# Non-Paper

## 1. Introduction

The objective of this non-paper is to collect and prepare technical arguments and background to support a broad and comprehensive use of the "Guidance on the applicability of the Convention to the lifetime extension of nuclear power plants", endorsed by the Meeting of the Parties to the Espoo Convention at its eighth session (Vilnius (online), 8–11 December 2020).

Several Nuclear Power Plants (NPPs) are near to reach the original design lifetime which is used as one essential parameter for safety analysis to apply the construction and the operation license. Design life extension seems possible, if safety concerns are properly addressed in full extent and current safety requirements and environment legislation are met.

Nuclear power plants undergo two types of time-dependent changes:

- Physical ageing of structures, system and components (SSCs), which results in degradation, i.e. gradual deterioration in their physical characteristics.
- Obsolescence of concepts and design for safety relevant SSCs regarding degree of Defence in Depth principles, physical separation, redundancy, functional diversification, ability for inspection and maintenance under state of the art knowledge and deviations from implementing safety objectives for new NPPs.

Both have to be considered by an effective long term operation and plant life management programme for lifetime extension beyond the original design life, which is much more challenging than just to proceed with operation under unchanged conditions within the established design lifetime and safety requirements for existing plants.

*The Western European Nuclear Regulator's Association (WENRA) has revised safety reference levels (SRLs) for existing reactors with the aim to integrate the lessons learned from the 2011 Fukushima Daiichi accident. A list of 342 SRLs has been published in 2014. In addition to the updated SRLs, the WENRA Reactor Harmonization Working Group (RHWG) provides several guidance documents on issues F (Design Extension Conditions) and T (Natural Hazards). According to the SRL F1.1, analysis of Design Extension Conditions (DEC) shall be under-taken with the purpose of further improving the safety of the nuclear power plant.* [UBA 2020]

EIA reporting for lifetime extension relevant for the Espoo Convention should include a comparison of the design and measures demonstrating fulfilment of all requirements of SRL F. In case of deviations, the reasons should be explained.

*The WENRA "Safety Objectives for New Power Reactors" have been elaborated for new reactors. Nevertheless, they should be used as a reference for identifying reasonably practicable safety improvements for existing plants.* [UBA 2020]

The most ambitious safety objective is to reduce potential radioactive releases to the environment from accidents with core melt. Accidents with core melt which would lead to early releases without enough time to implement off-site emergency measures or large releases which would require protective measures for the public that could not be limited in area or time including trans-boundary impact have to be practically eliminated. Practical elimination of an accident sequence cannot be claimed solely based on compliance with a general cut-off probabilistic value.

IAEA Safety Standards for protecting people and the environment – Safety of Nuclear Power Plants: Design, Specific Safety Requirements No. SSR-2/1 (Rev.1, 2016) requires: *The possibility of certain conditions arising may be considered to have been 'practically eliminated' if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise*. [IAEA SSR-2/1, 2016]

HERCA-WENRA defined its *Approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident, Stockholm, 22 October 2014* stating that *Fukushima has shown again that a severe nuclear accident anywhere in the world, including Europe, cannot be completely excluded.* [HERCA-WENRA 2014]

EIA reporting should consider and present all envisaged measures for lifetime extension to come as close as reasonably practicable to meet the WENRA safety objective O3 (avoiding accidents with core melt) for new NPPs.

A severe accident with large and early release can lead to significant trans-boundary impacts on foreign territory and justify the request for participation in EIA procedures for possibly affected Parties according to the Espoo Convention.

Chapter 2 provides general technical issues about the scope of the Guidance and the understanding of the term "Lifetime Extension" as background with reference to technical definitions.

Chapter 3 collects technical and regulatory arguments for "Situations" understood as a possible Lifetime Extension by the Guidance and the Convention.

## 2. Scope of the Guidance – Understanding of the Term Lifetime Extension

### 2.1 Factors limiting the lifetime of a nuclear power plant

Nuclear power plants undergo two types of time-dependent changes:

- Physical ageing of structures, system and components (SSCs), which results in degradation, i.e. gradual deterioration in their physical characteristics.
- Obsolescence of concepts and design for safety relevant SSCs regarding degree of Defence in Depth principles, physical separation, redundancy, functional diversification, ability for inspection and maintenance under state of the art knowledge and deviations from implementing safety objectives for new Nuclear Power Plants (NPPs).

Both have to be considered by an effective long term operation and plant life management programme for lifetime extension beyond the original established design life.

The UNECE Espoo *Guidance on the applicability of the Convention to the lifetime extension of nuclear power plants* (2020) describes situations considered as a Lifetime Extension (LTE) of NPPs by a combination of physical and legal conditions (see chapter 3).

The following aspects have to be considered in this regard:

- Physical status of safety critical SSCs according to the original licensing basis including detailed documentation and demonstration to fulfil all limits and conditions addressing effective ageing management

According to IAEA SSG-48, Para 3.11: *Ageing management should be addressed in the safety analysis report and other licensing documents. The description of ageing management in the safety analysis report should include general information on the following topics:*

*(a)* *The strategy for ageing management and prerequisites for its implementation;*

*(b)* *Identification of all SSCs that could be affected by ageing and are in the scope of the ageing management;*

*(c)* *Proposals for appropriate materials monitoring and sampling programmes in cases where it is found that ageing effects might occur that could affect the capability of SSCs to perform their intended functions throughout the lifetime of the plant;*

*(d)* *Ageing management for different types of in-scope SSCs (e.g. concrete structures, mechanical components and equipment, electrical equipment and cables, and* instrumentation and control equipment and cables) and the means to monitor their degradation;

(e) *Design inputs for equipment qualification […] of the in-scope SSCs, including required equipment and equipment functions that need to be qualified for service conditions in normal operation and associated with postulated initiating events;*

*(f)* *General principles stating how the environment of an SSC is to be maintained within specified service conditions (e.g. by means of proper location of ventilation, insulation of hot SSCs, radiation shielding, damping of vibrations, avoiding submerged conditions, and proper selection of cable routes);*

*(g)* *Appropriate consideration of the analysis of feedback on operating experience with respect to ageing.*

[IAEA SSG-48]

- Compliance of safety critical SSCs to the current licensing basis applied for new NPPs addressing state of the art safety concepts (obsolescence management)

*Each plant programme and analysis should be properly documented in safety analysis reports or in other current licensing basis documents, which should clearly and adequately describe the current licensing basis or the current design basis requirements for operation of the nuclear power plant.* [IAEA SSG-48]

The Contracting Parties meeting at the Diplomatic Conference at 9 February 2015 of the Convention on Nuclear Safety adopted the “Vienna Declaration on Nuclear Safety” including three principles, with the following two special relevance for LTE:

*1. New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.*

*2. Comprehensive and systematic safety assessments are to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the above objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.* [IAEA CNS VDNS 2015]

- Physical works to address physical ageing and obsolescence by safety upgrades for practical elimination of large and early release of radioisotopes with preference to severe accidents with off-site consequences above intervention levels.

*SSCs needed to cope with design extension conditions or to mitigate the consequences of severe accidents.* [IAEA SSG-48]

IAEA SSR-2/1 (Rev.1) defines that: *The possibility of certain conditions arising may be considered to have been ‘practically eliminated’ if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise.*

This definition of “practical elimination” should be strictly applied for new but also for existing NPP.

- Establishing and implementing comprehensive programmes for lifetime extension beyond the originally established design lifetime considering the entire period of LTE as required by IAEA Safety Guides including qualification, certification, (re)licensing and authorization procedures especially for safety relevant SCCs if they cannot be replaced (e.g. Reactor Pressure Vessel, Containment)

*Safety factor 3: Equipment qualification*

*5.37. Plant equipment important to safety (that is, SSCs) should be properly qualified to ensure its capability to perform its safety functions under all relevant operational states and accident conditions, including those arising from internal and external events and accidents (such as loss of coolant accidents, high energy line breaks and seismic events or other vibration conditions).* [IAEA SSG-25]

*5.40. Qualification of plant equipment important to safety should be formalized using a process that includes generating, documenting and retaining evidence that equipment can perform its safety functions during its installed service life. This should be an ongoing process, from its* *design through to the end of its service life. The process should take into account plant and equipment ageing and modifications, equipment repairs and refurbishment, equipment failures and replacements, any abnormal operating conditions and changes to the safety analysis. Although many parties (such as designers, equipment manufacturers and consultants) will be involved in the equipment qualification process, the operating organization has the ultimate responsibility for the development and implementation of an adequate plant specific equipment qualification programme.* [IAEA SSG-25]

*3.32. The operating organization should detail how the physical status of structures or components will be managed consistent with the current licensing basis for the planned period of long term operation.* [IAEA SSG-48]

Here it is necessary to insist in state of the art current licensing basis as valid for new nuclear reactors.

*3.18. The operating organization should collect baseline data and should also confirm that critical service conditions (as used in equipment qualification) are in compliance with the design. Analyses of such data should be subject to review by the regulatory body.* [IAEA SSG-48]

This implies that equipment requalification for Lifetime Extension should comply with current design requirements in full extent. Not only single components but the entire safety systems and conception should fulfil current safety requirements e.g. at all levels of Defence in Depth.

- Evaluation of safety aspects and risk assessment for actual conditions and plant site, changes in environment including new information on external hazards, land use, industrial applications or climate change.

*Terrorist attacks, airplane crashes and other disruptive actions as well as extreme natural events as a result of ramping climate change, can no longer be neglected, and represent risks. As such, they require special protective measures which were not foreseen in the design of the existing plants and can only be implemented to a very limited extent. Compliance with today's safety standards would practically require the development and construction of a completely new nuclear power plant.* [INRAG 2021]

<u>Ageing and obsolescence - background</u>

The INRAG 2021 Summary report on *Risks of Lifetime Extension of Ageing Nuclear Power Plants* [INRAG 2021] states in this regard: *In order to approach the ageing problem, a distinction is made between the physical ageing of materials and obsolescence (technological and conceptual ageing). In the case of physical ageing, ageing of components with manufacturing defects, physical ageing of special components, ageing management, time-dependent failure rates, and handling of ageing-related reportable events are analyzed, as well as countermeasures and their limitations. Technological ageing includes also the lack of spare parts, suppliers, industrial capacity of a component because it is no longer manufactured and conceptual ageing outdating of design (design obsolescence).*

*Although there is a requirement to retrofit old plants up to the current state of science and technology, the possibilities for technical retrofits are limited. Differences remain between the safety level achieved in old plants and the safety level required for new plants according to the current state of science and technology.*

*In addition, knowledge of older plant design and operation is generally dwindling. Knowledge of the original design is being lost and the generation of experts who designed and* *commissioned the plants is moving into retirement. In addition, the existing documentation is often incomplete and does not meet today's requirements.*

Physical ageing

*The ageing, which means the deterioration of material properties, and thus the decreasing functionality and reliability of structures, systems and components (SSCs) with increasing operating time of a plant inevitably leads to the reduction of original safety margins. This subsequently leads to a higher probability of failure, most importantly if special load cases occur. The dependence of the failure rate with the operating time can be described by the so-called bathtub curve, which basically applies to all technological systems. After a start-up phase, the failure rate generally remains constant at a comparatively low level over a further period of time until finally ageing processes lead to an increased number of failures.* [INRAG 2021]

Obsolescence

According to IAEA SSG-48 - Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants (2018), obsolescence can be classified in three main categories:

*o Technological*

*Manifestation:*

- *Lack of spare parts and technical support*
- *Lack of suppliers*
- *Lack of industrial capabilities*

*With the consequence:*

- *Declining plant performance and safety due to increasing failure rates and decreasing reliability*

*o Regulation, codes and standards*

*Manifestation:*

- *Deviations from current regulations, codes and standards for structures, components and software*
- *Design weaknesses (e.g. in equipment qualification, separation, diversity or capabilities for severe accident management)*

*With the consequences:*

- *Plant safety level below current regulations, codes and standards (e.g. weaknesses in defence in depth or higher risk of core damage (frequency))*

*o Knowledge*

*Manifestation:*

- *Knowledge of current regulations, codes and standards and technology relevant to SSCs not kept current*

*With the consequences:*

- *Opportunities to enhance plant safety missed*

[IAEA SSG-48]

Retrofitting and its limits

During operation history all structures, systems and components (SSCs) of NPPs, especially the safety critical SSCs have to be kept in a physical status to comply to limits and conditions as established by the regulatory framework as a mandatory requirement for the operation authorization.

The status of SSCs are regularly inspected and tested according to defined plans and procedures which are part or preconditions for the operation license. Results from inspections and testing has to be documented, accesses is to the nuclear safety regulator.

The safety regulator supervises the inspections of the operator and is in the position to initiate and perform own inspections to prove the organisational arrangement and quality of the inspection regime of the operator and the plant state itself. In case of discrepancies the nuclear safety regulator is obliged to force the operator for compliance, setting up a timeframe for corrective actions regarding safety relevance of deficiencies and affected SSCs in case also requesting immediate plant shut-down until safe operation can be demonstrated with validated methods by the operator.

All of these actions are part of normal operation and maintenance with focus on plant life management independent from intentions to operate beyond the original established design lifetime by LTE.

However inspection plans also refer to required retrofitting, refurbishment or replacement of components. In case SSC are qualified or requalified based on equipment production and operation data such as reactor passport and plant operation documentation.

*Change in material properties often cannot be tested non-destructively. Therefore, it is difficult to establish the condition of ageing materials with certainty. Calculation methods for the determination of loads and their effects on the material behaviour generally can only be validated on specimens, and uncertainties for results of said calculations for the nuclear power plant are therefore difficult to specify. Unknown damage mechanisms can occur with increasing age of the nuclear plants and cannot be taken into account in calculation models.* [INRAG 2021]

The following factors are relevant regarding the original established plant design lifetime:

- Plant investment is based on economic considerations including a minimum plant operation lifetime, with predicted overall output of electrical energy. During plant design lifetime return of investment, operational costs, benefits and financial risks has to be covered.

*Nuclear power plants were originally designed to operate for 30 to 40 years. Thus, the operating lifetime of many plants are approaching this limit, or has already exceeded it.* [INRAG 2021]

- All structures, systems and components (SSC) of the plant has be designed to sustain the established overall plant design lifetime under service conditions. If a SSC cannot fulfil limits and conditions for the established plant design lifetime, refurbishment, maintenance and replacement should be planned and foreseen by design.

- SSC design is based on qualified safety analysis by validated methods regarding all relevant parameters such as temperature, pressure, vibration, chemical corrosion, phase transition, irradiation and other factors for material degradation (physical ageing). All operation states as service conditions such as shutdown, cold and hot standby, modes of power operation and load cases e.g. start-up and shut-down, SCRAM, anticipated operational occurrences (AOO) and design basis accidents (DBA) should be considered taking into account estimated frequency during plant design lifetime (e.g. number of start-up and shut-down operation, number of SCRAMs and AOOs, etc.).

*4.3. Engineering design rules are related to the three characteristics of capability, reliability (dependability) and robustness:*

*(a) Capability is the ability of an SSC to perform its designated function as required;*

*(b) Reliability (dependability) is the ability of an SSC to perform its required function with a sufficiently low failure rate consistent with the safety analysis;*

*(c) Robustness is the ability to ensure that no operational loads or loads caused by postulated initiating events will adversely affect the ability of the SSC to perform its function.*

[IAEA SSG-30]

- SSCs are designed to comply with limits and conditions established by the regulatory framework and technical codes and standards including safety margins.

*Conservative safety margins shall be applied or other appropriate precautions shall be taken to compensate for possible unanticipated failures.* [IAEA SSR-2/1]

- Material degradation of SSCs are calculated for the design lifetime based on qualified methods. The design lifetime of safety critical SSCs shall be determined. This implies that service lifetime beyond the design life can be understood as Lifetime Extension on a technical and regulatory basis.

*Requirement 31: Ageing management*

*The design life of items important to safety at a nuclear power plant shall be determined. Appropriate margins shall be provided in the design to take due account of relevant mechanisms of ageing, neutron embrittlement and wear out and of the potential for age related degradation, to ensure the capability of items important to safety to perform their necessary safety functions throughout their design life.* [IAEA SSR-2/1]

- During plant lifetime inspection programmes are used to validate the predictions for physical ageing and material degradation. In case of discrepancies predictions require justified updates. According to the “Safety First” principle physical lifespan can be at maximum identical with the design lifetime used for licensing. In case of faster degradation or increased requirements requalification has to be performed also before design lifetime expires.

*5.52. Provision shall be made for monitoring, testing, sampling and inspection to assess ageing mechanisms predicted at the design stage and to help to identify unanticipated behaviour of the plant or degradation that might occur in service.* [IAEA SSR-2/1]

- Construction authorization is based on sufficient safety demonstration of the plant design including predicted lifetime of safety relevant SSCs.

*(e) Demonstration that the management of anticipated operational occurrences and design basis accidents is possible by safety actions for the automatic actuation of safety systems in combination with prescribed actions by the operator;*

*(f) Demonstration that the management of design extension conditions is possible by the automatic actuation of safety systems and the use of safety features in combination with expected actions by the operator.* [IAEA SSR-2/1]

- Economic consideration avoiding different design lifetime of SSCs as additional safety margins are expensive. Major SSC are therefore designed, qualified and certified for a common predicted design lifetime to be established. This can be understood as the technical design lifetime of the plant.

*Nuclear power plants were originally designed to operate for 30 to 40 years.* [INRAG 2021]

- Contracts between vendor, equipment producers and plant owner are usually restricted. They have also to cover responsibilities for equipment and construction certification as required by the nuclear safety and/or other regulators.

- Quality control and documentation including detailed description of methods and processes of production are part of the licensing documents and should provide warranty especially for nuclear safety relevant SSCs as "nuclear grade" under national legislation covering the design lifetime.

- Most of SSCs can be maintained, refurbished or replaced if necessary. However for some of them this seems not possible from an economic perspective, such as the Reactor Pressure Vessel or major structures of civil engineering such as the hermetic containment structure.

- If safety relevant SSCs cannot be maintained, refurbished or replaced to fulfil limits and conditions even taking benefits from compensatory actions, the plant has reached its physical lifetime.

- Physical lifetime may differ from design lifetime as the first one is based on inspection results and material properties or on increased regulatory requirements with technical implications, the second one on technical estimates and predictions. Usually the physical lifetime should not be reached by design lifetime under conservative approach with safety margins.

- Prolongation of design lifetime is possible for the entire plant and relevant SSCs, if safety critical components are (re-)qualified and (re-)certified. Based on comprehensive safety demonstration design life extension may be authorized if a vendor or operator takes warranty for the SSC to fulfil limits and conditions over the extended period, which therefore has to be defined.

<u>Safety upgrades and obsolescence</u>

*Theoretically, it is possible to counteract negative ageing processes by reducing thermal loads. In reality, however, reactor lifetime extensions are often linked to power increases or different upgrades and updates for economic rea-sons.* [INRAG 2021]

*With increasing knowledge and improved testing methods, manufacturing-related defects continue to be discovered, also because manufacturing-related defects often only have an effect after a certain period of operation. This shows by way of example that the presumed and claimed safety level of old NPPs does not necessarily correspond to the actual safety level simply due to the emergence of previously unknown defects.* [INRAG 2021]

*The differences that cannot be remedied generally relate to the degree of redundancy, diversity, functional independence and spatial separation of safety trains, as well as further protection of the plant against external impacts, including additional precautions against beyond-design-basis accidents. Thus, despite extensive retrofits, current safety standards are not and cannot be achieved in old nuclear power plants.* [INRAG 2021]

Improved inspection methods and modelling increases the efficiency also by enhancing quality of predictions. There is a trend to use higher precision to consume originally established safety margins for safety demonstration to be compliant under current limits and conditions including material properties. In reality an unchanged component might be requalified and reused even under stricter conditions with a long lasting operation history, more or less documented and documented after inspection procedures. This component or structure might fulfil all limits and conditions in full extent, however it is used with decreased safety margins which induces physically higher loss rates avoiding additional investment in system upgrade. Safety upgrades are only effective with physical implications to the plant and its operation such as system upgrades. Safety analysis, equipment qualification and licensing are only preconditions to plan actions and to prove its implementation.

*Only some countries planned new permanently installed and partially bunkered systems. Instead of extensive retrofits or permanent shutdown of particularly vulnerable nuclear plants, most countries are attempting to compensate for design deficiencies with the purchase of mobile equipment.* [INRAG 2021]

*Often 10 - 20 years pass between the recognition of safety deficits and their elimination. Safety improvements often are judged as not economical and are omitted with reference to the limited remaining operating life time of the plant.* [INRAG 2021]

Design life extension therefore has not only to address ageing management but also obsolescence management applying state of the art nuclear safety objectives and provisions by using actual validated methods.

Safety concepts and regulatory requirements relevant for lifetime extension

*On July 8, 2014, the Council of the European Union adopted Directive 2014/87/EURATOM amending Directive 2009/71/EURATOM establishing a Community framework for the nuclear safety of nuclear installations. However, this directive establishes a de facto double standard. The double standard consists in the specification for the technical design of the safety measures and facilities to achieve the radiological protection objective (Article 8a, Paragraph 1). Plants that have been granted the initial license for construction after August 14, 2014, must meet the safety objective defined in Article 8a as part of the design. For these facilities, it must be shown that releases of radioactive materials can only occur to a limited extent and that they will not occur early in the accident sequence. For existing plants, on the other hand, this requirement applies only as a "reference" for determining "reasonably practicable safety improvements" and implementing them in a "timely manner".* [INRAG 2021]

*The design of new plants must aim to prevent accidents and, in the event of an accident, to mitigate effects, as well as to prevent early releases that require off-site emergency response measures. Furthermore, large releases requiring protective measures that cannot be limited in space or time must be precluded. For existing facilities, these goals are considered a reference for the timely implementation of reasonably practicable safety improvements to be used in the periodic safety reviews. The periodic safety review (at least every ten years) is intended mainly* *to demonstrate compliance of the current design to the existing operating license. Further safety improvements are to be identified taking into account ageing, operating experience, recent research results and developments in international standards – provided their implementation is "reasonably practicable."* [INRAG 2021]

*This requirement for safety improvements in the 2014/87/EURATOM Directive is implemented differently by the regulatory authorities of the individual countries, because the Directive leaves open what "reasonably practicable" means and in what time frame an implementation is still "timely".* [INRAG 2021]

*The WENRA guidelines for new and existing nuclear power plants mean that new reactors are expected to meet higher overall safety levels, and new reactors must meet them – yet existing ageing reactors do not achieve the safety level of a new reactor in all respects, nor is this required.* [INRAG 2021]

## 2.2 Summary and Arguments

- Lifetime extension (LTE) has to address physical ageing of Structures, Systems and Components (SSCs) with safety relevance for the NPP.

- LTE has to consider outdating concepts and obsolescence regarding the three main safety objectives:
    - Criticality control
    - Heat removal/cooling
    - Confinement integrity, barrier against release of radioactive material

- Safety analysis and nuclear safety regulator authorization for operation beyond the originally established design lifetime is based on inspection of the physical plant state.

- Safety upgrades according to state of the art nuclear safety concepts using new NPP as reference. EU Post-Fukushima Stress Tests demonstrated that existing/old NPPs cannot fully comply with requirements for new NPP due to:
    - conceptual deficiencies of outdated design e.g. in:
        - Severe Accident Management (SAM) prevention
        - Severe Accident (SA) mitigation
        - Defence in Depth (DiD) systematic (decoupling of safe-ty layers)
        - Reaching state of the art Core Damage Frequency (CDF) and Large Release Frequency (LRF) based on Probabilistic Safety Assessment (PSA)
        - Implementation of concept of continuously improving Nuclear Safety
    - physical deficiencies e.g.:
        - Available systems (core catcher, ultimate heat sink (UHS), passive safety systems…)
        - Redundancy of Safety Systems
        - Physical separation of Safety Systems, general plant layout, zoning
        - Functional diversification
        - Materials properties

LTE beyond the originally established design lifetime should address these points.

- Most of SSCs can be maintained, refurbished or replaced if necessary. However for some of them this seems not possible from an economic perspective, such as the Reactor Pressure Vessel or major structures of civil engineering such as the hermetic containment structure.

- If safety relevant SSCs cannot be maintained, refurbished or replaced to fulfil limits and conditions even taking benefits from compensatory actions, the plant has reached its physical lifetime.

- Physical lifetime may differ from design lifetime as the first one is based on inspection results and material properties or on increased regulatory requirements with technical implications, the second one on technical estimates and predictions. Usually the physical lifetime should not be reached by design lifetime under conservative approach with safety margins.

- Prolongation of design lifetime is possible for the entire plant and relevant SSCs, if safety critical components are (re-)qualified and (re-)certified. Based on comprehensive safety demonstration design life extension may be authorized if a vendor or operator takes warranty for the SSC to fulfil limits and conditions over the extended period, which therefore has to be defined.

- Material degradation of SSCs are calculated for the design lifetime based on qualified methods. The design lifetime of safety critical SSCs shall be determined. This implies that service lifetime beyond the design life can be understood as Lifetime Extension on a technical and regulatory basis.

- LTE has to include safety upgrades for practical elimination of large and early release of radioisotopes with preference to severe accidents with off-site consequences above intervention levels using and complying with safety requirements for new NPP.

## 3. Guidance on how to determine if a Lifetime Extension presents an Activity or a Major Change to an Activity and on Characteristics of a Major Change

### 3.1 Situations understood as a possible lifetime extension

HERCA-WENRA mentions in its *Approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident*, (Stockholm, 22 October 2014) stating that *Fukushima has shown again that a severe nuclear accident anywhere in the world, including Europe, cannot be completely excluded.* [HERCA-WENRA 2014].

Lifetime extension should therefore be reviewed in transboundary context relevant under the Espoo Convention.

The Espoo Guidance describes *Situations understood as a possible lifetime extension*.

### 3.2 Situation 1: The end date of a time limited licence has been reached, but the plant is intended to continue operation

Depending on national legislation, there are cases with time limited authorization for the operation for the entire plant (e.g. in Ukraine). Different cases are possible under this Situation always with a direct effect of time limitation for the operation:

- Time limited operation license referring to a definite date when the license expires.
- Regulatory provisions with conditions able to limit the operation license in time e.g. quantity of full-power operation years, overall heat or electricity generated, amount of nuclear fuel consumption, maximum amount of generated spent nuclear fuel (SNF), integrated fuel burnup etc. based on safety requirements with technical justification but not on political decisions.

IAEA SSR-2/1 (rev1) proposes requirements for the design life of a NPP. The established design lifetime should be compatible with:

*Requirement 6: Design for a nuclear power plant:*

*The design for a nuclear power plant shall ensure that the plant and items important to safety have the appropriate characteristics to ensure that safety functions can be performed with the necessary reliability, that the plant can be operated safely within the operational limits and conditions for the full duration of its design life and can be safely decommissioned, and that impacts on the environment are minimized.*

*Long term operation should be justified by safety assessment and, depending on the State, this justification may take place within a broader regulatory process, such as licence renewal or a periodic safety review (see IAEA SSG-25).*

- Other legal provisions reflected within the operation authorization, e.g. from construction license relevant for the entire plant as defined by the approval of Preliminary or Provisional Safety Analysis Report (PSAR, POSAR).

*LTE is the extension of the design life which is the period of time during which a facility or component is expected to perform according to the technical specifications to which it was produced* [IAEA Safety Glossary 2018].

Whenever a new NPP project is or has been submitted for receiving its license by the authorities, information on the design life of the NPP is always part of the projects documentation. Any operation beyond that timeframe is to be seen as LTE and falls under the Espoo Convention.

- Time limits of other than the nuclear licensing such as site licensing, time limitations from general construction license, environment impact assessment.

The operation authorization includes more than only the operation license of the nuclear safety regulator, which is one essential license. According to the specific national legislation, other required licenses may be necessary and could include the site license, general construction license and environmental permit, which are not always under prime responsibility of the nuclear safety regulator. Some of these authorizations may be restricted to a defined time or operation end date and would need an explicit decision by the responsible authority to be prolonged.

*(j) The scope of the licence (the site, a nuclear installation, parts of a nuclear installation and activities, or a series of authorizations), its validity period and any incorporated conditions should be clearly defined by the regulatory body.* [IAEA SSG-12]

### 3.3 Situation 2: The nuclear power plant has a time unlimited licence, but the design life of irreplaceable safety critical structures, systems and components has been reached

This situation is a common situation and may appear for all NPPs independent from other limitations for life time.

- All NPPs have irreplaceable safety critical structures, systems and components (SSCs), which reach its design life at a certain state. Examples (irreplaceable or practical irreplaceable) are:
  - Structures: Hermetic confinement or containment
  - Systems: Reactor protection systems (relevant: Highest safety class) in its redundancy, physical separation and functional separation
  - Components: Reactor Pressure Vessel (most Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR)), Calandria for some heavy water reactors, reactor tank etc. for graphite moderated and some gas cooled reactors.

- Time restricted permits for Structures, Systems and Components (SSCs) may induce a time limited operation for the plant, but are usually on a lower regulatory level than licensing of the entire plant. National regulation establishes the legal structure for standardisation, qualification, verification and certification of SSCs depending on the safety function. If a safety critical SSC reaches the established design lifetime two approaches to renew the license can be distinguished:
  - Using the licensing basis of the original plant or with minor deviations according to requirements
  - Relicensing under state of the art provisions as applied for new NPPs with a view to the safety concept including safety upgrades.

Most of safety relevant SSCs can be requalified after inspections after defined time periods, and at the end of the original established design lifetime. If requalification is not possible because of physical ageing or incompliance with current nuclear safety concepts the SCC can be refurbished and maintained, if by this it not possible to reach applicable safety standards the relevant SSC has to be replaced. If replacement is not feasible and compensatory actions would be not effective, the plant is needed to shut-down. Even if the operation license for the entire plant is unlimited, time limited licenses, permits or certificates for safety critical SSCs may limit the overall operation license, if they represent mandatory preconditions. This criteria is valid independently from Situation 1, a plant may have a time limited license but in addition license of irreplaceable SSCs are binding. If these SSCs are irreplaceable and cannot be recovered regarding ageing and enhanced to fulfil actual safety standards and if no other compensatory actions (e.g. comprehensive additional safety systems) are possible the plant reaches its overall design lifetime and has to be shut-down.

IAEA SSR-2/1 establishes the following requirement regarding Design life of items important to safety:

*Requirement 31: The design life of items important to safety at a nuclear power plant shall be determined. Appropriate margins shall be provided in the design to take due account of relevant mechanisms of ageing, neutron embrittlement and wear out and of the potential for age related degradation, to ensure the capability of items important to safety to perform their necessary safety functions throughout their design life.*

Justification is based on physical mechanisms for ageing:

*5.51. The design for a nuclear power plant shall take due account of ageing and wear out effects in all operational states for which a component is credited, including testing, maintenance, maintenance outages, plant states during a postulated initiating event and plant states following a postulated initiating event.* [IAEA SSR-2/1]

*5.52. Provision shall be made for monitoring, testing, sampling and inspection to assess ageing mechanisms predicted at the design stage and to help to identify unanticipated behaviour of the plant or degradation that might occur in service.* [IAEA SSR-2/1]

The same Safety Standard establishes for design life of safety critical components in its Requirement 23: *Reliability of items important to safety: The reliability of items important to safety shall be commensurate with their safety significance.* [IAEA SSR-2/1]

*5.37. The design of items important to safety shall be such as to ensure that the equipment can be qualified, procured, installed, commissioned, operated and maintained to be capable of withstanding, with sufficient reliability and effectiveness, all conditions specified in the design basis for the items.* [IAEA SSR-2/1]

*5.38. In the selection of equipment, consideration shall be given to both spurious operation and unsafe failure modes. Preference shall be given in the selection process to equipment that exhibits a predictable and revealed mode of failure and for which the design facilitates repair or replacement.* [IAEA SSR-2/1]

- All operation states including normal operation and events should be considered for all safety relevant equipment. If refurbishment or repair is not feasible the equipment has to be replaced, if possible. It is therefore of special relevance to avoid degradation of safety relevant SSCs if they cannot be replaced, such as the hermetic containment building or the reactor pressure vessel (RPV) both with fundamental safety functions (barrier for primary coolant inventory and radioactive material, retention of radioactive release to the environment in case of events). The plant lifetime is limited by the lifetime of the unreplaceable safety relevant SSCs which have to demonstrate to fulfil always mandatory limits and conditions.

*Robustness is the ability to ensure that no operational loads or loads caused by postulated initiating events will adversely affect the ability of the SSC to perform its function.* [IAEA SSG-30]

*Requirement 5: Radiation protection in design:*

*The design of a nuclear power plant shall be such as to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits, that they are kept as low as reasonably achievable in operational states for the entire lifetime of the plant, and that they remain below acceptable limits and as low as reasonably achievable in, and following, accident conditions.* [IAEA SSR-2/1]

*Requirement 6: Design for a nuclear power plant:*

*The design for a nuclear power plant shall ensure that the plant and items important to safety have the appropriate characteristics to ensure that safety functions can be performed with the necessary reliability, that the plant can be operated safely within the operational limits and conditions for the full duration of its design life and can be safely decommissioned, and that impacts on the environment are minimized.* [IAEA SSR-2/1]

- According to inspection and analyses the operation lifetime of SSCs may differ from the original established design lifetime, however it may even be shorter depending on physical conditions and obsolescence. Both may therefor limit plant operation lifetime due to safety reasons.

IAEA SSR-2/1 underlines the function of *Design for safe operation over the lifetime of the plant* with:

*Requirement 29: Calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety Items important to safety for a nuclear power plant shall be designed to be calibrated, tested, maintained, repaired or replaced, inspected and monitored as required to ensure their capability of performing their functions and to maintain their integrity in all conditions specified in their design basis.* [IAEA SSR-2/1]

Design life extension of safety critical SSCs has to fulfil the following requirement under deterministic safety analysis as described in IAEA SSR-2/1:

Requirement 30: Qualification of items important to safety

*A qualification programme for items important to safety shall be implemented to verify that items important to safety at a nuclear power plant are capable of performing their intended functions when necessary, and in the prevailing environmental conditions, throughout their design life, with due account taken of plant conditions during maintenance and testing.*

*"Environmental conditions" are here used in terms of technical environment for SSCs such as temperature, pressure, chemical conditions, radiation levels etc.*

## 3.4 Situation 3: A periodic safety review is carried out in support of the decision-making process for a lifetime extension

Periodic Safety Review (PSR) sets up a regulatory framework for repeating and iterative safety reviews based on plant operation history and equipment performance.

- PSR is not limited to operation beyond the originally established design lifetime of the entire plant or reducing to irreplaceable safety critical SSCs. From an economic point of view PSR can provide useful guidance for investment planning.

- Several nuclear safety regulators established mandatory Periodic Safety Reviews for NPPs with repetition rate every 10 years usually for the entire plant. In this cases PSR approval might be a precondition to proceed with operation with an indicated time limit for the following PSR.

- PSR was developed as a regulatory instrument over the last decades based on operation experiences of the running and ageing fleet. For the current NPPs achieving the original established design lifetime of 30 or 40 years PSR was introduced as the plants were already running with the objective to define a systematic tool to formalize reassessment of plant safety in detail. The function of PSR is established according to the regulation and regulatory practice with national responsibility however Guidelines and Standards on best practice are available in international nuclear safety framework, e.g. from IAEA and WENRA.

IAEA SSG-25 introduces role and function of Periodic Safety Review (PSR) as following:
*2.10. A PSR can be used for various purposes:*

*As a systematic safety assessment carried out at regular intervals, as required […];*

o *In support of the decision making process for licence renewal*

o *In support of the decision making process for long term operation.*

Depending on national regulations, the regulatory body has the responsibility for:

o Specifying or approving the requirements to perform the PSR;

o Approving the documentation to be provided by the operating organization prior to the PSR (i.e. the PSR basis document including the project plan);

o Reviewing the actual scope, conduct and findings of the PSR and the resulting safety improvements;

o Assessing the prospects for safe operation for the period until the next PSR;

o Taking appropriate licensing actions;

o Informing the government and the general public about the results of the PSR and resulting safety improvements.

- It is a relatively young development that PSR is also used to support lifetime extension of NPPs reaching its original design lifetime or those induces by safety critical SSCs, especially if they are considered to be irreplaceable with the potential to limit the operation lifetime. PSR supporting LTE has to take into account the specific challenges to extent the original established design lifetime due to

o Termination of the license of the plant or irreplaceable safety critical SSCs

o Termination of vendor warranties and liabilities

o Completion of the original overall object of the plant

o Achieving the originally established time frame for safety analyses

- PSR is able to support LTE, if special challenges are properly ad-dressed and taking note that it cannot replace specific authorization requirements for LTE.

IAEA SSG-25 *underlines:*

*3.8. Where the PSR is to be used in decision making for long term operation or licence renewal, the review should pay particular attention to the following plant programmes and documentation, as these are of significant importance for continued safe operation:*

o Plant programmes to support the safety factors relating to plant design, the actual condition of SSCs important to safety, equipment qualification and ageing;

- A PSR and its findings can be used to support the decision making pro-cess for long term operation, lifetime extension or licence renewal. It has to consider

o Physical plant modification since the previous PSR and refer-ring to the original design basis for safety relevant SSCs

o Addressing physical conditions of SSCs including ageing and material degradation

o Enhanced safety requirements and standards

o To establish and justify a new extended design lifetime to re-place the original design lifetime including all technical, administrative and legal implications. It is definitely not enough to reduce to the 10 year scope of PSR.

IAEA SSG-25 notes in this regard:

*3.4. It is recognized that some States employ alternative arrangements to PSR, which may be equally adequate for justifying extension of the lifetime of a nuclear power plant. In such cases, the necessary plant modifications and related evaluations justifying licence renewal are generally performed separately from each other. If an alternative approach is followed, particular consideration should be given to the scope and objectives of the safety assessments con-ducted, which should be agreed with the regulatory body.*

*3.7. If the PSR is to be used to justify long term operation or licence renewal, the entire planned period of long term operation should be considered, and not just the ten years until the next PSR. Furthermore, if long term operation or licence renewal is approved, PSR should continue to be performed in a ten year cycle or at a frequency as required by the national regulatory body.* [IAEA SSG-25]

*3.8. Where the PSR is to be used in decision making for long term operation or licence renewal, the review should pay particular attention to the following plant programmes and documentation, as these are of significant importance for continued safe operation:*

*o Plant programmes to support the safety factors relating to plant design, the actual condition of SSCs important to safety, equipment qualification and ageing;*

*o A management system that addresses quality management and configuration management;*

*o Safety analyses involving time limiting assumptions relating to the proposed lifetime; […].*

In the frame of LTE in transboundary context its function for the plant safety and therefore for the (re-)qualification of equipment of the highest safety class under different factors is relevant.

IAEA SSG-25 mentions:

*Safety factor 3: Equipment qualification*

*5.37. Plant equipment important to safety (that is, SSCs) should be properly qualified to ensure its capability to perform its safety functions under all relevant operational states and accident* *conditions, including those arising from internal and external events and accidents (such as loss of coolant accidents, high energy line breaks and seismic events or other vibration conditions). The qualification should adopt a graded approach consistent with the safety classification of the SSC and should be an ongoing activity.* [IAEA SSG-25]

*5.40. Qualification of plant equipment important to safety should be formalized using a process that includes generating, documenting and retaining evidence that equipment can perform its safety functions during its installed service life. This should be an ongoing process, from its design through to the end of its service life. The process should take into account plant and equipment ageing and modifications, equipment repairs and refurbishment, equipment failures and replacements, any abnormal operating conditions and changes to the safety analysis. Although many parties (such as designers, equipment manufacturers and consultants) will be involved in the equipment qualification process, the operating organization has the ultimate responsibility for the development and implementation of an adequate plant specific equipment qualification programme.* [IAEA SSG-25]

Equipment (re-)qualification may also be part of a licensing process and there-fore involve the nuclear safety regulator with competence to decide.

PSR focused on design life extension of Structures, Systems and Components and qualification of equipment should consider [IAEA SSG-25]:

- *Whether installed equipment meets the qualification requirements;*
- *The adequacy of the records of equipment qualification;*
- *Procedures for updating and maintaining qualification throughout the service life of the equipment;*

*5.43. The review of equipment qualification should determine:*

- *Whether adequate assurance of the required equipment performance was initially provided;*
- *Whether current equipment qualification specifications and procedures are still valid (for example, initial assumptions regarding the service life of equipment and the environmental conditions);*

*IAEA SSG-25 states on Safety factor 4: Ageing*

*5.45. All SSCs important to the safety of nuclear power plants are subject to some form of physical change caused by ageing, which could eventually impair their safety functions and service lives.*

*5.46. The objective of the review of ageing is to determine whether ageing aspects affecting SSCs important to safety are being effectively managed and whether an effective ageing management programme is in place so that all required safety functions will be delivered for the design lifetime of the plant and, if it is proposed, for long term operation.*

### 3.5 Situation 4: Modification of a nuclear power plant not covered by the existing authorization to operate and therefore requiring a licence modification

Definition of a nuclear power plant not covered by the existing authorization to operate and therefore requiring a licence modification depends on national legislation and regulatory framework. Such modification may include (not exhaustive list):

- Power up-rates above certain absolute or relative values by improvements in process efficiency in the conventional part (e.g. replacement of turbine generator in PWR) or within the nuclear steam supply system (NSS) (e.g. steam generator replacement at PWR)

- Power increase or changes in the reactor core with special view on nuclear fuel and process parameters (neutron density, heat flux, primary coolant mass flow rate, core inlet and outlet temperature, primary circuit pressure, burn-up, criticality control by burnable and not-burnable absorbers etc.)
- Operation regime (e.g. base load to load-follow operation etc.)

- Exchange of major safety critical structures, systems and components (SSCs) due to:
  - Physical ageing and replacement
  - System upgrade or uprate, depending on dimension and safety importance of actions.

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# 5 GLOSSARY

| | |
|---|---|
| AMP | Ageing Management Programme |
| Bq | Becquerel |
| BWR | Boiling Water Reactor |
| CDF | Core Damage Frequency |
| DEC | Design Extension Conditions |
| DiD | Defence in Depth |
| EIA | Environmental Impact Assessment |
| ENSREG | European Nuclear Safety Regulators Group |
| EOP | Emergency Operating Procedures |
| EU | European Union |
| Euratom | European Atomic Energy Community |
| HERCA | Heads of European Radiological protection Competent Authorities |
| AEA | International Atomic Energy Agency |
| INRAG | International Nuclear Risk Assessment Group |
| LOCA | Loss of Coolant Accident |
| LTE | Lifetime Extension |
| LTO | Long Term Operation |
| LRF | Large Release Frequency |
| NPP | Nuclear Power Plant |
| PLiM | Plant Life Management |
| PLEx | Plant Life Extension |
| PSA | Probabilistic Safety Assessment |
| PSR | Periodic Safety Review |
| PWR | Pressurized Water Reactor |
| RCS | Reactor Coolant System |
| RHWG | Reactor Harmonization Working Group (WENRA) |
| RL | Reference Level |
| RPV | Reactor Pressure Vessel |
| SAM | Severe Accident Management |
| SSC | Structure, Systems and Components |
| SSG | Specific Safety Guide (IAEA) |
| SSR | Specific Safety Requirement (IAEA) |
| UNECE | United Nations Economic Commission for Europe |
| UNSCEAR | United Nations Scientific Committee on the Effects of Atomic Radiation |
| WENRA | Western European Nuclear Regulators´ Association |
| WHO | World Health Organization |