STRENGTH ANALYSES OF NUCLEAR POWER PLANT PRESSURE EQUIPMENT

continues

With regard to new nuclear facilities, this Guide shall apply as of 1 December 2013 until further notice. With regard to operating nuclear facilities and those under construction, this Guide shall be enforced through a separate decision to be taken by STUK. This Guide replaces Guide YVL 3.5.

First edition
Helsinki 2014

ISBN 978-952-309-130-6 (print) Kopijyvä 2014
ISBN 978-952-309-131-3 (pdf)
ISBN 978-952-309-132-0 (html)

STUK • SÄTEILYTURVAKESKUS
STRÅLSÄKERHETSCENTRALEN
RADIATION AND NUCLEAR SAFETY AUTHORITY

Osoite/Address • Laippatie 4, 00880 Helsinki
Postiosoite / Postal address • PL / P.O.Box 14, FIN-00881 Helsinki, FINLAND
Puh./Tel. (09) 759 881, +358 9 759 881 • Fax (09) 759 88 500, +358 9 759 88 500 • www.stuk.fi

Authorisation

According to Section 7 r of the Nuclear Energy Act (990/1987), *the Radiation and Nuclear Safety Authority (STUK) shall specify detailed safety requirements for the implementation of the safety level in accordance with the Nuclear Energy Act.*

Rules for application

The publication of a YVL Guide shall not, as such, alter any previous decisions made by STUK. After having heard the parties concerned STUK will issue a separate decision as to how a new or revised YVL Guide is to be applied to operating nuclear facilities or those under construction, and to licensees' operational activities. The Guide shall apply as it stands to new nuclear facilities.

When considering how the new safety requirements presented in the YVL Guides shall be applied to the operating nuclear facilities, or to those under construction, STUK will take due account of the principles laid down in Section 7 a of the Nuclear Energy Act (990/1987): *The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified considering operating experience, safety research and advances in science and technology.*

According to Section 7 r(3) of the Nuclear Energy Act, *the safety requirements of the Radiation and Nuclear Safety Authority (STUK) are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations. If the licensee can convincingly demonstrate that the proposed procedure or solution will implement safety standards in accordance with this Act, the Radiation and Nuclear Safety Authority (STUK) may approve a procedure or solution by which the safety level set forth is achieved.*

Translation. Original text in Finnish.

1 Introduction

101. It is of primary importance for the safety of a nuclear power plant that its pressure retaining structures and components, primary circuit components in particular, withstand with adequate margin the loadings and other environmental effects produced under design basis operational states. The technical solutions of system and component design provide protection against these effects.

102. The adequacy of strength is demonstrated by dimensioning calculations and strength analyses or equivalent experimental evidence included in the design documents. Data verifying their coverage and reliability is obtained by commissioning test programmes. Monitoring during operation ensures that loads remain within the limits set for strength analyses and that no harmful structural degradation occurs.

103. Government Decree on the Safety of Nuclear Power Plants (717/2013) sets forth legal justification applicable to strength analyses and verification of strength.

Under Section 3(2) of Government Decree (717/2013) *nuclear power plant safety and the technical solutions of its safety systems shall be assessed and substantiated analytically and, if necessary, experimentally. Analytical methods include transient and accident analyses, analyses of internal and external hazards, strength analyses, failure resistance analyses, failure mode and effects analyses, and probabilistic risk assessments. The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear power plants, the results of safety research, plant modifications, and the advancement of calculation methods.*

Under Section 3(3) of Government Decree (717/2013) *the analytical methods employed to demonstrate compliance with safety requirements shall be reliable and well qualified for the purpose. The analyses shall demonstrate the conformity with the safety requirements with high certainty. Any uncertainty in the results shall be assessed and considered in determining safety margins.*

Under Section 5 of Government Decree (717/2013) *the design, construction, operation, condition monitoring and maintenance of a nuclear power plant shall provide for the ageing of systems, structures and components important to safety in order to ensure that they meet the design-basis requirements with the necessary safety margins throughout the service life of the facility. Systematic procedures shall be in place for preventing the ageing of systems, structures and components which may deteriorate their availability, and for the early detection of the need for their repair, modification and replacement. Safety requirements and applicability of new technology shall be periodically assessed, in order to ensure that the technology applied is up to date, and the availability of the spare parts and the system support shall be monitored.*

Under Section 13(4) of Government Decree (717/2013) *in order to ensure primary circuit integrity*

1. *the primary circuit shall be designed and manufactured in compliance with high quality standards so that the probability of hazardous faults in structures and that of mechanisms threatening their integrity remains extremely low and any faults which occur can be detected reliably through inspections*
2. *the primary circuit shall, with sufficient margins, withstand the stresses arising in normal operational conditions, anticipated operational occurrences, postulated accidents and design extension conditions*
3. *the primary circuit and systems immediately connected to it, and components important to the safety of the secondary circuit of a pressurised water reactor, shall be reliably protected during anticipated operational occurrences and all accident scenarios, in order to prevent damage caused by over-pressurisation*
4. *the water chemistry conditions of a nuclear power plant's primary circuit and a pressurised water reactor's secondary circuit shall not cause mechanisms that threaten their integrity, and*

5. *in order to detect leakages, the facility shall be equipped with sufficient monitoring systems.*

Under Section 13(5)(1) of Government Decree (717/2013) *the containment shall be designed to maintain its integrity during anticipated operational occurrences and, with a high degree of certainty, during all accident scenarios.*

Under Section 17 of Government Decree (717/2013) *the design of a nuclear power plant shall take account of external hazards that may challenge safety functions. Systems, structures and components shall be designed, located and protected so that the impacts on plant safety of external hazards deemed possible remain minor. The operability of systems, structures and components shall be demonstrated in their design basis external environmental conditions. External events shall include exceptional weather conditions, seismic events, impact of accidents taking place in the plant's vicinity and other factors resulting from the environment or human activity. The design shall also consider unlawful actions with the aim of damaging the plant and a large commercial airplane crash.*

Under Section 18 of Government Decree (717/2013) *the design of a nuclear power plant shall take account of any internal hazards that may challenge safety functions. Systems, structures and components shall be designed, located and protected so that the probability of internal events remains low and impacts on plant safety minor. The operability of systems, structures and components shall be demonstrated in the room specific environmental conditions used as their design bases. Internal events to be considered shall include fire, flood, explosion, electromagnetic radiation, pipe breaks, container breakages, falling of heavy objects, missiles resulting from explosions and component failures, and other possible internal events.*

Under Section 21 of Government Decree (717/2013) *during construction, the holder of the nuclear power plant's construction licence shall ensure that the plant is constructed and implemented in compliance with the safety requirements and the approved plans and procedures. The licensee is obliged to also ensure that the plant supplier and sub-suppliers delivering services and products important to safety operate in compliance with the safety requirements.*

Under Section 22(1) of Government Decree 717/2013 *in connection with the commissioning of a nuclear power plant, the licensee shall ensure that the systems, structures and components and the plant as a whole operate as designed.*

2 Scope of application

201. The present Guide sets requirements for the loadings and strength analyses of the nuclear power plant's primary circuit and other nuclear pressure equipment important to safety. The requirements apply, to the appropriate extent, to the licence applicant or licensee, component or plant suppliers as well as other design, testing and expert organisations.

202. In the present Guide, loadings comprise the mechanical and thermal stress factors arising in the nuclear power plant's normal operation, and design basis and design extension operational conditions, against which the pressure equipment design shall present adequate safety margins, demonstrated by strength analyses. The strength analyses include stress analysis, brittle fracture analysis and analyses demonstrating the implementation of the leak before break principle (LBB). To be analysed are also loadings when their values are to be derived by calculations from the operation of systems, behaviour of buildings or loading conditions exerted on pressure equipment by local phenomena. The LBB principle is presented as a procedure for excluding pipe break induced impact loads from the design bases.

203. The present Guide discusses the aforementioned analyses from the viewpoint of participating organisations' activities as well. Quality management requirements are set for the making, reporting, procurement and review of these analyses. Plant construction and modification projects require systematic project management and the obtaining of approval for strength analyses significant for the design bases in connection with the project's licensing procedure.

204. For plant construction and modification projects, this Guide presents a strength analysis coverage and reliability verification procedure consisting of pre-operational tests and measurements. Monitoring during service life covers the maintenance of recurring loads within fatigue analysis assumptions, and the effects of the operating environment on the mechanical properties of materials, special emphasis being on radiation embrittlement of the reactor pressure vessel.

205. The following matters discussed in other YVL Guides pertain to the scope of application of the present Guide:

- dimensioning and simplified stress analyses, Guides YVL E.3, YVL E.8 and YVL E.9 on mechanical components
- containment strength analyses, Guides YVL B.6 and YVL E.6
- the LBB principle implications for safety systems design, Guide YVL B.5
- seismic design of structures and components as well as protection against pipe breaks by layout design, Guide YVL B.7
- qualification of non-destructive in-service inspections and acceptance of detected flaw indications by analytical evaluation, Guide YVL E.5.
- utilisation of the monitoring of loads and material properties in ageing management, Guide A.8
- nuclear power plant safety design and analyses, Guides YVL B.1 and YVL B.3.
- general requirements for management systems and quality plans, Guide YVL A.3.

206. Implementation of safety level in compliance with the present Guide is required, where necessary, also at other nuclear facilities equipped with a nuclear reactor and at other nuclear facilities to the extent appropriate for their estimated accident probability and consequences. Other nuclear facilities are discussed in Guides YVL D.3 and YVL D.5.

3 Strength analysis report

3.1 Contents and objective

301. A strength analysis report or an equivalent document made up of separate reports shall be submitted to STUK for approval concerning the pressure equipment and their parts referred to in para. 501. The strength analysis report shall present a stress analysis, with associated fatigue and accident condition analyses, as well as a brittle fracture analysis in the cases referred to in chapter 6.1.

302. Pursuant to para. 703 of chapter 7.2, the strength analysis report shall also present an LBB analysis for piping for which the LBB principle shall or can be demonstrated

303. The purpose of the strength analysis report is to demonstrate that the item of pressure equipment, when subjected to the design basis load conditions, fulfils the applicable strength analysis standard requirements as well as potential other requirements set for strength design.

3.2 Time of submission

304. The strength analysis report shall be submitted to STUK as part of the construction plan for the item of pressure equipment. On a case-by-case basis, and for substantiated reasons, STUK may allow a later submission date for the strength analysis report or a document belonging to it.

305. A construction plan submitted on pressure equipment modifications shall include strength analyses revised as necessary if the item of pressure equipment's design pressure or temperature, configuration, wall thicknesses, material values, operations, supporting or other factors relating to the item of pressure equipment change in a way that may compromise the safety margins to be attained.

306. If, during the manufacturing or modification of the item of pressure equipment or during its subsequent pre-operational testing, a deviation is detected from the strength analysis input data based on which the manufacturing or modification construction plan in question was approved, calculations revised as necessary shall be submitted to STUK.

307. Strength analyses shall be revised during service life if the item of pressure equipment exhibits load increases, reduction in wall thickness or decreasing toughness values deviating from the design bases. Revision may also become necessary due to an extension of service life, periodic safety assessment or an event affecting safety of which due account could not be taken during design.

308. The performance of strength analyses on primary circuit components shall begin in good time to make preliminary data on the results available as early as during the approval review of construction materials. In applying for a construction licence for a new nuclear power plant, evidence shall be given on these analyses being capable of fulfilling the set criteria. To be examined in particular are strength-determining structural elements and loading conditions as well as construction solutions deviating from other nuclear power plants of an equivalent type.

3.3 Format of presentation

309. The strength analysis report shall be so clearly written that the data provided enables the reviewer to ascertain correctness of the strength analyses conducted and the acceptability of their results. The strength analysis report shall unambiguously indicate the up-to-date input data and their sources, applicable standards, analysis methods and calculation models used, as well as the results, conclusions and their justification.

310. The results of the strength analyses shall be presented extensively enough for the easy conclusion and verification of the most stressed points in the structures analysed. Intermediate results to verify the correctness of essential analysis phases shall be presented. The results and intermediate results may be complemented by separately submitted digital documentation.

311. In submitting stress analyses of pressure equipment in accordance with the present Guide, to be presented in particular are:

a. acceptance markings by suppliers and contractors
b. summary of the review by licence applicant or licensee
c. identification markings, safety class and other classification data
d. operability-related requirements
e. description of the modelling method and progress of the analysis
f. computer programs and other software based procedures
g. construction drawings, isometrics and system diagrams or reference to them
h. design and service loadings as well as pressure tests
i. supports and mass distribution data
j. data on dimensioning calculations and wall thicknesses used in analysis
k. strength values of construction materials and other physical properties at room and design temperature as well as a definition of the fatigue curve
l. boundary conditions, symmetry conditions and other assumptions
m. formulae and units used as well as notation if not standard-compliant
n. temperature and stress distributions of most important cross sections as wellas displacement diagrams
o. stress classification into stress types with given acceptance limits
p. computer printouts required for reviews and provided with adequate explanations

4 Loadings

4.1 Documentation to be submitted

401. A strength analysis report shall present the pressure equipment's design loadings, service loadings, pressure tests and other tests required for reasons related to strength, as well as essential transportation-induced loads.

402. Detailed data on loadings and their computational modelling shall be submitted to STUK as separate analysis reports relating to the design of systems or civil structures.

403. Loading analyses significant to the design bases of a new nuclear power plant, those for the primary circuit in particular, shall be submitted to STUK when applying for a construction licence.

404. Obligations relating to the way of reporting the plans and results of load monitoring are determined on the basis of STUK requirements for pre-operational testing and ageing management.

4.2 Design loadings

405. Design loadings, to be determined in accordance with the applicable standard, comprise design pressure, design temperature and other mechanical design loads which in combination with design pressure produce the highest primary stresses in normal operational conditions.

4.3 Service loadings

406. Service loadings relate to a nuclear power plant's design-basis normal operational conditions, anticipated operational occurrences and postulated accidents as well as to internal and external events. For each item of pressure equipment to be analysed, the service conditions shall be determined for those conditions and events where integrity or operability is required of the item of pressure equipment.

Grouping

407. Service loadings shall be grouped into categories primarily based on how each loading condition is allowed to affect an item of pressure equipment's integrity or operability. In strength analyses conducted in accordance with the present Guide, grouping is as follows:

A. loading conditions under which the pressure equipment is in normal operation as designed

B. deviations from loadings generated by normal operation, which the pressure equipment is designed to withstand during its design service life without repairs

C. abnormal loading conditions not considered in the pressure equipment's service life calculations, which may lead to local deformations so extensive that inspection and possible repair of the pressure equipment is required before operation is continued

D. abnormal loading conditions which, while pressure-bearing capacity is further maintained, result in a large distortion of the shape of the pressure equipment and may require its removal from service.

408. In grouping service loadings, it shall be taken into account that postulated accidents must not compromise the operability of pressure equipment in safety systems whose operation is consequently required. An acceptable premise is that

- assigned to Category B are service loadings that must not compromise active operability requiring a component or its part to change position
- assigned to Category C are service loadings under which the component must maintain its shape as needed to permit a required flow rate to pass through the component (functional capability).

409. Extremely rare accident loads, e.g. vibrations and explosion pressure waves caused by a large commercial airliner crash or seismic conditions in excess of the design basis earthquake, may be addressed as design extension conditions applying best estimate methods. The effect thus analysed on the integrity of the pressure equipment or operability required of it in the accident in question must not exceed the effects allowable for service loadings in Categories D or C, respectively.

4.4 Loading analyses

410. The documents on service loadings shall present the mechanical and thermal loads exerted on an item of pressure equipment with their time-dependencies and numbers of occurrence calculated for the entire service life. These shall be derived in the necessary scope from specifications and analyses pertaining to the operation and behaviour of the nuclear power plant's structures and systems. Applicable operating experiences and experimental analyses may be provided as justification.

411. The documents on service loadings shall take into account local factors and phenomena significant for the strength of an item of pressure equipment and its structural elements. Thus e.g. compressive loads caused by external overpressure or other factors shall be accounted for to conduct a stability analysis on the pressure equipment in question or their parts.

412. If a service load is due to a plant event causing several load phenomena, the various partial loads shall be combined in a manner that reliably takes into account their potential combined effects.

Thermal loads

413. Thermal loads transferred to the structures of pressure equipment analysed in accordance with the present Guide shall be modelled in detail. To be established for the flexibility and fatigue analyses of piping are largest temperature changes as well as potential thermal stratification and temperature interface fluctuations. Time-dependent heat transfer and uneven temperature distribution shall be examined for rapid variations in temperature through large thickness materials, e.g. flanged joints.

414. Thermal loading analyses shall present the applied load specification, the analysis methods for flow and thermal fields and the way of determining the heat transfer coefficients.

Impact loads

415. Rapid pressure transients due to pipe breaks postulated based on para. 705 or due to safety and isolation valve actuation or due to equipment malfunctions shall be analysed with dynamic methods. These shall determine the loads acting on an equipment assembly as well as its supports and internals important to safety when subjected to the pressure transient and consequent mechanically transferred vibrations. Furthermore, accelerations experienced by important functional components shall be analysed.

416. If piping or connected components are provided with restraints against whipping of broken pipes, the thereby accomplished decrease in leakage flow and ensuing pressure transient can be taken into account when determining for postulated pipe breaks the loads on the equipment assemblies and internals referred to in para. 415.

417. For postulated pipe breaks, an analysis shall be conducted of the jet impingement forces on important components in close proximity, as well as of the impact absorbing capability of whip restraints and the pipe whip loads acting on them. If local pipe wall deformation capability is credited here, an account of it shall be given in the analysis. Pipe whip loads transferred to other components and structures shall be considered in providing protection against component failures as referred to in Guide YVL B.7.

418. Potential for condensation water hammer and water slug formation shall be examined in components where steam and cold water mix. Furthermore, the possibility of primary or secondary circuit water entering the main piping of the steam water systems shall be analysed.

419. Impact loading analyses shall present the assumptions used for pipe breaks and valve functioning or malfunctioning, the analysis methods pertaining to fluid and structural dynamics, as well as the way of computational modelling of forces induced by a pressure transient, jets and pipe whips.

Seismic loads

420. The dynamic effects of a design basis earthquake shall be analysed by seismic analysis conforming to a standard applicable to seismic category S1 pressure equipment and their supports in accordance with Guide YVL B.2. Best estimate methods can be applied to seismic category S2A pressure equipment. The loading condition data comprise the floor response resolved by analysing the dynamic behaviour of buildings. Its variations between different supporting points shall be analysed if significant additional stresses result from them.

421. The following shall be established as regards seismic analysis: the floor response spectrum applied, the equipment assembly analysed, dynamic modelling methods, damping ratios as well as free-vibration solutions and accelerations experienced by functional components.

4.5 Load monitoring

Pre-operational testing

422. Tests and measurements shall be conducted during the pre-operational testing of the nuclear power plant to ensure that the actual loads of pressure equipment do not exceed the values used as the input data for strength analyses, and that no detrimental loads have been ignored. The objective of the tests and measurements can also be the determination of stress states (experimental stress analysis) for an item of pressure equipment having an unusual structural configuration or manner of loading.

423. The items subject to the testing and measurement programmes shall include those maximum loading conditions and loading phenomena anticipated during service life whose computational analysis is uncertain. Among these are pressure transients caused by safety and isolation valves, temperature transients during startup and emergency cooling, thermal stratification, as well as the vibrations of piping and reactor pressure vessel internals [4] and the adequacy of clearances for thermal displacements.

424. The experimental analyses necessary to verify the strength of the nuclear power plant's safety-important pressure equipment during accidents shall be obtained by separate test arrangements representing accident conditions. These analyses shall be handed over to STUK with the design documentation of the plant project in question.

425. The most demanding analyses may be excluded from the pre-operational testing and measurement programme and the analyses referred to in para. 424, in case they have been previously implemented in a corresponding nuclear power plant and the reports presenting the results are submitted to STUK.

426. It shall be possible to compare the observations made during pre-operational testing with the results of load monitoring conducted during service. Supplementary measurement data on the behaviour of equipment assemblies and significant local stress conditions shall be obtained, which support the specification of these loadings based on the measurement data monitored during service.

Operation

427. Operational conditions and events inducing fatigue loads on the most important pressure equipment shall be recorded over the service life of the nuclear power plant. Sufficient measurement data shall be collected on their progress for later verification of essential factors. The monitoring shall be so arranged that unexpected load types occurring during operation are also detected.

428. The clearances for thermal displacements shall be controlled, particularly those for piping provided with whip restraints.

429. The primary circuit and, in a pressurised water reactor plant, the main piping of the steam and feed water systems shall be provided with permanent monitoring systems capable of producing loading data during operation. The variables to be monitored include i.a. process variables, surface temperatures, strains and vibrations.

430. The nuclear power plant's seismic monitoring equipment shall be implemented so as to permit establishing from the recorded measurement data the stresses imposed on primary circuit components by earthquakes and aeroplane crashes.

5 Stress analysis

5.1 Items to be analysed

501. A stress analysis in accordance with the present Guide shall be conducted on pressure equipment or their parts which are to be constructed in accordance with the highest safety and quality requirements applicable to a nuclear power plant. Items of special importance are

a. Safety Class 1 primary circuit components including main circulating pumps
b. in a pressurised water reactor plant, the secondary side of the steam generator as well as the main piping segments of the steam and feed water systems inside the containment building, up to and including the outer isolation valves
c. supporting structures of the primary circuit
d. internal structures of the reactor pressure vessel that support the reactor core and are important to its cooling capacity
e. piping penetrations important to containment leak tightness, which are subject to fatigue loads.

502. For other nuclear pressure equipment and pumps in nuclear facilities, the necessity of a stress analysis in accordance with the present Guide is determined primarily based on the applicable design standard. Further instructions are given in Guides YVL E.3, YVL E.8 and YVL E.9 in force for the component types in question. Even if a simpler approach (design rules, type testing) is used, the load definition and quality management shall be based on the requirements in chapters 4 and 8 of the present Guide, where applicable.

503. In their regulatory inspection, STUK or an authorised inspection body may also request a stress analysis accordant with the present Guide in other cases to verify an adequate level of safety. The arguments for this procedure may be e.g. the risk significance of an item of pressure equipment or pump, operability requirements or limited in-service inspection possibilities.

5.2 Applicable standards

504. In conducting a stress analysis of pressure equipment and pumps in accordance with the present Guide, the standard ASME Boiler and Pressure Vessel Code Section III, Division 1 (ASME III) [2] shall apply as a general rule. The mandatory regulations stated in its Articles NB 3200 and NB 3650 as well as in its Subsections NF and NG apply in these cases in so far as STUK has not presented more detailed requirements.

505. Alternatively, another design and strength analysis standard for Safety Class 1 pressure equipment, approved by a foreign regulatory authority and corresponding in principle, may be submitted to STUK for approval. One condition of approval is that the standard in question has been applied previously in the construction of nuclear power plants of the same type.

506. STUK may in its decisions specify in more detail the method of application of standards and their different editions acceptable for stress analysis. Among other things, account may be taken of how the requirement level of the applicable standard series is in its entirety fulfilled for the item of pressure equipment or pump to be analysed.

5.3 Modelling of structural configuration

Extent and detail needed

507. The structural configuration of an item of pressure equipment shall be modelled to an adequate extent by stress analysis. The model shall include the zones significantly affected by loadings and those parts of the item of pressure equipment significant to the development of the failure modes on which the limits of acceptability are based.

508. Computational modelling confined to individual parts of an item of pressure equipment may be used if the stress fields are not affected by structural discontinuities in close proximity, and if the correctness or conservatism of boundary conditions is substantiated. In applying symmetry conditions, the item of pressure equipment's geometry, loadings and other boundary conditions as well as material properties shall be simultaneously considered.

509. A locally refined modelling of the structural configuration is required if the item of pressure equipment is subject to strong temperature gradients and local mechanical loads, or contains essential structural discontinuity such as variable thermal expansion coefficients and wall thicknesses.

510. The models shall consider any strength behaviour 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.

511. The models shall consider non-linearities due to pretensioning, friction forces and clearances between structural elements, when significant to stress fields. The computational models of supports shall describe their strength characteristics essential for bearing the loads, the limitation of the movement of the item of pressure equipment in different directions, and attachment to civil structures.

Piping system models

512. The modelling of piping segments for stress analysis shall terminate at each end in a more rigid and massive other item of pressure equipment or structure, and the possibly transferred forced displacements shall be taken into account. Piping supports and pressure equipment following the movements of piping shall be so modelled that their effect on flexibility, damping and mass distribution corresponds to the actual configuration. For piping provided with whip restraints, clearances appropriate for all operational conditions and accidents shall be determined by calculations.

513. In piping system stress analysis, the associated fittings can be described by conservative simplifications in accordance with the applicable standard. The adequacy of the strength of the as-built configuration under design loadings shall then have been established with true load-bearing capacity determined by computational modelling or a test piece representing the manufacturing technology, geometry and wall thicknesses of the actual product.

514. The modelling by stress analysis of pressure-bearing parts connected to piping penetrations and to other points of attachment shall facilitate the reliable analysis of stresses induced by the piping and its limited movements as well as temperature variations in the points of attachment. The adequate strength of a penetration's other steel parts shall be demonstrated in accordance with their design standard.

Material properties

515. Stress analysis shall use material strength values and physical properties which are accordant with the applicable standard [1] and valid for the temperatures under examination. For other separately approved materials, these values shall be determined in accordance with the design standard.

5.4 Acceptance limits

516. Stress analysis shall demonstrate the fulfilment of the service limits set by the applicable standard for the item of pressure equipment in question or its part in accordance with its classification, service load grouping, number of pressure tests and the analysis methods used. The states of stress calculated by stress analysis shall, therefore, be classified into stress types which are significant for the maintenance of integrity and operability and relative to which each service limit is set. This procedure shall be repeated for all analysed structural parts and loads of the item of pressure equipment.

517. If other acceptance limits, e.g. relating to deformations, are set for the strength of an item of pressure equipment, these shall be included in the item's construction plan and their fulfilment shall be demonstrated by stress analysis.

518. Stress analyses of piping and the connected pressure equipment shall include a check that the loads imposed on their connecting nozzles do not exceed service limits set for each item to ensure their integrity and operability.

5.5 Fatigue analysis

519. In connection with the stress analysis, a fatigue analysis shall be conducted on pressure equipment parts subjected to recurring loads or loads of variable magnitude, unless the possibility of fatigue is excluded by conservative estimations in accordance with the applicable standard.

520. Among the baseline data for fatigue analysis shall be a design fatigue curve which applies to the material under the conditions under examination and presents the stress amplitude of harmonic load allowable in elastic analysis as a function of the load cycle number.

521. If repeated yielding occurs in an area larger than a local structural discontinuity, elastoplastic methods in accordance with the applicable standard shall be used in the fatigue analysis.

Stress concentrations

522. Peak stresses due to local structural discontinuities shall be defined in fatigue analyses either by detailed design modelling of geometry or by approximate analysis using experimental fatigue strength reduction coefficients or elastic stress concentration factors. Stress concentrations of welded joints shall be assessed on the basis of as-built geometry.

523. For claddings included in pressure equipment modelling based on para. 510, a fatigue analysis shall be performed to demonstrate avoiding detrimental fatigue in the cladding due to its greater thermal expansion.

Environmental effect

524. The fatigue analysis shall consider the environmental effects which arise from actual process conditions and decrease fatigue life. Environmental effect-related factors in the primary circuit include, among other things, dissolved oxygen content of coolant, operating temperature, material impurities and load-induced strain rate. Environmentally affected fatigue analysis shall primarily be performed using the methods given in [5]. Potential other procedures shall be submitted to STUK for approval.

Cumulative damage

525. The several types of fatigue loads exerted on pressure equipment during their service life shall be combined by a method specified in the applicable standard. Cumulative damage caused by the different specified partial loads jointly over the item of pressure equipment's entire service life shall be determined for the points most susceptible to fatigue.

6 Brittle fracture analysis

6.1 Items to be analysed

601. Brittle fracture analysis shall be performed on the most stressed ferritic steel parts of the nuclear power plant's Safety Class 1 pressure equipment. Most important are the core area of the reactor pressure vessel, its largest pipe nozzles and the flanged joint of its head. Other possible items requiring a brittle fracture analysis are the secondary side of a pressurised water reactor plant's steam generator, the steel containment, and the shafts and flywheels of the main circulating pumps.

6.2 Analysis methods

602. 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. The calculation method of the fracture mechanical parameters used shall be presented.

603. The analysis of other items may be more simplified and based on e.g. the difference between the lowest service temperature and the ductile-to-brittle transition temperature.

6.3 Toughness values

604. Brittle fracture analysis shall elaborate on the K_{Ic} values used and their dependency on the material temperature. Acceptable values for the most common reactor pressure vessel steel grades are provided as so-called reference curves in a standard [3]. The values to be used for other steel grades shall be substantiated separately. The ductile-to-brittle transition temperature shall be defined in accordance with the applicable material standard.

605. Fracture toughness can also be determined by creating a master curve fit to the toughness values measured from the pressure equipment material in accordance with reference [7]. The values applied to reactor pressure vessel materials in particular shall be verified in this way. The toughness values obtained as a result of quality control can be used on a case-by-case basis, taking into consideration the effect of the non-homogeneity and crystal orientation of steel as well as the number and scatter of the test results.

606. Brittle fracture analyses shall also take into account the reduction in toughness values due to possible ageing during service life. In dynamic load conditions, high material strain rate is another factor that reduces fracture toughness.

607. STUK shall be given the opportunity to oversee experimental analyses relating to the determination of toughness values.

6.4 Radiation embrittlement

608. The brittle fracture analysis of the reactor pressure vessel shall present a calculated prediction for a transition temperature shift resulting from neutron radiation. The conservative character of the prediction shall be verified during service life by means of a radiation embrittlement surveillance programme

609. The prediction of the radiation shift shall be based on an empirical correlation between the alloying elements of steel, chemical impurities and the fast neutron fluence. During service life, the prediction may be based on a fit to the results of the radiation embrittlement surveillance programme. The non-homogeneity of material and the scatter of test results shall then be taken into account. The values used shall be substantiated.

6.5 Allowable pressure versus temperature

610. In the brittle fracture analysis, the maximum allowable pressure for an item of pressure equipment at different temperatures in normal operational conditions shall be calculated. The calculated values shall be taken into account in determining the allowable pressure and temperature range in accordance with the Technical Specifications. In addition, the lowest allowable temperatures for the item of pressure equipment's pressure tests shall be calculated.

611. The postulated cracks and calculation methods presented in Appendix G of reference [2] may be used in the analysis. Other calculation methods shall be submitted to STUK for approval.

6.6 Operational occurrences and accidents

612. Transient and accident conditions involving the heavy cooling of the wall of an item of pressure equipment or the pressurisation of a cold item of pressure equipment shall be examined by brittle fracture analysis. The factors to be considered in the reactor pressure vessel analysis include

a. initiating events leading to emergency cooling
b. operation of plant systems
c. activities of plant operators
d. locally colder area due to emergency coolant
e. flow rates and heat transfer
f. stresses due to pressure and temperature differences
g. unequal thermal expansion and heat conduction of cladding and base material
h. residual stresses
i. crack size, shape, orientation and location
j. toughness values and radiation shift.

613. Relevant event sequences shall be examined in selecting loading conditions for analysis. The reactor pressure vessel's in-service inspection programme shall enable reliable detection of the cracks assumed in the analysis. The different factors and input data used shall be substantiated.

614. The acceptance criterion shall be in accordance with the safety margin required in the applicable standard as regards the growth of a postulated crack. An analysis of the safety margin provided by crack arrest may be considered as an additional justification on a case-by-case basis.

6.7 Probabilistic analysis

615. If, on the basis of a transient and accident condition analysis in accordance with chapter 6.6, the risk of brittle fracture cannot be determined to be negligible, the construction plan of the reactor pressure vessel shall also present an analysis of brittle fracture probability where corresponding factors are considered. The acceptance criterion implies the requirement that the brittle fracture probability is very low and only accounts for a small percentage of the overall probability of reactor core damage evaluated by means of a probabilistic risk assessment (PRA).

6.8 Other fast fracture considerations

616. In connection with the strength analysis of Safety Class 1 pressure equipment, an assessment shall be given on the potential for a fast fracture occurring in the upper shelf area where temperatures exceed the transition temperature zone. This could occur in thick-walled points of an item of pressure equipment which undergo rapid cooling under high pressure. The adequacy of the toughness values of the upper shelf shall be analysed, where necessary. The methods and criteria used are subject to STUK's approval.

7 Leak before break analysis

7.1 Safety principles

701. Break preclusion (BP) shall refer to the principle of using advanced technical and organisational procedures in piping construction, operations and maintenance in such a way that the LBB principle will be implemented and this may be credited for the nuclear power plant design and its provision against the consequences of a complete, instantaneous pipe break.

702. Leak before break (LBB) shall refer to a principle where no such failure mechanisms may be identified for the piping that could present a potential for its complete, instantaneous break. If a material defect remains undetected in piping inspections, only a small localised leak may result from it, even in the worst case, based on whose detection sufficient time remains to safely place the plant in an operational state where no danger of a complete pipe break exists.

7.2 Scope of application

703. This chapter presents the requirements for the demonstration of the LBB principle and for the technical and organisational implementation of the BP principle. Those piping for which the LBB principle shall or can be demonstrated are specified in Guide YVL B.5.

704. LBB principle can be taken into account, case-by-case, as further technical justification during the acceptance review of deviations related to the integrity of piping.

7.3 Protection requirements

705. If the LBB principle is not demonstrated, high energy pipe design shall take into account the impact loads referred to in paras 415, 416 and 417, which, in the event of a complete break of the piping in question, will exert on susceptible components and structures important to safety. The piping shall then be provided with whip restraints in accordance with [10] unless an other acceptable solution, e.g. based on layout design in accordance with Guide YVL B.7, is presented to protect the components and structures in question.

706. Threats arising from different break assumptions shall be extensively analysed in selecting the type, location and effective orientation of the whip restraints. Shields shall also be constructed, where necessary, to protect against a jet impingement from a completely broken pipe.

707. If the LBB principle has been demonstrated for piping, design need not be based on impact loads caused by a postulated complete break. Where necessary, shielding shall be implemented to protect against the maximum jet load that could impinge from postulated through-wall crack.

708. Other pipe break implications to be considered in the design of the nuclear power plant's safety systems are addressed in Guides YVL B.1, YVL B.5 and YVL B.7.

7.4 Documentation to be submitted

709. A description of the BP principle implementation and the exclusion of mechanisms leading to complete break shall be included when submitting to STUK the design documentation of the plant project in question. LBB analyses are submitted as part of the strength analysis reports of the piping in question.

7.5 Break preclusion implementation

Technical implementation

710. The technical implementation of the BP principle shall strive for a comprehensive structural design concept which is the best possible in terms of integrity and reliability, and fulfils the highest piping standard requirements by using the following measures [9], for example:

a. analysis and monitoring programmes verifying the strength of dissimilar metal joints unless these joints are limited to small-diameter piping segments
b. reducing the number of welded joints to a minimum and positioning them far away from stress concentrations by using large forgings and integrated assemblies

c. tightening of wall thickness tolerances for welded joints in particular
d. increasing of the bend radii of curved pipe segments and providing elbows with straight ends to improve inspection and reduce stresses of the welded joints
e. selection of materials and manufacturing techniques to optimise toughness properties, resistance to corrosion phenomena and inspectability by ultrasonic methods
f. reducing thermal loads to a minimum as well as avoiding nozzles susceptible to fatigue and providing them with thermal shields
g. optimised volumetric inspections during manufacturing

711. For the protection requirements to be based on the LBB principle, technical implementation shall also include:

a. unidentified-leakage control [6], which is sufficiently sensitive, implements the diversity principle and is provided with on-line alarms
b. monitoring of static and dynamic behaviour as well as assumed thermal fatigue points by means of in-service measurements
c. check of the adequacy of toughness properties by crack growth resistance testing.

Organisational implementation

712. The licence applicant or licensee shall see to it that the personnel contributing to the design, construction and operation of a nuclear power plant based on the BP principle have adequate instructions and training for operations advancing the preclusion and early detection of the breaks of pressure equipment important to safety.

7.6 Mechanisms of complete break

713. The description on BP principle implementation shall provide detailed justification that, due to its technical and organisational implementation, no direct or indirect mechanisms causing a complete break can be identified for piping. The justification may build on reference [11] and shall encompass the following alternatives:

a. manufacturing faults and ageing mechanisms as a result of which a defect that reduces load-bearing capacity would develop on pipe wall covering an area so extensive that with its further growth the capability of the piping to bear service loads in compliance with chapter 4.3 could be quickly lost.
b. manufacturing faults and ageing mechanisms resulting in the degradation of the mechanical properties of a piping material to such an extent that service loads in compliance with chapter 4.3 or a significant service temperature drop would cause the threat of a fast break.
c. an impact load induced by an internal or external event, malfunction or human error, which exceeds the load-bearing capacity of a pipe wall degraded in a conceivable manner.

714. A probabilistic evaluation of pipe failure frequency may be presented as further technical justification.

7.7 Analysis methods

Through-wall crack criticality and detection as leakage

715. LBB analysis shall demonstrate by analysis of fracture and fluid mechanics the safety margins required in reference [11] as regards leak detection and critical defect size.

716. LBB analysis shall be conducted assuming that through-wall cracks have developed in the piping. They must be in locations where the combination of stresses and the material properties given as input data is the least favourable.

717. Determination of critical defect size shall be based on crack growth resistance values representing the materials and welding procedures used in manufacturing, which are determined by testing in accordance with an applicable standard [8] at a temperature corresponding to normal operation. The crack growth resistance values must be valid for the stable crack extension based on which critical defect size and the safety margin for it are determined. In using extrapolation, the method applied and its qualification data shall be presented.

718. If crack growth resistance values are determined or the extrapolation method is experimentally validated for the plant project in question, STUK shall be given the possibility to oversee the tests. Material samples of the tests shall be available for the licensee's records.

719. Critical defect size shall be determined for service conditions that according to the stress analysis conducted cause maximum local stress, taking into account specified fast pressure transients and the design basis earthquake. The possibility of a temperature reduction that decreases toughness values shall be analysed for ferritic materials. Furthermore, to be taken into account are the effects of possible ageing during service life. The toughness value 100 J, demonstrated for a welded joint at room temperature, could be the primary circuit design premise.

720. For the determination of critical defect size and opening area for leakage, applicable methods shall be used, e.g. the elastoplastic methods presented in reference [12]. The calculation method of the fracture mechanical parameter, which the method is based on, shall be presented, as well as the stress-strain curve applied and the mathematical fit to it. When weld metal has significantly higher strength values than base material, these shall be used to calculate the opening area. Critical defect size shall then be based on the strength values of the base material.

721. The leak detection sensitivity assumed in LBB analysis shall be evaluated according to a system which belongs to the leak detection implementation and produces an alarm for an unidentified leakage no later than within an hour. The system's capability of yielding the values used in analyses shall have been qualified by testing. If the values used remain below 3.8 l/min [6], qualification shall be conducted by tests which represent as-built plant conditions and which STUK has been given an opportunity to oversee.

722. The method used for calculation of the leak rate shall have been qualified by applicable test results. The thermohydraulic modelling and the surface roughness values used for frictional leakage flow analysis shall be presented.

Crack propagation initiated from inner surface

723. LBB analysis shall demonstrate by fracture mechanics analyses based on fatigue crack growth that a crack postulated on the inner surface will not significantly grow under the fatigue loads specified for the entire service life of the piping.

724. Cracks to be examined by fatigue crack growth analysis shall be postulated in piping sections most susceptible to fatigue. Factors affecting accessibility for inspection may be taken into account in making selections.

725. The initial sizes of the fatigue cracks analysed shall be determined to be at least the size of the detection objective established for the non-destructive in-service inspections of the piping. Applicable experiences of defects undetected in inspections made during manufacturing can be taken into account.

8 Strength analysis quality management

8.1 Strength analysis producing organisation

801. The organisation conducting strength analyses shall have a quality management system [13] documented and implemented for this purpose, which determines the following, among other things:

a. personnel responsible for conducting, checking and approving the strength analysis, as well as the qualifications of this personnel
b. computer software used and their versions
c. procedures and responsibilities in purchasing, updating, developing and qualifying software
d. processes relating to input data, analysis performance, documentation and review
e. quality requirements for analyses of different categories of importance and technical difficulty.

802. The strength analyses of Safety Class 1 and 2 pressure equipment shall be conducted by sufficiently tested analysis methods. Fundamentals and limitations of the methods selected shall be well known, and their qualification for the analysis in question shall be checked. When using computer methods, such as the finite element method, the personnel conducting the calculations shall be well experienced in using the program and its features needed to perform the particular analysis. Reference calculations shall be conducted to verify the correctness of the analysis, and the checks performed by the program shall be reviewed.

803. Quality management and safety management in the strength analyses performed by component supplier, licence applicant or licensee shall be based on a certified or otherwise independently reviewed management system, and the requirements set by them shall be taken into account in resource management.

8.2 Evaluation and oversight of strength analysis suppliers

804. The licence applicant or licensee shall see to it that external suppliers of pressure equipment strength analyses have been evaluated before purchasing an analysis.

805. The licence applicant or licensee shall evaluate the suppliers of strength analyses of Safety Class 1 and 2 pressure equipment by auditing their quality management systems.

806. The licence applicant or licensee shall, by their own evaluation and supervision, ensure that suppliers of Safety Class 1 and 2 pressure equipment who subcontract strength analyses have in place an auditing procedure for subcontractor evaluation, as required by the quality management system, and that the procedure is observed.

807. Evaluation of the suppliers of strength analyses for Safety Class 3 pressure equipment can be based on a certified or otherwise independently evaluated quality management system.

808. The licence applicant or licensee shall maintain a list of the suppliers of strength analyses and strength analysis software approved during the evaluations. The maintenance of the prerequisites for approvability shall be followed.

8.3 Procurement of strength analyses

809. The licence applicant or licensee's quality management system shall include a guideline procedure defining the quality requirements to be complied with in purchasing strength analyses from approved suppliers.

810. The licence applicant or licensee shall, in connection with their strength analysis purchases, provide the supplier with the necessary input data on the pressure equipment to be analysed, as well as their design bases and the related safety regulations.

811. The licence applicant or licensee shall ensure that component suppliers have in place a corresponding procedure covering their subcontracted strength analyses and that the supplier receives the necessary input data in connection with the subcontract in question.

812. The organisation that purchased the strength analysis shall verify that the delivery corresponds to the safety and quality requirements of the purchase, as well as its input data, and that the results fulfil the criteria set for them. The correctness of the results shall be technically assessed and potential deviations or unexpected results shall be analysed.

8.4 Strength analysis review

813. The licence applicant or licensee shall independently review the strength analyses to be submitted for regulatory consideration. A summary of the review and the justification for acceptability shall accompany the submission. An independently conducted reference analysis to verify correctness and acceptability shall be acquired, where necessary.

814. The strength analysis reviews conducted by the licence applicant or the licensee shall be based on a certified or otherwise independently reviewed management system. Guidance shall be provided for matters to be addressed during the analysis reviews, and the requirements set by them shall be taken into account in resource management.

8.5 Quality management of loading analyses and load monitoring

815. The requirements in chapters 8.1–8.4 apply to loading analyses, as applicable.

816. In the monitoring of load and ageing effects, the quality management procedures determined for ageing management in nuclear power plants shall be followed. The personnel evaluating the results shall have good expert knowledge and experience in the fatigue analyses accepted for the nuclear power plant in question as well as its behaviour during different operational and transient conditions.

8.6 Quality management of strength analyses in plant projects

817. The prerequisites for operation of the most important suppliers of strength analyses for Safety Class 1 to 3 pressure equipment shall be systematically analysed at the start of a plant delivery or an extensive modification. The licence applicant or licensee shall ensure that the suppliers – the organisations performing strength analyses for the plant supplier in particular – are familiar with the project as well as the safety and quality policy, quality plan and the safety regulations to be observed.

818. After the plant delivery or an extensive modification has progressed to the component design phase, the licence applicant or licensee shall supervise the progress of the deliveries of strength analyses of Safety Class 1 to 3 pressure equipment, compliance with the quality management systems and safety regulations, as well as the acceptability of preliminary results and conformance to the requirement level of the present Guide through regular communication and follow-up meetings. Significant deviations and non-conformances shall be reported to STUK.

819. In supervising strength analyses pertaining to a plant delivery or an extensive modification, the licence applicant or licensee shall pay attention in particular to guidance provided for essential analysis processes, qualification of demanding methods as well as to ensuring that

a. design aims at technical solutions reducing pressure equipment stress factors, e.g. thermal fatigue or radiation embrittlement of the reactor pressure vessel, and that requirements pertaining to the BP principle are implemented
b. participating organisations comply with consistent design and reporting practices and standards
c. data transfers between different organisations and between computer programs and systems function reliably
d. an adequate administrative procedure is in place to ensure that the design data is correct, up-to-date and frozen
e. an efficient procedure is in place to cover the interfaces between components and responsible organisations, concerning e.g. loading specification and adjustment at pressure equipment connecting nozzles
f. efficient procedures are in place for manufacturing and installation non-conformances as well as for the management of changed data on mass, supports, etc.

820. The licence applicant or licensee shall ensure that a systematic procedure is in place to manage the delivery schedule of all the strength analyses required in a plant delivery or an extensive modification. A component-specific itemised plan for the delivery of strength analyses and regulatory approvals shall be in place when applying for a construction licence; when applying for an operating licence, a summary of the implementation of that plan shall be submitted to STUK.

8.7 Strength analysis register

821. The licensee shall keep a record during the nuclear power plant's service life of the strength and loading analyses approved by the regulatory authority. By means of the record it shall be possible to reliably determine pressure equipment related strength data for in-service inspection programmes, load and radiation embrittlement monitoring programmes as well as for potential modifications.

9 Regulatory oversight

9.1 Oversight by the Radiation and Nuclear Safety Authority

901. In connection with the review of licence applications and construction plans, STUK reviews loading analyses that comply with the present Guide as well as strength analyses of pressure equipment and pumps belonging to the scopes of application covered in paras 501, 503, 601, 616 and 703.

902. STUK oversees experimental analyses and testing relating to the demonstration of correctness of strength and loading analyses of pressure equipment, covered by its inspection responsibility, and also oversees potential experimental stress analyses.

903. STUK oversees the monitoring of fatigue loads and vibrations by inspection visits during pre-operational testing and operation, and also by reviewing annual reports.

904. STUK approves plans for the standards to be followed in strength analyses conducted in accordance with the present Guide and, where necessary, specifies by its decisions the method of application of the standards.

905. STUK assesses the quality of loading and strength analyses of pressure equipment covered by its oversight responsibility by conducting its own inspections and by having reference analyses made.

906. STUK reviews the licence applicant's or licensee's management systems relating to loading and strength analyses as well as strength verification.

907. In connection with plant deliveries and extensive modifications, STUK oversees the quality management system audits of the suppliers of loading and strength analyses of pressure equipment, covered by its oversight responsibility, and of subcontractor suppliers contributing to the above analyses, and also oversees the progress monitoring and project management of the most important strength analyses.

9.2 Oversight by an authorised inspection body

908. An authorised inspection body reviews the stress analyses of nuclear pressure equipment and pumps referred to in para. 502 in connection with the review of their construction plans.

909. An authorised inspection body assesses the quality of the stress analyses of nuclear pressure equipment and pumps referred to in para. 502 and assesses the need of stress analysis for items referred to in para. 503 by conducting its own inspections.

Definitions

Authorised inspection body

Authorised inspection body shall refer to an independent inspection organisation approved by the Radiation and Nuclear Safety Authority under Section 60 a of the Nuclear Energy Act to carry out inspections of the pressure equipment, steel and concrete structures and mechanical components of nuclear facilities in the capacity of an agency performing public administrative duties.

Dynamic analysis

Dynamic analysis shall refer to determining the time-dependent behaviour (vibrations) and stresses of a component or structure under an impact-type, seismic or cyclic loading. In particular, the analysis focuses on the resonance risk induced by the excitation of natural oscillations and the strengthening of stress in proportion to the stresses caused by an equivalent static load.

Finite element method

Finite element method (FEM) shall refer to a mathematical modelling method of physical phenomena in structures and other continua, in which the domain to be analysed and the distribution of the variables governing the phenomena are described numerically, by using a mesh of finite elements.

Brittle fracture

Brittle fracture shall refer to a rapid crack growth in a metal structure under tensile stress and with no substantial permanent deformation which, using the energy released, may proceed inside the structure and develop into a complete break.

Stress analysis

Stress analysis shall refer to a strength analysis based on the modelling of the actual structure and loads of pressure equipment, which is used to eliminate the risk of failure caused by the loss of the load-bearing capacity, excessive deformation and fatigue, when the acceptance limits that have been set for the calculated stresses governing these mechanisms, as stipulated in the applicable standard, are met.

Conservative analysis method

Conservative analysis method shall refer to a manner of preparing a safety analysis that considers the uncertainties related to the calculation models and initial assumptions so that, with a high level of certainty, the consequences of the event analysed would be milder than the analysis results.

Critical defect size

Critical defect size shall refer to a size of an assumed crack-like defect in pressure equipment which, if exceeded, would, according to the applicable fracture mechanical criterion, cause a risk of fast fracture in a load scenario where the combination of the stress states and material properties is the least favourable.

Loading analysis

Loading analysis shall refer to the computational analysis, covering the entire life cycle, of the mechanical and thermal loads (service loads) to which a component is subjected in the operational conditions and accidents used as the facility's design bases over the course of its entire life cycle, when the procedures, specifications and analyses concerning operation, required functions and sequences of events are taken into account.

Load grouping

Load grouping shall refer to applying the acceptance limits presented in the standard applied to stress analysis, graded by the severity of the load and the safety factors required, to the specified service loads so that the frequency of occurrence of the load, the post-load opportunities for inspection and repair, and the integrity and operability requirements set for the pressure equipment in the scenario in question are taken into consideration.

Quality management

Quality management shall refer to all of the coordinated and planned activities performed to ensure that the organisation, component, plant or activity meets the requirements and quality criteria set for it (SFS-EN ISO 9000).

Strength analysis report

Strength analysis report shall refer to a set of documents consisting of the pressure equipment strength analysis documentation submitted for regulatory review by authorities in connection with the construction plan.

Break preclusion

Break preclusion (BP) shall refer to the principle of using advanced technical and organisational procedures in piping construction, operations and maintenance in such a way that the LBB principle will be implemented and this may be credited for the nuclear power plant design and its provision against the consequences of a complete, instantaneous pipe break.

Whip restraint

Whip restraint shall refer to a steel structure constructed to provide protection against breaks in high-energy pipes by preventing impacts caused by the break on components and structures important to safety, and by limiting the hydrodynamic load caused on the internal structures of the same system by a sudden leakage flow.

Pressure equipment

Pressure equipment shall refer to a vessel, piping and other technical assembly, in which overpressure exists, or in which it may develop, as well as the technical assemblies designed to protect pressure equipment, including elements attached to pressure retaining parts such as flanges, nozzles, couplings, supports, lifting lugs etc.

Pressure equipment construction plan

The documentation to be submitted for regulatory consideration before the commencement of the manufacture of pressure equipment. It documents the structure of the pressure equipment, including calculations, manufacturing plans, inspection and testing plans, the suitability of other related components, and a summary of licensee's justification for acceptability.

Best estimate method

Best estimate method shall refer to a method of preparing a safety analysis where the physical modelling of any phenomenon studied is as realistic as possible, and the initial assumptions for the analysis are realistically selected.

Primary stress

Primary stress shall refer to stress generated in a pressure equipment structure that keeps the structure in balance in relation to the external mechanical loads exerted on it.

Secondary stress

Secondary stress shall refer to stress generated in a pressure equipment structure as a result of the limiting of deformation, or to stress controlling the compatibility without incremental distortion (shakedown) of components whose temperatures and rigidity differ.

Design load

Design loads, to be determined in accordance with the applicable standard, comprise design pressure, design temperature and other mechanical design loads which in combination with design pressure produce the highest primary stresses in normal operating conditions.

Radiation embrittlement

Radiation embrittlement shall refer to the embrittlement of the reactor pressure vessel's steel in the core region due to the microstructural damage and transition temperature increase (radiation shift) caused by neutron radiation.

Probabilistic risk assessment (PRA)

Probabilistic risk assessment (PRA) shall refer to a quantitative assessment of hazards, probabilities of event sequences and adverse effects influencing the safety of a nuclear power plant. (Government Decree 717/2013)

Transition temperature

Transition temperature shall refer to a temperature, which characterises an intrinsic substantial ductile-to-brittle transition of ferritic steels with decreasing temperature, and is determined through destructive testing in compliance with the applicable standard.

Complete, instantaneous break

Complete, instantaneous break shall refer to the breaking of a pipe or another type of break that results in a large leak due to structural damage, degradation of material properties, or overload.

Leak before break (LBB)

The leak before break (LBB) shall refer to a principle where piping does not have identified failure mechanisms creating the possibility of complete break, and faults that are not detected during inspections may at most create a small, local leak the detection of which allows the plant to be brought into a state where there is no risk of complete break.

Fatigue

Fatigue shall refer to damage propagating in local structural discontinuities as a result of cyclic mechanical or thermal loads, which is evaluated in a stress analysis by comparing the peak stresses calculated for these locations to the fatigue curve of the applicable standard.

References

1. ASME Boiler and Pressure Vessel Code, Section II, Materials, Part D - Properties, the American Society of Mechanical Engineers, New York, 2010.
2. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Facility Components, Division 1, the American Society of Mechanical Engineers, New York, 2010.
3. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, the American Society of Mechanical Engineers, New York, 2010.
4. Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing, Regulatory Guide 1.20, U.S. NRC, Rev.3, March 2007.
5. Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors, Regulatory Guide 1.207, U.S. NRC, Rev.3, March 2007.
6. Guidance on monitoring and responding to reactor coolant system leakage, Regulatory Guide 1.45, U.S. NRC, Rev.1, May 2008.
7. Standard Test Method for Determination of Reference Temperature, T_o, for Ferritic Steels in the Transition Range, ASTM E 1921-11.
8. Standard Test Method for Measurement of Fracture Toughness, ASTM E 1820–11.
9. RSK-Leitlinien für Druckwasserreaktoren mit Anhänge zu Kapitel 4.2, 1. Auflistung der Systeme und Komponenten und 2. Rahmenspezifikation Basissicherheit, GRS, Ausg. 3, 14. Okt. 1981.
10. Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Standard Review Plan 3.6.2, U.S. NRC, Rev.2, March 2007.
11. Leak-Before-Break Evaluation Procedures, Standard Review Plan 3.6.3, U.S. NRC, Rev.1, March 2007.
12. Leak-Before-Break Evaluation Procedures for Piping Components, K. Ikonen et al., STUK-YTO-TR 83, Helsinki, 1995.
13. ASME NQA-1-2008, Quality Assurance Requirements for Nuclear Facility Applications, the American Society of Mechanical Engineers, New York, 2008.
14. WENRA Reactor Safety Reference Levels PS/7.1.2010.
15. IAEA Specific Safety Requirements No. SSR-2/1, Safety of Nuclear Power Plants: Design.
16. Nuclear Energy Act (990/1987).
17. Government Decree on the Safety of Nuclear Power Plants (717/2013).