NATIONAL ASSESSMENT REPORT OF FINLAND

for the Purposes of Topical Peer-Review "Ageing Management"
under the Nuclear Safety Directive 2014/87/EURATOM

Säteilyturvakeskus
Strålsäkerhetscentralen
Radiation and Nuclear Safety Authority

Radiation and Nuclear Safety Authority **Report**

Nuclear Reactor Regulation

December 29, 2017

Contents

Nuclear Reactor Regulation

December 29, 2017

Abbreviations

312	Feed water system
321	Shut down cooling system
323	Reactor core spray system
AMP	Ageing management programme
APC	Airplane crash
BWR	Boiling Water Reactor
CASS	Cast austenitic stainless steels
CFD	Computational Fluid Dynamics
EPW	Explosion pressure wave
FAC	Flow-accelerated corrosion
ggbs	Ground granulated blast furnace slag
OPC	Ordinary Portland cement
PH	Precipitation-hardened
CRDM	Control rod drive mechanism
CUF	Cumulative usage factor
DMW	Dissimilar metal weld
ECCS	Emergency core cooling system
ECP	Electrochemical Potential
ENSREG	European Nuclear Safety Regulators Group
IAEA	International Atomic Energy Agency
IASCC	Irradiation accelerated Stress Corrosion Cracking
IGSCC	Intergranular Stress Corrosion Cracking
IGALL	IAEA's International Generic Ageing Lessons Learned
IRWST	In-containment Refuelling Water Storage Tank
KTO	Periodic inspection programme
LO1	NPP unit Loviisa 1
LO2	NPP unit Loviisa 2
LOCA	Loss of coolant accident
LTO	Long time operation
MCL	Main coolant line
MWe	Mega Watt electric power
MWth	Mega Watt thermal power
NAR	National Assessment Report
NPP	Nuclear Power Plant
NSD	Nuclear Safety Directive
OL1	NPP unit Olkiluoto 1
OL2	NPP unit Olkiluoto 2
OL3	NPP unit Olkiluoto 3
PAMS	Piping and component Analysis and Monitoring System of TVO
PSR	Periodic Safety Review
PWR	Pressurized Water Reactor
R&D	Research and Development
RCPB	Reactor coolant pressure boundary
RCSL	Reactor Control, Surveillance and Limitation
RI-ISI	Risk informed in-service inspection

RL	WENRA Reference Level for Existing Reactors
RPV	Reactor pressure vessel
SAFIR	The Finnish Research Programme on Nuclear Power Plant Safety
SC	Safety class
SCC	Stress corrosion cracking
SS	Stainless steel
SSC	System, Structure or Component important to safety
SSE	Safe shut-down earthquake
STUK	Radiation and Nuclear Safety Authority
TH	Low pressure safety injection system in Loviisa
TPR	Topical Peer Review
UT	Ultrasonic testing
VVER	A type of PWR (Water Water Energetic Reactor)
VT	Visual testing
VTT	Technical Research Centre of Finland
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association
YTN	STUK’s Advisory Committee on Nuclear Safety

1 General information

1.1 Nuclear installations identification

The Loviisa Nuclear Power Plant (NPP) houses two Soviet-designed (Atomenergoexport) VVER-440/213 pressurized water reactors (PWRs), i.e., Loviisa unit 1 and Loviisa unit 2. The current capacity after some modernisations is 2 x 502 MWe. A general view of Loviisa NPP is presented in Annex 1.The Loviisa NPP units started commercial operation in 1977 (Loviisa 1) and 1981 (Loviisa 2) respectively. The plant is operated by Fortum Oyj.

The Olkiluoto NPP consists of two Boiling Water Reactors (BWRs), i.e., Olkiluoto unit 1 and Olkiluoto unit 2 and one PWR, i.e., Olkiluoto unit 3. The current capacity of Olkiluoto units 1 and 2 is after several modernisations 2 x 880 MWe. The planned capacity of Olkiluoto unit 3 is 1600 MWe. A general view of Olkiluoto NPP is presented in Annex 1. Olkiluoto 1 started commercial operations in October 1979 and Olkiluoto 2 in July 1982. The designer of the units 1 and 2 was Swedish Asea-Atom which nowadays belongs to Westinghouse.

Olkiluoto unit 3 is EPR-type (European Pressurized Reactor), designed by a consortium of Areva and Siemens . The construction license was granted by the Government in February 2005. The plant unit is currently in commissioning phase. Current schedule is such that the fuelling is expected in August 2018 and the start of commercial operation in May 2019.

Olkiluoto NPP is owned and operated by Teollisuuden Voima Oyj (TVO), a subsidiary of Pohjolan Voima Oyj.

The research reactor FIR-1 of type TRIGA Mark II is located in Otaniemi Campus area. It was purchased for research purposes and it was started in 1962. Later on it was also used for producing isotopes for industry and medical purposes. Its thermal capacity is 250 kW. The responsible organization was originally Helsinki University of Technology and later on (after 1971) VTT Technical Research Centre of Finland Ltd. This reactor is not in the scope of this NAR due to its low thermal power. Furthermore, the reactor is now permanently shut down and preparing for decommissioning.

1.2 Process to develop the national assessment report

Coordination of NAR preparation in Finland was done by the Radiation and Nuclear Safety Authority (STUK). For NAR preparation a cross-sectional working group was set up representing three different disciplines, i.e., I&C, electrical, mechanical and civil engineering, and having knowledge and experience also in the area of aging management.

The licensees were invited to supply materials for the preparation of NAR. The NAR was prepared on the basis of these documents and the contributions from the members of the working group. Most of the material was already earlier delivered to STUK but also some new material was delivered by the licensees during the NAR preparation process.

The NAR was written directly in English. The draft of NAR was subjected to commenting procedure and comments were asked from the licensees and STUK's staff members. The draft of the NAR was also presented to the STUK's Advisory Committee on Nuclear Safety (YTN), whose members had opportunity to give their comments.

The NAR is to be delivered to the ENSREG and published at the STUK's website after its finalization.

2 Overall ageing management programme requirements and implementation

2.1 National regulatory framework

STUK Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2016) stipulates that the design, construction, operation, condition monitoring and maintenance of a nuclear power plant shall provide for the ageing of systems, structures and components (SSCs) important to safety in order to ensure that they meet the design-basis requirements with necessary safety margins throughout the service life of the facility. Furthermore referring to the STUK Regulation, systematic procedures shall be in place for preventing such ageing of SSCs which may deteriorate their operability, 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.

A new regulatory guide [Guide YVL A.8 Ageing Management of a Nuclear Facility] both imposes requirements on licensees related to the management of physical ageing and obsolescence of SSCs, and presents the regulatory oversight relevant to the licensees' duties. The Guide applies to all NPP life cycle phases and all SSCs important to nuclear and radiation safety. Regulatory requirements set in the Guide aim at ensuring both short and long term operability and technological conformance of SSCs whether in service or stand-by. The key documents for regulator's review are a conceptual plan for ageing management and an ageing management programme along with a construction and operating license applications, respectively, and ageing management follow-up reports issued annually by the licensees.

2.2 International standards

The basic rules of international standards and guide lines are followed in regulating and developing ageing management of Finnish NPPs, including:

- WENRA Safety Reference Levels for Existing Reactors; Issue I: Ageing Management;
- Safety of Nuclear Power Plants: Design; IAEA No SSR-2/1;
- Safety of Nuclear Power Plants: Commissioning and Operation; IAEA No SSR-2/2;
- Ageing Management for Nuclear Power Plants; IAEA No NS-G-2.12.

The mentioned guidelines are adapted to Finnish maintenance strategies and ageing management practices to an extent which has been deemed appropriate to ensure operability of SSCs at NPPs. Their utilization in the developing process of ageing management is addressed in the following paragraphs.

2.3 Description of the overall ageing management programme

2.3.1 Scope of the overall AMP

TVO

Ageing management is coordinated by a dedicated “Age” working group which consists of representatives of all technical disciplines and relevant plants’ functions such as nuclear safety, operation, maintenance and asset management. The main duties of this group include

- processing information related to the conditions and performance of SSCs;
- keeping up a life cycle database for SSCs (recommended major modifications, replacements, repairs and overhauls within next 20 years);
- coordinating input information for investment and outage planning;
- revising documentation for plant ageing management.

A person in charge “system responsible” is appointed for each plant system. This person is familiarized with both operation and safety functions of his system in all plant operation modes. His main duty is to analyse the performance and safety margins of the system, and to commit himself on the needs for development related to system modifications or the scope and frequency of inspections and tests.

SSCs are divided into groups and a person in charge “component responsible” is appointed for each group. This person is a key expert who is familiar with function, operability requirements and maintenance of plant components he is responsible for. He keeps record of maintenance works, inspections and tests, failure trends, operation hours etc. Furthermore, the component responsible monitors operability of his component group and takes appropriate actions whenever operability is endangered in the short or long period. He collects all information relevant to ageing management of his component group and draws up a status report on a regular basis.

Both the system and the component responsible persons communicate with and provide essential information to the AGE group to ensure that the overall AMP is being implemented and continuously developed.

Identifying SSCs within the scope of overall AMP follows the SSC safety classification. Mechanical, electrical, I&C or civil system, structure or components are within overall AMP if they are classified to safety class 1, 2 or 3. In addition, some non-nuclear safety classified SSCs having an consequential risk to nuclear safety are included in the overall AMP in Finland. Such SSCs may initiate an event by their failure or they protect safety functions against internal or external threats. In order to assure the integrity or functional capability of a particular SSC within the overall AMP, the measures for this assurance depend on the SSC’s significance to nuclear safety and availability for power production. For this purpose SSCs are assigned to four maintenance categories: category 1 “keep always operable”, category 2 “limited unavailability allowed”, category 3 “economically justified preventive maintenance allowed” and category 4 “no preventive maintenance”. Probabilistic risk analyses, including such important measures as Fussell-Vesely and Birnbaum, the Operational Limits and Conditions, gathered maintenance experience etc. have been utilized when categorizing the SSCs and planning appropriate actions to maintain their operability.

The AGE working group has the key role in the quality assurance of the overall AMP. Internal audits according to the Quality Management System of the licensee are made, and if necessary, external evaluations and regulatory inspections will be arranged. The AGE working group make regular overviews of the inspection and test results and maintenance data, assesses the effectiveness of the process and then decides on remedial measures if failure trends are rising or acceptance criteria are getting close to their limits. Rising long-term failure trends are generally counted as the most unambiguous indicator that reveals possible weaknesses in the implementation of the overall AMP.

Fortum

The lifetime management group of the engineering and maintenance division is in charge of ageing management at Loviisa NPP. On the SSC level, system engineers are appointed for each plant system, and within the dedicated plant systems, their duties are

- identification and follow-up of degradation mechanism;
- management of obsolescence;
- maintaining of data systems (LOAM, POMS) for ageing management;
- organizing of condition inspections and modification works;
- preparation of investment plans;
- management of SSC qualifications;
- contacts to internal and external interest groups;
- maintaining of SSC condition classification;
- input to procurement of spare parts and spare part stock strategy;
- plant walk downs.

Ageing management covers all safety classified mechanical, electrical, I&C and civil SSCs (system, structure or component) within overall AMP if they are classified to safety class 1, 2 or 3. Also non-nuclear safety classified SSCs are included in the overall AMP if they are considered to cause an consequential risk to nuclear safety. SSCs are classified into three groups A, B and C having graded procedures and scopes of ageing management each. The SSCs that are assumed to limit the plant lifetime are counted to group A, the SSCs that are highly significant to availability or safety of the plant are counted to group B and rest of the SSCs within the overall AMP belong to group C.

Furthermore, all the NPP components are classified into four criticality classes based on their significance on nuclear safety and power production. The criticality classes are 1 (high critical), 2 (critical), 3 (non-critical) and 4 (run to failure). The classification was originally introduced to optimize maintenance tasks but now utilized in the ageing management of SSCs, too.

2.3.2 Ageing assessment

TVO

IAEA No NS-G-2.12 and the Finnish regulatory guide YVL A.8 have been the main guidance as the overall AMP has been prepared.

Plant programmes, i.e., scheduled maintenance and follow-up programmes form the technical basis of ageing assessment. The plant programmes are the leading procedures for the identification of ageing mechanisms and their possible consequences. The evaluation of their performance for example in terms of SSC failure trends indicate if applied programmes are adequate either as such or modified to manage the adverse ageing effects for SSCs.

Scheduled maintenance programmes are implemented to SSCs belonging mainly to the maintenance categories 1 to 3. The maintenance tasks within the categories are determined by SSC's importance to nuclear safety and power production. The most comprehensive scheduled maintenance in the category 1 is typically executed to SSCs within Operation Limits and Conditions of the NPP. During the scheduled maintenance, or unscheduled, too, conditions of a SSC is monitored and feedback from the maintenance works is processed for any improvements. For example, replacement periods for spare parts and consumables or periods between the scheduled inspections and tests may be reconsidered based on the feedback.

Various follow-up programmes are executed in the assessment, including

- load monitoring (mechanical components);
- loading and stress calculations (piping and supports);
- in-service inspections (mechanical components and piping);
- erosion inspections (piping);
- periodic inspections (all SSCs);
- online monitoring (mechanical components);
- functional tests (mechanical, electrical and I&C components);
- surveillance (material samples);
- condition monitoring (electrical cables);
- water chemistry monitoring (mechanical components)

Plant programmes are used both to follow up and to maintain the operability of SSCs. Ageing progresses inevitably and impairs the operability regardless of mitigating measures. When establishing acceptance criteria for progressed ageing of a SSC the original design basis is normally applied. This is because basically SSCs are always to meet all the design basis requirements which have been set to them in any applicable service conditions. Then for having evidence on complying with the requirements the follow-up programmes are addressed and used to acquire data which can be compared to the acceptance criteria.

As far use of internal and external operating experience is concerned, TVO as a member of WANO (World Association of Nuclear Operators) obtains information about events that occur at other NPPs. WANO members also exchange recommendations and best practices worldwide with one of the subjects being ageing management of SSCs. Similarly, TVO exchanges regularly information with dedicated Swedish working groups (established by Swedish NPPs) for reactor pressure vessel and internals "Reactor Group" and turbine island "Turbine and Generator Group".

On the R&D side TVO participates in the national SAFIR (Safety of nuclear power plants - Finnish National Research Programme) that focuses on various research fields partly related to the ageing of NPPs. The SAFIR research programme provides TVO with access to valuable results of several large-scale international research projects. TVO maintains also continuous cooperation with VTT (Technical Research Centre of Finland) and Finnish technical universities. When necessary TVO's own expertise can be supplemented by experts and researchers from these organizations. Westinghouse Sweden coordinates the operation of NOG (Nordisk Owner Group), which offers TVO opportunities to join Swedish research projects concerning Olkiluoto 1 and 2 type BWRs. In addition, TVO is involved in EPROOG cooperation (EPR plants Taishan, Flamanville, Hinckley Point and Olkiluoto 3).

Fortum

Ageing is assessed regularly and reported to the authority acc. to YVL A.8 requirements in each technical discipline; Mechanical, Electrical, I&C, Cables and Civil Structures. Assessment is based on the results and feedback of the main plant programmes:

- Surveillance and testing programme;
- Maintenance programme;
- In-Service Inspection programme;
- Chemistry programme.

Feedback data (eg. failure trends) from the plant programmes is analysed annually to identify possible improvements areas of the ageing management programme.

A condition classification system has been introduced for the SSCs. It is derived from the following criteria

- unavailability (not planned);

- average Obsolete Value Ranking;
- relative change in failures and refurbishments over the previous three years;
- number of proposed major overhauls without an investment decision;
- number of temporary repairs;
- loss of power production resulting from a SSC failure.

Cooperation and utilization of operational experience in the field of ageing management takes place through the plant programme for internal and external operating experience.

2.3.3 Monitoring, testing, sampling and inspection activities

TVO

TVO's programmes for monitoring, testing, sampling and inspection activities are described in the following.

Load monitoring

Pressure and temperature transients occurring in the primary and the secondary circuit are monitored and recorded. The actual transients are thoroughly assessed by experts and then classified as design transients. The design transients are simplified plant events which stress SSCs with the pressure, temperature and flow impacts that have been specified in the original plant design. Load monitoring is important because the fatigue analyses that are used to estimate the plant's design service life have been performed through the design transients and expected amount of occurrences.

Loading and stress calculations

Loading and stress calculations of piping and supports inside the containment have been performed and reported with a software PAMS. Additionally, mechanical and thermal stresses, erosion, (stress) corrosion and other relevant failure mechanisms over the plant's service life are analysed for their effects on SSCs with PAMS. Results of PAMS analyses are then utilized when inspection areas are selected for risk-informed in-service (RI-ISI) inspections of piping.

In-service inspections (mechanical components and piping OL1, OL2, OL3)

ASME Code Section XI is applied to the periodic inspection programmes of piping and components. The objects of inspection and the division of the objects into inspection categories are selected based on Sub-sections IWB-2500, IWC-2500, R-2500 and Tables IWB-2500-1, IWC-2500-1, R2500-1, and the associated model drawings. The programmes cover components and structures assigned to safety classes 1 and 2 or otherwise assessed to be significant to safety, such as pressure tanks, pumps, valves and their support structures, as well as the reactor pressure vessel internals and the flywheels of the recirculation pumps, and piping inspection objects based on risk-informed targeting (RI-ISI). Changes in new revisions of ASME Code Section XI are monitored, and any amendments and changes that are considered necessary are added in the list of inspection objects. The selection principles applied to the inspection objects, methods and intervals as well as the reporting and assessment procedures for inspection results and defect indications are described in the summary programmes which provide references to detailed inspection programmes.

Ageing management covers the piping included in the RI-ISI programme. The risk-informed inspection method refers to the utilisation of data provided by failure analyses and the probabilistic safety analysis (PRA) in the selection of the inspection object. The method is based on ASME RI-ISI method – Boiler & Pressure Vessel Code Section XI, Annex R, method B. The method shall meet the requirements laid down for it in Guide YVL E.5. The risk resulting to the plant from a pipe break consists of two factors: the break probability and the adverse consequences of a break. Periodic inspections are designed to minimise this risk by affecting the break probabilities of pipes. The consequences of breaks cannot be influenced by the inspection programme. The use of the inspection programmes ensures the inclusion of the highest-risk welds in the programme, as well as the efficient use of resources. The objective of the method is to identify the piping segments of high risk significance, and to select in these segments the welds to be included in the inspection programme.

Erosion inspections (piping OL1, OL2, OL3)

Erosion inspections are targeted at areas susceptible to erosion and pit corrosion. The inspections are targeted at predefined pipelines. Every year, an expert group draws up the erosion inspection programme for the following year based on the results of the previous inspections. In addition, piping connected to pressure equipment included in periodic inspections as well as pipelines cut off in connection with modifications are selected as objects for visual inspections, as far as possible, even if they are not part of the monitored pipelines. The inspection methods are determined based on the monitored defect type. A measurement report based on the test procedure used is drawn up, and an isometric drawing with inspection area markings is attached to the report. The inspection results are on an annual basis imported to electronic isometric drawings used for the follow-up of erosion inspections.

Periodic inspections (all SSCs)

Some periodic inspections are regulatory requirements or they are within scheduled maintenance programmes or particularly set to be consist with the high maintenance category of a SSC. These inspections cover

- periodic inspections for pressure equipment;
- periodic inspections for hoisting equipment;
- periodic inspections for electrical components;
- periodic inspections for fire protection equipment;
- periodic inspections for civil structures.

In addition, inspections are made during regular walk-downs by operation personnel according to a check list. These inspections typically include visual inspections for leakages, checks for abnormal noises, lubricant level checks and vibration measurements with a portable device.

Online monitoring (mechanical components)

Online monitoring covers SSCs for which continuous monitoring is considered necessary for predictive maintenance. At NPP units Olkiluoto 1 and 2, online vibration monitoring has been installed to reactor recirculation pumps and turbine generator sets. At the NPP unit OL3, there will be a higher amount of online condition monitoring systems such as

- loose parts monitoring system primary circuit;
- vibration monitoring systems for main coolant line, main coolant pump, main feed water and steam line;
- leakage monitoring systems inside containment;
- diagnostic of rotating machinery;
- valve monitoring system;
- fatigue monitoring system.

The continuously monitored parameters are recorded and analysed in the data processing systems. The results are used to assess condition and performance of a SSC concerned, and if major changes are observed, to schedule future maintenance activities.

Functional tests (mechanical, electrical and I&C components)

Functional tests are performed following the scope set in Operational Limits and Conditions. These tests are normally to demonstrate the operability of stand-by SSCs including emergency power supply and safety injection systems. Functional tests are also required in the maintenance works before putting the SSC back into service in order to have full confidence in the SSC's functionality after repair or part replacement.

Surveillance (material specimen)

As irradiation embrittlement is considered a major degradation mechanism of the reactor pressure vessel, radiated material specimen are used to monitor the effect of radiation on the material properties of reactor pressure vessel parts in the beltline region. The beltline base material and weld metal properties are determined through the performance of mechanical tests for both non-irradiated and irradiated test specimens for reference. Specimens are located close to the core, are exposed to a higher rate of radiation and consequently the measured material properties are expected to be conservative. See also Section 5.1.3.

Condition monitoring (electrical cables)

See Section 3.1.3.

Chemistry monitoring (mechanical components)

Chemistry monitoring is performed by means of either periodic or continuous sampling and analyses. Parameters of chemistry do not usually provide immediate information of the SSC conditions but deviations from a target value indicate that remedial measures may be needed to avoid degradation in the SSC resulting from the deviation. Chemistry monitoring covers the following SSCs and monitored parameters

- Reactor coolant system
- Nuclear Island auxiliary systems including closed cooling water systems
- Water-Steam –cycle
- Turbine Island auxiliary systems including closed cooling water systems
- Emergency diesels
- Spent Fuel Storage systems

All system's parameters are divided into two categories, Control parameters and diagnostic parameters. All parameters have their normal values and three action limit values. Also systems in Turbine Island have so called tolerable values as an extra category.

Fortum

Fortum's programmes for monitoring, testing, sampling and inspection activities are described in the following.

Surveillance and testing program

Periodic functional testing of the safety systems is performed acc. to Technical Specifications which define testing of plant systems and components, operational status of the plant unit, testing frequency and test procedures. Daily surveillance is performed by operating staff walk-downs.

Maintenance program

Most of the condition monitoring activities are included in the maintenance programmes.

- on-line and off-line vibration monitoring
- motor current spectra monitoring
- sampling and condition monitoring of lubricants/oils
- Thermography monitoring (infrared)

Allocation of condition monitoring and maintenance activities is based on maintenance criticality classification(1-4). Condition monitoring is performed only for components in criticality classes 1-3. because the functional failure of class 1-3 components has an impact on plant safety. The main purpose of the condition monitoring is to detect the degradation before any functional failure occurs.

Chemistry programme

The programme of primary chemistry control is focused on maintaining primary system integrity and to reduce radiation levels in the primary system. The programme is based on original Technical Specifications and water chemistry instructions which reflect the recommendations from fuel vendor, plant supplier and the experiences gained at similar VVER units in the world.

The main purpose of the secondary chemistry control programme is to protect the steam generators and other components from corrosion related damages. It is a plant specific programme since the secondary side at Loviisa contains still some copper parts which prohibit the application of the high AVT-chemistry (project going-on, implementation after outage 2018). Otherwise the programme reflects the current understanding of the optimized water chemistry requirements based on operating experience of other plants and plant suppliers.

Inspection programme (ISI-program)

Loviisa plant has ISI-programmes for both pipelines and components. They are both based on ASME XI requirements and are prepared for ten year periods. Pipeline condition monitoring is a part of ISI-programme and it is mainly focused on monitoring flow accelerated corrosion in the secondary circuit of the plant.

Other activities for condition monitoring

- Monitoring of loads and transients which are the basis for strength analysis
- Monitoring programme for cable ageing
- Monitoring programme for RPV irradiation embrittlement

2.3.4 Preventive and remedial actions

The licensees have programmes in place defining the condition monitoring and maintenance of SSCs including schedules for the actions to be taken.

Any need for maintenance or repair (SSC in service or stand-by) shall be reliably detectable by means of condition monitoring (on-line monitoring or periodic inspections) before degradation of the operability incurs risk to nuclear safety. Condition monitoring is typically based on visual inspections, non-destructive testing, functional tests and pressure and leak tightness tests. Condition monitoring is also considered to include actions that provide information about parameters that are related to the SSC operability or have an effect on it such as cumulative fatigue, hydrochemistry parameters and material surveillance. When a SSC is refurbished or repaired, the licensee is to investigate whether the same degradation could be found in other similar SSCs at his plant (a common cause failure). Moreover, the licensee is to investigate how the degradation could be avoided in the future by improving the condition monitoring or maintenance of the SSC.

The condition monitoring and maintenance programmes and instructions pertaining to a SSC are be based on the applicable standards, manufacturer's recommendations or the operating experience feedback received in-house or from other nuclear facilities. The aim is that these programmes and instructions are unambiguously and clearly familiarized to the operation and maintenance staff.

2.4 Review and update of the overall AMP

The licensees review and assess the effectiveness of their overall AMP on a regular basis. The input for the assessment is typically gathered from the following sources

- condition monitoring of SSCs such as in-service inspections and tests;
- scheduled and unscheduled maintenance of SSSs;
- evaluation of plant specific and others' operating experiences
- evaluation of ageing analyses that are time limited;
- R&D results when applicable;
- regulator's feedback.

Whenever the assessment indicates weaknesses or impaired performance, for example in terms of increasing failure frequency of a SSC, the AM strategy is reconsidered and modified for that particular SSC.

The regulator reviews the overall AMP and related programmes when the licensee applies for an operating license (new or license renewal). The assessment of ageing management issues is also integrated into the review of Periodic Safety Review every tenth year.

2.5 Licensee's experience of application of the overall AMP

<u>TVO</u>

Normal way to manage any technical problem is to consider some corrective actions. Idea of all TVO's AMPs is by testing or monitoring SCCs to find out deficiencies. Corrective actions have to be started if a deficiency is indicated. This is a normal action to modify periodic or preventive maintenance or change the action itself. The pipeline management programme RI-ISI, or Erosion-programme etc. is updated based on the findings.

Sometimes a more detailed AMP is seen necessary. TVO will produce some more detailed AMPs in 2018, partly due to startup of Olkiluoto 3.

<u>Fortum</u>

Continuous improvement has been done over the years in the field of ageing management in Loviisa plant.

In 2002 major re-organizing was made and ageing management roles and responsibilities were clarified at that time. The scope of AMP was also clarified. Main reason for that was the upcoming license renewal of the plant. The new license was granted in 2007 and since then the ageing management organization, processes and scope have been quite stable until 2015.

In 2015 Guide YVL A.8 was published and the scope of the AMP had to be adjusted to meet the new requirements. Some new plant level guidelines and procedures were developed to support the effective ageing management. This work is under continuous development.

During the 40 years of plant operation there has been some unexpected ageing related incidents, e.g. the most severe were two feed water pipe ruptures in 1990's. Whenever this kind of incidents have occurred the corrective actions have been taken based on the case-specific lessons learned. In practice the scope of AMP or AM-related plant programme modification have been a typical corrective action.

At the moment the scope of AMP and plant level AM-related programmes are adequate to ensure the safe and reliable operation of the power plant.

2.6 Regulatory oversight process

The regulatory oversight of ageing in Finnish NPPs focuses on review of ageing management programmes along with operating license applications and periodic safety reviews (PSRs) where the conformance to the relevant STUK Regulations and YVL Guides, including experiences in licensee's recent ageing management, is investigated. STUK's findings from other regulatory control practices are used as verification.

The periodic inspections are performed on plant site according to annual planning and they tackle both the organizational issues and technical aspects of each discipline. Within the periodic inspection programme there is one dedicated inspection, called Plant Maintenance, which exclusively concentrates on the condition monitoring and maintenance activities and ageing management. The aim of this inspection is to evaluate and verify the procedures the licensee has for ensuring reliable integrity and performance of SSCs. STUK will also assess the implementation of the ageing management programmes based on the follow-up reports prepared annually by the licensees.

An expert group dedicated to ageing management has been established within STUK to oversee how the licensees perform their duties in the ageing management of SSCs. The group, which consists of mechanical, electrical, I&C, civil and human resource experts and resident inspectors, plans and coordinates STUK's regulatory duties pertaining to the ageing issues of Finnish NPPs. One of the major tasks is to evaluate ageing management programmes and annual follow-up reports. If shortcomings are found, for example in attending to the maintenance of a SSC in the long term, the group calls the licensee for further clarifications or possible corrective actions. The group also follows up findings from other countries and evaluates their possible applicability to the ageing management of the Finnish NPPs.

2.7 Regulator's assessment of the overall ageing management programme and conclusions

A regulatory guide for ageing management, Guide YVL A.8 has been recently issued and enforced. In the Guide, there is a requirement for the licensees to draw up an ageing management programme for regulator's approval. The main topics of the ageing management programme are summarized below

- Coordination, responsibilities and duties (organizational issues of ageing management);
- Measurement of effectiveness (evaluation of the performance of ageing management);
- Utilization of operational experience and research data;
- Graded approach (when applied to ageing management);
- Data of each SSC or commodity group within licensee's ageing management (design basis, ageing mechanisms, condition monitoring and maintenance programmes, specific AMPs, TLAAs.);
- Provisions for management of obsolescence.

In the beginning of 2017 TVO and Fortum issued their updated ageing management programmes according to Guide YVL A.8. Based on the review of these programmes STUK concluded that they both still need further development to some extent. Deadline for the revised ageing management programmes was set to the end of April 2018. Regardless of deviations from the new regulatory requirements both licensees have had satisfactory ageing management approaches since the commissioning of the plant units. However, the role of comprehensive ageing management programmes will be more emphasized as the operation of the NPP units is extended beyond the original design lifetime. The design basis operability of SSCs has to be maintained even though their degradation rate may be hard to anticipate.

A generic lesson learned in Finland is that the closer nuclear power plants get to the end of their design lifetime, the more challenging it is for the licensees to start large and expensive investments to modernise or modify the NPPs.

Instead of renewing a system or a component, modernisation may be postponed or realized only partially. A postponed decision to renew for instance an I&C system or an electrical system may result in an obsolescence of systems, i.e., spare parts or technical support are no longer available. This may lead to situations where the licensee may not be able to demonstrate the safety of operations to the regulator, or as far as the scope or adequacy of demonstration is concerned, opinions may differ between the licensee and the regulator. Finland has successfully applied periodic safety reviews (PSR) for the operating NPPs. The licensees are obliged to demonstrate that the safety of the operations can be ensured and improved also during the time before the next PSR. In a similar way, they have to commit to continuous safety improvements in terms of modernization projects in order to manage both physical and technological ageing in the long term.

3 Electrical cables

3.1 Description of ageing management programmes for electrical cables

NPP units Loviisa 1 and 2:

The ageing management programme (AMP) for electrical cables in Loviisa nuclear power plant involves all the cables and cable types used in Loviisa NPP. The AMP monitors the cables conditions in different buildings and at different environments. For the AMP of cables not only the cables are included. Also the cable ways (e.g. trays), connections, joints, terminals are included as well as the structures where the cables or their ways are in contact (e.g. concrete structures). The AM is ongoing programme and the reporting to Finnish Radiation and Nuclear Safety Authority is done every four years.

NPP units Olkiluoto 1, 2 and 3:

The ageing management of electrical cables of OL1 and OL2 is defined in the general ageing management programme of TVO's nuclear facilities OL1, OL2, OL3 and KPA: Document 117279. The programme describes the ageing management process for the SCCs of NPP units Olkiluoto 1, 2 and 3 and the spent fuel storage. However the more detailed facility specific programmes are focused on SSCs important to safety. The focusing is based on the technical specifications, safety classification and PRA.

Related to the general objectives of the ageing management (such as organization, choosing bases, analyses, qualification management and handling of the results) the ageing management programme of Olkiluoto do not handle cables separately. The cable sampling programme of Olkiluoto 1 and 2 containment cables and plans for OL3 cable sampling are referred in the general programme.

3.1.1 Scope of ageing management for electrical cables

NPP units Loviisa 1 and 2:

In Loviisa NPP the scope of AMP for cables are all MV (20 kV and 6 kV), LV (<400 V) and I&C (24 VDC, 48 VDC, 220 VDC) cables. Also special cables like LOCA, mineral, thermal, coaxial and fibre optic cables as well as cable joints and terminals are taken in to account of AMP in Loviisa NPP. Priority of AMP for cables lies in the safety related cables that are in the most harsh environment and unreachable during the operation, i.e. only reachable during outages, mainly the steam generator room in the containment.

Currently the used LOCA cables in the steam generator room are the power and I&C cables from Siemens (Sienopyr) and Habia. The other safety related cables in the steam generator room are from Nokia, Acome and Thermocoax. In the steam generator room lies also cable types (MHMS-Si, Monette and SSJS) that don't have safety function but are needed for operation. Their condition is also followed with samples.

Other cable types used in Loviisa NPP are from manufactures such as Reka, Prysmian and Helkama. It is recommended to use the Halogen-Free cable types for new installations and repairs.

The safety classified cable' suitability shown by analysis and tests. If possible, the cable chosen for qualification, it is reasonable to qualify the cable type for all the safety classes and a vast range of environmental conditions. Also if the whole cable type (meaning all its cable sizes) is qualified the right cable size should be available and therefore the electrical suitability is much easier to prove for the installations. In some places it is possible to only qualify a special cable type, like LOCA -cable, so the variety can't be that large.

The recognized ageing mechanisms for cables are: high temperature, short circuit or overcurrent, voltage load, ionization and UV radiation, lack of cooling and/or ventilation, mechanical damages and inappropriate substances getting involved with cable.

The above mentioned can be discovered with visual inspecting like change in the color of the cable, crack in the cables' jacket etc. Also inspections like thermal measuring are important. Sampling is also done for the cables in the steam generator room. Arrhenius method is used when the estimated lifetime of the cable is needed.

NPP units Olkiluoto 1 and 2:

The ageing management of electrical cables in NPP units Olkiluoto 1 and 2 concentrates on the nuclear safety classified (SC2 and SC3) cables in most demanding normal operating service environments and the ageing management procedures are more rigorous for the cables which in addition are required to operate under harsh environments in design basis accident (DBA) conditions (i.e. loss of coolant accident, LOCA).

The scope for condition monitoring of electrical cables of NPP units Olkiluoto 1 and 2 is defined in the instruction 108654.

As for the in-containment cables, the scope of condition monitoring covers the safety classified (SC2 and SC3) cable types which are in use in all areas of the reactor containment buildings (lower drywell (LDW), wetwell (WW) and upper drywell (UDW)) of OL1 and OL2. The cable samples for condition monitoring tests are located in most demanding normal operating service environments, primarily in UDW, in locations where the temperature and radiation levels are the highest. Cable loops for resistance measurements are installed also in WW. At present the sample cable tests of the condition monitoring programme cover the following "in-containment grade" cable families, which include low voltage power (< 1 kV), control and instrumentation cable types:

– Lipalon cables (Liljeholmens Kabelfabrik AB / Asea Kabel AB)

– Firewall III cables (The Rockbestos Company)

– "OL1/OL2 LOCA" cables (Habia Cable AB).

Similar tests and inspections are performed also on selected outside-containment cable types, which in general are Finnish national, PVC-insulated and -sheathed, standard cable types for low voltage power (< 1 kV), control and instrumentation installations (e.g. MMJ, MMO, MCMK, MMAO).

In addition, e.g. partial discharge (PD) measurements of non-nuclear safety classified medium voltage cables (6,6 kV system) important for plant operation have been performed as part of projects and as separate cable condition monitoring campaigns to prevent service interruptions and ensure plant availability.

The ageing stressors and degradation mechanisms applicable for electrical and I&C equipment are identified in TVO's Technical Requirements and Inspection Instructions for environmental qualification. The significance of the ageing stressors and degradation mechanisms for cables is evaluated in suitability analyses of the cables. The condition monitoring activities described above concentrate on the cables' insulation and sheathing structures manufactured of organic polymer materials, which are most sensitive to thermal and radiation induced ageing degradation. The loop resistance measurements cover also the conductor material (copper).

Test results and experiences from inspections and sample cable tests are utilized in evaluating the condition of the cabling in the plants generally.

The ageing management activities have been started in 1982. Processes, procedures, methods and criteria etc. have not been very precisely documented, but apparently all significant aspects have been considered.

NPP unit Olkiluoto 3:

The conditions affecting the ageing mechanism of electrical cables are identified and evaluated in technical documents, such as Project Specifications, Qualification Specifications and Suitability Analyses. The conditions include:

- ambient temperature
- temperature rise due to conductor loading
- voltage stress in insulation
- humidity
- mechanical stress and vibration
- UV radiation
- radioactive radiation.

Special attention in this context is given to effects of radioactive radiation together with other factors, assuming that engineers and cable manufacturers have enormous knowledge and long-time experience of ageing of different cable types from other branches of industry and utilities.

In OL3, the preliminary approach to define the scope of ageing management is to include all cables feeding or controlling equipment with safety functions and thus assigned to safety class SC2 or SC3 (quantity of cables approx. 16 000 of the total of approx. 47000 cables in NI). In addition the scope of ageing management shall include cables having important role in nuclear safety functions and cables installed in exceptionally harsh environment, as well as cables considered significant for plant availability.

Cables within the scope of ageing management are treated in different categories as follows:

- 10 kV (in IEC terminology Medium Voltage) cables for RCP motor feeders
- < 1 kV (in IEC terminology Low Voltage) power and control cables
- I&C cables
- Special cables e.g. for ECI, RPVL and N16 instrumentation cables
- Optical cables
- Coaxial cables.

For ageing management purposes the cables will be treated in groups, which will be formed according to category, ambient conditions, cable type, material etc. Grouping has not been defined yet, but it is expected that approximately 50 groups will be formed.

To define which cable types shall be installed in cable deposits, the following procedure has been used:

- extract data of cables and cable routes from the KADIS database (which is the project tool for managing cable data and routing)
- identify from the Room Conditions Report all rooms (inside the containment and out-side) where Room Radioactive Dose rate is 2 mSv/h or higher (so called red rooms)
- recognize from the total cable list of NI all cables which have origin or destination in such 'red' rooms or are routed through such rooms (total approx. 3 600 cables)
- from previous selection identify cables which belong to safety class SC2 or SC3 (total approx. 1 000 cables)
- consolidate list counting quantities of distinct cable types (61 types).

3.1.2 Ageing assessment of electrical cables

NPP units Loviisa 1 and 2:

At Loviisa NPP the ageing assessment of electrical cables is based on IAEA TecDoc 1188 and document NP-T-3.6. In qualification the standards used are IEEE 308, IEE 383, IEEE 323 and IEC 60780. With the help of these standards the qualification is done by emulating the artificial ageing of the cable. Also the regulatory guides [YVL A.8] and [YVL E.7] together with maintenance instructions are followed.

Fortum has been a part and/or studied of multiple programmes with other companies like VTT (Technical Research Centre of Finland Ltd), Aalto University and Elforsk (Swedish Energy Research Centre) with programmes like KaLiFi, POSSE and SAFIR. KaLiFi programme was done with Aalto University and it handles the ageing of the MV cables, POSSE was done by VTT and it handles the Polymers. SAFIR is a large programme that has been done in collaboration by STUK, Fortum, TVO, VTT, KTM (Ministry of trade and Industry), Tekes (Finnish Funding Agency for Technology and Innovation), TKK (Helsinki University of Technology) and LUT (Lappeenranta University of Technology). The SAFIR programme handles the Nuclear Power Plant safety.

Fortum is also part of the collaboration of other nuclear plants in Scandinavia. Fortum has been a part of seminars as an organizer and as a participant. Also the other Nuclear Operators and other events in the world are followed. If the events can be somehow related to Loviisa NP the event will be studied and it will be estimated if the event can happen in Loviisa. If it can some kind of actions will follow.

Some of the results from these programmes and collaborations has been taken in use. For example with collaboration with the Swedish was found that Acome 1E cable doesn't function in LOCA as it was supposed to and also the Habia cable doesn't function when sunk under water.

NPP units Olkiluoto 1 and 2:

Currently there is no Ageing Management Programme or documented assessment on ageing of electrical cables. Preparation of AMP is underway. Generic internal instructions of ageing management apply.

Key technical documentation of NPP units Olkiluoto 1 and 2 cabling is shown in Annex 2.

TVO has participated several national R&D programmes regarding cables. The most recent re-search projects related to ageing and condition monitoring of cables are listed below:

- COMRADE (Condition Monitoring, Thermal and Radiation Degradation of Polymers Inside NPP Containments, 2016 - 2017)
- FIRED (Ageing of flame retardant cables, 2016 - 2017)
- POSSE (Study on evaluation of radiation resistance and bases of inspectability of polymers, 2016)
- KaLiFi (Developing condition evaluation and determining lifetime of intermediate voltage cables by electrical and chemical methods, 2005 - 2007).

TVO has also been active in the Swedish Nuclear Power Plant companies' co-operation group "EKG Kabelgruppen". The co-operation gives valuable experiences and research data from other plants using same type of technology.

NPP unit Olkiluoto 3:

Preparation of Ageing Management Programme for electrical cables is currently underway.

In general, the principles and recommendations of IAEA document NP T 3.6 "Assessing and Managing Cable Ageing in Nuclear Power Plants" is followed when applicable.

Standards relevant to cable ageing:

IEC 60544	Determination of the effects of ionizing radiation – Procedures for assessing ageing in service
IEC 62465	Management of ageing of electrical cabling systems
IEC 62582-3	Electrical equipment condition monitoring methods – Elongation at break

Key technical OL3 project documentation relevant to cabling are shown in Annex 3.

3.1.3 Monitoring, testing, sampling and inspection activities for electrical cables

NPP units Loviisa 1 and 2:

There are instructions in which the Loviisa NPP buildings are categorized so that for every buildings' electrical installation (incl. cables) the monitoring, testing, sampling and inspections is done periodically. Mostly the monitoring and inspections are visual, but some actions e.g. forcing a load to a cable trays might be done.

The visual monitoring and inspection cover the condition of cables and trays, joints, connections, groundings, fire protection, possible mechanical damages, inspection that the cables are on the right tray etc. Special attention must be taken for installations that are in a humid environment. Every year is done the visual inspection for the thermal protection of the cables in the steam generator room. Radiation level of the steam generator room is followed as well.

For the MV cables some extra periodical testing is also done e.g. insulation resistance, tan-delta and off-line PD-tests.

In the steam generator room there are a three cable deposits where samples are taken. In these deposits are situated 0,5 m samples of the cable types used in the steam generator room. Two of these deposits are in normal operation environment and one is in harsh conditions. Samples are taken periodically (every 4 years). The samples are sent to an outside company. The company performs tests to each sample, elongation at break and tensile strength at break.

The same year when the samples are taken from the steam generator room is also done thermal measurement for cable trays and single cables. The thermal measurement from concrete structures in the steam generator room is done every year. The results are documented and also the source of the heat should be documented.

Determining if the cable is longer valid for its use is case-specific. For obvious reasons the more important to safety or production the cable is the more important it is to keep it functional. The cable samples that go through testing where the tensile strength and the elongation at break are measured. If the elongation at break value is higher than 50% the cable is considered to be suitable for use. If the elongation at break value begins to reach the 50% limit actions must be considered/taken.

The data is gathered from the monitoring, testing, sampling and inspections and saved for a later use. Afterwards the data can be used when e.g. information about the cables history is inspected. Some data will be never used but some is valuable all the time, eg. the radiation level and the thermal measuring are very important information when following the trends of steam generator room. That kind of information can be used when new cables are qualified.

NPP units Olkiluoto 1 and 2:

Visual and tactile inspections are performed for cables, conductors, connections and terminal boxes inside the containment. The inspections will be done during maintenance outages every second year so that approximately one third of the objects in the scope of inspections are inspected during one outage. Inspection interval for each object will thus be approximately six years. The inspections concentrate on effects of harsh environment and vicinity of hot objects and signs suggesting ageing phenomenon. The results are documented and analysed by experts following decisions on further action.

There are also specific test cables (type HHSO 3x1) in the containment of each plant unit, installed for the purpose of monitoring the performance of cables. Conductor loop resistance and insulation resistance is measured and recorded once a year.

Lipalon cable sample deposits are installed in six locations of each plant unit. The Lipalon cable samples have been in the containments since the commissioning of the plants. Installed cable types are HHO 3x1, HHO 4x1, HHSO 3x1 and HHSO 4x1. Cable samples (length approx. 1 m) will be taken at five year's intervals. In addition, there are nine samples on each plant unit of Habia cable type OL1/OL2 LOCA 1x(3x1)/screen, installed year 2011 and one sample of Rockbestos Firewall III cable type RXSR-G 3×AWG16 in OL1 installed in 1992.

Samples have also been taken from cables which have been in use outside containment and dismantled for the reason of some change work.

The samples are tested in an appropriate laboratory. Tests include tensile strength, elongation at break, voltage test and insulation resistance measurements. The instruction refers to standards IEC 811 and SFS-IEC 855. The general acceptance criteria for the sample cable tests are set as follows:

- Elongation at break: > 50% absolute (general acceptance criterion). In addition, for LOCA-cables the tensile test results of the LOCA-tests programme are taken into consideration.

- Voltage tests: No dielectric breakdown allowed.

- Insulation resistance: Reference values according to cable type specific standards and/or TVO's internal instructions and electrical safety regulations.

In addition, the test results are compared with the results of the previous tests for trending and actions will be taken based on analysis by experts.

NPP unit Olkiluoto 3:

Procedures for regular inspections for cables in use have not yet been defined. Inspections will include visual and probing inspections, measurements and PD tests for 10 kV cables.

Cable sample deposits will be installed in various locations of the plant. Exact locations have not yet been defined, but the target is to find conditions respecting the actual installation conditions of the respective cable type. The cable samples may be either approx. 35 m single lengths or 10 x 3,5 m ready cut lengths. Cable ends shall be sealed in either case to prevent potential local deterioration of the insulation material.

Preliminary list of cable types to be installed in deposits:

- N2XSH 10kV 1x240 RM/35

- N2XH-J 1kV 3x2,5 RE

- HXELCHXOE 1kV 3G2,5

- JE-HCH 4x2x0,5 BD SI

- JE-LiHCH 2x2x0,5 BD SI

- NHXCHX 1kV 3x6 RE/6 FRNC-BX

- (N)HXSCHXOE-J 1kV 4x2,5 FRNC-BX

- SISIF 500V 1x16 FRN*S

- SIHGLCSI 500V 4x2x1,5 FRNC*S

- Superscreened Coax 21AWG HFI 150 – 50 Ohm

- MI Thermocable Type K

- LN Coax 24AWG 50 Ohm

- JE-LiHXCHX 2x2x0,5 BD SI FRNC-BX

- JE-LiHXCHX 1x4x0,5 BD SI FRNC-BX EUPEN
- JE-HXCHX 20x2x0,8 BD FRNC-BX
- GORE GSC-02-25198-00.

Cable samples shall be extracted from the deposits at approx. five years intervals after an initial period of ten years.

Tests to be performed for each sample:

- Voltage test for Power Cables according to relevant product standards
- Conductor loop resistance test for I&C cables
- Insulation resistance adjusted to sample length (volume resistivity), test voltage according to relevant product standards
- Tensile test with elongation at break recording according to IEC 60811 or IEC/IEEE 62582-3.

Same tests shall be performed to un-aged cable samples at the beginning of deposit period, to collect reference values of tests.

Acceptance criteria for test results:

- PD measurement: No detectable PD at $2xU_0$
- Voltage test: Passed
- Conductor loop resistance: According to relevant data sheet or product standard
- Volume resistivity: According to relevant product standards, if available, or according to IEC 60885-1 (withdrawn standard), also generic value of $\geq 10^8$ Ωm may be used as reference
- Elongation at break: 50% absolute or relative indicates the end of life; other values may be justified for specific compound materials.

Multiple samples may be applied if there is an indication of dispersion in test results. Also more conservative alert criteria will be defined to initiate preventive action in time.

The elongation at break during a tensile test is the benchmark property by which the structural integrity of the cable is assessed. The purpose of the electrical tests described above is to verify that the sample is not damaged and is in operation condition.

3.1.4 Preventive and remedial actions for electrical cables

NPP units Loviisa 1 and 2:

Should the monitoring, testing, sampling and inspections show results that the cable is or may be in bad condition preventive/remedial actions must be considered. The actions might be as follows: more frequent monitoring of the cables condition, renewing the cable or assembling a cable joint if the renewal is very hard to make. In the steam generator room the cable joints are avoided.

If there is a minor damage in the cable (e.g. minor violation of jacket) then temporary repair should be taken, e.g. securing the crack with electrical tape. Or if a cable trays connection is loose, it should be tightened. After the temporary or a little repair the damage and/or action should be reported and then make the considerations if a future actions is necessary or if not.

If the samples tested (3.1.3) show that the cable type might be valid only for a short period of time, then the action of a new cable installation should be considered. Especially if the cable is safety related and the cable is no longer available (in the storage or by the manufacturer), then must be done a qualification for some other type of cable. Also requalification of cable type might be possible.

The requalification was done for Siemens Sienopyr power and I&C cables. The qualification of these cables was coming to an end (30 a from the first installations). The cables could have been replaced to some other type but it would have needed a lot of resources and time because the new cable type must have been qualified for LOCA conditions which involves a risk that the first cable type might not be valid so that a second type must be qualified. The replacement would have also risen the radiation levels of the employers because work would have taken tens of hours in the steam generator room. And of course the replacing all the cables would have been very expensive. The Sienopyr cables seemed to be in good condition so the requalification came a subject. The ageing and LOCA tests were done to the cables and the results showed that both the power and the I&C cable can be used for over 50 years.

As an example of more bigger action done is the steam generator room cable route change which was done because of the results from inspections. It was noticed that in some places the cables were ageing faster than expected. For preventive action ventilation and cooling were improved and the cable routes were moved in a place where the temperature and/or radiation is smaller. Also some obstacles were installed between the cable routes and the heat sources.

Other example of preventive/remedial actions is the Nokia LJNSM cable renewing. The Nokia LJNSM cables outer sheath was found to be in bad condition. Because of the AMP for cables the condition of the cables was discovered early enough to perform actions. The discovery was done by visual inspections and mechanical tests. The colour of the cables outer sheath had changed from grey to dark brown and the jacket was very stiff and fragile. Notice that the LJNSMs' insulation was in good condition. Because the temperature in LO2 is lower than in LO1, the LJNSM cables were in worse condition in LO1 than in LO2. For that reason all the LJNSM cables were changed in LO1 after the discovery of the bad condition of the outer sheath.

NPP units Olkiluoto 1 and 2:

The inspection and test results are analysed by a team of electrical experts case by case. In case of tests results indicating significant degradation, sample collection interval could be shortened and/or additional test methods, including repeating some of the original qualification tests, could be taken into use. Until now ageing-related factors have not initiated major changes in the cabling. All corrective cable replacement actions will be executed following generic instructions governing plant change engineering and implementation.

NPP unit Olkiluoto 3:

If the test results indicate significant degradation, possible actions are shortening the sample collection interval or to perform full LOCA test sequence to the sample to have a more accurate prediction of remaining lifetime.

Temperature and radiation will be monitored and recorded in the plant generally and at the locations of deposits. History conditions will be verified referring to conditions specified to qualification of cables.

3.2 Licensee's experience of the application of AMPs for electrical cables

NPP units Loviisa 1 and 2:

The ageing mechanisms for electric cables have gone mostly the way it was predicted or analysed. Some cables seem to be even more healthy than it was first analysed (requalification of Siemens Sienopyr Cables, Section 3.1.4). Some cable types seem to be ageing in a way that they probably won't last the whole operation life of Loviisa. These cables have functioned more than Loviisa NPP originally planned lifetime (30 a, now 50 a). These cables are situated in steam generator room and are not safety classified, except Nokia LJNSM (SC3). The main reason for ageing seem to be the temperature. The high temperatures weakens the cables outer sheath and makes it fragile. The conductors and the insulation of the cables have mostly been in good condition.

The AMP of cables in the steam generator room was originally based mainly on maintenance work done during the outage. Cable samples were taken and analyses were made every five years. And every sixth year a more profound visual inspection was done. Despite the continuous monitoring it was discovered in middle 90's that a number of cables were not in good condition. Mainly because the room temperature was higher than specified and there were hot spots in some part of the cable routes.

For those reasons the following has been done to the AMP of cables:

- More effort has been put on the environmental condition monitoring
- More temperature sensors have been installed for continuous monitoring
- Profound manual temperature monitoring on outage with infrared point measurements and thermographic camera are taken every 4 years.

- The radiation level is measured
- The deposits where the cable samples are taken have been installed
- Visual inspections are done now every 4 years instead of 6
- Knowledge has risen where the visual inspections should be done more profoundly.

Not a single failure caused by the cable degradation in the steam generator room was recognized. Usually cable failures are caused by some other work where the cable gets mechanical damage. The examples in Section 3.1.4 show that the AMP for cables is working properly in Loviisa NPP. The discoveries of cables in bad condition has been done early enough and the actions have been taken fast enough so that the all the cables have been functional before they were changed.

NPP units Olkiluoto 1 and 2:

A considerable amount of data has accumulated from the tests and inspections performed during the past years. This material is used as reference when evaluating future actions.

TVO's experiences on the in-containment cable types of OL1 and OL2 has mainly been good and major part of these cables are known to be in good condition. Most of the defects related to the in-containment cables have been mechanical damage caused e.g. by plant modification work near the cables. The damages have normally been limited to the outer jacket/screen layers of the cables. Electrical failures of in-containment cables have been very rare. However, as part of component renewals of ELMA project (Lifetime Management of Electrical and I&C Components Inside the Containment) also renewals of in-containment cables with long-term LOCA-requirements have been ongoing since 2011 and according to the present schedule, will be completed in 2019.

Also the outside containment cables have proven to be very reliable during the operating history of OL1 and OL2. Electrical failures of cables have been rare. Most of the cable defects and failures have been mechanical damage caused by plant modification work near the cables (e.g. cabling work on existing cable routes). Cable penetrations of type MCT or consisting of firestop mortar are typical locations for this kind of defects. Ageing related cable defects in OL1 and OL2 have been limited to cases in which the design temperature of a cable has been exceeded e.g. in case of connection to a warm object (i.e. design errors).

In addition, most of the medium voltage cables (6,6 kV system) of OL1 and OL2 have been replaced between 2005 and 2013 as part of switchgear renewal projects and later because of installation defects found in the terminations with PD tests. Also the 110 kV oil-filled cables have been replaced with XLPE-insulated cables in 2008 - 2010, because of obsolete technology and availability issues concerning repair parts and lack of installation expertise.

NPP unit Olkiluoto 3:

Preparation of Ageing Management Programme for electrical cables is currently underway. No experience gathered so far of the application of AMP.

3.3 Regulator's assessment and conclusions on ageing management of electrical cables

STUK's regulatory guide YVL E.7 for electrical and I&C equipment requires that a licensee has a plan for the monitoring of the ageing of the cables inside the containment. Ageing of other cables important to safety is managed by the general ageing management programmes according to the STUK guide YVL A.8 for ageing management.

STUK has inspected the ageing management programmes including the containment cable ageing monitoring plans. Implementation of the cable ageing management programmes has been verified during the STUK inspections.

The cable ageing management processes described in the ageing management programmes have been found satisfactory but the ageing management programme of cables on NPP unit Olkiluoto 3 (under construction) is still under preparation.

The ageing management process includes that the licensee regularly evaluate the adequacy of containment cable ageing monitoring plan and the coverage and effectiveness of the whole ageing management programme.

STUK reviews the annual ageing management follow-up reports of the licensees. Reporting of the containment cable ageing monitoring plan shall be presented at least every five years in connection of the follow-up report. STUK inspectors make observations about the cable ageing as part of inspections during operation and outages of the NPP units, in connection of test, maintenance and fault repairing works of cables and relating equipment. So far the findings of STUK observations of physical cables and reviews of relating reports of faults, maintenance, repair and renewal works of them have been in congruent with the reports of the ageing management programmes.

In the early years of Loviisa power plant operation the cable aging management processes were not efficient enough to confirm the preservation of the cable condition in all cases and locations. There were unpredicted hot spots exposing cables with high thermal and radiation stresses. The utility took the corrective and preventive actions as described in section 3.2.

To the strong features of the licensees' cable ageing management programmes belongs monitoring the temperatures in containment and using detection methods to find out the hot spots. Cable sample deposits are put in representative places and sample pieces are tested regularly. Test results have led to the necessary measures such as additional qualification or cable changes. In that way the ageing management processes for the safety related cables inside the containments of Loviisa and Olkiluoto NPPs are capable to maintain the qualified stage and predict the necessary measures for requalification or cable changes.

For the safety related cables outside the containment the periodical test and inspection programmes have turned out to be effective so that no significant degradation due to ageing has been reported.

For the reasons presented above STUK considers that the ageing management programmes concerning the safety related cables of the Finnish NPPs have been adequate.

4 Concealed pipework

There is actually not much of concealed pipework in the Finnish nuclear power plant units. In the following the few cases of buried or embedded pipework are shortly explained:

NPP units Loviisa 1 and 2:

Cooling waterways in NPP units Loviisa 1 and 2 are either rock or concrete tunnels, seawater piping is located in concrete buildings or in concrete channels. Concrete channels are either in cast-in-situ structures or made of prefabricated elements.

There are some piping of TG-system embedded in concrete in connection with the pools of reactor building. TG-piping has been handled in RI-ISI aging management program and there is only a few meters of piping that is embedded in concrete.

NPP units Olkiluoto 1 and 2:

Cooling water ways in NPP units Olkiluoto 1 and 2 are mainly concrete channels, which have their own AMPs. Cooling water piping is installed either in concrete buildings or channels so that it can be inspected. There are floor drainage pipes and sewage pipes in concrete or buried underground in NPP units Olkiluoto 1 and 2. There are also some safety classified pipes, which have penetration through thick concrete structures.

Figure 4.1 shows a case, in which a structure has had small leakage in a condensation water pipe between the turbine and concrete channel to seaside. Efflorescence in the surrounding concrete structure has been identified in the ageing management program and the corresponding flooding risk has been evaluated. Further maintenance works are scheduled accordingly.

Seawater cooling system in NPP unit Olkiluoto 3 is mainly using cast-in-situ concrete channels or piping installed in rock or concrete channels. Concrete seawater channels have their own AMP FIN005-CEC-6350-690132 “In service inspection plan of cooling water structures”.

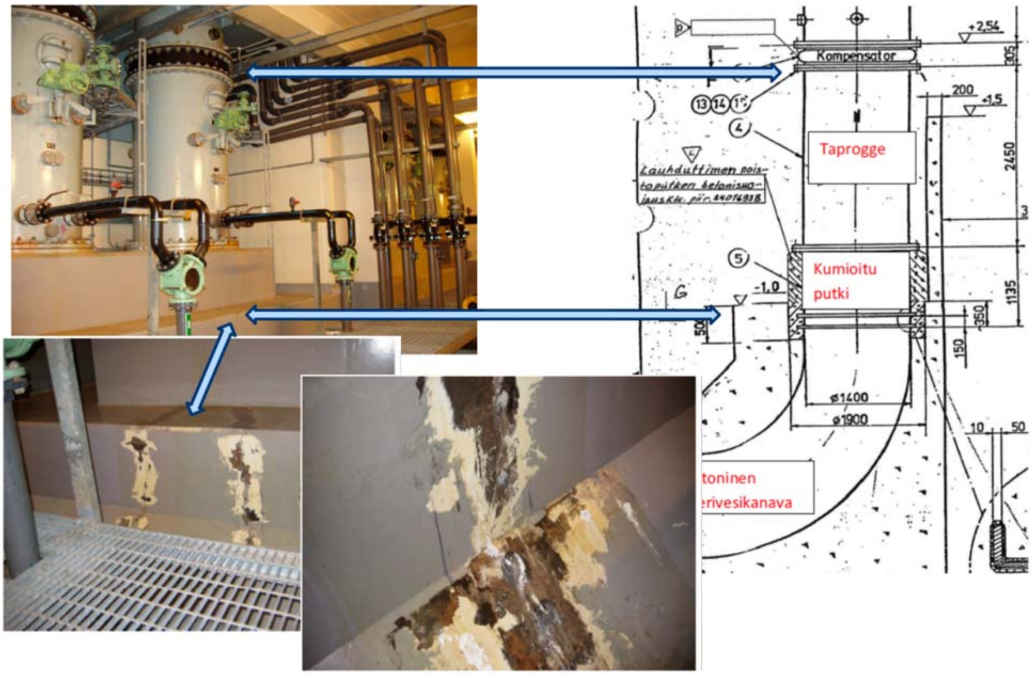


Figure 4.1 Case of flooding risk because of efflorescence.

NPP unit Olkiluoto 3

There are some safety classified piping inside concrete structures in NPP unit Olkiluoto 3.

In the concrete basemat of the Olkiluoto 3 containment inside the steel liner the stainless steel process pipes of IRWST tank (In-containment Refuelling Water Storage Tank) are inside of a sleeve pipe. The sleeve pipe forms a part of the liner. In order to allow the movement between the liner and the concrete basemat a stainless steel compensator was installed inside the liner. Outside the liner these straight IRWST process pipes (length 7 m) can be inspected inside of the pipe.

There are floor drainage pipes and sewage pipes in concrete or concealed underground in NPP unit Olkiluoto 3. Stainless steel drainage pipes of the fuel pools are inside thick concrete structures.

5 Reactor pressure vessels

5.1 Description of ageing management programmes for RPVs

The ageing management of NPP units Loviisa 1 and 2 RPVs is based on Special ageing management program [LO1-K822-00044 version 2.0].

The ageing management of NPP units Olkiluoto 1, 2 and 3 RPVs is defined in the general ageing management programme for TVO's nuclear facilities [OL1, OL2, OL3 and KPA: Document 117279]. The programme describes the ageing management process for the SSCs of TVO's NPP units Olkiluoto 1, 2 and 3 and the spent fuel storage as well. The scope of this document is quite wide and therefore it is also quite general including i.a. organizational aspects. However, it gives the general ageing management principles that are followed at Olkiluoto site.

5.1.1 Scope of ageing management for RPVs

It is common knowledge that achieving the optimal service design life of a SCC requires that it is operated within its allowable operating limits. The ageing of nuclear power plant SSCs that are important to plant safety and availability must be effectively managed to ensure that their respective design functions are maintained throughout the service life of the plant. This process involves the prediction and detection of equipment degradation when required safety margins are threatened, as well as the initiation of subsequent corrective or mitigation actions.

RPV, a SC1 component, forms a part of the RCPB and is within the scope of this AMP evaluated with respect to ageing degradation effects and their consequences. All pressure-retaining and surface welds are included in this report.

Ageing is a process by which the physical characteristics of a SCC change with time when subjected to a specific environment. Ageing degradation may proceed through the progression of a single ageing mechanism, or a combination of several ageing mechanisms. The ageing of materials in a nuclear power plant may lead to functional degradation of plant equipment including also the RPV.

The pressure vessel material must retain its fracture resistance even when the material is aging.

NPP units Loviisa 1 and 2

The RPVs have been ordered without any material specification according to the current requirements, so accurate material information and its impact on the performance characteristics of the pressure vessel has only been evaluated by analyses performed later.

The RPVs and associated primary nozzles are made of low alloy high-temperature Cr-Mo-V steel, 15X2MΦA. The old marking for this steel has been 12X2MΦA. The material of the RPV cover is high-temperature steel 20X2MΦA. The RPV flange is made of steel 25X2MΦA. The pressure vessel is mainly welded by submerged arc welding using filler material C10XMΦT and powder AH-42. Schematic drawings of LO1 and LO2 RPV and cover are shown in Annex 4. The compositions of base materials and welding additives are summarized in Table 5.1.

Table 5.1. Chemical composition of the main Loviisa RPV steels and welding filler metal

	C	Si	Mn	S	P	Cr	Ni	Mo	V	As	Co	Cu
15X2MΦA vessel, nozzles	≥0.13 ≤0.18	≥0.17 ≤0.37	≥0.30 ≤0.60	≤0.025	≤0.025	≥2.50 ≤3.3	≤0.40	≥0.60 ≤0.80	≥0.25 ≤0.35	≤0.08	≤0.025	≤0.3

20X2MΦA top	≥0.16 ≤0.21	≥0.17 ≤0.37	≥0.30 ≤0.60	≤0.025	≤0.025	≥2.50 ≤3.0	≤0.40	≥0.60 ≤0.80	≥0.25 ≤0.35	≤0.08	≤0.025	≤0.3
25X2MΦA top flange	≥0.22 ≤0.27	≥0.17 ≤0.37	≥0.30 ≤0.60	≤0.025	≤0.025	≥2.80 ≤3.30	≤0.40	≥0.60 ≤0.80	≥0.25 ≤0.35	≤0.08	≤0.025	≤0.3
b10XMΦT weld	≤0.11	≥0.35 ≤0.45	≥0.55 ≤1.30	≤0.03	≤0.03	≥1.1 ≤1.6	-	≥0.40 ≤0.60	≥0.15 ≤0.30	-	-	-

The pressure vessel consists of 7 cylindrical forgings. The RPV cover and bottom are made of two pieces that are welded to each other by an electro slag welding using welding filler material C13X2MTΦ and powder 0Φ-6.

To prevent corrosion caused by primary water the inner surface of the pressure vessel is coated with an austenitic steel cladding. The inner surface and partly the outer surface of the cover are also coated.

The first layer is welded by a powder coarse method using the so-called "superimposed" additive ribbon C07X25H13 and powder 0Φ-10. The second layer was made by the same method using the additive tape C07X19H10.

The welding parameters of the second coating layer are chosen so that the heat production normalizes the first coating layer, thereby improving the properties of the coating interface. The total thickness of the coating layer is 8-10 mm. The coating is welded according to welding standard NPS OP-1513-70.

For Loviisa 1 the copper concentrations of the weld and base materials are 0.18 % and 0.15 % and the phosphorus concentrations 0.030 % and 0.015 %, respectively.

For Loviisa 2 the figures are slightly smaller: 0.10 % and 0.13 % for copper and 0.025 % and 0.012 % for phosphorus, respectively.

There are 17 nozzles in the cylindrical body of the RPV. In the RPV cover, there are 37 penetrations for the CRDMs and 18 penetrations for instrumentation. The CRDM nozzles are welded to the cover from inside. So there is a crevice opening to outside between the nozzle and the cover. The instrumentation nozzles are welded to the outside coating. The instrumentation nozzles are made of ferritic (22K) forgings. The CRDM nozzles are made of pipe (CT20) and forged flange (22K). Inside the nozzles there are protective shield pipes made of austenitic stainless steel.

The nozzles of the main coolant lines are made of forgings that have been machined to the final shape. The main parts are interconnected by submerged arc welding method according to the welding standard NPS OP-1513-70.

NPP units Olkiluoto 1 and 2

The RPV is comprised of a cylindrical shell with a welded bottom head and a removable top head. The shell and top head are connected with flanges. The main process and auxiliary pipe nozzles are welded into the cylindrical shell. The control rod drive nozzles and main circulation pump nozzles are located in the bottom head. The RPV is supported by a support skirt that is welded to the cylindrical shell. The main internals of BWR RPVs include: steam dryer, steam separators, core shroud, core spray piping, fuel assemblies, control rods, control rod guide tubes and nozzles as well as the main recirculation pumps. The RPVs have been fabricated by ASEA Atom, which later became ABB Atom and presently it is Westinghouse. The height of a BWR RPV is approximately 19600 mm, the outer diameter 5660 mm, and the wall thickness is approximately 140 mm.

The most important changes relating to the working conditions of the RPV are the power upratings. The most important power uprating was implemented in 1997.

The operation of the RPV is deliberately continued beyond the original design lifetime. In this context also the transients relating to the fatigue calculations are re-evaluated.

Both of the above mentioned changes increase the lifetime neutron fluence. This has been taken into consideration in the surveillance programmes.

Support structures welded directly to external surface of the RPV are not in the scope of this AMP.

The pressure-retaining RPV subcomponents within the scope of this AMP include the lower and upper head, core shell, flange and nozzles.

A schematic drawing of the RPVs of the NPP units Olkiluoto 1 and 2 is shown in Annex 5 and a summary of its subcomponents' materials is shown in Table 5.2.

Table 5.2. Chemical composition of the main RPV steels and filler materials at Olkiluoto 1 and 2 [Nevander, O].

Reactor pressure vessel plate, of ASTM A533 Grade B, Class 1												
	C	Si	Mn	P	S	Cr	Ni	Mo	Cu	Co	Sn	
Max.	0.20	0.30	1.50	0.015	0.02	0.30	0.70	0.57	0.15	0.05	0.05	
Min.	-	0.15	1.20	-	-	-	0.45	0.47	-	-	-	
Flange and nozzle forgings, material equivalent to ASTM A508-69, Class 2												
	C	Si	Mn	P	S	Cr	Ni	Mo	Cu	V	Sn	Co
Max.	0.25	0.35	0.80	0.025	0.025	0.45	0.90	0.70	0.25	0.05	0.05	0.05
Min.	0.18	0.15	0.50	-	-	0.25	0.50	0.55	-	-	-	-
Nozzles in bottom end, of a modified specification of material to Swedish Standard SIS 2103												
	C	Si	Mn	P	S	Cr	Ni	Cu	Co	N		
Max.	0.16	0.50	1.60	0.025	0.025	0.25	0.025	0.20	0.05	0.009		
Min.	-	0.15	-	-	-	-	-	-	-	-		

The stainless steel cladding at the inside surface of the RPV cylindrical shell and the cover is 3 mm thick at least. The weld type is of 18/8 including not more than 0.06 % Carbon, 0.08 % Nitrogen and 0.05 % Cobalt.

The cladding at the inside surface of the bottom is 3 mm thick at least and the material is Inconel 182.

NPP unit Olkiluoto 3

The supports welded directly to external surfaces of the RPV are not in the scope.

The pressure-retaining RPV subcomponents within the scope of this AMP include the lower and upper head, core shell, transition ring, nozzles, flanges, and safe-ends.

A schematic drawing of the OL3 RPV is shown in Annex 6 and a summary of its subcomponents' materials is shown in Annex 7.

AMP evaluation boundaries for the RPV extend to their connection to the Main Coolant Lines and the CRDMs. The MCLs are connected to the RPV at the inlet and outlet nozzle safe-ends. The evaluation boundary for this AMP report ends at the RPV inlet and outlet nozzle safe-ends where they are connected to the MCL piping. The weld connecting RPV inlet/outlet nozzle safe-end and MCL piping is handled within the MCL SAMP.

The CRDMs are connected to the RPV at the CRDM Adaptor Flanges. The evaluation boundary for this AMP report ends at the CRDM Adaptor Flanges where they are connected to the CRDMs.

Only low-alloy steel, stainless steel and Ni-base alloys have been used to manufacture the RPV subcomponents. Therefore, only ageing mechanisms which are relevant for such materials are considered. Furthermore, only ageing mechanisms that could occur in primary water in combination with neutron fluence have to be evaluated. In Table 5.3 ageing mechanisms that are excluded in advance, are listed.

Table 5.3. Ageing mechanisms that are excluded in advance

Ageing Mechanisms	**Susceptible Material**	**Reason for Exclusion**
Caustic Corrosion	Carbon Steel, Low-Alloy Steel	Not in contact with primary water
Flow-Assisted Corrosion	Carbon Steel, Low-Alloy Steel	Not in contact with primary water
Denting	All	Only relevant for the Steam Generator tubes
Caustic Stress Corrosion Cracking	Carbon Steel, Low-Alloy Steel, Austenitic Stainless Steel	Caustic SCC is no longer a relevant ageing mechanism for PWRs
Outer Diameter Stress Corrosion Cracking	All	Only relevant for the Steam Generator tubes
Strain-Induced Corrosion Cracking	Carbon Steel, Low-Alloy Steel	Not in contact with primary water

5.1.2 Ageing assessment of RPVs

NPP units Loviisa 1 and 2

Ageing management of LO1 and LO2 RPVs is based on Special ageing management programme [LO1-K822-00044 version 2.0].

The original design lifetime of Loviisa RPVs was 40 years and now it is 50 years when the new licensing period has been granted.

The most important degradation mechanisms and critical areas of the RPV have been identified and listed in the table 5.4. Identification has been done by licensee's experts or expert panels and is based on internal and external operating experience. As the irradiation embrittlement of the core weld is the dominating degradation mechanism of RPV the corporate level R&D programmes have focused remarkable efforts on this issue.

Table 5.4 Critical areas and degradation mechanisms of Loviisa RPV

RPV section	Critical area	degradation mechanism
RPV-head	Penetration nozzles (instrumentation and CRDM nozzles)	Fatigue Fatigue of nozzle sleeve welds (effect: sleeve deformation) SCC of welds
	Outer surface	Boric Acid Corrosion
Main flange	Sealing surfaces	Boric Acid corrosion Stress corrosion cracking Fatigue of sealing grooves
	Threaded holes for main studs	Fatigue Corrosion Wearing of threads
Vessel	Vessel body	**Irradiation embrittlement (core area and weld)** Wearing of guiding surfaces
	Cladding	Growth of existing flaws
	Nozzles	Fatigue

The only important degradation mechanism of the RPV head is corrosion caused by primary water containing boric acid. This phenomenon is possible only in the crevice between the cover and the nozzle. This is possible only in case there is a leak in pipes that are located above the cover like CRDM nozzles.

Loosing leak tightness of the sealing surfaces of the CRDM and instrumentation flanges may lead to corrosion in ferritic material of the nozzles. The head may also corrode if the leak is large. SCC of austenitic material could also happen in some cases.

A protective corrosion sleeve is welded to the nozzle at the upper and lower ends. Due to different thermal expansion coefficients there exist cyclic thermal stresses in the construction. This may lead to loss of leak tightness of the welds due to thermal fatigue.

The threaded holes of main flange are in good condition. The threads are loaded by normal tightening and also by potential impurities in the holes. It is important to keep the threads in good condition because it is not easy to repair them.

The transition temperature shift due to radiation and the resulting change in the material fracture toughness are monitored by a surveillance programme. The radiation shift is influenced by the copper and phosphorus concentrations. The critical area is the core area weld 4.

As soon as the inspection programme was started, analyses of irradiation samples showed that the embrittlement rate of the RPV was higher than expected. The problem is crucial because the design of the VVER-440 RPV is narrower than most RPVs in western reactors. In addition, a disadvantageous (high Cu- and P- content) welded seam is located in the core area.

Since the embrittlement of the Loviisa 1 RPV was able to proceed too long before reduction of the core, Loviisa 1 RPV had to be heat treated in 1996.

The radial guides are welded to the cladding by fillet welds. Fatigue in these welds is an identified mechanism relating to ageing management. The risk for fatigue is greater at LO1 RPV with sharper welds compared to LO2 RPV where the welds have been grinded smoother.

Another identified ageing mechanism is wearing of the radial guide surfaces caused by movements of internals during operation and outages.

The integrity of the cladding especially in the core area has an important role in preventing brittle fracture.

The primary and ECCS nozzles are complicated structures with bimetallic welds. Fatigue is their most important identified failure mechanism. However, fatigue analyses show acceptable fatigue even for 50 years of service life.

NPP units Olkiluoto 1 and 2

The fatigue analyses of the RPVs of NPP units Olkiluoto 1 and 2 have been updated for 60 years of service life. At present, there are no known ageing mechanisms that could limit the technical service life of the RPVs after reaching the aforementioned planned lifetime.

The [IAEA Safety Standards Series No. NS-G-2.12] principles have been used in preparing the aging management programme. This programme is evaluated annually and updated as necessary.

The RPVs who have limited lifetime will need to be controlled by analyses or tests. As an example, load analyses are to be mentioned, which must be updated at the latest in LTO phase or if the regulations otherwise so require.

Based on operational experience the most important ageing mechanism at BWR plants is stress corrosion cracking (SCC). The material Inconel 182 and also some of the austenitic stainless steels used in the RPV require monitoring of SCC.

The most important subcomponents on SCC point of view are the nozzles of systems 312 (feed water system), 321 (shut-down cooling system) and 323 (core spray system). Nowadays the in-service inspections are made at 2 year intervals. The material Inconel 182 which is prone to SCC is also used e.g. in the lower nozzles of the RPV. However, the risk is here lower because of heat treatment.

NPP unit Olkiluoto 3

The ageing management of NPP unit Olkiluoto 3 RPV is based on Special ageing management programme [FGF NTCM-G/2009/en/1013 rev. B].

The fatigue analyses of the RPV is 60 years of service life. At present, there are no known ageing mechanisms that could limit the technical service life of the RPV after reaching the aforementioned planned lifetime.

Table 5.5 contains a summary of the ageing mechanisms that have been determined to require further evaluation. Table 5.5 also includes a list of the subcomponents that may be affected by this ageing mechanism.

Table 5.5. Ageing mechanisms that require further evaluation for in-scope subcomponents

Relevant Ageing Mechanisms	**Susceptible Material**	**Affected Subcomponents**
Thermal Ageing	Ni-Base Alloy	All DMWs
Neutron-Induced Ageing (Irradiation Embrittlement)	Low-Alloy Steel with Austenitic Stainless Steel Cladding	Core Beltline Region (Core Shell)
Ageing Under Cyclic or Transient Loading (Fatigue)	All Materials	All RPV Subcomponents

Boric Acid Corrosion	Low-Alloy Steel	External Surfaces
Pitting Corrosion	Austenitic Stainless Steel Low-Alloy Steel	All Austenitic Stainless Steel Subcomponents, External Surfaces
Intergranular Corrosion	Austenitic Stainless Steel	Vent Pipe, Vent Pipe Connection, Vent Valves
Crevice Corrosion	Austenitic Stainless Steel Low-Alloy Steel	All In-Scope Subcomponents
Intergranular Stress Corrosion Cracking	Austenitic Stainless Steel	Vent Pipe, Vent Pipe Connection, Vent Valves, Cold worked sub-components (Vent Pipe)
Primary Water Stress Corrosion Cracking	Ni-base Alloy	Adaptor tubes (both CRDM and Instrumentation), Vent pipe penetration, Dome thermocouple penetration, DMWs
Trans granular Stress Corrosion Cracking	Austenitic Stainless Steel	All Austenitic Stainless Steel Subcomponents
Environmentally-Assisted Fatigue	All Materials	All subcomponents in contact with medium

The information in Table 5.5 is addressed under, where each ageing mechanism undergoes evaluation to determine whether it is adequately managed by existing plant-specific Ageing Management activities and measures during the 60 years service period.

1. Thermal Ageing

Thermal Ageing has been identified as a relevant ageing mechanism for the OL3 DMWs between:

- radial guide to transition ring
- inlet and outlet nozzles and the nozzle safe-ends
- adaptors to upper head and instrumentation
- socket for vent pipe to upper head
- thermocouple penetration to upper head

The influence is limited to the DMWs between

- the ferritic RPV nozzles and austenitic stainless steel safe-end
- components of low-alloy base material with nickel-base alloy buttering
- nickel-base components
- components of low-alloy base material with austenitic stainless steel cladding
- nickel-base alloy components
- austenitic stainless steel components.

The initial assessment of RPV DMW thermal ageing susceptibility resulted in the conclusion that the only area that could be susceptible to thermal ageing during plant operation is the Carbon-depleted zone located along the HAZ fusion line of the low-alloy steel. This zone is induced by the diffusion of Carbon from the low-alloy steel base material to the weld metal, and formed during stress relieving heat treatments performed during component assembly. The adverse effect on the mechanical properties is related to the duration of heat treatment.

Thermal Ageing Surveillance Programme exists to evaluate the thermal ageing of the RPV DMWs. The programme is targeted toward providing projected material property results in the end-of-life condition, including the tearing resistance at 300°C and the equivalent RTNDT of the RPV nozzle to safe-end DMWs. Through this programme, the thermal ageing of the DMWs between the safe-ends and the RPV nozzles is evaluated to predict material property evolution.

This Thermal Ageing Surveillance Programme of RPV Dissimilar Metal Welds implemented to identify degraded DMWs during plant operation. If degradation is identified for the RPV nozzles and the nozzle safe-ends DMW, targeted inspections or testing should be performed for all other DMWs.

With the performance of this Thermal Ageing Surveillance Programme, it can be concluded that Thermal Ageing is adequately managed for in-scope components and subcomponents during the 60 years service period.

2. Neutron-Induced Ageing (Irradiation Embrittlement)

Irradiation Embrittlement has been identified as a relevant ageing mechanism for OL3 RPV. RPV subcomponents subjected to the highest amount of significant fast neutron exposure are the two core shells (item 4 of in Annex 7) and other adjacent subcomponents, such as transition ring (item 2 in Annex 7), flange-integrated nozzle shell (item 5 in Annex 7), and the corresponding welded joints and austenitic cladding.

Irradiation embrittlement is negligible for the safe-ends, CRDMs, CRDM instrumentation adaptor nozzles, vent pipe nozzles, and Nickel-based alloys used for the radial guides and welding filler material. In these cases, the end-of-life fluence is too low to introduce any change in mechanical properties or microstructure.

An irradiation surveillance is performed on OL3 RPV subcomponents which constitutes the core beltline region. Direct measurements after a representative irradiation of ferritic base material and deposited weld metal are determined through the use of a surveillance programme in order to verify the conservatism of the predicted material embrittlement (through calculation of RTNDT values).

With the performance of Monitoring of Irradiation Embrittlement for Core Beltline Region, it can be concluded that Irradiation Embrittlement is adequately managed for in-scope components and subcomponents during the 60 years service period.

3. Ageing Under Cyclic or Transient Loading (Fatigue)

RPV subcomponents are subject to Ageing under Cyclic or Transient Loading (Fatigue).

In-scope components and subcomponents were designed for a defined set of transient conditions. Existing Thermal and Mechanical Fatigue analyses are safety analyses that use time-limited assumptions, which are subject to revalidation when the original design life of a particular component is to be exceeded and/or real operational loads significantly differ from specified design loads.

Fatigue Monitoring System instrumentation is installed on inlet and outlet nozzles including the safe-end (i.e., mixing zone of cold and hot water) in order to capture the fatigue relevant thermal loads. The application of an appropriate fatigue monitoring approach constitutes the prerequisite to the determination of the fatigue relevant operational thermal loads. Using this kind of instrumentation and monitoring of operating parameters (i.e., temperature, pressure), it is possible to validate that the design assumptions are met during plant operation, also allowing indication that load transients not anticipated during the design stage have occurred or real loads are less severe and less frequent as design loads .

According to ASME XI inspections (VT, UT) are performed during ISI on susceptible in-scope subcomponents. This will also help to detect indications due to fatigue damage.

Therefore, although it is not expected that fatigue of in-scope RPV subcomponents will be a factor within the service life of the plant, this ageing mechanism has been classified as a relevant degradation mechanism for RPV subcomponents within the scope of SAMP. Actions should be taken to revisit this ageing mechanism if adverse results are obtained during monitoring, inspection or testing activities.

4. Boric Acid Corrosion (BAC)

A considerable loss of material caused by BAC would occur only in the case of accidental leakage resulting in the long-term presence of concentrated liquid Boric Acid at elevated temperatures on external carbon or low-alloy steel surfaces in an environment that contains oxygen. Relevant in-scope areas include external surfaces of the RPV composed of low-alloy ferritic steel (16 MND 5).

The long-term presence of Boric Acid on such surfaces is mitigated for RPV subcomponents through the use of a leakage detection system in the Reactor Building .

Annual visual inspections are also performed on external surfaces of accessible components in the primary circuit and adjacent systems prior to plant start-up following an outage, aiming to detect any Boric Acid deposits.

It can be concluded that Boric Acid Corrosion is adequately managed for in-scope components and subcomponents during the 60 years service period, and no further actions are necessary.

5. Pitting Corrosion

Pitting Corrosion is relevant for low-alloy steel with a passivating layer in high-temperature water with increased dissolved Oxygen. For austenitic stainless steel, Chloride-induced pitting is a relevant ageing mechanism. Critical conditions depend on local temperatures, the pH of the medium, and the concentration of Chlorides.

Pitting corrosion is mitigated in the primary circuit at OL3 because the main coolant has a sufficiently low corrosion potential. For internal surfaces made of austenitic stainless steel, the presence of high Chloride concentrations is mitigated by implementation of the OL3 Water Chemistry Monitoring. As described in the [Chemistry handbook part V], the Chloride concentration during steady-state normal operation is below 0.01 mg/kg, which is far below the Cl- concentration where Pitting is likely to occur.

Critical conditions that could promote Pitting Corrosion of external low-alloy steel surfaces could exist if leakage were to occur continuously over a long period of time. However, the visible effect of this incident is the presence of Boric Acid deposits. When Boric Acid deposits are present, the leading ageing mechanism is Boric Acid Corrosion. The activities or measures manage the occurrence of Boric Acid Corrosion during normal operation. Therefore, the occurrence of Pitting Corrosion is covered in this respect by these measures as well.

A significant amount of Pitting Corrosion of external austenitic stainless steel surfaces would only take place in the presence of external Chloride sources. These sources are mitigated at OL3 by the prohibition of Chloride-containing tapes, markings, fluids, etc.

So Pitting Corrosion is adequately managed for in-scope subcomponents and components through implementation of the Water Chemistry Monitoring, the use of Chloride-free chemicals and material.

6. Intergranular Corrosion

Intergranular Corrosion is a relevant ageing mechanism for the austenitic stainless steel vent pipe, vent pipe connection and vent valves.

Oxidizing conditions are mitigated at OL3 through the application of the Water Chemistry Programme. In addition, the application of Chloride sources is mitigated at OL3 through the use of chemicals or materials that have a low Chloride content.

So Intergranular Corrosion is adequately managed for auxiliary RPV in-scope components and subcomponents through implementation of the Water Chemistry Monitoring and the use of Chloride-free chemicals and materials.

7. Crevice Corrosion

Crevice Corrosion has been identified as a relevant ageing mechanism for RPV. Crevice Corrosion is not an independent corrosion mechanism, but the enhancement of common corrosion mechanisms (e.g., General Corrosion, Pitting Corrosion) in special crevice conditions.

External surfaces of the RPV made of low-alloy steel are only susceptible to General Corrosion in outage conditions when components or subcomponents are subject to the presence of moisture in local atmospheric conditions. In crevice conditions, this ageing mechanism can be enhanced so that it could also be relevant during normal outage periods. However, in-scope components and subcomponents are located in the Reactor Building, where the air is controlled in terms of humidity, temperature and purity.

Critical conditions that could promote Crevice Corrosion of external RPV low-alloy steel surfaces could exist if leakage were to occur continuously over a long period of time. However, the visible effect of this incident is the presence of Boric Acid deposits. When Boric Acid deposits are present, the leading ageing mechanism is Boric Acid Corrosion. The activities or measures stated earlier manage the occurrence of Boric Acid Corrosion during normal operation. Therefore, the occurrence of Crevice Corrosion is covered in this respect by those measures as well.

Leakage monitoring is implemented at OL3 to mitigate the occurrence of long-term leakage, serving to mitigate Crevice Corrosion on external surfaces. System Plant Walkdowns of external surfaces for all accessible components are also performed every year following an outage prior to plant start-up.

With respect to Crevice Corrosion of internal surfaces, the presence of high Chloride concentrations is generally mitigated by implementation of the Water Chemistry Monitoring. However, austenitic stainless steel may become sensitive to Crevice Corrosion when impurities (e.g., Chlorides) accumulate in the crevice during operation. This can lead to an increase in susceptibility to pitting or general attack, when the pH in the crevice changes. As water chemistry is monitored, Crevice Corrosion is not generally of concern for in-scope components and subcomponents.

So Crevice Corrosion is adequately managed for RPV in-scope components and subcomponents through implementation of the Water Chemistry Monitoring, Leakage Monitoring and the use of Chloride-free chemicals and materials.

8. Intergranular Stress Corrosion Cracking (IGSCC)

<u>Thermal Sensitization</u>

IGSCC does not affect components in the PWR primary and secondary circuit during normal operating conditions due to the low dissolved Oxygen content in the water. The water chemistry at OL3 is controlled in certain systems during normal operation, with a limited Oxygen content. However, Oxygen ingress could occur during outage or start-up conditions.

Visual Examination VT-1 will be performed on the ligaments between control rod penetrations of closure head. With this VT-1 examination, discontinuities and imperfections on the surfaces of the auxiliary subcomponents can be detected. If degradation is identified on the surface, targeted inspections or testing should be performed for the auxiliary subcomponents. Therefore, it can be concluded that VT-1 examination on the ligaments between control rod penetrations of closure head covers monitoring of:

- Vent pipe,
- Vent pipe connection, and
- Vent valves.

Once the scope and type of degradation has been identified, it should be determined if periodic inspections should be implemented within the ISI Plan to monitor this issue during the 60 years service period in susceptible areas (i.e., in the case that degradation due to ageing has occurred). Furthermore, VT-2 and System Plant Walk-downs will be performed after each outage, so that leakage will be detected.

<u>Cold Working</u>

IGSCC can occur in low ECP conditions in the event that a material has been subject to severe cold working. It is possible that the vent pipe within the scope of this SAMP report that is composed of austenitic stainless steel has been cold worked. As the severity of this effect is not currently known, these areas require further consideration or monitoring during the 60 years service period.

Most in-scope components and subcomponents are periodically inspected through walk-downs after each outage before plant start-up, and are also continuously monitored during operation using the Leakage Monitoring System.

So, IGSCC (due to cold working) is adequately managed for in-scope austenitic stainless steel RPV subcomponents and components, through the implementation of activities and measures that are in place at OL3 such as the Water Chemistry Monitoring, VT-1, VT-2, Leakage Monitoring System and System Plant Walk-downs performed following each outage.

9. Primary Water Stress Corrosion Cracking (PWSCC)

A long-term damage of PWSCC cannot be excluded on any of the today accepted materials in the case of adverse operation conditions. Therefore, PWSCC is a relevant ageing mechanism if a susceptible material condition exists (i.e., chemical composition and residual stresses) in contact with PWR main coolant.

Ultrasonic tests will be performed on the following components:

- Dissimilar metal weld (DMW) between radial guide to transition ring
- DMWs between the inlet and outlet nozzles and the nozzle safe-ends
- DMWs between adaptor flanges to sleeve weld and Instrumentation

The ultrasonic test on the welds includes also the testing of the HAZ. The welds and the HAZ are more susceptible to degradation through stress. If degradation is identified for the welds, targeted inspections or testing should be performed for all other welds and for the tubes. Therefore, it can be concluded that the ultrasonic inspection on the welds and their HAZ also covers monitoring of the corresponding tubes.

Visual Examination VT-1 will be performed on the ligaments between control rod penetrations of closure head. With this VT-1 examination, discontinuities and imperfections on the surface of components can be detected. If degradation is identified on the surface, targeted inspections or testing should be performed for the RPV penetrations. Therefore, it can be concluded that VT-1 examination on the ligaments between control rod penetrations of closure head covers monitoring of:

– Vent pipe penetration
– Dome thermocouple penetration
– Weld between thermocouple penetration to dome thermocouple guide
– DMWs between adaptor tubes to upper head and Instrumentation
– DMW between socket for vent pipe to upper head
– DMW between vent pipe to vent pipe connection
– DMW between thermocouple penetration to upper head and to thermocouple tube

So, PWSCC is adequately managed for in-scope components and subcomponents through performing UT and VT-1 examination, and implementation of Leakage Monitoring System.

10. Trans granular Stress Corrosion Cracking (TGSCC)

TGSCC is a relevant ageing mechanism for austenitic stainless steel with a Nickel content below 15 % when exposed to water above 50 °C that contains critical amounts of Chlorides and dissolved Oxygen.

Water chemistry of the main coolant is carefully monitored by the Water Chemistry Monitoring System. During steady-state normal operation, the concentration of dissolved Oxygen in the main coolant is below 0.005 ppm and the Chloride concentration is below 0.01 ppm [Chemistry Handbook Part V Monitoring]. These values are far below the critical concentrations discussed earlier, and therefore TGSCC of internal surfaces is not of concern during the 60 years service period.

In-scope austenitic stainless steel components and subcomponents in the vent pipe that are in contact with main coolant are periodically inspected through walk-downs after each outage before start-up, and continuously monitored during operation by the Leakage Monitoring System. Therefore, TGSCC is adequately managed for the vent pipe.

TGSCC on external surfaces due to the presence of external Chloride sources (e.g., adhesive tapes, lubricants), is managed by the prohibition of chemicals and materials which contain either dissolvable chloride or chlorine, which might be released as chloride during decomposition.

With the performance of these activities (Water Chemistry Monitoring, Leakage Monitoring, walk-downs and the use of Chloride-free chemicals and materials), it can be concluded that TGSCC is adequately managed for all in-scope components and subcomponents composed of austenitic stainless steel and Nickel-base alloys during the 60 years service period.

11. Environmentally-Assisted Fatigue

Environmentally-Assisted Fatigue is a relevant ageing mechanism for all components and subcomponents subjected to cyclic loading or repeated transient loading under simultaneous environmental impact.

Environmental effects can result in initiation and growth of cracks, which might lead to leakage or even rupture. Therefore, a reduction in the Fatigue life by environmentally-assisted fatigue shall be considered. So far there have been no documented occurrences of fatigue failure in PWRs due to environmental effects, that have not been covered by given fatigue design. However, the U.S. NRC requires the consideration of environmental factors for Fatigue assessment during the design phase of new plants and long-term operation in the U.S. [US NRC Regulatory Guide 1.207], as well as during the evaluation phase supporting the Long-Term Operation for existing plants [NUREG-1801].

Any environmental effects are considered during the design phase through the selection of appropriate materials, dimensioning, reduction in stresses by design, further procedural measures or measures during the fabrication phase (e.g. cladding, avoidance of crevices, etc.). Fatigue analysis results are available for in-scope components and subcomponents.

All sensitive zones by fatigue analyses are in ferritic steels with cladding, except for the sensitive zones in the CRDM adapter tubes. So ferritic steel zones are not wetted by primary fluid, because they are protected by the stainless steel cladding. Thus, they do not need to be considered in the environmental effect assessments. However, experimental evidence indicates the presence of an environmental effect on the Fatigue life of austenitic stainless steel claddings as well.

The most sensitive zone for the in-scope primary side are the Ni-Cr-Fe alloy CRDM adapter tubes, wetted by primary fluid. The cumulative usage factors (CUF) including environmental effect (Fen-factor) resulted in values less than 1.00 during the 60 years of service life for in-scope components and subcomponents.

Nevertheless, according to ASME XI ultrasonic testing and visual inspections are considered in the ISI of susceptible in-scope subcomponents.

Actions should be taken to revisit this ageing mechanism if adverse results are obtained during monitoring, inspection or testing activities.

So, it can be assumed that Environmentally-Assisted Fatigue is adequately managed for in-scope components and subcomponents during the 60 years service period.

5.1.3 Monitoring, testing, sampling and inspection activities for RPVs

NPP units Loviisa 1 and 2

In accordance with the requirements of [STUK – Guide YVL E.5], periodic inspection programmes have been prepared on the basis of ASME Code Section XI regulations. The inspection methods to be used correspond to the requirements of ASME XI or Finnish regulations on pressure equipment.

The selection of the inspection technique and the verification of inspection systems (equipment, instructions and personnel) are qualified on case-by-case basis and submitted to STUK for approval.

The inspection period is 10 years, corresponding to the requirements of ASME XI paragraph IWB-2400 and Finnish pressure equipment regulations.

Inspected objects are divided into categories based on ASME XI paragraph IWB-2500. Each inspection object is subject to the requirements of its category and approval limits.

The inspection results are assessed according to ASME XI paragraph IWA-3000 (dimensioning principles) and IWB-3000 (acceptance limits) if it is only possible for the component, structure and material.

Above the approval limits, the acceptability of the fault is solved on a case-by-case basis. The impact and acceptability of the fault can be demonstrated by compiling a separate strength analysis using analytical methods presented in ASME XI.

Primary circuit tightness and pressure tests are performed in accordance with regulations for Finnish pressure equipment and the requirements of ASME XI paragraphs IWA-5000 and IWB-5000.

IRRADIATION EMBRITTLEMENT

Neutron radiation causes embrittlement of ferritic materials. The embrittlement can be noticed as rising transition temperature and decreasing of upper shelf toughness. The sensitivity of the steel to radiation embrittlement depends on steel alloying and impurities, in Loviisa RPVs typically on Cu and P.

Embrittlement is a function of radiation dose. For this reason, embrittlement varies between different locations in the RPV wall, depending on the strength of the flux. In addition, embrittlement changes rapidly in the wall of the RPV as a function of thickness and is strongest at the surface between the base material and the coating.

Some of the structural changes in material structure caused by radiation are dynamically recovered by the reorganization of atoms in the material. Increasing temperature improves this material recovery. For this reason, the amount of embrittlement depends also on the operating temperature of the RPV.

Heat treatment, which takes place at temperatures 150 ° C to 200 ° C above the operating temperature, is able to eliminate most of the radiation-induced embrittlement. Due to the heat treatment, the toughness of the material is recovered.

Tests and results

During the manufacturing of the RPV several non-destructive tests have been carried out. In addition, non-destructive tests were performed also for test specimens manufactured simultaneously in the same way.

Non-destructive tests during the manufacturing of the RPV include ultrasonic testing and magnetic testing for the base material. The coating has been subjected to dye penetrant and ultrasonic inspections.

The welds have been x-rayed to the extent it has been possible. Other critical welds have been subjected to ultrasonic testing. The surface of all welds have been inspected. The inspections have been carried out in accordance with an approved quality control programme. The RPV is finally pressure tested after the last heat treatment at a pressure higher than the design pressure.

During fabrication, base and weld material of the core area has been reserved for brittle fracture studies. Test specimens made of this material are installed in the RPV in locations of high neutron flux. Radiation embrittlement is monitored with these irradiated surveillance specimens.

The original surveillance programme

The embrittlement caused by irradiation in forgings, welds and heat affected zones of welds is monitored by a surveillance programme.

The evaluation of embrittlement rate is based on Charpy-V toughness tests performed for base and weld metals and HAZ before and after irradiation. Embrittlement has also been studied by tensile testing and COD-fracture toughness testing.

Based on the first test results, changes to the reactor core were also made. Some of the outside fuel elements were replaced by so called dummy elements who do not contain fuel. Based on brittle fracture studies the temperature of emergency core cooling water was raised and changes were done in automation. The original surveillance programme has been expanded and modernised based on experience.

Also irradiated specimens that have been heat treated before re-irradiation have been placed in the sample chains. These samples have been used to investigate the re-embrittlement rate of the RPV base material and welds.

The new surveillance programme for Loviisa 1 RPV after heat treatment

After the heat treatment of the core weld seam in 1996, new surveillance specimens were installed in the Loviisa 1 RPV. This surveillance programme is limited only to monitoring the weld material. The programme consists of two thermal annealings and three irradiation cycles. The length of irradiation cycles equals to three operating cycles, and the entire programme lasts 9 years. The material for the programme has been acquired from the RPV manufacturer and, based on the examinations it is representative for the LO1 core area weld.

Based on the results of the new surveillance programme it has been demonstrated that the margin to brittle facture of RPV weld 4 can be considered acceptable even after 50 years of operation.

NPP units Olkiluoto 1 and 2

System 218 (Irradiation test specimens) consists of test specimen containers, test specimens and neutron detectors.

The test specimen containers consist of stainless steel pipes which, after having been charged with test specimens, are filled with inert gas (argon) and sealed with welded plugs. The containers are designed in order to be suspended in the special channels belonging to system 212 (Core support components) in the reactor down comer.

The test specimens are of two types:

- Charpy-V impact test specimens for the determination of the temperature at which the transition from ductile to brittle fracture takes place, and

- tensile test specimens (miniature test specimens) for the determination of the yield strength, tensile strength, rupture elongation and rupture contraction.

During the irradiation, the tensile test specimens are provided with copper fillers to give the same limiting volume as the impact test specimens.

The neutron detectors are of foil type and measure the neutron flow. The foils are contained in tightly welded containers.

Each type of reactor material is represented by five sets of test specimens. Each set consists of approximately 21 impact test specimens and four tensile test specimens.

One set is designed for reference and is tested in non-irradiated condition and one set is used for accelerated testing in a test position close to the outside of the moderator tank. The three remaining sets are irradiated in test positions close to the reactor vessel wall.

A deep (10 mm) axial indication was detected in OL2 system 312 nozzle / safe-end weld already in 2003 or even earlier. Since then the indication has been inspected from inside every second year until 2011. No change of the depth of the indication has been noticed during these years. The tests were comparable because they were always performed with the same qualified ordinary eddy-current and ultrasonic testing method with a manipulator.

In 2011 the nozzle was inspected with a phased array UT, which is a more sophisticated technique. Based on this technique the depth 15 mm was measured for the same indication. This new finding started a process that finally lead to repairing of the 312 nozzle / safe-end weld. Other RPV nozzles of same kind were also inspected. Based on these inspections it was also decided to repair one nozzle weld in system 323 (see also Section 5.1.4).

NPP unit Olkiluoto 3

Continued satisfaction of the basic safety requirements for OL3 RPV operation during the 60 year design life is assured through the application of operational monitoring activities, In- Service Inspections and additional surveillance programmes.

A. MONITORING MEASURES

The identification of ageing mechanisms relevant for the RPV is done in Section 5.1.2. The monitoring programmes corresponding these ageing mechanisms are specified below.

Monitoring programmes consider the direct or indirect recording of actual loadings and operational parameters during normal plant operation. The monitoring of operating parameters covers the continuous measurement of relevant parameters, such as pressure, temperature, flow rates, water chemistry and level measurements. Transients of these parameters, as well as the frequency of occurrence of specified loading conditions, are also recorded.

Monitoring systems are a part of the preventive ageing strategy employed for the OL3 NPP in order to mitigate the degradation of physical and functional properties of SSCs.

1. Monitoring of Thermal Ageing of Dissimilar Metal Welds (DMWs)

Thermal Ageing is a relevant Ageing Mechanism and is in general a degradation process that commonly causes deterioration in material's strength, hardness, ductility or toughness.

A surveillance programme is required for the DMWs of OL3 RPV to:

– Verify the hypotheses made during the design stage regarding the material property values used in the Leak Before Break and Fast Fracture Analyses

– Identify safety margins allowing the reduction of In-Service Inspection (ISI) scope or consideration of future plant life extension

The programme is targeted toward providing projected material property results in the end-of-life condition, including the tearing resistance at 300°C and the equivalent RTNDT of the RPV nozzle to safe-end DMWs.

2. Monitoring of Irradiation Embrittlement of Core Beltline Region

A further relevant Ageing Mechanism is irradiation embrittlement. During RPV operation, the vessel wall (core beltline region) is subjected to neutron irradiation, causing material embrittlement. This ageing mechanism is monitored using a surveillance programme.

The Irradiation Surveillance Programme considers 60 years design life of the OL3 RPV, and describes the methodology used to monitor the effect of in-service radiation on the material properties of RPV components in the beltline region. The Surveillance Programme determines the beltline component material properties through mechanical tests of both non-irradiated test specimens (initial state) and irradiated test specimens (post-irradiation state).

Specimens are located closer to the core and are consequently exposed to a higher rate of fast neutron exposure compared to the inside wall of the reactor vessel. Therefore, the mechanical properties of irradiated test specimens, and the related RTNDT shift, are representative of the vessel and allow a projection of its RTNDT shift later in the reactor vessel service life. These capsules can be withdrawn after the vessel head and the upper Internals have been removed.

The purpose of the RPV Irradiation Surveillance Programme is to experimentally verify the tensile and fracture toughness of the RPV shell materials after a representative irradiation of test specimens (post-irradiation testing).

3. Monitoring of Primary Coolant Water Chemistry

Primary coolant water chemistry monitoring is a supporting tool toward identifying the onset of corrosive ageing mechanisms in the RCS and the RPV as well. Monitoring of primary coolant chemistry is one of the preventive ageing measures applied to the OL3 NPP in order to mitigate the degradation of physical and functional properties of systems, structures, and components.

The main objectives of the primary side water chemistry control are:

- Minimize metal release rates and corrosion product concentration,
- Optimize corrosion product migration and re-deposition
- Limit the corrosion rate of fuel cladding material
- Suppress decomposition of water by radiolysis
- Prevent localized corrosion (SCC/pitting) through the limitation of impurities (chlorides, fluorides, sulphates)

4. Leakage Monitoring

With respect to the RPV, a Leakage Monitoring System provides humidity and air temperature instrumentation, installed on the RPV head. The function of the leakage detection equipment is to detect deviations from normal operation and to provide means to prevent the progression of initiating events to situations where a safety function is needed.

Between the Reactor Pressure Vessel and closure head a specific leakage monitoring is implemented. This joint is equipped with two O-rings to serve as a part of the RCS Pressure Boundary. A seal leak off line drains from the space between these two O-rings directly to the Vent and Drain System. If the inner O-ring is leaking, a temperature sensor will detect a temperature increase.

5. Loose Parts Monitoring

A Loose Parts Monitoring System (LPMS) is installed in the RPV, with three measuring locations at the bottom, four measuring locations on the top and two measuring locations on the head of the RPV. The LPMS (JYF) is applied to primary circuit components to ensure the RCS integrity through the early detection of loose and loosened parts.

6. Vibration Monitoring

Furthermore, a Vibration Monitoring System (VMS) with four absolute displacement transducers is installed on the RPV closure head. The VMS (JYG) is installed for early detection of changes in the vibration behaviour of the RPV internals and component structures.

7. Fatigue Monitoring

The fatigue behaviour of the primary inlet and outlet nozzles is monitored using appropriate tools. The measuring point is on the MCL loop 3 on the hot leg next to the RPV nozzle. The basic idea in automatic fatigue monitoring is the measurement and storage of temperature and pressure data with respect to time. One target is to identify such phenomena as thermal shock or thermal stratification.

8. Management of Boric Acid Corrosion

Boric Acid Corrosion is a type of uniform corrosion that attacks primarily carbon steel and low-alloy steel in the presence of hot, concentrated aqueous solutions of Boric Acid. Main coolant (i.e. borated water) leakage results in deposits of white Boric Acid crystals and the presence of moisture that can be observed by visual inspection. Concentrated Boric Acid in the form of a solution or crystals could be present where main coolant leakage is subject to evaporation. Exposure of carbon steel or low-alloy steel surfaces could result in corrosion and a decrease in the structural integrity of components and subcomponents that form a part of the Reactor Coolant Pressure Boundary.

[IAEA Safety Reports Series No.57] requires plant programmes for identification of preventive and mitigation actions. Timely corrective or mitigation actions for extent of degradation are possible through monitoring and trending. Operating experience should also be considered for ageing management.

According to [NUREG 1801] it is necessary to monitor the condition of the reactor coolant pressure boundary for borated water leakage. Therefore, it is assumed that a Boric Acid Corrosion management will be implemented at OL3. Boric Acid Corrosion management is an Ageing Management activity that monitors the effects of Boric Acid Corrosion on the intended function of an affected structure and component by detection of borated water leakage.

The Reactor Coolant System contains borated water that could affect external surfaces of structures and components in the primary circuit, as well as adjacent components. Therefore, these activities cover any structures or components with which borated water could come into contact.

For the management of Boric Acid Corrosion, both leakage monitoring and visual inspections (system plant walk-downs) should be implemented.

9. Mitigation of Chlorides

Due to the fact that chloride impurities have an impact on many ageing mechanisms, chlorides should be mitigated. Therefore, in addition to the monitoring of corrosion impurity concentrations, general Ageing Management activities are also in place to mitigate the contamination of mediums and steel surfaces with Chlorides.

Several additives or chemicals are introduced to the main coolant during normal operation. The handling and quality of these additives or chemicals are generally controlled according to their respective requirements to ensure that no critical concentration of harmful substances can enter the plant which could lead to the degradation of components and subcomponents.

The presence of external Chloride sources is prevented by prohibition of the use of Chloride-containing tapes, markings, fluids, etc. Chemicals should be listed in database, which are permissible or not permissible materials for different areas of use.

10. Foreign Material Exclusion

Fuel and equipment failures can occur due to foreign material intrusion into the plant systems and components. However, foreign material is not considered a risk for the RCPB.

B. PERIODIC PROGRAMMES FOR INSPECTION AND TESTING

The periodic inspection programmes are also a part of the preventive ageing management strategy.

1. Surveillance Programme

[IAEA Safety Standards Series No. NS-G-2.6] requires that a surveillance programme should provide assurance that the operational limits and conditions are met during operation. Therefore, a surveillance programme will also help to detect trends in ageing.

The objective of the OL3 RCSL system is to ensure the functional availability of systems and components that perform safety functions, within operational limits and conditions, to promptly detect SSC deterioration, as well as any trend that could lead to an unsafe condition [Nuclear Plant Chemistry Conference, Paper Reference n°167 046].

The RCSL system includes the performance of periodic inspections and testing to establish and direct monitoring, and inspection of the RPV and all SSCs generally.

For the ageing management assessment of RPV subcomponents the monitoring of certain plant parameters (e.g. water chemistry, temperature) and the special surveillance programme for thermal ageing is evaluated in this SAMP report with respect to surveillance.

2. Pre-Service Inspections

Pre-service Inspections (PSI) are important in determining the baseline condition of certain in-scope components, and were performed during plant construction and will be performed during commissioning. These results, in combination with those obtained during the construction and manufacturing phases, form the basis for comparison with subsequent In-Service Inspection results.

3. In-Service Inspections

In-Service Inspections (ISI) are performed for NPP components to provide continuing assurance that they will safely operate throughout the lifetime of the plant, contributing to plant availability and minimizing the probability that an event will occur. ISI activities allow the comparison of baseline information to assess changes or degradation in equipment condition which have occurred with time.

The component selection criteria have evolved to include inspections that should be performed at locations susceptible to developing in-service defects, and should be implemented based on [STUK – Guide YVL 3.8].

- Operating Experience feedback (both plant-specific and industry wide)
- Potential Failure Mechanisms
- Risk-informed arguments
- Operating Conditions and Stress analyses (e.g. stresses or high usage factors, geometrical defects)
- Equipment Design

The specific PSI / ISI requirements and ISI programme for the Main Primary Components are defined in relevant inspection instructions. The RPV ISI Plan is based on ASME XI inspection programme B, IWA 2432 [ASME Boiler & Pressure Vessel Code, Section XI].

The intervals for evenly distributed inspections may be chosen from a few years to about ten years. Intervals' length should be chosen on the basis of conservative assumptions, to ensure that any deterioration of the most exposed component is detected before it can lead to failure [IAEA Safety Standards Series No. NS-G-2.6].

General procedures or techniques applied during ISI activities are reported within the ISI summary document. Other relevant testing procedures for ISI are performed as follows:

- Volumetric Examinations:
  Ultrasonic Testing (UT) is used to detect volumetric defects within metallic materials (ASME XI IWA 2230).

- Surface Examinations:
  - Magnetic Particle Testing (MT) is used to detect surface defects in magnetic materials (ASME XI IWA 2220)
  - Liquid Penetrant Testing (PT) is used to detect surface defects (e.g. cracks) (ASME XI IWA 2220).

- Visual Examinations:
  Visual examination is performed to get information about the general condition of the part, component or surface. Defects like scratches, wear, cracks, corrosion or erosion on the surface, or evidence of leakage are observable.

- Eddy Current Testing:
  Eddy Current Examination (ET) is used to detect volumetric and surface defects (ASME XI IWA 2233).

5.1.4 Preventive and remedial actions for RPVs

NPP units Loviisa 1 and 2

The most important degradation mechanism of Loviisa RPVs is irradiation embrittlement of the RPV core area and its weld. Therefore the most important preventive and remedial actions are related to that. Already the very first surveillance programme results showed that the embrittlement rate of the Loviisa RPVs was much faster than expected. Therefore immediate mitigation actions had to be implemented in the early years of operation.

To mitigate the excessive embrittlement of the Loviisa RPVs following actions have been implemented:

- design modifications of the active core
  - core reduction by replacing the outmost fuel elements with dummy elements of stainless steel, flux reduction 83%
  - introducing a new fuel loading pattern to achieve a low leakage core design, flux reduction 18%

The following actions and plant modifications have been implemented in order to reduce the risk of brittle fracture of the RPV during accident or transient conditions:

- temperature increase of emergency core cooling water
- decrease of head and flow rate of high pressure injection pumps
- increase of pressurizer relief valve capacity
- introduction of a signal from high primary system pressure to stop the high pressure pumps
- modification of protection signals associated with occurrence of potential steam line break
- thermal annealing of Loviisa 1 RPV core area weld

Fatigue is another dominating degradation mechanism of the RPVs. RPV-head nozzle repairs have been performed because of fatigue damages and/or manufacturing defects in the nozzle sleeves of both units.

RPV main flange sealing surfaces of both units have been repaired. Main reason for these repairs have been low-cycle fatigue of the sealing grooves combined with manufacturing defects and corrosion effects. These failures have been detected in periodic inspections.

NPP units Olkiluoto 1 and 2

The preventive actions include:

– assessment of susceptibility to degradation mechanisms,
– assessment of propagation of degradation mechanisms, and
– protective actions.

The assessment of susceptibility to degradation mechanisms is mainly based on material type, local geometry, prevailing stresses or strains and process chemistry. Recently, a thorough study [Cronvall, O] on this issue has been done for the RPVs and their internals of NPP units Olkiluoto 1 and 2, see Annex 5 for the construction and subcomponents.

For the susceptible locations, degradation propagation analyses have been done, covering all significant degradation mechanisms. These include irradiation embrittlement, fatigue, SCC and irradiation accelerated SCC (IASCC). These analyses are presented in [Cronvall, O] and they cover also the LTO period, i.e. continued operation from 40 to 60 years. According to the conservative analysis results, the degradation in terms of crack growth is in most cases very slow.

Protective actions concern mainly shielding components. In case of RPV, the most significant shield is the feedwater sparger. It distributes the feedwater more evenly to the RPV down comer and also acts as a thermal shield to the feedwater nozzle and connecting components.

The remedial actions include:

– repair
– water chemistry control
– surface treatment
– component replacement

Repairing concerns mainly repair welding. For instance, in 2017 local repair welds have been done to one system 312 nozzle/safe-end weld and one system 323 nozzle/safe-end weld. The pipe nozzles were repaired by a method that was not previously used in Finland. TVO initiated the repair as planned by correcting the detected flaws in OL2 nozzles. The repair was carried out by machining the inside weld and welding a new filler coating with a filler material less susceptible to stress corrosion cracking.

Water chemistry control concerns improving the chemistry and oxygen content of the primary circuit water to decrease the effect of degradation mechanisms, mainly to reduce or prevent initiation and propagation of SCC.

Surface treatment, such as laser peening and shot peening, introduce a compressive surface residual stress. This effectively prevents the onset of both SCC and fatigue.

As for component replacement, it obviously does not concern the RPV. However, some internals are replaced periodically, e.g. control rods and fuel. Also the RPV main flange bolts are regularly inspected and replaced if necessary.

NPP unit Olkiluoto 3

Maintenance is defined as all activities necessary to establish and keep the plant in the condition required to fulfil its designated functions [IAEA Safety Standards Series No. NS-G-2.6]. The range of maintenance activities includes service, overhaul, repair and replacement of parts, and often, as appropriate, testing, calibration and inspection.

The relevant maintenance activities may be divided into [IAEA Safety Standards Series No. NS-G-2.6]:

- Preventive (including periodic, predictive and planned maintenance), and
- Corrective maintenance.

The task of maintenance is to prevent and to correct very early traces of wear out, material weaknesses or other types of degradation which might have a negative impact on safety or availability of the RPV.

Maintenance of certain SSCs will be performed to ensure that the condition and functionality of the SSC remain within acceptable limits. Therefore, TVO will apply a specific preventive condition assessment strategy for the OL3 RPV that involves the following activities:

- Recognize degradation processes in a timely manner (i.e. promptly), and
- Take measures in time to prevent a negative influence on nuclear safety, or to sufficiently minimize such influence.
- Should the monitoring, testing or inspections show results that the RPV may have suffered from unexpected degradation, preventive or remedial actions must be considered. The actions should be based on a root cause analysis which includes the possible degradation mechanisms, their origin, evolution scenario and impact to the plant safety and availability. The preventive or remedial actions are then evaluated and implemented using processes described in the TVO's operative instructions.

5.2 Licensee's experience of the application of AMPs for RPVs

NPP units Loviisa 1 and 2

Management of the embrittlement rate of the RPV core area weld has been the most challenging ageing management issue. Corrective actions have been taken and the most important one has been thermal annealing of the Loviisa 1 RPV core area weld.

The heat treatment of the core area weld of the Loviisa 1 RPV restored the toughness of the weld material close to the original level. The heat treatment was carried out at a temperature range of 475-500 °C. In safety analyses a value of 30 °C is applied to the post-heat residual brittleness. The re-embrittlement of the weld material is monitored by a new surveillance programme that was started in 1996.

As a result of extensive corporate level research work related to the irradiation embrittlement the licensee has obtained valuable know-how and competence in this area. This has had an important role in successful ageing management of the RPV.

Several linear indications have been found in the sealing faces of the RPV main flange in 2005 and 2006. The indications were opened and filled.

AMPs for the RPVs have been modified slightly during plant operation and according to current understanding they are comprehensive enough and adequate to manage the identified ageing mechanisms of the RPVs.

Operating experience has caused changes to the inspection programmes of the RPVs.
In the early 2000's RPV-head nozzle failures were discovered in one of the two VVER-440 reactors and based on this experience Loviisa inspection programme was updated. Later inspections showed other failures of similar kind in Loviisa reactors. However, after that no further failures of that kind have been detected.

In 2010 four indications close to each other were found in the circumferential weld (weld 8) in Loviisa 2 RPV cover. The location of the weld is shown in Annex 4. The combined size of these indications exceeded ASME XI acceptance criteria. The approval of the indications was shown later with fracture mechanical analysis. It was assumed that the indications were manufacturing defects and thus not prone to grow. However, their potential growth is monitored regularly with ISI.

In 2016 one linear axial indication was found in a low pressure safety injection nozzle (so called TH-nozzle) of Loviisa 1 RPV. The nozzle is located between the cold leg nozzles of MCLs. The indication is located in inside corner of the nozzle in 321^{0} location. The indication was sized to be14 mm (depth) and 44 mm (length). A fracture mechanical analysis according to ASME XI was performed for the indication and based on this analysis the indication could be considered acceptable. The indication was re-inspected in 2017 and the results did not show any growth. Due to development of the inspection technique the indication was re-sized more accurately and the new size of the indication was smaller, 10 mm (depth) and 30 mm (length). The indication was reported as indication located in cladding which most probably means several small old welding defects inside the cladding. The utility has inspected all nozzles of same kind at both RPVs. So far no corresponding indications have been found. The utility is developing the inspection technique further in order to improve its accuracy and reliability.

NPP units Olkiluoto 1 and 2

The basic evaluation and technical basis of the Ageing Management Programmes (AMPs) according to IAEA is described in the table below. Additionally a short explanation is provided on how the issue was handled in [Cronvall, O. Susceptibility of BWR RPV].

	Definition	Handled in [Cronvall, O. Susceptibility of BWR RPV]
1.	Scope of the ageing management programme based on understanding ageing	All possible degradation mechanisms as well as the associated component specific susceptibility. Fatigue is not an issue in the RPV except for the feedwater nozzle. The feedwater nozzle is handled in a separate report
2.	Preventive actions to minimize and control ageing degradation	Effect of improving water chemistry is described, not much else is covered concerning this issue. No new improvement methods have been recognized.
3.	Detection of ageing effects	Inspections and applied inspection techniques are described in detail
4.	Monitoring and trending of ageing effects	Computational modelling of significant degradation mechanisms is described in detail, the computational part covers them all for all significant internals and the RPV. The feedwater nozzle fatigue is handled in [OL1 and OL2 - TURO2018 - System 312]
5.	Mitigating ageing effects	Applicable mitigation techniques are described in detail
6.	Acceptance criteria	This issue is discussed when weighing the TLAA results for all significant internals, the RPV and the feedwater nozzle.
7.	Corrective actions	This issue is discussed only briefly.
8.	Operating experience feedback and feedback of research and development results	This issue is not included in [Cronvall, O. Susceptibility of BWR RPV]. Some damages have occurred in the history of the RPV and internals. Cracks have been observed in some nozzles, consoles and lifting lugs. These have all been handled at the time and documented in dedicated reports.
9.	Quality management	This issue is not included in [Cronvall, O. Susceptibility of BWR RPV] as it is not part of the scope.

Ageing Management Programme [IAEA Safety Reports Series No. 82] for the OL1 and OL2 RPV, internals and the main piping systems are listed below:

1. AMP101 Fatigue Monitoring:
   For all main piping systems in the containment a full reanalysis was made and all associated reports were submitted to STUK as part of the license renewal project.

Necessary actions following these analyses have been listed and will be carried out in the near future. Apart from the feedwater nozzle [OL1 and OL2 - TURO2018 - System 312] the fatigue usage of the RPV is insignificant, no specific AMP is necessary.

a. The purpose of this programme is to manage low-cycle fatigue considered in the nuclear plant design basis. This programme applies to piping components subject to cyclic loading. This programme manages also any other transients due to the occurrence of fluid conditions having the potential for inducing cyclic thermal stresses like:
    i. Stratified flow: according to the latest CFD analyses no stratification occurs, even in the last horizontal part of the feedwater pipe before the RPV.
    ii. Swirl penetration: non such is recognized in the piping or in the RPV.
    iii. Thermal mixing: thermal mixing is more related to high cycle fatigue and is handled in purpose made reports. Non such is possible in the RPV.

2. AMP102 In-service Inspection/Periodic Inspection:
   Full ASME ISI is performed. In case of findings the programme is intensified. This programme generally includes periodic visual, surface, and/or volumetric examination and leakage test of all Class 1, 2, and 3 pressure-retaining components and their integral attachments. Repair/replacement activities for these components and acceptance by analysis (such as flaw analysis) are also covered. The programme has proven to be effective in ageing management in Class 1, 2 or 3 piping and components.

3. AMP103 Water Chemistry:
   The main objective of this programme is to mitigate loss of material due to corrosion. This includes flow-accelerated corrosion (FAC), stress corrosion cracking (SCC) and related degradation mechanisms, and reduction of heat transfer due to fouling in components exposed to a treated water environment. The programme includes periodic monitoring of the treated water and control of known detrimental contaminants below the levels known to result in loss of material or cracking. The water chemistry programme for all types of nuclear power plants with water-cooled reactors relies on monitoring and control of reactor water chemistry based on several guidelines, such as IAEA Safety Guide SSG-13.

4. AMP104 Reactor Head Closure Stud Bolting:
   Regular ISI and exchange in case of findings. No specific AMP.

5. AMP105 BWR Vessel ID Attachment Welds:
   Regular ISI and analysis in case of findings. No specific AMP. This programme includes
   a. Inspections, inspection recommendations and flaw evaluation to provide reasonable assurance of the long-term integrity and safe operation.
   b. Evaluation methodologies for the attachment welds between the vessel wall and vessel ID brackets that attach safety-related components to the

vessel.

6. AMP106 BWR Feedwater Nozzle:
   Full ASME ISI, intensified due to findings. In 2017 repair of inner surface breaking axial IGSCC crack. This programme includes enhanced in-service inspections like periodic ultrasonic inspections of critical regions of the BWR feed-water nozzle.

7. AMP107 BWR Stress Corrosion Cracking in Coolant Pressure Boundary Components:
   The main objective of this programme is to manage SCC, particularly IGSCC, in BWR coolant pressure boundary components made of stainless steel (SS) and nickel-based alloys, including welds. Programmes and good practices to manage SCC in BWRs are delineated in various national and international reports, such as IAEA Technical Report No. NP-T-3.13. Full ASME ISI, intensified due to findings. Regular pre-analysis is performed for postulated IGSCC cracks. The programme includes:
   a. preventive measures to mitigate IGSCC
   b. inspection
   c. flaw evaluation to monitor IGSCC and its effects.

8. AMP109 BWR Vessel Internals:
   This programme includes inspection and flaw evaluations in conformance with the applicable requirements or guidance documents. It aims to provide reasonable assurance of the long-term integrity and safe operation of BWR vessel internals. In addition, this programme addresses ageing management of BWR vessel internals of cast austenitic stainless steels (CASSs). This programme considers loss of fracture toughness due to neutron embrittlement or thermal ageing embrittlement. This programme also addresses ageing degradation of X-750 alloy and precipitation-hardened (PH) martensitic SS materials, and those BWR vessel internals that are of martensitic SSs.

9. AMP138 Reactor Coolant Pump:
   All pumps will be exchanged 2016 – 2019. Normal inspection and maintenance, no further AMP. This programme on ageing management covers several degradation mechanisms it may be subjected to and the activities necessary to manage the ageing mechanisms. As such, this programme refers to other degradation-specific and/or monitoring type of programmes that deal with particular degradation mechanisms and degradation ageing effects. The body of the pump including its sealing parts is safety class 1, the internals of the pumps performing the active function of the pump are safety class 2, and they are included in the scope for LTO in accordance with the IAEA Safety Report No. 57.

NPP unit Olkiluoto 3

The AMPs for OL3 are under preparation. In 2017, OL3 is in the commissioning phase and the primary circuit hydrostatic test (233 barg) has been performed successfully in June 28, 2017. In addition, the loading condition monitoring programme has started and will continue until PTO conducted by the supplier and after PTO the monitoring will be carried out by TVO.

5.3 Regulator's assessment and conclusions on ageing management of RPVs

Requirements relating to ageing management are based on Section 5 of STUK Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2016):

> *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.*

More detailed requirements are given in [STUK – Guide YVL E.4]:

> *204. For plant construction and modification projects, this Guide presents a strength analysis coverage and reliability verification procedure consisting of pre-operational tests and measurements. Monitoring during service life covers the maintenance of recurring loads within fatigue analysis assumptions, and the effects of the operating environment on the mechanical properties of materials, special emphasis being on radiation embrittlement of the reactor pressure vessel.*
>
> *404. Obligations relating to the way of reporting the plans and results of load monitoring are determined on the basis of STUK requirements for pre-operational testing and ageing management.*
>
> *426. It shall be possible to compare the observations made during pre-operational testing with the results of load monitoring conducted during service. Supplementary measurement data on the behaviour of equipment assemblies and significant local stress conditions shall be obtained, which support the specification of these loadings based on the measurement data monitored during service.*
>
> *427. Operational conditions and events inducing fatigue loads on the most important pressure equipment shall be recorded over the service life of the nuclear power plant. Sufficient measurement data shall be collected on their progress for later verification of essential factors. The monitoring shall be so arranged that unexpected load types occurring during operation are also detected.*
>
> *816. In the monitoring of load and ageing effects, the quality management procedures determined for ageing management in nuclear power plants shall be followed. The personnel evaluating the results shall have good expert knowledge and experience in the fatigue analyses accepted for the nuclear*

power plant in question as well as its behaviour during different operational and transient conditions.

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

STUK has inspected the ageing management programmes including RPV ageing monitoring plans. Implementation of the ageing management programmes has been verified during STUK inspections.

The ageing management processes described in the ageing management programmes were found satisfactory at the operating NPPs.

The ageing management programmes of Olkiluoto 3 are practically finished although the plant unit is still under commissioning.

The ageing management processes require that the licensee regularly evaluates the adequacy of ageing monitoring plans and the coverage and effectiveness of the whole ageing management programme.

STUK reviews the annual ageing management follow-up reports of the licensees. STUK inspectors make observations about ageing as part of inspections during operation and outages of the NPP units, in connection with tests, maintenance or repairing works. So far the findings of STUK have been consistent with the ageing management programmes.

STUK considers that the ageing management programmes concerning the RPVs of the Finnish in-scope NPP units Loviisa 1, Loviisa 2, Olkiluoto 1, Olkiluoto 2 and Olkiluoto 3 are adequate.

Ageing management issues at NPP units Loviisa 1 and 2

There is one ageing management issue that has over the years required significant amount of work and attention from the licensees and STUK as well. This issue is the irradiation embrittlement of Loviisa RPVs and the thermal annealing of the core area weld of Loviisa 1 RPV in 1996. The embrittlement rate of the critical core area welds of both RPVs has to be carefully monitored by the surveillance programmes as long as the RPVs are in operation. If the licensee plans to continue operating the plant units after 50 years, some measures may be necessary to confirm safe operation of the RPVs.

So far one indication has been detected in a low pressure safety injection (TH) nozzle of Loviisa 1 RPV. It may become an ageing management issue if new indications will be detected in other nozzles of same kind in future inspections. However, it is also possible that the existing indication proves out to be a manufacturing defect.

Ageing management issues at NPP units Olkiluoto 1 and 2

The detected indications in the nozzle / safe-end welds of systems 312 (feed water) and 323 (reactor core spray) may become a significant ageing management issue. The licensee has decided to continue its work and inspect and, if necessary, repair the corresponding welds of systems 312, 323 and also system 321 (shut down cooling) at both units OL1 and OL2. So far one 312 nozzle and one 323 nozzle at OL2 have been repaired by a method that has not been previously used in Finland.

The operating license renewal process for NPP units Olkiluoto 1 and 2 is currently under way. The licensee TVO has applied permission to continue operating NPP units OL1 and OL2 until 60 years. In this context TVO has updated the strength and fatigue analyses of the RPVs. These analyses have been quite comprehensive including almost everything necessary except for brittle and fast fracture analyses that are required in Section 6 of [STUK – Guide YVL E.4]. TVO has been given time to complete the analyses. All relevant ageing mechanisms have to be considered in the analyses.

The analyses performed so far show that the most critical subcomponents of the RPVs are the feed water (312) nozzles. So the strength analyses and detected indications support each other. Based on these analyses, the damage risk of all other RPV's in-scope subcomponents is much lower. The analyses also show that IGSCC and IASCC can accelerate ageing of the RPVs.

Ageing management issues at NPP unit Olkiluoto 3

The unit is under commissioning. The long construction period has caused some minor ageing effects in some plant components but there has not been any real ageing issues relating to the RPV.

6 Calandria/pressure tubes (CANDU)

N/A

7 Concrete containment structures

7.1 Description of ageing management programmes for concrete structures

7.1.1 Scope of ageing management for concrete structures

The concrete structures within the scope of this Section are:

- concrete containment structures, with or without a liner, designed to withstand the pressure associated with a significant leakage of coolant from the reactor cooling system; and
- the concrete structure that surrounds:
  - a concrete containment structure as described in the first bullet; or
  - a (self-standing) steel containment designed to withstand the pressure associated with a significant leakage from the reactor cooling system.

This structure is often the outer wall of the reactor building.

Concrete structures (containments and reactor buildings) are grouped in three categories:

- concrete structures in the steel containments and reactor buildings that surround steel containments of Loviisa 1 and Loviisa 2 VVER-type of PWR nuclear power plant units,

- prestressed concrete containment and surrounding reactor building of Olkiluoto 1 (built 1976) and Olkiluoto 2 BWR nuclear power plant units,

- prestressed concrete containment and surrounding concrete reactor building of Olkiluoto 3 EPR-type of PWR nuclear power plant unit (under commissioning, most of the load bearing concrete structures have been cast during 2005-2011).

Concrete structures in the containment and reactor buildings of NPP units Loviisa 1 and 2

The Loviisa Nuclear Power Plant (NPP) is located close to the Finnish town of Loviisa. It houses two Soviet-designed (Atomenergoexport) VVER-440/213 PWR reactors (Loviisa 1 and Loviisa 2), each with a capacity of 502 MWe.

The reactors at Loviisa NPP went into commercial operation in 1977 (Loviisa 1) and 1981 (Loviisa 2) respectively. To comply with Finnish nuclear regulation, Westinghouse and Siemens supplied equipment and engineering expertise. The plant is operated by Fortum Oyj.

Loviisa 1 and Loviisa 2 nuclear power units have a self-standing, welded steel containments with a hemispherical dome, cylindrical middle part and concrete floor slab structure (level +10,45), where in the centre is located the reactor pit concrete structure. The concrete floor slab of the containment and the reactor pit are covered with steel liner in order to maintain the leak-tightness of the steel containment.

The self-standing steel containment is anchored to the reinforced concrete ring-shaped slab at the level +9.60. The cylinder part of the steel containment is welded as an angle connection to flat steel anchor flange ring, that is fixed with pre-tensioned bolts thru the concrete floor slab structure.

The self-standing steel containment is surrounded by outer cylindrical 0,6 m thick concrete wall of the reactor building (inner radius 23600 mm) and steel roof structure. The bearing structure of the roof is made of steel radial steel girders and tangential ring beam structures.

The floor slab at the level +9.60 m is a ring-shaped reinforced concrete slab, with inner radius 5360 mm in Loviisa 1 and radius 5410 mm in Loviisa 2 and outer radius 23750 mm. The slab is 1170 mm thick in Loviisa 1 and 1370 mm in Loviisa 2. In the concrete of the slab there are embedded I-profiles and liner plates of the slab are welded to the upper flange of the I-beams. Against lift caused by pressure the boundary of the ring-shape slab is supported to the reactor building wall with a structure, that is a combination of rubber bearing, steel profile and anchorage rods.

The inner concrete structures of the steel containment are reactor pit, floor slab +1,50, boxes and columns on the level +10,50 m, cylindrical reactor pit walls and reactor pools, slabs on the level +19,30 m and +22,20 m and +25,40 m. Slab on the level +9.60 m is supported by cylindrical walls, which have inner radius 5301, 12600, 20400 and 23400 mm. Additionally the +19,30 slab is supported by columns on the radius 9550 mm and 16300 mm. The reactor pit (level +0…+10,85) structure is a massive reinforced concrete cylinder, which supports the reactor and is the first biological shield.

Concrete structures are designed according to theory of elasticity (allowable stresses) using Finnish Concrete Code 1971, RIL 53C. Limit state design has been used in some analyses using Finnish Concrete Code RIL53d. Load combinations in the design were according to ACI 349-76.

Materials in NPP units Loviisa 1 and 2

The steel containment is fabricated using Rautaruukki carbon steel Raex 385 (Loviisa 1 unit) or Raex 305 AV (Loviisa 2 unit). Design of the steel containments has been done according to ASME Codes; ASME BPV Code Section II and Section III and Section VIII version 1968 for Loviisa 1 containment and version 1971 for Loviisa 2 containment.

The concrete class of inner structures of the containment are generally either concrete strength class AK300 or AK350 (C25/30 or C28/35 according to SFS-EN1992-1-1). The cylindrical wall of the reactor building is made of class AK300 (C25/30) concrete. The used cement in concrete structures has been mainly ordinary Portland cement (OPC) in inner containment and reactor building. Slag cement (OPC+ggbs) has been used in some structural parts (slab on the level +9,60 and reactor pit). The concrete strength class of reactor pit is partly $AK_{90}250$ (slag cement, strength tested at 90 days age, C20/25, EN1992-1-1) and partly $AK_{28}300$ (portland cement, strength tested at 28 days age, C25/30).

Heavyweight concrete (density > 3500 kg/m^3) with magnetite as an aggregate has been used in structures designed against gamma radiation. Cylindrical biological shield is partly covered by steel structure filled with heavyweight concrete with serpentinite aggregate.

Rebars of the concrete structures were ribbed bars either grade A40H or A40HS (weldable), fy= 400 MPa and round bars A22 or A22S (S235). Lap splices were mainly used but for 25 and 32 mm bars also coupler splices were used after 1974. Welded splices were used in the reactor pit of the Loviisa 1 unit.

Structures which needed absolute water tightness were covered with stainless steel plates grade AISI 304 L (reactor pit) and grade OX18H10T (pools and pits). Thickness of the liners was mainly 3 mm except in the bottom of the reactor pit where thickness was 10 mm. These liners were either prefabricated liner-elements installed as the formwork or welded to the steel members embedded in concrete.

Bearings have been used in the expansion joints between slabs on the level +9,60 and +10,50 and reactor pit. These bearings consist of two polyester plates and silicone grease installed between them. Thickness of the bearing is 4 mm. Between the bottom concrete surface and the bearing a neoprene rubber sheet is installed.

Concrete structures in the containment and reactor buildings of NPP units Olkiluoto 1 and 2

The Olkiluoto Nuclear Power Plant is on Olkiluoto Island, which is on the shore of the Gulf of Bothnia in the municipality of Eurajoki in western Finland. The plant is owned and operated by Teollisuuden Voima (TVO), a subsidiary of Pohjolan Voima Oy.

The Olkiluoto plant consists of two Boiling Water Reactors (BWRs) Olkiluoto1 and Olkiluoto2 with 880 MWe each.

Unit 1 achieved its initial criticality in July 1978 and it started commercial operations in October 1979. Unit 2 achieved its initial criticality in October 1979 and it started commercial operations in July 1982.

Olkiluoto 1 ja Olkiluoto 2 (OL1 and OL2) NPP units have prestressed concrete Inner Containments. Containments have been designed against load combinations given in the document VBB Design Criteria, Rev. H, 1977-11-15. Structures have been designed according to Finnish Building Code using "allowable stresses theory."

The main parts of the outer shell of the containment are the bottom part (slabs and walls), cylindrical outer wall and roof structure. All these parts are concrete structures. The outer wall contains a steel liner embedded in the concrete. The bottom part consists of basemat on the rock foundation, on the level -2.000 and a cylinder shell with inside radius 4500 mm on the level -2.000…+1.000 with outside radial walls 8 pcs, which support circular plate on the level +2.500.

Cylindrical outer wall consists of prestressed cylinder shell R_{inside}=11000 mm on the level +2.500…+35.000 and circular plate on the level +37.500 and above it walls of fuel tanks of which the walls in east-west direction are prestressed. Pool structures at the level +37.500 belong to containment roof slab and part of the containment walls vertical prestressing cables are anchored to upper part of the pool walls.

Inner structures of the containment consist the lower cylindrical wall from level 2,500 to level +20,00 (RPV pedestal) and the upper cylindrical wall (biological shield) from the level +20,00 to 0,200 m from the bottom of the containment roof structure. There is also a ring shape concrete slab structure between dry-well and wet-well.

The containment wall includes various penetrations such as equipment hatch D 2500 mm at level 37,500, personnel airlocks D 2500 mm at level +25.00 and -2.000 , piping D60…D810 mm and electrical cable penetrations D324 mm, enabling connections with the other buildings and different systems.

Materials of NPP units Olkiluoto 1 and 2

The strength class of the concrete used for the containment structures was AK40 (C32/40 according to EN1992-1-1). In all massive structures low heat cement (Kolari low heat cement in OL1 and Lohja ggbs/Portland-cement in OL2) has been used. For the reinforcement Swedish ribbed bars grade Ks40 or Ks40HS (weldable, fy= 400 MPa) were used. A lap splice was the predominant method used for splicing rebars but also some couplers (mechanical splices) were used.

The prestressing system was VSL (Vorspannsystem Losinger) in OL1 unit and BBRV in OL2 unit. Tendon ducts were grouted with cement based grout. Tendons used in OL1 unit and OL2 unit are given in the following Table 7.1.

Table 7.1. Tendon types used in Olkiluoto 1 and Olkiluoto 2 nuclear plants.

	Olkiluoto 1	Olkiluoto 2
prestressing system	VSL (Vorspannsystem Losinger)	BBRV
tendons of containment	122 horizontal 19 Bridon Supa LR wires d= 13 mm 40 vertical cables each with 19 wires and 18 vertical cables with 31 wires	122 horizontal cables each with 72 D6 mm strands 35 vertical cables each with 72 D6 strands and 12 vertical cables each with 115 D6 strands and 14 vertical cables each with 78 D6 strands
tendons in pool walls	48 horizontal cables each with 19 wires	48 vertical cables each with 72 D6 strands
anchors	VSL types E5-19 (for 19 D13) and E5-31 (for 31 D13)	active BBRV type R-268 (for 78 D 6) vertical and horizontal cables passive BBRV type R115 (for 115 D6)
specimen beams		BBRV type R-238 (72 D6)

Leak-tightness of the containment has been insured with 5 mm thick carbon steel plate HII/W N:o 1.0425 (DIN 17155) liner that has been embedded about 200-250 mm in the concrete structure. Thickness of the liner is 8mm in the wall structure of condensation pools (wet-well). The inner concrete layer protects liner against missiles , temperature variation and corrosion. The liner is not anchored to the concrete in the circular walls.

Fuel pools and condensation pools are covered with 3 mm thick stainless steel plate structure.

Concrete structures in the containment and reactor buildings of NPP unit Olkiluoto 3

The basemat of the unit Olkiluoto 3 was casted January-October 2005. Olkiluoto 3 unit , an EPR reactor, is still under commissioning.

The containment of the Olkiluoto 3 EPR project is a double-wall structure founded on a basemat. The inner containment shell is a prestressed containment with steel liner implemented on the inner surface including the basemat, thus forming a continuous surface.

In the primary containment (inside inner shell) the whole Reactor Coolant System (RCS), the In-Containment Refuelling Water Storage Tank (IRWST), the steam generators, and part of the main steam and main feedwater lines are situated. The inside radius of cylinder part in primary containment is 23,40 m. Design pressure of the primary containment is 5,3 bar absolute.

The primary containment is composed of

- a 6 mm steel liner,
- the liner anchorage system to the concrete, welded to the liner (stiffeners and studs),
- a prestressed 1,30 m thick cylindrical wall, with 1 or 2 layers of horizontal and 1 layer of vertical prestressing tendons,
- a prestressed dome approximately 1.00 m thick, with 2 sets of prestressing tendons

The containment liner consists of steel plates on the inner surface and an anchorage system on the concrete side with which the containment liner is connected to the inner containment wall.

Sleeves are welded into the containment liner penetrating the inner containment wall, which are used for piping systems, HVAC and electrical penetrations. Steel works inside the containment are welded onto anchor plates, which are also integrated part of the containment liner. The anchoring system of the containment liner consists of L-profiles (L-anchors), which are welded onto the steel plate by continuous fillet welds, limiting a containment liner field. Within the containment liner field headed studs are welded, which are arranged in a square pattern of 150 mm or 75 mm (dome).

The containment liner includes various penetrations such as equipment hatch, personnel and emergency airlocks, piping and electrical cable penetrations, and the fuel transfer tube, enabling connections with the other buildings and different systems.

Each horizontal tendon (totally 119) was jacked over full round of 360° from both ends on one vertical rib (three ribs in total). Pure vertical tendons (totally 47) were jacked by the lower anchorage located in prestressing gallery below the base mat. Dome tendons (totally 104) were jacked on both ends located in prestressing gallery and dome belt.

The outer containment shell is a reinforced concrete structure. It guarantees protection against external hazards such as airplane crash APC and explosion pressure wave EPW and withstands the loads of a safe-shutdown earthquake SSE.

The two shells are separated by the annulus, 1.80 m wide, placed under sub atmospheric pressure in order to collect the leaks through the inner containment and filter them before release to the environment.

The basemat is 3,15 m thick reinforced structure. The steel liner is implemented between the containment basemat and the basemat of the reactor building internal structure, which ensures any release of radioactivity to the ground water. A circumferential pre-stressing tendon gallery of vertical tendons of the inner containment is situated underneath the basemat.

The internal civil structures inside the inner containment, which house the reactor coolant system rest on the containment basemat. This interface between internal structures and containment basemat is provided with a leak tight steel liner. The steel liner is implemented on the inner surface of the inner containment shell.

Materials in NPP unit Olkiluoto 3

The concrete used for the inner containment prestressed shell is strength class C50/60 (EN1992-1-1) and all other concrete structures are strength class C32/40. In all massive structures binder with Portland cement and 50-70 % slag (ggbs) has been used.

For reinforcement hot rolled ribbed bars grade A500HW (Finnish standard SFS 1215) and A500SAS (Type Approval YM156/6221/2006, Stahlwerk Annahütte) were used (strength fy= 500 MPa, ductility class B). Couplers of the reinforcement were either Lenton (Erico B.V.) or SAS500 (Stahlwerk Annahütte). Couplers have been used both in the primary containment and in the secondary containment i.e. outer wall of the reactor building.

The prestressing system in OL3 containment is Freyssinet with European Technical Approval ETA-06/0226. Each tendon has 54 T15,7 low relaxation (class 2) strands. For prestressing steel material with strength fp0.1k/fpk = 1653 MPa / 1860 MPa has been chosen. The strands were 7-wire low relaxation type Y1860S7-15,7 fabricated by Arcelor Mittal. The applied code for pre-stressing steel material was prEN10138-3.

The anchors of the tendons are Freyssinet "C" Range Prestressing Anchorages with 55C15 trumplates. Trumplates are made of spheroidal ductile cast-iron grade EN GJS 500-7 according to French standard NF EN 1563.

Most of the tendon ducts are grouted with cement based grout, except the four pure vertical tendons fitted with dynamometers whose ducts are injected with wax. The cement grout complies with EN codes EN445, EN446 and EN447. The injection wax is Inject ELF CP-HF fulfilling ETAG013 requirements.

The interior of the Inner Containment was equipped with a 6 mm thick containment liner. The material of the liner was fine grained steel P275NL2 (EN 10028) fabricated by Rautaruukki Oy. The liner consists of the steel plates, the system of circumferential and meridian stiffeners and the liner studs. The stiffener profiles and the studs were embedded in the concrete. The anchoring system of the containment liner consists of L-profiles, which were welded onto the steel plate by continuous fillet welds, limiting a containment liner field. Within the containment liner field headed studs were welded. The material of L-anchors was S235JRG2 and headed studs were made of S235J2G3.

The anchor plates were made of carbon steel S235 in compliance with DIN EN 10025.

The sleeves welded into the containment liner and penetrating the inner containment wall were made of fine grained steel P275NL2 and P355NL2 (EN 10028).

The Reactor Cavity, Core Internal Storage and Transfer Compartment and Instrumentation Lances Storage Pools in the Reactor Building were equipped with pool liner, following stainless steel types were used in the fabrication

- martensitic stainless steel bars X20Cr13 (1.4021) QT700
- martensitic stainless steel X17CrNi16-2 (1.4057) QT800

- austenitic stainless steel X6CrNiTi18-10 (1.4541)
- austenitic stainless steel X6CrNiNb18-10 (1.4550)
- austenitic stainless steel X6CrNiMoTi17-12-2 (1.4571)
- austenitic stainless steels with steel grades A2, A3, A4 and A5 and strength grades 50, 70 and 80 for connecting elements
- stainless steel 1.4301 for headed standard studs
- for heavy loaded anchors non-alloy steel (1.0038, 1.0570) was used instead of the stainless steel

The concrete surfaces of the IRWST- tanks (In-Containment Refuelling Water Storage Tank) are covered with 5 mm thick austenitic steel liner (1.4571). Liner plates are welded to stainless steel supports (U-profiles 1.4571 and headed studs 1.4303) , which are fixed to the concrete with ferritic steel anchors.

7.1.2 Ageing assessment of concrete structures

This NAR describes the ageing assessments for each NAR example i.e. Loviisa 1 and 2, Olkiluoto 1 and 2 and Olkiluoto 3.

This NAR describes the outputs from the ageing assessment including the following:

a) Ageing mechanisms requiring management and identification of their significance;

In describing these outputs, the NAR explains how the following are used:

• Key standards and guidance used to prepare the SSC specific AMP, including a list of the main documents;

• Key design, manufacturing and operations documents used to prepare the SSC specific AMP;

• R&D programmes, by describing the objectives, contents and results of programmes managed by:

o Licensees;

o Industry or other relevant bodies;

• Internal and external operating experience, by describing why and how these experiences have been taken (or not) into account.

The safety related concrete structures in Finnish nuclear power plant and the degradation phenomena in those structures were identified in the report VTT-R-02323-08 [Vesikari 2008] reports under the auspices of SAFIR 2010 research programme.

NPP units Loviisa 1 and 2

Ageing mechanisms

Following concrete structures in Loviisa 1 and 2 containments and reactor buildings , their ageing mechanisms and identification of their significance to the plant safety were listed in SAFIR-research programme in the report VTT-R-02323-08 [Vesikari 2008]:

Aging mechanisms of concrete structures supporting the steel containment:

- T-weld of the anchor ring at level +9,60, possible lamellar tearing
- Ring layer +9,60, corrosion of the pre-tensioned fastening bolts of the steel containment thru the concrete floor slab structure, service-life maintained with inspection programme, considerable effects to the plant safety if periodic inspections and change of bolts are not conducted.

- Ring layer +9,60, aging of the slip bearing (glass fibre reinforced polyester structure), damages caused by thermal movement of the structures above and below the bearing, considerable effects on safety.

- Columns, walls, outer cylinder of the reactor pit, water leakage, cracking causing corrosion, moderate effect on safety.

Aging mechanisms of the inner concrete structures of the steel containment:

- Inner cylinder of the reactor pit, reactor pit and supporting area of the reactor, degradation of concrete due to elevated temperature and irradiation, considerable effects on safety.

- Pool structures, leakage of the pool liner due to corrosion, moderate effect on safety.

- Level +10,50 surface slab, damage in the floor surfacing and immersion of the leakage water into concrete causing corrosion, minor effect on safety.

- Area of the principal circuit valve, locally high temperature, cracking of concrete, minor effects on safety.

- Concrete structures of the ice condenser and crane wall , moisture and temperature variations, cracking of concrete, minor effects on safety.

- Columns and wall of the evaporator room and level +25,40 slab, , temperature variations, cracking of concrete, minor effects on safety

- Cylinder wall R=9550 (radiation shield) made partly of magnetite concrete, irradiation and high temperature, degradation of concrete.

- Level +19,30 (level of the evaporator support), variation in temperature, cracking of concrete, minor effects on safety

Aging mechanisms of the reactor building outer shell:

- Concrete cylinder wall cladded with corrugated steel plate, moisture stress, degradation and cracking of concrete, frost damage, corrosion of reinforcement, minor effects on safety.

- Rock anchors of the cylinder wall, reduction in alkalinity around rock anchors and corrosion, reduction in anchorage capacity, minor effect on safety.

- Roof supported by steel arches and frames, corrosion, loss of the roof tightness, minor effects on safety, diminishes the pressure between the outer shell and steel containment.

- Part of the cylinder wall without cladding, weathering, frost attack on concrete, corrosion of reinforcement

NPP units Olkiluoto 1 and 2

<u>Ageing mechanisms</u>

Following concrete structures in Olkiluoto 1/2 containments and reactor buildings , their ageing mechanisms and identification of their significance were listed in SAFIR-research programme in the report [VTT-R-02323-08, Vesikari 2008]. The aging mechanisms presented below are gathered either from the VTT report or from the information given by the licensee:

<u>Ageing mechanisms certain structures of the Containment</u>

Basemat on rock, top surface outside steel liner, tendon gallery

- Bottom surface, chloride initiated corrosion of reinforcement, cracking and spalling of concrete, only minor effects on safety

- Top surface outside steel liner, tendon gallery; corrosion of prestressing tendon anchors, reduction in prestressing force, reduction in pressure bearing capacity of containment

- Top surface inside the steel liner, corrosion of steel liner, reduction in leak tightness, radioactive leakage during accident

In-situ cast bottom parts inside and outside steel liner

- Corrosion of steel causing liner reduction in leak tightness, radioactive leakage during accident

- Leakage through the structure to the dry well, degradation of concrete, risk on the load bearing capacity of the inner cylinder wall

- High relative humidity of concrete (OL1), corrosion of steel liner and reduction in leak tightness

Pre-stressed, slip-form cast outer cylinder wall

- Corrosion of prestressing tendons, reduction in prestressing force (losses), reduction in pressure bearing capacity of containment due to larger deformations than expected

- Relaxation of prestressing tendons and shrinkage and creep of concrete, reduction in prestressing force, reduction in pressure bearing capacity of containment due to larger deformations than expected

- Corrosion of steel liner reduction in leak tightness, radioactive leakage during accident

- Repair of fire damages by shotcreting (OL1), detachment of shotcrete layer, risk on load bearing capacity and leak tightness of the outer cylinder wall

Slip-form cast inner cylinder wall

- Leakage through the structure to the lower dry well, degradation and cracking of concrete, risk on the loading capacity of the inner cylinder wall

Biological shield

- Inner surface, cracking of concrete, reduction in load-bearing capacity, reduction in the stability of the reactor support

- Outer surface, irradiation and high temperature, degradation of concrete, reduction in the stability of the reactor support

Roof structure and non-prestressed pool structures

- Corrosion of pool liners, reduction in leak tightness, radioactive leakage during accident

- Leakage of pool liners, degradation of concrete, only minor effects on safety. Wet-well pool leakage can be a problem for containment liner corrosion. Pool leakage can also start concrete degradation.

Intermediate floor

- Upper and lower surface, cracking of concrete, , reduction in leak tightness, reduction in leak tightness, risk on pressure suppression during accident.
- Leaktightness reduction: moving joint polymeric seal, transport joint polymeric seal, electrical penetrations and vicinity concrete cracking, door polymeric seals

Prestressed pool structures

- Relaxation of prestressing tendons, reduction in prestressing force, risk on displacements and cracking and leakage in pool structure
- Creep of concrete, reduction in prestressing force, risk on displacements and cracking and leakage in pool structure

- Load changes like temperature changes
- Gate operations

Containment and reactor building different vertical movement

- Vertical movement joints were too tight, they have been enlarged

Anchoring areas for prestressing cables

- Concrete cover cracking due to concrete shrinkage, risk for corrosion

Airlock

- Seal material ageing

Liner corrosion

- Construction time imperfections, concrete cleanness, possible problems due to liner surface organic impurity (gloves, thermal insulation etc.)

NPP unit Olkiluoto 3

<u>Ageing mechanisms</u>

Following description of the relevant aging mechanisms in Containment and Reactor Building of Olkiluoto 3 nuclear power plant is based on the AREVA report PESS-G/2011/en/1104.

<u>Leading Ageing Mechanism, Creep and Shrinkage of the Concrete and Relaxation of the Pre-stressing Steel</u>

With respect to the load bearing function of the Inner Containment concrete wall and the leak tightness function of the liner the leading ageing mechanisms are creep and shrinkage of the concrete in connection with the relaxation of the pre-stressing steel.

The creep and shrinkage of the concrete together with the relaxation of the pre-stressing steel lead to a decrease of the pre-stressing force and a decrease of the concrete compression stress. If this time dependent process would last unlimited it would finally cause tensile stresses and cracks in the concrete under certain design load conditions.

<u>Chemical Attack and Carbonation of the Concrete</u>

As the Outer Containment of Olkiluoto3 Reactor Building provides a comprehensive protection against detrimental weather conditions during the NPP lifetime, chemical attack and carbonation of the concrete therefore could take place only for short time. These mechanisms can't cause a relevant ageing of the concrete structure.

<u>Chemical Attack and Corrosion of the Reinforcement Bars and the Prestressing Steel</u>

Chemical attack and corrosion of the reinforcement bars and pre-stressing steel have been limited by suitable provisions in fabrication of concrete and injection grout:

- the chloride and sulphate content of the water and of admixtures in concrete and injection grout is limited
- the access of water and air to reinforcement bars and pre-stressing steel is averted
- after concreting during the containment lifetime. Possible chemical reactions are stopped shortly after concreting, because the reservoir of water and air is limited at the considered locations.
- air bubbles or voids between the grouted pre-stressing strands are impeded by properly injection procedures which have been approved by full-size mock-ups on curved tendon ducts before the application on the Inner Containment wall.
- chemical attack (sulphate) on the basemat is taken into account by using sulphate resistant (70 % ggbs)-slag concrete in the basemat and by protecting the underground boundaries of the basemat with bitumen coating.

<u>Corrosion of the Pre-stressing Anchorage Steel Parts</u>

- Corrosion of the pre-stressing anchorage steel parts is prevented by the installation of grouting caps and injecting of grout inside the caps after pre-stressing the tendons.

<u>Ageing Mechanisms for the Liner</u>

- Because the containment liner is connected rigid with the concrete wall, the relevant ageing mechanism for the concrete wall is linked to the containment liner regarding the leak tightness function. The liner is performed with a grid of studs and stiffeners which are embedded in the concrete of the containment wall. The deformation of the concrete wall due to creep and shrinkage is directly transferred to the liner plate by the liner studs and the stiffeners.

- At the beginning of the NPP lifetime the liner is already under compression due to the prestressing of the Inner Containment wall. The strains deriving from creep and shrinkage of the concrete increase the negative strains of the liner. The resulting negative liner strains can cause a blistering of the liner. Although the leak tightness of the liner is ensured even with blisters this deformation is regarded as an ageing mechanism.

- Liner corrosion is avoided by the coating both surfaces of the liner. The liner coating is not part of the AMP, because the removal and replacement of the coating is anyway in scope of the maintenance programme.

- Below the level 4.30 m the liner is embedded in the concrete and there is coating at both surfaces of the liner, replacement of coating or maintenance inspections are not possible.

7.1.3 Monitoring, testing, sampling and inspection activities for concrete structures

The NAR should describe the monitoring, testing, sampling and inspection activities for each specified element for each NAR example.

The monitoring, testing, sampling and inspection activities performed by the licensee and activities performed by 'third party certification organizations' should be described including:

a) Description of activities;

b) Frequencies;

c) Acceptance criteria.

In describing these outputs, the NAR should explain how the following are used:

- Programmes for monitoring and trending including test methods available for use in performing inspections;
- Key features of the inspection programmes;
- Surveillance programmes where appropriate;
- Inspection history identifying trends and progressive deterioration;
- Any provisions for identifying any unexpected degradation.

NPP units Loviisa 1 and 2

Description of monitoring and inspection activities in Loviisa 1 (LO1) and Loviisa 2 (LO2) nuclear power plants is based on the following guides of the licensee (Fortum); Y-05-00005 "Periodical civil engineering inspections".

<u>Pressure and leak-tightness tests</u>

Pressure test to the whole steel containment has been conducted twice during commissioning phase. The test pressures were 85,8 kPa in LO1 and for 78,9 kPa in LO2 containment. In LO1 and LO2 NPP leak tightness of the steel containment has been verified with pressure tests with design pressure (1,7 bar abs) every 4th year, after year 2015 the test interval has been increased to 8 years. . Acceptance criteria of the leakage rate for LO1 and LO2 is 0.2 Vol-% per 24 hours.

<u>Periodical inspections</u>

Guide Y-05-00005 defines those reinforced concrete structures and steel structures, which are under the scope of periodical civil engineering inspections and describes the inspection activities for each structure.

Inspections are conducted as visual inspections using also NDE-methods. Inspections are conducted as a general inspection for each building according to the inspection cards and objectives and criteria given in the Appendices of the Guide Y-05-00005. For safety critical structures inspection are conducted once a year, other structures every second year.

Cracks that are critical for safety, aging or structural integrity are recorded in the inspection cards. Damages which need special monitoring are photographed, measured and marked on the structure.

Classification of inspection findings in NPP units Loviisa 1 and 2

Inspection findings of are classified based on the criticality of the SSC to the nuclear and radiation safety and the urgency of remedial actions. The primary digit of the damage classification code (for example A1) describes the effect to the nuclear and radiation safety:

- Class 0 Critical structure to the nuclear and radiation safety (load bearing structures of the reactor, radioactive fuel pools etc.).
- Class A Critical structure to the safety, limiting plant life.
- Class B Structure has either a high importance to heath and safety or is an object to chemical or heat attack radioactive contamination etc.
- Class C Structure or its functionality and damage do not have an immediate effect to the safety.

The secondary digit of the classification describes the urgency of the remedial actions:

- 0: A serious damage which threatens load bearing capacity, stability and structural integrity of the structure
- 1: A serious damage which does not prevent the use of power plant unit. The damage is situated so that it threatens health and safety or the progress of the damage may cause a loss of load bearing capacity, stability and sustainability of the structure
- 2: A significant damage but the progress of it has only a minor effect to the load bearing capacity, stability and structural integrity of the structure
- 3: The damage is minor and the progress of the damage is slow
- 4: Remedial actions increase health and safety, decontaminability or otherwise improve structural functionality.

The classifications and colour code used in the inspection reports are following:

- 00: A serious damage which prevents the use of power plant unit, has to be repaired before the end of the outage (red)
- A1: Must be repaired immediately, structure is critical to safety (red)
- B1: Must be repaired immediately, damage that needs immediate remedial actions (red)
- A2: Must be repaired in the near future, structure is critical to safety (orange)

- B2: Must be repaired in the near future, damage that needs remedial actions (orange)
- B3: Must be repaired in due time (yellow)
- C1: No repair needed, progress of the damage Is monitored (green).

In the Appendices of the Guide Y-05-00005 inspection targets and criteria are given. For example in the Appendix 4 the following criteria for the inspection of the reactor building are presented:

Structures below level 9,60:

- Reinforced concrete slab at level +9,60, visual inspection of upper and bottom surface cracks and other damages are recorded, concrete around penetrations is inspected, slab's connection to the outer cylinder wall must be intact, possible cracks are recorded.
- Load bearing outer cylinder of the reactor pit from level +9,60 to the rock surface, possible cracks are measured and recorded.
- Columns and walls below level +9,60, possible cracks are measured and recorded.
- Equipment hatches 1R0301(LO1) and 2R0301 (LO2), pressure bearing structure, possible cracks in the concrete around the hatch are measured and recorded.
- Emergency water pools (boron water pools) bottom slab and walls, possible marks of leakage are recorded

Inner structures of the reactor building:

- Pressure resistant doors and hatches, the functionality of the doors and hatches are checked, and the condition of the sealants are visually inspected, functionality of the locking of reinforced concrete hatches locked with bolts is inspected
- Reinforced concrete and steel structures (functionality), general inspection of cracks on all concrete surfaces, over 0,2 mm wide cracks are recorded, connecting steel beam of the bottom structure of ice condenser and slab on the level +19,30 is inspected
- Heavy components; ring beam of the polar crane is visually inspected, condition of the concrete around the steel supports of the main coolant pumps is inspected. Concrete structure around the openings on the concrete slab is inspected.
- Outer cylinder wall and the roof structure; the roof structure and its connections to the cylinder wall are inspected visually. Damages causing deterioration the steel beams or their connection to the cylinder wall or increasing the leak tightness of the structure are recorded (rust of the steel structures, location and condition of neoprene slabs, condition of bolts, possible water leakage)

Expansion/movement joints

- Elasticity and integrity (air pressure, leak tightness) of the expansion joint sealants and the condition of concrete structure around the expansion joints are inspected (on the level +10,50, radius R= 4800 mm and on the level +25,40 radius R=4500 mm). Damages of the expansion joint sealants are classified either “2, repair in a due time” or “3, no urgent need for repair”.

NPP units Olkiluoto 1 and 2

According to the AMP Guide 107835 of the licensee description of monitoring and inspection activities in OL1 and OL2 NPP units is based on the following AMPs and guides of the licensee (TVO);

Deformation measurements

- AMP113016 Strain measurements of the containment
- AMP104730 Deformations of the containment,

Temperature measurements

- AMP111402 Temperature measurements of concrete structures
- AMP 111407 Temperature measurements of the turbine foundations

Cracking surveillance

- AMP104733 Surveillance of the cracks of the containment
- AMP 111407 Surveillance of the cracks of the turbine foundations

Leak and moisture detection

- AMP111409 Containment concrete humidity follow-up
- AMP 111555 Leakage detection of fuel and wetwell pools

Leak-tightness tests of the containment

- AMP104533 Leak-tightness tests of the containment

Surveillance of the containment materials

- AMP111556 Surveillance of the properties of the prestressed tendons
- AMP 104756 Surveillance of the condition of concrete in containment
- AMP 112739 Surveillance of the rubber seal materials of the containment

Periodical inspections of the buildings

Periodical inspections and measurements are concentrated in buildings and structures, that are important to the use and safety of the NPP´s. These buildings and structures are for example; the containment, reactor building, turbine foundations, seawater channels, pool structures and surfaces of the radioactive rooms.

Following items are inspected and recorded:

- crack mapping, cracks of the concrete structures
- moisture and corrosion damages
- malfunction of different types of steel structures for example doors, hatches, penetrations and anchorage plates
- changes in the quality, sustainability and colour of the paintings
- condition of facades and roof structures
- water leakages in the underground spaces

The correct functionality of buildings and critical structural members is ensured with inspections and measurements done yearly. Inspection and measurement results are gathered, analysed and reported. When needed remedial actions are started.

<u>Frequency of the inspections, graded approach</u>

Frequency of the visual inspections of the buildings is graded according to the importance of the building, for example; containment is inspected every year, reactor building, turbine building and waste building every second year, main control room every fourth year.

In the maintenance programme AMP103459 inspected rooms are classified according in a following way (inspection frequency is highest in the highest Class);

- Class 1S rooms in the containment
- Class I A spaces with high radioactivity on the ground floor, where is lots of maintenance work
- Class II A other spaces of high radioactivity, where frequent maintenance work is needed
- Class II B, spaces with rather low radioactivity (“green spaces”) or uncontrolled spaces, where frequent maintenance work is needed
- Class IV B, other ”ordinary” controlled spaces like switchgear or I&C cabinets, ventilation shafts and rooms, corridors, staircases
- Class E spaces with constant radioactivity, like radioactive shafts

Frequency of the inspections is dependent on the rapidity of the damage, importance of the space or structure. The condition of the rooms (spaces) is controlled in one , two or four years cycles depending on the importance of the space.

In the Appendices of the AMP103459 inspection targets and criteria are given. For example in the Appendix 3 the following criteria for the inspection of the containment (Class S , frequency once a year) are presented. Following items are checked;

- Load bearing and other important structures (cracking, spalling, deformations)
- Surfaces of the spaces (cracking, flaking), acceptance criteria:
    - surfaces shall be clean and shiny
    - no cracks are allowed
    - adhesion of the painting (tensile tests)
- Movement joints (cracking, hardening)
    - movement joint materials shall be in perfect condition
- Penetrations (cracking, spalling, leak tightness)
    - fire resistant penetrations
    - water tight penetrations
    - pressure resistant penetrations
- Steel platforms and –stairs (deformations, fixings)
- Doors of the wetwell, locking and seals

Current monitoring system of the NPP units Olkiluoto 1 and 2 containments

In the middle of the -90's containment structural monitoring was updated in Olkiluoto 1 (OL1) and Olkiluoto 2 (OL2) nuclear power plants. Concrete humidity sensor system and displacement system (mechanical clocks) between reactor building and containment structure were installed. Later also "eddy current" basis movable deformation monitoring system between containment inner and outer structures were installed, which are used during pressure tests. Concrete crack mapping was started as a part of the visual concrete surface inspection according to ASME XI.

In OL1 and OL2 displacements between the containment and reactor building frame and displacement between the middle floor slab and containment wall are monitored with measurements. Displacements are recorded once a year during outages. During leak tightness tests displacement are recorded before the test, during the test in maximum pressure and after the test. Displacement measurements are recorded using mechanical clocks (10 pcs). During leak tightness test displacements between the middle floor slab of the containment and roof top of the containment are measured constantly with "eddy current" basis movable deformation monitoring system.

Reinforcement strains in the containment are monitored. OL1 and OL2 containment original structural monitoring includes 26 strain gauges and temperature probes, which are fixed to reinforcement. Strain gauges are read once a year. During leak tightness test the strain gauges readings are recorded on pressure levels 100, 150, 250, 350, 400, 350, 250, 150, 100 kPa abs.

Temperature of the concrete structures of the containment are monitored using embedded thermocouples. Standard thermometry devices (10 pcs) are used to measure ambient temperature inside and outside the containment. Concrete temperatures are measured with 13 thermocouples, which are embedded in the concrete in the vicinity of the strain gauges. Temperatures are measured once a year during outages or during leak tightness tests at the same time as strains are recorded.

Concrete cracks are inspected once a year during outages and leak tightness tests. The purpose of the inspection is to observe possible new cracks and monitor the development of the existing cracks (length, width). During normal outages all concrete surfaces are inspected and the existing cracks which are under surveillance are analysed. During leak tightness test concrete surfaces are inspected before pressurization, during maximum pressure and upon completion of depressurization.

Acceptance criteria of the cracks is based on the Finnish Concrete Code BY50 [BY50,2012]. Exposure class of the containment is XC3. Requirements for the cracking of concrete are given in BY50 table 2.16a; cracking shall be limited to 0,2 mm under long-term loads. When cracks widths are over 0,20 mm, repair plan is done, if danger for corrosion of rebars or tendons is expected.

Containment prestressing forces have been tested and analysed with 3 m long test beams every five years during first 25 years.

Concrete quality has been tested with original test cylinders every five years during first 25 years and after that by taking core drills and with NDT methods.

All containment water pools include water leakage monitoring system. Visual concrete surface inspection has been performed yearly.

<u>Future monitoring system of OL1 and OL2 containment</u>

It is not possible to direct measure prestressing forces due to cement grouted tendons.

U:S. Regulatory Guide 1.90 [new Draft Regulatory Guide DG-1197, April 2011] part B gives an alternative way to determine prestressing forces for grouted tendons; “Measure containment deformations during pressure test and calculate prestressing forces according to displacements.” This method has been used in OL1 and OL2 first time and since 2002 (15 years ago). Displacement between containment and reactor building were measured with measurement clocks in the movement joints and deformations were studied using a finite element model of the containment structure.

Two kinds of FE-models were used;

- linear model, where half of the containment was modelled (180 degree model)
- wedge model that utilized non-linear reinforced concrete elements with capability to model concrete cracking and non-linear reinforcement response. FE-programme ANSYS was used in the analysis with solid65 elements.

The licensee (TVO) is updating containment deformation measurement system with Telemac pendulum type of measuring devices, which can measure also global horizontal and vertical movement in three levels. Test pendulums were installed year 2017 in OL2 before the outage. Measured deformations during leak tightness tests can be used to confirm the behaviour of the containment in more precise way.

Non-Destructive Examination (NDE) methods together with the destructive tests are also needed in the future to confirm containment condition. The licensee (TVO) has an AMP for destructive and NDE tests of the concrete structures of the containment. (TVO Guide 104756). The purpose of the AMP is to find out the effect of aging to the properties of the concrete structures of the containment. Another purpose of the programme is to confirm that the prestressed structure is behaving as designed. The earlier AMP of the licensee (0-TC-O-10/97) was purely based on the testing of core drills. New programme was started 2005-2006, when preliminary NDE tests with different methods were conducted 2005 and destructive tests were conducted during 2006. Based on the results of the tests, the new programme (104756) was drafted. The NDE research has been repeated first time 2010.

TVO participates the research programme SAFIR 2018 and the Wanda project, which is concentrated in concrete NDE methods investigations.

TVO also participates in Energiforsk research programmes together with Swedish utilities.

NPP unit Olkiluoto 3

Following description is based on AREVA reports PECC-G/2014/en/1023 "In-service Inspection Plan Civil Structures, 30UJA Building – Inner structures" and NGPM2/2004/en/0236 "In-service Inspection Plan Civil Structures Containment". The guides of the licensee TVO for OL3 are still under preparation and will probably unified with the guides for OL1 and OL2.

The previously mentioned In-service Inspection (ISI) plans comprises principal rules for periodic testing and inspection of the containment building structures of the Olkiluoto 3 Power Plant unit.

Containment structural behaviour under pressure is studied in two type of tests; in the initial structural integrity test (ISIT) and in the ISI leakage test. The ISIT during commissioning of OL3 nuclear power plant has been conducted.

Before the ISIT-test first overall leakage test (type A) was conducted. In the leakage test the pressure was increased up to the design pressure (5.3 bar absolute). Acceptance criteria of the leakage rate for OL3 was 0.5 Vol-% per 24 hours. The initial structural integrity test ISIT was performed after the determination of the leakage rate. After the initial structural integrity test (one hour in 1,15 times the containment design pressure) leak tightness test was repeated second time.

Following monitoring and inspection activities have been performed for inner containment structure in connection of ISIT:

- Measuring of local strains in concrete by means of embedded acoustic strain gauges C110. In the area of the cylindrical shell the measuring points are located at 3 different vertical axis (at opening angle of 67°, 204°, 294°) at 5 different altitudes (-2.3 m,+2.6 m, +10.0 m, +23.0 m, +38.0 m). In the dome part the measuring points are located at 3 opening angles (67°, 204°, 294°), each at 2 levels +50.91 m, +56.43 m and in the centre of the dome. Furthermore in the base mat and areas around the larger penetrations additional strain gauges have been implemented.

- Temperature measurements for the calibration of the strain gauge measurements. At each location of a set of three strain gauges (radial, tangential, orthogonal to this plane) there is one temperature probe implemented for the measurement of temperature in concrete.

- The elastic deformation response of the containment has been monitored in order to provide a basis for the assessment of the integrity and functionality of the containment structure. The global displacements of the containment shell are measured with direct pendulums, placed in 3 different axes (at opening angle of 67°, 204°, 294°) and installed at 3 different levels (+10.0 m, +23.0 m, +38.0 m). In addition, the global displacements of the dome area have been measured with laser.

- Monitoring of the prestressing force by dynamometers installed at the upper end of the four vertical tendons spaced out regularly over the circumference of the cylindrical shell.

- The moisture of the concrete shell is measured by three concrete elements stored in the annulus with similar moisture conditions as valid for the inner containment cylindrical concrete shell.

- Monitoring of containment liner strains with a system of optical strain gauges.

- Visual surface inspections and crack mapping of the inner containment concrete shell. Crack bridging transducers have been used for monitoring of crack width development during the pressure test.

- Visual surface examinations for the containment liner and penetrations, according to ASME Section XI.

Periodical ISI leakage rate test under pressure of 4.9 bar absolute will be performed every 5 years of operation (respectively 3 times in 15 years). Most of the measurements which were performed during the ISIT shall be repeated during ISI to get valid comparison values due to the elastic behaviour of the complete concrete structure. This indicates the stage of ageing of the containment.

Besides use for the tests with overpressure, the monitoring system is regularly used to follow long term properties of containment structure. The measurements during and between the tests shall supply the necessary data to verify that the safety margins provided in the design of the concrete structures have not been reduced as a result of operating and environmental conditions.

The inspections of containment internal structures are mainly visual. The regular inspections will start after the plant is in operation. The proposed intervals (varying from 1 to 5 years depending on the target) for the concrete surface inspections and other items of interest are listed in report PECC-G/2014/en/1023. Inspection subject are given in the following Table.

Subject	**Comment**
Condition and quality of the original concrete surfaces	Subject of inspection are the locations of concrete surface damage, e. g. caused by erosion of abrasion.
Condition and quality of repaired concrete surfaces	The kind of repair procedure has to be considered.
Development of existing concrete cracks	Crack propagation can be caused by loading or by concrete ageing phenomena. There is a special inspection area, described below this table.
Identification and description of new concrete cracks	New cracks can be caused by loading or by concrete ageing phenomena.
Abrasion of painting and coating	All concrete surface conditions and requirements are based on [4] and defined in the room book [5]. The functionality has to be checked and repair measures initiated, if required. The repair measures shall follow reference [6].
Functionality of doors and door anchorages	The subject of inspection is the indication of concrete and steel structure deformation. Door position drawings are defined in the room book [5]. The door supplier has provided a corresponding door maintenance report, which shall be followed. In case of damages to the painting, the documents [7], [8] and [9] shall be taken into account.
Joints between structural parts	In Inner Structures joints exist between precast elements. Inspection subjects are the gap widths. It shall also be ensured that the bolt connections of pre-cast wall elements are not loosened.
Pool liner surface	Inspection subject are possible corrosion as well as steel plate deformation.
Steel platform anchorages and connections	Inspection subject are possible corrosion as well as steel plate deformation.
External circumstances	E.g. temporary/accidental loading or fire.

Furthermore, several investigation programmes have been performed for OL3 concrete material. For example, creep and shrinkage, strength development and E-modulus of the concrete mix used in the containment have been investigated. Further samples for the possible testing during service lifetime of 60 years have been taken and stored in the prestressing gallery and the reactor building. In addition 36 special locations for drilling core specimens have been marked on the dome structure.

Inspection history of NPP unit OL3

The Initial Structural Integrity Test (ISIT) of the inner containment structure of Olkiluoto 3 has been conducted. The test started after a subpressure test phase on 2014-01-31 and ended on 2014-02-14. The maximum inner pressure was 6 bar absolute or 5 bar relative.

Measurements were preformed before, during and after the pressure test. Target values for pendulum displacements and strains of acoustical strain gauges were computed using FE-analysis by the Detail Designer of the containment. Besides the target values, also lower and upper limits have been determined. The limits for the displacements followed the criteria given in ASME Section III, Division 2. For the strains respective limits were determined.

All measurement results were based on respective zero measurements immediately before the pressure test. The measured strains in the current cylindrical part and in the dome part of the containment wall were within the upper and lower limits according to ASME for pressurization as well as for depressurization phase. In the vicinity of the openings the measured strains and calculated strains deviated from each other a small amount. Also the displacements measured by pendulums were in the limits according to ASME and the behaviour of displacements was linear both for pressurization as well as for depressurization phase.

It was concluded that the elastic behaviour as well as the integrity of the Inner Containment was ensured.

7.1.4 Preventive and remedial actions for concrete structures

The NAR should describe key preventive and remedial actions that have been identified for each NAR example.

Description should include:

- Criteria for taking actions;
- Procedures for taking actions;

Description of the actions to be taken.

NPP units Loviisa 1 and 2

Preventive actions

Inspections of the neoprene bearings between the steel girders and circular wall supporting the roof of the reactor building were conducted and follow –up of the deformations has been going on since 2016.

Research and inspection programme on radiation effects to the concrete (biological shield) is going on and specimens have been drilled.

The integrity of the paintings of the interior structures of the containment has been assured by repair and renovation works. DBA-tests and adhesion tests have been conducted for specimens.

In the dome part of the steel containment there have been damages of the coating caused by the water leakages of the roof structure of the reactor building. The damages were local and were repaired year 2011 as a normal maintenance work in order to prevent corrosion.

<u>Remedial actions</u>

There were a large study of the anchor plates and the capacity of drilled anchor bolts in Loviisa NPP units, a lot of the anchors had to be repaired.

The most critical spots in the steel containment are the T-welds of the anchorage ring and the mounting flange bolts. Steel containment mounting flange bolts have been inspected regularly according to inspection programme and no significant corrosion has been discovered. Outer shell mounting flange bolts will be replaced once during current licensing period according to existing replacement programme.

There has occurred several water leakages through the roof structure of the Loviisa 1 reactor building at recent years. The water insulation structure of the roof is a combination glass fibre and bitumen felts. There have occurred cracking and blistering of the glass fibre felts. Remedial actions have been designed on the year 2017. New water insulation layer will be installed and in the lower parts of the roof and all the water and heat insulation layers will be replaced with new layers. Water drainage sumps on the roof will be renewed and airtightness of the roof hatch will be assured with new sealing

NPP units Olkiluoto 1 and 2

<u>Remedial actions during construction period</u>

During construction phase of the OL1 and OL2 containments some construction errors have happened and results of the errors may have an effect on the aging of the structures. Some of the building non-conformances needed special studies and large remedial works. Such non-conformances and remedial actions were for example;

- In the cylinder part of the containment of Olkiluoto 1 unit there was some longer cracks that were closed, these cracks have been monitored during several decades. Some of them, when developing wider than criteria 0,2 mm, have been injected.

- There was a fire in the reactor building of Olkiluoto 1 unit during construction phase 8.2.1976. Fire caused damage to the upper part of the containment, large remedial works were needed and new structural analyses were conducted. Remedial works were done according to repair plans approved by STUK. According to the plans 1700 pcs of strengthening bolts (T25, A40, L=700) had to be installed in the concreting part 23, the thickness of the wall of the containment had to be increased during the repair work of the containment. There were also noticed some concreting errors near the liner. Based on the new structural calculations delivered to STUK it was concluded that the load bearing capacity of the structures was adequate.

<u>Other remedial actions</u>

Displacement monitoring system of Olkiluoto1 and 2 NPP´s was not initially exactly according to ASME Section III Div. 2 requirements. Measurement system devices were installed near the movement joints not in the positions where ASME requires. The reason for this has been partly the fact that the movement joints were rather tight and they had to be monitored. Using relative deformations and measurement results of the movement joints it has been possible to control how well the moving joints have been functioning. Finally the movement joints were so tight that they prevented movement. In order to guarantee the functionality, TVO finally had to make a repair plan and enlarge movement joints by cutting concrete around the joints.

Deformation measurements will be developed further with Pendulum-type of system. Installation of the Pendulum-system started 2017. When the measurement results of the Pendulum-system are compared to the FE-analysis results, better comprehension of the behaviour of the containment and pre-stressing system can be reached. With the pendulums, the deformation measurement system of Olkiluoto units 1 and 2 will satisfy the requirements of the standards ASME III Div.2. and ASME XI.

DBA-tests were conducted for the paint specimens drilled of the concrete walls of the OL1 and OL2 containments on the year 2009. Tests were part of a TBY research programme, in which common requirements and guidelines were developed for paintings inside containments in Nordic countries. Painting specimens of OL1 fulfilled the requirements but specimens of OL2 did not. As a result of the study inner surfaces of the OL1 and OL2 containments have been totally repainted 2010-2011 with a painting system that fulfilled the DBA-tests and the new painting programme AMP137614 of the licensee.

There have been several leakage cases of condensation pool (wet-well) of Olkiluoto 1 unit on the year 1996, 2003 and 2006 . Latest leakage of the steel liner of the condensation pool (wet-well) at Olkiluoto 1 was repaired on the year 2006 and joints of the upper boundaries of the liner of wet-well pool at the both units Olkiluoto 1 and 2 were also repaired.

NPP unit Olkiluoto 3

During construction phase of the OL3 containment and reactor building some construction errors have happened and results of the errors may have an effect on the aging of the structures. Following non-conformances and remedial actions were reported during construction;

- During construction on the year 2008 there occurred a fire in the annulus between inner and outer shell of the reactor building. The damage caused by the fire was actually rather minor and structures were repaired. During leak tightness tests (ISI-tests) the area of the fire damage is a special target of the visual inspection.

- During construction, cracks with considerable crack widths occurred at the reactor pit between elevations +1,50 m and +4,80 due to hydration heat. Temperature of the concreted area had been measured by temperature sensors. Non-conformance report and repair plan was written. Cracks were repaired by injection. Injected cracks are a specially marked target of the visual inspections of the inner structures of the Olkiluoto 3 containment during Leak-tightness (ISI) tests. In case of occurring new cracks in these marked areas or in the vicinity, the locations, the courses, the lengths and the widths shall be documented. All detected cracks at this special area with widths more than 0,3 mm shall be injected.

- Due to the fabrication technique of the liner (large prefabricated blocks) of Olkiluoto 3 unit the initial imperfections were rather large in the final product. There were also some welding errors done at the site, that needed to be repaired. The initial imperfections increase the buckling stresses in the liner during possible LOCA. That is why STUK demanded that the liner of the containment shall be equipped with monitoring devices i.e. additional strain measurement system.

7.2 Licensee's experience of the application of AMPs for concrete structures

NPP units Loviisa 1 and 2

Degradation mechanisms of steel containment and reactor building concrete structures have been well known since the start of plant operation.

Reactor building concrete structures have been inspected and monitored according to procedures and practices mentioned in Section 7.1.3 since 1978. Concrete structures important to safety are cast-in-place and there are no post-tensioned reinforced concrete structures in Loviisa plant. Supporting structures of RPV have been inspected from concrete material samples taken from both units when inspected containment steel liner condition. Based on the inspections both concrete and liner were in good condition.

No ageing related severe degradation have been detected in the containment concrete structures, steel containment or pool liners during the plant operation history.

Floor coatings are in good condition in reactor building and expansion joints have been refurbished based on the inspection results.

No changes have been done to the ageing management programme for containment structures based on the internal or external operating experiences.

Steel containment mounting flange bolts have been inspected regularly according to inspection programme and no significant corrosion has been discovered. Outer shell mounting flange bolts will be replaced once during current licensing period according to existing re-placement programme.

Steel containment and reactor building concrete structures periodic inspections, monitoring and testing is fairly comprehensive and unexpected degradation has not been identified.

NPP units Olkiluoto 1 and 2

OL1 and OL2 concrete containment in-service inspections (ISI) have been running during units operation. As a result of the ISI programmes status of the containment is controlled and ISI-programmes are in a way containment system "health" programmes.

<u>AMP104730 Containment deformation measurement between containment and reactor building</u>

Deformation measurements between containment and reactor building in OL1 and OL2 started the middle of 1990's. Measurement system has been operating well and TVO has collected 20 years information.

The reason why measurement system devices were installed near the movement joints has been the fact that the movement joints were rather tight. Using relative deformations and measurement results of the movement joints it has been possible to control how well the moving joints have been functioning. In order to guarantee the functionality of the movement joints, TVO finally had to make a repair plan and enlarge movement joints by cutting concrete around the joints.

As an example deformation measurements during containment leak tightness test 2016 in OL1 are presented in the Figures 7.1 below. TVO has developed together Energiforsk FE-model of the containment. Vertical measurement results (Figure 7.1) are almost the same as the calculated results (difference below 10%).

15,33	15,14	16,09	15,19			
Δt_{m0-m1}	Δt_{m1-x1}	Δt_{x1-x2}	Δt_{x2-m2}	Δt_{m2-x3}	Δt_{x3-x4}	Δt_{x4-m3}
128,25	9,97	6,37	0,92	11,15	6,17	3,08
k_1	k_2	k_3	x_1	x_2	x_3	x_4
-0,00148	0,15167	-0,28536	15,13	16,09	16,07	15,19

t [pvm hh:mm]	Siirtymä	Rostedt
23.5.2016	15,48	
31.5.2016	15,33	
5.6.2016 20:15	15,14	
6.6.2016 6:13	15,13	15,13
6.6.2016 12:35	16,09	15,97
6.6.2016 13:30	16,09	
7.6.2016 0:45	16,07	15,95
7.6.2016 6:55	15,19	15,19
7.6.2016 11:00	15,19	

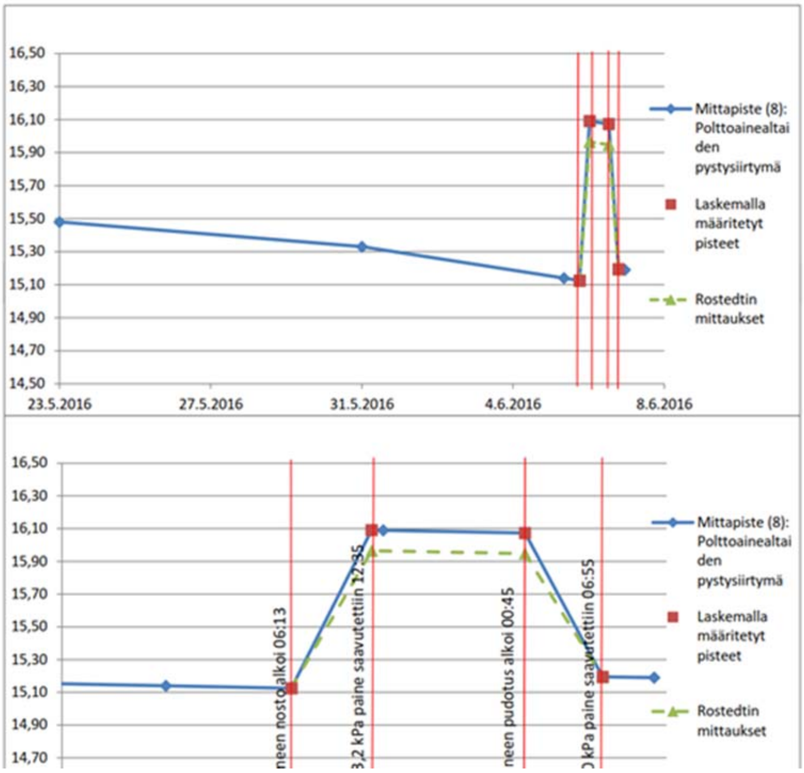


Figure 7.1. Measured vertical displacement of the fuel pools (blue line) and calculation results (red squares) during containment leak tightness test 2016 in OL1.

AMP104733 Containment concrete crack mapping

Concrete crack recording and mapping (width, length, in some case also depth) has been performed every year. The development of cracks is followed in order to find out if the width or length of the crack is increasing or staying constant. When needed developing cracks are injected. Crack mapping has been conducted according to ASME Section XI.

AMP104756 Containment concrete material follow-up

This AMP concrete material follow-up was started with test samples and continued with drilling cores and NDE tests. The NDE investigations have involved following methods:

- IE – Impact Echo

- SASW – Spectral Analysis of Surface Waves

• MASW – Multichannel Analysis of Surface Waves

<u>AMP111409 Containment concrete humidity follow-up</u>

Concrete humidity measurement started 20 years ago and it seems that no changes has happened. Humidity sensors monitor also leakages in the pools and gives reliable information for the concrete humidity and shrinkage. An example of temperature and humidity follow-up results during 2016 in OL1 is shown in Figure 7.2 below.

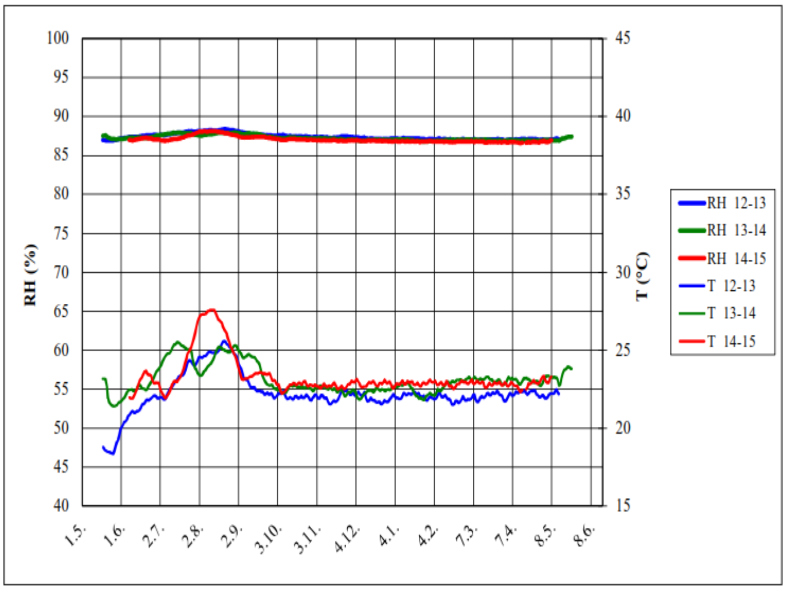


Figure 7.2. Temperature and humidity during 2016 in OL1.

<u>AMP111556 Containment prestressing steel follow-up</u>

In OL1 and OL2 prestressed test beams have been used to study the behaviour and give information of the prestressing tendon forces and corrosion rate.

In OL2 pendulum type of displacement measurement devices will be installed starting year 2017. With the pendulum measurements containment global deformations can be recorded in X-, Y- and Z-directions during leak tightness tests and in the long run. Pendulum-system can be used also in the prediction prestressing tendon forces. In order to succeed in this task, results of an accurate FE-model of the containment are needed.

According to the results the pendulum measurements are rather the same as the results of the measurement clocks and FE- analysis. An example of results showing deformations during pressure test in OL2 (2017) is shown below (Fig. 7.3).

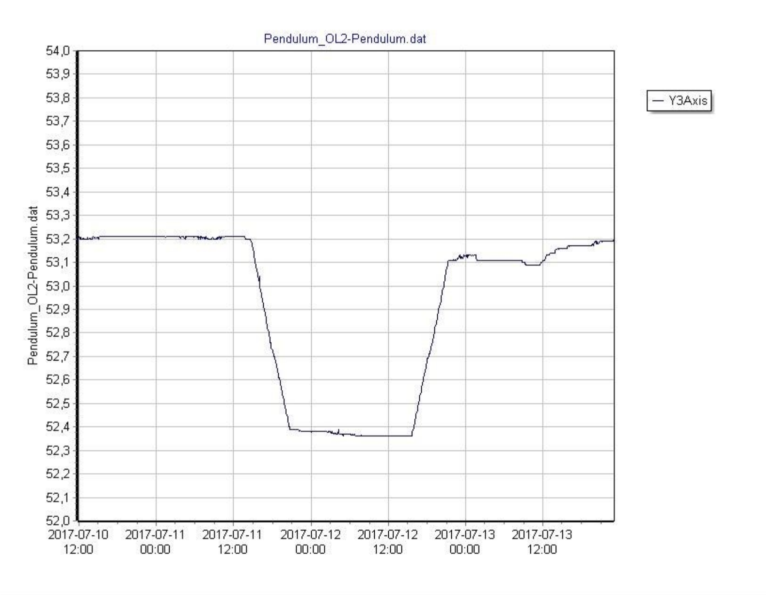


Figure 7.3 Horizontal displacement in the middle height of the cylinder wall of the OL2 containment. measured by pendulum installed 2017.

<u>AMP113016 Containment concrete strain gauges</u>

Strain gauges are installed to containment reinforcement in 26 positions. Strain gauge measurements are done during normal operation and pressure/leak tightness tests. An example of the strain measurement results during leak tightness test is shown in Figure 7.4 below. It can be seen that the strains in rebars are reversible and elastic.

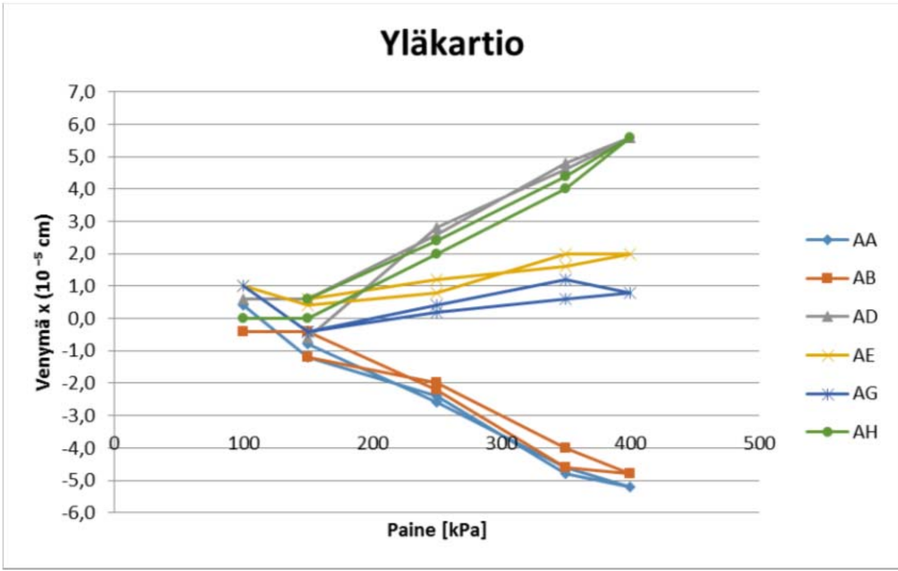


Figure 7.4 Strain in the upper gusset in the corner of the roof and circular wall as a function of pressure. Leak tightness test of OL2 year 2017.

NPP unit Olkiluoto 3

Some experiences from the most relevant AMPs for OL3 inner containment have been collected although the life cycle of the plant is only in the beginning.

Monitoring of the long term behaviour of the prestressed containment structure:

As stated above, with respect to the load bearing function of the inner containment concrete wall and the leak tightness function of the liner the leading ageing mechanisms are creep and shrinkage of the concrete in connection with the relaxation of the prestressing steel. The creep and shrinkage together with the relaxation lead to a decrease of the pre-stressing force and a decrease of the concrete compression stress. If this time dependent process would last unlimited it would finally cause tensile stresses and cracks in the concrete under certain design load conditions.

Therefore the long term monitoring of the loss of concrete compression stress can be seen as one of the most relevant AMP for OL3 containment. The monitoring is done by strain gauges and temperature probes embedded in the concrete. Additional information from relaxation of prestressing steel can be received through dynamometer measurements. The monitoring has been started at the time of post-tensioning in 2010 and it would be continued until the end of lifetime.

The measurement results are regularly compared to theoretical curves which have been calculated according to the formulas and assumptions used in the detailed design calculations of the containment, see example in the Figure below. Further investigation and/or assessment would be necessary if the measured losses would rise higher than the calculated ones.

So far the measured strains have stayed below the calculated values. The developments of the measured concrete strains also indicate that the long term behaviour of the prestressed containment structure correlates well to the calculations.

Another relevant AMP application is the monitoring of the behaviour of the prestressed containment structure during the tests with overpressure loading (ISIT, periodical ISI). The monitoring is done by strain gauges and temperature probes embedded in the concrete (concrete strains) as well as by pendulums (displacements) and by other instruments such as dynamometers (tendon force) and optical strain gauges (liner strains)

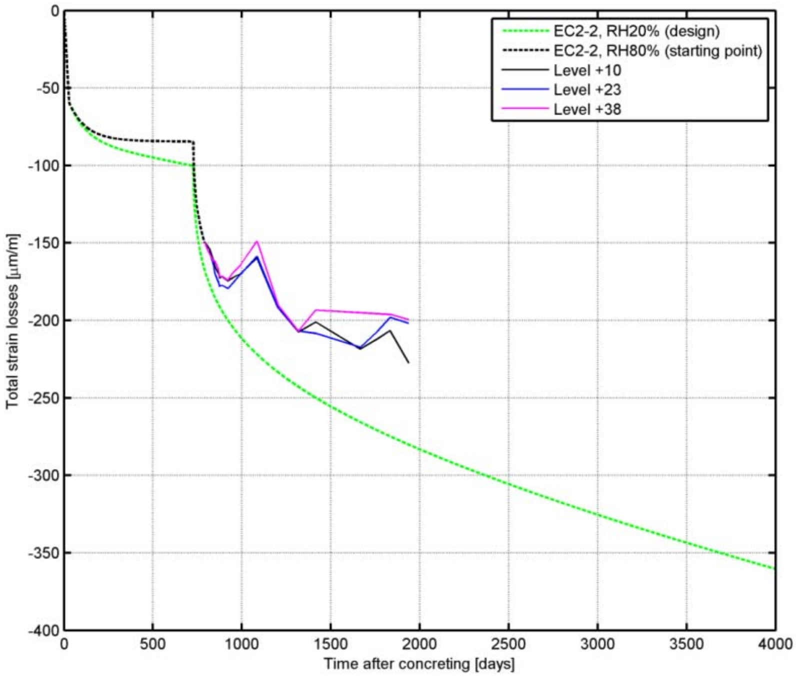


Figure 7.5 Measured tangential strains at different levels in the cylindrical wall compared to pre-calculated concrete strain losses (green curve according to assumptions used in the design).

As stated above, the ISIT was performed in 2014 and the monitoring results confirmed the elastic behaviour and the integrity of the inner containment. Similar investigations will be repeated in the future in connection of periodical ISI tests.

7.3 Regulator's assessment and conclusions on ageing management of concrete structures

<u>Regulator´s assessment of the ageing management processes</u>

STUK guide [STUK-Guide YVL E.6] requires that a licensee has a plan for the monitoring of the ageing deformations , temperature, humidity of the concrete structures of the containment. The in-service inspection plan shall present the inspections to be conducted on structures at specified intervals during plant operation, the manner of performance of the inspections, and the criteria for assessment and recording of the inspection results. The plan for the in-service inspection of reactor containment concrete structures shall include the following information:

- Inspection of displacements, strains and leak-tightness of structures at specified intervals and in conjunction with leakage and pressure tests.
- Inspection of the condition of post-tensioned containment tendons and anchorages at specified intervals.

- Inspection of structures essential for the containment's function by test loading or other reliable methods, if necessary.

STUK guide E.6 also has requirements of periodical inspections according to AMPs. Otherwise ageing of concrete structures is managed by the general ageing management programmes according to the STUK guide [STUK-Guide YVL A.8]. STUK has inspected and approved

- the ageing management programmes ,
- in-service inspection and monitoring plans of containments,
- leak-tightness and pressure test programmes of the containments.

The ageing management processes described in the ageing management programmes have been found satisfactory. In some cases STUK has required changes in the leak tightness or pressure test programmes or ISI-plans of containments.

For example STUK has required that the containment of Olkiluoto 3 NPP has to be equipped with strain measurement system of the steel liner. The reason for this requirement was large initial imperfections caused by the large module fabrication technique. After required changes to the programmes STUK approved the ISI-plan.

Implementation of the ageing management programmes and periodical inspections has been verified during the STUK inspections. The ageing management process includes that the licensee regularly evaluates the adequacy of containment ageing monitoring plan and the coverage and effectiveness of the whole ageing management programme. A good example of this is that installation of pendulum-type of deformation measurement system has started in the containments of OL2 and OL1 on the year 2017.

<u>Regulator´s experience from inspection and assessment as part of its regulatory oversight</u>

STUK reviews the annual ageing management follow-up reports of the licensees produced according to Guide YVL A.8. Results of the containment ageing monitoring shall be presented at least every three years in connection of the follow-up report.

STUK inspectors make observations about the ageing of the concrete structures in connection of inspections during operation and outages of the NPP units, and during the renovation works . STUK is doing inspections at least once a year during outages for each NPP unit and once a year a more thorough inspection either on civil structures or fire protection (KTO-programme of STUK).

STUK reviews following reports that are sent to STUK for information after tests and licensee´s inspections;

- leak-tightness and pressure test results,
- results of periodical inspections.

STUK also approves the construction plans of the major renovation works of safety classified concrete structures of the NPP´s. Usually STUK has made some requirements or remarks on the construction plan and possibly demanded some additional information . Licensee has made a new revision with corrective changes in the plan and has approved the new version.

<u>The main strengths and weaknesses that have been identified either by the licensees or the regulator on the effectiveness of the SSC specific AMPs</u>

The periodical test and inspection programmes and In-service inspection programmes have turned out to be effective so that no significant degradation due to ageing has been reported.

During construction phase of the containments of the different units some construction errors have happened and results of the errors may have an effect on the aging of the structures. Some of the building non-conformances needed special studies and large remedial works (see Section 7.1.4).

Containments of Olkiluoto 1 and 2 are functioning in the leak-tightness and pressure tests according to linear theory and deformations have been reversible in the limits of ASME III Div. 2. . The measured strains and deformations of the monitoring systems have been in a good congruence with the pre-calculated FE-analysis results.

So far the displacement measurement system of Olkiluoto 1 and 2 has not fulfilled the requirements of ASME Section III Div. 2. Deformation measurements will be developed further with Pendulum-type of system. With the pendulums the deformation measurement system of Olkiluoto 1 and 2 units will satisfy the requirements of the standards ASME III Div.2. and ASME XI.

Steel containment of Loviisa units are functioning in a linear way in leak tightness tests. Important mounting flange bolts have been inspected regularly according to inspection programme and no significant corrosion has been discovered in Loviisa 1 and 2. A certain amount of bolts are changed every year in order to be sure of the correct behaviour.

Concerning Olkiluoto 3 containment measured displacements and strains have been linear and reversible in the limits of ASME III Div. 2.

STUK considers that the ageing management programmes concerning the concrete containments and reactor buildings of the Finnish NPP units, Loviisa 1 and 2, Olkiluoto 1, 2 and 3, have been adequate.

8 Pre-stressed concrete pressure vessels (AGR)

N/A

9 Overall assessment and general conclusions

9.1 Overall ageing management

A new regulatory guide for ageing management, i.e., Guide YVL A.8 has been recently issued and enforced simultaneously with an overall revision of all STUK's regulatory guides in the end of 2013. In this Guide YVL A.8 there is a requirement for the licensees to draw up an ageing management programme for regulator's approval. Similar but less detailed requirements were stipulated also in the previous YVL guides.

In the beginning of 2017 TVO and Fortum issued their updated ageing management programmes ac-cording to the Guide YVL A.8. After a review of the programmes STUK concluded that the both ageing management programmes should be developed further to meet the new regulations. Deadline for the revised programmes was set to the end April 2018. Regardless of update needs of ageing management programmes both licensees have carried out reasonable ageing management for their SSCs since the commissioning of the plant units. However, the role of comprehensive ageing management programmes will now be more emphasized as the operation of the NPP units is extended beyond the original design lifetime. Even in this case the design basis operability of SSCs has to be maintained although degradation of SSCs may be more difficult to anticipate and control in the long term operation.

A generic lesson learned in Finland is that the licensees should make decisions to modernize the NPPs in due time before the end of their design lifetime. A postponed decision to renew for instance an I&C system or an electrical system may challenge purchase of new spare parts for the systems in service. This may lead to situations where the licensee may not be able to demonstrate the safety of continued operations to the regulator, or at least the licensee and regulator may understand the adequate scope of demonstration in a different way. Finland has successfully applied Periodic Safety Reviews (PSR) for the operating NPPs. In the Finnish practice the licensee is obliged to demonstrate that the safety of the operations can be ensured and improved also during the next 10 years. The licensee has to commit to safety improvements in terms of plant modernizations to address both physical and technological ageing of SSCs, too.

9.2 Electrical cables

Traditionally, since middle of 1980's, STUK's regulatory guides have required follow-up programmes for those electrical cables in the containment that are needed in accident situations. All the other cables important to safety are also included in the ageing management programme for electrical equipment as stipulated in the earlier regulatory guides for electrical equipment. Today, the regulatory requirements for the ageing management of cables important to safety are presented in the Guide YVL A.8 and Guide YVL E.7.

STUK has verified by various kind of inspections that the licensee's cable ageing management programmes are in accordance with the regulatory requirements without major deviations. At the moment (end of 2017), the ageing management programme of Olkiluoto 3 is under preparation.

The essential cable ageing mechanisms identified by the licensees are ambient heat loading, conductor current thermal loading, voltage load, ionizing and UV radiation loading, humidity and mechanical stresses. The cable environmental monitoring comprises usually of the ambient temperatures and radiation. Licensees' basic inspections are visual and tactile inspections of the cables. Those inspections can also be executed for cable joints, terminal boxes and cable trays specially in the containment. Heat loading is inspected by measurements and thermal imaging. In some cases tan delta and partial discharge tests are executed for MV-cables. Samples from containment cable deposits are tested using such criteria as elongation at break, tensile strength at break, and insulation resistance.

As remedial and preventive actions, minor defects of jackets have been temporarily repaired. If the inspections or sample test results indicate that qualified life of the cable is near the end, then a new similar cable or qualified new cable type has been installed or, if possible, the qualified age of cable is extended by requalification.

In certain cases the thermal or radiation load of a cable is reduced by changing the cable routes, by improving ventilation and cooling and by installing shielding structures between the cable and radiation source.

For instance starting from 1980's Loviisa NPP have executed relatively large extensive inspections and remedial actions for electrical cables. Actions include requalification of Siemens LOCA cables, improving steam generator room ventilation and cooling, rerouting and shielding of cables and replacing degraded cables with new ones.

At NPP units Olkiluoto 1 and 2 electrical failures of in-containment cables have been very rare. However, as part of containment component renewals project ELMA also renewals of in-containment cables with long-term LOCA-requirements have been ongoing since 2011.

As a summary the cable ageing management at the Finnish NPPs is currently implemented on such a level that failures caused by cable degradation caused by ageing are reported only in few cases were the design temperature of a cable is exceeded unforeseeably. Most often cable changes have been results of changes in the equipment types or due to the mechanical damages caused by the works near cabling. In some cases it was reported that a cable type had become obsolete because lack of repair parts and installation expertise**.**

9.3 Reactor pressure vessels

STUK has inspected Licensees' ageing management programmes including RPV ageing monitoring plans. Implementation of the ageing management programmes has been verified during STUK inspections. The ageing management processes described in the ageing management programmes were found satisfactory at the operating NPPs. The ageing management programmes of Olkiluoto 3 are practically completed although the unit is still under commissioning.

STUK considers that the ageing management programmes concerning the RPVs of the Finnish NPP units Loviisa 1, Loviisa 2, Olkiluoto 1, Olkiluoto 2 and Olkiluoto 3 are adequate.

One significant ageing management issue related to RPVs is the irradiation embrittlement of Loviisa RPVs and the thermal annealing of the core area weld of Loviisa 1 RPV in 1996. The embrittlement rate of the critical core area welds has to be carefully monitored by the surveillance programmes as long as the RPVs are in operation. If the licensee plans to operate the plant units beyond the current operating licenses, i.e., beyond 50 years of operation, some measures may be necessary to confirm safe operation of the RPVs.

So far one indication has been detected in a low pressure safety injection (TH) nozzle of Loviisa 1 RPV. It may become an ageing management issue if new indications will be detected in other nozzles of same kind in future inspections. However, it is also possible that the existing indication proves out to be a manufacturing defect.

The detected indications in the nozzle / safe-end welds of systems 312 (feed water) and 323 (reactor core spray) may become a significant ageing management issue at NPP units Olkiluoto 1 and 2. The licensee TVO has decided to continue its work and inspect and, if necessary, repair the corresponding welds of systems 312, 323 and also system 321 (shut down cooling) at both units OL1 and OL2.

The operating license renewal process for NPP units Olkiluoto 1 and 2 is currently under way. The licensee TVO has applied permission to continue operating NPP units OL1 and OL2 until 60 years. In this context TVO has updated the strength and fatigue analyses of the RPVs. These analyses have been quite comprehensive including almost everything necessary except for brittle and fast fracture analyses that are required in Section 6 of [STUK – Guide YVL E.4]. TVO has been given time to complete the analyses. All relevant ageing mechanisms have to be considered in the analyses.

The analyses performed so far show that the most critical subcomponents of the RPVs are the feed water (312) nozzles. Based on these analyses, the damage risk of all other RPV's in-scope subcomponents is much lower. The analyses also show that IGSCC and IASCC can accelerate ageing of the RPVs.

9.4 Concrete containment structures

STUK reviews the annual ageing management follow-up reports of the licensees produced according to Guide YVL A.8. The ageing management of buildings shall be presented at least every three years in connection of the follow-up report. STUK is also carrying out inspections concerning the ageing of the concrete structures during plant operation and outages.

Licensees are carrying out leak-tightness and pressure tests and periodical inspections for the containment. Result reports of these are sent to STUK for review.

During construction phase of the containments of NPP units Loviisa 1 and 2 and Olkiluoto units 1 and 3 some construction errors and accidents have happened which may have an effect on the aging of the structures. Some of the building non-conformances needed special studies and large remedial works, which were successful (see Section 7.1.4).

Containments of Olkiluoto units 1 and 2 are functioning in the leak-tightness and pressure tests according to linear theory and deformations have been reversible in the limits of ASME III Div. 2. The measured strains and deformations of the monitoring systems have been in a good congruence with the pre-calculated FE-analysis results.

So far the displacement measurement system of Olkiluoto 1 and 2 has not fulfilled the requirements of ASME Section III Div. 2. Deformation measurements will be developed further with Pendulum-type of system. When the measurement results of the Pendulum-system are compared to the FE-analysis results, better comprehension of the behaviour of the containment and pre-stressing system can be reached. With the pendulums, the deformation measurement system of Olkiluoto units 1 and 2 will satisfy the requirements of the standards ASME III Div.2. and ASME XI.

Steel containments of Loviisa units are functioning in a linear way in leak-tightness tests. The licensee has inspected important mounting flange bolts regularly according to inspection program and no significant corrosion has been discovered. A certain amount of bolts are changed every year in order to be sure of the correct behaviour.

Concerning the Olkiluoto 3 containment, measured displacements and strains have been linear and reversible in the limits of ASME III Div. 2 during the first pressure test (Initial Structural Integrity Test ISIT).

STUK considers that the ageing management programmes concerning the concrete containments and reactor buildings of the Finnish NPP units are adequate.

10 References

ACI 349-76, ACI Standard: Code Requirements for Nuclear Safety Related Concrete Structures 1976

ASME Boiler & Pressure Vessel Code, Section III: Rules for Construction of Nuclear Facility Components-Division 2-Code for Concrete Containments

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

BY50, Concrete Code 2012, Concrete Association of Finland

Chemistry Handbook Part V Monitoring, FGF, STC-G/2008/en/052, Rev. B

Cronvall, O. Susceptibility of BWR RPV and Its Internals to Degradation Mechanisms. To be published 2018.

Draft Regulatory Guide (DG)-1197, Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons (proposed revision 2 of Regulatory Guide 1.90, August 1977).

FGF NTCM-G/2009/en/1013 rev. B, Special Ageing Management Program, Reactor Pressure Vessel, Areva NP

IAEA Safety of Nuclear Power Plants: Design; No SSR-2/1

IAEA Safety of Nuclear Power Plants: Commissioning and Operation; No SSR-2/2

IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)

IAEA Safety Reports Series No.57, Safe Long Term Operation of Nuclear Power Plants, 2008

IAEA Safety Standards Series No. NS-G-2.6, Safety Guide, Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants, Vienna, 2002

IAEA Safety Standards Series No. NS-G-2.12 Ageing Management for Nuclear Power Plants

IAEA Safety Standards, Specific Safety Guide No. SSG-13, Chemistry Programme for Water Cooled Nuclear Power Plants,

IAEA Nuclear Energy Series No. NP-T-3.13, Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned

LO1-K822-00044 versio 2.0, Loviisa 1 ja 2, Reaktoripainesäiliöiden ikääntymisen hallinta; yhteenveto. 19.6.2014.

Nevander, O. OL1 – Final safety analysis report for system 211 – Reactor pressure vessel. FSAR, Report No. 106076, 18.6.2004. 44 p.

Nuclear Plant Chemistry Conference, Paper Reference n°167 046, The Application of Film-forming Amines in secondary side chemistry treatment of NPPs, Paris 2012

NUREG-1801, Rev. 2, Generic Aging Lessons Learned (GALL) Report (Volumes 1 and 2), U.S. Nuclear Regulatory Commission, 2010

OL1 and OL2 Document 108654: "OL1 ja OL2 - Suojarakennuksen kaapelien kunnonvalvonta (tarkastukset ja koestukset)".

OL1, OL2, OL3 and KPA Document 117279, "OL1/OL2/OL3 ja KPA Ikääntymisen hallintaohjelma"

OL1 and OL2 - TURO2018 - System 312 – Structural qualification of the feed water nozzle, FSD3020506-42 rev 01.

RIL 53d Finnish Concrete Code 1971 in Finnish; Betoninormit 1971 Suomen Rakennusinsinöörien Liitto.

RIL 53d, Finnish Concrete Code 1975 in Finnish; Betoninormit 1975, Betonielementtinormit 1975, Rajatilamitoitusohjeet 1975

SFS-EN 1992 Eurocode 2: Design of concrete structures (all parts)

STUK – Guide YVL 3.8: Nuclear Power Plant Pressure Vessel In-service Inspection with Non-Destructive Testing Methods, 22.09.2003

STUK – Guide YVL A.8: Ageing management of a nuclear facility, 20 May 2014

STUK – Guide YVL E.4: Strength analysis of nuclear power plant pressure equipment, 15 November 2013

STUK – Guide YVL E.5: In-service inspection of nuclear facility pressure equipment with non-destructive testing methods, 20 May 2014

STUK – Guide YVL E.6: Buildings and structures of a nuclear facility, 15 November 2013

STUK – Guide YVL E.7: Electrical and I&C equipment of a nuclear facility, 15 November 2013

US NRC Regulatory Guide 1.207, Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components due to the Effects of the Light-Water Reactor Environment for New Reactors, U.S. Nuclear Regulatory Commission, 2007

Vesikari E (2008), Degradation and service life of concrete structures in nuclear power plants. The Finnish Research Programme on Nuclear Power Plant Safety 2007-2010, Research Report VTT-R-02323-08.

11 ANNEX 1: Finnish NPP sites.

Loviisa 1 (right) and Loviisa 2 (left).

Olkiluoto 1 (middle), Olkiluoto 2 (left) and Olkiluoto 3.

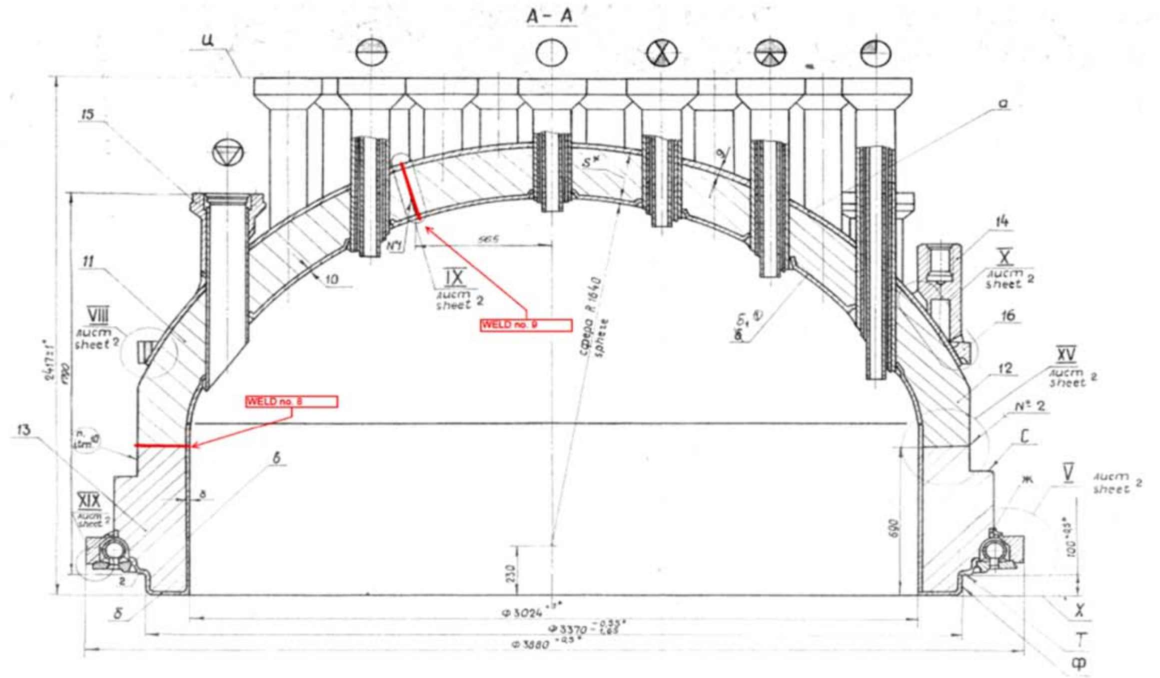
12 ANNEX 2: Key technical documentation of NPP units Olkiluoto 1 and 2 cabling

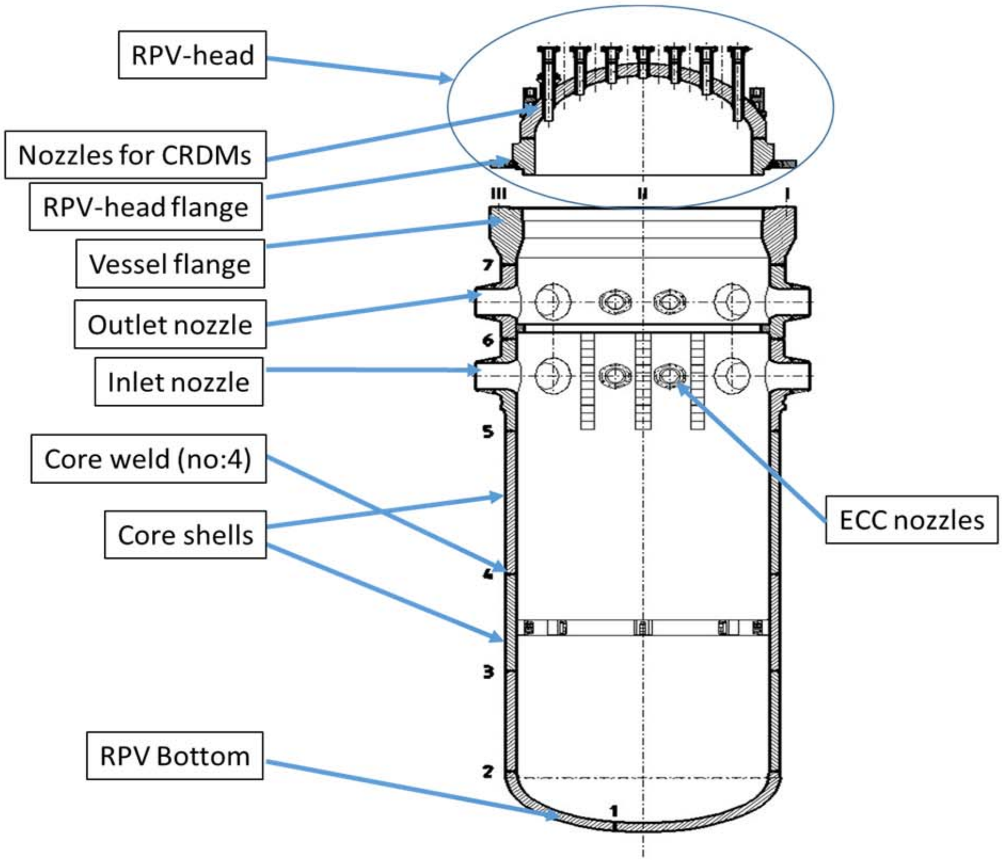
129251 (TE-Be-11e)	General Technical Requirements for design, manufacturing and delivery of cables for power supply, measuring and control purposes
107231 (TE-Be-1e)	Technical Requirements concerning transport and storage and normal operating conditions for electrical and control equipment
103394 (TE-Be-2e)	Technical Requirements concerning accident operation conditions for electrical and control equipment
129248 (TE-Ot-130e)	Inspection Instruction for DBE testing of cables
102478 (TE-Ot-135e)	Inspection Instruction for radiation and thermal exposure test of insulated wires
132743	Suitability analysis, “OL1/OL2 LOCA” cables, manufactured by Habia Cable AB
0-2/1/151	Pre-inspection documentation, Firewall III cables for in-containment use in LOCA-conditions, manufactured by The Rockbestos Company
0-2/531/26	Pre-inspection documentation, including the type test documentation for neutron flux measurement coaxial cables, manufactured by Habia Cable AB
135557	Type test documentation, Lipalon cables, manufactured by Liljeholmens Kabelfabrik AB and Asea Kabel AB
168775	Suitability analysis, halogen free standard cables for outside containment use, manufactured by Reka Kaapeli Oy
119125	Suitability analysis, standard cables for outside containment use, manufactured by Reka Kaapeli Oy
126857	Suitability analysis, standard cables for outside containment use, manufactured by Draka NK Cables Oy
0-2/515/1	Pre-inspection documentation, standard instrumentation cables for outside containment use, manufactured by Nokia Cables Ltd
168543	Lifetime assessment, in-containment cables

13 ANNEX 3: Key technical OL3 project documentation relevant to cabling

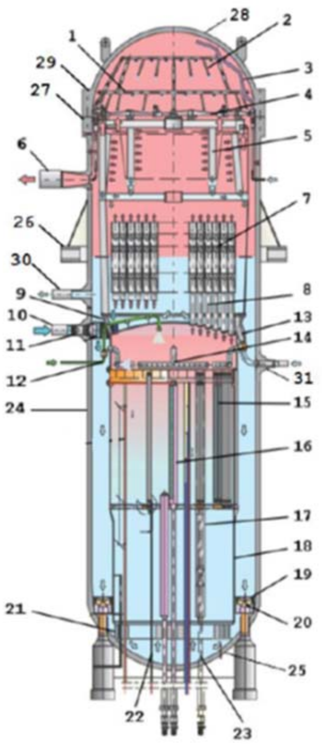
EZE-100032	Project Specification, normal power and control cables
NLEE-G/2006/en/1019	Project Specification, normal I&C cables
EYTM-100252	Project Specification, accident proofed power and control cables
EYTM-100366	Project Specification for Accident Proofed I&C-Cables < 60 V of the NI
PELES-G/2010/en/1001	Technical Specification, special accident proofed I&C cables
EZE-100050	Equipment Qualification Specification, normal power and control cables
NLEE-G/2008/en/1002	Equipment Qualification Specification, normal I&C cables
EYTM-100430	Equipment Qualification Specification, accident proofed power and I&C cables
NLQ-G/2007/en/1005	Suitability analysis, normal power and control cables
NLQ-G/2007/en/1010	Suitability analysis, accident proofed power and control cables
NLM-G/2009/en/1001	Suitability analysis, normal I&C cables
NLM-G/2009/en/1002	Suitability analysis, accident proofed I&C cables, manufacturer LEONI
IBQ-G/2011/en/1007	Suitability analysis, I&C extension cables, manufacturer HEW
IBQ-G/2011/en/1008	Suitability analysis, special accident proofed I&C cables, manufacturer EUPEN
NLLN-G/2008/en/1081	Suitability analysis, ECI connecting line
NLLN-G/2008/en/1082	Suitability analysis, RPVL connecting line
IBQ-G/2014/en/1001	Suitability analysis, N16 cables

14 ANNEX 4: A Schematic drawings of Loviisa 1 and Loviisa 2 RPVs





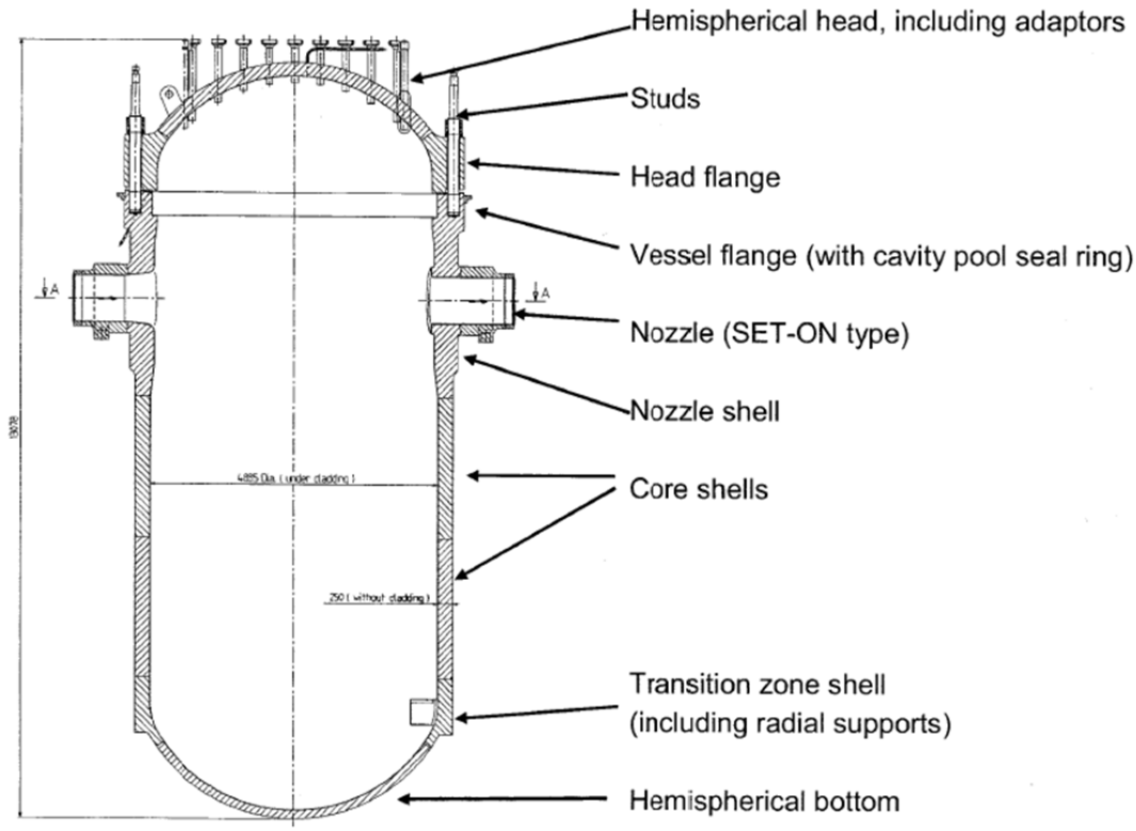
15 ANNEX 5: A schematic drawing of Olkiluoto 1 and Olkiluoto 2 RPVs



Considered components

ID	Component	ID	Component
1	Flange cooling spray piping	16	Control rods
2	Long nozzle pipes in cooling spray piping	17	Control rod guide tubes
3	Evacuation pipe	18	Core shroud, Core shroud support
4	Spring beams and support brackets	19	Pump deck
5	Steam dryer	20	Main circulation pump nozzles
6	Steam outlet nozzles	21	Core shroud support legs
7	Steam separator stand pipes	22	Instrumentation guide tubes and nozzles
8	Steam separator pipe bundles	23	Control rod guide tubes and nozzles at RPV bottom
9	Steam separator support legs	24	Cylindrical RPV shell
10	Feedwater nozzles	25	RPV bottom
11	Feedwater spargers	26	RPV support skirt
12	Boron spray nozzles and piping	27	RPV flange
13	Core spray piping outside core shroud cover	28	RPV-head
14	Core spray piping inside core shroud cover	29	RPV-head bolts
15	Fuel assembly	30	Shutdown cooling nozzles
		31	Core spray nozzles

16 ANNEX 6: A schematic drawing of Olkiluoto 3 RPV

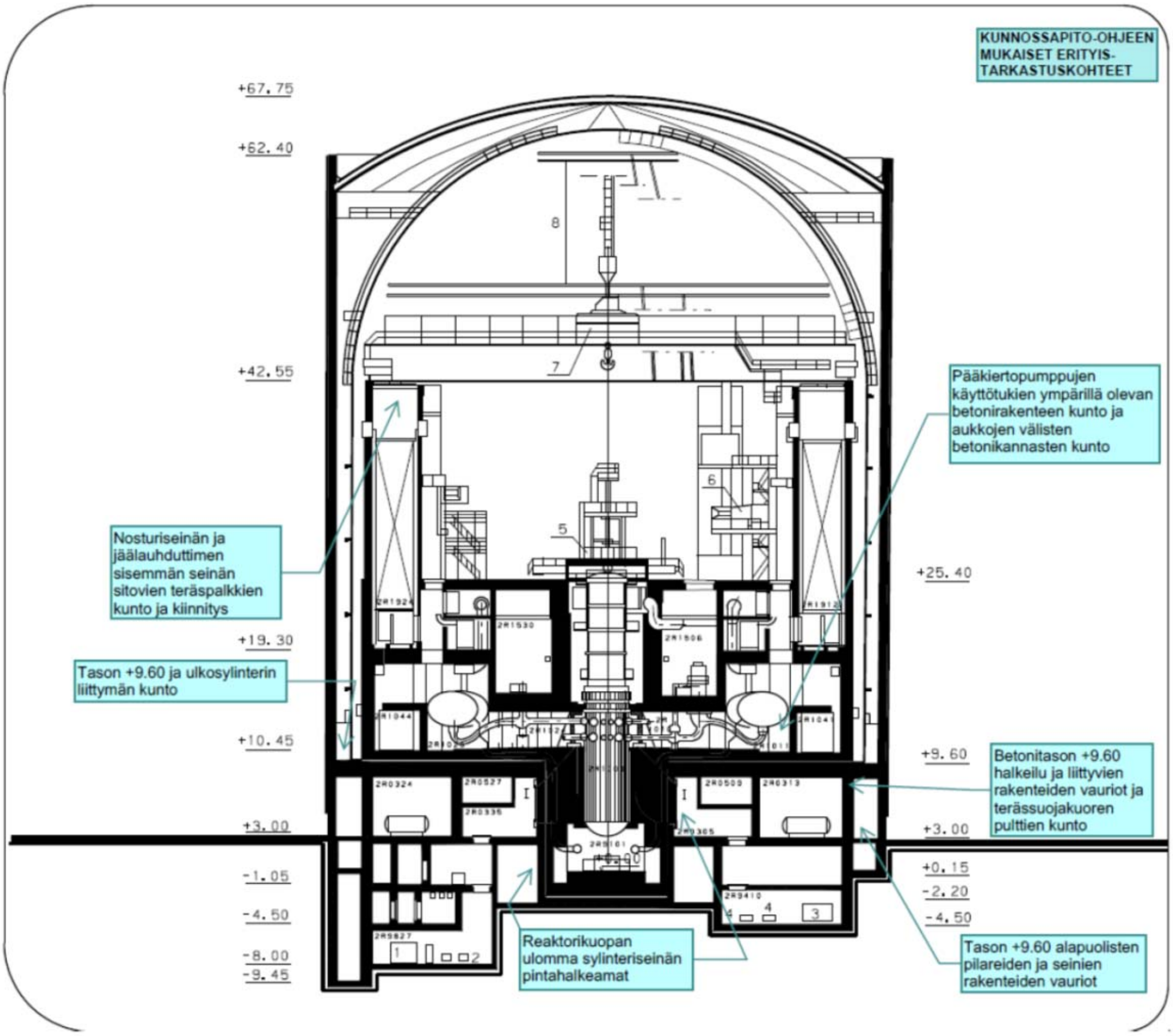


17 ANNEX 7 Summary of Olkiluoto 3 RPV subcomponents' materials

Item no.	Description	Material / Filler Metal	Weld Process
1	**Lower Head**	Low-Alloy Steel 16 MND 5	
	Weld – Lower Head to Transition Ring	Low-Alloy Steel US-56B/MF-27	SAW
	Cladding Lower Head to Transition Ring	Austenitic Stainless Steel USB-309LK, USB-308LK, PFB-7FK, NC-39LK, NC-38LK	SMAW, SAW
	Cladding Lower Head (Central Zone)	Austenitic Stainless Steel NC-39LK, NC-38LK	SMAW
2	**Transition Ring**	Low-Alloy Steel 16 MND 5	
	Weld – Core Shell to Transition Ring	Low-Alloy Steel US-56B/MF-27	SAW
	Cladding Core Shell and Transition Ring	Austenitic Stainless Steel USB-309LK, USB-308LK, PFB-7FK, NC-39LK, NC-38LK	SMAW, SAW
	Weld – Radial Guide to Transition Ring (DMW)	Nickel-base + Low-Alloy WEL AC 152	SMAW
	Buttering for Radial Guides	Low-Alloy Steel WEL AC 152	SMAW
4	**Core Shell**	Low-Alloy Steel 16 MND 5	
	Weld – Core Shell to Integrated Shell, Weld – Core Shell	Low-Alloy Steel US-56B/MF-27	SAW
	Cladding Core Shell and Integrated Shell	Austenitic Stainless Steel USB-309LK, USB-308LK, PFB-7FK, NC-39LK, NC-38LK	SMAW, SAW
5	**Nozzle / Flange-integrated Shell**	Low-Alloy Steel 16 MND 5	
	Weld – Leak Detection Tube to Integrated Shell, **Weld – Seal Ledge to Integrated Shell**	Austenitic Stainless Steel TGS-308LK	TIG
	Buttering for Leak Detection Tube Weld to Integrated Shell	Austenitic Stainless Steel NC-39LK, NC-38LK	SMAW

	Cladding Integrated Shell key way, **Cladding Integrated Shell**	Austenitic Stainless Steel NC-39LK, NC-38LK	SMAW
	Cladding Integrated Shell, **Cladding Integrated Shell internals support ledge**	Austenitic Stainless Steel USB-309LK, USB-308LK, PFB-7FK, NC-39LK, NC-38LK	SMAW, SAW
6, 7	**Inlet / Outlet Nozzle**	Low-Alloy Steel 16 MND 5	
	Weld – Inlet / Outlet Nozzle to Integrated Shell	Low-Alloy Steel US-56B/MF-27	SAW
	Cladding of Inlet / Outlet Nozzle Bore	Austenitic Stainless Steel USB-309LK, USB-308LK, PFB-7FK, NC-39LK, NC-38LK	SMAW, SAW
	Cladding of Inlet / Outlet Nozzle and Integrated Shell	Austenitic Stainless Steel TGS-309LK, TGS-308LK, NC-39LK, NC-38LK	SMAW, TIG
		Austenitic Stainless Steel NC-39LK, NC-38LK	SMAW
8	**Safe-End**	Austenitic Stainless Steel (nitrogen controlled) Z2 CND 18-12	
	Weld – Inlet / Outlet Nozzle to Safe-End (DMW)	Nickel-base Alloy WEL TIG 52	TIG
11	**Head Flange**	Low-Alloy Steel 16 MND 5	
	Weld – Head Flange to Upper Head	Low-Alloy Steel US-56B/MF-27	SAW
	Cladding Head Flange and Upper Head, **Cladding Head Flange**	Austenitic Stainless Steel USB-309LK, USB-308LK, PFB-7FK, NC-39LK, NC-38LK	SMAW, SAW
	Cladding Head Flange key way	Austenitic Stainless Steel NC-39LK, NC-38LK	SMAW
12	**Upper Head**	Low-Alloy Steel 16 MND 5	
	Weld – Handling Lug to Upper Head	Low-Alloy Steel LBL-96	SMAW
	Cladding Upper Head (Central Zone)	Austenitic Stainless Steel NC-39LK, NC-38LK	SMAW

18 ANNEX 8: A schematic drawing of Loviisa 1 and 2 reactor building



19 ANNEX 9: A schematic drawing of Olkiluoto 1 and 2 containment

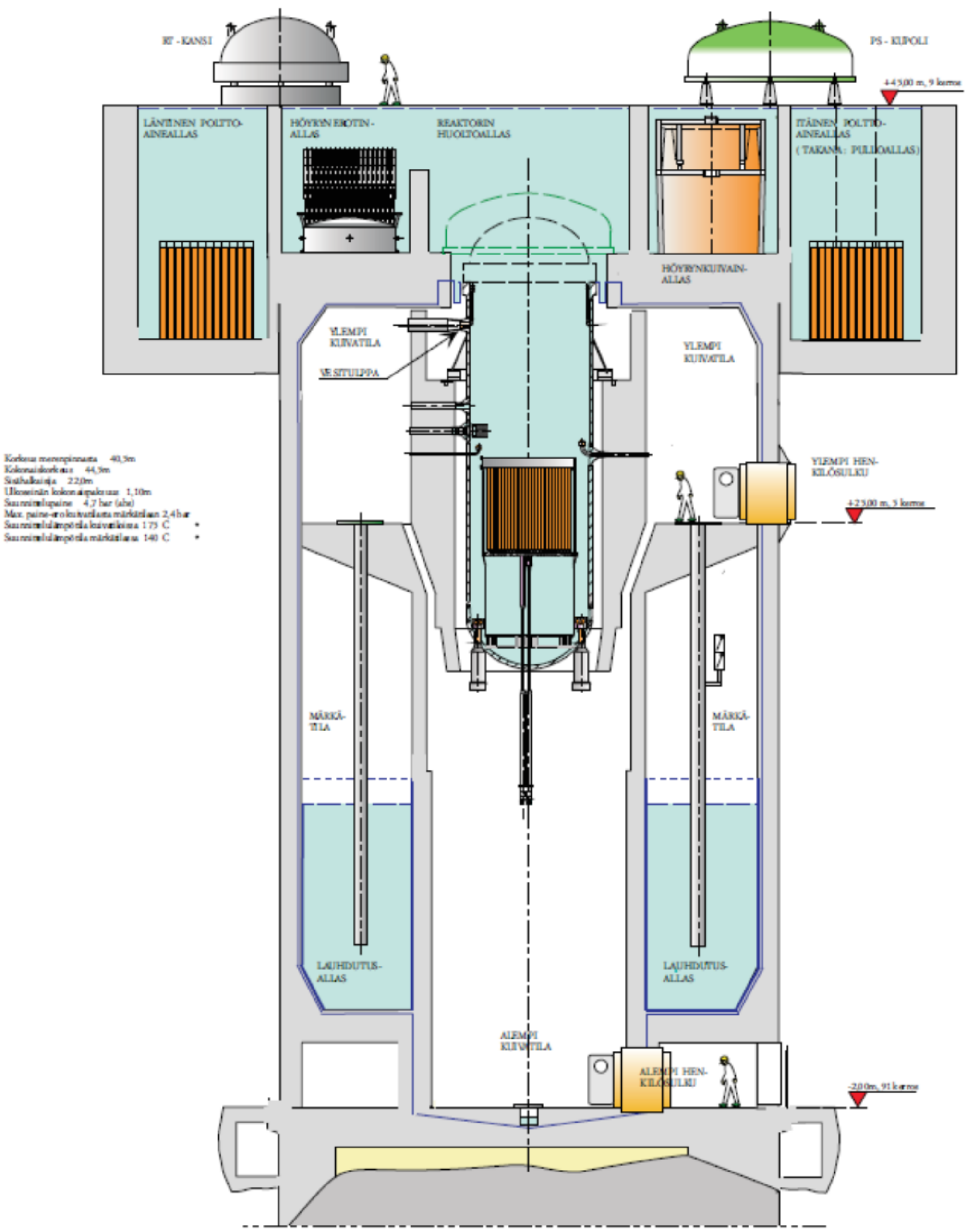


Figure Schematic drawing presenting Olkiluoto1 and 2 cross-section of the containment.

20 ANNEX 10: A schematic drawing of Olkiluoto 3 reactor building

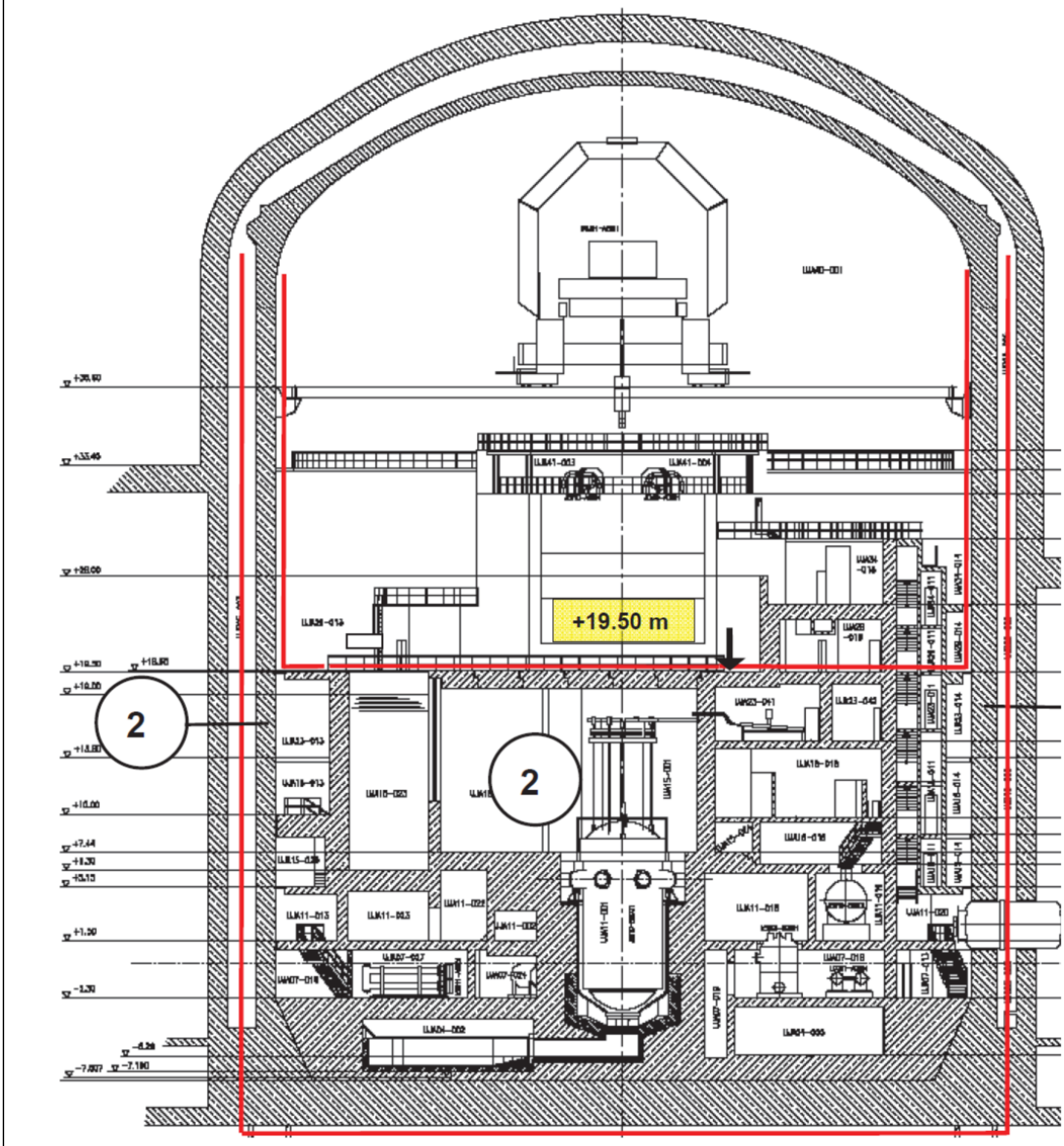


Figure Schematic drawing presenting Olkiluoto 3. cross-section of the reactor building with inner and outer containment. In the Figure boundaries between Safety Class 2 and 3 structures has been presented.