sCO2-4-NPP: Innovative sCO2-Based Heat Removal Technology for an Increased Level of Safety of Nuclear Power Plants

Deliverable 3.5

Independent review of the sCO2-4-NPP system licensing roadmap for real nuclear power plant

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R	Document, report excluding the periodic and final reports	X
DEM	Demonstrator, pilot, prototype, plan designs	
DEC	Websites, patents filing, press & media actions, videos, etc.	
OTHER	Software, technical diagram, etc.	
Dissemination level		
PU	PUBLIC, fully open, e.g. web	X
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Deliverable Contributors

Authors

Partner	Name
JSI	ANDREJ PROŠEK
JSI	LEON CIZELJ
JSI	IVO KLJENAK
JSI	IZTOK TISELJ

Internal Reviewers

Partner	Name
EDF	ALBANNIE CAGNAC
USTUTT	JÖRG STARFLINGER

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Abbreviation / Acronym	Description / meaning
AFCEN	Association Française pour les règles de Conception, de construction et de surveillance en exploitation des matériels des Chaudières Electro Nucléaire
ALT	alternative
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASG	emergency feedwater system
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire
BNI	Basic Nuclear Installation
BPVC	Boiler and Pressure Vessel Code
BSR	Basic Safety Rules
BT	bezpečnostní třída (in English 'safety class')
BWR	boiling water reactor
CDF	core damage frequency
CHX	Compact Heat Exchanger
CNRA	Committee on Nuclear Regulatory Activities
COFRAC	Comité français d'accréditation (French accreditation committee)
CSS	Commission on Safety Standards
ČEZ	NPP operator in Czech
DBA	Design Basis Accident
DBC	Design Basis Condition
DBE	Design Basis Earthquake
DBH	Design Basis external Hazards
DEC	Design Extension Condition
DiD	Defence in Depth
DIV	Diverzní prostředky (diverse components)
DPS	Diverse Protection System
DUHS	Diverse Ultimate Heat Sink
EDF	Électricité de France

Abbreviation / Acronym	Description / meaning
EIP	Élément important pour la protection (in English 'element important for protection')
EMC	Electromagnetic Compatibility
EN	Europäische Norm (in English 'European Standard')
EPR	European Pressurised Reactor
ESPN	équipements sous pression nucléaires (in English 'nuclear pressure equipment'
ETSON	European Technical Safety Organization Network
EU	European Union
FSF	fundamental safety function
GOR	General Operating Rules
GPR	Groupe Permanent chargé des Réacteurs nucléaires
GSG	General Specific Guide
GSR	General Safety Requirement
HEPR	High Energy Pipe Break
HFE	Human Factors Engineering
HMI	Human-Machine Interface
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
IAEA-TECDOC	IAEA Technical Document
IBN	installation nucléaire de base (in English 'Basic Nuclear Installation)'
ICRP	International Commission on Radiological Protection
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
INSAG	International Nuclear Safety Advisory Group
IP	important pour la protection (in English 'important for protection')
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
ISO	International Organization for Standardization
KTA	Kerntechnischer Ausschuss (in English 'German Nuclear Standard Committee')
LUHS	Loss of Ultimate Heat Sink
MDE	Maximum Design Earthquake
MSIS	Macro Seismic Intensity Scale

Abbreviation / Acronym	Description / meaning
MTSI	Maintenance, Testing, Surveillance and Inspection
NC	non classé (in English 'unclassified class'
NEA	Nuclear Energy Agency
NO	normal operation
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
OTS	Operating Technical Specifications
PAMS	Post-Accident Monitoring System
PCS	Plant Control System
PGA	Peak Ground Acceleration
PIE	postulated initiating event
PRPS	Primary Reactor Protection System
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Seismic Hazard Assessment
PWR	Pressurized Water Reactor
RBS	safety boration system
RCC	Rules for Design and Construction of Components of PWR Nuclear Islands
RCC-CW	Règles de Conception et de Construction du Génie Civil REP (in English 'Design and construction rules for civil works in PWR nuclear islands')
RCC-E	Règles de Conception et de Construction des matériels Electriques des îlots nucléaires REP (in English 'Design and construction rules for electrical equipment of PWR nuclear islands')
RCC-M	Règles de Conception et de Construction des Matériels mécaniques des îlots nucléaires REP (in English, 'Design and Construction Rules for the Mechanical Components of PWR Nuclear Islands')
RCS	reactor coolant system
RHRS	residual heat removal system
RHWG	Reactor Harmonisation Working Group
RIS	safety injection system
RL	reference level
RQM	Requirements for Qualified Materials

Abbreviation / Acronym	Description / meaning
RSOME	Supervision in Operation of Mechanical Equipment
SA	severe accident
SAP	Safety Assessment Principle
SBO	station blackout
SSB	Systémy Související s Bezpečností (safety related systems)
SSC	stricture, system and component
SF	Safety Fundamentals
SFC	single failure criterion
SL	seismic level
SRL	safety reference level
SRS	Safety Reports Series
SSR	Specific Safety Requirement
SÚJB	Státní úřad pro jadernou bezpečnost (State Office for Nuclear Safety)
TCS	Turbo-Compressor System
TOS	Technical Operating Specifications
UE	L' Union Européenne (in English 'European Union')
VDA	Main Steam Atmospheric Dump System
VIV	Main Steam Isolation Valve
VVER	Vodo-Vodjanoj Energetičeski Reaktor (in English 'water-water energetic reactor')
VVP	Vapeur Vive Principale (Main Steam System)
VyDiD	Významné z hlediska DiD (significant for Defence in Depth)
WENRA	Western European Nuclear Regulators Association
ZBF	Základní Bezpečnostní Funkce (in English 'fundamental safety function')

2 Executive Summary

The objective of this deliverable is to present the results of independent review of the proposed sCO2-4-NPP passive residual heat removal system considering the international experience in licensing similar systems. In the scope of the independent review were deliverables D3.2 ("Requirements for reference plant modifications for installation of sCO2-4-NPP ") [2], D3.3 ("Design bases and safety analyses for system and components") [3] and D3.4 ("Requirements for testing and operation, including requirements for the preoperational and initial start-up test programmes for the system") [4], prepared by project partners from EDF, France and NRI (UJV), Czech Republic. The safety/licensing requirements described of Czech Republic and France, in D3.2 [2], D3.3 [3] and D3.4 [4], were assessed against International Atomic Energy Agency (IAEA) safety standards to ensure that applicable regulations were considered for the sCO2-4-NPP system development. It should be also noted that the methodology used in this report is applicable to several other country regulations.

Deliverable D3.2 [2] deals with requirements for reference plant modifications for installation of the sCO2-4-NPP system. It provides the set of requirements for the implementation of the system in the nuclear power plant (NPP) in selected reactor types from project partner countries (VVER in Czech Republic and PWR in France). The licensing requirements depend on the country regulations, design requirements and other factors that have been considered in the analysis. The starting point for identification of the nuclear regulatory elements was the setup of a hierarchy of regulatory requirements proposed to be used for the sCO2-4-NPP project, both for Czech Republic and France. Then, more specific guidelines are presented for the general safety approach, requirements for structures, systems and components, and requirements for plant modifications.

Deliverable D3.3 [3] provides detailed requirements for the design and operation of the sCO2-4-NPP system in Czech and French NPPs. The specific safety classification of the sCO2-4-NPP system in Czech and French legislation is presented. Then, all the requirements for the design basis of structures, systems and components (SSCs), such as functions to be performed, internal and external hazards, reliability, monitoring and control, etc., are described. Next, the requirements for the qualification of the SSCs are provided. At the end, some of the most important requirements regarding the operation are presented.

Deliverable D3.4 [4] presents the regulations related to the operation and testing of nuclear power plant equipment in the Czech Republic and France, with an extrapolation to the sCO2-4-NPP system developed in the sCO2-4-NPP project.

In this report, the results of independent assessment of the requirements and criteria set up in D3.2 [2], D3.3 [3] and D3.4 [4] are presented. Czech and French requirements were compared to the IAEA safety standards, which represent international consensus on best international practices to achieve a high level of safety. The assessment methodology, based on the IAEA safety assessment process, is described first. Then, the independent review of the regulatory areas has been performed. The regulatory areas considered in this report were the nuclear regulatory framework, general safety approach, requirements for SSCs, requirements for plant modifications, requirements for design basis, requirement for equipment qualification, and requirements for operation and maintenance.

For each regulatory area independent review, the criteria were first described. The criteria are requirements set in the IAEA safety standards (subsection 'IAEA requirements'). Then, the Czech and French requirements for each selected regulatory area were reviewed and summarized (subsections 'Review of…Czech Republic' and 'Review of…France'). Finally, results of the independent review are presented (subsection 'Assessment').

3 Introduction

The objective of this deliverable is to present the results of independent review of the requirements and criteria set up in D3.2 [2] through D3.4 [4] reports for potential installation of the passive heat removal device sCO2-4-NPP. The following areas have been in the scope of the independent review:

- Nuclear regulatory framework;
- General safety approach;
- Requirements for the structures, systems and components (SSCs);
- Requirements for the plant modifications;
- Safety classification of SSCs;
- Requirements for design basis;
- Requirements for qualification;
- Requirements for operation and maintenance.

The OECD/NEA/CNRA survey on the regulatory practice to assess passive safety systems used in new nuclear power plant design [5], presented in D3.1 [1], covered several of the above areas. The purposes of this survey were:

- to improve the regulatory review and assessment of passive safety systems that are used in new nuclear power plant designs by identifying good practices and knowledge gaps, and by sharing experience and guidance;
- to compare national approaches to defining and regulating the use of these passive safety systems.

Based on the above, this survey is of interest for this deliverable, firstly because several national approaches have been compared for assessment of passive safety systems and secondly because of the areas of interest for safety assessment. Several areas presented in D3.2 [2] through D3.4 [4] reports are covered by the survey, consisting of the following five thematic areas:

- requirements for passive safety systems;
- testing and analyses of passive safety systems;
- regulatory review of passive safety systems;
- commissioning and periodic verification testing;
- experience with passive safety systems.

The results of the survey for the five thematic areas are briefly presented in the following.

1. Requirements for passive safety systems

Regarding requirements for passive safety systems, it was observed that many countries do not have specific requirements. The survey showed that there are no differences in the regulatory treatment of systems, irrespectively whether they are passive or active in the following areas:

- providing system descriptions in the safety analysis report;
- protection from tampering;
- establishing operational limits and conditions;
- safety classification;
- protection against external events;
- functional failures identification and consideration;
- substantiation of system parameters;

- instrumentation and control;
- demonstration of the maximum number of passive safety system actuations (including false actuations), and consideration of the equipment design life and environment that it is operating in;
- false actuation considerations and system starting considerations;
- testing during commissioning;
- testing during operation.

2. Testing and analyses of passive safety systems

Survey respondents were asked what safety principles must be demonstrated through testing and analyses, their expectations for the validation of computer codes and the conduct of testing used to demonstrate safety performance. There are no significant differences in the approaches applicable for active and passive systems. Nevertheless, some countries indicate that passive safety systems as a rule require more emphasis on experimental substantiation than on analytical approaches.

The next point of interest was the concurrent operation of several different passive safety systems (trains), in particular the expectations for the testing and analyses required to be demonstrated by the licensee. The same question was formulated for concurrent operation of passive and active safety systems. Responding countries did not report any difference between the passive or active nature of a system. The possible negative effects from concurrent operation of safety systems (either passive or active) shall be analysed and if necessary tested.

3. Regulatory review of passive safety systems

The first question was how are functional failures identified and considered in the safety demonstration (specifically, containment isolation challenged by a leaking passive system penetrating the containment wall and non-condensable gases collecting in heat exchangers). No differences in approaches for active and passive systems were reported.

Then the respondents provided information on their expectations for the applicant or licensee to provide substantiation of passive safety system parameters (e.g. pressure, temperature, flow rates, inventory, procedures, concentration of neutron absorbers, pressure resistance of lines, valve characteristics).

No differences in expectations for active and passive systems were reported.

The third question was related to expectations for the applicant or licensee to provide the results of quantitative and qualitative analysis of passive safety systems reliability. Additionally, any special considerations for the reliability analysis because of passivity should be described by respondents.

A number of respondents indicated that there are no significant differences between passive and active systems here.

The next question was on regulatory expectations for the review of instrumentation and controls supporting passive systems. No differences in expectations for instrumentation and controls supporting active and passive systems were reported.

The next two questions in this chapter were related to expectations for the applicant (licensee) to demonstrate the maximum number of passive safety system actuations (including false actuations) and to evaluate the impact of false actuation (starting) of passive safety systems. Countries were also asked to share any special considerations in the starting procedure of passive safety systems. The countries did not report any differences in expectations for active and passive systems.

The final question was related to expectations for the applicant or licensee to submit information on the response of liquids and gases in the passive flooding systems tanks, such as droplet carryover and possibility for blowdown. The countries reported that there are no differences on this subject.

4. Commissioning and periodic verification testing

The fourth set of questions was devoted to commissioning and periodic verification testing. Questions were asked in the following areas:

- expectations for testing passive safety systems during commissioning;
- expectations for the interval and scope of periodic checks and testing of passive safety systems during nuclear power plant operations.

No differences between active and passive systems were reported by respondents.

5. Experience with passive safety systems

A number of passive safety systems were listed by countries that participated in the survey: passive core flooding systems, passive residual heat removal systems, passive containment cooling systems, passive auxiliary feedwater systems, passive autocatalytic recombiners, etc. Given this experience, it is evident that the regulatory framework and bodies of various countries have dealt with passive safety systems and are in a good position to address them in new reactor applications.

This D3.5 report is organised as following. After introduction in this section, assessment methodology used for independent review of the requirements and criteria set up D3.2 [2] through D3.4 [4] report is presented in Section 4. In Section 5, the assessment results are presented for Czech and French requirements for sCO2-4-NPP system design and operation. Section 5 consists of seven subsections, covering the following regulatory areas:

- nuclear regulatory framework (see Section 5.1),
- general safety approach (see Section 5.2),
- requirements for SSCs (see Section 5.3),
- requirements for plant modifications (see Section 5.4),
- requirements for design basis (see Section 5.5),
- requirement for equipment qualification (see Section 5.6),
- requirements for operation and maintenance (see Section 5.7).

For each regulatory area independent review, the criteria were first described. The criteria are requirements set in IAEA safety standards (see subsection 'IAEA requirements'). Then, the Czech and French requirements for selected regulatory area were reviewed and summarized (see subsections 'Review of…Czech Republic' and 'Review of…France'). Finally, results of the independent review are presented (see subsection 'Assessment').

Finally, In Section 6 conclusions are drawn.

4 Assessment methodology

The task of independent review is to review the safety/licensing requirements of Czech Republic and France, described in D3.2 [2] through D3.4 [4] to ensure that they satisfy applicable regulations. In this report International Atomic Energy Agency (IAEA) safety standards have been chosen for judging applicability of Czech and French requirements and criteria. Namely, IAEA safety standards represent international consensus on best international practices to achieve a high level of safety. Based on presentation [24] the IAEA standards were formally adopted (i.e. China, Netherlands), they were directly used to establish regulation (i.e. Canada, Czech Republic, Germany, India, Korea, Russian Federation), they were used as reference for review of national standards and situations (by all States, also by Industry) and used by international organizations (European Safety Directive, WENRA).

WENRA safety reference levels (SRLs) are a key driver for developing nuclear safety by a continuous improvement and harmonization of regulatory approaches in Europe [8]. WENRA highly appreciates the work of the IAEA and is grateful for its important contributions to enhance nuclear safety world [9]. For example, in [10] it is stated that a set of Reference Levels (RLs) identifying the main relevant requirements on reactor safety was developed for 18 safety issues. These Reference Levels were primarily based on IAEA safety standards. In WENRA RLs 2014 [8] only one new safety issue was added compared to WENRA 2008 RLs [6]. In WENRA RL 2020 [12] a few existing RLs have been extended, but they still comprise 19 issues. Review against changes in knowledge, international standards and other factors have identified the need to introduce the notion of leadership into Issue C (Leadership and Management for Safety) and obsolescence into Issue I (Ageing Management). There was also a need to complete the hazards to be addressed in the safety demonstration. To achieve this, Issue S (Protection against Internal Fires) has been extended to cover all internal hazards (Issue SV), and Issue T (Natural Hazards) has been extended to address all external hazards (Issue TU). All other issues remain unchanged from the previous version. The WENRA RLs 2014 [8] identified in the D3.1 [1] were Issue C (Management System), Issue E (Design Basis Envelope for Existing Reactors), Issue F (Design extension of existing reactors), Issue G (Safety Classification of Structures, Systems and Components), Issue K (Maintenance, In-Service Inspection and Functional Testing), Issue Q (Plant Modifications) and Issue T (Natural Hazards). Quality assurance is mentioned in D3.2 [2] through D3.4 [4] report, but it is not considered as separate regulatory area, therefore Issue C extensions do not influence this report (also, WENRA extensions have been done based on IAEA approach for leadership and management for safety [15]). Issue I is not very much relevant for sCO2-4-NPP projects at this stage. Finally, hazards have been extended by Issue SV and Issue TU. Hazards have some influence on design basis area elements like protection against the effect of internal and external hazards (again some references have been made by WENRA RL 2020 [12] to IAEA safety standards).

It is also important to note that WENRA RLs from 2014 [8] considered the latest IAEA standards at the time of publication (i.e. from 2014 and before). On the other hand, both Czech Republic and France are harmonizing their legislation against WENRA RLs. As of 1 January 2021 the status [11] on WENRA RLs 2014 [8] implementation into regulation as reported by the WENRA countries shows that all RLs have been harmonized in Czech Republic, while in France 73 out of 342 RLs have not yet been harmonized, but they were in progress. Based on these facts, it is reasonable to review compliance of selected country regulations to IAEA standards only, which is also more complete set of safety requirements than WENRA RLs, originally based on IAEA standards (note: IAEA standards cover all regulatory and operational aspects of nuclear and radiation safety).

The IAEA safety standards are not legislation but reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which have three categories: safety fundamentals, safety standards and safety guides. The IAEA safety requirements are divided into general safety requirements (GSRs) and specific safety requirements (SSRs). GSRs are applicable to all facilities and activities, while SSRs are applicable to specified facilities and activities. The IAEA guidance documents are divided into general specific guides (GSG) and specific safety guides (SSGs), applicable to all facilities and activities and to specified facilities and activities, respectively. The IAEA safety requirements used for the assessment are listed in Section 7.4, while IAEA safety guides are listed in Section 7.7.

Safety assessment as proposed by IAEA GSR Part 4 (Rev. 1) [17] is performed at different stages in the lifetime of a facility, including development of the design and modification of the design (i.e. plant modification). For example, the safety assessment process is to be implemented by the designer to fulfil the fundamental safety functions with the appropriate level of defence in depth or to be used by the reviewer of the design to assess the safety of the design. The overview of IAEA safety assessment requirements in accordance with IAEA GSR Part 4 (Rev. 1) [17] is shown in Figure 1. A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out for any particular facility, consistent with the magnitude of the possible radiation risks arising from the facility. During the preparation of the safety assessment it shall be ensured that the necessary resources, information, data, analytical tools as well as safety criteria are identified and are available. As can be seen, part of the assessment process is also (general) safety approach with requirements for defence in depth, multiple barriers and safety margins. The safety assessment includes safety analysis, which consists of a set of different quantitative analyses for evaluating and assessing challenges to safety by means of deterministic and also probabilistic methods.

On the left side of Figure 1, features to be assessed are shown. Possible radiation risks include the radiation exposure to people and the release of radioactive material to the environment following the occurrence of abnormal or accident conditions that lead to a loss of control. Safety functions include the safety functions associated with the engineered safety SSCs, any natural barriers as applicable, and any human actions required to ensure the safety of the facility. The functions have to be carried out with an adequate level of reliability, there should be no vulnerability to a single failure or to a common cause failure for engineered equipment, and any SSC or barrier provided to carry out a safety function has to have an adequate level of redundancy, diversity, separation, segregation, equipment qualification, etc. as appropriate. The site characteristics related to the safety of the facility include: (a) the physical and chemical characteristics that will affect the dispersion or migration of radioactive materials released in normal operation or due to an incident or accident; (b) natural and man-made hazards of the area that have the potential to affect the safety of the facility; and (c) the site demographic characteristics in regard to any siting policy of the Member State and the need to determine an emergency plan. For radiation protection, adequate measures need to be in place to control the occupational radiation exposure within any relevant dose limit and that the protection is optimized such that the magnitude of individual doses, the number of people exposed and the likelihood of incurring exposures have all been kept as low as reasonably achievable, economic and social factors being taken into account. Engineering aspects according to IAEA include implementation of defence in depth, operating experience, radiation protection, classification of SSCs, safety classification of SSCs, aging and wear-out mechanism, protection against internal and external hazards, materials, equipment qualification. Human factors are related to design and operation of any facility and the procedures for any activities. Requirements to comply with human factors include also the ergonomic design of all the areas and man-machine interfaces.

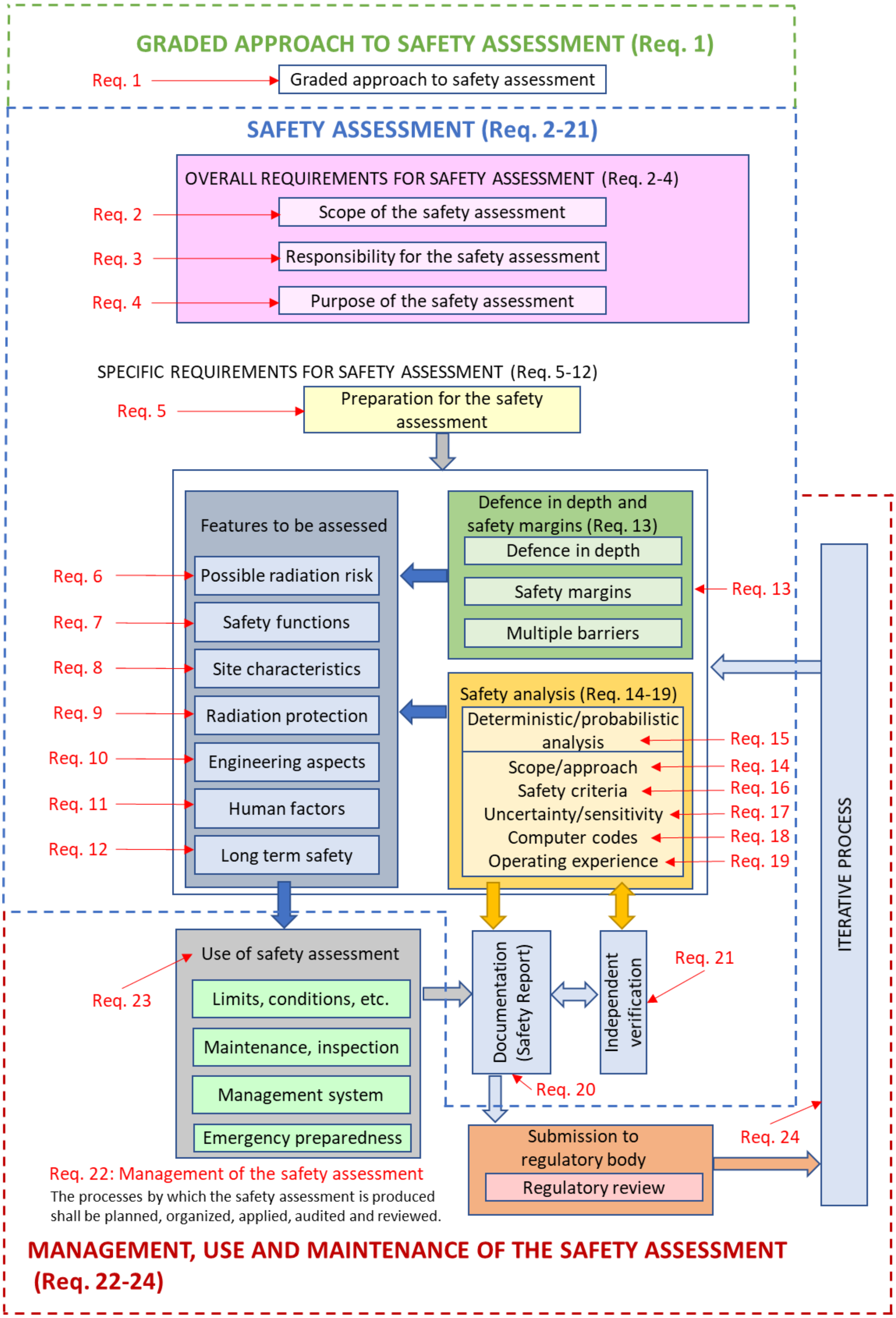

Figure 1: Overview of safety assessment requirements in accordance with GSR Part 4 (Rev. 1) [17]) standard (extended Fig. 1 of IAEA GSR Part 4 (Rev. 1) [17])

Finally, ageing management for nuclear power plants is implemented to ensure that the effects of ageing will not prevent structures, systems and components (SSCs) from being able to accomplish their required safety functions. It takes account of changes that occur with time and use. This requires addressing both the effects of physical ageing of SSCs, resulting in degradation of their performance characteristics, and the non-physical ageing (obsolescence) of SSCs.

At the end of assessment process, safety analysis report should be prepared and submitted to the regulatory body.

IAEA safety requirements standards used in the performed independent review of D3.2 through D3.4 were:

- IAEA SF-1: IAEA Fundamental Safety Principles [13];
- IAEA GSR Part 1 (Rev. 1): Governmental, Legal and Regulatory Framework for Safety General Safety Requirements [14];
- IAEA GSR Part 2 (Rev. 1): Leadership and Management for Safety [15];
- IAEA GSR Part 3: Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [16];
- IAEA GSR Part 4 (Rev.1): Safety Assessment for Facilities and Activities [17];
- IAEA GSR Part 7: Safety Assessment for Facilities and Activities [18];
- IAEA SSR-1: Site Evaluation for Nuclear Installations [19];
- IAEA SSR-2/1 (Rev.1): Safety of Nuclear Power Plants: Design [20];
- IAEA SSR-2/2 (Rev. 1): Safety of Nuclear Power Plants – Commissioning and Operation [21].

5 Independent review of Czech and French requirements for sCO2-4-NPP design and operation

5.1 Nuclear regulatory framework

5.1.1 IAEA structure of safety standards

IAEA safety standards are primarily addressed to national regulatory authorities and cover all regulatory and operational aspects of nuclear and radiation safety. They cover all facilities and activities that can give rise to radiation exposure (only peaceful facilities and activities are covered). Safety standards are non-binding on IAEA member states but may be adopted by them. These standards are obligatory for IAEA's own activities. The hierarchy of IAEA standards is shown in Figure 2.

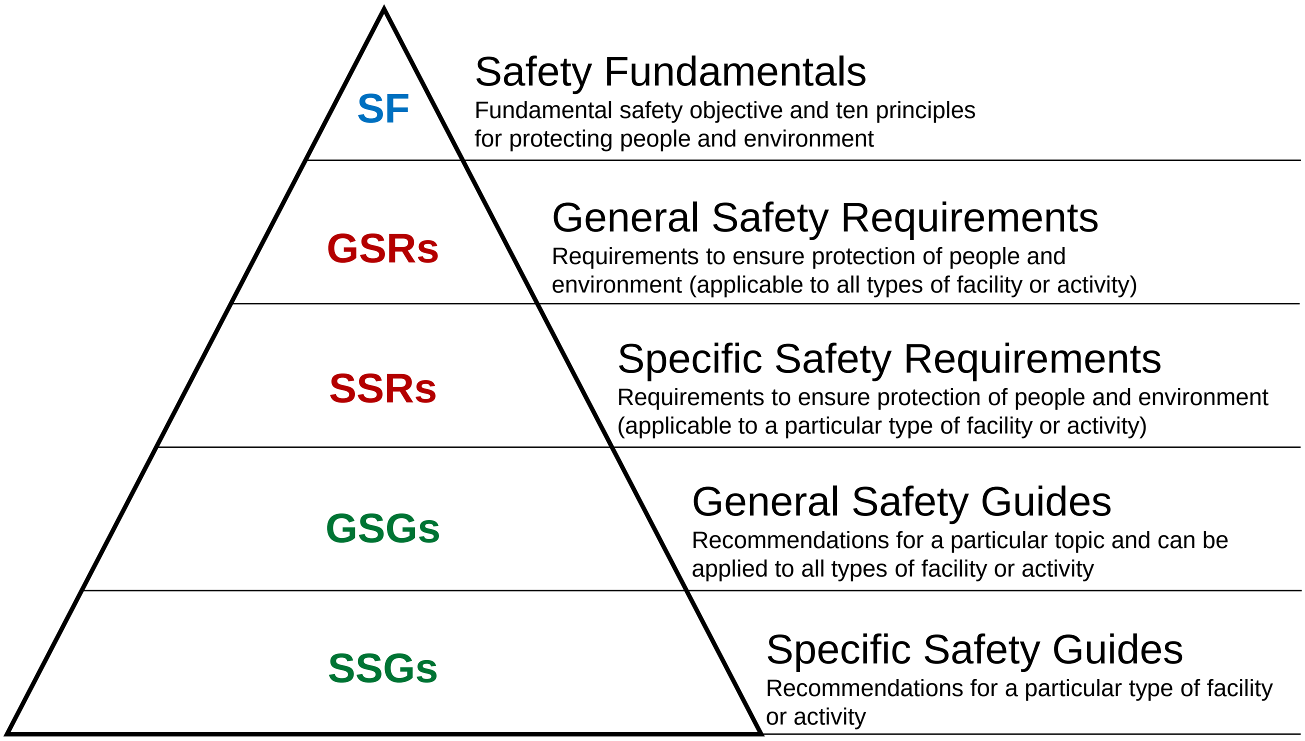

Figure 2: The hierarchy of IAEA standards (adapted per [99]

Figure 2 shows the long term structure of the IAEA safety standards series, with list of standards with safety requirements. The requirements are governed by the objective and principles of the Safety Fundamentals. It should be noted that the format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Also, IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions [14]. Therefore, they are appropriate to assess to what extent the national requirements meet the set IAEA requirements. There are seven IAEA standards on general safety requirements (identified as Parts 1 to 7). As explained in Figure 2, general safety requirements are applicable to all types of facilities (for the purpose of this report the focus is on nuclear power plants). From the point of sCO2-4-NPP system for heat removal development in the frame of sCO2-4-NPP project, the most relevant parts are GSR Part 1 [14] with requirements for setting up the regulatory framework, GSR Part 2 [15] with requirements for leadership and management system (including quality assurance) and GSR Part 4 [17] with requirements for safety assessment. Among the IAEA safety standards with specific safety requirements the most relevant are specific safety requirements for safety of nuclear power plants, which are the target type of facility. As shown in Figure 3, there are two such standards, SSR-2/1 [20] for design and SSR-2/2 [21] for commissioning and operation. Finally, it should be noted that the whole set of Requirements is in line with the revised long-term structure (the publication of the new SSR-1 [19] in 2019 marks the completion of whole set of requirements). On the other hand, the revision of safety guides is an ongoing process.

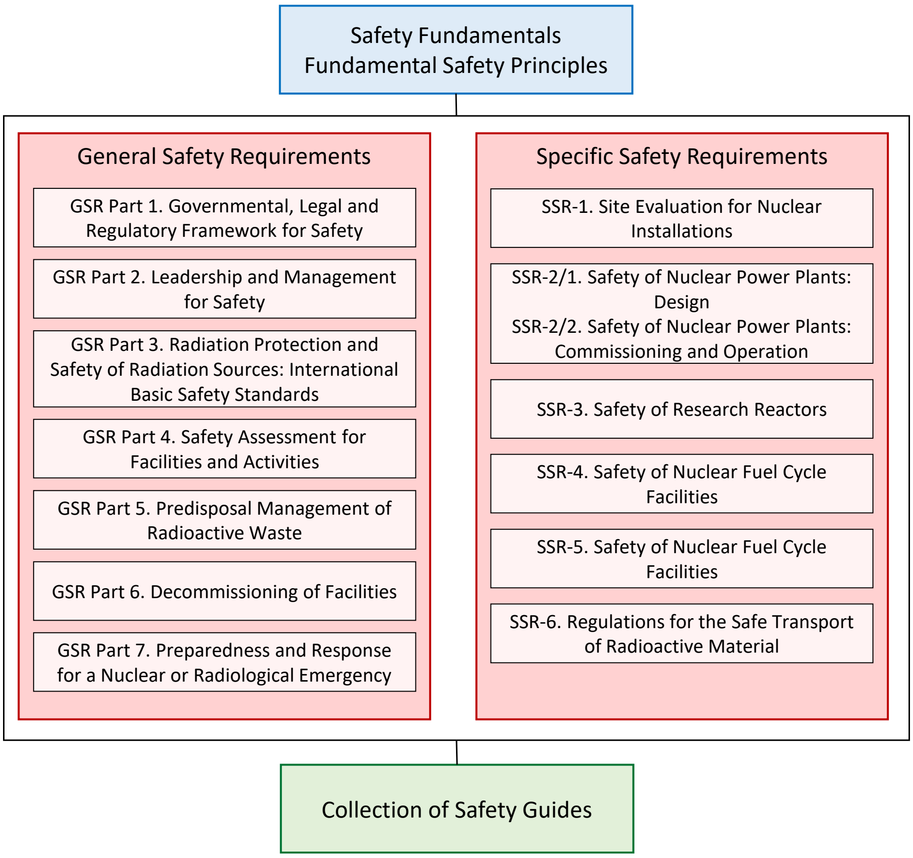

Figure 3: The long term structure of the IAEA Safety Standards Series (adapted per FIG. 1 of [14])

In the following the comparison of Czech Republic and France nuclear regulatory framework to IAEA structure is done. In addition, the nuclear regulatory framework of Czech Republic and France is generally assessed against selected requirements set in IAEA GSR Part 1 [14], which requires that the government shall establish and maintain an appropriate governmental, legal and regulatory framework for safety within which responsibilities are clearly allocated. This framework for safety among other shall set out also the following:

"*(1) The safety principles for protecting people — individually and collectively — society and the environment from radiation risks, both at present and in the future;*

(2) The types of facilities and activities that are included within the scope of the framework for safety;

(8) Provision for the review and assessment of facilities and activities, in accordance with a graded approach."

5.1.2 Review of nuclear regulatory framework in Czech Republic

As explained in Section 4.1 of D3.2 [2], the requirements can be arranged into a 5-level "pyramid". The hierarchy of nuclear regulations in Czech Republic is shown in Figure 4. This first level includes Czech legislation of all degrees currently in force, international conventions and agreements which are bounding for the Czech Republic, and directly bounding EU regulations. All these are generally binding documents. They were issued by Czech government, Czech national nuclear regulator (SÚJB), EU Commission, Euratom and other international institutions.

Some of the important Czech Republic documents are listed in Section 4.1 of D3.2 [2]. For the sCO2-4-NPP design the most relevant documents are:

a) Act No. 18/1997, Act on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act);
b) the new ACT No. 263/2016 of Coll. Of 14th July 2016 Atomic Act [25],
c) IMPLEMENTING DECREE No. 162 of 25th May 2017 On The Requirements for Safety Assessment According to the Atomic Act [27], and
d) IMPLEMENTING DECREE No. 329 of 26th September 2017 On The Requirements For Nuclear Installation Design [29].

Decree No. 329/2017 [29] incorporates the relevant Euratom legislation and governs:

a) the requirements for the contents of documentation for licensed activities,
b) the list of safety functions that must be performed by nuclear installations and classification of the functions into categories according to their impact to nuclear safety,
c) safety classes and the criteria for classifying selected equipment into these classes,
d) the method of ensuring defence-in-depth, and
e) the content of the requirements for nuclear installation design.

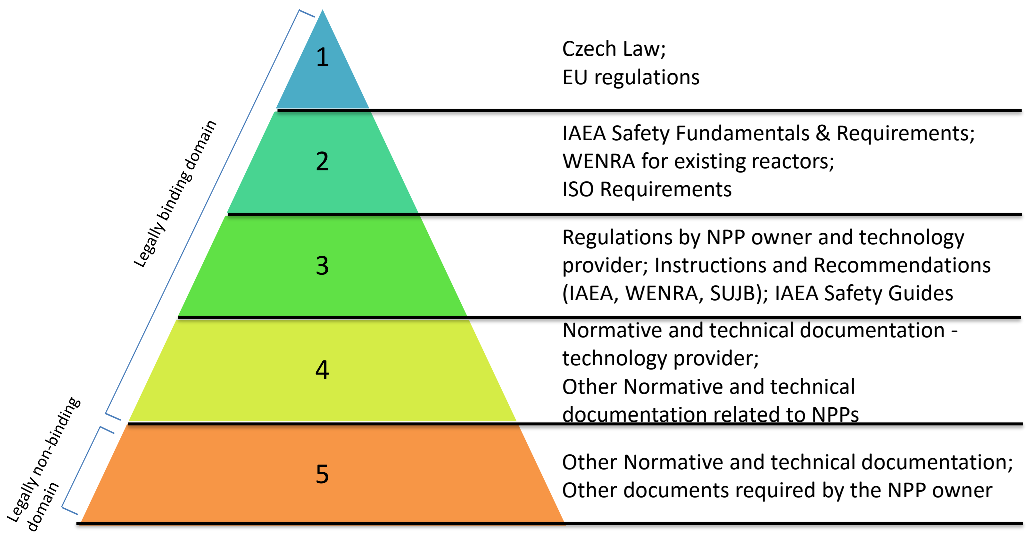

Figure 4. Hierarchy of nuclear regulations in Czech Republic [2]

Second level comprise the IAEA, WENRA and ISO documents. Some of the most important documents considered in the Czech Republic are:

- SF-1. IAEA Fundamental safety principles (2006) [13],
- SSR-2/1 Rev.1, IAEA Safety of Nuclear Power Plants: Design (2016) [20],
- SSR-2/2 Rev.1, IAEA Safety of Nuclear Power Plants: Commissioning and Operation (2016) [21],

- SSR-1, IAEA Site evaluation for nuclear installations (2019) [19],
- GSR Part 3, IAEA Radiation Protection and Safety of Radiation Sources: International basic safety standards (2014) [16],
- GSR Part 4 Rev. 1, IAEA Safety Assessment for Facilities and Activities (2016) [17],
- Safety of new NPP designs 01/2013, WENRA [7],
- WENRA/RHWG Updating WENRA Reference Levels for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons (2014) [8].

As of 2019, the Czech Republic has implemented all WENRA 2014 requirements.

Level 3 includes Nuclear Safety Regulations of the country of origin of the technology, IAEA Safety Guides, Instructions and recommendations of SÚJB, IAEA (including TECDOC) and WENRA, and Fundamental requirements for the NPPs set by the NPP owner.

Finally, level 4 includes normative and technical documentation of the country of origin of the technology related to nuclear energy, and other normative and technical documentation related to nuclear energy.

It can be judged that nuclear regulations in Czech Republic in general comply with IAEA modern standards structure. At the top are national and EU regulations, followed by IAEA safety requirements and WENRA requirements for existing reactors, which are legally binding documents. As already stated, the Czech Republic has implemented all WENRA 2014 requirements. The IAEA standards listed under level 2 include most important safety standards like safety fundamentals (SF-1), general safety requirements (GSR Part 3 and GSR Part 4) and specific safety requirements (SSR-1, SSR-2/1 and SSR-2/2). It was identified that no information regarding GSR Part 2 is given. Also, no detail on implemented IAEA requirements is given. Nevertheless, by implementing WENRA reference levels (RLs) from 2014, which presented harmonized European requirements for existing reactors, it is expected that Czech Republic legislation is a good representative for licensing process. The potential implementation of IAEA internationally established safety standards for design of nuclear power plants, with the broader scope than that of WENRA, presents complemented requirements to that of WENRA.

Finally, the Czech Republic level 3 (like IAEA) includes safety guides with recommendations. Legally binding domain is also level 4 with normative and technical documentation from technology provider and other documentation related to NPPs. This is much more detailed information comparing to WENRA RLs or IAEA safety standards. These documents were issued by A.S.I. (Association of Mechanical Engineers, Czech Republic), ANSI (American National Standards Institute), IEEE (Institute of Electrical and Electronics Engineers), Gosgortechnadzor, NRC, KTA (Kerntechnischer Ausschuss - German Nuclear Standard Committee) etc.

5.1.3 Review of nuclear regulatory framework in France

As explained in Section 4.2 of D3.2 [2], nuclear legislation in France was developed in successive stages alongside technological advances and growth in the atomic energy field. The hierarchy of nuclear regulations in France is shown in Figure 4. The legal framework specific to nuclear activities finds its source in norms, standards or recommendations established by various international bodies (IAEA, International Standard Organisation (ISO) and European Union). Based on these recommendations and directives, the French government can then issue the decrees and orders necessary to regulate nuclear activities in France. The responsibilities belong to the French government. Decrees and orders are taken at ministerial level. ASN proposes or gives advice on these ministerial decisions. The ASN issues regulatory decisions (mainly technical rules) are to be endorsed by the French government. Once it’s done, the ASN publish Guides that explain how to consider the corresponding regulation. Theses Guides are not prescriptive.

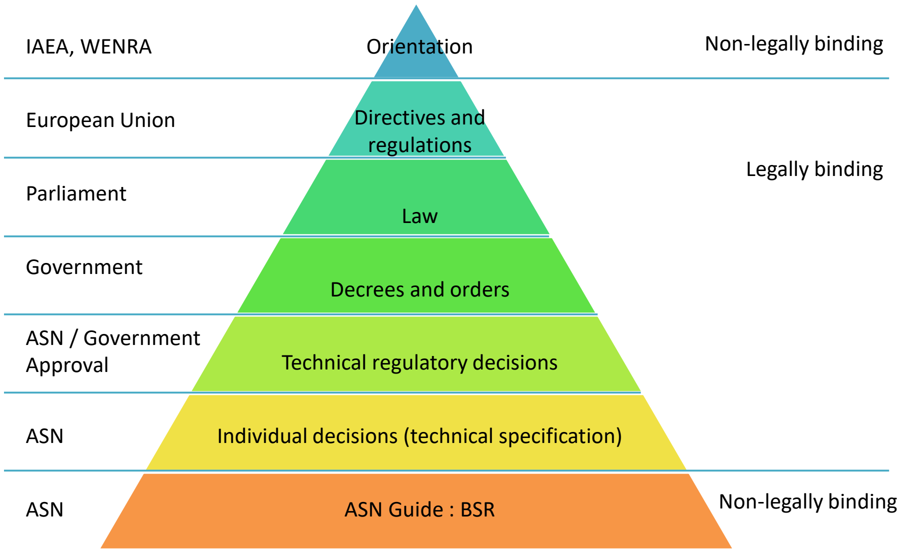

Figure 5. Hierarchy of nuclear regulations in France (adapted Figure 7 of [2])

Several legislative and regulatory provisions relating to BNIs (Basic Nuclear Installation) are derived from or incorporate international conventions and standards, in particular those of the IAEA [13]. Several community texts are applicable to BNI. The most important of these are Euratom Treaty and Directive of June 25, 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Directive 2014/87/Euratom of July 8, 2014.

The Euratom Treaty, signed in 1957 and entered into force in 1958, aims to develop nuclear energy by ensuring the protection of the population and workers against the harmful effects of ionizing radiation (the information provisions of the European Commission have been integrated into the decrees and the environmental code.

The Directive defines the fundamental obligations and general principles in this field. It strengthens the role of national regulatory bodies, contributes to the harmonization of safety requirements between Member States for the development of a high level of safety of installations and promotes transparency on these issues. It takes into account the harmonization work carried out by the Western European Nuclear Regulators Association (WENRA).

The provisions of Chapters III, V and VI of Title IX of Book V of the Environment Code provide the basis for the authorization and control regime for BNIs. Chapter III of Title IX of Book V of the Environment Code (regulatory part) defines the framework within which procedures relating to BNI are conducted and deals with the entire life cycle of a BNI, from the definition of safety options, its creation authorization and commissioning, to its final shutdown and dismantling, and then its decommissioning. It sets out the relationship between the Minister in charge of nuclear safety and the ASN in the field of BNI safety. These provisions define the procedures applicable for the adoption of general regulations and the taking of individual decisions relating to BNIs. It also defines the specific conditions for the application of certain administrative regimes within the perimeter of BNIs.

General technical regulations, provided for in Article L. 593-4 of the French Environment Code, include all the texts of general scope setting technical rules in the field of nuclear safety.

In 2012, the French government implemented the order of February 7, 2012 ("INB Order") [41] which defines the requirements applicable to basic nuclear installations for the protection of the interests listed by law: public safety, health and hygiene, protection of nature and the environment. These requirements are applicable to the design, construction, operation, decommissioning, dismantling, maintenance and monitoring of BNIs. The order recalls the principles protection of all the interests mentioned in Article L. 593-1 of the

Environmental Code (public safety, health and hygiene or the protection of nature and the environment), beyond the mere prevention of accidents. The order also recalls "*the priority given to the protection of the above-mentioned interests, first and foremost through the prevention of accidents and the limitation of their consequences in terms of nuclear safety*".

In application of Article L. 592-20 of the Environment Code, the ASN may take regulatory decisions of a technical nature to specify decrees and orders issued in matters of nuclear safety or radiation protection. The most important in sCO2-4-NPP are the following:

- 2016-DC-0571 of October 11, 2016 on various provisions relating to the compliance of nuclear pressure equipment [47];
- 2017-DC-0616 of November 30, 2017 on significant modifications to basic nuclear installations [48];
- 2014-DC-0417 of January 28, 2014, known as the "fire decision": rules applicable to basic nuclear installations for the control of risks related to fire [49];
- 2014-DC-0462 of October 7, 2014: control of criticality risk in basic nuclear installations [50];
- 2015-DC-0532 of November 17, 2015 relating to the safety report for basic nuclear installations [51].

The Basic Safety Rules (BSR) are recommendations that specify safety objectives and describe practices that the ASN considers satisfactory. As part of the current restructuring of the general technical regulations applicable to basic nuclear installations, the BSR are gradually being replaced by ASN guides. The collection of ASN guides currently includes more than thirty guides of a non-prescriptive nature. The ASN guide that interests us most in the sCO2-4-NPP project will be Guide n°22 (ASN) [43]. It presents ASN and IRSN recommendations for the "Design of Pressurized Water Reactors" (PWR).

5.1.4 Assessment

Based on the review of Czech requirements in Section 5.1.2 it can be judged that nuclear regulations in France in large extent met the intent of IAEA modern standards structure. At Level 2 are IAEA Safety Fundamentals and Requirements, and at Level 3 are IAEA Safety Guides. The same hierarchy is followed for WENRA documents, at Level 2 are WENRA requirements and at Level 3 are WENRA recommendations (i.e. guides). The fact that as for 2019, the Czech Republic has implemented all WENRA 2014 requirements is also important. Also, most important IAEA safety standards with requirements are listed (compare Czech list of IAEA documents with long term structure of IAEA safety standards shown in Figure 3).

Based on review of French requirements in Section 5.1.3 it can be judged that nuclear regulations in France in general comply with IAEA modern standards structure. At the top are European Union directives and regulations. Several legislative and regulatory provisions are derived from or incorporate IAEA standards. The next level is the law (environmental code), which defines the framework within which procedure related to BNIs are conducted. Under law are general technical regulations, i.e. "INB Order" [41] and ASN decisions. "INB Order" sets general rules relating to basic nuclear installations and ANS specify decrees and orders issued in matters of nuclear safety or radiation protection. At the bottom are non-legally binding basic safety rules and ASN guides. It should be also noted that levels of regulation in France in the nuclear field are broader in scope than IAEA. Also, it is the environmental code which defines the main concepts for nuclear activities (several countries adopt an atomic act). When comparing French legislation with Czech legislation, it can be judged that French regulation is more complex. The IAEA standards are not directly used to establish regulation, but they are used for orientation.

5.2 General safety approach

5.2.1 IAEA general safety approach

The IAEA general safety approach, as defined in IAEA GSR Part 4 (Rev. 1) [17] is mainly based on the concept of defence in depth. High quality of SSCs, conservatism and safety margins are used. Plant is deterministically designed against a broad set of postulated events according to established design criteria. There is also capability to deal with conditions that are not considered in the design basis.

The defence in depth concept consists of a set of procedures as well as components, classified in levels, to maintain the effectiveness of physical boundaries placed between radioactive materials and workers, the public and the environment. Each level should prevent degradation of the next level and mitigate the consequences of failure of the previous level. The efficiency of mitigation must not lead to cutbacks in prevention, which takes precedence.
The application of the defence in depth principle to the design is secured by a series of defence levels (intrinsic characteristics, components, procedures) aimed at preventing accidents and at providing an appropriate protection should the prevention fail. Five different levels of defence have been identified.

5.2.1.1 IAEA fundamental safety objective and principles

According to IAEA SF-1 [13] the fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation. IAEA will pursue this fundamental safety objective in accordance with the following ten safety principles:

Principle 1: Responsibility for safety

Principle 2: Role of government

Principle 3: Leadership and management for safety

Principle 4: Justification of facilities and activities

Principle 5: Optimization of protection

Principle 6: Limitation of risks to individuals

Principle 7: Protection of present and future generations

Principle 8: Prevention of accidents

Principle 9: Emergency preparedness and response

Principle 10: Protective actions to reduce existing or unregulated radiation risks

The fundamental safety objective applies to all circumstances that give rise to radiation risks. They provide the basis for requirements and measures for the protection of people and the environment against radiation risks and for the safety of facilities and activities that give rise to radiation risks, including, in particular, nuclear installations and uses of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste. For more detailed information the reader is referred to IAEA SF-1 [13].

5.2.1.2 Defence in depth

In the paragraph 3.31 of IAEA SF-1 [13] under the "Principle 8: Prevention of accidents" it is stated: "*The primary means of preventing and mitigating the consequences of accidents is 'defence in depth'*".

IAEA GSR Part 4 (Rev. 1) [17] standard gives requirements on defence in depth in the Requirement 13:

"*Requirement 13: Assessment of defence in depth*

It shall be determined in the assessment of defence in depth whether adequate provisions have been made at each of the levels of defence in depth."

Associated paragraphs 4.45 to 4.48A to Requirement 13 deal with assessment that: a) adequate provisions have been made at each of the levels of defence in depth, b) the identification of the necessary layers of protection, including physical barriers to confine radioactive material at specific locations, c) whether defence in depth has been adequately implemented, d) whether there are adequate safety margins in the design and operation of the facility and e) there are adequate margins to avoid cliff edge effects.

Section 2 of IAEA SSR‑2/1 (Rev. 1) [20] is devoted to applying the IAEA safety principles [13] and concepts. The concept of defence in depth is described in paragraphs 2.12-2.14 of IAEA SSR‑2/1 (Rev. 1) [20]. IAEA SSR‑2/1 (Rev. 1) [20] requirements on defence in depth are given in the Requirement 23 (with associated paragraphs 4.9–4.13A):

"*Requirement 7: Application of defence in depth*

The design of a nuclear power plant shall incorporate defence in depth. The levels of defence in depth shall be independent as far as is practicable."

The paragraphs 4.9–4.13A of IAEA SSR‑2/1 (Rev. 1) [20] deal with requirements that: a) defence in depth concept shall be applied to provide several levels of defence, b) the existence of multiple levels of defence is not a basis for continued operation in the absence of one level of defence, c) ensure that the concept of defence in depth is maintained, and d) levels of defence in depth shall be independent as far as practicable. The requirements for design are specified in paragraph 4.11 of IAEA SSR‑2/1 (Rev. 1) [20]:

"*4.11. The design:*

(a) Shall provide for multiple physical barriers to the release of radioactive material to the environment;

(b) Shall be conservative, and the construction shall be of high quality, so as to provide assurance that failures and deviations from normal operation are minimized, that accidents are prevented as far as is practicable and that a small deviation in a plant parameter does not lead to a cliff edge effect;

(c) Shall provide for the control of plant behaviour by means of inherent and engineered features, such that failures and deviations from normal operation requiring actuation of safety systems are minimized or excluded by design, to the extent possible;

(d) Shall provide for supplementing the control of the plant by means of automatic actuation of safety systems, such that failures and deviations from normal operation that exceed the capability of control systems can be controlled with a high level of confidence, and the need for operator actions in the early phase of these failures or deviations from normal operation is minimized;

(e) Shall provide for systems, structures and components and procedures to control the course of and, as far as practicable, to limit the consequences of failures and deviations from normal operation that exceed the capability of safety systems;

(f) Shall provide multiple means for ensuring that each of the fundamental safety functions is performed, thereby ensuring the effectiveness of the barriers and mitigating the consequences of any failure or deviation from normal operation."

The current IAEA approach to defence in depth in IAEA SSR-2/1 (Rev. 1) [20] is presented in Table 1 [96]. The main difference with the original table of INSAG-10 [90] is represented by the introduction of the DECs. A slight elaboration of the third and fourth levels of defence in depth and minor changes in the wording have been done not to impair the general approach. The column of the 'essential means' has been split in two to better indicate essential means related to design and those related to operation. Table 1 displays in addition the plant states (see column 'Objectives') associated with each level of defence in depth for the two different approaches. It is important to note that currently there is no unanimous understanding among IAEA Member States about the association of all the levels of defence in depth with the plant states defined in IAEA SSR-2/1 (Rev.1) [20]. The point of discrepancy is the association of DECs without fuel degradation to one of the levels of defence in depth defined in IAEA SSR-2/1 (Rev. 1) [20]. Some Member States associate them to the level 3 and others associate them to the level 4.

Table 1: Levels of defence in depth for the design of new nuclear power plants (adapted TABLE 4 of [96])

Level of defence Approach 1	Objective	Essential design means	Essential operational means	Level of defence Approach 2
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction of normal operation systems, including monitoring and control systems	Operational rules and normal operating procedures	Level 1
Level 2	Control of abnormal operation and detection of failures	Limitation and protection systems and other surveillance features	Abnormal operating procedures/emergency operating procedures	Level 2
Level 3 — 3a	Control of design basis accidents	Engineered safety features (safety systems)	Emergency operating procedures	Level 3
Level 3 — 3b	Control of design extension conditions to prevent core melt	Safety features for design extension conditions without core melt	Emergency operating procedures	Level 4 — 4a
Level 4	Control of design extension conditions to mitigate the consequences of severe accidents	Safety features for design extension conditions with core melt. Technical Support Centre	Complementary emergency operating procedures/ severe accident management guidelines	Level 4 — 4b
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	On-site and off-site emergency response facilities	On-site and off-site emergency plans	Level 5

Approach 1, i.e. the association of DECs without core melt to level 3, has the advantage that each level has clear objectives regarding the progression of the accident and the protection of the barriers, i.e. level 3 to prevent damage to the reactor core and level 4 to mitigate severe accidents for preventing off site contamination.

Approach 2, i.e. the grouping of DECs without core melt and with core melt in level 4, facilitates however the differentiation between the set of rules for design and safety assessment to be applied for DECs from those for DBA.

Regardless of the approach used, the subject of fundamental importance is the appropriate definition of the rules and criteria to be applied in the design and safety assessment of safety features for DEC and the consistent implementation of requirements for independence of safety provisions for DBA and DEC. For example, WENRA [7] revised structure of the levels of DiD in accordance with Approach 1. The question has been discussed by RHWG whether for multiple failure events, a new level of defence should be defined, because the safety systems which are needed to control the postulated single initiating events fail and thus another level of defence should take over. At the stage of the discussion of WENRA document [7] preparation, it has been proposed to treat the multiple failures conditions as part of the 3rd level of DiD, but with a clear distinction between means and conditions (sub levels 3a and 3b).

5.2.1.3 Plant states

IAEA Fundamental Safety Principles [13] in the design of a NPP are required to be demonstrated for the broad spectrum of plant states including: operational states (normal operation and anticipated operational occurrences) and accident conditions. As shown in Figure 6, after publication of IAEA Fundamental Safety Principles [13] in 2006 the original plant states were revised first in 2012 by introducing design extension conditions (DECs) and then in 2016 by dividing DEC into DEC without significant fuel degradation and DEC with core melting (see Revision 1 of IAEA SSR-2/1 [20]). It should be noted that the concept of DEC is not completely new since some multiple failures of safety systems have been considered in the design and safety assessment of existing NPPs or their importance was recognized and requirements were issued to back fit the existing designs.

The control of DECs is expected to be achieved primarily by features implemented in the design and not only by accident management measures that are using equipment designed for other purposes. Such features are called safety features for DECs. The ‘safety features for DECs’ are obviously items important to safety but, although their safety functions are similar to those performed by ‘safety systems,’ they are considered separately since they may be designed with rules and acceptance criteria different from those used for safety systems.

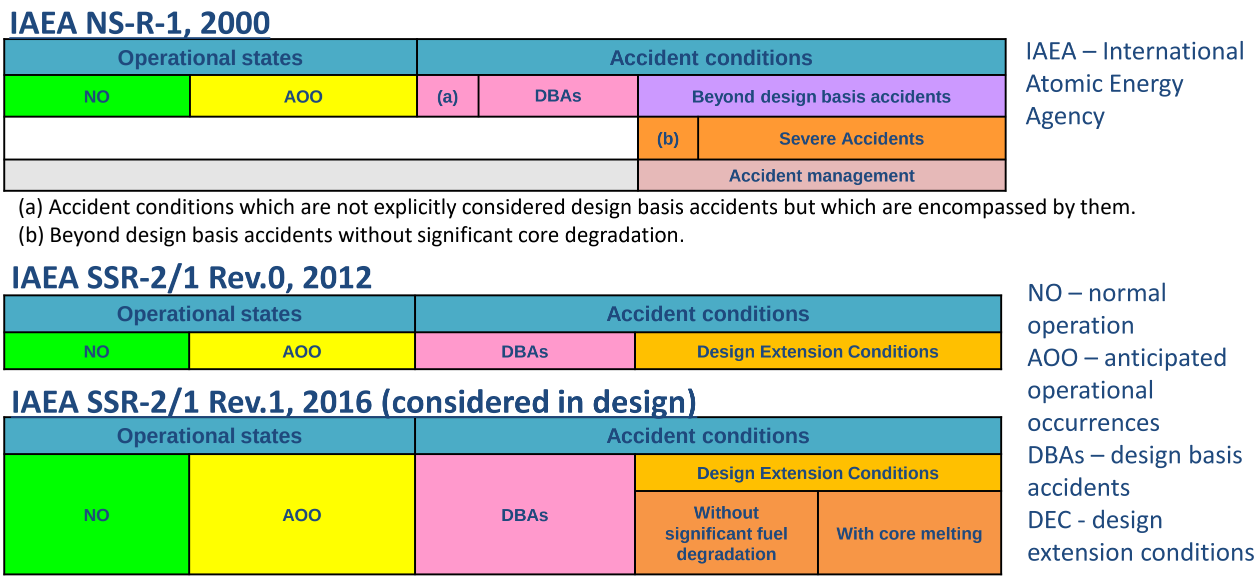

Figure 6. Evolution of IAEA plant states

5.2.1.4 Acceptance criteria

The demonstration of adequacy of the design to cope with different plant states includes the demonstration of the compliance with the acceptance criteria, which are established, following a graded approach, for each plant state. The application of the graded approach leads to acceptance criteria more restrictive for events with higher probability of occurrence.

Acceptance criteria are established in terms of acceptable radiological consequences and in terms of degree of integrity of barriers against releases of radioactive substances (fuel matrix, fuel cladding, reactor coolant pressure boundary or containment) – see Table 2. High level criteria are typically expressed in terms of discharges or releases of radioactive material to the environment, whole body effective doses, equivalent doses for selected organs or tissues, and radioactivity or contamination levels of ground, water, crops and food items. Derived criteria are typically expressed in terms of surrogate variables determining integrity of barriers, such as pressures, temperatures, stresses, strains, etc.

Table 2: Examples of acceptance criteria for different plant states (adapted TABLE 5 of [96])

Level of defence	Objective	Associated plant state	Criteria for maintaining integrity of barriers	Criteria for limitation of radiological consequences
Level 1	Prevention of abnormal operation and failures	Normal operation	No failure of any of the physical barriers except minor operational leakages	Negligible radiological impact beyond immediate vicinity of the plant. Acceptable effective dose limits are bounded by the general radiation protection limit for the public (1 mSv /year20 commensurate with typical doses due to natural background), typically in the order of 0.1 mSv/year.
Level 2	Control of abnormal operation and detection of failures	Anticipated operational occurrence	No failure of any of the physical barriers except minor operational leakages	Negligible radiological impact beyond immediate vicinity of the plant. Acceptable effective dose limits are similar as for normal operation, limiting the impact per event and for the period of 1 year following the event (0.1 mSv/y)
Level 3a	Control of design basis accidents (DBAs)	Design basis accident	No consequential damage of the reactor coolant system, maintaining containment integrity, limited damage of the fuel	No or only minor radiological impact beyond immediate vicinity of the plant, without the need for any off-site emergency actions. Acceptable effective dose limits are typically in the order of few mSv.
Level 3b	Control of DECs without significant fuel degradation (prevention of accident progression into severe accident)	Design extension condition without significant fuel degradation	No consequential damage of the reactor coolant system, maintaining containment integrity, limited damage of the fuel.	The same or similar radiological acceptance criteria as for the most unlikely design basis accidents
Level 4	Control of DECs with core melt (mitigation of consequences of severe accidents)	Design extension condition with core melt (severe accident)	Maintaining containment integrity	Only emergency countermeasures that are of limited scope in terms of area and time are necessary
Level 5	Mitigation of radiological consequences of significant releases	Accident with releases requiring implementation of emergency countermeasures	Containment integrity severely impacted, or containment disabled or bypassed	Off-site radiological impact necessitating emergency countermeasures

5.2.1.5 Safety analyses

IAEA SF-1 [13] Principle 8: Prevention of accidents (with associated paragraphs 3.30–3.33) states that all practical efforts must be made to prevent and mitigate nuclear or radiation accidents. It is also explained that the primary means of preventing and mitigating the consequences of accidents is ‘defence in depth’.

IAEA GSR Part 4 (Rev. 1) [17] requirements for safety analysis are Requirements 14-21. Both deterministic and probabilistic approaches shall be included in the safety analysis. The aim of the deterministic approach is to specify and apply a set of deterministic rules and requirements for the design and operation of facilities or for the planning and conduct of activities. The objectives of a probabilistic safety analysis are to determine all significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined.

IAEA SSR-2/1 (Rev. 1) [20] Requirement 42: Safety analysis of the plant design states:

"*A safety analysis of the design for the nuclear power plant shall be conducted in which methods of both deterministic analysis and probabilistic analysis shall be applied to enable the challenges to safety in the various categories of plant states to be evaluated and assessed*."

The guidance for deterministic safety analysis is given in IAEA SSG-2 (Rev. 1) [57] and probabilistic safety analysis in IAEA SSG-3 [58] and IAEA SSG-4 [59] (it should be noted that both safety guides are before 2013, what means that they will be updated in the future; currently draft specific safety guide DS523 [87] is available presenting revision of SSG-3 [58] and document preparatory profile DS528 [88] for revision of SSG-4 [59]).

According to IAEA SSG-2 (Rev. 1) [57] deterministic safety analyses are primarily required to demonstrate adequate fulfilment of safety functions by the design, to ensure that barriers to the release of radioactive material will prevent an uncontrolled release to the environment for all plant states, and to demonstrate the validity of the operational limits and conditions. There are four options how to perform deterministic safety analysis. For design extension conditions realistic option is recommended.

Table 3: Options for performing deterministic safety analysis (adapted TABLE 1 of [57])

Option	**Computer code type**	**Assumptions about systems availability**	**Type of initial and boundary conditions**
Conservative	Conservative	Conservative	Conservative
Combined	Best estimate	Conservative	Conservative
Best estimate plus uncertainty	Best estimate	Conservative	Best estimate Partly most unfavourable conditions
Realistic*	Best estimate	Best estimate	Best estimate

* For simplicity, the terms ‘realistic approach’ or ‘realistic analysis’ are used in this Safety Guide to mean best estimate analysis without quantification of uncertainties.

5.2.2 Review of general safety approach in Czech Republic

5.2.2.1 Safety analysis

The Czech general safety approach described in D3.2 [2] consider safety analysis, defence in depth, plant states and acceptance criteria.

As explained in Section 5.1.1 of D3.2 [2] safety analyses are judged to be a key part of the safety assessment process, which follows Decree No. 162/2017 [27] (see Section 5.1.2). Necessary analyses must be performed during the design, operation and modifications of the NPP. Safety analyses should demonstrate the fulfilment of basic safety functions, consider the occurrence of the most-limiting single failure of safety systems with an active actuation, verify the effectiveness of the intervention of the safety systems in the least favourable conditions envisaged by the design after the considered PIE, demonstrate that the influence of initial parameters and calculation procedures uncertainties was taken into account in the NPP evaluation, and be performed both with deterministic and probabilistic approach.

Deterministic safety analyses are one of the key demonstrations of the safety of a given nuclear installation as a whole. Their deterministic character is given both by a deterministically defined set of PIEs and by deterministically applied assumption of a most-limiting single failure. The most important safety objectives are determined as acceptance criteria for safety analyses, given for individual categories of initiating events. The criteria are usually different depending on the frequency of occurrence of the given initiating event, more restrictive criteria are set for events with a higher frequency of occurrence. The criteria are divided into radiation (ensuring the acceptability of radiation consequences) and technical (ensuring the functionality of physical barriers against the release of radioactive substances). Radiation acceptance criteria are set by the regulator. The technical acceptance criteria are set by the designer of the nuclear power plant and must be accepted by the regulator.

Deterministic analyses of Anticipated Operational Occurrences (AOO) and Design Basis Accidents (DBA) are performed by conservative computational analyses using realistic codes. Conservatism can be ensured by choosing input data, boundary and initial conditions (including possible uncertainties in parameter settings) and accepting additional limiting assumptions (e.g. application of single failures).

Deterministic analyses for design extension conditions (DEC) are performed by realistic calculations, given that it is important to get the best possible approximation to the actual response of the power plant. Realistic assumptions about the configuration and behaviour of the equipment and mitigated acceptance criteria compared to design accident analyses can be used. For DEC-B analyses, the appropriate assumptions are made according to the individual sets of measures for mitigating the consequences of severe accidents.

The requirements are based on IAEA SSG-2 (Rev. 1) [57], Decree No. 162/2017 [27] and Decree No. 329/2017 [29]. The above approach for deterministic safety analysis is judged that in general conforms to IAEA safety standards. IAEA SSG-2 (Rev. 1) [57] safety standard recommends conservative analyses for AOO and DBA, while for DEC realistic approach is proposed (see Table 3 above presenting four IAEA options).

The probabilistic safety assessment must include two levels of analysis, Level 1 and Level 2. Level 1 of a probabilistic safety assessment is performed to identify the sequence and likelihood of events that could lead to damage to the nuclear fuel or other SSC resulting in radioactive releases. Level 2 of the probabilistic safety assessment includes the analysis of scenarios identified in PSA Level 1 as leading to damage to nuclear fuel or other important SSC. The basis for the evaluation of probabilistic safety assessment is a complex model of a given nuclear facility structured into event trees and fault trees. Unlike deterministic safety analyses, PSA studies are not limited to analyses of a defined PIE spectrum, but concern all realistically possible emergency situations and their development scenarios, including situations with simultaneous multiple component and system failures where important safety features are not fully available or not available at all. The numerical result of these probabilistic analyses in the final phase is usually the probability of core melting (PSA level 1), and the probability of release of radioactive substances into the environment (PSA level 2).

Probabilistic safety analyses of nuclear power plants must be performed according to IAEA SSG-3 [58].

5.2.2.2 Defence in depth

Defence in depth (DiD) is the basic safety principle of nuclear facilities. In general, the design of a nuclear installation must set out requirements for systems, structures and components and procedures to implement safety functions to protect the integrity and functionality of physical safety barriers at various levels of DiD so that these barriers prevent a radiological release. In accordance with Law 263/2016 Coll. [25] and Decree No. 329/2017 [29] and with international requirements (IAEA SSR-2/1 (Rev. 1) [20], WENRA Safety Reference Levels for Existing Reactors [8]) compliance with the basic objectives of nuclear safety must be ensured. The DiD strategy must also cover the role and influences of the human factor and must be included in the procedures, regulations and instructions for the normal and emergency conditions of the NPP.

5.2.2.3 Plant states

For nuclear facilities with a reactor (NPP unit including auxiliary equipment) there is a classification of the operating conditions, according to the requirements of Decree No. 329/2017 [29] the following states are considered:

- normal state (normal operation - NO),
- abnormal conditions (anticipated operational occurrence - AOO),
- basic emergency conditions (design basis accidents - DBA),
- extended emergency conditions (design extension conditions - DEC):
    - DEC-A (DEC without fuel damage),
    - DEC-B (severe accident – SA),
- Unlikely extreme events.

The transients classified as DEC-A can be further divided into two groups [2]:

Local complex transients typical for one block, e.g. the abnormal conditions or design basis accident with subsequent failure of one or more safety systems.
Global complex transients that may affect more units or the entire NPP. These are typically SBO, LUHS and combinations of the above.

Design extension conditions without significant fuel damage are solved within the level of protection to the depth of DiD Level 3b. These kinds of accidents are particularly important from the point of view of sCO2-4-NPP project.

5.2.2.4 Acceptance criteria

In Section 5.1.4 of D3.2 [2] it is stated that the determination of the criteria is based on the IAEA TECDOC-1791 [96]. The specific acceptance criteria for radiation protection in Czech Republic are given in the Annex to the SÚJB letter no. SÚJB/RO/9326/2019 and are 0.1 mSv/1 y, 1 mSv/1 y, 10 mSv/1 y and 100 mSv/1 y for AOO, DBA, DEC A and DEC B, respectively.

5.2.3 Review of general safety approach in France

5.2.3.1 Safe objectives to be achieved

As explained in Section 5.1.2 of D3.2 [2] Article 1.2 of the "INB Order" [41] states that the operator of an NBI must ensure that the provisions adopted for the exercise of the activities mentioned in Article 1.1. (i.e. the design, construction, operation, decommissioning, dismantling, maintenance and surveillance of basic nuclear installations) make it possible to achieve, taking into account the state of knowledge, practices and environmental vulnerability, a level of risks and inconveniences [...] as low as possible under economically acceptable conditions. These provisions, which apply for the entire lifetime of the plant, must therefore also be complied with when applying for a modification. For this reason, it is recommended that the same approach be applied to the design of the sCO2-4-NPP system as to the design of a power plant.

According to IAEA safety standard SF-1 (IAEA, 2006) [13], the fundamental safety objective is to protect people and the environment against the effects of ionising radiation (see Section 5.2.1.1). This fundamental safety objective of the IAEA as well as the Fundamental Safety Principles that must be applied to achieve it must be retained in all stages of the design of the sCO2-4-NPP system.

The safety objectives for nuclear power plants have been defined by WENRA [7] on the basis of a review of the IAEA Fundamental Safety Principles [13]. These safety objectives defined by WENRA and consistent with the "INB Order" [41] will be applied for the design of the sCO2 system.

In the Table 2 of D3.2 [2], it was summarised the declination adopted by EDF for WENRA's safety objectives. This declination has been used in the preparation and design of the latest French power plant concepts (EPR and EPR New Model). For brevity reason, the deviations in description are given in Table 4, while for deviations in explanation the reader is referred to Table 2 of D3.2 [2].

Table 4: WENRA safety objectives in French nuclear regulation

WENRA objective	WENRA Description	French description
O1	Normal operation, abnormal events and prevention of accidents	Normal operation and prevention of incidents and accidents
O2	Accidents without core melt	Accidents without core or fuel meltdown
O3	Accidents with core melt	Accidents with core or fuel meltdown
O4	Independence between all levels of defence-in-depth	Sufficient independence between levels of defence in depth
O5	Safety and security interfaces	Interfaces between nuclear safety and security
O6	Radiation protection and waste management	Radiation protection and waste management
O7	Management of safety	Safety management

5.2.3.2 Safety requirements

Based on information in Section 5.2.2 of D3.2 [2] the French operators opt for a deterministic approach based on the principle of defense in depth supplemented by a probabilistic verification for their safety approach at the design stage. The design of the sCO2-4-NPP system and the modification request will have to follow the same approach.

The principles adopted by the French operators for defense in depth are the same as those for the Czech plants (see Section 5.2.2.2).

The basic safety functions, which must not be degraded by the sCO2-4-NPP system are the following (INB Order [41]):

- Control of nuclear chain reactions;
- Evacuation of thermal power from radioactive substances and nuclear reactions;

- Containment of radioactive substances;
- Protection of people and the environment against ionising radiation.

Compliance with the first three fundamental safety functions makes it possible to ensure the fourth. These functions must be ensured for the different levels of defence in depth.

The function of containment of radioactive substances is ensured by the interposition, between these substances and people and the environment, of one or more successive barriers sufficiently independent of each other (INB Order [41]). Thus, three containment barriers are provided between the fuel located in the reactor core and the environment:

- First barrier: fuel cladding;
- Second barrier: primary circuit jacket;
- Third barrier: containment and associated isolation devices.

General safety approach (deterministic approach based on the principle of defense in depth supplemented by a probabilistic verification for their safety approach at the design stage) requires determining the events likely to affect a barrier or a safety function and then defining the measures to be implemented on the installation to prevent these events and limit their consequences if they are plausible.

For the design of the sCO2-4-NPP system, it will also be necessary to apply this approach whose steps are as follows:

- Identification of the events that may affect the nuclear safety of the installation,
- Consideration of events that may affect the nuclear safety of the installation,
- Analysis within the Reference Design Domain (reference operating conditions),
- Analysis within the Extended Design Domain (the plant is faced with more complex or severe initiating events than those considered in the reference design domain).

5.2.3.3 Plant states

The reference operating conditions design basis conditions (DBC) are classified into five categories:

- DBC1: normal operating transients;
- DBC2: incidents that may occur at least once in the life of the installation;
- DBC3: accidents with a low probability of occurring during the life of the installation;
- DBC4: hypothetical accidents that are assumed to be unlikely to occur during the life of the facility;
- DBC5: hypothetical accidents that are assumed to occur during the life of the facility life of the installation.

The extended design domain includes multiple failure operating conditions (DEC A) as well as core meltdown operating conditions (DEC B). DEC A are associated with level 3b of defense in depth. The safety objective associated with DEC-A multiple failure operating conditions is the absence of core meltdown and no or minor radiological impact. The radiological consequences of DEC A accident conditions are compared to the dose limits defined for accidents without core meltdown or fuel meltdown. DEC B accident conditions are associated with level 4 of defence in depth. The safety objective associated with DEC B core meltdown accident conditions is that public protection measures remain limited in space and time. The radiological consequences of the DEC B accident conditions are compared to the dose limits defined for core meltdown accidents.

The sCO2-4-NPP system developed in the sCO2-4-NPP project will have to be part of the overall safety strategy for a power plant. It must:

- Be part of one of the levels of defence in depth set up by the operator,

- Be developed to improve the response of operators to the second fundamental function and thus participate in the fourth function (for fundamental safety functions see Section 5.2.3.2),
- Do not interfere with containment barriers. Particular attention will be paid to its installation in relation to the containment in order not to weaken it.
- Be developed following the same principles of the general operator safety approach.

5.2.4 Assessment

Based on the review of Czech requirements in Section 5.2.2 it can be judged that general safety approach meets the intent of IAEA general safety approach. Safety analyses are based also on IAEA SSG-2 (Rev. 1) [57] and IAEA SSG-3 [58] recommendations. The defence in depth comprise of five levels: DiD1, DiD2, DiD3a, DiD3b, DiD4 and DiD5 (for more details see Section 5.1.2 of D3.2 [2]). These levels correspond to Approach 1 in above Table 1, i.e. revised structure of the levels of DiD proposed by WENRA RHWG RLs 2014 [8]. It is judged that the plant states defined by Czech Republic follows the WENRA approach, which was based on IAEA definitions of DEC. It can be judged that Czech Republic regarding integrity of barriers follows IAEA TECDOC-1791 [96], while criteria for limitation of radiological consequences are slightly different (stricter for DBA and slightly less strict for DEC A). However, it should be noted that according to IAEA TECDOC-1791 [96] the example acceptance criteria shown in Table 2 are to be understood as design targets rather than as regulatory acceptance criteria. The approach for probabilistic safety analysis is judged that in general conforms to IAEA safety standards, especially as probabilistic safety analyses must be performed according to IAEA SSG-3 [58].

Based on the review of French requirements in Section 5.2.3 it can be judged that the general safety approach in general conform to international approach by IAEA and WENRA.

The largest deviation seems to be independence between levels of defence in depth. WENRA objective is "*enhancing the effectiveness of the independence between all levels of defence-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels separately as addressed in the previous three objectives) to provide, as far as reasonably achievable, an overall reinforcement of defence in depth*". French objective is "*ensure sufficient independence between the different levels of defense in depth, in particular by resorting where necessary to diversification, to ensure an overall strengthening of defense in depth*".

The WENRA safety objectives [7] are grounded on IAEA SF-1 fundamental safety principles (see Section 5.2.1.1), as they were recognized to be a good basis. These fundamental safety principles have been used to ground the proposed safety objectives for new reactors. In the context of the WENRA study [7], the following fundamental safety principles have been found to be especially relevant for improvement of safety of new reactors:

Principle 3: Leadership and management for safety (for O5 and O7)

Principle 5: Optimization of protection (for O1 to O4 and O6)

Principle 6: Limitation of risks to individuals (for O2, O3 and O6)

Principle 8: Prevention of accidents (for O1 to O5 and O7)

It is judged that French safety objectives (with some deviation) in general conform to WENRA safety objectives [7], grounded on IAEA SF-1 fundamental safety principles.

It is judged that the French approach conform to IAEA approach relying on deterministic and probabilistic analysis, defence in depth and fundamental safety functions performed by SSCs important to safety. The events to be considered in the frame of sCO2-4-NPP are station blackout and loss of ultimate heat sink, which may affect a barrier.

The French approach regarding DEC A and DEC B is judged to be conformed with by WENRA and follows IAEA approach (defence in depth uses Levels 3a and 3b, which comply with IAEA Approach 1 – see Table 1).

5.3 Requirements for structures, systems and components (SSCs)

5.3.1 IAEA requirements for SSCs

5.3.1.1 Categorization of safety functions and classification of SSCs

IAEA GSR Part 4 (Rev. 1) [17], paragraph 4.30 states (associated to Requirement 10, Assessment of engineering aspect):

"*It has to be determined in the safety assessment whether a suitable safety classification scheme has been formulated and applied to structures, systems and components. It has to be determined whether the safety classification scheme adequately reflects the importance to safety of structures, systems and components, the severity of the consequences of their failure, the requirement for them to be available in anticipated operational occurrences and accident conditions, and the need for them to be adequately qualified. It also has to be determined in the safety assessment whether the scheme identifies the appropriate industry codes and standards and the regulatory requirements that need to be applied in the design, manufacturing, construction and inspection of engineered features, in the development of procedures and in the management system for the facility or activity*."

IAEA SSR-2/1 (Rev. 1) [20] Requirement 22 for safety classification states:

"*All items important to safety shall be identified and shall be classified on the basis of their function and their safety significance.*"

The reliability of items important to safety shall be commensurate with their safety significance (Requirement 23 of IAEA SSR-2/1 (Rev. 1) [20]).

The objective of IAEA SSG-30 [64] is to provide recommendations and guidance on how to meet the requirements established in paragraph 4.30 of IAEA GSR Part 4 (Rev. 1) [17] and Requirement 22 (with associated paragraphs 5.34-5.36) of IAEA SSR-2/1 (Rev. 1) [20] for the identification of SSCs important to safety and for their classification on the basis of their function and safety significance.

IAEA SSG-30 [64] proposes a structured process for identifying and classifying the SSCs, which is illustrated in Figure 7. For easier understanding the boxes were numbered as steps in Figure 7 (not used in original FIG. 1. of IAEA SSG-30 [64]) and IAEA TECDOC-1787 [95] was used for explanation of these steps.

Step 1: Prior to starting the safety classification process, it is necessary to understand how the plant is designed and to know the radiological release limits (consequences) established by the regulatory body for operational conditions and for the different accident conditions.

Step 2: One of the goals of the IAEA SSG-30 [64] is also to make the whole classification of systems and associated SSCs clearer and more consistent by suggesting defining first the required functions to be accomplished for all of the plant states. Defining functions also enables identification of all of the systems that have to operate together to accomplish a particular function, and consequently makes the classification clearer and more consistent by assigning all of the systems requested for one function to the same safety class.

Step 3: According to IAEA SSG-30 [64] three factors should be used to categorize the identified safety functions into safety categories according to their safety significance:

1) The consequences of failure to perform the function;

2) The frequency of occurrence of the postulated initiating event for which the function will be called upon;

3) The significance of the contribution of the function in achieving either a controlled state or a safe state.

Step 4: The objective of safety classification is to link the safety significance of functions to design requirements (capability, reliability and robustness) of the SSCs performing these functions. The safety significance at the component level is expected to be correctly reflected considering both the functional role and the barrier confinement role (if relevant) of the component. All SSCs performing the categorized functions need to be identified and classified once the categorization of the safety functions is completed. The identification and classification of SSCs is made in two sub steps. The classification starts at the system level and continues to the component level.

Step 5: Using this information, the functions and design provisions required to fulfil the main safety functions are systematically identified for all plant states, including all modes of normal operation. An SSC implemented as a design provision should however be classified directly, because the significance of its postulated failure fully defines its safety class without any need for detailed analysis of the category of the associated safety function. See Table 5 below on differences between function and design provision.

Step 6: As explained in IAEA SSG-30 [64] paragraph 2.9, design provisions can be directly classified according to the severity of consequences of their failures:

Safety class 1 - Any SSC whose failure would lead to consequences of ‘high’ severity;

Safety class 2 - Any SSC whose failure would lead to consequences of ‘medium’ severity;

Safety class 3 - Any SSC whose failure would lead to consequences of ‘low’ severity.

Step 7: The adequacy of the safety classification should be verified by using deterministic safety analysis, which should be complemented by insights from probabilistic safety assessment and/or supported by engineering judgment. See also Figure 8 showing that design provisions are implemented primarily to decrease the probability of an accident and functions are implemented to make the consequences acceptable with regard to its probability. For most initiating events, a combination of both design provisions and functions is implemented to decrease the frequency of occurrence of an accident and to make its consequences acceptable and also as low as practicable.

Step 8: See Section 5.3.1.2 below.

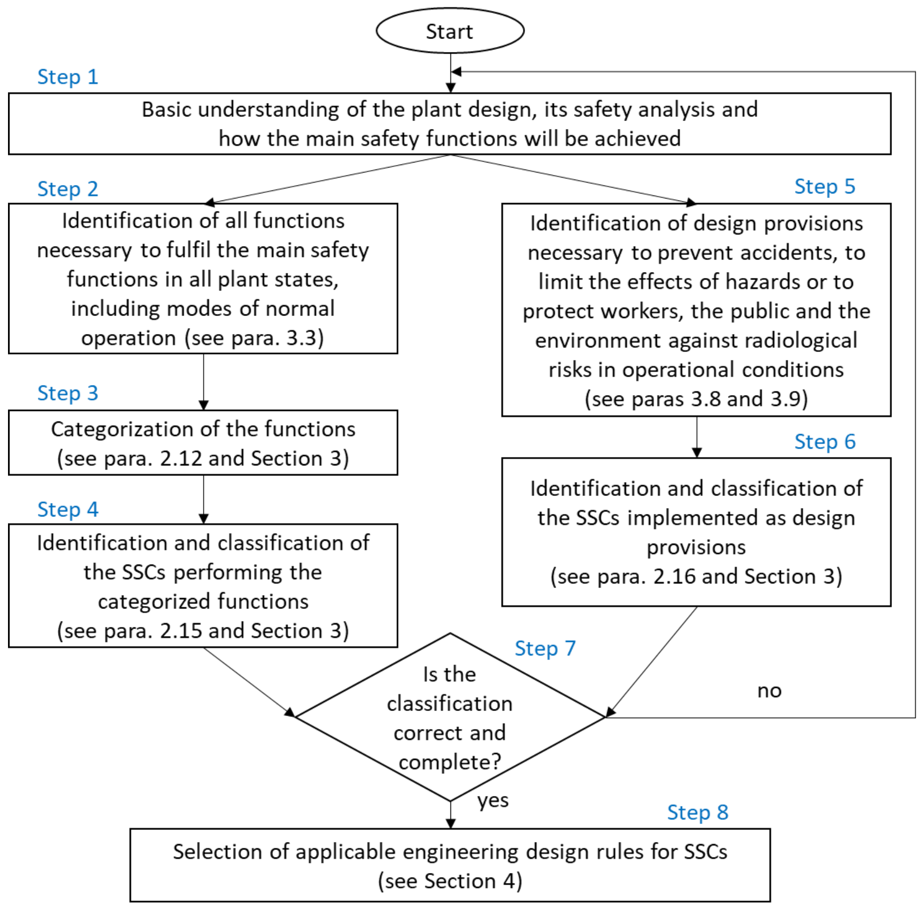

Figure 7. Flow chart indicating the classification process (adapted per FIG. 1 of [64])

Table 5: Difference between 'Function' and 'Design provision' [95]

Function	Design provision
Generally accomplished by a safety functional group consisting of several SSCs (including supporting systems)	Generally linked to a single SSC or to a limited number of SSCs
Generally called upon during an event (i.e. the function is actuated after the occurrence of the event) Note: Some functions are also used in normal operation (e.g. control of main plant parameters).	Not called upon during an event but provides its inherent characteristics during all operational and accidental plant states (i.e. typically, the design provision is not actuated by an I&C signal).

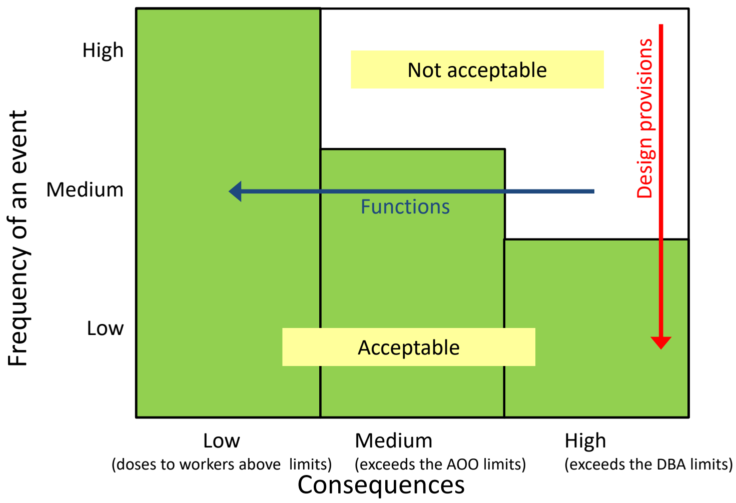

Figure 8. The basic principle of frequency versus consequences (adapted per FIG. 2 of [64])

5.3.1.2 Requirements for SSCs

Once the safety classes of the SSCs have been established, corresponding engineering design rules will have to be specified and applied. Engineering design rules are the relevant national or international codes, standards and proven engineering practices that are applied, as appropriate, to the design of SSCs to meet the overall objective that the most frequent postulated initiating events yield little or no adverse consequences, while more extreme events (those having the potential for the greatest consequences) have a very low probability of occurrence. Depending on its safety significance, reflected by its safety class, each SSC is designed, manufactured and operated according to appropriate engineering rules defined to give confidence that its capability, reliability and robustness will be adequate.

It is reasonable to distinguish between design requirements that apply at the system level and those that apply to individual structures and components:

- Design requirements applied at the system level may include specific requirements, such as single failure criteria, independence of redundancies, diversity and testability, but also general requirements for environmental, seismic and hazard qualification.
- Design requirements applied for individual structures and components define the needs with regard to environmental and seismic qualification, and manufacturing quality assurance procedures. They are typically expressed by specifying the codes or standards that apply.

5.3.1.2.1 Requirements applicable to system

Document IAEA TECDOC-1787 [95] gives as example a set of typical generic design requirements for systems, as shown in Table 6.

Moreover, seismic requirements and environmental qualification are essential to ensure the integrity of buildings and structures or the operability of components if required in case of an earthquake or during accident conditions with harsh ambient conditions. Any system designed to mitigate the consequences of an accident is expected to be designed according to requirements ensuring its operability when challenged. Nevertheless, qualification requirements are defined at the individual component level taking into account the relevant environmental conditions at the component location and its mission time.

Table 6: Example of typical safety requirements for systems (TABLE 17 of [95])

System safety class	Single failure criterion	Physical & electrical separation	Emergency power supply	Periodic tests	Protected against or designed to withstand hazard loads	Environmental qualification
SC1	Yes	Yes	Yes	Yes	Yes	Yes
SC2 (1)	Yes (1)	Yes	Yes	Yes	Yes	Yes
SC2 (2)	Not required (3) (4)	Yes for redundant equipment	Yes	Yes	Yes	Yes
SC3 (5)	Not required (6)	Yes for redundant equipment	Yes	Yes	Yes	Yes
SC3 (7)	Not required	Not required	According to functional analysis	Yes (8)	According to functional analysis	According to functional analysis

(1): Systems necessary to reach and maintain a safe state. Reaching a safe state should be possible despite one single failure.

(2): Systems designed for design extension conditions as a backup of a system assigned to safety class 1. Independence from the safety class 1 system is necessary.

(3): System designed as a backup of a system assigned to safety class 1 already provides an alternate means to accomplish the same safety function as that performed by the safety class 1 system. Nevertheless, reliability of such system needs to be adequate to meet the total core damage frequency (CDF) target.

(4): Might be needed for I&C backup system to prevent spurious actuation (e.g. for the diverse actuation system).

(5): Systems designed to mitigate the consequences of design extension conditions but not assigned to safety class 2.

(6): Compliance with the single failure criterion is not required in design extension conditions. However redundant active components might be necessary to achieve the reliability expected for the function to be accomplished by the system (e.g. active components of systems required to preserve the containment integrity in case of a severe accident with core melt).

(7): Systems not required meeting the acceptance criteria established for design basis accidents or design extension conditions but that are in the group of systems important to safety according to the IAEA Safety Glossary. A common set of requirements to be systematically applied cannot be established, but the relevant and specific requirements are generally defined on the basis of a functional analysis supplemented by probabilistic insights.

(8): Unless necessary for normal operation. Redundant divisions of a single safety system need to be independent and separated from each other to prevent a failure in one redundancy from propagating to the non-affected redundancies, or the loss of all of the redundancies caused by a hazard. Independence and separation of systems belonging to different levels of defence are also fundamental design elements for achieving a high level of safety, and are therefore expected to be implemented adequately between the different levels of defence. However, independence of levels of defence is a complex issue taking into account that full independence is not practically feasible, and is therefore not addressed in this TECDOC.

5.3.1.2.2 Requirements applicable to individual structures and components:

Generic consideration: Document IAEA TECDOC-1787 [95] explains that by assigning a safety class to every individual SSC, a set of design and manufacturing requirements needs to be established to meet the requested quality and reliability objectives. Adequate and proven codes or standards need to be used for the design and manufacturing of the structures and components to ensure that they will be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected according to their safety significance. These industry codes and standards indicate the methodologies, rules and criteria to be used for procurement, design, construction, inspection and testing of components.

Seismic requirements: Document IAEA TECDOC-1787 [95] explains that apart from the safety classification, it is also important to factor in the requirements related to classification of SSCs with respect to the importance of their integrity/performance/failure during a seismic event.

Environmental qualification: Document IAEA TECDOC-1787 [95] explains that qualification of equipment contributes to provide evidence that safety classified equipment is able to fulfil its required function(s) during accident conditions (design basis accidents or design extension conditions), despite the harsh environmental conditions (pressure, temperature, moisture, irradiation) prevailing prior to or at the time they are requested to operate. Specifications need to be defined taking into the following factors:

- The location of the item (environmental conditions are building dependent);
- The mission(s) of the item in accident conditions.

Pressure retaining equipment: Document IAEA TECDOC-1787 [95] gives as an examples of well-established codes defining design and manufacturing requirements for pressure retaining equipment the following:

- ASME Boiler and Pressure Vessel Code, Section III, Division 1
- French Association for Design, Construction and In-Service Inspection Rules for Nuclear Island Components (AFCEN) (RCC-M)
- Safety Standards of the German Nuclear Safety Standards Commission (KTA).

All the above standards are explained in more detail in Section 6.4.1 of D3.1 [1]. Table 7 shows relationship between safety classes SC2 and SC3 and code requirements for pressure retaining equipment.

Table 7: Relationship between safety class and code requirements for pressure retaining equipment (adapted per Table 18 of [95])

Safety Class	Safety classified pressure retaining equipment items	Code requirement	Example of SSCs
SC2	Components providing Cat. 3 functions with a safety barrier class 2	ASME Code, Section III, Division 1, Subsection NC RCC-M2	Residual heat removal system
	Components providing Cat. 2 functions	ASME Code, Section III, Division 1, Subsection ND RCC-M3	Spent fuel pool cooling system
SC3	Components providing Cat. 3 functions with a safety barrier class 3	ASME Code, Section III, Division 1, Subsection ND RCC-M3	Systems containing radioactive fluids in normal operation, e.g. chemical volume and control system, waste processing systems
	Components providing Cat. 3 functions unless specific codes and requirements are applied for specific reasons	Conventional codes like: • European Pressure Directive 97/23/EC • ASME Code, Section VIII, Division 1 for pressure vessels • ANSI B31.1 for piping	Systems providing make-up to feedwater tanks in postulated design extension conditions

For more details on safety classification refer to D3.1 [1], in which document IAEA TECDOC-1787 [95] is described or directly to IAEA TECDOC-1787 [95]. Namely, on the request of some Member States, the Commission on Safety Standards (CSS) requested the IAEA to consider developing a TECDOC-1787 to provide more technical detail in support of the methodology set out in IAEA SSG-30 [64].

5.3.2 Review of requirements for SSCs in Czech Republic

5.3.2.1 Categorization of safety functions

The concept of fundamental safety functions in Temelín NPP is in line with the Atomic Act of the Czech Republic 263/2016, the IAEA standard SSR-2/1 (Rev. 1) [20] and the WENRA report - Safety Reference Levels for Existing Reactors [8]. The fundamental safety functions (in Czech 'Základní Bezpečnostní Funkce' - ZBF) for nuclear facilities with a nuclear reactor are:

- reactivity control (ZBF1),
- heat removal from the reactor core and from the spent fuel outside the reactor (ZBF2),
- retention of radioactive isotopes, radiation shielding, control of routine releases of radioactive isotopes and reduction of radioactive releases in emergency situations (ZBF3).

The requirement for the definition of SSC functions, especially for safety functions, is based on the following regulations and documents:

- 329/2017; SÚJB on requirements for the design of a nuclear facility [29];
- 408/2016; SÚJB on requirements for the management system [31];
- IAEA documents SSR-2/1 rev. 1 [20], SSG-30 [64], SRS 46 [91], TECDOC-1791 [96];
- Safety philosophy for NPPs operated by ČEZ, a. s., Archive number EGP 5010-F-090427, 2009;
- SÚJB safety instructions (JB-1.0, 11/2011 [32], JB-1.7, 12/2010 [33]).

Safety functions can be divided into three categories, namely:

- Category I - the passive functions (properties) of the SSC of the primary circuit pressure boundaries;
- Category II - safety functions with the highest requirements for reliability;
- Category III - safety functions not included in categories I and II, which are substitutable for achieving the safety goals.

5.3.2.2 Classification of SSCs and requirements for SSCs

According to the "new legislation" (Act No. 263/2016 and its Implementing Decree No. 329/2017 [29]), the basic safety classification is updated as follows. Within the framework of ensuring the fulfilment of safety functions in accordance with their categorization, the design of a nuclear facility must divide SSCs into:

- SSCs with no impact nuclear safety,
- SSCs with an impact on nuclear safety, which are not selected equipment,
- selected equipment with the impact on nuclear safety, namely:
    - selected devices that are not safety systems,
        - systems related to nuclear safety, which are not safety systems,
        - various safety means and measures, which are not safety systems,
    - safety systems.

where selected equipment means a system, structure, component or other part of a nuclear installation affecting nuclear safety and the performance of safety functions.

Next, the SSC with the impact on nuclear safety are divided into three safety classes (in Czech 'bezpečnostní třída' - BT), i.e. Safety class 1 (BT1), Safety class 2 (BT2) and Safety class 3 (BT3) fulfilling the safety functions of Category I, Category II and Category III, respectively. For more details refer to Section 6.1.2.1 of D3.2 [2].

According to Section 6.1.2.2 of D3.2 [2] it seems that seismic classification is done according to the IAEA standards. Two seismic levels (SL) are considered:

- SL-1, operating basis earthquake (average peak acceleration in a given locality with a repeatability of 100 years),
- SL-2, maximum calculated earthquake/safe shutdown earthquake, (peak acceleration in a given locality with a repeatability of 10 000 years).

SSC are divided into the following four seismic categories Category 1a, Category 1b, Category 1c and Category NC (not classified).

According to Czech legislation, sCO2-4-NPP components would be classified as system fulfilling the **fundamental safety function (ZBF2),** i.e. the long-term heat removal from the containment vessel. It can be assigned to the **safety category III** and **seismic category 1a**.
For passive safety systems (such as sCO2 loop), the required standard is 2 × 100% redundancy. The 2 × 100% principle is based on a Russian technical design and is based on the IAEA recommendation (SAFETY SERIES No. 50-P-1 Application of the Single Failure Criterion [94]). Exceptions to the application of this rule must be justified by design and safety analyses. The passive safety systems rely on simple physical principles and do not require any external control or power supply to activate the safety function. Therefore, their reliability in terms of fulfilling the required safety function is significantly higher.

5.3.3 Review of requirements for SSCs in France

Section 6.2 of D3.2 [2] explains that in accordance with the "INB order" [41], SSCs classified as safety are part of the Important Elements for Protection (EIP). The safety classification of SSCs ensures that they are designed, manufactured and monitored in operation with a level of quality corresponding to their importance in demonstrating nuclear safety. The SSCs to be classified are mainly identified on the basis of the functions they perform. These functions are categorized according to their importance for safety.

The main steps to be followed to classify SSCs and assign requirements to them are:

- a functional analysis is used to categorize the functions according to the conditions under which they are required;
- the SSCs required to perform a categorized function are identified and classified accordingly;
- requirements are associated with the classified SSCs in order to guarantee that they will be able to satisfy their safety function with the required level of performance and reliability.

5.3.3.1 Categorization of safety functions

As explained in Section 6.2.1 of D3.2 [2], safety functions, including support functions, are categorized according to their importance, in three categories. The categorization of the safety functions is established in coherence with:

- the consequences of their failure for the demonstration of nuclear safety;
- the estimated frequency of solicitation;
- the time available to implement them as well as the duration during which they must be ensured, in particular for the achievement of a controlled or safe state.

The highest category is associated with a maximum level of requirements. The highest is Safety Category 1 (Cat1), followed by Safety Category 2 (Cat 2) and Safety Category 3 (Cat 3). Any function not categorized as Cat1, Cat2 or Cat3 is categorized as Category NC.

According to technical guidelines for next generation PWR [46] a possible way to define an appropriate classification is to assess the different reference transients, incidents and accidents, according to their estimated frequencies, with consideration of two physical states:

a) in the **controlled state**, the core is subcritical (short term recriticality before operator actions with only low neutron power could be accepted for a few events on a case by case basis), the heat removal is ensured in the short term e.g. by the steam generators, the core coolant inventory is stable, the activity releases remain tolerable;

b) in the **safe shutdown state**, the core is subcritical, the decay heat is removed durably, the activity releases remain tolerable.

For the multiple failures conditions, a final state can be defined: the core is subcritical, the decay heat is removed by primary or secondary systems, the activity releases remain tolerable.

Cat1 functions are related to controlled state, requiring short term heat removal for DBC2-4 conditions. Cat2 functions are related to safe shutdown state required before 24 hours for DBC2-4 conditions, while Cat3 functions are required for DBC2-4 after 24 hours, DEC A and DEC B to limit consequences etc.

5.3.3.2 Classification of SSCs and requirements for SSCs

A safety function is generally provided by a set of EIPs, including the EIPs performing the function and the associated support EIPs. Each EIP contributing to a safety function (Cat1, 2 or 3) is assigned a safety class reflecting its importance to safety.

Ranking rule for safety classification assigned to a EIP is defined by the highest safety category of the functions to which it contributes. EIP may also be ranked according to other criteria, such as the significance of the consequences of its failure, regardless of the functions they perform. The classification of the SSCs is therefore made up of three safety classes (Safety Classes S1, S2 and S3) and one unclassified safety class (NC). A system or component required to perform a Cat1, Cat2 and Cat3, is classified as Safety Class 1 (S1), Safety Class 2 (S2) and Safety Class 3 (S3), respectively. In addition, safety significance is also reflected in safety classes (refer to Section 6.2.3 of D3.2 [2]).

This safety class helps to define requirements for:

- the level of quality assurance;
- operating requirements (Periodic Tests, Technical Operating Specifications (TOS), maintenance...);
- depending on the type of equipment, the appropriate level of requirement in the design code or equipment specifications;
- the qualification proportional to the stakes, guaranteeing the ability of the EIP to carry out the functions assigned to it with respect to the solicitations and ambient conditions associated with the situations in which it is required.

Reliability of a function is primarily achieved by designing in accordance with a set of requirements. These requirements are derived from the design principles and design rules (e.g. application of the Single Failure Criterion (SFC), electrical back-up, sizing to external hazards, etc.). In order to simplify the analyses, the requirements can be defined in a decoupled manner. The decoupled design requirements associated with the Cat1, Cat2 and Cat3 safety functions required for the study of DBC2-4, DEC A, DEC B operating conditions and for the study of internal and external hazards are presented in the following table (Table 8):

Table 8. Decoupled design requirements associated with Cat1, Cat2 and Cat3 safety functions (Table 4 of [2])

Condition of operation	Category of the function	Application of the SFC	Electrical back-up	Sizing / Protection against external hazards
DBC	Cat 1 / Cat 2	Yes	Yes	Yes
DEC A	Cat 3	-	As per study	Yes
DEC B	Cat 3	-	As per study	Yes
Internal hazards	Cat 3	Yes	As per Hazard	
External hazards	Cat 3	Yes	As per Hazard	

In the case of French power plants, as for Czech power plants, the sCO2-4-NPP system is part of Category 3 (S3) for the classification of SSCs, as it provides a Category 3 safety function. This classification will allow to describe the requirements for the design, qualification and operation of the sCO2-4-NPP system in Section 5.3.4.

5.3.4 Review of sCO2-4-NPP system safety classification

As explained in Section 3 of D3.3 [3] the purpose of this subsection is to present different categories to which the sCO2-4-NPP system belongs, in order to establish the design conditions and regulations that will be linked to these categories.

Among the different possible classifications of the system, the most important one will be the classification related to safety. This classification has been already presented in Sections 5.3.2.2 and 5.3.3.2 for Czech Republic and France, respectively and will be discussed here in more detail.

5.3.4.1 Review and summary of information from D3.3

In this section information from Section 3.1 of D3.3 [3] is summarized after performed review. It considers different categories to which the sCO2-4-NPP system belongs, in order to establish the design conditions and regulations that will be linked to these categories. Among the different possible classifications of the system, the most important one will be the classification related to safety.

First in Section 3.1 of D3.3 [3] it is explained that the IAEA SSR-2/1 (Rev. 1) [20] specific safety requirements also concerns the requirements for residual heat removal[1] from the reactor core and heat transfer to the ultimate heat sink[2].

Then, it is explained that Section 3 of IAEA SSG-56 [69] entitled 'Design basis of the reactor coolant system and associated systems' is used as design basis, because residual heat removal systems are associated to reactor coolant system (RCS). In addition, Section 6 of IAEA SSG-56 [69] entitled 'Specific considerations in the design of associated systems for PWR technology' is used, because it further explains the recommendations for heat removal system. The heat removal systems are described separately for each operational state. For residual

[1] "The residual heat transfer chain includes the intermediate cooling systems and the cooling system directly associated with the ultimate heat sink. The intermediate cooling system is designed as a closed loop system that transfers heat from residual heat removal systems to the cooling system directly associated with the ultimate heat sink. The cooling system directly associated with the ultimate heat sink is an open loop system that takes water from the ultimate heat sink (pumping station), provides cooling to the intermediate cooling system and discharges transferred heat loads to the ultimate heat sink."

[2] "The ultimate heat sink is the medium into which residual heat is discharged in the different plant states after shutdown of the reactor, and it normally consists of a large body of water or the atmosphere, or both."

heat removal in hot shutdown modes for design extension conditions without significant fuel degradation, as a design provision, consideration should be given also to implementation of a secondary side passive heat removal system. By IAEA glossary definition [89] the residual heat is "*the sum of the heat originating from radioactive decay and shutdown fission and the heat stored in reactor related structures and in heat transport media*".

Further in Section 3.1 of D3.3 [3] it is explained that the system associated with the reactor coolant system is also a residual heat removal system, which could be also passive (the KTA 3301 [106] definition states that the residual heat removal systems of light-water reactors comprise those systems that transfer heat from the reactor coolant and containment vessel to a heat sink whenever the operation-related main heat sink is not in use anymore). Here it should be noted that in the case of sCO2-4-NPP system the passive residual system is associated to the secondary side, what according to IAEA terminology used in [69] it is a secondary side passive heat removal system (e.g. compact heat exchanger and turbo compressor system). The components may have a confinement barrier role in addition to its functional role [20]. The importance of the barrier role is reflected by assigning the SSC to a barrier safety class.

On the other hand, Section 4 of IAEA SSG-56 [69], entitled 'Ultimate heat sink and residual heat transfer systems' provides recommendations on meeting Requirement 53 of IAEA SSR-2/1 (Rev. 1) [20] with regard to the systems designed to transfer residual heat from the different decay heat removal systems to the ultimate heat sink. As the recommendations for the design of the above two systems are different, they are treated separately (especially recommendations for structures). Nevertheless, the design and manufacture of the heat transfer chains and associated systems and components (e.g. heat exchanger to diverse ultimate heat sink) should apply the design recommendations derived from the safety class of these structures, systems and components. This means that the same mechanical code (e.g. RCC-M [52] from AFCEN) can be used, if the components are of the same safety class. Finally, it should be noted that in accordance with IAEA SSG-56 [69], "*for an ultimate heat sink that relies on the atmosphere, cooling towers or spray ponds, with their associated structures and systems, are the usual equipment designed to transfer heat to the atmosphere*". In the case of sCO2-4-NPP, by analogy, this is Diverse Ultimate Heat Sink (DUHS), which is in accordance with IAEA SSG-56 [69] cooling system directly associated with the ultimate heat sink (open loop). Size and characteristics of the DUHS are crucial for the implementation and operation of the planned sCO2-4-NPP heat removal system in a nuclear power plant. However, only the design of the heat sink heat exchanger in within the scope of the project.

Additional requirements arise from ensuring the reliability of ultimate heat sink. The reliability of the heat transfer function can be ensured by a number of safety provisions, including high quality of SSCs, redundancy, diversity, physical separation, etc.

Finally, in Section 3.1 of D3.3 [3] it is explained that the code class is determined by the classification specified by the owner (or his designee) and is included in the design specification that establishes the rules for design and construction of items, including structures, systems and components (SSC). In Table 9 requirements for mechanical equipment are given, related to codes and standards like RCC-M [52] or ASME Boiler and Pressure Vessel Code (BPVC) [104], Section III or KTA standards [105] (hereafter term 'ASME code' is used for ASME BPVC).

Table 9. Selection of code or standard with requirements for pressure retaining equipment (Table 18 of [95])

Safety class	Equipment description	Code requirement
Class 2	Components providing Cat. 3 functions with a safety barrier class 2	RCC-M, Section I, Subsection C ASME code, Section III, Division 1, Subsection NC
Class 3	Components providing Cat. 3 functions with a safety barrier class 3	RCC-M, Section I, Subsection D ASME code, Section III, Division 1, Subsection ND
	Components providing Cat. 3 functions unless specific codes and requirements are applied for specific reasons	Conventional codes like: • European Pressure Directive 97/23/EC • ASME Code, Section VIII, Division 1 for pressure vessels • ANSI B31.1 for piping

At the end of Section 3.1 of D3.3 [3] it is explained that it should be noted that the ASME mechanical code requirements cover only safety of pressure integrity (these codes do not address other safety issues relating to the construction of nuclear components, or the in-service inspection of nuclear components; users of the code should refer to the pertinent codes, standards, laws, regulations, or other relevant documents for safety issues other than those relating to pressure integrity). Requirements for residual heat removal system are given also in KTA 3301 [106], which is applicable to design basis accidents (also should be noted that the KTA definition of “residual heat removal system (RHRS)” is broader than IAEA definition, where RHRS takes suction water from the RCS and injects water back into the RCS after being cooled by the heat exchanger). Appendix A of KTA 3301 [106] list possible systems within the scope of standard for PWR and BWR. Possible systems of light water reactors residual heat removal systems on secondary side are emergency feedwater system for feeding the steam generator and main steam safety and discharge control valve for the main steam blowdown from the steam generators.

5.3.4.2 Other classification in Czech Republic

Section 3.2 of D3.3 [3] explains that the specific requirements regarding the sCO2-4-NPP system will be based on the classification of the system parts. The general classification of SSC in a nuclear power plant according to Czech legislation was discussed in Section 5.3.2.2. The specific division of sCO2-4-NPP components is the following:

- The part connecting the system to the steam generator including the closing valves will be classified as **selected equipment with safety class 2 (BT2),**
- The piping driving the water to the Compact Heat Exchanger (CHX) and the CHX itself will be classified as **selected equipment with safety class 3 (BT3),**
- The remaining parts of the system, including the Turbo-Compressor System (TCS) and the Diverse Ultimate Heat Sink (DUHS) will be classified as **not selected equipment and with no safety class (BT0).**

The requirements for selected equipment are given in Decrees No. 358/2016 [30] and No. 329/2017 [29].

Also, the whole sCO2-4-NPP system will be classified as alternative (ALT) equipment, which means "*the system, structure, component or organizational measure to manage design extended conditions in situations where, due to a common-cause failure, a loss of the function of the safety system or the function of the diversion means specified in the nuclear installation design may occur when ensuring basic safety functions*". Usually, the alternative means are not selected, but the sCO2-4-NPP system is an innovative solution, different from those used previously in Czech NPPs. Therefore, the classification can differ from the well-known standards.

The Czech legislation also defines the pressure equipment, as "*selected equipment under stress from pressure exerted by a process medium with maximum operating pressure in excess of 0.05 MPa, including elements connected to parts exposed to pressure, safety and pressure equipment and other equipment that ensures its functionality*". Some specific requirements regarding pressure equipment are given in Annex 1 to Decree No. 358/2016 [30].

Regarding seismic classification, the whole sCO2-4-NPP system belongs to category 1a, which means that it must retain full functionality, including integrity during and after a seismic event, up to the level of the maximum design earthquake.

5.3.4.3 Other classifications in France

Section 3.3 of D3.3 [3] explains the other possible classifications in France. In the case of French power plants, operators and manufacturers can base themselves on two other complementary classifications to define the necessary regulations: a classification relating to the necessary mechanical quality and a classification of pressure equipment.

In addition to the initial design requirements detailed above in Section 5.3.3.2, designers of equipment for nuclear power plants must meet mechanical quality levels. These mechanical quality levels are three (Q1, Q2, and Q3) and an unclassified level (Qc) for pressure equipment and are assigned as follows:

- Equipment having a barrier role:
  - Quality level Q1 for:
    - Equipment forming the pressure boundary of the main primary circuit,
    - Equipment subject to the principle of exclusion of breakage.
  - Quality level Q2 for:
    - Pressure equipment carrying fluid in contact with the primary fluid and therefore integrity is required for a DBC3-4 or DEC-A/DEC-B event with possible fuel damage,
    - Equipment forming the pressurized envelope of the main secondary circuit,
    - Equipment forming a pregnant penetration.
  - Quality level Q3 for:
    - Equipment carrying activity and whose failure in normal operation would lead to radiological consequences greater than those of normal operation,
    - Equipment whose level of quality of mechanical realization Q is valorized in the studies of aggressions.
- Safety classified equipment without barrier role:
  - Equipment S1 and S2 must respect at least the quality level Q3;
  - Equipment S3 must comply with at least quality level Qc (reinforced quality level, non-nuclear).
- Equipment belonging to emergency feedwater system (ASG) (including downstream of pumps), safety injection system (RIS) and safety boration system (RBS) (including downstream of pumps): this equipment respects at least quality level Q2.

For the particular case of interfaces between two components, the principle is as follows:

- The interface takes the highest safety requirement of the two equipment,
- Where equipment is separated by redundant means, the same requirement applies to both means.

5.3.5 Assessment

Based on the review of Czech requirements in Section 5.3.2 it is judged that Czech safety function categorization is comparable to IAEA approach. Both IAEA and Czech Republic define three safety categories, but the meaning of categories is different. Further, it is judged that Czech safety function classification is comparable to IAEA approach. Both IAEA and Czech Republic define three safety classes for the three safety function categories. Regarding seismic classification it seems that seismic classification is done according to the IAEA standards. Finally, safety classes define requirements what is also the intent of IAEA requirements.

Based on the review of French requirements in Section 5.3.3 and comparison with IAEA requirements the following has been judged. As stated in Section 5.3.1.1 for Step 3 of classification process, IAEA uses three factors to categorize the identified safety functions into safety categories according to their safety significance. IAEA safety factor 1) concerns consequences of failure to perform the function, while French approach concerns consequences of their failure for the demonstration of nuclear safety. IAEA safety factor 2) concerns frequency occurrence of postulated initiating events, while French approach considers estimated frequency of solicitation. IAEA safety factor 3) concerns functions intended to reach a particular plant state. Generally, two plant states are distinguished by IAEA, namely a controlled state and a safe state. French safety function categorization uses controlled state and safe shutdown state what is similar to IAEA. It can be judged that intent of IAEA requirements is followed by French requirements for safety classification.

Regarding classification of SSCs it is judged that French classification is comparable to IAEA approach. Both define three safety classes for the three safety function categories, where the meaning of categories is different. Seismic classification requirements have not been included into D3.2 [2] for France. Nevertheless, the safety classes define requirements what is also the intent of IAEA requirements.

5.4 Requirements for plant modifications

5.4.1 IAEA approach for plant modification

The IAEA SF-1 principle 3 states that "*the process of safety assessment for facilities and activities is repeated in whole or in part as necessary later in the conduct of operations in order to take into account changed circumstances (such as the application of new standards or scientific and technological developments), the feedback of operating experience, modifications and the effects of ageing*."

The safety assessment process is performed in accordance with IAEA GSR Part 4 (Rev. 1) [17], which also requires that safety assessment is performed at different stages in the lifetime of a facility, including modification of the design (i.e. plant modification).

The requirement for programme to manage modifications is established in IAEA SSR-2/2 (Rev. 1) [21]: "*Requirement 11: Management of modifications*

The operating organization shall establish and implement a programme to manage modifications."

As IAEA NS-G-2.3 safety guide [76] for plant modifications is from 2001 (i.e. before above requirements have been established) and all IAEA safety standards before 2013 will be revised, draft safety standard IAEA DS497B [82], which has been developed and submitted to Commission on Safety Standards (CSS) for approval, has been used. IAEA DS497B [82] gives specific recommendations on controlling activities relating to modifications to nuclear power plants. In the following these recommendations are summarized with respect to sections of IAEA DS497B [82].

Section 2: MODIFICATION MANAGEMENT PROGRAMME

Plant modifications are required to be performed in accordance with a management system that is established and implemented by the operating organization in accordance with the requirements established in IAEA GSR Part 2 [15] (for further details refer also to Section 9.2.1.1 of D3.1 [1]). Plant modifications are required to be characterized on the basis of their safety significance.

Plant modifications that might affect safety should be divided into the following two categories:

a) Modifications directly relating to plant configuration,
b) Modifications to the operating organization.

Plant modifications directly in relation to plant configuration include modifications to structures, systems and components or process software (including the relevant documentation).

Section 3: ROLES AND RESPONSIBILITIES FOR PLANT MODIFICATIONS

Roles and responsibilities are described for the operating organisation and relation to contractors and other external organizations. The operating organization retains the prime responsibility for safety, including all safety implications of modifications (see Requirement 1 of SSR-2/2 (Rev. 1.) [21]).

Section 4: MODIFICATIONS RELATING TO PLANT CONFIGURATION

This section recommends that the proposed modifications should be categorized in accordance with their safety significance. Three categories are proposed (i.e. Category 1, Category 2 and Category 3). Modifications in Category 1 are capable of having a significant effect on safety, or involve an alteration of the principles and conclusions on which the design and the licensing of the plant were based. Modifications in Category 2 include changes in items important to safety, and in associated operational approaches and/or procedures, and usually necessitate an update of the safety analysis report or other licensing documents. Modifications in Category 3 are minor modifications.

In accordance with paragraphs 4.6 and 5.2 of IAEA GSR Part 4 (Rev. 1) [17], the safety assessment of a nuclear power plant is required to be updated as necessary so as to take into account the modifications to the design or operation of the plant.

Guidance is also given on review of proposed modifications, design considerations (in particular, the capability to fulfil the fundamental safety functions), modifications to operational limits and conditions, modifications to procedures and documentation), modifications to computer based systems, and configuration control.

Section 5 MODIFICATIONS TO THE OPERATING ORGANIZATION and Section 6 TEMPORARY MODIFICATIONS are of less relevance for this deliverable.

Section 7 IMPLEMENTATION OF MODIFICATIONS RELATING TO PLANT CONFIGURATION

This section provides recommendations for administrative control of modifications to plant configuration, specific safety considerations of modifications to plant configuration, testing and commissioning of modifications to plant configuration, and putting modifications to plant configuration into operation.

The last three sections are Section 8 IMPLEMENTATION OF ORGANIZATIONAL CHANGES, Section 9 TRAINING OF PERSONNEL and 10 MANAGEMENT OF DOCUMENTATION.

For further details on above information refer to IAEA DS497B [82].

5.4.2 Review of modification process in Czech Republic

Section 7.1 of D3.2 [2] provides information on the modification process in Czech Republic. No change made to a nuclear power plant shall affect its ability to be operated safely in accordance with the assumptions and goals of the project. The NPP operator is responsible for the modification management. The proposed changes must be divided into categories according to their impact on nuclear safety, radiation protection, emergency preparedness, technical safety and physical protection. SÚJB must be informed about this process. Changes affecting the NPP configuration and operating limits and conditions must follow the requirements and related regulations. In particular, the operability of the safety systems must be maintained. All related documents must be updated and the NPP staff must be informed about the modifications and trained appropriately [34].

5.4.2.1 Categorization of changes according to their significance

Section 7.1.1 of D3.2 [2] states that all proposed changes must be evaluated and categorized according to their influence on nuclear safety, physical barriers, radiation protection, emergency preparedness etc. These items must be properly identified, described and documented. The classification of the modifications is determined by the operator. In Attachment 1 of SÚJB JB-1.10 [34] three categories are proposed.

Category 1 modifications are those changes that have a direct impact on plant safety. These changes may affect the safety analysis results, technical measures to ensure compliance with safety requirements or directly affect the operating regimes.

Category 2 modifications affect the SSC important for plant safety. Such changes will result in changes to the Safety Report or other documentation submitted in the permit procedure. However, they have only limited impact on the basic principles of the project, and they don't affect the results of the safety analyses.

Category 3 changes have little or no effect on nuclear safety. They are performed either on SSC of safety class 3 or on other devices that are not directly listed in the license documentation. Their implementation will not affect the requirements for plant safety and even if they are incorrectly implemented, the risk will not increase. Category 3 includes replacement of the equipment with an approved equivalent.

5.4.2.2 Responsibilities

Section 7.1.2 of D3.2 [2] explains that the NPP operator (ČEZ) is responsible for the safety aspects of the changes and for obtaining the appropriate assessment and approval of regulatory body in accordance with the state regulations. The operator must establish the process to ensure the appropriate design, assessment, management and implementation of all permanent and temporary changes. The operator must ensure that appropriate safety analyses are performed with the relevant approach. The operator must ensure a systematic safety assessment which confirms on an ongoing basis that NPP safety analyses are still valid after the modifications.

SÚJB, which proceeds with the assessment and approval of the modifications to the NPPs in accordance with the Atomic Act and other regulations related to Atomic Act (SÚJB decrees).

The operator may outsource the performance of technical, analytical, design or production activities to another organization. Responsibility for meeting the requirements regarding the safety of the NPP is borne only by the operator. If the modification is made by supplier, a professional level must be demonstrated, the experience and qualifications of all the workers who are to perform the work.

5.4.2.3 Implementation of modifications

Section 7.1.3 of D3.2 [2] explains that the operator of the nuclear installation is responsible for the management processes of all changes. For major modifications it must include the establishment of objectives and organizational structure, determination of responsibilities, project leader, management and control procedures and allocation of adequate resources. For further details on administrative procedures, consideration of specific safety aspects, tests and start-up, operation, quality assurance and training refer to section 7.1.3 of D3.2 [2].

5.4.2.4 Safety assessment

Section 7.1.4 of D3.2 [2] explains that the initial safety assessment must be performed before the change is made, in order to determine what impact the proposed modification will have on safety and whether it is in accordance with the requirements and applicable legislation. This initial safety assessment must be carried out by sufficiently qualified staff with a systematic approach on the issue and reviewed by an independent safety expert. For Category 1 and 2 modifications, the safety assessment must be provided for independent review of SÚJB. Based on the results of the safety assessment, more detailed information can be required and a more comprehensive safety analysis, depending on the impact of the change on the plant safety. If the primary safety assessment clearly demonstrates that the implementation of the change will not affect safety, it is not necessary to provide additional supporting documentation.

Typical content of the safety assessment submitted to SÚJB is:

- description and justification of the modification,
- assessment of the impact of the modification on the nuclear safety,
- information on the updating of the documentation concerned,
- the estimated schedule for implementing the proposed modification,
- human factor impact assessment.

The scope, safety analysis and consequences of the proposed modification must be assessed by an independent person or institution, not directly involved in the design and implementation of the change.

The possible deviations from the basic characteristics of the project resulting from the modification should be minimized. If deviations are unavoidable, they must be assessed from the point of view of safety requirements and their acceptability must be demonstrated.

5.4.3 Review of modification process in France

Section 7.2 of D3.2 [2] states that the regulations relating to modifications to a nuclear power plant are described in Decision No. 2017-DC-0616 [48] of the Nuclear Safety Authority of November 30, 2017 on significant modifications to basic nuclear installations. This decision describes the regulations relating to different types of modifications.

5.4.3.1 Categorization of changes according to their significance

Section 7.2.1 of D3.2 [2] explains that the Decision No. 2017-DC-0616 [48] lists the different categories of modifications relating to a nuclear power plant in France. These categories are documentary modification, physical modification, significant modification, organizational modification, and substantial modification (for more detailed description refer to Section 7.1.2 of D3.2 [2]).

It is the responsibility of the operator to determine the category of the change and then to design the file accompanying the change.

Within the framework of the sCO2-4-NPP project, considering the function of the sCO2-4-NPP system, a French operator will have to constitute a significant modification file, because the installation of the sCO2-4-NPP system will lead to a documentational modification (new operating rules in case of accident), a physical modification (the system can be considered as an EIP) and an organizational modification. This modification will be subject to authorization by ASN and not to declaration to ASN.

5.4.3.2 Responsibilities

Section 7.2.2 of D3.2 [2] explains that the NPP operator is responsible for the safety aspects of the changes and for obtaining the appropriate assessment and approval of regulatory body in accordance with the state regulations. The operator will be responsible for compiling the modification request file with all the necessary documents.

The operator's requirements include several actions to be carried out like to determine whether or not any proposed modification is significant, give reasons for any significant modification envisaged, design the significant proposed modification and possible actions at the end of design, determine the possible tests to be carried out, analyse the compatibility with the regulatory requirements, analyse the impact of any significant changes on documentation, determine the possible provisions allowing the control of the implementation of a planned modification, formalize the operator's decision to implement any significant changes, prepare the documentary changes, implement the amended documents, monitor the completion of the significant change and its compliance with the defined requirements applicable to it, control the effective training of the persons, and draw and take into account the feedback from the implementation of the significant change.

Any significant change is subject to systematic verification of the defined requirements covering the achievement of the defined requirements for the management of significant changes. This verification is prior to any decision by the operator to implement the modification in question. It is carried out by an internal control body made up of persons with the appropriate skills to examine the modification in question.

The operator may outsource the performance of technical, analytical, design or production activities to another organization. Responsibility for meeting the requirements regarding the safety of the NPP is borne only by the operator. The operator must have his own staff with sufficient knowledge and experience to lead and supervise the activities carried out by hired organizations. If the modification is made by supplier, a professional level must be demonstrated, the experience and qualifications of all the workers who are to perform the work. Supplier staff must be adequately trained in accordance with pre-approved procedures in relation to the part of the NPP in which they are to operate. The supplier organization must have a proven quality assurance system and must be regularly audited by the NPP operator.

5.4.3.3 Implementation of modifications

Section 7.2.3 of D3.2 [2] explains that any modification (addition, modification or removal of at least one important element for protection (EIP) within the meaning of the order of February 7, 2012 (“INB Order”), or the addition, modification or removal of at least one element whose presence, operation or failure may affect the operation or integrity of an EIP) on a nuclear power plant must be the subject of a request for modification by the operator and be validated by the safety authorities in order to maintain the operating license.

The description of the modifications desired by the operator is part of the regulatory file. The impact study is an important part of the file and must be as exhaustive as possible in order to answer all possible questions from the ASN and other assessors of the file (IRSN). The elements of the file are also updated version of the safety report and updated risk management study. For details on these elements and few other elements to be included into the file refer to Section 7.2.3 of D3.2 [2].

The nature and the role of the sCO2-4-NPP heat recovery module with regard to nuclear safety makes it an important element for protection (EIP) and therefore its installation will have to be the subject of a complete file from the operator.

5.4.3.4 Safety assessment

Regarding safety assessment in Section 7.2.4 of D3.2 [2] information is given that IRSN is the assessors of the file. No details on safety assessment are provided in D3.2 [2]. The verification is prior to any decision by the operator to implement the modification in question. It is carried out by an internal control body made up of persons with the appropriate skills to examine the modification in question. The organization to ensure the independence of this verification from the persons directly responsible for the operation or modification is proportionate to the issues that the modification is likely to present for the protection of interests.

5.4.4 Assessment

Based on review in Section 5.4.2 presenting Czech requirements for plant modification, it is judged that Czech requirements comply with international practice. Czech three categories of modifications are comparable to IAEA. In addition, Attachment 3 of SÚJB JB-1.10 [34] shows Sections of SÚJB JB-1.10 [34] covering each item of WENRA Reference levels Issue Q: Plant modifications. Responsibilities are defined as recommended in Section 3 of IAEA DS497B [82]. Implementation of modifications is also required as recommended in Section 7 of IAEA DS497B [82]. Finally, safety assessment is required by Czech Republic. IAEA DS497B [82] requires safety assessment in accordance with paragraphs 4.6 and 5.2 of IAEA GSR Part 4 (Rev. 1) [17] (see also Figure 1 presenting IAEA safety assessment process). Also, Czech regulatory framework considers IAEA GSR Part 4 (Rev. 1) [17] (see Section 5.1.2).

Based on review in Section 5.4.3 presenting French requirements for plant modification, it is judged that French requirements comply with international practice. The categories of modifications are documentary modification, physical modification, significant modification, organizational modification, and substantial modification. By this the intent of IAEA to have different categories is fulfilled. Responsibilities are defined as recommended in Section 3 of IAEA DS497B [82]. Implementation of modifications is also required as recommended in Section 7 of IAEA DS497B [82]. IRSN is the assessors of the file, but no further information has been provided in Section 7.2 of D3.2 [2]. It should be noted that IRSN is European Technical Safety Organization Network (ETSON) member. ETSON developed its own safety assessment guide [100]. Approach proposed by IAEA GSR Part 4 (Rev. 1) [17] may also be used.

5.5 Requirements for design basis

The design and construction of the sCO2-4-NPP system must meet certain requirements and regulations. In Section 4 of D3.3 [3] the authors already discussed the processes and regulations for France and for Czech Republic, according to the IAEA SSG-56 [69] recommendations. The IAEA requirements for design in IAEA SSR -2/1 (Rev. 1) [20] and recommendations to satisfy design requirements in IAEA SSG-56 [69] have been already reviewed. For this reason, the information from D3.3 [3] was summarized and complemented, when needed. According to paragraph 3.7. of IAEA SSG-56 [69], the design basis for every SSC should specify the following:

(a) Function(s) to be performed by the structure, system or component;
(b) Postulated initiating events that the structure, system or component has to cope with;
(c) Loads and load combinations the structure or component is expected to withstand;
(d) Protection against the effects of internal hazards;
(e) Protection against the effects of external hazards;
(f) Design limits and acceptance criteria applicable to the design of structures, systems and components;
(g) Reliability;
(h) Provisions against common cause failures within a system and between systems belonging to different levels of defence in depth;
(i) Safety classification;
(j) Environmental conditions for qualification;
(k) Monitoring and control capabilities;
(l) Materials;
(m) Provisions for testing, inspection, maintenance and decommissioning.

The above design considerations are applicable to all water cooled reactors. In Section 6.3.2.2 of D3.1 [1] information is given, how IAEA SSG-56 [69] specific safety guide supports the IAEA design requirements in SSR-2/1 (Rev. 1) [20] (link is given to paragraphs of IAEA SSG-56 [69] to get further guide).

5.5.1 Functions to be performed by the system

5.5.1.1 IAEA requirements

IAEA SSR‑2/1 (Rev. 1) [20] standard for design requires fulfilment of fundamental safety functions to be ensured.

"*Requirement 4: Fundamental safety functions*

Fulfilment of the following fundamental safety functions for a nuclear power plant shall be ensured for all plant states: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel store; and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases."

Paragraph 3.8 of SSG-56 [69] provides recommendations on meeting Requirement 4 of SSR‑2/1 (Rev. 1) [20].

Paragraph 2.1 (d) of IAEA SSG-56 [69] recommendations also apply to the associated systems to RCS designed to fulfil the function "*to remove decay heat from the core and to transfer residual heat from the reactor coolant system to the ultimate heat sink in operational states and in accident conditions*". The sCO2-4-NPP system will also perform or contribute to the functions stated in Paragraphs 2.1 (a), 2.1 (c), 2.1 (f) and 2.1 (g) of IAEA SSG-56 [69]:

(a) To provide confinement of radioactive material for the protection of workers, the public and the environment;

(c) To maintain sufficient coolant inventory and cooling conditions to prevent significant fuel damage in design basis accidents and to mitigate the consequences of design extension conditions to the extent practicable;

(f) To limit overpressure of the reactor coolant system in operational states, design basis accidents and design extension conditions without significant fuel degradation;

(g) To shut down the reactor and to control the core reactivity to ensure compliance with fuel design limits in operational states and in accident conditions.

According to IAEA SRS No. 46 [91], there are three fundamental safety functions (FSFs):

- FSF(1) Controlling the reactivity,
- FSF(2) Cooling the fuel,
- FSF(3) Confining the radioactive material.

For example, during station blackout heat removal system will limit overpressure during design extension condition and by this also maintain reactor in safety shutdown condition (contributing to FSF(1)), maintain sufficient coolant inventory (contributing to FSF(2)) and contribute to confinement of radioactive material (contributing to FSF(3)).

5.5.1.2 sCO2-4-NPP system functions

In Section 4.1 of D3.3 [3] it is explained that the sCO2-4-NPP system will allow the heat generated in the reactor core to be removed under certain multiple failure operating conditions (DEC-A) by discharging the steam produced in the steam generators to the atmosphere. It is not intended to have a functional role in design basis conditions.

The sCO2-4-NPP system will participate in the following three fundamental safety functions:

- the control of nuclear chain reactions,
- the evacuation of thermal power from the reactor core,
- the prevention of radioactive releases.

Compliance with the latter makes it possible to ensure the fourth fundamental safety function, which is the protection of people and the environment against ionizing radiation.

5.5.1.3 Assessment

Based on the information in Section 5.5.1.2 it can be judged that the intent of IAEA SSR-2/1 (Rev. 1) [20] standard requirement to ensure fundamental safety functions is met.

5.5.2 Postulated initiating events that the system must cope with

5.5.2.1 IAEA requirements

IAEA SSR-2/1 (Rev. 1) [20] standard for design requires to identify a comprehensive set of postulated initiating events.

"*Requirement 16: Postulated initiating events*

The design for the nuclear power plant shall apply a systematic approach to identifying a comprehensive set of postulated initiating events such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design."

Paragraphs 3.10–3.12 of SSG-56 [69] provide recommendations on meeting Requirement 16 of SSR-2/1 (Rev. 1) [20].

Paragraph 3.10 of SSG-56 [69] recommends that "*from the list of postulated initiating events established for the design of the plant, those events that affect the design of the reactor coolant system and associated systems should be identified and categorized on the basis of their estimated frequency of occurrence.*"

Paragraph 3.11 of SSG-56 [69] recommends that "*for each of the conditions caused by the postulated initiating events, a list of the reactor coolant system and associated systems that are necessary to bring the plant to a safe and stable shutdown condition should be established*."

Paragraph 3.12 of SSG-56 [69] recommends that "*bounding conditions caused by the postulated initiating events should be determined in order to define the capabilities and performance of the reactor coolant system and associated systems and related equipment.*"

5.5.2.2 Review of postulated initiating events that sCO2-4-NPP system must cope with

In Section 4.2 of D3.3 [3] it is explained that the Horizon 2020 project sCO2-HeRo develops and proves the concept of a heat removal backup technology based on sCO2-4-NPP that can safely, reliably and efficiently remove residual heat from nuclear fuel without the need for external power sources making it an excellent backup cooling system for the reactor core in the case of a station blackout and loss of ultimate heat sink (postulated accident conditions, considered as design extension conditions - DEC). The concept consists of several modular sCO2-4-NPP systems, attached to the existing heat removal system, to remove decay heat from the reactor.

sCO2-4-NPP will provide a heat removal solution for Nuclear Power Plants that will increase the grace period in case of above mentioned accidents to beyond 72 hours, delaying the need for human intervention in case of an accident and thus decreasing the risk of human errors and the spread of radioactive material into the surrounding environment, ultimately reducing harm on both workers and citizens.

5.5.2.3 Assessment

Based on information provided in Section 5.5.2.2, the identified postulated initiating events are station blackout and loss of ultimate heat sink (postulated accident conditions, considered as design extension conditions - DEC). By this the intent of IAEA SSR‑2/1 (Rev. 1) [20] standard design requirement to identify a comprehensive set of postulated initiating events is judged to be met.

5.5.3 Loads and load combinations the system is expected to withstand

5.5.3.1 IAEA requirements

The highest IAEA GSR Part 4 (Rev. 1) [17] requirement regarding loads and load combinations is:

"*Requirement 10: Assessment of engineering aspects*

It shall be determined in the safety assessment whether a facility or activity uses, to the extent practicable, structures, systems and components of robust and proven design."

Engineering aspects according to IAEA include implementation of defence in depth, operating experience, radiation protection, safety classification of SSCs, aging and wear-out mechanism, protection against internal and external hazards, materials, equipment qualification.

More detailed requirements for assessment of internal events are given in paragraph 4.32 of IAEA GSR Part 4 (Rev. 1) [17]:

"*4.32. The internal events that could arise for a facility shall be addressed in the safety assessment, and it shall be demonstrated whether the structures, systems and components are able to perform their safety functions under the loads induced by normal operation and the anticipated operational occurrences and accident conditions that were taken into account explicitly in the design of the facility. Depending on the radiation risks* *associated with the facility or activity, this could include consideration of specific loads and load combinations, and environmental conditions (e.g. temperature, pressure, humidity and radiation levels) imposed on structures and components as a result of internal events, such as pipe breaks, impingement forces, internal flooding and spraying, internal missiles, load drop, internal explosions and fire.*"

IAEA SSR-2/1 (Rev. 1) [20] standard for design requires (see Requirement 17 below) that hazards shall be considered in the determining the generated loadings for use in the design of relevant items important to safety for the plant.

"*Requirement 17: Internal and external hazards*

All foreseeable internal hazards and external hazards, including the potential for human induced events directly or indirectly to affect the safety of the nuclear power plant, shall be identified and their effects shall be evaluated. Hazards shall be considered in designing the layout of the plant and in determining the postulated initiating events and generated loadings for use in the design of relevant items important to safety for the plant."

Paragraphs 3.76–3.86 of SSG-56 [69] provide recommendations on meeting Requirement 17 (and associated paragraphs 5.15A–5.21A) of SSR-2/1 (Rev. 1) [20] in relation to loads and load combinations and for brevity reasons the reader is referred to Section 4.3 of D3.3 [3], which summarizes recommendations in paragraphs 3.78-3.80, 3.82 and 3.86 or directly to IAEA SSG-56 [69].

5.5.3.2 Review of requirements for Czech plants

As explained in Section 4.3.2 of D3.3 [3], at the potential location of sCO2-4-NPP Compact Heat Exchanger (CHX) in Temelín NPP the design loads are 110 °C for temperature, steam-gas mixture for humidity and 0.12 MPa for pressure.

The CHX should be able to withstand the same design loads. Also, it must not worsen these conditions.

The design load of the climatic conditions assumes a frequency of occurrence once in 100 years (see Table 10). The extreme case of maximum design weather load considers a frequency of occurrence once in 10000 years. Buildings of the 1st seismic category must withstand the effects of extreme design conditions without posing risk to the functioning of systems relevant for nuclear safety. Other buildings must withstand the design level of weather conditions.

Table 10. Design loads from external hazards for Temelín NPP (Table 4 of [3])

Event (weather conditions)	**Design level (expected once in 100 years)**	**Extreme design load (expected once in 10 000 years)**
Extreme wind speed	49 m/s	68 m/s
Maximum temperature (peak value)	39.0 °C	45.6 °C
Minimum temperature (peak value)	-32.3 °C	-45.9 °C
Snow (equivalent water column)	92 mm	157 mm

In accordance with worldwide practice there are two design basis types of earthquakes for the NPP Temelín project:

- MDE (Maximum Design Earthquake, referred to also as SL-2 (seismic level) earthquake according to the IAEA Safety Standards Series No. NS-G-3.3 [78] and NS-G-1.6 [74]),
- DE (Design Earthquake, referred to also as SL-1 earthquake according to the IAEA Safety Standards Series No. NS-G-3.3 [78] and NS-G-1.6 [74]).

Table 11. Seismic levels for Temelín NPP (Table 5 of [3])

DBE	Level	Acceleration (PGA)	Duration	Comparable to $I_{state.}$
MDE	DBE – 2_{hor}	0.1 g	4-8 sec.	7° MSIS-64
	DBE – 2_{ver}	0.07 g	4-8 sec.	
DE	DBE – 1_{hor}	0.05 g	4-8 sec.	6° MSIS-64
	DBE – 1_{ver}	0.035 g	4-8 sec.	

PGA – maximum value of acceleration in horizontal and vertical direction at the level of free terrain (peak ground accelaration)

The frequency of occurrence of MDE is assumed to be once in 10000 years, while the frequency of DE once in 100 years.

Regardless of the magnitude of acceleration, which results from the assessment of the location, the design complies with recommendations of IAEA (NS-G-3.3 [78], section 2.6) for minimum value of acceleration in horizontal direction PGA_{hor} = 0.1 g.

5.5.3.3 Review of requirements for French plants

As explained in Section 4.3.3 of D3.3 [3], the French regulations concerning the loads and load combinations that the sCO2-4-NPP system must meet are presented in the RCC-M [52] for the mechanical equipment of the system. Class 2 Mechanical Equipment (heat recovery heat exchanger, turbo-machine) for type of loads, definition of operating conditions, loading rules, Levels of criteria, minimum criteria levels, requirements for stress report are in RCC-M, Section I, Subsection C (for details refer to Table 6 of D3.3 [3]). Class 3 Equipment design rules, pump design, piping design, valves design and overpressure protection are RCC-M, Section I, Subsection D (for details refer to Table 7 of D3.3 [3]).

5.5.3.4 Assessment

From Section 5.5.3.2 with Czech requirements, following IAEA NS-G-3.3 [78] and IAEA NS-G-1.6 [74], it should be noted that IAEA NS-G-3.3 [78] and IAEA NS-G-1.6 [74] standards have been superseded by IAEA SSG-9 (Rev. 1) [61] and IAEA SSG-67 [71] standards, respectively. IAEA SSG-9 [60] standard mention that it should also be recognized that when geological and seismological data have deficiencies, the value of 0.1 g will not represent a sufficiently conservative estimate of the hazard. IAEA SRS No. 66 [92] from 2011 complements IAEA NS-G-1.6 [74] and reference to definition of SL-1 and SL-2 in IAEA NS-G-1.6 [74]. IAEA SSG-67 [71] standard from 2021 states that the minimum level for seismic design (SL-2) should correspond to a peak ground acceleration of 0.1 g at the free field or foundation level and should be not less than the values established by the national seismic codes for conventional installations. The Czech value of SL-2 of 0.1 g follows the value of SL-2 as defined by IAEA.

From Section 5.5.3.3 with French requirements, it can be seen that French regulations concerning the loads and load combinations that the sCO2-4-NPP system must meet are presented in the RCC-M [52]. The latest IAEA SSG-9 (Rev. 1) [61] many times refer to codes and standards. Based on the information provided it is difficult to judge the extent the French regulations conform to IAEA standards.

However, IAEA TECDOC-1956 [98] in the context of instances in France for seismic alarm systems explains:

"*In case of exceedance of the level of earthquake causing an alarm to be generated by the triggers in the control room of the various units on the site (0.01 g), the reading of the data from the monitoring device and a first analysis of time-histories recorded will determine whether the peak acceleration corresponding to half of the amplitude of the design spectrum adapted to the site (SL-1 = 1/2 site specific SL-2) is exceeded at any of* *the measuring points. In case of exceeding this seismic level in any of the records, the operator needs to immediately reach the shutdown state considered as the safest for each unit. The restart of operation can be performed only after justification to the French regulator of the innocuousness of the earthquake for the future behaviour of the plant.*

Currently, in the framework of new periodic safety reviews of Électricité de France nuclear power plants since 2012, the SL-1 is re-defined as an 'inspection earthquake' corresponding to a fraction of SL-2 set to 0.05 g PGA in the horizontal direction, which is equal or lower than the previous SL-1 depending on the site."

5.5.4 Protection against the effects of internal hazards

5.5.4.1 IAEA requirements

The highest IAEA GSR Part 4 (Rev. 1) [17] requirement regarding internal hazards is:

"*Requirement 10: Assessment of engineering aspects*

It shall be determined in the safety assessment whether a facility or activity uses, to the extent practicable, structures, systems and components of robust and proven design."

More detailed requirements for assessment of internal events are given in paragraph 4.32 of IAEA GSR Part 4 (Rev. 1) [17], which requires that "*the internal events that could arise for a facility shall be addressed in the safety assessment.*"

IAEA SSR-2/1 (Rev. 1) [20] standard Requirement 17 (see Section 5.5.3.1 above) requires identification of internal hazards and evaluation of its effect:

"*Requirement 17: Internal and external hazards*

All foreseeable internal hazards and external hazards, including the potential for human induced events directly or indirectly to affect the safety of the nuclear power plant, shall be identified and their effects shall be evaluated."

More explicit requirements for design are given in paragraph 5.16 of IAEA SSR-2/1 (Rev. 1) [20], which is associated to Requirement 17:

"*5.16. The design shall take due account of internal hazards such as fire, explosion, flooding, missile generation, collapse of structures and falling objects, pipe whip, jet impact and release of fluid from failed systems or from other installations on the site. Appropriate features for prevention and mitigation shall be provided to ensure that safety is not compromised.*"

Paragraphs 3.14–3.17 of IAEA SSG-56 [69] provide recommendations on meeting Requirement 17 and paragraph 5.16 of SSR-2/1 (Rev. 1) [20] in relation to internal hazards and for brevity reasons the reader is referred to Section 4.4.1 of D3.3 [3], which summarizes recommendations in paragraphs 3.14-3.17 or directly to IAEA SSG-56 [69].

IAEA SSG-64 [70] may also be referred, as also gives recommendations on Requirement 17 of IAEA SSR-2/1 (Rev. 1) [20], especially Section 3 providing general design recommendations for protection against internal hazards in nuclear power plants. Other sections of IAEA SSG-64 [70] provides the following information. Section 2 outlines general considerations for protection against internal hazards in nuclear power plants. Section 4 provides specific recommendations for protection against fires, explosions, missiles, pipe breaks, flooding, collapses of structures and falling objects with a focus on heavy load drop, electromagnetic interference and release of hazardous substances originating within the site boundary. Appendix I provides guidance on dealing with hazard combinations. Appendix II provides detailed guidance on protection against internal fires.

5.5.4.2 Review of requirements for Czech plants

As explained in Section 4.4.2 of D3.3 [3], the requirements for the protection against the internal hazards base on IAEA DS494 (Safety guide on protection against internal hazards in the design of NPPs; note by authors of report: in 2021 IAEA DS494 was published as IAEA SSG-64 [70]).

These are all foreseeable effects that can directly or indirectly affect the safety of a nuclear power plant. These effects must be clearly identified, and their effects should be evaluated. Appropriate initiating events and induced loads must be determined for the design and protection of SSC important for the safety of the power plant.

5.5.4.3 Review of requirements for French plants

As explained in Section 4.5.3 of D3.3 [3], the sCO2-4-NPP system will need to be protected against internal attacks that could induce a DEC event for which the system would be required. If the system is placed inside each building of the back-up auxiliaries, the necessary trains can be duplicated according to the redundancy principle. Each train can be located in a specific bunker in a backup building. This will contribute to the protection against internal aggressions.

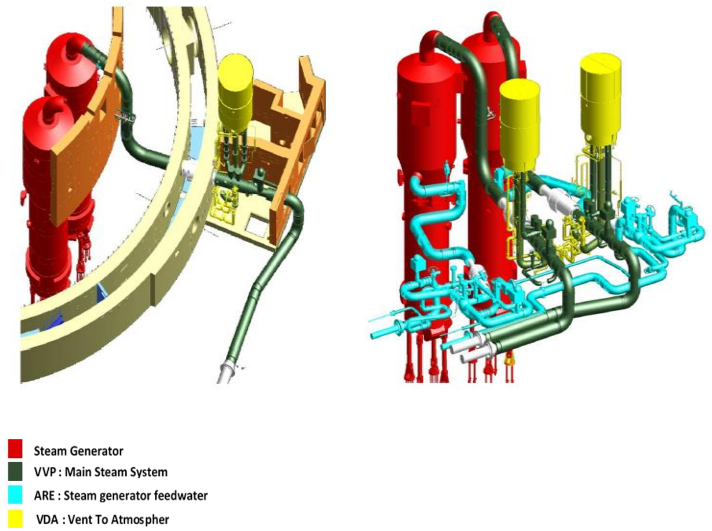

Figure 9. Secondary Loop (Figure 2 of [3])

In addition, it should be noted that:

- the main piping of the VVP (Vapeur Vive Principale - Main Steam System, see Figure 9) system of a power plant is considered high energy, however the concept of rupture exclusion applies here. If the sCO2-4-NPP system is placed on this piping, the rupture of the main piping of the VVP system will therefore not be considered as a hazard of the sCO2-4-NPP system.
- On the other hand, the rupture of the other pipes considered as high energy in the casemate is to be assumed (case of the main steam isolation valve (VIV) bypass line or the Vent To Atmosphere (VDA) system. The complete analysis will have to be carried out during the High Energy Piping Rupture (HEPR) studies.
- The sCO2-4-NPP system is a system made up of:

- components of quality Q1 or Q3: consequently, no missile is postulated for the equipment but could be investigated for the piping;
- high energy components (piping): the consequences of their rupture will be analyzed within the framework of the HEPR studies.

5.5.4.4 Assessment

From Section 5.5.4.2 with Czech requirements is can be seen that the requirements follow IAEA DS494 draft standard, which was later published as IAEA SSG-64 [70] in 2021. By this conformance to IAEA may be judged.

From Section 5.5.4.3 with French requirements the focus is on high energy break. With the information provided for other internal hazards it is difficult to judge conformance to IAEA safety standards. Nevertheless, it should be noted that France was reviewer of IAEA DS494 [80] (later published as IAEA SSG-64 [70]).

5.5.5 Protection against the effects of external hazards

5.5.5.1 IAEA requirements

The highest IAEA GSR Part 4 (Rev. 1) [17] requirement regarding external hazards is Requirement 10 (see Section 5.5.4.1). More detailed requirements for assessment of external events are given in paragraph 4.31 of IAEA GSR Part 4 (Rev. 1) [17], which requires:

"*4.31 The external events that could arise for a facility or activity shall be addressed in the safety assessment, and it shall be determined whether an adequate level of protection against their consequences is provided. This could include natural external events, such as extreme weather conditions, and human induced events, such as aircraft crashes, depending on the possible radiation risks associated with the facility or activity. Where applicable, the magnitude of the external events that the facility is required to be able to withstand (sometimes referred to as design basis external events) shall be established for each type of external event on the basis of historical data for the site for natural external events and a survey of the site and the surrounding area for human induced events. Where appropriate, the safety assessment shall demonstrate that the design is adequately conservative so that margins are available to withstand external events more severe than those selected for the design basis.*"

IAEA SSR‑2/1 (Rev. 1) [20] standard Requirement 17 (see also Section 5.5.3.1 above) requires that *"All foreseeable internal hazards and external hazards, including the potential for human induced events directly or indirectly to affect the safety of the nuclear power plant, shall be identified and their effects shall be evaluated.*" More explicit requirements for design are given in paragraphs 5.17-5.21A of IAEA SSR‑2/1 (Rev. 1) [20], which are associated to Requirement 17. For example:

"*5.17. The design shall include due consideration of those natural and human induced external events (i.e. events of origin external to the plant) that have been identified in the site evaluation process. Causation and likelihood shall be considered in postulating potential hazards. In the short term, the safety of the plant shall not be permitted to be dependent on the availability of off-site services such as electricity supply and firefighting services. The design shall take due account of site specific conditions to determine the maximum delay time by which off-site services need to be available.*"

Paragraphs 3.19–3.26 of SSG-56 [69] provide recommendations in relation to external hazards on meeting Requirement 17 and paragraphs 5.17–5.21A of SSR-2/1 (Rev. 1) [20] and for brevity reasons the reader is referred to Section 4.5.1 of D3.3 [3], which summarizes recommendations in paragraphs 3.19–3.26 or directly to IAEA SSG-56 [69].

IAEA SSG-68 [72] may be referred too, as it also gives recommendations on Requirement 17 of IAEA SSR-2/1 (Rev. 1) [20]. Throughout IAEA SSG-68 [72] standard, the term 'external events' always excludes earthquakes, for which recommendations are provided in IAEA SSG-67 [71].

IAEA SSG-68 [72] in Section 2 presents requirements for the general concepts and application of safety criteria to the design of nuclear installations for protection against external events, including the relevant safety requirements; SSCs to be protected against external events; and recommendations on design and evaluation for design basis external events and beyond design basis external events and for the determination of adequate margins.

5.5.5.2 Review of requirements for Czech plants

As explained in Section 4.5.2 of D3.3 [3], the requirements regarding the protection against external hazards base on the IAEA DS498: External Events Excluding Earthquakes in the Design of Nuclear Installations [84] (note by authors: in 2021 the DS498 draft standard was published as IAEA SSG-68 [72]).

External influences of natural origin include, for example, geological influences (e.g. seismicity), meteorological influences (wind, snow, temperature extremes, heavy rains) and hydrological influences. The combination and intensity of the load caused by external natural influences on the SSC must be determined according to the importance of the SSC from the point of view of ensuring nuclear safety.

External influences caused by human activity include, for example, the fall of aircraft and other objects, explosions and fires that originate in human activity, negative effects of road, rail and water transport, the effects of pipelines and power lines, operation of equipment containing substances, which are flammable, explosive, toxic, asphyxiating, corrosive or radioactive.

The design must take into account the following principles:

- SSC must be located and designed in such a way as to minimize the effects of external influences,
- mutual interactions between individual SSC due to external influences must be minimized,
- SSC must survive the load caused by external influences or appropriate protective measures must be included, e.g. in the form of passive barriers,
- in the case of several units, the design must consider a situation when several blocks are affected by external influences at the same time.

5.5.5.3 Review of requirements for French plants

As explained in Section 4.5.3 of D3.3 [3], sCO2-4-NPP as a system participating in the mitigation of DEC operating conditions will need to be sized to remain operational in the event of a DBH (Design Basis external Hazards) reference external attack. Furthermore, by convention, the system shall also be sized to be operable during and after the reference earthquake of the power plant where it will be installed (see Section 5.2.3.4 of D3.2 [2]).

Concerning the other hazards:

- If the system is installed in the buildings of the back-up auxiliaries, it will be protected from external explosion, the effects of snow and wind, projectiles associated with the tornado and lightning. The discharge piping to the system atmosphere (outside the building) is protected by suitable guards to ensure the discharge function.
- If the system is installed elsewhere, its building will have to be designed according to the same rules as the buildings of the back-up systems.

- In the case of outside temperatures of DBH level, the system valves must be operable. The conditioning requirement for DBH temperature can be analysed at a later stage of the design (depending on the plant).

Requirements for the building could be find in the RCC-CW [54].

5.5.5.4 Assessment

From Section 5.5.5.2 with Czech requirements it can be seen that the requirements follow IAEA DS498 draft standard [84], which was published as IAEA SSG-68 [72] in 2021. By this conformance to IAEA may be judged, except that IAEA SSG-67 [71] is recommended to be followed for earthquakes.

From Section 5.5.5.3 with French requirements it can be seen that several hazards need to be considered. Specific requirements are given in RCC-CW [54]. With the information provided it is difficult to judge conformance to IAEA safety standards. Nevertheless, it should be noted that France was reviewer of IAEA DS490 [79] (later released as IAEA SSG-67 [71]) and IAEA DS498 draft standard (later released as IAEA SSG-68 [72]).

5.5.6 Design limits and acceptance criteria applicable to the design of SSC

5.5.6.1 IAEA requirements

IAEA SSR-2/1 (Rev. 1) [20] standard gives requirements on design limits in the Requirement 15:

"*Requirement 15: Design limits*

A set of design limits consistent with the key physical parameters for each item important to safety for the nuclear power plant shall be specified for all operational states and for accident conditions."

There is one associated paragraph 5.4 regarding consistency of design limit:

"5.4. The design limits shall be specified and shall be consistent with relevant national and international standards and codes, as well as with relevant regulatory requirements."

The acceptance criteria are mentioned in the Requirement 14 IAEA SSR-2/1 (Rev. 1) [20]:

"*Requirement 14: Design basis for items important to safety*

The design basis for items important to safety shall specify the necessary capability, reliability and functionality for the relevant operational states, for accident conditions and for conditions arising from internal and external hazards, to meet the specific acceptance criteria."

Paragraphs 3.44-3.45 of SSG-56 [69] provides recommendations on meeting Requirement 15 of SSR-2/1 (Rev. 1) [20]: associated systems should be designed so that the relevant design limits and criteria for fuel are not exceeded and should be designed so as not to cause unacceptable stresses on the reactor coolant pressure boundary.

5.5.6.2 Review of requirements for Czech plants

As explained in Section 4.6.2 of D3.3 [3], the design limit is defined paragraph 3 of Decree 329/2017 [29]:

"*k) 'design limit' means the acceptance criterion used to assess the capability of a nuclear installation or its structure, system or component to perform its function as intended in the nuclear installation design; design limit is, in particular, a limit set out by legislation or an acceptance criterion derived therefrom, which* *corresponds to the method of assessment of the capability of the nuclear installation to perform its function as intended in the nuclear installation design,*"

The design basis shall determine the acceptance criteria relevant to the categories of the anticipated states of the nuclear installation and the consequences of these states (see Table 1 of D3.2 [2]). Nuclear installation design shall determine the safety limits and acceptance criteria for parameters characterising the state of the nuclear installation.

The design limits for the sCO2-4-NPP system will be derived from the final detailed design of the system. After the safety analysis proves that the equipment fulfils its safety functions and meets the acceptance criteria in all plant states, all the parameters for corresponding SSC will be defined.

5.5.6.3 Review of requirements for French plants

As explained in Section 4.6.3 of D3.3 [3], acceptance criteria are defined for the equipment of each of the French power plants according to the operating parameters. Within the framework of the sCO2-4-NPP project, it seems premature to define acceptance criteria for the sCO2-4-NPP system, as its operating parameters are not yet defined precisely enough.

Nevertheless, this step will have to be integrated in the roadmap for the development of the system by the system designers.

5.5.6.4 Assessment

Czech Republic and France stated that design limits will be derived from the final detailed design of the system and that it seems premature to define acceptance criteria for the sCO2-4-NPP system, respectively. IAEA SSR‑2/1 (Rev. 1) [20] requires that a set of design limits for the nuclear power plant shall be specified for all operational states and for accident conditions. As stated in Section 5.5.6.3, this step will have to be integrated in the roadmap for the development of the system by the system designers.

5.5.7 Reliability

5.5.7.1 IAEA requirements

IAEA SSR‑2/1 (Rev. 1) [20] standard gives requirements on reliability in the Requirement 23:

"*Requirement 23: Reliability of items important to safety*

The reliability of items important to safety shall be commensurate with their safety significance."

The associated paragraphs 5.37 and 5.38 to Requirement 23 of IAEA SSR‑2/1 (Rev. 1) [20], providing more detailed requirements for design, are:

"*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.*

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."

Paragraphs 3.47–3.56 of SSG-56 [69] provide recommendations on meeting Requirements 21–26, 29 and 30 of SSR-2/1 (Rev. 1) [20] and for brevity reasons the reader is referred to Section 4.7.1 of D3.3 [3], which summarizes the safety factors to be considered to achieve the necessary reliability or directly to IAEA SSG-56 [69]. Paragraph 3.47 of SSG-56 [69] is the most relevant for Requirement 23 of SSR-2/1 (Rev. 1) [20] (also presented in Section 4.7.1 of D3.3 [3]):

"*3.47. To achieve the necessary reliability of the reactor coolant system and associated systems to control the reactivity of the core, to maintain sufficient inventory in the reactor coolant system, to remove residual heat from the core and to transfer residual heat to the ultimate heat sink, the following factors should be considered:*

(a) Safety classification and the associated engineering requirements for design and manufacturing;
(b) Design criteria relevant for the systems (e.g. number of redundant trains, seismic qualification, qualification to harsh environmental conditions, and power supplies);
(c) Prevention of common cause failures by the implementation of suitable measures such as diversity, physical separation and functional independence;
(d) Layout provisions to protect the reactor coolant system and associated systems against the effects of internal and external hazards;
(e) Periodic testing and inspection;
(f) Ageing effects;
(g) Maintenance;
(h) *Use of equipment designed for fail-safe behaviour.*"

5.5.7.2 Review of requirements for Czech plants

As explained in Section 4.7.2 of D3.3 [3], alternative equipment must have sufficient reliability corresponding to the required actuation time and the period of time for which they must be able to perform their function. They must be maintained in such a way that they are available and functional at the required timing.

The reliability and efficiency of systems at the various levels of DiD must be ensured in such a way as to:

- minimize the likelihood of deviations from normal operation,
- provide the highest reliability of the plant's control and safety systems,
- ensure the efficiency of DBA and DEC management systems and procedures to minimize the likelihood of core damage and radioactive releases.

The reliability of systems, structures and components relevant to nuclear safety shall be ensured through:

- a system ensuring their environment qualification,
- the method of ensuring resilience of systems to failures and
- the method of maintaining and testing them.

5.5.7.3 Review of requirements for French plants

As explained in Section 4.7.3 of D3.3 [3], ASN Guide No. 22 [43] specifies the following expectations for the reliability of equipment important to safety:

"*EIPs and IP systems shall be designed to ensure that the safety functions they perform are provided with appropriate reliability, taking into account their role for nuclear safety. This reliability shall be achieved by an appropriate combination:*

- *design, construction, installation, monitoring and maintenance arrangements;*
- *redundancy, separation, and diversification between EIPs, in particular in order to reduce the probability of common cause failures.*

In the design of EIPs, consideration should be given to:

- *aging and wear mechanisms (possibly related to the maintenance program);*
- *uncertainties about the physical parameters of the installation;*
- *operating experience feedback.*

Insofar as this does not introduce excessive complexity and where a single state favorable to nuclear safety is identified, the IP systems must be designed in such a way as to switch automatically to this state (principle of directed failure) when some of their components fail (including due to the failure of a possible system performing a support function)".

In our case, this means that the sCO2-4-NPP system will have to be dimensioned to be able to accomplish its mission in the event of failure of one of the modules (one or two redundant modules may therefore have to be provided), that the necessary measures must be taken so that a possible failure of the system does not harm the rest of the power plant (control of valves, prevention of CO_2 leakage, etc.), and that the I&C system must be designed in such a way that it automatically switches to this state (principle of oriented failure) when some of its components fail (including due to the failure of a possible system performing a support function).

5.5.7.4 Assessment

From Section 5.5.7.2 with Czech requirements the reliability of SSCs relevant to nuclear safety shall be ensured through environment qualification, resilience of systems to failures and maintaining and testing them. Czech requirements cover some factors to be considered to achieve necessary reliability, recommended by paragraph 3.47 of IAEA SSG-56 [69]. The information in Section 4.7.2 of D3.3 [3] seems to be very limited and it is recommended to verify if Czech legislation considers also other factors, recommended by paragraph 3.47 of IAEA SSG-56 [69].

From Section 5.5.7.3 with French requirements the reliability shall be achieved by an appropriate combination of design, construction, installation, monitoring and maintenance arrangements; and redundancy, separation, and diversification between EIPs, in particular in order to reduce the probability of common cause failures. French requirements cover several factors to be considered to achieve necessary reliability, recommended by paragraph 3.47 of IAEA SSG-56 [69]. It is suggested to verify French legislation regarding a few remaining factors, recommended by paragraph 3.47 of IAEA SSG-56 [69].

5.5.8 Provisions against common cause failures within a system and between systems belonging to different levels of defence in depth

5.5.8.1 IAEA requirements

IAEA SSR-2/1 (Rev. 1) [20] standard gives requirements on common cause failures in the Requirement 24:

"*Requirement 24: Common cause failures*

The design of equipment shall take due account of the potential for common cause failures of items important to safety, to determine how the concepts of diversity, redundancy, physical separation and functional independence have to be applied to achieve the necessary reliability."

As stated in Section 4.8 of D3.3 [3], this requirement is applicable to system as a whole (e.g. several modules of sCO2-4-NPP) and not to the components of the sCO2-4-NPP. D3.3 [3] also states that no specific guide is given in IAEA SSG-56 [69]. Nevertheless, common cause failures are discussed at paragraphs related to internal hazards, design basis accidents, reliability, defence in depth, and systems for core cooling in accident conditions. The most relevant recommendation seems to be paragraph 7.24 of IAEA SSG-56 [69], related to core cooling in design extension conditions:

"*7.24. The capability to cool the core adequately in the event of design extension conditions should be focused on ensuring that such conditions do not result in core melting. As such, the primary focus should be on ensuring that the most probable common cause failure sequences identified for consideration as part of the design extension conditions without significant fuel degradation can be successfully mitigated utilizing on-site equipment.*"

5.5.8.2 Review of requirements for Czech Republic and France

As explained in Section 4.8 of D3.3 [3], it was indicated in Section 5.2.4 of D3.2 [2] report that the sCO2-4-NPP system will have to be integrated into the defence-in-depth strategy of the plants in which it will be installed. The system will have to be considered as a whole and thus can be added in the safety demonstration.

Also, as explained in Section 4.8 of D3.3 [3] Requirement 24 of IAEA SSR-2/1, Rev. 1 [1] was identified. This requirement is applicable to system as a whole (e.g. several modules of sCO2-4-NPP) and not to the components of the sCO2-4-NPP. It was also stated that "No specific guide is given in IAEA SSG-56 [69]".

5.5.8.3 Assessment

Based on the review of requirements provided by Czech Republic and France, it is deemed, that by identifying Requirement 24 of IAEA SSR-2/1, Rev. 1 [1] the international practice is followed. It is recommended to follow also paragraph 7.24 of IAEA SSG-56 [69] recommendation regarding ensuring that the most probable common cause failure sequences identified for consideration as design extension conditions without significant fuel degradation can be successfully mitigated utilizing on-site equipment.

5.5.9 Safety classification

5.5.9.1 IAEA requirements

Requirement 10 of IAEA GSR Part 4 (Rev. 1) [17] (see Section 5.5.3.1) is related to engineering aspects, which include also safety classification. More specific requirement regarding safety classification is given in paragraph 4.30 (see Section 5.3.1.1), which is associated to Requirement 10 of IAEA GSR Part 4 (Rev. 1) [17].

IAEA SSR-2/1 (Rev. 1) [20] standard gives requirements on safety classification in the Requirement 22 (with associated paragraphs 5.34-5.36):

"*Requirement 22: Safety classification*

All items important to safety shall be identified and shall be classified on the basis of their function and their safety significance."

Paragraph 5.34 of IAEA SSR-2/1 (Rev. 1) [20] on method for classifying the safety significance of items important to safety states:

"*5.34. The method for classifying the safety significance of items important to safety shall be based primarily on deterministic methods complemented, where appropriate, by probabilistic methods, with due account taken of factors such as:*

(a) The safety function(s) to be performed by the item;
(b) The consequences of failure to perform a safety function;
(c) The frequency with which the item will be called upon to perform a safety function;
(d) *The time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function.*"

Paragraph 5.35 of IAEA SSR‑2/1 (Rev. 1) [20] on interfaces between items important to safety states:

"*5.35. The design shall be such as to ensure that any interference between items important to safety will be prevented, and in particular that any failure of items important to safety in a system in a lower safety class will not propagate to a system in a higher safety class.*"

Paragraph 5.36 of IAEA SSR‑2/1 (Rev. 1) [20] on equipment that performs multiple functions states:

"*5.36. Equipment that performs multiple functions shall be classified in a safety class that is consistent with the most important function performed by the equipment.*"

Paragraphs 3.63–3.66 of SSG-56 [69] provide recommendations on meeting Requirement 22 of SSR-2/1 (Rev. 1) [20] and for brevity reasons the reader is referred to IAEA SSG-56 [69]. According to paragraph 3.62 of IAEA SSG-56 [69] the recommendations for safety classification provided in IAEA SSG-30 [64] should also be considered (see Section 5.3.1.1).

5.5.9.2 Review of requirements for Czech plants

As explained in Section 4.9.2 of D3.3 [3], according to Czech legislation (decree 329/2017) [29], "*nuclear installation design shall ensure automatic activation and control of safety systems or implementation of a safety function using passive function systems, structures or components so that intervention by operators is not necessary until 30 minutes after the initiating event has occurred.*" The categorization of safety functions and safety classification have already been discussed in Sections 5.3.2.1 and 5.3.2.2, respectively. Safety functions are divided into Categories I, II and III.

Selected equipment performing Category II safety functions, which are safety functions with the highest reliability requirements, shall be classified as safety class 2. These SSCs perform the functions:

- for the removal of heat from the core and to limit damage to nuclear fuel in the event of a design basis accident involving the failure of the primary circuit boundary,
- necessary for the removal of residual heat from the core during operational states and in the event of a design basis accident not involving a failure of integrity of the primary circuit boundary.

5.5.9.3 Review of requirements for French plants

As explained in Section 4.9.3 of D3.3 [3], the classification rule applicable for France is as follows:

Are said to be "safety classified" the mechanical equipment, electrical systems, structures and civil engineering works which are necessary to achieve, under operating conditions considered plausible, the objectives listed below, with exceptions to be justified in the case of additional operating conditions:

- maintenance of the integrity of the pressure boundary of the main primary circuit;
- ability to shut down the reactor and maintain it in a safe shutdown state;
- ability to prevent accidents or limit their radiological consequences.

Due to its function, the sCO2-4-NPP system will have to be classified as safety equipment.

In Section 5.3.3.2 the classification of SSCs into three safety classes is presented. Here, it is described more specifically for mechanical equipment carrying a pressurized fluid to which safety class is belongs. Of interest for sCO2-4-NPP system are Safety class 2 and Safety class 3.

To Safety class 2 belong the materials of the systems or parts of systems:

- which, in the event of an accident, directly provide core cooling and heat extraction in the containment;
- or which indirectly provide these cooling functions if they are not accessible under accident conditions.

To Safety class 3 belong pressure vessels, which, according to the rules defined above, do not belong to safety classes 1 or 2.

In the case of sCO2-4-NPP system, we can define that the mechanical components of the sCO2-4-NPP system belong to Safety class 2 by virtue of their role in extracting heat in accidental situations. In report D3.2 [2], a first classification in category S3 it was established, but given the presence of mechanical equipment, a classification in category S2 seems more appropriate. The final classification can only be determined once the system start-up and operation procedures are more advanced. Indeed, depending on how the system is coupled to the power plant, and the procedures induced for its start-up, it will be easier to determine whether a class 2 or 3 is more appropriate for some of the components. Because of this difficulty, it was recommended to continue the development of the components by applying the most conservative approach: a class 2 for the most critical components (secondary loop/sCO2-4-NPP exchangers and the turbocompressor). As the piping is classified as high energy pipe break, the associated regulations will have to be applied to them.

5.5.9.4 Assessment

Based on the review in Sections 5.5.9.2 and 5.5.9.3 for Czech Republic and France, respectively, the safety classification is primarily done by the safety function(s) to be performed by the item. Each country defines three safety function categories and three corresponding safety classes. The safety classes do not depend only on the safety function category. The intent of Requirement 22 of IAEA SSR-2/1 (Rev. 1) [20] to identify items important to safety and to classify them on the basis of their function and their safety significance seem to be fulfilled.

5.5.10 Environmental qualification

5.5.10.1 IAEA requirements

According to IAEA Glossary [89], the term ‘Equipment qualification’ refers to the "*generation and maintenance of evidence to ensure that equipment will operate on demand, under specified service conditions, to meet system performance requirements.*" For information, IAEA Glossary [89] explains that "*More specific terms are used for particular equipment or particular conditions; for example, seismic qualification is a form of equipment qualification that relates to conditions that could be encountered in the event of earthquakes.*" By analogy, it can be assumed that 'environmental qualification' is a form of equipment qualification that relates to the 'environmental conditions'. As such, it is one of the forms of equipment qualification.

The highest IAEA GSR Part 4 (Rev. 1) [17] requirement regarding equipment qualification is Requirement 10 (see Section 5.5.3.1) on engineering aspects, including environmental qualification.

IAEA SSR-2/1 (Rev. 1) [20] standard gives requirements on qualification of items important to safety in the Requirement 30 (with associated paragraphs 5.48-5.50):

"*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."

Paragraph 5.48 of IAEA SSR-2/1 (Rev. 1) [20] on environmental conditions considered in the qualification programme for items important to safety states:

"*5.48. The environmental conditions considered in the qualification programme for items important to safety at a nuclear power plant shall include the variations in ambient environmental conditions that are anticipated in the design basis for the plant.*"

Paragraph 5.49 of IAEA SSR‑2/1 (Rev. 1) [20] on aging effects by environmental factors and replicating the conditions by the natural external event states:

"*5.49. The qualification programme for items important to safety shall include the consideration of ageing effects caused by environmental factors (such as conditions of vibration, irradiation, humidity or temperature) over the expected service life of the items important to safety. When the items important to safety are subject to natural external events and are required to perform a safety function during or following such an event, the qualification programme shall replicate as far as is practicable the conditions imposed on the items important to safety by the natural external event, either by test or analysis, or by a combination of both.*"

Paragraph 5.50 of IAEA SSR‑2/1 (Rev. 1) [20] on environmental conditions in specific operational states requires the following:

"*5.50. Any environmental conditions that could reasonably be anticipated and that could arise in specific operational states, such as in periodic testing of the containment leak rate, shall be included in the qualification programme.*"

Paragraphs 3.68–3.75 of SSG-56 [69] provide recommendations on meeting Requirements 30 of SSR-2/1 (Rev. 1) [20] related to equipment qualification.

Paragraph 3.68 of SSG-56 clarifies IAEA meaning of 'environmental qualification' is to be qualified to perform functions in the entire range of environmental conditions:

"*3.68. The components and instrumentation for the reactor coolant system and associated systems are required to be qualified to perform their functions in the entire range of environmental conditions that might prevail prior to or during their operation, or should otherwise be adequately protected from those environmental conditions (see Requirement 30 of SSR‑2/1 (Rev. 1) [1]).*"

Paragraph 3.70 of SSG-56 [69] recommends that the environmental qualification should be carried out by means of testing, analysis and the use of experience, or through a combination of these.

Paragraph 3.71 of SSG-56 [69] recommends that the environmental qualification should include the consideration of such factors as temperature, pressure, humidity and radiation levels.

Paragraph 3.72 of SSG-56 [69] recommends that the techniques to accelerate the testing for ageing and qualification can be used, provided that there is adequate justification to do this.

For other paragraphs the reader is referred to IAEA SSG-56 [69]. Also, further recommendations on ageing management are given in IAEA SSG-48 [67] and IAEA SRS No. 106 [93], which complements IAEA SSG-48 [67].

Even more complete set of recommendations to satisfy IAEA GSR Part 4 (Rev. 1) [17] and IAEA SSR‑2/1 (Rev. 1) [20] requirements for equipment qualification can be found in IAEA SSG-69 [73]. This guide applies to electrical equipment, instrumentation and control and active mechanical equipment, as well as components associated with this equipment (e.g. seals, gaskets, lubricants, cables, connections, mounting and anchoring structures). The equipment qualification process described in IAEA SSG-69 [73] comprises three phases:

(a) Establishment of appropriate design inputs;

(b) Establishment of equipment qualification process steps;

(c) Preservation of the status of qualified equipment.

These three phases and the relationship of activities within each phase are considered in Sections 3, 4 and 5 IAEA SSG-69 [73], respectively. For this stage of the project important seems to be Annex INTERNATIONAL

STANDARDS RELATING TO EQUIPMENT QUALIFICATION of IAEA SSG-69 [73]. Namely, Requirement 9 of IAEA SSR‑2/1 (Rev. 1) [20] dealing with proven engineering practices states:

"*Requirement 9: Proven engineering practices*

Items important to safety for a nuclear power plant shall be designed in accordance with the relevant national and international codes and standards."

In Table A–1 of IAEA SSG-69 [73] international standards relating to equipment qualification are listed. A few significant for the sCO2-4-NPP project are listed in Table 12. According to IAEA SSG-69 [73], there are important differences between the IEC and the IEEE standards. The IEC standards take the IAEA Safety Requirements publications and Safety Guides as fundamental inputs for their development. As a result, the IEC standards deal with items important to safety and use IAEA recommendations and guidance on instrumentation and control systems as a basis. In contrast, the IEEE standards focus mostly on items important to safety. The IEEE standards can be applied to safety related items (i.e. items important to safety that are not safety systems) using a graded approach.

Table 12: Selected international standards for relating to equipment qualification (adapted TABLE A-1 of [73])

Standard No.	**Standard title**
IEC/IEEE 60980-344:2020	Nuclear Facilities — Equipment Important to Safety — Seismic Qualification
IEC 61513:2011	Nuclear Power Plants — Instrumentation and Control Important to Safety — General Requirements for Systems
IEC/IEEE 60780-323:2016	Nuclear Facilities — Electrical Equipment Important to Safety — Qualification
IEEE 308-2020	IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE 344-2013	IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations
IEEE 603-2018	IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations
IEEE 627-2019	IEEE Standard for Qualification of Equipment Used in Nuclear Facilities
ASME QME-1-2017	Qualification of Active Mechanical Equipment Used in Nuclear Facilities

5.5.10.2 Review of requirements for Czech plants

Section 4.10 of D3.3 [3] is entitled 'Environmental conditions for qualification' (paragraph 3.7 of IAEA SSG-56 [64], recommends that a design basis should specify also element 'Environmental conditions for qualification'; see Section 5.5.10.4 for further explanation). As explained in Section 4.10.2 of D3.3 [3], environment qualification means the ability of a system, structure or component to meet the requirements set out by technical specifications for its functioning in the working environment and in conditions triggered by the characteristics of the area surrounding the nuclear installation.

The return period of 10000 years is based on Decree 329/2017 [29] and WENRA Issue T: Natural hazards [8]. The measurement requirement for at least thirty years is based on IAEA SSG-18 [62].

The determination of seismic hazard of the NPP was prepared in accordance with IAEA standards NS-R-3 (Rev.1) [22], SSR-1 [19], SSR-2/1 rev.1 [20] and SSG-9 [60], using the Probabilistic Seismic Hazard Assessment (PSHA) approach.

5.5.10.3 Review of requirements for French plants

As explained in Section 4.10.3 of D3.3 [3] for French requirements for design basis element 'environmental conditions for qualification', ASN Guide No. 22 [43] tells us that equipment important to safety must be qualified to ensure its ability to meet its defined requirements for the conditions under which it is needed. These conditions must include conditions related to the environment (such as temperature, pressure, humidity, impact of fluid jets, irradiation, vibrations, chemical phenomena, electromagnetic interference and any plausible combination of these factors), as well as conditions related to the fluid conveyed (such as radioactive fluid, particle-laden water, thermal shock).

The large number of reactors in France means that the environmental requirements are adapted according to the location of the plant for which a dossier is filed. For example, power plants on the seaside will have to present an analysis for the risk of tidal waves that is very different from that for power plants on the river (which will have a flood risk). Similarly, the seismic risk does not have the same value depending on the location of the power plant.

This process implies that we cannot provide a table of tolerated values, as these will be calculated from the different fundamental safety rules and guides published by the ASN.

This qualification must be acquired during the expected duration of operation of the equipment, when replacement is possible during operation (in operation or during the shutdown and dismantling phase).

The qualification must in particular be based on design, construction, testing, inspection or maintenance provisions. These provisions are discussed in Section 5.6.

5.5.10.4 Assessment

D3.3 [3] in Section 4.10 deals with 'environmental conditions for qualification'. It seems that this title was selected due to paragraph 3.7 of IAEA SSG-56 [69], which recommends that a design basis should specify 'Environmental conditions for qualification'. Also, from paragraph 3.52 of IAEA SSG-56 [69] it can be judged that environmental conditions are addressed in paragraphs 3.68-3.75. However, paragraph 3.67 of IAEA SSG-56 [69] under heading "Environmental qualification of items important to safety" states that paragraphs 3.68–3.75 provide recommendations on meeting Requirement 30 of SSR-2/1 (Rev. 1) [20], which is entitled "Qualification of items important to safety" (also in the table of contents the wording '*Environmental qualification of items important to safety (3.67–3.75)*' is used). Therefore, it seems that paragraph 3.7 is not completely harmonized with the rest of IAEA SSG-56 [69] safety guide. The consequence of this was that for Czech Republic only '*environmental conditions for qualification*' were considered, which present the input to environmental qualification.

Also, according to paragraph 3.68 of IAEA SSG-56 [69] "*the components and instrumentation for the reactor coolant system and associated systems are required to be qualified to perform their functions in the entire range of environmental conditions that might prevail prior to or during their operation, or should otherwise be adequately protected from those environmental conditions.*" It means that SSCs have to be qualified to environmental conditions. According to Section 5.5.10.1 (i.e. '*environmental qualification*' is understood as one form of '*equipment qualification*').

This seems to be the root cause that Czech Republic provides only '*environmental conditions*', while requirements for environmental qualification are not provided (see Section 5.5.10.2). IAEA SSG-56 [69], IAEA SSG-69 [46] and IAEA SSG-67 [58] recommendations for environmental qualification specified in Section 5.5.10.1 may be followed.

From Section 5.5.10.3 for French requirements ASN Guide No. 22 [43] tells us that equipment important to safety must be qualified to ensure its ability to meet its defined requirements for the conditions under which it is needed. These conditions must include conditions related to the environment as well as conditions related to the fluid conveyed. As such, requirements of ASN Guide No. 22 [43] for qualification of equipment important to safety is broader then just '*environmental conditions*' of IAEA SSG-56 [69], but narrower than '*equipment qualification*' described in IAEA SSG-69 [73], which provides recommendations to meet the Requirement 30 of IAEA SSR-2/1 (Rev. 1) [20]. In addition, seismic risk was mentioned but this is out of the scope of environmental qualification according to IAEA. Namely, IAEA recommendations on seismic qualification for nuclear power plants are provided in IAEA SSG-67 [71].

Finally, it should be noted that '*equipment qualification*' requirements are considered in separate Section 5.6.

5.5.11 Monitoring and control capabilities

5.5.11.1 IAEA requirements

IAEA SSR-2/1 (Rev. 1) [20] standard gives requirements on instrumentation in the Requirement 59 (with associated paragraph 6.31):

"*Requirement 59: Provision of instrumentation*

Instrumentation shall be provided for: determining the values of all the main variables that can affect the fission process, the integrity of the reactor core, the reactor coolant systems and the containment at the nuclear power plant; for obtaining essential information on the plant that is necessary for its safe and reliable operation; for determining the status of the plant in accident conditions; and for making decisions for the purposes of accident management."

IAEA SSR-2/1 (Rev. 1) [20] standard gives requirements on instrumentation in the Requirement 60:

"*Requirement 60: Control systems*

Appropriate and reliable control systems shall be provided at the nuclear power plant to maintain and limit the relevant process variables within the specified operational ranges."

Section 3 entitled DESIGN BASIS FOR INSTRUMENTATION AND CONTROL SYSTEMS of IAEA SSG-39 [65] provide recommendations for design basis. For more details the reader is referred to IAEA SSG-39 [65].

Paragraphs 7.10-7.15 of IAEA SSG-39 [65] provide recommendations on meeting Requirement 60 of SSR-2/1 (Rev. 1) [20] for control systems.

5.5.11.2 Review of requirements for Czech plants

As explained in D3.3 [3], the main systems for monitoring and control in Temelín NPP are PRPS (Primary Reactor Protection System), PCS (Plant Control System), PAMS (Post-Accident Monitoring System) and DPS (Diverse Protection System). The task of PRPS and PCS system are briefly described. PAMS need to satisfy requirements of requirements of NRC Regulatory Guide 1.97 [103].

DPS is used in case of failure of PRPS.

Non-essential protection and control systems provide automatic and manual control functions (for SSC classified as SSB, DIV, ALT and VyDiD). Non-essential protection and control systems must have acceptable functional reliability, the level of which results from the set of functions provided. The sCO2-4-NPP system will have to be equipped with such control means to meet these objectives.

5.5.11.3 Review of requirements for French plants

As explained in D3.3 [3], in France instrumentation must be provided to measure the main quantities characterizing the nuclear reactions, the tightness of the fuel cladding, the efficiency of fuel cooling and the state of the containment of the buildings on the nuclear island, and to obtain the information on the installation that is necessary to operate it reliably and safely while limiting damage to the interests mentioned in article L. 593-1 of the environmental code [39].

The instrumentation must be adapted (measurement range, location, qualification, uncertainty...) to the situations in which it is required. The instrumentation and the methods for automatic recording of the quantities relevant for assessing nuclear safety must be chosen and designed in order to have the necessary information and to detect an incident or accident, monitor its evolution as well as the state of the containment barriers and safety functions.

Requirements about the I&C and monitoring equipment are defined in the RCC-E guide (cf. [53]). These requirements depend mainly on the safety class of the components and the system to which the I&C will be connected. Refer to Table 10 of D3.3 [3] which summarizes the rules according to the safety class.

5.5.11.4 Assessment

From Section 5.5.11.2 for Czech Republic requirements it can be seen that NRC Regulatory Guide 1.97 [103] is used for PAMS, which is consistent with the following documents:

- IAEA Nuclear Energy Series No. NP-T-3.16, “Accident Monitoring Systems for Nuclear Power Plants,” issued February 2015
- IAEA Safety Standard Series No. TECDOC-1818, “Assessment of Equipment Capability to Perform Reliably under Severe Accident Conditions,” issued July 2017
- IAEA Safety Standards Series No. SSG-39, “Design of Instrumentation and Control Systems for Nuclear Power Plants,” issued April 2016
- IEEE 63147-2017 - IEEE/IEC International Standard, “Criteria for accident monitoring instrumentation for nuclear power generating stations,” December 22, 2017

It is judged that Czech requirements met the intent of IAEA requirements.

From Section 5.5.11.3 for French requirements about the I&C and monitoring equipment it can be seen that there are defined in the RCC-E guide (cf. [53]). IAEA SSG-39 [65] list mainly IEC, IEEE and ISO/IEC standards. Relationship between the topic areas of IAEA SSG-39 [65] and international standards is given. It cannot be verified due to not provided information but it is expected that RCC-E guide [53] used for design of nuclear power plants in France (including EPR) in general met the intent of IAEA requirements for I&C.

5.5.12 Materials

5.5.12.1 IAEA requirements

Under Principle 8 of IAEA SF-1 [13] (see Section 5.2.1.2) in paragraph 3.32 it is stated that defence in depth is provided by an appropriate combination of an effective management system, adequate site selection and the incorporation of good design and engineering features providing safety margins, diversity and redundancy, mainly by the use of a) Design, technology and materials of high quality and reliability, b) Control, limiting and protection systems and surveillance features and c) An appropriate combination of inherent and engineered safety features.

Requirement 47 (with associated paragraphs 6.13-6.16) of IAEA SSR-2/1 (Rev. 1) [20] is related to the design of reactor coolant system (RCS):

"*Requirement 47: Design of reactor coolant systems*

The components of the reactor coolant systems for the nuclear power plant shall be designed and constructed so that the risk of faults due to inadequate quality of materials, inadequate design standards, insufficient capability for inspection or inadequate quality of manufacture is minimized."

In the case the sCO2-4-NPP will not be connected to RCS, this requirement seems not to be relevant. Paragraphs 3.88–3.92 of SSG-56 [69] provide recommendations on meeting Requirement 47 of IAEA SSR-2/1 (Rev. 1) [20]. It should be noted that paragraph 3.88 of SSG-56 [69] includes also associated systems as it states:

"*3.88. The materials used for the pressure retaining boundary of the reactor coolant system and associated systems should be specified with regard to chemical composition, microstructure, mechanical–thermal properties, heat treatment, manufacturing requirements and activation of materials, as applicable. The materials should be appropriately homogeneous and should be compatible with the coolant they contain, as well as with joining materials (e.g. welding materials) and with adjoining components or materials, such as sliding surfaces, spindles and stuffing boxes (packing boxes), overlay or radiolysis products.*"

Paragraph 3.89 recommends properties and characteristics of the material, used for RCS and associated systems.

As stated in Section 5.5.10.1, the Requirement 9 of IAEA SSR-2/1 (Rev. 1) [20] is dealing with proven engineering practices and it states:

“*Requirement 9: Proven engineering practices*

Items important to safety for a nuclear power plant shall be designed in accordance with the relevant national and international codes and standards.”

One of engineering factors is also selection of materials. As mentioned in Section 6.3.2.2 of D3.1 [1], codes and standards have been developed by various national and international organizations, covering different areas, including materials.

Paragraph 6.70 of IAEA SSR-2/1 (Rev. 1) [20] is related to design for radiation protection:

"*6.70. Materials used in the manufacture of structures, systems and components shall be selected to minimize activation of the material as far as is reasonably practicable.*"

5.5.12.2 Review of requirements for Czech plants

As explained in D3.3 [3], the main requirements for Czech Republic specific requirements for selected equipment and pressure equipment in the NPP are given in Decree 358/2016 [30].

Requirements on pressure equipment materials and materials of parts of pressure equipment exposed to pressure are specified. Only approved basic and auxiliary materials permitted for this use may be used to manufacture, repair, or modify pressure equipment. The list of materials must be drawn up with respect to classification of pressure equipment in the appropriate safety class.

In choosing material for manufacture, installation, repair, or modification of pressure equipment, it is necessary to take into account its chemical composition, physical and mechanical properties, weldability and ability to operate under operating conditions in which the pressure equipment is to perform its function. The material used in the manufacture, installation, repair, or modification of pressure equipment must be on the list of materials permitted for the given use. If the proposed material is not on the list of materials permitted for the given use, a specific assessment of the proposed material must take place; for pressure equipment specified in § 12(2), Decree 358/2016 [30], an authorized person must arrange the specific assessment of the proposed material.

5.5.12.3 Review of requirements for French plants

As explained in D3.3 [3], in France general material provisions are covered in chapters 2000 of the various sub-sections of RCC-M, Section I [52]. These Chapters 2000 entitled "MATERIALS" in the subsections of SECTION I, which may be supplemented by requirements given in the equipment specification, specify how the requirements of SECTION II are to be applied to components subject to the RCC-M.

These chapters include general rules on selection of grades according to inter-granular corrosion susceptibility and cobalt content limitation, and lists of applicable procurement specifications presented in Section II of the RCC-M.

RCC-M specified chemical compositions are generally in conformance with ASME II requirements for equivalent grades.

The differences between the RCC-M and ASME codes concern essentially the use of complementary analyses and additional restrictions, which are required in order to improve the selected properties (for details see Section 4.12.3 of D3.3 [3].

From the mechanical properties point of view, the requirements of the RCC-M are equivalent to those of the ASME code for equivalent grades. The RCC-M specifies, in addition, the verification at temperature of mechanical properties consistent with ASME tabulated values for design use.

In addition to the ASME code prescriptions, for low alloy steels RCC-M specifies a verification of mechanical properties after heat treatment, for mechanical properties at room and elevated temperature and not only after simulated stress-relief treatment.

5.5.12.4 Assessment

From Section 5.5.12.2 for Czech Republic requirements it can be seen that main requirements for Czech Republic specific requirements for selected equipment and pressure equipment in the NPP are given in Decree 358/2016 [30]. Only approved basic and auxiliary materials permitted for this use may be used to manufacture, repair, or modify pressure equipment. The list of materials must be drawn up with respect to classification of pressure equipment in the appropriate safety class. This seems to be in accordance with IAEA, that materials are selected in accordance with the relevant national and international codes and standards. At present no details are provided which codes and standards are used for safety classes.

From Section 5.5.12.3 for French requirements it can be seen that general material provisions are covered in chapters 2000 of the various sub-sections of RCC-M, Section I [52]. The selection of RCC-M [52] is in accordance with IAEA requirement that materials are selected in accordance with the relevant national and international codes and standards.

5.5.13 Provisions for testing, inspection, maintenance and decommissioning

The provisions below are related to the design requirements for the sCO2-4-NPP system. For requirements for testing and maintenance of sCO2-4-NPP system during operation, refer to Section 5.7.

5.5.13.1 IAEA requirements

Requirement 29 (with associated paragraphs 6.13-6.16) of IAEA SSR-2/1 (Rev. 1) [20] states:

"*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."

Paragraphs 3.100–3.115 of SSG-56 [69] provide recommendations on meeting Requirement 29 of SSR-2/1 (Rev. 1) [20]. The design should establish a technical basis for structures, systems and components that require in-service inspection, examination, testing, maintenance and monitoring. The design should incorporate provisions to facilitate examination, testing, in-service inspection, maintenance, repair and monitoring to be carried out during the construction, commissioning and operation stages (the reader may refer also to DS497E [83] on maintenance, testing, surveillance and inspection in nuclear power plants during operation). Structures, systems and components important to safety should be designed and located to make surveillance and maintenance simple.

Requirement 12 (with associated paragraphs 4.20) of IAEA SSR-2/1 (Rev. 1) [20] deals with decommissioning:

"*Requirement 12: Features to facilitate radioactive waste management and decommissioning*

Special consideration shall be given at the design stage of a nuclear power plant to the incorporation of features to facilitate radioactive waste management and the future decommissioning and dismantling of the plant."

Considerations of decommissioning during design stage are given in paragraphs 7.5-7.9 of IAEA SSG-47 [66]. Relevant features and aspects that should be considered during the design stage of a facility to facilitate decommissioning are recommended.

5.5.13.2 Review of requirements for Czech plants

As explained in D3.3 [3], the main requirements for Czech Republic are given in Decree 358/2016 [30] on requirements for assurance of quality and technical safety and assessment and verification of conformity of selected equipment. Requirements for final assessment and checks of selected equipment upon completion of manufacturing and installation, and checks of selected equipment performed within the scope of conformity assessment following repair, maintenance, or re-installation following repair, or maintenance of selected equipment are given in Annex 6 of Decree 358/2016 [30]. Final assessment of pressure equipment must include:

1) a final test;
2) a pressure test, a tightness test, or other equivalent check; and
3) a check of safety equipment and equipment ensuring the functionality of pressure equipment.

Supervision by an authorized person must ensure that a manufacturer, importer, or person installing selected equipment fully complies with requirements that follow from the approved management system, including manufacturing quality assurance requirements.

5.5.13.3 Review of requirements for French plants

As explained in D3.3 [3], in France the purpose of the Operating Technical Specifications (OTS) of a power plant is to define the equipment required in operation as well as the conduct to be followed in case of unavailability of an SSC and the test programs. They must be updated to be adapted to the operation of the sCO2 system.

The inspection and maintenance programs required for each SSC should be defined according to the characteristics of the system and/or component.

The programmed unavailability of equipment shall be taken into account in the design of the systems. That is to say that the impact of the authorized unavailability times for line maintenance, tests and scheduled repair work will be included in the assessment of the reliability of the SSCs and, if necessary, in the reference studies for the modification file.

Although the system is not yet precisely defined, we can already expect periodic tests to be carried out at the same frequency as for similar systems (emergency diesels, vent to atmosphere circuit for the EPR, etc.).

5.5.13.4 Assessment

From Section 5.5.13.2 for Czech Republic requirements it can be seen that requirements for final assessment and checks of selected equipment upon completion of manufacturing and installation, and checks of selected equipment performed within the scope of conformity assessment following repair, maintenance, or re-installation following repair, or maintenance of selected equipment are given. It can be judged that final assessment is a sort of acceptance testing. It seems that no provisions for other testing, inspection, maintenance and decommissioning are given for designing of sCO2-4-NPP.

From Section 5.5.13.3 it can be seen that Operating Technical Specifications of a power plant had to define the equipment required in operation as well as the conduct to be followed in case of unavailability of an SSC and the test programs. The inspection and maintenance programs are required for each SSC to be defined according to the characteristics of the system and/or component. The programmed unavailability of equipment shall be taken into account in the design of the systems. No guidance how to satisfy these design requirements are given.

In cases when national regulations are not known, it is recommended to follow IAEA safety standards for design of sCO2-4-NPP system.

5.6 Requirements for equipment qualification

5.6.1 IAEA requirements

IAEA requirements for equipment qualification are extensively described in Section 5.5.10.1, which focus on environmental qualification requirements. For clarity reasons it should be noted that in the scope of equipment qualification could be e.g. seismic qualification, environmental qualification, electromagnetic qualification. Therefore, recommendations on equipment qualification of IAEA SSG-69 [73] not presented in Section 5.5.10.1 will be described in this section.

In Section 3 of IAEA SSG-69 [73], the design inputs that are necessary for equipment qualification should be established and documented in a specification that includes the following:

(a) The performance requirements necessary to accomplish the intended safety functions;

(b) The specified environmental conditions and operating conditions expected in operational states and accident conditions, including for seismic events;
(c) The safety class (see IAEA SSG-30 [64]) assigned to the equipment and the corresponding supplemental classifications (e.g. seismic classification, quality classification);
(d) The acceptance criteria for equipment qualification.

Examples of performance requirements include requirements for accuracy, resolution, range, sample rate and response time. Relevant environmental conditions for operational states typically ambient temperature and pressure, humidity and steam, radiation level, submergence, chemical leakages (e.g. boric acid, steam spray), chemicals in the atmosphere (e.g. salt mist, oil aerosols, dust), induced vibrations from neighbouring equipment or from a seismic event and SL-1 vibration.

Relevant operating conditions for operational states typically power surges, operating cycles (e.g. electrical, mechanical, water hammer), electrical loading parameters (e.g. voltage, frequency, current), mechanical loads (e.g. thrust; torque; displacement; non-seismic vibration including flow induced vibration, condensing mode vibration and quenching vibration), process fluid conditions (e.g. pressure, temperature, chemical composition, flow rate, water hammer), chemical composition, loads and duty cycles, self-heating, submergence, electromagnetic interference and electromagnetic fields.

Service conditions recommendations are specified for equipment located in mild environments, for harsh environments resulting from design basis accidents and conditions resulting from design extension conditions with core melting.

In Section 4 of IAEA SSG-69 [73], the following equipment qualification methods are recommended:

(a) Type tests;
(b) Analysis;
(c) Evaluation of operating experience;
(d) Where appropriate, an assessment of equipment capability for design extension conditions;
(e) A combination of the above methods.

For type testing the recommendations are given on the following:

- General recommendations,
- Test specification for equipment qualification by type testing,
- Test specimens for equipment qualification by type testing,
- Simulation of ageing effects (pre-ageing) in type tests for equipment qualification,
- Simulation of seismic conditions in type testing for equipment qualification,
- Simulation of specified service conditions in type testing for equipment qualification, and
- Margins for test profiles in type testing for equipment qualification.

Information on suitable margins for conducting type tests on electrical equipment important to safety is provided in IEC/IEEE 60780-323 [107]).

The qualification requirements in IEC/IEEE 60780-323-2016 [107] are intended, when met, to demonstrate and document the safety of electrical equipment under applicable service conditions and to reduce the risk of environmentally induced, common-cause equipment failure. In 2017 IEC/IEEE 60780-323-2016 [107] was adopted as European Norm EN 60780-323 “Nuclear facilities - Electrical equipment important to safety - Qualification - IEC/IEEE 60780-323:2016” [108]. EN 60780-323 describes the basic requirements for qualifying electrical equipment important to safety and interfaces (electrical and mechanical) that are to be used in nuclear facilities. The principles, methods, and procedures described are intended to be used for qualifying equipment, maintaining and extending qualification, and updating qualification, as required, if the equipment is modified. The qualification requirements in this standard, when met, demonstrate and document the ability of equipment to perform safety function(s) under applicable service conditions, including design basis events and certain design extension conditions, and reduce the risk of environmentally induced common-cause equipment failure.

Qualification by analysis alone is recommended by IAEA SSG-69 [73] only for the analysis of the structural integrity of the equipment and its mounting; it is not recommended for analysing equipment functionality.

Operating experience may be used to help demonstrate the reliability of equipment to perform safety functions. Equipment cannot be qualified on the basis of operating experience feedback only, and this should therefore be combined with other qualification methods.

Equipment qualification may be achieved through a combination of type testing, analysis and operating experience. The combination of methods used for equipment qualification should be justified and documented.

The equipment qualification programme should be subject to a quality assurance programme that includes a variety of elements, such as equipment design control, procurement document control, manufacturing quality control, qualification assessment (e.g. testing, analysis, combined testing and analysis, experience), storage, installation and commissioning, installation surveillance and maintenance, periodic testing and documentation. Equipment qualification activities, including the assessment or reassessment of the status of qualified equipment, should be performed in accordance with approved procedures and controls. Traceability should be established between the qualification documentation.

Modelling and/or simulations of specified service conditions should be used to derive the parameters needed as inputs for the qualification process. Recommendations on conducting such modelling and simulations are provided in IAEA SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [57].

Finally, IAEA SSG-69 [73] does not specify seismic qualification methods and processes. Recommendations on seismic qualification for nuclear power plants are provided in IAEA SSG-67 [71]. Section 6 of IAEA SSG-67 [71] entitled SEISMIC QUALIFICATION gives recommendations on:

- Qualification methods (paragraphs 6.3–6.10),
- Qualification by analysis (paragraphs 6.11–6.18),
- Qualification by testing (paragraphs 6.19–6.26),
- Qualification by a combination of analysis and testing (paragraphs 6.27–6.29),
- Qualification by indirect methods (paragraphs 6.30, 6.31).

5.6.2 Review of requirement for Czech plants

As explained in Section 5.1 of D3.3 [3], the manner in which selected equipment and parts of selected equipment are designed, manufactured, and installed must be documented in a way that permits conformity assessment[3]. Conformity assessment must be documented in conformity assessment documentation pursuant to requirements stipulated in Decree 358/2016 [30]. Conformity assessment documentation must be archived for the entire service life of the selected equipment.

Conformity assessment must take place prior to the use of selected equipment in a nuclear installation.

[3] § 3 (4) of Decree 358/2016 [30] explains the meaning of conformity assessment following: The selected equipment list must indicate selected equipment for which pursuant to § 12(2) its conformity with technical requirements (hereinafter 'conformity assessment') is performed by an authorised or accredited person.

The conformity assessment of a pressure equipment assembly must be performed with regards to the uppermost safety class into which one of the pieces of selected equipment that are part of the assembly is classified.

A manufacturer or importer must ensure within performing conformity assessment that the selected equipment is marked with a conformity mark along with manufacturer’s identification; if the conformity assessment procedure stipulates participation of an authorized or accredited person, this person must also be identified.

In Annex 7 of Decree 358/2016 [30] conformity assessment procedures are described.

5.6.3 Review of requirements for French plants

5.6.3.1 Testing Qualification Strategy

As explained in Section 5.2.1 of D3.3 [3], component and system qualification are two important points for the justification and validation of the use of a sCO2-4-NPP system. Indeed, this qualification will feed the modification file and must be included in the operator's integrated management system.

The purpose of the qualification is to provide proof that the equipment meets all the requirements requested according to its classification, and according to the ambient and environmental conditions. As far as the qualification of components is concerned, part of the expectations will depend on their qualification. Worldwide, the classification of equipment is not standardized. In France, equipment is classified according to its level of requirement for nuclear safety. There are three levels of qualification:

- Category K1: it concerns components, located inside the reactor building, having to ensure their functions in environmental conditions corresponding to normal, accidental and/or post-accidental operating conditions, and under seismic stress. Equipment in the Severe Accident (SA) category, located inside the reactor building, having to perform its functions under environmental conditions corresponding to normal, accidental and/or post-accident operating conditions of the reactor core, and under seismic stress.
- Category K2: it concerns the components, located inside the reactor building, having to carry out their functions in environmental conditions corresponding to normal operating conditions and under seismic stress.
- Category K3: Are concerned the components, located outside the reactor building having to ensure their functions in environmental conditions corresponding to normal operating conditions and under seismic stress. This category includes a sub-category K3ad, which concerns components located outside the reactor building that have to perform their functions in environmental conditions corresponding to operating conditions in a degraded environment (pressure, temperature, irradiation, etc.) and under seismic stress.

Testing qualification strategy process consists in determining a demonstration strategy that can be done either by analysis, tests, or a combined method (analysis + tests). In the case of tests, a list and the order of tests are defined by notifying the severities to be applied and the acceptance criteria.

The IEC 60780 standard defines in a general way the qualification of electrical safety equipment. In addition, manufacturers' standards (RCC-M, RCC-E) define and describe the tests to be carried out for a qualification as well as the associated documents.

The demonstration of an equipment's ability to operate under the conditions for which it was intended is based on a qualification strategy. In the case of tests, it is important to respect the defined sequence (order and criteria). Several types of tests are carried out:

- Reference tests (visual examinations, electrical tests, functional tests);
- Tests of functional limits of use (voltage variation, climatic test, EMC humidity);
- Tests of robustness in time (climatic test, mechanical tests, irradiation tests);
- Accidental tests (earthquake, accidental irradiation, thermodynamic accident).

5.6.3.2 Numerical Qualification Strategy

As explained in Section 5.2.2 of D3.3 [3], the strategy of qualification through testing will enable the system components to be qualified in terms of mechanical strength, but to qualify the system as a whole, it will be necessary to use digital tools.

This numerical strategy will depend on the numerical codes used and calculations such as PSA. Both the codes used and the PSAs will have to follow certain regulations to be able to assure the safety authorities of the value of the results obtained.

The probabilistic analyses mentioned in article 3.3 of the decree of February 7, 2012 [INB Order] and the probabilistic safety assessments (PSA) mentioned in article 8.1.2 of this decree are carried out in order to guide or support the design choices of systems ensuring a safety function or a support function, particularly in terms of redundancy and diversification, with regard to safety objectives.

In the case of the qualification of the sCO2-4-NPP system, Probabilistic Safety Assessment studies may be carried out, integrating the sCO2-4-NPP system in the possible scenarios.

Qualification of computer codes incorporates a procedure aimed at justifying the validity of results and stipulating the respective responsibilities of the supplier of the code, the subcontractor (if the support study is carried out under contract) and of the plant operator with regard to code implementation.

If the manufacturers or the operator used computer codes specifically for the design of equipment or structures, a code description must be provided. The use of qualified codes will be necessary for those used in uncertainty calculations and more generally for the demonstration of nuclear safety.

5.6.3.3 Qualification Quality requirements

The nuclear field imposes rigor in the traceability from the design to the operation put in operation on site in order to be able to demonstrate at any time the respect of the requested requirements. Certain documents are analyzed by the nuclear safety authorities of each country using nuclear power. Required is traceability for all steps, from design to operation. The qualification specimen must be representative of the equipment assembled on site. The tests are based on international standards, codes and norms. It is requested to carry out the tests under a COFRAC (name of the French committee for accreditation) accreditation (or equivalent to ISO/IEC 17025 [109]). Failure to comply with the defined criteria leads to the management of a non-conformity.

5.6.4 Assessment

Based on information in Section 5.6.2 and the fact that Decree 358/2016 [30] prescribes requirements for assurance of quality and technical safety and assessment and verification of conformity of selected equipment, it is judged that these Czech requirements are focused mainly on qualification quality requirements (i.e.

conformity with technical requirements; see also French qualification quality requirements in Section 5.6.3.3). In Annex 7 of Decree 358/2016 [30] conformity assessment procedures are described. Conformity assessment involves a set of processes that show your product, service or system meets the requirements of a standard.

French qualification requirements are given for testing qualification strategy, numerical qualification strategy and qualification quality (see Sections 5.6.3.1, 5.6.3.2 and 5.6.3.3, respectively). Testing qualification strategy process consists in determining a demonstration strategy that can be done either by analysis, tests, or a combined method (analysis + tests). Such approach is comparable to IAEA approach, which includes also evaluation of operating experience (may help to demonstrate the reliability, but equipment cannot be qualified on the basis of it only). The IEC 60780 standard, recommended also by IAEA is used for qualification of electrical equipment. For numerical qualification strategy to qualify the system as a whole it will be necessary to use digital tools (deterministic and probabilistic). In respect to digital tools, IAEA recommends simulations of specified service conditions to be used to derive the parameters needed as inputs for the qualification process. According to IAEA deterministic safety analyses are primarily required to demonstrate adequate fulfilment of safety functions by the design, to ensure that barriers to the release of radioactive material will prevent an uncontrolled release to the environment for all plant states, and to demonstrate the validity of the operational limits and conditions. The objectives of a probabilistic safety analysis are to determine all significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined (see Section 5.2.1.5). It seems that the intent of French use of deterministic and probabilistic is similar to IAEA regardless the fact that the term 'numerical qualification strategy' is not used in the frame of IAEA safety standards and safety analysis is not part of equipment qualification, rather part of safety assessment process (see Figure 1). The French qualification quality requirements on traceability, accredited testing laboratories and management of a non-conformity conform to IAEA which requires quality assurance program.

5.7 Requirements for operation and maintenance

5.7.1 IAEA requirements

The IAEA safety standards in the domain of NPP operational safety have been discussed already in Section 3.2 of D3.4 [4]. The description includes the Specific Safety Requirements publication IAEA SSR-2/2 (Rev. 1) [21]) from 2016 and a number of Safety Guides. The IAEA NS-G-2.2 [75], IAEA NS-G-2.3 [76] and IAEA NS-G-2.6 [77] safety guides were identified for implementation of Requirements 10, 11 and 13 of IAEA SSR-2/2 (Rev. 1) [21], respectively.

In Section 3.2 of D3.4 [4] it was also explained that in the IAEA NS-G-2.2 [75] the operational limits and conditions should be expanded to also cover design extension conditions, that IAEA NS-G-2.3 [76] is dealing with configuration control and management of modifications, and IAEA NS-G-2.6 [77] should be implemented to address adequately Equipment Qualification in relation to activities needed during operation, including realistic performance targets under DEC conditions. Then, summary of Requirements 6, 25 and 31 of IAEA SSR-2/2 (Rev. 1) [21] has been also provided.

IAEA NS-G-2.2 [75], IAEA NS-G-2.3 [76] and IAEA NS-G-2.6 [77] safety standards were briefly summarized, too.

The above information from Section 3.2 of D3.4 [4] need to be complemented with the current situation in July 2022. According to IAEA document preparation profile DS497 [86] the above guides were published in the period 2000–2002 and represent the international consensus on operational safety which existed at that time, when design extension conditions had not yet been introduced (i.e. no guidance on DEC is given in below described IAEA guides). Also, all IAEA safety standards before 2013 will be revised. The guides would benefit from amendments to take into consideration revisions implemented in the other safety standards and, in particular, the IAEA SSR-2/2 (Rev. 1) [21]) and lessons from the Fukushima Daiichi accident. Therefore, draft safety guide revisions of NS-G-2.2 (DS497A [81]), NS-G-2.3 (DS497B [82] and NS-G-2.6 (DS497E [83]) are used in the following to cover guidance on DEC and lessons from the Fukushima Daiichi accident. Namely, according to CSS page (https://www-ns.iaea.org/committees/css/) these drafts (after Step 12) are endorsed by the CSS awaiting publication (it means that editorial changes are expected, while the content is not expected to be changed).

In the following Requirements 10, 11, 13, 18 and 32 of IAEA SSR-2/2 (Rev. 1) [21]) are presented. For Requirements 6 and 28 of IAEA SSR-2/2 (Rev. 1) [21]) description refer to Section 5.7.1.1 on operational limits and conditions, while Requirements 25 and 31 of IAEA SSR-2/2 (Rev. 1) [21]) are described in Section 5.7.1.2 on maintenance during operation.

IAEA SSR-2/2 (Rev. 1) [21]) Requirement 10 gives the following requirements for control of plant configuration:

"*Requirement 10: Control of plant configuration*

The operating organization shall establish and implement a system for plant configuration management to ensure consistency between design requirements, physical configuration and plant documentation."

DS497B [82] (draft safety guide revision of NS-G-2.3) paragraphs 4.27-4.32 give recommendations for configuration control. Requirement 10 of IAEA SSR-2/2 (Rev. 1) [21] and paragraph 4.38 require consistency between modifications, design requirements and plant documentation. When modifications are made to structures, systems and components and process software, the relevant plant documentation should be modified accordingly. When modifications are to be made to operational limits and conditions, the associated operating instructions and procedures should be modified accordingly, and in some cases the associated structures, systems and components might also be subject to modification. Configuration management should also be used to ensure that the implementation of the modification is in accordance with the design requirements as established in the design documentation. Any updates to the configuration of the plant simulator or training facilities should be included within the modification programme, to ensure that this programme accurately reflects all modifications and changes made to the plant.

IAEA SSR-2/2 (Rev. 1) [21]) Requirement 11 gives requirements for management of modifications. Modifications can be done on the original design (new advanced reactor) and on existing operating reactors. Refer to Section 5.4.1, where plant modifications have been already considered.

IAEA SSR-2/2 (Rev. 1) [21]) Requirement 13 gives following requirements for equipment qualification:

"*Requirement 13: Equipment qualification*

The operating organization shall ensure that a systematic assessment is carried out to provide reliable confirmation that safety related items are capable of the required performance for all operational states and for accident conditions."

Here the focus is on preservation of equipment qualification (see also to Section 5.5.10.1, where Requirement 30 of IAEA SSR-2/1 (Rev. 1) [20] for qualification requirements of items important to safety are given and Section 5.6.1, where guidance for this requirement is given by IAEA SSG-69 [73]). Guidance for preservation of equipment qualification is also given in IAEA SSG-69 [73] (see Section 5 of IAEA SSG-69 [73], which gives recommendations also for Requirement 13 (with associated paragraph 4.48) of IAEA SSR-2/2 (Rev. 1) [21]). To meet the requirements of Requirement 13 (with associated paragraph 4.48), qualified equipment should be designed, manufactured, procured, stored, installed, commissioned, inspected, operated, maintained and replaced or modified in a manner that helps to ensure that the equipment qualification is preserved for the lifetime of the installation.

Paragraph 5.7 of IAEA SSG-69 [73] list factors that can adversely impact the established equipment qualification. In the following only the factors of relevance for this report are listed:

- Changes in the design basis or safety analysis;
- Changes in regulatory requirements or in licensing conditions;
- Modifications to the nuclear installation;
- Deviations in service conditions from those assumed in the equipment qualification.

For other factors that can adversely impact the established equipment qualification refer to IAEA SSG-69 [73].

Section 6.2.3 of D3.3 [3] specify French requirements for emergency preparedness. For emergency preparedness IAEA prepared IAEA GSR Part 7 [18] safety standard.

IAEA SSR-2/2 (Rev. 1) [21]) standard for operation provides Requirement 18 with the following requirements for emergency preparedness:

"*Requirement 18: Emergency preparedness*

The operating organization shall prepare an emergency plan for preparedness for, and response to, a nuclear or radiological emergency."

Paragraphs 5.2-5.7 are associated to Requirement 18 of IAEA SSR-2/2 (Rev. 1) [21]). Current document DS504 [85] is a revision of the GS-G-2.1 [55] which was published in 2007. GS-G-2.1 [55] supported the Safety Requirements Preparedness and Response for a Nuclear or Radiological Emergency (GS-R-2 [23]), which was superseded and replaced by GSR Part 7 [18] (same title) published in 2015. DS504 [85] complements the Safety Guide Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency (GSG-2) [56] published in 2011.

Section 6.2.1 of D3.3 [3] specify French requirements for human factor. IAEA SSR-2/1 (Rev. 1) [20]) standard for human factors specify Requirement 32 (with associated paragraphs 5.53-5.62), giving the following requirements:

"*Requirement 32: Design for optimal operator performance*

Systematic consideration of human factors, including the human–machine interface, shall be included at an early stage in the design process for a nuclear power plant and shall be continued throughout the entire design process."

Section 4 of IAEA SSG-51 [68] provide recommendations for Requirement 32 and associated paragraphs (paragraphs 5.55, 5.56, 5.60 and 5.61 of IAEA SSR-2/2 (Rev. 1) [21] are specifically mentioned in IAEA SSG-51 [68]).

5.7.1.1 Operational limits and conditions

Similarly like for equipment qualification, requirements for operational limits and conditions (OLCs) are given both for design and operation of nuclear power plant. Requirement 28 of IAEA SSR-2/1 (Rev. 1) [20] requires to establish a set of OLCs, while Requirement 6 of IAEA SSR-2/2 (Rev. 1) [21]) requires that the plant is operated in accordance with the established set of OLCs.

IAEA SSR-2/1 (Rev. 1) [20]) Requirement 28 gives following requirement for establishment of OLCs:

"*Requirement 28: Operational limits and conditions for safe operation.*

The design shall establish a set of operational limits and conditions for safe operation of the nuclear power plant."

Associated paragraph 5.44 to Requirement 28 of IAEA SSR-2/1 (Rev. 1) [20] states that the requirements and operational limits and conditions established in the design for the nuclear power plant shall include safety limits, limiting setting for safety systems, limits and conditions for normal operation, surveillance and testing requirements, specified operational configurations (including operational restrictions) in the event of the unavailability of safety systems or safety related systems, and action statements for deviations from normal operation (see Requirement 6 of IAEA SSR-2/2 (Rev. 1) [21] below).

IAEA SSR-2/2 (Rev. 1) [21]) Requirement 6 gives following requirement for operation in accordance with the established set of OLCs:

"*Requirement 6: Operational limits and conditions*

The operating organization shall ensure that the plant is operated in accordance with the set of operational limits and conditions."

The purpose of DS497A [81] (draft safety guide revision of NS-G-2.2 [75]) is also to provide recommendations on the development, content and implementation of OLCs and operating procedures for nuclear power plants, to meet Requirement 6 (and associated paragraphs 4.6-1.15) of SSR-2/2 (Rev. 1) [21].

The operating organization is required to ensure compliance with OLCs. A major contribution to compliance with OLCs is the provision of operating procedures that are consistent with the OLCs. Some OLCs might be directly stated in procedures or in other documents, and, if so, this should be clearly indicated in the relevant document. Verifications of the compliance with OLCs should be regularly performed by the operating organization. Procedures shall be updated periodically and in a timely manner in the light of operating experience and the actual plant configuration.

The OLCs should be defined in such a way that the independence of the levels of defence in depth as far as is practicable (for Requirement 7 of IAEA SSR-2/1 (Rev. 1) [20] dealing with application of defence in depth see Section 5.2.1.2) and their adequate reliability is ensured. The OLCs should define operational requirements to ensure that items important to safety perform their functions in all operational states, in design basis accidents and in design extension conditions for which they are necessary. This includes permanently installed, portable and mobile equipment used for accident management (including for severe accident management). The OLCs should be based on a safety analysis of the individual plant and its environment. The use of deterministic safety analysis should be complemented by probabilistic safety analysis, as appropriate. The initial OLCs should normally be developed by the operating organization in cooperation with the plant designers well before commencement of operation to ensure that adequate time is available for an independent assessment commissioned by the operating organization.

5.7.1.2 Maintenance during operation

IAEA SSR-2/2 (Rev. 1) [21]) Requirement 25 gives following requirements for commissioning programme:

"*Requirement 25: Commissioning programme*

The operating organization shall ensure that a commissioning programme for the plant is established and implemented."

Associated paragraphs 6.1-6.15 to Requirement 25 of IAEA SSR-2/2 (Rev. 1) [21]) provide more detailed requirements. Recommendations how to conform to Requirement 25 of IAEA SSR-2/2 (Rev. 1) [21]) are given in IAEA SSG-28 [63]. Section 2 of IAEA SSG-28 [63] provides recommendations for the commissioning process (including development), while Section 4 of IAEA SSG-28 [63] provides recommendations for implementation of commissioning programme. Designers and other specialists should be involved in the development and review process for commissioning and test programmes, procedures and results. Implementation of the commissioning tests include: commissioning tests; preparation for testing; prerequisites for testing; testing stages and sequences; review, evaluation and reporting of testing results; and handling of deviations during commissioning.

IAEA SSR-2/2 (Rev. 1) [21]) Requirement 31 gives following requirements for establishment and implementation of effective programmes for maintenance, testing, surveillance and inspection:

"*Requirement 31: Maintenance, testing, surveillance and inspection programmes*

The operating organization shall ensure that effective programmes for maintenance, testing, surveillance and inspection are established and implemented."

IAEA DS497E [83] (the draft safety guide revision of IAEA NS-G-2.6 [77]) entitled "Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants" provide specific recommendations on maintenance, testing, surveillance and inspection (hereafter collectively referred to as 'MTSI activities'). The purpose of this safety guide is to provide recommendations on MTSI activities in nuclear power plants, to meet Requirements 28, 31 and 32 of IAEA SSR-2/2 (Rev. 1) [21]). Recommendations are also provided on maintaining equipment qualification, and on MTSI activities to meet Requirement 13 IAEA SSR-2/2 (Rev. 1) [21].

Recommendations for establishment of MTSI programme are given in Section 2 of IAEA DS497E [83] (paragraphs 2.2-2.28 for maintenance, 2.19-2.22 for surveillance, 2.22-2.24 for inspections and 2.25 for testing). Recommendations for implementation of MTSI are given in Section 5 of IAEA DS497E [83] (paragraphs 5.1-5.63).

IAEA SSR-2/2 (Rev. 1) [21]) Requirement 32 gives following requirements for establishment and implementation of arrangements to ensure outage management:

"*Requirement 32: Outage management*

The operating organization shall establish and implement arrangements to ensure the effective performance, planning and control of work activities during outages."

Recommendations are given in paragraphs 5.27-5.38 of IAEA DS497E [83]. Outage management should be based on the application of the concept of defence in depth. The operating organization should ensure the safe and effective implementation and control of all MTSI activities during both planned and unplanned outages. The outage safety reviews should be based on a well-defined set of OLCs for shutdown states. After each outage a comprehensive review shall be performed to draw lessons to be learned.

5.7.2 Review of Czech regulations for operation and maintenance

As explained in Section 4.1 of D3.4 [4], Act No. 263/2016 (Atomic Act) [25] states that "limits and conditions mean a set of requirements, compliance with which means that the performance of activities [related to the use of radioactive materials] is considered safe".

Limits and conditions for Temelín NPP must meet the legislative requirements given in particular by Act 263/2016 [25] and Decree 106/1998 [26]. The requirements for testing are given in Decree 358/2016 [30]. The reader is referred to Section 6.1 of D3.3 [3], where requirements for ensuring compliance when operating selected equipment and parts thereof are presented (see also Annex 2 Decree 358/2016 [30], section E. Requirements for ensuring compliance when operating selected equipment and parts thereof). Among other it is also required that "*during the operation of selected equipment, a maintenance system and a system of* *checks performed during the operation of selected equipment must be put into place, which must be implemented with respect to operating conditions affecting the technical safety of this equipment; and stipulate technical and organisational measures for conformity assurance*".

The philosophy of plant operation limits and conditions is consistent with NUREG 1431, Vol. 1 [101] and Vol. 2 [102].

5.7.2.1 Limits and conditions for Temelín NPP

As explained in Section 4.2 of D3.4 [4], the set of Limits and conditions for Temelín NPP must be clear and the internal structure of the documentation must guarantee unambiguous interpretation. This requirement is achieved by the use of NUREG 1431 Vol. 1 [101] and Vol. 2 [102]. Therefore, the requirements for Temelín NPP are divided into two volumes: Part A (TL001): Limits and conditions [35] and Part B (TL002): Limits and conditions – justification [36]. Content of both volumes corresponds to NUREG 1431 to the applicable extent.

Limits and conditions, Part A contains the conditions for the safe operation of the safety systems of Temelín NPP based on safety analyses that demonstrate the accidents mitigation within abnormal operation and emergency conditions. This part forms clearly defined conditions within which the safety of the Temelín NPP is maintained. Failure to comply with the conditions set out in Part A means immediate disruption of the safe operation of the power plant.

Limits and conditions, Part B describes the conditions for the safe operation of the systems related to nuclear safety, during normal operation of Temelín NPP. Violation of the conditions set in Part B does not cause an immediate threat to plant safety, however, long-term power operation is not possible without their fulfilment.

The sCO2-4-NPP system is formally not a safety system. It serves to mitigate an accident’s consequences as an alternative system and therefore it can have an impact on plant safety.

During normal operation of the unit at all power levels, the sCO2-4-NPP system is not utilized and should be in a stand-by mode. The system should be actuated automatically in case of an SBO, LUHS and combinations of the above if other means for the decay heat removal fail.

The operability of all modules of the sCO2-4-NPP system is required to ensure the sufficient heat removal in accident conditions after the reactor shutdown. The pre-commissioning test (final test) requirements and the expected supervision by an authorized person were already described in the Section 4.13.2 of D3.3 [3]. During the lifetime of the plant, the system should be controlled in a regular manner to prove its operability.

The regular check is required to ensure the operability of the system. The control is performed by a physical check. This control is part of the program of so-called cyclic controls. For the systems, which are redundant according to the principle 3 x 100% each division will be verified once every 3 months. If the system is redundant, it should be controlled every 31 days. The second type of test, conducted once every 18 months, should demonstrate that the sCO2-4-NPP system *as a whole* is operational.

5.7.2.2 Requirements for operational tests of SSCs

As explained in Section 4.3 of D3.4 [4], the next level of the requirements is the document “Requirements for operational tests of SSC according to plant Limits and conditions – primary circuit systems and auxiliary systems” (1TC014/2 [37], parts A and B). All the important systems and components and their testing procedures are described in two parts of this document.

The following elements are considered for each SSC: explanation, test objective, testing frequency, initial state, list of monitored parameters, test procedure, final state of the unit and considered system, and success criteria and test evaluation.

5.7.2.3 Requirements for the operation of the specific system

As explained in Section 4.4 of D3.4 [4], there are separate documents for each system. Such detailed description will have to be prepared for the sCO2-4-NPP system if it is to be installed in one of the Czech NPPs. This document consists of the following chapters:

- Introduction
- System description
- System components
- System operation
- General system requirements

For further details refer to Section 4.4 of D3.4 [4].

5.7.3 Review of French regulations for operation and maintenance

As explained in Section 6.2 of D3.3 [3], the regulations to be applied for the sCO2-4-NPP system in the context of plant and system operation can be defined once the development of the system is more advanced in terms of determining the procedures and scenarios for system operation. Nevertheless, it is already possible to determine which documents or areas may be impacted by the sCO2-4-NPP system.

As explained in Section 5.1 of D3.4 [4], legislation on operational safety and testing of safety-related equipment in nuclear power plants in France is derived from the international texts (IAEA, WENRA). In France, it is set out in the following texts:

- Environment French Code [40]
- BNI Decree [44]
- Order of 10 November 1999 [45]
- ESPN Order [42]

Based on these texts, the operator will apply the regulations through internal documents specifying the characteristics and parameters to be respected for the systems, the general operating rules and the maintenance and test procedures.

The "safety report" is the document that the operator must submit to the ASN in support of an application for authorisation to create, commission or dismantle a basic nuclear installation (BNI).

General Operating Rules (GOR) for the plant are a collection of rules approved by the ASN which define the authorised area of operation of the installation and the associated control prescriptions. Chapter III of GOR describes the "technical operating specifications" (TOS). Chapter VI of GOR consists of procedures for operating the reactor in the event of an incident or accident. Chapter IX of GOR defines the periodic inspection and test programmes for equipment and systems important to safety, implemented to verify their availability. Finally, Chapter X of GOR defines the physical test programme for the reactor core.

5.7.3.1 Operating Technical Specifications

As explained in Section 6.2.2 of D3.3 [3], the operating technical specifications (OTS) form part of the operating documentation that must be developed for the plant operation (note that 'operational limits and conditions' in some countries are called 'technical specifications'). The general objective of the OTS is to set out the rules that must be followed to ensure that during normal operation the reactor remains within the limits justified by the safety case. For this purpose, the OTS must:

- Specify the normal operating limits on the parameters which will ensure compliance with the parameter values assumed in the safety analyses contained in the safety case,
- Determine the operability requirements for the safety systems, structures and components (SSCs) necessary to mitigate transients, incidental scenarios and accidental scenarios considered in the safety case,
- Define in the event of inoperability of the required safety SSCs or any abnormal change in an operating limit, the recovery actions that are required so that the main safety functions are achieved. Regarding each inoperability condition or event and its associated recovery action, the OTS specify a completion time, during which the plant can be maintained in the degraded condition without compromising plant safety.

In the case of the integration of the sCO2-4-NPP system in a power plant, these specifications will have to be re-evaluated.

5.7.3.2 Requirements during operation

As explained in Section 5.2.1 of D3.4 [4], the operating boundary conditions of a plant are given in the GOR. For each power plant operated by EDF, these boundary conditions were calculated on the basis of the type of power plant when the safety report was initially written. Then, during the life of the plant, and depending on the modifications and new regulations, these boundary conditions may be recalculated.

For the sCO2-4-NPP system, boundary conditions will also have to be calculated so that they can be integrated into the system monitoring parameters in the control room. It will therefore also be necessary to ensure that the control system can be integrated into the control room of the power plant and to provide the necessary measurements for monitoring the system when it is operating.

As explained in Section 5.2.2 of D3.4 [4], in terms of maintenance, the objective is to keep all the functions necessary for the safe and economical operation of reactors available and reliable. In essence, maintenance makes it possible to monitor the abnormal behaviour of equipment, particularly following inspections or visits during operation, and, if necessary, to bring them back into conformity by corrective action (repairs, adjustments, replacements, renovations).

The chapter related to the maintenance of GOR (Chapter VII) is completed by the Requirements for Qualified Materials (RQM) and Rules for the Supervision in Operation of Mechanical Equipment (RSOME). These maintenance-related documents are specific to each plant (as they are established for each of the plant components). In the framework of the sCO2-4-NPP system, it will also be necessary to define these rules to integrate them into the different documents (GOR, RQM and RSOME). As the system is not yet precisely and definitively defined, it is difficult to establish these rules at this stage.

As explained in Section 5.2.3 of D3.4 [4], when the plant is in operation, some components, such as diesels, may be subject to periodic testing (as defined in the GOR) to verify their correct start-up.

In the context of the sCO2-4-NPP system, start-up tests may be planned (in addition to more tests during the plant's shutdowns) and will then be integrated into the plant's GOR.

5.7.3.3 Requirements for Maintenance, Tests and Shutdown

As explained in Section 5.3.1 of D3.4 [4], reactors have to be shut down periodically to renew the fuel, which is gradually depleted during the operating cycle. At each shutdown, one third or one quarter of the fuel assemblies are renewed. The length of the operating cycles depends on the fuel management adopted.

These shutdowns make parts of the plant temporarily accessible that are not accessible during operation. They are therefore used to check the state of the installation by carrying out inspection and maintenance operations and to implement planned modifications to the installation.

As explained in Section 5.3.2 of D3.4 [4], the maintenance carried out during plant shutdowns is mainly preventive maintenance (corrective maintenance operations must be carried out as soon as possible to allow the component or system to function).

The sCO2-4-NPP system will also be affected by these maintenance operations. The developers of the main components should therefore consider possible non-destructive testing in parallel with the development of the component. Some components, such as the turbocharger, will probably be able to benefit from technologies already used for components of the same type, but for other components, such as the compact heat exchanger between the secondary loop and the sCO2-4-NPP system, these non-destructive tests will have to be precisely defined and the means of carrying them out will also have to be qualified.

The preventive maintenance operations will also have to be determined in order to be integrated in the various documents related to the operation of the plant.

As explained in Section 5.3.3 of D3.4 [4], the shutdown period is used, in addition to the maintenance that cannot be done during operation, to carry out various regulatory tests on the plant's equipment. In this section, all the tests performed in a nuclear power plant are not presented in detail but the types of tests to which the sCO2-4-NPP system may be subjected. These are commissioning tests and periodic tests.

Any equipment subjected to pressure must pass, after its assembly and before its operation, a series of tests to verify the good conformity of the assembly and that it did not deteriorate the integrity of the system or component. For the sCO2-4-NPP system, commissioning tests will have to be performed to validate the proper installation and operation of the system after its installation.

In addition to the design and qualification requirements applied to classified safety equipment, the periodic tests imposed on this equipment make it possible to guarantee, during operation of the unit, that there is no unfavorable change in the performance required to comply with the assumptions made in the accident studies of the safety report. Some of these periodic tests are carried out during partial outages (during outages for refueling, which take place about every 18 months) and ten-yearly outages. But most of the tests have a higher calendar frequency. In the operation of the sCO2-4-NPP system, it will be necessary to define the list of periodic tests that the system must undergo. Some elements (such as valves or equipment that allow the system to start working properly) will have to be tested with a higher frequency than other components (such as the sCO2/external heat exchanger for example).

5.7.3.4 Human Factors

As explained in Section 6.2.1 of D3.3 [3], relevant international Human Factors standards and guidelines could be applied to the design of the sCO2-4-NPP system and its integration in a plant operation, including

International Standards Organization (ISO) standards, United States Nuclear Regulatory Commission (US NRC) guidance, International Electrotechnical Commission (IEC) standards, Institute of Electrical and Electronic Engineers (IEEE) standards and Electric Power Research Institute (EPRI) guidance. French norms and operator proprietary procedures could also be used.

The design requirements for Human Factors Engineering (HFE) and for the Human-Machine Interface (HMI) are specified in the Design & Construction Rules Applicable to Electrical Equipment (see Sub-chapter 3.8 of RCC-E [53]).

5.7.3.5 Emergency plans

As explained in Section 6.2.3 of D3.3 [3], Emergency Plans should be written in compliance with the requirements of WENRA reference level issue R and Health and Safety Executive (HSE) Safety Assessment Principle (SAP) Fundamental Principle: FP 7 and Accident Management and Emergency Preparedness Principle: AM.1.

The Emergency Plans are regularly reviewed, tested and updated and all employees would be given appropriate training to ensure that the plans can be implemented efficiently and effectively.

The sCO2-4-NPP system will have to be integrated into the emergency plans of the plant where it will be installed. Indeed, the use of CO2-4-NPP could constitute a health situation in case of leakage, and moreover, its presence could imply safety measures for the teams in charge of intervention.

5.7.4 Assessment

Based on information in Section 5.7.2.1, it can be judged that requirements for OLCs Temelín NPP in Czech Republic meet the intent of IAEA requirements. This is achieved by the use of NUREG 1431 Vol. 1 [101] and Vol. 2 [102], which conform to IAEA requirement for OLCs. Regarding requirements for operational tests of SSCs (see Section 5.7.2.2) and requirements for the operation of the specific system (see Section 5.7.2.3), not much information is provided. Nevertheless, based on consideration of IAEA SSR-2/2 (rev. 1) of nuclear regulatory framework in Czech Republic (see Section 5.1.2) it may be assumed that the intent of maintenance, testing, surveillance and inspection programmes requirements of IAEA is in general met.

Review of French requirements for operation and maintenance (summarized in Section 5.7.3) showed that legislation on operational safety and testing of safety-related equipment in nuclear power plants in France is derived from the international texts (IAEA, WENRA). This in principle suggests that the intent of IAEA requirements is met.

Operating technical specification (IAEA uses term Operational Limits and Conditions) are only briefly presented in Section 5.7.3.1. However, from information in Section 5.7.3 it can be concluded that Chapter III of GOR describes the "technical operating specifications" (TOS). It can be assumed that GOR satisfies the IAEA requirements for OLCs.

The operating boundary conditions of a plant are given in the GOR (see Section 5.7.3.2). Similarly, periodic testing is defined in GOR (see Section 5.7.3.2). Maintenance is defined in Chapter VII of GOR (see Section 5.7.3.2). The intent of IAEA requirements seems to be met.

As shown in Section 5.7.3.3, several maintenance and test activities are performed during shutdown. Information on commissioning tests and periodic tests is provided. Some of the periodic tests are carried out during partial outages (during outages for refueling, which take place about every 18 months) and ten-yearly outages. But most of the test have a higher calendar frequency. The intent of IAEA requirements seems to be met.

Some information on French human factors requirements is also given in Section 5.7.3.4. The design requirements for Human Factors Engineering (HFE) and for the Human-Machine Interface (HMI) are specified in the Design & Construction Rules Applicable to Electrical Equipment (see Sub-chapter 3.8 of RCC-E [53]), therefore it is judged that the intent of IAEA requirements is met.

Finally, based on information in Section 5.7.3.5 that French emergency plans should be written in compliance with the requirements of WENRA reference level issue R, it is judged that the intent of IAEA requirements for emergency plans is met.

6 Conclusions

This report presents the results of independent review of the proposed licensing roadmap for safety assessment of the sCO2-4-NPP system considering the international experience in licensing similar systems. In the scope of the independent review were deliverables D3.2 ("Requirements for reference plant modifications for installation of sCO2-4-NPP ") [2], D3.3 ("Design bases and safety analyses for system and components") [3] and D3.4 ("Requirements for testing and operation, including requirements for the preoperational and initial start-up test programmes for the system") [4].

Based on the fact that both Czech Republic and France are harmonizing their legislation against the Western European Nuclear Regulators Association (WENRA) reference levels (RLs) (Czech Republic harmonized all requirements, while in France 73 out of 342 RLs have not yet been harmonized) and that International Atomic Energy Agency (IAEA) safety standards represent international consensus on best international practices to achieve a high level of safety, independent review of compliance of selected country regulations to IAEA standards has been made. Also, IAEA safety standards cover all regulatory and operational aspects of nuclear and radiation safety and are by this more complete safety requirements than selected WENRA RLs, which were originally based on IAEA standards.

The reviewed deliverables D3.2 [2], D3.3 [3] and D3.4 [4] present regulatory requirements and criteria in Czech Republic and in France. The nuclear regulatory framework in these countries is based on the international requirements, mainly by IAEA and WENRA. However, there are different ways of incorporating these rules into national legislation. The regulatory areas considered in the independent review were the nuclear regulatory framework, general safety approach, requirements for SSCs, requirements for plant modifications, requirements for design basis, requirement for equipment qualification, and requirements for operation and maintenance.

The assessment results showed that the nuclear regulatory framework, the general approach to safety, the requirements for structures, systems and components, and plant modifications in an NPP in general conform to the intent of the relevant IAEA standards. Many regulations regarding nuclear safety are similar in these countries as they are based on the same international rules (IAEA, WENRA, etc.). This fact makes it easier for the sCO2-4-NPP system designers to make it possible to implement the system in more European countries.

In the area of design basis for some elements, limited information has been provided to be able to judge that the intent of IAEA relevant requirements is met (i.e., load and load combinations for France, environmental qualification requirements for Czech Republic, seismic qualification requirements for France, reliability requirements for both Czech Republic and France). For element 'design limits', the Czech Republic and France regulations state that design limits will be derived from the final detailed design of the system and that it seems premature to define acceptance criteria for the sCO2-4-NPP system, respectively. Regarding equipment qualification, the Czech Republic requirements focus mainly on qualification quality requirements, while for France, testing qualification and numerical qualification strategy requirements have been provided. Considering the provided information, the intent of IAEA requirements is deemed to be met.

For the provided information regarding operation requirements, the intent of IAEA requirements is deemed to be met in both countries. Here it should be noted that the recommendations on the maintenance and tests to which the sCO2-4-NPP system will be subjected, cannot be established in detail at the time of D3.4 [4] preparation, because of the ongoing development of the system (and therefore the fact that the main components are still at a design stage). However, some basic rules have been presented, in order to consider as much as possible the future constraints in the development phase.

The D3.3 [3] deliverable concludes that the nuclear regulatory framework in the Czech Republic and France is based on the international requirements, mainly by IAEA and WENRA. Nevertheless, there are differences in the regulatory approach, although both are based on the same international texts. Here it should be noted that for the Czech Republic, the IAEA safety requirements standards are legally binding, whereas in France they are meant for orientation.

Finally, the high level requirements for design and operation were reviewed, as provided in D3.2 [2], D3.3 [3] and D3.4 [4] reports (note that information on national safety guides was very limited). When the compliance to these high level requirements are assessed by the national regulators, the recommendations in the national safety guides would be typically followed to demonstrate the compliance to the requirements (safety guides, not legally binding, provide guidance to licensees and applicants on implementing specific parts of the regulations; alternative ways are allowed). The safety guides are typically very detailed and for the sake of this review the IAEA safety guides were identified giving recommendations how to satisfy IAEA safety requirements.

7 References

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