REACTOR COOLANT SYSTEM STRENGTH AND RELIABILITY AND SAFETY PROBLEMS

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1 ABSTRACT

The analysis of regulatory document requirements to RP WWER safety from the viewpoint of reactor coolant system component strength and reliability has been provided. Logic relations between strength and safety requirements are specified for regulatory documents of different level. It is proved that WWER-1000 reactor coolant system strength and reliability is fully provided.

2. INTRODUCTION

Safety is a foreground task for the nuclear power plant (NPP) as a man-caused hazardous object. Safety (nuclear and radiation) is NPP qualitative characteristic that determines the limits of radiation load for personnel, population and environment under all design operating conditions. Safety analysis is based on safety criteria OPB [PNAE G-01-011-97] that are of a quality character in essence and have a number of recommendations for designers, manufacturers and plant operators. NPP safety is also provided “by meeting the requirements of federal standards and regulations in the field of atomic energy and other regulatory documents” [PNAE G-01-011-97]. Terms of the reliability theory: operability of physical barriers, reliability of systems (components) important to safety, single failure and reliability level are used in [PNAE G-01-011-97] for safety basic criteria and principles. Operability of the third safety barrier (for normal operating conditions), i.e. reactor coolant system means fulfillment of its functions specified in regulatory and design documentation. Integrity of safety barriers under an accident is not considered since loss of one safety barrier integrity, i.e. an actual failure assumes other safety level.

3. STANDARD SAFETY REQUIREMENTS

The requirement of bearing both dynamic and temperature loads under all considered initiating events without failures is specified for reactor coolant system equipment and pipelines (as to the physical barrier - the third safety barrier preventing from propagation of ionizing radiation and radioactive materials to the environment). Reactor coolant system (the primary circuit) in [PNAE G-01-011-97] is a circuit including a pressurization system purposed for coolant circulation through the core under design operating conditions. The term “main circulation circuit” is also used for WWER-1000 type reactors. Auxiliary system pipelines which are also under the primary coolant pressure, e.g., chemical and volume control pipelines are not referred to the primary circuit. Initiating event is considered in [PNAE G-01-011-97] as a single failure resulting in anticipated operational occurrence that can cause violation of safe operation limits and/or conditions that can lead to an accident situation in the course of the event development. It is specified in [PNAE G-01-011-97] that the number of initiating events does not include equipment and vessel body breaks that allows to assume pipeline breaks as initiating events that can be considered contradicting to the reactor coolant system requirements.

The requirement of safety analysis for RP primary circuit components and systems is specified in regulatory standards PBYa [NP-082-07]. Formulation [GOST 27.002-89] specifies that reliability is an "object property to keep in time and assigned limits all the parameters characterizing the capability to fulfill required functions under specified design conditions". Loss of one of the functions is considered a failure. The reliability theory considers a failure as a casual event and the reliability as a system probabilistic characteristic. Probability of system non-failure operation is the most convenient measure of reliability. Primary components and systems can be divided into two not equivalent groups from the viewpoint of reliability. The first most dominant group includes monitoring and control systems and also valves. The first group reliability indexes are determined on-delivery and in many respects are based on service life and experimental experience. The second group includes the elements defining RP safety to a lesser degree, such as equipment and pipelines, i.e. reactor vessel and top head, nozzles of the top head, RCP casing, ECCS and PRZ, SG steam header and heat-exchanging tubes and also the pipelines of: RCS, ECCS, PRZ connecting and injection. The structures specified shall not be damaged according to OPB [PNAE G-01-011-97], i.e. the third safety barrier integrity shall not be lost that are the assigned functions in determination of reliability indexes.

The next requirement of [NP-082-07] is reactor pressure vessel strength ensuring under normal operation conditions (NOC), anticipated operational occurrences (AOO) and design basis accidents (DBA) within the whole service life. OPB [PNAE G-01-011-97] also specifies limiting value of reactor vessel failure probability $1{,}0 \cdot 10^{-7}$ per reactor/year alongside with the requirement of vessel failure-free operation under loads. The latter requirement can be referred to a reliability index. At the same time the requirement of strength provision shall be specified in terms of nomenclature. Deformable solid body mechanics (DSBM) specifies strength as material resistance to failure and also non-recoverable variation of shape (plastic strain) under external loads. In wide extent, strength failure means reaching of the condition when item design function is lost resulting in operability failure. Generally, strength failure means only separation to pieces [Rabotnov (1979)]. At the same time, formation of a through-wall crack, e.g. in the vessel is also a failure of strength, i.e. a damage.

Comparison of requirements for the reactor coolant system [PNAE G-01-011-97, NP-082-07] and requirements [Regulatory Guide 1.70] shows that the general requirement of "primary pressure integrity" comes to application of ASME code (section III is used for the components of class I) for the areas within the primary pressure boundaries. Section III of ASME code generally satisfies "Regulations" [Regulatory Guide 1.70] and "Strength standards" [GS-G-4.1]. Safety analysis requirements are also non-available according to IAEA [PNAE G-7-002-87.] The requirement for integrity (tightness) of pressure boundaries (of pipelines and equipment) within NPP service life are specified in [Analysis of Compliance of AES 92 Design Versus. Chapter 4. Design Basis. European Utility requirements for LWR Nuclear Power Plants] similarly to [PNAE G-01-011-97]. Non-availability of direct requirements for primary components reliability can be considered common for regulations [Regulatory Guide 1.70, PNAE G-7-002-87].

4. REGULATORY STRENGTH REQUIREMENTS

Regulatory document [PNAE G-7-002-87] specifies that equipment and pipeline designs shall meet the requirements of proper "Regulations" [PNAE G-8-89] and "Strength standards" [GS-G-4.1] and also serviceability, reliability and safety of their operation within the service life shall be provided. Serviceability or serviceable condition according to [GOST 27.002-89] means that "all the parameters characterizing given functions satisfy the requirements of standard-technical or design documentation", i.e. these terms refer to the reliability theory. Requirement [PNAE G-8-89] of operational safety shall be referred to the scope of requirements [PNAE G-01-011-97, NP-082-07].

"Safety standards" [GS-G-4.1] specify the issues of strength and related thermal-mechanical calculations most exactly. These standards are used for NPP equipment and pipelines and refer to the objects which designing, manufacturing and mounting are provided according to Regulations [PNAE G-8-89]. All reactor internal structures, valves as well as equipment and pipeline supports and suspenders are outside normative calculations. This explains independent approaches to component strength calculations relating and not relating to reactor coolant pressure boundaries.

The tendency of enhanced entirety and profundity of RP structures (components) strength analysis is observed for the recent 30 years of WWER-1000 design evolutionary development. On the one part, it is promoted by development of numerical methods for DSBM problem solving, e.g. FEM and computer

codes [Shary et al. (2004)] and on the other part - by the development of new and advancing of previous normative and regulatory documents with specified requirements for safety analysis in which strength problems are formulated more exactly. Generally it shall be noted that the problems of NPP component strength are of major importance for safety analysis and licensing. Experience of various thermal-mechanical analyses (calculations) and related strength analysis proves that systematic consideration of regulatory document requirements for strength and reliability calculations needs further specification from the viewpoint of safety and considering the experience of RP safety analyses for WWER-1000 as the most widely spread in modern Russia. Elaboration of the hierarchical diagram for provided calculations and supporting analyses is an initial stage of RP safety and strength requirements regulation. Diagram in the figure, which elaborates and develops the previous diagram of [Shary et al. (2004)] is suggested for discussion. Equipment and pipelines thermal-mechanical calculations can be conventionally divided according to Figure 1 into normative strength calculations, special strength calculations, calculations of reliability, experiment-calculation verification of detachable joint tightness and residual service life calculation analyses.

Normative strength calculations are provided strictly according to Strength standards [PNAE G-7-002-87] Possible material damage mechanisms and limiting conditions of these structures are developed on the basis of RP structure operational analysis and calculation experience. The term “damage” is not one of reliability concepts here but is considered according to DSBM [Rabotnov (1979)].

The following limiting conditions are assumed for WWER-1000 RP components with creeping non-available:

- static failure (loss of bearing strength over vessel or pipeline cross-section);
- brittle fracture, related or non-related to radiation damage;
- cyclic damage, resulting in macro crack sometimes with concurrent corrosion effect;
- damages under bearing stress in component contact points usually typical for detachable joints.

Loss of stability is also considered in some rather occasional events. Strength calculations are provided also for all design loads [NP-031-01] and the conditions divided in design documentation into NOC, AOO, and DBA. Accident situation concepts are used in [GS-G-4.1, PNAE G-8-89] as an individual case of DBA enlarged concept. The clarification in [PNAE G-8-89] stating that "accident situation" is referred only to equipment and pipelines is not sufficiently substantiated.

Consideration of DBA conditions in cyclic strength analyses as the dominating mechanism of RP component damage can be interpreted in combination with NOC and AOO from the viewpoint of safety. Firstly, RP is designed for all spectrum of accident conditions and, therefore, it shall withstand repeated loads typical for accident scenarios within the whole service life. Secondly, repeated break of pipelines or equipment vessels which are additional failures resulting in beyond design-basis accident under the accident scenario related e.g. to the primary leak, i.e. conformity of OPB requirements is not permissible. At the same time, the check strength calculation of cyclic loading is purposed to prevent macro cracks in the conventional calculated point according to [GS-G-4.1]. Loss of bearing strength in the calculated point does not mean a failure according to the fatigue mechanism and the problem of crack development to limit sizes that implies a possibility of additional cyclic loading for a structure is under discussion at present. Importance of accident conditions for accumulated cyclic damageability of RP components is not similar as for different RP components so for calculated points inside the components proper. Insignificant contribution of DBA on the background of general low damageability is typical for reactor vessel and especially for steam generator. Pressurizer (PRZ) is representative from the viewpoint of cyclic damageability level. Number of the conditions considered in the fatigue calculation is 19620 for NOC and 471 for AOO, but the number of considered accident conditions is 42, i.e. significantly lower non-comparable number. Thus the contribution of accident conditions for two typical calculated points of primary coolant main injection nozzle into total fatigue damageability is 0,26 and 0,57 respectively, that amounts to 35% and 44%. It proves that WWER-1000 has the margin not only from the viewpoint of fatigue strength, but safety also.

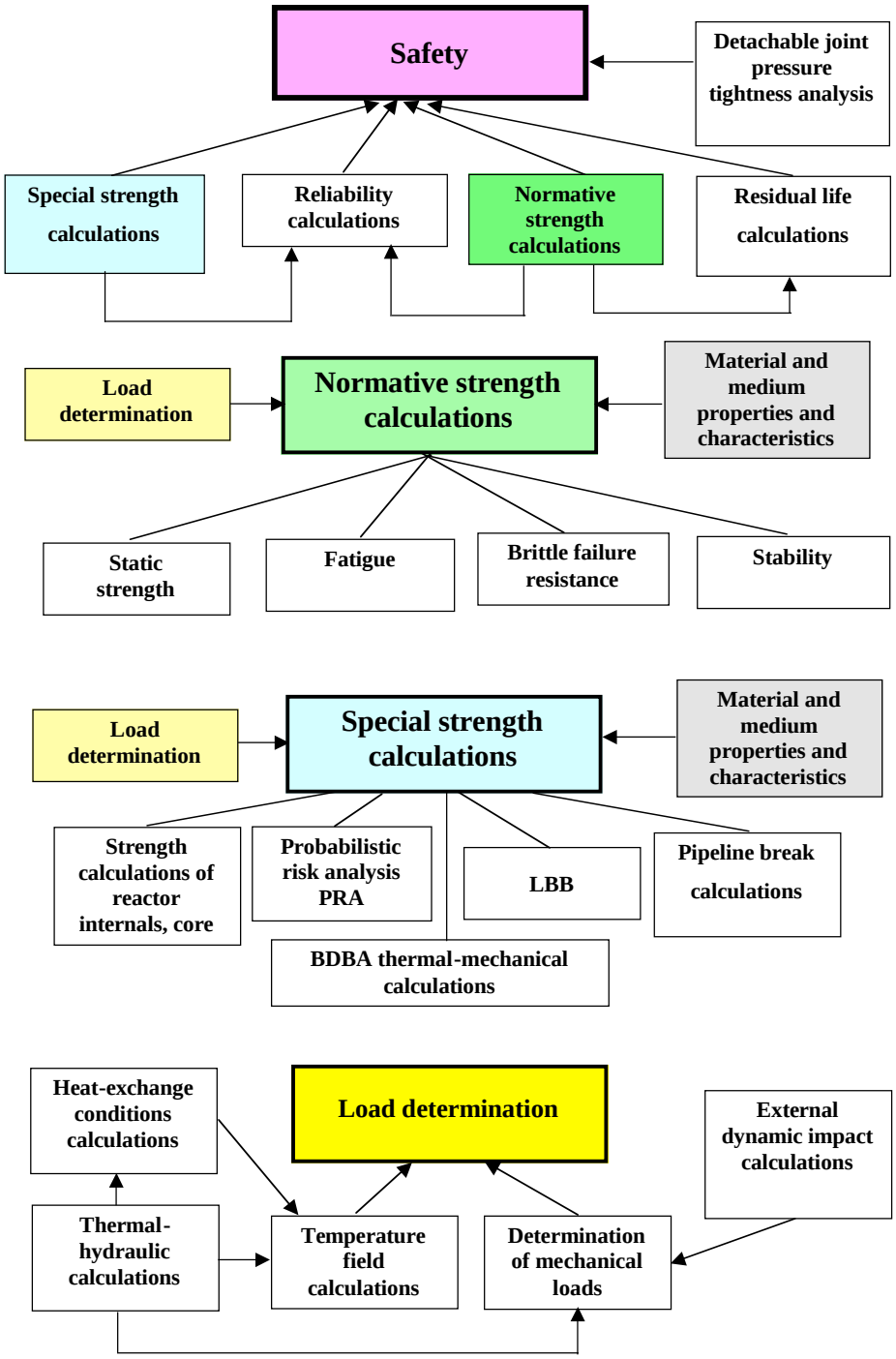

Figure 1. Main types of thermo-mechanical analyses applicably to WWER-1000

Permissible stress ranges for fatigue strength calculation are determined proceeding from cyclic strength experimental data with introduction of strength margins for durability and stresses. Cyclic strength calculation result is a permissible number of repeated operating conditions and service life or permissible thermal and mechanic loads for assigned number of repeated operating conditions and service life [GS-G-4.1]. Used terms service life and durability are also referred to reliability. Thus, durability according to [GOST 27.002-89] is an object property to keep operable state to the limiting condition. The durability is characterized by useful operating life according to non-failure operation and by service life according to time schedule. The ratio between the conditions and service life is provided by the recalculation of conditions number per a time unit.

5. NON-STANDARD THERMOMECHANICAL ANALYSES

Special strength calculations for reactor internals, core, supports, etc. are out of the ranges of valid regulatory documents. Mechanisms of damage related to high-frequency vibration fatigue, loss of stability, accumulation of significant plastic strains and cross-section failure of various type rods can be additionally realized in these structures. All these strength calculations are similar in one factor – they are not directly related to reactor coolant system integrity. Deformation or failure of these structures can determine additional loads or edge conditions in RP vessel or pipeline calculations or influence the core heat removal.

Probabilistic risk analysis (PRA) of reactor coolant components is an additional requirement to the limiting value of reactor vessel failure probability. PRA calculations are not qualified as independent analyses in regulatory documents [PNAE G-01-011-97, NP-082-07]. Attribution of PRA to the field of probabilistic safety assessment (PSA) makes apparent a practical importance of these calculations. The main thing is determination of design weak points and improvement of equipment and pipeline inspection and repair schedules. PRA used for assessment of initiating events related to the failures resulting in leaks can be additionally considered. PRA calculations are based on the concept that equipment and pipelines have crack-type flaws of technological nature, and such flaws can be not revealed during manufacturing according to QP and the flaws resulted from damage accumulation, e.g. under cyclic loading. Probabilistic analysis of crack growth up to critical condition is provided using a hypothesis of statistical distribution of this type flaws. The results of such kind probabilistic assessment have considerable uncertainty still permissible in the frames of PSA. On the other hand, PRA cannot be used as a source for normative regulation within the frames of PSA because of its high uncertainty.

Regulatory documents [PNAE G-01-011-97, NP-082-07] do not specify LBB concept usage, with exception of the requirements for determination of coolant leak location and size. Protection from breaks over pipeline cross-section allows to prevent pipeline end whipping, reactive and active forces from pipeline flow jet to the pipelines, equipment and civil engineering structures and also the effects of pressure increase in building blocks. LBB application is based on regulation [R-TPR-01-99] which specifies the requirements of pipeline material sufficient viscosity, determination of the conditions when possible leak is not resulted in pipeline complete failure and of absolute (undoubted) leak detection provided by available leak monitoring systems long time before the leak crack reaches limiting sizes. Thermo-mechanical calculations considering LBB are based on application of linear and nonlinear fracture mechanics, i.e. DSBM areas being under development at present. It is especially related to the mechanisms of crack extension based on known Paris dependences and treatment of experimental data that results in significant uncertainties which can be presently overcome only using increased conservatism.

LBB is applied to WWER-1000 design for main coolant pipelines Dnom 850, pressurizing system connecting pipeline Dnom 350 and ECCS passive pipelines Dnom 300, i.e. for the pipelines of large diameter. LBB procedures are based on the reverse concept. Firstly, size of the crack for which coolant leak can be surely detected is determined with margin 10 and then this crack is compared to the critical one. Crack length margin shall not be below 2. Thus it shall be previously proved that considered pipelines satisfy to brittle failure resistance (BFR) and cyclic strength criteria according to "Strength standards" [GS-G-4.1]. Extra high requirements for manufacturing and mounting and metal condition in-service inspection provide that the probability of guillotine rupture for the pipelines of large diameter is significantly low though it is not used for LBB concept.

Primary pipelines for which LBB concept is not used shall be equipped with the facilities of monitoring and protection from impermissible displacements under jet forces originating from breaks [NP-082-07]. Missiles can be resulted from pipeline breaks being hazardous for NPP safety within the containment. Very conservative condition of pipeline break locations determination is assumed. Pipeline break is postulated in the section with damageability point above 0,1, and a through-wall crack appears if cumulative fatigue damage exceeds 0,5. Calculations of broken pipeline dynamics have high uncertainty and are significantly conservative that results in expensive design solutions for protection from dynamic effects of this type.

Safety analysis report (SAR) shall include the data on reliability factors for the systems of normal operation, important to safety, and their components referred to safety classes 1 and 2, and also safety systems and components. Thus reliability analysis shall be provided considering common-cause failures

and personnel errors. Reliability analysis is also required for primary systems and components operation in [NP-082-07]. Determination of the functions for RP system and component purposes as specified above also results in the fact that reliability factors for RP equipment and pipelines amount to some part of total factor from the viewpoint of their integrity. The approach based on operational experience and sometimes considering bench tests, e.g. for CPS drive is usually used in the experience of WWER-1000 RP systems and components analysis. Availability of CPS moving components and complicated motion system is the reason that probability for CPS loss of tightness (failure) caused by rupture of housing, stud or loss of tightness in the gasket area (loss of sealing unit tightness) is theoretically low and does not exceed 10^{-6}. RCP set reliability analysis is also representative. Failure rate of vessel components is a theoretical value (about 10-6), resulted from cumulative fatigue damage of vessel components non-exceeding 0,5 with average probability of failure-free operation ~0,89.

The approach based on the concept that crack failures of technological nature are available in the equipment and pipelines [Shary et al. (2004)] is developed recently since the accumulated experience of equipment and pipelines failures is rather limited. Flaws of this type can be not revealed during manufacturing according to QP. Since, probability of the flaws revealing is at least available. Failure probability is suggested to be determined on the basis of statistical spread of the data relating to unsoundness, mechanical properties, crack-resistant characteristics, etc. Acceptance of the concept of equipment process flaw availability assumes that the mechanism of crack-type flaw extension (increase or kinetics) is known and flaw limit sizes and permissible values are known after introduction of safety factors. Paris equation is traditionally used. It is also assumed that in-service inspection of fracture development, which can be even analytical, shall be available. Failure intensity as one of reliability factors is determined from the failure probability. Failure probability calculations for reactor vessel individual areas are below 10^{-8}, and total probability of the vessel failure does not exceed 10^{-7} that satisfies the requirements of [PNAE G-01-011-97].

Additionally, it is also obtained that failure probability for all WWER-1000 RP vessels, including SG steam header does not exceed 10^{-7}. At the same time failure probabilities of vessels and pipelines resulting in leaks of various equivalent diameter can be below 10^{-7} that is taken as initial data for safety analyses.

6. OPERATIONAL STRENGTH CALCUALTIONS

Nuclear safety assurance during operation is required both in OPB [PNAE G-01-011-97] and PBYa [NP-082-07]. Operating organization shall provide for this purpose the process specification of NPP unit safe operation (on the basis of design documentation), including elaboration of test and inspection instructions in one of the items. Using of the system of residual cyclic life computerized monitoring (SACOR) is an important component of safe operation. Main purpose of this system is recording of damageability accumulation according to fatigue mechanism by actual realization of operating conditions. Residual service life is determined as a result of nontrivial problem solution by determination of the conditions combinations permissible for realization and not resulting in exceeding of fatigue damageability in the most loaded (control) points of equipment more than 1. Mechanisms of crack growth under cyclic loading are considered in expanded version of SACOR.

In case a flaw is detected during in-service inspection which sizes exceed acceptance criteria for a certain type of inspection (exceed the permissible sizes in other words), the specified flaw shall be repaired or damaged component removed if the repair is inexpedient. Flaw permissible sizes specified in [PNAE G-7-009-89/ PNAE G-010-89] which are rather conservative as are initially intended for use at the stage of equipment manufacturing are actually used as acceptance criteria for in-service inspection. Since, there is a possibility to allow further operation of damaged structure component (without replacement or repair) if it is possible to prove that the flaw does not impair component strength. In this case a calculation analysis for detected flaw admissibility shall be provided proceeding from [125-02-95, -02-91]. Decision of damaged structural component admissibility for further operation is generally substantiated by additional (more frequent that usual) nondestructive test of the component purposed to the monitoring of possible flaw increase during operation. Analysis of schematized (simulated) flaw cyclic extension is provided using a well-known Paris ratio. Thus medium corrosion effect onto cyclic crack extension rate is taken into account in specified cases [-02-91].

Analysis of flaw admissibility is provided using stability criteria for a schematizing (simulated) crack considering cyclic extension to the time moment for which the analysis is provided (generally it is either the end of design service life or the period of next inspection). Stability criteria for brittle or quasi-brittle fracture and also limiting plastic conditions are specified in [PNAE G-7-009-89/ PNAE G-010-89]. In the first case the analysis is provided using the methods of linear fracture mechanics. In the second case, typical for austenitic steel structural components, simplified dependences are used for plastic stability of a cross-section with flaw available or a ligament between the flaw and structural component opposite surface, obtained using the methods of nonlinear fracture mechanics [-02-91]. Stability criterion of the crack simulating a flaw should be tested for the whole spectrum of design conditions including accident situations and seismic effects.

7. CONCLUSION

The analysis above of regulatory document requirements for reactor coolant system thermo-mechanical calculations proves comprehension of WWER-1000 RP safety assessment regarding thermo-mechanical analyses. Sufficient and sometimes excessive conservatism of the requirements for strength and safety can be emphasized. Russian and western regulation requirements for reactor coolant system integrity are similar except for the requirements for safety analysis. Variations can be available in standards and rules for designing, manufacturing, mounting and operation and somehow in criteria requirements for strength that demands additional research. The demand for more exact differentiation of the requirements for thermo-mechanical calculations of reactor coolant system integrity and calculation analyses of reliability and safety is general. Differentiation between standard and non-standard calculations of reactor coolant system components shall decrease the scope of obligatory calculation analysis and lower criteria requirements for non-standard calculations.

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