigma_{eq}; \qquad \text{(G.3)}$$

$$f_1(\sigma_{eff}) = 1 + P \cdot \sigma_{eff}; \qquad \text{(G.4)}$$

$$f_2(æ_p) = \exp(-\eta_2 \cdot æ_p). \qquad \text{(G.5)}$$

With a changing stress level and plastic deformation the swelling rate is calculated by the equations:

$$\dot{S} = \begin{cases} \dot{S}_0 \cdot f_1(\sigma_{eff}) \cdot f_2(æ_p), \\ 0, \text{ if } S \le 0 \text{ and } f_1(\sigma_{eff}) < 0 \end{cases}, \qquad \text{(G.6)}$$

where

$$\dot{S}_0 = c \cdot n_v \cdot F^{(n_v - 1)} \cdot \Phi \cdot \exp[(-r \cdot (T_{irr} - T_m)^2]. \qquad \text{(G.7)}$$

In the Eqs. (G.1) - (G.7): $æ_p = \int d\varepsilon_{eq}^p$; n_v=1.88; r = $1.825 \cdot 10^{-4}$ $^oC^{-2}$; T_m = 470 °C; η_1 = 0.15; P = $5.4 \cdot 10^{-3}$ MPa^{-1}; c=$2.588 \cdot 10^{-4}$, η_2=8.75; Φ is a damage dose rate of neutron irradiation, dpa/s; F is a damage dose of neutron irradiation, dpa; T_{irr} is a irradiation temperature, °C.

Note: The calculation of SSF of a structural component with regard for swelling is carried out, if the value S_0 calculated by equation G2 with T=T_m exceeds $1.0 \cdot 10^{-5}$.

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ANNEX H TO APPENDIX C

RADIATION CREEP OF RVI MATERIALS

H-1. The equation of radiative creep at different temperatures and damage dose rates of neutron irradiation is calculated by the equation:

$$\xi^{c}_{eq}\left(\Phi, \sigma_{eq}\right) = (B\Phi + D\dot{S}) \cdot \sigma_{eq}, \tag{H.1}$$

where $\dot{S}$ is swelling rate at the given level of stress and plastic strain calculated according to Equation (G.6); Φ is a damage dose rate of neutron irradiation, dpa/s; $B = 1 \cdot 10^{-6}$ (MPa·dpa)$^{-1}$; $D = 2.7 \cdot 10^{-3}$ MPa^{-1}.

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[2] MARGOLIN, B.Z., MURASHOVA, A.I., NEUSTROEV, V.S., Effect of stress on radiation swelling of austenitic steels, “Voprosy Materialovedeniya” (Problems of materials science), №4(68), (2011), 124–139 (in Russian).

ANNEX I TO APPENDIX C

A PROCEDURE OF CYCLE FORMATION UNDER NON-RADIAL LOADING

I-1. A PROCEDURE OF DETERMINATION OF A CYCLIC LOADING PROFILE AND STRAIN RANGES UNDER NON-RADIAL LOADING.

I-1.1. The kinetics of SSF in the investigated zone of a structural component is determined under the investigated loading conditions or a group of loading conditions.

I-1.2. Equivalent deformation $\varepsilon_{eq}^{(1)}(\tau)$ is calculated at the first loading step at the current instant of time τ:

$$\begin{aligned}\varepsilon_{eq}^{(1)}(\tau) = \frac{\sqrt{2}}{2(1+\nu)} \cdot \Big[& \left(\varepsilon_x(\tau) - \varepsilon_y(\tau)\right)^2 + \left(\varepsilon_y(\tau) - \varepsilon_z(\tau)\right)^2 + \left(\varepsilon_z(\tau) - \varepsilon_x(\tau)\right)^2 + \\ & + \frac{3}{2}\left(\left(\gamma_{xy}(\tau)\right)^2 + \left(\gamma_{yz}(\tau)\right)^2 + \left(\gamma_{zx}(\tau)\right)^2\right)\Big]^{1/2}\end{aligned} \tag{I.1}$$

where $\varepsilon_x(\tau)$, $\varepsilon_y(\tau)$, $\varepsilon_z(\tau)$, $\gamma_{xy}(\tau)$, $\gamma_{yz}(\tau)$,$\gamma_{zx}(\tau)$ are strain components in a XYZ coordinate system calculated with regard for elastic, plastic strains and creep strain; ν is Poisson ratio, $\nu = 0.3$.

The strain vector $\begin{Bmatrix} \varepsilon_x \\ \varepsilon_y \\ \varepsilon_z \\ \gamma_{xy} \\ \gamma_{yz} \\ \gamma_{zx} \end{Bmatrix}^{(1)}$ that corresponds to the maximum value $\varepsilon_{eq}^{(1)}(\tau)$ at $\tau=t^{(1)}$ is determined and the equivalent strain range at the first step is calculated:

$$\Delta\varepsilon^{(1)} = \max_{\tau>0} \varepsilon_{eq}^{(1)}(\tau) = \varepsilon_{eq}^{(1)}(\tau)\Big|_{\tau=t^{(1)}} \tag{I.2}$$

where $t^{(1)}$ is the time at which the maximum value $\varepsilon_{eq}^{(1)}(\tau)$ is reached. The value $\Delta\varepsilon^{(1)}$ is assigned the plus sign.

I-1.3. The equivalent strain range at the second step is calculated:

$$\Delta\varepsilon_{eq}^{(2)}(\tau)=\frac{\sqrt{2}}{2(1+\nu)}\cdot\Big[\Big(\Delta\varepsilon_x^{(2)}(\tau)-\Delta\varepsilon_y^{(2)}(\tau)\Big)^2+\Big(\Delta\varepsilon_y^{(2)}(\tau)-\Delta\varepsilon_z^{(2)}(\tau)\Big)^2+\Big(\Delta\varepsilon_z^{(2)}(\tau)-\Delta\varepsilon_x^{(2)}(\tau)\Big)^2+$$
$$+\frac{3}{2}\Big(\Big(\Delta\gamma_{xy}^{(2)}(\tau)\Big)^2+\Big(\Delta\gamma_{yz}^{(2)}(\tau)\Big)^2+\Big(\Delta\gamma_{zx}^{(2)}(\tau)\Big)^2\Big)\Big]^{1/2} \qquad \text{(I.3)}$$

where $\begin{Bmatrix}\Delta\varepsilon_x(\tau)\\ \Delta\varepsilon_y(\tau)\\ \Delta\varepsilon_z(\tau)\\ \Delta\gamma_{xy}(\tau)\\ \Delta\gamma_{yz}(\tau)\\ \Delta\gamma_{zx}(\tau)\end{Bmatrix}^{(2)}=\begin{Bmatrix}\varepsilon_x(\tau)\\ \varepsilon_y(\tau)\\ \varepsilon_z(\tau)\\ \gamma_{xy}(\tau)\\ \gamma_{yz}(\tau)\\ \gamma_{zx}(\tau)\end{Bmatrix}-\begin{Bmatrix}\varepsilon_x\\ \varepsilon_y\\ \varepsilon_z\\ \gamma_{xy}\\ \gamma_{yz}\\ \gamma_{zx}\end{Bmatrix}^{(1)}$; $\tau > t^{(1)}$.

The strain increment vector $\begin{Bmatrix}\Delta\varepsilon_x\\ \Delta\varepsilon_y\\ \Delta\varepsilon_z\\ \Delta\gamma_{xy}\\ \Delta\gamma_{yz}\\ \Delta\gamma_{zx}\end{Bmatrix}^{(2)}$ is determined as well as the corresponding to it equivalent strain range at the second step

$$\Delta\varepsilon^{(2)}=\max_{\tau>t^{(1)}}\Delta\varepsilon_{eq}^{(2)}(\tau)=\Delta\varepsilon_{eq}^{(2)}(\tau)\Big|_{\tau=t^{(2)}} \qquad \text{(I.4)}$$

The value $\Delta\varepsilon^{(2)}$ is assigned the minus sign and the strain vector corresponding to the end of the second step is determined by:

$$\begin{Bmatrix}\varepsilon_x\\ \varepsilon_y\\ \varepsilon_z\\ \gamma_{xy}\\ \gamma_{yz}\\ \gamma_{zx}\end{Bmatrix}^{(2)}=\begin{Bmatrix}\varepsilon_x\\ \varepsilon_y\\ \varepsilon_z\\ \gamma_{xy}\\ \gamma_{yz}\\ \gamma_{zx}\end{Bmatrix}^{(1)}+\begin{Bmatrix}\Delta\varepsilon_x\\ \Delta\varepsilon_y\\ \Delta\varepsilon_z\\ \Delta\gamma_{xy}\\ \Delta\gamma_{yz}\\ \Delta\gamma_{zx}\end{Bmatrix}^{(2)} \qquad \text{(I.5)}$$

I-1.4. At the n-th step of loading the procedure of determination of $\Delta\varepsilon_{eq}^{(n)}$ is similar to the procedure I-1.3.

The equivalent strain range is calculated by the equation:

$$\Delta\varepsilon_{eq}^{(n)}(\tau) = \frac{\sqrt{2}}{2(1+\nu)} \cdot \Big[\left(\Delta\varepsilon_x^{(n)}(\tau) - \Delta\varepsilon_y^{(n)}(\tau)\right)^2 + \left(\Delta\varepsilon_y^{(n)}(\tau) - \Delta\varepsilon_z^{(n)}(\tau)\right)^2 + \left(\Delta\varepsilon_z^{(n)}(\tau) - \Delta\varepsilon_x^{(n)}(\tau)\right)^2 + \\ + \frac{3}{2}\left(\left(\Delta\gamma_{xy}^{(n)}(\tau)\right)^2 + \left(\Delta\gamma_{yz}^{(n)}(\tau)\right)^2 + \left(\Delta\gamma_{zx}^{(n)}(\tau)\right)^2\right)\Big]^{1/2} \tag{I.6}$$

where $\begin{Bmatrix} \Delta\varepsilon_x(\tau) \\ \Delta\varepsilon_y(\tau) \\ \Delta\varepsilon_z(\tau) \\ \Delta\gamma_{xy}(\tau) \\ \Delta\gamma_{yz}(\tau) \\ \Delta\gamma_{zx}(\tau) \end{Bmatrix}^{(n)} = \begin{Bmatrix} \varepsilon_x(\tau) \\ \varepsilon_y(\tau) \\ \varepsilon_z(\tau) \\ \gamma_{xy}(\tau) \\ \gamma_{yz}(\tau) \\ \gamma_{zx}(\tau) \end{Bmatrix} - \begin{Bmatrix} \varepsilon_x \\ \varepsilon_y \\ \varepsilon_z \\ \gamma_{xy} \\ \gamma_{yz} \\ \gamma_{zx} \end{Bmatrix}^{(n-1)}$; $\tau > t^{(n-1)}$;

$\begin{Bmatrix} \varepsilon_x \\ \varepsilon_y \\ \varepsilon_z \\ \gamma_{xy} \\ \gamma_{yz} \\ \gamma_{zx} \end{Bmatrix}^{(n-1)}$ is the strain vector corresponding to the end of a step (n-1).

The strain increment vector $\begin{Bmatrix} \Delta\varepsilon_x \\ \Delta\varepsilon_y \\ \Delta\varepsilon_z \\ \Delta\gamma_{xy} \\ \Delta\gamma_{yz} \\ \Delta\gamma_{zx} \end{Bmatrix}^{(n)}$ is determined as well as the corresponding to it equivalent strain range at the n-th step:

$$\Delta\varepsilon^{(n)} = (-1)^{n-1} \cdot \max_{\tau > t^{(n-1)}} \Delta\varepsilon_{eq}^{(n)}(\tau) = (-1)^{n-1} \cdot \Delta\varepsilon_{eq}^{(n)}(\tau)\Big|_{\tau = t^{(n)}} \tag{I.7}$$

The strain vector corresponding to the end of the n-th step is determined by:

$$\begin{Bmatrix} \varepsilon_x \\ \varepsilon_y \\ \varepsilon_z \\ \gamma_{xy} \\ \gamma_{yz} \\ \gamma_{zx} \end{Bmatrix}^{(n)} = \begin{Bmatrix} \varepsilon_x \\ \varepsilon_y \\ \varepsilon_z \\ \gamma_{xy} \\ \gamma_{yz} \\ \gamma_{zx} \end{Bmatrix}^{(n-1)} + \begin{Bmatrix} \Delta\varepsilon_x \\ \Delta\varepsilon_y \\ \Delta\varepsilon_z \\ \Delta\gamma_{xy} \\ \Delta\gamma_{yz} \\ \Delta\gamma_{zx} \end{Bmatrix}^{(n)} \quad \text{(I.8)}$$

Note: When determining kinetics of SSF of an investigated structural component an XYZ coordinate system should be fixed.

I-2. LOADING CYCLE FORMATION.

I-2.1. Loading cycle under non-radial loading is formed by the „rain flow method". Figure I.1 shows the diagram of the use of the „rain flow method".

I-2.2. The dependence $\varepsilon^{(n)}$ on the loading step number is constructed:

$$\varepsilon^{(n)} = \varepsilon^{(n-1)} + \Delta\varepsilon^{(n)} \quad \text{(I.9)}$$

In doing this it is assumed that at the initial moment of the investigated loading period and at the end moment $\varepsilon \ = 0$, i.e.:

$$\varepsilon^{(0)} = \varepsilon^{(N)} = 0 \quad \text{(I.10)}$$

where N is the last loading step of the investigated loading period.

I-2.3. The step number at which $\varepsilon^{(n)}$=max, as well as the step number at which $\varepsilon^{(n)}$ = min are found (in Figure I.1 these are steps 7 and 4, correspondingly). A „rain flow" line (the „steepest descent" line) is constructed from the maximum to minimum value (from step 7 to step 4 in Figure I.1). Then the „steepest ascent" line is constructed from the minimum value $\varepsilon^{(n)}$ to the maximum value (from step 4 to step 1 in Figure I.1), except for the value from which the „steepest descent" line has already been constructed. Subsequent lines of the „steepest descent and ascent" are constructed in a similar way each time connecting the step numbers where $\varepsilon^{(n)}$ = max and $\varepsilon^{(n)}$ = min, except for the step numbers where the lines of the „steepest descent" and „steepest ascent" have already been constructed.

I-2.4. The loading cycles with their ranges are formed after drawing the „steepest descent" lines and „steepest ascent" lines. The procedure of the loading cycles formation is shown in Figure I.1.

The main principles of cycle formation are the following.

a) Each subsequent formed cycle corresponds to the maximum strain range of the remained cycles;

b) For each loading half cycle an unloading half cycle should be found. In other words, each cycle should be closed. The range on unloading is equal to the strain range on loading.

Based on the outlined principle it follows from Figure I.1 that:

— Cycle 1. b-f-l-m strain range $\Delta\varepsilon_{\text{range}}^{(1)} = \varepsilon^{(7)} - \varepsilon^{(4)}$;

— Cycle 2. 0-a-b strain range $\Delta\varepsilon_{\text{range}}^{(2)} = \varepsilon^{(1)} - 0$;

— Cycle 3. c-d-e strain range $\Delta\varepsilon_{\text{range}}^{(3)} = \left|\varepsilon^{(2)} - \varepsilon^{(3)}\right|$;

— Cycle 4. g-j-k strain range $\Delta\varepsilon_{\text{range}}^{(4)} = \varepsilon^{(5)} - \varepsilon^{(6)}$.

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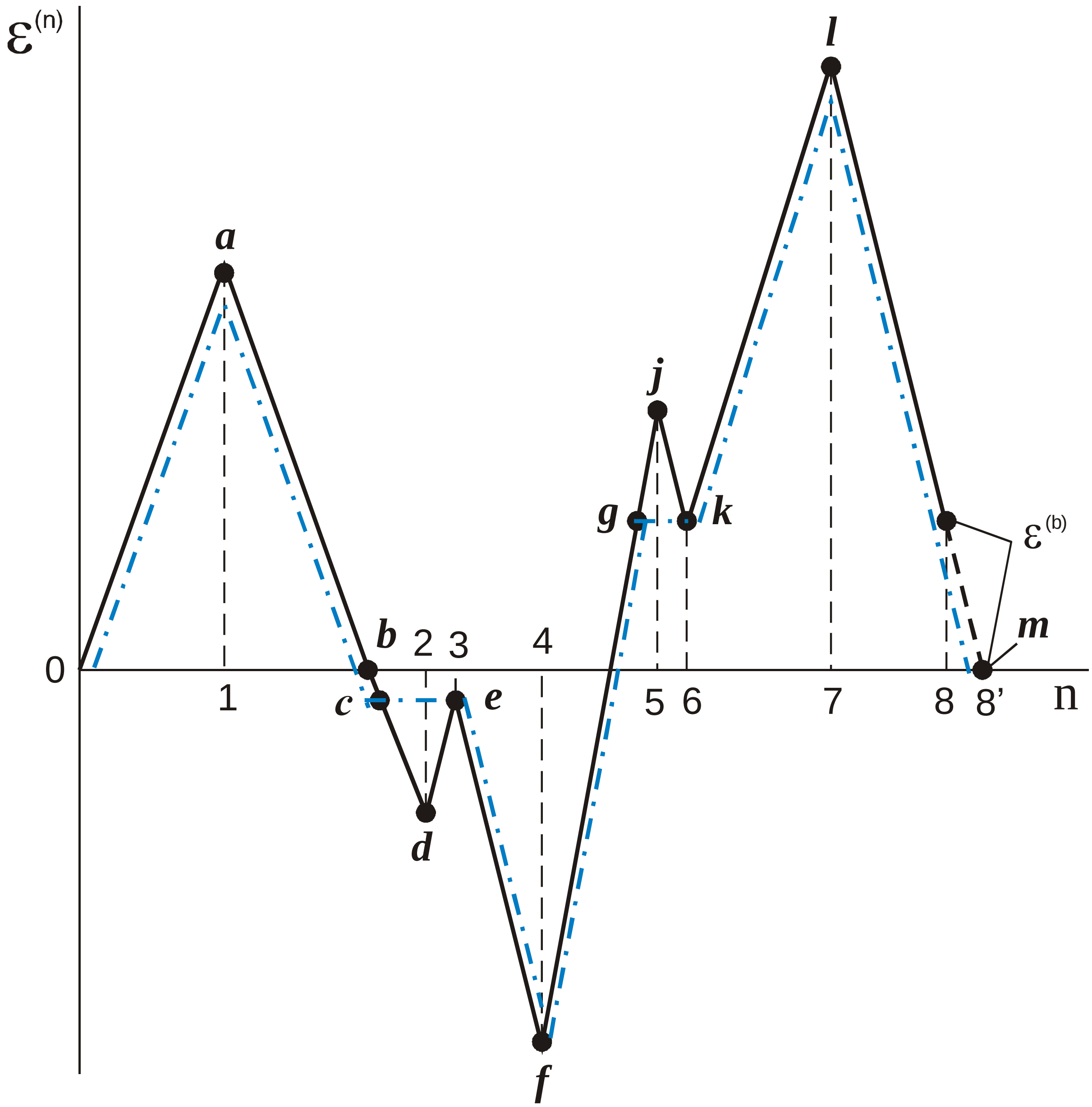

FIG. I.1. Dependence of $\varepsilon^{(n)}$ on a loading step number and the application of the „rain flow method" in order to loading cycles formation: N=8, the calculated value $\varepsilon^{(8)}$ corresponds to n=8, the assumed value $\varepsilon^{(8)}=0$ corresponds to n=8':

- - - - the lines of the steepest descent and ascent.

ANNEX J TO APPENDIX C

PROCEDURE OF REFERENCE STRESS CALCULATION

The geometrical characteristics of the investigated structural component and the relationship between the mean radius (r_m) of a cylindrical surface of a component wall and its thickness (h) should be taken into account when the scheme of reference stress σ_{ref} calculation is chosen. When $r_m/h \geq 20$ the reference stress σ_{ref} is calculated based on the design diagram of a semielliptical or a one-quarter elliptic surface crack in a plate. When $r_m/h < 20$ the reference stress σ_{ref} is calculated depending on the assumed crack orientation: according to the design diagram of the inner semielliptical surface crack in a diametral plane of a cylinder or according to the design diagram of the inner semielliptical or a one-quarter elliptic surface crack on the cylinder element.

Only primary stresses (from force loads) are taken into consideration when SSF is calculated. The calculation of SSF is performed by means of solution of an elastic problem for a structural component without a crack.

J-1. SEMIELLIPTICAL SURFACE CRACK IN A PLATE

J-1.1. The distribution of stresses $\sigma_n(x,t)$ normal to a section plane at the instant of time t is determined by the numerical or analytical methods for an estimated cross-section of a structural component (see Figure J.1) where a crack take place or assumed (a crack plane coincide with a section plane).

J-1.2. The calculated distribution of stresses $\sigma_n(x,t)$ is represented at the instant of time t as membrane and bending components as calculated by Eqs. (J.1) and (J.2), (see Figure J.2).

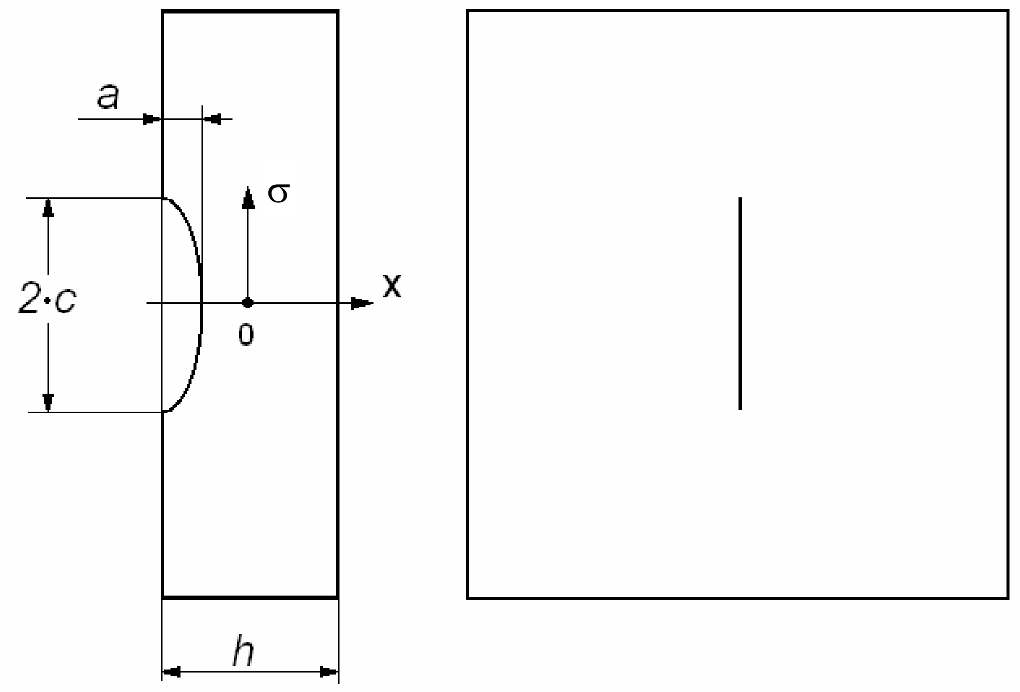

FIG. J.1. Semielliptical surface crack in a plate.

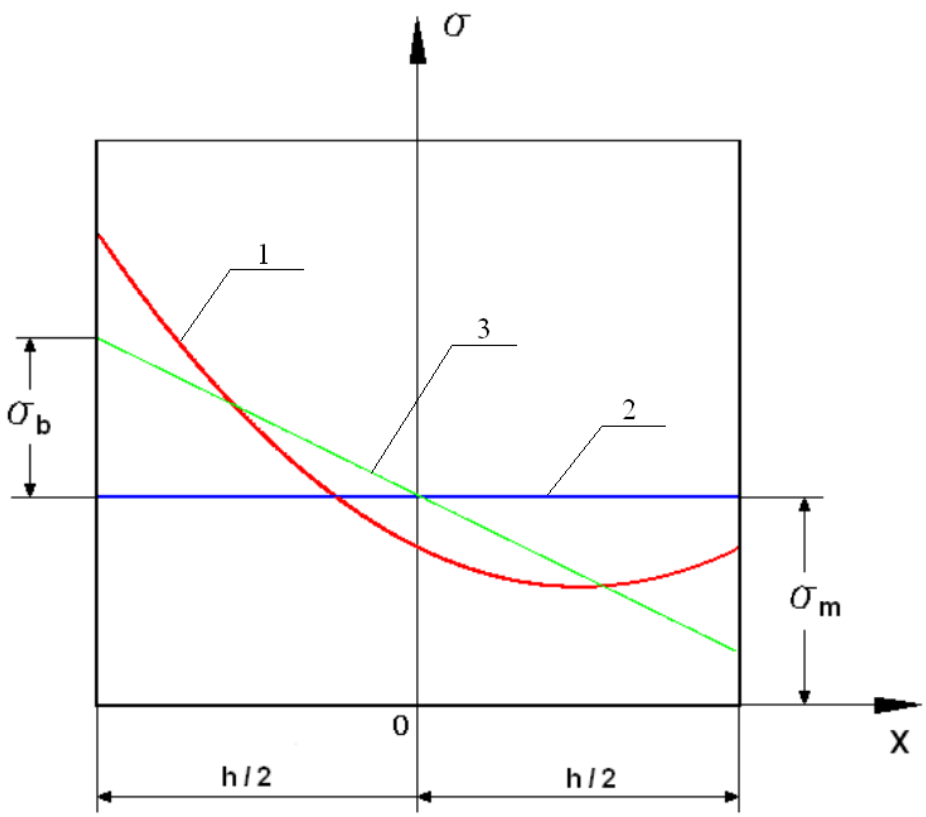

FIG. J.2. Cross-section distribution of the stress σ_n (1) and its membrane (2) and bending (3) components.

J-1.3. Membrane stress is calculated by the equation:

$$\sigma_M = \frac{1}{h}\int_{-h/2}^{h/2} \sigma(x)\cdot dx \tag{J.1}$$

J-1.4. Maximum bending stress is calculated by the equation:

$$\sigma_b = -\frac{6}{h^2}\int_{-h/2}^{h/2} \sigma(x)\cdot x \cdot dx \tag{J.2}$$

J-1.5. General stress is presented in the form $\sigma_n(x) \approx \sigma_m - \sigma_b\cdot(2\cdot x/h)$.

J-1.6. Reference stress is calculated by the equation:

$$\sigma_{ref} = \frac{(1-\xi)^{1.58}\cdot\frac{\sigma_b}{3} + \sqrt{(1-\xi)^{3.16}\cdot\frac{\sigma_b^2}{9} + (1-\xi)^{3.14}\cdot\sigma_M^2}}{(1-\xi)^2} \tag{J.3}$$

where $\xi = \dfrac{a\cdot c}{h\cdot(c+h)}$.

If $\sigma_m < 0$, then it is assumed that $\sigma_m = 0$.

If $\sigma_b < 0$, then it is assumed that $\sigma_b = 0$.

Note: It is allowed to calculate using Equation (J.3) with changes of a / h from 0 to 0.8.

J-2. INNER ONE-QUARTER ELLIPTIC SURFACE CRACK IN A NODE FORMED BY TWO PLATES

J-2.1. When considering a one-quarter elliptic crack it is assumed that the minor and major ellipse semiaxes are equal.

J-2.2. Distribution of stresses $\sigma_n(x,t)$ normal to a section plane at instant of time t is determined by the numerical or analytical methods for an estimated cross-section of a structural component (see Figure J.3) where a crack take place or assumed (a crack plane coincide with a section plane).

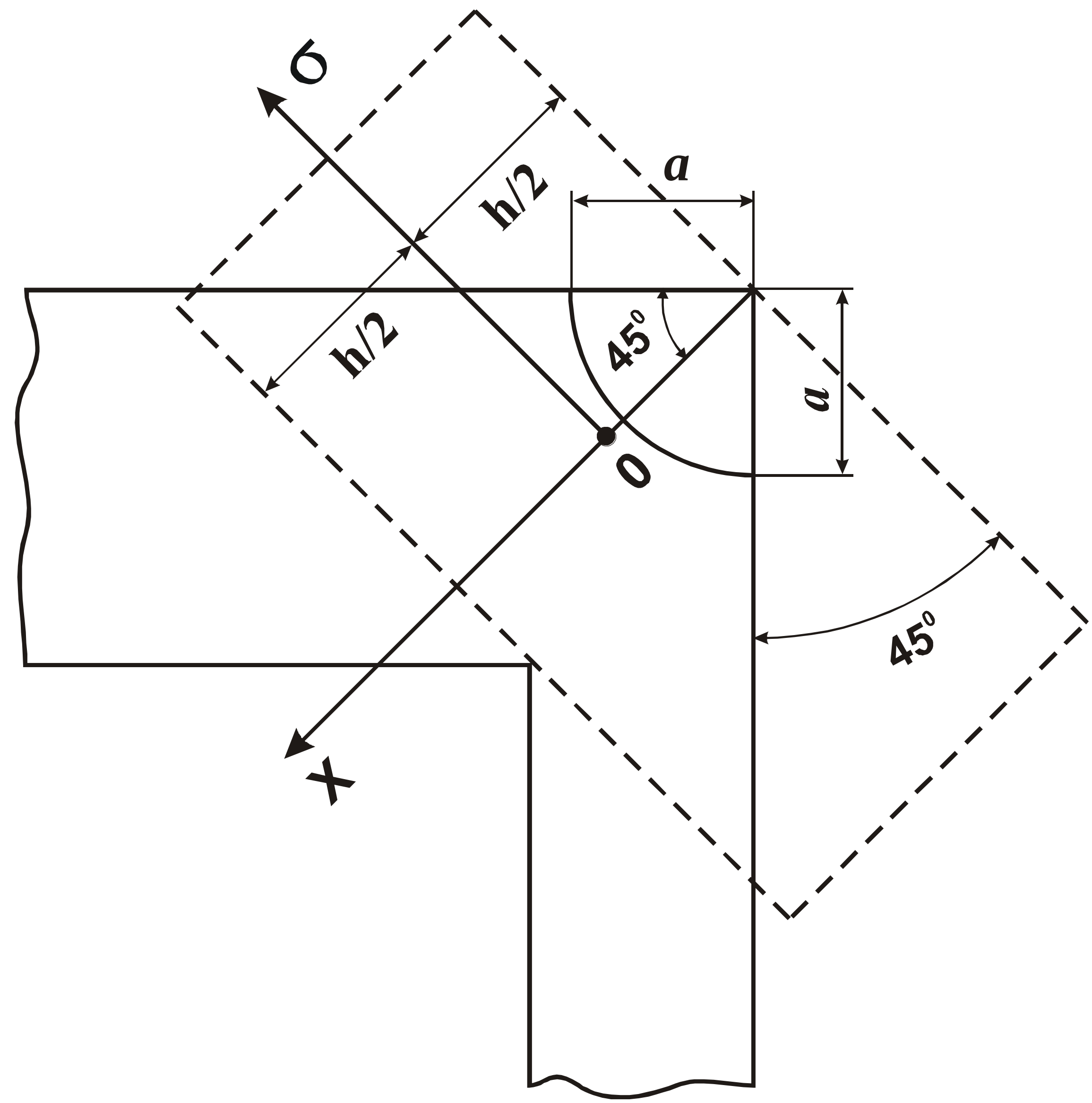

FIG. J.3. One-quarter elliptic surface crack in a node formed by two plates and diagram of a calculated parameter h: the axis X is a coordinate from which distribution of stresses normal to a crack is given.

J-2.3. The calculated distribution of stresses $\sigma_n(x,t)$ is represented at instant of time t as membrane and bending components as calculated by Eqs. (J.1) and (J.2), (see Figure J.2).

J-2.4. Reference stress is calculated by Equation (J.3) provided that h is determined from Figure J.3 and c=a.

J-3. INNER SEMIELLIPTICAL SURFACE CRACK LOCATED IN A DIAMETRAL PLANE OF A CYLINDER

J-3.1. Geometry of the structure, crack and loadings is shown in Figure J.4.

J-3.2. Reference stress is calculated by the equation:

$$\sigma_{ref} = \sqrt{\frac{\left(\frac{\sigma_M}{s'_M}\right)^2 + \left(\frac{\sigma_{bg}}{s'_{bg}}\right)^2}{\left(\frac{s_M}{s'_M}\right)^2 + \left(\frac{s_{bg}}{s'_{bg}}\right)^2}} \tag{J.4}$$

where

σ_m and σ_{bg} are correspondingly membrane and global bending stresses:

$$\sigma_M = \frac{N}{S_{tb}},\ \sigma_{bg} = \frac{M}{W_{tb}};$$

N is axial force;
M is a bending moment in the investigated cross-section of a tube;
S_{tb} is a cross-section area of a tube;
W_{tb} is a modulus of section in the investigated cross-section of a tube.

The parameters s_M and s_{bg} and β are found from solution of the system of the equations:

$$\begin{cases} s_M = 1 - 2 \cdot \frac{\beta}{\pi} - \frac{a}{h} \cdot \frac{\theta}{\pi} \\ s_{bg} = \frac{4}{\pi} \cdot \sin\beta - \frac{2}{\pi} \cdot \frac{a}{h} \cdot \sin\theta \\ \sigma_M \cdot s_{bg} - \sigma_{bg} \cdot s_M = 0 \end{cases} \tag{J.5}$$

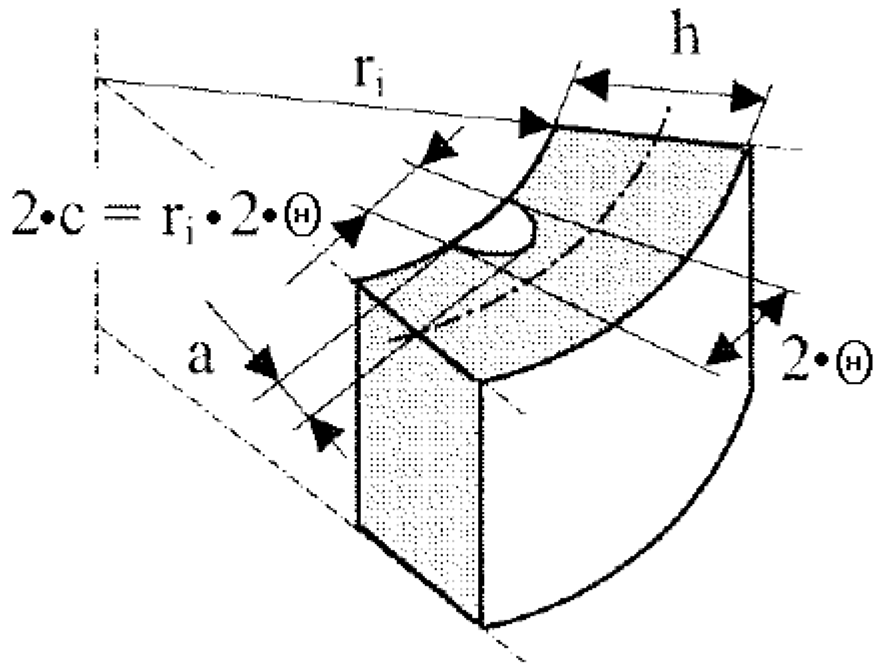

FIG. J.4. Layout of a crack in a diametral plane of a cylinder;

where a is a crack depth; 2·c is a crack length; 2·θ is a crack angle (in radians) located symmetrically to a plane of a bending moment action.

Parameters s'_M and s'_{bg} are found from the equations:

$$s'_M = 1 - \frac{2}{\pi} \cdot \arcsin\left(\frac{a}{2h} \cdot \sin\theta\right) - \frac{a}{h} \cdot \frac{\theta}{\pi},$$

$$s'_{bg} = \frac{4}{\pi} \cdot \sin\left[\frac{1}{2} \cdot \left(\pi - \frac{a}{h} \cdot \theta\right)\right] - \frac{2}{\pi} \cdot \frac{a}{h} \cdot \sin\theta \qquad \text{(J.4)}$$

Note : It is allowed to calculate using Equation (J.5) with changes of a/h from 0 to 0.8 and $\theta \leq \pi - \beta$.

J-4. THROUGH-THICKNESS CRACK IN A PLATE. THROUGH-THICKNESS CRACK IN A CYLINDER, WITH CRACK IS ORIENTED ALONG THE COMPONENT OF A CYLINDER

J-4.1. The geometry of the structure, crack and loadings is shown in Figure J.5.

J-.4.2. Reference stress is calculated by the formula:

$$\sigma_{ref} = \frac{\sigma_M}{1-\xi} \tag{J.6}$$

where $\xi = \frac{c}{W}$.

J-4.3. Membrane stress is calculated by the formula:

$$\sigma_M = \frac{1}{2 \cdot W \cdot h} \int_{-W}^{W} \int_{-h/2}^{h/2} \sigma(x, z) dzdx \tag{J.7}$$

where $\sigma(x,z)$ is stress normal to the plane of a crack.

J-5. INNER SEMIELLIPTICAL SURFACE CRACK IN A CYLINDER, WITH CRACK ORIENTED ALONG THE COMPONENT OF A CYLINDER

J-5.1. The geometry of the structure, crack and loadings is shown in Figure J.6.

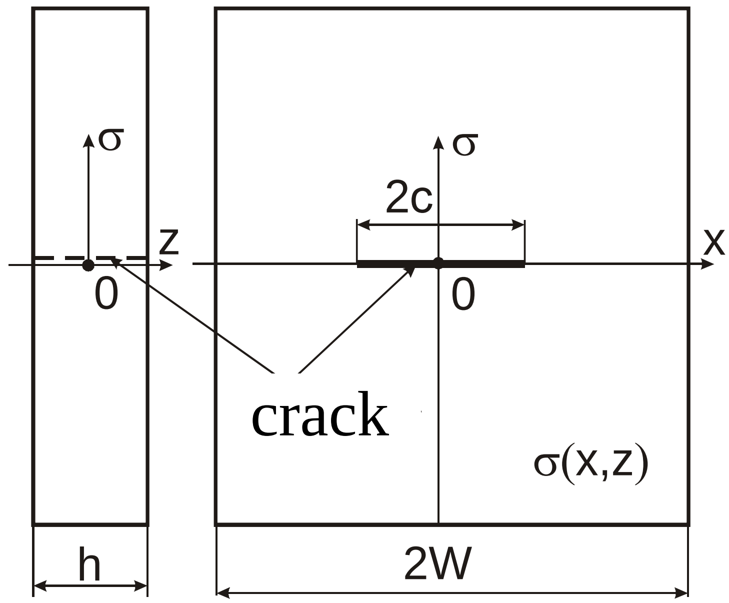

FIG. J.5. Through-thickness crack in a plate.

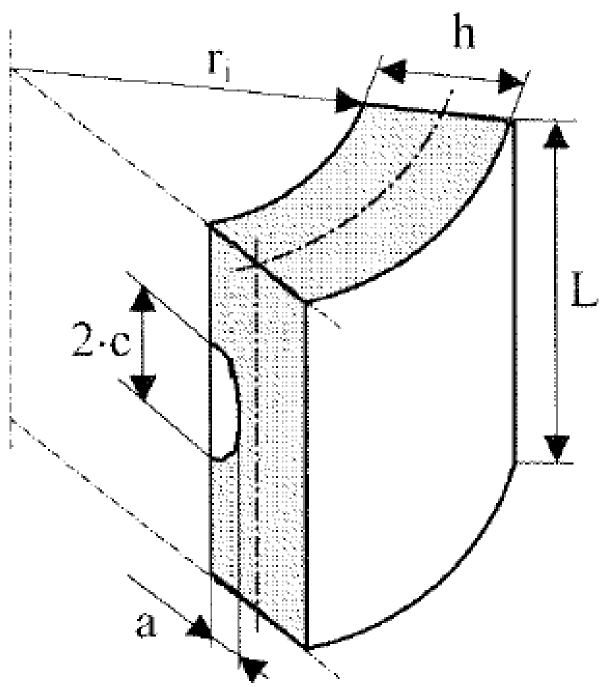

FIG. J.6. Layout of a crack along the component of a cylinder;where a is a crack depth; 2·c is a crack length.

J-5.2. Reference stress is calculated by the formula:

$$\sigma_{ref} = \frac{\sigma_M}{\sqrt{\left(1-\xi^{3.11}\right)^{1.9}}} \qquad (J.8)$$

where $\xi = \dfrac{a \cdot c}{h \cdot (c+h)}$; σ_m is a membrane stress component oriented perpendicular to a crack plane in a cylinder without crack.

Note: It is allowed to calculate using Equation (J.8) under changes of a / h from 0 to 0.8.

J-6. INNER ONE-QUARTER ELLIPTIC SURFACE CRACK IN A NODE FORMED BY INTERSECTION OF TWO CYLINDERS

J-6.1. When considering a one-quarter elliptic crack it is assumed that ellipse minor and major semiaxes are equal.

J-6.2. Geometry of the structure and crack is shown in Figure J.7.

J-6.3. Reference stress is calculated by Equation (J.8) provided that h is determined from Figure J.7 and c=a.

J-7. THROUGH-THICKNESS CRACK LOCATED IN A DIAMETRAL PLANE OF A CYLINDER

J-7.1. Geometry of the structure and crack is shown in Figure J.8.

J-7.2. Reference stress is calculated by equation:

$$\sigma_{ref} = \sqrt{\frac{\left(\dfrac{\sigma_M}{s'_M}\right)^2 + \left(\dfrac{\sigma_{bg}}{s'_{bg}}\right)^2}{\left(\dfrac{s_M}{s'_M}\right)^2 + \left(\dfrac{s_{bg}}{s'_{bg}}\right)^2}} \qquad (J.9)$$

where

σ_m and σ_{bg} are correspondingly membrane and global bending stresses:

$$\sigma_M = \frac{N}{S_{tb}},\ \sigma_{bg} = \frac{M}{W_{tb}};$$

N is axial force;

M is a bending moment in the investigated cross-section of a tube;

S_{tb} is a cross-section area of a tube;

W_{tb} is a modulus of section in the investigated cross-section of a tube.

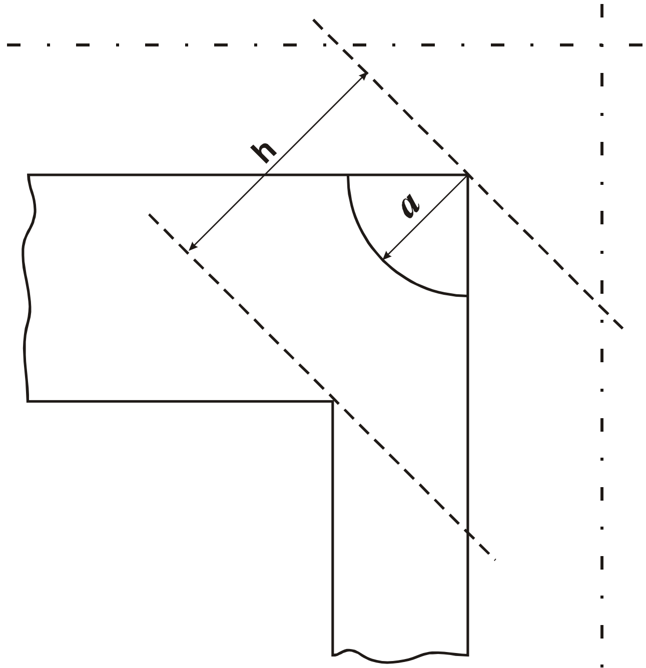

FIG. J.7. One-quarter elliptical surface crack in a node formed by two intersecting cylinders and diagram of the calculated parameter h.

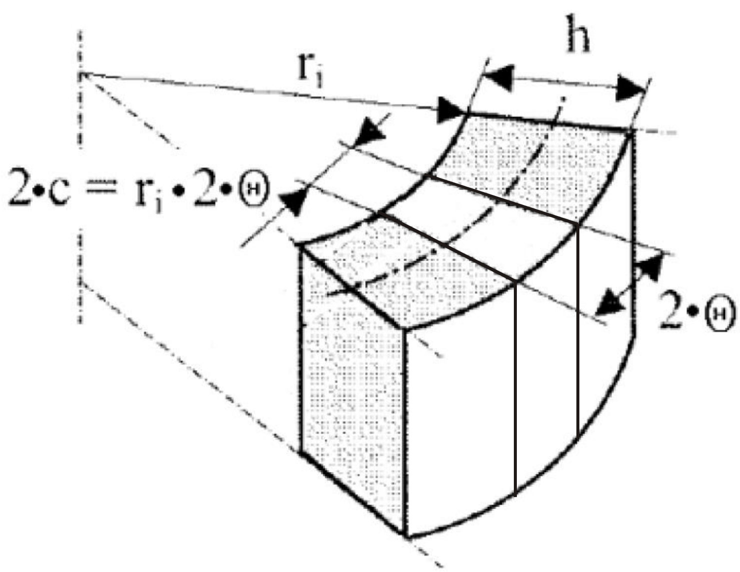

FIG. J.8. Layout of a crack in a diametral plane of a cylinder;
where $2 \cdot c$ is a crack length; $2 \cdot \theta$ is a crack angle (in radians) located symmetrically to the plane of a bending moment action.

Parameters s_M and s_{bg} and β are found from solution of the system of the equations:

$$\begin{cases} s_M = 1 - 2 \cdot \dfrac{\beta}{\pi} - \dfrac{\theta}{\pi} \\ s_{bg} = \dfrac{4}{\pi} \cdot \sin\beta - \dfrac{2}{\pi} \cdot \sin\theta \\ \sigma_M \cdot s_{bg} - \sigma_{bg} \cdot s_M = 0 \end{cases} \tag{J.10}$$

Parameters s'_M and s'_{bg} are found from the expressions:

$$s'_M = 1 - \frac{2}{\pi} \cdot \arcsin\left(\frac{1}{2} \cdot \sin\theta\right) - \frac{\theta}{\pi},$$

$$s'_{bg} = \frac{4}{\pi} \cdot \sin\left[\frac{1}{2} \cdot (\pi - \theta)\right] - \frac{2}{\pi} \cdot \sin\theta .$$

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[1] ASSOCIATION FRANÇAISE POUR LES RÈGLES DE CONCEPTION ET DE CONSTRUCTION DES MATÉRIELS DES CHAUDIÈRES ÉLECTRONUCLÉAIRES, Design and Construction Rules for Mechanical Components of FBR Nuclear Islands, Appendix A16, RCC-MR, Edition 2002, AFCEN, Paris, (2002).

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ANNEX K TO APPENDIX C

CONSTITUTIVE EQUATIONS TO CALCULATE VISCOELASTIC-PLASTIC PROBLEMS BY FEM

Calculation of viscoelastic-plastic problems is made in an unsteady quasi-static non-isothermic statement.

K-1. STRESSES AND STRAINS

Stresses and strains are determined by the following equations:

$$\sigma_{ij} = s_{ij} + \delta_{ij}\sigma_m \tag{K.1}$$

$$d\varepsilon_{ij} = d\varepsilon_{ij}^{e} + d\varepsilon_{ij}^{p} + d\varepsilon_{ij}^{c} + \delta_{ij}d\varepsilon^{T} + \delta_{ij}d\varepsilon^{sw} \tag{K.2}$$

$$\sigma_m = \frac{\sigma_{ii}}{3} \tag{K.3}$$

$$\delta_{ij} = \begin{cases} 1, \text{for } i = j \\ 0, \text{for } i \neq j \end{cases} \tag{K.4}$$

where

σ_{ij}, s_{ij}	are, correspondingly, components of stress tensor and deviatoric stress tensor;
σ_m	is a hydrostatic stress;
$d\varepsilon_{ij}$, $d\varepsilon_{ij}^{e}$, $d\varepsilon_{ij}^{p}$, $d\varepsilon_{ij}^{c}$	are, correspondingly, tensor component increments of total strains ε_{ij}, elastic strains ε_{ij}^{e}, plastic strains ε_{ij}^{p} and creep strains ε_{ij}^{c};
$d\varepsilon^{t}$, $d\varepsilon^{sw}$	are strain increments determined by temperature and swelling, correspondingly.

Equations (K.1) - (K.4) make it possible to investigate the kinetics of SSF in a structure with regard for viscoelastic-plastic deformation of a material, given its known properties and specified boundary conditions, and strains caused by thermal expansion and swelling. Temperature strains are determined from thermal fields calculated by means of solution of a steady or unsteady problem by the certified programme codes. The following equations are used to describe material behavior in the region of elastic, plastic and viscous deformations, as well as deformation caused by swelling.

K-2. EQUATIONS OF THE ELASTICITY THEORY

The equation describing the relationship between stresses and elastic strains is written in the form:

$$\varepsilon_{ij}^{e} = s_{ij} / 2G + \delta_{ij} K \sigma_m \tag{K.5}$$

where

$$G = \frac{E}{2(1+\nu)} \tag{K.6}$$

$$K = \frac{1-2\nu}{E} \tag{K.7}$$

E=E(T)	is Young's modulus of elasticity;
G=G(T)	is shear modulus;
K=K(T)	is volume compressibility;
ν	is Poisson ration, ν=0.3;
T	is temperature.

The dependence of the Young modulus of elasticity E on temperature is presented in the Annex B.

K-3. EQUATIONS OF THE THEORY OF PLASTICITY

In order to describe plastic strain incremental theory is used. The tensor components of plastic strain increments under loading are determined by the associated flow rule in the form:

$$d\varepsilon_{ij}^{p} = \frac{3}{2}\frac{d\varepsilon_{eq}^{p}}{\beta_{eq}}\beta_{ij} \tag{K.8}$$

where

$$d\varepsilon_{eq}^{p} = \sqrt{\frac{2}{3} d\varepsilon_{ij}^{p} d\varepsilon_{ij}^{p}} \tag{K.9}$$

$$\beta_{ij} = s_{ij} - \rho_{ij} \tag{K.10}$$

$$\beta_{eq} = \sqrt{\frac{3}{2}\beta_{ij}\beta_{ij}} \tag{K.11}$$

β_{ij} are components of tensor of effective stresses; ρ_{ij} are microstresses tensor components, ρ_{ii}=0.

In order to describe elastic-plastic strain of RVI component material it is recommended to use the equations on the basis of nonlinear kinematic hardening in the following form:

$$\left[\frac{3}{2}(s_{ij}-\rho_{ij})(s_{ij}-\rho_{ij})\right]^{1/2}-C_{p0}=0 \quad (K.12)$$

$$C_{p0}=\sigma_Y(F,T,T_{irr}) \quad (K.13)$$

$$\dot{\rho}_{ij}=g_1\xi_{ij}^p-g_2\rho_{ij}\,\dot{æ}_p \quad (K.14)$$

$$\xi_{ij}^p=\frac{d\varepsilon_{ij}^p}{dt} \quad (K.15)$$

where

C_{p0}	is the radius of a yield surface;
T	is the current operational temperature;
T_{irr}	is an irradiation temperature;
F	is a damage dose of neutron irradiation;
$æ_p=\int d\varepsilon_{eq}^p$	
$\sigma_Y(F,T,T_{irr})$	is yield strength;
$\dot{\rho}_{ij}$	is the rate of a microstress tensor;
g_1 and g_2	are material parameters (functions of temperature and damage dose of neutron irradiation);
ξ_{ij}^p	are components of tensor of plastic strain rates.

The dependence of yield strength $\sigma_Y(F,T,T_{irr})$ is presented in the Annex A. The parameters g_1 and g_2 are determined in section K-6.

K-4. EQUATIONS OF THE CREEP THEORY

The components of tensor of creep strain increments are determined by the following equations:

$$d\varepsilon_{ij}^c=\frac{3}{2}\frac{\xi_{eq}^c dt}{\sigma_{eq}}s_{ij} \quad (K.16)$$

where

$$\xi_{eq}^c=\sqrt{\frac{2}{3}\xi_{ij}^c\xi_{ij}^c} \quad (K.17)$$

$$\xi_{ij}^c=\frac{d\varepsilon_{ij}^c}{dt} \quad (K.18)$$

$$\sigma_{eq} = \sqrt{\frac{3}{2} s_{ij} s_{ij}} \quad (K.19)$$

ξ_{ij}^{c} are components of tensor of creep strain rates; dt is a time increment.

The equation of radiative creep at different temperatures and the rates of neutron irradiation damage dose is:

$$\xi_{eq}^{c}\left(\Phi, \sigma_{eq}\right) = (B\Phi + D\dot{S}\ ) \cdot \sigma_{eq} \quad (K.20)$$

where $\dot{S}$ is a swelling rate at the given level of stresses calculated according to Equation (G.6) in the Annex G.

K-5. EQUATIONS OF SWELLING

Swelling without account for the effect of stress and plastic strain is calculated by the equation:

$$S_0 = c \cdot F^{n_v} \cdot \exp[-r \cdot (T_{irr} - T_m)^2] \quad (K.21)$$

Calculation of the change in swelling strain $d\varepsilon^{sw}$ in the time dt is made by the formula:

$$d\varepsilon^{sw} = \frac{1}{3} \dot{S} \cdot dt \quad (K.22)$$

where $\dot{S}$ is the rate of swelling depending on an irradiation temperature, stress and plastic strain. It is written in the form:

$$\dot{S} = \begin{cases} \dot{S}_0 \cdot f_1(\sigma_m) \cdot f_2(æ_p), \\ 0, \text{ if } S \le 0 \text{ and } f_1(\sigma_m) < 0 \end{cases} \quad (K.23)$$

$$\dot{S}_0 = c \cdot n_v \cdot F^{(n_v - 1)} \cdot \Phi \cdot \exp[(-r \cdot (T_{irr} - T_m)^2] \quad (K.24)$$

$$f_1(\sigma_{eff}) = 1 + P \cdot \sigma_{eff} \quad (K.25)$$

$$\sigma_{eff} = (1-\eta_1) \cdot \sigma_m + \eta_1 \cdot \sigma_{eq} \quad (K.26)$$

$$f_2(æ_p) = \exp(-\eta_2 \cdot æ_p) \quad (K.27)$$

where

c, n_v, r, P, K, η_1, η_2, T_m	are material constants;
Φ	is the damage dose rate of a neutron irradiation, dpa/s;
F	is a damage dose of neutron irradiation, dpa;
T_{irr}	is an irradiation temperature, °C.

The values of the constants in Eqs. (K.21), (K.23) - (K.27) are given in the Annex G.

K-6. DETERMINATION OF MATERIAL PARAMETERS G1 AND G2 IN THE EQUATION OF PLASTICITY WITH NONLINEAR KINEMATIC HARDENING

Parameters g_1 and g_2 may be determined from the diagram of elastic-plastic strain with the use of Eqs. (K.12) - (K.15).

In accordance with the Annex A in order to describe the diagrams of elastic-plastic strain the dependence is used which for a uniaxial tension may be presented in the form:

$$\sigma_{11} - \sigma_Y = A \cdot (\varepsilon_{11}^p)^n \tag{K.28}$$

where for the uniaxial case ε_{11}^p = $æ_p$, σ_{11} = σ_{eq}; $A(F,T,T_{irr})$ and $n(F,T,T_{irr})$ are material parameters depending on the current test temperature T, fluence F and irradiation temperature T_{irr}, the parameter values are given in the Annex A.

For uniaxial tension and kinematic hardening from Equation (K.12) we have:

$$\rho_{11} = \frac{2}{3} \cdot \left(\sigma_{11} - C_{p0}\right) \tag{K.29}$$

From Equation (K.28) with regard to Equation (K.29) we obtain:

$$\rho_{11} = \frac{2}{3} \cdot A \cdot \left(\varepsilon_{11}^p\right)^n \tag{K.30}$$

Integration of Equation (K.14) on tension ($d\varepsilon_{11}^p > 0$) gives:

$$\rho_{11} = \frac{g_1}{g_2} \cdot \left[1 - \exp\left(-g_2 \varepsilon_{11}^p\right)\right] \tag{K.31}$$

Parameters g_1 and g_2 may be determined by the least squares technique on the basis of Equation (K.31) where numerical values ρ_{11} and ε_{11}^p obtained by Equation (K.30) are used as inputs.

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ANNEX L TO APPENDIX C

DETERMINATION OF A LIMIT EMBRITTLEMENT AREA AND DIAGRAM OF A POSTULATED CRACK

L-1. An area with limit embrittlement (LEA) is a material area in an RVI component that is bordered by one or several closed surfaces, for each point of which neutron damage dose F at the considered time meets the condition

$$F=F_{crit}(T_{irr}) \tag{L.1}$$

and for all the points of LEA the condition below is satisfied.

$$F \geq F_{crit}(T_{irr}) \tag{L.2}$$

The critical value of damage dose, F_{crit}, for a given irradiation temperature T_{irr} is determined as the damage dose above which $\gamma \rightarrow \alpha$–transformation occurs. The dependence $F_{crit}(T_{irr})$ is shown in Figure L.1.

L-2. If some domains exist inside a closed surface bordering the LEA for which condition (L.2) is not satisfied, it may be assumed that such domains also belong to the LEA.

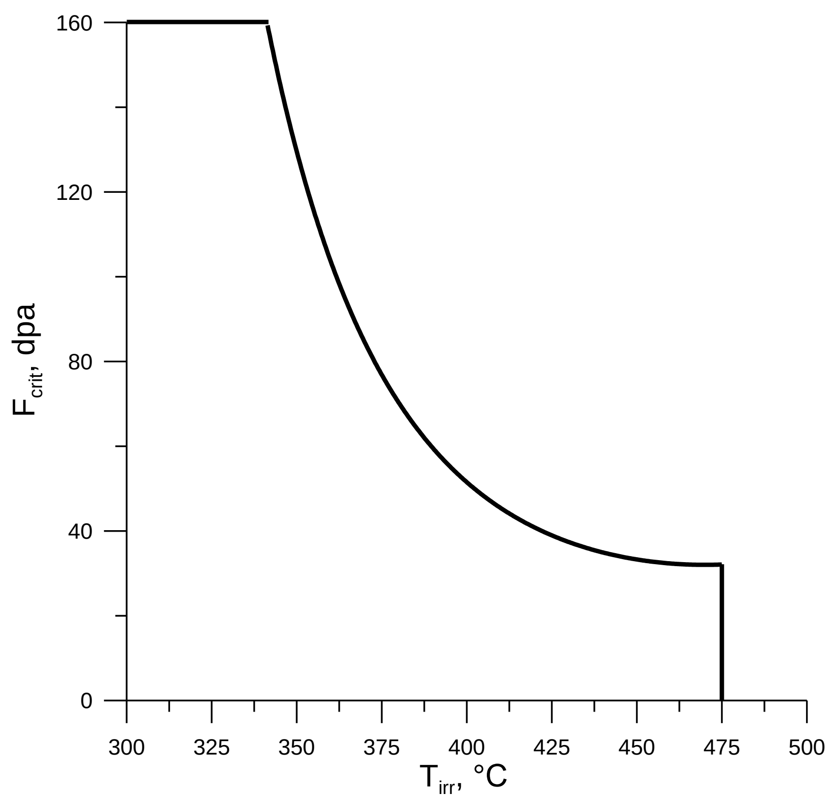

FIG. L.1. Dependence of critical dose F_{crit} on T_{irr}.

L-3. It is assumed that in the LEA there are crack-like cracks with maximum sizes that are restricted by the LEA boundary.

L-4. The postulated crack in LEA is defined from the following:

L-4.1. If the maximum size of the LEA does not exceed 2 mm, it may be assumed that there are no cracks in the LEA, and that the properties of material inside the LEA are the same as for austenitic material outside the LEA.

L-4.2. The postulated crack in LEA is a plane elliptic crack, a semi-elliptic surface crack or an one-quarter elliptical crack (if the LEA boundary coincides with a surface of the considered internal components).

L-4.3. The secant plane in LEA is postulated corresponding to the following requirements:

L-4.3.1. The plane is orientated in such a way that the growth of a crack in this plane calculated by IASCC and fatigue mechanisms is a maximum for the considered time interval.

L-.4.3.2. For the secant plane orientated according to section L-4.3.1, its intersection with LEA is found in such a way to provide the maximum area of LEA domain lying in the secant plane.

L-4.4. Elliptic, semi-elliptic or one-quarter elliptical cracks with the sizes a_0 and c_0 are taken as the postulated crack surrounding the LEA lying in the secant plane (see Figure L.2). If a minimum size of the postulated crack does not exceed 2 mm then this size is taken to be equal to 2 mm.

REFERENCES

[1] MARGOLIN, B.Z., KURSEVICH, I.P., SOROKIN, A.A., LAPIN, A.N., KOHONOV, V.I. NEUSTROEV, V.S., About the problem of irradiation swelling and embrittlement of austenitic steels. Part I. Experimental results, “Voprosy Materialovedeniya” (Problems of materials science), №2(58), (2009), 89–98 (in Russian).

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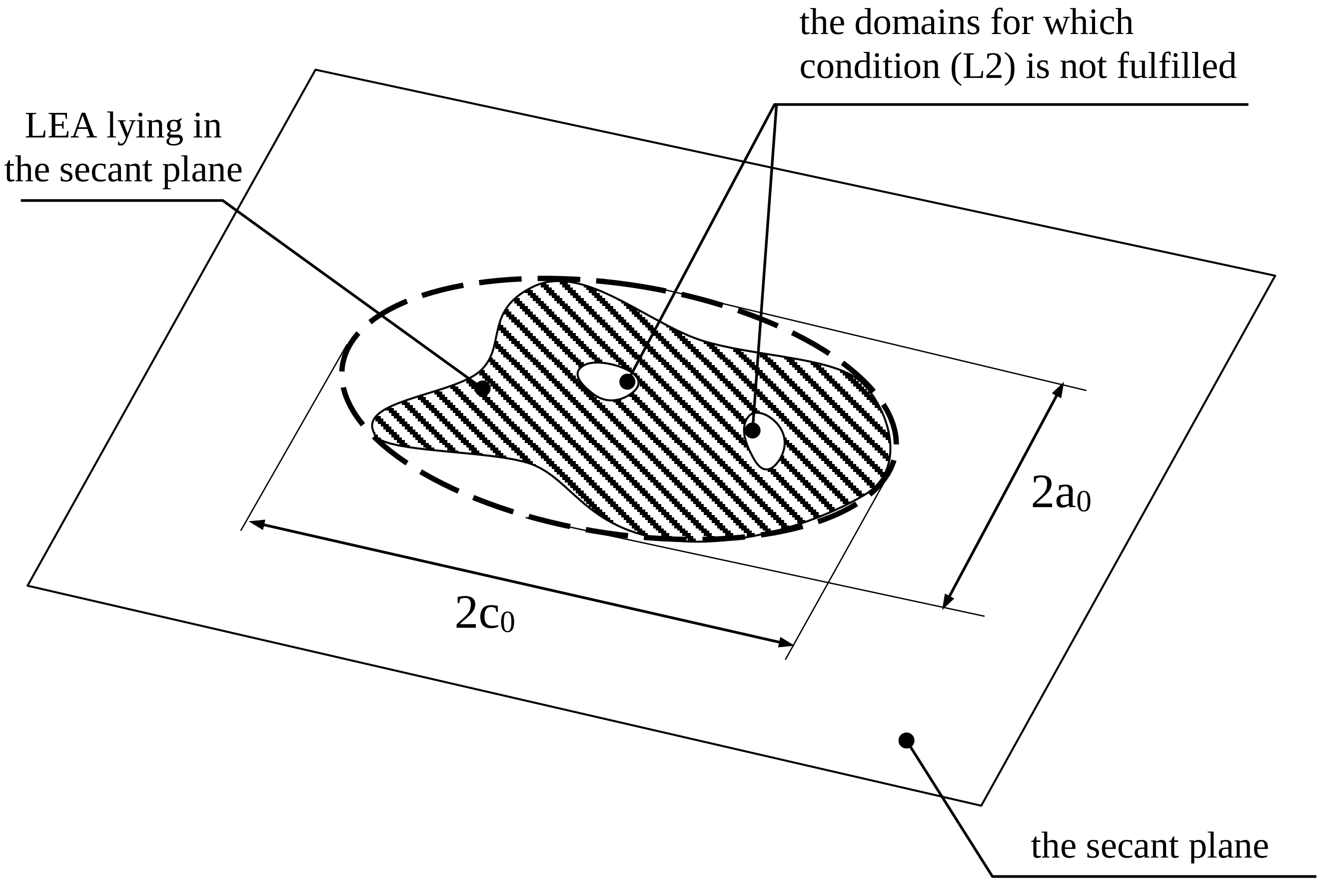

FIG. L.2. Postulated crack surrounding the LEA lying in the secant plane.

(– – – – – postulated crack)

APPLICATION 1 TO APPENDIX C

VIBRATION LOADING AND WEAR OF RVI COMPONENTS

A1-1. TERMS AND DEFINITIONS

Component means

Any independent part, combination of parts, assemblies, sub-assemblies or units of RVI that functions in a defined mode to ensure correct working of the reactor.

Computational fluid dynamics (CFD)

One of the branches of fluid mechanics that uses numerical methods to analyze problems that involve fluid flows. The fundamental basis of almost all CFD problems are the Navier-Stokes equations, which define any single-phase fluid flow. For solution of these equations, many state-of-art computer methods are used which simulate the interaction of liquids and gases with structural surfaces defined by boundary conditions.

Coupled coordinates

Degrees of freedom of a structure used as variables in differential equations of motion (coupled dynamic equilibrium equations); in finite analysis methods, these are, as a rule, nodal displacements.

Damping

Means a phenomenon of energy dissipation (energy loss) during motion of a structure; it results in reduction of amplitude of structural oscillations with time.

Damping ratio

A dimensionless measure describing level of damping in a structure, i.e. how oscillations of a structure decay after a disturbance. The damping ratio is defined as the ratio of the damping constant to the critical damping constant:

$$\xi = \frac{c}{c_{cr}} .$$

The damping ratio is also related to the logarithmic decrement (see below) via the relation

$$\xi = \frac{1}{\sqrt{1+\left(\frac{2\pi}{\delta}\right)^2}} .$$

Fluid-elastic instability

A class of phenomena caused by effect of structure-fluid coupling forces; for example, it manifests itself in the form of initiation and self-sustaining of structural oscillations in fluid flow, and also in abrupt rising of oscillation amplitude at a slight increasing of fluid velocity.

Logarithmic decrement

A parameter characterized damping at oscillation of a structure. The logarithmic decrement is the natural log of the amplitudes ratio for any two successive oscillations:

$$\delta = \frac{1}{n} \ln \frac{x_0}{x_n},$$

where x_0 is the greater of the two amplitudes and x_n is the amplitude *n* periods away.

Mode shape

The eigenvector relating to a certain normal mode of structural oscillations; it presents spatial distribution of displacement amplitudes at oscillations on this mode; is obtained as a result of solution of the eigenvalue problem.

Natural frequency

A square root of the eigenvalue relating to a certain normal mode of structural oscillations; it is frequency of oscillations on this mode; is obtained as a result of solution of the eigenvalue problem.

Normal coordinates

Degrees of freedom of a structure used as independent variables in a system of uncoupled differential dynamic equilibrium equations; these equations may be obtained by means of modal coordinate transformation applied to the coupled coordinates and based on solution of the eigenvalue problem.

Turbulence or **turbulent flow**

A fluid regime characterized by chaotic, stochastic property changes. This includes rapid variation of the fluid pressure and velocity in space and time. Turbulence causes the formation of eddies of many different length scales.

Vibration

Mechanical oscillation, i.e. the repetitive (periodical) motion of a structure under applied time-depending loading about an equilibrium point. The term ‘vibration’ is used, as a rule, more narrowly to mean a mechanical oscillation which occurs with relatively high frequency.

Vortex shedding

Means an unsteady flow which is caused when a fluid (or gas) flows past a blunt body. In this flow, vortices are created at the back of the body and detach periodically from either side of the body.

Wear

The erosion (removal) of material from a solid surface by the action of another surface.

A1-2. PRINCIPAL SYMBOLS AND ABBREVIATIONS

A1-2.1. TECHNICAL SYMBOLS

C_y	Hydrodynamic cross resistance (lift) coefficient
f	Reference geometrical dimensions (diameter and length) of an RVI component, mm
F_N	Vibration frequency, Hz
H	Normal force, N
m	Hardness, N/mm^2
Re	Structural mass and associated fluid mass per length unit, kg/m
Sh	Reynolds number
$\overline{u}$	Strouhal number
ΔV	Flow velocity, m/s
y_{max}	Volume increment, m^3
δ	Amplitude of structural vibrations, mm
Ξ	Logarithmic decrement
ρ	Damping ratio
$\sigma^2(x)$	Fluid density, kg/m^3
ω	Mean square response (response parameter is designated by x)
ϕ	Vibration circular frequency, rad/s
	Mode shape

A1-2.2. ABBREVIATIONS OF TERMS

AC	Aircraft crash
DA	Design accident
DE	Design earthquake
FEM	Finite element method
MDE	Maximum design earthquake
NOC	Normal operation conditions
SWA	Shock wave action
VNOC	Violation of normal operation conditions

A1-3. GENERAL

This application covers vibration loading and vibration-induced wear of RVI components during operation.

Generally, sources of vibrations (oscillations) of RVI components are the following:

- Seismic events;
- Loads in accidents caused by external and internal dynamic actions (aircraft crash – AC, shock wave action – SWA, hydraulic shock, double ended guillotine break of primary piping and other design accidents – DA);
- Hydrodynamic excitation in coolant fluid flow.

Accordingly, for strength assessment of PRV internals the following combinations of loading modes shall be considered:

- NOC + MDE/AC/SWA[1];
- VNOC + MDE/AC/SWA;
- NOC + DE;
- VNOC + DE;
- DA + DE.

As a rule, seismic events and loads in accidents are caused by a few cycles of vibration loading, so they determine rather the static strength of RVI components, and do not impact or determine cyclic strength or vibration-induced wear. Therefore, in accordance with the subject of this Application, only hydrodynamic excitation will be taken into account, and it shall be considered within the load range applicable for NOC and VNOC.

In case of necessity, vibrations caused by earthquakes or accidents can be analyzed by means of dynamic analysis methods described in Section 5 of this Application. It be noted that the loading mode AC might not have been included in the design basis for older WWER NPPs.

Vibration loading of an RVI component is characterized by time variation of applied surface forces, inertia forces and internal structural friction forces, as well as kinematical parameters caused by these forces (displacements, velocities, accelerations), and stresses and strains in component points.

Vibration loading impacts on

- Maximum values of stresses and strains (in the form of additives to quasi-static values of these parameters caused by weight, pressure, temperature and other loads slowly varying in time; because of small values of these additives, as a rule, they cannot be taken into account).
- Cyclic loading parameters (in the form of additional cycles of alternating-sign stresses; they can result in fatigue damage of RVI components).
- Self-loosening of bolts (due to vibration-induced decrease of friction forces).

[1] Sign „/“ means „or“

— Wear of metal at contact zones of movable and (or) weakened mechanical joints (in the form of surface crumpling, scuffing, chipping and other factors of initial geometry changes due to vibration-induced micro-impacts).

Vibration loading parameters (frequencies and amplitudes of forces, stresses and strains) in various RVI components and at various zones of a component can differ significantly. Vibration loads and kinematical parameters in RVI components shall be obtained by dynamic analysis. It is allowed to determine vibration loads and kinematical parameters by means of recording and generalization of relevant experimental data.

Stresses and strains in RVI components can be calculated on the basis of vibration loads and kinematical parameters obtained by analyzing of element quasi-static equilibrium under external and internal loads.

Assessment of vibration loading (for cyclic strength analysis), changes of properties of structural materials and self-loosening of bolts due to vibrations as well as assessment of wear shall be carried out for those RVI components whose structural condition (structural health) changes in dependence on the mentioned factors and impacts sufficiently on reactor life.

A1-4. VIBRATION HYDRODYNAMIC EXCITATION MECHANISMS

Hydrodynamic-induced vibrations of RVI components arise and are supported due to energy of fluid coolant flow. Vibrations are excited by the following mechanisms:

— Turbulent pulsations of flow pressure;

— Vortex shedding;

— Fluid-elastic instability.

Each of these mechanisms appears in certain areas of flow velocity. For an example see Figure A1.1 below for a chart of vibrations versus flow veolocity for steam generator tubes.

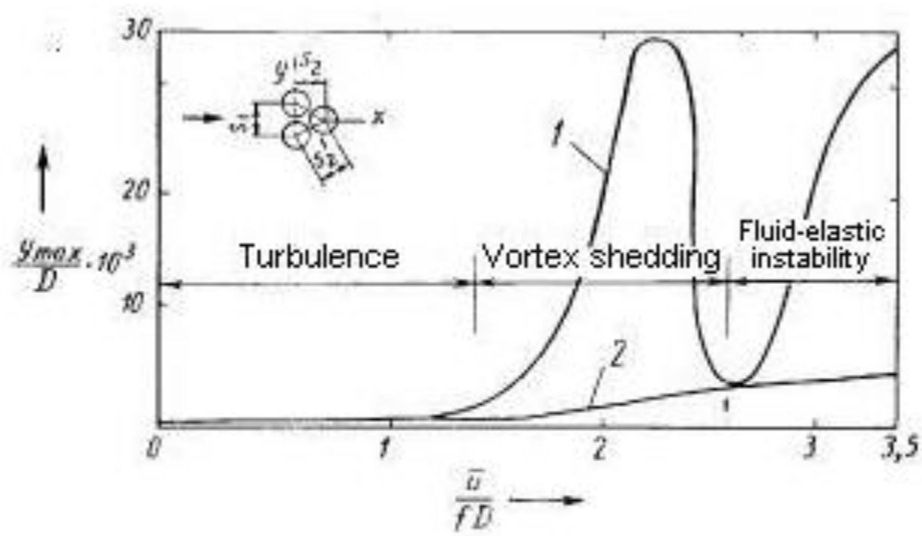

FIG.A1.1 Mechanisms of vibration hydrodynamic excitation in dependence on flow velocity for steam generator tubes (1 – 1st row, 2 – 5th row of tubes [2]*).*

Vibration analysis shall be carried out for each of the mechanisms of hydrodynamic excitation.

A1-5. VIBRATION ANALYSIS METHODS

A1-5.1. GENERAL APPROACH

Generally, for vibration analysis, a mathematical model of the reactor structure including RVI shall be built. This model shall reflect real technical conditions of the structure within the considered operation interval, as well as structural dynamics features within the vibration frequency band under investigation, taking into account structure-fluid coupling forces. For mathematical modeling, the finite element method (FEM) is recommended.

Calculation of the vibration parameters can be carried out by the following dynamic analysis methods:

- Dynamic analysis in the coupled coordinates (linear or nonlinear analysis);
- Dynamic analysis in the normal coordinates (modal linear analysis);
- Linear spectral method;
- Complex frequency response method.

Linear dynamic analysis in the coupled coordinates is based on the following differential equation of structure motion in matrix form:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F\} \qquad \text{(A1.1)}$$

where

$[M]$, $[C]$, $[K]$	are inertia, damping and stiffness matrices expressed in terms of variables (degrees of freedom) in nodes of a structure numerical model;
$\{x\}$, $\{\dot{x}\}$, $\{\ddot{x}\}$	are column vectors of nodal displacements, velocities and accelerations;
$\{F\}$	is column vector of nodal forces representing external loading as function of time.

A nonlinear behavior of a structure may be caused by either a material nonlinearity (i.e. a nonlinear stress-strain relationship of the material), a geometrical nonlinearity (an excessive deformation, a support gap, etc. which significantly changes the geometry of the structure), or a mixture of the two.

The differential equation of motion for nonlinear problem is:

$$[M(x)]\{\ddot{x}\} + [C(x)]\{\dot{x}\} + [K(x)]\{x\} = \{F\} \qquad \text{(A1.2)}$$

where [$M(x)$], [$C(x)$], [$K(x)$] are inertia, damping and stiffness matrices as functions of structural displacements.

Nonlinear problem may be expressed in terms of nodal increments:

$$[M]_{t,t+\Delta t}\,\Delta\{\ddot{x}\}_{t+\Delta t} + [C]_{t,t+\Delta t}\,\Delta\{\dot{x}\}_{t+\Delta t} + [K]_{t,t+\Delta t}\,\Delta\{x\}_{t+\Delta t} = \{F\}_{t+\Delta t} \tag{A1.3}$$

where $[M]_{t,t+\Delta t}$, $[\tilde{N}]_{t,t+\Delta t}$, $[K]_{t,t+\Delta t}$ are matrices of inertia, damping and stiffness coefficients established for the time interval (t, $t+\Delta t$); $\Delta\{x\}_{t+\Delta t} = \{x\}_{t+\Delta t} - \{x\}_t$, etc.

The Equation (A1.3) is more convenient for the problems with large nonlinearities.

Dynamic analysis in the normal coordinates assumes that a modal transformation is applied to inertia, stiffness and damping matrices – the such that

$$\begin{aligned}
[\phi]^T[M][\phi] &= [I] \\
[\phi]^T[K][\phi] &= [\omega^2] \\
[\phi]^T[C][\phi] &= [2\xi\omega]
\end{aligned} \tag{A1.4}$$

where

the right-hand sides	are diagonal matrices;
[ϕ]	is the eigenvector (modal) matrix;
[ω^2]	is the eigenvalue matrix (ω is the natural circular frequency);
ξ	designates the damping ratio.

Let $\{\eta\}$ be the normal coordinates such that the displacement $\{x\}$ can be defined by the transformation:

$$\{x\} = [\phi]\{\eta\} \tag{A1.5}$$

Equation (A1.5) can be written by means of this transformation as

$$\{\ddot{\eta}\} + 2\xi\omega\{\dot{\eta}\} + \omega^2\{\eta\} = [\phi]^T\{f\} \tag{A1.6}$$

which represents a decoupled system of individual modal equation. Response in each mode may be obtained by solving the differential equations individually for the corresponding mode. The total response of a structure is therefore the result of combining the responses of all its component modes.

Linear spectral method and complex frequency response method are described in [1] and [3] accordingly.

Structure-fluid interaction in the case of small vibrations of a structure in an incompressible fluid can be taken into account in matrices [M] and [C] by means of added mass and added damping. Methods for determination of added mass and added damping for structural elements of simple analytical shapes are presented in [3]. For analysis of vibrations of a complex shaped structure interacting with non viscous incompressible fluid which fills narrow channels between its components, e.g. core barrel of the reactor, a numerical method described in [7] can be used.

Generally, hydrodynamic forces impacting RVI components in coolant flow enter in the right-hand side of Equation (A1.1); determination of these forces can be carried out by sophisticated methods of Computational fluid dynamics (CFD) – see, for example [8].

A1-5.2. ANALYSIS PROCEDURE

Input data for the analysis of hydrodynamic-induced vibrations are the following:

- Geometry parameters of RVI;
- Mechanical properties of RVI materials;
- Boundary conditions including geometry of reactor's surrounding structure;
- Values and directions of mechanical forces applied to RVI;
- Velocity and pressure of primary coolant, pulsation amplitude and frequency.

Procedure of the analysis includes:

- Determination of dynamic characteristics for RVI components;
- Determination of flow features for RVI components;
- Calculation of flow parameters and hydrodynamic forces effecting on RVI components;
- Identification of RVI components which are under conditions of flow-induced instability and resonance;
- Calculation of frequencies and amplitudes of RVI components forced vibrations;
- Estimation of vibration strength for RVI components.

The analysis of hydrodynamic-induced vibrations is carried out for NOC. If needed the analysis shall be carried out for VNOC.

A1-5.3. DETERMINATION OF DYNAMIC CHARACTERISTICS OF RVI

Dynamic characteristics for RVI components are the following:

- Natural frequencies f1, f2, …;
- Mode shapes ϕ_1, ϕ_2, …;

— Values of logarithmic decrement ξ_1, ξ_2 …

The natural frequencies and mode shapes are calculated as a result of solution of the eigenvalue problem; this problem is defined practically on the basis of the mathematical modeling of RVI (see Clause A1-5.1). Values of logarithmic decrement are obtained on the basis of relevant experimental data. In the absence of such data, logarithmic decrement can be taken in accordance with damping values given in available regulatory documents – see, for example, [1-3].

Number of natural modes to be taken into account shall be necessary and sufficient for simulation of structural dynamics features of RVI components within the vibration frequency band under investigation.

A1-5.4. TURBULENT PULSATIONS OF FLOW PRESSURE

Turbulent pulsations of coolant flow pressure are generated mainly by the main coolant pump of the reactor. The main components of these pulsations are harmonics multiple of rotation and blade frequencies of the pump, laying in the frequency range up to 100 Hz.

Vibrations of RVI components due to turbulent pulsations of coolant flow pressure can be determined on the basis of the general approach described in Clause A1-5.1 that involves sophisticated calculation methods. Below, a simplified method applicable in many practical cases is given.

For vibration analysis, the assumption of uniform mean velocity and homogeneous turbulence (homogeneous and ergodic pressure field) is reasonable.

The mean square response $\sigma^2(x)$ is the most useful measure for the amplitude of vibration displacements and is found by integration of the power spectral density of the response, $S_y(x,\omega)=S_y(x,x,\omega)$ over the frequency band:

$$\sigma^2(x)=\int_{-\infty}^{\infty} S_y(x,\omega)\,d\omega \qquad \text{(A1.7)}$$

where *x* designates spatial coordinate, ω means vibration circular (radian) frequency.

Assuming a Gaussian fluctuating pressure distribution, a Rayleigh distribution is expected for the absolute amplitude of response [3].

In the case of lightly damped structures with well separated modes, cross modal contribution to the response can be ignored, and Equation (A1.7) can be analytically evaluated as the following [3]:

$$\sigma^2(x)=\sum_j 2\pi\xi_j\omega_j S_y(\omega_j)=\sum_j \pi\xi_j f_j G_y(f_j) \qquad \text{(A1.8)}$$

where ω_j and ξ_j are the circular natural frequency and damping ratio of the *j*-th vibration mode of an RVI component, respectively f_j designates natural frequency of this mode (in Hz); $G(f)$ is the single-sided power spectral density of turbulent flow pressure pulsations as a function of f .

Under the assumption of light structural damping, the Equation (A1.8) can be transformed in the following formula [3]:

$$\sigma^2(x) = \sum_j \frac{L^3 G_f(f_j)\phi_j^2(x)}{64\pi^3 M_j^2 f_j^3 \xi_j} J_{jj}^2 \tag{A1.9}$$

where

L	is reference geometrical dimension of a RVI component;
$G_f(f_j)$	in MPa^2/Hz is generated by the turbulent pressure field at the natural frequency f_j of the *j*-th vibration mode;
ϕ_j	is the *j*-th mode shape function;
M_j	is the *j*-th modal masse;
J_{jj}	designates the joint cross acceptance which reflects contributions into structural response due to coupling between different modes.

When vibration analyzing, the power spectral density of turbulent flow pressure pulsations $G(f)$ may be given in the form of an analytical approximation of available experimental data obtained during operation of WWERs or at special tests.

On the basis of mean square vibration displacement amplitude $\sigma^2(x)$ calculated in accordance with Equation (A1.9), amplitudes of stresses and strains can be obtained as is described in Section A1-6 of this Application.

A1-5.5. VORTEX SHEDDING

In general, hydrodynamic forces caused by vortex shedding can be obtained by a numerical simulation using CFD methods (see Clause A1-5.1). Vibrations of RVI components due to these forces can be determined by solving the equations of structure motion given in Clause A1-5.1. Below, a simplified method which does not involve complex calculations is given.

The frequency of the hydrodynamic force (the frequency of vortex shedding) is calculated as:

$$f_V = Sh\frac{\bar{u}}{d} \tag{A1.10}$$

where $\bar{u}$ is flow velocity in narrow channels formed by RVI components and the reactor's surrounding structure; *d* designates reference dimension of an RVI component.

Strouhal number *Sh* is defined experimentally and for Reynolds number $10^3 < Re < 2\times10^5$ lays in the range 0.2 – 0.36.

Maximum amplitude of forced vibrations due to vortex shedding for an element of cylindrical shape subject to cross flow is evaluated by the following formula [2]:

$$y_{\max} = \frac{C_y d\rho \bar{u}^2}{4\pi^2 f_1^2 m \sqrt{\left[1-\left(\frac{f_V}{f_1}\right)^2\right]^2 + \left(\frac{\delta}{\pi}\right)^2 \left(\frac{f_V}{f_1}\right)^2}} \tag{A1.11}$$

where

C_y	is hydrodynamic cross resistance (lift) coefficient to be defined experimentally; commonly, it shall not exceed 0.7 [2];
ρ	is fluid density;
f_1	is the basic, as a rule, the first natural frequency which determines vibration loading level of a RVI component;
m	designates structural mass of a RVI component and associated fluid mass per length unit;
δ	is the logarithmic decrement expected equal for all natural frequencies under consideration.

Maximum possible amplitude of vibrations (in the case of resonance) can be calculated by the formula [2]:

$$y_{\max} = \left(\frac{C_y d\rho}{8\pi}\right)^2 \frac{\bar{u}^2}{f_1^2} \frac{1}{m\delta} \tag{A1.12}$$

This approach is applicable in the cases of cross flow, e.g. for flow across the protective tube unit.

When vortex shedding, vibration excitation due to simultaneous turbulent pulsations is possible. In this case, maximum amplitude of vibrations with the probability 0.997 is evaluated by the formula [2]:

$$y_{\Sigma\max} \le \sqrt{(y_{\max})^2 + (3\sigma_y)^2} \tag{A1.13}$$

where σ_y is mean square vibration displacement amplitude due to turbulent pulsations, for example, obtained in accordance with Equation (A1.9).

A1-5.6. FLUID-ELASTIC INSTABILITY

Excitation of structural vibrations due to fluid-elastic instability occurs when:

$$\bar{u} \geq \bar{u}^* \tag{A1.14}$$

where $\bar{u}^*$ is the critical fluid velocity.

In general, analysis of these vibrations can be carried out using CFD and structural dynamics methods described in Clause A1-5.1. A simplified calculation of the critical fluid velocity can be based on the following equation [3]:

$$\frac{\bar{u}^*}{f_1 d} = C\left[m\delta / \rho d^2\right]^a \tag{A1.15}$$

The use of this equation with a = 0.5 is recommended [2, 3]. Coefficient C (C ≥ 1) shall be obtained experimentally.

A1-6. CYCLIC STRENGTH ASSESSMENT FOR RVI UNDER VIBRATION LOADING

Input data for cyclic strength analysis at vibration loading are time history functions of vibration displacements, accelerations and stresses at RVI component points or spectra of these parameters obtained for the frequency band which defines requirements for strength assessment – see Section A1-5. If dynamic analysis results are in the form of kinematical parameters (vibration displacements and accelerations) then vibration stresses shall be calculated on the basis of quasi-static equilibrium equations for an RVI component under mechanical and kinematic loads.

Maximum stresses in RVI components (taking into account vibration loading) are obtained by the summation of maximum quasi-static values caused by weight, pressure, temperature and other slow time varying loads, and the maximum values of vibration stresses. Summation shall be carried out in accordance with the adopted methods [3].

Maximum stresses in RVI components (taking into account vibration loading) shall not exceed allowable values. Determination of allowable values and stress estimation for the stress category groups shall be carried out in accordance with Russian regulatory document [1].

Cyclic strength assessment for RVI components (taking into account vibration loading) is to be carried out excluding fatigue failure during the forthcoming operational period. Methods for cyclic strength assessment for RVI components can follow regulations given in [1], [2], [4] taking into account the application area of the method used.

A1-7. ASSESSMENT OF SELF-UNTIGHTENING OF BOLTS DUE TO VIBRATION LOADING

Self-loosening of bolts in RVI components due to vibration loading shall be estimated on the basis of experimental data obtained during operation of WWERs or from special tests.

A1-8. ASSESSMENT OF VIBRATION WEAR

Vibration wear at contact zones of movable and (or) weakened mechanical joints during micro-impacts caused by vibrations occurs, as a rule, according to the mechanism of fretting wear. For assessment of vibration wear the Archard theory [4] or the Blevins model [5] can be used.

The first model is based on the following assumptions:

— Wear volume V produced in a sliding distance L is proportional to the true area of contact;
— True area of contact is formed by local plastic deformation;
— Particles removed by the sliding motion are hemispherical and all have the same diameter.

The Archard wear equation derived for un-lubricated sliding surfaces due to fretting relates the worn volume to the normal force and the sliding distance as follows:

$$\Delta V = \frac{s\, F_N\, l}{K\, H} \tag{A1.16}$$

where

ΔV	is wear volume [m^3];
s	is wear coefficient, dimensionless;
F_N	is normal force on the contacting surfaces [N];
H	is hardness [N/mm^2];
l	is total sliding distance [m];
K	is shape factor and conversing constant, shall be determined experimentally.

Wear coefficient s is to be defined experimentally. Normal force on the contacting surfaces F_N shall be obtained on the basis of a structural vibration calculation using a non-linear approach that includes simulation of variable contact conditions.

The Blevins model is proposed for heat exchanger tubes; it allows the expression of fretting wear analytically via the following parameters:

— Frequencies and amplitudes of vibrations of the contacting structural parts;

— Gaps between the contacting parts;

— Reference structural dimensions;

— Load applied to the structure.

Expanding this model to RVI components, the mass loss within one vibration cycle can be described by the following equation:

$$\Delta m = \alpha_1 f^{\alpha_2} \left(\frac{A_C}{D} \right)^{\alpha_3} \left(\frac{A_G}{D} \right)^{\alpha} \left(\frac{t}{D} \right)^{\alpha_5} e^{\alpha_6 (R_S - W - P_L)} \tag{A1.17}$$

where all coefficients are empirical ones; their definition and determination follows [5].

A1-9. INSPECTION OF VIBRATIONS AND WEAR DURING OPERATION

A1-9.1. INSPECTION OF VIBRATIONS FOR DETERMINATION OF A MODEL OF VIBRATION LOADING

A model of vibration loading for RVI components is a mathematical form which represents rated and current values of vibration parameters during operation.

Rated values of vibration parameters for RVI components are established on the basis of theoretical (design) and experimental data, confirmed by start-up and adjustment measurements carried out on the large number of reactors in operation.

Current values of vibration parameters for RVI components are obtained on the basis of in-service inspections of their vibration parameters during outage or refueling, supported by the vibro-noise diagnostics system of the reactor.

The model of vibration loading for RVI components can be built e.g. in terms of acceleration as a function of the frequency *f* for each component under consideration, taking into account operation mode, operation time, etc.

A1-9.2. INSPECTION OF VIBRATION FOR DETERMINATION OF TECHNICAL CONDITION OF RVI COMPONENTS INCLUDING WEAR

The condition of frictional contact surfaces of RVI components which are subject to wear during operation is estimated on the basis of in-service inspections of their vibration parameters (dynamic characteristics, mainly frequencies). This is carried out by the vibro-noise diagnostic system of the reactor. If deflections in vibration parameters from their rated values are absent then the condition of frictional contact surfaces of RVI components is satisfactory.

If any deflections in vibration parameters of RVI components from the rated values are revealed, then analysis of the vibro-noise measurement data shall be carried out in order to determine the cause of these deflections.

If the cause of deflections in vibration parameters of RVI components cannot be discovered in this way, then it is needed to carry out television measurements of wear in frictional contact surfaces of RVI components. This is aided by the standard control system of the reactor. The measurement data is used for analysis of the technical condition of RVI components.

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APPENDIX D

RISK INFORMED IN-SERVICE INSPECTION

FOREWORD

Appendix D provides an overview of RI-ISI methodology and basic requirements on RI-ISI applications with special emphasis on WWER type piping systems and other passive component applications.

As documented in [1], over one hundred nuclear power plants have implemented some form of RI-ISI. These RI-ISI applications have ranged from small scope applications (e.g. reactor coolant pressure boundary), to medium scope applications, up to large scope applications (e.g. large portions of safety and non-safety piping, so called „full scope applications”).

Because of the flexibility of RI-ISI methodologies to address various scopes of application, plant owners including all WWER operators have the option to learn and apply this technology in a measured manner, reflecting the limited resources that may be available. In support of this, this Appendix reviews the different disciplines that are needed to develop and implement an RI-ISI programme (e.g. PSA, system engineers, ISI personnel). One conclusion of the report [1] is that most plants already have the basic infrastructure (PSA, system engineers, ISI personnel) necessary to develop and implement an RI-ISI programme.

As witnessed by the large number of plants that have implemented RI-ISI programs, RI-ISI has the ability to focus finite plant resources of those components that are most important from a public health and safety perspective. In addition, experience with already approved and implemented RI-ISI programmes has confirmed that focusing resources can reduce burden when compared to traditional deterministic ISI approaches. As such, it is recommended that plants that have not adopted an RI-ISI programme actively pursue developing and implementing such programmes. These programmes would be particularly useful during licence renewal and LTO of existing plants, and could potentially be beneficial for future plants.

1. ACRONYMS AND ABBREVIATIONS

ASME	American Society Of Mechanical Engineers
CC	Crevice cracking
CCDF	Conditional Core Damage Frequency
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CLERF	Conditional Large Early Release Frequency
CLERP	Conditional Large Early Release Probability
E-C	Erosion-cavitation
ECSCC	External chloride stress corrosion cracking
ENIQ	European Network for Inspection and Qualification
EPRI	Electric Power Research Institute
FAC	Flow-accelerated corrosion
FMEA	Failure Modes and Effects Analysis
FS	Flow Sensitive
FSAR	Final safety analysis report
HSS	High Safety Significance
IGSCC	Intergranular stress corrosion cracking
ISI	In-service inspection
LC	Localized Corrosion
LERF	Large Early Release Frequency
LOCA	Loss of coolant accident
LSS	Ligh Safety Significance
MIC	Microbiologically influenced corrosion
NDE	Non Destructive Evaluation
NDT	Non Destructive Testing
NPP	Nuclear Power Plant
NUREG	Nuclear Regulatory Commission Regulation
OL	Operating Loads
P&IDs	Pipe and instrumentation drawings
PIT	Pitting
POD	Probability of detection
PSA	Probabilistic Safety Assessment
PWR	Pressurized water reactor
PWROG	PWR Owners' Group
PWSCC	Primary water stress corrosion cracking
R&D	Research and Development
RAW	Risk Achievement Worth

RCPB	Reactor Coolant Pressure Boundary
RI-ISI	Risk-informed in-service inspection
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
RT	Radiographic Testing
SCC	Stress corrosion cracking
SRM	Structural Reliability Modelling
TASCS	Thermal stratification cycling and striping
TC	Technical Cooperation
TF	Thermal fatigue
TGR	Task Group on Risk
TGSCC	Transgranular stress corrosion cracking
TVO	Teollisuuden Voima Oy
US NRC	Nuclear Regulatory Commission
WWER	Water-Cooled, Water-Moderated Energy Reactor

2. INTRODUCTION

During the design phase of the first nuclear power plants, it was believed that the high standards used to design and fabricate passive components would allow for problem-free operation throughout their lifetime. When such plants became operational, it was discovered that components still degraded over time despite such high design standards. The industry thus began to develop inspection programs. ASME developed a standard, the ASME Boiler and Pressure Vessel Code, Section XI: Rules for Inservice Inspection of Nuclear Power Plant Components. This standard initially provided rules for inspection of Class 1 systems only, but later included rules for the inspection of Class 2 and 3 systems. Due to the sweeping breadth of these rules, nuclear plants were obliged to devote significant manpower, radiation exposure, and financial resources to examine many locations, some with low failure potential and/or little safety significance.

In the past, while Section XI Inservice Inspection was based on a sampling approach, WWER operating countries followed Russian rules and standards, and utilized the experience of designers and equipment manufacturers. The traditional Russian approach was based on hydrostatic pressure tests, surface NDE methods, and only partially on radiographic testing (RT) due to the lack of other reliable NDT volumetric methods. A too high reliance on the experience of manufactures led to partially non-systematic development of ISI programs, as different manufacturers and designer groups applied their specific philosophies, taking into account their own practical manufacturing rules. Experience not integrated into an overall ISI programme approach. ISI programmes developed in such a way became mandatory ISI programs approved by the Regulatory Bodies of individual countries. For this reason there were no generally accepted sampling rules for piping systems, and variations in the percentage and locations of piping welds selected for examination would occur. In recent years, this situation has become more divergent in different countries. In some countries there are ISI programs that strictly follow Russian standards, while in other countries programs follow new national standards and approaches that take into account worldwide experience and different standards.

Traditional ISI requirements looked for generic degradation. Industry experience has shown that degradation is typically not a random occurrence. Degradation occurs where the conditions necessary for a particular mechanism exist. Over time a better understanding of degradation typically found in systems at a nuclear power plant has been developed. A potential degradation mechanism can be assigned to those locations where appropriate conditions may exist. Thus locations that have a higher failure potential can be targeted.

Risk informed technology allows plants to inspect those systems or portions of systems that are most risk important. Risk-informed in-service inspection (RI-ISI) reflects recent developments in Probabilistic Safety Assessment (PSA) technology, structural reliability, as well as experience gained from nearly 13,000 reactor years operating experience of Nuclear Power Plants.

This Appendix D introduces the general approach of RI-ISI technology, discusses related issues, and describes on-going research developments for new plants. The objective of this document is to provide guidance on RI-ISI technology. In Section 3 a generic approach is given to RI-ISI. Section 4 describes process and organizational issues.

3. GENERAL APPROACH TO RI-ISI

3.1. TECHNICAL PERSPECTIVE

An overview of the fundamental aspects of most RI-ISI methodologies is depicted in Figure 1. This figure reflects the basic technical elements of the risk-informed concept as relevant to developing an ISI programme. From a technical perspective, the following main steps summarize a typical RI-ISI process:

- Definition of RI-ISI programme scope;
- Collection and analysis of input data required;
- Evaluation of piping failure consequence;
- Identification and evaluation of piping failure potential;
- Risk ranking;
- Inspection element selection;
- Evaluation of risk impact of changes to inspection programme;
- Long term management of the RI-ISI programme.

Each step of the process is further discussed in the following sub-sections.

3.1.1 Definition of RI-ISI programme scope

The first step is to decide on the scope of the RI-ISI programme. The scope of an RI-ISI programme is subject to the goals of the plant operator as well as feedback received from the regulatory authority. Options include:

- Large scope applications including all safety class piping systems (e.g. ASME Class 1, 2 and 3) and non-safety classified piping;
- Selection of individual piping system(s) (e.g. reactor coolant system), or alternative piping system scope (e.g. ASME Class 1 only);
- A subset of a class of piping systems (see sections 3.1 and 5.1 of Ref [1]).

Selection of the RI-ISI scope may strongly influence the results, and therefore the process used to define the scope should be properly understood by both the operator and regulator. Any system not selected for inclusion in the RI-ISI programme scope is retained within the current (deterministic) in-service inspection programme for existing plants.

An additional decision has to be made on whether piping systems and degradation mechanisms covered within the plant's other inspection programs, if any, will be incorporated into the RI-ISI programme. Examples of other inspection programs that may or may not be incorporated into the RI-ISI programme include intergranular stress corrosion cracking (IGSCC), localized corrosion (e.g. MIC), and flow accelerated corrosion (FAC).

3.1.2. Collection and analysis of input data required

The process of RI-ISI brings together a large amount of information from many different sources, which needs to be collected and analysed. This information can be classified in the following five categories:

- Equipment data;
- Plant operating data;
- General nuclear industry information;
- Safety Analysis Report and technical specifications;
- PSA data.

Data collection is an essential part of the RI-ISI process, as it constitutes the basis for the whole analysis and decision process. Data collection is likely to be a resource-demanding phase in the RI-ISI process, however the data gathered should be of considerable value for many safety or reliability related activities.

3.1.3. Evaluation of piping failure consequence

Consequence analysis is normally performed on a system-by-system basis and leads to the preliminary definition of piping segments.

The term „piping segment" may have different meaning for different RI-ISI methodologies and can include: common potential for failure, common consequence potential, or both. In practice a segment includes a region of the pressure boundary for which a failure would lead to the same consequences. Later, this classification can be refined to take into account insights from failure potential analysis.

Consequences of pipe rupture are typically measured in terms of the conditional probability of core damage given a pipe rupture (CCDP) and the conditional probability of large early release given a pipe rupture (CLERP). These measurements require quantitative risk estimates which would be obtained from the plant specific PSA models. This is accomplished by identifying the impacts of the pipe rupture in terms of initiating events, system mitigation and containment response.

The use of PSA in RI-ISI is further discussed in section 5.3 of Ref [1].

3.1.4. Identification and evaluation of piping failure potential

The first step in the assessment of the probability of failure of a structural element or segment is the identification of the potential degradation mechanisms. This requires the qualitative evaluation of a range of influential parameters, such as, design and fabrication information, loadings, environmental conditions, and inspection results. This analysis should be supported with a review of operating experience from the plant, its sister units, similar plants, as well as insights from world-wide generic data. Such an analysis phase is very important in order to correctly classify or quantify failure potential.

Piping failure potential can be assessed in different ways, ranging from purely qualitative assessment to quantification, with either statistical analysis of service data or structural reliability models.

If the evaluation is carried out at a segment level, the initial consequence based segmentation could be refined at this stage to take into account differences in the degradation mechanisms, or in the severity of the failure potential.

A discussion on the identification of potential degradation mechanisms can be found in section 5.4 and 5.6 of Ref [1]. The use of qualitative versus quantitative pipe failure potential is further discussed in section 5.5. of Ref [1].

3.1.5. Risk ranking

The segments or structural elements are ranked according to their associated risk, which is determined from their failure potential and the severity of failure consequences. The ranking or categorization criteria can be expressed as thresholds in terms of CDF and LERF or pertinent importance measures such as risk reduction worth and risk achievement worth. Each segment is placed onto the appropriate place of a risk characterization matrix or plot. See examples in Figure 1 and Figure 2.

Pipe elements within each segment are candidate locations to be selected for the inspection programme based on risk characterization of the segment to which each element belongs.

3.1.6. Inspection element selection

In this step, the revised set of inspection requirements is defined. Specific locations are selected for the inspection programme based on the segment’s risk ranking and a set of practical considerations that bear on the feasibility and effectiveness of the specific inspection. The number of locations selected for inspection would be a function of the RI-ISI methodology selected for use. For those locations selected for NDE inspections, inspections are focused on the type of degradation mechanism identified earlier. The ability to focus the examination on specific damage mechanism(s) enhances the effectiveness of the retained inspections. All locations, regardless of risk classification and element selection results are typically also subjected to pressure and leak testing requirements.

3.1.7. Evaluation of risk impact of changes to inspection programme

This step deals with showing the impact of the proposed RI-ISI on plant safety (i.e. the change in risk). It should be confirmed that the initial selection of elements for the RI-ISI programme does not produce an unfavorable and unacceptable risk impact. Depending upon the RI-ISI application and its results, qualitative criteria, bounding estimates of risk impacts, or realistic estimates of risk impacts may need to be developed. If unacceptable risk impacts do occur, then adjustments to the selection of elements to meet the risk acceptance criteria may be needed.

The RI-ISI methodology used and the accompanying acceptance criteria will most likely be determined by the plant and its respective regulator. As an example, when using USNRC Regulatory Guide 1.174[R1], it must be shown that the changes in risk due to changes in the inspection programme do not pose a significant risk impact as determined by changes in CDF or LERF. In the Czech Republic, a Safety Guide has been developed for risk-informed applications. While similar to Reg Guide 1.174, it does have slightly different acceptance criteria. Other countries may have different acceptance criteria. This is further discussed in section 5.10 of Ref. [1].

3.1.8. Long term management of a RI-ISI programme

The final step of the process is to document the RI-ISI programme and implement monitoring strategies, so that the programme is managed on a long term basis. The frequency and content of these updates will most likely be agreed upon by the plant and its regulator. As time passes these updates may need to be more frequent, if dictated by updates to the PSA or as new degradation mechanisms are identified.

Additionally, as changes to plant design are implemented, changes to the inputs associated with RI-ISI programme segment definition and element selections may occur. It may be important to address such changes to the inputs used in any assessment that may affect resultant pipe failure potentials used to support the RI-ISI segment definition and element selection. Some examples of these inputs would include:

— Operating characteristics (e.g., changes in water chemistry control);

— Material and configuration changes;

— Welding techniques and procedures;

— Construction and pre-service examination results;

— Stress data (operating modes, pressure, and temperature changes).

In addition, plant design changes could result in changes to a plant's CDF or LERF, which in turn could result in a change in consequence of failure for system piping segments.

			Conditional Consequence				
			Very Low	Low	Medium	High	Very High
			$<10^{-6}$	10^{-6}-10^{-5}	10^{-5}-10^{-4}	10^{-4}-10^{-3}	$>10^{-3}$
Failure potential	Very High	$>10^{-4}$					
	High	10^{-5}-10^{-4}					
	Medium	10^{-6}-10^{-5}					
	Low	10^{-7}-10^{-6}					
	Very Low	$<10^{-7}$					

FIG. 1. Example of Risk matrix. The qualitative and quantitative values are for illustrative purposes only.

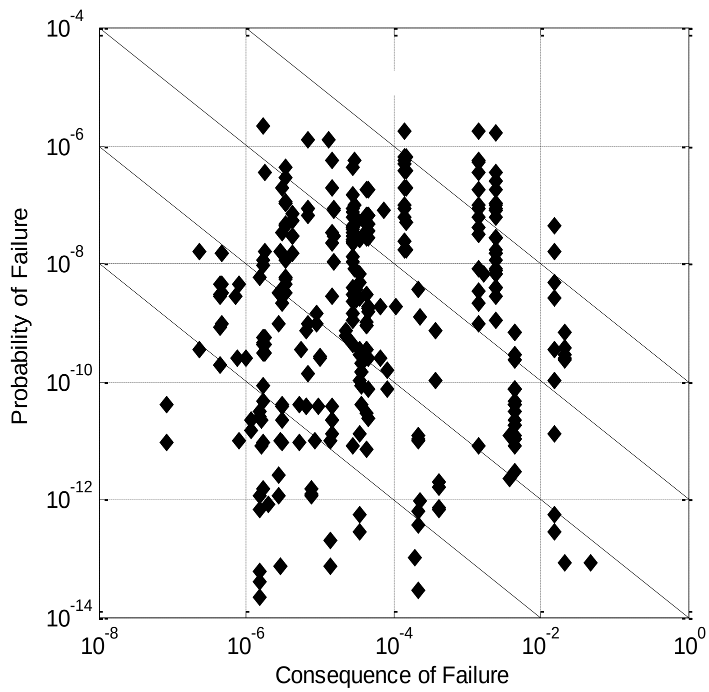

FIG. 2 Example of Risk Plot.

4. RI-ISI IMPLEMENTATION PROCESS AND ORGANIZATION

Implementation of a successful RI-ISI programme requires the use and integration of inputs and analyses from multiple disciplines within the plant organization. This section discusses the various required inputs and provides suggestions for documenting these inputs for use in developing the RI-ISI programme, as well as supporting future updates and improvements.

4.1. ORGANIZATION

The overall responsibility for the RI-IS programme lies with the plant operator (licensee). It is recognised that utilities in different countries will have different organizational structures. The following provides an example of an organizational structure and interfaces that can be used as a guide to what is required in order to implement a risk-informed ISI programme. A similar structure has been suggested in the ENIQ Framework Document on RI-ISI.

With reference to Figure 3, the main parties/personnel involved are described in the following:

- RI-ISI Responsible Person:

  The RI-ISI Responsible Person is responsible for the development and acceptance of the final RI-ISI programme.

- RI-ISI Team:

  The RI-ISI team has the responsibility of developing the RI-ISI programme and following it through to implementation.

- RI-ISI Review Panel:

  The purpose of the Review Panel is to provide an independent review function of the RI-ISI programme including supporting evaluations and calculations.

- Inspection Qualification Team:

  The Inspection Qualification Team has the responsibility of identifying appropriate inspection locations, inspection techniques and qualification procedures for the proposed ISI Programme.

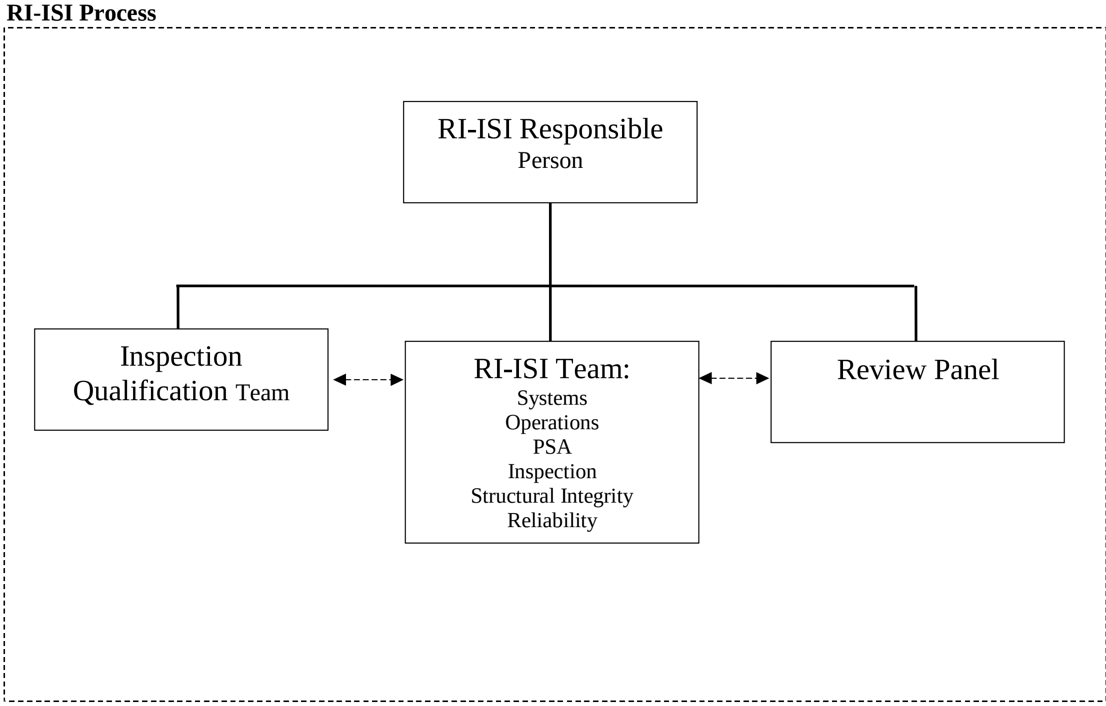

FIG. 3. Possible organisational structure for the RI-ISI process.

4.2. PLANT INFORMATION REQUIREMENTS

The evaluation process begins by assembling information based on the scope of the application. This information is necessary for defining the systems to be analyzed and for identifying the elements to be evaluated within the system boundaries. The information and support that is needed for the evaluation process is summarized as follows:

- Evaluation Documentation:
  - ISI programme plan;
  - ISI isometrics;
  - P&IDs (for applicable systems);
  - Spatial databases;
  - Flooding/spray studies;
  - System training manuals;
  - PSA Information;
  - Material specifications;
  - Line lists;
  - Valve lists;
  - System design basis documents;

  - Inspection programs for active degradation (e.g. FAC, IGSCC);
  - Plant service experience (i.e., cracking, water hammer, FAC event history, etc.);
  - Normal, abnormal and emergency operating procedures;
  - Alarm response procedures.
- Evaluation Personnel Support:
  - ISI personnel;
  - PSA personnel;
  - Operations personnel;
  - System/design engineers;
  - Material engineers.
- Miscellaneous:
  - Cost drivers (i.e., which systems, locations, etc.);
  - Worker exposure (i.e. hot spots, time consuming examinations);
  - Walkdown access.

4.3 DOCUMENTATION

Documentation of the RI-ISI application involves two general requirements:

- Documentation of the risk-informed evaluation, and
- Documentation of the revised ISI programme.

This section discusses those attributes that will be useful in providing the necessary documentation. As ISI programs for all plants must meet their country's specific documentation requirements, the information below is provided as illustration.

Two sets of documentation are normally required. The first set involves developing the RI-ISI application and the second set involves regulatory review and approval.

The first set of documentation is required to develop the analysis, to support the plant review of the RI-ISI evaluations, as well as to provide a means for future re-creation and/or modification of the risk-informed application. This type of documentation tends to consist of marked-up drawings and supplemental calculations that will need to be retained in hard copy, microfilmed/fiched or scanned media.

The second set of documentation is the actual submittal to the regulatory body and subsequent regulatory approval.

Both sets of information would generally be expected to be retained and be retrievable on site.

4.3.1. RI-ISI evaluation documentation

This documentation is required to support the development and review of the RI-ISI application and to provide a means of re-creating and/or modifying the results as part of maintaining a living RI-ISI programme. This information generally consists of marked-up drawings and supplemental calculations that need to be retained in hard copy, microfilm or scanned media. It would typically contain the following types of information:

- Scope:
  - Definition;
  - Basis for scope definition.
- Piping System Configuration:
  - Current ISI programme;
  - ISI isometric drawings;
  - Piping and Instrumentation Diagrams (P&ID);
  - Piping design specification;
  - Material and fabrication specification;
  - Inspection cost data.
- Failure Potential Evaluation:
  - Degradation mechanisms identified;
  - System training manuals;
  - Design basis documents;
  - Operating conditions (e.g. operating temperatures, water chemistry);
  - Plant-specific and industry service experience;
  - Segment properties (e.g. material, insulation).
- Consequence Evaluation:
  - PSA analyses;
  - FSAR;
  - Spatial studies (e.g. internal flooding, pipe whip, spray);
  - Spatial databases (e.g. component locations, plant layout);
  - Results of direct and indirect effects analysis (e.g. initiating events, mitigating equipment failures).
- Risk Evaluation:
  - Consequence and failure potential results;
  - Risk Ranking results;
  - Expert Panel results.

- Element Selection:
  - Number and location;
  - Examination methods and qualification requirements;
  - Previous ISI programme;
  - Integration with other inspection programs.
- Risk Impact:
  - Evaluation of results;
  - Comparison with acceptance criteria.

4.3.2. Regulatory Submittal Documentation

The results of the RI-ISI analysis may be required to be submitted to the plant's regulatory body for approval. Although each regulatory body may require different information, the following list provides an example of the information contained in a so called „Template Submittal" that has been used:

- Justification statement that PSA is of sufficient quality;
- Summary of risk impact;
- Summary of all inspection programs are impacted;
- Revised FSAR pages impacted by the change, if any;
- Process followed, and exceptions to methodology, if any;
- Summary of results of each step (e.g., number of segments in each risk category, number of locations to be inspected, etc.);
- Compliance with regulatory guides and principles (or any exceptions);
- Summary of changes from current ISI programme.

5. RI-ISI REVIEW AT THE REGULATOR

Each country will need to determine how and what must be reviewed by the regulatory authorities, depending upon the ISI requirements in each regulatory framework. Early and close cooperation between licensees and the regulatory body is recommended to effectively and efficiently implement RI-ISI in each country. Prior agreement of several fundamental principles (scope of the application, risk acceptance criteria, applicability of other inspection activities) should be achieved before significant plant-specific analyses are attempted. Once these agreements are in place, the plant operator and regulator can determine the appropriate review and approval process.

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SUPPLEMENT 1 TO APPENDIX D

RISK INFORMED SELECTION PROCESS-- METHOD A

S1-1. INTRODUCTION AND SCOPE

This Supplement provides a risk-informed selection process to be used for selection of piping segments and piping structural elements (including connections) for preservice and inservice inspection.

S1-2. EXPERT PANEL REQUIREMENTS

S1-2.1. GENERAL

Each Owner shall establish an expert panel to implement the risk-informed selection process described in this Supplement. The expert panel shall be indoctrinated in the specific requirements to be used under this risk-informed selection process. Risk analysis techniques shall include the use of applicable risk-importance measures, threshold values, failure probability models, failure mode and degradation mechanism assessments, and the use of expert judgment. Each of these techniques shall be covered in the indoctrination to the extent necessary to provide the expert panel with a level of knowledge needed to adequately evaluate and approve the scope of the risk-informed selections.

S1-2.2. EXPERT PANEL FORMATION

Panel members selected for this risk-informed selection process shall include members of the expert panel established to implement other Probabilistic Risk Assessment (PRA) applications such as those associated with maintenance, quality assurance, or inservice testing activities, if such a panel was used. The panel for this risk-informed selection process shall include individuals having expertise in the following fields:

(a) Probabilistic safety assessment;

(b) In-service examination;

(c) Nondestructive examination;

(d) Stress and material considerations;

(e) Plant operations;

(f) Plant and industry maintenance, repair, and failure history;

(g) System design and operation.

The Owner shall define and document quorum requirements. Members may be experts in more than one field, but the Owner shall consider the diversity of the panel make-up, avoiding heavy reliance on any one member's judgment. The Owner is responsible for ensuring adequate experience levels for each expert panel member. This experience shall be documented and maintained by the Owner.

S1-2.3. EXPERT PANEL LEADER SELECTION

The Owner shall select a panel leader who is familiar with the requirements of this risk-informed selection process. The panel leader shall facilitate the panel activities and shall be responsible for ensuring accomplishment of this risk-informed selection process.

S1-2.4. EXPERT PANEL RESPONSIBILITIES

The expert panel shall be responsible for evaluation and approval of all risk-informed selection results (i.e., system, segment, structural element, and inspection selections) by utilizing their expertise (including knowledge of plant operation, prior inspection results, industry data, and any available stress and fracture mechanics results) and PRA insights to make the final decision on the High Safety Significant (HSS) structural elements to be included for inservice inspection. Selections made in accordance with this process, or any other required input where the judgment of the expert panel is needed, shall be reached by consensus. The expert panel shall be provided documentation to support a decision making process based on a complete description of the functions endangered by the failure of piping within the scope and the operating conditions of the piping. The documentation should include the functions performed by the system or system parts included in the scope, the degradation mechanism identified in the system, the operator recovery actions credited in the analysis, and all PRA results.

S1-2.5. MAINTENANCE OF THE EXPERT PANEL

The Owner shall maintain the expert panel to allow changes, as necessary, to the risk-informed selections, when new information is applied, as directed by the requirements of this Supplement. Members may be added or removed as needed as long as the requirements of S1-2.2 are satisfied.

S1-2.6. PFM USER TRAINING AND QUALIFICATION.

To ensure that input parameters are consistently assigned and the Probabilistic Fracture Mechanics (PFM) model used in this methodology is properly executed, users of the PFM model shall be trained and qualified. Acceptable qualification and the scope of training is to be determined by the Owner based on the background and experience of the individuals using the model. Qualification should cover the following topics:

(a) Overall risk-informed inspection process;

(b) How PFM-calculated probabilities are used in the piping segment risk calculations;

(c) Capabilities and limitations of the PFM model;

(d) Expertise and type of information required, including applicable sources;

(e) How potential degradation mechanisms are considered and combined;

(f) The importance of each input parameter on each degradation mechanism and failure mode;

(g) Examples of PFM model use for different degradation mechanisms and failure modes;

(h) How detailed PFM input (e.g., uncertainty) is developed and used.

S1-3. BOUNDARY REQUIREMENTS

S1-3.1. BOUNDARY IDENTIFICATION

(a) The operator shall define the system boundaries included in the scope of the risk-informed inspection programme evaluation. Within each system boundary, the risk-informed evaluation may include Class 1, 2, or 3 piping defined in the deterministic inservice inspection programme, if applicable, and piping outside the current deterministic programme examination boundaries, if applicable. Piping, or portions thereof, included for evaluation shall be based on the deterministic programme Class 1, 2, or 3 examination boundaries, if applicable.

(b) Piping, or portions thereof, within the Class 1, 2, or 3 boundaries, if applicable, and known from PRA insights to have a high consequence contribution, shall be included.

S1-3.2. USE OF THE APPLICABLE PRA.

The boundary requirements of chapter S1-3.1 shall be used to identify the piping systems, or portions thereof, to be considered for risk-informed selections of HSS and Low Safety Significant (LSS) piping segments and piping structural elements in accordance with this process. The Operator's PRA and its evaluated safety functions, which consist of core damage protection, large early release protection, and the risk measures associated with these safety functions (core damage frequency and large early release frequency), provide the necessary information for the piping system PRA boundaries to be used in this process.

S1-4. RISK-INFORMED PROCESS

S1-4.1. GENERAL

The risk-informed selection of nuclear power plant piping segments and piping structural elements shall be performed using the process described in this Supplement. The final result of this process is to identify those HSS piping structural elements that will be examined in accordance with Table S1-1.

TABLE S1-1. EXAMINATION CATEGORIES

item	Parts examined	Examination requirement	Examination method	Extent and frequency for 1st interval	Extent and frequency for successive interval
R0	High safety significant piping structural elements				
R1	Elements subject to thermal fatigue	Note (1)	Volumetric + notes (8), (9)	Element Note (2),(4)	same as 1st
R2	Elements subject to high cycle mechanical fatigue		Visual VT2 + notes (8), (9)	each refueling	same as 1st
R3	Elements subject to erosion cavitaion	Note (6)	Volumetric + notes (7)	Element Note (2)	same as 1st
R4	Elements subject to crevice corrosion cracking	Note (5)	Volumetric + notes (8), (9)	Element Note (2)	same as 1st
R5	Elements subject to PWSCC	Note (1)	Volumetric + notes (8), (9)	Element Note (2),(4)	same as 1st
R6	Elements subject to IGSCC or TGSCC	Note (1)	Volumetric+notes (7-9)	Element Note (2),(4)	same as 1st
R7	Elements subject to localised MIC or pitting corrosion		VT3+Vol. notes (6-7)	Element Note (2)	same as 1st
R8	Elements subject to FAC	Note (7)	Note (7)	Note (7)	Note (7)
R9	Elements subject to ECSCC		Surface	Element Note (2)	same as 1st
R10	Elements not subject to a degradation mechanism	Note (1)	Volumetric + notes (8), (9)	Element Note (2),(4)	same as 1st

Legend:

PWSCC primary water stress corrosion cracking

IGSCC/TGSCC intergranular/transgranual stress corrosion cracking

MIC microbiologically influnced corrosion

FAC flow accelerated corrosion

ECSCC external chloride stress corrosion cracking

Notes:

a. The length of the examination volume shall be increased by enough distance (approx. 13 mm) to include each side of the base metal thickness transition or counterbore.

b. Includes examination locations and Class 1 weld examination requirement figures applicable to Class 1, 2,3 or Non-Class welds identified in accordance the risk-iformed selection process described in Supplement 1 or 2.

c. Includes 100 % of the examination location. When the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Acceptance of limited examinations or volumes shall not invalidate the results of the risk-informed evaluation. Areas with acceptable limited examinations, and their bases, shall be documented.

d. The examination shall include any longitudinal welds at the location selected for examination in note (2) for the intersecting circumferential welds.

e. The examination volume shall include the volume surrounding the weld, weld heat affected zone (HAZ), and base metal as applicable, in the crevice region. Examination shall focus on detection of cracks iniatitng and propagating from the inner surface.

f. The examination volume shall include base metal, welds and weld heat affected zone (HAZ) in the affected regions of carbon or low alloy steel and the welds and weld HAZ of austenitic steel. Examinations shall verify the minimum wall thickness required. Acceptance criteria for localized thinning are in course of preparation. The examination method and examination region shall be sufficient to characterize the extent of the element degradation.

g. In accordance with the Owner’s existing augmented programme such as IGSCC, MIC or FAC programmes, as applicable.

h. Sockets welds of any size and branch pipe connection welds DN 50 and smaller require only VT-2 visual examination.

i. VT-2 visual examinations shall be conducted during a system pressure test or a pressure test specific to that element or segment and shall be performed during each refueling outage or at a frequency consistent with the time between refueling outages.

The basic overview of this process is provided in Figure S1-1.

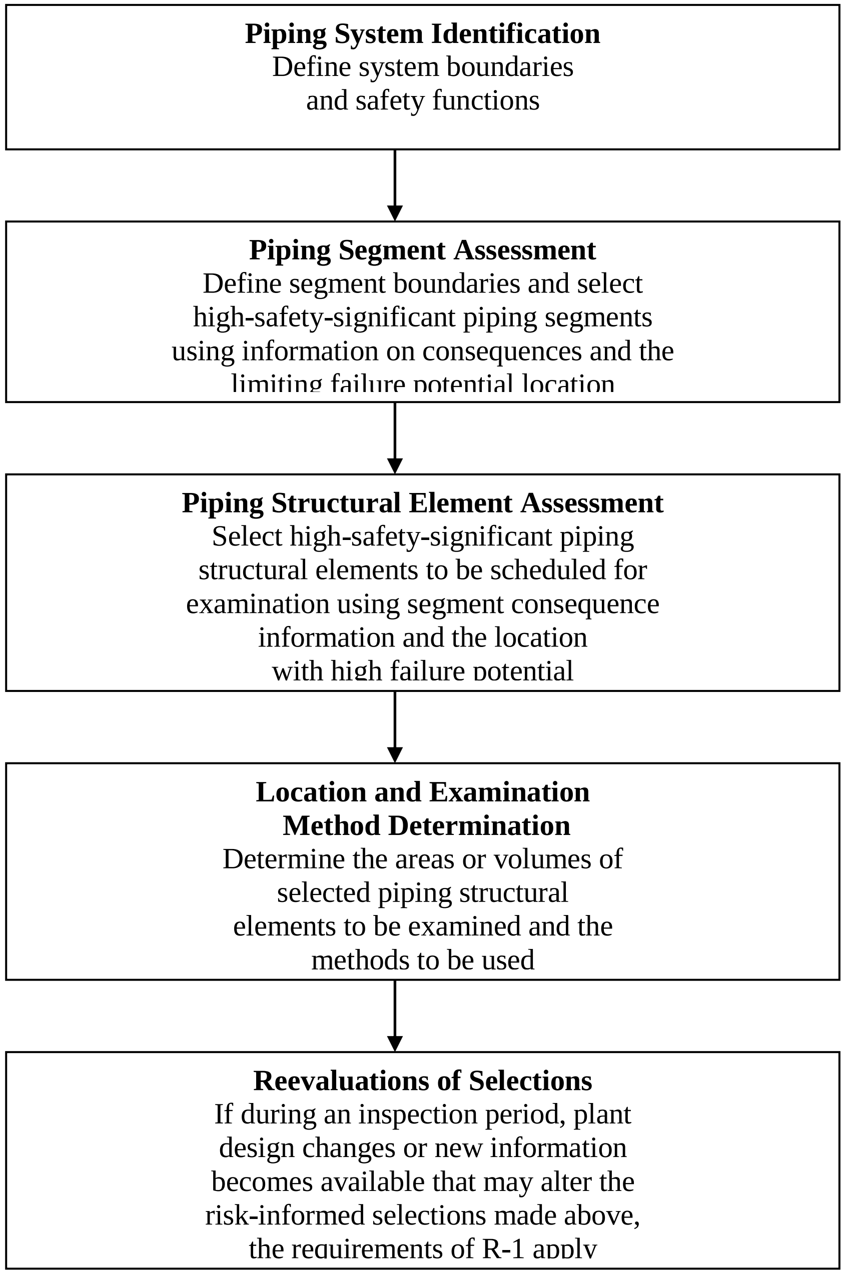

FIG. S1-1. Overview risk-informed selection process.

S1-4.2. QUANTITATIVE APPROACH

S1-4.2.1. General

The process for this quantitative approach uses risk-based ranking calculational methods with established threshold values, and risk-informed considerations of operational and deterministic insights to select a final list of HSS piping structural elements to be included in a risk-informed inspection programme. This approach is divided into four major tasks.

(a) Identify and define the piping system boundaries and portions that will be considered in this risk-informed selection process in accordance with the boundary requirements of chapter S1-3.

(b) Define, calculate, rank, and select the HSS piping segments within these identified systems using the failure modes and effects analysis (FMEA) technique and relevant plant information, including the plant PRA results.

(c) Assess or calculate, rank, and select the HSS piping structural elements, such as welds, elbows, and tees, within the HSS piping segments mat will form the risk-informed inspection programme for piping.

(d) Determine the areas or volumes of the selected piping structural elements to be scheduled for examination, and the appropriate examination methods or monitoring techniques to be used.

S1-4.2.2. Risk Importance Measures

Risk Reduction Worth (RRW) shall be used as the primary risk importance measure in this risk-informed selection process. In addition, the Risk Achievement Worth (RAW) importance measure shall be considered in accordance with S1-4.2.6(b)(1). Four RRW and RAW values shall be calculated for each segment, two for CDF and two for LERF. One calculation of RRW and RAW based on CDF and on LERF shall be performed assuming all reasonable recovery actions to isolate the failed segment and mitigate the spatial effects are successful. A second calculation of RRW and RAW based on CDF and on LERF shall be performed assuming that no recovery actions are performed to isolate the failed segment and mitigate the spatial effects. RRW and RAW are used in failure consequence calculations, as discussed in S1-4.2.2(a) and (b).

(a) Risk Reduction Worth. RRW indicates the reduction factor in risk if the piping is assumed perfectly reliable for all failure modes. The RRW is calculated by reevaluating the PRA model and substituting a value of zero for the unavailability for all modeled components that would be placed in a failed state if the segment failed for each piping segment or structural element of interest. Thus, RRW is represented as follows:

$$RRW = R_o / R_i \qquad \text{(S1.1)}$$

where

Ri is decreased risk level (total core damage frequency or large early release frequency from piping pressure boundary failures) with the component *i* assumed to be perfectly reliable;

R_0 is base risk level (total core damage frequency or large early release frequency from piping pressure boundary failures only).

Fussell-Vesely. Fussell-Vesely (F-V) importance may be used in lieu of RRW because of the mathematical relationship between the measures. The following relationship allows translation of F-V results to RRW if the F-V is less than 0.1:

$$RRW = \frac{1}{[1-(F-V)]} \tag{S1.2}$$

(b) *Risk Achievement Worth.* RAW indicates the increased factor in risk if the piping is assumed failed for all failure modes. The RAW is calculated by reevaluating the PRA model and substituting a value of unity for the unavailability for all modeled components that would be placed in a failed state if the segment failed belonging to the piping segment of interest. Thus, RAW is represented as follows:

$$RAW = R_i^{+} / R_o \tag{S1.3}$$

where

R_i^{+} is increased risk level (core damage frequency or large early release frequency from piping pressure boundary failures) without component *i*, or with component i assumed failed;

R_0 is base risk level (core damage frequency or large early release frequency from piping pressure boundary failures only).

S1-4.2.3. Selection of Systems

The expert panel shall determine, from the boundary requirements of chapter S1-3, the systems and portions thereof that will be considered in this risk-informed selection process. The final system list, along with the rationale for any decisions, including those affected by other PRA application considerations, such as risk significance determinations, shall be documented.

S1-4.2.4. Piping Segment Risk Ranking and Selection

The selected systems (as identified in S1-4.2.3) shall be further evaluated at the piping segment level. The ranking process is discussed in S1-4.2.5 and S1-4.2.6. The ranking process shall consider the calculated conditional CDF and conditional LERF determined by the evaluation of piping pressure boundary failures. Four calculations shall be performed, two for CDF and two for LERF. One calculation of CDF and LERF shall be performed assuming all reasonable recovery actions to isolate the failed segment and mitigate the spatial effects are successful. A second calculation of CDF and LERF shall be performed assuming that no recovery actions are performed to isolate the failed segment and mitigate the spatial effects. The following calculations shall be applied as applicable:

(a) *Expanded Equations For Use in Risk Evaluation*

For a given segment,

$$CDF = FP(FR)_{leak} * CCDF(CCDP)_{leak} + FP(FR)_{disabling\ leak} * CCDF(CCDP)_{disabling\ leak} + FP(FR)_{break} * CCDF(CCDP)_{break} \tag{S1.4}$$

where

$FP(FR)_{leak}$	is probability (dimensionless) or rate (per yr) for the failure mode of small leaks;
$FP(FR)_{disabling\ leak}$	is probability (dimensionless) or rate (per yr) for the failure mode of disabling leaks;
$FP(FR)_{break}$	is probability (dimensionless) or rate (per yr) for the failure mode of breaks;
$CCDF(CCDP)_{leak}$	is conditional core damage frequency or core damage probability given a small leak;
$CCDF(CCDP)_{disabling\ leak}$	is conditional core damage frequency or core damage probability given a disabling leak
$CCDF(CCDP)_{break}$	is conditional core damage frequency or core damage probability given a break.

Similar calculations apply to the calculation of LERF for piping segments.

(b) Initiating Event Consequence CDF Calculations:

$$CDF_{PB} = FR_{PB} * CCDP_{IE} \qquad \text{(S1.5)}$$

where

CDF_{PB} is CDF from piping failure (events/yr);

FR_{PB} is piping failure rate (no deterministic inservice inspection) (events/yr);
$CCDP_{IE}$ is conditional core damage probability (dimensionless)

$$FR_{PB} = FP_{E0L} / EOL \qquad \text{(S1.6)}$$

where

FP_{EOL} is failure probability at end of life (EOL).

(c) Mitigating System Consequence CDF Calculations:

$$CDF_{PB} = FP_{PB} * CCDF_{PB} \qquad \text{(S1.7)}$$

where

CDF_{PB} is the Core Damage Frequency from a piping failure (in events/yr);
$CCDF_{PB}$ is Conditional CDF with segment failed (=1) (in events/yr);
FP_{PB} is piping failure probability (dimensionless).

$$CCDF_{PB} = CDF_{PB=1} - \mathrm{CDF}_{\mathrm{BASE}} \qquad \text{(S1.8)}$$

where

$CDF_{PB=1}$ is new total plant CDF with surrogate component = 1 (in events/yr)
$\mathrm{CDF}_{\mathrm{BASE}}$ is base total plant CDF (events/yr).

(1) Continuously Operating Systems

$$FP_{PB} = FR_{PB} * T_m \tag{S1.9}$$

where

FR_{PB}	is the failure rate (in events per unit time);
T_m	is the total defined mission time (24 hr for most PRAs).
FR_{PB}	$= FP_{E0L} /(EOL$ yr* 8760 hr/yr)

(2) Standby Systems

$$FP_{PB} = 1/2\ (FR_{PB})\ T_t + (FR_{PB})\ T_m \tag{S1.10}$$

where

FR_{PB}	is the failure rate (in events per unit time);
T_t	is the interval between tests that would identify a piping failure;
T_m	is the total defined mission time (24 hr for most PRAs).

(d) Initiating Event and System Degradation Consequence CDF Calculations:

$$CDF_{PB} = FR_{PB} * CCDP_{IE,SEG=1} \tag{S1.11}$$

where

CDF_{PB}	is the Core Damage Frequency from a piping failure (events per yr);
$CCDP_{IE,SEG=1}$	is conditional core damage probability for the initiator with mitigating system component assumed to fail (initiating event and mitigating system component =1);
FR_{PB}	is piping failure rate (in events per yr)

S1-4.2.5. Calculate Piping Segment Risk Importances

The FMEA technique shall be used to rank piping segments within the selected systems on the basis of core damage frequency and large early release frequency. Relevant plant information used for initial formulation of the FMEA shall be realistic and shall reflect current plant operational practices. The FMEA technique shall include at least the following information.

(a) *Piping Segment.* A location and boundary description of the segment that includes consideration of a number of structural elements being evaluated. Piping segment typically consist of welds, may include other elements such as elbows, flow reducers, etc.

(b) *Degradation Mechanism.* Identification of the full range of potential degradation mechanisms, such as mechanical fatigue, thermal fatigue, stress corrosion cracking, and flow accelerated corrosion (FAC), mat may occur within the piping segment, and the identification of the particular structural elements where these failures are most likely to occur.

(c) *Failure Probability.* Estimates of the failure probability of a piping segment under consideration assuming no inservice inspection. Failure rates (on demand, per hour, or per year) are required inputs to the risk-importance calculations. The piping segment failure rate is analogous to the active component failure rates that are used in the PRA, where the rate is the number of observed failures divided by the number of years.

TABLE S1-2. DEFINITION OF FAILURE PROBABILITY ESTIMATES FOR PIPE SEGMENTS

Definition	**Failure Probability (per year)**
An event that individually may be expected to occur more than once during the lifetime of the pipe segment.	10^{-1}
An event that individually may be expected to occur during the lifetime of the pipe segment.	10^{-2}
An event that individually is not expected to occur during the lifetime of the pipe segment; however, when considering all piping systems, an event in this category has the credibility of happening once.	10^{-4}
An event of such low probability that an event in this category is rarely expected to occur.	10^{-6}
An event of such extremely low probability that an event in this category is considered to be incredible.	10^{-8}

Historical or service data, expert judgment, or validated PFM calculations shall be used to estimate the limiting piping segment failure probabilities. The PFM calculations shall be the primary method used to estimate failure probabilities unless the piping materials and operating characteristics assessed are not compatible. When using expert judgment to estimate failure rates, the selected experts shall have sufficient structural reliability knowledge to estimate the failure probability. The process shall integrate information from relevant disciplines. Table S1-2 provides definitions that have been found useful in having the selected experts relate their knowledge of piping failures to a failure probability. PFM calculations that are used to estimate piping failure probabilities shall contain the following fundamental parameters:

(1) An appropriate geometric characterization of the piping segment of interest;

(2) Flaw density and size distribution after preservice inspection;

(3) Characterization of loading conditions, including mean stress, cyclic stress, number of cycles for both expected and postulated events; a probabilistic treatment of the frequency and loading uncertainty of these events shall also be considered;

(4) Failure modes and degradation mechanisms that are identified to potentially occur within the piping segment of interest shall be characterized over the lifetime of the piping system.

(5) Failure criteria shall be included, such as a limited loss of pipe wall, leaks, and rupture.

The above-noted fundamental parameters for the PFM calculations should also be considered if historical or service data or expert judgment processes are used to estimate piping failure probabilities. Only estimates of limiting failure probabilities for the pipe segments are needed. Estimates should be based on the scope of structural elements within the piping segment, and consideration of particular structural elements that will dominate the overall failure probability for the piping segment. To estimate the limiting failure probability of a piping segment, all significant degradation mechanisms, material attributes, and operating characteristics shall be combined to calculate the failure probability of the segment, regardless of the number of elements in the segment.

(d) *Failure Consequence.* Failure consequences are:

(1) Those pressure boundary failures affecting the function of the system in question, often referred to as direct effects. The following direct effects shall be considered:

(a) Failures that cause an initiating event such as a LOCA or reactor trip;

(b) Failures that disable a single train or system;

(c) Failures that disable multiple trains or systems;

(d) Failures that cause any combination of the above failures.

(2) Pressure boundary failures affecting other systems, components, or piping segments, often referred to as spatial or indirect effects, such as failures that cause pipe whip, jet impingement, or flooding. The total effect on core damage and large early release, given failure of the piping segment under consideration, shall be assessed. Consideration shall be given to the failure mode postulated for the piping segment. Consequences must then be measured in the correct terms to ensure proper calculations of risk measures. The spatial effects of piping segment failure on other systems, components, or piping segments shall be approximated. Previous plant hazard evaluations are useful in this process, along with a plant walkdown. Any assessment performed to determine that the effect on impacted targets would not cause any additional effects, interfere with system operation, or prevent plant shutdown, shall be documented.

(e) *Uncertainty*. To address the potential impact of uncertainty in the estimated segment failure probabilities and PRA results, a simplified uncertainty analysis shall be performed.

(1) The following range factors shall be assigned to each of the values estimated in S1-4.2.4. If the value is less than 10^{-4}, a range factor of 20 shall be assigned. If the value is greater than 10^{-2}, a range factor of 5 shall be assigned. Otherwise, a range factor of 10 shall be assigned.

(2) Propagation of uncertainty using an acceptable Monte-Carlo technique shall be performed for each of the four calculations.

(3) Any segment for which any of the four RRW values increased from below to above 1.005, due to the uncertainty propagation, shall be targeted for special consideration by the expert panel.

(f) *Recovery Action.* Consideration shall be given to evaluating the consequence of piping failure with and without recovery action.

(g) *Core Damage Frequency and Large Early Release Frequency.* The core damage value provides the risk, in terms of core damage frequency (events per yr), associated with the failure of the piping segment under consideration. The conditional core damage frequency or probability per failure is multiplied by the segment failure probability or rate to obtain a core damage frequency due to failures for each segment. The large early release frequency shall be evaluated in a similar manner.

(h *Piping Segment Importance.* The risk-importance measures defined in S1-4.2.2 shall be used to assist in piping segment risk selection.

(i) *Other Modes of Operation.* Any other information, including evaluation of external events and other plant operating modes other than at-power, that are appropriate to establish the importance of the piping segment shall be considered by the expert panel.

S1-4.2.6. Select HSS Piping Segments

(a) The expert panel shall apply the risk-importance measure RRW as described in para. S1-4.2.2. Any piping segment that has any of its four RRW values (two for CDF and two for LERF) exceeding 1.005 shall initially be considered HSS.

	Low Safety Significant	High Safety Significant
High Failure Importance	Owner Defined Programme **3**	(A) Susceptible Location(s) (100%) (B) Inspection Location Selection Process **1**
Low Failure Importance	Only System Pressure Test & Visual Examination **4**	Inspection Location Selection Process **2**

FIG. S1-2. HSS and LSS segment selection matrix.

(b) Ideally, the screening criteria established by this quantitative approach should capture the HSS piping segments, but the following condition shall also be considered by the expert panel.

(1) All piping segments not exceeding the screening criteria shall be evaluated by the expert panel to determine whether any piping segment was inappropriately ranked below these threshold values. RAW insights may be considered. Considerations shall be given to the limitations of the PRA implementation approach resulting from PRA structure and to limitations in the meaning and uncertainty associated with the importance measures. The expert panel shall also consider defense-in-depth, aging, deterministic and operational insights from inspection results, industry data, available pipe failure data, and other PRA application impacts. The expert panel shall then determine if additional piping segments must be considered HSS.

(c) The final HSS and LSS piping segments list, along with the rationale for any adjustments and decisions, shall be documented. The HSS and LSS piping segments shall be put into the appropriate regions of Figure S1-2.

S1-4.2.7. Process for Selecting Piping Structural Elements

The final list of HSS piping segments, as identified in S1-4.2.6 shall be selected for further evaluation at the piping structural elements level. The selection process is described in S1-4.2.8.

S1-4.2.8. Piping Structural Element Selection

(a) To complete the element selection process, a determination of segment high and low failure-importance shall be used to rank failure potential for the elements in that segment. HSS segments shall be considered to have a high failure-importance when a piping segment or its elements has either a degradation mechanism that is known to exist, which may be currently monitored as part of an existing owner's augmented programme, or is determined to be highly susceptible to a degradation mechanism that could lead to leakage or rupture. PFM calculation results may be used to determine this high failure importance if any location within which the segment exceeds the following indicator:

Probability of Large Leak $>10^{-4}$ per 40 years of operation

(b) A set of inspection locations or elements shall be identified for which (1) failures will have the greatest potential impact on safety, and (2) there is a greater likelihood of detectable degradation and consequently a greater potential for identifying, through NDE, piping degradation prior to failure. The final list of structural elements and rationale for any decisions made in establishing this list shall be documented. The following criteria shall be used to make this determination as shown in Figure S1-2:

(*1*) Region 1(A) includes all high-failure-importance locations in each HSS piping segment identified as likely to be susceptible to a known or postulated degradation mechanism and shall be examined. Exceptions include those locations already being examined under existing augmented programs. Region 1(B) includes other portions of these same HSS piping segments containing locations not affected by a known degradation mechanism and evaluated using a statistical evaluation such as the process described in S1-4.2.8(b)(2). At least one element in this portion of each HSS piping segment shall be examined.

(2) Region 2 includes all HSS piping segments with low-failure-importance locations. For these segments a statistical evaluation shall be used to define the number of random locations to be examined. A sampling plan shall be selected for each of these segments that achieves at least a 95% confidence (no more than 5% risk) of not exceeding an estimated leak (through-wall crack) frequency defined from industry operating experience, based upon estimates for piping leak frequencies in Table S1-3. In statistical calculations, a leak is a visible leak that does not influence system operation. It shall be estimated as the frequency of a through-wall flaw. This estimate shall be obtained from a PFM model with suitable input and output parameters. In cases where a PFM model cannot be used due to model limitations, such as application to socket welds or specific materials and to account

for uncertainty and the possibility of unknown degradation mechanisms in these segments, at least one element in each HSS piping segment shall be examined.

TABLE S1-3. ESTIMATES FOR PIPING LEAK FREQUENCIES

Material	Size		
	≤NPS 1(DN 25)	NPS 1(DN 25) < Size < NPS 4 (DN 100)	≥ NPS 4(DN 100)
Stainless steel	10^{-5}	10^{-5}	10^{-6}
Ferritic steel	10^{-5}	10^{-6}	5×10^{-6}

(3) Region 3 includes all LSS piping segments that have a high failure-importance. Locations selected for examination should be based on owner-defined programs.

(4) Region 4 includes all LSS segments and locations with low failure-importance.

(5) System pressure tests and VT-2 visual examinations are required for all Class 1, 2, and 3 piping, as applicable.

S1-4.2.9. Change-In-Risk Evaluation

(a) If a prior deterministic inservice inspection programme has been used, a change-in-risk evaluation shall be performed prior to initial implementation of a risk-informed inspection programme.

(b) Proposed inspection programme changes shall be assessed to quantitatively determine if any adjustments or compensatory measures to the proposed risk-informed inspection programme are necessary to provide assurance that the effect of the proposed change results in a risk decrease, risk neutrality, or acceptably small increase. The quantitative assessment shall consider CDF and LERF with and without operator action. Operator recovery action credited in the calculations shall assume perfect performance, which means no human error probabilities are required.

(c) The quantitative assessment shall modify the failure probability used (see S1-4.2.5(c)) in calculating change in CDF and LERF as follows:

(1) For piping segments that are part of augmented programs (such as erosion-corrosion and stress corrosion cracking), the failure probabilities with examinations credited are used.

(2) For piping segments that have NDE selections proposed or selected, the failure probabilities with examinations credited are used.

(3) For piping segments that have no NDE selections proposed or selected, the failure probabilities without examinations credited are used.

(4) For piping segments within containment, the failure probabilities with and without examinations credited based upon the proposed or selected NDE shall be used along with credit for leak detection. Credit for leak detection may be taken for reactor coolant system leaks and other system leaks that have analogous impact and detection possibilities as primary system leaks.

(5) No additional credit for inspection shall normally be given to piping segments that contain both augmented inspections and inservice inspections in the change-in-risk evaluation. However, for selected piping segments that are in the inspection programme and an owner's augmented programme, when the inspection programme requires additional or more stringent examinations beyond the augmented programme, an additional factor of 3 improvement shall be used to adjust the failure probability used in the change-in-risk calculation.

(d) The criteria for an acceptable change-in-risk evaluation are as follows:

(1) The total change in piping risk shall be a risk reduction or risk neutral when moving from a deterministic inspection programme to a risk-informed inspection programme. If not, the dominant system and piping segment contributors should be reviewed to determine additional examinations needed. If additional examinations are proposed, the change-in-risk calculation is to be revised until a risk-neutral position is achieved. This may require several iterations. For the purpose of this evaluation, risk-neutral is defined as essentially equivalent values, which may include acceptably small increases defined by the regulatory authority having jurisdiction at the plant site.

(2) The system results shall be reviewed to identify any systems for which there is a risk increase when moving from a deterministic inspection programme to a risk-informed inspection programme. For these systems, the following evaluation shall be performed to determine if additional examinations are required:

(a) If the CDF increase for the system is greater than one percent of the risk-informed inspection CDF for that system or greater than 10^{-8}, (whichever is higher), at least one dominant segment in that system shall be reevaluated to identify additional examinations.

(b) If the LERF increase for the system is greater than one percent of the risk-informed inspection LERF for that system or greater than 10^{-9}, whichever is higher, at least one dominant segment in that system shall be reevaluated to identify additional examinations.

(3) Dominant systems (e.g., system contribution to the total risk is greater than 10%), remaining essentially risk-neutral when moving from a deterministic inspection programme to a risk-informed inspection programme, shall be reevaluated to determine if it is practical to add examinations to reach system risk reduction. Impractical situations shall be part of the evaluation documentation.

(4) If additional examinations are added as a result of the change-in-risk evaluation, the evaluation shall be updated to reflect the additional examinations. Segments for which additional examinations are added shall be categorized HSS, and no statistical evaluation for a sample size or expert panel review and approval is required.

S1-4.2.10. Location and Examination Method Determinations

Once the piping structural elements list is completed in accordance with S1-4.2.8 and S1-4.2.9, the areas or volumes of concern for each of the HSS piping structural elements shall be determined and documented. This determination is based on the postulated degradation mechanisms identified in S1-4.2.5 and the configuration of each piping structural element. The examination methods and techniques for these identified areas or volumes of concern shall be determined in accordance with the requirements of Table S1-1 and documented.

S1-5. REEVALUATION OF RISK-INFORMED SELECTIONS

S1-5.1. GENERAL

Examination selections made in accordance with a risk-informed inspection programme shall be reevaluated on basis of inspection periods and inspection intervals that coincide with the inspection programme requirements for Inspection Programme A or B, as applicable. For Inspection Programme B, the third-inspection-period reevaluation will serve as the subsequent inspection interval reevaluation. The performance of each inspection-period or inspection-interval reevaluation may be accelerated or delayed by as much as one year. The reevaluation shall determine if any changes to the risk-informed inspection programme examination selections need to be made, by evaluation of the following:

(a) Plant design changes (e.g., physical: new piping or equipment installation; programmatic: power uprating / 18 to 24 month fuel cycle; procedural: pump test frequency changes, operating procedure changes)

(b) Changes in postulated conditions or assumptions (e.g., check valve seat leakage greater than previously assumed)

(c) Examination results (e.g., discovery of leakage or flaws)

(d) Piping failures (e.g., plant-specific or industry occurrences of through-wall or through-weld leakage, failure due to a new degradation mechanism, or a nonpostulated mechanism)

(e) PRA updates (e.g., new initiating events, new system functions, more detailed model used, initiating event and failure data changes)

S1-5.2. PERIODIC UPDATES

(a) If the periodic reevaluations of S1-5.1 indicate that piping structural elements, systems, or portions of systems may now be HSS, and the risk-informed inspection programme needs to be updated, the owner shall update the programme by adding examination selections in accordance with the requirements for HSS piping structural elements in S1-4.2.8, or by using the applicable portions of the same risk-informed selection process previously used to establish the risk-informed inspection programme. This reevaluation of the selections shall be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to reperform the entire risk-informed selection process, but the evaluation for the changes to the selections that occur shall be documented.

(b) If the reevaluations indicate that piping structural elements, systems, or portions of systems may now be LSS, the risk-informed inspection programme may remain unchanged, or examination selections may be deleted by using the applicable portions of the same risk-informed selection process that previously established the risk-informed inspection programme. This reevaluation of the selections shall be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to reperform the

entire risk-informed selection process, but the evaluation for the changes to the selections that occur shall be documented.

(c) If any portion of the risk-informed selection process is reperformed, a change-in-risk evaluation shall be completed in accordance with S1-4.2.9.

S1-5.3. INTERVAL UPDATES

If changes occur during periodic updates, based on qualitative reevaluation results, those changes shall be cumulatively evaluated for inclusion in the subsequent inspection interval update. The subsequent inspection interval update shall include a reevaluation using the applicable portions of the same risk-informed selection process used to establish the risk-informed inspection programme. This reevaluation of the selections shall be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to reperform the entire risk-informed selection process, but the evaluation for the changes to the piping selections that occur shall be documented. A change-in-risk evaluation shall be completed in accordance with S1-4.2.9.

SUPPLEMENT 2 TO APPENDIX D

RISK INFORMED SELECTION PROCESS--METHOD B

S2-1. INTRODUCTION

This Supplement provides the risk-informed selection process to be used for selection of piping segments and piping structural elements (including connections) for pre-service and in-service inspection. This selection process is based on the risk-significance of locations within an individual system. Figure S2-1 illustrates the evaluation process that is summarized in the following text.

S2-1.1. SYSTEM IDENTIFICATION

Systems shall be selected for analysis, and system boundaries and functions shall be identified.

S2-1.2. SEGMENT RISK ASSESSMENT

Each selected system shall be divided into piping segments determined to have similar consequence of failure and potential for failure (common degradation mechanisms, etc.). These segments shall be placed into risk categories based on combinations of consequence and failure potential. Risk-significant segments shall be identified.

S2-1.3. ELEMENT ASSESSMENT

Potential locations (elements) within the risk-significant segments shall be selected for inspection based on the specific degradation mechanism identified for the segment.

S2-1.4. INSPECTION VOLUME AND EXAMINATION METHODS

The inspection volume and method used for each element shall be determined based on the degradation mechanism associated with the element.

S2-1.5. DOCUMENTATION

The results of this alternative selection process shall be documented. This process shall include a review incorporating plant-specific and industry experience, as well as results of plant-specific inspections.

S2-2. BOUNDARY IDENTIFICATION

The Plant operator shall define the system boundaries included within the scope of the risk-informed inspection programme evaluation. Within each system boundary, the risk-informed evaluation may include Class 1, 2, or 3 piping defined in the deterministic in-service inspection programme, and piping outside the current deterministic programme examination boundaries.

Piping, or portions thereof, included for evaluation shall be based on the deterministic programme Class 1, 2, or 3 examination boundaries, determined in accordance with the national Code requirements.

S2-3. SEGMENT RISK ASSESSMENT

Piping within a system shall be grouped into segments of common failure consequence and susceptibility to common degradation mechanisms. To accomplish this grouping for each pipe segment within a system, both the potential for failure (i.e., susceptibility to potential degradation mechanisms) and the direct and indirect consequence of failure, shall be assessed in accordance with paras. S2-3.1 and S2-3.2.

S2-3.1. FAILURE POTENTIAL ASSESSMENT

S2-3.1.1. Identification of Degradation Mechanisms

Potential active degradation mechanisms for each pipe segment within the selected system boundaries shall be identified. The following conditions shall be considered:

(a) design characteristics, including material, pipe size and schedule, component type and other attributes related to system configuration

(b) fabrication practices, including welding and heat treatment

(c) operating conditions, including temperatures and pressures, fluid conditions (e.g., stagnant, laminar flow, turbulent flow), fluid quality (e.g., primary water, raw water, dry steam, chemical control), and service environment (e.g., humidity, radiation)[1]

(d) industry-wide service experience with the systems being evaluated

(e) results of pre-service, in-service, and supplemental (so called augmented) examinations, and the presence of prior repairs in the system

(*f*) degradation mechanisms identified in Table S2-1

S2-3.1.2. Failure Potential Categories

Degradation mechanisms shall be categorized as described in Table S2-2 in accordance with their probability of causing a large pipe break. Segments susceptible to Flow Accelerated Corrosion (FAC) shall be classified in the high failure potential/large break category. Segments susceptible to any of the other degradation mechanisms shall be classified in the medium failure potential/small leak category. Segments having degradation mechanisms listed in the small leak category shall be upgraded to the high failure potential/large break category, if the pipe segments also have the potential for water hammer loads.

System
Select system and define boundaries for evaluation

↓

Consequence Assessment
Direct & Indirect Effects
Identify Impact Groups:
Initiating Evenly System/Train, Combination*
Containment

↓

Failure Assessment
Design & Operating Conditions
Service Experience
Degradation Mechanisms

↓

Segment
Consequence + Degradation Mechanism

↓

Consequence Category
PRA/IPE Based Impact Group Tables

↓

Failure Potential Category
Degradation Mechanism
Loading
Service Experience

↓

Risk Categorization
High-Cat. 1, 2, and 3
Medium – Cat. 4 and 5
Low – Cat. 6 and 7

↓

Safety Significant Segments
Risk important segments to be included in Rl-ISI Programme

↓

Element Assessment
Defines Number of Inspections Required ([Note (1)]
25% - Categories 1, 2, and 3
10% -Categories 4 and 5

↓

Inspection Location
Procedure Selection Criteria and Service Experience

↓

Examination Requirements
Apply Prescriptive Requirements:
Examination Method and Volume
Acceptance Criteria and Evaluation Standards

↓

Document Revised ISI Plan

FIG. S2-1 Risk evaluation process.

(NOTE(1): If the chosen scope of the application applies only to Examination Category B-J welds, excluding socket welds, the following may be applied: Element Assessment: 10% - Examination Category B-J.)

TABLE S2-1. DEGRADATION MECHANISMS

Mechanisms		**Chapter Two Attributes**	**Susceptible Regions**
TF	*TASCS*	- piping > NPS 1 (DN 25); and - pipe segment has a slope < 45 deg. from horizontal (includes elbow or tee into a vertical pipe), and - potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids, or - potential exists for leakage flow past a valve (i.e., in-leakage, out-leakage, cross-leakage) allowing mixing of hoi and cold fluids, or - potential exists for convection healing in dead-ended pipe sections connected to a source of hot fluid, or - potential exists for two phase (steam / water) flow, or - potential exists for turbulent penetration in branch pipe connected to header piping containing hot fluid with high turbulent flow, and - calculated or measured AT > 50°F (10°C), and - Richardson number > 4,0	nozzles, branch pipe connections, safe ends, welds, heat affected zones (HAZ), base metal, and regions of stress concentration
	TT	- operating temperature > 270°F (130°C) for stainless steel, or operating temperature > 220°F (105°C) for carbon steel, and - potential for relatively rapid temperature changes including cold fluid injection into hot pipe segment or hot fluid injection into cold pipe segment, and - $\|\Delta T\|$ >200°F (93°C) for stainless steel, or $\|\Delta T\|$ > 150°F (65°C) for carbon steel, or $\|\Delta T\| > \Delta T$ allowable (applicable lo both stainless and carbon)	
SCC	*IGSCC (BWR)*	- evaluated in accordance with existing plant IGSCC programme per NRC Generic Letter 88-01	austenitic stainless steel welds and HAZ
	IGSCC (PWR)	- operating temperature > 200°F (93°C), and - susceptible material (carbon content > 0.035%). and - tensile stress (including residual stress) is present, and - oxygen or oxidizing species are present OR	
		- operating temperature < 200°F (93 °C), the attributes above apply, and - initiating contaminants (e.g., thiosulfate, fluoride, chloride) are also required to be present	
	TGSCC	- operating temperature > 150°F (65°C), and - tensile stress (including residua! stress) is present, and - halides (e.g.. fluoride, chloride) arc present, or caustic (NaOH) is present, and - oxygen or oxidizing species are present (only required to be present in conjunction w/halides, not (required w/caustic)	austenitic stainless steel base metal, welds, and HAZ

TABLE S2-1. DEGRADATION MECHANISMS (CONT.)

Mechanisms		Chapter Two Attributes	Susceptible Regions
SCC	*ECSCC*	- operating temperature > 150°F (65°C), and - tensile stress is present, and - an outside piping surface is within five diameters of a probable leak path (e.g.. valve stems) and is covered with non-metallic insulation that is not in compliance with Reg. Guide 1.36, or - an outside piping surface is exposed to wetting from chloride bearing environments (e.g., seawater, brackish water, brine)	austenitic stainless steel base metal, welds, and HAZ
	PWSCC	- piping/weld material is {Alloy 600/82/182), and - exposed to primary water at T > 570°F (300°C), and - the material is mill-annealed and cold worked, or cold worked and welded without stress relief	nozzles, welds, and HAZ without stress relief
LC	*MIC*	- operating temperature < 150°F (65°), and - low or intermittent flow, and - pH < 10, and - presence/intrusion of organic material (e.g., raw water system), or water source is not treated w/biocides {e.g., refueling water lank)	fittings, welds, HAZ, base metal, dissimilar metal joints (e.g., welds, flanges), and regions containing crevices
	PIT	- potential exists for low flow, and - oxygen or oxidizing species are present, and - initiating contaminants (e.g., fluoride, chloride) are present	
	CC	- crevice condition exists (e.g., thermal sleeves), and - operating temperature 150°F (65°C), and - oxygen or oxidizing species are present	
FS	*E-C*	- existence of cavitation source (i.e., throttling or pressure reducing valves or orifices); and - operating temperature < 250°F (120°C), and - flow present > 100 hr/yr, and - velocity > 30 ft/s (9.1 m/s), and - $(P_d - P_v)/\Delta P < 5$ where, P_d = static pressure downstream of the cavitation source, P_v - vapor pressure, and ΔP - pressure difference across the cavitation source	fittings, welds, HAZ, and base metal
	FAC	- evaluated in accordance with existing plant FAC programme	per plant FAC programme
Water Hammer (Note (1))		- potential for fluid voiding and relief valve discharge	

NOTE (1): Water hammer is a rare, severe loading condition, as opposed to a degradation mechanism, but its potential at a location, in conjunction with one or more of the listed mechanisms, could be a cause for a higher examination zone ranking.

LEGEND:

Thermal Fatigue	(TF)	Localized Corrosion	(LC)
Thermal Stratification, Cycling, and Striping	(TASCS)	Microbiologically-Influenced Corrosion	(MIC)
Thermal Transients	(TT)	Pitting	(PIT)
Stress Corrosion Cracking	(SCC)	Crevice Corrosion	(CC)
Intergranular Stress Corrosion Cracking	(IGSCC)	Flow Sensitive	(FS)
Transgranular Stress Corrosion Cracking	(TGSCC)	Erosion-Cavitation	(E-C)
External Chloride Stress Corrosion Cracking	(ECSCC)	Flow-Accelerated Corrosion	(FAC)
Primary Water Stress Corrosion Cracking	(PWSCC)		

TABLE S2-2. DEGRADATION MECHANISM CATEGORY

Failure Potential	Conditions	Degradation Category	Degradation Mechanism
High (Note (1))	Degradation mechanism likely to cause a large break	Large Break	Flow-Accelerated Corrosion
Medium	Degradation mechanism likely to cause a small leak	Small Leak	Thermal Fatigue Erosion-Cavitation Corrosion, Stress Corrosion Cracking
Low	None	None	None

NOTE (1): Refer to S2-3.1.2

S2-3.2. CONSEQUENCE EVALUATION

S2-3.2.1. Failure Modes and Effects Analysis (FMEA)

Potential failure modes for each pipe segment shall be identified, and their effects shall be evaluated. This evaluation shall consider the following:

(a) *Break Size.* Consequence analysis shall be performed assuming a large break for most segments. Exceptions are piping for which a smaller leak is more conservative, or when a small leak can be justified through a leak-before-break analysis in accordance with the criteria specified in NUREG-1061, Volume 3, and 10CFR50, Appendix A, General Design Criterion 4.

(b) *Isolability of the Break.* A break can be automatically isolated by a check valve, a closed isolation valve, or an isolation valve which closes on a given signal or by operator action.

(c) Indirect Effects. Includes spatial and loss of inventory effects.

(d) Initiating Events. Identified using a plant-specific list of initiating events from the plant Probabilistic Risk Assessment/Individual Plant Examination (PRA/IPE) and the plant design basis.

TABLE S2-3. CONSEQUENCE CATEGORIES FOR INITIATING EVENT IMPACT OF GROUP

Design Basis event Category	**Initiating Event Type**	**Representative Initiating Event Frequency Range (1/yr)**	**Example Initiating Events**	**Consequence Category (Note(1))**
I	Routine operation	>1		N/A
II	Anticipated event	$>10^{-1}$	Reactor trip, turbine trip, partial loss of feedwater	Low/Medium
III	Infrequent event	10^{-1} to 10^{-2}	Excessive feedwater Steam removal, loss of off-site Power	Low/Medium Medium/High
IV	Limiting fault or accident	$<10^{-2}$	Small LOCA, steam line break, feedwater line break, large LOCA	Medium/High

NOTE (1) Refer to S2-3.2.2 (a)(3)

(e) System Impact/Recovery. Means of detecting a failure, and the Technical Specifications associated with the system and other impacted systems. Possible automatic and operator actions to prevent a loss of systems shall also be evaluated.

(f) System Redundancy. Existence of redundant flowpaths for accident mitigation purposes shall be considered.

S2-3.2.2. Impact Group Assessment

FMEA impacts for each pipe segment shall be classified into one of three impact groups: initiating event, system, or combination. Consequence category (high, medium, low, none) shall then be selected in accordance with S2-3.2.2(a) through (f).

(a) Initiating Events (IE) Impact Group Assessment. When a postulated break results in only an initiating event (e.g., loss of coolant accident, loss of feedwater, reactor trip), the consequence shall be classified into one of four categories: high, medium, low, or none. Initiating event categories shall be assigned according to the following:

(1) The initiating event shall be placed into one of the categories in Table S2-3. These shall include all applicable design basis events previously analyzed in the Plant Operator's updated final safety analysis report, PRA, or IPE.

(2) Breaks that cause an initiating event classified as routine operation (Category I) are not relevant to this analysis.

(3) For piping segment breaks that result in Category II (Anticipated Event), Category III (Infrequent Event), or Category IV (Limiting Fault or Accident), the consequence category shall be assigned to the initiating event according to the conditional core damage probability (CCDP) criteria specified in Table S2-4.

(b) System Impact Group Assessment. The consequence category of a pipe segment failure that does not cause an initiating event, but that degrades or fails a system essential to plant safety, shall be based on the following three attributes:

(1) Frequency of challenge, which determines how often the mitigating function of the system is called upon. This corresponds to the frequency of initiating events that require the system's operation.

(2) Number of backup systems available, which determines how many unaffected systems are available to perform the same mitigating function as the degraded or failed system.

(3) Exposure time, which determines the time that the system would be unavailable before the plant is changed to a different mode in which the failed system's function is no longer required, the failure is recovered, or other compensatory action is taken. Exposure time is a function of the detection time and Allowed Outage Time, as defined in the plant Technical Specification.

TABLE S2-4. QUANTITATIVE INDICES FOR CONSEQUENCE CATEGORIES

Consequence Category	**Corresponding Conditional Core Damage Probability Range**	**Corresponding Conditional Large Early Release Probability Range**
High	$>10^{-4}$	$>10^{-5}$
Medium	$10^{-6}<\text{CCDP}\leq 10^{-4}$	$10^{-7}<\text{CLERP}\leq 10^{-5}$
Low	$\leq 10^{-6}$	$\leq 10^{-7}$

TABLE S2-5. GUIDELINES FOR ASSIGNING CONSEQUENCE CATEGORIES TO FAILURES RESULTING IN SYSTEM OR TRAIN LOSS

Affected Systems		**Number of Unaffected Backup Trains**							
Frequency of Challenge	**Exposure Time to Challenge**	0.0	0.5	1.0	1.5	2.0	2.5	3.0	≥ 3.5
Anticipated (DB Cat. II)	All Year	HIGH	HIGH	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW
	Between tests (1-3 months)	HIGH	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW
	Long AOT (≤ 1 week)	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW
	Short AOT (≤ 1 day)	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW	LOW
Infrequent (DB Cat. III)	AIL Year	HIGH	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW
	Between tests (1-3 months)	HIGH	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW
	Long AOT (≤ 1 week)	HIGH	MEDIUM*	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Short AOT (≤ 1 day)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
Unexpected (DB Cat. IV)	All Year	HIGH	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW
	Between tests (1-3 months)	HIGH	MEDIUM	MEDIUM	LOW*	LOW	LOW	LOW	LOW
	Long AOT (≤ 1 week)	HIGH	MEDIUM	LOW*	LOW	LOW	LOW	LOW	LOW
	Shan AOT (≤ 1 day)	HIGH	LOW*	LOW	LOW	LOW	LOW	LOW	LOW

GENERAL NOTE: Containment Performance: If there is no containment barrier and the consequence category is marked by an asterisk (*), the consequence category should be increased (medium to high or low to medium).

Consequence categories shall be assigned in accordance with Table S2-5 as High, Medium, or Low. Consistent with the initiating event group (Table S2-3), frequency of challenge is grouped into design basis event categories (II, III, and IV) unless initiating event frequency ranges are not consistent with Table S2-3. If this is the case, the frequency of the initiating event shall be used to determine the event category. Exposure time shall be obtained from Technical Specification and system operating configuration limits. In lieu of Table S2-5, quantitative indices based on conditional core damage probability may be used to assign consequence categories on the basis of the plant's PRA/IPE in accordance with Table S2-4. The owner shall ensure that the quantitative basis of Tables S2-4 and S2-5 (e.g., train unavailability approximately 10^{-2}) is consistent with the pipe failure scenario under evaluation.

(c) *Combination Impact Group Assessment.* The consequence category for a pipe segment whose failure results in both an initiating event and the degradation or loss of a system shall be determined using Table S2-6. The owner shall ensure that the quantitative basis of Table S2-5 (e.g., train unavailability approximately 10^{-2}) is consistent with the pipe failure scenario under evaluation. The consequence category is a function of two factors:

(1) Use of the system as a mitigating function for the induced initiating event;

(2) Number of unaffected backup systems or trains available to perform the same function.

(d) *Containment Performance.* The previous evaluations determine pipe failure importance relative to core damage. Pipe failure shall also be assessed for impact on containment performance. This shall be accomplished by addressing two issues both of which are based on an approximate conditional value of $\leq 10^{-1}$ between the CCDP and the likelihood of large early release from containment as shown in Table S2-4. If there is no margin (i.e., conditional large early release probability (CLERP) given core damage is $> 10^{-1}$), the assigned consequence category shall be increased by one level. The two issues are as follows:

TABLE S2-6. CONSEQUENCE CATEGORIES FOR COMBINATION IMPACT GROUP

Event	**Consequence Category**
Initiating event and 1 unaffected train of mitigating system available	High
Initiating event and 2 unaffected trains of mitigating systems available	Medium (Note (1)) (or IE Consequence Category from Table S2-3)
Initiating event and more than 2 unaffected trains of mitigating systems available	Low (Note(1)) (or IE Consequence Category from Table S2-3)
Initialing event and no mitigating systems affected	IE Consequence Category from Table S2-3

NOTE (l): The higher consequence category from Table S2-3 or Table S2-6 shall be assigned.

(1) CCDP values for initiating events and safety functions shall be evaluated to determine whether the potential for large early containment failure requires the consequence category to be increased.

(2) The impact on containment isolation shall be evaluated. If there is a containment barrier available, the consequence category from the core damage assessment shall be retained. If there is no containment barrier or the barrier failed in determining the consequence category from the core damage assessment, a margin of at least 10^{-1} in the core damage consequence category assignment shall be present for the consequence category to be retained.

For example, if the CCDP for core damage is less than 10^{-5} (i.e., a „Medium" consequence assignment) and there is no containment barrier, the „Medium" consequence assignment is retained because there is a margin of 0.1 to the „High" consequence category threshold (i.e., 10^{-4}). However, if the CCDP for core damage is $5x10^{-5}$ (i.e., a „Medium" consequence assignment) and there is no containment barrier, the consequence category is increased to „High" because the margin to the „High" consequence category threshold (i.e., 10^{-4}) is less than 10^{-1}. Table S2-7 shall be used to assign consequence categories for those piping failures that can lead to a LOCA outside containment.

(e) *Other Modes of Operation.* Any other information, including evaluation of external events and other plant operating modes other than at-power, that are appropriate to establish the importance of the piping segment shall be considered.

S2-3.3. SEGMENT RISK CATEGORIZATION

S2-3.3.1. Risk Matrix

The risk of pipe segment failure shall be evaluated on the basis of me expected likelihood of the event and the expected consequence. The likelihood of failure shall be estimated based on the segment exposure to varying degradation mechanisms, and shall be represented by the degradation category assigned to the segment in accordance with S2-3.1.

TABLE S2-7. CONSEQUENCE CATEGORIES FOR PIPE FAILURES RESULTING IN INCREASED POTENTIAL FOR AN UNISOLATED LOCA OUTSIDE OF CONTAINMENT

Protection Against LOCA Outside Containment	**Consequence Category**
One active barrier [Note(1)]	HIGH
One passive barrier [Note(2)]	HIGH
Two active barriers	MEDIUM
One active and one passive barrier	MEDIUM
Two passive barriers	LOW
More than two passive barriers	NONE

NOTES:

(1) An active barrier is presented by a valve that needs to close on demand.
(2) A passive barrier is presented by a valve that needs to remain closed.

Consequence shall be represented by the consequence category assigned to the segment in accordance with S2-3.2. The structure used to document the results of this analysis is called a Risk Matrix and is shown in Table S2-8. Each pipe segment shall be assigned to one of the risk categories in Table S2-8 based on its degradation and consequence categories.

S2-3.3.2. Risk Categories

The three failure potential (i.e., degradation mechanism) categories and four consequence categories shall be combined into seven risk categories, as follows. Piping segments and piping structural elements determined to be in Risk Category 1, 2, 3, 4, or 5 are HSS, and those that are in Risk Category 6 or 7 are LSS.

TABLE S2-8. RISK MATRIX

<table>
<tr><td colspan="2" rowspan="2">RISK GROUPS
High: Cat. 1, 2, and 3
Medium: Cat. 4 and 5
Low: Cat. 6 and 7</td><td colspan="4">CONSEQUENCE CATEGORY</td></tr>
<tr><td>NONE</td><td>LOW</td><td>MEDIUM</td><td>HIGH</td></tr>
<tr><td rowspan="3">FAILURE POTENTIAL</td><td>HIGH</td><td>Category 7</td><td>Category 5</td><td>Category 3</td><td>Category 1</td></tr>
<tr><td>MEDIUM</td><td>Category 7</td><td>Category 6</td><td>Category 5</td><td>Category 2</td></tr>
<tr><td>LOW</td><td>Category 7</td><td>Category 7</td><td>Category 6</td><td>Category 4</td></tr>
</table>

Risk Category	**Risk Area**
1	High Consequence and High Failure Potential Category
2	High Consequence and Medium Failure Potential Category
3	Medium Consequence and High Failure Potential Category
4	High Consequence and Low Failure Potential Category
5	Medium Consequence and Medium Failure Potential Category, or Low Consequence and High Failure Potential Category
6	Medium Consequence and Low Failure Potential Category, or Low Consequence and Medium Failure Potential Category
7	Low Consequence and Low Failure Potential Category, or No Consequence and Any Failure Potential Category

S2-4. ELEMENT ASSESSMENT

S2-4.1. STANDARD ELEMENT ASSESSMENT

The number of elements to be examined in each risk category shall be as follows:

(a) For segments in Risk Categories 1, 3, or 5, in an existing FAC inspection programme, with no known degradation mechanisms other than FAC, the number, location, and frequency of inspections shall be in accordance with the existing FAC inspection programme. The existing FAC inspection programme shall remain unchanged.

(b) For segments in Risk Category 1, 2, 3, or 5, in an existing IGSCC inspection programme with no known degradation mechanisms other than IGSCC, the number, location, and frequency of inspections shall be the same as the existing IGSCC inspection programme. If IGSCC welds are to be selected and credited under the risk-informed inspection programme they shall be treated in accordance with S2-4.1(c) or (d).

(c) For segments determined to have degradation mechanisms other than those included in the existing FAC and IGSCC inspection programs, segments that remain High or Medium risk, with FAC or IGSCC removed from consideration, or segments with no known degradation mechanisms, the following number of locations shall be examined as part of the risk-informed inspection programme.

  (1) For Risk Category 1, 2, and 3, the number of inspection locations in each category shall be at least 25% of the total number of elements in each risk category applicable to the system evaluated.

  (2) For Risk Category 4 or 5, the number of inspection locations in each category shall be at least 10% of the total number of elements in each risk category applicable to the system evaluated.

(d) For segments in Risk Category 6 or 7, volumetric and surface examinations are not required.

(e) All Class 1, 2, and 3 elements, regardless of risk category, require pressure tests and VT-2 visual examinations, as applicable, in accordance with national Code requirements.

S2-4.2. OPTIONAL ELEMENT ASSESSMENT

Rules should be discussed. Examinations may be concentrated on systems with more high-risk segments, such that a larger percentage of the elements in the High Risk Categories 1, 2, and 3 are examined. No more than 50% of the examinations required by other supplemental/augmented programs should be credited toward the required 10% inspection population.

S2-5. INSPECTION LOCATIONS AND EXAMINATION METHODS

S2-5.1. SELECTION OF ELEMENTS FOR INSPECTION

The selection of elements within each risk category shall be documented and shall be based on the following:

(a) Elements identified as susceptible to the specific degradation mechanisms in Table S2-1.

(b) Plant-specific in-service cracking or flaw experience.

(c) Availability of previous examination results for baseline, pre-service, and historical records.

(d) Inspections shall be required for each degradation mechanism and combination of degradation mechanisms (e.g., thermal fatigue and IGSCC) identified. Relative degradation severity for specific degradation mechanisms, when applicable, shall be considered (e.g., wear or erosion rates for flow accelerated corrosion, ΔT or Richardson number for thermal fatigue). Examination for elements in Risk Category 4 segments shall be based on areas of significant stress concentration, geometric discontinuities, or terminal ends.

(e) Availability of access to the element to ensure the examination method for the relevant degradation mechanism can be used effectively to achieve required coverage for the defined examination volumes.

(f) Elements should be selected to minimize personnel radiation exposure during inspection.

(g) Elements should be selected to minimize support services such as scaffolding, insulation, and rigging.

S2-5.2. EXAMINATION VOLUMES AND METHODS

The selection of examination volumes and methods for each element within a Risk Category will depend on the degradation mechanism present. Examination programs developed in accordance with this Appendix shall use NDE techniques suitable for specific degradation mechanisms and examination locations. The examination volumes and methods that are appropriate for each degradation mechanism are given by national Code requirements. The methods and procedures used for the examinations shall be qualified to reliably detect and size the relevant degradation mechanisms identified for each element. Personnel performing the examinations shall be qualified, and examinations shall be conducted and documented, in accordance with national Code requirements.

S2-6. CHANGE-IN-RISK EVALUATION

(a) If a prior deterministic inspection programme has been used, a change-in-risk evaluation shall be performed prior to initial implementation of a risk-informed inspection programme.

(b) Proposed risk-informed inspection programme changes shall be qualitatively or quantitatively assessed to determine if any adjustments to the proposed inspection programme or compensatory measures are needed to provide assurance that the effect of the proposed change is either risk-neutral, provides a risk reduction, or adds negligible increases in CDF and LERF.

i. *Qualitative Evaluation:* For segments categorized as Low Risk (Category 6 or 7), any changes to the number of examinations shall have a negligible impact on risk (EPRI TR-112657 Revision B-A, December 1999, Section 3.7.1). This qualitative assessment shall review all other risk categories to determine if the number of examinations is greater, the same, or less than the previous deterministic inspection programme, if applicable. For risk categories for which the number of risk-informed inspection examinations is greater or the same, the risk impact shall be considered a decrease, or at worse, risk-neutral, and therefore acceptable. For risk categories for which the number of examinations is less than the deterministic inspection programme, as applicable, a quantitative evaluation shall be performed.

ii. *Quantitative Evaluation:* The change in risk shall be estimated for changes in examinations of welds in segments in medium or high risk categories (1, 2, 3, 4, or 5).

(c) *Bounding Failure Frequency*: The failure frequencies of $2x10^{-6}$/weld-yr for welds in the high-failure-potential category, $2x10^{-7}$/weld-yr for welds in the medium-failure-potential category, and 10^{-8}/weld-yr in the low-failure-potential category may be used.

i. *Location-Specific Failure Frequency*: Degradation-mechanism-specific failure frequencies developed from an integrated analysis of observed pipe failure data may be used to develop location- and degradation-specific failure frequencies. An acceptable source of data is provided in EPRI TR-112657, Revision B-A, December 1999, Section 3.7.1. Any other evaluation of data used shall be of comparable scope and quality. These failure frequency estimates shall be used directly or in a sound and appropriate first-order Markov model of pipe rupture. The Markov model shall incorporate degradation-specific failure frequencies, time between opportunities to detect a leak, time between inspections, and the probability of (flaw) detection. The Markov model may be used to estimate location-specific failure frequencies, or to estimate the factor that inspection could reduce the location-specific failure frequency of the inspection programme. When failure frequencies based on degradation mechanisms are used, mechanisms that are in an augmented programme unaffected by the proposed risk-informed inspection programme need not be included in the location-specific estimates. Failure frequencies for other degradation mechanisms simultaneously present (except for IGSCC and FAC) shall be summed. If IGSCC and FAC are present with any other degradation mechanism, and if the deterministic inspection programme or the risk-informed inspection programme provides additional or more stringent examination beyond that required by the owner's augmented inspection programme, an additional factor of 3 improvements may be used to adjust the failure probability used in the change-in-risk calculation.

ii. *Conditional Risk Estimates:* The conditional core damage probability (CCDP) and conditional large early release probability (CLERP) estimated for each segment may be used, if available. Bounding values of the highest estimated CCDP and CLERP for high-consequence segments, and 10^{-4} (CCDP) and 10^{-5} (CLERP) for medium-consequence segments, and 10^{-6} (CCDP) and 10^{-7} (CLERP) for low-consequence segments shall be used, if segment-specific estimates are not available.

iii. The following general equations shall be used to estimate the change-in-risk. One estimate shall be made for the change in CDF and one for LERF. The equations illustrate only the change in CDF; the change in LERF due to application of the risk-informed inspection process shall be estimated by substituting the conditional large early release probability

(CLERP) for CCDP in the equations.

$$\Delta R_{CDF} = \Sigma_j (I_{rj} - I_{ej}) * PF_j * CCDP_j \tag{S2.1}$$

where

ΔR_{CDF}	is change in CDF due to replacing a deterministic inspection programme with a risk-in formed inspection programme;
I_{rj}	is factor of reduction in pipe rupture frequency at location j, associated with a risk-informed inspection programme;
I_{ej}	is factor of reduction in pipe rupture frequency at location j, associated with a deterministic inspection programme;
PF_j	is piping failure frequency at location j, without inspection;
$CCDP_j$	is conditional core damage probability at location j.

In terms of probability of detection [$POD_j = (I - I_j)$], the equation becomes:

$$\Delta R_{CDF} = \Sigma_j (POD_{ej} - POD_{rj}) * PF_j * CCDP_j \tag{S2.2}$$

where

POD_{ej}	is probability of detection at location *j*, in a deterministic inspection programme;
POD_{rj}	is probability of detection at location *j*, in a risk-informed inspection programme.

It is acceptable to use bounding estimates for pipe failure frequency from Equation (S2.1) and conditional core damage and large early release probabilities from Equation (S2.2) to simplify the calculations. If the bounding estimates for both pipe failure frequencies and conditional probabilities are used, the equation becomes:

$$\Delta R_{CDF} = [(POD_e * N_{efc} - POD_r * N_{rfc})] * PF_f * CCDP_c (POD_e * N_{efc} - POD_r * N_{rfc}) \tag{S2.3}$$

where

POD_e	is probability of detection in the previous deterministic inspection programme, if applicable (may be degradation mechanism specific);
N_efC	is number of inspection locations in the consequence f and failure frequency c categories in the previous deterministic inspection programme;
POD_r	is probability of detection in the risk-informed inspection programme (may be degradation mechanism specific);
N_{rfc}	is number of inspection locations in the consequence f and failure frequency c categories in the risk-informed inspection programme;
PF_f	is piping failure frequency for the high-, medium-, and low-failure frequency estimates;
$CCDP_c$	is conditional core damage probability for the high-, medium-, and low-consequence estimates.

(d) *Acceptance Criteria:* The estimated change in CDF and LERF for each system shall be less than 10^{-7}/yr and 10^{-8}/yr, respectively, and the total change in CDF and LERF shall be less than 10^{-6}/yr and 10^{-7}/yr respectively. If these requirements are not met, inspection locations shall be added. If this Supplement is applied to only Class 1 welds, the individual systems and system parts in Class 1 may be considered a single system, and the system-level guidelines shall be applied to the total change.

S2-7. REEVALUATION OF RISK-INFORMED SELECTIONS

S2-7.1. GENERAL

Examination selections made in accordance with a risk-informed inspection programme shall be reevaluated on the basis of inspection periods and inspection intervals that coincide with national inspection programme requirements. The performance of each inspection period, or inspection interval reevaluation may be accelerated or delayed according to national requirements (usually by as much as one year). The reevaluation shall determine if any changes to risk-informed inspection programme examination selections need to be made, by evaluation of the following:

(a) Plant design changes (e.g., physical: new piping or equipment installation; programmatic: power uprating / 18 to 24 month fuel cycle; procedural: pump test frequency changes, operating procedure changes).

(b) Changes in postulated conditions or assumptions (e.g., check valve seat leakage greater than previously assumed).

(c) Examination results (e.g., discovery of leakage or flaws).

(d) Piping failures (e.g., plant specific or industry occurrences of through-wall or through-weld leakage, failure due to a new degradation mechanism, or a nonpostulated mechanism).

(e) PRA updates (e.g., new initiating events, new system functions, more detailed model used, initiating event and failure data changes).

S2-7.2. PERIODIC UPDATES

(a) If the periodic reevaluations of S2-7.1 indicate that piping structural elements, systems, or portions of systems may now be HSS and the risk-informed inspection programme needs to be updated, the owner shall update the programme by adding examination selections in accordance with the requirements for Risk Category 1, 2, 3. 4, or 5 element selections in chapter S2-4, or by using the applicable portions of the same risk-informed selection process previously used to establish the risk-informed inspection programme. This reevaluation of the selections shall be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to reperform the entire risk-informed selection process, but the evaluation for the changes to the selections that occur shall be documented.

(b) If the reevaluations indicate that piping structural elements, systems, or portions of systems may now be LSS, Risk Category 6 or 7, the risk-informed inspection programme may remain unchanged, or examination selections may be deleted by using the applicable portions of the same risk-informed selection process that previously established the risk-informed inspection programme. This reevaluation of the selections shall be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to reperform the entire risk-informed selection process, but the evaluation for the changes to the piping selections that occur shall be documented.

(c) If any portion of the risk-informed inspection process is reperformed, a change-in-risk evaluation shall be completed in accordance with chapter S2-6.

S2-7.3. INTERVAL UPDATES

If changes occur during periodic updates based on qualitative reevaluation results, those changes shall be cumulatively evaluated for inclusion in the subsequent inspection interval update. The subsequent inspection interval update shall include a reevaluation using the applicable portions of the same risk-informed selection process used to establish the risk-informed inspection programme. This reevaluation of the selections shall be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to re-perform the entire risk-informed selection process, but the evaluation for the changes to the piping selections that occur shall be documented. The inspection interval update shall meet the requirements of national Codes, and a change-in-risk evaluation shall be completed in accordance with chapter S2-6.

APPENDIX E

NDE SYSTEM QUALIFICATION

SUMMARY

Integrity of the primary circuit is fundamental for the safe operation of any nuclear power plant. In-service inspection (ISI) and in particular non-destructive tests (NDT) play a key role in maintaining primary circuit integrity.

„Qualification“ is widely used to assess the capabilities and limitations of nondestructive in-service inspection systems. Qualification of an in-service inspection system means the assessment of the considered combination of NDT procedures, equipment and personnel to demonstrate that such an inspection system is fit for its purpose.

This appendix provides a methodology for qualification of ISI systems that might be used by WWER operating countries as a commonly accepted basis for further development of the necessary qualification related infrastructure.

This appendix also provides several qualification principles defining the administrative framework needed for the practical implementation of the methodology, a description of the process of qualification of an inspection system, according to that methodology, and specifys its minimum technical and documentation related requirements. Additionally it provides several specific requirements with regard to the NDT procedures, equipment and personnel to be qualified and to the test specimens to be used in practical trials. Finally, the Appendix suggests an appropriate distribution of responsibilities, among all parties involved in a qualification process, based on international practice and proposes a basic procedure for recognition of NDE qualification certificates.

It should be noted that this appendix does not provide defined criteria for the scope of a qualification process in terms of required inspection area(s) and NDT method(s), nor for the type(s) of flaws and required inspection effectiveness.

Nevertheless, qualification scope is a matter to be agreed upon between the licensee and the regulatory body having jurisdiction at the plant site before starting any qualification process.

This appendix has been prepared within the frame of the IAEA Technical Co-operation Project RER/4/030 and of the Extrabudgetary Programme on the Safety of WWER and RBMK NPPs.

1. INTRODUCTION

Radiation induced reactor pressure vessel embrittlement, the application of the leak-before-break concept to primary circuit piping and components and steam generator integrity problems both pose stringent requirements on the capability and effectiveness of in-service inspections performed at WWER plants. The capability and effectiveness of in-service inspection (ISI) has been identified and ranked, in the framework of the IAEA's Extrabudgetary Programme activities, as one of the most important (Category III) safety issues for WWER plants.

The present valid methodology [1] for qualification of in-service inspection systems for WWER NPPs was developed using the approaches and experience of several WWER operating countries, from the USA (ASME/PDI [2]), and from the European Commission and other western European countries (ENIQ [3], NRWG [4]). It should be noted that those qualification approaches, methodologies and activities are different in a number of aspects due to different industry and regulatory environments ([4], [5]). In this respect, this qualification methodology is intended to be a pragmatic synthesis, appropriate in the short and medium terms, to the specific circumstances of various WWER operating countries with relation to other guidelines (see [6]).

Implementation of this qualification methodology through all WWER operating countries would enable them to reach a common level of qualification related infrastructures, data bases, experience and expertise. It should be pointed out that qualification of in-service inspection systems is a complex and resource consuming task and hence WWER operating countries are strongly encouraged to coordinate and optimize their qualification related initiatives and resources.

2. OBJECTIVE

The objective of this Appendix is to provide a methodology for qualification of ISI systems which might be used as a commonly accepted basis for further development of necessary qualification-related infrastructures in WWER operating countries.

3. SCOPE

This Appendix refers to any considered non-destructive testing method and defines how non-destructive in-service inspection systems (NDT procedures, equipment and personnel) should be assessed in order to demonstrate that a given inspection system is able to reliably detect discontinuities with specified parameters.

It is not the intent of this report to provide criteria for definition of the extent of a qualification process neither in terms of required inspection area(s) nor NDT method(s) nor the type(s) of flaws and required inspection effectiveness. These are matters to be agreed upon between the licensee and the regulatory body having jurisdiction at the plant site before starting the qualification process.

4. QUALIFICATION PRINCIPLES

The detailed scope of a qualification process, in terms of required inspection area(s) and NDT method(s), as well as discontinuities being sought and required inspection effectiveness, is a matter to be agreed between the plant operator (licensee) and the regulatory body having. The scope should take into account the safety significance of each particular case and consider relevant national and international experience. This scope, or technical specification of the inspection required to be qualified, should be agreed before starting the qualification process, and should form part of the qualification process documentation.

Any organization managing, conducting, evaluating and certifying an in-service inspection system's qualification process (known as qualification body) should be independent from any commercial or operational consideration (e.g. independence criteria equivalent to those specified in ISO/IEC 17020: 1998 for a type A inspection body). Qualification bodies may also be an independent part of the licensee's organization (e.g. independence criteria equivalent to those specified in ISO/IEC 17020: 1998 for a type B inspection body).

The qualification body should operate according to a written quality system which guarantees its independence, impartiality and confidentiality. The quality system should be developed and reviewed by the qualification body, but be agreed to by the the regulatory body having jurisdiction at the plant site.

Any qualification process should be carried out according to written qualification procedures which clearly define the administrative interfaces and the types (unrestricted, restricted, confidential), paths, and timing of information to be exchanged between the involved parties (regulatory body, qualification body, licensee, inspection organization) as a consequence of the qualification process (see 7.4).

Written qualification procedures should be developed by the licensee, reviewed by the qualification body, and agreed to by the involved parties. The procedures should specify:

- In-service inspection area for which the NDE system is required to be qualified;
- Number, type, geometry, materials, manufacturing technology and surface conditions of test specimens to be used in practical trials;
- Type(s) and ranges of geometrical parameters of flaws to be detected and/or sized in practical trials;
- Conditions of the practical trials (open, blind);
- Minimum and maximum number of flawed and unflawed grading units;
- Grading criteria for detection and sizing of flaws;
- Acceptance criteria for detection and sizing;
- Special requirements where applicable (i.e. time limitations, access restrictions, environmental conditions, etc.).

Upon successful qualification of the ISI system, the qualification body should issue a qualification certificate to the licensee and/or inspection organization. The regulatory body is then to ask for permission to apply this NDE system as written on the certificate.

The qualification certificate should be indefinitely valid unless changes, affecting essential variables and/or parameters, are made to the equipment and/or procedure or to any mandatory document whose requirements must be met.

Personnel certificates, complementary to national certificates, should be issued by the qualification body to the inspection organization separately for each successful candidate. Validity of personnel certificates should be limited in time. Personnel certificates should be invalidated when a certified individual ceases to work for the associated inspecting organization, or when the inspection organization cannot produce documentary evidence of continuous satisfactory involvement of its certified individual in the qualified inspection. Conditions for validity of the qualification certificate shall be clearly defined in the qualification body's quality system.

Personnel certificates should clearly specify their scope of applicability (procedure, detection, sizing, etc.)

Responsibility for the ultimate approval of an NDT inspection system remains with the licensee on the basis of evidence provided by the qualification body as a result of the qualification process.

5. QUALIFICATION APPROACH

Qualification of a non-destructive inspection system (NDT procedures, equipment and personnel) should be carried out through a combination of technical justifications and practical trials. The relative weight of each of these elements is a matter to be agreed with the licensee and the qualification body. This would be based on a comprehensive assessment of the technical justification prepared either by the licensee or by the inspection organization on behalf of the licensee. It is recommended to separate NDT procedures and NDT equipment qualification from personnel qualification.

The qualification process is described in more detail in Section 8.

5.1. QUALIFICATION OF PROCEDURES AND ASSOCIATED EQUIPMENT

The purpose of this qualification process is to demonstrate that a proposed NDT method, technique, procedure and associated equipment is adequate for its purpose i.e. to detect and/or size specific type(s) of flaws, in a specific component or inspection area, with the required effectiveness.

Where an item of the equipment falls within the scope of a national, European or other international NDT standard or other written specification, the qualification should include, where appropriate:

— A paper study to determine the relevance of the standard or specifications to the specific case;

— Proof of compliance with the standard(s).

Where an item of the equipment does not fall within the scope of an appropriate standard or specification, or the plant operator does not want to use existing standards or specifications, the qualification should ensure that provision is made to measure the equipment essential parameters, identified during the analysis of the influential parameters. The NDT procedure should identify the essential parameters and should specify allowable values and tolerances.

Qualification of NDT procedures using technical justification involves the following:

- Assessment of the technical adequacy of the NDT procedure;
- Assessment of the analysis of the essential parameters;
- Checking that all those NDT procedure essential parameters which affect the outcome of the NDT significantly, and the ranges within which they can vary, are specified and that they are, if necessary, considered in the practical trials;
- Checking that the NDT procedure is written in a sufficiently systematic and unambiguous way that its application is reproducible.

If the assembled technical justification cannot provide a full demonstration of the adequacy of the proposed procedure and associated equipment, then supplementary practical trials shall be required. In this case, these supplementary practical trials may be carried out under „blind“ or „non-blind“ conditions.

5.2. QUALIFICATION OF PERSONNEL

The purpose of this qualification process is to demonstrate that inspection personnel, certified along a national or international scheme, using a previously qualified NDT procedure and associated equipment, are able to detect, discriminate and size flaws with the required effectiveness when subject to relevant conditions resembling those to be encountered in an actual inspection. Qualification of personnel should be carried out through practical trials under „blind“ conditions (see for details Section 6.4.2.).

For automated non-destructive inspections, carried out by a team of inspectors, it may be necessary for only certain designated members of the team to be qualified, for example those carrying out data analysis and interpretation.

6. QUALIFICATION PROCESS

Qualification of an NDT inspection system may be required by the regulatory body having jurisdiction at the plant site, by the plant owner (licensee) or by the inspection organization. Any qualification process should be carried out according to the following steps.

6.1. TECHNICAL SPECIFICATION

The detailed scope of qualification, in terms of required inspection area(s) and NDT method(s) as well as flaws and required inspection effectiveness, is a matter to be agreed between the licensee and regulatory body, before starting any qualification process, taking into account the safety significance of each particular case, and considering relevant national and international experience.

This technical specification should clearly state:

- Applicable code requirements;
- Components or areas to be inspected and its essential parameters (geometry, material compositions and structure, surface finish, etc.);
- Non-destructive inspection method to be applied (UT, RT, EC, etc.);
- Inspection conditions and their essential parameters (time and/or access restrictions, relevant environmental conditions, etc.);
- Expected or postulated flaws or degradation conditions which should be detected in the actual component and their essential parameters (type, morphology, geometry, position, orientation, size and any other factor which could affect the response of the required NDT method).
- Flaw parameters which have to be measured as position, length, depth and their expected or postulated ranges;
- Required inspection effectiveness in terms of:
  - flaw detection rate;
  - acceptable false-call rate;
  - allowable tolerances between reported and actual flaw parameters;
- Minimum quality assurance requirements applicable to the qualification process.

6.2. INSPECTION PROCEDURE

Based on the technical specification the inspection organization will propose a particular inspection procedure and associated equipment deemed adequate to fulfill inspection requirements. The inspection procedure must be described and associated equipment should be identified.

This includes:

- In-service inspection area for which inspection procedure is to be qualified;
- Application range for which inspection procedure is to be qualified;
- Applied ISI standard (national or international);
- Description of proposed NDT techniques and of their physical principles;
- Description of essential parameters;
- Description of operational conditions of the technique/equipment;
- Description of calibration process;
- Indication's reporting and discrimination criteria;
- Description of equipment and software (if applicable) associated to the inspection procedure including a list of their essential parameters, necessary data verification , measurement methods and acceptable values;

— Statement on whether the equipment, or part of it, is under the scope of any national or international standard.

6.3. PRELIMINARY REVIEW OF THE INSPECTION PROCEDURE

The purpose of this step is to provide a detailed review and subsequently preliminary approval or rejection, by the qualification body, of the proposed inspection procedure before proceeding with qualification.

Requirements listed in Sections 9.1, 9.2 and 9.3 are relevant for this review.

6.4. QUALIFICATION PROCEDURE

The qualification procedure describes the entire process of qualification of the inspection system (NDT procedure, equipment and personnel) and specifies the technical details to carry out the practical trials.

It contains the description of the:

— In-service inspection area for which qualification procedure is to be prepared;

— Technical justification;

— Practical trials (open and/or blind);

— Evaluation of qualification results.

The qualification procedure should be produced by the qualification body in cooperation with the licensee and submitted to the Regulatory Body, if required.

6.4.1. Technical justification

Practical reasons limit the number of test pieces that can be used for inspection qualification. Therefore, test piece trials can only provide limited information on the performance of an inspection system. The purpose of the technical justification is to overcome these limitations by:

— Citing all the evidence which supports an assessment of the capability of the inspection system to perform to the required level, and therefore to provide better confidence in the inspection;

— Complementing and generalizing practical trials results by providing systematic evidence that the results obtained on specific defects in the test pieces have a wider range of validity to other defects and conditions within the inspection scope;

— Providing a sound basis for designing effective test piece trials;

— Providing a technical basis for selection of essential parameters of the inspection system and their valid ranges.

Important elements with regard to the technical justification of the NDT procedure and associated equipment capabilities are given below in Sections 6.4.1.1 and 6.4.1.2.

6.4.1.1. Available evidences

All available theoretical and/or practical evidence which support the total or partial adequacy of the proposed procedure and equipment should be detailed in the technical justification.

Theoretical evidence may include:

— Review of the NDT procedure to verify that it is written to allow its reproducible application in the field while minimizing operator error, and so that all essential variables are identified;

— Results from application of mathematical models of the test (where available). In this case it has to be demonstrated that models used have been verified and validated for the particular conditions of the actual inspection. Mathematical models are very useful when they allow one to extrapolate or interpolate inspection results obtained on certain test blocks to actual components or inspection areas. This can justify the use of simplified test blocks, instead of actual components or mock-ups, for the supplementary practical trials which are deemed necessary. Mathematical models can be used in technical justifications if their predictive capability, accuracy and limitations are adequately documented.

Practical evidence may include:

— Results from similar qualifications previously performed;

— Feedback from field experience;

— Feasibility studies and industrialization trials;

— Description of equipment by the manufacturer;

— Experimental development results;

— Applicable results of national or international research programmes;

— Practical (field) experience on the application of the procedure and associated equipment in actual inspections when this information has been confirmed through destructive tests;

— Applicable laboratory tests.

6.4.1.2. Assessment of the procedure and associated equipment capabilities

Based on available theoretical and/or practical evidence the capabilities of the proposed procedure and associated equipment must be assessed, within the technical justification, in relation to:

— Inspection objectives;

— Essential parameters of the component or area to be inspected;

— Essential parameters of the flaws to be detected and/or sized;

— Essential variables of the inspection equipment;

Results of this assessment should be documented in terms of:

— Performance of the proposed inspection procedure and associated equipment (detection rate, sizing accuracy and false call rate) justified on the basis of the results obtained over the defect population, taking into account both the representativness of those defects in relation to the actual defects being sought and appropriate statistical criteria.

— Maximum range of variation of essential parameters where the applicability of the proposed procedure and equipment can be justified on the basis of theoretical and practical evidence provided.

— Essential parameters (i.e. those ones which have a significant effect on the outcome) should be identified and the range in which they can vary should be defined. For defects and components the essential parameters are to be defined in the input information provided prior to inspection qualification, and the qualification is only valid within the defined boundaries. For NDT equipment and procedures it should be verified during qualification that requirements are included (e.g. calibration requirements) which ensure that the essential parameters remain within the defined boundaries in order not to invalidate the qualification.

— An important element of the technical justification is the feedback of field experience. This source of information can become most important if the population of similar components or plants is large enough. This feedback must however be validated. The information generated should not be biased by experts' impressions. Evaluation, possibly involving destructive examination, is often necessary to validate the information coming from plant inspections.

6.4.2. Practical trials

Practical trials may be proposed by the licensee as a part of the qualification procedure, or may be deemed necessary as a result of the assessment of the technical justification as supplementary practical trials by the qualification body.

Practical trials are of two major types:

— Open trials, generally considered for the qualification of inspection procedures and of associated equipment;

— Blind trials more specifically considered for personnel complementary qualification; blind trials are carried out using blind test pieces or qualification data, or via a combination of both.

Specific test pieces requirements are listed in Section 8.4**,** requirements for qualification data are in Section 8.5.

Practical trials may involve test pieces replicating the component being inspected in size and geometry. The defective condition may also be accurately replicated. If metallurgical flaws are involved, the test piece will be designed to contain flaws of the type judged to be possible in appropriate positions and will normally include the 'worst case' defects judged most difficult to detect and size for the given defect situation. Such test pieces produce realistic results but are expensive and time-consuming to make and can usually only replicate a small fraction of the flaws which might actually occur.

Simpler test pieces, i.e. test pieces of simpler geometry and/or containing less realistic defects, can also be used but the results need to be extrapolated to real situations using physical reasoning and modeling. When this is possible it offers a quicker and less expensive route to inspection qualification.

The qualification body should assess the use of flaws incorporated in test pieces as producing either realistic or conservative responses relative to the defects specified by the plant operator.

It is recommended that personnel qualification be separated from procedure/equipment qualification. This will aid in exact identification of where any weaknesses lie. Procedure / equipment qualification is preferably done using open trials, both for detection and sizing. An important aspect of using open trials for procedure/equipment is that inspection results obtained have to be explained and justified in full detail to the qualification body. A blind trial can be a realistic way of assessing performance, particularly in terms of whether the combined personnel, equipment and procedure or some combination of these can in practice produce satisfactory results.

6.4.3. Evaluation of the practical tests

Results of the qualification process will be evaluated by the qualification body and reported to the licensee.

The degree of fulfillment of each of the specified inspection requirements (technical specifications) by the proposed ISI system must be summarized in the qualification procedure. These degrees of fulfillment should be derived from a qualitative and/or quantitative comparison between each inspection requirement and the corresponding procedure, equipment and personnel capability as supported by the theoretical and/or practical evidence generated by the technical justification and practical trials.

6.5. CERTIFICATION AND APPROVAL

After completion of a qualification process, the qualification body should issue appropriate certificates clearly identifying the particular combination of NDT procedure, equipment and personnel which has been successfully qualified. If experience demonstrates that a qualified inspection system or individual is not meeting inspection requirements, then the qualification certificate should be withdrawn by the qualification body.

Based on the evidence provided by the qualification body, the licensee should approve the utilization of the particular inspection system (NDT procedure, associated equipment and personnel) at its plant(s).

Depending on the particular circumstances in each country a review and endorsement of the qualification process by the regulatory body may be required prior to approval of the licensee.

7. DOCUMENTATION OF THE QUALIFICATION PROCESS

Any qualification process should be thoroughly documented in a qualification dossier. The qualification dossier should include:

- Technical specification of the inspection to be qualified;
- Inspection procedure;
- Preliminary review of the inspection procedure;
- Qualification procedure which consists of:
  - Technical justification;
  - Description of practical trials;
  - Results of practical trials including ranges of considered essential parameters and limitations.
- Evaluation of qualification process;
- Conclusion(s) of qualification.

The qualification dossier should also include, in the appropriate order, all assessments, evaluations and certificates issued by the qualification body, along the qualification process and the licensee's approvals and/or endorsements. The qualification process and its relation with the qualification dossier is summarized in Figure 1.

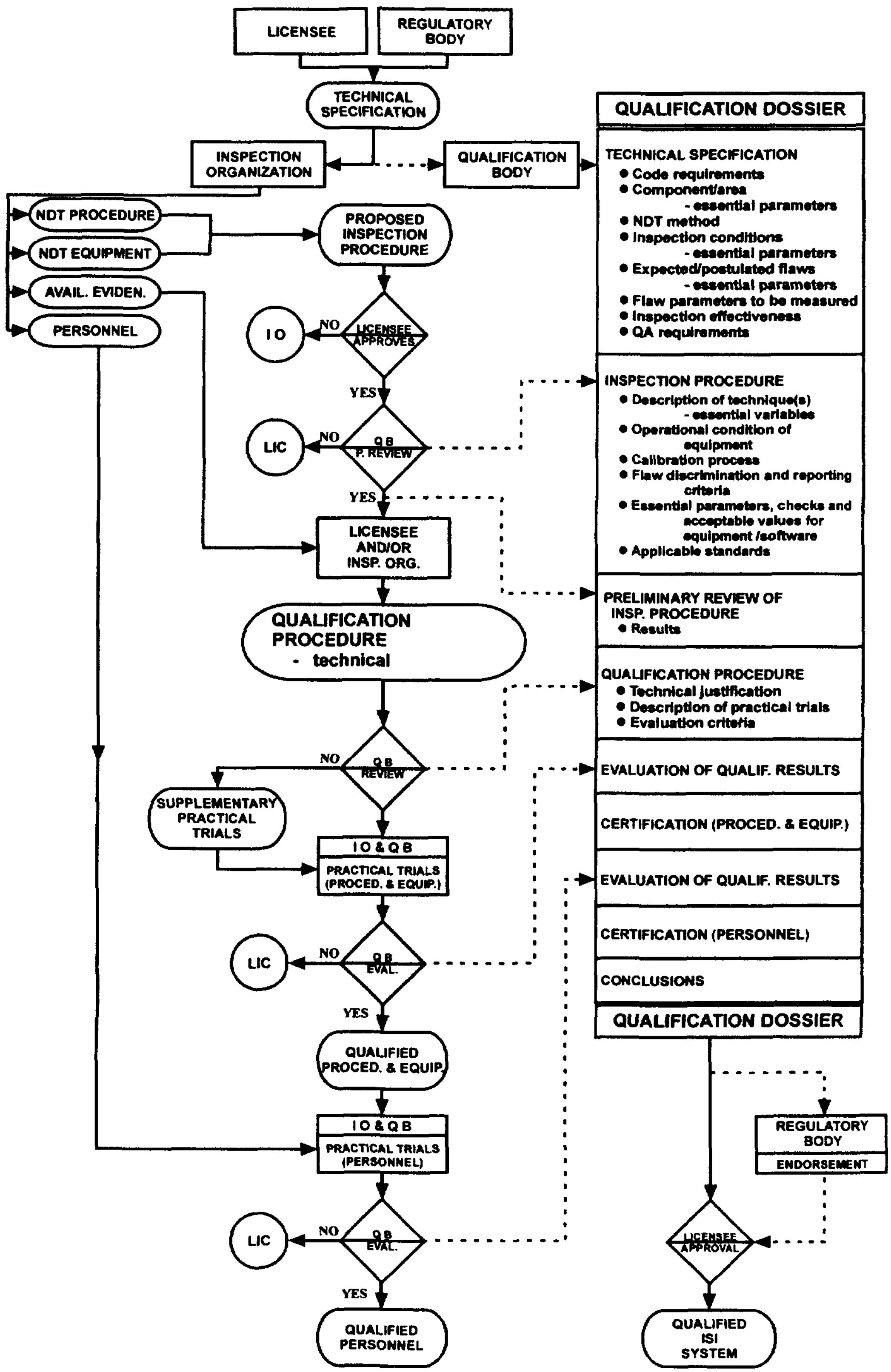

FIG.1. Qualification process and its relationship with the qualification dossier.

8. SPECIFIC QUALIFICATION RELATED REQUIREMENTS

8.1. INSPECTION PROCEDURE

Inspection procedures should be written in a step-by-step way which allows for its reproducible application while minimizing risk of operator misinterpretation. Inspection procedures should specifically define their scope and applicability (materials, product form, thickness, diameter, etc.) and essential parameters.

Inspection procedures should clearly specify the value or range of values of essential variables which can significantly affect, in a negative way, the outcome of the test for the particular NDT method and technique used. Inspection procedures should clearly specify the calibration and inspection data to be recorded, the recording method, and recording equipment where used.

When calibration is required, inspection procedures should clearly specify calibration methods for detection and sizing of flaws.

Inspection procedures should clearly state the flaw detection, discrimination and sizing methods/techniques used.

8.2. EQUIPMENT

Inspection procedures should include a specification of the equipment's essential variables, methods for their measurement, and required performance values. Inspection equipment should be assessed against the performance values specified in the inspection procedure.

8.3. PERSONNEL

Inspection procedures should require personnel with appropriate certifications according to a recognized national or international certification scheme.

Only inspection personnel who detect, record, evaluate and perform sizing of flaws, during inspections are required to be additionally qualified through practical trials according to this methodology.

8.4. TEST PIECES

Test pieces (test blocks) should be produced according to requirements given in technical specifications and be representative of the material, original technology, configuration details of inspection areas, and appropriate flaws and degradation mechanisms of concern.

Tests specimens used in non-blind practical demonstration should not be used in blind practical demonstrations.

All test specimens used in blind practical demonstration shall be identified in an adequate way so that identification markings should be obscured during qualification.

Flaws connected to the examination surface of test specimens shall be obscured during performance of blind practical trials.

All information regarding test specimens used in blind practical demonstrations (manufacturing drawings; type, number, location and sizes of flaws) should be considered restricted except to the authorized staff of the qualification body.

Realistic defects should be used in test specimens whenever possible. Artificial defects may be used in test specimens when they have been previously validated with respect to both the characteristics of the simulated defects and their response to the actual NDT method.

For each known or potential component degradation mechanism the minimum defect dimensions considered must be a fraction of the maximum allowable defect size, determined by structural integrity assessment, taking into account different defect geometries and in-service defect growth between inspections, as well as any other applicable requirement (LBB concept application, etc.). Defect growth rate must be based either on conservative generic data or on specific test data for the component, its material composition, heat treatment, operating conditions, etc. Appropriate safety margins in relation with minimum defect size and detection and sizing capabilities should be demonstrated in accordance with regulations and codes accepted by the national regulatory body. All such information is essential to any inspection required to be qualified.

8.5. QUALIFICATION DATA FILES

Qualification data files contain data previously measured on real components or on test pieces with realistic or artificial flaws. For qualification it is also possible to use data obtained by computer simulation.

Data obtained in this way be restricted in such a manner, so that the personnel to be qualified using the data are not aware of its details during their qualification.

Acceptance criteria for the evaluation of practical trials with the utilization of qualification data files shall be defined in the technical specification and/or qualification procedure for each qualification task.

If qualification data files are processed in such a way whereby several data files are combined for the qualification of an individual personnel to be qualified, the qualification body will retrieve a set of qualification data files by random selection.

A set of qualification data shall contain data with the indication of flaws, and also data without indications. Data with the indication of flaws shall be in compliance with indications which are characteristic of the typical degradation mechanisms of individual components.

Qualification data should be preferably in the format corresponding to the software for the measurement and analysis of the measured data. To ensure confidentiality of information, identical processes are to be used for the test pieces as for practical blind trials.

9. RESPONSIBILITIES

The responsibilities of each of the involved parties is a matter to be agreed upon before starting any qualification process, taking into account international practice and the specific circumstances of each particular country. Appropriate distribution of responsibilities may be as follows:

9.1. PLANT OPERATOR (LICENSEE)

Plant operator is responsible for overall facility safety. This requires among other things performing in-service inspections, and therefore the plant operator is responsible for the required efficiency of the inspections.

Plant operator responsibilities include:

— Select components that require inspection qualification and the defects to be detected in particular components. This list is updated according to national and international field experience.

— Ensure that any internal or external inspection organization performing non-destructive in-service inspections (inspection organization) at the operator's plant(s) has been previously qualified, if so required, according to this methodology.

— Approve inspection procedures proposed by the inspection organization, and take ultimate responsibility for NDT procedures and technical justifications. In some cases these documents will be written by the plant operator, in others by the inspection organization.

— Define the technical specifications of in-service inspections required to be qualified and the required effectiveness for each particular case; to give the inspection organization and to the qualification body all required input information (components, defects, qualification objectives) pertaining to the inspection(s) to be qualified, including the inspection performance to be met.

— Ensure that results of qualification trials (including any limitations) are taken into account in, and remain applicable to, the subsequent inspection.

— Review and approve (if applicable) the final qualification dossier; the plant operator takes the necessary steps to enable the qualification body to keep the qualification dossier updated with national and international field experience.

— Supervise inspection activities that affect performance, especially receipt and verification of the equipment, qualification of personnel, contents of procedures, logistics of operations and evaluation of results.

9.2. INSPECTION ORGANIZATION

Inspection organization (inspection vendor) responsibilities include:

— Develop detailed inspection procedures;

— Assemble technical justifications for proposed inspection procedures and associated equipment;

— Assure certification of its inspection personnel in relevant NDE methods according to relevant national or international schemes;

— Participate in the qualification process in close cooperation with the qualification body, providing all necessary information according to applicable qualification procedures (see also [1] IAEA-EBP-WWER-11 document);

— Provide necessary documentation and information elements allowing the qualification body to set up the qualification dossier;

— Participate in the qualification of the NDT procedure, equipment and personnel, if requested (inspection organization developing inspection procedure and other documents can be ordered by the licensee only for these activities, the other inspection organization can be required to be qualified);

— Assist the qualification body in keeping the qualification dossier up to date.

9.3. QUALIFICATION BODY

Qualification body responsibilities include:

— Develop qualification procedures;

— Assess, review and comment on inspection procedures and technical justifications;

— Assess results of practical qualification tests performed on test assemblies;

— Identify and/or design required test specimens for supplementary and personnel qualification practical trials;

— Manage procurement of test specimens according to the qualification body's quality system;

— Conduct and supervise the qualification process, as described in Section 6, including:

  — Initial review and preliminary approval of proposed inspection procedures;

  — Invigilation of practical trials;

  — Evaluation of results;

  — Assembly of qualification dossier;

  — Issue and withdrawal of certificates.

The need for the qualification body to be separate from the plant operator is to be determined by the plant operator and by the regulatory body if qualification is carried out as a result of regulatory requirements. Where it is necessary for the qualification body to be independent, but it is within the plant operator's organization, the qualification body should have a quality system which guarantees its independence from commercial or operational considerations.

9.4. REGULATORY BODY

The regulatory body has been assigned the task of monitoring and evaluating safety and ensuring that the licensees fulfill the conditions of their site licensees.

Regulatory body responsibilities include:

- Define or review basic qualification requirements that must be met from a safety and defense in depth point of view;
- Approve and periodically audit the quality system of the qualification body;
- Review and approve qualification procedures; if required
- Review and endorse (if applicable) the final qualification dossier.

10. PROCEDURE FOR THE RECOGNITION OF QUALIFICATION CERTIFICATES OF FOREIGN/ALIEN INSPECTION ORGANIZATIONS

For cases where the plant operator intends to perform an in-service inspection using a foreign/alien inspection vendor, which has the NDT system for specific component qualified abroad, the following procedure shall be followed:

Plant operator provides technical specification in accordance with section 7.1 to the foreign/alien inspection vendor organization.

The inspection organization submits to the operator and the qualification body the detailed inspection procedure, the technical justification and the qualification certificate approved by the foreign/alien qualification body.

Qualification body will review appropriateness and completness of the submitted documentation regarding qualification of the inspection procedure, inspection equipment, and competency of inspection personnel in accordance with the technical specification. On the basis of the qualification dossier the qualification body will issue a recognition statement of the qualification certificate. In some cases the qualification body may determine that supplementary documents and/or evidence may be needed to ensure that the technical specification and national qualification body quality system requirements are met.

Based on the statement of the national qualification body, the national regulatory body will confirm the validity of the qualification certificate of the foreign/alien inspection organization.

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SUPPLEMENT 1 TO APPENDIX E

GLOSSARY

For the purpose of this report the following definitions apply:

Blind trial/test

Practical demonstration in which inspection personnel have:

- No detailed knowledge of the number, position and size of any flaw;
- No knowledge of whether a test specimen contains any flaw;
- No access to flaws connected to the examination surface;
- No access to test specimen identifications.

Discrimination

The process and rules which allow inspection personnel to disregard indications arising from sources other than relevant flaws.

Essential parameter

Those parameters of components, of defects that the test is intended to detect and/or to size and of the environmental conditions in which the actual inspection has to be performed which are significant in determining the outcome of the test.

Essential variable

Those parameters of NDT equipment which can significantly affect in a negative way, the outcome and quality of a particular test.

False call

The reporting of a non-defective area as defective.

Flaw

Defective condition in which the metallurgical structure contains discontinuities such as cracks.

Inspection system

All parts of a non-destructive testing system (NDT procedure, associated equipment, software and personnel).

Inspection organization (inspection vendor)

Organization performing In-service inspections.

ISI System qualification

The systematic assessment, by all available methods, to provide a reliable confirmation that an inspection system is capable of achieving the required effectiveness under real inspection conditions.

NDT method

Physical principle which defines a family of non-destructive tests (i.e. ultrasonics, eddy current, etc.).

NDT procedure

Step-by-step sequence of rules describing how a specific NDT technique or combination of techniques is to be applied to a specific inspection area in order to detect, evaluate and size flaws.

NDT technique

Each one of the specific ways an NDT method can be applied (i.e. pulse-echo, rotating probe, etc.).

Open (non-blind) trial/test

Practical demonstration in which inspection personnel are previously informed on the type, number and characteristics of the test specimens as well as on the type, morphology, position and dimensions of the flaws to be detected and/or sized.

Qualification body

Independent organization that conducts the NDT qualification processes.

Qualification certificate

Document issued by a qualification body, under the rules of its quality system, stating that a duly identified combination of NDT procedures, equipment and personnel is capable of achieving the objectives stated for the inspection.

Qualification data

Data (data acquisition files) previously measured on real components or on test pieces with realistic or artificial flaws.

Qualification procedure

An orderly sequence of steps describing how a specific combination of NDT procedures, equipment and personnel applied to a specific inspection area has to be qualified including: generation of the technical justification, required test specimens, flaws, conditions of practical trials, grading and success criteria for practical trials, and any other special requirements where applicable.

Qualification protocol

Document describing administrative actions and interfaces, as well as the types (unrestricted, restricted, confidential) paths and timing of the information to be exchanged, between all involved parties, as a consequence of a qualification process.

Quality system

Document describing the structure, staff and facilities of a qualification body including rules of procedure and management for carrying out all its activities.

Supplementary practical trials

Those practical trials which might be deemed necessary for qualification of NDT procedure and NDT equipment in addition to practical trials proposed by the licensee.

Technical justification

Documented evidence which supports the assessment of proposed non-destructive inspection system capabilities justifying the selection of essential parameters of the inspection system and their valid ranges, the extent of practical trials needed, real or artificial defects to be used, and any other particular requirement.

Complementary personnel qualification

Qualification, which applies only to those inspection personnel who detect, record, evaluate and size flaws.

APPENDIX F

COMPONENT AND PIPING SUPPORTS

1. SCOPE

This Appendix contains requirements for structure design and development, materials, manufacturing, installation, repair, operation (including in-service inspection), operation life time extension and decommissioning of supports and hangers (hereinafter referred to as supports) intended for elements of WWER nuclear power plants (NPP).

The requirements are applicable to supports of equipment and pipelines of 1st, 2nd, 3rd and 4th safety classes in accordance with the classification given in Section 4.

The requirements are not applicable to:

— Supports of safety or containment shells;

— Support structures of pressurized vessel internals including core supports;

— Penetrations (tunnels).

This Appendix does not cover special support or restraint structures (the function of which is based on plastic deformation), loss of stability, or other part failures, with the exception of pipe-whip restraints intended for minimization of pipe rupture effects. Requirements for pipe-whip restraints differ from the main contents of this Appendix, and are given in Section 12.

2. TERMS AND DEFINITIONS

Component

Any independent part, combination of parts, assemblies, sub-assemblies or units which function in a defined mode to ensure correct working of a system. Equivalent term: unit (block).

Certification

A form of conformance confirmation of a product to requirements of technical regulations, standards, codes of rules, technical specifications or terms of contracts; it is carried out by a certification body.

Defect

Each separate discrepancy of a product to established requirements.

Product

Output (result) of an activity presented in a material form and intended for further application with economic or other purposes.

Regulatory document

A document approved in established order, which contains juridical and technical regulations for various activities or their results, e.g. general technical regulation, international standard, national standard, code of rules, classifier, standard of organization.

Safety (of a product, an object, a process)

A state characterized by absence of inadmissible risk connected with damaage to life or health of people, property of physical or corporate bodies, government assets and environment.

Standard

A document, which determines for the purposes of voluntary reusability, features of a product, rules for implementation and features of processes of design and development (including investigations), manufacturing, building, installation, adjustment, operation, storage, transportation, selling and utilization, carrying out works or services. Also, standard can contain rules and methods for investigations (experimental tests) and measurements, sample selection rules, requirements to terminology, symbols, packing, marking or labels, and rules of their application.

Support structure (support, hanger)

A structure intended for installation, orientation and attachment of reactor vessel components, equipment components, pipelines and fittings (elements) of NPP units; supports take load from supported (suspended or hanged) elements and transmits them to building structures. Supports are not pressure boundary components.

Support structure (support, hanger) of linear type

A support or hanger which can be approximated as part of strength analysis as a set of beams or rods.

Support structure (support, hanger) of shell type

A support or hanger which can be approximated as part of strength analysis as a set of plates and shells.

Technical specification (TS)

A document approved in established order in accordance with manufacturer’s decision or consumer’s (customer’s) request which specifies requirements of a product, processes of its manufacturing, marking, acceptance, inspection methods, transportation, storage, using and utilization. Also, TS can contain rules and methods for investigations (experimental tests) and measurements, sample selection rules, requirements for terminology, symbols, packing, marking or labels, and rules of their application.

3. PRINCIPAL SYMBOLS AND ABBREVIATIONS

3.1. TECHNICAL SYMBOLS

a_N	Cumulative fatigue damage
E	Young modulus, [MPa]
E^T	Young modulus at design temperature, [MPa]
F	Load, or force, [N]
$F_{кр}$	Critical load, or critical force, [N]
F_L	Limit load (load combination), [N]
Fx, Fy, Fz	Components of load vector along the axes of coordinate system, [N]
$I_{\min}$	Minimum inertia moment of a cross section, [mm^4]
i_σ	Stress (cycle asymmetry) ratio
k	Shape factor
K_I	Stress intensity factor, [MPa·m$^{1/2}$]
K_{Ic}	Critical stress intensity factor, [MPa·m$^{1/2}$]
K_σ	Theoretical stress concentration factor
L	Reference length, mm
Mx, My, Mz	Components of moment vector relative to the axes of coordinate system, [N·m]
N	Number of loading cycles
n_N	Safety factor on number of cycles at cyclic strength analysis
n_σ	Safety factor on reference local stresses at cyclic strength analysis
n_{PL}	Safety factor at limit analysis
R_m^T	Minimum value of ultimate strength at design temperature, [MPa]
$R_{p0,2}^T$	Minimum value of yield stress at design temperature, [MPa]
R_{-1}^T	Endurance limit at symmetric cycle and design temperature, [MPa]
s	Rated wall thickness, mm
T	Design temperature, [K, ºC]
T_k	Critical brittleness temperature, [K, ºC]
T_{k0}	Critical brittleness temperature for material in initial condition, [K, ºC]

ΔT_N	Shift of critical brittleness temperature due to cyclic damageability, [K, ºC]
ΔT_T	Shift of critical brittleness temperature due to thermal ageing, [K, ºC]
Z^T	Reduction of area of sample cross section at static tensile failure at design temperature
$[a_N]$	Allowable cumulative fatigue damage
$[N]$	Allowable number of loading cycles
ζ	Coefficient which takes into account manufacturing imperfections
σ	Stress, [MPa]
σ_b	General bending stress due to mechanical loads, [MPa]
$\sigma_{b\Sigma}$	Total bending stress due to mechanical and kinematic loads, [MPa]
σ_{bL}	Local bending stress, [MPa]
σ_{bw}	Mean bending stress in threaded part section, [MPa]
σ_d	General kinematic stress, [MPa]
σ_m	General membrane stress due to mechanical loads, [MPa]
$\sigma_{m\Sigma}$	Total membrane (total mean) stress due to mechanical and kinematic loads, [MPa]
σ_{mL}	Local membrane stress, [MPa]
σ_{mw}	Mean membrane stress in threaded part section, [MPa]
σ_T	General thermal stress, [MPa]
σ_{TL}	Local thermal stress, [MPa]
σ_Y	Yield stress, [MPa]
τ_j	Tangential stress in reference section of a corner weld with incomplete penetration, [MPa]
τ_{sw}	Torsion stress in threaded part, [MPa]
τ_{th}	Tread shear stress, [MPa]
φ_j	Static strength reduction coefficient in welded joint
φ_s	Cyclic strength reduction coefficient in welded joint
$[\sigma]$, $[\sigma]_j$, $[\sigma]_w$	Rated allowable stresses for support parts, [MPa]
$(\sigma)_1$, $(\sigma)_2$, $(\sigma)_{3w}$, $(\sigma)_{4w}$	Groups of stress categories, [MPa]
$(\sigma)_R$	Range of reference stresses, [MPa]

(σ_{aF}), $(\sigma_{aF})_w$	Amplitudes of local reference elastic equivalent stresses in support components, [MPa]
$(\sigma_F)_{max}$	Maximum local reference elastic equivalent stresses cycle stress calculated taking into account stress concentration, [MPa]

3.2. ABBREVIATIONS OF TERMS

AF	Airplane fall
AMP	Ageing management programme
AMR	Ageing management review
CAD	Computer Aided Design, software type
CAE	Computer Aided Engineering, software type
CAM	Computer Aided Manufacturing, software type
CAPP	Computer Aided Process Planning, software type
DA	Design accident
DE	Design earthquake
ERP	Enterprise Resource Planning, software type
FEM	Finite element method
GOST	Standard of USSR in Russian transcription
ILS	Integrated Logistics System, software type
LRA	Licence renewal application
MDE	Maximum design earthquake
NOC	Normal operation conditions
NOC (S)	Normal operation conditions (short-term)
NPP	Nuclear power plant
OST	Industry standard of USSR or RF in Russian transcription
PDM	Product Data Management, software type
RE	Radiographic examination
SWA	Shock wave action
TS	Technical specifications
TU	Technical specifications in Russian transcription

USE	Ultrasonic examination
VE	Visual examination
VNOC	Violation of normal operation conditions
WWER	Water-water energy reactor

4. GENERALITIES

4.1. REQUIREMENTS FOR SUPPORT STRUCTURES

4.1.1. Design and development of supports intended for elements of WWER NPPs shall be carried out in accordance with the requirements of this Appendix. Additional information can be obtained in Russian regulatory documents including PNAE G-1-011-97 [1], PNAE G-7-002-86 [2], PNAE G-7-008-89 [3], NP-031-01 [4], SPiR-O-2008 [5].

4.1.2. Supports for elements of WWER NPPs shall ensure working capacity, reliability and operational safety within the specified life time given in technical specification (TS) for a support and/or in a registration certificate of the supported element.

If the specified life time of a supported element exceeds the specified life time of its support then a process shall be provided for timely support replacement.

Life time of supports can be extended for a period exceeding that given in the TS and/or in the registration certificate of the supported element in accordance with the procedure described in Section 9.

4.1.3. It is recommended to use a support arrangement that ensures the practicality of its surveyance, repair, and examination of base metal and welded joints by non-destructive methods after manufacturing (installation) and during operation.

4.1.4. Supports shall be designed in such manner that excludes moisture collection in secluded corners and hollows.

4.1.5. Design of sliding supports shall ensure that friction coefficients do not exceed values used in strength analysis. Sliding surfaces made from special materials with low friction shall be free of contamination.

4.1.6. When designing a pipeline support system, standard supports shall be preferred.

4.1.7. For bearing dynamic loads, use of shock absorbers, vibration dampers or rigid struts is recommended. Shock absorbers and vibration dampers do not restraint thermal movements of the supported element, while rigid struts restrain movement of the supported element in one direction only.

4.1.8. Validation of support strength is based on the requirements given in Sub-section 5.4.

4.2. BOUNDARY BETWEEN SUPPORTS AND SUPPORTED COMPONENTS

4.2.1. Boundaries between supports and the supported element are defined by the design organization taking into account the rules given in this Sub-section below.

4.2.2. The boundary between a supported element and its support is a contact surface of connected components of the supported element and its support structure.

4.2.3. Welded joint which connects the support and its supported element is not a part of the support structure but belongs to the supported element.

4.2.4. An integrated support (i.e. manufactured jointly with an element without using detachable or non-detachable joints) is not a part of the support structure but belongs to the supported element.

4.2.5. A pad for attachment of supports to an element manufactured together with the element without using detachable or non-detachable joints is not a part of support structure but belongs to the supported element.

4.2.6. Connecting parts of detachable joints (bolts, pins, etc.) between the supported element and its support are a part of the support structure.

4.2.7. Clamps used for fixing elements, and detachable joints of clamps, including the connecting parts, are a part of the support structure.

4.2.8. Examples of dividing components between support structures and supported element are shown in Figures 1 – 3 where the following designations are used:

1 = supported element

2 = support

3 = welded joint relating to element

4 = welded joint relating to support structure

5 = fasteners relating to support structure

6 = site for attachment of support structure relating to supported element.

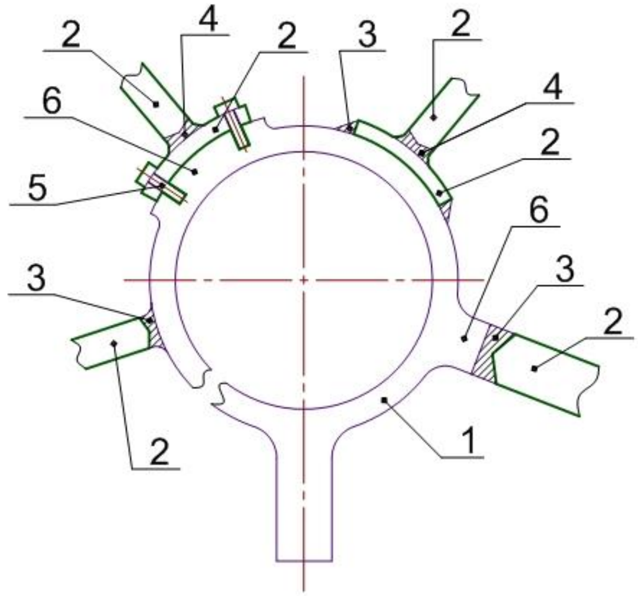


FIG. 1. Example of dividing components between support structures and supported element.

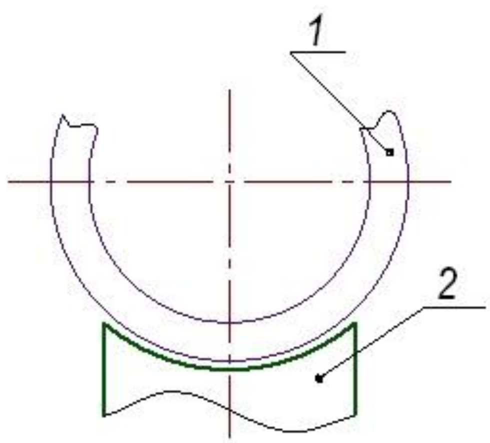


FIG. 2. Example of dividing components between support structures and supported element.

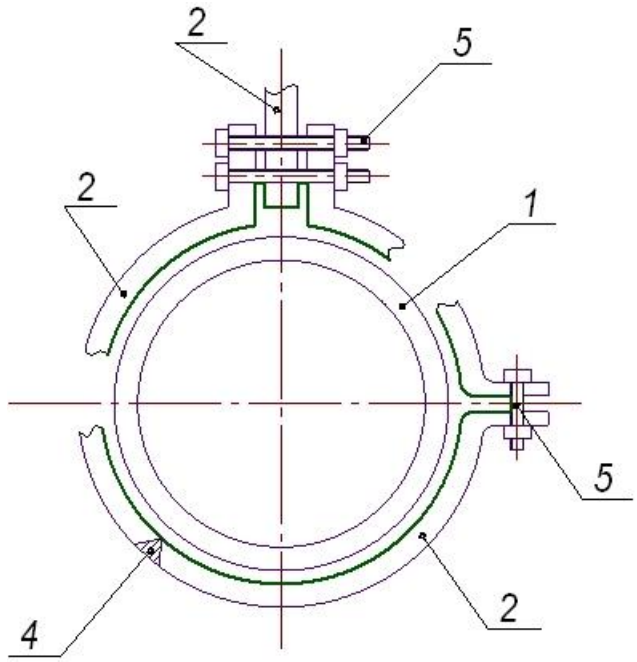


FIG. 3. Example of dividing components between support structures and supported element.

4.3. BOUNDARY BETWEEN SUPPORTS AND BUILDING STRUCTURES

4.3.1. Boundaries between supports and building structures are defined by the design organization taking into account the rules given in this Sub-section below.

4.3.2. Boundary between a support and building structure is a surface of the building structure contacting with the support structure.

4.3.3. Welded joints for welding supports to the building structure are a part of the support structure.

4.3.4. Connecting parts of detachable joints (bolts, pins, etc.) between the support and building structure are not a part of support structure.

4.3.5. Metal plates or parts manufactured from iron sections preliminarily embedded into building structures for future attachment of supports to building structures are not a part of the support structure.

4.3.6. Examples of dividing components between support structures and building structures are shown in Figures 4 – 6 where the following designations are used:

1 = supported element

2 = support

3 = building structure

4 = welded joint relating to support structure

5 = fasteners relating to building structure

6 = metal plates or parts manufactured from iron sections intended for attachment of supports to reinforced concrete building structure.

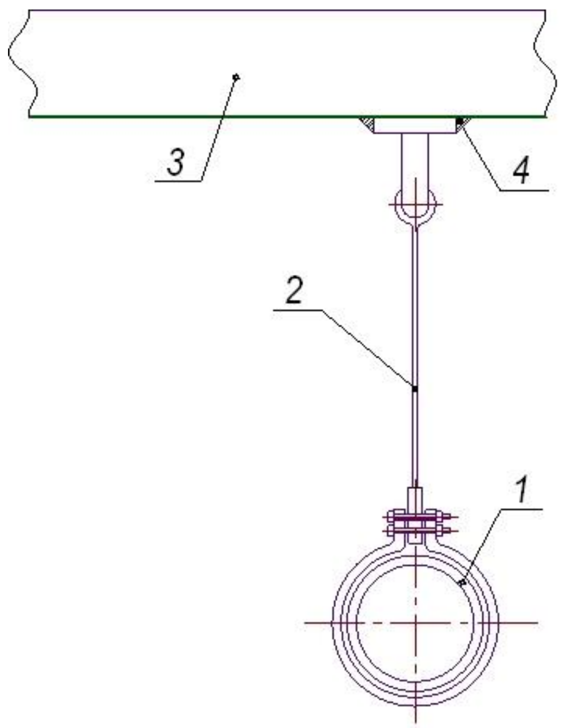


FIG. 4. Example of dividing components between support structures and building structures.

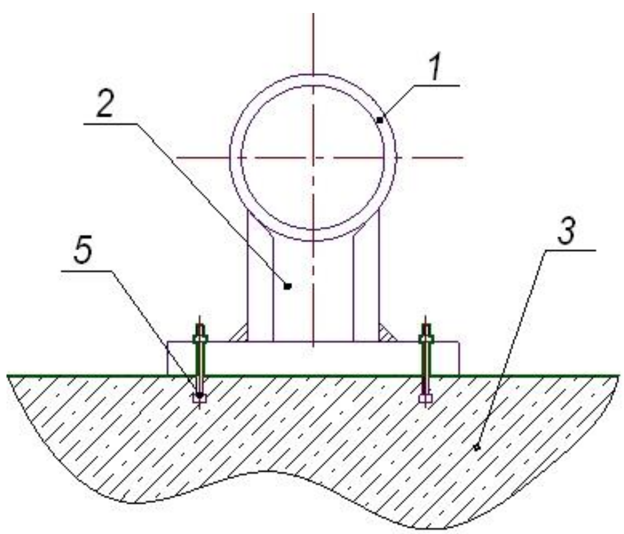


FIG. 5. Example of dividing components between support structures and building structures.

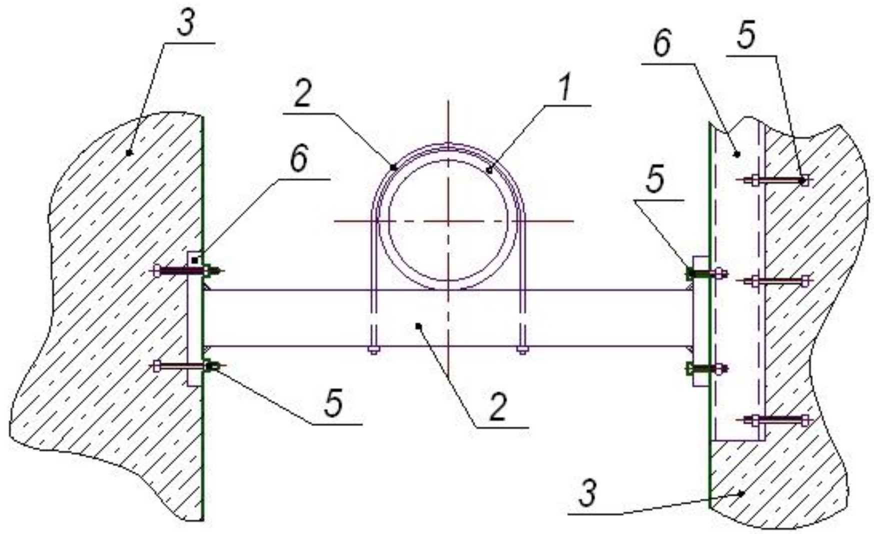


FIG. 6. Example of dividing components between support structures and building structures.

4.4. DOCUMENTATION

4.4.1. All documentation: design, engineering, technological, installation, operational and repair documents developed for supports shall meet requirements of this document.

4.4.2. Any changes in documentation on supports which may be necessary during manufacturing, installation, surveillance, operation, extension of operation life time, and decommissioning shall be made by organizations which developed the documentation, and shall be brought to the notice of the manufacturing organization, installation organization and operating organization.

4.4.3. Operating organization on the basis of design (engineering) documentation shall develop working instructions on operation of supports, surveillance, in-service inspection, repair and replacement.

4.5. CLASSIFICATION

4.5.1. Support classification

4.5.1.1. Supports for elements of WWER NPPs are subdivided into four safety classes and three categories of seismic resistance.

4.5.1.2. Safety class for a support shall be assigned in accordance with the safety class of supported element under classification given in NP-001-97 (PNAE G-01-011-97) [1].

4.5.1.3. Category of seismic resistance for a support shall be assigned in accordance with the category of seismic resistance of the supported element under classification given in NP-031-01 [4].

4.5.1.4. If a support is used simultaneously to support several components of equipment or pipelines of different classes and/or categories then for the support, the higher safety class and/or category of seismic resistance of supported equipment or pipeline shall be assigned.

4.5.1.5. It is allowed to design, manufacture, install, repair, operate, extend the life time and decommission supports in accordance with requirements of supports of a higher safety class.

4.5.1.6. Supports of each safety class (with the exception of supports of the 1st safety class due to the fact that at WWER NPPs only reactor vessel supports belong to this class) are subdivided into equipment supports and pipeline supports. Pipeline supports, in turn, are divided into standard and non-standard supports.

4.5.1.7. Non-standard supports require individual designs.

4.5.1.8. Standard supports are supports composed of components which are catalogued according to structural shape (design), dimensions, and allowable loads, and are qualified (certified). Manufacturing of standard supports shall be certified and shall meet established repeatable requirements (for example, series manufacturing).

4.5.2. Categories of welded joints

4.5.2.1. The category of welded joints for a support shall be the same as the safety class of the support.

4.5.3 Loading mode classification

4.5.3.1. Basic design loading modes of supports for elements of NPPs with WWER are the following:

- Normal operating conditions: long-term – NOC and short-term – NOC (S);
- Violation of normal operating conditions (VNOC);
- Design accident (DA);
- Maximum design earthquake (MDE);
- Design earthquake (DE);
- Airplane fall (AF);
- Shock wave action (SWA).

4.5.3.2. The following combinations of operating loading modes and dynamic forces (loading cases) shall be considered:

- For supports of 1st category of seismic resistance:
  - NOC + MDE;
  - VNOC + MDE;
  - NOC + AF;
  - VNOC + AF;
  - NOC + SWA;

- VNOC + SWA;
- DA + DE;
- NOC + DE;
- VNOC + DE.

— For supports of 2nd category of seismic resistance:

- NOC + DE;
- VNOC + DE.

— For supports of 3rd category of seismic resistance:

- NOC + DE.

5. DESIGN AND DEVELOPMENT OF SUPPORTS

5.1. EQUIPMENT SUPPORTS

5.1.1. Support structures for WWER NPP equipment are non-standard structures.

5.1.2. Equipment supports for WWERs are recommended to be designed in the form of two support units spaced along the height. The bottom support unit limits linear displacements and rotation of an equipment component in all directions. The top support unit does not restraint upward thermal expansion, but limits horizontal displacement of the equipment component. Such a restraining system is recommended for the WWER reactor vessel and pressurizer. Installation of equipment components on one support is allowed with appropriate justification.

For restraining of the upper reactor block against external dynamic loading, it is recommended to install a support at the upper crossbeam level that limits horizontal displacements of the upper block steel structure and the covers of the protection control system. Due to the necessity to compensate for the thermal expansion of the reactor coolant pipe, it is impossible to ensure rigid fixing of the steam generator and the main circulating pump. Thus to withstand seismic loadings the use of shock absorbers and tie-rods is required.

5.1.3. Example of arrangement of the reactor main circulation circuit in the reactor compartment building is shown in Figure 7.

The main bearing support of the reactor is the supporting truss, which is composed of radial pairs of supporting ribs connected by upper and lower plates, inner and outer shells, rider sheets and bridge sheets. Each pair of connected supporting ribs form a support beam with a box cross section which varies along its length. Circular connections between support beams create the circular stiffness of the supporting truss which is needed to withstand vertical and horizontal loads from the reactor. Supporting truss components are shown in Figure 8. The whole inner volume of the supporting truss is filled with concrete.

In order to withstand the horizontal seismic load, and to prevent the reactor vessel from overturning, it is recommended to use the supporting truss at the reactor vessel flange connector level. The basic components of the supporting truss are a ring with upper and lower disks, and ribs fixed in concrete with anchor rods, as well as ribs welded into box structures (brackets). A drawing of the supporting truss is shown in Figure 9.

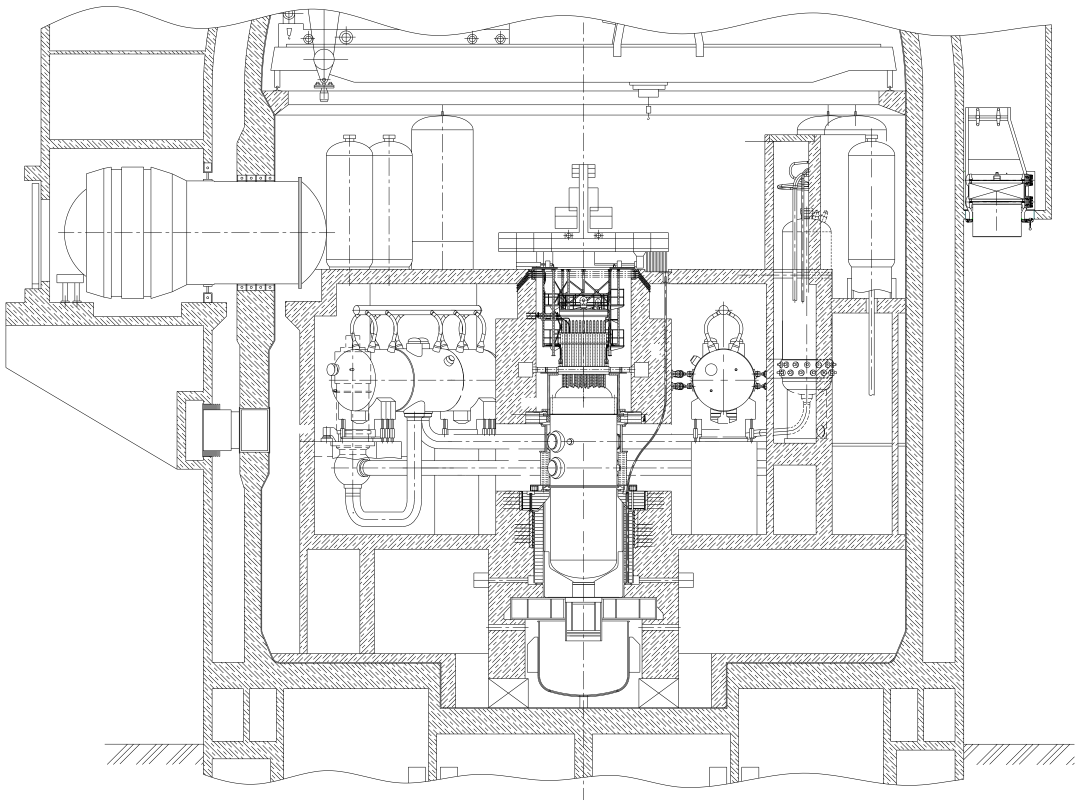


FIG. 7. Example of arrangement of WWER equipment in the reactor building.

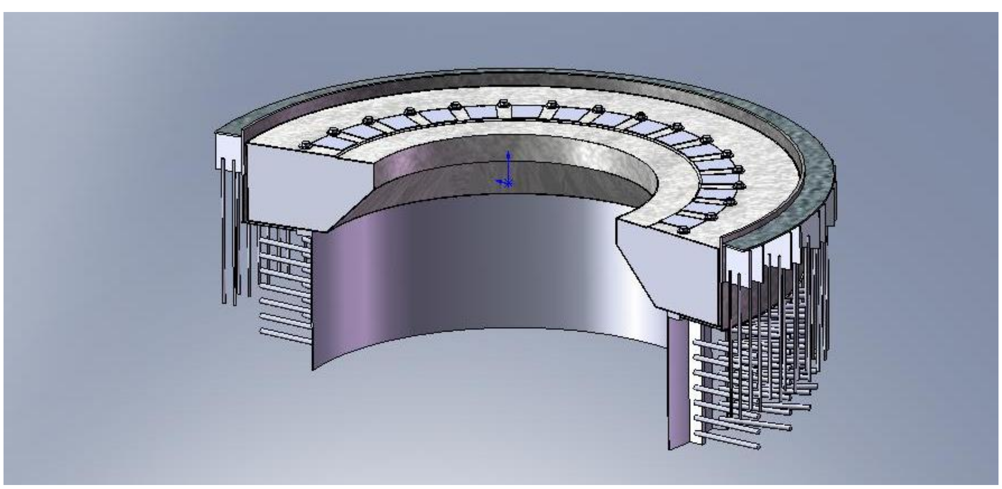


FIG. 8. Supporting truss.

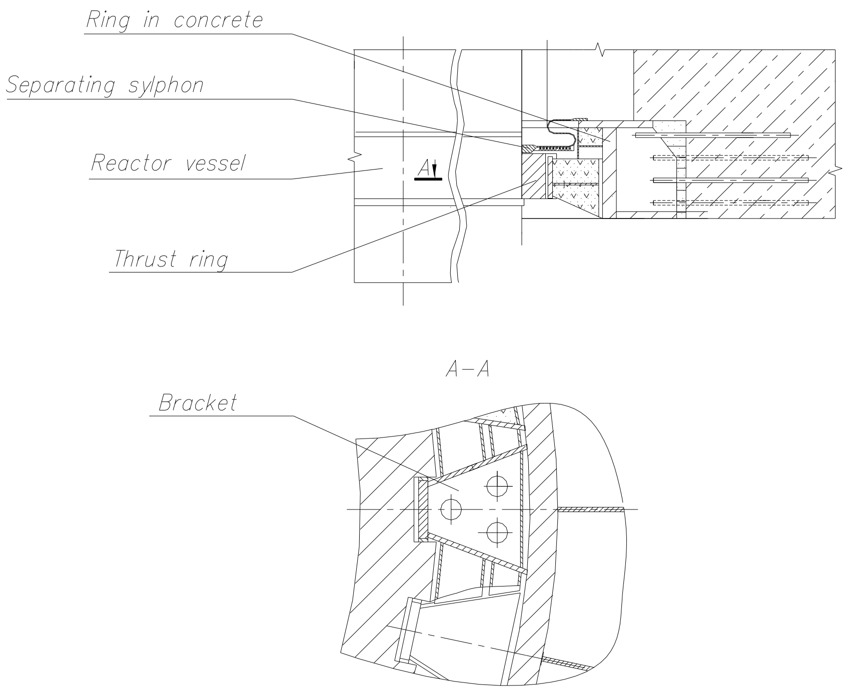


FIG. 9. Drawing of the supporting truss.

5.1.4. It is recommended to form the pressurizer support system using two independent parts: the lower support and the upper supporting unit.

For the lower support is recommended to use a cylinder shell with its upper end welded to the bottom of the pressurizer, and with its lower end to the cast-in ring. A drawing of the lower support is shown in Figure 10.

The upper supporting unit is intended to bear horizontal forces only; it prevents the pressurizer body from overturning. The recommended design of the upper supporting unit includes a thrust ring fitted onto the top part of the pressurizer body, brackets welded evenly along the perimeter of the thrust ring, stops welded to the cast-in ring, and external beams with a box cross section. A drawing of the upper support is shown in Figure 11.

5.1.5. Other equipment supports are usually structures composed of welded plates and sheets. In zones of attachment of a support to equipment housing, use of a support shoe (saddle) is recommended. An example of equipment support is shown in Figure 12.

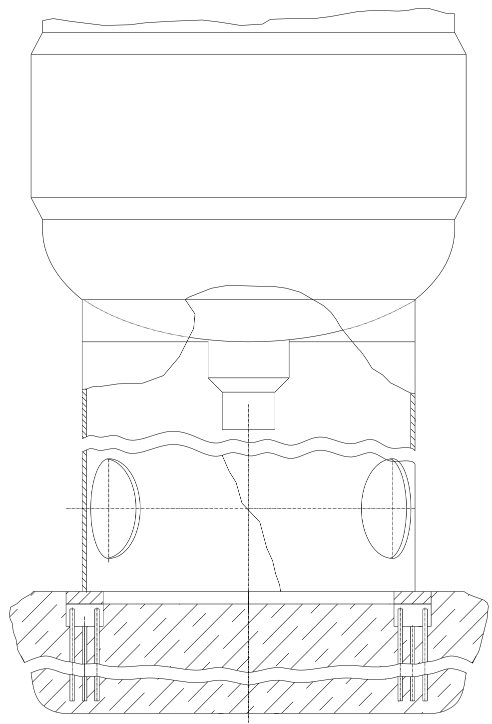


FIG. 10. Drawing of components of the pressurizer lower support.

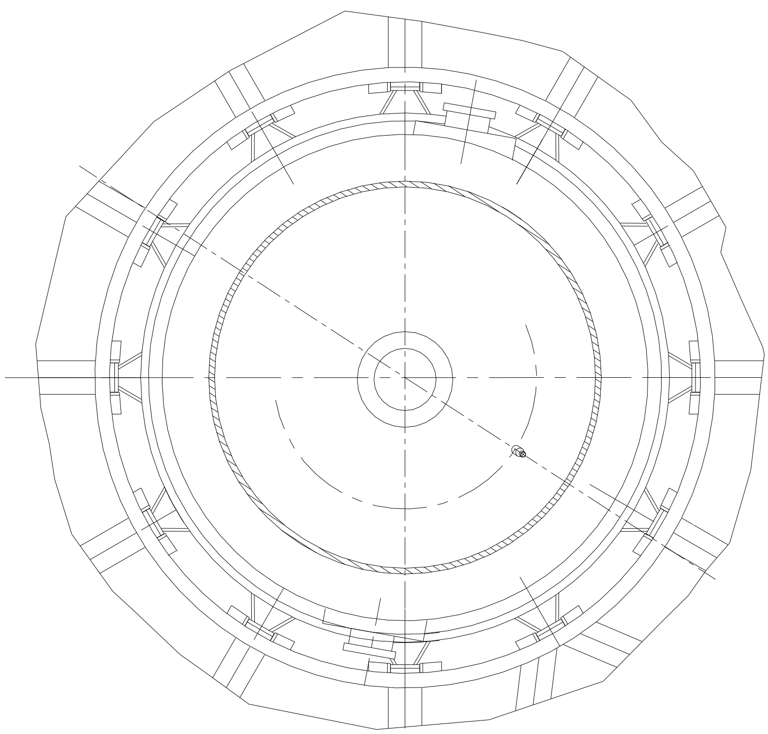


FIG. 11. Drawing of components of the pressurizer upper supporting unit.

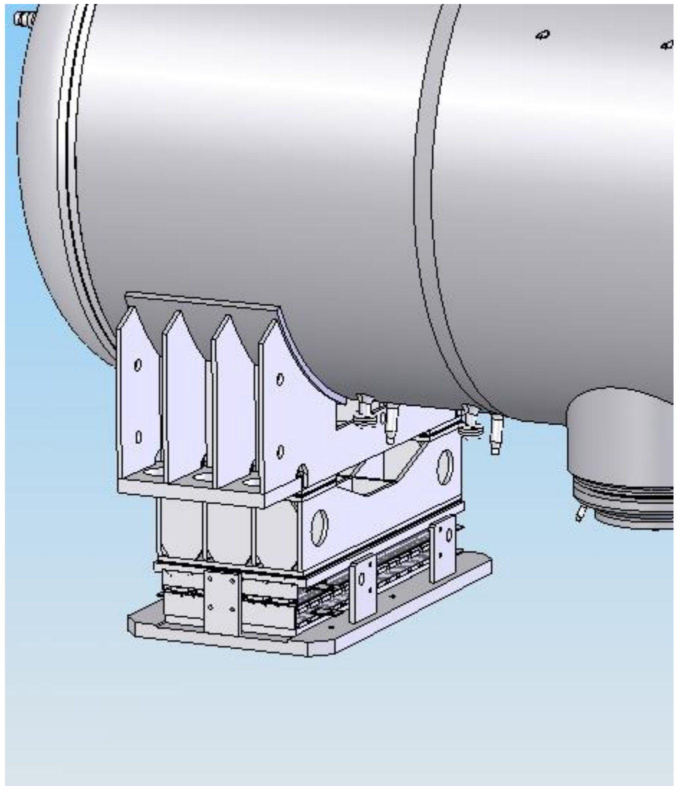


FIG. 12. Support of the steam generator.

5.2. PIPELINE SUPPORTS

5.2.1. Types and structure designs of standard supports and hangers for pipelines as well as their components (standardized units and parts) are given in Appendix O-2 of SPiR-O-2008 [5].

Other catalogues containing qualified standard supports and hangers for pipelines, their standardized units and parts are allowed to be used.

5.2.2. Structure designs of standard supports and hangers shall be chosen from Appendix O-2 of SPiR-O-2008 [5] or catalogues in regard to allowable loads.

Catalogues of standard supports and hangers for pipelines shall regulate structure design of a support or hanger (type, version, material), basic dimensions as well as parameters of to be supported pipeline (diameter, pressure, temperature and material).

5.2.3. Standard support units shall be formed combining standard supports or standard components of supports using standard methods for attachment of supports to building structures in order to ensure specified design requirements to certain pipelines and supporting systems.

5.2.4. The following standard components of supports are recognized:

— Components of weld-on supports (stanchions);
— Clamp bases;
— Clamp blocks;
— U-shaped bolts (clamps);
— Variable spring blocks;
— Constant spring blocks;
— Guide plates;
— Supporting beams;
— Roller bearing blocks ;
— Shock absorbers (snubbers);
— Dampers;
— Rigid struts;
— Fins;
— Threaded tie-rods;
— Rod couplings, turnbuckles;
— Connecting parts (eye nuts, eye plates, clevises);
— Trapezes, pipe saddles, pipe trays, lift-off restraints.

Standard components of supports are shown in Figures 13–25.

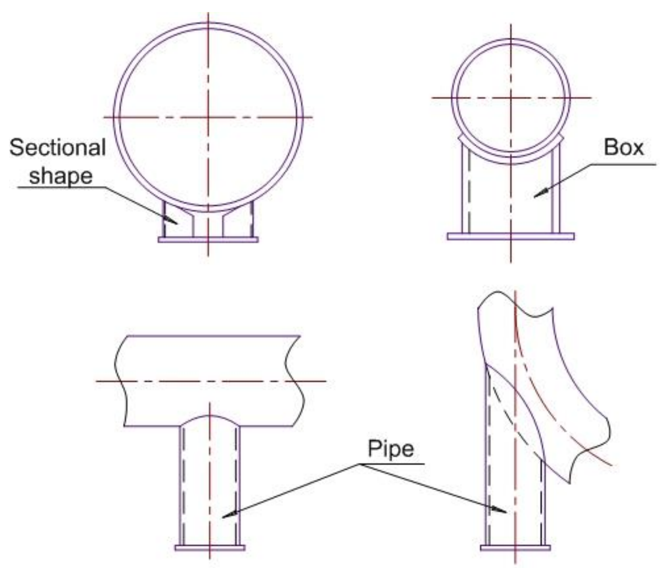


FIG. 13. Components of weld-on supports (stanchions) for horizontal pipelines and connector bends.

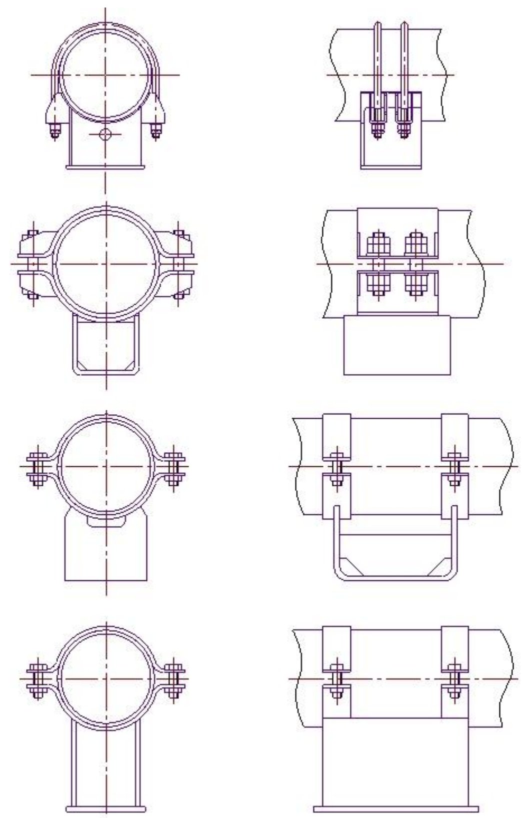


FIG. 14. Clamp bases of various designs.

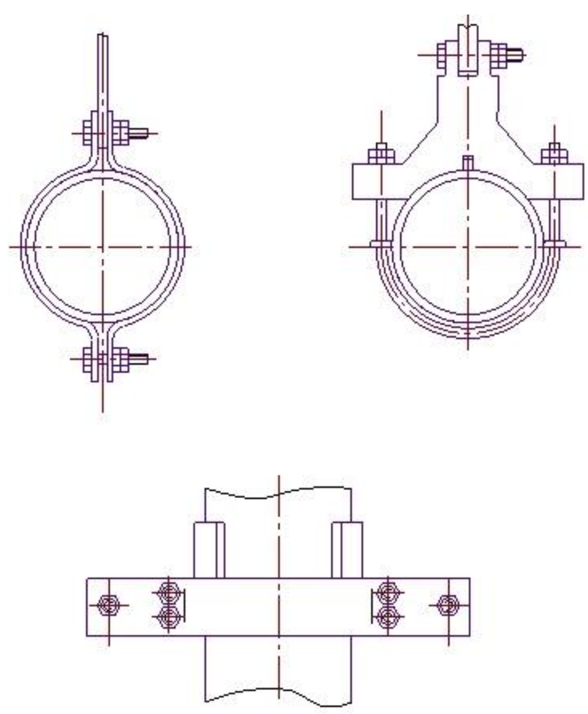


FIG. 15. Clamps of various designs.

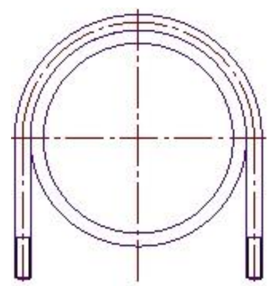


FIG. 16. U-shaped bolt (clamp).

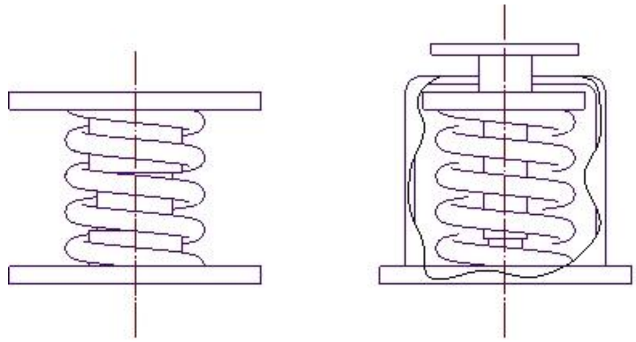


FIG. 17. Variable spring blocks of various designs (with and without housing).

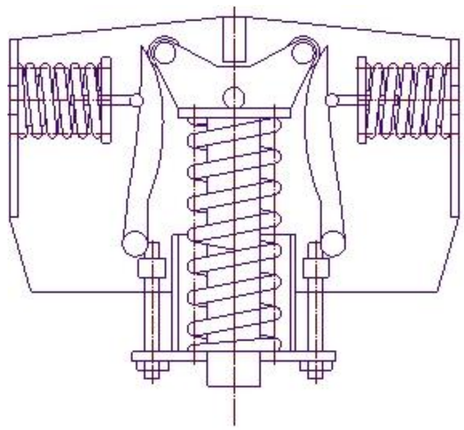


FIG. 18. Constant spring block.

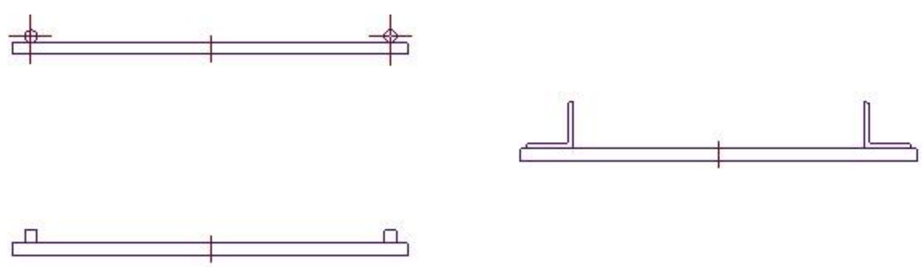


FIG. 19. Guide plates of various designs.

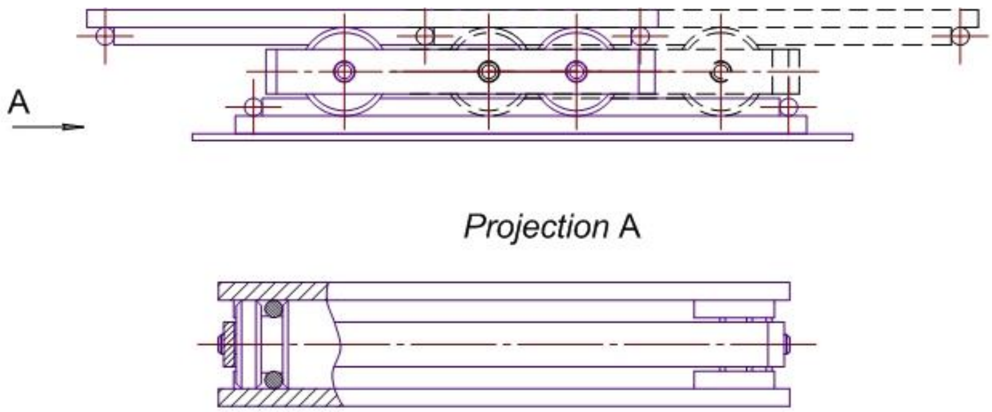


FIG. 20. Roller bearing blocks.

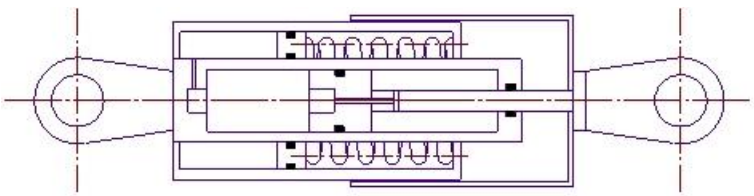


FIG. 21. Shock absorber (snubber).

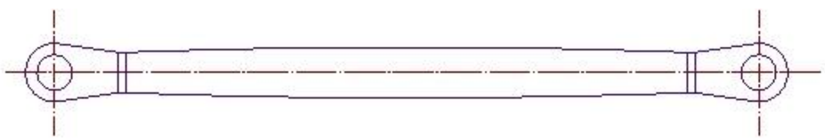


FIG. 22 . Rigid strut.

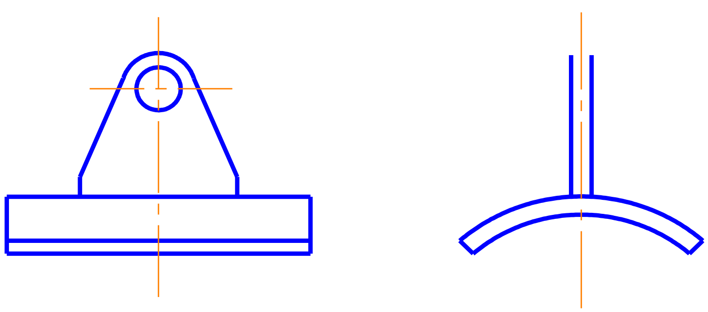


FIG. 23. Fin.

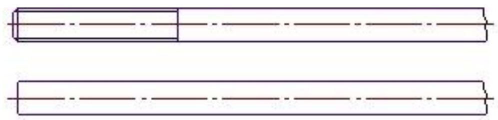


FIG. 24. Threaded tie-rod.

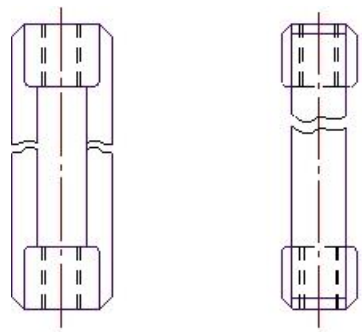


FIG. 25. Rod coupling and turnbuckle.

5.2.5. The following types of standard supports composed from standard components, units and parts are recognized:

— Fixed supports (welded and clamped);

— Sliding supports (welded and clamped);

— Guide supports (welded and clamped);

— Roller supports;

— Spring supports;

— Constant supports;

— Dynamic supports (shock absorbers, dampers, struts).

Fixed supports are formed by welding components of welded supports or clamp bases to the under-support building structure.

Sliding supports are formed by installing components of welded supports or clamp bases onto a sliding plate fixed on the under-support building structure.

Guide supports are formed by installing components of welded supports or clamp bases onto a guiding plate fixed on the under-support building structure.

Roller supports are formed by installing components of welded supports or clamp bases onto roller block(s) fixed on the under-support building structure.

Spring supports are formed by installing components of welded supports or clamp bases onto a spring cartridge fixed on the under-support building structure.

Constant supports are formed by installing components of welded supports or clamp bases onto a constant spring cartridge fixed on the under-support building structure.

Dynamic supports are formed by combining shock absorbers, dampers or rigid struts with clamp blocks.

Specific combinations of support structure components, units and parts used in standard supports designs are given in relevant catalogues.

5.2.6. During design of a pipeline it is recommended to avoid the use of fixed supports. To compensate for pipeline thermal expansion, it is recommended to use variable spring supports and hangers or (at large thermal movements) constant spring supports and hangers.

5.2.7. A hanger structure is usually assembled from standard parts (tie-rods, rod couplings and turnbuckles) that allow for the design of chains of the required length and complexity.All tie-rods within a chain shall be of the same diameter.

Hanger type is characterised by the structure of the attachment unit to the pipeline, and the hanger chain structure.

5.2.8. Rod couplings are used for extension of hanger tie-rod length. Both ends of rod coupling shall have right-hand threads. Two lock nuts shall be ordered for each rod coupling.

5.2.9. Turnbuckles are used to change the length of tie-rods and also to regulate the tightening of springs. Turnbuckles shall have a right-hand thread on one side and a left-hand thread on the other side. Turnbuckles shall be locked with one locking nut from the right-hand side thread.

In order to regulate the tightening of springs, it is recommended to provide one turnbuckle for one or two spring blocks, and two turnbuckles for three or more spring blocks. If turnbuckles are used for the complete tightening of springs, it is required to provide one turnbuckle for every spring block.

5.2.10. In hangers, depending on allowable load spring blocks and tie-rods of several dimensions are used. The choice of spring block defines the tie-rods for the whole chain. In order to increase the allowable load on a hanger, cross beams are used. Possible variants of connecting tie-rods to a cross beam are presented in relevant catalogues.

5.2.11. Examples of standard pipeline hangers are shown in Figures 26–28, where the following designations are used:

1 = pipeline

2 = attachment unit (clamp)

3 = tie-rod

4 = rod coupling

5 = turnbuckle

6 = hanger spring block

7 = cross beam

8 = support spring block

9 = beam

10 = spring block of constant hanger.

5.2.12. Installation of non-standard supports is allowed if technically required or if it is not practical to install standard supports.

5.2.13. When repairing standard supports, the need may arise to reconstruct, modernize, or reinforce, etc. In this case, a support may change from a standard category support into a non-standard category support.

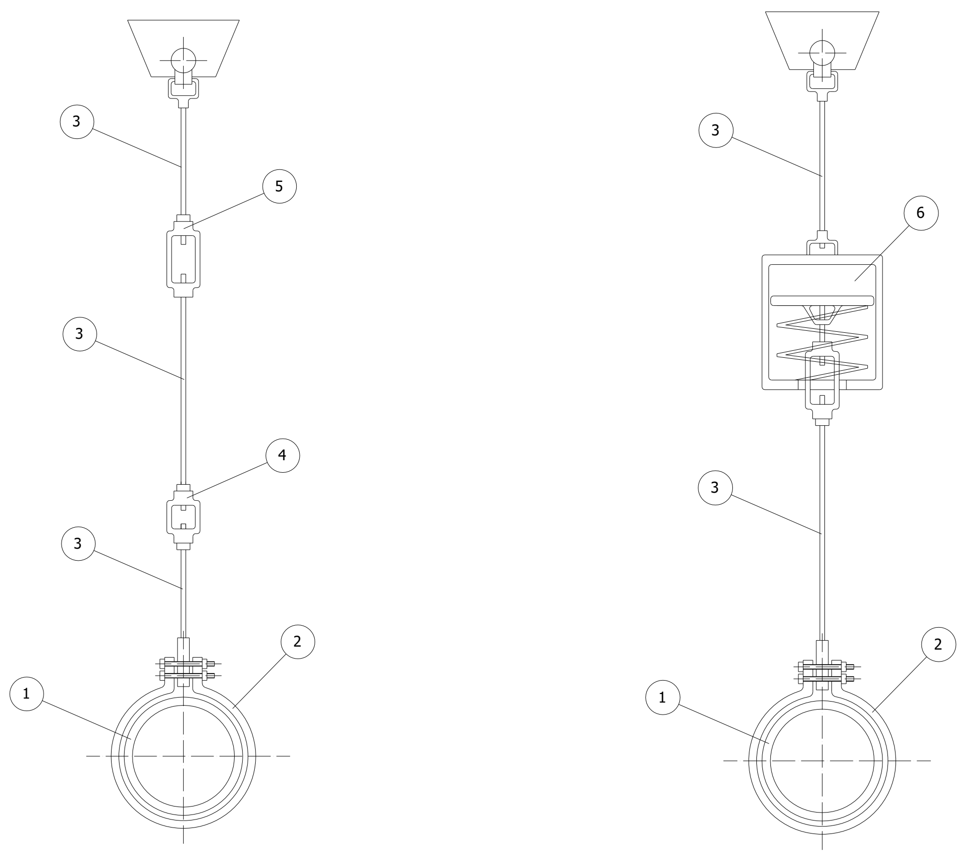


FIG. 26. Rigid and spring hangers with single tie-rods.

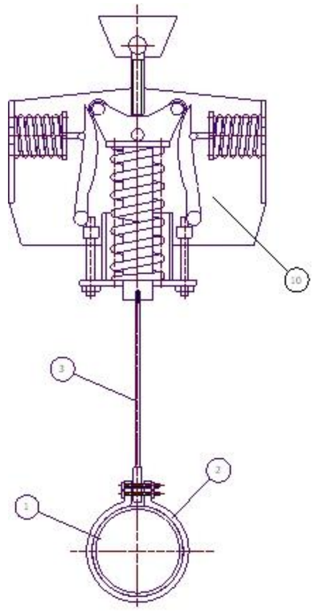


FIG. 27. Constant hanger with single tie-rod.

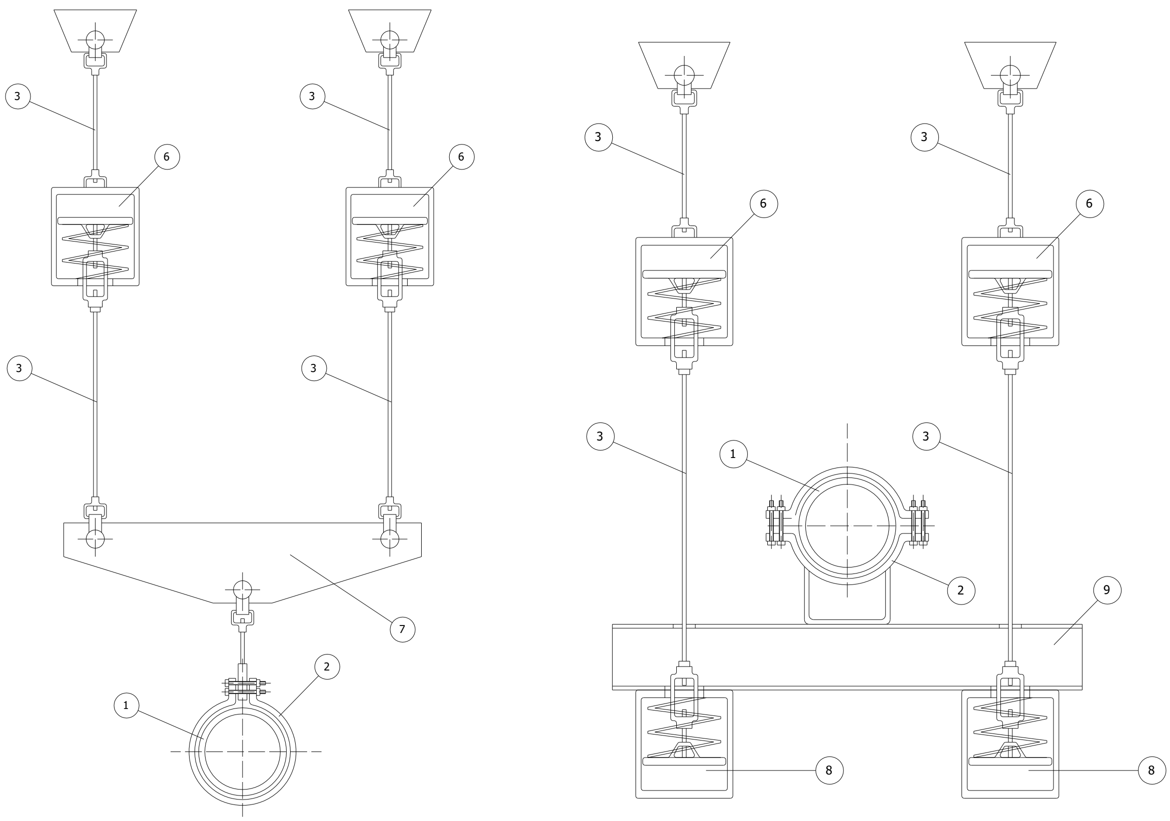


FIG. 28. Spring hangers with cross beam and beam.

5.3. STRUCTURAL MATERIALS

5.3.1. Materials for manufacturing supports shall be chosen taking into account physical and mechanical properties required in a project, manufacturability and usability at operational conditions within the specified life time.

5.3.2. For manufacturing, installation and repair of supports, basic materials specified in Section O-2000 of SPiR-O-2008 [5] shall be used.

5.3.3. Quality and properties of basic materials (semi-finished items and blanks) shall meet requirements given in Appendix O-1 (for Russian materials) or Appendix O-6 (for foreign materials) of SPiR-O-2008 [5] and shall be confirmed by manufacturer certificates.

5.3.4. The organization manufacturing supports shall ensure incoming quality examination of incoming basic materials for conformance with requirements of technical specifications and/or requirements given in above mentioned regulatory documents. If necessary, by agreement with the material knowledgable organisation, it is allowed to establish higher requirements for materials (than that given in standards or technical specifications).

5.3.5. Methods and extent of examination of basic materials shall be established (and indicated) in design documentation on the basis of technical specifications or requirements given in Russian regulatory documents including PNAE G-7-002-86 [2], PNAE G-7-008-89 [3], SPiR-O-2008 [5].

5.3.6. Incoming examination of welding materials shall be carried out in accordance with requirements and directions of Russian regulatory documents including PNAE G-7-002-86 [2], PNAE G-7-008-89 [3], SPiR-O-2008 [5].

5.3.7. Materials and semi-finished items intended for manufacturing of supports shall be protected against damage during transportation and storage.

Methods for protecting materials and semi-finished items during transportation and storage shall be developed by the organizations manufacturing supports.

5.4. ALLOWABLE STRESSES AND STRENGTH ANALYSIS

5.4.1. Rated allowable stresses

5.4.1.1 .Rated allowable stresses for components of shell type supports are calculated in accordance with relationship:

$$[\sigma] = \min\left(\frac{R_m^T}{2{,}6}; \frac{R_{p0,2}^T}{1{,}5}\right) \tag{1}$$

5.4.1.2. Rated allowable stresses for components of linear type supports are calculated in accordance with relationship:

$$[\sigma] = 0{,}75 \times \min\left(\frac{R_m^T}{2{,}0};\ \frac{R_{p0,2}^T}{1{,}5}\right) \text{ for components loosened by holes for pins, studs, etc.}$$

$$[\sigma] = \min\left(\frac{R_m^T}{2{,}0};\ \frac{R_{p0,2}^T}{1{,}5}\right) \text{ in other cases.} \qquad (2)$$

5.4.1.3. Rated allowable stresses for threaded parts are calculated in accordance with relationship:

$$[\sigma]_w = \min\left(\frac{R_m^T}{2{,}0};\ \frac{R_{p0,2}^T}{1{,}5}\right) \qquad (3)$$

5.4.1.4. Rated allowable stresses for shock absorber components made from casts are calculated in accordance with relationship

$$[\sigma] = 0{,}75 \times \min\left(\frac{R_m^T}{2{,}0};\ \frac{R_{p0,2}^T}{1{,}5}\right) \qquad (4)$$

5.4.1.5. Rated allowable stresses for metal of welded joints with full penetration are calculated in accordance with the same relationship as for the basic metal.

5.4.1.6. Rated allowable stresses for corner welds and lap-welded joints with incomplete penetration are calculated in accordance with relationship

$$[\sigma]_j = \frac{R_m^T}{1{,}6} \text{ for weld metal}$$

$$[\sigma] = \min\left(\frac{R_m^T}{2{,}6};\ \frac{R_{p0,2}^T}{1{,}5}\right) \text{ for metal on the fusion boundary} \qquad (5)$$

5.4.2. Regulations on strength analysis

5.4.2.1. General regulations

Strength analysis (stress calculation) for support structures shall be carried out based on rated dimensions, with the exception of cases when consideration of tolerances leads to fundamental changes in the analytical model of a structure and, as a consequence, influences the calculation results.

The basic design loads for supports are the following:

- Weight of equipment or pipeline and its contents, including heat insulation and other connected items;
- Support structure's own weight;
- Bolt and pin tightening loads;
- Loads and forces transmitted on the support from the supported element (including loads due to thermal expansion of the element);
- Loads due to environmental effects including wind and snow loads;
- External dynamic loads including seismic loads, loads due to airplane fall or shock wave action;
- Loads and forces caused by relative movement of support components;
- Loads due to extension or compression of the supported element as a result of coolant pressure action on the supported element.

When carrying out stress calculation using the linear elastic approach, all stresses in the support are divided into categories. Stresses related to different categories are united into groups of stress categories.

Equivalent stresses for a group of stress categories shall be compared with allowable stresses. In the cases specially mentioned in this document, manufacturing residual stresses shall be taken into account.

When carrying out stress calculations for supports of different safety classes, various stress classification divisions are used.

5.4.2.2. Stress classification

The following stress categories are used:

- Total membrane (total mean) stress $\sigma m\Sigma$;
- Total bending stress $\sigma b\Sigma$;
- General membrane stress σm;
- Local membrane stress σml;
- General bending stress σb;
- Local bending stress σbl;

— General thermal stress σm;

— Local thermal stress σtl;

— General kinematic stress σd;

— Local stress in concentration zone.

Total membrane (total mean) stress $\sigma_{m\Sigma}$ is caused by mechanical and kinematical loads; it is uniformly distributed on the support component cross section and is equal to the mean value of stresses in this section due these loads.

Total bending stress $\sigma_{b\Sigma}$ is caused by mechanical and kinematical loads; it varies from a minimum negative value to a maximum positive value on the support component section.

General membrane stress σ_m is caused by mechanical loads; it is uniformly distributed on the support component cross section and is equal to the mean value of stresses in this section due these loads.

Local membrane stress σmL is membrane stress caused by mechanical loads in an irregularity zone, e.g. connection of shell and flange, shell and plate, etc.

General bending stress σ_b is caused by mechanical loads; it varies from a minimum negative value to a maximum positive value on the support component cross section.

Local bending stress σ_{bL} is bending stress caused by mechanical loads in an irregularity zone.

General thermal stress σ_T is caused by a non-uniform temperature distribution on the support component volume or by differences in linear thermal expansion coefficients of dissimilar materials; in extreme cases, it results in inadmissible residual changes of support shape or dimensions. General thermal stresses are part of general kinematic stresses – see below.

Local thermal stress σ_{TL} is thermal stress caused by a non-uniform temperature distribution on the support component volume or by differences in linear thermal expansion coefficients of dissimilar materials; it does not result in inadmissible residual changes of support shape or dimensions.

General kinematic stress σ_d is caused by constraints of support movements; in extreme cases, it results in inadmissible residual changes of support shape or dimensions.

Local stress in concentration zone is stress caused by mechanical and/or kinematic loads in zones of holes, fillets, threads, etc. taking into account stress concentration.

Groups of stress categories which are used when analysing strength with regard to relevant (being under consideration) zones of support components are given in Table 1.

TABLE 1. EXAMPLES OF GROUPS OF STRESS CATEGORIES IN ZONES OF SUPPORTS

Zone of support	Loading mode	Stress categories	Calculation group of stress categories
Cylindrical or conical part of shell (plain part)	Applied forces and moments (including tightening)	σ_m	$(\sigma)_1$
		$\sigma_m \oplus \sigma_b$	$(\sigma)_2$
	Applied forces and moments (including tightening), deformation because of pressure or restraint of movements (including thermal movements)	$\sigma_m \oplus \sigma_b \oplus \sigma_d$ (including σ_T)	$(\sigma)_R$
		$\sigma_m \oplus \sigma_b \oplus \sigma_d$ (including σ_T) $\oplus\ \sigma_{TL}$	(σ_{aF})
Cylindrical or conical part of shell (in discontinuity zone)	Applied forces and moments (including tightening)	$\sigma_{mL} \oplus \sigma_b$	$(\sigma)_2$
	Applied forces and moments (including tightening), deformation because of pressure or restraint of movements (including thermal movements)	[σ_m or σ_{mL}] $\oplus\ \sigma_b \oplus \sigma_{bL} \oplus \sigma_d$ (including σ_T)	$(\sigma)_R$
		[σ_m or σ_{mL}] $\oplus\ \sigma_{pb} \oplus \sigma_{bL} \oplus \sigma_d$ (including σ_T) $\oplus\ \sigma_{TL}$ in consideration of concentration	(σ_{aF})
Plate	Applied forces and moments (including tightening)	σ_m	$(\sigma)_1$
		$\sigma_m \oplus \sigma_b$	$(\sigma)_2$
	Applied forces and moments (including tightening), deformation because of pressure or restraint of movements (including thermal movements)	$\sigma_m \oplus \sigma_b \oplus \sigma_d$ (including σ_T)	$(\sigma)_R$
		$\sigma_m \oplus \sigma_b \oplus \sigma_d$ (including σ_T) $\oplus\ \sigma_{TL}$ in consideration of concentration	(σ_{aF})

TABLE 1. EXAMPLES OF GROUPS OF STRESS CATEGORIES IN ZONES OF SUPPORTS (cont.)

Zone of support	Loading mode	Stress categories	Calculation group of stress categories
Rod or beam	Applied forces and moments (including tightening)	σ_m	$(\sigma)_1$
		$\sigma_m \oplus \sigma_b \oplus \tau_s$	$(\sigma)_2$
	Applied forces and moments (including tightening), deformation because of pressure or restraint of movements (including thermal movements)	$\sigma_{m\Sigma} \oplus \sigma_{b\Sigma} \oplus \tau_s$	$(\sigma)_R$
		$\sigma_{m\Sigma} \oplus \sigma_{b\Sigma} \oplus \tau_s$ in consideration of concentration	(σ_{aF})
Bolts or stud	Tightening, applied forces and moments, thermal and kinematic loads	σ_{mw}	$(\sigma)_{3w}$
		$\sigma_{mw} \oplus \sigma_{bw} \oplus \tau_{sw}$	$(\sigma)_{4w}$
		$\sigma_{mw} \oplus \sigma_{bw} \oplus \tau_{sw}$ in consideration of concentration	$(\sigma_{aF})_w$

In Table 1 the following designations are used in addition to the indicated above:

σ_{bw} = mean bending stress in threaded part cross section (bolt or stud)

σ_{mw} = mean tensile stress in threaded part cross section

τ_{sw} = torsion stress in threaded part cross section.

Sign $\oplus$ in Table 1 means that summation of stresses can be carried out on the basis of the strength theory chosen.

5.4.2.3. Static analysis

(a) Elastic analysis

Values of groups of stress categories calculated when performing static strength analysis for support components (except corner welds with incomplete penetration) shall not exceed those given in Tables 2 and 3. It is allowed to determine [σ] and $[\sigma]_w$ not for room temperature but for maximum average temperature on the wall thickness (cross section) of the support component for the loading mode under consideration.

Value of shape factor *k* used in Table 2 depends on structural shape of a component and loading type. Values of shape factor *k* are given in Table 4. For cases not mentioned in Table 4, the value k = 1,3 shall be accepted.

TABLE 2. LIMIT VALUES ON GROUPS OF STRESS CATEGORIES FOR SUPPORT COMPONENTS EXCEPT BOLTS AND STUDS

Loading mode	$(\sigma)_1$	$(\sigma)_2$	$(\sigma)_R$
NOC	[σ]	k [σ]	$\left(2{,}5-\frac{R^T_{p0,2}}{R^T_m}\right)R^T_{p0,2}$, but no more $2R^T_{p0,2}$
NOC (S)	1,1 [σ]	1,1 k [σ]	$\left(2{,}5-\frac{R^T_{p0,2}}{R^T_m}\right)R^T_{p0,2}$, but no more $2R^T_{p0,2}$
VNOC	1,2 [σ]	1,2 k [σ]	–
DA	1,4 [σ]	1,4 k [σ]	–

TABLE 3. LIMIT VALUES ON GROUPS OF STRESS CATEGORIES FOR BOLTS AND PINS OF SUPPORTS

Loading mode	$(\sigma)_{3w}$	$(\sigma)_{4w}$	τ_{th}
NOC	$[\sigma]_w$	1,3 $[\sigma]_w$	0,5 $[\sigma]_w$
NOC (S)	1,1 $[\sigma]_w$	1,4 $[\sigma]_w$	0,55 $[\sigma]_w$
VNOC	1,2 $[\sigma]_w$	1,6 $[\sigma]_w$	0,6 $[\sigma]_w$
DA	1,4 $[\sigma]_w$	1,8 $[\sigma]_w$	0,7 $[\sigma]_w$

TABLE 4. VALUES OF SHAPE FACTOR *k*

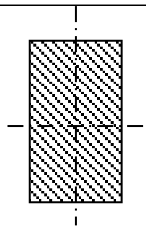
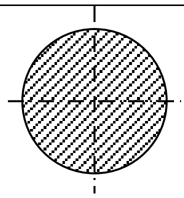
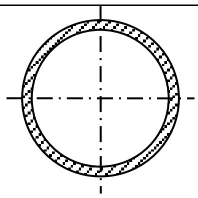
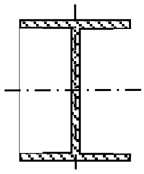
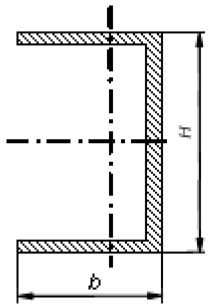
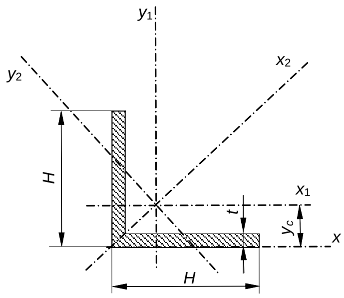
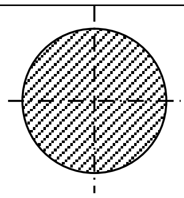
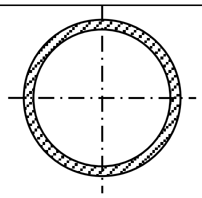
Component shape	Loading type	Type of cross section	*k*
Beam	General bending + tension (compression)		1,5
			1,7
			1,3
	General bending (relative to horizontal axis) + tension (compression)		1,0
	General bending (relative to vertical axis) + tension (compression)		1,5

TABLE 4. VALUES OF SHAPE FACTOR *k* (cont.)

Component shape	Loading type	Type of cross section	*k*
Beam	General bending (relative to horizontal axis) + tension (compression)		1,0
	General bending (relative to vertical axis) + tension (compression)		See Figure 29
	General bending (relative to x_1 or y_1 axis) + tension (compression)		2,0
	General bending (relative to x_2 or y_2 axis) + tension (compression)		1,5
	Torsion		1,3
			1,0
Plate	General bending + tension (compression)	–	1,5
Shell, plane curved	General bending + tension (compression)	–	1,3
	Local bending	–	1,5

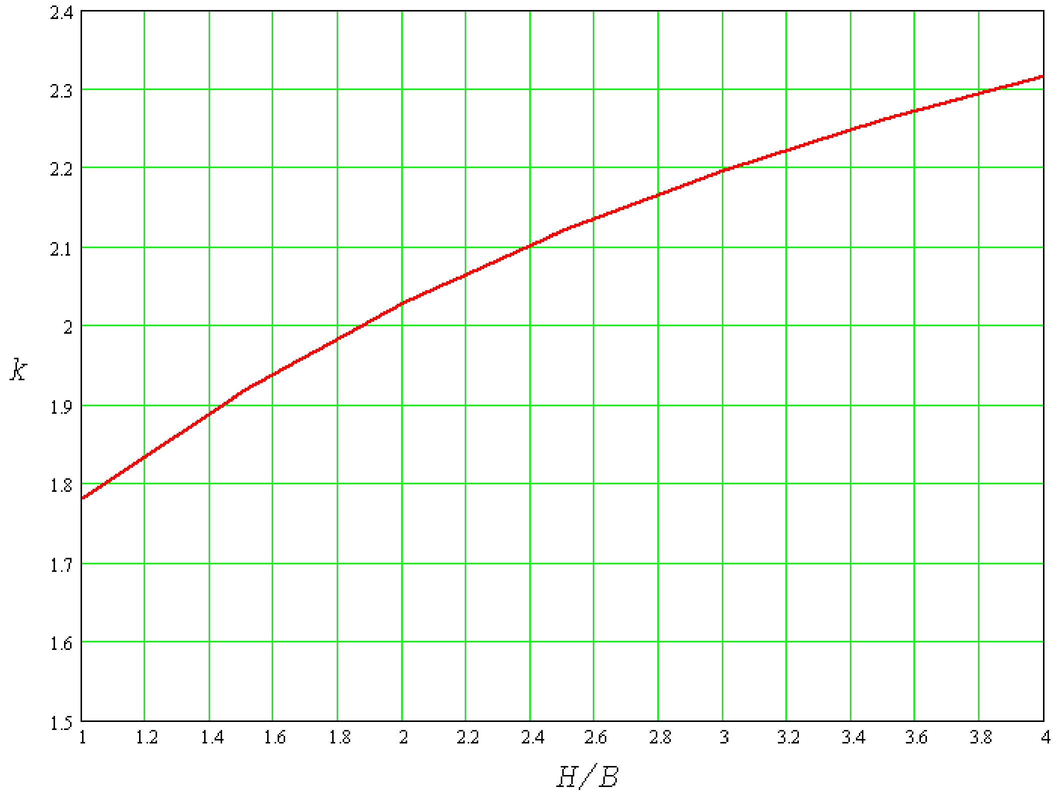


FIG. 29. Values of shape factor k *for channel beam.*

Average crumpling stresses on contact surface shall not exceed 1,5 $R^T_{p0,2}$. If distances from an edge of load application zone to free edge exceed dimensions of load application zone, then allowable stresses may be increased by 25%.

Average tangential stresses on thread of support component shall not exceed the values given in Table 3.

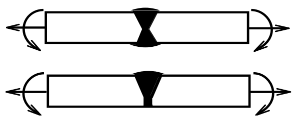
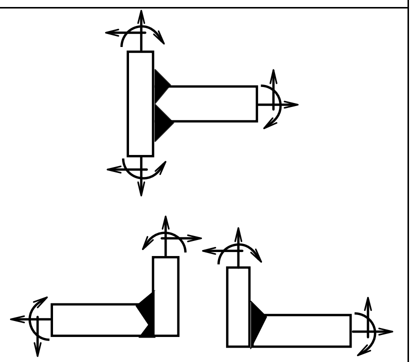
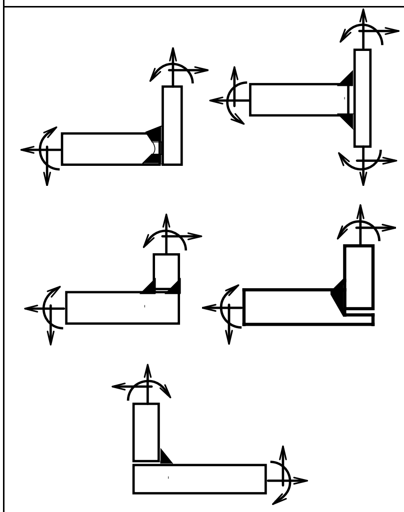
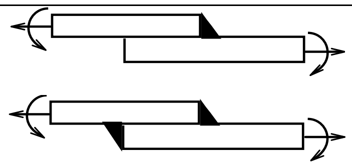
Maximum reference tangential stresses in reference section of corner weld with incomplete penetration shall not exceed the values given in Table 5.

TABLE 5. LIMIT STRESS VALUES IN REFERENCE SECTION OF CORNER WELD WITH INCOMPLETE PENETRATION

Loading mode	τ_j
NOC	0,5 $[\sigma]_j$
NOC (S)	0,55 $[\sigma]_j$
VNOC	0,6 $[\sigma]_j$
DA	0,7 $[\sigma]_j$

For welded joints, the value of rated allowable stress [σ] or $[\sigma]_j$ used when analyzing shall be decreased according to strength reduction coefficient φ_j, which depends on implementation of the welded joint, its loading pattern and the extent of non-destructive examination of the welded joint at manufacturing. Values of strength reduction coefficient φ_j are given in Table 6.

TABLE 6. VALUES OF STRENGTH REDUCTION COEFFICIENT IN WELDED JOINTS $\varphi_{\boldsymbol{j}}$

Joint type	Loading type	Loading pattern	Examination method	Extent of examination	φ_j
Butt joint with complete penetration	Tension, bending		VE + (RE or USE)	100 %	1,0
				≥ 50 %	0,9
				≥ 25 %	0,85
				≥ 10 %	0,8
				< 10 % or VE only	0,7
Corner and T-joint with complete penetration	Tension, bending, shear			100 %	0,9
				≥ 10 %	0,8
				< 10 % or VE only	0,7
Corner and T-joint with incomplete penetration			VE	100 %	0,8
				≥ 10 %	0,7
				< 10 %	0,6
Lap joint				100 %	0,8
				≥ 10 %	0,7
				< 10 %	0,6
Notes: RE – radiographic examination, USE – ultrasonic examination, VE – visual examination					

(b) Bearing capacity analysis (limit analysis)

Bearing capacity analysis (limit analysis) is allowed as an alternate check of static strength on groups of stress categories $(\sigma)_1$ and $(\sigma)_2$.

The check of static strength on other groups of stress categories shall be carried out by elastic analysis in accordance with requirements of Sub-sub-clause 5.4.2.3(a). Bearing capacity analysis (limit analysis) may be carried out for all support components except for threaded joints and corner welds with incomplete penetration, taking into account monotone changing mechanical loads at all operation modes (loading modes).

Threaded joints and corner welds with incomplete penetration shall be analyzed by calculations using a linear elastic approach.

In this analysis, the combination of mechanical loads applied to a support or its component, which are changed in proportion to one parameter – load coefficient, shall be considered.

When carrying out bearing capacity analysis (limit analysis), the following are used:

- Material tensile diagram is considered as an ideal elastic-plastic one (see Figure 30) with fictitious yield stress $\sigma_Y = 1{,}5[\sigma]$ for basic metal and $\sigma_Y = 1{,}5\varphi_j\,[\sigma]$ for metal of weld joint with complete penetration (values of φ_j are given in Table 6).

- Yield criterion is the von Mises criterion.

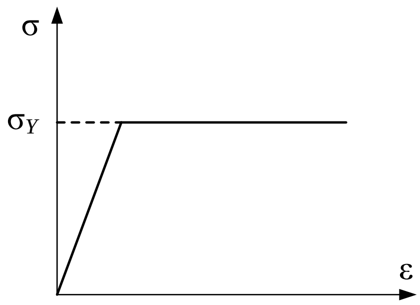


FIG. 30. Ideal elastic-plastic tensile diagram of material.

Limit load (load combination) F_L is determined as the maximum load (load combination) for which the stress field exists in any point of the support component satisfying static equilibrium equations.

Value of applied load (load combination) F shall not exceed value of limit load (load combination) F_L taking into account safety factor:

$$F \le \frac{F_L}{n_{PL}} \tag{6}$$

where n_{PL} denotes safety factor which depends on support loading mode and may be taken in accordance with Table 7.

TABLE 7. VALUES OF SAFETY FACTOR n_{PL}

Loading mode	n_{PL}
NOC	1,50
NOC (S)	1,36
VNOC	1,25
DA	1,07

5.4.2.4. Stability analysis

(a). Elastic analysis for supports of linear type

This stability analysis procedure is applicable for supports of linear type under axial compression forces. When stability is analyzed, the value of critical load shall be determined by calculations in accordance with Table 8.

In Table 8, the following designations are used:

F = axial compression force, N

$I_{\min}$ = minimum inertia moment of support cross section, mm^4

E = Young modulus, MPa

L = reference length of support, mm.

TABLE 8. FORMULAS FOR CALCULATION OF CRITICAL LOAD

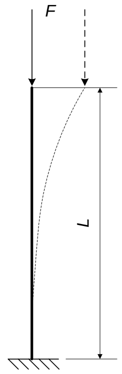
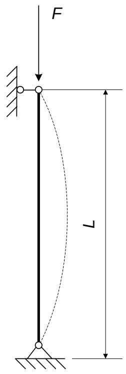
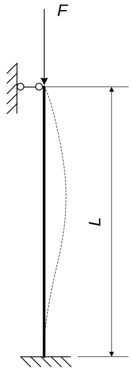
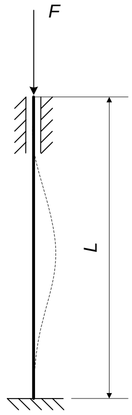
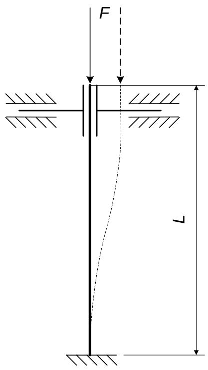
Component shape	Loading type	Analytical model	Formula for calculation of critical load
Beam	Axial compression force		$F_{кр} = \frac{\pi^2 EI_{\min}}{4L^2}$
			$F_{кр} = \frac{\pi^2 EI_{\min}}{L^2}$
			$F_{кр} = \frac{\pi^2 EI_{\min}}{0{,}489L^2}$
			$F_{кр} = \frac{\pi^2 EI_{\min}}{0{,}25L^2}$

TABLE 8. FORMULAS FOR CALCULATION OF CRITICAL LOAD (cont.)

Component shape	Loading type	Analytical model	Formula for calculation of critical load
Beam	Axial compression force		$F_{кр} = \frac{\pi^2 EI_{\min}}{L^2}$

In the case is not provided by analytical models of Table 8, the critical load shall be determined by other calculation methods, e.g. finite element method (FEM).

(b) Elastic analysis for supports of shell type

This procedure is applicable for supports subjected to loading by arbitrary load vector $\overline{F}$. Only forces that are dangerous from a loss of stability point of view shall be considered:

$$\overline{F} = \begin{Bmatrix} Fx \\ Fy \\ Fz \\ Mx \\ My \\ Mz \end{Bmatrix}$$

The main task of the calculation is to determine critical values of $\overline{F}_{кр}$.

Calculation may be carried out analytically or by FEM.Support stability is ensured provided the following condition is satisfied:

$$\overline{F} \leq \frac{\overline{F}_{кр}}{2{,}5\,\varsigma} \tag{7}$$

where ζ denotes coefficient which takes into account support manufacturing imperfections that correspond to differences between actual and rated (design) dimensions of support.

Values of coefficient ζ are given in Table 9.

TABLE 9. VALUES OF COEFFICIENT ζ

Ratio of imperfection size to length of a support,%	ζ
less than 1	1,0
1	3,0
2	5,0

5.4.2.5. Cyclic strength analysis

The allowable number of cycles for given stress amplitudes, or the allowable stress amplitudes for a given number of cycles shall be determined based on

— Reference fatigue curves describing in the scope of their applicability a dependence between allowable amplitudes of reference elastic stresses and allowable number of cycles.

— Formulas connecting allowable amplitudes of reference elastic stresses and allowable number of cycles.

Stress amplitude during operation shall not exceed allowable stress amplitude $[\sigma_{aF}]$ obtained for given number of cycles N and maximum cycle stress $(\sigma_F)_{\max}$. If stress amplitude and maximum cycle stress are given, then the number of cycles during operation N shall not exceed allowable number of cycles $[N]$.

If the loading process consists of a row of cycles, each characterized by stress amplitude $(\sigma_{aF})_i$, maximum stress $(\sigma_F)_{\max i}$ and relevant number of cycles N_i, then the strength condition on cumulative fatigue shall be satisfied.

Allowable amplitude of reference elastic stress $[\sigma_{aF}]$ or allowable number of cycles $[N]$ at maximum cycle stress $(\sigma_F)_{\max}$ for steels at $[N] \leq 10^{12}$ is equal to the minimum of two values obtained from Eqs. (8) and (9):

$$[\sigma_{aF}] = \frac{E^T e_c^T}{(4\,n_N[N])^m} + \frac{R_c^T - [\sigma_{F\max}] i_\sigma}{(4n_N[N])^{m_e} - i_\sigma} \qquad (8)$$

where the following shall be accepted:

$[\sigma_{F\max}] = R_p^T$ at $[\sigma_{F\max}] \geq R_p^T$

$i_\sigma = 0$ at $[\sigma_{aF}] \geq [\sigma_{F\max}]$ or $[\sigma_{aF}] \geq R_p^T$

$i_\sigma = 1$ at $[\sigma_{aF}] < [\sigma_{F\max}] < R_p^T$

$$[\sigma_{aF}] = \frac{E^T e_c^T}{n_\sigma (4[N])^m} + \frac{R_c^T - n_\sigma [\sigma_{F\max}] i_\sigma}{n_\sigma \left((4[N])^{m_e} - i_\sigma\right)} \qquad (9)$$

where the following conditions are satisfied:

$n_\sigma\,[\sigma_{F\,\max}] = R_p^T$ at $n_\sigma\,[\sigma_{F\,\max}] \geq R_p^T$

$i_\sigma = 0$ at $[\sigma_{aF}] \geq [\sigma_{F\,\max}]$ or $n_\sigma\,[\sigma_{aF}] \geq R_p^T$

$i_\sigma = 1$ at $n_\sigma\,[\sigma_{aF}] < n_\sigma\,[\sigma_{F\,\max}] < R_p^T$.

Here n_σ, n_N designate safety factors on stresses and on number of cycles accordingly; m, m_e denote material properties,

$$R_c^T = R_m^T\,(1 + 1{,}4 \cdot 10^{-2} Z^T) \text{ means strength characteristic} \qquad (10)$$

e_c^T is plasticity characteristic defined by formulas:

$$e_c^T = 1{,}15\lg\frac{100}{100 - Z_c^T} - \frac{\left|(\sigma_F^*)_{\max}\right| - R_{p0,2}^T}{2E^T} \qquad \text{if } \left|(\sigma_F^*)_{\max}\right| \geq R_{p0,2}^T \qquad (11)$$

$$e_c^T = 1{,}15\lg\frac{100}{100 - Z_c^T} \qquad \text{if } \left|(\sigma_F^*)_{\max}\right| < R_{p0,2}^T \qquad (12)$$

Value R_p^T for steels is accepted based on properties at minimum temperature of cycle:

$$R_p^T = 0{,}5[R_{p0,2}^{(T_{\min})} + R_m^{(T_{\min})}] \qquad (13)$$

It is allowed to accept $T_{\min} = 20^\circ$C.

Calculation of $[\sigma_{aF}]$ or $[N]$ taking into account maximum effect of average stress of cycle shall be carried out by Equation (8) at $[\sigma_{F\,\max}] = R_p^T$ and Equation (9) at $n_\sigma\,[\sigma_{F\,\max}] = R_p^T$.

Using data from relevant strength design codes which contain guaranteed mechanical properties, at $Z^T \leq 50\%$ it is accepted: $Z_c^T = Z^T$. At $Z^T > 50\%$, it is accepted: $Z_c^T = 50\%$.

If plasticity characteristic e_c^T is determined at actual value of Z^T obtained from static tensile tests, then the following formulas are used:

$$e_c^T = 0{,}005 Z^T - \frac{\left|(\sigma_F^*)_{\max}\right| - R_{p0,2}^T}{2E^T} \qquad \text{if } \left|(\sigma_F^*)_{\max}\right| \geq R_{p0,2}^T \qquad (14)$$

$$e_c^T = 0{,}005 Z^T \quad \text{if} \ \left|(\sigma_F^*)_{\max}\right| < R_{p0,2}^T \tag{15}$$

When analysing, characteristics E^T, Z^T, R_m^T are assumed to be equal to minimal values within interval of operation temperatures taking into account degradation of material properties. Safety factor for stresses n_σ is assumed equal to 2, and for number of cycles $n_N = 10$.

If allowable number of cycles $[N]$ is less than or equal to 10^6, then determination of allowable amplitude of reference elastic stress $[\sigma_{aF}]$ or allowable number of cycles $[N]$ is permitted to be carried out by Eqs. (16) and (17):

$$[\sigma_{aF}] = \frac{E^T e_c^T}{(4 n_N [N])^m} + \frac{R_{-1}^T (R_m^T - [\sigma_{F\max}] i_\sigma)}{R_m^T - R_{-1}^T i_\sigma} \tag{16}$$

$$[\sigma_{aF}] = \frac{E^T e_c^T}{n_\sigma (4[N])^m} + \frac{R_{-1}^T (R_m^T - n_\sigma [\sigma_{F\max}] i_\sigma)}{n_\sigma (R_m^T - R_{-1}^T i_\sigma)} \tag{17}$$

taking into account conditions at Eqs. (8) and (9).

Indices of power (exponents) m, m_e and values of endurance (fatigue) limit R_{-1}^T are assumed in accordance with Table 10.

Of two values $[N]$ or $[\sigma_{aF}]$ obtained from Eqs. (16) and (17) the minimum shall be chosen as the allowable value.

TABLE 10. VALUES OF EXPONENTS m, m_e AND ENDURANCE (FATIGUE) LIMIT R_{-1}^T

	$R_m^T \le 700$ MPa	700 MPa < $R_m^T \le 1200$ MPa
R_{-1}^T	0,4 R_m^T	$(0{,}54 - 2\cdot10^{-4} R_m^T)\ R_m^T$
m	0,5	$0{,}36 + 2\cdot10^{-4} R_m^T$
m_e	$0{,}132 \lg\left[\frac{R_m^T}{R_{-1}^T}(1 + 1{,}4\cdot10^{-2} Z^T)\right]$	

Residual stresses shall be taken into account if they are tensile stresses, and in the considered component support zone their amplitude of local reference elastic stress due to mechanical or kinematic loads at no loading cycle types exceeds the yield stress at 20°C. It is conservatively assumed that residual stresses are equal to the yield stress at 20°C.

Allowable amplitude of stresses for welded joints $[\sigma_{aF}]_s$, with the exception of welded joints with incomplete penetration, shall be calculated by the formula:

$$[\sigma_{aF}]_s = \varphi_s\,[\sigma_{aF}] \tag{18}$$

where

$[\sigma_{aF}]$ designates amplitude of reference elastic stresses which is obtained on the basis of reference fatigue curve or the relevant formula for basic material at given number of cycles;

φ_s denotes coefficient which depends on welding type, welded materials and thermal treatment after welding ($\varphi_s \le 1$).

Values of φ_s for a number of welding methods are given in Table 11. Coefficient φ_s is used together with reference fatigue curve or equation of fatigue curve for basic material in relation to which φ_s is determined.

For other welding methods, welding and welded materials that are not presented in Table 11, value for φ_s shall be obtained by an experiment in accordance with requirements given in PNAE G-7-002-86 [2], Appendix 2 (mandatory).

In absence of experimental values for φ_s, the data from Table 12 can be used.

Values of φ_s for manual welding austenitic steels by electrodes EA-395/9 and EA-400/10U can be used when analyzing dissimilar welded joints for pearlitic steels with austenitic steel for an overlay disposed onto pearlitic steel, using the reference fatigue curve for austenitic steel.

TABLE 11. VALUES OF COEFFICIENT φ_s

Basic metal	Welding method	Welding material	Type of thermal treatment after welding	φ_s
Steels 20, 22K, 20K	Manual welding	Electrodes UONI-13/45, UONI-13/45A	Without thermal treatment, tempering, normalizing and tempering	1,0
		Electrode UONI-13/55	Without thermal treatment, tempering up to 10 h	0,8 at $(\sigma_{aF}) > 400$ MPa 1,46 – 0,261 lg (σ_{aF}) at 60 MPa $< (\sigma_{aF}) \leq 400$ MPa 1,0 at $(\sigma_{aF}) \leq 60$ MPa
			Tempering more than 15 h	1,0
	Automatic submerged-arc welding	Welding wires Sv-08A, Sv-08GSMT, EP-458	Without thermal treatment	0,9
			Tempering	1,0
	Electroslag welding	Welding wires Sv-10G2, Sv-08GSMT	Normalizing and tempering, quenching and tempering	1,0
High-alloy chrome-nickel steels of austenitic class	Manual welding	Electrode EA-395/9	Without thermal treatment	1,0
		Electrodes EA-400-10U, EA-898-21B, EIO-8	Austenitizing	1,0
	Argon-arc welding	Welding wire Sv-04Cr19Ni11Mo3	Without thermal treatment, austenitizing	1,0

TABLE 12. VALUES OF COEFFICIENT φ_s

Basic material	φ_s for welded joint	
	After tempering	Without tempering
Carbon steel, silicon-manganese steel, alloyed steel, $R_m^T \leq 380$ MPa	0,75	0,75
Alloyed steel, 380 MPa < $R_m^T \leq 520$ MPa	0,70	0,65
Alloyed steel, 520 MPa < $R_m^T \leq 700$ MPa	0,60	0,50
Austenitic steel	0,70	0,60

For threaded segments of bolts and studs, the reference fatigue curves given in PNAE G-7-002-86 [2] shall be used or Eqs. (1) – (4). Safety factors n_σ and n_N shall be accepted equal to 1,5 and 5 accordingly.

Input data concerning high-frequency loading is obtained from measurements during support operation or by calculation.

Strength condition at various loadings shall be checked by the formula:

$$\sum_{i=1}^{k} \frac{N_i}{[N]_i} = a_N \leq [a_N] \tag{19}$$

where

N_i designates number of cycles of i-th type during operation term;

k denotes general number of cycle types;

$[N]_i$ is allowable number of cycles of i-th type;

a_N is cumulative fatigue damage, its limit value is as follows: $[a_N] = 1$.

5.4.2.6. Brittle fracture strength analysis

(a) General requirements

i. This sub-clause does not apply to analysis of threaded segments of bolts and studs.

ii. Brittle fracture strength analysis for support components shall be carried out with respect to to design loads indicated in sub-clause 6.4.2.1, and in all operational modes.

iii. When analyzing, the basic characteristics of structural materials are critical stress intensity factor K_{Ic}, critical brittleness temperature T_k and yield stress $R_{p0,2}^T$.Change of material properties during operation shall be taken into account by introducing into the analysis a shift of critical brittleness temperature due to various operational effects.

iv. If thickness of the analyzed components is less than that required for determination of K_{Ic} values in accordance with GOST 25.506 [6] then it is allowed when analyzing crack growth resistance to use critical crack opening δ_c or critical J-integral J_c determined in accordance with this standard. Analysis procedures using the above characteristics shall be agreed with the material knowledge organization.

v. Brittle fracture resistance is ensured if the following condition is met for the chosen reference crack-like defect in the considered operational mode:

$$K_I \le [K_I] \tag{20}$$

where $[K_I]$ denotes allowable value of stress intensity factor.

vi. Analysis of brittle fracture resistance is allowed to pass if any of the following conditions are met:

- Support components are manufactured from corrosion-resistant austenitic steels .
- Yield stress values for materials of support components including welded joints at 20 ºc are less than 300 mpa and component wall thickness is not greater than 25 mm.
- Yield stress values for support component materials including welded joints at 20 ºc are less than 600 mpa and component wall thickness is not greater than 16 mm.
- Thickness of support component under consideration meets the condition:

$$s \le 8 \cdot 10^3 \left(\frac{[K_I]}{R^T_{p0,2}} \right)^2 \tag{21}$$

where s designates wall thickness in mm; $[K_I]$ denotes the allowable value of stress intensity factor at NOC (in MPa·m^{½}), $R^T_{p0,2}$ and is dimensioned in MPa.Values of $[K_I]$ and $R^T_{p0,2}$ shall be accepted at minimum operational temperature and critical brittleness temperature T_k corresponding to the end of operation.

(b) Stress intensity factor

i. Stress intensity factor (in MPa·m½) for cylindrical, spherical, conical and plane components of supports is allowed to be determined by formula:

$$K_I = \eta \cdot \left(M_m \sigma_{(m)} + M_b \sigma_{(b)} \right) \frac{\sqrt{\pi \frac{a}{10^3}}}{Q} \tag{22}$$

where

η	designates coefficient that takes into account effect of stress concentration (is accepted 1,0 when analyzing zones without stress concentration);
$\sigma_{(m)}$	denotes average tensile stresses on wall thickness (in MPa);
$\sigma_{(b)}$	is bending stress on wall thickness (in MPa).

$$M_m = 1 + 0{,}12\left(1 - \frac{a}{c}\right) \quad (23)$$

$$M_b = 1 - 0{,}64\frac{a}{h} \quad (24)$$

where

a means crack depth (in mm);

c means crack semi-length (in mm);

h is length of the zone where bending stress keeps positive value (is accepted to be s/2, where s denotes wall thickness in mm).

$$Q = \sqrt{1 + 4{,}6 \cdot \left(\frac{a}{2c}\right)^{1{,}65}} \quad (25)$$

Formula (25) is used at $a \leq 0{,}25s$ and $a / c \leq ⅔$.

Stresses $\sigma_{(m)}$ and $\sigma_{(b)}$ are calculated in accordance with formulas given in PNAE G-7-002-86 [2] taking into account that integral is calculated for wall thickness of a component which is assumed in Equation (2): $x = s/2$.

ii. When determining KI, the semi-elliptical crack with the depth a = 0,25s and relation a / c = ⅔ is considered as reference defect.

iii. Taking into account item 5.4.2.6.2.2, Equation (22) becomes the following form:

$$K_I = \eta \cdot \left(0{,}7\,\sigma_{(m)} + 0{,}45\,\sigma_{(b)}\right)\sqrt{\frac{s}{10^3}} \quad (26)$$

where $\sigma_{(m)}$ and $\sigma_{(b)}$ are dimensioned in MPa; s in mm; K_I in MPa·m^{½}.

iv. Coefficient η for zones of stiffness jump (connection of flanges with shell, corners, fillets, etc.) shall be calculated by formulas:

$$\text{at } 0 < s / R_1 \leq 5 \qquad \eta = 1 + (K_\sigma - 1)^{0{,}7}\frac{1{,}8}{s / R_1} \quad (27)$$

$$\text{at } s / R_1 > 5 \qquad \eta = 1 + (K_\sigma - 1)^{0{,}7}\frac{9{,}0}{(s / R_1)^2} \quad (28)$$

where R_1 designates radius of curvature of stress concentrator in a section under consideration; K_σ denotes theoretical stress concentration factor.

It is assumed at $\eta > K_\sigma$ that $\eta = K_\sigma$.

v. Coefficient η for hole zones shall be calculated by formulas:

$$\text{at } s / R_2 \leq 0{,}8 \quad \eta = \sqrt{1 + 5(K_\sigma - 1)\exp(-0{,}86\,s / R_2)} \quad (29)$$

$$\text{at } s / R_2 > 0{,}8 \quad \eta = \sqrt{1 + \frac{2\,(K_\sigma - 1)}{s / R_2}} \tag{30}$$

where R_2 means hole radius.

vi. Use of other methods for determination of stress intensity factor and reference crack-like defects of other shapes and sizes is allowed under a proper validation.

(c) Allowable values of stress intensity factors

i. Allowable values of stress intensity factors depend on reference temperature ($T – Tk$) and loading mode. Dependence of [KI] on ($T – Tk$) is issued as an envelope of two curves defined on the basic dependence (curve) for KIc. The first curve is obtained as result of dividing the basic curve by safety factor nk, the second one – by shifting the basic curve along the temperature safety factor (reserve) ΔT axis. Values of safety factors for various loading modes of supports are given in Table 13.

ii. Basic temperature dependences for KIc shall be assumed on the data given in relevant qualification reports on materials (basic material, welded joints) or on technical decisions approved by material knowledge organization.

For steels of pearlitic class or high-chromium steels and their welded joints with yield stress at 20°C not greater than 600 MPa ascertained on the basis of regulatory documents, it is possible to use generalized curve for stress intensity factors:

$$K_{Ic} = 23 + 20 \cdot \exp\big(0.019 \cdot (T - T_k)\big) \tag{31}$$

TABLE 13. VALUES OF SAFETY FACTOR n_k AND TEMPERATURE RESERVE ΔT

Loading mode	n_k	ΔT, ºC
NOC	2	30
NOC (S)	1,5	30
VNOC	1,5	30
DA	1,0	0

(d) Critical temperature of material brittleness

i. Critical temperature of material brittleness shall be obtained by formula:

$$T_k = T_{k0} + \Delta T_T + \Delta T_N \tag{32}$$

where

T_{k0} designates critical temperature of material brittleness in initial condition;

ΔT_T denotes shift of critical brittle temperature due to temperature ageing;

ΔT_N means shift of critical brittle temperature due to cyclic fatigue damaging.

ii. Values of T_{k0}, ΔT_T, ΔT_N shall be assumed based on data given in relevant qualification reports on materials (basic material, welded joints), technical specifications on materials or on the basis of technical decisions approved by a material knowledge organization.

iii. It is allowed to use the values of $Tk0$, ΔTT given in Table 14.

iv. It is allowed to obtain values of ΔTN by formula:

$$\Delta T_N = 20\sum_{i=1}^{m}\frac{N_i}{[N_i]} \tag{33}$$

where

N_i designates number of loading cycles during i-*th* operation mode;

$[N_i]$ denotes allowable number of loading cycles for *i*-th operation mode;

m means number of modes.

TABLE 14. VALUES OF CRACK STRENGTH CHARACTERISTICS

Basic material	Welding type, welding material	Standard or technical specifications	T_{k0}, °C	ΔT_T, °C
22K	Automatic-arc welding, wire Sv-06A	TU 14-1-1569-75	0	0
	Fluxes AN-42, AN-42M	–		
	Wire Sv-08GSMT, Sv-10GSMT	GOST 2246	40	20
	Flux AN-42	–		
	Wire Sv-08GS	GOST 2246	15	30
	Flux FC-16	TU 108.949-80		
	Wire Sv-08GSMT, Sv-08GS	GOST 2246	0	0
	Flux KF-30	TU 5.965-11090-78		
	Manual arc welding, electrodes UONII-13/45, UONII-13/45A, UONII-13/55	–	20	20
	Electroslag welding, wire Sv-10G2	GOST 2246	40	20
	Flux AN-8M	GOST 9087		

(e) Criterion of brittle fracture strength

i. Brittle fracture strength is ensured if the following condition is met:*KI* ≤ [*KI*].

ii. Calculation shall be carried out only up to reference temperature [*T* – *Tk*]*, its maximum value in the relation [*KI*] = *f* [*T* – *Tk*] corresponds to the value [*KI*]* obtained in accordance with formula:

$$[K_I]^* = 0{,}35 R^T_{p0,2} \sqrt{\frac{s}{10^3}} \qquad (34)$$

where $R^T_{p0,2}$ is dimensioned in MPa; *s* in mm; [K_I]* in MPa·m½.

5.4.2.7. Analysis of external dynamic loads

(a) General provisions

i. When analyzing external dynamic loading (including seismic loading) for supports, the following terms, definitions and designations are used:

— *Accelerogram* is a dependence of vibration acceleration on time under external dynamic loading (time history).

— *Response accelerogram* is an accelerogram obtained by calculation of structural vibrations caused by external dynamic loading.

— *Floor accelerogram* is a response accelerogram for a certain elevation mark (floor elevation), on which a supported element is installed.

— *Element mark* is the fastening element height relative to the bottom plane of the building foundation.

— *Seismicity of NPP site* is the intensity of potential seismic loads (earthquake magnitude) for DE or MDE events at the NPP site, which is measured on the MSK-64 scale.

— *Response spectrum* is an array of absolute values of maximum response accelerations for linear oscillations under dynamic loading given by accelerogram taking into account natural frequency and damping of the oscillator.

— *Floor response spectrum* is an array of absolute values of maximum response accelerations for linear oscillations under dynamic loading given by floor accelerograms.

— *Generalized response spectrum* is a spectrum obtained as result of processing of response spectra for a set of real accelerograms.

ii. Analysis on seismic loadings shall be carried out if seismicity of NPP site is 5 or higher. If seismicity of NPP site is less than 5, then the necessity of analyzing is determined by the design organization.

iii. Necessity of analyzing external dynamic loadings other than seismic loading is determined by the design organization.

iv. When analyzing, all supports are subdivided into three categories of seismic resistance (I, II and III) according to requirements given in Section 4.

v. Below, stresses caused by dynamic (seismic) loading are marked with the lower index s.

(b) Requirements for analysis

i. Forces applied to supports during external dynamic loading shall be obtained through analysis of supports and supported elements as a united mechanical system.

ii. Input data for analysis is the following:

— Parameters of external dynamic loadings (DE, MDE, AF, SWA) in the form of floor accelerograms, floor response spectra or generalized response spectrum, determined for three orthogonally related directions (vertical and two horizontal).

— Loads in operation modes.

iii. Dynamic loads are applied to the mechanical system under consideration (supports and supported element) simultaneously in two orthogonally related horizontal directions and vertical direction.

iv. Dynamic loads include:

— Inertia forces due to vibration of the mechanical system under external dynamic loading ;

— Forces caused by relative movements of supports or their parts under external dynamic loadings.

v. Logarithmic decrement of vibration is assumed $k_{log} = 0{,}02$. It is allowed to use other values of k_{log} with appropriate validation by calculation or experimental tests.

vi. Supports shall be analyzed depending on their category of seismic resistance at the load combinations given in Sub-clause 4.5.3.3.

vii. Calculation shall be carried out by linear spectral method (using response spectra) and/or dynamic analysis method (using accelerograms). If the first natural frequency of the mechanical system is greater than 20 Hz, then the calculation is allowed to be carried out by the static method, multiplying accelerations obtained on response spectrum by 1,3 for frequency within the range 20 – 33 Hz and by 1,0 for frequency greater than 33 Hz.

viii. Determination of stresses in support parts is allowed to be carried out supposing static application of maximum values of dynamic loads obtained by calculation to these parts.

ix. Stresses in support parts shall not exceed allowable values given in Tables 5.15 – 5.18.

TABLE 15. ALLOWABLE STRESSES FOR SUPPORT PARTS EXCEPT BOLTS AND PINS

Category of seismic resistance	Load combination	Group of stress categories	Allowable stress
I	NOC+MDE/AF/SWA VNOC+MDE/AF/SWA	$(\sigma_s)_1$	1,4 [σ]
		$(\sigma_s)_2$	1,4 k [σ]
	DA+DE	$(\sigma_s)_1$	1,5 [σ]
		$(\sigma_s)_2$	1,5 k [σ]
	NOC+DE VNOC+DE	$(\sigma_s)_1$	1,2 [σ]
		$(\sigma_s)_2$	1,2 k [σ]
II	NOC+DE VNOC+DE	$(\sigma_s)_1$	1,5 [σ]
		$(\sigma_s)_2$	1,5 k [σ]
III	NOC+DE	$(\sigma_s)_1$	1,5 [σ]
		$(\sigma_s)_2$	1,5 k [σ]

TABLE 16. ALLOWABLE STRESSES FOR BOLTS AND PINS OF SUPPORTS

Category of seismic resistance	Load combination	Group of stress categories	Allowable stress
I	NOC+MDE/AF/SWA VNOC+MDE/AF/SWA	$(\sigma_s)_{3w}$	1,4 $[\sigma]_w$
		$(\sigma_s)_{4w}$	1,8 $[\sigma]_w$
	DA+DE	$(\sigma_s)_{3w}$	1,5 $[\sigma]_w$
		$(\sigma_s)_{4w}$	2,0 $[\sigma]_w$
	NOC+DE VNOC+DE	$(\sigma_s)_{3w}$	1,2 $[\sigma]_w$
		$(\sigma_s)_{4w}$	1,6 $[\sigma]_w$
II	NOC+DE VNOC+DE	$(\sigma_s)_{3w}$	1,5 $[\sigma]_w$
		$(\sigma_s)_{4w}$	2,0 $[\sigma]_w$
III	NOC+DE	$(\sigma_s)_{3w}$	1,5 $[\sigma]_w$
		$(\sigma_s)_{4w}$	2,0 $[\sigma]_w$

TABLE 17. ALLOWABLE CRUMPLING STRESSES $(\sigma_s)_s$

Category of seismic resistance	Load combination	Allowable stress
I	NOC+MDE/AF/SWA VNOC+MDE/AF/SWA	2,7 [σ]
	DA+DE	2,9 [σ]
	NOC+DE VNOC+DE	2,3 [σ]
II	NOC+DE VNOC+DE	2,9 [σ]
III	NOC+DE	2,9 [σ]

TABLE 18. ALLOWABLE TANGENTIAL STRESSES

Category of seismic resistance	Load combination	Allowable stress	
		τ_{ths} for bolts and pins	τ_{js} for corner welds with incomplete penetration
I	NOC+MDE/AF/SWA VNOC+MDE/AF/SWA	0,7 $[\sigma]_w$	0,7 $[\sigma]_j$
	DA+DE	0,75 $[\sigma]_w$	0,75 $[\sigma]_j$
	NOC+DE VNOC+DE	0,6 $[\sigma]_w$	0,6 $[\sigma]_j$
II	NOC+DE VNOC+DE	0,75 $[\sigma]_w$	0,75 $[\sigma]_j$
III	NOC+DE	0,75 $[\sigma]_w$	0,75 $[\sigma]_j$

x. For welded joints, the value of rated allowable stress [σ] and [σ]j used when evaluating stress according to Tables 5.15 and 5.18 shall be decreased taking into account strength reduction coefficient φj which depends on implementation of welded joint, loading pattern, and extent of non-destructive examination of welded joint. Values of strength reduction coefficient φj are given in Table 6.

xi. Cyclic strength analysis shall be carried out in accordance with Subclause 5.4.2.5. The calculation is allowed to be carried out using maximum stress amplitude obtained, taking into account loadings which correspond to NOC+DE. Number of loading cycles is assumed to be equal to 50. This analysis is not allowed to be carried out if cumulative fatigue damaging without external dynamic loadings does not exceed 0,8.

6. MANUFACTURING, INSTALLATION, REPAIR AND REPLACEMENT OF SUPPORTS

6.1. GENERAL REQUIREMENTS

6.1.1. Manufacturing, installation, repair and/or replacement of supports shall be carried out in accordance with supporting technological (processing) documentation (process instructions, operation cards, etc.). Technological documentation shall be developed by the manufacturer, installation, or repair organization in accordance with design documentation taking into account requirements of Russian regulatory documents including PNAE G-7-008-89 [3] and SPiR-O-2008 [5].

6.1.2. Technological documentation for thermal cutting, forming, welding and thermal treatment shall meet requirements of Russian regulatory documents in force including PNAE G-7-008-89 [3] and SPiR-O-2008 [5].

6.1.3. All operations on preparing and assembling for welding, making welded joints, following thermal treatment (if needed), as well as quality assurance of welded joints, shall be carried out in accordance with requirements of Russian regulatory documents including PNAE G-7-008-89 [3] and SPiR-O-2008 [5].

6.1.4. Manufacturing, installation, repair and/or replacement of supports shall be carried out by organizations, which have licenses from the national oversight body for such work.

The licence holder may reassemble supports from completed supports or from parts of dissembled supports that have not been in service provided all documentation is available and relevant regulatory requirements are met.

6.2. THERMAL TREATMENT

6.2.1. Thermal treatment of supports (support components) shall be carried out in accordance with requirements of Russian regulatory documents including PNAE G-7-008-89 [3] and SPiR-O-2008 [5], design and technological documentation.

7. TESTS AND OPERATION OF SUPPORTS

7.1. TESTS

7.1.1. Supports are not allowed to be subject to separate tests. Tests of supports may be combined with hydraulic tests for supported elements in accordance with requirements given in Russian regulatory documents including PNAE G-7-008-89 [3] and SPiR-O-2008 [5].

7.2. OPERATION

7.2.1. Operating organization shall ensure reliable and safe operation of NPP element supports, technical surveillance, in-service inspection (if possible) and repair (if needed).

7.2.2. NPP element supports which are within the scope of requirements of this Appendix shall have a surveillance programme by the operating organization after installation. In order to carry out surveillance for NPP element supports, a responsible person shall be appointed from the technical and engineering employees of operating organization.

7.2.3. NPP element supports which are designed and manufactured in accordance with the requirements of this Appendix are permitted for operation.

7.2.4. The operating organization shall on the basis of this Appendix, and design and engineering documentation, develop operating instructions (operating manuals) for supports.

Operating instructions for supports shall contain:

- Procedures for preparing for operation and maintenance during normal operation;
- Procedures for technical surveillance;
- Actions to take upon detecting damage or failure of;
- Procedure for withdrawal of supports for repair.

7.2.5. Operating instructions for supports may be updated by the operating organization on the basis of pre-commissioning results with design organization agreement.

7.2.6. In the case of a change in technical conditions or loading conditions of supports, the operating organization shall make relevant alterations in operating instructions for NPP element supports and inform personal about these alterations.

7.2.7. NPP element supports shall be subject to technical surveillance following installation (or during installation if after it supports become inaccessible), before pre-commissioning, periodically (if possible) during operation, and ahead of schedule if needed. Periodicity (terms) of surveillance of supports is specified in Section 8.

7.2.8**.** In-service inspection of the metal of NPP element supports shall be carried out in accordance with requirements specified in Section 8 and should precede technical surveillance. Results of in-service inspections shall be analyzed before the technical surveillance.

7.2.9. If technical surveillance reveals defects which give rise to doubts about the operability of the supports, then the operating organization shall document the causes of defects, and the possibility and conditions required for further operation of the supports.

8. IN-SERVICE INSPECTION OF SUPPORTS

8.1. IN-SERVICE INSPECTION FOUNDATIONS

8.1.1. Supports, both standard and non-standard, involving relative movements shall be subjected to in-service inspection during operation (in-service inspection). In-service inspection shall be carried out by NPP personnel.

8.1.2. In-service inspection is necessary due to ageing, that is an irreversible adverse change of support technical conditions (mechanical properties, geometry, functional parameters) under operational factors. Ageing mechanisms for basic metal and welded joints, as well as ageing effects observed are presented in Table 19.

TABLE 19. AGEING MECHANISMS AND AGEING EFFECTS OBSERVED FOR BASIC METAL AND WELDED JOINTS OF SUPPORTS

Ageing effects observed	Ageing mechanisms					
	Thermal ageing	Fatigue (low-cycle, high-cycle, thermal)	General corrosion	Local corrosion	Relaxation of springs and bolts	Wear in movable joints
Change of mechanical properties and structure of metal	×	×			×	
Cracking		×				
Thinning			×			×
Erosion			×			×
Pitting				×		

Ageing mechanisms and ageing effects for supports shall be subjected to monitoring during operation under the ageing management procedure described in Section 11.

8.1.3. In-service inspection of supports is aimed at:

- Detection and registration of defects (caused by cracking, thinning, erosion and pitting);
- Detection and registration of changes in mechanical properties and structure of metal;
- Detection and registration of welding defects (cracking);
- Check of geometrical parameters and relative position of movable parts and units;
- Check of function of spring supports, constant supports, shock absorbers and vibration dampers;
- Check of oil filling level for hydraulic shock absorbers and vibration dampers;
- Condition evaluation and validation of decisions on continued operation, repair or replacement of supports.

8.1.4. In-service inspection is divided into pre-operational, periodic and out-of-turn.

8.1.5. The goal of pre-operational inspection is to register the initial (before operation begins) technical condition of supports. Pre-operational inspection data may be used as a basis for comparison of the actual condition of supports in operation with their initial state.

8.1.6. Periodic in-service inspection is carried out in accordance with a relevant plan during NPP operation. Performance shall be regulated by the manual on in-service inspection for the nuclear power facility.

8.1.7. Out-of-turn in-service inspection is carried out:

- After an earthquake equivalent of seismic intensity at or above the design earthquake;
- Following violation of normal operational conditions, or following accidents accompanied by loadings, in particular dynamic loads, with levels exceeding design loadings;
- At the decision of the npp owner, management, or a state oversight body.

8.1.8. In-service inspection of supports may be carried under the framework for in-service inspection of supported pipelines and equipment, or separately in accordance with the manual on in-service inspection for particular nuclear power facility. In particular, the manual shall determine extent of inspection, intervals of inspection, examination methods, inspection procedures, as well as used technical tools, accessories and required documentation.

8.1.9. The operating organization is responsible for establishment of the needed organizational structure which shall carry out in-service inspection of supports, as well as related financial provisions, logistics, supply of regulatory, technical and scientific documentation, and recruitment and training of personnel.

8.1.10. The frequency and extent of in-service inspections shall be specified in inspection programs developed by NPP management, and approved by the state oversight body. Inspection programs shall meet the regulatory documents in force. When developing inspection programs, support manufacturer's recommendations regarding inspection shall be taking into account, including references to qualification test data.

8.2. IN-SERVICE INSPECTION METHODS

8.2.1. Structure design of supports and hangers shall provide for the possibility of carrying out in-service inspections by one or several existing technical methods.

8.2.2. In-service inspection of supports, carried out by technical methods, shall include:

- In-service inspection for structural elements;
- Measurement of relative position of movable parts;
- Geometry measurement of contact surfaces;
- Function testing for spring supports, constant supports, shock absorbers and vibration dampers.

8.2.3. In-service inspection during operation is carried out, as a rule, by non-destructive examination methods.

8.2.4. During in-service inspection, the following non-destructive examination methods are used:

- Visual examination;
- Ultrasonic examination;
- Radiographic examination;
- Other examination methods ensuring detection of metal defects.

8.2.5. Relative position of movable parts shall be examined by instrument methods, in particular by optical examination methods.

8.2.6. Geometry measurement of contact surfaces shall be carried out by instrument contact and non-contact (optical) methods.

8.2.7. Functional testing of spring supports, constant supports, shock absorbers and vibration dampers is carried out, as a rule, free-standing (after demounting) on special check and test equipment. This equipment shall ensure:

- For spring supports and constant supports: follow-up of rated (design) load/travel diagram over the specified travel range and recording of its actual profile;
- For shock absorbers: measurements of dynamic characteristics according to manufacturer's specifications or requirements given in spir-o-2008 [5];
- For vibration dampers: measurements of total travel range and damping resistance parameters according to manufacturer's specifications or requirements given in spir-o-2008 [5].

8.2.8. For hydraulic shock absorbers and vibration dampers, visual examination shall be carried out in order to determine fluid loss level, critical for their function.

8.3. IN-SERVICE INSPECTION TOOLS

8.3.1. For in-service inspection, the following tools are applicable:

- Transportable examination and measuring devices (ultrasonic scanners, range finders, goniometers, etc.);
- Stationary examination and measuring (systems), installed permanently or temporarily on support components;
- Facilities for examination of demounted supports.

8.3.2. Number, type, placement locations and other requirements for examination and measuring devices are determined by the design organization on the basis of specific operating conditions and regulations.

8.3.3. Installation diagrams of examination and measuring devices shall provide for the possibility of periodical checking their function in laboratory environment and (or) at their field locations. Procedures and frequency of checking shall be stated in labour operating instructions for the supports.

8.3.4. At places where inspection can be not carried out due to radiation or equipment location, appropriate remote tools shall be provided for in-service inspection. A list of remote tools and specifications for their development shall be given in the engineering design of the support system; development shall be carried out by a specialized organization or the design organization.

8.4. EXTENT AND PERIODICITY OF IN-SERVICE INSPECTION

8.4.1. Extent of inspection, placement locations of sensing devices, examination methods, measurement accuracy, safe operating limits for supports shall be determined by the design organization and specified in design documentation.

8.4.2. Extent of in-service inspection for each type of support shall be determined in a relevant inspection programme.

8.4.3. Type inspection programme shall include:

- Indication of type of checked supports;
- List of zones checked by non-destructive examination methods;
- List of zones checked by destructive examination (if needed);
- Check modes and their extent for each of checked zones;
- Check procedures (references to documents, containing description of check procedures, or description of procedures directly);
- Periodicity of each check mode;
- Requirements to resolve capacity of examination and measuring devices;
- Canons for evaluation of results (on all check modes);
- List of special inspection tools.

Type inspection programme shall be agreed with the organizations having responsibility for the NPP.

8.4.4. On the basis of type inspection programme, the organization possessing the equipment and piping shall develop a working inspection programme (inspection instructions).

8.4.5. The following shall be included in the working inspection programme (inspection instructions):

- List of checked supports specific for NPP;
- List and coordinates of zones of non-destructive examination for specific types of supports;
- Coordinates of zones of specimen cut-offs for destructive examination (if required);
- Description of check procedures (or reference to relevant documents);
- List of technical and other measures required for inspection;
- Inspection personal demand;
- Name and position of person responsible for inspection;
- Safety requirements;
- Organizational directions concerning inspection;
- Directions for processing inspection results and developing documentation.

8.4.6. In-service inspection of supports shall include the following visual examinations:

- Contamination check;
- Wear check;
- Damage check;
- Examination on visible shape changes;
- Free movement examination;
- Check of actual position in the cold condition, and as far as possible, in the hot condition;

— Installation (basic) settings of the support.

8.4.7. In-service inspection of pipeline supports shall include:

— General condition examination;

— Load and displacement examinations for variable spring supports and hangers;

— Function tests for constant supports and hangers using mobile test equipment at the npp, or at the manufacturer's factory.

8.4.8. Inspection of supports installed on a pipeline shall include:

— General condition examination and if needed, inspection of geometrical position;

— Examination of pipeline system for possibility of movement in all three planes;

— Determination of vertical displacement at all support locations.

8.4.9. Intervals of in-service inspections are regulated by the manual on in-service inspection for the nuclear power facility. Values shall be fixed on the basis of required safety level of the NPP, which takes into account state-of-the-art nuclear science and engineering, available examination methods and tools, technical specifications and manufacturer's recommendations.

8.4.10. Periodic in-service inspections for supports, performed by non-destructive examination methods, shall be carried out as a minimum every 100000 h of operation.

8.5. APPLICATION OF IN-SERVICE INSPECTION DATA

8.5.1. Results of in-service inspections are applied within the framework of ageing management of supports, for example, as a basis for maintenance, decisions about operational life time extension, and repair or replacement of supports.

8.5.2. Results of in-service inspection containing information on actual condition of supports may be used as input data when performing strength analysis during the operation.

9. OPERATIONAL LIFE EXTENSION FOR SUPPORTS

9.1. INTEGRATED SURVEY AND PREPARATION FOR OPERATIONAL LIFE EXTENSION

9.1.1. Special work on supports shall be included within the programme of integrated surveillance, and the programme for preparation to extend the operational life for elements of WWER NPPs. This work will be carried out in accordance with requirements of regulatory documents.

9.1.2. The programme of integrated surveillance shall include the following:

— Analysis of design, manufacturing and maintenance documentation on supports in order to determine equivalence of design and actual operation modes, and to reveal the most loaded zones of support structures;

— Condition survey of supports in accordance with requirements of section 8;

- Determination of actual mechanical properties for structural materials of supports in order to compare their values with that used in design calculations;
- Strength analyses aimed at substantiation the possibility to extend the operational life of supports.

9.1.3. The programme of preparation to extend the operational life shall be developed on the basis of integrated survey data. Within the programme of preparation to extend the operational life, the following shall be provided:

- Research aimed at determination of residual life (taking into account requirements of Clause 9.2.1) and, if necessary, implementation of measures to increase residual life;
- In case of amendment or repair, development of design and engineering documentation;
- Repair or amendment in accordance with requirements given in Subsection 9.3;
- Update / reissuance of strength analyses according to actual support structure conditions.

9.2. REQUIREMENTS FOR STRENGTH ANALYSIS FOR OPERATIONAL LIFE EXTENSION

9.2.1. When analyzing support strength in order to substantiate operational life extensions, it is necessary to take into account the following:

- Actual dimensions and shape of support components, deviations from nominal dimensions;
- Defects revealed at installation and during operation;
- Data concerning changes of mechanical properties of structural materials in operation;
- Actual operational modes and support loading parameters, in particular:
    - Load changes as compared with design values;
    - Loads in addition to those taken in to account in original strength analyses (for example, vibration);
    - Loading conditions different from those considered in original strength analyses;
    - Actual movement of supported elements;
    - Mounting and repair tightness;
    - Changes of supports and supported elements due to repair or amendment;
    - Operating modes and loading conditions of supports within the extended life period.

9.2.2. Strength analysis when preparing for operational life extension shall be carried out according to the requirements given in Sub-section 5.4, taking into account the regulations of Clause 9.2.1.

9.3. REQUIREMENTS FOR REPAIR AND AMENDMENT

9.3.1. If during integrated surveillance of supports metal failures and/or distortions and/or plastic deformations and/or cracks with unallowable sizes have been detected, then repair or amendment shall be carried out.

9.3.2. Supports may be repaired by replacing the failed component with a new one. Also, supports may be amended in accordance with a specially developed programme for repair or amendment.

9.3.3. The programme for repair or amendment of supports shall be developed taking into account the requirements of Section 4.

9.4. PERMIT-TO-OPERATE FOR ADDITIONAL LIFE

9.4.1. Operational life extension is only possible if work on integrated surveillance and preparing to extend operational life has been carried out completely and successfully.

9.4.2. If as result of support strength analysis and residual life evaluation factors have been revealed which impede subsequent operation for the additional life period, then the support shall be replaced be a new or similar one. The new support shall meet the requirements of Section 4.

9.4.3. It is allowed to permit to support operation for an additional life time period with detected defects which exceed that given in Section 8, on the condition that strength is substantiated according to the requirements of Sub-section 9.2.

10. DECOMMISSIONING

10.1. PROGRAMME OF DECOMMISSIONING AND INTEGRATED SURVEILLANCE

10.1.1. For decommissioning of NPP, it is necessary to develop a decommissioning programme that includes integrated surveillance of NPP elements.

10.1.2. If NPP elements are planned to be exposed for a long time, it is necessary to provide within the programme of integrated surveillance (a constituent part of the NPP decommissioning programme) special attention to supports, including the following:

- Analysis of support design, manufacturing and maintenance documentation in order to compare of design and actual operating conditions, and to reveal the most loaded zones of support structures;
- Condition surveys of supports (it is allowed to use the requirements of section 4 as condition criteria for supports);
- Determination of actual mechanical properties of support structural materials;
- Strength analysis in accordance with sub-section 10.2 in order to determine the actual fatigue loading of supports.

10.1.3. The NPP decommissioning project shall provide within its integrated surveillance programme:

- In the case of amendment or repair, development of design and engineering documentation;
- If necessary, repair in accordance with the requirements of section 6;
- Development of documents on strength substantiation for the exposure time after decommissioning in accordance with sub-section 10.2.

10.2. REQUIREMENTS FOR STRENGTH ANALYSIS AT DECOMMISSIONING

10.2.1. When analyzing support strength at decommissioning, the following shall be taken into account:

- Actual dimensions and shape of support components, and deviations from nominal dimensions;
- Defects revealed at installation and during operation;
- Data concerning changes in mechanical properties of structural materials during operation;
- Actual operating modes and support loading parameters, in particular:
    - Load changes as compared with design values;
    - Loads in addition to that taken in to account in early issued strength analyses (for example, vibration);
    - Loading conditions different from that considered in early issued strength analyses;
    - Actual movements of supported elements;
    - Mounting and repair tightness;
    - Changes in supports and supported elements due to repair or amendment;
    - Parameters of use and loading of supports during the exposure time after decommissioning.

10.2.2. Strength analysis at decommissioning is recommended to be carried out in accordance with the requirements given in sub-section 5.4**,** taking into account the provisions of clause 10.2.1. Mode of the exposure time after decommissioning shall be considered as a mode of normal operation.

11. AGEING MANAGEMENT FOR SUPPORTS AND LIFE INFORMATION MANAGEMENT (DATAWARE)

11.1. AGEING MANAGEMENT FOR SUPPORTS

11.1.1. Ageing management for supports is carried out within the framework of ageing management for NPPs. Ageing management means a system of measures ensuring the required safety level throughout the whole life of the NPP, taking into account changes that occur with time and use. This requires addressing both physical ageing of structures, resulting in degradation of their performance characteristics, and obsolescence of structures, methods, procedures, tools, hardware and software, i.e. becoming out of date in comparison with current knowledge, regulation/standards and technology [7].

11.1.2. A foundation for effective ageing management is that ageing is properly taken into account at each stage of NPP life, i.e. design, construction, commissioning, operation and decommissioning.

11.1.3. Strategy of ageing management for supports shall follow provisions of IAEA Safety Guide [7].

11.1.4. An important part of ageing management is in-service inspections as described in Section 8. In-service data acquisition and record keeping shall be carried out with the aid of life time information management (dataware) – see Sub-section 11.2.

11.1.5. With the endorsement of the national regulatory body, ageing management for supports can follow the guidelines of the Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (NUREG-1800) [8].

In this case, ageing management consists of an Ageing Management Review (AMR) and Ageing Management Programme (AMP). AMR and AMP applicable to supports are described in Chapters II and III of the Report 'Generic Ageing Lessons Learned' (GALL, NUREG-1801) [9]. The responsible review organization is to review the license renewal application (LRA). LRA should identify any enhancements that are needed to permit LRA. The reviewer is to confirm that the enhancement, when implemented, would allow existing LRA to be consistent with the provisions of GALL. The reviewer should document the disposition of all enhancements.

11.2. INFORMATION MANAGEMENT (DATAWARE) FOR SUPPORT LIFE

11.2.1. Information management (dataware) shall cover the entire life of supports: design, manufacturing, installation, operation (including repair and replacement), life extension and decommissioning.

11.2.2. Information management for support life consists of data acquisition, record keeping and exchange between all participants: designers, manufacturers, suppliers, customers, installers, operating or replacement personnel, etc. Information management covers ageing management data.

The main goal of information management is the enhancement of operational reliability and thus the reduction of maintenance costs for supports due to the elimination of information loss, errors or distortion during support life.

11.2.3. The information management system is a combination of software and hardware and includes the following:

- Integrated information environment;
- Information models of supports applicable to various life stages;
- Software for Product Data Management (PDM);
- Software for Computer Aided Design (CAD);
- Software for Computer Aided Engineering (CAE);
- Software for Computer Aided Manufacturing (CAM) and Computer Aided Process Planning (CAPP);
- Software for Enterprise Resource Planning (ERP);
- Integrated Logistics System (ILS);
- Other software programs for resource management and process control.

11.2.4. In general, an integrated information environment is a system of allocated data bases connected via telecommunications and provided with control and interface tools. The data bases contain necessary information on design, manufacturing, operation (including repairs) and decommissioning of supports. This information shall be accessible (subject to permissions) to all participants during support life time.

11.2.5. Sets of information models shall cover all stages of support life and include:

- 3D models of support structures;
- 3D models of relevant supported components and piping, as well as environmental conditions;
- Mathematical (simulation) models intended for thermal, strength and other analyses;
- Models describing manufacturing and operating processes, possible failures, reliability characteristics, etc.

All of these models are information objects in an integrated information environment with various data structures: digital, text of graphic files, data base records, executable modules, etc.

Information models shall represent parameters of supports and environmental characteristics in a reasonable, easy-to-use way. They shall allow for obtaining and inputting new information or data on supports by way of analytical or numerical calculations.

3D models of support structures define the geometry and properties of a support needed for its manufacturing, inspection, acceptance, assembling, operation, repair and decommissioning. This model is used for:

- Data interpretation in automatic systems;
- Visual representation of support structures when design, manufacturing, installation, repair or other operations are carried out;
- Release of drawings and operating documentation (in electronic or paper form).

3D models of relevant supported components and piping and environmental conditions shall provide necessary data for thermal, strength and other analyses, as well as for simulation of manufacturing and operating processes, etc.

Mathematical (simulation) models intended for thermal, strength and other analyses are built on the basis of 3D models of supports, and contain relevant physical, mechanical and control data to carry out numerical calculations.

Models for manufacturing and operating processes, possible failures, reliability characteristics, etc. shall be built in the form most appropriate to obtain necessary information in accordance with state-of-art nuclear science and engineering.

11.2.6. PDM, CAD, CAE, CAM, ERP, ILS and other software programs used shall ensure information management during support life by the most effective way in accordance with state-of-art nuclear science and engineering.

11.2.7. For most responsible supports, information management during operation shall include the following functions related to structural health monitoring and diagnostics:

- Data acquisition on actual operating conditions and loading parameters;

— Determination of current structure conditions, taking into account degradation of material properties, initiation and development of defects, untightening of bolts, changes of initial geometry and positioning of support parts;

— Estimation of exhausted life, provided by information on actual structure conditions and loading;

— Forecasting of residual life based on actual loading and expected future operating conditions;

— Definition of recommendations for repair or replacement of supports in order to ensure required operational parameters for the specified life.

Diagnostics, estimation of exhausted life, forecasting of residual life, and definition of recommendations shall be based on effective mathematical methods ensuring reliable results with minimal calculation costs. Among such methods, FEM and neural networks shall be mentioned.

For structural health monitoring and diagnostics of supports, a rational realization version shall be chosen in dependence of location and application way: mobile or stationary, inside or outside NPP, on-line or delayed analysis.

Mathematical (simulation) models intended for structural health monitoring and support diagnostics are generated by CAD and CAE and used initially during the design stage. In the manufacturing stage, the models shall be subjected to verification and correction based on actual support implementation including possible initial manufacturing imperfections (dimensional deviations, allowable technological defects, etc.). In the operating stage, in order to obtain more precise life estimations, periodic verifications and corrections may be needed based on actual structural health monitoring and diagnostic data.

11.2.8. All data generated or obtained during the support life shall be information objects within integrated information environment.

12. PIPE-WHIP RESTRAINTS

12.1. GENERAL REQUIREMENTS

12.1.1. The contents of this Section follow Annex D Pipe-whip restraints in Safety Standard KTA 3205.1 [10] taking into account general methodology of SPiR-O-2008 [5] and loading mode classification given in Clause 4.5.3 of this Appendix.

12.1.2. Pipe-whip restraints are structures to prevent broken pipe from whipping.

12.1.3. Pipe-whip restraints are required if none of the following conditions is satisfied:

— Proof that for postulated pipe ruptures only admissible pipe movements can occur due to the energy stored in the piping;

— Sufficient spatial separation is provided between the broken pipe run and safety-related components and systems;

— Limitation of possible effects of pipe rupture by building or structural parts (e.g. Ceilings, walls, platforms, platform supports);

— Exclusion of supercritical failure of pipe welds.

12.1.4. Where snubbers or other shock suppressors take over the function of a pipe-whip restraint, the requirements of this Section shall apply in the load case „pipe rupture" (relating to loading mode DA, see Clause 4.5.3 of this Appendix).

12.2. DESIGN PRINCIPLES

12.2.1. The principal design feature of pipe-whip restraints is the ability to withstand single short-term thrust loadings – caused by jet thrust reaction forces that are variable in time during pipe rupture – and to limit deformation. Design measures shall be taken such that operation of the entire plant is assured, i.e.:

- The pipe is not inadmissibly restrained during regular operation;
- No inadmissible thermal bridges are formed.

12.2.2. The design shall lead to simple geometries of the pipe-whip restraints. Sufficient accessibility to the piping for inspection purposes shall be provided.

12.2.3. Pipe-whip restraints shall be designed for single incident control. To this end, plastic deformations of pipe-whip restraints are permitted. Therefore, damping elements shall be provided between the piping and the pipe-whip restraint to absorb energy. All components of a pipe-whip restraint designed to undergo plastic deformation shall be replaced after having been subjected to a specific design load.

12.2.4. Adequate functioning of pipe-whip restraints shall be ensured as follows:

- By loose enclosure of the piping;
- By frictional connections (integral or non-integral connections).

12.2.5. Principally strains occurring shall be verified.

12.2.6. Where strain limitation of the pipe-whip restraint is required, this shall be indicated on the design data sheet. Indepent of this protective goal the pipe-whip restraint shall be classified under load cases specified in Clause 4.5.3.

12.3. DESIGN LOADS

12.3.1. Notations as to the forces occurring during pipe rupture are shown below in Figure 31 with the example of a crack.

12.3.2. The arrangement and design of pipe-whip restraints shall be subject to the effects of rupture occurring.

12.3.3. The quantities governing the loading of pipe-whip restraints are:

- Jet reaction force;
- Pipe stiffness;
- Pipe mass (including fluid, insulation);
- Pipe guidance;
- Maximum unimpeded movement (free whip) of pipe until limitation by pipe-whip restraint;

— Stiffness of pipe-whip restraint;

— Plastic deformability of piping and pipe-whip restraint.

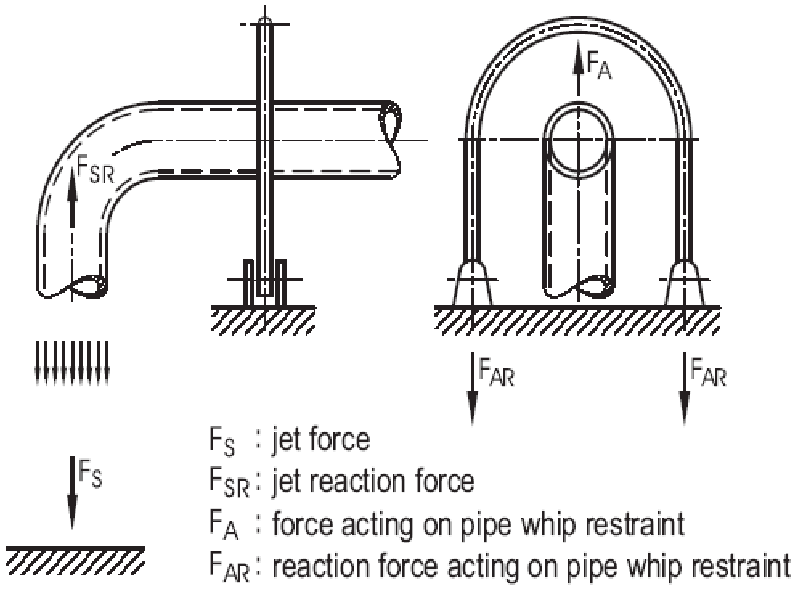


FIG. 31. Notation of forces caused by leaking fluid due to cracking.

12.3.4. For analysis of the pipe-whip restraint behavior (load as function of time) due to pipe rupture the loading function (qualitative course in Figure 32) can be divided into two time ranges.

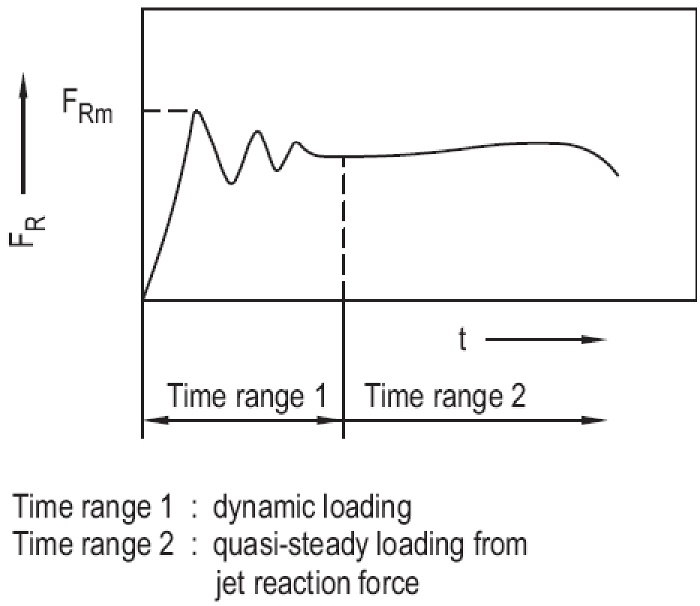


FIG. 32. Loading of pipe-whip restraint.

12.3.5. The pipe-whip restraint loading within time range 1 shall be proven by a dynamic analysis. The possibility of rebound due to the elastic energy stored in the pipe/pipe-whip restraint shall be taken into account.

12.3.6. For determination of thrust loading the maximum possible free movement *s* of the piping shall be used, if required by adding the elastic deformation. In the case of loose enclosures *s* may be determined by means of Figure 33.

12.4. CALCULATION PRINCIPLES

12.4.1. The basic calculation method for pipe-whip restraint behavior under loading caused by pipe rupture is dynamic analysis.

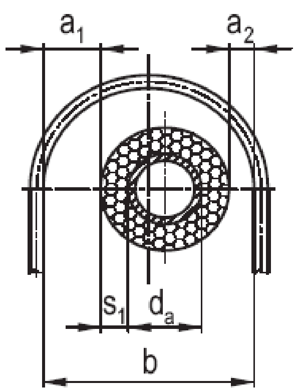
12.4.2. Verification by static calculation of thrust-loaded pipe-whip restraints is possible on the basis of load factors derived from the dynamic calculation of a similar structure.

12.4.3. Where the plastic behavior of the load supporting structure of pipe-whip restraints is considered, it shall be based on a load deformation diagram which may be determined by means of an assured calculation procedure or representative tests.

12.4.4. Determination of the load deformation characteristic curve used in the calculation shall consider the following:

— Upper stiffness and strength values shall be used to calculate transferred loadings and in the stress analysis;

— Lower stiffness and strength values shall be used for deformation analysis.



$s = b - d_a - s_1 - a_1$ (if heat insulation is fully deformed)

b : clear width of whip direction

d_a : pipe outside diameter

s_1 : insulation thickness

a_1, a_2 : distance between heat insulation and whip direction, depending on direction of movement

FIG. 33. Maximum free whip s.

12.4.5. For pipe-whip restraints subject to elastic deformations the principles of sub-section 5.4 of this Appendix shall be applied to design and calculation.

12.4.6. For load supporting structures of pipe-whip restraints subject to plastic deformation, suitable design and calculation methods shall be applied, e.g. given in DIN 18800-1 [11].

12.4.7. The stability (against column buckling, overturning, buckling) shall be verified by analysis.

12.5. MATERIALS

12.5.1. The materials listed in Section O-2000 of SPiR-O-2008 [5] are permitted; also materials listed in Appendix O-6 to this regulatory document are recommended for use on pipe-whip restraints.

12.5.2. For components strained in the plastic range only materials shall be used for which sufficient deformability can be proved, e.g. elongation at fracture exceeding 20%.

12.6. MANUFACTURING AND INSTALLATION

The requirements of Section 6 apply.

12.7. TESTS AND INSPECTIONS

The general requirements of Sections 7 and 8 apply.

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