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Technical Report
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VVER Working Group

Regulatory approaches and oversight practices related to reactor pressure vessel and primary components

Regulators involved in the VVER working group discussions:	AERB, HAEA, NNSA, Rostechnadzor, STUK and TAEK
Regulators which support the present report:	AERB, HAEA, NNSA, Rostechnadzor, STUK and TAEK
Compatible with existing IAEA related documents:	Yes

CONTENT

I. Introduction

In June 2014, during the 2nd VVER Working Group (VVERWG) meeting [1], the representative from STUK expressed the interest of the VVER's family to understand differences in regulatory approaches and oversight practices used in different countries related to reactor pressure vessel and primary components.

It was suggested to establish the technical experts subgroup on reactor pressure vessel and primary circuit (TESG on RPV&PC) to have further discussions between regulators to better understand differences in regulatory approaches and oversight practices as well as to identify commendable practices in this area.

In March 2015 [2], it was agreed that the TESG on RPV&PC would conduct a discussion and prepare a technical report on the following topics:

- Regulatory requirements related to application of leak before break (LBB) concept;
- Requirements and regulatory oversight on manufacturing of primary components;
- Radiation embrittlement of RPV regarding use of new base materials including influence of Ni and Mn;
- Regulatory requirements related to pre- and in-service inspection of primary components (including hydrostatic pressure test);
- Regulatory requirements related to design basis of primary components (loadings and their combinations);
- Regulatory requirements related to cladding of primary circuit;
- Regulatory requirements related to protection against overpressure of primary circuit.

This report has been prepared on the basis of answers given by VVERWG members to the questionnaire elaborated by STUK (Finland). The regulatory bodies from following countries have participated in the preparation of this Technical report: AERB (India), HAEA (Hungary), NNSA (China), Rostechnadzor (Russian Federation), STUK (Finland) and TAEK (Turkey).

This technical report is divided into a comparative summary (Chapter II), main findings (Chapter III), conclusions and recommendations (Chapter IV) and one appendix from each member country corresponding their regulatory requirements (Appendix A - F). Identified commonalities, main differences and challenges are discussed in Chapter III.

II. Comparative summary

2.1. Application of LBB concept

Finnish regulatory requirements related to application of leak before break (LBB) concept is presented in the chapter 1 of the appendix A. The overall scope of LBB is addressed in the YVL guides B.5, B.7 and E.4. In addition, leak detection system is safety classified electrical and automation system that has to be qualified according to YVL E.7 *"Electrical and I&C equipment of a nuclear facility".*

When LBB is demonstrated for main coolant line (MCL), the effects of blowdown from double-ended break (DEGB) of a pipe connected to MCL and having the most adverse pressure impact are used as a design basis load for reactor internals and all structures and components which are part of or connected to primary circuit. In addition, blowdown effects on some primary circuit items still need consideration as a design extension condition (DEC) with realistic assumptions. Among these items are the reactor pressure vessel internals and their supports, PWR steam generator tubes and the PWR main coolant pump flywheel.

Impact loads of whipping MCL need not be considered in the plant design. Effects of jet impingement forces from postulated through-wall crack of MCL shall be evaluated for structures, systems, and components important to safety.

YVL E.4 formulates the qualification for LBB in the frame of a more comprehensive German break preclusion (BP) concept. It gives considerable attention to the prerequisites of LBB by adopting advanced technical and organizational procedures in piping construction, operations and maintenance. LBB may be also applied to surge line, if the technical prerequisites of the German BP concept can be fulfilled. However, where systems are loaded with excessive or unusual loads e.g. water hammer, thermal stratification or any significant degradation mechanisms, e.g. erosion-corrosion, fatigue, creep, and brittle fracture, sufficient LBB safety margins are not expected.

LBB may not be credited for the design of safety systems that would be needed following a postulated instantaneous DEGB of the LBB candidate piping.

Pipe whip restraints for MCL to protect structures and components important to safety from effects of complete MCL break are not necessary. Where necessary, shielding shall be implemented to protect against the maximum jet loading that could impinge from postulated through-wall crack.

Russian Federation regulatory requirements are presented in the chapter 1 of the appendix B.

The Russian regulatory document for application of the LBB concept RD 95 10547-99 was in force till 2013. According to RD 95 10547-99, deterministic procedures from US NRC standard review plan and German BP were recommended for the LBB analysis. Then instead of RD 95 10547-99, a regulatory guideline of the concern “Rosenergoatom” for application of LBB concept at operating power plants RD EO 1.1.2.05.0939-2013 was in force from 2013 till 2017. Then a new document – RD EO 1.1.2.05.0939-2016 “Safety concept "leak before break” for coolant circuit of nuclear plants. Guideline" has been in force in April 2017. RD EO 1.1.2.05.0939-2016 has requirements for operating power plants and new plant design. It should be noticed that technical part of all these documents in general corresponds to RD 95 10547-99.

Application of the LBB concept for reactor coolant system is obligatory since 2016 according to Federal regulation NP-001-15.

According to RD EO 1.1.2.05.0939-2013, it was recommended to apply the LBB concept to the following pipelines at VVER primary and secondary circuit (within containment only): MCL, pressurizer surge line, main steam line and main feedwater line. If LBB requirements are fulfilled, the dynamic effects of pipe breaks may be not considered as design basis for the affected components. The LBB methodology is not applied to the systems in which the following aspects can cause their failure: excessive or unusual loads, degradation mechanisms as intergranular corrosion cracking, corrosion-erosion wear, etc.

LBB may be used at "old"-type NPP units to demonstrate the confidence in structural integrity of components which is difficult to inspect.

Hungarian regulatory requirements are presented in the chapter 1 of the appendix C. LBB concept is a requirement for new design according to the Annex 3a to Govt. Decree 118/2011 (VII.11).

LBB concept could be accepted as a technical justification for eliminating dynamic effects of DEGBs in high energy piping systems.

Turkish regulatory requirements are presented in the chapter 1 of the appendix D. There is no specific requirement related to LBB concept in Turkish regulations which means that the LBB concept shall be covered by the IAEA safety standards (NS-G-1.9, NS-G-1.11), the regulations of the designer country (NP-006-98, PNAE G-01-036-95, RD 95 10547-99) or vendor.

According to Russian regulations, LBB concept can be used for following pipelines that contain primary coolant: MCL and connecting lines to adjoin systems within the primary circuit boundary. The LBB concept application shall be justified.

The justification methods for LBB are the same that have been referred in IAEA Safety Standard NS-G-1.11 (EUR 18549 - European safety practices on the application of the LBB concept, SRP 3.6.3 - United States Nuclear Regulatory Commission's LBB application, RSK guidelines - Germany BP or JEAG 4613 - LBB guideline used in Japan). According to the Safety Standard, LBB may be used to demonstrate the confidence in structural integrity of certain components (pressure vessels, piping and rotating equipment). If LBB is applied, the protection against missiles and pipe whip need not be considered in the design.

Chinese regulatory requirements are presented in the chapter 1 of the appendix E. Chinese nuclear regulatory documents related to LBB involve references to US NUREG/1061 Volume 3 and US standards review program SRP 3.6.3. For the VVER units, Russian LBB guidelines are also widely referred.

LBB can be applied to primary pipes (diameter more than 6 inches) including connected auxiliary pipes, surge lines, main steam lines and main feed water lines.

The dynamic effects of pipe breaks (pipe break fluid force, pipe whip, jet impingement, compartment pressurization and pressure change inside the pipe) need not be considered as design basis for the affected components. However, in design of emergency core cooling system and containment and equipment qualification LOCA effects should be considered.

Pipe whip restraints for MCL to protect structures and components important to safety from effects of complete MCL break are not necessary.

Indian regulatory requirements are presented in the chapter 1 of the appendix F. The overall scope of LBB is addressed in AERB/NPP-LWR/SC/D Section-5A and -6B.

LBB can be applied to primary coolant pipelines, pressurizer pipelines, emergency core cooling system passive part pipelines etc. LBB may not be credited for the design of engineered safety systems (emergency core cooling system capacity, internal missile effect on primary containment wall and containment pressure build up in accident condition) that would be needed following a postulated instantaneous DEGB of the LBB candidate piping.

The transient pressure differential (blow-down) effects on certain safety related primary circuit items (measuring instruments etc.) and reactor pressure vessel internals (RPVI) shall be analyzed for DEGB loads. The break size for LBB pipelines should be limited to complete DEGB of the largest connected pipeline where LBB criteria are not met.

Implementation of the LBB permits the removal of pipe whip restraints and jet impingement barriers.

2.2. Manufacture of primary components

Safety classification of equipment

According to the Turkish regulations, equipment in nuclear facilities are classified as “Equipment important to safety” and “equipment other than those important to safety”. “Equipment important to safety” is defined as equipment that the failure of which may cause exposure of site personnel or public to radiation exceeding the predetermined levels, or that prevents escalation of anticipated operational occurrences into accidents or mitigates accident consequences.

In Finland, according to YVL B.2, the nuclear facility’s systems, structures and components shall be grouped into the Safety Classes 1, 2 and 3 and Class EYT (non-nuclear safety).

According to Russian Federation regulatory requirements, four safety classes of SSCs are identified depending on their influence on NPP safety as Safety Classes 1, 2, 3 and 4. Components of Safety Class 4 are NPP normal operating SSCs those do not affect on safety and are not included into Safety Classes 1, 2, 3. Additionally for the equipment and pipes there are three groups depending on consequences their failure and damage.

According regulatory requirements of India given in the Appendix F, the safety systems are grouped and ranked into safety classes taking into consideration the consequences of failure of the safety function performed by the SSC and the probability of its occurrence. Three factors which is required to assign safety class are defined in related regulation.

In China and Hungary, there are regulatory requirements related to the safety classification.

Manufacturing of equipment

In Turkey, in order to initiate procurement process for equipment to be used in a nuclear facility, a permit for initiation of procurement shall be obtained by the Owner. To obtain procurement permit, the Owner shall apply to the Authority with procurement system documents defined in the quality management system. Before initiation of procurement, the Owner which has the procurement permit shall notify to the Authority by submitting a procurement plan prepared on equipment basis as well as a list of equipment constituting each unit including their safety, quality and seismic classification. The Owner shall submit a notification to the Authority for each equipment important to safety at least two (2) months before the manufacturing begins. For manufacturing of each equipment important to safety whose manufacture needs to start before obtaining limited work permit due to long procurement process, the Owner shall apply to the Authority to obtain manufacturing approval, instead of manufacturing notification.

According to Finnish regulation, there is an “Approval of construction plan” phase. Construction plan is a document or set of documents, where design bases, calculations, analyses, materials, drawings, manufacturing procedures, qualifications and inspection and testing procedures are presented. The construction plan must be approved by STUK or by authorized inspection body prior to start of manufacturing.

In Russian Federation, there are requirements related to conformity assessment of equipment, qualification of welders, NDT personnel and welding procedures and accreditation of testing laboratories and certification centers.

According to the Indian regulations, there are regulatory requirements related to verification, corrective functions and quality assurance records for manufacturing components.

In Hungary, the Government Decree No. 118/2011 (VII. 11) on the nuclear safety of nuclear facilities, related requirements of regulatory activities annexed Nuclear Safety Codes (NSC) and Hungarian Atomic Energy Authority Guidelines cover the requirements related to the manufacturing of equipment. Before the start of manufacturing, the product to be manufactured shall be identified with documents, i.e. specifications and drawings, to the extent as required for the manufacture and the certification of compliance.

Inspection management regulations for civil nuclear safety facilities and HAF604 contain the regulatory requirements about manufacturing of equipment in China.

Approval of manufacturers and testing organizations

In Turkey, manufacturers involved in the manufacturing of equipment important to safety and used in a nuclear facility have to get approval from the TAEK and subject to regulatory inspections. Manufacturer approval is granted for a five years term. During the approval process for manufacturers, TAEK seeks for established Quality Management System in the manufacturers' organization and facilities. The test, examination and supervision organizations involved in equipment manufacturing process has to be accredited by Turkish Accreditation Agency or national accreditation organizations listed in the Mutual Recognition Agreement of International Accreditation Forum.

In Finland, suppliers of safety-significant products shall have in place a management system that is appropriately certified or independently evaluated by a third party. Evaluated (certified) SFS-EN ISO 9001 or a similar quality management system is approved. Suppliers of products in safety class 1 and 2 shall also comply with the management system requirements of YVL A.3. In safety class 1, YVL A.3 requirements shall be included directly in supplier's management system. For special processes, STUK gives an approval on application. The approval is valid for 5 years, but an annual report on quality system audits shall be sent to STUK for information. For testing organisations, the primary way to prove that a testing organisation is technically competent and it has a working quality system is accreditation. However, for NDT in safety classes 1 and 2, STUK's approval on application is required. Approval is also needed for manufacturer's own testing organisations which are not accredited.

In Russian Federation, nuclear safety authority Rostechnadzor licences the manufacturers and designers of components. There is no time-limit for licence term but every 10 years safety justification must be updated, and according to the results of this justification, the licence may be continued or cancelled. State Corporation “Rosatom” carries out the accreditation of testing laboratories and certification centers.

According to the Chinese requirements mentioned in Appendix E, NNSA nuclear equipment division is responsible for registration management for overseas organizations, carry out inspection for their civil nuclear safety facilities’ design, manufacturing, installation and NDT. Overseas organizations which intend to be involved in civil nuclear safety equipment design, manufacturing, installation should submit application according to activity and equipment category and nuclear safety classification. NNSA nuclear equipment division will review registration application and allows eligible organizations to register. Registered overseas organizations should accept the inspection by NNSA and its Northern Regional Office during their design, manufacturing, installation and NDT for civil nuclear safety equipment.

In India, the manufacturers of items important to safety are responsible for the establishment and implementation of a quality assurance programme, as specified in procurement documents, through a contractual arrangement. The overall responsibility for the effectiveness of the quality assurance programme remains with the Responsible Organisation (RO) without prejudice to the manufacturer's obligations and the legal requirements imposed on the manufacturer. Methods for ensuring effectiveness of Quality Assurance (QA) Programme of the manufacturer could be surveillance and audit. The manufacturing organisation's QA programmes shall be reviewed by RO as and when necessary. However, this review shall be carried out at least once in 3 years.

In Hungary, the Government Decree No. 118/2011 (VII. 11) on the nuclear safety of nuclear facilities, related requirements of regulatory activities annexed Nuclear Safety Codes (NSC) and Hungarian Atomic Energy Authority Guidelines cover the requirements related to the manufacturers of equipment. The manufacturer shall have a quality management programme for the entire manufacturing process and provide for the certified qualification of its tools, procedures and workforce required for manufacturing and inspection.

Manufacturing inspections

In Turkey, before the start of manufacturing of equipment the Owner submits quality plan of the equipment to the Authority which is prepared by the manufacturer and includes inspections to be implemented by the Owner. The Authority plans nuclear safety inspections including hold points and witness points of equipment manufacturing and notifies the Owner about the plan. To conduct inspections, the Authority sends nuclear safety inspectors to the Owner and manufacturers; and if necessary, to test, examination and supervision organizations.

In Finland, in the inspection plan all supervision sequences for the regulator, the licensee and for third parties are presented. Manufacturing of primary components shall be extensively witnessed by the licensee, and certain special process sequences also by a third party, whose expertise is evaluated when reviewing the construction plan. Construction inspection of primary components is always STUK’s hold point. It consists of document review, visual inspection of the components and factory tests (e.g. pressure test, leak tightness test, performance test).

In Russian Federation, Acceptance Inspection and Oversight of Manufacturing is obligatory for 1st – 3rd safety class components and their parts (with some exceptions for 3rd class components), and 4th safety class components included in the “List of nonnuclear production subjected to mandatory acceptance inspection” and performed by Notified Organization. Regulatory supervision (of Manufacturer) is performed by Territorial Division of Rostechnadzor.

According to the Chinese requirements which are given in Appendix E of this report, stationed inspection will be carried out for main equipment. During the equipment manufacturing, periodic comprehensive inspection on nuclear safety mechanical equipment will be carried out. H points will be selected for important procedure and special technics. H points witness includes documents inspection, welding, heat treatment, NDT and factory testing such as pressure test, leak tightness test and function test.

In India according to the Appendix F, inspection and tests are implemented throughout the manufacturing cycle and these inspection and tests shall be performed in accordance with written procedures or work instructions, in sequential order, as set forth in a quality plan.

In Hungary, the Government Decree No. 118/2011 (VII. 11) on the nuclear safety of nuclear facilities and related requirements of regulatory activities annexed Nuclear Safety Codes (NSC) contain the requirements related to the manufacturing inspections.

2.3. Radiation embrittlement of RPV regarding use of new base materials including influence of Ni and Mn

Finland

Regulatory document YVL E.4 "Strength analyses of nuclear power plant pressure equipment" includes Chapter 6 "Brittle fracture analysis".

The brittle fracture analysis of Safety Class 1 pressure vessels shall be performed by methods of fracture mechanics:

- for cracks postulated in potential fracture points, margins with regard to their sudden growth shall be evaluated by comparing the stress intensity factor K_I with the material's fracture toughness $K_{Ic,}$
- elastoplastic methods shall be used in case of larger yielding zone,
- a calculation method of the fracture mechanical parameters used shall be given.

RPV steel T – K_{Ic} reference curves from ASME XI or from separately substantiated code and standard are used. DBTT –curve is defined

- in design stage according to the standard in question,
- during manufacturing quality control according to the standard in question as well as,
- according to the master curve approach, ASTM E 1921, and
- under STUK´s oversight.

Prediction of radiation shift

- in design stage brittle fracture analysis shall be based on empirical correlation between steel alloying elements, impurities and fast neutron fluence,
- shall be verified for conservatism during plant service life using surveillance program measurements,
- may be recalculated during plant service life using surveillance results provided that material inhomogeneity and measurement scatter are properly taken into account and values substantiated.

Russia

Regulatory documents

- RD EO 1.1.2.09.0789-2012. Guideline for determination of fracture toughness using test results of surveillance for strength analysis and residual life of VVER-1000,
- RD EO 1.1.3.99.0871-2012. Guideline for calculation of brittle fracture resistance of reactor pressure vessels VVER-1000 during extended to 60 years lifetime,
- RD EO 1.1.2.99.0920-2014 Calculation of brittle fracture resistance of reactor pressure vessels for designed VVER. Guideline.

VVER-1000 RPV integrity assessment:

- postulated cracks,
- margins (inhomogeneity, specimen type, specimen number, K_{JC} test results),

- structural integrity condition - $K_1 < K_{JC}$,
- elastoplastic methods is be used in plastic zone.

Unified curve method:

$$K_{JC(med)} = K_{JC}^{shelf} + \Omega\left[1 + th\left(\frac{T-130}{105}\right)\right].K_{JC}^{shelf} = 26MPa\sqrt{M}, for\ B_N = 25\ MM, P_f = 0.5$$

- Ω depends on material embrittlement,
- Ω decreases with increasing of steel embrittlement

Prediction of VVER-1000 steel shift:

$$\Delta T_K(F,t) = \Delta T_t(t) + \Delta T_F(F)$$

$$\Omega(F,t) = \Omega_0 \exp\left(-\left(C_T + C_F\left(\frac{F}{F_0}\right)^m\right)\right)$$

for thermal ageing:

$$C_T(t) = \frac{2}{105°C}\Delta T_t\ (t)$$

for radiation embrittlement:

$$C_F = \frac{2}{105°C}A_F$$

$$\Delta T_F(F) = A_F\left(\frac{F}{F_0}\right)^m,$$

base metal:

m = 0.8,

A_F = 1.45 °C,

weld:

m = 0.8,

$A_F = \alpha_1 \exp(\alpha_2 \cdot C_{eq})$

$$C_{eq} = \begin{cases} C_{Ni} + C_{Mn} - \alpha_3 C_{Si},\ if\ C_{Ni} + C_{Mn} - \alpha_3 C_{Si} \geq 0 \\ 0,\ if\ C_{Ni} + C_{Mn} - \alpha_3 C_{Si} < 0 \end{cases}$$

$\alpha_1 = 0.703$; $\alpha_2 = 0.883$; $\alpha_3 = 3.885$

RD EO 1.1.2.09.0789-2012 predicts the $K_{JC}(T)$ curve shape for VVER-1000 RPV integrity assessment on the basis of trend curves or/and surveillance results

Hungary

Regulatory documents

Material input data temperature dependent material properties (thermal conductivity, specific heat, thermal expansion coefficient, Young modulus and density) were defined for the heat transfer and stress-deformation calculations as these are given. These data are the same as was used in the designer's strength calculations (the data are the same for all the RPVs). The fracture toughness reference curve of RPV materials was taken as follows:

$$[K_{IC}]_3(T) = \min\{26 + 36\cdot\exp[0.020\cdot(T-T_k)];\ 200\},\quad \mathrm{MPa}\sqrt{m}$$

Methodology of Allowable Service Time Calculations Allowable service time values were derived for all RPVs from calculated T_k allow values and from RPV specific T_k data. Allowable service lifetime for a designed location on the RPV is the maximum lifetime (measured in years), for which predicted value of T_k is less or equal than the calculated T_k allow value. Allowable service lifetime for one selected RPV has been defined as the minimum of calculated service time values for its various components.

Codes, Standards, Guidelines Applied

Evaluation of brittle-fracture resistance of VVER-440/213 reactor pressure vessel for normal operation, hydrostatic test, pressurized thermal shock (PTS) and unanticipated operating occurrences, Regulatory Guide No 3.18 (Ver. 2), HAEA, Budapest, October 2009 [2.1]

PNAE G-7-002-86: Calculation Standard for Strength of Equipment and Pipes of Nuclear Power Units, Energoatomizdat, Moscow (1989).

DBTT shift prediction is similar to Russian procedure:

$$\Delta T_K(F,t) = \Delta T_t(t) + \Delta T_F(F)$$

$$\Delta T_F(F) = A_F\left(\frac{F}{F_0}\right)^m$$

Turkey

Regulatory documents

There are no any specific requirements in Turkish Regulation related to radiation embrittlement assessment.

IAEA safety standards:

- Safety of Nuclear Power Plants: Design, Specific Safety Requirements, No. SSR-2/1, IAEA, Vienna, 2012,
- Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety Guide, NS-G-1.9, IAEA, Vienna, 2008.

Russian standard:

- Rules for design and safe operation of equipment and pipelines of nuclear power plant, PNAE G-7-008-89 and PNAE G-7-002-86.

China

Regulatory documents for VVER RPVs:

- Russian standard – PNAE G-7-002-86,
- Methods of analysis must be approved by national Regulator (NNSA).

VVER-1000 RPV integrity assessment according PNAE G-7-002-86:

- postulated cracks (0.25S),
- margins,
- structural integrity condition - $K_1 \leq K_{1C}$.

VVER-1000 reference curve for PTS:

$$[K_1]_3=74+11 \exp [0.038(T-T_k)]$$

Prediction of shift:

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

$$\Delta T_F = A_F F^{1/3}$$

VVER-1000 steels	T_{ko} [°C]	ΔT_T [°C]	ΔT_N [°C]	A_F [°C]
Base metal	-25	0	0	23
Weld	0	0	0	20

- no Mn & Ni,
- Tianwan RPVs surveillance data will be used for NPP service life assessment.

India

Regulatory documents

1. Indian national regulatory documents

2. In addition to Indian regulation, the specific requirements given in following standards:

- IAEA safety standards “Safety of Nuclear Power Plants: Design”, Series No. SSR-2/1 (Rev. 1), 2016;
- USNRC Regulatory Guide, “Radiation Embrittlement of Reactor Vessel Materials”, RG 1.99, Rev. 2, 1988;
- ASME Boiler and Pressure Vessel Code, Section III, Division-1, 2013;
- ASME Boiler and Pressure Vessel Code, Section XI, Division-1, Appendix-G, 2013.

Two approaches to RPV integrity assessment to prevent a brittle failure are based on:

- Transition temperature concept:
  - defines a safe service temperature to provide protection against fracture initiation (and crack propagation),
  - impact test results (RT_{NDT}, ΔT_{41}, USE),
  - surveillance data;
- Fracture mechanics concept:
  - ASME reference K_{IC} (and K_{Ia}) curve,
  - postulated cracks (0.25S)

Prediction of radiation shift:

- based on US NRC RG 1.99, Rev. 2 (only Ni and Cu)
- IAEA TECHDOC-1441“Effects of nickel on irradiation embrittlement of light water RPV steels” is used on case-by-case basis (Mn influence is considered)

2.4. Pre- and in-service inspection of primary components (including hydrostatic pressure test)

Finland

Pre- and in-service inspections in Finland are based on the requirements given in the Finnish legislation:

- Nuclear Energy Act (990/1987),
- Nuclear Energy Decree (161/1988),
- Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant (Regulation STUK Y/1/2016),
- STUK Guide E.5 (previous version YVL Guide 3.8)

The basic requirement level of in-service inspections is based on the standard

- ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components, Division 1, (ASME Code, Section XI. latest Edition).

<u>Acceptance criteria</u>

The requirements for pre- and in-service inspection acceptance criteria are given in STUK Guide E.5.

<u>Hydrostatic test</u>

Pressure of hydrostatic tests should be not less than 1.3 times the maximum allowable operating pressure.

Interval of in-service pressure test is typically 8 years in Finland.

<u>Qualification of NDT inspection systems</u>

In Finland the most technically demanding requirements are presented for the plant supplier in Guide YVL E.5 - The requirements for qualification of NDT inspection systems.

Qualification process in Finland follows the standards of The European network for inspection and qualification (ENIQ).

The risk-informed in-service inspection programmes for piping shall be drawn up on the basis of the risk-informed selection process presented in YVL Guide E.5 chapter 4.

Russian Federation

Regulatory requirements related to pre-service inspection of primary circuit components including hydrostatic pressure test are given in:

- PNAE G-7-010-89 «Equipment and Piping of Nuclear Power Installations. Weld Joints and Weld Overlays. Rules of inspection» (is now under revision).

Regulatory requirements related to in-service inspection of primary circuit components including hydrostatic pressure test are given in:

- NP-089-15 «Rules for Arrangement and Safe Operation of Equipment and Piping of Nuclear Power Installations» (were enforced on 23 February 2016 instead of PNAE G-7-008-89),
- NP-084-15 «Rules for in-service inspection of base metal, welds and cladding of equipment, pipelines and other elements of nuclear power plants».

<u>Intervals of inspection and hydrostatic tests</u>

New intervals for in-service non-destructive inspection of primary circuit components of nuclear power plants, according to NP-084-15:

- First inspection - no later than 3 years after commissioning of NPP unit
- Second inspection - no later than 7 years after the first inspection
- Third inspection - no later than 10 years after the second inspection
- ....
- N^{th} inspection - no later than 10 years after the $(N\text{-}1)^{th}$ inspection
- ....

If the service life of NPP Unit is near to the end of design life (10 years and less):

- Last inspection - during 3 years before the end of design life of the NPP unit
- Next to last inspection - during 7 years after the last inspection.

<u>Acceptance criteria</u>

Requirements for pre-service inspection acceptance criteria are given in PNAE G-7-010-89.

Requirements for in-service inspection acceptance criteria are given in NP-084-15.

<u>Hydrostatic test</u>

Pressure of hydrostatic tests should be not less than (P - working pressure) $1{,}25 \cdot P \cdot [\sigma]^{Th} / [\sigma]^{Tr}$ (low value) and not more than the pressure under which overall membrane stresses equal to $1{,}35 \cdot [\sigma]^{Th}$ occur in the tested component and a sum of overall or local membrane stresses and overall bend stresses reaches $1{,}7 \cdot [\sigma]^{Th}$ (upper value).

The metal temperature for hydrostatic tests of equipment and pipelines should be not less than the minimal permissible temperature defined according to the Standards on Strength Analysis. At the same time it should be not less than 5 ^{0}C in all cases.

<u>Qualification of NDT inspection systems</u>

According to NP-084-15, NDT inspection systems (methods, equipment, personnel) should be subjected to conformity assessment.

The risk informed in-service inspection (RI-ISI) should be done for primary circuit components at NPP units, according to NP-084-15.

Specific requirement for in-service inspection:

- According to par. 89 NP-084-15, the results of non-destructive inspection of areas (zones) of primary circuit components of nuclear power plant must be selectively retested by inspectors of higher or equal qualification whom did not participate in the inspection of specified areas of equipment or pipeline.

Hungary

Regulatory requirements and guidelines related to pre-service and in-service inspection of primary circuit components including hydrostatic pressure test are given in:

- Government Decree 118/2011 (VII.11.) on the nuclear safety requirements for nuclear facilities and related requirements of regulation activities,
- Guideline «Licensing documentation on manufacturing and acquisition of component of nuclear power plants» № 1.7.

The requirements are based on the standards:

- ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components (2001 edition),
- ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components.

In-service inspection requirements – adaptation of ASME Code Section XI. 2001 edition (Hungarian standard MSZ 27011).

Intervals of in-service inspection

Interval of in-service inspection for equipment and pipelines is every 10 in Hungary.

Hydrostatic test

Interval of in-service pressure test is 10 years.

Qualification of NDT inspection systems

Qualification of NDT inspection systems (equipment, technology, personal) is mandatory in Hungary.

Methodology of Qualification process follows ENIQ (European Network for Inspection Qualification).

Personal qualification EN ISO 9712 and Eddy current QDA examination (Qualified Date Analyser).

Ref: European Methodology for Qualification of Non-Destructive Testing (third issue). EUR 17299 EN, Luxembourg, 2007.

Methodology for Qualification of In-Service Inspection Systems for WWER Nuclear Power Plants. IAEA-EBP-WWER 11, IAEA, Vienna, 1998.

The risk informed in-service inspection (RI-ISI) hasn't been used at NPP units.

Turkey

General requirements for in-service inspection are given in:

- Regulation on Design Principles for Safety of Nuclear Power Plants No: 27027, 17/10/2008,
- Regulation on Specific Principles for Safety of Nuclear Power Plants, No: 27027, 17/10/2008.

As these requirements do not cover the aspect, the following documents are used:

<u>IAEA Safety standards</u>

- Safety of Nuclear Power Plants: Design, Series No. SSR-2/1 (Rev. 1), 2016,
- NS-G-1.9 Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety Guide,
- SSG-3 Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Specific Safety Guide,
- NS-G-1.12 Design of the Reactor Core for Nuclear Power Plants Safety Guide,
- NS-G-2.5 Core Management and Fuel Handling for Nuclear Power Plants Safety Guide,
- NS-G-1.10 Design of Reactor Containment Systems for Nuclear Power Plants Safety Guide,
- NS-G-1.11 Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants Safety Guide.

<u>Russian Federation regulations</u>

- NP-082-07 Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants,
- NP-045-03 Rules for design and safe operation of steam and hot water pipelines at nuclear power facilities,
- NP-068-05 Pipeline valves for NPPs. General technical requirements,
- PNAE G-7-008-89 Rules of arrangement and safe operation of equipment and pipelines for nuclear power facilities,
- Equipment and pipelines of nuclear power plants. Welded joints and claddings. Inspection rules. PNAE G-7-010-89, Moscow, 2000,
- NP-006-98 (PNAE G-01-036-95) Requirements to Contents of Safety Analysis Report of Nuclear Power Plant with VVER Reactors.

China

General requirements and guidelines for in-service inspection are given in:

- Nuclear safety regulation HAF001/01 «Application and issuing for nuclear power plant safety license», Chapter 5 Section 19,
- HAF102-2016 «Nuclear power plant design safety regulation», Section 5,
- HAD103/07 «In-service inspections at nuclear power plants»,
- HAF601 «Civil nuclear safety equipment's design, manufacture, installation and nondestructive inspection supervision and management regulation»,
- HAF602 «Nondestructive inspection of civil nuclear safety equipment qualification management regulation».

China nuclear safety guide HAD103/07 gives guidelines for nuclear power plant in-service inspection scope, qualification standards, review of the inspection results, check the equipment technology and methods, and organizational management.

Intervals of in-service inspection

Russian code PNAE G-7-008-89 is used for the following aspects of in-service inspection at nuclear power plants and in-service inspection of VVER units:

- The major pressure welds of the primary equipment (such as reactor pressure vessel cylinder ring weld, steam generator housing weld, the primary pipe butt weld etc.) go through in-service inspection every 4.5 years with inspection methods including ultrasonic examination, penetration inspection and visual inspection;
- Tubes of one steam generator go through eddy current inspection during each refueling outage with steam generators in rotation and the number of tubes randomly inspected during each inspection is not less than 1792;
- The primary pressure weld at nuclear 2, 3 levels go through in-service inspection every 6-8 years with inspection methods including ultrasonic examination, penetration inspection and visual inspection.

Acceptance criteria

Previously in-service inspection acceptance standards of the VVER units in China were adopted acceptance criteria for manufacturing stage (Russian code PNAE G-7-010-89). In 2015, operators developed a special in-service inspection table for allowed defect size for VVER reactor pressure vessels, pressurizers, steam generator pressure welds. NNSA has currently approved it.

Hydrostatic test

A reinforced hydrostatic test should be carried out for the primary circuit every 4.5 years after the unit enters commercial operation.

The pressure of the leak tightness hydrostatic test is 17.6 MPa while the pressure of the reinforced hydrostatic test is 24.5 MPa lasting for at least 10 minutes.

During reinforced hydrostatic test, the medium temperature is at least 88 ℃ during the early years of the reactor operating life and at least 125 ℃ during the last years of the reactor operating life.

Qualification of NDT inspection systems

China nuclear safety regulation HAF601 and HAF602 provide qualification provisions on the organizations and personnel engaged in in-service inspection at NPPs.

The risk informed in-service inspection (RI-ISI) has been started at VVER units in China.

India

Mainly, in-service inspection requirements are given in:

- AERB code “Design of Light Water Reactor based Nuclear Power Plants”, AERB/NPP-LWR/SC/D in Section-6.3.2: In-Service Inspection of the Reactor Coolant Pressure Boundary.

Additional information for in-service inspection is given in the related guidelines:

- AERB guide, “In-Service Inspection of Nuclear Power Plants”, AERB/NPP/SG/O-2.

The basic requirement level of in-service inspections in India is based on the standard:

- ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-service Inspection of Nuclear Power Plant Components.

Qualification of NDT inspection systems

The qualification body should manage, conduct, evaluate and certify an in-service inspection system’s qualification process. Upon successful qualification of a non-destructive testing procedure and the associated equipment, the qualification body should issue a certificate to the licensee and/or inspection organization which clearly identifies the aspects of the procedure and the equipment that have been qualified. The responsibility for acceptance of an inspection system using nondestructive testing remains with the operating organization. The criteria given in the IAEA Safety Guide No-NS-G-2.6 may be used for the acceptance of result of ISI.

2.5. Design basis of primary components (loadings and their combinations)

General provisions

Finnish (YVL Guides), Russian (PNAE G and NP Regulations), Hungarian (NSC 3a Nuclear Safety Code), Turkish (Regulation on Design Principles for Safety of NPP), Chinese (HAF102 Regulation), and Indian (AERB/NPP) Regulations require to evaluate primary components integrity under the action of all possible loadings (mechanical, temperature, irradiation). The loads shall be identified and grouped to limited number of loading combinations.

The loading combinations for different service conditions (normal and abnormal operation) as well as different strength criteria for these service conditions are considered only by Russian (PNAE G-7-002-86 and NP-031-01) and Finnish Regulations. The requirements on strength analysis procedure are also only in Russian and Finnish Regulations. Finnish requirements are more general and include the links to worldwide recognized standards (ASME code, for instance), but other standards may be used in the case of STUK approving. Russian Regulations PNAE G-7-002-86 and NP-031-01 include the detailed guidance on strength analysis. Turkish and Chinese Regulations allow using of PNAE G-7-002-86 and NP-031-01. The standard practice in Hungary is to use ASME code.

There are no requirements for use of numerical strength analysis methods (FEM, for instance) in any considered Regulations, the requirements are in the state-of-art standards (ASME code, etc.).

LBB features

In accordance with all considered regulatory approaches, pipe break loadings shall be considered only if LBB is not implemented. In this case the implementation of protective devices (for instance, whip restraints) shall be done. *Note: LBB is discussed in topic 1*.

Finnish YVL B.5 explicitly prescribes to consider as DBA loading for core support structures the loading (pressure impact) from break of largest primary circuit piping for which LBB concept was not implemented. It also prescribes to consider as BDBA loading for RPV and other primary equipment internals the loading from break of main coolant piping.

Combination of dynamic and static loadings

All considered Regulations require evaluating of earthquake impact on mechanical components. The NPP site-specific ground acceleration shall be determined. The horizontal ground acceleration, used for NPP design, shall not be less than 0.1 g in Russia and Finland, 0.15 g in Turkey and China, and 0.25 g in Hungary regardless to NPP site-specific value.

The maximum earthquake (SSE), used for NPP design, shall be with occurrence less than 1 time per 100 000 years in Finland, 10 000 years in other countries.

Earthquake which is stronger than SSE shall be evaluated only in accordance with Finnish regulatory documents. There is no OBE consideration in Finnish regulatory documents.

Aircraft crash loadings shall be considered in NPP design in accordance with all Regulations. Finnish and Turkish requirements are more specific; small aircraft crash is DBA, large commercial aircraft crash is BDBA. The mass and speed of aircrafts determined only by Turkish regulations.

2.6. Cladding of primary components

Finland

Manufacturing

In terms of manufacturing regulations in YVL Guides, no special requirements are given for the cladding of surfaces but e.g. the following requirements for welding are applied:

- requirement for the approval of manufacturers for special processes is valid also, for cladding (for the approval process, see topic 2, chapter 2.3)
- requirements for welding consumables
- requirements for qualification of processes and personnel

Welding consumables - classified according to EN, ASME or other standards may be accepted, but a specification shall be approved by STUK in advance.

Welding procedure qualification shall be supervised by a third party. Welding procedure specifications (WPS) are approved by STUK.

Inspections

There are no specific requirements for inspection of cladding. However, the construction plan shall include requirements and methods for inspections according to national or international standards. Typical requirements are LT, PT, MT and UT.

Strength analysis

YVL E.4 guide contain requirements related to the stress analysis. The modelling of structural configuration shall consider any strength behavior essentially deviating from the shell theory and the non-linear distribution of thermal stresses in areas with large wall thicknesses. If the thickness of internal austenitic cladding is at least 10% of total wall thickness, the cladding shall be included in the models for thermal loading analysis and stress analysis of ferritic pressure equipment under normal operational conditions and anticipated operational occurrences. A fatigue analysis shall be performed on claddings to demonstrate that the claddings do not undergo detrimental fatigue due to unequal thermal expansion of cladding and base material.

Russia

Introduction

The regulatory requirements related to cladding of primary circuit are included in the following Federal Regulations and Rules for the use of nuclear energy:

- NP-089-15. Rules for design and safe operation of NPP equipment and piping ;
- PNAE G-7-009-89. Welding of NPP equipment and piping. General requirements;
- PNAE G-7-010-89. Welding of NPP equipment and piping. Inspection rules;

- PNAE G-7-003-87. NPP components welders qualification rules;
- PNAE G-7-002-86. Regulations for Strength Analysis of Equipment and Piping of NPPs.

General Requirements

In accordance with NP-089-15:

List of standardization documents for the use of nuclear energy;

- welding, cladding and heat treatment of welded joints of components fabricated from steels grades 10GN2MFA, 10GN2MFAL, 15Kh2NMFA, 15Kh2NMFA-A, 15Kh2NMFA class 1 (materials of primary circuit components) shall be carried out used standardized procedure.

Strength Analysis

Strength analysis requirements for cladding are included in PNAE G-7-002-86 and operating organization standard RD EO 1.1.2.99.0920-2014.

PNAE G-7-002-86 requirements

In accordance with PNAE G-7-002-86:

- the thickness of the corrosion-resistant cladding shall be disregarded for the dimensioning;
- the stresses in the wall and in the cladded layers shall be considered with regard to the thermal stress, resulting from the difference between the linear expansion coefficients of the base metal and the cladded metal, for the design by analysis of cladded walls,
- the fatigue damage of cladding may be determined used base metal fatigue curve together with fatigue strength reduction factor for the fatigue analysis of cladded walls;
- the wall thickness used for calculations shall exclude the cladding for the fast fracture analysis.

RD EO 1.1.2.99.0920-2014 requirements

In accordance with RD EO 1.1.2.99.0920-2014 procedure the statements shall be used.

In accordance with RD EO 1.1.2.99.0920-2014 the strength of cladding shall be evaluated. The procedure of evaluation based on the following data:

- crack growth law due to cyclic loading for cladded material;
- mechanical properties data (elasticity modulus, linear expansion coefficient, heat conductivity coefficient, etc.);
- fracture toughness and yield strength data (with consideration of irradiation influence) for cladded material;

- fracture toughness criteria for cladded material (based on critical J-integral value together with necessary safety margins for different operating conditions);
- recommendation on the distribution of residual stress through the wall (including cladding layer) thickness.

Fabrication. General provisions

Requirements for personnel
In accordance with PNAE G-7-009-89 the following requirements: PNAE G-7-003-87.

Requirements for process and technological documentation: PNAE G-7-009-89

Types of cladding
In accordance with PNAE G-7-009-89 corrosion-resistant cladding is divided into:
- homogeneous - double materials,

Homogeneous corrosion-resistant cladding is divided into:
- single layer, - multi layers,

Double materials corrosion-resistant cladding is divided into:
- two layers, - multi layers.

Requirements for thickness of cladding
PNAE G-7-009-89 includes the requirements for cladding thickness:

Requirements for cladding processes
The following cladding processes (techniques) may be used in accordance with PNAE G-7-009-89:

- automatic submerged-arc cladding;
- manual shielded metal arc cladding;
- manual, semiautomatic, automatic gas (argon) metal arc cladding;
- plasma-arc cladding.

The special requirements if necessary.

Fabrication. Inspection Rules

Inspection of cladding process
The quality control of cladded components in accordance with PNAE G-7-010-89.

Inspection of welding consumables
The quality control of welding consumables in accordance with PNAE G-7-010-89.

DT is performed in accordance with standards or manuals of Material Organization.

Cladding process inspection
The cladding process inspection shall performed in accordance with PNAE G-7-010-89.

NDT of Cladding
NDT cladding examination methods VT, UT, Dimensional exam. The acceptance criteria are given in PNAE G-7-010-89 chapter 11.

Personnel and cladding process qualifications
Qualification of personnel, making cladding, as well as cladding process qualification shall be done in accordance with rules similar to welders/welding qualification. These rules are given in PNAE G-7-003-87 and PNAE G-7-010-89.

Hungary

No special requirements are given for the cladding, but the Reg. Guide contain some requirement for welding applications, for example:

- approval of manufacturers for special processes;
- requirements for base material;
- requirements for welding consumables EN ISO 10204;
- requirements for qualification of processes (WPS - WPQR)- EN ISO 15614;
- personnel EN ISO 9606-1.

There are no specific requirements for inspection of cladding (LT, PT, MT, UT).

There are no specific requirements for strength analysis of cladding of new units.

Turkey

There is not any specific requirement related to cladding of primary circuit.

The base of application are IAEA and Russian documents (NS-G-1.9, PNAE G-7-008-89, PNAE G-7-009-89, PNAE G-7-010-89).

China

Structure and material of cladding

VVER 1000/428 reactor pressure vessel and steam generator coolant collecting tube, pipe and container of the pressurizer, corrosion resistant cladding of the primary pipeline's inner surface, to avoid corrosion caused by direct contact between the alloy steel base material of the primary equipment and the reactor coolant. The equipment of the reactor coolant circuit has the inner surface with the low carbon austenitic stainless steel with strip electrode except the primary coolant pipe to form double layers of alloy steel base material, including transition layer and surface layer. Finally the total thickness of cladding is between 7-13 mm after machining according to different equipment or welding in different regions.

The design, manufacture and inspection of cladding in VVER primary equipment are mainly based on the following Russian standards: PNAE G-7-008-89, PNAE-G-7-009-89； PNAE-G-7-010-89； PNAE-G-7-016-89, PNAE-G-7-018-89, PNAE-G-7-030-91, PNAE-G-7-031-91.

Quality inspection of cladding NDT method and inspection acceptance at cladding manufacturing stage should comply with PNAE-G-7-010-89.

In-service inspection for cladding

ISI every 4 years before the implementation of the long-term refueling outage;

ISI every 6 years after the implementation of long-term refueling outage.

The inspection methods are the following: VT, PT, UT.

India

The specific requirements given in the following codes, guides and standards are used for safety review of cladding of primary circuits:

- ASME Boiler and Pressure Vessel Code, Section XI, Division-1, 2013;
- Control of stainless steel weld cladding of low-alloy steel components, US RG 1.43 (Rev-1), 2011;
- Integrity of Reactor Pressure Vessels in Nuclear Power Plants: Assessment of Irradiation Embrittlement Effects in Reactor Pressure Vessel Steels, IAEA No-NPT-3.11, 2009.

Manufacturing

During manufacturing weld cladding for safety-related components should comply with the fabrication requirements specified in ASME BPVC Section III, Rules for Construction of Nuclear Power Plant Components, and Section-IX, Welding, Brazing, and Fusing Qualifications of the ASME BPVC. For the cladding procedure qualification, US RG 1.43 (Rev-1) may be used.

NDE (Non-destructive examination) inspections of the cladding should be performed following PWHT.

2.7. Protection against overpressure of primary circuit

Finnish regulatory requirements related to protection against overpressure of primary circuit in the chapter 7 of the appendix A. Finnish legislation authorizes STUK to stipulate safety requirements for the nuclear power plants. These requirements, including ones concerning protection of primary circuit from overpressure, are published in STUK's YVL Guides.

The basic idea is that the primary circuit would never loose its integrity during any anticipated operational occurrence or accident scenario due to excessive pressure. The regulatory requirements have been written down for this purpose and they cover the following topics about overpressure protection devices, i.e., safety relief and bleed valves

- design and configuration;
- exhaust capacity;
- quality control of manufacturing;
- qualification;
- maintenance;
- in-service inspection and testing.

In addition to relieving valves, devices that may pressurize the primary circuit are equipped with a system to switch off their function in case of excessive pressure increase.

STUK as a regulatory body pays special attention to oversight of primary circuit integrity, and so is closely involved in all inspections from the system level to in-service testing of overpressure protection devices.

Russian regulatory requirements related to protection against overpressure of primary circuit in the chapter 7 of the appendix B. The following regulations are referred to

- General Regulations on Safety of NPP, NP-001-15,
- Nuclear Safety Rules for Reactor Units of NPP, NP-052-07,
- Rules for Design and Safe Operation of NPP Components, NP-089-15,
- Piping Armature for NPP. General Technical Requirements, NP-068-05.

Generally, the mentioned regulations stipulate, for example, that automatic protection system against unallowable overpressure in the primary circuit during normal operation and violations of normal operation (including design basis accidents) shall be provided. Pressure relief devices of the primary circuit, if actuated, shall not pose a release of the primary coolant medium outside the containment.

The number of pressure relief valves, their capacity and set points shall be designed to prevent the pressure from rising above 1,15 x the service pressure. Once opened a pressure relief valve shall close at a pressure not less than 0,9 x service pressure.

Functioning and set points of pressure relief valves shall be tested periodically. Settings shall be sealed.

Hungarian regulatory requirements related to protection against overpressure of primary circuit in the chapter 7 of the appendix C. The following regulations are referred to

- Act CXVI of 1996 on Atomic Energy,
- Government Decree 118/2011. (VII. 11.) on the nuclear safety requirements of nuclear facilities and related regulatory activities,
- Appendixes of the Govt. Decree as Nuclear Safety Codes,
- Regulatory Guides,
- National Standards.

The pressure retaining equipment and pipeline shall be fitted with pressure relief device if the pressure may exceed the acceptable level. Pressure relief devices shall be designed so that the amount of radioactive material released into the environment during their operation shall be the lowest reasonably achievable.

Hungarian regulations emphasize apparently periodical test. For example, class 1 (primary circuit) pressure relief valves shall be tested at least once every 5 years, starting with initial electric power generation. No maximum limit is specified for the number of valves to be tested within each interval; however, a minimum of 20% of the valves from each valve group shall be tested within any 24-month interval. This 20% shall consist of valves that have not been tested during the current 5-year interval, if they exist. The test interval for any individual valve shall not exceed 5 years.

Turkish regulatory requirements related to protection against overpressure of primary circuit in the chapter 7 of the appendix D. The following regulations are referred to

- Regulations and Guides of The Republic of Turkey,
- IAEA Safety of Nuclear Power Plants: Design, Series No. SSR-2/1,
- IAEA NS-G-1.9 Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety Guide,
- NP-006-98 (PNAE G-01-036-95) Requirements to Contents of Safety Analysis Report of Nuclear Power Plant with VVER Reactors,
- NP-082-07 Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants,
- Rules for design and safe operation of equipment and pipelines of nuclear power plant, PNAE G-7-008-89.

Turkish regulations stipulate that primary coolant system shall designed to protect its integrity during the operational lifetime of the plant. Analyses shall be carried out to demonstrate that structures such as the pressure vessel, systems and components can withstand the stresses likely to be imposed under the more extreme expected loading conditions. In-service inspection and maintenance features shall be taken into consideration in design to control the primary coolant system integrity during the operational lifetime of the plant. For specific requirements, international standards are applied for overpressure protection devices of NPP primary circuit.

In the **Chinese** regulations overpressure protection requirements for NPPs are specified for power operation and cold pressurizing. No specific legislative or regulatory requirements are reported so far.

In the **Indian** NPPs, the primary coolant system, its associated auxiliary systems, and the pressure control/over pressure protection systems are designed with a sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences, and that at the same time the relief system is not actuated. No specific legislative or regulatory requirements are reported so far.

III. Main findings

3.1 Regulatory requirements related to application of leak before break (LBB) concept

Commonalities identified

Finnish, Russian, Turkish and Chinese requirements for leak before break (LBB) are based mainly on US NRC standard review plan and German break preclusion (BP) concept. Indian requirements of LBB are addressed in AERB/NPP-LWR/SC/D.

If LBB is applied, pipe whip restraints for main coolant line (MCL) are not necessary. Shielding shall be implemented to protect against the maximum jet loading, that could impinge from postulated through-wall crack. These requirements are quite similar for all member countries. However, according to Indian regulations LBB permits the removal of jet impingement barriers.

Main differences

The main differences are design basic loads for pressure transient differential (blowdown) effects of a double-ended break (DEGB). In Hungary, Russia and China the dynamic effects of DEGB need not be considered in the design, if LBB concept is applied.

In Finland the design basis are DEGB of a pipe connected to MCL and having the most adverse pressure impact. In addition, blowdown effects on some primary circuit items still need consideration as a design extension condition (DEC) with realistic assumptions. In India the transient pressure differential (blow-down) effects on reactor pressure vessel internals (RPVI) and certain safety related primary circuit items (measuring instruments etc.) shall be analyzed for DEGB loads. The break size for LBB pipelines should be limited to complete DEGB of the largest connected pipeline where LBB criteria are not met.

Challenges identified

Material selections play an important part in defining the physical boundaries of LBB application area. For some PWR designs it has been argued from the standpoint of balanced safety design that reactor coolant lines need no whip restraints since they are made from cladded ferritic material like the RPV in whose design complete breaks are neither considered. With this material selection the obvious physical boundary for LBB application in PWR are thus the dissimilar welds that connect the reactor coolant lines to other pipelines made from austenitic stainless steel. Due to their larger potential for manufacturing errors and ageing, dissimilar welds have proven to be less reliable.

In some cases austenitic cladding of primary piping on the inner surface of primary components may cause difficulties to leak detection.

3.2 Requirements and regulatory oversight on manufacturing of primary components

Commonalities identified

All the member states have the requirements related to the equipment classification, oversight and inspection of manufacturing. The requirements differ from each other in details.

Main differences

Identified difference about equipment classification is that Russian Federation and Finland have 4 safety classes, depending on equipment's influence on safety, including one safety class for normal operation systems (non-nuclear) but Turkey is grouped the equipment as "equipment important to safety" and "equipment other than those important to safety".

Some member countries such as Turkey and Finland have requirements related to initiating of manufacturing of equipment.

Russian Federation, Finland and Turkey have the process of approval of manufacturer. Related to this process there are some differences such as required documents, license validity periods and etc.

All member countries have their own approach about manufacturing inspections.

3.3 Radiation embrittlement of RPV regarding use of new base materials including influence of Ni and Mn

Commonalities identified

The basic reason of RPV mechanical properties degradation is the neutron irradiation, resulting in hardening and embrittlement of the steel from which the RPV is made. The prediction of radiation embrittlement is performed usually in accordance with relevant regulatory documents that are based on a large amount of information from surveillance and test irradiation researches, see comparative summary Table 1. The approach for the RPV steel irradiation embrittlement analysis is similar in all countries – using the steel chemical composition information, surveillance data, etc.

It is generally accepted that the presence of nickel in RPV steels increases its sensitivity to radiation embrittlement even at low deleterious impurity concentrations. The potential effect of manganese and nickel on radiation embrittlement of RPV materials was presented in IAEA-TECDOC-1441 *"Effects of Nickel on Irradiation Embrittlement of Light Water Reactor Pressure Vessel Steels"*. It was demonstrated that high manganese content leads to much greater irradiation-induced embrittlement than low manganese content for both VVER and PWR materials.

Table 1. Comparative summary table for VVER RPV integrity assessment

Country	Regulatory documents	Brittle fracture analysis	Radiation shift factors	Surveillance results for service life assessment
Finland	YVL E.4 Chapter 6 (ASME XI, ASTM E 1921)	ASME K_{IC} reference + Master curve	Fluence, steel alloying elements, impurities	Yes
Russia	RD EO 1.1.2.09.0789-2012	VVER reference curve + Unified curve	Fluence, Ni, Mn	Yes
Hungary	VERLIFE	VVER reference curve + Master curve	Fluence, Ni, Mn	Yes
Turkey	IAEA standards PNAE G-7-002-86 PNAE G-7-008-89	VVER reference curve + ΔT_F	A_F max	?
China	PNAE G-7-002-86	VVER reference curve + ΔT_F	A_F max	Yes
India	Indian IAEA ASME	ASME K_{1C} reference curve + Master curve	Fluence, Cu, Ni	Yes

Differences identified

Requirement to probability analysis of RPV failure is obligatory in Russian Federation. In Finland the construction plan shall include an analysis of brittle fracture probability, if the risk of brittle fracture cannot be concluded as negligible according to deterministic safety analysis.

3.4 Regulatory requirements related to pre- and in-service inspection of primary components (including hydrostatic pressure test)

Commonalities identified

1. The scope of inspection for primary circuit components is similar (not much difference).
2. Intervals for in-service non-destructive inspection and hydrostatic pressure test of primary circuit components of VVER units vary from 3 to 10 years.
3. The values of hydrostatic test pressure are approximately the same for all countries.
4. The quality assessment regulations (**ac**ceptance criteria) to asses the results of inspection are similar for all countries nevertheless there is some difference.

Differences identified

There is some difference in the following aspects for Finland and Russian Federation:

- The difference in classification of equipment and pipelines and categories of welded joints which influence on scope of inspection
- Scope and intervals of in-service non-destructive inspection of primary circuit components of NPP differs in Russian Federation (according to new regulations, NP-089-15 and NP-084-15 frequency of in-service non-destructive inspection of primary circuit components of nuclear power plants is considerably reduced from

every 4 years to every 10 years, except the beginning and the end of design lifetime) and Finland (basically, every 10 years)

- Hydrostatic test pressure differs (Finland - not less than 1.3 times the maximum allowable operating pressure; Russia - not less than $1{,}25 \cdot P \cdot [\sigma]^{Th} / [\sigma]^{Tr}$ - *low bound,* and not more than the pressure under which overall membrane stresses equal to $1{,}35 \cdot [\sigma]^{Th}$ occur in the tested component and a sum of overall or local membrane stresses and overall bend stresses reaches $1{,}7 \cdot [\sigma]^{Th}$) as well as Interval of in-service pressure test - typically 8 years in Finland, in Russia – as for non-destructive inspection (from 3 to 10 years)
- The quality assessment regulations (acceptance criteria) to assess the results of inspection are different (Russian Federation - are given in NP-084-15 and PNAE G-7-010-89) and according to ASME Code 11 in Finland
- According to new Russian regulations, the results of non-destructive inspection of areas (zones) of primary circuit components of nuclear power plant must be selectively retested by inspectors of higher or equal qualification whom did not participate in the inspection of specified areas of equipment or pipeline
- In Finland the most technically demanding requirements are presented for the plant supplier in Guide YVL E.5 - The requirements for qualification of NDT inspection systems. Non destructive techniques and personnel should be qualified in Russia

3.5 Regulatory requirements related to design basis of primary components (loadings and their combinations)

Commonalities identified

All considered regulatory documents require to evaluate primary components integrity under the action of all possible loadings (mechanical, temperature, irradiation).

All regulatory documents require evaluating of earthquake impact on components and aircraft crash loadings.

There are no requirements for use of numerical strength analysis methods (FEM, for instance) in any considered regulatory documents.

Differences identified

The loading combinations for different service conditions (normal and abnormal operation) as well as different strength criteria for these service conditions are considered only by Russian and Finnish regulatory documents.

The maximum earthquake (SSE), used for NPP design, shall be with occurrence less than 1 time per 100 000 years in Finland, 10 000 years in other countries. The 10 times difference in SSE occurrence does not lead to the significant difference in earthquake parameters.

Earthquake which is stronger than SSE shall be evaluated only in accordance with Finnish regulatory documents. There is no OBE consideration in Finnish regulatory documents.

3.6 Regulatory requirements related to cladding of primary circuit

Commonalities identified

It seems that the basis for regulatory requirements related to cladding of primary circuit is quite similar. In terms of manufacturing regulations in different countries, in general no special requirements are given for the cladding. Turkey and China use Russian codes and standards referenced directly. Finland and Hungary applies the EN and ASME standards for requirements, for example for welding and personnel qualification. India uses ASME Code Section XI and the 1.43 Reg Guide as requirements.

In Russian codes strength analysis requirements for cladding are included in PNAE G-7-002-86 and RD EO 1.1.2.99.0920-2014. However, some country apply additional requirements to applied codes.

Differences identified

The main differences in regulatory requirements are related to the welding technology and personnel qualifications.

3.7 Regulatory requirements related to protection against overpressure of primary circuit

Commonalities identified

A common self-evident requirement is that NPP primary circuit shall be protected from excessive pressurization, which is primarily realized by safety relief valves. Number of relief valves, set points and exhaust capacities are specified by a system designer. Maintenance and testing programs are drawn up for relief valves and followed during operation in order to ensure their operability if actuation is required.

Differences identified

A major identified difference is that design exhaust capacity of primary circuit safety relief devices is not unambiguously specified by all member countries. Also some technical details, e.g., set points or periodic test intervals vary country by country.

3.8 Remarks

Russian Regulation PNAE G-7-002-86 was developed in 1980´s. At this time the maximum NPP lifetime considered in design was 30 years. PNAE G-7-002-86 is under revision now and it will be taking into account the increased design lifetime of NPP unit up to 60 years (the revision is planning to be finished in 2018).

It should be noted that following Russian Regulations are under revision now: PNAE G-7-002-86, PNAE G-7-009-89, PNAE G-7-010-89 and NP-031-01.

IV. Conclusions and Recommendations

The current report presents the overview of the existing national regulatory practices of certain aspects of reactor pressure vessel and primary circuit components in condensed manner which are used by the regulatory authorities in support of licensing of the NPP with VVER designs.

The information presented in the report will assist in appreciation of national practices used in safety assessment of reactor pressure vessel and primary circuit components and could be used as a basis for the further activity of the VVER TESG on RPV&PC for elaboration of the common position regarding the items of common interest for regulators of the MDEP VVER member countries.

V. References

1. Summary Record of the 2nd Meeting of the VVER Working Group NEA MDEP, June 2014, Paris, France.

2. Summary Record of the 1st Meeting of the VVER Technical Expert Subgroup on Reactor Pressure Vessel and Primary Circuit NEA MDEP, March 2015, Helsinki, Finland.