Radiation and Nuclear Safety Authority 121/0002/2016
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Guide YVL E.4, Strength analyses of nuclear power plant pressure equipment

1 Scope of application

Guide YVL E.4 “Strength analyses of nuclear power plant pressure equipment” sets forth requirements concerning the loads and strength analyses of a nuclear power plant’s primary circuit and other nuclear pressure equipment important for 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.

2 Justifications of the requirements

2.1 Stress analysis

Chapter 5 of Guide YVL E.4 discusses the stress analysis. As regards the stress analysis, the requirements in Guide YVL E.4 are related to the design by analysis method as described in Article NB 3200 of the American ASME III standard. This method focuses primarily on failure mechanisms where the structure of the pressure equipment loses its load-bearing capacity or deforms excessively under a strong single-stage load or repeated loads causing permanent deformations, or shows structural fatigue in stress concentrations. Of the states of stress calculated for the structure, the stress types (primary, secondary and peak stresses) governing the development of the aforementioned failure mechanisms shall be identified and the intensities describing their combined effects checked for compliance with the limits allowed for each stress type in the applicable standard. The severity classification of the load condition (the loads are divided in groups A, B, C and D) and the maximum amount of damage selected as the safety margin basis affect the range of acceptance limits applied in the checks as well as the stress classes in relation to which the compliance of each load condition with respect the limits shall be checked.

To calculate the states of stress required for the design by analysis method, a detailed computational model is first constructed of the analysed structure and its physical properties using, nowadays almost invariably, a computer software based on the finite element method (FEM). The loads are introduced in the model as input data unless the same model is also used in the analysis of the load or the connections between the load and structural response (time-dependent thermal loads, interaction between fluid and structure). Stress classification requires computational reprocessing of the stress states. Loads that cause fatigue are examined as a function of time to identify the number and amplitude of the governing stress periods.

Guide YVL E.4 takes into account the pressure equipment type-specific E series Guides YVL E.3 “Pressure vessels and piping of a nuclear facility” and E.8 “Valves of a nuclear facility” that set requirements also for the design. These requirements are to cover by formulae (design by rule) the dimensioning procedures suitable for the different types of equipment, and the so-called simplified stress analyses where the equipment is expected to comply with the basic geometry idealised for the type of equipment in question. This mainly includes pressure equipment in Safety Class 3 subject to regulatory inspections by an approved inspection organisation according to Section 60 a of the Nuclear Energy Act (990/1987). The above is taken into account in the regulatory oversight definition provided in Chapter 9 of Guide YVL E.4. Standards applicable to the dimensioning and the simplified analyses of said pressure equipment types are specified in the E series component guides. In case of exceptional situations outside the scope of the standards in question, the design shall be conducted using a detailed stress analysis pursuant to Guide YVL E.4. The experimental stress analysis mentioned in Chapter 4 of the Guide may also be considered in this case. These options may also have to be adopted in case the equipment’s risk significance proves to be higher than expected, or due to some other technical reason, such as insufficient inspectability of the pressure equipment.

As regards the scope of application, the aforementioned also applies to Guide YVL E.9 “Pumps of a nuclear facility”. Pump housings subject to high-pressure piping system loads at nuclear power plants are designed similar to the pressure equipment, making the inclusion of the respective detailed stress analyses in Guide YVL E.4 appropriate. This is even though in EU legislation pumps generally fall within the scope of the Machinery Directive (not the Pressure Equipment Directive, or PED).

ASME III remains the primary reference standard for stress analyses pursuant to Guide YVL E.4. The requirements set in the standard are based on the widest experience from nuclear power plants in different countries, and it provides good coverage of safety principles applicable to the different operational conditions. Relevant sections are now those intended for ASME III Code Class 1 components and those applicable to the aforementioned Safety Class 2 structures (NB, NF, NG). The possibility to apply for approval of a similar standard approved in another country has been useful in the Olkiluoto 3 project, and the possibility should, therefore, be preserved in future as well. For substantiated reasons, this can also be applied to a smaller-scale target, even a single equipment acquisition. As regards the main secondary piping of the pressurised water reactor, the above ASME reference means in practical terms that the highest quality requirements of the design standard must be met even though the pressure equipment in question is in Safety Class 2. In the case of the Olkiluoto 3 plant unit, this procedure has been considered a good technical and qualitative basis to apply the LBB principle to these piping systems. As regards the definition of the scope of application of the stress analysis, Guide YVL E.4 sets out to establish that quality planning shall also be considered when defining the standard requirements to be met.

Attention has been paid, for example, to the load-bearing piping penetration fittings that simultaneously double as anchors and the dimensioning of standard-compliant fittings with design loadings as STUK is of the opinion that the applied standards for dimensions as referred to in Article ASME III NB 3132 do not specify the wall thicknesses adequately.

Fatigue analysis of the cladding is only required when it has been taken into account in the modelling of the structural configuration, or, in practice, when the thickness of the cladding is at least 10% of the overall wall thickness. A similar procedure shall be observed in thermal load modelling. Factoring in the environmental effect in the fatigue analysis has proven to be challenging. The extensive observation material from fatigue tests conducted in different countries, however, serves as a technical justification. It is an established administrative practice in many countries to require environmental effect to be considered at new facilities and when extending the plants' operating life or assessing any fatigue damage occurred. Environmentally affected fatigue analysis is included as a requirement in Guide YVL E.4, with the primary procedure referenced being Regulatory Guide (RG) 1.207 published by the United States regulatory authority, NRC. According to the requirement laid down to ensure operability, Guide YVL E.4 makes a distinction in accordance with the application practice adopted in the Olkiluoto 3 plant project between active operability (group B acceptance limits) and passive operability (group C acceptance limits). The requirement to maintain active operability is a valid starting point because, as demonstrated by the Fukushima nuclear power plant accident, post-accident management may take a long time. However, these requirements need not be met for loads limited to one subsystem of the safety system if this subsystem is not required before bringing the plant into a safe state, and provided that it is properly ensured that the components in question are in good working order in case of possible recommissioning of the components. The load caused by a large commercial airliner crash shall be considered as design extension condition applying best estimate methods according to Section 2 of the Nuclear Safety Authority Regulation STUK Y/1/2018 (requirement level pursuant to group C acceptance limits if operability is required, otherwise group D acceptance limits).

The requirements concerning the format of presentation of the strength analysis report have proved useful in controlling the quality of the stress analyses. Chapter 3.3 of Guide YVL E.4 sets out to ensure an adequate requirement level. A typical example of this is the presentation of a computational stress distribution in such detail that the selection of cross-sections for secondary stress and fatigue analysis can be verified as correct. For quality assurance purposes, the specific information to be provided for the stress analysis includes acceptance markings by suppliers and contractors. This requirement aims at obtaining reliable information on quality assurance and approvals even in cases where the report in question is originally intended to be submitted to another project with restricted disclosure of information to a third party.

2.2 Brittle fracture analysis

Chapter 6 of Guide YVL E.4 discusses the brittle fracture analysis. Brittle fracture shall refer to a rapid crack growth in a metal structure under tensile stress and with no substantial permanent deformation. Using the elastic energy released along the growth line with minimum crystal structure resistance to the fracture mechanism, such a fracture may proceed inside a structure and develop into a complete break. Ferritic steels are vulnerable to brittle fracture due to the inherent decrease in their deformation capacity and fracture toughness as the temperature drops to the so-called transition range. In the reactor pressure vessel (RPV) of a nuclear power plant, this possibility is increased during operation by an increase in the transition temperature shift, due to a change in the crystal structure caused by rapid neutrons hitting the RPV wall. To constitute an actual threat, the crack on the radiation-embrittled section would already have to be of significant size and the wall should either rapidly cool down in the transition range during an emergency cooling situation or similar cooling should have already taken place during a cold shutdown, with the RPV allowed to be pressurised for some reason. In the case of austenitic cast steel, on the other hand, experience has shown that so-called thermal embrittlement may occur over time at the primary circuit operation temperatures. In practice, also other factors, such as the large size of the pressure equipment or high strain rate due to load, reduce the toughness.

In the early years of operation, extensive improvement measures were required at the Loviisa nuclear power plant to control the faster than expected radiation embrittlement. A successful heat treatment applied in 1996 nearly recovered the RPV of the number 1 unit, more vulnerable to this mechanism due to the greater material impurity levels, back to its original state. The prerequisites for continued use of the reactor pressure vessels have been determined at multiannual intervals by safety assessments based essentially on brittle fracture analysis performed by methods of fracture mechanics and the associated verification programme (so-called surveillance programme). The required fracture toughness values have been determined by the so-called master curve method developed by VTT and widely known as the ASTM E-1921 standard. The differences between scopes of application and methods and the fact that stress states determined by stress analysis are the input data for the brittle fracture analysis have contributed in the brittle fracture analysis being performed as a separate strength analysis type independent of the stress analysis.

As regards the Loviisa plant case, the most important thing to be followed is the speed of the re-embrittlement in the RPV of the number 1 unit and the question of whether heat treatment will be required in the number 2 unit. The management of radiation embrittlement is organised as part of the plant-wide service life management, and safety assessments thereof will be provided in conjunction with the plant-wide periodic safety reviews. Olkiluoto 1 and 2 units have had no problems with radiation embrittlement even from the outset due to the low RPV design pressure and the main circulating pump system extending inside and increasing the distance between the wall and the reactor core, thus allowing the water space in between the two to decrease the fast neutron fluence received by the key welded joints. Based on the experience of the Olkiluoto 3 construction project, it is also evident that radiation embrittlement is better managed in the design of new plants. Improved manufacturing processes have reduced the level of impurities in the pressure vessel steels, and the optimisation of the water gap in the PWR plants is no longer a logistical issue but rather a question associated with the manufacturing technology requirements related to large forgings. At Olkiluoto 3, the reactor pressure vessel wall is also protected by a heavy reflector surrounding the fuel assemblies, and there are no welded joints between the shell’s annular forgings at the location of the reactor core. Therefore, the most significant aspect to be monitored is likely to be the possibility of thermal embrittlement in the dissimilar metal joint between the RPV and main coolant piping. At the operating plants, provision against thermal embrittlement is to be increased due to the projects designed to extend the service life of the plants from that originally planned.

As regards the RPV brittle fracture analysis, Guide YVL E.4 requires that a probabilistic analysis shall be performed when the risk of brittle fracture cannot be determined to be negligible based on a deterministic transient and accident condition analysis. The requirement level for the deterministic assessment is still in line with guide RG 1.154, even though the guide was withdrawn in the United States by NRC in 2011 as technically outdated and unnecessarily conservative. After finding the brittle fracture risk at the operating US plants to be lower than previously thought, NRC now considers the minimum temperatures defined in terms of pressure tests, normal operation and anticipated operational occurrences, which include a margin to the reference temperature determined pursuant to the ASME III and ASTM E 208 standards, sufficient. However, the increase in the wall thicknesses at the new plants is an unfavourable factor for the brittle fracture, which is further emphasised in Finnish conditions by the lower temperature of the emergency cooling water in our plants. STUK still considers the approach of RG 1.154 advantageous as regards the systematic fracture mechanics-based analysis of plant event sequences, including accident situations, which are currently performed at a high level of competence in Finland. STUK will also continue to insist that the toughness values shall be verified using the master curve.

Fast fracture may also occur at temperatures above the brittle-ductile transition temperature in the so-called upper shelf area in case of a respective significant decrease in the fracture toughness values due to ageing or a manufacturing method that proves to be inadequate. At the US plants, the low toughness values of certain submerged arc welds have caused concerns of this type. In the case of cracked ductile material, a fast fracture may occur through a ductile tearing mechanism in, for example, operational occurrence or accident situations where a thick-walled structural part is subjected to rapid cooling under high pressure. In the Olkiluoto 3 project, this was taken as a starting point for the required analysis of fast fracture in accordance with the French RCC-M standard in temperatures exceeding the temperature criteria set for the brittle fracture analysis. As regards the transition area between the brittle and ductile mechanisms, an analysis was required in terms of both mechanisms. The possibility of a ductile fracture is discussed in the “Other fast fracture considerations” subsection in Chapter 6 of Guide YVL E.4. The procedures and criteria to be used shall be approved by STUK because, at this stage, development work is needed to present specified fracture mechanical requirements for ductile fracture analysis. As regards piping, the requirements set out in Chapter 7 of the Guide on LBB analysis of cracks that have already penetrated the wall shall be considered. These requirements apply to the design phase. Monitoring of the toughness values in the ductile area during operation is part of the ageing management in accordance with Guide YVL A.8 “Ageing management of a nuclear facility” which shall in future consider the various mechanisms that decrease the toughness values in a more comprehensive manner.

The oversight of the brittle fracture analyses and the leak before break analyses discussed in Chapter 7 also includes experimental studies associated with these analyses. Tests with special safety significance include testing of the fracture-mechanical properties of materials and welded joints performed by test programmes that supplement normal manufacturing quality control and are often extensive and long-lasting.

The update proposes that paras. 611 and 612 should include the following further specification as regards postulated defects: *as well as [cracks] that open to the inner and outer surface*. This does not change the requirement level because, in addition to internal defects, it has been a practice in applying the current Guide also to assume surface cracks. However, the properties of postulated cracks are now defined in the referenced “ASME III appendix G” material. In order to clarify the matter, the important crack-determining property has been included in the Guide itself. Other properties determining postulated cracks shall still be checked from the aforementioned reference. In order to avoid confusion, it is deemed necessary to include this important crack-determining property in the Guide itself.

2.3 Leak before break analysis

Chapter 7 of Guide YVL E.4 discusses the leak before break analysis. As a physical concept, LBB refers to such damaged piping behaviour where a complete break is not possible because the failure would, in any case, extend through the wall in a very short distance and would be detected in time due to the leak thereof. Developed in Germany, the break preclusion (BP) safety principle is aimed at new plant designs and includes improved technical and organisational procedures that are expected to eliminate the possibility of a break, thereby allowing LBB behaviour and its demonstration by calculation. In Germany, the effects of BP on the plant's design basis as regards protection against pipe breaks are described in detail. In the United States, NRC guide SRP 3.6.3 likewise presents LBB as a comprehensive safety principle but mainly for the purpose of changing the design basis of the operating plants to eliminate the need for protection against impacts due to a break in the main coolant pipe. In this case, the adequacy of the quality and condition of components built according to the original standard requirements shall first be assessed to ensure that there are no mechanisms creating any risk of break and that the conditions allow LBB to be applied. LBB behaviour is demonstrated based on more detailed analysis of the material fracture properties and leak detection and conservative safety factors set for the critical size of a leaking crack, while the German calculation method puts more emphasis on the fatigue crack growth through the wall prior the leak. In the United States, the effects on the plant design basis have been associated with pipe impacts and omitting of whip restraints. NRC has developed mitigations for the dimensioning of the safety systems in connection with, for example, the "Large Break LOCA Redefinition" project, but they have not received wide international recognition.

In Finland, LBB was discussed already in the 1980s in the Finnish Research Programme on Nuclear Power Plant Safety, including full-scale tests using decommissioned conventional pressure vessels. Procedures in line with the Finnish regulatory control practices were already defined in the 2002 Guide YVL 3.5 "Ensuring the firmness of pressure vessels of a NPP" by describing LBB as a safety matter subject to separate approval by STUK and linked to the plant's construction licence processing and general protection against pipe breaks. At the Olkiluoto 3 plant, LBB principle has been implemented in conjunction with special defence in depth strategy that also includes whip restraints. The operating plants have used LBB as an additional justification providing supplementary defence in depth for normal defect assessments in situations with unusual uncertainty factors associated with the defect assessment.

Guide YVL E.4 focuses on technical procedures and is based on established international practices. The requirements in Chapter 7 of the Guide are intended to apply to plant projects pursuant to both the German BP and US LBB principle. Safety improvements required by BP and relevant to a new plant built in accordance with the German practice are described in Chapter 7.5. Requirements described in Chapter 7.6 apply to the elimination of breaking mechanisms also when applying the US LBB concept. The mechanisms are presented on a general level (mechanisms that cause structural damage, deteriorate material properties, exceed load-bearing capacity) because new mechanisms may emerge and the perception of the risk significance of the known mechanisms may change. The increased stress corrosion cracking under PWR plant primary circuit conditions (PWSCC) even at LBB-approved plants provides for a cautionary example of the above. Provided in Chapter 7.7, the requirement level for the analysis methods includes a combination of the best features of both the aforementioned practices as applied to the Olkiluoto 3 plant unit. The chapter on penetrative fractures emphasises the US procedure, and the chapter on surface cracks emphasises the German procedure. To facilitate a satisfactory result, the former allows for certain computational assumptions (determination of critical fracture size, improved leakage control) the correctness of which must be verified experimentally.

The effects of BP and LBB applications to the plant's design basis in Guide YVL E.4 could be considered for pipe impacts only. The objective is to have the primary circuit of a new nuclear power plant constructed by implementing structural improvements in accordance with the BP safety principle, para. 710, as regards the good practices even in the case that LBB behaviour is not demonstrated by analyses pursuant to Chapter 7.7. In this case, impacts caused by pipe breaks provide the design basis in accordance with para. 705. Their dynamic effects must be analysed, with the installation of whip restraints as the primary means of protection. LBB-related material studies and the special leakage control and in-service inspection requirements linked to the LBB analyses are, then, unnecessary. On the other hand, if LBB behaviour is demonstrated in accordance with para. 707, pipe impacts are no longer the design basis. Consideration of primary circuit pressure transients due to impacts as design extensions, and other physical effects of pipe breaks on the design of the primary circuit, are discussed in Guide YVL B.5 “Reactor coolant circuit of a nuclear power plant”.

2.4 Quality management

Chapter 8 of Guide E.4 discusses the quality management of the strength analyses. Regulatory oversight of strength analyses has traditionally been based on reviewing the reports submitted of these analyses. With the development of numerical methods of analysis and information technology, the review process is challenged by the continuously expanding scope and increasingly routine-like production of the strength and load analyses.

In the Olkiluoto 3 project, the importance of strength-related quality management has been highlighted by the long supply chains typical for multinational projects, interfaces between the different design phases and the need to coordinate various design practices and technical capabilities within a busy schedule. Amidst all this, the operating environment of the licensee has diversified to cover assessments and oversight of the providers of the strength analyses, procurement and inspection of analyses and, to some extent, also reference calculations and analyses performed in its own strength calculation unit. By supplementing the documentation reviews with oversight of the strength analysis operations in the form of project work, STUK, for its part, has been able to ensure beforehand the quality of the design and compliance with the safety regulations even in the case of strength analyses submitted belatedly from the original schedule. In future projects, the share of this activity is likely to increase and be performed at an earlier stage of the project if the completion status of the plant's design is more advanced in the construction licence phase.

The 2013 Guide YVL E.4 and its current update continue along the same line, taking into account the changes in the quality management-related practices and reference standards. Strength analyses project management may have implications on the progress of the plant project as a whole, as STUK will continue to consider it necessary to submit the analyses already in conjunction with the pressure equipment construction plan, i.e. prior to the commencement of manufacture. When submitting the strength analyses to STUK, the licensee shall include a written summary of the inspections it has carried out and, if necessary, commission reference calculations in support of these inspections while, however, preserving the same possibility for STUK. This procedure is implemented to utilise the licensee's overall view of the technical location requirements of the component and to reduce the risk of detailed regulatory inspections of the strength analysis and their updates forming a bottleneck for future plant projects implemented in a tighter schedule.

According to Guide YVL A.3 “Leadership and management for safety”, a general management system forms a natural basis for the activities of both the licensee and plant supplier, ensuring compliance with comparable quality principles in any independent production units carrying out design activities. As regards management systems, Guide YVL A.3 is referenced.

In para. 801 on the strength analysis quality management, Guide YVL A.3 is presented as a new quality management system reference to which the other Guides should rely on. The requirement in Guide YVL E.4 on ASME NQA-1-2008 certified quality system for the design organisation is in contrast to the general line of the YVL Guides which also accept alternative verification methods if this can be considered to achieve an equivalent level of safety.

3 International provisions concerning the scope of the Guide

The Western European Nuclear Regulators' Association (WENRA), a joint body established to harmonise nuclear safety oversight procedures of the national regulatory authorities in Western European countries, has published its recommendations in January 2008 in so-called reference levels PS/7.1.2010.

Providing specific instructions on strength analyses of pressure equipment and based on established industry standards, Guide YVL E.4 does not set out to quote WENRA requirements as they are. However, Guide YVL E.4 includes several requirements that further specify the more general requirements laid down by WENRA, ensuring compliance with the requirements from the strength engineering point of view.

A similar assessment has also been made of how Guide YVL E.4 complies with the specific safety requirements SSR-2/1 set out by the International Atomic Energy Agency (IAEA) for the design of nuclear power plants. As regards strength engineering, Guide YVL E.4 meets more than 20 IAEA safety criteria, including topics such as external hazards, operational state and load grouping, operability, pressure and temperature limits, radiation embrittlement, the internals of pressure equipment, penetrations, fatigue, quality and project management, interfaces and established practices.

The primary safety class 1 design standard specified in Guide YVL E.4, the “ASME Boiler and Pressure Vessel Code, Section III”, is a standard specially designed for nuclear power plant components. In the control of manufacturing, however, the practice according to ASME where the undertaking itself is responsible for, for example, the personnel qualifications and the oversight is performed by an authorized inspector is not approved as such. Instead, the basic requirement is STUK’s own oversight and inspections.

4 Impacts of the Tepco Fukushima Dai-ichi accident

In the update assessment of the YVL Guides due to the nuclear power plant accident at the Fukushima Dai-ichi nuclear power plant in Japan on 11 March 2011, no need to modifications in Guide YVL E.4 was found. The implementation of the requirement level pursuant to Guide YVL E.4 will have a bearing on the impact of external hazards on the mechanical structures and equipment of the nuclear power plant as well as any safety functions dependant on their integrity. The impact of the previous major earthquake in Japan on 16 July 2007 on the mechanical structures of the Kashiwazaki-Kariwa nuclear power plant proved to be minor despite the significant overshoot of the design basis which, according to later analyses, was due to conservative assumptions in the structural design.

In Guide YVL E.4, external hazards are considered as loads analysed in the strength analyses included in the scope of the Guide. The design basis for such loads is an earthquake and design extension assumptions are vibration and pressure waves caused by a large commercial airliner crash. These are transmitted to pressure equipment through vibration and forces in their supports and nozzles to be determined by dynamic analysis of buildings and assemblies. The dynamic analysis shall also determine accelerations and loads on the mountings experienced by any equipment significant for safety functions and connected to vibrating structures or pressure equipment, and the results are then compared to the limit values defined in the seismic qualification or type tests of the equipment to verify their acceptability.

The adequacy of the strength of the pressure equipment and its supports under the above load conditions shall be verified in the stress analysis pursuant to Guide YVL E.4. Compliance with the acceptance limits for the supporting structures of the primary circuit shall be verified to ensure that the failure sequence cannot extend in case the main component supports or their anchoring to the buildings should fail in an earthquake or airplane crash situation. The reliability of the safety systems is ensured by the requirements set for their operability. Plastic deformations shall be limited in such a way that the required cross-sectional flow areas are not substantially reduced (passive operability) or the movements of the functional components’ (pumps, valves) internal parts necessary for active safety functions are not compromised. With the group B acceptance limits to be set, earthquakes are equated with the loads used as the basis for the service life calculations of the components, ensuring reliable structural long-term availability of the safety function in the event of, for example, decay heat removal. In the case of design extension accidents, the above is to be achieved using best estimate methods. Severe reactor accidents are discussed in other YVL Guides. Stress analyses of the key penetrations as regards the containment leak tightness are discussed in Guide YVL E.4. Other penetrations are covered in Guide E.6 “Buildings and structures of a nuclear facility”.

A further assurance on the integrity of the primary circuit in the specified load conditions is provided, even in the event of a significant crack or a crack extending through the wall on the main components, by the fast fracture and LBB principle analysis performed in accordance with the requirements set out in Guide YVL E.4. The analyses are performed using design earthquake-level loading assumptions, thus providing some margin in case of crack propagation into a complete break in that situation. The break preclusion principle, which is linked to the LBB principle, includes both technical solutions and organisational procedures that improve the reliability of the main components and prevent failure mechanisms that could lead to large cracks.

The LBB principle has not been applied to the plants in use in Finland where full-scale primary circuit breaks are the design basis. Whip restraints and jet impingement shields constructed to protect against the aforementioned also provide protection in other accident situations causing dynamic loads. This design basis is not excluded in the LBB analysis guidelines for new plants proposed in Guide YVL E.4. The defence in depth strategy implemented in the Olkiluoto 3 unit includes both the LBB analyses and structures that protect against pipe breaks.

In the aforementioned section on protection against accident situations, Guide YVL E.4 sets out strength engineering procedures and acceptance criteria that provide sufficient structural safety. The input data on the progress of accident situations needed in the application of the aforementioned are specified in requirements set out in other YVL Guides.

5 Needs for changes taken into account in the revision

The needs for changes due to changes made to international and national laws/regulations and the change proposals made in connection with the preparation of the YVL Guide implementation decisions (SYLVI) together with others recorded in STUK's change proposal database have been considered when updating the requirements. In addition, the possibilities to reduce the so-called administrative burden have been considered.

Changes to para. 801 facilitate the operation of the licensee and may reduce the administrative burden. With the proposed modification, Guide YVL E.4 observes the same procedures with regard to quality management as other E series Guides. The requirements of the Guide do not contain other possibilities for administrative burden reduction.

Regulatory references to Regulation STUK Y/1/2018 have been updated. In line with other YVL Guides, publication years of the ASME pressure equipment standards are not included in the reference list. Similar to the Olkiluoto 3 project, however, it will be possible to have a specific version of the standard approved for use in connection with future construction projects. The same applies to references to the material standards ASTM E 1921 and ASTM E 1820.

References to Guide YVL B.5 “Reactor coolant circuit of a nuclear power plant” have been clarified.