GUIDE **YVL A.8** / 20 May 2014

# AGEING MANAGEMENT OF A NUCLEAR FACILITY



With regard to new nuclear facilities, this Guide shall apply as of 1 June 2014 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 A.8, 15 November 2013.


Second, revised edition
Helsinki 2014

ISBN 978-952-309-153-5 (print) Kopijyvä Oy 2014
ISBN 978-952-309-154-2 (pdf)
ISBN 978-952-309-155-9 (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.*

In accordance with 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**. The systems, structures and components of a nuclear facility are subjected to a number of stresses, as a result of which their integrity and performance may be impaired. The requirements related to systems, structures and components may also change during the service life of the nuclear facility, and advances may be made in the available technology so that its systems, structures and components are no longer up to the required standard. Provision shall be made in respect of the aforementioned factors i.e. the ageing of systems, structures and components already in the design stage, and they shall be controlled by monitoring and maintaining the operability of systems, structures and components up until their decommissioning.

**102**. The following regulations serve as the legal basis of the present Guide:

Nuclear Energy Act (990/1987) [1]

Section 7 a: *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 and safety research and advances in science and technology.*

Section 7 f: *Safety shall take priority during the construction and operation of a nuclear facility.*

*The holder of a construction licence, as referred to in Chapter 5 herein, shall be responsible for the nuclear facility's construction in accordance with safety requirements.*

*The holder of an operating licence, as referred to in Chapter 5 herein, shall be responsible for the nuclear facility's operation in accordance with safety requirements.*

*Moreover, the condition and operating experiences of any nuclear facility shall be systematically monitored and assessed.*

Nuclear Energy Decree (161/1988) [2]

Section 111: *The Radiation and Nuclear Safety Authority (STUK) controls the operation of a nuclear facility to ensure that the operation of the facility is safe and complies with the licence conditions and the approved plans and that the operation also in other respects adheres to the Nuclear Energy Act and to the regulations issued by virtue of the Act. The control of the operation of a nuclear facility also involves the maintenance, repairs, inspections and tests of the nuclear facility systems, structures and components.*

Government Decree on the Safety of Nuclear Power Plants (717/2013) [3]

Section 3: *The safety of a nuclear power plant shall be assessed when applying for a construction licence and operating licence, in connection with plant modifications, and periodic safety reviews during the operation of the plant. It shall be demonstrated in connection with the safety assessment that the nuclear power plant has been designed and implemented in a manner that meets the safety requirements. The safety assessment shall cover all the nuclear power plant states. The safety of a nuclear power plant shall be assessed also after accidents and, whenever necessary, on the basis of the safety research results.*

Section 4: *The systems, structures and components that implement or are related with safety functions shall be designed, manufactured, installed and used so that their quality level, and the assessments, inspections and tests, including environmental qualification, required to verify their quality level, are sufficient considering the safety significance of the item in question.*

Section 5: *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.*

Section 24: *Safety-significant operational events shall be investigated for the purpose of identifying the root causes as well as defining and implementing the corrective measures. The nuclear power plant's own as well as operational experience feedback from other nuclear power plants shall be collected, safety research results and advances in technology monitored, and assessed for the purpose of continuously enhancing safety. Opportunities for technical and organisational safety improvements resulting from operating experience, safety research and advances in technology shall be assessed and implemented to the extent justified on the basis of the principles laid down in Section 7 a of the Nuclear Energy Act (990/1987).*

Section 26: *The systems, structures and components important to the safety of a nuclear power plant shall be available as detailed in the design basis requirements. Their availability and the impact of the operating environment shall be supervised by means of inspections, tests, measurements and analyses. Availability shall be confirmed in advance by means of regular maintenance, and preparations shall be made for maintenance and repair to avoid reduced availability. Condition monitoring and maintenance shall be designed, instructed and implemented in a manner that can reliably ensure the integrity and operability of the systems, structures and components throughout their service life.*

Government Decree on the Safety of the Disposal of Nuclear Waste (736/2008) [4]

Section 18: *The nuclear waste facility shall have a condition monitoring and maintenance programme for ensuring the integrity and reliable operation of systems, structures and components. Written orders and appended instructions shall be issued for the service and repair of components.*

**103.** The requirements set forth in the present Guide are based on the international levels of requirement specified in [5] and [6].

# 2 Scope of application

**201.** The present Guide sets forth the requirements pertaining to the design, operation and maintenance activities conducted by the licensee (licence applicant prior to the granting of the construction licence) with regard to the ageing management of the systems, structures and components (SSC) of a nuclear facility, and describes the regulatory oversight by means of which STUK controls compliance with said requirements.

**202.** The provisions of the present Guide shall apply to all nuclear facilities at all stages of their life cycle to the extent necessary for ensuring the operability of systems, structures and components important to nuclear and radiation safety.

**203.** Requirements related to the ageing management of a nuclear facility concerning design, operation and maintenance activities and their technical and administrative implementation, reporting and supervision are set out in the following Guides:

- YVL A.1, Regulatory oversight of safety in the use of nuclear energy
- YVL A.3, Management system for a nuclear facility
- YVL A.5, Construction and commissioning of a nuclear facility
- YVL A.7, Probabilistic risk assessment and risk management of a nuclear power plant
- YVL A.6, Conduct of operations at a nuclear power plant
- YVL A.9, Regular reporting on the operation of a nuclear facility
- YVL A.10, Operating experience feedback of a nuclear facility
- YVL B.1, Safety design of a nuclear power plant
- YVL B.5, Reactor coolant circuit of a nuclear power plant
- YVL B.7, Provisions for internal and external hazards at a nuclear facility
- YVL C.3, Limitation and monitoring of radioactive releases from a nuclear facility

- YVL C.6, Radiation monitoring at a nuclear facility
- YVL D.2, Transport of nuclear materials and nuclear waste
- YVL D.3, Handling and storage of nuclear fuel
- YVL D.5, Disposal of nuclear waste
- YVL E series, Structures and equipment of a nuclear facility.

# 3 General requirements

**301.** The licensee shall describe ageing management as part of the management system of the nuclear facility.

**302.** The licensee shall define an ageing management programme for the nuclear facility comprising the functions, duties and responsibilities for assuring the operability and technological conformance of systems, structures and components (hereinafter 'SSC', see the definition below) related to the safety of the nuclear facility throughout their service life.

**303.** The licensee shall target the ageing management of the nuclear facility at all SSC.

**304.** The licensee shall have systematic procedures in place for identifying both foreseen and new ageing mechanisms of SSC. Research data, operating experience feedback from other nuclear facilities as well as the feedback received from the licensee's in-house operating and maintenance organisation shall be utilised to identify ageing mechanisms. Annex A presents typical ageing mechanisms that may occur at a nuclear facility.

**305.** The individual ageing of SSC shall be prepared for. When a system has redundant subsystems, the operability of a subsystem shall be assured separately and independently from others.

**306.** The physical degrading of a nuclear facility shall be managed by means of SSC's design solutions, condition monitoring and maintenance, and by avoiding any operating modes and conditions that give rise to unnecessary stresses. Provision shall be made for any repairs and modifications necessitated by an atypical degradation.

**307.** The obsolescence of a nuclear facility shall be managed by regularly assessing the conformity and up-to-dateness of SSC. Any SSC in which deviations from the adoptable safety requirements are detected shall be brought to conformance with ones. Furthermore, the applicability of new technical solutions and methods to enhance the safety of the nuclear facility shall always be investigated.

**308.** For the ageing management of the nuclear facility's SSC the licensee shall have in its organisation designated responsible individuals from different disciplines, with defined duties and coordinated responsibilities. The responsible individuals shall possess the necessary competence and experience to discover such solutions that the adverse effects of ageing mechanisms on the operability of SSC may be prevented.

**309.** The licensee shall have documented procedures in place for ensuring that any information and knowledge necessary for discharging the duties is not lost in the event of personnel changes.

**310.** The licensee shall assess the comprehensiveness and effectiveness of the ageing management system on a regular basis. Ageing management shall, in particular, be improved whenever increasing overhaul needs or failure frequency of SSC are found.

**311.** The licensee shall keep current with new developments in design solutions, manufacturing techniques and condition monitoring and maintenance methods for the purpose of improving the ageing management of SSC. This requirement also applies to any amendments made to the relevant regulations and standards.

**312.** The licensee shall record and keep up-to-date the reference information necessary for the ageing management of SSC. These include design documents (design bases, technical specifications, construction materials, drawings, functional descriptions), qualification data, the manufacturing result documentation, and modification records.

# 4 Design and procurement

**401.** SSC shall meet the design bases defined as the requirements applicable to them both during normal operation and under transient and accident conditions throughout their service life. The requirements shall be defined for the design, dimensioning, quality, functioning, operation and operating environment of SSC, as well as for their inspectability and maintability.

**402.** The design solutions shall minimise the ageing of SSC and create the necessary preconditions for ensuring that the operability of the SSC can be monitored and maintained throughout their service life. Consequently:

a. The materials selected for the SSC shall be known to withstand the mechanical, chemical and other potential stresses of design basis operational and ambient conditions.
b. The design and dimensioning of SSC shall provide for the foreseen ageing mechanisms. Examples of such mechanisms include the radiation embrittlement of the reactor pressure vessel, and thermal fatigue in regions where hot and cold water flows are mixed.
c. It shall be possible to verify SSC's integrity. Provisions shall be made (in terms of geometry, selection of material and accessibility) for non-destructive testing methods by means of which the conformance of the SSC integrity can be periodically assured.
d. It shall be possible to verify SSC's performance. Provisions shall be made (in terms of measurements, process connections and testing equipment) for functional testing or other periodic checks by means of which the conformance of the SSC performance can be assured.
e. The design shall provide for real-time condition monitoring of the SSC (device compatibility, instrumentation readiness, capacity of information systems) if real-time information about the operability of the SSC materially improves the safety of the nuclear facility.
f. It shall be possible to maintain the SSC. The operability over the SSC lifetime shall be possible to ensure by means of regular service and replacement of parts where necessary.

**403.** The required operability of the SSC shall be reliably maintained regardless of the effects of ageing even in the most adverse design basis operational conditions. The operability-related uncertainties involved in design, dimensioning, fabrication, operating conditions, condition monitoring and maintenance shall be investigated and reduced commensurate with the safety significance of the SSC.

**404.** The design solutions of a SSC shall be qualified by means of operating experience, analyses and testing. If the operability of a SSC cannot be reliably assured by other means, the design solutions shall be qualified experimentally (by means of testing).

**405.** The licensee shall possess a technically reasoned estimate of the service life for SSC susceptible to ageing. The service life can be extended if the operability of the SSC can be verified over a service life longer than the original estimate, and it must be shortened if the ageing process is found to be faster. The procedures based on strength analyses are discussed in Guide YVL E.4.

**406.** The licensee shall ensure in connection with the procurement of a SSC that the delivery comprises proper operating, condition monitoring and maintenance instructions for SSC, as well as recommendations for spare parts and supplies, education and potential technical support.

**407.** The design solutions of a nuclear facility shall ensure that condition monitoring and maintenance duties of SSC will not expose the personnel to any unnecessary radiation or occupational safety risks. Provision shall be made in the layout design for the replacement of SSC designed to be replaced, in order to allow the necessary lifting and transfer operations to be accomplished safely.

**408.** The licensee shall define in its design and procurement instructions requirements for the ageing management of SSC.

# 5 Fabrication

**501.** Fabrication methods that do not contribute to the development of adverse ageing effects shall be employed in the fabrication of SSC or any repairs made during their fabrication.

**502.** The inspections and testing performed during the fabrication of a SSC shall ensure that the SSC is free of any defects in materials and workmanship that could result in failure or unforeseen ageing in design basis operational conditions. The inspection and testing scope shall be commensurate with the safety significance of the SSC.

**503.** The licensee shall ensure that a reference or baseline level for the data used for the subsequent in-service monitoring of the operability of the SSC concerned is recorded prior to its placement in service. Reference levels are typically needed for the non-destructive testing of pressure equipment, piping wall thickness measurements and vibrations of rotating machines. Photographs can be used for those objects of inspection for which visual inspections are carried out.

**504.** The licensee shall at least collect and store material samples for those SSC that are not designed to be replaced during the service life of the nuclear facility, and in respect of which reference data on their construction material may later be needed for the purpose of evaluation of the operability or repairing method of the SSC.

**505.** The licensee shall ensure that transportations, storage and installation do not endanger the operability of the SSC or contribute to the development of adverse ageing mechanisms.

# 6 Operation

**601.** The operating parameters and loading process and ambient conditions of the SSC shall be monitored and kept within the limits of design basis operational conditions.

**602.** The licensee shall utilise operating experience feedback from nuclear facilities as well as the feedback received from the licensee's in-house operating and maintenance organisation. This information shall be used for evaluating alternative modes of operation for SSC that can be used for reducing the adverse effects of ageing for the operability of the SSC.

**603.** The licensee shall instruct and train its operating personnel to optimise the modes of operation of the nuclear facility. Using improper operation modes, unnecessary stresses, fast process transients (e.g. pressure, temperature and hydrochemistry) and other unfavourable operating conditions shall be avoided.

**604.** If a SSC is found to have been subjected to loads beyond design basis or the operability of the SSC may for some other reason have been compromised, the preconditions for the continued use of the SSC shall be ascertained by means of inspections, testing, computational analysis or other examinations.

**605.** If a SSC has lost its operability and it cannot be restored, it shall be removed from service.

**606.** The licensee shall ensure that the preconditions for the qualification of SSC continue to be met. A new qualification shall be initiated on a timely basis prior to the expiry of its validity.

# 7 Condition monitoring and maintenance

## 7.1 Condition monitoring

**701.** The licensee shall monitor the operability of the nuclear plant's SSC. It shall be possible to reliably detect the need for overhaul of a SSC before the SSC fails, and the failure of a SSC in real time. Condition monitoring may be based on visual inspections, non-destructive testing, functional tests and pressure and leaktightness tests, or other actions that provide direct information concerning the operability of the SSC. Condition monitoring is also considered to include measurements and samples that provide indirect information concerning the operability of the SSC, or information concerning the conditions that affect operability (such as cumulative usage, hydrochemistry parameters, and material samples that have been subjected to operating environment).

**702**. The comprehensiveness of the condition monitoring of a SSC, as well as the methods employed, monitoring intervals, resourcing, acceptance limits, and other factors affecting the effectiveness of condition monitoring, shall be commensurate with the safety significance of the SSC being monitored, the assessment of which shall be based on a probabilistic risk assessment. Provision shall also be made for identifying the kind of ageing and for detecting the types of failures that cannot be foreseen based on operating experience.

**703**. The condition monitoring of SSC shall be conducted in compliance with the requirements set out in the “E” series YVL Guides regarding component-specific scope and the in-service inspection of pressure equipment.

**704**. The condition monitoring of a SSC shall be conducted on a real-time basis whenever this materially improves the effectiveness of the condition monitoring of the SSC as compared to condition monitoring conducted at regular intervals. For example, this is of relevance to the monitoring of vibrations, leaks, fatigue or loose parts, and valve operation diagnostics.

**705**. The licensee shall record the essential condition monitoring data and monitor their development trends against the reference levels. Whenever changes are detected, their significance in terms of the operability and service life of the SSC shall be determined.

**706**. The licensee shall monitor the operational conditions and events giving rise to fatigue loads on important pressure equipment. The load monitoring results shall be used for assessing the ageing of pressure equipment by comparing the actual load history against the design basis usage of the SSC concerned. More detailed requirements are specified in Guide YVL E.4.

**707**. The measuring and analysis instruments used in condition monitoring shall be calibrated at regular intervals or otherwise checked for their required accuracy. It shall be possible to verify the validity of the calibration from device-specific calibration certificates.

## 7.2 Maintenance

**708**. The licensee shall ensure that any SSC that is currently in or ready for operation meets the applicable operability requirements in normal operational conditions as well as in all design basis transient and accident conditions.

**709**. Whenever a SSC is found to be in need of overhaul, the refurbishment shall be conducted before operability is lost. No SSC shall be deliberately used until failure.

**710**. The comprehensiveness of the maintenance of a SSC, as well as the methods employed, maintenance intervals, resourcing and other factors affecting the effectiveness of maintenance shall be commensurate with the safety significance of the SSC, the assessment of which shall be based on a probabilistic risk assessment.

**711**. When a SSC is refurbished or repaired, the licensee shall determine whether a similar need for overhaul or a similar fault exists, either manifestly or latently, in other SSC that are similar in terms of structure, function or operating conditions (a common cause failure). At the same time, the licensee shall determine whether the observed degradation or loss of operability can be avoided in the future by improving the condition monitoring or maintenance of the SSC.

**712**. The approval procedures specified in the “E” series YVL Guides shall be complied with in the overhaul and repair work of the SSC. However, the approval procedures can be agreed upon on a case-by-case basis in the event of urgent repair work that needs to be carried out without delay to bring or maintain the nuclear facility in a safe state.

**713**. To avoid any direct or indirect risk to the personnel or safety, process, power supply and I&C connections shall be secured for the duration of the maintenance work, and the altered connections shall be restored when the maintenance work is brought to completion so as. More detailed requirements are specified in Guide YVL A.6.

**714.** The operability of a serviced, refurbished or repaired SSC shall be ensured by means of adequate inspections or tests before the SSC is placed in service.

**715.** The licensee shall record the operation and maintenance history of SSC. Each SSC shall be tracked individually to ensure that all maintenance, repair work and modifications performed as well as any stresses and failures the SSC has undergone are traceable throughout the service life of the SSC.

**716.** Any SSC whose service place may change in maintenance rotation, for example, shall be furnished with indelible identification marking to ensure traceability.

**717.** The tools used in maintenance operations shall be suitable for their intended purpose, and they shall be serviced and stored appropriately. Any tools requiring adjustments shall be inspected at regular intervals.

## 7.3 Programmes and instructions

**718.** The licensee shall have programmes in place defining the condition monitoring and maintenance of SSC complete with timetables or intervals for the actions to be taken. Risk-informed methods shall be employed in the preparation of the programmes as specified in Guide YVL A.7.

**719.** The condition monitoring and maintenance of a SSC shall be unambiguously and clearly instructed. The licensee shall ensure by means of training and induction that the actions taken are in compliance with the instructions.

**720.** The condition monitoring and maintenance programmes pertaining to a SSC, as well as the associated instructions, shall be based on the applicable standards, the manufacturer's recommendations or the operating experience feedback received in-house or from other nuclear facilities.

**721.** The licensee shall have operating practices and information systems in place to ensure that the condition monitoring and maintenance of the SSC is carried out as planned.

**722.** The condition monitoring and maintenance programmes as well as the associated instructions shall be continuously assessed and revised based on the feedback received on ageing management. Any identified needs for modification shall be analysed and the necessary steps to improve effectiveness taken in the subsequent programme and instruction updates.

## 7.4 Spare parts

**723.** The licensee shall regularly monitor and assess the availability and operability of the spare parts necessary for the maintenance of SSC at the nuclear facility. This also applies to the spare parts for the measurement and analysis instruments as well as the tools used in the condition monitoring and maintenance of SSC.

**724.** The licensee shall have spare parts available on site for the functions necessary for ensuring that the nuclear facility is kept in a safe state during prolonged transients and accidents. As far as such functions are concerned, the spare part stock shall cover at least one subsystem (redundancy) of 100% capacity that is alone sufficient for sustaining the function. The spare part stock shall include spare or replacement parts for SSC that may fail in prolonged operation and are critical for the operability of the functions concerned.

**725.** The spare parts and consumables shall satisfy the design basis requirements imposed on them. The conformance of a spare part or consumables shall be carefully examined, and an approval specified in the "E" series YVL Guides shall be obtained for it prior to its installation.

**726.** The licensee shall have documented procedures in place for the procurement, reception and storage of spare parts and supplies to ensure the conformance of the spare parts and supplies used in SSC.

**727.** Controls shall be put in place to monitor the remaining shelf and service life of the spare parts and consumables kept at a nuclear facility.

**728.** The licensee shall have procedures in place for reliably preventing counterfeit products in the spare part supply chain.

**729.** The licensee shall ascertain the availability of spare parts and technical support for the SSC on a regular basis. If the availability is coming to an end, the licensee shall initiate the necessary steps for addressing the shortages under changed circumstances well in advance.

# 8 Modifications

**801.** The licensee shall systematically control the physical degrading and potential obsolescence of the nuclear facility to identify the need for modifications at the facility. The replacement of a SSC or its spare part with a different one, or the addition of a new SSC into the nuclear facility, are also considered modifications.

**802.** The planning of foreseeable major modification or repair projects shall be commenced well on time to permit timely execution of the projects in accordance with the requirements and predefined plans. Requirements pertaining to the planning of modifications are specified in Guide YVL A.5, Construction and commissioning of a nuclear facility.

**803.** The licensee shall initiate the preparation of the modifications provided under the enforced safety requirements without delay.

**804.** The licensee shall assess the safety impact of the modification at the planning stage. The modification may not compromise the safety of the nuclear facility or the preconditions for the condition monitoring or maintenance of SSC.

**805.** When the modification is planned, the current system and component design, their actual design bases and structures shall be known, and any factors posing restrictions on the modification of the SSC shall be identified based on them. The effects of the modification on other nuclear plant SSC shall be determined at the planning stage, too.

**806.** The licensee shall prepare system and component-level plans for the modification. The approval procedures specified in the other YVL Guides shall be complied with in the pre-inspection and implementation of the modification.

**807.** When SSCs are modified, drawings, instructions and other documents rendered out of date by the modification shall be identified and they shall be promptly updated.

**808.** The licensee shall ensure by means of training that the effects of the modification on the operation, condition monitoring and maintenance of the SSC and the nuclear facility concerned are communicated to the operation and maintenance organisations.

**809.** The licensee shall maintain a register of the design bases of the SSC and the modifications performed.

**810.** The licensee shall keep STUK informed of any extensive future modification of SSC foreseen by the licensee during the service life of the nuclear facility.

# 9 Documents to be submitted

## 9.1 Plan for principles of ageing management

**901.** When applying for a construction licence for a nuclear facility, the licensee shall submit to STUK for approval a plan for principles of the ageing management of SSC.

**902.** The conceptual plan for ageing management shall describe the principles according to which the ageing management of the SSC of the foreseen nuclear facility is planned to be implemented. The plan shall accommodate the following main points:

a. the preconditions for implementing ageing management at the nuclear facility:
   - administrative organisation;
   - securing of the necessary expertise;
b. the provisions made in respect of ageing in the design, procurement and fabrication of SSC:
   - the design solutions to reduce ageing;
   - the inspectability and maintenability of SSC;
   - the quality requirements pertaining to the procurement and fabrication;

 – the utilisation of reference data from other nuclear facilities;

c. ageing management during the construction and operation of the nuclear facility:
 – the condition monitoring and maintenance concepts for the SSC;
 – the avoidance of any operating modes and conditions that give rise to unnecessary stressors;

d. the qualification of SSC against ageing;
 – the measures to demonstrate that the SSC satisfy the requirements imposed on them in design basis operational conditions up until the end of their service life.

## 9.2 Ageing management programme

**903**. When applying for an operating licence for a new nuclear facility, the licensee shall submit to STUK for approval the ageing management programme for the nuclear facility that is to be complied with during the operation of the facility. The program shall present the ageing management process to the following level of detail:

a. the organisation of ageing management (responsibilities, duties, coordination);
b. the measurement of the effectiveness of ageing management (e.g. failure trends of SSC, load history, hydrochemistry parameters and other similar factors based on which the changes in the SSC operability can be assessed in the long term);
c. the utilisation of condition monitoring and maintenance feedback data, operating experience feedback from other nuclear facilities, and related research data in the ageing management of SSC;
d. the relationship between the ageing management of SSC and their safety significance (e.g. extent of condition monitoring and maintenance based on the safety significance of the SSC);
e. the following information on each SSC:
 – service place code;
 – operating conditions, design bases, materials and structure, complete with any other details relevant to ageing management;
 – identified ageing mechanisms;
 – condition monitoring and maintenance programmes complete with the associated instructions;
 – qualification plans (for those SSC in respect of which qualification is required even after the commissioning of the nuclear facility);
 – spare part stocks (for those SSC that are needed in functions used for managing prolonged transient and accident conditions);
f. procedures for managing the obsolescence of SSC;
g. table of contents of the ageing follow-up report.

**904**. As to the details pertaining to SSC, at least references to the documents containing the required information shall be provided in the ageing management programme. The reference documents shall have been submitted to STUK or be available to STUK's inspectors at the plant site.

## 9.3 Ageing management follow-up report

**905**. During the operation of the nuclear facility, the licensee shall annually submit a follow-up report on the ageing management of SSC to STUK for information during the first trimester of each year.

**906**. The follow-up report shall provide the following information on the SSC covered by the ageing management regime:

a. long-term trends in the number of failures and failure types;
b. major servicing, repair, replacement and modification work carried out during the reporting period;
c. an assessment of operability and ageing-induced changes to the safety margins;
d. any development needs related to ageing management (condition monitoring, maintenance, need for replacement of SSC, research) in both the short and long term;
e. the validity of qualifications and changes thereto during the reporting period;
f. summary of spare parts (stock, condition).

Information pertaining to components other than the primary components of the nuclear facility concerned can be provided in the form of a summary that is appropriately grouped by system, for example.

**907.** Where applicable, the follow-up report may make reference to the result documentation of other follow-up procedures that has separately been submitted to STUK or is available to STUK's inspectors at the plant site.

**908.** If any changes have been made to the ageing management programme, the updated programme shall be submitted to STUK for information along with the follow-up report.

# 10 Regulatory oversight by the Radiation and Nuclear Safety Authority

**1001.** At the construction licence stage, STUK will review the plan for principles of the ageing management of the nuclear facility. An approved conceptual plan for ageing management is one of the prerequisites for any endorsement by STUK of the application for a construction licence.

**1002.** At the operating licence stage, STUK will review the ageing management programme. An approved ageing management programme is one of the prerequisites for any endorsement by STUK of the application for an operating licence.

**1003.** During the operation of the nuclear facility, STUK will assess the implementation of the ageing management programme based on the follow-up report prepared annually by the licensee.

**1004.** STUK will assess the effectiveness of the ageing management of the nuclear facility as part of the periodic inspection programme and other inspections, and in connection with the processing of the periodic safety reviews described in Guide YVL A.1 and the operating licence applications.

# Definitions

**Physical degrading**

Physical degrading shall refer to the degradation of structural or functional characteristics in use or with time as a result of physical, chemical and/or biological mechanisms. Physical degrading may lead to the loss of operability of a SSC.

**Ageing**

Ageing shall refer to the potential physical degrading and obsolescence of SSCs at a nuclear facility.

**Ageing management**

Ageing management shall refer to assuring the operability of SSCs throughout the service life of a nuclear facility. It shall also refer to assuring the conformance with adoptable requirements and current level of technological development.

**Ageing management programme**

Ageing management programme shall refer to the functions and duties defined by the licensee, pursuant to which the licensee implements the ageing management of a nuclear facility.

**Qualification**

Qualification shall refer to a process to demonstrate the ability to fulfil specified requirements (corresponds to the qualification process of the ISO 9000 standard).

**Repair**

Repair shall, in the context of Guide YVL A.8, refer to restoring the operability of a faulty SSC.

**Condition monitoring**
Condition monitoring shall refer to the determining of the operability of a SSC.

**Maintenance**
Maintenance shall refer to the planned service of SSC to reduce the probability of failure in advance, or the overhaul or repair of a SSC undertaken on the basis of observed needs.

**Overhaul**
Overhaul shall refer to the elimination of the non-conformities or shortcomings observed in the structure or performance of a SSC as the SSC still fulfils the requirements set for its operability.

**Service life**
Service life shall refer to the period of time during which a SSC installed at its service place is estimated to reliably retain its operability.

**Operability**
Operability shall refer to the integrity and performance of SSC in conformance with its design bases.

**Systems, structures and components (SSC)**
Systems, structures and components (SSC) shall refer to any mechanical, electrical, I&C or civil system, structure or component in safety classes 1, 2 and 3 or in class EYT/STUK (non-nuclear).

**Modification**
Modification shall refer to introducing changes to a system, structure or component so that it no longer corresponds to previous specifications.

**Obsolescence**
Obsolescence shall mean that a plant SSC fails to meet the newly adopted safety requirements or no longer represents the current technological development in terms of the assurance of safety. The lack of technical support or spare parts is also regarded as a manifestation of the obsolescence of a SSC.

**Fault, failure**
Fault or failure, in the context of Guide YVL A.8, shall mean that a SSC no longer meets the operability requirements.

# References


1. Nuclear Energy Act (990/1987).
2. Nuclear Energy Decree (161/1988).
3. Government Decree on the Safety of Nuclear Power Plants (717/2013).
4. Government Decree on the Safety of Disposal of Nuclear Waste (736/2008).
5. IAEA, Ageing Management for Nuclear Power Plants, Safety Guide No. NS-G-2.12, 2009.
6. WENRA Reactor Safety Reference Levels, Issue I: Ageing Management, Issue K: Maintenance, Inservice Inspections and Functional Testing.

# ANNEX A Typical ageing mechanisms

| Physical degrading | Susceptible materials, regions and components |
|---|---|
| **Mechanical components** | |
| **A01. Stress corrosion cracking (SCC)** – Cracking in metal induced by the combined influence of corrosion and tensile stress. Tensile stress, on the other hand, may be induced by internal tensions and/or an external load. The corrosive environment leading to stress corrosion cracking is specific to each individual material. | Austenitic stainless, carbon and low-alloy steels, Ni-based alloys including welded joints and their vicinity; reactor internals, primary and secondary circuit pipings, reactor support structures, steam generators, pressuriser. |
| **A02. Intergranular stress corrosion cracking (IGSCC)** – Stress corrosion cracking that propagates along grain boundaries in sensitised austenitic material (see stress corrosion cracking). | Welded joints in austenitic stainless steels; primary and secondary circuit pipings, reactor internals, bellows. |
| **A03. Transgranular stress corrosion cracking (TGSCC)** – Stress corrosion cracking that propagates through the grains (see stress corrosion cracking). | Austenitic stainless and carbon steels, including welded joints and their vicinity; steam generator casing, pipings, for example reactor coolant pipes, control rod drive mechanisms; pipings connected to the containment steel liner; stainless steel bellows, bolts in flanged joints. |
| **A04. External chloride stress corrosion cracking (ECSCC)** – The external surface of a component subjected to tensile stress, usually piping, is exposed to chloride-containing water due to pipe leaks and chloride-containing insulation material, for example (see stress corrosion cracking). | Austenitic stainless steels and welded joints; all pipings. |
| **A05. Primary water stress corrosion cracking (PWSCC)** – Stress corrosion cracking occurring in high-temperature oxygen-free water (see stress corrosion cracking). | Ni-based alloys; non-stress relief annealed or cold-worked fittings and welded joints; the primary side of steam generator heat transfer tubes, control rod drive mechanism fittings, pressuriser. |
| **A06. Strain-induced corrosion cracking (SICC)** – A monotonically rising dynamic load in oxygenous water or steam may induce cracking akin to stress corrosion cracking. | Low-alloy ferritic steels; feedwater nozzles and horizontal pipings, thin-walled pipes and pipe bends. |
| **A07. Boric acid corrosion** – A primary water leak may induce boric acid corrosion in carbon and low-alloy steels. The mechanism is a common form of corrosion and/or material wastage. | Low-alloy ferritic steels; control rod drive mechanism penetrations and the reactor pressure vessel head, containment steel lining, bolted joints. |
| **A08. Erosion corrosion** – In erosion corrosion, flowing liquid dissolves the protective layer (corrosion product) on a metal surface, thereby accelerating the corrosion when the flow rate exceeds its critical value. | Carbon and low-alloy steels; flow discontinuities that cause vortices, such as pipe bends and branches, flow inlets and reducers, welds that are poorly formed on the inside, outlet sides of flow measurement flanges. |
| **A09. Water hammering** – Water hammering may induce high dynamic loads. | Flow arrest in the event of quick valve closure, for example. |
| **A10. Microbiologically-influenced corrosion** – Water contaminated by organic matter in a raw water system, for example, may induce local corrosion at elevated temperatures and low flow rates in crevice conditions, in particular. | Austenitic and ferritic steels, including welded joints and dissimilar metal welds; various auxiliary pipings, containment steel lining, reinforcing steels. |
| **A11. Pitting** – Local corrosion pits may develop in pipings where the flow rate is occasionally or continuously low and the water is oxygenous and contaminated (by fluorides or chlorides, for example). | Austenitic and ferritic steels and Ni-based alloys, including welded joints and dissimilar metal welds; steam generator heat transfer tubes, bolts in the reactor pressure vessel head, steam generator frame. |
| **A12. Crevice corrosion** – Form of corrosion occurring in crevices at elevated temperatures and oxidising conditions. | Austenitic and ferritic steels and Ni-based alloys; thermal shields, flanged joints, containment steel lining, etc. |

| Physical degrading | Susceptible materials, regions and components |
|---|---|
| **A13. Erosion-Cavitation** – In a fluid flow, the pressure drops locally to or below the steam pressure corresponding to the ambient temperature. The steam bubbles thus formed suddenly implode when the steam pressure is again exceeded in the flow. The phenomenon causes local pressure shocks that accelerate corrosion by dissolving the protective layer on the surface or, when strongest, cause mechanical damage to the surfaces. | All metal-based materials in components where the pressure may fall below the steam pressure as a result of high flow rates or exceptionally high temperature; valves, pumps, steam generator internals. |
| **A14. Inter-Granular attack (IGA)** – Corrosion in sensitised austenitic material that propagates along the grain boundaries in the material; usually cracking induced by inter-granular stress corrosion cracking (see stress corrosion cracking; inter-granular stress corrosion cracking). | Austenitic stainless steels and Ni-based alloys, including welded joints; steam generator heat transfer tubes on the secondary side. |
| **A15.** Galvanic corrosion – An electrochemical corrosion process involving two electrically conductive metals and an electrolyte. The metal with a lower electrode potential acts as anode and dissolves into the electrolyte. | The pair of ferritic and stainless steel in a seawater system whose cathodic protection is not working. In addition to metal, graphite (seal) may also act as cathode and induce corrosion of the metal surface. |
| **A16. General corrosion** – Metal corrodes at a consistent rate evenly from the entire surface. | Unprotected carbon and low-alloy steels, hard chromated coatings at high temperatures. |
| **A17. Fatigue** – Damage propagating in a structure subjected to mechanical alternating load or temperature changes, the stages of which are microcrack initiation, crack growth, and cracking. | All metal-based constructions. Susceptible materials, regions and components include vibrating and rotating constructions, regions where flows become mixed, and welds in nozzles and other similar stress concentration areas. |
| **A18. Thermal fatigue** – Variations in temperature due to a variety of reasons (the mixing of hot and cold water, for example) induce alternating loads, which eventually results in metal fatigue (see fatigue). | Austenitic and ferritic steels, Ni-based alloys, parent metal and welded joints; nozzles, tees, stress concentration areas in pipings. |
| **A19. Corrosion fatigue** – When metal is subjected to vibrations, alternating loads or temperature variations, its fatigue resistance decreases (the crack initiates and/or grows faster) because of a corroding environment. | Austenitic stainless and low-alloy ferritic steels, including welded joints especially in nozzles and other stress concentration areas. |
| **A20. Thermal ageing and embrittlement** – High operating temperatures induce thermal ageing resulting in embrittlement of the metal. | Stainless steels with austenitic-ferritic structure, such as cast austenitic and duplex stainless steels, ferritic steels that contain impurities, precipitation-hardening strong stainless steels; reactor internals, control rod drive mechanism, pipings, valves, pumps, axles. |
| **A21. Stress relaxation** – The yielding of metal subjected to stress at elevated temperatures; elastic strain converts to plastic strain. Neutron radiation may accelerate the process. | Austenitic and ferritic steels, Ni-based alloys; pre-tightened bolts in reactor internals and flanged joints, for example. |
| **A22. Radiation embrittlement** – The strength of a material, usually steel, increases while its ductility decreases when exposed to neutron radiation. The purity of steel has a significant impact on the intensity of the embrittlement. | Reactor pressure vessel steels; parent metal and welded joints; austenitic stainless steels, Ni-based alloys; reactor internals. |
| **A23. Irradiation-assisted stress corrosion cracking (IASCC)** – Cracking induced by a neutron dose in excess of a certain threshold value when the other preconditions for stress corrosion cracking are met (see stress corrosion cracking). | Austenitic stainless steels; reactor internals such as various kinds of bolts. |
| **A24. Irradiation-induced swelling** – Large neutron doses may induce notches or macroscopic structural deformation in certain austenitic stainless steels. | Austenitic stainless steels; reactor internals. |

| Physical degrading | Susceptible materials, regions and components |
|---|---|
| **A25.** Hydrogen damage – Damage related to the effects of hydrogen on metal. Examples include hydrogen embrittlement, hydrogen blistering and hydrogen or cold cracking due to welding. | Most commonly ferritic steels. Hydrogen embrittlement causes reduced deformation capacity. Hydrogen blistering and cracking cause delayed cracking. The hydrogen present in molten metal in castings and welds may also cause the formation of pores. Hydrogen embrittlement may arise in service when radiolysis or the hydrogen released in corrosion, for example, act as sources of hydrogen. |
| **A26. Erosion, wear and wastage** – Material comes off from a surface subjected to wear through the action of various mechanisms. This results in weight losses, dimensional changes and deformations as well as causing degradation in the quality of the surface. | Austenitic and ferritic steels, Ni-based alloys; reactor internals, control rod drive mechanisms, steam generator heat transfer tubes, pipings in general, valves, pumps, etc. |
| **A27. Fretting** – A process that occurs at the contact area between two surfaces pressed together when the surfaces are subjected to a relative sliding motion by vibration. As a result, metal may wear and undergo corrosion or fatigue. | Austenitic and ferritic steels, Ni-based alloys; reactor and steam generator internals. |
| **A28. Denting** – The denting of thin-walled steam generator tubes occurs at the location of support plates as a result of the corrosion product layer that forms between the tube and the support plate. The process is accelerated if the support plate is of carbon steel and the secondary side water contains impurities such as chlorides. | Austenitic stainless steels, Ni-based alloys; steam generator heat transfer tubes. |
| **A29.** Creep – Time-dependent deformation that occurs at high temperatures ($T > 0.3 \times T_{\text{melting point (K)}}$) under the influence of constant stress or load. Creep may be accelerated by neutron radiation. | All metal-based materials; the mechanical components of light water reactors normally operate within a temperature range where creep is of no major significance. The bolts in reactor internals may be susceptible to radiation-accelerated creep. Creep is of relevance for reactor internals in the event that fuel becomes overheated. |
| **A30. Ageing of lubricants and greases** – The impairment of flow or lubrication characteristics as a result of, for example, impurities, oxidation, radiation, electric current, separation or polymerisation. | Bearings and slide and guide surfaces that require lubrication. |
| **A31.** Vibrations of machine foundations – The vibrations induced by the machine foundations cause damage (indentations) in contact surfaces. | Bearings of standing pumps and motors. |
| **Electrical and instrumentation and control components** | |
| **A32. Thermal ageing** – Temperature causes the degradation of the electrical, chemical and, in particular, mechanical properties of insulation materials in forms such as embrittlement when, for example, adipic acid separates from polymeric materials. | Insulation materials, penetrations and connectors. |
| **A33. Electric ageing** – Voltage causes degradation in the dielectric strength of the insulant. Combined with thermal ageing and partial discharges, electric ageing may result in the loss of electric strength. | Insulation materials. |
| **A34. Degradation of mechanical properties** – Vibration, tension, torsion, thermal expansion and contraction, and overvoltage on connection and disconnection cause degradation in the ductility and strength of the material. | Insulation materials, conductor connections and cooling fans for electronics. |
| **A35. Ageing due to humidity** – High relative humidity or water condensed from the air may induce punctures in the insulation material and corrosion. This may result in the swelling of the insulation material and in the formation of water trees. | Insulation materials, penetrations and connectors. |

| Physical degrading | Susceptible materials, regions and components |
|---|---|
| **A36. Ageing due to ionising radiation** – Ionising radiation causes embrittlement in insulation materials and degradation in their mechanical properties. | Insulation materials, penetrations and connectors. |
| **A37. Corrosion** – Chemical reactions on a metal surface induce impedance growth in contact surfaces or break the circuit. | Relay and switch contactors, contact surfaces of contactors, conductor connections. |
| **A38. Whiskers** – A process occurring in zinc, tin and silver coatings whereby very thin hair-like metal crystals grow off the surface. Whiskers may cause short circuits. | Electric cabinets, cable trays, relay contacts. |
| **A39. Metallic diffusion** – Change in the composition of electrical connection materials due to current-induced heating that may result in the degradation of electrical conductivity and/or mechanical properties. | Soldered joints in electrical and I&C components. |
| **A40. Electrical erosion** – The constant discharge of current pulses via bearings may gradually wear the rolling surface of bearings and induce bearing damage. The sparkling of opening contacts also wears the contact surfaces. | Contactor contacts, bearings of electric motors and generators. |
| **A41. Drying of electrolytic capacitors** – The capacitance of a capacitor collapses as the electrolyte volume decreases. This may also cause the thinning of the insulating aluminium oxide film, which results in breakdown and potential capacitor explosion. | Electrolytic capacitors |
| **Concrete structures** | |
| **A42. General corrosion** – See mechanical components, A15. | Reinforcing steels, anchor bolts, steel lining; carbon and ferritic low-alloy steels. |
| **A43. Pitting** – See mechanical components, A11. | Reinforcing steels, anchor bolts, steel lining; carbon and low-alloy ferritic steels. |
| **A44. Hydrogen embrittlement stress cracking (HESC)** – Corrosion accelerated by acidic water solutions, chlorides, sulphites or electric currents results in cathodic release of hydrogen that induces hydrogen embrittlement in steel. Constant tensile stress of steel causes cracking. | Cold-drawn reinforcing steels, anchor bolts. Corrosion requires that there is humidity in the structure. The binder of concrete contains sulphites; steel is affected by the chemical stress arising from chlorides, ammonium compounds or sulphites. Galvanism or electric stray currents accelerate the corrosion. The oxide layer that protects steels in a concrete structure has been damaged as a result of some other damage mechanism, or the weather protection during construction is insufficient prior to the injection of tendons. |
| **A45. Freeze-thaw deterioration** – Water inside the capillary pores of concrete expands when it freezes. The resulting pressure weathers the concrete surface. | Outdoor concrete structures where the ratio of protective pores of $\phi$ 0.01–0.8 mm to other pores is small. |
| **A46. Carbonation of concrete** – The carbon dioxide dissolved in water in a chemical reaction reacts with the alkaline hydroxides present in concrete, thereby reducing the alkalinity of the concrete. The carbonation rate depends on the impermeability, cement content and relative humidity of the concrete.<br><br>The corrosion of steels begins when the alkalinity of the concrete is reduced to a sufficiently low level (pH <9) around reinforcement steels. | Concrete constructions in humid conditions, 40% < RH < 90%. |

| Physical degrading | Susceptible materials, regions and components |
|---|---|
| **A47. Chloride attack on concrete** –The penetration of chloride ions into concrete causes corrosion even if the pH of the concrete is high. The chlorides break the oxide film providing protection against corrosion, and the corrosion of steel begins. | The aggregates, binder and water used in the concrete may have contained chlorides in harmful amounts, or chlorides penetrate into the concrete from the environment in the form of aqueous solution either by way of diffusion through pores or directly through the cracks in the concrete (seawater, de-icing salt, chemicals). |
| **A48. Delayed Ettringite Formation, DEF** – If the curing temperature of concrete is high, the hydration of the concrete is disturbed and ettringite crystallises into its pore structure which leads to volume expansion. This results in cracking and the disintegration of the concrete. Furthermore, frost attack resistance is impaired when the pore structure fills up. | Heat-treated prefabricated concrete units and massive concrete structures with a setting temperature of more than 70°C. The cement contains tricalcium aluminate. |
| **A49. Sulphate attack on concrete** – Sulphate ions from an external source react with the cement hydration products, forming swelling compounds such as ettringite. The swelling-induced cracking further facilitates the penetration of sulphates into the concrete, as a result of which the entire structure may disintegrate. Thaumasite may also form in the sulfate attack (thaumasite form of sulfate attack, TSA), which causes expansion in the concrete and weakens its strength. | This occurs when concrete is exposed to running water (groundwater, sewage water) that contains sulphates ($SO_4^{2-}$ > 200 mg/L). The binder of concrete contains calcium hydroxide and calcium laminates. |
| **A50. Alkali-Aggregate Reaction, AAR** – Certain silica in the aggregate dissolve in the alkaline environment of the concrete, and in the chemical reactions with pore water alkalis (Na+ and K+) and hydroxyl ions, hygroscopic alkali gel is produced. This causes expansion and, eventually, cracking.<br><br>Three different alkali-aggregate reactions have been identified: the alkali-silica reaction, the alkali-silicate reaction, and the alkali-carbonate reaction. | AAR occurs when the aggregate of the concrete contains certain types of amorphic or weakly crystallised forms of silica (opal, chalcedony, flint and certain types of deformed quartz) or mix-phases of silica and silicate minerals (which may be present in granites, slates and greywacke, for example). |
| **A51. Demineralised water on concrete** – Soft water effectively dissolves the calcium hydroxide present in cement stone. When the water impermeability of concrete is poor due to a high water-cement ratio or cracking, water finds its way inside the concrete and dissolves the calcium hydroxide present in cement stone. As a result, the strength and impermeability of the concrete are impaired. | Concrete structures that come into contact with soft water (hardness ≤3°dH). |
| **A52. Acid attack on concrete** – Solutions with a low pH value dissolve the cement stone. Acidic solutions are present in nature or come into contact with concrete surfaces during operation. | Concrete surfaces inside the plant that are exposed to acidic solutions (pH < 6.5), foundations, pools. |
| **A53. Chemical attack** – Several chemical substances, such as magnesium sulphate, magnesium chloride and ammonium sulphate, are harmful to concrete. They act on the cement binder by changing its physical properties. | Underground concrete constructions when the groundwater contains dissolved chemical substances (ammonium $NH_4^+$ > 15 mg/L, magnesium Mg2+ > 300 mg/L, aggressive $CO_2$ > 15 mg/L). |
| **A54. Biological organisms** – A biological effect may arise directly when an organism penetrates inside the structure or as a result of the chemicals produced by the biological process of the organism that cause damage to the structure, such as sulphates. | Seawater constructions, humid conditions. |

| Physical degrading | Susceptible materials, regions and components |
|---|---|
| **A55. Restraint forces** – Restraint forces arise as a result of thermal movements, changes in humidity and the shrinkage of concrete. Thermal movements may arise during concrete setting or use. If the movement has been prevented and the structure is unable to accommodate the tension caused, the structure may sustain damage or crack. | Concrete structures where no provision has been made for restraint forces. |
| **A56. Stray current corrosion** – An increase in the corrosion rate caused by electric stray currents. | Underground concrete constructions and metal pipes that are exposed to electric stray currents (power transmission lines, cathodic protection in the adjacent structures). |
| **A57. Erosion** – The intensity of the wear caused by running water – erosion – depends, among other things, on the amount of erosive particles present in the water. | Constructions exposed to the effects of running water. |
| **A58. High temperature** – A high temperature causes the water present in hardened cement paste to vaporise. This has an adverse effect on the strength characteristics of concrete. At higher temperatures, calcium hydroxide breaks down and changes occur in the aggregate. A change in volume causes internal tension and spalling of the structure. The concrete surface disintegrates and presents a risk of further damage. | Damage to concrete structures caused by a fire or high temperature of >90°C. |
| **A59. Ionising radiation** – Ionising radiation induces a loss of strength and an increase in volume. The loss of strength may result from changes in the structure or from the heating effect of the radiation. | Radiation protection structures next to the reactor pressure vessel, biological shields. |
| **A60. Relaxation** – In relaxation, the stress of reinforcing steel decreases when the strain remains constant (see stress relaxation, mechanical components). | Reinforcing steels. |
| **A61. Creep** – Deformation of concrete under the influence of stress that propagates as a function of time after the initial stage. The deformation due to creep is not recoverable. | Concrete constructions under high stress and operating temperature. |
| **A62. Shrinkage** – Concrete shrinks when it dries and expands when humidity increases. The shrinkage that occurs during setting is not recoverable. | All concrete constructions. |

| Obsolescence |
|---|
| **A63. National and international regulations** – SSC fail to meet the requirements specified in national or international regulations. The non-conformities may relate to the design basis requirements, qualification, safety aspects and/or redundancy or diversity of the SSC, for example. |
| **A64. Standards** – SSC fail to meet the updated or new standards that are used as reference in the requirements concerning the design, fabrication and materials of SSC. |
| **A65. Equipment technology** – SSC no longer represent the current level of technology. New qualified technology may become available that materially improves the safety of the nuclear facility concerned. |
| **A66. Condition monitoring and maintenance technology** – The condition monitoring and maintenance of SSC no longer represents the current level of technology. New methods may become available that materially enhance the condition monitoring or maintenance of SSC. |
| **A67. Technical support** – The technical support for a SSC ends because the manufacturer or supplier goes out of business. |
| **A68. Availability of spare parts** – The availability of spare parts ends because the manufacturer of the component or other spare part manufacturers go out of business. |